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NL-13-122

September 27, 2013

U.S. Nuclear Regulatory Commission
Document Control Desk
11545 Rockville Pike, TWFN-2 F1
Rockville, MD 20852-2738

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, SET 2013-04" dated July 26, 2013.

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 response to RAI 16-A includes new Commitment 50 that concerns the planned replacement of the IP2 splits pins. The response to RAI 11-B includes a revision to the implementation date for Commitment 47. These new and revised commitments are included in the latest list of regulatory commitments provided in Attachment 2. This list has also been updated to reflect closure of all the IP2 commitments required to be implemented prior to the PEO and closure of select IP3 commitments.

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 9/27, 2013.

Sincerely,

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 22

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
Ms. Kimberly Green, 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-13-122

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

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

REACTOR VESSEL INTERNALS PROGRAM AND INSPECTION PLAN

RAI 11-B

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

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

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

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

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

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

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

5) MRP-227-A, Section 4.2.7, requires the plant-specific analysis for Applicant/Licensee Action Item 7 demonstrating that the lower support column bodies (expansion components) will maintain their functionality during the period of

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

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

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

Response to RAI 11-B

1) The assumption that only a limited number of columns will contain flaws of significant size is based on the qualitative factors discussed in detail below. These factors are the lack of significant flaws in the columns at manufacturing, the lack of a credible relevant flaw enhancement mechanism during service, and the operational experience that shows a lack of cracking and loose parts that would be expected from failed columns.

   The size and number of potential pre-existing flaws in the lower core support column caps is considered to be limited because, prior to component assembly, all of the columns were inspected using dye penetrant and radiography. All columns met ASTM E-71 standards. All columns were considered defect free to this level and were deemed to exhibit zero surface-breaking flaws. Based on this inspection, any remaining flaws would be expected to be of small size and number. Therefore, the potential number of flaws of sufficient size to be relevant to embrittlement-related fracture processes would be small.

   Flaw development due to other screened-in mechanisms occurring during service is not considered a viable mechanism for the production of a significant number of size-relevant flaws either by itself or from the original as-manufactured distribution of flaws. Potential mechanisms for the development of new flaws or growth of existing flaws are irradiation-assisted stress corrosion cracking (IASCC) and fatigue. Per the following, neither mechanism is expected to be viable for significant development of new flaws or growth of existing flaws. Per MRP-175 [1], IASCC is a mechanism for service aging degradation of cast austenitic stainless steel (CASS). However, under the conditions of operation, it is not expected that IASCC can cause sufficient additional degradation to increase the susceptibility to embrittlement-driven fracture. A detailed discussion of the factors controlling IASCC of wrought stainless steel and CASS is provided in Appendix B of [1]. This discussion demonstrates that IASCC processes are only significant for wrought stainless steel and CASS at relatively high stresses and neutron exposures. Even at several tens of dpa, the threshold stress for the onset of IASCC is over 40 ksi, while at the lower neutron exposures expected for the column cap regions, the threshold stress for IASCC would be approximately 70 ksi or greater. Since the nominal stresses developed in the columns during normal plant operation are significantly below these values, on the order of less than 20 ksi, IASCC is not expected to contribute significantly to the development of flaws. Fatigue is a potential aging mechanism that has been evaluated for Indian Point Unit 2. The fatigue evaluation, which determined that all environmentally adjusted cumulative usage factors (CUF_{en}s) for the support columns are less than 1.0, has demonstrated that the Indian Point Unit 2 lower support columns are acceptable for fatigue through the period of extended operation. Because of this evaluation, we do not expect to generate or grow any structurally significant flaws as a result of fatigue during the period of extended operation.

Operating experience also supports the view that the number of cracked columns will be limited. Although the limited access to lower core support column cap sections has precluded extensive observation and inspection, no cracked columns have been observed to date. Furthermore, extensive column cracking would be expected to produce loose parts, and there has been no evidence of such parts found in the reactor coolant system. Reference [2] summarizes a survey of operating experience of operating pressurized water reactor designs in the U.S. The survey included responses from similar operating plants worldwide. The survey specifically requested reporting of any relevant operating experience with MRP-227-A components, including failures or inspections that have not detected off-normal conditions in the components. As a result of the survey, there was no reported degradation or off-normal conditions noted in the lower support columns for the operating fleet. A summary of the survey, WCAP-17435-NP, was provided for information to the US NRC by the Pressurized Water Reactor Owners Group (PWROG.)

Based on the preceding discussion, it is expected that no column failures would occur during the period of extended operation. As noted in the response to item 5 of this RAI, Entergy plans to provide a plant-specific functionality analysis of this component. As part of this analysis, a quantitative assessment of the impact of potentially failed columns will be performed.

2) Industry guidance (including NUREG-1801 Rev. 1 Section XI.M13) [3,4] specifies tensile stress levels to be considered as significant in performing screening evaluations. However, in the plant-specific screening analysis, no complete columns will be screened out based on the stress criteria. Therefore, a functionality analysis will be performed as noted in the response to item 5 of this RAI.

3) The grade of stainless steel used in the upper sections of the lower support columns is ASTM 296 Grade CF-8. This material designation is consistent with the chemistries of the columns as identified in plant CMTRs. No special casting processes were designated; thus, it is determined that the lower core support column caps were statically cast. After casting, surface mechanical clean up (grinding) was permitted to meet the requirements of a 250-microinch finish. No specific surface finishing process was designated or disallowed. After heat treatment, the bolt holes were centerline bored and machined to allow fitting to the lower core support column forging and to allow bolting at the correct position to the lower core plate. Finally, the cast upper

section of the lower support column was welded to the wrought lower section of the core support column with a circumferential weld.

The circumferential weld that joins the upper (cast) section of the lower support column to the lower (wrought) section of the lower support column is approximately 18 inches below the upper section to core plate interface.

4) The most recent ASME Code Section XI inservice inspection of the core support structure (ASME Section XI Category B-N-3) was performed at IP2 in May 2006. The inspection utilized a camera attached to a remote underwater examination vehicle (submarine) and only the portion of the lower support column bodies below the dome lower support plate (specifically the exterior bottom of the core barrel) were inspected. The portion of the lower support column bodies below the dome lower support plate is the end of the column body that extends past the lower support column nut. The portion of the lower support column bodies that was inspected was the wrought lower section. All accessible surfaces were inspected with no limitations noted; however, a specific amount of coverage was not documented on the data sheets. All inspections were satisfactory with no recordable or rejectable indications noted.

   The most recent ASME Code Section XI inservice inspection of the core support structure (ASME Section XI Category B-N-3) was performed at IP3 in March 2009. The inspection utilized a camera attached to a remote underwater examination vehicle (submarine) and only the portion of the lower support column bodies below the dome lower support plate (specifically the exterior bottom of the core barrel) were inspected. The portion of the lower support column bodies below the dome lower support plate is the end of the column body that extends past the lower support column nut. The portion of the lower support column bodies that was inspected was the wrought lower section. All accessible surfaces were inspected; however, a specific amount of coverage was not documented on the data sheets. The lower internals exterior (core barrel) bottom section and sides (approximately 350 degrees clockwise thru 100 degrees) were restricted from examination due to the core barrel location relative to the refueling cavity wall and the stand for the internals. All inspections were satisfactory with no recordable or rejectable indications noted.

5) In order to provide the NRC staff with the requested 18 month review time, Entergy is revising Commitment 47 as follows.

   Commitment 47 – revision to implementation date

   The implementation date for commitment 47 for IP2 is being revised from March 1, 2015 to August 15, 2014.

RAI 15-B

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

Response to RAI 15-B

New sections A.2.2.2.3 and A.3.2.2.3 have been created for the discussion of the reactor vessel internals. "Reactor vessel internals" has been deleted from the list of Class 1 components in Sections A.2.2.2.1 and A.3.2.2.1.

The commitment to complete the revised fatigue cumulative usage factor analyses accounting for environmental effects (Commitment 49) is discussed in new Sections A.2.2.2.3 and A.3.2.2.3.

The commitment to complete the revised fatigue cumulative usage factor analyses accounting for environmental effects for reactor vessel internals also affects LRA Section B.12, Fatigue Monitoring. This section is revised to include Subsection NG for reactor vessel internals.

LRA Appendix A Sections A.2.2.2 and A.3.2.2 are revised as shown below. New sections are added for reactor vessel internals. Changes are shown as strikethroughs for ~~deletions~~ and underlines for <u>additions.</u>

Section A.2.2.2.1, first paragraph, is revised as follows:

A.2.2.2.1 Class 1 Metal Fatigue

Class 1 components evaluated for fatigue and flaw growth include the reactor pressure vessel (RPV), ~~reactor vessel internals,~~ pressurizer, steam generators, reactor coolant pumps, control rod drive mechanisms, regenerative letdown heat exchanger, and Class-1 piping and in-line components.

New Section A.2.2.2.3 is added; existing section A.2.2.2.3 is renumbered to A.2.2.2.4:

<u>A.2.2.2.3 Subsection NG Fatigue Analysis of Reactor Pressure Vessel Internals</u>

<u>The reactor vessel internals were designed to meet the intent of Subsection NG of the ASME Boiler and Pressure Vessel Code, Section III. Subsequent plant uprate evaluations determined CUFs for some reactor vessel internals components. These evaluations were performed to the intent of Subsection NG. The Fatigue Monitoring Program manages the effects of aging related to these TLAAs (fatigue analyses) in accordance with 10 CFR 54.21(c)(1)(iii).</u>

Each of the limiting CUFs for the reactor vessel internals will be recalculated prior to September 28, 2013, to include the reactor coolant environment effects (F_{en}) as provided in the Fatigue Monitoring Program using NUREG/CR-5704 or NUREG/CR-6909. Corrective actions specified in the Fatigue Monitoring Program include further CUF re-analysis and/or repair or replacement of the affected components prior to the CUF_{en} reaching 1.0.

A.2.2.2.~~3~~4 Environmental Effects on Fatigue

Section A.3.2.2.1, first paragraph, is revised as follows:

A.3.2.2.1 Class 1 Metal Fatigue

Class 1 components evaluated for fatigue and flaw growth include the reactor pressure vessel (RPV), ~~reactor vessel internals,~~ pressurizer, steam generators, reactor coolant pumps, control rod drive mechanisms, regenerative letdown heat exchanger, and Class-1 piping and in-line components.

New Section A.3.2.2.3 is added; existing section A.3.2.2.3 is renumbered to A.3.2.2.4:

A.3.2.2.3 Subsection NG Fatigue Analysis of Reactor Pressure Vessel Internals

The reactor vessel internals were designed to meet the intent of Subsection NG of the ASME Boiler and Pressure Vessel Code, Section III. Subsequent plant uprate evaluations determined CUFs for some reactor vessel internals components. These evaluations were performed to the intent of Subsection NG. The Fatigue Monitoring Program manages the effects of aging related to these TLAAs (fatigue analyses) in accordance with 10 CFR 54.21(c)(1)(iii).

Each of the limiting CUFs for the reactor vessel internals will be recalculated prior to December 12, 2015, to include the reactor coolant environment effects (F_{en}) as provided in the Fatigue Monitoring Program using NUREG/CR-5704 or NUREG/CR-6909. Corrective actions specified in the Fatigue Monitoring Program include further CUF re-analysis and/or repair or replacement of the affected components prior to the CUF_{en} reaching 1.0.

A.3.2.2.~~3~~4 Environmental Effects on Fatigue

Section B.1.12, Program Description, third paragraph, is revised as follows:

The analysis methods for determination of stresses and fatigue usage will be in accordance with an NRC endorsed Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III Rules for Construction of Nuclear Power Plant Components Division 1 Subsection NB, Class 1 Components, Sub articles NB-3200 or NB-3600 and Subsection NG, Requirements for Class CS Components, Core Support and Internal Structures as applicable to the specific component. IPEC will utilize design transients from IPEC Design Specifications to bound all operational transients. The numbers of cycles used for evaluation will be based on the design number of cycles and actual IPEC cycle counts projected out to the end of the license renewal period (60 years).

RAI 16-A

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

Response to RAI 16-A

Entergy provides the following commitment for providing a detailed inspection plan for the IP2 split pins if the planned replacement of the IP2 split pins is not to be implemented in 2016.

Commitment 50

If the planned replacement of the IP2 split pins will not be accomplished in 2016, provide the NRC staff a detailed inspection plan for the IP2 split pins, including inspection methods, inspection coverage, and inspection frequency, by March 31, 2015.

Because the new commitment only affects IP2, LRA Section A.3.1.41 does not require revision. Changes to the RVI Program description also affect LRA Section B.1.42.

The following paragraph is added to LRA Section A.2.1.41 as the fourth paragraph (additions are underlined):

The IP2 guide tube support pins (split pins) are scheduled to be replaced during the 2016 refueling outage. If the planned replacement of the IP2 split pins will not be accomplished in 2016, Entergy will provide the NRC staff a detailed inspection plan, including inspection methods, inspection coverage, and inspection frequency, no later than March 31, 2015.

The following paragraph is added to LRA Section B.1.42, Reactor Vessel Internals Program, under "Evaluation/1. Scope of Program," new last paragraph:

The IP2 guide tube support pins (split pins) are plant-specific components as discussed in MRP-227-A, Section 4.4.3, "Westinghouse Components." The split pins are scheduled to be replaced during the 2016 refueling outage. See letter NL-12-166, Entergy to NRC, response to RAI 16, dated November 20, 2012, for further discussion. If the planned replacement of the IP2 split pins will not be accomplished in 2016, Entergy will provide the NRC staff a detailed inspection plan, including inspection methods, inspection coverage, and inspection frequency, no later than March 31, 2015.

RAI 17

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

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

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

Response to RAI 17

Entergy provides the following technical justification for the adequacy of the existing inspection requirement to manage the effects of PWSCC.

The main function of the lower radial support system (LRSS) is to prevent tangential or rotational motion of the lower internals assembly while permitting axial displacement and differential radial expansion. Indian Point Units 2 and 3 have six radial supports spaced at 60 degree intervals around the circumference of the vessel (see Reference [5], Figure 1). Because of the small tangential clearance between the radial keys and the clevis insert, the keys are potentially subjected to flow-induced vibration loads and wear at the key-to-keyway (clevis) interface. These supports are designed to prevent excessive tangential displacement of the lower internals during seismic and loss of coolant accident (LOCA) conditions. The supports also limit displacements and misalignments in order to avoid overstressing the core barrel and to ensure that the control rods can be freely inserted. Therefore, providing the clevis inserts remain in place, the design function of the LRSS will be maintained during seismic and LOCA conditions.

Crack detection prior to bolt failure is not required due to inherent design redundancy. The ability of the LRSS to perform its intended design function under seismic and LOCA condition loadings is unrelated to the integrity of the cap screws and pins that are used to hold the clevis insert in place. The cap screws and the dowel pins hold the clevis inserts in place so as to minimize long term vibration and wear of the mating parts.

Should cap screws fail during operation, it could result in potential increased wear of mating surfaces. Any increased wear, which would occur over several operating cycles, will not impact the function of the reactor internals components. This is based on operating experience with damaged bolts and one dowel pin as described in the InfoGram [5] which showed no discernible change in the clevis insert wear surfaces after operation for two additional cycles.

Complete disengagement of one of the clevis inserts is highly unlikely based on the available gaps with surrounding components (see Figure 1). Even if it were postulated that one of the clevis inserts becomes non-functional, the other lower radial supports are capable of resisting all of the internal and external asymmetric loads. Wear or some degradation of a key might occur, but the key would still be expected to maintain functionality. Taken as a whole, the core barrel and LRSS system are expected to maintain their design function with degraded clevis insert bolts. Based on the evaluations performed to date, there are no safety or operability concerns with clevis insert bolt failure.

As described in the InfoGram [5], Westinghouse performed evaluations of the potential for loose parts with failed clevis insert bolts for the plant referenced in this RAI. The loose parts evaluation concluded that the separated cap screw heads will remain captured in the clevis insert counterbores and will not impact operation. However, lock bars at the degraded cap screw locations have experienced wear-related degradation; therefore, the potential for loose parts from the lock bars to affect other locations in the reactor vessel was also evaluated. Westinghouse concluded that no significant degradation of mechanical components is expected as a result of potential loose parts from the lock bars in the primary system.

The MRP-227-A categorization for wear only is based on the primary concern for clevis insert looseness and wear of the clevis insert and radial key interfacing surfaces that could potentially lead to increased motion at the bottom end of the core barrel, rather than bolt material cracking. The video camera visual inspections at a ten-year interval by qualified personnel that are specified in the ASME Code Section XI and MRP-227-A are capable of identifying wear or dislodged components of the clevis insert cap screws or dowel pins at any location, if they exist.

The susceptibility of Alloy X-750 to PWSCC and low-temperature crack propagation may have been a contributor to the observed degradation detected in 2010; however, at this time, this has not been confirmed by metallurgical analysis. It was also indicated by the Staff that most failures of Alloy X-750 material have occurred in material with heat treatment condition AH. The Alloy X-750 material used at Indian Point Units 2 and 3 for clevis insert bolts is not heat treatment condition AH.

Industry operating experience, such as metallurgical test results, will continue to be evaluated for applicability as part of the operating experience program at Indian Point Energy Center.

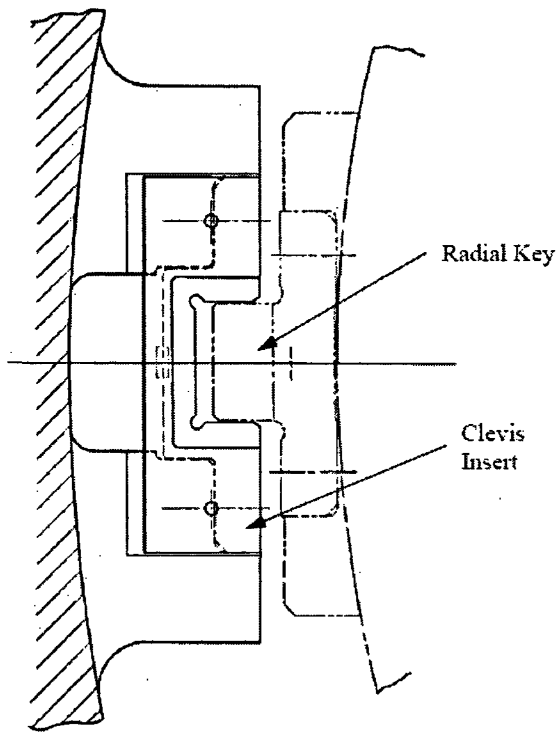

Figure 1 Lower Radial Support System Engagement

Entergy References

1. Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175). EPRI, Palo Alto, CA: 2005. 1012081.

2. Westinghouse Report, WCAP-17435-NP, Rev. 1, "Results of the Reactor Internals Operating Experience Survey Conducted under PWROG Project Authorization PA-MSC-0568," October 22, 2012.

3. BWRVIP-234: BWR Vessel and Internals Project, Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steels for BWR Internals. EPRI, Palo Alto CA: December 2009. 1019060.

4. U. S. Nuclear Regulatory Commission NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," July 2001.

5. Westinghouse InfoGram, IG-10-1, "Reactor Internals Lower Radial Support Clevis Insert Cap Screw Degradation," March 31, 2010.

NRC References

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

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

Footnote to NRC RAI 17:

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

ATTACHMENT 2 TO NL-13-122

LICENSE RENEWAL APPLICATION
IPEC LIST OF REGULATORY COMMITMENTS

Rev. 22

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

List of Regulatory Commitments

Rev. 22

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

Changes are shown as strikethroughs for ~~deletions~~ and underlines for additions.

#	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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122	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~~ Complete IP3: ~~December 12, 2015~~ Complete	NL-07-039 NL-07-153 NL-13-122	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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122 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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122 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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122	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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122 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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122	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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122 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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122	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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122 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~~ Complete IP3: ~~December 12, 2015~~ Complete	NL-07-039 NL-13-122	A.2.1.18 A.3.1.18 B.1.19
13	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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122 NL-07-153 NL-08-057 NL-13-077	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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122	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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122 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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122 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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122 NL-07-153	A.2.1.24 A.3.1.24 B.1.25 Audit item 173

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122 NL-11-101	A.2.1.25 A.3.1.25 B.1.26
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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122 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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122 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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122	A.2.1.28 A.3.1.28 B.1.29

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122	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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122 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~~ Complete IP3: ~~December 12, 2015~~ Complete	NL-07-039 NL-13-122	A.2.1.34 A.3.1.34 B.1.35
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)	IP2: ~~September 28, 2013~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122 NL-07-153 NL-08-057 NL-13-077	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
	• 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 • new fuel storage racks • sumps 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.		NL-13-077	

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
	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. 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. 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-08-127 NL-11-032 NL-11-101	Audit Item 360 Audit Item 358
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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122 NL-07-153	A.2.1.36 A.3.1.36 B.1.37 Audit item 173

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
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~~ Complete IP3: ~~December 12, 2015~~ Complete	NL-07-039 NL-13-122 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~~ Complete IP3: ~~December 12, 2015~~ Complete	NL-07-039 NL-13-122 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~~ Complete	NL-07-039 NL-13-122	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~~ Complete IP3: ~~December 12, 2013~~ Complete	NL-07-039 NL-13-122 NL-11-107	A.2.1.41 A.3.1.41
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~~ Complete IP3: December 12, 2015	NL-07-039 NL-13-122	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

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
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~~ Complete IP3: ~~December 12, 2013~~ Complete	NL-07-039 NL-13-122 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
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-13-122 NL-07-078 NL-08-074 NL-11-101	2.1.1.3.5

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
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~~ Complete IP3: December 12, 2015	NL-08-127 NL-13-122 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~~ Complete	NL-08-127 NL-11-101 NL-13-122 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~~ Complete IP3: ~~December 12, 2015~~ Complete	NL-08-127 NL-13-122	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~~ Complete IP3: December 12, 2015	NL-08-143 NL-13-122	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~~ Complete IP3: December 12, 2015	NL-09-106 NL-13-122	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~~ <u>Complete</u> IP3: Prior to December 12, 2015	NL-11-032 <u>NL-13-122</u> 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~~ Complete IP3: Prior to December 12, 2015	NL-11-032 NL-11-101 NL-13-122	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~~ Complete IP3: Prior to December 12, 2015	NL-11-032 NL-11-101 NL-13-122	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~~ Complete	NL-11-032 NL-11-074 NL-13-122	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 ~~March 1, 2015~~ August 15, 2014 IP3: Prior to December 12, 2015	NL-12-089 NL-13-052 NL-13-122	N/A

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
48	Entergy will visually inspect IPEC underground piping within the scope of license renewal and subject to aging management review prior to the period of extended operation and then on a frequency of at least once every two years during the period of extended operation. This inspection frequency will be maintained unless the piping is subsequently coated in accordance with the preventive actions specified in NUREG-1801 Section XI.M41 as modified by LR-ISG-2011-03. Visual inspections will be supplemented with surface or volumetric non-destructive testing if indications of significant loss of material are observed. Consistent with revised NUREG-1801 Section XI.M41, such adverse indications will be entered into the plant corrective action program for evaluation of extent of condition and for determination of appropriate corrective actions (e.g., increased inspection frequency, repair, replacement).	IP2: ~~Prior to September 28, 2013~~ Complete IP3: Prior to December 12, 2015	NL-12-174 NL-13-122	N/A
49	Recalculate each of the limiting CUFs provided in section 4.3 of the LRA for the reactor vessel internals to include the reactor coolant environment effects (F_{en}) as provided in the IPEC Fatigue Monitoring Program using NUREG/CR-5704 or NUREG/CR-6909. In accordance with the corrective actions specified in the Fatigue Monitoring Program, corrective actions include further CUF re-analysis, and/or repair or replacement of the affected components prior to the CUF_{en} reaching 1.0.	IP2: ~~Prior to September 28, 2013~~ Complete IP3: Prior to December 12, 2015	NL-13-052 NL-13-122	A.2.2.2 A.3.2.2
50	If the planned replacement of the IP2 split pins will not be accomplished in 2016, provide the NRC staff a detailed inspection plan for the IP2 split pins, including inspection methods, inspection coverage, and inspection frequency, by March 31, 2015.	IP2: Prior to March 31, 2015 IP3: N/A	NL-13-122	A.2.1.41 B.1.42