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Fred Dacimo
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NL-12-140

October 17, 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, SET 2012-02" dated September 10, 2012

Dear Sir or Madam:

Entergy Nuclear Operations, Inc is providing, in Attachment 1, a reply to the additional information requested in Reference 1 pertaining to NRC review of the License Renewal Application (LRA) for Indian Point 2 and Indian Point 3.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on 10/17/12.

Sincerely,

FRD/rw

A128
NRR

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

cc: Mr. William Dean, Regional Administrator, NRC Region I
Mr. Sherwin E. Turk, NRC Office of General Counsel, Special Counsel
Mr. Dave Wrona, NRC Branch Chief, Engineering Review Branch I
Mr. John Daily, 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-12-140

REPLY TO NRC REQUEST FOR ADDITIONAL INFORMATION

REGARDING THE

LICENSE RENEWAL APPLICATION

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

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

RAI 13

The response to request for additional information (RAI) 3 states that the ASME Section XI Inservice Inspection (ISI) Program is the existing program credited with managing cracking of the "Upper Support Plate, Support Assembly (Including Ring)" in Tables 3.1.2-2-IP2 and 3.1.2-2-IP3 of LRA Amendment 9, which provide the results of the aging management review (AMR) of the reactor vessel internals (RVI) for Indian Point Nuclear Generating Units 2 and 3 (IP2 and IP3). The staff compared LRA Tables 3.1.2-2-IP2 and 3.1.2-2-IP3 for consistency with Table 5-4 of the RVI Inspection Plan, which identifies the applicant's "Existing Programs" components corresponding to Table 4-9 of MRP-227-A. This review identified some apparent inconsistencies between Table 5-4, "Existing Program Components at IPEC Units 2 and 3," of the RVI Inspection Plan and Tables 3.1.2-2-IP2 and 3.1.2-2-IP3, with respect to the existing program credited with managing the aging effect. The table below compares the aging effects and aging management programs (AMPs) identified in Table 5-4 of the RVI Inspection Plan versus those identified in Tables 3.1.2-2-IP2 and 3.1.2-2-IP3.

Furthermore, Table 5-4 of the RVI Inspection Plan identifies the aging mechanism causing the loss of material aging effect as wear, which the Water Chemistry- Primary and Secondary AMP does not address. The staff notes that the aging effects and mechanisms, AMPs, examination methods and coverage identified in Table 5-4 of the RVI Inspection Plan are consistent with those recommended in Table 4-9 of MRP-227-A. The staff also notes that there is a component named "Bottom Mounted Instrumentation – Flux Thimble Tube" in Tables 3.1.2-1-IP2 and 3.1.2-1-IP3, "Reactor Vessel," of the applicant's LRA.

Item	IPEC Name	Effect/ Mechanism – Table 5-4	Aging Effect Requiring Management – Tables 3.1.2-2-IP2 and –IP3	AMP – Table 5-4	AMP – Tables 3.1.2-2-IP2 and –IP3
Bottom Mounted Instrumentation System – Flux Thimble Tubes	Flux Thimble Guide Tube	Loss of Material/ Wear	Loss of Material	NUREG-1801, Rev.1	Water Chemistry – Primary and Secondary
Alignment and Interfacing Components – Clevis Insert Bolts	Lower Internals Assembly – Clevis Insert Bolt	Loss of Material/ Wear	Loss of Material	ISI	Water Chemistry – Primary and Secondary

Requested Information

1. Clarify the inconsistency between Table 5-4 of the RVI Inspection Plan and the AMR tables, with respect to the two components noted in the table above.

2. Clarify whether the component named "Bottom Mounted Instrumentation – flux thimble tube" in Tables 3.1.2-1-IP2 and 3.1.2-1-IP3, "Reactor Vessel," of the IPEC LRA, is the same component as the component named "flux thimble guide tube," in Tables 3.1.2-2-IP2 and 3.1.2-2-IP3 of Amendment 9 to the LRA.

Response to RAI 13

1. In the IPEC license renewal aging management review, the flux thimble tubes, and the flux thimble guide tubes external to the reactor vessel, were evaluated as part of the reactor vessel and the aging management review results were presented in LRA Tables 3.1.2-1-IP2 and -IP3. The flux thimble guide tube listed in LRA Tables 3.1.2-2-IP2 and IP3 as part of the reactor vessel internals, refers to the short extension of the guide tubes internal to the reactor vessel that are part of the BMI (bottom mounted instrumentation) column bodies. Consequently, the Bottom Mounted Instrumentation System – Flux thimble tubes listed in Table 5-4 of the RVI Inspection Plan, are the same as the Bottom mounted instrumentation – flux thimble tube listed in LRA Tables 3.1.2-1-IP2 and -IP3. These LRA tables identify loss of material – wear as an applicable aging effect and identify the Flux Thimble Tube Inspection Program as the aging management program. This is consistent with Table 5-4 of the RVI Inspection Plan. The LRA table lines indicating loss of material managed by the Water Chemistry – Primary and Secondary Program, refer to loss of material due to pitting and crevice corrosion, but not due to wear.

   In LRA Tables 3.1.2-2-IP2 and -IP3, loss of material due to wear was identified as an aging effect for the clevis inserts, but not for the clevis insert bolts. This is consistent with NUREG-1801, Rev. 1 which does not identify an aging effect of loss of material due to wear for the bolts. However, MRP-227-A identifies loss of material due to wear as an aging effect for the clevis insert bolts, and the RVI Inspection Plan manages this aging effect accordingly. The LRA table lines for clevis insert bolts indicating loss of material managed by the Water Chemistry – Primary and Secondary Program, refer to loss of material due to pitting and crevice corrosion, but not due to wear. For consistency the following line item is added to LRA Tables 3.1.2-2-IP2 and -IP3.

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
Lower internals assembly • clevis insert bolt	Structural support	Nickel alloy	Treated borated water	Loss of material – wear	Inservice Inspection	--	--	H

2. See the response to Item 1. The component named "Bottom Mounted Instrumentation – flux thimble tube" in LRA Tables 3.1.2-1-IP2 and -IP3, is not the same component as the "flux thimble guide tube," in LRA Tables 3.1.2-2-IP2 and -IP3.

RAI 15

The response to RAI 12 states that, for RVI components that are not covered by a time-limited aging analysis, Entergy will use the RVI Program to manage the effects of aging due to fatigue on the reactor vessel internals. The response also states that, 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, Entergy will manage the effects of aging due to fatigue through its Fatigue Monitoring Program in accordance with 10 CFR 54.21(c)(1)(iii).

In its response, the applicant also stated that the Fatigue Monitoring Program as described in LRA Section B.1.12 provides assurance that the cumulative usage factors (CUFs) remain below the allowable limit of 1.0 and that, consistent with Section 3.5.1 of the safety evaluation for MRP-227-A, prior to entering the period of extended operation, Entergy will review the existing RVI fatigue calculations to evaluate the effects of the reactor coolant system water environment on the CUF. Specifically, under Commitment 43, Entergy stated that it will review the units' 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 IP2 and IP3. The applicant stated that this review will also include ASME Code Class 1 fatigue evaluations for reactor vessel internals. Based on this review, if more limiting locations are identified, Entergy will evaluate the most limiting location for the effects of the reactor coolant environment on fatigue usage. The applicant's response is not clear regarding how the "ASME Code Class 1 fatigue evaluations for reactor vessel internals" will account for the effects of the reactor coolant environment, nor what actions will be taken if CUF's for RVI components exceed 1.0.

Requested Information

1. Clarify whether, as a result of the review described in the response to RAI 12, CUF calculations for RVI components that incorporate environmental factors (F_{en}) will be performed in response to Applicant/Licensee Action Item 8 of the Staff SE of MRP-227-A. If such calculations will not be performed, discuss how the effects of the reactor water environment on the existing CUF analyses for RVIs will be evaluated in response to Applicant/Licensee Action Item 8 of the Staff SE of MRP-227-A.

2. Clarify what action(s) will be taken if the consideration of environmental effects results in a CUF exceeding 1.0 for any RVI component.

3. Since ASME Code Class 1 components are designed to ASME Section III, Subsection NB (i.e., reactor coolant pressure boundary components, not reactor vessel internals), provide necessary revisions to clarify the term "ASME Code Class 1 fatigue evaluations for reactor vessel internals" and any inconsistency in the response to RAI 12.

4. For the purposes of clarity, provide a new commitment and an associated new UFSAR Supplement to address the review of reactor vessel internals for environmentally-assisted fatigue as part of the Fatigue Monitoring Program in response to Applicant/Licensee Action Item 8 of the Staff SE of MRP-227-A, in lieu of your proposal to use Commitment 43.

Response to RAI 15

Indian Point Units 2 and 3

1. Section 3.5.1 of the NRC's Safety Evaluation (SE) of MRP-227, Revision 0 requires that reactor vessel internals time-limited aging analyses (TLAAs) be submitted along with the application to implement the approved version of MRP-227-A. For locations with a fatigue TLAA, Entergy stated in the response to RAI 12 that Indian Point Energy Center (IPEC) would manage the effects of aging due to fatigue through the Fatigue Monitoring Program in accordance with 10 CFR.54.21(c)(1)(iii). Although MRP-227-A does not directly address TLAAs for the reactor vessel internals, Section 3.5.1 of the SE requires that fatigue TLAAs that are evaluated as acceptable by any means other than a periodic component inspection program must account for the effects of the reactor coolant system environment in the associated fatigue analyses. The environmental F_{en} factors cited in the SE were developed to apply to fatigue in ASME Class 1 pressure boundary components. Application of these F_{en} factors to reactor internals fatigue analyses may not provide realistic cumulative usage factors (CUFs) for a Fatigue Monitoring Program. Given the large degree of conservatism included in the analysis of fatigue, it is clear that application of environmental factors to current licensing basis CUF values for reactor vessel internals may produce values greater than 1.0. Therefore, Entergy hereby amends the previously submitted RAI 12 response to indicate that cracking due to fatigue in the reactor vessel internals will be managed by periodic inspections under the Reactor Vessel Internals Program (MRP-227-A program) and under the Inservice Inspection (ISI) Program. This approach will manage the effects of aging due to fatigue in accordance with 10 CFR 54.21(c)(1)(iii).

   Although the original design specifications during plant construction did not require explicit fatigue analyses of the reactor vessel internals, subsequent plant uprate evaluations for Indian Point Unit 2 (IP2) [1] and Indian Point Unit 3 (IP3) [2] determined CUFs for some reactor vessel internal components. These analyses are treated as time-limited aging analyses per 10 CFR 54.21(c)(1). Table 1 lists components with a fatigue analysis for each unit.

Table 1
RV Internal Components with Fatigue Analyses

IP2	IP3
• Lower Core Support Plate	• Lower Core Plate
• Lower Support Columns	• Upper Core Plate
• Mid Core Barrel	• Lower Support Columns
• Upper Core Barrel	• Instrument Columns
• Core Barrel Outlet Nozzle	• Core-Barrel-to-Lower-Support-Plate Junction
• Core Barrel Flange	• Thermal Shield
• Lower Radial Key Base	• Top Hat Structure
• Lower Radial Key	
• Upper Support Assembly – Perforated Plate	
• Upper Support Assembly – Skirt	

The CUFs in these analyses were calculated using the nuclear steam supply system (NSSS) design transient cycle numbers shown in LRA Table 4.3-1 in the column titled "Analyzed Numbers of Cycles." From the IPEC license renewal application, the analyzed design transient cycle numbers were shown to remain valid for 60 years of plant operation based on cycles to date and projected cycles through the period of extended operation. Since the analyzed NSSS design transient cycle numbers remain valid for the period of extended operation, the original CUFs for the reactor internal components also remain valid with values less than 1.0. Therefore, the fatigue analyses without considering environmental effects would remain valid through the period of extended operation. Since the Fatigue Monitoring Program was credited to ensure the analyzed numbers of cycles remain valid, the LRA credited the Fatigue Monitoring Program to manage the effects of aging due to fatigue on the reactor internals components with CUFs, consistent with 10 CFR 54.21(c)(1)(iii). As discussed below, the RVI Program is now credited to address the effects of fatigue of reactor vessel internals components, including the effects of the reactor coolant environment, also consistent with 10 CFR 54.21(c)(1)(iii).

There are thirteen reactor vessel internals components included in the Reactor Vessel Internals Program MRP-227-A inspection plan for the Indian Point units. For these thirteen components, MRP-227-A lists fatigue as one of the potential aging related degradation mechanisms. There is a close correspondence between the TLAA components listed in Table 1 and the MRP-227-A components. Table 2 provides a guide to the nomenclature and illustrates this correspondence. Further details on the aging management strategies for each of the components with TLAA listed in Table 1 are provided below.

1. Core-Barrel-to-Lower-Support-Plate Junction: MRP-227-A lists the lower core barrel flange weld as a Primary Inspection. In the Entergy inspection plan, it is correctly noted that the Indian Point core barrels do not have a flange at this location. However the plant-specific RVI program clearly requires inspection of the core barrel to lower support casting weld, which is the equivalent location. The effects of aging due to fatigue on the core-barrel-to-lower-support-plate junction for IP3 are adequately addressed through the RVI Program inspections, and therefore, no additional actions are required to address the effects of aging due to fatigue during the period of extended operation.

2. Thermal Shield: MRP-227-A requires a Primary visual inspection of the thermal shield flexures. The flexure has been determined to be the lead fatigue location on the thermal shield. The effects of aging due to fatigue on the thermal shield for IP3 are adequately addressed through the RVI Program inspections, and therefore, no additional actions are required to address the effects of aging due to fatigue during the period of extended operation.

3. Upper Core Plate: The RVI Program includes the upper core plate as an Expansion item with the lead item being the nearby control rod guide tube (CRGT) lower flange welds. Placement of the upper core plate as an expansion item is reasonable as the reported CUF is only 0.062. The effects of aging due to fatigue on the upper core plate for both IP2 and IP3 are adequately addressed through the RVI Program inspections, and therefore, no additional actions are required to address the effects of aging due to fatigue during the period of extended operation.

4. Lower Support Columns: The fatigue analyses for the lower support columns apply to the lower support column bolts. MRP-227-A identifies the lower support column bolts as an Expansion item. The lower support column bolts are listed as an expansion item because baffle-former bolts experience both higher irradiation levels and more fatigue loading during operation. The effects of aging due to fatigue on the lower support columns for both IP2 and IP3 are adequately addressed through the RVI Program inspections, and therefore, no additional actions are required to address the effects of aging due to fatigue during the period of extended operation.

5. Instrument Columns: MRP-227-A identifies the bottom-mounted instrumentation (BMI) columns as an expansion item related to SCC and fatigue concerns in the control rod guide tube. The two degradation mechanisms produce similar cracking effects. The MRP-227-A inspection recommendation calls for a visual (VT-3) examination of the BMI columns "as indicated by difficulty of insertion/withdrawal of flux thimbles." The recommendation is based on the observation that the BMI columns are not part of the core support structure and loss of function would be evident during normal operation. Since the MRP-227-A inspection requirements indicate the consequences of failure would be noted during normal operation and the expert panel has determined that the control rod guide tube assemblies are adequate leading indicators for cracking mechanisms in the instrument columns no additional actions are required. The effects of aging due to fatigue on the instrument columns for IP3 are adequately managed by the RVI Program inspections during the period of extended operation.

6. Upper Support Assembly (Top Hat, Perforated Plate, Skirt): Entergy has identified that the MRP-227-A requirement for inspection of the upper support ring or skirt will be applied to the Top Hat structure in the Indian Point units. This examination will be conducted as part of the Inservice Inspection Program (ref. ASME Section XI BN-3). The effects of aging due to fatigue on the Upper Support Assembly – Perforated Plate and the Upper Support Assembly – Skirt for IP2 and the Top Hat Structure for IP3, are adequately addressed through the RVI Program inspections, and therefore, no additional actions are required to address the effects of aging due to fatigue during the period of extended operation.

7. Lower Core Plate: The RVI Program includes the lower core plate as an existing examination. This is clearly a core support structure that is accessible for visual examination. The examination is conducted as part of the Inservice Inspection Program (ASME Section XI, BN-3). The effects of aging due to fatigue on the Lower Core Support Plate for IP2 and the Lower Core Plate for IP3 are adequately addressed through the Inservice Inspection Program, and therefore, no additional actions are required to address the effects of aging due to fatigue during the period of extended operation.

8. Core Barrel Welds: The Indian Point 2 list of TLAA components includes multiple core barrel locations. The SE process for MRP-227-A promoted most of the core barrel to a status requiring Primary inspections. These MRP-227-A Primary inspections incorporate the effects of aging due to fatigue on the Mid Core Barrel, the Upper Core Barrel, and the Core Barrel Flange. The Core Barrel Outlet

Nozzle is an Expansion component linked to fatigue concerns in the Lower Core Barrel Flange Weld and the Upper and Lower Core Barrel Girth Welds. Fatigue concerns listed in Table 1 for IP2, are adequately addressed through the RVI Program inspections, and therefore, no additional actions are required to address the effects of aging due to fatigue in the core barrel components during the period of extended operation.

9. Lower Radial Keys: These components are inspected as part of the ASME Section XI BN-3 inservice inspection program. The effects of aging due to fatigue on the Lower Radial Key Base and the Lower Radial Key for IP2 are adequately addressed through the Inservice Inspection Program, and therefore, no additional actions are required to address the effects of aging due to fatigue during the period of extended operation.

10. Upper Core Barrel Flange: Although fatigue was not identified as a degradation mechanism for the upper flange weld in MRP-227-A, a Primary inspection is required for stress corrosion cracking (SCC). This program is adequate to identify cracking due to fatigue in the Upper Core Barrel Flange.

The RVI Program outlined in NL-12-037 [3] provides an integrated approach to managing the effects of aging due to the eight aging related degradation mechanisms that are potentially active in the core internals. Fatigue is one of three mechanisms that can cause cracking in the internals. The screening process used in the MRP-227 evaluation considered typical cumulative usage factors as indicators of potential susceptibility to fatigue failure. It is not therefore surprising that the list of components compiled by the MRP-227 expert panel included the IP2 and IP3 components with CUF analyses. The MRP-227-A list of components included in the Reactor Vessel Internals Program for fatigue related degradation includes all of the IP2 and IP3 components listed in Table 1.

In accordance with MRP-227-A, the RVI Program divides the RVI inspections into three inspection categories, Primary, Expansion and Existing. The intention of the RVI Program is to provide an integrated approach to managing the effects of aging in the reactor vessel internals including potential cracking due to fatigue.

Components placed in the Primary inspection category have been identified as lead items, where degradation, if any, is expected to appear first. There are three Primary inspections listed in Table 2 below (Upper and lower core barrel cylinder girth welds, core plate-to-lower support plate junction and thermal shield). Inspection of these components is required within two operating cycles of having entered into the period of extended operation. The periodicity of these Primary examinations is based on the established practice of conducting ASME Section XI in-service inspections on ten-year intervals. MRP-227-A states that:

"The intent is to provide sufficient flexibility for integration with ongoing inspection programs. Implementation of these recommendations will provide data on a broad spectrum of plants and conditions. The I&E Guidelines currently being developed by the U. S. industry are intended to be a living document. Data collected in the on-going industry program may provide the basis for adjustments to the inspection requirements and provide a definitive basis for the inspection interval."

A similar statement about the inspection interval is contained in the response to RAI-1 in the MRP-227 SE.

Components with less severe conditions, where manifestation of the degradation is expected to take more time have been placed in the Expansion category. Inspection of Expansion components is only required if degradation in the Primary components is observed. There are four expansion inspection locations (core barrel nozzle, upper core plate, lower support columns, and instrument columns) listed in Table 2. For example, the core barrel nozzle inspection would be triggered by observation of cracking in other core barrel welds. The re-inspection periodicity for the expansion items was set by the NRC in the MRP-227-A SE:

"*The staff has concluded that the NRC-approved version of MRP-227 shall specify a baseline periodicity of subsequent re-examination for all "Expansion" inspection category components. A baseline 10-year interval between examinations of "Expansion" inspection category components is required once degradation is identified in the associated "Primary" inspection category component.*"

When there was a potential aging related degradation mechanism that was adequately managed in the scope of the current ASME Section XI BN-3 examinations or other established programs, the components were placed in the Existing category. The Existing examinations listed in Table 2 are all related to examinations of the lower core plate and upper support structure. In both cases, a broader VT-3 examination to identify damage to the structure was deemed appropriate. This exam is already required as part of the ASME Section XI BN-3 examinations for all accessible core support structures. Based on this understanding, MRP-227-A simply requires that the regions of concern are included in the Inservice Inspection Program. The periodicity for these exams is specified by ASME Section XI.

The comparison clearly demonstrates that the RVI Program inspections based on MRP-227-A, in conjunction with the Inservice Inspection Program adequately manage the effects of aging due to fatigue on the reactor vessel internals for Indian Point Units 2 and 3.

Table 2
MRP-227-A Inspection Locations for Fatigue Susceptible Components

Assembly	Component	Identified TLAA Component	Inspection Type and Timing
MRP-227 Primary Inspections			
Control Rod Guide Tube Assembly	Lower Flange Welds	None	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.
Core Barrel Assembly	Upper and lower core barrel cylinder girth welds	Mid Core Barrel (IP2) Upper Core Barrel (IP2)	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.
Core Barrel Assembly	Lower core barrel flange weld (At IPEC this weld is the lower core barrel to lower support casting weld. IPEC does not have a lower core barrel flange)	Core-Barrel-to-Lower-Support-Plate Junction (IP3)	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.
Baffle-Former Assembly	Baffle-edge bolts	None	Visual (VT-3) examination, with baseline examination between 20 and 40 effective full power years(EFPY) and subsequent examinations on a ten-year interval.
Baffle-Former Assembly	Baffle-former bolts	None	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a ten-year interval.

Table 2
MRP-227-A Inspection Locations for Fatigue Susceptible Components

Assembly	Component	Identified TLAA Component	Inspection Type and Timing
Thermal Shield Assembly	Thermal shield flexures	Thermal Shield (IP3)	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten year interval.
MRP-227 Expansion Inspection			
Upper Internals	Upper Core Plate	Upper Core Plate (IP3)	Expansion from CRGT lower flange weld. Enhanced visual (EVT-1) examination Re-inspection every 10 years following initial inspection.
Core Barrel Assembly	Barrel-former bolts	None	Expansion from baffle-former bolts. Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.
Lower Support Assembly	Lower support column bolts	Lower Support Columns (IP2 & IP3)	Expansion from baffle-former bolts. Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.
Core Barrel Assembly	Core barrel outlet nozzle welds	Core Barrel Outlet Nozzle (IP2)	Expansion from upper core barrel flange weld. Enhanced visual (EVT-1) examination Re-inspection every 10 years following initial inspection.
Bottom-mounted Instrumentation System	Bottom-mounted instrumentation (BMI) column bodies	Instrument Columns (IP3)	Expansion from Control Rod Guide Tube (CRGT) lower flanges. Visual (VT-3) examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles. Re-inspection every 10 years following initial inspection. Flux thimble insertion/withdrawal to

Table 2
MRP-227-A Inspection Locations for Fatigue Susceptible Components

Assembly	Component	Identified TLAA Component	Inspection Type and Timing
			be monitored at each inspection interval.
MRP-227 Existing Programs (ASME Section XI BN-3)			
Upper Internals Assembly	Upper support ring or skirt (This item is N/A because IPEC has a tophat design, therefore there is no support ring or skirt, however the vertical sections of the tophat will be inspected)	Upper Support Assembly: Perforated Plate (IP2) Skirt (IP2) Top Hat Structure (IP3)	ASME Code Section XI. Visual (VT-3) examination.
Lower Internals Assembly	Lower core plate	Lower Core Support Plate (IP2) Lower Core Plate (IP3)	ASME Code Section XI. Visual (VT-3) examination of the lower core plates to detect evidence of distortion and/or loss of bolt integrity.
Fatigue Degradation not Identified in MRP-227A			
	None	Core Barrel Flange (IP2)	Upper core barrel flange is a Primary component for SCC. Requires EVT-1 exam.
	None	Lower Radial Key Base (IP2)	ASME Code Section XI. Visual (VT-3)
	None	Lower Radial Key (IP2)	ASME Code Section XI. Visual (VT-3)

2. As discussed above in item 1, the RVI components with existing CUFs provided in the License Renewal Application (LRA) are included in the RVI Program based on MRP-227-A or under the Inservice Inspection Program mandated by the ASME Section XI Code. Since fatigue cracking was one of the degradation mechanisms considered in the development of these inspection requirements, no additional actions are required to manage the effects of aging due to fatigue for the reactor vessel internals. The effects of aging due to fatigue are managed in accordance with 10 CFR 54.21(c)(1)(iii) throughout the PEO.

3. Indian Point reactor vessel internals were designed prior to the release of ASME Section III, Subsection NG design requirements. As a result, no explicit fatigue evaluations were required. Subsequent plant uprate evaluations for Indian Point determined CUFs for some reactor vessel internal components. These evaluations were performed to the intent of ASME Section III, Subsection NG requirements. As discussed in the response to item 1 of this RAI, the effects of fatigue are managed through a combination of the MRP-227-A and the ASME Section XI Inservice Inspection Programs for all the reactor vessel internal components with existing CUFs listed in the IPEC LRA through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii). No additional

evaluations are required to include the environmental effects of the reactor coolant system environment.

4. A new paragraph, below, is added to UFSAR Supplement Sections A.2.2.2.1 and A.3.2.2.1 indicating that the Reactor Vessel Internals and Inservice Inspection Programs manage the effects of aging due to fatigue on the reactor vessel internals components with an associated fatigue TLAA. No new commitment is necessary since the UFSAR Supplement Sections A.2.1.41 and A.3.1.41 delineate the Reactor Vessel Internals Program and the Inservice Inspection Program is an established program described in UFSAR Supplement Sections A.2.1.17 and A.3.1.17.

   The following changes (identified by underline) are made to LRA Section A.2.2.2.

   A.2.2.2 Metal Fatigue

   A.2.2.2.1 <u>/ A.3.2.2.1</u> 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.

   The Fatigue Monitoring Program will assure that the analyzed number of transient cycles is not exceeded. The program requires corrective action if the analyzed number of transient cycles is approached. Consequently, the effects of aging related to these TLAA (fatigue analyses) based on those transients will be managed by the Fatigue Monitoring Program in accordance with 10 CFR 54.21(c)(1)(iii).

   <u>As indicated in EPRI MRP-227-A, the effects of aging due to fatigue were considered in determining the necessary inspections for reactor vessel internals components. Consistent with MRP-227-A, during the period of extended operation, component inspections performed under the Reactor Vessel Internals Program and the Inservice Inspection Program will manage the effects of aging due to fatigue of reactor vessel internals components in accordance with 10 CFR 54.21(c)(1)(iii).</u>

<u>References</u>

1. Westinghouse Report, WCAP-16156-P, Rev. 1, "Indian Point Nuclear Generating Unit No. 2, Stretch Power Uprate, NSSS Engineering Report," March 2004.

2. Westinghouse Report, WCAP-16211-P, Rev. 0 "Power Uprate Project, Indian Point Unit 3 Power Plant, NSSS Engineering Report," June 2004.

3. Entergy Letter, NL-12-037, "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," February 17, 2012.