ENCLOSURE 2 TO NL-14-010

LTR-RIAM-13-117, Revision 0, Attachment 2, "Final Response to U.S. NRC RAI 6-A Items 1 and 2 on the RVI Program and RVI Inspection Plan for Indian Point Units 2 and 3 (Non-Proprietary)

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

Attachment 2: Final Response to U.S. NRC RAI 6-A Items 1 and 2 on the RVI Program and RVI Inspection Plan for Indian Point Units 2 and 3 (Non-Proprietary)

Final Response to U.S. NRC RAI 6-A Items 1 and 2 on the RVI Program and RVI Inspection Plan for Indian Point Units 2 and 3 (Non-Proprietary)

RAI 6-A Item 1:

Do the IP2 or IP3 reactor vessel internals (RVI) have non-weld or bolting austenitic stainless steel components with 20 percent cold work or greater, and if so, do the affected components have operating stresses greater than 30 ksi? If so, perform a plant-specific evaluation to determine the aging management requirements for the affected components.

RAI 6-A Item 1 Response:

The EPRI guidance document for MRP-227-A applicability (MRP 2013-025) was followed for this evaluation [3]. Indian Point Unit 2 and Unit 3 have evaluated reactor internals components according to the MRP-191 [1] industry generic component listings and screening criteria (including consideration of cold work as defined in MRP-175 [2], noting the requirements of Section 3.2.3). In addition to consideration of the material fabrication, forming, and finishing process, a general screening definition of a resulting reduction in wall thickness of 20% was applied as an evaluation limit. It was confirmed that all of the Indian Point Unit 2 and Unit 3 components, as applicable for the design, are included directly in the MRP-191 component lists, except for the components identified in Table 1.

The evaluation included a review of all plant modifications affecting reactor internals and the plant operating history. The components were procured according to ASTM International or ASME material specifications through applicable quality controlled protocols. The evaluation performed concluded that the reactor internals Category 1, 2, and 3 (non-bolting) components at Indian Point Unit 2 and Unit 3 contain no cold work greater than 20% as a result of construction. Category 4 components were already assumed to have the potential for cold work in the generic assessments and no category 5 components were identified. The evaluation therefore concluded that there was no impact to the MRP-227-A sampling inspection aging management requirements based on the detailed evaluation for Applicant/Licensee Action Item 1.

Table 1: Component Categories for MRP-191 Material Differences

MRP-227 Component	Material (Form/Fabrication)	Category(1)	Cold-worked 20% Assessment(2)	Comments
Mixing Devices	[ ]a,c Type 304 SS; [ ]a,c Grade CF8	3 1	N	
Fuel Alignment Pins	[ ]a,c Type 304 SS	3	N	
Brackets, Clamps, Terminal Blocks, and Conduit Straps	[ ]a,c Grade CF8	1	N	
Locking Caps	[ ]a,c Type 304L SS	3	N	
BMI Column Cruciforms	[ ]a,c Type 304A SS [ ]a,c Type 304 SS	3	N	
Flux Thimble Tube Plugs	[ ]a,c AISI 308	2	N	
Fuel Alignment Pins	[ ]a,c Type 304 SS	3	N	
Lower Support Column Bolts	[ ]a,c Type 316 SS	4	Y	
Thermal Shield Dowels	[ ]a,c Type 304 SS	3	N	
Radial Support Key Bolts	[ ]a,c	4	Y	
Lock Keys	[ ]a,c Type 304 SS	3	N	

Notes:

(3) Categories include the following:
- CASS (Category 1)
- hot-formed austenitic stainless steel (Category 2)
- annealed austenitic stainless steel (Category 3)
- fasteners austenitic stainless steel (Category 4)
- cold-formed austenitic stainless steel without subsequent solution annealing (Category 5)

(4) Cold work potential based on MRP-227-A generic criteria:
- N applies to categories 1, 2, and 3.
- Y applies to categories 4 and 5.

RAI 6-A Item 2:

Has IP2 or IP3 ever utilized atypical fuel design or fuel management that could make the assumptions of MRP-227-A regarding core loading/core design non-representative for that unit, including power changes/uprates? If so, describe how the differences were reconciled with the assumptions of MRP-227-A, or provide a plant-specific aging management program for affected components as appropriate.

RAI 6-A Item 2 Response:

Neither Indian Point Unit 2 nor Indian Point Unit 3 has ever utilized atypical fuel design or fuel management that could make the assumptions of MRP-227-A regarding core loading/core design non-representative for that unit, including power changes/uprates that have occurred over the operating lifetime of both units. This conclusion is based on comparisons of the Indian Point Units 2 and 3 core geometry and operating characteristics with the MRP-227-A applicability guidelines for Westinghouse-designed reactors specified in [3].

Specifically, the following comparisons with the MRP-227-A applicability guidelines were established for the key reactor internals components at Indian Point Units 2 and 3:

Components Located Beyond the Outer Radius of the Reactor Core

Guideline 1 - The reactor has been operated with out-in fuel management for thirty effective full-power years or less and all future operation will use low-leakage fuel management.

Comparison - Indian Point Unit 2 initiated low-leakage fuel management strategy in the sixth fuel cycle following 5.2 effective full-power years of operation and has been implementing low-leakage core designs since that time. There are no current plans to return to out-in fuel management.

Indian Point Unit 3 initiated low-leakage fuel management strategy in the fourth fuel cycle following 3.2 effective full-power years of operation and has been implementing low-leakage core designs since that time. There are no current plans to return to out-in fuel management.

Guideline 2 - For operation going forward the average power density of the reactor core (as defined in [3]) shall not exceed 124 W/cm^3.

Comparison - For the last four operating fuel cycles (Cycles 18 through 21), Indian Point Unit 2 has been operating at a rated power level of [ ]^{a,c}. For the [ ]^{a,c} fuel assembly Indian Point Unit 2 core geometry, the [ ]^{a,c} power level corresponds to a core power density of [ ]^{a,c}. This level of power generation is also representative of anticipated future operation.

For the last four operating fuel cycles (Cycles 15 through 18) Indian Point Unit 3 has been operating at a rated power level of [ ]^{a,c}. For the [ ]^{a,c} fuel assembly Indian Point Unit 3 core geometry, the [ ]^{a,c} power level corresponds to a core power density of [ ]^{a,c}. This level of power generation is also representative of anticipated future operation.

Guideline 3 - For operation going forward, the nuclear heat generation rate figure of merit (HGR-FOM) (as defined in [3]) shall not exceed 68 W/cm^3.

Comparison - For the last four operating fuel cycles at Indian Point Unit 2, the HGR-FOM at key baffle locations has ranged between [ ]a,c. This range of HGR-FOM is representative of anticipated future operation.

For the last four operating fuel cycles at Indian Point Unit 3, the HGR-FOM at key baffle locations has ranged between [ ]a,c. This range of HGR-FOM is representative of anticipated future operation.

Components Located Above the Reactor Core

Guideline 1 - Considering the entire operating lifetime of the reactor, the average power density of the core (as defined in [3]) shall not exceed 124 W/cm^3 for a period of more than two effective full-power years.

Comparison - Over the operating lifetime of the Indian Point Unit 2 reactor, the rated core power level, including power uprates, has varied between [ ]a,c. This variation of rated power level corresponds to a power density range of [ ]a,c.

Over the operating lifetime of the Indian Point Unit 3 reactor, the rated core power level, including power uprates, has varied between [ ]a,c. This variation of rated power level corresponds to a power density range of [ ]a,c.

Guideline 2 - Considering the entire operating lifetime of the reactor, the distance between the top of the active fuel stack and the bottom of the upper core plate (UCP) shall not be less than 12.2 inches for a period of more than two effective full-power years.

Comparison - For the Indian Point Unit 2 reactor internals and fuel assembly geometry, the nominal distance between the top of the active fuel stack and the bottom of the upper core plate (UCP) averaged over the first 21 fuel cycles of operation was [ ]a,c. During that period of time the nominal distance between the UCP and the top of the active fuel was not less than 12.2 inches.

For the Indian Point Unit 3 reactor internals and fuel assembly geometry, the nominal distance between the top of the active fuel stack and the bottom of the upper core plate (UCP) averaged over the first 18 fuel cycles of operation was [ ]a,c. During that period of time the nominal distance between the UCP and the top of the active fuel was not less than 12.2 inches.

Components Located Below the Reactor Core

Based on the discussion provided in [3], plant-specific applicability of MRP-227-A for components located below the reactor core with no further evaluation required is demonstrated by meeting the MRP-227-A, Section 2.4 criteria.

References

1. *Materials Reliability Program: Screening Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191).* EPRI, Palo Alto, CA: 2006. 1013234.
2. *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175).* EPRI, Palo Alto, CA: 2005, 1012081.
3. Materials Reliability Program, MRP 2013-025, "MRP-227-A Applicability Template Guideline," October 14, 2013