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W3F1-2015-0038

June 18, 2015

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: Waterford Steam Electric Station, Unit 3 Response to Request for Additional Information Regarding Reactor Vessel Internals Aging Management Program
Waterford Steam Electric Station, Unit 3 (Waterford 3)
Docket No. 50-382
License No. NPF-38

References:

1. Entergy Letter W3F1-2013-0070, Submittal of Reactor Vessel Internals Aging Management Program Consistent with MRP-227-A, dated December 16, 2013. (ADAMS Accession No. ML13352A041)
2. Letter from NRC, Request for Additional Information Regarding the Reactor Vessel Internals Aging Management Program (TAC No. MF3247), dated October 21, 2014. (ADAMS Accession No. ML14232A023)
3. Letter from NRC, Waterford Steam Electric Station, Unit 3 – Request for Additional Information Regarding the Reactor Vessel Internals Aging Management Program (TAC NO. MF3247), dated April 22, 2015, (ADAMS Accession No. ML15107A411)

Dear Sir or Madam:

In letter dated December 16, 2013 (Reference 1), Entergy Operations, Inc. (Entergy) submitted a request for the NRC to review and Waterford 3's Reactor Vessel Aging Management Program (AMP) developed to implement MRP-227-A, Rev 0.

In letter dated October 21, 2014 (Reference 2), NRC requested Entergy to provide additional information to support review of the Reactor Vessel Internals Aging Management Program. This letter provides the response to RAI 2 in Attachment 2, 3, and 4, and RAI 4 in Attachment 5.

In letter dated April 22, 2015 (Reference 3), NRC requested Entergy to provide additional information to support review of the Reactor Vessel Internals Aging Management Program. This letter provides the response to RAI 5 Attachment 1.

**Attachment 3 to the letter contains proprietary information – Attachment 3 is to be withheld from public disclosure per 10 CFR 2.390.
When separated from the attachment this document is decontrolled.**

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This correspondence contains no commitments. If you have any questions or require additional information, please contact the Regulatory Assurance Manager, John Jarrell, at 504-739-6685.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 18, 2015.

Sincerely,

MRC/SWM

Attachments:

1. RESPONSE SUMMARY OF RAI 2 AND 4 AND RESPONSE TO RAI 5
2. PWROG-15041 Appendix A (TAC No.MF3247) RAI 2A *NON – PROPRIETARY*
3. PWROG-15039-P (TAC No.MF3247) RAI 2B *PROPRIETARY*
4. PWROG-15039-NP (TAC No.MF3247) RAI 2B *NON – PROPRIETARY*
5. PWROG-15054-NP (TAC No.MF3247) RAI 4 *NON – PROPRIETARY*
6. Affidavit

cc: Mr. Marc L. Dapas, Regional Administrator
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Attachment 1 to

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RESPONSE SUMMARY RAI 2 AND 4 AND RESPONSE TO RAI 5

RESPONSE SUMMARY OF RAI 2 AND 4 AND RESPONSE TO RAI 5

RAI-2

MRP-2013-025, "MRP-227-A Applicability Template Guidelines," dated October 14, 2013 (ADAMS Accession No. ML 13322A454), was developed by Westinghouse Electric Company LLC (Westinghouse) as a part of its response to the NRC staff's Action Item 1 of the safety evaluation for the MRP-227 -A report. In MRP-2013-025, two issues were addressed (A) Cold Worked materials which is addressed in Appendix A of the report; and (B) Fuel Design or Fuel Management which is addressed in Appendix B of the report. With regard to the above, the NRC staff requests the following information:

A. Please clarify if the WF3 RVI components have non-weld or bolting austenitic stainless steel components with 20 percent cold work or greater, and if so, whether the affected components have operating stresses greater than 30 ksi. In particular, please provide the plant-specific information on the extent of cold work on its RVI components. The licensee can apply "Option 1" or "Option 2," as addressed in Appendix A of the report. If "Option 2" is applicable to WF3, the licensee should list plant-specific RVI components that have been exposed to cold work equal to or greater than 20 percent. Plant-specific information related to this issue as addressed in "Option 2" in Appendix A, should be provided.

B. Please explain if WF3 has ever utilized atypical design or fuel management that could make the assumptions of MRP-227-A regarding core loading/core design non-representative for that plant, including power changes/uprates such as the extended power uprate implemented in 2005. The following guidelines provided by MRP should be followed. The licensee is requested to use the MRP document dated October 14, 2013, MRP-2013-025, and it can apply "Option 1" or "Option 2," as addressed in Appendix B of the report.

   <u>Option 1</u>
   WF3 complies with the MRP-227-A assumptions regarding core loading/core design. Neutron fluence and heat generation rates are concluded to be Option A or Option B.
   - Option A: acceptable based on the following assessment to the limiting MRP guidance threshold values.
   - Option B: unacceptable based on an assessment to the limiting MRP guidance threshold values.

   If Option A as addressed under "Option 1" is applicable, the following plant specific values should be submitted: (a) active fuel to fuel alignment plate distance; (b) average core power density; and, (c) heat generation figure of merit.
   If Option B under "Option 1" is applicable to WF3, the licensee should justify the usage of its fuel management program.

   <u>Option 2</u>
   The licensee should provide a technical justification for the application of MRP-227 -A criterion to WF3.

RAI-2A

Please clarify if the WF3 RVI components have non-weld or bolting austenitic stainless steel components with 20 percent cold work or greater, and if so, whether the affected components have operating stresses greater than 30 ksi. In particular, please provide the plant-specific information on the extent of cold work on its RVI components. The licensee can apply "Option 1"

or "Option 2," as addressed in Appendix A of the report. If "Option 2" is applicable to WF3, the licensee should list plant-specific RVI components that have been exposed to cold work equal to or greater than 20 percent. Plant-specific information related to this issue as addressed in "Option 2" in Appendix A, should be provided.

RAI 2A Response:
Westinghouse evaluated and provided response to RAI 2A. See Attachment 2.

RAI-2B
Please explain if WF3 has ever utilized atypical design or fuel management that could make the assumptions of MRP-227-A regarding core loading/core design non-representative for that plant, including power changes/uprates such as the extended power uprate implemented in 2005. The following guidelines provided by MRP should be followed. The licensee is requested to use the MRP document dated October 14, 2013, MRP-2013-025, and it can apply "Option 1" or "Option 2," as addressed in Appendix B of the report.

RAI 2B Response:
Westinghouse evaluated and provided response to RAI 2B. See Attachment 3 and 4.

RAI-4
As discussed in Section 3.3.7 of Revision 1 to the safety evaluation for MRP-227 dated December 16, 2011 (ADAMS Accession No. ML 11308A770), Action Item 7 requires that the licensees of Westinghouse reactors develop plant-specific analyses to be applied for their facilities to demonstrate that lower support column cast austenitic stainless steel, (CASS) bodies will maintain their function during the extended period of operation. However, licensees are observing degradation in CASS bodies during the current operating license. MRP-227 –A Table 3-2 (Final disposition of Combustion Engineering internals) classifies CASS lower support columns as Primary Components based on susceptibility to irradiation embrittlement (IE) and irradiation assisted stress-corrosion cracking and thermal embrittlement (TE). After further review of the existing literature data for the threshold limits for IE and TE of CASS materials, the NRC staff developed a new position for screening of CASS materials for combined IE and TE. The bases for the staff's new threshold limits are described in the document "NRC position on Management of CASS Reactor Vessel Internal Components," at ADAMS Accession No. ML 14163A112.
To enable an assessment of the adequacy of aging management for the lower support columns in response to TE and IE, the NRC staff requests that the licensee address Action Item 7 of the December 16, 2011, safety evaluation for Revision 1 of MRP-227 -A under the current operating license.

RAI 4 Response:
Westinghouse evaluated and provided response to RAI 4. See Attachment 5.

RAI-5
Section 7.3 of the December 26, 2013 submittal referenced a plant-specific in-core instrumentation thimble tube program report. The staff requests that the licensee provide a brief summary of the plant-specific in-core instrumentation thimble tube program report. The summary should include: (a) the aging degradation; (b) licensee's inspection methods of identifying aging effects; (c) the inspection results

Response to RAI-5

Brief Summary of Plant-Specific In-Core Instrumentation Thimble Tube Program Report: Greater than expected, In-Core Instrumentation (ICI) Thimble growth problem was discovered at several plants in the industry including Waterford 3. ICI thimble growth was caused by high level of neutron radiation exposure experienced by the zircaloy section of the ICI Thimbles due to extensive time in the reactor core.

The long term solution of the ICI thimble growth was to replace the ICI thimbles with a shorter ICI Thimble design that are also subject to growth due to neutron radiation exposure.
In-core instrumentation thimble tubes were replaced at Waterford 3 during RF16 to address the elongation of ICI thimble tubes by neutron irradiation in the reactor core. The report provided recommendations for periodic measurement of the replacement ICI thimbles to monitor the future growth and determine if the thimbles will require replacement in future.
In discussion with the OEM Vendor, it was further explained that Thimbles grow on an exponential curve. At this time only 6 years removed from Thimble replacement the thimbles have not grown enough to be measured. There were 9 ICI Thimbles measured during RF16 in 2009 for a baseline. Those same 9 locations will be measured during RF24 to quantify Thimble growth.

Following the replacement of ICI thimble tubes, OEM Vendor issued a letter to Entergy on June 4, 2012 that recommended inspection of the thimbles for wear during the next refueling outage. It was decided to evaluate ICI thimble tube inspection results from ANO fall 2012 outage to determine applicability to WF3. The results of the evaluation determined results from ANO-2's ICI thimble inspection showed little to no wear on the tubes.
Based on similarities in ANO-2's and Waterford 3's thimble design and the recent replacement of the thimbles, it was not necessary to add this inspection to the RF18 outage scope.
The visual Non Destructive exam will be performed by Westinghouse in RF20 accordance with an approved procedure using an underwater zoom camera with lighting. A video recording will be made of the inspections. Thimbles will be viewed from the periphery of the Upper Guide Structure Lift Rig (UGSLR) at the elevation of the thimble opposite the wear sleeve in the fuel assembly. All thimbles will be viewed from a minimum of one (1) side. Approximately 75% will be viewed from two (2) or three (3) sides.

Table 1 lists the licensee response to the following parts of RAI 5:
5(a) aging degradation
5(b) licensee's inspection methods of identifying aging effects of the ICI thimble tubes
5(c) inspection results

Table 1: In Core Instrumentation		
Component Type	**Aging Degradation**	**Inspection Method of identifying aging effects**
In Core Instrumentation thimble tubes	Radiation induced growth of ICI tube	Visual Non Destructive Examination
Inspection Results: A project letter report summarizing the inspection results will be provided by Westinghouse within thirty (30) days of the inspection.		

Attachment 2 to

W3F1-2015-0038

PWROG-15041 Appendix A

APPENDIX A

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As a result of the review of the Applicant/Licensee Action Items (A/LAIs) 1, 2, and 7 responses submitted by the industry, the Nuclear Regulatory Commission (NRC) has requested that additional information regarding cold work in the Reactor Vessel Internals (RVI) be provided to support A/LAI 1 plant-specific demonstration of MRP-227-A [1] applicability. The issue of cold work in stainless steel relates to the criteria in MRP-175 [2] for stress corrosion cracking (SCC). The specific NRC question is focused on whether the materials of original construction for the domestic fleet RVI components contain "significant" cold work (greater than 20 percent), and if so, if the component is subjected to stresses greater than 30 ksi. Component conformance with the cold work 20 percent criterion is completed for this evaluation; specific stress levels are not addressed in this evaluation. A guideline template (MRP 2013-025 [3]) was completed by Westinghouse and the Electric Power Research Institute (EPRI) to define a process for evaluating cold work in the RVI component materials. For the purpose of this evaluation, it is noted that the assessments are based on the screening and binning process, which is based exclusively on material specifications. This assessment did not specifically investigate any other avenues, such as field installation. Field fit up and auxiliary processes that could introduce cold work are expected to provide only minor increase of cold work and minor increases of potential susceptibility to SCC [3].

In Request for Additional Information (RAI) Part 2, Item A [4] for Waterford Steam and Electric Station Unit 3 (WF3), the U.S. NRC specifically requested that WF3 addresses the following:

RAI-2, Item A:

"Please clarify if the WF3 RVI components have non-weld or bolting austenitic stainless steel components with 20 percent cold work or greater, and if so, whether the affected components have operating stresses greater than 30 ksi. In particular, please provide the plant-specific information on the extent of cold work on its RVI components. The licensee can apply "Option 1" or "Option 2", as addressed in Appendix A of the report. If "Option 2" is applicable to WF3, the licensee should list plant-specific RVI components that have been exposed to cold work equal to or greater than 20 percent. Plant-specific information related to this issue as addressed in "Option 2" in Appendix A, should be provided."

The MRP-227-A [1] Applicability Template Guideline, as summarized in MRP 2013-025 [3], is followed to support this assessment and response to the NRC.

Waterford Steam and Electric Station Unit 3 – Reactor Internals Cold Work Assessment

Westinghouse has evaluated the WF3 reactor internals components according to industry guideline MRP 2013-025 [3] and the MRP-191 [5] industry generic component listings and screening criteria (including the consideration of cold work as defined in MRP-175 [2], noting the requirements of subsection 3.2.3). In addition to consideration of the material fabrication, forming, and finishing process, a general screening definition of "significant cold work", as that which results in a 20 percent reduction in wall

thickness, was applied as an evaluation limit. It was confirmed that all WF3 components, as applicable for design, are either included in the MRP-191 [5] component lists or have been evaluated accordingly. The evaluation included a review of all plant modifications affecting reactor internals and the plant operating history. The components were procured according to American Society for Testing and Materials (ASTM) International or American Society of Mechanical Engineers (ASME) material specifications through applicable quality controlled protocols that were called out in the original plant construction drawings. Thus, material identification based on the material call-outs and notes in the component drawings was an efficient and reasonable approach to identify the material of construction of components at WF3.

Based on the specifications called out on the WF3 component drawings, the RVI components are binned into the five material categories identified in MRP 2013-025 [3].

The categories based on MRP 2013-025 [3] include:

- Cast austenitic stainless steel (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)

The potential for cold work is directly controlled by the materials specifications. Essentially all of the components that are binned (based on their specified materials) as Categories 1, 2, and 3 are non-cold worked; therefore, they have less than 20 percent cold work according to the NRC criterion. Similarly, any component binned under Category 5 has the potential to contain greater than 20 percent cold work. Category 4 materials are fasteners that may have been intentionally strain-hardened.

During the fabrication of fasteners, the strain hardening was typically intentionally restricted to less than 20 percent. These restrictions, if present, were noted on engineering drawings. A restriction or limitation on the material yield stress (e.g., a maximum of 90 ksi) would indicate that the material cold work would be limited to be less than 20 percent. In the absence of a restriction on the maximum yield stress of strain-hardened material, a conservative approach has been taken to assume the potential for greater than 20 percent cold work.

Where multiple options existed for a component or assembly, the bounding condition was taken as the option that had the greater potential to include greater than 20 percent cold work. This option was then employed in the assessment of the component, and was selected for the purposes of the assessment. In some instances sequential fabrication would appear to mitigate any potential for cold work. However, since the historical record was not detailed, the potential is noted and a conservative approach was selected for this assessment.

The evaluation, performed consistently with MRP 2013-025 [3] guidelines, concluded that the reactor internals Categories 1, 2, and 3 (non-bolting) components at WF3 contain no cold work greater than 20 percent as a result of material specification and controlled fabrication construction. Category 4 components were already assumed to have the potential for cold work in the MRP-191 [5] generic assessments. Material fabrication specifications used for WF3 suggest that processes were limiting, which precluded the introduction of significant cold work; thus, no Category 4 and Category 5 components with significant cold work were identified for WF3. The detailed evaluation for the A/LAI for the WF3 cold work assessments concluded that the plant-specific material fabrication and design were consistent with the MRP-191 [5] basis and that the MRP-227-A [1] sampling inspection aging management requirements, as related to cold work, are directly applicable to WF3.

References

1. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)*. EPRI, Palo Alto, CA: 2011. 1022863.

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

3. EPRI Letter, MRP 2013-025, "MRP-227-A Applicability Template Guideline," October 14, 2013.

4. U.S. NRC RAI, "Waterford Steam Electric Station, Unit 3 - Request for Additional Information Regarding the Reactor Vessel Internals Aging Management Program (TAC NO. MF3247)," October 21, 2014. (Accession No. ML14232A023).

5. *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.