UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 6, 2015

Site Vice President
Entergy Operations, Inc.
Waterford Steam Electric Station, Unit 3
17265 River Road
Killona, LA 70057-3093

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 – SAFETY EVALUATION REGARDING THE AGING MANAGEMENT PROGRAM FOR REACTOR VESSEL INTERNALS (TAC NO. MF3247)

Dear Sir or Madam:

By letter dated December 16, 2013, as supplemented by letters dated January 19, June 18, July 9, 2015, Entergy Operations, Inc. (the licensee), submitted an aging management program (AMP) for the reactor vessel internals (RVI) at Waterford Steam Electric Station, Unit 3 (WF3). The licensee submitted the AMP to fulfill its regulatory commitment made in a letter dated February 27, 2012, stating that it would submit the AMP for the RVI components at WF3 for U.S. Nuclear Regulatory Commission (NRC) review and approval no later than December 16, 2013.

The Materials Reliability Program (MRP)-227-A, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," and its supporting reports were used as the technical bases for developing WF3's AMP for the RVI components. The NRC safety evaluation for MRP-227 was issued on December 16, 2011, and contained seven topical report conditions and eight applicant/licensee action items. The MRP-227-A is the NRC-approved version of MRP-227 that incorporates the NRC safety evaluation. The NRC staff reviewed the AMP for the RVI components and found that the licensee appropriately addressed all eight applicant/licensee action items, as specified in MRP-227-A for WF3.

The NRC staff's safety evaluation is enclosed.

If you have any questions, please contact me at 301-415-3229 or via e-mail at Michael.Orenak@nrc.gov.

Sincerely,

Michael D. Orenak, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

UNITED STATES
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SAFETY EVALUATION FOR THE AGING MANAGEMENT PROGRAM

FOR REACTOR VESSEL INTERNALS

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 INTRODUCTION AND BACKGROUND

By letter dated December 16, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13352A041), as supplemented by letters dated January 19, June 18, and July 9, 2015 (ADAMS Accession Nos. ML15019A026, ML15170A377, and ML15190A302, respectively), Entergy Operations, Inc. (the licensee), submitted an aging management program (AMP) for the reactor vessel internals (RVI) at Waterford Steam Electric Station, Unit 3 (WF3). The licensee submitted the AMP to fulfill its regulatory commitment made in a letter dated February 27, 2012 (ADAMS Accession No. ML12059A077) regarding the WF3 extended power uprate, stating that it would submit an AMP for the RVI components at WF3 for U.S. Nuclear Regulatory Commission (NRC) review and approval no later than December 16, 2013. The AMP was submitted for review under the current WF3 licensing basis.

The Materials Reliability Program (MRP)-227-A, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," and its supporting reports were used as the technical bases for developing WF3's AMP for the RVI components. The NRC safety evaluation (SE) for MRP-227 was issued on December 16, 2011 (ADAMS Accession No. ML11308A770), and contained seven topical report conditions and eight applicant/licensee action items. The licensee's December 16, 2013, submittal, as supplemented by letters dated January 19, June 18, and July 9, 2015, addressed the seven topical report conditions and the eight action items.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a, "Codes and standards," provides the materials specifications, controls on welding, and inspection of RVI components and core support structures.

Appendix A to 10 CFR Part 50, "General Design Criteria [GDC] for Nuclear Power Plants," states, in part that "[t]hese General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission." GDC 1, "Quality standards and records," states, in part that "[a] quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions."

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), Section 4.5.2, "Reactor Internal and Core Support Structure Materials," Revision 3, dated March 2007 (ADAMS Accession No. ML063190005) and Matrix 1 of Review Standard (RS)-001, "Review Standard for Extended Power Uprate," Revision 0, December 2003 (ADAMS Accession No. ML033640024) provide specific review criteria for the evaluation of the reactor vessel internals that will be affected during the extended power uprate.

The MRP-227, Revision 0, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," (ADAMS Accession No. ML090160204) contains a discussion of the technical basis for implementing inspection requirements for pressurized water reactor (PWR) RVI components that are subject to any of the applicable degradation mechanisms (e.g., stress corrosion cracking (SCC), intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking, wear, fatigue, thermal and/or neutron embrittlement, void swelling, and irradiation-enhanced stress relaxation) during the license renewal period. MRP-227 also provides a brief, high-level summary of flaw evaluation guidelines for RVI components that exhibit active degradation mechanisms, and establishes requirements for inspection of additional components if an active degradation mechanism is discovered (i.e., expansion of the scope of RVI component inspections). The NRC issued the SE for MRP-227 on December 16, 2011. The NRC-approved version of the topical report, MRP-227-A, was received by the NRC by letter dated January 9, 2012 (ADAMS Accession Nos. ML12017A193 through ML12017A197 and ML12017A199) and the NRC endorsed MRP-227-A by letter dated February 3, 2012 (ADAMS Accession No. ML120270374).

3.0 TECHNICAL EVALUATION

In the December 16, 2013, submittal, the licensee provided an AMP for the WF3 RVI components that is consistent with MRP-227-A. The NRC staff reviewed the seven technical sections of the AMP. Sections 1 through 5 of the AMP directly complied with MRP-227-A; therefore, the NRC staff finds these sections acceptable. The full NRC staff review of the sections is available in the SE for MRP-227, dated December 16, 2011.

The sections of the licensee's AMP that did contain WF3-specific information are the following:

(1) Section 6, "Operational Experience and Additional Considerations." The information for Section 6 was provided in Attachment A of the December 16, 2013 submittal;

(2) Section 7, "Responses to the NRC Safety Evaluation Report Applicant/Licensee Action Items;" and

(3) Materials susceptible to aging degradation, which are addressed in Appendix A of MRP-227-A.

The NRC staff's review of these three sections is provided below.

3.1 Evaluation of the Licensee's AMP related to Operating Experience

Appendix A of MRP-227-A addresses the AMP for RVI components that are susceptible to various aging degradation mechanisms. In Attachment A of the December 16, 2013, submittal, the licensee included its operating experience of the following RVI components: (a) in-core Instrumentation (ICI) thimbles; and (b) core barrel alignment key (clevis bolting).

In the supplement dated January 19, 2015, the licensee provided additional information that the RVI components, with the exception of ICI thimbles and clevis bolting, did not experience any aging degradation to date. The licensee stated that it implemented a design change, which included replacement of the existing thimble tubes with shorter thimble tubes that would accommodate any increase in the growth of the thimble due to neutron void swelling.

In the June 18, 2015, supplement, the licensee provided information on the AMP for the ICI thimble tubes, specifically: (a) the aging degradation; (b) licensee's inspection methods of identifying aging effects; (c) the inspection results; and, (d) frequency of subsequent inspections. The licensee stated that it would implement the following as recommended by the manufacturer:

(a) Periodic measurement of the replaced thimble tubes to monitor the growth of the thimble tubes.

(b) A visual inspection of the thimble tubes for wear, which would be accomplished from the periphery of upper guide structure lift rig. Visual inspection will be implemented for thimble tubes to monitor the aging degradation mechanisms due to void swelling and wear.

The NRC staff reviewed the supplement and determined that by using shorter thimble tubes, the contact between the thimble and the fuel guide tube is minimized. Periodic inspections of the replaced thimble tubes to monitor wear and void swelling ensure that the degradation of the thimble tubes is adequately managed by the licensee. The NRC staff concludes that the AMP adequately addresses the aging degradation due to neutron void swelling and wear in ICI thimble tubes at WF3.

In the January 19, 2015, supplement, the licensee stated that it implemented visual testing per the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, inspection requirements for monitoring the aging effects on clevis bolting. During the inspections for the previous 10-year interval, the alignment key was backed off by 3/16 inches from the back of the core barrel key way and the licensee's evaluation determined that this condition was acceptable as is. The core barrel was successfully installed after the refueling outage, demonstrating the accuracy of the licensee's evaluation conclusion. The NRC staff reviewed the AMP for the clevis insert assembly and found that routine ASME Code, Section XI, inspections of this component would identify the aging effect in a timely manner. Furthermore, since the licensee found no active aging degradation to date in the clevis insert, the NRC staff concludes that the existing AMP with routine inspections of this component would enable the licensee to effectively monitor future aging degradation. Based on the information provided, the NRC staff concludes that the AMP adequately manages the aging degradation of ICI thimble tubes and core barrel clevis insert key bolts.

3.2 Evaluation of the Licensee's Resolution of Action Items through 1 through 8 Addressed in the SE for MRP-227-A

3.2.1 Applicant/Licensee Action Item 1 of the SE for MRP-227

Section 4.2.1 of the NRC's SE for MRP-227 states that:

> Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the FMECA [failure modes, effects and criticality analysis] and functionality analyses for reactors of their design (i.e., Westinghouse, CE [Combustion Engineering], or B&W [Babcock and Wilcox]) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227. This is Applicant/Licensee Action Item 1.

To resolve Action Item 1, a series of public and non-public meetings were conducted between the NRC, Westinghouse, the Electric Power Research Institute (EPRI), and utility representatives to discuss regulatory concerns and determine a path forward for a comprehensive and consistent utility response to demonstrate applicability of MRP-227-A, specifically for Westinghouse and CE-design RVI components. A summary of the proprietary meeting presentations and supporting proprietary generic design-basis information is contained in Westinghouse report WCAP-17780-P, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs" (not publicly available; proprietary information). WCAP-17780-P provides background proprietary design information regarding variances in stress, fluence, and temperature in the plants designed by CE to support NRC reviews of utility submittals to demonstrate plant-specific applicability of MRP-227-A. As a result of the technical discussions with the NRC staff, a technical basis was developed for the response to Action Item 1.

In a request for additional information (RAI-2) by letter dated October 21, 2014 (ADAMS Accession No. ML14232A023), the NRC staff requested, in part, that the licensee provide the following information related to verification of the applicability of MRP-227-A to WF3:

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 [thousand pounds per square inch]. In particular, please provide the plant-specific information on the extent of cold work on its RVI components.

B. 1) Please explain if WF3 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 plant, including power changes/uprates such as the extended power uprate implemented in 2005.

The EPRI MRP group reviewed these questions and performed a generic evaluation of the RVI components designed by Westinghouse and CE. By letter dated October 14, 2013, EPRI provided to licensees MRP-2013-025, "MRP-227-A Applicability Template Guideline" (ADAMS Accession No. ML13322A454), a non-proprietary document containing guidance for responding to the two above questions. Regarding Question 1, EPRI provides guidance for licensees to assess whether RVI components at their plant, other than those identified in the generic evaluation, have the potential for cold work greater than 20 percent. Because cold work in austenitic stainless steel material increases its susceptibility to SCC, Question 1 relates to the effectiveness of the AMP in monitoring SCC in the RVI components at WF3.

In Attachment 2 to Appendix A of the supplement dated June 18, 2015, the licensee provided additional information regarding WF3 RVI austenitic stainless steel non-weld or bolting components with 20 percent cold work or greater and operating stresses greater than 30 ksi. The licensee performed an evaluation using the MRP-2013-025 guidelines. Based on the evaluation, the licensee concluded that no cold work greater than 20 percent was performed on cast austenitic stainless steel (Category 1), hot-formed austenitic stainless steel (Category 2), and annealed stainless steel (Category 3) materials at WF3. Bolting materials (Category 4) were already assumed to undergo greater than 20 percent cold work as part of their fabrication. Similar to bolting materials, cold formed austenitic stainless steel without solution annealing (Category 5) are also susceptible to SCC. The licensee's review of material fabrication specifications at WF3 indicated that no significant cold work was performed on the most susceptible materials classified under Categories 4 and 5 materials. Furthermore, the licensee stated that its evaluation was consistent with MRP-191, Revision 0, "Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Designs (MRP-191)," November 2006 (ADAMS Accession No. ML091910130). MRP-191, Revision 0, was used as a technical basis in developing MRP-227-A.

The NRC staff reviewed the information in Attachment 2 of Appendix A of the licensee's supplement dated June 18, 2015. WF3 has no RVI components that were exposed to cold work of 20 percent or greater. Furthermore, the licensee stated that all the RVI component's welds were monitored for the presence of SCC. The licensee evaluated the effect of cold work on SCC in RVI components by using the technical bases provided in MRP-191, Revision 0. Since MRP-191, Revision 0, was used as a basis for MRP-227, the NRC staff finds that the licensee satisfied the MRP-2013-025 guidelines with respect to the evaluation of the effect of cold work on the occurrence of SCC in austenitic stainless steels at WF3.

The MRP-2013-025 guidelines provide quantitative criteria to allow a licensee to assess whether a particular plant has atypical fuel design or fuel management. For a CE design plant such as WF3, these criteria are:

(1) The heat generation rate must be less than or equal to (≤) 68 Watts/cm^3.
(2) The maximum average core power density must be less than 110 Watts/cm^3.
(3) The active fuel to bottom of the fuel alignment plate (FAP) distance must be greater than 12.4 inches.

In the June 18, 2015, supplement, the licensee provided the plant-specific values of the heat generation rate, the maximum average core power density, and the FAP distance. The NRC staff reviewed these values and determined that they comply with the values in MRP-2013-025. Therefore, the staff concludes that the licensee satisfied the guidelines related to fuel management issues addressed in MRP-2013-025.

3.2.2 Applicant/Licensee Action Item 2 of the SE for MRP-227

The licensee submitted their response to Action Item 2 in the June 18, 2015, supplement. Action Item 2 is only applicable to the period of extended operation. The NRC staff is not reviewing the submitted information for Action Item 2 in this SE. If necessary, the information will be reviewed for a future WF3 license renewal application.

3.2.3 Applicant/Licensee Action Item 3 of the SE for MRP-227

Action Item 3 in the SE for MRP-227 recommends that the licensees with the CE-designed RVI components perform a plant-specific evaluation for the thermal shield positioning pins and ICI thimble tubes. In the December 16, 2013, submittal, the licensee stated that WF3 does not have thermal shield components; therefore, the AMP does not include an inspection plan for thermal shield components. The NRC staff's assessment related to plant-specific evaluation of ICI thimble tubes are addressed in Section 3.1 in this SE.

3.2.4 Applicant/Licensee Action Items 4 and 6 of the SE for MRP-227

The licensee stated that Action Items 4 and 6 are not applicable to RVI components designed by CE. The NRC staff finds this response acceptable.

3.2.5 Applicant/Licensee Action Item 5 of the SE for MRP-227

The licensee submitted its response to Action Item 5 in the December 16, 2013, submittal. Action Item 5 is only applicable to the period of extended operation. The NRC staff is not reviewing the submitted information for Action Item 5 in this SE. If necessary, the information will be reviewed for a future WF3 license renewal application

3.2.6 Applicant/Licensee Action Item 7 of SE for MRP-227-A

Action Item 7 states, in part, that:

> the applicants/licensees of B&W, CE, and Westinghouse reactors are required to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W IMI [incore monitoring instrumentation] guide tube assembly spiders and CRGT [control rod guide tube] spacer castings, CE lower support columns, and Westinghouse lower support column bodies will maintain their functionality during the period of extended operation or for additional RVI components that may be fabricated from CASS [cast austenitic stainless steel], martensitic stainless steel or precipitation hardened stainless steel materials.

In the June 18, 2015, supplement, the licensee stated that the WF3 lower support columns were fabricated as wrought austenitic stainless steel, and not CASS materials, as described in the original December 16, 2013, submittal. Wrought austenitic stainless steel is not susceptible to thermal embrittlement (TE); therefore, the WF3 lower support columns are not susceptible to TE.

Irradiation embrittlement (IE) results in the cracking of RVI components. The intensity of the applied stress determines the severity of the cracking. Regarding IE in the wrought stainless steel lower support columns, the NRC staff notes that in Table 4-2 of MRP-227-A, the wrought stainless steel lower support columns welds are binned under the "Primary" inspection category and are to be inspected every 10 year interval per MRP-227-A guidelines. Therefore, the NRC staff finds that any cracking due to IE would be identified in the lower core support column welds in a timely manner to allow for proper aging management. Since the lower support columns are subject to similar IE degradation as lower support column welds, any emerging degradation identified in the lower support column welds would provide observable indications that would facilitate the licensee to effectively monitor any aging effects in lower support columns. Since the lower support column welds are part of the AMP that is consistent with the MRP-227-A guidelines, the NRC staff finds that the degradation of the wrought lower support columns is adequately managed at WF3. The NRC staff concludes that the licensee has satisfactorily addressed Action Item 7.

3.2.7 Applicant/Licensee Action Item 8 of the SE for MRP-227

The licensee stated in the December 16, 2013, submittal, that the information for Action Item 8 would be submitted at the time of the license renewal application; therefore, no information was provided for the NRC staff to review.

3.3 Materials Susceptible to Degradation

Historically, the following materials used in the PWR RVI components were known to be susceptible to some of the aging degradation mechanisms that are identified in Appendix A of MRP-227-A: 1) nickel base alloys, (2) precipitation hardened stainless steel materials, (3) alloy A-286, A453, and (4) type 431 stainless steel materials. In the supplement dated January 19, 2015, the licensee stated that the following materials were not used in the RVI components at WF3: (1) type 347 stainless steel; (2) type 431 stainless steel; (3) 17-4 or 15-5 precipitation hardened stainless steels; and alloy A-286 and ASTM A-453 Grade 660. The NRC staff reviewed the licensee's supplemental information and found that an AMP for these materials is not applicable to WF3.

In the supplement dated June 18, 2015, the licensee stated that nickel base alloys were used in the core support barrel assembly (snubber assembly bolts, shims and pins) and ICI thimble tube couplings. The core support barrel assembly is inspected under the ASME Code, Section XI, inservice inspection program that monitors aging degradation in the core support barrel assembly. Similarly, ICI thimbles are monitored under the "Existing" program per MRP-227. The NRC staff reviewed the information regarding the use of nickel base alloys and concludes that the current routine inspections of these components would identify active or emerging aging degradation in a timely manner.

3.4 Conditions in the SE for MRP-227

The NRC SE for MRP-227 contains seven conditions that the licensee must follow to receive credit for MRP-227-A implementation. The NRC staff reviewed the licensee's submittal against these seven conditions.

Condition 1: The licensee, in the AMP provided in the December 16, 2013, submittal, added the upper core support barrel assembly and upper core barrel flange weld to the AMP. This addition is in accordance with the guidelines addressed in Table 4-5 of the MRP-227-A report, and, therefore, the NRC staff finds Condition 1 to be met.

Condition 2: In accordance with the guidelines provided in Table 4-2 of MRP-227-A, the licensee included the lower core cylinder girth welds and core support barrel assembly welds in its AMP. Therefore, the NRC staff finds Condition 2 to be met.

Condition 3: In accordance with the guidelines provided in Table 4-2 of the MRP-227-A report, the licensee included the core support column welds in its AMP. Therefore, the NRC staff finds Condition 3 to be met.

Condition 4: Condition 4 states that a minimum of 75 percent coverage of the entire examination volume (i.e., including both accessible and inaccessible regions) of the RVI components and their welds, and a minimum sample size of 75 percent of the total population of like components (i.e. bolts), should be inspected. The licensee included this guideline in its AMP; therefore, the NRC staff finds Condition 4 to be met.

Condition 5: Condition 5 is applicable for a bolted core shroud assembly. Since the core shroud assembly in the reactor vessel at WF3 is a welded assembly, the NRC staff finds Condition 5 is not applicable for WF3.

Condition 6: Condition 6 states that subsequent re-examination for all "Expansion" inspection category components should be at a 10 year interval once degradation is identified in the associated "Primary" inspection category component. The licensee included this guideline in the AMP; therefore, the NRC staff finds Condition 6 to be met.

Condition 7: In Section 6.0 of the AMP, the licensee stated that the operating experience related to the aging degradation of the RVI components in the PWR fleet would be periodically documented. Furthermore, Attachment A of the December 16, 2013, submittal included the operating experience related to the aging degradation of some of the RVI components at WF3. See Section 3.1 of this SE. The NRC staff reviewed Attachment A and found that the licensee provided the information required by MRP-227-A. Therefore, the NRC staff finds Condition 7 to be met.

Based on the review of the licensee's responses to the seven above conditions, the NRC staff concludes that the AMP for the RVI components adequately addressed all of the conditions in the NRC staff's safety evaluation for MRP-227.

4.0 CONCLUSION

The NRC staff has reviewed the inspection plan for the WF3 RVI components against MRP-227-A and the seven conditions and eight applicant/licensee action items of the NRC safety evaluation for MRP-227. The NRC staff finds that the WF3 AMP is consistent with MRP-227-A and that the seven conditions and eight applicant/licensee action items of the NRC SE for MRP-227 are appropriately addressed for the current licensing basis. The NRC staff concludes that the WF3 AMP will adequately manage RVI components' aging effects, such that, they will perform their intended functions in accordance with the current licensing basis.

Principal Contributor: G. Cheruvenki

Date: October 6, 2015

The NRC staff's safety evaluation is enclosed.

If you have any questions, please contact me at 301-415-3229 or via e-mail at Michael.Orenak@nrc.gov.

Sincerely,

/RA/

Michael D. Orenak, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosure:
Safety Evaluation

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