UNITED STATES
NUCLEAR REGULATORY COMMISSION
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May 27, 2016

Mr. Shane M. Marik
Site Vice President and Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station
9610 Power Lane, Mail Stop FC-2-4
Blair, NE 68008

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 – SAFETY EVALUATION RELATED TO REACTOR VESSEL INTERNALS INSPECTION PLAN BASED ON MRP-227-A (CAC NO. MF3412)

Dear Mr. Marik:

In its letter dated September 27, 2012, as supplemented by letters dated August 22, 2014, and April 13 and November 30, 2015, Omaha Public Power District (OPPD, the licensee), submitted an aging management program (AMP) for the reactor vessel internals (RVI) at Fort Calhoun Station, Unit No. 1 (FCS). The Electric Power Research Institute technical report, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," December 2011 (MRP-227-A), and its supporting reports were used as technical bases for developing the FCS AMP. The licensee submitted the AMP to meet License Renewal Commitment 16 (implementation of MRP-227-A at FCS), as listed in FCS Updated Safety Analysis Report (USAR) Section 15.4. The AMP included the inspection and evaluation guidelines for the RVI components at FCS.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the AMP for the RVI components at FCS, and concludes that it is acceptable because it is consistent with the inspection and evaluation guidelines of MRP-227-A, and the licensee has appropriately addressed all seven conditions and eight licensee action items specified in the staff's safety evaluation for MRP-227-A. Therefore, the NRC staff concludes that OPPD meets License Renewal Commitment 16 as listed in FCS USAR Section 15.4. The staff's safety evaluation of the FCS AMP for the RVI components is enclosed with this letter.

If you have any questions, please contact me at 301-415-2296 or via e-mail at Fred.Lyon@nrc.gov.

Sincerely,

CF Lyon

Carl F. Lyon, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

INSPECTION PLAN FOR REACTOR VESSEL INTERNALS

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

## 1.0 INTRODUCTION AND BACKGROUND

In a letter dated September 27, 2012 (Agencywide Documents Access and Management System (ADAMS Accession No. ML12276A006), as supplemented by letters dated August 22, 2014, and April 13 and November 30, 2015 (ADAMS Accession Nos. ML14234A530, ML15103A642, and ML15350A018, respectively), Omaha Public Power District (OPPD, the licensee), submitted to the U.S. Nuclear Regulatory Commission (NRC) an aging management program (AMP) for the reactor vessel internals (RVI) at Fort Calhoun Station, Unit No. 1 (FCS). This submittal is in accordance with the licensee's Commitment 16 addressed in Section 15.4 of the Updated Safety Analysis Report (USAR). The Electric Power Research Institute's (EPRI's) technical report, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," December 2011 (MRP-227-A) (ADAMS Package Accession No. ML120170453), and its supporting reports were used as technical bases for developing FCS's AMP. The NRC staff's safety evaluation (SE) dated December 6, 2011, for the AMP for the pressurized-water reactor (PWR) RVI components is addressed in the MRP-227-A report. The licensee submitted its AMP for the RVI components in FCS, consistent with the requirement addressed in Section 3.1.2.3.2, "Reactor Vessel Internals Inspection Program," of NUREG-1782, "Safety Evaluation Report Related to the License Renewal of Fort Calhoun Station, Unit 1" (SER), October 2003 (ADAMS Accession No. ML033020438). The AMP included the inspection and evaluation (I&E) guidelines for the RVI components at FCS. In this SE, the terms AMP and I&E guidelines are used interchangeably.

## 2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," addresses the requirements for the plant license renewal (LR) process. The regulation at 10 CFR Section 54.21 requires that each application for LR contain an integrated plant assessment (IPA) and an evaluation of time-limited aging analyses (TLAA). The plant-specific IPA shall identify and list those structures and components subject to an aging management review and demonstrate that the effects of aging (e.g., cracking, loss of material, loss of fracture toughness, dimensional changes, and loss of preload) will be adequately managed so that their intended functions will be maintained consistent with the current licensing basis for the period of extended operation (PEO) as required by

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10 CFR 54.29(a). In addition, 10 CFR 54.22 requires that an LR application include any technical specification changes or additions necessary to manage the effects of aging during the PEO as part of the LR application.

Structures and components subject to an AMP shall encompass those structures and components that (1) perform an intended function, as described in 10 CFR 54.4, without moving parts or without a change in configuration or properties and (2) are not subject to replacement based on a qualified life or specified time period. These structures and components are referred to as "passive" and "long-lived" structures and components, respectively. The scope of components considered for inspection under MRP-227-A includes core support structures (typically denoted as Examination Category B-N-3 by American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code*, Section XI) and those RVI components that serve an intended LR safety function pursuant to criteria in 10 CFR 54.4(a)(1). The scope of the program does not include consumable components such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation because these components are not typically within the scope of the components that are required to be subject to an AMP, as defined by the criteria set in 10 CFR 54.21(a)(1).

The FCS inspection plan was developed by the licensee based on MRP-227-A with the intent to meet the requirements addressed in Section 3.1.2.3.2 of the NRC staff's October 2003 SER (NUREG-1782). The EPRI's MRP-227-A summarized the most recent industry recommended I&E guidelines for PWR RVI components. The SE for MRP-227-A was issued on December 16, 2011, with seven topical report (TR) conditions and eight applicant/licensee action items. The TR conditions were specified to ensure that certain information was revised generically in the published MRP-227-A, and the applicant/licensee action items were specified for applicants/licensees to address plant-specific issues which could not be resolved generically in the December 16, 2011, SE for MRP-227-A. In fact, almost all actions related to the above-mentioned LR commitment have been accomplished as a result of issuance of the December 16, 2011, SE for MRP-227-A, or are being addressed by the licensee in its plant-specific inspection plan based on MRP-227-A in response to the eight applicant/licensee action items.

## 3.0 TECHNICAL EVALUATION

In its September 27, 2012, submittal, the licensee included inspection requirements of RVI components that are consistent with MRP-227-A. The NRC staff reviewed the AMP for the RVI components addressed in the submittal and determined that a major portion of the licensee's AMP had no specific technical information which would affect the review and approval of the FCS inspection plan. Therefore, the focus of the NRC staff's evaluation is related to the following issues: (1) operating experience (OE) at FCS; (2) Applicant/Licensee Action Items 1 through 8 that are addressed in the staff's SE for MRP-227-A; (3) the staff's Conditions as addressed in the SE for MRP-227-A; (4) the staff's evaluation of the licensee's AMP for the RVI components under the current ASME Code, Section XI inservice inspection (ISI) program; (5) the staff's evaluation of the licensee's LR commitments; and (6) alloy materials in RVI components that are susceptible to aging degradation. The following sections of the SE include the staff's evaluation of the six issues addressed above.

3.1 <u>Evaluation of the Licensee's AMP for RVI Components - Operating Experience</u>

Appendix A in MRP-227-A addresses the AMP for RVI components that are susceptible to various aging degradation mechanisms. Appendix A also includes OE related to the identification of various aging effects in some RVI components that were manifested during plant operation. In this context, in a request for additional information (RAI) dated July 8, 2014 (ADAMS Accession No. ML14190A211), the NRC staff requested the licensee to explain how it would evaluate and disposition relevant plant-specific or generic OE applicable to RVI components at FCS. The staff requested that the licensee identify any and all generic and plant-specific OE that is applicable to the design of the RVI components including, but not limited to, OE that is applicable the following components at FCS: panel to former bolts, core barrel bolting, thermal shields (including positioning pins), fuel alignment pins, guide lug inserts and bolts, guide lugs, core support barrel girth welds, in-core instrumentation (ICI) flux thimble tubes, core barrel at the prior thermal shield bracket attachment areas, and reactor vessel flow skirt.

By letter dated August 22, 2014, in response to RAI-4, the licensee stated that it reviewed OE applicable to the aforementioned RVI components and concluded that, currently, there are no failures associated with these components. Furthermore, the licensee stated that OE at FCS is documented each trimester. The licensee includes any generic OE from the nuclear industry in its corrective action program. The NRC staff considers this response acceptable because: (1) any emerging OE from the nuclear industry is continuously addressed by the licensee; and (2) information applicable to OE is upgraded every trimester. Based on the licensee's response, the staff concludes that the licensee has addressed the issue adequately by including the previously observed aging effects. In addition, the licensee would update its AMP by including any emerging (future) aging degradation of RVI components identified in the PWR fleet. Therefore, the staff considers that the issue associated with OE has been adequately addressed by the licensee.

3.2 <u>Applicant/Licensee Action Item 1 of SE for MRP-227-A</u>

Section 4.2.1 of the SE for MRP-227-A states, in part, that:

> Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the [failure modes, effects and criticality analysis (FMECA)] and functionality analyses for reactors of their design (i.e., Westinghouse, CE [Combustion Engineering], or B&W [Babcock and Wilcox]) which support MRP-227... 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 the generic issue of the information needed from licensees to resolve Action Item 1, a series of public and non-public meetings were conducted, at which the NRC, Westinghouse, EPRI, and utility representatives discussed regulatory concerns and determined a path 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 proprietary report WCAP-17780-P, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs," June 2013. This report provides background proprietary design information regarding variances in stress, fluence, and temperature in the plants designed by Westinghouse and CE to support NRC reviews of utility submittals that demonstrate the 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. For a plant to demonstrate that it is bounded by the MRP-227-A evaluation for originally licensed and uprated conditions, it would need to provide a satisfactory response to the following two questions by each Westinghouse and CE unit.

> Question 1: Does the plant have non-weld or bolting austenitic stainless steel (SS) components with 20 percent cold work or greater, and, if so, do the affected components have operating stresses greater than 30 ksi? [If both conditions are true, additional components may need to be screened in stress corrosion cracking (SCC)].
>
> Question 2: Does the plant have atypical fuel design or fuel management that could render the assumptions of MRP-227-A, regarding core loading/core design, non-representative for that plant?

By MRP Letter 2013-025 dated October 14, 2013, EPRI provided to licensees "MRP-227-A Applicability Guidelines for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs," a non-proprietary document (ADAMS Accession No. ML13322A454) containing guidance for responding to the two questions above. The NRC staff assessed MRP Letter 2013-025 and the technical basis that was used for the guidance contained in WCAP-17780-P, and concluded that if an applicant or licensee demonstrates that its plant(s) comply with the guidance in MRP Letter 2013-025, there is reasonable assurance that the I&E guidance of MRP-227-A will be applicable to the specific plant(s). The details of the staff's assessment (non-proprietary) of WCAP-17780-P and MRP Letter 2013-025 can be found in ADAMS Accession No. ML14309A484. The guidance in MRP Letter 2013-025 provides an acceptable basis for the licensees to respond to generic questions 1 and 2 addressed above.

In RAI 2 dated July 8, 2014, the NRC staff requested that the licensee provide the information discussed in NRC meeting summaries dated February 21, 2013,[1] and March 15, 2013,[2] related to verification of the applicability of MRP-227-A to FCS.

Regarding Question 1, the EPRI provides guidance for licensees to assess whether RVI components at their plant(s), other than those identified in the generic evaluation, have the potential for cold work greater than 20 percent. The NRC staff considers that MRP

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[1] Golla, J., memorandum to Anthony Mendiola, U.S. Nuclear Regulatory Commission, "Summary of January 22-23, 2013, Closed Meeting with the Electric Power Research Institute and Westinghouse," dated February 21, 2013 (ADAMS Accession No. ML13042A048).

[2] Golla, J., memorandum to Anthony Mendiola, U.S. Nuclear Regulatory Commission, "Summary of February 25, 2013 Telecom with the Electric Power Research Institute and Westinghouse Electric Company," dated March 15, 2013 (ADAMS Accession No. ML13067A262).

Letter 2013-025, with respect to the effect of cold work on the AMP for the RVI components at FCS, is acceptable because of the following reasons.

In its RAI response dated November 30, 2015, the licensee evaluated the effect of cold work on SCC in RVI components by using technical bases addressed in Westinghouse's generic aging evaluation of RVI components contained in MRP-191, Revision 0, "Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design," November 2006 (ADAMS Accession No. ML091910130).

Cold work during the original fabrication of the RVI components could potentially increase the susceptibility to SCC in austenitic stainless steels. Non-welded RVI components at FCS were binned under various categories based on the extent to which these components were subjected to cold work during the original fabrication. The licensee stated that it considered plant modifications and operating history of FCS, and binned the RVI components under the following categories: (1) Category 1 - cast austenitic stainless steel (CASS); (2) Category 2 – hot-formed stainless steel; (3) Category 3 - annealed austenitic stainless steels; (4) Category 4 - fasteners austenitic stainless steels; and (5) Category 5 – cold-formed austenitic stainless steels without subsequent solution annealing. The licensee stated that RVI components that were binned under Categories 1, 2, and 3 at FCS were not subjected to cold work greater than 20 percent. This criterion is consistent with MRP-191 and Option 2 addressed in Appendix A of MRP Letter 2013-025. Therefore, the licensee concluded that the I&E guidelines in MRP-227-A are valid for the RVI components under Categories 1, 2, and 3.

The RVI components binned under Category 4 were already assumed to have been exposed to cold work greater than 20 percent. These components were already evaluated in accordance with the MRP-191. According to the licensee, RVI components binned under Category 5 were not subject to severe cold work. Therefore, the licensee concluded that the material fabrication and design related to the RVI components binned under Categories 4 and 5 were consistent with the guidelines addressed in MRP-191. Therefore, the I&E guidelines in MRP-227-A are valid for the RVI components under Categories 4 and 5.

Based on the information provided by the licensee, the NRC staff determined that the categorization process (Categories 1 through 5) of the RVI components at FCS, takes into account the effect of cold work on the SCC. Based on the licensee's response, the staff concluded that the RVI components binned under Categories 1 through 5 met the criterion, which is consistent with MRP-191 and the criterion addressed by Option 2 in Appendix A of MRP Letter 2013-025. Based on the review, the staff determined that I&E guidelines developed in MRP-227-A remain valid for RVI components that were binned under Categories 1 through 5 at FCS.

With respect to Question 2, MRP Letter 2013-025 provides quantitative criteria to allow a licensee to assess whether a particular plant has an atypical fuel design or fuel management. For a CE design plant such as FCS, these criteria are:

(1) The heat generation rate must be less than or equal to 68 Watts per cubic centimeter (/$cm^3$).

(2) The maximum average core power density must be less than 110 Watts/$cm^3$.

(3) The active fuel to fuel alignment plate (FAP) distance must be greater than 12.4 inches.

In its response to Question 2, the licensee stated that it does not comply with the MRP-227-A assumptions regarding core loading/core design. Neutron fluence and heat generation rates are concluded to be outside the limiting MRP guidance threshold values addressed in MRP Letter 2013-025, but are determined to be acceptable based on plant-specific information provided in PWROG-15030-P, Revision 0, "Evaluation of Fort Calhoun Fuel Alignment Plate Fluence for MRP-227-A," included with the licensee's November 30, 2015, submittal (non-proprietary version available at ADAMS Accession No. ML15350A020). The licensee stated that the FAP distance at FCS was less than 12.4 inches for multiple fuel cycles. However, the licensee provided the power densities of the fuel cycles, and there is a substantial margin to the power density acceptance criterion of less than 110 Watts/$cm^3$. The comparison of the smaller fuel-to-FAP distance with the lower power density shows that the increase in fast neutron fluence from the decreased fuel-to-FAP distance is offset by the reduced power density for the first 23 operating cycles at FCS. Although the licensee did not meet the fuel-to-FAP distance screening criterion, the fast neutron fluence is still below the damaging level of fast neutron fluence value used in developing MRP-227-A. In addition, the licensee provided the heat generation rate values for FCS, and there is a substantial margin to heat generation value acceptance criterion of less than or equal to 68 Watts/$cm^3$.

Therefore, FCS is still bounded for the RVI components receiving less fast neutron fluence than addressed in MRP Letter 2013-025. Based on the submitted response, the NRC staff concludes that the licensee satisfied the guidelines related to the fuel management issue addressed in MRP Letter 2013-025 and that implementation of I&E guidelines addressed in MRP-227-A would be valid during the PEO. Based on this review, the staff considers that the licensee addressed Action Item 1 satisfactorily.

### 3.3 Evaluation of the Licensee's Resolution of Action Item 2 in the SE for MRP-227-A

In its letter dated November 30, 2015, the licensee stated that as part of the FCS AMP, it included a list of all RVI components that required an aging management review. This list of RVI components is addressed in Table 2.3.1.1-1 in the LR application dated January 9, 2012 (ADAMS Accession Nos. ML020290333 and ML020290338). Furthermore, the licensee stated that no modifications in RVI components were implemented to date; therefore, no additional aging evaluation in the RVI components was necessary at FCS. The RVI components identified by the licensee in the LR application are consistent with those listed in the MRP-191 report, which was used as a supporting document for developing MRP-227-A. Based on this evaluation, the licensee concluded that no revisions are required to the AMP for the RVI components at FCS. The NRC staff reviewed the licensee's evaluation and concludes that: (1) the licensee's AMP for the RVI components is consistent with MRP-227-A I&E guidelines; (2) no additional RVI components at FCS were screened in due to the usage of different type of materials that were not prescribed in MRP-191/MRP-227-A; (3) the licensee satisfied the guidelines addressed in Action Item 1, and (4) no modifications in RVI components were implemented to date; therefore, no additional evaluations were necessary for the RVI

components during the PEO. Details of the staff's evaluation of Action Item 1 are addressed in Section 3.2 of this SE. Based on this assessment, the staff considers that the licensee addressed Action Item 2 satisfactorily.

### 3.4 Evaluation of the Licensee's Resolution of Action Item 3 in the SE for MRP-227-A

In Section 6.2.3 of its letter dated September 27, 2012, the licensee stated that plant-specific AMPs were implemented for the following RVI programs at FCS: (1) thermal shield integrity program and (2) ICI thimble tube program. In the supplement dated April 13, 2015, in response RAI-2-1 dated March 3, 2015 (ADAMS Accession No. ML15057A015), the licensee provided additional information on the aforementioned RVI components.

The licensee stated that the thermal shield is monitored for fatigue damage, cracking, reduction in fracture toughness, change in dimensions, and loss of preload. Visual inspections (VT-3) are performed to identify these degradation mechanisms. The licensee stated that loss of preload was observed previously in this component during a repair in 1992. The licensee determined to perform a follow-up VT-3 during refueling outage 28 in 2016. The licensee stated that the future inspection frequency will be determined based on the extent of the observed aging degradation during refueling outage 28.

Similarly, ICI thimble tubes are monitored for cracking, loss of material, reduction in fracture toughness, and change in dimensions. The licensee stated that consistent with MRP-227-A guidelines, the ICI tubes will be inspected for the first time during refueling outage 28. The licensee stated that based on the inspection results, it will determine which of the ICI thimble tubes will be replaced. The licensee stated that 22 ICI thimble tubes are scheduled to be replaced during refueling outage 27 in 2015. Six ICI thimble tubes are scheduled to be replaced during refueling outage 28. The licensee subsequently confirmed that 22 ICIs were replaced during RFO 27 beginning in April 2015, and the remaining 6 ICIs are scheduled for replacement in RFO 28. ICIs are replaced within a 9 year interval. The next scheduled replacement of all 28 ICIs is in RFO 30 (ADAMS Accession No. ML16131A731).

Based on the information provided, the NRC staff determined that the licensee is implementing the AMPs related to the thermal shield and ICI thimble tubes satisfactorily. The staff accepts the licensee's response to Action Item 3 addressed in the staff's SE for MRP-227-A because: (1) the components are being inspected in accordance with their plant-specific inspection criterion, which is based on the extent of aging degradation of thermal shield and ICI thimble tubes, and (2) the licensee has demonstrated that it is adequately taking corrective action that is necessary to ensure that the functionality of these components would be maintained during the PEO.

### 3.5 Evaluation of the Licensee's Resolution of Action Items 4 and 6 in the SE for MRP-227-A

Action Items 4 and 6 of the NRC staff's SE for MRP-227-A are applicable to RVI components designed by B&W and, therefore, are not applicable to FCS.

3.6 Evaluation of the Licensee's Resolution of Action Item 5 in the SE for MRP-227-A

In its submittal dated September 27, 2012, the licensee stated that the FCS design has a core shroud assembly with full-height bolted core shroud plates and, therefore, physical measurement examination is not required. The NRC staff accepts this response because the physical measurement is to be implemented to verify the gap between the top and bottom core shroud segments for core barrel shrouds assembled in two vertical sections. Since the FCS core shroud assembly consists of full-height bolted core shroud plates, this measurement is not required.

3.7 Evaluation of the Licensee's Resolution of Action Item 7 in the SE for MRP-227-A

In Section 6.2.7 of its submittal dated September 27, 2012, the licensee stated that the FCS lower support columns were fabricated with CASS with a calculated ferrite value of less than or equal to 20 percent. CASS materials with delta ferrite greater than 20 percent would be susceptible to loss of fracture toughness due to thermal embrittlement. Since the delta ferrite in lower support columns at FCS is below the threshold limit of 20 percent, thermal embrittlement is unlikely to occur in these columns. However, these columns are susceptible to irradiation embrittlement. Irradiation embrittlement results in the cracking of RVI components based on the intensity of the applied stress, which determines the severity of the cracking. With respect to irradiation embrittlement in the CASS lower support columns, the NRC staff noted that in Table 4-2 of MRP-227-A, the stainless steel lower support column welds were binned under the "Primary" inspection category and are to be inspected during every 10-year interval in accordance with MRP-227-A guidelines. Therefore, the staff believes that any cracking due to irradiation embrittlement would be identified in the lower core support column welds in a timely manner. Since the lower support columns are subject to similar aging degradation due to irradiation embrittlement as the lower support column welds, any emerging degradation identified in the lower support column welds would provide observable indications that would facilitate the licensee's effective monitoring of any aging effects in the lower support columns. Since the lower support column welds are part of the AMP that is consistent with the MRP-227-A guidelines, the staff finds that the degradation of the CASS lower support columns is adequately managed at FCS. The staff concludes that the licensee has satisfactorily addressed Action Item 7.

3.8 Evaluation of the Licensee's Resolution of Action Item 8 in the SE for MRP-227-A

Action Item 8 of the NRC staff's SE for MRP-227-A requires that the licensee submit an AMP for RVI components consistent with I&E guidelines addressed in MRP-227-A. In Section 5.1 of its submittal dated September 27, 2012, the licensee addressed the ten AMP elements. The licensee's AMP program elements were reviewed and the staff concluded that they comply with NUREG-1801, Revision 2, "Generic Aging Lessons Learned," December 2011 (ADAMS Accession No. ML103490041), AMP program elements.

Time-limited aging analysis (TLAA) related to fatigue evaluation applies to the following two RVI components binned under the "Primary" inspection category at FCS: (1) lower flange weld and

(2) core support plate. By letter dated August 22, 2014, in response to RAI-7 dated July 8, 2014, the licensee stated that in lieu of performing a TLAA, the aging effects due to fatigue in these components would be monitored by frequent inspections in accordance with the requirements of MRP-227-A. Frequent inspections provide reasonable assurance that any cracking due to fatigue would be identified in a timely manner in these RVI components. The NRC staff considers that implementation of I&E guidelines would be as effective as performing a TLAA to monitor fatigue. Based on the response, the staff determined that implementation of I&E guidelines consistent with MRP-227-A to monitor fatigue in these components is acceptable.

The NRC staff concludes that the licensee has satisfactorily addressed the Action Item 8.

### 3.9 TR Conditions in the Staff's SE for the MRP-227-A

The NRC staff's SE for MRP-227-A 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 submittal dated September 27, 2012, 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; 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: This condition states that a 10-year inspection frequency for core shroud bolts in CE-designed reactors should be implemented following the initial or baseline inspection. The licensee satisfied this condition by including this criterion in its September 27, 2012, submittal, and, therefore, the NRC staff finds Condition 5 to be met.

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 5.10 of the September 27, 2012, submittal, the licensee stated that the OE related to the aging degradation of the RVI components in the PWR fleet would be periodically documented. Furthermore, the licensee included OE related to the aging degradation of some of the RVI components at FCS. The staff's review of the OE at FCS is addressed in Section 3.1 of this SE. Based on the review, the staff found that the licensee provided the necessary information required by MRP-227-A. The NRC staff finds Condition 7 to be met.

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

### 3.10 RVI Components in the ASME Code, Section XI, ISI Program

In Appendix B of the letter dated September 27, 2012, the licensee included a list of RVI components binned under the ASME Code, Section XI ISI program. Consistent with ASME Code, Section XI requirements, the licensee monitors the aging degradation of the RVI components of the vessel interior and core support structures. By monitoring the aging degradation in RVI components under its ASME Code, Section XI ISI program, the licensee would be likely to identify any emerging degradation in a timely manner. The licensee stated that the ASME Code, Section XI, inspections using VT-3 technique to date have revealed no aging degradation in these RVI components. Based on this information, the NRC staff concludes that continuing the ASME Code, Section XI inspections provide assurance that the licensee is adequately managing the aging degradation in the RVI components at FCS. In addition, as stated in Section 3.1 of this SE, the licensee is monitoring emerging OE from the nuclear industry and the information applicable to OE is evaluated by the licensee. Therefore, the staff considers that the licensee had adequately implemented its AMP related to RVI components that were categorized under its ASME Code, Section XI ISI program at FCS.

### 3.11 Evaluation of License Renewal Commitments Addressed in Appendix A of the NRC Staff's SER for NUREG-1782

The licensee made three commitments in Section 15.4 in the USAR and these commitments are related to the AMP for RVI components at FCS. These commitments are applicable to some of the RVI components which require an appropriate AMP to ensure that aging degradation is adequately monitored during the PEO. Commitments 16, 17, and 18 are included in FCS USAR Section 15.4, "License Renewal Commitment Listing," which is part of the FCS licensing basis and is subject to the requirements of 10 CFR 50.59. In addition, completion of these commitments is subject to future NRC inspection under Inspection Manual Chapter 0305, "Operating Reactor Assessment Program." The commitments and staff evaluation of each are discussed below:

(a) Commitment 16: The licensee made a commitment that it will participate in the industry-developed inspection program for the RVI components. The licensee's commitment requires implementation of an AMP developed by the industry. In its August 22, 2014, response to RAI-6 dated July 8, 2014, the licensee stated that it incorporated I&E guidelines addressed in MRP-227-A, which requires inspections to be performed on RVI components. In addition, in this

commitment, the licensee stated that aging degradation in core shroud bolts will be monitored during its PEO. The core shroud bolts are susceptible to irradiation-assisted stress-corrosion cracking, irradiation embrittlement, irradiation stress relaxation, and void swelling. These shroud bolts are to be inspected every 10-year interval using ultrasonic testing. Based on this response, the NRC staff concludes that the licensee complied with this commitment.

(b) Commitment 17: Aging degradation in thermal shield bolts or pins is to be monitored during the PEO. In its August 22, 2014, response to RAI-6 dated July 8, 2014, the licensee stated that it incorporated an augmented inspection program to monitor thermal shield vibrations during the PEO. The NRC staff finds this response is acceptable.

(c) Commitment 18: Monitoring of aging degradation in the reactor vessel flow skirt is required. In its August 22, 2014, response to RAI-6 dated July 8, 2014, the licensee stated that Commitment 18 will be met by implementing the MRP-227-A inspection program. Implementation of the inspection program replaces the original commitment, which included fracture mechanics evaluations to ensure the functionality and structural integrity of the reactor vessel flow skirt. The NRC staff noted that the licensee's inspection program (per MRP-227-A) addressed in Appendix C of WCAP-17347-NP, Revision 1, "PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Fort Calhoun Station," August 2012 (ADAMS Accession No. ML12276A006), did not include an AMP for the reactor vessel flow skirt. Therefore, the staff identified this discrepancy and requested the licensee to clarify how an effective AMP is implemented for the reactor vessel flow skirt at FCS.

In Table B-1 of the September 27, 2012, submittal, the licensee identified the following aging degradation mechanisms for the reactor vessel flow skirt: (1) primary-water stress-corrosion cracking; (2) void swelling; (3) fatigue; and (4) reduction in fracture toughness. Primary-water stress-corrosion cracking is known to occur in Alloy 600 materials.

In an e-mail dated September 17, 2015 (ADAMS Accession No. ML16039A261), the licensee stated that condition report 2015-11011 was written to address this issue. The licensee's disposition stated that the reactor vessel flow skirt will be inspected during refueling outage 28. Based on the information provided by the licensee, the staff determined that the licensee's proposed continued inspections, consistent with MRP-227-A I&E guidelines during the PEO, would provide reasonable assurance that the licensee is adequately monitoring the aging degradation of reactor vessel flow skirt at FCS.

### 3.12 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 identified in Appendix A of MRP-227-A: (1) nickel-base alloys, (2) precipitation-hardened stainless steel materials,

(3) A286, (4) A453, and (5) type 431 stainless steel materials. In this context, the NRC staff requested that the licensee provide a list of any additional RVI components (not listed in MRP-227-A and MRP-191, Revision 0) that are manufactured from those materials. If any of these materials is identified in additional RVI components at FCS, the licensee was asked in the NRC RAI dated March 3, 2015, to provide information on the type of aging effects that were detected, and the AMP implemented for these components.

In its RAI response dated November 30, 2015, the licensee stated that the materials used in the RVI components at FCS are consistent with the materials listed in MRP-191 and MRP-227-A. The NRC staff reviewed materials addressed in Table 4-7 in MRP-191 and, based on the review, concludes that the materials listed above are not used at FCS. Therefore, based on the information provided, the staff concludes that the licensee has adequately addressed this issue and that the issue is closed.

## 4.0 CONCLUSION

The NRC staff has reviewed the AMP for the RVI components at FCS and concludes that it is acceptable because it is consistent with the I&E guidelines of MRP-227-A, and the licensee has appropriately addressed all seven conditions and eight licensee action items specified in the staff's SE for MRP-227-A. Therefore, the NRC staff concludes that OPPD meets License Renewal Commitment 16 (implementation of MRP-227-A at FCS) as listed in FCS USAR Section 15.4.

Principal Contributor: G. Cheruvenki, NRR/DE/EVIB

Date: May 27, 2016

If you have any questions, please contact me at 301-415-2296 or via e-mail at Fred.Lyon@nrc.gov.

Sincerely,

*/RA/*

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Office of Nuclear Reactor Regulation

Docket No. 50-285

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