# OPPD Response to Remaining Questions From NRC Request for Additional Information (RAI) Regarding Fort Calhoun Station, Reactor Vessel Internal Component Aging Management Program (Non-Proprietary Version)

REQUEST FOR ADDITIONAL INFORMATION
AGING MANAGEMENT PROGRAM FOR THE
REACTOR VESSEL INTERNALS
FORT CALHOUN STATION, UNIT 1
OMAHA PUBLIC POWER DISTRICT
DOCKET NO. 50-285

**RAI 1:**

**Historically, the following materials used in the pressurized water reactor (PWR) reactor vessel internals (RVI) components were known to be susceptible to some of the aging degradation mechanisms that are identified in the MRP-227-A report. In this context, the U.S. Nuclear Regulatory Commission (NRC) staff requests 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 the following materials. If any of these materials are identified as an additional RVI component at FCS, provide information on the type of aging effect that was detected, and the type of AMP implemented on these components.**

**Nickel base alloys—Inconel 600; Weld Metals—Alloy 82 and 182 and Alloy X-750.**

**Stainless steel type 347 material (excluding baffle-former bolts).**

**Precipitation hardened (PH) stainless steel materials—17-4 and 15-5.**

**Type 431 stainless steel material.**

**Alloy A-286, ASTM A 453 Grade 660, Condition A or B.**

**OPPD Response to RAI 1:**

There were no additional components identified for Fort Calhoun Station (FCS) in the comparison of components to MRP-191. A number of components designated as within the scope of license renewal by the FCS License Renewal Application (LRA) Table 2.3.1.1-1 did not have a precise analog in the component names given in MRP-191, Table 4-5. However, each of these components was covered by the generic failure modes effects and criticality analysis (FMECA) and functionality analysis because the development of the MRP-227-A aging management strategies considered these component s as assemblies. Therefore, the FCS RVI components and material are covered by the list of generic Combustion Engineering-designed PWR RVI components within MRP-191 and MRP-227-A.

**RAI 2:**

**Related to MRP-2013-025, "MRP-227-A Applicability Template Guidelines," (ADAMS Accession No. ML13322A454), the staff has identified two additional questions that all CE and Westinghouse design plants referencing topical report MRP-227-A must answer to close Applicant/Licensee Action Item (AI) 1 related to plant-specific applicability of the topical report. If the answer to either or both questions is yes, then further evaluation will be necessary to demonstrate the applicability of MRP-227-A to FCS. The staff therefore requests the following information:**

1. **Do the FCS RVI have non-weld or bolting austenitic stainless steel components with 20% cold work or greater, and if so do the affected components have operating stresses greater than 30 ksi? In particular, the staff is interested in 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 FCS, the licensee should list plant-specific RVI components that have been exposed to cold work equal to or greater than 20%. Plant-specific information related to this issue as addressed in "Option 2" in Appendix A, should be provided.**

**OPPD Response to RAI 2-1:**

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

Westinghouse has evaluated the Fort Calhoun reactor internals components according to industry guideline MRP 2013-025 and the MRP-191 industry generic component listings and screening criteria (including consideration of cold work as defined in MRP-175, 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 "severe cold work," a resulting reduction in wall thickness of 20%, was applied as an evaluation limit. It was confirmed that all Fort Calhoun components, as applicable for design, are included in the MRP-191 component lists. 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 Materials (ASTM) International or American Society of Mechanical Engineers (ASME) material specifications that were called out in the original plant construction drawings. Material and component procurement was through applicable quality-controlled protocols.

Fort Calhoun components were binned according to the following categories for the materials used in component fabrication.

Cold work categories based on MRP 2013-025 include the following:

- Cast austenitic stainless steel (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)

Cold work potential is based on MRP-227-A generic criteria:

- No (N) typically applies to Categories 1, 2, and 3
- Yes (Y) typically applies to Categories 4 and 5

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% cold work. Then, this option was employed in the assessment of the component, and was selected for the purpose 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, but a conservative approach was selected for this assessment.

The evaluation, performed consistently with MRP 2013-025 guidelines, concluded that the reactor internals Categories 1, 2, and 3 (non-bolting) components at Fort Calhoun contain no cold work greater than 20% as a result of material specification and controlled fabrication construction, and therefore Option 1 is applicable. Category 4 components were already assumed to have the potential for cold work in the MRP-191 generic assessments. Material fabrication specifications used for Fort Calhoun suggest that processes were limiting, which precluded the introduction of severe cold work in some of the Categories 4 and 5 components. In these cases, the components were conservatively considered to be cold-worked for the purpose of this assessment. No Category 5 components with severe cold work were identified for Fort Calhoun. The detailed evaluation for the A/LAI for the Fort Calhoun cold work assessments concluded that the plant-specific material fabrication and design were consistent with the MRP-191 basis and that the MRP-227-A sampling inspection aging management requirements, as related to cold work, are directly applicable to Fort Calhoun.

2. **Has FCS 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? The following guidelines provided by modification/rework package (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.**

   **Option 1:**

   **FCS 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 FCS, the licensee should justify the usage of its fuel management program.**

**Option 2:**

**FCS does not comply with the MRP-227-A assumptions regarding core loading/core design. The licensee should provide a technical justification for the application of MRP-227-A criterion to FCS.**

**OPPD Response to RAI 2-2:**

Fort Calhoun Station (FCS) is applying Option 2 and has not utilized atypical design or fuel management, including power changes/uprates, which are non-representative of the assumptions of MRP-227-A.

To support this conclusion, the assumptions of MRP-227-A, along with the additional guidance provided by the MRP 2013-025, were evaluated. The assumptions of MRP-227-A were evaluated against fuel design changes, core designs, and plant operation. The FCS calculated reactor core power density remains well below the screening criterion of 110.0 W/cm$^3$. It should be noted that during final reviews prior to submittal of this response, two (2) errors were discovered in PWROG-15030-P, Rev. 0, "Evaluation of Fort Calhoun Fuel Alignment Plate Fluence for MRP-227-A." The issue was documented in OPPD's Corrective Action Program and a technical evaluation was completed. EC 67484, "Evaluation of PWROG-15030-P Acceptability for Use at Fort Calhoun Station" reviewed and dispositioned the errors as follows:

- PWROG-15030-P incorrectly assumed that the fresh fuel active fuel heights provided by OPPD to Westinghouse in December 2014 were cycle-average active fuel heights. However, the Cycle 24-27 cores were mixed-core designs, containing fuel from previous cycles with two active fuel heights (i.e., [ ] and [ ]). The actual power density for Cycles 24-27 will range between 79.5 W/cm$^3$ and 80.4 W/cm$^3$ (PWROG-15030-P Table 1) based on the mixed core designs. The most limiting (highest) power density for Cycles 24-27 is [ ] W/cm$^3$ based on a full core with an active fuel length of [ ] inches, which is shown to have a 37% margin to the 110 W/cm$^3$ screening criterion (PWROG-15030-P, Table 3). Therefore, this calculation remains acceptable.
- An incorrect but conservative core support plate (CSP) to fuel alignment plate (FAP) distance was used to calculate the Cycle 1-19 active fuel to FAP gap. The gap distances documented in PWROG-15030-P, Table 1 for Cycles 1-10 and Cycles 11-19 are [ ] and [ ], respectively whereas the correct gap distances for Cycles 1-10 and Cycles 11-19 are [ ] and [ ] respectively. Because the error is conservative, the statement in Section 4 of PWROG-15030-P: "Therefore, even with the active fuel slightly closer to the upper alignment plate than analyzed for MRP 2013-025, FCS reactor vessel upper internals would have received less fluence than analyzed in MRP 2013-025." remains valid.

Based on the technical evaluation conducted by EC 67484, the conclusions of PWROG-15030-P were found to remain valid and the document is acceptable for use at Fort Calhoun Station. PWROG-15030-P shows that the screening criterion of a 12.4 inch distance between the active fuel and the fuel alignment plate was not met for multiple fuel cycles.

For fuel cycles prior to fuel Cycle 20 the distance between the active fuel and the fuel alignment plate was slightly less than the assumed 12.4 inches (the gap varied from [ ]$^{a,c}$ to [ ]$^{a,c}$). Fuel cycles from Cycle 24 to present also do not meet the screening criterion with a gap of [ ]$^{a,c}$.

Even though these screening criteria were not met, FCS is still bounded by the MRP 2013-025 analysis because the screening criteria used a fuel power density that was between 37% and 45% higher than FCS's fuel power density during the fuel cycles where the screening criterion was not met. This margin more than offsets the increased fluence due to the active fuel being closer to the alignment plate.

Therefore, even with the active fuel slightly closer to the upper alignment plate than analyzed for MRP 2013-025, FCS reactor vessel upper internals would have received less fluence than analyzed in MRP 2013-025.

RAI 3:

**Action Item 2 in the staff's SE for the MRP-227-A requires the licensee to confirm that Table 4-4 of the MRP-191, Revision 0, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs," report includes all of RVI components for a licensee's Combustion Engineering-designed (CE-designed) reactor facility, or else to identify the missing components and propose any necessary modifications to the program defined in the MRP-227-A report for CE-designed reactors.**

OPPD Response to RAI 3:

All required components in the FCS LRA Table 2.3.1.1-1 are consistent with those contained in MRP-191. No components were missing; therefore, no modifications to the program defined in MRP-227-A need to be proposed.

RAI 5:

**As discussed in Section 3.3.7 of Revision 1 to the Safety Evaluation for MRP-227, AI 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. MRP-227-A Table 3-2 (Final disposition of CE internals) classifies CASS lower support columns as Primary Components based on susceptibility to irradiation embrittlement (IE) and irradiation assisted stress corrosion cracking (IASCC) and thermal embrittlement (TE). After further review of the existing literature data for the threshold limits for IE and TE of CASS materials, the staff developed a new position on these limits. The bases for the staff's new threshold limits are described in Attachment 1 of this document.**

**To enable evaluation of the susceptibility of the lower support columns to TE and IE, the staff requests that the licensee should provide the following information:**

**(d) provide the neutron exposures for these columns and assess their susceptibility to IE and TE consistent with Attachment 1, or provide a functionality evaluation of the lower support columns considering the effects of IE and TE,**

**(e) provide the ferrite content for each lower support column and,**

**(f) provide the casting method for the column (static or centrifugal), if known.**

**OPPD Response to RAI 5:**

The FCS lower support columns are fabricated from static-cast, low molybdenum, ASTM A351 Grade CF8. For the FCS lower support columns' material heats, the maximum ferrite content (as estimated by Hull's factors) is 10.5 percent, the minimum is 3.1 percent, and the mean is 6.7 percent. Since the maximum delta ferrite content is 10.5 percent, all of the FCS lower support columns have ferrite content below the 15 percent threshold in the NRC screening position. Therefore, per the guidance of Attachment 1 to the RAI, the Fort Calhoun lower support columns are not expected to undergo significant TE and can be treated as wrought stainless steel.

MRP-227-A, Table 3-2 classifies CASS lower support columns as Primary Components based on susceptibility to IE, IASCC, and TE. The FCS lower support columns are not susceptible to TE. In MRP-191, the lower support columns screened as susceptible to IE and IASCC. During the period of extended operation, portions of the CASS lower support columns are expected to exceed the NRC screening threshold of 1.5 dpa and so aging management is required per the guidance of Attachment 1 to the RAI. The MRP-227-A inspection of the lower support column welds constitutes aging management for potential material degradation of the FCS lower support columns.