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"Fort Calhoun Station Summary Report for the Fuel Design / Fuel Management Assessments to Demonstrate MRP-227-A Applicability"

(Non-Proprietary)

PRESSURIZED WATER REACTOR OWNERS GROUP

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WESTINGHOUSE NON-PROPRIETARY CLASS 3

Fort Calhoun Station Summary Report for the Fuel Design / Fuel Management Assessments to Demonstrate MRP-227-A Applicability

Materials Committee

PA-MSC-0983, Revision 1, Task 7

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Westinghouse

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PA-MSC-0983, Revision 1, Task 7

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1 MRP 2013-025 GUIDELINES TO DEMONSTRATE MRP-227-A APPLICABILITY FOR THE FORT CALHOUN STATION REACTOR INTERNALS AGING MANAGEMENT FUEL DESIGN / FUEL MANAGEMENT ASSESSMENTS

The Safety Evaluation (Reference 1) issued on Materials Reliability Program (MRP) technical report MRP 227 by the U.S. Nuclear Regulatory Commission (NRC) contained eight Applicant/Licensee Action Items (A/LAIs). These eight action items must be completed in the implementation of the Inspection and Evaluation (I&E) Guidelines outlined in MRP-227-A (Reference 2).

On November 28, 2012, a public meeting was held (Reference 3) at the NRC office to discuss staff expectations and concerns regarding industry responses to A/LAIs 1 and 2. The concerns were addressed to owners of currently operating pressurized water reactor (PWR) plants designed by Westinghouse and Combustion Engineering (CE). A series of proprietary and public meetings were conducted from January to June of 2013 (References 4 through 7). At these meetings, the NRC, Westinghouse, the Electric Power Research Institute (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.

The NRC staff indicated in Reference 7 that the information provided by the industry to the NRC staff demonstrated that the MRP-227-A I&E Guidelines are applicable for the range of conditions expected at the currently operating Westinghouse and CE-designed plants in the United States. As a result of the technical discussions with the NRC staff, the basis for a plant to respond to the NRC's Request for Additional Information (RAI) to demonstrate compliance with MRP-227-A for originally licensed and uprated conditions was determined to be satisfied with plant-specific responses to the following two questions (References 5 and 7):

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 for 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?

To address the NRC staff's concerns, Westinghouse summarized the proprietary meeting presentations and supporting proprietary generic design basis information in [ ]^{a,c} (Reference 8), and provided it to the NRC. [ ]^{a,c} 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 to demonstrate plant specific applicability of MRP-227-A.

Based on the sensitivity studies documented in [ ]^{a,c}, EPRI developed guidance in MRP 2013-025 (Reference 9) for licensees to respond to the two questions. The objective of MRP 2013-025 is to provide a simple, non-proprietary means of demonstrating plant specific applicability of the MRP-227-A inspection sampling recommendations for managing aging in currently operating U.S. CE and Westinghouse plants for NRC Questions 1 and 2. Plants that exceed the thresholds defined in this document do not necessarily fall outside the MRP-227-A recommendations. Instead, they may require additional plant specific evaluations to fully demonstrate plant specific applicability.

In Reference 10, the NRC staff documented the technical evaluation of [ ]^{a,c} and MRP 2013-025. The staff concluded that if an applicant or licensee demonstrates that its plant(s) comply with the guidance in MRP 2013-025, there is reasonable assurance that the I&E guidance of MRP-227-A will be applicable to the specific plant(s).

The NRC formally transmitted these questions to Omaha Public Power District (OPPD) in RAI 2 (Reference 11).

1.1 FORT CALHOUN STATION EVALUATION

Westinghouse has evaluated the Fort Calhoun Station reactor internals components with regard to fuel designs and fuel management according to industry guideline MRP 2013-025 (Reference 9).

An additional evaluation is required for the Fort Calhoun Station to demonstrate the applicability of the MRP-227-A Inspection and Evaluation Guidelines. This conclusion is based on comparisons of the Fort Calhoun Station core geometry and operating characteristics with the MRP-227-A applicability guidelines for CE-designed reactors specified in Reference 9. The MRP 2013-025 guideline stating that the distance between the top of the active fuel stack and the bottom of the fuel alignment plate (FAP) shall not be less than 12.4 inches for a period of more than two effective full-power years is not met for the Fort Calhoun Station.

Performing the MRP 2013-025 guideline assessment using a calendar years basis is not required but would provide bounding results. Demonstrating that the fluence guideline compliance requirement is met based on effective full-power years, as was done in this evaluation, is acceptable and provides a technically robust and accurate assessment of plant-specific historical operating conditions.

Specifically, the following comparisons with the MRP-227-A applicability guidelines in MRP 2013-025 were established for the key reactor internals components at the Fort Calhoun Station.

1.1.1 Components Located Beyond the Outer Radius of the Reactor Core

Guideline 1 The reactor has been operated with out-in fuel management for 30 effective full-power years or less and all future operation will use low-leakage fuel management.

Comparison The Fort Calhoun Station initiated low-leakage fuel management strategy in the eighth fuel cycle following 5.9 effective full-power years of operation and has been implementing low-leakage core designs since that time. There are no current plans to return to out-in fuel management.

Guideline 2 For operation going forward, the average power density of the reactor core (as defined in MRP 2013-025) shall be less than 110 W/cm^3.

Comparison For the last five operating fuel cycles (Cycles 23 through 27), the Fort Calhoun Station has been operating at a rated power level of 1500 MWt. For the 133 fuel assembly Fort Calhoun Station core geometry, the 1500 MWt power level corresponds to a core power density of 79.5 W/cm^3. Until implementation of the Extended Power Uprate, this level of power generation is also representative of anticipated future operation. Following implementation of the Extended Power Uprate to 1780 MWt, the corresponding core power density will be 94.4 W/cm^3.

Guideline 3 For operation going forward, the nuclear heat generation rate figure of merit (HGR-FOM) (as defined in MRP 2013-025) shall not exceed 68 W/cm^3.

Comparison For the last five operating fuel cycles at the Fort Calhoun Station, the HGR-FOM at key baffle locations has ranged between [ ]^{a,c}. Until implementation of the Extended Power Uprate, this range of HGR-FOM is representative of anticipated future operation. Following implementation of the Extended Power Uprate to 1780 MWt, the corresponding HGR-FOM is expected to range between [ ]^{a,c}.

1.1.2 Components Located Above the Reactor Core

Guideline 1 Considering the entire operating lifetime of the reactor, the average power density of the core (as defined in MRP 2013-025) shall be less than 110 W/cm^3 for a period of more than two effective full-power years.

Comparison Over the operating lifetime of the Fort Calhoun Station reactor, the rated core power level, including power uprates, has varied between 1420 MWt and 1500 MWt. This variation of rated power level corresponds to a power density range of 76.1 W/cm^3 to 80.4 W/cm^3. Following implementation of the Extended Power Uprate to 1780 MWt, the corresponding core power density will be 94.4 W/cm^3.

Guideline 2 Considering the entire operating lifetime of the reactor, the distance between the top of the active fuel stack and the bottom of the fuel alignment plate (FAP) shall be greater than 12.4 inches for a period of more than two effective full-power years.

Comparison For the Fort Calhoun Station reactor internals and fuel assembly geometry, the nominal distance between the top of the active fuel stack and the bottom of the

FAP was less than 12.4 inches for a period of more than two effective full-power years. The lifetime-average distance is estimated as [ ]^{a,c} inches, with the distance for Cycles 24 and beyond at [ ]^{a,c} inches. Therefore, per the guidance in MRP 2013-025, a plant-specific analysis is required to demonstrate that the fluence above the FAP does not exceed the irradiation embrittlement screening threshold.

1.1.3 Components Located Below the Reactor Core

Based on the discussion provided in MRP 2013-025, plant-specific applicability of MRP-227-A for components located below the reactor core with no further evaluation required is demonstrated by meeting the MRP-227-A, Section 2.4 criteria.

2 REFERENCES

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3. U. S. Nuclear Regulatory Commission Letter, "Summary of November 28, 2012, Category II Public Meeting with the Electric Power Research Institute and Industry Representatives," January 29, 2013. (ADAMS Accession No. ML13009A066)
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7. U. S. Nuclear Regulatory Commission Presentation, "Status of Resolution of MRP-227-A Action Items 1 and 7," June 5, 2013. (ADAMS Accession No. ML13154A152)
8. [

   ]a,c.
9. Electric Power Research Institute Materials Reliability Program Letter MRP 2013-025, "MRP-227-A Applicability Template Guideline," October 14, 2013. (ADAMS Accession No. ML13322A454)
10. U. S. Nuclear Regulatory Commission Document, "Office of Nuclear Reactor Regulation Evaluation of WCAP-17780-P, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering [(CE)] and Westinghouse [Electric Company (Westinghouse)] Pressurized Water Reactor Designs," and MRP-227-A, Applicability Guidelines for CE and Westinghouse Pressurized Water Reactor Designs," November 10, 2014. (ADAMS Accession No. ML14309A484)
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