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U. S. Nuclear Regulatory Commission
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Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant – Units 1 and 2
Response to Request for Additional Information Regarding
License Aging Management Program

References:

1. SNC Letter NL-15-1507, *Joseph M. Farley Nuclear Plant – Units 1 and 2 License Renewal Commitment Item 6,* dated August 12, 2015.

2. NRC Letter, *Joseph M. Farley Nuclear Plant, Units 1 and 2 - Request for Additional Information Regarding License Aging Management Program (CAC NOS. MF8730 AND MF8731),* dated January 19, 2017. (ML17010A014)

Ladies and Gentlemen:

By letter dated August 12, 2015, the Southern Nuclear Operating Company, Inc. (SNC), submitted an aging management program (AMP) for the reactor vessel internals per License Renewal Commitment Item 6 for Joseph M. Farley Nuclear Plant (FNP), Units 1 and 2 (Reference 1).

The Materials Reliability Program (MRP)-227-A report, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," and its supporting reports were used as technical bases for developing FNP units' AMP and for the U.S. Nuclear Regulatory Commission (NRC) staff's review.

The NRC staff determined that a request for additional information regarding the aging management program is needed as discussed in their letter dated January 19, 2017 (Reference 2). The enclosure provides the requested information.

This letter contains no new NRC commitments. If you have any questions, please contact Ken McElroy at 205.992.7369.

Mr. C. R. Pierce states he is Regulatory Affairs Director of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and, to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

C. R. Pierce

C. R. Pierce
Regulatory Affairs Director

CRP/GLS/lac

Sworn to and subscribed before me this 13th *day of* February *, 2017.*

Notary Public

My commission expires: 1-2-2018

Enclosure: Response to Request for Additional Information – MRP-227-A

cc: Southern Nuclear Operating Company
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. D. G. Bost, Executive Vice President & Chief Nuclear Officer
Mr. D. R. Madison, Vice President – Farley
Mr. M. D. Meier, Vice President – Regulatory Affairs
Mr. B. J. Adams, Vice President – Engineering
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U. S. Nuclear Regulatory Commission
Ms. C. Haney, Regional Administrator
Mr. S. A. Williams, NRR Senior Project Manager – Farley
Mr. P. K. Niebaum, Senior Resident Inspector – Farley

Alabama Department of Public Health
David Walter, Director, Alabama Office of Radiation Control

Joseph M. Farley Nuclear Plant – Units 1 and 2
Response to Request for Additional Information Regarding
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Enclosure

Response to Request for Additional Information – MRP-227-A

By letter dated August 12, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML 15226A227), Southern Nuclear Operating Company (SNC) submitted an aging management program (AMP) for the reactor vessel internals per License Renewal Commitment Item 6 for Joseph M. Farley Nuclear Plant (FNP), Units 1 and 2.

The U.S. Nuclear Regulatory Commission (NRC) staff is reviewing the submittal and has determined that the additional information is needed to complete its review.

RAI No. 1 Introduction:

Applicant/Licensee Action Item (AI) 1 was addressed in the staff's safety evaluation of the MRP-227-A report. AI 1 is related to plant-specific applicability of MRP-227-A to Combustion Engineering (CE) and Westinghouse RVI components. The MRP had issued generic guidelines, MRP-2013-025, "MRP-227-A Applicability Template Guidelines," (ADAMS Accession No. ML 13322A454) to address two issues related to AI 1. These two issues are:
(a) effect of cold work on the occurrence of stress corrosion cracking in RVI components, and, (b) 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. In this context, the staff requests that the licensee provide response to the following RAIs.

RAI No. 1(a):

(a) Do the FNP units' RVI components 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?. The staff requests that the licensee provide the plant-specific information on the extent of cold work on the RVI components. The licensee can apply "Option 1" or "Option 2," as addressed in Appendix A of MRP-2013-025. If "Option 2" is applicable to FNP units, 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.

SNC Response to RA No. 1(a):

FNP participated in Pressurized Water Reactor Owner's Group (PWROG) generic work which led to the publication of topical report PWROG-15105-NP, PWR RV Internals Cold-Worked Assessment. This report provides a basis for providing reasonable assurance that highly cold worked (i.e. >20%) components do not exist in the Westinghouse (WEC) and Combustion Engineering (CE) designed plants. Specific record searches of the majority of WEC plants over the range of early, median and later plants designs consistently show no use of cold-worked internals materials in non-fastener applications. The material specifications and associated discussion in the PWROG report appendices A and C were compared to material specifications for FNP and are consistent with no

disparate materials specified. It can therefore be reasonably concluded that the generic report is applicable to FNP.

RAI No. 1(b):

(b) Have FNP units 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? If the fuel design complied with the assumptions of MRP-227-A, the following plant-specific values for FNP units should be submitted: (a) active fuel to upper core plate distance; (b) average core power density; and, (c) heat generation figure of merit. If the fuel design did not comply with the assumptions of MRP-227-A regarding core loading/core design, the licensee should provide a technical justification for the application of MRP-227-A criterion to FNP units.

SNC Response to RA No. 1(b):

FNP has not utilized an atypical design or fuel management that would invalidate the core loading/core design assumptions of MRP-227-A. Specifically, the active fuel to upper core plate distance, average core power density and heat generation figure of merit are all within the bounds presented in MRP letter 2013-025.

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 thirty effective full-power years or less and all future operation will use low leakage fuel management.

Comparison - FNP Unit 1 initiated low leakage fuel management strategy in the fifth fuel cycle following 3.08 effective full-power years (EFPY) of operation and has been implementing low leakage core designs since that time. FNP Unit 2 initiated low leakage fuel management strategy in the third fuel cycle following 1.91 effective full-power years (EFPY) 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 remain less than 124 W/cm^3.

Comparison - For the last thirteen operating fuel cycles (Cycles 16 through 28), FNP Unit 1 has been operating at a rated power level of 2775 MWt. For the 157 fuel assembly FNP Unit 1 core geometry, the 2775 MWt power level corresponds to a core power density of 104.5 W/cm^3. For the last thirteen operating fuel cycles (Cycles 13 through 25), FNP Unit 2 has been operating at a rated power level of 2775 MWt. For the 157 fuel assembly FNP Unit 2 core geometry, the 2775 MWt power level corresponds to a core power density of 104.5 W/cm^3. FNP may implement

a Measurement Uncertainty Recapture (MUR) power uprate of less than 2% within the next several years. However, even a 2% increase in core power density beyond 104.5 W/cm3 would still be far below 124 W/cm^3 during anticipated future operation.

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 thirteen operating fuel cycles at FNP Unit 1, the HGR-FOM at key baffle locations has ranged between 42.43 to 67.27 W/cm^3. For the last thirteen operating fuel cycles at FNP Unit 2, the HGR-FOM at key baffle locations has ranged between 43.19 to 71.22 W/cm^3. The violation of the 68 W/cm^3 criteria takes place only late in Cycle 22. With low leakage fuel management being implemented within the first 10 years instead of the first 30 years, the integrated peripheral power is still quite conservative relative to the MRP. This range of HGR-FOM is representative of anticipated future operation. Therefore, FNP Units 1 and 2 do not exceed this limit for more than two years, which demonstrates compliance with the applicability of MRP-227-A.

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 remain less than 124 W/cm^3 for a period of more than two effective full-power years.

Comparison - Over the operating lifetime of the FNP Unit 1 and FNP Unit 2 reactors, the rated core power level, including power uprates, has varied between 2652 MWt and 2775 MWt. This variation of rated power level corresponds to a power density range of about 98.4 W/cm^3 to 104.5 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 upper core plate (UCP) shall not be less than 12.2 inches for a period of more than two effective full-power years.

Comparison - For the FNP Unit 1 and Unit 2 reactor internals and fuel assembly geometry, the nominal distance between the top of the active fuel stack and the bottom of the upper core plate (UCP) averaged over the first 20 fuel cycles of operation was 16.545 inches. The minimum distance between the top of the active fuel and the bottom of the UCP would be 12.535 inches accounting in maximum irradiation growth through the cycle. Thus during that period of time the nominal distance between the UCP and the top of the active fuel was not less than 12.2 inches for an operating period of more than two effective full-power years.

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.

RAI-No. 2:

Action Item 7 of the staff's SE for MRP-227-A addresses irradiation embrittlement (IE) in austenitic stainless steel components in the lower support columns (LSCs) of the Westinghouse units. Functionality of the LSCs would be affected if the structural integrity of these columns is compromised due to IE. The industry developed a functionality report, Pressurized Water Reactor Owner's Group (PWROG)-14048-P, Revision 0, "Functionality Analysis: Lower Support Columns [LSC]," which was submitted to staff for information. The staff assessed this report (ADAMS Accession No. ML 15334A462), and based on its assessment, the staff is requesting the following RAI:

The NRC staff has determined that the flaw tolerance analysis contained in report PWROG 14048-P utilized conservative assumptions to demonstrate that the likelihood of failure of the LSCs is low during the period of extended operation (PEO). It is reasonable to infer that the functionality of the LSCs will be maintained during the PEO if the likelihood of failure of the LSCs is shown to be low. Therefore, the staff requests the licensee to demonstrate how the flaw tolerance analysis in PWROG-14048-P is applicable to the FNP units' LSCs. The flaw tolerance analysis should contain plant specific parameters (such as LSC geometry and number of LSCs) and conditions (such as loading conditions and LSC stresses). If the licensee determines that PWROG-14048-P is not applicable to the FNP's LSCs or chooses not to apply it, the staff requests that the licensee identify its approach to demonstrating that the functionality of the LSCs will be maintained during the PEO.

SNC Response to RAI-No. 2:

The flaw tolerance analysis performed in PWROG-14048-P is applicable to FNP Unit 1 and 2. The demonstration that this report applies to FNP is to be published as part of a generic industry effort to extend the flaw tolerance analysis to the entire CE/WEC domestic fleet. The PWROG Materials Committee authorized this additional analysis in 2016. As of this date, the technical work has been completed and the results show that all but one domestic PWR are bounded by the results and conclusion. Both FNP units' are included in the list of plants that are bounded by the draft revision to PWROG-14048-P, therefore the results and conclusions are applicable to FNP Unit 1 and 2. The result of this additional work will be published in a revision to PWROG-14048-P. The revision is currently under review by the PWROG materials committee members as a "final draft" with an expected publication by the end of March 2017. SNC participates in this PWROG committee and has reviewed the draft. The conclusion that both FNP units are bounded has been verified by SNC review of the draft, by recent

PWROG meeting materials, and re-confirmed by emails with the technical lead at WEC. Similarly, it has been confirmed that the intent of the PWROG is to provide the revised version of PWROG-14048-P to the staff for information.