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

January 19, 2017

Mr. C. R. Pierce
Regulatory Affairs Director
Southern Nuclear Operating Co., Inc.
P.O. Box 1295, Bin 038
Birmingham, AL 35201-1295

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AGING MANAGEMENT PROGRAM (CAC NOS. MF8730 AND MF8731)

Dear Mr. Pierce:

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, Units 1 and 2 (FNP).

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 has determined that additional information is needed as discussed in the Enclosure. We request that SNC respond within 30 days of the date of this letter. Please note that the NRC staff's review is continuing and further requests for information may be developed.

Sincerely,

Shawn Williams

Shawn Williams, Project Manager
Plant Licensing Branch, II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-348, 50-364

Enclosure:
Request for Additional Information

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

AGING MANAGEMENT PROGRAM FOR REACTOR VESSEL INTERNALS

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NOS. 50-348 AND 50-364

By letter dated August 12, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15324A297), Southern Nuclear Operating Company (the licensee) submitted an aging management program (AMP) for the reactor vessel internals per License Renewal Commitment Item 6 for Joseph M. Farley Nuclear Plant, Units 1 and 2 (FNP).

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:

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. ML13322A454) 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.

(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.

(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.

Enclosure

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. ML15334A462), 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.

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/RA/
Shawn Williams, Project Manager
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ADAMS Accession No.: ML17010A014 ***by internal memo**

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