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

June 2, 2017

Mr. James J. Hutto
Regulatory Affairs Director
Southern Nuclear Operating Company, Inc.
P.O. Box 1295 / Bin 038
Birmingham, AL 35201-1295

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 – STAFF EVALUATION OF THE REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM (CAC NOS. MF8730 AND MF8731)

Dear Mr. Hutto:

By letter dated August 12, 2015, as supplemented by letter dated February 13, 2017, Southern Nuclear Operating Company, Inc., (SNC, the licensee) submitted a reactor vessel internals (RVI) aging management program (AMP) for the Joseph M. Farley Nuclear Plant (FNP), Units 1 and 2. The licensee submitted the RVI AMP with the intent to meet the license renewal Commitment No. 6 addressed in Appendix A of NUREG-1825, "Safety Evaluation Report Related to the License Renewal of the Joseph M. Farley, Units 1 and 2." The RVI AMP is based on Electric Power Research Institute Technical Report No. 1022863 "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)."

The U.S. Nuclear Regulatory Commission (NRC) staff's review of the FNP, Units 1 and 2, RVI AMP is provided in the enclosed staff evaluation. The NRC staff concludes that the licensee has adequately addressed Action Items 1, 2, 3, 4, 5, 6, and 8. However, Action Item 7 remains open. Action Item 7 requires licensees of Westinghouse reactors to develop plant-specific analyses applied for their facilities to demonstrate that cast austenitic stainless steel lower support columns will maintain their function during the period of extended operation. By letter dated February 13, 2017, the licensee stated in its response to Request for Additional Information (RAI)-2, that the industry revised the Pressurized Water Reactor Owner's Group (PWROG)-14048-P report, which includes additional analysis that confirms that with the exception of one unit, all Combustion Engineering (CE) and Westinghouse units are bounded by the revised analysis. In a letter dated March 1, 2017, the PWROG submitted this report to the NRC staff. The NRC staff is currently reviewing this report for its validity in providing adequate bounding functionality analyses for CE and Westinghouse fleets (with the exception of one unit). The NRC staff's review of the PWROG-14048-P report will be used in assessing the applicability of the report for FNP units. Therefore, Action Item 7 will remain open until the staff completes its assessment of PWROG-14048-P report.

The NRC staff's evaluation of the RVI AMP does not reduce, alter, or otherwise affect the current American Society of Mechanical Engineers Boiler Pressure &Vessel Code, Section XI, Inservice Inspection (ISI) requirements, or any specific licensing requirements related to ISI. The licensee must follow the implementation requirements as defined in Section 7.0, "Implementation Requirements," of MRP-227-A, which require that the NRC be notified of any deviations from the "needed" requirements.

If you have any questions concerning this matter, please contact the Project Manager, Shawn Williams, at 301-415-1009 or by e-mail at Shawn.Williams@nrc.gov.

Sincerely,

Shawn Williams

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

Docket Nos. 50-348 and 50-364

Enclosure:
Staff Evaluation

cc w/enclosure: Distribution via Listserv

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 – STAFF EVALUATION OF THE REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM (CAC NO. MF8730 AND MF8731) DATED JUNE 2, 2017

DISTRIBUTION:
PUBLIC
LPL2-1 R/F
RidsACRS_MailCTR Resource
RidsNrrKGoldstein Resource
RidsNrrPMFarley Resource
RidsNrrDorlLpl2-1 Resource
RidsRgn2MailCenter Resource
RidsNrrDeEvib Resource
GCheruvenki, NRR

ADAMS Accession No. ML17135A252 *by e-mail

OFFICE	NRR/DORL/LPL2-1/PM	NRR/DORL/LPL2-1/LA	NRR/DE/EVIB/BC*
NAME	SWilliams	KGoldstein (PBlechman for)	DRudland
DATE	05/03/17	05/30/17	05/17/17
OFFICE	NRR/DORL/LPL2-1/BC	NRR/DORL/LPL2-1/PM	
NAME	MMarkley	SWilliams	
DATE	06/02/17	06/02/17	

OFFICIAL RECORD COPY

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

STAFF EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NOS. 50-348 AND 50-364

1.0 INTRODUCTION

By letter dated August 12, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15226A225), as supplemented by letter dated February 13, 2017 (ADAMS Accession No. ML17044A437), Southern Nuclear Operating Company, Inc. (SNC, the licensee), submitted a reactor vessel internals (RVI) aging management program (AMP) for the Joseph M. Farley Nuclear Plant (FNP), Units 1 and 2. The licensee applied Electric Power Research Institute (EPRI) Technical Report (TR) No. 1022863. "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," as the technical bases for developing FNP's AMP (ADAMS Package Accession No. ML120170453). The licensee submitted the AMP with the intent to meet the license renewal (LR) Commitment No. 6 addressed in Appendix A of NUREG-1825, "Safety Evaluation Report Related to the License Renewal of the Joseph M. Farley Nuclear Plant, Units 1 and 2" (ADAMS Accession No. ML051250126) The AMP included Inspection and Evaluation (I&E) guidelines (inspection plan) for the RVI components at FNP.

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 LR process. The regulation at 10 CFR 54.21, "Contents of application-technical information," requires that each application for LR contain an integrated plant assessment (IPA) and an evaluation of time limited aging analyses. In addition, 10 CFR 54.22, "Contents of application-technical specifications," requires that a license renewal application (LRA) include any technical specification changes or additions necessary to manage the effects of aging during the period of extended operation (PEO) as part of the LRA.

Structures and components subject to an aging management review (AMR) shall encompass those structures and components that (1) perform an intended function, as described in 10 CFR 54.4, "Scope," 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 the American Society of Mechanical Engineers (ASME) Boiler Pressure & Vessel Code (ASME Code), Section XI) and those RVI components that serve an intended LR safety function pursuant to the 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 required to be subject to an AMP, as defined by the criteria set in 10 CFR 54.21(a)(1).

The plant-specific IPA shall identify and list those structures and components subject to an AMR 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 PEO as required by 10 CFR 54.29(a).

The NRC staff's evaluation and basis for approving the previous LRA for FNP, Units 1 and 2, in accordance with 10 CFR Part 54, is documented in NUREG-1825. The FNP's inspection plan was developed by the licensee and is based on MRP-227-A. MRP-227-A summarized most recent industry recommended I&E guidelines for pressurized-water reactor (PWR) RVI components. Revision 1 to the safety evaluation (SE) for MRP-227-A was issued on December 16, 2011 (ADAMS Accession No. ML11308A770), with eight applicant/licensee action items. The applicant/licensee action items were specified for applicant/licensees to address plant-specific issues, which could not be resolved generically in the December 16, 2011, MRP-227-A SE.

3.0 TECHNICAL EVALUATION

The NRC staff's evaluation and approval of the RVI AMP Program, as submitted in the LRA for FNP, Units 1 and 2, is documented in Section 3.0.3.2.7, "Reactor Vessel Internals Program," of NUREG-1825. However, the staff's prior approval of the AMP, was tied to the fulfillment of the following actions in Commitment No. 6 of the Updated Final Safety Analysis Report (UFSAR) supplement commitment table for the LRA:

> The new FNP Reactor Vessel Internals (RVI) Program will be implemented prior to entering the period of extended operation to provide an integrated inspection program that addresses the reactor internals. It will supplement the inspection requirements of ASME Section XI, IWB Category B-N-3 to ensure that aging effects do not result in a loss of intended function of internal components during the period of extended operation.
>
> SNC will continue to participate in industry initiatives intended to clarify the nature and extent of aging mechanisms potentially affecting the FNP reactor internals. SNC will incorporate the results of these initiatives into the RVI Program.
>
> FNP will submit an inspection plan for the RVI Program for NRC review and approval at least 24 months prior to entering the periods of extended operation for the FNP units.
>
> The RVI Program will be consistent with the NUREG-1801 Programs XI.M13 and XI.M16....

The NRC staff verified that the licensee has been following and participating in the EPRI programmatic aging activities for PWR RVI components and that licensee's submittal includes an updated version of the RVI Program that is based on the program element criteria in generic aging lessons learned (GALL) AMP Section XI.M16A, "PWR Vessel Internals," as updated in

NRC Final License Renewal Interim Staff Guidance (ISG) Document No. LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors" (ADAMS Accession No. ML12270A436).

The program in GALL AMP Section XI.M16A, as updated in LR-ISG-2011-04, provides the NRC staff's most recent aging management criteria for PWR RVI components and supersedes the previous set of aging management guidelines for PWR RVI components in the versions of GALL AMP Section XI.M13 and GALL AMP Section XI.M16 in NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL)" Report (ADAMS Accession No. ML103490041). Therefore, based on these verifications, the NRC staff finds that the licensee's submittal fulfills the actions in Commitment No. 6 for participating in the industry's aging initiatives on aging of PWR RVI components and for supplementing the RVI Program to be consistent with NRC's current AMP for the components, which, for initial plant license extensions, is now given in the version of GALL AMP Section XI.M16A in LR-ISG-2011-04.

The NRC staff also verified that the applicant included an inspection plan for the RVI components (RVIIP) in the submittal of August 12, 2015. The staff noted that the RVIIP was consistent with the augmented I&E criteria specified. MRP-227-A and the applicant's augmented I&E criteria for these will supplement any mandated ASME Code, Section XI, IWB Category B-N-3 inspection criteria for the components. Therefore, based on these verifications, the staff concludes that the licensee's submittal fulfills the actions in Commitment No. 6 for submitting an RVIIP for staff approval and for using the RVI Program to supplement the aspects of the licensee's inservice inspection (ISI) program that applies to the RVI B-N-3 components.

The FNP, Units 1 and 2, RVI AMP in the licensee's August 12, 2015, submittal contains Section 1, "Purpose"; Section 2, "Background"; Section 3, "PWR Vessel Internals Program Owner"; Section 4, "Description of the Farley Nuclear Plant [Unit 1 and 2] Reactor Internals Aging Management Programs and Industry Programs"; Section 7, "Program Enhancement and Implementation Schedule"; Section 8, "Implementing Documents"; Appendix A, "Illustrations," and Appendix B, "Farley [Unit 1 and Unit 2] License Renewal Aging Management Review Summary Table." These Sections contain no specific technical information, which would affect the review and approval of the FNP's inspection plan. Therefore, the focus of the NRC staff's evaluation is related to: Section 5, "Farley Nuclear Plant Reactor Internals Aging Management Program Attributes RVI Aging Management Program Attributes," which includes operating experience (OE), Appendix C, "MRP-227-A Augmented Inspections" and, Section 6.2, "Demonstration of Applicant/Licensee Action Item Compliance to SE on MRP-227-A, Revision 0."

3.1 FNP RVI Aging Management Program Attributes: Section 5.0

Licensee Evaluation

In the submittal dated August 12, 2015, the licensee stated that the AMP for the RVI components at FNP is in compliance with all the attributes addressed in NUREG-1801, Revision 2, Section XI.M16A. The licensee further stated that FNP fully utilized the GALL process contained in NUREG-1801 in performing an AMR of the RVI components in the LR process. The AMP for the RVI components at FNP includes consideration of the augmented inspections identified in MRP-227-A and fully meets the requirements of Section XI.M16A in the GALL Report.

NRC Staff Evaluation

The NRC staff reviewed the ten program elements for FNP, Units 1 and 2, AMP and compared them with the program elements in the version of GALL AMP Section XI.M16A that was updated in NRC ISG LR-ISG-2011-04, and concluded the following: (1) all ten program elements described in Section 5.0 of the licensee's submittal dated August 12, 2015, are consistent with the program elements addressed in Section XI.M16A, in the GALL Report, Revision 2, (2) all the program elements included extensive discussions on the implementation of MRP-227-A, (augmented inspections) I&E guidelines and ASME Code, Section XI, ISI criteria, (3) the licensee's discussions included applicability of its Action Items addressed in the staff's SE for MRP-227-A, and (4) the licensee complied with the provisions addressed in MRP-227-A, specifically with respect to the following program elements: (a) parameters monitored or inspected, (b) detection of aging effects, and (c) monitoring and trending.

3.2 Operating Experience: Section 5.10

Section 5.10 of the licensee's submittal dated August 12, 2015, addressed OE related to aging degradation, reported to date, in RVI components at FNP. In this context, the licensee identified aging degradation mechanism for one RVI component at the FNP units: control rod guide tube (CRGT) split pins, which are susceptible to primary water stress corrosion cracking (PWSCC). In Section 5.10, the licensee listed industry-wide OE with respect to an active aging degradation in CRGT guide cards (wear) and irradiation assisted stress corrosion cracking (IASCC) in baffle bolts.

3.2.1 CRGT Split Pins

Licensee Evaluation

CRGT split pins at FNP were fabricated from Alloy X-750, which did not receive high temperature heat treatment (HTH). Alloy X-750 material without undergoing HTH is more susceptible to PWSCC. In Section 5.10 of the submittal dated August 12, 2015, the licensee stated that as part of its corrective action, it replaced Alloy X-750 with Type 316 stainless steel material, which has superior resistance to PWSCC.

NRC Staff Evaluation

The licensee's action in replacing the split pins with Type 316 stainless steel material that is more resistant to PWSCC, provides reasonable assurance that the aging degradation in this component is being monitored adequately by the licensee during the PEO.

3.2.2 Baffle Bolts

Licensee Evaluation

Based on previous OE, Type 347 baffle bolts in PWR units, are susceptible to IASCC. Each FNP unit has a total of 1088, Type 316, cold worked bolts. During the fall 1998 outage, and fall 1999 outage, the licensee proactively replaced 211 baffle bolts in FNP Unit 1, and 203 baffle bolts in FNP Unit 2. The material used for the replacement was Type 316 bolts, which are less susceptible to IASCC than Type 347 material. In addition, the licensee reversed the secondary coolant flow pattern from "downflow" to "upflow" resulting in reduction in flow induced vibration initiated by reactor coolant crossflow jetting through the joints between the baffle plates. In addition, it reduces the pressure difference across the baffle, which reduces the loading on the bolt. With respect to baffle bolts at FNP Unit 1, the licensee stated that it will perform initial inspections (ultrasonic testing—UT) of these bolts during the spring 2018 outage. Similarly, for FNP Unit 2, the licensee stated that it will perform initial inspections of these bolts during the spring 2022 outage.

NRC Staff Evaluation

The NRC staff noted that baffle bolt replacement in the FNP units with Type 316 materials in conjunction with modified bolt design and upflow conversion, which reduces loads on the baffle bolts, provides reasonable assurance that the licensee is adequately managing the aging degradation in these bolts. In order to maintain the adequacy of the AMP for these bolts, the licensee will be inspecting the baffle bolts at the FNP units during the PEO. The staff noted that the inspection frequency and the inspection techniques that would be used for these bolts are consistent with I&E guidelines addressed in MRP-227-A. Therefore, the staff concludes that the licensee's proposed plan provides reasonable assurance that the aging degradation in this component is being monitored adequately by the licensee during the PEO at the FNP units.

Based on the emerging OE associated with IASCC on these bolts, on August 1, 2016, Westinghouse issued Nuclear Safety Advisory Letter (NSAL)-16-1, Revision 1, "Baffle-Former Bolts," which provides recommendations to manage the aging degradation in baffle-former bolts (ADAMS Accession No. ML16225A729). Recommendations addressed in NSAL-16-1 were developed based on the OE of the baffle-former bolt failures. Each plant that experienced bolt failures was binned under a tier based on the extent of aging degradation of the bolts. The FNP units were binned under "Tier 3" category, and the NRC staff expects that the licensee will follow the guidelines recommended in Nuclear Energy Institute (NEI) NEI 03-08, "Guidelines for the Management of Materials Issues," and NSAL-16-1.

3.2.3 Control Rod Guide Cards

Licensee Evaluation

In Table 7-1 of the licensee's submittal dated August 12, 2015, the licensee indicated that the control rod guide cards at the FNP units are to be inspected no later than two refueling outages from the beginning of the PEO for each unit. The licensee further stated that it will inspect control rod guide cards in accordance with MRP-227-A and WCAP-17451-P, "Reactor Internals Guide Tube Wear." To date, the licensee did not identify any active aging degradation in guide cards at FNP units.

NRC Staff Evaluation

The NRC staff reviewed the licensee's evaluation of the guide cards and determined that the licensee's plan to follow the guidelines addressed in MRP-227-A and WCAP-17451-P provides reasonable assurance that the AMP for the guide cards would be effectively implemented at the FNP units. The staff's basis for this conclusion is addressed below. The staff's assessment of WCAP-17451-P was included in the staff's SE for the WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," December 2009 (ADAMS Accession No. ML101460157). In the SE for the WCAP-17096-NP, Revision 2 (ADAMS Accession No. ML16061A243), the staff stated that the evaluation methodology and acceptance criteria for the guide cards is acceptable because it provides a methodology for measuring wear that is based on ensuring functionality of the rod cluster control assemblies, and the acceptance criteria provide margin for future wear. In addition, the WCAP-17451-P report provides a rigorous and comprehensive basis for the methods and criteria for guide card wear evaluation. Therefore, the staff concludes that the licensee's proposed plan provides reasonable assurance that the aging degradation in guide cards is being monitored adequately by the licensee during the PEO at the FNP units. The staff noted that the MRP is proactively reviewing emerging issues related to this issue.

3.2.4 Clevis Insert Bolts

Licensee Evaluation

In the August 12, 2015, submittal, Appendix C, Table C-3 "MRP-227-A Components Existing Inspection and Aging Management Programs Credited in Recommendations for Westinghouse-Designed Internals," the licensee stated that it is implementing I&E guidelines for the clevis insert bolts, which are categorized as "Existing Programs," ASME Code, Section XI Inservice Inspection Program. This program is credited for managing aging due to wear only in these bolts.

NRC Staff Evaluation

The NRC staff noted that page A-2 of Appendix A to MRP-227-A stated that failures of Alloy X-750 clevis insert bolts were reported by one licensee. Alloy X-750 material was used for clevis insert bolts in Westinghouse units. If these bolts did not receive HTH they might be susceptible to PWSCC. Appendix A to MRP-227-A indicates that failures of Alloy X-750 clevis insert bolts were reported by one Westinghouse-designed plant in 2010. Appendix A to MRP-227-A also stated that the most likely cause for failure was PWSCC. Sections 5.10 of FNP-1 and FNP-2 AMPs addressed in the August 12, 2015, submittal stated that during the previous ASME Code, Section XI inspections, no aging degradation in the RVI components was observed.

Based on the information provided in Section 5.10 of the August 12, 2015, submittal, the NRC staff concludes that the AMP of the clevis inserts is adequately managed at FNP because: (1) no relevant indications were found to date the clevis insert bolts; (2) the inspected area of coverage is likely be one hundred percent, therefore, the staff believes that any aging degradation in clevis insert assembly would be identified during the examinations during the PEO. Therefore, the staff considers that the licensee's AMP for the clevis inserts provides reasonable assurance that the clevis insert's safety function would be maintained during the PEO.

3.2.5 Materials Susceptible to Degradation

Operating experience in the PWR fleet to date, identified that the following nickel-based and stainless steel alloys are susceptible to some of the aging degradation mechanisms addressed in MRP-227-A: Nickel based alloys (i.e., Alloy 600), weld metals (i.e., Alloy 82 and 182), Alloy A-286, ASTM A 453 (Grade 660, Condition A or B), Type 347 stainless steel material (excluding baffle-former bolts), precipitation hardened stainless steel materials (i.e., 17-4 and 15-5) and Type 431 stainless steel material.

NRC Staff Evaluation

The NRC staff is concerned about the functionality of the RVI components that were fabricated from the aforementioned materials. MRP-227-A contains a list of the type of materials used in RVI components, which was originally developed in a supporting technical document, EPRI Topical Report, Revision 0, "Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Designs (MRP-191)" (ADAMS Accession No. ML091910130). If the materials used in the FNP units are different from MRP-191, they require new I&E guidelines other than MRP-227-A. The licensee reviewed its list of materials used in the RVI components at the FNP units. In the August 12, 2015, submittal, Section 6.2.2, "SE Applicant/Licensee Action Item 2: PWR Vessel Internal Components within the Scope of License Renewal," the licensee stated that most of RVI materials, with the exception of cast austenitic stainless steel (CASS) materials, used at the FNP units are consistent with the materials specified in MRP-191. This issue is discussed in detail in Section 3.4.2 of this SE. Since the RVI materials used at FNP are consistent with that of MRP-191, I&E guidelines used in MRP-227-A are still valid for these units. The following nickel base RVI components made from Alloy X-750 were used in the FNP units and they are susceptible to PWSCC: (1) CRGT guide cards/plates; (2) CRGT tube support pins; (3) Clevis insert bolts. CRGT guide cards/plates and Clevis insert bolts are inspected in accordance with MRP-227-A, and ASME Code, Section XI criteria, respectively. CRGT tube support pins were replaced with Type 316 stainless steel material, which is more resistant to PWSCC. Clevis inserts were fabricated with Alloy 600 material and they were welded to the reactor vessel and these welds and the clevis inserts are routinely inspected per ASME Code, Section XI criteria.

Based on the review of the AMP for the materials susceptible to degradation, the NRC staff determined that the licensee has demonstrated that it is adequately managing the aging degradation in the RVI components at the FNP units during the PEO. The staff's conclusion is based on the following technical bases: (1) the nickel-based materials used in RVI components at the FNP units would be inspected in accordance with MRP-227-A I&E guidelines, which apply to non-ASME Code, Section XI components; (2) similarly, the ASME Code, Section XI RVI components would be inspected in accordance with ASME Code, Section XI criteria for the ASME Code Section XI vessel attachment welds; (3) to date, no active aging degradation was identified in these components. Therefore, the staff concludes that aging degradation in RVI components described above are monitored adequately using routine inspections as required by ASME Code, Section XI criteria (for ASME Code, Section XI components), and by the criteria addressed in I&E guidelines of MRP-227-A (for non-ASME Code, Section XI components).

3.3 Inspection and Evaluation Guidelines Consistent with MRP-227-A

August 12, 2015, submittal, Appendix B, includes the "License Renewal Aging Management Review Summary Table" which is consistent with MRP-227-A. In this section, the licensee listed RVI components that are to be inspected per MRP-227 I&E guidelines, and RVI components that are scheduled to be inspected per ASME Code, Section XI. Consistent with AMP Section XI.M16A in the GALL report, the licensee included a partial list of RVI components, which are part of its ASME Code, Section XI, ISI program at FNP. In addition, in Section 4.1.2 of the submittal dated August 12, 2015, the licensee stated that it implemented ASME Code, Section XI, 2001 Edition with 2003 Addenda for the FNP units.

The NRC staff reviewed Appendix B and Section 4.1.2 of the licensee's submittal dated August 12, 2015, and concludes that ASME Code, Section XI, RVI components will be inspected in accordance with requirements of the ASME Code, Section XI criteria. Based on the information provided, the staff concludes that the licensee's compliance with ASME Code, Section XI criteria for ASME Code, RVI components provides reasonable assurance that any aging degradation in the ASME Code, RVI components (B-N-1 and B-N-2) would be monitored adequately by the licensee during the PEO.

3.4 Applicant/Licensee Action Item 1 of the SE for MRP-227-A

3.4.1 Evaluation of the Licensee's Resolution of Action Item 1

Section 4.2.1, "Applicability of FMECA [Failure Modes, Effects, and Criticality Analyses] and Functionality Analysis Assumptions," of the SE for MRP-227-A states, in part:

> Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the 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....

To resolve the generic issue of the information needed from licensees to resolve Action Item 1, 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 PWR RVI.

A summary of the proprietary meeting presentations and supporting proprietary generic design bases information are 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." This report provides background on the proprietary design information regarding variances in stress, fluence, and temperature in the RVI components that were designed by Westinghouse and CE to support NRC reviews of utility submittals to demonstrate plant-specific applicability of MRP-227-A.

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:

Question 1:

Do the plants have non-weld or bolting austenitic stainless steel components with 20 percent cold work or greater, and, if so, do the affected components have operating stresses greater than 30 kilo pounds per square inch (ksi)? (If both conditions are true, additional components may need to be screened in for stress corrosion cracking (SCC).)

Question 2:

Do the plants 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" (MRP-227-A Applicability Guidelines), a non-proprietary document (ADAMS Accession No. ML13322A454) containing guidance for responding to the two questions above.

The NRC staff assessed, and concludes 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 non-proprietary assessment of WCAP-17780-P and MRP Letter 2013-025 can be found at ADAMS Accession No. ML14309A484. The guidance in MRP Letter 2013-025 provides an acceptable basis for licensees to respond to the generic Questions 1 and 2 addressed above. With respect to Question 1, the staff issued RAI-1(a), which is addressed below.

In RAI-1(a) by letter dated January 19, 2017 (ADAMS Accession No. ML17010A014), the NRC staff requested that the licensee provide information addressed in the aforementioned Question 1 (i.e., effect of cold work on the occurrence of SCC in RVI components at FNP units).

RAI-1(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 the MRP Letter 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.

By letter dated February 13, 2017, the licensee provided the following response. The licensee stated that it is adopting the generic guidance related to the effect of cold work on SCC in PWR RVI components. This guidance was developed by the PWR Owner's Group (PWROG) as topical report, PWROG-15105-NP, "PWR RV Internals Cold Work Assessment," and was submitted for the NRC staff's assessment. Furthermore, the licensee stated that it compared the material specifications of the FNP units' RVI components with Appendices A and C addressed in the PWROG-15105-NP report and concluded the following. The material specifications in the FNP units are consistent with the report with no deviations. Based on this comparison, the licensee concluded that the generic report is applicable to FNP and no cold work greater than 20 percent was imposed on RVI components at the FNP units.

The NRC staff reviewed the licensee's evaluation and concluded the following: (1) the staff's assessment of PWROG-15105-NP states that non-fastener RVI components in Westinghouse units were not subject to cold work greater than 20 percent during the construction phase. This conclusion was based on the PWROG's review of material specifications (Appendix C of the report) and drawings of Westinghouse RVI components, (2) the licensee's material specifications of the RVI components at FNP units are consistent with material specifications listed in Appendix C of the report without any exceptions. Therefore, the staff concludes that non-fastener RVI components at FNP units were not exposed to cold work greater than 20 percent during the construction phase. In the absence of any cold work greater than 20 percent, the staff concludes that the RVI components at FNP units are less susceptible to SCC. The NRC staff considers that the licensee's response to RAI-1(a) is acceptable.

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

(1) The heat generation rate must be ≤ 68 Watts/cm^3.
(2) The maximum average core power density must be less than 124 Watts/cm^3.
(3) The active fuel to upper core plate (UCP) distance must be greater than 12.2 inches.

In RAI-1(b) by letter dated January 19, 2017, the NRC staff requested that the licensee provide the information addressed in aforementioned Question 2 (fuel management) related to the verification of the applicability of MRP-227-A to FNP.

RAI-1(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.

In the letter dated February 13, 2017, the licensee stated that, to date, it complied with the criteria stated above for the essential attributes (i.e., heat generation rate, maximum average core power density, and active fuel distance to UCP). The licensee provided plant-specific values related to heat generation rate, maximum average core power density, and active fuel distance to UCP. The NRC staff reviewed the submitted values and concludes that the licensee complied with the guidelines related to the fuel management issue addressed in MRP Letter 2013-025. Based on this review, the NRC staff considers that the licensee addressed Action Item 1 satisfactorily.

3.4.2 Evaluation of the Licensee's Resolution of Action Item 2

Section 4.2.2, "PWR Vessel Internal Components Within the Scope of License Renewal," of the SE for MRP-227-A states, in part:

> ...each applicant/licensee is responsible for identifying which RVI components are within the scope of LR for its facility. Applicants/licensees shall review the information in...Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RVI components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. If the tables do not identify all the RVI components that are within the scope of LR for its facility, the applicant or licensee shall identify the missing component(s) and propose any necessary modifications to the program defined in MRP-227, as modified by this SE, when submitting its plant-specific AMP. The AMP shall provide assurance that the effects of aging on the missing component(s) will be managed for the period of extended operation....

In the submittal dated August 12, 2015, the licensee stated that it performed the scoping and screening of the RVI components as per the requirements of the LR process. Most RVI materials (with few exceptions—described in the next paragraph) used at the FNP units are consistent with the materials specified in MRP-191, which was used as a technical basis document for the development of I&E guidelines in MRP-227-A. The licensee further stated that for the FNP units, it did not have any modifications to the AMP addressed in MRP-227-A. The following paragraph addresses the issues related to the materials used in some RVI components, which are not consistent with MRP-191.

The licensee stated that for FNP Unit 1 the material used in the CRGT plates was potentially cast austenitic stainless steel (i.e., ASME grade (CF8) material). The material addressed in MRP-191 for CRGT plates was Type 304 wrought stainless steel. Unlike a wrought material, CASS is prone to thermal embrittlement (TE), hence the licensee evaluated the CASS material for its susceptibility to TE. Using the FMECA process, the licensee evaluated the CRGT plates for its susceptibility to TE and concluded that the MRP-227-A aging management strategy is acceptable for the CRGT plates. CASS CF8 grade material was identified in the following RVI components in FNP Units 1 and 2: Instrumentation conduits and support brackets; clamps; terminal blocks; and conduit straps. The licensee performed evaluation of these RVI components using the FMECA process and concluded that these components will be categorized under "No Additional Measure." RVI components classified under this category are not susceptible to any aging degradation, hence no inspections are necessary during the PEO. Based on this evaluation, the licensee concluded that no revisions are required to the AMP for the CASS RVI components at FNP.

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 FNP were screened in due to the usage of different type of materials. Hence, the materials used at FNP units are consistent with MRP-191. CASS materials (stated above) were evaluated under the FMECA process and was categorized under "No Additional Measure." Therefore, they are not susceptible to TE; (3) the licensee complied with the guidelines addressed in Action Item 1, and (4) FNP units do not have any martensitic stainless steel or martensitic precipitation hardened stainless steel in their RVI components. Details of the staff's evaluation of the Action Item 1 are addressed in Section 3.4.1 of this SE. Based on this assessment, the NRC staff considers that the licensee addressed Action Item 2 satisfactorily.

3.4.3 Evaluation of the Licensee's Resolution of Action Item 3

Section 4.2.3, "Evaluation of the Adequacy of Plant-Specific Existing Programs," of the SE for MRP-227-A states, in part:

> ...applicants/licensees of....Westinghouse are required to perform plant-specific analysis either to justify the acceptability of an applicant's/licensee's existing programs, or to identify changes to the programs that should be implemented to manage the aging of these components for the period of extended operation. The results of this plant-specific analyses and a description of the plant-specific programs being relied on to manage aging of these components shall be submitted as part of the applicant's/licensee's AMP application. The...Westinghouse component identified for this type of plant-specific evaluation include...Westinghouse guide tube support pins (split pins).

Action Item 3 of the NRC staff's SE for the MRP-227-A report stated that the licensee is required to perform a plant-specific evaluation of its existing program on CRGT support split pins at FNP. The NRC staff evaluation of Action Item 3 is addressed in Section 3.2.1 of this SE.

3.4.4 Evaluation of the Licensee's Resolution of Action Items 4 and 6

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

3.4.5 Evaluation of the Licensee's Resolution of Action Item 5

Section 4.2.5, "Application of Physical Measurements as Part of I&E Guidelines for B&W, CE, and Westinghouse RVI Components," of the SE for MRP-227-A states, in part:

> ...applicants/licensees shall identify plant-specific acceptance criteria to be applied when performing physical measurements required by the NRC-approved version of MRP-227 for loss of compressibility for Westinghouse hold down springs.... The applicant/licensee shall include its proposed acceptance criteria and an explanation of how the proposed acceptance criteria are consistent with the plants' licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation during the period of extended operation as part of their submittal to apply the approved version of MRP-227....

Action Item 5 in the NRC staff's SE for MRP-227-A states that the licensee should identify a plant-specific acceptance criterion to be applied while performing the physical measurement of loss of compressibility for hold-down springs. Hold down springs at FNP were fabricated with 304 austenitic stainless steel material, which is susceptible to loss of preload due to irradiated assisted creep/stress relaxation.

Licensee's Evaluation

In the submittal dated August 12, 2015, the licensee stated that consistent with I&E guidelines addressed in MRP-227-A, it is planning to perform inspections/physical measurements on the hold down springs at FNP Unit 1 between spring 2018 and spring 2021, and FNP Unit 2 between spring 2022 and spring 2025. As part of its corrective action, the licensee will obtain the plant-specific acceptance criteria prior to performing the inspections/physical measurements on the hold down springs at FNP units.

NRC Staff Evaluation

The NRC staff reviewed the licensee's plan and determined that the licensee complied with the requirements of MRP-227-A. These conclusions are based on the following: (1) the licensee is planning to perform inspections/measurement of the hold down springs during PEO; (2) the licensee is planning to develop the acceptance criteria prior to performing physical measurement of the hold down springs. Based on the above, the NRC staff concludes that the issue related to Action Item 5 of the staff's SE for the MRP-227-A is resolved.

3.4.6 Evaluation of the Licensee's Resolution of Action Item 7

Action Item 7 was discussed in Section 3.3.7, "Plant-Specific Evaluation of CASS Components," of the staff's SE for the MRP-227-A, and it requires licensees of Westinghouse reactors to develop plant-specific analyses applied for their facilities to demonstrate that CASS lower support columns (LSCs) will maintain their function during the PEO. These components are subject to irradiation embrittlement (IE) and TE. CASS materials with delta ferrite greater than 20 percent would be susceptible to loss of fracture toughness due to TE. CASS materials that are exposed to neutron fluence value greater than 1×10^{17} n/cm^2 (E>1 MeV) are susceptible to IE. The licensee addressed TE, however, IE in LSCs was not addressed.

Occurrence of TE depends on the ferrite content, which in turn depends on the chemical composition of CASS material, Molybdenum content and casting process. In Table C-2 of the August 12, 2015, submittal, the licensee stated that the LSCs at FNP units are wrought (non-cast) materials, hence, they are not susceptible to TE. However, the NRC staff noted that IE is still an active aging degradation in the LSCs. Functionality of the LSCs would be affected if the structural integrity of these columns is compromised due to IE. To address the functionality of the LSCs, the staff determined that the licensee may use a generic functionality report. The industry developed a functionality report, PWROG-14048-P, Revision 0, "Functionality Analysis: Lower Support Columns," which was submitted to staff for information. The staff assessed this report (ADAMS Accession No. ML15334A462), and based on its assessment, by letter dated January 19, 2017, the staff, issued the following RAI:

RAI-2:

> 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 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.

In response to RAI-2, by letter dated February 13, 2017, the licensee stated that the industry revised the PWROG-14048-P report to include additional analysis, which confirms that with the exception of one unit, all CE and Westinghouse units are bounded by the revised analysis. The licensee reviewed the new revised version of the PWROG-14048-P report and concluded that both FNP units are bounded by this report. In a letter dated March 1, 2017 (ADAMS Accession No. ML17066A266), the PWROG submitted this report to the NRC staff for information. The staff is currently reviewing the report for its validity in providing adequate bounding functionality analyses for CE and Westinghouse fleets (with the exception of one unit). The staff review of the PWROG-14048-P report will be used in assessing the applicability of the report for the FNP units. Therefore, Action Item 7 will remain open until the NRC staff completes its assessment.

3.4.7 Evaluation of the Licensee's Resolution of Action Item 8

Farley's LRA was based on GALL Revision 0 and was submitted prior to the review of MRP-227, Revision 0, and the NRC staff's SE on MRP-227-A. Therefore, Action Item 8 in the staff's SE for MRP-227-A does not apply to the FNP units. Instead, any ties to updating their AMP for consistency with GALL and submitting the RVI I&E guidelines was tied strictly only to the provisions in Commitment No. 6, as specified previously in the UFSAR Supplement commitment table for the previous LRA.

3.4.8 Evaluation of the Conditions in the NRC Staff's SE for MRP-227-A

The NRC staff noted that the licensee incorporated all the seven conditions addressed in the staff's SE for MRP-227-A. Therefore, the NRC staff concludes that the licensee had adequately addressed all the conditions stated in the NRC staff's SE for MRP-227-A.

4.0 CONCLUSION

The NRC staff concludes that the licensee has adequately addressed Action Items 1, 2, 3, 4, 5, 6, and 8. However, Action Item 7 remains open. Action Item 7 requires licensees of Westinghouse reactors to develop plant-specific analyses applied for their facilities to demonstrate that cast austenitic stainless steel lower support columns will maintain their function during the period of extended operation. By letter dated February 13, 2017, the licensee stated in its response to Request for Additional Information (RAI)-2, that the industry revised the Pressurized Water Reactor Owner's Group (PWROG)-14048-P report, which includes additional analysis that confirms that with the exception of one unit, all Combustion Engineering (CE) and Westinghouse units are bounded by the revised analysis. In a letter dated March 1, 2017, the PWROG submitted this report to the NRC staff for information. The NRC staff is currently reviewing this report for its validity in providing adequate bounding functionality analyses for CE and Westinghouse fleets (with the exception of one unit). The NRC staff's review of the PWROG-14048-P report will be used in assessing the applicability of the report for FNP units. Therefore, Action Item 7 will remain open until the staff completes its assessment of PWROG-14048-P.

Principal Contributor: G. Cheruvenki, NRR

Date: June 2, 2017