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

May 18, 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 – INSERVICE INSPECTION ALTERNATIVE (FNP-ISI-ALT-20) (CAC NOS. MF8910 AND MF8911)

Dear Mr. Hutto:

By letter dated December 7, 2016, as supplemented by letter dated April 13, 2017, Southern Nuclear Operating Company, Inc., (SNC, the licensee) submitted a relief request (FNP-ISI-ALT-20, Version 1) for the Joseph M. Farley Nuclear Plant, Units 1 and 2. SNC proposed to use an inservice inspection (ISI) alternative to perform the examination of selected Class I piping and valves at plant conditions other than those required by American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, IWB-5222(b) by using alternative boundaries permitted in ASME Code Case N-800 "Alternative Pressure Testing Requirements for Class 1 Piping Between the First and Second Injection Valves."

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), the licensee proposed to use an alternative on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the proposed alternative and concludes, as set forth in the enclosed safety evaluation, that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC staff authorizes the use of alternative request FNP-ISI-ALT-20, Version 1, for Joseph M. Farley Nuclear Plant, Units 1 and 2, for the fourth 10-year ISI interval, which is scheduled to end on November 30, 2018.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the staff remain applicable, including the third party review by the Authorized Nuclear In-service Inspector.

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

Sincerely,

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-348, 50-364

Enclosure:
Safety Evaluation

cc w/enclosure: Distribution via Listserv

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST FNP-ISI-ALT-20, VERSION 1.0, REGARDING SYSTEM LEAKAGE TEST

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

JOSEPH M. FARLEY NUCLEAR PLANT UNITS 1 AND 2

DOCKET NUMBERS 50-348 AND 50-364

1.0 INTRODUCTION

By letter dated December 7, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16342C529), as supplemented by letter dated April 13, 2017 (ADAMS Accession No. ML17103A576), Southern Nuclear Operating Company, Inc. (SNC, the licensee) submitted a relief request (FNP-ISI-ALT-20, Version 1) for the Joseph M. Farley Nuclear Plant (FNP), Units 1 and 2. SNC proposed to use an inservice inspection (ISI) alternative to perform the examination of selected Class I piping and valves at plant conditions other than those required by American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, IWB-5222(b) by using alternative boundaries permitted in ASME Code Case N-800 "Alternative Pressure Testing Requirements for Class 1 Piping Between the First and Second Injection Valves."

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), the licensee proposed to use the alternative on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

2.0 REGULATORY EVALUATION

The regulation in 10 CFR 50.55a(g)(4), *Inservice inspection standards requirement for operating plants,* states in part, that: Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) that are classified as ASME Code Class 1, Class 2, and Class 3, must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions and addenda of the ASME BPV Code ... that become effective subsequent to editions specified in paragraphs (g)(2) and (3) of this section and that are incorporated by reference in paragraph (a)(1)(ii) ... of this section, to the extent practical within the limitations of design, geometry, and materials of construction of the components.

The regulation in 10 CFR 50.55a(g)(4)(ii), *Applicable ISI Code: Successive 120-month intervals,* states in part, that: Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (a) of this section 12 months before the start of the 120-month inspection interval (or the optional ASME Code Cases listed in NRC Regulatory Guide (RG) 1.147, Revision 17, *"Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1"* (ADAMS Accession No. ML13339A689), when using Section XI, that are incorporated by reference in paragraphs (a)(3)(ii) of this section, subject to the conditions listed in paragraph (b) of this section. However, a licensee whose inservice inspection interval commences during the 12 through 18-month period after July 21, 2011, may delay the update of their Appendix VIII program by up to 18 months after July 21, 2011.

Pursuant to 10 CFR 50.55a(z), *"Alternatives to codes and standards requirements,"* alternatives to the requirements of paragraphs (b) through (h) of this section [50.55a] or portions thereof may be used when authorized by the Director, Office of Nuclear Reactor Regulation. A proposed alternative must be submitted and authorized prior to implementation. The applicant or licensee must demonstrate that: (1) *"Acceptable Level of Quality and Safety,"* the proposed alternative would provide an acceptable level of quality and safety; or (2) *"Hardship without a Compensating Increase in Quality and Safety,"* compliance with the specified requirements of this section [50.55a] would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request and the NRC to authorize the alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 Component Affected

The components affected are ASME Code Class 1 piping boundary between and including the first and second isolation valves in the injection and return path of the safety injection system (SIS) High (HI) Head and Low (LO) Head, residual heat removal (RHR) system supply lines, the accumulator discharge lines to the reactor coolant system (RCS), and the RHR suction lines from the RCS. In accordance with Table IWB-2500-1 of Section XI, they are classified as Examination Category B-P, Item Number B15.10.

The licensee stated on pages E-1 and E-3 of the December 7, 2017, application, the Class 1 piping to be tested per Code Case N-800 pipes are identified in Table 2 for FNP Unit 1 and Table 3 for FNP Unit 2. However, the NRC staff notes that the Table for Unit 2 was incorrectly labeled as Table 2, Unit 2, thus, NRC Staff will refer to the piping to be tested for FNP Unit 2 as Table 3 in this safety evaluation.

In Tables 2 and 3, the licensee provided a detailed description of the affected line segments and isolation valves. The lines for each unit are described as:

SIS to RCS Cold Leg Loop 1
SIS to RCS Cold Leg Loop 2
SIS to RCS Cold Leg Loop 3

SIS HI and LO Head to Cold Leg Loop 1
SIS HI and LO Head to Cold Leg Loop 2
SIS HI and LO Head to Cold Leg Loop 3

SIS HI and LO Head to Hot Leg Loop 1
SIS HI and LO Head to Hot Leg Loop 2
SIS HI and LO Head to Hot Leg Loop 3

RHR suction from Hot Leg Loop 1
RHR suction from Hot Leg Loop 3

The licensee stated that materials of construction of these pipes and associated components (valves, fittings, and welded connections) are stainless steel based on IWB-5240(c).

3.2 Applicable Code Edition and Addenda

The Code of record for the fourth10-year ISI interval is the 2001 Edition through 2003 Addenda to the ASME Code, Section XI.

3.3 Duration of Relief Request

The licensee submitted this relief request for the fourth 10-year ISI interval, which commenced on December 1, 2007, and is scheduled to end on November 30, 2018.

The licensee stated that an extension of one year was applied to the fourth 10-year ISI interval in accordance with IWA-2430, and all applicable requirements in IWA-2430 have been met.

3.4 ASME Code Requirement

The ASME Code requirement applicable to this request originates in Table IWB-2500-1 of Section XI (Examination Category B-P, Item Number B15.10). Under this requirement, the piping shall be subjected to a system leakage test according to IWB-5220 and the VT-2 visual examination according to IWA-5240 prior to plant startup following a reactor refueling outage. Under IWB-5220, the requirements (boundary and pressure) for conducting system leakage tests at or near the end of each inspection interval are as follows:

- According to IWB-5222(b), the pressure retaining boundary during system leakage test conducted at or near the end of each inspection interval shall extend to all Class 1 pressure retaining components within the system boundary. The visual examination shall extend to and include the second closed valve at the boundary extremity.

- According to IWB-5221(a), the system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100 percent rated reactor power.

3.5 Proposed Alternative

The licensee proposed an alternative to IWB-5222(b). The alternative is to implement provisions of ASME Code Case N-800 to conduct system leakage test during refueling outage at or near the end of each inspection interval. The requirement of ASME Code Case N-800 is as follows:

> For portions of the Class 1 boundary between the first and second isolation valves in the injection and return path of standby safety systems, the system leakage test may be conducted by pressurization of the Class 1 volume using the Class 2 safety system to pressurize the volume. Such alternative tests shall be performed each inspection interval. The system leakage test shall be conducted using the pressure associated with the Class 2 system function that provides the highest pressure between the Class 1 isolation valves.

This code case has not been incorporated by reference into 10 CFR 50.55a by inclusion in RG 1.147, Revision 17.

In the April 13, 2017, supplement, the licensee proposed that,

> For the Safety Injection System (SIS) to Reactor Coolant System (RCS) Cold legs associated with the Accumulator Tanks, [piping segments 1, 2, and 3] the examinations will be performed using the outboard Class 2 highest system functional pressure associated with the tanks. The accumulator piping will be examined during the reactor shutdown for refueling outage at a pressure between 601 and 649 psig.
>
> For the SIS HI and LO Head, Hot leg/Cold Leg injection lines [piping segments 4, 5, 6, 7, 8,and 9] the examinations will be performed using the outboard Class 2 system functional pressure associated with the SIS HI Head (Charging Pumps), during the full flow test. The charging pump comprehensive testing is performed at a flow of 610 gpm and a discharge pressure typically between 1200 and 1600 psig. All 4 charging pump paths are tested each outage, and the two Residual Heat Removal (RHR) paths can be examined as well since this alignment will also pressurize those paths.
>
> For the RHR suction lines, Hot Leg Loops 1 and 3 [piping segments 1 0 and 11], the examinations will be performed using the outboard Class 2 highest system functional pressure associated with the RHR suction lines. During cooldown at the beginning of an outage, this section of piping is in service and can be examined. Per FNP Procedure SOP-7.0, RCS pressure is less than 375 psig.

3.6 Basis for Use

The licensee stated that the ASME Code required extended pressure boundary conditions in accordance with IWB-5222(b) is to detect evidence of leakage, and thereby verify the integrity of the reactor coolant pressure boundary (RCPB) beyond the first isolation valve. All the piping in this request is located inside the containment and in high radiation areas. The activities associated with the IWB-5222(b) required system leakage test would cause personnel to incur increased radiation dose and would pose potential worker safety hazards. Furthermore, compliance with IWB-5222(b) requires temporary plant configurations (bypassing interlocks and safety systems) that could place plant in abnormal conditions. The licensee stated that the proposed alternative (ASME Code Case N-800) allows the system leakage test be conducted using the pressure associated with the Class 2 system that provides the highest pressure between the Class 1 isolation valves.

The licensee stated that in the unlikely event of a through wall leak in the piping runs and segments identified in Tables 2 and 3 during normal operation, the leak would result in unidentified RCS leakage. The RCS leakage detection instrumentation has been designed to aid the operator in differentiating between possible sources of detected leakage within the containment and identifying the physical location of the leak. The RCS leakage detection instrumentation consists of the Containment Air Particulate Monitor, the Containment Radioactive Gas Monitor, Specific Humidity Monitoring Devices, the Condensate Measuring System, and the Dew point Temperature Monitoring System described in the Final Safety Analysis Report (FSAR) Section 5.2.7. These systems are in place to assure that the Technical Specifications for Farley, Units 1 and 2, Limiting Condition for Operation 3.4.13 "RCS Operational LEAKAGE" limit system operation in the presence of leakage from the RCS components to amounts that do not compromise safety.

In supplement dated April 13, 2017, the licensee stated that sections of pipe closer to the RCS loops are insulated. This insulation is metal reflective insulation. Any leakage, if it were to occur, should be apparent (i.e., it won't soak in). Additionally, the insulated segments are inspected every refueling outage after a 4-hour hold time during the Class 1 walkdown in Mode 3.

In supplement dated April 13, 2017, the licensee stated there are approximately 650 welded connections in the program scope. These welds were originally classified as Examination Category B-J during the first period of the fourth 10-year ISI interval. But they are now included in the risk-informed (RI)-ISI program and classified as Category R-A. Of the sixty-one welded connections that have been examined during the current 10-year ISI interval, 55 were volumetric examination and 6 were surface examination. In the upcoming fall 2017 refueling outage (Unit 2), five welded connections are scheduled for volumetric examinations. In all the examinations performed, there have been no recordable indications documented. There has not been any through-wall leakage reported in any of these piping segments. Identified leaks are tracked in the Boric Acid Corrosion Control Program (BACCP) leak management database.

3.7 Basis for Hardship

The licensee stated that an alternative is requested on the basis that a hardship and an unusual difficulty exists in establishing a system configuration necessary to pressurize the subject Class 1 components to the RCS operating pressure to accommodate for system leakage test in accordance with IWB-5222(b).

The licensee stated that existing design of the Class 1 piping in the standby safety systems requires substantial effort to extend the boundary subject to the RCS operating pressure where check valves or non-redundant components serve as the first system isolation from the RCS. Such configurations would require manually opening the inboard isolation valves (NRC staff notes that the licensee incorrectly referred to these as vent and drain type valves on page E-3 of the application), or bypassing the inboard isolation valves by installation of temporary piping installations, such as high-pressure hoses, and/or other unusual temporary system configurations in order to achieve test pressure at upstream piping as required by IWB-5222(b).

Manually opening of the inboard isolation valves defeats the double isolation barrier. By establishing and restoring temporary configurations or defeating double isolation barriers in order to test these lines in accordance with IWB-5222(b) could introduce potential hazards to personnel safety. Examples include occupational hazards, risk for spills, contaminations, and additional radiation exposure. In addition, these temporary configurations could require bypassing the FSAR specified safety system protective features and interlocks that could create abnormal plant conditions. Therefore, extension of the boundary subjected to the RCS operating pressure to all Class 1 pressure retaining components within the injection and return path of standby safety systems represents a hardship and an unusual difficulty.

The licensee stated that all the components listed in this relief request (Tables 2 and 3) are located inside the containment and in areas involving occupational radiation exposure. In Table 1 of the enclosure to this relief request, the licensee provided a history of radiation exposure and person-hour for conducting the IWB-5222(b) system leakage test during the previous (third) 10-year ISI interval. The activities associated with this test included scaffold erection, insulation removal, valve manipulations, examinations, re-installation of insulation, and scaffold removal. As a result of the above activities, the estimated total radiation dose incurred by personnel is 80 milli roentgen equivalent man (mrem) (Unit 1) and 84 mrem (Unit 2).

3.8 NRC Staff Evaluation

The NRC staff has evaluated this relief request pursuant to 10 CFR 50.55a(z)(2). The NRC staff focuses on whether (1) compliance with the specified requirements results in hardship or unusual difficulty; (2) performance of the testing for leakage is adequate; and (3) the licensee's proposed alternative (accepting the reduced test pressure in this case) provides reasonable assurance of structural integrity and leak tightness of the subject piping and components. The NRC staff finds that if these three criteria are met that the requirements of 10 CFR 50.55a(z)(2) will also be met.

Hardship

The NRC staff assessed the licensee's basis for hardship and/or unusual difficulties from compliance with the ASME Code system leakage test (IWB-5222(b) boundary and IWB-5221(a) pressure requirements) prior to plant startup following a reactor refueling outage at or near the end of each inspection interval. The NRC staff confirmed that:

- Opening or bypassing the first isolation valve (inboard isolation valve closer to primary loop piping) for the purpose of pressurizing the pipe segments to the RCS operating pressure to perform the ASME Code required system leakage test defeats the double isolation criteria, and reduces safety of the plant operation.

- Alternatively, establishing temporary configurations (e.g., using a temporary external hydro pump, non-code connections, and compatible medium) to pressurize the pipe segments to the RCS operating pressure for the purpose of performing the ASME Code required system leakage test would pose unnecessary safety hazards to personnel operating the equipment and performing the test and to those working nearby in case of a break in any temporary non-code connections.

- Activities associated with establishing temporary configurations and restoring to the normal operating condition would expose personnel to additional radiation dose.

- Establishing temporary configurations could also defeat the safety system protective features and interlocks that could create abnormal plant conditions and reduce safety of the plant operation.

Based on the above, the NRC staff determines that concerns from defeating the double isolation criteria, subjecting personnel to unnecessary safety hazards, exposing personnel to additional radiation dose, and defeating the safety system protective interlocks constitute hardship and/or unusual difficulty.

Test Pressure

In evaluating the licensee's proposed alternative, the NRC staff assessed whether it appeared that the licensee used the highest achievable test pressure to conduct system leakage test during outage at or near the end of each inspection interval, and the manner in which the licensee adequately performed the testing and the associated VT-2 visual examinations of the piping for leakage. The NRC staff verified that:

- The SIS to RCS Cold Leg lines associated with the accumulator tanks will specifically be subjected to a test pressure between 601 psig and 649 psig. This pressure is achieved using the outboard Class 2 highest system functional pressure associated with the tanks during reactor shutdown for refueling.

- The SIS HI and LO Head, Hot Leg / Cold Leg injection lines will specifically be subjected to a test pressure between 1200 psig and 1600 psig. This pressure is associated with the SIS HI Head during full flow testing of the charging pumps each refueling outage.

- The RHR suction lines, Hot Leg Loops 1 and 3 will specifically be subjected to a test pressure of less than 375 psig. This pressure is associated with the RHR suction lines that are in service during cooldown at the beginning of a refueling outage.

- As part of this leakage testing, the licensee will perform the associated VT-2 visual examination of the insulated and noninsulated pipe segments in accordance with the IWA-5240 to identify any leak or boron residue.

Based on the above, the NRC staff determines that the licensee will accomplish the proposed leakage testing with use of the maximum obtainable pressure without major modifications to existing configuration of piping, placing the plant in abnormal conditions, and exposing plant personnel to safety hazards and additional radiation dose. Therefore, the NRC staff finds the licensee's proposed testing adequate because the highest possible pressure is used.

Safety Significance of Alternative Test Pressure

In addition to the analysis described above, the NRC staff evaluated the safety significance of performance of the system leakage test at an alternative reduced pressure. The NRC staff confirmed that:

- All welded connections in the subject piping segments (650 welds) have been included in the Farley's RI-ISI program. As a part of the RI-ISI process, the licensee assesses susceptibility of these welded connections (stainless steel welds) to potential degradation mechanisms, and the risk significant locations are identified for inspections. In all the examinations (volumetric and surface) performed during the fourth 10-year ISI program, there have not been any recordable indications found.

- There has not been any through wall leakage reported in any of the welded connections during the current interval.

- If in an unlikely event, the subject piping segments develop a through wall flaw and a leak, the Farley existing reactor coolant leakage detection systems will be able to identify the leakage during normal operation, and the licensee will take appropriate corrective actions in accordance with the plant technical specifications.

Therefore, the NRC staff determines that based on the alternative system leakage testing that subject these piping segments to the maximum attainable pressure and the performance of the ASME Code required VT-2 visual examinations, it is reasonable to conclude that if significant service induced degradation occurs, evidence of that degradation will be detected either by the proposed examinations or the RCS leakage detection systems.

Therefore, the NRC staff finds that the proposed system leakage testing using the proposed test pressure is adequate to provide a reasonable assurance of structural integrity and leak tightness of the subject piping. Complying with the requirement specified in IWB-5222(b) would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff concludes that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the subject piping segments, and complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC staff authorizes the use of FNP-ISI-ALT-20, Version 1, at Farley, Units 1 and 2, for the fourth 10-year ISI interval, which is scheduled to end on November 30, 2018. The authorization of the proposed alternative does not imply or infer the NRC approval of ASME Code Case N-800."

All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the staff remain applicable, including the third party review by the Authorized Nuclear In-service Inspector.

Principal Contributor: A. Rezai, NRR

Date: May 18, 2017

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 – INSERVICE INSPECTION ALTERNATIVE (FNP-ISI-ALT-20) (CAC NOS. MF8910 AND MF8911) DATED MAY 18, 2017

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