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

August 8, 2013

Mr. C. R. Pierce
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 - REQUEST FOR ALTERNATIVE FNP-ISI-ALT-13 REGARDING DEFERRAL OF INSERVICE INSPECTION OF REACTOR PRESSURE VESSEL COLD LEG NOZZLE DISSIMILAR METAL WELDS (TAC NOS. ME9739 AND ME9740)

Dear Mr. Pierce:

By letters dated October 1, 2012, May 6, May 24, and July 19, 2013, Southern Nuclear Operating Company (the licensee) submitted for the U.S. Nuclear Regulatory Commission (NRC) approval of the request for alternative (RFA) FNP-ISI-ALT-13. The licensee proposed an alternative to a certain requirement of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i), the licensee proposed an alternative frequency of examination for the reactor pressure vessel cold leg nozzle dissimilar metal welds on the basis that the alternative provide an acceptable level of quality and safety.

The NRC staff has reviewed the subject request, and concludes that the proposed alternative provides an acceptable level of quality and safety. Therefore, the NRC staff authorizes the proposed alternative in accordance with 10 CFR 50.55a (a)(3)(i). The NRC staff's safety evaluation is enclosed.

Sincerely,

R. Pascarelli

Robert Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosure: Safety Evaluation

cc w/encl: Distribution via Listserv

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ALTERNATIVE FNP-ISI-ALT-13

REGARDING DEFERRAL OF INSERVICE INSPECTION OF REACTOR PRESSURE VESSEL

COLD LEG NOZZLE DISSIMILAR METAL WELD

SOUTHERN NUCLEAR OPERATING COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS.: 50-348 AND 50-364

1.0 INTRODUCTION

By letters dated October 1, 2012, May 6, May 24, and July 19, 2013, (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML12276A110, ML13130A119, ML13149A021, and ML13200A341, respectively), Southern Nuclear Operating Company (the licensee) submitted for the U.S. Nuclear Regulatory Commission (NRC) approval the request for alternative (RFA) FNP-ISI-ALT-13. The licensee proposed an alternative to a certain requirement of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. RFA FNP-ISI-ALT-13 relates to the inservice inspection (ISI) requirements for the reactor pressure vessel (RPV) cold leg nozzle dissimilar metal (DM) welds at the Joseph M. Farley Nuclear Plant (Farley), Units 1 and 2.

In the initial request, the licensee proposed to inspect the RPV cold leg nozzle DM welds every ten years in lieu of the required every second inspection period not to exceed 7 years. By letter dated July 19, 2013, the licensee revised the initial request in order to expedite approval of RFA FNP-ISI-ALT-13 due to the upcoming fall 2013 refueling outage at Farley, Unit 1. In the revised request, the licensee proposed deferral of the Farley, Unit 1, RPV cold leg nozzle DM welds examination which is scheduled for the fall 2013 to the spring of 2015 refueling outage and the Farley, Unit 2, RPV cold leg nozzle DM welds examination which is scheduled for spring 2016 to the fall of 2017 refueling outage. The deferral of examination will not exceed 5 refueling outages based on a nominal cycle length of approximately 1.5 calendar years (approximately every 7.5 calendar years or 7.1 effective full power years (EFPY).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i), the licensee proposed an alternative frequency of examination for the RPV cold leg nozzle DM welds on the basis that the alternative provide an acceptable level of quality and safety.

Enclosure

2.0 REGULATORY EVALUATION

10 CFR 50.55a(g)(4) specifies that ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for In-service Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that in-service examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code, incorporated by reference in 10 CFR 50.55a(b), 12 months prior to the start of the 120-month interval, subject to the conditions listed therein.

10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used, when authorized by the NRC, if the licensee demonstrates (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

10 CFR 50.55a(g)(6)(ii)(F) "Examination requirements for Class 1 piping and nozzle dissimilar-metal butt welds" requires licensees of existing operating pressurized-water reactors (PWR) implement the requirements of ASME Code Case N-770-1, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated With UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities Section XI, Division 1," subject to the conditions specified in paragraphs (g)(6)(ii)(F)(2) through (g)(6)(ii)(F)(10) of section 50.55a, by the first refueling outage after August 22, 2011.

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

3.1 TECHNICAL EVALUATION

3.2 The Licensee's Request for Alternative

The components affected by this request include ASME Code Class 1 RPV cold leg nozzle DM butt welds in the reactor coolant system. These welds contain Alloy 82/182 materials. Welds containing Alloy 82/182 are susceptible to the primary water stress corrosion cracking (PWSCC).

The Farley, Units 1 and 2, code of record for the fourth 10-year interval ISI program is the ASME Code 2001 Edition through 2003 Addenda.

ASME Code Case N-770-1 (Table 1, Inspection Item B) requires unmitigated butt welds at cold leg temperatures greater than or equal to 525°F and less than 580°F to be volumetrically examined every second inspection period not to exceed 7 years.

The licensee initially proposed to inspect the subject RPV cold leg nozzle DM welds every ten years in lieu of the ASME Code Case N-770-1 required every second inspection period not exceeding 7 years. However, due to preparation for the upcoming refueling outage of Farley, Unit 1, in fall 2013, the licensee revised the initial request in order to expedite approval of RFA FNP-ISI-ALT-13. The revised proposal defers the examination to the fifth refueling outage based on a nominal cycle length of approximately 1.5 calendar years (approximately every 7.5 calendar years or 7.1 EFPY).

The licensee's overall bases (Enclosures 1 and 2 to RFA FNP-ISI-ALT-13) for this request are discussed below.

- The subject RPV cold leg nozzle DM welds are typically inspected from the inside diameter (ID). For this inspection to be performed at the required frequency, the vessel lower internals assembly (core barrel) must be removed for access on a six or seven year interval. However, if the inspection frequency of these welds is extended to a ten year interval to coincide with the frequency of inspection of the RPV shell welds that is typically done every interval, the core barrel removal evolutions will be minimized. The removal of the core barrel is considered to be a critical lift due to various reasons such as weight of the component, tight clearances involved, risks associated with equipment damage, and risk of personnel exposure to radiation emitted by the assembly.

- The licensee performed the baseline examinations of the subject DM welds using remotely operated and mechanized technique from the ID surface with the procedure developed in accordance with the requirements of Appendix VIII. The technique used in site-specific examinations achieved 100 percent coverage in the axial and circumferential scans. The licensee used encoded techniques. The licensee did not need to use site-specific mockups. The licensee thus acceptably demonstrated its Appendix VII qualification.

- There have been no known instances of PWSCC occurring in large bore pipes that operate at or near cold leg temperatures noted within the nuclear industry.

To support extension of the reexamination interval for the subject welds, the licensee developed a generic technical basis document (i.e., Enclosure 2 to RFA FNP-ISI-ALT-13) by compiling all existing flaw tolerance analyses performed to date on Alloy 82/182 welds. Figure 5-4 of Enclosure 2 of the October 1, 2012, letter which is relevant to this request shows the results of a generic flaw tolerance analysis for a circumferential flaw in the RPV inlet nozzle DM welds. These results are not representative of a single plant in the Westinghouse PWR fleet, rather they are based on the limiting thickness in the Westinghouse PWR fleet combined with limiting piping loads from another plant in the Westinghouse PWR fleet. The underlying analysis assumptions included a 25 percent ID weld repair, a postulated initial ID surface-connected circumferential flaw of 10 percent through wall thickness, and a short and a long stainless steel safe end. The analysis was performed for the cold legs with high and low temperature of 565°F and 535°F, respectively.

A flaw tolerance analysis was not performed for an axial through-wall flaw in Enclosure 2 to RFA FNP-ISI-ALT-13. The licensee's justification was that the maximum length of an axial flaw is limited to the width of the Alloy 82/182 weld which is significantly less than the calculated critical axial crack length.

In response to an NRC request for additional information, the licensee provided the plant-specific axial and hoop weld residual stress (WRS) profiles. The licensee also provided plant-specific axial flaw tolerance analysis. The licensee's axial flaw analysis assumed an initial flaw of 7.5 percent through wall thickness. The licensee showed that the assumed initial 7.5 percent through wall axial flaw will grow to become a through wall leak flaw in 10.5 EFPY.

The licensee submitted this request for the fourth 10-year ISI interval which commenced on December 1, 2007, and will end on November 30, 2017.

3.3 NRC Staff Evaluation

The NRC staff (or hereafter called the staff) has evaluated RFA FNP-ISI-ALT-13 pursuant to 10 CFR 50.55a(a)(3)(i). The staff evaluation focuses on whether the proposed alternative provides an acceptable level of quality and safety.

The staff performed an independent flaw evaluation to verify the licensee's flaw tolerance analysis. The staff's review identified certain limitations as follows:

a. The licensee's flaw tolerance analysis and results presented in Enclosure 2 of the October 1, 2012, letter were generic and did not show sufficient information to bound the Farley's RPV cold leg nozzle DM welds.

b. The results shown in Figure 5-4 of Enclosure 2 of the October 1, 2012, letter assumed a 25 percent ID weld repair. Section 3.6, Attributes of an Acceptable Residual Stress Analysis, of MRP-287, "Materials Reliability Program: Primary Water Stress Corrosion Cracking (PWSCC) Flaw Evaluation Guidance," identifies the expectation that a 50 percent ID weld repair would be used to support analysis for NRC review. The 25 percent assumption is potentially non-conservative since all repairs made to the welds during fabrication may not necessarily be recorded.

c. The licensee considered that a flaw tolerance evaluation for an axial flaw was unnecessary since the maximum length of a through wall axial crack would be limited to the width of the Alloy 82/182 weld and significantly less than the calculated critical axial crack length. The staff found this assumption to be inadequate because an axial crack could exceed the ASME Code allowable 75 percent through-wall thickness of the weld before the required end of the inspection period. Therefore, the staff requested the licensee perform an axial flaw analysis.

d. In letter dated May 24, 2013, the licensee provided additional information by submitting a plant-specific flaw evaluation for an axial flaw. The licensee assumed a postulated initial axial flaw with depth of 7.5 percent through-wall thickness. This assumption is potentially non-conservative since it may not be reasonable to expect that all flaws with depth less than 10 percent through-wall thickness would be reliably detected and recorded.

The staff addressed the potential limitations noted above by performing an independent flaw analysis that is discussed as follows. The staff's independent flaw analyses determined the maximum flaw depth, leak, and rupture characteristics of the subject welds to a postulated initial ID surface-connected (circumferential or axial) flaw. The analyses were performed based on the requirements of the ASME Code, Section XI IWB-3640 and an assumed postulated initial flaw due to PWSCC. The staff used the Farley's WRS distributions provided by the licensee. The staff also performed a sensitivity study to understand the effects of certain uncertainties concerning the WRS on the results.

In these analyses, the staff used the piping loads available in the leak before break database and the licensee's provided operating temperature for the cold leg at Farley. In all cases, the staff assumed an initial ID surface-connected flaw depth of 10 percent through-wall thickness. The staff's analyses assumed an idealized semi-elliptical surface flaw with initial aspect ratio (i.e., length divided by depth) of 2 and 10 for an axial and a circumferential flaw, respectively. The WRS profile for the axial and the hoop direction were curve-fit by a fourth order polynomial approximation. For the PWSCC crack growth, the staff used the 75th percentile crack growth rate data for Alloy 182.

The staff based its assessment of the licensee's proposed alternative on the time for a 10 percent deep initial crack to grow to the Code allowable crack depth of 75 percent through-wall thickness. The staff's analysis found sufficient margin to conclude that inspecting these welds once every five refueling outages based on a nominal cycle length of approximately 1.5 calendar years (approximately every 7.5 calendar years or 7.1 EFPYs), would provide an acceptable level of quality and safety.

4.0 CONCLUSION

Pursuant to 10 CFR 50.55a(a)(3)(i), alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the NRC, if the licensee demonstrates that the proposed alternatives will provide an acceptable level of quality and safety. As set forth above, the staff determines that the licensee's proposed alternative provides an acceptable level of quality and safety for the RPV cold leg nozzle DM welds subject to the ISI examinations being performed once every five refueling outages based on a nominal cycle length of approximately 1.5 calendar years (approximately every 7.5 calendar years or 7.1 EFPY). Accordingly, the staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(i). Therefore, pursuant to10 CFR 50.55a(a)(3)(i), the staff authorizes the Farley, Units 1 and 2, RPV cold leg nozzle DM welds examinations once every five refueling outages based on a nominal cycle length of approximately 1.5 calendar years (approximately every 7.5 calendar years or 7.1 EFPY).

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 Contributors: A. Rezai
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Date: August 8, 2013

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/RA/
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Docket Nos. 50-348 and 50-364

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