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

May 5, 2015

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, UNIT 2, ALTERNATIVE TO INSERVICE INSPECTION REGARDING REACTOR CLOSURE HEAD NOZZLE AND PARTIAL PENETRATION WELDS (TAC NO. MF4990)

Dear Mr. Pierce,

By letter dated October 6, 2014, as supplemented on April 21, 2015, Southern Nuclear Operating Company, Inc., (SNC) requested approval to use an alternative to the American Society of Mechanical Engineers (ASME) Code Case N-729-1, Inspection Item B4.40 for reactor pressure vessel closure head nozzle and partial penetration welds of primary water stress corrosion cracking resistant materials. Specifically, SNC requested the welds be reexamined once every 15-years for Farley Nuclear Plant (FNP) Unit 2, in lieu of the 10-year examination requirement outlined in ASME Code Case N-729-1.

The application was submitted pursuant to Section 50.55a(z)(1) of Title 10 of the *Code of Federal Regulations* (10 CFR), which requires that the applicant demonstrate that the proposed alternative provides an acceptable level of quality and safety.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the subject request, and concludes that SNC has adequately addressed all of the regulatory requirements and 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(z)(1). The NRC staff's safety evaluation is enclosed.

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

Sincerely,

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

Docket No. 50-364

Enclosure: Safety Evaluation

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ALTERNATIVE TO ASME CODE REQUIREMENTS

FOURTH 10-YEAR INSERVICE INSPECTION PROGRAM INTERVAL

SOUTHERN NUCLEAR OPERATING COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-364

## 1.0 INTRODUCTION

By letter dated October 6, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14280A260), as supplemented by letter dated April 21, 2015 (ADAMS Accession No. ML15111A387), Southern Nuclear Operating Company, Inc., (SNC, the licensee) requested an alternative from requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code). The alternative request FNP-ISI-ALT-17, Versions 1 and 2, pertains to frequency of examination of the reactor pressure vessel (RPV) closure head penetration tubes and vent pipe and the associated attachment partial penetration dissimilar metal (DM) welds made of the primary water stress corrosion cracking (PWSCC) resistant materials at the Joseph M. Farley Nuclear Plant (Farley), Unit 2.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee proposed an alternative frequency of examination for the RPV closure head penetrations and their attachment DM welds on the basis that the alternative provides an acceptable level of quality and safety.

## 2.0 REGULATORY EVALUATION

By Federal Register Notice 79 FR 65776, dated November 5, 2014, which became effective on December 5, 2014, the paragraphs headings in 10 CFR 50.55a were revised. Accordingly, relief requests that had been previously covered by 10 CFR 50.55a(a)(3)(i) are now covered under the equivalent 10 CFR 50.55a(z)(1) and relief requests that had been previously covered by 10 CFR 50.55a(a)(3)(ii) are now covered under the equivalent 10 CFR 50.55a(z)(2).

In FNP-ISI-ALT-17, the licensee requested relief from the frequency of inservice inspection (ISI) of penetration nozzles and associated partial penetration DM welds of PWSCC resistant materials in the reactor closure head as required by ASME Code Case N-729-1 "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1."

Enclosure

Pursuant to 10 CFR 50.55a(g)(6)(ii)(D), Augmented ISI requirements: Reactor vessel head inspections - (1) All licensees of pressurized water reactors must augment their ISI program with ASME Code Case N-729-1, subject to the conditions specified in paragraphs (g)(6)(ii)(D)(2) through (6) of 10 CFR 50.55a.

Pursuant to 10 CFR 50.55a(g)(4), the ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components.

Pursuant to 10 CFR 50.55a(z), alternatives to the requirements of paragraph (g) of 10 CFR 50.55a 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 licensee must demonstrate (1) the proposed alternative would provide an acceptable level of quality and safety; or (2) compliance with the specified requirements of this section 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 RPV closure head penetration tubes and vent pipe and their attachment partial penetration (J-groove) DM welds made of PWSCC resistant materials. In accordance with ASME Code Case N-729-1 (Table 1), they are classified as Inspection Item B4.40.

The material of construction of the penetration tubes and vent pipe is Alloy 690. Each penetration tube and vent pipe is welded to the RPV closure head by Alloy 52/152 weld materials. Alloy 690 base material and Alloy 52/152 weld materials have been known to be resistant to PWSCC.

### 3.2 Applicable Code Edition and Addenda

The code of record for the fourth 10-year ISI interval is the 2001 Edition through 2003 Addenda of the ASME Code.

### 3.3 Duration of Relief Request

The licensee submitted this relief request for the ISI that is expected to take place in the spring of 2016 refueling outage in the fourth 10-year ISI interval which began on December 1, 2007, and will end on November 30, 2017. In the April 21, 2015 letter, the licensee stated that approval of this relief request would permit deferral of examinations of the RPV closure head penetration tubes and vent pipe and their partial penetration DM welds currently scheduled for

the spring of 2016 refueling outage in the fourth 10-year ISI interval to be moved to the fall of 2020 refueling outage (U2R27) in the fifth 10-year ISI interval.

3.4 ASME Code Requirement

ASME Code Case N-729-1, Table 1, Item Number B4.40, requires that the reactor vessel closure head penetration nozzles and the associated attachment partial penetration welds of the PWSCC resistant materials be subjected to volumetric and surface examinations during every 10-year ISI interval (nominally 10 calendar years).

3.5 Proposed Alternative

The licensee proposed alternative frequency of examination. In the April 21, 2015 letter, the licensee stated that the proposed alternative is to perform the volumetric and surface examinations of the RPV closure head penetration tubes and vent pipe and their associated attachment partial penetration DM welds not later than fifteen (15) calendar years.

3.6 Basis for Use

The licensee's basis for use of the proposed alternative is based primarily on three topics of consideration. The first topic addresses the concept that the inspection interval in ASME Code Case N-729-1 is based on PWSCC crack growth rates for Alloy 600/82/182 materials. The second topic addresses a bare metal visual VE examination of the licensee's replacement RPV closure head performed according to Item Number B4.30 in Table 1 of Code Case N-729-1. The third topic addresses a plant-specific factor of improvement (FOI) analysis that the licensee conducted.

In addressing its first basis for use of the proposed alternative, the licensee asserts that the inspection intervals contained in Code Case N-729-1 for Alloy 600/82/182 are based on re-inspection years (RIY) equal to 2.25 and that this value is based on PWSCC crack growth rates as defined in the 75th percentile curve contained in Electric Power Research Institute Materials Reliability Program (MRP)-55, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Material", and MRP-115, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds." The licensee further asserts that the PWSCC crack growth rates of Alloy 690/52/152 are significantly lower than those of Alloy 600/82/182, therefore, merit a longer inspection interval. The licensee bases that assertion on: (a) the lack of cracking in other Alloy 690 components such as steam generators and pressurizers in the approximately 20 years that Alloy 690 has been in service in these components; (b) the failure to observe cracking in inspections already performed in replacement heads (9 of 40 replacement heads have been examined which includes heads which operate at higher temperatures than the head under consideration); (c) the similarity of the inspected heads to the head under consideration regarding configuration, manufacturing, design and operating conditions; and (d) laboratory test data for Alloy 690/52/152 as contained in MRP-375, "Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles."

In addressing its second basis for use of the proposed alternative, the licensee stated in April 21, 2015, letter that the bare metal visual VE examination performed on the Farley, Unit 2, replacement RPV closure head in the spring of 2010 and fall of 2014 refueling outages showed

no indication of leakage. The licensee will continue to perform a bare metal visual VE examination on the Farley, Unit 2, replacement RPV closure head in accordance with all requirements specified in Code Case N-729-1, Table 1, Item Number B4.30. The upcoming bare metal visual VE examination on the Farley, Unit 2, replacement RPV closure head is scheduled for the spring of 2019 refueling outage. The licensee stated that it does not propose any alternative examination processes to those specified in Code Case N-729-1 as required by 10 CFR 50.555a(g)(6)(ii)(D). The bare metal visual VE examination and acceptance criteria required by Item Number B4.30 in Table 1 of Code Case N-729-1 are not affected by this request and the licensee will continue to perform this examination on a frequency not to exceed every 5 calendar years.

Furthermore, the licensee stated that the results from the 2005 preservice volumetric and surface examinations performed on the Farley, Units 1 and 2, replacement heads showed no detectable defects. The volumetric and surface examinations that were recently completed successfully on the Farley, Unit 1, replacement head showed no detectable indications. The Farley, Unit 1, replacement head is of the same design and manufacturer with comparable age and operating conditions as the Unit 2 replacement head.

In addressing its third basis for use of the proposed alternative, the licensee made a plant specific calculation of the required FOI in the crack growth rate of Alloy 690/52/152 as compared to the crack growth rate of Alloy 600/82/182. In making this calculation, the licensee used the actual temperature of the head and conservatively assumed that calendar years were equal to effective full power years. Based on this calculation and as documented in the April 21, 2015 letter, the licensee determined that an FOI of 7 was required to meet the proposed and desired inspection interval of 15 calendar years. The licensee then proposed that because the required FOI of 7 was smaller than the FOI of 20 which bounded most of the MRP-375 data for Alloy 690/52/152, the use of a FOI of 7 would not result in a reduction in safety, therefore, was justified.

The licensee stated that their analysis showed significant margin to ensure the potential for PWSCC in Alloy 690 nozzles and their Alloy 52/152 attachment welds is remote. As such, the licensee found that technical basis sufficient to provide reasonable assurance of the structural integrity and leak tightness of the Farley, Unit 2, replacement RPV closure head by extending the inspection frequency of the head from a maximum of 10 years to a new maximum of 15 calendar years.

## 3.7 NRC Staff Evaluation

The NRC staff has evaluated the licensee's request pursuant to 10 CFR 50.55a(z)(1). The NRC staff focuses on whether the proposed alternative provides an acceptable level of quality and safety.

In evaluating the technical sufficiency of the licensee's proposed alternative (i.e., one time extension of the volumetric and surface examination interval contained in ASME Code Case N-729-1 from 10 calendar years to not later than 15 calendar years), the NRC staff considered each of the three aspects of the licensee's basis for use of the proposed alternative.

Due to PWSCC concerns, many pressurized water reactor (PWR) plants in the United Stated and overseas have replaced RPV closure heads containing Alloy 600/182/82 nozzles with

heads containing Alloy 690/152/52 nozzles. The Alloy 690/152/52 materials have been resistant to PWSCC. The inspection frequencies developed in Code Case N-729-1 for RPV closure head penetration nozzles using Alloy 600/182/82 were based, in part, on those material's crack growth rate equations documented in MRP-55 and MRP-115. The licensee's primary technical basis is to present crack growth rate data for the Alloy 690/152/52 materials that are resistant to PWSCC, and demonstrate a FOI of these materials versus the Alloy 600/82/182 materials. This FOI would then provide the basis for extension of the ISI frequency requested by the licensee in its proposed alternative.

In evaluating the licensee's first technical basis for use of the proposed alternative, the NRC staff notes that the licensee uses MRP-375. This document, in part, summarizes numerous Alloy 690/152/52 crack growth rate data from various sources to develop FOI for the crack growth rate equations provided in MRP-55 and MRP-115. While the NRC staff finds that the licensee's assertions and interpretations are reasonable, MRP-375 is not an NRC approved document. Furthermore, a more detailed review of the data provided in MRP-375 is being performed by an international group of experts as part of an Alloy 690 Expert Panel and is scheduled to be completed in the 2016-2017 timeframe.

In the interim, the NRC staff's review will rely upon Alloy 690/152/52 crack growth rate data from two NRC contractors: Pacific Northwest National Laboratory (PNNL) and Argonne National Laboratory (ANL). This data is documented in a data summary report (ADAMS Accession No. ML14322A587). Furthermore, this data generally supports the contention that the crack growth rate of Alloy 690/52/152 is more crack resistant but differs from the MRP-375 data in some respects.

The PNNL and ANL data summary report includes crack growth rate data up to approximately 20 percent cold work based on the observation of local strains in welds and weld dilution zone data. However, the NRC staff did not consider the weld dilution zone data in its assessment of the licensee's proposed alternative. This is because the limited weld dilution zone data that is currently available has shown higher crack growth rates than are commonly observed for Alloy 690/152/52 materials. The high crack growth rates in weld dilution zones may be due to the reduced chromium present in these areas. The NRC staff chose to exclude the weld dilution zone data from this analysis due to the limited number of data points available, the variability in results, and due to the limited area of continuous weld dilution for flaws to grow through. For example, in the case of the highest measured crack growth rates, a flaw would have to travel in the heat affected zone of a J-groove weld along the low alloy steel head interface. It is not fully apparent to the NRC staff how accelerated crack growth in very small areas of weld dilution zone would result in a significantly increased probability of leakage or component failure during a relatively short extension of the required inspection interval. Exclusion of these data may be reevaluated as additional data become available; a better understanding of the existing data is obtained; or if a longer extension of the inspection interval is requested. Therefore, the NRC staff finds that the impact of these weld dilution zone crack growth rates on the change in volumetric inspection frequency, as requested by the licensee's proposed alternative, is not considered to be relevant for this specific relief request.

In evaluating the licensee's second basis for use of the proposed alternative, the NRC staff finds that the past bare metal visual VE examination on the head under consideration is a reasonable means to demonstrate the absence of leakage through the nozzle or J-groove weld, or both, prior to the time the examination was conducted. The NRC staff also finds that performance of

future bare metal visual VE examinations in accordance with Code Case N-729-1 is adequate to demonstrate the absence of leakage at or prior to the time the examinations are conducted. Finally, the NRC staff finds that the proposed alternative's frequency for bare metal visual VE examinations in conjunction with the new frequency of volumetric examinations is sufficient to provide reasonable assurance of the structural integrity of the RPV closure head.

In evaluating the licensee's third basis for use of the proposed alternative, the NRC staff found that the licensee's calculated FOI of 7, to support an extension of the Code Case N-729-1 inspection frequency of 2.25 RIY to 15 calendar years, was acceptable by NRC staff calculation. The NRC staff also found that the application of a FOI of 7 to the 75th percentile curves in MRP-55 and MRP-115 bounded essentially all of the NRC data included in the PNNL and ANL data summary report. Therefore, the NRC staff found that this analysis supports the concept that a volumetric inspection interval for the RPV closure head of not more than 15 calendar years does not pose a higher risk than that associated with an Alloy 600/182/82 RPV closure head inspected at intervals of 2.25 RIY. Hence, the NRC staff found the licensee's technical basis to be acceptable.

Therefore, the NRC staff finds that the licensee has provided adequate technical basis to demonstrate that its proposed alternative examination frequency (i.e., not exceeding fifteen (15) calendar years) would provide an acceptable level of quality and safety.

## 4.0 CONCLUSION

As set forth above, the NRC staff determines that the licensee's proposed alternative provides an acceptable 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)(1). Therefore, the NRC staff authorizes the use of the licensee's proposed alternative at Farley, Unit 2, for duration of up to and including the fall of 2020 refueling outage in the fifth 10-year ISI interval.

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 Inservice Inspector.

Principal Contributor: Ali Rezai, NRR/DE/EPNB

Date of issuance: May 5, 2015

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

Sincerely,

**/RA/**

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

Docket No. 50-364

Enclosure: Safety Evaluation

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