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SOUTHERN COMPANY

APR 2 1 2015

Docket No.: 50-364 NL-15-0718

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant, Unit 2
Proposed Inservice Inspection Alternative FNP-ISI-ALT-17, Version 2.0

Ladies and Gentlemen:

In a letter dated October 6, 2014 (NL-14-1355), Southern Nuclear Operating Company (SNC) requested Nuclear Regulatory Commission (NRC) approval of proposed Inservice Inspection (ISI) Alternative FNP-ISI-ALT-17, Version 1.0. This alternative would allow ASME Code Case N-729-1, Inspection Item B4.40 for Reactor Pressure Vessel (RPV) Closure Head nozzle and partial penetration welds of Primary Water Stress Corrosion Cracking (PWSCC)-resistant materials to be re-examined once every 20 years for Farley Nuclear Plant (FNP) Unit 2 in lieu of the 10-year examination requirement outlined in ASME Code Case N-729-1.

On April 8, 2015 a conference call was held with the NRC to discuss the 20 year proposed alternative inspection interval. Based on discussions during the call, SNC is submitting this supplement to NL-14-1355 to request the inspection interval for the RPV Closure Head nozzle and partial penetration welds be changed to a nominal 15 years, which would permit completing the required examinations in refueling outage U2R27, currently scheduled for the fall of 2020.

Changes from the previous proposed alternative version include the following:

- Updated reference to the CFR section pertaining to alternatives throughout
- Requested 20-year inspection interval has been changed to 15 years throughout
- Clarifications in Enclosure 1 in regards to inspection results for the baseline inspection performed in 2005 and bare metal visual inspections
- Reference to known 2015 FNP Unit 1 head inspection results in Enclosure 1

- Updated Precedents section in Enclosure 1
- Enclosure 2 Factor of Improvement calculation has been updated to reflect 15 years versus 20 years
- Enclosure 3 has been deleted from this alternative version since it has already been submitted and has not changed

SNC continues to request approval of Version 2.0 of the proposed alternative by June 30, 2015.

This letter contains no NRC commitments. If you have any questions, please contact Ken McElroy at (205) 992-7369.

Sincerely,

C. R. Pierce

C. R. Pierce
Regulatory Affairs Director

CRP/JMC/

Enclosure 1: Proposed Alternative FNP-ISI-ALT-17, Version 2.0
In Accordance with 10 CFR 50.55a(z)(1)

Enclosure 2: Minimum Factor of Improvement (FOI) Calculation for Farley Unit 2

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Joseph M. Farley Nuclear Plant, Unit 2
Proposed Inservice Inspection Alternative FNP-ISI-ALT-17, Version 2.0

Enclosure 1

Proposed Alternative FNP-ISI-ALT-17, Version 2.0
In Accordance with 10 CFR 50.55a(z)(1)

Enclosure 1

Proposed Alternative FNP-ISI-ALT-17, Version 2.0
In Accordance with 10 CFR 50.55a(z)(1)

Plant Site-Unit:	Joseph M. Farley Nuclear Plant (FNP) – Unit 2.
Interval Dates:	4th Inservice Inspection (ISI) Interval – December 1, 2007 through November 30, 2017.
Requested Date for Approval :	Approval is requested by June 30, 2015.
ASME Code Components Affected:	The affected components are American Society of Mechanical Engineers (ASME) Class 1 Pressurized Water Reactor (PWR) Reactor Vessel Upper Head (Closure Head) nozzles and partial-penetration welds fabricated with Primary Water Stress Corrosion Cracking (PWSCC)-resistant materials. FNP Unit 1 and 2 penetration tubes and vent pipe are fabricated from Alloy 690 with Alloy 52/152 attachment welds.
Applicable Code Edition and Addenda:	The applicable Code edition and addenda (for the 4th ISI interval) is ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant components," 2001 Edition through the 2003 Addenda. Note that FNP will adopt a later applicable code edition for the 5th ISI Interval, but that this alternative still applies to the frequency of the next Reactor Vessel Closure Head (RPVCH) Exam.
Applicable Code Requirements:	10 CFR 50.55a(g)(6)(ii)(D) required licensees of existing, operating pressurized-water reactors by December 31, 2008 to implement the requirements of ASME Code Case N-729-1. Code Case N-729-1, Inspection Item B4.40 for ASME Class 1 PWR Reactor Vessel Upper Head, i.e., closure Head, (RPVCH) nozzles and partial-penetration welds fabricated with PWSCC-resistant materials, requires volumetric and/or surface examination of essentially 100% of the required volume or equivalent surface of the nozzle tube each inspection interval (nominally 10 calendar years). A demonstrated volumetric or surface leak assessment through all J-groove welds is required.
Reason for Request:	Treatment of Alloy 690 RPV Closure Heads in Code Case N-729-1 was intended to be conservative and subject to reassessment once additional laboratory data and plant experience on the performance of Alloy 690 and Alloy 52/152 weld metals became available. Using plant and laboratory data, ERPI document Materials Reliability Program (MRP) - 375 (EPRI 3002002441) was developed to support a technically based volumetric / surface reexamination interval using appropriate analytical tools. This technical basis demonstrates that the re-examination interval can be extended to the requested interval length while maintaining an acceptable level of quality and safety. Therefore, Southern Nuclear Operating Company (SNC) is requesting approval of this alternative to allow the use of the ISI interval extension for the affected FNP-Unit 2 components.

Proposed Alternative:	SNC is requesting extension of the requirements of Code Case N-729-1, Inspection Item B4.40 for performing volumetric/surface exams of the FNP Unit 2 RPVCH. Specifically, this would allow volumetric/surface examinations currently scheduled for the spring of 2016 (baseline exams were performed in the fall of 2005 with satisfactory results) to be moved to the fall of 2020. This request applies to the inspection frequencies not the inspection techniques, as the inspection techniques may change with later editions of ASME Section XI and 10 CFR 50.55a.
Basis for Use:	The basis for the inspection frequency for ASME Code Case N-729-1 comes, in part, from the analysis of laboratory and plant data presented in report MRP-111, which was summarized in the safety assessment for RPVCHs in MRP-110. The material improvement factor for PWSCC of Alloy 690/52/152 materials over that of mill-annealed Alloys 600 and 182 was shown by this report to be on the order of 26 or greater. Further evaluations were performed to demonstrate the acceptability of extending the inspection intervals for Code Case N-729-1, Inspection Item B4.40 components and documented in MRP- 375. In summary, the basis for extending the intervals from nominally 10 calendar years to nominally 15 calendar years is based on plant service experience, factor of improvement studies using laboratory data, deterministic study results, and probabilistic study results. Per MRP-375, much of the laboratory data indicated a factor of improvement of 100 for Alloys 690/52/152 versus Alloys 600/182/82 (for equivalent temperature and stress conditions) in terms of crack growth rates (CGRs). In addition, laboratory and plant data demonstrate a factor of improvement in excess of 20 in terms of the time to PWSCC initiation. This reduced susceptibility to PWSCC initiation and growth supports elimination of all volumetric exams throughout the plant service period. However, since work is still ongoing to determine the performance of Alloy 690/52/152 metals, the determination of the proposed inspection interval is based on conservatively smaller factors of improvement. Deterministic calculations demonstrate that the alternative volumetric re-examination schedule is sufficient to detect any PWSCC before it could develop into a safety significant circumferential flaw that approaches the large size (i.e., more than 300°) necessary to produce a nozzle ejection. The deterministic calculations also demonstrate that any base metal PWSCC would likely be detected prior to a through-wall flaw occurring. Probabilistic calculations based on a Monte Carlo simulation model of the PWSCC process, including PWSCC initiation, crack growth, and flaw detection via ultrasonic testing show a substantially reduced effect on nuclear safety compared to a head with Alloy 600 nozzles examined per current requirements.

Basis for Use: (Continued)	Service Experience As documented in MRP-375, the resistance of Alloy 690 and corresponding weld metals Alloy 52 and 152 is demonstrated by the lack of any PWSCC indications reported in these materials, in up to 24 calendar years of service for thousands of Alloy 690 steam generator tubes, and more than 20 calendar years of service for thick-wall and thin-wall Alloy 690 applications. This excellent operating experience includes service at pressurizer and hot-leg temperatures and includes Alloy 690 wrought base metal and Alloy 52/152 weld metal. This experience includes ISI volumetric/surface examinations performed in accordance with ASME Code Case N-729-1 on 14 of the 40 replacement RPVCHs currently operating in the U.S. fleet. This data supports a factor of improvement in time of at least 5 to 20 to detectable PWSCC when compared to service experience of Alloy 600 in similar applications. Factors of Improvement (FOI) for Crack Initiation Alloy 690 is highly resistant to PWSCC due to its approximate 30% chromium content. Per MRP-115, it was noted that Alloy 82 CGR is 2.6 slower than Alloy 182. There is no strong evidence for a difference in Alloy 52 and 152 CGRs. Therefore data used to develop factors of improvement for Alloy 52/152 were referenced against the base case Alloy 182, as Alloy 182 is more susceptible to initiation and growth when compared to Alloy 82. A simple factor of improvement approach was applied in a conservative manner in MRP-375 using multiple data. As discussed in MRP-375, laboratory and plant data demonstrate a factor of improvement in excess of 20 in terms of the time to PWSCC initiation. Factors of Improvement (FOI) for Crack Growth MRP-375 also assessed laboratory PWSCC crack growth rate data for the purpose of assessing FOI values for growth. Data analyzed to develop a conservative factor of improvement include laboratory specimens with substantial levels of cold work. Similar processing, fabrication, and welding practices apply to the original (Alloy 600) and replacement (Alloy 690) components. It is important to note that much of the data used to support Alloy 690 CGRs was produced using materials with significant amounts of cold work, which tends to increase the CGR. MRP-375 considered the most current worldwide set of available PWSCC CGR data for Alloy 690/52/152 materials. Figure 3-2 of MRP-375, compares data from Alloy 690 specimens with less than 10% cold work and the statistical distribution from MRP-55 describing the material variability in CGR for Alloy 600. Most of the laboratory comparisons were bounded by a factor of improvement of 20, and all were bounded by a factor of 10. Most data support a FOI of much larger than 20. This is similar for testing of the Alloy 690 Heat Affected Zone (HAZ) as shown in Figure 3-4 of MRP-375 and for the Alloy 52/152 weld metal as shown in Figure 3-6 of

Basis for Use: (Continued)	MRP-375. Based on the data, it is conservative to assume a FOI of between 10 and 20 for CGRs. <u>Design Features Further Increasing the Resistance of the FNP Replacement Heads to PWSCC</u> Methods were used to reduce residual stress caused by J-groove welding by applying a narrow gap for weld preparation (less weld deposit) and employing spray cooling from inside the CRDM nozzle while producing the weld. Mock-up testing performed to confirm the residual stress reduction showed a maximum of approximately 40 ksi, a 30% reduction over the conventional groove/Shielded Metal Arc Welding (SMAW) process. These methods substantially reduce PWSCC susceptibility beyond that assumed in the generic MRP-375 study, resulting in additional assurance that the FNP Unit 2 head can be operated for 15 years prior to its next volumetric/surface examination with an acceptable level of quality and safety. <u>Deterministic Modeling</u> A deterministic crack growth evaluation is commonly applied to assess PWSCC risks for specific components and operating conditions. The deterministic evaluation is intended to demonstrate the time from an assumed initial flaw to some adverse condition. Deterministic crack modeling results were presented in MRP-375 for previous references in which both growth of part-depth surface flaws and through-wall circumferential flaws were evaluated and normalized to an adjusted growth of 613 degrees Fahrenheit (°F) to bound the PWR fleet. The time for through-wall crack growth in Alloy 600 nozzle tube material, when adjusted to a bounding temperature of 613°F, ranged between 1.9 and 3.8 Effective Full Power Years (EFPY). Assuming a growth FOI of 10 to 20 as previously established for Alloy 690/52/152 materials, the median time for through-wall growth was 37.3 EFPY. In a similar manner, crack growth results for through-wall circumferential flaws were tabulated and adjusted to a temperature of 613°F. Applying a growth FOI of 20 resulted in a median time of 176 EFPYs for growth of a through-wall circumferential flaw to 300 degrees of circumferential extent. The results of the generic evaluation are summarized in Table 4-1 of MRP-375. All cases were bounding and support an inspection interval greater than is being proposed. It is important to note that FNP RPV head temperatures are 600 °F and well within the bounds of the assumptions. Note that for a head with Alloy 600 nozzles and Alloy 82/182 attachment welds operating at a temperature of 600°F, the reinspection years (RIY) = 2.25 constraint on the volumetric/surface reexamination interval of ASME Code Case N-729-1 correspond to an interval of 2.25 EFPYs. Thus, a nominal interval of 15 calendar years for the FNP replacement heads implies a FOI of less than 7 versus the standard interval for heads with Alloy 600 nozzles. It is emphasized that the FOI of 7 (reference Enclosure 2) implied by

Basis for Use: (Continued)	the requested extension period represents a level of reduction in PWSCC crack growth rate versus that for Alloys 600/82/182 that is completely bounded on a statistical basis by the laboratory data compiled in EPRI MRP-375. Given the lack of PWSCC detected to date in any PWR plant applications of Alloys 690/52/152, the simple FOI assessment clearly supports the requested period of extension. Deterministic calculations performed in MRP-375 demonstrate that the alternative volumetric re-examination interval is sufficient to detect any PWSCC before it could develop into a safety significant circumferential flaw that approaches the large size necessary to produce a nozzle ejection. The deterministic calculations also demonstrate that any base metal PWSCC would likely be detected prior to a through-wall flaw occurring. <u>Probability of Cracking or Through-Wall Leaks</u> Probabilistic calculations are based on a Monte Carlo simulation model of the PWSCC process, including PWSCC initiation, PWSCC crack growth, and flaw detection via ultrasonic testing and visual examinations for leakage. The basic structure of the probabilistic model is similar to that used in the MRP-105 technical basis report for inspection requirements for heads with Alloy 600 nozzles, but the current approach includes more detailed modeling of surface flaws (including multiple flaw initiation for each nozzle on base metal and weld surfaces) and the initiation module has been calibrated to consider the latest set of experience for U.S. heads. The outputs of the probabilistic model are leakage frequency (i.e., frequency of through-wall cracking) and nozzle ejection frequency. Even assuming conservatively small factors of improvement for the crack growth rate for the replacement nickel-base alloys (with no credit for improved resistance to initiation), the probabilistic results with the alternative inspection regime show: 1. An effect on nuclear safety substantially within the acceptance criterion applied in the MRP-117 technical basis for Alloy 600 heads, and 2. A substantially reduced effect on nuclear safety compared to that for a head with Alloy 600 nozzles examined per current requirements. Furthermore, the results confirm a low probability of leakage if some modest credit is taken for improved resistance to PWSCC initiation compared to that for Alloys 600 and 182. <u>Conclusion</u> In summary, the basis for extending the intervals from nominally 10 calendar years to nominally 15 calendar years is based on plant service experience, factor of improvement studies using laboratory initiation and growth data, deterministic modeling, and probabilistic study results. The results of the analysis show that the alternative proposed frequency results in a substantially reduced effect on nuclear safety when compared to a head with Alloy 600 nozzles and examined per the current requirements. The proposed revised interval will continue to provide reasonable assurance of structural

Basis for Use: (Continued)	integrity. Additional assurance of structural integrity is provided by the design features of the FNP heads such as the narrow weld preparation design, by the results from the 2005 pre-service volumetric/surface examinations (no detectable defects), and by the successful volumetric/surface examinations recently completed of the FNP Unit 1 head. The FNP Unit 1 head is of the same design and manufacturer with comparable age and operating conditions as Unit 2. Furthermore, the visual examinations and acceptance criteria as required by Item B4.30 of Table 1 of ASME Code Case N-729-1 are not affected by this request and will continue to be performed on a frequency of every third refueling outage or 5 calendar years, whichever is less. As discussed in Section 5.2.3 of MRP-375, the visual examination requirement of the outer surface of the head for evidence of leakage supplements the volumetric/surface examination requirement and conservatively addresses the potential concern for boric acid corrosion of the low-alloy steel head due to PWSCC leakage. Visual examinations were completed on FNP Unit 2 head during the Spring 2010 and Fall 2014 refueling outages with no indication of leakage and must be conducted again by the Spring 2019 refueling outage. For the reasons noted above, it is requested that the NRC authorize this proposed alternative in accordance with 10 CFR 50.55a(z)(1) as the alternative provides an acceptable level of quality and safety.
Duration of Proposed Alternative:	The 4th and 5th ISI Interval, because utilizing the proposed examination frequency will require the examination to be performed in the 5th ISI Interval.
Precedents:	ML14283A045 supplemented by ML15069A227 – ASME Section XI Inservice Inspection Program Proposed Inservice Inspection Alternative N2-14_NDE-002, North Anna Power Station Unit 2.
References:	MRP-375, Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles (EPRI 3002002441)
Status:	Awaiting NRC Review / Approval

Joseph M. Farley Nuclear Plant
Proposed Inservice Inspection Alternative FNP-ISI-ALT-17 Version 2.0

Enclosure 2

Minimum Factor of Improvement (FOI) Calculation for Farley Unit 2

Enclosure 2

Minimum Factor of Improvement (FOI) Calculation for Farley Unit 2

Minimum FOI Implied by Requested Inspection Interval

ASME Code Case N-729-1 is based upon conclusions reached [3] that a reexamination interval between volumetric/surface examinations of one 24-month operating cycle is acceptable for a head with Alloy 600 nozzles and operating at a temperature of 605°F. The inspection period for heads with Alloy 690 nozzles in Case N-729-1 is a nominal 10 years, which represents a minimum implied factor of improvement (FOI) of 5 over Alloy 600.

Per the technical basis documents for ASME Code Case N-729-1 for heads with Alloy 600 nozzles ([2], [3], and [4]), the effect of differences in operating temperature on the required volumetric/surface reexamination interval for heads with Alloy 600 nozzles can be easily addressed on the basis of the Re-Inspection Years (RIY) parameter. The RIY parameter adjusts the effective full power years (EFPYs) of operation between inspections for the effect of head operating temperature using the thermal activation energy appropriate to PWSCC crack growth. For heads with Alloy 600 nozzles, ASME Code Case N-729-1 as conditioned by 10 CFR 50.55a(g)(6)(ii)(D)(2) limits the interval between subsequent volumetric/surface inspections to RIY = 2.25. The RIY parameter, which is referenced to a head temperature of 600°F, limits the time available for potential crack growth between inspections.

The RIY parameter for heads with Alloy 600 nozzles is adjusted to the reference head temperature using an activation energy of 130 kJ/mol (31 kcal/mol) [1]. Based on the available laboratory data, the same activation energy is applicable to model the temperature sensitivity of growth of a hypothetical PWSCC flaw in the Alloy 690/52/152 material of the replacement RVCH. Key laboratory crack growth rate testing data for Alloy 690 wrought material investigating the effect of temperature are as follows:

(1) Results from ANL indicate that Alloy 690 with 0-26% cold work has an activation energy between 100 and 165 kJ/mol (24-39 kcal/mol) [5]. NUREG/CR-7137 [5] concludes that the activation energy for Alloy 690 is comparable to the standard value for Alloy 600 (130 kJ/mol).
(2) Testing at PNNL found an activation energy of about 120 kJ/mol (28.7 kcal/mol) for Alloy 690 materials with 17-31% cold work [6].
(3) Additional PNNL testing determined an activation energy of 123 kJ/mol (29.4 kcal/mol) for Alloy 690 with 31% cold work [7].

These data show that it is reasonable to assume the same crack growth thermal activation energy as was determined for Alloys 600/82/182 (namely 130 kJ/mol (31 kcal/mol)) for modeling growth of hypothetical PWSCC flaws in Alloy 690/52/152 PWR plant components.

As discussed in the MRP-117 [3] technical basis document for heads with Alloy 600 nozzles, effective time for crack growth is the principal basis for setting the appropriate reexamination interval to detect any PWSCC in a timely fashion. U.S. PWR inspection experience for heads with Alloy 600 nozzles has confirmed that the RIY = 2.25 interval results in a suitably conservative inspection program. There have been no reports of nozzle leakage or of safety-significant circumferential cracking for times subsequent to the time that the Alloy 600 nozzles in a head were first examined by non-visual inservice non-destructive examination ([8] and [9]).

For a replacement head with alloy 690/52/152 material that operates at 600 °F the implied FOI needed to support 15-yr inspections is equal to 15 divided by 2.25 which is approximately 7 where 2.25 is RIY.

References:

1. 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," Approved March 28, 2006.

2. *Materials Reliability Program: Reactor Vessel Closure Head Penetration Safety Assessment for U.S. PWR Plants (MRP-110NP)*, EPRI, Palo Alto, CA: 2004. 1009807-NP. [ML041680506]

3. *Materials Reliability Program: Inspection Plan for Reactor Vessel Closure Head Penetrations in U.S. PWR Plants (MRP-117)*, EPRI, Palo Alto, CA: 2004. 1007830. [freely available at www.epri.com; NRC ADAMS Accession No. ML043570129]

4. *Materials Reliability Program: Probabilistic Fracture Mechanics Analysis of PWR Reactor Pressure Vessel Top Head Nozzle Cracking (MRP-105 NP)*, EPRI, Palo Alto, CA: 2004. 1007834. [ML041680489]

5. U.S. NRC, *Stress Corrosion Cracking in Nickel-Base Alloys 690 and 152 Weld in Simulated PWR Environment – 2009*, NUREG/CR-7137, ANL-10/36, published June 2012. [ML12199A415]

6. *Materials Reliability Program: Resistance of Alloys 690, 152, and 52 to Primary Water Stress Corrosion Cracking (MRP-237, Rev.2): Summary of Findings Between 2008 and 2012 from Completed and Ongoing Test Programs*, EPRI, Palo Alto, CA: 2013. 3002000190. [freely available at www.epri.com]

7. M. B. Toloczko, M. J. Olszta, and S. M. Bruemmer, "One Dimensional Cold Rolling Effects on Stress Corrosion Crack Growth in Alloy 690 Tubing and Plate Materials," *15th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors*, TMS (The Minerals, Metals & Materials Society), 2011.

8. EPRI MRP Letter 2011-034, "T_{cold} RV Closure Head Nozzle Inspection Impact Assessment," dated December 21, 2011. [ML12009A042]

9. G. White, V. Moroney, and C. Harrington, "PWR Reactor Vessel Top Head Alloy 600 CRDM Nozzle Inspection Experience," presented at *EPRI International BWR and PWR Material Reliability Conference*, National Harbor, Maryland, July 19, 2012.