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

March 8, 2012

Mr. Mark J. Ajluni
Manager, Nuclear Licensing
Southern Nuclear Operating Company, Inc
40 Inverness Center Parkway
Birmingham, Alabama 35201

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT 2, SAFETY EVALUATION OF ALTERNATIVE HNP-ISI-ALT-11, FOR THE FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL, REACTOR PRESSURE VESSEL CIRCUMFERENTIAL SHELL WELD EXAMINATIONS (TAC NO. ME6290)

Dear Mr. Ajluni:

By letter to the U.S. Nuclear Regulatory Commission (NRC), dated May 17, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML111380211), as supplemented by letter dated August 16, 2011 (ADAMS Accession No. ML112290732), Southern Nuclear Operating Company, Inc. (the licensee) submitted Request for Alternative HNP-ISI-ALT-11 to certain examination requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) at the Edwin I. Hatch Nuclear Plant, Unit 2 (Hatch-2). Specifically, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(a)(3)(i), the licensee proposed to continue the use of the alternative probabilistic fracture mechanics methods for the Reactor Pressure Vessel (RPV) circumferential shell weld examinations at Hatch-2.

Based on the review of the information the licensee provided, the NRC staff concludes that the licensee's request to implement the provisions of the Boiling Water Reactor Vessel and Internals Project (BWRVIP) BWRVIP-05 and Generic Letter (GL) 98-05 for continuing the use of an alternative to the ASME Code, Section XI examination requirements for the RPV circumferential shell welds through the end of the period of extended operations (PEO) will provide an acceptable level of quality and safety at Hatch-2. Therefore, the licensee's proposed alternative in HNP-ISI-ALT-11 is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the remainder of licensed operating period at Hatch-2, which ends on June 13, 2038.

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

If you have any questions concerning this matter, please contact Patrick Boyle at (301) 415-3936.

Sincerely,

Nancy Salgado

Nancy Salgado, Branch Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-366

Enclosure: Safety Evaluation

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR PROPOSED ALTERNATIVE HNP-ISI-ALT-11

REACTOR PRESSURE VESSEL CIRCUMFERENTIAL SHELL WELD EXAMS

EDWIN I. HATCH NUCLEAR PLANT, UNIT 2

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NO.: 50-366 (TAC. NO. ME6290)

1.0 INTRODUCTION

By letter dated May 17, 2011, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML111380211), Southern Nuclear Operating Company, Inc. (the licensee) submitted Request for Alternative HNP-ISI-ALT-11 for the Edwin I. Hatch Nuclear Plant, Unit 2 (Hatch-2). In HNP-ISI-ALT-11, the licensee proposed an alternative to the examination requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, for the reactor pressure vessel (RPV) circumferential shell welds at Hatch-2. The proposed alternative would continue the elimination of the reactor vessel circumferential weld inspection requirements as previously approved by the staff for Hatch-2. The licensee supplemented this request by letter dated August 16, 2011 (ADAMS Accession No. ML112290732) in response to a staff request for additional information (RAI) dated July 28, 2011 (ADAMS Accession No. ML11206A001). The staff has reviewed and evaluated the licensee's request pursuant to the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(a)(3)(i).

2.0 REGULATORY REQUIREMENTS

2.1 Inservice Inspection Requirements

Inservice Inspection (ISI) of the ASME Boiler and Pressure Vessel Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable addenda as required by 10 CFR 50.55a(g), except where specific relief has been granted by the Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 50.55a(g)(6)(i). Title 10 CFR, Section 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if: (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.

Enclosure

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall 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 Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulation requires that inservice examination of components and system pressure tests conducted during the first ten-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) twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable Code of Record for the fourth 10-year interval ISI program at Hatch-2 is the 2001 Edition through the 2003 Addenda of the ASME Code, Section XI.

2.3 Additional Regulatory Guidance

2.3.1 BWRVIP-05 Report

By letter dated September 28, 1995, as supplemented by letters dated June 24 and October 29, 1996, May 16, June 4, June 13, and December 18, 1997, and January 13, 1998 (ADAMS Accession Nos. 9510030130, 9606270011, 9610310079, 9808200035, 9706060278, 9706180413, 9712240085, 9801150081 respectively), the Boiling Water Reactor Vessel and Internals Project (BWRVIP), a technical committee of the BWR Owners Group (BWROG), submitted the Electric Power Research Institute (EPRI) proprietary report TR-105697, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)." The BWRVIP-05 report evaluates the current inspection requirements for RPV shell welds in BWRs, formulates recommendations for alternative inspection requirements, and provides a technical basis for these recommended requirements. As modified, the BWRVIP-05 report proposed to reduce the scope of inspection of BWR RPV welds from essentially 100 percent of all RPV shell welds to examination of 100 percent of the axial (i.e., longitudinal) welds and essentially zero percent of the circumferential RPV shell welds, except for the intersections of the axial and circumferential welds. In addition, the report includes proposals to provide alternatives to ASME Code, Section XI requirements for successive and additional examinations of circumferential welds, provided in paragraph IWB-2420 and IWB-2430, respectively, of Section XI of the ASME Code.

On July 28, 1998, the NRC staff issued a Safety Evaluation (SE) on BWRVIP-05. As a part of its review of the report, the NRC conducted an independent probabilistic fracture mechanics assessment of the results presented in the BWRVIP-05 report. This evaluation concluded that the failure frequency of RPV circumferential welds in BWRs was sufficiently low to justify elimination of ISI of these welds. In addition, the evaluation concluded that the BWRVIP proposals on successive and additional examinations of circumferential welds were acceptable. The evaluation indicated that examination of the circumferential welds will be performed if axial weld examinations reveal an active degradation mechanism. The NRC staff supplemented this evaluation in an SE to the BWRVIP dated March 7, 2000 (ADAMS Accession No. ML031430372). In this SE, the NRC staff updated the interim probabilistic failure frequencies for RPV axial shell welds and revised the Table 2.6-4 to correct a typographical error in the 32 effective full power years (EFPY) mean RT_{NDT} value cited for the limiting Chicago Bridge and Iron (CB&I) case study for circumferential welds.

2.3.2 Generic Letter 98-05

On November 10, 1998, the NRC issued Generic Letter (GL) 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds." GL 98-05 states that BWR licensees may request permanent relief from the ISI requirements of 10 CFR 50.55a(g) for the volumetric examination of RPV circumferential welds (ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, "Circumferential Shell Welds") by demonstrating conformance with the following safety criteria:

1. At the expiration of the operating license, the licensees will have demonstrated that the limiting probability of failure for their limiting RPV circumferential welds will continue to satisfy (i.e., be less than) the limiting conditional failure probability for circumferential weld assessed in the applicable BWRVIP-05 limiting case study.

2. Licensees have implemented operator training and established procedures that limit the frequency of cold overpressure events to the amount specified in the NRC staff's July 28, 1998, SE.

In GL 98-05, the NRC staff stated that licensees applying the BWRVIP-05 criteria would need to continue performing the volumetric inspections of all axial RPV shell welds that are required by the ASME Code, Section XI, Table IWB-2500-1, Inspection Category B-A, Item B1.12, and the augmented volumetric inspections of the RPV axial shell welds that were then required under 10 CFR 50.55a(g)(6)(ii)(A)(2). For plants that are currently licensed to operate in accordance with their initial 40-year operating licenses, the limiting case studies are provided in Table 2.6-4 of the SE on BWRVIP-05 dated July 28, 1998. For plants that have been granted renewed licenses to operate for a 20 year extended period, the limiting case studies are provided in Table 2.6-5 of the SE. In addition to meeting the above criteria, plants granted renewed operating licenses must also demonstrate that the failure probability for their limiting axial shell welds at the end of the period of extended operation is bounded by the limiting axial weld failure frequency of 5×10^{-6} per reactor-year from Table 3 of March 7, 2000 supplemental SE on BWRVIP-05.

3.0 EVALUATION

3.1 Request for Alternative HNP-ISI-ALT-11

The Request for Alternative HNP-ISI-ALT-11 proposed an alternative to the volumetric examination requirements for the RPV circumferential shell welds that would remain in effect for the remainder of the licensed operating period as well as the period of extended operation (PEO), extending from July 31, 2007 through June 13, 2038. The proposed alternative would allow for the elimination of the RPV circumferential shell weld volumetric examinations required by the ASME Code, Section XI in accordance with the alternative probabilistic fracture mechanics methods discussed in the BWRVIP-05 report and GL 98-05. Currently, as the weld-examination relief was only approved through July 31, 2007, examinations would be required during the current ISI interval.

3.2 NRC Staff Evaluation

The 2001 Edition through the 2003 Addenda of the ASME Code, Section XI, Article IWB-2500 requires that components be examined and tested as specified in Table IWB-2500-1 of the ASME Code, Section XI. Table IWB-2500-1, Examination Category B-A, Item No. B1.11 requires a volumetric examination of the RPV circumferential shell welds, with essentially 100 percent volumetric coverage of the examination volume specified in Figure IWB-2500-1 of the ASME Code, Section XI for the entire length of the weld.

Pursuant to 10 CFR 50.55a(a)(3)(i), HNP-ISI-ALT-11 proposed an alternative to the requirements of the ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, pertaining to circumferential shell welds at Hatch-2. Specifically, the licensee requested authorization to obtain relief from the ASME Code, Section XI volumetric examination requirements for the RPV circumferential shell welds in accordance with the alternative probabilistic fracture mechanics methods discussed in the BWRVIP-05 report and with the NRC's guidelines for proposing these alternative programs, as established in GL 98-05.

The licensee has been operating under relief from the above ASME Code, Section XI RPV circumferential shell weld examination requirements based on NRC authorization of a previous request in accordance with the same BWRVIP-05 and GL 98-05 criteria discussed above. This request, submitted as Relief Request (RR) No. RR-38 (ADAMS Accession No. ML040910183) was authorized by the NRC staff through July 31, 2007, for reasons discussed below. The licensee initially submitted this request in a letter dated March 29, 2004 (ADAMS Accession No. ML040910183). In its March 29, 2004 letter, the licensee requested authorization for circumferential weld examination relief for the remainder of the original 40-year licensed operating period. Subsequently, the licensee determined that the request should be amended to include the period of extended operation. Therefore, by letter dated September 13, 2004 (ADAMS Accession No. ML042590229), RR-38 was amended to request authorization for relief for the entire 60-year renewed licensed operating period. The NRC staff evaluated RR-38 for the entire 60-year licensed operating period, as documented in its January 28, 2005 SE (ADAMS Accession No. ML050130317).

For any given RPV weld material, the mean RT_{NDT} value and the conditional probability of failure for the weld increase with the material's neutron fluence. The licensee used the Radiation Analysis Modeling Application (RAMA) fluence methodology for calculating neutron fluence for the amended RR-38. At the time, RAMA was not yet considered conforming to the guidance found in Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The staff determined that the licensee neutron fluence estimates for August 1, 2007, were more conservative than the 54 EFPY projected fluence value based on the nonconforming methodology, even considering a standard 40 percent fluence uncertainty adjustment for nonconforming methodologies. Based on this determination, the staff found that the licensee's August 1, 2007, fluence estimates were sufficient to warrant approval of the fluence values used in the 54 EFPY limiting circumferential weld mean RT_{NDT} analysis through July 31, 2007, and that GL 98-05 Criterion 1 would be satisfied at least up to this date. Otherwise the staff concluded that the licensee had provided a sufficient basis in RR-38 for satisfaction of GL 98-05 Criterion 2 and granted approval for RR-38 until July 31, 2007.

As a result of the conditional approval of RR-38 the relief from the ASME Code, Section XI examination requirements for the RPV circumferential shell welds will expire during this ISI interval. The licensee proposes to extend relief from August 1, 2007, through the end of the PEO. As part of its basis for extending this relief, the licensee stated in HNP-ISI-ALT-11 that the NRC has now addressed the approval and use of the RAMA fluence methodology. The NRC approved the RAMA fluence methodology via SE dated May 13, 2005 (ADAMS Accession No. ML051380572). The NRC staff's conditions for implementation of the RAMA code and the licensee's response to these conditions in HNP-ISI-ALT-11 were as follows:

Condition 1: For plants that are similar in core, core shroud, and downcomer-vessel geometry to that of the Susquehanna and Hope Creek plants, the RAMA methodology can be applied without a bias factor for the calculation of the RPV neutron fluence.

The licensee stated in the submittal that Hatch-2 has a core, core shroud, and downcomer-vessel geometry that is similar to that of the Susquehanna and Hope Creek plants. Specifically this is acceptable as Hatch-2 is a GE-4 design, as are Susquehanna and Hope Creek. Therefore, the RAMA methodology may be applied without a bias factor for the calculation of RPV neutron fluence at Hatch-2.

Condition 2: For plants (or plant groups) with a different geometry than that of the Susquehanna or Hope Creek plants, a plant-specific application for RPV neutron fluence calculations is required to establish the value of a bias factor.

The licensee stated in the submittal that, in meeting Condition 1, Condition 2 is not applicable to Hatch-2.

Condition 3: Relevant benchmarking will be required for application of RAMA fluence calculations to core shroud and reactor internals applications.

The licensee stated in the submittal that HNP-ISI-ALT-11 does not address the core shroud or reactor internals and therefore Condition 3 is not applicable.

The staff reviewed the licensee's responses to these three conditions and determined that Hatch-2 meets the stated provisions in Condition 1 and that Conditions 2 and 3 are not applicable to HNP-ISI-ALT-11. Several further expectations of RAMA users were outlined in the RAMA SE, and the licensee confirmed that they adhered to these in their response to the staff RAI. In particular the licensee confirmed that they used detailed plant-specific geometry, core operating history, the BUGLE-96 nuclear data library, and a P_5 Legendre polynomial approximation for the iron inelastic scattering. Therefore, the staff found that the licensee is eligible to apply the RAMA fluence methodology without a bias factor for determining RPV neutron fluence values at Hatch-2.

The licensee utilized the RAMA fluence methodology to calculate a new projected fluence value for the limiting circumferential weld at Hatch-2. The licensee provided an updated calculation of the mean RT_{NDT} value for the limiting RPV circumferential weld at Hatch-2 as Table 1 of Enclosure 1 of the application, reproduced here:

	Limiting 64 EFPY CE-VIP Case Study Table 2.6-5 of NRC SER for BWRVIP-05	Limiting 64 EFPY CEOG Case Study Table 2.6-5 of NRC SER for BWRVIP-05	Hatch-2 Limiting Circ. Weld at License Expiration (54 EFPY) from RR-38	Hatch-2 Limiting Circ. Weld at License Expiration (50.1 EFPY) from HNP-ISI-ALT-11
Cu%	0.13	0.183	0.047	0.047
Ni%	0.71	0.704	0.049	0.049
Chemistry Factor (CF)	151.7	172.2	31.0	31.0
Fluence (10^{19} n/cm^2)	0.40	0.40	0.244	0.324
ΔRT_{NDT}	113.2	128.5	19.2	18.9
Initial RT_{NDT}	0	0	-50	-50
Mean RT_{NDT}	113.2	128.5	-30.8	-31.1
P(F/E)	1.99×10^{-4}	4.38×10^{-4}	---	---

The licensee's updated calculation of the mean RT_{NDT} value for the limiting circumferential weld at Hatch-2, based on the new projected PEO fluence (determined using the RAMA methodology), is shown in the last column of the above table. The staff independently calculated the updated mean RT_{NDT} value, noting that the ΔRT_{NDT} for 50.1 EFPY would be 21.4°F, not 18.9°F as submitted. Regardless the mean RT_{NDT} value is bounded by both CE case studies from Table 2.6-5 of the NRC SE for BWRVIP-05, as shown in the first two data columns of the above table. As discussed in the NRC SE for BWRVIP-05, P(F/E) represents the circumferential weld conditional failure probability calculated by the NRC for each of the CE case studies. The actual weld failure frequency is determined in each instance by multiplying the P(F/E) value by the frequency of occurrence for a low temperature over-pressure event, which is 1×10^{-3} per reactor operating year. Thus, the actual weld failure frequency for each of the limiting case studies is calculated to be no more than 1.99×10^{-7} for the CE-VIP case study (first data column above) and 4.38×10^{-7} for the CEOG case study (second column above). The CEOG value is considered to be an acceptable bounding limit on the circumferential weld failure frequency, and the staff's calculated mean RT_{NDT} value of -28.6 °F conclusively demonstrates that the Hatch-2 limiting circumferential weld failure frequency at the end of the PEO will be substantially less than the case study value. It should be noted that the licensee has revised the number of EFPYs corresponding to the new calculated EOL fluence value from 54 EFPY to 50.1 EFPY. The licensee explained that this change resulted from a more detailed evaluation of plant capacity factor over the remainder the plant's 60 year extended operating life. Based on the above evaluation, the staff determined that the data provided by the licensee adequately demonstrated that the limiting circumferential weld at Hatch-2 will satisfy the provisions of Criterion 1 from GL 98-05 through the end of the period of extended operation.

The licensee had provided an evaluation in RR-38 for demonstrating that the failure probability for the limiting axial weld at Hatch-2 was bounded by that calculated by the NRC in the March 7, 2000, supplement to the BWRVIP-05 SE at the end of the PEO. The licensee provided a new fluence value for the limiting axial weld at Hatch-2 calculated using the NRC-approved RAMA fluence methodology in the submittal. The staff determined that the new limiting axial weld fluence value resulted in a mean RT_{NDT} value that was less than the bounding RT_{NDT} value used in the axial weld failure probability analysis from the NRC's March 7, 2000, supplemental SE. Specifically the staff calculated mean RT_{NDT} would be 15.6°F, well below the bounding value of

114°F. Therefore, the requirements for the axial weld failure probability will continue to be met through the end of the PEO.

Regarding the GL 98-05 Criterion 2, the staff determined in the RR-38 SE dated December 6, 2007 (ADAMS Accession No. ML073130188) that, "based on the licensee's information provided about the systems that inject at high pressures, operator training, and plant-specific procedures at Hatch, Units 1 and 2, the possibility of a low temperature overpressurization event will be minimized, and thus, the licensee provided a sufficient basis to support the NRC staff's approval of the alternative examination request for circumferential shell welds in the Hatch, Units 1 and 2 RPVs." In addition, as the licensee continues to implement the operator training and established procedures that limit the frequency of cold overpressure events as substantively detailed in RR-38, the staff considers that this fulfills Criterion 2 of GL 98-05.

Based on the above evaluation, the staff determined that the licensee adequately demonstrated that Hatch-2 will remain in compliance with the acceptance criteria for circumferential weld examination relief from the NRC SE on BWRVIP-05, and Generic Letter 98-05 through the end of the PEO. Furthermore, the licensee has demonstrated that the limiting axial weld failure probability will remain in compliance with the acceptance criteria from the NRC supplemental SE on BWRVIP-05, dated March 7, 2000, and that axial welds and intersecting regions of circumferential welds will be examined in accordance with 10 CFR 50.55a requirements. Therefore, the staff found that the licensee's request to extend the RPV circumferential weld examination relief through the end of the period of extended operation is acceptable, and that the proposed alternative in HNP-ISI-ALT-11 will provide an acceptable level of quality and safety.

4.0 CONCLUSION

The staff concludes that the licensee's request to implement the provisions of BWRVIP-05 and GL 98-05 pertaining to the continuation of relief from the ASME Code, Section XI examination requirements for the RPV circumferential shell welds through the end of the PEO will provide an acceptable level of quality and safety at Hatch-2. Therefore, the licensee's proposed alternative in HNP-ISI-ALT-11 is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the remainder of licensed operating period at Hatch-2, which ends on June 13, 2038. All other requirements of the ASME Code, Section XI, for which relief has not been specifically requested and approved, remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: Dan Widrevitz

Date: March 8, 2012

If you have any questions concerning this matter, please contact Patrick Boyle at (301) 415-3936.

Sincerely,

/RA/

Nancy Salgado, Branch Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-366

Enclosure: Safety Evaluation

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