December 6, 2007

Mr. Dennis R. Madison
Vice President - Hatch
Edwin I. Hatch Nuclear Plant
11028 Hatch Parkway North
Baxley, GA 31513

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1 (HATCH 1), SAFETY EVALUATION FOR ALTERNATIVE ISI-ALT-08 (TAC NO. MD4704)

Dear Mr. Madison:

By letter to the Nuclear Regulatory Commission (NRC) dated March 8, 2007, Southern Nuclear Operating Company, Inc., submitted a request for relief, alternative ISI-ALT-08, to the examination requirements of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code), Section XI, for the reactor vessel (RV) circumferential shell welds at Hatch 1. A previous request for relief, RR-38, was authorized by the NRC staff on January 28, 2005 for a limited time based on the licensee's usage of a methodology that had not, at that time, been approved. The licensee's March 8, 2007, submittal addressed the outstanding issues identified in the NRC staff's evaluation of January 28, 2005.

Based on its review the NRC staff concludes, as stated in the enclosure, that the licensee's request pertaining to the continuation of relief from the ASME Code, Section XI, examination requirements for the Hatch 1 RV circumferential shell welds through the end of the period of extended operation will provide an acceptable level of quality and safety. Therefore, the licensee's proposed alternative in ISI-ALT-08 is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the remainder of the licensed operating period at Hatch 1, which ends on August 6, 2034. 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.

Sincerely,

/RA/

Evangelos C. Marinos, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-321

Enclosure:
Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR PROPOSED ALTERNATIVE ISI-ALT-08

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NO. 50-321

1.0 INTRODUCTION

By letter dated March 8, 2007, Southern Nuclear Operating Company, Inc. (the licensee), submitted Request for Alternative ISI-ALT-08 for Edwin I. Hatch Nuclear Plant, Unit 1 (Hatch 1). In ISI-ALT-08, the licensee proposed an alternative to the examination requirements of the American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code* (ASME Code), Section XI, for the reactor vessel (RV) circumferential shell welds at Hatch 1. The Nuclear Regulatory Commission (NRC) staff (staff) has reviewed and evaluated the licensee's request pursuant to the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.55a(a)(3)(i).

2.0 REGULATORY REQUIREMENTS

2.1 Inservice Inspection Requirements

Inservice Inspection (ISI) of the ASME 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 NRC pursuant to 10 CFR 50.55a(g)(6)(i). 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 of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety."

Section 50.55a(g)(4), requires that " . . . components (including supports) which are classified as ASME Code Class 1, Class 2 and Class 3 must meet the requirements, except design and access provisions and pre-service examination requirements, set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda . . . to the extent practical within the limitations of design, geometry, and materials of construction of the components." Section 50.55a(g)(4) specifies the edition and addenda of the ASME Code for conducting inservice examination of components and system pressure tests during the first 10-year interval and subsequent intervals. The applicable Code of record for the fourth 10-year interval ISI program at Hatch 1 is the 2001 edition through the 2003 addenda of ASME Code, Section XI. The fourth 10-year interval ISI program at Hatch 1 began on January 1, 2006, and ends on December 31, 2015.

2.2 Augmented Inservice Inspection Requirements for RV Shell Welds

Section 50.55a(g)(6)(ii)(A)(2) requires that "All licensees shall augment their reactor vessel examination by implementing once, as part of the inservice inspection interval in effect on September 8, 1992, the examination requirements for reactor vessel shell welds specified in Item B1.10, of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," in Table IWB-2500-1 of subsection IWB" of the ASME Code. Section XI, Item B1.10 includes the volumetric examination requirements for both circumferential RV shell welds, as specified in Item B1.11, and longitudinal RV shell welds, as specified in Item B1.12. Section 50.55a(g)(6)(ii)(A)(2) defines "essentially 100% examination" as covering 90 percent or more of the specified examination volume of each weld.

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, 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 RV 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 RV welds from essentially 100 percent of all RV shell welds to examination of 100 percent of the axial (i.e., longitudinal) welds and essentially zero percent of the circumferential RV 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 paragraphs 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. This evaluation concluded that the failure frequency of RV 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. In this SE, the NRC staff updated the interim probabilistic failure frequencies for RV axial shell welds and revised the Table 2.6-4 to correct a typographical error in the 32 effective full power years (EFPY) Mean nil-ductility reference temperature (RT_{NDT}) value cited for the limiting Chicago Bridge and Iron (CB&I) case study for circumferential welds. The correction changed the 32 EFPY Mean RT_{NDT} value for the CB&I case study from 109.5 °F to 134.9 °F.

In the BWRVIP-05 report, the BWRVIP committee concluded that the conditional probabilities of failure for BWR RV circumferential welds are orders of magnitude lower than that of the axial welds. As a part of its review of the report, the staff conducted an independent probabilistic fracture mechanics assessment of the results presented in the BWRVIP-05 report. The staff's assessment conservatively calculated the conditional failure probability values for RV axial and circumferential welds during the initial 40-year license period and at conditions approximating an 80-year vessel lifetime for a BWR nuclear plant. The failure frequency is calculated as the product of the frequency for the critical (limiting) transient event and the conditional probability of failure for the weld.

The staff determined the conditional probability of failure for axial and circumferential welds in BWR vessels fabricated by CB&I, Combustion Engineering (CE), and Babcock and Wilcox (B&W). The analysis identified a cold overpressure event that occurred in a foreign reactor as the limiting event for BWR RVs, with the pressure and temperature from this event used in the probabilistic fracture mechanics calculations. The NRC staff estimated that the probability for the occurrence of the limiting overpressurization transient was 1×10^{-3} per reactor year. For each of the vessel fabricators, Table 2.6-4 of the NRC staff's SE of July 28, 1998, identifies the conditional failure probabilities for the plant-specific conditions with the highest projected reference temperature (for that fabricator) through the expiration of the initial 40-year license period. Table 2.6-5 of NRC staff's SE of July 28, 1998, identifies the conditional failure probabilities for the plant-specific conditions with the highest projected reference temperature (for that fabricator) through the expiration of an 80-year license period, which constitutes the licensing basis if two 20-year extended periods of operation would have been granted for a BWR-designed nuclear power plant.

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 (i.e., for the remaining term of operation under the existing, initial, license) relief from the inservice inspection requirements of 10 CFR 50.55a(g) for the volumetric examination of circumferential reactor pressure vessel welds (ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item 1.11, Circumferential Shell Welds) by demonstrating that: (1) at the expiration of their license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the staff's July 30, 1998, safety evaluation, and (2) licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the staff's July 30, 1998, safety evaluation. Licensees will still need to perform their required inspections of "essentially 100 percent" of all axial welds."

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 RV shell welds that are required by ASME Code, Section XI, Table IWB-2500-1, Inspection Category B-A, Item B1.12, and the augmented volumetric inspections of the RV axial shell welds that are 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 this 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 the March 7, 2000, supplemental SE on BWRVIP-05.

3.0 EVALUATION

3.1 Request for Alternative ISI-ALT-08

Request for Alternative ISI-ALT-08 proposed an alternative to the volumetric examination requirements for the RV circumferential shell welds that would remain in effect for the remainder of the licensed operating period, including the period of extended operation, at Hatch 1. The proposed alternative would allow for the elimination of the RV 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.

3.2 Staff's 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 RV circumferential shell welds, with essentially 100% volumetric coverage of the examination volume specified in Figure IWB-2500-1 of ASME Code, Section XI for the entire length of the weld.

Pursuant to 10 CFR 50.55a(a)(3)(i), ISI-ALT-08 proposed an alternative to the requirements of ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, pertaining to circumferential shell welds at Hatch 1. Specifically, the licensee requested authorization to obtain relief from the ASME Code, Section XI volumetric examination requirements for the RV 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 RV circumferential shell weld examination requirements and the augmented examination requirements of 10 CFR 50.55a(g)(6)(ii)(A)(2) 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 (Reference (Ref.) 1), was authorized by the NRC staff for only a limited duration. Consequently, this circumferential weld examination relief expired on July 31, 2007, for reasons discussed below. The licensee initially submitted this request in a letter dated March 29, 2004. In its March 29, 2004, letter, the licensee requested authorization for circumferential weld examination relief for the remainder of its 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, RR-38 was amended to request authorization for circumferential weld examination relief for the entire 60-year renewed licensed operating period. The NRC staff evaluated RR-38 for the entire 60-year licensed operating period, and its evaluation was documented in its January 28, 2005, SE (Ref. 2).

In its amended submittal for RR-38 (Ref. 1), the licensee provided a technical basis for relief from the ASME Code, Section XI requirements for examination of RV circumferential shell welds. The licensee based RR-38 on the NRC's provisions for obtaining relief from these requirements in GL 98-05 and the guidelines of BWRVIP-05. In RR-38, the licensee had cited the following acceptance criteria as the bases for evaluating the acceptability of RR-38:

Per the NRC SE on BWRVIP-05 dated July 28, 1998, and GL 98-05, BWR licensees may request relief from the ISI requirements of 10 CFR 50.55a(g) for volumetric examination of reactor vessel (RV) circumferential welds by demonstrating conformance with two safety criteria as discussed above in section 2.3.2 of this evaluation.

As previously established in BWRVIP-05, the NRC SE for BWRVIP-05, and GL 98-05, the limiting conditional failure probability for each of the circumferential welds assessed in the limiting case study from the NRC SE for BWRVIP-05 is associated with a specific mean RT_{NDT} value for that weld. This mean RT_{NDT} value was essentially derived from the conditional failure probability for each limiting case study. The limiting case studies, conditional failure probabilities, and associated mean RT_{NDT} values corresponding to a 64 EFPY operating period are listed in Table 2.6-5 of the NRC SE for BWRVIP-05. According to GL 98-05, Criterion 1 above may be met by demonstrating that the end-of-life (EOL) mean RT_{NDT} value for the limiting circumferential weld at the plant is less than that in the applicable limiting case study from Table 2.6-5 of the NRC SE for BWRVIP-05. In a letter dated March 29, 2004, in support of RR-38, the licensee provided its calculations for demonstrating conformance with the Criterion 1.
In its evaluation of RR-38 (Ref. 2), the NRC staff had performed an independent calculation of the mean RT_{NDT} value for the limiting circumferential weld at Hatch 1 through the end of the period of extended operation (54 EFPY). This calculation confirmed that the EOL mean RT_{NDT} value for the limiting circumferential weld at Hatch 1 was correctly calculated by the licensee and is less than the mean RT_{NDT} value corresponding to the limiting CE circumferential weld failure probability from Table 2.6-5 of the NRC SE for BWRVIP-05.

For any given RV weld material, the mean RT_{NDT} value and the conditional probability of failure for the weld increases with the material's neutron fluence, as projected to the expiration of the operating license for the facility. Accordingly, the neutron fluence values for the RV circumferential welds at the inside surface of the RV are critical inputs to the mean RT_{NDT} calculations for the plant. Therefore, notwithstanding the staff's determination above that the licensee correctly calculated the mean RT_{NDT} values for the limiting circumferential weld at Hatch 1, the ultimate validity of these calculations for satisfying Criterion 1 above are dependent on the neutron fluence inputs to the calculation being based on a valid neutron fluence calculation methodology. The licensee's method for calculating neutron fluence at the time did not conform to the NRC staff's recommended methodology in Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The licensee had committed to have a RG 1.190-conforming neutron calculation methodology (i.e., the Radiation Analysis Modeling Application (RAMA) computer code) for Hatch 1 by December 15, 2004. However, the approval of the RAMA code was delayed, and in a letter dated July 13, 2004, the licensee requested to revise its commitment date for having a neutron fluence calculation methodology that conforms with RG 1.190 from December 15, 2004, to July 31, 2007. The licensee stated, as justification for this extension, that the December 15, 2004, commitment date was selected arbitrarily (based on the expectation that RAMA would be approved by this date) and that there was sufficient conservatism in the pressure-temperature (P-T) limit curves calculated using fluence values based on the non-conforming fluence methodology to operate the plant safely until July 31, 2007. At that time, the NRC staff was reviewing the RAMA code and determined that the review would be complete before July 31, 2007. Furthermore, the licensee had demonstrated that the estimated fluence value on August 1, 2007, based on the non-conforming fluence methodology, would be only 44.8 percent of the 54 EFPY neutron fluence value estimated by this non-conforming methodology for the limiting circumferential welds at Hatch 1. The staff subsequently determined that this neutron fluence estimate for August 1, 2007, was more conservative than the 54 EFPY projected fluence value based on the nonconforming methodology, even considering a standard 40 percent fluence uncertainty adjustment for the nonconforming methodology. Based on this determination, the staff had 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. As a result, the staff had limited its approval of the licensee's GL 98-05 Criterion 1 analysis of the limiting circumferential weld conditional failure probability for RR-38 only through July 31, 2007. In its submittal for RR-38 (Ref. 1), the licensee had stated its bases for meeting Criterion 2 of GL 98-05 above. The licensee stated that its bases for meeting this criterion are documented in a section of the enclosure to a December 2, 1998, letter entitled, "Consideration of Low Temperature – Over Pressurization Events," and that this documentation was applicable to the evaluation of RR-38 for conformance with Criterion 2. This documentation included a detailed description of the operational controls, system design considerations, procedural considerations, and operator training programs that are in place to minimize the possibility of a low temperature overpressurization event. In its evaluation of RR-38 (Ref. 2), the NRC staff had reviewed these programs and determined that, based on the information provided by the licensee regarding the plant's high pressure coolant injection systems, operating training programs, and plant-specific procedures, the possibility of a low temperature overpressurization event will be minimized. The staff therefore concluded that the licensee had provided a sufficient basis in RR-38 for satisfaction of Criterion 2 from GL 98-05.

Based on the staff's determination that the licensee's analysis for Criterion 1 would be acceptable only through July 31, 2007, and that the licensee had fully satisfied Criterion 2, the staff authorized the alternative provisions of RR-38 only through July 31, 2007. Accordingly, the relief from the ASME Code, Section XI examination requirements for the RV circumferential shell welds has now expired, and the licensee has requested in ISI-ALT-08 to extend this relief from August 1, 2007 through the end of the period of extended operation, August 6, 2034. As part of its basis for extending this relief, the licensee stated in ISI-ALT-08 that the NRC staff has addressed the approval and use of the RAMA fluence methodology. The licensee cited a May 13, 2005, letter (Ref. 3) from W. H. Bateman, NRC, to Bill Eaton, BWRVIP Chairman, granting conditional approval for the use of the RAMA code. The NRC staff's conditions for implementation of the code and the licensee's response to these conditions in ISI-ALT-08 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 RV neutron fluence.

In its response to Condition 1, the licensee stated that Hatch 1 has a core, core shroud, and downcomer-vessel geometry that is similar to that of the Susquehanna and Hope Creek plants. Therefore, the RAMA methodology may be applied without a bias factor for the calculation of RV neutron fluence at Hatch 1.

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 RV neutron fluence calculations is required to establish the value of a bias factor.

In its response, the licensee indicated that, in meeting Condition 1, Condition 2 is not applicable to Hatch 1.

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

In its response to Condition 3, the licensee indicated that ISI-ALT-08 does not address the core shroud or reactor internals. Therefore, Condition 3 is not applicable for the use of the RAMA methodology in ISI-ALT-08.

The staff reviewed the licensee's responses to these three conditions and determined that Hatch 1 meets the stated provisions in Condition 1. Conditions 2 and 3 are not applicable to ISI-ALT-08 at Hatch 1. Therefore, the staff found that the licensee is eligible to apply the RAMA fluence methodology without a bias factor for determining RV neutron fluence values at Hatch 1.

The licensee utilized the RAMA fluence methodology to calculate a new projected fluence value for the limiting circumferential weld at Hatch 1. However, the licensee did not report this fluence value. The licensee indicated only that the RAMA calculation report showed a small increase in this projected fluence value over the previous value used for Hatch 1 in RR-38 and that the limiting circumferential welds at Hatch 1 will continue to satisfy the provisions of Criterion 1 from GL 98-05 through the end of the period of extended operation. The staff found that this statement alone did not provide an adequate basis for concluding that the licensee had satisfied Criterion 1 using the new fluence data.

In a request for additional information (RAI) issued on July 6, 2007, the staff requested that the licensee provide an updated calculation of the mean RT_{NDT} value for the limiting RV circumferential weld at Hatch 1 that utilizes the new projected fluence value calculated using the RAMA methodology. The staff also requested that this updated mean RT_{NDT} value be compared with the mean RT_{NDT} value corresponding to the limiting conditional failure probability for circumferential welds in CE-fabricated RVs from the applicable limiting case study in Table 2.6-5 of the NRC SE for BWRVIP-05. In its August 3, 2007, RAI response, the licensee provided the requested mean RT_{NDT} analysis, which is shown below.

	Limiting 64 EFPY CE-VIP Case Study Table 2.6-5 of NRC SE for BWRVIP-05	Limiting 64 EFPY CEOG Case Study Table 2.6-5 of NRC SE for BWRVIP-05	Hatch 1 Limiting Circ. Weld at License Expiration (54 EFPY) from RR-38	Hatch 1 Limiting Circ. Weld at License Expiration (49.3 EFPY) from ISI-ALT-08
Cu%	0.13	0.183	0.197	0.197
Ni%	0.71	0.704	0.060	0.060
Chemistry Factor (CF)	151.7	172.2	91.0	91.0
Fluence (10^{19} n/cm^2)	0.40	0.40	0.236	0.296
Delta RT_{NDT}	113.2	128.5	48.5	53.3
Initial RT_{NDT}	0	0	-10	-10
Mean RT_{NDT}	113.2	128.5	38.5	43.3
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 1, based on the new projected EOL fluence (determined using the RAMA methodology), is shown in the last column of the above table. The staff independently confirmed that the licensee correctly calculated this updated mean RT_{NDT} value. This 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 1.99×10^{-7} for the CE-VIP case study (first data column above) and 4.38×10^{-7} for the Combustion Engineering Owners Group (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 licensee's calculated mean RT_{NDT} value of 43.3 °F from the last column conclusively demonstrates that the Hatch 1 limiting circumferential weld failure frequency at EOL will be substantially less. 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 49.3 EFPY. In its RAI response, 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 in its RAI response adequately demonstrated that the limiting circumferential weld at Hatch 1 will satisfy the provisions of Criterion 1 from GL 98-05 through the end of the period of extended operation. Therefore, the staff found that the issue addressed by the RAI was resolved.

The licensee had provided an evaluation in RR-38 for demonstrating that the failure probability for the limiting axial weld at Hatch 1 was bounded by that calculated by the NRC in the March 7, 2000, supplement to the BWRVIP-05 SE at the end of the period of extended operation. As with the circumferential weld evaluation discussed above, this evaluation involved demonstrating that the EOL mean RT_{NDT} value for the limiting axial weld is less than that used by the NRC for calculating a bounding axial weld failure frequency. However, the limiting axial weld fluence value used for the axial weld evaluation in RR-38 was determined using the same fluence methodology as that used for determining the circumferential weld fluence values (i.e., the methodology that did not conform to RG 1.190). By letter dated, November 13, 2007, the licensee provided a new fluence value for the limiting axial weld at Hatch 1 that was calculated using the NRC-approved RAMA fluence methodology. The staff found 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 staff's March 7, 2000, supplemental SE. Therefore, the requirements for the axial weld failure probability will continue to be met through the end of the period of extended operation. The licensee previously indicated in RR-38 that axial welds and intersecting regions of circumferential welds will be examined in accordance with ASME Code, Section XI requirements to the extent practical, dependent upon interference by another component or restrictions due to geometrical configuration. For those cases where the reduction in axial weld examination coverage is greater than 10 percent, relief from full examination coverage will be requested pursuant to 10 CFR 50.55a requirements.

Based on the above evaluation, the staff determined that the licensee adequately demonstrated that it will remain in compliance with both of the acceptance criteria for circumferential weld examination relief from the NRC SE on BWRVIP-05 dated July 28, 1998, and GL 98-05 through the end of the period of extended operation (49.3 EFPY). 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 RV circumferential weld examination relief through the end of the period of extended operation is acceptable, and the proposed alternative in ISI-ALT-08 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 RV circumferential shell welds through the end of the period of extended operation will provide an acceptable level of quality and safety at Hatch 1. Therefore, the licensee's proposed alternative in ISI-ALT-08 is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the remainder of licensed operating period at Hatch 1, which ends on August 6, 2034, and extends up to 49.3 EFPY of facility operation. All other requirements of 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.

5.0 REFERENCES

1. Letter from H. L. Sumner, Southern Nuclear Operating Company, Inc., to USNRC Document Control Desk, "Hatch Nuclear Plant Third 10-Year Interval Inservice Inspection Program Submittal of Revised Relief Request RR-38," September 13, 2004, ADAMS ML042590229.

2. Letter from J. A. Nakoski, USNRC, to H. L. Sumner, Southern Nuclear Operating Company, Inc., "Edwin I. Hatch Nuclear Plant, Units 1 and 2 – Evaluation of Relief Request (RR) Number 38 (TAC NOS. MC2381 and MC2382)," January 28, 2005, ADAMS ML050130317.

3. Letter from W. H. Bateman, USNRC, to B. Eaton, BWRVIP Chairman, Entergy Operations, Inc., "Safety Evaluation of Proprietary EPRI Reports, 'BWR Vessel and Internals Project, RAMA Fluence Methodology Manual (BWRVIP-114),' 'RAMA Fluence Methodology Benchmark Manual – Evaluation of Regulatory Guide 1.190 Benchmark Problems (BWRVIP-115),' 'RAMA Fluence Methodology – Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5 (BWRVIP-117),' 'RAMA Fluence Methodology Procedures Manual (BWRVIP-121),' and 'Hope Creek Flux Wire Dosimeter Activation Evaluation for Cycle 1 (TWE-PSE-001-R-001)' (TAC No. MB9765)," May 13, 2005, ADAMS ML051380572.

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