November 10, 2005

Mr. H. L. Sumner, Jr.
Vice President - Nuclear
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Company, Inc.
P.O. Box 1295
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SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT 1 UPDATED ANALYSIS OF CORE SHROUD VERTICAL WELDS (TAC NO. MC5500)

By letter dated December 3, 2004, the Southern Nuclear Operating Company, Inc. (SNC, the licensee) submitted its re-evaluation of core shroud vertical welds V5 and V6 for Edwin I. Hatch Nuclear Plant (Hatch), Unit 1, considering the ultrasonic inspection results for these two welds obtained in the unit's Spring 2004 refueling outage. The information in the December 3, 2004, letter superceded the licensee's letter dated November 14, 2003. The licensee intended to demonstrate through a flaw evaluation using a proposed fluence-dependent crack growth rate and a revised fracture mechanics methodology that the unit can be operated without repair of the V5 and V6 welds for more than a fuel cycle.

The Nuclear Regulatory Commission (NRC) staff has completed its review and found that the licensee's flaw evaluation meets the intent of Section XI of the American Society of Mechanical Engineers (ASME) Code. Since the safety factors associated with the detected cracks for the limiting faulted condition are greater than the value of 1.5 specified in the ASME Code, the NRC staff concludes that Hatch, Unit 1 can be operated without repair of the V5 and V6 welds for a reinspection interval of 10 years. The NRC staff noted, however, that the NRC staff's review of BWRVIP-76, "BWR Vessel and Internals Project: BWR Core Shroud Inspection and Flaw Evaluation Guidelines (BWRVIP-76)," which contains core shroud inspection and flaw evaluation guidelines, is ongoing and will be completed in the near future. At that time, the NRC staff expects that the licensee will evaluate whether the conclusions in the NRC staff's forthcoming safety evaluation (SE) for BWRVIP-76 will have any impact on the 10-year reinspection interval for the Hatch, Unit 1 welds. The licensee stated in its response to the NRC staff request for additional information that it will inform the NRC staff within 45 days of issuance of the SE for BWRVIP-76 if it intends to deviate from the approved guidelines.

If you have questions regarding this evaluation, please contact me at 301-415-1055.

/RA/

Christopher Gratton, Sr. Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-321

Enclosure: As stated

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

EDWIN I. HATCH NUCLEAR PLANT, UNIT 1

REEVALUATION OF THE CORE SHROUD WELDS V5 AND V6

SOUTHERN NUCLEAR OPERATING COMPANY, INC

DOCKET NO. 50-321

1.0 INTRODUCTION

By letter dated December 3, 2004 (Ref. 1), the Southern Nuclear Operating Company, Inc. (the licensee) submitted its reevaluation of core shroud vertical welds V5 and V6 for Edwin I. Hatch Nuclear Plant (Hatch), Unit 1 using the ultrasonic (UT) inspection results for these two welds obtained in the unit's Spring 2004 refueling outage. The licensee intended to demonstrate that the unit can be operated without repair of the V5 and V6 welds for more than a fuel cycle by proposing the use of a flaw evaluation using a bounding, fluence-dependent crack growth rate (CGR) and an elastic-plastic fracture mechanics (EPFM) analysis. The Nuclear Regulatory Commission (NRC) review includes the additional information provided by the licensee in a letter dated April 15, 2005, in response to the NRC staff's request for additional information (RAI).

There are no specific regulations governing shroud crack propagation, but the NRC has requested that licensees provide for the structural stability of the shroud. This review does not involve a licensing amendment request. The specific objective of this review is to establish the acceptability of the fast (E > 1.0 MeV) neutron fluence on the shroud and its correlation to the CGRs in BWRVIP-99 (Ref. 3). The analysis documented in Ref. 1 contains responses to NRC staff questions related to the November 14, 2003, submittal by Structural Integrity Associates, Inc., and the fluence analysis by the TransWare Enterprises Inc., using the RAMA code.

The RAMA code has recently been approved by the NRC staff for use in licensing actions with certain limitations. One of the limitations requires benchmarking for the specific application for which the code is being used. The licensee did not submit benchmarking information with the shroud fluence calculations, therefore, the NRC staff used information from outside the submittal to corroborate and substantiate its conclusion.

2.0 REGULATORY EVALUATION

The inservice inspection of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Code and applicable editions and addenda as required by Title 10 of the Code of *Federal Regulations* (10 CFR) 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(I).

However, since reactor pressure vessel (RPV) internals are not part of the RPV pressure boundary (and only the failure of some attachments directly connected to the RPV can affect the pressure boundary), they are not ASME Code Class 1, 2, and 3 components. As a result, the Boiling Water Reactor (BWR) Vessel and Internals Project (BWRVIP) has developed its guidelines for conducting inspections and evaluating inspection results for various BWR vessel internals where Section XI of the ASME Code does not apply. These inspection and evaluation guidelines are contained in numerous approved BWRVIP reports. The guidelines, as accepted or modified by the staff in the NRC Safety Evaluations (SEs) for these BWRVIP reports, have become standards for inspection and flaw evaluation for BWR vessel internals.

BWRVIP-76, “BWR Vessel and Internals Project: BWR Core Shroud Inspection and Flaw Evaluation Guidelines (BWRVIP-76),” which is currently under NRC staff review, documents the latest BWR core shroud inspection and flaw evaluation guidelines. Once approved, the next inspection of the Hatch, Unit 1 core shroud welds will be conducted in accordance with BWRVIP-76, as accepted or modified by the NRC at the completion of its review. A BWRVIP commitment letter dated October 30, 1997, requires the licensee to inform the NRC through the BWRVIP within 45 days of the issuance of the approved SE for a BWRVIP report if it intends to deviate from the approved guidelines.

3.0 TECHNICAL EVALUATION

3.1 RAMA Assessment

The fluence calculations are provided in a separate report by TransWare Enterprises, Inc. (Ref. 2). For assessment of the fast fluence (excluding the reactor pressure vessel), calculations were performed with the RAMA code. The relevant shroud benchmarking data are those related to the Hatch, Unit 1 pump riser pad. These data were verified with measured dosimetry scrapings taken from the Hatch, Unit 1 pump riser pad. The fact that the submittal and the scrapings were taken from the Hatch,Unit 1 is coincidental. The scrapings predated the submittal by several years. The pump riser pad scrapings are well within the vessel inner radius and off the core mid-plane and, therefore, provide data points to be compared to the corresponding values calculated using the RAMA code. Comparison of the results indicate very good agreement of the measured and the calculated fast fluence values. The NRC staff review of the Hatch, Unit 1 pump riser pad scrapings fluence data is documented in NUREG/CR-6880 (Ref. 3) and provides the most relevant data for benchmarking the capability of the RAMA code to calculate shroud fluence as it relates to estimating the irradiation-assisted stress-corrosion cracking propagation rates. The NRC staff’s assessment of the RAMA code does not extend to the calculation of vessel fluence within the scope of RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.”

The following is the NRC staff’s evaluation of the assumptions and approximations made during the fluence calculation for the Hatch, Unit 1 shroud.

- In TWE HATCH1-001-R-001 (Ref. 2) the region outside the pressure vessel was not modeled. This assumption does not have any effect on the shroud fluence. The shroud is at a considerable distance from the outer surface of the vessel and as such reflected neutrons have a negligible chance of reaching the shroud.

- In TWE HATCH1-001-R-001 the azimuthal location of the cracked vertical welds V5 and V6 are brought into the first core octant. This assumption does not affect the calculated fluence value due to azimuthal symmetry.

- The licensee estimated a 17 percent conservatism between measured and calculated fluence values for the Hatch, Unit 1 pump riser pad. NRC staff review indicates that the conservatism between the measured and calculated fluence values is only 7 percent. While there is less conservatism based on the NRC staff's review, the overall conclusion remains unaffected in that the RAMA code calculates conservative fluence values when compared to relevant benchmarking data.

- The source data used in TWE HATCH1-001-R-001 are the same as those used in the benchmark for the pump riser pad. The pump riser pad is located about 12 inches above the core mid-plane. As a result the axial source variation is not accounted for. This effect is not quantified, but the NRC staff estimates it to be of the order of a few percent.

- For cycles 1 to 12, the axial power distribution was represented by four segments. This is particularly important for the top and bottom core segments because the outer assembly flux gradient is high. It is estimated that the source at the bottom portion of the core may be overestimated by 200 to 300 percent. For cycles 13 to 19, the axial power distribution was represented by 25 axial segments and provided excellent accuracy. The accuracy of the axial flux distribution is important because weld H5 is at an elevation affected by the lower core power distribution.

- The core operating data assumed a 40 percent core moderator void fraction. This value is representative of core conditions well above the core mid-plane. The effect of this assumption is that the flux will be over estimated in the upper part of the core.

- The core operating data used in the pump riser pad benchmark were calculated for Discrete Ordinates Radiation Transport/Three Dimensional Discrete Ordinates Radiation Transport (DORT/TORT) based on a calculated spectrum and fuel exposure. RAMA on the other hand assumes fissile number densities and calculates the spectrum and the neutron source. If there is a difference in the calculation of the spectrum and the fissile number densities, it was estimated to be of the order of few percent.

- No plant specific operating reactor data were provided for water densities outside the core (for example in the downcomer and the jet pump and pump risers). In TWE HATCH1-001-R-001 generic numbers from NUREG-6115 (Ref. 5) were employed. The effect of this assumption cannot be quantified without plant specific data. It is estimated that it of the order of a few percent.

Summarizing the effect of the assumptions and approximations the NRC staff concludes that the shroud fluence is likely to be conservatively calculated using the RAMA code.

The calculated fluence values for the entire extent of the V5 and V6 welds exceed $5x10^{20}$ n/cm^2 but are lower than the upper limit of $3x10^{21}$ n/cm^2, therefore, the CGR shown in BWRVIP-99 (Ref. 4) should be applied. In addition, vertical welds V4 and V8 indicate one crack each and, in view of the discussion above, can be considered within the range of Ref. 4. Finally, the lower plate horizontal weld H5 is well within the guidance of Ref. 4.

Note: The axial range of the cracks in vertical welds V5 and V6 extend from about 8 inches to about 109 inches above the bottom active fuel plane. It is remarkable to observe that both vertical welds with cracks are those closest to corner fuel assemblies with the highest exposure.

3.2 CGR Assessment

A typical flaw evaluation for detected flaws includes five elements: (1) flaw sizing, (2) the applied stress intensity factor ($K_{applied}$) calculation and the associated crack growth evaluation, (3) a driving force evaluation for the final flaw size using linear elastic fracture mechanics (LEFM), EPFM, or limit load analysis according to the projected failure mode, (4) a failure resistance evaluation considering embrittlement due to various environmental conditions and the projected failure mode, and (5) a stability evaluation using NRC approved acceptance criteria including appropriate structural factors (SFs). The five elements of the licensee's flaw evaluation are evaluated in the following sections.

3.2.1 Flaw Sizing

The licensee's sizing of the flaws in RPV core shroud V5 and V6 welds is based on the 2004 UT inspection results. The inspection results indicate that the flaw lengths range from 1.26 inches to 20.31 inches and flaw depths from 50 to 75 percent of the shroud thickness. The licensee assumed in its flaw evaluation that all detected flaws are through-wall, making the 20.31-inch flaw in the V6 weld the most limiting. These flaws were originally detected in 1996, and the fracture mechanics evaluation of the flaws was submitted to the NRC on July 16, 1996. The flaw evaluation was updated in 1999 and in 2002.

To verify the flaw sizing, the NRC staff compared the 2004 UT inspection results for the V6 weld with the 1999 inspection results of the same weld as reported in an earlier submittal dated November 14, 2003. The NRC staff found that the characterized flaws based on the 2004 UT inspection results are drastically shorter than those based on the 1999 inspection results. To explain this significant discrepancy, the licensee clarified in its April 15, 2005, response to the NRC staff's RAI that, instead of using UT, the 1999 inspection results were obtained from a two-sided enhanced visual examination (EVT). The licensee provided procedural information for the EVT and identified factors which could lead to measurement discrepancies. Separately, the response stated that the 2004 UT inspection results compared favorably with earlier UT results obtained in 1997 with respect to flaw location, depth, and length, giving additional credibility to the 2004 UT results. Extra EVT of the V5 and V6 welds from the outer diameter of the core shroud also confirmed the 2004 UT results in both location and length. Based on the additional information discussed above regarding the use of different nondestructive examination techniques in the 1999 and 2004 examinations, and the 1997 UT inspection results supporting the 2004 UT results, the NRC staff concludes that the 2004 UT inspection results are credible and the associated flaw sizing can be used in the subsequent flaw evaluation.

3.2.2 The Applied Stress Intensity Factor and the Crack Growth Rate

The licensee used a bounding CGR of 5×10^{-5} inch/hour in its evaluation and, therefore, did not calculate the applied stress intensity factors for the growing crack. The submittal indicates that

although portions of the V5 and V6 welds will exceed 1 x 10^{21} n/cm^2 (E $ 1 one million electron volts (MeV)) before the end of the 10-year reinspection interval, this CGR is still applicable.

The CGR of 5 x 10^{-5} inch/hour has been approved by the NRC as the bounding rate for a variety of BWR internals subject to intergranular stress-corrosion cracking and has been used in past plant-specific flaw evaluations for core shrouds under normal water chemistry conditions with fluence under 5 x 10^{20} n/cm^2. Out of the concern regarding the adequacy of applying this bounding CGR to the current evaluation of the core shroud vertical welds considering the additional CGR data reported in BWRVIP-99, "BWR Vessel and Internals Project - Crack Growth Rates in Irradiated Stainless Steels in BWR Internal Components," the NRC staff requested that the licensee examine the new information in BWRVIP-99 to ensure that the CGR of 5 x 10^{-5} inch/hour remains bounding for the current high fluence application. This RAI was issued based on a prior version of the December 3, 2004, submittal. However, since all RAIs for the prior version are applicable to the current submittal, the licensee provided its response to these RAIs in Enclosure 1 of Ref.1.

Enclosure 1 of the submittal states, "the [applied] stress intensity factor is calculated to be about 13 ksi / in for the longest through-wall crack in the V6 weld. Using the normal water chemistry disposition curve [of BWRVIP-99], the crack growth rate is about 5 x 10^{-5} in/hr." This $K_{applied}$ of 13 ksi / in is based on the operating load. Although the major driving force for circumferential flaws (the subject of BWRVIP-99) is residual stresses, the NRC staff concludes that the CGR disposition curve of BWRVIP-99 is still applicable to V5 and V6 flaws because the key parameter for CGR is the $K_{applied}$ regardless of whether the stresses are residual or operational.

Acceptance of the normal water chemistry disposition curve of BWRVIP-99 in/hr for circumferential flaws was documented in the NRC SE dated July 29, 2005. The BWRVIP-99 normal water chemistry disposition curve is for the fluence range of 5 x 10^{20} n/cm^2 (E $ 1 MeV) to 3 x 10^{21} n/cm^2 (E $ 1 MeV). To evaluate the applicability of using this disposition curve to the Hatch, Unit 1 core shroud, the NRC staff has evaluated the plant-specific fluence calculations for Hatch, Unit 1 core shroud and concluded in Section 3.1 of this evaluation that the shroud fluence, which is well within the applicable fluence range for the disposition curve, is likely to be conservatively calculated using the RAMA neutron transport code. Therefore, the NRC staff finds that the Hatch, Unit 1 fluence is acceptable for the purpose of estimating the CGR using the disposition curve of BWRVIP-99, and the additional CGR data reported in BWRVIP-99 support the use of the bounding rate of 5 x 10^{-5} in/hr in the current application.

3.2.3 Elastic Plastic Fracture Mechanics (Driving force, Failure Resistance, and Flaw Stability)

The licensee used the guidelines provided by BWRVIP-76, as supplemented by BWRVIP-100, "BWRVIP-100: BWR Vessel and Internals Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds," to perform the crack stability analysis. BWRVIP-76 documents a core shroud cracking analysis which identifies the evaluation methodology, i.e., LEFM or limit load analysis, to be used for analyzing cracked BWR core shrouds at various fluence levels. This core shroud cracking analysis was supplemented recently by BWRVIP-100 to include the EPFM methodology as an additional analytical tool for core shroud flaw evaluation. Table 1 of this SE highlights the differences in the core shroud cracking analyses of BWRVIP-76 and BWRVIP-100.

Based on the SE dated March 1, 2004, for BWRVIP-100, which accepted EPFM as a flaw evaluation methodology for core shrouds with fluence less than 3 x 10^{21} n/cm^2 (E $ 1 MeV), the NRC staff concludes that using EPFM in the current application and using the J-integral resistance (J-R) curves from BWRVIP-100 in the current EPFM evaluation is appropriate. The licensee's plant-specific EPFM analysis for Hatch, Unit 1 contains some special features, and these features will be evaluated in the following paragraphs.

Table 1: Appropriate Methodologies for BWR Core Shroud Flaw Evaluation

<table>
<tr><td>Fluence (n/cm^2, E $ 1.0 MeV)</td><td colspan="6">3E20 – 5E20 – 1E21 – 3E21 –</td></tr>
<tr><td rowspan="2">BWRVIP-76</td><td rowspan="2">limit load</td><td>limit load</td><td>LEFM (K_{Ic} = 150 ksi / in)</td><td colspan="3" rowspan="2">plant-specific analysis for NRC review</td></tr>
<tr><td colspan="2">Note: the limiting one</td></tr>
<tr><td rowspan="3">BWRVIP-100</td><td>EPFM or</td><td colspan="2">EPFM or</td><td colspan="2">EPFM or</td><td rowspan="3">LEFM (K_{Ic} = 50 ksi / in)</td></tr>
<tr><td rowspan="2">limit load</td><td>limit load</td><td>LEFM (K_{Ic} = 150 ksi / in)</td><td>limit load</td><td>LEFM (K_{Ic} = 112 ksi / in)</td></tr>
<tr><td colspan="2">Note: the limiting one</td><td colspan="2">Note: the limiting one</td></tr>
</table>

To simplify the postprocessing of the finite element analysis results to calculate applied J, the driving force for EPFM, the licensee employed a crack tip opening displacement (CTOD) approach in this application. Since the applied J calculation based on the CTOD is an indirect approach, the NRC staff requested that the licensee provide a comparison of the applied J determined from a direct approach (i.e., a volume integral enveloping the crack front), with that from the CTOD approach. Enclosure 1 of Ref. 1 provides two examples for validation: (1) a part-through-wall circumferential crack in a cylinder of R/t = 10 (R is radius and t is thickness) under remote tension and (2) the same example but with a through-wall circumferential crack. The results indicate that the indirect and direct approaches give similar applied J values for the plane strain condition, but different applied J values (about 25 percent difference) for the plane stress condition. The NRC staff has no concern about the inconsistent results for the plane stress and plane strain conditions because this application used the results based on the more conservative plane strain assumption. The applied load selected by the licensee is the faulted condition, which was reviewed by the licensee and determined to be limiting. As to SFs, the licensee used the ASME Code-specified SF of 1.5 for an axial flaw in a component under the faulted condition even though the BWR core shroud is not subject to the ASME Code inspection. Further, although EPFM analysis could be executed using applied J values and the

J-R curve, the licensee used the J-integral/Tearing modulus (J-T) approach. Both methods are permitted by the ASME Code for specific applications using EPFM and have been accepted by the NRC in the past.

As another check of the validity of the finite element modeling, the licensee provided additional calculated applied J-T data for much longer flaws and demonstrated that the near vertical applied J-T curve for this core shroud application is caused primarily by the low applied stresses in its April 15, 2005, response to the NRC staff's RAI. This clarification has resolved the NRC staff's concern regarding the physical meaning of the steep slope of the applied J-T curve.

3.2.4 Evaluation of the EPFM Results

The licensee performed the EPFM evaluation based on the 2004 UT inspection results for two cases: the flaw with 8 years of growth and the flaw with 15 years of growth. In the evaluation, flaw instability is defined as the intersection of the applied J-T curve and the material J-T curve and the corresponding J at the intersection is designated as $J_{instability}$. The licensee defined the calculated SF as $(J_{instability}/J_{applied})^{1/2}$ and presented its EPFM results by this single parameter. The results show that the calculated SF for a flaw after 15 years of growth is 1.91, having extra margin when compared to the ASME Code-specified SF of 1.5. Hence, a 10-year reinspection interval is justified.

This NRC staff considered the licensee's use of the calculated SF as a key parameter in the EPFM evaluation to compare with the ASME Code-specified SF for the LEFM evaluation conservative and acceptable. The licensee defined the calculated SF as $(J_{instability}/J_{applied})^{1/2}$. This is appropriate because the $K_{applied}$ is proportional to the square root of the applied J. Since the licensee has demonstrated that the flaw after 15 years of growth will not become unstable, the NRC staff agrees with the licensee that an analytical basis exists to justify a 10-year reinspection interval.

4.0 CONCLUSIONS

The NRC staff reviewed the analyses contained in the submittals and found that the licensee's flaw evaluation, which is in accordance with the guidance in BWRVIP-76 as supplemented by BWRVIP-99 and BWRVIP-100, meets the intent of the rules in Section XI of the ASME Code. Based on the licensee's evaluation, the NRC staff approves a 10-year reinspection interval for the V5 and V6 core shroud welds for Hatch, Unit 1. The NRC staff notes, however, that the NRC staff's review of BWRVIP-76, which contains core shroud inspection and flaw evaluation guidelines, is ongoing and will be completed in the near future. At that time the NRC staff expects that the licensee will evaluate whether the conclusions in the NRC staff's forthcoming SE for BWRVIP-76 will have any impact on the 10-year reinspection interval for the Hatch, Unit 1 welds. As clarified in the licensee's April 15, 2005, response to the NRC staff's RAI, the licensee will inform the NRC staff within 45 days of the issuance of the BWRVIP-76 SE if it intends to deviate from those approved core shroud inspection guidelines.

5.0 REFERENCES

1. Letter from H.L. Sumner, Jr. Southern Nuclear Operating Company to US Nuclear Regulatory Commission "Edwin I. Hatch Nuclear Plant Unit 1 Updated Analysis of Core Shroud Vertical welds and Supplemental Information" December 3, 2004.

2. TWE-HATCH1-001-R-001, “Evaluation of > 1.0 MeV Fluence in the Hatch 1 Shroud” by D.B. Jones TransWare Enterprises Inc. September 10, 2003.

3. NUREG/CR-6880, “DORT/TORT Analyses of the Hatch, Unit 1 Jet Pump Riser Brace Pad Neutron Dosimetry Measurements with Comparisons to Predictions Made with RAMA” US Nuclear Regulatory Commission, to be published.

4. BWRVIP-99, “BWR Vessel and Internal Project Crack Growth Rates in Irradiated Stainless Steel in BWR Internal Components” EPRI, December, 2001.

5. NUREG/CR-6115, “PWR and BWR Pressure Vessel Fluence Calculation Benchmark Problems and Solutions” US Nuclear Regulatory Commission, September, 2001.

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Date: November 10, 2005