February 13, 2009

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. 2 (HATCH-2), SAFETY EVALUATION AND REQUEST FOR ADDITIONAL INFORMATION FOR REACTOR CORE SHROUD RELIEF REQUEST (TAC NO. MD9579)

Dear Mr. Madison:

By letter to the Nuclear Regulatory Commission (NRC) dated September 3, 2008, as supplemented on December 19, 2008, and January 9, 2009, Southern Nuclear Operating Company, Inc., submitted a request for authorization, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i), to use modified core shroud stabilizer assemblies and to inspect the upper support arm inner and outer corner radius locations using an American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code), Section XI, VT-1, examination technique at the first refueling outage following installation of the modified assemblies and to subsequently re-examine the locations at 10-year intervals.

Based on the NRC staff's review of the information provided in the submittals listed above, the staff finds that the proposed modification and the proposed inspection of the upper support arm will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative, is authorized for the remainder of the licensed operating period at Hatch-2, which ends on June 13, 2038. 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.

Sincerely,

/RA/

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

Docket No. 50-366

Enclosure: Safety Evaluation

cc w/encl: Distribution via ListServ

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Vice President - Hatch
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR REACTOR CORE SHROUD RELIEF REQUEST

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NO. 50-366

1.0 INTRODUCTION

In its letter dated September 3, 2008, from the Southern Nuclear Operating Company (SNC, the licensee), to the Nuclear Regulatory Commission (NRC, the Commission) requesting authorization under the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.55a(a)(3)(i), to modify the core shroud stabilizer assemblies for the Edwin I. Hatch Nuclear Plant, Unit 2 (Hatch-2). Additional information was submitted in letters dated December 19, 2008, and January 8, 2009. The September 3, 2008, letter proposed to replace all four Hatch-2 core shroud upper support assemblies and tie rod top nuts with a modified design during the unit's 2009 Refueling Outage due to the potential for cracking of the original upper support assemblies. The September 3, 2008, letter indicates that the proposed modification is being submitted to NRC for review and approval as an alternative repair pursuant to 10 CFR 50.55a(a)(3)(i).

2.0 REGULATORY REQUIREMENTS

The inservice inspection of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME Code and applicable editions and addenda as required by the 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). 10 CFR 50.55a(a)(3) states in part that alternatives to the requirements of paragraph (g) may be used, when authorized by NRC, if the licensee demonstrates that: (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.

The licensee submitted this relief request for NRC staff's review and approval to use as an alternative repair under the provisions of 10 CFR 50.55a(a)(3)(i).

3.0 TECHNICAL EVALUATION

3.1 ASME Code Requirement

The core shroud tie rod assemblies are not included as a repair option under Section XI of the ASME Code. However, the licensee's proposed alternative was designed to comply with the

requirements of the ASME Code, Section III, Division 1, Subsection NG, 2001 Edition and 2003 Addenda and Code Case N-60-5, "Materials for Core Support Structures, Section III, Division 1."

Enclosure

3.2 Licensee's Basis for Requesting Relief

Industry experience has shown that boiling-water reactor (BWR) core shroud welds are subject to intergranular stress corrosion cracking (IGSCC) and/or irradiation-assisted stress corrosion cracking. As a result, NRC staff issued Generic Letter 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds at Boiling Water Reactors," which led BWR facilities to perform detailed inspections and analyses of the core shrouds. By letters dated July 3, and August 25, 1995, SNC requested NRC approval of a repair of the Hatch-2 core shroud by installation of four tie rods. NRC staff, in its letter dated September 25, 1995, found the proposed repair acceptable as allowed by the requirements of 10 CFR 50.55a(a)(3)(i). SNC installed the core shroud tie rod assemblies at Hatch-2 in 1995 on a pre-emptive basis in lieu of ultrasonic inspection of the core shroud H1 through H8 horizontal welds. The tie rods functionally replace the core shroud horizontal H1 through H10 welds. The General Electric Company (GE) designed and installed the Hatch-2 tie rod assemblies. GE provided core shroud repairs using tie rods to many other domestic BWR plants.

In Spring 2006, cracking was discovered in two of the four tie rod upper supports at Hatch Nuclear Plant (Hatch-1) during an in-vessel visual inspection. The apparent root cause was IGSCC in the Alloy X-750 tie rod upper support material. Alloy X-750 material is susceptible to IGSCC if subjected to sustained, large peak stress conditions. GE conducted an evaluation to determine if the potential for IGSCC in the Alloy X-750 tie rod structural components of other BWR core shroud repairs designed by GE could be a reportable condition under 10 CFR Part 21. GE used the criterion provided in the BWR Vessel and Internals Project (BWRVIP) report BWRVIP-84, "BWR Vessel and Internals Project, Guidelines for Selection and Use of Materials for Repairs to BWR Internals," for the IGSCC susceptibility assessment of the Alloy X-750 components in the tie rod vertical load path.

Based on the Hatch-1 finding, GE revised the assessment of the GE tie rod upper support design life and determined that the potential for a high peak surface stress existed for the Hatch-1 tie rod design. The potential for high peak stress in the original tie rod upper bracket design at Hatch-1 was attributed to the lack of a specified radius at the corner junction between horizontal and vertical legs of the bracket which created a high stress concentration. This high peak stress reduced the design life of the original tie rod upper support. Tie rod inspections performed during each refueling outage could justify continued operation on a cycle-by-cycle basis. However, the licensee has determined that the most prudent course of action is pre-emptive replacement of the tie rod upper supports with a modified upper support design capable of operation through the end of the unit's renewed operating license term.

GE conducted an extent of condition review to determine if other Alloy X-750 tie rod components had similar potential for high peak stresses. GE identified that the root radii of the threads in the tie rod threaded components of the original Hatch-1 design may be smaller than the nominal values used in previous design evaluations. Hence, the licensee proposed to include a modified tie rod nut that incorporates an improved locking mechanism. To improve IGSCC resistance, the new tie rod nuts will include a specified root radius sufficient to minimize the peak principle stress to within the same criterion as used for the upper support.

3.3 Technical Evaluation of Modified Tie Rod Assembly

3.3.1 Design Objectives

The objectives of the proposed tie rod repair modifications were to design and install replacement upper support assemblies and tie rod top nuts that will remain resistant to IGSCC through the end of the Hatch-2's current operating license and to ensure that the replacement components interface correctly with the existing core shroud repair hardware.

3.3.2 Design Criteria

The modified upper supports and tie rod top nuts comply with the criteria delineated in the BWRVIP-02-A, "BWR Vessel and Internals Project, BWR Core Shroud Repair Design Criteria, Revision 2," and the BWRVIP-84 reports with no exceptions taken. The codes and design standards used for construction of the original tie rod assemblies were delineated in GE Specification 25A5718, Rev. 0, which was included in the licensee's September 3, 2008, core shroud repair submittal. These codes and design standards remain applicable to the proposed modifications, as well as other more recent standards (e.g., the BWRVIP-84 report).

3.3.3 Description of Repair Components and Design Features

The geometry of the upper support replacement hardware (upper support and tie rod nut) was shown in Figure 1 of the GE stress analysis report GE-NE-0000-0080-0259-R2 (Enclosure 3 to the licensee's September 3, 2008, letter). These newly-designed components incorporate features that improve their ability to resist IGSCC. These features included: (1) a large fillet radius at the corner of the upper support; (2) a geometry change for the upper horizontal arm, (3) elimination of sharp edges; (4) a larger root radius of the tie rod nut threads; and (5) use of a more IGSCC resistant material (XM-19) for some upper support components and the tie rod nut. The original tie rod installation required that cutouts be made in the shroud head flange to accommodate the upper support. The geometry of the cutouts will be changed to accommodate the simplified design of the modified supports.

The licensee indicated in its September 3, 2008, letter that the proposed modification has an insignificant effect on the following attributes that were evaluated in the original design analysis:

(1) seismic effect of the reactor internals under all design basis earthquake loadings;

(2) original seismic dynamic analysis ;

(3) analysis related to original cracked shroud weld;

(4) load case and load combinations that include normal, upset, emergency and faulted conditions;

(5) original analysis related to horizontal and vertical deflections of the shroud;

(6) effect of the tie rod modification on the original analysis related to tie rod thermal preloads;
(8) capturing of loose parts (i.e., all threaded parts, pins etc.,) that may affect the flow;

(9) flow-induced vibration associated with a change in annulus flow velocity; and

(10) original evaluation related to radiation effects.

3.3.4 Structural Evaluation

Finite element analysis (FEA) and/or manual calculations were used to structurally analyze the modified upper support assembly components. The original FEA of the upper support brackets used the ANSYS [software] finite element code. The original model was not capable of capturing peak stresses. A revised FEA of the replacement upper support bracket has been performed using the ANSYS computer program. Details of the analysis, such as input criteria, applied loading, material properties, boundary conditions, and analysis methods were described in the GE stress analysis report (Enclosure 3 to the licensee's September 3, 2008, letter). The licensee also contracted Structural Integrity Associates, Inc. (SIA) to perform an independent third-party review of the GE upper support FEA. SIA developed a separate ANSYS model and their results compared favorably to the GE results for the maximum principle tensile stress. The ANSYS program is widely used for analysis of safety-related components.

The replacement hardware components (upper support and tie rod nut) were evaluated for their susceptibility to IGSCC. The BWRVIP-84 report limits the allowable peak stress for Alloy X-750 material to 0.8 times the minimum yield stress (S_y) at operating temperature. The design goal established by the licensee was to maintain total stress, which includes peak stress, well below the GE 0.8 S_y criterion for all the new Alloy X-750 upper support components, thereby providing margin to the criteria in the BWRVIP-84 report.

The replacement hardware components were also evaluated against ASME Code allowable stresses. The design stress intensity and S_y values for Alloy X-750 material were specified in accordance with Code Case N-60-5. This is consistent with the BWRVIP-84 report. The membrane and bending stresses were calculated for these components and shown to meet the ASME Code allowable stress limits.

3.3.5 Materials and Fabrication

For the proposed modification, the licensee used the following materials:

(1) Alloy X-750 -Tie rod upper support and retainer spring, and

(2) XM-19 -Tie rod nut, top support bracket, retainer pin, socket head screw cap and dowel pin.

The above-listed materials have been used for many other reactor internal components and have demonstrated good resistance to stress corrosion cracking in laboratory testing and long-term service experience in non-welded and low sustained operating stress conditions. Nickel-based Alloy X-750 and XM-19 austenitic stainless steel are acceptable per the BWRVIP-84 report and Section III of the ASME Code.

Consistent with the fabrication requirements specified in the BWRVIP-84 report, the licensee proposed to not utilize any avoidable crevices in the upper bracket design. If crevices are inherently present, the licensee proposed to implement the following requirements:

(1) design of such features should avoid sensitized areas and should utilize IGSCC resistant materials

(2) such features should be vented to the extent practical to minimize stagnant conditions

The licensee stated that there are no welds in the replacement upper support assemblies, and they will be procured and processed to prevent sensitized material by meeting the requirements of the BWRVIP-84 report.

3.4 Pre-Modification Inspection

Since the Hatch-2 core shroud stabilizers were recently inspected in 2007, the licensee proposed to inspect only the new hardware that will be installed during the proposed modification.

3.5 Post-Modification Inspection

A post-modification inspection prior to reactor pressure vessel (RPV) reassembly will include a general post-maintenance visual inspection and recording of the fit of the core shroud support hardware onto the core shroud to confirm that there are no interferences at the support locations and that the installation is in accordance with the requirements of the modification drawings and GE Installation Specification 26A5716. The licensee’s inspection will verify, as a minimum, the following attributes:

(a) all retainer clips and latches are in place for the upper spring, the mid-support, the lower spring, and the tie rod nut,

(b) upper stabilizer, the mid-support, and the lower spring are all in contact with the RPV wall,

(c) contact exists between the lower support clevis pin and hook and on both sides of the hook,

(d) contact exists between the mid-support and shroud, and between upper support and shroud and,

(e) "as-left" inspection cleanliness is equal to or better than the "as-found" inspection.

3.6 Inspections During Subsequent Refueling Outages

In the first refueling outage following installation of the modified tie rod upper supports, the licensee will inspect the tie rod assemblies in accordance with the requirements defined in report BWRVIP-76, "BWR Vessel and Internals Project BWR Core Shroud Inspection and Flaw Evaluation Guidelines," Section 3.5, Option 1 or 2. The licensee will also repeat the post-modification inspections (items a through e).

4.0 STAFF EVALUATION

4.1 Safety Evaluation of the Modified Tie Rod Assembly

4.1.1 Structural Design Evaluation

The licensee's vendor, GE, used ANSYS, Version 10.0, to determine the total stress for safety-related components. NRC staff discussed the GE method of validating this computer code in a September 27, 2007, SE to M. A. Balduzzi, Entergy Nuclear Operations (Agencywide Documents Access and Management System (ADAMS) Accession No. ML072420161). GE indicated that ANSYS has been used for several RPV internals evaluations (e.g., Clinton Power Station, Unit 1 - Core Shroud Repair, Docket No. 50-461). In addition, ANSYS's applicability and validity was demonstrated by running a series of verification cases (over 200) that exercise the elements and options used in the finite element code. The verification cases were extracted from textbooks in which classical or theoretical solutions are published. Since GE has validated the ANSYS computer code, NRC staff concluded that its use for determining principle stresses in the safety analysis is acceptable.

The original analysis included loadings for normal operating conditions, two upset conditions, three emergency conditions and three faulted conditions. In the original analysis, one of the emergency conditions (Emergency 2) and one of the faulted conditions (Faulted 2) were considered bounding and were also used for the proposed modification. NRC staff finds this acceptable because the licensee used bounding conditions based on a methodology that is consistent with the original design analysis.

The design analysis indicates that the licensee used the following aspects of the design criteria for the modification of the upper core shroud support and tie rod nuts:

(a) maximum tensile principal stress due to sustained normal condition loads for IGSCC evaluation of Alloy X-750 materials, and

(b) stress intensities for the upper core shroud support for ASME Code compliance

For Alloy X-750 material, the licensee established a criterion for IGSCC susceptibility for the proposed modification which specifies that total stress (including peak stress) be below, and thus more conservative, than the GE $0.8S_y$ criterion for all new Alloy X-750 upper support components. NRC staff reviewed the stress values for all Alloy X-750 materials that are used in the proposed modification. The limiting location is at the large fillet radius of the upper support which resulted in a peak stress intensity, due to the sustained normal condition loading, that is less than the BWRVIP-84 established allowable value of $0.8S_y$ for IGSCC of Alloy X-750. It is noted that the IGSCC susceptibility of the Hatch-2 original upper support was identified based on the maximum calculated stress exceeding the BWRVIP-84 criterion of $0.8S_y$. By creating an upper core shroud support large fillet radius, the applied stresses are reduced significantly in the upper support which ultimately reduces the susceptibility to IGSCC. Based on its review, NRC staff concludes that for all Alloy X-750 materials that are used in the proposed modification, the peak stress intensity values comply with the IGSCC design criterion.

For XM-19 material, which is more IGSCC resistant, BWRVIP-84 does not contain any specific criterion. However, in accordance with GENE-0000-0063-5939, "Assessment of SCC Crack Initiation in Hatch-2 and Pilgrim Type XM-19 Tie Rods," dated February 22, 2007, the GE stress analysis report (Enclosure 3 to the licensee's September 3, 2008, letter) limits the maximum plastic strain for XM-19 material. NRC staff reviewed the analysis results for the XM-19 components of the proposed modification. The XM-19 (replacement) tie rod nut and (existing) tie rod threads each resulted in a total strain (elastic and plastic) which meets the set strain limit criterion. All other XM-19 components remained in the elastic range, thereby, meeting the strain limit criterion.

As noted in Section 3.3.3, the shroud flange cutout slots will be changed to accommodate the proposed modified upper support horizontal arm. In response to NRC staff's request for additional information (RAI), the licensee provided figures showing the original and proposed replacement upper support, a figure showing the existing versus the proposed shroud flange cutout slots and a summary of the bearing stresses on the shroud head flange due to the proposed modification and analyzed loading conditions (normal, upset, emergency and faulted). The stress summary shows the bearing stresses to be less than the ASME code allowable values. Therefore, NRC staff finds the proposed changes to the shroud head flange acceptable.

The licensee used the ASME Code design criteria that are specified in the ASME Code, Section III, Subsection NG, Core Support Structures, 2001 Edition through and including the 2003 Addenda for the proposed core shroud tie rod modifications. All the stress intensity values and fatigue usage factors for the components that are used in the proposed core shroud tie rod modifications complied with the ASME Code, Section III, Subsection NG, design criteria. Therefore, the staff concluded that the licensee's proposed core shroud tie rod modifications are structurally qualified to meet the ASME Code, Section III, design criteria, and the design is consistent with the original design basis of the tie rod repair.

To determine the principle stress for the design analysis, GE used the computer code ANSYS for performing a FEA. The stress analysis modeled the bearing interface of the horizontal arm of the upper support with that of the flange using contact elements with a coefficient of friction, as indicated in Section 5.3.1 of Enclosure 3 to the licensee's September 3, 2008 submittal. The selected coefficient of friction used is consistent with the GE standard design specification for core support structures. NRC staff finds the value of the friction factor used to be acceptable because it is a reasonable typical value for static conditions. Based on the above, NRC staff concludes that the licensee's stress analysis has adequately modeled the interaction of the horizontal arm of the upper support and the shroud flange.

The licensee's contractor SIA performed an independent third-party review of the GE's analysis by developing a separate ANSYS FEA. In its response to an NRC staff RAI, the licensee stated that the comparison of the maximum principle tensile stress between the two calculations was within 0.5 percent. The 0.5 percent variation is insignificant. The comparison of the two analyses provides further assurance of the accuracy of the GE results.

Based on its above review, NRC staff finds the proposed modification of the core shroud stabilizer assemblies for Hatch-2 structurally adequate.

4.1.2 Materials and Fabrication

Section 6.3, "Materials Fabrication," in Enclosure 1 to the licensee's September 3, 2008, letter indicates that the replacement hardware conforms specifically to the conditions described in Sections 3.5.2, 3.6.2 and 3.6.3 of NRC's safety evaluation (SE) for the BWRVIP-84 report, dated September 6, 2005. These sections of the staff's SE describe requirements for surface preparation techniques to be followed after cold working of XM-19, and Alloy X-750 materials and electrical discharged machining of Alloy X-750 material to reduce susceptibility to IGSCC and fatigue.

In an RAI dated December 8, 2008, NRC staff requested that the licensee identify the following:

(1) RAI-1(a) - all Alloy X-750 components, excluding the replacement tie rod upper support, in the primary vertical and horizontal load paths of the core shroud stabilizer assembly and,

(2) RAI-1(b) - design changes that are proposed to reduce the total stress in the Alloy X-750 components, excluding the replacement tie rod upper support, to reduce their sustained, peak stresses to a value below the IGSCC susceptibility criteria.

In response to RAI-1(a), the licensee, in a letter dated December 19, 2008, provided a list and drawings of all Alloy X-750 components that are exposed to vertical and horizontal load paths of the core shroud assembly. With respect to RAI-1(b), the licensee stated that there are no Alloy X-750 components, excluding the replacement tie rod upper support, that have peak sustained stresses above the IGSCC susceptibility criteria of $0.8S_y$ and the stresses in the replacement components have been reduced by increasing the radii in the corners and in the thread roots. NRC staff reviewed the responses and finds them acceptable because the new design stress levels in Alloy X-750 components are below the IGSCC threshold level of $0.8S_y$ in these components. Therefore, NRC staff's concern related to RAI-1(a) and RAI-1(b) are adequately resolved.

In an RAI dated December 8, 2008, NRC staff requested that the licensee provide information regarding the water chemistry at Hatch-2, including the use of hydrogen and noble metal chemical addition (NMCA). In response to NRC staff's RAI-2, in a letter dated December 19, 2008, the licensee stated that it uses hydrogen water chemistry with NMCA. Since the effectiveness of the hydrogen/NMCA is limited in the upper core regions, the licensee is not taking any credit for IGSCC mitigation due to hydrogen/NMCA. NRC staff concurs with this response and considers that its concern related to RAI-2 is resolved adequately.

For this modification, the licensee proposed to use XM-19 material which is more resistant to IGSCC than typical austenitic stainless steel materials (i.e., 304/316). With regards to the proposed modification, in an RAI dated December 8, 2008, NRC staff requested that the licensee provide the following:

(1) RAI-4 - the heat treatment and surface cold work limitations for machining that were instituted to prevent IGSCC. Describe the tests performed to demonstrate that the heat treatment and surface cold work limitations will prevent IGSCC and,

(2) RAI-5 - data and analyses that demonstrate that the maximum plastic strain limit for XM-19 material will assure that the XM-19 material is not susceptible to IGSCC.

In response to RAI-4, in a letter dated December 19, 2008, the licensee stated that to reduce IGSCC XM-19 it is always used in the solution annealed condition and the machining of this material is controlled per the requirements of Paragraph C9.6.7 in the BWRVIP-84 report.

In response to RAI-5, in a letter dated December 19, 2008, the licensee stated that based on the laboratory test data and as well as GE BWR experience, XM-19 materials demonstrated IGSCC resistance at stress levels above yield strength which in turn produced plastic strain levels well beyond the licensee's acceptance criterion. Therefore, the licensee justified its claim to limit the amount of cold work to a plastic strain acceptance criterion to prevent IGSCC in XM-19 materials. NRC staff reviewed the responses and concludes that by complying with the limits specified in the BWRVIP-84 report and by maintaining the cold work limits below the licensee's plastic strain limit criterion, occurrence of IGSCC in XM-19 materials can be minimized. Therefore, the staff's concerns related to RAI-4 and RAI-5 are resolved.

4.1.3 Pre-Modification Inspection

Since the Hatch-2 core shroud stabilizers were recently inspected in 2007, the licensee proposed to inspect the only hardware that will be installed during the proposed modification. In response to NRC staff's RAI-1(c), in a letter dated December 19, 2008, the licensee confirmed that the results of the inspection performed during Hatch-2's 2007 refueling outage are satisfactory. Since there are no indications in the existing core shroud assembly, NRC staff agrees with the licensee's proposal to limit the inspection to only the proposed core shroud modification that will be installed.

4.1.4 Post-Modification Inspection

NRC staff reviewed the licensee's proposed post-modification inspection of the core shroud assembly items (with the exception of the gusset plates and attachment welds) listed in Section 3.5 of this SE. NRC staff concludes that since the proposed inspections of these items are consistent with NRC staff-approved inspections performed at Hatch-1 for a similar core shroud tie rod modification they are, therefore, acceptable. Regarding the gusset plates and attachment welds, in response to the staff's RAI-3, the licensee in a letter dated December 19, 2008, stated that the shroud support plate is a thick, low alloy steel plate with stainless steel clad weld which is unique to Hatch-2 and this thick plate replaces thinner stainless steel support plate/gussets commonly used at other BWR units. Absent the gusset and support welds, no examinations are required for these components. NRC staff accepts this response and finds that its concern related to RAI-3 is satisfactorily resolved.

4.1.5 Inspections During Subsequent Refueling Outages

The licensee committed to perform inspections at the first refueling outage following the installation of the modified tie rod upper supports in accordance with the requirements defined in the staff-approved BWRVIP-76 report. In response to NRC staff's RAI-1(d), in a letter dated December 19, 2008, the licensee confirmed its commitment to re-inspect the core shroud tie rod assembly per the requirements specified in the BWRVIP-76 report. The licensee's implementation of the inspection guidelines of the BWRVIP-76 report provide adequate information regarding the structural integrity of the core shroud tie rod modifications.

In a letter dated December 8, 2008, in RAI-1(e), NRC staff requested that the licensee provide an explanation why VT-1 was chosen as an inspection method for detecting IGSCC at the upper support arm inner and outer corner radius locations. In a letter dated December 19, 2008, the licensee stated that VT-1 was used for the inspection of the Hatch-1 core shroud stabilizer upper supports as approved by NRC staff. To maintain consistency, the licensee will use VT-1 for the inspection of the Hatch-2 core shroud stabilizer upper supports. NRC staff finds this response acceptable and finds that its concern related to RAI-1(e) is resolved.

5.0 CONCLUSIONS

NRC staff found that the modified core shroud stabilizer assemblies for Hatch-2 are acceptable for the following reasons:

(A) The newly-designed upper supports have a large fillet radius at the corner of the support arm and are wider and thicker with no sharp edges. The newly-designed tie rod nuts have threads with a larger root radius and are made of XM-19 material which is more IGSCC resistant. These upper support and tie rod nut design changes reduce the peak stress and thereby increase resistance to IGSCC.

(B The licensee's modified core shroud stabilizer assemblies are structurally qualified to meet ASME Code, Section III, design criteria, and the design is consistent with the original design basis for the tie rod repair.

(C) Implementation of the inspection guidelines of the BWRVIP-76 report during subsequent outages will provide adequate information regarding the structural integrity of the core shroud tie rod modifications.

(D) The inclusion of the inspection of the upper support arm inner and outer corner radius locations using an ASME Code, Section XI, VT-1 examination technique at 10-year intervals in the ISI program will provide for detection of IGSCC, if it were to occur.

Based on the above discussion, NRC staff concludes that the modified core stabilizer assemblies and the proposed inspection of the upper support arm inner and outer corner radius locations using an ASME Code, Section XI, VT-1, examination technique at the first refueling outage following installation of the modified assemblies and at subsequent 10-year intervals will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative, is authorized for Hatch-2 for the remainder of the licensed operating period, which ends on June 13, 2038.

Principal Contributors: G. Cheruvenki
A. Tsirigotis

Date of Issuance: February 13, 2009