February 10, 2003

Mr. H. L. Sumner, Jr.
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Hatch Project
Southern Nuclear Operating
Company, Inc.
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SUBJECT: RELIEF REQUEST FOR THE THIRD 10-YEAR INSERVICE INSPECTION PROGRAM RE: EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. MB5009 AND MB5010)

By letter dated September 28, 2001, Southern Nuclear Operating Company, Inc. (SNC, the licensee) submitted Relief Requests RR-34 for inner nozzle radii examinations and Relief Request RR-35 for reactor pressure vessel stud examinations at the Edwin I. Hatch Nuclear Plant (Hatch), Units 1 and 2. The staff authorized Relief Request RR-35 in a Safety Evaluation (SE) dated July 2, 2002. By letter dated May 3, 2002, the licensee submitted Relief Request RR-34, Revision 1, which superceded the earlier version of Relief Request RR-34 in its entirety. This letter also included Relief Request RR-37 for review and approval by the U. S. Nuclear Regulatory Commission (NRC) staff. A Request for Additional Information was sent via E-mail to the licensee on November 8, 2002 (ADAMS Accession # ML023120447).

By letter dated November 21, 2002, the licensee submitted Relief Request RR-34, Revision 2 and Relief Request RR-37, Revision 1, superceding the earlier versions of these relief requests included in the May 3, 2002, letter. The licensee has proposed to use visual examinations with demonstrated resolution capabilities to the width of a 1-mil (0.001 inch) diameter wire or equivalent in lieu of the volumetric examinations required by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). Relief Request RR-34, Revision 2, proposes to perform a visual examination of “essentially 100 percent” of the Code-required inner nozzle radius surface area. Relief Request RR-37, Revision 1, proposed to perform a visual examination of “less than essentially 100 percent” of the Code-required inner nozzle radius surface area. The subject relief requests are for the third 10-year interval at Hatch, Units 1 and 2, which began January 1, 1996, and ends December 31, 2005.

The NRC staff has reviewed Relief Request RR-34, Revision 2, and Relief Request RR-37, Revision 1, and the associated proposed alternative testing methods against the requirements of the ASME Code, 1989 Edition, that was referenced in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a, 12 months prior to the start of the Hatch third 10-year interval. Our findings are provided in the enclosed SE.

Pursuant to 10 CFR 50.55a(a)(3)(i), Relief Request RR-34, Revision 2, is authorized based on the alternatives providing an acceptable level of quality and safety.

Pursuant to 10 CFR 50.55a(a)(3)(ii), Relief Request RR-37, Revision 1, is authorized because the specified ASME Code requirements would result in a hardship without a compensatory increase in the level of quality and safety and the proposed alternative provides a reasonable assurance of structural integrity of the subject components.

Sincerely,

/RA/

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

Enclosure: As stated

cc w/encl: See next page

Pursuant to 10 CFR 50.55a(a)(3)(ii), Relief Request RR-37, Revision 1, is authorized because the specified ASME Code requirements would result in a hardship without a compensatory increase in the level of quality and safety and the proposed alternative provides a reasonable assurance of structural integrity of the subject components.

Sincerely,

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John A. Nakoski, Chief, Section 1
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Docket Nos. 50-321 and 50-366

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** See previous concurrence
*No Major Changes to SE

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THIRD 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

The inservice inspection (ISI) of American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code) Class 1, Class 2, and Class 3 components are to be performed in accordance with Section XI of the ASME Code and applicable edition and addenda as required by Title 10 *Code of the Federal Regulations* (10 CFR) Section 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). In 10 CFR 50.55a(a)(3) it states, in part, that alternatives to the requirements of paragraph (g) may be used, when authorized by the U. S. Nuclear Regulatory Commission (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.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) will meet the requirements (except the design and access provisions and the preservice examination requirements) set forth in the ASME Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” to the extent practical within the limitations of design, geometry, and construction materials of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME Code incorporated by the reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to Commission approval.

By letter dated September 28, 2001, Southern Nuclear Operating Company, Inc. (SNC, the licensee) submitted Relief Requests RR-34 for inner nozzle radii examinations and Relief Request RR-35 for reactor pressure vessel (RPV) stud examinations at the Edwin I. Hatch Nuclear Plant (Hatch), Units 1 and 2. The staff authorized Relief Request RR-35 in a Safety Evaluation (SE) dated July 2, 2002. By letter dated May 3, 2002, the licensee submitted Relief Request RR-34, Revision 1, which superceded the earlier version of Relief Request RR-34 in its entirety. This letter also included Relief Request RR-37 for review and approval by the NRC staff.

A Request for Additional Information was sent via e-mail to the licensee on November 8, 2002 (ADAMS Accession # ML023120447).

By letter dated November 21, 2002, the licensee submitted Relief Request RR-34, Revision 2 and Relief Request RR-37, Revision 1, superceding the earlier versions of these relief requests included in the May 3, 2002, letter. The licensee has proposed to use visual examinations with demonstrated resolution capabilities to the width of a 1-mil (0.001 inch) diameter wire or equivalent in lieu of the Code-required volumetric examinations. Relief Request RR-34, Revision 2, proposes to perform a visual examination of “essentially 100 percent” of the Code-required inner nozzle radius surface area. Relief Request RR-37, Revision 1, proposed to perform a visual examination of “less than essentially 100 percent” of the Code-required inner nozzle radius surface area. The subject relief requests are for the third 10-year interval at Hatch, Units 1 and 2, which began January 1, 1996, and ends December 31, 2005.

2.0 REGULATORY EVALUATION

2.1 Relief Request RR-34, Revision 2

2.1.1 Component Function/Description

This request covers Unit 1 nozzles N1A, N1B, N3A, N3B, N3C, N3D, N6A, N6B, N7, and N9; and Unit 2 nozzles 2N1A, 2N1B, 2N3A, 2N3B, 2N3C, 2N3D, 2N6A, 2N6B, 2N7, and 2N9.

2.1.2 Code Requirements for Which Relief is Requested

The 1989 Edition of ASME Code, Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.100, requires a volumetric examination of the RPV inner nozzle radius section. Relief is requested from the requirements to perform the volumetric examination of the inner nozzle radii for the nozzles listed in Section 2.1.1 of this SE.

2.1.3 Licensee’s Proposed Alternative

The licensee states:

> [SNC] proposes the substitution of a visual, VT-1 [Section XI Code, Paragraph IWA-2210] type examination in lieu of the volumetric examination requirements. Direct visual examination of the RPV head spray (N6A(B)) and RPV head vent (N7) nozzles will be performed and the remaining nozzles inner radii regions will be examined using remote visual examination techniques. For both direct and remote visual examinations, the resolution sensitivity will be established using a 1-mil (0.001 inch) wire standard, or equivalent. The visual examination coverage will include virtually 100% [percent] of the surface M-N as shown in ASME XI Figures IWB-2500-7(a) through (d). No examination coverage limitations exist for the listed RPV nozzle inner radius regions [in Section 2.1.1 of this SE].
>
> If crack-like surface flaws are detected by visual examination, the flaws will be characterized in accordance with Table IWB-3512-1. When applying Table IWB-3512-1 criteria, the crack depth will be assumed to be equal to one-half the measured crack

> length. Once the flaw characteristics are established, the flaws will be evaluated in accordance with ASME Section XI Code [sic] [Paragraph] IWB-3140.

2.1.4 Licensee's Basis for Alternative

The licensee states:

> Early in the development of ASME Section XI, examination requirements were applied to all nozzles welded with full penetration welds. RPV nozzle inner radius examinations are the only non-welded areas requiring ultrasonic examination, and no service related cracking or degradation has ever been found in the nozzle inner radius region in any of the BWR [Boiling Water Reactor] fleet plant nozzles other than on Feedwater or operational CRD [control rod drive] return line nozzles. Examination of Feedwater nozzles will continue to be performed in accordance with augmented examination program commitments (i.e. NUREG-0619). For all nozzles other than Feedwater, there is no significant thermal cycling during operation, therefore, from a risk perspective, there is no need to perform volumetric examination on any other nozzles. Southern Nuclear Operating Company (SNC) believes that application of a visual examination alternative for the subject nozzle inner radius regions ensures an acceptable level of quality and safety.

2.1.5 Evaluation

In the mid 1970s, fatigue-initiated cracking was discovered in the nozzle inner radius section of feedwater nozzles at 18 BWRs. The cracks were found using visual examination. Ultrasonic testing (UT) failed to reveal the presence of these cracks. The shortcomings with UT prompted the NRC to issue NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," that modified inspection requirements for these components.

In NUREG-0619, the NRC staff concluded that UT of the vessel nozzle inner radius section involves complex geometries, long examination metal paths, and inherent UT beam spread, scatter, and attenuation. During the intervening years, improvements in UT technologies were introduced (e.g., computer modeling, tip diffraction, and phased array scanning) that improved the quality of the examination for this component. However, the area remains difficult to examine completely.

The NRC staff finds that even with vessel examinations using improved nondestructive examination technology from the outside surface, the complex geometry of the RPV head nozzle inner radius sections prevents complete UT coverage. For the RPV head nozzles, the licensee proposed to perform an enhanced direct VT-1 (EVT) visual examination with "essentially 100-percent coverage" in lieu of UT. The enhancement refers to using a procedure and personnel that have the capability of detecting a 1-mil wire standard or equivalent at 2 feet.

The demonstration provides assurance that an examiner would recognize a crack if one were to exist. The estimated coverage for each nozzle is provided in the licensee's submittal dated November 21, 2002. The licensee indicated that measures to assure examination conditions,

including adequate lighting, will be consistent with the conditions used for the demonstration of examiner competency.

The primary degradation mechanism in RPV nozzles is fatigue that produces hairline surface indications along the circumference of the nozzle at the inner radius section. The licensee will use high magnification cameras or direct visual examinations (with magnification if needed) that have demonstrated resolution capability of detecting a 1-mil wire or equivalent and will perform the examination over 100 percent of the nozzle inner radius surface area. Given the 1-mil resolution capability of the EVT, this examination method has a very high probability of detecting all detrimental flaws. The staff has determined that the high resolution image from the camera or direct visual examination as demonstrated will provide adequate assurance of structural integrity and may be used in lieu of UT for the inner nozzle radius region.

2.1.6 Conclusion

Based on the information provided in the licensee's submittal, the NRC staff has determined that the proposed alternative will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the staff authorizes Relief Request RR-34, Revision 2, for the remainder of the third 10-year ISI interval at Hatch, Units 1 and 2, which began January 1, 1996, and ends December 31, 2005.

2.2 Relief Request RR-37, Revision 1

2.2.1 Component Function/Description

This request covers Unit 1 nozzles N2A, N2B, N2C, N2D, N2E, N2F, N2G, N2H, N2J, N2K, N5A, N5B, N8A, and N8B; and Unit 2 nozzles 2N2A, 2N2B, 2N2C, 2N2D, 2N2E, 2N2F, 2N2G, 2N2H, 2N2J, 2N2K, 2N5A, 2N5B, 2N8A, and 2N8B.

2.2.2 Code Requirements for Which Relief is Requested

The 1989 Edition of ASME Code, Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.100, requires volumetric examination of the RPV inner nozzle radius section. Relief is requested from the requirements to perform the volumetric examination of the inner nozzle radii for the nozzles listed in Section 2.2.1 of this SE.

2.2.3 Licensee's Proposed Alternative

The licensee proposes the substitution of a modified visual, VT-1 type examination in lieu of the volumetric examination requirements. All nozzle inner radius regions will be examined using remote visual examination techniques. These remote visual examinations will be performed in accordance with the 1989 Edition of the ASME Code, Section XI, Paragraph IWA-2211(c) with the exception that the resolution sensitivity will be established using a 1-mil wire standard or equivalent.

If crack-like surface flaws are detected by visual examination, the flaws will be characterized in accordance with Table IWB-3512-1. When applying Table IWB-3512-1 criteria, the crack depth will be assumed to be equal to one-half the measured crack length.

Once the flaw characteristics are established, the flaws will be evaluated in accordance with ASME Section XI Code, Paragraph IWB-3140.

2.2.4 Licensee's Bases for Alternative (as submitted)

The licensee believes that application of a visual examination alternative for the subject nozzle inner radius regions ensures an acceptable level of quality and safety. The basis for the licensee's proposed alternative is stated below:

> The visual examination coverage will include all accessible areas of the surface M-N as shown in Figures IWB-2500-7(a) through (d). RPV internal component configurations (e.g., thermal sleeves, spargers, vessel internal attachments, instrumentation lines, etc.) prevent placement of the remote visual examination camera in positions necessary to examine surface M-N over the full circumference of the nozzle inner radius region. However, examinations will be performed on the accessible nozzle inner radius regions to the maximum extent practicable.
>
> All nozzle forgings were examined during the fabrication process (volumetric and surface techniques) and have subsequently been examined in accordance with inservice inspection program requirements. No indication of fabrication defects or service induced cracking has been detected by these examinations to date.
>
> Obtaining additional visual examination coverage would result in significant hardship due to the limitations of existing remote visual examination equipment and inability to remove or alter RPV internals components to allow additional coverage. Removal, or alteration, of the internal interference could result in damage to the components, requires specialized removal equipment, could require replacement with new components which are not readily available and are extremely expensive, and removal/re-installation requires significant expenditure of man-power. The tables in the letter dated November 21, 2002, provide estimates of the examination coverage in the circumferential direction for each affected nozzle inner radius region.
>
> The limited visual examination coverage does not significantly reduce the level of plant quality and safety for the following reasons:
>
> 1. There are no mechanisms of damage other than fatigue for the nozzle inner radius section, and for other than Feedwater nozzles, there is no significant thermal cycling. Therefore, the primary flaw of concern would be a flaw that was not detected during the manufacturing process. All of the nozzles were examined during and after fabrication by surface and volumetric examination techniques. Additionally, preservice and inservice ultrasonic examinations have detected no flaws. It is very unlikely that any flaws would be initiated by a fatigue mechanism.
>
> 2. After approximately 27 years of reactor operation for Unit 1, and 23 years for Unit 2, no cracking of any kind has been detected in the subject nozzle inner radius regions.

3. Approximately 42% of the total nozzle population will receive a complete (100%) examination of the inner radius region (see Relief Request RR-34).

4. Visual examination of the accessible nozzle inner radius surface (zone M-N) provides reasonable assurance that deep flaws are not present. Additionally, when flaws are initiated by fatigue mechanisms, they are typically encountered over a significant portion of the nozzle circumference as was the case for cracking of feedwater nozzles addressed in NUREG-0619.

2.2.5 Evaluation

As stated in Section 2.1.5 of this SE, the staff finds that, even with vessel examinations using improved UT from the outside surface, the complex geometry of the RPV nozzle inner radius regions prevents complete UT coverage. At the same time, performance of UT on these components requires the examiner to be in very close proximity to the RPV shell or head. Some of the nozzle configurations (e.g., thermal sleeves) also allow for entrapment of radioactive material that could result in high personnel dose rates due to “hot-spots.” The licensee estimates the personnel exposure time would be reduced approximately 1-3 hours per nozzle by use of visual examination. Visual examinations performed using a remote camera eliminate direct personnel exposure. Therefore, visual examination of the subject (non-head) inner radius regions are expected to have less dosage than UT examinations.

The primary degradation mechanism in the inner radius region is fatigue that produces hairline surface indications along the circumference of the nozzle inner radius. Given the resolution capabilities of the enhanced remote VT-1, there is a very high probability that the licensee would detect such flaws using high magnification cameras that can examine the accessible portions of the nozzle inner radius region surface areas. As stated in Section 2.1.5 of this SE, the staff has determined that the high resolution image from the camera may be used in lieu of UT (which is difficult to perform) of the inner nozzle radius region and provides adequate assurance of structural integrity. For the components listed in Section 2.2.1 of this SE, the licensee proposes to perform an enhanced VT-1 examination on the accessible portion of the nozzle inner radius region in lieu of UT. The licensee stated that the estimated coverage for some nozzles will be in the range of 40 to 50 percent. The resolution sensitivity for this remote in-vessel examination will be established using a 1-mil diameter wire or equivalent.

The staff finds that compliance with the ASME Code coverage requirements would result in a significant hardship due to the limitations of the remote visual examination equipment and inability to remove or alter the RPV internal components to allow additional coverage. While the proposed visual examination on these components is estimated to be between 40 to 50 percent of the Code-required surface area, the NRC staff believes that this coverage provides reasonable assurance that flaws of significant size will be detected. When flaws are initiated by the fatigue mechanism, they form crack networks that are encountered over a portion of the nozzle circumference, as shown in NUREG-0619. The NRC staff also recognizes that the industry has stated that it has experienced no reported cracking in the subject nozzle inner radius regions, and that the subject nozzles are not exposed to significant thermal cycling. In addition, the staff notes that at least 20 nozzles in the RPV nozzle population will receive a complete visual examination of the nozzle inner radius region (see Enclosure to the licensee’s letter dated November 21, 2002).

Based on the licensee's ability to demonstrate and perform a visual examination within 2 feet of the surface and the capability of resolving a 1-mil wire or equivalent width flaw, the NRC staff has determined that the licensee's proposed alternative to use an enhanced remote visual examination of the subject RPV Nozzle inner radius regions will provide reasonable assurance of structural integrity of the subject components.

2.2.6 Conclusion

Based on the information provided in the licensee's submittal, the NRC staff has concluded that the proposed alternative provides reasonable assurance of the structural integrity of the subject components and that compliance with the specified ASME Code requirements would result in hardship without a compensating increase in the level of quality and safety. Therefore, RR-37, Revision 1, is authorized at Hatch, Units 1 and 2, pursuant to 10 CFR 50.55a(a)(3)(ii) for the remainder of the third 10-year ISI interval which began January 1, 1996 and ends December 31, 2005.

3.0 CONCLUSION

Relief Request RR-34, Revision 2, is authorized pursuant to 10 CFR 50.55a(a)(3)(i) based on the alternative providing an acceptable level of quality and safety. Relief Request RR-37, Revision 1, is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) because the specified ASME Code requirements would result in a hardship without a compensatory increase in the level of quality and safety and the proposed alternative provides a reasonable assurance of structural integrity of the subject components. Both relief requests are authorized for Hatch, Units 1 and 2, for the remainder of the third 10-year ISI interval which began January 1, 1996, and ends December 31, 2005.

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Date: February 10, 2003

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