October 28, 2003

Mr. Stephen A. Byrne
Senior Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88
Jenkinsville, South Carolina 29065

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION — SECOND 10-YEAR INSERVICE INSPECTION PLAN REQUEST FOR RELIEF RR-II-08 (TAC NO. MB6647)

Dear Mr. Byrne:

By a letter dated October 30, 2002, as supplemented January 29 and April 21, 2003, South Carolina Electric & Gas Company (licensee) submitted Relief Requests (RRs) RR-II-08 through RR-II-12 from certain requirements specified in American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). By letter dated September 11, 2003, the licensee withdrew RR-II-09 through RR-II-12. The remaining RR-II-08 requested relief from ASME Code, Section XI, IWB-2500-1, Examination Category B-D, Item B3.140. In lieu of the Code requirements, the licensee proposed an alternative examination using enhanced remote visual equipment that is capable of a 1-mil (0.001-inch) wire resolution. The enhanced visual examination will be performed on, essentially, 100 percent of the inner nozzle radius. The RR is for the second 10-year interval at Virgil C. Summer Nuclear Station.

The U.S. Nuclear Regulatory Commission staff authorizes the proposed alternatives in RR-II-08 pursuant to Title 10 of the *Code of Federal Regulations*, Section 50.55a(a)(3)(ii), on the basis that compliance would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Sincerely,

/RA/

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

Docket No. 50-395

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SECOND 10-YEAR INTERVAL INSERVICE INSPECTION

REQUEST FOR RELIEF RR-II-08

VIRGIL C. SUMMER NUCLEAR STATION

SOUTH CAROLINA ELECTRIC AND GAS COMPANY

DOCKET NO. 50-395

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 is 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 U.S. Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 50.55a(g)(6)(i). According to 10 CFR 50.55a(a)(3) it is stated, in part, that alternatives to the requirements of paragraph (g) may be used, when authorized by the 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 materials of construction 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 ISI Code of record for Virgil C. Summer Nuclear Station (VCSNS) second 10-year ISI interval is the 1989 Edition. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to Commission approval.

By a letter dated October 30, 2002, as supplemented January 29 and April 21, 2003, South Carolina Electric & Gas Company (licensee) submitted Relief Requests (RRs) RR-II-08, RR-II-09, RR-II-10, RR-II-11 and RR-II-12 from certain requirements specified in the ASME

Code. By letter dated September 11, 2003, the licensee withdrew RR-II-09 through RR-II-12. The remaining item, RR-II-08, requested relief from ASME Code, Section XI, IWB-2500-1, Examination Category B-D, Item B3.140. In lieu of the Code requirements, the licensee proposed an alternative examination using enhanced remote visual equipment that is capable of a 1-mil (0.001-inch) wire resolution. The enhanced visual examination will be performed on essentially 100 percent of the inner nozzle radius. The NRC staff has reviewed and evaluated the licensee's request for relief pursuant to 10 CFR 50.55a(a)(3)(i) and 10 CFR 50.55a(a)(3)(ii).

2.0 DISCUSSION (RELIEF REQUEST RR-II-08)

2.1 Components Affected

Class:	1
Reference:	IWB-2500, Table IWB-2500-1
Examination Category:	B-D
Item Number:	B3.140
Description:	Steam Generator Primary Nozzle Inner Radius Section
Components:	6 Sections: 1-3100-12IR-A, 1-3100-12IR-B, 1-3100-12IR-C, 1-3100-13IR-A, 1-3100-13IR-B, 1-3100-13IR-C

2.2 Code Requirement

Table IWB-2500-1 for Examination Category B-D requires the primary nozzle inner radius section, Code Item Number B3.140, to be ultrasonically inspected each interval.

2.3 Licensee's Proposed Alternative

Perform an enhanced VT-1 examination of the nozzle inner radius section each interval. Examinations are to be performed only when the interior surfaces are made accessible for maintenance, repair, or concurrent with eddy current examination.

2.4 Licensee's Basis for Relief

VCSNS is presently required to schedule component inspections in accordance with the 1989 ASME Section XI. The steam generator primary nozzles are manufactured as non-welded integrally cast components. Ultrasonic preservice examination of this component required the use of low angle circumferential scan parameters to provide adequate coverage. The general design of the component exterior surfaces has ultrasonic scan interference and the primary side drain requires recording of geometric indications. During the operational phase of the component, considerable man-hours are required to provide access, insulation removal and restoration, surface preparation and ultrasonic examination in a high radiation area. The alternative examination will perform a primary inspection method, VT-1, to the interior inner radius surface normally accomplished during the performance of steam generator eddy current testing. The qualified visual examination of the inner radius surface along with considerable reduction in personnel radiation exposure will provide adequate flaw detection capabilities and enhanced As Low As Reasonably Achievable initiatives.

The ultrasonic examination of the steam generator nozzle inner radius is performed in a radiation area of approximately 200 millirem (mrem) per hour. Each examination requires the following items to complete the task:

1. Mirror insulation removal, approximately 2 man-hours. This task is typically done in respirators.

2. Cleaning and/or buffing of the surface, approximately 4 man-hours. This task is typically done in respirators.

3. Ultrasonic examination, approximately 2 man-hours.

4. Mirror insulation installation, approximately 2 man-hours.

5. Health Physics support, approximately ½ man-hour (using remote surveillance).

This evolution of 10½ man-hours should be expected to cause an exposure of 2100 mrem per nozzle. There are six nozzles to be inspected for a total of 12,600 mrem for completion of all ultrasonic examinations each interval.

The ultrasonic examinations performed for the detection of corner flaws per IWB-2500-7(d) consists of two circumferential scans, one clockwise and one counterclockwise. Both exams are performed with a specially designed 28 degree longitudinal wave alternative style transducer. This examination technique has scan interferences from manufacturing pads and the internally fabricated primary head drain hole. The combination of these interferences has limited the maximum achievable coverage to 80.4 percent of the examination volume.

Visual examination, VT-1, of the steam generator nozzle inner radius is performed by remotely utilizing either a robotic camera or utilization of the Eddy Current tooling end effectors for a robotic camera. Each visual examination requires approximately 1 man-hour to complete the task.

The least dose efficient method [for the VT-1] is expected to cause an exposure of 200 mrem per nozzle. There are six nozzles to be inspected for a total of 1200 mrem for completion of all visual examinations each interval. The preferred method is the use of the Eddy Current tooling to perform this visual inspection. With the use of this tooling the expected total dose associated with the Visual Inspection should be zero (0) mrem.

The proposed inspection activity will be implemented by the use of a dedicated, qualified procedure. The guidelines of a typical VT-1 activity will be detailed in the component specific procedure to include the aspects of personnel qualification, maximum distance requirements, lighting, detection of the “1-mil” wire gauge to verify resolution prior to each use, camera type and model, video recording, acceptance criteria and reporting. These specific procedural requirements will ensure the inspection system will be capable of accurate and repeatable defect detection along with substantial reduction in personnel radiation exposure.

3.0 <u>EVALUATION</u>

Title10 of the Code of Federal Regulations Section 50.55a allows licensees, when using the 1999 Addenda through the latest addenda of the ASME Code, to use enhanced magnification visual examination in lieu of ultrasonic examination. It states, in part, in 10 CFR 50.55a(b)(2)(xxi)(A), “The provisions of Table IWB-2500-1, Examination Category B-D, Full Penetration Welded Nozzles in Vessels, Items B3.140. . . . A visual examination with enhanced magnification that has a resolution sensitivity to detect a 1-mil width wire or crack . . . may be performed in place of an ultrasonic examination."

In the mid-1970s, fatigue-initiated cracking was discovered in the nozzle inner radius section of feedwater nozzles of 18 boiling water reactor (BWR) vessels. The cracks were found using visual examinations. 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," which 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) which 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 steam generator inner radius sections prevents complete UT coverage. In addition, the NRC staff recognizes the difficulties of performing accurate ultrasonic examination of cast nickel alloys. For the steam generator nozzle inner radii, the licensee proposed to perform an enhanced visual examination with essentially 100-percent coverage in lieu of UT. The enhancement refers to using personnel and a procedure which have a demonstrated capability of a 1-mil wire resolution.

Based on the resolution capability of the enhanced VT-1, it is highly unlikely that the licensee would fail to detect any detrimental flaws. The NRC staff has determined that the high resolution image from the camera, as demonstrated, will provide adequate assurance of structural integrity and may be used in lieu of UT for the inner nozzle radius region.

4.0 CONCLUSION

Based on the information provided in the licensee's submittal, the NRC staff has determined that the proposed alternative RR-II-08, as submitted on October 30, 2002, and supplemented January 29 and April 21, 2003, supports that performance of the 1989 Code required UT examination would result in hardship without a compensating increase in the level of quality and safety due to unnecessary dose exposure. Therefore, pursuant to 10 CFR50.55a(a)(3)(ii), the staff authorizes an enhanced VT-1 examination in lieu of the 1989 Code required volumetric examination for the remainder of the second 10-year ISI interval at VCSNS. The enhancement refers to using personnel and a procedure which have a demonstrated capability of a 1-mil wire resolution. This authorization is limited to those components described in Section 2.1 above. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

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Date: October 28, 2003

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