March 19, 2004

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

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION - SECOND 10-YEAR INSERVICE INSPECTION, REQUEST FOR RELIEF RR-II-15, RR-II-18, AND RR-II-19 (TAC NO. MC0108)

Dear Mr. Byrne:

By a letter dated July 11, 2003, South Carolina Electric & Gas Company (the licensee) submitted Relief Request (RR)-II-15, RR-II-16, RR-II-17, RR-II-18, RR-II-19, RR-II-20, RR-II-20 Addenda, and RR-II-21 for Virgil C. Summer Nuclear Station (VCSNS). On September 16, 2003, the licensee provided supplemental information for RR-II-15, RR-II-16, RR-II-17, and RR-II-19. On October 23, 2003, the licensee provided supplemental information for RR-II-15 and RR-II-18 and withdrew RR-II-17. On November 13, 2003, the licensee provided supplemental information for RR-II-18. The requests pertain to certain requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). Specifically, the licensee requested relief from ASME Code, Section XI, nozzle-to-vessel weld volume, nozzle inner radius examination, vessel ultrasonic qualification criteria and examination coverage requirements. The request for relief is for the second 10-year inservice inspection interval which ends December 31, 2003. The enclosed safety evaluation includes the technical review for RR-II-15, RR-II-18, and RR-II-19. RR-II-16, was issued February 11, 2004. RR-II-20, RR-II-20 Addenda, and RR-II-21 were issued February 3, 2004.

The Nuclear Regulatory Commission staff authorizes the proposed alternatives in RR-II-15, and RR-II-19 for the second 10-year inservice inspection interval of VCSNS, pursuant to Title 10, Code of Federal Regulations (10 CFR), Section 50.55a(a)(3)(i), on the basis that the alternatives provide an acceptable level of quality and safety. RR-II-18 is granted pursuant to 10 CFR 50.55a(g)(6)(i), on the basis that performance would result in a significant burden and the proposed alternative would provide reasonable assurance of integrity.

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-15, RR-II-18, AND RR-II-19

VIRGIL C. SUMMER NUCLEAR STATION

SOUTH CAROLINA ELECTRIC AND GAS COMPANY

DOCKET NUMBER 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 of the *Code of 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). As stated, in part, in 10 CFR 50.55a(a)(3), 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 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 Power Station (VCSNS) second 10-year ISI interval, which ends December 31, 2003, 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 July 11, 2003, South Carolina Electric & Gas Company (the licensee) submitted Relief Request (RR)-II-15, RR-II-16, RR-II-17, RR-II-18, RR-II-19, RR-II-20, RR-II-20 Addenda, and RR-II-21 for VCSNS. On September 16, 2003, the licensee provided supplemental information for RR-II-15, RR-II-16, RR-II-17, and RR-II-19. On October 23, 2003, the licensee provided supplemental information for RR-II-15 and RR-II-18 and withdrew RR-II-17. On November 13, 2003, the licensee provided supplemental information for RR-II-18. The requests pertain to certain requirements of the ASME Code. Specifically, the licensee requested relief from the 1989 Edition of Section XI of the ASME Code for selected requirements of nozzle-to-vessel weld volume, nozzle inner radius examinations, vessel ultrasonic qualification criteria and examination coverage. The request for relief is for the second 10-year ISI interval which ends December 31, 2003.

2.0 DISCUSSION FOR RR-II-15

2.1 Components for Which Relief is Requested

ASME Code Class 1, reactor vessel flange-to-upper shell weld.

2.2 Code Requirements

The 1989 Edition of ASME Code, Section XI, Appendix I, I-2100 requires "ultrasonic examinations of vessel welds greater than 2-inch thickness shall be conducted in accordance with Article 4 of Section V, as supplemented by this Appendix. Supplements identified in Table I-2000-1 shall apply." The specific requirements are in the Table submitted in the licensee's letter dated October 23, 2003.

Regulatory Guide 1.150, Revision 1, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations," augments the Code requirements.

2.3 Licensee Proposed Alternative

The licensee proposed using qualified personnel and procedures for remote mechanized examination of the reactor vessel flange-to-shell weld in accordance with the 1995 Edition with 1996 Addenda of the ASME Code, Section XI, Appendix VIII, Supplements 4 and 6, in lieu of Section V, Article 4 requirements.

2.4 Licensee Basis for Relief

Although Appendix VIII is not required for this weld, using an examination procedure and personnel qualified in accordance with Appendix VIII will provide an increased margin of safety and surpass the quality of the generic examination techniques specified by the referencing Code edition. Compliance with these requirements will assure the requisite level of quality is maintained.

The September 22, 1999, revision of 10 CFR 50.55a required implementation of ASME Section XI, Appendix VIII, Supplement 4 (clad-base metal interface) and Supplement 6 (vessel welds other than clad-base metal interface). The reactor vessel shell welds are subject to examination in accordance with these supplements, however, the flange-to-shell weld is the only reactor vessel shell weld not included in Appendix VIII.

For the VCSNS reactor vessel examination planned for 2003, the licensee will be employing procedures, equipment, and personnel qualified by performance demonstration in accordance with ASME Section XI, 1995 Edition, 1996 Addenda, as amended by 10 CFR 50.55a. The Code requirements as amended by the final rule will be complied with by using the WesDyne International procedure PDI-ISI-254, Revision 5.

The Appendix VIII procedure is technically superior to the standard ASME Code, Section V, Article 4 methodologies that are amplitude based. Enhanced performance is possible by (a) increased sensitivity to flaws, (b) demonstrated flaw measurement capability using amplitude independent sizing techniques, and (c) compatibility of the Appendix VIII examination technique with VCSNS flange-to-shell weld joint geometry resulting in good ultrasonic beam coverage. An additional benefit is reduction in radiation exposure to the exam team and VCSNS support personnel. This is possible because different examination devices will not have to be installed on the robot just to perform the flange to shell weld examination.

(a) Increase Sensitivity to Flaw: The Appendix VIII procedure is more sensitive to flaws because the exam sensitivity level compares to the ASME DAC [distance amplitude correction] level of 5-10 percent DAC. This is the highest practical sensitivity for ultrasonic testing. Previous examinations on the reactor vessel shell welds in accordance with ASME Section V were conducted at the less sensitive level of 50 percent DAC for flaws resident in the outer 80 percent of the material thickness and 20 percent DAC for flaws resident from the clad-base metal interface to a depth of about 20 percent thickness.

The Appendix VIII procedure offers an additional level of assurance in the detection of flaws because the procedure requires that all signals interpreted by the analyst as flaws regardless of response amplitude shall be measured and assessed in accordance with the applicable criteria. The Appendix VIII procedure recognizes that some flaws can exhibit low amplitude response depending on orientation. This evidence has not been factored into the ASME Section V techniques that have traditionally had a flaw response cut-off point of 20 percent DAC.

(b) Demonstrated Flaw Measurement Capability Using Amplitude Independent Sizing Techniques: Westinghouse Procedure PDI-ISI-254, “Remote Inservice Examination of Reactor Vessel Shell Welds,” in accordance with ASME Section XI, Appendix VIII, Supplements 4 and 6 was demonstrated in 2001 to the Electric Power Research Institute (EPRI) Performance Demonstration Initiative (PDI). The reference number for the performance demonstration test is PDQS No. 407.

The procedure complies with ASME Code, Section XI, 1995 edition, 1996 Addenda as modified by the final rule. The procedure was qualified using amplitude independent sizing techniques such as tip diffraction measurement and sizing by measurement of the flaw secondary response signals (a proven method for volumetric-type defects). The amplitude based flaw bounding criteria specified in ASME Section V procedures have been proven inaccurate as the size of the reflection is measured which may or may not accurately reflect true flaw size.

(c) Compatibility of the Appendix VIII technique to the VCSNS flange-to-shell weld joint and synergy with the previous examination. The Appendix VIII shell weld examination procedure requires the use of one beam angle, 45 degrees, applied to the weld and volume using 3 different transducer types each covering a specified depth range. The procedure requires the exam volume to be cross hatched with sound beams in four orthogonal directions. The increment size is 0.5 inch. Coverage is estimated in the attached sketch [see licensee's submittal for the sketch]. From the sketch, the critical inner 15 percent is well interrogated with the exception of area directly beneath the curved surface above the weld. This is a common limitation for the flange-to-shell weld joint.

The last remote mechanized exam of the flange-to-shell weld was conducted in 1993. At that time 45, 60, and 70 degree exam angles were used, and the results were acquired and analyzed using an automated ultrasonic exam system. The increment size was 0.5 inch, and the exam method was contact. Results from the exam were that no indications were found exceeding the allowable limits of Section XI. There is excellent data archival from the 1993 exam, and the licensee is confident that reasonable comparisons can be made with the Appendix VIII examination if any questions arise concerning indications.

The licensee will ensure that the flange-to-shell weld of the VCSNS reactor vessel will be examined with proven qualified ultrasonic examination techniques in lieu of standard amplitude based ultrasonic examination techniques currently specified. Examination will be conducted to the maximum extent practical in four orthogonal directions. When these results are combined with the manual examination performed from the flange seal surface, the coverage is 95 percent. The examination sensitivity and flaw measurement capability of the proposed alternative are superior to the method prescribed and coverage will be good considering the difficult geometric presentation.

2.5 <u>NRC Staff's Evaluation</u>

The 1989 Edition of Section XI IWA-2232 states, "Ultrasonic examination shall be conducted in accordance with Appendix I." Number I-2100 of Appendix I states, "Ultrasonic examination of vessel welds greater than 2-inch thickness shall be conducted in accordance with Article 4 of Section V, as supplemented by this Appendix [Appendix I of Section XI]. Supplements identified in Table I-2000-1 shall be applied." Section V, Article 4 as supplemented by Appendix I provides a prescriptive-based process for qualifying ultrasonic testing (UT) procedures. In lieu of Section XI requirements, the licensee proposed using procedures and personnel qualified in accordance with the performance-based criteria as implemented by the EPRI PDI program for the examination of reactor pressure vessels, Section XI, Appendix VIII, Supplements 4 and 6. The licensee contracted the services of Westinghouse to perform the examinations using Westinghouse procedure PDI-ISI-254.

When qualified prescriptive-based UT procedures are applied in a controlled setting containing real flaws in mockups of reactor vessels and the results are statistically analyzed according to the screening criteria in Appendix VIII of Section XI of the ASME Code, the procedures are equal to or less effective than UT Appendix VIII, Supplement 4 and 6 qualified procedures. A tabulation of the differences between the performance-based Westinghouse procedure

PDI-ISI-254, Revision 5 and Section V, Article 4 requirements is shown in Table 1 submitted in the licensee's letter dated October 23, 2003. Whereas, the performance-based UT uses fewer transducers than Section V, the performance-based UT is performed with higher sensitivity that increases the chances of detecting a flaw when compared to prescriptive-based Section V, Article 4 requirements. Also, flaw sizing is more accurately determined with the echo-dynamic motion and tip diffraction criteria used by performance-based UT as opposed to the less accurate amplitude criteria for prescriptive-based Section V, Article 4 requirements. Procedures, equipment, and personnel qualified through the EPRI PDI program have shown high probability of detection levels. This has resulted in an increased reliability of inspections for weld configurations within the scope of the PDI program.

2.6 Conclusion

Based on the licensee's proposed alternative to use UT procedures and personnel qualified to the 1995 Edition with 1996 Addenda of Section XI of the ASME Code, Appendix VIII, Supplements 4 and 6 as modified by 10 CFR 50.55a(b)(2)(xv) for the reactor pressure vessel shell-to-flange weld, the staff has determined that the proposed alternative examination with PDI qualified procedures and personnel of the shell-to-flange weld provides an equivalent or better examination than the current ASME Code requirements or the Regulatory Guide 1.150 recommendations and would provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative RR-II-15 is authorized for the subject flange-to-vessel weld at VCSNS for the second 10-year ISI interval which ends December 31, 2003.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this RR remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

3.0 DISCUSSION FOR RR-II-18

3.1 Components for Which Relief is Requested

ASME Code Class 1, Reactor Vessel Lower Head Circumferential Weld, CGE-1-1100B-5.

3.2 Code Requirements

1989 Edition of ASME Code, Section XI, Table IWB-2500-1, Item B1.21 specific volumetric examination for circumferential head welds 100 percent of the accessible length. The examination volume is defined in Figure IWB-2500-3.

3.3 Licensee Proposed Alternative

The licensee proposes to examine the lower head circumferential weld to the maximum extent practical using advanced robotics for contact examinations.

3.4 Licensee Basis for Relief

The request is to allow examination coverage to the maximum extent practical in consideration of component geometry.

The reactor vessel lower head circumferential weld was examined with automated ultrasonic techniques. UT scanning was conducted between obstructing penetrations with the scan boundaries maximized by visually assisted positioning of the examination head so that scanning starts and stops as close to the tubes penetrations as the configuration allowed. A small portion of the weld length cannot be examined due to obstructions from the periphery of the lower head penetrations. The examination coverage was 88 percent of the weld length.

3.5 NRC Staff's Evaluation

The 1989 Edition of ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item B1.21 specifies that the accessible length of the circumferential head weld be essentially 100 percent volumetrically examined. The examination volume is defined in Figure IWB-2500-3.

The staff reviewed the data submitted for the subject weld and determined that the tube penetrations would interfere with the movement of the UT transducers. In order to increase the examination coverage, the licensee would have to change the design of the tube penetrations, which is impractical. The licensee successfully examined 88 percent of the circumferential weld length. The coverage that was achieved would have detected any pattern of degradation if present.

3.6 Conclusion

The staff concludes that requiring the licensee to perform a design modification to obtain essentially 100 percent coverage is impractical and would result in a significant burden, and that the testing performed provides reasonable assurance of the structural integrity of the weld. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), relief RR-II-18 is granted to VCSNS for the second 10-year ISI interval which ends December 31, 2003. The relief is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this RR remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

4.0 DISCUSSION FOR RR-II-19

4.1 Component for Which Relief Is Requested

ASME Code Class 1 reactor vessel shell and head welds (Section XI, Examination Category B-A, Items B1.10 and B1.20).

4.2 <u>Code Requirements</u>

Relief is requested from the 1995 Edition with 1996 Addenda, ASME Section XI, Appendix VIII, Supplement 4, subparagraph 3.2(c) that states “performance demonstration results reported by the candidate, when plotted on a two-dimensional plot (Fig. VIII-S4-1) with the depth estimated by ultrasonic plotted along the ordinate and the true depth plotted along the abscissa, satisfy the following statistical parameters: (1) slope of the linear regression line is not less than 0.7; (2) the mean deviation of flaw depth is less than 0.25 inch; and (3) correlation coefficient is not less than 0.70.”

4.3 <u>Licensee Proposed Alternative</u>

The licensee’s proposed alternative is to use the root mean square (RMS) value of 10 CFR 50.55a(b)(2)(xv)(C)(1), that changes the required depth sizing criterion of Subparagraph 3.2(a) to Supplement 4 of Appendix VIII of Section XI of the ASME Code in lieu of Subparagraph 3.2(c).

4.4 <u>Licensee Basis for Relief</u>

ASME Code Section XI, Appendix VIII, Supplement 4, Subparagraph 3.2(c) imposes three statistical parameters for depth sizing. The first parameter, 3.2 (c)(1), pertains to the scope of a linear regression line. The linear regression line is the difference between actual versus true value plotted along a through-wall thickness. For Supplement 4 performance demonstrations, a linear regression line of the data is not applicable because the performance demonstrations are performed on test specimens with flaws located in the 15-percent through-wall thickness. The differences between actual versus true value produces a tight grouping of results, which resemble a shotgun pattern. The slope of a regression line from such data is extremely sensitive to small variations, thus making the parameter of 3.2(c)(1), an inappropriate criterion. The second parameter, 3.2(c)(2), pertains to the mean deviation of flaw depth. The value used in the ASME Code is too lax with respect to evaluating flaw depths within the inner 15-percent of the wall thickness. Therefore, the licensee proposes to use the more appropriate criterion of 0.15-inch RMS of 10 CFR 50.55a(b)(2)(xv)(C)(1), which modifies Subparagraph 3.2(a), as the acceptance criterion. The value of the correction coefficient in Subparagraph 3.2(c)(3) is inappropriate for this application since it is based on the linear regression from Subparagraph 3.2(c)(1).

The licensee believes the proposed alternative to use the RMS value of 10 CFR 50.55a(b)(2)(xv)(C)(1), which modifies the criterion of ASME Code, Appendix VIII, Supplement 4, Subparagraph 3.2(a), in lieu of Subparagraph 3.2(c), will provide an acceptable level of quality and safety.

4.5 <u>NRC Staff’s Evaluation</u>

As stated in the licensees bases for relief section above, Subparagraph 3.2(c) imposes three statistical parameters for depth sizing. The staff agrees with the licensee’s assessment of the inappropriateness of the statistical parameters of Subparagraph 3.2(c) for analyzing the test data from the Supplement 4 performance demonstration. Subparagraph 3.2(c)(1) pertains to the slope of a linear regression line. The linear regression line is the difference between actual versus true value plotted along a through-wall thickness. For Supplement 4 performance demonstrations, a linear regression line of the data is not applicable because the performance demonstrations are performed on test specimens with flaws located in the inner 15 percent through-wall. The differences between actual versus true value produce a tight grouping of results which resemble a shot gun pattern. The slope of a regression line from such data is extremely sensitive to small variations, thus, making the parameter of Subparagraph 3.2(c)(1) a poor and inappropriate, acceptance criterion. The second parameter, 3.2(c)(2), pertains to the mean deviation of flaw depth. The value used in the code is too lax with respect to evaluating flaw depths within the inner 15 percent of wall thickness. Therefore, the licensee proposed to use the more appropriate criterion of 0.15 inch RMS of 10 CFR 50.55a(b)(2)(xv)(C)(1), which modifies Subparagraph 3.2(a), as the acceptance criterion. The third parameter, 3.2(c)(3), pertains to a correlation coefficient. The value of the correlation coefficient in Subparagraph 3.2(c)(3) is inappropriate for this application since it is based on the linear regression from Subparagraph 3.2(c)(1).

Based on the above, the NRC staff believes that the use of Subparagraph 3.2(c) requirements in this context is inappropriate and that the proposed alternative to use the RMS value of 10 CFR 50.55a(b)(2)(xv)(C)(1), which modifies the criterion of Appendix VIII, Supplement 4, Subparagraph 3.2(a), in lieu of Subparagraph 3.2(c) will provide an acceptable level of quality and safety.

4.6 Conclusion

Based on the discussion above, the staff has concluded that the alternative proposed in RR-II-19 will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the staff authorizes the proposed alternative for VCSNS second 10-year ISI interval which ends December 31, 2003.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this relief request remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: Donald Naujock

Date: March 19, 2004

cc:

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