December 13, 2005

Mr. Jeffery Archie
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, UNIT 1 - ISSUANCE OF AMENDMENT REGARDING REACTOR COOLANT SYSTEM PRESSURE TEMPERATURE LIMITS (TAC NO. MC7375)

Dear Mr. Archie:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 174 to Renewed Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit 1. The amendment changes the Technical Specifications in response to your application dated June 22, 2005.

This amendment for Virgil C. Summer replaces the current reactor coolant system pressure-temperature limits for 32 effective full power years with the proposed limits for 56 effective full power years.

A copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's Biweekly *Federal Register* notice.

Sincerely,

/RA/

Robert E. Martin, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosures:
1. Amendment No. 174 to NPF-12
2. Safety Evaluation

cc w/enclosures: See next page

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SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

DOCKET NO. 50-395

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 174
Renewed License No. NPF-12

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by South Carolina Electric & Gas Company (the licensee), dated June 22, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (I) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-12 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 174, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment: Changes to the Technical Specifications

Date of Issuance: December 13, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 174

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-12

DOCKET NO. 50-395

Replace the following pages of the Appendix A Technical Specifications and associated Bases with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3/4 4-31	3/4 4-31
3/4 4-32	3/4 4-32
B 3/4 4-6	B 3/4 4-6
B 3/4 4-14	B 3/4 4-14

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 174 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1

DOCKET NO. 50-395

1.0 INTRODUCTION

By application dated June 22, 2005, South Carolina Electric & Gas Company (the licensee) requested changes to the Technical Specifications (TS) for the Virgil C. Summer Nuclear Station (VCSNS). The proposed changes would replace the current pressure and temperature (P/T) limits of 32 effective full power years (EFPYs) with the proposed P/T limits of 56 EFPY. The calculations of the revised P/T limit curves are delineated in Westinghouse Commercial Atomic Power report (WCAP), WCAP-16305-NP, "V. C. Summer Heatup and Cooldown Limit Curves for Normal Operation," dated August 2004. The proposed changes affect TS Section 3/4.4.9 and the associated Bases Section B 3/4.4.9. The proposed P/T limits were based on the use of the 1998 Edition through the 2000 Addenda of the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G and ASME Code Case –641, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements."

2.0 REGULATORY EVALUATION

The Nuclear Regulatory Commission (NRC) staff evaluates the acceptability of a facility's proposed P-T limits based on the following regulations and guidance:

Title 10 of Code of Federal Regulations (10 CFR) Part 50.60(a) states:

> Except as provided in paragraph (b) of this section, all light-water nuclear power reactors, other than reactor facilities for which the certifications under §50.82(a)(1) have been submitted, must meet the fracture toughness and material surveillance program requirements for the reactor coolant program pressure boundary set forth in appendices G and H to this part.

Appendix H to 10 CFR Part 50, "Reactor Vessel Material Surveillance Requirements," establishes requirements related to facility reactor pressure vessel (RPV) material surveillance programs. Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," requires that facility P/T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Code. The most recent version of Appendix G to Section XI of the ASME Code which has been endorsed in 10 CFR 50.55a, and therefore by reference in 10 CFR Part 50, Appendix G, is the 2001 Edition through the 2003 Addenda of the ASME Code. This edition of Appendix G to Section XI continues to incorporate the provisions of ASME Code Cases –588 and –640 (which later became part of ASME Code Case – 641). Additionally, Appendix G to 10 CFR Part 50 imposes minimum head flange temperatures when system pressure is at or above 20% of the preservice hydrostatic test pressure.

Regulatory Guide (RG) 1.99, Revision (Rev.) 2, “Radiation Embrittlement of Reactor Vessel Materials,” contains methodologies for determining the increase in transition temperature resulting from neutron radiation. Generic Letter (GL) 92-01, Revision 1, requested that licensees submit the RPV data for their plants to the NRC staff for review, and GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. NUREG-0800, “Standard Review Plan,” (SRP) Section 5.3.2, “Pressure Temperature Limits,” provides guidance on using these regulations and documents in the NRC staff’s review. Additionally, Section 5.3.2 provides guidance to the NRC staff in performing check calculations of the licensee’s submittal.

Pressure Vessel Fluence and Pressurized Thermal Shock

The current VCSNS P/T limit curves are valid for 32 effective full power years (EFPYs) of operation. This request is based in part, on the results of surveillance capsule Z removed from the VCSNS reactor vessel at the end of cycle 14 after 16.36 EFPYs of irradiation. The capsule analysis report (WCAP-16298NP, Ref. 2) was submitted to the NRC staff on October 22, 2004.

The NRC staff has issued Regulatory Guide (RG) 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” which describes the attributes of the calculational methods the NRC staff finds acceptable for the calculation of the pressure vessel fluence. In addition, RG 1.190 outlines methods for benchmarking and uncertainty analysis. RG 1.190 is based on the provision of general design criteria 14, 30, and 31, regarding fracture prevention of the reactor coolant pressure boundary.

Pressurized thermal shock (PTS) is reviewed relative to the requirements of 10 CFR 50.61, supplemented by RG 1.99, Revision 2 and RG 1.190.

3.0 TECHNICAL EVALUATION

3.1 Background

The basic parameter of the methodology of Appendix G to Section XI of the ASME Code (the Appendix G methodology) addressed in SRP 5.3.2 is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration. The Appendix G methodology requires a factor of 2.0 on stress intensity factors resulting from reactor pressure during normal and upset operating conditions and a factor of 1.5 on the stress intensity factor during hydrostatic testing. The Appendix G methodology also requires a factor of 1.0 on stress intensity factors resulting from thermal loads for normal and upset operating conditions as well as for hydrostatic testing. This methodology postulates the existence of a sharp elliptical surface flaw in the RPV. The flaw is assumed to be axially orientated for plates and axial welds and circumferentially orientated for circumferential welds. This flaw is postulated to have a depth that is equal to one quarter of the RPV beltline wall thickness (1/4T) and a length equal to six times its depth. The critical locations in the RPV beltline region for calculating heatup and cooldown P/T limit curves are the 1/4T and 3/4T locations, which correspond to the depth of the postulated inside surface and outside surface defects, respectively.

The Appendix G methodology requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}) at the 1/4T and 3/4T locations. The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the shift in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin term. Guidance on the determination of ΔRT_{NDT} and the margin term is given in RG 1.99, Rev. 2. ΔRT_{NDT} is a product of a chemistry factor (CF) and a fluence factor (FF). The CF is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The FF is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence, and the calculational procedures.

The licensee indicated in its submittal that Appendix G to Section XI of the 1998 Edition through 2000 Addenda of the ASME Code was used in generating the P/T limit curves. In addition, the element of Code Case - 641 which allows the use of the plane-strain fracture toughness (K_{Ic}) instead of the crack arrest fracture toughness (K_{Ia}) in the P/T limit calculations was used in the submittal. Consequently, the licensee's P/T limit methodology is equivalent to Appendix G to Section XI of the 2001 Edition through the 2003 Addenda of the ASME Code, as endorsed in 10 CFR 50.55a.

3.2 Licensee's Evaluation

The licensee's proposed ART values and P/T limit curves valid for up to 56 EFPYs of facility operation are based on the information in WCAP-16305-NP. WCAP-16305-NP considered additional information in the surveillance capsule report WCAP-16298-NP, (Reference 2), and concluded that the surveillance data for the intermediate shell plate that was fabricated from plate heat A9154-1 (Plate A9154-1) are not credible. Consequently, the licensee used Position 1.1 (based on tables) for evaluating the CF for Plate A9154-1 and identified it as the new limiting material for the proposed P/T limit curves. The licensee calculated the ART values for the limiting material for both the 1/4T and 3/4T locations. The key parameters for the licensee's ART determination for these locations from WCAP-16305-NP are reproduced in the table below.

WCAP-16305-NP documented detailed thermal and fracture mechanics evaluations to establish the proposed VCSNS P/T limits. The numerical representations of the proposed P/T limits can be found in Table 17 and Table 18 of the Westinghouse Commercial Atomic Power (WCAP) report, and additional information on thermal stress intensity factor, K_{It}, values at the 1/4T and 3/4T locations for the 100E F/hr cooldown curve and the 100E F/hr heatup curve can be found in Appendix A to the WCAP. These applied K_{It} values at the tip of the postulated axial flaw were derived using the thermal stresses caused by the temperature distribution across the RPV wall.

Based on these applied K_{It} and K_{Ic} values at the crack tips, the WCAP calculated the corresponding applied pressure stress intensity factors (K_{Ip}) at the tip of the postulated flaw at the 1/4T and 3/4T locations, and subsequently the pressure itself. The licensee stated that the proposed P/T limit methodology, as applied to RPV beltline materials, is in accordance with the Appendix G methodology.

3.3 NRC Staff Evaluation

The evaluation of the embrittlement of the RPV beltline materials relies on neutron fluence prediction acceptable to the NRC staff. The remaining issue of P/T limits is reviewed in the following table:

Summary of Key Information in WCAP-16305-NP for ART Calculations

Applicable Curves	Limiting Material	Location	Initial RT_{NDT} (EF)	Fluence (n/cm^2)	Chemistry Factor [1] (EF)	ΔRT_{NDT} (EF)	Margin [2] (EF)	ART (EF)
Cooldown	Intermediate Shell Plate A9154-1	1/4 T	30	4.27×10^{19}	65	89.05	34 ($\sigma_I = 0$, $\sigma_\Delta = 17$)	153
Heatup	Intermediate Shell Plate A9154-1	3/4 T	30	1.69×10^{19}	65	74.36	34 ($\sigma_I = 0$, $\sigma_\Delta = 17$)	138

(1) The chemistry factors were determined from the chemistry factor table using Regulatory Guide 1.99, Revision 2 Position 1.1 because the surveillance data is not credible

(2) The margin term for each ART calculation was based on the establishment of initial material property uncertainty (σ_I) and shift in material property uncertainty (σ_Δ) consistent with the guidance in Regulatory Guide 1.99, Revision 2.

To evaluate the proposed P/T limits for VCSNS, the NRC staff performed an independent calculation of the ART values for the limiting material of the VCSNS RPV, Plate A9154-1, using the methodology in RG 1.99, Rev. 2. The NRC staff effort includes a credibility evaluation of each surveillance data for Plate A9154-1 to verify the identification of this material as the new limiting material for the proposed P/T limits caused by the surveillance capsule information in WCAP-16298-NP. The NRC staff found that four of the ten surveillance data are not credible and, therefore, confirmed that the licensee's use of Position 1.1 (tables) for Plate A9154-1 is appropriate. The ART values for the limiting material at the 1/4T and 3/4T locations calculated by the NRC staff using RG 1.99, Rev. 2, and materials information for VCSNS in the NRC Reactor Vessel Integrity Database agree with the licensee's calculated values.

Finally, the NRC staff evaluated the licensee's P/T limit curves for acceptability by performing independent calculations using the Appendix G methodology based on information submitted by the licensee. The licensee stated that the proposed P/T limits were based on the element of Code Case - 641 which permits the use of the ASME Code K_{Ic} curve instead of the K_{Ia} curve for the RPV materials in the P/T limit calculations. As discussed in Section 2.0 of this safety evaluation, the NRC staff determined that the licencee's P/T limit methodology is in accordance with the 2001 Edition through the 2003 Addenda of the ASME Code, as endorsed in

10 CFR 50.55a. ASME Code, Section XI, Appendix G permits two approaches to calculate K_{It}: use of the bounding K_{It} formulas based on heatup and cooldown rates, and use of the K_{It} formulas based on the thermal stress distribution from a thermal model (e.g., a finite element model) for the heatup and cooldown. The WCAP used the latter approach and provided the RPV coolant temperatures, the metal temperatures at the 1/4T and the 3/4T locations, and their associated K_{It} values during the 100E F/hr cooldown and the 100E F/hr heatup. Based on the WCAP information, the NRC staff verified that the licensee's proposed P/T limit methodology is in accordance with the Appendix G methodology.

In addition to beltline materials, Appendix G to 10 CFR Part 50 also imposes a minimum temperature requirement for the closure head flange based on the most limiting reference temperature for the flange material. Section IV.A.2 of 10 CFR Part 50, Appendix G, states that when the pressure exceeds 20% of the pre-service system hydrostatic test pressure, the temperature of the closure flange regions which are highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120EF for core-not-critical, and at least 160EF for core-critical operation. The latter could be replaced by the inservice hydrostatic test temperature if it is more limiting. WCAP-16305-NP reported that the most limiting reference temperature for the flange material is 10EF. Based on this, the NRC staff determined that the vertical segment (10EF + 120EF) of the heatup and cooldown curves for core-not-critical and the vertical segment (the inservice hydrostatic test temperature of 210EF) of the heatup curves for core-critical operation satisfy the closure flange requirement of Appendix G to 10 CFR Part 50. Therefore, the licensee's proposed P/T limit curves satisfy all requirements of Appendix G to 10 CFR Part 50 and are acceptable for operation of the VCSNS RPV through 56 EFPY of operation.

<u>Technical Evaluation - Pressure Vessel Fluence and Pressurized Thermal Shock</u>

The analysis was performed using the discrete ordinates radiation transport (DORT) two-dimensional discrete ordinates code (Reference 4). The associated approximations were: P_5 for the expansion of the scattering cross section and S_{16} for the modeling of the angular quadrature, respectively, both of which exceed the RG 1.190 recommendation. With DORT, the licensee used the BUGLE- 96, which is a 47 energy group cross sections library that is based on the ENDF/B-VI data file (Reference 5). Bugle-96 is the recommended cross section library in RG 1.190.

Plant specific sources were derived from fuel cycle design reports and plant operating history. The capsule dosimetry for a set of six dosimeters is reported for measured and calculated values. The cycle 15 loading was projected to 56 EFPYs. Future projections for 25, 32, 36, 48, 54, and 56 EFPYs were calculated. Those projections are based on the current power level of 2900 MWt.

The six dosimeters used with capsule Z yielded measured values in excellent agreement with the corresponding calculated values. The average measured to calculated (M/C) ratio is 0.97 ± 5.2% (1σ) in the range of 0.90 to 1.03. The submittal (Ref. 2) included three additional levels of benchmarking, i.e., comparison of calculations with measured values in the pool critical assembly, comparison of calculations with the H. B. Robinson benchmark surveillance experiment, and an analytical sensitivity uncertainty analysis, regarding important input parameters used in the neutron transport solution. Accounting for the contributions from all the benchmarks, the licensee estimated an overall uncertainty of 13%. This value is well within the guidance in RG 1.190. The submittal recalculated corrections and updates to the four surveillance capsules (U, V, X, and W) that were removed previously. The five withdrawn and measured capsules yielded M/C ratios in the range of 0.96 to 0.99 and a standard deviation in the range 5.2% to 8.0%. These results are well within the guidance in RG 1.190 and can also be seen as plant specific benchmarking.

TS 3.4 Reactor Coolant System, Figures 3.4-2 and 3.4-3 are replaced with limit curves valid to 56 EFPYs. The new figures correctly reflect the new applicability limit.

The submittal also included WCAP-16306-NP (Reference 6), a pressurized thermal shock (PTS) analysis. The PTS rule 10 CFR 50.61, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events,” requires that the projected reference temperature of all materials in the belt-region at the end of life of the vessel RT_{PTS} meet the following screening criteria:

- be less than or equal to 270EF for plates, forgings and axial weld materials
- be less than or equal to 300EF for circumferential weld materials

The calculation of the RT_{PTS} is detailed in RG 1.99 and involves material properties and the peak vessel fluence for each material in the vessel belt-region.

WCAP-16306 lists the peak fluence values for the beltline materials derived in the same manner as the fluence values for the P/T limit curves. Those vessel fluence values likewise are acceptable. The licensee calculated RT_{PTS} values are well with in the 10 CFR 50.61 screening criteria.

4.0 SUMMARY

Based on the above evaluations, the NRC staff concludes that the proposed P/T limits for the pressure test, core-not-critical, and core-critical conditions satisfy the requirements in Appendix G to 10 CFR Part 50 and Appendix G to Section XI of the ASME Code. Therefore, the proposed P/T limits are approved for incorporation into the VCSNS TSs and shall be valid until 56 EFPYs of facility operation.

Summary - Pressure Vessel Fluence and Pressurized Thermal Shock

The NRC staff reviewed the VCSNS surveillance capsule Z analysis report regarding the estimation of projected vessel fluence values to 25, 32, 36, 48, 54, and 56 EFPYs. The analytical methodology and calculational assumptions and approximations meet NRC staff recommendations (RG 1.190) and the plant specific comparisons of measured and calculated results are in excellent agreement, thus, the NRC staff finds the projected fluence values to be acceptable. Therefore, for the reasons stated above the NRC staff finds the proposed TS changes for the P/T limit curves, and PTS to be acceptable

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of South Carolina official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (70 FR 56504). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. Letter from J. B. Archie South Carolina Electric and Gas Company to U.S. Nuclear Regulatory Commission “License Amendment Request Reactor Coolant System - Heatup/Cooldown Curves,” June 22, 2005.

2. WCAP-16298NP, “Analysis of Capsule Z from the South Carolina Electric and Gas Company Virgil C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program,” Westinghouse Electric Company LLC, October, 2004.

3. Regulatory Guide (RG) 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” U. S. Nuclear Regulatory Commission, March 2001.

4. DOORS 3.1, “One-, Two- and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System,” Reactor Shielding Information Center (RSIC) Computer Code Collection CCC-650, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

5. BUGLE-96, “Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications,” Reactor Shielding Information Center (RSIC) Computer Code Collection CCC-185, Oak Ridge National Laboratory, Oak Ridge, Tennessee, March 1996.

6. WCAP-16306-NP, Revision 0, “Evaluation of Pressurized Thermal Shock for V. C. Summer,” by C. M. Barton, Westinghouse Electric Company LLC, August 2004.

Principal Contributors: LLois
SSheng

Date: December 13, 2005

cc:
Mr. R. J. White
Nuclear Coordinator
S.C. Public Service Authority
c/o Virgil C. Summer Nuclear Station
Post Office Box 88, Mail Code 802
Jenkinsville, South Carolina 29065

Resident Inspector/Summer NPS
c/o U.S. Nuclear Regulatory Commission
576 Stairway Road
Jenkinsville, South Carolina 29065

Chairman, Fairfield County Council
Drawer 60
Winnsboro, South Carolina 29180

Mr. Henry Porter, Assistant Director
Division of Waste Management
Bureau of Land & Waste Management
Dept. of Health & Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

Mr. Thomas D. Gatlin, General Manager
Nuclear Plant Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88, Mail Code 300
Jenkinsville, South Carolina 29065

Mr. Robert G. Sweet, Manager
Nuclear Licensing
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88, Mail Code 830
Jenkinsville, South Carolina 29065

Ms. Kathryn M. Sutton
Morgan, Lewis & Bockius LLP
111 Pennsylvania Avenue, NW.
Washington, DC 20004