October 29, 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, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: ONE-TIME EXTENSION OF THE STEAM GENERATOR INSPECTION FREQUENCY (TAC NO. MB7312)

Dear Mr. Byrne:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 165 to Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1. The amendment changes the Technical Specifications in response to your application dated January 14, 2003, as supplemented by letters dated July 1, 2003, and August 20, 2003.

This amendment revises the steam generator inservice inspection frequency requirements in Technical Specification 4.4.5.3.a, from 40 months to 58 months after two consecutive inspections that were classified as C-1.

A copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's Biweekly *Federal Register* notice. This completes the staff's efforts on TAC No. MB7312.

Sincerely,

/RA/

Karen R. Cotton, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosures:
1. Amendment No. 165 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 FACILITY OPERATING LICENSE

Amendment No. 165
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 January 14, 2003, as supplemented by letters dated July 1, 2003, and August 20, 2003, 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 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. 165, 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/

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

Attachment: Changes to the Technical Specifications

Date of Issuance: October 29, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 165

TO FACILITY OPERATING LICENSE NO. NPF-12

DOCKET NO. 50-395

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3/4 4-13	3/4 4-13

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 165 TO FACILITY OPERATING LICENSE NO. NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-395

1.0 INTRODUCTION

By application dated January 14, 2003, as supplemented by letters dated July 1, and August 20, 2003, South Carolina Electric & Gas Company (the licensee) requested changes to the Technical Specifications (TSs) for the Virgil C. Summer Nuclear Station. The proposed changes would provide a one time extension of steam generator inspection frequency. Specifically, the proposed one-time change to TS 4.4.5.3.a revises the maximum inspection interval from 40 months to 58 months after two consecutive inspections that were classified as C-1. The license amendment request (LAR) also modifies Limiting Condition for Operation (LCO) 3.4.6.2.C, "Reactor Coolant System Operational Leakage," and Bases 3/4.4.5, "Steam Generators," to reduce the primary-to-secondary leakage limit for each SG from 500 gallons per day (gpd) to 150 gpd.

The supplemental letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the application.

2.0 REGULATORY EVALUATION

As stated in the August 20, 2003, submittal, the licensee complies with the regulatory requirements as stated below.

> 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," 1, 2, 4,14, 30, 31, and 32, define requirements for the reactor coolant pressure boundary (RCPB) with respect to structural and leakage integrity. SG [steam generator] tubing and tube repairs constitute a major fraction of the RCPB surface area. SG tubing and associated repair techniques and components, such as tube plugs, must be capable of maintaining reactor coolant inventory and pressure.
>
> General Design Criteria 19 of 10 CFR 50, Appendix A, defines requirements for the control room and for the radiation protection of the operators working within it. Accidents involving the leakage or burst of SG tubing comprise a challenge to the habitability of the control room. SG tubing and associated repair techniques and components, such as tube plugs, must be capable of maintaining reactor

coolant inventory and pressure in order to prevent excessive leakage and the resulting radiation doses to the control room operator.

10 CFR 50, Appendix B, establishes quality assurance requirements for the design, construction, and operation of safety related components. The pertinent requirements of this appendix apply to all activities affecting the safety related function of SGs that include inspection, testing, operating, and maintaining.

10 CFR 50.65, the Maintenance Rule, classifies SGs as risk significant components because they are relied upon to remain functional during and after design basis events. SGs are monitored, under a (2) of the Maintenance Rule against industry established performance criteria. If performance criteria are not met, a cause determination shall be done and the results evaluated to determine if goals should be established per a (1) of the Maintenance Rule.

10 CFR 100, Reactor Site Criteria, establishes the reactor-siting criteria with respect to the risk of public exposure to the release of radioactive fission products. Accidents involving the leakage or burst of SG tubing may comprise a challenge to containment and therefore involve increased risk of radioactive release. SG tubing and associated repair techniques and components, such as tube plugs, must be capable of maintaining reactor coolant inventory and pressure in order to prevent excessive leakage.

3.0 TECHNICAL EVALUATION

The U.S. Nuclear Regulatory Commission (NRC) Staff's evaluation covers: the replacement steam generator's improved design features; steam generator inspection scope and results from inservice inspections since replacement in Fall 1994; and related industry operating experience.

3.1 Steam Generator Design Improvements

The replacement steam generators incorporate both design and material enhancements to address service related degradation issues the industry experienced with the original steam generator design. Several examples of these improvements are discussed below.

- The replacement SG tubing is made of thermally-treated Alloy 690, which has an increased resistance to stress corrosion cracking (SCC) over the original mill annealed Alloy 600 steam generator tubing. The thermally-treated Alloy 690 material has a 13 percent higher chromium content and correspondingly reduced nickel content than the original mill annealed Alloy 600 steam generator tubing. The higher chromium content and thermal processing reduces the degree of sensitization of the material, thus increasing resistance to corrosion attack at the grain boundaries. Extensive laboratory tests have been performed by the industry that have demonstrated thermally treated Alloy 690 material is superior to mill annealed Alloy 600 material in its resistance to both primary and secondary system SCC, pitting and general corrosion.

- The replacement SG tube support plate (TSP) material is Type 405 stainless steel that shows improved corrosion resistance over the carbon steel TSPs used in the original

SGs. Corrosion resistant tube support plate material limits the potential for crevice corrosion product buildup, and subsequent denting and degradation of the SG tube.

- The trifoil shaped, broached, flat contact TSPs improve axial fluid flow within the tube bundle and minimize tube-to-tube support contact area. An enhanced anti-vibration bar (AVB) design and minimum gap U-bend construction provides for a more stable tube bundle, and limits potential for wear. Stress relief was performed on the U-bend tubes in the innermost 17 rows, which reduces the tube's susceptibility to stress corrosion cracking.

- Full depth hydraulic tube expansions minimize the depth of the crevice between the tubes and the tubesheet. The full depth expansion minimizes the accumulation of contaminants in the tubesheet crevice and the hydraulic expansion process minimizes the residual stresses in the steam generator tubes. Both these improvements reduce the susceptibility of the steam generator tube within the tubesheet to corrosion.

- Foreign material entry into the steam generators is minimized by the design of the feedwater distribution headers. The potential for loose parts is minimized by the feedwater ring spray nozzle assemblies, each consisting of a series of 0.25 inch diameter outlet holes, which function to trap potential foreign objects that may be introduced from the feedwater systems.

The NRC Staff finds the replacement steam generator's design and material improvements should enhance the steam generator tubing resistance to service induced degradation of the type experienced with the original steam generators, especially during the early service life.

3.2 Steam Generator Inspection Scope

Since steam generator replacement during Refueling (RF)-8, the licensee has conducted three partial (RF-9, RF-10, RF-11) and one 100 percent (RF-12) steam generator tube inspections. The licensee stated that during the RF-12 Fall 2000 refueling outage, 100 percent of the tubing in all three steam generators was inspected full-length (i.e., hot leg tube end to cold leg tube end) with an eddy current probe containing a bobbin coil. In addition, an approximately 5 percent sample of the hot leg top-of-tubesheet transitions in Steam Generator B and approximately 20 percent of the U-bend region of row 1 tubes in Steam Generator C were inspected with a rotating pancake coil (RPC) probe containing a +Point™ coil.

The NRC Staff concluded that the eddy current inspection scope (i.e., bobbin and RPC inspection) during the fall 2000 outage was comprehensive and supports the requested inspection interval extension.

3.3 Steam Generator Inspection Results

The licensee stated that there were no indications of stress corrosion cracking and no other degradation mechanisms have been detected since steam generator replacement. Three wear-like indications were detected during the RF-12 outage but historical review indicated each of these was present prior to service. Other eddy current indications observed during the RF-12 outage were determined by the licensee to be freespan dings, dents, manufacturing burnish marks and freespan signals unchanged from the preservice inspection. The licensee provided a discussion indicating that there was no active degradation in the steam generators through RF-12. The NRC Staff did not identify any concerns.

The licensee also performed inspections looking for foreign objects and tube degradation due to foreign objects. Inspections performed included: 1) eddy current inspections of all tubes; 2) foreign object search and retrieval of regions most likely to experience high levels of wear should an object be present in the steam generator; and 3) inspections of the material removed via sludge lancing from the steam generators.

The eddy current inspection program included a 100 percent bobbin coil inspection (which is capable of detecting volumetric indications). A loose part that was identified during RF-11 (a ½ inch long piece of wire) was removed during RF-12. There was no tube damage caused by the wire. This was the only loose part identified in the steam generator tube bundle since steam generator replacement in 1994. Various pieces were removed from the 3 feed rings in RF-12 and a visual inspection was performed to verify cleanliness within each feed ring. All pieces collected from the feed ring were small but had one dimension greater than ¼ inch, which prevented them from passing into the steam generator tube bundle. The licensee discussed corrective actions taken to prevent future introduction of debris from the secondary systems.

The licensee also performed an operational assessment to evaluate the predicted condition of the steam generator tubing after the proposed extended inspection interval. In this assessment, the largest wear-like through-wall indication (discussed above) was conservatively assumed to have not been present prior to service, although actual analysis indicated it was present at baseline and had not grown. The assumed maximum growth rate indicated a structurally significant flaw size would not be reached for approximately 18 cycles from RF-12. Therefore, the licensee concluded that the operational assessment supports the inspection interval extension.

The NRC Staff concluded that the inspection results and operational assessment results provide assurance that unexpected degradation of steam generator tubing has not occurred and is not expected to occur over the proposed inspection interval extension.

3.4 Related Industry Operating Experience

The licensee indicated that industry data for steam generators containing thermally treated Alloy 690 tubing has shown that no degradation mechanism, other than mechanical wear, has been identified. These plants include steam generators of the same vintage as V.C. Summer. Although Alloy 690 tubing has only been used in more recent steam generator replacements, laboratory tests indicate Alloy 690 should perform at least as well as, if not better than, thermally treated Alloy 600. Alloy 600 thermally treated steam generator tubing has demonstrated good resistance to in-service degradation and has been in service for much longer than Alloy 690. The NRC Staff also notes that Alloy 690 also has an excellent service history at plants outside the United States, including plants with greater operating time than V.C. Summer. Laboratory testing in environments much more aggressive than actual field service conditions indicate that Alloy 690 should continue to perform well during the extended period of operation. The NRC Staff agrees with the licensee's assessment that Alloy 690 has demonstrated excellent resistance to corrosion in service and in laboratory testing.

Wear is the only potential degradation mode currently identified for V.C. Summer. The licensee stated that there has been no instances of wear at AVBs, tube support plates or other locations in the V.C. Summer replacement steam generators. Based on this information, the licensee concluded there is reasonable assurance that wear indications will not become structurally significant over the proposed inspection interval extension. The NRC Staff agrees with this assessment.

The NRC Staff concluded that the industry operating experience with replacement steam generators and laboratory corrosion testing of thermally treated Alloy 690 tubing supports the licensee's proposed inspection interval extension.

3.5 Primary-To-Secondary Leakage Limit Revision

The licensee proposed changes to LCO 3.4.6.2.c, "Reactor Coolant System, Operational Leakage," and Bases 3/4.4.5, "Steam Generators," would reduce the primary-to-secondary leakage limit for each steam generator from 500 gpd to 150 gpd. This change would make TS operational leakage limit for the steam generators consistent with the V.C. Summer Steam Generator Management Program. The NRC Staff recognizes the reduced leakage limit conservatively changes the requirements for shutdown due to steam generator tube leakage and therefore agrees with the proposed change.

4.0 <u>STATE CONSULTATION</u>

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.

5.0 <u>ENVIRONMENTAL CONSIDERATION</u>

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 and/or changes surveillance requirements. 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 (68 FR 10280). 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.

6.0 <u>CONCLUSION</u>

The replacement steam generator's improved design features, the scope and results of the first inservice steam generator inspection, and related industry operating experience were evaluated as part of the review of the proposed LAR.

The NRC Staff concluded that the replacement steam generators incorporate both design and material improvements which are expected to improve the steam generator's tubing resistance to all forms of service induced degradation. In addition, the comprehensive fall 2000 inspection scope, the results of the inspection, and the conclusions of the operational assessment indicate the tubing is not experiencing any service induced degradation and can be expected to maintain tube integrity during the proposed extension. Industry operating experience with both the thermally treated Alloy 690 tubing and the improved Westinghouse design provides added assurance that the steam generators will maintain tube integrity over the proposed period of operation without an inspection of the steam generator tubing. Finally, the licensee has reduced the LCO for operational leakage from each steam generator from 500 gpd to 150 gpd, which the NRC Staff agrees conservatively changes the requirements and is accordingly acceptable. Therefore, the proposed technical specifications changes are acceptable.

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.

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Date: October 29, 2003

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