April 12, 19

Mr. Gary J. Taylor
Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88
Jenkinsville, South Carolina 29065

Distribution

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Summer Rdg.	S. Varga
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SUBJECT: ISSUANCE OF AMENDMENT NO. 133 TO FACILITY OPERATING LICENSE NO. NPF-12 REGARDING POWER UPRATE - VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1 (TAC NO. M93404)

Dear Mr. Taylor:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 133 to Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1. The amendment changes the Facility Operating License and Technical Specifications (TS) in response to your application dated August 18, 1995, as supplemented on November 1, 1995, February 14, March 14 (there are two supplemental letters dated March 14), and March 25, 1996.

The amendment increases the authorized core power level from 2775 Megawatts thermal (MWt) to 2900 MWt. The amendment also approves changes to the TS to implement uprated power operation.

A copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's Bi-weekly <u>Federal</u> <u>Register</u> notice. This completes the staff's efforts on TAC No. M93404.

Sincerely,

Original signed by Stephen Dembek for
Jacob I. Zimmerman, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosures: 1. Amendment No.133 to NPF-12
2. Safety Evaluation

cc w/enclosures: See next page

DOCUMENT NAME: G:\SUM93404.AMD

*SEE PREVIOUS CONCURRENCE:

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OFFICE	OGC*	DRPE:D	ADP	NRR		
NAME	RBachmann	SVarga	RZimmerman	WRussell		
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PDR ADOCK 05000395
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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

J. B. Knotts, Jr., Esquire
Winston & Strawn Law Firm
1400 L Street, N.W.
Washington, D.C. 20005-3502

Resident Inspector/Summer NPS
c/o U.S. Nuclear Regulatory Commission
Route 1, Box 64
Jenkinsville, South Carolina 29065

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta St., NW., Ste. 2900
Atlanta, Georgia 30323

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

Mr. Virgil R. Autry
Director of Radioactive Waste Management
Bureau of Solid & Hazardous Waste Management
Department of Health & Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

Mr. Robert M. Fowlkes, Acting Manager
Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station, Mail Code 303
Post Office Box 88
Jenkinsville, South Carolina 29065

Mr. George A. Lippard, Acting Manager
Nuclear Licensing & Operating Experience
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station, Mail Code 830
Post Office Box 88
Jenkinsville, South Carolina 29065

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 133
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 August 18, 1995, as supplemented on November 1, 1995, February 14, (there are two supplemental letters with this date), and March 25, 1996, 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, Facility Operating License No. NPF-12 condition 2.C(1) is hereby amended to read as follows:

   (1) Maximum Power Level

   SCE&G is authorized to operate the facility at reactor core power levels not in excess of 2900 megawatts thermal in accordance with the conditions specified herein and in Attachment 1 to this license. The preoperational tests, startup tests and other items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

3. Further, 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. 133 , 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.

4. This amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

W. Russell

William T. Russell, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Page 3 of Facility Operating License
   (included for convenience)
2. Changes to the Technical
   Specifications

Date of Issuance: April 12, 1996

(4) SCE&G, pursuant to the Act and 10 CFR Parts 30, 40 and 70 to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) SCE&G, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) SCE&G, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain, and is subject to, the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

SCE&G is authorized to operate the facility at reactor core power levels not in excess of 2900 megawatts thermal in accordance with the conditions specified herein and in Attachment 1 to this license. The preoperational tests, startup tests and other items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 133 , 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.

ATTACHMENT TO LICENSE AMENDMENT NO. 133

TO FACILITY OPERATING LICENSE NO. NPF-12

DOCKET NO. 50-395

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revisions are indicated by marginal lines.

Remove Pages	Insert Pages
3/4 1-5	3/4 1-5
3/4 4-31	3/4 4-31
3/4 4-32	3/4 4-32
3/4 11-5	3/4 11-5
6-16a	6-16a

DEFINITIONS

PURGE - PURGING

1.23 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2900 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.28 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SLAVE RELAY TEST

1.29 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

1.30 Not Used

SOURCE CHECK

1.31 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: LOWER SHELL
INITIAL RT_{NDT}: 10°F
ART AFTER 13 EFPY: 1/4T, 96°F
3/4T, 83°F

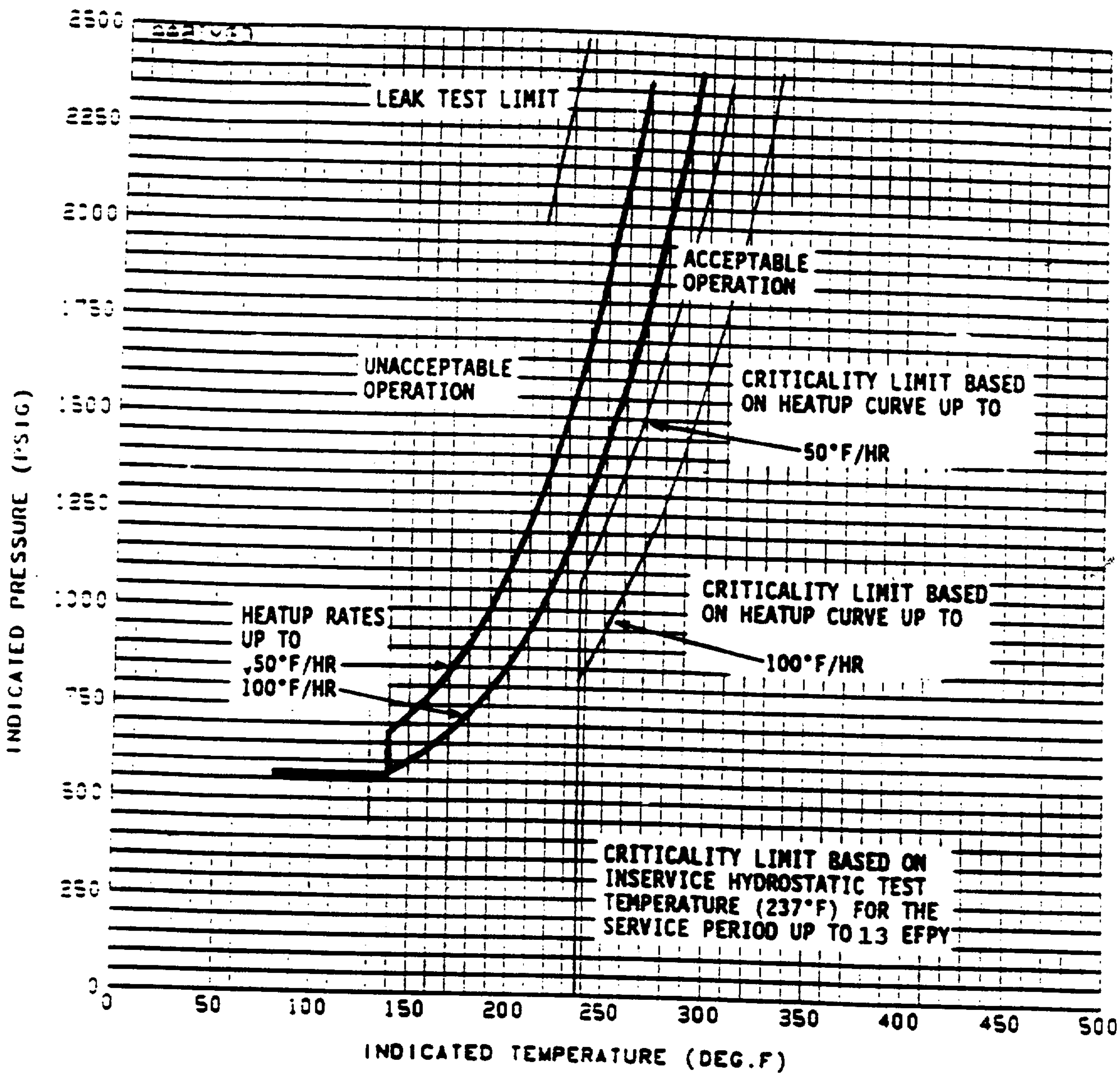

Figure 3.4-2 V. C. Summer Unit 1 Reactor Coolant System Heatup Limitations (Heat up rates up to 50 and 100°F/hr) Applicable for the First 13 EFPY (With Margins 10°F and 60 psig For Instrumentation Errors)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: LOWER SHELL
INITIAL RT_{NDT}: 10°F
ART AFTER 13 EFPY: 1/4T, 96°F
3/4T, 83°F

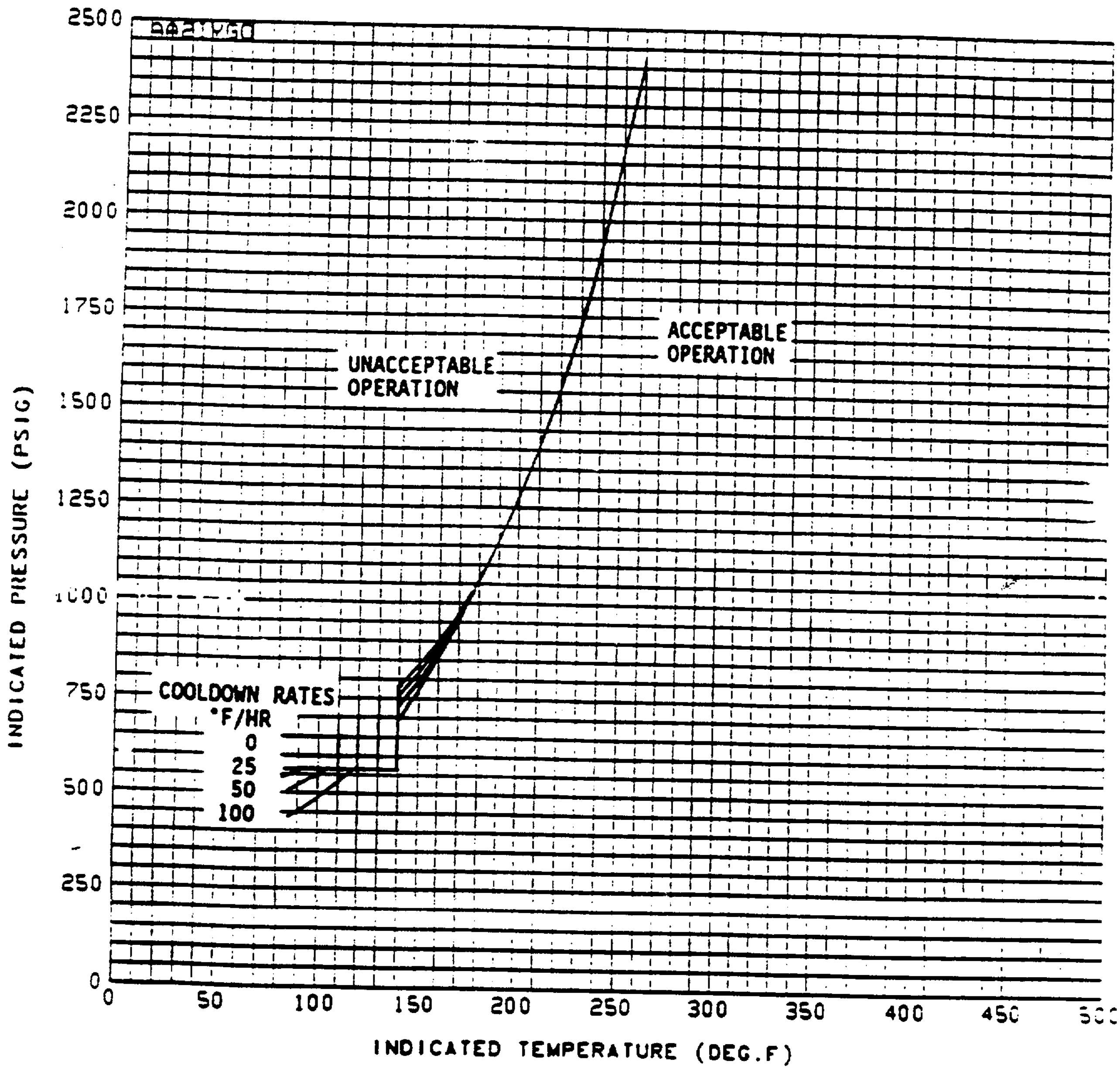

Figure 3.4-3 V. C. Summer Unit 1 Reactor Coolant System Cooldown (Cooldown rates up to 100°F/hr) Limitations Applicable for the First 13 EFPY (With Margins 10°F and 60 psig For Instrumentation Errors)

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 131,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least one per 24 hours when radioactive materials are being added to the tank.

CORE OPERATING LIMITS REPORT (Continued)

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limit, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

b. WCAP-10216-P-A, Rev. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION", February 1994 (W Proprietary).

(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (FQ Methodology for W(Z) surveillance requirements).)

c. WCAP-10266-P-A, Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987; Including Addendum 2-A, "BASH METHODOLOGY IMPROVEMENTS AND RELIABILITY ENHANCEMENTS," MAY 1988, (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements there to shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 133 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 letter dated August 18, 1995, as supplemented on November 1, 1995, and February 14, March 14 (there are two supplemental letters with this date), and March 25, 1996, (hereafter, collectively referred to as power uprate submittal) South Carolina Electric & Gas Company (the licensee) requested changes to the Facility Operating License (FOL) and Technical Specifications (TS) for the Virgil C. Summer Nuclear Station, Unit 1 (VCSNS). The proposed amendment would revise the FOL and TS to increase allowed core power level from 2775 Megawatts thermal (MWt) to 2900 MWt.

The original Federal Register notice included information in the licensee's November 1, 1995 supplemental letter. The February 14, March 14, and March 25, 1996 supplemental letters provided clarification and amplification of the analysis in the November 1, 1995 letter and were not outside the scope of the original Federal Register notice.

2.0 BACKGROUND

License Amendment No. 119, issued November 18, 1994, implemented changes to support VCSNS operation with replacement steam generators. The majority of the supporting analyses for the steam generator replacement were performed at the proposed core uprate power level of 2900 MWt. Also, several TS changes necessary for power uprate were approved in Amendment No. 119. This safety evaluation (SE) covers the power uprate issues that were not addressed in the staff's SE supporting Amendment No. 119. The FOL and TS changes requested by the licensee in their power uprate submittal are:

FOL Paragraph 2.C.1 - Revise maximum power level to 2900 MWt core power.

TS Definition 1.25 - Revise Rated Thermal Power definition to incorporate the increased power level.

TS Figures 3.4-2 and 3.4-3 - Revise applicability from 14 effective full power years (EFPY) to 13 EFPY due to increased neutron fluence effect.

ENCLOSURE

TS 3.11.2.6 - Revise maximum quantity of radioactivity in each gas storage tank from 160,000 curies to 131,000 curies of Noble gas in order to reference the current large break loss-of-coolant accident analysis.

TS 6.9.1.11.c - Revise methodology referenced by the core operating limits report that is used to determine the heat flux hot channel factor.

3.0 EVALUATION

3.1 Uprate Issues Evaluated for Amendment No. 119

The following table lists items previously evaluated in Amendment No. 119 and found acceptable at the uprated power level of 2900 MWt. These items will not be reevaluated for this amendment.

Evaluation	SE Section
Primary Components and Piping Support Considerations	2.2
Leak-Before-Break	2.2
Nuclear Steam Supply System Design Transients	2.3
Protection System Setpoints	2.3
Small Break Loss-of-Coolant Accident (LOCA)	2.3
Post LOCA Long Term Core Cooling Subcriticality	2.3
Hot Leg Switchover	2.3
Containment Considerations	2.4
Equipment Qualification Inside Containment	2.4
Radiological Consequences	2.5

3.2 Uprate Issues Not Previously Evaluated for Amendment No. 119

3.2.1 Large Break Loss-of-Coolant Accident (LBLOCA)

In its power uprate submittal, the licensee stated the licensing basis analyses have consistently shown the double-ended cold leg guillotine (DECLG) break with C_D=0.4 is the most limiting DECLG break. Previous analyses also showed that reduced vessel average temperature produces the most limiting results. Therefore, the licensee analyzed a DECLG break with a C_D=0.4 and a reduced vessel average temperature of 572°F using the Westinghouse 1981 Evaluation Model with BASH (WCAP-10266-P-A, Rev.2, 1987, Including Addendum 2-A, 1988). This analysis has been approved by the NRC for licensing applications and is applicable to VCSNS. The calculated peak cladding temperature is 2099°F, the calculated maximum local metal/water reaction is

7.9 percent, and the calculated core-wide metal/water reaction is less than 1 percent. These results are within the criteria specified in 10 CFR 50.46(b) (1 through 3, respectively) of 2200°F, 17 percent, and 1 percent. The results ensure the core will remain amenable to cooling, as required by 10 CFR 50.46(b)(4). In its submittal for Amendment No. 119, the licensee stated the time of emergency core coolant system (ECCS) hot leg switchover was determined by analysis to be within 8 hours. This, combined with the VCSNS ECCS design, assures continued conformance with the long-term cooling requirement of 10 CFR 50.46(b)(5). The licensee analyzed LBLOCA using bounding assumptions with an NRC approved methodology and concluded it met the acceptance criteria of 10 CFR 50.46. Therefore, the licensee's proposal is acceptable. The licensee proposed to revise TS 6.9.1.11.c to add "Including Addendum 2-A, 'BASH METHODOLOGY IMPROVEMENTS AND RELIABILITY ENHANCEMENTS,' MAY 1988." The staff agrees that the TS Administrative Controls section should include reference to the BASH addendum. Therefore, the licensee's proposed TS change is acceptable.

3.2.2 Residual Heat Removal (RHR) System

The licensee indicated the RHR system still has the ability to bring the plant from hot standby to cold shutdown (defined as less than or equal to 200°F) within the TS required time of 30 hours. Specifically, the licensee calculated the RHR system will require 21 hours to cool the reactor coolant system to 140°F with two RHR heat exchangers and two RHR pumps available. The staff agrees the RHR system can continue to perform its intended function in the uprated condition.

3.2.3 Increased Neutron Fluence

The licensee indicated the increase in core power will have an associated increase in the neutron fluence which interacts with the reactor vessel. To account for this increase in neutron fluence, the licensee has proposed changing the applicability of the TS heatup and cooldown curves (TS Figures 3.4-2 and 3.4-3, respectively) from 14 effective full power years (EFPY) to 13 EFPY. The staff agrees that a 1 EFPY reduction in applicability is appropriate for the proposed power uprate. Therefore, the licensee's requested TS changes are acceptable.

3.2.4 Waste Gas Decay Tank Rupture

In the Amendment No. 119 SE, the staff independently assessed the potential consequences of the release of the contents of a waste gas decay tank. The staff concluded the licensee's assumed release of 160,000 curies (Ci) of ^{133}Xenon (Xe) was nonconservative and should instead be approximately 131,000 Ci of ^{133}Xe. Specifically, the staff stated "While this particular issue is not associated with the replacement of the D3 steam generators, the licensee should reevaluate the determination of the allowable TS quantity of ^{133}Xe in the waste gas decay tank."

As requested by the staff, the licensee reevaluated this issue. The licensee determined that a change to the maximum quantity of radioactivity that can be stored in the Waste Gas Storage Decay Tank is required. In its power uprate submittal, the licensee requested to change TS 3.11.2.6 by replacing "160,000 curies" with "131,000 curies." The licensee's proposal is consistent with the staff's previous conclusion and is therefore acceptable. It is also noted that the licensee stated VCSNS has never exceeded an administrative limit of 90,000 curies of ^{133}Xe in a gas storage tank.

3.2.5 Spent Fuel Pool Cooling System

The spent fuel pool cooling system (SFPCS) was designed to remove the decay heat released from the stored spent fuel assemblies and maintain the spent fuel pool (SFP) water temperature at acceptable levels. The licensee evaluated three offload scenarios as part of its uprate analysis. The scenarios are 1) a partial offload with a single failure, 2) a routine refueling outage full core offload, and 3) an abnormal full core offload that occurs 36 days after a refueling outage in which 72 fuel assemblies were placed in the pool.

The following are the SFPCS heat loads resulting from the partial and full core offloads and their corresponding calculated peak SFP temperatures resulting from plant operations at the uprate power level:

	SFPCS Heat Loads (10^6 Btu/Hr)	Peak Pool Temperature
Partial Core Offloaded	21.23	150.2°F (assuming a single active failure)
Full Core Offloaded (offload occurs 36 days after 72 fuel assemblies were placed in SFP)	44.76	152.6°F (no single failure)
Full Core Offloaded	38.21	145.7°F (no single failure) 186.1°F (assuming a single active failure)

For the full core offload 36 days after 72 fuel assemblies have been placed in the SFP (i.e., abnormal offload), the calculated peak SFP temperature is 152.6°F which is below the guidance in Standard Review Plan (SRP) 9.1.3. Therefore, the staff finds the licensee's proposal acceptable.

With the partial core offload SFPCS heat load case (assuming a single active failure), the calculated peak SFP temperature is 150.2°F. Also, during routine refueling outages, the peak temperature could reach 186.1°F if a single active failure occurs. These temperatures are higher than the current VCSNS SFP Updated Final Safety Analysis Report design temperature and the SRP 9.1.3 guidance of 140°F for the SFP temperature limit during normal operating conditions. To address these higher temperatures, the licensee performed evaluations of the:

1. Structural integrity of the SFP and the SFP liner
2. SFPCS pipe stresses
3. SFPCS components
4. SFP ventilation system
5. Margin to localized boiling
6. Adequacy of net positive suction head available for the SFPCS pumps.

Based on its evaluation, the licensee concluded the above peak temperatures were acceptable. The licensee provided its bases for this conclusion in a March 14, 1996 supplemental letter. The licensee also stated that sufficient time exists to restore the SFPCS or provide make-up water to prevent the spent fuel from being uncovered if boiling were to occur. Based on the information provided by the licensee, the staff finds the licensee's proposal regarding normal SFP operations complies with General Design Criterion 61-*Fuel storage and handling and radioactivity control*, and is therefore acceptable.

Based on our review, the licensee's evaluations listed above, and the experience gained from our review of power uprate applications for similar pressurized water reactor (PWR) plants, we conclude that VCSNS operations at the proposed uprated power level is acceptable.

In a related matter, an issue associated with spent fuel pool cooling adequacy was identified in NRC Information Notice 93-83 and its Supplement 1, "Potential Loss of Spent Fuel Pool Cooling Following a Loss of Coolant Accident (LOCA)," dated October 7, 1993 and August 24, 1995, respectively, and in a 10 CFR Part 21 notification, dated November 27, 1992. The staff is evaluating this issue, as well as broader issues associated with spent fuel storage safety, as part of the NRC generic issue evaluation process. If the generic review concludes that additional requirements in the area of spent fuel pool safety are warranted, the staff will address those requirements to the license under separate cover.

3.2.6 Component Cooling Water System

The component cooling water system (CCWS) provides cooling water to various safety and non-safety systems during all phases of normal plant operation, including startup through cold shutdown and refueling, as well as following a station blackout event, loss-of-coolant accident (LOCA) or main steam line break accident. The CCWS is a closed loop system which serves as an intermediate barrier between the service water system and systems which contain radioactive or potentially radioactive fluids in order to eliminate the possibility of an uncontrolled release of radioactivity. The licensee stated that the CCWS heat loads resulting from plant operations at the proposed uprated power level will increase slightly. The increases in heat loads are from the SFPCS during both power and refueling operations, and residual heat removal (RHR) system during plant shutdown. The licensee performed evaluations of the effects of these increases in heat loads on CCWS and concluded that the CCWS has adequate capacity to accept the minimal increases from SFPCS and CCWS heat loads.

Based on our review and the experience gained from our review of power uprate applications for similar PWR plants, the staff finds that plant operations at the proposed uprated power level do not change the design aspects and operations of the CCWS. Therefore, the staff concludes that VCSNS operations at the proposed uprated power level is acceptable.

3.2.7 Service Water System

The service water system (SWS) is designed to supply cooling water to various non-safety related components and heat exchangers in the turbine, reactor, and radwaste buildings during normal plant operation, and to supply cooling water to safety related systems and other essential equipment during a station blackout event and a LOCA or main steam line break accident. Based on its performed evaluations, the licensee stated that the SWS as designed will supply sufficient water to remove the additional heat loads resulting from plant operations at the proposed uprated power level.

Based on our review and the experience gained from our review of power uprate applications for similar PWR plants, the staff finds that plant operations at the proposed uprated power level do not change the design aspects and operations of the SWS. Therefore, the staff concludes that VCSNS operations at the proposed uprated power level is acceptable.

3.2.8 Main Steam System

The licensee stated that the main steam system and its associated components (e.g. main steam isolation valves, turbine steam bypass system, etc.) were evaluated for a reactor power level of 2912 MWt and that VCSNS operations at the proposed uprated power level have an insignificant or no impact on the main steam system and its associated components.

Based on our review and the experience gained from our review of power uprate applications for similar PWR plants, the staff finds that plant operations at the proposed uprated power level do not change the design aspects and operations of the main steam system. Therefore, the staff concurs with the licensee that VCSNS operations at the proposed uprated power level is acceptable.

3.2.9 Condensate and Feedwater Systems

The licensee evaluated the condensate and feedwater systems for the plant operations at 2912 MWt reactor power level to support the above cited replacement steam generator TS change request and concluded that these systems satisfy their design bases for plant operations at the proposed uprated power level. Since these systems do not perform any safety related function, the staff has not reviewed the impact of plant operations at the proposed uprated power level on the design and performance of these systems.

3.2.10 Circulating Water/Main and Auxiliary Condensers/Turbine Auxiliary Systems

The circulating water, main and auxiliary condensers, and turbine auxiliary systems are designed to remove the heat rejected to the condenser by turbine exhaust and other exhausts over the full range of operating loads, thereby maintaining adequately low condenser pressure. The licensee stated that performance of these systems were evaluated for power uprate and determined that these systems are adequate for uprated power level operation. Also, in order to solve the problem of corrosion and fouling, the open cycle cooling system for cooling various turbine auxiliary systems will be converted to a closed system cooled with a modular forced draft cooling tower. This modification will solve the fouling problem, enhance performance, increase reliability and take a heat load off the circulating water system.

Since the circulating water, main and auxiliary condensers, and turbine auxiliary systems do not perform any safety function, the staff has not reviewed the impact of the uprated power level operation on the designs and performance of these systems.

3.2.11 Turbine-Generator

The licensee stated that evaluations for turbine operations with respect to design acceptance criteria to verify the mechanical integrity under the conditions imposed by the power uprate were performed. Results of the evaluations showed that there would be no increase in the probability of turbine overspeed nor associated turbine missile production due to plant operations at the proposed uprated power level. Therefore, the licensee concluded that the turbine could continue to be operated safely at the proposed uprated power levels.

Based on our review, the staff agrees with the licensee that operation of the turbine at the proposed uprated power level is acceptable.

3.2.12 Systems Not Affected By Power Uprate

The licensee stated that various systems were evaluated and determined that these systems were not affected by the power uprate. Those systems were evaluated for respective capacities, heat removal capabilities, and in many cases no direct connection to plant uprate was found. The following are major plant systems that were not affected by power uprate: auxiliary steam, condenser air removal, emergency diesel generators and auxiliaries, solid and liquid waste, fire service, station/instrument air, reactor building cooling, generator gas and vent, non-nuclear drains, plant waste, reactor building spray, and heating, ventilation and air-conditioning systems.

Since plant operations at the proposed uprated power level do not change the design aspects and operations of these systems and from the experience gained from our review of power uprate applications for other plants, the staff concludes that plant operations at the proposed uprated power level is acceptable.

3.2.13 Equipment Qualification Outside Containment

In a letter dated October 17, 1994, the licensee stated that impacts on environmental conditions (inside and outside containment) due to high energy line breaks were reconciled to ensure applicable equipment qualification requirements continue to be met. The licensee also outlined the process of ensuring environmental qualification of equipment after replacement steam generators. This 1994 letter was part of the licensee's submittal for steam generator replacement. This issue was evaluated in this SE because the staff did not review this aspect of the licensee's submittal for Amendment No. 119

Based on our review, the staff concludes that safety-related equipment outside the containment will be qualified to operate in an accident environment resulting from plant operations at the proposed uprated power level.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on April 12, 1996 (61 FR 16272). In this finding, the Commission determined that issuance of these amendments would not have a significant effect on the quality of the human environment.

6.0 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: D. Shum and S. Dembek

Date: April 12, 1996