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.

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