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

November 30, 1984

Docket No. 50-395

Mr. O. W. Dixon, Jr.
Vice President Nuclear Operations
South Carolina Electric & Gas Company
P.O. Box 764
Columbia, South Carolina 29218

Dear Mr. Dixon:

Subject: Issuance of Amendment No. 34 to Facility Operating License NPF-12 Virgil C. Summer Nuclear Station, Unit No. 1

The Nuclear Regulatory Commission has issued Amendment No. 34 to Facility Operating License NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1 located in Fairfield County, South Carolina. This amendment is in response to your letter dated May 23, 1984, and supplemented November 27, 1984.

The amendment modifies the Technical Specifications to allow installation of a P-9 interlock which would prevent a direct reactor trip following a turbine trip at or below 50% reactor power. The amendment is effective as of its date of issuance.

A copy of the related safety evaluation supporting Amendment No. 34 to Facility Operating License NPF-12 is enclosed.

Sincerely,

Elinor G. Adensam

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

Enclosures:
1. Amendment No. 34
2. Safety Evaluation

cc w/enclosure:
See next page

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SUMMER

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November 30, 1984

AMENDMENT NO. 34 TO FACILITY OPERATING LICENSE NO. NPF-12 - Virgil C. Summer Unit 1

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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. 34
License No. NPF-12

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

   A. The application for amendment to the Virgil C. Summer Nuclear Station, Unit No. 1 (the facility) Facility Operating License No. NPF-12 filed by the South Carolina Electric & Gas Company acting for itself and South Carolina Public Service Authority (the licensees), dated May 23, 1984, and supplemented November 27, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the 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 set forth in 10 CFR Chapter I;

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

   E. The issuance of this license 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 hereby amended by page changes to the Technical Specifications as indicated in the attachments 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

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 34 , are hereby incorporated into this license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

Enclosure:
Technical Specification Change

Date of Issuance: November 30, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 34

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 revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. Corresponding overleaf pages are also provided to maintain document completeness.

Amended Pages	Overleaf Pages
2-7	
B2-7	
B2-8	
3/4 3-5	3/4 3-6
3/4 3-13	3/4 3-14

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	S	Trip Setpoint	Allowable Value
18. Safety Injection Input from ESF	NA	NA	NA	NA	NA
19. Reactor Trip System Interlocks					
A. Intermediate Range Neutron Flux, P-6	NA	NA	NA	$\geq 1 \times 10^{-10}$ amps	$\geq 6 \times 10^{-11}$ amps
B. Low Power Reactor Trips Block, P-7					
a. P-10 input	7.5	4.56	0	≤10% of RTP	≤12.2% of RTP
b. P-13 input	7.5	4.56	0	≤10% turbine impulse pressure equivalent	≤12.2% of turbine impulse pressure equivalent
C. Power Range Neutron Flux P-8	7.5	4.56	0	≤38% of RTP	≤40.2% of RTP
D. Low Setpoint Power Range Neutron Flux, P-10	7.5	4.56	0	≥10% of RTP	≥7.8% of RTP
E. Turbine Impulse Chamber Pressure, P-13	7.5	4.56	0	≤10% turbine impulse pressure equivalent	≤12.2% turbine pressure equivalent
F. Power Range Neutron Flux, P-9	7.5	4.56	0	≤50% of RTP	≤52.2% of RTP
20. Reactor Trip Breakers	NA	NA	NA	NA	NA
21. Automatic Actuation Logic	NA	NA	NA	NA	NA

RTP = RATED THERMAL POWER

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level (Continued)

level setpoint, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide reactor core protection against DNB as a result of complete loss of forced coolant flow. The specified set points assure a reactor trip signal is generated before the low flow trip set point is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip set point is reached shall not exceed 0.6 seconds. On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10 percent of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10 percent of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine Trip initiates a reactor trip. On decreasing power, the reactor trip from the turbine trip is automatically blocked by P-9 (a power level less than or equal to 50% of RATED THERMAL POWER); and on increasing power, reinstated automatically by P-9.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. The ESF instrumentation channels which initiate a safety injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range reactor trip and de-energizing of the high voltage to the detectors. On decreasing power, Source Range level trips are automatically reactivated and high voltage restored.

P-7 On increasing power P-7 automatically enables reactor trips on low flow in more than one primary coolant loop, more than one reactor coolant pump breaker open, reactor coolant pump bus undervoltage and underfrequency, turbine trip, pressurizer low pressure and pressurizer high level. On decreasing power the above listed trips are automatically blocked.

P-8 On increasing power P-8 automatically enables reactor trips on low flow in one or more primary coolant loops, and one or more reactor coolant pump breakers open. On decreasing power the P-8 automatically blocks the above listed trips.

P-9 On increasing power P-9 automatically enables reactor trip on turbine trip. On decreasing power P-9 automatically blocks reactor trip on turbine trip.

P-10 On increasing power P-10 allows the manual block of the Intermediate Range reactor trip and the low setpoint Power Range reactor trip; and automatically blocks the Source Range reactor trip and de-energizes the Source Range high voltage power. On decreasing power the Intermediate Range reactor trip and the low setpoint Power Range reactor trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
19. Reactor Trip System Interlocks					
A. Intermediate Range Neutron Flux, P-6	2	1	2	2#	7
B. Low Power Reactor Trips Block, P-7 — P-10 Input	4	2	3	1	7
P-13 Input	2	1	2	1	7
C. Power Range Neutron Flux, P-8	4	2	3	1	7
D. Power Range Neutron Flux, P-10	4	2	3	1, 2	7
E. Turbine First Stage Pressure, P-13	2	1	2	1	7
F. Power Range Neutron Flux, P-9	4	2	3	1	7
20. Reactor Trip Breakers	2	1	2	1, 2	8
	2	1	2	3*, 4*, 5*	9
21. Automatic Trip Logic	2	1	2	1, 2	8
	2	1	2	3*, 4*, 5*	9

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

**The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped condition.

#The provisions of Specification 3.0.4 are not applicable.

##Below the P-6 (Intermediate Range Neutron Flux Interlock) setpoint.

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

****Values left blank pending NRC approval of 2 loop operation.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1.

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
D. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)	M (8)	N.A.	N.A.	1, 2
E. Turbine Impulse Chamber Pressure, P-13	N.A.	R	M (8)	N.A.	N.A.	1
F. Low Power Range Neutron Flux, P-9	N.A.	R(4)	M (8)	N.A.	N.A.	1
20. Reactor Trip Breaker	N.A.	N.A.	N.A.	M (7, 11)	N.A.	1, 2, 3*, 4*, 5*
21. Automatic Trip Logic	N.A.	N.A.	N.A.	N.A.	M (7)	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

- Below P-6 (Intermediate Range Neutron Flux Interlock) setpoint.

- Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) setpoint.

(1) - If not performed in previous 7 days.

(2) - Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2 percent. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) - Single point comparison of incore to excore axial flux difference above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3 percent. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - Detector plateau curves shall be obtained, evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux Channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) - Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) - Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) - With power greater than or equal to the interlock setpoint the required OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window.

(9) - Monthly Surveillance in MODES 3*, 4* and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.

(10) - Setpoint verification is not required.

(11) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include an independent verification of the undervoltage and shunt trips.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 34 TO FACILITY OPERATING LICENSE NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

I. INTRODUCTION

By letter dated May 23, 1984, South Carolina Electric and Gas Company requested a change to Technical Specifications to allow installation of a P-9 interlock which would prevent a direct reactor trip following a turbine trip (anticipatory reactor trip) at or below 50% reactor power. The purpose of this change is to prevent needless challenges to the reactor protection system and unnecessary transients during reactor startup. Additional information relating to this request was provided by letter dated November 27, 1984.

II. EVALUATION

The V. C. Summer steam dump system, including condenser dump valves, steam generator power relief valves (safety grade), and atmospheric dump valves (non-safety grade) has sufficient capacity to pass 85% of the main steam flow at full load temperature and pressure. A turbine trip can occur due to a generator trip, a malfunction of the turbine or its ancillary systems (e.g., low hydraulic or oil pressure, turbine overspeed, excessive vibration, low condenser vacuum), high-high steam generator level, and other secondary or primary plant malfunctions. Also, a loss of electrical load would probably result in a turbine overspeed trip. Normally, the steam dump system would accommodate the excess steam generation, preventing any significant increase in reactor coolant system (RCS) temperature and pressure. However, in the event of a failure of the steam dump valves to open following a loss of load or turbine trip, secondary pressure would rise, which may result in lifting the steam generator safety valves (SGSVs). The RCS pressure rise would be limited by pressurizer spray if the reactor coolant pumps (RCPs) are operating, or by the power operated relief valves (PORVs). Above 50% power, the turbine trip would also trip the reactor.

If the anticipatory reactor trip is bypassed at or below 50% power, the reactor may trip on high RCS pressure or overtemperature ΔT. A loss of offsite power (LOOP) may also occur during the bus transfer following turbine trip, resulting in a loss of RCS flow transient. The licensee considers this transient as bounding if it occurs at full power since it results in the minimum acceptable DNBR. The pressurizer safety valves and SGSVs are sized to protect the RCS and steam generators against overpressure, respectively, for all load losses without assuming operation of the steam dump system, pressurizer spray, PORVs, automatic rod control and anticipatory reactor trip.

In order to justify bypassing the anticipatory reactor trip at and below 50% power, the licensee submitted analyses for three cases involving turbine trip. The LOFTRAN, FACTRAN and THINC codes were used. Cases 1 and 2 assume turbine trip at 60% power, minimum reactivity feedback, manual reactor control, and no credit for steam dump, main and auxiliary feedwater. Case 1 assumes actuation of the PORVs and SGSVs to limit primary and secondary pressure increase. The reactor trips at 30 seconds due to bus undervoltage and LOOP occurs at this time. Case 2 is the more severe case. It assumes that the PORVs are unavailable and the LOOP occurs at 6.5 seconds to produce the most limiting transient with respect to departure from nucleate boiling (DNB). The reactor trips at 8.0 seconds due to high pressure. The pressurizer safety valves are actuated. Minimum departure from nucleate boiling ratio (DNBR) (2.5) and peak primary pressure (2523 psia) occur in 9.5 seconds. For both cases the RCS pressure is maintained at less than 110% of design pressure and the DNBR is maintained above the acceptable 95 percent probability at a 95 percent confidence level (95/95) DNBR limit of 1.3. The acceptance criteria for anticipated operational occurrences are thus met.

In order to demonstrate that deletion of the anticipatory reactor trip at or below 50% power does not significantly increase the probability of a small break loss of coolant accident (SBLOCA) resulting from a stuck open PORV, the licensee has performed a "better estimate transient" analysis. (Case 3) This analysis assumes turbine trip at 50% reactor power and operability of the network control. Beginning of core life reactivity feedback was assumed. The results of this case showed smooth control rod insertion to hot shutdown conditions without reactor trip, and a peak pressure of about 2300 psia, well below the PORV setpoint. The licensee concludes that installation of the P-9 interlock to prevent an anticipatory reactor trip at or below 50% power would not substantially affect the probability of a SBLOCA resulting from a stuck open PORV. We concur with this conclusion based on the licensee's analyses, which indicate that, except for unusual conditions, the PORVs would not be actuated for turbine trip at and below 50% power, (due to the high capacity of the V. C. Summer steam dump system), and challenges to the reactor protection system would be reduced.

The configuration of the bistable/relay driver board and transformer, which receives the input to the P-9 interlock circuitry from the power range detectors, is the same as that for the P-8 interlock circuitry currently installed at the V. C. Summer Nuclear Station. Therefore, isolation of the P-9 modification is in accordance with existing plant design. Therefore, from the above evaluation, the staff concludes that the proposed change is acceptable.

III. ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 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 this amendment.

IV. CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (49 FR 33370) on August 22, 1984, and consulted with the state of South Carolina. No public comments were received, and the state of South Carolina did not have any comments.

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Jon B. Hopkins, Licensing Branch No. 4, DL
Bernard Mann, Reactor Systems Branch, DSI

Dated: November 30, 1984