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

April 30, 1985

Docket No. 50-395

Mr. O. W. Dixon, Jr.
Vice President Nuclear Operations
South Carolina Electric & Gas Company
P.O. Box 764 (Mail Code 167)
Columbia, South Carolina 29218

Dear Mr. Dixon:

Subject: Issuance of Amendment No. 40 to Facility Operating License NPF-12 Virgil C. Summer Nuclear Station, Unit No. 1

The Nuclear Regulatory Commission has issued Amendment No. 40 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 August 24, 1984, and revised November 14, 1984.

The amendment modifies the Technical Specifications to change time constant T_1 in the overtemperature delta-T setpoint equation from 33 seconds to 28 seconds and to change the reactor trip setpoint for the steam generator water level low-low signal. The amendment is effective seven days after its date of issuance.

A copy of the related safety evaluation supporting Amendment No. 40 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. 40
2. Safety Evaluation

cc w/enclosure:
See next page

DESIGNATED ORIGINAL
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SUMMER (Amendment)

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April 30, 1985

AMENDMENT NO. 40 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. 40
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 August 24, 1984, and revised November 14, 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. 40 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 seven days after 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 Changes

Date of Issuance: April 30, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 40

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
2-6	2-5
2-8	
3/4 3-28	3/4 3-27

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	S	Trip Setpoint	Allowable Value
1. Manual Reactor Trip	Not Applicable	NA	NA	NA	NA
2. Power Range, Neutron Flux High Setpoint	7.5	4.56	0	$\leq$109% of RTP	$\leq$111.2% of RTP
Low Setpoint	8.3	4.56	0	$\leq$25% of RTP	$\leq$27.2% of RTP
3. Power Range, Neutron Flux High Positive Rate	1.6	0.5	0	$\leq$5% of RTP with a time constant $\geq$2 seconds	$\leq$6.3% of RTP with a time constant $\geq$2 seconds
4. Power Range, Neutron Flux High Negative Rate	1.6	0.5	0	$\leq$5% of RTP with a time constant $\geq$2 seconds	$\leq$6.3% of RTP with a time constant $\geq$2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.4	0	$\leq$25% of RTP	$\leq$31% of RTP
6. Source Range, Neutron Flux	17.0	10.0	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT	7.1	2.94	1.8	See note 1	See note 2
8. Overpower ΔT	4.5	1.4	1.2	See note 3	See note 4
9. Pressurizer Pressure-Low	3.1	0.71	1.5	$\geq$1870 psig	$\geq$1859 psig
10. Pressurizer Pressure-High	3.1	0.71	1.5	$\leq$2380 psig	$\leq$2391 psig
11. Pressurizer Water Level-High	5.0	2.18	1.5	$\leq$92% of instrument span	$\leq$93.8% of instrument span
12. Loss of Flow	2.5	1.0	1.5	$\geq$90% of loop design flow*	$\geq$89.2% of loop design flow*

*Loop design flow = 98,000 gpm
RTP = RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	S	Trip Setpoint	Allowable Value
13. Steam Generator Water Level Low-Low	12.0	9.18	1.5	≥12% of span from 0 to 30% RTP increasing linearly to ≥30.0% of span from 30% to 100% RTP	≥10.2% of span from 0 to 30% RTP increasing linearly to ≥28.2% of span from 30% to 100% RTP
14. Steam/Feedwater Flow Mis-Match Coincident With	16.0	13.24	1.5/ 1.5	≤40% of full steam flow at RTP	≤42.5% of full steam flow at RTP
Steam Generator Water Level Low-Low	12.0	9.18	1.5	≥12% of span from 0 to 30% RTP increasing linearly to ≥30.0% of span from 30% to 100% RTP	≥10.2% of span from 0 to 30% RTP increasing linearly to ≥28.2% of span from 30% to 100% RTP
15. Undervoltage - Reactor Coolant Pump	2.1	1.28	0.23	≥4830 volts	≥4760
16. Underfrequency - Reactor Coolant Pumps	7.5	0	0.1	≥57.5 Hz	≥57.1 Hz
17. Turbine Trip					
A. Low Trip System Pressure	NA	NA	NA	≥800 psig	≥750 psig
B. Turbine Stop Valve Closure	NA	NA	NA	≥1% open	≥1% open

RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \leq \Delta T_o \{K_1 - K_2 \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} [T - T'] + K_3(P - P') - f_1(\Delta I)\}$$

Where:	ΔT	=	Measured ΔT by RTD Manifold Instrumentation
	ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
	K_1	=	1.090
	K_2	=	0.01450
	$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
	τ_1, & τ_2	=	Time constants utilized in the lead-lag controller for T_{avg}, τ_1 = 28 secs., τ_2 = 4 secs.
	T	=	Average temperature °F
	T'	$\leq$	587.4°F Reference T_{avg} at RATED THERMAL POWER
	K_3	=	.0006728
	P	=	Pressurizer pressure, psig
	P'	=	2235 psig, Nominal RCS operating pressure
	S	=	Laplace transform operator, sec^{-1}.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	S	Trip Setpoint	Allowable Value
4. STEAM LINE ISOLATION					
a. Manual	NA	NA	NA	NA	NA
b. Automatic Actuation Logic and Actuation Relays	NA	NA	NA	NA	NA
c. Reactor Building Pressure-High 2	3.0	0.71	1.5	≤6.35	≤6.61
d. Steam Flow in Two Steamlines-High, Coincident with	20.0	13.16	1.5/ 1.5	≤ a function defined as follows: A ΔP corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load	≤ a function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 114.0% of full steam flow at full load.
T_{avg} - Low-Low	4.0	1.12	1.2	≥553°F	≥550.6°F
e. Steamline Pressure - Low	20.0	10.71	1.5	≥675 psig	≥635 psig(1)

(1) Time constants utilized in lead lag controller for steamline pressure low are as follows: $\tau_1 \geq 50$ secs. $\tau_2 \leq 5$ secs.

TABLE 3.3-4

	Functional Unit	Total Allowance (TA)	Z	S	Trip Setpoint	Allowable Value
5.	TURBINE TRIP AND FEEDWATER ISOLATION					
	a. Steam Generator Water Level - High-High	5.0	2.18	1.5	$\leq$82.4% of narrow range instrument span	$\leq$84.2% of narrow range instrument span
6.	EMERGENCY FEEDWATER					
	a. Manual	NA	NA	NA	NA	NA
	b. Automatic Actuation Logic	NA	NA	NA	NA	NA
	c. Steam Generator Water Level - Low-Low	12.0	9.18	1.5	$\geq$12% of span from 0% to 30% RTP increasing linearly to $\geq$30.0% of span from 30% to 100% RTP	$\geq$10.2% of span from 0% to 30% RTP increasing linearly to $\geq$28.2% of span from 30% to 100% RTP
	d. & f. Undervoltage-ESF Bus				$\geq$5760 Volts with a $\leq$0.25 second time delay	$\geq$5652 Volts with a $\leq$0.275 second time delay
					$\geq$6576 volts with a $\leq$3.0 second time delay	$\geq$6511 volts with a $\leq$3.3 second time delay

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 40 TO FACILITY OPERATING LICENSE NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

I. INTRODUCTION

By letter dated August 24, 1984, South Carolina Electric and Gas Company (the licensee) requested Technical Specification changes in the trip setpoint requirements for the low-low steam generator reactor trip and the overtemperature delta-T trip. Data taken during the startup test program revealed that the plant did not have the capability to handle a full load rejection without tripping using the original setpoints. The requested modifications, together with a modification to the steam dump control setpoint, would allow the plant to accommodate a complete loss of external load in accordance with the original design as described in the FSAR. Modification to the steam dump control setpoint does not involve a Technical Specification change. In response to NRC staff questions, the licensee submitted additional information by letter dated November 14, 1984.

This amendment request of August 24, 1984, was noticed in the Federal Register (50 FR 12162) on March 27, 1985, as including a change to T_4 in the overtemperature delta-T trip setpoint equation. By amendment No. 28 to NPF-12 (October 12, 1984), time constant T_4 was renumbered to be T_1, and T_4 was eliminated. Therefore, time constant T_1 is actually being changed, but it is the same time constant as that requested originally and noticed. Therefore, this amendment request is not being renoticed.

II. EVALUATION

Steam generator level (narrow range) for the Westinghouse Model D steam generator is measured within the downcomer region. Sudden reductions in steamflow cause a rapid drop in downcomer level. The level change is caused by the redistribution of water within the steam generator and not by change in the total water mass.

The licensee requested that the reactor trip setpoint be changed for the steam generator water level low-low signal. Currently, this setpoint is linear from 12% to 54.9% of span for 30% to 100% of rated thermal power (RTP). This would be changed to 12% to 30% of span for 30% to 100% of RTP. Also, the allowable value associated with the trip setpoint is being changed a corresponding amount. The low-low level trip setpoint is relied upon in the FSAR to trip the reactor and initiate auxiliary feedwater following loss of main feedwater. A 0% level on the narrow range was assumed for the setpoint in the FSAR. The revised setpoint is 5.8 feet above the FSAR assumption. The revised setpoint is within the bounds of the current safety analysis and is, therefore, acceptable.

The change proposed by the licensee to the overtemperature delta-T setpoint equation involves changing the time constant T_1 from 33 seconds to 28 seconds. The decrease in the time constant T_1 will decrease the compensation to the reactor core average temperature and, consequently, reduce the penalty to the overtemperature delta-T setpoint caused by increases in the core average temperature.

The licensee states that, of the seven safety analyses listed under overtemperature delta-T in FSAR Table 7.2-4, only four take credit for a reactor trip initiated by the overtemperature delta-T protection circuit. These events are:

1. Uncontrolled rod withdrawal at power
2. Uncontrolled boron dilution at power
3. Loss of load transient
4. Accidental depressurization of the reactor coolant system

The other three safety analyses (excessive heat removal, excessive load increase, and accidental depressurization of the main steam system) do not take credit for a reactor trip based on the overtemperature delta-T protection circuit. The overtemperature delta-T trip does, however, provide a backup trip. For these three transients and the accidental depressurization of the reactor coolant system, the core average temperature decreases resulting in a credit to the overtemperature delta-T setpoint. The decrease in the time constant T_1 delays this credit and is conservative for these four events. The proposed decrease in T_1 is, therefore, acceptable for these four events. The remaining three events are evaluated below.

Uncontrolled Rod Withdrawal at Power

This event is protected by the overtemperature delta-T trip for low reactivity insertion rates and by the high flux trip for high reactivity insertion rates. The limiting departure from nucleate boiling ratio (DNBR) can be plotted for a given core power and moderator feedback as a function of reactivity insertion rate. The two trips overlap to some degree. The effect of decreasing the time constant T_1 will cause the high flux trip to become effective for this event at a somewhat lower reactivity insertion rate. The DNBR, however, never becomes less than the Technical Specification limiting value. We find, therefore, that the decrease in T_1 is acceptable for this event.

Uncontrolled Boron Dilution at Power

This event requires operator action both to recognize that the reactor is undergoing the event and to terminate the event with acceptable consequences.

The safety analysis indicates that the operator has 43.2 minutes after a trip to stop the dilution. A change in the time constant T_1 which is proposed here changes this time to 43 minutes. The decrease in the time available to the operator to terminate the boron dilution is insignificant. We find, therefore, that the decrease in T_1 is acceptable for this event.

Loss of Load Transient

This event is protected by the overtemperature delta-T trip when the pressurizer pressure control is assumed to function and by the high pressurizer pressure trip when the pressurizer pressure control is not assumed to be operable. For a beginning of the life event with pressurizer pressure control, the decrease in T_1 results in a slight delay in the overtemperature delta-T trip and a slightly lower minimum DNBR of 1.50 which is well above the Technical Specification limiting DNBR value of 1.30. For the end of life case with pressurizer pressure control, the decrease in T_1 again results in a slight delay in the overtemperature delta-T trip. The DNBR does not decrease below its initial value because of the decrease in reactor power caused by the large negative moderator coefficient and the increase in pressurizer pressure. We find, therefore, that the decrease in T_1 is acceptable for this event.

Based on our review, we conclude that the proposed changes are acceptable because the effect on FSAR safety analyses has been demonstrated to not exceed applicable criteria.

III. ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the use 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). Purusant 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 (50 FR 12162) on March 27, 1985, 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.

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Dated: April 30, 1985