November 18, 1994

Mr. John L. Skolds
Senior Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88
Jenkinsville, South Carolina 29065

Dear Mr. Skolds:

SUBJECT: ISSUANCE OF AMENDMENT NO. 120 TO FACILITY OPERATING LICENSE NO. NPF-12 REGARDING REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS- VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1 (TAC NO. M90041)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 120 to Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1. The amendment changes the Technical Specifications (TS) in response to your application dated July 20, 1994.

The amendment changes the Technical Specifications to modify TS Table 2.2-1, Reactor Trip System Instrumentation Setpoints, and Table 3.3-4, Engineered Safety Features Actuation System Instrumentation Trip Setpoints and several associated bases. The change would remove specific rack and sensor allowable drift values by removing three columns from the tables.

A copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's Bi-weekly Federal Register notice.

Sincerely,

Original signed by:

George F. Wunder, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosures:
1. Amendment No. 120 to NPF-12
2. Safety Evaluation

cc w/enclosures:
See next page
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AMENDMENT NO. 120 TO FACILITY OPERATING LICENSE NO. NPF-12 - SUMMER, UNIT NO.1

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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. 120
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 , complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-12 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 120 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

William H. Bateman, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 18, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 120

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 revised pages are indicated by marginal lines.

Remove Pages	Insert Pages
2-4	2-4
2-5	2-5
2-6	2-6
2-7	2-7
2-8	2-8
2-9	2-9
2-10	2-10
B 2-1	B 2-1
B 2-3	B 2-3
B 2-4	B 2-4
B 2-6	B 2-6
3/4 3-15	3/4 3-15
3/4 3-15a	3/4 3-15a
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
3/4 3-28a	3/4 3-28a
3/4 3-28b	3/4 3-28b
B 3/4 3-1	B 3/4 3-1
B 3/4 3-1a	B 3/4 3-1a

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor trip system instrumentation and interlocks setpoints shall be consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a reactor trip system instrumentation or interlock setpoint less conservative than the value shown in the Trip Setpoint column of Table 2.2-1 adjust the setpoint consistent with the Trip Setpoint value.

b. With the reactor trip system instrumentation or interlock setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirements of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint Value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
1.	Manual Reactor Trip	NA	NA
2.	Power Range, Neutron Flux High Setpoint Low Setpoint	 ≤109% of RTP ≤25% of RTP	 ≤111.2% of RTP ≤27.2% of RTP
3.	Power Range, Neutron Flux High Positive Rate	≤5% of RTP with a time constant ≥2 seconds	≤6.3% of RTP with a time constant ≥2 seconds
4.	DELETED		
5.	Intermediate Range, Neutron Flux	≤25% of RTP	≤31% of RTP
6.	Source Range, Neutron Flux	≤10^5 cps	≤1.4 x 10^5 cps
7.	Overtemperature ΔT	See note 1	See note 2
8.	Overpower ΔT	See note 3	See note 4
9.	Pressurizer Pressure-Low	≥1870 psig	≥1859 psig
10.	Pressurizer Pressure-High	≤2380 psig	≤2391 psig
11.	Pressurizer Water Level-High	≤92% of instrument span	≤93.8% of instrument span
12.	Loss of Flow	≥90% of loop design flow*	≥88.9% of loop design flow*

* Loop design flow = 94,500 gpm
RTP - RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
13.	Steam Generator Water Level Low-Low Barton Transmitter Rosemount Transmitter	 ≥27.0% of span ≥27.0% of span	 ≥26.1% of span ≥25.7% of span
14.	Steam/Feedwater Flow Mis-Match Coincident With Steam Generator Water Level Low-Low Barton Transmitter Rosemount Transmitter	≤40% of full steam flow at RTP ≥27.0% of span ≥27.0% of span	≤42.5% of full steam flow at RTP ≥26.1% of span ≥25.7% of span
15.	Undervoltage - Reactor Coolant Pump	≥4830 volts	≥4760 volts
16.	Underfrequency - Reactor Coolant Pumps	≥57.5 Hz	≥57.1 Hz
17.	Turbine Trip A. Low Trip System Pressure B. Turbine Stop Valve Closure	 ≥800 psig ≥1% open	 ≥750 psig ≥1% open

RTP - RATED THERMAL POWER

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
18.	Safety Injection Input from ESF	NA	NA
19.	Reactor Trip System Interlocks		
	A. Intermediate Range Neutron Flux, P-6	$\geq 7.5 \times 10^{-6}$ % indication	$\geq 4.5 \times 10^{-6}$ % indication
	B. Low Power Reactor Trips Block, P-7 a. P-10 input	$\leq$10% of RTP	$\leq$12.2% of RTP
	b. P-13 input	$\leq$10% turbine impulse pressure equivalent	$\leq$12.2% of turbine impulse pressure equivalent
	C. Power Range Neutron Flux P-8	$\leq$38% of RTP	$\leq$40.2% of RTP
	D. Low Setpoint Power Range Neutron Flux, P-10	$\geq$10% of RTP	$\geq$7.8% of RTP
	E. Turbine Impulse Chamber Pressure, P-13	$\leq$10% turbine impulse pressure equivalent	$\leq$12.2% turbine pressure equivalent
	F. Power Range Neutron Flux, P-9	$\leq$50% of RTP	$\leq$52.2% of RTP
20.	Reactor Trip Breakers	NA	NA
21.	Automatic Actuation Logic	NA	NA

RTP - RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE ΔT

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

Where:	ΔT	=	Measured ΔT by RTD Instrumentation
	ΔT_o	$\leq$	Indicated ΔT at RATED THERMAL POWER
	K_1	$\leq$	1.23
	K_2	$\geq$	0.0292/°F
	$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	The funtion generated by the lead-lag controller for T_{avg} dynamic compensation
	τ_1, τ_2	=	Time constants utilized in lead-lag controller for T_{avg}, $\tau_1 \geq 28$ secs., $\tau_2 \leq 4$ secs.
	T	=	Average temperature, °F
	T'	$\leq$	Indicated T_{avg} at RATED THERMAL POWER, 572.0°F $\leq$ T' $\leq$ 587.4°F
	K_3	$\geq$	0.00161/psi
	P	=	Pressurizer pressure, psig
	P'	$\geq$	2235 psig, Nominal RCS operating pressure
	S	=	Laplace transform operator, sec^{-1}.

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (continued)

NOTE 1: (Continued)

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between -35 percent and +6 percent $f_1(\Delta I) = 0$ where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER.

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -35 percent, the ΔT trip setpoint shall be automatically reduced by 2.46 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +6 percent, the ΔT trip setpoint shall be automatically reduced by 3.29 percent of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 2.2 percent ΔT Span.

NOTE 3: OVERPOWER ΔT

$$\Delta T \leq \Delta T_o \left[K_4 - K_5 \frac{(\tau_3 S)}{(1 + \tau_3 S)} T - K_6 \left[T - T'' \right] \right]$$

Where:	ΔT	=	as defined in Note 1
	ΔT_o	=	as defined in Note 1
	K_4	≤	1.078
	K_5	≥	0.02/°F for increasing average temperature and 0 for decreasing average temperature
	$\frac{\tau_3 S}{1 + \tau_3 S}$	=	The function generated by the rate-lag controller for T_{avg} dynamic compensation

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (continued)

NOTE 3: (continued)

τ_3	=	Time constant utilized in rate-lag controller for T_{avg}, $\tau_3 \geq 10$ secs.
K_6	$\geq$	0.00198/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$
T	=	as defined in Note 1
T''	$\leq$	Indicated T_{avg} at RATED THERMAL POWER, 572.0°F $\leq T'' \leq$ 587.4°F
S	=	as defined in Note 1

NOTE 4: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 2.3 percent ΔT Span.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio (DNBR) defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability with 95 percent confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. In addition, margin has been maintained in the design by meeting safety analysis DNBR limits in performing safety analyses.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the design DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The setpoint for a reactor trip system or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the reactor trip setpoints have been specified in Table 2.2-1. Operation with setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error.

The methodology to derive the trip setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the trip setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various reactor trip circuits automatically open the reactor trip breakers whenever a condition monitored by the Reactor Protection System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Protection System which monitors numerous system variables, therefore, providing protection system functional diversity. The Reactor Protection System initiates a turbine trip signal whenever reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive reactor system cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Protection System includes manual reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a high and low range trip setting. The low setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the high setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The low setpoint trip may be manually blocked above P-10 (a power level of approximately 10 percent of RATED THERMAL POWER) and is automatically reinstated below the P-10 setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup to mitigate the consequences of an

Pressurizer Pressure (Continued)

On decreasing power the low setpoint trip is automatically blocked by P-7 (a power level of approximately 10 percent of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10 percent of full power equivalent); and on increasing power, automatically reinstated by P-7.

The high setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The pressurizer high water level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the pressurizer high water level trip is 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 equivalent); and on increasing power, automatically reinstated by P-7.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10 percent of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10 percent of full power equivalent), an automatic reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 38 percent of RATED THERMAL POWER) an automatic reactor trip will occur if the flow in any single loop drops below 90 percent of nominal full loop flow. Conversely on decreasing power between P-8 and the P-7 an automatic reactor trip will occur on loss of flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The steam generator water level low-low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified setpoint provides allowances for starting delays of the auxiliary feedwater system.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The steam/feedwater flow mismatch in coincidence with a steam generator low water level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Protection System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than or equal to 40% of full steam flow at RTP. The Steam Generator Low Water level portion of the trip is activated when the water level drops below the low

3/4 3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint Column but more conservative than the value shown in the Allowable Value Column of Table 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value Column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to its OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

c. With an ESFAS instrumentation channel or interlock inoperable take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the engineered safety feature actuation system instrumentation surveillance requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

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TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
1.	SAFETY INJECTION, REACTOR TRIP, FEEDWATER ISOLATION, CONTROL ROOM ISOLATION, START DIESEL GENERATORS, CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER.		
	a. Manual Initiation	NA	NA
	b. Automatic Actuation Logic	NA	NA
	c. Reactor Building Pressure-High 1	≤3.6 psig	≤3.86 psig
	d. Pressurizer Pressure--Low	≥1850 psig	≥1839 psig
	e. Differential Pressure Between Steamlines--High	≤97 psig	≤106 psi
	f. Steamline Pressure--Low	≥675 psig	≥635 psig(1)
2.	REACTOR BUILDING SPRAY		
	a. Manual Initiation	NA	NA
	l. Automatic Actuation Logic and Actuation Relays	NA	NA
	c. Reactor Building Pressure-High 3 (Phase 'A' isolation aligns spray system discharge valves and NaOH tank suction valves)	≤12.05 psig	≤12.31 psig

(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 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
3.	CONTAINMENT ISOLATION		
	a. Phase "A" Isolation		
	1. Manual	NA	NA
	2. Safety Injection	See 1 above for all safety injection setpoints	See 1 above for all allowable values
	3. Automatic Actuation Logic and Actuation Relays	NA	NA
	b. Phase "B" Isolation		
	1. Automatic Actuation Logic and Actuation Relays	NA	NA
	2. Reactor Building Pressure-High 3	$\leq$12.05 psig	$\leq$12.31 psig
	c. Purge and Exhaust Isolation		
	1. Safety Injection	See 1 above for all safety injection setpoints	See 1 above for all allowable values
	2. Containment Radioactivity High	*	*
	3. Automatic Actuation Logic and Actuation Relays	NA	NA

* Trip setpoints shall be set to ensure that the limits of ODCM Specification 1.2.2.1 are not exceeded.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
4.	STEAM LINE ISOLATION		
	a. Manual	NA	NA
	b. Automatic Actuation Logic and Actuation Relays	NA	NA
	c. Reactor Building Pressure-High 2	$\leq$6.35	$\leq$6.61
	d. Steam Flow in Two Steamlines-High, Concident with	$\leq$ 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	$\leq$ 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	$\geq$552.0°F	$\geq$548.4°F
	e. Steamline Pressure-Low	$\geq$675 psig	$\geq$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 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
5.	TURBINE TRIP AND FEEDWATER ISOLATION		
	a. Steam Generator Water Level - High-High Barton Transmitter Rosemount Transmitter	 $\leq$79.2% of span $\leq$79.2% of span	 $\leq$81.0% of span $\leq$81.0% of span
6.	EMERGENCY FEEDWATER		
	a. Manual	NA	NA
	b. Automatic Actuation Logic	NA	NA
	c. Steam Generator Water Level - Low-Low Barton Transmitter Rosemount Transmitter	 $\geq$27.0% of span $\geq$27.0% of span	 $\geq$26.1% of span $\geq$25.7% of span
	d. & f. Undervoltage-ESF Bus	$\geq$5760 Volts with a $\leq$0.25 second time delay $\geq$6576 Volts with a $\leq$3.0 second time delay	$\geq$5652 Volts with a $\leq$0.275 second time delay $\geq$6511 Volts with a $\leq$3.3 second time delay

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
	e. Safety Injection	See 1 above (all SI Setpoints)	See 1 above (all SI Setpoints)
	g. Trips of Main Feedwater Pumps	NA	NA
	h. Suction transfer on Low Pressure	$\geq$442 ft. 4 in. (2)	$\geq$441 ft. 3 in.
7.	LOSS OF POWER		
	a. 7.2 kv Emergency Bus Undervoltage (Loss of Voltage)	$\geq$5760 volts with a $\leq$0.25 second time delay	$\geq$5652 volts with a $\leq$0.275 second time delay
	b. 7.2 kv Emergency Bus Undervoltage	$\geq$6576 volts with a $\leq$3.0 second time delay	$\geq$6511 volts with a $\leq$3.3 second time delay
8.	AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP		
	a. RWST Level Low-Low	$\geq$18%	$\geq$15%
	b. Automatic Actuation Logic and Actuation Relays	NA	NA

(2) Pump suction head at which transfer is initiated is stated in effective water elevation in the condensate storage tank.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
9.	ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS		
	INTERLOCKS		
	a. Pressurizer Pressure, P-11	1985 psig	$\geq$1974 psig & $\leq$1996 psig
	b. T_{avg} Low-Low, P-12	552°F	$\geq$548.4°F & $\leq$555.6°F
	c. Reactor Trip, P-4	NA	NA

3/4.3.1 and 3/4.3.2 REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Protection System and Engineered Safety Feature Actuation System Instrumentation and interlocks ensure that 1) the associated action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoints, 2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and, 3) sufficient system functions capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses. The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for Reactor Protection Instrumentation System," and supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

The Engineered Safety Feature Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A setpoint is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the setpoints have been specified in Table 3.3-4. Operation with setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error.

The methodology to derive the trip setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the trip setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this

REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (continued)

will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the reactor trip and the engineered safety feature actuation associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite, or offsite test measurements or 2) utilizing replacement sensors with certified response times.

The Engineered Safety Features response times specified in Table 3.3-5 which include sequential operation of the RWST and VCT valves (Notes 2 and 3) are based on values assumed in the non-LOCA safety analyses. These analyses are for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction isolation valves are closed following opening of the RWST charging pumps suction valves. When the sequential operation of the RWST and VCT valves is not included in the response times (Note 1) the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response times specified in Table 3.3-5 will assure that the assumptions used for the LOCA and non-LOCA analyses with respect to the operation of the VCT and RWST valves are valid.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those engineered safety features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss of coolant accident 1) safety injection pumps start and automatic valves position, 2) reactor trip, 3) feedwater isolation, 4) startup of the emergency diesel generators, 5) containment spray pumps start and automatic valves position, 6) containment isolation, 7) steam line isolation, 8) turbine trip, 9) auxiliary feedwater pumps start and automatic valves position, 10) containment cooling fans start and automatic valves position, 11) essential service water pumps start and automatic valves position, and 12) control room isolation and ventilation systems start.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 120 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 July 20, 1994, South Carolina Electric & Gas Company (the licensee), submitted a request for changes to the Virgil C. Summer Nuclear Station, Unit No. 1, (Summer Station) Technical Specifications (TS).

The proposed change would modify TS Tables 2.2-1, "Reactor Trip System Instrumentation Setpoints," and 3.3-4, "Engineered Safety Features Actuation System Instrumentation Trip Setpoints," and several associated bases. The proposed change would remove three columns from the Tables. The columns contain specific rack and sensor allowable drift values.

2.0 EVALUATION

Before the issuance of 10 CFR 50.73 on July 26, 1983, the reportability of reactor trip system and engineered safety features actuation system discrepancies was governed by the TS. The TS were specifically designed so that if an instrument setpoint was determined to be less conservative than a specified value, the condition would not be reportable as long as certain instrument drift conditions were met. Whether or not these conditions were met was determined by substituting the instruments individual allowable drift values, as specified in the TS, into an equation. If the equation was satisfied, the incident was not reportable.

Since the issuance of 10 CFR 50.73, a reactor trip system or engineered safety features actuation system instrument discrepancy is not reportable unless instrument function is lost. The licensee no longer has a need for the drift values specified in the three columns of the TS tables covering these instruments because the reportability of a discrepancy is specifically governed by 10 CFR 50.73. The removal of the three columns from the TS has no impact on plant operation or safety and is acceptable.

The proposed amendment also makes a correction to the definition of OVERPOWER DELTA T on page 2-10 due to a typographical error. This change is acceptable.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (59 FR 47181). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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

Principal Contributor: G. Wunder

Date: November 18, 1994