December 15, 1992

Docket No. 50-395

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Mr. John L. Skolds
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. 108 TO FACILITY OPERATING LICENSE NO. NPF-12 REGARDING REVISED ENGINEERED SAFETY FEATURES RESPONSE TIMES - VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1, (TAC NO. M83224)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 108 to Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application dated April 15, 1992.

The amendment changes the Technical Specifications to revise Engineered Safety Features response times to account for sequential stroking of the outlet isolation valves on the refueling water storage tank and volume control tank.

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

Enclosures:
1. Amendment No. 108 to NPF-12
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	LA:PD21:DRPE	PM:PD21:DRPE	SICB	SRXB	OGC
NAME	SLittle	GWunder:tms	SNewberry	RJones	MYoung
DATE	11/5/92	11/06/92	11/12/92	11/16/92	12/3/92
OFC	D:PD21:DRPE				
NAME	EAdensam				
DATE	12/15/92				

Document Name: SUM83224.AMD

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Mr. John L. Skolds
South Carolina Electric & Gas Company

Virgil C. Summer Nuclear Station

cc:

Mr. R. J. White
Nuclear Coordinator
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AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. NPF-12 - SUMMER, UNIT NO. 1

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

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. 108
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 December 15, 1992, 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. 108 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, 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: December 15, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 108

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
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-33	3/4 3-33
3/4 3-34	3/4 3-34

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION			RESPONSE TIME IN SECONDS
1.	Manual		
	a.	Safety Injection	Not Applicable
	b.	Reactor Building Spray	Not Applicable
	c.	Containment Isolation Phase "A" Isolation	Not Applicable
	d.	Steam Line Isolation	Not Applicable
	e.	Feedwater Isolation	Not Applicable
	f.	Emergency Feedwater	Not Applicable
	g.	Essential Service Water	Not Applicable
	h.	Reactor Building Cooling Fans	Not Applicable
	i.	Control Room Isolation	Not Applicable
2.	Reactor Building Pressure-High		
	a.	Safety Injection (ECCS)	$\leq$27.0(2)/27.0(1)
	b.	Reactor Trip (from SI)	$\leq$3.0
	c.	Feedwater Isolation	$\leq$10.0
	d.	Containment Isolation-Phase "A"	$\leq$45.0(4)/55.0(5)

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
e. Reactor Building Purge and Exhaust Isolation	Not Applicable
f. Emergency Feedwater Pumps	Not Applicable
g. Service Water System	71.5(4)/81.5(5)
h. Reactor Building Cooling Units	76.5(4)/86.5(5)
i. Control Room Isolation	Not Applicable
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤27.0(2)/27.0(1)
b. Reactor Trip (from SI)	≤3.0
c. Feedwater Isolation	≤10.0
d. Containment Isolation -Phase "A"	≤45.0(4)/55.0(5)
e. Reactor Building Purge and Exhaust Isolation	Not Applicable
f. Emergency Feedwater Pumps	Not Applicable
g. Service Water System	71.5(4)/81.5(5)
h. Reactor Building Cooling Units	76.5(4)/86.5(5)
i. Control Room Isolation	Not Applicable
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	≤27.0(2)/37.0(3)
b. Reactor Trip (from SI)	≤3.0
c. Feedwater Isolation	≤10.0
d. Containment Isolation -Phase "A"	≤45.0(4)/55.0(5)

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
	e. Reactor Building Purge and Exhaust Isolation	Not Applicable
	f. Emergency Feedwater Pumps	Not Applicable
	g. Service Water System	$\leq$71.5(4)/81.5(5)
	h. Reactor Building Cooling Units	$\leq$76.5(4)/86.5(5)
	i. Control Room Isolation	Not Applicable
5.	Steam Line Pressure-Low	
	a. Safety Injection - ECCS	$\leq$27.0(2)/37.0(3)
	b. Reactor Trip (from SI)	$\leq$3.0
	c. Feedwater Isolation	$\leq$10.0
	d. Containment Isolation - Phase "A"	$\leq$45.0(4)/55.0(5)
	e. Reactor Building and Purge and Exhaust Isolation	Not Applicable
	f. Emergency Feedwater Pumps	Not Applicable
	g. Service Water System	$\leq$71.5(4)/81.5(5)
	h. Reactor Building Cooling Units	$\leq$76.5(4)/86.5(5)
	i. Steam Line Isolation	$\leq$10.0
	j. Control Room Isolation	Not Applicable
6.	Steam Flow in Two Steam Lines - High Coincident with T_{avg}--Low-Low	
	a. Steam Line Isolation	$\leq$12.0
7.	Reactor Building Pressure-High-2	
	a. Steam Line Isolation	$\leq$9.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
13.	Trip of Main Feedwater Pumps	
	a. Motor-driven Emergency Feedwater Pumps	Not Applicable
14.	Loss of Power	
	a. 7.2 kv Emergency Bus Undervoltage (Loss of Voltage)	$\leq$10.3
	b. 7.2 kv Emergency Bus Undervoltage (Degraded Voltage)	$\leq$13.3
15.	Containment Radioactivity--High	
	a. Purge and Exhaust Isolation	Not Applicable
16.	RWST Level--Low-Low	
	a. Automatic Switchover to Containment Sump	Not Applicable
17.	Aux Feed Suction Pressure Low	
	a. Suction transfer	Not Applicable
Note:	Response time for Motor-driven Emergency Feedwater Pumps on all SI signal starts	$\leq$60.0

TABLE 3.3-5 (Continued)

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays from under voltage included. Response time limit includes positioning of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps and RHR pumps. Sequential transfer of centrifugal charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is not included.

(2) Diesel generator starting delay not included. Sequence loading delay included. Offsite power available. Response time limit includes positioning of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of centrifugal charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(3) Diesel generator starting and sequence loading delays from under voltage included. Response time limit includes positioning of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of centrifugal charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(4) Diesel generator starting delay not included. Sequence loading delay included. Offsite power available.

(5) Diesel generator starting and sequence loading delays from undervoltage included.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 108 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 April 15, 1992, 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 requested changes would revise Engineered Safety Features (ESF) response times in Table 3.3-5 to account for sequential stroking of the outlet isolation valves on the refueling water storage tank (RWST) and the volume control tank (VCT).

2.0 EVALUATION

Table 3.3-5 in the Summer Station TS, Engineered Safety Features Response Times, provides the interval from when a monitored ESF parameter exceeds its actuation setpoint until the ESF equipment must be capable of performing its safety function. These response times are measured at least once every 18 months.

In its normal configuration, the chemical and volume control system (CVCS) is aligned such that the charging pumps, which double as the high-head safety injection pumps, take suction from the VCT. When a safety injection (SI) signal is generated by the protection logic, a start signal is sent to the charging pumps (which then begin to function as SI pumps). In addition, a signal is sent to open the RWST outlet isolation valves, thus lining up this source of borated water to the suction of the high-head SI pumps. Only after the RWST outlet isolation valves are fully open does the logic tell the VCT outlet isolation valves to shut. The logic is designed this way to ensure that there is always an adequate net positive suction head to the SI pumps. The entire switchover from VCT suction to RWST suction can take as long as 25 seconds. Since the RWST contains heavily borated water and the VCT does not, it is conservative to assume that the RWST serves as the source of water to the SI pumps only after the VCT outlet isolation valves are fully shut.

Current TS times take into account only the opening of the RWST outlet isolation valves; the requested change would include the time for the VCT outlet isolation valves to shut. In order to evaluate this change, its impact on large and small break loss-of-coolant accidents (LOCA's), post-LOCA plant performance, steam generator tube rupture (SGTR) accidents, and non-LOCA transients was assessed.

Loss of Coolant Accidents

Large Break LOCA

After a large break LOCA, the formation of steam voids in the core adds enough negative reactivity to cause the reactor to go subcritical. Thus, the immediate injection of boron to reduce nuclear generated heat is not necessary. This is why large break LOCA analyses assume that the immediate function of the SI system is to provide water to the primary in order to cool the core. Whether or not the water is borated or unborated is not of immediate concern. Water will be available from the SI system when the SI pumps reach full speed regardless of the position of the VCT outlet isolation valves.

The injection of borated water will be necessary to maintain long-term subcriticality; however, the additional delay due to RWST switchover is not long-term. Since borated water is not immediately necessary to mitigate a large break LOCA, the proposed change does not have any impact on the Summer Station large break LOCA analyses.

Small Break LOCA

Small break LOCA analyses assume that the reactor will become subcritical due to the insertion of all but the most reactive control rod, and no credit is taken for increased boration due to injection of water from the RWST. As with large break LOCA, then, the assumption is that the immediate function of the SI system is to provide water to cool the core.

The analyses take credit for the injection of borated water in maintaining long-term subcriticality; however, the additional delay due to RWST switchover is not long-term. Since borated water is not immediately necessary to mitigate a small break LOCA, the proposed change does not have any impact on the Summer Station small break LOCA analyses.

Control Rod Ejection Accident

Since the SI flow following a control rod ejection is modeled under similar assumptions as in the LOCA analyses, the delay in borated water injection will have negligible impact. The analysis of the control rod ejection accident in the Final Safety Analysis Report (FSAR) will remain valid.

Post-LOCA Plant Performance

Reactor Vessel and Loop Blowdown Forces

Since the maximum blowdown forces are generated within the first few seconds of break initiation, the initiation of ECCS flow is not considered in the hydraulic forces modeling. For this reason, the delay time in the initiation of borated water flow has no impact on these analyses.

Short- and Long-Term Mass and Energy Release

For the containment subcompartment analyses the short duration of the transient (<3 seconds) leads to the assumption that there is not yet any SI flow to the reactor coolant system (RCS). Therefore, any delay in the injection of borated water could not have any effect on the analyses.

Post-LOCA Long-Term Core Cooling

The Summer Station licensing position regarding long-term core cooling is that, following a postulated LOCA, the reactor will remain subcritical due to borated ECCS water residing in the sump. Since no credit is taken for control rod worth, the core must remain subcritical due to the boron concentration of the SI water, assuming it is diluted by all other water that could be in the sump.

In the normal configuration, the charging pumps are aligned to the VCT. It is conservative to assume that, following an SI signal, the SI pumps inject only VCT water into the primary until the VCT outlet isolation valves are fully shut. The effect of injecting non-borated water for the duration of the RWST switchover has been considered in the long-term core cooling evaluation. Assuming maximum flow from all SI pumps, the amount of unborated water that could be injected before switchover to the RWST is complete has been estimated at about 3363 lbs. This amount of water would result in a reduction in sump boron concentration of about 2.1 ppm. Since the boron concentration in the RWST is 2300 - 2500 ppm and the amount of water that would be injected following a LOCA would be about 3,200,000 lbs, the additional delay in RWST switchover has no impact on the Summer Station long-term cooling analysis.

Hot Leg Switchover to Prevent Boron Precipitation

Post-LOCA hot leg recirculation switchover time is determined to ensure that no boron precipitates in the reactor vessel following boiling in the core. The time to initiate hot leg switchover is not affected by the delay in RWST switchover; therefore, the proposed TS change does not affect these analyses.

Steam Generator Tube Rupture

For the SGTR, primary to secondary break flow was assumed to terminate 30 minutes after event initiation, and operator action to cool down the plant was not modeled. With no RCS cooldown, sufficient shutdown margin is assumed to be available initially, and long-term shutdown margin is assumed to be maintained with borated water. The increase in delay time will have no impact on these assumptions.

Non-Loca Transients

Steam Line Rupture Accident

The only non-LOCA event affected by the increased time delay is the steam line break. This is the only event which relies on short-term boration. Based on the current steam line break analysis, the additional delay time is acceptable since the delay in the availability of borated water occurs early in the transient when RCS pressure is still high and SI flow rates are low. A sensitivity study conducted by Westinghouse shows that delays of the magnitude under consideration will result in maximum increases of 0.2% in reactor power, 0.6°F in RCS temperature, and 10 psi in pressure. A Summer Station plant specific analysis has shown that there is sufficient design margin to accommodate these small increases; therefore, the analyses in the FSAR remain valid.

Sensitivity studies on the steam line break superheated mass/energy release outside containment show that these releases are not sensitive to large changes in SI flow. The additional time delay is small when compared to the large change in total SI flow; therefore, the impact of the delay on superheated mass/energy releases outside containment is insignificant.

The proposed additions of 10 to 15 seconds to the response times given in the Table 3.3-5 of the TS in conjunction with the modifications to the notation at the back of the Table reflect both the opening of the RWST outlet valves and the closing of the VCT outlet valves. As the above evaluation indicates, these changes will not invalidate the analyses or subsequent conclusions in the FSAR for any LOCA or non-LOCA transient; therefore, the staff has found the proposed TS changes 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 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 (57 FR 24679). 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: December 15, 1992