July 22, 1987

Docket No. 50-395

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

Dear Mr. Nauman:

SUBJECT: ISSUANCE OF AMENDMENT NO. 67 TO FACILITY OPERATING LICENSE NO. NPF-12 REGARDING ENGINEERED SAFETY FEATURES RESPONSE TIMES - VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1 (TAC NO. 64150)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 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 December 12, 1986, as supplemented April 9, 1987.

The amendment clarifies the Service Water System and Reactor Building Cooling Unit response times.

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

Sincerely,

/S/

Jon B. Hopkins, Project Manager
Project Directorate II-1
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 67 to NPF-12
2. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrence

LA:PD21:DRPR*	PM:PD21:DRPR*	OGC-B*	D:PD21:DRPR*
PAnderson	JHopkins/dsf	MYoung	EAdensam
7/14/87	7/14/87	7/16/87	7/14/87

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2600 Bull Street
Columbia, South Carolina 29201

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. 67
License No. NPF-12

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

   A. The application for amendment filed by South Carolina Electric & Gas Company and South Carolina Public Service Authority (the licensees), dated December 12, 1986, as supplemented April 9, 1987, 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. 67 , 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

B. C. Buckley for

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 22, 1987

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 67 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 identified by amendment number and contain vertical lines indicating the areas of change. Corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
B 3/4 3-1b	B 3/4 3-1b
--	B 3/4 3-1c
B 3/4 6-4	B 3/4 6-4

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)	≤ 12.0[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)	≤ 12.0[2]/22.0[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	$\leq 12.0^{(2)}/22.0^{(3)}$
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 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	≤ 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	≤ 12.0
7. Reactor Building Pressure-High-2	
a. Steam Line Isolation	≤ 9.0

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

Several automatic logic functions included in this specification are not necessary for Engineered Safety Feature System actuation but their functional capability at the specified setpoints enhances the overall reliability of the Engineered Safety Features functions. These automatic actuation systems are purge and exhaust isolation from high containment radioactivity, turbine trip and feedwater isolation from steam generator high-high water level, initiation of emergency feedwater on a trip of the main feedwater pumps, automatic transfer of the suctions of the emergency feedwater pumps to service water on low suction pressure, and automatic opening of the containment recirculation sump suction valves for the RHR and spray pumps on low-low refueling water storage tank level.

The service water response time includes: 1) the start of the service water pumps and, 2) the service water pumps discharge valves (3116A,B,C-SW) stroking to the fully opened position. This condition of the valves assures that flow will become established through the component cooling water heat exchanger, diesel generator coolers, HVAC chiller, and to the suction of the service water booster pumps when these components are placed in-service. Prior to this time, the flow is rapidly approaching required flow and sufficient pressure is developed as valves finish their stroke. Each of the above-listed components will be starting to perform their accident mitigation function, either directly or indirectly depending upon the use of the component, and will be operational within the service water response time of 71.5/81.5 seconds[1/]. Only the service water booster pumps have a direct impact on the accident analysis via the RBCUs' heat removal capability as discussed below.

The RBCU response time includes: 1) the start of the RBCU fan and the service water booster pumps and, 2) all the service water valves which must be driven to the fully opened or fully closed position. This condition of the valves allows the flow to become fully established through the RBCU. Prior to this time, the flow is rapidly approaching required flow as the

1/Total time is 1.5 second instrument response after setpoint is reached, plus 10 seconds diesel generator start, plus 10 seconds to reach service water pump start and begin 3116-SW opening via Engineered Safety Features Loading Sequencer, plus 60 seconds stroke time for 3116-SW. During this total time, the service water pumps start and the service water pump discharge valve begins to open at 11.5 seconds and the pump discharge valve is fully open at 71.5 seconds without a diesel generator start required and 21.5 seconds and 81.5 seconds including a diesel generator start.

valves finish their stroke. Although the RBCU would be removing heat throughout the Engineered Safety Features response time, the accident analysis does not assume heat removal capability from 0 to 71.5 seconds [2/] because the industrial cooling water system is not completely isolated until 71.5 seconds. A linear ramp increase from 95% full heat removal capability to 100% full heat removal capability is assumed by the accident analysis to start at 71.5 seconds and end at 86.5 seconds[3/]. Full heat removal capability is assumed at 86.5 seconds based on the position of the valve 3107-SW.

2/Total time is 1.5 second instrument response after setpoint is reached, plus 10 second diesel start plus 60 seconds* for valves to isolate industrial cooling water system.

3/Total time is 1.5 second instrument response after setpoint is reached, plus 10 second diesel generator start plus 75 seconds to stroke valves 3107A,B-SW.

*During this time period, the Engineered Safety Features Loading Sequencer starts the RBCU fans at 25 seconds and service water booster pumps at 30 seconds after the valves begin to stroke.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the spray additive system ensures that sufficient NaOH is added to the reactor building spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 7.8 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

3/4.6.2.3 REACTOR BUILDING COOLING SYSTEM

The OPERABILITY of the reactor building cooling system ensures that 1) the reactor building air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the reactor building spray systems during post-LOCA conditions.

The reactor building cooling system and the reactor building spray system are redundant to each other in providing post accident cooling of the reactor building atmosphere. As a result of this redundancy in cooling capability, the allowable out of service time requirements for the reactor building cooling system have been appropriately adjusted. However, the allowable out of service time requirements for the reactor building spray system have been maintained consistent with that assigned other inoperable ESF equipment since the reactor building spray system also provides a mechanism for removing iodine from the reactor building atmosphere.

The accident analysis requires the service water booster pump to be passing 4,000 gpm to both RBCU's within 86.5 seconds. This time encompasses the driving of all necessary service water valves to the correct positions, i.e., fully opened or fully closed. Reference Technical Specification Bases B3/4.3.1 and B3/4.3.2 for additional details.

3/4.6.3 PARTICULATE IODINE CLEANUP SYSTEM

The OPERABILITY of the containment filter trains ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 67 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 December 12, 1986, South Carolina Electric & Gas Company submitted a request for changes to the Virgil C. Summer Nuclear Station Technical Specifications (TS).

The amendment would revise Table 3.3-5 "Engineered Safety Features Response Times," and its associated Basis. This change would clarify the Service Water System and Reactor Building Cooling Unit response times.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on January 14, 1987 (52 FR 1557). No comments or requests for hearing were received.

By letter dated April 9, 1987, the licensee submitted additional information which clarified certain aspects of the December 12, 1986 application. Since the additional information did not change the requested TS revision or affect the staff's initial determination, the amendment was not renoticed in the Federal Register.

2.0 EVALUATION

The proposed changes will increase the required response time for the service water system (SWS) and the reactor building cooling units (RBCU) to "as-built" values. The proposed SWS response time is 81.5 seconds (including diesel generator delay) or 71.5 seconds (not including diesel generator delay). The response time of 81.5 seconds includes 1.5 seconds for instrument response, 10 seconds to start the diesel generator, 10 seconds to start the service water pump and 60 seconds to open the SWS pump discharge valves (3116A, B, C-SW).

The main function of the SWS is to deliver water to the following components:

(1) component cooling water/SW heat exchanger;
(2) diesel generator coolers;
(3) heating, ventilation, and air conditioning chiller; and
(4) the suction of the SW booster pumps.

Each of the above listed components will be operational within the SWS response time. The licensee identified that only the service water booster pumps have a direct safety impact on the containment analysis via the RBCU's heat removal capability. This impact was evaluated in conjunction with the effect of the RBCU response time as discussed below.

The proposed RBCU response time is 86.5 seconds (including generator delay) or 76.5 seconds (not including diesel generator delay). The response time of 86.5 seconds includes 1.5 seconds for instrument response, 10 seconds to start the diesel generator and 75 seconds to open the slowest valves (3707A, B-SW) in the SW line. The licensee performed two containment response analysis cases, i.e., the worst pressure case and the worst temperature case. In the analysis, no flow to the RBCU was assumed until 71.5 seconds because valves 3110A, B-SW would remain completely closed throughout this time period to isolate the SWS from the non-safety related, non-seismic industrial cooling water (CI) system. At 71.5 seconds, the RBCU discharge valves 3107A, B-SW would still be opening and would be passing 3800 gpm of service water. At 86.5 seconds, the valves 317A, B-SW would be fully open and passing 4000 gpm. In the analysis, the licensee assumed a linear ramp increase of RBCU heat removal capability from 95% at 71.5 seconds to 100% at 86.5 seconds.

As indicated in the licensee's letter of April 9, 1987, the analyses used the mass and energy release data that were calculated by the Westinghouse LOFTRAN code, April 1986 revision. This revision of the LOFTRAN code was found acceptable by the staff in its safety evaluation transmitted to Westinghouse by letter dated May 27, 1986. The computer code CONTEMPT LT/26 was used by the licensee to calculate containment pressure and temperature responses. This is consistent with the staff's recommendation in SRP Section 6.2.1.1.A. A peak pressure of 46 psig was calculated which is less than the design pressure of 57 psig with some margin. The peak calculated temperature of 322°F is bounded by the equipment qualification temperature profiles specified in FSAR Figures 3.11-8, 3.11-9 and 3.11-10.

The staff has reviewed the licensee's analyses, including the assumptions, methodologies, and results. Based on the above review, the staff finds that the proposed TS changes to increase the response time of the SWS and RBCU to the specified values have only a minor impact on the containment pressure and temperature responses. The resulting containment pressure and temperature were determined in accordance with SRP Section 6.2.1.1.A and are within the design values specified in the FSAR. The staff, therefore, concludes that the proposed TS changes are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the surveillance requirement of a facility component located within the restricted area, as defined in 10 CFR Part 20. The staff has determined that these amendments involve 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 these amendments involve no significant hazard consideration, and there has been no public comment on such finding. Accordingly, this amendments 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 this amendment.

4.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration which was published in the Federal Register (52 FR 1557) on January 14, 1987, and consulted with the State of Alabama. No public comments or requests for hearing were received and

The Commission has issued a "Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Opportunity for Prior Hearing" which was published in the FEDERAL REGISTER on January 14, 1987 (52 FR 1557) and consulted with the state of South Carolina. No public comments or requests for hearing were received, and the state of South Carolina did not have any comments.

The staff 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, 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:

J. Hopkins, Project Directorate II-1
C. Li, Plant Systems Branch

Dated: July 22, 1987