November 7, 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

SUBJECT: ISSUANCE OF AMENDMENT NO. 118 TO FACILITY OPERATING LICENSE NO. NPF-12 REGARDING ACCIDENT MONITORING INSTRUMENTATION - VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1 (TAC NO. M88546)

Dear Mr. Skolds:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 118 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 (TS) in response to your application dated December 17, 1993.

The change revises TS 3/4.3.3.6, Accident Monitoring Instrumentation, and its associated bases; relocates TS 3/4.6.5.1, Hydrogen Monitors, and TS 3/4.3.3.1, Tables 3.3-6 and 4.3-3, Item 1.c, Reactor Building Area High Range Radiation Monitors, into the Accident Monitoring TS.

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. 118 to NPF-12
2. Safety Evaluation

cc w/enclosures:
See next page

DOCUMENT NAME: G:\SUMMER\SUM88546.AMD *See Previous Concurrence

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

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Senior Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
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Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosures:
1. Amendment No. 118 to NPF-12
2. Safety Evaluation

cc w/enclosures:
See next page

cc:

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AMENDMENT NO. 118 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. 118
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 17, 1993, 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. 118 , 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

Michael L. Boyle

Michael L. Boyle, Acting 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 7, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 118

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
VII	VII
3/4 3-42	3/4 3-42
3/4 3-45	3/4 3-45
3/4 3-56	3/4 3-56
3/4 3-57	3/4 3-57
3/4 3-57a	3/4 3-57a
3/4 3-58	3/4 3-58
3/4 3-59	3/4 3-59
3/4 6-21	3/4 6-21
B 3/4 3-3	B 3/4 3-3
B 3/4 3-3a	B 3/4 3-3a
B 3/4 3-3b	B 3/4 3-3b
B 3/4 3-3c	B 3/4 3-3c
B 3/4 3-3d	B 3/4 3-3d

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Pool Area (RM-G8)	1	*	$\leq$ 15 mR/hr	10^{-1} - 10^{4} mR/hr	25
b. Reactor Building Manipulator Crane Area (RM-G17A or RM-G17B)	1	6	$\leq$ 1 R/hr	1 - 10^{5} mR/hr	28
2. PROCESS MONITORS					
a. Spent Fuel Pool Exhaust - Ventilation System (RM-A6)					
i. Gaseous Activity	1	**	$\leq 1 \times 10^{-5}$ μCi/cc (Kr-85)	10 - 10^{6} cpm	27
ii. Particulate Activity	1	**	N/A	10 - 10^{6} cpm	27
b. Containment					
i. Gaseous Activity - Purge & Exhaust Isolation (RM-A4)	1	6	$\leq$ 2 x background***	10 - 10^{6} cpm	28
ii. Particulate and Gaseous Activity (RM-A2) - RCS Leakage Detection	1	1, 2, 3 & 4	N/A	10 - 10^{6} cpm	26
c. Control Room Isolation (RM-A1)	1	ALL MODES	$\leq$ 2 x background	10 - 10^{6} cpm	29

* With fuel in the storage pool or building

** With irradiated fuel in the storage pool

*** Alarm/trip setpoint will be per the Operational Dose Calculation Manual when purge exhaust operations are in progress.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. AREA MONITORS				
a. Spent Fuel Pool Area (RM-G8)	S	R	M	*
b. Reactor Building Manipulator Crane Area (RM-G17A or RM-G17B)	S	R	M	6
2. PROCESS MONITORS				
a. Spent Fuel Pool Exhaust Area - Ventilation System (RM-A6)				
i. Gaseous Activity	S	R	M	**
ii. Particulate Activity	S	R	M	**
b. Containment				
i. Gaseous Activity - Purge & Exhaust Isolation (RM-A4)	S	R	M	6
ii. Particulate and Gaseous Activity - RCS Leakage Detection (RM-A2)	S	R	M	1, 2, 3 & 4
c. Control Room Isolation (RM-A1)	S	R	M	All MODES
d. Noble Gas Effluent Monitors (High Range)				
i. Main Plant Vent (RM-A13)	S	R	M	1, 2, 3 & 4
ii. Main Steam Lines (RM-G19A, B, C)	S	R	M	1, 2, 3 & 4
iii. Reactor Building Purge Supply & Exhaust System (RM-A14)	S	R	M	1, 2, 3 & 4

* With fuel in the storage pool or building

** With irradiated fuel in the storage pool

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown on Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 30 days or submit a Special Report within the following 14 days from the time the action is required. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels to operable status.

b.1 With the number of OPERABLE Reactor Building radiation monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

i) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

ii) Submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

b.2 With the number of Hydrogen monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, restore at least one monitor to operable status within 72 hours or be in at least HOT STANDBY within the next 6 hours, and in HOT SHUTDOWNwithin the next 12 hours.

b.3 With the number of OPERABLE accident monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, either restore the inoperable channels to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performing a monthly CHANNEL CHECK and a CHANNEL CALIBRATION every refueling outage. The Reactor Building Radiation Level Instrumentation CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for the range decades above 10R/hr and a single point calibration of the detector below 10R/hr with an installed or portable gamma source.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1.	Reactor Building Pressure - Narrow Range Instrument Loop/Indicator: Channel D IPT-951/IPI-951 Channel B IPT-952/IPI-952	2	1
2.	Reactor Building Pressure - Wide Range Instrument Loop/Indicator: Channel D IPT-954A/IPI-954A Channel E IPT-954B/IPI-954B	2	1
3.	Reactor Building Radiation Level - High Range Instrument Loop/Indicator: Channel A RMG-18 Channel B RMG-7	2	1
4.	Reactor Building Hydrogen Concentration Instrument Loop/Indicator: Channel A IAE-8263A/ICI-8257 Channel B IAE-8263B/ICI-8258	2	1
5.	Reactor Building/RHR Sump Level Instrument Loop/Indicator: Channel A ILT-1969/ILI-1969 Channel B ILT-1970/ILI-1970	2	1
6.	Reactor Coolant Outlet Temperature - T_{Hot}- Wide Range Instrument Loop/Indicator: Channel A ITE-413/ITI-413 Channel A ITE-423/ITI-423 Channel E ITE-433/ITR-413	2	1
7.	Reactor Coolant Inlet Temperature - T_{Cold} - Wide Range Instrument/Loop Indicator: Channel E ITE-410/ITI-410 Channel E ITE-420/ITI-420 Channel E ITE-430/ITR-410	2	1

TABLE 3.3-10 (continued)

ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
8.	Reactor Coolant Pressure - Wide Range Instrument Loop/Indicator: Channel E IPT-402/IPI-402 Channel A IPT-403/IPI-403	2	1
9.	Pressurizer Water Level Instrument Loop/Indicator: Channel A ILT-459/ILI-459A Channel D ILT-460/ILI-460 Channel B ILT-461/ILI-461	2	1
10.	Reactor Coolant System Subcooling Margin Instrument Loop/Indicator: Channel A ITM-499A Channel B ITM-499B	2	1
11.	Reactor Vessel Level Instrument Loop/Indicator: Channel A ILT-1311/ILI-1311, ILT-1312/ILI-1312 Channel B ILT-1321/ILI-1321, ILT-1322/ILI-1322	2	1
12.	Core Exit Temperature Instrument Loop/Indicator: Channel A: ITEs 2, 4, 9, 12, 13, 15, 19, 21, 22, 23, 24, 25, 26, 27, 28, 29, 31, 32, 33, 35, 39, 41, 42, 45, 46, 47 Channel B: ITEs 1, 3, 5, 6, 7, 8, 10, 11, 14, 16, 17, 18, 20, 30, 34, 36, 37, 38, 40, 43, 44, 48, 49, 50, 51 (Primary display is the plant computer) (Backup displays are ITM 499 A&B)	4/core quadrant/ channel	2/core quadrant/ channel
13.	Neutron Flux Instrument Loop/Indicator: Channel 1 INI-35 Channel 2 INI-36	2	1

TABLE 3.3-10 (continued)

ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
14.	Steam Line Pressure Instrument Loop/Indicator: SG A IPTs-474, 475, 476/IPIs-474, 475, 476 SG B IPTs-484, 485, 486/IPIs-484, 485, 486 SG C IPTs-494, 495, 496/IPIs-494, 495, 496	2/stm. gen.	1/stm. gen.
15.	Steam Generator Water Level - Wide Range Instrument Loop/Indicator: SG A ILT-477/ILI-477 SG B ILT-487/ILI-487 SG C ILT-497/ILI-497	1/stm. gen.	1/stm. gen.
16.	Steam Generator Water Level - Narrow Range Instrument Loop/Indicator: SG A ILTs 474, 475, 476/ILIs 474, 475, 476 SG B ILTs 484, 485, 486/ILIs 484, 485, 486 SG C ILTs 494, 495, 496/ILIs 494, 495, 496	2/stm. gen.	1/stm. gen.
17.	Emergency Feedwater Flow Instrument Loop/Indicator: Channel A SG A IFT-3561/IFI-3561 SG B IFT-3571/IFI-3571 SG C IFT-3581/IFI-3581 Channel B SG A IFT-3561A/IFI-3561B SG B IFT-3571A/IFI-3571B SG C IFT-3581A/IFI-3581B	2/stm. gen.	1/stm. gen.
18.	Refueling Water Storage Tank Level Instrument Loop/Indicator: Channel A ILT-990/ILI-990 Channel B ILT-992/ILI-992	2	1

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3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The PAM Instrumentation LCO provides OPERABILITY requirements for Regulatory Guide 1.97 Type A monitors, which provide information required to perform certain manual actions specified in the Emergency Operating Procedures. These manual actions ensure that a system can accomplish its safety function and are credited in the safety analyses. Additionally, this LCO addresses Regulatory Guide 1.97 instruments that have been designated Category I, non-Type A.

The OPERABILITY of the PAM instrumentation ensures there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident.

LCO 3.3.3.6 requires two OPERABLE channels for most Functions. Two OPERABLE channels ensure no single failure prevents operators from getting the information necessary for them to determine the safety status of the unit, and to bring the unit to and maintain it in a safe condition following an accident.

Furthermore, OPERABILITY of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information.

ACCIDENT MONITORING INSTRUMENTATION (Continued)

Type A and Category I variables are required to meet Regulatory Guide 1.97 Category I design and qualification requirements for seismic and environmental qualification, single failure criterion, utilization of emergency standby power, immediately accessible display, continuous readout, and recording of display.

The following are discussions of specified instrument functions listed in Table 3.3-10.

1. & 2. Reactor Building Pressure

Reactor Building Pressure is provided for verification of RCS and containment OPERABILITY. Reactor Building Pressure is used to verify closure of main steam isolation valves (MSIVs), and containment spray Phase B isolation. Other manual actions based on Reactor Building Pressure include: stopping the RCPs, stopping containment spray pumps, and starting RHR pumps. Reactor Building Pressure indications are also required to calculate reactor vessel vent times.

3. Reactor Building Radiation Level

Reactor Building Radiation Level is provided to monitor the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans. Reactor Building Radiation Level is used to determine if a high energy line break (HELB) has occurred and whether the event is inside or outside of containment.

4. Reactor Building Hydrogen Concentration

Reactor Building Hydrogen Concentration Monitors are provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. This variable is also important in verifying the adequacy of mitigating actions. Reactor Building Hydrogen concentration is used by the operator to calculate reactor vessel vent time and is monitored as a criterion for continuing vessel venting when attempting to collapse voids.

5. Reactor Building/RHR Sump Water Level (Wide Range)

Reactor Building/RHR Sump Water Level is provided for verification and long term surveillance of RCS integrity. Reactor Building/RHR Sump Water Level is used to determine: containment sump level accident diagnosis; when to begin the recirculation procedure; and whether to terminate SI, if still in progress.

6. & 7. Reactor Coolant System (RCS) Hot and Cold Leg Temperatures

RCS Hot and Cold Leg Temperatures variables provide verification of core cooling and long term RCS surveillance. In addition, RCS cold leg temperature is used in conjunction with RCS hot leg temperature to verify the unit conditions necessary to establish natural circulation in the RCS.

ACCIDENT MONITORING INSTRUMENTATION (Continued)

8. Reactor Coolant Pressure

RCS pressure provides verification of core cooling and RCS integrity long term surveillance. RCS pressure is used to verify delivery of SI flow to RCS from at least one train when the RCS pressure is below the pump shutoff head. RCS pressure is also used to verify closure of manually closed spray line valves and pressurizer power operated relief valves (PORVs). RCS pressure can be used: to determine whether to terminate actuated SI or to reinitiate stopped SI; to determine when to reset SI and shut off low head SI; to manually restart low head SI; as reactor coolant pump (RCP) trip criteria; and to make a determination on the nature of the accident in progress and where to go next in the procedure. RCS pressure is also related to three decisions about depressurization. They are: to determine whether to proceed with primary system depressurization; to verify termination of depressurization; and to determine whether to close accumulator isolation valves during a controlled cooldown/depressurization. A final use of RCS pressure is to determine whether to operate the pressurizer heaters. RCS pressure is used by the operator to monitor the cooldown of the RCS following a steam generator tube rupture (SGTR) or small break LOCA. Operator actions to maintain a controlled cooldown, such as adjusting steam generator (SG) pressure or level, would use this indication. Furthermore, RCS pressure is one factor that may be used in decisions to terminate RCP operation.

9. Pressurizer Level

Pressurizer Level is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Knowledge of pressurizer water level is also used to verify the unit conditions necessary to establish natural circulation in the RCS and to verify that the unit is maintained in a safe shutdown condition.

10. Reactor Coolant System Subcooling Margin

RCS hot and cold leg temperatures are used to determine RCS subcooling margin. RCS subcooling margin will allow termination of safety injection (SI), if still in progress, or reinitiation of SI if it has been stopped. RCS subcooling margin is also used for unit stabilization and cooldown control.

11. Reactor Vessel Level

Reactor Vessel Level is provided for verification and long term surveillance of core cooling. It is also used for accident diagnosis and to determine reactor coolant inventory adequacy.

Reactor Vessel Level Monitoring provides direct measurement of collapsed liquid level above the fuel alignment plate. Collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. Measurement of collapsed water level is selected because it is a direct indication of the water inventory.

ACCIDENT MONITORING INSTRUMENTATION (Continued)

12. Core Exit Temperature

Core Exit Temperature is provided for verification and long term surveillance of core cooling. Core Exit Temperature is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Core Exit Temperature is also used for unit stabilization and cooldown control.

Two OPERABLE channels of Core Exit Temperature, in each quadrant, provide indication of radial distribution of coolant temperature rise across representative regions of the core. Power distribution symmetry was considered in determining the specific number and locations provided for diagnosis of local core problems. Two randomly selected thermocouples are not sufficient to meet the two thermocouples per channel requirement in any quadrant. The two thermocouples in each channel must meet the additional requirement that one is located near the center of the core and the other near the core perimeter, such that the pair of Core Exit Temperatures indicate the radial temperature gradient across their core quadrant. Two sets of two thermocouples ensure a single failure will not disable the ability to determine the radial temperature gradient.

13. Neutron Flux

Neutron Flux indication is provided to verify reactor shutdown. Neutron flux is used for accident diagnosis, verification of subcriticality, and diagnosis of positive reactivity insertion.

14. Steam Line Pressure

Steam Line Pressure is used to: identify and isolate a faulted steam generator; maintain an adequate reactor heat sink; verify that EFW to steam generator associated with pipe rupture is isolated; and monitor secondary side steam pressure to: verify operation of pressure control steam dump system, monitor RCS cooldown rate, and maintain plant in cold shutdown condition.

ACCIDENT MONITORING INSTRUMENTATION (Continued)

15. & 16. Steam Generator Water Level

SG Water Level is provided to monitor operation of decay heat removal via the SGs. Temperature compensation of this indication is performed manually by the operator. Redundant monitoring capability is provided by two trains of instrumentation. The uncompensated level signal is input to the unit computer, a control room indicator, and the Emergency Feedwater Control System.

SG Water Level is used to: identify the faulted SG following a tube rupture; verify that the intact SGs are an adequate heat sink for the reactor; determine the nature of the accident in progress (e.g., verify an SGTR); and verify unit conditions for termination of SI during secondary unit HELBs outside containment. Operator action is based on control room indication of SG level. The RCS response during a design basis small break LOCA depends on the break size. For a certain range of break sizes, SG-condenser mode of heat transfer is necessary to remove decay heat. Operator action is required to manually raise and control SG level to establish heat transfer. Operator action is initiated on a loss of subcooled margin. Feedwater flow is increased until the indicated level reaches its setpoint.

17. Emergency Feedwater Flow

EFW Flow is provided to monitor operation of decay heat removal via the SGs. EFW Flow to each SG is determined from a differential pressure measurement calibrated for a range of 0 gpm to 1000 gpm. Redundant monitoring capability is provided by two independent trains of instrumentation for each SG. Each differential pressure transmitter provides an input to a control room indicator and the unit computer. Since the primary indication used by the operator during an accident is the control room indicator, the PAM specification deals specifically with this portion of the instrument channel. EFW flow is used three ways: to verify delivery of EFW flow to the SGs; to determine whether to terminate SI if still in progress, in conjunction with SG water level (narrow range); and to regulate EFW flow so that the SG tubes remain covered. Operator action is required to throttle EFW flow during an SLB accident to prevent the EFW pumps from operating in runout conditions. EFW flow is also used by the operator to verify that the EFW System is delivering the correct flow to each SG. However, the Primary indication used by the operator to ensure an adequate inventory is SG level.

18. Refueling Water Storage Tank (RWST) Level

RWST Level is provided to ensure water supply for ECCS. RWST low level indications are used by the operator as the basis for aligning ECCS suction to the containment sump and stopping all pumps taking suction from the RWST on low-low level.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 118TO 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 17, 1993, as supplemented July 22, 1994, South Carolina Electric & Gas Company (the licensee) submitted a request for changes to the Virgil C. Summer Nuclear Station, Unit No. 1, Technical Specifications (TS). The proposed changes would:

(1) Delete TS 3.6.5.1, "Hydrogen Monitors," and its associated Surveillance Requirement and move the hydrogen monitor requirements to the Accident Monitoring Instrumentation TS,

(2) Extend mode applicability for the hydrogen monitors to include mode 3,

(3) Modify TS Table 3.3-10, "Accident Monitoring Instrumentation," to include Type A variables and Category 1 instrumentation in this table and to delete variables that are not Type A and instrumentation that is not Category 1 from the table,

(4) Delete Table 4.3-7, "Accident Monitoring Instrumentation Surveillance Requirements," and move the requirements imposed by this table to Surveillance Requirement 4.3.3.6,

(5) Increase the allowable time for which the number of channels of available accident monitoring instrumentation (with the exception of the reactor building hydrogen concentration monitor) can be less than the minimum channels operable requirement of Table 3.3-10 from 48 hours to 7 days and increase indefinitely the allowable time for which the number of available channels of accident monitoring instrumentation can be less than the Required Number of Channels requirement of Table 3.3-10,

(6) Remove the TS and associated Surveillance Requirement for reactor building area high range monitors RM-G7 and RM-G18 in mode 4.

The July 22, 1994, submittal contained a correction and did not change the staff's finding of no significant hazards considerations.

2.0 EVALUATION

Regulatory Guide 1.97 defines Type A variables as "...those variables that provide primary information needed to permit the control room operating personnel to take the specified manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accident events." The Regulatory Guide does not identify specific variables as being Type A as these may vary from plant to plant. Regulatory Guide 1.97 goes on to provide design and qualification criteria for the instrumentation used to measure various variables; Category 1 provides the most stringent requirements. Category 1 instrumentation is usually reserved for monitoring key variables, hereinafter referred to as Category 1 variables.

On April 15, 1985, South Carolina Electric & Gas Company provided its "Summary Report on Regulatory Guide 1.97." This report listed the specific Type A Category 1 accident monitoring variables for Summer Station. In letters dated November 13, 1987, and July 27, 1988, the staff responded to and approved the licensee's list of Type A and Category 1 variables.

The bases for the NUREG-1431 Standard Technical Specifications (STS) state that it is appropriate to have TS for all Type A and Category 1 variable instrumentation. The licensee's amendment request would ensure that the instrumentation for all variables identified as either Type A or Category 1 are listed in Table 3.3-10 of the TS. The amendment would remove variables that are neither Type A nor Category 1 from the TS. The proposed TS differ from the STS in that neither Condensate Storage Tank level nor Containment Isolation Valve position are considered either Type A or Category 1 for Summer Station. The proposed changes are, however, consistent with the bases of the STS and are acceptable.

The requested amendment would move the TS for hydrogen monitors to the accident monitoring TS and expand mode applicability to include mode 3. The amendment would also remove Table 4.3-7 from the TS and move the associated requirements to Surveillance Requirement 4.3.3.6. Table 4.3-7 lists the specific surveillances that are to be conducted on each accident monitoring instrument. Surveillance Requirement 4.3.3.6 would change neither the type nor reduce the frequency of any surveillance. These changes do not relax current requirements and are, therefore, acceptable.

The requested amendment would extend from 48 hours to 7 days the time that plant operation is allowed with the number of channels of instrumentation listed in Table 3.3-10 (with the exception of hydrogen monitors) less than the "Minimum Channels Operable" requirement. The outage time of 7 days is based on the relatively low probability of an event requiring accident monitoring instrument operation and the availability of alternate means to obtain the required information. This change is consistent with the STS and is acceptable.

The requested amendment would extend indefinitely the time that plant operation is allowed with the number of channels of instrumentation less than the "Required No. of Channels" condition of Table 3.3-10. It would require the licensee to submit a special report outlining the actions that have been taken to measure the variable in question and to provide a plan for a long-term solution. This action is appropriate since alternative actions are identified before loss of functional capability and since the likelihood that the instrumentation will be required is small. This request is consistent with the STS and is, therefore, acceptable.

The proposed amendment would remove the TS associated with Reactor Building Area High Radiation Monitors RM-G7 and RM-G18 in mode 4. These monitors do not initiate any automatic mitigation system and are required only to provide indication that will help operators in mitigating design basis accidents that severely degrade the reactor coolant pressure boundary. Since design basis accidents that severely degrade the reactor coolant pressure boundary are not postulated during mode 4, the staff finds this change 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 and changes the Surveillance Requirements. 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 7699 cite). 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.

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Date: November 7, 1994