May 19, 1986

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
Columbia, South Carolina 29218

Dear Mr. Nauman:

The Commission has issued Amendment No. 49 to Facility Operating License NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1. This amendment is in response to your application dated March 15, 1985.

The amendment modifies the Technical Specifications to reflect administrative changes. The amendment is effective seven days after its date of issuance.

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

Sincerely,

Lester S. Rubenstein, Director
PWR Project Directorate #2
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 49
2. Safety Evaluation

cc w/enclosures:
See next page

LA:PAD#2	PM:PAD#2	D:PAD#2	OELD
DMiller	JHopkins:hc	LRubenstein	MYoung
4/29/86	4/29/86	5/16/86	5/12/86

OELD note: reported revisions to SE & notice. Done. 5/16

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PDR ADOCK 05000395
P PDR

Mr. D. A. Nauman
South Carolina Electric & Gas Company

Virgil C. Summer Nuclear Station

cc:
Mr. William A. Williams, Jr.
Technical Assistant - Nuclear Operations
Santee Cooper
P.O. Box 764 (Mail Code 167)
Columbia, South Carolina 29218

J. B. Knotts, Jr., Esq.
Bishop, Liberman, Cook, Purcell
and Reynolds
1200 17th Street, N.W.
Washington, D. C. 20036

Resident Inspector/Summer NPS
c/o U.S. Nuclear Regulatory Commission
Route 1, Box 64
Jenkinsville, South Carolina 29065

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission,
101 Marietta Street, N.W., Suite 2900
Atlanta, Georgia 30323

Chairman, Fairfield County Council
P.O. Box 293
Winnsboro, South Carolina 29180

Attorney General
Box 11549
Columbia, South Carolina 29211

Mr. Heyward G. Shealy, Chief
Bureau of Radiological Health
South Carolina Department of Health
and Environmental Control
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. 49
License No. NPF-12

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

A. The application for amendment by the South Carolina Electric & Gas Company acting for itself and South Carolina Public Service Authority (the licensees), dated March 15, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

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

C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 49 , are hereby incorporated into this license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective seven days after its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel G McDonald

Acting for Lester S. Rubenstein, Director
PWR Project Directorate #2
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 16, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 49

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. The corresponding overleaf page is also provided to maintain document completeness.

Remove Pages	Insert Pages
X	X
XV	XV
XVII	XVII
3/4 3-16	3/4 3-16
3/4 3-35	3/4 3-35
3/4 3-42	3/4 3-42
3/4 3-45	3/4 3-45
B 3/4 6-2	B 3/4 6-2
6-11	6-11
6-18	6-18

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

BASES

SECTION PAGE

DESIGN FEATURES

ADMINISTRATIVE CONTROLS

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.

*A one time extension of the frequency of response time tests is granted until June 30, 1983 for all tests due to be completed before this date. Surveillance tests for response time will be conducted on or before June 30, 1983.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION, REACTOR TRIP, FEEDWATER ISOLATION, CONTROL ROOM ISOLATION, START DIESEL GENERATORS, CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER.					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Reactor Building Pressure - High-1	3	2	2	1, 2, 3	15*
d. Pressurizer Pressure - Low	3	2	2	1, 2, 3#	15*
e. Differential Pressure Between Steam Lines - High	3/steam line	2/steam line twice and 1/3 steam lines	2/steam line	1, 2, 3	15*

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. SAFETY INJECTION, REACTOR TRIP FEEDWATER ISOLATION, CONTROL ROOM ISOLATION START DIESEL GENERATORS, CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Reactor Building Pressure-High-1	S	R	M	N.A.	N.A	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure--Low	S	R	M	N.A	N.A.	N.A.	N.A.	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. Steam Line Pressure Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. REACTOR BUILDING SPRAY								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Reactor Building Pressure-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. CONTAINMENT ISOLATION								
a. Phase "A" Isolation								
1) Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Safety Injection	See 1 above for all Safety Injection Surveillance Requirements							
3) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
b. Phase "B" Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
2) Reactor Building Pressure--High-High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Purge and Exhaust Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
2) Containment Radio-activity-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
3) Safety Injection	See 1 above for all Safety Injection Surveillance Requirements.							

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-3.

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
c. Reactor Building Area					
i. High Range RM-G7 and High Range RM-G18	2	1, 2, 3 & 4	N/A	10 - 10^{7} R/hr 1 - 10^{7} R/hr	30
2. PROCESS MONITORS					
a. Spent Fuel Pool Exhaust - Ventilation System (RM-A6)					
i. Gaseous Activity	1	**	$\leq$ 1 x 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
c. Reactor Building Area				
i. High Range (RM-G7)	S	R***	M	1, 2, 3 & 4
ii. High Range (RM-G18)	S	R***	M	1, 2, 3 & 4
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

***Channel Calibration will consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source.

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The movable incore detection system shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of 2 detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the movable incore detection system is used for:

a. Recalibration of the excore neutron flux detection system,

b. Monitoring the QUADRANT POWER TILT RATIO using a full-core flux map per Specification 4.2.4.2, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and F_{xy}

ACTION:

With the movable incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The movable incore detection system shall be demonstrated OPERABLE at least once per 24 hours, by normalizing each detector output when required for:

a. Recalibration of the excore neutron flux detection system, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$, and F_{xy}.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates (including those used in demonstrating a 30 day water seal) ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or 0.75 L_t, as applicable, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50.

3/4.6.1.3 REACTOR BUILDING AIR LOCKS

The limitations on closure and leak rate for the reactor building air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on reactor building internal pressure ensure that 1) the reactor building structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.5 psig and 2) the reactor building peak pressure does not exceed the design pressure of 57 psig during steam line break conditions.

The maximum peak pressure expected to be obtained from a steam line break event is 47.1 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 47.1 psig which is less than design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on reactor building average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a steam line break accident.

3/4.6.1.6 REACTOR BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 47.1 psig in the event of a steam line break accident. The measurement of containment tendon lift off force, the tensile tests of the tendon wires, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.

The tendon lift off forces are evaluated to ensure that 1) the rate of tendon force loss is within predicted limits, and 2) a minimum required prestress level exists in the containment. In order to assess the rate of force loss, the lift off force for a tendon is compared with the force predicted for the tendon times a reduction factor of 0.95. This resulting force is referred to as the 95% Base Value. The predicted tendon force is equal to the original stressing force minus losses due to elastic shortening of the tendon, stress relaxation of the tendon wires, and creep and shrinkage of the concrete. The 5% reduction on the predicted force is intended to compensate for both uncertainties in the prediction techniques for the losses and for inaccuracies in the lift-off force measurements.

RECORDS

6.5.2.10 Records of NSRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NSRC meeting shall be prepared, approved and forwarded to the Vice President, Nuclear Operations within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Vice President, Nuclear Operations within 14 days following completion of the review.

c. Audit summary reports encompassed by Section 6.5.2.8 above, shall be forwarded to the NSRC and to the Vice President, Nuclear Operations. Full audits shall be forwarded to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.3.1 Activities which affect nuclear safety shall be conducted as follows:

a. Procedures required by Technical Specification 6.8 and other procedures which affect plant nuclear safety, and changes thereto, shall be prepared, reviewed and approved. Each such procedure or procedure change shall be reviewed by an individual/group other than the individual/group which prepared the procedure or procedure change, but who may be from the same organization as the individual/group which prepared the procedure or procedure change. Procedures other than Administrative Procedures will be approved as delineated in writing by the Director, Nuclear Plant Operations. The Director, Nuclear Plant Operations will approve administrative procedures, security implementing procedures and emergency plan implementing procedures. Temporary approval to procedures which clearly do not change the intent of the approved procedures can be made by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License. For changes to procedures which may involve a change in intent of the approved procedures, the person authorized above to approve the procedure shall approve the change.

b. Proposed changes or modifications to plant nuclear safety-related structures, systems and components shall be reviewed as designated by the Director, Nuclear Plant Operations. Each such modification shall be designed as authorized by Technical Services and shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modifications. Implementation of modifications to plant nuclear safety-related structures, systems and components shall be concurred in by the Director, Nuclear Plant Operations.

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from site to unrestricted areas of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the Process Control Program (PCP) made during the reporting period.

MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORV's or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted as set forth in 6.5 above.

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.11 The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) shall be provided to the Regional Administrator of the Regional Office of Inspection and Enforcement, with a copy to the Director, Nuclear Reactor Regulation, Attention Chief of the Core Performance Branch, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555 for all core planes containing bank "D" control rods and all unrodded core planes at least 60 days prior to cycle initial criticality. In the event that the limit would be submitted at some other time during core life, it shall be submitted 60 days prior to the date the limit would become effective unless otherwise exempted by the Commission.

Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Office of Inspection and Enforcement Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 49 TO FACILITY OPERATING LICENSE NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

I. INTRODUCTION

South Carolina Electric and Gas Company (SCE&G), the licensee of the Virgil C. Summer Nuclear Plant, in a letter written to this Commission, dated March 15, 1985, proposed changes to the Operating License (NPF-12). These were administrative changes to the Technical Specifications presented as six attachments. As discussed below, attachments 1, 2, 3, 4, 5, and 6 are acceptable.

II. EVALUATION

MONTHLY OPERATING REPORT
(Section 6.9.10)

Discussion and Evaluation

SCE&G proposed to change the addressee of the Monthly Operating Report to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission. The proposed change is consistent with current NRC reporting requirements and is purely administrative in nature.

Conclusion

The proposed change is accepted as submitted and is incorporated into the Technical Specifications.

FACILITY ORGANIZATION
(Section 6.5.3)

Discussion and Evaluation

SCE&G proposed to change the organizational title of the Nuclear Engineering Department to the Technical Services Department. This accurately reflects the new title of the department. This proposed change is purely administrative in nature and does not effect any of the functions currently performed by the department.

Conclusion

The proposed change is accepted as submitted and is incorporated into the Technical Specifications.

SPENT FUEL ASSEMBLY STORAGE
(Section 3/4.9-12)

Discussion and Evaluation

SCE&G proposed adding Section 3/4.9-12 to the Indices for both the "Limiting Conditions for Operation and Surveillance" and the "Bases" portions of the Technical Specifications. This addition is necessitated due to changes approved in Amendment 27. The proposed changes are administrative in nature.

Conclusion

The proposed change is accepted as submitted and is incorporated into the Technical Specifications.

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
(Tables 3.3-3 and 4.3-2)

Discussion and Evaluation

SCE&G proposed to clarify the terminology describing reactor building pressure trip setpoints delineated in Tables 3.3-3 and 4.3-2 of the Technical Specifications. The proposed changes are administrative in nature.

Conclusion

The proposed change is accepted as submitted and is incorporated into the Technical Specifications. Also, we have identified two other terminology inconsistencies which shall be reviewed by the licensee for possible future change. They are the following:

Table 3.3-3, Item 3, Containment Isolation, Item b(2) is termed "Reactor Building Pressure High-3." However, the corresponding item in Table 4.3-2, Item 3.b.(2), is termed "Reactor Building Pressure High-High-High."

Table 3.3-3, Item 4, Steam Line Isolation, Item C is termed "Reactor Building Pressure High-2." However, the corresponding item in Table 4.3-2, Item 4.3, is termed "Reactor Building Pressure High-High."

DESIGN NEGATIVE PRESSURE OF THE REACTOR BUILDING
(Section 3/4.6.1.4)

Discussion and Evaluation

SCE&G proposed to change the value contained in the Bases for the design negative pressure differential of the reactor building with respect to the outside atmosphere. This change is required to incorporate a value that is consistent with FSAR Sections 3.8.1.3.2.1 and 6.2.1.2.2 values. This change is administrative in nature.

Conclusion

The proposed change is acceptable as submitted and is incorporated into the Technical Specifications.

RADIATION MONITORING INSTRUMENTATION
(Tables 3.3-6 and 4.3-3)

Discussion and Evaluation

SCE&G proposed to clarify the required channels currently needed to meet Technical Specification 3.4.6.1. These changes are administrative in nature.

Conclusion

The proposed changes are accepted as submitted and are incorporated into the Technical Specifications.

III. ENVIRONMENTAL CONSIDERATION

This amendment involves changes in recordkeeping, reporting or administrative procedures or requirements. This amendment also involves changes in the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. We have 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 made a proposed determination that the amendment involves no significant hazards considerations and there have been no comments on that proposal. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(9) and (10). 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.

IV. CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (50 FR 31072) on July 31, 1985, and consulted with the state of South Carolina. No public comments were received, and the state of South Carolina did not have any comments.

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:

Michael F. Runyan, Region II
Jon B. Hopkins, PWR Project Directorate #2, Division of PWR Licensing-A

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