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

July 24, 1989

Docket No. 50-395

Mr. O. S. Bradham
Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
P. O. Box 88
Jenkinsville, South Carolina 29065

Dear Mr. Bradham:

SUBJECT: ISSUANCE OF AMENDMENT NO. 79 TO FACILITY OPERATING LICENSE NO. NPF-12 - VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1, REGARDING REMOVAL OF ORGANIZATION CHARTS AND VARIOUS ADMINISTRATIVE CHANGES (TAC NOS. 65778 AND 67340) AND DELETION OF FIRE PROTECTION TECHNICAL SPECIFICATIONS (TAC NO. 65060)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 79 to Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1. The amendment consists of changes to Section 6.0, Administrative Controls, of the Technical Specifications (TS) in response to your application dated July 6, 1987, as supplemented May 16 and July 18, 1988, and an application dated January 6, 1988. In addition, you provided a December 22, 1988 submittal to consolidate and to ensure the accuracy of the desired changes resulting from the previous submittals. This consolidation is administrative in nature and does not change the previous proposed no significant hazards considerations determinations which were issued.

The amendment changes the TS by revising: (1) TS 6.1.1, 6.5.1, and 6.5.3 to change the title for the jobs currently titled Director, Nuclear Plant Operations to General Manager, Nuclear Plant Operations (2) TS 6.4 to meet training requirements specified in ANSI 3.1-1981, "Qualification and Training of Personnel for Nuclear Power Plants," and 10 CFR 55.59, "Requalification"; (3) TS 6.2.1 to reflect the recommendations of Generic Letter (GL) 88-06, "Removal of Organization Charts from Technical Specifications Administrative Control Requirements"; (4) deletion of TS Figures 6.2-1 and 6.2-2 consistent with the recommendations of GL 88-06; and (5) TS 6.5-1 and 6.5-3 to reflect new titles and new constituency resulting from the reorganization which occurred in January 1988.

The amendment also changes Section 2.C.(18) of the operating license; deletes TS 3/4.3.3.7, Fire Detection Instrumentation and TS 3/4.7.9 and 3/4.7.10, Fire Suppression Systems and Fire Rated Assemblies, respectively; and revises Section 6.0 involving support to the Fire Protection Program and the fire brigade in response to your application dated March 31, 1987, as supplemented December 21, 1987 and January 31, 1989. The January 31, 1989 submittal provided your commitments to modify existing surveillance test procedures and existing station administrative procedures and did not affect the TS changes or the licensing condition change previously prepared.

DF01 1/1

The amendment requires Commission approval if the change to the fire protection program would adversely affect the capability of the station to achieve and maintain safe shutdown in the event of fire.

This amendment is effective as of its date of issuance and shall be implemented within 90 days of the date of issuance.

Enclosure 1 is the Safety Evaluation for this amendment concerning the removal of the Organization Charts, deletion of the existing TS on fire protection, modification of License Condition 2.C.(18), and various administration changes.

The Notice of Issuance will be included in the Commission's Bi-weekly Federal Register notice.

Sincerely,

Original Signed By:

John J. Hayes, Jr., Project Manager
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 79 to NPF-12
2. Safety Evaluations

cc w/enclosures:
See next page

[SUM AMEND 65778/67340/65060]

OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	:D:PD21:DRPR	:	:	:	:
NAME	:PAnderson	:JHayes:bd	:EAdensam	:	:	:	:
DATE	:06/22/89	:06/22/89	:06/20/89	:	:	:	:

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AMENDMENT NO. 79 TO FACILITY OPERATING LICENSE NO. NPF-12 - SUMMER, UNIT 1

Docket File
NRC PDR
Local PDR
PDII-1 Reading
S. Varga (14E4)
G. Lainas
E. Adensam
P. Anderson
J. Hayes
OGC
D. Hagan (MNBB 3302)
E. Jordan (MNBB 3302)
B. Grimes (9A2)
T. Meeks (8) (P1-137)
W. Jones (P-130A)
J. Calvo
McCracken (8H7)
ACRS (10)
GPA/PA
ARM/LFMB

cc: Licensee/Applicant Service List

DFO1
1/1

Mr. O. S. Bradham
South Carolina Electric & Gas Company

Virgil C. Summer Nuclear Station

cc:

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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. 79
License No. NPF-12

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

   A. The applications for amendment by South Carolina Electric & Gas Company (the licensee), dated July 6, 1987, as supplemented May 16, July 18, and December 22, 1988; dated January 6, 1988; and dated March 31, 1987, as supplemented December 21, 1987 and January 31, 1989 comply 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 paragraphs 2.C.(2) and 2.C.(18) of Facility Operating License No. NPF-12 are 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. 79 , 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.

(18) Fire Protection System (Section 9.5.1, SSER 4)

Virgil C. Summer Nuclear Station shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility, and as approved in the SER (Safety Evaluation Report) dated February 1981 (and Supplements dated January 1982 and August 1982) and Safety Evaluations dated May 22, 1986, November 26, 1986, and July 17, 1987 subject to the following provisions:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of fire.

3. This amendment is effective as of its date of issuance, and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By:

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:

*SEE PREVIOUS CONCURRENCE (OGC concurrence was by C. Barth on 2/13/89 on Fire Protection sections and C. Woodhead on 4/4/89 on Organization Charts and administrative change sections)

OFC	LA:PD21:DRPR	PM:PD21:DRPR	OGC*	D:PD21:DRPR			
			B Woodhead				
NAME	PAnderson	JHayes:bd	See note	EAdensam			
DATE	04/22/89	04/27/89	04/04/89	04/20/89			

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ATTACHMENT TO LICENSE AMENDMENT NO. 79

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
III	III (overleaf)
IV	IV
VIII	VIII
XIV	XIV
3/4 3-61	3/4 3-61
3/4 3-62 through 3/4 3-66d	-
3/4 7-25	3/4 7-25
3/4 7-26 through 3/4 7-36	-
B3/4 3-3	B3/4 3-3 (overleaf)
B3/4 3-4	B3/4 3-4
B3/4 7-6	B3/4 7-6
B3/4 7-7	-
6-1	6-1
6-1a	6-1a
6-2	6-2
6-3	6-3
6-4	6-4
6-5	6-5
6-6	6-6
6-7	6-7
6-8	6-8
6-9	6-9
6-10	6-10
6-11	6-11
6-12	6-12
6-13	6-13
6-14	6-14
6-14a	----
6-15	6-15
6-16	6-16
6-17	6-17
6-18	6-18
6-18a	----
6-19	6-19
6-20	6-20
6-21	6-21
6-22	----
6-23	----

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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Pages 3/4 7-26 through 3/4 7-36 have been deleted.

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Pages 3/4 3-62 through 3/4 3-66d have been deleted.

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

The Engineered Safety Feature Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates turbine trip, closes main feedwater valves on T_{avg} below setpoint, prevents the opening of the main feedwater valves which were closed by a safety injection or high steam generator water level signal, allows safety injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of safety injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates safety injection actuation on low pressurizer pressure. On decreasing pressure, P-11 allows the manual block of safety injection actuation on low pressurizer pressure.

P-12 On increasing primary coolant loop temperature, P-12 automatically reinstates safety injection actuation and steam line isolation on low steam line pressure, and removes a blocking signal from the steam dump system. On decreasing primary coolant loop temperature, P-12 allows the manual block of safety injection actuation and steam line isolation on low steam line pressure and automatically provides a blocking signal to the steam dump system.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the excore neutron flux detection system, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Channel is inoperable.

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 OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

3/4.7.8 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.9 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of 2°F.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The General Manager, Nuclear Plant Operations shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor shall be responsible for unit operations. A management directive to this effect, signed by the Vice President, Nuclear Operations, shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

Offsite and Onsite organizations shall be established for unit operation and corporate management, respectively. The offsite and onsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts, functional descriptions of department responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. The organizational charts will be documented in the FSAR and updated in accordance with 10 CFR 50.71(e).

b. The General Manager, Nuclear Plant Operations, shall be responsible for overall unit safe operation and shall have control over onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President, Nuclear Operations, shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out the health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 UNIT STAFF

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one Licensed Senior Reactor Operator shall be in the Control Room.

c. A health physics technician# shall be on site when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel.

Adequate shift coverage shall be maintained without routine heavy use of overtime. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

c. A break of at least eight hours should be allowed between work periods, including shift turnover time.

d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the General Manager, Nuclear Plant Operations, or his deputy, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Except during extended shutdown periods, controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the General Manager, Nuclear Plant Operations, or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

#The health physics technician composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

SUMMER UNIT 1

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3, & 4	MODES 5 & 6
SS	1	1
CRF	1	None
RO	2	1
AO	2	1
STA	1	None

SS - Shift Supervisor with a Senior Reactor Operators License on Unit 1
CRF - Control Room Supervisor with a Senior Reactor Operators License on Unit 1
RO - Individual with a Reactor Operators License on Unit 1
AO - Auxiliary Operator
STA - Shift Technical Advisor

Except for the Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Control Room Supervisor from the Control Room while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room command function. During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 5 or 6, an individual with a valid RO or SRO license shall be designated to assume the Control Room command function.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 The ISEG shall be composed of a multi-disciplined dedicated onsite group with a minimum assigned complement of five engineers or appropriate specialists.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

6.2.3.4 The ISEG shall make detailed recommendations for procedure revisions, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to the General Manager, Nuclear Safety.

6.2.4 SHIFT TECHNICAL ADVISOR

The Shift Technical Advisor shall provide technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the unit.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 letter to all licensees as clarified in NUREG-0737, Section I.A.2.1, except for the Associate Manager, Health Physics who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, the Shift Technical Advisor who shall meet or exceed the qualifications referred to in Section 2.2.1.b of Enclosure I of the October 30, 1979 NRC letter to all operating nuclear power plants, and the members of the Independent Safety Engineering Group, each of whom shall have a Bachelor of Science degree or registered Professional Engineer and at least two years experience in their field. At least one year experience shall be in the nuclear field.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained and shall meet or exceed the requirements and recommendations of Sections 5.2 and 5.5 of ANSI 3.1-1981 and 10 CFR 55.59, as committed to in Appendix 3A of the Final Safety Analysis Report.

*Not responsible for sign-off function.

6.5 REVIEW AND AUDIT

6.5.1 PLANT SAFETY REVIEW COMMITTEE (PSRC)

FUNCTION

6.5.1.1 The PSRC shall function to advise the General Manager, Nuclear Plant Operations on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Plant Safety Review Committee shall be composed of the:

Chairman:	General Manager, Nuclear Plant Operations or General Manager, Operations and Maintenance
Member:	Manager, Operations
Member:	General Manager, Station Support
Member:	Manager, Maintenance Services
Member:	Manager, Core Engineering and Nuclear Computer Services
Member:	Manager, Chemistry and Health Physics
Member:	Manager, Design Engineering

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PSRC Chairman to serve on a temporary basis; however, no more than two alternates including the Chairman's alternate, if applicable, shall participate as voting members in PSRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PSRC shall meet at least once per calendar month and as convened by the PSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the PSRC necessary for the performance of the PSRC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and three members including alternates.

RESPONSIBILITIES

6.5.1.6 The Plant Safety Review Committee shall review:

a. Station administrative procedures and changes thereto,

b. The safety evaluations for 1) procedures, 2) changes to procedures, equipment or systems, and 3) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question and all programs required by Specification 6.8 and changes thereto.

c. Proposed procedures and changes to procedures, equipment or systems which may involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed tests or experiments which may involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

e. Proposed changes to Technical Specifications or the Operating License.

f. Reports of violations of codes, regulations, orders, Technical Specifications, or Operating License requirements having nuclear safety significance or reports of abnormal degradation of systems designed to contain radioactive material.

g. Reports of significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

h. Review of all REPORTABLE EVENTS.

i. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

j. The plant Security Plan and changes thereto.

k. The Emergency Plan and changes thereto.

l. Items which may constitute a potential nuclear safety hazard as identified during review of facility operations.

m. Investigations or analyses of special subjects as requested by the Chairman of the Nuclear Safety Review Committee.

n. The unexpected offsite release of radioactive material and the report as described in 10 CFR 50.73.

o. Changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL.

p. The plant Fire Protection Program and revisions there to.

AUTHORITY

6.5.1.7 The Plant Safety Review Committee shall:

a. Recommend in writing to the General Manager, Nuclear Plant Operations, approval or disapproval of items considered under 6.5.1.6a, c, d, e, j, and k above.

b. Render determinations in writing to the General Manager, Nuclear Plant Operations, with regard to whether or not each item considered under 6.5.1.6a, c, and d above constitutes an unreviewed safety question.

c. Make recommendations in writing to the General Manager, Nuclear Plant Operations, that actions reviewed under 6.5.1.6(b) above did not constitute an unreviewed safety question.

d. Provide written notification within 24 hours to the Vice President, Nuclear Operations and the Nuclear Safety Review Committee of disagreement between the PSRC and the General Manager, Nuclear Plant Operations however, the General Manager, Nuclear Plant Operations shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The Plant Safety Review Committee shall maintain written minutes of each PSRC meeting that, at a minimum, document the results of all PSRC activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Vice President Nuclear Operations and the Chairman of the Nuclear Safety Review Committee.

6.5.2 NUCLEAR SAFETY REVIEW COMMITTEE (NSRC)

FUNCTION

6.5.2.1 The Nuclear Safety Review Committee shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

COMPOSITION

6.5.2.2 NSRC shall consist of a Chairman and four or more other members appointed by the Executive Vice President, Operations, in consultation with the Vice President, Nuclear Operations. No more than a minority of the members of the NSRC shall have line responsibility for the operation of the unit.

The NSRC members shall hold a Bachelor's degree in an engineering or physical science field or equivalent experience and a minimum of five years of technical experience of which a minimum of three years shall be in one or more of the disciplines of 6.5.2.1a through h. In the aggregate, the membership of the committee shall provide specific practical experience in the majority of the disciplines of 6.5.2.1a through h.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the Executive Vice President, Operations, in consultation with the Vice President, Nuclear Operations; however, no more than two alternates shall participate as voting members in NSRC activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NSRC Chairman to provide expert advice to the NSRC.

MEETING FREQUENCY

6.5.2.5 The NSRC shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 A quorum of the NSRC necessary for the performance of the NSRC review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least 3 NSRC members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.2.7 The NSRC shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems, and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

g. All REPORTABLE EVENTS.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.

i. Reports and meetings minutes of the Plant Safety Review Committee.

AUDITS

6.5.2.8 The NSRC shall have cognizance of the audits listed below. Audits may be performed by using established SCE&G groups such as the ISEG and QA or by outside groups as determined by the NSRC. Audit reports or summaries will be the basis for NSRC action:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire unit staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Emergency Plan and implementing procedures at least once per 12 months.

f. The Security Plan and implementing procedures at least once per 12 months.

g. Any other area of unit operation considered appropriate by the NSRC, Executive Vice President, Operations, or the Vice President, Nuclear Operations.

h. The Fire Protection Program and implementing procedures at least once per 24 months.

i. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified offsite licensee personnel or a qualified outside firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 3 years.

k. The radiological environmental monitoring program and the results thereof, including the performance of activities required by the quality assurance program per R.G. 4.15 Rev. 1, February 1979, at least once per 12 months.

l. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

m. The PROCESS CONTROL PROGRAM and implementing procedures for solidification of radioactive wastes at least once per 24 months.

AUTHORITY

6.5.2.9 The NSRC shall report to and advise the Executive Vice President, Operations on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

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 Executive Vice President, Operations, and 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 Executive Vice President, Operations, and 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, the Executive Vice President, Operations, and 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 General Manager, Nuclear Plant Operations. The General Manager, 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 General Manager, Nuclear Plant Operations. Each such modification shall be designed as authorized by Engineering 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 General Manager, Nuclear Plant Operations.

c. Proposed tests and experiments which affect plant nuclear safety and are not addressed in the Final Safety Analysis Report shall be reviewed by an individual/group other than the individual/group which prepared the proposed test or experiment.

d. Events reportable pursuant to the Technical Specification 6.9 and violations of Technical Specifications shall be investigated and a report prepared which evaluates the event and which provides recommendations to prevent recurrence. Such report shall be approved by the General Manager, Nuclear Plant Operations and forwarded to the Chairman of the Nuclear Safety Review Committee.

e. Individuals responsible for reviews performed in accordance with 6.5.3.1.a, 6.5.3.1.b, 6.5.3.1.c and 6.5.3.1.d shall be members of the plant staff that meet or exceed the qualification requirements of Section 4 of ANSI 18.1, 1971, as previously designated by the General Manager, Nuclear Plant Operations. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the review personnel of the appropriate discipline.

f. Each review will include a determination of whether or not an unreviewed safety question is involved.

RECORDS

6.5.3.2 Records of the above activities shall be provided to the General Manager, Nuclear Plant Operations, PSRC and/or NSRC as necessary for required reviews.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PSRC and the results of this review shall be submitted to the NSRC and the Vice President, Nuclear Operations.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Vice President, Nuclear Operations and the NSRC shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRC and the Vice President, Nuclear Operations within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.
b. Refueling operations.
c. Surveillance and test activities of safety-related equipment.
d. Security Plan.
e. Emergency Plan.
f. Fire Protection Program.
g. PROCESS CONTROL PROGRAM.
h. OFFSITE DOSE CALCULATION MANUAL.
i. Effluent and environmental monitoring program using the guidance in Regulatory Guide 4.15, Revision 1, February 1979.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed prior to implementation as set forth in 6.5 above.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the chemical and volume control, letdown, safety injection, residual heat removal, nuclear sampling, liquid radwaste handling, gas radwaste handling and reactor building spray system. The program shall include the following:

(i) Preventive maintenance and periodic visual inspection requirements, and
(ii) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

(i) Training of personnel,
(ii) Procedures for monitoring, and
(iii) Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(i) Identification of a sampling schedule for the critical variables and control points for these variables,

(ii) Identification of the procedures used to measure the values of the critical variables,

(iii) Identification of process sampling points, including monitoring the discharge of the condensate pumps for evidence of condenser in-leakage.

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for all off-control point chemistry conditions,

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Postaccident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(i) Training personnel,
(ii) Procedures for sampling and analysis,
(iii) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORT

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated manrem exposure according to work and job functions,[1/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

[1/] This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

This report shall also include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.6 Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

6.9.1.7 The annual radiological environmental operating reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The annual radiological environmental operating reports shall include summarized and tabulated results in the format of Regulatory Guide 4.8, December 1975 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.8 Routine radioactive effluent release reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

6.9.1.9 The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The radioactive effluent release report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, and atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of the public due to their activities inside the site boundary (Figures 5.1-3 and 5.1-4) during the report. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. Historical annual average meteorology or meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The radioactive effluent release report to be submitted within 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1.

The radioactive effluents release shall include the following information for each type of solid waste shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms),

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 AFD limits, the W(z) Functions for RAOC and Base Load operation and the value for APL^{ND} (as required) shall be established for each reload core and implemented prior to use.

The methodology used to generate the W(z) functions for RAOC and Base Load Operation and the value for APL^{ND} shall be those previously reviewed and approved by the NRC.* If changes to these methods are deemed necessary they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

A report containing the AFD limits, the W(z) functions for RAOC and Base Load operation and the value for APL^{ND} (as required) shall be provided to the NRC document control desk with copies to the regional administrator and the resident inspector within 30 days of their implementation.

Any information needed to support W(z), $W(z)_{BL}$ and APL^{ND} will be by request from the NRC and need not be included in this report.

*WCAP-10216 P-A "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification."

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.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Unit Operating License.

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities as specified in the NRC's approved SCE&G position on Regulatory Guide 1.88, Rev. 2, October 1976.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PSRC and the NSRC.

l. Records of the service lives of all hydraulic and mechanical snubbers defined in Section 3.7.7 including the date at which the service life commences and associated installation and maintenance records.

m. Records of secondary water sampling and water quality.

n. Records of analysis required by the radiological environmental monitoring program.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP).* Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

*Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they otherwise comply with approved radiation protection procedures for entry into high radiation areas.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit.

6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area. The maximum allowable stay time for individuals in that area shall be established prior to entry. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem** that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP direct or remote (such as use of closed circuit TV cameras) continuous surveillance shall be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the semi-annual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

    a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

    b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

**Measurement made at 18" from source of radioactivity.

c. Documentation of the fact that the change has been reviewed and found acceptable by the PSRC.

2. Shall become effective upon review and acceptance as set forth in 6.5 above.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Monthly Operating Report within 90 days of the date the change(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations, and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the PSRC.

2. Shall become effective upon review and acceptance as set forth in 6.5 above.

6.15 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Liquid, Gaseous and Solid)

6.15.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid).

1. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was reviewed by the Plant Safety Review Committee. The discussion of each change shall contain:

   a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

   b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

   d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differs from those previously predicted in the license application and amendments thereto;

e. An evaluation of the change which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable by the PSRC.

2. Shall become effective upon review and acceptance as set forth in 6.5 above.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 79 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

This Safety Evaluation (SE) is divided into two parts. The first part addresses the portion of the amendment dealing with the deletion of the existing Technical Specifications (TS) on fire protection and the modification to the License Condition 2.C.(18). The second part addresses that portion of the amendment associated with the removal of the organizational charts from the TS and various changes to titles in the Administrative Controls section of the TS.

1.1 Fire Protection

By letter dated March 31, 1987, South Carolina Electric & Gas Company, the licensee, requested an amendment to the Virgil C. Summer Nuclear Station TS. In this request, the licensee proposed to remove all fire protection requirements from the Technical Specifications. In addition, it was proposed that TS 6.5.1.6, Plant Safety Review Committee (PSRC) Responsibilities, be revised to include the fire protection program and its revisions. Finally, it was proposed that a requirement be added to License Condition 2.C.(18) that would allow the licensee to make changes to the fire protection program only if the changes did not adversely affect the capability of the plant to achieve and maintain safe shutdown in the event of a fire. These changes were proposed as a result of the issuance of NRC Generic Letter 86-10, "Implementation of Fire Protection Requirements," and were noticed in the Federal Register on July 29, 1987 (52 FR 28388). On November 9, 1987 the staff provided comments on License Condition 2.C.(18) and on December 21, 1987 the licensee modified their proposed wording of the License Condition. A re-notice was issued in the Federal Register on February 24, 1988 (53 FR 5496). On August 2, 1988 GL 88-12, "Removal of Fire Protection Requirements from Technical Specifications", was issued for the purpose of providing guidance for the preparation of a license amendment request to implement GL 86-10. The licensee made a submittal on January 31, 1989 in which they committed to incorporating the Fire Protection Evaluation Report (FPER) into Chapter 9 of the Final Safety Analysis Report (FSAR). In addition, the licensee committed to retaining the existing Surveillance Test Procedures associated with the fire protection TS and converting them to the Preventative/Fire Test Procedures. The licensee

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committed to taking the limiting conditions for operation in the TS and the remedial action portions of these sections of the fire protection TS, which were being removed from TS, and placing them into an existing Station Administrative Procedure. The licensee also committed to having a level of protection equal to that currently existing in the TS. This submittal did not affect the TS changes previously submitted nor the proposed change to the license condition 2.C.(18).

1.2 Organization Charts

By letter dated July 6, 1987, South Carolina Electric & Gas Company (the licensee) submitted a request for changes to the Virgil C. Summer Nuclear Station, Unit No. 1 (TS). This letter proposed to delete the organization charts from Section 6.0 and to make changes to the training and requalification requirements of the TS. Specifically, the licensee proposed to: (1) delete the reference to the Offsite Organization from Section 6.2.1; (2) delete Figures 6.2-1 and 6.2-2, which reflect the Offsite Organization and the Functional Organization, respectively; (3) renumber Sections 6.2.2 - 6.2.4 as the result of the deletion of Section 6.2.1; and (4) revise Section 6.4 to require the retraining and replacement training program to meet or exceed the requirements of Sections 5.2 and 5.5 of ANSI 3.1-1981, "Qualification and Training of Personnel for Nuclear Power Plants," and 10 CFR 55.59, "Requalification."

The deletion of the Organization Charts from the TS was being handled by the NRC staff utilizing the lead plant concept, with the Shearon Harris Nuclear Power Plant fulfilling the role of the lead plant for this item. In December 1987, the licensee indicated in discussions with the staff that a reorganization was to occur in January 1988. Thus, on January 6, 1988 they proposed to change TS 6.2 in order not to differentiate between the offsite and onsite organizations. At this time, the entire nuclear staff for the licensee was moving onsite. In addition, the licensee provided a revised Figure 6.2-2 to detail the new functional organization. This submittal also proposed to renumber Sections 6.2.2 - 6.2.4, as a result of the proposed change to Section 6.2, and proposed to modify Sections 6.5.1 and 6.5.3, as a result of the change in titles and constituency in makeup of the Plant Safety Review Committee (PSRC) brought about as a result of the reorganization. In their January 6, 1988, letter the licensee stated that they preferred to have the organization charts deleted from the TS. However, they proposed this revised Figure 6.2-2 to be consistent with existing NRC staff policy until completion of the lead plant action.

On January 27, 1988, an amendment was issued for the Shearon Harris Plant to delete the organization charts from the TS. Following that action, on March 22, 1988, the NRC staff issued Generic Letter (GL) 88-06, "Removal of Organizational Charts from Technical Specification Administrative Control Requirements." This GL provided additional guidance on the removal of the organization charts from the TS.

In a May 16, 1988 letter, the licensee followed the guidance of GL 88-06 and proposed to modify TS 6.2.1 by addressing both the offsite and onsite organizations and by deleting the reference to TS Figure 6.2-2, but leaving the remainder of TS 6.2.2 unchanged.

In a July 18, 1988 submittal, the licensee proposed modifying Section 6.1.1 to address another change in title and to delete a statement in that section which stated that other titles referenced, such as Plant Manager, Station Manager, or Manager, Virgil C. Summer Nuclear Station would be synonymous with the new title. The incorporation of that new title was also proposed for TS 6.2.2, 6.5.1.1, 6.5.1.2, 6.5.1.7, 6.5.3.1, and 6.5.3.2. TS 6.4 was still to be changed as noted in the July 6, 1987 submittal.

On December 22, 1988, a submittal was made to consolidate and to ensure the accuracy of the previously proposed changes resulting from the previous submittals and to reflect title changes and to correct omissions. This consolidation was administrative in nature and did not change the staff's previous no significant hazards determination.

2.0 EVALUATION

2.1 Fire Protection

The staff has reviewed the licensee's submittals. The licensee has proposed to delete TS 3.3.3.7, Fire Protection Instrumentation; TS 3/4.7.9, Fire Suppression Systems; 3/4.7.10, Fire Rated Assemblies; and TS 6.2.2.e, Site Fire Brigade Requirements. The requirements of each of these specifications are proposed to be incorporated into plant procedures. Technical Specification 6.5.1.6, Plant Safety Review Commission (PSRC), is also proposed to be revised to make specific reference to the requirement that the PSRC will review the Summer Fire Protection Program and its revisions. The licensee indicated that this reinforces the importance of the Fire Protection Program on plant safety and assures a multi-discipline review by the PSRC of proposed changes to those requirements that are removed from the TS. The licensee has also proposed to modify License Condition 2.C.(18) to state:

> Virgil C. Summer Nuclear Station shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility, and as approved in the SER (Safety Evaluation Report) dated February 1981 (and Supplements dated January 1982 and August 1982) and Safety Evaluations dated May 22, 1986, November 26, 1986, and July 27, 1987 subject to the following provisions:
>
>> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

Since the fire protection program, as described in the Final Safety Analysis Report (FSAR) and as approved in the NRC Safety Evaluation Report and its Supplements and as approved in various NRC Safety Evaluations, will be completely described and controlled through the FSAR/Fire Protection Evaluation Report and Station Administrative procedures, the staff finds that the proposed changes to delete the fire protection requirements from the TS acceptable. The staff believes that the PSRC will exercise proper oversite to ensure that changes which are made to the fire protection program do not adversely affect the ability of the station to achieve and maintain safe shutdown in the event of fire.

2.2 Organizational Charts

The staff has reviewed the licensee's submittals to delete the organizational charts and to modify various sections of the TS as a result of the change in titles resulting from the 1988 reorganization and the new constituency of the PSRC and modifications to training TS (6.4).

The licensee proposed to change the title Director, Nuclear Plant Operations, to General Manager, Nuclear Plant Operations. In addition, this title change will no longer be synonymous with the titles Manager, Virgil C. Summer Nuclear Station, Station Manager, or Plant Manager. The licensee has proposed these changes in TS 6.1.1, 6.2.1, 6.5.1.1, 6.5.1.2, 6.5.1.7, 6.5.3.1, and 6.5.3.2. The licensee has also proposed to change the titles of those members of the PSRC to reflect changes in titles resulting from the new organization and to increase the membership of the PSRC to reflect an appropriate constituency. The PSRC will now consist of the General Manager, Nuclear Plant Operations or General Manager, Operations and Maintanence; Manager, Operations; General Manager, Station Support; Manager, Maintenance Services; Manager, Core Engineering & Nuclear Computer Services; Manager, Chemistry and Health Physics; and Manager, Design Engineering. The PSRC will be chaired by the General Manager, Nuclear Plant Operations, or the General Manager, Operations and Maintenance. The licensee also proposed other changes to Section 6 of the TS to reflect changes in titles and organization resulting from the reorganization. These changes include Group Manager, Technical Services to General Manager, Nuclear Safety (TS 6.2.3.4), and Technical Services to Engineering Services (TS 6.5.3.1). The staff has reviewed these changes and finds that each of these changes proposed, as a result of the reorganization at the Summer Station, to be appropriate and acceptable because the change only involves a change in title and adequate controls will be maintained.

The licensee also requested that TS 6.4 be modified to indicate that the retraining and replacement training program for the unit staff shall be maintained and shall meet or exceed the requirements and recommendations of Section 5.2 and 5.5 of ANSI 3.1-1981 and 10 CFR 55.59 as committed to in Appendix 3A of the FSAR. The proposed changes to TS 6.4 are acceptable because the change only involves a change in title and adequate controls will be maintained.

The licensee proposed to delete the organization charts from TS 6.2 through the deletion of Figures 6.2-1 and 6.2-2. In addition, the licensee proposed to modify TS 6.2.1, Offsite and Onsite Organizations, and TS 6.2.2, Unit Staff, to be consistent with the recommendations of GL 88-06. The staff has reviewed these proposed changes which are consistent with GL 88-06 and are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to a surveillance requirement. 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 this amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

This amendment also changes recordkeeping, reporting or administrative procedures or requirements. Accordingly, this amendment meets the eligibility criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(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.

4.0 CONCLUSION

4.1 Fire Protection

Based upon a review of the licensee's amendment request to remove the fire protection requirements from the TS, to modify the responsibilities of the PSRC to include the fire protection program and its revisions, and to modify License Condition 2.C.(18) so that changes to the fire protection program can only be made if they do not adversely affect the capability of the Station to achieve and maintain safe shutdown in the event of a fire; the staff has concluded that these proposed changes are appropriate and consistent with Generic Letter 86-10 and are, therefore, acceptable.

The Commission made proposed determinations that the amendment involves no significant hazards consideration which were published in the Federal Register on July 29, 1987 (52 FR 28388) and on February 24, 1988 (53 FR 5496), and consulted with the State of South Carolina. No public comments were received, and the State of South Carolina did not have any comments.

4.2 Organizational Charts

Based upon a review of the licensee's amendment request to (1) delete the organizational charts from the TS, (2) modify the titles in the Administrative Controls section of the TS, and (3) review the retraining and replacement training program; the staff has concluded that these proposed changes are appropriate and consistent with GL 88-06.

The Commission made proposed determinations that the amendment involves no significant hazards consideration which were published in the FEDERAL REGISTER on September 9, 1987 (52 FR 34019), May 4, 1988 (52 FR 15916), August 10, 1988 (53 FR 30142), and August 24, 1988 (53 FR 32296), 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 comments.

The staff has concluded, based on the considerations described 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 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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Dated: July 24, 1989