February 9, 1989

Docket No. 50-424

Mr. W. G. Hairston, III
Senior Vice President -
Nuclear Operations
Georgia Power Company
P.O. Box 1295
Birmingham, Alabama 35201

Dear Mr. Hairston:

SUBJECT: ISSUANCE OF AMENDMENT NO.17 TO FACILITY OPERATING LICENSE NPF-68
VOGTLE ELECTRIC GENERATING PLANT, UNIT 1 (TAC 71404)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 17 to Facility Operating License No. NPF-68 for the Vogtle Electric Generating Plant, Unit 1. The amendment is being issued in response to your letter dated December 6, 1988.

The amendment modified the Technical Specifications (TS) to reflect the control room configuration for two-unit operation. These changes were necessitated by differences between the current Unit 1 limiting conditions for operation, action statements, and surveillance requirements, and those proposed for the combined Unit 1 and Unit 2 TS. The amendment is effective as of its date of issuance.
A copy of the related safety evaluation supporting Amendment No. 17 to Facility Operating License NPF-68 is enclosed.

Notice of issuance of the amendment and opportunity for hearing will be included in the Commission's next bi-weekly Federal Register notice.

Sincerely,

/S/

Jon B. Hopkins, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 17 to NPF-68
2. Safety Evaluation

cc w/enclosures:
See next page

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MRood	JSchiffgens	JHopkins:ls	DMatthews
12/23/88	12/23/88	12/23/88	1/9/89

DF01 1/1

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

February 9, 1989

Docket No. 50-424

Mr. W. G. Hairston, III
Senior Vice President -
Nuclear Operations
Georgia Power Company
P.O. Box 1295
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Sincerely,

Jon B. Hopkins

Jon B. Hopkins, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reacator Regulation

Enclosures:
1. Amendment No. 17 to NPF-68
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. W. G. Hairston, III
Georgia Power Company

Vogtle Electric Generating Plant

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Attorney General
Law Department
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Atlanta, Georgia 30334

DATED: February 9, 1989

AMENDMENT NO. 17 TO FACILITY OPERATING LICENSE NPF-68 - Vogtle Electric Generating Plant, Unit 1

DISTRIBUTION:

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ACRS (10)	P-135
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DHagan	MNBB-3302
BGrimes	9-A-2
CSchulten	11-F-23

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 17
License No. NPF-68

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 1 (the facility) Facility Operating License No. NPF-68 filed by the Georgia Power Company acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, (the licensees) dated December 6, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, 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 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; 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 hereby amended by page 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-68 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 17 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. GPC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects
Office of Nucelar Reactor Regulation

Attachment:
Technical Specification Changes

Date of Issuance: February 9, 1989

2. Accordingly, the license is hereby amended by page 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-68 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 17 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. GPC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification Changes

Date of Issuance: February 9, 1989

OFFICIAL RECORD COPY

LA:PDII-3	PM:PDII-3	PM:PDII-3	OGC-WF	D:PDII-3
*MRood	*JSchiffgens	*JHopkins:ls	*	*DBMatthews
12/23/88	12/23/88	12/23/88	12/28/88	01/09/89

*SEE PREVIOUS CONCURRENCE

2. Accordingly, the license is hereby amended by page 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-68 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No.   , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. GPC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification Changes

Date of Issuance:

OFFICIAL RECORD COPY

LA:PDII-3	PM:PDII-3	PM:PDII-3	OGC-WF	D:PDII-3
MRood	for JSchiffgens	for JHopkins:ls		DBMatthews
12/23/88	12/23/88	12/23/88	12/29/88	1/9/89

ATTACHMENT TO LICENSE AMENDMENT NO. 17

FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

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 pages are also provided to maintain document completeness.

Amended Page	Overleaf Page
3/4 3-17	
3/4 3-24	
3/4 3-27	
3/4 3-27a	
3/4 3-28	
3/4 3-34	3/4 3-33
3/4 3-36	3/4 3-35
3/4 3-43	
3/4 3-44	
3/4 3-46	
3/4 3-63	3/4 3-64
3/4 7-14	
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3/4 7-16	
B 3/4 3-3	
B 3/4 3-5	B 3/4 3-6
B 3/4 7-3	
B 3/4 7-4	
3/4 7-15	

TABLE 3.3-2

ENGINEERED SAFETY FEATURES 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, Component Cooling Water, Control Room Emergency Filtration System Actuation, Start Diesel Generators, Containment Cooling Fans, Nuclear Service Cooling Water, Containment Isolation, Containment Ventilation Isolation, and Auxiliary Feedwater Motor-Driven Pumps).					
a. Manual Initiation	2	1	2	1, 2, 3, 4	19
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure--High-1* (PI-0934, PI-0935, PI-0936)	3	2	2	1, 2, 3, 4	15[d]
d. Pressurizer Pressure--Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	4	2	3	1, 2, 3[a]	20[d]
e. Steam Line Pressure--Low*	3/steam line	2/steam line any steam line	2/steam line	1, 2, 3[a]	15[d]

(LOOP1	LOOP2	LOOP3	LOOP4
PI-0514A,B&C	PI-0524A&B	PI-0534A&B	PI-0544A,B&C
PI-0515A	PI-0525A	PI-0535A	PI-0545A
PI-0516A	PI-0526A	PI-0536A	PI-0546A)

*See Specification 3.3.3.6

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. Control Room Emergency Filtration System Actuation					
a. Manual Initiation	2	1	1 in either Unit	Either Unit in 1, 2, 3, 4, 5^e, 6^e	26^d
b. Automatic Actuation Logic and Actuation Relays	2	1	2	Either Unit in 1, 2, 3, 4, 5^e, 6^e	27^d
c. Safety Injection	See Functional Unit 1 above for all Safety Injection initiating functions and requirements.				
d. Intake Radiogas Monitor (RE-12116, RE-12117)	2	1	2	Either Unit in 1, 2, 3, 4, 5^e,6^e	28^d
11. Fuel Handling Building Post Accident Ventilation Actuation					
a. Manual Initiation	2	1	1	i	j
b. Fuel Handling Building Exhaust Duct Radiation Signal (ARE-2532 A&B ARE-2533 A&B)	4	1	1	i	j
c. Automatic Actuation Logic and Actuation Relays	4	1	1	i	j

TABLE 3.3-2 (Continued)

ACTION STATEMENTS (Continued)

ACTION 24 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 25 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 26 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore at least one channel to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the Control Room Emergency Filtration System in the Emergency mode#.

ACTION 27 - a. With one channel inoperable in a unit, restore the inoperable channel to OPERABLE status within 7 days OR within the next 6 hours initiate and maintain operation of one Control Room Emergency Filtration System (CREFS) in the unaffected unit in the emergency mode#.

b. With on channel inoperable in each unit, restore each inoperable channel to OPERABLE status within 7 days OR within the next 6 hours initiate and maintain operation of one CREFS in each unit in the emergency mode#.

c. With two channels inoperable in a unit, within 1 hour either 1) initiate and maintain operation of the two CREFS in the unaffected unit OR 2) initiate and maintain operation of one CREFS in each unit in the emergency mode#.

d. With three channels inoperable, within 1 hour initiate and maintain operation of the two CREFS in the emergency mode in the unit with only one channel inoperable.

e. With four channels inoperable, within 1 hour initiate and maintain operation of two CREFS in the emergency mode.

ACTION 28 - a. With one channel inoperable in a unit, restore the inoperable channel to OPERABLE status within 7 days, OR within the next 6 hours either 1) lock closed the affected and lock open the unaffected outside-air (OSA) intake dampers OR 2) initiate and maintain operation of one CREFS in the emergency mode#.

#The initiated CREFS shall be Train B unless Train B is inoperable.

TABLE 3.3-2 (Continued)

ACTION STATEMENTS (Continued)

b. With one channel inoperable in each unit, restore each inoperable channel to OPERABLE status within 7 days OR within the next 6 hours initiate and maintain operation of one CREFS in the emergency mode#.

c. With two channels inoperable in a unit, within 1 hour either 1) lock closed the affected and lock open the unaffected OSA intake dampers OR 2) initiate and maintain operation of one CREFS in each unit in the emergency mode#.

d. With three channels inoperable, within 1 hour either 1) lock closed the OSA intake dampers of the units with two inoperable channels and lock open the other OSA intake dampers and either restore the remaining affected channel to OPERABLE status within 7 days OR initiate and maintain operation of one CREFS in the emergency mode# in the following 6 hours OR 2) initiate and maintain operation of one CREFS in each unit in the emergency mode#.

e. With four channels inoperable, within 1 hour initiate and maintain operation of the CREFS in each unit in the emergency mode#.

#The initiated CREFS shall be Train B unless Train B is inoperable.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Feedwater Isolation, Component Cooling Water, Control Room Emergency Filtration System Actuation, Start Diesel Generators, Containment Cooling Fans, Nuclear Service Cooling Water, Containment Isolation, Containment Ventilation Isolation, and Auxiliary Feedwater Motor-Driven Pumps)					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High 1 (PI-0934, PI-0935, PI-0936)	2.7	0.71	1.67	$\leq$ 3.8 psig	$\leq$ 4.4 psig
d. Pressurizer Pressure--Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	13.1	10.71	1.67	$\geq$ 1870 psig***	$\geq$ 1860 psig
e. Steam Line Pressure--Low	13.0	10.71	1.67	$\geq$ 585 psig*	$\geq$ 570 psig
(LOOP1 PI-0514A,B&C PI-0515A PI-0516A; LOOP2 PI-0524A&B PI-0525A PI-0526A; LOOP3 PI-0534A&B PI-0535A PI-0536A; LOOP4 PI-0544A,B&C PI-0545A PI-0546A)					
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-3 (PI-0934, PI-0935, PI-0936, PI-0937)	3.1	0.71	1.67	$\leq$ 21.5 psig	$\leq$ 22.4 psig

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
7. Semi-Automatic Switchover to Containment Emergency Sump (Continued)					
b. RWST Level--Low-Low Coincident With Safety Injection (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A)	3.5	0.71	1.67	≥275.3 in. from tank base (≥39.1% of instrument span)	≥264.9 in. from tank base (≥37.4% of instrument span)
8. Loss of Power to 4.16 kV ESF Bus					
a. 4.16 kV ESF Bus Undervoltage-Loss of Voltage	N.A.	N.A.	N.A.	≥2975 volts with a ≤ 0.8 second time delay.	≥2912 volts with a ≤ 0.8 second time delay.
b. 4.16 kV ESF Bus Undervoltage-Degraded Voltage	N.A.	N.A.	N.A.	≥3746 volts with a ≤20 second time delay.	≥3683 volts with a ≤20 second time delay.
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11 (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	N.A.	N.A.	N.A.	≤ 1970 psig	≤ 1980 psig
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
10. Control Room Emergency Filtration System Actuation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Safety Injection	See Functional Unit 1 above for all Safety Injection Trip Setpoints and Allowable Values.				
d. Intake Radiogas monitor (RE-12116, RE-12117)	N.A.	N.A.	N.A.	$\leq$ 3X background	N.A.
11. Fuel Handling Building Post-Accident Ventilation System Actuation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Fuel Handling Building Exhaust Duct Radiation Signal (ARE-2532A&B, ARE-2533A&B)	N.A.	N.A.	N.A.	C	N.A.
c. Automatic Actuation Logic And Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

**The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate-High is greater than or equal to 50 seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

***Until resolution of the Veritrak transmitter uncertainty issue this setpoint will be set at ≥1885 psig.

#Until resolution of the Veritrak transmitter uncertainty issue the setpoint will be set at ≥22.5% of narrow range instrument span.

##Feedwater isolation only. Turbine trip occurs on reactor trip.

[a]During refueling operations.

[b]During power operation. This is an initial setpoint only. The trip setpoint will be set at 50 times background level. Background level should be determined at or near the end of the first fuel cycle.

[c]Setpoints will not exceed the limits of Specification 3.11.2.1.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL 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, Component Cooling Water, Control Room Emergency Filtration System Actuation, Start Diesel Generators, Containment Cooling Fans, Nuclear Service Cooling Water, Containment Isolation, Containment Ventilation Isolation, and Auxiliary Feedwater Motor-driven Pumps).								
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. Containment Pressure-High-1 (PI-0934, PI-0935, PI-0936)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Pressurizer Pressure-Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

*See Specification 4.3.3.6

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL 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
9. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11 (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A)	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
10. Control Room Emergency Filtration System Actuation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	Either Unit in 1, 2, 3, 4. 5#, 6#
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	N.A.	N.A.	Either Unit in 1, 2, 3, 4, 5#, 6#

*See Specification 4.3.3.6
#During movement of irradiated fuel or movement of loads over irradiated fuel.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL 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
10. Control Room Emergency Filtration System Actuation (Continued)								
c. Safety Injection	See Functional Unit 1 above for all Safety Injection Surveillance Requirements							
d. Intake Radiogas Monitor (RE-12116, RE-12117)	S	R	M	N.A.	N.A.	N.A.	N.A.	Either Unit in 1, 2, 3, 4, 5#, 6#
11. Fuel Handling Building Post Accident Ventilation Actuation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	(2)
b. Fuel Handling Building Exhaust Duct Radiation Signal (ARE-2532 A&B ARE-2533 A&B)	S	R	M	N.A.	N.A.	N.A.	N.A.	(2)
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	N.A.	N.A.	(2)

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
(2) Whenever irradiated fuel is in the storage pool.
During movement of irradiated fuel or movement of loads over irridated fuel.

TABLE 3.3-4

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment Area (High Range) (RE-0005, RE-0006)	1	See Table 3.3-8	1, 2, 3, 4	100 R/hr	See Table 3.3-8
b. RCS Leakage Detection					
1) Gaseous Activity (RE-2562C)	1	1	1, 2, 3, 4	$\leq$ 2 x back-ground	29
2) Particulate Activity (RE-2562A)	1	1	1, 2, 3, 4	$\leq$ 2 x back-ground	29
2. Containment Ventilation	1	2^{c}	1, 2, 3, 4, 6^{a}	See Table 3.3-3	See Table 3.3-2
Area Low Range (RE-0002, RE-0003) Gaseous Activity (RE-2565C) Particulate Activity (RE-2565A) Iodine Activity (RE-2565B)					
3. Control Room Air Intake					
(RE-12116, RE-12117)	1	2	Either Unit in 1, 2, 3, 4, 5^{b}, 6^{b}	See Table 3.3-3	See Table 3.3-2

INSTRUMENTATION

CHLORINE DETECTION SYSTEMS

Specification 3/4.3.3.7 Deleted.

INSTRUMENTATION

LOOSE PARTS DETECTION SYSTEM

Specification 3/4 3.3.3.8 Deleted

3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM (Common System)

LIMITING CONDITION FOR OPERATION

3.7.6 Four independent Control Room Emergency Filtration Systems (CREFS) shall be OPERABLE.

APPLICABILITY: Either Unit in MODES 1, 2, 3, and 4. MODES 5 and 6 during movement of irradiated fuel or movement of loads over irradiated fuel.

ACTION:

With both units in MODES 1, 2, 3, or 4:

a. With one Control Room Emergency Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days OR initiate and maintain operation of one train of the CREFS in the unaffected unit in the emergency mode# OR within 1 hour lock closed the affected and lock open the unaffected unit's outside-air (OSA) intake dampers and place the affected unit in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one Control Room Emergency Filtration System inoperable in each unit, restore each inoperable system to OPERABLE status within 7 days OR initiate and maintain operation of both remaining CREFS in the emergency mode.

c. With two Control Room Emergency Filtration Systems inoperable in a unit, initiate and maintain operation of both trains of CREFS in the unaffected unit in the emergency mode OR within 1 hour lock closed the affected and lock open the unaffected unit's OSA intake dampers and place the affected unit in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

With only one unit in MODES 1, 2, 3, or 4:

a. With one Control Room Emergency Filtration System inoperable in an operating unit, restore the inoperable system to OPERABLE status within 7 days OR initiate and maintain operation of one train of the CREFS in the shutdown unit in the emergency mode#.

b. With one Control Room Emergency Filtration System inoperable in a shutdown unit, restore the inoperable system to OPERABLE status within 7 days OR either 1) lock closed the affected and lock open the unaffected unit's OSA intake dampers OR 2) initiate and maintain operation of one train of CREFS in the operating unit in the emergency mode#.

#The initiated CREFS shall be Train B unless Train B is inoperable.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one Control Room Emergency Filtration System inoperable in each unit, restore each inoperable system to OPERABLE status within 7 days OR lock closed the shutdown unit's OSA intake dampers and lock open the operating unit's OSA intake dampers and initiate and maintain operation of the remaining CREFS train in the shutdown unit in the emergency mode.

d. With two Control Room Emergency Filtration Systems inoperable in an operating unit, initiate and maintain operation of both CREFS trains in the shutdown unit in the emergency mode.

e. With two Control Room Emergency Filtration Systems inoperable in a shutdown unit, either 1) lock closed the affected and lock open the unaffected unit's OSA intake dampers or 2) initiate and maintain operation of both trains of the CREFS in the operating unit in the emergency mode.

With both units in MODES 5 or 6, during movement of irradiated fuel or movement of loads over irradiated fuel in either unit:

a. With one Control Room Emergency Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days OR either 1) lock closed the affected and lock open the unaffected unit's OSA intake dampers OR 2) initiate and maintain operation of one train of the CREFS in the unaffected unit in the emergency mode#.

b. With one Control Room Emergency Filtration System inoperable in each unit, restore inoperable system to OPERABLE status within 7 days OR initiate and maintain operation of one train of CREFS in the emergency mode.

c. With two Control Room Emergency Filtration Systems inoperable in a unit, either 1) lock closed the affected and lock open the unaffected unit's OSA intake dampers, OR 2) initiate and maintain operation of one train of the CREFS in the unaffected unit in the emergency mode.

d. With three Control Room Emergency Filtration Systems inoperable, either 1) lock closed the OSA intake dampers of the unit with two inoperable systems and lock open the OSA intake dampers of the unit with one inoperable system and restore the one inoperable system to OPERABLE status within 7 days OR initiate and maintain operation of the remaining train of CREFS in the emergency mode; OR 2) initiate and maintain operation of the remaining train of CREFS in the emergency mode.

#The initiated CREFS shall be Train B unless Train B is inoperable.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

e. With four Control Room Emergency Filtration Systems inoperable or with the OPERABLE CREFS required to be in the emergency mode by ACTION a, b, c, or d above not capable of being powered by an OPERABLE emergency power source, suspend all operations involving movement of irradiated fuel or movement of loads over irradiated fuel.

SURVEILLANCE REQUIREMENTS

4.7.6 Each Control Room Emergency Filtration System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 85°F

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow (FI-12191, FI-12192) through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heater control circuit energized.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

    1) Verifying that the filtration system satisfies the in-place testing acceptance criteria of greater than or equal to 99.95% filter retention while operating the system at a flow rate of 19,000 cfm ±10% and performing the following tests:

        (a) A visual inspection of the control room emergency filtration system shall be made before each DOP test or activated carbon adsorber section leak test in accordance with Section 5 of ANSI N510-1980.

        (b) An in-place DOP test for the HEPA filters shall be performed in accordance with Section 10 of ANSI N510-1980.

        (c) A charcoal adsorber section leak test with a gaseous halogenated hydrocarbon refrigerant shall be performed in accordance with Section 12 of ANSI N510-1980.

SURVEILLANCE REQUIREMENTS (Continued)

2) Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Section 13 of ANSI N510-1980 meets the laboratory testing criterion of greater than or equal to 99.8% when tested with methyl iodide at 30°C and 70% relative humidity.

3) Verifying a system flow rate of 19,000 cfm ± 10% during system operation when tested in accordance with Section 8 of ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Section 13 of ANSI N510-1980 meets the laboratory testing criterion of greater than or equal to 99.8% when tested with methyl iodide at 30°C and 70% relative humidity.

e. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters, charcoal adsorber banks and cooling coil is less than 7.1 inches Water Gauge while operating the system at a flow rate of 19,000 cfm ± 10%;

2) Verifying that on a Control Room Isolation Test Signal, the system automatically switches into an emergency mode of operation with flow through the HEPA filters and charcoal adsorber banks;

3) Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch Water Gauge at less than or equal to a pressurization flow of 1500 cfm relative to adjacent areas during system operation; and

4) Verifying that the heaters dissipate 118 ± 6 kW when tested in accordance with Section 14 of ANSI N510-1980;

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in place in accordance with Section 10 of ANSI N510-1980 while operating the system at a flow rate of 19,000 cfm ± 10%; and

g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the charcoal absorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when tested in-place in accordance with Section 12 of ANSI N510-1980 while operating the system at a flow rate of 19,000 cfm ± 10%.

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

(7) steam line isolation, (8) turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment fan coolers start and automatic valves position, (11) Nuclear Service Cooling and Component Cooling water pumps start and automatic valves position, and (12) Control Room Emergency Filtration System start.

The Engineered Safety Features 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 With pressurizer pressure below the P-11 setpoint, allows manual block of safety injection actuation on low pressurizer pressure signal. Allows manual block of safety injection actuation and steam line isolation on low compensated steam line pressure signal and allows steam line isolation on high steam line negative pressure rate. With pressurizer pressure above the P-11 setpoint, defeats manual block of safety injection actuation on low pressurizer pressure and safety injection and steam line isolation on low steam line pressure and defeats steam line isolation on high steam line negative pressure rate.

P-14 On increasing steam generator water level, P-14 automatically trips all feedwater isolation valves, initiates a turbine trip, and inhibits feedwater control valve modulation.

The Source Range High Flux at Shutdown Alarm Setpoint is an analysis assumption for mitigation of a Boron Dilution Event in MODES 3, 4, and 5.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

REMOTE SHUTDOWN SYSTEM (Continued)

control, and transfer switches necessary to eliminate effects of the fire and allow operation of instrumentation, and control circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shut down the reactor. This capability is consistent with General Design Criterion 3 and CMEB 9.5.1.

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," Revision 2, December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980. The instrumentation listed in Table 3.3-8 corresponds to the Category 1 instrumentation for which selection, design, qualification and display criteria are described in Regulatory Guide 1.97, Rev. 2.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The Chlorine Detection System is not required because the quantity of chlorine gas stored on site is small (< 20 lbs.) and utilized for laboratory and calibration purposes. This applicability is consistent with the exclusions and recommendations of Regulatory Guide 1.95, Revision 1, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," January 1977.

3/4.3.3.8 LOOSE PARTS DETECTION SYSTEM

Not used.

3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.10 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the GASEOUS WASTE PROCESSING SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50. The sensitivity of any noble gas activity monitors used to show compliance with the gaseous effluent release requirements of Specification 3.11.2.2 shall be such that concentrations as low as 1×10^{-6} µCi/ml are measurable.

3/4.3.3.11 HIGH ENERGY LINE BREAK ISOLATION SENSORS

The operability of the high energy line break isolation sensors ensures that the capability is available to promptly detect and initiate protective action in the event of a line break. This capability is required to prevent damage to safety-related systems and structures in the auxiliary building.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety-related components, equipment or structures.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 60°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 NUCLEAR SERVICE COOLING WATER (NSCW) SYSTEM

The OPERABILITY of the Nuclear Service Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level, temperature, minimum required number of OPERABLE fans, and NSCW transfer pumps ensure that sufficient cooling capacity is available to either: (1) provide normal cooldown of the facility or (2) mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level, maximum temperature, minimum required number of OPERABLE fans, and NSCW transfer pumps are based on providing an adequate cooling water supply to safety-related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," Revision 2, January 1976.

3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

The OPERABILITY of the Control Room Emergency Filtration System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system during accident conditions, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. Operation of the system with the heater control circuit energized for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of

CONTROL ROOM EMERGENCY FILTRATION SYSTEM (Continued)

moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rems or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

3/4.7.7 PIPING PENETRATION AREA FILTRATION AND EXHAUST SYSTEM

The OPERABILITY of the Piping Penetration Area Filtration and Exhaust System ensures that radioactive materials leaking from the containment mechanical penetration rooms and ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. Operation of the system with the heater control circuit energized for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

3/4.7.8 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Plant Review Board. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 17 TO FACILITY OPERATING LICENSE NPF-68

GEORGIA POWER COMPANY, ET AL.

DOCKET NO. 50-424

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

1.0 INTRODUCTION

By letter dated December 6, 1988, Georgia Power Company, et al., (GPC) (the licensee) proposed to delete the Vogtle, Unit 1 Technical Specifications (TS) Section 3/4.3.3.7, Chlorine Detection Systems, and amend Section 3/4.3.2, Control Room Emergency Filtration System, and the relevant portions of Section 3/4.3.2, Engineered Safety Features Actuation System Instrumentation, and Bases 3/4.3, Instrumentation, to reflect the control room configuration for two-unit operation. This amendment request was necessitated by differences between the current Unit 1 limiting conditions for operation, action statements, and surveillance requirements, and those proposed for the combined Unit 1 and Unit 2 TS. In summary, the differences between the current Unit 1 TS requirements and the proposed combined TS requirements are due to: (1) operation of the Unit 2 CREFS and associated Unit 2 actuation instrumentation; (2) disabling of the circuitry for control room isolation on a chlorine signal; (3) revision of the automatic start logic to prevent more than two CREFS trains from automatically operating; and (4) various administrative changes to delete obsolete footnotes and maintain consistent nomenclature.

2.0 EVALUATION

The Vogtle combined control room envelope will be served by four Control Room Emergency Filtration System (CREFS) trains. Two trains will be powered from Unit 1 and two powered from Unit 2. Each train has sufficient capacity to remove the combined control room heat load and pressurize the combined control room to 1/8 inch water gauge relative to adjacent areas. The bases of the combined Unit 1 and Unit 2 TS meet the same criteria as the current Unit 1 TS. That is, the system has been designed with sufficient heat removal capability so that equipment qualification temperatures will not be exceeded and the control room will remain habitable during and following all credible accident conditions, as well as meet single failure criteria.

During normal plant operation the combined control room is to be served by a non-safety related HVAC system which draws in outside air makeup via either of two outside air (OSA) intakes. There is one intake associated with each unit. Each air intake is provided with two redundant radiation monitors that are powered by the associated unit. Upon detection of radiation in one OSA intake at or above the radiation monitor setpoint, each monitor will automatically initiate both of the associated units' Control Room Isolation signals (CRI-A and CRI-B). Each CRI signal functions to start its associated CREFS and isolate the normal HVAC system. The lead/lag logic employed will permit only one CREFS to start. If high radiation is detected in both OSA intakes, all four CRI signals will be generated and two CREFSs will start. Upon receipt of a Safety Injection (SI) signal in a unit, the associated train's CRI signal will be generated. Just as for high radiation initiated CRI signals, if an SI-A and an SI-B initiate CRI signals, the lead/lag logic will permit only one CREFS to start.

The radiological consequences of the limiting LOCA have been analyzed by the licensee for the two-unit control room configuration. These analyses showed slight increases in control room doses; however, the total doses remained within the limits of 10 CFR 50, Appendix A, General Design Criterion 19. The NRC staff approved the revised control room doses in the safety evaluation for Amendment 11 to the Vogtle Unit 1 TS. The bases for the operability of the CREFS actuation instrumentation also remain applicable. These instruments provide redundant and diverse means for initiating a CRI in response to credible accidents in either unit, as well as meet single failure criteria.

By Final Safety Analysis Report, Amendment 39, the licensee committed to not store liquified gaseous chlorine in excess of twenty pounds onsite. As a result of this commitment the automatic switchover of control room HVAC system to the isolation mode is no longer required and, therefore, it was deleted from the system design. Based on its review, the staff concludes that automatic isolation of the control room on chlorine detection is no longer required at the Vogtle plant and the requirements of GDC 19 are still met for the control room HVAC system. Thus, the need to consider an on-site chlorine release as a credible accident has been eliminated; thereby, permitting the deletion of TS Section 3/4.3.3.7 and the disabling of the circuitry for Control Room Isolation on a chlorine signal.

The limiting condition for operation and mode applicability for TS 3/4.3.2 have been revised to reflect that four CREFSs are required and that the specification is applicable when either unit is in Mode 1, 2, 3 or 4 or in Mode 5 or 6 during movement of irradiated fuel or movement of loads over irradiated fuel. That is, the specification is applicable at all times unless both units are shut down and no movement of irradiated fuel or movement of loads over irradiated fuel is occurring.

The NRC staff has reviewed the licensee's proposed changes and finds that they are adequate and acceptable. The changes do not affect safety limits or limiting safety system settings. The bases of the current Unit 1 Technical Specifications are maintained for the two-unit control room configuration.

The current Unit 1 criteria for control room heat removal, pressurization, radiation protection, and safety injection actuation are met, including single failure capability. The proposed surveillance requirements and allowed outage times are also consistent with current Unit 1 requirements. Finally, control room habitability is maintained consistent with the current TS requirements.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. 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 exposure. The NRC staff has made a determination that the amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register on December 16, 1988 (53 FR 50607), and consulted with the state of Georgia. No public comments were received, and the state of Georgia 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.

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Dated: February 9, 1989