4/6/78

Docket No. 50-321

Georgia Power Company
Oglethorpe Electric Membership Corporation
Municipal Electric Association of Georgia
City of Dalton, Georgia
ATTN: Mr. Charles F. Whitmer
Vice President - Engineering
Georgia Power Company
Atlanta, Georgia 30302

Gentlemen:

The Commission has issued the enclosed Amendment No. 51 to Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant Unit No. 1. The amendment consists of changes to the Technical Specifications and is in response to your letters dated August 4, 1976 (as supplemented April 26, 1977), November 16, 1977, December 2, 1977, December 9, 1977 and February 3, 1978.

The amendment consists of changes to the Technical Specifications and involve: (1) instrumentation setpoint changes resulting from startup testing, (2) the Hatch Unit No. 1 Containment Leak Rate Test Program, (3) deletion of specific snubbers from the Table of Safety Related Shock Suppressors, (4) revision of action statements associated with exceeding LHGR, APLHGR and MCPR limits, (5) revision to the performance requirements of the SGTS and Control Room filter systems, (6) modification of the Control Room ventilation system and (7) deletion of the respiratory protection program.

Your letter dated November 16, 1977 included a proposal to hydrostatically test the valves in the RHR suction lines and to revise the reporting requirements for leakage from the RHR suction line valves and certain other valves that you stated were sealed from the primary containment atmosphere. We have not yet completed our review of these items.

Cont. 1
GD

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

/s/

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 51 to DPR-57
2. Safety Evaluation
3. Notice

cc w/enclosure:
see next page

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Georgia Power Company
Oglethorpe Electric Membership Corporation
Municipal Electric Association of Georgia
City of Dalton, Georgia

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION
MUNICIPAL ELECTRIC ASSOCIATION OF GEORGIA
CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 51
License No. DPR-57

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

   A. The applications for amendment by Georgia Power Company, et al, (the licensee) dated August 4, 1976 (as supplemented April 26, 1977), November 16, December 2 and 9, 1977, and February 3, 1978, 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 paragraph 2.C.(2) of Facility Operating License No. DPR-57 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 51, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 6, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 51

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

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 area of change.

Remove	Replace
3.2-5	3.2-5
3.2-6	3.2-6
3.2-8	3.2-8
3.2-9	3.2-9
3.2-17*	3.2-17*
3.2-18	3.2-18
3.2-19	3.2-19
3.2-20*	3.2-20*
3.6-10c	3.6-10c
3.6-10d	3.6-10d
3.6-10e	3.6-10e
3.7-11	3.7-11
3.7-23	3.7-23
3.11-1	3.11-1
3.11-2	3.11-2
3.12-1	3.12-1
3.12-2	3.12-2
3.12-4	3.12-4
3.12-5	3.12-5
3.12-6	3.12-6
6-17	6-17
6-18	6-18
6-19	6-19
6-22	6-22
6-23	6-23
6-24	6-24

*Overleaf page (no changes)

Amendment No. ~~37~~, 51

Table 3.2-2

INSTRUMENTATION WHICH INITIATES OR CONTROLS HPCI

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Remarks
1.	Reactor Water Level (Yarway)	Low Low (LL2)	2	≥ -38 inches	Initiates HPCI; Also initiates RCIC.
2.	Drywell Pressure	High	2	≤ 2 psig	Initiates HPCI; Also initiates LPCI and Core Spray and provides a permissive signal to ADS.
3.	HPCI Turbine Overspeed	Mechanical	1	≤ 5000 rpm	Trips HPCI turbine
4.	HPCI Turbine Exhaust Pressure	High	1	≤ 150 psig	Trips HPCI turbine
5.	HPCI Pump Suction Pressure	Low	1	≤ 15" Hg vacuum	Trips HPCI turbine
6.	Reactor Water Level (Narrow Range)	High	2	≤ +58 inches	Trips HPCI turbine
7.	HPCI System Flow (Flow Switch)	High	1	> 800 gpm	Closes HPCI minimum flow pass line to suppression chamber.
		Low	1	≤ 500 gpm	Opens HPCI minimum flow bypass line if pressure permissive is present.
8.	HPCI Equipment Room Temperature	High	1	≤ 175°F	Closes isolation valves in HPCI system, trips HPCI turbine.
9.	HPCI Equipment Room Differential Temperature	High	1	≤ 50°F	Closes isolation valves in HPCI system, trips HPCI turbine.

Table 3.2-2 (Cont.)

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Remarks
10.	HPCI Steam Line Pressure	Low	2	$\geq$ 100 psig	Closes isolation valves in HPCI system, trips HPCI turbine.
11.	HPCI Steam Line ΔP (Flow)	High	1	$\leq$ 216" water (300% Flow)	Close isolation valves in HPCI system, trips HPCI turbine.
12.	HPCI Turbine Exhaust Diaphragm Pressure	High	1	$\leq$ 10 psig	Close isolation valves in HPCI system, trips HPCI turbine.
13.	Suppression Chamber Area Air Temperature	High	1	$\leq$ 175°F	Close isolation valves in HPCI system, trips HPCI turbine.
14.	Suppression Chamber Area Differential Air temperature	High	1	$\leq$ 50°F	Close isolation valves in HPCI system, trips HPCI turbine.
15.	Condensate Storage Tank Level	Low	2	$\geq$0 inches	Automatic interlock switches suction from CST to suppression chamber.
16.	Suppression Chamber Water Level	High	2	$\leq$0 inches	Automatic interlock switches suction from CST to suppression chamber.
17.	HPCI Logic Power Failure Monitor		1	Not Applicable	Monitors availability of power to logic system.

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between items in Table 3.2-2 and items in Table 4.2-2.

Table 3.2-3
INSTRUMENTATION WHICH INITIATES OR CONTROLS RCIC

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Remarks
1.	Reactor Water Level (Yarway)	Low Low (LL2)	2	≥-38 inches	Initiates RCIC; also initiates HPCI.
2.	RCIC Turbine Overspeed	Electrical	1	≤110% rated	Trips RCIC turbine.
		Mechanical	1	≤125% rated	Trips RCIC turbine.
3.	RCIC Turbine Exhaust Pressure	High	1	≤+25 psig	Trips RCIC turbine.
4.	RCIC Pump Suction Pressure	Low	1	≤15" Hg Vacuum	Trips RCIC turbine.
5.	Reactor Water Level (Narrow Range)	High	2	≤+58 inches	Trips RCIC turbine.
6.	RCIC System Flow (Flow Switch)	High	1	>80 gpm	Closes RCIC minimum flow bypass line to suppression chamber.
		Low	1	≤40 gpm	Opens RCIC minimum flow bypass line if pressure permissive is present.
7.	RCIC Equipment Room Temperature	High	1	≤175^{0}F	Closes isolation valves in RCIC system, trips RCIC turbine.
8.	RCIC Equipment Room Differential Temperature	High	1	≤50^{0}F	Closes isolation valves in RCIC system, trips RCIC turbine.

Table 3.2-3 (cont.)

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Remarks
9.	RCIC Steam Line Pressure	Low	2	$\geq$ 50 psig	Closes isolation valves in RCIC system, trips RCIC turbine.
10.	RCIC Steam Line Flow (Upstream and Downstream Elbow Taps)	High	1	$\leq$ 300% Flow	Closes isolation valves in RCIC system, trips RCIC turbine.
11.	RCIC Turbine Exhaust Diaphragm Pressure	High	1	$\leq$ 10 psig	Closes isolation valves in RCIC system, trips RCIC turbine.
12.	Suppression Chamber Area Air Temperature	High	1	$\leq 175^0$F	Closes isolation valves in RCIC system, trips RCIC turbine.
13.	Suppression Chamber Area Differential Air Temperature	High	1	$\leq 50^0$F	Closes isolation valves in RCIC system, trips RCIC turbine.
14.	RCIC Logic Power Failure Monitor		1	Not Applicable	Monitors availability of power to logic system.

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between items in Table 3.2-3 and items in Table 4.2-3.

b. Whenever any CCCS subsystem is required to be operable by Section 3.5, there shall be two operable trip systems. If the required number of operable channels cannot be met for one of the trip systems, that system shall be repaired or the reactor shall be placed in the Cold Shutdown Condition within 24 hours after this trip systems is made or found to be inoperable.

Notes for Table 3.2-7

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between items in Table 3.2-7 and items in Table. 4.2-7.

b. For the START & HOT STANDBY position of the Mode Switch, there shall be two operable or tripped systems for each potential trip condition. If the requirements established by the column cannot be met for one of the two trip systems, the condition may exist for up to seven days provided that during that time the operable systems is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the requirements established by this column cannot be met for both trip systems, the systems shall be tripped.

c. One of the four SRM inputs may be bypassed.

d. The SRM and IRM blocks need not be operable in the Run Mode. This function is bypassed when the Mode Switch is placed in the RUN position.

e. The APRM and RBM rod blocks need not be Operable in the Start & Hot Standby Mode (Except 12% APRM Rod Block)

f. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.

g. This trip bypassed when the reactor power is $\leq$ 30%.

h. One channel of the RBM may be inoperative or bypassed if this condition does not persist longer than 24 hours in a 30 day period.

Table 3.2-8
RADIATION MONITORING SYSTEMS WHICH LIMIT RADIOACTIVITY RELEASE

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Action to be taken if there are not two operable or tripped trip systems	Remarks
1.	Off-gas Post Treatment Radiation Monitors	Upscale/ Downscale	1	At a value not to exceed the equivalent of the stack release limit indicated in Environmental Tech Specs	(c) (d)	2 upscales, or 1 downscale and 1 upscale, or 2 downscales will isolate the SJAE off-gas
2.	Refueling Floor Exhaust Vent Radiation Monitors	Upscale	2	At a value not to exceed the equivalent of the stack release limit indicated in Environmental Tech Specs	Cease refueling operations, if in progress. Isolate the secondary containment and start the standby gas treatment system.	2 upscale will isolate the secondary containment and initiate the standby gas treatment system.
3.	Reactor Bldg. Exhaust Vent Radiation Monitors	Upscale	2	$\leq$ 20 mr/hr	Isolate the secondary containment, start standby gas treatment system, close primary containment and vent valves.	2 upscale will isolate the secondary containment and initiate the standby gas treatment system.
4.	Control Room Intake Radiation Monitors	Downscale Hi	2	$\geq$0.015 mr/hr $\leq$1.0 mr/hr	Refer to Specifications 3.12.C and 3.12.D.	1 upscale or 2 downscale will isolate the main control room from outside air.

Table 3.2-8 (cont.)

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Action to be taken if there are not two operable or tripped trip systems	Remarks
5.	Main Steam Line Radiation Monitor	Hi	2	$\leq$3 times normal full power background	Isolate the mechanical vacuum pump and the gland seal condenser exhauster	One trip per trip logic system will isolate the mechanical vacuum pump and the gland seal condenser exhauster.

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between items in Table 3.2-8 and items in Table 4.2-8.

b. Whenever the systems are required to be operable, there shall be two operable or tripped trip systems. If this cannot be met, the indicated action shall be taken.

c. In the event that both off-gas post treatment radiation monitors become inoperable, the reactor shall be placed in the Cold Shutdown within 24 hours unless one monitor is sooner made operable, or adequate alternative monitoring facilities are available.

d. From and after the date that one of the two off-gas post treatment radiation monitors is made or found to be inoperable, continued reactor power operation is permissible during the next fourteen days (the allowable repair time), provided that the inoperable monitor is tripped in the downscale position.

Table 3.2-9

INSTRUMENTATION WHICH INITIATES RECIRCULATION PUMP TRIP

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Remarks
1	Reactor Water Level	LowLow (LL2)	1	$\geq$-38 inches	Power must be reduced and the mode switch placed in a mode other than the RUN Mode.
2	Reactor Pressure	High	1	$\leq$1120 psig	Power must be reduced and the mode switch placed in a mode other than the RUN Mode.

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between items in Table 3.2-9 and items in Table 4.2-9.

b. Whenever the reactor is in the RUN Mode, there shall be one operable trip system for each parameter for each operating recirculation pump, except that one trip system may remain inoperable for up to 14 days. If this cannot be met, the indicated action shall be taken.

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

A. SNUBBERS NOT ACCESSIBLE DURING REACTOR OPERATION

MAIN STEAM SEISMIC RESTRAINTS

SNUBBER #	SIZE (kips)	LOCATION	ELEVATION	SHUTDOWN ACCESSIBILITY CODE
SS- 1	20	72°	150'	RA
2	20	72°	150'	RA
3	10	0°	140'	RA
4	10	5°	128'	RA
5	10	0°	140'	RA
6	30	108°	150'	RA
7	20	90°	150'	RA
8	3	160°	151'	RA
9	3	95°	144'	RA
10	3	90°	142'	DA
11	10	100°	138'	DA
12	10	85°	144'	DA
13	10	82°	143'	RA
14	10	315°	150'	RA
15	10	90°	145'	RA
16	10	90°	144'	RA
17	3	170°	145'	N
18	3	160°	145'	N
19	3	160°	134'	RA
20	3	160°	145'	N
21	10	135°	158'	RA
22	10	160°	152'	RA
23	30	278°	150'	RA
24	50	270°	150'	RA
26	3	275°	145'	RA
27	10	270°	138'	RA
28	10	270°	142'	DA
29	3	270°	140'	RA
31	3	270°	139'	RA
32	3	275°	125'	RA
33A	3	285°	123'	RA
33B	10	285°	123'	RA
34	3	280°	120'	RA
35	3	292°	120'	RA
36	20	307°	148'	RA
37	30	315°	150'	RA
38	10	9°	128'	RA
39	10	9°	128'	RA
40	20	0°	123'	RA
41	3	345°	155'	RA
42	3	347°	151'	RA
43	3	340°	155'	RA
44	3	343°	155'	RA
45	10	67°	118'	RA
46	3	67°	120'	RA

SNUBBER #	SIZE (kips)	LOCATION	ELEVATION	*SHUTDOWN ACCESSIBILITY CODE
		Nuclear Boiler Sys.		
MVVH-23	10	270°	144'	DA
24	10	270°	144'	DA
25	10	270°	144'	DA
27	10	225°	124'	RA
28	10	250°	128'	RA
29	10	250°	128'	RA
31	10	140°	149'	RA
32	10	140°	148'	RA
33	10	140°	145'	RA
35	10	90°	126'	RA
36	10	90°	126'	RA
37	10	90°	126'	RA
FDH-11	10	16°	147'(2)	RA
12	10	0°	147 (2)	RA
13	10	40°	148'	RA
14	10	75°	148'	RA
15	20	53°	148'	RA
16	10	79°	146'	RA
17	10	280°	148'	RA
18	10	281°	146'	RA
19	10	98°	150'	RA
21	3	210°&150°	165' (2)	RA
22	3	120°&240°	165' (2)	RA
23	10	60°	164'	RA
24	10	30°	167'	RA
25	10	330°	164'	RA
26	10	310°	167'	RA
DFDH-28	3	4'SR7-8'ERA	132'	RA
30	3	4'NR7-8'ERA	132'	RA
32	3	14°	132'	RA
36	3	0°	132'	RA
		RHR SYSTEM		
S-1	30	270°	141'	RA
2	30	270°	141'	RA
4	30	210°	141'	RA
5	30	240°	141'	RA
15	20	185°	139'	DA
SM-1	30	180°	134'	RA
2	10	180°	140'	DA
3	30	225°	146'	N
4	30	225°	146'	N
8	30	90°	146'	RA
RHRH-255*	3	6'NR7-13'ERF	207'	DA
256*	3	6'NR7-13'ERF	207'	DA
257*	3	6'NR7'13'ERF	200'	DA
258*	3	2'NR7-13'ERF	194'	DA

*Mechanical type snubber; provisions of T.S. 4.6.L. do not apply

SS-A1	35	315°	123'	RA
A2	50	315°	123'	RA
A3	50	315°	123'	RA
A4	50	310°	131'	RA
A5	50	320°	131'	RA
A6	50	315°	134'	RA
A7	21	15°	134'	RA
A8	35	10°	134'	RA
A13	35	270°	145'	RA
A14	50	270°	122'	RA
A12	35	270°	145'	RA
SS-B1	21	140°	120'	RA
B2	50	135°	123'	RA
B3	50	135°	123'	RA
B4	50	145°	131'	RA
B5	50	135°	131'	RA
B6	50	135°	137'	DA
B7	21	185°	140'	RA
B8	35	180°	140'	RA
B12	35	90°	145'	RA
B13	35	90°	145'	RA
B14	50	90°	116'	RA

SNUBBERS ACCESSIBLE DURING REACTOR OPERATION

CORE SPRAY SYSTEM

CSH-75	3	10'NR3-7'WRL	125'	RA
71	10	7'NR13-10'WRL	121'	RA
79	10	2'NR9-7'WRH	172'	RA

HPCI SYSTEM

HPCIH-9	20	13'SR1-6'ERC	88'	RA
13	50	7'SR1-2'WRL	94'	RA
HPSEH-2	10	12'NR2-10'WRL	92'	RA
8	10	6'NR2-4NRC	112' (2)	RA
12	10	5'NR3-3'ERF	123'8"	RA
13	10	4'NR3-3'ERF	123'6" (2)	RA
17	10	5'NR3-14'ERF	123'6"	RA
57	3	1'SR1-18'WRL	99 1/2'	RA
58	3	4'SR1-18'WRL	99'	RA
60	3	4'NR2-4' NRC	120' (2)	RA
61	10	3'NRS-11'	123'	RA
62	10	3'NRS-11'	123'	RA

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d. Automatic initiation of each branch of the standby gas treatment system.

e. Manual operability of the bypass valve for filter cooling.

3.7.B.2. Performance Requirements

a. The results of the in-place DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal absorber banks shall show $\geq$ 99% DOP removal when tested in accordance ANSI N510-1975.

b. The results of laboratory carbon sample analysis shall show $\geq$ 90% radioactive methyl iodide removal when tested in accordance with RDT-M16-1T (80^{o}C, 95% R.H.).

c. Fans shall be shown to operate within $\pm$10% design flow when tested in accordance with ANSI N510-1975.

2. Filter Testing

a. The tests and analysis shall be performed at least once per operating cycle, not to exceed 18 months, or after every 720 hours of system operation or following painting, fire or chemical release in any ventilation zone communicating with the system.

b. DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal absorber bank or after any structural maintenance on the system housing.

d. Each circuit shall be operated with the heaters on at least 10 hours every month.

Table 3.7-4

Primary Containment Testable Isolation Valves

Penetration Number	Valve Designation	Notes
X-7A	B21-F022A & F028A Main Steam Isolation Valves	(1) (2) (3) (5) (9)
X-7B	B21-F022B & F028B " " " "	"
X-7C	B21-F022C & F028C " " " "	"
X-7D	B21-F022D & F028D " " " "	"
X-8	B21-F016 & F019	(1) (2) (4) (5) (9)
X-9A	B21-F010A	(1) (2) (3) (5) (10)
X-9A	B21-F032A	"
X-9A	E41-F006, F007, & F008	(1) (2) (4) (5) (9)
X-9B	B21-F010B	(1) (2) (3) (5) (10)
X-9B	B21-F032B	"
X-9B	E51-F012, F013, & F022	(1) (2) (4) (5) (9)
X-9B	G31-F042	(1) (2) (4) (5) (10)
X-10	E51-F007, F008	(1) (2) (4) (5) (9)
X-11	E41-F002 & F003	"
X-12	E11-F008 & F009	"
X-13A	E11-F015A & F017A	"
X-13B	E11-F015B & F017B	"
X-14	G31-F001 & F004	"
X-16A	E21-F004A & F005A	"
X-16B	E21-F004B & F005B	"
X-17	E11-F022 & F023	"
X-18	G11-F019 & F020	"
X-19	G11-F003 & F004	"

3.11 FUEL RODS

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figure 3.11-1, sheets 1 thru 3. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, then reduce reactor power to less than 25% of rated thermal power within the next four (4) hours. If the limiting condition for operation is restored prior to expiration of the specified time interval, then further progression to less than 25% of rated thermal power is not required.

B. Linear Heat Generation Rate (LHGR)

During power operation, the LHGR as a function of core height shall not exceed the limiting value shown in Figure 3.11-2, sheets 1 and 2. If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the

4.11 FUEL RODS

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

B. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.

3.11.B. Linear Heat Generation Rate (LHGR) (Continued)

LHGR is not returned to within the prescribed limits within two (2) hours, then reduce reactor power to less than 25% of rated thermal power within the next four (4) hours. If the limiting condition for operation is restored prior to expiration of the specified time interval, then further progression to less than 25% of rated thermal power is not required.

C. Minimum Critical Power Ratio (MCPR)

The MCPR limit is specified as a function of fuel average exposure. From BOC2 to 1500 MWD/t before EOC2 the MCPR limit is 1.20 for 7x7 and 1.23 for 8x8 fuels. From 1500 MWD/t before EOC2 to EOC2 the MCPR limit is 1.25 for 7x7 and 1.32 for 8x8 fuels. During power operation, MCPR shall be as above at rated power and flow. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, then reduce reactor power to less than 25% of rated thermal power within the next four (4) hours. If the Limiting Condition for Operation is restored prior to expiration of the specified time interval, then further progression to less than 25% of rated thermal power is not required. For core flows other than rated the MCPR shall be K_f times the MCPR value applicable above, where K_f is as shown in Figure 3.11-3.

D. Reporting Requirements

If any of the limiting values identified in Specifications 3.11.A., B., or C. are exceeded, a Reportable Occurrence report shall be submitted.

If the corrective action is taken, as described, a thirty-day written report will meet the requirements of this specification.

4.11.C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.F.

3.12 MAIN CONTROL ROOM ENVIRONMENTAL SYSTEM

Applicability

The Limiting Conditions for Operation apply to the operating status of the main control room environmental system.

Objective

The objective of the Limiting Conditions for Operation is to assure the availability of the main control room environmental system under conditions for which its capability is required to protect plant operators.

Specifications

A. Ventilations System Operability Requirements

1. Operability Requirement

a. Two independent control room air treatment systems shall be operable at all times when secondary containment integrity is required. However, from and after the date that one circuit of the control room air treatment system is made or found to be inoperable for any reason, reactor operation or refueling operation is permissible only during the succeeding seven days. If the system is not made fully operable within 7 days, reactor shutdown shall be initiated and the reactor shall be in cold shutdown within the next 36 hours and irradiated fuel handling operations shall be terminated within 2 hours.

b. The control room ventilation system shall be capable of maintaining the control room at a positive pressure of 1/8 inches W.G. relative to the turbine building when in the pressurization mode.

4.12 MAIN CONTROL ROOM ENVIRONMENTAL SYSTEM

Applicability

The Surveillance Requirements apply to the periodic tests and examinations of the main control room environmental system.

Objective

The objective of the Surveillance Requirements is to verify the operability, availability or efficiency of the main control room environmental system under conditions for which its capability is required to protect plant operators.

Specifications

A. Ventilation System Tests

At least once per operating cycle, not to exceed 18 months, the following shall be demonstrated:

a. The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at system design flow rate ($\pm$10%).

b. Automatic initiation of the control room air treatment system.

c. Each circuit shall be operated for at least 15 minutes each month.

3.12.A.2. Performance Requirements

a. The results of the in-place DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal absorber banks shall show $\geq$99% DOP removal and $\geq$99% halogenated hydrocarbon removal, respectively when tested in accordance with ANSI N510-1975.

b. The results of laboratory carbon sample analysis shall show $\geq$90% radioactive methyl iodide removal when tested in accordance with RDT-M16-1T (25°C, 95% R.H.).

c. Fans shall be shown to operate within $\pm$10% design flow when tested in accordance with ANSI N510-1975.

4.12.A.2 Filter Testing

a. The tests and analysis shall be performed at least once per operating cycle, not to exceed 18 months, or after every 720 hours of system operation or following painting, fire or chemical release in any ventilation zone communicating with the system.

b. DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank of after any structural maintenance on the system housing.

B. Isolation Valve Operability and Closing Time

The control room air intake isolation valves shall be operable whenever the ventilation system is required to be operable by Specification 3.12.A and shall be required to be closed within seven seconds from receipt of an isolation signal. One valve may be considered inoperable for a period not to exceed seven days.

B. Isolation Valve Testing

The control room air intake isolation valves shall be tested for operability every three months and for the required closing time at least once per operating cycle.

3.12 MAIN CONTROL ROOM ENVIRONMENTAL SYSTEM

The control room air treatment system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation or pressurization conditions.

A. Ventilation System Operability Requirements

The control room air treatment system operates on emergency power and is designed to filter the control room atmosphere for intake air and/or recirculation during control room isolation or pressurization conditions. The control room air treatment system is designed to automatically start upon receipt of an initiation signal and to align the system dampers to either provide for pressurization of the control room or isolation of the control room, depending on the source of the initiating signal.

Pressurization will be initiated upon receipt of any one of the following signals: High radiation at control room intake, LOCA signal from Unit 1 or 2, main steam line high radiation from Unit 1 or 2, main steam line high flow from Unit 1 or 2, or refueling floor high radiation from Unit 1 or 2. In this mode the normal control room exhaust fan is stopped and outside air is taken in through one of the charcoal filters to pressurize the control room with respect to the surrounding turbine building.

Isolation of the control room will be initiated upon receipt of a high chlorine concentration at the control room intake signal. In this mode the control room is isolated from the normal outside air intake and the control room atmosphere is recirculated through one of the charcoal filters. The normal control room exhuast fan is stopped in this mode also. If one system is found to be inoperable, there is no immediate threat to personnel in the control room. Therefore, reactor operation or refueling operation may continue for a limited period of time while repairs are being made.

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP, respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the fans significiantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performances are as specified, the calculated doses would be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50.

B. Isolation Valve Operability and Closing Time

The control room air intake isolation valves would receive signals to close from a high chlorine concentration in the control room air intake duct. A valve closing time requirement of seven seconds or less assures that any chlorine detected in the intake duct, and transported at normal air intake rate would not reach the first isolation valve prior to the valve being closed. The second isolation valve offers a considerably longer transport time should the first valve fail to close.

C. Radiation Monitors

At least one channel (detector) in the control room air intake radiation monitoring system must be operable at all times for indication-alarm of radioactivity being drawn into the main control room. Main control room intake air filtration is required when a trip signal from the detectors is given via failure or pressurization signals from both channels or a failure signal in one channel and a pressurization signal in the other channel.

D. Shutdown Requirements

Shutdown requirements are based on the need to ensure habitability for operations personnel during normal plant operation and subsequent to a postulated design basis accident.

E. Chlorine Monitors

At least one channel (detector) in the control room air intake chlorine monitoring system must be operable at all times for indication-alarm of chlorine being drawn into the control room. Main control room isolation from outside air intake is required upon receipt of a chlorine detection signal or a failure of the chlorine monitor in either channel. The chlorine monitors are also discussed in Reference 2.

4.12 MAIN CONTROL ROOM ENVIRONMENTAL SYSTEM

A. Ventilation System Tests

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A test frequency of once per operating cycle established system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52. The charcoal adsorber efficiency test procedures shall allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If the iodine removal efficiency test results are unacceptable, all adsorbent in the system should be replaced. Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

Operation of the system every month will demonstrate operability of the filters and adsorber system. Operation for 15 minutes demonstrates operability and removes the moisture build-up during testing.

If painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis should be performed as required for operational use.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

B. Isolation Valve Testing

The operability test for the control room air intake isolation valves consists of a closing-opening cycle. Since these valves are not expected to be used often, they are also not expected to degrade in performance, thus the closing time verification once per operating cycle is adequate.

C. Radiation and Chlorine (3.12.5) Monitors

Bases for the control room air intake radiation monitors are specifically discussed in Bases for Limiting Conditions for Operation, Specification 3.2.H.4, and are generally discussed in Bases for Surveillance Requirements, Specification 4.2. The test interval of the chlorine monitors is based on the same criteria as for the radiation monitors.

D. References

1. Deleted.

2. FSAR Questions 11.6.3 and 11.6.4.

3. ANSI Standard N101.1, 1972, "Efficiency Testing of Air-Cleaning Systems Containing Devices for Removal of Particulates".

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.

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6.13 HIGH RADIATION AREA

6.13.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

a. 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 issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.13.1(a) above, and in addition locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the shift supervisor on duty.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 51 TO FACILITY OPERATING LICENSE NO. DPR-57

GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION
MUNICIPAL ELECTRIC ASSOCIATION OF GEORGIA
CITY OF DALTON, GEORGIA

EDWIN I. HATCH NUCLEAR PLANT UNIT NO. 1

DOCKET NO. 50-321

Introduction

By letters dated August 4, 1976 (as supplemented by letter dated April 26, 1977), November 16, 1977, December 2, 1977, December 9, 1977 and February 3, 1978, Georgia Power Company (GP or licensee) proposed amendments to Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant Unit No. 1. The proposed amendments consist of changes to the Technical Specifications and involve: (1) instrumentation setpoint changes resulting from startup testing, (2) the Hatch Unit No. 1 Containment Leak Rate Test Program, (3) deletion of specific snubbers from the Table of Safety Related Shock Suppressors, (4) revision of action statements associated with exceeding LHGR, APLHGR and MCPR limits, (5) revision to the performance requirements of the SGTS and Control Room filter systems and (6) modification of the Control Room ventilation system. In the license amendment supported by this evaluation, the Commission has deleted the respiratory protection program in accordance with the revocation provision of the current Technical Specifications.

Evaluation

1. Instrumentation Setpoints

   Prior to initial startup of Hatch Unit No. 1, several instruments which monitored air temperature and radiation levels throughout the plant had trip setpoints specified as either "90°F + ambient" or "to be determined during startup". Georgia Power Company (GP) has reviewed data accumulated during startup and has proposed definite trip settings for these instruments.

The current Technical Specifications state that the trip settings for HPCI Equipment Room Temperature, RCIC Equipment Room Temperature and Suppression Chamber Area Air Temperature will be ≤90°F + Ambient. The intent of this setting on these particular instruments is to provide a control signal which would be indicative of gross steam leakage. Such trips are diverse from other instrumentation which performs the same functions, e.g., high steam flow on HPCI and RCIC. The licensee stated that from the results of the startup testing, the equipment area cooling system reduced the ambient temperature to 85°F; thus, he proposed a trip setting of ≤175°F. We have reviewed the licensee's submittal and concluded that the proposed revision is comparable to and more specific than that currently specified and is acceptable. It is noted that this trip setting is consistent with the staff position for such a setpoint[1].

The current Technical Specifications do not set forth a specific trip setpoint for Suppression Chamber Area Differential Air Temperature. The purpose of a differential air temperature measurement between the inlet and outlet ducts which ventilates the suppression chamber area is to detect a steam line break in the exhaust line from either the HPCI or RCIC system. The functions of a trip for high differential temperature are to close the isolation valves and to trip the turbines associated with these systems. The licensee proposed a trip setpoint of ≤50°F based on his operating experience of the observed differential air temperature for the HPCI and RCIC Equipment Rooms. We have reviewed the licensee's submittal and determined that a 50°F differential setpoint is high enough to preclude unnecessary isolation in the event of small steam leaks, such as occur occasionally in valve packing glands, flanges or fittings. Automatic isolation as a result of small leaks is undesirable since the HPCI & RCIC should remain available to perform their functions in the presence of leaks which have no significant consequences. The 50°F setting is also low enough to yield positive indication of gross steam leakage for which automatic isolation of these systems is warranted. The 50°F setting is consistent with the staff position for such a setting[1] and is acceptable.

The current Technical Specifications do not set forth specific trip setpoints for the radiation monitors associated with the Reactor Building Exhaust Vent, the Control Room Intake and the Main Steam Line. The purpose of these instruments and their associated setpoints is to limit the release of radioactivity via the normal ventilation path, limit the amount of radioactivity entering the control room, and preclude the release of radioactivity from the main condenser. Based on data accumulated during startup, the licensee proposed trip setpoints at 3 times normal, full power background levels. We have reviewed the licensee's data on background levels and conclude, based on the staff's position (Reference 1) of 3 times background level setpoints for these instruments, that the licensee's proposed specifications are acceptable.

2. Containment Leakage Test Program

The licensee's submittal dated November 16, 1977 requested changes to the Hatch Unit No. 1 Containment Leak Rate Test Program. Included in this request was the deletion of RCIC and HPCI Turbine exhaust drain line valves from the containment leakage test program. We have reviewed the licensee's submittal and determined that these valves (E51-F025, E51-F026, E41-F028 and E41-F029) do not perform a containment leakage barrier function for their respective penetration. Each valve is located downstream from two containment isolation valves that do serve as leakage limiting barriers and are Type C tested in accordance with the provisions of 10 CFR Part 50, Appendix J. Therefore, deletion of the named valves from the containment leak rate test program is acceptable.

Also included in the licensee's request was a proposal to revise the reporting requirements for the leakage of the RHR suction line valves and certain other valves that the licensee determined to be sealed from the primary containment atmosphere. The licensee stated that such revision was in accordance with the requirements of 10 CFR 50, Appendix J, Article III.A.1.d. The staff determined that such a proposal constitutes an exemption to 10 CFR Apendix J and is currently under review as part of the staff's evaluation of the Hatch Unit No. 2 request for exemption from the requirements of Appendix J. Upon completion of that review, revision to the Hatch Unit No. 1 Technical Specifications, as appropriate, will be authorized and supported by a separate safety evaluation.

3. Snubbers

The amendment proposed by the licensee would revise the Table of safety related shock suppressors (snubbers) by deleting six snubbers. During a discussion with representatives of GP on March 6, 1978, the staff was advised that the proposed amendment is essentially an editorial correction to the Table of safety related snubbers. Prior to the issuance of Amendment No. 37 to Operating License No. DPR-57, snubber designated SS-25 was deleted and HPCIH-12 was changed to a spring hanger after a reanalysis by Bechtel Corporation. Also, snubbers RHRH-255 through RHRH-258 are of the mechanical rather than hydrualic type.

We have independently verified that shock suppressors designated SS-25 and HPCIH-12 were (and still are) non-existent as safety related shock suppressors prior to the issuance of Amendment No. 37 and that their deletion from the Table of Safety Related Shock Suppressors is an editorial correction which is acceptable. We have determined that the shock suppressors designated RHRH-255 through RHRH-258, are of the mechanical rather than hydraulic type, and are required for assurance that structural integrity of the RHR system would be maintained during dynamic loading. Thus, their deletion from the limiting condition for operation (Specification 4.6.L) could not be authorized without staff review and approval of a justification for deletion. However, since Specification 4.6.L relates solely to surveillance of hydraulic snubbers, we have annotated the Table of Safety Related Shock Absorbers to indicate that snubbers RHRH-255 through RHRH-258 are of the mechanical type and that Specification 4.6.L does not apply. At the present time there are no surveillance requirements mechanical snubbers. However, possible future surveillance requirements are under generic consideration in the category A-13 task action plan. This change to the licensee's submittal was discussed with representatives of GP and they are in agreement with the staff's recommended change.

4. LHGR, APLHGR and MCPR Limits

In the event that a Linear Heat Generation Rate (LHGR), Average Planar Linear Heat Generation Rate (APLHGR), or Minimum Critical Power Ratio (MCPR) limit is exceeded, the existing Technical Specifications require that the reactor will be placed in Cold Shutdown within 36 hours following the event if the parameter exceeded is not returned to within prescribed limits within two hours. The licensee indicated that there does not appear to be a firm basis for the requirement to bring the reactor to cold shutdown. The licensee's proposed amendment would retain the time period allowed to bring the parameter to within prescribed limits but would replace the requirement to proceed to cold shutdown within 36 hours with a requirement to reduce reactor power to less than 25% within the succeeding four hours. The licensee further indicated that the revised action statement is consistent with the staff's Standard Technical Specification for boiling water reactors.

We have reviewed the licensee's submittal and agree with his conclusion. Previous staff reviews have indicated that even under low reactor pressure (<800 psia) or low core flow (<10%) conditions, a core thermal power limit of 25% of rated power is conservative with respect to core peaking factors. There are no plant unique factors for Hatch Unit No. 1 that would require operating limitations more conservative than those specified in the staff's standard Technical Specifications for Boiling Water Reactors[1]. Therefore, the licensee's request is acceptable, as proposed.

5. <u>SGTS and Control Room Filters</u>

The current Technical Specifications relative to the performance requirements for the Standby Gas Treatment System and Control Room filters specifies that tests of laboratory sample analysis for methyl iodide removal should be tested in accordance with ANSI N510-1975. Since the ANSI standard is to be used for in-place DOP and halogenated hydrocarbon tests, and does not apply to carbon sample analysis, the licensee proposed the use of RDT-M16-IT as the correct standard. The request specified the test conditions of 80°C, 95% R.H for the SGTS and 25°C, 70% R.H for the Control Room Environmental System.

We have reviewed the licensee's proposal and determined that the use of RDT-M16-IT is an acceptable test method and is consistent with the staff's position as set forth in Regulatory Guide 1.52[2]. The test conditions for the SGTS are also consistent with this staff position and represents a conservative evaluation of methyl iodide removal.

The licensee's submittal indicated that the Control Room filters are not equipped with moisture elimenators and that the carbon bed heaters are de-energized when the fans are energized. During discussions with the licensee, he indicated that humidity is controled by heaters located upstream of the preheaters in each filter train. However, the staff has determined that the test conditions for these filters should also be at 95% R.H (in lieu of 70%) to assure a conservative evaluation of performance. This position is consistent with that set forth in R.G. 1.52. The revision to the licensee's submittal was discussed with representatives of GP and they do not object. Accordingly, we find the licensee's proposed change, as amended by the staff, to be acceptable.

6. Control Room Ventilation System

The main control room environmental control system is designed to maintain the control room within the thermal and air quality limits required for operation of the plant controls, and interrupted safe occupancy of required manned areas during normal operation, shutdown and post-accident conditions. The main control room is shared between Hatch Unit No. 1 and Hatch Unit No. 2. During the design of Hatch Unit No. 2, the design bases for the control room ventilation system were changed to add a pressurization mode of operation under certain plant conditions. These conditions include: high radiation at control room intake, loss of coolant accident signal from either Unit, main steam line high radiation from either Unit, main steam line flow from either Unit or refueling floor high radiation from either Unit. The incorporation of the pressurization mode will improve control room habitability by minimizing control room radiation levels. Based on our evaluation[3] of the control room ventilation system for Hatch Unit No. 2, we find the licensee's proposal to incorporate this mode of operation into the Technical Specifications for Hatch Unit No. 1 to be acceptable.

Respiratory Protection Program

By our letter dated July 29, 1977, we advised Georgia Power Company (GP) of an amended Section 20.103 of 10 CFR 20 which became effective on December 29, 1976. GP was further advised that one effect of the revision is that in order to receive credit for use of respiratory protection equipment at their facility after December 28, 1977, such use must be as stipulated in Regulatory Guide 8.15 rather than as specified in the current Technical Specifications for Hatch Unit No. 1. In view of the provisions of Section 6.11 of the Technical Specifications which require conformance with 10 CFR 20, and in the absence of written objection from GP, this amendment executes the revocation provision of the current specifications on respiratory protection by deleting Section 6.12.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Dated: April 6, 1978

REFERENCES

1. Standard Technical Specifications for General Electric Boiling Water Reactors; Revision of August 15, 1976, NUREG-0123.

2. Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for Engineered-Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants", Revision 1, July 1976.

3. Report to the Advisory Committee on Reactor Safeguards by the Office of Nuclear Reactor Regulation, NRC, in the matter of Georgia Power Company, et al; Edwin I. Hatch Nuclear Plant, Unit No. 2, Docket No. 50-366, January 1978.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-321

GEORGIA POWER COMPANY, ET AL

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 51 to Facility Operating License No. DPR-57 issued to Georgia Power Company, Oglethorpe Electric Membership Corporation, Municipal Electric Association of Georgia and City of Dalton, Georgia, which revised Technical Specifications for operation of the Edwin I. Hatch Nuclear Plant, Unit No. 1, located in Appling County, Georgia. The amendment is effective as of its date of issuance.

The amendment consists of changes to the Technical Specifications and involve: (1) instrumentation setpoint changes resulting from startup testing, (2) the Hatch Unit No. 1 Containment Leak Rate Test Program, (3) deletion of specific snubbers from the Table of Safety Related Shock Suppressors, (4) revision of action statements associated with exceeding LHGR, APLHGR and MCPR limits, (5) revision to the performance requirements of the SGTS and Control Room filter systems, (5) modification of the Control Room ventilation system and (7) deletion of the respiratory protection program.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated August 4, 1976 (as supplemented April 26, 1977), November 16, 1977, December 2 and 9, 1977 and February 3, 1978, (2) Amendment No. 51 to License No. DPR-57 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Appling County Public Library, Parker Street, Baxley, Georgia 31513. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 6 day of April 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors