January 13, 1988

Docket No.: 50-366

Mr. James P. O'Reilly
Senior Vice President - Nuclear Operations
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. O'Reilly:

Subject: Issuance of Amendment No. 88 to Facility Operating License NPF-5 - Edwin I. Hatch Nuclear Plant, Unit 2 (TAC 65641)

The Commission has issued the enclosed Amendment No. 88 to Facility Operating License NPF-5 for the Edwin I. Hatch Nuclear Plant, Unit 2. The amendment consists of changes to the Technical Specifications in response to your application dated March 20, 1987, and supplemented November 23, 1987.

The amendment modifies the Technical Specifications to permit a temporary increase in the main steam line high radiation scram and isolation setpoints to facilitate the testing of hydrogen addition water chemistry.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

/S/

Lawrence P. Crocker, Project Manager
Project Directorate II-3
Division of Reactor Projects-I/II

Enclosures:
1. Amendment No. 88 to NPF-5
2. Safety Evaluation

cc w/enclosures:
See next page

	LC	LC
PD#II-3/DRP-I/II	PD#II-3/DRP-I/II	PD#II-3/DRP-I/II
MRood/mac	LCrocker	Acting PD
12/28/87	12/29/87	12/29/87

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 88
License No. NPF-5

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 2 (the facility) Facility Operating License No. NPF-5 filed by Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, (the licensee) dated March 20, 1987, and supplemented November 23, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-5 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 88 , 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 its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Lawrence P. Crocker, Acting Director
Project Directorate II-3
Division of Reactor Projects-I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 13, 1988

PD#II-3/DRP-I/II	PD#II-3/DRP-I/II	OGC-Bethesda	PD#II-3/DRP-I/II
MRood/mac	LCrocker		Acting PD
12/28/87	12/29/87	12/30/87	12/29/87

ATTACHMENT TO LICENSE AMENDMENT NO. 88

FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

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 Page	Insert Page
2-4	2-4
3/4 3-2	3/4 3-2
3/4 3-5	3/4 3-5
3/4 3-11	3/4 3-11
3/4 3-15	3/4 3-15
3/4 3-16	3/4 3-16
3/4 3-58a	3/4 3-58a
3/4 3-58b	3/4 3-58b
3/4 3-58c	3/4 3-58c

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High (2C51-K601 A,B,C,D,E,F,G,H)	≤ 120/125 divisions of full scale	≤ 120/125 divisions of full scale
2. Average Power Range Monitor: (2C51-K605 A,B,C,D,E,F)		
a. Neutron Flux-Upscale, 15%	≤ 15/125 divisions of full scale	≤ 20/125 divisions of full scale
b. Flow Referenced Simulated Thermal Power-Upscale	≤ (0.58 W + 59% - 0.58ΔW)** with a maximum ≤ 113.5% of RATED THERMAL POWER	≤ (0.58 W + 62% - 0.58ΔW)** with a maximum ≤ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux-Upscale, 118%	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High (2B21-N678 A,B,C,D)	≤ 1054 psig	≤ 1054 psig
4. Reactor Vessel Water Level - Low (Level 3) (2B21-N680 A,B,C,D)	≥ 10 inches above instrument zero*	≥ 10 inches above instrument zero*
5. Main Steam Line Isolation Valve - Closure (NA)	≤ 10% closed	≤ 10% closed
6. Main Steam Line Radiation - High (2D11-K603A,B,C,D)	≤ 3 x full-power background***	≤ 3 x full-power background***
7. Drywell Pressure - High (2C71-N650A,B,C,D)	≤ 1.92 psig	≤ 1.92 psig

*See Bases Figure B 3/4 3-1.

** W = Total loop recirculation flow rate in percent of rated. Rated loop recirculation flow is equal to 34.2 MLB/hr.

ΔW = Maximum measured difference between two-loop and single-loop drive flow for the same core flow in percent of rated recirculation flow for single-loop operation. The value is zero for two-loop operation.

***Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater than 20% rated power, the normal full-power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of hydrogen injection and prior to establishing reactor power levels below 20% rated power.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(a)	ACTION
1. Intermediate Range Monitors: (2C51-K601, A, B, C, D, E, F, G, H)			
a. Neutron Flux - High	2[c], 5[b]	3	1
	3, 4	2	2
b. Inoperative	2,5[b]	3	1
	3, 4	2	2
2. Average Power Range Monitor: (2C51-K605 A, B, C, D, E, F)			
a. Neutron Flux - Upscale, 15%	2, 5	2	1
b. Flow Referenced Simulated Thermal Power - Upscale	1	2	3
c. Fixed Neutron Flux - Upscale, 118%	1	2	3
d. Inoperative	1, 2, 5	2	4
e. Downscale	1	2	3
f. LPRM	1, 2, 5	(d)	NA
3. Reactor Vessel Steam Dome Pressure - High (2B21-N678 A, B, C, D)	1, 2[e]	2	5
4. Reactor Vessel Water Level - Low (Level 3) (2B21-N680 A, B, C, D)	1, 2	2	5
5. Main Steam Line Isolation Valve - Closure (NA)	1[f]	4	3
6. Main Steam Line Radiation - High (2D11-K603 A, B, C, D)	1, 2[e], [j]	2	6
7. Drywell Pressure - High (2C71-N650 A, B, C, D)	1, 2[g]	2	5

ACTION 9 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, lock the reactor mode switch in the Shutdown position within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS or positive reactivity changes and fully insert all insertable control rods within one hour.

TABLE NOTATIONS

a. A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

b. The "shorting links" shall be removed from the RPS circuitry during CORE ALTERATIONS and shutdown margin demonstrations performed in accordance with Specification 3.10.3.

c. The IRM scrams are automatically bypassed when the reactor vessel mode switch is in the Run position and all APRM channels are OPERABLE and on scale.

d. An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than eleven LPRM inputs to an APRM channel.

e. These functions are not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed.

f. This function is automatically bypassed when the reactor mode switch is in other than the Run position.

g. This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

h. With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.11.1 or 3.9.11.2.

i. These functions are bypassed when turbine first stage pressure is ≤250* psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

j. Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater than 20% rated power, the normal full-power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of hydrogen injection and prior to establishing reactor power levels below 20% rated power.

*Initial setpoint. Final setpoint to be determined during startup testing.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1. Low (Level 3) (2B21-N680 A, B, C, D)	2, 6, 10, 11, 12	2	1, 2, 3	20
2. Low-Low (Level 2) (2B21-N682 A, B, C, D)	5, *	2	1, 2, 3	20
3. Low-Low-Low (Level 1) (2B21-N681 A, B, C, D)	1	2	1, 2, 3	20
b. Drywell Pressure - High (2C71-N650 A, B, C, D)	2, 6, 7, 10, 12, *	2	1, 2, 3	20
c. Main Steam Line				
1. Radiation - High (2D11-K603 A, B, C, D)	1, 12, (d)	2	1, 2, 3,[k]	21
2. Pressure - Low (2B21-N015 A, B, C, D)	1	2	1	22
3. Flow - High (2B21-N686 A, B, C, D) (2B21-N687 A, B, C, D) (2B21-N688 A, B, C, D) (2B21-N689 A, B, C, D)	1,	2/line	1, 2, 3	21
d. Main Steam Line Tunnel Temperature - High (2B21-N623 A, B, C, D) (2B21-N624 A, B, C, D) (2B21-N625 A, B, C, D) (2B21-N626 A, B, C, D)	1	2/line[e]	1, 2, 3	21
e. Condenser Vacuum - Low (2B21-N056 A, B, C, D)	1	2	1, 2,[f], 3[f]	23
f. Turbine Building Area Temperature - High (2U61-R001, 2U61-R002, 2U61-R003, 2U61-R004)	1	2[e]	1, 2, 3	21
g. Drywell Radiation - High (2D11-K621 A, B)	[j]	1	1, 2, 3	29

TABLE 3.3.2-1 (Conti)

ISOLATION ACTUATION INSTRUMENTATION

ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 21 - Be in at least STARTUP with the main steam line isolation valves closed within 2 hours or be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 22 - Be in at least STARTUP within 2 hours.

ACTION 23 - Be in at least STARTUP with the Group 1 isolation valves closed within 2 hours or in at least HOT SHUTDOWN within 6 hours.

ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

ACTION 25 - Isolate the reactor water cleanup system.

ACTION 26 - Close the affected system isolation valves and declare the affected system inoperable.

ACTION 27 - Verify power availability to the bus at least once per 12 hours or close the affected system isolation valves and declare the affected system inoperable.

ACTION 28 - Close the shutdown cooling supply and reactor vessel head spray isolation valves unless reactor steam dome pressure ≤ 145 psig.

ACTION 29 - Either close the affected isolation valves within 24 hours or be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the next 30 hours.

NOTES

* Actuates the standby gas treatment system.

** When handling irradiated fuel in the secondary containment.

a. See Specification 3.6.3, Table 3.6.3-1 for valves in each valve group.

b. A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

c. With a design providing only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.

d. Trips the mechanical vacuum pumps.

e. A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

f. May be bypassed with all turbine stop valves closed.

g. Closes only RWCU outlet isolation valve 2G31-F004.

h. Alarm only.

i. Adjustable up to 60 minutes.

j. Isolates containment purge and vent valves.

k. Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater than 20% rated power, the normal full-power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of hydrogen injection and prior to establishing reactor power levels below 20% rated power.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1. Low (Level 3)	≥ 10 inches*	≥ 10 inches*
2. Low Low (Level 2)	≥ -47 inches*	≥ -47 inches*
3. Low Low Low (Level 1)	≥ -113 inches*	≥ -113 inches*
b. Drywell Pressure - High	≤ 1.92 psig	≤ 1.92 psig
c. Main Steam Line		
1. Radiation - High	≤ 3 × full-power background**	≤ 3 × full-power background**
2. Pressure - Low	≥ 825 psig	≥ 825 psig
3. Flow - High	≤ 138% rated flow	≤ 138% rated flow
d. Main Steam Line Tunnel Temperature - High	≤ 194°F	≤ 194°F
e. Condenser Vacuum - Low	≥ 7" Hg vacuum	≥ 7" Hg vacuum
f. Turbine Building Area Temp.-High	≤ 200°F	≤ 200°F
g. Drywell Radiation - High	≤ 138 R/hr	≤ 138 R/hr
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Exhaust Radiation - High	≤ 60 mr/hr	≤ 60 mr/hr
b. Drywell Pressure - High	≤ 1.92 psig	≤ 1.92 psig
c. Reactor Vessel Water Level - Low Low (Level 2)	≥ -47 inches*	≥ -47 inches*
d. Refueling Floor Exhaust Radiation - High	≤ 20 mr/hr	≤ 20 mr/hr

*See Bases Figure B 3/4 3-1.

**Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater than 20% rated power, the normal full power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of hydrogen injection and prior to establishing reactor power levels below 20% rated power.

TABLE 3.3.6.7-1 (SHEET 1 OF 2)

MCRECS ACTUATION INSTRUMENTATION

	TRIP FUNCTION	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(a)(b)	APPLICABLE OPERATIONAL CONDITION	ACTION
1.	Reactor Vessel Water Level - Low Low Low (Level 1) (c) 2B21-N691 A, B, C, D	2	1, 2, 3	52
2.	Drywell Pressure - High (c) 2E11-N694 A, B, C, D	2	1, 2, 3	52
3.	Main Steam Line Radiation - High (c) 2D11-K603 A, B, C, D	2	1, 2, 3,[e]	53
4.	Main Steam Line Flow - High (c) 2B21-N686 A, B, C, D 2B21-N687 A, B, C, D 2B21-N688 A, B, C, D 2B21-N689 A, B, C, D	2/line	1, 2, 3	53
5.	Refueling Floor Area Radiation - High (c) 2D21-K002 A, D	1	1, 2, 3, 5, *	54
6.	Control Room Air Inlet Radiation - High (c) 1Z41-R615 A, B	1	1, 2, 3, 5, *	54
7.	Control Room Air Inlet Chlorine Level - High (d) 1Z41-N022 A, B	1	1, 2, 3, 4, 5	55

MCRECS ACTUATION INSTRUMENTATION

ACTION

ACTION 52 - Take the ACTION required by Specification 3.3.3.

ACTION 53 - Take the ACTION required by Specification 3.3.2.

ACTION 54 -

a. With one of the required radiation monitors inoperable, restore the monitor to OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of the MCRECS in the pressurization mode of operation.

b. With no radiation monitors OPERABLE, within 1 hour initiate and maintain operation of the MCRECS in the pressurization mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

ACTION 55 -

a. With one of the required chlorine detectors inoperable, restore the inoperable detector to OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of the MCRECS in the isolation mode of operation.

b. With no chlorine detectors OPERABLE, within 1 hour initiate and maintain operation of the MCRECSs in the isolation mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

NOTES

* When handling irradiated fuel in secondary containment.

a. A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition, provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

b. With a design providing only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.6.7-1 for that Trip Function shall be taken.

c. Actuates the MCRECS in the control room pressurization mode.

d. Actuates the MCRECS in the control room isolation mode.

e. Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater than 20% rated power, the normal full power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of hydrogen injection and prior to establishing reactor power levels below 20% rated power.

TABLE 3.3.6.7-2

MCRECS ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low Low (Level 1)	≥ -113 inches	≥ - 113 inches
2. Drywell Pressure - High	≤ 1.92 psig	≤ 1.92 psig
3. Main Steam Line Radiation - High	≤ 3 x full-power background*	≤ 3 x full-power background*
4. Main Steam Line Flow - High	≤ 138% rated flow	≤ 138% rated flow
5. Refueling Floor Area Radiation - High	≤ 20 mr/hour	≤ 20 mr/hour
6. Control Room Air Inlet Radiation - High	≤ 1 mr/hour	≤ 1 mr/hour
7. Control Room Air Inlet Chlorine Level - High	≤ 5 ppm chlorine	≤ 5 ppm chlorine

*Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater 20% rated power, the normal full power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of hydrogen injection and prior to establishing reactor power levels below 20% rated power.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 88 TO

FACILITY OPERATING LICENSE NPF-5

GEORGIA POWER COMPANY
OGLETHORPE POWER CORPORATION
MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA
CITY OF DALTON, GEORGIA

EDWIN I. HATCH NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-366

1.0 INTRODUCTION

By submittal dated March 20, 1987, (Reference 1) the Georgia Power Company proposed Technical Specification changes to permit a temporary increase in the Edwin I. Hatch Nuclear Plant Unit 2 main steam line high radiation scram and isolation setpoints to facilitate the testing of hydrogen addition water chemistry. The addition of hydrogen to BWR reactor coolant has been shown to reduce problems associated with intergranular stress corrosion cracking of the stainless steel piping. These proposed changes are necessary since, on the basis of prior experience at Hatch Unit 1, it is anticipated that main steam line radiation levels may increase by a factor of three to eight during the tests over the routinely experienced dose rates. In addition, the changes would correct a typographical error in Table 3.3.6.7-1 of the Technical Specifications.

Receipt of the March 20, 1987 amendment request was noticed in the Federal Register on July 15, 1987 (52 FR 26586). That notice described the changes requested by the licensee and discussed the basis for a proposed no significant hazards consideration determination.

Since that time, the NRC staff has had several telephone conversations with representatives of the licensee, primarily pertaining to the results of the hydrogen addition tests that previously had been approved for Hatch Unit 1. A November 23, 1987 submittal from the licensee (Reference 2) provides information regarding the higher in-plant radiation levels that have been measured during the Unit 1 hydrogen addition tests, confirms that the guidelines of the Electric Power Research Institute (EPRI) for installation of BWR hydrogen water chemistry have been followed for Unit 2, and corrects an erroneous reference to a Technical Specification table.

The November 23, 1987 submittal does not change the licensee's March 20, 1987 original request and does not in any way affect the NRC staff's proposed finding of no significant hazards consideration.

2.0 EVALUATION

2.1 High Radiation Scram and Isolation Setpoints

The Main Steam Line Radiation Monitors (MSLRMs) provide reactor scram and reactor vessel and primary containment isolation signals upon detection of high activity levels in the main steam lines. Additionally, these monitors serve to limit radioactivity releases in the event of fuel failures. The proposed Technical Specification changes (to Tables 2.2.1-1, 3.3.1-1, 3.3.2-1, 3.3.2-2, 3.3.6.7-1 and 3.3.6.7-2) would allow adjustments to the normal background radiation level and associated trip setpoints for the MSLRMs at reactor power levels greater than 20% rated power. The adjustments are needed to accommodate the expected increase in main steam activity levels as a result of hydrogen injection into the primary system. This is primarily due to increased nitrogen-16 (N-16) levels in the steam phase.

The licensee states that the only transient or postulated accident which takes credit for the main steam line high radiation scram and isolation signals is the control rod drop accident (CRDA). The staff notes that for a CRDA, the MSLRMs' primary function is to limit the transport of activity released from failed fuel to the turbine and condensers by initiating closure of the main steam isolation valves and thus isolating the reactor vessel. Main steam line high radiation will also produce a reactor scram signal (reactor scram in the event of a CRDA, however, would be initiated by signals from the Neutron Monitoring System) and will isolate the mechanical vacuum pump and the gland seal steam exhaust system to reduce leakage of fission products to the atmosphere from the turbine and condensers.

Generic analyses of the consequences of a CRDA have shown that fuel failures are not expected to result from a CRDA occurring at greater than 10% power. As power increases, the severity of the rod accident rapidly decreases due to the effects of increased void formation and increased Doppler reactivity feedback. Since the setpoint adjustments will be restricted to power levels above 20% of rated power, the staff concludes that the currently approved CRDA analysis for Hatch 2 remains appropriately bounding.

2.2 Radiation Protection/ALARA

The staff also has reviewed the proposed Technical Specification change to assure that the licensee has considered the radiological implications of the dose rate increases associated with N-16 equilibrium changes during hydrogen addition at BWRs. The review was also intended to determine that the licensee has adequately considered radiation protection/ALARA measures for the course of the test, in accordance with 10 CFR 20.1(c) and Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable."

An overall objective of the test is to determine general in-plant and site boundary dose rate increases due to hydrogen addition. The licensee has indicated that normal health physics/ALARA practices and procedures for Hatch will be continued throughout the test. Additionally, main steam system dose rates will be monitored by surveys on a routine basis. The licensee also indicated that specific locations will be identified where temporary shielding may be needed for long-term implementation of hydrogen injection.

A similar test was previously approved for Hatch Unit 1 (Amendment 125, issued May 2, 1986). The licensee has provided a radiological assessment of the Unit 1 test by letter dated November 23, 1987. Dose control and surveillance efforts planned for the Unit 2 test are similar to those previously approved for the Unit 1 hydrogen addition test. Tests of this type have been proposed and conducted at other operating BWRs following formal staff review and approval of similar Technical Specification changes. The test conditions, as identified by the vendor, as well as the measures proposed for radiation protection/ALARA at the Edwin I. Hatch Nuclear Plant Unit 2, are consistent with those utilized at the other BWRs during their successful hydrogen addition tests. None of these tests involved any significant, unanticipated, radiological exposures or releases.

2.3 Compressed Hydrogen Storage and Distribution System

In its letters dated March 20 and November 23, 1987, (References 1 and 2), the licensee provided information on the hydrogen and oxygen storage and distribution system to facilitate the Hydrogen Water Chemistry pre-implementation test. The licensee's hydrogen and oxygen storage and distribution system is designed to minimize the potential hazard to safety related systems and meets the applicable parts of the BWR Owners Group, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations," 1987 Revision.

The pre-implementation test will be conducted with the guidance of General Electric (GE), taking into consideration the lessons learned from approximately eight other hydrogen injection tests previously performed with GE assistance. Compressed hydrogen will be supplied and stored onsite in a gaseous tube trailer. The separation distance of the hydrogen tube trailer and safety related structures meets the BWR Owners Group (BWROG) Guidelines. With respect to the hydrogen distribution system, an excess flow valve is provided in the 1 inch flexible metallic hose connecting the hydrogen storage tanks with the injection system. The purpose of this valve is to limit the release rate of hydrogen in the event of a pipe break. In the hydrogen injection area inside the plant, hydrogen monitors are provided at the booster pump and the hydrogen injection control valves. These monitors are set to alarm and isolate the hydrogen injection system when hydrogen concentrations exceed 2%.

Since the licensee stores substantial amounts of chlorine onsite for the purpose of water treatment, the staff evaluated the potential synergistic effect associated with the storage of hydrogen. The combination of hydrogen gas and chlorine gas can explode in the presence of any form of energy, such as sunlight or heat (250°C). Therefore, it is prudent to maintain an adequate separation distance between the chlorine and hydrogen storage facilities. The hydrogen tube tank trucks will be parked a distance of 100 feet away from the chlorine tank car. The licensee has stated that the closest approach the hydrogen supply truck will make to the chlorine tank car is approximately 50 feet. The 50 foot separation distance is judged to be sufficient to prevent interaction in the unlikely event of a simultaneous chlorine and hydrogen release, since it meets the requirements of NFPA 50A-2984, Standards for Gaseous Hydrogen Systems at Consumer Sites.

On the basis of the above evaluation, we find that the proposed Technical Specification changes are in accordance with the BWROG (1987 Revision) "Guidelines for Permanent Hydrogen Water Chemistry Installations," are bounded by prior accident analysis, are consistent with Regulatory Guide 8.8, and meet General Design Criteria 3, and Branch Technical Position CMEB 9.5.1 of NUREG-0800. The changes are therefore acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 should be no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 20 CFR 51.22(b), no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of this 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 July 15, 1987 (52 FR 26586), and consulted with the state of Georgia. No public comments were received, and the state of Georgia did not have any comments.

We have 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 REFERENCES

1. Letter from J. P. O'Reilly, Georgia Power Company, to the NRC, dated March 20, 1987.

2. Letter from L. T. Gucwa, Georgia Power Company, to the NRC, dated November 23, 1987.

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Dated: January 13, 1988

DATED January 13, 1988

AMENDMENT NO. 88 TO FACILITY OPERATING LICENSE NPF-5, EDWIN I. HATCH, UNIT 2

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