May 21, 1986

Docket No. 50-321

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Mr. J. T. Beckham, Jr.
Vice President - Nuclear Generation
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Beckham:

The Commission has issued the enclosed Amendment No. 125 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 (TSs) in response to your application dated March 7, 1986.

The amendment revises the main steam line high radiation scram and isolation setpoints, on a one time short term basis, to facilitate test injections of hydrogen into the reactor coolant. The test is expected to be completed within about 20 days of its start.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next Biweekly notice.

Sincerely,

George W. Rivenbark, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

Enclosures:
1. Amendment No. 125 to DPR-57
2. Safety Evaluation

cc w/enclosures:
See next page

BWR:PD#2	BWR:PD#2	OELD	BWR:PD#2
SNorris	GRivenbark:nc	Goddard	DMuller
5/16/86	5/16/86	5/19/86	5/21/86

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Mr. J. T. Beckham, Jr.
Georgia Power Company

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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. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 125
License No. DPR-57

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

   A. The application for amendment by Georgia Power Company, et al., (the licensee) dated March 7, 1986 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;

   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:

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Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 125, 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 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 21, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 125

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 a vertical line indicating the area of change.

Remove	Insert
3.1-5	3.1-5
3.1-6	3.1-6
-	3.1-6a
3.2-2	3.2-2
3.2-4	3.2-4
3.2-19	3.2-19

Table 3.1-1 (Cont'd)

Scram Number (a)	Source of Scram Trip Signal		Operable Channels Required Per Trip System (b)	Scram Trip Setting	Source of Scram Signal is Required to be Operable Except as Indicated Below
8	APRM	Downscale	2	≥3/125 of full scale	The APRM downscale trip is active only when the Mode Switch is in RUN. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not tripped.
		15% Flux	2	≤15/125 of full scale Tech Spec 2.1.A.1.b.	The APRM 15% Scram is automatically bypassed when the Mode Switch is in the RUN position.
9	Main Steam Line Radiation		2	≤3 times normal background at rated thermal power (c).	Not required if all steam lines are isolated.
10	Main Steam Line Isolation Valve Closure		4	≤10% valve closure from full open Tech Spec 2.1.A.5.	Automatically bypassed when the Mode Switch is not in the RUN position. The design permits closure of any two lines without a scram being initiated.
11	Turbine Control Valve Fast Closure		2	Within 30 milliseconds of the start of control valve fast closure Tech Spec 2.1.A.4.	Automatically bypassed when turbine steam flow is below that corresponding to 30% of rated thermal power as measured by turbine first stage pressure.

Table 3.1-1 (Cont'd)

Scram Number (a)	Source of Scram Trip Signal	Operable Channels Required Per Trip System (b)	Scram Trip Setting	Source of Scram Signal is Required to be Operable Except as Indicated Below
12	Turbine Stop Valve Closure	4	$\leq$10% valve closure from full open Tech Spec 2.1.A.3.	Automatically bypassed when turbine steam flow is below that corresponding to 30% of rated thermal power as measured by turbine first stage pressure.

Notes for Table 3.1-1

a. The column entitled "Scram Number" is for convenience so that a one-to-one relationship can be established between items in Table 3.1-1 and items in Table 4.1-1.

b. There shall be two operable or tripped trip systems for each potential scram signal. If the number of operable channels cannot be met for one of the trip systems, that trip system shall be tripped. However, one trip signal channel of a trip system may be inoperable for up to two (2) hours during periods of required surveillance testing without tripping the associated trip system, provided that the other remaining channel(s) monitoring that parameter within that trip system is (are) operable.

c. 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.

Notes for Table 3.1-1 (cont)

For SCRAMS 1 thru 7 and 8 APRM 15% Flux, if the number of operable channels is not met for both trip systems; initiate insertion of all control rods capable of being moved by control rod drive pressure and complete their insertion within four (4) hours.

For SCRAM 8 (APRM High Trips, Inoperative, and Downscale), if the number of operable channels in not met for both trip systems; initiate insertion of all control rods capable of being moved by control rod drive pressure and complete their insertion within four hours or reduce power to the IRM range and go to the START & HOT STANDBY position of the Mode Switch within eight hours.

For SCRAMS 9 and 10, if the number of operable channels is not met for both trip systems; reduce turbine load and close main steam line isolation valves within eight hours or initiate insertion of all control rods capable of being moved by control rod drive pressure and complete their insertion within four hours.

For SCRAMS 11 and 12, if the number of operable channels is not met for both trip systems, reduce reactor power to 25% of rated thermal power or less within eight hours.

TABLE 3.2-1

INSTRUMENTATION WHICH INITIATES REACTOR VESSEL AND PRIMARY CONTAINMENT ISOLATION

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System (b)	Trip Setting	Action to be taken if number of channels is not met for both trip systems (c)	Remarks (d)
1	Reactor Vessel Water Level	Low (level 3) Narrow Range	2	≥8.5 inches	Initiate an orderly shutdown and achieve the Cold Shutdown Condition within 24 hours or isolate the shutdown cooling system.	Initiates Group 2 & 6 isolation.
		Low Low (Level 2)	2	≥-55 inches	Initiate an orderly shutdown and achieve the Cold Shutdown Condition within 24 hours.	Starts the SGTS, initiates Group 5 isolation, and initiates secondary containment isolation.
		Low Low Low (Level 1)	2	≥-121.5 inches	Initiate an orderly shutdown and achieve the Cold Shutdown Condition within 24 hours.	Initiates Group 1 isolation.
2	Reactor Pressure (Shutdown Cooling Mode)	High	1	≤135 psig	Isolate shutdown cooling.	Isolates the shutdown cooling suction valves of the RHR system.
3	Drywell Pressure	High	2	≤2 psig	Initiate an orderly shutdown and achieve the Cold Shutdown Condition within 24 hours.	Starts the standby gas treatment system, initiates Group 2 isolation and secondary containment isolation.
4	Main Steam Line Radiation	High	2	≤3 times normal full power background (e)	Initiate an orderly load reduction and close MSIVs within 8 hours.	Initiates Group 1 isolation.

Notes for Table 3.2-1

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between lines in Table 3.2-1 and items in Table 4.2-1.

b. Primary containment integrity shall be maintained at all times prior to withdrawing control rods for the purpose of going critical, when the reactor is critical, or when the reactor water temperature is above 212^{0}F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 MWt.

   When primary containment integrity is required, there shall be two operable or tripped trip systems for each function.

c. If the number of operable channels cannot be met for one of the trip systems, that trip system shall be tripped. However, one trip signal channel of a trip system may be inoperable for up to two (2) hours during periods of required surveillance testing without tripping the associated trip system, provided that the other remaining channel(s) monitoring that same parameter within that trip system is (are) operable.

d. The valves associated with each Group isolation are given in Table 3.7-1.

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 3.2-8 (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	≤3 times normal full power background (e)	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.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 125 TO

FACILITY OPERATING LICENSE NO. DPR-57

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

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-321

1.0 INTRODUCTION

By submittal dated March 7, 1986 the Georgia Power Company has proposed a Technical Specification change to permit a temporary increase in the Edwin I. Hatch Nuclear Plant Unit No. 1 main steam line high radiation scram and isolation setpoints to facilitate the testing of hydrogen addition water chemistry at their Hatch plant. This proposed change is necessary since, on the basis of prior experience, it is anticipated that main steam line radiation levels may increase during the test by a factor of five over the routinely experienced dose rates. In addition, in response to discussions with the staff, GPC, by letter dated April 22, 1986 provided details concerning the dose control measures and radiological surveillance efforts planned in support of the testing.

2.0 EVALUATION

2.1 HIGH RADIATION SCRAM AND ISOLATION SETPOINTS

The Main Steam Line Radiation Monitoring (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 3.1-1, 3.2-1 and 3.2-8) 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 1 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.

The staff also has discussed with licensee representatives the details of the dose control measures and surveillance efforts planned for the 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 1, 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.

On the basis of the adequacy of the licensee's radiation protection/ALARA program, utilization of special surveys to monitor dose rate increases on site and at the site boundary, the capability to monitor for fuel failures, as well as the success of similar efforts at other operating BWRs, the staff finds the licensee's request acceptable. Hence, the staff recommends that the Technical Specification change be approved as requested.

2.3 COMPRESSED HYDROGEN STORAGE AND DISTRIBUTION SYSTEM

The licensee's hydrogen addition system is designed to reduce the potential hazard to certain safety related systems from intergranular stress corrosion cracking. Central storage of gaseous hydrogen is located outdoors near the existing hydrogen storage facility for turbine generators. Potential release of gaseous hydrogen outdoors is not expected to be a significant hazard. Past reviews of compressed hydrogen storage at other sites indicate that the release and ignition of pressurized hydrogen does not produce significant overpressures. The factors that determine this finding are the limited quantities of hydrogen that are available in compressed hydrogen storage containers and, given a release, the relatively high dispersion rate due to hydrogen buoyancy. 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%. On the basis of the above considerations, we conclude that the licensee's hydrogen addition system meets Section C.5.d of BTP CMEB 9.5.1 (NUREG 0800) and is, 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

On the basis of the considerations discussed above, the staff has concluded 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.

Principal Contributor: M. Lamastra

Date: May 21, 1986