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Docket No. 50-366

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. 29 to Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Unit 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 23, 1982, and your application dated April 29, 1982, as supplemented May 3, 1982.

These changes to the TSs involve a surveillance requirement related to the safety-relief valve tail-pipe pressure switch setpoint, implementation of inerting of Unit No. 2, and replacement of the recirculation pump fire protection sprinkler system by the inerted drywell atmosphere.

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

Sincerely,

ORIGINAL SIGNED BY
JOHN F. STOLZ

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 29 to NPF-5
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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4/23/82, 4/29/82, 5/3/82.

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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. 29
License No. NPF-5

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

   A. The applications for amendment by Georgia Power Company, et al., (the licensee) dated April 23, 1982, and April 29, 1982, as supplemented May 3, 1982, 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 applications, 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. 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. 29 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 13, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 29

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. The overleaf pages are provided to maintain document completeness.

Remove	Insert
3/4 3-5	3/4 3-5
3/4 4-4	3/4 4-4
3/4 6-3	3/4 6-3
3/4 6-9	3/4 6-9
—	3/4 6-45
3/4 7-25	3/4 7-25
B3/4 6-6	B3/4 6-6

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; this function may be bypassed when necessary for containment inerting or de-inerting (purging).

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. Also trips reactor coolant system recirculation pump MG sets.

k. Also trips reactor coolant system recirculation pump motors.

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

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High*	NA
b. Inoperative	NA
2. Average Power Range Monitor:*	
a. Neutron Flux - Upscale, 15%	NA
b. Flow Referenced Simulated Thermal Power - Upscale	≤ 0.09**
c. Fixed Neutron Flux - Upscale, 118%	≤ 0.09
d. Inoperative	NA
d. Inoperative	NA
e. Downscale	NA
f. LPRM	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 0.55
4. Reactor Vessel Water Level - Low	≤ 1.05
5. Main Steam Line Isolation Valve -Closure	≤ 0.06
6. Main Steam Line Radiation - High	NA
7. Drywell Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
9. Turbine Stop Valve - Closure	≤ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≤ 0.08#
11. Reactor Mode Switch in Shutdown Position	NA
12. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from detector output or input of first electronic component in channel.

**Not including simulated thermal power time constant.

#Measured from start of turbine control valve closure.

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.3 An idle recirculation loop shall not be started unless the temperature differential between the reactor coolant within the dome and the bottom head drain is $\leq$ 145°F, and

a. The temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is $\leq$ 50°F when both loops have been idle, or

b. The temperature differential between the reactor coolant within the idle and operating recirculation loops is $\leq$ 50°F when only one loop has been idle, and the operating loop flow rate is $\leq$ 50% of rated loop flow.

APPLICABILITY: CONDITIONS 1, 2, 3 and 4.

ACTION:

With temperature differences and/or flow rate exceeding the above limits, suspend startup of any idle recirculation loop.

SURVEILLANCE REQUIREMENTS

4.4.1.3 The temperature differential and flow rate shall be determined to be within the limit within 30 minutes prior to startup of an idle recirculation loop.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of at least 9 of the following reactor coolant system safety/relief valves shall be OPERABLE with lift settings within ± 1% of the indicated pressures*.

4 Safety-relief valves @ 1090 psig.
4 Safety-relief valves @ 1100 psig**.
3 Safety-relief valves @ 1110 psig**.

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

a. With the safety valve function of one of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief valve tail-pipe pressure switches inoperable and the associated safety/relief valve(s) otherwise indicated to be open, place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.4.2 The tail-pipe pressure switch of each safety/relief valve shall be demonstrated OPERABLE by performance of:

a. CHANNEL FUNCTIONAL TEST:

1. At least once per 31 days, except that all portions of the channel inside the primary containment may be excluded from the CHANNEL FUNCTIONAL TEST, and

2. At each scheduled outage of greater than 72 hours during which entry is made into the primary containment, if not performed within the previous 31 days.

b. CHANNEL CALIBRATION and verifying the setpoint to be 85 ± 5 psig at least once per 18 months.

* The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperature and pressure.
** Up to two inoperable valves may be replaced with spare OPERABLE valves with lower setpoints of 1090 or 1100 psig, respectively, until the next refueling outage.

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. $\leq L_a$, 1.2 percent by weight of the containment air per 24 hours at P_a, 57.5 psig, or

2. $\leq L_t$, 0.849 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 28.8 psig.

b. A combined leakage rate of:

1. $\leq$ 0.60 L_a for all penetrations and valves, except for main steam isolation valves, subject to Type B and C tests when pressurized to P_a, and

2. $\leq$ 0.009 L_a for the following penetrations*:

(a) Main steam condensate drain, penetration 8;

(b) Primary feedwater, penetrations 9A and 9B;

(c) Reactor water cleanup, penetration 14;

(d) Equipment drain sump discharge, penetration 18;

(e) Floor drain sump discharge, penetration 19;

(f) Chemical drain sump discharge, penetration 55; and

(g) Torus drainage and purification, penetration 234A.

c. 11.5 scf per hour for any one main steam isolation valve when tested at 28.8 psig.**

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

* Potential bypass leakage paths.
**Exemption to Appendix J of 10 CFR 50.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

With:

a. the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or,

b. the measured combined leakage rate for all penetrations and valves, except main steam isolation valves, subject to Type B and C tests exceeding 0.60 L_a or with the measured combined leakage rate for all specified potential bypass leakage path penetrations exceeding 0.009 L_a, or

c. the main steam isolation valve measured leak rate exceeding 11.5 scf per hour for any one MSIV,

Restore:

a. the overall integrated leakage rate(s) to < 0.75 L_a or < 0.75 L_t as applicable, and

b. the combined leakage rate for all penetrations and valves, except main steam isolation valves, subject to Type B and C tests to $\leq$ 0.60 L_a and the combined leakage rate for the specified potential bypass leakage path penetrations to $\leq$ 0.009 L_a, and

c. the leakage rate to $\leq$ 11.5 scf per hour for any one main steam isolation valve,

Prior to increasing the reactor coolant temperature above 212°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 - (1972):

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 $\pm$ 10 month intervals during shutdown at either P_a, 57.5 psig or at P_t, 28.8 psig during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

PRIMARY CONTAINMENT INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Primary containment internal pressure shall not exceed 0.75 psig.*

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

With the primary containment internal pressure in excess of the specified limit, restore the internal pressure to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The primary containment internal pressure shall be determined to be less than or equal to 0.75 psig at least once per 12 hours.

*Except when performing the test required by Specification 4.6.4.1.b or the Special Startup Test authorized by Amendment No. 2, or when either inerting or de-inerting (purging) primary containment as required by 3.6.6.4.

CONTAINMENT SYSTEMS

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Drywell average air temperature shall not exceed 135°F*.

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

With the drywell average air temperature > 135°F*, reduce the average air temperature to within the limit with 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The drywell average air temperature shall be the volumetric average of the temperatures at the following drywell elevations and shall be determined to be within the limit at least once per 24 hours:

	Elevation	Azimuth
a.	128-130 feet	115° or 255° or 315° or 320°
b.	162 feet	90° or 270°
c.	175 feet	90° or 270°
d.	187 feet	90° or 270°

* From July 26, 1979 until the end of the first refueling outage, a limit of 145°F applies.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.6.6.4 The primary containment atmosphere oxygen concentration shall be less than 4% by volume.

APPLICABILITY: OPERATIONAL CONDITION 1, during the time period:

a. Within 72 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER, following startup, to

b. Within 72 hours prior to reducing THERMAL POWER to less than 15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.

ACTION:

With the oxygen concentration in the primary containment exceeding the limit, be in at least STARTUP within 8 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.4 The oxygen concentration in the primary containment shall be verified to be within the limit within 72 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER and at least once per 7 days thereafter.

SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.6.2 The following sprinkler systems shall be OPERABLE:

Turbine lube oil storage
Turbine lube oil reservoir
RCIC room
HPCI room
West cableway
East cableway
Recirculation pump motor generator room
Cable spreading room
RPS vertical cableway
Control Building Corridor (El. 130 ')
Reactor Building HVAC room

APPLICABILITY : Whenever equipment in the sprinkler protected areas is required to be OPERABLE.

ACTION:

a. With one or more of the above required sprinkler systems inoperable, establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s) within 1 hour, provided radiation levels permit personnel access; restore the system to OPERABLE status eithin 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SPRINKLER SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7.6.2 Each of the above required sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 12 months by performing a system functional test which includes simulated automatic actuation of the system and verifying that the automatic sprinkler valves in the flow path actuate to their correct positions. Deluge system valves will be checked for auto actuation but will not be tripped through a complete cycle.

b. At least once per 18 months:

1. By inspection of the spray headers to verify their integrity, and

2. By a visual inspection of each nozzle's spray area to verify spray pattern is not obstructed.

c. At least once per 3 years by performing an air flow test through each open head sprinkler header and verifying each open head sprinkler nozzle is unobstructed.

BASES

3/4.6.4 VACUUM RELIEF

Vacuum relief breakers are provided to equalize the pressure between the suppression chamber and drywell and between the reactor building and suppression chamber. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There are an adequate number of valves to provide some redundance so that operation may continue with no more than three vacuum breakers inoperable in the closed position.

Each set of vacuum breakers between the reactor building and the suppression chamber provides 100% relief, so operation may continue with one valve out-of-service for 7 days.

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The reactor building provides secondary containment during normal operation when the drywell is sealed and in service. When the reactor is shutdown or during refueling the drywell may be open and the reactor building then becomes the primary containment.

Establishing and maintaining a vacuum in the building with the standby gas treatment system once per 18 months, along with the surveillance of the doors, hatches and dampers, is adequate to ensure that there are no violations of the integrity of the secondary containment. Only one closed damper in each penetration line is required to maintain the integrity of the secondary containment.

3/4.6.6 CONTAINMENT ATMOSPHERE CONTROL

The OPERABILITY of the containment iodine filter trains ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses.

CONTAINMENT ATMOSPHERE CONTROL (Continued)

The OPERABILITY of the systems required for the detection and control of hydrogen gas ensures that these systems will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner is capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. The hydrogen mixing system is provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

The requirement for the primary containment atmosphere oxygen concentration to be less than 4% by volume is being added for fire protection considerations. This is being done in lieu of the installation of sprinkler for the recirculation pumps inside the drywell.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 29 TO FACILITY OPERATING LICENSE NO. NPF-5

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

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

DOCKET NO. 50-366

Introduction

By letters dated April 23 and 29 and May 3, 1982, Georgia Power Company (GPC or the licensee) requested changes to the Technical Specifications (TSs) appended to Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Unit No. 2. The changes involve TS 4.4.2.b (April 23 letter) regarding the surveillance requirements for the safety-relief valve (SRV) tail-pipe pressure switch setpoint; TS Table 3.3.1-1, and TSs 3.6.1.6, 3.6.6.4 and 4.6.6.4 (April 29 letter) regarding implementation of inerting of Unit No. 2; and TSs 3.6.1.2 and 3.7.6.2 (May 3 letter) which replace the recirculation pump fire protection sprinkler system with an inerted drywell atmosphere.

Evaluation

1. April 23, 1982 Letter

The licensee has determined that the present setpoint on the SRV tail-pipe pressure switches of 20 psig is not high enough to provide correct information on the "open" or "closed" status of the SRV to a reactor operator during and following a loss-of-coolant accident (LOCA). In order to provide correct status to the operators, the licensee proposes to revise the Hatch Unit 2 setpoint to 85 psig. The 85 psig setpoint properly accounts for the drywell LOCA pressure of 65 psig plus a 10 psig static head pressure in the tail-pipe and a 10 psig margin above these two pressure components to provide assurance against a spurious "SRV open" indication. Based on the above summation of pressure components, plus margin, we conclude that the 85 psig setpoint change provides more reliable information to the operators and is thus acceptable.

2. April 29, 1982 Letter

GPC committed to inert the Unit 2 drywell as part of the Fire Protection Program in order to meet 10 CFR 50.48, Appendix R. However, inerting is also required as part of 10 CFR 50.44, Interim Requirements Related to Hydrogen Control. Hatch Unit 1 has always operated with an inerted drywell. The TSs proposed for Unit 2 provide for an extension of the current 24-hour limit on exceeding 0.75 psig primary containment internal pressure, to 72 hours. In addition, there is a proposal to extend the period of inoperability of the High Drywell Pressure trip function from the periods when containment operability is not required to those additional periods when inerting and

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deinerting is in progress. There is a very low likelihood of any off-normal operating conditions during these periods; this however, is compensated for by a reduction in potential unnecessary scrams and resultant thermal cycles on the primary system. We conclude for the above reasons, that the TS changes needed to implement the Hatch 2 inerting program are acceptable.

3. May 3, 1982 Letter

The licensee proposes to replace the recirculation pump sprinkler system with the inerted containment as an equivalent fire protection system for these pumps. As the containment atmosphere will be inerted to 4% or less oxygen, it would not be able to support combustion. GPC will seal the existing sprinkler penetration and proposes to delete the penetration leak test from the TSs as it is no longer needed. We conclude that the inerted Unit 2 containment serves as a replacement for the recirculation pump sprinkler system.

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: May 13, 1982

Morton B. Fairtile of the NRC staff prepared this Safety Evaluation.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-366

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. 29 to Facility Operating License No. NPF-5, issued to Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, which revised Technical Specifications (TSs) for operation of the Edwin I. Hatch Nuclear Plant, Unit No. 2 (the facility) located in Appling County, Georgia. The amendment is effective as of the date of issuance.

These changes to the TSs involve a surveillance requirement related to the safety-relief valve tail-pipe pressure switch setpoint, implementation of inerting of Unit No. 2, and replacement of the recirculation pump fire protection sprinkler system by the inerted drywell atmosphere.

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.

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The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR Section 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 April 23, 1982, and April 29, 1982, as supplemented May 3, 1982, (2) Amendment No. 29 to License No. NPF-5, 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, 301 City Hall Drive, 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 Licensing.

Dated at Bethesda, Maryland, this 13th day of May 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing