UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 137
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 the Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees), dated April 14, 1995, as supplemented by letters dated June 22 and July 18, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 hereby amended by page 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:

    Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 137 , 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 from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: August 23, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 137

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 areas of change.

Remove Pages	Insert Pages
3.3-6	3.3-6
3.3-41	3.3-41
3.3-42	3.3-42
3.3-43	3.3-43
3.3-44	3.3-44
3.3-55	3.3-55
3.5-6	3.5-6
-	3.5-6a
-	3.5-6b
B 3.3-31	B 3.3-31
--	B 3.3-31a
--	B 3.3-31b
B 3.3-32	B 3.3-32
B 3.3-105	B 3.3-105
B 3.3-133	B 3.3-133
B 3.3-134	B 3.3-134
B 3.3-174	B 3.3-174
B 3.5-16	B 3.5-16
--	B 3.5-16a
--	B 3.5-16b

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.16	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Functions 3 and 4, channel sensors are excluded. 3. For Function 5, "*n*" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. -- Verify the RPS RESPONSE TIME is within limits.	18 months on a STAGGERED TEST BASIS

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------

1. Refer to Table 3.3.5.1-1 to determine which SRs apply for each ECCS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed as follows: (a) for up to 6 hours for Functions 3.c and 3.f; and (b) for up to 6 hours for Functions other than 3.c and 3.f provided the associated Function or the redundant Function maintains initiation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.5.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.5.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.5.1.3	Perform CHANNEL CALIBRATION.	92 days
SR 3.3.5.1.4	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.5.1.5	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months

Table 3.3.5.1-1 (page 1 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3, 4(a), 5(a)	4(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -113 inches
b. Drywell Pressure - High	1,2,3	4(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 1.92 psig
c. Reactor Steam Dome Pressure - Low (Injection Permissive)	1,2,3	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 390 psig and ≤ 476 psig
	4(a), 5(a)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 390 psig and ≤ 476 psig
d. Core Spray Pump Discharge Flow - Low (Bypass)	1,2,3, 4(a), 5(a)	1 per subsystem	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 570 gpm and ≤ 745 gpm
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3, 4(a), 5(a)	4(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -113 inches

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Also required to initiate the associated diesel generator (DG) and isolate the associated plant service water (PSW) turbine building (T/B) isolation valves.

Table 3.3.5.1-1 (page 2 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
b. Drywell Pressure - High	1,2,3	4[b]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 1.92 psig
c. Reactor Steam Dome Pressure - Low (Injection Permissive)	1,2,3	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 390 psig and ≤ 476 psig
	4[a], 5[a]	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 390 psig and ≤ 476 psig
d. Reactor Steam Dome Pressure - Low (Recirculation Discharge Valve Permissive)	1[c],2[c], 3[c]	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 335 psig
e. Reactor Vessel Shroud Level - Level 0	1,2,3	2	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -202 inches
f. Low Pressure Coolant Injection Pump Start - Time Delay Relay	1,2,3, 4[a], 5[a]	1 per pump	C	SR 3.3.5.1.4 SR 3.3.5.1.5	
Pumps A,B,D					≥ 9 seconds and ≤ 11 seconds
Pump C					≤ 1 second

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Also required to initiate the associated DG and isolate the associated PSW T/B isolation valves.

(c) With associated recirculation pump discharge valve open.

Table 3.3.5.1-1 (page 3 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
g. Low Pressure Coolant Injection Pump Discharge Flow - Low (Bypass)	1,2,3, $4^{(a)}$, $5^{(a)}$	1 per subsystem	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 1675 gpm and ≤ 2215 gpm
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level - Low Low, Level 2	1, $2^{(d)}$, $3^{(d)}$	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -47 inches
b. Drywell Pressure - High	1, $2^{(d)}$, $3^{(d)}$	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 1.92 psig
c. Reactor Vessel Water Level - High, Level 8	1, $2^{(d)}$, $3^{(d)}$	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 56.5 inches
d. Condensate Storage Tank Level - Low	1, $2^{(d)}$, $3^{(d)}$	2	D	SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 2.61 ft
e. Suppression Pool Water Level - High	1, $2^{(d)}$, $3^{(d)}$	2	D	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 154 inches

(continued)

(a) When the associated subsystem(s) are required to be OPERABLE.

(d) With reactor steam dome pressure > 150 psig.

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------
1. Refer to Table 3.3.6.1-1 to determine which SRs apply for each Primary Containment Isolation Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains isolation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.6.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.6.1.3	Perform CHANNEL CALIBRATION.	92 days
SR 3.3.6.1.4	Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.6.1.5	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.6.1.6	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months
SR 3.3.6.1.7	------------------NOTE------------------ Channel sensors are excluded. -- Verify the ISOLATION SYSTEM RESPONSE TIME is within limits.	18 months on a STAGGERED TEST BASIS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.9	--------------------NOTE-------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 165 psig, the HPCI pump can develop a flow rate ≥ 4250 gpm against a system head corresponding to reactor pressure.	18 months
SR 3.5.1.10	--------------------NOTE-------------------- Vessel injection/spray may be excluded. --- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	18 months
SR 3.5.1.11	--------------------NOTE-------------------- Valve actuation may be excluded. --- Verify the ADS actuates on an actual or simulated automatic initiation signal.	18 months
SR 3.5.1.12	--------------------NOTE-------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify each ADS valve opens when manually actuated.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.13	-------------------NOTE------------------- ECCS injection/spray initiation instrumentation response time may be assumed from established limits. -- Verify each ECCS injection/spray subsystem ECCS RESPONSE TIME is within limits.	18 months

This page intentionally left blank.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.14

The Average Power Range Monitor Flow Biased Simulated Thermal Power — High Function uses an electronic filter circuit to generate a signal proportional to the core THERMAL POWER from the APRM neutron flux signal. This filter circuit is representative of the fuel heat transfer dynamics that produce the relationship between the neutron flux and the core THERMAL POWER. The time constant is specified in the COLR and must be verified to ensure that the channel is accurately reflecting the desired parameter.

The Frequency of 18 months is based on engineering judgment considering the reliability of the components.

SR 3.3.1.1.15

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods (LCO 3.1.3), and SDV vent and drain valves (LCO 3.1.8), overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.1.1.16

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. This test may be performed in one measurement or in overlapping segments, with verification that all components are tested. The RPS RESPONSE TIME acceptance criteria are included in Reference 10.

Note 1 allows neutron detectors to be excluded from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.16 (continued)

Note 2 allows channel sensors for Reactor Vessel Steam Dome Pressure - High and Reactor Vessel Water Level - Low, Level 3 (Functions 3 and 4) to be excluded from RPS RESPONSE TIME testing. This allowance is supported by Reference 12 which concludes that any significant degradation of the channel sensor response time can be detected during the performance of other Technical Specifications SRs.

(continued)

This page intentionally left blank.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.16 (continued)

RPS RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. Note 3 requires STAGGERED TEST BASIS Frequency to be determined based on four channels per trip system, in lieu of the eight channels specified in Table 3.3.1.1-1 for the Main Steam Line Isolation Valve—Closure Function. This Frequency is based on the logic interrelationships of the various channels required to produce an RPS scram signal. This Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

REFERENCES

1. FSAR, Section 7.2.
2. FSAR, Chapter 15.
3. FSAR, Section 6.3.3.
4. FSAR, Supplement 5A.
5. FSAR, Section 15.1.12.
6. NEDO-23842, "Continuous Control Rod Withdrawal in the Startup Range," April 18, 1978.
7. FSAR, Section 15.1.38.
8. P. Check (NRC) letter to G. Lainas (NRC), "BWR Scram Discharge System Safety Evaluation," December 1, 1980.
9. NEDO-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.
10. Technical Requirements Manual.
11. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
12. NEDO-32291, "System Analyses for Elimination of Selected Response Time Testing Requirements," January 1994.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

ECCS instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 6). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the ECCS instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.5.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint). Table 3.3.5.1-1, footnote (b), is added to show that certain ECCS instrumentation Functions are also required to be OPERABLE to perform DG initiation and actuation of the PSW T/B isolation.

Allowable Values are specified for each ECCS Function specified in the table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis, where applicable. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined, accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.1.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.5.1, LCO 3.5.2, LCO 3.7.2, LCO 3.8.1, and LCO 3.8.2 overlaps this Surveillance to complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 5.2.
2. FSAR, Section 6.3.
3. FSAR, Chapter 15.
4. NEDC-31376-P, "Edwin I. Hatch Nuclear Power Plant, SAFER/GESTR-LOCA, Loss-of-Coolant Accident Analysis," December 1986.
5. NEDC-30936-P-A, "BWR Owners' Group Technical Specification Improvement Analyses for ECCS Actuation Instrumentation, Part 2," December 1988.
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1.7 (continued)

ISOLATION SYSTEM RESPONSE TIME acceptance criteria are included in Reference 6. This test may be performed in one measurement, or in overlapping segments, with verification that all components are tested.

A Note to the Surveillance states that the channel sensors are excluded from ISOLATION SYSTEM RESPONSE TIME testing. The exclusion of the channel sensors is supported by Reference 8 which indicates that the sensors' response times are a small fraction of the total response time. Even if the sensors experienced response time degradation, they would be expected to respond in the microsecond to millisecond range until complete failure.

ISOLATION SYSTEM RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. This Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience that shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

REFERENCES

1. FSAR, Section 6.3.

2. FSAR, Chapter 15.

3. FSAR, Section 4.2.3.4.2.

4. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

5. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

6. Technical Requirements Manual.

7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

8. NEDO-32291, "System Analyses for Elimination of Selected Response Time Testing Requirements," January 1994.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.12 (continued)

SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.5.1.13

This SR ensures that the ECCS RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in Reference 14. A Note to the Surveillance states that the instrumentation portion of the response time may be assumed from established limits. The exclusion of the instrumentation from the response time surveillance is supported by Reference 15, which concludes that instrumentation will continue to respond in the microsecond to millisecond range prior to complete failure.

The 18 month Frequency is based on the need to perform the Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 6.3.2.2.3.
2. FSAR, Section 6.3.2.2.4.
3. FSAR, Section 6.3.2.2.1.
4. FSAR, Section 6.3.2.2.2.
5. FSAR, Section 15.1.39.
6. FSAR, Section 15.1.40.
7. FSAR, Section 15.1.33.

(continued)

BASES

REFERENCES (continued)

8. 10 CFR 50, Appendix K.

9. FSAR, Section 6.3.3.

10. NEDC-31376P, "E.I. Hatch Nuclear Plant Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Analysis," December 1986.

11. 10 CFR 50.46.

12. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

13. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

14. Technical Requirements Manual.

15. NEDO-32291, "System Analyses for Elimination of Selected Response Time Testing Requirements," January 1994.

This page intentionally left blank.