July 21, 1995

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Mr. J. T. Beckham, Jr.
Vice President - Plant Hatch
Georgia Power Company
P. O. Box 1295
Birmingham, AL 35201

SUBJECT: ISSUANCE OF AMENDMENTS - EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M90617 AND M90618)

Dear Mr. Beckham:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 196 to Facility Operating License DPR-57 and Amendment No. 136 to Facility Operating License NPF-5 for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated October 13, 1994, as supplemented by letters dated January 13 and May 4, 1995.

The amendments revise the TS to lower the anticipated transient without scram-recirculation pump trip (ATWS-RPT) setpoint by approximately 2 feet 2 inches to minimize the potential for RPTs following reactor scram, and allow restarting the recirculation pump following an RPT when the temperature differential between the coolant at the reactor bottom head and the reactor steam dome cannot be obtained, provided certain conditions are met.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly <u>Federal Register</u> notice.

Sincerely,

/s/

Kahtan N. Jabbour, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

Enclosures:
1. Amendment No. 196 to DPR-57
2. Amendment No. 136 to NPF-5
3. Safety Evaluation

cc w/encl:
See next page

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

July 21, 1995

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Vice President - Plant Hatch
Georgia Power Company
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Kahtan N. Jabbour

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cc w/encl: See next page

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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. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 196
License No. DPR-57

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 1 (the facility) Facility Operating License No. DPR-57 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 October 13, 1994, as supplemented by letters dated January 13 and May 4, 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. DPR-57 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 196 , 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, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: July 21, 1995

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. 136
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 October 13, 1994, as supplemented by letters dated January 13 and May 4, 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. 136 , 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: July 21, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 196

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

AND

TO LICENSE AMENDMENT NO. 136

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
Unit 1	3.3-30	3.3-30
	3.3-32	3.3-32
	B 3.3-89	B 3.3-89
	B 3.3-91	B 3.3-91
	B 3.3-92	B 3.3-92
	B 3.4-50	B 3.4-50
	---	B 3.4-50a
	---	B 3.4-50b
	B 3.4-52	B 3.4-52
Unit 2	3.3-31	3.3-31
	3.3-33	3.3-33
	B 3.3-89	B 3.3-89
	B 3.3-91	B 3.3-91
	B 3.3-92	B 3.3-92
	B 3.4-50	B 3.4-50
	---	B 3.4-50a
	---	B 3.4-50b
	B 3.4-52	B 3.4-52

3.3 INSTRUMENTATION

3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

LCO 3.3.4.2 Two channels per trip system for each ATWS-RPT instrumentation Function listed below shall be OPERABLE:

a. Reactor Vessel Water Level — ATWS-RPT Level; and

b. Reactor Steam Dome Pressure — High.

APPLICABILITY: MODE 1.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Restore channel to OPERABLE status. <u>OR</u> A.2 --------NOTE--------- Not applicable if inoperable channel is the result of an inoperable breaker. --------------------- Place channel in trip.	14 days 14 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.4.2.3	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Reactor Vessel Water Level — ATWS-RPT Level: ≥ -73 inches; and b. Reactor Steam Dome Pressure — High: ≤ 1095 psig.	18 months
SR 3.3.4.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation.	18 months

B 3.3 INSTRUMENTATION

B 3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

BASES

BACKGROUND

The ATWS-RPT System initiates an RPT, adding negative reactivity, following events in which a scram does not (but should) occur, to lessen the effects of an ATWS event. Tripping the recirculation pumps adds negative reactivity from the increase in steam voiding in the core area as core flow decreases. When Reactor Vessel Water Level — ATWS-RPT Level or Reactor Steam Dome Pressure — High setpoint is reached, the recirculation pump drive motor breakers trip.

The ATWS-RPT System (Ref. 1) includes sensors, relays, bypass capability, circuit breakers, and switches that are necessary to cause initiation of an RPT. The channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an ATWS-RPT signal to the trip logic.

The ATWS-RPT consists of two independent trip systems, with two channels of Reactor Steam Dome Pressure — High and two channels of Reactor Vessel Water Level — ATWS-RPT Level in each trip system. Each ATWS-RPT trip system is a two-out-of-two logic for each Function. Thus, either two Reactor Water Level — ATWS-RPT Level or two Reactor Pressure — High signals are needed to trip a trip system. The outputs of the channels in a trip system are combined in a logic so that either trip system will trip both recirculation pumps (by tripping the respective drive motor breakers).

There is one drive motor breaker provided for each of the two recirculation pumps for a total of two breakers. The output of each trip system is provided to both recirculation pump breakers.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The individual Functions are required to be OPERABLE in MODE 1 to protect against common mode failures of the Reactor Protection System by providing a diverse trip to mitigate the consequences of a postulated ATWS event. The Reactor Steam Dome Pressure — High and Reactor Vessel Water Level — ATWS-RPT Level Functions are required to be OPERABLE in MODE 1, since the reactor is producing significant power and the recirculation system could be at high flow. During this MODE, the potential exists for pressure increases or low water level, assuming an ATWS event. In MODE 2, the reactor is at low power and the recirculation system is at low flow; thus, the potential is low for a pressure increase or low water level, assuming an ATWS event. Therefore, the ATWS-RPT is not necessary. In MODES 3 and 4, the reactor is shut down with all control rods inserted; thus, an ATWS event is not significant and the possibility of a significant pressure increase or low water level is negligible. In MODE 5, the one rod out interlock ensures that the reactor remains subcritical; thus, an ATWS event is not significant. In addition, the reactor pressure vessel (RPV) head is not fully tensioned and no pressure transient threat to the reactor coolant pressure boundary (RCPB) exists.

The specific Applicable Safety Analyses and LCO discussions are listed below on a Function by Function basis.

a. <u>Reactor Vessel Water Level — ATWS-RPT Level</u>

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the ATWS-RPT System is initiated at a low level to aid in maintaining level above the top of the active fuel. The reduction of core flow reduces the neutron flux and THERMAL POWER and, therefore, the rate of coolant boiloff.

Reactor vessel water level signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Reactor Vessel Water Level — ATWS-RPT Level (continued)

Four channels of Reactor Vessel Water Level — ATWS-RPT Level, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. The Reactor Vessel Water Level — ATWS-RPT Level Allowable Value is chosen so that the system will not be initiated after a Level 3 scram with feedwater still available, and for convenience with the reactor core isolation cooling initiation.

b. Reactor Steam Dome Pressure — High

Excessively high RPV pressure may rupture the RCPB. An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This increases neutron flux and THERMAL POWER, which could potentially result in fuel failure and overpressurization. The Reactor Steam Dome Pressure — High Function initiates an RPT for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power generation. For the overpressurization event, the RPT aids in the termination of the ATWS event and, along with the safety/relief valves, limits the peak RPV pressure to less than the ASME Section III Code limits.

The Reactor Steam Dome Pressure — High signals are initiated from four pressure transmitters that monitor reactor steam dome pressure. Four channels of Reactor Steam Dome Pressure — High, with two channels in each trip system, are available and are required to be OPERABLE to ensure that no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. The Reactor Steam Dome Pressure — High Allowable Value is chosen to provide an adequate margin to the ASME Section III Code limits.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1 (continued)

cooldown operations and RCS inservice leakage and hydrostatic testing.

SR 3.4.9.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

SR 3.4.9.3 and SR 3.4.9.4

Differential temperatures within the applicable limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits ensures that the assumptions of the analysis for the startup of an idle recirculation loop (Ref. 7) are satisfied.

Performing the Surveillance within 15 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start.

If the 145°F temperature differential specified in SR 3.4.9.3 cannot be determined by direct indication, an alternate method may be used as described below:

The bottom head coolant temperature and the RPV coolant can be assumed to be ≤ 145°F if the following can be confirmed:

a. One or more loop drive flows were > 40 percent of rated flow prior to the RPT,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.3 and SR 3.4.9.4 (continued)

b. High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems have not injected since the RPT,

c. Feedwater temperature has remained > 300°F since the RPT, and

d. The time between the RPT and restart is < 30 minutes.

General Electric test data from BWR plants shows that stratification up to the 145°F differential does not occur any sooner than 1 hour following the RPT (Refs. 10 and 11). Adding HPCI and RCIC injection, and feedwater temperature constraints provides assurance that the temperature differential will not be exceeded within 30 minutes of the RPT.

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.9.4 is to compare the temperatures of the operating recirculation loop and the idle loop.

SR 3.4.9.3 and SR 3.4.9.4 have been modified by a Note that requires the Surveillance to be performed only in MODES 1, 2, 3, and 4. In MODE 5, the overall stress on limiting components is lower. Therefore, ΔT limits are not required.

(continued)

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BASES

REFERENCES (continued)

3. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," July 1982.

4. 10 CFR 50, Appendix H.

5. Regulatory Guide 1.99, Revision 2, May 1988.

6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

7. FSAR, Section 14.3.6.2.

8. George W. Rivenbark (NRC) letter to J. T. Beckham, Jr. (GPC), Amendment 126 to the Operating License, dated June 20, 1986.

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

10. GE-NE-668-13-0393, "Recirculation Pump Restart Without Vessel Temperature Indication for E.I. Hatch Nuclear Plant," April 9, 1993.

11. DRF A00-05834/6, "Safety & 10 CFR 50.92 Significant Hazards Consideration Assessment for RPV Stratification Prevention Improvements at Edwin I. Hatch Nuclear Plant Units 1 and 2," April 1994.

3.3 INSTRUMENTATION

3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

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a. Reactor Vessel Water Level — ATWS-RPT Level; and

b. Reactor Steam Dome Pressure — High.

APPLICABILITY: MODE 1.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Restore channel to OPERABLE status. <u>OR</u> A.2 --------NOTE--------- Not applicable if inoperable channel is the result of an inoperable breaker. --------------------- Place channel in trip.	14 days 14 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.4.2.3	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Reactor Vessel Water Level — ATWS-RPT Level: $\geq$ -73 inches; and b. Reactor Steam Dome Pressure — High: $\leq$ 1095 psig.	18 months
SR 3.3.4.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation.	18 months

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There is one drive motor breaker provided for each of the two recirculation pumps for a total of two breakers. The output of each trip system is provided to both recirculation pump breakers.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The individual Functions are required to be OPERABLE in MODE 1 to protect against common mode failures of the Reactor Protection System by providing a diverse trip to mitigate the consequences of a postulated ATWS event. The Reactor Steam Dome Pressure — High and Reactor Vessel Water Level — ATWS-RPT Level Functions are required to be OPERABLE in MODE 1, since the reactor is producing significant power and the recirculation system could be at high flow. During this MODE, the potential exists for pressure increases or low water level, assuming an ATWS event. In MODE 2, the reactor is at low power and the recirculation system is at low flow; thus, the potential is low for a pressure increase or low water level, assuming an ATWS event. Therefore, the ATWS-RPT is not necessary. In MODES 3 and 4, the reactor is shut down with all control rods inserted; thus, an ATWS event is not significant and the possibility of a significant pressure increase or low water level is negligible. In MODE 5, the one rod out interlock ensures that the reactor remains subcritical; thus, an ATWS event is not significant. In addition, the reactor pressure vessel (RPV) head is not fully tensioned and no pressure transient threat to the reactor coolant pressure boundary (RCPB) exists.

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Reactor vessel water level signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Reactor Vessel Water Level — ATWS-RPT Level (continued)

Four channels of Reactor Vessel Water Level — ATWS-RPT Level, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. The Reactor Vessel Water Level — ATWS-RPT Level Allowable Value is chosen so that the system will not be initiated after a Level 3 scram with feedwater still available, and for convenience with the reactor core isolation cooling initiation.

b. Reactor Steam Dome Pressure — High

Excessively high RPV pressure may rupture the RCPB. An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This increases neutron flux and THERMAL POWER, which could potentially result in fuel failure and overpressurization. The Reactor Steam Dome Pressure — High Function initiates an RPT for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power generation. For the overpressurization event, the RPT aids in the termination of the ATWS event and, along with the safety/relief valves, limits the peak RPV pressure to less than the ASME Section III Code limits.

The Reactor Steam Dome Pressure — High signals are initiated from four pressure transmitters that monitor reactor steam dome pressure. Four channels of Reactor Steam Dome Pressure — High, with two channels in each trip system, are available and are required to be OPERABLE to ensure that no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. The Reactor Steam Dome Pressure — High Allowable Value is chosen to provide an adequate margin to the ASME Section III Code limits.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1 (continued)

cooldown operations and RCS inservice leakage and hydrostatic testing.

SR 3.4.9.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

SR 3.4.9.3 and SR 3.4.9.4

Differential temperatures within the applicable limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits ensures that the assumptions of the analysis for the startup of an idle recirculation loop (Ref. 7) are satisfied.

Performing the Surveillance within 15 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start.

If the 145°F temperature differential specified in SR 3.4.9.3 cannot be determined by direct indication, an alternate method may be used as described below:

The bottom head coolant temperature and the RPV coolant can be assumed to be ≤ 145°F if the following can be confirmed:

a. One or more loop drive flows were > 40 percent of rated flow prior to the RPT,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.3 and SR 3.4.9.4 (continued)

b. High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems have not injected since the RPT,

c. Feedwater temperature has remained > 300°F since the RPT, and

d. The time between the RPT and restart is < 30 minutes.

General Electric test data from BWR plants shows that stratification up to the 145°F differential does not occur any sooner than 1 hour following the RPT (Refs. 10 and 11). Adding HPCI and RCIC injection, and feedwater temperature constraints provides assurance that the temperature differential will not be exceeded within 30 minutes of the RPT.

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.9.4 is to compare the temperatures of the operating recirculation loop and the idle loop.

SR 3.4.9.3 and SR 3.4.9.4 have been modified by a Note that requires the Surveillance to be performed only in MODES 1, 2, 3, and 4. In MODE 5, the overall stress on limiting components is lower. Therefore, ΔT limits are not required.

(continued)

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BASES

REFERENCES (continued)

3. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," July 1982.

4. 10 CFR 50, Appendix H.

5. Regulatory Guide 1.99, Revision 2, May 1988.

6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

7. FSAR, Section 15.1.26.

8. Kahtan N. Jabbour (NRC) letter to W. G. Hairston, III (GPC), Amendment 118 to the Operating License, dated January 10, 1992.

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

10. GE-NE-668-13-0393, "Recirculation Pump Restart Without Vessel Temperature Indication for E.I. Hatch Nuclear Plant," April 9, 1993.

11. DRF A00-05834/6, "Safety & 10 CFR 50.92 Significant Hazards Consideration Assessment for RPV Stratification Prevention Improvements at Edwin I. Hatch Nuclear Plant Units 1 and 2," April 1994.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 196 TO FACILITY OPERATING LICENSE DPR-57

AND AMENDMENT NO. 136 TO FACILITY OPERATING LICENSE NPF-5

GEORGIA POWER COMPANY, ET AL.

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

By letter dated October 13, 1994, as supplemented by letter dated January 13 and May 4, 1995, Georgia Power Company, et al. (the licensee or GPC), proposed license amendments to change the Technical Specifications (TS) for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The proposed changes would revise the TS to lower the anticipated transient without scram-recirculation pump trip (ATWS-RPT) setpoint by approximately 2 feet 2 inches to minimize the potential for RPTs following reactor scram, and allow restarting the recirculation pump following an RPT when the temperature differential between the coolant at the reactor bottom head and the reactor steam dome cannot be obtained, provided certain conditions are met. If a thermally stratified condition were to develop in the reactor vessel following an unplanned reactor trip and RPT, and a depressurization to atmospheric conditions was required, a delay in restart of at least 24 hours could occur. Such events have occurred at Hatch twice before. The January 13 and May 4, 1995, letters provided clarifying information that did not change the scope of the October 13, 1994, application and the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

Boiling Water Reactor operating experience indicates that the bottom head region of the reactor pressure vessel (RPV) experiences a rapid cooldown following a reactor scram coincident with RPT. The reduction in metal temperature in the bottom head region, while the steam dome remains near rated temperature and pressure, causes an approach toward the TS pressure - temperature limits. This reduction in the bottom head coolant temperature prevents restart of the reactor coolant pumps if the temperature difference between the steam dome and the bottom head exceeds 145 degrees F. The 145 degrees F requirement is specified to avoid thermal fatigue on the control rod drive (CRD) penetration nozzle stub tubes in the bottom head due to a thermal shock by the reactor hot water on the cold stub tubes during the start of the recirculation pumps.

For TS purposes, the bottom head drain line temperature is used as a representation of coolant temperature in the bottom head region. The temperature in the drain line would not be meaningful if the Reactor Clean-up System is secured and would result in no flow through the drain line. If the drain line temperature cannot be determined or it exceeds 145 degrees F limit, RPV is depressurized as required by the TS. This scenario happened at Hatch in August 1992 and October 1993.

To alleviate the thermal stratification problem, two approaches were proposed by the licensee: (1) Reduce the probability of an RPT, and (2) Provide alternatives to the drain line temperature measurement, if the recirculation pumps trip.

The licensee with the assistance of their consultants from General Electric Company (GE) evaluated the proposed changes.

2.1 Lowering of RPV Water Level Setpoint for ATWS-RPT

To reduce the probability of RPTs, GPC proposed to reduce the low-water level ATWS-RPT setpoint from -47 inches (level 2 setpoint) to -73 inches (level 3 scram setpoint is 0). GE evaluated the plant ATWS response that could potentially be affected by the proposed changes, and determined that the most limiting event is the loss-of-feedwater (LOFW) ATWS event. The LOFW was selected, since the low level RPT setpoint is reached during an ATWS event. The staff approved the REDY model that was used for the analysis. Level 2 was reached and the high pressure coolant injection (HPCI) and the reactor core isolation coolant (RCIC) were initiated and the RPV level was stabilized. Since the reactor pressure and the heat flux do not increase above their initial value and no safety/relief valve opens, there is no increase in peak fuel cladding temperature or in suppression pool temperature.

The ATWS analysis acceptance guidelines of RPV discussed in NUREG-0460, "Anticipated Transients Without Scram For Light Water Reactors," dated April 1978, (less than 1500 psig) and the suppression pool temperature (less than 200 degrees F) are satisfied.

It should be noted that the RPT on reactor high pressure set at 1095 psig, during an ATWS event, will mitigate the rapid pressurization transients. Therefore, the setpoint will not impact these ATWS scenarios.

The ATWS-RPT is supplied by the analog transmitter trip system (ATTS). Reducing the setpoint may require installation of new slave trip units in ATTS. According to the licensee, the new trip units are functionally identical to the trip units being used at Hatch and within the design capabilities of the ATTS.

The plant transient analysis takes credit only for the reactor high pressure ATWS-RPT which remain unchanged from the current design. Since plant ATWS response will not be significantly impacted, the staff finds the proposed change of lowering the RPV water level setpoint for ATWS-RPT acceptable.

2.2 Restart of the Reactor Recirculation Pump Within 30 Minutes of RPT

When the differential temperature cannot be confirmed, the proposed alternative to the current criteria is to start the recirculation pump within 30 minutes of the RPT, if the drive flow in either loop is above 40% of rated prior to the pump trip. This alternative will be used only when the differential temperature cannot be determined and in situations where neither HPCI nor RCIC injection occurs and feedwater temperature does not decrease below 300 degrees F.

The licensee with the assistance of its consultants from GE evaluated the proposed change. The staff agrees with the licensee (based on GE's review of other operating BWR startup test data) that stratification up to the 145-degree F level does not develop until one hour following the RPT. Restarting the pumps within 30 minutes in situations where neither HPCI nor RCIC injection occurs and the feedwater temperature does not decrease below 300 degrees F, assures sufficient mixing in the lower plenum to avoid thermal stratification and maintain the differential temperature between the steam dome and the bottom head within 145 degrees F. Allowing the early restart is acceptable because the conditions for restart assure that a stratified condition has not yet developed. Therefore, the structures at the vessel bottom will not experience a severe thermal shock resulting from the injection of hot water following the pump restart. The proposed change should reduce the number of RPV blowdowns, reducing thermal cycles on the RPV. It may be noted that the restriction on pump start specified in TS 3.6.D that states "The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50 degrees F of each other" is not changed.

GE's review of plant transient and accident analyses identified no adverse affect due to this change. Therefore, GPC's proposal to restart the recirculation pumps within 30 minutes is acceptable.

The low level RPT setpoint change is justified at both Hatch units based on the GE LOFW ATWS analyses performed. The restart of the pumps after 30 minutes is acceptable under the special conditions based on the BWR operating experience.

Based on its review, the staff finds that the above changes have no adverse impact on safety and does not pose an undue risk to public health and safety. Therefore, the proposed changes to TS 3.3.4.2 and associated Bases for Unit 1 and TS 3.3.4.2 and associated Bases for Unit 2 are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Georgia State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (59 FR 65813 dated December 21, 1994). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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