March 21, 1997

Mr. Jack D. Woodard
Senior Vice President -
Nuclear Operations
Georgia Power Company
P. O. Box 1295
Birmingham, Alabama 35201

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SUBJECT: ISSUANCE OF AMENDMENTS - EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M96752 AND M96753)

Dear Mr. Woodard:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 204 to Facility Operating License DPR-57 and Amendment No. 145 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 7, 1996.

The amendments revise Surveillance Requirements (SRs) 3.1.7.7 and 3.4.3.1, and Limiting Conditions for Operation 3.4.3, 3.5.1, and 3.6.1.6 to increase the nominal mechanical pressure relief setpoints for all of the 11 safety/relief valves (SRVs) to 1150 psig and allow operation with one SRV and its associated functions inoperable. The change will reduce the potential for SRV pilot leakage and the potential for forced outages due to an inoperable SRV during a fuel cycle.

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

Sincerely,

ORIGINAL SIGNED BY:

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. 204 to DPR-57
2. Amendment No. 145 to NPF-5
3. Safety Evaluation

cc w/encl: See next page

SRXB/SC
LPhillips
3/18/97

DOCUMENT NAME: G:\HATCH\HAT96752.AMD

OFFICE	DRPE/PD22/PM	DRPE/PD22/LA	OGC *	DRPE/PD22/D
NAME	K.JABBOUR:cn	L.BERRY		H.BERKOW
DATE	3/18/97	3/18/97	3/18/97	3/18/97
COPY	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY

* See previous concurrence

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

March 21, 1997

Mr. Jack D. Woodard
Senior Vice President -
Nuclear Operations
Georgia Power Company
P. O. Box 1295
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SUBJECT: ISSUANCE OF AMENDMENTS - EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M96752 AND M96753)

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The amendments revise Surveillance Requirements (SRs) 3.1.7.7 and 3.4.3.1, and Limiting Conditions for Operation 3.4.3, 3.5.1, and 3.6.1.6 to increase the nominal mechanical pressure relief setpoints for all of the 11 safety/relief valves (SRVs) to 1150 psig and allow operation with one SRV and its associated functions inoperable. The change will reduce the potential for SRV pilot leakage and the potential for forced outages due to an inoperable SRV during a fuel cycle.

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Sincerely,

Kahtan N. Jabbour

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. 204 to DPR-57
2. Amendment No. 145 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

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. 204
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 7, 1996, 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 Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 204 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Enviromental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to startup from the Unit 1 refueling outage scheduled for fall 1997.

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: March 21, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 204

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 vertical lines indicating the areas of change.

Remove Pages	Insert Pages
3.1-23	3.1-23
3.4-7	3.4-7
3.4-8	3.4-8
3.5-1	3.5-1
3.5-2	3.5-2
3.5-3	3.5-3*
3.5-4	3.5-4*
3.5-5	3.5-5*
3.5-6	3.5-6*
3.6-18	3.6-18

*no change - overflow page

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.5	Verify the concentration of sodium pentaborate in solution is within the Region A limits of Figure 3.1.7-1.	31 days AND Once within 24 hours after water or sodium pentaborate is added to solution AND Once within 24 hours after solution temperature is restored within the Region A limits of Figure 3.1.7-2
SR 3.1.7.6	Verify each SLC subsystem manual and power operated valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position.	31 days
SR 3.1.7.7	Verify each pump develops a flow rate $\geq$ 41.2 gpm at a discharge pressure $\geq$ 1232 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	18 months on a STAGGERED TEST BASIS

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (S/RVs)

LCO 3.4.3 The safety function of 10 of 11 S/RVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Two or more S/RVs inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the S/RVs are as follows: Number of S/RVs — Setpoint (psig) 11 — 1150 ± 34.5 Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program
SR 3.4.3.2	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify each S/RV opens when manually actuated.	18 months

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS — Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six of seven safety/ relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable.	A.1	Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	36 hours
C. HPCI System inoperable.	C.1	Verify by administrative means RCIC System is OPERABLE.	1 hour
	<u>AND</u>		
	C.2	Restore HPCI System to OPERABLE status.	14 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. HPCI System inoperable. AND One low pressure ECCS injection/spray subsystem is inoperable.	D.1	Restore HPCI System to OPERABLE status.	72 hours
	OR		
	D.2	Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours
E. Two or more ADS valves inoperable. OR Required Action and associated Completion Time of Condition C or D not met.	E.1	Be in MODE 3.	12 hours
	AND		
	E.2	Reduce reactor steam dome pressure to ≤ 150 psig.	36 hours
F. Two or more low pressure ECCS injection/spray subsystems inoperable. OR HPCI System and two or more ADS valves inoperable.	F.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.1.2	-------------------NOTE------------------- Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than the Residual Heat Removal (RHR) low pressure permissive pressure in MODE 3, if capable of being manually realigned and not otherwise inoperable. -- Verify each ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	 31 days
SR 3.5.1.3	Verify ADS air supply header pressure is $\geq$ 90 psig.	31 days
SR 3.5.1.4	Verify the RHR System cross tie valve is closed and power is removed from the valve operator.	31 days
SR 3.5.1.5	Verify each LPCI inverter output voltage is $\geq$ 570 V and $\leq$ 606 V while supplying the respective bus.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.6	-------------------NOTE------------------- Only required to be performed prior to entering MODE 2 from MODE 3 or 4, when in MODE 4 > 48 hours. -- Verify each recirculation pump discharge valve cycles through one complete cycle of full travel or is de-energized in the closed position.	31 days
SR 3.5.1.7	Verify the following ECCS pumps develop the specified flow rate against a system head corresponding to the specified reactor pressure. SYSTEM / FLOW RATE / NO. OF PUMPS / SYSTEM HEAD CORRESPONDING TO A REACTOR PRESSURE OF CS / ≥ 4250 gpm / 1 / ≥ 113 psig LPCI / ≥ 17,000 gpm / 2 / ≥ 20 psig	In accordance with the Inservice Testing Program
SR 3.5.1.8	-------------------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 ≤ 1058 psig and ≥ 920 psig, the HPCI pump can develop a flow rate ≥ 4250 gpm against a system head corresponding to reactor pressure.	92 days

(continued)

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 system 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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
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

3.6 CONTAINMENT SYSTEMS

3.6.1.6 Low-Low Set (LLS) Valves

LCO 3.6.1.6 The LLS function of three of four safety/relief valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Two or more LLS valves inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours

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. 145
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 7, 1996, 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 Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 145 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to startup from the Unit 2 refueling outage currently scheduled for March 15, 1997.

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: March 21, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 145

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
3.1-23	3.1-23
3.4-7	3.4-7
3.4-8	3.4-8
3.5-1	3.5-1
3.5-2	3.5-2
3.5-3	3.5-3*
3.5-4	3.5-4*
3.5-5	3.5-5*
3.5-6	3.5-6*
3.5-6a	---
3.5-6b	---
3.6-18	3.6-18

*no change - overflow page

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.5	Verify the concentration of sodium pentaborate in solution is within the Region A limits of Figure 3.1.7-1.	31 days AND Once within 24 hours after water or sodium pentaborate is added to solution AND Once within 24 hours after solution temperature is restored within the Region A limits of Figure 3.1.7-2
SR 3.1.7.6	Verify each SLC subsystem manual and power operated valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position.	31 days
SR 3.1.7.7	Verify each pump develops a flow rate $\geq$ 41.2 gpm at a discharge pressure $\geq$ 1232 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	18 months on a STAGGERED TEST BASIS

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (S/RVs)

LCO 3.4.3 The safety function of 10 of 11 S/RVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Two or more S/RVs inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the S/RVs are as follows: Number of S/RVs — Setpoint (psig) 11 — 1150 ± 34.5 Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program
SR 3.4.3.2	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify each S/RV opens when manually actuated.	18 months

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS — Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six of seven safety/ relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. HPCI System inoperable.	C.1 Verify by administrative means RCIC System is OPERABLE. AND C.2 Restore HPCI System to OPERABLE status.	1 hour 14 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. HPCI System inoperable. AND One low pressure ECCS injection/spray subsystem is inoperable.	D.1 Restore HPCI System to OPERABLE status. OR D.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours 72 hours
E. Two or more ADS valves inoperable. OR Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3. AND E.2 Reduce reactor steam dome pressure to ≤ 150 psig.	12 hours 36 hours
F. Two or more low pressure ECCS injection/spray subsystems inoperable. OR HPCI System and two or more ADS valves inoperable.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.1.2	-------------------NOTE------------------- Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than the Residual Heat Removal (RHR) low pressure permissive pressure in MODE 3, if capable of being manually realigned and not otherwise inoperable. -- Verify each ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.1.3	Verify ADS air supply header pressure is ≥ 90 psig.	31 days
SR 3.5.1.4	Verify the RHR System cross tie valve is closed and power is removed from the valve operator.	31 days
SR 3.5.1.5	Verify each LPCI inverter output voltage is ≥ 570 V and ≤ 606 V while supplying the respective bus.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.6	-------------------NOTE------------------- Only required to be performed prior to entering MODE 2 from MODE 3 or 4, when in MODE 4 > 48 hours. --- Verify each recirculation pump discharge valve cycles through one complete cycle of full travel or is de-energized in the closed position.	31 days
SR 3.5.1.7	Verify the following ECCS pumps develop the specified flow rate against a system head corresponding to the specified reactor pressure. SYSTEM / FLOW RATE / NO. OF PUMPS / SYSTEM HEAD CORRESPONDING TO A REACTOR PRESSURE OF CS / ≥ 4250 gpm / 1 / ≥ 113 psig LPCI / ≥ 17,000 gpm / 2 / ≥ 20 psig	In accordance with the Inservice Testing Program
SR 3.5.1.8	-------------------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 ≤ 1058 psig and ≥ 920 psig, the HPCI pump can develop a flow rate ≥ 4250 gpm against a system head corresponding to reactor pressure.	92 days

(continued)

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

3.6 CONTAINMENT SYSTEMS

3.6.1.6 Low-Low Set (LLS) Valves

LCO 3.6.1.6 The LLS function of three of four safety/relief valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Two or more LLS valves inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 204 TO FACILITY OPERATING LICENSE DPR-57

AND AMENDMENT NO. 145 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 7, 1996, 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 allow the units to be operated with one Safety/Relief Valve (SRV) out of service (Reference 1), and increased mechanical lifting setpoints from between 1120 and 1140 psi to 1150 psi. GPC had the General Electric Company (GE) perform a safety evaluation of the proposed revision and the GE report is included in the GPC submittal and is referenced as the justification for the proposed revision.

The proposed revision affects the pressure rise of the vessel during transient reactor behavior. Both of these changes will tend to increase the peak pressure. The staff has evaluated the proposed revision by performing a series of audit calculations to assess the effects of the changes. The staff used its TRAC-BF1[2] evaluation model (Reference 2) with a generic BWR4 input deck. The deck design and its validation are discussed in Reference 3. The staff modified the base deck by adding a more realistic control system model and adding a model of the steam line. Also, the SRV model was modified to model 11 valves instead of 13 for application to Hatch Units 1 and 2.

2.0 EVALUATION

The licensee enclosed in its submittal (Reference 1) an analysis performed by GE of the impact of the proposed changes to the SRV setpoints and allowing one SRV to be out of service. The analysis considered a wide spectrum of events and equipment performance characteristics (High Pressure Coolant Injection, Automatic Depressurization System, etc.). The limiting Anticipated Operational Occurrence (the Turbine Trip without Bypass) was evaluated with the approved ODYN code (Reference 4). The analysis demonstrates that a margin of more than 50 psi exists to the American Society of Mechanical Engineers (ASME) Code limit of 1375 psi. Finally, GE considered the effect of the modification on the Loss-of-Coolant Accident and the plant's Anticipated Transient without SCRAM (ATWS) mitigation capability. GE concluded that the effect of the changes on the LOCA results would be minimal and that the ATWS response remains within acceptable limits.

As part of its evaluation, the staff performed the following audit calculations:

1. The Main Steam Isolation Valve (MSIV) Closure Anticipated Transient Without SCRAM (ATWS) using original SRV setpoints.

2. MSIV Closure ATWS assuming all SRV's open at the relief setpoint of 1150 psi plus a 3% tolerance to account for valve drift.

3. Same as 2 above except that one SRV is assumed out of service.

4. A turbine trip without bypass transient with all valves at the relief setpoint of 1150 psi plus a 3% tolerance to account for valve drift. The recirculation pump trip is disabled.

5. Same as 4 above except that one SRV is assumed out of service.

These calculations provide the limiting pressure rise. The acceptance criterion for cases 1 through 3 is that the peak vessel pressure should remain below 1500 psi. For cases 4 and 5, the acceptance criterion is that the peak vessel pressure should remain below 1325 psi (the 1375 psi ASME code limit minus a 50 psi margin). The use of the relief setpoints plus a 3% tolerance and, for the case of the turbine trip analyses, assuming a malfunction of the recirculation pump trip, are conservative assumptions because they increase the calculated peak pressure.

The licensee stated in its submittal that the purpose of the amendment is to reduce the potential for pilot valve leakage and the potential for forced outages due to an inoperable SRV during a fuel cycle. The staff concludes that by allowing one SRV to be out of service, the likelihood of having to shut down due to SRV failures will be reduced. Furthermore, the proposed modification increases the margin between the mechanical relief setpoints and the dome pressure to a value consistent with other units and should reduce the likelihood of pilot valve leakage.

The results of the staff's evaluation show that all the applicable acceptance criteria are met with considerable margin. The staff's audit calculation results are less conservative than the licensee's results, but this is consistent with the best-estimate nature of the staff's model.

3.0 TS CHANGES

The following discussion of changes to the Limiting Conditions of Operation (LCO) and Surveillance Requirements (SRs) applies to both Hatch Units 1 and 2.

- LCO 3.4.3 is changed to read that "10 out of 11" SRVs shall be OPERABLE from "10." This change is acceptable based upon both the licensee's calculations, and the staff's audit calculations, which demonstrate that all of the applicable limits described above continue to be met.

- SR 3.4.3.1 is changed to read that the nominal mechanical trip setpoints for all of the 11 SRV's are 1150 psig ± 34.5 psi. This change is acceptable based upon both the licensee's calculations and the staff's audit calculations, which demonstrate that all of the applicable limits continue to be met. This also has an effect on the performance of Emergency Core Cooling Systems (ECCS) and the Standby Liquid Control System (SLCS). The licensee stated that the SLCS pumps are positive displacement pumps and the increase in dome pressure is within their capacity. The licensee's submittal states that the ECCS pumps can deliver the required flow at the elevated pressures associated with this change. This conclusion is based upon examination of the pump performance curves.

- LCO 3.5.1 is changed to read that "six of seven" automatic depressurization system (ADS) valves shall be OPERABLE instead of "seven." The licensee stated that one less ADS valve will not affect the Large Break Loss-of-Coolant Accident (LBLOCA) results for Hatch because the plant depressurizes very fast. The staff agrees with the licensee's statement. GE studied the effect of one less ADS valve on the Small Break Loss-of-Coolant accident (SBLOCA) results for Hatch and concluded that the effect is minimal and the Large Break results remain limiting (with calculated LBLOCA Peak Clad Temperature (PCT) 400 degrees centigrade higher than SBLOCA results). This is due to the fact that during SBLOCA events, heat-up rates are very small (on the order of several degrees per second), and increasing the depressurization time by allowing one SRV to be out of service will not raise the limiting SBLOCA PCT by 400 degrees centigrade. The staff agrees with this conclusion.

- LCO 3.6.1.6 is changed to state that "three of four" Lower Level Setpoint (LLS) valves shall be operable instead of "four." The LLS system is intended to reduce the probability of valve failures by reducing the number of valves that cycle. The LLS system consists of 4 SRVs designated to function in the LLS mode. LLS logic is armed following one full cycle of SRVs in the normal mode of operation. The reduction of the number of LLS valves could impact the probability of an SRV sticking open because the number of SRVs forced to open to maintain the system pressure could increase. The probability of an SRV sticking open is directly proportional to the number of valves open. However, this is acceptable because control of the reactor if an SRV sticks open is provided for by the plant's Emergency Operating Procedures.

- SR 3.1.7.7 is changed to reflect the fact that the SLCS pumps have to deliver their flow at a higher pressure (1232 psi). This change is consistent with the discussion of the changes to SR 3.4.3.1 and is, therefore, acceptable.

Based on its review of the licensee's submittal, the staff concludes that the proposed changes to Hatch Units 1 and 2 TS are acceptable.

4.0 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.

5.0 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 (62 FR 129 dated January 2, 1997). 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.

6.0 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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RREFERENCES

1. Letter from J. D. Woodward (GPC) to USNRC, "Request to Revise Technical Specifications: Safety/Relief Valve Setpoint Change," October 7, 1996, and attachments.

2. Borkowski, J. A., TRAC-BF1: An Advanced Best Estimate Computer Program for BWR Accident Analysis, NUREG/CR-4356, August 1992.

3. Letter from David B. Matthews (USNRC) to K. P. Donovan (BWROG), "ACCEPTANCE OF PROPOSED MODIFICATIONS TO THE BOILING WATER REACTOR (BWR) EMERGENCY PROCEDURE GUIDELINES (TAC NOS. M89489 AND M89629)," June 6, 1996.

4. Letter R. L. Tedesco (NRC) to G. G. Sherwood (GENE), "ACCEPTANCE FOR REFERENCING GENERAL ELECTRIC LICENSING TOPICAL REPORT NEDO-24154/NEDE-24154P,"