October 2, 1991

Docket Nos. 50-321
and 50-366

Distribution
See next page

Mr. W. G. Hairston, III
Senior Vice President -
Nuclear Operations
Georgia Power Company
P. O. Box 1295
Birmingham, Alabama 35201

Dear Mr. Hairston:

SUBJECT: ISSUANCE OF AMENDMENT NO. 174 TO FACILITY OPERATING LICENSE DPR-57 AND AMENDMENT NO. 114 TO FACILITY OPERATING LICENSE NPF-5 EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 (TACs 76027/76028)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 174 to Facility Operating License No. DPR-57 and Amendment No. 114 to Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated January 26, 1990, as supplemented January 15, 1991.

The amendments revise TS 3/4.7 for Hatch 1 and TS 3/4.6.2.1 for Hatch 2 which would incorporate into the TSs a means for determining the suppression pool bulk temperature. The TS Bases for both units are also revised. Based on a July 22, 1991, telephone conversation between F. Rinaldi and A. Angelo, NRC, and K. McElroy, GPC, you agreed to withdraw paragraph (d) of the January 15, 1991, letter because this item is covered by other TSs.

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

Sincerely,

/s/
Kahtan N. Jabbour, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 174 to DPR-57
2. Amendment No. 114 to NPF-5
3. Safety Evaluation

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NAME	LBerry	FRinaldi:rst	KJabbour	CMCracken	MZobler	DMatthews
DATE	9/11/91	9/11/91	9/11/91	9/13/91	9/16/91	10/1/91

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DATED: October 2, 1991

AMENDMENT NO.174TO FACILITY OPERATING LICENSE DPR-57 - Edwin I. Hatch Nuclear Plant, Unit 1
AMENDMENT NO.114TO FACILITY OPERATING LICENSE NPF-5 - Edwin I. Hatch Nuclear Plant, Unit 2

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

October 2, 1991

Docket Nos. 50-321
and 50-366

Mr. W. G. Hairston, III
Senior Vice President -
Nuclear Operations
Georgia Power Company
P. O. Box 1295
Birmingham, Alabama 35201

Dear Mr. Hairston:

SUBJECT: ISSUANCE OF AMENDMENT NO. 174 TO FACILITY OPERATING LICENSE DPR-57 AND AMENDMENT NO. 114 TO FACILITY OPERATING LICENSE NPF-5 EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 (TACs 76027/76028)

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The amendments revise TS 3/4.7 for Hatch 1 and TS 3/4.6.2.1 for Hatch 2 which would incorporate into the TSs a means for determining the suppression pool bulk temperature. The TS Bases for both units are also revised. Based on a July 22, 1991, telephone conversation between F. Rinaldi and A. Angelo, NRC, and K. McElroy, GPC, you agreed to withdraw paragraph (d) of the January 15, 1991, letter because this item is covered by other TSs.

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

Sincerely,

Kahtan N. Jabbour

Kahtan N. Jabbour, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 174 to DPR-57
2. Amendment No. 114 to NPF-5
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. W. G. Hairston, III
Georgia Power Company

Edwin I. Hatch Nuclear Plant

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 174
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 Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees) dated January 26, 1990, as supplemented January 15, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this license 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.

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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. 174, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: October 2, 1991

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 114
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 Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees) dated January 26, 1990, as supplemented January 15, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this license 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. 114, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: October 2, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 174

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
3.7-1	3.7-1
3.7-1a	3.7-1a
3.7-28	3.7-28
3.7-29	3.7-29
--	3.7-29a
--	3.7-29b

3.7. CONTAINMENT SYSTEMS

Applicability

The Limiting Conditions for Operation associated with containment systems apply to the operating status of the primary and secondary containment systems.

Objective

The objective of the Limiting Conditions for Operation is to assure the integrity of the primary and secondary containment systems.

Specifications

A. Primary Containment

1. Pressure Suppression Chamber

At any time that irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression chamber water level and average water temperature shall be maintained within the following limits except while performing low-power physics tests at atmospheric pressure at power levels not to exceed 5 Mwt.

a. Minimum water level - 12 feet, 2 inches.

b. Maximum water level - 12 feet, 6 inches.

c. During normal power operation, the average suppression chamber water temperature shall be maintained ≤ 100°F. If this temperature limit is exceeded, pool cooling shall be initiated immediately.

If the average water temperature cannot be restored to ≤ 100°F within 24 hours, the reactor shall be shut down using normal shutdown procedures.

4.7. CONTAINMENT SYSTEMS

Applicability

The Surveillance Requirements associated with containment systems apply to the primary and secondary containment integrity.

Objective

The objective of the Surveillance Requirements is to verify the integrity of the primary and secondary containment.

Specifications

A. Primary Containment

1. Pressure Suppression Chamber

a. The pressure suppression chamber water level, average water temperature and air temperature shall be measured and recorded daily. The average suppression chamber water temperature shall be determined using a weighted average of the suppression pool temperature sensors, as described in Bases paragraph 3.7.A.1.

b. The interior painted surfaces above the level 1 foot below the normal water line of the pressure suppression chamber shall be visually inspected once per operating cycle. In addition, the external surfaces of the pressure suppression chamber shall be visually inspected on a routine basis for evidence of corrosion or leakage.

c. Whenever there is indication that a significant amount of heat is being added to the pressure suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

d. During relief valve operation or testing of RCIC, HPCI, or other testing which adds heat to the suppression pool, the maximum average water temperature shall not exceed 105°F. In connection with such testing, the average pool temperature must be reduced within 24 hours to ≤ 100°F.

e. The reactor shall be scrammed from any operating condition when the average suppression pool temperature reaches 110°F. Operation shall not be resumed until the pool temperature is reduced to below the normal power operation limit specified in c. above.

f. During reactor isolation conditions the reactor pressure vessel shall be depressurized to < 200 psig at normal cooldown rates if the average pool temperature reaches 120°F.

d. Whenever there is indication that there was relief valve operation with the average temperature of the suppression pool exceeding 160°F and the reactor primary coolant system pressure greater than 200 psig, an external visual examination of the pressure suppression chamber shall be conducted before resuming power operation.

3.7 CONTAINMENT SYSTEMS

A. Primary Containment

The integrity of the primary containment and operation of the emergency core cooling systems in combination, limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the initial startup test program is being conducted. There will be no pressure on the system at this time, which greatly reduces the changes of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect to minimize the probability of an accident occurring. Procedures for rod withdrawal and patterns programmed into the Rod Worth Minimizer and Rod Sequence Control System would limit control rod worth to less than 1.25% Δk. A drop of such a rod does not result in any fuel damage. In addition in the unlikely event that an excursion did occur, the secondary containment (reactor building) and standby gas treatment system, which shall be operational during this time, offers a sufficient barrier to keep off-site doses well below 10 CFR 100 limits.

1. Pressure Suppression Chamber

The pressure suppression chamber water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdowns.

Since all of the non-condensable gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compressure plus the vapor pressure of the liquid must not exceed 62 psig, the maximum pressure. The design volume of the pressure suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. Reference FSAR Section 5.2.3.

Using the minimum or maximum water levels given in the specification, containment pressure during the design basis accident is less than 59 psig which is below the maximum pressure of 62 psig. The minimum water level of 12 ft 2 in. corresponds to a water volume of 87,300 ft^3 and a downcomer submergence of 3 ft 8-1/2 in. The maximum water level of 12 ft 6 in. corresponds to a water volume of 90,380 ft^3. The corresponding downcomer submergence is 4 ft 1/2 in. Since the majority of the Bodega tests (Reference 1) were run with a submergence of 4 ft and with complete condensation, this specification is adequate with respect to downcomer submergence.

3.7.A.1. Pressure Suppression Chamber (Continued)

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the pressure suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high pressure suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define that action to be taken in the event a relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

The average (or bulk) suppression pool temperature limits specified in paragraphs 3.7.A.1 and 4.7.A.1 are normally monitored using a weighted average of 15 temperature sensors. Four sensors, T48-N009A through N009D, are located in the lower half of the suppression pool and 11 sensors, T48-N301 through N311, are located in the upper half of the suppression pool. The 4 lower sensors are averaged and the 11 upper sensors are averaged. The bulk suppression pool temperature is the average of the upper and lower average temperature. Should more than two of the 300 series sensors be determined inoperable when the suppression chamber is required, a preplanned alternate method of determining average temperature may be used. The table below illustrates the correction factor (if any) to be added to the operable N009 series elements in this condition. These correction factors for plant conditions (a), (b), and (c) were developed from a detailed review of Plant Hatch suppression pool temperature data.

Plant Condition (See Notes)	Correction Factor (°F) to Operable N009A-D Elements
(a) Normal Operation; Torus Cooling not Operating (Note 1); No HPCI Testing (Note 2); No Leaking SRV(s) (Note 3)	5
(b) Normal Operation; With or without Torus Cooling Operating; HPCI Testing; With or without Leaking SRV(s)	(Note 4)
(c) Normal Operation; Torus Cooling Operating; No HPCI Testing; With or without Leaking SRV(s)	0

Notes:

(1) Torus cooling is at least one loop of RHR in pool cooling or torus spray mode.

(2) The Technical Specification limit for this condition is 105°F.

(3) A leaking SRV is defined as an SRV experiencing significant steam leakage past the seat. All the steam is not condensed in the SRV discharge line and, therefore, results in steam expulsion into pool.

(4) Without upper pool temperature indication, HPCI testing time should be limited to assure the bulk pool temperature does not exceed 105°F. Pool temperature data should still be recorded each 5 minutes as instructed by Technical Specifications, but the run time should be administratively controlled by the following:

Max Run Time in Minutes = $(105 - T_{initial}) \times 2$,

where $T_{initial}$ is the pool temperature taken prior to the test with torus cooling operating. This equation assumes a 30°F/hour rise in bulk pool temperature.

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ATTACHMENT TO LICENSE AMENDMENT NO. 114

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
3/4 6-11	3/4 6-11
3/4 6-12	3/4 6-12
3/4 6-13	3/4 6-13
B 3/4 6-4	B 3/4 6-4
--	B 3/4 6-4a
--	B 3/4 6-4b

3/4 6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with the pool water:

a. Volume between 87,300 ft^3, and 90,550 ft^3, equivalent to a level between 12 ft 2 in. and 12 ft 6 in., and a

b. Maximum average temperature of 100°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

1. 105°F during testing which adds heat to the suppression chamber during OPERATIONAL CONDITION 1 or 2,

2. 120°F with the main steam line isolation valves closed following a scram from OPERATIONAL CONDITION 1 or 2.

c. Level instrumentation channels alarms adjusted to actuate at:

1. High water level of ≤ 12 ft 6 in.

2. Low water level of ≥ 12 ft 2 in.

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

a. With the suppression chamber water volume outside the above limits, restore the volume to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 1 or 2 with the average suppression chamber water temperature > 100°F, except as permitted above, initiate suppression pool cooling and restore the temperature to ≤ 100°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. In OPERATIONAL CONDITION 1 or 2 with the average suppression chamber water temperature > 105°F during testing which adds heat to the suppression chamber, stop all testing, initiate suppression pool cooling, and restore the temperature to ≤ 100°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. In OPERATIONAL CONDITION 1 or 2 with THERMAL POWER > 1 percent of RATED THERMAL POWER and the average suppression chamber water temperature > 110°F, place the reactor mode switch in the Shutdown position.

e. With the average suppression chamber water temperature > 120°F and the main steam isolation valves closed following a scram from OPERATIONAL CONDITION 1 or 2, depressurize the reactor pressure vessel to < 200 psig at normal cooldown rates.

f. With one suppression chamber water level instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

g. With both suppression chamber water level instrumentation channels inoperable, restore at least one inoperable channel to OPERABLE status within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be between 12 ft 2 in. and 12 ft 6 in. at least once per 24 hours

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the average* suppression chamber water temperature to be ≤ 100°F.

c. At least once per 5 minutes in OPERATIONAL CONDITION 1 or 2 during testing which adds heat to the suppression chamber, by verifying the average* suppression chamber water temperature ≤ 105°F.

d. At least once per 60 minutes when THERMAL POWER > 1 percent of RATED THERMAL POWER and average* suppression chamber water temperature > 100°F, by verifying average* suppression chamber water temperature < 110°F.

*The average suppression chamber water temperature shall be determined using a weighted average of the suppression pool temperature sensors, as described in BASES subsection 3/4.6.2.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 30 minutes following a scram from OPERATIONAL CONDITION 1 or 2 with the main steam line isolation valves closed, and average* suppression chamber water temperature > 100°F, by verifying average* suppression chamber water temperature < 120°F.

f. By an external visual examination of the suppression chamber after there has been indication of safety/relief valve operation with the average* suppression chamber water temperature ≥ 160°F and reactor coolant system pressure > 200 psig.

g. At least once per 18 months by a visual inspection of the accessible interior and exterior of the suppression chamber.

h. By verifying two suppression chamber water level instrumentation channels (2T48-R607A,B) OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

3. CHANNEL CALIBRATION at least once per 6 months.

*The average suppression water temperature shall be determined using a weighted average of the suppression pool temperature sensors, as described in BASES subsection 3/4.6.2.

DEPRESSURIZATION SYSTEMS (Continued)

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety/relief valve inadvertently opens or sticks open. As a minimum, this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety/relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety/relief valve to assure mixing and uniformity of energy insertion to the pool.

The average (or bulk) suppression pool temperature limits specified in paragraphs 3.6.2.1 and 4.6.2.1 are normally monitored using a weighted average of 15 temperature sensors. Four sensors, 2T48-N009A through N009D, are located in the lower half of the suppression pool and 11 sensors, 2T48-N301 through N311, are located in the upper half of the suppression pool. The 4 lower sensors are averaged and the 11 upper sensors are averaged. The bulk suppression pool temperature is the average of the upper and lower average temperature. Should more than two of the 300 series sensors be determined inoperable while in Conditions 1, 2, or 3, a preplanned alternate method of determining average temperature may be used. The table below illustrates the correction factor (if any) to be added to the operable N009 series elements in this condition. These correction factors for plant conditions (a), (b), and (c) were developed from a detailed review of Plant Hatch suppression pool temperature data.

Plant Condition (See Notes)	Correction Factor (°F) to Operable N009A-D Elements
(a) Normal Operation; Torus Cooling not Operating (Note 1); No HPCI Testing (Note 2); No Leaking SRV(s) (Note 3)	5
(b) Normal Operation; With or without Torus Cooling Operating; HPCI Testing; With or without Leaking SRV(s)	(Note 4)
(c) Normal Operation; Torus Cooling Operating; No HPCI Testing; With or without Leaking SRV(s)	0

Notes:

1. Torus cooling is at least one loop of RHR in pool cooling or torus spray mode.

2. The Technical Specification limit for this condition is 105°F.

3. A leaking SRV is defined as an SRV experiencing significant steam leakage past the seat. All the steam is not condensed in the SRV discharge line and, therefore, results in steam expulsion into pool.

4. Without upper pool temperature indication, HPCI testing time should be limited to assure the bulk pool temperature does not exceed 105°F. Pool temperature data should still be recorded each 5 minutes as instructed by Technical Specifications, but the run time should be administratively controlled by the following:

   Max Run Time in Minutes = $(105 - T_{initial}) \times 2$,

   where $T_{initial}$ is the pool temperature taken prior to the test with torus cooling operating. This equation assumes a 30°F/hour rise in bulk pool temperature.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the primary containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the primary containment atmosphere or pressurization of the containment. Primary containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA. Only one closed valve in each penetration line is required to maintain the integrity of the containment.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 174 TO FACILITY OPERATING LICENSE DPR-57

AND AMENDMENT NO. 114 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 January 26, 1990, as supplemented January 15, 1991, the Georgia Power Company, et al. (the licensee), submitted a request for changes to the Edwin I. Hatch Nuclear Plant, Units 1 and 2, Technical Specifications (TSs). The requested changes would revise Technical Specification (TS) 3/4.7 for Hatch Unit 1 and TS 3/4.6.2.1 for Hatch Unit 2. The TS Bases for both units are also revised. The proposed changes incorporate into the TSs a means for determining the suppression pool bulk temperature. This value will be used to show compliance with existing TS operational limits. The licensee's proposal, originally described in a January 26, 1990 submittal and revised per the subject letter, describes the preferred method of using an average of all 15 temperature elements as read from the Safety Parameter Display System (SPDS) to determine the bulk pool temperature. The licensee also proposed preplanned alternate methods of temperature monitoring when the preferred method is unavailable due to inoperable temperature elements of SPDS.

2.0 EVALUATION

Temperature limits have been placed on the suppression pool for BWRs with Mark I containments to minimize the potential for high amplitude plant loads during transient or accident events. Because the suppression pool is the intermediate heat sink for this plant design, it is important that its water temperature be monitored during normal operations as well as during transient/accident conditions. The NRC's requirements are based on "local" pool temperature responses. Correlations are used to relate "local" to "bulk" conditions. Today, the "bulk" or average water temperature in the suppression pool is used as the initial condition for many safety analyses, and as such, it is an important plant parameter which must be monitored during normal, abnormal and emergency conditions.

The existing Plant Hatch TSs makes no reference to "average" or "bulk" temperatures and does not describe any methods used to monitor them. The licensee's proposal specifies that the "bulk" pool temperature will be used to show compliance with existing TS limits. Furthermore, the TS change describes the preferred method for obtaining this critical parameter as the average of all 15 temperature elements (eleven N300 series upper sensors, four N009 series lower sensors) in each unit. The new TS Bases will describe the preplanned alternatives for measuring the "bulk" pool temperature during periods when more than two of the upper temperature elements (TEs) are inoperable.

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In its submittal, the licensee stated that the N300 (upper) series sensors feed into a single-point recorder and are not single-failure proof. Also, Plant Hatch's SPDS, from which the bulk pool temperature is usually monitored, is neither single-failure proof nor subject to TS requirements for operability. Therefore, Georgia Power postulated the failures of more than two of the upper TEs during normal and abnormal operating conditions and provided alternative methods for conservative suppression pool temperature monitoring. The proposed alternatives to using the SPDS for various operating conditions are as follows:

(a) Normal Operation; Torus cooling not operating; No high pressure coolant injection (HPCI) testing; No leaking safety relief valve(s) (SRV(s))

In the event that more than two of the eleven N300 series sensors become inoperable, the licensee proposed to add 5 degree Fahrenheit correction factor to the average of the four operable N009A-D elements. For the above plant condition, the licensee's 1989 study demonstrated that while suppression pool water is not being circulated, thermal stratification caused only minor temperature variances between the upper and lower regions of the pool. The largest temperature differential observed in either unit during the surveillance checks was 3.5 degrees Fahrenheit with the average being about 2 degrees. This is bounded by the 5 degree correction factor the licensee has proposed for normal operation with no significant heat energy being added to the pool. Based on the above, the NRC staff finds the licensee's proposed suppression pool monitoring alternative for this plant condition to be acceptable.

(b) Normal Operation; With or without torus cooling operating; HPCI testing on; with or without leaking SRV(s)

Should more than two of the upper suppression pool temperature sensors be deemed inoperable, the licensee has proposed that HPCI testing during this plant condition be administratively limited to assure the bulk pool temperature does not exceed 105 degrees Fahrenheit. Pool temperature data should still be recorded each 5 minutes as instructed by the TSs, but the HPCI run time should be administratively controlled by the following equation:

> Max Run Time in Minutes = (105 - T initial) x 2, where T initial is the pool temperature taken prior to the test with torus cooling operating.

This equation assumes a 30 degree/hour rise in bulk pool temperature. In its submittal, the licensee stated that T initial can be determined by using only the lower TEs, if necessary, since the RHR system will be aligned in the suppression pool cooling (SPC) mode prior to running HPCI. On that basis, the lower TEs' accuracy would be offset by an effective thermal mix in the suppression pool caused by running RHR in SPC mode. Worst case data taken during the licensee's 1989 surveillance runs indicate that since the HPCI system has a large capacity for steam flow, thermal stratification may be significant during HPCI testing. From Figure 2 in the licensee's January 1991 submittal it is shown that the calculated bulk pool temperature rose at an estimated 22.5 degrees/hour during HPCI. This rate of temperature increase is bounded by the 30 degree/hour rate assumed in the proposed equation for administratively controlling HPCI testing. Based on this review, the NRC staff finds the above alternative method for suppression pool temperature monitoring to be acceptable during normal operations with HPCI testing.

(c) Normal Operation; Torus cooling operating; No HPCI testing; With or without leaking SRV(s)

In the event that more than two of the eleven N300 series sensors become inoperable, the licensee proposed to use the average of the four operable N009 elements with no correction factor. The licensee maintains that a correction factor would not be necessary due to at least one RHR pump being in the SPC mode and providing enough flow to effect thermal mixing during normal operations, testing of SRVs, or testing of the RCIC System. Results from the 1989 study indicate that the temperature differential between the average of the N009 Series elements and the bulk pool temperature is on the order of 1 degree Fahrenheit or less during these conditions. Based on the NRC staff's review of the subject submittal, it finds the above-defined method for monitoring suppression pool temperature during normal operations with no significant heat addition to the pool to be acceptable.

(d) Abnormal Operation; With or without torus cooling operating; Significant heat addition to the suppression pool

Based on a telephone conversation with the licensee of July 22, 1991, the licensee has agreed to withdraw this paragraph. Technical Specification (TS) paragraphs 3.7.A.1 for Unit 1 and 3.6.2.1.G for Unit 2 address the suppression pool temperature monitoring during normal power operation with or without HPCI surveillance in process. Paragraph (d) of the licensee's letter dated January 15, 1991, addresses an abnormal plant condition with significant heat addition to the suppression pool. This would be, as a minimum, a condition where a single SRV (or more SRVs) is (or are) discharging to the pool potentially under ATWS conditions.

Therefore, for the licensee to be in a postulated condition similar to paragraph (d), the licensee would have already entered or be about to enter other action statements on high suppression pool temperature and the plant operators would be in process of dealing with the primary cause of the event such as an ATWS or open SRV. The TS paragraphs in question here are monitoring requirement which assure that the suppression pool is operable and capable of receiving the heat input from the postulated transient. If the N300 series sensors, and SPDS become inoperable, and a significant heat addition to the pool was postulated to occur, other TS LCOs would be exceeded requiring a shutdown on the reactor.

On the basis of the above evaluation, the staff concludes that the licensee employed a sound methodology in analyzing suppression pool temperature surveillance data to develop conservative alternative methods for determining this critical parameter. Therefore, the licensee's proposal to change the TSs to specify that a "bulk" temperature be used to show compliance with existing TS limits is acceptable as described in paragraphs (a), (b), and (c) only of the licensee's letter dated January 15, 1991. In addition, the proposed revision to the TS Bases for both units, describing the "preferred method" of using 15 TEs as well as the alternative methods for the various operating conditions, is also acceptable. The licensee's proposal in paragraph (d) of Enclosure 1 to the January 15, 1991, letter was withdrawn by the licensee.

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 requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 (56 FR 27044). 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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