Docket Nos: 50-321
50-366

July 18, 1989

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. 165 TO FACILITY OPERATING LICENSE DPR-57 AND AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NPF-5 - EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 (TACS 69406/69407)

The Commission has issued the enclosed Amendment No. 165 to Facility Operating License DPR-57 and Amendment No. 102 to Facility Operating License NPF-5 for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated September 6, 1988.

The amendments change the action level regarding suppression pool temperature from 95° F to 100° F.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

/S/

Lawrence P. Crocker, Project Manager
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 165 to DPR-57
2. Amendment No. 102 to NPF-5
3. Safety Evaluation

cc w/ enclosures:
See next page

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cc:
G. F. Trowbridge, Esq.
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N. W.
Washington, D.C. 20037

Mr. L. T. Gucwa
Engineering Department
Georgia Power Company
P. O. Box 1295
Birmingham, Alabama 35201

Nuclear Safety and Compliance Manager
Edwin I. Hatch Nuclear Plant
Georgia Power Company
P. O. Box 442
Baxley, Georgia 31513

Mr. Louis B. Long
Southern Company Services, Inc.
P. O. Box 1295
Birmingham, Alabama 35201

Resident Inspector
U.S. Nuclear Regulatory Commission
Route 1, Box 725
Baxley, Georgia 31513

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, Suite 2900
Atlanta, Georgia 30323

Mr. Charles H. Badger
Office of Planning and Budget
Room 610
270 Washington Street, S.W.
Atlanta, Georgia 30334

Mr. J. Leonard Ledbetter, Director
Environmental Protection Division
Department of Natural Resources
205 Butler Street, S.E., Suite 1252
Atlanta, Georgia 30334

Chairman
Appling County Commissioners
County Courthouse
Baxley, Georgia 31513

Mr. R. P. McDonald
Executive Vice President -
Nuclear Operations
Georgia Power Company
P.O. Box 1295
Birmingham, Alabama 35201

Mr. Alan R. Herdt, Chief
Project Branch #3
U.S. Nuclear Regulatory Commission
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

DATED July 18, 1989

AMENDMENT NO. 165 TO FACILITY OPERATING LICENSE DPR-57, EDWIN I. HATCH, UNIT 1
AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NPF-5, EDWIN I. HATCH, UNIT 2

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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 NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 165
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 licensee) dated September 6, 1988, 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 amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-57 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 165, 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 of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

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

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: July 18, 1989

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ATTACHMENT TO LICENSE AMENDMENT NO. 165

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 area of change.

Remove Page	Insert Page
3.7-1	3.7-1
3.7-1a	3.7-1a
3.7-30	3.7-30

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 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 suppression chamber water temperature shall be maintained $\leq$ 100°F. If this temperature limit is exceeded, pool cooling shall be initiated immediately.

If the water temperature cannot be restored to $\leq$ 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, water temperature and air temperature shall be measured and recorded daily.

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.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
d. During relief valve operation or testing of RCIC, HPCI, or other testing which adds heat to the suppression pool, the maximum water temperature shall not exceed 105°F. In connection with such testing, the pool temperature must be reduced within 24 hours to $\leq$ 100°F.	d. Whenever there is indication that there was relief valve operation with the 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.
e. The reactor shall be scrammed from any operating condition when the 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 pool temperature reaches 120°F.	

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

The maximum pool temperature based on the consideration of complete condensation has been determined by evaluating the blowdown test data from the Mark I Full Scale Test Facility. Based on these analyses, a pool temperature of 195°F can provide complete steam condensation (conservatively assumes no pressurization of the air space over the pool). Analyses for Plant Hatch have shown that with an initial pool temperature of 110°F, the pool temperature following a blowdown will be below that needed for complete condensation. Therefore, the 100°F limit on operating pool temperature is justified.

For an initial suppression pool temperature of 110°F and assuming that one loop of the RHR system is available for containment cooling (2 RHR and 2 RHR service water pumps) adequate net positive suction head (NPSH) is maintained for the core spray, RHR, and HPCI pumps. Therefore, the 100°F limit on operating pool temperature is justified.

Limiting pressure suppression chamber water temperature to 120°F during RCIC, HPCI or relief valve operation when decay heat and stored energy are removed from the primary system by discharging reactor steam directly to the suppression chamber assures adequate margin for controlled blowdown anytime during RCIC operation.

Using a 50°F rise (Table 5.2-1 FSAR) in the pressure suppression chamber water temperature and an initial temperature of < 120°F, the 195°F limit is not exceeded.

If a loss-of-coolant accident were to occur when the reactor water temperature is below 330°F, containment pressure will not exceed the 62 psig maximum pressure even if no condensation were to occur. The maximum allowable pressure suppression chamber water temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus specifying combinations of water volume and temperature requirements applicable for reactor-water temperatures above 212°F provides additional margin above that available at 330°F.

Should it be necessary to drain the pressure suppression chamber, this should only be done when there is no requirement for core standby cooling systems operability, as explained in basis 3.5.G.

2. Primary Containment Integrity

Discussed under Bases for Specification 3.7.A., Primary Containment.

3. Reactor Building to Pressure Suppression Chamber Vacuum Relief System

The purpose of the reactor building to pressure suppression chamber vacuum relief system is to equalize pressure so that the structural integrity of the containment is assured.

The vacuum relief system from the reactor building to the pressure suppression chamber consists of two 100-percent vacuum relief lines, each of which has an air operated valve and a vacuum breaker (check valve) in series. Operation of either line will maintain the pressure differential less than 2 psid, the external design pressure. Reference Section 5.2.3.6.2 of the FSAR.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 102
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 licensee) dated September 6, 1988, 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 amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-5 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 102, 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 of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 18, 1989

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6/29/89	6/29/89	/89	7/3/89	7/10/89	7/17/89

ATTACHMENT TO LICENSE AMENDMENT NO. 102

FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Page	Insert Page
3/4 6-11	3/4 6-11
3/4 6-12	3/4 6-12
3/4 6-13	3/4 6-13
B3/4 6-3	B3/4 6-3

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 temperature of 100°F during OPERATIONAL CONDITION 1 or 2, except that the maximum 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 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 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 suppression chamber water temperature > 110°F, place the reactor mode switch in the Shutdown position.

e. With the 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 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 suppression chamber water temperature ≤ 105°F.

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

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 suppression chamber water temperature > 100°F, by verifying 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 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.

3/4.6.2 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the maximum allowable internal pressure of 62 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1040 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a LOCA, the pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the 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.

Using the minimum or maximum water levels given in the specification, containment pressure during the design basis accident is approximately 57.5 psig which is below the maximum allowable internal pressure of 62 psig. Maximum water level results in a downcomer submergence of 4 ft 4 in. and the minimum water level results in a submergence approximately 4 in. less. The Mark I Full Scale Test Facility tests were performed at several submergence levels which bound this variance, all with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate.

The maximum pool temperature based on the consideration of complete condensation has been determined by evaluating the blowdown test data from the Mark I Full Scale Test Facility. Based on these analyses, a pool temperature of 195°F can provide complete steam condensation (conservatively assumes no pressurization of the air space over the pool). Analyses for Plant Hatch have shown that with an initial pool temperature of 110°F, the pool temperature following a blowdown will be below that needed for complete condensation. Therefore, the 100°F limit on operating pool temperature is justified.

For an initial suppression pool temperature of 110°F and assuming that one loop of the RHR system is available for containment cooling (2 RHR and 2 RHR service water pumps), adequate net positive suction head (NPSH) is maintained for the core spray, RHR, and HPCI pumps. Therefore, the 100°F limit on operating pool temperature is justified.

When it is necessary to make the suppression chamber inoperable, this shall only be done as provided in Specification 3.5.4.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 165 AND 102 TO

FACILITY OPERATING LICENSES DPR-57 AND NPF-5

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

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

By letter dated September 6, 1988, Georgia Power Company, the licensee for the Edwin I. Hatch Nuclear Plant, Units 1 and 2 requested changes to Technical Specifications (TS) 3.7 and associated Bases 3.7.A.1 for Hatch Unit 1, and TS 3.6.2.1 and 4.6.2.1 and associated Bases 3/4.6.2 for Hatch Unit 2. These specifications deal with the limiting conditions of operation (LCO) of the suppression pool (SP) during normal plant operation at conditions 1, 2 and 3 for both the units and the associated surveillance requirement for Unit 2. Specifically, the proposed change would raise the suppression pool temperature limit during normal operation from 95° F to 100° F. The 105° F limit on allowable pool temperature during safety system testing, which adds heat to the suppression pool, will not be changed. Also, the suppression pool temperature limits (SPTL) requiring immediate plant shutdown (110° F) and vessel depressurization (120° F), will remain unchanged.

In recent years, high summertime temperatures have caused the temperature of the Altamaha River, which serves as the ultimate heat sink for the plant service water and residual heat removal (RHR) systems, to rise to the point where an insufficient differential temperature is available to maintain the suppression pool temperature below 95° F. Request for emergency relief from the TS LCO has been imminent on a number of occasions, and processing of an emergency TS change to increase the 95° F limit was in progress during August 1987 when the LCO was cleared.

To avoid the necessity of submitting emergency TS changes regarding the 95° F limit, the licensee proposes to raise the limit from 95° F to 100° F during normal operation. In support of this increase in the suppression pool temperature limit during normal operation, the licensee provided the General Electric (GE) Company's safety evaluation (EAS-19-0388, dated March 1988) of the suppression pool temperature limit for Mark I containment and its applicability to Hatch Units. The GE report discussed the impact of the proposed increase in the pool's operational temperature limit on (1) containment response, (2) safety-relief valve (SRV) operation, (3) emergency core cooling system (ECCS) performance, (4) NPSH for safety system pumps, (5) Hatch emergency operating procedures (EOPs), and (6) anticipated transient without scram (ATWS) evaluations.

2.0 EVALUATION

The events which involve the suppression pool can be divided into two general categories: safety relief valve (SRV) discharge to the pool via the SRV discharge lines and T-Quenchers, and discharges to the pool via the drywell to wetwell vent pipes during design basis loss-of-coolant accidents (LOCA). These are evaluated in sections 2.1 and 2.2 below.

2.1 LOCA-RELATED CONTAINMENT LOADS

The GE safety evaluation of the suppression pool (SP) temperature limit for Hatch Units 1 and 2 discussed the ranges for operational temperature limits for SP water under LOCA conditions to ensure that containment pressures and temperatures and hydrodynamic loads under such conditions do not exceed the design values. The GE evaluation concludes that a normal operating suppression pool temperature up to 100° F for the Hatch units will not affect the design loads. The following paragraphs (a) through (d) summarize these evaluations and discuss their application to the Hatch units.

(a) Containment Pressure and Temperature Design Limits

The GE report compared the pressure and temperature design limits for several Mark I plants (including Hatch) to the predicted maximum containment pressure and temperatures during a LOCA. The report noted that because the design limits are very high for such containments, there is a large margin between the predicted values under LOCA conditions and the design values that would support a large increase in the normal operational pool temperature. Specifically, the report pointed out that based on design pressure and temperature consideration alone, an operational pool temperature in the range of 133° F to 161° F should be acceptable.

(b) Steam Condensation

With regard to the ability of the suppression pool to ensure complete steam condensation following a LOCA, the report stated that based on an analysis of test data for the Mark I full scale test facility (FSTF), GE determined that a normal operational pool temperature in the range of 118° F to 133° F would ensure complete steam condensation because it would correspond to the tested maximum pool temperatures for which complete steam condensation was confirmed.

(c) Condensation Oscillation Loads

The report pointed out that condensation oscillation (CO) loads are primarily affected by two hydrodynamic parameters, i.e., pool temperature and the enthalpy flux through the downcomer vents. Using the

GE-developed correlation between these two parameters and the CO loads under transient conditions, the CO loads for the expected LOCA conditions and the conditions simulated during the FSTF test were determined and compared with plant-specific predictions to determine the margin between the expected and the design CO loads and, subsequently, the associated margin in the pool temperature. The licensee stated that consideration of Hatch plant-specific bounding hydrodynamic parameters would result in a CO load that is less than that assumed in the containment loads evaluation even with a normal operational pool temperature of 110° F (the shutdown limit).

(d) Chugging Loads

The GE report stated that a review of chugging data obtained during the Mark 1 FSTF tests (NEDE 24539-P) indicated that chugging occurs only with small-break LOCAs and relatively low pool temperatures (less than 135° F). The report concluded that the proposed increase in the normal operational pool temperature limit will have no impact on chugging loads.

On the basis of the GE information, the staff concludes that the LOCA-related containment loads resulting from the proposed increase in normal operational pool temperature limit will be within the containment design loads.

2.2 SRV OPERATIONAL LOADS

The SRV operational loads can be divided into two categories. The SRV air clearing load and SRV condensation loads.

(a) SRV Air Clearing Loads

The SRV air clearing loads result from the expulsion of air out of the SRV discharge line into the suppression pool. The expansion and contraction of the air bubble creates an oscillatory load on the containment wall and submerged structures. The SRV air clearing load will increase with a higher initial pool temperature. However, the staff notes that the US Mark I containment program requires that the limiting SRV air clearing load to be considered in containment structural evaluations be determined on the basis of the first actuation of an SRV at the maximum pool temperature permitted by the Mark I plant TS (120° F) with the reactor at operating pressure. The Hatch units also have the same TS limit for suppression pool that would require the reactor to be depressurized. Therefore, the staff agrees with the licensee that the SRV air clearing load resulting from the proposed increase of normal operational pool temperature from 95° F to 100° F will be bounded by the limiting SRV air clearing load for the Hatch units.

(b) SRV Condensation Loads

The licensee referred to GE Topical Report NEDO-30832, "Elimination of Limit on BWR Suppression Pool Temperature for SRV Discharge with Quenchers" submitted to the NRC by the BWR Owners' Group in March 1985. This report had concluded that the local pool temperature limits for the suppression pool to ensure steam condensation under stable conditions during SRV steam discharge into the pool specified in NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments" dated November 1981, could be eliminated for BWRs that utilized T or X-quencher devices. GE concluded the above, based on their findings (tabulated in the NEDO-30832 report) that the SRV condensation loads with the above devices were low in comparison with other loads (e.g., SRV air clearing loads) considered in containment structural evaluation. The staff has not yet completed its evaluation of the above report. Therefore, for this safety evaluation, the staff has used the criterion for local pool temperature limit during SRV steam discharge into the pool that is identified in NUREG-0783 to assess whether the peak local pool temperature resulting from the proposed initial pool temperature of 100° F will meet the criteria given in the NUREG. In January 1983, and in February 1983, the licensee provided plant-unique analysis reports for Hatch, Units 1 and 2 long term containment programs. In these reports, using an initial pool temperature of 95° F and other hydrodynamic parameters, the licensee calculated a bounding local pool temperature of 199° F for the Hatch units during transients involving SRV actuations. The licensee concluded that the Hatch units, therefore, complied with the NUREG-0783 limit for local pool temperature during SRV steam discharge into the pool (200° F). Based on the review of these reports, the staff concluded (SER, dated January 25, 1984) that the licensee employed a conservative methodology to analyze pool temperature transients involving SRV actuations to demonstrate the plant's compliance with NUREG-0783. The staff, therefore, found the calculated temperatures acceptable.

By providing credit for quencher submergence as allowed by the NUREG, the staff has reevaluated the local pool temperature limit for the Hatch units, and concluded that a limit of 204° F is appropriate (Hatch units have about 8 feet quencher submergency; steam flux through quencher perforations is less than 42 lbs m/ft^2-sec, when the peak local pool temperature is reached). The staff has determined that the proposed increase of operational pool temperature by 5° F will not result in a peak pool local temperature higher than the estimated allowable limit of 204° F. Therefore, the staff concludes that there is reasonable assurance that the proposed normal operational pool temperature limit of 100° F will not compromise the ability of the suppression pool to condense steam under stable conditions during SRV discharge of steam into the pool and, therefore, meets the criteria of NUREG-0783. Furthermore, the staff notes that the proposed TS changes will not alter the existing requirements for (1) pool cooling whenever the pool temperature exceeds 100° F, (2) scramming the reactor whenever the pool temperature exceeds 110° F, and (3) depressurizing the reactor whenever the pool temperature exceeds 120° F.

2.3 ECCS PERFORMANCE

The core cooling capability of the ECCS pumps is determined by the ability to keep the peak clad temperature of the fuel to less than 2200° F for all postulated loss of coolant accident (LOCA) events, considering an arbitrary single failure. For the Hatch units, the most limiting LOCA event is a large break in the discharge line of the recirculation loop coupled with a single failure of the low pressure coolant injection (LPCI) valve on the other loop. For this postulated event, the two core spray pumps are the only effective means for core cooling.

The GE report (EAS-19-0388) presented the results of an ECCS analysis using 110° F as the initial pool temperature instead of the 95° F used in the original ECCS calculations. The results indicate that there is no significant impact on the LOCA analysis. Thus, the proposed TS change would not adversely affect ECCS performance.

On the basis of the GE information, the staff concludes that ECCS performance will remain within the limits set by 10 CFR 50, Appendix K, and thus is acceptable.

2.4 NPSH FOR SAFETY SYSTEM PUMPS

In accordance with Regulatory Guide 1.1, it is required that the RHR and core spray pumps have adequate net positive suction head (NPSH) without dependence on positive containment pressure during the worst case LOCA with a single failure.

The initial NPSH calculations for the Hatch units were performed using an initial suppression pool water temperature of 95° F and assuming that all the energy in the reactor pressure vessel was absorbed by the suppression pool water following a LOCA. Using these and other assumptions, the peak suppression pool temperature was calculated to occur at 6.9 hours following a LOCA. At that time, the NPSH margins for both the RHR pumps and the core spray pumps were determined to be adequate (3.94 ft. and 1.34 ft., respectively).

The GE report (EAS-19-0388) presents the results of a re-analysis using all of the assumptions of the initial analysis except that the initial pool temperature was assumed to be 110° F and realistic energy source terms were used. The energy input to the suppression pool was taken to be the blowdown energy from the LOCA plus decay heat calculated using the May-Witt decay heat correlation, which includes a 10% factor for conservatism. The energy input also was calculated using the 1979 ANS decay heat correlation which represents the best estimate decay heat correlation, and results in a calculated peak pool temperature of about 190° F.

Using the revised assumptions and the May-Witt decay heat correlation, GE calculated that the maximum suppression pool temperature would be approximately 212° F which would still result in adequate NPSH for the RHR pumps. At this temperature, the core spray pumps may operate with some cavitation since the required NPSH is about 0.05 feet higher than the available NPSH.

However, the GE report points out that the time at which the peak pool temperature occurs is more than 6 hours into the accident, by which time only about 10% of the core spray rated flow would be required to remove the decay heat. The required NPSH at such reduced flow is significantly less than the NPSH required at full flow. The report also notes that Revision 4 to the Emergency Procedure Guidelines (EPGs) instructs the plant operators to reduce the ECCS pump flow and to turn off unneeded pumps when adequate core cooling is assured. The GE report concludes that, based on the actual NPSH requirements for the core spray pumps at high water temperatures and the required mode of pump operation, the increase in initial pool temperature will still result in adequate NPSH for the core spray pumps.

Based on the GE report, and noting the conservatism built into the May-Witt correlation plus the fact that the calculation was run using 110° F rather than the proposed 100° F as the initial pool temperature, the staff concludes that the RHR and core spray pumps will have adequate NPSH. The NPSH evaluation is limited to the RHR and core spray pumps because neither the HPCI or the RCIC pumps would be operated beyond 6 hours into a LOCA event. The peak pool temperature and the resultant minimum NPSH availability do not occur until after 6 hours into the event.

The staff therefore concludes that the increase in suppression pool temperature requested by the licensee would not have an adverse impact upon the operation of the safety system pumps.

2.5 EMERGENCY OPERATING PROCEDURES (EOPs)

The GE report points out, correctly, that the proposed change in the suppression pool temperature limit would result in some needed changes to the EOPs. However, the staff is not now reviewing the adequacy of EOPs prior to implementation. Thus, this SER does not address changes to the EOPs. As a matter of interest, however, the licensee now is revising the Hatch EOPs to be in accordance with Revision 4 to the Emergency Procedure Guidelines (EPGs). The staff expects that any changes to the EOPs required as a result of this proposed change will be incorporated as a part of the ongoing EOP revision, which will be subject to later staff inspection for adequacy.

2.6 ATWS EVALUATION

The TS for each of the Hatch units now require that the reactor be scrammed by placing the mode switch in the Shutdown position whenever the suppression pool temperature exceeds 110° F. This TS requirement is not changed as a result of the requested TS amendment. Therefore, we conclude that the proposed change has no impact on the ATWS evaluation.

2.7 SUMMARY

In summary, the staff has examined the impacts of the proposed TS changes on (1) LOCA-related containment loads, (2) safety-relief valve (SRV) operational loads, (3) ECCS performance calculations, (4) NPSH for safety system pumps, (5) Emergency Operating Procedures, and (6) ATWS evaluation, and has concluded that the proposed changes are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The 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. 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 these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that these amendments involve no significant hazards consideration which was published in the Federal Register on November 2, 1988 (53 FR 44251), and consulted with the state of Georgia. No public comments were received, and the state of Georgia did not have any comments.

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

Principal Contributors: Raj K. Anand, SPLB, DEST, NRR
George Thomas, SRXB, DEST, NRR
Lawrence P. Crocker, PD II-3, DRP I/II, NRR

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