October 4, 1988

Docket No. 50-424

Mr. W. G. Hairston, III
Senior Vice President -
Nuclear Operations
Georgia Power Company
P.O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Hairston:

SUBJECT: ISSUANCE OF AMENDMENT NO. 11 TO FACILITY OPERATING LICENSE NPF-68 VOGTLE ELECTRIC GENERATING PLANT, UNIT 1 (TAC 68164)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 11 to Facility Operating License No. NPF-68 for the Vogtle Electric Generating Plant, Unit 1. The amendment is being issued in response to your letter dated May 19, 1988, as supplemented August 12 and October 3, 1988.

The amendment modified the Technical Specifications to allow a slightly positive moderator temperature coefficient and revised shutdown margin requirements. The amendment is effective as of its date of issuance.

A copy of the related safety evaluation supporting Amendment No. 11 to Facility Operating License NPF-68 is enclosed.

Notice of issuance of the amendment will be included in the Commission's next bi-weekly Federal Register notice.

Sincerely,

/S/

Jon B. Hopkins, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II

Enclosures:
1. Amendment No. 11 to NPF-68
2. Safety Evaluation

cc w/enclosures:
See next page

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MRood	JHopkins:	DMatthews
09/30/88	10/4/88	10/4/88

*See Previous concurrence

DF01
'11 C/P1

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Georgia Power Company

Vogtle Electric Generating Plant

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DATED: October 4, 1988

AMENDMENT NO. 11 TO FACILITY OPERATING LICENSE NPF-68 - Vogtle Electric Generating Plant, Unit 1

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 11
License No. NPF-68

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 1 (the facility) Facility Operating License No. NPF-68 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 May 19, 1988, as supplemented August 12 and October 3, 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, as amended, 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 attachments to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-68 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 11, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. GPC 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 within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By:

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects

Attachment:
Technical Specification Changes

Date of Issuance: October 4, 1988

Footnote to TS agreed to by J. Scinto OGC 10/4/88

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LA:PDII-3	PM:PDII-3	OGC-WF	D:PDII-3
MRood	JHopkins:		DBMatthews
9/19/88	9/19/88	9/26/88	10/4/88

ATTACHMENT TO LICENSE AMENDMENT NO. 11

FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

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. The corresponding overleaf pages are also provided to maintain document completeness.

Amended Page	Overleaf Page
3/4 1-3a	-
3/4 1-3b	-
3/4 1-4	-
3/4 1-11	-
3/4 1-12	-
3/4 3-13	3/4 3-14
3/4 5-1	3/4 5-2
3/4 5-10	3/4 5-9
B 3/4 1-3	B 3/4 1-4
B 3/4 3-3	B 3/4 3-4
B 3/4 5-1	-
B 3/4 5-2	-
B 3/4 6-3	
B 3/4 9-1	B 3/4 9-2

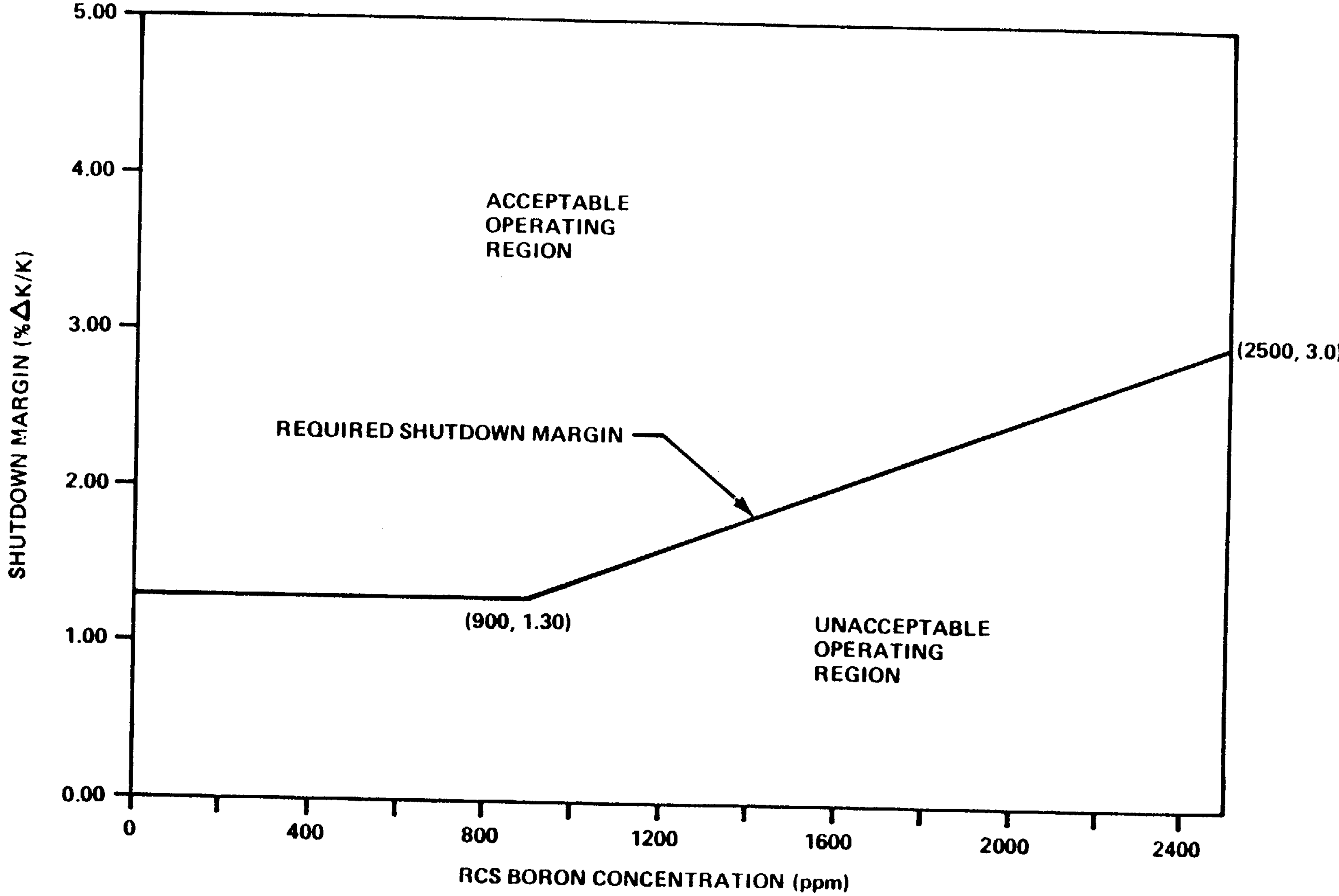

FIGURE 3.1-1 REQUIRED SHUTDOWN MARGIN FOR MODES 3 AND 4 (MODE 4 WITH AT LEAST ONE REACTOR COOLANT PUMP RUNNING)

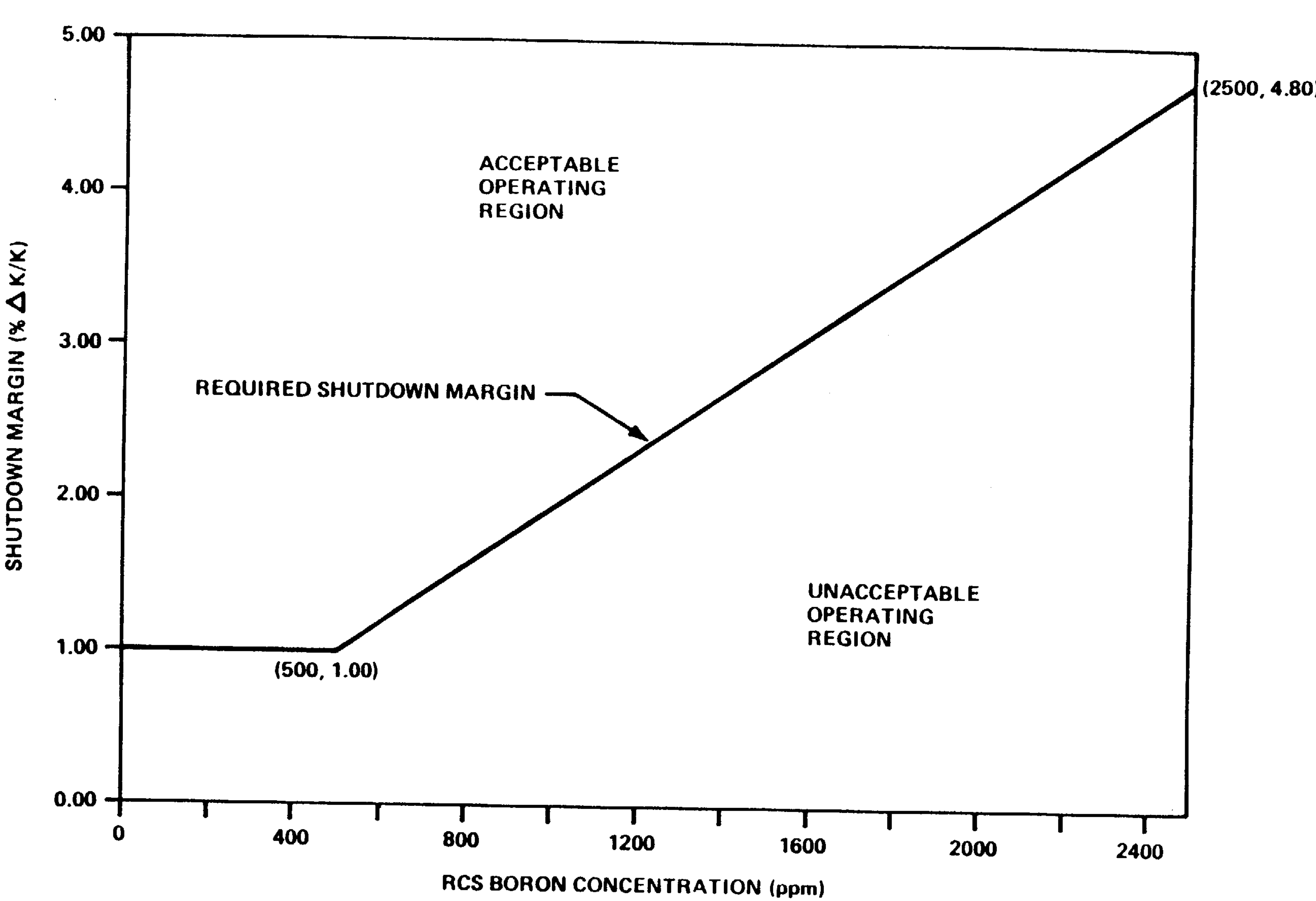

FIGURE 3.1-2 REQUIRED SHUTDOWN MARGIN FOR MODE 5 (MODE 4 WITH NO RCPs RUNNING)

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than + 0.7 x 10^{-4} $\Delta k/k/^{\circ}F$ for the all rods withdrawn, beginning of core life (BOL), condition for power levels up to 70% RATED THERMAL POWER with a linear ramp to 0 $\Delta k/k/^{\circ}F$ at 100% RATED THERMAL POWER; and

b. Less negative than - 4.0 x 10^{-4} $\Delta k/k/^{\circ}F$ for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2* only**.
Specification 3.1.1.3b. - MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a. above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to within the above limits within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.8.2, within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.

**See Special Test Exceptions Specification 3.10.3.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage Tank with:

1) A minimum contained borated water volume of 9504 gallons (19% of instrument span) (LI-102A, LI-104A),

2) A boron concentration between 7000 ppm and 7700 ppm, and

3) A minimum solution temperature of 65°F (TI-0103).

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 99404 gallons (9% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A),

2) A boron concentration between 2400 ppm and 2600 ppm, and

3) A minimum solution temperature of 54°F (TI-10982).

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) When the boric acid storage tank is the source of borated water and the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20903) is $\leq$72°F, verify the boric acid storage tank solution temperature is $\geq$ 65°F.

b. At least once per 24 hours by verifying the RWST temperature (TI-10982) when it is the source of borated water and the outside air temperature is less than 50°F.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A Boric Acid Storage Tank with:

1) A minimum contained borated water volume of 36674 gallons (81% of instrument span) (LI-102A, LI-104A),

2) A boron concentration between 7000 ppm and 7700 ppm, and

3) A minimum solution temperature of 65°F (TI-0103).

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 631478 gallons (86% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A),

2) A boron concentration between 2400* ppm and 2600 ppm,

3) A minimum solution temperature of 54°F, and

4) A maximum solution temperature of 116°F (TI-10982).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the Boric Acid Storage Tank inoperable and being used as one of the above required borated water sources, restore the tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN as required by Figure 3.1-2 at 200°F; restore the Boric Acid Storage Tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* Until concentration is initially raised to 2400 ppm from the maximum limit authorized prior to Amendment No. 11, the minimum boron concentration limit is 2000 ppm.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

a When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

b Above P-9 (Reactor Trip on Turbine Trip Interlock) Setpoint.

c Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

d Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

e Above P-7 (Low Power Reactor Trip Block) Setpoint.

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable to entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. For the purpose of this surveillance requirement, monthly shall mean at least once per 31 EFPD.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) Detector plateau curves shall be obtained, and evaluated. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. This is the determination of the response of the excore power range detectors to the incore measured axial power distribution to generate setpoints for the CHANNEL CALIBRATION. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. For the purpose of this surveillance requirement, quarterly shall mean at least once per 92 EFPD.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) Not used

(9) Quarterly surveillance in MODES 3[a], 4[a], and 5[a] shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive window. Quarterly surveillance shall include verification of the Source Range High Flux at Shutdown Alarm Setpoint of less than or equal to 2.30 times background.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the OPERABILITY of the Undervoltage and Shunt trip of the Reactor Trip Breaker.

(12) CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

(13) Not used

(14) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

(15) Local manual shunt trip prior to placing breaker in service.

(16) Automatic undervoltage trip.

(17) Each channel shall be tested at least every 92 days on a STAGGERED TEST BASIS.

(18) The surveillance frequency and/or MODES specified for these channels in Table 4.3-2 are more restrictive and, therefore, applicable.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 6616 (36% of instrument span) and 6854 (64% of instrument span) gallons (LI-0950, LI-0951, LI-0952, LI-0953, LI-0954, LI-0955, LI-0956, LI-0957),

c. A boron concentration of between 1900 and 2600 ppm, and

d. A nitrogen cover-pressure of between 617 and 678 psig. (PI-0960A&B, PI-0961A&B, PI-0962A&B, PI-0963A&B, PI-0964A&B, PI-0965A&B, PI-0966A&B, PI-0967A&B)

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open. (HV-8808A, B, C, D)

---

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume (67 gallons) by verifying the boron concentration of the accumulator solution; and

c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that the circuit breaker supplying power to the isolation valve operator is open.

4.5.1.2 Each accumulator water level and pressure channel shall be demonstrated OPERABLE at least once per 18 months by the performance of a CHANNEL CALIBRATION.

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

SAFETY INJECTION PUMPS

LIMITING CONDITION FOR OPERATION

3.5.3.2 All Safety Injection pumps shall be inoperable.

APPLICABILITY MODES 4, 5, and 6 with the reactor vessel head on.

ACTION:

With a Safety Injection pump OPERABLE, restore all Safety Injection pumps to an inoperable status within 4 hours.

SURVEILLANCE REQUIREMENTS

4.5.3.2 All Safety Injection pumps shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, and at least once per 31 days thereafter.

* An inoperable pump may be energized for testing or for filling accumulators provided the discharge at the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

BORON INJECTION SYSTEM

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 631,478 gallons (86% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A).

b. A boron concentration of between 2400*ppm and 2600 ppm of boron,

c. A minimum solution temperature of 54°F, and

d. A maximum solution temperature of 116°F (TI-10982).

e. RWST Sludge Mixing Pump Isolation valves capable of closing on RWST low-level.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the RWST inoperable except for the Sludge Mixing Pump Isolation Valves, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a Sludge Mixing Pump Isolation Valve(s) inoperable, restore the valve(s) to OPERABLE status within 24 hours or isolate the sludge mixing system by either closing the manual isolation valves or deenergizing the OPERABLE solenoid pilot valve within 6 hours and maintain closed..

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 50°F.

c. At least once per 18 months by verifying that the sludge mixing pump isolation valves automatically close upon an RWST low-level test signal.

*Until concentration is initially raised to 2400 ppm from the maximum limit authorized prior to Amendment No. 11, the minimum boron concentration limit is 2000 ppm.

BORATION SYSTEMS (Continued)

MARGIN from expected operating conditions as defined by Specification 3/4.1.1.1 (MODES 1 and 2) and Specification 3/4.1.1.2 (MODES 3 and 4) after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 31740 gallons usable volume of 7000 ppm borated water from the boric acid storage tanks or 178182 gallons usable volume of 2400 ppm borated water from the refueling water storage tank (RWST).

With the RCS temperature below 200°F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The boration capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN as defined by Specification 3/4.1.1.2 (MODE 5) after xenon decay and cooldown from 200°F to 140°F. This condition requires either 4570 gallons usable volume of 7000 ppm borated water from the boric acid storage tanks or 41202 gallons usable volume of 2400 ppm borated water from the RWST.

The contained water volume limits provided in Specifications 3/4.1.2.5 and 3/4.1.2.6 include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.0 and 10.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section are necessary to ensure that the following requirements are met at all times during normal operation. By observing that the RCCAs are positioned above their respective insertion limits during normal operation,

1. At any time in life for MODE 1 and 2 operation, the minimum SHUTDOWN MARGIN will be maintained. For operational MODES 3, 4, 5, and 6, the reactivity condition consistent with other specifications will be maintained with all RCCAs fully inserted by observing that the boron concentration is always greater than an appropriate minimum value.

2. During normal operation the enthalpy rise hot channel factor, $F_{\Delta H}$, will be maintained within acceptable limits.

MOVABLE CONTROL ASSEMBLIES (Continued)

3. The consequences of an ejected RCCA accident will be restricted below the limiting consequences referred to in the ejected rod analysis.

4. The core can be made subcritical by the required SHUTDOWN MARGIN with one RCCA stuck. In the event of an RCCA ejection, the core can be made subcritical with two RCCAs stuck, where one of the RCCAs is assumed to be the worst ejected rod control assembly.

5. The trip reactivity assumed in the accident analysis will be available.

6. Dropping an RCCA into the core or statically misaligning an RCCA during normal operation will not violate the thermal design basis with respect to DNBR.

7. The uncontrolled withdrawal of an RCCA will result in consequences no more severe than presented in the accident analysis.

8. The uncontrolled withdrawal of a control assembly bank will not result in a peak power density that exceeds the center line melting criterion.

OPERABILITY of the control rod position indicator channels (LCO 3.1.3.2) is required to determine control rod positions and thereby ensure compliance with the control rod alignment.

OPERABILITY of the Demand Position Indication System (LCO 3.1.3.2) is required to determine bank demand positions and thereby ensure compliance with the insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that some of the original criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER, either of these restrictions provide assurance of fuel rod integrity during continued operation provided no further abnormal condition develops.

For Specification 3.1.3.1 ACTIONS b and c it is incumbent upon the plant to verify the trippability of the inoperable control rod(s). This may be by verification of a control system failure, usually electrical in nature, or that the failure is associated with the control rod stepping mechanism. In the event the plant is unable to verify the rod(s) trippability, it must be assumed to be untrippable and thus fall under the requirements of ACTION a. Assuming a controlled shutdown from 100% RATED THERMAL POWER, this allows approximately four hours for this verification.

The maximum rod drop time permitted by (LCO 3.1.3.4) is consistent with the assumed rod drop time used in the accident analyses. Measurement with T_{avg}

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

(7) steam line isolation, (8) turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment fan coolers start and automatic valves position, (11) Nuclear Service Cooling and Component Cooling water pumps start and automatic valves position, and (12) Control Room Ventilation Emergency Actuation Systems start.

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 With pressurizer pressure below the P-11 setpoint, allows manual block of safety injection actuation on low pressurizer pressure signal. Allows manual block of safety injection actuation and steam line isolation on low compensated steam line pressure signal and allows steam line isolation on high steam line negative pressure rate. With pressurizer pressure above the P-11 setpoint, defeats manual block of safety injection actuation on low pressurizer pressure and safety injection and steam line isolation on low steam line pressure and defeats steam line isolation on high steam line negative pressure rate.

P-14 On increasing steam generator water level, P-14 automatically trips all feedwater isolation valves, initiates a turbine trip, and inhibits feedwater control valve modulation.

The Source Range High Flux at Shutdown Alarm Setpoint is an analysis assumption for mitigation of a Boron Dilution Event in MODES 3, 4, and 5.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range channel is inoperable.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit safe shutdown of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

The OPERABILITY of the Remote Shutdown System ensures that a fire will not preclude achieving safe shutdown. The Remote Shutdown System instrumentation,

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met. The minimum boron concentration must ensure that the reactor core will remain subcritical during the accumulator injection period of a small break LOCA.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

ECCS SUBSYSTEMS (Continued)

The limitation for all safety injection pumps to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses and (4) to ensure that centrifugal charging pump injection flow which is directed through the seal injection path is less than or equal to the amount assumed in the safety analysis.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident, or a steam line rupture.

The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, 2) the reactor will remain subcritical in the cold condition following a small LOCA or steamline break, assuming complete mixing of the RWST, RCS, and ECCS water volumes with all control rods inserted except the most reactive control assembly (ARI-1), and 3) the reactor will remain subcritical in the cold condition following a large break LOCA (break flow $\geq$ 3.0 FT^2) assuming complete mixing of the RWST, RCS, ECCS water and other sources of water that may eventually reside in the sump, post-LOCA with all control rods assumed to be out.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.0 and 10.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

CONTAINMENT VENTILATION SYSTEM (Continued)

The use of the containment purge lines is restricted to the 14-inch purge supply and exhaust isolation valves since, unlike the 24-inch valves, the 14-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline of 10 CFR Part 100 would not be exceeded in the event of an accident during containment PURGING operation. Only safety-related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, should be used to justify the opening of these isolation valves.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or steam line break. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Cooling System both provide post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 8.0 and 10.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The solution volume limits (3700-4000 gallons) represent the required solution to be delivered (i.e., the delivered solution volume is that volume above the tank discharge). These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portions of the Reactor Coolant System. This action prevents flow to the RCS of unborated water by closing flowpaths from sources of unborated water. These limitations are consistent with the initial conditions assumed for the Boron Dilution Accident in the safety analysis. The boron concentration value of 2000 ppm or greater ensures a K_{eff} of 0.95 or less and includes a conservative allowance for calculational uncertainty of 100 ppm boron.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements of the refueling machine and auxiliary hoist ensure that:

(1) The refueling machine will be used for the movement of fuel assemblies and/or rod control cluster assemblies (RCCA) or thimble plug assemblies, and the auxiliary hoist will be used for the movement of control rod drive shafts,

(2) the refueling machine will have sufficient load capacity to lift a fuel assembly and/or a rod control cluster assembly or thimble plug assembly, and the auxiliary hoist will have sufficient load capacity to lift a control rod drive shaft and attached RCCA, and

(3) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE AREAS

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) train be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR trains OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR train will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR train, adequate time is provided to initiate emergency procedures to cool the core.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 11 TO FACILITY OPERATING LICENSE NPF-68

GEORGIA POWER COMPANY, ET AL.

DOCKET NO. 50-424

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

1.0 INTRODUCTION

By letter dated May 19, 1988 (Ref. 1), Georgia Power Company (the licensee) made application to amend the Technical Specifications of Vogtle Electric Generating Plant, Unit 1 (Vogtle 1). The proposed changes would modify the Technical Specifications (TS) concerning (1) the moderator temperature coefficient (MTC) and (2) shutdown margin (SDM) requirements. The MTC change would allow a slightly positive MTC below 100 percent of full rated power. The principal benefit of this change is that it would facilitate the design of future reload fuel cycles. TS changes are required to meet SDM requirements to accommodate the positive MTC and future 18 month reload fuel cycles. To assure that subcriticality requirements are met following a postulated loss-of-coolant accident (LOCA), the boron concentration is increased for the refueling water storage tank (RWST) and the accumulators. An increase in the RWST water volume requirement for Modes 5 and 6 is also proposed. To meet subcriticality requirements for boron dilution events, the SDM limits for Modes 3, 4, and 5 and the high flux at shutdown alarm setpoint are changed.

The applicable safety analysis for Vogtle 1, which is in its initial cycle, is that documented in the Final Safety Analysis Report (FSAR). This safety analysis is based on a 0 pcm/deg F MTC at all times when the reactor is critical. (Note that a pcm is equal to a reactivity of 10^{-5} delta k/k.) The proposed change to the TS would allow a + 7 pcm/deg F MTC below 70 percent power, with a linear variation in the MTC of + 7 pcm/deg F at 70 percent power to 0 pcm/deg F at 100 percent power. The licensee has reevaluated the FSAR safety analysis using this positive MTC as well as an increase in the boron concentration for the RWST to a concentration range of 2400-2600 ppm and an increase in the boron concentration for the accumulators to a concentration range of 1900-2600 ppm. This reevaluation is provided in Enclosure 1 (Ref. 2) of Reference 1.

Additional information was submitted by letter dated August 12, 1988 (Ref. 14). Included in this submittal was a change to the TS bases Section 3/4.9.1, 'Boron Concentration' which clarified the calculational uncertainty for reactivity conditions during refueling. This small clarifying change to the bases did not substantially affect the amendment request as noticed or the staff's initial determination; therefore, the amendment was not renoticed.

Additional information was submitted by letter dated October 3, 1988 (Ref. 15) and by telecon of October 4, 1988 (Ref. 16). This information described the licensee's intended implementation of this amendment. To implement this amendment, a footnote is being added to Technical Specification 3.5.4, "Refueling Water Storage Tank" and Technical Specification 3.1.2.6, "Borated Water Sources-Operating" that states, "Until concentration is initially raised to 2400 ppm from the maximum limit authorized prior to Amendment No. 11, the minimum boron concentration limit is 2000 ppm." This footnote prevents the licensee from being in immediate noncompliance when this Technical Specification amendment is implemented.

This footnote regarding implementation does not substantially affect the amendment request as noticed or the staff's initial determination; therefore, the amendment was not renoticed.

The NRC staff has reviewed the proposed amendment for Vogtle 1 and its evaluation follows.

2.0 EVALUATION

2.1 Effect of Positive MTC on Transient and Accident Analyses

The licensee has evaluated the effect of the proposed change to the MTC TS on the transients and accidents previously analyzed in the FSAR. Events which were sensitive to a positive or near zero MTC were reanalyzed. These events were the transients which cause the reactor coolant temperature to increase (or equivalently the density to decrease). The events reanalyzed used the identical analysis methods, computer codes, and assumptions used for the FSAR analysis. Any differences in the present analysis to the previous FSAR analysis will be noted in the following discussion.

A number of transients were not reanalyzed for the proposed change in the TS to a positive MTC. These transients are listed below, along with the applicable FSAR Sections, and the reason given by the licensee for not requiring a reanalysis.

Transient	FSAR Section	Reason for not Reanalyzing
1. Feedwater System Malfunctions		
(a) Decrease in Feedwater Temperature	15.1.1	Limiting case with negative MTC
(b) Increase in Feedwater Flow	15.1.2	Limiting case with negative MTC
2. Increase in Steam Flow	15.1.3	Analysis assumes large negative MTC, minimum DNBR not sensitive to MTC

Transient	FSAR Section	Reason for not Reanalyzing
3. (a) Inadvertent Opening of a Steam Generator Relief or Safety Valve	15.1.4	Limiting case with negative MTC
(b) Steam System Piping Failure	15.1.5	Limiting case with negative MTC
(c) Steamline break mass/energy release inside containment	6.2.1.4	Limiting case with negative MTC
4. Feedwater System Pipe Break	15.2.8	Analyzed with negative MTC, not sensitive to positive MTC since reactor trip occurs near beginning of transient before reactor coolant temperature increases
5. Control Rod Misoperation	15.4.3	Misalignment cases are not affected by a positive MTC, single rod withdrawal analyzed at steady-state conditions until the reactor trips on the overtemperature delta-T signal and is not dependent on a positive MTC, dropped rod limiting analysis unaffected since MTC must be close to zero or negative at 100% power.
6. Startup of an Inactive Loop at an Incorrect Temperature	15.4.4	Limiting case with negative MTC
7. Inadvertent operation of the ECCS During Power Operation	15.5.1	A positive MTC is less limiting than FSAR analysis. Positive MTC causes power to decrease more rapidly and increases margin to DNB.
8. (a) Small Break LOCA (b) Large Break LOCA (c) LOCA forces (d) Containment Integrity Analysis	6.2 15.6.5	Positive MTC has negligible impact on small break LOCA; during large break LOCA a positive MTC will not be significant due to core voiding during blowdown period; peak LOCA forces occur before any impact from positive MTC occurs;

Transient	FSAR Section	Reason for not Reanalyzing
		containment integrity analysis is performed at 100% power where the analysis is based on a zero MTC; the steamline break mass and energy analysis is based on a negative MTC.
9. Steam Generator Tube Failure or Rupture (SGTR)	15.6.3	Effect of a positive MTC was determined using a LOFTRAN result; minimum DNBR remains above limits; increase in reactor power reduces reactor trip time and increases primary to secondary leakage but doses remain well below NRC limits of a small fraction of 10 CFR 100.
	WCAP-11731 (Ref. 3)	Revised SGTR analysis was performed with Vogtle positive MTC and boron requirements with acceptable results.
10. Fuel Misloading Error	15.4.7	Analysis is performed with steady-state methods and is not affected by a positive MTC.

The NRC staff has reviewed the evaluation presented by the licensee for each of the transients listed above and concurs with the licensee's assessment that these transients do not require a reanalysis.

The transient analyses for those events that were reanalyzed used the same computer codes that were used for the Vogtle 1 FSAR analyses. These codes are LOFTRAN (Ref. 4), TWINKLE (Ref. 5), FACTRAN (Ref. 6), and THINC (Refs. 7 and 8). The initial conditions, instrument errors, and setpoint errors are essentially the same as those found in Chapter 15 of the Vogtle 1 FSAR. The allowance on pressurizer pressure given in FSAR Section 15.0.3.2 has been changed from ± 30 psi to a more conservative ± 45 psi. Also, the pressurizer and steam generator water levels uncertainties have been increased from 5% to 6.6% to bound calculated increases in associated transmitter uncertainties.

The LOFTRAN code was used to reanalyze all the events affected by a positive MTC except for the rod ejection accident and rod withdrawal from a subcritical condition which were analyzed using the TWINKLE code. The analyses with the LOFTRAN code conservatively used a + 7 pcm/deg F MTC over the entire power range except for the locked rotor accident. The MTC was conservatively assumed to remain constant for variations in temperature for all transients. The analyses with the TWINKLE code used at least a + 7 pcm/deg F MTC at zero power nominal average temperature conditions. This coefficient becomes less positive at higher temperatures since the TWINKLE formulation does not allow the moderator temperature feedback to artificially be held constant.

The evaluation of each transient that was reanalyzed for a positive MTC is discussed below.

<u>Uncontrolled Rod Bank Withdrawal From a Subcritical or Low-Power Startup Condition (FSAR Section 15.4.1)</u>

The uncontrolled rod bank withdrawal from a subcritical condition transient leads to a power excursion. This power excursion is terminated, after a fast power rise, by the negative Doppler reactivity coefficient of the fuel, and a reactor trip on source, intermediate or power range flux, or high positive neutron flux rate. The power excursion results in a heatup of the moderator/coolant and the fuel. The positive MTC causes an increase in the rate of reactivity addition, resulting in an increase in peak heat flux and peak fuel and clad temperature. The analysis used the same reactivity insertion rate of 60 pcm/sec as the Vogtle 1 FSAR. This reactivity insertion rate is greater than two sequential control banks withdrawing at the maximum speed of 45 inches/minute. The neutron flux overshoots the nominal full power value; however, the peak heat flux is much less than the full power nominal value because of the inherent thermal lag in the fuel. The minimum Departure from Nucleate Boiling Ratio (DNBR) remains above the limiting value at all times throughout the transient. Therefore, the conclusions of the FSAR remain valid.

<u>Uncontrolled Rod Bank Withdrawal at Power (FSAR Section 15.4.2)</u>

The uncontrolled rod bank withdrawal from a power condition transient leads to a power increase. The transient results in an increase in the core heat flux and an increase in the reactor moderator/coolant temperature. This transient could result in departure from nucleate boiling (DNB) unless terminated by manual or automatic action. The Power Range High Neutron Flux and Overtemperature Delta-T reactor trips are assumed in the analysis to provide protection against DNB. The positive MTC causes an increase in the rate of reactivity addition. The minimum reactivity feedback cases presented in the FSAR were reanalyzed with the positive MTC. The maximum reactivity feedback cases were not reanalyzed because these assumed a large negative MTC. The licensee presented results for the minimum reactivity feedback case for power levels of 10%, 60%, and 100% power for a range of reactivity insertion rates.

The results indicate that the departure from nucleate boiling ratio (DNBR) limit is met for all the cases that were analyzed and the conclusions of the FSAR remain valid.

Loss of Forced Reactor Coolant Flow (FSAR Sections 15.3.1 and 15.3.2)

The loss of flow transient causes the reactor power to increase until the reactor trips on a low flow trip signal or a reactor coolant pump power supply undervoltage signal. The reactor power increase causes a reactor moderator/coolant temperature increase. This initial coolant temperature increase causes a positive reactivity insertion because of the positive MTC. The licensee reanalyzed both a partial loss of flow (loss of 2 pumps with four coolant loops in operation) transient and a complete loss of flow transient. For the partial loss of flow transient, DNB does not occur. The average fuel and clad temperatures do not increase significantly above their initial values because the primary coolant maintains its ability to remove heat from the fuel. For a partial loss of flow the reactor will stabilize at a condition of power, flow rate, and pressure from which a normal plant shutdown may proceed. For the complete loss of flow transient DNB does not occur. The average fuel and clad temperatures do not increase significantly above their initial values because the primary coolant maintains its ability to remove heat from the fuel. For this complete loss of flow, the flow will coastdown until natural circulation flow is established and normal plant shutdown may proceed. The results indicate that the DNBR limit is met for the partial and complete loss of flow events and the conclusions of the FSAR remain valid.

Reactor Coolant Pump Shaft Seizure (FSAR Section 15.3.3)

The FSAR analysis for the reactor coolant pump seizure event assumed DNB to occur at the beginning of the event. Consequently, the positive MTC will not affect the time to DNB. Existing sensitivity studies, which were used for the FSAR analysis, show that a zero MTC at full power is more limiting than a positive MTC at lower power. The sensitivity results covered the proposed positive MTC for the Vogtle 1 plant. The amount of fuel in DNB that is assumed to fail is, therefore, the same as that previously assumed for the FSAR analysis and the FSAR radiological consequences evaluation remains valid. The reactor coolant pump seizure event was reanalyzed with a positive MTC to evaluate the effect of the power transient on peak reactor coolant system pressure and clad temperatures. This reanalysis was performed both with and without offsite power. For both of these cases, reactor trip occurs when the low flow trip setpoint is reached. The peak reactor coolant system pressure reached during both cases is less than that which would cause stresses to exceed the faulted condition stress limits. The peak clad surface temperature for both cases is about 1750 deg F so that the amount of zirconium-water reaction is small. These two cases demonstrate that the conclusions of the FSAR remain valid with respect to peak pressure and clad temperature for the reactor coolant pump shaft seizure event.

Turbine Trip Events (FSAR Sections 15.2.3, 15.2.4, and 15.2.5)

A turbine trip event is more limiting than other events which lead to a turbine trip. The minimum reactivity feedback cases were analyzed for the positive MTC. These cases occur at beginning-of-cycle (BOC). The maximum reactivity feedback cases were not reanalyzed because they occur near end-of-cycle (EOC) and assume a negative MTC. For one case, full credit is taken in the analysis for the pressurizer spray and power operated relief valves (PORV). For the other case, no credit is taken in the analysis for the operation of the pressurizer spray or PORVs. Both cases assume the availability of the pressurizer safety valves. For both pressure control cases, the reactor trips on reaching the High Pressurizer Pressure Trip setpoint. The results show that the primary system pressure remains below the 110% design value and that DNBR remains well above its limit. This transient remains the limiting Condition 2 transient with respect to peak pressure. These two cases demonstrate that the conclusions of the FSAR remain valid with respect to peak pressure and DNBR for the turbine trip event. In addition, the conclusions of the Overpressure Protection Report remain valid because the system pressure remains below 110% of the design value.

Rod Ejection Accident (FSAR Section 15.4.8)

The rod ejection accident is analyzed at full power and hot standby conditions for both BOC and EOC in the FSAR. The EOC cases are those with large negative MTCs. Therefore, the reanalysis was performed for the positive MTC for BOC conditions. The reanalysis used ejected rod worths and transient peaking factors that are conservative with respect to the actual values for the current fuel cycle. In the analysis, reactor trip occurred when the power range high neutron flux setpoint was reached. The peak hot spot clad average temperature was reached in the hot zero power case. The peak hot spot value of 2490 deg F was below the limit specified in the FSAR. The maximum fuel temperature and enthalpy occurred for the hot full power case. For both cases, the peak fuel enthalpy was well below the staff criterion of 280 cal/gm. For the hot full power case, the peak fuel centerline temperature at the hot spot exceeded the melting temperature, but the extent of melting was less than the innermost 10% of the fuel pellet. Because fuel and clad temperatures and the fuel enthalpy do not exceed the limits in the FSAR, the conclusions of the FSAR remain valid.

Loss of Normal Feedwater Flow/Loss of Nonemergency AC Power to the Plant Auxiliaries (FSAR Sections 15.2.7 and 15.2.6)

FSAR Section 15.2.7 presents the analysis of a loss of normal feedwater with offsite power available. FSAR Section 15.2.7 presents the analysis of a loss of normal feedwater which assumes offsite power is lost. This event was reanalyzed for BOC conditions for the positive MTC. The reanalysis used a conservative core decay heat model based on the ANSI/ANS-5.1-1979 decay heat standard (Ref. 9). The pressurizer pressure control system, sprays and PORVs were assumed to be available since a lower pressure results in a greater system expansion. For the cases with and without offsite power, reactor trip occurred when the low-low steam generator water level trip setpoint was reached. For the case with the loss of offsite power assumption, power was assumed lost to the reactor coolant pumps following control rod motion. The reanalysis showed that a loss of normal feedwater does not adversely affect the reactor core, the reactor coolant system, the steam system, and that the auxiliary feedwater system is sufficient to prevent water relief through the pressurizer relief or safety valves. For the case without offsite power available, the natural circulation capability of the reactor coolant system is sufficient to remove decay heat following a reactor coolant pump coastdown to prevent fuel or clad damage. For both cases, the pressurizer does not fill and, therefore, the conclusions of the FSAR remain valid.

Inadvertent Opening of a Pressurizer Safety or Relief Valve (FSAR Section 15.6.1)

The inadvertent opening of a pressurizer safety or relief valve event causes a moderator/coolant density reduction. Because a positive MTC can be considered to be a negative density coefficient, that is, equivalent to a temperature increase, the density reduction due to a reactor coolant system depressurization causes a positive reactivity insertion. For this event, the nuclear power increases until a reactor trip occurs when the overtemperature delta-T trip setpoint is reached. The analysis is performed with the control rods in the manual mode so that control rod insertion does not compensate for the reactivity insertion caused by the coolant density reduction. The results indicate that the DNBR remains above the limit of 1.30 throughout the transient and, therefore, the conclusions of the FSAR remain valid.

Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (FSAR Sections 15.5.2 and 15.4.6)

The boron dilution event for Mode 1 assumes manual rod control and is dependent upon the results of the Uncontrolled Rod Bank Withdrawal at Power Analysis. A reactivity insertion rate is calculated for this Mode 1 boron dilution event and compared to those analyzed for the rod withdrawal at power event. This analysis shows that a minimum of 16.9 minutes are available for the operator to terminate the event from the time of reactor trip which is the first indication of the event. The analysis for Modes 3, 4, and 5 was performed in the same manner as the FSAR analysis. The high flux at shutdown alarm setpoint was revised to 2.3 times background, and the makeup flow control valve setpoint was changed to 100 gpm. The change to a 100 gpm setpoint allows a lower dilution flow of 110 gpm for the analysis compared to the previous analysis. The analysis results in curves of required shutdown margin as a function of reactor coolant system boron concentration. Meeting the shutdown margin requirements will provide at least 15 minutes for operator action from the high flux at shutdown alarm to the time when shutdown margin is lost. These shutdown margin curves include the increased reload boron concentration and will be placed in the Technical Specifications as Figures 3.1-1 and 3.1-2. For Mode 6, administrative procedures require that certain valves be locked closed to prevent a boron dilution event. The Mode 2 boron dilution event was not reanalyzed because it is not affected by a positive MTC. The results provided by the licensee demonstrate that sufficient time is available to the operator to terminate a boron dilution event in the reanalyzed modes and, therefore, the conclusions of the FSAR remain valid.

In summary, the NRC staff has reviewed each of the reanalyzed events discussed above and concludes that the licensee's assessments and results obtained are acceptable.

In meetings with the staff on the ATWS MTC, Westinghouse and the Westinghouse Owners Group have presented data for their plants which show that there has been no adverse trend in the ATWS MTC. The data include data from plants with extended fuel cycles and positive MTCs similar to the planned future operation of Vogtle 1. Because ATWS is considered for a class of plants and the Vogtle 1 class of plants have ATWS MTCs which are not more positive than the previous ATWS analysis basis MTC, the staff concludes that ATWS need not be considered in this review of a change to a more positive MTC for Vogtle 1.

2.2 RWST and Accumulator Boron Increase

The implementation of a positive MTC and extending operating cycles to 18 months require changes to the boron concentration of the refueling water storage tank (RWST) and the accumulators. The licensee states that these changes to the boron concentration are required to meet the long term core cooling requirements of 10 CFR 50.46. The maximum boron concentration of the RWST and accumulators has been increased to 2600 ppm. The minimum boron concentration of the RWST has been increased to 2400 ppm. The minimum boron concentration of the accumulators remains unchanged at 1900 ppm. The licensee has considered the effect of these boron concentration changes in a number of areas.

Non-LOCA Transient and Accident Analyses which Model the RWST

The various steamline break accidents discussed in FSAR Sections 15.1.4, 15.1.5, and 6.1.2.4 all take credit for the RWST boron injected into the reactor coolant system by the Safety Injection (SI) system. Increasing the minimum RWST boron concentration leads to a less limiting analysis than provided in the FSAR which used a smaller value for the boron concentration. The increased minimum boron concentration will insert more negative reactivity into the core for these events, thus providing less limiting results. The FSAR analysis will remain bounding and the conclusions in the FSAR remain valid.

The feedwater system pipe break (FSAR Section 15.2.8) analysis models the SI system. This transient is not sensitive to the RWST boron concentration. Increasing the RWST boron concentration would, however, give less limiting results. The purpose of the SI is to provide a source of cool water to aid in cooling the primary system and to help ensure that the core remains covered. The FSAR analysis remains bounding and the FSAR conclusions remain valid.

The minimum RWST boron concentration has been considered for boron dilution events (FSAR Section 15.4.6) for Modes 3, 4, and 5 in the reanalysis with a positive MTC discussed above. The shutdown margin limits have been defined for the possible higher reactor coolant system boron concentrations allowed by the increased RWST boron concentrations. New Technical Specification limits for the shutdown margin as a function of reactor coolant system boron are being proposed for Modes 3, 4, and 5.

The inadvertent operation of the ECCS during power operation event (FSAR Section 15.5.1) would result in an increased boron concentration in the SI water. This would cause the nuclear power to decrease at a somewhat faster rate. The core average temperature and pressure would decrease at a somewhat faster rate. The decreasing power and temperature more than offset the decrease in pressure so that the trend of increasing DNBR for this event will not change. The FSAR analysis remains bounding and the FSAR conclusions remain valid.

The only non-LOCA transients which take credit for the accumulator boron concentration are the steamline break event. The minimum boron concentration is modeled for conservatism. There is no change in the steamline break analyses because the minimum accumulator boron concentration has not been changed.

The NRC staff has reviewed the effect of the proposed changes to the RWST and accumulators boron concentrations on affected non-LOCA transients and concludes that the licensee's assessments, as discussed above, are acceptable.

The licensee requested 60 days to implement this amendment. As discussed in references 15 and 16, the licensee intends to commence implementation of this amendment prior to shutdown for refueling in order to assure that coolant is adequately mixed to ensure uniform boron concentration for refueling operations. As a result, RWST boron concentration will be increased above the old TS limit of 2000-2100 ppm and yet will still be below the new TS limit of 2400-2600 ppm for a period of time. Therefore, by telecon of October 4, 1988 (Ref. 16), a footnote was agreed to be added to TS to allow the minimum boron concentration to be 2000 ppm until concentration is initially raised to 2400 ppm.

The increase in RWST boron concentration will be accomplished by utilizing a gravity feed of approximately 100 gpm from the RWST to the spent fuel pool with a return to the RWST via a spent fuel pool cooling pump bypass loop. During this operation, RWST level will be maintained approximately 36,000 gallons above the minimum TS limit; however, if a small break LOCA occurs, operators will secure this operation long before 6 hours pass which is the time that the 36,000 gallons is based on. Also, prior to exceeding 2100 ppm, the procedure for hot leg recirculation switchover will be changed to occur at 11 hours post accident.

A review of LOCA, non-LOCA, and system related transients was performed regarding implementation of the amendment prior to Mode 5. It has been determined that conclusions regarding amendment acceptability reached in previous licensee submittals remain valid. From this, the NRC staff concludes that implementation of this amendment to increase boron concentration in the RWST may commence prior to shutdown and that therefore the footnote allowing the minimum boron concentration to be 2000 ppm is acceptable.

The submittal of October 3, 1988 (Ref. 15) requested discretionary enforcement from the previous TS upper boron concentration limit of 2100 ppm to allow the boron concentration to be raised to 2400 ppm for this amendment. Since this amendment, as footnoted, covers raising boron concentration, discretionary enforcement is not necessary.

LOCA

The small break LOCA analysis (SBLOCA) does not explicitly model core reactivity or account for boron provided by the Emergency Core Cooling System. The analysis assumes that the reactor remains subcritical following control rod insertion. The analysis further assumes that the boron provided by the ECCS will keep the reactor subcritical for the all rods in minus 2 (ARI-2) condition (one rod out for rod ejection caused SBLOCA and one rod out for stuck rod assumption). The additional RWST and accumulator boron concentrations would make the SBLOCA less limiting with regard to core reactivity and, therefore, the FSAR conclusions regarding the SBLOCA analysis remain valid.

The large break LOCA analyses do not take credit for the negative reactivity of the boron in the ECCS water. During a time to just beyond the time of peak clad temperature (PCT), core voiding keeps the reactor subcritical. The FSAR conclusions remain valid regarding the large break LOCA for an increased RWST and accumulator boron concentration.

For the post-LOCA long term cooling (FSAR Section 15.6.5) the increase in the RWST minimum boron concentration gives an increase of about 280 ppm of boron in the post LOCA reactor coolant system/sump boron concentration. The licensee states that this increased boron concentration is enough to offset the effect of a positive MTC.

The licensee states that LOCA hydraulic vessel and loop forces (FSAR Section 3.6) are not affected by the increase in the minimum RWST boron concentration because the maximum loads are generated within the first few seconds of a break initiation. For this reason, the ECCS, including the RWST, is not modeled when considering LOCA forces.

For the FSAR analysis of the steam generator tube rupture (SGTR) event (FSAR Section 15.6.3), sufficient shutdown margin is assumed to be available initially because of control rod insertion following a reactor trip and adequate shutdown margin is assumed to be maintained for the long term by the borated safety injected water. An increased RWST minimum boron concentration will result in more negative reactivity insertion for this event and will, therefore, have no adverse impact on the FSAR analysis. For the revised SGTR analysis in WCAP-11731, operator actions are modeled in the analysis. It is assumed that sufficient shutdown margin will be provided initially by the insertion of control rods on reactor trip and will be maintained during the reactor coolant system cooldown by the borated safety injection water. The increased negative reactivity insertion rate will, therefore, have no adverse impact on the WCAP-11731 SGTR analysis.

For the containment integrity analysis (FSAR Section 6.2), the short term mass and energy subcompartment pressure analyses are not affected by the increase in the RWST boron concentration because, for the short duration of the transient, safety injection flow from the RWST is not considered. The long term mass and energy release and containment response calculations following a LOCA do not take credit for the soluble boron in the safety injected water from the RWST and, therefore, the increased RWST boron concentration will have no effect on these analyses. For secondary system pipe ruptures inside containment, the RWST is modeled in the mass and energy analysis. The increased RWST boron concentration will insert more negative reactivity in the core and result in less limiting mass and energy releases. The licensee states that the conclusions presented in the Vogtle FSAR remain valid.

For the increased concentration of boron in the RWST, the licensee established the time for hot leg switchover from cold leg injection (FSAR Section 6.3.2). This switchover is required to prevent boron precipitation from occurring for a cold leg break. The licensee states that 11 hours from the start of a LOCA, hot leg switchover of RWST borated water injection must be initiated before loss of solubility of boron in the water. This hot leg switchover time covers the complete break spectrum.

The rod ejection accident (FSAR Section 15.4.8) mass and energy releases use similar assumptions in modeling the RWST safety injection flow as the small and large break LOCA analyses. Therefore, the increased RWST boron concentration will have no adverse effect on the FSAR rod ejection accident.

The NRC staff has reviewed the effect of the proposed increases to the RWST and accumulators boron concentrations on LOCA and concludes that the licensee's assessments, as discussed above, are acceptable.

<u>Effect of RWST and Accumulator Boron Concentrations on Fluid Systems and LOCA Radiological Consequences</u>

The licensee evaluated other effects of the increased boron concentrations in the RWST and accumulators. The volume of boric acid solution required in the RWST during Modes 5 and 6 is increased, as well as the boron concentration, to meet the requirements of an 18 month cycle. The volume of boric acid solution in the RWST required to bring the plant to cold shutdown conditions is increased (affects a TS Bases). No changes are required to the Boric Acid

Storage Tank. The Reactor Makeup Control System automatic flow rate is decreased from 120 to 100 gpm so that the makeup control system will be able to blend approximately 35 gpm of 7000 ppm boron solution from the boric acid storage tank with 65 gpm of reactor makeup water to deliver 100 gpm of 2500 ppm liquid to the reactor.

The current licensing basis for Vogtle 1 limits the long term sump pH to between 8.5 and 10.5 and the containment spray pH to less than 11.0. The increased RWST and accumulators boron concentrations reduce the long term sump pH. The licensee established that the minimum sump pH is 8.15. This value of the sump pH is less than the current licensing basis of 8.5. The licensee evaluated the effect of this change in the minimum sump pH. The licensee determined that the increase in the corrosion rate between a pH of 8.0 and 8.5 is not significant for the construction materials in containment and, therefore, the source of post-LOCA hydrogen generation is not increased by a reduction in the minimum sump pH. The licensee states that equipment qualification is not affected by the reduction from a pH of 8.5 to a pH of 8.0 because equipment is qualified at a pH of 10.7 and any move towards a pH of 7.0 would provide a more neutral environment.

For chloride induced stress corrosion cracking of stainless steel, Westinghouse recommends a minimum sump pH of 7.5. The NRC recommends in Reference 10 that the minimum equilibrium sump pH should be between 7.0 and 9.5, with the higher values providing greater assurance that no stress corrosion cracking will occur. The minimum pH of 8.0 is consistent with both the Westinghouse and NRC recommendations.

The licensee evaluated the effect of the reduction in minimum sump pH to 8.0 on LOCA thyroid doses. Reduced conservatisms were assumed for (1) deposition removal of elemental iodine from the containment (Ref. 11), (2) spray removal of particulate iodine from the containment atmosphere (Ref. 11), and (3) rate of unfiltered inleakage into the control room (Refs. 12 and 13). Revised performance for the Control Room Emergency HVAC was also used in the reanalysis. The revised LOCA doses, including effects of the reduced conservatisms and increased RWST boron concentration, meet the acceptance criteria.

The NRC staff has reviewed the effect of the proposed increase in the RWST boron concentration on other fluid systems and LOCA doses and concludes that the licensee's assessments, as discussed above, are acceptable.

2.3 Technical Specifications

TS changes are required to incorporate the changes to a more positive MTC and increased boron concentrations of the RWST and accumulators. All of these TS changes are acceptable per the discussion in the preceding evaluation section. These changes are discussed below.

Specification 3/4.1.1.2

New shutdown margin curves as a function of reactor coolant system boron concentration have been generated for Modes 3, 4, and 5. These new curves, given by Figures 3.1-1 and 3.1-2, are based on a higher reactor coolant system boron concentration, a reduction in the assumed dilution flow rate, and a reduction in the high flux at shutdown alarm setpoint.

Specification and Basis 3/4.1.1.3

The MTC is changed to be less positive than + 7 pcm/deg F for the all rods withdrawn, beginning of life condition for power levels up to 70% of rated thermal power with a linear ramp to 0 pcm/deg F at 100% rated thermal power.

Specifications and Bases 3/4.1.2.5 and 3/4.1.2.6

The RWST minimum and maximum boron concentrations are changed, the minimum is increased from 2000 to 2400 ppm, the maximum is increased from 2100 or 2200 to 2600 ppm. The minimum RWST volume is increased from 70,832 to 99,404 gallons for Modes 5 and 6.

The Bases are changed to require the RWST to have 178,182 gallons of borated water at a boron concentration of 2400 ppm for Modes 1 through 4. For Modes 5 and 6, the Bases are changed to require the RWST to have 41,202 gallons of borated water with a boron concentration of 2400 ppm.

The lower limit pH of the recirculated borated water solution during a LOCA is changed from 8.5 to 8.0 in the Bases.

Specification and Basis 3/4.3.1

The Source Range High Flux at Shutdown Alarm Setpoint in Table 4.3-1, Footnote 9 was changed from 3.16 to 2.3 times background.

A statement was added to the Bases to state that this setpoint is an assumption of the boron dilution event in Modes 3, 4, and 5.

Specification and Basis 3/4.5.1

The maximum boron concentration for the accumulators is increased from 2100 to 2600 ppm.

A statement was added to the Bases to state that the minimum boron concentration of the accumulators is required to maintain the reactor subcritical during the accumulator injection period of a small break LOCA.

Specification and Basis 3/4.5.4

The RWST minimum and maximum boron concentrations are changed, the minimum is increased from 2000 to 2400 ppm, and the maximum is increased from 2100 to 2600 ppm.

The lower limit pH of the recirculated borated water solution during a LOCA is changed from 8.5 to 8.0 in the Bases.

Basis 3/4.6.2

The Bases lower the pH value from 8.5 to 8.0 of the recirculated solution within containment after a LOCA.

Basis 3/4.9.1

The licensee provided a clarification of the change to this basis in Reference 14. This change is acceptable.

2.4 SUMMARY

The NRC staff has reviewed the submittal for the operation of Vogtle 1 with a more positive MTC and with increased boron concentrations for the RWST and accumulators. Based on its review, the NRC staff concludes that appropriate material was submitted and that the transients and accidents that were evaluated and reanalyzed are acceptable. The TS changes submitted for this license amendment suitably reflect the necessary modifications for the operation of Vogtle 1 for extended cycles with a more positive MTC. Therefore, the NRC staff finds that the proposed amendment is acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the use of a facility component located within the restricted area as defined in Part 20 and changes in surveillance requirements. The staff has determined that the amendment involves 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 exposure. The NRC staff has made a determination that the amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendment meets 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 amendment.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register on June 29, 1988 (53 FR 24509), and consulted with the state of Georgia. No public comments were received, and the state of Georgia did not have any comments.

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

5.0 REFERENCES

1. Letter (SL-4682) from R. P. McDonald (GPC) to USNRC, dated May 19, 1988.

2. "Positive Moderator Temperature Coefficient and RWST/Accumulator Boron Concentration Increase Licensing Report for Vogtle Electric Generating Plant Units 1 and 2," Westinghouse Electric Corporation, April 1988 (this report was provided as Enclosure 1 to Reference 1).

3. Letter from C. Rossi (NRC) to A. Ladieu (Westinghouse Owners Group), dated March 30, 1987 (letter approved methodology of WCAP-11731 for performing SGTR analysis).

4. Burnett, T.W.T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984.

5. Risher, D.H., Jr. and Barry, R.F., "TWINKLE - A Multidimensional Neutron Kinetics Computer Code," WCAP-7979-P-A (Proprietary) and WCAP-8028-A (Non-Proprietary), 1975.

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7. Hochreiter, L.E., Chelemer, H.; and Chu, P.T., "Subchannel Thermal Analysis of Rod Bundle Cores," WCAP-7015, Revision 1, 1969.

8. Chelemer, H.; Weisman J.; and Tong L.S., "Subchannel Thermal Analysis of Rod Bundle Cores," WCAP-7015, Revision 1, 1969.

9. "American National Standard for Decay Heat Power in Light Water Reactors," ANSI/ANS-5.1-1979, August 1979.

10. Branch Technical Position MTEB 6-1, "pH for Emergency Coolant Water for PWR's."

11. "Technological Bases for Model of Spray Washout of Airborne Contaminants in Containment Vessels," NUREG/CR-009, October 1978.

12. Georgia Power letter to NRC (Bailey to Youngblood), GN-808, February 19, 1986.

13. "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," Regulatory Guide 1.78, June 1974.

14. Letter from W. G. Hairston, III (GPC) to NRC, dated August 12, 1988.

15. Letter from W. G. Hairston, III (GPC) to NRC, dated October 3, 1988.

16. Telecon between J. Swartzwelder (GPC) and J. Hopkins (NRC) on October 4, 1988.

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