FEBRUARY 19, 1993

Docket Nos. 50-424
and 50-425

Distribution
See next page

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

Dear Mr. Hairston:

SUBJECT: ISSUANCE OF AMENDMENTS - VOGTLE NUCLEAR GENERATING PLANT, UNITS 1 AND 2 (TAC NOS. M77466, M77467, M77391, AND M77392)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 56 to Facility Operating License NPF-68 and Amendment No. 35 to Facility Operating License NPF-81 for the Vogtle Electric Generating Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated November 18, 1991, as supplemented March 2, 1992.

The amendments revise the TS in response to Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," which was issued by the NRC on June 25, 1990.

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

Sincerely,
ORIGINAL SIGNED BY:
Darl S. Hood, Project Manager
Project Directorate II-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.56 to NPF-68
2. Amendment No.35 to NPF-81
3. Safety Evaluation

cc w/enclosures:
See next page

DM 2/10/93

OFC	PDII-3/LA	PDII-3/PM	OGC NLO	PDII-3/D	SRXB	EMEB
NAME	LBerry	DHood/cw DSH	MZOBLER	DMatthews	RJones	JNorberg
DATE	12/30/92	1/6/93	2/17/93	2/19/93	2/4/93	2/10/93

minor, editorial change

added DSH with comments noted

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

February 19, 1993

Docket Nos. 50-424
and 50-425

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

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The amendments revise the TS in response to Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," which was issued by the NRC on June 25, 1990.

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Sincerely,

Darl Hood

Darl S. Hood, Project Manager
Project Directorate II-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. **56** to NPF-68
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3. Safety Evaluation

cc w/enclosures:
See next page

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

Vogtle Electric Generating Plant

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P. O. Box 572
Waynesboro, Georgia 30830

DATED: FEB 19 1993

AMENDMENT NO. 56 TO VOGTLE ELECTRIC GENERATING PLANT, UNIT 1
AMENDMENT NO. 35 TO VOGTLE ELECTRIC GENERATING PLANT, 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

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 56
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 November 18, 1991, as supplemented March 2, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-68 is hereby amended to read as follows:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 56 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

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

Attachment:
Technical Specification
Changes

Date of Issuance: February 19, 1993

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

VOGTLE ELECTRIC GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 35
License No. NPF-81

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 2 (the facility) Facility Operating License No. NPF-81 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 November 18, 1991, as supplemented March 2, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-81 is hereby amended to read as follows:

   <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 35 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

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

Attachment:
Technical Specification
Changes

Date of Issuance: **February 19, 1993**

ATTACHMENT TO LICENSE AMENDMENT NO. 56

FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

AND

TO LICENSE AMENDMENT NO. 35

FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NO. 50-425

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

Remove Pages	Insert Pages
3/4 4-10	3/4 4-10
B 3/4 4-3	B 3/4 4-3
3/4 4-34	3/4 4-34
B 3/4 4-16	B 3/4 4-16

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 Both power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or both PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one or both PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve, and

   1. With only one PORV OPERABLE, restore at least a total of two PORVs to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

   2. With both PORVs inoperable, restore at least one PORV to OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

c. With one or both block valve(s) inoperable, within 1 hour restore the block valve(s) to OPERABLE status or place its associated PORV(s) in manual control. Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; restore any remaining inoperable block valve to OPERABLE status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 Each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Operating the valve through one complete cycle of full travel, and

b. Performing a CHANNEL CALIBRATION.

BASES

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. The PORVs are equipped with automatic actuation circuitry and manual control capability. No credit is taken for accident mitigation by automatic PORV operation in the analyses for MODE 1, 2, and 3 transients. The PORV(s) are considered OPERABLE in either the manual or automatic mode. The automatic mode is the preferred configuration, since pressure relieving capability is provided without reliance on operator action.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

REACTOR COOLANT SYSTEM

COLD OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following groups of Cold Overpressure Protection Devices shall be OPERABLE when the reactor coolant system (RCS) is not depressurized through a vent path capable of relieving at least 670 gpm water flow at 470 psig.

a. Two power-operated relief valves (PORVs) with lift settings which vary with RCS temperature and which do not exceed the limits established in Figure 3.4-4a (Unit 1), Figure 3.4-4b (Unit 2), or

b. Two residual heat removal (RHR) suction relief valves each with a setpoint of 450 psig ± 3%, or

c. One RHR SRV and one PORV with setpoints as described above.

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

ACTION:

a. In MODE 4, with only one PORV or one RHR SRV OPERABLE, restore one additional valve to OPERABLE status within the next 7 days or depressurize and vent the RCS, as specified in 3.4.9.3 above, within the next 8 hours.

b. In MODES 5 and 6, with only one PORV or one RHR SRV OPERABLE, restore one additional valve to OPERABLE status within the next 24 hours or depressurize and vent the RCS, as specified in 3.4.9.3 above, within the next 8 hours.

c. In MODES 4, 5, or 6 with none of the PORVs or RHR SRVs OPERABLE, depressurize and vent the RCS as specified in Specification 3.4.9.3 above, within the next 8 hours.

d. In the event that the PORVs and/or RHR SRVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.8.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, the RHR suction relief valves or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

e. The provisions of Specification 3.0.4 are not applicable.

PRESSURE/TEMPERATURE LIMITS (Continued)

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

COLD OVERPRESSURE PROTECTION SYSTEMS

The OPERABILITY of two PORVs, two RHR suction relief valves, a PORV and RHR SRV, or an RCS vent capable of relieving at least 670 gpm water flow at 470 psig ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 350°F. The PORVs have adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of all three charging pumps and subsequent injection into a water-solid RCS. The RHR SRVs have adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary to primary water temperature difference of the steam generator less than or equal to 25°F at an RCS temperature of 350°F and varies linearly to 50°F at an RCS temperature of 200°F or less, or (2) the start of all three charging pumps and subsequent injection into a water-solid RCS. A combination of a PORV and an RHR SRV also provides overpressure protection for the RCS.

The Maximum Allowed PORV Setpoint for the Cold Overpressure Protection System (COPS) is derived by analysis which models the performance of the COPS assuming various mass input and heat input transients. Operation with a PORV Setpoint less than or equal to the maximum Setpoint ensures that the nominal 13 EFPY for Unit 1 and 16 EFPY for Unit 2 Appendix G reactor vessel NDT limits criteria will not be violated with consideration for a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of all safety injection pumps while in MODES 4, 5, and 6 with the reactor vessel head installed and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature. Additional temperature limitations are placed on the starting of a Reactor Coolant Pump in Specification 3.4.1.3. These limitations assure that the RHR system remains within its ASME design limits when the RHR relief valves are used to prevent RCS overpressurization.

The Maximum Allowed PORV Setpoint for the COPS will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and in accordance with the schedule in Table 16.3-3 of the VEGP FSAR.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 56 TO FACILITY OPERATING LICENSE NPF-68

AND AMENDMENT NO. 35 TO FACILITY OPERATING LICENSE NPF-81

GEORGIA POWER COMPANY, ET AL.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

1.0 INTRODUCTION

By letter dated November 18, 1991, as supplemented March 2, 1992, Georgia Power Company, et al. (the licensee), proposed license amendments to change the Technical Specifications (TSs) for the Vogtle Electric Generating Plant (Vogtle or the facility), Units 1 and 2. The proposed changes are in response to Generic Letter (GL) 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," which was issued by the NRC on June 25, 1990. A discussion of the proposed changes and the NRC staff evaluation and findings relative to each are addressed in section 3.0 of this Safety Evaluation.

The licensee's earlier response to GL 90-06 dated December 20, 1990, also provided specific commitments regarding the GL 90-06 recommendations for quality assurance, maintenance, and testing of the power-operatered relief valves (PORVs) and block valves.

2.0 BACKGROUND

Generic Letter 90-06 represents the technical resolution of two generic issues and includes changes which are safety enhancements. Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," involves the evaluation of the reliability of PORVs and block valves and their safety significance in PWR plants. The generic letter discussed how PORVs are increasingly being relied on to perform safety-related functions and the corresponding need to improve the reliability of both PORVs and their associated block valves. Proposed staff positions and improvements to the plant's TS were recommended to be implemented at all affected facilities.

Generic Issue 94, "Additional Low-Temperature Overpressure protection for Light-Water Reactors," addresses concerns with the implementation of the requirements set forth in the resolution of Unresolved Safety Issue (USI) A-26, "Reactor Vessel Pressure Transient Protection (Overpressure Protection)." The generic letter discussed the continuing occurrence of overpressure events and the need to further restrict the allowed outage time for a low-temperature overpressure protection channel in operating modes 4, 5, and 6.

3.0 EVALUATION

3.1 Generic Issue 70

In response to Generic Issue 70, the licensee proposed changes to TS 3/4.4.4, "Relief Valves," that require power to be maintained to the associated block valves when closed, and to place the PORVs in manual control when the block valves are not operable. Other changes are editorial and are proposed to better agree with the model TS in GL 90-06. Specifically, the proposed amendments would change the following portions of TS 3.4.4:

(1) In the Limiting Condition For Operation (LCO) statement, change "All power-operated relief valves ..." to "Both power-operated relief valves ..."

(2) In Action Statement a, change "With one or more PORV(s) inoperable ..." to "With one or both PORV(s) inoperable ..." Also, change "... or close the associated block valve(s)" to "... or close the associated block valve(s) with power maintained to block valve(s)."

(3) In Action Statement b, change "With one or more PORV(s) inoperable ..." to "With one or both PORV(s) inoperable ..."

(4) In Action Statement b.2, change "With no PORVs OPERABLE ..." to "With both PORVs inoperable ..."

(5) In Action Statement c, replace the existing statement with the following: "With one or both block valve(s) inoperable, within 1 hour restore the block valve(s) to OPERABLE status or place its associated PORV(s) in manual control. Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; restore any remaining inoperable block valve to OPERABLE status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours."

(6) Supplement TS Bases 3/4.4.4 to include the following: "... The PORV(s) are equipped with automatic actuation circuitry and manual control capability. No credit is taken for accident mitigation by automatic PORV operation in the analyses for MODE 1, 2, and 3 transients. The PORV(s) are considered OPERABLE in either the manual or automatic mode. The automatic mode is the preferred configuration since pressure relieving capability is provided without reliance on operator action."

The NRC staff has reviewed the licensee's proposed modifications to the Vogtle TSs. Since the proposed modifications are consistent with the staff's position previously stated in the generic letter, the staff finds the proposed modifications to be acceptable. The staff also finds that the specific commitments regarding quality assurance, maintenance, and testing of the PORVs and block valves are consistent with the GL and, therefore, are acceptable.

3.2 Generic Issue 94

In response to Generic Issue 94, the licensee proposed changes to TS 3.4.9.3, "Cold Overpressure Protection Systems." The first is an editorial change to relocate the depressurizing of the reactor coolant system (RCS) through a RCS vent from statement c of the LCO to the initial LCO statement. The new action statement c would allow the combination of one residual heat removal (RHR) suction relief valve (SRV) and one PORV to be used for cold overpressure protection. The action statement proposed for Modes 5 and 6 would decrease the allowed out-of-service time (AOT) from 7 days to 24 hours with only one valve available to provide cold overpressure protection. Specifically, the proposed amendments would change the following portions of TS 3.4.9.3:

(1) In the LCO statement, change "At least one of the following Cold Overpressure Protection Systems shall be OPERABLE:" to "At least one of the following groups of Cold Overpressure Protection Devices shall be OPERABLE when the reactor coolant system (RCS) is not depressurized through a vent path capable of relieving at least 670 gpm water flow at 470 psig." (This statement for the vent path was relocated from LCO statement c.)

(2) In LCO statement c, change "The Reactor Coolant System (RCS) depressurized with an RCS vent capable of relieving at least 670 gpm water flow at 470 psig." to "One RHR SRV and one PORV with setpoints as described above."

(3) In Action Statement a, change "With one PORV and one RHR suction relief valve inoperable, either restore two PORVs or two RHR suction relief valves to OPERABLE status within 7 days or depressurize and vent the RCS as specified in Specification 3.4.9.3.c above, within the next 8 hours." to "In Mode 4, with only one PORV or one RHR SRV OPERABLE, restore one additional valve to OPERABLE status within the next 7 days or depressurize and vent the RCS, as specified in 3.4.9.3 above, within the next 8 hours."

(4) Add a new Action Statement b which states, "In MODES 5 and 6, with only one PORV or one RHR SRV OPERABLE, restore one additional valve to OPERABLE status within the next 24 hours or depressurize and vent the RCS, as specified in 3.4.9.3 above, within the next 8 hours."

(5) Relocate the current Action Statement b to Action Statement c. Change "With both PORVs and both RHR suction relief valves inoperable, depressurize and vent the RCS as specified in Specification 3.4.9.3.c, above, within 8 hours." to "In MODES 4, 5, or 6 with none of the PORVs or RHR SRVs OPERABLE, depressurize and vent the RCS as specified in 3.4.9.3 above, within the next 8 hours."

(6) Renumber Action Statement c as Action Statement d, and change "In the event either the PORVs, the RHR suction relief valves, or the RCS vent(s) are used ..." to "In the event that the PORVs and/or RHR SRVs, or the RCS vent(s) are used ..."

(7) Renumber Action Statement d as Action Statement e.

(8) Change TS Bases under "Cold Overpressure Protection Systems" (page B 3/4 4-16) by replacing the first paragraph with the following:

> The OPERABILITY of two PORVs, two RHR suction relief valves, a PORV and RHR SRV, or an RCS vent capable of relieving at least 670 gpm water flow at 470 psig ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 350 degrees F. The PORVs have adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50 degrees F. above the RCS cold leg temperatures, or (2) the start of all three charging pumps and subsequent injection into a water-solid RCS. The RHR SRVs have adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary to primary water temperature difference of the steam generator less than or equal to 25 degrees F. at an RCS temperature of 350 degrees F. and varies linearly to 50 degrees F. at an RCS temperature of 200 degrees F. or less, or (2) the start of all three charging pumps and subsequent injection into a water-solid RCS. A combination of a PORV and a RHR SRV also provides overpressure protection for the RCS.

Also revise the second paragraph of this Bases by changing "Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that the nominal 16 EFPY Appendix G reactor vessel NDT limits ..." to "Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that the nominal 13 EFPY for Unit 1 and 16 EFPY for Unit 2 Appendix G reactor vessel NDT limits ..."

The staff has reviewed the licensee's proposed modifications to the Vogtle TSs. Since the proposed modifications are consistent with the staff's position previously stated in the generic letter, the staff finds the proposed modifications to be acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (57 FR 34585 and 57 FR 34587 both dated August 5, 1992). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: D. Hood
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Date: **February 19, 1993**