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Georgia Power
the southern electric system

November 18, 1991

ELV-02517
1130

Docket Nos. 50-424
50-425

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Gentlemen:

VOGTLE ELECTRIC GENERATING PLANT
GENERIC LETTER 90-06

In our letter ELV-02269 dated December 20, 1990, Georgia Power Company (GPC) indicated that 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)," a Technical Specification change was expected to be submitted prior to the end of the next refueling outage, which occurred 6 months after the issuance of GL 90-06. In accordance with our commitment in ELV-02269 and with the provisions of 10 CFR 50.90 and 10 CFR 50.59, GPC hereby proposes an amendment to the Vogtle Electric Generating Plant (VEGP) Unit 1 and Unit 2 Technical Specifications, Appendix A to Operating Licenses NPF-68 and NPF-81.

Technical Specification 3/4.4.4, "Relief Valves," is being changed to make the wording of the Technical Specifications similar to that proposed by the NRC in GL 90-06. Also, a revision to the bases is being proposed to more accurately describe the function of the power operated relief valves (PORVs).

Technical Specification 3.4.9.3, "Cold Overpressure Protection Systems," currently allows the use of either the pressurizer power-operated relief valves, the residual heat removal suction relief valves (RHR SRVs), or a reactor coolant system (RCS) vent to provide cold overpressure protection, which ensures that the RCS is protected as required by 10 CFR 50, Appendix G and the RHR design limits are not exceeded. A sample Technical Specification for this configuration was not provided in enclosure B of GL 90-06. Additionally, a revision to the bases of 3.4.9 is being proposed to identify plant conditions where overpressure protection is provided by the RHR SRVs and PORVs.

Our proposed revision to Technical Specification 3.4.9.3 reflects the plant configuration at VEGP and allows qualified equipment to provide overpressure protection. The proposed Technical Specification adopts a 24-hour allowed outage time when only a single channel of cold overpressure protection is available in Modes 5 and 6, which is consistent with the staff's position in enclosure B to GL 90-06. The proposed revision to the Technical Specifications

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will require that at least two of these devices be operable; i.e., two PORVs or two RHR SRVs, or one PORV and one RHR SRV when cold overpressure protection is required.

Enclosure 1 provides a description of the proposed change and the basis for the change request.

Enclosure 2 provides the basis for a determination that the proposed change does not involve significant hazards considerations. Enclosure 3 provides the environmental impact determination.

Enclosure 4 provides instructions for incorporating the proposed change into the Technical Specifications. The proposed revised pages are also provided in enclosure 4.

In accordance with 10 CFR 50.91, the designated state official will be sent a copy of this letter and all enclosures.

Mr. C. K. McCoy states that he is a Vice President of Georgia Power Company and is authorized to execute this oath on behalf of Georgia Power Company and that, to the best of his knowledge and belief, the facts set forth in this letter and enclosures are true.

GEORGIA POWER COMPANY

By: CKM'Cy
C. K. McCoy

Sworn to and subscribed before me this 15th day of November, 1991.

Mary M. Bentley
Notary Public

CKM/PAH/gmb

Enclosures:
1. Basis for Proposed Change
2. 10 CFR 50.92 Evaluation
3. Environmental Impact Determination
4. Instructions for Incorporation and Revised Pages

xc: (See next page.)

MY COMMISSION EXPIRES MAY 6, 1995

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Georgia Power

c(w): <u>Georgia Power Company</u>
Mr. W. B. Shipman
Mr. M. Sheibani
NORMS

<u>U. S. Nuclear Regulatory Commission</u>
Mr. S. D. Ebneter, Regional Administrator
Mr. D. S. Hood, Licensing Project Manager, NRR
Mr. B. R. Bonser, Senior Resident Inspector, Vogtle

<u>State of Georgia</u>
Mr. J. D. Tanner, Commissioner, Department of Natural Resources

ENCLOSURE 1

VOGTLE ELECTRIC GENERATING PLANT
GENERIC LETTER 90-06

BASIS FOR PROPOSED CHANGE

Proposed Change

Technical Specification 3/4.4.4 addresses relief valves and their surveillance requirements. In the limiting condition for operation (LCO) statement it is proposed that "all power-operated relief valves..." be changed to "both power-operated relief valves...." In action statement a., it is proposed that "With one or more PORVs..." be changed to "With one or both PORVs..." and the following underlined phrase will be added: "associated block valve(s) with power maintained to the block valve(s); otherwise...." Additional changes are proposed for action statements b., b.2., and c.: in action statement b., "with one or more PORV(s) inoperable" will be changed to "with one or both PORV(s) inoperable; and in b.2., "with no PORV(s) operable" is changed to "with both PORV(s) inoperable"; and c. is changed from "with one or more" to "with one or both."

The following insert is proposed for the bases section of Technical Specification 3/4.4.4.

> The PORV(s) are equipped with automatic actuation circuitry and manual control capability. No credit is taken for 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.

Currently, Technical Specification 3.4.9.3 does not group the relief capacity of a power-operated relief valve (PORV) and a residual heat removal (RHR) system safety relief valve (SRV). The proposed change will relocate the depressurizing of the reactor coolant system (RCS) through a RCS vent from action statement c. of the LCO statement to the initial LCO statement for operability, and a new action statement c. will allow the combination of one RHR SRV and one PORV to be used for cold overpressure protection. An action statement is proposed for Modes 5 and 6 that decreases the allowed out-of-service time (AOT) from 7 days to 24 hours with only one valve available to provide cold overpressure protection. This is consistent with the guidance of 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)."

ENCLOSURE 1 (CONTINUED)

GENERIC LETTER 90-06

BASIS FOR PROPOSED CHANGE

Basis

The LCO statement and action statement are being clarified by replacing "all" with "both," "more" is being changed to "both," and action statements "b.2" and c. are being changed to indicate "both," since the design for the Vogtle Electric Generating Plant has two PORVs and two block valves. Additionally, action statement a. now specifically includes the requirement to maintain power to the closed block valves because by maintaining power to the block valves, the block valves can be readily opened from the control room, and the PORVs could then be utilized for controlling reactor pressure. Closure of the block valves establishes reactor coolant pressure boundary (RCPB) integrity for a PORV that has excessive seat leakage. (Integrity of the RCPB takes priority over the capability of the PORV to mitigate an overpressure event.) The applicability requirements of the LCO to operate with the block valves closed with power maintained to the block valves are intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (Mode 6) so that maintenance can be performed on the PORVs to eliminate the seat leakage condition.

The change to action statement c. establishes remedial measures consistent with the function of the block valves. The block valves' main function is to isolate a stuck-open PORV. Therefore, if the block valves cannot be restored to operable status within 1 hour, the remedial action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck-open PORV at a time that the block valves are inoperable. The time allowed to restore the block valves to operable status is based upon the time limit for inoperable PORVs in action statements b.1. and b.2., since the PORVs are not capable of automatically mitigating an overpressure event when placed in manual control. These actions are also consistent with the use of the PORVs to control reactor coolant system pressure if the block valves are inoperable at a time when they have been closed to isolate PORVs that have excessive seat leakage. The modified action statement does not specify closure of the block valves because such action would not likely be possible when the block valves are inoperable. Likewise, it does not specify either the closure of the PORV, because it would not likely be open, or the removal of power from the PORV. When a block valve is inoperable, placing the PORV in manual control is sufficient to preclude the potential for having a stuck-open PORV that could not be isolated because of an inoperable block valve. For the same reasons, reference is not made to action statements b. and c. for the required remedial actions.

ENCLOSURE 1 (CONTINUED)

GENERIC LETTER 90-06

BASIS FOR PROPOSED CHANGE

In GL 90-06, surveillance requirement 4.4.4.3 was proposed to demonstrate the operability of the emergency power supply for the PORVs and block valves by manually transferring motive and control power from the normal to the emergency power bus. At VEGP, the block valves are powered from safety-related, 480-V busses, which are also tied to the diesel generators. Additionally, the PORVs are electrically solenoid operated, and the solenoids for the PORVs are powered from the Class 1E 125-Vdc system. Therefore, the normal power supplies are from Class 1E sources, and no emergency power supply transfer is required.

The change to bases page B 3/4 4-3 clarifies PORV operability. If one PORV is inoperable due to causes other than excessive seat leakage, within 1 hour the PORV must be restored to operable status or the associated block valve must be closed and power removed from the block valve. In the accident analyses, no credit is taken for the actuation of PORVs for overpressure protection. The only conditions analyzed are those where the actuation of the PORVs would make operation more severe. The pressurizer code safety valves are assumed to provide overpressure protection. The PORVs can be considered operable in either the manual or automatic mode. By maintaining power to the block valve, the PORV can be manually opened from the control room. This condition was analyzed in NUREG/CR-5230, "Shutdown Decay Heat Removal Analysis - Plant Case Studies and Special Issues: Summary Report." In this study, feed and bleed cooling of the primary system was evaluated as an alternative measure for removing decay heat. The study indicated that current Technical Specifications which require that the block valves be closed with power removed upon discovering that a PORV has excessive seat leakage make it unlikely that feed and bleed operations could be initiated in a timely manner. It was proposed that the Technical Specifications require that power be maintained to the block valve, thus increasing the likelihood that timely feed and bleed operations could be initiated from the control room.

Technical Specification 3.4.9.3 allows the use of several means to mitigate the effects of overpressurization of the RCS at reduced temperatures. The operability of two PORVs or two RHR SRVs 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. Other plants, because of their design, utilize either PORVs or RHR SRVs to provide overpressure protection. The proposed Technical Specification will allow the use of either safety grade valve (PORV or RHR SRV) or a combination of the safety grade valves for the purposes of cold overpressure protection. Operation of these valves assures that the nominal Appendix G reactor vessel nondestructive testing (NDT) limits and the RHR design limits will not be exceeded.

ENCLOSURE 1 (CONTINUED)

GENERIC LETTER 90-06

BASIS FOR PROPOSED CHANGE

The following underlined phrase is being added to bases page B 3/4 4-16:

"The OPERABILITY of two PORVs, two RHR suction relief valves, a PORV and RHR SRV, or a RCS vent...."

Additionally, the bases are being revised to indicate specifically the conditions for which the PORVs and RHR SRVs provide overpressure protection. The following paragraph is proposed to be added:

"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 water-solid 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 a RHR SRV with the above notes also provides overpressure protection for the RCS."

Additionally, the bases indicate that the "nominal 16 EFPY Appendix G reactor vessel NDT limits criteria will not be violated..." is being changed to "nominal 13 EFPY for Unit 1 and 16 EFPY for Unit 2 Appendix G reactor vessel NDT limits criteria will not be violated...." These changes reflect the existing limits of the pressure temperature curves.

ENCLOSURE 2

VOGTLE ELECTRIC GENERATING PLANT
REVISION OF TECHNICAL SPECIFICATION 3.4.9.3

10 CFR 50.92 EVALUATION

Pursuant to 10 CFR 50.92, Georgia Power Company (GPC) has evaluated the attached proposed amendment to the Vogtle Electric Generating Plant (VEGP) Unit 1 and Unit 2 Technical Specifications and has determined that operation of the facility in accordance with the proposed amendment would not involve a significant hazards consideration.

Background for Enclosure A

Enclosure A to GL 90-06 discusses the staff positions resulting from the resolution of Generic Issue 70, "Evaluation of Power-Operated Relief Valve and Block Valve Reliability in PWR Nuclear Power Plants." The technical findings and regulatory analysis are discussed in NUREG-1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70." In their discussion, the NRC staff indicated that, over a period of time, the role of power-operated relief valves (PORVs) had changed such that PORVs performed one, or more, of the following safety-related functions:

1. Mitigation of steam generator tube rupture accident,
2. Low-temperature overpressure protection of the reactor vessel during startup and shutdown, or
3. Plant cooldown.

At VEGP, the PORVs and block valves are safety-grade, while at many plants licensed earlier, the valves are not safety-grade. Based upon their studies, the NRC staff proposed changes to Technical Specification 3/4.4.4, "Relief Valves." At VEGP, many of these changes were already incorporated in the Technical Specification, although the wording was slightly different. To provide a consistent approach to the Technical Specifications, it was decided to incorporate most of the proposed wording changes into VEGP Technical Specification 3.4.4. The major change to the Technical Specifications was to specify that power be maintained to the block valve when the PORV was declared inoperable due to excess seat leakage. Maintaining power to the block valve would make it easier for the operator to establish feed and bleed operations in a timely manner.

Result for Enclosure A

1. The proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated. Maintaining power to the block valve actually decreases the probability of core melt as noted in the NRC studies of this subject.

2. The proposed change does not create the possibility of a new or different kind of accident than any previously evaluated. There is no change to the design of the plant.

3. The proposed change does not involve a significant reduction in any margin of safety. Keeping power to the block valves, as noted in the NRC studies, actually reduces the probability of core melt, thus improving the margin of safety.

ENCLOSURE 2 (CONTINUED)

REVISION OF TECHNICAL SPECIFICATION 3.4.9.3

10 CFR 50.92 EVALUATION

Background for Enclosure B

Enclosure B to Generic Letter 90-06 discusses the staff positions resulting from resolution of Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors." Historically, low-temperature overpressure protection (LTOP) was designated as Unresolved Safety Issue A-26 in 1978 (NUREG-0371). Procedures were implemented by pressurized water reactor (PWR) licensees to reduce the potential for LTOP events, and equipment modifications were installed to mitigate such events. Staff guidelines for LTOP are in Standard Review Plan section 5.2.2, "Overpressure Protection," and in its attached Branch Technical Position (BTP) RSB 5-2,"Overpressure Protection of Pressurized Water Reactor While Operating at Low Temperatures."

Major overpressurization of the reactor coolant system while at low temperature, if combined with a critical crack in the reactor pressure vessel welds or plate material, could result in a brittle fracture of the pressure vessel. As long as the fracture resistance of the reactor pressure vessel material is relatively high, these events are not expected to cause vessel failure. The fracture resistance of reactor pressure vessel materials decreases with exposure to fast neutrons, and the rate of decrease is dependent on the metallurgical composition of the vessel walls and welds. If the fracture toughness of the vessel has been reduced sufficiently by neutron irradiation, low-temperature overpressure events could cause propagation of fairly small flaws that might exist near the inner surface. The assumed initial flaw may propagate into a crack that may threaten vessel integrity and core cooling capability. The safety significance of low-temperature transients that have occurred and the unavailability of LTOP protection channels was designated as Generic Issue 94, "Additional Low-Temperature Overpressure Protection."

In this discussion in GL 90-06, it was noted that with the exception of a few plants, the LTOP protection systems consist of either redundant PORVs or redundant safety relief valves (SRVs) in the residual heat removal (RHR) system.

One of the exceptions noted was that newer Westinghouse plants allow the two PORVs and the two RHR SRVs to provide overpressure protection. In reviewing the evaluations assessing the risk of operation, section 5.1.2.1, "Risk Reduction Estimates" in NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,'" states that newer Westinghouse plants that allow either PORVs or RHR SRVs were placed in Group 2; i.e., these plants were evaluated with plants that use safety relief valves in the residual heat removal system for protection. Thus, newer Westinghouse plants that have four qualified LTOP overpressure protection channels available for overpressure protection are being treated as if they have only two. The Technical Specification prepared by the NRC in enclosure B to GL 90-06 does not allow newer Westinghouse plants to take credit for the four qualified LTOP channels. In GL 90-06, the PWR licensee is requested to inform the NRC if, in particular, we intend to follow the staff positions in enclosures A and B, or if we intend to propose alternative measures. This proposed Technical Specification is the proposed alternative measure of Georgia Power Company.

In their proposed Technical Specification, the NRC reduces to 24 hours the allowed out-of-service time for when only one channel is available for cold overpressure protection in Modes 5 and 6. The proposed VEGP Technical Specification also allows only 24 hours out-of-service time when only one channel is available for cold overpressure protection in Modes 5 and 6.

Analysis

The current VEGP Technical Specifications allow the use of either two PORVs, two RHR SRVs, or an RCS vent to provide overpressure protection. The proposed revised Technical Specification allows the use of one PORV and one RHR SRV. One PORV or one RHR SRV is capable of providing cold overpressure protection provided the following restrictions are adhered to:

1. Minimize the time the reactor coolant system (RCS) is maintained in a water-solid condition.

2. Require the intermediate head safety injection pumps to be inoperable when the RCS is in the LTOP condition.*

3. Prior to starting a reactor coolant pump, ensure that the steam generator to RCS temperature difference is less than 25°F at a RCS temperature of 350°F and varies linearly to 50°F at a RCS temperature of 200°F or less.

4. Set the PORV setpoint and have surveillance that checks the PORV actuation electronics and setpoint.

These conditions ensure that the relief capacity of the PORVs and RHR SRVs is sufficient to ensure cold overpressure protection. The proposed Technical Specification will ensure that with at least one of four valves operable, the plant is allowed 7 days in Mode 4 to repair an additional valve. Consistent with NRC guidelines in GL 90-06, with only one valve operable (of four), the plant will be placed in a 24-hour LCO for Modes 5 or 6.

* A proposed amendment (ELV-01567, May 13, 1991) has been submitted which would allow one intermediate head safety injection pump to be operable with the RCS loops not filled.

ENCLOSURE 2 (CONTINUED)

REVISION OF TECHNICAL SPECIFICATION 3.4.9.3

10 CFR 50.92 EVALUATION

Result for Enclosure B

1. The proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated. The proposed change will allow the combination of one RHR SRV and one PORV to be used for cold overpressure protection and reduces the allowed out-of-service time in Modes 5 and 6 to 24 hours from 7 days when only one RHR SRV or PORV is available. These changes do not affect the probability of any initiating event, and therefore, the probability of any previously evaluated accident is not affected. Furthermore, cold overpressure protection will be maintained in accordance with 10 CFR 50, Appendix G. Therefore, there is no effect on the consequences of any accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident than any previously evaluated. Cold overpressure protection is maintained, no new modes of operation are involved, and no new failure modes will be created by the proposed change.

3. The proposed change does not involve a significant reduction in any margin of safety. The limits of 10 CFR 50, Appendix G will continue to be met as before under the existing requirements. The allowed out-of-service time for the case where only one PORV or RHR SRV is available will be more restrictive under the proposed change, requiring corrective action or compensatory measures in 24 hours rather than 7 days under the existing requirements. Therefore, there will be no reduction in any margin of safety.

Conclusion

Based on the preceding analyses, GPC has determined that the proposed change to the Technical Specifications does not involve a significant increase in the probability or consequences of accidents previously evaluated, create the possibility of a new or different kind of accident from any previously evaluated, or involve a significant reduction in a margin of safety. Therefore, GPC concludes that the proposed change meets the requirements of 10 CFR 50.92 (c) and does not involve a significant hazards consideration.

ENCLOSURE 3

VOGTLE ELECTRIC GENERATING PLANT
REVISION OF TECHNICAL SPECIFICATION 3.4.9.3

ENVIRONMENTAL IMPACT DETERMINATION

The criteria for categorical exclusion from the requirement for a specific environmental assessment per 10 CFR 51.21 are specified in 10 CFR 51.22 (b). This amendment request meets the criteria specified in 10 CFR 51.22(c)(9). Specific criteria contained in this section are discussed below.

(i) the amendment involves no significant hazards consideration

As demonstrated in the significant hazards consideration determination in enclosure 2, the requested license amendment does not involve any significant hazards considerations.

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The requested license amendment involves no change to the facility and does not significantly alter the manner of operation in a way which could cause an increase in the amounts of effluents or create new types of effluents.

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes do not impact plant design features or operations that affect radiation protection, radioactive effluent processing, radioactive waste handling, or radiological environmental monitoring. The changes do not result in additional exposure to personnel nor affect levels of radiation present. The proposed changes do not result in significant individual or cumulative occupational radiation exposure.

Based on the above, it is concluded that there will be no impact on the environment resulting from this change, and the change meets the criteria specified in 10 CFR 51.22 for a categorical exclusion from the requirements of 10 CFR 51.21 relative to specific environmental assessment by the NRC.

ENCLOSURE 4

VOGTLE ELECTRIC GENERATING PLANT
REVISION OF TECHNICAL SPECIFICATION 3.4.9.3

INSTRUCTION FOR INCORPORATION

The proposed amendment would be incorporated as follows:

Replace	Insert
3/4 4-9*/4-10	3/4 4-9*/4-10
B 3/4 4-3/4-4*	B 3/4 4-3/4-4*
3/4 4-33*/4-34	3/4 4-33*/4-34
B 3/4 4-15*/4-16	B 3/4 4-15*/4-16

* Overleaf Page