

Georgia Power Company
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201
Telephone 205 877-7122

C. K. McCoy
Vice President, Nuclear
Vogtle Project

October 3, 1994



LCV-0321-B

Docket Nos. 50-424
50-425

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

Gentlemen:

VOGTLE ELECTRIC GENERATING PLANT
REQUEST TO REVISE TECHNICAL SPECIFICATIONS
REVISION TO REACTOR PRESSURE LIMITS

In accordance with the provisions of 10 CFR 50.90 and 10 CFR 50.59, Georgia Power Company (GPC) hereby proposes to amend the Vogtle Electric Generating Plant (VEGP) Unit 1 and Unit 2 Technical Specifications (TS), Appendix A to Operating Licenses NPF-68 and NPF-81. This change will replace the reactor coolant system heatup and cooldown limitations for VEGP Units 1 and 2, contained in TS figures 3.4-2a through 3.4-3b, and the maximum allowable nominal power operated relief valve (PORV) setpoint for the cold overpressure protection system, figures 3.4-4a and 3.4-4b. These changes are the results of new analyses that account for the nonconservatisms identified in NRC Information Notice 93-58, the results of reactor pressure vessel surveillance capsule examinations, and recently issued ASME Code Case N-514.

The surveillance capsule examination results were previously transmitted to the NRC with letter LCV-0321 dated April 4, 1994. Since Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability," has not yet been revised to incorporate Code Case N-514, it is necessary to request an exemption to 10 CFR 50.60 in accordance with 10 CFR 50.12. The exemption request is included in Enclosure 4 of this letter.

The revised figures are based on analyses utilizing the same analytical methods and design basis events as those in the current Technical Specifications. The changes account for nonconservatisms due to the pressure difference between the pressure transmitter and the reactor vessel midplane. The 10 percent relaxation of allowable pressure limits as allowed by ASME Code Case N-514 was used in developing cold overpressure mitigation system (COMS) setpoint curves. The maximum setpoint curve protects against exceeding either the Appendix G limit or an 800 psig limit on the PORV discharge piping, whichever is smaller. Since the revised curves are consistent with previously accepted limits and

9410110194 941003
PDR ADDCK 05000424
P PDR

Handwritten initials: H-031
111

analytical methods, Georgia Power Company requests this change be processed by March 31, 1995. The proposed change and the bases for the changes are described in Enclosure 1 to this letter. Enclosure 2 provides an evaluation pursuant to 10 CFR 50.92 showing that the proposed changes do not involve significant hazards considerations. The marked up Technical Specification changes are provided in Enclosure 3. Enclosure 4 provides a request for an exemption to allow the use of ASME Code Case N-514.

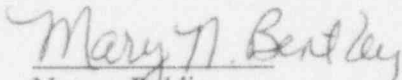
In accordance with 10 CFR 50.91, the designated state official will be sent a copy of this letter and all enclosures.

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: 
C. K. McCoy

Sworn to and subscribed before me this 3rd day of October, 1994.


Notary Public

Enclosures:

1. Basis for Proposed Change
2. 10 CFR 50.92 Evaluation
3. Marked Up Technical Specification Pages
4. Exemption Request - ASME Code Case N-514

cc: (See next page)

U. S. Nuclear Regulatory Commission

Page 3

c(w): Georgia Power Company

Mr. J. B. Beasley, Jr.

Mr. M. Sheibani

NORMS

U. S. Nuclear Regulatory Commission

Mr. S. D. Ebnetter, Regional Administrator

Mr. D. S. Hood, Licensing Project Manager, NRR

Mr. B. R. Bonser, Senior Resident Inspector, Vogtle

State of Georgia

Mr. J. D. Tanner, Commissioner, Dept. of Natural Resources

ENCLOSURE 1
VOGTLE ELECTRIC GENERATING PLANT
REQUEST TO REVISE TECHNICAL SPECIFICATIONS
REVISION TO REACTOR PRESSURE LIMITS

BASIS FOR PROPOSED CHANGE

Proposed Change

The proposed changes to the Technical Specifications (TS) will replace reactor coolant heatup and cooldown curves, figures 3.4-2a, 3.4-2b, 3.4-3a, and 3.4-3b with new heatup and cooldown curves. It will also replace TS figures 3.4-4a and 3.4-4b with new maximum allowable nominal power operated relief valve (PORV) setpoint curves for the cold overpressure protection.

The bases sections for these curves are also being revised to indicate the revisions to the reactor vessel material properties and the addition of conservative assumptions to account for pressure instrument location.

Basis

Westinghouse Nuclear Safety Advisory Letter, NSAL 93005A, and the subsequent NRC Information Notice 93-58 identified nonconservatisms in low temperature overpressure protection setpoint calculations due to a failure to consider the static and dynamic pressure effects due to the location of the pressure sensors. These nonconservatisms affected the cold overpressure protection setpoints and the heatup and cooldown limits for Vogtle Electric Generating Plant (VEGP) Units 1 and 2. The NSAL and information notice described interim administrative restrictions that were intended to provide actions until the Technical Specification's were revised. The heatup and cooldown limits as well as the cold overpressure mitigation system (COMS) setpoints have been recalculated, accounting for the nonconservatisms. These heatup and cooldown curve calculations also included the results of the Reactor Pressure Vessel Material Surveillance Program (WCAP-13931) which were submitted to the NRC with letter LCV-0321 dated April 4, 1994. The 10 percent relaxation in pressure limits which is allowed by ASME Code Case N-514 was utilized in developing the COMS setpoints. The COMS setpoint curve protects against exceeding either the Appendix G limit or an 800 psig limit on the PORV discharge piping, whichever is smaller. The 800 psig limit on the discharge piping becomes limiting at temperature greater than 140°F for VEGP-1 and temperature greater 150°F for VEGP-2.

ENCLOSURE 2

VOGTLE ELECTRIC GENERATING PLANT REQUEST TO REVISE TECHNICAL SPECIFICATIONS REVISION TO REACTOR PRESSURE LIMITS

10 CFR 50.92 EVALUATION

Pursuant to 10 CFR 50.92, Georgia Power Company (GPC) has evaluated the proposed revision to the Technical Specifications and has determined that operation of the facility in accordance with the proposed amendment would not involve any significant hazards considerations.

Background

The ability of the steel pressure vessel containing the reactor core and its primary coolant to resist fracture constitutes an important factor in ensuring plant safety. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to significant fast neutron bombardment. Generally, low alloy ferritic materials (such as A533 Grade B Class 1, which is the base material of the VEGP Units 1 and 2 reactor pressure vessels) show an increase in hardness and tensile properties and a decrease in ductility and toughness during high-energy irradiation. The VEGP Unit 1 reactor vessel radiation surveillance program, designed by Westinghouse, is described in WCAP-11011, "Georgia Power Company Alvin W. Vogtle Unit No. 1 Reactor Vessel Radiation Surveillance Program." The surveillance program was planned to cover the 40 year design life of the reactor pressure vessel and was based on ASTM E185-82, "Standard Practice for Conducting Surveillance Test for Light-Water Cooled Nuclear Power Reactor Vessel." The VEGP Units 1 and 2 heatup and cooldown operational limits are designed to prevent the reactor coolant system (RCS) from being exposed to pressure and temperatures that may result in a component failure due to material embrittlement. The VEGP Units 1 and 2 heatup and cooldown curves are based on the radiation embrittlement changes in the mechanical properties of the reactor pressure vessel specimens, which are periodically removed from the surveillance capsules and evaluated as part of the surveillance program.

The VEGP Unit 1 heatup and cooldown curves and COMS setpoints are being revised to address analysis results of surveillance capsule Y. The VEGP Unit 2 heatup and cooldown curves are based on the methodology of Regulatory Guide 1.99 Revision 2.

ENCLOSURE 2 (CONTINUED)

VOGTLE ELECTRIC GENERATING PLANT REQUEST TO REVISE TECHNICAL SPECIFICATIONS REVISION TO REACTOR PRESSURE LIMITS

10 CFR 50.92 EVALUATION

In addition, Westinghouse has performed a reanalysis of the heatup and cooldown curves and the COMS to determine revised PORV setpoints for VEGP Units 1 and 2. The previous analysis did not consider the pressure difference between the reactor vessel midplane, where Appendix G pressure/temperature limits are defined, and the wide range pressure transmitters located in the RCS loop piping. Therefore, the potential existed for violation of the Appendix G limits. The COMS setpoints also include the low temperature relaxation allowed by ASME Code Case N-514.

Analysis

COMS Setpoints

Revised PORV setpoints for VEGP Units 1 and 2 have been determined for the COMS to address the previously identified nonconservatism. The pressure difference between the reactor vessel midplane, where Appendix G pressure/temperature limits are defined, and the wide range pressure transmitters located in the RCS loop piping had not been considered in the COMS setpoint development. Thus, the potential existed for violation of the Appendix G limits.

This revised analysis accounts for the pressure difference between the pressure transmitter and the reactor vessel midplane. Factors considered in defining the non-conservatism include the static head due to elevation differences and the dynamic head effect of four reactor coolant pump (RCP) operation. The magnitude of the non-conservatism was generically calculated to be 74 psi for four loop plants such as the VEGP units. Application of this 74 psi value is therefore valid for use in this analysis. To offset the 74 psi nonconservatism, revised heatup/cooldown curves were generated based on 16 EFPY for both units. For Unit 1, the Appendix G limits also incorporate the results of the capsule Y material surveillance program described in WCAP-13931. The COMS setpoints were then generated based on the relaxed Appendix G limit allowed by Code Case N-514 at steady-state conditions in conformance with the Westinghouse COMS setpoint development methodology. This methodology maximizes the available operating margin for setpoint selection while maintaining the appropriate level of protection against brittle failure at cold conditions.

ENCLOSURE 2 (CONTINUED)

VOGTLE ELECTRIC GENERATING PLANT REQUEST TO REVISE TECHNICAL SPECIFICATIONS REVISION TO REACTOR PRESSURE LIMITS

10 CFR 50.92 EVALUATION

The design basis transients for determination of COMS setpoints continues to be both a mass injection event with no letdown and a heat injection event from the startup of a RCP in a loop with a resultant secondary-to-primary heat transfer, as described in the Technical Specification Bases. Operation of only one PORV is assumed to mitigate these transients, thus preserving a single failure criteria. PORV setpoints were selected within a range of allowable pressures at nine different temperatures between 70 and 460°F. For temperatures above the COMS arming temperature of 350°F, the highest setpoint ramps rapidly up to an allowable pressure of 2,335 psig. This is intended to prohibit PORV lift for an inadvertent COMS arming at power. The maximum allowable setpoint curve protects against exceeding either the Appendix G limit or an 800 psig limit on the PORV discharge piping, whichever is smaller. The selected setpoints are adjusted to account for temperature streaming and instrument uncertainty as well as thermal transport effects during the heat injection transient. The minimum allowable pressure established by RCP seal integrity criteria is unaffected by this reanalysis.

Two sets of setpoints are selected at each temperature, one for each PORV. The setpoint program for each PORV is staggered to prevent simultaneous opening which could cause an exaggerated minimum pressure (undershoot).

The Vogtle Technical Specifications also allow for the utilization of the RHR relief valves for mitigation of a cold overpressurization transient. The current basis for use of RHR relief valve for cold overpressure mitigation was established assuming an RHR relief valve setpoint of 450 psig plus 10 percent accumulation (45 psig) for a total of 495 psig. It was determined that a single RHR relief valve could mitigate the effects of the design basis mass injection transient and the 50°F ΔT heat injection transient below an RCS temperature of 200°F. Between 200° and 350°F, a single RHR relief valve will mitigate a heat injection transient with a 25°F ΔT . In order to maintain the applicability of these conclusions, it must be shown that the revised Appendix G limits, adjusted up for the 10 percent relaxation and then down for the 74 psi non-conservatism, remain above the 495 psig value. It has been confirmed that the revised Appendix G limits adjusted to account for the 74 psi non-conservatism remain above 495 psig at all temperatures. Therefore, the basis for utilization of the RHR relief valves for cold overpressure mitigation remains valid.

ENCLOSURE 2 (CONTINUED)

VOGTLE ELECTRIC GENERATING PLANT REQUEST TO REVISE TECHNICAL SPECIFICATIONS REVISION TO REACTOR PRESSURE LIMITS

10 CFR 50.92 EVALUATION

Heatup and Cooldown Curves

A method for performing analyses to guard against fast fracture in reactor vessels has been presented in "Protection Against Nonductile Failure," Appendix G to Section III of the ASME Boiler and Pressure Vessel Code. The method uses fracture mechanics concepts and is based on the reference nil-ductility transition temperature (RT_{NDT}).

RT_{NDT} is defined as the greater of either the drop weight nil-ductility transition temperature (NDTT) per ASTM E-208) or the temperature 60°F less than the 50 ft-lb (and 35 mil lateral expansion) temperature as determined from Charpy specimens oriented normal (transverse) to the major working direction of the plate. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IR} curve) which appears in Appendix G to the ASME Code. The K_{IR} curve is a lower bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined using these allowable stress intensity factors.

RT_{NDT} and, in turn, the operating limits of nuclear power plants can be adjusted to account for the effect of radiation on the reactor material properties. The radiation embrittlement changes in mechanical properties of a given reactor pressure vessel shell can be monitored by a reactor surveillance program, such as the VEGP Unit 1 Reactor Vessel Radiation Surveillance Program, in which surveillance capsules are periodically removed from the operating nuclear reactor and then encapsulated specimens tested.

The increase in the average Charpy V-notch temperature (ΔRT_{NDT} at 30 ft-lbs) due to irradiation is added to the initial RT_{NDT} (ART) for radiation embrittlement. This ART (RT_{NDT} initial + ΔRT_{NDT}) is used to index the material to the K_{IR} curve and, in turn, to set operating limits for the nuclear power plant that take into account the effects of irradiation on the reactor vessel materials.

ENCLOSURE 4 (CONTINUED)

VOGTLE ELECTRIC GENERATING PLANT REQUEST TO REVISE TECHNICAL SPECIFICATIONS REVISION TO REACTOR PRESSURE LIMITS

10 CFR 50.92 EVALUATION

Capsule Y was removed from VEGP Unit 1 after about 4.6 EFPY of exposure and was subject to post-irradiation mechanical testing of the Charpy V-notch impact and tensile specimens. Capsule Y was the second capsule to be removed. The summary of the testing and postirradiation data obtained from surveillance capsule Y is contained in WCAP-13931, "Analysis of Capsule Y from the Georgia Power Company Vogtle Unit 1 Reactor Vessel Radiation Surveillance Program."

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G. As described in the Bases, the heatup and cooldown curves define limits to assure prevention of nonductile failure only. Based on the findings from the testing and postirradiation data obtained from surveillance capsule Y, new heatup and cooldown operational limits were established that take into account the effects of irradiation on the reactor vessel materials for Vogtle Unit 1, as well as to reflect the 74 psi pressure difference. The Unit 2 heatup and cooldown curves have also been revised to address the 74 psi pressure.

The assumptions and results in the Final Safety Analysis Report (FSAR) remain valid with the new heatup and cooldown curves, and the modified PORV setpoints. In addition, the changes will have no impact on loss of coolant accident (LOCA), non-LOCA, steam generator tube rupture, and containment integrity analyses.

Conclusion

Conformance of the proposed amendment with the standards for a determination of no significant hazards, as defined in the three factor test of 10 CFR 50.92, is shown in that the proposed amendment:

1. Does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The revised heatup and cooldown limits and PORV setpoints ensure that the Appendix G pressure/temperature limits are not exceeded and, therefore, help ensure that RCS integrity is maintained. The changes do not result in a condition where the design, material, and construction standards of the RCS are altered. In addition, the safety

ENCLOSURE 2 (CONTINUED)

VOGTLE ELECTRIC GENERATING PLANT REQUEST TO REVISE TECHNICAL SPECIFICATIONS REVISION TO REACTOR PRESSURE LIMITS

10 CFR 50.92 EVALUATION

function of the COMS, which is related to accident mitigation, has not been degraded. Therefore, the probability of an accident is not increased by the PORV setpoint change.

The changes do not adversely affect the integrity of the RCS such that its function in the control of radiological consequences is affected. In addition, the changes do not affect any fission barrier. The changes do not degrade or prevent the response of the COMS or other safety-related system to accident scenarios, as described in FSAR chapter 15. In addition, the changes do not alter any assumption previously made in the radiological consequence evaluations nor affect the mitigation of the radiological consequences of an accident described in the FSAR. Therefore, the consequences of an accident previously evaluated in the FSAR will not be increased.

Thus, operation of VEGP Units 1 and 2 in accordance with the proposed license amendment, does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The changes do not cause the initiation of any accident nor create any new credible limiting single failure for safety-related systems and components. The changes do not result in any event previously deemed incredible being made credible. As such, it does not create the possibility of an accident different than any evaluated in the FSAR.

The changes do not have any effect on the ability of the safety-related systems to perform their intended safety functions. The changes do not create failure modes that could adversely impact safety-related equipment. Therefore, it will not create the possibility of a malfunction of equipment important to safety different than previously evaluated in the FSAR. Thus, the proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

ENCLOSURE 2 (CONTINUED)

VOGTLE ELECTRIC GENERATING PLANT REQUEST TO REVISE TECHNICAL SPECIFICATIONS REVISION TO REACTOR PRESSURE LIMITS

10 CFR 50.92 EVALUATION

3. Does not involve a significant reduction in a margin of safety.

The evaluation has shown that the PORV setpoints ensure that the Appendix G pressure/temperature limits are not exceeded. The analysis to support the proposed PORV setpoint change demonstrates that the appropriate criteria, including that of ASME Code Case N-514, are met for the postulated RCS pressures and temperatures. An adequate margin of safety against vessel failure is assured, in part, by the safety factors identified in Appendix G to Section III of the ASME Boiler and Pressure Vessel Code, and the basis for ASME Code Case N-514 as well as added margin to prevent lifting of the PORVs. The heatup and cooldown limits are designed to prevent ductile failure of the reactor vessel and take into account the results of surveillance capsule Y on the reactor vessel materials for VEGP Unit 1. The actuation of the safety-related components and responses of the safety-related systems will remain as modeled in the safety analyses. The changes will have no adverse affect on the availability, operability, or performance of the COMS. Therefore, the changes will not reduce the margin of safety, as described in the bases to any Technical Specification.

Thus, these proposed license amendment does not involve a significant reduction in a margin of safety.

Based on the preceding analysis it is concluded that the operation of VEGP Units 1 and 2 in accordance with the proposed amendment does not result in an increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, nor result in a significant reduction in margins to plant safety. Therefore, the license amendments do not involve a significant hazards consideration, as defined in 10 CFR 50.92.