June 4, 2003

Mr. J. T. Gasser
Vice President
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MB5046 AND MB5047)

Dear Mr. Gasser:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 128 to Facility Operating License NPF-68 and Amendment No. 206 to Facility Operating License NPF-81 for the Vogtle Electric Generating Plant (VEGP), Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated May 8, 2002, as supplemented by letters dated November 26, 2002, and April 10, 2003.

The amendments revise the Reactor Core Safety Limits curve in TS Figure 2.1.1-1, and the Overtemperature Delta Temperature (OTDT) and Overpower Delta Temperature (OPDT) reactor trip functions described in TS Table 3.3.1-1. These changes will provide VEGP, Units 1 and 2 with increased operating margins that will increase the OTDT and OPDT setpoints to account for hot leg temperature fluctuations that are part of the VEGP Setpoint Margin Recovery Program.

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,

/RA/

Frank Rinaldi, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosures:
1. Amendment No. 128 to NPF-68
2. Amendment No. 106 to NPF-81
3. Safety Evaluation

cc w/encls: See next page

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Dear Mr. Gasser:

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The amendments revise the Reactor Core Safety Limits curve in TS Figure 2.1.1-1, and the Overtemperature Delta Temperature (OTDT) and Overpower Delta Temperature (OPDT) reactor trip functions described in TS Table 3.3.1-1. These changes will provide VEGP, Units 1 and 2 with increased operating margins that will increase the OTDT and OPDT setpoints to account for hot leg temperature fluctuations that are part of the VEGP Setpoint Margin Recovery Program.

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DATE	06/04/03	06/04/03	03/03/03	05/02/03	05/12/03	05/29/03	06/04/03

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SOUTHERN NUCLEAR OPERATING COMPANY, INC.

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. 128
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 Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated May 8, 2002, as supplemented by letter dated November 26, 2002, and April 10, 2003, 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. 128, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear 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 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: June 4, 2003

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

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. 106
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 Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated May 8, 2002, as supplemented by letters dated November 26, 2002, and April 10, 2003, 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:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No.   , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear 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 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: June 4, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 128

FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

AND

TO LICENSE AMENDMENT NO. 106

FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NO. 50-425

Replace the following pages of the Appendix A Technical Specifications and associated Bases with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
2.0-2	2.0-2
3.3.1-14	3.3.1-14
3.3.1-15	3.3.1-15
3.3.1-16	3.3.1-16
3.3.1-17	3.3.1-17
3.3.1-18	3.3.1-18
3.3.1-19	3.3.1-19
3.3.1-20	3.3.1-20
3.3.1-21	3.3.1-21
-----------	3.3.1-22
B 2.1.1-7	B 2.1.1-7
B 3.3.1-19	B 3.3.1-19
B 3.3.1-21	B 3.3.1-21
B 3.3.1-65	B 3.3.1-65
------------	B 3.3.1-66

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 128 TO FACILITY OPERATING LICENSE NPF-68

AND AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NPF-81

SOUTHERN NUCLEAR OPERATING COMPANY, INC., ET AL.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

1.0 INTRODUCTION

By letter dated May 8, 2002, as supplemented by letters dated November 26, 2002, and April 10, 2003, Southern Nuclear Operating Company, Inc., et al. (SNC, or the licensee) proposed license amendments to change the Technical Specifications (TS) for the Vogtle Electric Generating Plant (VEGP), Units 1 and 2.

The proposed changes would revise the Reactor Core Safety Limits curve in TS Figure 2.1.1-1, and the Overtemperature Delta Temperature (OTDT) and Overpower Delta Temperature (OPDT) reactor trip functions described in TS Table 3.3.1-1. These changes will provide VEGP, Units 1 and 2 with increased operating margins that will increase the OTDT and OPDT setpoints to account for hot leg temperature fluctuations that are part of the VEGP Setpoint Margin Recovery Program (MRP).

The licensee's November 26, 2002, letter responded to an NRC staff's request for additional information (RAI) dated July 12, 2002, related to Westinghouse document WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions" (Reference 4), Updated Final Safety Analysis Report (UFSAR) Chapter 15 transients, and other clarifications to the NRC staff's RAI of July 12, 2002. The licensee's April 10, 2003, letter documents the NRC staff's questions asked to the licensee during a conference call on February 3, 2003, and the licensee's responses to the NRC staff's questions. The NRC staff's questions addressed the effects of steam line breaks in the main steam isolation valve (MSIV) compartment on safety related equipment to ensure that breaks in this compartment will not result in exceeding the environmental qualification limits of safety related instrumentation. Also, on April 3, 2003, during a conference call between the NRC staff and the licensee, the licensee confirmed that the revised setpoints and time constants continue to meet the applicable acceptance criteria.

The supplemental letters dated November 26, 2002, and April 10, 2003, provided clarifying information that did not change the scope of the May 8, 2002, application nor the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criterion (GDC) 10 requires that specified acceptable fuel design limits (SAFDLs) are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis (95/95 DNB criterion) that DNB will not occur on the limiting fuel rods, and by requiring that fuel centerline temperature stays below the melting temperature. The reactor core safety limits are established to preclude violation of these criteria. Automatic enforcement of the reactor core safety limits is provided by the Reactor Protection System that includes a number of reactor trip functions, two of which are the OTDT and OPDT reactor trips.

The design of the OTDT reactor trip function provides protection against violating the TS safety limit for DNB ratio (DNBR). Westinghouse has designed plants using this trip to ensure that transients that are slow with respect to delays from the core to the instrumentation system, do not result in damage to the core. The OTDT reactor trip function inputs include coolant temperature, pressure, axial flux, and reactor power based on the coolant temperature increase (coolant delta temperature). The licensee credits the OTDT trip in certain UFSAR Chapter 15 safety analyses.

The OPDT reactor trip function provides protection to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1-percent cladding strain) under all possible overpower conditions. Westinghouse has designed plants using the OPDT trip function to ensure that the allowable heat generation rate (kW/ft) of the fuel is not exceeded during normal operation and AOOs. This ensures that the fuel melt temperature is not exceeded. The licensee credits the OPDT trip in the Main Steam Line Break at power accident in its analysis.

GDC 4, “Environmental and Dynamic Effects Design Bases,” requires that structures, systems, and components important to safety shall be designed to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Also, GDC 4 specifies that the dynamic effects of pipe breaks on structures should be considered. Further, NRC document NUREG-0800, “Standard Review Plan (SRP),” Section 6.2.1.2, “Sub-compartment Analysis,” specifies the review criteria for pipe breaks in sub-compartments within containment. The NRC staff has used these same criteria for the analysis of pipe breaks outside the primary containment.

NRC document NUREG-0588, Rev. 1, “Interim Staff Position on Environmental Qualification of Safety Related Equipment,” (Reference 9) provides guidance on ensuring the environmental qualification of Class IE safety-related equipment. This document specifies methods of predicting the adverse conditions to which safety related equipment could be exposed as a result of a design basis accident. The licensee has deviated from the guidance in this document for predicting these adverse conditions. The licensee utilized the GOTHIC computer code to calculate these conditions, rather than the method specified in NUREG-0588, Rev. 1. NUREG-0588, Rev. 1 is not an NRC staff requirement, and is only a guidance document. As discussed in Section 3.3, the NRC staff finds the licensee’s use of the GOTHIC computer code acceptable.

The NRC staff finds that these proposed license amendments satisfy the requirements of GDC 4 and 10, and the criteria of NRC document NUREG-0800, SRP, Section 6.2.1.2 (Reference 7).

3.0 TECHNICAL EVALUATION

The licensee is proposing to revise TS Figure 2.1.1-1, “Reactor Core Safety Limits,” and also proposes revisions to the OTDT (Table 1) and OPDT (Table 2) setpoint parameters for both VEGP, Units 1 and 2 . The OTDT and OPDT setpoint allowable values are also being revised. The licensee calculated the proposed setpoint parameter values in accordance with the NRC approved methodology of Westinghouse document WCAP-8745-P-A (Reference 4).

TABLE 1 - OTDT Setpoint Parameters:

PARAMETER	CURRENT VALUE	REVISED VALUE
K_1	1.12	1.149
K_2	0.0224/°F	0.0224/°F
K_3	0.00115/psi	0.00177/psi
τ_1	8 sec	0 sec
τ_2	3 sec	0 sec
τ_3	2 sec	6 sec
τ_4	28 sec	28 sec
τ_5	4 sec	4 sec
τ_6	0 sec	6 sec

TABLE 2 - OPDT Setpoint Parameters :

PARAMETER	CURRENT VALUE	REVISED VALUE	COMMENT
K_4	1.095	1.10	
K_5	0.02/°F 0.0	0.02/°F 0.0	For increasing T_{AVG} For decreasing T_{AVG}
K_6	0.002/°F 0.0	0.00244/°F 0.0	$T>T''$ $T \leq T''$
τ_7	10 sec	10 sec	
τ_1	8 sec	0 sec	
τ_2	3 sec	0 sec	
τ_3	2 sec	6 sec	
τ_6	0 sec	6 sec	
f_2(AFD) penalty	0.0%	0.0%	

The licensee is requesting approval of the proposed TS changes to address steady-state aperiodic hot leg temperature fluctuations experienced at both VEGP, Units 1 and 2. This is not a unique VEGP phenomenon, as similar effects have been noted at other Westinghouse plants. Although no definitive causes for the temperature fluctuations have been identified, they are believed to be caused by upper plenum flow anomalies. The temperature fluctuations of interest are in the increasing direction, because these can reduce the margin to OTDT and OPDT trip setpoints. To accommodate the effects of streaming and the associated hot leg temperature fluctuations, the licensee proposes to increase the OTDT and OPDT setpoints.

SNC's program to increase the setpoints is referred to as the OTDT and the OPDT Setpoint MRP. The intent of the MRP is to revise the OTDT and OPDT setpoints to increase operating margin. This is accomplished by increasing the steady-state setpoints and by revising the dynamic compensation time constants in the setpoint equations. The setpoint allowable values and core safety limits are also revised to support the MRP. Additionally, the licensee's analyses supporting the MRP include a revision to the Relaxed Axial Offset Control (RAOC) band and the inclusion of a limit or clamp on the compensated temperature difference term in the OTDT trip setpoint. The NRC staff previously reviewed and approved the modification of the RAOC band and the inclusion of the temperature difference clamp. The NRC staff review of these two changes is documented in an NRC staff Safety Evaluation (SE) dated August 9, 2002 (Reference 3), and will not be re-evaluated in this SE.

The increase in the OTDT and OPDT setpoints has been achieved through revised core thermal analyses and revised core thermal limits. The NRC staff evaluation of the proposed setpoint changes included review of the core thermal-hydraulic analyses and non loss-of-coolant accident (LOCA) transient analyses that supports the proposed setpoint changes.

The NRC staff has also reviewed specific changes to the dynamic compensation time constants and allowable values in the OTDT and OPDT setpoint equations. Alteration of the dynamic compensation time constants in the OTDT and OPDT protection circuits could have a significant effect upon safety margin. From the isolated perspective of the electronics alone, such changes can affect both the speed and the magnitude of the system response. The exact nature of the effect upon safety margin, in terms of magnitude and in regard to whether it results in an increase in safety margin or a reduction of it, can be determined only in the context of overall process dynamics.

The licensee has presented to the NRC staff the results of detailed analyses that show that the proposed setpoints and dynamic time constants result in a minimum DNBR that is acceptable to the NRC staff. Hence, a detailed analysis of the instrumentation and protection circuit dynamics is not required, and the NRC staff finds that the proposed modifications to the system time constants are acceptable. In addition, the licensee has indicated in its letter of November 26, 2002, that any adjustments to setpoints and allowable values, if required, will be carried out in accordance with a methodology already accepted by the NRC.

The NRC staff review of the proposed TS changes ensured that: (1) the instrumentation located in the MSIV area will continue to function under the adverse conditions of a main steam line break, (2) all applicable acceptance criteria for those UFSAR Chapter 15 transients that credit the OTDT and OPDT reactor trips continue to be satisfied, and (3) the licensee incorporated NRC approved methods for all analyses associated with the proposed TS revisions to the Safety Limit Curves and the OTDT and OPDT setpoints.

3.1 <u>Core Thermal-Hydraulic Analyses</u>

To support the increase of the OTDT and OPDT setpoints, the licensee proposed a revision to the DNBR limit lines in TS Figure 2.1.1-1, “Reactor Core Safety Limits.” This figure, also referred to as the Core Thermal Limits, defines the acceptable operating regions, assuming various reactor protection systems functions, such that the SAFDLs, (i.e., the design DNBR limit and the centerline fuel melt temperature limit), are satisfied during normal operation and AOO’s. The figure shows the loci of points of Reactor Coolant System (RCS) average temperature as a function of rated thermal power and pressure for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid. As discussed in WCAP-8745-P-A (Reference 4), this figure forms the basis for the calculation of the OTDT and OPDT setpoints.

The licensee accomplished the proposed revision to TS Figure 2.1.1-1 by eliminating overly conservative assumptions in the thermal-hydraulic analyses and reallocating margins built into the current analyses. The licensee incorporated revised analysis parameters for the VEGP MRP that are more representative of current VEGP operating conditions. These changes included use of a minimum measured flow value for the DNB analysis that is consistent with the current limit in the TS and a revised bypass flow fraction that is consistent with the use of thimble plugs in both VEGP cores. The licensee gains additional margin by performing the DNB analyses for the MRP assuming that future VEGP core designs are primarily VANTAGE+ fuel. The licensee proposed to reduce margin between the safety analysis DNBR limit and the design DNBR limit for VANTAGE+ fuel. The licensee’s safety analysis DNBR limit maintains adequate margin to the design DNBR limit to offset known DNBR penalties (i.e., rod bow and transition core penalties). The net remaining DNBR margin after considering any penalties, is available for operating and design flexibility. Additionally, the licensee proposed to gain margin to the OTDT and OPDT setpoints by increasing the overpower limit assumed in the analyses from 118-percent to 120-percent of Rated Thermal Power (RTP).

For the current VEGP core thermal limits, LOPAR fuel is the most limiting fuel design. The revised DNB core thermal limits for the VEGP MRP are based on VANTAGE+ fuel as the most limiting fuel design. The VANTAGE+ safety analysis DNBR limits for the MRP analysis were reduced by decreasing the DNBR margin that was retained in the current safety analysis DNBR limits. The DNBR margin that was previously retained was necessary to offset the transition core DNBR penalty associated with the first LOPAR-to-VANTAGE-5 transition cycle. Because the current core designs are primarily VANTAGE+ (or all VANTAGE+), the amount of offset DNBR margin needed to address any limited reinsertion of LOPAR fuel assemblies is significantly reduced. In response to a NRC staff RAI (Reference 2, question 4), the licensee stated that the amount of DNBR margin included in the MRP analyses for the VANTAGE+ fuel was greater than 15-percent. Prior to the MRP analyses, approximately 17-percent had been included in these analyses. The known DNBR penalties that had to be addressed for the MRP analyses were less than 5-percent (not including any transition core penalty). Therefore, approximately 10-percent margin remains available for operating and design flexibility to offset any cycle-specific DNBR penalties. The NRC staff determined that this amount of margin is acceptable.

The licensees DNBR analyses for the revised core thermal limits assumed that the VEGP core designs are primarily VANTAGE+ fuel. To ensure that the LOPAR fuel is not limiting with respect to DNB, the licensee reduced the LOPAR $F_{\Delta H}$ limit from the current value of 1.53 specified in the COLR to a value of 1.30. Only a limited number of LOPAR fuel assemblies may be reinserted in low power core locations in future VEGP core designs. Operation with a mixed core of VANTAGE+ and LOPAR fuel is still addressed using the current VEGP transition core DNB methodology, as described in UFSAR Section 4.2.2. The LOPAR fuel safety analysis DNBR limits are unchanged from the limits in the current safety analyses. The licensee will continue to evaluate the use of LOPAR fuel on a cycle-specific basis during the reload core design process.

The licensee increased the overpower limit used in the analyses from 118 percent to 120-percent of RTP. The NRC staff questioned the use of a 120-percent overpower limit and the licensee provided additional information in the response to question 6 of Reference 2. The methodology described in WCAP-8745-P-A (Reference 4) utilizes an overpower limit of 118-percent and notes that this is a typical value. The NRC staff's SER for that WCAP noted this and states, “typically at 118 percent nominal power level.” The true limit is the kW/ft value that results in fuel centerline melting, which is 22.4 kW/ft. The licensee stated that for VEGP, even considering the 120-percent overpower limit, there is margin to the limit of 22.4 kW/ft. Based on this, the NRC staff determined that the use of an overpower limit of 120-percent RTP is acceptable.

The licensee incorporated existing thermal-hydraulic design criteria and methods in performing the analyses for the VEGP MRP. The methods used are the same as those presented in the VEGP UFSAR and include the improved THINC-IV PWR design modeling method (Reference 5) and the revised thermal design procedure (RTDP) (Reference 6). The current RTDP design limit DNBR values and the DNBR correlations approved for the VEGP units remain unchanged for the MRP. The licensee employed the methodology described in

WCAP-8745-P-A (Reference 4) to calculate the revised the OTDT and OPDT setpoints and confirmed that the limitations and restrictions identified in this methodology are satisfied (Reference 2, question 1).

The NRC staff has reviewed the licensee's thermal-hydraulic analyses that support the proposed changes to TS Figure 2.1.1-1 and the OTDT and OPDT setpoints. The licensee incorporated NRC approved methods in performing these analyses. The revisions to TS Figure 2.1.1-1 and the OTDT and OPDT setpoints are based on elimination of overly conservative assumptions and reallocating margins in the current analyses. Although some of the thermal margin has been reduced, the licensee has not removed all available thermal margin. Acceptability of the proposed changes is also demonstrated through reanalysis of the UFSAR Chapter 15 non-LOCA transients that credit the OTDT or OPDT trip functions (discussed in Secton 3.2). Based on this, the NRC staff determined that the proposed change to TS Figure 2.2.1-1 is acceptable.

3.2 <u>Non-LOCA Transients</u>

The licensee's proposed MRP related TS changes for core thermal limits and OTDT and OPDT setpoints have an impact on the VEGP UFSAR Chapter 15 non-LOCA transient analyses. The licensee's revised core thermal limits, as discussed in 3.1 above, were used to calculate the safety analysis limit OTDT and OPDT setpoints. The licensee calculated these safety analysis OTDT and OPDT setpoints in accordance with the NRC approved methodology described in WCAP-8745-P-A (Reference 4). The licensee credits the OTDT and OPDT reactor trip functions in certain non-LOCA transient analyses, and as such, must evaluate the impacts of the setpoint changes on the affected events and their specific acceptance criteria. The NRC staff reviewed these events to ensure that their individual acceptance criteria, as specified in the SRP (Reference 7), remain satisfied considering the proposed OTDT and OPDT setpoints.

The licensee reanalyzed or evaluated the following VEGP Units 1 and 2 UFSAR Chapter 15 safety analyses that rely on the OTDT and OPDT trip functions for primary protection:

- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power - OTDT trip
- Uncontrolled Boron Dilution - OTDT trip
- Loss of External Electrical Load and /or Turbine Trip - OTDT trip
- Accidental Depressurization of the Reactor Coolant System - OTDT trip
- Steamline Break Core Response at Power - OPDT trip

The licensee reanalyzed all of these events except for the Uncontrolled Boron Dilution event, which was evaluated with respect to the setpoint changes. The licensee addressed the evaluation performed for this event in response to the NRC staff's RAI (Reference 2, questions 8.a and 8.e). The safety analysis acceptance criteria evaluated for this event is the minimum amount of time available for the operators to terminate an inadvertent boron dilution prior to complete loss of shutdown margin. The licensee determined that in the automatic rod control mode, the modifications associated with the MRP do not impact the time of occurrence of an alarm, and therefore, do not impact the analysis of record as described in the UFSAR. For the reactor in manual rod control, the licensee determined that from time of indication (OTDT trip) until loss of shutdown margin, the operator has 30.5 minutes to terminate the event. This time exceeds the minimum acceptable time requirement of 15 minutes, as documented in the SRP (Reference 7) and is, therefore, acceptable.

The licensee reanalyzed the four remaining affected transients in accordance with the current licensing basis methodology as described in the VEGP UFSAR. The events were reanalyzed using the LOFTRAN computer code (Reference 8). The licensee provided quantitative results for these analyses in its submittal (Reference 1), and supplemented that information in the response to question 8 of the RAI (Reference 2). The specific acceptance criteria for these events are identified in the SRP (Reference 7), and include DNBR, fuel temperature, RCS pressure and secondary pressure. The licensee provided results (Reference 1 and Reference 2, question 8) that demonstrated for each of these transients that their respective SRP safety analysis acceptance criteria are satisfied. Based on these acceptable results, the NRC staff determined that the proposed OTDT and OPDT setpoint changes are acceptable.

3.3 Main Steamline Break Outside Containment

In order to ensure that the instrumentation located in the MSIV area will continue function under the adverse conditions of a main steam line break in this area, the licensee calculated a bounding curve of temperature as a function of time. The licensee's April 10, 2003, letter describes the calculation of this curve. The bounding curve is a composite of 85 cases covering initial power levels ranging from 102-percent to 0-percent power, and break sizes ranging from 1.0 ft^2 to 0.1 ft^2. The largest postulated break considered in the VEGP UFSAR (Section 3.11.B) is 1.0 ft^2. The smallest break (0.1 ft^2) is less than the smallest area currently analyzed in the UFSAR. The licensee's analyses ensured that all safety related equipment is qualified to operate at temperatures up to the temperature of this composite curve.

The LOFTRAN computer code is used to calculate the mass and energy released from the ruptured pipe. In the calculation of the response to this steam line break outside containment, the steam generator tubes are uncovered. LOFTRAN has been previously found acceptable by the NRC for this purpose.

The licensee calculated the bounding curve using the GOTHIC computer code. GOTHIC was developed by Numerical Applications Incorporated for the Electric Power Research Institute. GOTHIC has not previously been used as part of the VEGP licensing basis. However, GOTHIC has been used in other licensing applications that have been approved by the NRC on a case specific basis.

The subject of this review was limited to steam line breaks in the MSIV area and, therefore, the NRC staff did not review the acceptability of GOTHIC for calculating phenomena associated with other sub-compartment analysis that were not pertinent to this review, but would be part of a more comprehensive sub-compartment analysis, such as liquid entrainment in exit flows, critical flow correlations, and destruction or non-destruction of vents. For this reason, the acceptability of GOTHIC for this application does not apply to other sub-compartment analyses.

The effects on structures, such as walls and ducts, were not considered in the licensee's analyses because as stated in the licensee's April 10, 2003, letter:

> the analyses performed for this application generate only low pressures, approximately 15 psia, which are well below the structural design pressure (18 psia) and the minimum equipment qualification pressure (15 psig).

Based on the low pressures encountered during this postulated accident and the margins to structural design and equipment qualification pressures, the NRC staff finds this acceptable.

UFSAR Section 3F contains the sub-compartment analyses to support the integrity of compartments inside and outside containment with respect to a high energy line break. These analyses have not changed as a result of these TS amendments.

Figure 1 of the licensee's April 10, 2003, letter provides the composite temperature curve. The letter also describes how the curve was constructed. The results of a representative calculation shown in Figure 3 of this letter and the accompanying discussion demonstrate that this composite curve is conservative. This is based on the fact that for the temperature vs. time curves for the individual cases the vapor temperature falls off rapidly following the temperature spike due to steam generator tube uncovery while the composite curve does not fall off but remains at a higher temperature.

SRP Section 6.2.1.2, "Sub-compartment Analysis," provides criteria for a sub-compartment analysis acceptable to the NRC. Note that these are given in terms of GDC 50, "Containment Design Basis," that applies to sub-compartments within containment. However, the criteria of SRP 6.2.1.2 can be applied to sub-compartments outside containment since their purpose is to ensure a conservative analysis.

SRP Subsection 6.2.1.II.B.1 addresses the initial conditions assumed for the analyses. The licensee assumed an initial temperature of 120 °F based on two standard deviations of the measured temperature. Other input parameters were unchanged from previous analyses. The outside air temperature is assumed to be 95 °F. The NRC staff determined these initial conditions are conservative and acceptable.

SRP Subsection 6.2.1.II.B.2 addresses the adequacy of compartment nodalization. The licensee's April 10, 2003, letter states that the nodalization used with GOTHIC is the same as that in UFSAR Section 3F, except that Node 1 is divided into two matching adjacent nodes. This is an improvement and is acceptable.

SRP Subsection 6.2.1.II.B.3 discusses vent flow paths that connect the break compartment with other compartments. The licensee states that the flow paths are modeled as open areas bounded by concrete walls and slabs and major structural steel. No heating ventilation and air conditioning ducts are modeled as vent pathways. They may be modeled as flow obstructions. Since the pressure increase in the break compartment is minimal, the NRC staff determined that the details of the vents are not significant.

SRP Subsection 6.2.1.II.B.4 discusses the flow behavior between compartments. Since the pressure in the break room is low, the NRC staff determined that the details of the modeling of critical flow and liquid entrainment are not important to the VEGP MSIV area calculations.

The licensee included heat transfer to walls and internal surfaces in the calculation. Walls, floors and ceilings are used in the Uchida Condensation Heat Transfer Correlation. This correlation is generally conservative and is acceptable to the NRC staff. This correlation was combined with radiation and convective heat transfer, when applicable.

The GOTHIC drop-liquid conversion model was included in these calculations. The licensee stated that one case was run with and one case was run without this model. The results showed no discernible difference as noted in the licensee's April 10, 2003, submittal.

Based on the NRC staff's review of the licensee's submittals following the guidance of SRP Section 6.2.1.2, as appropriate, the NRC staff determined that the licensee's calculation provides reasonable assurance of conservative environmental conditions in the MSIV area.

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 requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 (67 FR 48221). 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.

7.0 REFERENCES

1. Letter from J. B. Beasly, Jr. (SNC) to U.S. Nuclear Regulatory Commission, "Request to Revise Technical Specifications, Reactor Trip System Instrumentation Overtemperature Delta Temperature and Overpower Delta Temperature Reactor Trip Functions (LCV-1617)," dated May 8, 2002.

2. Letter from J. T. Gasser, (SNC) to U.S. Nuclear Regulatory Commission, "Request to Revise Technical Specifications, Reactor Trip System Instrumentation Overtemperature Delta Temperature and Overpower Delta Temperature Reactor Trip Functions Request For Additional Information (LCV-1617-A)," dated November 26, 2002.

3. Letter from U.S. Nuclear Regulatory Commission to J. B. Beasly, Jr. (SNC), "Vogtle Nuclear Generating Plant, Units 1 and 2 RE: Issuance of Amendment (TAC NOS. MB3568 and MB3569)," ADAMS Accession No. ML022270088, dated August 9, 2002.

4. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," dated September, 1986.

5. WCAP-12330-A, "Improved THINC-IV Modeling for PWR Core Design," dated September 1991.

6. WCAP-11397-P-A, "Revised Thermal Design Procedure," dated April 1989.

7. NUREG-0800, "Standard Review Plan."

8. WCAP-7907-P-A, "LOFTRAN Code Description," dated April, 1984.

9. NUREG-0588, Rev. 1, "Interim Staff Position on Environmental Qualification of Safety Related Equipment."

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