October 19, 1990

Dockets Nos. 50-424
and 50-425

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

Dear Mr. Hairston:

SUBJECT: ISSUANCE OF AMENDMENT NO. 36 TO FACILITY OPERATING LICENSE NPF-68 AND AMENDMENT NO. 16 TO FACILITY OPERATING LICENSE NPF-81 - VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 (TACs 71564/71565)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. to Facility Operating License No. NPF-68 and Amendment No. to Facility Operating License No. NPF-81 for the Vogtle Electric Generating Plant, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated July 28, 1989.

The amendments revise TS 3.4.9, "Pressure/Temperature Limits," and the associated bases concerning the number of effective full power years (EFPY) for which the Vogtle Unit 1 heatup and cooldown curves are applicable. The applicability of the Vogtle Unit 1 curves decreased from 16 EFPY to 13 EFPY based on an analysis performed by Westinghouse Electric Corporation in response to Generic Letter 88-11.

A copy of the related Safety Evaluation supporting the amendments is also enclosed. Notice of issuance of the amendments will be included in the Commission's biweekly <u>Federal Register</u> notice.

Sincerely,

Original signed by:

Timothy A. Reed, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 36 to NPF-68
2. Amendment No. 16 to NPF-81
3. Safety Evaluation

cc w/enclosures:
See next page

CP-1

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DATED: October 19, 1990

AMENDMENT NO. 36 TO FACILITY OPERATING LICENSE NPF-68 - Vogtle Electric Generating Plant, Unit 1

AMENDMENT NO. 16 TO FACILITY OPERATING LICENSE NPF-81 - 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. 36
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 July 28, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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. 36 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. GPC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David B. Matthews

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: October 19, 1990

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. 16
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 July 28, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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. 16 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. GPC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days 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: October 19, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 36

FACILITY OPERATING LICENSE NO. NPF-68

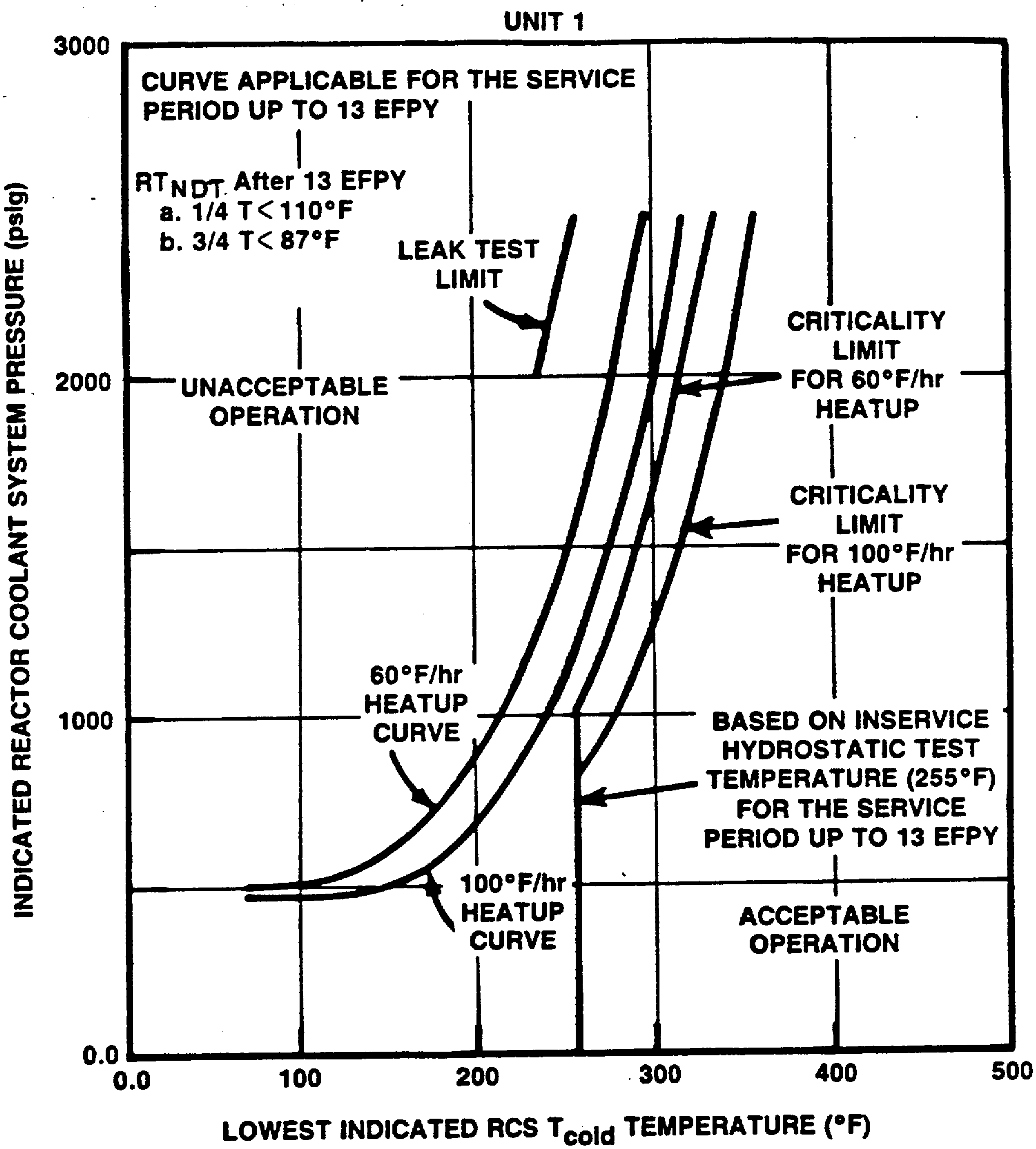
AND LICENSE AMENDMENT NO. 16

FACILITY OPERATING LICENSE NO. NPF-81

DOCKETS NOS. 50-424 AND 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. The corresponding overleaf pages are also provided to maintain document completeness.

Amended Page	Overleaf Page
3/4 4-31	
3/4 4-32	3/4 4-31a
B 3/4 4-8	B 3/4 4-7
B 3/4 4-10	B 3/4 4-9a

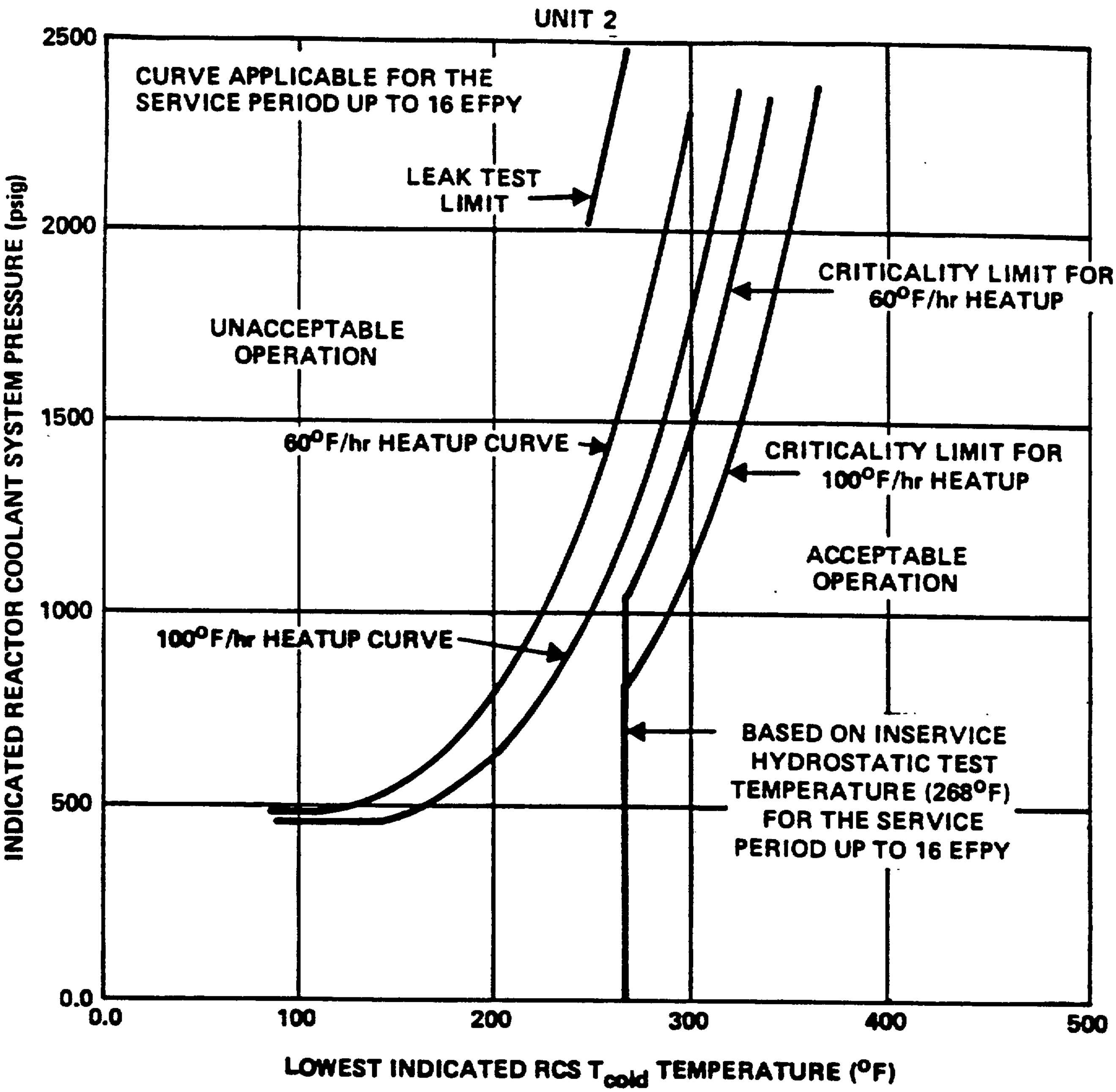


MATERIAL BASIS

Copper Content:	Assumed - 0.10 Wt % (Actual - 0.06 Wt %)
RT_{NDT} Initial:	Assumed - 40°F (Actual - 30°F)
RT_{NDT} After 13 EFPY	@ 1/4 T<110°F @ 3/4 T<87°F

FIGURE 3.4-2a

UNIT 1 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 13 EFPY



MATERIAL BASIS

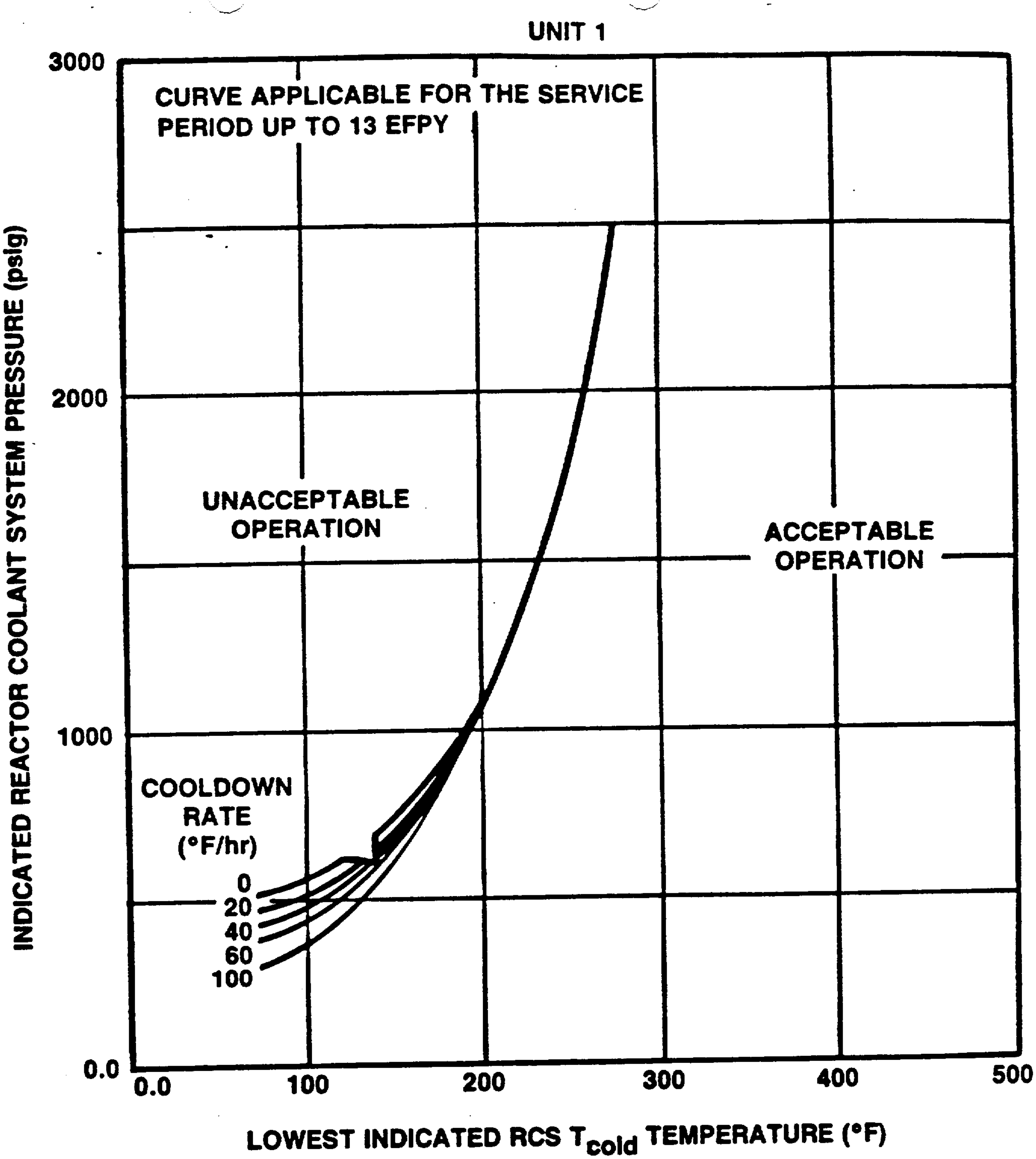
Copper Content: Assumed - 0.10 Wt % (Actual - 0.05 Wt %)

RT_{NDT} Initial: Assumed - NA °F (Actual - 50°F)

RT_{NDT} After 16 EFPY @ 1/4 T = 123°F
@ 3/4 T = 97°F

FIGURE 3.4-2b

UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 16 EFPY



MATERIAL BASIS

Copper Content:	Assumed - 0.10 Wt % (Actual - 0.06 Wt %)
RT_{NDT} Initial:	Assumed - 40°F (Actual - 30°F)
RT_{NDT} After 13 EFPY	@ 1/4 T<110°F @ 3/4 T<87°F

FIGURE 3.4-3a

UNIT 1 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 13 EFPY

SPECIFIC ACTIVITY (Continued)

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the specific count should be made for gases (i.e., xenons and kryptons) and particulates (i.e., cobalt and cesiums) in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

The identification of 95% of the gross specific activity by definition does not obligate VEGP into calculating $\bar{E}$ every time gross activity is determined.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

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:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2a and 3.4-3a (Unit 1), Figures 3.4-2b and 3.4-3b (Unit 2) for the service period specified thereon:

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

   b. Figures 3.4-2a and 3.4-3a (Unit 1), Figures 3.4-2b and 3.4-3b (Unit 2) define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

PRESSURE/TEMPERATURE LIMITS (Continued)

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The auxiliary spray shall not be used if the temperature difference between the pressurizer and the auxiliary spray fluid is greater than 625°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975.

The heatup and cooldown limit curves shown in Figures 3.4-2a and 3.4-3a are applicable to Unit 1 for up to 13 EFPY and are based on Westinghouse-developed generic curves which were developed assuming a 40°F initial RT_{NDT} and a copper content of 0.10 WT% for the most limiting material. These curves are applicable to Unit 1 since its most limiting material (Table B 3/4.4-1a) has both a lower initial RT_{NDT} (30°F) and a lower copper content (0.06 WT%). These curves, however, are not applicable to Unit 2, since its most limiting material (Table B 3/4.4-1b) has a higher initial RT_{NDT} (50 compared to 40°F). Separate heatup and cooldown limit curves were developed based on the actual material properties of the most limiting material for Unit 2 up to 16 EFPY. The Unit 2 curves are shown in Figures 3.4-2b and 3.4-3b.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of the Effective Full Power Years (EFPY) of service life. The EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

TABLE B 3/4.4-1b

UNIT 2 REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	ASME MATERIAL TYPE	CU (%)	Ni (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)	AVERAGE NMWD* (FT-LB)
Closure Head Dome	R9-1	A533B CL. 1	0.07	0.61	0.008	-40	-30	123
Closure Head Torus	R10-1	A533B CL. 1	0.07	0.64	0.010	-30	0	84
Closure Head Flange	R7-1	A508 CL. 2	-	0.72	0.011	10	10	130
Vessel Flange	R1-1	A508 CL. 2	-	0.87	0.011	-60	-60	115
Inlet Nozzle	B9806-1	A508 CL. 2	0.07	0.84	0.010	-50	-50	119
Inlet Nozzle	B9806-2	A508 CL. 2	0.06	0.83	0.009	-40	-40	128
Inlet Nozzle	R5-1	A508 CL. 2	0.09	0.87	0.008	-20	-20	147
Inlet Nozzle	R5-2	A508 CL. 2	0.08	0.85	0.009	-20	-20	134
Outlet Nozzle	R6-3	A508 CL. 2	-	0.69	0.011	-10	-10	122
Outlet Nozzle	R6-4	A508 CL. 2	-	0.66	0.010	-10	-10	140
Outlet Nozzle	B9807-3	A508 CL. 2	-	0.66	0.005	-30	-30	116
Outlet Nozzle	B9807-4	A508 CL. 2	-	0.64	0.010	10	10	132
Nozzle Shell	R3-1	A533B CL. 1	0.20	0.67	0.015	0	20	79
Nozzle Shell	R3-2	A533B CL. 1	0.20	0.67	0.015	0	40	79
Nozzle Shell	R3-3	A533B CL. 1	0.15	0.62	0.010	-10	60	84
Intermediate Shell	R4-1	A533B CL. 1	0.06	0.64	0.009	-20	10	95
Intermediate Shell	R4-2	A533B CL. 1	0.05	0.62	0.009	-10	10	104
Intermediate Shell	R4-3	A533B CL. 1	0.05	0.59	0.009	0	30	84
Lower Shell	B8825-1	A533B CL. 1	0.05	0.59	0.006	-20	40	83
Lower Shell	R8-1	A533B CL. 1	0.06	0.62	0.007	-20	40	87
Lower Shell**	R8628-1	A533B CL. 1	0.05	0.59	0.007	-20	50	85
Bottom Head Torus	R12-1	A533B CL. 1	0.17	0.64	0.012	-20	-20	89
Bottom Head Dome	R11-1	A533B CL. 1	0.10	0.62	0.008	-30	-30	115
Intermediate & Lower Shell Vertical Weld Seams	G1.60	SAW	0.07	0.13	0.007	-10	-10	147
Intermediate to Lower Shell Girth Weld Seam	E3.23	SAW	0.06	0.12	0.007	-50	-30	90

*Upper Shelf energy, normal to major working direction.

**Limiting material.

PRESSURE/TEMPERATURE LIMITS (Continued)

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown for Units 1 and 2 in Table B 3/4.4-1a and b, respectively. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by either Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2. The heatup and cooldown limit curves of Figures 3.4-2a and 3.4-3a (Unit 1), Figures 3.4-2b and 3.4-3b (Unit 2) include predicted adjustments for this shift in RT_{NDT} at the end of 13 (Unit 1) and 16 (Unit 2) EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 16.3-3 of the VEGP FSAR. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in the following paragraphs.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 36 TO FACILITY OPERATING LICENSE NPF-68

AND AMENDMENT NO. 16 TO FACILITY OPERATING LICENSE NPF-81

GEORGIA POWER COMPANY, ET AL.

DOCKETS NOS. 50-424 AND 50-425

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

1.0 INTRODUCTION

By letter dated July 28, 1989, Georgia Power Company, et al. (the licensee), requested a change to the Technical Specifications (TSs) for the Vogtle Electric Generating Plant (VEGP), Units 1 and 2. The proposed change responds to Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations" and revises TS 3.4.9, "Pressure/Temperature Limits," and the associated bases concerning the number of effective full power years (EFPY) for which the Unit 1 heatup and cooldown curves are applicable. Specifically, TS Figures 3.4-2a and 3.4-3a, as well as pages B3/4 4-8 and 4-10, are revised to reflect the change of EFPY from 16 to 13. The Vogtle Unit 2 pressure/temperature (P/T) limits for 16 EFPY remain unchanged.

To evaluate the P/T limits, the NRC staff used the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide TSs for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the TSs. The P/T limits are among the limiting conditions of operation in the TSs for all commercial nuclear plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature.

Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

2.0 EVALUATION

The NRC staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Vogtle 1 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with Section 1 of Regulatory Guide (RG) 1.99, Revision 2. The staff has determined that the material with the highest ART at 13 EFPY for Vogtle 1 was the intermediate shell plate B8805-2 with 0.08% copper (Cu), 0.59% nickel (Ni), and an initial RT_{ndt} of 20°F. The material with the highest ART at 16 EFPY for Vogtle 2 was the lower shell plate B8628-1 with 0.05% Cu, 0.59% Ni, and an initial RT_{ndt} of 50°F.

For the limiting beltline material in Vogtle 1, intermediate shell plate B8805-2, the NRC staff calculated the ART to be 101.2°F at 1/4T (T = reactor vessel beltline thickness) and 85.9°F at 3/4T for 13 EFPY. The staff used a neutron fluence of 7.7E18 n/cm^2 at 1/4T and 2.7E18 n/cm^2 at 3/4T.

For the limiting beltline material in Vogtle 2, the NRC staff calculated the ART for lower shell plate B8628-1 to be 111°F at 1/4T and 93.4°F at 3/4T for 16 EPFY. The staff used a neutron fluence of 9.4E18 n/cm^2 at 1/4T and 3.4E18 n/cm^2 at 3/4T.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 110°F at 1/4T and 87°F at 3/4T for the limiting material for 13 EFPY in Vogtle 1. Because the licensee's ART of 110°F is more conservative than the staff's ART of 101.2°F, the staff judges that the difference of about 9°F is acceptable. Substituting the ART of 110°F into equations in SRP 5.3.2, the staff verified that the proposed Vogtle 1 P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

For Vogtle 2, the licensee calculated an ART of 123°F at 1/4T and 97°F at 3/4T for the limiting material for 16 EFPY. Because the licensee's ART of 123°F is more conservative than the staff's ART of 111°F, the staff judges that the difference of about 12°F is acceptable. Substituting the ART of 123°F for 1/4T and 97°F for 3/4T into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 20°F for Vogtle 1 and 10°F for Vogtle 2, the NRC staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. The material with the lowest unirradiated USE in Vogtle 1 is intermediate shell plate B8805-1 with 90 ft-lb. Using the method in RG 1.99, Rev. 2, the predicted Charpy USE of shell plate B8805-1 after being exposed to a fluence of 3.17E19 n/cm^2 (32 EFPY) is 68.4 ft-lb.

The material with the lowest unirradiated USE in Vogtle 2 is lower shell plate B8825-1 with 83 ft-lb. Using the method in RG 1.99, Rev. 2, the predicted Charpy USE of shell plate B8825-1 after being exposed to a fluence of 3.17E19 n/cm^2 (32 EFPY) is 63.1 fl-lb. Both predicted USEs are greater than 50 ft-lb and, therefore, are acceptable.

The NRC staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 13 EFPY for Vogtle 1 and 16 EFPY for Vogtle 2 because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. Since the licensee used RG 1.99, Rev. 2, to calculate ART, the proposed P/T limits also satisfy GL 88-11. Hence, the proposed P/T limits may be incorporated into the Vogtle TSs.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes in requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The 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. 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.

4.0 CONCLUSION

The Commission's proposed determination that the amendments involve no significant hazards consideration was published in the Federal Register on November 1, 1990 (54 FR 46149). The Commission consulted with the State of Georgia. No public comments were received, and the State of Georgia did not have any comments.

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

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