June 8, 1995

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Mr. C. K. McCoy
Vice President - Nuclear
Vogtle Project
Georgia Power Company
P. O. Box 1295
Birmingham, AL 35201

SUBJECT: ISSUANCE OF AMENDMENTS - VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 (TAC NOS. M90966 AND M90967)

Dear Mr. McCoy:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 87 to Facility Operating License NPF-68 and Amendment No. 65 to Facility Operating License NPF-81 for the Vogtle Electric Generating Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated October 3, 1994, as supplemented by letter dated March 1, 1995.

The amendments revise TS 3/4.4.9, Pressure/Temperature Limits, and its associated Bases, to provide new reactor coolant system heatup and cooldown limitations and new power-operated relief valve setpoints for the low temperature overpressure protection system.

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

Sincerely,

Original signed by:
Louis L. Wheeler, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosures:
1. Amendment No. 87 to NPF-68
2. Amendment No. 65 to NPF-81
3. Safety Evaluation

cc w/encl: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 8, 1995

Mr. C. K. McCoy
Vice President - Nuclear
Vogtle Project
Georgia Power Company
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Mr. C. K. McCoy
Georgia Power Company

Vogtle Electric Generating Plant

cc:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 87
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 October 3, 1994, as supplemented by letter dated March 1, 1995, 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. 87 , 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 from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert E Martin

for ~~Herbert~~ N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: June 8, 1995

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 65
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 October 3, 1994, as supplemented by letter dated March 1, 1995, 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. 65 , 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 from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert E Martin
for ~~Herbert N. Berkow~~, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: June 8, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 87

FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

AND

TO LICENSE AMENDMENT NO. 65

FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NO. 50-425

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

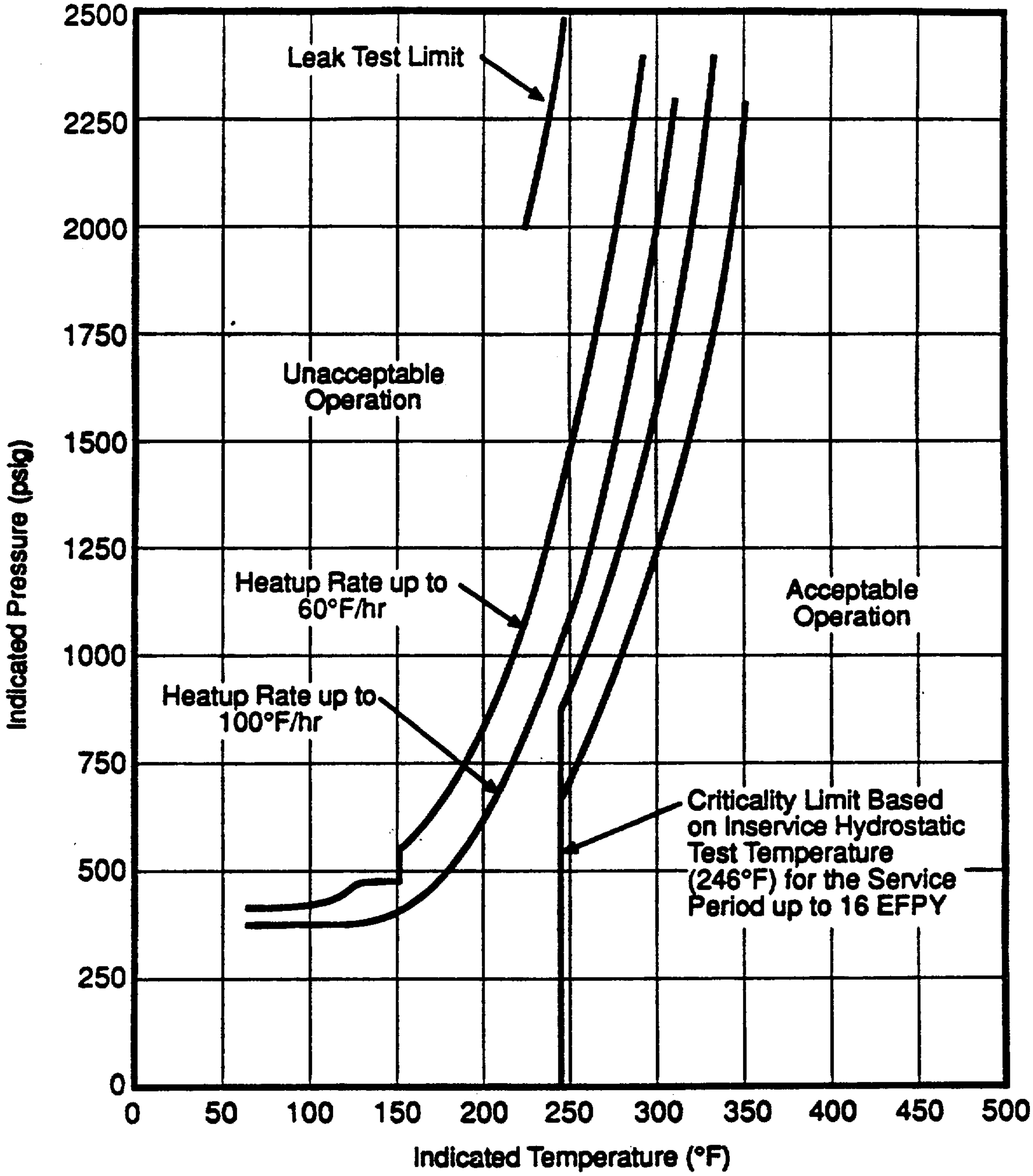
Remove Pages	Insert Pages
Index XVI	Index XVI
3/4 4-31	3/4 4-31
3/4 4-31a	3/4 4-31a
3/4 4-32	3/4 4-32
3/4 4-32a	3/4 4-32a
3/4 4-35	3/4 4-35
3/4 4-35a	3/4 4-35a
B 3/4 4-8	B 3/4 4-8
B 3/4 4-9	B 3/4 4-9
B 3/4 4-9a	B 3/4 4-9a
B 3/4 4-10	B 3/4 4-10
B 3/4 4-11	----
B 3/4 4-12	----
B 3/4 4-13	B 3/4 4-11
B 3/4 4-14	B 3/4 4-12
B 3/4 4-15	B 3/4 4-13
B 3/4 4-16	B 3/4 4-14
B 3/4 4-17	B 3/4 4-15

BASES

SECTION	PAGE

3/4.4 REACTOR COOLANT SYSTEM

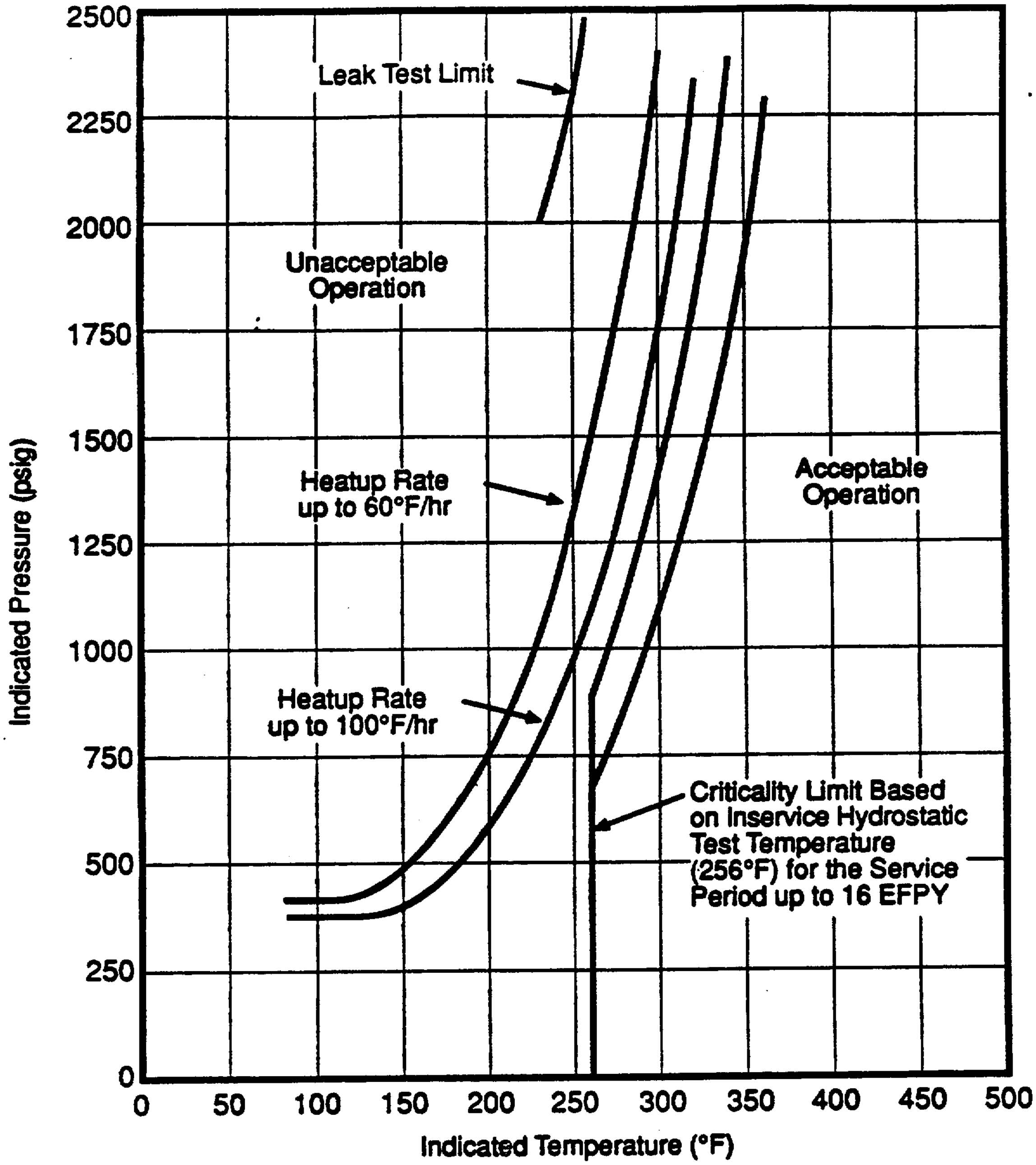
3/4.5 EMERGENCY CORE COOLING SYSTEMS



MATERIAL BASIS

Copper Content:	Assumed - NA WT% (Actual - 0.083 WT%)
RT_{NDT} Initial:	Assumed - NA °F (Actual - 20°F)
RT_{NDT} At 16 EFPY:	@ 1/4T = 100.7°F @ 3/4T = 84.1°F

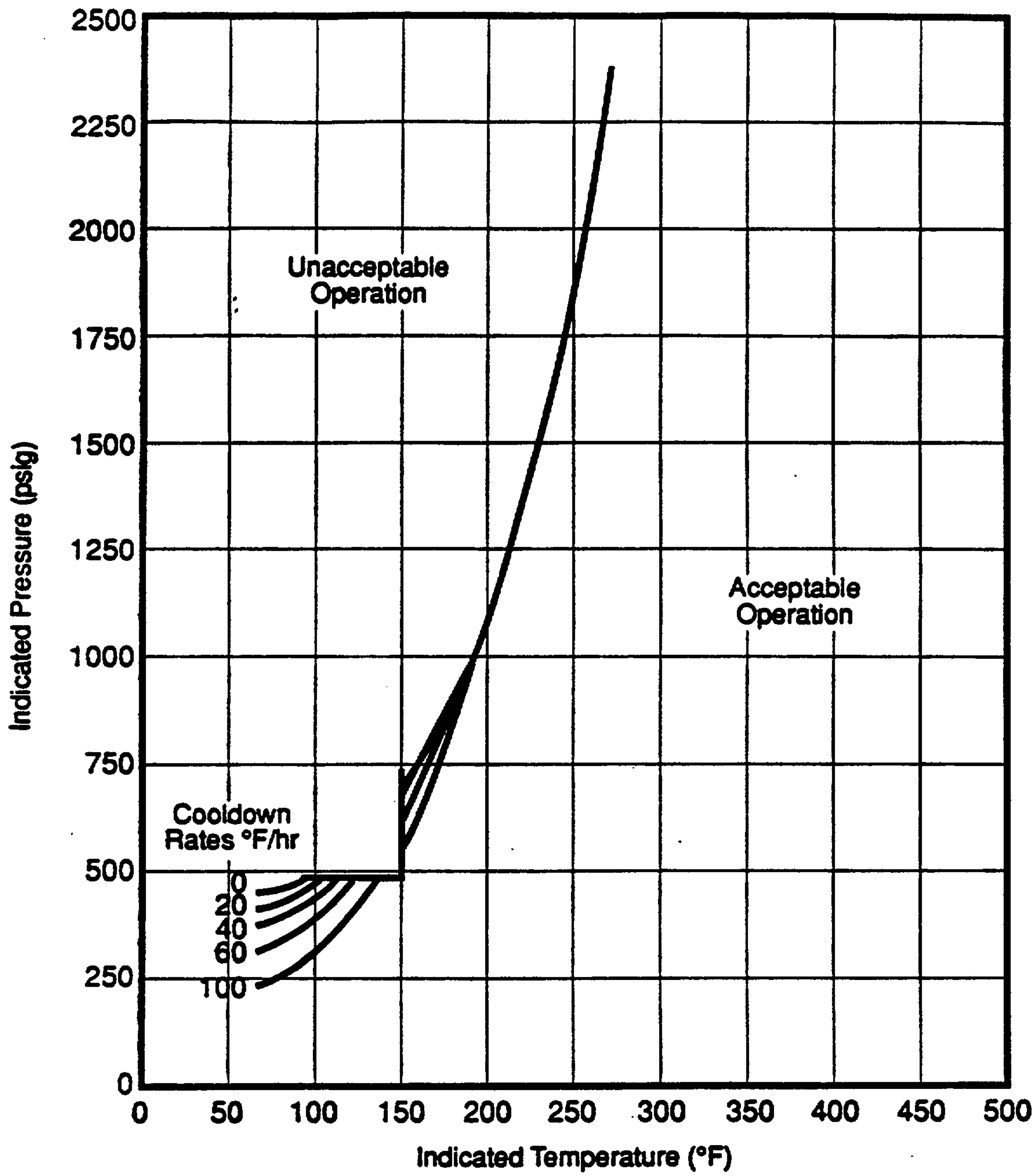
Figure 3.4-2a
Unit 1 Reactor Coolant System Heatup Limitations (Heatup rates up to 100°F/hr) Applicable for the First 16 EFPY (With Margins of 10°F and 60 psig for Instrumentation Errors and Margin of 74 psig for Pressure Difference Between Pressure Instrumentation and Reactor Vessel Beltline Region).



MATERIAL BASIS

Copper Content:	Assumed - NA WT% (Actual - 0.05 WT%)
RT_{NDT} Initial:	Assumed - NA °F (Actual - 50°F)
RT_{NDT} At 16 EFPY:	@ 1/4T = 112°F @ 3/4T = 94°F

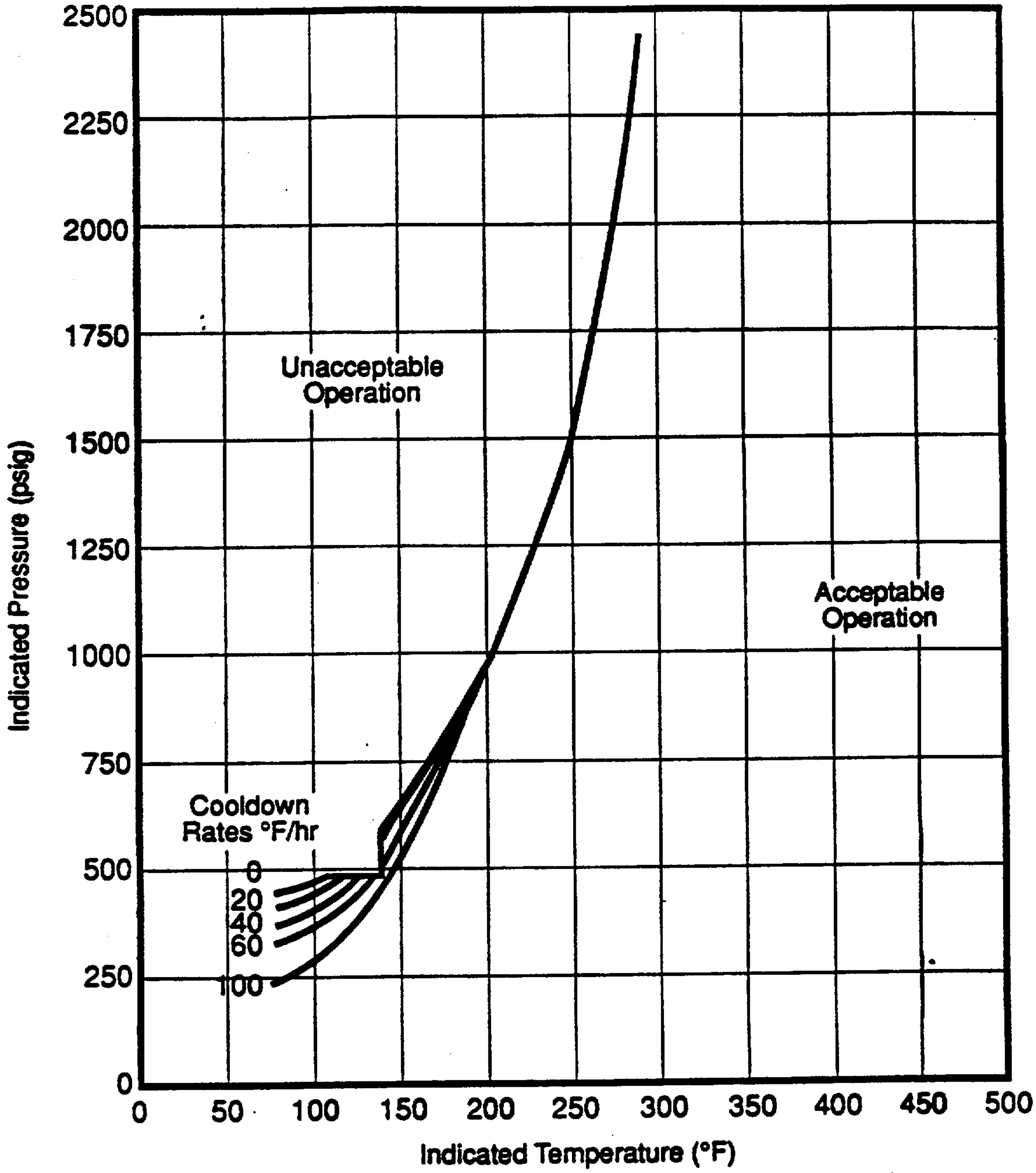
Figure 3.4-2b
Unit 2 Reactor Coolant System Heatup Limitations (Heatup rates up to 100°F/hr) Applicable for the First 16 EFPY (With Margins of 10°F and 60 psig for Instrumentation Errors and Margin of 74 psig for Pressure Difference Between Pressure Instrumentation and Reactor Vessel Beltline Region).



MATERIAL BASIS

Copper Content:	Assumed - NA WT% (Actual - 0.083 WT%)
RT_{NDT} Initial:	Assumed - NA °F (Actual - 20°F)
RT_{NDT} At 16 EFPY:	@ 1/4T = 100.7°F @ 3/4T = 84.1°F

Figure 3.4-3a
Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown rates up to 100°F/hr) Applicable for the First 16 EFPY (With Margins of 10°F and 60 psig for Instrumentation Errors and Margin of 74 psig for Pressure Difference Between Pressure Instrumentation and Reactor Vessel Beltline Region).



MATERIAL BASIS

Copper Content:	Assumed - NA WT% (Actual - 0.05 WT%)
RT_{NDT} Initial:	Assumed - NA °F (Actual - 50°F)
RT_{NDT} At 16 EFPY:	@ 1/4T = 112°F @ 3/4T = 94°F

Figure 3.4-3b
Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown rates up to 100°F/hr) Applicable for the First 16 EFPY (With Margins of 10°F and 60 psig for Instrumentation Errors and Margin of 74 psig for Pressure Difference Between Pressure Instrumentation and Reactor Vessel Beltline Region).

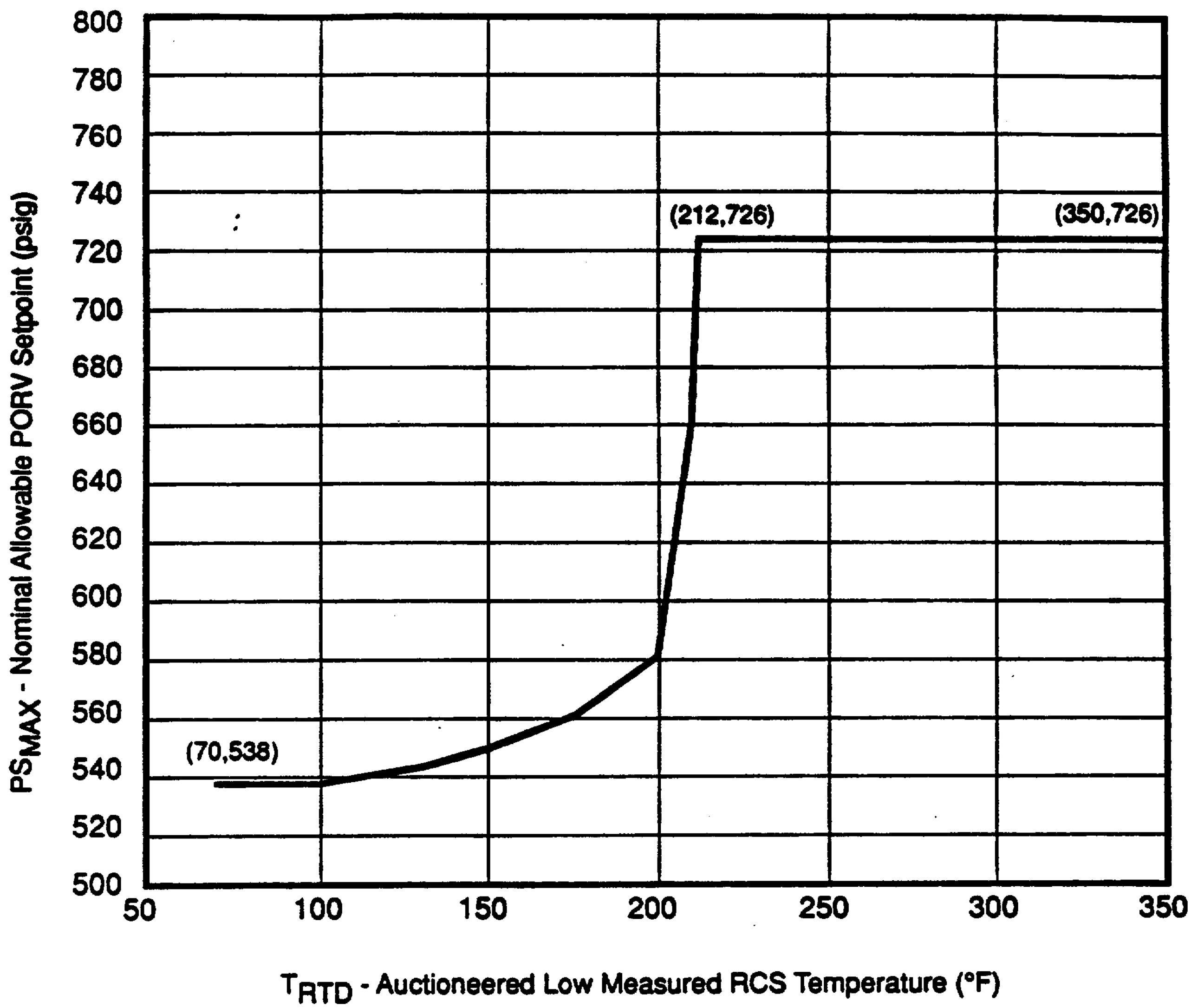


Figure 3.4-4a
Unit 1 Maximum Allowable Nominal PORV Setpoint for the Cold Overpressure Protection System

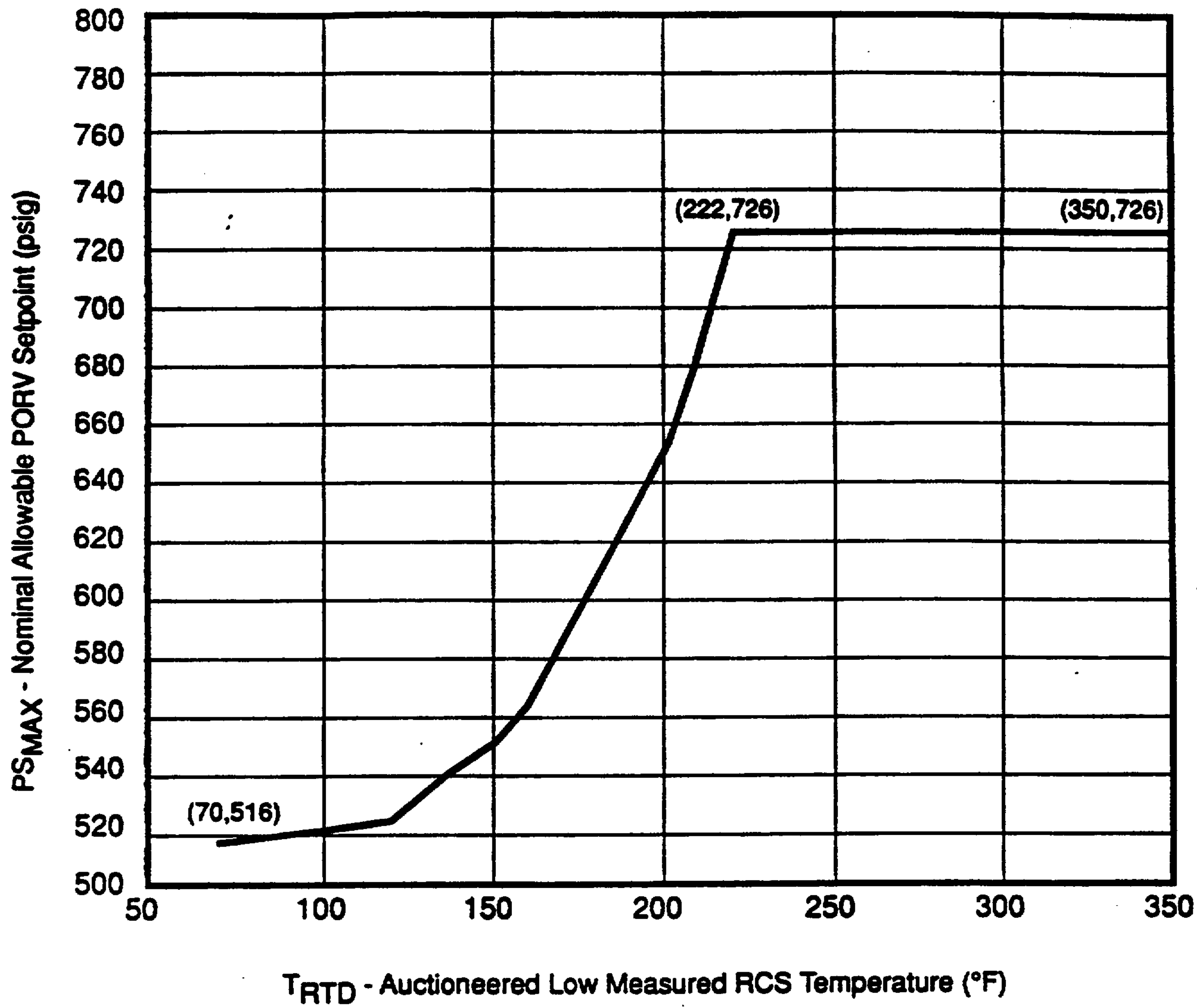


Figure 3.4-4b
Unit 2 Maximum Allowable Nominal PORV Setpoint for the Cold Overpressure Protection System

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 ASME Boiler and Pressure Vessel Code, Section III, Division 1 - Appendices, "Rules for Construction of Nuclear Power Plant Components, Appendix G, Protection Against Nonductile Failure," 1986 Edition 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 for Unit 1 and Figures 3.4-2b and 3.4-3b for Unit 2 are applicable for up to 16 EFPY and were developed based on the actual material properties of the most limiting material. The most limiting material are shown in Table B 3/4.4-1a for Unit 1 and Table B 3/4.4-1b for Unit 2.

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 3/4.4-1a

UNIT 1 REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	CU (%)	NI (%)	INITIAL RT_{NDT}(°F)	16 EFPY RTNDT 1/4-t(°F)	16 EFPY RTNDT 3/4-t(°F)
Closure Head Flange	--	--	0.70	20	---	---
Vessel Flange	--	--	0.71	0	---	---
Intermediate Shell	B8805-1	0.083	0.597	0	80.7	64.1
Intermediate Shell*	B8805-2	0.083	0.610	20	100.7	84.1
Intermediate Shell	B8805-3	0.062	0.598	30	97.5	76.4
Lower Shell	B8606-1	0.053	0.593	20	77.6	57.6
Lower Shell	B8606-2	0.057	0.600	20	81.9	62.5
Lower Shell	B8606-3	0.067	0.623	10	80.8	60.6
Circ. Weld	101-171	0.039	0.102	-80	-21.7	-39.9
Long. Weld	101-124A	0.039	0.102	-80	-31.8	-48.5
Long. Weld	101-124B	0.039	0.102	-80	-30.0	-47.0
Long. Weld	101-124C	0.039	0.102	-80	-30.0	-47.0
Long. Weld	101-142A	0.039	0.102	-80	-30.0	-47.0
Long. Weld	101-142B	0.039	0.102	-80	-31.8	-48.5
Long. Weld	101-142C	0.039	0.102	-80	-30.0	-47.0

* Limiting material

TABLE B 3/4.4-1b

UNIT 2 REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	CU (%)	NI (%)	INITIAL RT_{NDT} (°F)	16 EFPY RTNDT 1/4-t(°F)	3/4-t(°F)
Closure Head Flange	--	--	0.72	10	--	--
Vessel Flange	--	--	0.87	-60	--	--
Intermediate Shell	R4-1	0.06	0.64	10	81	62
Intermediate Shell	R4-2	0.05	0.62	10	72	54
Intermediate Shell	R4-3	0.05	0.59	30	92	74
Lower Shell	B8825-1	0.05	0.59	40	102	84
Lower Shell	R8-1	0.06	0.62	40	111	92
Lower Shell*	B8628-1	0.05	0.59	50	112	94
Circ. Weld	--	0.06	0.12	-30	55	31
Long. Weld	--	0.07	0.13	-10	83	56

* 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 Tables 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 nickel content of the material in question, can be predicted using Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." 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 16 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments of 60 psig and 10°F, respectively. In addition, these curves include a pressure adjustment of 74 psig to account for the pressure differential between the wide range pressure transmitter and the belt line region.

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

3/4.4.10 STRUCTURAL INTEGRITY (Continued)

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through the 1975 Winter Addenda.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of at least one Reactor Coolant System vent path from the reactor vessel head, ensures that the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

PRESSURE/TEMPERATURE LIMITS (Continued)

calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \quad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{IT} \leq K_{IR} \quad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{IT} = the stress intensity factor caused by the thermal gradients,

K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of

BASES

PRESSURE/TEMPERATURE LIMITS (Continued)

the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{IT}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

BASES

PRESSURE/TEMPERATURE LIMITS (Continued)

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Next, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Finally, the new 10 CFR 50 Appendix G Rule which addresses the metal temperature of the closure head flange and vessel flange regions is considered. This rule states that the minimum metal temperature of the closure flange regions should be at least 120°F higher than the limiting RT_{NDT} for these regions when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Unit 1). For Unit 1 the minimum temperature of the closure flange and vessel flange regions is 140°F, since the limiting RT_{NDT} is 20°F (see Table B 3/4-4.1a). For Unit 2, the minimum temperature of the closure flange and vessel flange regions is 130°F, since the limiting RT_{NDT} is 10°F (Table B 3/4-1b). These values include margin of 10°F and 60 psig for instrumentation errors. The heatup and cooldown curves as shown in Figures 3-4.2a and 3-4.3a for Unit 1 and the heatup and cooldown curves as shown in Figures 3-4.2b and 3-4.3b for Unit 2 are impacted by the new 10 CFR 50 rule.

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

COLD OVERPRESSURE PROTECTION SYSTEMS

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

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

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

3/4.4.10 STRUCTURAL INTEGRITY

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NPF-68

AND AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NPF-81

GEORGIA POWER COMPANY, ET AL.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

1.0 INTRODUCTION

By letter dated October 3, 1994, as supplemented by letter dated March 1, 1995, Georgia Power Company, et al. (GPC or the licensee) proposed license amendments to change the Technical Specifications (TS) for Vogtle Electric Generating Plant (VEGP), Units 1 and 2. The proposed changes would revise the pressure-temperature (P-T) limits for both units. The March 1, 1995, letter provided supporting technical data that did not change the scope of the October 3, 1994, application and initial no proposed significant hazards consideration determination. Based on the results of recent surveillance capsule Y examinations, the applicable period of the P-T limits would be extended from a current 13 effective full power years (EFPY) to 16 EFPY for Unit 1. The period for the Unit 2 P-T limits would remain at 16 EFPY. The revised P-T limits include an adjustment in the calculations to account for the pressure difference between the pressure transmitter and the reactor vessel midplane.

The proposed revisions would replace the current reactor vessel heatup and cooldown curves (TS Figures 3.4-2a through 3.4-3b) with revised curves and would modify the pressure setpoint curves (TS Figures 3.4-4a and 3.4-4b) for the low-temperature, overpressure protection system (LTOPS). In addition, these revisions reflect actions taken by the licensee in response to NRC Information Notice (IN) 93-58, "Non-conservatism in Low-Temperature Overpressure Protection for Pressurized-Water Reactors," and the use of an updated methodology to determine applicable limits in accordance with the provisions of Appendix G to 10 CFR Part 50.

In addition to Appendix G to 10 CFR Part 50, the staff evaluated the P-T limits based on the following NRC regulations and guidance: Generic Letters (GLs) 88-11 and 92-01; Regulatory Guide (RG) 1.99, Rev. 2; and Standard Review Plan (SRP) Section 5.3.2. Appendix G to 10 CFR Part 50 requires that P-T limits for the reactor vessel must be at least as conservative as those obtained by Appendix G to Section III of the American Society of Mechanical Engineers (ASME) Code. GL 88-11 provides that licensees may use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation by calculating adjusted reference temperature (ART) of reactor vessel materials. The ART is defined as the sum of initial nil-ductility transition reference temperature (RT_{ndt}) of the material, the increase in RT_{ndt} caused by neutron irradiation, and a margin to account for uncertainties in the prediction method. The increase in RT_{ndt} is calculated from the product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the vessel material. GL 92-01 requires licensees to submit reactor vessel materials data for staff evaluation during its review of licensing actions related to P-T limits.

SRP 5.3.2 provides guidance on calculation of the P-T limits using linear elastic fracture mechanics methodology specified in Appendix G to Section III of the ASME Code. The linear elastic fracture mechanics methodology postulates sharp surface defects that are normal to the direction of maximum stress and have a depth of one-fourth of the reactor vessel beltline thickness (1/4T) and a length of 1-1/2 times the beltline thickness. The critical locations in the vessel for this methodology is the 1/4T and 3/4T locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

In addition, IN 93-58 alerted licensees of a potential non-conservatism associated with the LTOPS setpoint calculations for Westinghouse facilities. This non-conservatism is the result of measured reactor coolant system (RCS) pressure (measured in the hot leg) being lower than the pressure at the location where the reactor vessel is most susceptible to non-ductile failure (typically at the mid-core level). This pressure difference is attributed to non-recoverable (friction and minor) losses and fluid acceleration between these two locations and the hydrostatic head due to corresponding elevation differences. Consequently, the actual margin between the peak pressure occurring during the limiting design-basis event and the Appendix G pressure limit will be less than the corresponding value based on measured RCS pressure. The potential for exceeding Appendix G limits therefore exists.

2.0 EVALUATION

For the Unit 1 reactor vessel, the licensee determined that intermediate shell, B8805-2, is the limiting material for both the 1/4T and 3/4T locations. The licensee calculated an ART of 100.7°F at the 1/4T location and 84.1°F at the 3/4T location. For the Unit 2 reactor vessel, the licensee determined that lower shell, B8825-1, is the limiting material for both the 1/4T and 3/4T locations. The licensee calculated an ART of 112°F at the 1/4T location and an ART of 94°F at the 3/4T location.

The staff verified that the copper and nickel contents and the initial RT_{ndt} of the Unit 1 and Unit 2 reactor vessel materials agreed with those in the licensee's response to GL 92-01 for Units 1 and 2. However, the nickel content for Unit 1 intermediate shell, B8805-5, increased from 0.59% to 0.61% from the GL 92-01 submittal to this submittal. The increase in nickel results in a more conservative restrictive ART value. Therefore, the change is acceptable. The staff used the material properties to perform an independent calculation of the ART values for the limiting materials using RG 1.99, Revision 2. Based on the staff's calculation, the staff verified that the licensee's calculated ARTs for Units 1 and 2 are acceptable.

Substituting the ARTs of Units 1 and 2 limiting materials into equations in SRP 5.3.2, the staff verified that the proposed Units 1 and 2 P-T limits for heatup, cooldown, criticality, and inservice hydrostatic test satisfy the requirements in Paragraphs IV.A.2 and IV.A.3 of Appendix G to 10 CFR Part 50.

In addition to beltline materials, Appendix G also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.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 RT_{ndt} of 20°F for Unit 1 and 10°F for Unit 2 provided by the licensee, the staff has determined that the proposed P-T limits satisfy the requirement for the closure flange region during normal operation, hydrostatic pressure test and leak test.

To address the non-conservatism identified in IN 93-58, the licensee has revised TS Figures 3.4-2a through 3.4-3b to include a 74 psi adjustment to the Appendix G pressure limits. This value is the sum of a 4 psi hydrostatic head between the mid-core and hot leg centerline elevations, a 59.1 psi pressure drop due to non-recoverable losses, and a 9.7 psi pressure drop due to fluid acceleration. Pressure drops were calculated between the downcomer (at mid-core level) and the residual heat removal suction line connection to the hot leg (where the wide range pressure transmitter is located). Operation of all four reactor coolant pumps was assumed in this calculation, whereas the TS permit a maximum of three pumps to operate during RCS heatup or cooldown. Accordingly, the calculated pressure drops represent overestimates and the 74 psi adjustment to the P-T operating limits is conservative. We find that the issue identified in IN 93-58 has been adequately addressed by the licensee.

The maximum allowable LTOPS setpoints (Figures 3.4-4a and b) are intended to prevent either the Appendix G limits or an 800 psig limit on the PORV discharge piping (whichever is smaller) from being exceeded during the limiting design basis transients. For purposes of LTOPS setpoint determination, these transients are the mass injection and heat injection events. In generating the revised setpoint curves, the licensee employed P-T limits which differ in several respects from the revised Appendix G curves (Figures 3.4-2a through 3.4-3b) used as plant operating limits. On the basis of ASME Code Case N-514, a 10 percent relaxation of the pressure limit has been applied to the revised Appendix G curves for RCS temperatures up to 200°F. Above temperatures of 140°F and 150°F for Units 1 and 2, respectively, the 800 psig limit on the power-operated relief valve piping becomes limiting. Additionally, the pressure and temperature instrument uncertainties of 60 psi and 10°F incorporated in Figures 3.4-2a through 3.4-3b are not included in the P-T limits used to generate the LTOPS setpoints. Instead, a 27°F uncertainty is applied to these limits when the maximum overpressure is computed for the mass injection event and a 77°F uncertainty (including 50°F for thermal transport) is applied for the heat injection event computation. However, no pressure instrument uncertainty has been applied to these limits. The effect of instrument uncertainty, in general, is considered insignificant in light of the very conservative assumptions used in the development of the Appendix G limits. As such, neither Appendix G nor the ASME Code require that margins for instrument uncertainty be incorporated into the P-T limits. The inclusion of this uncertainty is not needed for ensuring that the operating limits are conservative. The licensee's supplementary submittal dated March 1, 1995, provides data which indicate that margin exists between the revised maximum allowable pressure and the peak pressure attained during the limiting design basis transient for each of the revised LTOPS setpoints. We therefore find the licensee's approach to the development of revised LTOPS setpoints to be acceptable.

Based on an independent analysis to verify the licensee's proposed P-T limits for Units 1 and 2, the staff concludes that the proposed P-T limits for heatup, cooldown, inservice hydrostatic test and criticality are valid for 16 EFPY because: 1) the limits conform to the requirements of Appendix G to 10 CFR Part 50 and GL 88-11; and 2) the material properties and chemistry used in calculating the P-T limits are consistent with or conservative compared to data submitted under GL 92-01. Therefore, the proposed P-T limits for Units 1 and 2 may be incorporated in the VEGP Units 1 and 2 TS. In addition, the proposed editorial changes in the Bases section of the TS are consistent with the P-T limits changes and therefore are acceptable.

Based on the above evaluation, we also find that the issue identified in IN 93-58 has been adequately addressed by the licensee and that the revised LTOPS setpoints (TS Figures 3.4-4a and 3.4-4b) are acceptable.

3.0 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.

4.0 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