January 10, 1992

Docket Nos. 50-321
and 50-366

Distribution
See next page

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. 177 TO FACILITY OPERATING LICENSE DPR-57 AND AMENDMENT NO. 118 TO FACILITY OPERATING LICENSE NPF-5 - EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 (TACs M71501, M71502, M81149 and M81152)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 177 to Facility Operating License No. DPR-57 and Amendment No. 118 to Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated July 15, 1991.

The amendments revise Hatch Unit 1 Bases Section 3.6.B, "Reactor Vessel Temperature and Pressure," and Hatch Unit 2 TS 3/4.4.6, "Pressure/Temperature Limits," and its associated Bases.

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:

Kahtan N. Jabbour, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 177 to DPR-57
2. Amendment No. 118 to NPF-5
3. Safety Evaluation

cc w/enclosures:
See next page

as modified

OFC	PDII-3/LA	PDII-3/PM	OGC	PDII-3/D
NAME	LBerry	KJabbour	J Hull	~~DMatthews~~ D. Lange
DATE	12/6/91	12/9/91	12/26/91	1/10/92

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DATED: JANUARY 10, 1992

AMENDMENT NO. 177 TO FACILITY OPERATING LICENSE DPR-57 - Hatch Nuclear Plant, Unit 1
AMENDMENT NO. 118 TO FACILITY OPERATING LICENSE NPF-5 - Hatch Nuclear Plant, Unit 2

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

January 10, 1992

Docket Nos. 50-321
and 50-366

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Senior Vice President -
Nuclear Operations
Georgia Power Company
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3. Safety Evaluation

cc w/enclosures:
See next page

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Georgia Power Company

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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

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 177
License No. DPR-57

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 1 (the facility) Facility Operating License No. DPR-57 filed by Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees) dated July 15, 1991, 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. DPR-57 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 177, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Lange

David J. Lange, Acting Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: January 10, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 177

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

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

Remove Page	Insert Page
3.6-16	3.6-16

Test pressures for inservice hydrostatic and leak testing required by the ASME B&PV Code, Section XI, are a function of testing temperature and component material. For the Hatch 1 reactor pressure vessel, the ISI hydrostatic test pressure would be approximately 1.1 times operating pressure, or about 1106 psig, depending on the reactor water temperature. The temperatures for pressures above 440 psig are determined by the RPV core beltline with a shift in RT_{NDT} of 123°F, appropriate for operation up to 16 effective full power years (EFPY).

Figure 3.6-2 provides appropriate limitations for plant heatup and cooldown when the reactor is not critical. Figure 3.6-2 is also applicable to low power physics tests. These curves assume heatup and cooldown rates up to 100°F per hour. Temperatures for pressures above 300 psig represent the limits of the RPV core beltline with a shift in RT_{NDT} of 123°F, appropriate for 16 EFPY of operation.

Figure 3.6-3 establishes operating limits when the core is critical. Figure 3.6-3 is not applicable to low power physics tests. These limits include a margin of 40°F as required by 10CFR50 Appendix G. In accordance with the May 1983 revision of 10CFR50 Appendix G, core critical operation may be initiated at temperatures at or above (RT_{NDT} + 60°F) of the closure flange region, or 76°F. Temperatures for pressures above 300 psig represent the limits of the RPV core beltline with a RT_{NDT} shift of 123°F, appropriate for 16 EFPY of operation.

During inservice hydrostatic or leak testing, Figure 3.6-2 is used for nonnuclear heatup until the test temperature is achieved. After the test temperature is reached, performance of the inservice hydrostatic or leak test is governed by Figure 3.6-1. After test completion, vessel cooldown is in accordance with Figure 3.6-2.

The fracture toughness of all ferritic steels gradually and uniformly decreases with exposure to fast neutrons above a threshold value, and it is prudent and conservative to account for this in the operation of the RPV. Two types of information are needed in this analysis: (a) a relationship between the change in fracture toughness of the RPV steel and the neutron fluence (integrated neutron flux); and (b) a measure of the neutron fluence at the point of interest in the RPV wall. A method of relating shift in RT_{NDT} to accumulated fast nuetron (>1 MeV) fluence is contained in Regulatory Guide 1.99, Revision 1. Experimental results of irradiated surveillance specimens taken from the RPV show a shift in RT_{NDT} greater than predicted by Regulatory Guide 1.99, so the surveillance results were used with the methods of 1.99 to establish the RT_{NDT} shift. The shift for 16 EFPY was added to the unirradiated RPV core beltline curves, resulting in the beltline being the limiting region in the vessel for higher pressure-temperature conditions.

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 118
License No. NPF-5

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 2 (the facility) Facility Operating License No. NPF-5 filed by Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees) dated July 15, 1991, 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-5 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 118, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Lange

David J. Lange, Acting Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: January 10, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 118

FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

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
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
3/4 4-15	3/4 4-15
3/4 4-16	3/4 4-16
3/4 4-17	---
B 3/4 4-4	B 3/4 4-4
B 3/4 4-5	B 3/4 4-5
B 3/4 4-6	B 3/4 4-6

REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and reactor vessel pressure shall be limited in accordance with the limit lines shown on (1) Figure 3.4.6.1-1 for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; (2) Figure 3.4.6.1-2 for operations with a critical core other than low power PHYSICS TESTS; and (3) Figure 3.4.6.1-3 for inservice hydrostatic or leak testing, with:

a. A maximum heatup of 100°F in any one hour period, and

b. A maximum cooldown of 100°F in any one hour period.

APPLICABILITY: At all times.

ACTION:

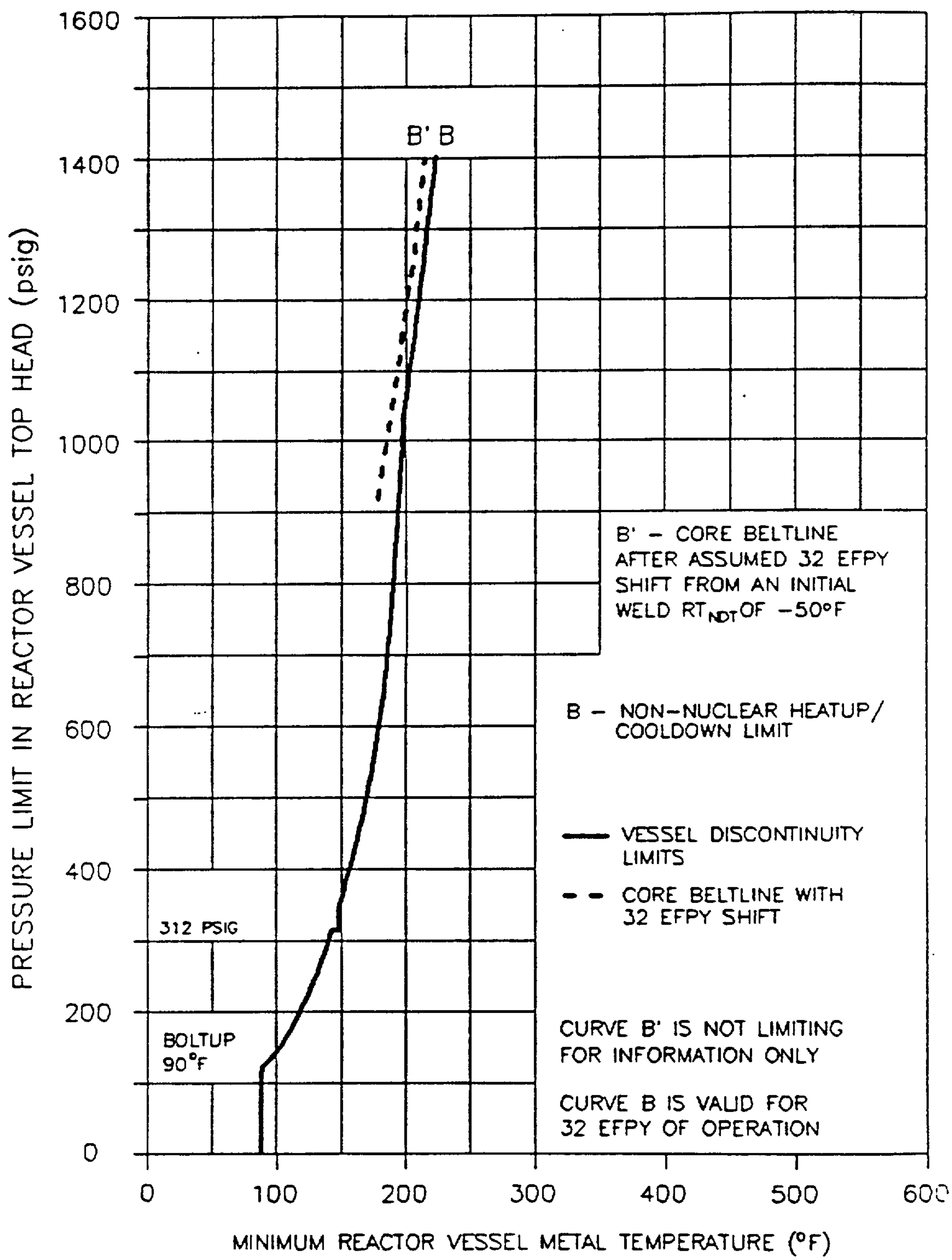
With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

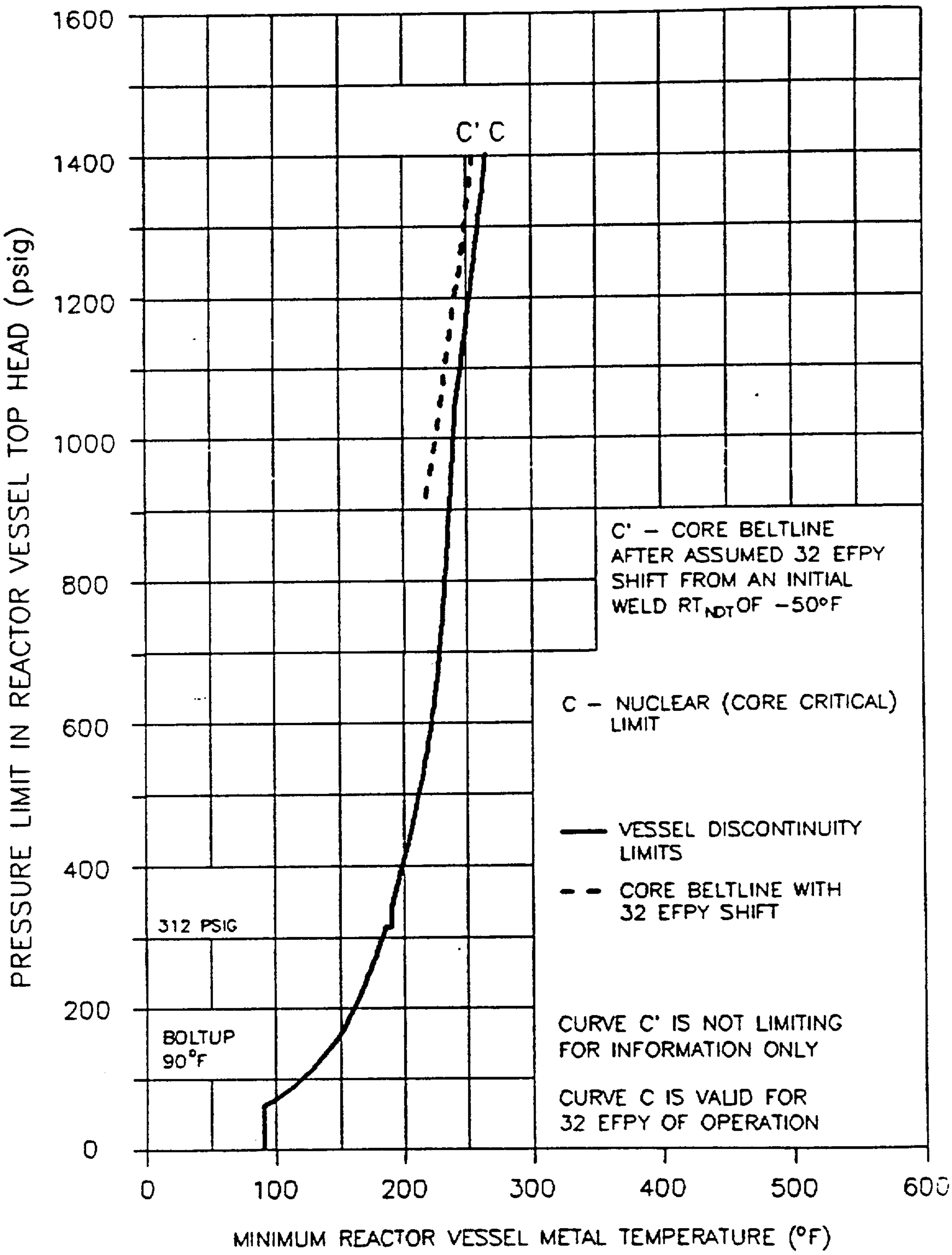
4.4.6.1.1 The reactor coolant system temperature and reactor vessel pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown and inservice leak and hydrostatic testing operations.

4.4.6.1.2 The reactor coolant system temperature and reactor vessel pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-2 within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality.

4.4.6.1.3 The reactor material irradiation surveillance specimens shall be removed and examined to determine changes in material properties, as required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update Figures 3.4.6.1-1, 3.4.6.1-2 and 3.4.6.1-3.

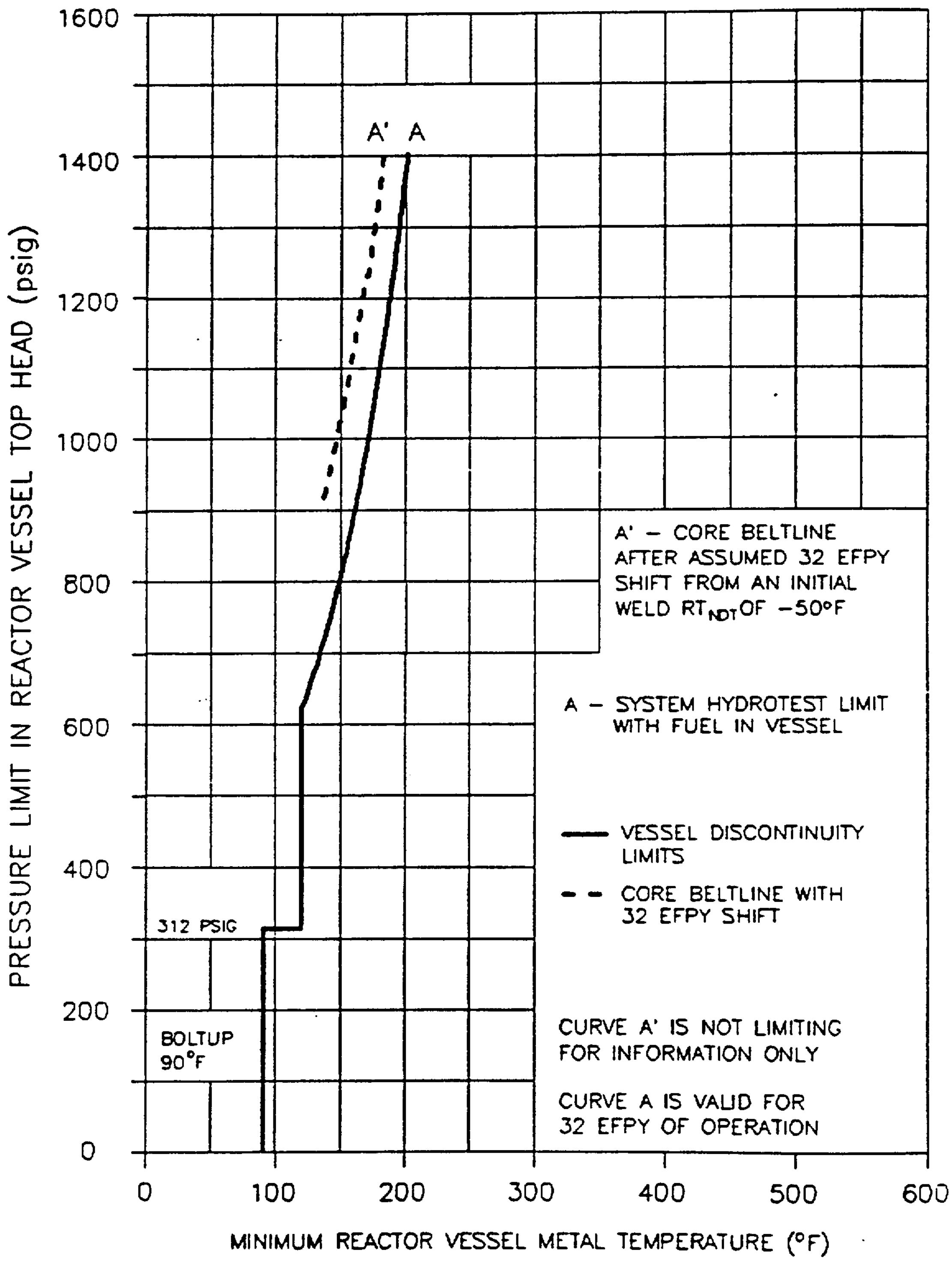


TEMPERATURE PRESSURE LIMITS FOR NON-NUCLEAR HEATUP, LOW POWER PHYSICS TESTS AND COOLDOWN FOLLOWING A SHUTDOWN
FIGURE 3.4.6.1-1



TEMPERATURE-PRESSURE LIMITS FOR CRITICALITY (INCLUDES ADDITIONAL 40°F MARGIN REQ'D BY 10CFR50, APP-G)

FIGURE 3.4.6.1-2



TEMPERATURE-PRESSURE LIMITS FOR INSERVICE HYDROSTATIC TEST
FIGURE 3.4.6.1-3

TABLE 4.4.6.1.3-1

Deleted

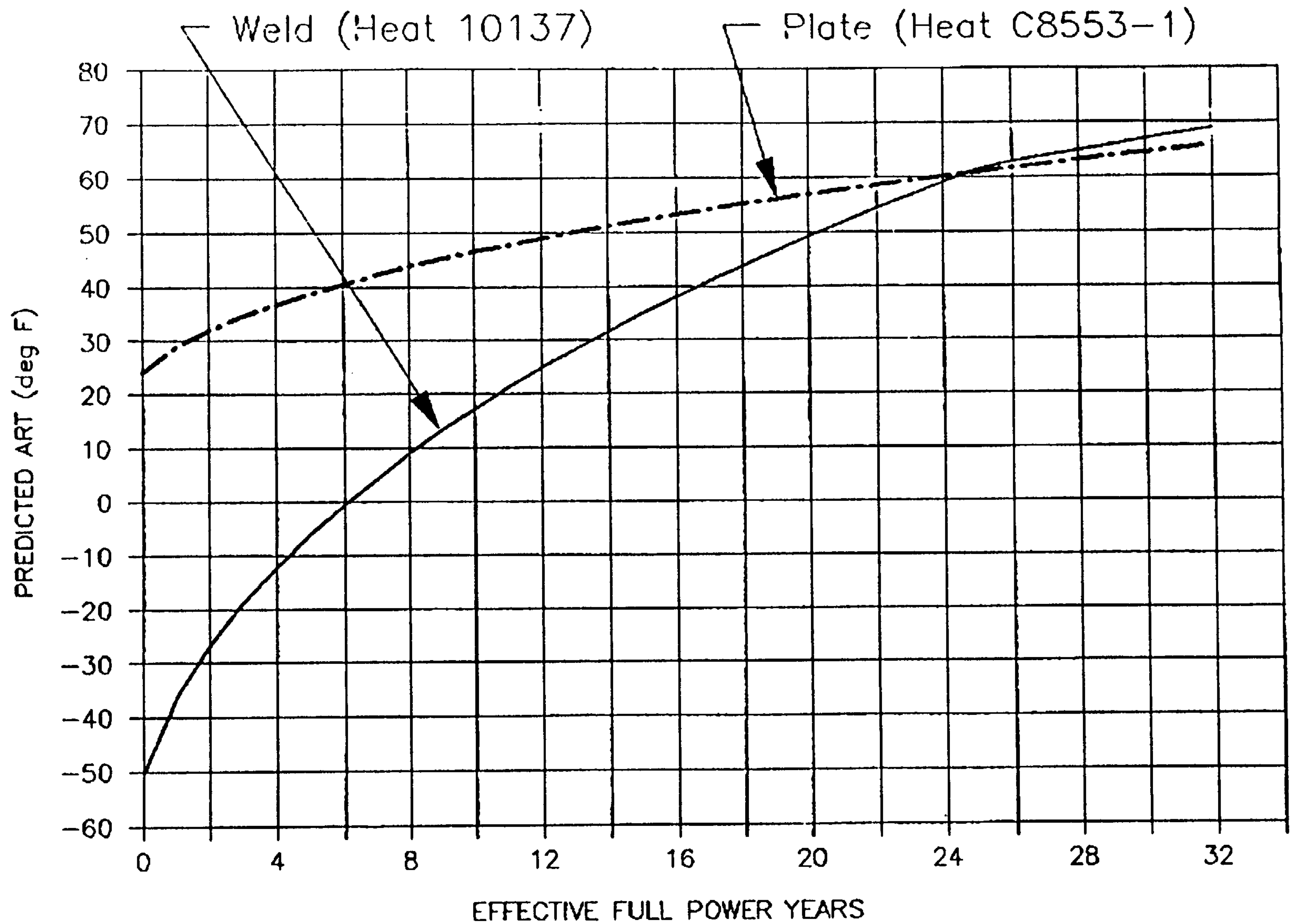
3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup or cooldown, the thermal gradients in the reactor vessel wall produce thermal stresses. During heatup, the stresses are compressive in the inner wall and tensile in the outer wall. During cooldown, the opposite is the case. Thus, the one quarter of the material thickness (1/4 T) location is critical during cooldown and the 3/4 T location is critical during heatup. However, a conservative simplification is made by applying the absolute value of the thermal stress at the 1/4 T location, where the Reference Temperature-Nil-ductility temperature (RT_{NDT}) is the highest. In this manner, a single curve is generated for both the heatup and cooldown conditions.

The reactor vessel materials have been tested, and analyses performed to determine their initial RT_{NDT} values. Reactor operation and resultant fast neutron (E>1 MeV) irradiation will cause an increase in the RT_{NDT} values of the beltline materials. Adjusted reference temperatures, based upon the fluence, have been predicted using Regulatory Guide 1.99, Revision 2, and the beltline material test results. The adjusted reference temperature for the most limiting beltline material is plotted versus EFPY of operation in Bases Figure B 3/4.4.6-1. The pressure/temperature limit curves, shown in Figures 3.4.6.1-1 through 3.4.6.1-3, include predicted adjustments in RT_{NDT} for 32 EFPY of operation. Comparison with the nonbeltline curves shows the nonbeltline curves are limiting for 32 EFPY of operation.

During inservice hydrostatic or leak testing, Figure 3.4.6.1-1 is used for nonnuclear heatup until the test temperature is achieved. After the test temperature is reached, performance of the inservice hydrostatic or leak test is governed by Figure 3.4.6.1-3. After test completion, vessel cooldown is in accordance with Figure 3.4.6.1-1.



Bases Figure B 3/4.4.6-1
Adjusted Reference Temperature for Limiting Beltline Materials

PRESSURE/TEMPERATURE LIMITS (Continued)

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown in Figures 3.4.6.1-2 and 3.4.6.1-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Unit 2 FSAR Section 5.2. The removal of the specimens meets the requirements of Appendix H to 10 CFR Part 50.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 177 TO FACILITY OPERATING LICENSE DPR-57

AND AMENDMENT NO. 118 TO FACILITY OPERATING LICENSE NPF-5

GEORGIA POWER COMPANY, ET AL.

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

By letter dated July 15, 1991, the Georgia Power Company, et al. (the licensee), submitted a request for changes to the Edwin I. Hatch Nuclear Plant, Units 1 and 2, Technical Specifications (TS). The requested changes would revise the pressure/temperature (P/T) limits in the Hatch 2 TS 3/4.4.6, "Reactor Vessel Temperature and Pressure." In addition, the licensee requested: (1) to revise Unit 2 Bases Section 3/4.4.6 to reflect the changes in TS 3/4.4.6 and to include a brief description of the use of revised TS curves during inservice hydrostatic leakage testing, and (2) to add this brief description to Unit 1 Bases Section 3.6.B. The proposed P/T limits were requested for 32 effective full power years (EFPY). The proposed P/T limits were developed using Regulatory Guide (RG) 1.99, Revision 2, Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," which recommends RG 1.99, Revision 2, be used in calculating P/T limits, unless the use of different methods can be justified. The P/T limits provide for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff uses 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, Revision 2; Standard Review Plan (SRP) Section 5.3.2; and GL 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide TS for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the TS. The P/T limits are among the limiting conditions of operation in the TS for all commercial nuclear plants in the United States. 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 use the methods in RG 1.99, Revision 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.

The licensee also proposed to delete the withdrawal schedule of the RPV surveillance capsules from the Unit 2 TS in accordance with NRC GL 91-01. GL 91-01 allows the removal of the surveillance capsule withdrawal schedule from the TS but requires the schedule be incorporated in the Final Safety Analysis Report (FSAR). The licensee should incorporate the schedule in TS Table 4.4.6.1.3-1 into the updated Hatch Unit 2 FSAR, in accordance with GL 91-01. We conclude this proposal is acceptable.

2.0 EVALUATION

The NRC staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Hatch 2 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Revision 2. The staff determined that the material with the highest ART at 32 EFPY at 1/4T (T = reactor vessel beltline thickness) was the lower longitudinal weld, 101-842, with 0.23% copper (Cu), 0.50% nickel (Ni), and an initial RT_{ndt} of -50°F. At 3/4T, the limiting material at 32 EFPY was plate C8553-1 with 0.08% copper (Cu), 0.58% nickel (Ni), and an initial RT_{ndt} of 24°F.

The licensee has removed surveillance capsule No. 3 from Hatch 2. The results from capsule No. 3 were published in General Electric Report SASR 90-104. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, weld 101-842, the staff calculated the ART to be 68.8°F at 1/4T. For plate C8553-1, the ART was 51.9°F for 3/4T. The staff used a neutron fluence of 1E18 n/cm² at 1/4T and 5.2E17 n/cm² at 3/4T. The ART was determined by Section 1 of RG 1.99, Revision 2, because the licensee has removed only one surveillance capsule from the Hatch 2 reactor vessel.

The licensee used the method in RG 1.99, Revision 2, to calculate an ART of 69°F at 32 EFPY at 1/4T for the same limiting weld metal. The licensee's ART 69°F is more conservative than the staff's ART of 68.8°F and, therefore, is acceptable. Substituting the ART of 68.8°F 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 pre-service 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. Paragraph IV.A.3 of Appendix G states "an exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the pre-service system hydrostatic test pressure. In this case, the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload." Based on the flange reference temperature of 10°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.A of Appendix G requires that the predicted Charpy USE at end of life (EOL) be above 50 ft-lb. The material with the lowest initial USE is the lower intermediate shell plate C8579-2 with 70 ft-lb. Based on Figure 2 of RG 1.99, Revision 2, the staff predicted the USE at EOL to be 61.9 ft-lb. This is greater than 50 ft-lb. and, therefore, is 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 32 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The proposed P/T limits also satisfy GL 88-11 because the method in RG 1.99, Revision 2 was used to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Hatch 2 TS.

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 and change surveillance requirements. 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 (56 FR 60117). 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.

5.0 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.

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REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988

2. NUREG-0800, Standard Review Plan, Section 5.3.2: Pressure-Temperature Limits

3. July 15, 1991, Letter from J. T. Beckham (GP) to USNRC Document Control Desk, Subject: Plant Hatch - Units 1 and 2, NRC Dockets 50-321 and 50-366, Operating Licenses DPR-57 and NPF-5, Request to Revise Technical Specifications: Reactor Vessel Temperature and Pressure Limits

4. T. A. Caine, "E. I. Hatch Nuclear Power Station, Unit 2 Vessel Surveillance Materials Testing and Fracture Toughness Analysis,"SASR 90-104, General Electric Company, May 1991