DCR 016

June 20, 1986

Docket No. 50-321

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Mr. J. T. Beckham, Jr.
Vice President - Nuclear Generation
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Beckham:

The Commission has issued Amendment No. 126 to Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated January 7, 1986.

The amendment revises the TSs to change the reactor vessel operating temperature and pressure limits and to make associated editorial changes.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next Bi-Weekly Federal Register Notice.

Sincerely,

/S/

George W. Rivenbark, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

Enclosures:
1. Amendment No. 126
2. Safety Evaluation

cc w/enclosures
See next page

BWR#2	BWR#2	BWR#2	OELD
for SNorris	GRivenbark:dd	DMuller	
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Mr. J. T. Beckham, Jr.
Georgia Power Company

Edwin I. Hatch Nuclear Plant,
Units Nos. 1 and 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

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 126
License No. DPR-57

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

   A. The application for amendment by Georgia Power Company, et al., (the licensee) dated January 7, 1986 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;

   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 amended by 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. 126 , 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 and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 20, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 126

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

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 area of cahnge.

Remove	Insert
x	x
3.6-1	3.6-1
3.6-2	3.6-2
3.6-15	3.6-15
3.6-16	3.6-16
3.6-17	3.6-17
Figure 3.6-1	Figure 3.6-1
Figure 3.6-2	Figure 3.6-2
Figure 3.6-3	Figure 3.6-3
Figure 3.6-4	(Blank Page)

LIST OF FIGURES

3.6 PRIMARY SYSTEM BOUNDARY

Applicability

The Limiting Conditions for Operation apply to the operating status of the reactor coolant system.

Objective

The objective of the Limiting Conditions for Operation is to assure the integrity and safe operation of the reactor coolant system.

Specifications

A. Reactor Coolant Heat-Up and Cooldown

The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period.

B. Reactor Vessel Temperature and Pressure

1. The reactor vessel shell temperatures during inservice hydrostatic or leak testing shall be at or above the temperatures shown on the curve of Figure 3.6-1.

4.6 PRIMARY SYSTEM BOUNDARY

Applicability

The Surveillance Requirements apply to the periodic examination and testing requirements for the reactor coolant system.

Objective

The objective of the Surveillance Requirements is to determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specifications

A. Reactor Coolant Heat-Up and Cooldown

The reactor coolant system temperature and pressure shall be determined to be within the limits of Specifications 3.6.A. and 3.6.B. at least once every 30 minutes during reactor coolant heatup and cooldown.

B. Reactor Vessel Temperature and Pressure

Reactor vessel metal temperature at the outside surface of the bottom head in the vicinity of the control rod drive housing and reactor vessel shell adjacent to shell flange shall be recorded at least every 15 minutes during in-service hydrostatic or leak testing when the vessel pressure is $\geq$312 psig.

3.6.B. Reactor Vessel Temperature and Pressure (Continued)

2. During heatup by non-nuclear means, cooldown following nuclear shutdown or low level physics tests, the reactor vessel shell and fluid temperatures of Specification 4.6.A. shall be at or above the temperatures shown on the curve of Figure 3.6-2.

3. During all operation with a critical core, other than for low level physics tests, the reactor vessel shell and fluid temperatures of Specification 4.6.A. shall be at or above the temperatures shown on the curve of Figure 3.6-3.

4.6.B. Reactor Vessel Temperature and Pressure (Continued)

Test specimens representing the reactor vessel, base weld and weld heat affected zone metal were installed in the reactor vessel adjacent to the vessel wall at the core midplane level before the start of operation. The number and type of specimens are in accordance with GE report NEDO-10115. The specimens meet the intent of ASTM E185-70.

The next surveillance capsule shall be removed from the vessel at approximately 15 EFPY of operation, as recommended in ASTM E185-82, but not to exceed 16 EFPY.

3.6.C. Reactor Vessel Head Stud Tensioning

The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 76°F.

C. Reactor Vessel Head Stud Tensioning

When the reactor vessel head studs are under tension and the reactor is in the Cold Shutdown Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

D. Idle Recirculation Loop Startup

The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.

D. Idle Recirculation Loop Startup

Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be compared and permanently recorded.

3.6 PRIMARY SYSTEM BOUNDARY

A. Reactor Coolant Heatup and Cooldown

The vessel has been analyzed for stresses caused by thermal and pressure transients. Heating and cooling transients throughout plant life at uniform rates of 100°F per hour were considered in the temperature range of 100 to 546°F and were shown to be within the requirements for stress intensity and fatigue limits of Section III of the ASME Boiler and Pressure Vessel Code (1965 Edition including Winter 1966 addenda).

B. Reactor Vessel Temperature and Pressure

Operating limits for the reactor vessel pressure and temperature during normal heatup and cooldown, and during inservice hydrostatic and leak testing were established using 10CFR50 Appendix G, May 1983 and Appendix G of the Winter 1984 Addenda to Section III of the ASME Boiler and Pressure Vessel Code. In addition, operating limits reflecting discontinuity effects were calculated by adjusting BWR/6 discontinuity analyses to reflect the appropriate Hatch 1 RT_{NDT} values. Together, these operating limits assure that a postulated surface flaw, having a depth of 0.24 inch at the flange-to-vessel junction and one-quarter of the material thickness at all other reactor vessel locations can be safely accommodated. For the purpose of setting these operating limits, the RT_{NDT} of the vessel material was estimated from impact test data taken in accordance with requirements of the Code to which this vessel was designed and manufactured (1965 Edition including Winter 1966 Addenda). A General Electric Company procedure, designed to evaluate fracture toughness requirements for older plants where information may be incomplete, was used to estimate RT_{NDT} values on an equivalent basis to the new requirements for plants which have construction permits after August 15, 1973.

The limiting initial RT_{NDT} value of the RPV core beltline region is 10°F, based on Charpy V-Notch data for plate material. The closure flange region RT_{NDT} is limited by the upper vessel shell plate with a value of 16°F based on Charpy data. The non-beltline discontinuity limits for hydrotest (Curve A in GE Topical Report NEDC-30997) are based on the RT_{NDT} for the steam outlet nozzle of 40°F, based on the dropweight test temperature. The non-beltline discontinuity limits for heatup/cooldown (Curve B in GE Topical Report NEDC-30997) and core critical operation (Curve C in GE Topical Report NEDC-30997) are based on the 40°F RT_{NDT} of the steam outlet nozzle, determined by Charpy data.

Figure 3.6-1 establishes minimum temperature requirements for leak testing and hydrostatic testing required by the ASME Boiler and Pressure Vessel Code, Section XI.

Am. No. 59

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.

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.

3.6.B. Reactor Vessel Temperature and Pressure (Continued)

The expected neutron fluence at the reactor vessel wall can be determined at any point during plant life based on the linear relationship between the reactor thermal power output and the corresponding number of neutrons produced. Accordingly, neutron flux wires were removed from the reactor vessel with the suveillance test specimens to establish the correlation at the capsule location by experimental methods. The flux distribution at the vessel wall and 1/4 T depth was analytically determined as a function of core height and azimuth to establish the peak flux location in the vessel and the lead factor of the surveillance specimens. Relating the flux wire data to the vessel peak flux analysis location gives a conservative estimate of maximum 1/4 T depth flux of 1.86×10^9 (n/cm^2-sec).

The first capsule containing test specimens was withdrawn in November 1984 after 5.75 EFPY of operation. The specimens were tested according to ASTM E185-82 and the results are in GE report NEDC-30997. The curves of Figures 3.6-1 through 3.6-3 include the findings of the test report related to the copper-phosphorus content of the RPV core beltline materials, the flux wire test and fluence distribution analysis results, and the Charpy V-Notch specimen test results.

C. Reactor Vessel Head Stud Tensioning

The requirements for cold bolt-up of the reactor vessel closure are based on the RT_{NDT} temperature plus 60°F which is derived from the requirements of the ASME Code to which the vessel was built. The maximum RT_{NDT} of the closure flanges, adjacent head and shell material and stud material is 16°F. The minimum temperature for bolt-up is therefore 16 + 60 = 76°F. The neutron radiation fluence at the closure flanges is well below 10^{17} nvt (>1 Mev) and therefore radiation effects will be minor and will not influence this temperature.

D. Idle Recirculation Loop Startup

Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temperature before a recirculation pump is started prevents the potential seizure of the pump impeller within the wear rings because of the more rapid dimensional increase of the impeller during heatup arising from thermal capacity.

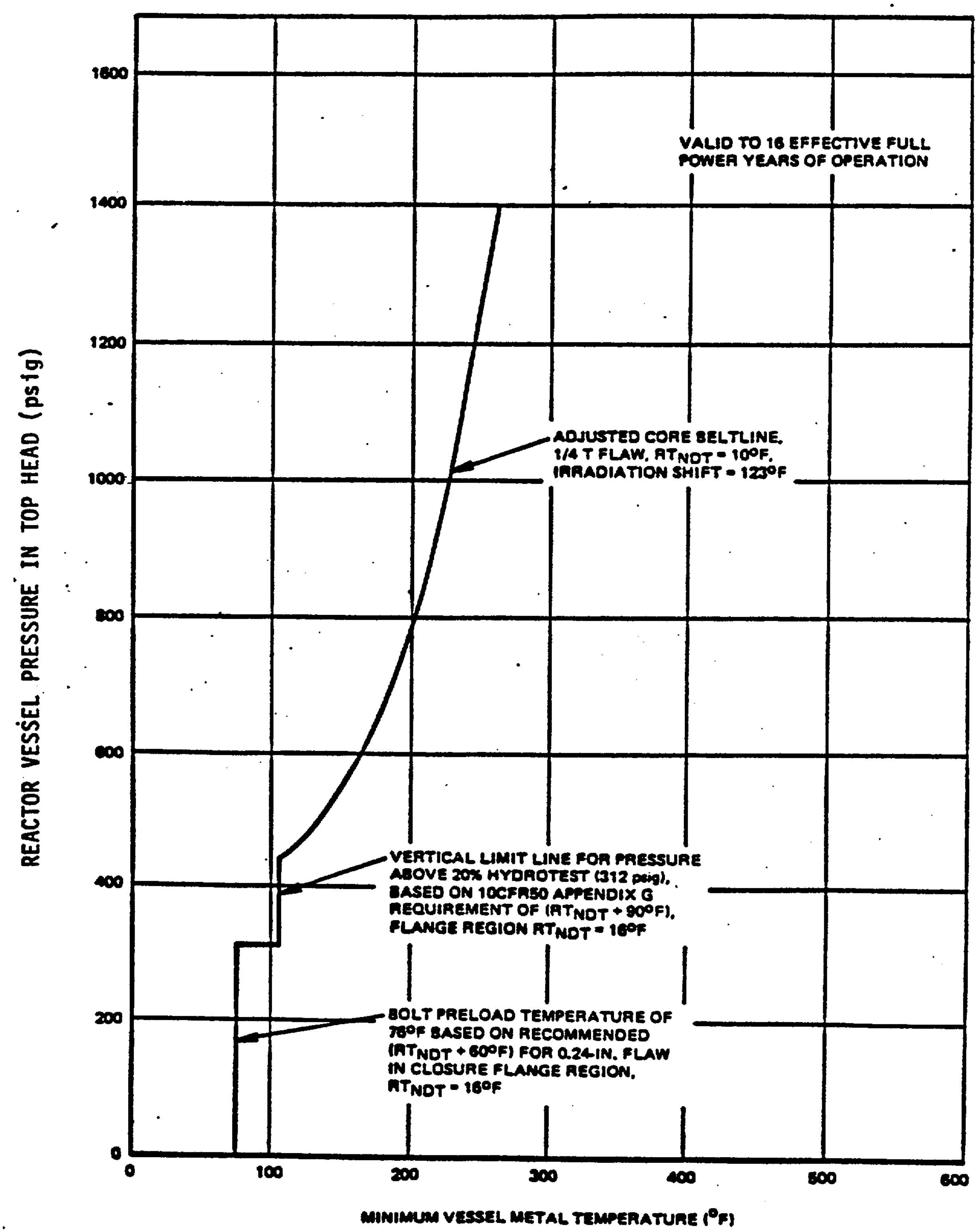

Figure 3.6-1. Pressure versus Minimum Temperature for Pressure Tests, Such as Required by ASME Section XI

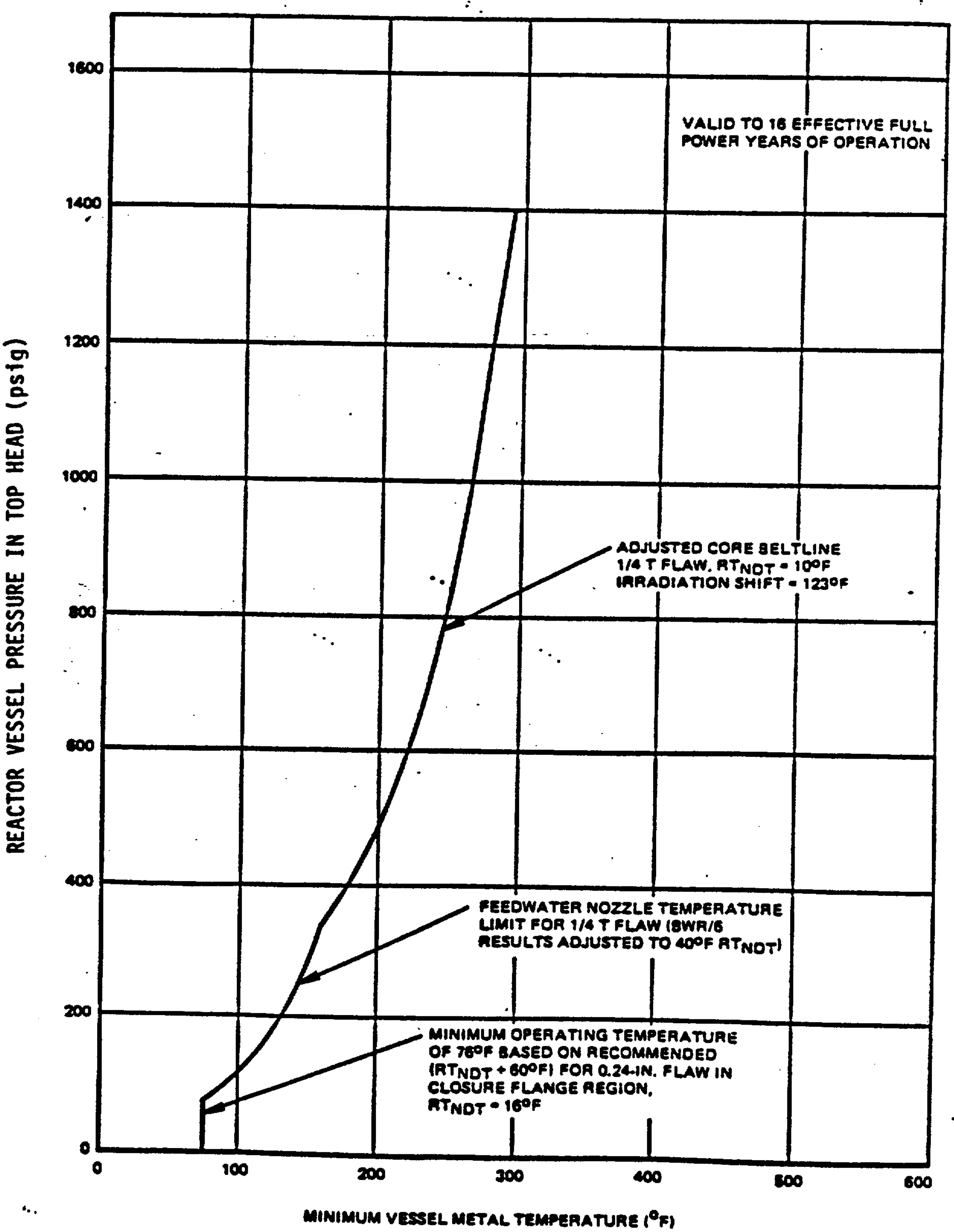

Figure 3.6-2. Pressure versus Minimum Temperature for Non-Nuclear Heatup/Cooldown and Low Power Physics Tests

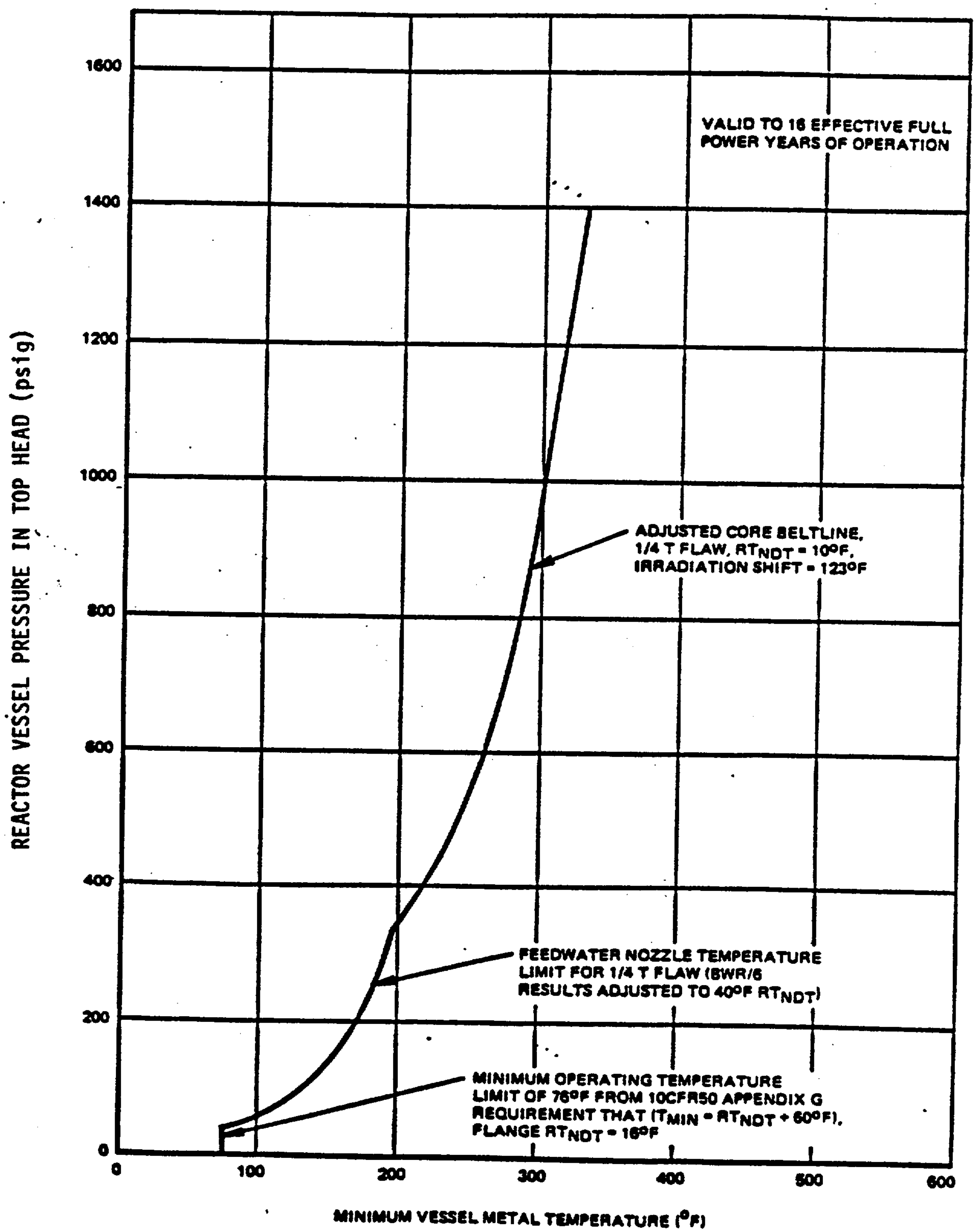

Figure 3.6-3. Pressure versus Minimum Temperature for Core Critical Operation other than Low Power Physics Tests (Includes 40°F Margin Required by 10CFR50 Appendix G)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 126 TO FACILITY OPERATING LICENSE NO. DPR-57

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-321

Introduction

By letter dated January 7, 1986, Georgia Power Company (GPC) requested modification of the Hatch Unit 1 Technical Specifications (TSs) regarding the reactor vessel temperature and pressure limits for pressure tests, non-nuclear heatup/cooldown, core critical operation and vessel stud tensioning. The proposed modifications reflect the decrease in materials fracture toughness as measured in the first surveillance capsule removed from Hatch Unit 1 after Fuel Cycle 10 in November, 1984. As a supporting document to the request, GPC enclosed General Electric Report NEDC-30997 dated October 1985.

Evaluation

The capsule received a fluence of 2.4×10^{17} n/cm^2 ($E \geq 1.0$MeV) during 5.75 effective full power years (EFPY) irradiation, based on dosimetry wire measurements. Chemical analyses of reactor vessel materials indicated the following:

Limiting plate: 0.17% Cu, 0.011% P
Limiting weld: 0:28% Cu, 0.013% P
Surveillance plate: 0.13% Cu, 0.010% P

The shift in fracture toughness properties is a function of fluence and chemical analyses, particularly copper and phosphorus, and is calculated by a specific relationship prescribed in Regulatory Guide 1.99, Revision 1. The shift in RT_{NDT} in the surveillance plate was 47°F, compared to the calculated shift of 17°F, using the methodology described in the Regulatory Guide. The shift in RT_{NDT} in the weld metal was not measured because the fracture toughness properties of the unirradiated weld were not available. Since the measured shift in the surveillance plate exceeded the predicted shift by a factor of 47/17, the coefficient representing the materials from the equation in Regulatory Guide 1.99, Revision 1, was increased by an amount equivalent to a factor of 2.76.

The proposed changes to the Technical Specification relating to the pressure-temperature limits for reactor vessel test and operation meet the requirements of 10 CFR 50, Appendices G and H, ASTM E-185, Regulatory Guide 1.99, Revision 1, and Appendix G, Section III of the ASME Code.

We have reviewed Figure 3.6-1, "Pressure Versus Minimum Temperature for Pressure Tests, Such as Required by ASME Section XI," Figure 3.6-2, "Pressure Versus Minimum Temperature for Non-Nuclear Heatup/Cooldown and Low Power Physics Tests," and Figure 3.6-3, "Pressure Versus Minimum Temperature for Core Critical Operation (Includes 40°F Margin Required by 10 CFR 50, Appendix G) of NEDC-30997. We conclude that these curves are conservative, preserve a margin of safety for Hatch Unit 1 reactor vessel test and operation and are acceptable. Although we find the proposed curves acceptable in this case, we do not agree that the methodology used in conducting the analysis would be applicable for general use. Therefore, our acceptance of these specific limit curves does not constitute generic approval of the methodology.

The staff currently has issued draft Regulatory Guide 1.99, Revision 2, for public comment. Following the review and issuance of the Regulatory Guide, review of and possible changes to the Technical Specifications may be required.

Environmental Consideration

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves 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 amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets 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 amendment.

Conclusion

We have 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: June 20, 1986

Principal Contributor: F. Litton