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AUGUST 14 1978

Docket No. 50-321

Georgia Power Company
Oglethorpe Electric Membership Corporation
Municipal Electric Association of Georgia
City of Dalton, Georgia
ATTN: Mr. Charles F. Whitmer
Vice President - Engineering
Georgia Power Company
Atlanta, Georgia 30302

Gentlemen:

The Commission has issued the enclosed Amendment No. 59 to Facility License No. DPR-57 for the Edwin I. Hatch Nuclear Plant Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your request dated May 26, 1978.

The amendment revises the Technical Specifications to provide operating temperature and pressure limits in accordance with Appendix G, 10 CFR Part 50.

Copies of the related Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Original signed by

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 59 to DPR-57
2. Safety Evaluation
3. Notice

cc w/enclosures:
see next page

Const. 1
GD

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SURNAME	SSheppard	DVerrelli:acr	B Smith	TIppolito	V. Noonan
DATE	7/28/78	7/30/78	7/10/78	8/14/78	8/10/78

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

Georgia Power Company
Oglethorpe Electric Membership Corporation
Municipal Electric Association of Georgia
City of Dalton, Georgia

cc:
G. F. Trowbridge, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Ruble A. Thomas
Vice President
P. O. Box 2625
Southern Services, Inc.
Birmingham, Alabama 35202

Mr. Harry Majors
Southern Services, Inc.
300 Office Park
Birmingham, Alabama 35202

Charles H. Badger
Office of Planning and Budget
Room 610
270 Washington Street, S. W.
Atlanta, Georgia 30334

Mr. H. B. Lee, Chairman
Appling County Commissioners
County Courthouse
Baxley, Georgia 31513

Mr. L. T. Gucwa
Georgia Power Company
Engineering Department
P. O. Box 4545
Atlanta, Georgia 30302

Mr. C. T. Moore
Georgia Power Company
Power Generation Department
P. O. Box 4545
Atlanta, Georgia 30302

Chief, Energy Systems Analysis Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

Mr. D. P. Shannon
Georgia Power Company
Edwin I. Hatch Plant
P. O. Box 442
Baxley, Georgia 31513

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street, N. E.
Atlanta, Georgia 30308

Appling County Public Library
Parker Street
Baxley, Georgia 31513

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

GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION
MUNICIPAL ELECTRIC ASSOCIATION 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. 59
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 May 26, 1978, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 59, 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 14, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 59

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace the following pages of the Technical Specifications contained in Appendix A of the above indicated license with the attached pages bearing the same numbers, except as otherwise indicated. The changed areas on the revised pages are reflected by a marginal line.

Remove	Insert
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	Figure 3.6-4

3.6 PRIMARY SYSTEM BOUND.

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

The reactor vessel shell temperatures during inservice hydrostatic or leak testing shall be at or above the higher of the temperatures shown on the curves of Figure 3.6-2 where the dashed line curve, RPV core beltline region is increased by the expected shift in RT_{NDT} from Figure 3.6-1.

4.6 PRIMARY TEM 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 ≥ 312 psig.

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

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 higher of the temperatures shown on the curves of Figure 3.6-3 where the dashed line curve, initial core beltline region is increased by the expected shift in RT_{NDT} from Figure 3.6-1.

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 higher of the temperatures shown on the curves of Figure 3.6-4 where the dashed line curve, initial core beltline region is increased by the expected shift in RT_{NDT} from Figure 3.6-1.

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

Test specimens representing the reactor vessel, base weld and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The number and type of specimens will be in accordance with GE report NEDO-10115. The specimens shall meet the intent of ASTM E 185-70. Samples shall be withdrawn at one-fourth and three-fourths service life.

3.6.C. Reactor Vessel Head Study 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 70°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.

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 on the reactor vessel pressure and temperature during normal heatup and cooldown, and during inservice hydrostatic testing, were established using Appendix G of the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition, as a guide. These operating limits assure that a large postulated surface flaw, having a depth of one-quarter of the material thickness, can be safely accommodated in regions of the vessel shell remote from discontinuities. Also, discontinuity effects were considered by adjusting BWR/6 discontinuity analyses to Hatch 1 nil ductility temperature conditions. For the purpose of setting these operating limits the reference temperature, 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 to Winter 1966 addenda).

The limitations for the RPV core beltline region and the initial core beltline region were established assuming RT_{NDT} = +10°F established from dropweight tests for base metal and Charpy V notch 30 ft. lb. values for weld metal and heat affected weld zones.

The limitations for areas of the vessel shell remote from regions of discontinuities and the beltline were established assuming RT_{NDT} = +40°F, which was the highest NDTT permissible by the reactor vessel purchase specification. The limitations for areas of the vessel shell in the feedwater nozzle region were established from BWR/6 discontinuity analysis results adjusted to 10°F RT_{NDT}. As shown in Figures 3.6-2 and 3.6-3, one or more of these limitations were used to establish the initial temperature limit.

Figure 3.6-2 establishes minimum temperature requirements for leak testing and hydrostatic testing required by the ASME Boiler and Pressure Vessel Code, Section XI.

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.

Figure 3.6-3 provides appropriate limitations for plant heatup and cooldown when the reactor is not critical. These curves assume heatup and cooldown rates up to 100°F per hour.

Figure 3.6-4 establishes limitations for critical operations. These limits are determined from Figure 3.6-3 using a 40°F margin, and Figure 3.6-2 using a minimum temperature limit based on the inservice hydrostatic test pressure, in order to comply with 10 CFR 50 Appendix G.

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.

The relationship shown in Figure 3.6-1 is between neutron fluence and the predicted adjustment of reference temperature (RT_{NDT}). This relationship is based on NRC recommendations in Regulatory Guide 1.99, Revision 1 (April, 1977) and data contained in General Electric report NEDO-21708. In turn, this change in transition temperature can be related to a change in the temperature ordinate shown in Figure G 2110-1 in Appendix G of Section III of the Boiler Code.

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

The neutron fluence at any point in the pressure vessel wall can be computed from core physics data. The neutron fluence can also be measured experimentally on the ID of the vessel wall. At present valid experimental measurements can be made only over time periods of less than 5 years because of the limitations of the dosimeter materials. This causes no problem because of the exact relationship between thermal power produced and the number of neutrons produced from a given core geometry. A single experimental measurement in a time period of one year can be used to predict the fluence for the life of the plant in terms of thermal power output if no great changes in core geometry are made.

The vessel pressurization temperature at any time period can be determined from the thermal power output of the plant and its relation to the neutron fluence and from Figures 3.6-1 and 3.6-2. For heatup or cooldown and core operation, see also curves on Figures 3.6-3 and 3.6-4. During the first fuel cycle only calculated neutron fluence values were used. At the first refueling, neutron dosimeter wires, which were installed adjacent to the vessel wall, were removed to verify the calculated neutron fluence. As more experience is gained in calculating the fluence the need to verify it experimentally will disappear. Figure 3.6-1 will conservatively predict the shift in transition temperature as a function of neutron fluence. In addition, industry wide shift curves are currently being developed based on information gathered from light water reactors. In case verification of Figure 3.6-1 is required for other reasons, three sets of mechanical test specimens representing the base metal, weld metal and weld heat affected zone metal have been placed in the vessel. These can be removed and tested as required.

C. Reactor Vessel Head Stud Tensioning

The requirements for cold bolt-up of the reactor vessel closure are based on the NDT temperature plus 60°F which is derived from the requirements of the ASME Code to which the vessel was built. The NDT temperature of the closure flanges, adjacent head and shell material and stud material is a maximum of 10°F. The minimum temperature for bolt-up is therefore 10 + 60 = 70°F. The neutron radiation fluence at the closure flanges is well below 10^{17} nvt > 1Mev 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.

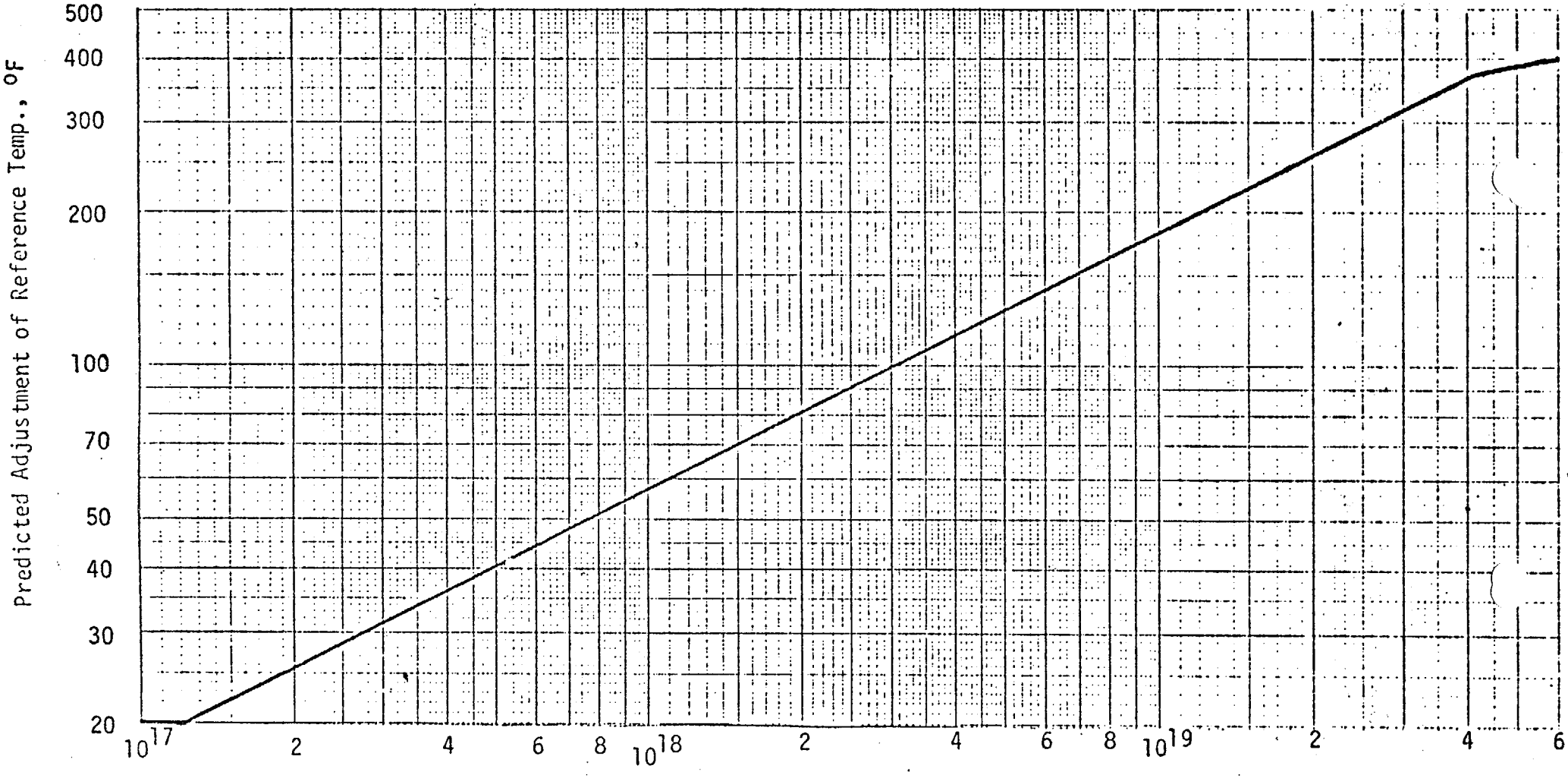

FIGURE 3.6-1 Predicted Adjustment of Reference Temperature as a Function of Fluence. (20°F minimum temperature shift required)

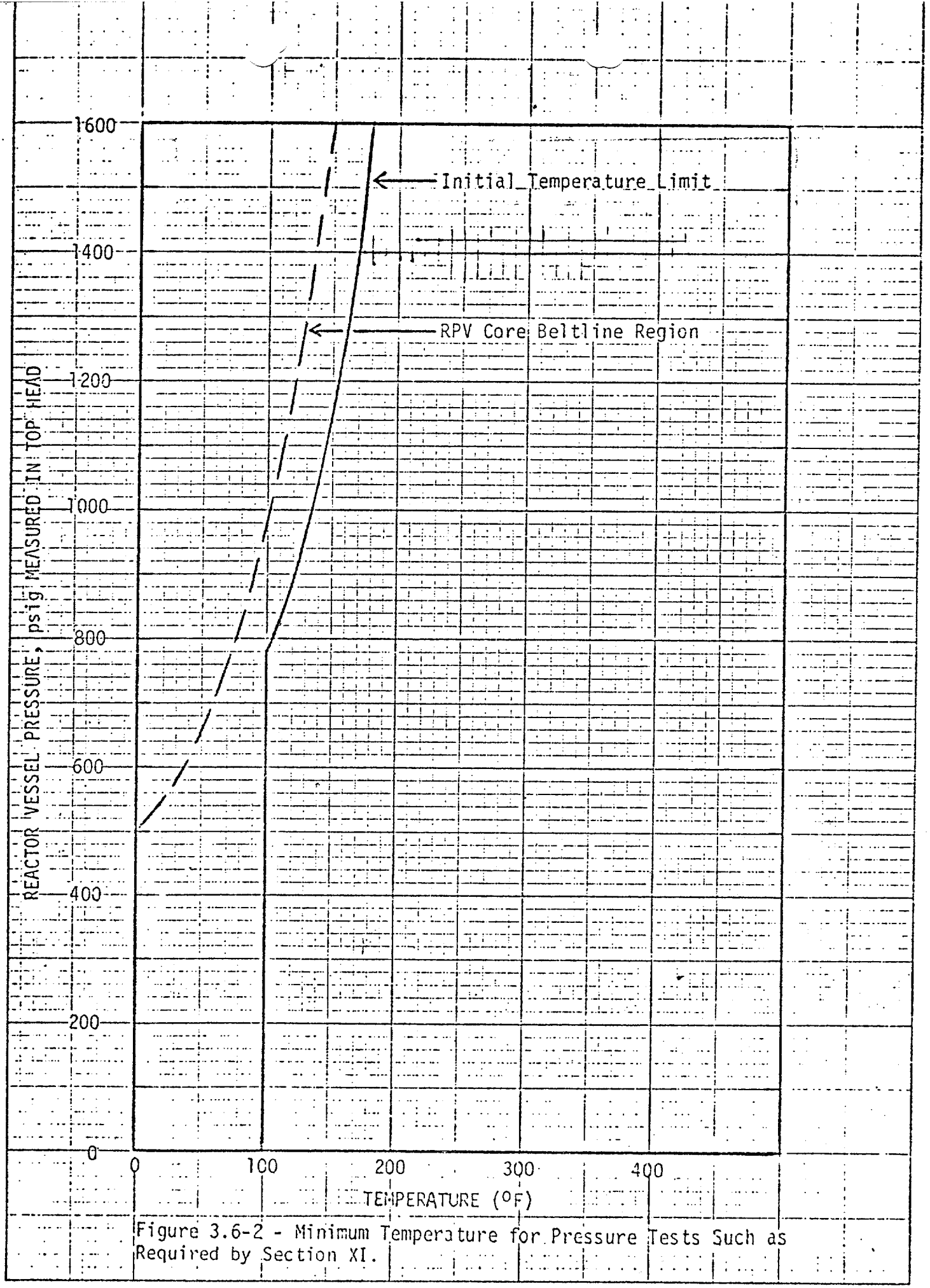

Figure 3.6-2 - Minimum Temperature for Pressure Tests Such as Required by Section XI.

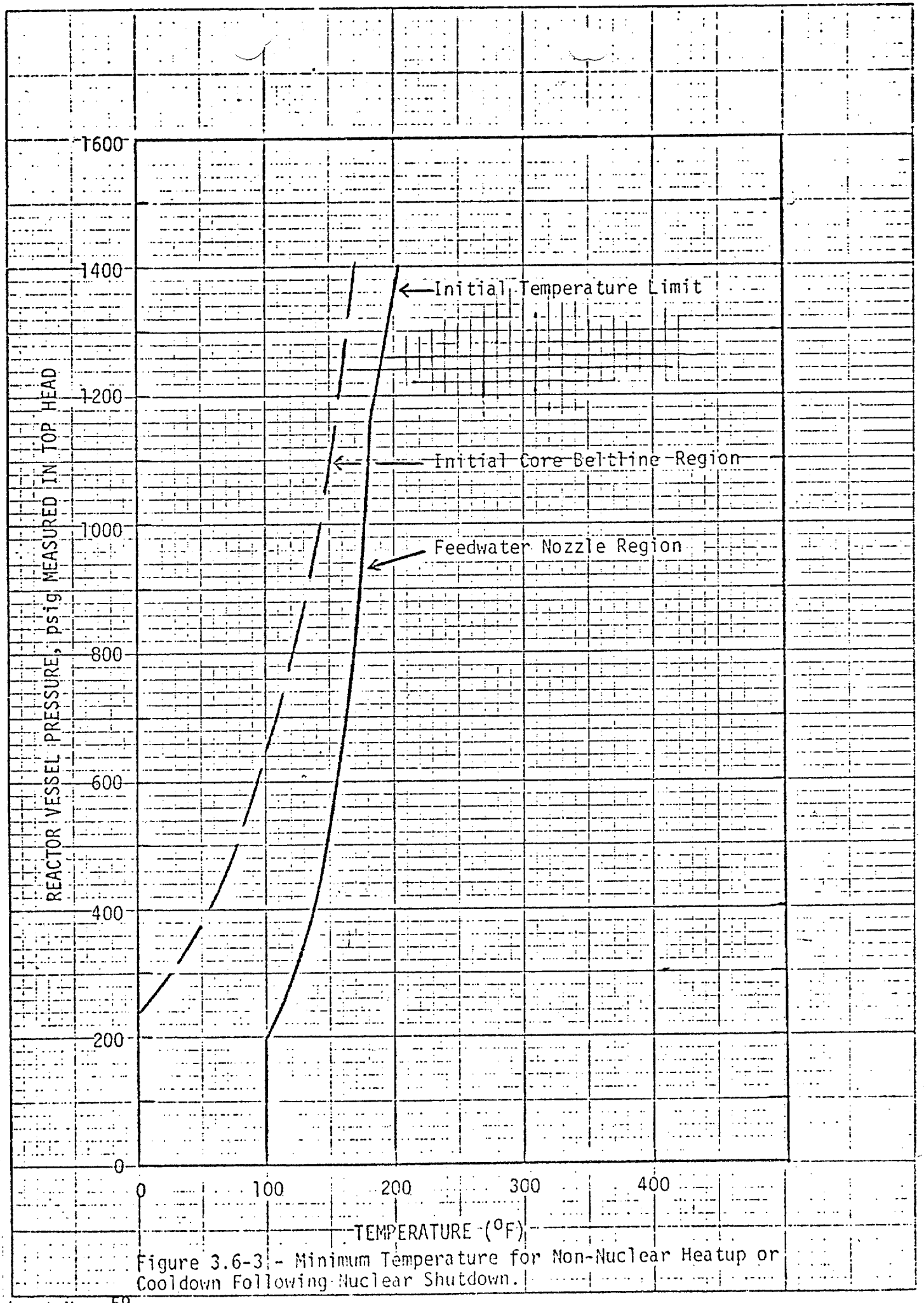

Figure 3.6-3 - Minimum Temperature for Non-Nuclear Heatup or Cooldown Following Nuclear Shutdown.

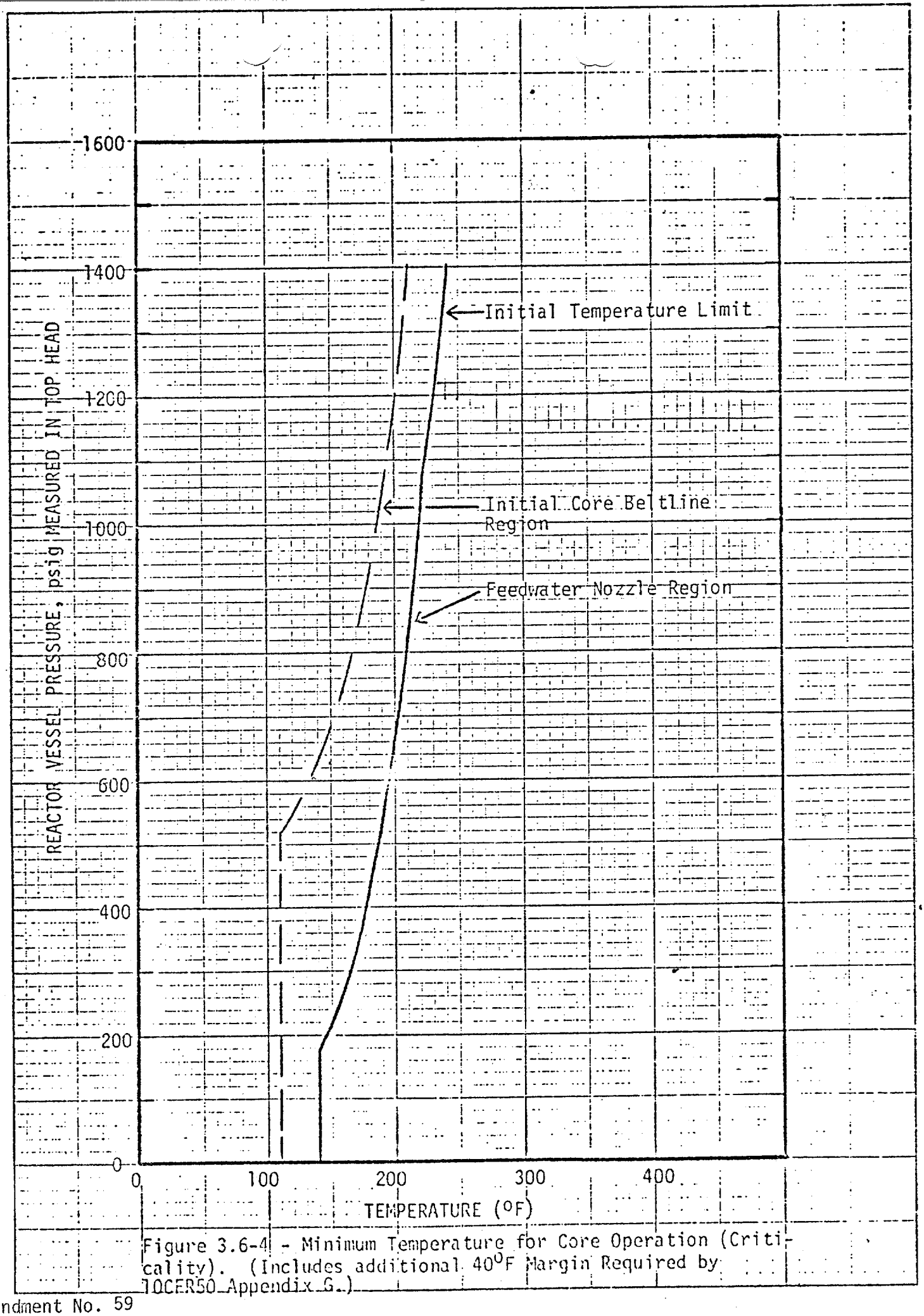

Figure 3.6-4 - Minimum Temperature for Core Operation (Criticality). (Includes additional 40°F Margin Required by 10CFR50 Appendix G.)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 59 TO FACILITY OPERATING LICENSE NO. DPR-57

GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION
MUNICIPAL ELECTRIC ASSOCIATION OF GEORGIA
CITY OF DALTON, GEORGIA

EDWIN I. HATCH NUCLEAR PLANT UNIT NO. 1

DOCKET NO. 50-321

Introduction

By letter dated May 26, 1978, Georgia Power Company proposed a change to the Technical Specifications appended to Operating License No. DPR-57 for Hatch Unit No. 1. The changes would modify the reactor coolant system thermal and pressurization limitations to account for irradiation induced increases in reactor vessel metal nil ductility temperature (RT_{NDT}). The Georgia Power Company submittal was based on their determination that certain changes were necessary to bring the reactor coolant system pressure-temperature limits into conformity with the requirements of paragraph IV A of Appendix G to 10 CFR 50.

Discussion

Title 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements", requires that pressure-temperature limits be established for reactor coolant system heatup and cooldown operations, inservice leak and hydrostatic tests, and reactor core operation. These limits are required to ensure that the stresses in the reactor vessel remain within acceptable limits. They are intended to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences.

The specific pressure-temperature limits which are initially established depend upon the metallurgical properties of the reactor vessel material and the design service conditions. However, the metallurgical properties vary over the lifetime of the reactor vessel because of the effects

of neutron irradiation. One principal effect of the neutron irradiation is that it causes the reactor vessel nil ductility temperature $(RT_{NDT})^1$ to increase or shift with time. The practical results of the RT_{NDT} shift is that for any given value of reactor pressure, the reactor vessel metal temperature must be maintained at higher values during the heatup and cooldown process. By periodically revising the pressure-temperature limits to account for neutron irradiation induced increases in RT_{NDT} the stresses in the reactor vessel are maintained within acceptable limits.

Evaluation

The Georgia Power Company submittal dated May 26, 1978 included pressure-temperature limits for hydrostatic testing, mechanical heatup and cooldown, minimum temperature for core operation (criticality), and damage estimate curves predicting the effect of neutron fluence on RT_{NDT}. These proposed operating limits include pressure-temperature operating curves based on an analysis of regions of discontinuities that are remote from the beltline region. The stress intensities used for this analysis were obtained by adjusting the results of the analysis made on the BWR/6 251 reactor vessel to account for differences between the design and material properties of these vessels. Since both vessels were designed to the same rule, i.e., Section III of the ASME Code, we find this procedure acceptable. Currently, the feedwater nozzle is the limiting region for the operating limits. This region will see very low radiation levels so that there will be no degradation of its materials; i.e., no increase in RT_{NDT}. On the other hand the materials in the beltline region are subjected to high radiation levels that will cause their RT_{NDT} values to increase. Because of this radiation damage, the beltline material will become the limiting material in about 4 EFPY. To account for this shift from feedwater nozzle to beltline region as the limiting material, the proposed operating limit curves, Figures 3.6.2, 3 and 4, provide curves based on both the nozzle region analysis and on the beltline region analysis. The pressure-temperature curves for the beltline region will be increased by the shift in reference temperature produced by irradiation as predicted in Figure 3.6.1. The estimated increases in reference temperature presented in Figure 3.6.1 were obtained from Regulatory Guide 1.99, Revision 1 using a copper content of 0.2%.

---

[1] RT_{NDT} is the temperature associated with the transition from ductile to brittle fracture mode of failure.

We have reviewed the proposed amendment to Technical Specifications 3/4.6A, B and C regarding the reactor vessel pressure-temperature operating limits. We conclude that the proposed operating limits provide adequate margin of protection against fracture, taking into account factors as the potential for anticipated overstress and thermal shock during operationl occurrences in the control of reactivity. Thus, the revised Technical Specifications are in accordance with the requirements of Section IV A of Appendix G to 10 CFR 50 and are acceptable. Compliance with Appendix G in establishing safe operating limitations will ensure adequate safety margins during operation, testing, maintenance and postulated accident conditions and constitute an acceptable basis for satisfying the requirements of NRC General Design Criterion 31, Appendix A, 10 CFR Part 50.

Other Technical Specification Changes

In addition to the Technical Specification changes discussed above, the only other change authorized by the amendment is to delete the surveillance requirement regarding neutron flux wires. These wires were removed (as required) and tested during the first refueling outage. Thus, this specification is no longer relevent to Hatch Unit No. 1 and its deletion is acceptable.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: August 14, 1978

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-321

GEORGIA POWER COMPANY, ET AL

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 59 to Facility Operating License No. DPR-57 issued to Georgia Power Company, Oglethorpe Electric Membership Corporation, Municipal Electric Association of Georgia and City of Dalton, Georgia, which revised Technical Specifications for operation of the Edwin I. Hatch Nuclear Plant, Unit No. 1, located in Appling County, Georgia. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications to provide operating temperature and pressure limits in accordance with Appendix G, 10 CFR Part 50.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant

to 10 CFR Section 51.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated May 26, 1978, (2) Amendment No. 59 to License No. DPR-57, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Appling County Public Library, Parker Street, Baxley, Georgia 31513. A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 14th day of August 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors