Georgia Power Company
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201
Telephone 205 877-7279

J. T. Beckham, Jr.
Vice President - Nuclear
Hatch Project

Georgia Power
the southern electric system

June 20, 1995

Docket Nos. 50-321
50-366

HL-4865

TAC Nos. M91077
M91078

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Edwin I. Hatch Nuclear Plant
Response to Second Request for Additional Information:
Power Uprate

Gentlemen:

On May 18, 1995, Georgia Power Company (GPC) received a second request for additional information (RFAI) regarding the Plant Hatch power uprate amendment request dated January 13, 1995. Responses to the May 18th NRC request are provided in Enclosure 1.

The original power uprate amendment request, dated January 13, 1995, was submitted utilizing the proposed Improved Technical Specifications (ITS) pages which were current at the time. Since the January 13th submittal, the NRC issued Technical Specifications Amendments 195 and 135 to the Hatch Unit 1 and Unit 2 Technical Specifications, respectively, which approve the ITS.

As a consequence of the ITS conversion, this letter also transmits the previously proposed power uprate changes incorporated into the Technical Specifications as issued by Amendments 195 and 135. The technical justifications and evaluations provided in the January 13, 1995, request remain valid. Enclosure 2 contains the basis for GPC's determination that the proposed changes do not involve a significant hazards consideration. Enclosure 3 provides the Environmental Assessment. Both Enclosures 2 and 3 are based on GPC's January 13th submittal and additional information supplied to the NRC by GPC on April 5, 1995.

Enclosure 4 provides page change instructions for incorporating the proposed changes for power uprate, the revised Technical Specifications pages (utilizing Amendments 195 and 135 as the basis), and the corresponding marked up pages. Enclosure 5 provides, for your information, the Bases changes which reflect the proposed change.

230023

A001
111

In accordance with the requirements of 10 CFR 50.91, a copy of this letter and the enclosure will be sent to Mr. J. D. Tanner of the Environmental Protection Division of the Georgia Department of Natural Resources.

Mr. J. T. Beckham, Jr. states that he is a Vice President of Georgia Power Company and is authorized to execute this oath on behalf of Georgia Power Company, and to the best of his knowledge and belief, the facts set forth in this letter are true.

Sincerely,

J. T. Beckham, Jr.

J. T. Beckham, Jr.

Sworn to and subscribed before me this 20th *day of* June *1995.*

Doris Delene Brown
Notary Public

My Commission Expires November 3, 1997

GKM/eb

References:

1. HL-4738, J. T. Beckham, Jr., to NRC, "Power Uprate Implementation Schedule," dated December 1, 1994.
2. HL-4724, J. T. Beckham, Jr., to NRC, "Power Uprate Operation," dated January 13, 1995.
3. K. N. Jabbour to J. T. Beckham, Jr., "Request for Additional Information Regarding Power Uprate Program for Hatch Nuclear Plant, Units 1 and 2," dated March 10, 1995.
4. HL-4812, J. T. Beckham, Jr., to NRC, "Response to Request for Additional Information - Power Uprate Submittal," dated April 5, 1995.
5. K. N. Jabbour to J. T. Beckham, Jr., "Second Request for Additional Information Regarding Power Uprate Program for Hatch Nuclear Plant Units 1 and 2," dated May 18, 1995.

Georgia Power

Enclosures:

1. Response to Second RFAI
2. 10 CFR 50.92 Evaluation
3. Environmental Assessment
4. Page Change Instructions and Proposed Technical Specifications Pages
5. Bases Changes

cc: *Georgia Power Company*
Mr. H. L. Sumner, Jr., Nuclear Plant General Manager
NORMS

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. K. Jabbour, Licensing Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. S. D. Ebneter, Regional Administrator
Mr. B. L. Holbrook, Senior Resident Inspector - Hatch

State of Georgia
Mr. J. D. Tanner, Commissioner - Department of Natural Resources

Enclosure 1

Edwin I. Hatch Nuclear Plant
Response to Second RFAI
Power Uprate Submittal

The NRC questions contained in the May 18, 1995, letter are repeated for clarity and relate to GPC's power uprate submittal dated April 5, 1995.

NRC Question 1A:

The licensee stated that Regulatory Guide (RG) 1.99, Revision 1, was used to calculate the adjusted reference temperature (ART). In accordance with Generic Letter 88-11, Revision 2 of RG 1.99 must be used, not Revision 1. Discuss this discrepancy.

GPC Response:

All ART calculations used RG 1.99, Revision 2. A typographical error was made in GPC's April 5th letter.

NRC Question 1B:

Based on an independent calculation, the staff was not able to verify the limiting ART values listed in the table for Hatch Units 1 and 2. Provide the initial RT_{NDT}, the increase in RT_{NDT}, and the margins that were used in calculating the limiting ARTs, using RG 1.99, Revision 2.

GPC Response:

The following table shows the data used to calculate the limiting ART for the materials:

Limiting ART for Current Power

Component	ID	Heat	Initial RT_{NDT}	32 EFPY Delta RT_{NDT}	Margin	32 EFPY ART
Hatch Unit 1						
Plate: Lower Shell	G-4805-2	C4112-2	10	49	34	93
Weld: Lower to Low-Int Girth	1-313	90099	-10	113.2	56	159.2
Hatch Unit 2						
Plate: Lower Shell	G-6603-2	C8553-1	24	20.8	20.8	65.6
Weld: Lower Long.	101-842	10137	-50	63	56	69.0
Weld: Low-Int Long.	101-834	51874	-50	57.9	56	63.9

Limiting ART for Uprated Power

Component	ID	Heat	Initial RT_{NDT}	32 EFPY Delta RT_{NDT}	Margin	32 EFPY ART
Hatch Unit 1						
Plate: Lower Shell	G-4805-2	C4112-2	10	51.1	34	95.1
Weld: Lower to Low-Int Girth	1-313	90099	-10	117.9	56	163.9
Hatch Unit 2						
Plate: Lower Shell	G-6603-2	C8553-1	24	21.8	21.8	67.6
Weld: Lower Long.	101-842	10137	-50	65.9	56	71.9
Weld: Low-Int Long.	101-834	51874	-50	60.6	56	66.6

NRC Question 2A:

The licensee stated that the second capsule will be removed from Unit 1 either during the 1996 spring or 1997 fall outages. Discuss the uncertainty of 1.5 years in the removal schedule of the second capsule.

GPC Response:

The uncertainty in the removal schedule was based on how well Unit 1 would run during its current fuel cycle (Cycle 16). The unit has run well, and removal of the second capsule is now scheduled for Spring 1996.

Unit 1 FSAR Section 4.2 states that removal of the second surveillance capsule is planned after 15 EFPY of operation. As of May 18, 1995, Unit 1 had accumulated approximately 13.5 EFPY and has run at a high capacity factor for the first 7 months of this 16-month operating cycle. Therefore, GPC intends to remove the capsule during the Spring 1996 Unit 1 outage.

NRC Question 2B:

The 1 effective full-power year (EFPY) used in the licensee's ART calculation may not be conservative. The testing of a capsule and documentation of the test results usually takes about 1 year. If the capsule was removed in the fall of 1997, the test results would not be available until the fall of 1998. In such a case, the use of 1 EFPY may not be sufficiently conservative. Discuss this discrepancy.

GPC Response:

The schedule for removing the capsule is Spring 1996. Therefore, the new P-T curves will be available Spring 1997. Power uprate is scheduled to be fully implemented by Spring 1996. The following shows the predicted number of full power years for Unit 1:

> As of May 18, 1995, Unit 1 accumulated 13.5 EFPY at current power. By Spring 1996, Unit 1 will accumulate less than 15 EFPY (14.3 EFPY) at current power when the capsule will be withdrawn. By Spring 1997, when the new P-T curves will be available, Unit 1 will accumulate less than 1 EFPY at uprated power. Therefore, with a withdrawal schedule of Spring 1996, the calculation using 16 EFPY at current power and 1 EFPY at uprated power is conservative.

NRC Question 2C:

Provide detailed calculations to show how ART of 132° F for the uprate case and 133° F for the current P-T limits were derived for Hatch Unit 1.

GPC Response:

17 EFPY ART calculation of 132°F for the uprate case:

ART	= Initial RT_{NDT} + 17 EFPY ΔRT_{NDT} + Margin, where:
Initial RT_{NDT}	= -10°F
17 EFPY ΔRT_{NDT}	= (CF) $f^{(0.28-0.1\log f)}$ = 85.1°F, where:
CF	= 207
$f^{(0.28-0.1\log f)}$	= 0.4113 = Fluence factor. Therefore,
f	= 0.0973
	= 1/4 T fluence at current power for 16 EFPY +
	= 1/4 T fluence at uprated power for 1 EFPY
	= 1.82e18 n/cm^2 (16 yr)/32 yrs/*1e19* *n*/cm^2 +
	= 1.82e18 n/cm^2 * 1.1 (1 yr)/32 yrs/1e19 n/cm^2
Margin	= 56°F

The 133°F calculations were submitted to the NRC in GPC letter SL-189, dated January 10, 1986. This GPC submittal provided GE Report NEDC-30997, "Edwin I. Hatch Nuclear Power Plant, Unit 1, Reactor Pressure Vessel Surveillance Materials Testing and Fracture Toughness Analysis," October 1985. The NRC reviewed this report and approved the Unit 1 P-T curves as Amendment 126 to the Unit 1 Technical Specifications. Pertinent pages of NEDC-30997 are provided in Attachment A to this enclosure.

<u>NRC Question 2D</u>:

Provide the current (as of May 18, 1995) EFPY and neutron fluence for Units 1 and 2.

<u>GPC Response</u>:

Georgia Power reported the following EFPY as of May 18, 1995:

> The current EFPY for Unit 1 is 13.5 EFPY. Peak inside diameter (ID) fluence is 1.18e18 n/cm^2.
>
> The current EFPY for Unit 2 is 10.9 EFPY. Peak ID fluence is 4.8e17 n/cm^2.

These fluence values are consistent with the 32 EFPY fluence values provided to the NRC in GPC letter HL-4636, "Edwin I. Hatch Nuclear Plant Response to Open Issues - Generic Letter 92-01, Rev. 1, Reactor Vessel Structural Integrity," dated July 1, 1994.

<u>NRC Question 2E</u>:

Provide detailed calculations to show that for unit 2 the non-beltline curves are still limiting even when evaluating the ART for 32 EFPY at 110% power uprate.

<u>GPC Response</u>:

As can be seen in the tables in Question 1B, the ART for Unit 2 increases by approximately 3°F (from 69.0°F to 71.9° F for the largest difference) when considering the power uprate condition. The beltline curves shown in Unit 2 Technical Specifications figures 3.4.9-1 through 3.4.9-3 can easily accommodate an additional 3°F shift without becoming the limiting curves. (The smallest temperature difference between the beltline and non-beltline curve is 10°F.) The following table shows the beltline and non-beltline temperatures corresponding to 1400 psi for each curve. The A', A, B', B, C', and C curves refer to the labels on the Technical Specification figures.

Tech Spec Figure No.	Curve	Beltline Temperature (°F)	Non-Beltline Temperature (°F)	Delta Temperature (°F)
3.4.9-1	A' and A	184	201	17
3.4.9-2	B' and B	215	225	10
3.4.9-3	C' and C	255	265	10

Therefore, the non-beltline curves remain limiting.

NRC Question 3 - Turbine Overspeed Protection:

The impact of Hatch operation at the proposed uprated power level on turbine overspeed protection has not been addressed in your previous submittals related to this subject. Provide the analyses that have been performed to ensure that there will be no increase in the probability of turbine overspeed or the associated turbine missile production due to plant operation at the proposed uprated power level.

GPC Response:

The overspeed calculation compares the entrapped steam energy contained within the turbine and the associated piping after the stop valves trip and the sensitivity of the rotor train for the capability of overspeeding. The entrapped energy increases slightly for the uprated conditions but not to the degree that an adjustment of the overspeed trip setting is needed. Based on the revised steam specifications for this uprate, the overspeed potential for these units was reviewed, and no changes to the current mechanical trip settings are required. As stated in GPC's response to Question No. 29 in the April 5, 1995 submittal, the probability of turbine missile production was also reevaluated and found to remain within acceptable limits.

NRC Question 4 - Reactor Vessel Internals:

The licensee's response to Question 6 stated that Section 3.3.2 of NEDC-32405P specifies the Code and Edition used for evaluation of the reactor internals. However, Section 3.3.2 specifies the specific applicable Code Editions for the reactor vessel, and not for the reactor internals. Provide the specific applicable Code Edition used for the evaluation of the reactor vessel internals.

GPC Response:

The evaluation of the reactor vessel internals uses ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, in conjunction with the Hatch FSAR requirements, as a guide for design acceptance criteria. No specific ASME Code edition was applicable to reactor internals during the time of Hatch design and construction. The specific applicable Code edition for reactor vessel internals is cited in Section 3.3.2 of NEDC-32405P.

NRC Question 5 - Fatigue Usage Factor:

The licensee's response to Question 10 stated that because the initial estimate of the cumulative fatigue usage factor (CUF) of the Unit 2 feedwater nozzle exceeded 1.0 for the power uprate condition, the CUF was reevaluated and determined to be 0.93 based on the actual operating cycle data combined with the design basis CUF calculated for the power uprate period. Provide an evaluation for other RPV components if the operating cycle counting data are used to calculate the fatigue usage factor.

GPC Response:

Operating cycle counting data were used only for the Unit 2 feedwater nozzle; however, if operating cycle counting data were used for other components, it is expected that the fatigue usage factor would be less than that shown in the report since actual cycles to date are less than the estimated cycles used in the analysis.

NRC Question 6 - Evaluation of Reactor Coolant Pressure Boundary:

Based on the licensee response to Question 11, it appears that the evaluations of the reactor coolant pressure boundary (RCPB) piping systems were performed only for the main steam and recirculation lines. Provide evaluation for the remaining RCPB piping systems.

GPC Response:

The following table presents a summary of the piping evaluations that were necessary for the Hatch Units 1 and 2 piping power uprate scope of services.

Unit	Category	Description	Evaluation Needed
1	A	Main Steam	All piping
1	B	Feedwater	None (Note 1)
1	C	Recirculation	None (Note 2)
1	D	RWCU	None (Note 2)
1	E	Core Spray	None (Note 2)
2	A	Main Steam	All piping
2	B	Feedwater	None (Note 1)
2	C	Recirculation	All piping
2	D	RWCU	Inside Containment
2	E	Core Spray	None (Note 1)

Notes:

1. Increases due to power uprate are insignificant. For these systems, either the increase was zero due to the use of conservative design pressures and temperatures in the existing analysis or the increase was small such that the impact on piping analyses was negligible.

2. Design adequacy is demonstrated by reanalysis.

NRC Question 7 - Stress Ratios:

Table 16-1 provided by the licensee in response to Questions 16 and 17 shows that stress ratios computed from Equation 10 of the ASME Section III, Subsection NB, for the Safety and Relief Valve Discharge Lines exceed 1.0 for both Hatch Units 1 and 2. Stress ratios calculated from Equation 12 are 0.93 and 0.76 for Units 1 and 2 respectively. Provide the stress ratios based on Equations 13 and 14 of the ASME Code.

GPC Response:

The Hatch Units 1 and 2 SRVDLs are Class 3 piping. If Equation 10 exceeds 1.0, the piping is considered acceptable if Equation 11 is less than 1.0. In Table 16-1, the values of 0.93 and 0.76 should have been shown as Equation 11. Therefore, the SRVDLs are acceptable. Table 1 presents a revised summary of the limiting stress ratios from the power uprate balance of plant piping evaluations.

NRC Question 8 - Reliability of High Pressure Coolant [sic] Injection and Reactor Core Isolation Cooling Systems:

The licensee should provide assurance that the high pressure coolant [sic] injection and the reactor core isolation cooling systems will be capable of injecting their design flow rates at the higher reactor operating pressures associated with power uprate. Additionally, the licensee should provide assurance that the reliability of the systems will not be decreased by the higher loads placed on the systems or because of any modifications made to the systems to compensate for the increased loads.

GPC Response:

The effect of power uprate on the reliability of the reactor core isolation cooling (RCIC) and the high pressure coolant injection (HPCI) systems can be divided into two parts:

1. Availability of each system. Periodic surveillance testing at the slightly higher reactor operating pressures combined with infrequent demands for the system to operate under design conditions will result in an insignificant change in component failure

rates. Thus, power uprate will not have any effect on the availability of the two systems.

2. Reliability of each system to successfully start and operate at the higher design pressure required with power uprate. Each turbine and pump were analyzed to confirm they have the capability to perform at higher pressure. The reduction in the overspeed trip margin resulting from the higher rated speeds was evaluated, and the reduction in the overspeed trip margin is insignificant in terms of its effect on reliability. Modifications that significantly reduce the severity of startup transients, even with the increased steam pressures on the HPCI and RCIC turbines, have either been installed or are planned to be installed prior to power uprate implementation. (Reference SILs 480 and 377.) Performance tests will be conducted to ensure that the HPCI and RCIC systems can function at the uprated conditions.

Table 1

Hatch Piping Power Uprate
Summary of Highest Ratios
(Calculated to Allowable)

Unit	System	Eq. 8	Eq. 9:N/U	Eq. 9:Emer	Eq. 9:Faulted	Eq. 10	Eq. 11	Eq. 12	Eq. 13	Eq. 14
1	Main Steam		0.72	0.53	0.52	1.35		0.99	0.68	0.66
1	SRVDLs	0.32	0.99	0.79	0.69	1.32	0.93			
1	HPCI		0.78	NA *	0.71	0.95				
2	Main Steam		0.52	0.41	0.33	1.34		0.81	0.61	0.28
2	SRVDLs	0.2	0.81	0.65	0.51	1.09	0.76			
2	Turbine Bypass	0.29	0.29	0.38	0.16	0.36				
2	Recirculation		0.96	0.72	0.66	1.09		0.73	0.69	0.08
2	RWCU		0.38	0.77	0.6	0.51				

* Not Available

Attachment A

Enclosure 1

Excerpts from NEDC-30997, "Edwin I. Hatch Nuclear Power Plant, Unit 1
Reactor Pressure Vessel Surveillance Materials Testing
and Fracture Toughness Analysis,"
GENE, San Jose, CA, October 1985

Pages 7-4 - 7-6, 7-12 - 7-16

~~erature mentioned in Subsection 7.4 for Curve C is an exception for BWRs, allowing critical operation at temperatures below the hydrostatic test temperature.~~

7.6 EVALUATION OF RADIATION EFFECTS

The shift in fracture toughness properties in the beltline materials is a function of neutron fluence and the presence of certain elements, such as copper (Cu) and phosphorus (P). The specific relationship from Reference 5 is:

$$\text{SHIFT } (^\circ F) = [40 + 1000\ (\%\ Cu - 0.08) + 5000\ (\%\ P - 0.008)] * (f/10^{19})^{1/2} \tag{7-1}$$

where:

% Cu = wt % of Cu present,
% P = wt % of P present,
f = fluence (n/cm^2) at selected EFPY.

The beltline chemical compositions in Tables 3-1 and 3-3 show 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

This material information is used to evaluate irradiation shift versus fluence.

7.6.1 Measured Versus Predicted Surveillance Shift

Table 5-3 presents a measured shift for the base metal RT_{NDT} of 47°F. No measured shift is available for the weld metal. The predicted shift of

the surveillance plate, calculated according to Equation (7-1), assumes 0.13% Cu, 0.010% P and a fluence from Table 4-2 of:

$f = (2.4 \times 10^{17}\ n/cm^{17}\ n/cm^2) * (1.25 \text{ uncertainty factor})$, or
$f = 3.0 \times 10^{17}\ n/cm^2$.

The predicted shift is 17°F.

7.6.2 Modification of the Shift Relationship

Since the measured shift exceeds the predicted shift,, the Reference 5 method is not conservative for the base metal and must be modified. The shift calculated is proportional to the material characteristics and to the square root of the fluence. Assuming that the fluence relationship is correct means that the coefficient representing the materials in Equation (7-1) must be increased by the factor (47/17) or 2.76.

7.6.3 Radiation Shift Versus EFPY

Equation (7-1) can be simplified and expressed as a function of EFPY for the base metal and weld metal. Subsection 4.3 concludes that the EOL 1/4 T fluence (32 EFPY) is $1.9 \times 10^{18}\ n/cm^2$. Therefore, in terms of EFPY, the fluence is

$$f = 5.94 \times 10^{16} * \text{EFPY} \qquad (7-2)$$

Equation (7-2) is used in Equation (7-1) with the appropriate Cu, P values.

For the weld metal:

$$\text{SHIFT}_W = 20.42 \quad \text{EFPY} . \qquad (7-3)$$

For the base metal: (including 2.76 factor)

$$\text{SHIFT}_B = 30.84 \quad \text{EFPY} . \qquad (7-4)$$

The initial RT_{NDT} of the beltline plates of 10°F is higher than the -10°F RT_{NDT} of the beltline welds. Therefore, the plates are the limiting beltline material. The shift relationship in Equation (7-4) is plotted on Figure 7-4.

7.6.4 End-Of-Life Conditions

Paragraph IV.B of Reference 1 sets limits on the adjusted reference temperature (initial RT_{NDT} plus SHIFT) and on the upper shelf energy (USE) of the beltline materials. The adjusted reference temperature must be less than 200°F, and the USE must be above 50 ft-lb. Based on Figure 7-4, the EOL SHIFT of 174°F plus the initial RT_{NDT} of 10°F results in an adjusted reference temperature of 184°F, which is acceptable. Calculations of USE using data from Table 5-3 and following Reference 5 procedures are summarized in Table 7-1. The equivalent transverse USE of the plate material is taken as 65% of the longitudinal USE, according to Reference 6. The weld metal USE is not adjusted because weld metal has no orientation effect. The EOL plate and weld USE values are estimated as 66 ft-lb and 72 ft-lb, respectively, which are above the minimum limit. Therefore, irradiation effects are not severe enough to necessitate RPV annealing before 32 EFPY.

7.7 OPERATING LIMITS CURVES VALID TO 16 EFPY

The shift applied to the unirradiated core beltline curves in Figures 7-1 through 7-3 depends on the amount of operation for which the curves are valid. Sixteen EFPY was selected because Reference 3 requires withdrawal of the second surveillance capsule at 15 EFPY. The beltline shift estimated with Figure 7-4 is 123°F. Adding this to the unirradiated beltline curves in Figures 7-1 through 7-3, and considering the non-beltline curves, gives the operating limits valid to 16 EFPY, as shown in Figures 7-5 through 7-7. These curves are also presented in the Technical Specification revision in Appendix A.

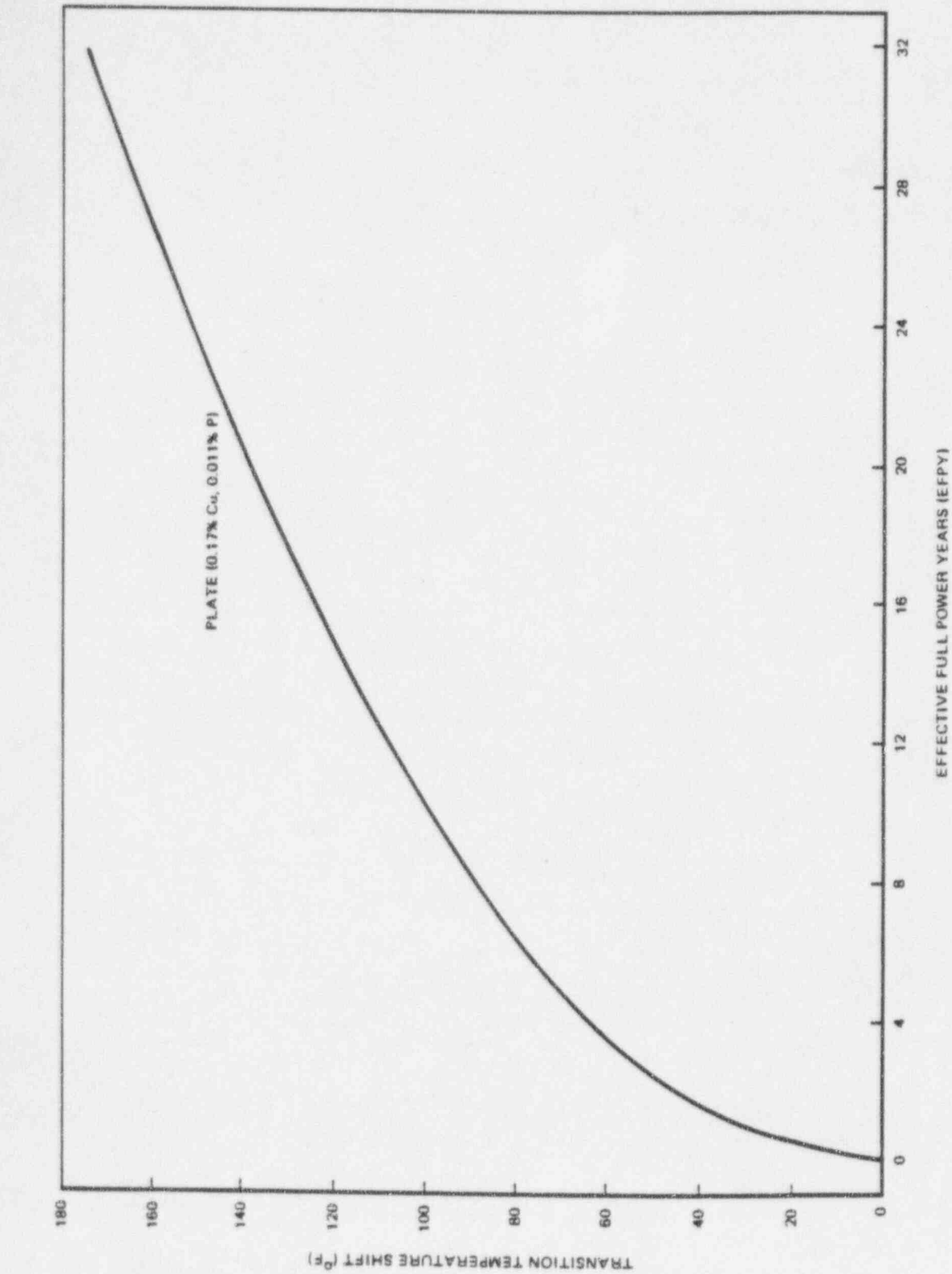

Figure 7-4. Transition Temperature Shift Based on Surveillance Specimen Test Results

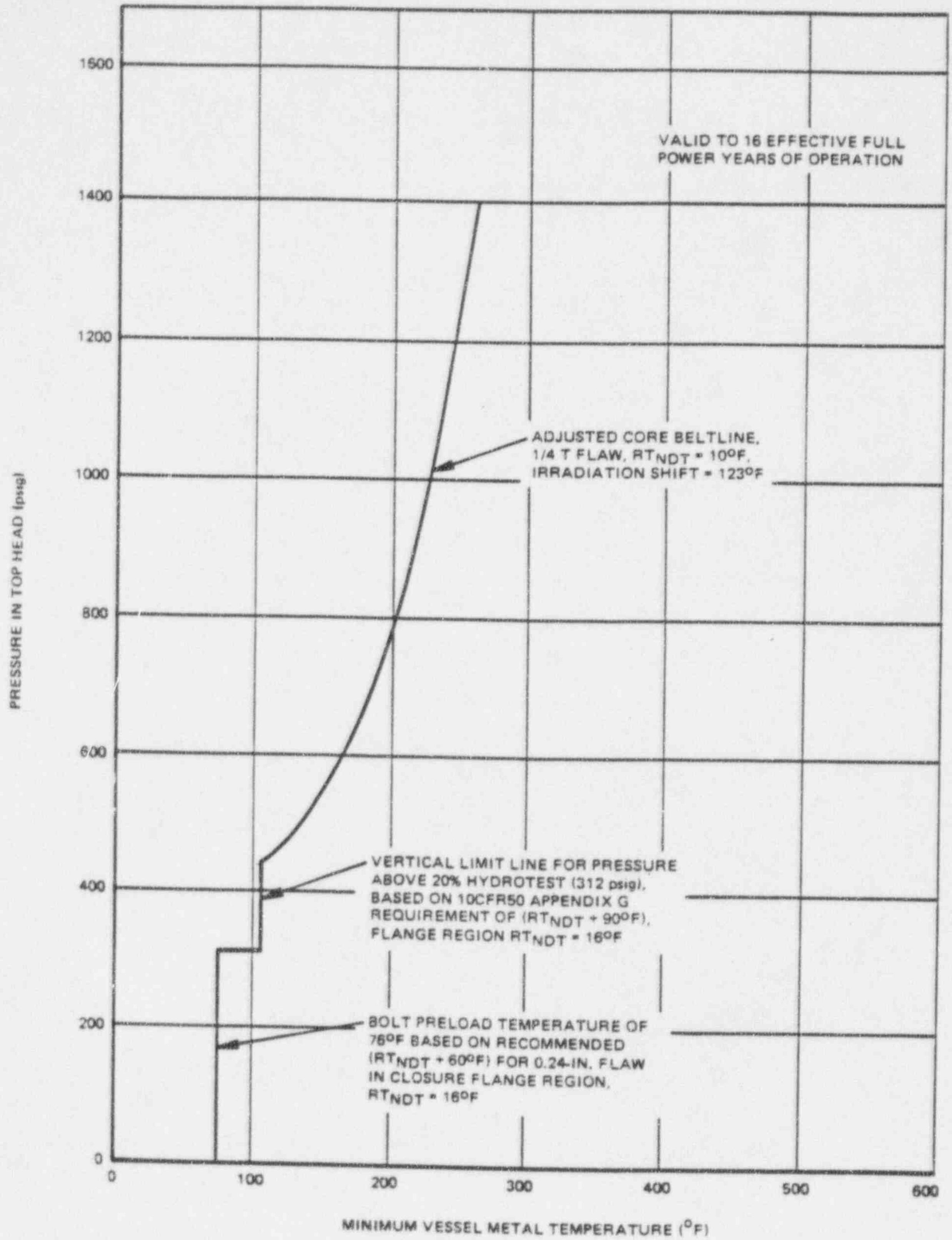

Figure 7-5. Pressure versus Minimum Temperature for Pressure Tests, Based on Surveillance Test Results

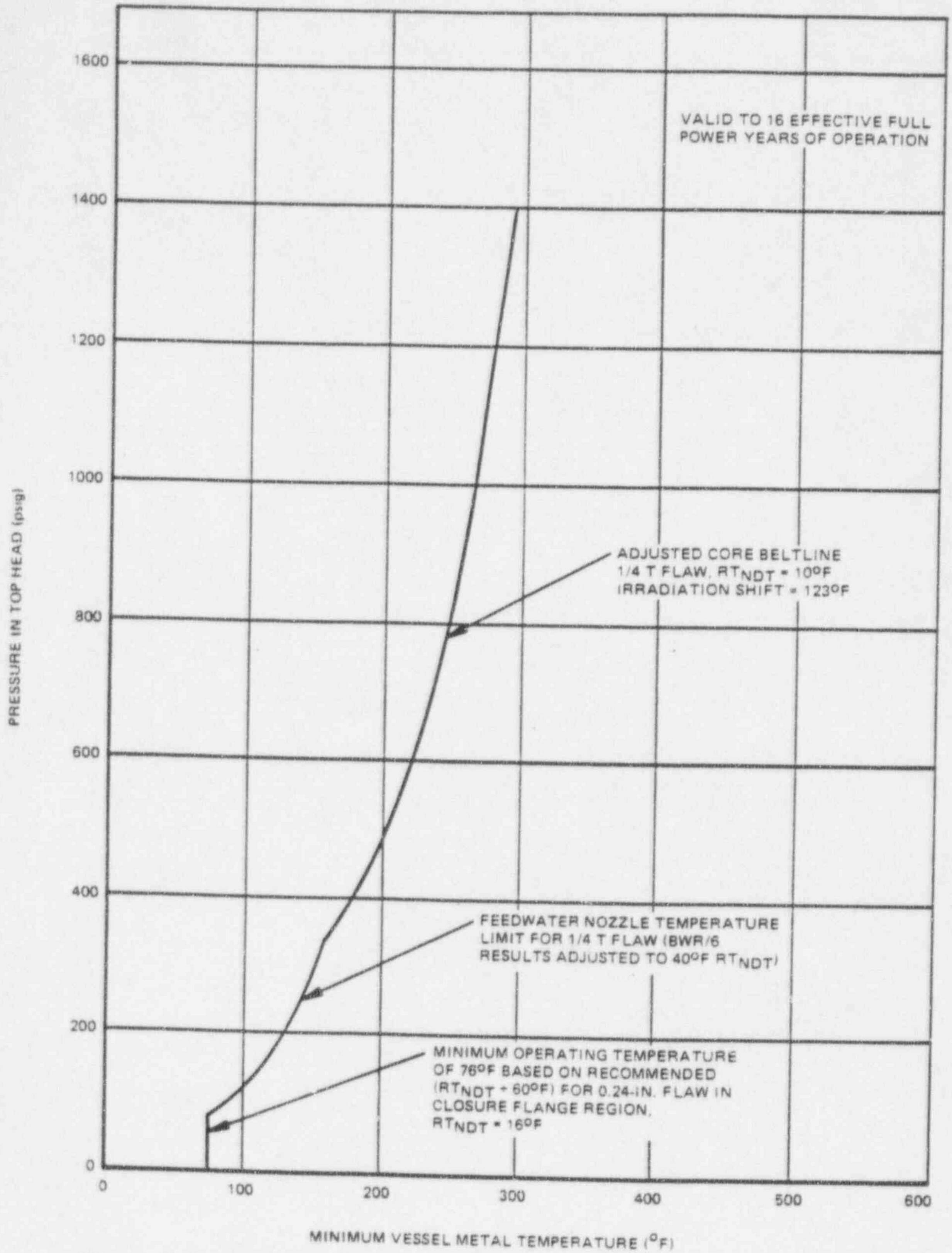

Figure 7-6. Pressure versus Minimum Temperature for Non-Nuclear Heatup/Cooldown and Low Power Physics Tests, Based on Surveillance Test Results

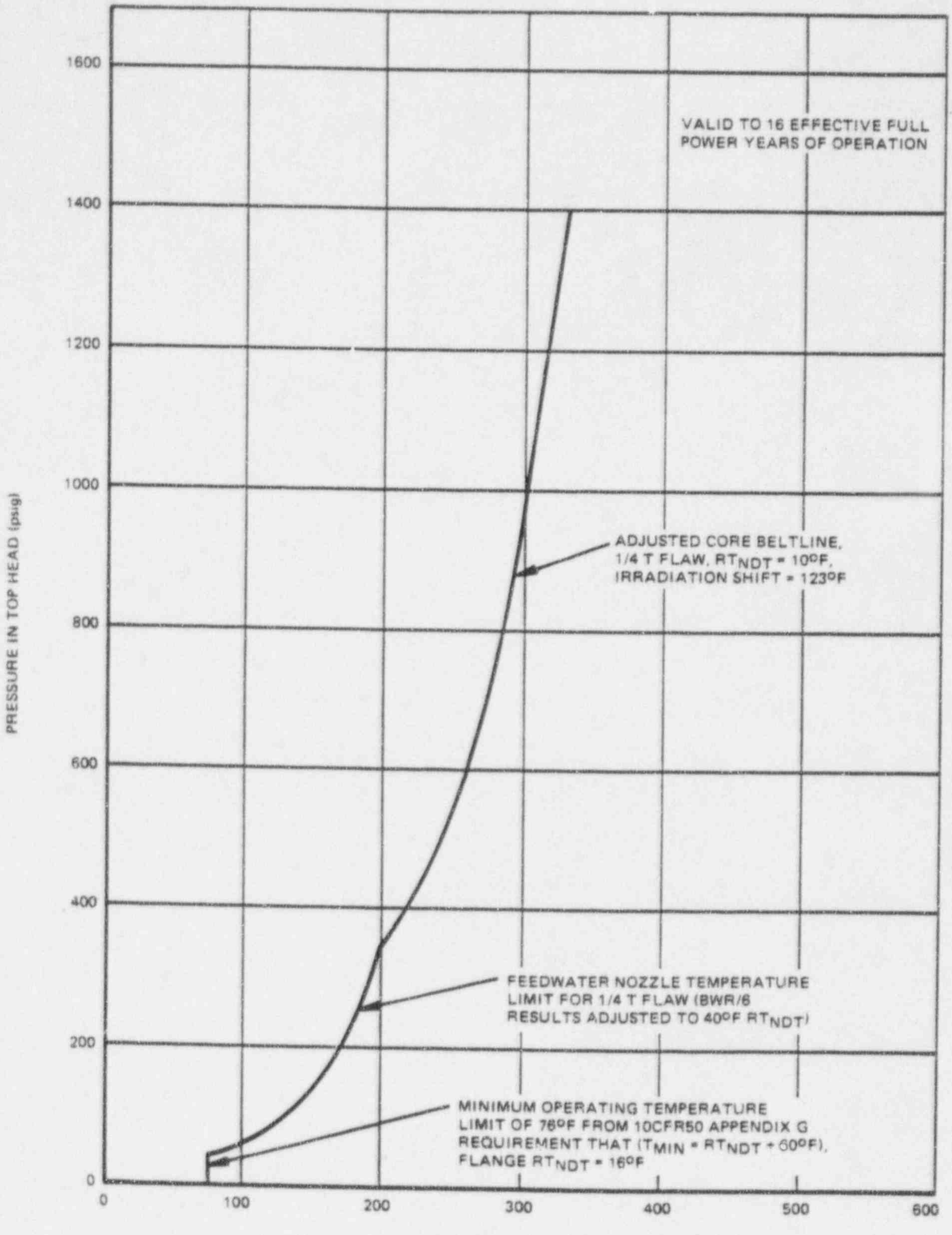

Figure 7-7. Pressure versus Minimum Temperature for Core Critical Operation, Based on Surveillance Test Results

Enclosure 2

Edwin I. Hatch Nuclear Plant
Response to Second RFAI
Power Uprate Submittal

10 CFR 50.92 Evaluation

The NRC has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists in a proposed license amendment. A proposed license amendment does not involve a significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in the margin of safety.

Georgia Power Company has reviewed the proposed license amendment and Technical Specifications changes and has determined that its adoption does not involve a significant hazards consideration. The basis for this determination is given below.

Description of the Proposed Changes

The proposed amendment increases the licensed core thermal power from 2436 MWt to 2558 MWt, which represents an increase of 5% over the current licensed power level. This request is in accordance with the generic boiling water reactor (BWR) power uprate program established by the General Electric Company (GE) and approved by the U.S. Nuclear Regulatory Commission (NRC) staff in a letter from W. T. Russell, NRC, to P. W. Marriott, GE, dated September 30, 1991. Implementation of the proposed power uprate at Plant Hatch will result in an increase of steam flow to approximately 106% of the current value but will require no changes to the basic fuel design. Core reload design and fuel parameters will be modified as power uprate is implemented to support the current 18-month reload cycle. The higher power level will be achieved by expanding the power/flow map and by slightly increasing reactor vessel dome pressure. The maximum core flow limit will not be increased over the pre-uprate value. Implementation of this proposed power uprate will require minor modifications, such as, resetting the safety relief setpoints, as well as calibrating of plant instrumentation to reflect the uprated power. Plant operating, emergency, and other procedure changes will be made where necessary to support uprated operation.

Basis for No Significant Hazards Consideration Determination

1. **Will the changes involve a significant increase in the probability or consequences of an accident previously evaluated?**

   A. Rated Thermal Power is increased to 2558 MWt on page 3 of the Unit 1 Operating License, page 4 of the Unit 2 Operating License, and in Section 1.1 (Definitions) of the Units 1 and 2 Technical Specifications.

   Evaluation

   The changes in the Operating Licenses and Technical Specifications were evaluated and it was determined that the probability (frequency of occurrence) of design basis accidents occurring is not affected by the increased power level, as the regulatory criteria established for plant equipment (e.g., ASME Code, IEEE standards, NEMA standards, Regulatory Guide criteria) will still be complied with at the uprated power level. Scram setpoints (equipment settings that initiate automatic plant shutdowns) will be established such that there is no significant increase in scram frequency due to uprate. No new challenges to safety-related equipment will result from power uprate.

   The changes in consequences of hypothetical accidents which would occur from 102% of the uprated power, compared to those previously evaluated, are in all cases insignificant, because the power uprate accident evaluations will not result in exceeding any NRC-approved acceptance limits. Enclosure 4 of Reference 1, General Electric Report NEDC-32405P, "Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2," December 1994, investigated the spectrum of hypothetical accidents and transients, and showed the plant's current regulatory criteria are satisfied at power uprate. For example, in the area of core design, the fuel operating limits will still be met at the uprated power level, and fuel reload analyses will show plant transients meet the criteria accepted by the NRC as specified in NEDO-24011, "GESTAR II." Challenges to fuel or emergency core cooling system (ECCS) performance were evaluated (Section 4.2 of NEDC-32405P) and shown to still meet the criteria of 10 CRF 50.46 and Appendix K. Challenges to the containment were evaluated (Section 4.1 of NEDC-32405P) and shown to still meet 10 CFR 50 Appendix A, Criterion 38, Long Term Cooling, and Criterion 50, Containment. Radiological release events were evaluated (Section 9.2 of NEDC-32405P) and shown to meet the criteria of 10 CFR 100 (Unit 1 FSAR Chapter 14 and Unit 2 FSAR Chapter 15).

   The results of the analyses discussed above demonstrate that operation at the power uprate level does not significantly increase the probability or consequences of an accident previously evaluated.

B. The surveillance test discharge pressure for the standby liquid control pump at 41.2 gpm is increased from 1190 psig to 1201 psig. This value appears in Surveillance Requirement (SR) 3.1.7.7 and the corresponding Bases Section B 3.1.7 in the Unit 1 and Unit 2 Technical Specifications.

<u>Evaluation</u>

Power uprate operation will result in a 30 psi increase in reactor operating pressure. As will be discussed in these proposed changes, several pressure-dependent setpoints (including safety relief valve [SRV] setpoints) will be increased to preserve current margins. Increasing the pressure 11 psi, at which a 41.2 gpm flow rate is developed, assures continued conformance to anticipated transient without scram (ATWS) criteria at uprated conditions. The surveillance test pressure is based on the maximum pressure for an ATWS event during the time period when the standby liquid control pump is in operation. Section 6.5 of NEDC-32405P discusses the capability of these positive displacement pumps. A small increase in the SRV setpoints will have no effect on the rated injection flow to the reactor. This change, therefore, will not increase the probability or consequences of a previously evaluated accident.

C. The reactor vessel steam dome high pressure allowable value for reactor protection system (RPS) instrumentation is increased 31 psi, consistent with the nominal pressure increase for power uprate. The allowable value appears in Section 3.3.1.1, Table 3.3.1.1-1, Function 3, in the Unit 1 and Unit 2 Technical Specifications.

<u>Evaluation</u>

The reactor vessel steam dome high pressure scram limit is increased because the steam dome operating pressure is increased. Operating pressure for uprated power is increased to assure that satisfactory reactor pressure control is maintained. The operating pressure was chosen on the basis of steam line pressure drop characteristics and the steam flow capability of the turbine. Satisfactory reactor pressure control requires an adequate flow margin between the uprated operating condition and the steam flow capability of the turbine control valves at their maximum stroke. An operating dome pressure of 1035 psig, which is 30 psi higher than the current operating dome pressure, is expected. Therefore, the high pressure scram is increased approximately the same amount to preserve existing margins to reactor trips.

The high pressure scram terminates a pressurization transient not terminated by direct scram or high neutron flux scram. The setting is maintained above the nominal reactor vessel operating pressure and below the specified analytical trip

limit used in the safety analyses. The revised high pressure scram setpoint will preserve the hierarchy of pressure setpoints. This means that the high pressure scram setpoint will remain below the opening setpoint of the SRVs. The SRV nominal setpoints are also increased 30 psi, as discussed in Item G below. This hierarchy of setpoints provides assurance that the probability of opening more than one SRV without scram intervention is low.

Since the scram function and the current margins to trip avoidance are maintained with revised setpoints, there is no significant increase in the probability or consequences of an accident previously evaluated.

D. The ATWS reactor vessel steam dome high pressure recirculation pump trip (RPT) allowable value is raised 80 psi. The allowable value appears in Section 3.3.4.2, SR 3.3.4.2.3, in the Unit 1 and Unit 2 Technical Specifications.

<u>Evaluation</u>

The ATWS-RPT high pressure setpoint initiates a trip of the recirculation pumps, thereby adding negative reactivity following events in which a scram does not (but should) occur. Section 5.1.3.2 of NEDC-32405P discusses this function in detail.

The current analytical limit for the ATWS-RPT high pressure trip is 1150 psig. This value was increased 30 psi in the power uprate ATWS safety evaluations to account for the 30 psi increase in vessel operating pressure, SRV setpoints, etc. The current allowable value in the Technical Specifications is 1095 psig. This allowable value was not set by the current analytical limit, but by the range of the installed pressure instruments. As part of the power uprate plant changes, these pressure instruments will be replaced to accommodate higher pressure, and the allowable value, in conjunction with the analytical limit used in the safety analysis, will be increased.

Sections 5.1 and 9.3 of NEDC-32405P show the system can adequately perform its ATWS function with the new setpoint. Therefore, the proposed change does not cause a significant increase in the probability or consequences of an accident previously evaluated.

E. The low-low set (LLS) SRV arming pressure allowable value is increased 31 psi, consistent with the increase in operating pressure and high pressure scram allowable value. The LLS arming pressure allowable value appears in Section 3.3.6.3, Table 3.3.6.3-1, Function 1, in the Unit 1 and Unit 2 Technical Specifications.

Evaluation

The allowable value for the LLS SRV high pressure arming setpoint is increased because the high pressure scram setpoint is increased. No changes to the LLS arming logic associated with the SRV tailpipe pressure switches and the LLS opening and closing pressure setpoints are proposed.

The LLS relief logic mitigates the postulated containment loads of subsequent SRV actuations during small or intermediate loss of coolant accidents (LOCAs) by extending the time between actuations. The LLS logic requires two separate signals to arm itself for operation. Specifically, the LLS logic arms when an SRV opens (i.e., tailpipe pressure switch) and reactor pressure concurrently exceeds the scram setpoint. To preserve the hierarchy of pressure setpoints, the high pressure input to the LLS SRV arming logic has the same setpoint as the high pressure scram, thus minimizing the potential for a spurious SRV opening through the LLS logic without occurrence of a reactor scram.

Increasing the arming setpoint is consistent with increasing the high pressure scram setpoint and will not increase the probability or consequences of an accident previously evaluated.

F. Lower the permissible rod line for single-loop operation (SLO) below 45 percent core flow from the 80 percent rod line to the 76 percent rod line. This Technical Specifications limit appears in Section 3.4.1 (Figure 3.4.1-1) and the corresponding Bases Section B 3.4.1 of the Unit 1 and Unit 2 Technical Specifications.

Evaluation

During development of the generic power uprate program, GE and the NRC agreed to maintain the current exclusion region in the power-to-flow map related to thermal-hydraulic stability. The current limit for SLO is the 80 percent rod line. Power uprate will redefine 100 percent rated power and, therefore, rated rod or flow control lines. The 76 percent rod line at uprated conditions closely corresponds on an absolute, rather than percentage basis, to the existing 80 percent rod line.

Therefore, this proposed Technical Specifications change ensures that power uprate operation will not cause a significant increase in the probability or consequences of accident previously evaluated.

G. The SRV lift setpoints in the Units 1 and 2 Technical Specifications SR 3.4.3.1 will be increased 30 psi.

Evaluation

The SRVs are designed to prevent overpressurization of the reactor pressure vessel during abnormal operational transients. The SRV lift setpoints are increased to accommodate the increase in operating pressure that accompanies power uprate. The increase in SRV setpoints ensures that adequate margins are maintained so that the increase in dome pressure during normal operation does not result in an increase in the number of unnecessary SRV actuations. The setpoint increase also maintains the hierarchy of pressure setpoints described in these proposed changes. Transient evaluations include a + 3 percent tolerance to the nominal setpoints. As described in Section 3.2 of NEDC-32405P, peak vessel pressure increases by 3 percent, but remains well below the 1375 psig ASME Code limit.

Although not credited in the transient analysis, GPC installed a pressure transmitter system which can electronically actuate the SRVs on high vessel pressure. The nominal trip setpoints for its actuation correspond with the nominal mechanical lift setpoints in the Technical Specifications. The SRV pressure transmitter system nominal setpoints will also be increased 30 psi.

General Electric generically evaluated the adequacy of BWR SRVs to operate at uprated temperatures and pressures. The reactor operating pressure and temperature increases of less than 40 psi and 5°F, respectively, used in that evaluation bound the uprated Hatch operating conditions.

The impact of power uprate on the Hatch containment dynamic loads due to SRV discharge has also been evaluated. As discussed in Section 4.1.2 of NEDC-32405P, the vent thrust loads with power uprate were calculated to be less than the loads used in the containment analysis. The effects of power uprate on SRV air-clearing, the discharge line, the pool pressure boundary, and submerged structure drag loads are discussed in Section 4.1.2 of NEDC-32405P which concludes that the small increase in the setpoint pressure is well within the margin in the SRV loads defined in the Mark I Containment Long-Term Program. Therefore, power uprate does not impact the Hatch SRV load definitions used in the containment analysis, and no significant increase in the probability or consequences of an accident previously evaluated is caused by this proposed change.

H. The Limiting Condition for Operation (LCO) and SRs for the maximum reactor steam dome pressure will be increased from 1020 psig to 1058 psig. This requirement appears in LCO 3.4.10, SR 3.4.10.1, and the corresponding Bases in the Unit 1 and Unit 2 Technical Specifications.

Evaluation

As discussed in the Technical Specifications Bases and NEDC-32405P, the maximum reactor dome pressure is an initial condition of the vessel overpressure protection analysis, which assumes a fast isolation of all four main steam lines by the main steam isolation valves (MSIVs). The reactor scram signal generated directly by the valve closure is assumed defeated for this analysis. Instead, the scram signal is generated by high neutron flux. The overpressure analysis for power uprate assumed an initial dome pressure of 1058 psig, which represents an increase of 38 psig. This initial pressure was chosen approximately 2 percent above the 1035 psig steam dome operating pressure expected for power uprate operation. The analysis also included the other changes (including SRV setpoints) discussed in these proposed changes. Therefore, there is no significant increase in the probability or consequences of an accident previously evaluated.

I. The HPCI and RCIC surveillance test pressures in Units 1 and 2 Technical Specifications SRs 3.5.1.8 and 3.5.3.3, respectively, are increased 38 psi.

Evaluation

The allowable HPCI and RCIC surveillance test pressure is increased to correspond with the increase in normal reactor operating pressure and LCO/SR on maximum reactor pressure that accompanies power uprate. (As discussed in Item H above, the LCO on reactor steam dome pressure is increased 38 psi.) The change is needed to ensure that pressure and power reductions are not required to perform surveillance testing. The requested changes will allow the quarterly demonstration of the HPCI and RCIC systems' capability to perform at normal reactor operating pressures, which meets the original intent of the Technical Specifications.

The HPCI and RCIC systems have been evaluated and demonstrated to be capable of injecting design flow rate at the higher reactor pressure as discussed in Sections 4.2 and 3.8 of NEDC-32405P and in Reference 2.

Therefore, these changes will ensure that power uprate operation will not cause a significant increase in the probability or consequences of an accident previously evaluated.

J. Bases Changes

Several changes to the Hatch Units 1 and 2 Technical Specifications Bases are proposed for consistency with the power uprate safety analyses. These proposed

changes are in addition to the Bases changes corresponding to proposed changes A through I.

i. The main steam line flow differential pressure setpoints (Bases Section B 3.3.6.1.c) and the HPCI/RCIC high flow differential pressure setpoints (Bases Section B 3.3.6.3.a and B 3.3.6.4.a) are changed for both units.

   The allowable values (in percent of rated) will not change for power uprate operation. However, the actual differential pressure will change due to the increase in steam flow and pressure.

ii. The HPCI and RCIC upper design pressure in Bases Sections B 3.5.1 and B 3.5.3, respectively, is increased 34 psi for both units.

   The Bases changes support the design of these high pressure systems to pump rated flow from approximately 150 psig up to a pressure associated with the first group of SRV setpoints. This proposed design pressure conservatively considers the 30 psi higher nominal setpoints and 3 percent setpoint drift. The capability of the HPCI and RCIC systems to deliver design flows at these pressures is discussed in Reference 2, and was reviewed by GE for the Unit 1 and Unit 2 systems.

   Note that the upper design pressure for HPCI and RCIC is different from the surveillance test pressure for HPCI and RCIC discussed previously in item I. The maximum surveillance test pressure corresponds to reactor operating pressure since the surveillance test is performed when the unit is operating. The HPCI and RCIC upper design pressure reflects the capability to inject water to the vessel following a reactor scram and isolation.

iii. The peak post accident containment pressure (P_a) is changed to 49.6 psig (Unit 1) and 45.5 psig (Unit 2). These values appear in Bases Sections B 3.6.1.1, B 3.6.1.2, and B 3.6.1.4 in each unit's Technical Specifications.

   Section 4.1.1.3 of NEDC-32405P discusses the peak short-term containment pressure response which was recalculated for power uprate conditions. Containment pressure and temperatures remain below design limits and are essentially unchanged.

iv. The main condenser offgas gross gamma activity rate limit of 240 mci/second will not be changed for power uprate. A statement that the current limit is conservative for power uprate conditions was added to Bases Section 3.7.6 for both units.

The Bases derive the current 240 mci/second limit using a rated core thermal power limit of 2436 MWt. A slightly higher limit could be justified using the uprated power level. However, adequate margin exists with the current limit.

v. The inservice hydrostatic and leak testing pressures shown in Bases Section 3.10.1 are increased 33 psi and 30 psi, respectively. This change affects each unit's Bases.

This change is a direct result of the 30 psi increase in normal operating pressure proposed for power uprate. The leakage test is normally performed at operating pressure and the hydrostatic test at approximately 110 percent of operating pressure.

The above Bases changes Items i-v have been evaluated and will not increase the probability or consequences of an accident previously evaluated.

2. Will the changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Evaluation

The Operating License changes in power level and the associated Technical Specifications changes discussed previously will not create the possibility of a new or different kind of accident from any accident previously evaluated, as summarized below.

Equipment that could be affected by power uprate was evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode were identified. The full spectrum of accident considerations defined in RG 1.70 was evaluated, and no new or different kind of accident was identified. Uprate uses already-developed technology and applies it within the capabilities of existing plant equipment in accordance with presently existing regulatory criteria to include NRC-approved codes, standards, and methods. GE has designed BWRs of higher power levels than the uprated power of any of the currently operating BWR fleet, and no new power dependent accidents have been identified.

The Technical Specifications changes required to implement power uprate require only minor modifications to the plant's configuration. All changes were evaluated and found to be acceptable.

3. Will the changes involve a significant reduction in the margin of safety?

A. Rated Thermal Power is increased to 2558 MWt on page 3 of the Unit 1 Operating License, page 4 of the Unit 2 Operating License, and in Section 1.1 (Definitions) of the Unit 1 and Unit 2 Technical Specifications.

Evaluation

The events analyzed in the FSAR were re-evaluated to demonstrate that power uprate can be implemented without exceeding any regulatory limit. Because the applicable safety analysis criteria and limits are satisfied for power uprate, the margin of safety associated with the safety limits and other limits identified in the Technical Specifications will be maintained.

As discussed in NEDC-32405P, the safety margins prescribed by the Code of Federal Regulations are maintained by meeting the appropriate regulatory criteria. Similarly, the margins provided by the application of the ASME design criteria are maintained. Section 11.4.2 of NEDC-32405P discusses the effects of power uprate on safety margins for the following:

- Fuel thermal limits.
- Design basis accidents and the challenges to fuel, containment, and radiological releases.
- Transient analyses.
- Non-LOCA radiological releases.
- Environmental consequences.

These evaluations conclude that applicable safety analysis criteria and limits are satisfied, and thus, the margin of safety will not be significantly reduced.

B. The surveillance test discharge pressure for the SLC pump at 41.2 gpm is increased from 1190 psig to 1201 psig. This value appears in SR 3.1.7.7 and corresponding Bases Section B 3.1.7 in the Unit 1 and Unit 2 Technical Specifications.

Evaluation

Power uprate operation will result in a 30 psi increase in reactor operating pressure. Several pressure-dependent setpoints (including SRV setpoints) will be increased to preserve current margins. Increasing the pressure 11 psi, at which a 41.2 gpm flow rate is developed, assures continued conformance to ATWS criteria at uprated conditions. The surveillance test pressure is based on the maximum pressure for an ATWS event during the time period when the SLC pump is in operation. Section 6.5 of NEDC-32405P discusses the capability of these positive

displacement pumps. A small increase in the SRV setpoints will have no effect on the rated injection flow to the reactor.

For power uprate, the capability of the SLCS to respond with adequate margin to an ATWS event was confirmed. The results are reported in Section 9.3.1 of NEDC-32405P. The limiting ATWS event was an inadvertent MSIV closure. The event was reanalyzed at uprate conditions with the higher SRV setpoints and ATWS-RPT setpoints. Peak vessel pressure was well below the ASME emergency limit of 1500 psig. The effect of power uprate on peak clad temperature and maximum suppression pool temperature was judged to be negligible, because the calculations showed no increase in fuel surface heat flux or integrated SRV flow.

In summary, all ATWS criteria are satisfied and the SLC pumps are capable of injecting the required amounts of sodium pentaborate at uprated conditions. Therefore, there is no significant decrease in the margin of safety.

C. The reactor vessel steam dome high pressure allowable value for RPS instrumentation is increased 31 psi, consistent with the nominal pressure increase for power uprate. The allowable value appears in Section 3.3.1.1, Table 3.3.1.1-1, Function 3, in the Unit 1 and Unit 2 Technical Specifications.

Evaluation

The reactor vessel steam dome high pressure scram limit is increased because the steam dome operating pressure is increased. Operating pressure for uprated power is increased to assure that satisfactory reactor pressure control is maintained. The operating pressure was chosen on the basis of steam line pressure drop characteristics and the steam flow capability of the turbine. Satisfactory reactor pressure control requires an adequate flow margin between the uprated operating condition and the steam flow capability of the turbine control valves at maximum stroke. An operating dome pressure of 1035 psig, which is 30 psi higher than the current operating dome pressure, is expected. Therefore, the high pressure scram is increased approximately the same amount to preserve existing margins to reactor trips.

The increases in the steam dome high pressure scram instrument setpoints for uprated power were evaluated by determining whether the high pressure scram, which is used as a backup to other scram signals, provides adequate overpressure protection. The evaluation demonstrates that the backup protection function, with the revised setpoints, continues to provide adequate overpressure protection at uprated power conditions by meeting the applicable ASME Code criteria. Therefore, there is no significant decrease in the margin of safety.

D. The ATWS reactor vessel steam dome high pressure RPT allowable value is raised 80 psi. The allowable value appears in Section 3.3.4.2, SR 3.3.4.2.3, in the Unit 1 and Unit 2 Technical Specifications.

Evaluation

The ATWS-RPT high pressure setpoint initiates a trip of the recirculation pumps, thereby adding negative reactivity following events in which a scram does not (but should) occur. Section 5.1.3.2 of NEDC-32405P discusses this function in detail.

For power uprate, the capability of the SLCS to respond to a postulated ATWS event with adequate margin was confirmed (Section 9.3.1 of NEDC-32405P). By reducing reactor power until the SLCS can inject the required amounts of sodium pentoborate to achieve full shutdown, the RPT also reduces suppression pool temperature for isolation cases (also shown to be acceptable for power uprate conditions in Section 9.3.1 of NEDC-32405P). Therefore, there is no significant decrease in a margin of safety.

E. The LLS SRV arming pressure allowable value is increased 31 psi, consistent with the increase in operating pressure and high pressure scram allowable value. The LLS arming pressure allowable value appears in Section 3.3.6.3, Table 3.3.6.3-1, Function 1, in the Unit 1 and Unit 2 Technical Specifications.

Evaluation

The allowable value for the LLS SRV high pressure arming setpoint is increased, because the high pressure scram setpoint is increased. No changes to the LLS arming logic associated with the SRV tailpipe pressure switches, and the LLS opening and closing pressure setpoints are proposed.

Since this proposed change only affects one of two arming signals for LLS, the safety analyses are not affected; therefore, there is not a significant change in the margin of safety.

F. Lower the permissible rod line for SLO below 45 percent core flow from the 80 percent rod line to the 76 percent rod line. This Technical Specifications limit appears in Section 3.4.1 (Figure 3.4.1-1) and corresponding Bases Section B 3.4.1 of the Unit 1 and Unit 2 Technical Specifications.

Evaluation

This change to the power versus flow map restricted zone is made to maintain the same operating constraints and stability margin that were established for the current power level. This change avoids any increase in the possibility of occurrence or any increase in the potential effects of power oscillations. Therefore, there is no significant decrease in a margin of safety.

G. The SRV lift setpoints in Surveillance Requirement 3.4.3.1 (both units) will be increased 30 psi.

Evaluation

The SRVs are designed to prevent overpressurization of the reactor pressure vessel during abnormal operational transients. The SRV lift setpoints are increased to accommodate the increase in operating pressure that accompanies power uprate. The increase in SRV setpoints ensures that adequate margins are maintained so that the increase in dome pressure during normal operation does not result in an increase in the number of unnecessary SRV actuations. The setpoint increase also maintains the hierarchy of pressure setpoints described in these proposed changes. Transient evaluations include a + 3 percent tolerance to the nominal setpoints. As described in Section 3.2 of NEDC-32405P, peak vessel pressure increases by 3 percent but remains well below the 1375 psig ASME Code limit. Therefore, there is no significant decrease in the margin of safety.

H. The Limiting Condition for Operation (LCO) and Surveillance Requirements for the maximum reactor steam dome pressure will be increased from 1020 psig to 1058 psig. This requirement appears in LCO 3.4.10, SR 3.4.10.1, and the corresponding Bases in the Unit 1 and Unit 2 Technical Specifications.

Evaluation

As discussed in the Technical Specifications Bases and in Section 3.2 of NEDC-32405P, the maximum reactor dome pressure is an initial condition of the vessel overpressure protection analysis, which assumes a fast isolation of all four main steam lines by the main steam isolation valves. It is also used as a sensitivity study parameter for certain transient and LOCA events.

With this revised limit, peak vessel pressure remains below ASME Code criteria, transient limits are maintained, and LOCA fuel performance satisfies the requirements of 10 CFR 50.46 and 10 CFR 50, Appendix K. Therefore, there is no significant decrease in a margin of safety.

I. The HPCI and RCIC surveillance test pressures in SRs 3.5.1.8 and 3.5.3.3, respectively, (both units) are increased 38 psi.

<u>Evaluation</u>

The allowable HPCI and RCIC surveillance test pressure is increased to correspond with the increase in normal reactor operating pressure and LCO/SR on maximum reactor pressure that accompanies power uprate. (As discussed previously, the LCO on reactor steam dome pressure is increased 38 psi.)

The purpose of the HPCI and RCIC surveillance test is to provide periodic demonstration of the systems' ability to perform consistent with the requirements of the analyses at the higher operating pressure associated with power uprate conditions. An evaluation of the HPCI and RCIC systems confirmed their ability to operate at slightly higher turbine speed and provide design flow at power uprate conditions. System performance will be confirmed during the initial power ascension to uprated conditions (and periodically thereafter per the Technical Specifications). Therefore, there is no significant decrease in the margin of safety.

J. <u>Bases Changes</u>

Several changes to the Hatch Units 1 and 2 Technical Specifications Bases are proposed for consistency with the power uprate safety analyses. These proposed changes are in addition to the Bases changes corresponding to proposed changes A through I.

i. The main steam line flow differential pressure setpoints, as shown in Bases Section B 3.3.6.1.c, and the HPCI/RCIC high flow differential pressure setpoints (Units 1 and 2 Bases Sections B 3.3.6.3.a and B 3.3.6.4.a) are changed.

   The allowable values (in percent of rated) will not change for power uprate operation. However, the actual differential pressure will change due to the increase in steam flow and pressure.

ii. The HPCI and RCIC upper design pressure in Units 1 and 2 Bases Sections B 3.5.1 and B 3.5.3, respectively, is increased 34 psi.

   The Bases changes support the design of these high pressure systems to pump rated flow from approximately 150 psig up to a pressure associated with the first group of SRV setpoints. This proposed design pressure conservatively considers the 30 psi higher nominal setpoints and 3 percent setpoint drift. The

capability of the Unit 1 and Unit 2 HPCI and RCIC systems to deliver design flows at these pressures was reviewed by GE and is discussed in Reference 2.

iii. The peak post accident containment pressure (P_a) is changed to 49.6 psig (Unit 1) and 45.5 psig (Unit 2). These values appear in Units 1 and 2 Bases Sections B 3.6.1.1, B 3.6.1.2, and B 3.6.1.4.

Section 4.1.1.3 of NEDC-32405P discusses the peak short-term containment pressure response which was recalculated for power uprate conditions. Containment pressure and temperatures remain below design limits and are essentially unchanged.

iv. The main condenser offgas gross gamma activity rate limit of 240 mci/second will not be changed for power uprate. A statement that the current limit is conservative for power uprate conditions was added to Units 1 and 2 Bases Section 3.7.6.

The Bases derive the current 240 mci/second limit using a rated core thermal power limit of 2436 MWt. A slightly higher limit could be justified using the uprated power level. However, adequate margin exists with the current limit.

v. The inservice hydrostatic and leak testing pressures shown in Units 1 and 2 Bases Section 3.10.1 are increased 33 psi and 30 psi, respectively.

This change is a direct result of the 30 psi increase in normal operating pressure proposed for power uprate. The leakage test is normally performed at operating pressure and the hydrostatic test at approximately 110 percent of operating pressure.

The above Bases changes i-v were evaluated, and there is no significant decrease in the margin of safety.

References

1. Georgia Power Company letter HL-4724, J. T. Beckham, Jr., to NRC, "Power Uprate Operation," dated January 13, 1995.

2. Georgia Power Company letter HL-4812, J. T. Beckham, Jr., to NRC, "Response to Request for Additional Information - Power Uprate Submittal," dated April 5, 1995.

Environmental Assessment

This Environmental Assessment is similar to the one contained in Georgia Power Company's' submittal dated January 13, 1995 (Reference 1). Information has been added based on GPC's Reference 2 submittal and formatted similar to Reference 3.

Georgia Power Company has evaluated the potential radiological and nonradiological environmental impacts associated with this proposed action. The conclusions are that the proposed uprate will not result in a significant adverse environmental impact and is not an unreviewed environmental question.

Identification of the Proposed Action

This Environmental Assessment addresses potential environmental issues related to GPC's application to amend the Plant Hatch Units 1 and 2 Operating Licenses. The proposed amendment increases the licensed core thermal power from 2436 MWt to 2558 MWt, which represents an increase of 5 percent over the current licensed power level. This request is in accordance with the generic boiling water reactor (BWR) power uprate program established by the General Electric Company (GE) and approved by the U.S. Nuclear Regulatory Commission (NRC) staff in a letter from W. T. Russell, NRC, to P. W. Marriott, GE, dated September 30, 1991. Implementation of the proposed power uprate at Plant Hatch will result in an increase of steam flow to approximately 106 percent of the current value, but will not require changes to the basic fuel design. Core reload design and fuel parameters will be modified as power uprate is implemented to support the current 18-month reload cycle. The higher power level will be achieved by expanding the power/flow map and slightly increasing reactor vessel dome pressure. The maximum core flow limit will not be increased over the pre-uprate value. Implementation of this proposed power uprate will require minor modifications, such as resetting of the safety relief setpoints, as well as calibrating plant instrumentation to reflect the uprated power. Plant operating, emergency, and other procedure changes will be made where necessary to support uprated operation.

The proposed action involves NRC issuance of a license amendment to uprate the authorized power level by changing the Operating Licenses, including Appendix A (Technical Specifications). Appendix B of the Operating License does not require revision as a result of power uprate.

The Need for the Proposed Action

The proposed action would authorize GPC to increase the potential electrical output of Plant Hatch by approximately 40 megawatts per unit and thus would provide additional electrical power to service domestic and commercial areas of GPC's grid.

Environmental Impacts of the Proposed Action

The "Final Environmental Statement" (FES) related to operation of Plant Hatch Units 1 and 2 (Reference 6) evaluates the nonradiological impact of operation at a maximum design reactor power level of 2537 MWt per unit. By letter dated January 13, 1995 (Reference 1), GPC submitted the proposed amendment to implement power uprate for Hatch Units 1 and 2 which is the subject of this environmental assessment. Enclosure 2 of that submittal provided information on the nonradiological environmental aspects of the amendment request. Enclosure 4 was the Plant Hatch power uprate licensing report (GE report NEDC-32405P) which provided information on the radiological environmental impact of power uprate.

Georgia Power Company concludes that the proposed amendment allowing power uprate operation will not have a significant impact on the environment and that the change does not constitute an unreviewed environmental question. The nonradiological and radiological effects of the proposed action on the environment are described below.

Nonradiological Environmental Assessment

Power uprate will not change the method of generating electricity nor the method of handling any influents from the environment or effluents to the environment. Therefore, no new or different types of environmental impacts are expected.

The detailed evaluation presented below and in Reference 1 concludes that nonradiological parameters affected by power uprate will remain within the bounding conditions cited in the FES, which concludes that no significant environmental impact will result from operation of Plant Hatch. This conclusion remains valid for power uprate.

The FES evaluated the nonradiological impact at a maximum design reactor power level of 2537 MWt per unit (approximately 104 percent of the current licensed power level). The parameters evaluated in the Environmental Report and the subsequent FES (References 4 through 6) were re-evaluated at 2558 MWt to determine whether the proposed change is significant relative to adverse environmental impact. Table E2-1 of Reference 1 provided a comparison of environmental-related operation parameters at rated and uprated power.

Both units at Plant Hatch utilize a closed-loop circulating water system and forced air cooling towers for dissipating heat from the main turbine condenser. Other equipment is cooled by the plant service water (PSW) and residual heat removal (RHR) service water systems. The cooling towers and service water systems are operated in accordance with the requirements of the National Pollutant Discharge Elimination System (NPDES) Permit No. GA 0004120, which expires October 31, 1997. No notification changes or other action relative to the NPDES Permit are required.

The withdrawal of cooling water from the Altamaha River is expected to increase slightly, primarily due to the increase in the evaporation rate from the cooling towers. Emergency system flows are expected to remain generally unchanged. Although increased heat loads are expected for nonsafety-related loads, such as the main generator stator coolers, hydrogen coolers, and exciter coolers, heat loads will remain within the existing design heat loads of the service water systems.

The circulating water system design flow rate is the primary basis for determining makeup water for the Plant Hatch cooling towers. Other factors affecting tower makeup are tower performance and meteorological conditions. Based on the review of cooling tower performance parameters associated with power uprate, the design flow rate of the cooling towers will not change. Makeup requirements may increase slightly due to increased heat load on the towers and the associated increase in evaporation. As discussed in Enclosure 2 of Reference 1, the increase in makeup (withdrawal) rate is expected to be approximately 5 percent or 500 gpm. This projected increase associated with the uprate is not significant and is enveloped by the river water withdrawal rates discussed in the FES and the rates approved under the current Georgia Surface Water Withdrawal Permit for Plant Hatch. Intake canal velocity will not be significantly affected. No measurable effects on fish impingement or plankton entrainment are expected.

Changes in cooling tower blowdown rate and cooling tower chemistry as a result of the uprate are not significant. Any changes in blowdown rate and cooling tower cycles of concentration resulting from uprated power operation are enveloped by the existing design criteria discussed in FES.

Cooling tower drift does not increase as a result of the uprate since the circulating water flow rate does not change. Cooling tower blowdown temperature associated with power uprate operation will increase slightly (<1°F), thereby producing a slight increase in river discharge temperature. A review of the increase in the river discharge temperature relative to the conclusions of the FES and thermal studies required to support licensing of the plant indicates the slight temperature increase is not significant.

The thermal plume characteristics are not expected to change significantly as a result of power uprate. Circulating water and service water flow rates remain unchanged. The discharge temperature to the cooling towers should increase by no more than 1°F due to

operation at power uprate conditions The corresponding change in discharge temperature at the river will not significantly impact the size or characteristics of the thermal plume. Thermal plume studies conducted during original licensing and the FES conclusions relative to thermal impacts remain valid for the uprated condition.

No significant change in discharge flow rate, velocity, or chemical composition will occur due to the proposed power uprate. Power uprate does not impact the discharge characteristics upon which the NPDES Permit is based. No notification, changes, or other actions relative to the NPDES Permit are required.

No change in the groundwater withdrawal required to supply the Hatch treatment plant or fire protection system will result from the proposed uprate.

The evaluation also considered the flow rate required by the liquid radwaste system (e.g., floor and equipment drains) due to the proposed uprate. No significant change in liquid radwaste quantities or activity levels which would increase the required radwaste dilution flow are expected. Therefore, the impact on the environment from these systems as a result of operation at the uprate power levels is not significant.

Plant operation at uprated power conditions will not effect current noise levels. Major plant equipment is housed within structures located on the plant site and is not a major contributor to surrounding noise levels. Equipment, such as the main turbines/generators and the cooling towers, will continue to operate at the current speed and noise level. The generator step-up transformers will operate at an increased KVA level; however, the overall noise level will not increase significantly.

Georgia Power Company concludes that the proposed uprate will not result in any significant environmental impact and is not an unreviewed environmental question In addition, no actions relative to the Environmental Technical Specifications (ETS), NPDES permit or other environmental documents are required.

Radiological Environmental Assessment

Georgia Power Company has evaluated the impact of the proposed power uprate amendment and has concluded that the applicable regulatory acceptance criteria relative to radiological environmental impacts will continue to be satisfied for the uprated power conditions. Existing Technical Specifications limits on radiological effluents will be maintained. In conducting this evaluation, GPC considered the effect of the higher power level on liquid radioactive wastes, gaseous radioactive wastes, and radiation levels both in the plant and offsite during both normal operation and post-accident.

Enclosure 4 of Reference 1 provides the power uprate safety analyses report for Plant Hatch, as well as an assessment of the radiological effects of power uprate operation

during both normal and postulated accident conditions. Sections 8.1 and 8.2 discuss the potential effect of power uprate on the liquid and gaseous radwaste systems. Sections 8.3, 8.4, and 8.5 discuss the potential effect of power uprate on radiation sources within the plant and radiation levels during normal and post-accident conditions. Section 4.4 discusses the standby gas treatment system (SGTS). Section 9.2 presents the results of the calculated whole body and thyroid doses at the exclusion area boundary and the low population zone that might result from the postulated design basis radiological accidents. All offsite doses remain below established regulatory limits for power uprate operation.

The floor drain collector subsystem and the waste collector subsystem both receive inputs from a variety of sources (e.g., leakage from component cooling water system, reactor coolant system, condensate and feedwater system, turbine, and plant cooling water system). However, leakages from these systems are not expected to increase significantly since the operating pressures of these systems are either being maintained constant or are being increased only slightly due to the proposed power uprate.

The largest single source of liquid radioactive waste is from the backwash of the condensate demineralizers. These demineralizers remove activated corrosion products which are expected to increase proportionally to the proposed power uprate. However, the total volume of processed waste is not expected to increase significantly, since the only appreciable increase in processed waste will be due to the slightly more frequent cleaning of these demineralizers. Based on a review of plant effluent reports and the slight increase expected due to the proposed power uprate, GPC has concluded that the slight increase in the processing of liquid radioactive wastes will not have a significant increase in environment impact and that requirements of 10 CFR 20 and 10 CFR 50, Appendix I, will continue to be met.

Gaseous radioactive effluents are produced during both normal operation and abnormal operation occurrences. These effluents are collected, controlled, processed, stored, and disposed of by the gaseous radioactive waste management systems which include the various building ventilation systems, the offgas system, and the SGTS. The concentration of radioactive gaseous effluents released through the building ventilation systems during normal operation is not expected to increase significantly due to the proposed power uprate since the amount of fission products released into the reactor coolant (and subsequently into the building atmosphere) depends on the number and nature of fuel rod defects and is not dependent on reactor power level. The concentration of activation products contained in the reactor coolant is expected to remain unchanged, since the linear increase in the production of these activation products will be offset by the linear increase in steaming rate. Therefore, based on its review of the various building ventilation systems, GPC has concluded that there will not be a significant adverse effect on airborne radioactive effluents as a result of the proposed power uprate.

Radiolysis of the reactor coolant causes the formation of hydrogen and oxygen, the quantities of which increase linearly with core power. These additional quantities of hydrogen and oxygen would increase the flow to the recombiners by 5 percent during uprated power conditions. However, the operational increases in hydrogen and oxygen remain within the design capacity of the system.

The SGTS is designed to minimize offsite and control room radiation dose rates during venting and purging of both the primary and secondary containment atmospheres under accident or abnormal conditions. This is accomplished by maintaining the secondary containment at a slightly negative pressure with respect to the outside atmosphere and discharging the secondary containment atmosphere through high-efficiency particulate air (HEPA) filters and charcoal adsorbers. The SGTS charcoal absorbers are designed for a charcoal loading capacity of 2.5 mgI/gC for the 30-day loss-of-coolant accident (LOCA) scenario. The proposed power uprate will increase the post-LOCA iodine loading by 5 percent; however, the charcoal loading will remain within the 2.5 mgI/gC design limit. Therefore, there will be no significant increase in environmental impact.

Georgia Power Company has evaluated the effects of the power uprate on in-plant radiation levels for Plant Hatch during both normal operation and post-accident. The conclusions are that radiation levels during both normal operation and post-accident may increase slightly (approximately proportional to the increase in power level). The slight increases in in-plant radiation levels expected due to the proposed power uprate should not affect radiation zoning or shielding requirements. Individual worker occupational exposures will be maintained within acceptable limits by the existing as low as reasonably achievable (ALARA) program which GPC uses to control access to radiation areas. Therefore, the slightly increased in-plant radiation levels will not have a significant environmental impact.

The offsite doses associated with normal operation are not significantly affected by operation at the proposed uprated power level and are expected to remain well within the limits of 10 CFR 20 and 10 CFR 50, Appendix I. Existing Technical Specifications limits will not be changed due to uprate. Therefore, offsite doses due to power uprate conditions will not result in a significant environmental impact.

Georgia Power Company performed dose evaluations for design basis accidents at or above 102% of the uprated power level and reported these results in Reference 1. The offsite doses remain below regulatory limits and the increase due to power uprate is 5% or less.

Conclusions

Georgia Power Company concludes that the proposed uprate will not result in a significant adverse environmental impact and is not an unreviewed environmental question. No

changes to Appendix B of the Operating Licenses are required. Based on the above evaluation, the plant operating parameters impacted by the proposed power uprate remain within the bounding conditions on which the FES was based. The FES concluded that no significant environmental impact would result from the operation of Plant Hatch. This conclusion remains valid for power uprate.

References

1. Georgia Power Company HL-4724, J. T. Beckham, Jr., to NRC, "Power Uprate Operation," dated January 13, 1995.

2. Georgia Power Company HL-4812, J. T. Beckham, Jr., to NRC, "Response to Request for Additional Information - Power Uprate Submittal," dated April 5, 1995.

3. Federal Register Vol. 60, No. 41, "Niagra Mohawk Corporation; Environmental Assessment and Finding of No Significant Impact", dated March 2, 1995.

4. Georgia Power Company "Final Environmental Statement for Edwin I. Hatch Nuclear Plant Units 1 and 2," October 1972.

5. Georgia Power Company Edwin I. Hatch Nuclear Plant - Unit 2 Environmental Report, Operating License Stage, July 1975.

6. Georgia Power Company "NUREG-0417," Final Environmental Statement Related to Operation of Edwin I Hatch Nuclear Plant Unit No. 2, March 1978.