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April 5, 1995

Docket Nos. 50-321 HL-4812
50-366

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

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

Gentlemen:

By letter dated January 13, 1995, Georgia Power Company (GPC) submitted a Technical Specifications amendment request for the Edwin I. Hatch Nuclear Plant - Units 1 and 2. The proposed amendment increases the authorized maximum power level for both units from the current limit of 2436 MWt to 2558 MWt. On February 22, 1995, the NRC Staff met with GPC representatives to discuss the proposed Technical Specifications revisions.

By letter dated March 10, 1995, the NRC requested GPC to provide additional information based on the January 13th submittal and data supplied in the February 22nd meeting. Enclosure 1 is GPC's response to the Request for Additional Information. Enclosure 2 provides information relative to Plant Hatch setpoint methodology and calculations as requested by the NRC.

Please contact this office if you have questions or comments.

Sincerely,

J. T. Beckham, Jr.

GKM/eb

Enclosures:

1. Response to RFAI
2. Setpoint Methodology and Selected Setpoint Calculations

cc: (See next page.)

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1/1

Georgia Power

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

Enclosure 1

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

Enclosure 1

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

As stated in the NRC's Request for Additional Information dated March 10, 1995, the section numbers referenced in this enclosure are applicable to General Electric Report NEDC-32405P, "Power Uprate Safety Analysis for Edwin I. Hatch Nuclear Plant Units 1 and 2," dated December 1994.

NRC Question 1:

(Section 2.5.1) State the basis for determining the acceptability of the control rod drive mechanism (CRDM) regarding compliance with the design Code. The information provided should include the Code and Edition, the Code allowables, the most critical component, the calculated maximum stresses, deformation, and fatigue usage factor for the uprated power conditions, and assumptions and load combinations used in the calculations.

GPC Response:

Section 2.5.1 states that the CRD mechanism structural and functional integrity is acceptable for at least 1250 psig (which is above the uprate operating pressure and the high pressure scram setpoint, including hydrostatic head). Also, included is information on the design Codes and Editions for both Units 1 and 2.

The limiting component of the CRD mechanism is the indicator tube which has a calculated stress of 20,790 psi; the allowable stress is 26,060 psi. The maximum stress on this component results from a maximum CRD internal hydraulic pressure of 1750 psig with no other event having a significant impact on the total load.

The cyclic operation of the CRD was conservatively evaluated in accordance with ASME Code, N-415.1 (NB-3222.4). The analysis was performed based on the loads from scram with a leaking scram discharge valve and a failed scram buffer. The limiting component was found to be the CRD main flange. The fatigue usage factor, calculated based on NB-3222.4(E) is 0.15, which is less than the allowable limit of 1.0. All requirements of N-415.1 (NB-3222.4) are satisfied even when considering the increased power uprate vessel bottom head pressure, thereby satisfying the peak stress intensity limits governed by fatigue.

Deformation of components was not specifically calculated for power uprate conditions. However, the CRD mechanism has been subjected to intensive testing at 1250 psi, which is higher than the maximum power uprate pressure. Based on the demonstrated performance of the mechanism at these high pressures, it is concluded that deformation resulting from the power uprate pressure increase is of no significant consequences.

NRC Question 2:

(Section 3.2) No relief flow for ASME overpressure analysis should be assumed. Only the safety portion of the safety/relief valve (SRV) should be used in the analysis. Figure 3-1 in this section is not clear as to whether the relief flow is assumed to be zero. Provide assurance that zero relief flow was assumed in the analysis.

GPC Response:

The safety analysis does not take credit for "relief" flow. Both Plant Hatch units are equipped with 11, two-stage Target Rock SRVs, each providing safety and relief functions. The setpoints for both functions are identical. Therefore, the relief valve flow shown in Figure 3-1 of NEDC-32405P represents only the flow associated with the safety function.

NRC Question 3:

(Section 3.3.1) Provide the following: (1) the increased end-of-life fluence values for reactor vessel materials due to power uprate; (2) the limiting adjusted reference temperature under the increased neutron fluence using Regulatory Guide 1.99, Revision 2, to demonstrate that with an increased fluence and operating pressure, the heatup, cooldown, and hydrotest pressure/temperature limit curves are bounding; (3) the hydrotest pressure resulting from the increased operating pressure; (4) the revised hydrostatic/leak test curve resulting from increased operating pressure; and (5) an assessment of the capsule withdrawal schedule and the number of capsules in the surveillance capsule program under the increased neutron fluence using Appendix H to 10 CFR 50.

GPC Response:

1. The increased end-of-life fluence values for reactor vessel materials due to power uprate.

   The fluence was conservatively increased by 10% to consider the influence of power uprate. The calculation of fluence for the power uprate condition are as follows:

		32 EFPY Fluence 110% Power Uprate	
Identification	Thickness (inches)	Peak ID (n/cm^2)	Peak 1/4 T (n/cm^2)
Hatch Unit 1			
Lower Intermediate Shell	5.38	2.8e18	2.0e18
Lower Shell	6.38	2.8e18	1.9e18
Hatch Unit 2			
Lower Intermediate Shell	5.38	1.5e18	1.1e18
Lower Shell	6.38	1.5e18	1.1e18

2. The limiting adjusted reference temperature under the increased neutron fluence using Regulatory Guide 1.99, Revision 1, to demonstrate that with an increased fluence and operating pressure, the heatup, cooldown, and hydrotest pressure/temperature limit curves are bounding.

   The following table shows the limiting adjusted reference temperature (ART) for the materials:

LIMITING ART

			32 EFPY Fluence	
Component	Identification	Heat	Current Power (°F)	110% Uprated Power (°F)
Hatch Unit 1				
Plate: Lower Shell	G-4805-2	C4112-2	93.0	95.1
Weld: Lower to Low-Int Girt	1-313	90099	159.2	163.9
Hatch Unit 2				
Plate: Lower Shell	G-6603-2	C8553-1	65.6	67.5
Weld: Lower Long	101-842	10137	69.0	71.9
Weld: Low-Int Long	011-834	51874	63.9	66.6

3. The hydrotest pressure resulting from the increased operating pressure.

   The proposed hydrostatic and leakage test pressures are provided in GPC's January 13, 1995 submittal in proposed Technical Specification Bases 3.10.1 (both units). The leakage test pressure is increased 30 psi from 1005 psig to 1035 psig. The hydrostatic test pressure is increased 33 psi from 1106 psig to 1139 psig.

4. The revised hydrostatic/leak test curve resulting from increased operating pressure.

   The pressure versus temperature curves provided in the Technical Specification remain limiting for both Hatch 1 and Hatch 2.

   For Hatch 1 the Technical Specification curves were conservatively done for an ART provided for 16 effective full power years (EFPY) and are based on a previously calculated ART of 133°F. For power uprate the ART was evaluated for 17 EFPY using a combination of 16 EFPY at 100% power and 1 EFPY at 110% power uprate. This uprated ART resulted in a limiting ART of 132°F which is bounded by the 133°F ART used for the curves in the Technical Specification. Note that the pressure versus temperature curves will require revision following removal and testing of the second surveillance capsule. The second capsule will be removed from Unit 1 either during the Spring of 1996 or Fall of 1997.

   For Hatch 2 the non-beltline curves are still limiting even when evaluating the ART for 32 EFPY at 110% power uprate.

5. An assessment of the capsule withdrawal schedule and the number of capsules in the surveillance capsule program under the increased neutron fluence using Appendix H to 10 CFR 50.

   The change in ART for a conservatively assumed 10% increase in fluence is less than 5°F. A review of ASTM E185-82 indicates that uprate will not have a significant impact on the current surveillance capsule program. The withdrawal schedule is provided in both units' FSARs and there are no plans to change this schedule for power uprate. Originally, each unit contained three capsules.

<u>NRC Question 4</u>:

(Section 3.3.2) Clarify whether the recent Hatch Unit 1 shroud modification was analyzed for the uprated condition and would satisfy, in that condition, the recommendations of Generic Letter (GL) 94-03. Discuss the planned shroud modification at Unit 2 with respect to the uprated conditions and GL 94-03. Provide an assessment of power uprate impact on the feedwater nozzle cracking and control rod drive return line as discussed in NUREG-0619.

<u>GPC Response</u>:

The Unit 1 shroud repair was designed and analyzed for the uprate condition. However, a design discrepancy, which could result in a small gap in the shroud during normal operation if a complete through-wall circumferential crack is assumed, was recently identified. For the shroud design, GPC committed to adhere to criteria which do not allow for gaps during normal operation. As stated in GPC's letter to the NRC dated February 20, 1995, the resolution of, and conformance, to the shroud repair criteria will consider the power uprate conditions and will be in place prior to startup from the Spring 1996 Unit 1 outage. Unit 1 will not be operated above 100% power without 1) modifying the repair such that no separation occurs, 2) performing additional analysis showing that no separation occurs or, 3) making a separate submittal for review and approval should the criteria not be met.

The Unit 2 shroud repair design will be fully analyzed for uprated conditions.

The feedwater nozzle cracking identified in NUREG-0619 is primarily associated with startup and shutdown cycles which are not significantly affected by power uprate. High cycle fatigue is not significantly impacted, since final feedwater temperature is not reduced at uprated conditions. The resolution of NUREG-0619 for the CRD return line includes rerouting of the piping -- a physical change not affected by power uprate.

<u>NRC Question 5</u>:

(Section 3.3.2) Discuss the effects of power uprate on the reactor internals responses associated with loss-of-coolant accident (LOCA), SRV discharge, annulus pressurization and jet reaction loads.

<u>GPC Response</u>:

Reactor internal loads associated with loss-of-coolant accident, SRV discharge, annulus pressurization, and jet reaction loads are not specifically described in NEDC-32405P. These reactor internal loads are defined as "new loads" and, therefore, are not part of the current Plant Hatch licensing basis for both units. Power uprate analyses for Plant Hatch were performed consistent with the current plant licensing basis. This approach is in agreement with the Generic Power Uprate Report (LTR-1), NEDC-31897P-1.

Main Steam Line and Recirculation Line break LOCA loads which are part of the current Hatch licensing basis have been evaluated for power uprate and resulting reactor internals stresses are within allowable limits as discussed in NEDC-32405P.

<u>NRC Question 6</u>:

(Section 3.3.2) List the codes used for evaluation of stresses and allowables for the reactor internals. List the maximum stresses, fatigue usage factors, and the location of highest stressed areas for both the current design and the uprated power conditions.

GPC Response:

Section 3.3.2 of NEDC-32405P specifies the code and edition used for evaluation of the reactor vessel and internals. Table 3-1 provides a comparison of maximum stresses and locations for reactor internals. Table 3-5 provides fatigue usage factors of limiting reactor vessel components. For power uprate, the highest fatigue usage factor was 0.93 for the Hatch Unit 1 CRD nozzle and 0.93 for the Hatch Unit 2 feedwater nozzle. Using actual plant data and a cycle counting approach for pre-uprate conditions, some conservatism was removed from the existing feedwater nozzle analysis.

NRC Question 7:

(Section 3.3.2) Are Table 3-1 stresses affected by the shroud modification? If so, provide an evaluation of the reactor internal components for the configuration with the shroud modification at the uprated power conditions.

GPC Response:

Table 3-1 of NEDC-32405P and the power uprate evaluation of reactor internal components includes the Unit 1 shroud modification configuration. The other reactor component stresses are not affected by the shroud modification. Comparable evaluations will be performed for the Unit 2 shroud modification, and these evaluations will consider power uprate conditions.

NRC Question 8:

(Section 3.3.2) Provide assurance that core shroud modification will not have a negative impact on the thermal-hydraulic analysis of the reactor coolant system.

GPC Response:

The thermal hydraulic analysis of the reactor coolant system was evaluated for the Unit 1 shroud repair at power uprate conditions. The effect of the shroud modification consists of increased leakage through the repair slots. The steam portion of this leakage increases the total steam carryunder draining from the steam separators to the downcomer annulus. The impact of this carryunder on the steam separation system and jet pump performance, fuel thermal margin, ECCS performance, and fuel cycle length has been evaluated.

The shroud leakage increases the total carryunder by about 0.02%; however, carryunder remains within the steam separator criteria used for the plant safety analysis. The increased carryunder reduces the margin to jet pump cavitation; but remains within the design condition for jet pump performance. The increased carryunder results in a favorable effect regarding fuel thermal margin. Therefore, fuel thermal limits are not effected.

The increased carryunder has a very small effect on core inlet enthalpy; and there is no significant degradation of ECCS performance. There is also a small but insignificant increase of fuel peak cladding temperature (PCT), but the calculated PCT is still within the licensing basis PCT reported in NEDC-32405P. There is also a minor effect on fuel cycle length which is not significant. The design basis limit for carryunder is still met.

A comparable thermal hydraulic evaluation at power uprate conditions will be performed for the Unit 2 shroud repair design.

NRC Question 9:

(Section 3.3.3.2) This section stated that "Elastic-plastic methods were implemented for some components; the Code requirements for these methods were met." Discuss in detail the analysis methodology, assumptions, and compliance with the Code. Include the Code Edition and the Code allowables used in the evaluation.

GPC Response:

The evaluation of the reactor vessel and component integrity uses Section III, Class 1 subsections NB-3222 and NB-3223, of the ASME boiler and pressure vessel code. The Unit 1 feedwater nozzle, control rod drive nozzle, and vessel shell were reanalyzed for uprated conditions. The feedwater nozzle evaluation uses the 1974 ASME Code edition with addenda to and including Summer of 1976. The control rod drive nozzle and vessel shell evaluations use the 1965 ASME Code edition with addenda to and including Winter 1966.

The Unit 2 closure vessel shell, closure region bolts, feedwater nozzle, and basin seal skirt were also reanalyzed for uprated conditions. The closure vessel shell, closure region bolts, and basin seal skirt evaluations use the 1968 ASME Code edition with addenda to and including Summer 1973. The feedwater nozzle evaluation used the 1971 ASME Code edition with addenda to and including Summer 1973.

According to the ASME Code subsections, structural adequacy is met if the maximum primary plus secondary stress intensity at a location on a component is less than $3S_m$ of the material. If the $3S_m$ limit is not met, plastic behavior is assumed and the simplified elastic-plastic analysis of ASME Code, paragraph NB-3228.3, can be used to determine structural adequacy. Specifically, Code criteria are as follows:

1. The calculated range of primary plus secondary membrane plus bending stress intensity, excluding thermal bending stresses, shall be below $3S_m$.

2. The value of S_a used for entering the design fatigue curve is multiplied by the factor K_e.

3. The rest of the fatigue evaluation stays the same.

4. The component meets the thermal ratcheting requirements.

5. The material temperature does not exceed the maximum temperature permitted for the material.

6. The material shall have a specified minimum yield strength to specified minimum tensile strength ratio of less than 0.80.

The analyses were done in conformance with the code and code allowables were met.

<u>NRC Question 10</u>:

(Section 3.3.3.2) Table 3-5 provides fatigue usage factors of the limiting components of the reactor vessel and its support for Hatch Units 1 and 2. State why the cumulative usage factor (CUF) for the feedwater nozzle was calculated using the cycle counting approach for the pre-uprated conditions, but not the uprated conditions. Because the table shows that the CRD nozzle is the most critical component at the uprated condition for Hatch Unit 1, provide the CUF values for the CRD nozzle for Hatch 2 at both the current and uprated power conditions.

<u>GPC Response</u>:

The use of conservative power uprate scaling factors on the Unit 2 feedwater nozzle fatigue evaluations shows an end-of-life CUF of 1.1. Since this CUF exceeds the allowable limit of 1.0, a more realistic CUF was calculated by combining the fatigue usage factor based on actual plant operating data with a design basis fatigue usage factor calculated for power uprate conditions. That is, credit was taken for actual operating thermal cycle counting information to estimate the actual fatigue usage factor <u>thus far</u> in the plant's life. The actual usage factor is lower than the design basis fatigue usage factor originally predicted for this portion of the plant's life. From September 1995, until end of life, the power uprate design fatigue usage factors were used, resulting in a 0.93 CUF for uprate, versus the current design basis CUF of 0.94.

The cumulative fatigue usage factor for the Unit 2 CRD nozzle is 0.12 at current power conditions. This value is significantly lower than the allowable fatigue usage factor limit of 1.0. Therefore, the CRD nozzle was not considered to be a limiting component for Unit 2 and was not reanalyzed for power uprate.

NRC Question 11:

(Section 3.5) Provide analysis methodology and assumptions regarding the evaluation of Reactor Coolant Pressure Boundary piping systems, in-line components (pumps, valves, nozzles, penetrations, etc.) and supports (hangers, snubbers, struts including anchorage, etc.), thermal and vibration displacements. Discuss compliance with the Code and Edition, and the Code allowables for stresses and fatigue limits at the uprated power conditions.

GPC Response:

The evaluation of piping at uprated power conditions in Reactor Coolant Pressure Boundary (RCPB) systems for compliance with the ASME Code was performed on the recirculation and main steam piping. Also, Class 1 and Class 2 piping of Core Spray, Feedwater, Standby Liquid Control, Reactor Water Cleanup (inside containment), High Pressure Coolant Injection (outside containment), and RHR (outside containment) systems, and the Reactor Pressure Vessel (RPV) vent line were evaluated for acceptability at the uprated conditions, and shown to be adequate as currently designed. Small-bore RCPB lines (e.g., instrument lines) were also evaluated. The original Code of record, Code allowables, and analytical techniques were used. No new assumptions were introduced.

The methodology used in the evaluation is as follows:

a. Existing design basis documents; e.g., design specifications and piping stress reports, were reviewed to determine the design and analytical basis for RCPB piping systems. The power uprate parameters of RCPB systems were compared with the existing analytical basis to determine increases in temperature, pressure, and flow due to power uprate conditions.

b. ASME B&PV Code, Section III, Subsection NB-3600, Code equations 9, 10, 12, 13 and 14 were reviewed to determine the equations impacted by temperature, pressure, and flow increases due to power uprate condition on Class 1 piping systems.

c. General Electric (GE) performed a parametric study for the RCPB systems to determine the percent increases in applicable Code stresses, displacements, CUFs, and pipe interface component (including supports) loads as a function of percentage increase in pressure, temperature, and flow due to power uprate conditions. The percent increases were applied to the highest calculated stresses, displacements, and the CUF at applicable RCPB piping system node points to conservatively determine the maximum power uprate calculated stresses, displacements and usage factors. This approach is conservative since power uprate does not affect all dynamic loads; e.g., seismic loads are not affected by power uprate.

Piping interfaces with RPV nozzles, anchors, struts, penetrations, flanges, pumps, and valves were evaluated in a similar manner. The effect of power uprate conditions on thermal and vibration displacement limits was also evaluated, and the results indicate that the piping within these systems satisfies the displacement limitation criteria.

The results of these evaluations demonstrate that the requirements of ASME Code, Section III, Subsection NB-3600, are satisfied for both the Main Steam and Recirculation piping systems at the power uprate conditions. Interface loads on system components, which have increased due to power uprate conditions, do not exceed component acceptance criteria.

NRC Question 12:

(Section 3.5) Provide assurance that power uprate will not have negative effects on the reactor vessel level instrumentation system purging system.

GPC Response:

Power uprate will not adversely affect the reactor vessel water level instrumentation system purging system (keep-fill system). The system may require a slight adjustment of the flow regulator valve due to the increase in reactor pressure; however, the flow rate is now being checked periodically and adjusted as necessary to meet flow requirements. Any adjustment required for power uprate will be made as a matter of course.

NRC Question 13:

(Section 3.5) GPC must commit to evaluating plant-specific data related to the recirculation pump vibration as stated in the generic topical, NEDC-31897P-1, Section 5.6.2. GPC should also address the power uprate testing plan that is discussed in Section 5.11.9. This is also addressed in NEDC-31984P, Section 4.5.

GPC Response:

General Electric and GPC reviewed plant-specific data relating to recirculation system performance and vibration. (Reference NEDC-31897-1, Section 5.6.2, and NEDC-31984, Section 4.5.) The results of this review indicate that the expected speed increase necessary to maintain the same core flow is less than 1-percent. System capability and vibration are not expected to be significantly affected.

Both Plant Hatch units are licensed for 105-percent increased core flow (ICF). GPC's January 13 power uprate submittal does not propose to increase the licensed core flow limit. The primary use of ICF is near the end of the fuel cycle. During this mode of operation, Plant Hatch has experienced no problems, and none are anticipated for power uprate operation.

Georgia Power Company's testing plan is currently under development and will address Section 5.11.9 of NEDC-31897P-1. Regarding specific testing for recirculation pump vibration, GPC will monitor existing instrumentation on the recirculation pump motors during and after power ascension; however, GPC does not plan to install any new or special testing equipment.

NRC Question 14:

(Section 3.8) GPC should commit to the recommendations and modifications of GE SIL 377. This should include a discussion on the reactor core isolation cooling (RCIC) testing to ensure that there is no adverse impact on the reliability of RCIC as a result of any power uprate modifications or operation.

GPC Response:

Service Information Letter (SIL) 377 modifications were incorporated into the Unit 1 RCIC System. The steam bypass line modification was installed on Unit 2; however, GE and GPC determined the existing line and bypass valve configuration was less than optimum. Although the Unit 2 RCIC system performs well during system startup, further enhancements recommended in SIL 377 will be incorporated during the Fall 1995 outage, thereby providing full compliance with SL 377. Therefore, both RCIC systems will meet the commitments specified in the generic topical report (LTR-1) prior to startup at the uprated power level.

A RCIC system flow test, performed at uprated power and pressure conditions and in accordance with the power uprate testing program recommended by GE, will be included in the power uprate startup testing program for each unit. The RCIC systems will take suction from and discharge back to the condensate storage tank.

NRC Question 15:

(Section 3.9) What is the increased time to shutdown temperature? Discuss how power uprate impacts the 36-hour criterion for cold shutdown discussed in Regulatory Guide 1.139. What is the increased time to cold shutdown?

GPC Response:

The original design goal for normal reactor shutdown was to cool the reactor vessel to 125°F within 20 hours (assuming two loops operating). This 20 hour criterion was established with the objective of preventing the cool down from being a critical path activity during plant shutdown. The time needed to achieve cold shutdown at power uprate conditions will be increased by approximately 5-percent (1 hour) of the prepower uprate time for cold shutdown and it is expected that the original objective will still be met. (This has been confirmed by calculation for a BWR similar to Plant Hatch). [Additionally, margin in the original design assumptions (e.g., heat exchanger fully fouled, maximum expected shutdown cooling service water temperature) is expected to minimize the impact of this small increase in cooldown time.]

Regulatory Guide 1.139 requires demonstration of cold shutdown (<212°F in 36 hours) assuming the most limiting single failure. Although GPC is not formally committed to RG 1.139, evaluations of shutdown cooling for a BWR similar to Plant Hatch, using RG 1.139 assumptions, indicated that shutdown time as 21 hours. Based on this information it is expected that both Plant Hatch units will meet the 36 hour criteria with a large margin.

NRC Question 16:

(Section 3.12) Identify the Code and Edition used for the power uprate evaluation of balance-of-plant (BOP) piping, in-line components (valves and nozzles, etc.), and pipe supports including anchorages. List the limiting BOP piping systems and components with respect to the maximum stresses and safety margin as a result of the power uprate.

GPC Response:

The Code and Edition used for the power uprate evaluations of BOP piping, in-line components (pumps, valves, nozzles, penetrations, etc.), and pipe supports, including anchorages, are the same as those committed to in the Plant Hatch FSARs.

The evaluation of the BOP piping and supports was performed in a manner similar to the evaluation of RCPB piping systems and supports. GPC's response to NRC Question 11 provides a detailed discussion of this analysis method. Table 16-1 provides a summary of the limiting stress ratios from the BOP piping evaluations for power uprate.

NRC Question 17:

(Section 3.12) This section provided the Code and Edition used for the power uprate evaluation of BOP piping, in-line components (pumps, valves, nozzles, penetrations, etc.), and pipe supports including anchorages. List the limiting BOP piping systems and components with respect to the maximum stresses and safety margin as a result of the power uprate.

GPC Response:

See the response to NRC Question 16 above.

NRC Question 18:

(Section 3.12) Provide an evaluation of the effect caused by the reactor power uprate on erosion/corrosion of carbon steel components exposed to single and two phase fluids. The proposed increase of reactor power would cause a corresponding increase in the operating temperatures, pressures, and flows in BOP systems. These parameters would have a significant effect on the rates at which the components susceptible to erosion/corrosion are degraded. In order to ensure safe operation of the plant, the present erosion/corrosion analyses should be revised.

GPC Response:

Erosion/corrosion concerns may be affected by increased flow rates, higher operating temperatures, and a change in moisture content of two-phase flow streams. However, the differences are considered to be small and the existing Plant Hatch Flow Assisted Corrosion (FAC) Program monitors and identifies such concerns. Power uprate should not create new FAC problems; however, in location(s) where erosion/corrosion wall thinning exist (monitored or unmonitored), the uprate may cause a slight increase in the rate of loss of pipe wall material.

Of the identified FAC mechanisms, (fluid velocity, temperature, moisture, oxygen level, pH, pipe metallurgy, and piping geometry), power uprate may cause minor changes only in velocity, temperature, and moisture content (two-phase fluids). GPC identified areas of potential FAC using present industry methods, specifically EPRI's CHECKWORKS program. GPC has a formal inspection program for monitoring pipe wall thickness. Any increase in the rate of thinning that occurs as a result of uprated conditions will be identified and monitored through this formal program.

TABLE 16-1

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	Eqn. 10	Eqn. 12	Eqn. 13	Eqn. 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	N/A*	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

NRC Question 19:

(Section 4.1.1) List the computer codes used for the containment pressure and temperature response. Provide figures showing temperature and pressure changes. Confirm that all other key input parameters except the decay heat codes are the same as in the current Final Safety Analysis Report. If not, provide details and discuss their effects on the pressure and temperature response.

GPC Response:

Containment analyses are performed for both short-term and long-term response. Using the M3CPT computer code, short-term containment pressure and temperature response analyses were performed at various reactor operating points. These operating points included rated conditions (uprated power/rated flow) and off-rated conditions (extended load line limit [ELLL] and final feedwater temperature reduction [FFWTR]). For the rated point, the analysis method and input assumptions are identical to those used in the Mark I Long-Term Program (LTP). For off-rated points and the rated point, a detailed blowdown model (LAMB) was used to calculate break flow rates and enthalpies. These LAMB break flow values were used as input to M3CPT (containment method used in the LTP). The reason for using LAMB flow values is that the blowdown model built in M3CPT is overly conservative at off-rated conditions. Relative to the long-term response evaluation, the SHEX computer code was used. No significant differences between SHEX and the previous analysis method were identified.

With regard to key input parameter differences (aside from the reactor power level and dome pressure associated with power uprate), the initial drywell temperature increased from 135°F to 150°F. The initial average drywell pressure increased to 1.75 psig, and the initial suppression pool temperature remained at 100°F. (The initial allowable pool temperature increased from 95°F to 100°F in 1988.) For the long-term temperature evaluation, the ANS/ANSI 5.1 decay heat model was used to calculate decay heat values as input to SHEX; whereas, the May-Witt decay heat model was most likely used in the previous analysis. The ANS 5.1 decay heat model is considered more realistic than the other model. In addition, the RHR flow rate is degraded 10 percent to 6900 gpm/pump, consistent with the degradation assumed in the existing SAFER/GESTR-LOCA analysis. All other key input parameters for the power uprate analysis are identical to those for the previous analysis.

Figures 19-1 through 19-4 provide the drywell pressure and temperature responses for Units 1 and 2. The long-term pool temperature response, applicable to both units, is plotted in Figure 19-5.

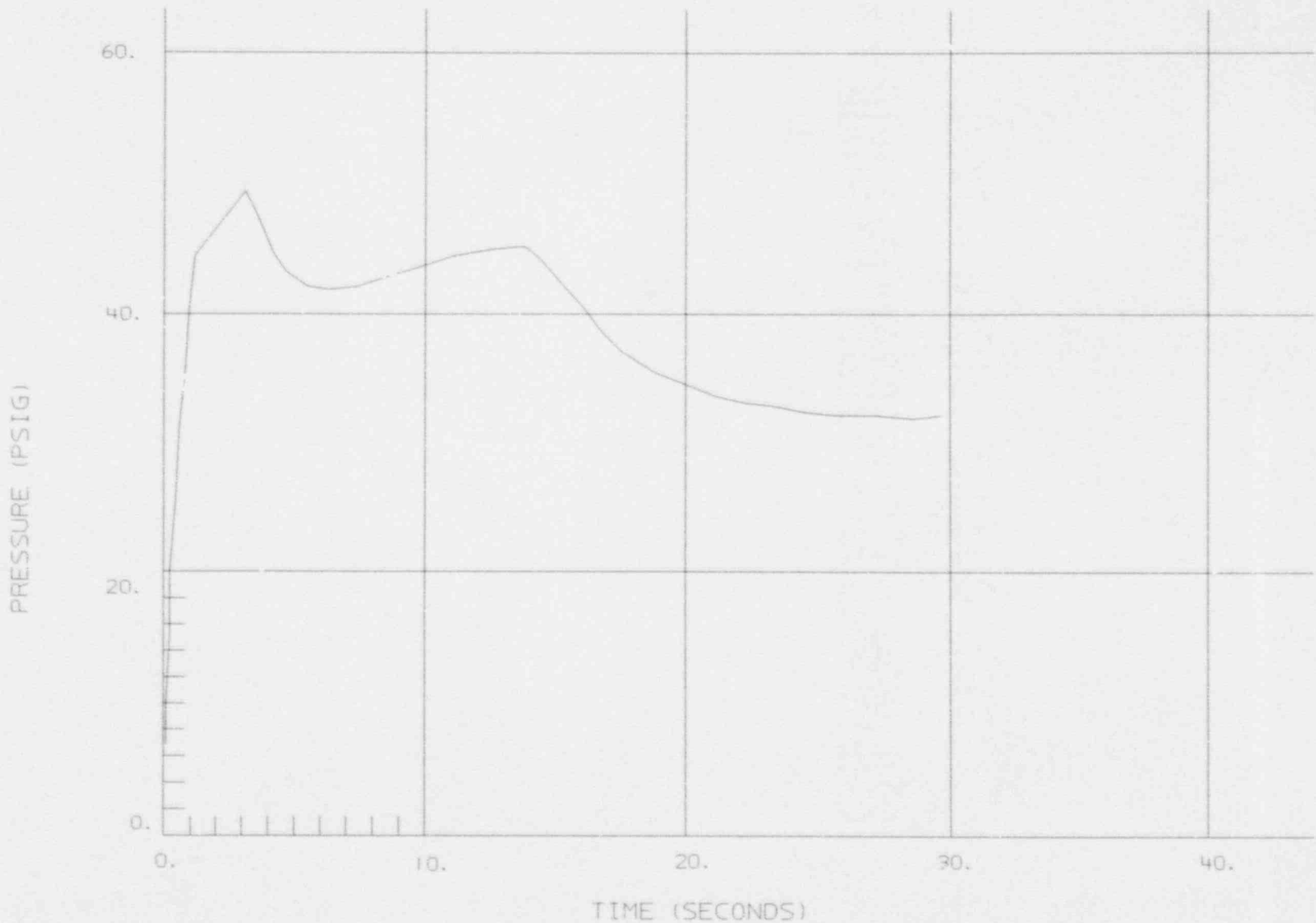

Figure 19-1 -CALCULATED DRYWELL PRESSURE RESPONSE FOLLOWING A RECIRCULATION LINE BREAK - UNIT 1 (POWER UPRATE)

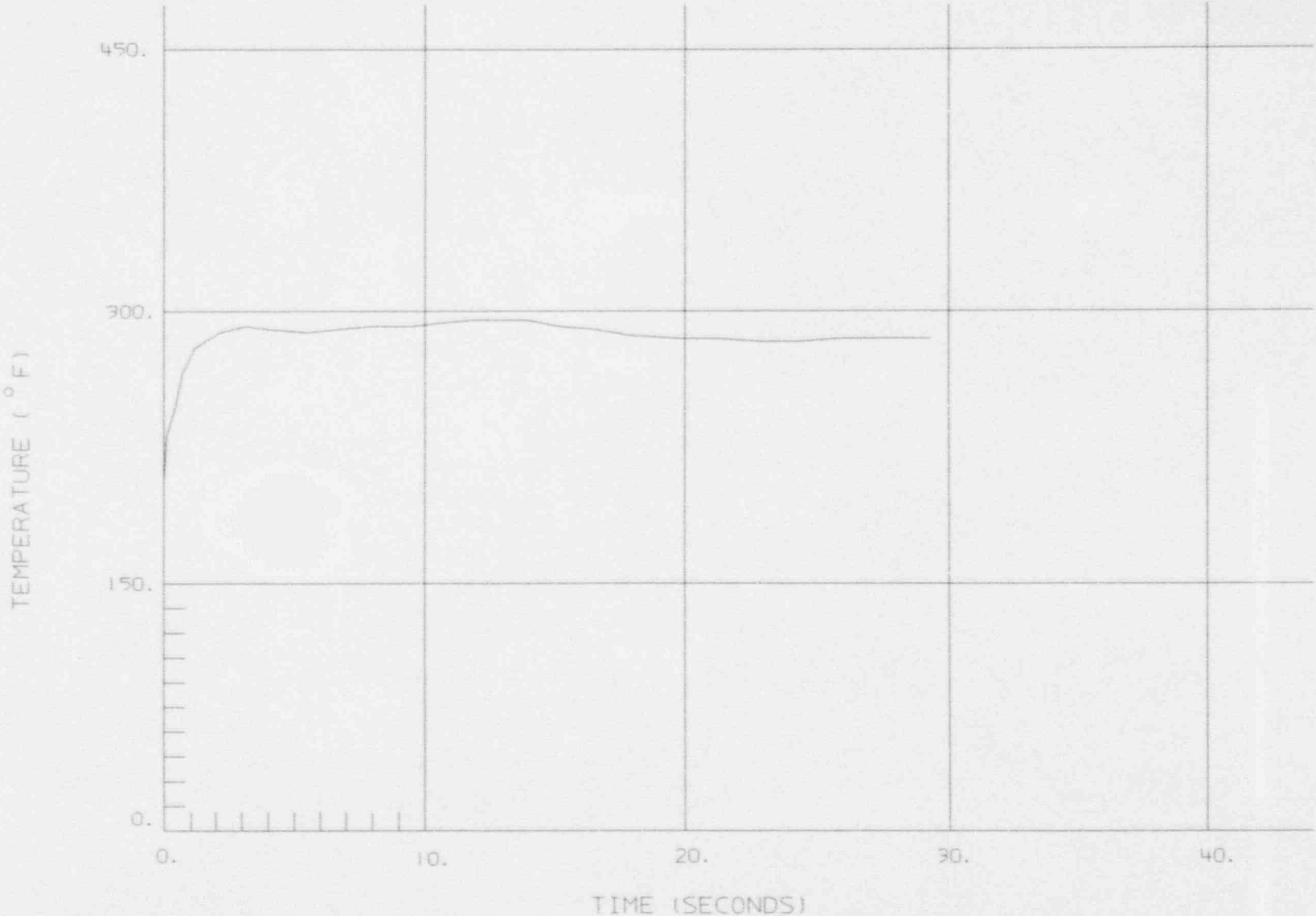

Figure 19-2 -CALCULATED DRYWELL TEMPERATURE RESPONSE FOLLOWING A RECIRCULATION LINE BREAK - UNIT 1 (POWER UPRATE)

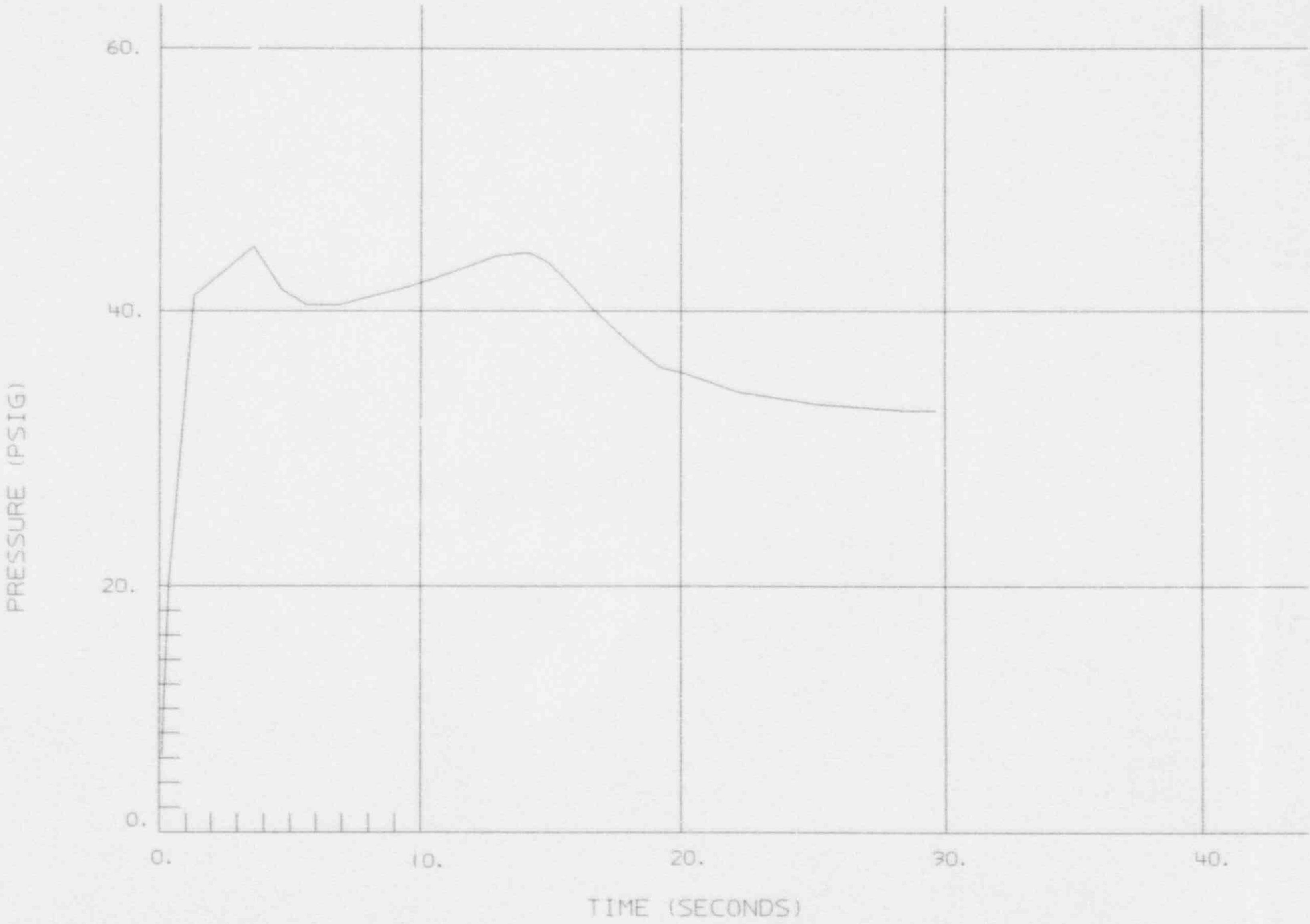

Figure 19-3 -CALCULATED DRYWELL PRESSURE RESPONSE FOLLOWING A RECIRCULATION LINE BREAK - UNIT 2 (POWER UPRATE)

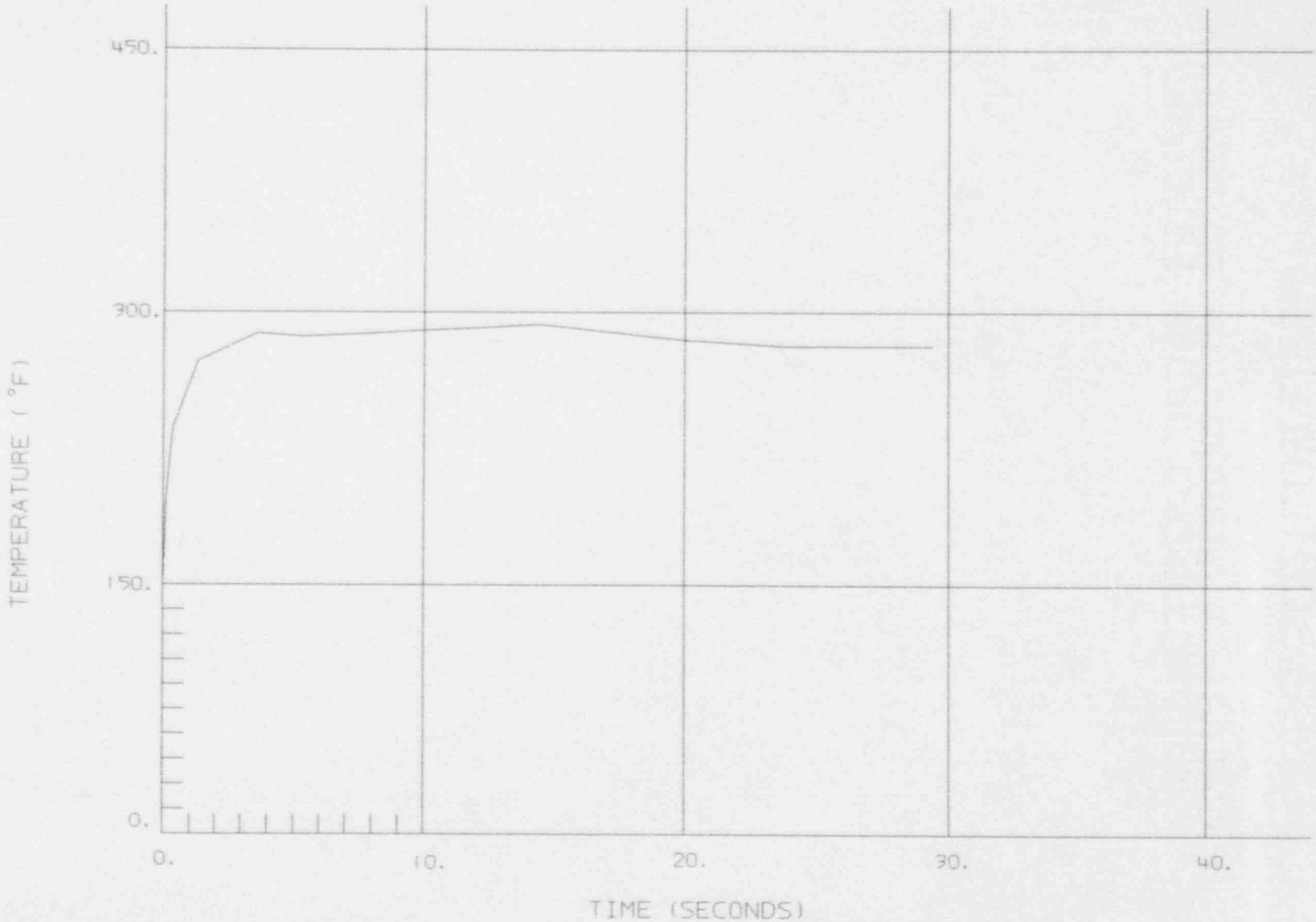

Figure 19-4 -CALCULATED DRYWELL TEMPERATURE RESPONSE FOLLOWING A RECIRCULATION LINE BREAK - UNITS 1 AND 2 (POWER UPRATE)

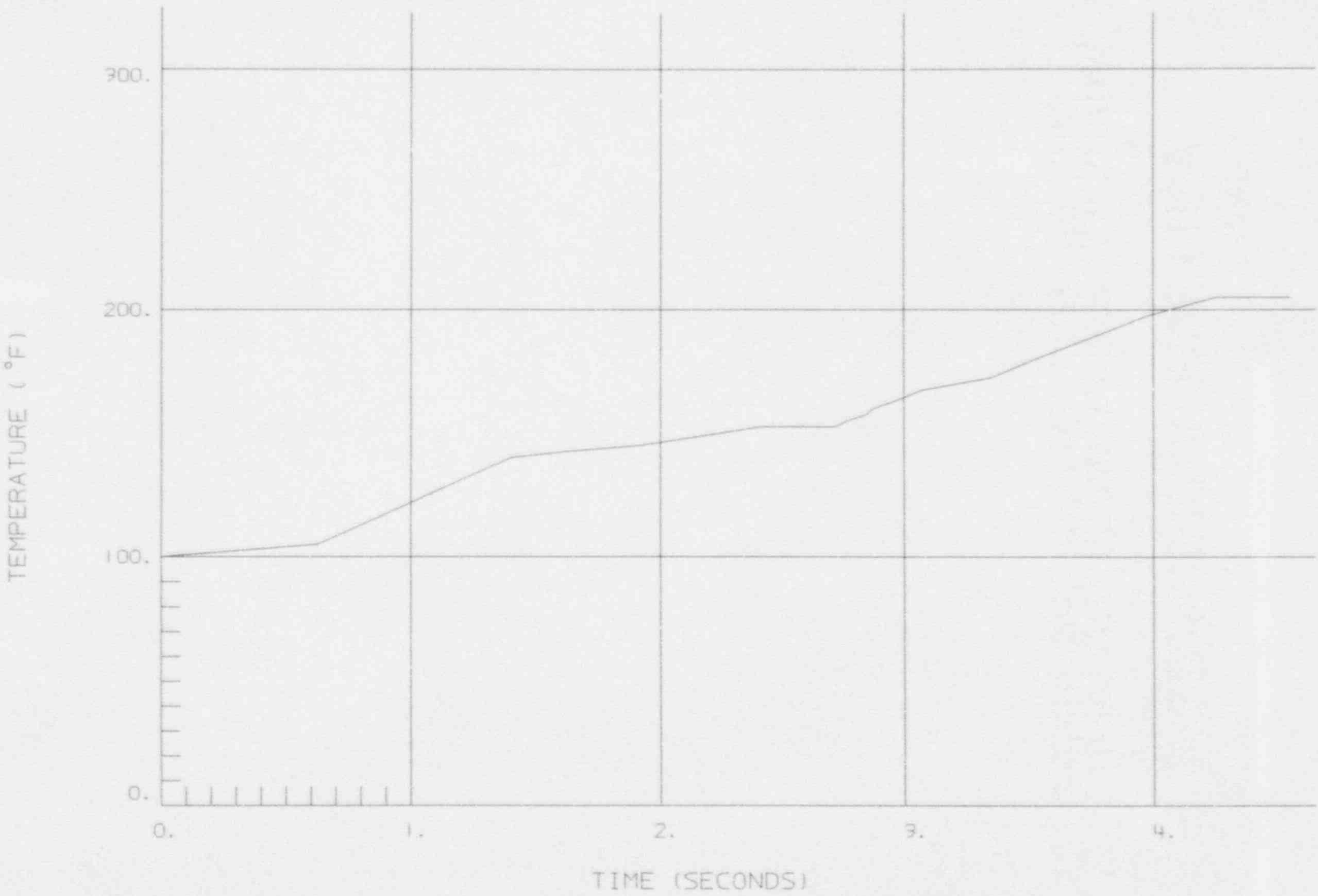

Figure 19-5 -CALCULATED SUPPRESSION POOL TEMPERATURE RESPONSE FOLLOWING A RECIRCULATION LINE BREAK - UNITS 1 AND 2 (POWER UPRATE)

NRC Question 20:

(Section 4.1.1.1) The submittal indicated that peak pool temperature goes up 4°F due to power uprate based on current methodology and remains below the wetwell structural design temperature of 281°F for Unit 1 and 340°F for Unit 2. Provide the peak pool design temperature for each unit based on the emergency core cooling system (ECCS) net positive suction head (NPSH) requirements.

GPC Response:

Unit 1 NPSH calculations take credit for wetwell pressure, because the unit is not committed to Regulatory Guide 1.1. The peak pool temperature allowed to satisfy NPSH requirements for given pump flows was determined as a function of wetwell pressure, as follows:

	Containment Pressure		
	0 psig	5 psig	10 psig
Core Spray Pump	202°F	222°F	236°F
RHR Pump	202°F	220°F	234°F

For Unit 2, no credit is taken for wetwell pressure, because the unit is committed to Regulatory Guide 1.1. Therefore, the peak pool design temperature for NPSH requirements was calculated assuming wetwell pressure is 0 psig. This value was determined to be approximately 220°F for both the Core Spray pumps and RHR pumps.

NRC Question 21:

(Section 4.1.1.1) The submittal indicated that the available NPSH for Core Spray (CS) and the residual heat removal (RHR) pumps is adequate for both units. Provide quantitative results to show margins. It is also indicated that for Unit 1, the NPSH for RHR and CS pumps is allowed to take credit for containment pressure. Compare this pressure credit at uprated power with the current/original credit. Provide information regarding the method used to calculate this pressure and the figures that show its variations.

GPC Response:

The design basis LOCA containment pressure and temperature responses used in the NPSH evaluation were calculated using the same input assumptions used in the peak pool temperature evaluation (Table 4.1 of NEDC-32450P), except for the following input assumptions to minimize the pressure response:

a. 0 psig initial pressure for both the drywell and wetwell.

b. 100 percent initial relative humidity for both the drywell and wetwell.

c. Drywell and wetwell sprays are actuated at 10 minutes and remain turned on throughout the event. This reduces the calculated wetwell pressure.

These assumptions have a negligible impact on long-term peak pool temperature but significantly decrease wetwell pressure. Figures 21-1 and 21-2 show the wetwell pressure and suppression pool temperature responses to the DBA-LOCA that were calculated with such assumptions. The peak pool temperature was calculated to be 202°F, with a wetwell pressure of 8.2 psig.

For Unit 1, the wetwell pressure required to satisfy NPSH requirements at 202°F peak pool temperature is approximately 0 psig, as compared to the 8.2 psig wetwell pressure. (See GPC Response to Question 20.) This means that at least an 8.2 psi margin exists for Unit 1, as compared to a 5 psi margin shown in the current FSAR.

For Unit 2, the peak pool design temperature for NPSH requirements is approximately 220°F for both CS and all RHR pumps, as compared to the 202°F peak pool temperature. (See GPC Response to Question 20.) Namely, an 18°F margin exists for Unit 2.

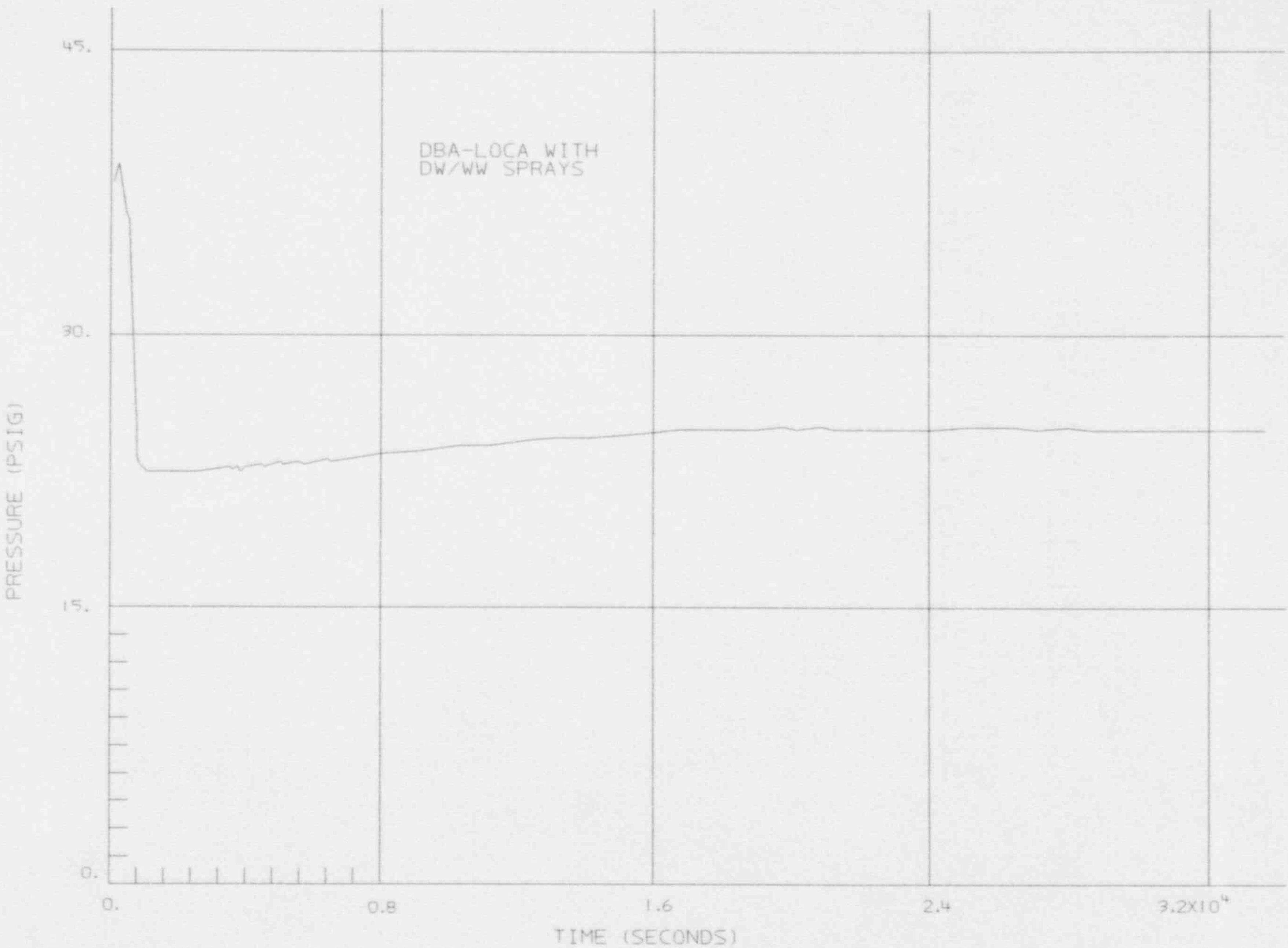

Figure 21-1 -CALCULATED LONG-TERM WETWELL PRESSURE RESPONSE FOLLOWING A RECIRCULATION LINE BREAK - UNITS 1 AND 2 (POWER UPRATE)

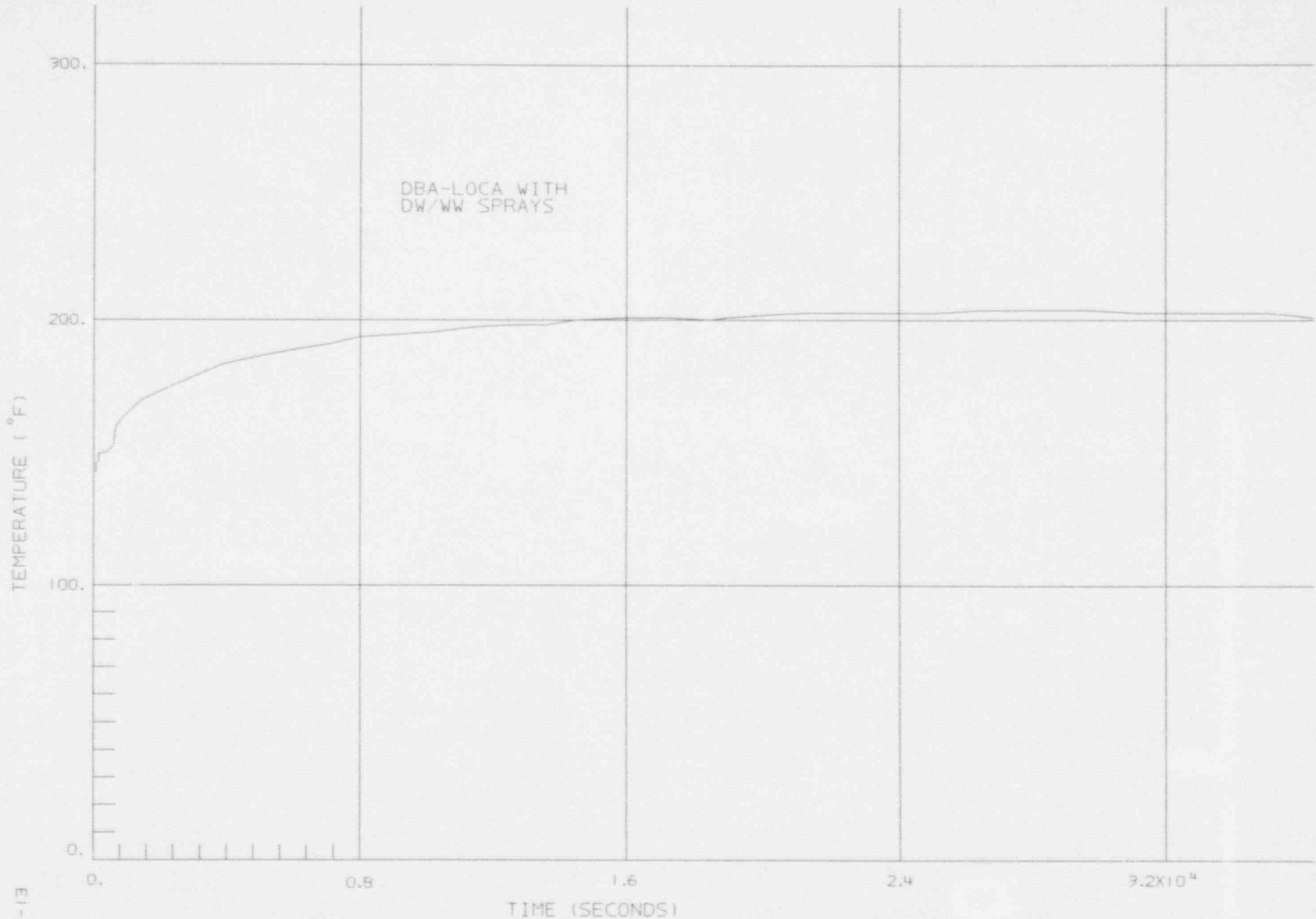

Figure 21-2 -CALCULATED LONG-TERM WETWELL PRESSURE RESPONSE FOLLOWING A RECIRCULATION LINE BREAK - UNITS 1 AND 2 (POWER UPRATE)

NRC Question 22:

(Section 4.1.2.2) The submittal stated that SRV setpoint will increase approximately 6%. Discuss the impact of the increased SRV discharge loading resulting from this setpoint increase on the SRV discharge, the torus-attached piping and pipe supports.

GPC Response:

Nominal SRV setpoints are actually being increased by approximately 3 percent due to power uprate. However, for the power uprate SRV loads evaluation, an additional 3 percent increase to the SRV open setpoint was included to support the incorporation of a ± 3 percent tolerance on the SRV open setpoint pressure. Industry operating experience indicates that the opening tolerance on the SRVs may be more than the 1 percent assumed during the Mark I Long-Term Program (LTP).

The SRV load definitions used for structural evaluations performed during the Mark I LTP are documented in the Unit 1 and Unit 2 Plant Unique Analyses Reports (PUARs), which were based on hydrodynamic models described in the Mark I Containment Load Definition Report (NEDO-21888). The values of the SRV setpoints used in the original PUAR analyses and a comparison to the values used in the power uprate evaluation are given in Table 22-1. The impact of an assumed increase in the setpoint pressure is approximately a linear increase in load for the first actuation of an SRV. (Subsequent actuations are controlled by low-low set logic; these pressure setpoints are not being changed for power uprate). Based on a review of the methods described in NEDO-21888 and the results of the structural evaluations documented in the PUARS, GPC determined there is sufficient margin to accommodate the increase in the SRV opening pressure with power uprate.

TABLE 22-1

Plant Hatch Units 1 and 2 SRV Setpoint Analytical Limits (AL) at Power Uprate Conditions Versus Values Used in PUARs Analyses

SRV Group	Values Used in PUAR (psig)		New AL at Power Uprate + 3% Tolerance (psig)		Maximum Allowable Value (psig)	
	Unit 1	Unit 2	Unit 1	Unit 2	Unit 1	Unit 2
1	1080	1090	1143.3	1153.3	1195.0	1195.0
2	1090	1100	1153.6	1163.9	119.50	1195.0
3	1100	1110	1163.9	1174.2	1195.0	1195.0

NRC Question 23:

(Section 4.2.1) GPC should commit to the recommendations of GE SIL 480 on the high-pressure coolant injection (HPCI) modifications and testing. GPC must also provide assurance that reliability will not be adversely impacted as a result of any power uprate modification or operation.

GPC Response:

Service Information Letter 480 was implemented in the HPCI systems of both units. Testing during startup will be similar to that described for the RCIC system. (See GPC Response to NRC Question 14).

The generic licensing topical report for power uprate (LTR-1) documents that HPCI reliability will not be adversely impacted by the relatively small increase in vessel pressure and turbine speed needed for uprate, provided that SIL 480 is implemented.

NRC Question 24:

(Section 4.3) ECCS analysis for power uprate, increased core flow, extended load line limit analysis, single-loop operation (SLO), break spectrum, and single failure as specified in LTR-1 page D-5, must be evaluated to assure that the limiting break, single failure, and break spectrum is not changed as a result of power uprate or fuel reload. Any multipliers for SLO must also be evaluated.

GPC Response:

The ECCS analysis, which was reviewed and approved by the NRC in 1987, was performed at a power level within 1 percent of the power level requested for power uprate. The analysis specifically accounted for the same performance improvement programs as the power uprate report (i.e., increased core flow, extended load line limit analysis, final feedwater temperature reduction). SLO was considered but was not recalculated using SAFER/GESTR, because the 0.75 multiplier on maximum average planar linear heat generation rate (MAPLHGR) calculated with the older SAFE/REFLOOD methodology was conservative. For example, SLO ECCS analyses using SAFER/GESTR on similar BWR plants resulted in MAPLHGR multipliers near 1.0.

The power uprate program and the fuel reload for uprate will not change the limiting break, single failure, or break spectrum as compared to the existing analysis. The performance improvement programs are also unchanged. The SLO 0.75 MAPLHGR or multiplier remains very conservative.

NRC Question 25:

(Section 5.0) The submittal indicated that the SRV setpoint with power uprate will increase approximately 6% in opening pressure. However, Table 5-1 shows an increase of 30 psig in SRV setpoints. Discuss this discrepancy. What was the highest design SRV setpoint for which dynamic loads were originally calculated?

GPC Response:

The SRV setpoints for power uprate were increased 3%. For analytical purposes an additional 3% was included. See the GPC response to NRC question 22.

NRC Question 26:

(Section 5.0) Discuss your plant-specific methodology and how it is different from that in NEDC-31366. What is the confidence level in your plant-specific methodology?

GPC Response:

Originally, Hatch setpoints were created by conservatively adding margin to the analytical limits. Beginning in 1984, the methodology was changed for those instruments associated with the analog trip system.

The plant-specific methodology was approved by the NRC under the Plant Hatch Analog Transmitter Trip System Program. The methodology was reviewed and approved by the NRC as follows:

1. Unit 1 - Submittals are documented in GPC letters dated September 5, 1984, and July 24, 1985, with NRC approval in Technical Specifications Amendments 103 and 121, respectively.

2. Unit 2 - Submittals are documented in GPC letters dated February 23, 1983; January 23, 1984; and June 14, 1984.

This methodology, although plant-specific, is similar to the setpoint methodology given in GE topical report NEDC-31336, "Instrument Setpoint Methodology," October 1986.

Included in Enclosure 2 to this Response to RFAI is a description of Hatch's plant-specific methodology. The power uprate program is based on Plant Hatch's existing licensing basis. Note that when GPC submitted the setpoint methodology to the NRC, it was considered proprietary by GE. However, GE has reclassified this information and it is no longer considered proprietary.

Specific differences between the Plant Hatch method and NEDC-31336 are as follows:

a. Process measurement and process element accuracy are terms associated with the physical ability to measure a parameter within a device. At Plant Hatch these inaccuracies are accounted for between the process safety limit and analytical limit for Plant Hatch. NEDC-31336 addresses these terms between the allowable value and analytical limit.

b. The Plant Hatch methodology differs from NEDC-31336 regarding the calculation of the nominal trip setpoints (NTSP) and tends to be conservative as shown below:

$$\text{Hatch NTSP} = \text{Allowable Value} \pm (\,(\text{Trip Unit Drift})^2 + (\text{Transmitter Drift})^2\,)^{1/2}$$

$$\text{NEDC-31336 NTSP} = \text{Analytical Limit} \pm 1.645/2(\,(\text{Uncertainty 1})^2 + (\text{Uncertainty 2})^2 + \ldots)^{1/2}$$

The Plant Hatch methodology is provided in Enclosure 2.

NRC Question 27:

(Section 5.1.3) Provide the calculations and related documentation used to develop the setpoints for the following parameters: (a) Reactor Vessel Hi Pressure Scram, and (b) High Pressure Anticipated Transient Without Scram-Recirculation Pump Trip (ATWS-RPT).

GPC Response:

The subject setpoint calculations for Unit 2 are included in Enclosure 2. Similar calculations for Unit 1 are being revised to support implementation in Spring 1996.

NRC Question 28:

(Section 5.1.3) Provide the necessary information to determine if power uprate would result in any decrease in the margins between the allowable value for instrument setpoints and the analytical limits. Provide justification for any instrument where such condition occurs.

GPC Response:

The setpoint calculations that determine allowable values from analytical limits utilize vendor information on loop accuracy and calibration accuracy. The vendor information is constant and is not affected by power uprate. Therefore, a decrease in margins between the allowable value and the analytical limit should not occur.

NRC Question 29:

(Section 7.1) Provide assessments to demonstrate that: (1) the turbine missile failure probability will be within the staff's criteria for the uprated conditions; (2) the increased steam will not negatively affect the integrity of the long blades in the last few stages of the low pressure turbines; and (3) turbine vibration will be within the acceptable range under the increased steam flow.

GPC Response:

1. The missile analysis was calculated using the new uprated conditions and is within the guidelines set by the NRC for missile failure probability which are: P1<1x10E-5 for each unit and P1<1x10E-6 for each rotor. The results are acceptable.

2. All stages of the turbine were screened for excessive loadings and/or stress, and results show that the increased steam flow will not negatively affect the integrity of the long blades in the last few stages of the low pressure turbine.

3. A mechanical review of the turbine rotors was conducted to evaluate steady-state vibrational and upset stress conditions affected by uprated steam conditions. The results are acceptable. Therefore, turbine vibrations will be within the acceptable range under the increased steam flow.

NRC Question 30:

(Section 8.4) In accordance with Reference 3, paragraph 2.7(b), describe any changes to the reactor fuel design that will impact the isotopic concentrations of the radionuclides in the irradiated fuel. [Reference 3 is the NRC approval of NEDC-31897P-1.]

GPC Response:

General Electric has reviewed Reference 3, paragraph 2.7 (b) relative to the Plant Hatch power uprate program. Any changes made to core loading or design parameters to optimize power uprate operation will not significantly impact the isotopic concentrations of the radionuclides in the irradiated fuel.

NRC Question 31:

(Section 8.5.3) Some effects of the increased post-accident source term, due to the uprated power, are addressed in the January 13, 1995, submittal (Reference 1) with regard to the Control Room and the Technical Support Center. However, other vital areas that are required to have post-accident accessibility (per item II.B.2 of NUREG-0737) are missing. The most notable of these is the Post-accident Sample Sink. In accordance with Table 2, page T2-3, Reference 2, [GE Report NEDC-31984P] describe the impact on the accessibility of all the vital areas identified in response to NUREG-0737, item II.B.2, at Plant Hatch.

GPC Response:

Shielding analyses were performed for Plant Hatch to comply with the post-accident accessibility requirements of NUREG-0737, Section II.B.2. The analyses determined the dose rates and shielding requirements to ensure that personnel can access areas of the plant to aid in the mitigation of or recovery from an accident and that dose rates in unrestricted areas are within permitted levels. Among the areas analyzed, in addition to the Control room and the Technical Support Center, were the Post Accident Sampling System (PASS) Stations, and various locations on Elevation 130' of the Reactor Building, including the Hot Machine Shop and Health Physics Area.

Although the licensed reactor power was 2436 MWt, all the calculations used TID-14844 or equivalent source terms with a conservative power level of either 2537 or 2550 MWt. As the source terms are proportional to the reactor power, the uprated power level of 2558 MWt represents an increase of less than 1% in the source terms used in these analyses. Such a change in source terms is within the accuracy of the calculations and is negligible in light of the inherent conservatism in the source terms and the methodologies utilized. Shielding recommendations and time-motion studies would also be unaffected.

Based on the above discussions, it can be concluded that the change in source terms due to power uprate does not compromise the accessibility of vital areas of the plant following an accident.

NRC Question 32:

(Section 9.2) Tables 15-2 and 15-6 of Reference 4 lists the NRC staff's assumptions used to recalculate the consequences of a LOCA to support a recent license amendment for Unit 2. The staff notes that the results of the analysis documented in Reference 4 [NRC SER for Amendment 132] are significantly higher than the corresponding LOCA results in Table 9-3 of Reference 1 [GPC Power Uprate Submittal] even though the Reference 1 analysis is based on the uprated power level. Identify the assumptions and parameters that were used in calculating the LOCA consequences in Table 9-3 of Reference 1, and provide justification for those that are not consistent with the assumptions and parameters used by the NRC staff in Reference 4.

GPC Response:

The dose values in the power uprate submittal, Table 9-3 under the "2537 MWt" column were those submitted to the NRC on January 6, 1994, regarding GPC's request to delete the MSIV Leakage Control System and Increase Allowable MSIV Leakage. The "2664 MWt" column of Table 9-3 provides the uprated parameters. This column was derived by multiplying the "2537 MWt" column by 1.05 (5%). The NRC provided the results of its confirmatory analysis to GPC on March 17, 1994, in the SER for Amendment 132 to NPF-5. Table 32-1 summarizes the LOCA dose values from all three evaluations.

A review of Table 32-1 and the NRC SER indicates the most critical difference in the reported doses are those affiliated with the Main Control Room (MCR). This is because the NRC dose value for the MCR is reported as 29 rem. If this value is increased by 5% to account for power uprate conditions, it would exceed the 30 rem 10 CFR 50 Appendix A limit. The MCR dose calculated by GPC is considerably lower (15.0 rem).

The remaining discussions focus on the assumptions and parameters used in the GPC analysis as compared to those reported by the NRC in the SER for Amendment 132. Table 15.2 of the SER for Amendment 132 shows that the NRC used X/Q information from the original SER for E. I. Hatch Unit 2. The MSIV Leakage Increase submittal by GPC used X/Q information provided in Unit 2 FSAR Table 2.3-11.

A discussion of the formulation of the X/Q values used for the OFF-SITE LOCA results is provided in Section 2.3 of Hatch Unit 2 FSAR. The OFF-SITE LOCA dose discrepancy of 266 rem versus 214 rem is attributed primarily to the differences in these values. A review of the NRC assumptions (Table 15.2 of the NRC SER) versus those used by GPC indicates that GPC used 95% efficiency for the Standby Gas Treatment System (SGTS) versus the 99% efficiency assumed by the NRC. Other than the X/Q values and the SGTS efficiency, the assumptions are similar. Either the GPC or NRC results could be increased by 5% for Power Uprate conditions and remain within 10 CFR Part 100 limits.

MCR DOSE CALCULATION

Table 32-2 compares the assumptions between the NRC and GPC evaluations for the MCR. The NRC values are from Table 15.6 of the NRC SER for Amendment 132. The assumptions were similar with the exception of the X/Q values and Iodine Conversion Factors. GPC used Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Release of Reactor Effluents for the Purpose of Evaluating Compliance with

10 CFR 50, Appendix I" to determine iodine conversion factors. The differences in the X/Q calculations are detailed below.

The Main Control Room thyroid dose discrepancy of 29 rem (NRC) versus 15.0 (15.8 rem for Power Uprate) is also attributed primarily to X/Q differences. It appears the NRC utilized the Murphy-Campe (M-C) method. GPC derived its X/Q values for the Main Control Room air intake calculations using NUREG/CR-5055: Atmospheric Diffusion for Control Room Habitability.

Bechtel performed two different calculations based on M-C (circa 1974) and NRC NUREG/CR-5055 (circa 1988) to assess control room habitability following a postulated LOCA at Plant Hatch. To be conservative, only half of the building wake dispersion credit generated by the contiguous composite building complex for both generating units was accounted for in the calculation using the M-C method. It is worth noting that the results of the Bechtel calculation using M-C are very similar to the NRC's results reported in the Amendment 132 SER. The second calculation, which used NUREG/CR-5055 is the basis for the values in the Power Uprate submittal, Table 9-3, and GPC's submittal to the NRC to increase allowable MSIV leakage.

The appropriateness of using the more recent NUREG guidance and the differences in the resulting X/Q values when compared to the M-C method is summarized in Table 32-3. NUREG/CR-5055 is the industrial recognized NRC guidance for performing this type of analysis. The model formulation was developed from a wide spectrum of plant configurations and field measurements. Therefore, the X/Q values calculated by the NUREG method are more representative of the actual plant characteristics than those calculated by the M-C method.

Table 32-4 provides the GPC-developed X/Q values for the Main Control Room dose calculations. It is these X/Q values which determined the Power Uprate Main Control Room thyroid dose of 15.8 rem.

Table 32-1

LOCA Accident

	Power Uprate Submittal Table 9-3 Values (2664 MWt)	MSIV Leakage Increase Submittal by GPC Table 2 (Sheet 1 of 2)	NRC-SER Amendment 132 Table 15.1 and 15.6 Information
Thyroid Dose(1) Exclusion Area (0-2 hrs.)	95 rem	90.517 rem	60 rem
Thyroid Dose(1) LPZ (30 days)	224.6 rem	213.59 rem	266 rem
Thyroid Dose(2) Main Control Room (30 days)	15.8 rem	15.0 rem	29 rem

(1) Offsite Doses Remain Below 10 CFR Part 100 Limits at Power Uprate Conditions for both the GPC and NRC Evaluations.

(2) NRC Dose for MCR would be 30.5 rem if multiplied by 1.05 for Power Uprate. The 10 CFR 50 Appendix A limit is 30 rem.

Table 32-2

Comparison of NRC and GPC Assumptions for
MCR Dose Determination Following a LOCA

Parameter	*NRC*	*GPC*
Control room free volume	9.35 E + 4 ft^3	Same
Recirculation Rates Filtered Intake Unfiltered Intake Filtered Recirculation Filter Efficiency (2 inch charcoal)	 400 CFM 0 CFM 2100 CFM 95%	 Same
Unfiltered control room infiltration rate (assumed)	10 CFM	Same
Duration of accident	30 days	Same
Breathing rate of operators in control room for the course of the accident	3.47E-04 m^3/sec	Same
Meteorology (wind speeds for all sectors: 0 - 8 hours (sec/m^3) 8 - 24 hours (sec/m^3) 24 - 96 hours (sec/m^3) 96 - 720 hours (sec/m^3)	 1.7E-03 1.0E-03 5.6E-04 1.3E-04	 3.75×10^{-4} 2.55×10^{-4} 1.85×10^{-4} 1.05×10^{-4}
Iodine protection factor	100	80[(c)]
Iodine dose conversion factors[(a)]	ICRP-30	RG 1.109
Control Room operator occupational factors 0 - 8 hours 8 - 24 hours 24 - 96 hours 96 - 720 hours	 1 1 0.6 0.4	 Same
Doses to control room operators: Thyroid dose[(a)] Whole body dose[(b)]	 29 rem <1 rem	 15.01 rem <1 rem

(a) Unweighted dose equivalent.
(b) Unweighted dose equivalent (red bone marrow) due to immersion in an infinite cloud.
(c) Used actual flow model in lieu of protection factor. The value of 80 assumes 10 CFM unfiltered control room infiltration.

Table 32-3

Difference	Murphy-Campe	NUREG/CR-5055
1. Data Base	EBR-II wind tunnel tests (Hilitsky, 1963)	7 field experiments: EBR-II, MTR-ETR, TMI, Rancho Seco, EOCR, Duane Arnold, and Millstone.
2. Reactor Bldg.	Round dome-type	Dome/Block-type.
3. Model Formulation	Not known	Paramatization/ Regression Analysis.
4. Conservatism		
a. Release building shape	Assume round dome structure	Not limited to round structures (Block-type building is known to create more turbulence than round objects. Thus enhanced dispersion is achieved).
b. Wind speed dispersion credit	For ground releases, model used wind speed at 10m level to calculate X/Qs which result in over conservative values. Instead, wind speed at a level above the reactor equivalent to the EBR-II tests should be used. The Murphy-Campe application of wind speed yields X/Qs that are conservative by a factor of 2 or higher than their corresponding tunnel measurements.	For ground releases, model was formulated based on wind speed at 10m level.
c. Simulation ability	Wind tunnel results often overestimate field measurements due to its inability to account for the additional dispersion generated by the wind random fluctuations.	Model development was based on field data that fully accounted for the dispersion capability of the environment.

Table 32-4

GPC-Developed Main Control Room X/Qs

Averaging Period	X/Q (sec/m^3)	
Ground Level	Unit 1	Unit 2
0-2 hours	4.38 E-4	4.38 E-4
2-8 hours	3.35 E-4	3.35 E-4
8-24 hours	2.60 E-4	2.55 E-4
1-4 days	1.85 E-4	1.75 E-4
4-30 days	1.15 E-4	1.05 E-4
Stack Releases	Both Units	
0-2 hours	7.41 E-6	
2-8 hours	4.41 E-6	
8-24 hours	3.41 E-6	
1-4 days	1.94 E-6	
4-30 days	8.68 E-7	

NRC Question 33:

(Sections 10.1 and 10.1.1) Discuss the methodologies and assumptions used in the evaluation regarding the effects of the power uprate on the design basis analyses of the high energy line break locations, and pipe-whip and jet impingement loads, in accordance with the plant licensing criteria.

GPC Response:

1. *High Energy Line Break (HELB) Locations*

   The HELB locations for Units 1 and 2 were evaluated in accordance with the limits specified in Branch Technical Position MEB 3-1 for the appropriate classification of piping. These reviews included maximum stress, maximum stress range, and cumulative fatigue usage as specified in the Branch Technical Position. No new break locations are postulated, because the power uprate values do not exceed any Branch Technical Position criteria.

2. *Pipe Whip Restraints*

   a. *Unit 1 Pipe Whip Restraints*

      The Unit 1 pipe whip restraints are made of structural steel members. Energy absorbing materials and U-bolts were not used in the design. The pipe whip restraints are designed in accordance with AISC Code (reference FSAR Section N.5.1.1.b) for an equivalent static jet force using a 50-percent increase in allowable stresses.

      Yield strength of ductile materials, such as A36 (used for Unit 1 pipe whip restraints) is highly influenced by the rate of loading. Bechtel Topical Report, BN-TOP-2, "Design for Pipe Break Effects" is NRC-approved and part of the Hatch licensing basis. It allows for a 10% increase in yield strength, although other data showed a significantly higher increase. The actual increase in jet force due to power uprate was typically determined to be less than 4-percent; however, a bounding value of 5-percent (maximum) increase in load was used to perform the uprate evaluation. Therefore, based on the above, it can be concluded that the Unit 1 pipe whip restraints can accommodate an increase of 5% in the jet force, with a minimum of 4 percent margin remaining between actual and allowable (minimum yield) stress.

b. Unit 2 Pipe Whip Restraints

The Unit 2 pipe whip restraints were designed in accordance with the energy balance approach outlined in BN-TOP-2, which assumes a maximum steady-state force. To calculate the design force on the pipe whip restraint, the jet force is multiplied by a dynamic load factor. This is a conservative approach, since this force is not a steady-state force but a time-history function which peaks quickly and then decreases with time. Generic analysis performed for whip restraints similar to Hatch, show substantial load reductions result if the time-history approach is used. This reduction in calculated load is much greater than the 4-5% increase expected for power uprate conditions. This generic analysis was used for evaluation of the Hatch Unit 2 restraints. The paragraph below discuss the energy balance approach and the force time-history approach used for the Unit 2 reanalysis.

The energy balance approach considers elastic, elasto-plastic, and plastic design limits for the pipe whip restraint components. The energy balance approach is an iterative process based on balancing the kinetic energy of the pipe following the break with the potential energy of the pipe whip restraint. The kinetic energy of the pipe is based upon a steady-state force.

For pipe whip restraints with zero gap during normal plant operation, the dynamic load factor of 2.0 is used in accordance with BN-TOP-2 (reference FSAR 15.A.5.1.1). For pipe whip restraints with a gap between the pipe and the restraints, the dynamic load factor is calculated using the energy balance approach. The non-linear time-history analysis approach has not been used in the design of Unit 2 pipe whip restraints.

Based on the previous generic analysis of pipe whip restraints similar to those installed at Plant Hatch, it was determined that, if the non-linear time-history approach is utilized, a lesser amount of deformation of the energy-absorbing material is required to balance the pipe whip force than previously calculated using the energy balance approach. This equates to additional capacity in the energy-absorbing material/U-bolt structure, since the previous deformation calculations were based on the pipe whip restraint properties which did not change. The size, shape, materials, and jet forces used for whip restraints used at Plant Hatch were compared and found similar to those used in the previous generic analysis. Based on the conclusions of the analysis, the deformation margin (which is a function of the jet force, gap between the pipe whip restraints and pipe, and height of the energy-absorbing material or length of the U-bolt) for a representative sample of pipe whip restraints varies between 1.34 and 12.5.

Therefore, it can be concluded that a 5-percent increase in jet force is acceptable for the deformation margins of 1.34 and 12.5, since additional energy required to be absorbed by the pipe whip restraints due to 5-percent increase in jet force is much smaller than spare energy available if a time-history analysis is used.

3. *Jet Impingement Barriers*

The jet impingement loads for Plant Hatch Units 1 and 2 are derived using methods described in FSAR Sections N5.1.1 and 15.A.5.1.1, with a dynamic load factor of 1.25 and 2.0, respectively. Ductile behavior of steel was not considered in the design. In addition, energy balance of the time-history approach has not been used in the design.

The maximum allowable span lengths of conduit and instrument tubing subjected to jet impingement forces conservatively use a shape/curvature factor of 1.0 for conduits/tubing. However, the Hatch methodology (Reference BN-TOP-2) allows a shape of $2/\pi$ or 0.64. Use of a curvature factor of 0.64 results in a design margin of ≈ 36 percent. In addition, the Plant Hatch licensing basis allows a 50-percent increase in allowable stresses for jet impingement loads. Based on this, a design margin of at least 50 percent exists for jet impingement barriers with the exception of the 1-in.-diameter wedge anchors for the recirculation pump discharge valve jet impingement barrier which has a design margin of ≈11 percent.

Although the actual increase in jet force due to power uprate was typically determined to be less than 4 percent, a bounding value of 5-percent (maximum) increase in load was used to perform the uprate evaluation. Therefore, it is concluded that the jet impingement barriers are acceptable for power uprate.

<u>NRC Question 34</u>:

(Section 10.2) The power uprate submittal indicated that the qualification of mechanical equipment/components (pumps, valves, heat exchangers, etc.) are affected by operation at the uprated power conditions. Provide evaluation of the effect of the increased SRV loads and jet impingement on these components.

<u>GPC Response</u>:

To assess the mechanical design adequacy of equipment/components, all equipment in the systems impacted by power uprate was reviewed to determine the acceptability for operation at the new uprate conditions. In all cases, the designed equipment capability bounds the marginal increases in system pressure-, temperature-, and flow-associated loads due to power uprate.

SRV loads are addressed in Section 4.1.2.2 of NEDC-32405P. Evaluations determined that there is no additional impact due to power uprate (other than opening SRV setpoints) on the SRV actuation loads.

HELB and jet impingement are addressed in Sections 10.1.1 and 10.1.2 of NEDC-32405P. The results of the original HELB and jet impingement analyses will not be impacted as a result of power uprate.

NRC Question 35:

(Section 10.3) A recent abnormal reactor recirculation pump vibration problem was identified at Susquehanna during startup testing at the power uprate conditions. Provide an evaluation of the increased flow-induced dynamic loads on the recirculation piping and components such as pumps and flow control valves. Provide assurance that similar excessive recirculation pump vibration problem will not occur at Hatch for the power uprate conditions.

GPC Response:

The Susquehanna vibration problem was associated with an increase in their licensed core flow rather than power uprate. As discussed in the response to Question 13, GPC is not requesting an increase in licensed core flow and expects no recirculation pump vibration problems.

NRC Question 36:

(Section 10.4) Address whether the power uprate will have any effect on operator reliability and performance. Other submittals related to power uprate have addressed this with respect to operator action times assumed in the Individual Plant Examination [IPE] Human Reliability Analysis.

GPC Response:

GPC compared the potential impact of power uprate on operator actions modeled in the IPE with the GE generic evaluation presented in Supplement 2 of NEDC-31984P. The conclusion is that power uprate will not significantly impact operator reliability and performance. (The Plant Hatch IPE used the Success Likelihood Index Method (SLIM) for human reliability analysis.) This conclusion was based on a qualitative review of the human reliability analysis with recalculation of selected operator response times. The actual human error probabilities were not requantified, because the SLIM used in the IPE is based on expert elicitation rather than time-dependent equations. Two examples of operator actions potentially sensitive to power uprate are discussed below.

1. One of the most risk-significant actions in the IPE is the action to perform emergency depressurization. The time available to perform this action, given a loss of all high pressure injection, is approximately 36 minutes from the initial scram. Power uprate conditions which could affect this timing are the slightly higher initial inventory temperature, the potentially higher flow rate through the SRVs, and the higher decay heat present after the scram. Based on 105-percent initial power, the time available for this action is reduced from 36 minutes to 32 minutes. The decrease in available response time is judged to have an insignificant impact on the human error probability for this action. The dynamic action to depressurize requires only a fraction of the time available, and is influenced primarily by other factors such as the presence of high drywell pressure or temperature, and the success of the preceding action to inhibit the Automatic Depressurization System (ADS).

2. A second action judged to be potentially sensitive to initial power is the action to initiate Standby Liquid Control (SLC) during an MSIV closure and ATWS, before the Boron Injection Initiation Temperature (BIIT) is exceeded. Failure of all rods to insert from 105 percent will result in more steam being dumped to the suppression pool, resulting in a more rapid increase of pool temperature. The human error probability for the IPE was based on a general time frame of 1 to 3 minutes. Because the suppression pool temperature is only affected by the energy transferred from the vessel via the SRVs the time available for SLC system initiation is estimated to be inversely proportional to the initial power. (HPCI and RCIC turbine exhaust will tend to reduce the energy transferred to the pool.) Thus, the time available for SLC initiation will be reduced by approximately 5 percent. This is within the uncertainty bounds provided to the SLIM evaluators (1 to 3 minutes), and thus, GPC concluded that the original SLIM human error rate will remain applicable.

<u>NRC Question 37</u>:

(Section 11.1.2.3) Discuss whether the power uprate will change the type, scope or time requirements of operator actions needed for accident mitigation, including the type and scope of plant procedure changes, and any anticipated changes in the scope or nature of operator response. Provide copies of the procedure steps in the emergency and abnormal operating procedures which will change as a result of the above question. Also, describe these procedural changes resulting from the power uprate.

GPC Response:

The EOPs for Plant Hatch are symptom based. Changes to the Emergency Operating Procedures (EOPs) and the abnormal operating procedures (accident mitigation procedures) required for power uprate implementation are revisions to previously defined numerical values only (e.g., RPV high pressure scram setpoint value). The definition of these parameters has not been altered, only the numerical value of the parameter has changed. As such, the type, scope, and nature of the operator actions required for accident mitigation are unchanged. No new types of operator actions are necessary.

As discussed in the GPC Response to Question 36, the response time for some operator actions during dynamic accident events at power uprate conditions will be slightly shorter when compared to the same events at pre-uprate conditions. However, the change in response time is not significant. The operating crew will still be able to successfully implement EOP actions. The type and scope of the operator actions remain unchanged. The accident mitigation strategy of the EOPs will not change. A procedure revision, other than numerical value changes, is not required.

ADDITIONAL REQUESTED INFORMATION

At the request of the NRC program manager, GPC is providing responses to eight additional questions addressing environmental impact. The questions are the same as the NRC requests for additional information regarding Philadelphia Electric's power uprate submittal for its Peach Bottom Plant.

NRC Question 38:

(Section 11.3) Is additional water to be withdrawn from the Altamaha River or other water sources, such as ground water, in order to support the Power Uprate? If so, what are the effects of additional water withdrawal?

GPC Response:

The flow rates of the circulating water systems are not changed due to power uprate, as discussed in the GPC power uprate submittal. Other plant water requirements, such as makeup to the water treatment plant and fire protection system, are also unaffected.

The river water withdrawal rate is expected to increase slightly due to increased cooling tower evaporation and the corresponding increase in cooling tower make up. The slight increase in makeup due to evaporation is not significant and is enveloped by the river water withdrawal rates discussed in the final environmental statement (FES) and rates approved under the current Georgia Surface Water Withdrawal Permit.

NRC Question 39:

(Section 11.3) Are changes to the environmental protection plan (EPP) required for uprate?

GPC Response:

A proposed EPP for Plant Hatch is currently under Staff review as part of the Improved Technical Specifications. Nc substantive changes to the current Environmental Technical Specifications (ETS) or the proposed EPP are anticipated for power uprate.

NRC Question 40:

(Section 11.3) What are the effects of any increased noise levels attributed to the power uprate, including the noise from the cooling tower?

GPC Response:

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 stepup transformers will operate at an increased KVA level; however, the overall noise level will not increase significantly.

NRC Question 41:

(Section 11.3) Will there be an increase in the amount of fuel used for the power uprate? If so, what are the potential environmental effects (e.g., waste volume, curie content, radiation exposure)? Will the quantity of U-235 be consumed during operation of the plant increase significantly?

GPC Response:

The increased energy requirements associated with power uprate can be accommodated by either an increase in the reload batch size or an increase in fuel bundle enrichment for the same reload batch size. Although the initial reloads for power uprate will contain more fuel bundles, energy requirements in future cycles are expected to be accommodated primarily by an increase in the reload fuel enrichment. Thus, the number of fuel assembles requiring ultimate disposal should not be significantly impacted by power uprate. Furthermore, the improvements in reload fuel nuclear efficiency, since the FES was issued, should offset the increased U-235 requirements associated with power uprate and result in approximately the same overall U-235 consumption.

Due to the higher steady-state operating power associated with power uprate, the curie content of the reactor fuel will increase; however, the change in environmental impact of radioactive material releases due to operation at uprated power levels was reviewed, and found to be insignificant. These releases will remain well within the regulatory limits. More detailed discussions on changes in radiation levels are discussed in section 8.5 (Radiation Levels) and 9.2 (Design Basis Accidents) of the GPC January 13th submittal.

NRC Question 42:

(Section 11.3) If there are any, what are the changes and the effects from changes to the river water discharge flowrate, velocity, temperature and thermal plume, or chemical composition of the discharge? What are the effects to the various aquatic plant and fish species (e.g., will there be an increase in entrainment of plankton organisms? Will there be an increase in impingement of fish?

GPC Response:

The river water discharge flow rate and velocity from the plant are not substantively changed by operation at power uprate conditions. Operating at power uprate will result in slightly higher heat loads being rejected by the cooling towers. The resulting contaminant concentration in the towers will increase slightly due to evaporation; however, concentration changes in the cooling tower are not significant. No significant change to the chemical concentration of discharge will result.

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 remain valid for the uprated condition.

Changes in intake canal velocity resulting from the slight increase in river water withdrawal rates to accommodate increased cooling tower makeup requirements from power uprate are not significant. No measurable effects on fish impingement or plankton entrainment are expected.

NRC Question 43:

(Section 11.3) Is the cooling tower discharge temperature and discharge temperature to the river expected to increase with uprate? Is this within FES limits?

GPC Response:

The temperature of cooling tower blowdown will increase slightly (<1°F) due to power uprate. A corresponding slight increase in discharge temperature to the river will occur. The slight increase in temperature is bounded by thermal studies conducted during the licensing of the plant. The conclusions of these studies and the FES relative to temperature remain valid for uprated power conditions.

Enclosure 2 of the January 13th GPC power uprate submittal details the expected increase of 1°F, and documents the expected temperature of 93°F projected for uprate.

NRC Question 44:

(Section 11.3) What are the effects on the terrestrial environmental (vegetation and solids) due to the additional emissions from the cooling towers? What is the expected increase in the amount of cooling drift fog due to the power uprate and what are the environmental effects?

GPC Response:

The circulating water and service water flow through the cooling towers is not changed due to power uprate. The cooling towers' duty cycle will increase due to power uprate, resulting in increased evaporation. Any increase in cooling tower drift associated with uprate is enveloped by the bounding conditions of the FES. The conclusions of the FES relative to cooling tower drift impacts remain valid for power uprate conditions.

NRC Question 45:

(Section 11.3) Are there any increases in the makeup requirements for various plant systems (condensate system, feedwater system, component cooling water, recirculating system, etc.) and if so what are the environmental effects?

<u>**GPC Response**</u>:

Makeup water requirements will not change as the result of operating at uprate power levels for any of the systems listed. The only potential change is due to increased reactor operating pressure, which could slightly increase leakage through valve packing. This higher leakage rate slightly increases the liquid radwaste processing load, which is processed and returned to the condensate storage tank for reuse.

Enclosure 2

Edwin I. Hatch Nuclear Plant
Setpoint Methodology and Selected Setpoint Calculations

Enclosure 2

Edwin I. Hatch Nuclear Plant
Setpoint Methodology and Selected Setpoint Calculations

This enclosure provides the additional information requested in NRC Questions 26 and 27 of Enclosure 1.

A. Setpoint Methodology

B. Calculations

- ATWS/RPT High Reactor Pressure
- Reactor High Pressure Scram

SETPOINT METHODOLOGY

1.0 General Description

The Hatch technical specification trip setpoints and allowable values are derived from the analytical limits. The analytical limits are process variables associated with the analyzed abnormal plant transients or accidents described in the Hatch FSAR. In performing the safety analysis for the original Hatch FSAR, analytical limits were selected, following an iterative process, to ensure that safety limits identified in the technical specifications were not exceeded in postulated transient events or accidents. Except for those setpoints which are proposed to be revised, with justification provided in the Hatch License Submittal, the original FSAR analytical limits were used in the determination of allowable values and trip setpoints. In essence the analytical limits utilized to generate Hatch ATTS instrument allowable values and trip setpoints are consistent with the licensing basis documented in the Hatch FSAR.

As illustrated in Figure 1, once the analytical limit is identified, the allowable value is determined by including the margin to account for instrument and calibration accuracy. Typically the allowable value is determined so that there is at least a 95% confidence level of providing the trip action before the process variable reaches the analytical limit assuming the maximum loop drift. The instrument accuracy is defined in the performance specification prepared by the manufacturer. These specifications envelope Plant Hatch's specific requirements. Operating experience has shown that, using standard techniques and test equipment, plant technicians can calibrate the instrument with an accuracy 0.25% of span. The instrument accuracy and calibration accuracy are considered to be independent variables. Thus, these two margins are com-

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bined statistically to determine the allowable value. To statistically combine the independent variables, the square root of the sum of square (SRSS) method was used. To achieve a 95% confidence level, two standard deviation values on instrument accuracy ($2\sigma_I$) and two standard deviation values on calibration accuracy ($2\sigma_C$) are combined; (i.e.,

$$\sigma_{AV} = \pm 2\left(\sqrt{\sigma_I^2 + \sigma_C^2}\right)$$

where σ_{AV} is the combined standard deviation for allowable value.

Once the allowable value is determined, the trip setpoint is established by including the margin to account for instrument drift. The drift specified in the manufacturer's issued performance specification was used. The linear extrapolation technique was used to derive the drift over the technical specification specified surveillance interval (e.g., if a transmitter performance specification specifies a drift value based on a 6-month interval, the transmitter drift to be considered over a 18-month period is the 6-month value multiplied by 3).

After the trip setpoint is established, statistical tests are performed to ensure that the trip setpoint has acceptable margin. (Margins are added to avoid exceeding the allowable value and to avoid spurious trips during normal power operation.) The acceptable level for avoiding violation of allowable values and undesirable reactor trips is set at equal to or greater than 90 percent assuming a normal setpoint distribution.

Field data and operating experience have shown that transmitters and trip units tend to become more stable and drift less if unnecessary adjustments are reduced. To improve the reliability of the system the trip unit setpoints are monitored in situ monthly. A "leave-alone range" is added as a dependent variable to the total margin for drift. Figure 1 provides an illustration of the safety limit and setpoint relationship.

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1.1 Location of Major Components

All ATTS sensors are mounted in the reactor building, and would be subjected to the harsh environment resulting from a high energy line break (HELB) outside primary containment. Trip units and power supplies are located in the main control room and are subjected to the mild environment.

1.2 Methodology for Accuracy, Calibration and Drift

Based on vendor supplied performance data, the 95% confidence level (i.e., 2 standard deviations over the instrument's calibrated span) design margins are used for the Hatch setpoint calculations. The instrument loop margin is determined by taking the SRSS of the margins of the sensor (i.e., transmitter or RTD) and its associated trip unit. The margins of the transmitter and of the trip unit are independent variables thus they are statistically combined; i.e.,

$$\pm 2 \sqrt{\sigma^2_{\text{sensor}} + \sigma^2_{\text{trip unit}}}$$

The dominant factor in determining the design margin for instrument accuracy is the ambient temperature effect on the transmitter. The temperature effect is a function of the mounting location of the transmitter. Based on the results of the previously performed high energy line break (HELB) analysis, ambient temperature vs. time curves for all GE supplied transmitters were plotted to identify the maximum ambient temperature for each transmitter. The maximum transmitter operating temperatures were determined by the vendor from the temperature profiles and thermal time constant of the transmitter housing.

The harsh environments for HELB include temperature and radiation effects. However, the mechanisms of these two parameters on the transmitter performance are significantly different. The temperature

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effect on transmitter performance is almost instantaneous because the temperature peaks in the order of 10 seconds following an HELB. On the other hand, the radiation effect is accumulative i.e., radiation will not affect transmitter performance until it has accumulated a significant total integrated dose. The Barton transmitter qualification test shows that at a rate of 10^6 rad per hour for a period of more than an hour, (a condition close to the design basis event) there is no effect on the transmitter's performance. In many cases, the period is as long as four to five hours. Hatch HELB analysis shows that the automatic function for all maximum HELB events occurs within 20 minutes. For a smaller line break, the event duration could exceed 20 minutes. However, the dose rate would be less and the required accumulative period would be longer. Consequently the radiation effect for instrument setpoint is negligible and it is excluded from the Hatch setpoint calculations.

1.2.1 Design Margin for Sensor Accuracy

In calculating the design margin for the transmitter accuracy the following factors are included; reference accuracy, power supply effect, temperature effect, and static pressure effect. The static pressure effect is only applicable to differential pressure transmitters. The uncertainty due to reference accuracy, power supply, temperature, and static pressure as defined in vendor supplied performance specifications (which envelop Plant Hatch specific requirements) are used for the calculations. Applying the SRSS method, the transmitter design margin is calculated as follows:

Transmitter accuracy margin

$$= \pm \sqrt{\begin{pmatrix}\text{Reference}\\ \text{Accuracy}\end{pmatrix}^2 + \begin{pmatrix}\text{Power Supply}\\ \text{Effect}\end{pmatrix}^2 + \begin{pmatrix}\text{Temperature}\\ \text{Effect}\end{pmatrix}^2 + \begin{pmatrix}\text{Static}\\ \text{Pressure Effect}\end{pmatrix}^2}$$

For the RTD's, the design margin for sensor accuracy is calculated as follows:

RTD accuracy margin

$$= \pm \sqrt{\begin{pmatrix}\text{Reference}\\ \text{Inaccuracy}\end{pmatrix}^2 + \begin{pmatrix}\text{Interchange}\\ \text{Ability}\end{pmatrix}^2 + (\text{Repeatability})^2}$$

The trip units are located in the control room (mild environment), thus the only variable affecting the trip unit accuracy is the reference accuracy determined by the vendor. The combined sensor and trip unit accuracy is established by taking the SRSS of the sensor accuracy and trip unit accuracy.

1.3 Design Margin for Calibration Accuracy

The trip unit is calibrated using the Calibration Unit (CU) and Readout Assembly (RA). The CU generates a ramp signal for input to the trip unit. When the ramp signal reaches the trip unit setpoint, the trip and trip status outputs change state. When the trip status output changes state, the ramp signal is latched and displayed by the RA as the value at which the trip setpoint is set. An error source is present due to the time delay of the trip unit. The time delay is primarily a function of the input filter on the trip unit. The trip unit input filter has a break point of 250 Hz which corresponds to a 4 msec time delay. The ramp signal has a maximum ramp rate of 1.1 mA/sec, which translates into an error of 0.0044 mA, or a calibration uncertainty of 0.0275% FS (for a 4-20 mA range trip unit input, the calibrated span is 16 mA).

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The uncertainty of the Readout Assembly (RA) which is used to display the values of the trip setpoint also contributes to trip unit calibration uncertainty. The RA has an accuracy of ±0.01 mA over the range of 0-20 mA. This tolerance is equivalent to a calibration uncertainty of 0.05% FS. By using the SRSS method, the trip unit calibration allowance was calculated by statistically combining the time delay error and RA error.

It is required that calibration equipment has an accuracy equal to or less than ±0.25% of the span of the instrument under test. The transmitter calibration design margin is established by statistically combining the calibration equipment accuracy margins. The transmitter calibration design margin is equal to ±0.35% of span.

The loop calibration design margin is established using the SRSS method combining both trip unit and transmitter calibration design margins. By design and operation principle, the RTD's do not require any calibration. To be conservative and consistent, it is assumed that the combined RTD/trip unit margin for calibration is identical to that of the transmitter/trip unit.

The equations used in calculating the calibration margins are provided in Section 1.5.

1.4 Design Margin for Drift

The long term drift specified by the transmitter vendor performance specifications are linearly extrapolated to determine the design margin drift for transmitter. Ongoing vendor test data indicated that linear extrapolation is a conservative approach. The trip unit 6-month drift specification was used even though the trip units are monitored monthly. The transmitter drift and trip unit drift are statistically combined to determine the loop drift. A ±0.25% of span "leave alone" range is added as a dependent variable to the loop drift to determine the nominal trip setpoint. If a trip unit setpoint is

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found to be within the "leave alone" range during the monthly surveillance test, the trip unit setpoint is left as is to prevent the accelerated wear of the adjustment potentiometer. If the setpoint is outside the "leave alone" range, the trip unit is recalibrated to the nominal trip setpoint. The combined loop drift including the "leave alone" band is calculated as follows:

ATTS loop drift

$$= \pm \left[\sqrt{(\text{Trip Unit Drift})^2 + (\text{Transmitter Drift})^2} + 0.25\% \right]$$

The probability of avoiding an undesirable trip or of not exceeding the allowable value are evaluated for combined ATTS accuracy to ensure that better than 90% probability is attained. The equations used in these evaluations are discussed in Section 1.5.

1.5 Example - Main Steam Line High Flow Trip (Hatch 2)

Setpoint calculated per Reference 5.

System Values

Transmitter Range: 0-150 psid
Maximum Operating Temp: 125°F
Calibration Reference Temp: 70°
Analytical Flow Limit: 140% of Rated Flow
Analytical dP Limit (Based on Flow Element Curve): 127 psid
Extreme Steady State Value in dP: X_o = 71.44 psid (105% Rated Flow)
Magnitude of Limiting Transient: T = 3% Rated Flow (4.14 psid)
Standard Deviation of Limiting Transient: σ_m = 1% Rated Flow
Exceeding Tech Spec Limit Avoidance Probability: 90%
System Unacceptable Trip Avoidance Probability: 90%

Transmitter Accuracy Values

Reference Accuracy: $\pm 0.5\%$ of Span

Temperature Effect: $\pm 1\% \left(\frac{125-70}{100}\right) = \pm 0.55\%$ of Span

Static Pressure Effect: $\pm 0.5\%$ of Span for 1000 psig

*Power Supply Effect: $\pm .025\%$ per volt + $\pm .075\%$ of Span

Transmitter Drift: $\pm 1\%$ of Span per 12 Months

Transmitter Calibration: $\pm 0.35\%$ of Span

*Power supply is 25 VDC $\pm$ 1.5 VDC or 3 volt maximum deviation.

Trip Unit Accuracy Values**

Reference Accuracy: $\pm 0.32\%$ of Span

Trip Unit Drift: $\pm 0.35\%$ of Span per 6 months

Read Out Resolutions: $\pm 0.05\%$ of Span

Calibration Current: $\pm 0.0275\%$ of Span

**All values are for the MTU and STU combined.

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ATTS Accuracy

ATTS Accuracy = $\pm\sqrt{(\text{loop ACCY})^2 + (\text{Cal ACCY})^2}$; ACCY = Accuracy, CAL = Calibration

Loop ACCY = $2\sigma_A = \pm\sqrt{(\text{Tx ACCY})^2 + (\text{Tu ACCY})^2}$; Tu = Trip unit, Tx = Transmitter

Tx ACCY = $\pm\sqrt{(\text{Tx REF.})^2 + (\text{TEMP. EFF.})^2 + (\text{STAT. PRES. EFF.})^2 + (\text{Ps. EFF})^2}$; EFF = Effect, TEMP = Temperature, STAT = Static, PRES = Pressure, Ps = Power supply

Tu ACCY = TRIP UNIT REFERENCE ACCURACY

Tu CAL ACCY = $\pm\sqrt{(\text{READ OUT RESOLUTION})^2 + (\text{CAL CURRENT})^2}$

Determine Transmitter Accuracy; Tx ACCY

Tx ACCY = $\pm\sqrt{(\text{REF ACCY})^2 + (\text{TEMP EFF})^2 + (\text{STAT PRES EFF})^2 + (\text{Ps EFF})^2}$

Tx ACCY = $\pm\sqrt{(0.5)^2 + (0.55)^2 + (0.5)^2 + (0.075)^2} = \pm 0.90$

Determine Loop Accuracy; $2\sigma_A$

LOOP ACCY = $2\sigma_A = \pm\sqrt{(\text{Tx ACCY})^2 + (\text{REF ACCY Tu})^2}$

$2\sigma_A = \pm\sqrt{(0.90)^2 + (0.32)^2} = \pm 0.96$

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Determine Calibration Accuracy; $2\sigma_C$

CAL accuracy; $2\sigma_C = \pm\sqrt{(\text{Tx CAL ACCY})^2 + (\text{READ OUT ACCY})^2 + (\text{CAL CUR})^2}$; CUR=Current

$$2\sigma_C = \pm\sqrt{(0.35)^2 + (0.05)^2 + (0.0275)^2} = \pm 0.355$$

$$\text{ATTS ACCY} = \pm\sqrt{(\text{LOOP ACCY})^2 + (\text{CAL ACCY})^2}$$

$$\text{ATTS ACCY} = \pm\sqrt{(0.96)^2 + (0.355)^2} = \pm 1.0 \text{ of Span}$$

Determine the Allowable Value (AV)

AV = AL-ATTS ACCY

AV = 127-150 (0.01) = 125.5 psid (rounded off in safe direction to 125 psid)

Determine ATTS Drift (for 24 Month Xmtr Cal)

$$\text{ATTS DRIFT} = 2\sigma_D = \pm\left[\sqrt{(\text{Tx DRIFT})^2 + (\text{Tu DRIFT})^2} + 0.25\right]$$

$$\text{ATTS DRIFT} = \pm\left[\sqrt{(2\text{x}1)^2 + (0.35)^2} + .25\right] = \pm 2.28\% \text{ of Span}$$

Determine the Nominal Trip Setpoint (NTS)

NTS = AV-ATTS DRIFT

NTS = 125-150 (0.0228) = 121.58 psi (rounded in safe direction to 120 psid)

Test for Exceeding AV Avoidance (EAVA) Probability

$$\text{EAVA } \% = \frac{1}{2\pi}\int_{-\infty}^{Z} e^{-0.5t^2}\,dt$$

$$\text{WHERE } Z = \frac{AV - NTS}{\sqrt{(\sigma_A)^2 + (\sigma_C)^2 + (\sigma_D)^2}} = \frac{83.3 - 80.0}{\sqrt{(0.48)^2 + (0.18)^2 + (1.14)^2}} = 2.64$$

$$\text{EAVA } \% = \frac{1}{2\pi}\int_{-\infty}^{2.64} e^{-0.5t^2}\,dt \cong 100\% \text{ (FROM PROBABILITY TABLES)}$$

<u>Test for Unacceptable Trip Avoidance (UTA) Probability</u>

$$\text{UTA } \% = \frac{1}{2\pi}\int_{-\infty}^{Z} e^{-0.5t^2}\,dt$$

$$\text{WHERE } Z = \frac{NTS-(Xo + T)}{\sqrt{(\sigma_M)^2 + (\sigma_A)^2 + (\sigma_C)^2 + (\sigma_D)^2}} = \frac{80.0 - (47.63 + 2.76)}{\sqrt{(0.93)^2 + (0.48)^2 + (0.18)^2 + (1.14)^2}}$$

$$= 19.0$$

$$\text{UTA } \% = \frac{1}{2\pi}\int_{-\infty}^{19.0} e^{-0.5t^2}\,dt \cong 100\% \text{ (from probability tables)}$$

<u>The Exceeding AV and Trip Avoidance Probabilities are met.</u>
<u>Trip Setpoints are Acceptable</u>. (See Figure 2)

<u>NOTE</u>: If exceeding AV avoidance probability criterion is not met, the AV is moved in the conservative direction until it is met. (This adds more conservatism in the NTS value.)

If the undesirable trip avoidance probability criterion is not acceptable, system considerations are taken into account. Examples:

1. Add delay time.
2. Change transmitter range to obtain better accuracy.

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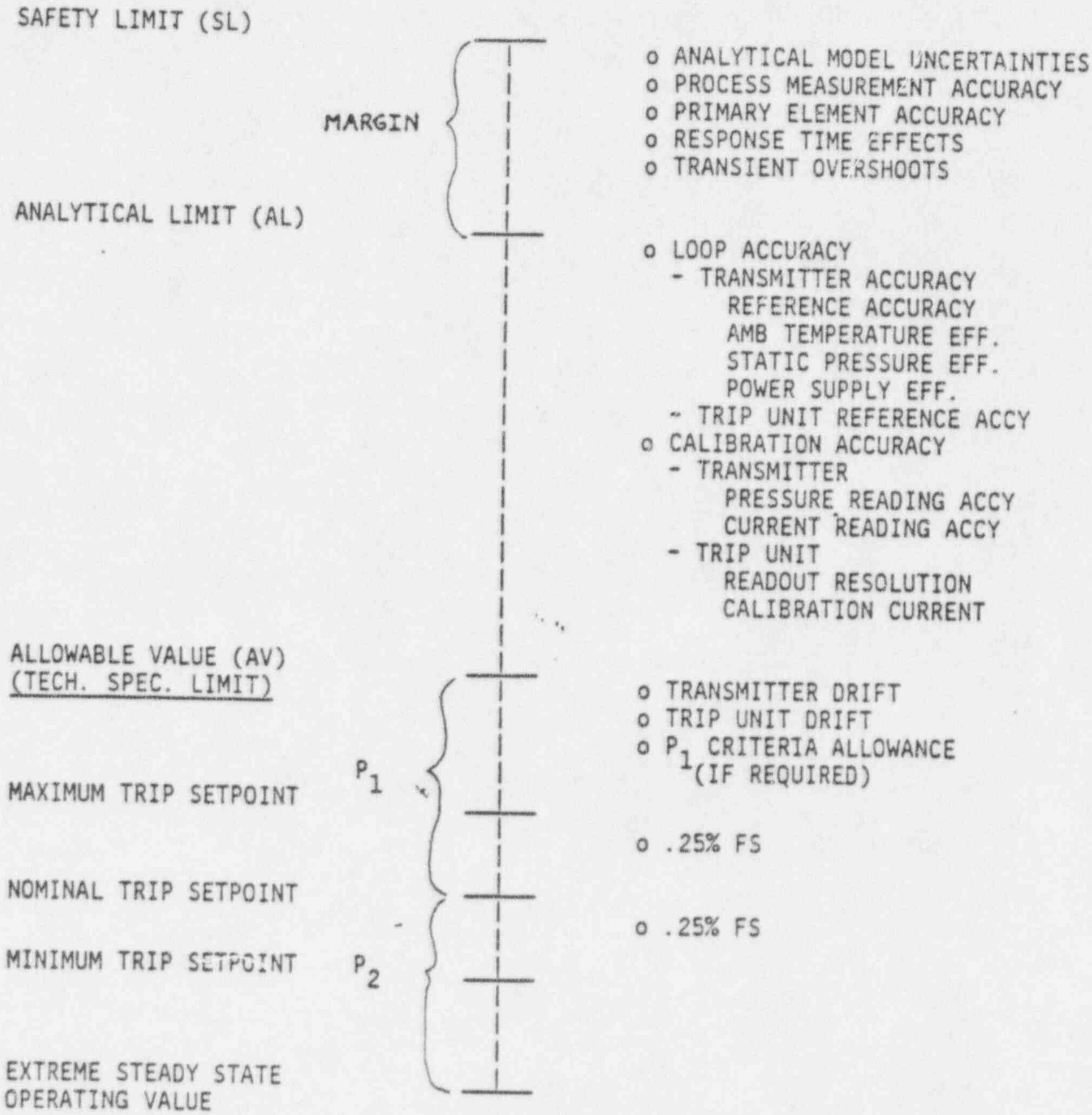

P_1 = EXCEEDING AV AVOIDANCE PROBABILITY

P_2 = UNDESIRABLE TRIP AVOIDANCE PROBABILITY

FIGURE 1. SETPOINT METHODOLOGY SAFETY LIMIT AND SETPOINT RELATIONSHIPS

JUN 0 7 1984