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WCAP 10650

FRACTURE MECHANICS

EVALUATION OF INSERVICE INSPECTION INDICATION

INDIAN POINT UNIT 2 REACTOR VESSEL

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BY

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SECTION 1

INTRODUCTION

A fracture analysis per ASME Section XI has been carried out to investigate the acceptability of an indication discovered during the inservice inspection of the Indian Point Unit 2 reactor vessel. The indication was found in the longitudinal weld in the lower shell region of the vessel near the outside surface at an azimuthal angle of 345° to the cardinal axis. The analyses presented in this report are based on the information presently available for the indication, and have been structured to be applicable to it as presently characterized, or for any smaller indication in the same location, near the outside surface of the vessel.

Three types of calculation are necessary for the evaluation of an indication per the requirements of Section XI, article IWB 3600:

1. Fatigue crack growth
2. Critical Flaw depth-normal conditions
3. Critical Flaw depth-faulted conditions

Fatigue crack growth evaluation must be carried out on the indication as characterized and then the indication after growth must be assessed relative to the critical flaw depths calculated. The criteria given in Section XI are clearly specified, and two alternative approaches are available, either a margin based on flaw depth, or a margin based on applied stress intensity factor. The second option has been adopted here, and the required margins are:

$$K_I < \frac{K_{Ia}}{\sqrt{10}} \quad \text{for normal, upset and test conditions}$$

$$K_I < \frac{K_{Ic}}{\sqrt{2}} \quad \text{for emergency and faulted conditions}$$

The fatigue and fracture toughness properties used in the analyses were taken from the reference properties provided in Appendix A of Section XI. The irradiation damage accumulated in this region of the vessel is a function of the integrated neutron fluence, which was obtained as described in Section 3.

The indication was also evaluated relative to the primary stress limits of Section III, NB 3000, as required by Section XI to demonstrate its acceptability.

SECTION 2

INSPECTION RESULTS

During the interval 1, period 3 inservice examination of the Indian Point Unit 2 reactor vessel, a ultrasonic reflector was detected in the lower shell longitudinal weld at 345° vessel azimuth. The reflector was initially discovered via 45° and 60° circumferential shear wave scanning of the adjacent base material on the lower shell side of the intermediate-to-lower shell circumferential weld. Indications of recordable amplitude were apparent in the clockwise and counter-clockwise scan directions, approximately three inches below the centerline of the circumferential weld seam. These results were confirmed during subsequent 45° and 60° examinations of the lower shell longitudinal weld at 345° vessel azimuth.

Initially, there was some speculation that multiple reflectors might be involved, however subsequent investigations have confirmed one reflector, located near 345° vessel azimuth, oriented axially with respect to the vessel, extending downword for a length of 1.96 inches from a point approximately 2.25 inches below the centerline of the intermediate-to-lower shell weld. Raw data predict a 2.03 inch through-wall dimension. Beam spread correction reduces that size to 1.2 inches. Peak amplitude sweep locations for the indications from this reflector place it at or very near the OD surface (within .25").

Recognizing that ASME XI 50% DAC sizing techniques tend to exaggerate through-wall dimensions of small reflectors at or near the opposite surface; in some cases as much as 10 to 1, a program for further investigation of this reflector was requested by Consolidated Edison Co. The program which was implemented included 1) application of a 45° pitch-catch transducer arrangement to determine the extent of "shadowing" by the reflector and 2) a delta transducer arrangement to more accurately establish the depth or through-wall dimension of the indication.

Results from these additional investigations suggest the reflector through-wall dimension is significantly smaller than the 2.03 inches predicted by ASME XI 50% DAC sizing techniques. The delta technique measurements predict a maximum depth of 0.3 inches. Additional confirmatory studies are planned to further demonstrate the efficacy of the delta transducer arrangement for reflector depth measurements, but for this analysis the maximum corrected depth was used.

SECTION 3

FRACTURE TOUGHNESS DETERMINATION

As described in the previous section, the indication is located in the vertical weld seam of the lower shell, oriented axially, and centered about 2 inches below the centerline of the intermediate to lower shell weld. The location of the indication is shown in Figure 3-1, where the reactor vessel weld locations as well as their designation numbers are also shown. The indication is in weld seam 3-042A shown in this figure, located at an angle of 345° from the cardinal axis.

The properties of the weld seam of interest were not obtained when the vessel was constructed, since no characterization was required. It is known that this weld was made with RACO3 weld wire (heat number W5214) and Linde 1092 flux, with a Nickel 200 wire addition. A review was made of welds made with this heat of weld wire combined with the Nickel wire and 1092 flux, and it was found that several characterizations have been made. A summary of the chemistry results is given in Table 3-1. Based on these nine chemistry analyses, it was concluded that the average values of copper, nickel and phosphorous should be used, and these are provided in Table 3-2. Since the actual RT_{NDT} of this weld was not measured, a generic value of -56°F was used [1]. For reference the chemistry and RT_{NDT} values for the two adjacent base metal plates have also been included in Table 3-2 [2].

Fluence was determined for both present life and end-of-life in the cross-section of the vessel where the indication was located, at the 345° degree azimuthal angle. Results are presented in Figure 3-2 for the current exposure and in Figure 3-3 for the projected end-of-life (32 EFPY) exposure of the vessel.

Exposure informatiom is supplied both in terms of neutron fluence (E > 1.0 MeV) and of fluence equivalent dPa. Here "fluence equivalent dPa" refers to the use of the shape of the energy dependent displacements per iron atom damage function through the vessel wall normalized to the fast neutron

(E > 1.0 MeV) fluence at the vessel inner radius. The use of the fluence equivalent dpa permits the application of energy dependent damage gradients in conjunction with available trend curves to assess vessel integrity.

Test neutron (E > 1.0 MeV) fluence profiles were taken from Reference [3]. Displacements per atom calculations were carried out using the calculated neutron data presented in ASTM Standard Practice E 693-79, "Characterizing Neutron Exposures in Ferrite Steels in Terms of Displacements Per Atom".

Although Indian Point Unit 2 has recently implemented low leakage fuel management, the flux reduction at the 345° azimuthal location resulting from the new fuel management scheme is not large. Therefore, for conservatism the fluence projections provided in Figures 3-2 and 3-3 were based on the non low leakage neutron flux profiles.

The effect of the fluence on the fracture toughness of the vessel is not severe, because of the location of the indication near the outside of the vessel. Therefore, the end-of-life values were used in determining the final value of RT_{NDT}. The fluence values are listed in Table 3-3, for the region at the tip of the assumed flow, a location of 18.2 cm from the inside surface.

The irradiation damage calculations recomended by the NRC [1] were used to determine the end-of-life RT_{NDT} value for the weld material. RT_{NDT} at end-of-life fluence is determined as the lower of the results given by equations (3-1) and (3-2).

Equation 3-1:

$$RT_{NDT} = I + M + [-10 + 470\ Cu + 350\ Cu\ Ni]\ f^{0.270}$$

Equation 3-2:

$$RT_{NDT} = I + M + 283\ f^{0.194}$$

"I" means the initial reference temperature of the unirradiated material measured as defined in the ASME Code, NB-2331. If a measured value is not available, the following generic mean values must be used: 0°F for welds made with Linde 80 flux, and -56°F for welds made with Linde 0091, 1092 and 124 and ARCOS B-5 weld fluxes.

"M" means the margin to be added. In Equation 1, M=48°F if a measured value of I was used, and M=59°F if the generic mean value of I was used. In Equation 2, M=0°F if a measured value of I was used, and M=34°F if the generic mean value of I was used.

"Cu" and "Ni" mean the weight percent of copper and nickel in the material. The source of these values must be reported. The relationship of the material on which any measurements were made to the actual material in the pressure vessel must be described.

"f" means the maximum neutron fluence, in units of $10^{19} n/cm^2$ (E greater than or equal to 1MeV), at the location of interest.

The calculated value of RT_{NDT} at end of life for the material where the indication is located is shown in Table 3-3. Using this final value of RT_{NDT}, and the temperature, the toughness of the vessel can be determined from the reference curves of the ASME Code, which are shown in Figure 3-4.

TABLE 3-1

SUMMARY OF AVAILABLE WELD METAL CHEMISTRY MEASUREMENTS

Weld Source	Cu (Wt.%)	Ni (Wt.%)	P (Wt.%)
Indian Point Unit 3	.15	1.02	.019
Indian Point Unit 3	.16	1.06	.017
Indian Point Unit 3	.15	1.11	.018
Indian Point Unit 3	.15	1.09	.018
Millstone Unit 1	.19	.98	-
H. B. Robinson Unit 2	.154	.99	.012
H. B. Robinson Unit 2	.163	.90	.011
H. B. Robinson Unit 2	.152	1.08	.014
H. B. Robinson Unit 2	.166	1.00	.012

TABLE 3-2

PROPERTIES OF INDIAN POINT UNIT 2 LOWER SHELL

Property	Weld 3-042A	Plate B2003-1	Plate B2003-2
Copper, wt.%	0.16	0.20	0.19
Nickel, wt.%	1.03	0.66	0.48
Phosphorous, wt.%	0.015	0.011	0.010
Initial RT_{NDT}	-56°F	20°F	-20°F

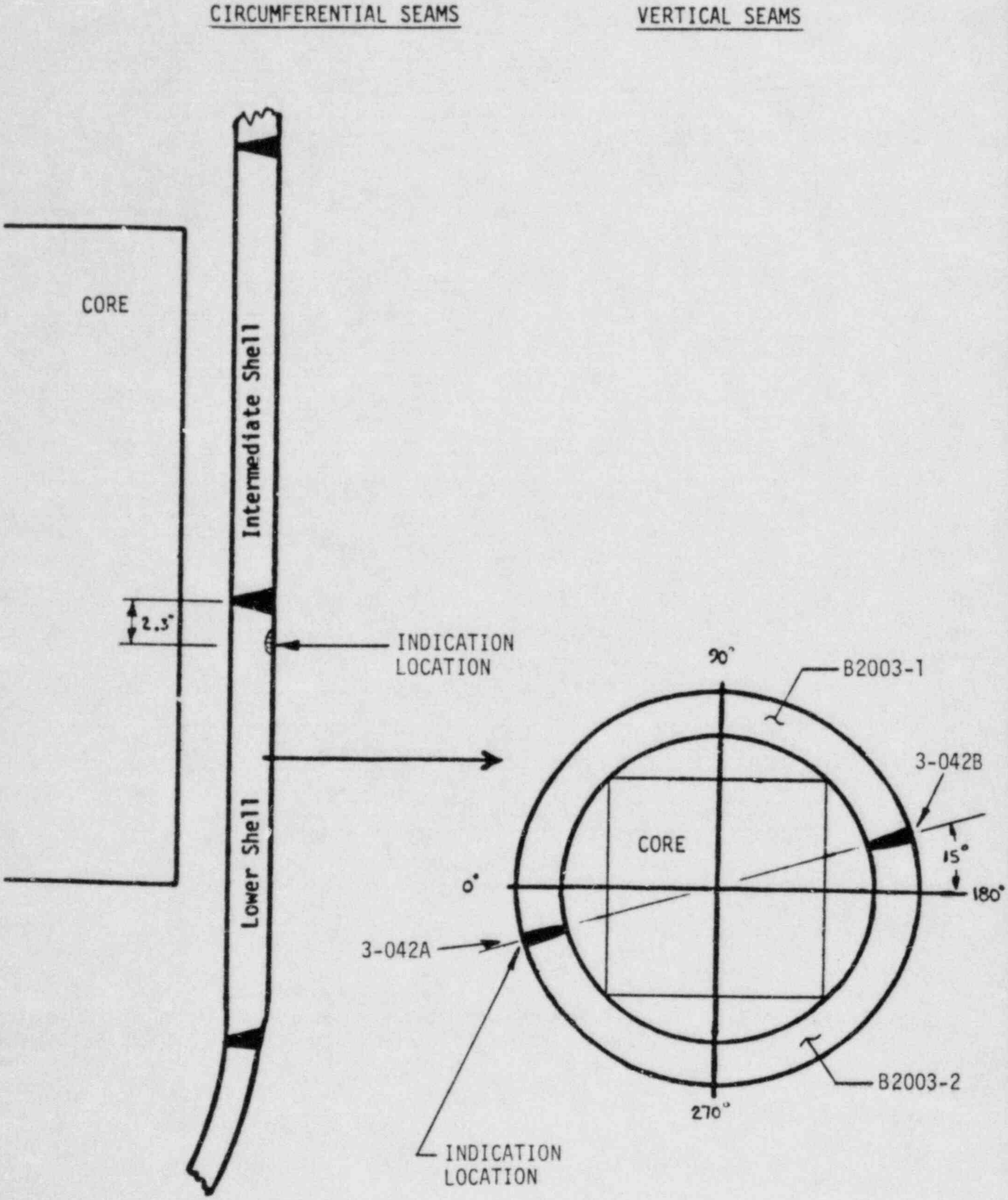
TABLE 3-3

END OF LIFE FLUENCE AND RT_{NDT} AT INDICATION

Fluence (E>1 MeV)	7.5×10^{17} n/cm^2
Fluence (dPa)	1.7×10^{18} n/cm^2
RT_{NDT} (weld)	79°F

FIGURE 3-1

IDENTIFICATION AND LOCATION OF BELTLINE REGION MATERIAL FOR THE INDIAN POINT UNIT NO. 2 REACTOR VESSEL



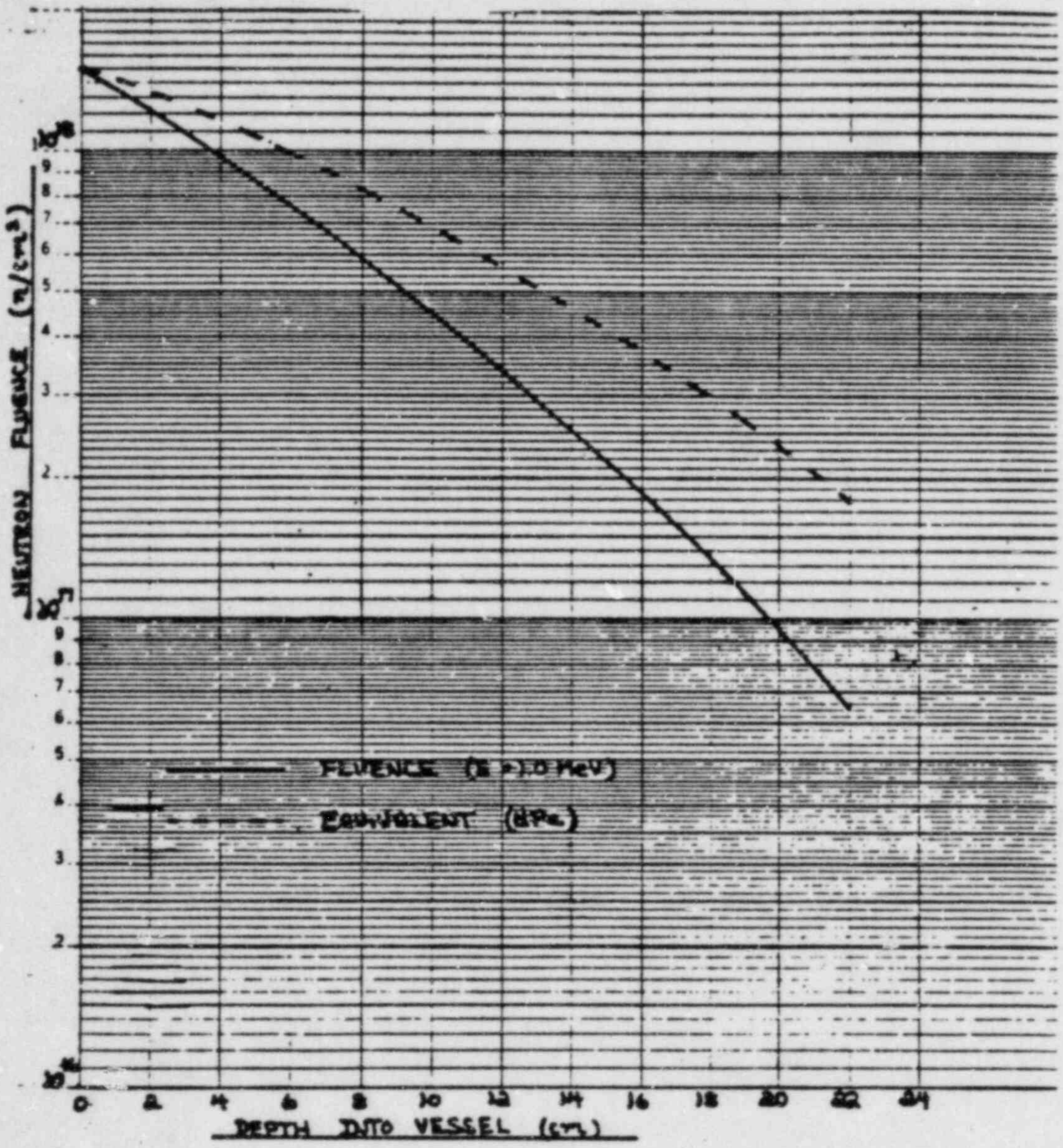


Figure 3-2 Current Fast Neutron Exposure as a Function of Depth within the Indian Point Unit 2 Pressure Vessel - 345° Azimuthal Angle - 5.33 EFPY

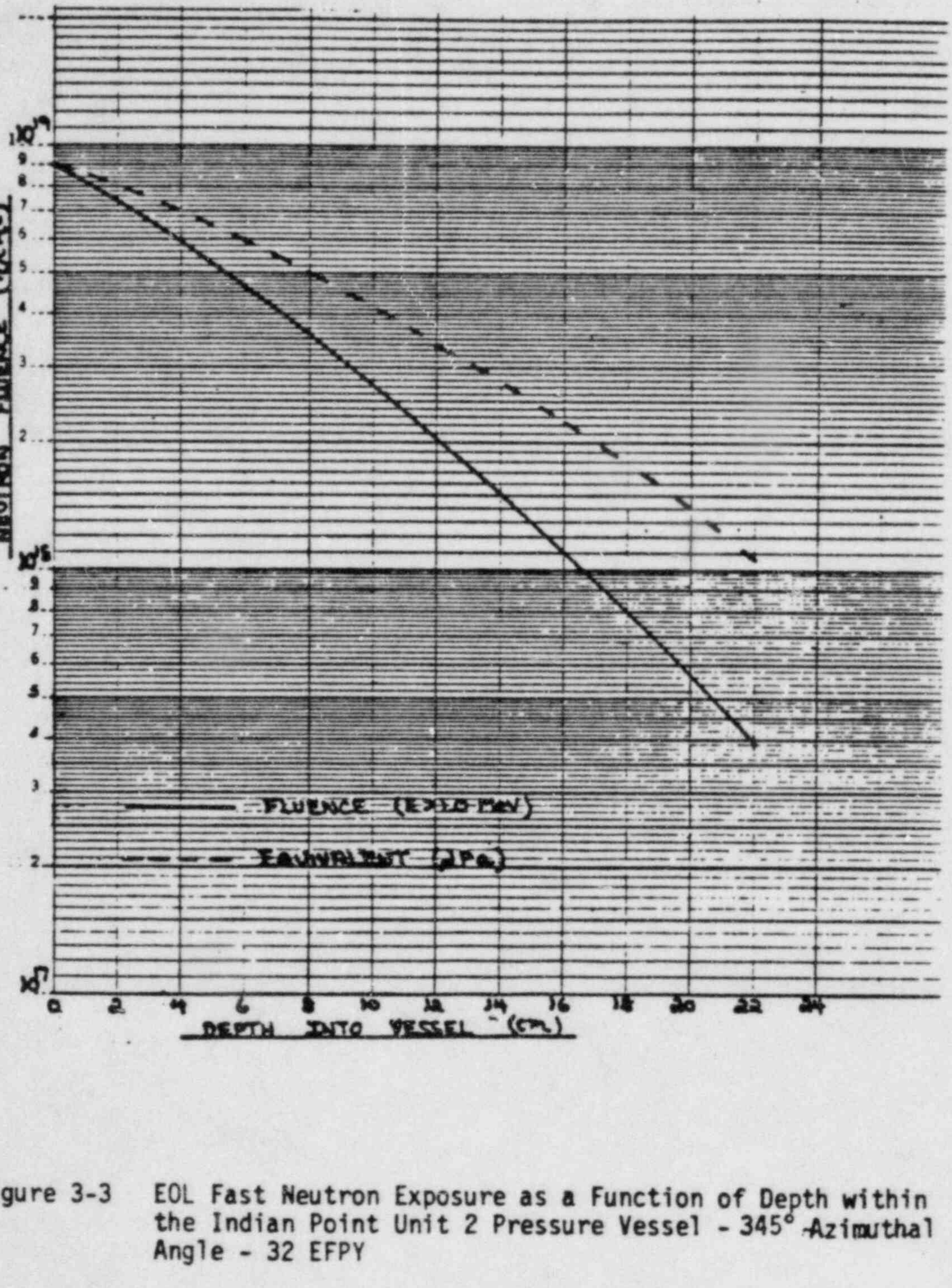


Figure 3-3 EOL Fast Neutron Exposure as a Function of Depth within the Indian Point Unit 2 Pressure Vessel - 345° Azimuthal Angle - 32 EFPY

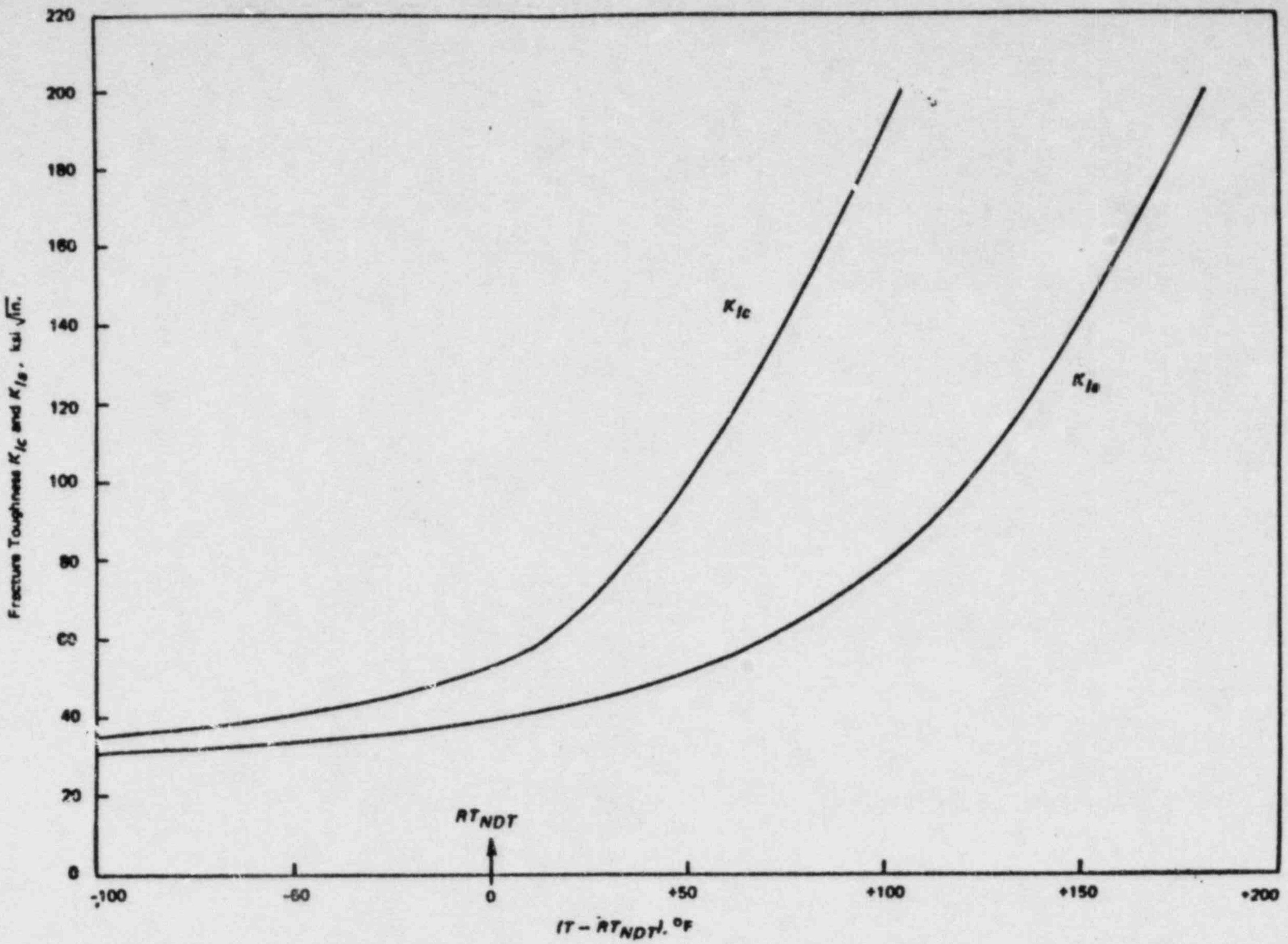


Figure 3-4 - K_{Ia} and K_{Ic} from Section XI, ASME Code

SECTION 4

FATIGUE CRACK GROWTH CONSIDERATIONS

The growth of the indication of interest was assessed based on the assumption that it was a flaw, located at the outside surface of the vessel. The analyses reported here utilized the reactor coolant system transients detailed in the original equipment specification of the reactor vessel as well as additional transients which have been developed since that time. These transients are listed in Table 4-1. All the transients are described in detail in Appendix A. Since the assumed flaw was not exposed to primary coolant, the fatigue crack growth was calculated using the air environment crack growth reference curves of Section XI Appendix A, as shown in Figure 4-1. The calculated crack growth was insignificant, as will be shown in this section.

4.1 STRESS ANALYSIS

Thermal and stress analyses were performed for the beltline region of the reactor vessel during 25 Level A, Level B and Test Condition Transients. The temperature analysis was done for a total of 221 time points. The thermal and pressure stress analyses were also done for each of these time points.

The thermal and stress analyses were performed with the WECAN finite element program[4], using a two-dimensional finite element model. To assure conservative results, the insulating effect of the cladding was not included.

4.2 DESCRIPTION OF LEVEL A, LEVEL B AND TEST CONDITION TRANSIENTS

The design transients used in the evaluation of the reactor vessel beltline are based on conservative estimates of the magnitude and frequency of the temperature and pressure variations resulting from various operating conditions in the plant. These are representative of operating conditions which are considered to occur during plant operation and are sufficiently severe or frequent to be included in an analysis of cyclic stress conditions. Further, these are regarded as a conservative representation of transients which, when used as a basis for component fatigue evaluation, provide confidence that the component is appropriate for its application over the design life of the plant.

Transients were picked to give an upper bound to the temperature limits. For example, the heatup and cooldown technical specification is for 100°F/hr, whereas the actual rate is closer to 50°F/hr. As another example, loading and unloading at 15-100% power is analyzed for 13,200 cycles. It actually would occur about 500 times in a typical plant. In general the transients were digitized for input to the computer, the representations where simplified, were chosen such that the resultant stresses would be higher than for the actual transient, thus, giving conservative results.

The operational transients are broken down into the following categories:

1.) <u>Level A Service Limits</u>

Level A Service Limits are applicable to any transient in the course of system startup,operation in the design power range, hot standby and system shutdown, other than Level B Service Limits or Test Conditions.

2.) <u>Level B Service Limits</u>

Level B Service Limits are applicable to transients which deviate from those controlled by Level A Service Limits and which are anticipated to occur often enough that the design should include a capability to withstand the limits without operational impairment.

3.) <u>Test Conditions</u>

Test conditions are those pressure overload tests including hydrostatic tests and leak tests which occur in the course of testing the system both before and after initial startup.

The total number of cycles for each transient exclusive of the preoperational test cycles was distributed over the 40-year operating life of the plant.

4.3 VESSEL GEOMETRY AND FINITE ELEMENT MODEL

Figure 4-2 shows the cross-sectional view of the four-loop reactor vessel. The vessel is axisymmetric and supported at the nozzles in such a fashion as to allow radial expansion and contraction. The basic geometric data required for the thermal and stress analyses input are shown in Figure 4-3.

The model for this analysis uses two-dimensional four node quadrilateral isoparametric elements. The symmetry of the region to be analyzed permitted a 2D axisymmetric finite element idealization. The outline of the idealized structure is shown in Figure 4-4.

The model consists of a mesh of rectangular elements with 9 elements through the thickness of the vessel wall and one hundred thirty-seven elements in height. The selection of the finite element mesh is based on considerable practical experience with thermal and pressure transients. It is selected to efficiently provide acceptable engineering accuracy without being excessively large. In order to model the stress profiles through the thickness, the nine rows of elements are distributed such that the first four rows starting at the inner surface, are half the thickness of the remaining five rows. This distribution was chosen because the highest stresses and greatest changes in stress generally occur at the inner surface, where the temperature and pressure changes are applied and where a finer mesh will provide a more accurate calculation. The model consists of 1380 nodes and 1233 elements, resulting in 2760 gross total degrees of freedom.

The stainless steel reactor vessel cladding was not included in the model. This assumption is considered to add conservatism to the analyses primarily because the insulating effect of the cladding is not included. This means that the heat transfer is higher at the vessel inner surface which causes the stresses on the inside and outside surfaces to be conservatively higher during thermal transients.

4.4 BOUNDARY CONDITIONS FOR STRESS ANALYSIS

The reactor vessel is subjected to axisymmetric thermal and pressure loadings so that the lower end of the model representing the center of the lower head is restrained from moving laterally ($U_x = 0$), and the upper end of the model representing the shell at the nozzle region is restrained from moving vertically ($U_y = 0$) as shown in Figure 4-5. Pressure is applied to the internal surface and no mechanical boundary conditions are applied to the external surface.

Thermal boundary conditions are shown in Figure 4-6. The outside surface is assumed to be completely insulated. When the inside surface of the vessel is subjected to thermal transients, the primary mechanism of heat transfer is forced convection. The thermal properties of the metal are input as a linear function of the temperature. The heat transfer coefficient associated with forced convection is obtained by using the Dittus-Boelter formula.[5] The Reynolds number and Prandtl number are obtained based on the flow cross-section geometry, the flow rate and the temperature of the coolant.

The boundary conditions for the thermal analysis are summarized below:

a) Initial temperature is 557°F
b) External surface is insulated
c) Fluid temperatures associated with the transients are according to those shown in Appendix A.
d) Heat transfer coefficient associated with the transients is 7000 Btu/hr-ft^2•F.

4.5 STRESS ANALYSIS RESULTS

Obtained in the stress analysis are the sum of the stresses due to temperature effects and the stresses due to internal pressure. The WECAN computer program[4] was used to perform the stress analysis, using the same finite element model that was used to obtain the temperatures. The temperature and pressure input data for the stress analysis were given in Appendix A. The pressure induced stress analysis was performed for each time step and is combined with the temperature induced stresses.

The results of the stress analysis provide the stress distribution history for the entire region of the finite element model. Maximum and minimum, inside and outside combined thermal and pressure hoop stresses for the beltline are given in Table 4-2 for the twenty-five transients and the steady-state condition. Note that in some cases, the steady-state value is used in the calculation. This occurs when the steady state condition is either the maximum or minimum condition in the transient, for example when it occurs from steady state conditions.

Temperature and hoop stress contour plots are shown in Figures 4-7 through 4-16. These figures include hoop stresses for pressure only at p = 3105 psig for the cold hydrostatic pressure test, the maximum combined hoop stress for Heatup and Cooldown as an example of a slow thermal transient, and the maximum combined stress for Excessive Feedwater Flow as an example of fast thermal transient.

4.6 FATIGUE CRACK GROWTH RESULTS

The fatigue crack growth analysis was carried out using the guidelines provided in Appendix A, Section Xi of the ASME Code. A semi-elliptic surface flaw was assumed to exist at the outside surface of the reactor vessel where the actual indication was observed. The assumed flaw was oriented axially, and was assumed to be exposed to an air environment. Crack growth was calculated using the stress intensity factor expression of Section XI Appendix A, as well as the reference fatigue crack growth law for air environments contained therein. The transients listed in Table 4-1 were used in the analysis, and the stresses used as input are listed in Table 4-2. Results of the faitgue crack growth analysis showed that a flaw initially 1.45 inches deep would grow to 1.454 inches in ten years, to 1.455 inches in twenty years, and to 1.457 inches in 30 years, the remaining design lifetime. Therefore, the crack growth is insignificant.

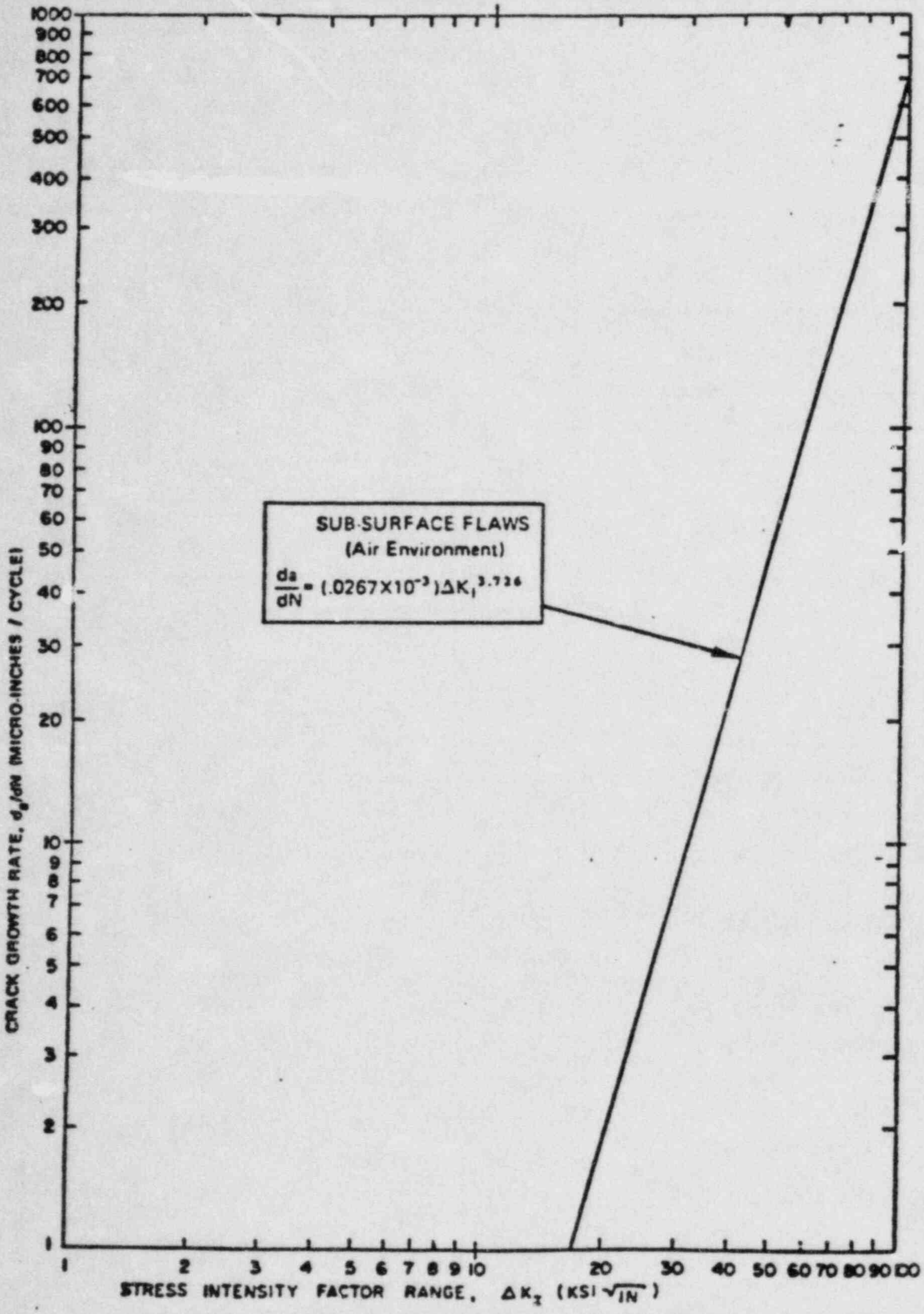
TRANSIENT	OCCURRENCE (CYCLES)	OPERATING CONDITION SERVICE LIMITS
Heatup and Cooldown	200	Level A
Loading and Unloading	13,200	Level A
Reduced Temp. Return to Power	2,000	Level A
Step Load Decrease and Increase	2,000	Level A
Large Step Load Decrease	200	Level A
Initial Steady State Fluct.	150,000	Level A
Random Steady State Fluct.	3,000,000	Level A
Feedwater Cycling	2,000	Level A
Loop Out of Service	80	Level A
Loss of Load	80	Level B
Loss of Power	40	Level B
Partial Loss of Flow	80	Level B
Reactor Trip		
with No Cooldown	230	Level B
with Cooldown No Safety Inj.	160	Level B
with Cooldown and Safety Inj.	10	Level B
Inadvert. Depressurization	20	Level B
Inadvert. Startup Inactive Loop	10	Level B
Inadvert. Safety Injection Actuation	60	Level B
Control Rod Drop	80	Level B
Excessive Feedwater Flow	30	Level B
Boron Concentration	26,400	Level A
Refueling	80	Level B
Turbine Roll	20	Test
Hot Hydrostatic	280	Test
Cold Hydrostatic Test	10	Test

TABLE 4-1 REACTOR DESIGN TRANSIENTS

TABLE 4-2

BELTLINE SURFACE HOOP STRESSES FOR LEVEL A, LEVEL B AND TEST TRANSIENTS

TRANSIENT	Maximum Outside Stress ksi	Corresponding Inside Stress ksi	Minimum Outside Stress ksi	Corresponding Inside Stress ksi
Heatup and Cooldown	16.22	26.10	8.70	-6.98
Loading and Unloading	22.35	23.36	21.99	26.89
Reduced Temp. Return to Power	23.01	23.49	19.97	29.88
Step Load Decrease and Increase	23.09	23.17	21.93	26.28
Large Step Load Decrease	23.50	21.91	22.06	24.98
Initial Steady State Fluct	22.55	26.17	22.17	23.42
Random Steady State Fluct	22.40	25.01	22.31	24.54
Feedwater Cycling	23.75	21.98	20.43	31.93
Loop Out of Service	21.61	23.5	21.22	27.47
Loss of Load	25.99	10.99	22.35	24.77
Loss of Power	26.52	19.26	25.27	27.15
Partial Loss of Flow	23.16	24.92	21.39	18.79
Reactor Trip				
with No Cooldown	22.34	24.77	20.86	18.87
with Cooldown No Safety Inj.	20.69	21.17	17.53	27.86
with Cooldown and Safety Inj.	20.5	20.77	16.50	42.28
Inadvert. Depressurization	-0.824	1.631	-7.045	46.20
Inadvert. Startup Inactive Loop	22.53	21.24	21.28	29.17
Inadvert. S.I. Actuation	23.48	25.59	22.05	19.43
Control Rod Drop	22.34	24.88	20.07	21.51
Excessive Feedwater Flow	15.37	21.8	11.35	62.88
Boron Concentration	22.65	25.10	22.35	24.77
Refueling	0.909	-32.11	-.926	32.66
Turbine Roll	14.38	22.04	7.437	42.30
Hot Hydrostatic	21.54	27.07	6.80	-2.35
Cold Hydrostatic Test	29.97	33.21	0.0	0.0



UPPER BOUND FATIGUE CRACK GROWTH DATA FOR REACTOR VESSEL STEELS

Figure 4-1 ASME Code Reference Fatigue Crack Growth Law, Air Environment

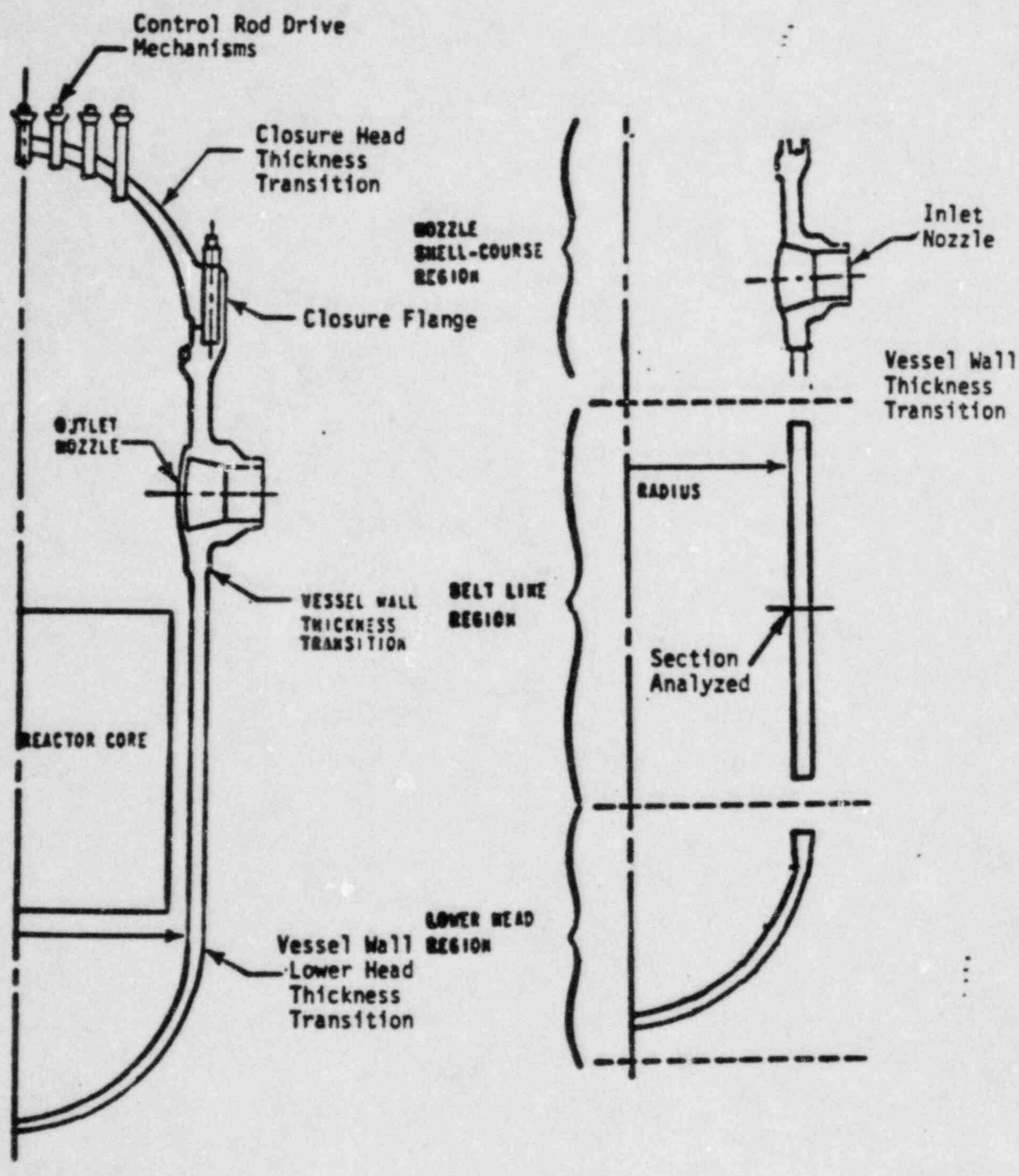


Figure 4-2 Pressurized Water Reactor Vessel Showing Beltline Region

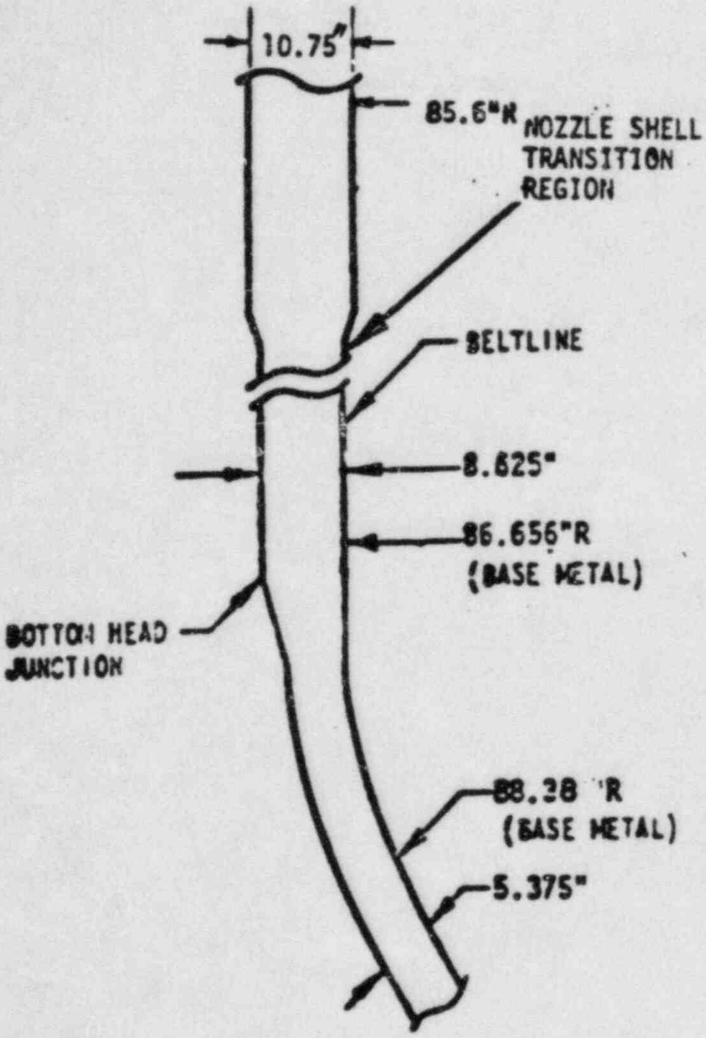


Figure 4-3 Nozzle Transition, Beltline and Lower Head Regions-Dimensions

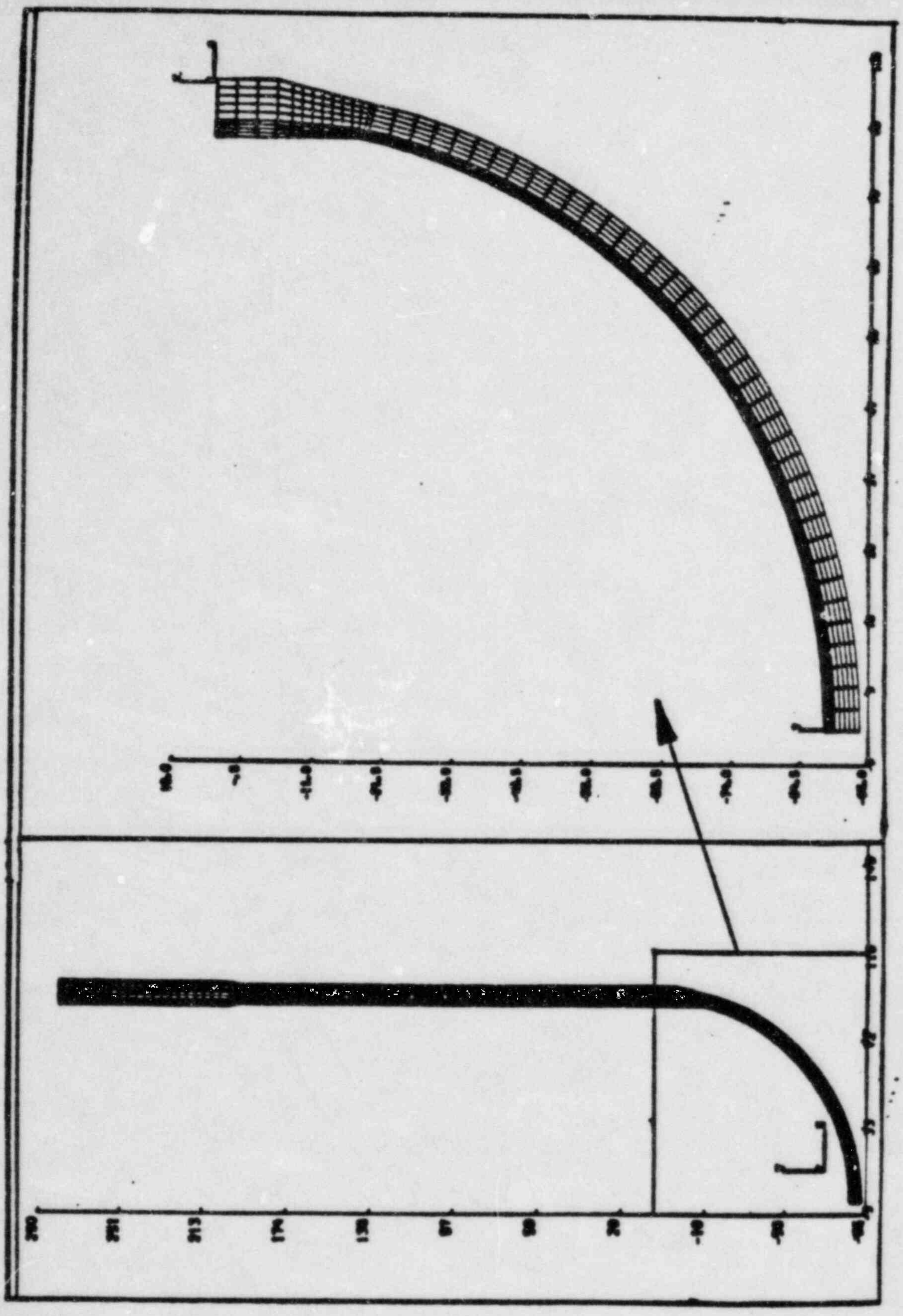


Figure 4-4 WECAN Finite Element Geometry Model

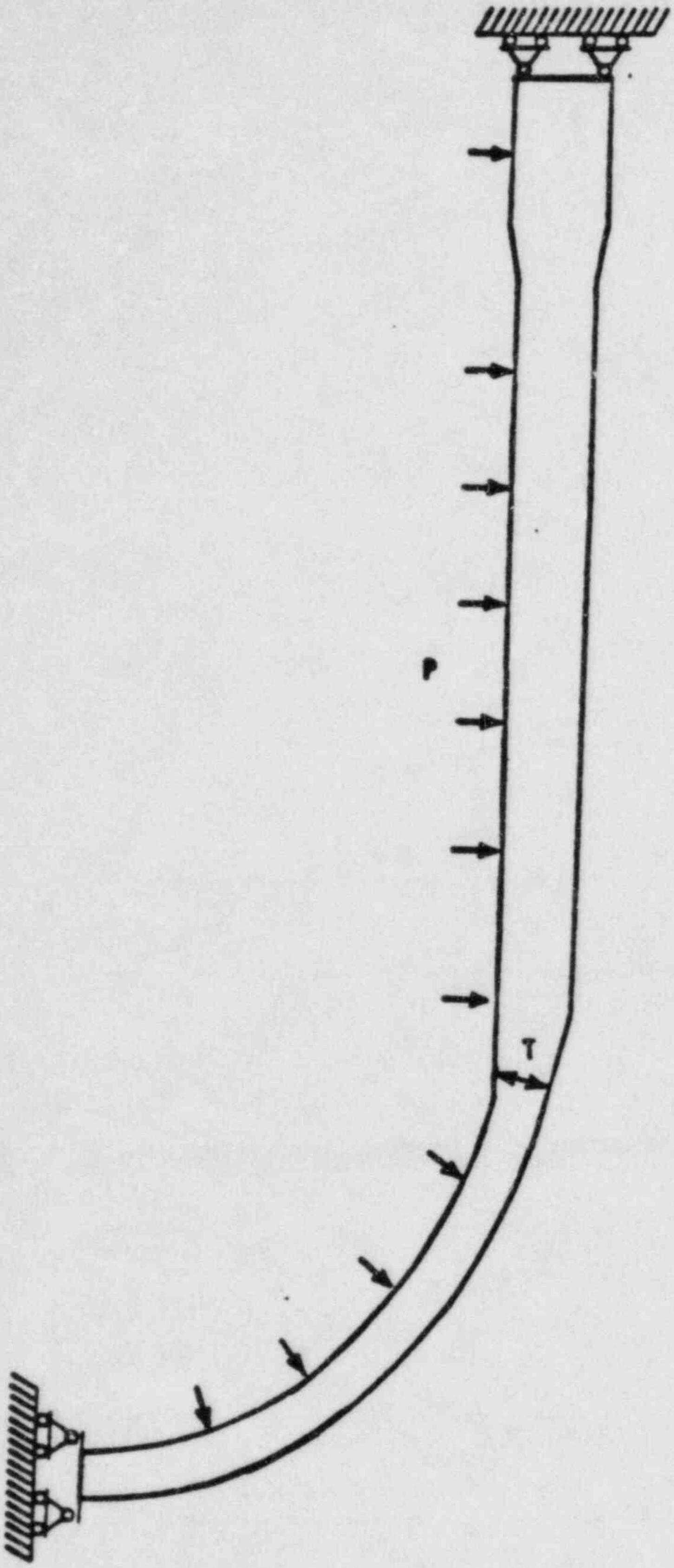


Figure 4-5 4 Loop Reactor Vessel Mechanical Boundary Conditions

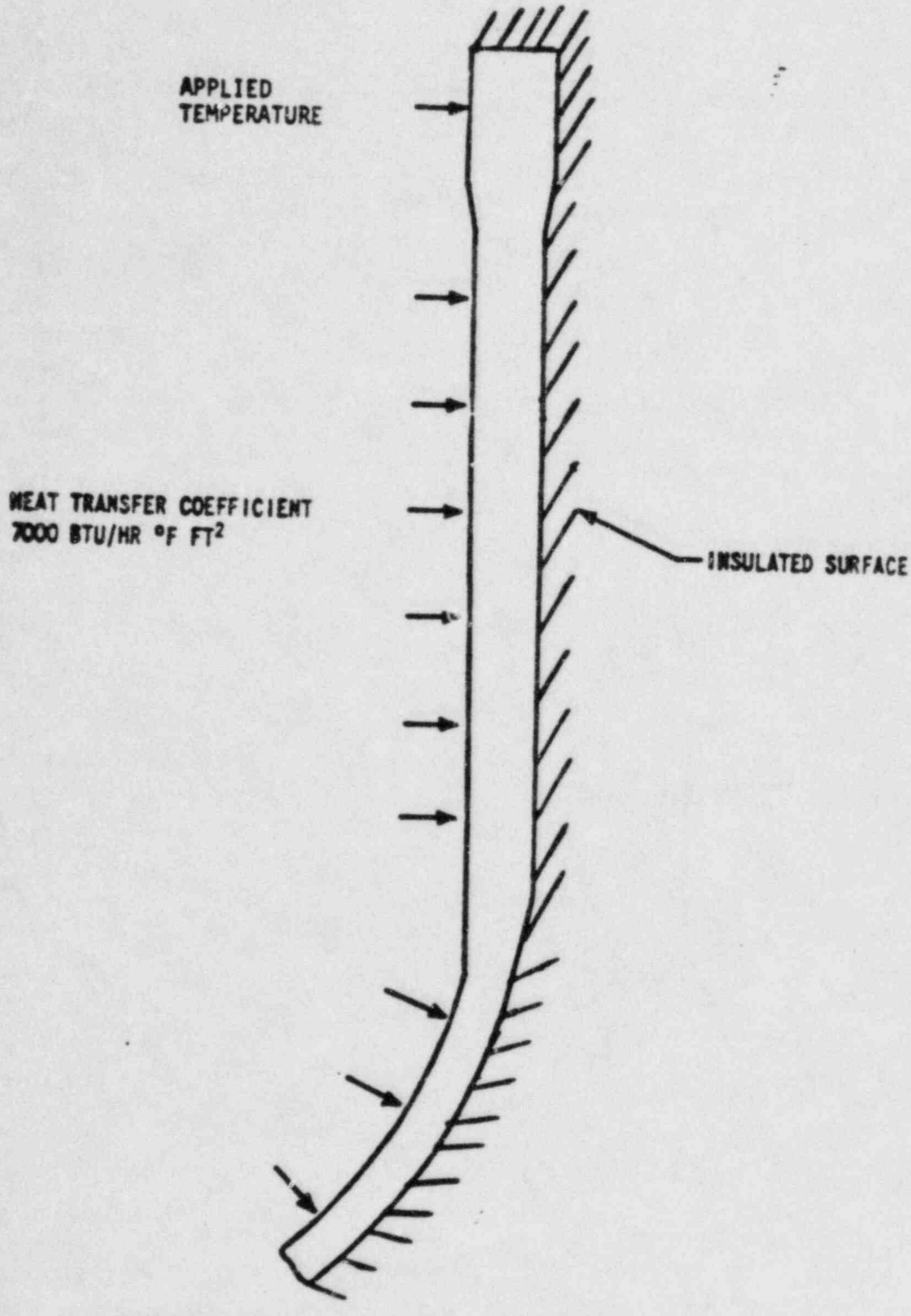
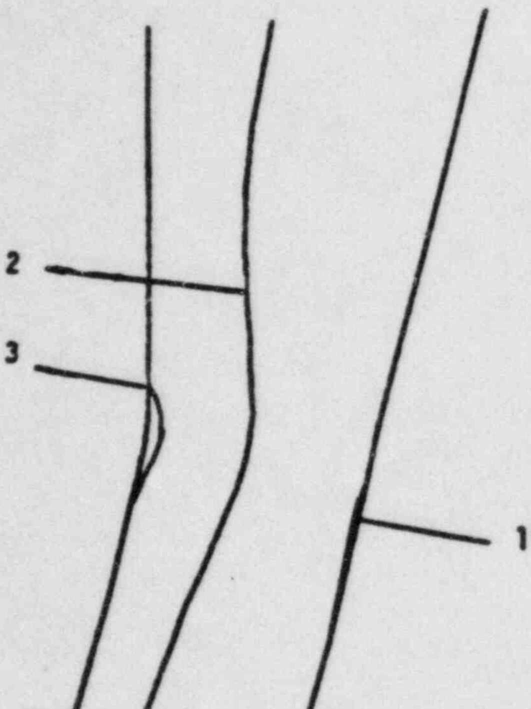


Figure 4-6 4-Loop Reactor Vessel Thermal Boundary Conditions

Contour Number	Stress (PSI)
1	21000
2	24000
3	27000
4	30000
5	33000

X = Max Stress = 33215 PSI



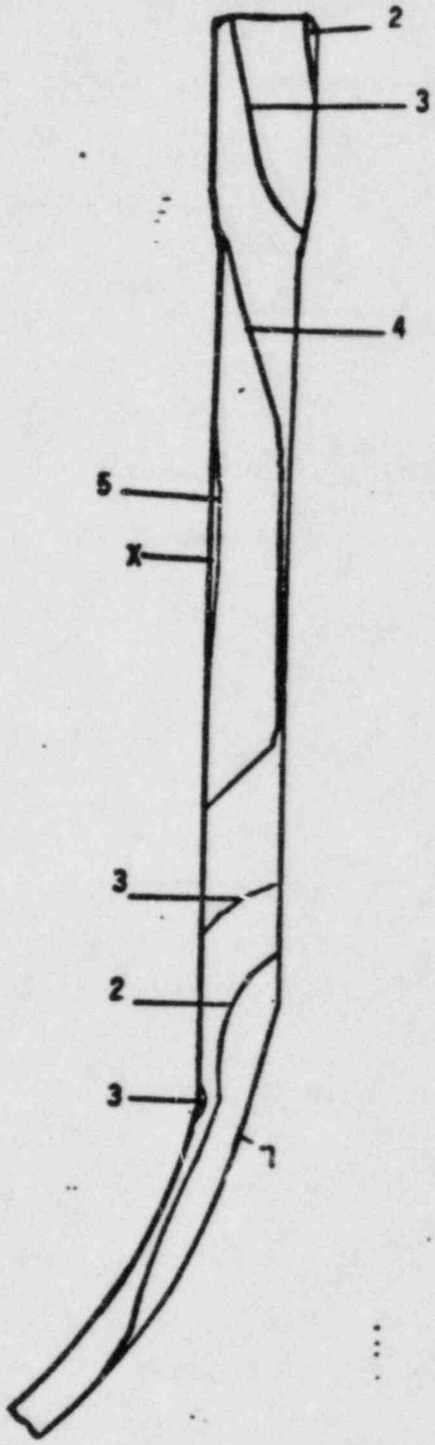


Figure 4-7 Hoop Stress Contours - Pressure Only (3105 PSI)

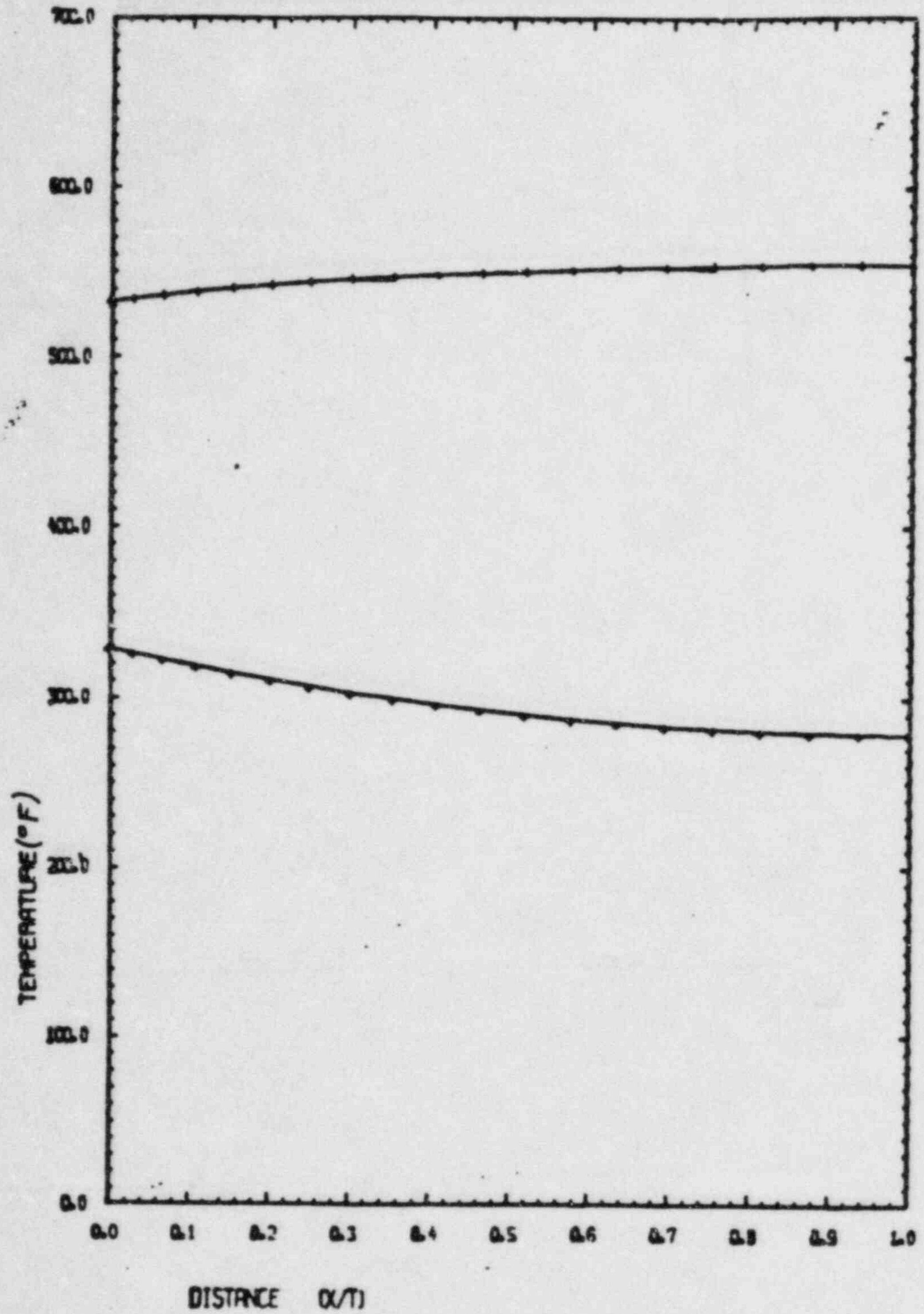


Figure 4-8 Beltline Heatup and Cooldown Temperature

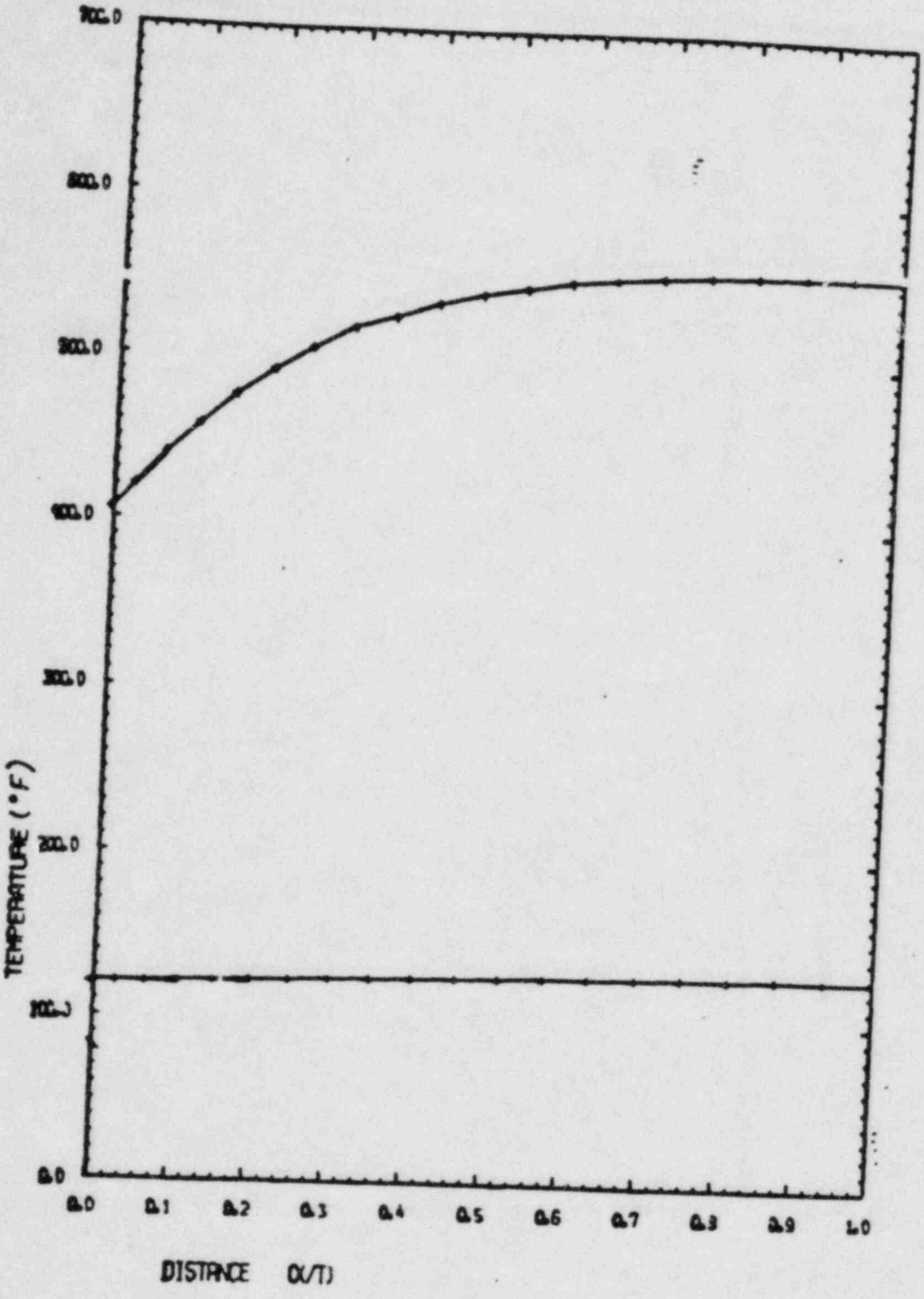


Figure 4-9 Beltline Inadvertant RCS Depressure Temperature

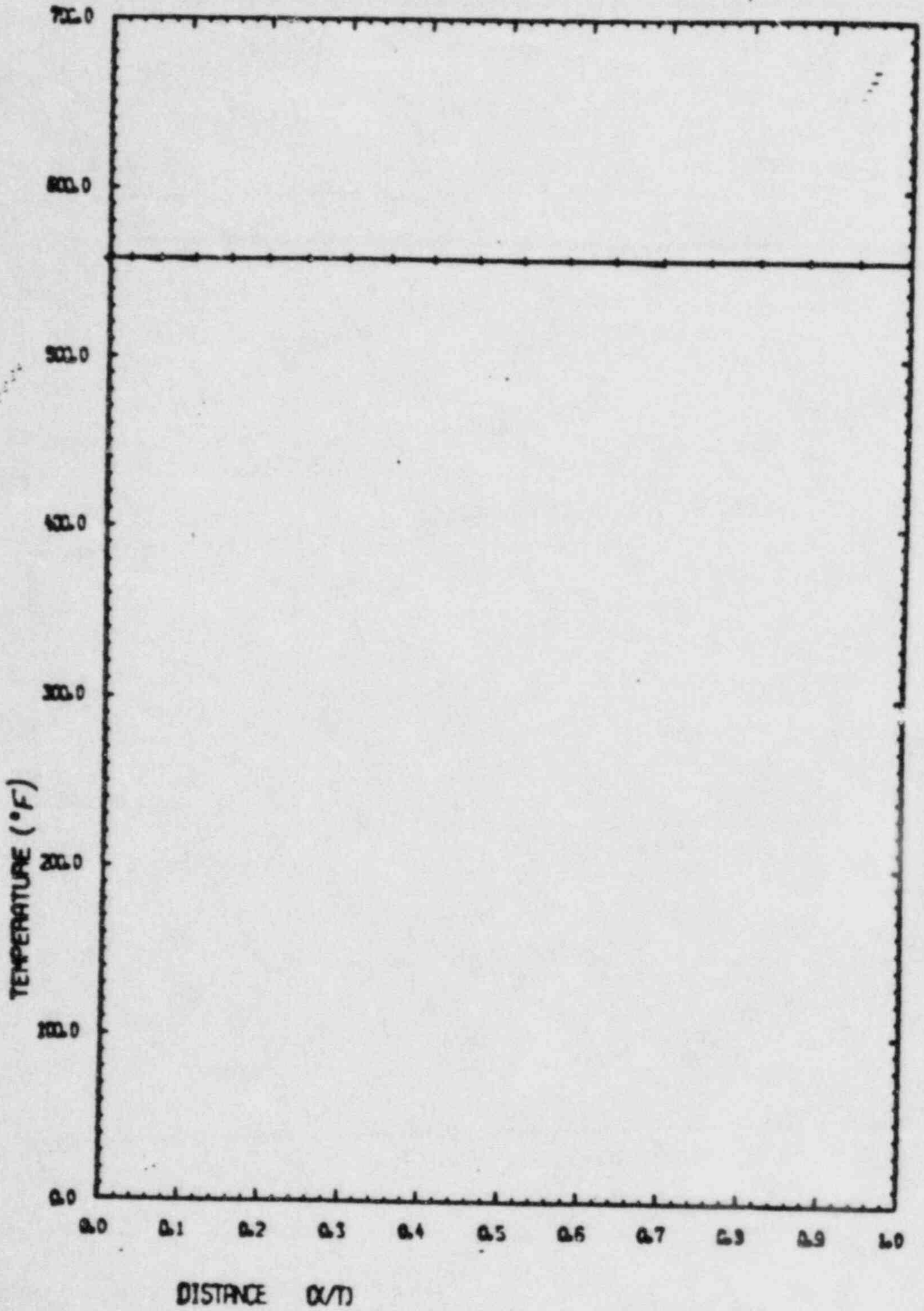


Figure 4-10 Beltline Boron Concentration Temperature

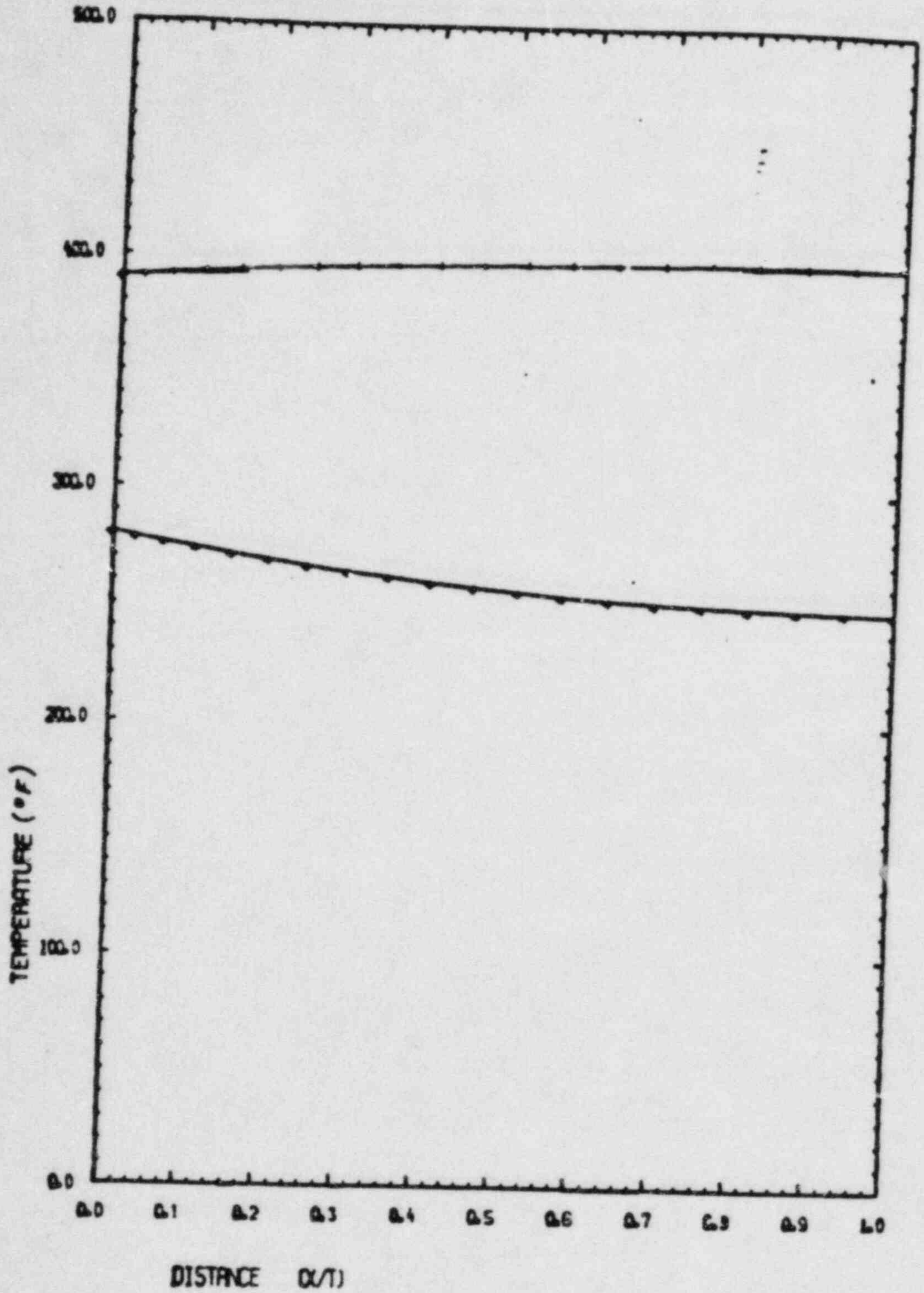
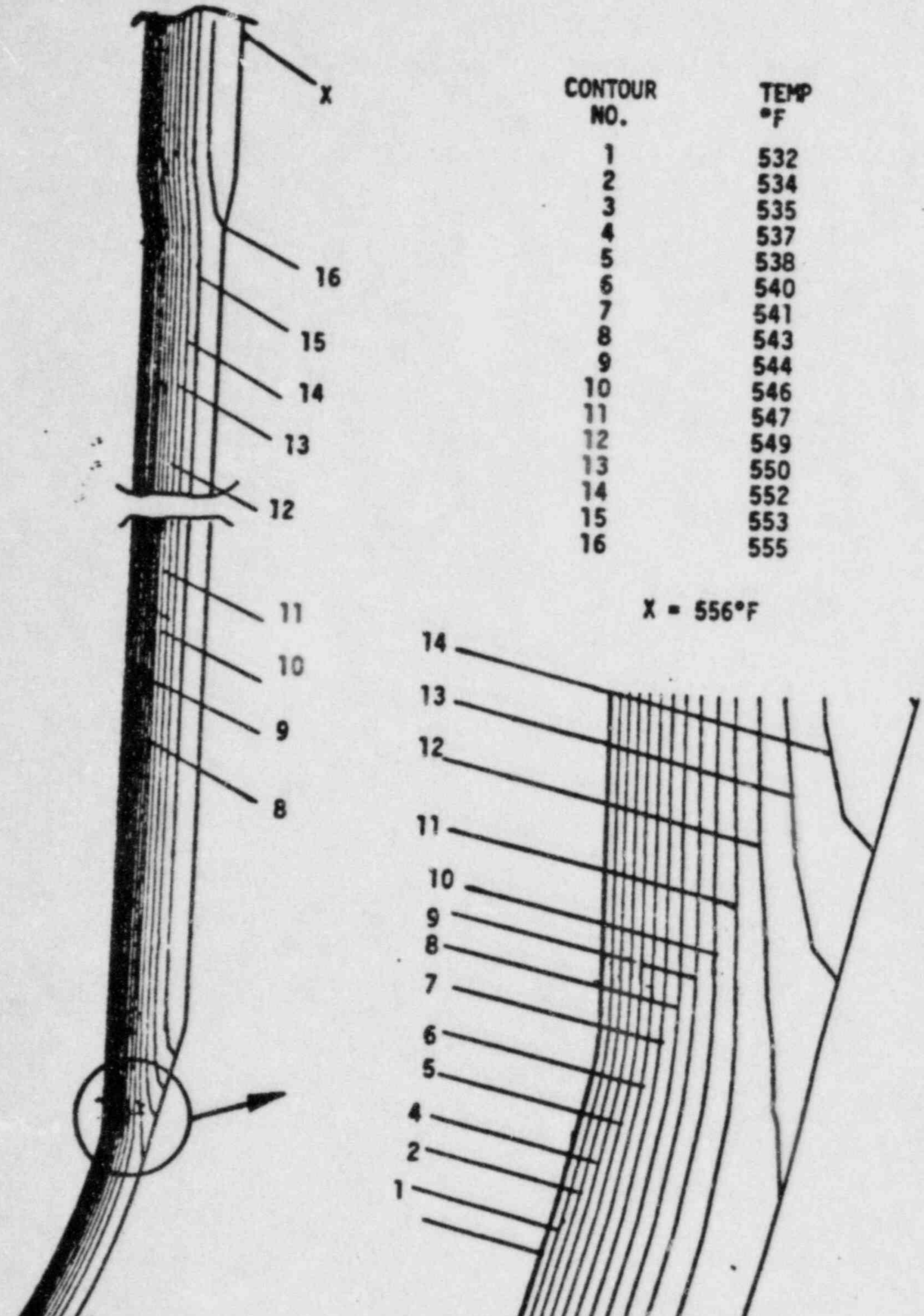


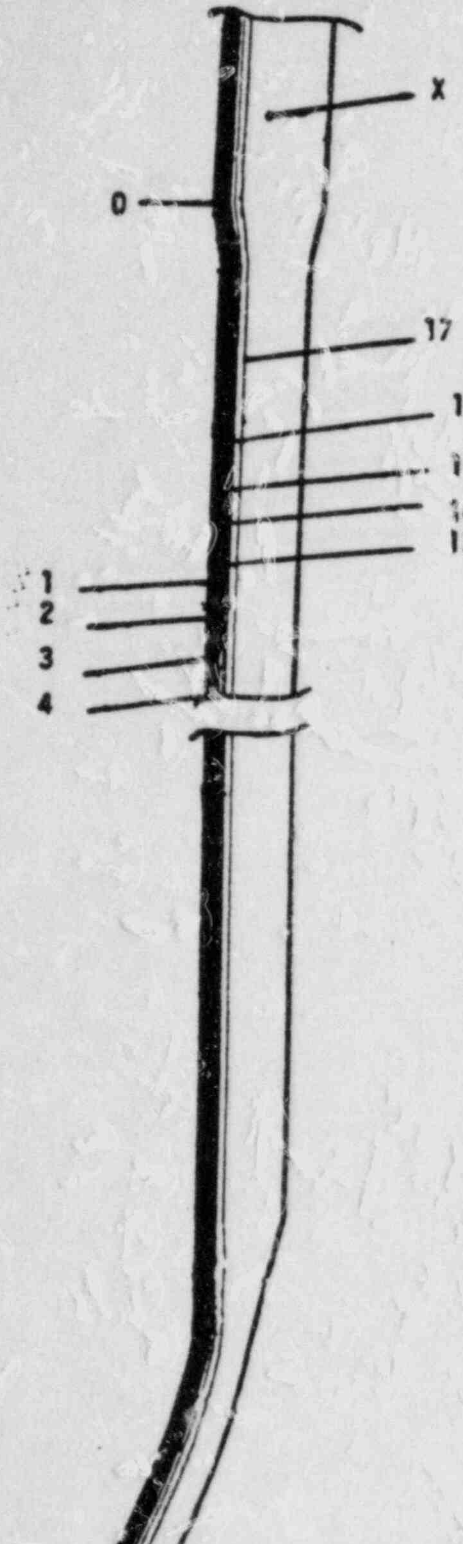
Figure 4-11 Beltline Hot Hydro Test Temperature



CONTOUR NO.	TEMP °F
1	532
2	534
3	535
4	537
5	538
6	540
7	541
8	543
9	544
10	546
11	547
12	549
13	550
14	552
15	553
16	555

X = 556°F

Figure 4-12 Cooldown Temperature Contours - Example of a Slow Thermal Transient



Contour No.	Temp °F
1	435
2	442
3	450
4	457
5	465
6	472
7	480
8	487
9	495
10	502
11	510
12	517
13	525
14	532
15	540
16	547
17	555

X = 558°F

0 = 430

Figure 4-13 Excessive Feedwater Flow Temperature Contours - Example of a Fast Thermal Transient

Contour No.	Temp °F
1	435
2	442
3	450
4	457
5	465
6	472
7	480
8	487
9	495
10	502
11	510
12	517
13	525
14	532
15	540
16	547
17	555

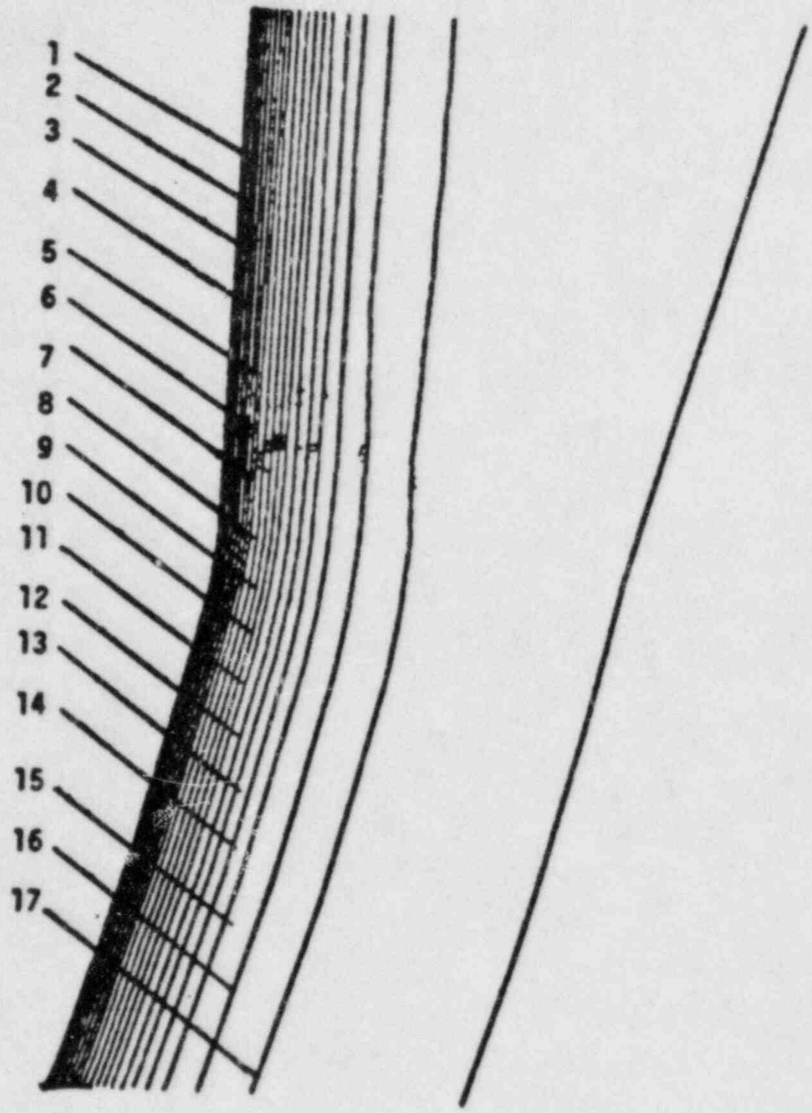


Figure 4-14 Expanded View of Thermal Contours for Excessive Feedwater Flow Transient (See Figure 4-13)

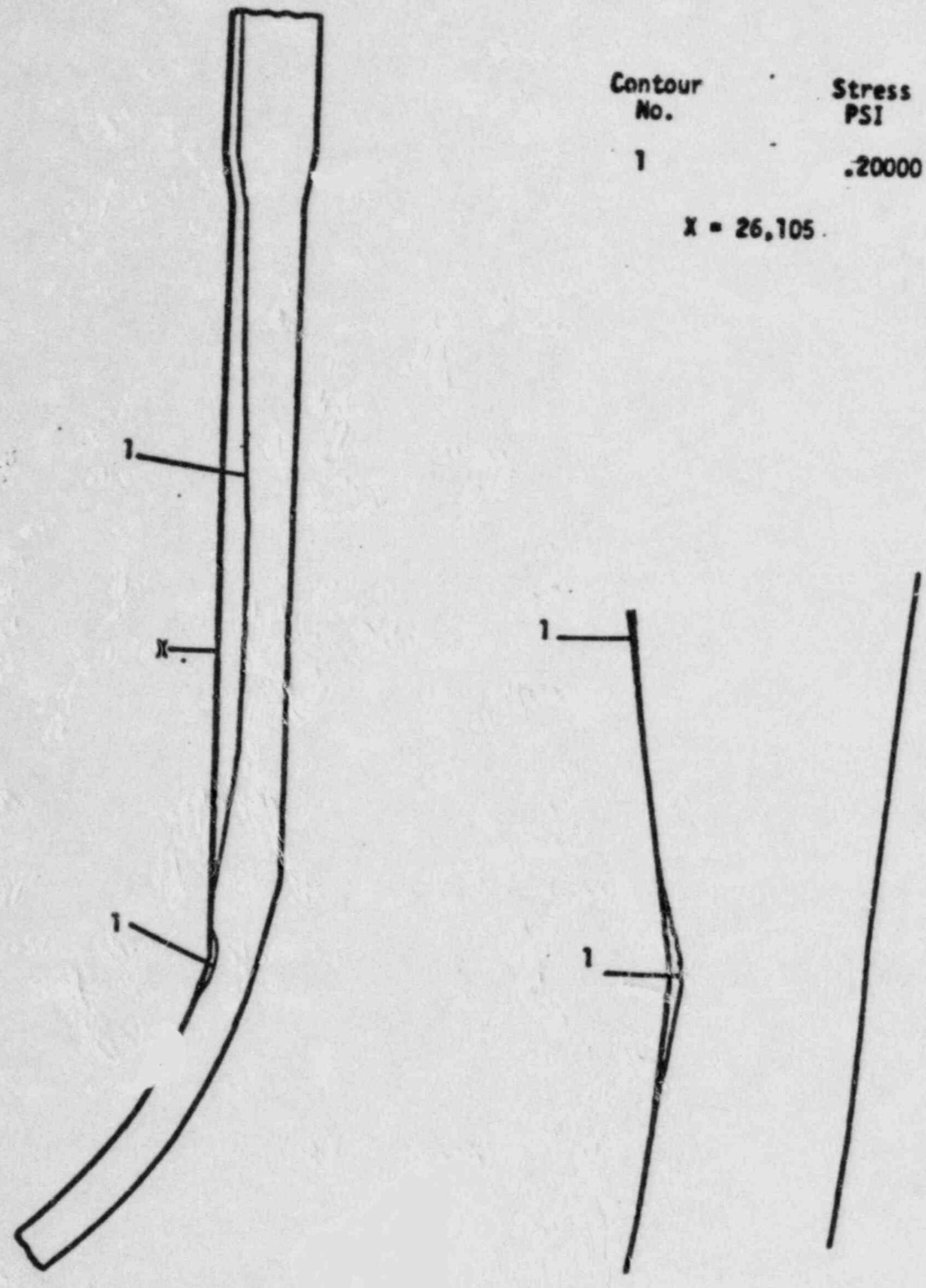


Figure 4-15 Hoop Stress Contours for Cooldown Transient - Example of Slow Therm Transient

SECTION 5

ALLOWABLE FLAW DEPTH CALCULATIONS

5.1 NORMAL, UPSET AND TEST CONDITIONS (LEVEL A AND B)

For these conditions the stress analyses for the fatigue crack growth considerations in the previous section were used. The criterion which must be met for these conditions is

$$K_I \leq \frac{K_{Ia}}{\sqrt{10}}$$

The most severe of all normal, upset and test conditions is the cold hydrotest condition, a constant temperature pressure test, generally carried out only at the beginning of plant life. All the other normal, upset and test transients are less severe, because no pressure exceeds the 3105 psi used for the hydrotest, and thermal stresses will always result in compressive or near zero stresses near the outer surface of the vessel.

The fracture toughness K_{Ia} was determined from the reference toughness curves in Appendix A of Section XI, which were shown in Section 3. To use these curves, RT_{NDT} must be calculated, and in Section 3 it was shown that the RT_{NDT} at end-of-life was 79°F. The leak test temperature for this vessel is required to be at least 310°F for up to 5 effective full power years of operation [6], and thereafter will increase. Hydrotest temperature would be higher than this value, but at any higher temperature the fracture toughness would still be on the upper shelf, which has been set at 200 ksi$\sqrt{\text{in}}$ for this steel.

Therefore using the criterion, we find

$$\frac{200}{\sqrt{10}} = 63.2 \text{ ksi}\sqrt{\text{in}} > K_I$$

and the maximum size of an allowable flaw will be that for which the maximum calculated stress intensity factor is 63.2 ksi$\sqrt{\text{in}}$.

The stress intensity factor for an outside surface flaw in the Indian Point Unit 2 reactor vessel was calculated using the expression published recently by Newman and Raju [7]. This expression is applicable for a range of aspect ratios, and for this analysis the length to depth ratio was set at 2, even though the actual dimensions result in a ratio of 1.4. This assumption provides some small conservatism in the results obtained.

The hoop stress distribution for an internal pressure was calculated directly in the finite element analyses detailed in Section 4.

Results of the stress intensity factor calculations using the 3105 psi hydrotest condition are shown in Figure 5-1. Using the relationship and the previously calculated allowable stress intensity factor, we find that a flaw with a depth up to 31 percent of the wall, or 2.67 inches is acceptable.

5.2 EMERGENCY AND FAULTED CONDITIONS

The emergency and faulted transient categories considered in this evaluation are:

- Large Steamline Break (LSLB)
- Small Steamline Break (SSLB)
- Large LOCA
- Small LOCA
- Steam Generator Tube Rupture (SGTR)

Since plant specific transients were not available, the generic transients developed in References 8, 9 and 10 for the Westinghouse Owners Group (WOG) were used. All of these transients involve severe cooling of the inside vessel surface which results in compressive loadings near the outer surface. For example, the low temperature and low pressure characterizing the small LOCA transient result in compressive stresses near the outer surface as shown in Figure 5-2. For transients with low temperature and high pressure, there is a potential for tensile stresses to exist near the outer surface.

For all of the emergency and faulted transients considered, the lowest temperature at the outer half of the vessel wall for which the stresses are tensile is approximately 240°F. This result is obtained from the evaluation of the small steamline break transient. Figures 5-3 and 5-4 show the temperature and stress distribution for this transient. The temperature, 240°F, results in the fracture toughness K_{Ic} occurring on the upper shelf which has been assumed to equal 200 ksi$\sqrt{\text{in}}$.

Using the Section XI criterion for faulted and emergency conditions (level C and D), we have

$$\frac{K_{Ic}}{\sqrt{2}} > K_I$$

or

$$\frac{200}{\sqrt{2}} = 141.4 \text{ ksi}\sqrt{\text{in}} > K_I$$

Stress intensity factor calculations were carried out for the worst case faulted condition, small steamline break transient, which includes a repressurization to 2350 psi, using the same methods as previously described for normal, upset and test conditions.

For a postulated outside surface flaw,the results are shown in Figure 5-1. The K_I values for this transient never reach 141.1 ksi$\sqrt{\text{in}}$, therefore the emergency and faulted conditions considered do not affect the integrity of Indian Point 2 vessel.

To verify the acceptability of the flaw indication relative to vessel integrity following emergency and faulted conditions, the results of available probabilistic analysis were reviewed. The review shows that on a probabilistic basis the affect of the indication on the risk of significant flaw extension or vessel failure is negligible. The details of this analysis are discussed in Appendix B.

5.3 PRIMARY STRESS LIMITS

In addition to satisfying the fracture criteria, it is required that the primary stress limits of Section III, paragraph NB 3000 be satisfied. A local area reduction of the pressure retaining membrane must be used, equal to the area of the detected indication. Specifically, two criteria must be met:

$$P_\ell + P_b < 1.5\ S_m \quad \text{for design conditions}$$

$$P_\ell + P_b < 3.0\ S_m \quad \text{for normal, upset and test conditions.}$$

To evaluate these criteria it was assumed that the indication extended along the entire length of the vessel, reducing the net section by 1.45 inches. Even under this extremely conservative assumption the indication can easily be shown to be acceptable.

From Table I-1.1 of NB 3000, S_m for A533B Class 1 steel at 600°F was found to be 26.7 ksi. The applied surface stresses for the governing transients were taken directly from Table 4.2, and increased linearly in proportion to the reduction in area. For design conditions a pressure of 2500 psi was used. From Table 4-2, the maximum surface stress for a pressure of 2500 psi can be obtained from the results for "Hot Hydro Test", and can be shown to be a conservative value for $P_m + P_b$. After reducing the net section appropriately, we find that

$$P_\ell + P_b = 32.14 \text{ ksi} < 1.5\ S_m = 40 \text{ ksi}$$

The second criterion, for all normal, upset and test conditions, is less limiting. The governing condition is the "Cold Hydrostatic Test" found in Table 4.2. After reducing the net section appropriately,

$$P_\ell + P_b = 39.92 \text{ ksi} < 3.0\ S_m = 80.1 \text{ ksi}$$

Therefore, the primary stress limits of NB 3000 are met.

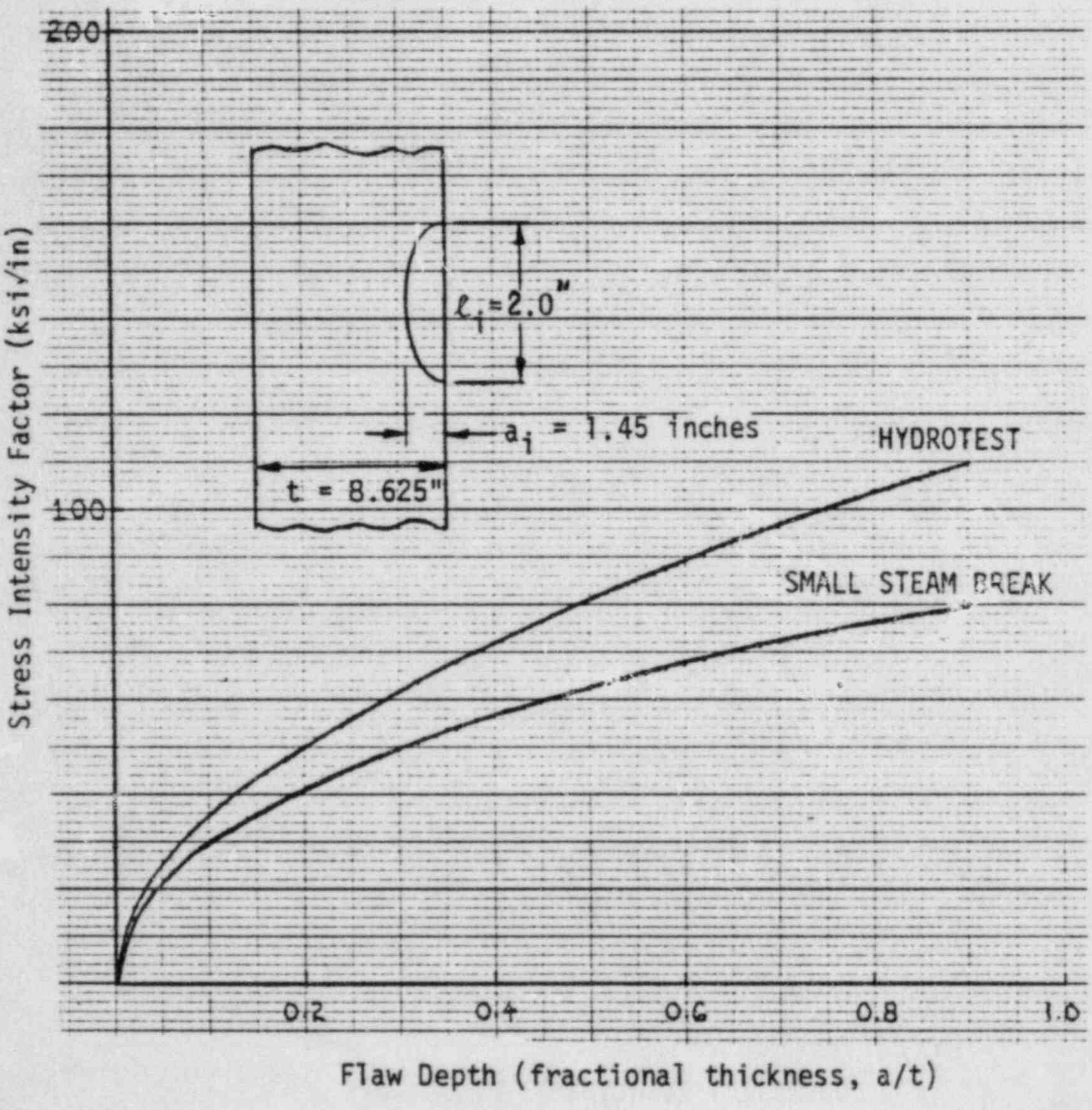


Figure 5-1 Stress Intensity Factor Calculations - Surface Flaw

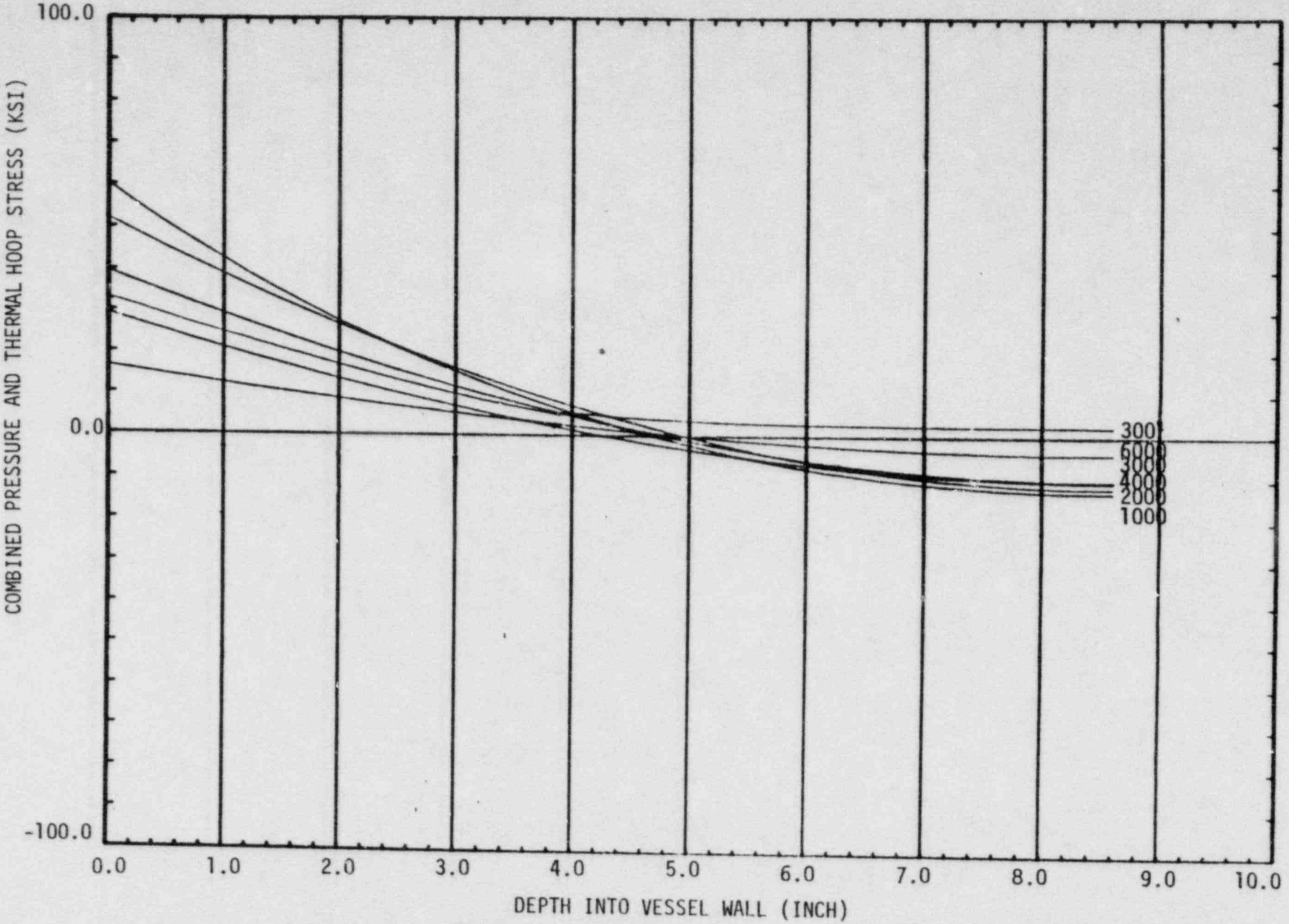


Figure 5-2 Combined Pressure and Thermal Hoop Stress for the Small LOCA Transient (time in seconds)

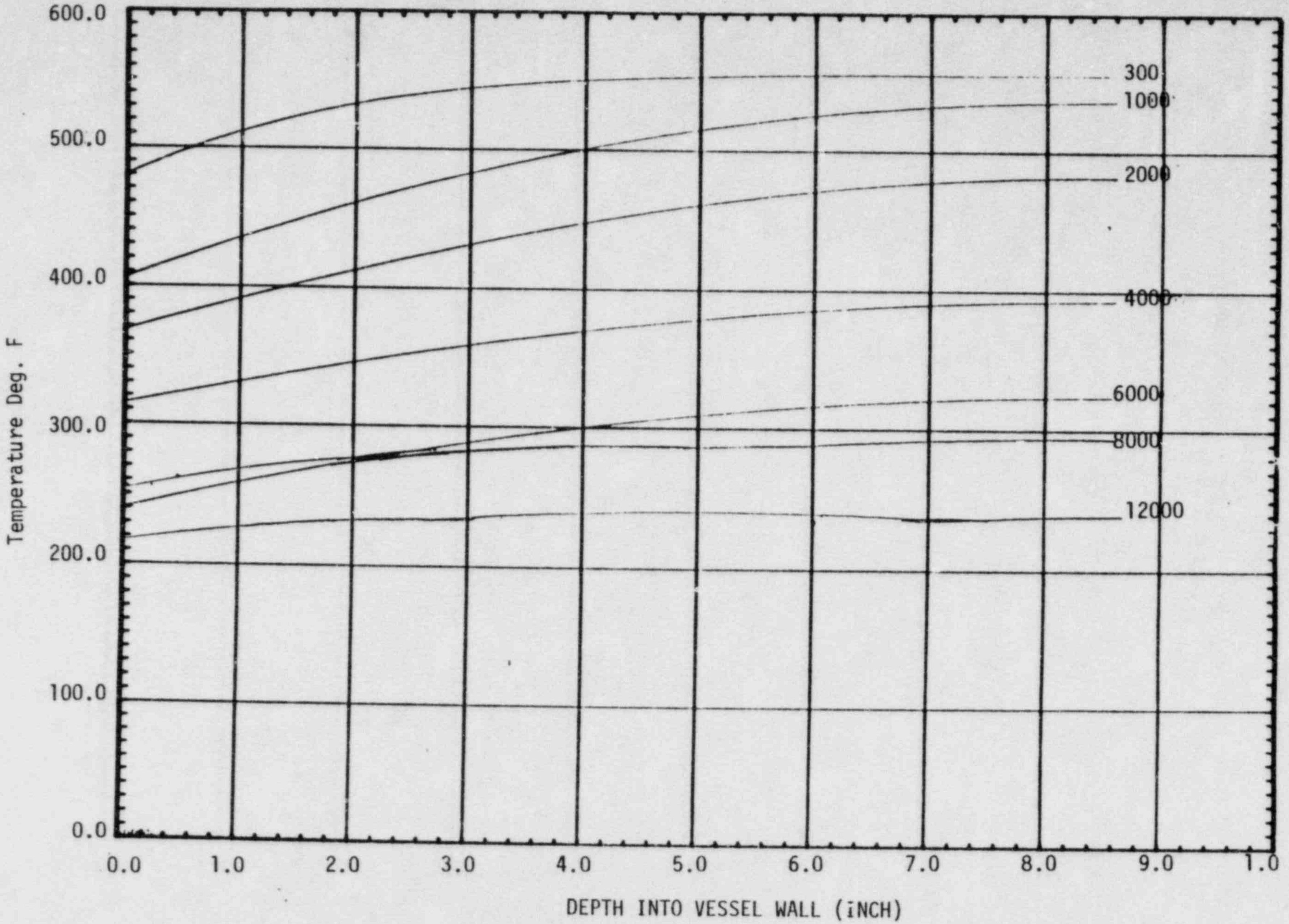


Figure 5-3 Temperature Distribution for Small Steamline Break (time in seconds)

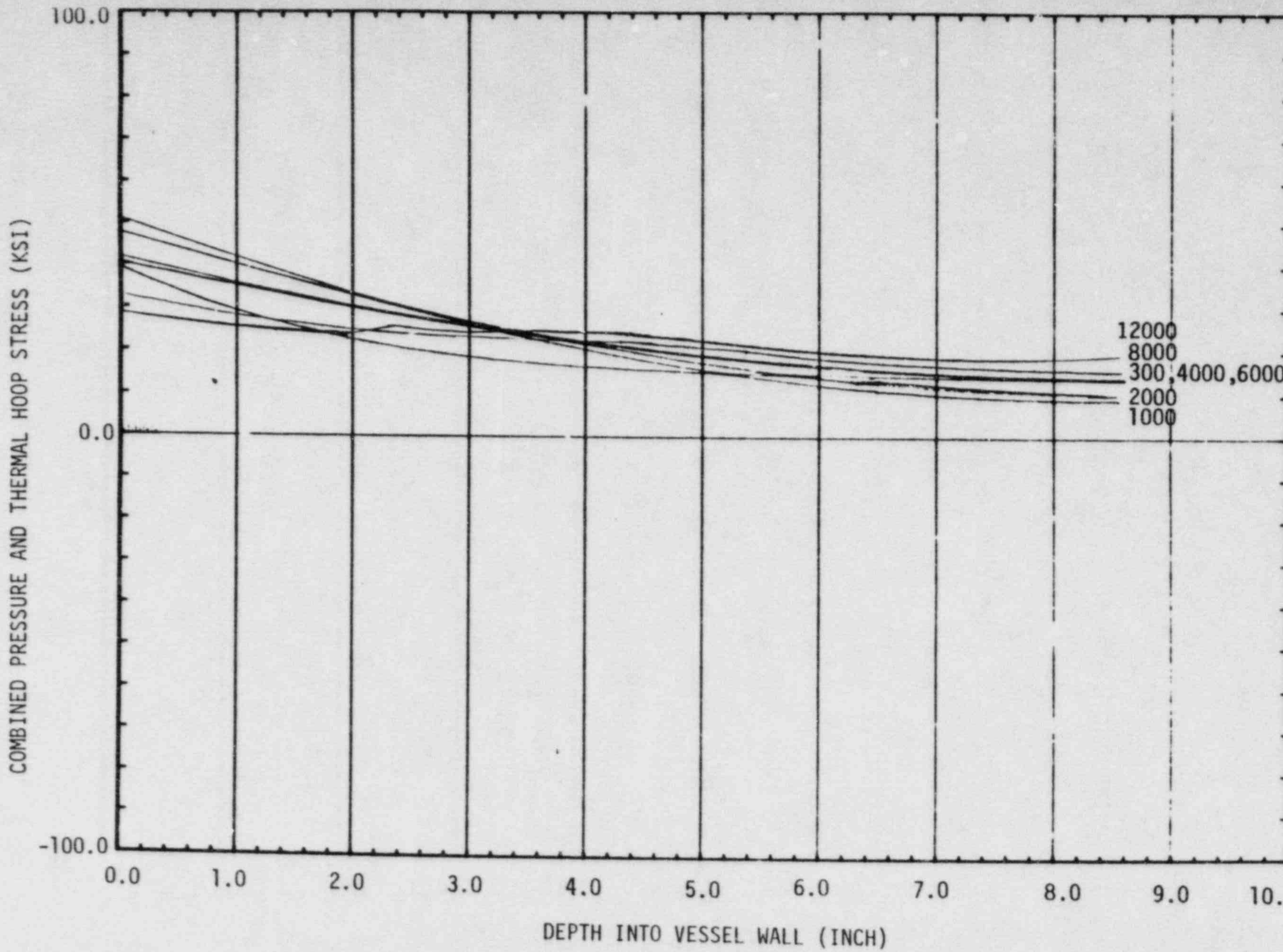


Figure 5-4 Combined Pressure and Thermal Hoop Stress for the Small Steamline Break (time in seconds)

SECTION 6

EXTERNAL THERMAL SHOCK FROM CAVITY FLOODING

Since the indication in the Indian Point Unit 2 reactor vessel is located near the outside surface, the effect of an external thermal shock due to inadvertent flooding of the reactor vessel cavity is of interest, because such an event would produce positive thermal stresses in the region of the indication. Even though such an event is not a design transient, it has occurred, at Indian Point Unit 2, and will be evaluated for completeness.

A detailed fracture analysis of such an incident was recently completed [11], and those results will be reported here, to show that this type event is not of concern to structural integrity.

In the earlier analysis a flaw was assumed to exist at the junction of the lower shell and bottom head region, oriented axially. This is a somewhat more severe stress location than that of the actual indication. The cavity flooding water boiled as it came in contact with the vessel, which was assumed to be operating at steady-state pressure of 2250 psi, at 550°F. The stress and temperature results for this case are shown in Table 6-1.

The fracture toughness for this case is equal to the upper shelf toughness, which is 200 ksi$\sqrt{\text{in}}$. Using the emergency and faulted condition criteria, the allowable flaw depth is

$$\frac{200}{\sqrt{2}} = 141.4 \text{ ksi}\sqrt{\text{in}} > K_I$$

The stresses were linearized through the vessel wall thickness, and the stress intensity factor expression of Section XI of Appendix A was used for an external surface flaw. The stress intensity factor is presented as a function of flaw depth in Figure 6-1. It can be seen here that the allowable flaw depth is 1.77 inches, and therefore the external thermal shock from cavity flooding is not a threat to the integrity of the Indian Point Unit 2 vessel, even with the observed indication.

TABLE 6-1

TEMPERATURES AND STRESSES FOR CAVITY FLOODING CORE (REF 11)

Node Temperature:	200°F (Outer Wall)	284°F	370°F	458°F	550°F (Inner Wall)

Hoop Stresses:	Outer Wall	Inner Wall
	84.4 ksi	-41.3 ksi

Linearized Stress Components

$\sigma_m = + 21.6$ psi $\qquad$ $\sigma_b = + 62.8$ ksi

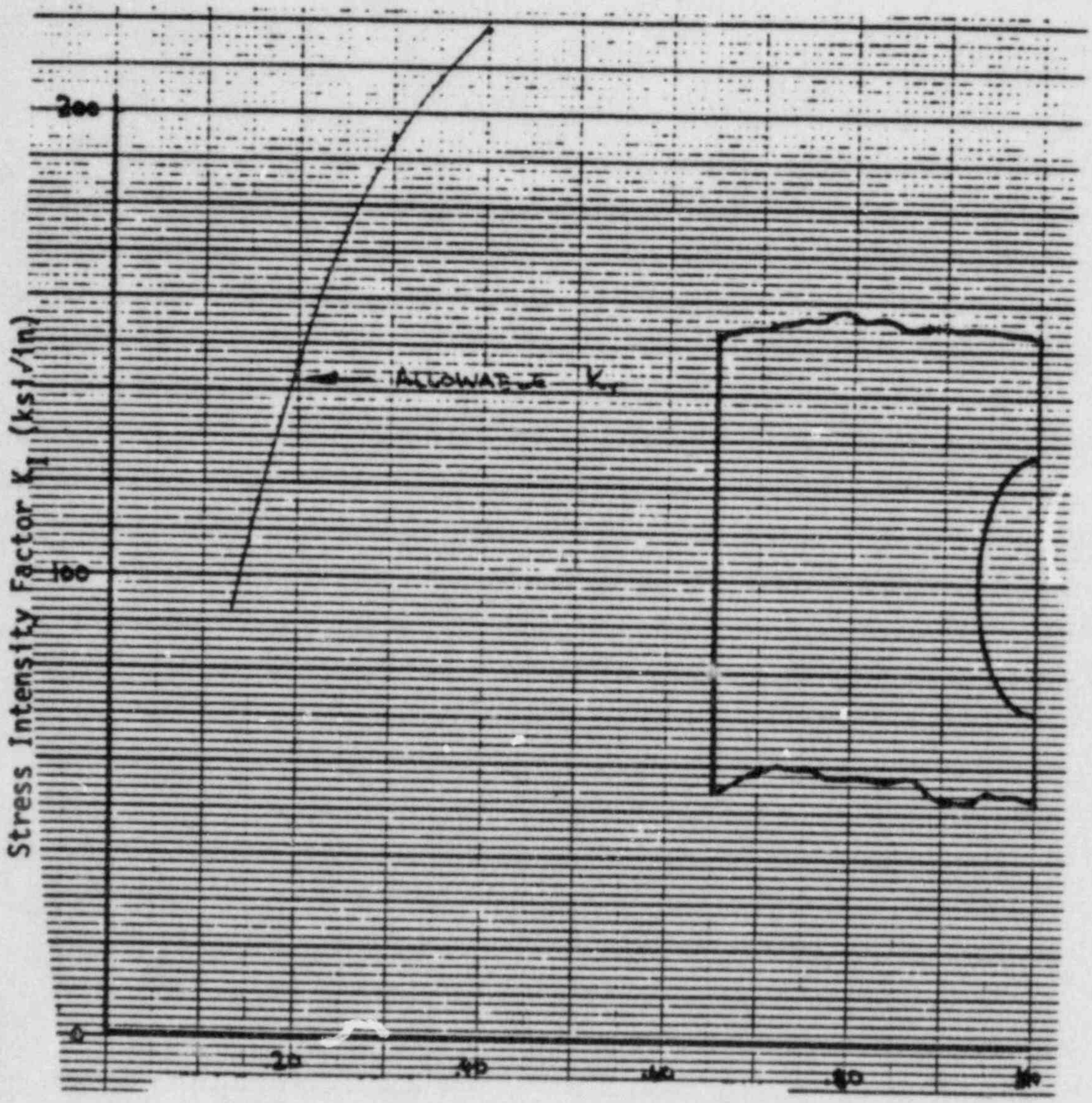


Figure 6-1 Fracture Results of Cavity Flooding

SECTION 7

SUMMARY AND CONCLUSIONS

An evaluation of the recently observed indication in the beltline region of the Indian Point Unit 2 reactor vessel has been completed. The indication is located near the outside surface of the vessel, oriented axially, and for the purpose of this evaluation is assumed to be a flaw. The through-wall dimension is 1.2 inches, axial length 2.0 inches and the edge of the indication is 0.25 inches from the vessel surface. The proximity of the outside surface requires the indication to be characterized as an outside surface flaw. The characterized dimensions of the surface flaw are therefore a depth of 1.45 inches and length equal to 2.0 inches. The characterized depth is 16.8% of the wall thickness.

The initial RT_{NDT} for the longitudinal weld in the beltline region is estimated by the NRC thermal shock report[2] to be -56°F, and the copper content is 0.16 wt. percent. Using radiation damage estimation methods the change in RT_{NDT} was conservatively assessed to be 79°F. With this information, along with the temperatures of the applicable transients, (T_{min} > 240°F) the vessel steel toughness will be on the upper shelf in the region of the indication. Therefore,

$$K_{Ia} = 200 \text{ ksi}\sqrt{\text{in}}$$

$$K_{Ic} = 200 \text{ ksi}\sqrt{\text{in}}$$

The allowable crack size was obtained from IWB 3600 of ASME Section XI. Specifically, for normal, upset and test conditions we have:

$$K_I < \frac{K_{Ia}}{\sqrt{10}} = 63.2 \text{ ksi}\sqrt{\text{in}} \text{ (normal, upset, test)}$$

For emergency and faulted conditions:

$$K_I < \frac{K_{Ic}}{\sqrt{2}} = 141.4 \text{ ksi}\sqrt{\text{in}} \text{ (emergency, faulted)}$$

The stress intensity factor, K_I, is the driving force for the crack after it is subjected to a fatigue crack growth analysis. The fatigue crack growth analysis of Section 4 indicates a growth of about less than 1% in depth for the remaining lifetime of the vessel. This growth is essentially insignificant, and was based on a conservative, up-to-date set of design transients.

For normal and upset conditions, the worst case condition (hydrostatic test) shows an estimated allowable crack depth of 31% of the wall thickness. The flaw as characterized is only 16.8%, so this criterion is met.

For emergency and faulted conditions, the governing condition is a small steamline break transient, as described in Section 5. The results of the fracture evaluation for this transient show that an external surface flaw would not be affected by this transient, since the applied K_I never reaches the allowable, so the criterion for normal, upset and test conditions will be governing.

Therefore, it is concluded that the indication is acceptable without repair.

SECTION 8

REFERENCES

1. NRC Policy Issue, Enclosure A, "NRC Staff Evaluation of Pressurized Thermal Shock," SECY-82-465, Nov. 23, 1982.

2. "Calculation of Operating and NTOL Vessel RT_{NDT} Values," Letter WOG-82-290, December 1982.

3. "Analyses of Neutron Flux Levels and Surveillance Capsule Lead Factors for the Indian Point Unit 2 Reactor," Westinghouse NTD letter SAO-RSA-655, July 1979.

4. WECAN, Westinghouse Electric Computer Analysis User's Manual, Westinghouse R&D, Pittsburgh, Pa., Sept. 17, 1979.

5. Dittus, F. W., and Boelter, L. M. K., "Heat Transfer in Automobile Radiators of the Tubular Type," Calif. Univ. Publication in Eng. 2 No. 13, pp. 443-461, 1930.

6. Norris, E. B., "Reactor Vessel Material Surveillance Program for Indian Point Unit 2 - Analysis of Capsule T," Southwest Research Institute Project 02-4531 Final Report, June 30, 1977.

7. J. C. Newman and I. S. Raju "Stress Intensity Factors for Internal Surface Cracks in Cylindrical Pressure Vessels," Trans. ASME, Vol. 102, Nov. 1980.

8. Cheung, A. C., et. al., "A Generic Assessment of Significant Flaw Extension, Including Stagnant Loop Conditions, from Pressurized Thermal Shock of Reactor Vessels on Westinghouse Nuclear Power Plant," WCAP 10319, December, 1983.

9. "Summary of Small Steamline Break Analysis," performed for the Westinghouse Owners Group, January 1984.

10. Meyer, T. A., "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants", WCAP 10019, December 1981.

11. D. T. Entenmann, et. al., "Indian Point #2 Reactor Vessel Structural Evaluation for Accumulation of Water in Containment Incident," Westinghouse Electric Corp., WCAP 9822, November 1980.

12. Jouris, G. M. and Witt, F. J., "An Application of Probabilistic Fracture Mechanics to Reactor Pressure Vessels Including Multiple Initiation and Arrest Events," Transaction of the Seventh International Conference on Structural Mechanics in Reactor Technology, August 1983.

APPENDIX A

TRANSIENTS DESCRIPTIONS

LEVEL A AND B TRANSIENTS - FIGURES A-1 TO A-46

LEVEL C AND D TRANSIENTS - FIGURES A-47 TO A-51

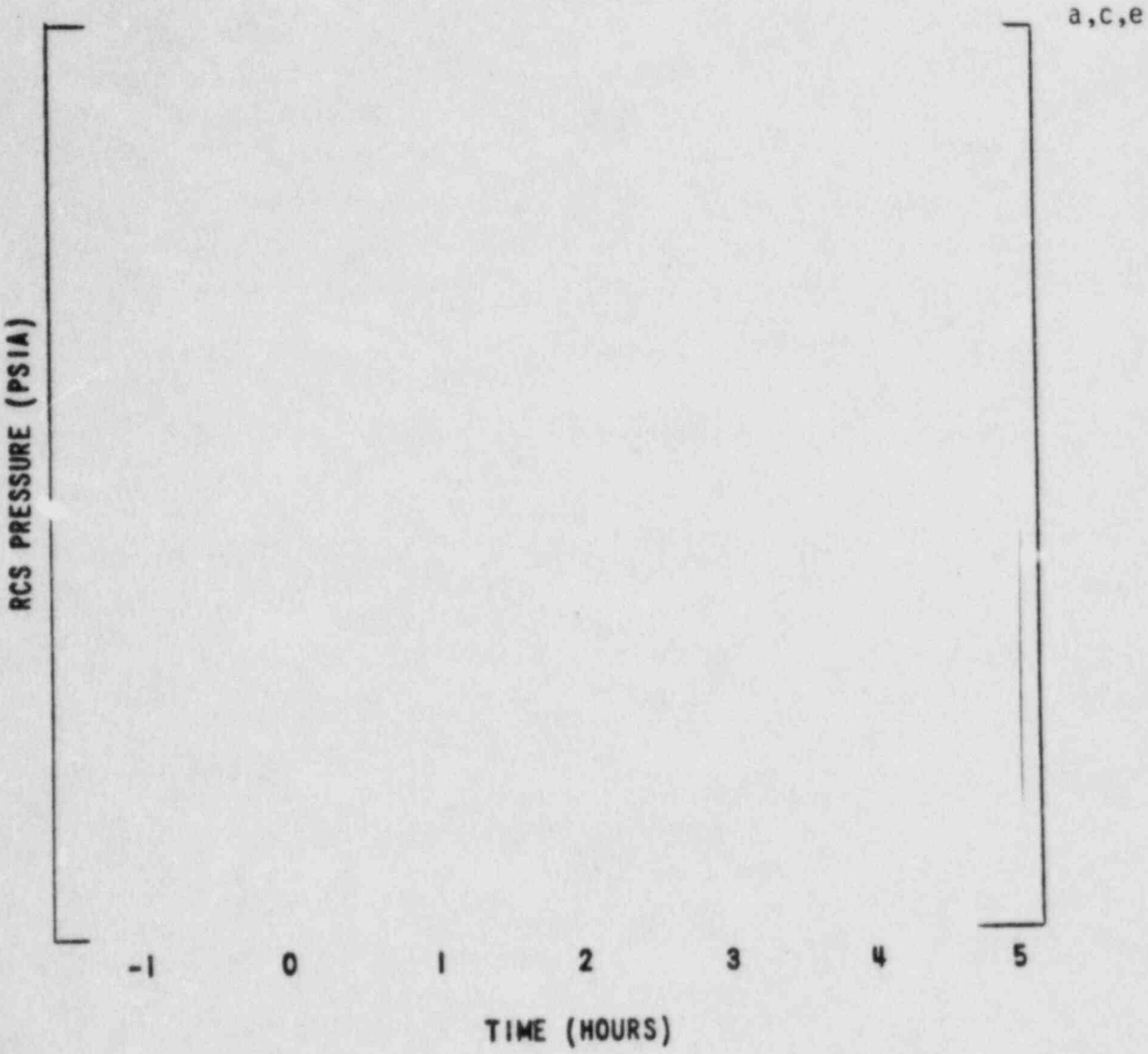


Figure A-1 Plant Heatup - Reactor Coolant System Pressure Versus Time

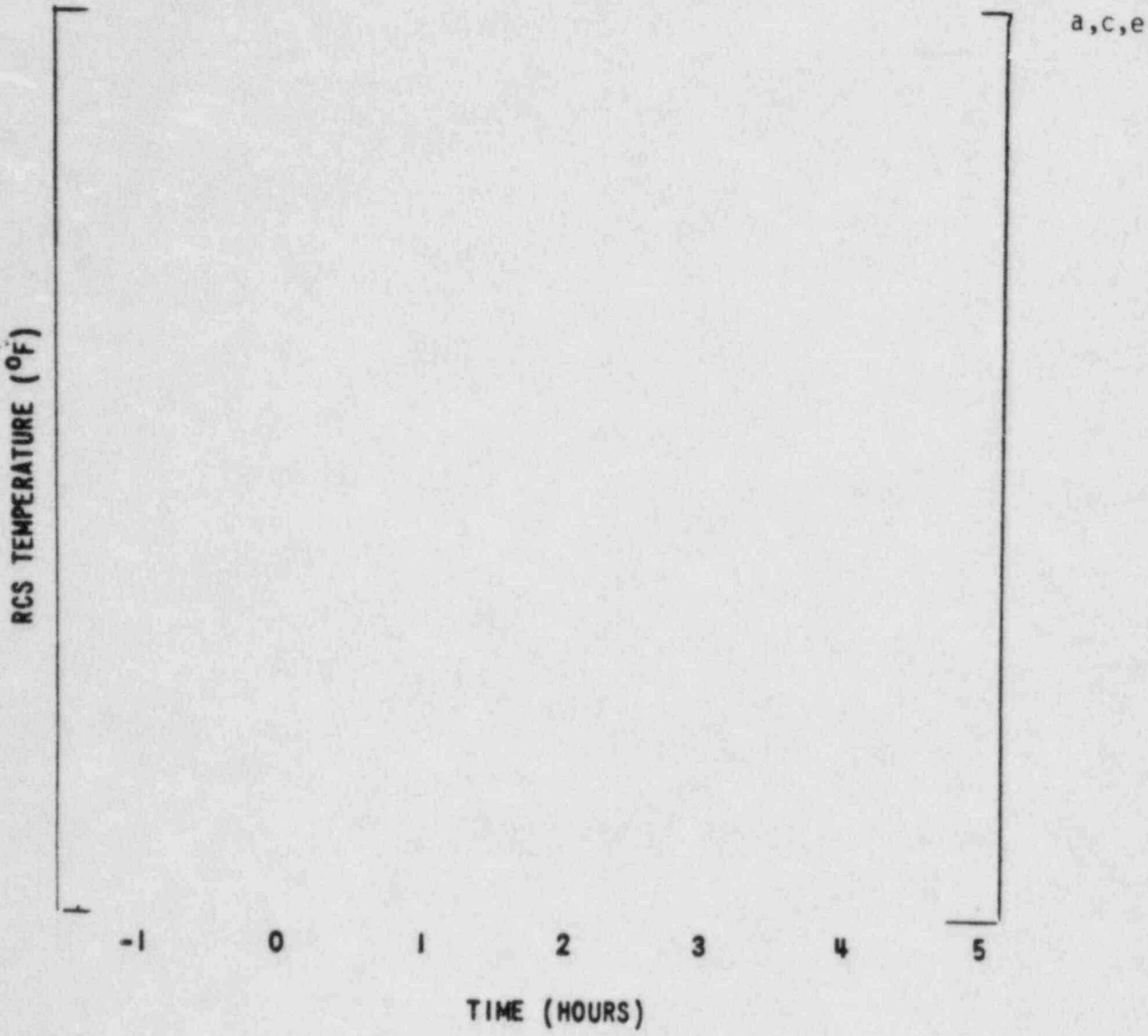


Figure A-2 Plant Heatup - Reactor Coolant System Temperature Versus Time

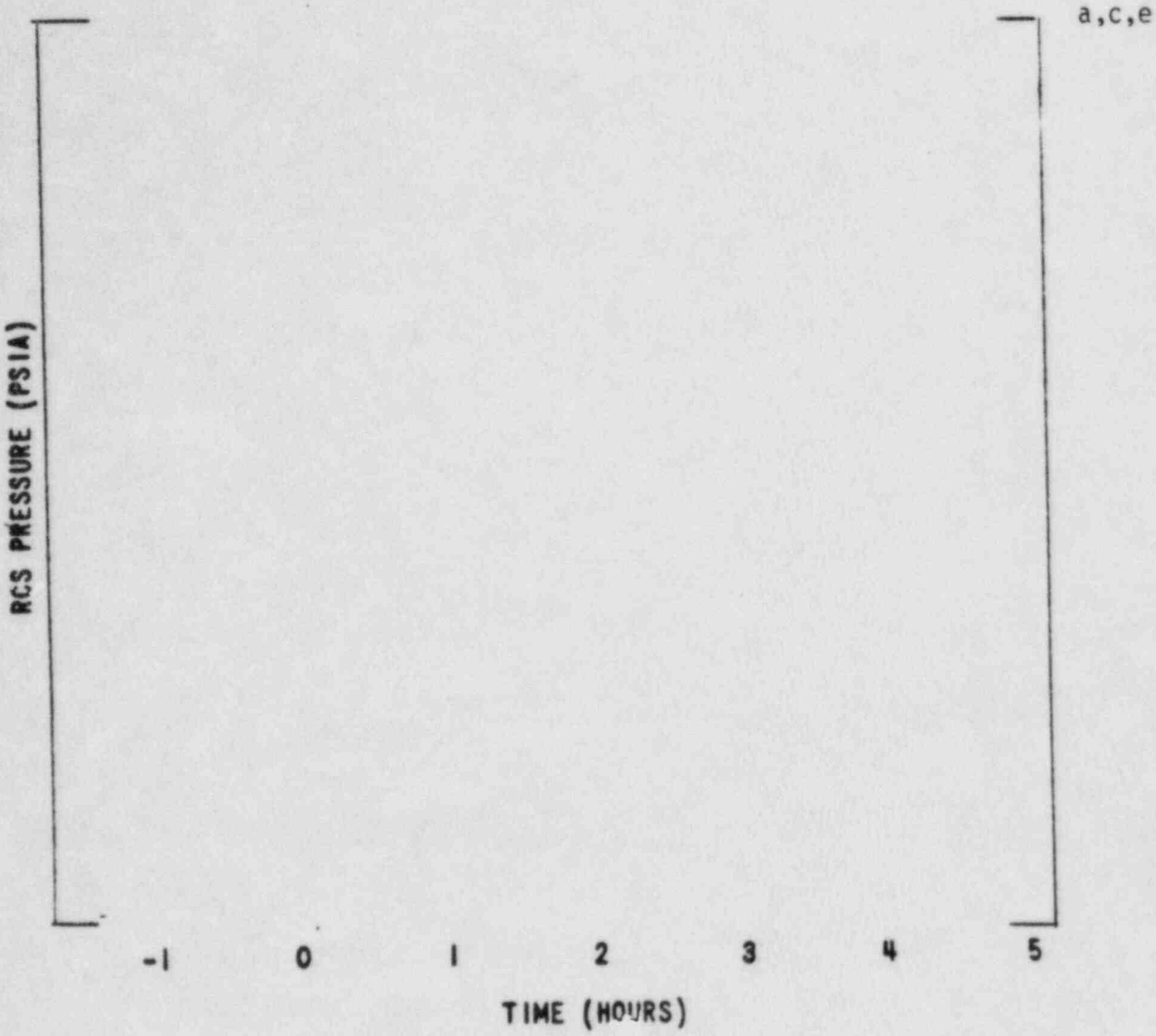


Figure A-3 Plant Cooldown - Reactor Coolant System Pressure Versus Time

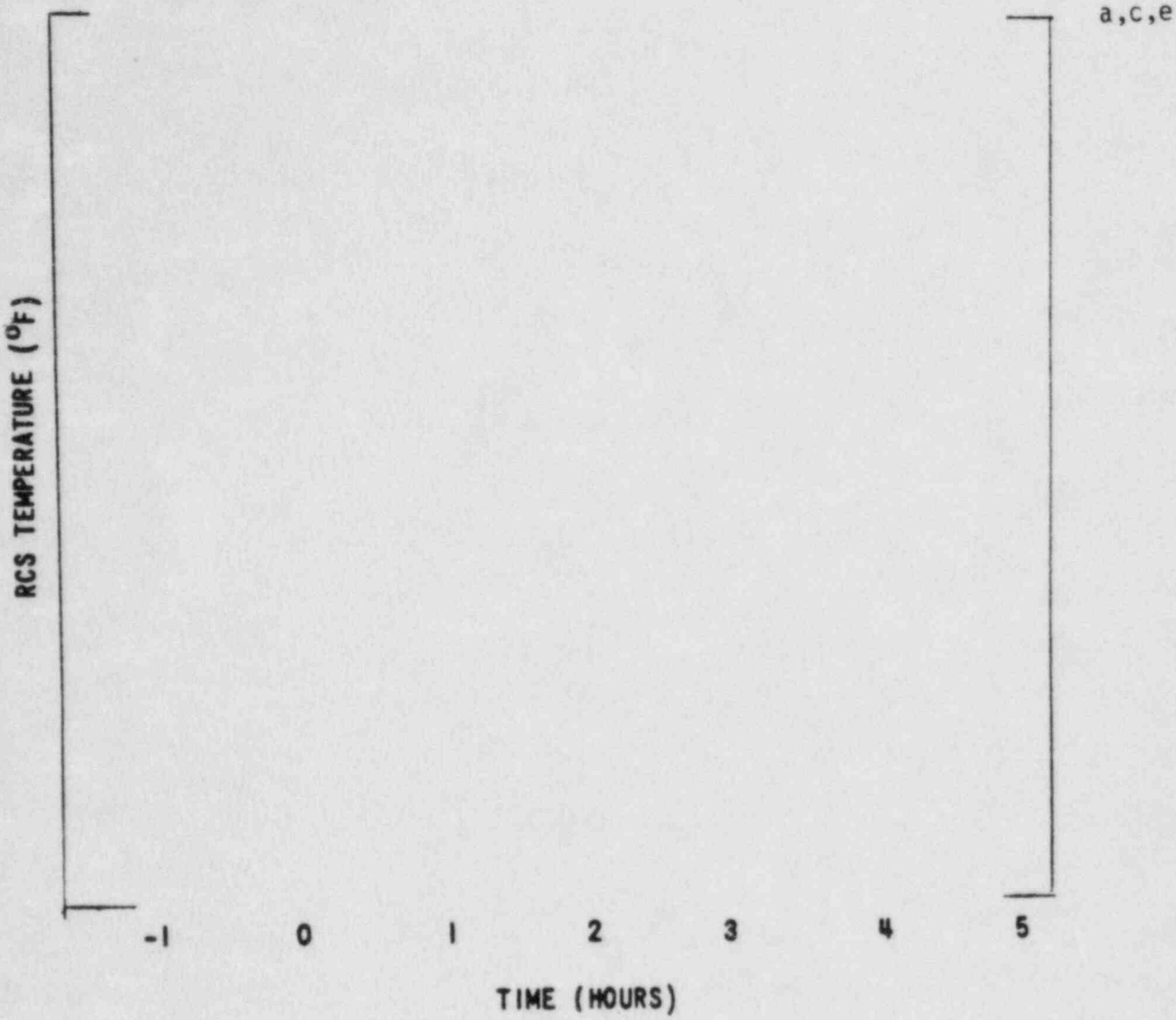


Figure A-4 Plant Cooldown - Reactor Coolant System Temperature Versus Time

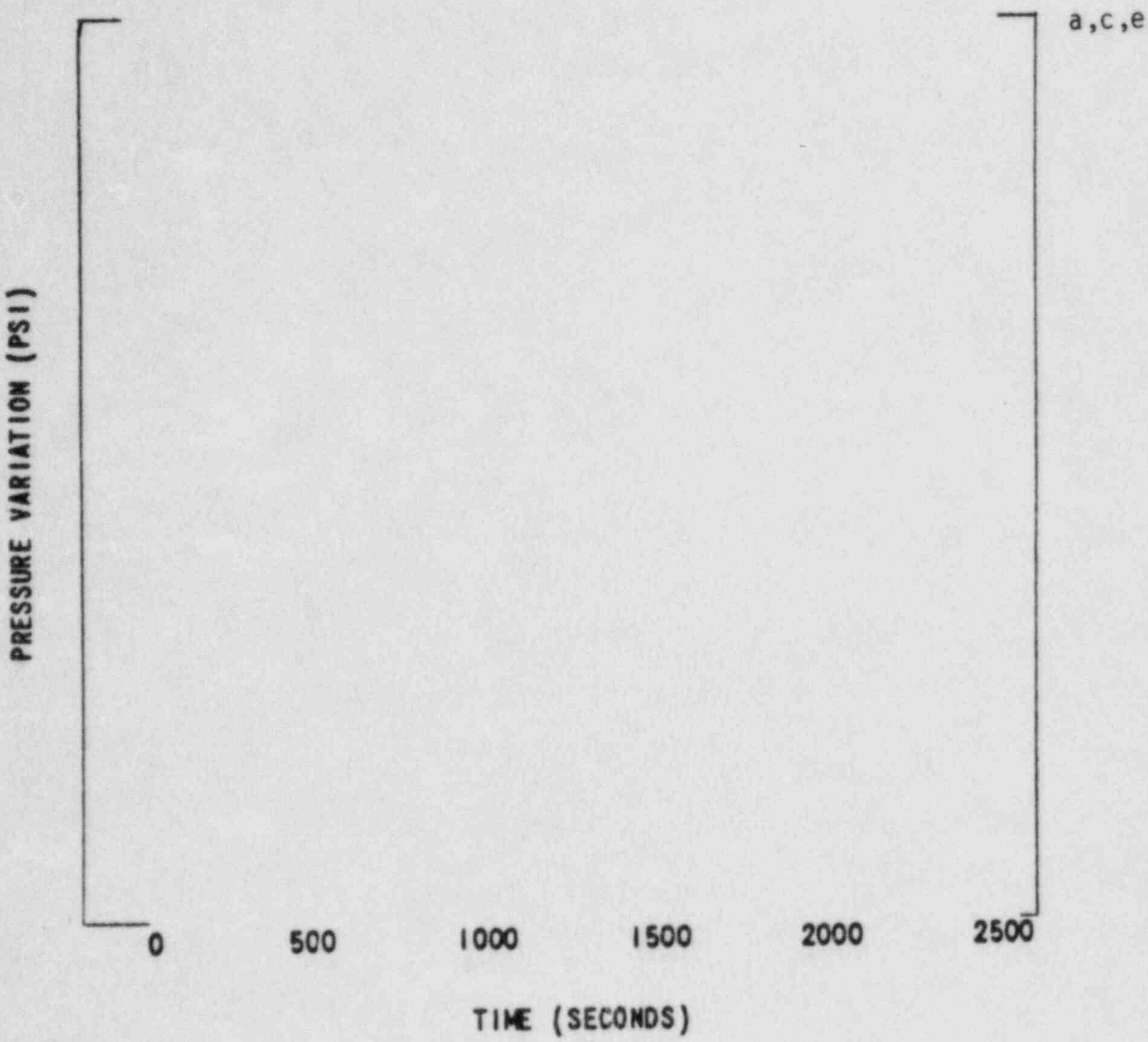


Figure A-5 Unit Loading - Reactor Coolant Pressure Versus Time

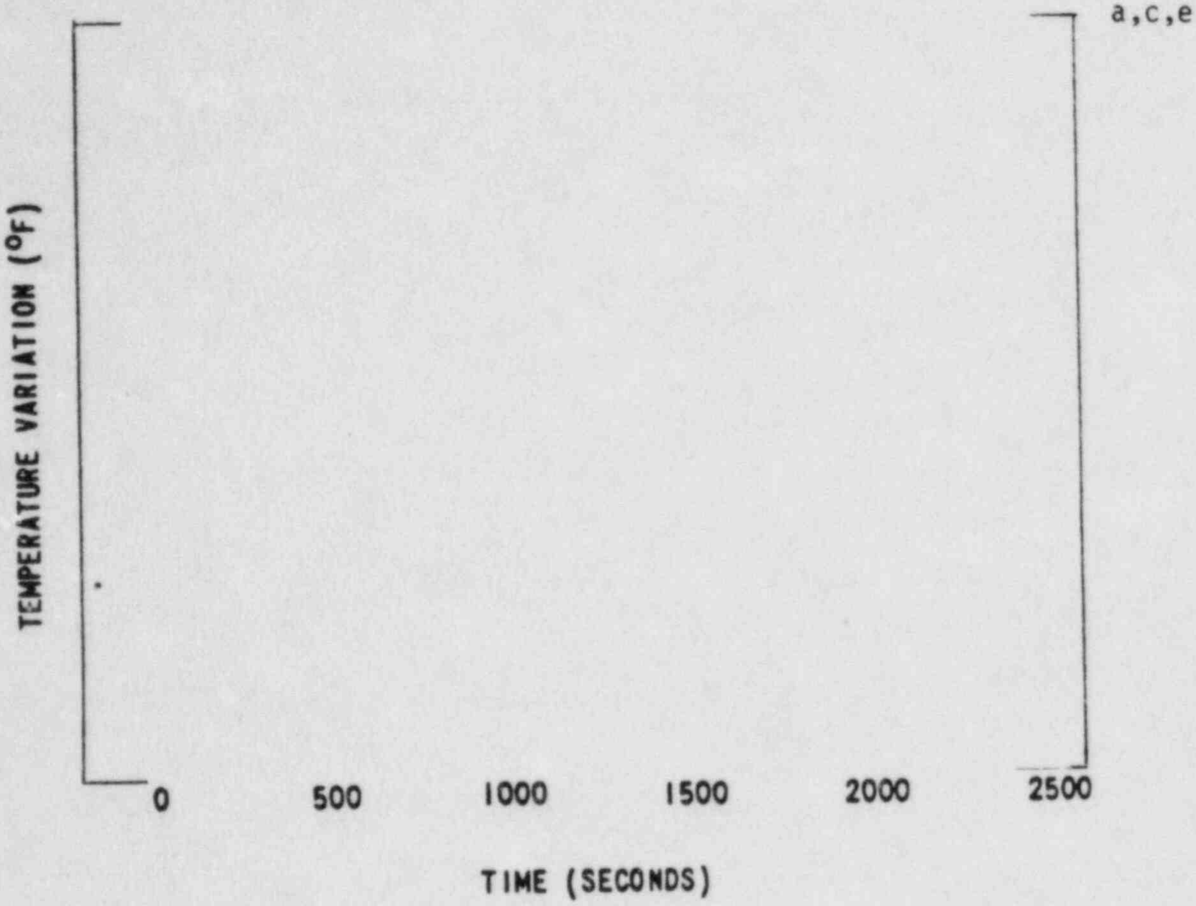


Figure A-6 Unit Loading - Cold Leg Temperature Versus Time

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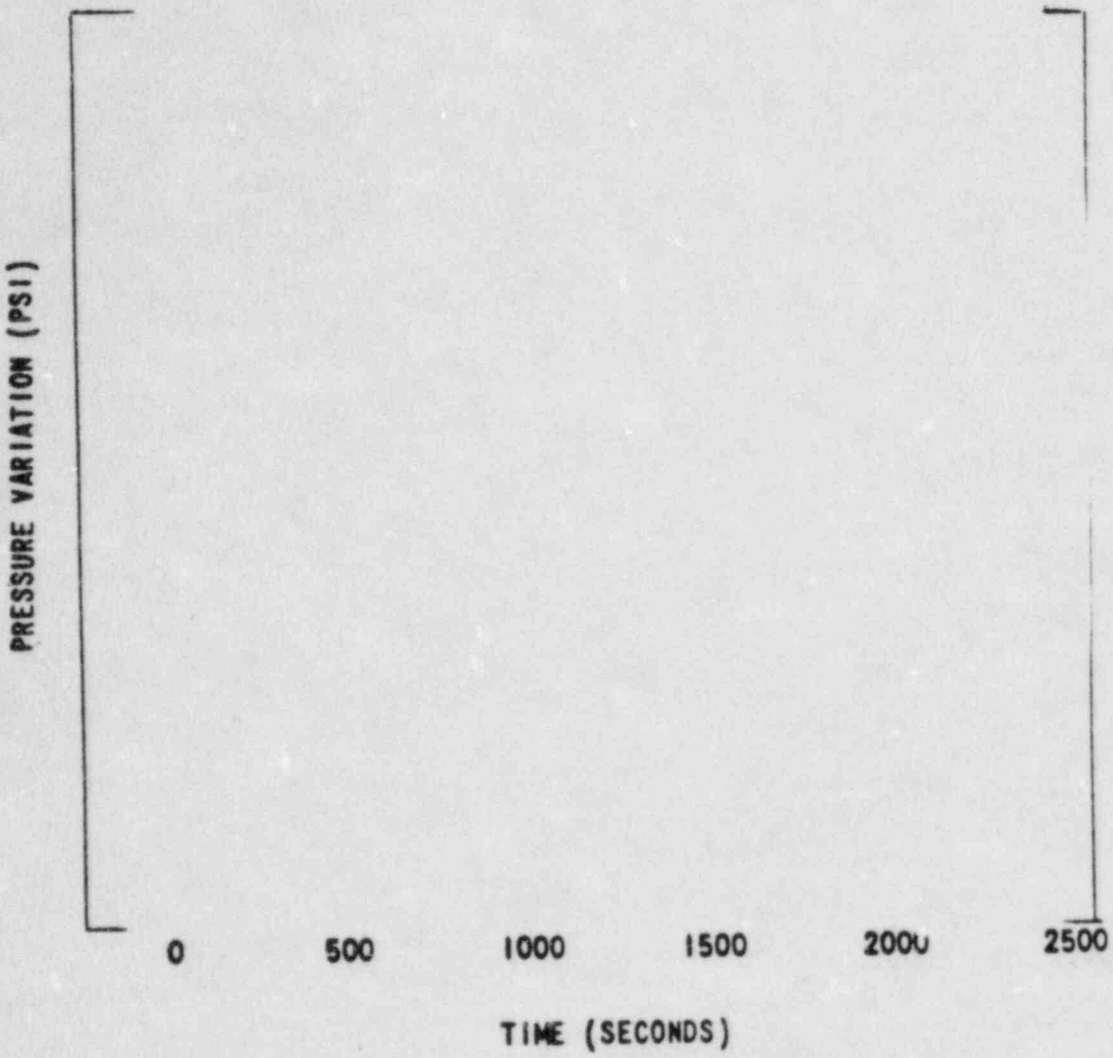


Figure A-7 Unit Unloading - Reactor Coolant Pressure Versus Time

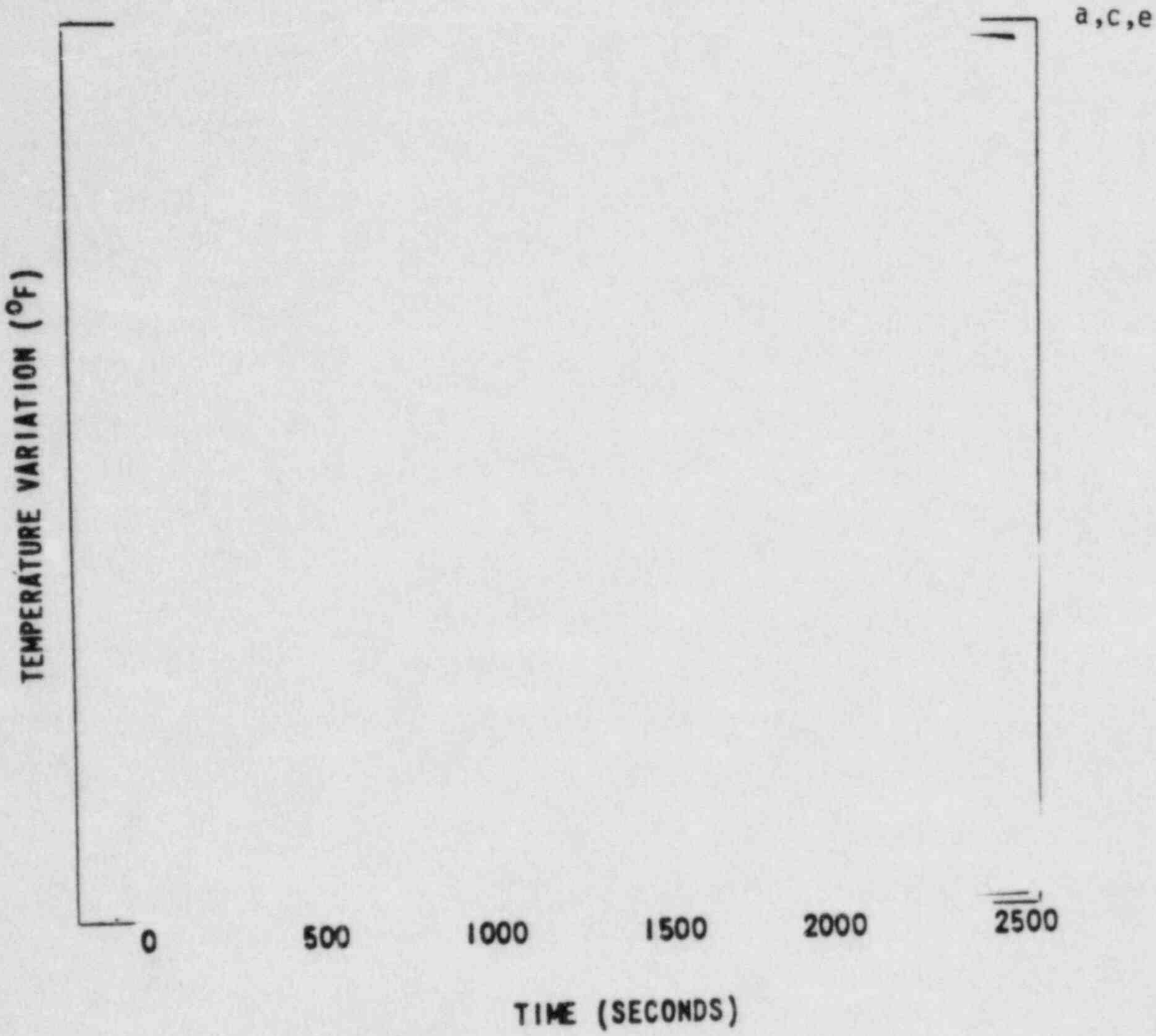


Figure A-8 Unit Unloading - Cold Leg Temperature Versus Time

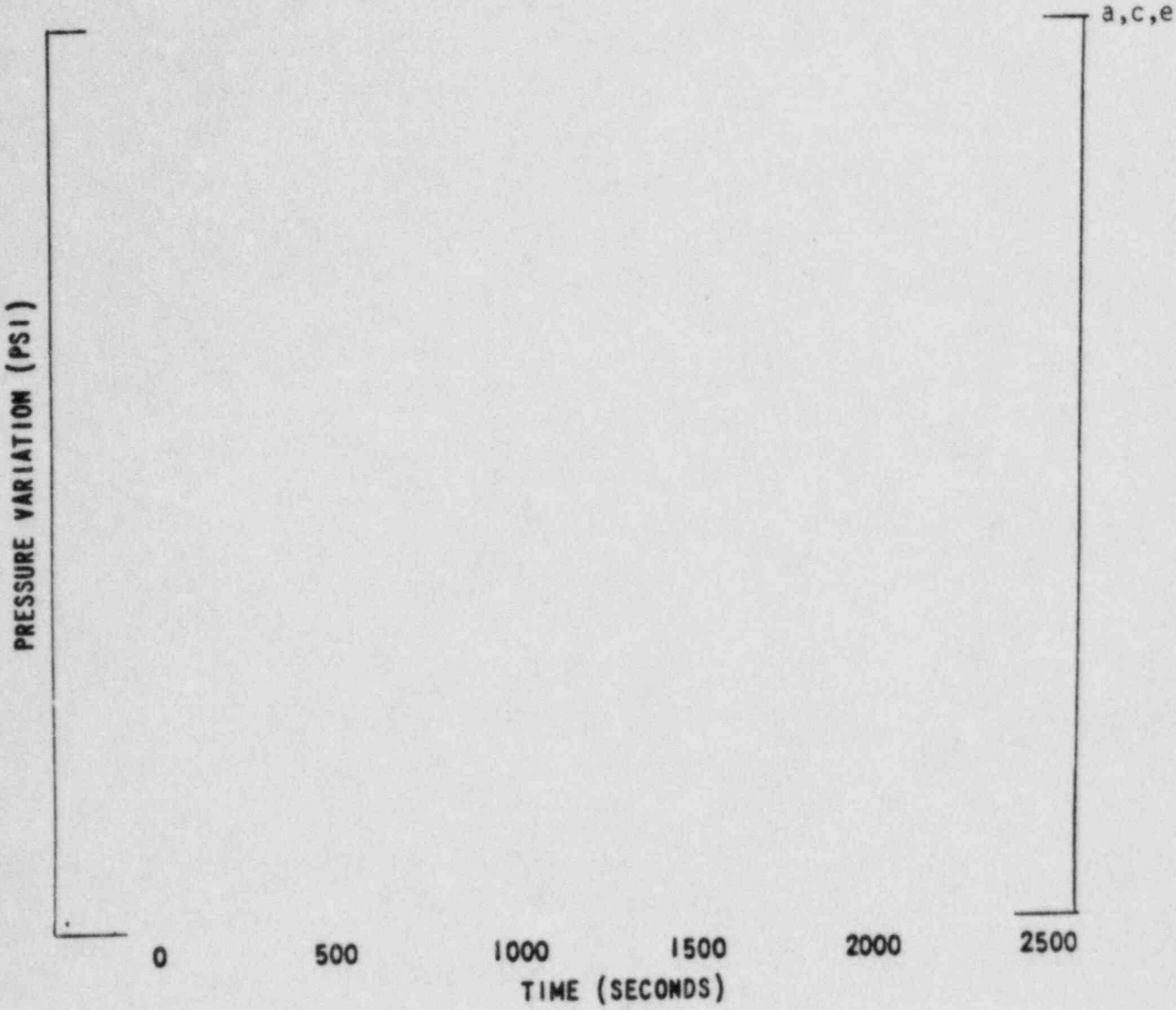


Figure A-9 Reduced Temperature Return to Power - Reactor Coolant Pressure Versus Time

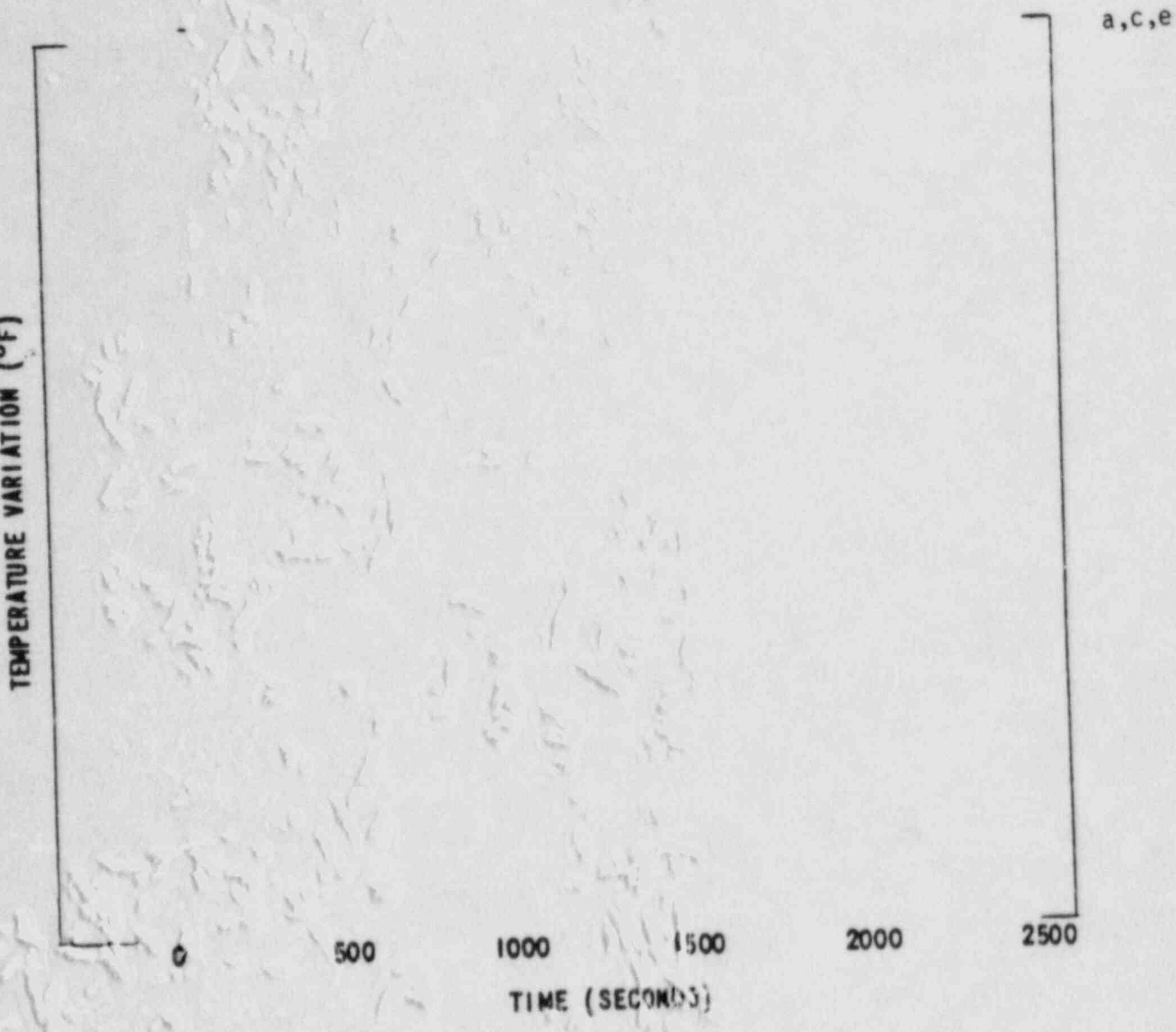
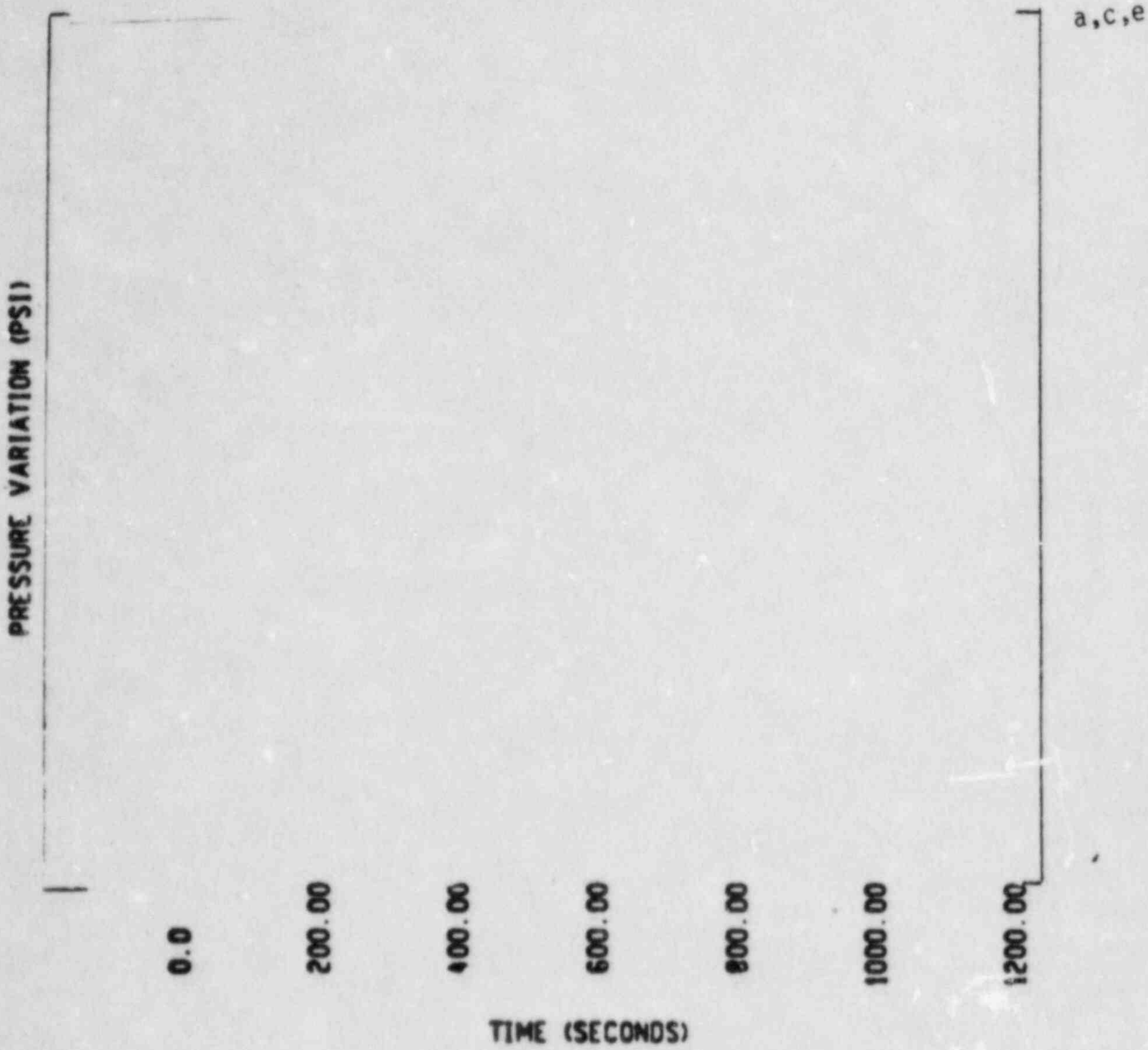


Figure A-10 Reduced Temperature Return to Power - Cold Leg Temperature Versus Time



NOTES: 1. TRANSIENT ASSUMED TO BEGIN AT 90 PERCENT OF FULL POWER.

2. AFTER TRANSIENT IS COMPLETED, PLANT WILL BE AT PROGRAMMED OPERATING CONDITIONS AT THE INCREASED POWER LEVEL (100 PERCENT POWER).

Figure A-11 Step Load Increase - Reactor Coolant Pressure Versus Time

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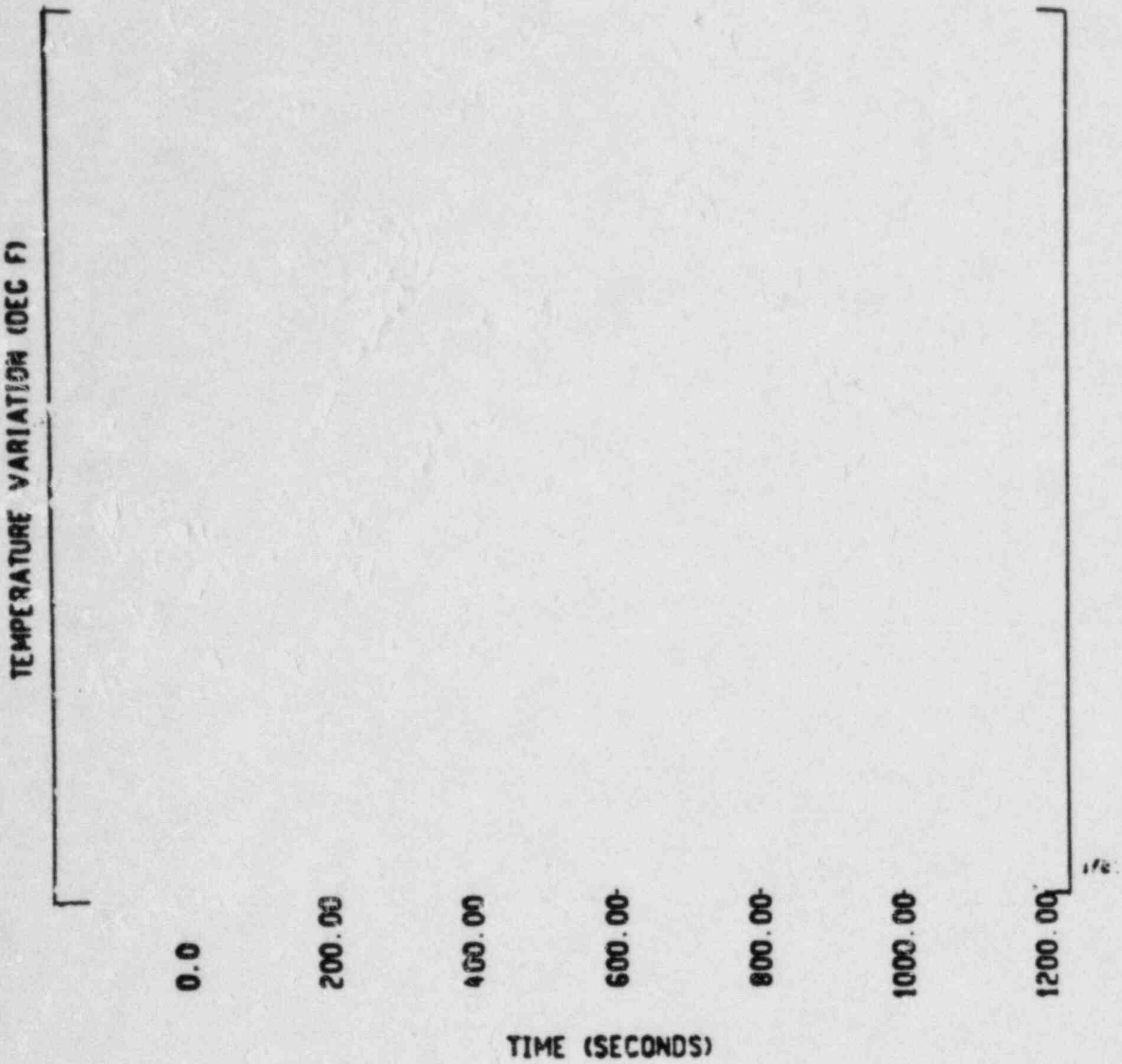
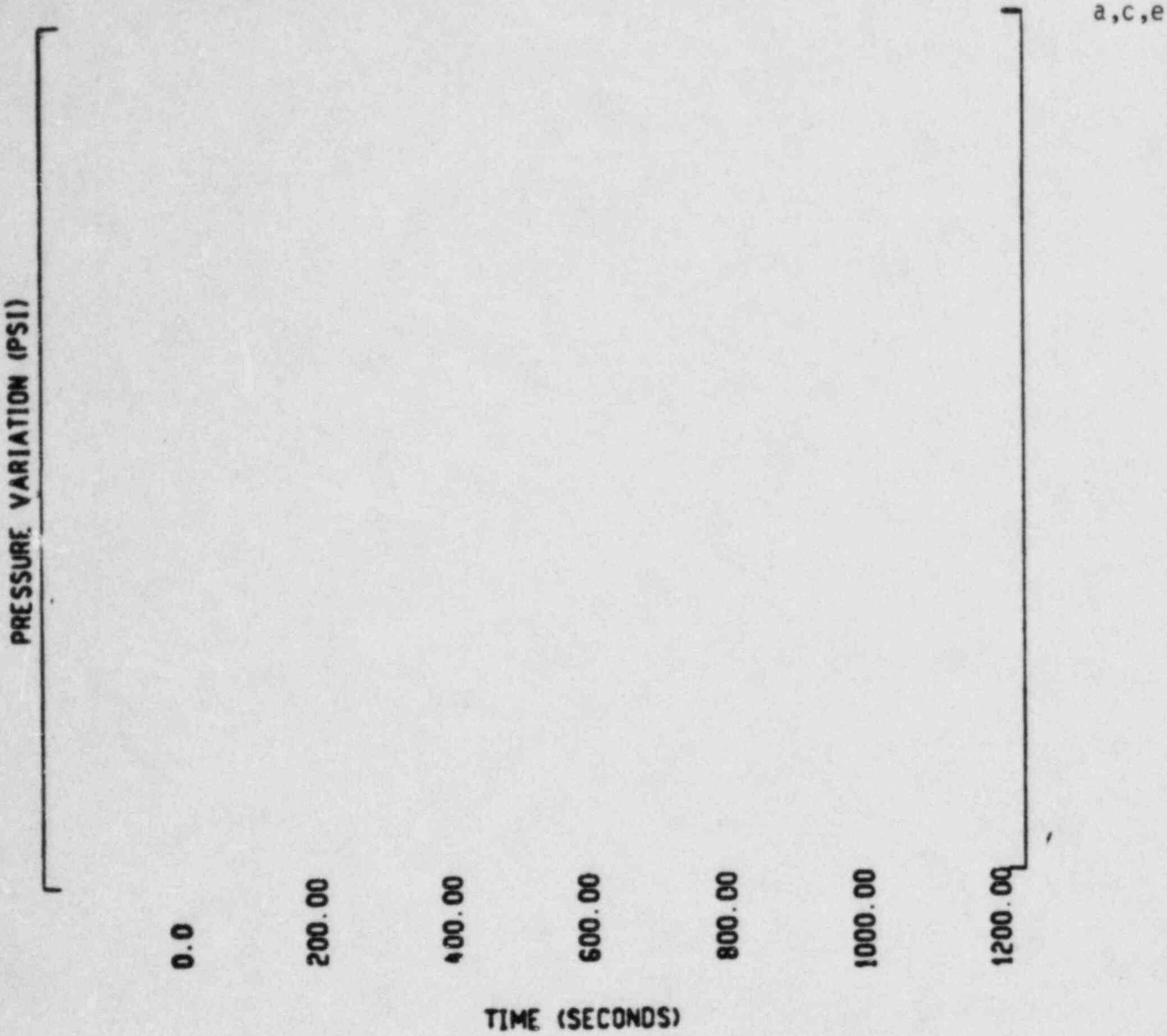


Figure A-12 Step Load Increase - Cold Leg Temperature Versus Time



NOTE: 1. TRANSIENT ASSUMED TO BEGIN AT 100 PERCENT OF FULL POWER.

2. AFTER TRANSIENT IS COMPLETED, PLANT WILL BE AT PROGRAMMED OPERATING CONDITIONS AT THE REDUCED POWER LEVEL (90 PERCENT POWER).

Figure A-13 Step Load Decrease - Reactor Coolant Pressure Versus Time

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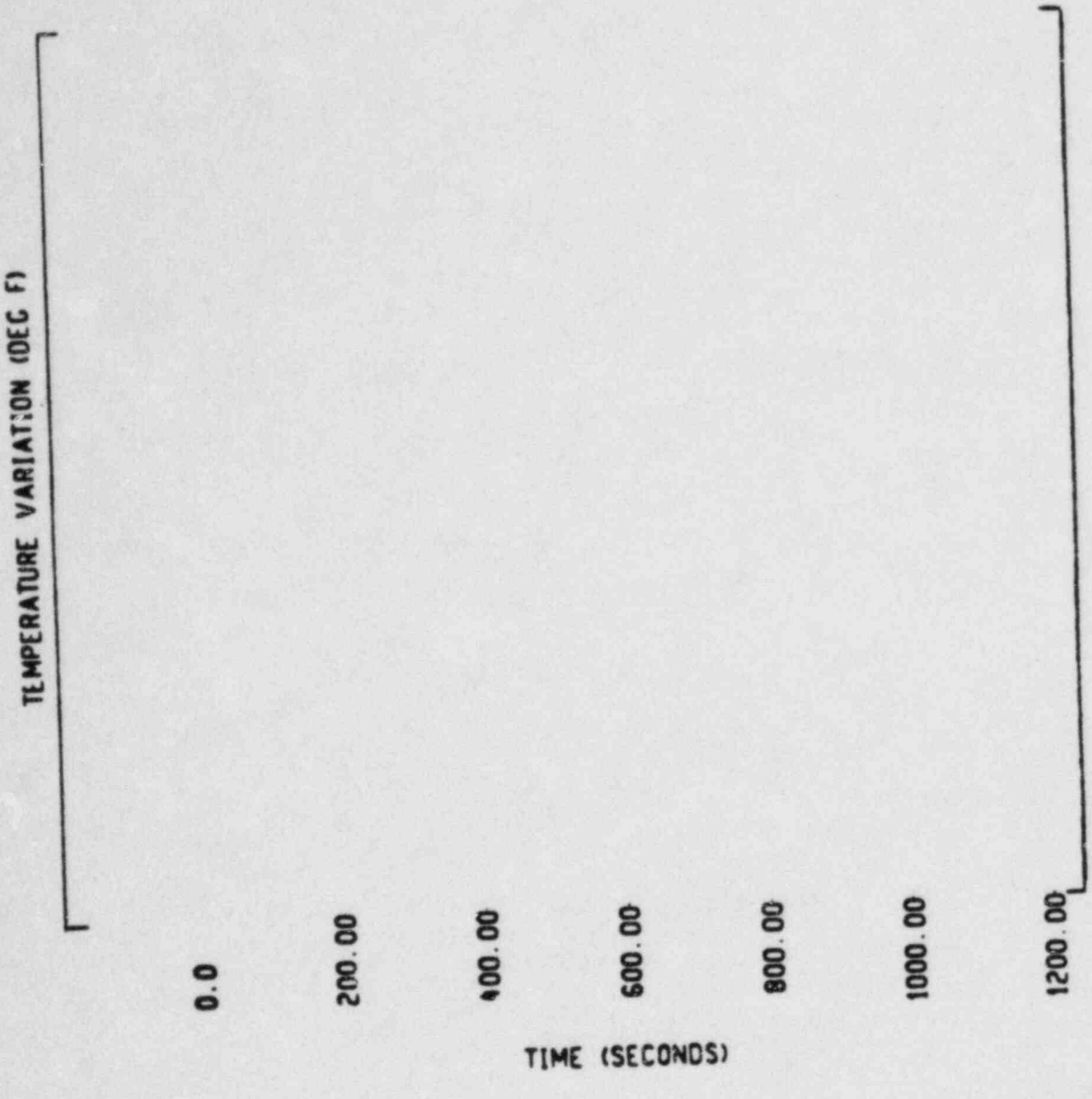

Figure A-14 Step Load Decrease - Cold Leg Temperature Versus Time

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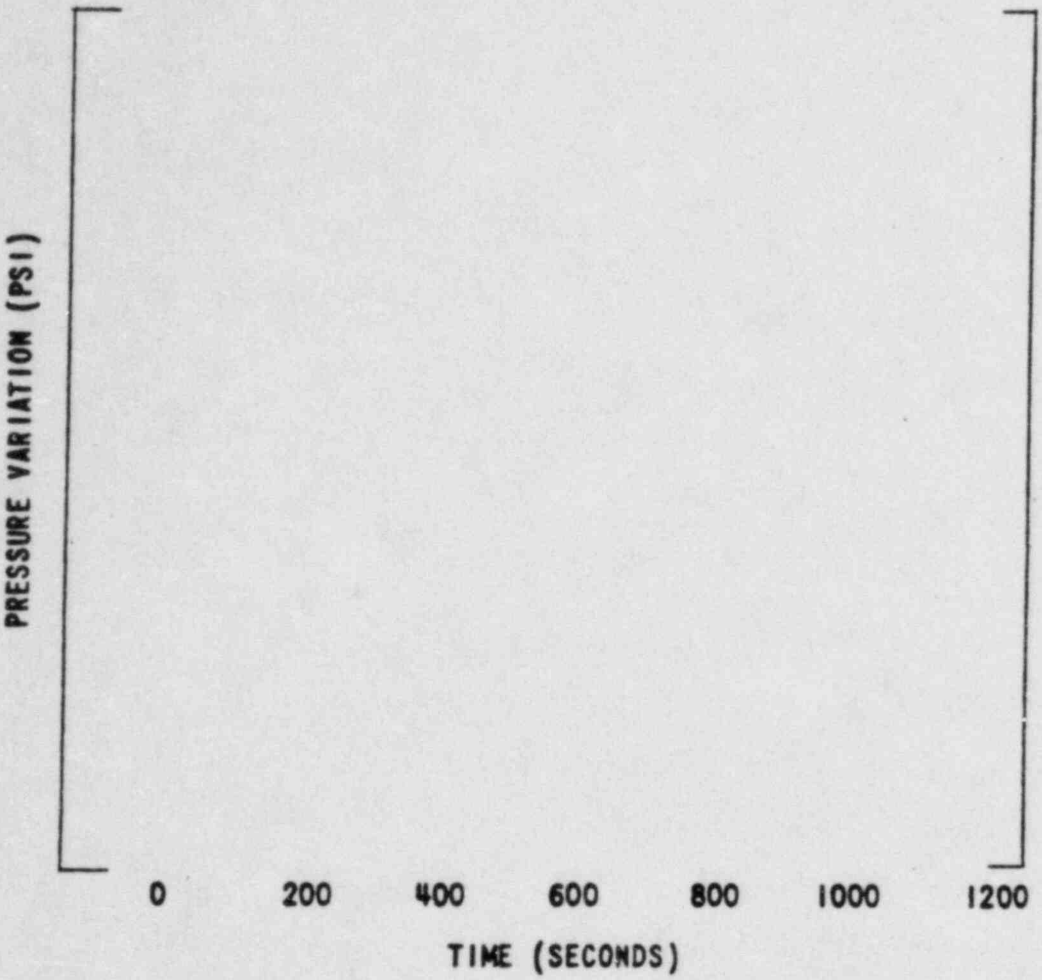

Figure A-15 Large Step Load Decrease with Steam Dump - Reactor Coolant Pressure Versus Time

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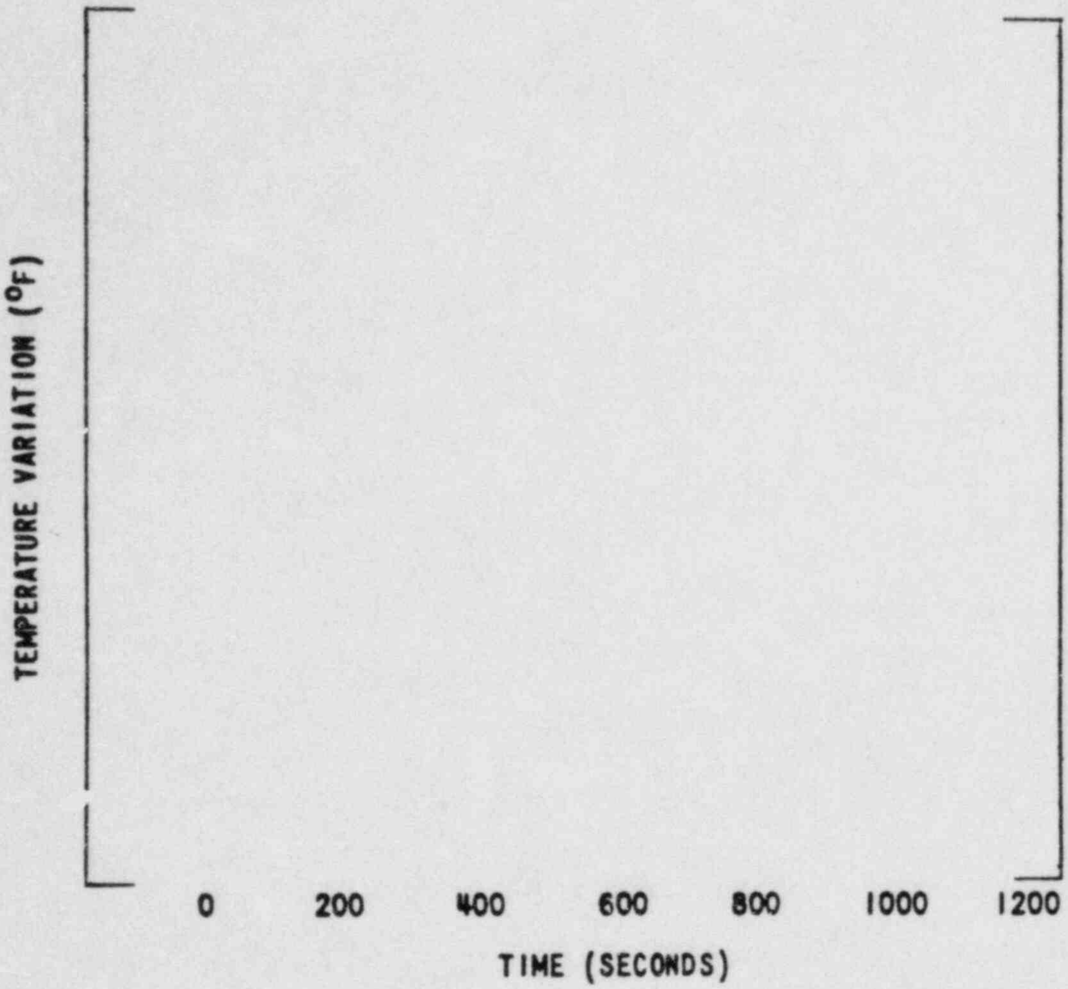

Figure A-16 Large Step Load Decrease with Steam Dump - Cold Leg Temperature Versus Time

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RCS PRESSURE VARIATION

RCS TEMPERATURE VARIATION

Figure A-17 Steady State Fluctuations - Reactor Coolant Pressure and Temperature Versus Time

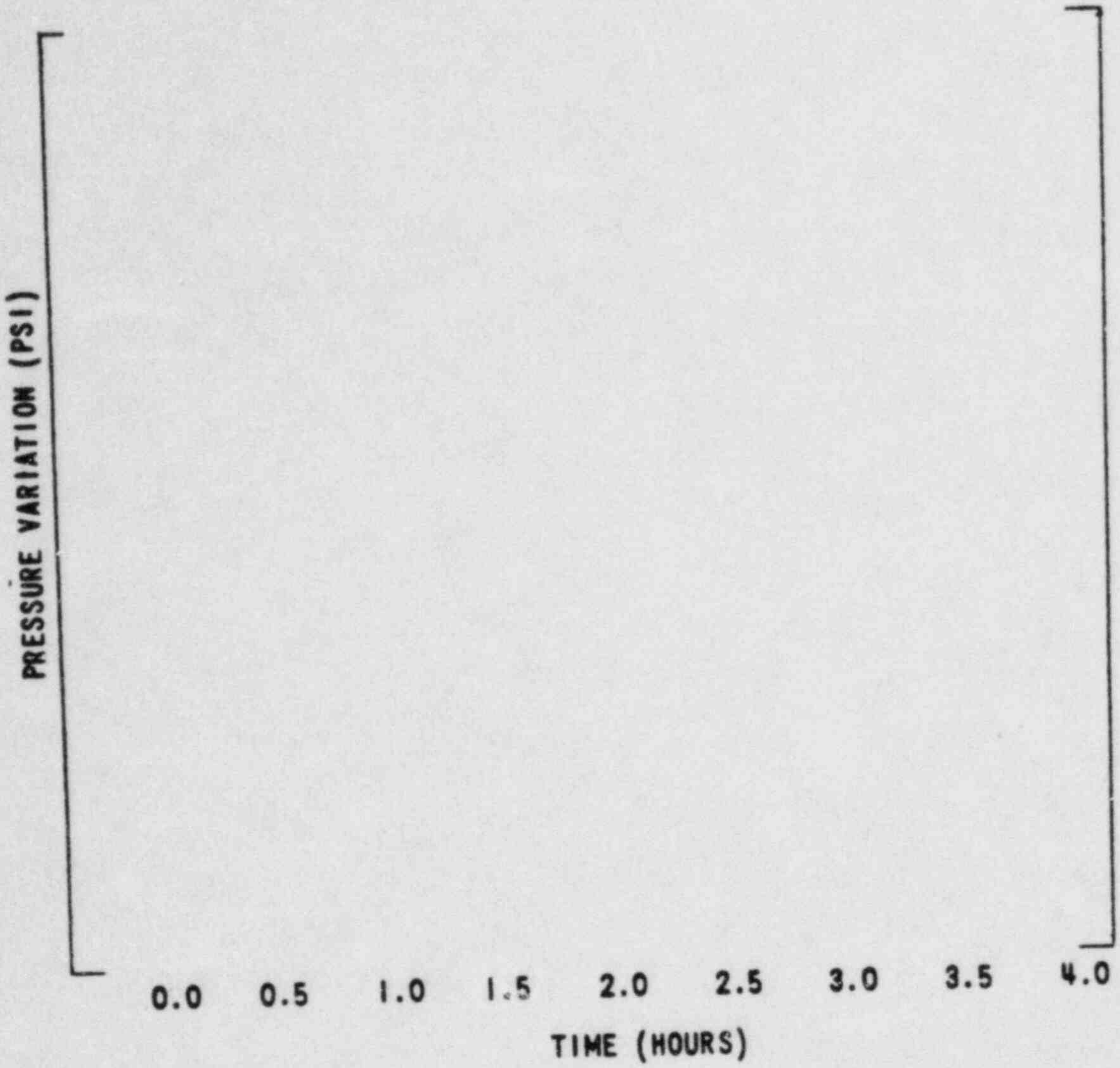

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Figure A-18 Feedwater Cycling - Reactor Coolant Pressure Versus Time

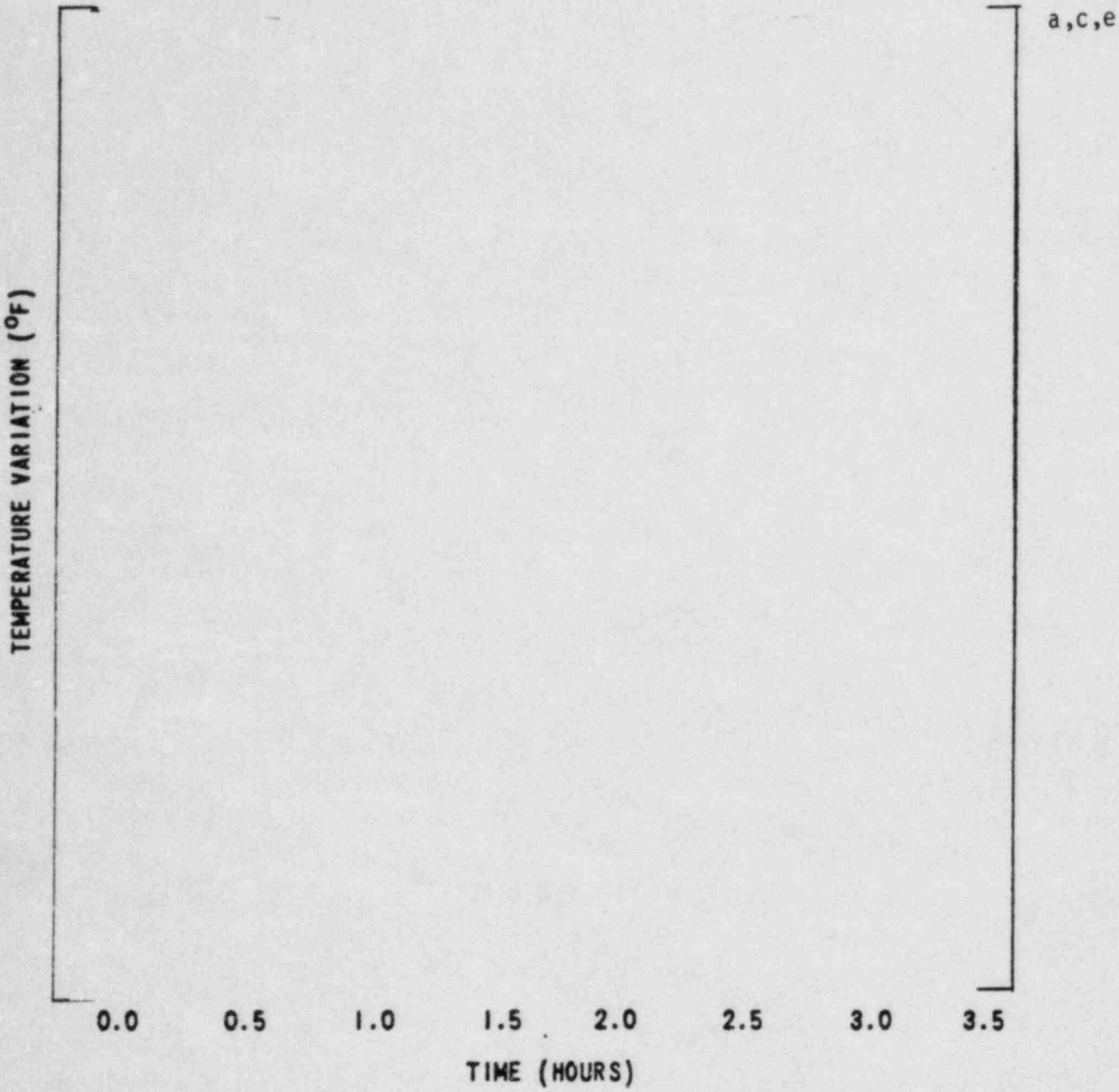

Figure A-19 Feedwater Cycling - Cold Leg Temperature Versus Time

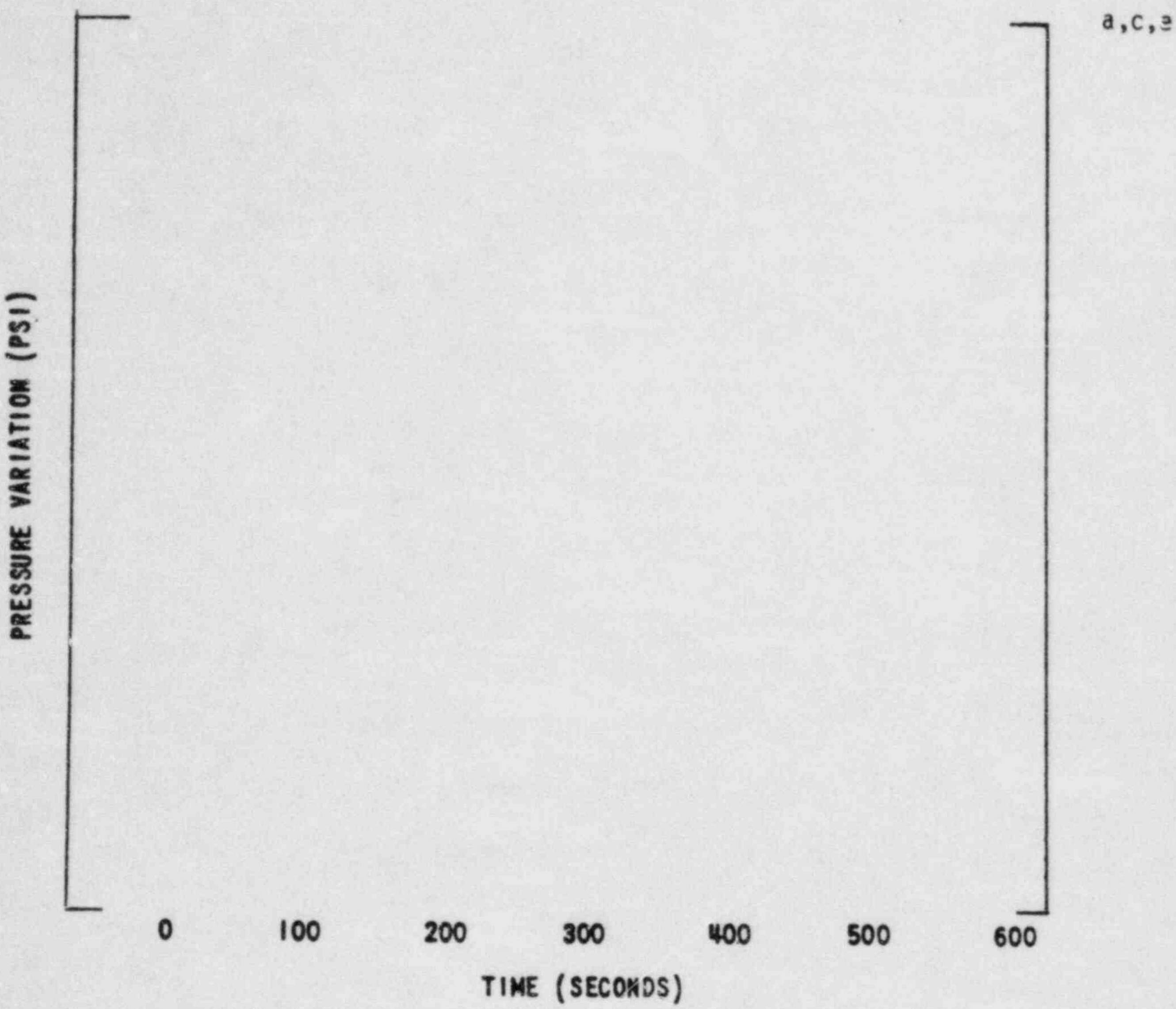

Figure A-20 Loop Out of Service, Normal Loop Shutdown - Reactor Coolant Pressure Versus Time

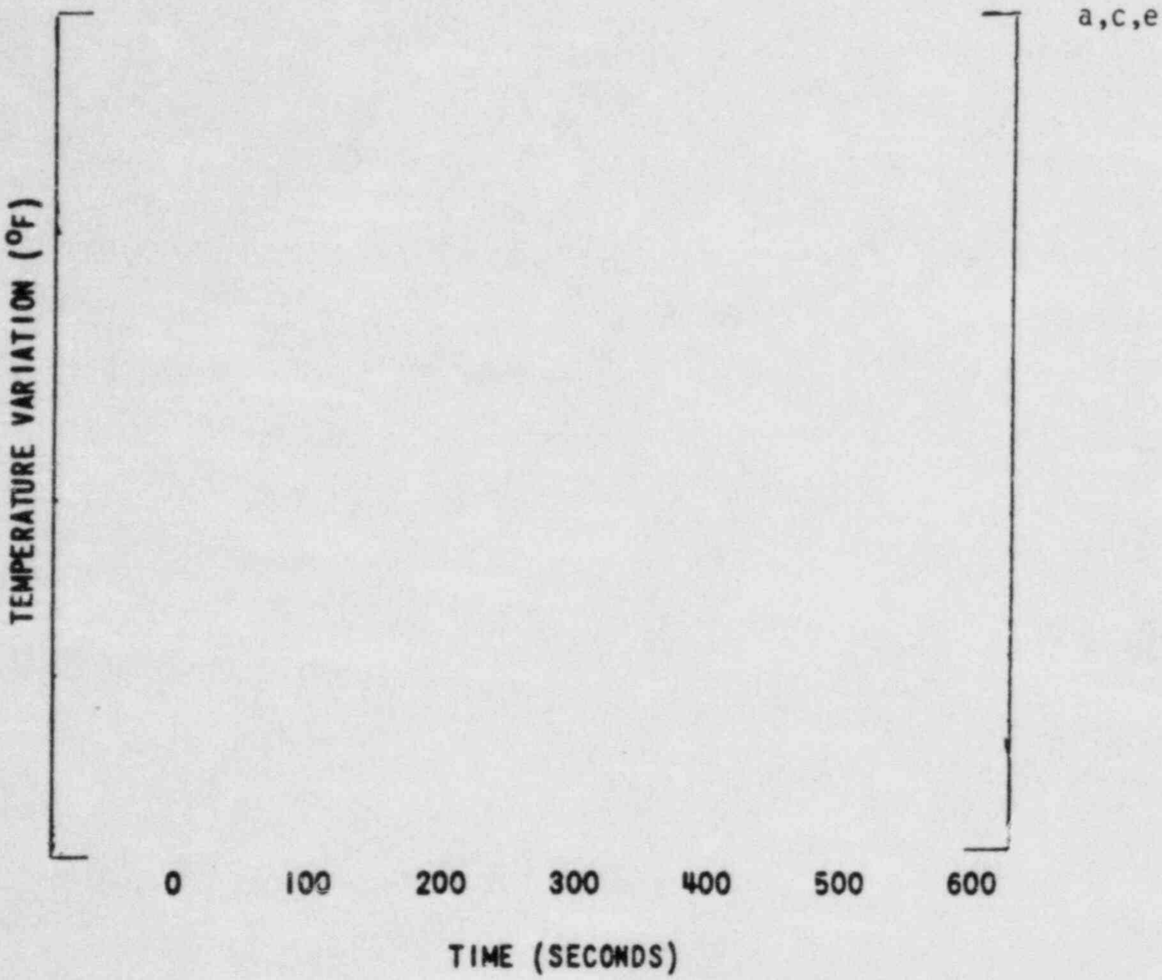

Figure A-21 Loop Out of Service, Normal Loop Shutdown - Cold Leg Temperature Versus Time

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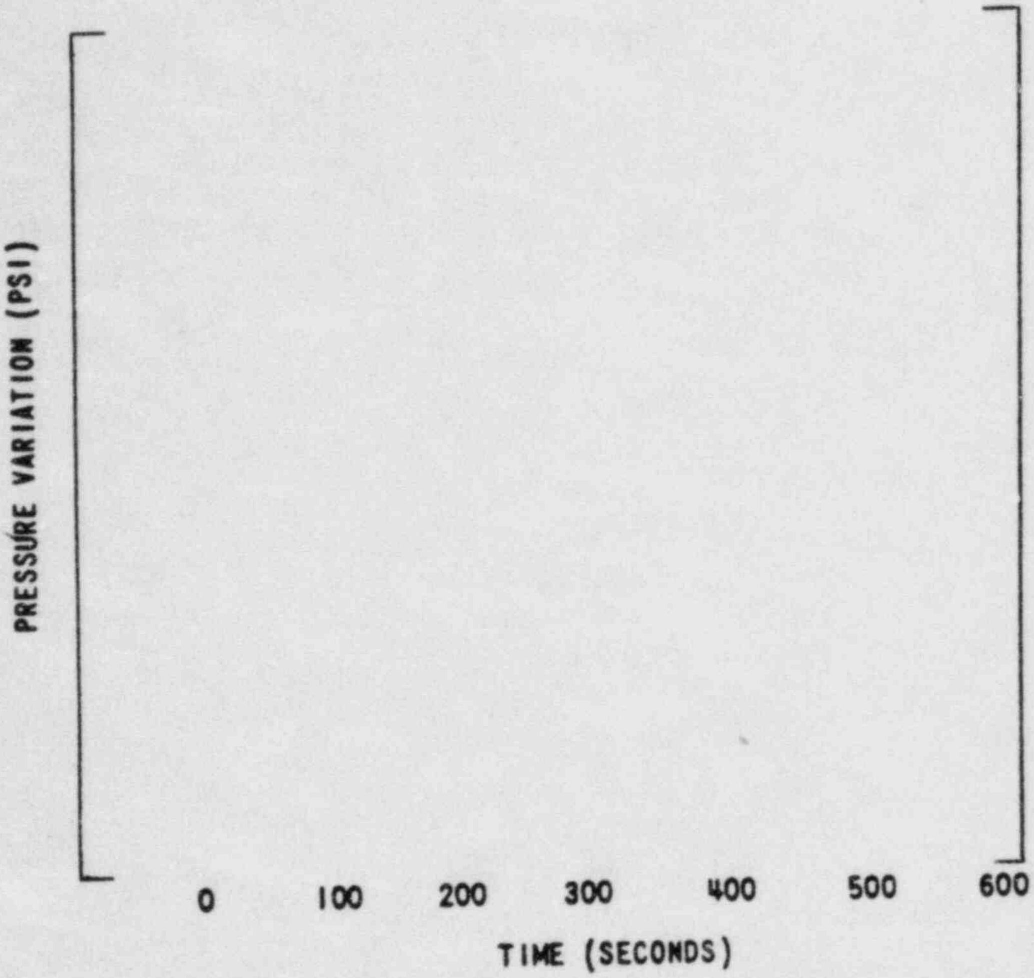

Figure A-22 Loop Out of Service, Normal Loop Startup - Reactor Coolant Pressure Versus Time

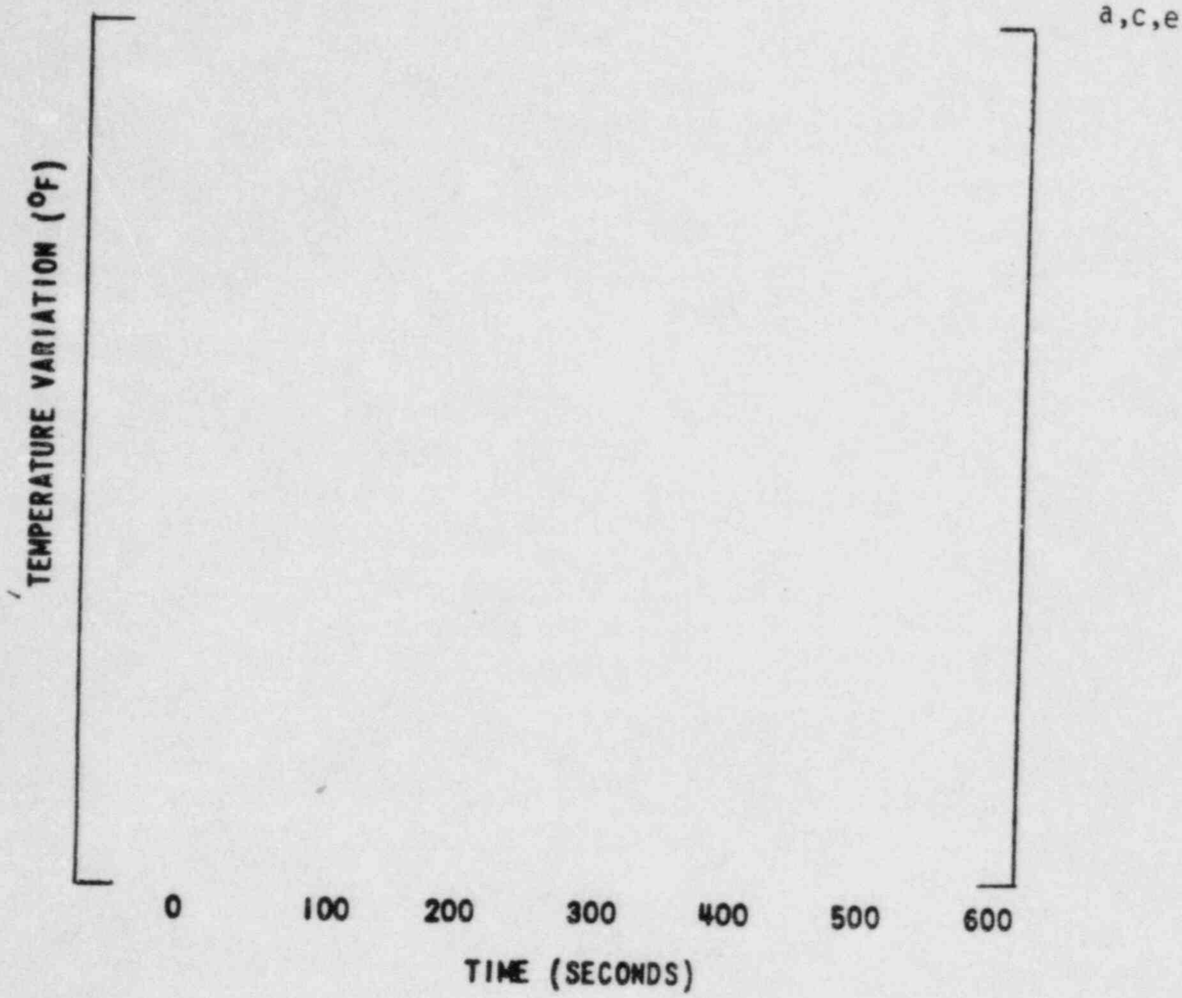

Figure A-23 Loop Out of Service, Normal Loop Startup - Cold Leg Temperature Versus Time

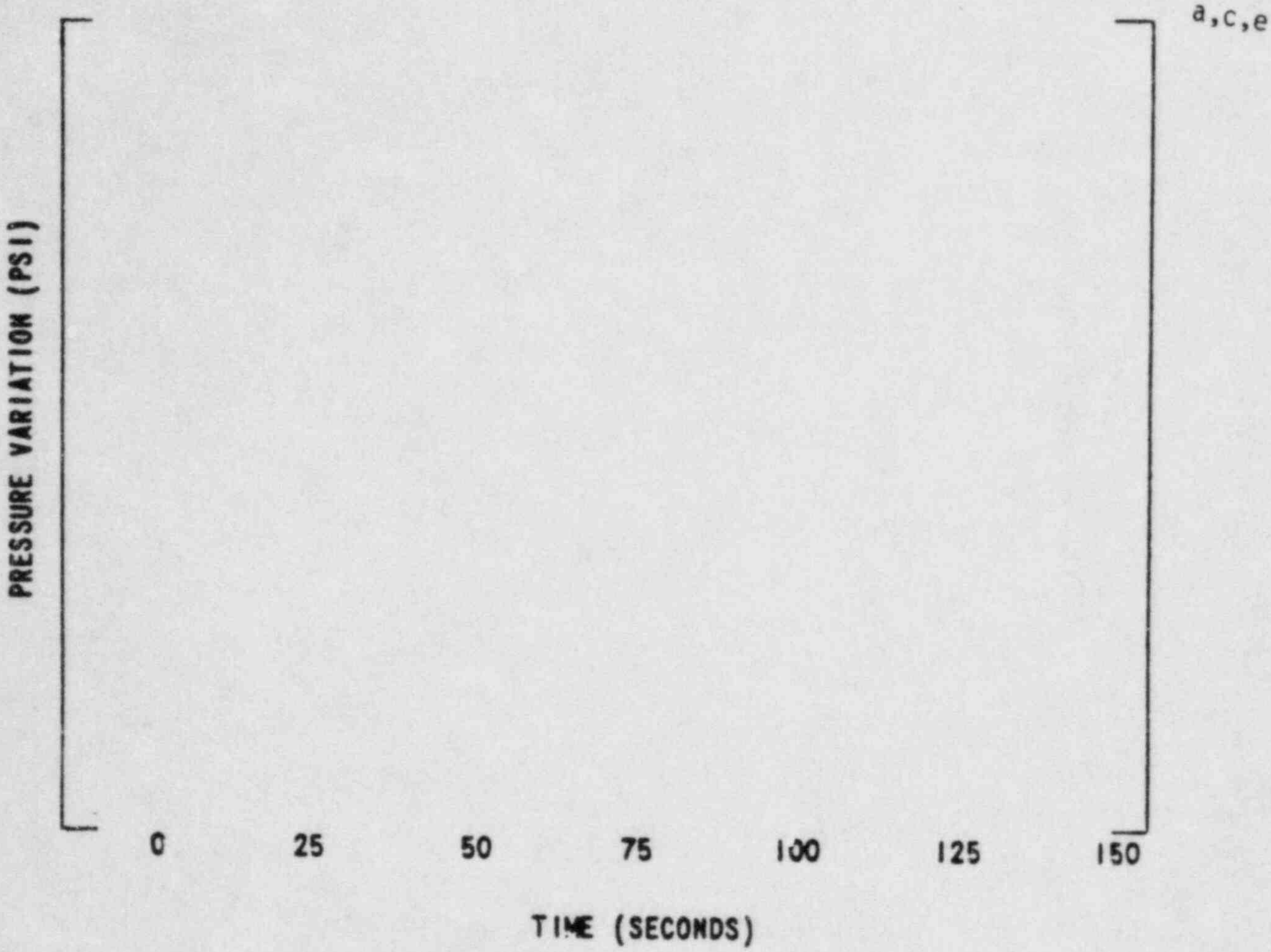

Figure A-24 Loss of Load - Reactor Coolant Pressure Versus Time

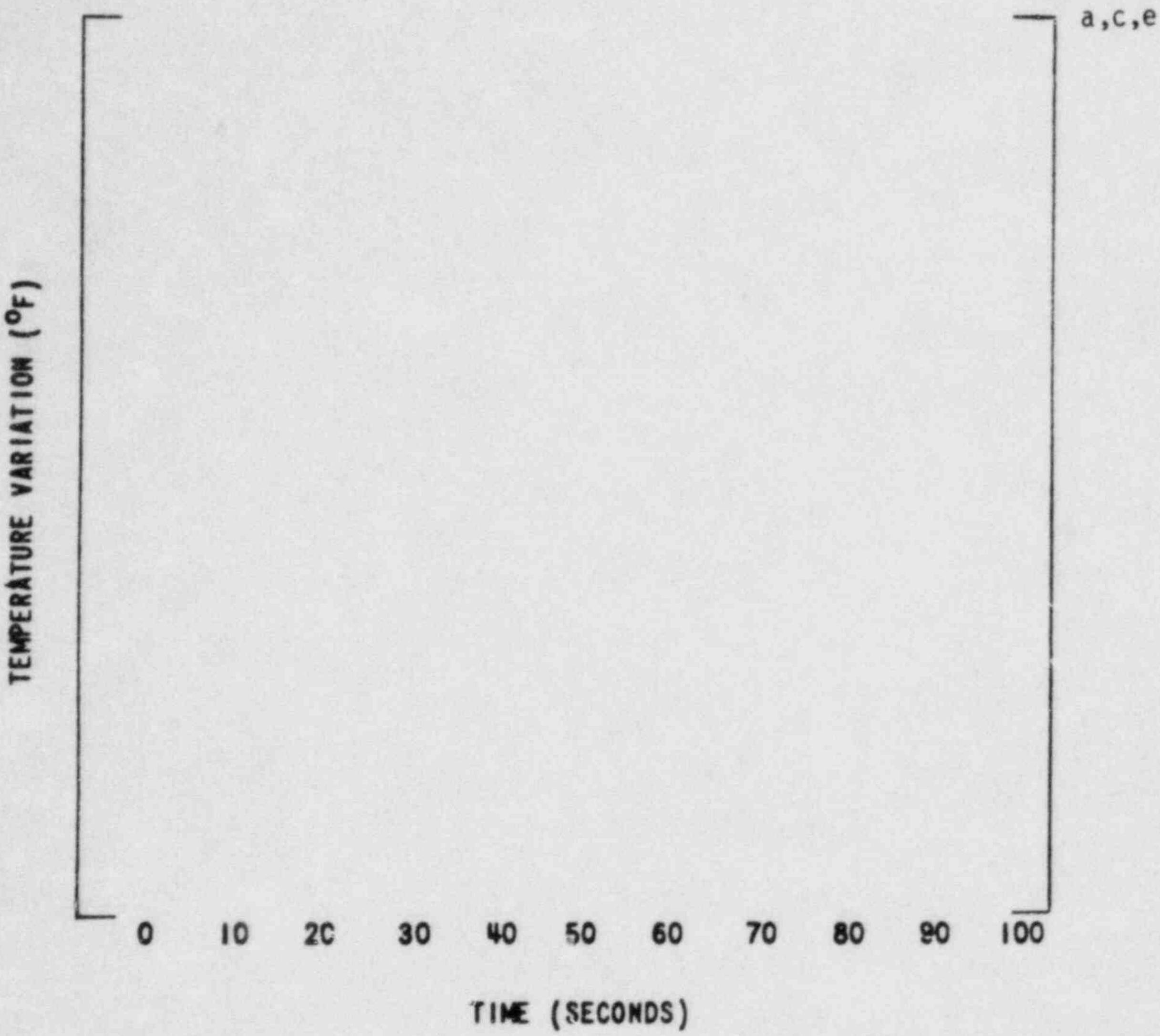

Figure A-25 Loss of Load - Reactor Coolant Temperature Versus Time

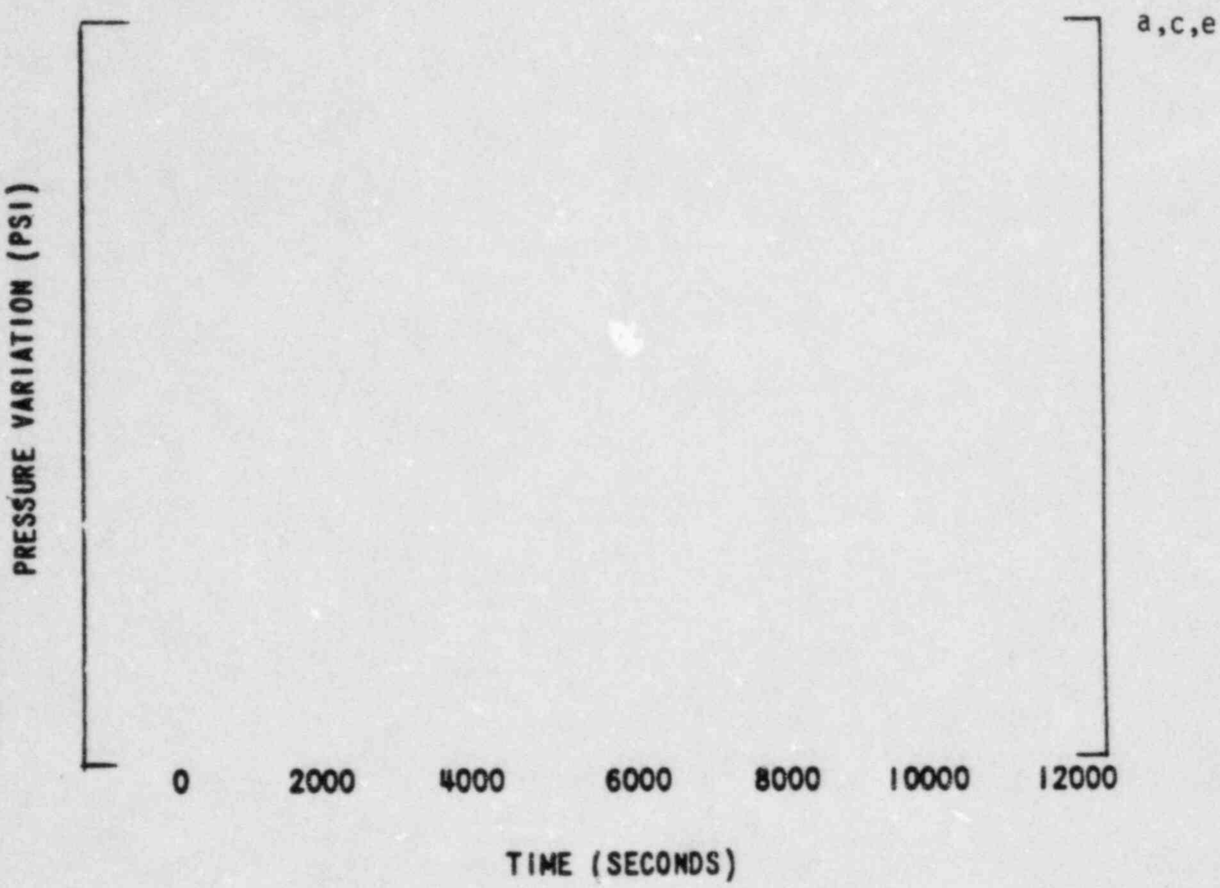

Figure A-26 Loss of Power - Reactor Coolant Pressure Versus Time

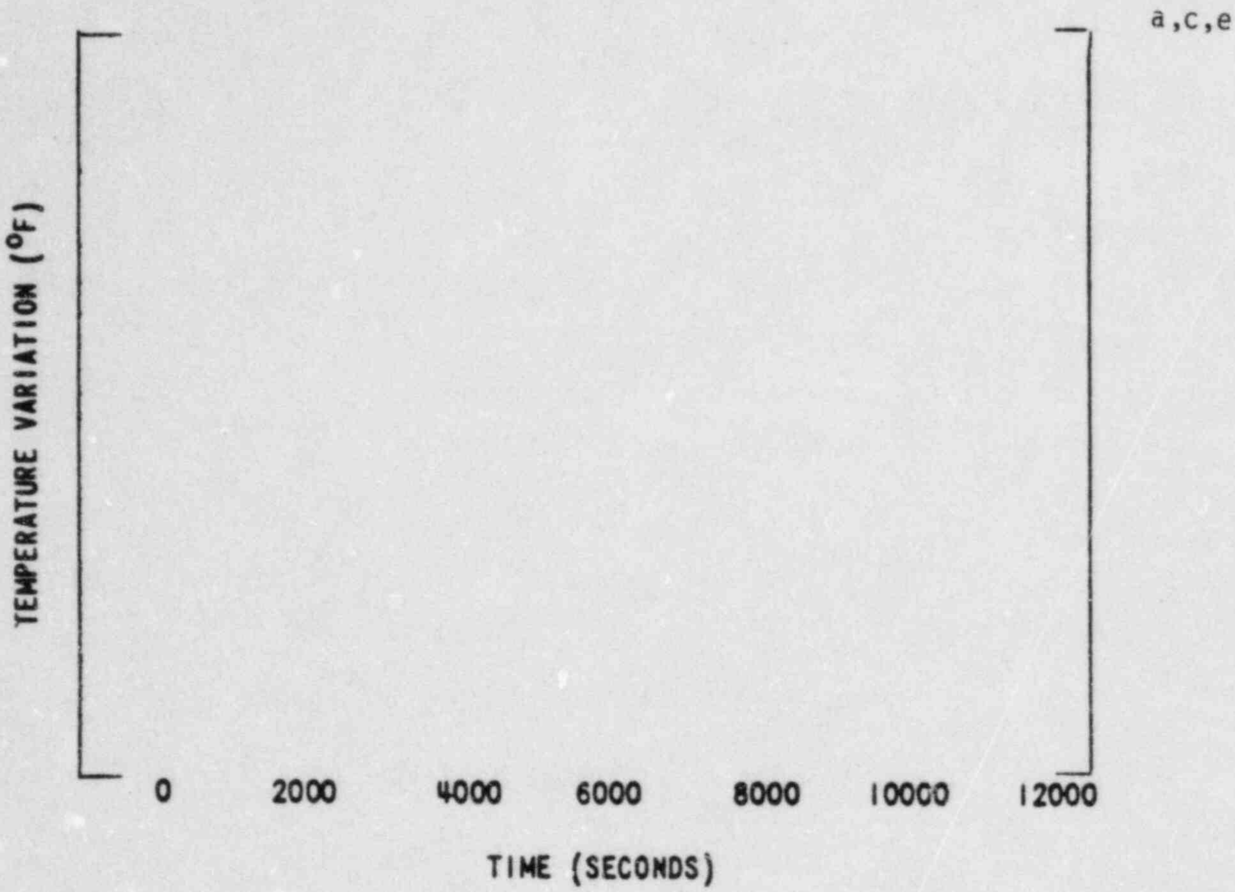

Figure A-27 Loss of Power - Cold Leg Temperature Versus Time

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PRESSURE VARIATION (PSI)

Figure A-28 Partial Loss of Flow - Reactor Coolant Pressure Versus Time

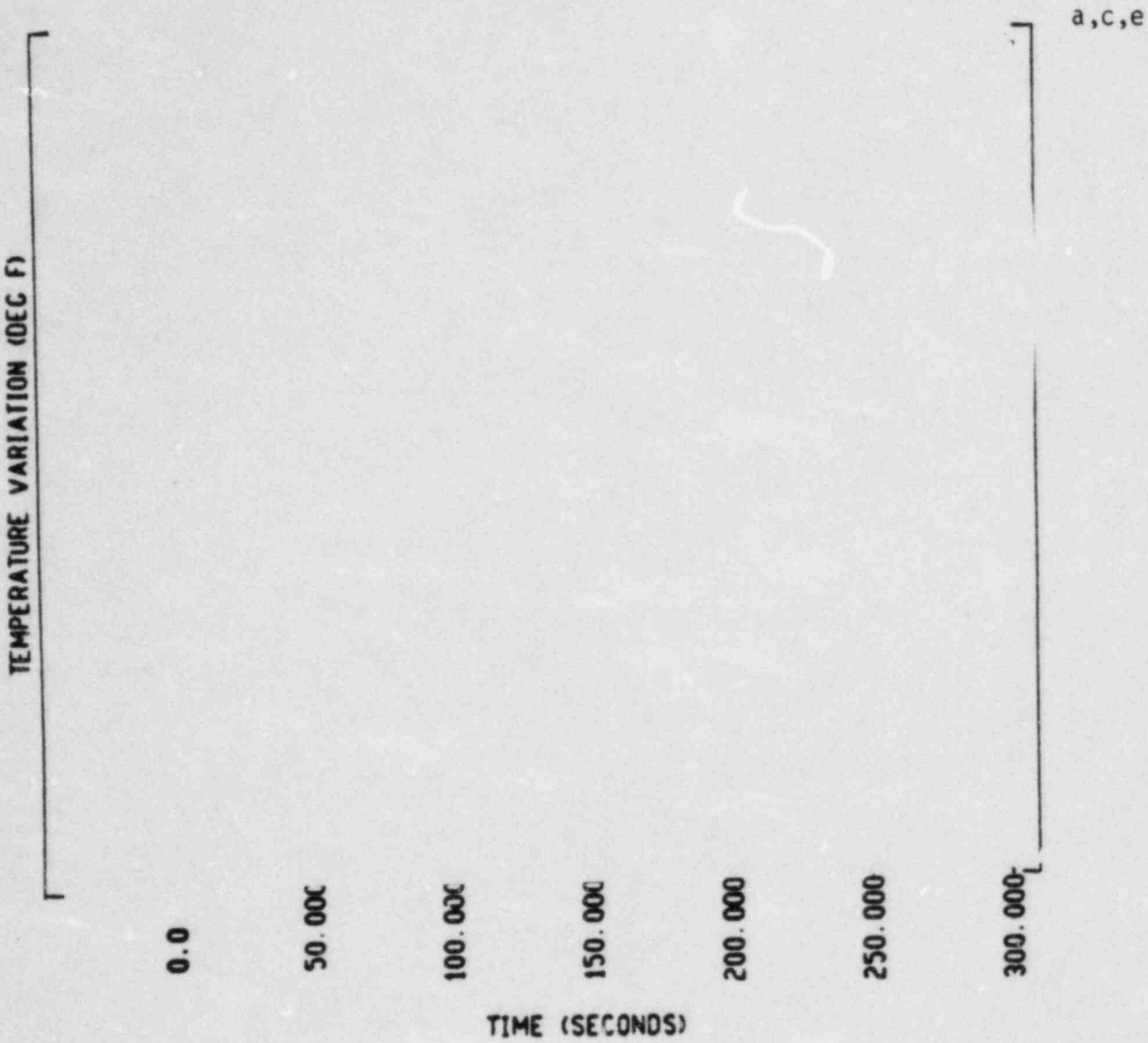
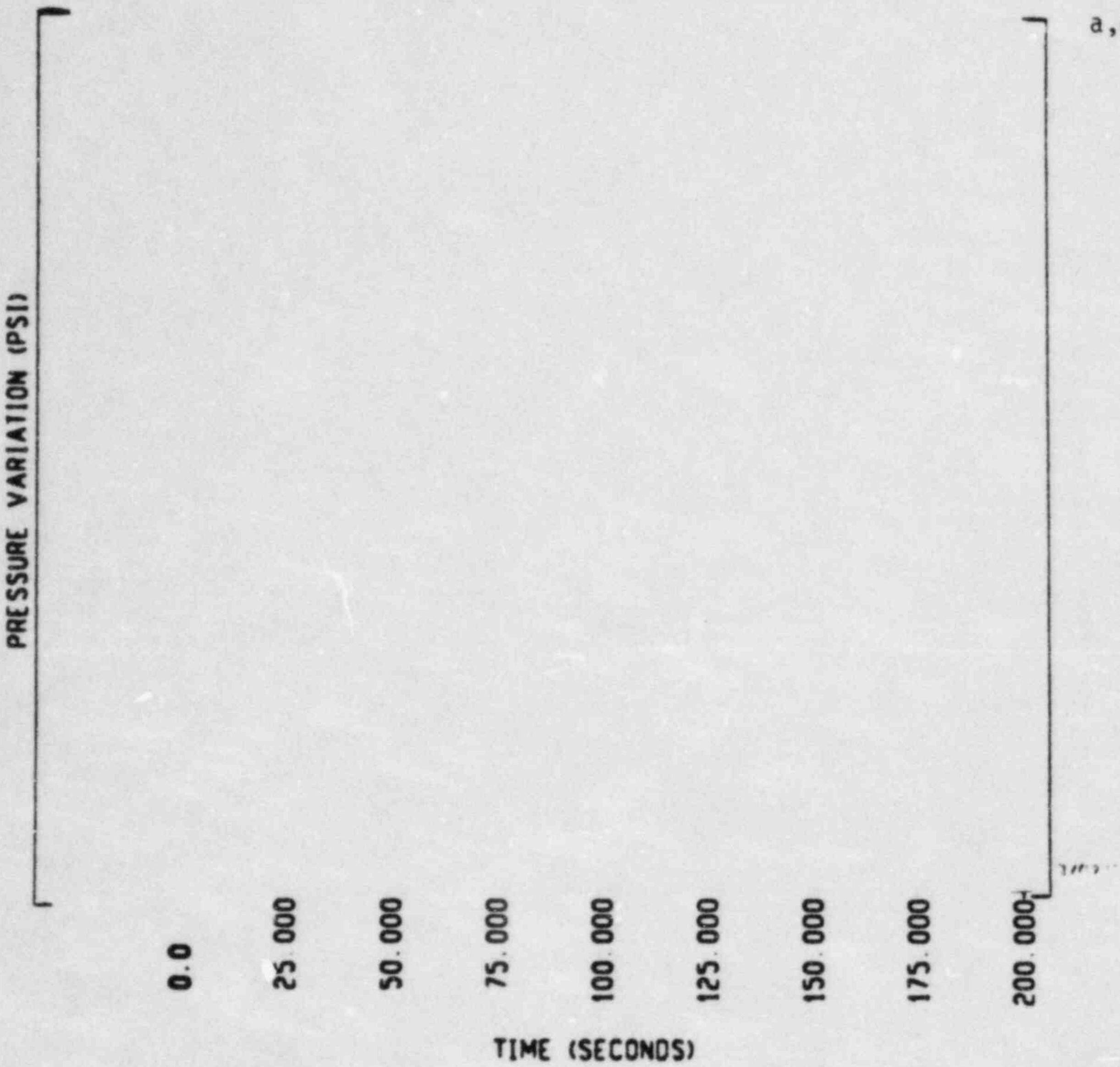


Figure A-29 Partial Loss of Flow - Cold Leg Temperature Versus Time

a,c,e



NOTE: PRESSURE IS RETURNED TO THE INITIAL VALUE AFTER 9 MINUTES, AT A RATE CONSISTENT WITH NORMAL PRESSURIZER HEATUP.

Figure A-30 Reactor Trip with No Cooldown - Reactor Coolant Pressure Versus Time

a,c,e

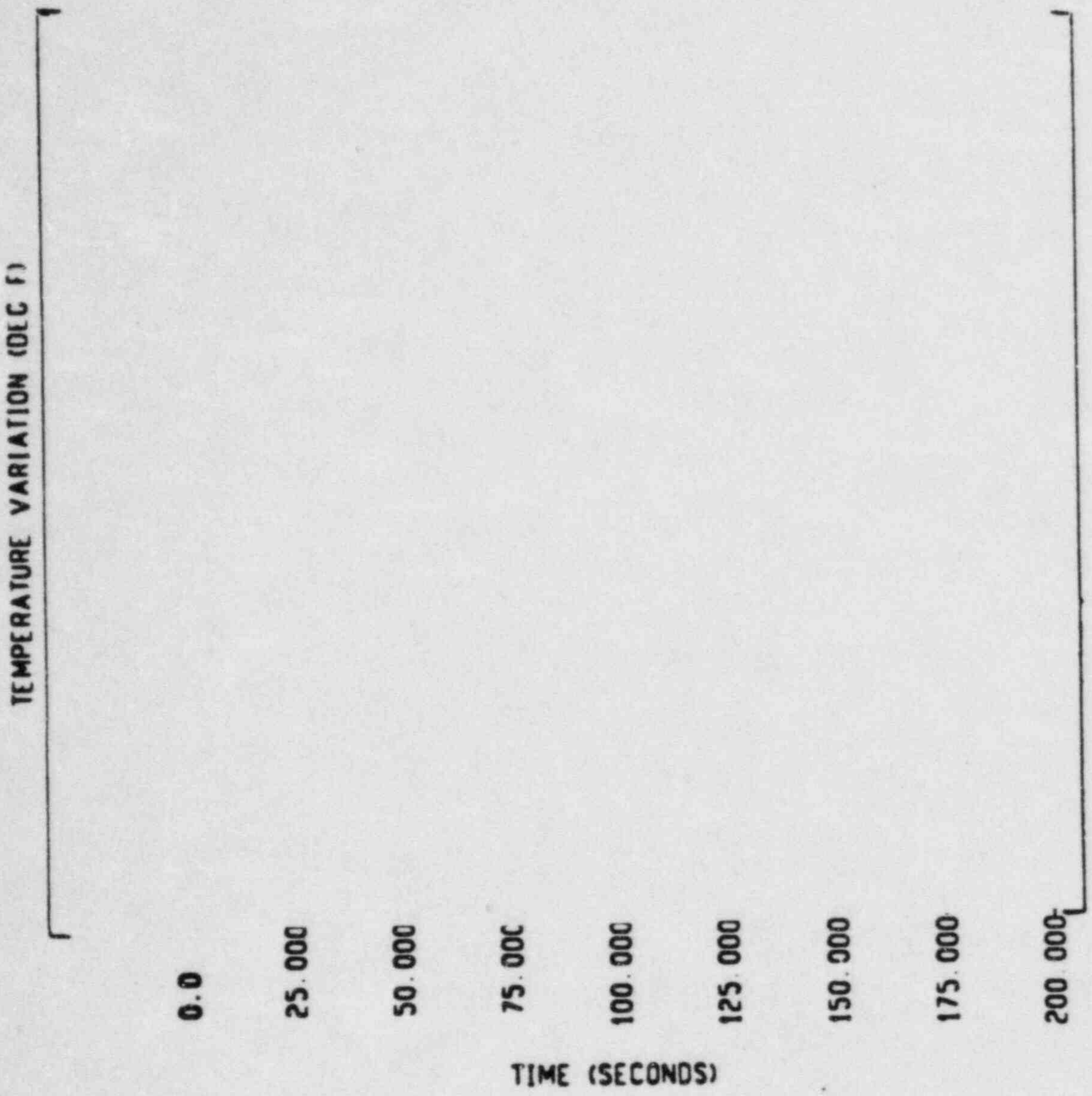


Figure A-31 Reactor Trip with No Cooldown - Cold Leg Temperature Versus Time

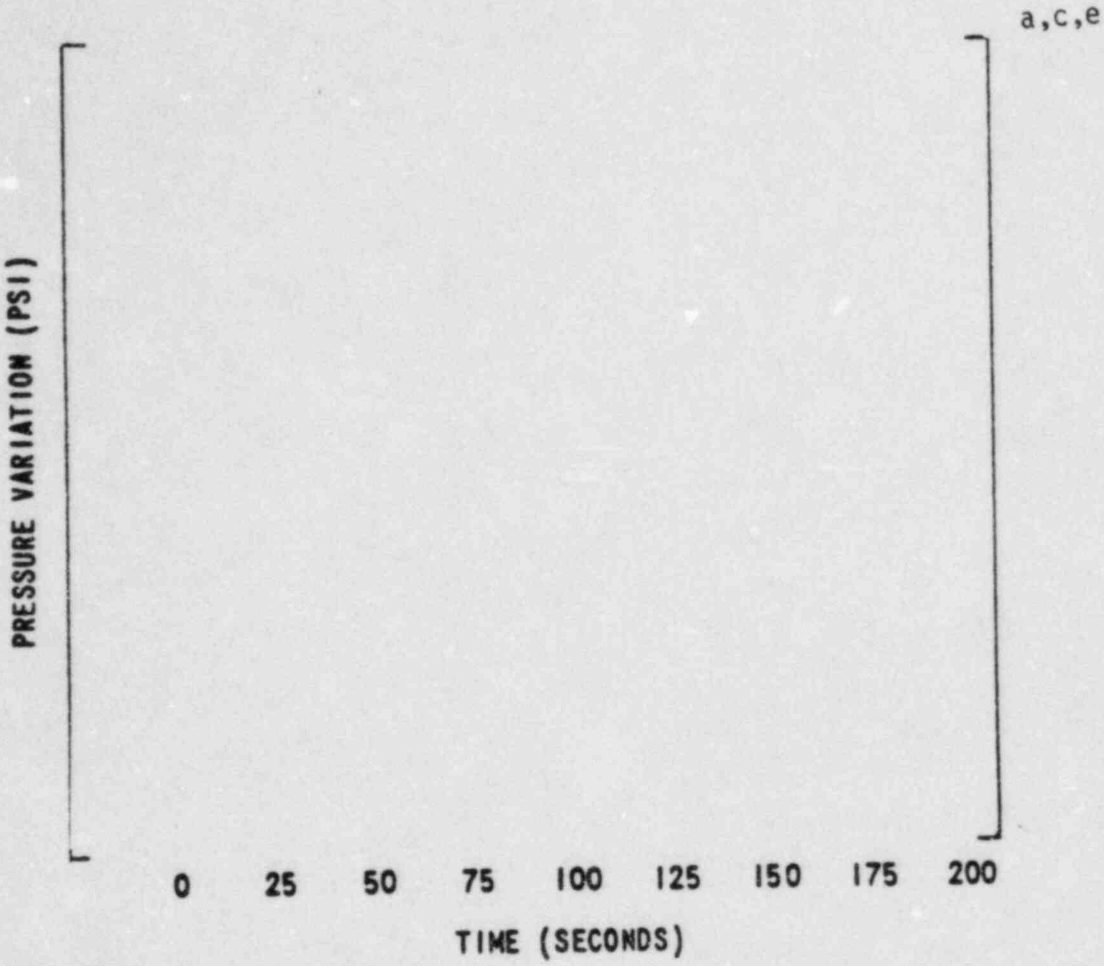


Figure A-32 Reactor Trip with Cooldown, No SI - Reactor Coolant Pressure Versus Time

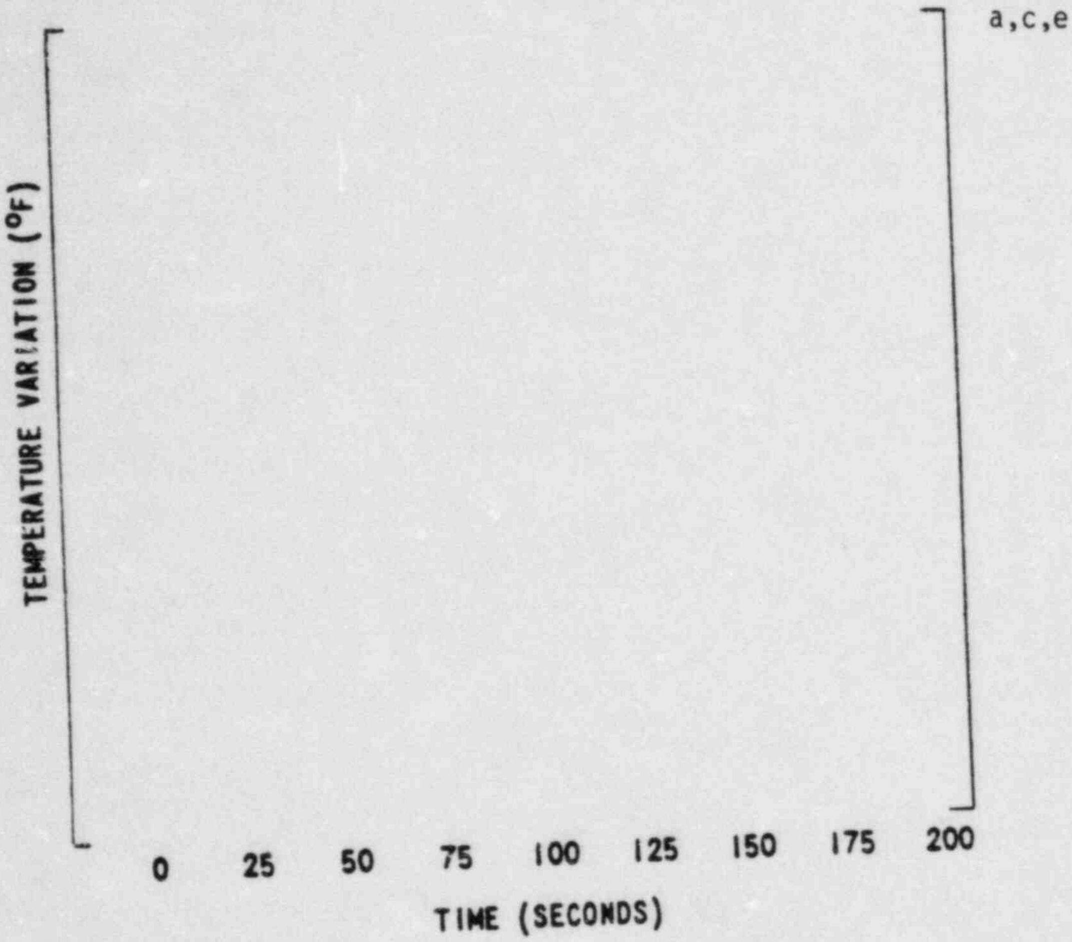


Figure A-33 Reactor Trip with Cooldown, No. SI - Cold Leg Temperature Versus Time

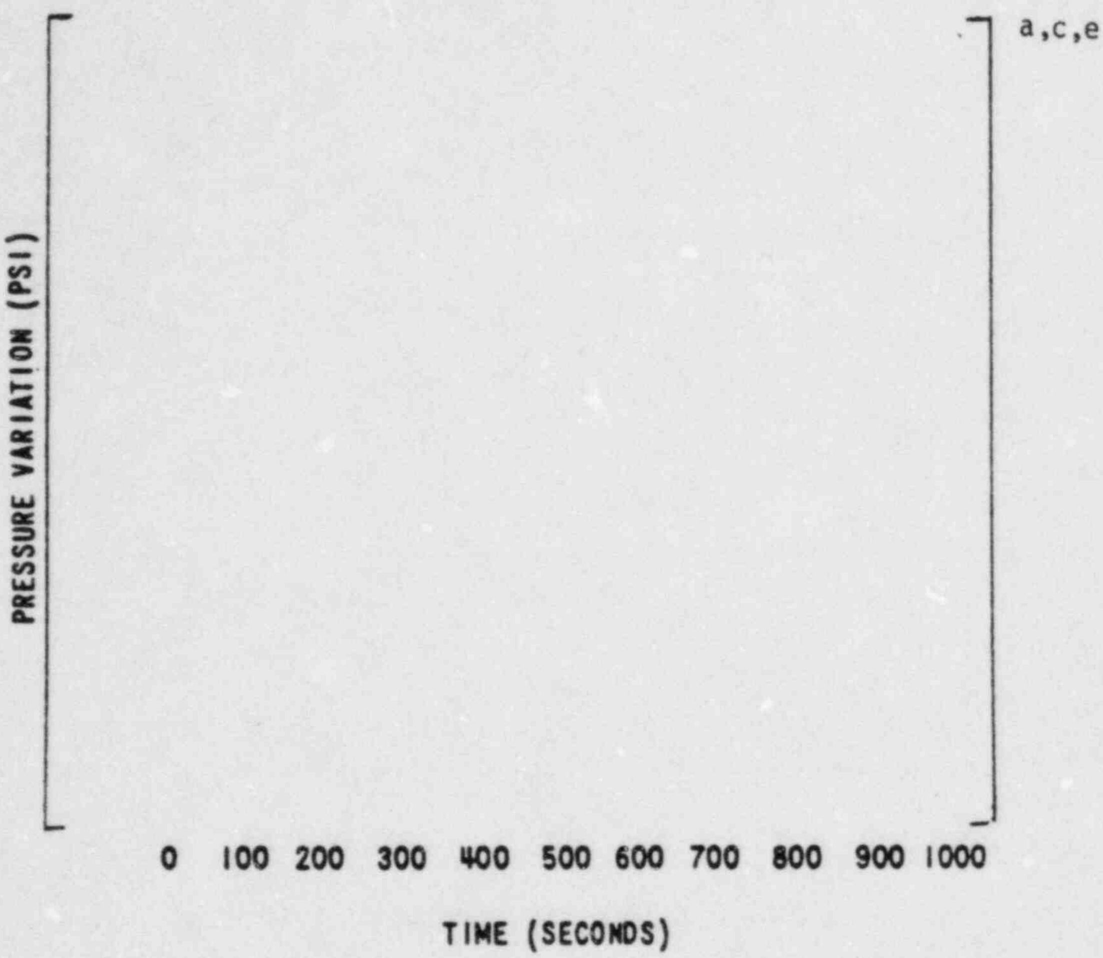


Figure A-34 Reactor Trip with Cooldown and SI - Reactor Coolant Pressure Versus Time

a,c,e

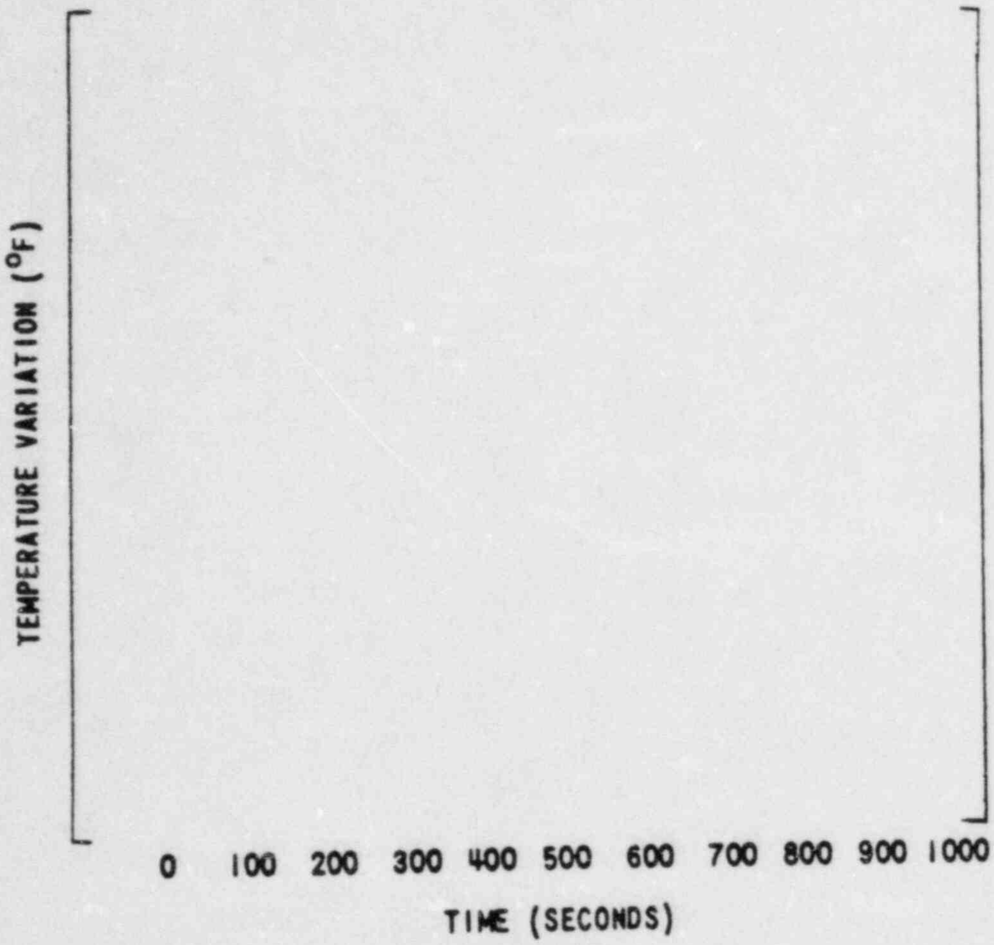


Figure A-35 Reactor Trip with Cooldown and SI - Cold Leg Temperature Versus Time

a,c,e

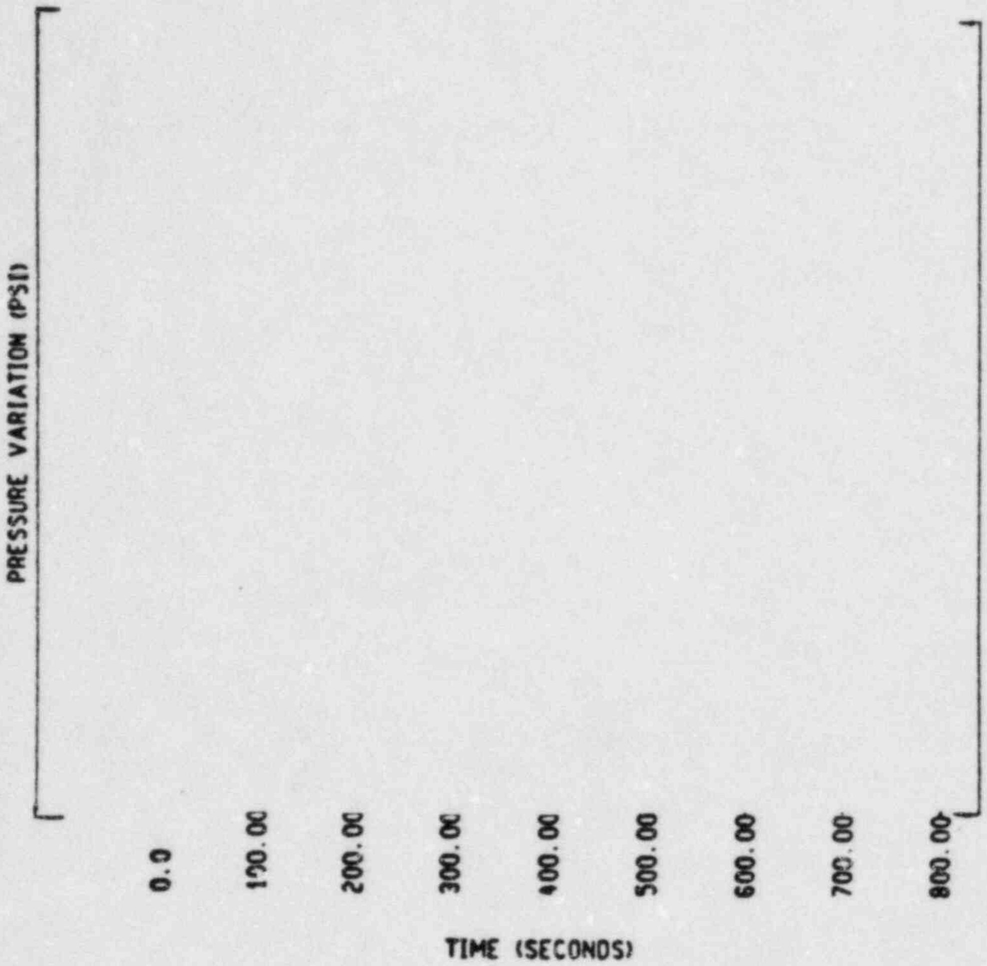
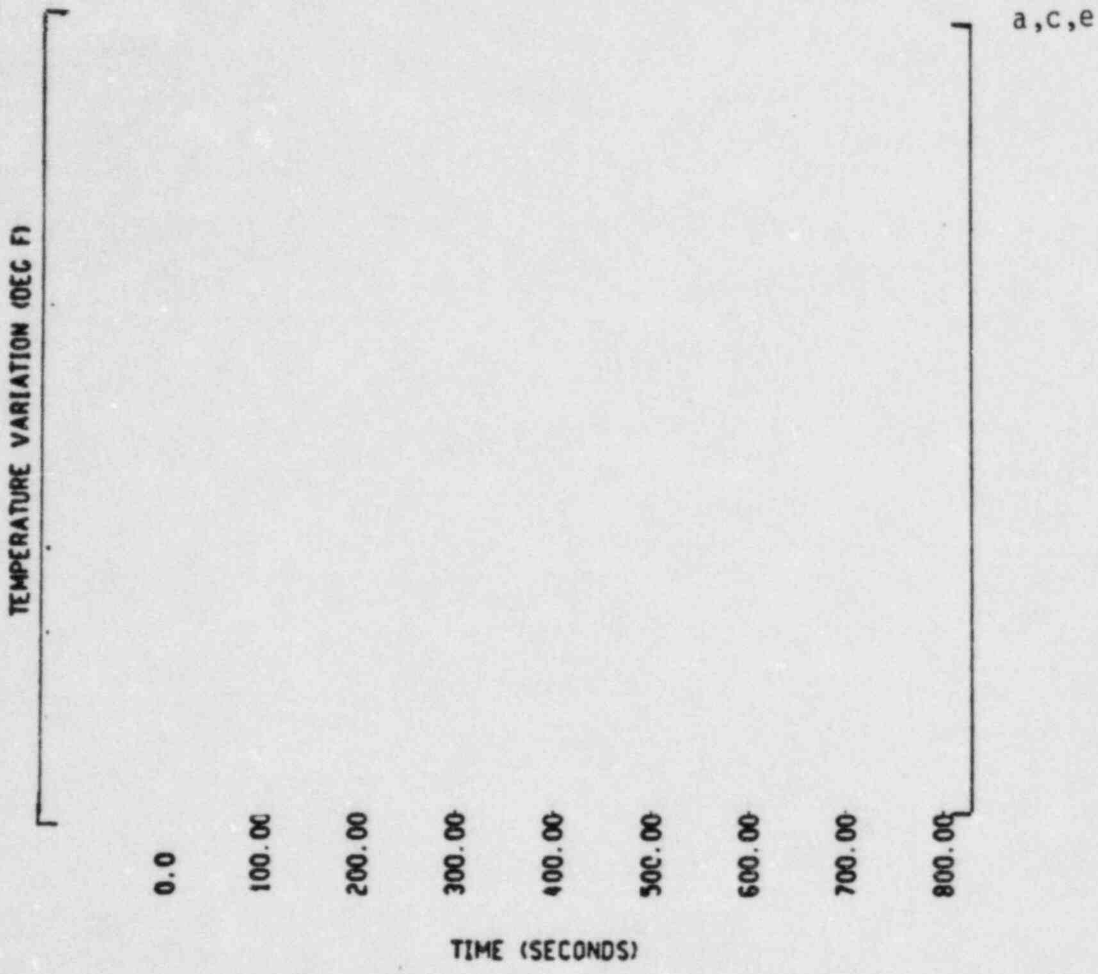


Figure A-36 Inadvertent RCS Depressurization - Reactor Coolant Pressure Versus Time



NOTE: TEMPERATURE GOES TO COLD SHUTDOWN VALUE CONSISTENT WITH NORMAL COOLDOWN RATE.

Figure A-37 Inadvertent RCS Depressurization - Cold Leg Temperature Versus Time

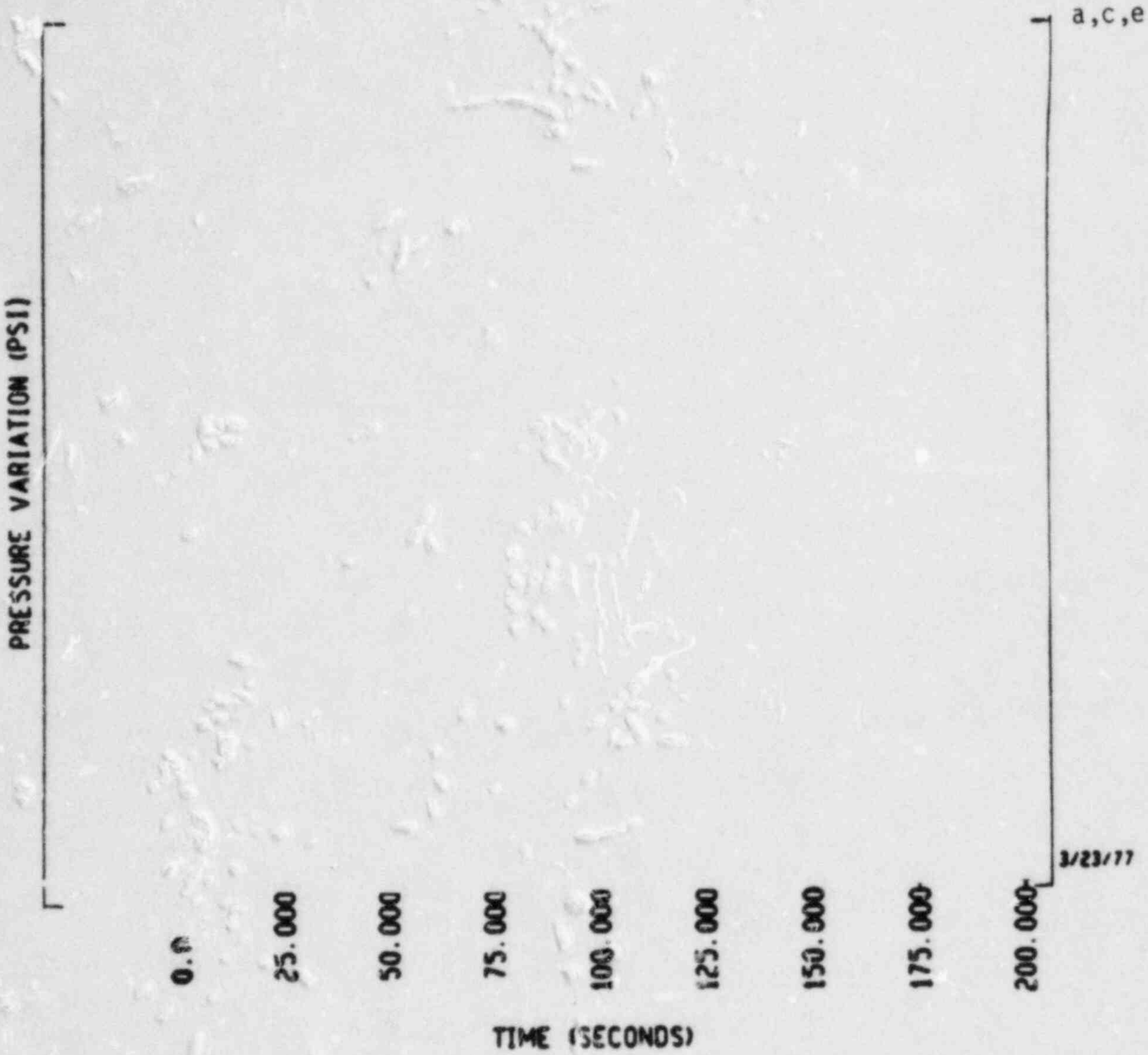


Figure A-38 Inadvertent Startup of an Inactive Loop - Reactor Coolant Pressure Versus Time

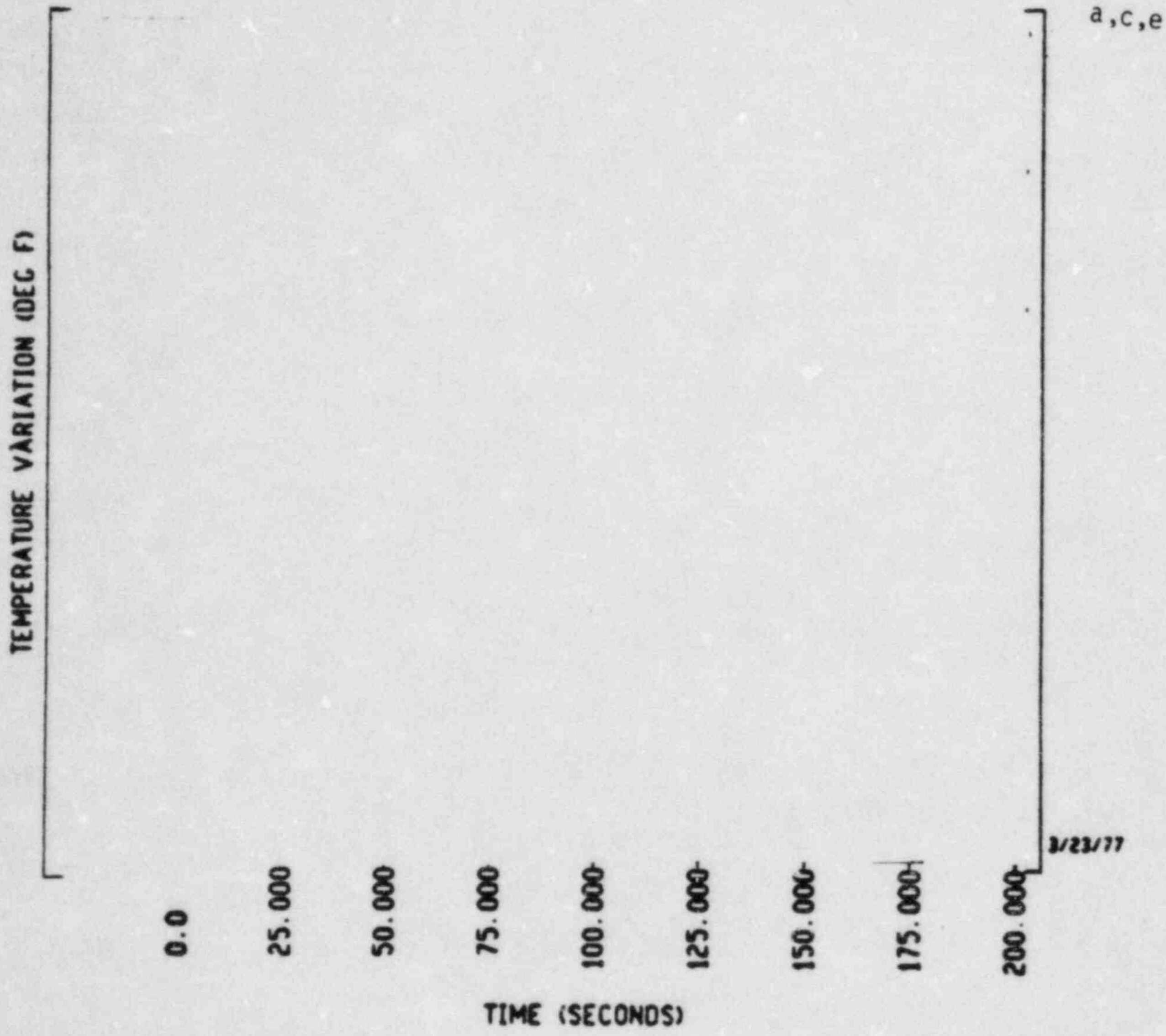


Figure A-39 Inadvertent Startup of an Inactive Loop - Cold Leg Temperature Versus Time

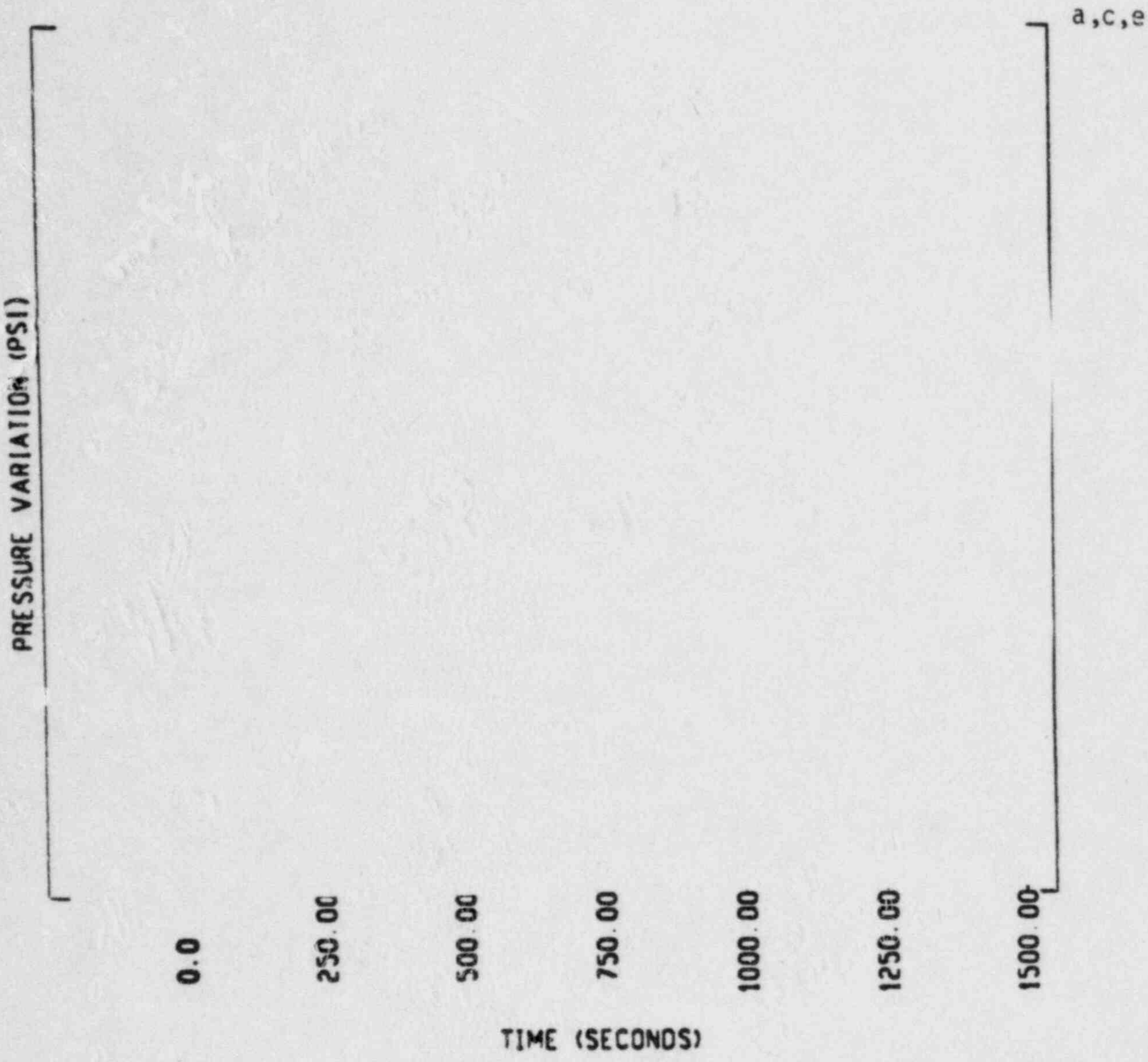


Figure A-40 Inadvertent SI Actuation - Reactor Coolant Pressure Versus Time

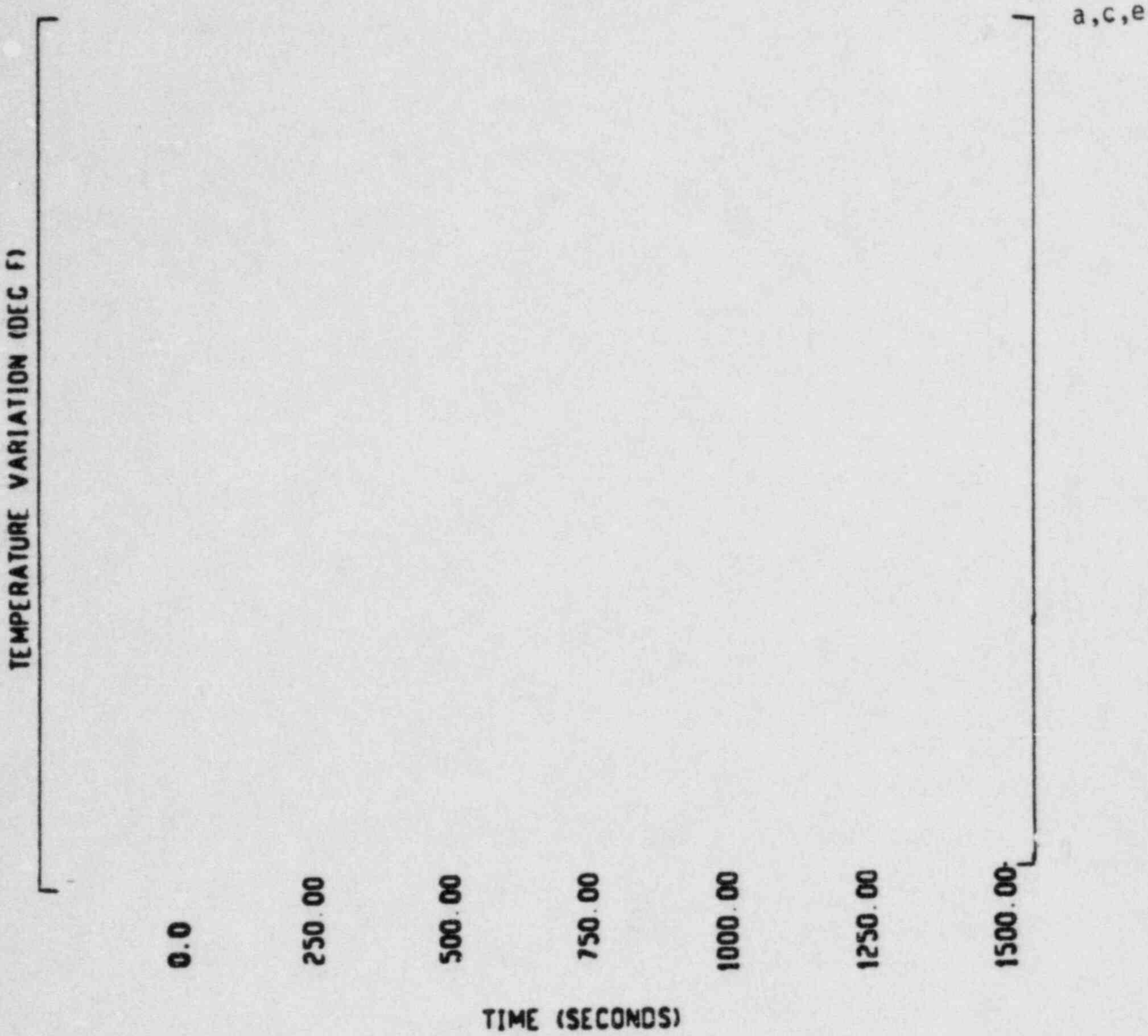
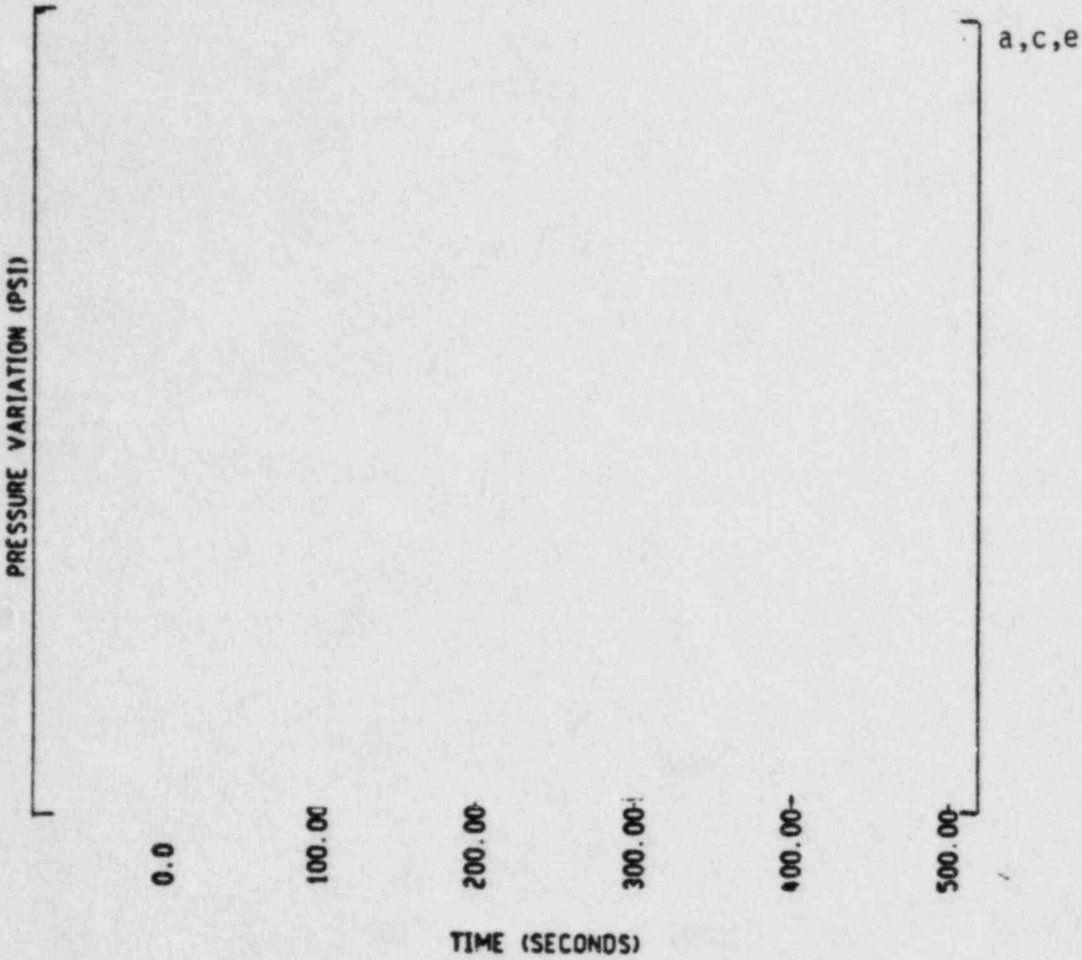


Figure A-41 Inadvertent SI Actuation - Cold Leg Temperature Versus Time



NOTE: PRESSURE RETURNS TO THE INITIAL VALUE AT A RATE CONSISTENT WITH NORMAL PRESSURIZER HEATUP.

Figure A-42 Control Rod Drop - Reactor Coolant Pressure Versus Time

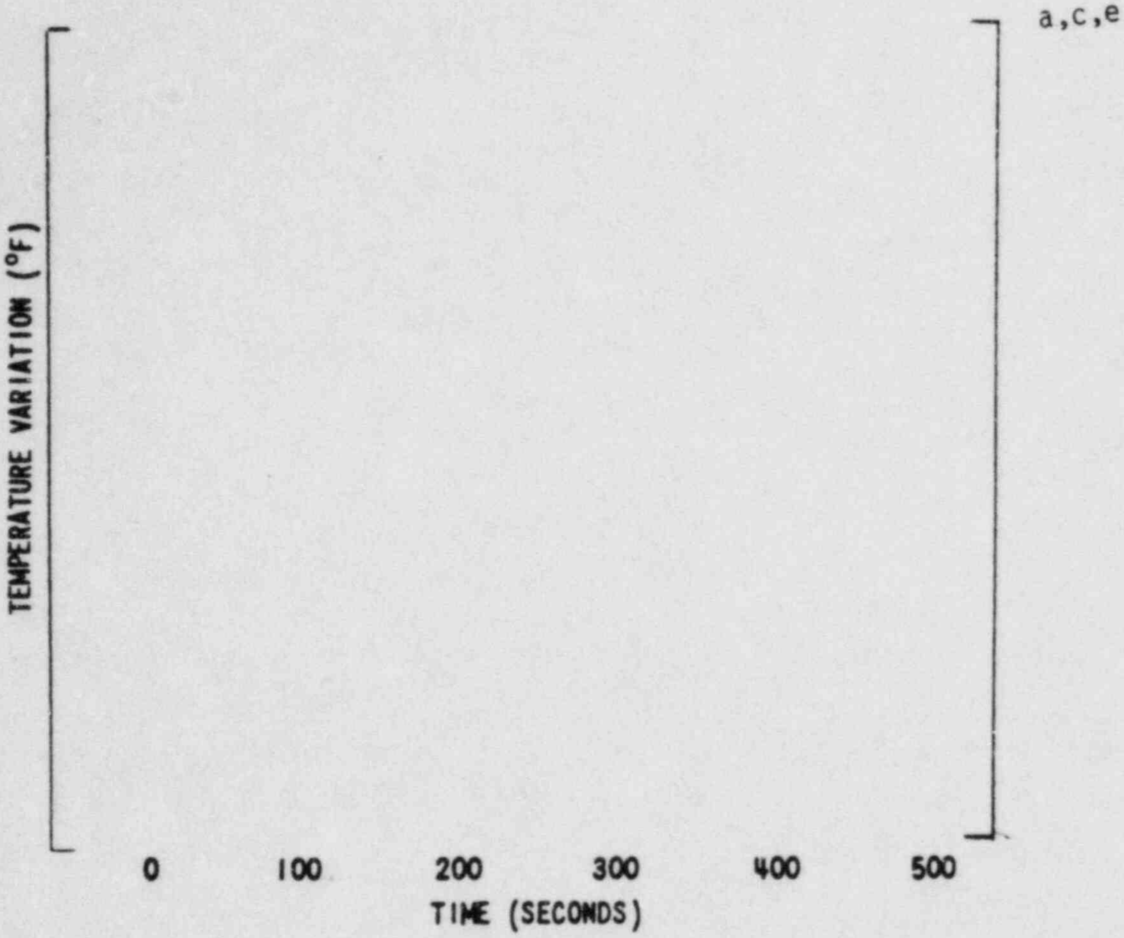
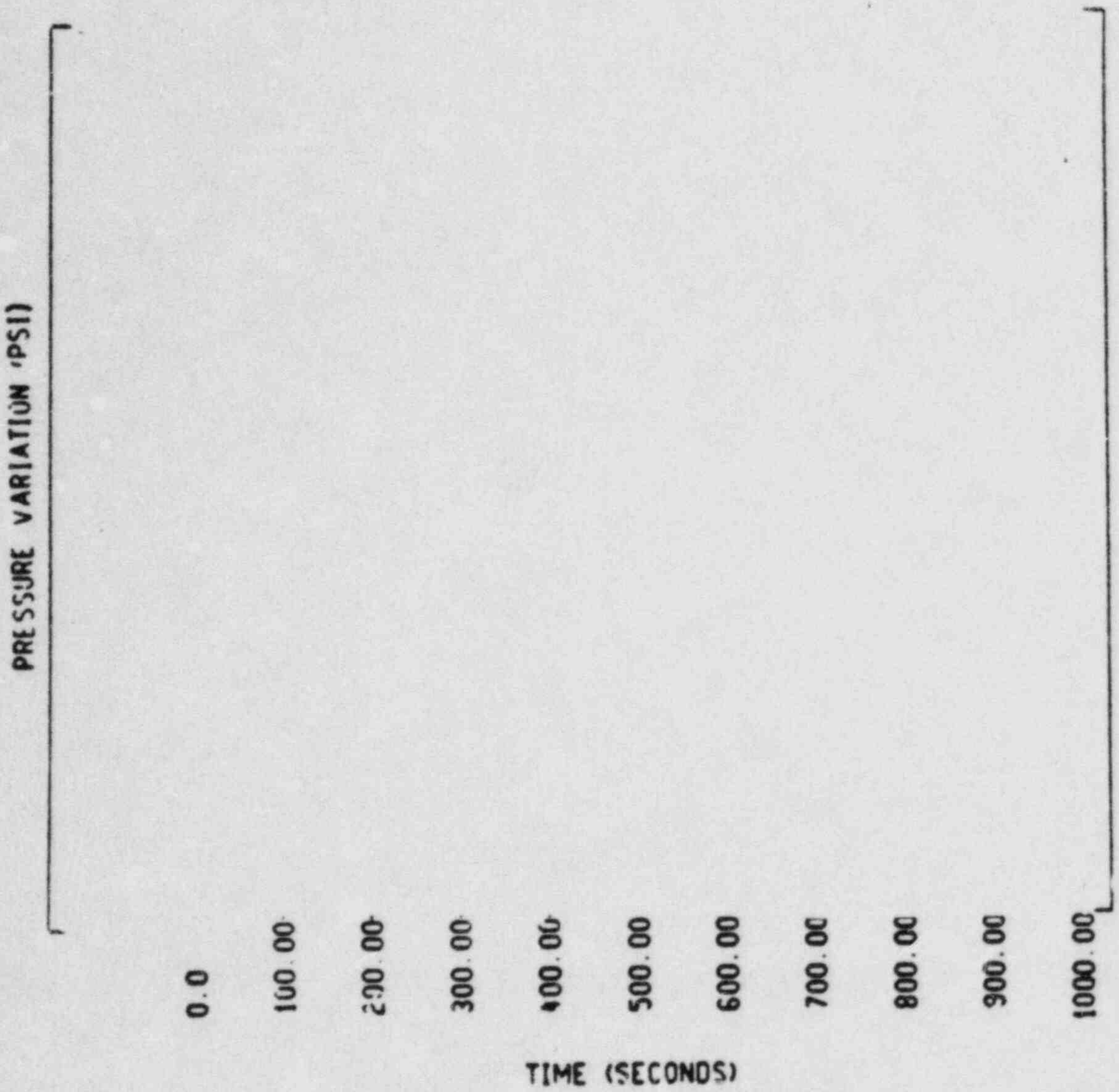


Figure A-43 Control Rod Drop - Cold Leg Temperature Versus Time

a,c,e



NOTE: PRESSURE RETURNS TO INITIAL VALUE AT A RATE CONSISTENT WITH NORMAL PRESSURIZER HEATUP.

Figure A-44 Excessive Feedwater Flow - Reactor Coolant Pressure Versus Time

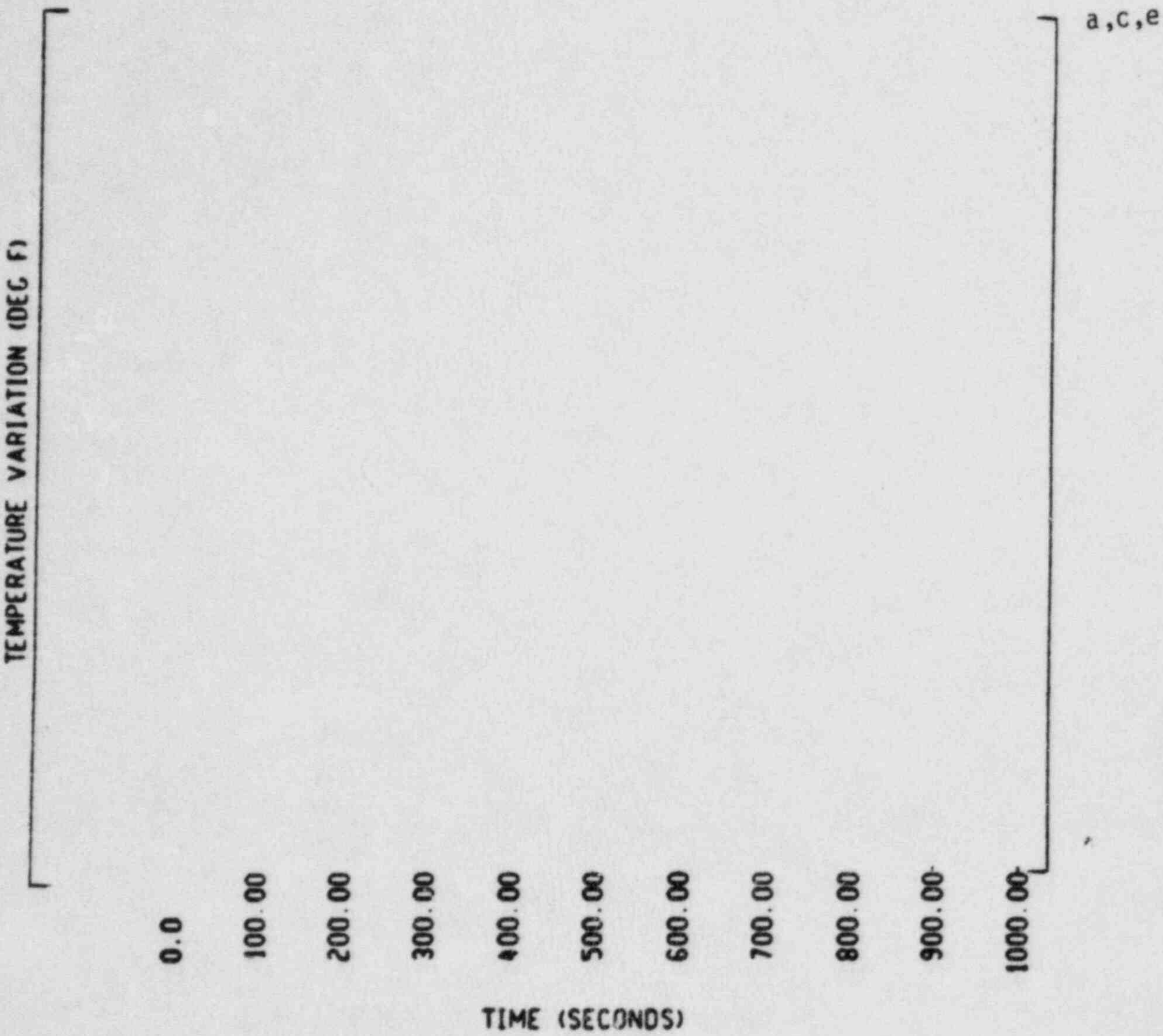


Figure A-45 Execssive Feedwater Flow - Cold Leg Temperature Versus Time

a,c,e

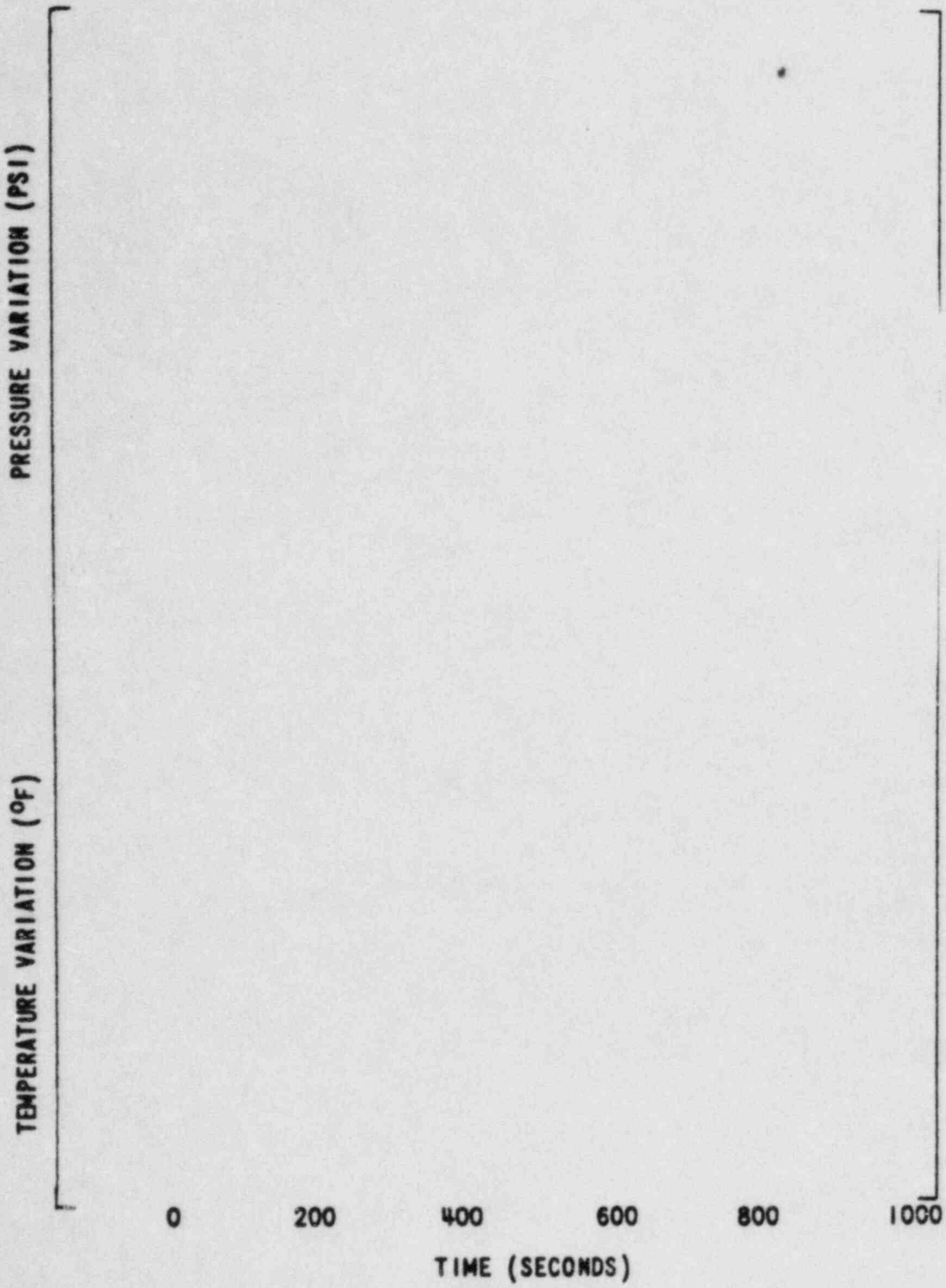


Figure A-46 Turbine Roll Test - Reactor Coolant Pressure and Temperature Versus Time

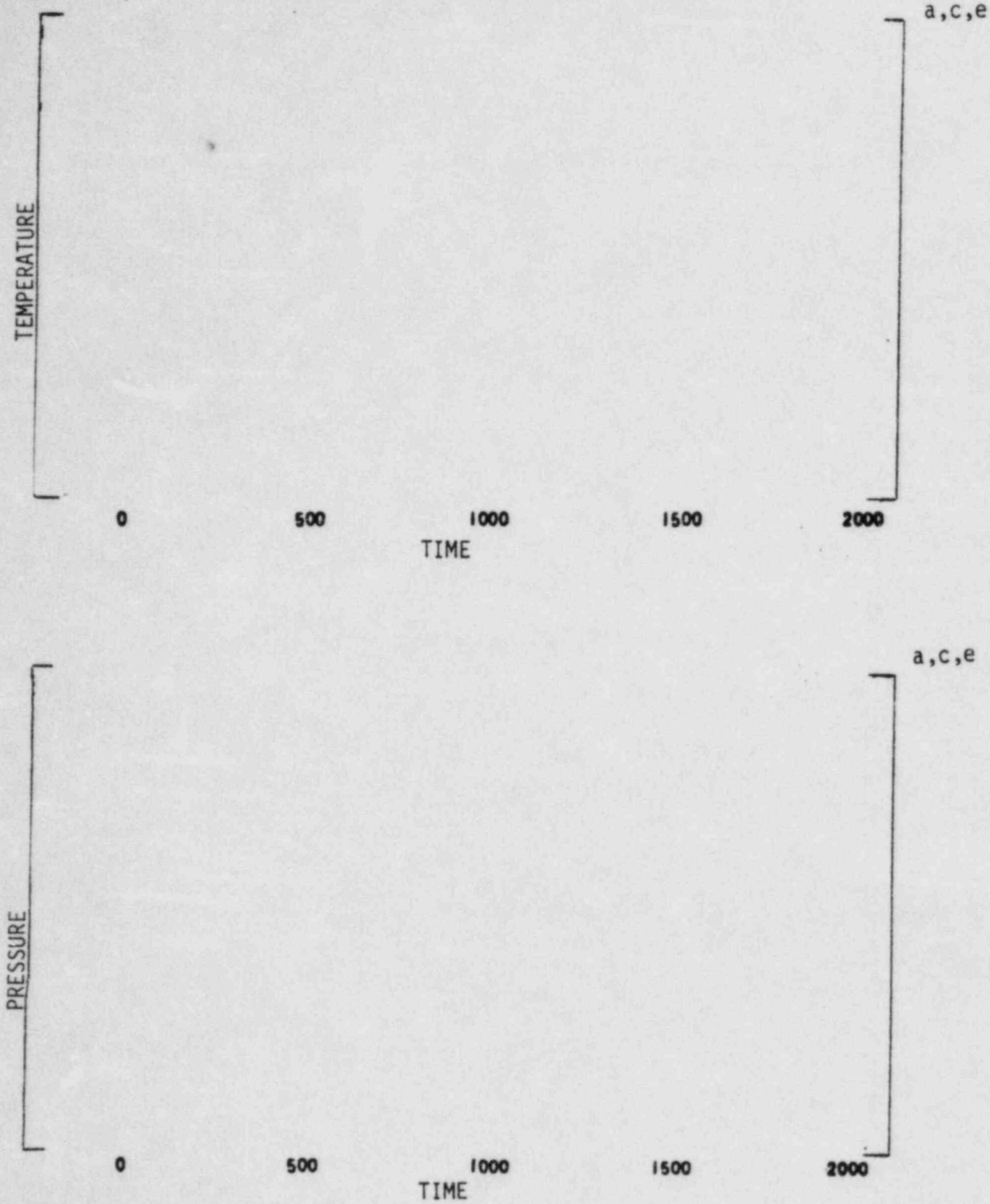
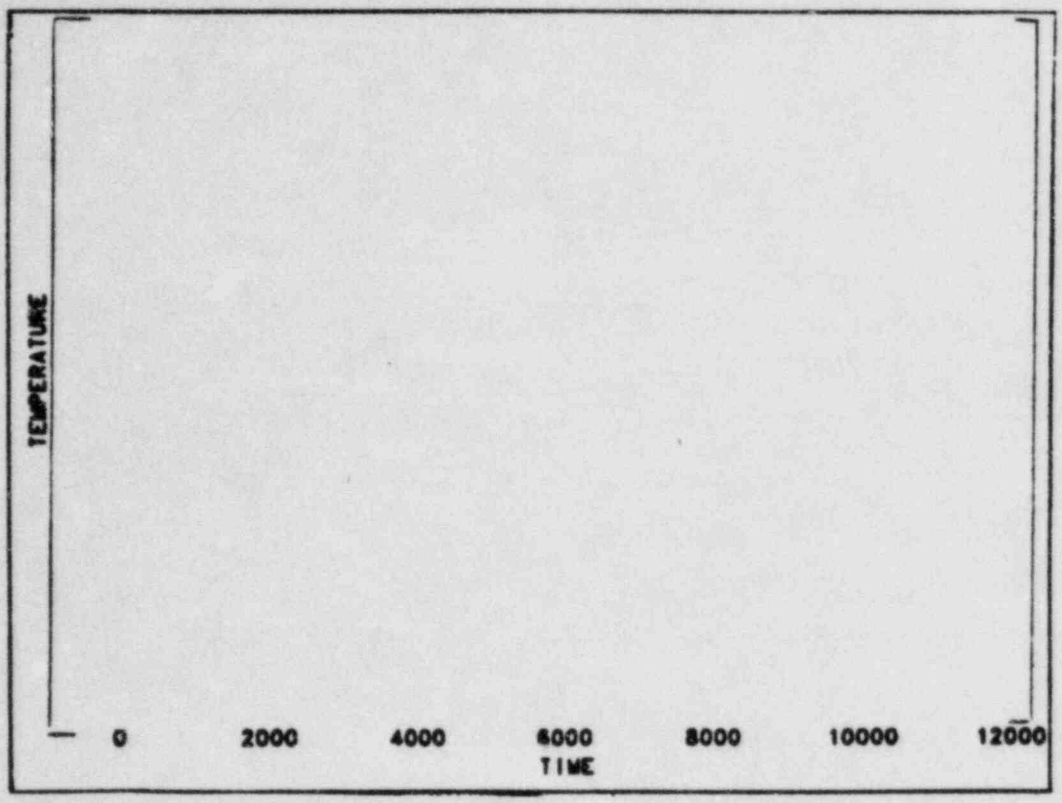


Figure A-47 Large Steamline Break - Temperature and Pressure Versus Time

a,c,e



a,c,e

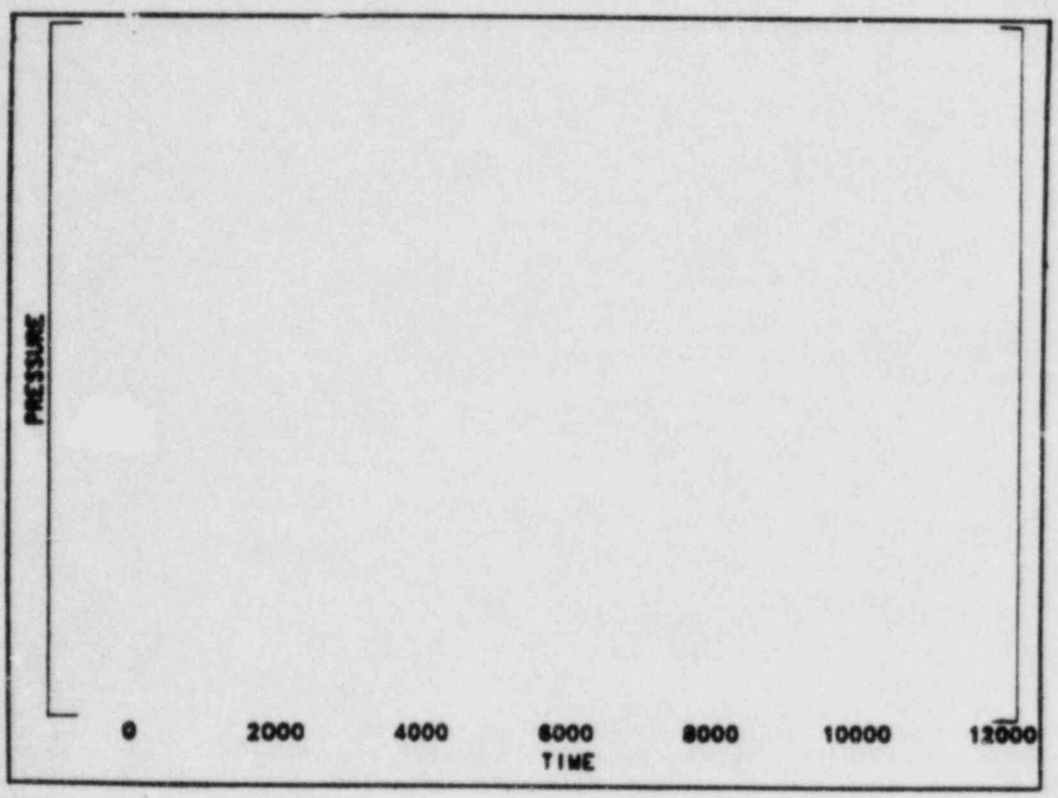


Figure A-48 Small Steamline Break - Temperature and Pressure Versus Time

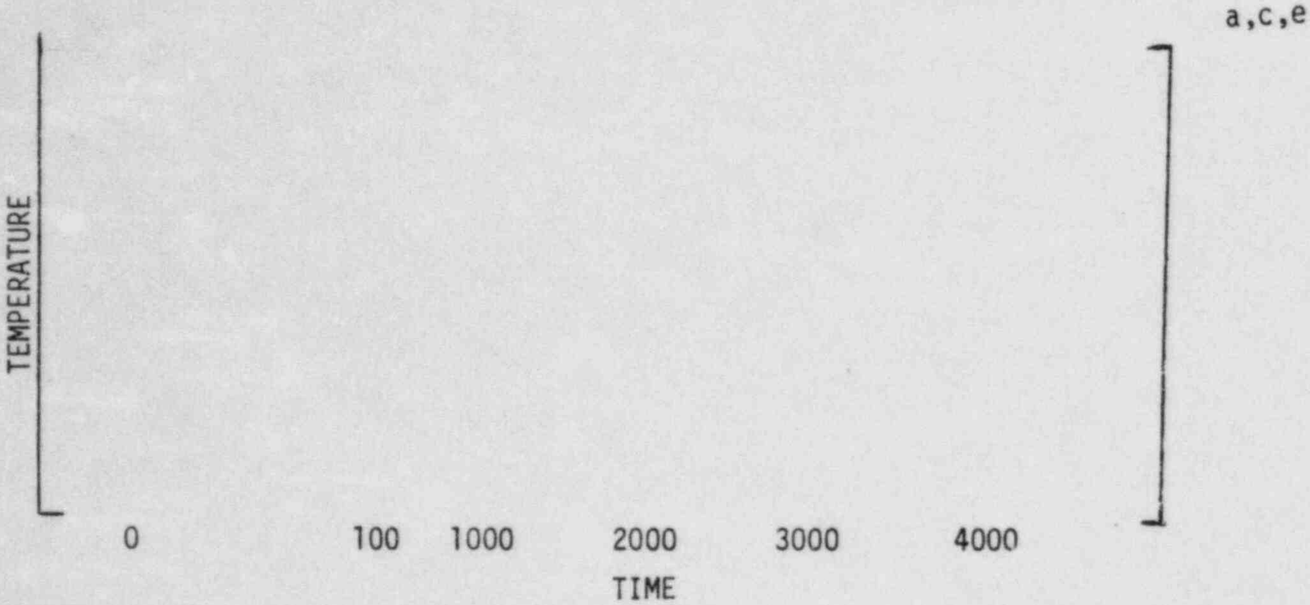
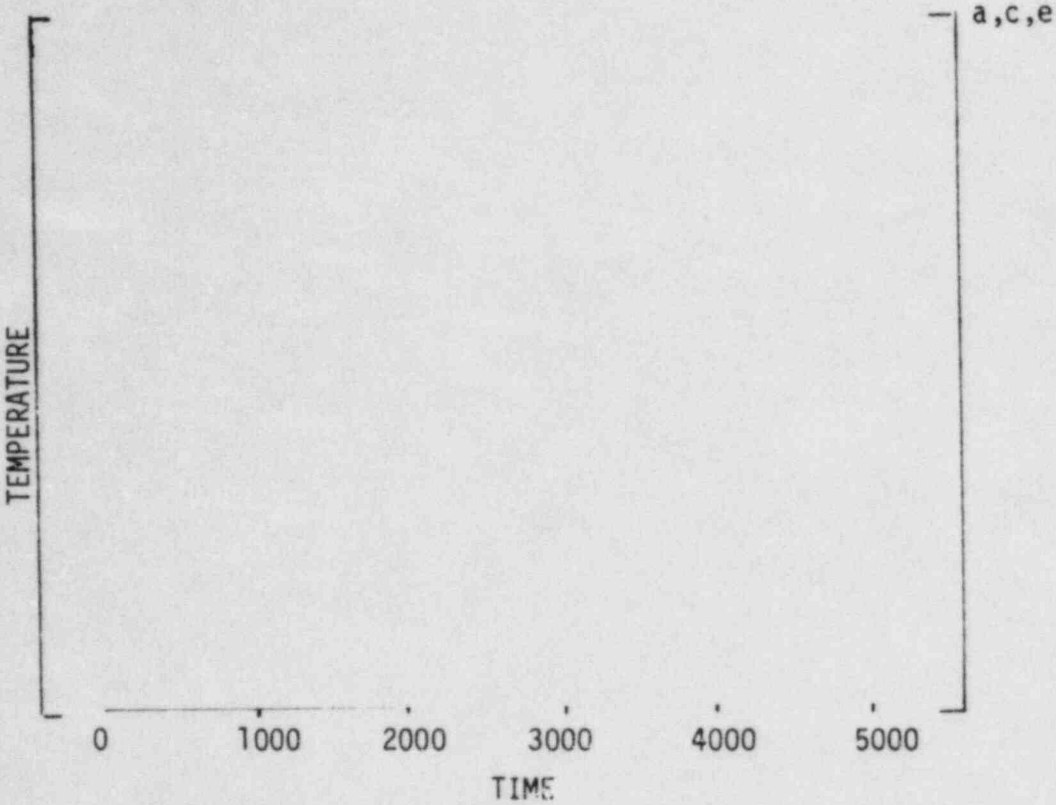


Figure A-49 Large LOCA - Temperature and Pressure Versus Time



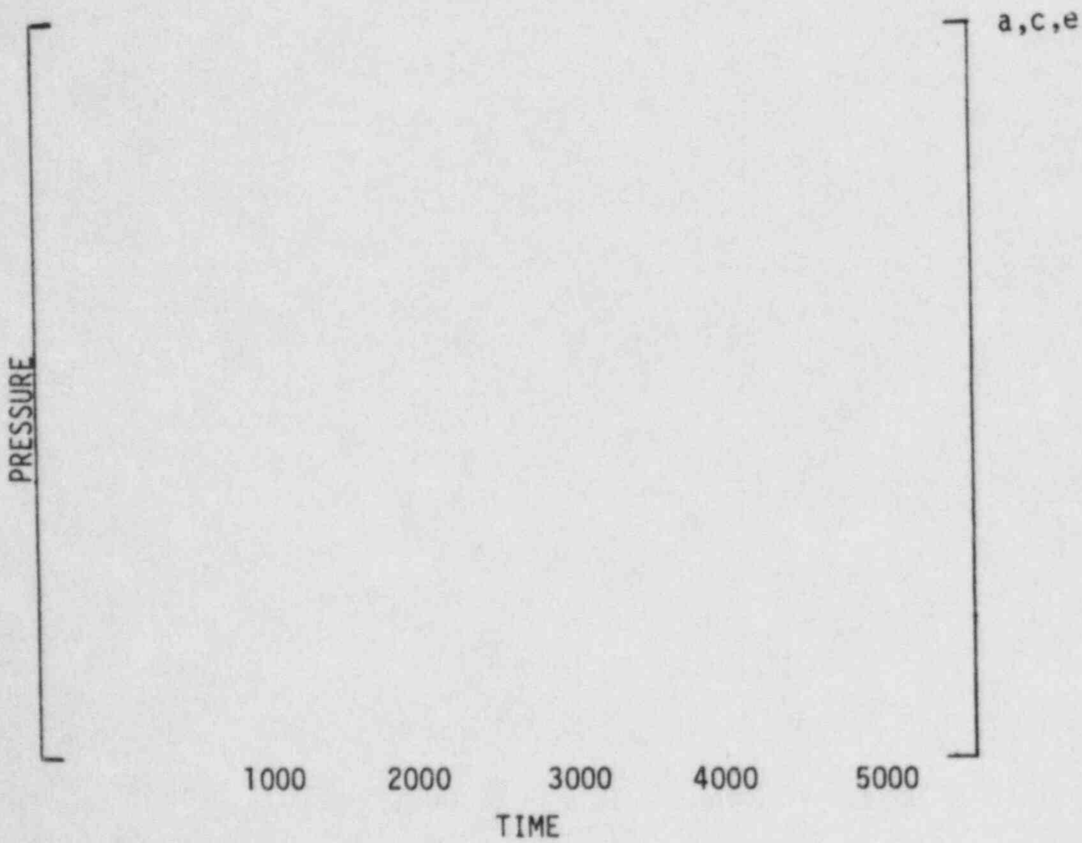
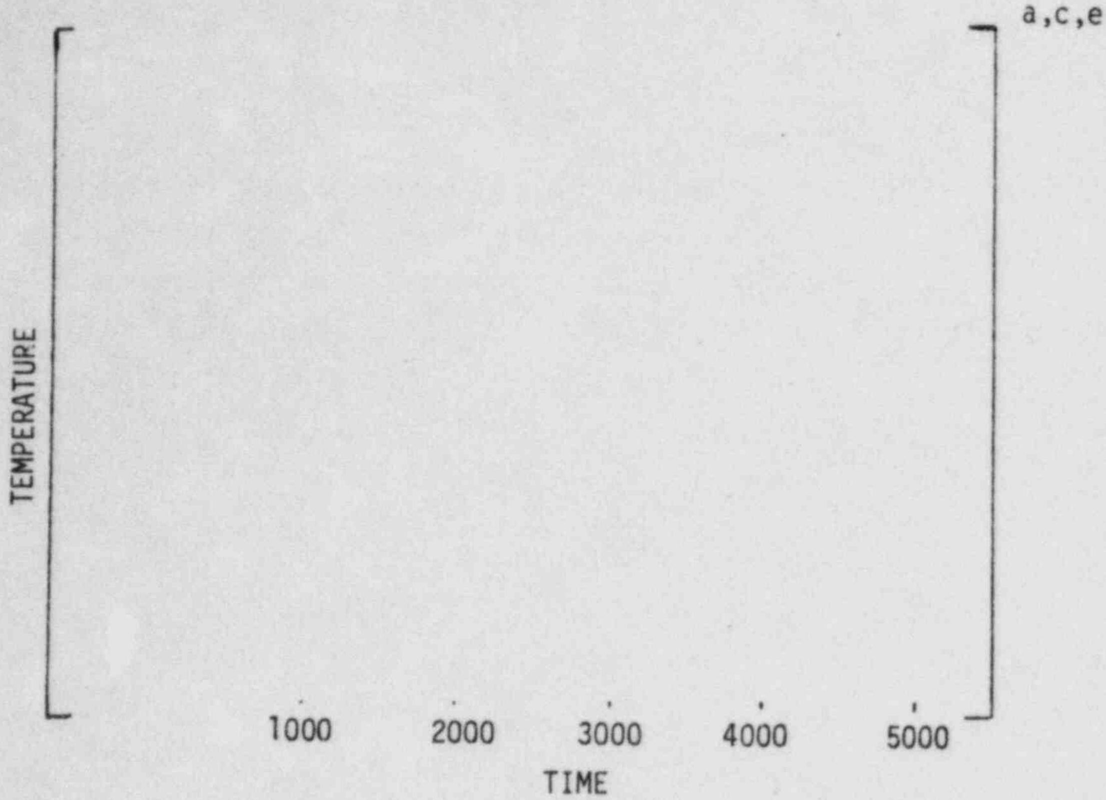


Figure A-50 Small LOCA - Temperature and Pressure Versus Time



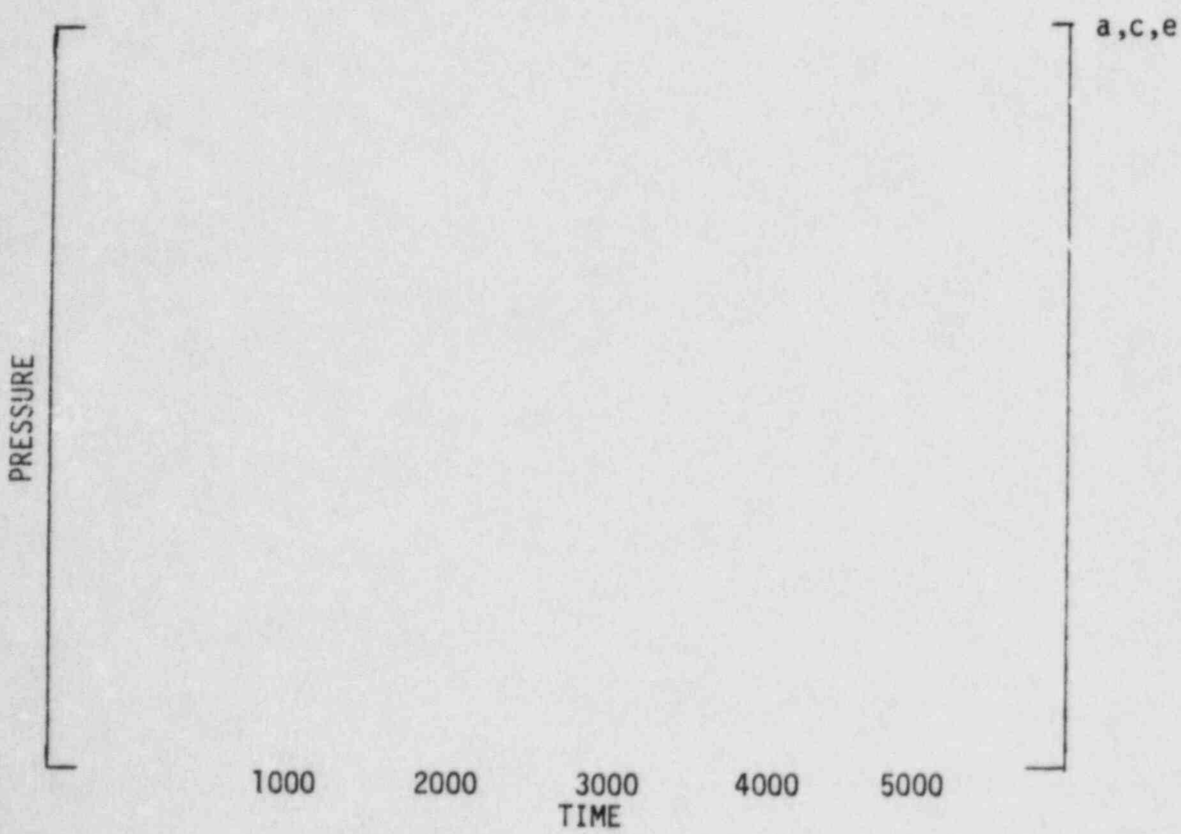


Figure A-51 Steam Generature Tube Rupture - Temperature ard Pressure Versus Time

APPENDIX B

PROBABILISTIC ASSESSMENT

The results of a Westinghouse development program [1,12] can be used in order to assess the deterministic results obtained in the previous section. Using probabilistic fracture mechanics (PFM), an evaluation was performed during 1983 to quantify the risk of vessel failure from transients with high pressures while at relatively low temperatures. The probabilistic model used in this evaluation is similar to the NRC model [1] except for the following:

- Constant Flaw Size

  In the NRC model, a distribution of flaw sizes was used, however, it is more appropriate in this evaluation of an indication to use a constant value estimate of the indication. Also in this evaluation, the flaw is assumed to be semi-elliptic with an aspect ratio of 1 to 6 on the inside surface, however, for transients in which the pressure stresses are predominant the stresses can be assumed to be constant through the thickness, therefore, an indication near the outside surface can be modeled by a flaw on the inside surface.

- K_{Ic} and K_{Ia} Curves

  The K_{Ic} and K_{Ia} curves developed by Westinghouse were used [2] since it is more conservative in the higher transition region.

- Fluence Attenuation

  The fluence distribution included the effects of dPa. This is especially important when evaluating flaws near the outer surface. The surface fluence used in the PFM evaluation was 1.9×10^{19} n/cm^2 which is higher than the surface flaw fluence predicted at end-of-life (EOL) for the longitudinal weld [2].

References 1 and 12 contain a much more detailed discussion of the PFM model and assumptions.

From the many transients which were evaluated in the development program, the transient which best models the small steamline break (found to be the most limiting of the emergency and faulted conditions in Section 5.0) is represented by a cooldown from 550°F to 200°F at 100°F/hr with a pressure of 2560 psi. Using the previously discussed assumptions, the conditional probability of significant flaw extension was found to be 8.0×10^{-7} occurrences/reactor year given that the event occurs. This is a very conservative number because (1) it is obtained using an RT_{NDT} which is much greater than the predicted EOL RT_{NDT} for the longitudinal weld of concern [2], (2) the frequency of the event occurring has not been taken into consideration, and (3) the potential benefit of arrest of a propagating flaw is not considered. Therefore, the results of the probabilistic analysis support the conclusions of the deterministic analyses in that the risk of vessel failure or significant flaw extension from the indication is negligible.