Affect of Thermal Transient on Reactor Coolant System
Steam Generator Tube Rupture Incident

R.E. Ginna Nuclear Power Plant

April 26, 1982

TABLE OF CONTENTS

1.0 INTRODUCTION

At 9:25 a.m. on January 25, 1982, the Ginna B-Steam Generator experienced a tube failure. Prior to the transient, the plant was operating at full power with no primary to secondary leakage. The plant transient resulting from the tube failure included a significant primary system depressurization and actuation of the safety injection system.

As a result of this incident, the impact of the transient on the structural integrity of the reactor vessel beltline, inlet nozzle, and safety injection nozzle was evaluated. A summary of the analyses was presented in Section 6.4 of the Incident Evaluation Report, which was submitted to the NRC by letter dated April 13, 1982. This report presents additional details of the analysis.

2.0 TRANSIENT DESCRIPTION

At 9:25 a.m. on January 25, 1982, the Ginna B-Steam Generator experienced a tube failure. Reactor trip and safety injection system actuation occurred at 9:28 a.m. The reactor coolant pumps were tripped at 9:29 a.m. and one pump was restarted at 11:22 a.m. The primary system wide range pressure and B-loop cold leg temperature transients which resulted are shown in Figures 2-1 and 2-2 and the flow rate history is provided in Figure 2-3. Time zero for the figures was arbitrarily chosen as 09:22 which is the last data point at nominal conditions. (A complete description of the event, including a detailed chronology is provided in our Incident Evaluation Report.)

During the period after which the B-Steam Generator had been isolated and the reactor coolant pumps had been stopped, the measured cold leg temperature in the isolated loop may not have been an accurate indication of vessel beltline fluid temperature. Consequently, for the vessel nozzle and vessel beltline, two cases were analyzed. First, assuming natural circulation did occur in the faulted loop, perfect mixing between the safety injection and primary flows would be likely to occur and the measured B-loop cold leg temperature given in Fig. 2-2 is assumed to be the actual thermal transient experienced by the vessel. Second, if natural circulation did not occur, then streaming of the safety injection (SI) flow could have occurred and the vessel could have experienced the Refueling Water Storage Tank (RWST) temperature. This is the minimum possible temperature and is considerably below the isolated cold leg measured temperature. It presumes that no mixing of injection water with warmer loop and downcomer water occurs from the point of injection to the vessel nozzle and vessel beltline. It is expected that some mixing would occur and, therefore, is a conservative lower bound on which to base the analysis. The transients used in the perfect mixing case were the ones shown in Figures 2-1 to 2-3. For the latter case (heretofore labeled as the no mixing case), the pressure and flow rate histories were kept the same but the thermal transient was modified. From the point at which the reactor coolant pumps were tripped, the "bulk" temperature experienced by the vessel was assumed to be the minimum RWST temperature of 60°F, and this case is also shown in Figure 2-2. The flow rates while the Reactor Coolant pumps were operational were assumed to be full flow of 180,000 gpm and 7% of this flow or 12,600 gpm was assumed with the pumps not running.

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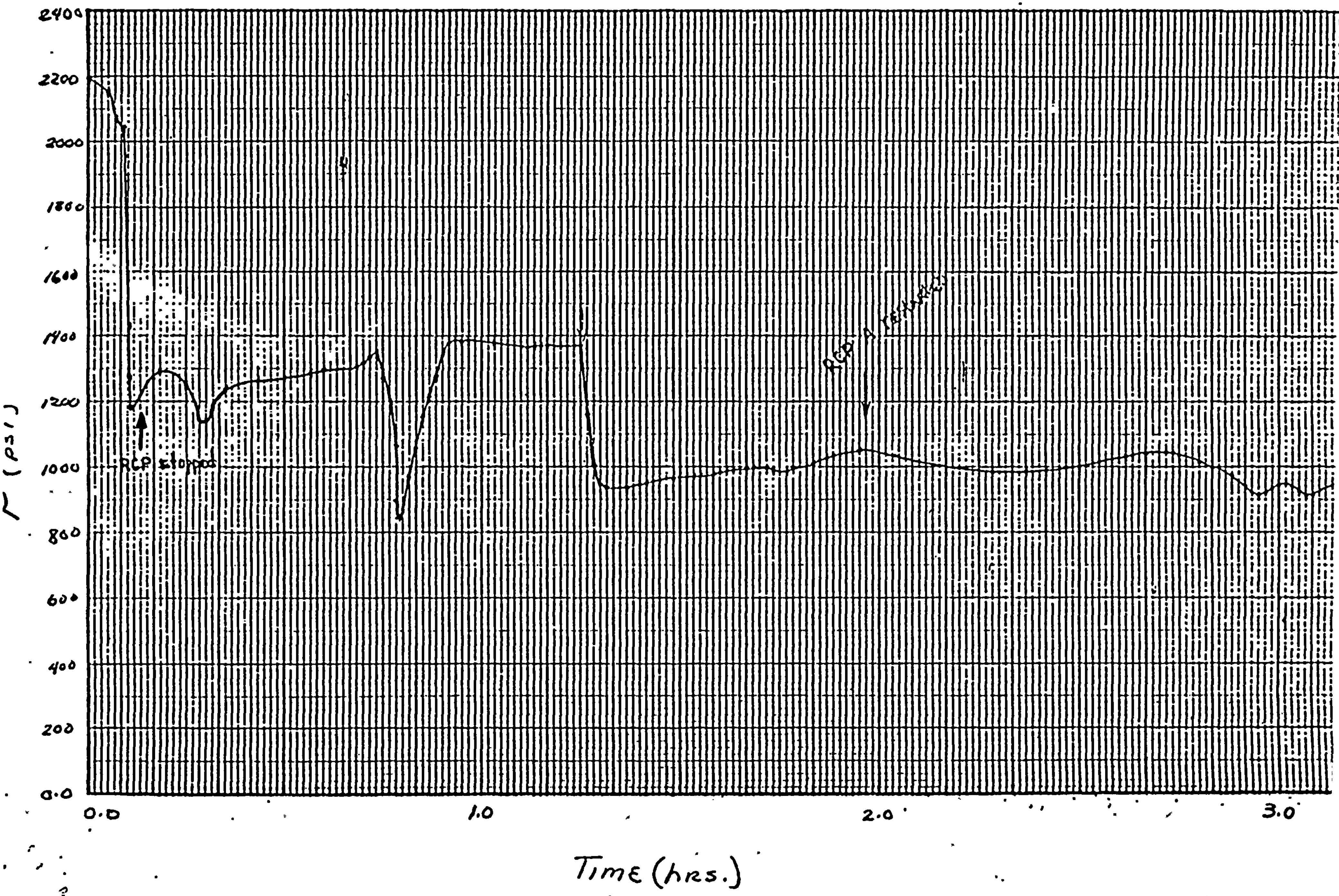


Figure 2-1   Cold Leg Pressure History - January 25 Incident

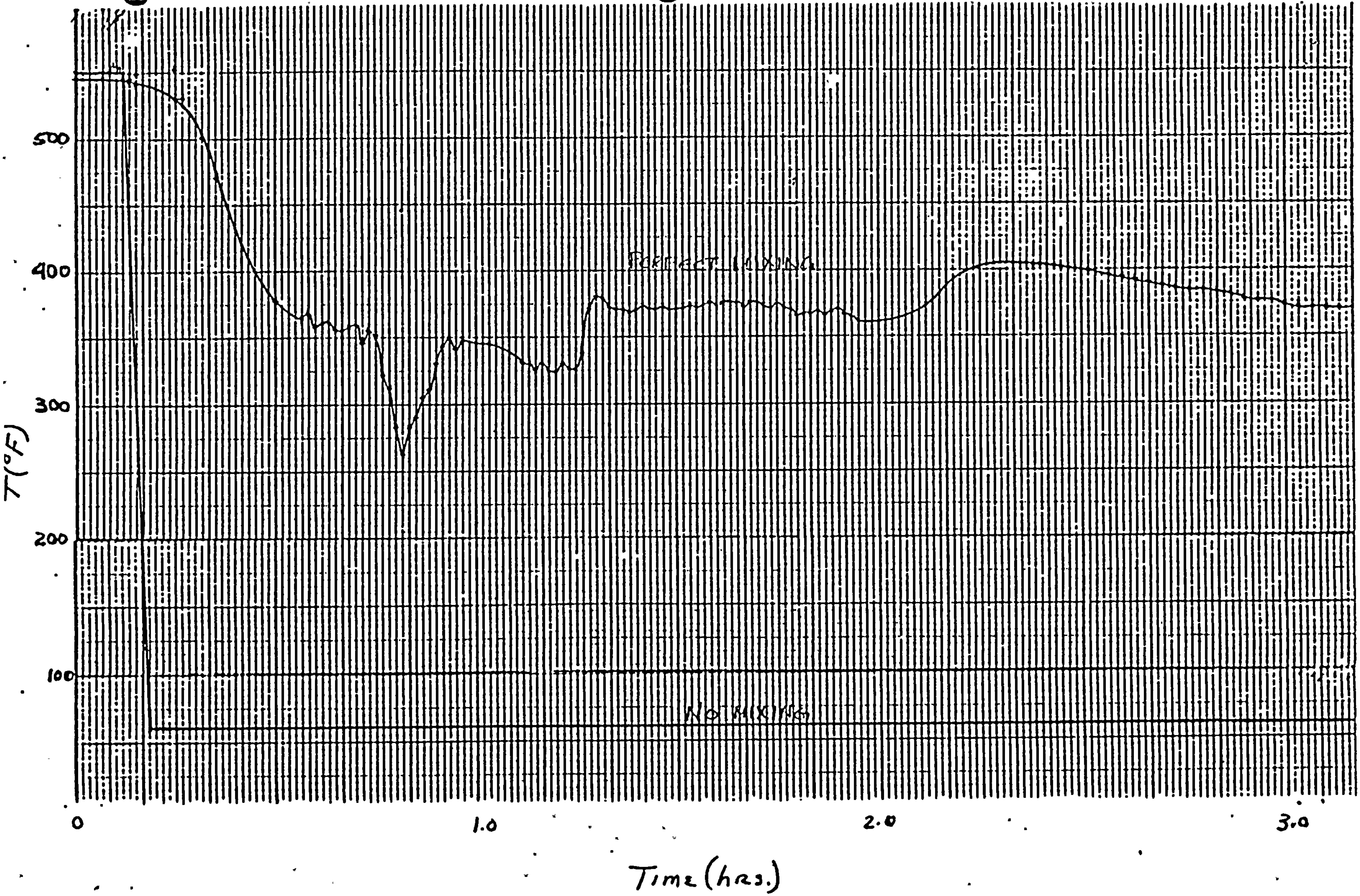


Figure 2-2 Cold Leg Temperature History, January 21 Incident

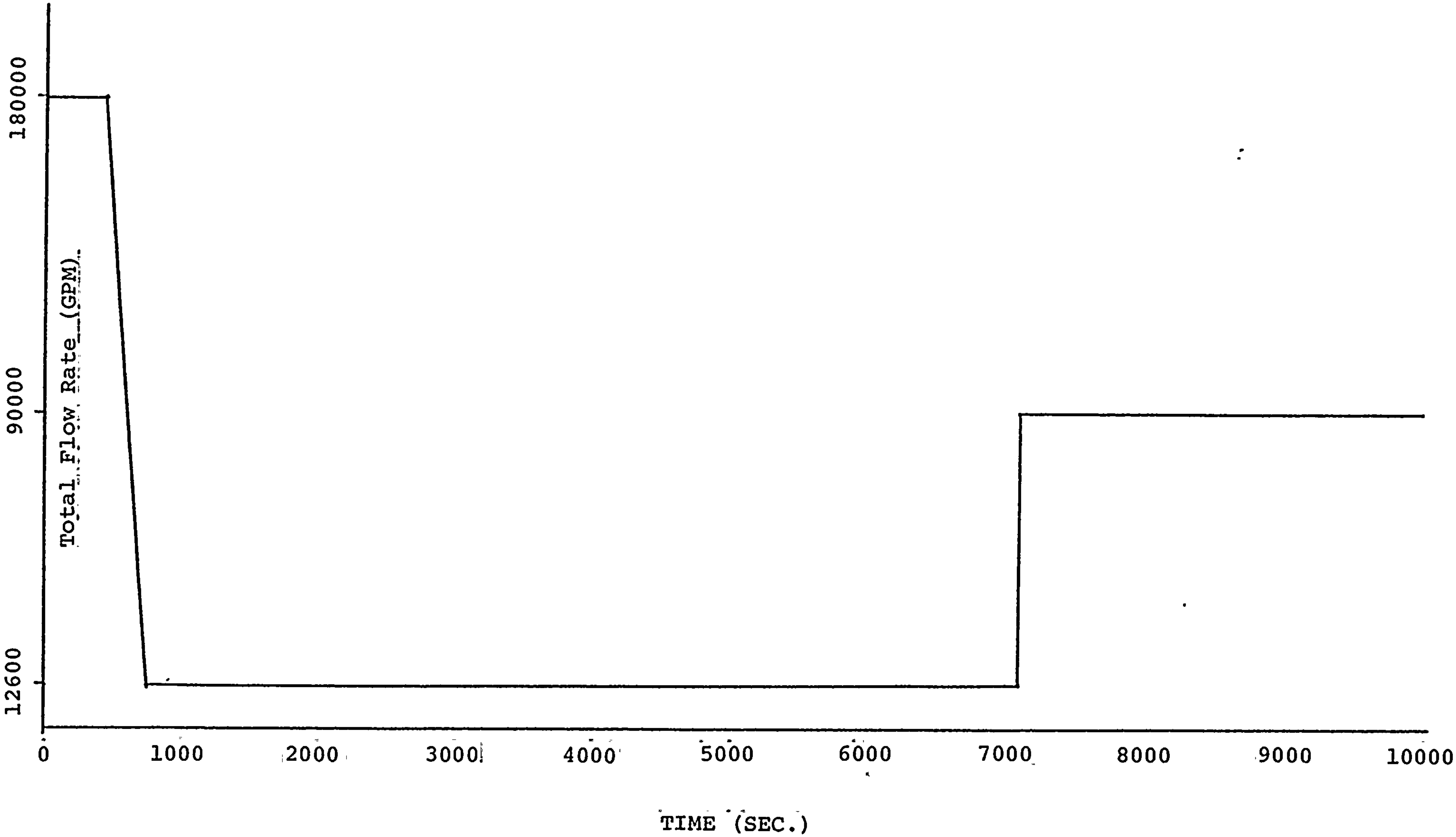


Figure 2-3 Flow Rate History, January 25 Incident

3.0 MATERIAL PROPERTIES AND IRRADIATION EFFECTS

The Ginna reactor vessel was manufactured by Babcock and Wilcox Co. and was shipped to the site in October 1967. The vessel was fabricated from SA508 C2 ring forgings jointed with automatic submerged arc welds. The numerical designations of the ring forgings and circumferential welds in the vessel are shown in Figure 3-1, along with the location of the reactor core relative to the vessel.

3.1 Beltline Region

A detailed listing of the reactor vessel core region forgings and welds is given in Table 3-1, along with their heat treatment history. The chemistry of all the materials is given in Table 3-2, and the mechanical properties are given in Table 3-3.

Three surveillance capsules have been tested and analyzed from this vessel, and have included two forgings and a single weld. The results of these investigations are summarized in Table 3-4, along with the sources for the data. Results show that the forgings incur very little irradiation damage as measured by the shift in charpy energy, which is expected because of their very low copper content. The surveillance weld showed an initial shift approximately in line with the predictions of Regulatory Guide 1.99, revision 1 [1] as shown in Figure 3-2. The results at higher fluence levels have shown that this trend does not continue, but in fact saturates with a shift of about 170°F. This saturation effect has been observed on other materials of this type, and was discussed recently by Yanichko [2].

Since the surveillance weld had the same weld wire as the governing weld in the reactor vessel beltline, its properties were used in the fracture analysis of the beltline region. The trend curve used is shown in Figure 3-2 as a heavy dashed line. The curve used in the analysis matches the Regulatory Guide curve for 0.23 wt percent copper up to a shift value of 170°F, after which it becomes horizontal.

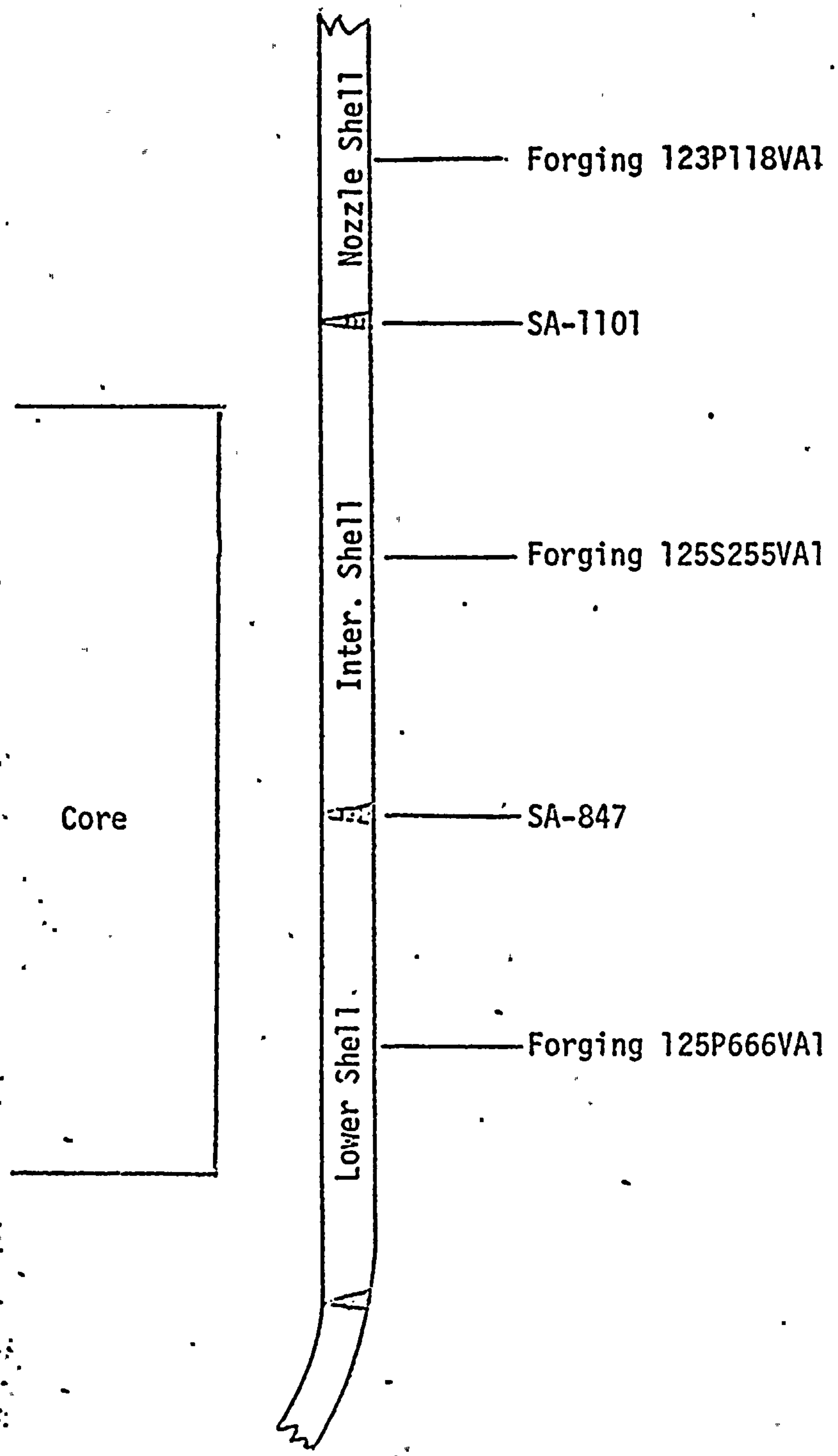
3.2 Inlet Nozzles

The inlet nozzles for the vessel were fabricated with SA508 C2 forgings. Each nozzle heat number is given in Table 3-1. Full charpy curves were obtained for each of the nozzle forgings, and they are presented in Figures 3-3 and 3-4. Since drop weight test results were not obtained on these materials, the initial RT_{NDT} was assumed to be 60°F. This is expected to very conservative, because the material has very good toughness properties as evidenced by the charpy curves.

The nozzle region is sufficiently removed from the core that no significant irradiation damage is experienced, and the RT_{NDT} remains at 60°F throughout life.

FIGURE 3-1

IDENTIFICATION AND LOCATION OF BELTLINE REGION MATERIAL
GINNA UNIT NO. 1 REACTOR VESSEL



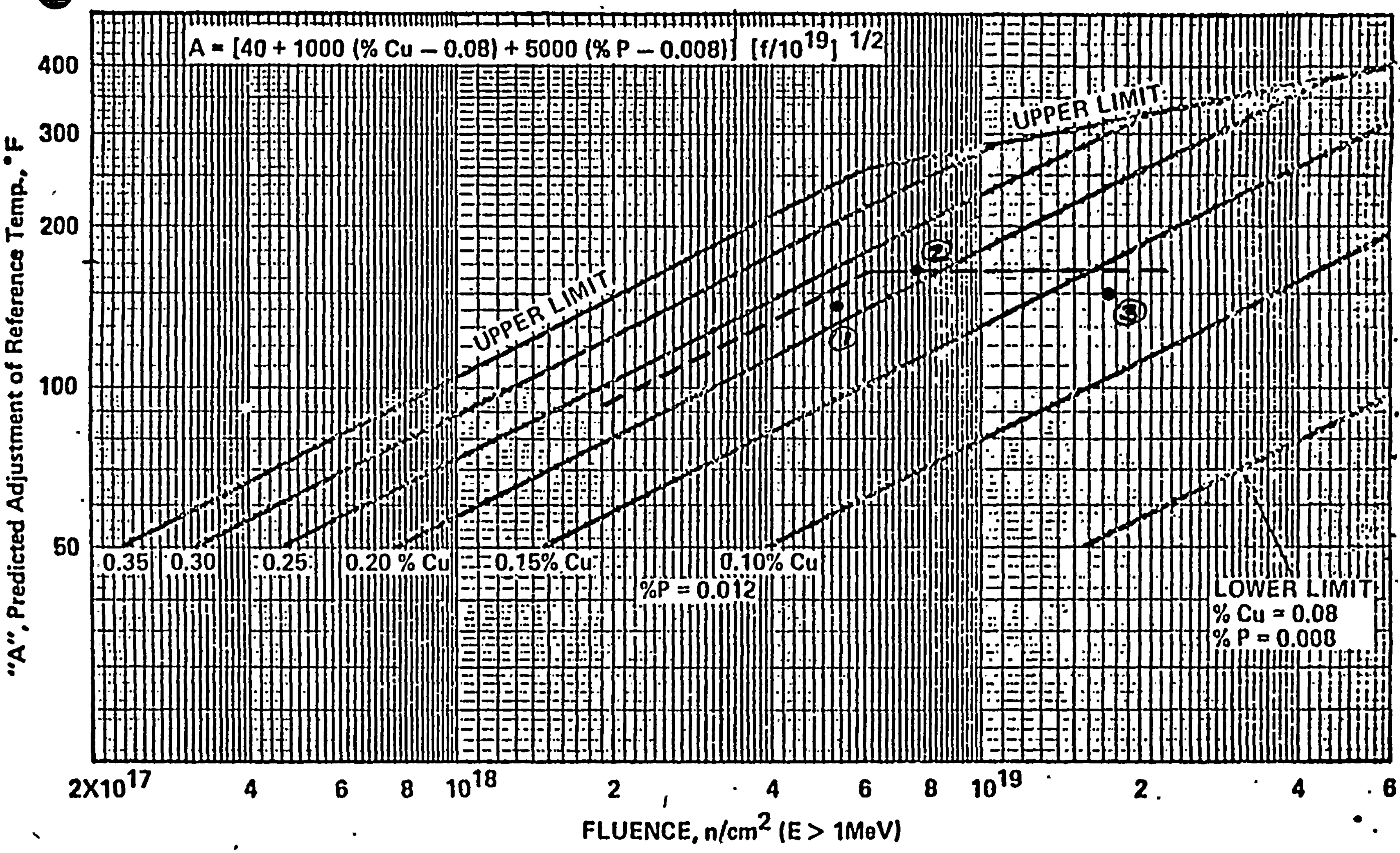


Figure 3-2 Irradiation Damage Curve Used for Beltline Fracture Analysis Compared with Regulatory Guide 1.99 [1]. (Note surveillance results which are numbered)

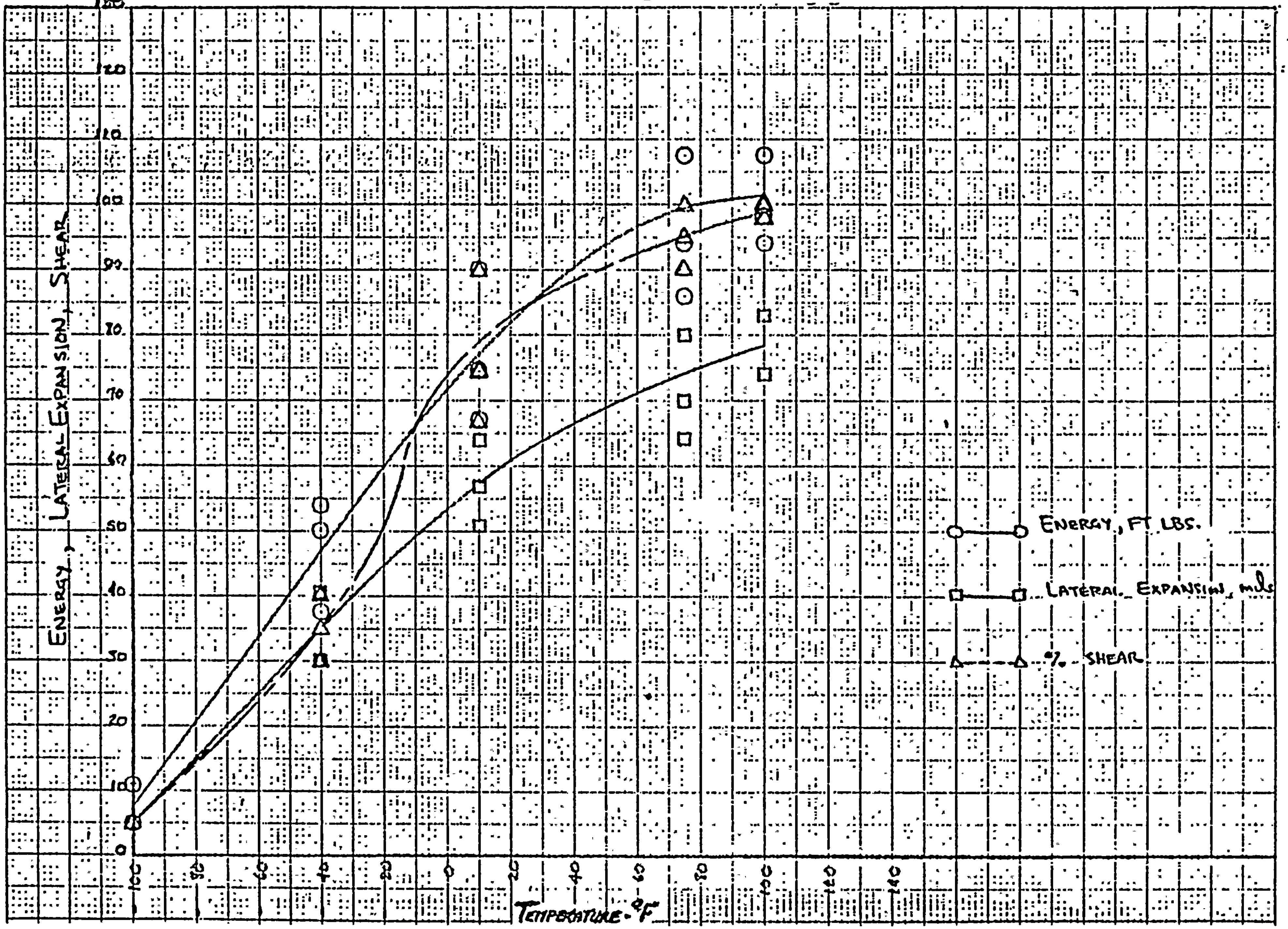


Figure 3-3 Charpy results for
Inlet Nozzle Forging
Heat ZT 2254-2

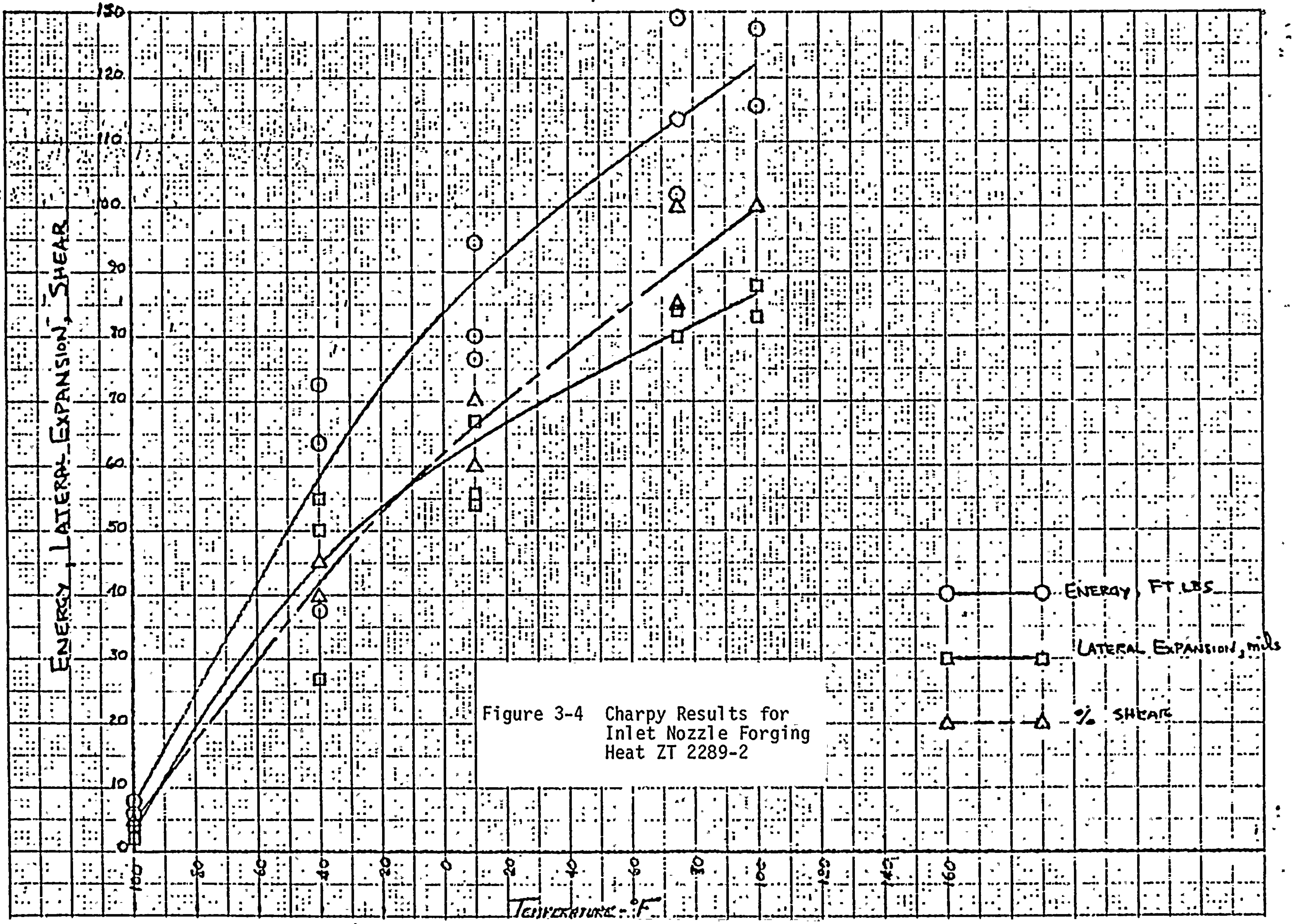


Figure 3-4 Charpy Results for Inlet Nozzle Forging Heat ZT 2289-2

TABLE 3-1

IDENTIFICATION OF BELTLINE MATERIALS

WELDS

Weld Location	Weld Process	Weld Control No.	Weld Wire Type	Weld Wire Heat No.	Flux Type	Flux Lot No.	Post Weld Heat Treatment
Nozzle Shell to Inter. Shell	Submerged Arc	SA-1101	Mn-Mo-Ni	71249	Linde 80	8445	1100-1125°F-48 Hrs.-FC
Inter. Shell to Lower Shell	Submerged Arc	SA-847	Mn-Mo-Ni	61782	Linde 80	8350	1100-1125°F-48 Hrs.-FC
Surveillance Weld	Submerged Arc	SA-1036	Mn-Mo-Ni	61782	Linde 80	8436	1100°F-11-1/4 Hrs.-FC

FORGINGS

Component	Forging No.	Heat No.	Material Spec.	Supplier	Heat Treatment: Austenitize	Heat Treatment: Temper	Heat Treatment: Stress Relief
Nozzle Shell	123P118VA1	123P118	A336	Bethlehem Steel	1550°F-11 Hrs-WQ	1220°F-22 Hrs-AC	1125°F-30 Hrs-FC
Inter. Shell	125S255VA1	125S255	A508 CL2	Bethlehem Steel	1550°F-15-1/2 Hrs-WQ	1210°F-18 Hrs-AC	1125°F-30 Hrs-FC
Lower Shell	125P666VA1	125P666	A508 CL2	Bethlehem Steel	1550°F-9 Hrs-WQ	1220°F-12 Hrs-AC	1125°F-30 Hrs-FC
Surveillance Forgings	125S255VA1 125P666VA1	125S255 125P666	A508 CL2 A508 CL2	Bethlehem Steel Bethlehem Steel	1550°F-15-1/2 Hrs-WQ 1550°F-9 Hrs-AC	1210°F-18 Hrs-AC 1220°F-12 Hrs-AC	1100°F-11-1/4 Hrs-FC 1100°F-11 Hrs-FC
Inlet Nozzle	ZT 2254-2		A508 CL2	Midvale-	Not Available		
Forgings	ZT 2289-2		A508 Cls	Hepenstall Co.	Not Available		

TABLE 3-2

BELTLINE MATERIAL CHEMICAL COMPOSITION

	Weight Percent									
Forging No.	C	P	S	Mn	Si	Mo	Ni	Cr	Cu	V
123P118VA1	.19	.010	.009	.65	.23	.60	.69	.42	---	---
125S255VA1	.18	.010	.007	.66	.23	.58	.69	.33	.07	.02
125P666VA1	.19	.012	.011	.67	.20	.57	.69	.37	.05	.02
ZT 2254-2	.19	.012	.014	.59	.21	.58	.71	.37	.09	-
ZT-2289-2	.20	.011	.014	.66	.20	.60	.69	.30	.09	-
Weld Control No.										
SA1101	.07	.021	.014	1.28	.52	.36	.57	.17	.21	---
SA 847	.082	.012	.012	1.34	.45	.39	.39	.06	.20	---
Surv. weld	.075	.012	.016	1.31	.59	.36	.56	.59	.23	---

TABLE 3-3

Mechanical Properties of Beltline Materials

Forging No.	T_{NDT} °F	RT_{NDT} °F	Upper Shelf Energy ft-lb	YS ksi	UTS ksi	Elong. %	RA %	
123P118VA1	30	30*	117*	66.87	88.00	25.50	73.50	
125S255VA1	20	20*	106*	67.25	88.25	26.25	70.10	
125P666VA1	40	40*	114*	63.50	85.00	26.25	71.05	
125S255VA1	20	20*	91*	78.22	97.19	23.30	66.85	Surveillance Test Results
125P666VA1	40	40*	120*	62.72	83.65	26.35	70.75	

Weld Control No.	Weld Wire		Flux		T_{NDT} °F	Energy at 10°F ft-lb	RT_{NDT} °F	Shelf Energy ft-lb	YS ksi	UTS ksi	Elong. %	RA %
	Type	Heat No.	Type	Lot No.								
SA-1484	Mn-Mo-Ni	71249	Linde 80	8445	0*	45, 45, 46	0*	----	68.63	84.26	28.5	----
SA-1101	Mn-Mo-Ni	61782	Linde 80	8350	0*	58, 60, 36	0*	----	67.00	81.88	29.5	----
Surveillance Weld					0*	54, 66.5, 71**	0*	79.0	73.52	87.35	22.8	62.0

* Estimated based on NRC Standard Review Plan Section 5.3.2 and MTEB 5-2

** Energy at 60°F

TABLE 3-4

SUMMARY OF SURVEILLANCE CAPSULE RESULTS - R. E. GINNA

Material	Cu (wt%)	30 ft-lb Temperature Shift After Fluence of 5.32×10^{18} n/cm^{2}*	7.6×10^{18} n/cm^{2}**	1.75×10^{19} n/cm^{2}***
Weld SA1036	0.23	140°F	165 F	150 F
Forging 125P666VA1	0.05	25 F	25 F	35 F
Forging 125S255VA1	0.07	25 F	0 F	0 F

*"Analysis of Capsule V from the Rochester Gas and Electric R.E. Ginna Unit No. 1 Reactor Vessel Radiation Surveillance Program", T. R. Mager et.al. Westinghouse Nuclear Energy Systems report FP-RA-1, April 1973.

**"Analysis of Capsule R from the Rochester Gas and Electric R.E. Ginna Unit No. 1 Reactor Vessel Radiation Surveillance Program", S. E. Yanichko et.al. Westinghouse Nuclear Energy Systems report WCAP 8421, Nov. 1974.

***"Analysis of Capsule T, to be published by S. E. Yanichko, April 82.

TABLE 3-5

R. E. GINNA MATERIAL IRRADIATON EFFECTS

COMPONENT	CONTROL NO.	Cu (wt%)	INITIAL RT_{NDT} (°F)	AFTER 9 EFPY OPERATION SURFACE FLUENCE (cn/cm^2)	AFTER 9 EFPY OPERATION RT_{NDT} (°F)
Nozzle to Interim. Shell Weld	SA-1101	0.21	0	2.16×10^{18}	95
Interim. to Lower Shell Weld	SA-847	0.20	0	1.125×10^{19}	170
Nozzle Shell	123P118VA1	0.19	30	2.16×10^{18}	95
Inter. Shell	1255255VA1	0.07	20	1.125×10^{19}	45
Lower Shell	125P666VA1	0.05	40	1.125×10^{19}	75
Surveillance Weld	SA-1036	0.23	0	2.16×10^{18}	170

4.0 FRACTURE ANALYSIS AND RESULTS

Detailed evaluations have been made of each of the regions which were exposed to the safety injection water during the January 25 incident. Three regions have been considered:

1. Reactor Pressure Vessel Beltline

2. Reactor Pressure Vessel Inlet Nozzle

3. Safety Injection Nozzle in Cold Leg Piping

The regions of most interest are the reactor vessel inlet nozzle and beltline regions, and these will be discussed first. The safety injection nozzle is fabricated of stainless steel; therefore, fracture is not of concern as discussed later.

The basis for the thermal stress and fracture analyses used for this transient has been discussed in [6] and will not be repeated in this report.

4.1 Reactor Pressure Vessel Beltline

Two bounding cases of perfect mixing and no mixing have been considered in the analyses for the beltline. Since this particular accident transient occurred at approximately 8 EFPY in plant life, all analyses were done for 9 EFPY. Circumferential flaws were assumed to exist in the peak neutron flux region, in the circumferential weld.

The transients used in the analyses shown in Fig. 2-1 through 2-3 have already been discussed. In the perfect mixing assumption, conservatisms such as use of the Reg. Guide 1.99 [1] trend curve and exclusion of warm prestressing considerations have been imposed while performing the fracture analyses. Conversely, due to the conservatism implied in the no mixing assumption, less restrictive assumptions such as a modified Reg. Guide 1.99 trend curve and use of the warm prestressing principle were applied in the fracture analyses.

The warm prestressing principle is based on empirical observations and has been previously discussed [6].

In the perfect mixing case, it has been found that no flaw will initiate anywhere in the beltline. This has been based on the conservative analysis using the actual Reg. Guide 1.99 and without the use of warm prestressing. For the no mixing case, using the modified Reg. Guide 1.99 trend curve and the warm prestressing principle, no flaw was found to initiate. This can be seen on Fig. 4-1 which displays the minimum initiation flaw depth, maximum initiation flaw depth, maximum arrest flaw depth, including warm prestressing effects. It can be seen that the material is warm prestressed before any initiation occurs and therefore no flaw can initiate.

It can therefore be concluded that no flaw would have initiated in the vessel beltline region due to the transient that the plant experienced. This result is applicable to the end of life conditions.

4.2 Reactor Pressure Vessel Inlet Nozzle

The geometry of the nozzle is shown in Figure 4-2, and the material properties have been previously discussed in Section 3. The transient evaluated has been described in Section 2, and as with the beltline, two cases were examined, one for perfect mixing and the other for no mixing. Although the beltline analyses used the actual transients, for the nozzle region both analyses have made use of previous work on other vessels, which have employed different transients. These transients are conservative compared to the January 25 transient. Detailed comparisons are given between the analysis results reported and the actual transient, to demonstrate the applicability of the reference transients.

4.2.1 Perfect Mixing

A previous analysis was completed of a postulated large steamline break (LSB) transient applied to a nozzle of identical construction, and those results will be shown to conservatively envelope the January 25 incident. In addition use will be made of a previous analyses of a safety injection actuation at Ginna [7].

Figure 4-3 shows the January 25 temperature transient compared to the two previously mentioned transients. It can be seen that the reference LSB transient is much more severe from the standpoint of temperature effects.

This is not true for the pressure transient, as seen in Figure 4-4, where the January 25 incident has higher pressure than the reference LSB transient for the first 400 seconds. The January 25 transient is however very similar to the pressure transient which occurred in the previous incident of October 1973 [7] for the first 600 seconds. The stresses obtained in the nozzle corner region for this previous incident are much lower than those for the reference LSB case during the same time frame, so it can be concluded that the more severe temperature transient overwhelms the fact that the pressure is lower during the first 400 seconds. This can be seen by comparing the stress results for the earlier incident in Figure 4-6 with those for the reference LSB in Figure 4-7.

The material fracture toughness properties of the Ginna inlet nozzles were used to determine the appropriate fracture toughness gradient in conjunction with the temperature gradient for the reference LSB transient. The results of the analysis indicated that at the most limiting time in the transient the critical flaw depth for initiation was at least 1.4 inches, as shown in Figure 4-8.

4.2.2 Imperfect Mixing

For the assumption of no mixing, two previous analyses were utilized, and the results were merged to develop a conservative approximation for the January 25 incident. To develop the applied stresses and stress intensity factors, a previous analysis of a large steamline break transient was used. The temperature transient appears in Figure 4-9, and the pressure transient in Figure 4-10. Comparison with the January 25 transient shows that the temperature transient is more severe than the perfect mixing case, but less severe than the case of imperfect mixing. Consideration of the thermal and pressure loadings together led to the conclusion that the stresses resulting from this transient would be applicable for the case of imperfect mixing, and these are summarized in Figures 4-11 and 4-12.

The temperatures in the nozzle corner region resulting from the no mixing assumption will be lower than those calculated for the above reference LSB case, so another analysis was used to obtain the required temperature profile. The analysis chosen was a postulated large loss of coolant accident for a nozzle with essentially identical geometry. The temperature transient is shown in Fig. 4-13, and the resultant temperature distribution in the nozzle corner region was used with the material properties to determine the fracture toughness profile for use in the analysis.

The results of the analyses are shown for the governing time step in Figure 4-14. The applied stress intensity factor was calculated using a three-dimensional formulation, and results are shown for both the surface and the deepest point for a range of flaw sizes. Comparison with the appropriate values of fracture toughness leads to the conclusion that a flaw deeper than 0.75 inches could initiate at the surface and propagate in length, but that a flaw deeper than 1.9 inches would be necessary to have any propagation through the thickness of the nozzle.

4.3 Safety Injection Nozzle

Two nozzle geometries are involved in the safety injection line for the Ginna Plant. The safety injection line feeds a larger 10-inch line which, in turn, feeds the reactor coolant cold leg piping. The 10-inch nozzle in the reactor coolant piping is bounding for two reasons. It is located on the top of the reactor coolant pipe and therefore is always at the temperature of the cold leg (540°F). The nozzle where the 2-inch safety injection line meets the 10-inch line is not in contact with the reactor coolant and thus sees a much smaller temperature change due to its lower initial temperature. Further, the 10-inch nozzle is much more massive than the other nozzle and thus higher thermal stresses will be produced by an injection of cold water.

The 10-inch nozzle was previously analyzed under conditions of full flow such as would be produced during a loss-of-coolant accident. The analysis and results are reported in WCAP 8321, "A Summary Analyses of the Loss of Offsite Power at the Robert E. Ginna Generating Station, October 21, 1973". [7] That analysis has been determined to bound the January 25 incident and demonstrates the acceptability of the safety injection nozzle.

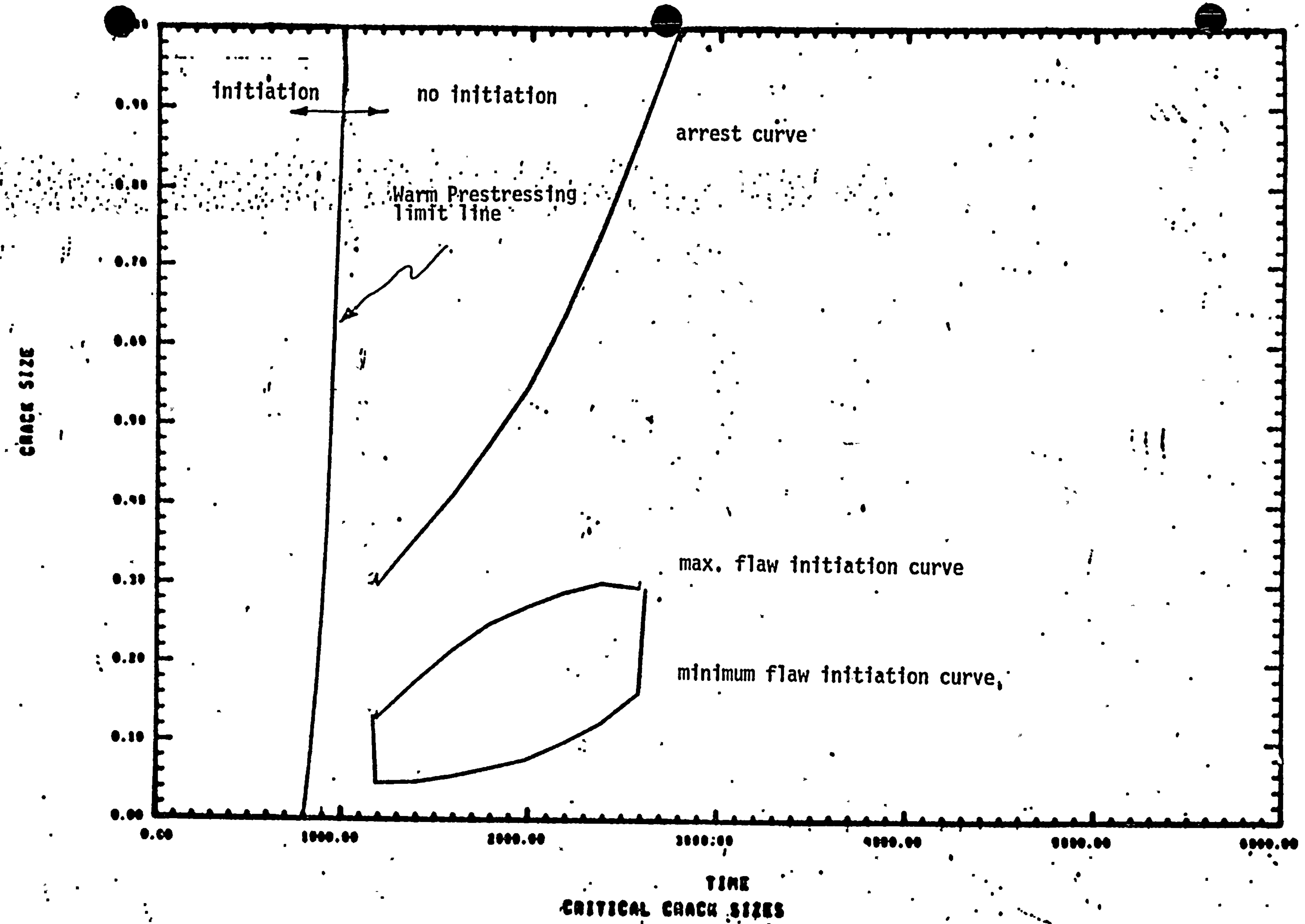


Figure 4-1 Results of the Fracture of Analysis of the January 25 incident for the Reactor Vessel Beltline for the case of imperfect mixing

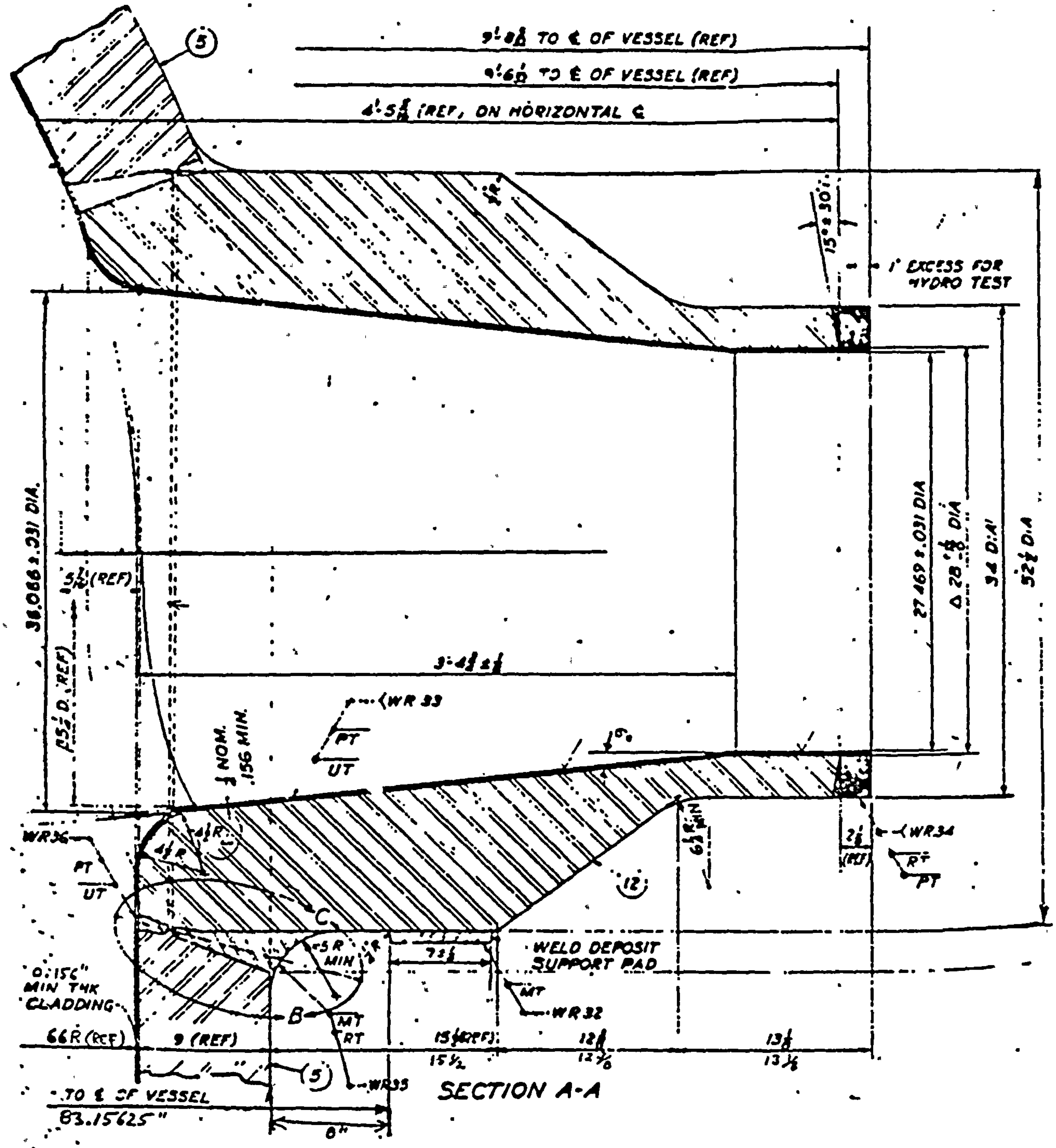


Figure 4-2   Geometry of Inlet Nozzle, R.E. Ginna Unit 1

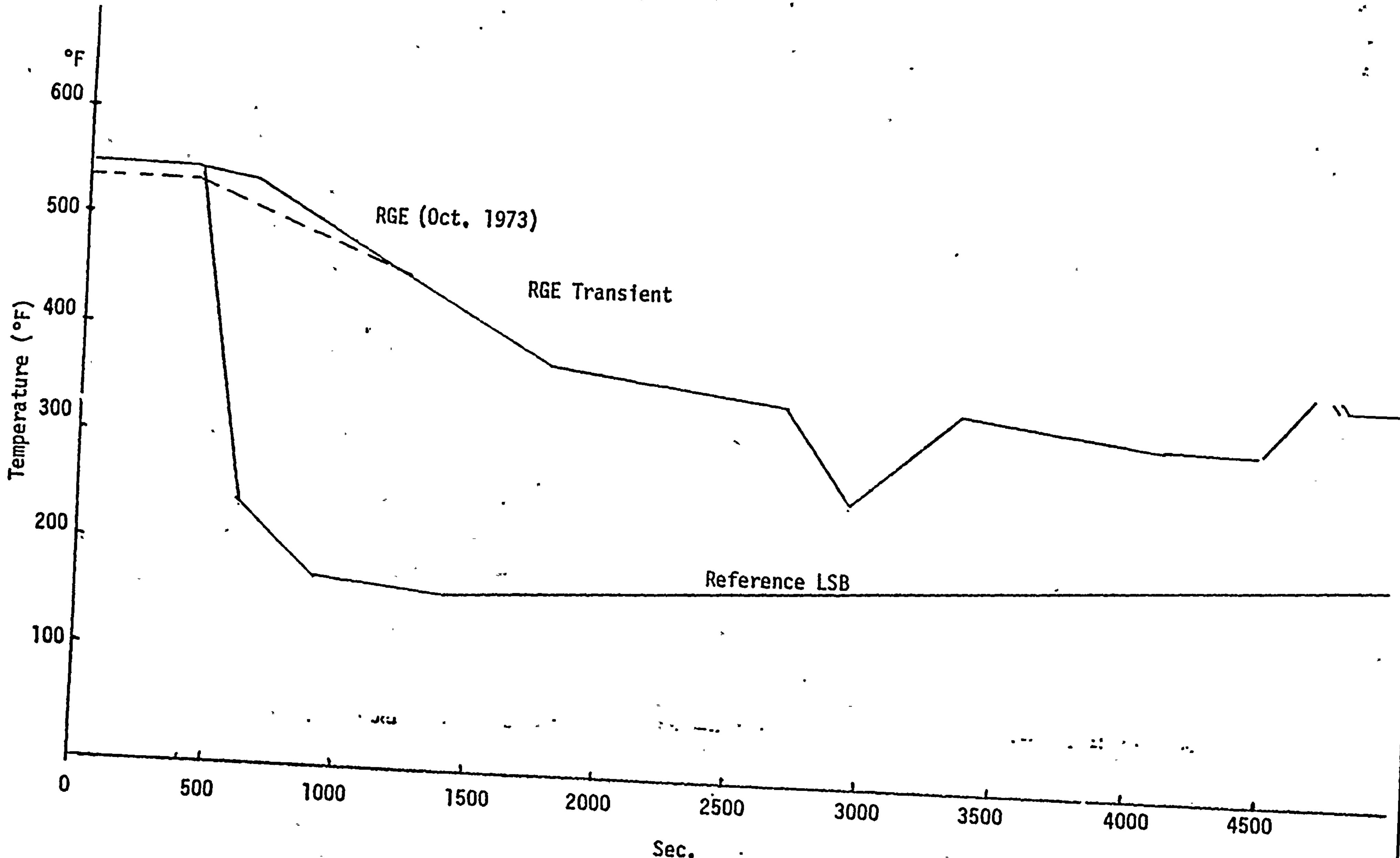


Fig. 4-3 Comparison of Temperature Transient for Reference LSB and RGE

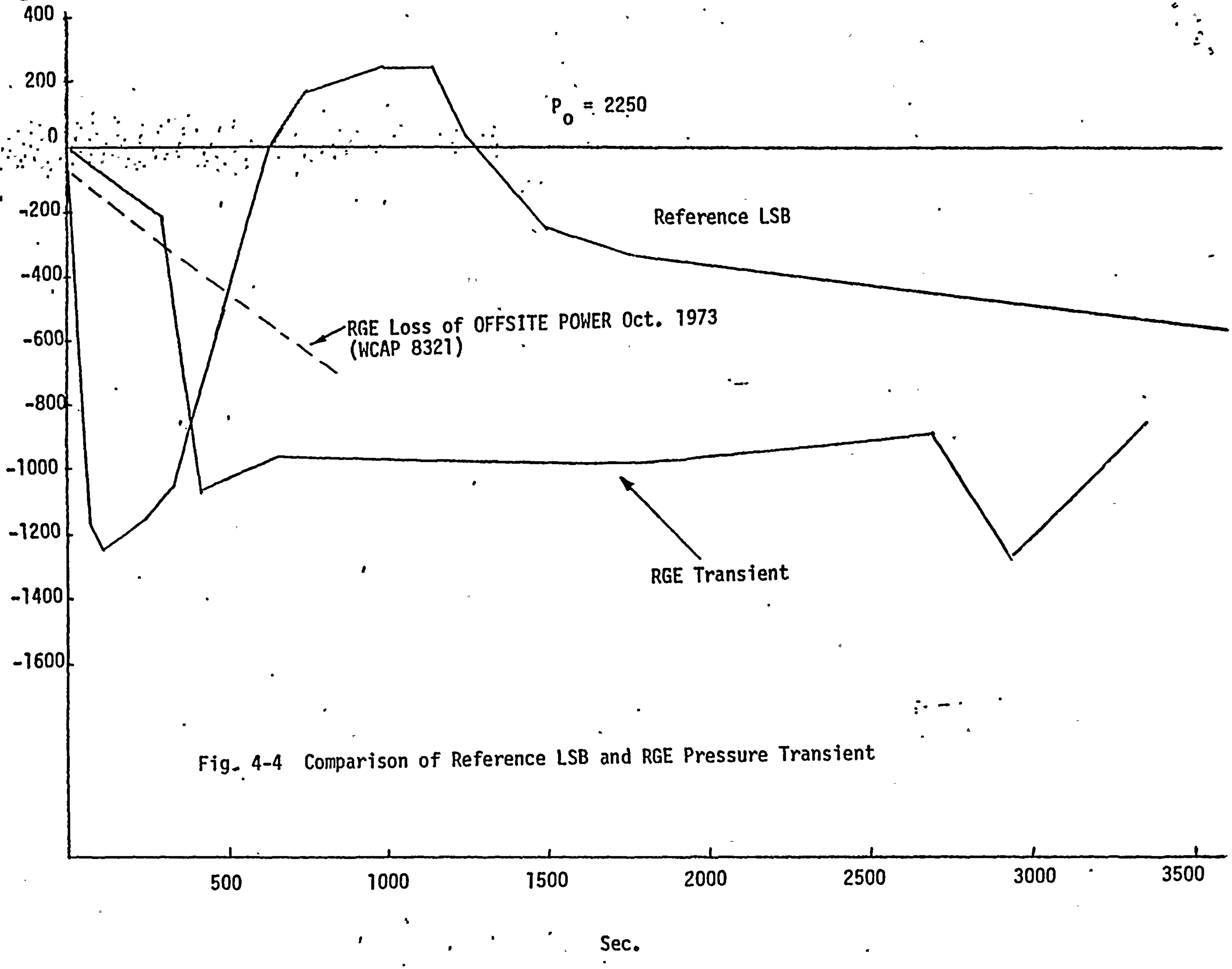


Fig. 4-4 Comparison of Reference LSB and RGE Pressure Transient

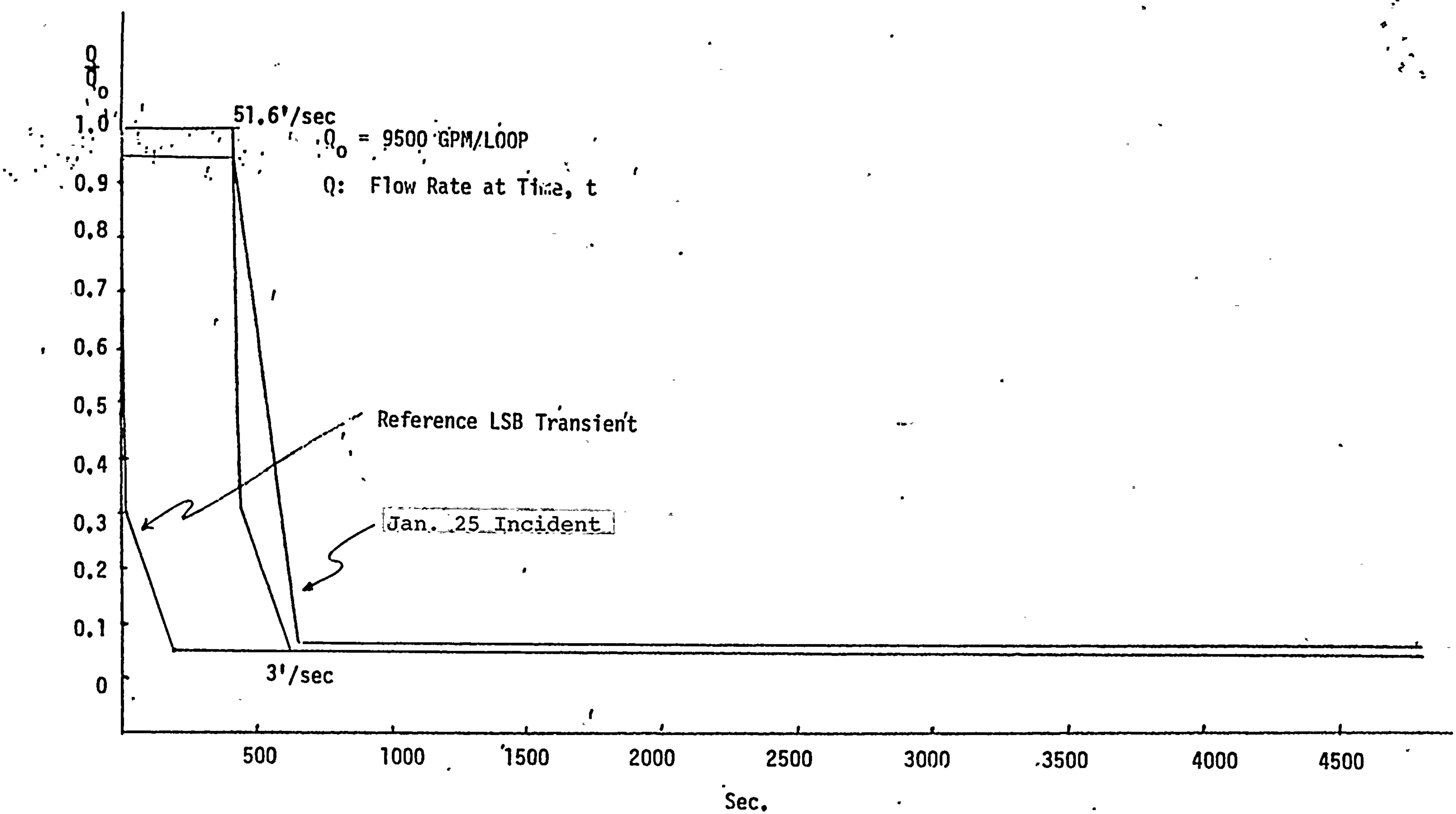


Fig. 4-5 Flow Rate Variation Due to Reference LSB and RGE Current Transient

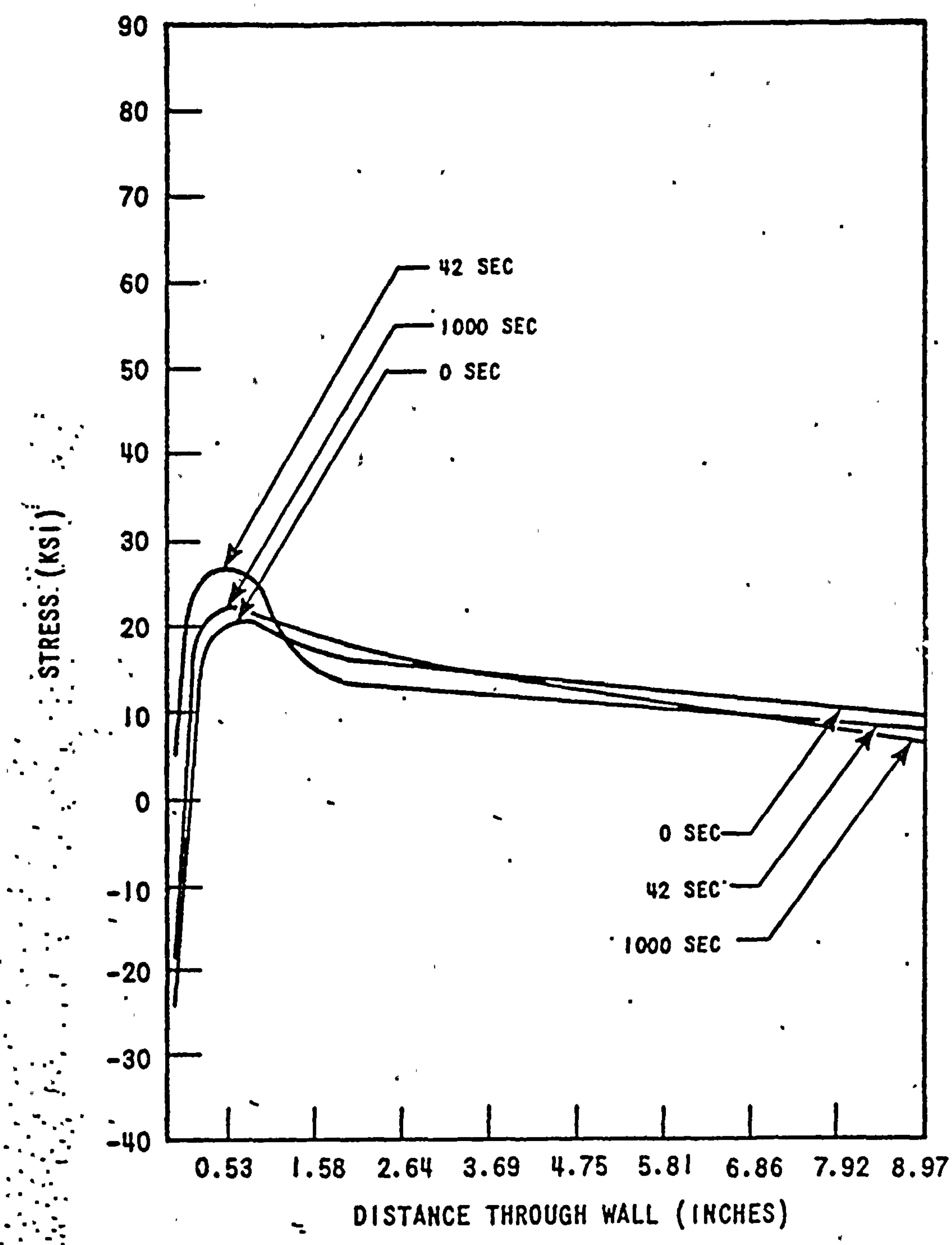


Figure 4-6 Stress Profiles During the 90°F ΔT Transient, R.E. Ginna, Reactor Vessel Inlet Nozzle - Corner - Reference 7

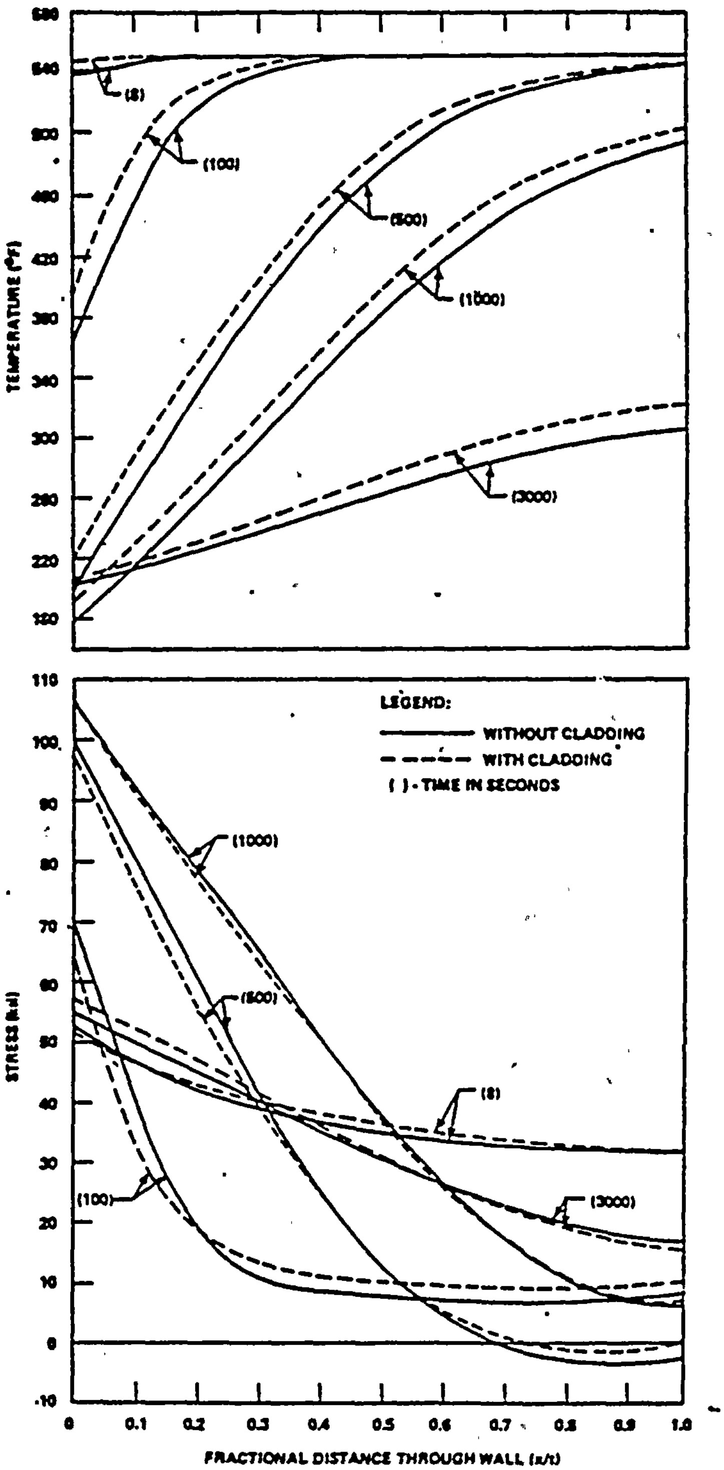


Figure 4-7 Variation of Temperatures and Stresses through the Thickness at Various Times for Both the Unclad and Clad Conditions of the Inlet Nozzle Corner Location during the Reference LSB

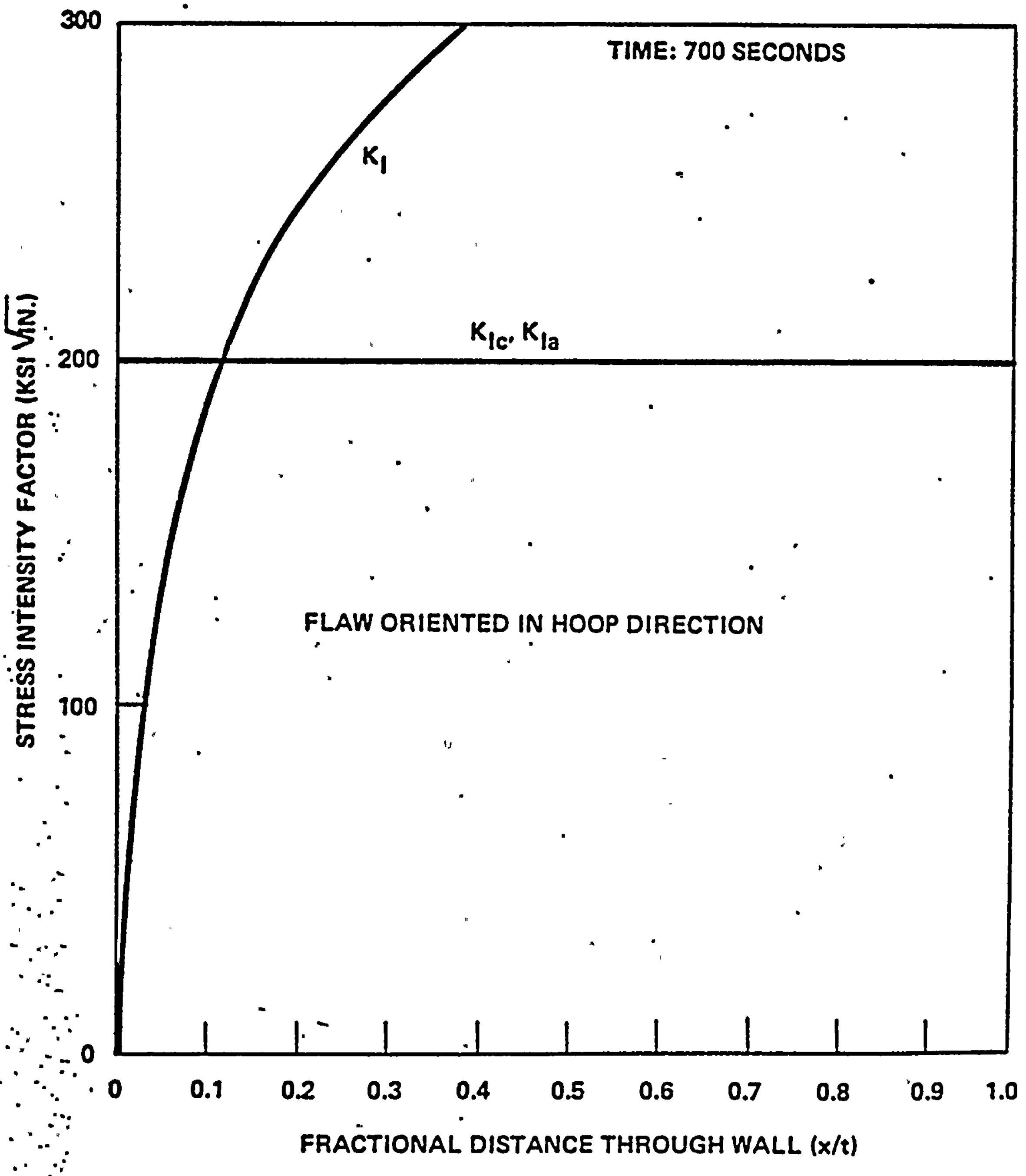


Figure 4-8 Variation of Stress Intensity Factor and Failure Criteria through the Thickness at the Critical Initiation Time for the Unclad Inlet Nozzle Corner Location

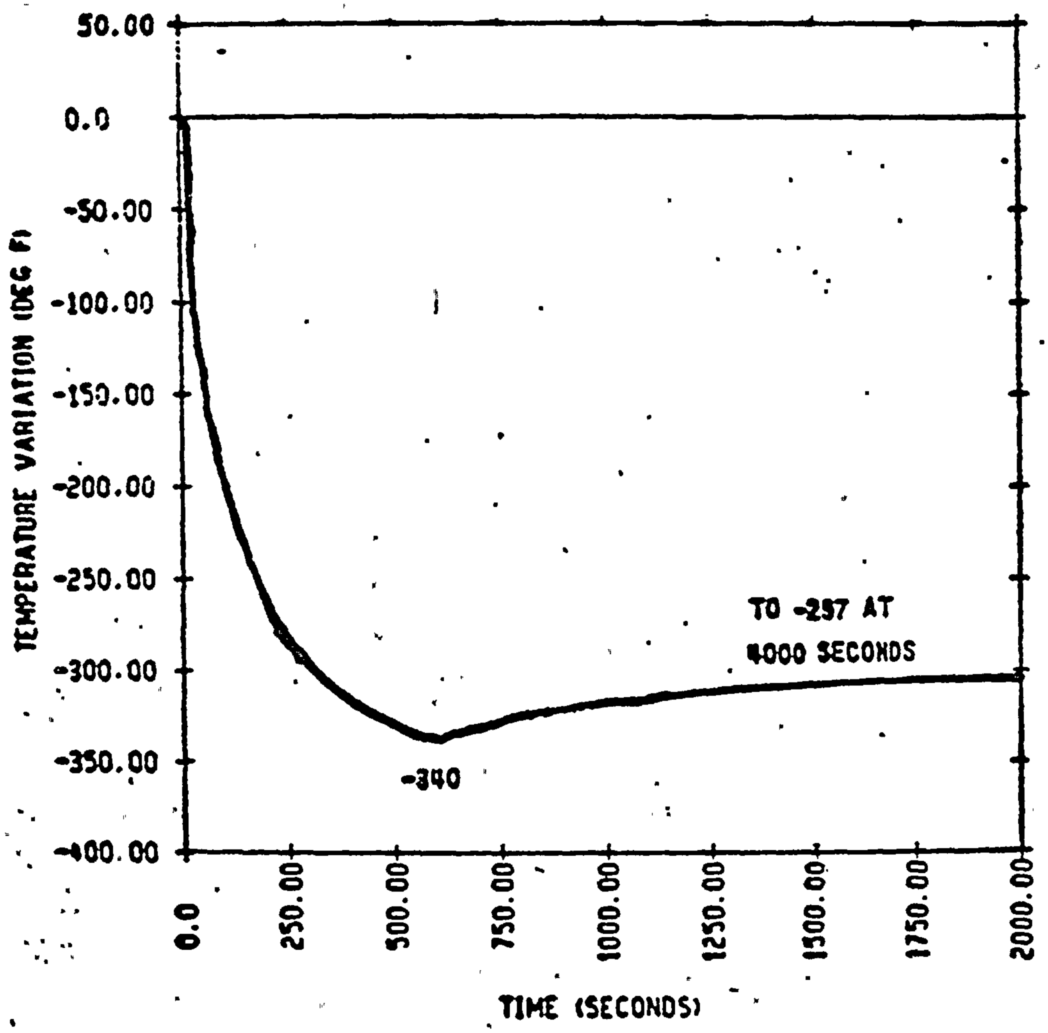


Figure 4-9 Cold Leg Temperature History
for Reference LSB

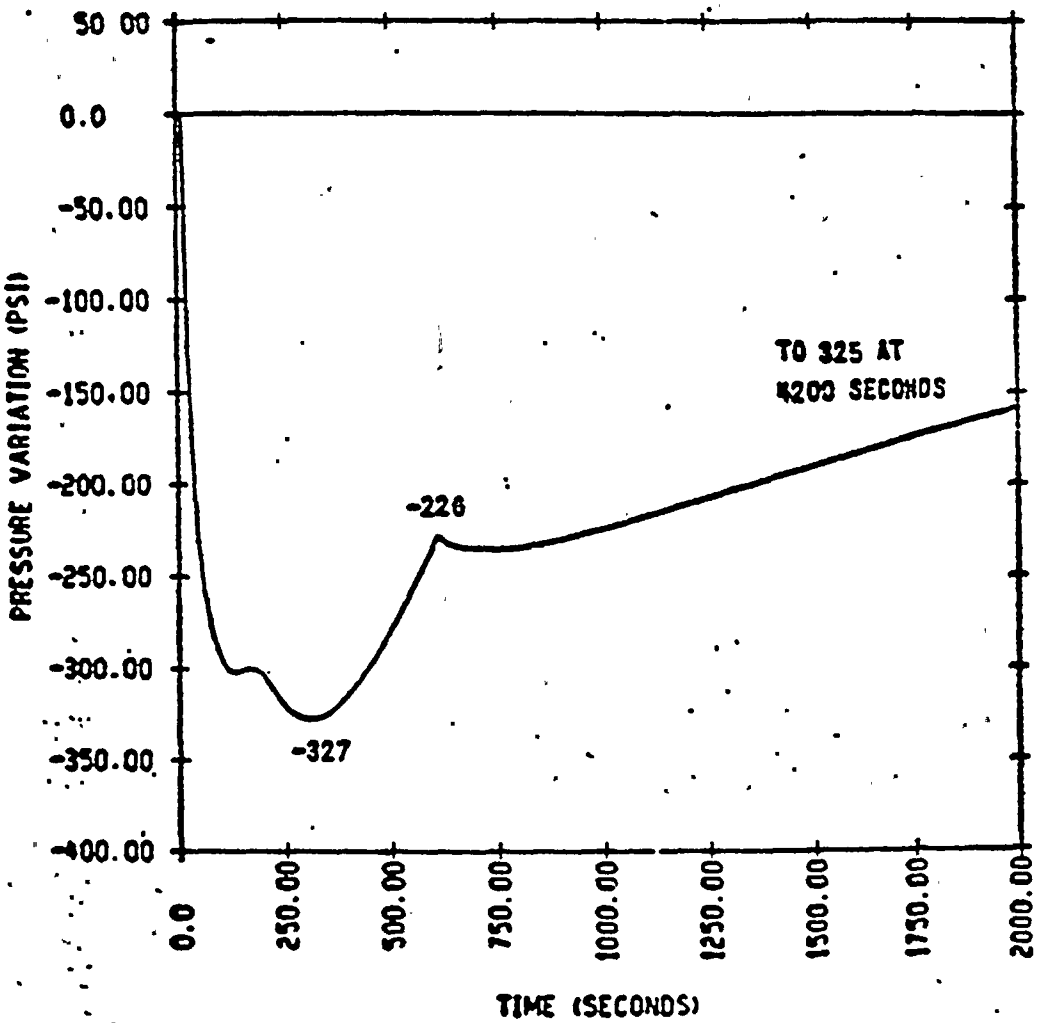


Figure 4.10 Reactor Coolant Pressure History - for Reference LSB

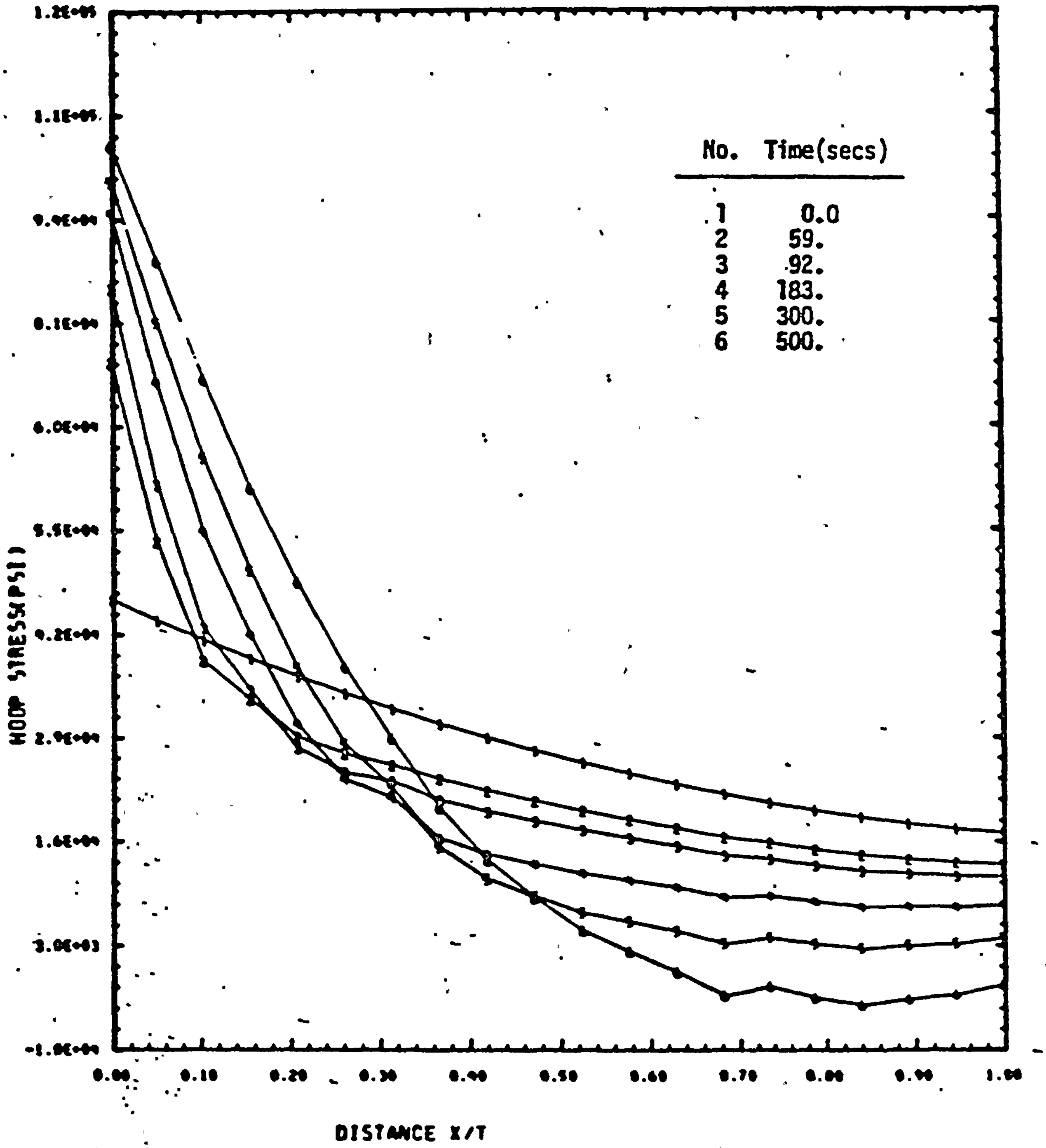


Fig. 4-11 Hoop Stress Profiles, Inlet Nozzle Corner, Reference LSB

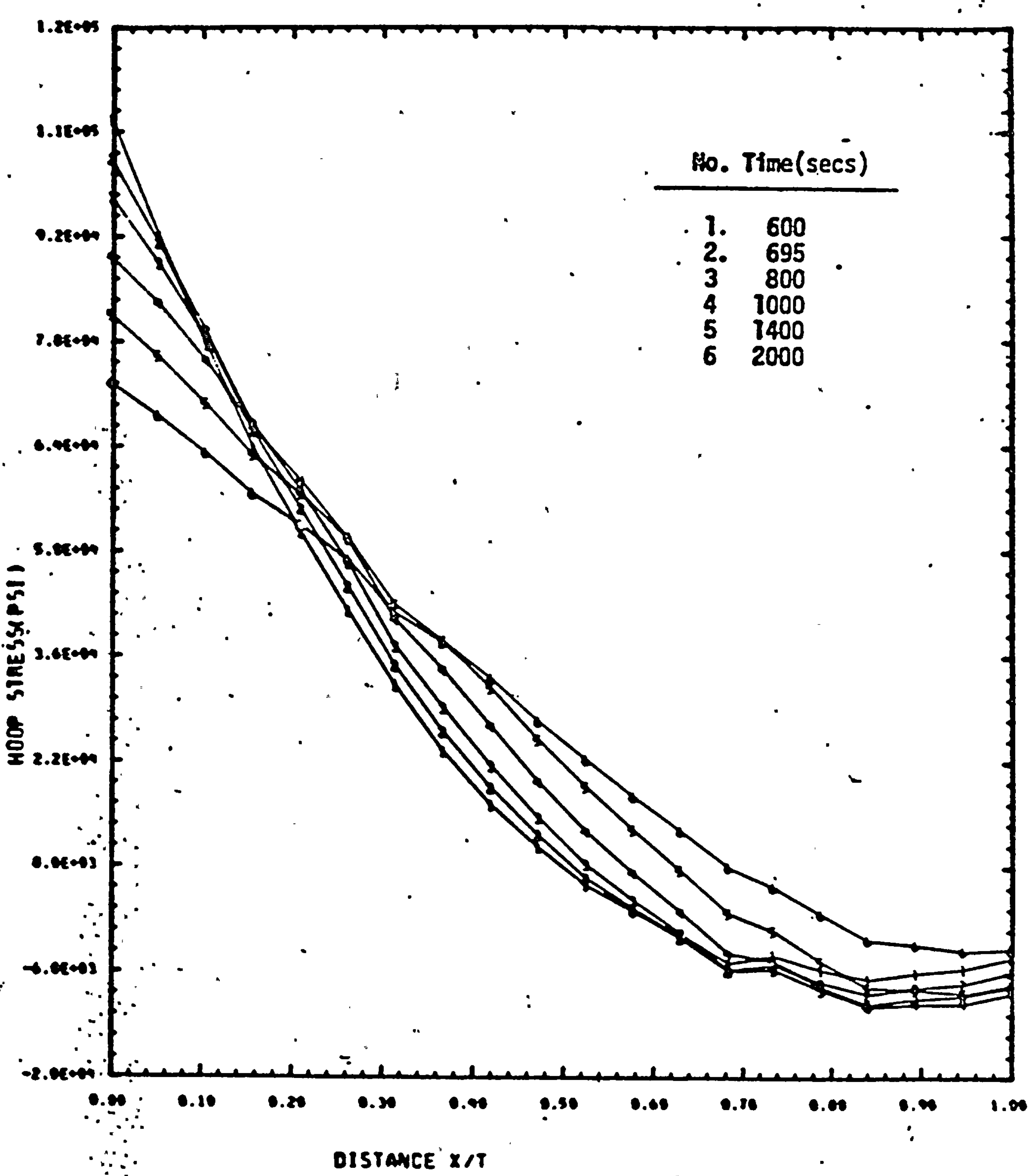


Fig. 4-12 Hoop Stress Profiles (cont'd) for Inlet Nozzle Corner, Reference LSB

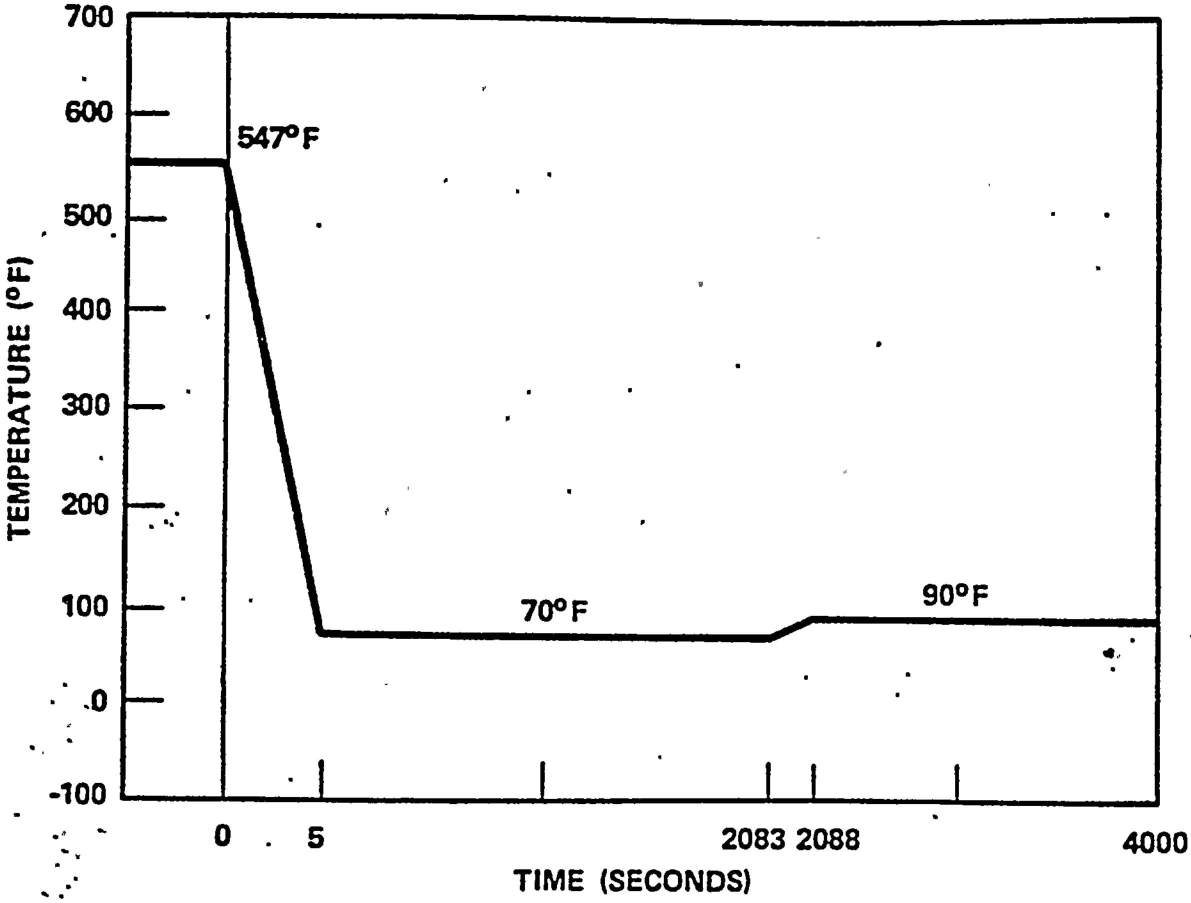


Figure 4-13 Cold Leg LOCA Temperature History for Reference Loss of Coolant Transient

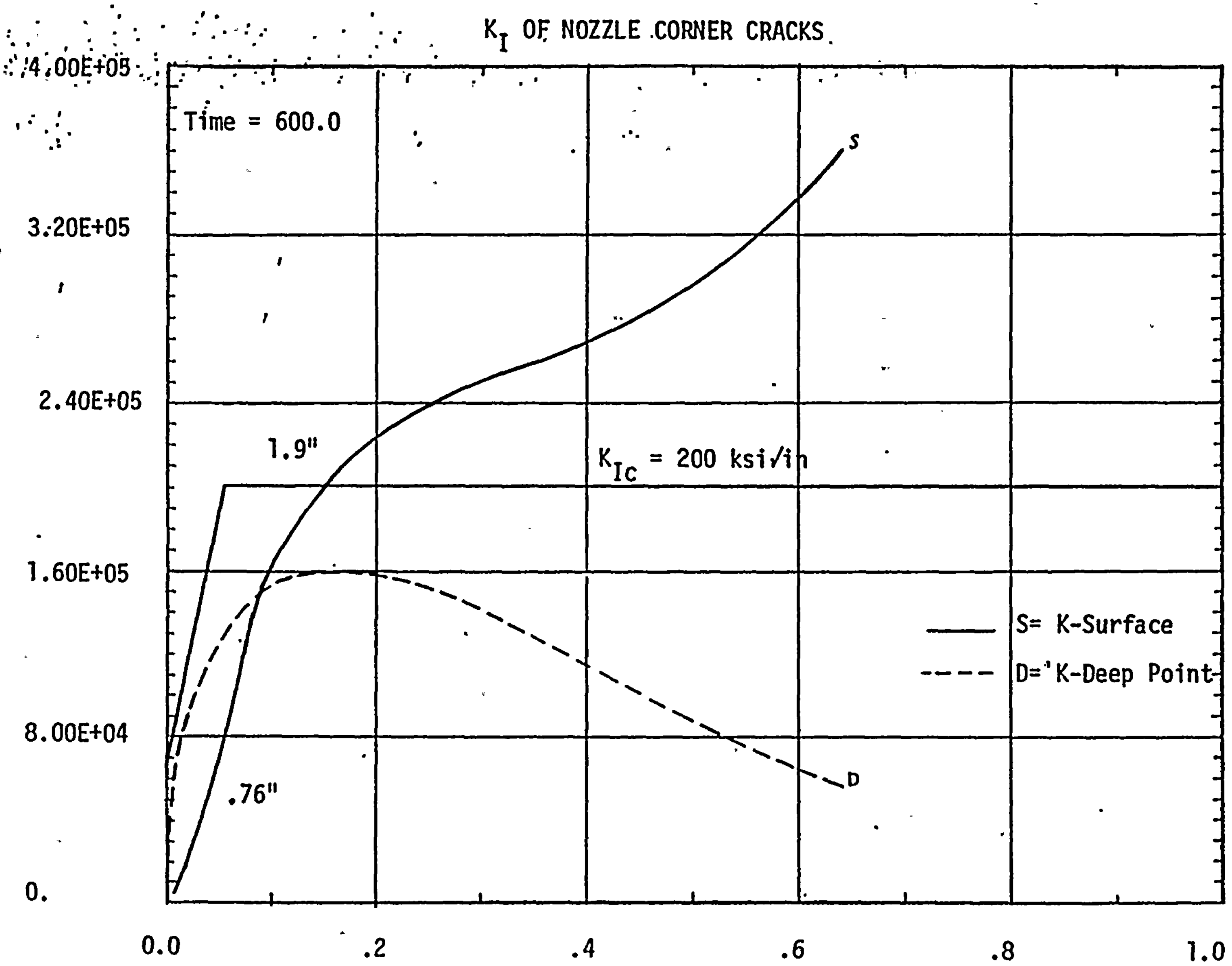


Figure 4-14 Results of Fracture Analysis of January 25 Incident, Reactor Vessel Inlet Nozzle Corner, assuming no mixing

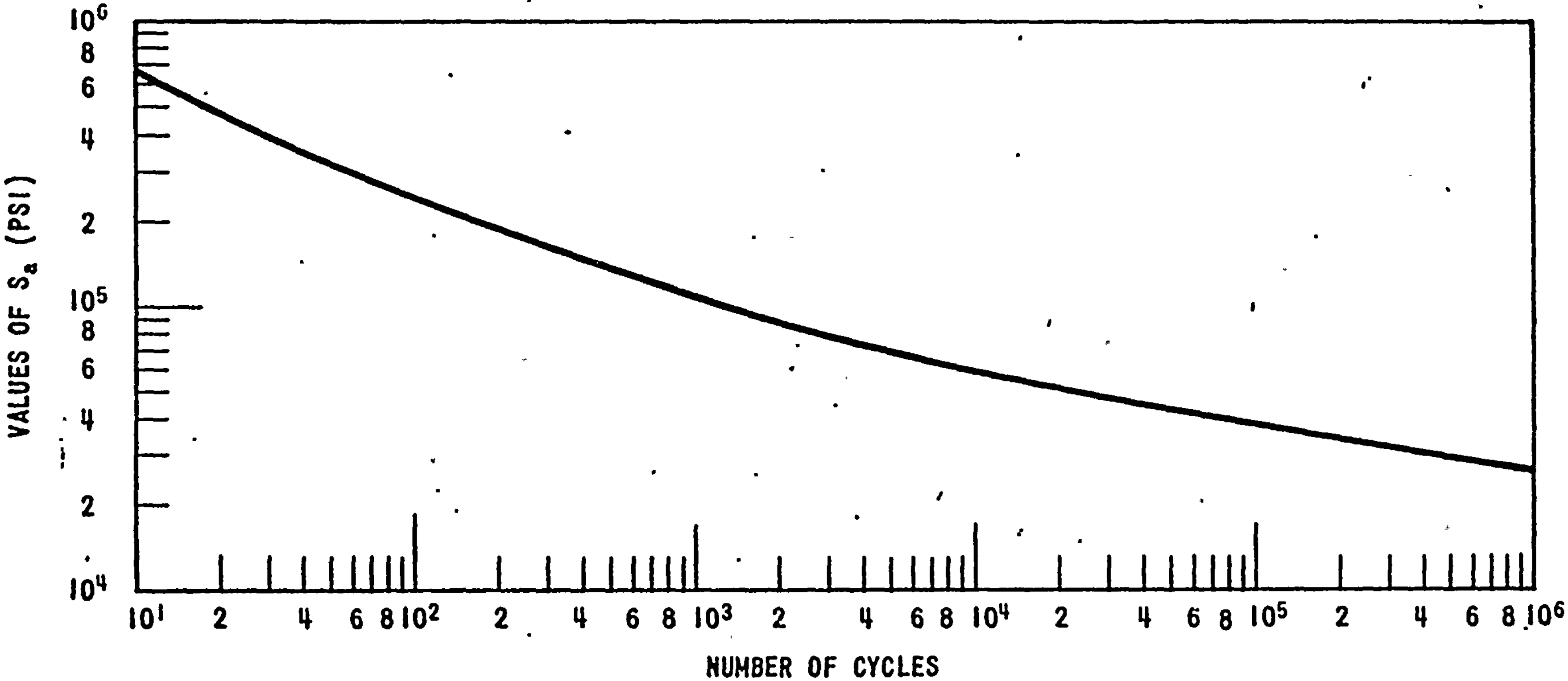


Figure 4-15 Design Fatigue Curve for Austenitic Steels (Nickel-Chromium-Iron Alloy, Nickel-Iron-Chromium Alloy and Nickel-Copper Alloy)

6284-14

5.0 SUMMARY AND CONCLUSIONS

An integrity analysis has been completed of the January 25, 1982 transient on the Ginna plant for the following regions:

1. Reactor vessel beltline
2. Reactor vessel inlet nozzle
3. Safety Injection nozzle

For the beltline a detailed analysis was made assuming two extremes in fluid mixing, i.e., perfect mixing and no mixing. For the inlet nozzle, the evaluation was based on existing analyses performed on other plants again assuming two extremes in mixing. It is judged that detailed analyses will confirm the conclusions drawn. The safety injection nozzle was evaluated by referring to a previous analysis that is applicable to this transient. The analyses followed the general methods outlined in WCAP 8510. [8]

The results of the evaluations indicate that the January 25, 1982 transient did not impair the integrity of the regions evaluated with either no crack initiation or calculated large critical flaw sizes.

6.0 REFERENCES

1. U.S. Nuclear Regulatory Commission Regulatory Guide 1.99, Revision 1, April 1977.

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GINNA 50-244

Subject: Steam generator tube rupture

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