VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

March 6, 1990

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No.	90-094
NO/JZL:	
Docket No.	50-280
	50-281
License No.	DPR-32
	DPR-37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
RUPTURE OF A MAIN STEAM PIPE
PROPOSED OPERATING LICENSE AMENDMENT

Licensee Event Report (LER) No. 89-043-00 identified that the instrument loop uncertainty associated with the low pressurizer pressure safety injection setpoint may exceed the margin between the setpoint assumed in the accident analysis and the actual setpoint value under adverse environment conditions. The instrument loop uncertainty in an adverse environment is sufficiently large such that safety injection initiated by low pressurizer pressure could not be assured. The existing analysis for the main steam line break takes credit for low pressurizer pressure safety injection in certain small steam line break scenarios.

As noted in LER 89-043-00, a formal analysis was performed in which no credit was taken for low pressurizer pressure safety injection. This analysis confirms that other sources of automatic safety injection provide adequate protection when required, or that no safety injection is necessary.

During our review of the instrument loop uncertainty associated with LER 89-043-00, it was determined that the probability of malfunction of the low pressurizer pressure safety injection had increased, and hence, an unreviewed safety question existed as defined by 10 CFR 50.59. Accordingly, our analysis which shows that low pressurizer pressure safety injection is not required for small steam line breaks is being submitted for NRC approval per 10 CFR 50.59(c); along with a request for the appropriate license amendment per 10 CFR 50.90. Attachment 1 is a copy of our aforementioned analysis and Attachment 2 is our proposed amendment to the operating licenses for Surry Units 1 and 2.

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This request has been reviewed and approved by the Station Nuclear Safety and Operating Committee and the Corporate Nuclear Safety Staff. It has been determined that the proposed change does not constitute a significant hazard as defined in 10 CFR 50.92. The basis for our determination of no significant hazard is provided as Attachment 3.

Very truly yours,

W L Stewart

W. L Stewart
Senior Vice President - Nuclear

Attachments
1. Analysis of Small Steamline Break Performance Without Low Pressurizer Pressure Safety Injection, Surry Units 1 and 2
2. Proposed Surry Units 1 and 2 Operating License Amendments
3. Significant Hazards Determination

cc: U.S. Nuclear Regulatory Commission
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Mr. W. E. Holland
NRC Senior Resident Inspector
Surry Power Station

Commissioner
Department of Health
Room 400
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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by W. L. Stewart who is Senior Vice President - Nuclear, of Virginia Electric and Power Company. He is duly authorized to execute and file the foregoing document in behalf of that Company, and the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 6TH day of March, 1990.

My Commission Expires: May 31, 1994.

Vicki L. Hull
Notary Public

(SEAL)

ATTACHMENT 1

Analysis of Small Steamline Break Performance Without Low Pressurizer Pressure Safety Injection Surry Units 1 and 2

ATTACHMENT 1

ANALYSIS OF SMALL STEAMLINE BREAK PERFORMANCE WITHOUT LOW PRESSURIZER PRESSURE SAFETY INJECTION SURRY UNITS 1 AND 2

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1.0 INTRODUCTION

In References 1 and 2, Virginia Electric and Power Company identified the potential for the low pressurizer pressure safety injection (SI) function to be delayed relative to the safety analysis assumptions or to not function for events which result in a harsh environment inside containment. It has been shown that the error in the pressure transmitter output signal induced by a harsh environment may exceed the difference between the safety injection actuation setpoint and the bottom of the pressure range over which the instrument is calibrated (i.e., the channel span). In such a case actuation of safety injection cannot be guaranteed. This results from the fact that the transmitter output is highly nonlinear for input pressures which are significantly outside the calibration span. A summary of the calculated pressurizer pressure channel errors and the available margin between the actuation setpoint and the bottom of the calibration span is shown in Table 1.

As discussed in References 1 and 2, an evaluation showed that existing safety analyses remain bounding for both large and small break Loss of Coolant Accidents (SBLOCA) and for the large, hypothetical main steam line break events. The evaluation further concluded that the results for a small steam line break inside containment would continue to be acceptable, but that safety injection actuation might be delayed with respect to the assumption in the safety analysis. As a result, an unreviewed safety question was determined to exist in that the probability of a malfunction of equipment important to safety has increased with respect to the currently reviewed and approved licensing basis. The purpose of this analysis is to present the technical information required for NRC resolution of the unreviewed safety question. Following NRC approval, the analysis results will be incorporated into the UFSAR.

For the SBLOCA, the low pressurizer pressure SI occurs at times less than one minute into the transient. For the current SBLOCA licensing analysis, the low pressure reactor trip time is 21.2 seconds for a 3 inch break and 13.4 seconds for a 4 inch break. Low-low pressure SI is generated very shortly after the trip (i.e., within a few seconds). The mass and energy releases for the SBLOCA are approximately an order of magnitude less than the mass and energy releases for the LBLOCA. For the double ended cold leg guillotine large break LOCA, it takes about 20 seconds to reach the peak containment temperature. Thus, the anticipated rise in containment temperature for the SBLOCA is minimal for the time interval in which the pressurizer low pressure SI is expected to actuate. In addition, the temperature time constant for the pressure transmitters being used, Rosemount Model 1153 Series D, is 4.8 minutes. Based on this information, it is expected that the pressure transmitters would function in what is basically a mild environment. Therefore it is not necessary to apply the harsh environment errors to generation of this SI signal for SBLOCA.

For the large main steam line breaks examined in the UFSAR, safety injection is initiated based on secondary side indications and not on low pressurizer pressure.

The small steam line break event currently analyzed and presented in the UFSAR (Reference 3) corresponds to a 247 lb/sec steam release rate at 1100 psia. This break size was chosen to bound the effects of a stuck open secondary relief, dump or safety valve. It is of interest to note that none of these events, which are normally classified as ANS Condition II, or anticipated transients, would result in a harsh containment environment since the release point is outside containment.

However, while such a case is not specifically analyzed in the UFSAR, one could postulate a small steam line break inside containment which results in releases equivalent to or greater than the UFSAR "credible" case. In this case, a harsh environment could be created and the low-low pressurizer pressure safety injection might not function for the reason described in References 1 and 2. In this case safety injection initiation would result from one of the following (see Table 2):

1) High steam flow coincident with either low RCS Tavg or low steam pressure for break sizes in excess of about 0.2 sq. ft. per loop.

2) High header to steam line delta-P (expected to function if the main steam non-return valves close as designed).

3) High containment pressure.

Virginia Power has performed a bounding analysis of a small steam line break inside containment to show the impact of failure of the low pressurizer pressure SI function in this case. The analysis is summarized in this report.

2.0 METHOD OF ANALYSIS

The following assumptions were made for the small SLB analysis:

1) The low-low pressurizer pressure SI is assumed not to be available for a break inside containment.

2) A break size is selected which is just underneath the size which would actuate high steam flow protection (see Table 2). The case being examined is a break at hot zero power (HZP) at end of life for the standard reasons discussed in the UFSAR. The break examined was a 0.19 sq. ft. per loop split break.

3) A conservative estimate of safety injection actuation on High-1 containment pressure was made based on existing North Anna containment analyses (Ref. 4). Examination of the available data shows that the time to High-1 containment pressure actuation can be correlated well with the inverse of the initial break flow. A plot of such a correlation is shown in Figure 1. Also shown on the Figure is the value of the correlating parameter (39,000 lbm/sec divided by initial break flow rate) for the 0.19 sq. ft. per loop break. The estimated Hi-1 actuation time is about 30 seconds. The North Anna data is considered a valid approximation for Surry based on the similarities in containment design, as demonstrated by a detailed comparison of the two designs. However, in order to conservatively account for uncertainties, the actuation time from Figure 1

was doubled and a time of 60 seconds was used.

A detailed review of the Reference 4 analysis was performed to confirm the applicability of the analysis results to the current evaluation. It was concluded that the methods and assumptions used maximize the calculated time to High-1 actuation, and the use of these actuation times in the current analysis is appropriate and conservative. Nevertheless the additional 100% factor discussed above was applied.

4) Safety injection pump acceleration and valve stroke was conservatively modeled to take 15 seconds.

5) The safety injection flow capability corresponding to the operation of one HHSI/charging pump (minimum safeguards) was assumed. The time delay to purge boron-free water in the piping from the RWST to the cold legs was modeled. The minimum RWST boron concentration of 2000 ppm was assumed, corresponding to the Technical Specification 3.3 limit.

6) The non-return valves in the main steam lines were assumed NOT to function. This is consistent with way the High-1 actuation time (see Assumption 3) was estimated. If the NRV's function, the total amount of secondary inventory available for discharge is significantly reduced, and safety injection on header-to-line DP would be expected very quickly (in about

20 seconds for the 0.19 sq. ft. per loop break).

The analysis was performed using the Virginia Power RETRAN two loop model documented in Reference 5 and the RETRAN02, MOD003 transient analysis code documented in Reference 6.

Table 1
Presentation of Pressurizer Pressure Channel Errors and Margins

	Bottom of Channel Span, psia	Nominal Safety Injection Setpoint, psia	Margin, psi	Total Channel Statistical Allowance (%/Psi)* Normal Env.	Total Channel Statistical Allowance (%/Psi)* Harsh Env. #
A.	SURRY UNITS 1 AND 2				
	1700.0	1718.0	18.0	+,-1.97/15.8	+19.60/156.8 -18.10/-144.8
B.	NORTH ANNA UNITS 1 AND 2				
	1700.0	1765.0	65.0	+,-1.76/14.1	+19.7/157.7 -17.9/-143.6

* Error (psi) = Error (% Span) x 800 psi span/100

Based on vendor's stated performance characteristics for operation during a design bases event (DBE). Applicability of these performance characteristics to Surry and North Anna has been demonstrated in accordance with 10 CFR 50.49 and associated regulatory guidance.

Table 2
Steam Line Break Protection for Surry

A. Safety Injection

Source	Tech. Spec Setpoint*	Notes
1. Low-low pressurizer pressure-2/3 channels	1700 psig	*May not function for breaks inside containment (Ref. 1).
2. High steam line flow in 2/3 line coinc w/	40% of full load steam flow (at no-load)	*May not actuate for
A) Low-low RCS Tavg, or	541 F	a) breaks inside containment if NRV's close
B) Low Steam Line Pressure	600 psig	b) breaks less than about 0.2 sq. ft. per loop.
3. High delta-P between steam line and header	150 psid	* May not actuate for break inside containment if NRV's don't close
4. High-1 containment pressure	5 psig	* Will not actuate for breaks outside containment

* Nominal setpoint. The impact of uncertainties is included in the analysis.

Table 2 (CONT.)
Steam Line Break Protection for Surry

B. Steam Line Isolation

Source	Tech. Spec Setpoint*	Notes
1. High steam line flow in 2/3 line coinc w/	40% of full load steam flow (at no-load)	*May not actuate for
A) Low-low RCS Tavg, or	541 F	a) breaks inside containment if NRV's work
B) Low Steam Line Pressure	500 psig	b) breaks less than about 0.2 sq. ft. per loop.
2. High-2 containment pressure	10.3 psig	* Will not actuate for breaks outside contaiment

* Nominal setpoint. The impact of uncertainties is included in the analysis.

3.0 ANALYTICAL RESULTS

Results of the analysis are shown in Figures 2-11. Because the case modeled is a symmetric blowdown, results for the other steam generators and reactor coolant loops are essentially the same. A sequence of events is presented in Table 3.

Figure 2 shows the total break flow coming from the two "intact" steam generators. The flow path is from the generators to the header, then in a reverse direction back through the NRV in the faulted loop to the break. If the NRV were to close as designed, the SI signal on high header to line differential pressure (See Table 2) would be rapidly generated.

This can be seen clearly from Figure 3. The header to line differential pressure setpoint is 150 psid (Table 2). Figure 3 shows that the pressure in a single loop drops by 150 psi very quickly (about 20 seconds in this case). Thus from the standpoint of delaying safety injection and increasing the severity of the cooldown, failure of the NRV's to close is the limiting assumption.

Core heat flux is shown in Figure 4. The peak heat flux is reached at about 280 seconds. The turnaround in heat flux results from a combination of the continued decay in steam flow (Figure 2), a leveling off of the secondary pressure decay (Figure 3) and the negative reactivity effects of soluble boron addition to the core via safety injection (Figure 10). The peak heat flux attained is only about 60% of that resulting from the current limiting hypothetical case (inside break with offsite power available) presented in the UFSAR (see Table 4 for a detailed comparison).

Figure 5 shows the total core kinetics reactivity in dollars. The design end of cycle (EOC) shutdown reactivity of 1.77% was assumed to be inserted at the beginning of the transient. Recriticality occurs at about 100 seconds. A reactivity balance was performed which showed that with a more realistic shutdown margin assumption (representative of recent reload cores), recriticality would most likely not occur at all.

The cold leg temperature on the faulted loop is shown in Figure 6. The intact loop temperature response is essentially identical since the case being modeled is a symmetrical blowdown.

Figure 7 shows the pressurizer pressure response. The initial rapid depressurization is retarded somewhat as the pressurizer drains and the upper head begins to flash. The effect of safety injection terminates the depressurization at about 200 seconds and a gradual repressurization begins.

Hot leg temperature is shown in Figure 8 and is very simlar to the cold leg response (Figure 6). Figure 9 shows the faulted loop steam generator inventory. At the end of ten minutes about 60% of the initial inventory has been discharged. This would result in significant dryout effects on the secondary side, which would tend to retard heat transfer and hence the cooldown rate. This effect has not been modeled. Instead the standard Virginia Power assumption of a constant heat transfer coefficient representative of nucleate boiling over the entire secondary surface of the tubes has been made. This conservatively accentuates the cooldown rate.

Figure 10 shows the boron concentration at the inlet plenum, as previously discussed. Figure 11 shows core power, which correlates closely with core heat flux (Figure 4).

Table 4 provides a comparison of the limiting statepoint (maximum core heat flux) for the current case with the UFSAR hypothetical break (inside containment with offsite power). From a DNB perspective, the hypothetical case will bound the current case for the following reasons:

- the heat flux for the UFSAR case is more than 165% of that for the current (0.19 sq. ft. per loop) case.

- The inlet temperature gradient across the core is negligible for the current case, where the UFSAR case has over a 70 F gradient. A large temperature gradient results in higher radial power peaking.

- RCS pressures are comparable for the two cases (within 100 psi).

As a result of these factors, and because of the inherent conservatism of the current analysis, particularly in the area of the overall core reactivity calculation, as discussed previously, explicit power peaking and DNB calculations are not required for this case. Since the UFSAR case meets the ANS Condition II DNBR criterion with margin, so will the current case.

The analysis case presented above is expected to bound those which would result for other break sizes. As discussed previously, larger break sizes will generate a high steam flow/ low steam pressure or /low Tavg signal much earlier than the Hi-1 actuation assumed in the above case. For smaller break sizes, the Hi-1 containment actuation signal would be delayed due to the lower mass and energy releases to the containment. However, the cooldown and return to power would be expected to be less severe for the smaller breaks. In the limit, there will exist some break size small enough such that Hi-1 actuation will not occur.

Review of available North Anna containment analysis data shows that the smallest break size examined yielded an initial break flow of 430 lb/sec and resulted in Hi-1 actuation at 97 seconds. This break flow corresponds to a RETRAN break area of about .064 sq. ft. per loop.

An additional case was examined to show the effects of no SI on very small steam breaks. A break size of .064 sq. ft. per loop, as discussed above, was simulated. No SI was assumed, although the North Anna containment results showed SI on Hi-1 actuation would occur. The most significant results are shown in Figures 12-14. As expected, the small break size limits the amount of power generation that can be achieved in the core (Figure 12). As a result, the conclusions drawn for the .19 sq. ft. per loop case are valid for the entire spectrum of break sizes.

FIGURE 1

HI-1 ACTUATION TIME

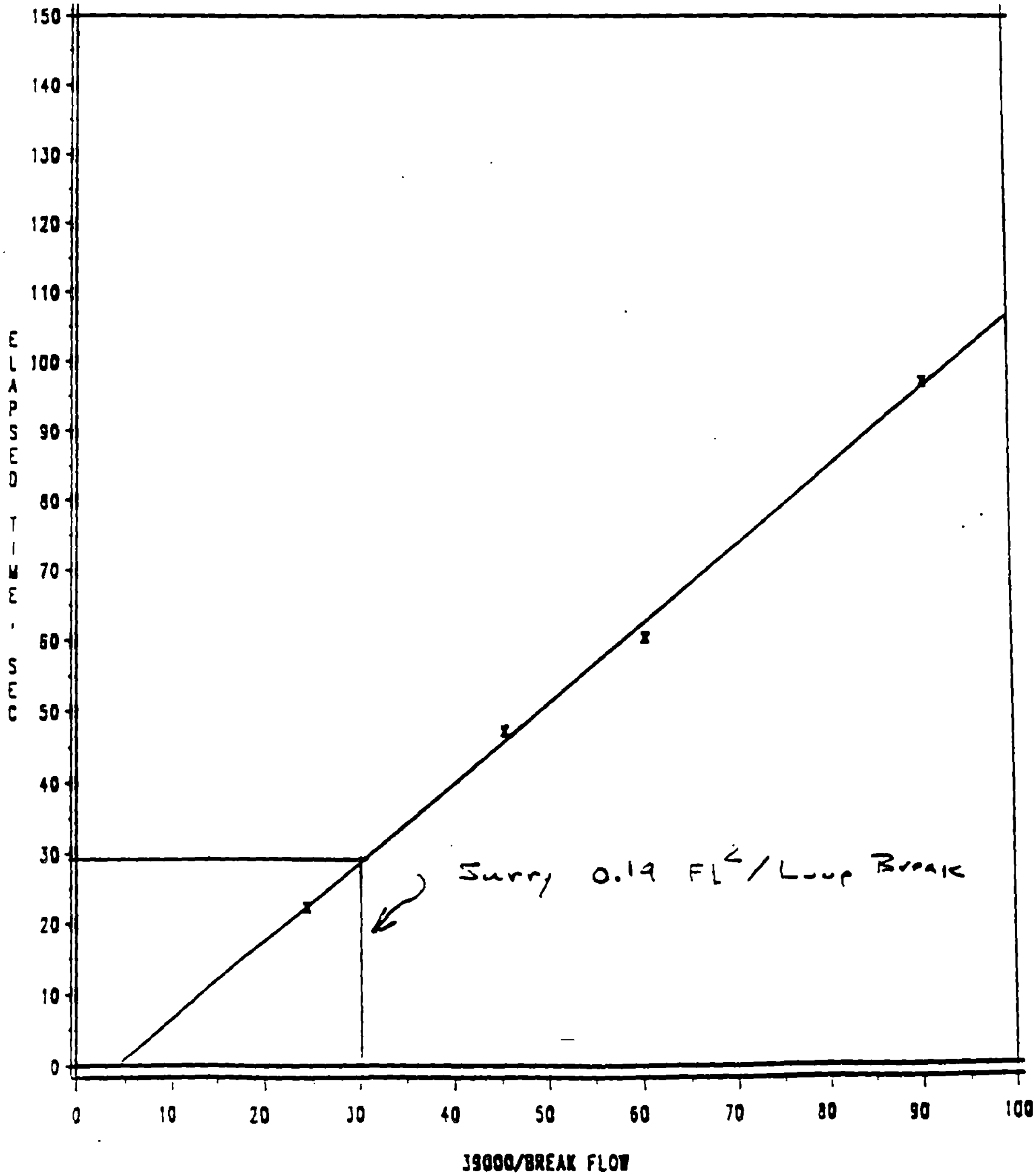

FIGURE 2
BREAK FLOW FROM TWO STEAM GENERATORS
0.19 SQ. FT./LOOP BREAK

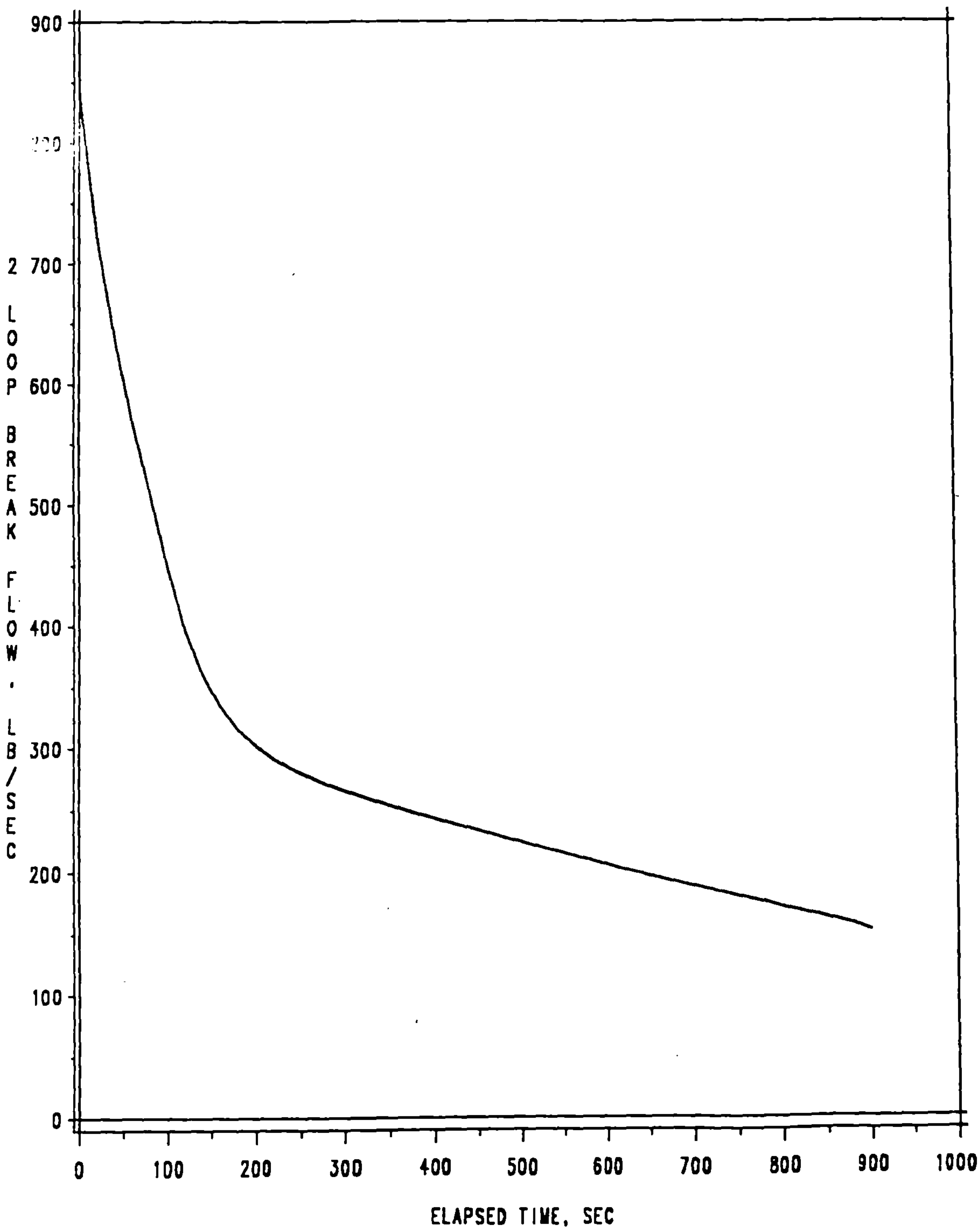

FIGURE 3
SINGLE LOOP STEAM PRESSURE
0.19 SQ. FT./LOOP BREAK

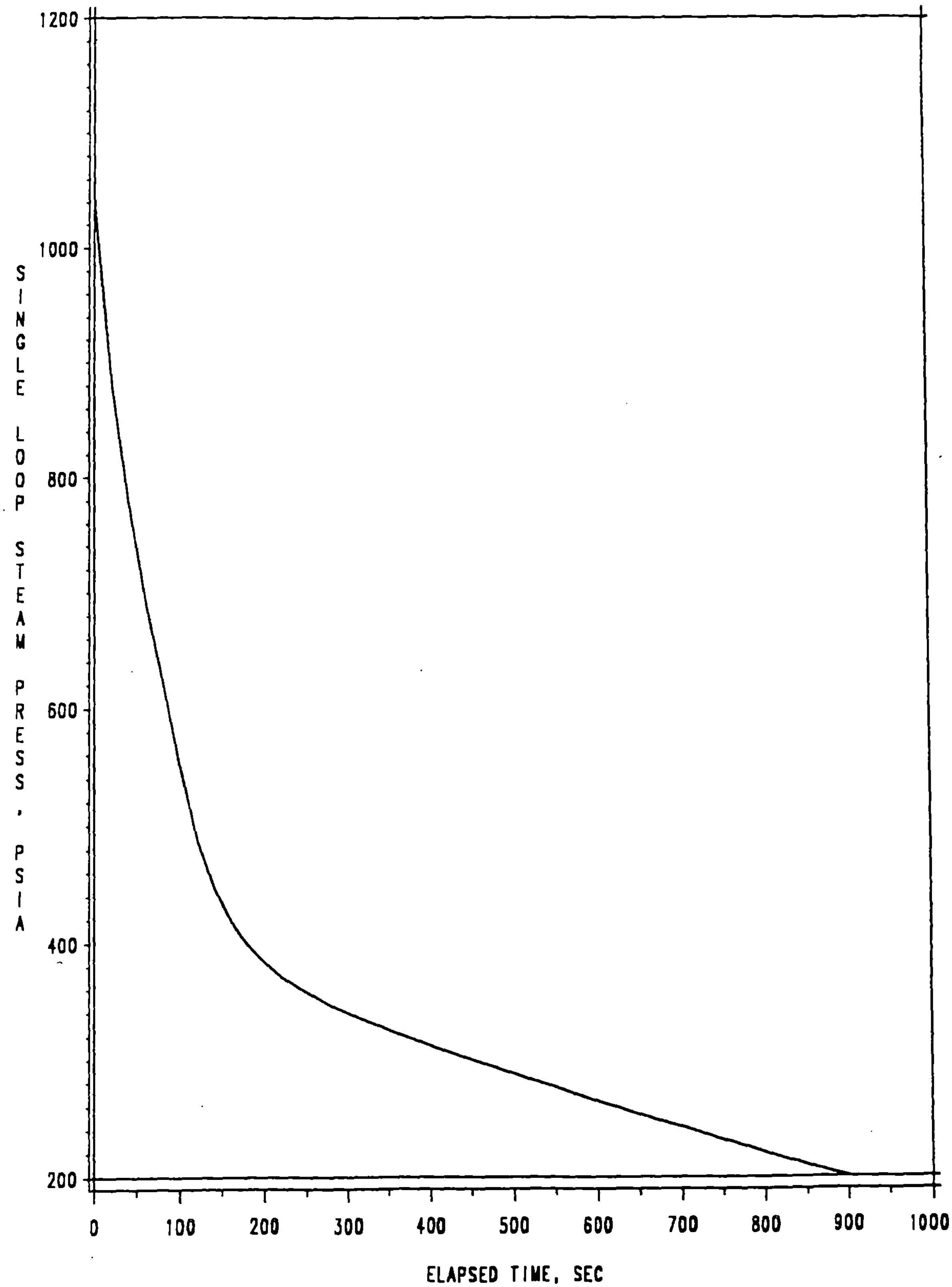

FIGURE 4
CORE HEAT FLUX (FRACTION OF HFP)
0.19 SQ. FT./LOOP BREAK

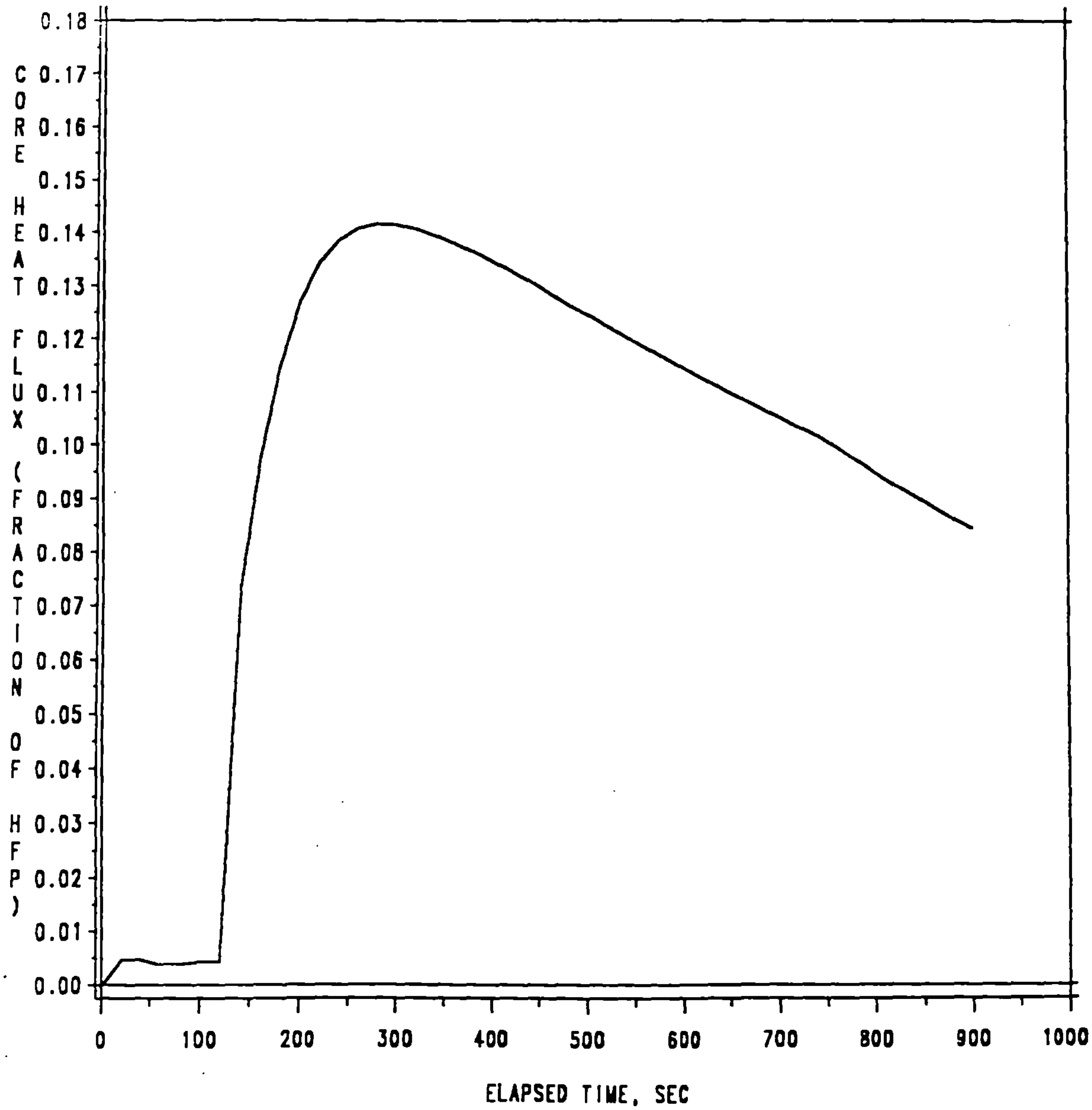

FIGURE 5
TOTAL REACTIVITY ($)/BETA=.0044
0.19 SQ. FT./LOOP BREAK

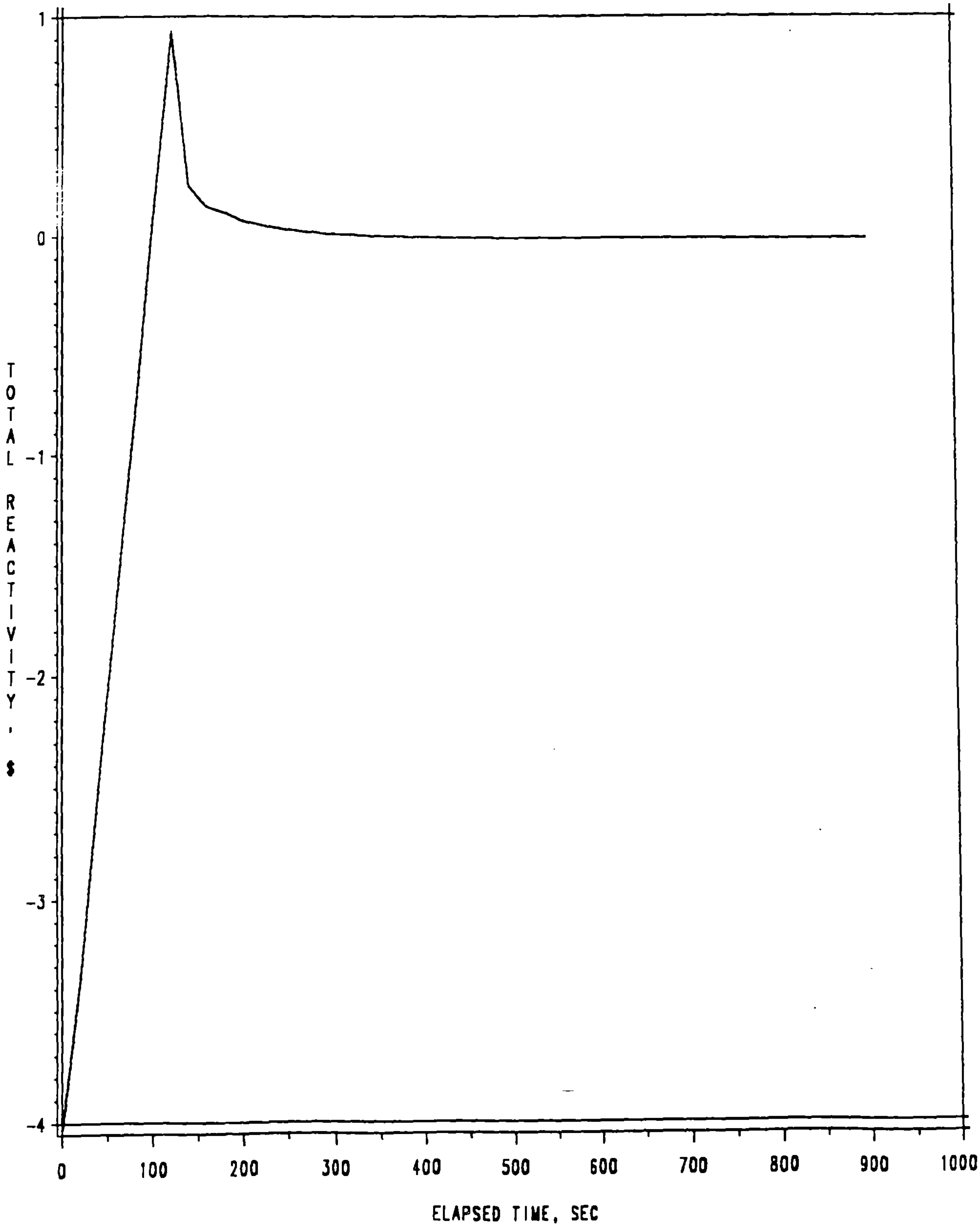

FIGURE 6
1–LOOP COLD LEG TEMPERATURE
0.19 SQ. FT./LOOP BREAK

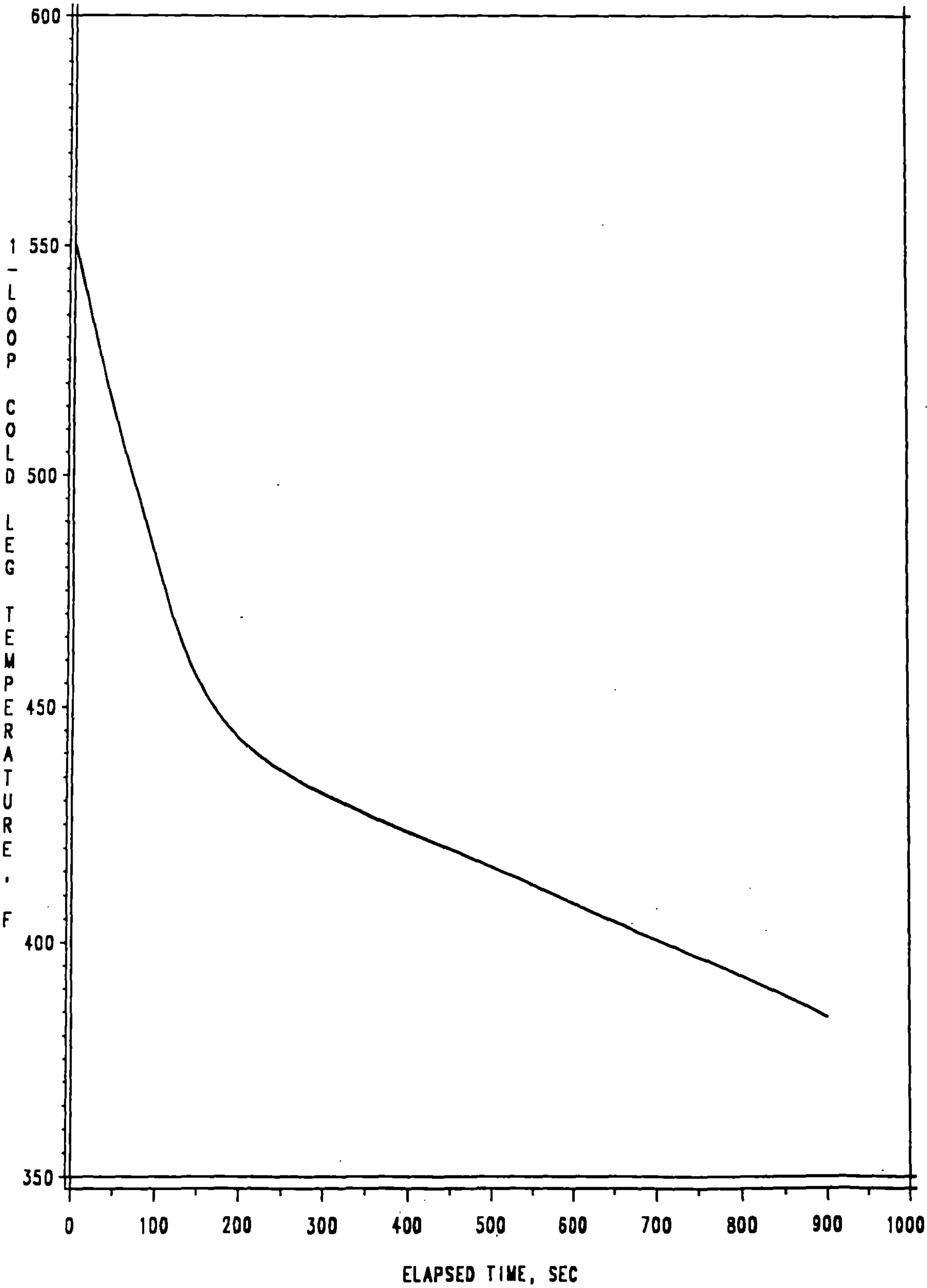

FIGURE 7
PRESSURIZER PRESSURE
0.19 SQ. FT./LOOP BREAK

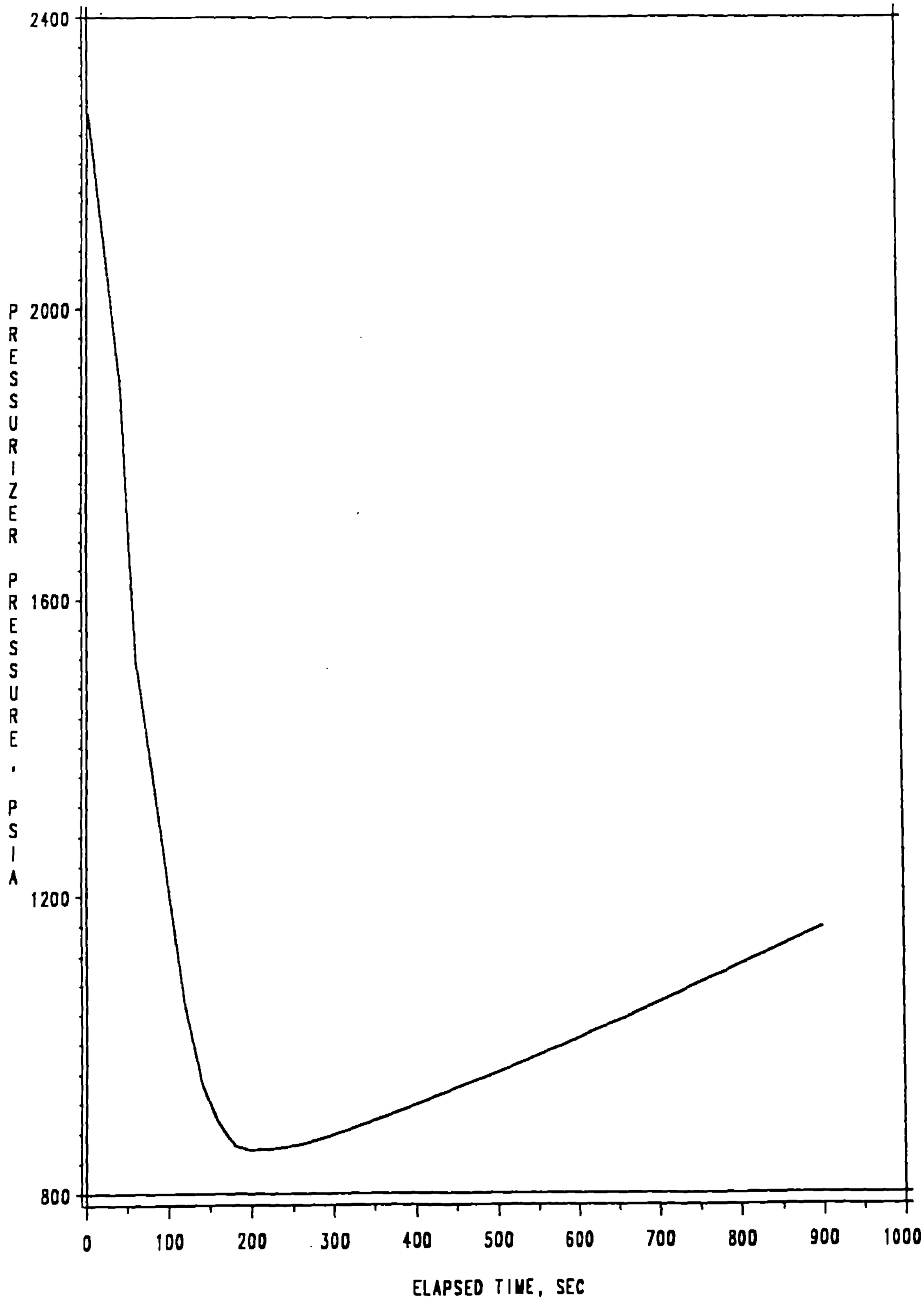

FIGURE 8
1-LOOP HOT LEG TEMPERATURE
0.19 SQ. FT./LOOP BREAK

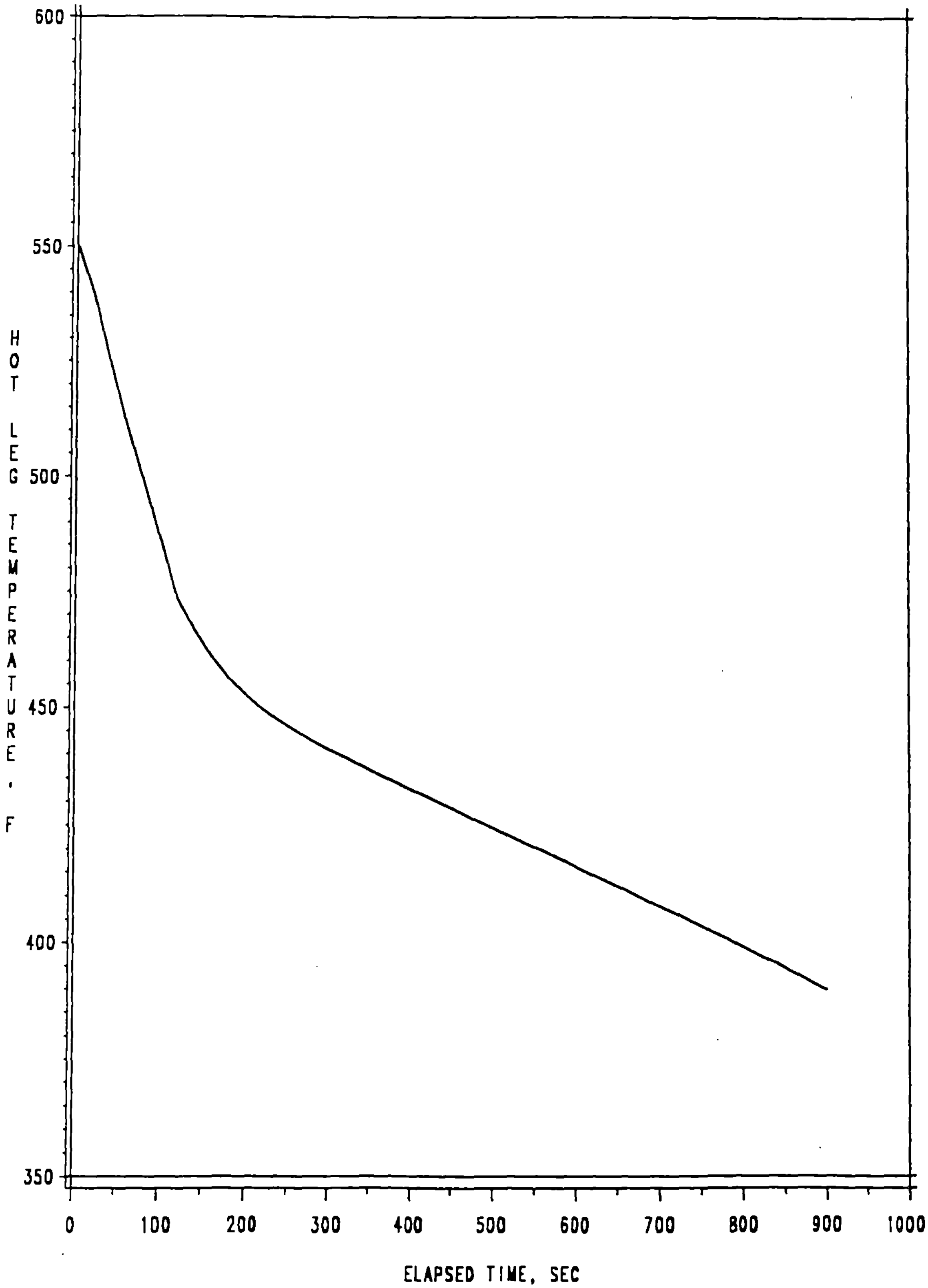

FIGURE 9
1-LOOP SG INVENTORY
0.19 SQ. FT./LOOP BREAK

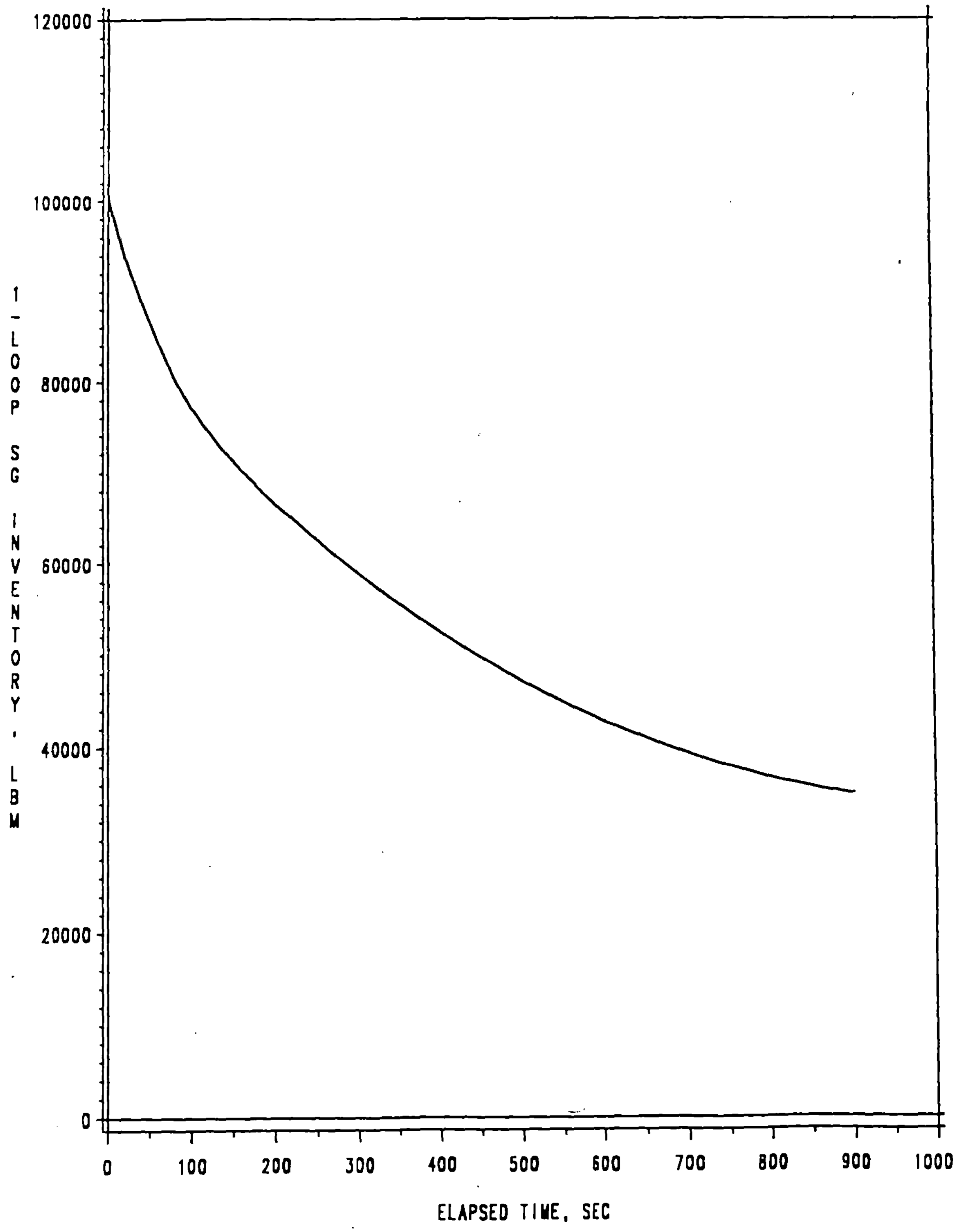

FIGURE 10
CORE INLET BORON CONCENTRATION, PPM
0.19 SQ. FT./LOOP BREAK

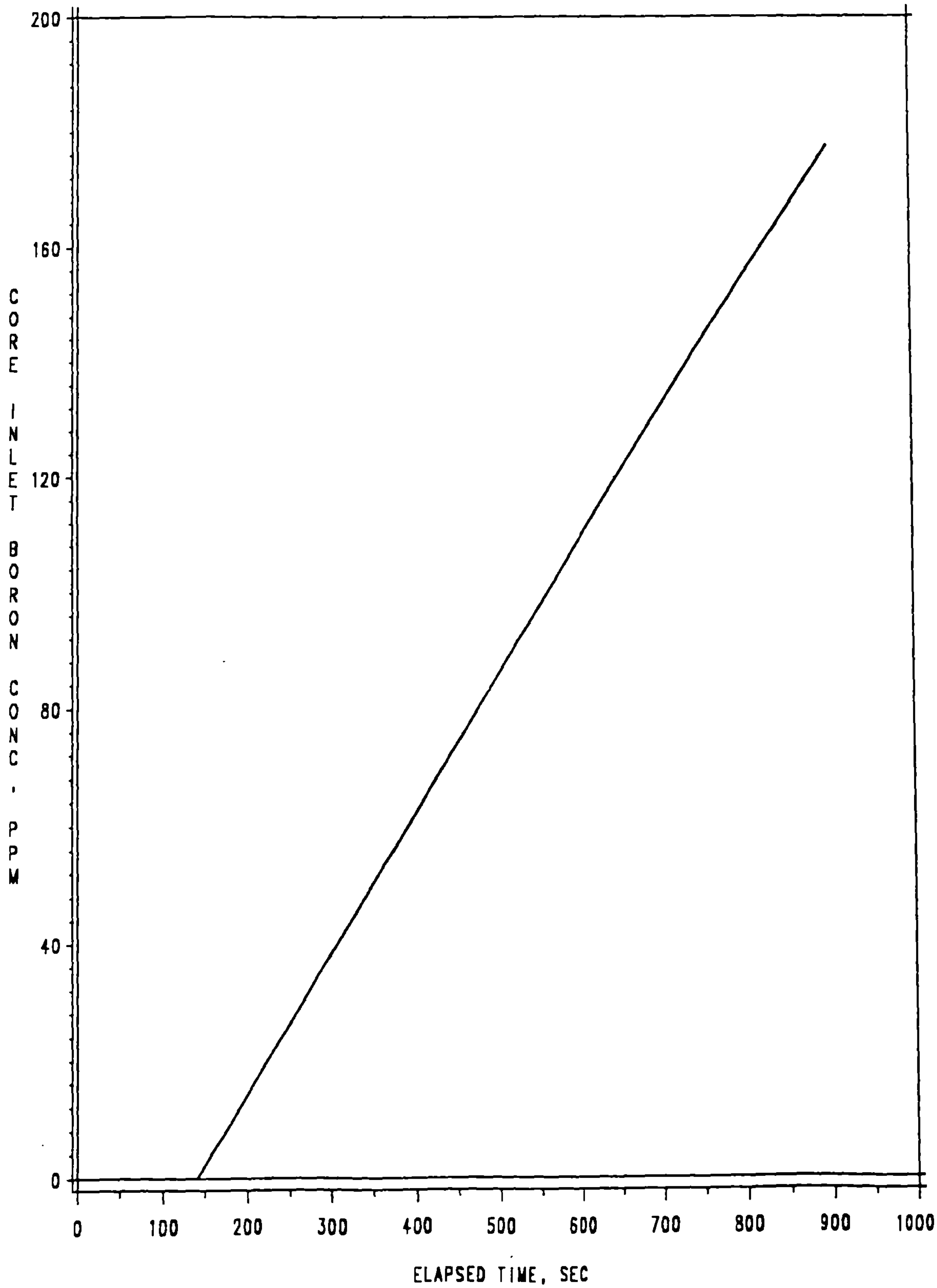

FIGURE 11
CORE POWER (MWT)
0.19 SQ. FT./LOOP BREAK

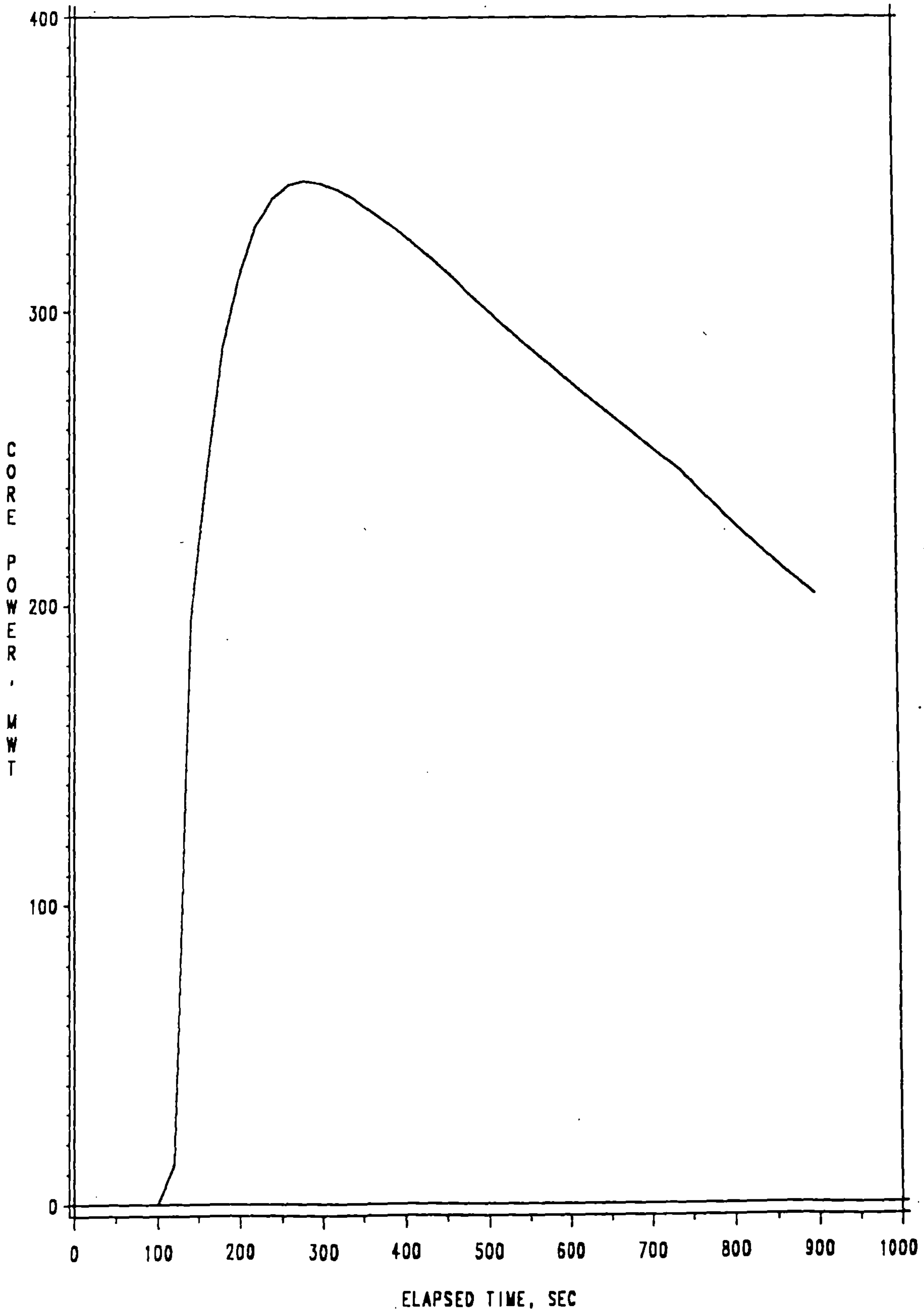

FIGURE 12
CORE HEAT FLUX (FRACTION OF HFP)
0.064 SQ. FT./LOOP BREAK

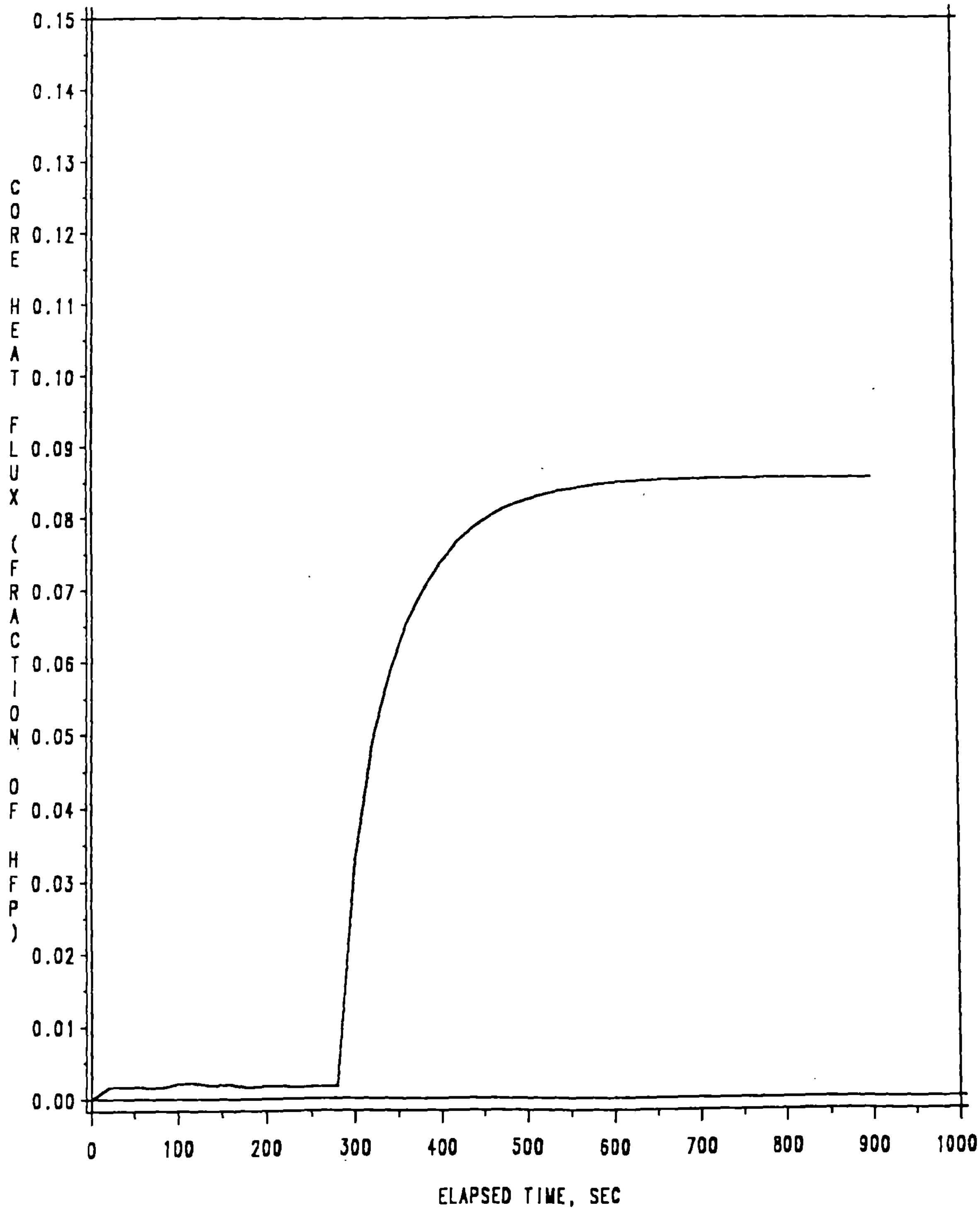

FIGURE 13
1-LOOP COLD LEG TEMPERATURE
0.064 SQ. FT./LOOP BREAK

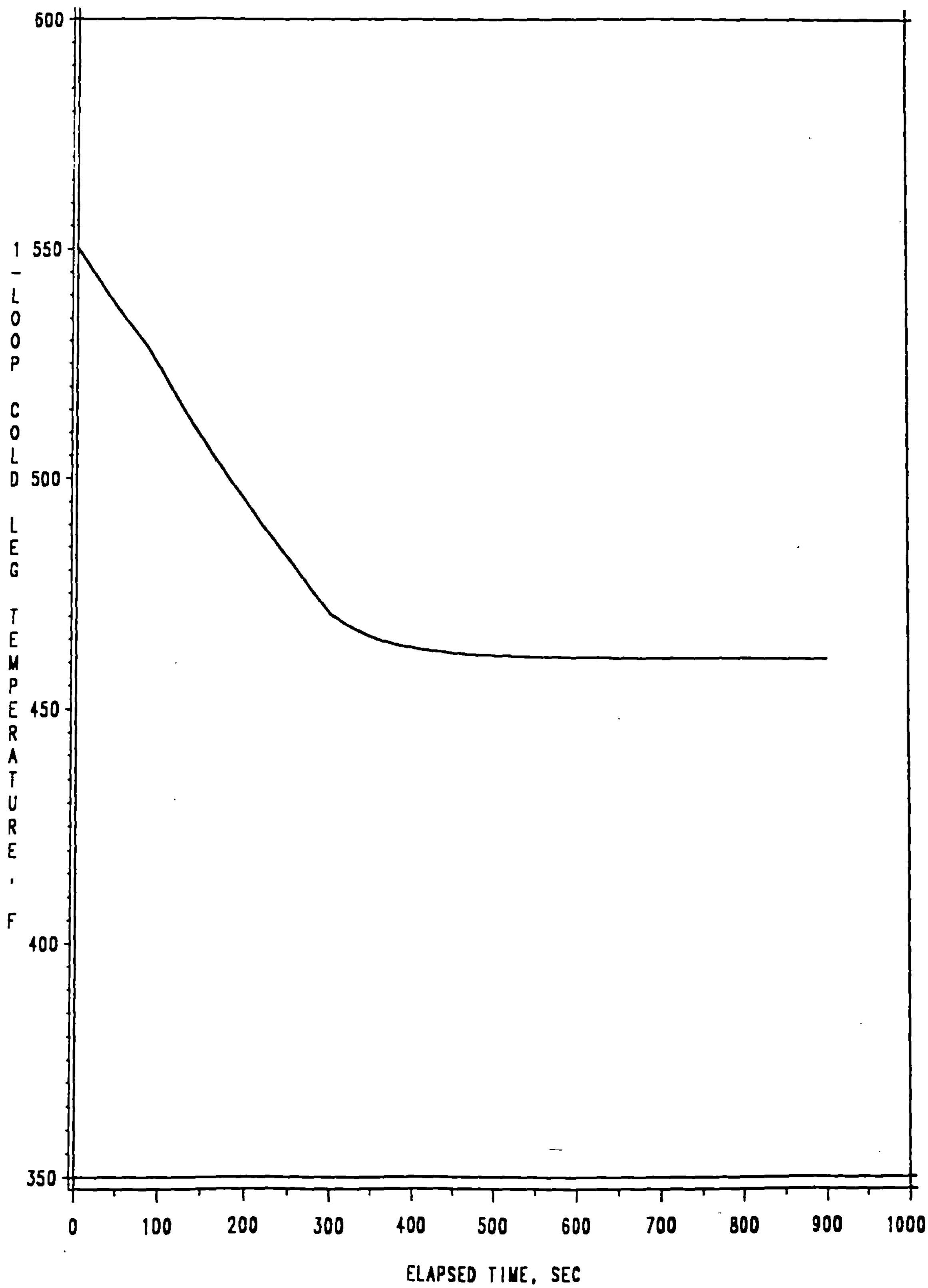

FIGURE 14
PRESSURIZER PRESSURE
0.064 SQ. FT./LOOP BREAK

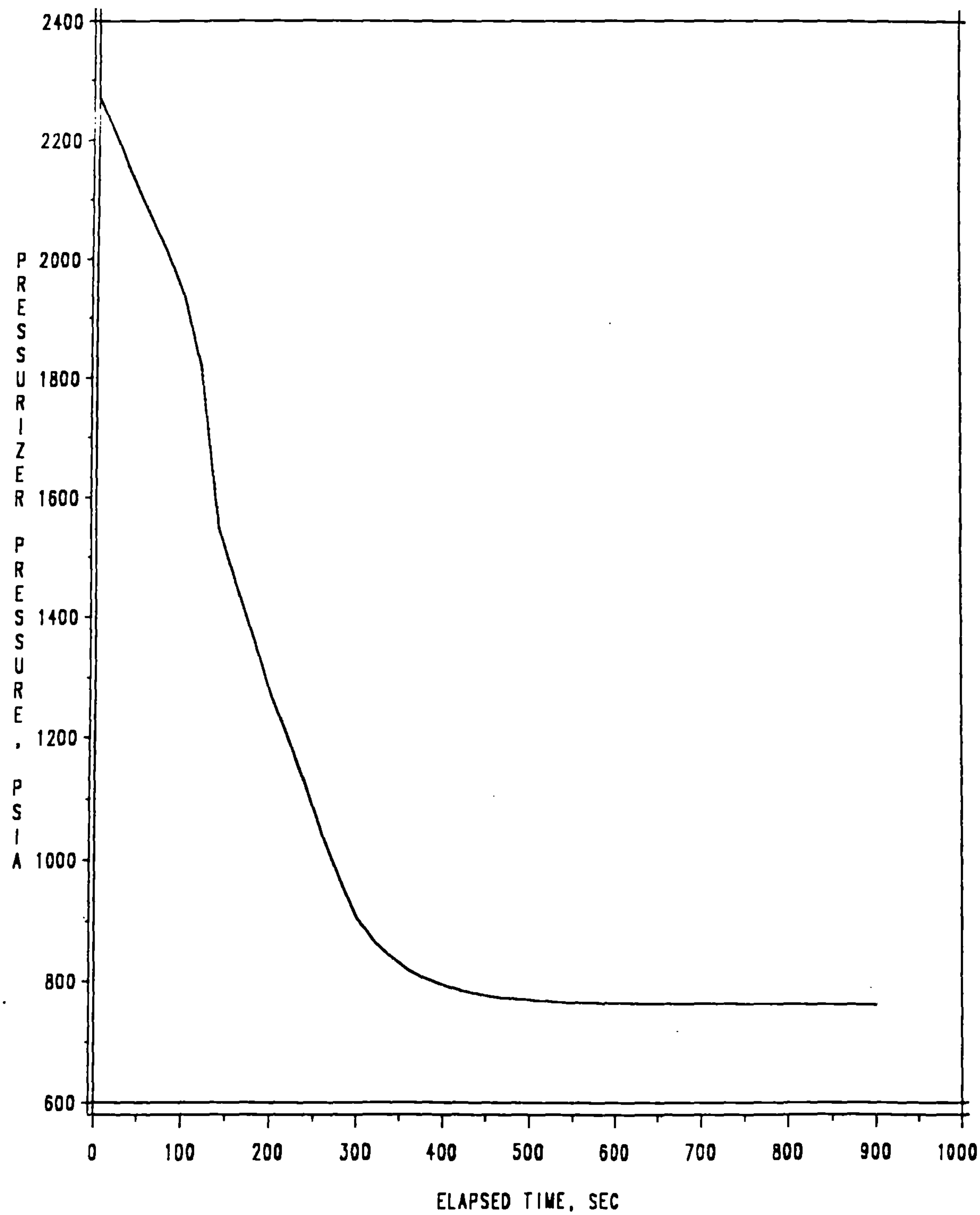

TABLE 3
SEQUENCE OF EVENTS- 0.24 SQ. FT. PER LOOP STEAM LINE BREAK

Time, seconds	Event
0.0	Steam line break occurs
20.1	Low-low RCS Tavg setpoint reached
60.0	Safety Injection on High-1 containment pressure
100.7	Low steamline pressure setpoint reached
280	Peak heat flux
900.0	End of simulation

TABLE 4

STEAMBREAK ACCIDENT STATEPOINTS

	Hypothetical Break With Power UFSAR Case A	Hypothetical Break Without Power UFSAR Case B	0.19 Sq. Ft. per Loop With Power Current Case
Core Heat Flux, % of 2441 MWT	23.7	8.1	14.2
RCS Pressure, Psia	959	853	872
Loop A Inlet Temp, F	398	276	433
Loop B Inlet Temp, F	469	497	432
Core Boron Concentration, PPM	0.0	0.4	33.6
RCS Flow, %	100	7.2	100
Reactivity, % deltaK/K	.007	.003	6.4E-5
Time, sec.	201	250	280
DNBR	>W-3 DNBR Limit	>W-3 DNBR Limit	>W-3 DNBR Limit

4.0 CONCLUSIONS

An analysis of small steam line break (SLB) events inside containment has demonstrated that there is adequate protection to ensure that the applicable accident analyses acceptance criteria are met for the entire spectrum of break sizes without relying on the low-low pressurizer pressure safety injection initiating function. Recent evaluations have shown that this function might not function in a harsh containment environment.

The analyses have shown that other sources of safety injection, i.e. high-1 containment pressure, high steam flow with low steam pressure or low Tavg, or high steam header to line differential pressure, provide adequate protection when required and the low-low pressurizer pressure may therefore be considered a diverse source of protection which need not be relied on to demonstrate acceptable results.

Assuming that the low-low pressurizer pressure safety injection function does not actuate in a harsh environment does not result in calculated conditions for any steam line break which are more limiting than those calculated for the large (hypothetical) steam line break case examined in the UFSAR. As a result, NRC review and approval of these analyses will resolve the outstanding unreviewed safety question involving operation of low-low pressurizer pressure safety injection.

5.0 REFERENCES

1. Licensee Event Report (LER) 89-043-00, "Low Pressure SI May Not Actuate During a Harsh Environment in Containment Due to Instrument Loop Inaccuracies", Surry Power Station Units 1 and 2, January 12,1990.

2. Licensee Event Report (LER) 89-018-00, " Uncertainty Associated With Harsh Environment Below ESF Transmitter Range", North Anna Power Station Units 1 and 2, January 11,1990.

3. UFSAR Section 14.3.2, "Rupture of a Main Steam Pipe", Rev. 3, 6/85.

4. WCAP-11431, "Mass and Energy Releases Following a Steam Line Rupture For North Anna Units 1 and 2", February 1987 (Proprietary).
   WCAP-11432, "Mass and Energy Releases Following a Steam Line Rupture For North Anna Units 1 and 2", February 1987 (Non-Proprietary).

5. VEP-FRD-41A, "Reactor System Transient Analyses Using the RETRAN Computer Code," May 1985.

6. EPRI NP-1850-CCM-A, "RETRAN-02- A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," November, 1988.

ATTACHMENT 2

Proposed Surry Units 1 and 2 Operating License Amendments