ATTACHMENT A

Revise the Technical Specification pages as follows:

Remove	Insert
3.6-1	3.6-1
3.6-3	3.6-3

3.6 Containment System

Applicability:

Applies to the integrity of reactor containment.

Objective:

To define the operating status of the reactor containment for plant operation.

Specification:

3.6.1 Containment Integrity

a. Except as allowed by 3.6.3 containment integrity shall not be violated unless the reactor is in the cold shutdown condition.

b. The containment integrity shall not be violated when the reactor vessel head is removed unless the boron concentration is greater than 2000 ppm.

c. Positive reactivity changes shall not be made by rod drive motion or boron dilution whenever the containment integrity is not intact unless the boron concentration is greater than 2000 ppm.

3.6.2 Internal Pressure

If the internal pressure exceeds 1 psig or the internal vacuum exceeds 2.0 psig, the condition shall be corrected within 24 hours or the reactor rendered subcritical.

Basis:

The reactor coolant system conditions of cold shutdown assure that no steam will be formed and hence there would be no pressure buildup in the containment if the reactor coolant system ruptures.

The shutdown margins are selected based on the type of activities that are being carried out. The (2000 ppm) boron concentration provides shutdown margin which precludes criticality under any circumstances. When the reactor head is not to be removed, a cold shutdown margin of 1%Δk/k precludes criticality in any occurrence.

Regarding internal pressure limitations, the containment design pressure of 60 psig would not be exceeded if the internal pressure before a major steam break accident were as much as 1 psig.[1] The containment is designed to withstand an internal vacuum of 2.5 psig.[2] The 2.0 psig vacuum is specified as an operating limit to avoid any difficulties with motor cooling.

References:

(1) Westinghouse Analysis, "Report for the BAST Concentration Reduction for R. E. Ginna," August 1985.

(2) UFSAR - Section 6.2.1.4.

Attachment B

The original licensing basis for containment integrity at Ginna was the Loss of Coolant Accident (LOCA). During the Systemmatic Evaluation Program (SEP) this licensing basis was reviewed. The results of the review concluded that the large steam break inside containment was more limiting than the LOCA for containment integrity. Therefore, the licensing basis for containment integrity became the large steam break supported by analysis done for the Staff by Lawrence Livermore National Laboratory and by Rochester Gas and Electric Corp. (RG&E)

Recently RG&E has contracted Westinghouse Electric Corp. to perform analysis to evaluate the possibility of reducing boron concentration in the Boric Acid Storage Tanks (BAST). A byproduct of this evaluation is a new containment integrity analysis (Enclosure 1) This analysis does not invalidate the previously approved SEP analysis. The new analysis uses a different methodology, different assumptions, different codes, and is better documented than the SEP analysis. It is proposed that the new analysis become the design basis containment integrity analysis. Since the SEP analysis used different assumptions, substituting the new analysis as the licensing basis necessitates revising the Technical Specifications. To be consistant with the initial conditions assumed in the new analysis, containment pressure should be limited to 1 psig.

In accordance with 10 CFR 50.91, this change to the Technical Specifications has been evaluated against three criteria to determine if the operation of the facility in accordance with the proposed amendment would:

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

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

3. involve a significant reduction in a margin of safety.

The proposed change would decrease the initial containment pressure before a major steam break inside containment and therefore does not increase the probability or consequences of a previously evaluated accident or create the possibility of a new or different kind of accident or involve a significant reduction in a safety margin. Therefore, a no significant hazards finding is warranted for the proposed change.

ENCLOSURE 1

REPORT FOR THE BAST CONCENTRATION REDUCTION FOR R. E. GINNA

August 1985

INTRODUCTION

Westinghouse has developed improved analytical techniques which allow a reduction in the Boric Acid Storage Tank (BAST) concentration. This report provides background information on the BAST design basis, reasons why boron reduction may be desirable, plant design features which allow the change to be proposed, as well as a summary of analytical results which demonstrate the feasibility of this option on the BAST system for Ginna.

BACKGROUND

The two BASTs are components of the Chemical and Volume Control System which also provides concentrated boric acid to the reactor coolant to mitigate the consequences of postulated steamline break accidents. In this function, they act as part of the Safety Injection System. Although the BASTs act to mitigate steamline break of various sizes occurring from any power level, the cases which serve as the Westinghouse steamline break licensing basis, and which define the existing requirements on the minimum BAST boron concentration, are as follows:

- For the "hypothetical" steamline break, i.e., double ended rupture of a main steamline, the radiation releases must remain within the requirements of 10CFR Part 100. This is the ANSI N18.2 criterion for Condition IV events, "Limiting Faults." Westinghouse conservatively meets this for Ginna by demonstrating that the DNB design basis is met, the criterion typically used for Condition II events.

- For the "credible" steamline break, i.e., the failure open of a single steam generator relief, safety, or turbine bypass valve, that radiation releases must remain within the requirements of 10CFR Part 20. This is the ANSI N18.2 criterion for Condition II events, "Faults of Moderate Frequency." Westinghouse conservatively meets this criterion by showing that the DNB design basis is met.

In order to assure the validity of the safety analyses performed to verify that the evaluation criteria are met, Technical Specifications have been applied to the BAST and associated equipment. Specifically, these assure that the boric acid concentration is maintained in excess of 20,000 ppm, approximately a 12 weight percent solution. In order to maintain this high concentration, heat tracing of the tanks and associated piping is required. Furthermore, the safety-related nature of the boric acid system requires that the heating systems be redundant.

The required solubility temperature imposes a continuous load on the heaters, and low-temperature alarm actuation and heater burnout have occurred in some operating plants. Violation of the Technical Specification on concentration in the BAST poses availability problems in that recovery is required within a very short time. If the concentration is not restored within 24 hours, the plant must be taken to the hot shutdown condition. Thus, this requirement has a potentially serious impact on plant availability.

These potential difficulties unfavorably affecting plant availability, operability, and maintainability can be drastically reduced in severity or eliminated by reducing the boron concentration to a minimum level at which heat tracing would no longer be required. The effect of this change is discussed in the following section.

DESCRIPTION OF THE ANALYSES

The only accident analyses which are significantly affected by boron concentration reduction are the steamline break transients. Since the steam break affects the core and the containment responses, both of these were considered in the boron concentration reduction analysis. The following analysis consists of a core analysis and a containment mass-energy analysis.

CORE ANALYSIS

The following cases must be considered for the BAST boron concentration reduction with respect to the core analysis.

a. Complete severance of a pipe inside the containment at the outlet of the steam generator at initial no-load conditions with outside power available and two loops in service. The equivalent break area is 4.37 sq. ft.

b. Case (a) above with loss of outside power simultaneous with the steamline break.

c. A break equivalent to steam release through one steam generator safety valve with outside power available and two loops in service.

d. Case (a) above with only one loop in service.

e. Case (c) above with only one loop in service.

The severance of a pipe downstream of the steam flow measuring nozzle is not analyzed. The equivalent break area (1.4 sq. ft.) is less than that of case (a) and would result in a less severe cooldown. Thus, this break is bounded by cases (a) and (b).

Of these cases, cases (a) through (c) were analyzed with the BAST concentration at 2000 ppm in the Reload Transition Safety Report (RTSR)[1] and approved by issuance of a Technical Specification change[2]. The results of these analyses in the RTSR show that the DNB design is met. Thus, only cases (d) and (e) need be considered here.

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1. NRC Letter, R. W. Kober (NRC) to D. M. Crutchfield (RG&E), Application for Amendment to Technical Specifications, December 20, 1983.

2. Amendment No. 61 to the R. E. Ginna Technical Specifications dated May 1, 1984.

Analysis Method

As in the Ginna RTSR steamline break analysis, the system transient parameters, i.e., RCS pressure, temperatures, steam flow, core boron concentration and core power are calculated using the LOFTRAN[(3)] system transient analysis computer code. This computer code includes models of the reactor core, steam generators, pressurizer, primary piping, protection systems and engineered safeguards systems.

The results presented are a conservative indication of the events which would occur assuming a steamline rupture. The worst case assumes that <u>all</u> of the following occur simultaneously.

1. Minimum shutdown reactivity margin equal to 2.45 percent (1 loop in service).

2. The most negative moderator temperature coefficient for the rodded core at end-of-life.

3. The rod having the most reactivity stuck in its fully withdrawn position.

4. One safety injection pump fails to function as designed.

The plant is initially assumed to be at hot zero power at the minimum required shutdown margin. Following the break, the RCS temperatures and pressures decrease rapidly, and in the presence of a large End-of-Life (EOL) moderator coefficient of reactivity, the reactor returns critical with the rods inserted, assuming the most reactive RCCA in the fully withdrawn position. The reactor power increases at a decreasing rate until boron from the safety injection system reaches the core and begins to offset the positive reactivity insertion caused by the cooldown. The core is subsequently brought subcritical with boron injection, aided by the abatement and eventual termination of steam flow from the broken steam generator.

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3. WCAP-7907, T.W.T. Burnett, et. al., "LOFTRAN Code Description," October, 1972.

Figures 1 through 5 show the transient behavior for the 4.37 sq. ft. Hypothetical Break with one loop in service with the BAST concentration equal to 2000 ppm (case d). A comparison of the RTSR cases (fig. 14.2.5-18) with Figures 1-5 reveals that the reactor coolant system transients are similar, with the single exception of core power, which is understandably higher for the case with reduced boron concentration in the BAST. The effect of the boron on the total reactivity is both delayed and damped in Figure 1 because the boron source is both colder and of a lower boron concentration. This causes the heat flux to initially rise to a higher peak (33% of 1520 MWth) and to subsequently decay at a slower rate after the boron reaches the core. A DNB analysis for this transient shows that the minimum DNBR is above the limit value, thus no fuel failure is predicted due to DNB.

Figures 6 through 8 depict transient parameters for the Condition II steamline break, assuming 2000 ppm in the BAST (case e). In the RTSR, the reactivity plot in Figure 14.2.5-25 shows that the reactor remains subcritical. This assures that the DNB design basis is met in a very conservative manner. The reactor also remains subcritical when the BAST is at 2000 ppm. Boron enters the core while the reactor is still significantly shut down, as can be seen in Figure 8. Since the reactor remains subcritical, the DNB design basis is met.

The sequence of events is presented in the attached table.

In conclusion, calculations have been performed for Ginna which show that from the DNB standpoint BAST concentration can be reduced to 2000 ppm since the DNB design basis is met. For 2 loop operation, this analysis is contained in the RTSR. The analyses presented here show that the results are acceptable for operation with one loop in service.

MASS AND ENERGY ANALYSIS

Steamline ruptures occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures. The quantitative nature of the releases following a steamline rupture is dependent upon the many possible configurations of the plant steam system and containment designs as well as the plant operating conditions and the size of the rupture.

The following cases have been reanalyzed for the BAST boron concentration reduction.

- Large or Full Double-Ended Steamline Ruptures

- Small Double-Ended Steamline Ruptures

- Split Steamline Ruptures

The large breaks analyzed are listed in Table 2; the small break analyzed are listed in Table 3; and the split breaks analyzed are listed in Table 4. These break sizes were chosen because the 4.37 sq. ft. is the largest break that can occur. The 1.4 sq. ft. break is the largest break that can occur downstream of the flow restrictor. The split break is chosen to be the largest break which can occur such that protection is actuated by the containment signals, rather than the primary signals (low steam pressure, high steam flow, etc.). The hot zero power and hot full power cases have been analyzed since these have been previously defined by the NRC to be the steamline break mass and energy release inside containment licensing requirement[4] for R. E. Ginna.

4. NRC Letter, D. M. Crutchfield (NRC) to John E. Maier (RG&E), "Evaluation Report on SEP Topic VI-2.D and VI-3," November 3, 1981.

Analysis Method

The mass and energy analysis is initiated by using the LOFTRAN[3] code to determine the mass and energy released to the containment during a steamline break. The mass and energy data is then used by the COCO[5] code to determine temperature and pressure response in the containment following a steamline break accident. The basic initial conditions, heat sink model and fan cooler parameters employed in the containment response calculation are outlined in Tables 5 through 7. The following conservative assumptions are made for each mass and energy release analysis:

1. Maximum decay heat equivalent to 120% of ANS finite model.

2. No credit is taken for water entrainment in the blowdown results.

3. Conservatively high values for reverse steam generator heat transfer.

4. The most negative moderator temperature coefficient for the rodded core at end-of-life.

5. One containment spray pump fails to function.

6. Offsite power is available throughout the transient.

Figure 9 provides the pressure and temperature curves for the limiting large break case providing the highest peak containment pressure and temperature of those cases listed in Table 2. Figure 10 provides the pressure and temperature curves for the limiting small break case providing the highest peak containment pressure and temperature of those cases listed in Table 3. Figure 11 provides the pressure and temperature curves for the limiting split break case providing the highest peak containment pressure and temperature of those cases listed in Table 4. These latter two curves are not representative of the split break accident with a single failure assumed since all three

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5. Bordelon, F. M., and Murphy, E. T., Containment Pressure Analysis Code (COCO), WCAP-8326, June 1974.

failures were included. Because of the margin observed in the peak pressures and temperatures of the large and small steamline breaks when a single failure was assumed, all those single failures: containment spray pump, MSIV and FIV were assumed in the split break analyses. Thus, the peak values illustrated in Figure 11 are conservative due to the multiple failures.

Figure 9 contains the containment pressure and temperature response for the 4.37 sq. ft. HZP double-ended rupture. Note that the HZP case proved to be more limiting than the HFP case analyzed. This is primarily due to the large mass of water in the steam generator under HZP conditions which is available for discharge through the break. For this particular case analyses were performed which examined the consequences of two single failures: a single containment spray pump failure, and an auxiliary feedwater runout failure. The case presented in Figure 9 represents the containment spray pump failure. This case was analyzed assuming a BAST boron concentration of 20000 ppm. Figure 12 contains the mass and energy release rates for this case.

Figure 10 shows the containment pressure and temperature response for the 1.4 sq. ft. HFP double-ended rupture. Note that the peak pressure and temperature are significantly lower for the 1.4 sq. ft. break than for the 4.37 sq. ft. break. This is due to the smaller break area which reduces the blowdown mass and energy release rate, this in turn results in a lower peak containment pressure and temperature than the 4.37 sq. ft. case. Due to the significant margin available to the containment pressure design limit only the containment spray pump failure was considered for the 1.4 sq. ft. cases. This case was analyzed assuming a BAST boron concentration of 6000 ppm.

Figure 11 contains the containment pressure and temperature response profiles for the 0.6 sq. ft. HFP split rupture. As discussed above this case contains three single failures: a containment spray pump failure, a main steam isolation valve failure, and a feedwater isolation valve failure. This case was analyzed assuming a BAST boron concentration of 6000 ppm.

The large break mass and energy calculations were proven to be the limiting cases because of the higher pressures reached. The temperatures and pressures reached in the large breaks with the assumed BAST concentration of 20000 ppm fall below the containment design limits.

Therefore, from a mass and energy point of view for the cases analyzed, it does not appear possible to reduce the BAST boron concentration below the current value of 20000 ppm due to the lack of significant available margin to the peak containment pressure limit of 60 psig.

A sensitivity study was performed to determine the impact of superheat for the RGE steamline break containment analysis. This sensitivity was performed on the limiting pressure case, 4.37 sq. ft. double-ended rupture at hot zero power, utilizing updated mass and energy releases modeling superheat characteristics. The results from this case revealed no diversion from the results of the non-superheat case.

CONCLUSIONS

Plant specific analyses have been performed for the R. E. Ginna steamline break transients and have shown that while the current boron concentration of 20000 ppm will ensure that the peak containment pressure limit of 60 psig is not exceeded, there is not a sufficient amount of margin to the containment pressure limit to allow a reduction in the Boric Acid Storage Tank boron concentration requirement.

TABLE 1

TIME SEQUENCE OF EVENTS

Case	Event	Time (seconds)
d	Steamline ruptures	0
	Pressurizer empties	9
	Criticality attained	22
	Boron enters core	45
e	Safety valve fails open	0
	Pressurizer empties	93
	Low pressurizer pressure SI setpoint reached	99
	Boron enters core	183

TABLE 2

4.37 FT^2 FULL-DOUBLE-ENDED BREAK

Power Level	Single Failure	Offsite Power
102%	Containment Spray Pump	Available
0%	Containment Spray Pump	Available
0%	Auxiliary Water Runout	Available

TABLE 3

4.37 FT^2 DOUBLE-ENDED BREAK

Power Level	Single Failure	Offsite Power
102%	Containment Spray Pump	Available
0%	Containment Spray Pump	Available

TABLE 4

SPLIT BREAK THAT WILL NEITHER GENERATE A PRIMARY STEAMLINE ISOLATION SIGNAL NOR RESULT IN ENTRAINMENT

Power Level	Break Area	Single Failure	Offsite Power
102%	0.6 ft^2	Containment Spray Pump MSIV (Main Steam Isolation Valve) FIV (Feedwater Isolation Valve)	Available
0%	0.3 ft^2	Containment Spray Pump MSIV FIV	Available

TABLE 5

ASSUMPTIONS FOR CONTAINMENT ANALYSIS

Refueling water temperature (°F)	80
Initial Containment	
Temperature (°F)	120
Initial pressure (psia)	15.7
Initial relative humidity (%)	30
Net free volume (ft^3)	1.00×10^6
Safeguard System	
Number of fan coolers	4
Pressure set point (psig)	6
Delay time (sec)	32
Number of spray pumps	1
Maximum spray flow (gpm)	1200.
Pressure set point (psig)	33.5
Delay time (sec)	35.5

TABLE 7

FAN COOLER HEAT REMOVAL

Containment Temp (°F)	RCFC Heat Removal/Fan Cooler Btu/sec
200.	4416.67
210.	4833.3
220.	5750.0
230.	7166.67
240.	8500.0
250.	9583.3
260.	10583.2
270.	11583.3
280.	12500.0
290.	13416.7
300.	14083.3

TABLE 6

PASSIVE HEAT SINKS

	Wall Description	Heat Transfer Area (ft^2)	Material	Thickness (ft)
1.	Insulated portion of dome and containment wall	36181.0	Insulation	0.1042
			steel	0.03125
			Concrete	2.5
2.	Uninsulated portion of dome	12474.0	Concrete	2.5
			steel	0.03125
3.	Basement floor	7955.0	Concrete	2.0
			Steel	0.03125
			Concrete	2.0
4.	Walls of sump in basement floor	2342.0	Concrete	5.0
			Steel	0.03125
			Concrete	3.5
5.	Floor of sump	297.0	Concrete	2.0
			Steel	0.03125
			Concrete	2.0
6.	Inside of refueling cavity	3800.0	Stainless Steel	0.020833
			Concrete	2.5

TABLE 6 (Continued)

PASSIVE HEAT SINKS

Wall Description	Heat Transfer Area (ft^2)	Material	Thickness (ft)
7. Bottom of refueling cavity	1117.0	Stainless Steel	0.020833
		Concrete	2.5
8. Area on outside of refueling cavity walls	5952.0	Concrete	2.5
9. Area inside of loop and steam generator compartment*	12463.0	Concrete	2.5
10. Floor area intermediate level*	6170.0	Concrete	0.5
11. Operating floor*	6540.0	Concrete	2.0
12. 1 1/2" thick I-beam**	3151.0	Steel	0.125
13. 1" thick I-beam**	5016.0	Steel	0.0833
14. 1/2" thick I-beam	8138.0	Steel	0.04167
15. Cylindrical supports for S.G. and MCP's	430.0	Steel	0.04167

TABLE 6 (Continued)

PASSIVE HEAT SINKS

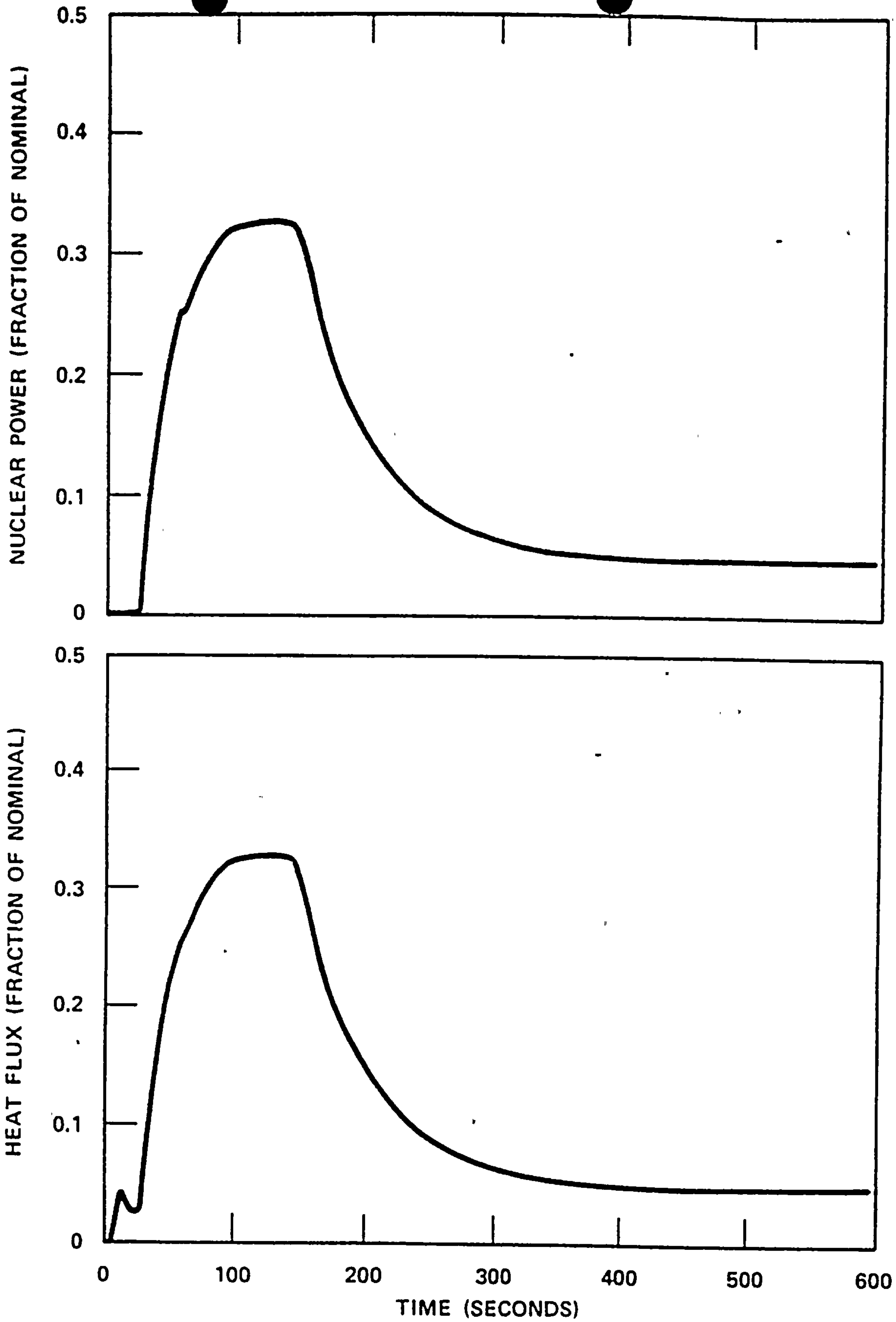
Wall Description	Heat Transfer Area (ft^2)	Material	Thickness (ft)
16. Plant crane rectangular support columns	5756.0	Steel	0.0625
17. Beams used for crane structure**	6023.0	Steel	0.125
18. Structure on operating floor	2622.0	Concrete	2.0
19. Grating, stairs, misc. steels	7000.0	Steel	0.0104

* Both sides exposed, valve represents area for one side.

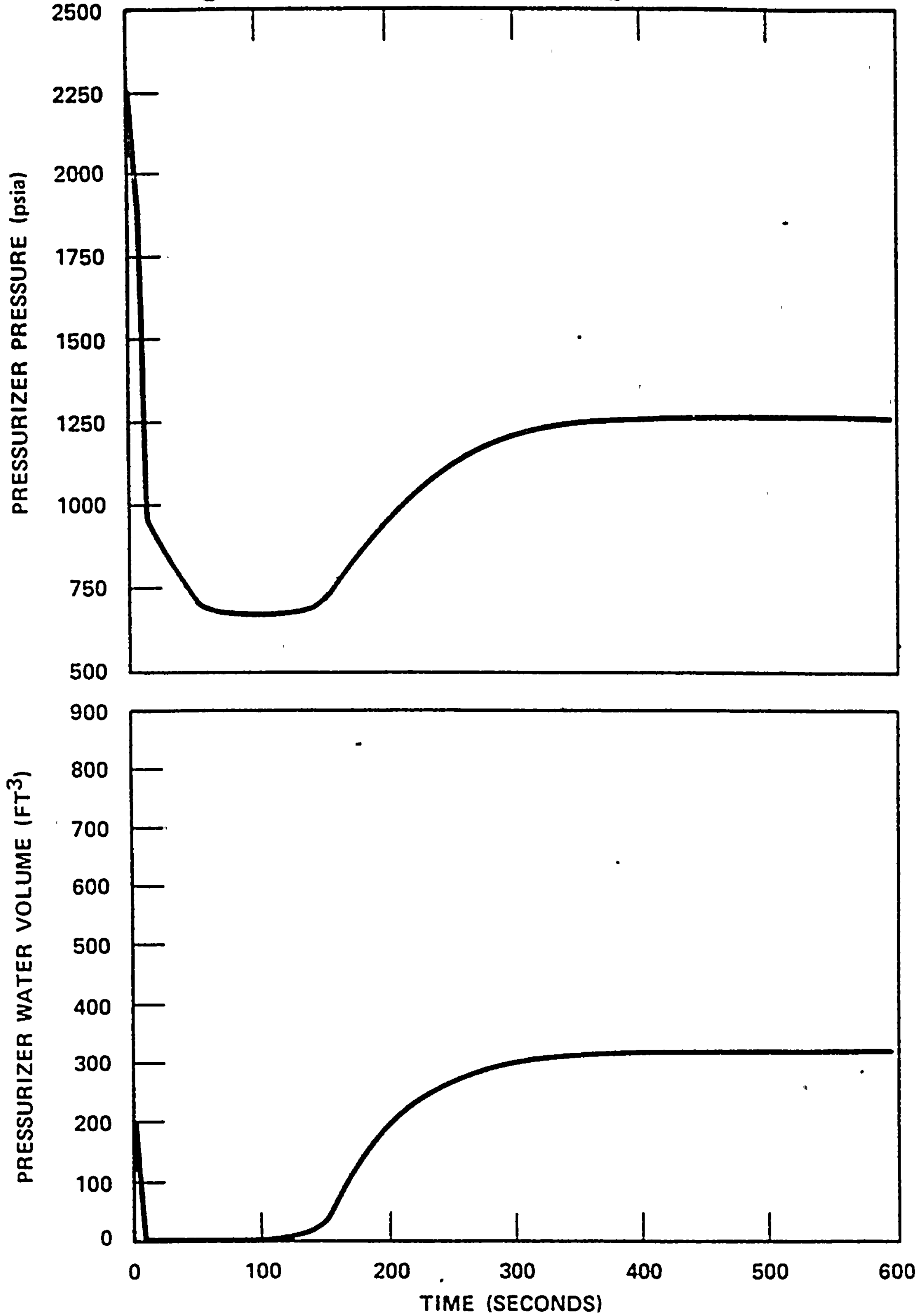
** Both sides exposed, valve represents area for both sides.

<u>Thermophysical Properties of Containment Heat Sinks</u>

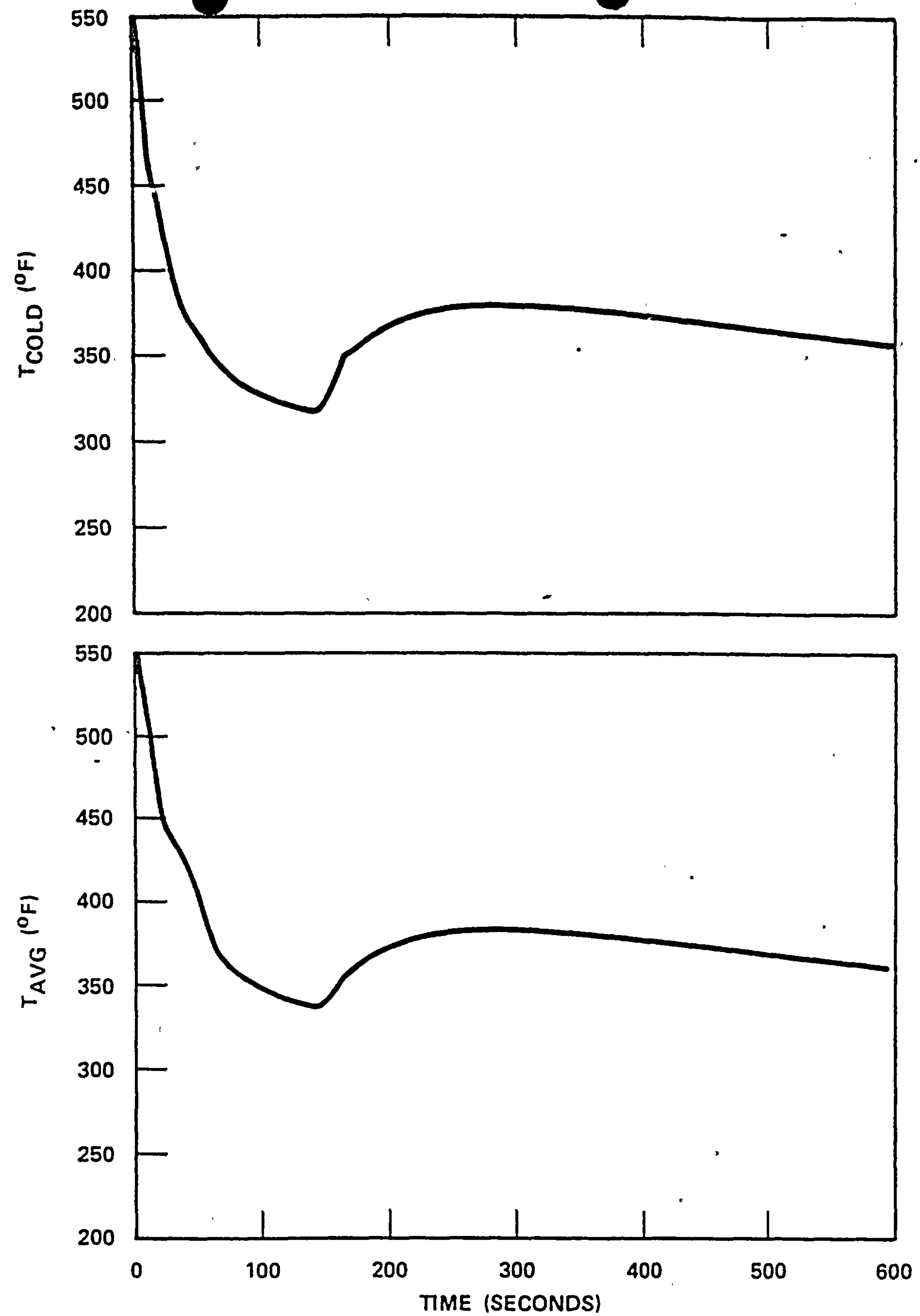
Material	Thermal Conductivity (Btu/hr-ft-°F	Volumetric Heat Capacity Btu/ft3-°F
Insulation	0.0208	2.0
Steel	28.0	58.8
Concrete	0.9	32.9



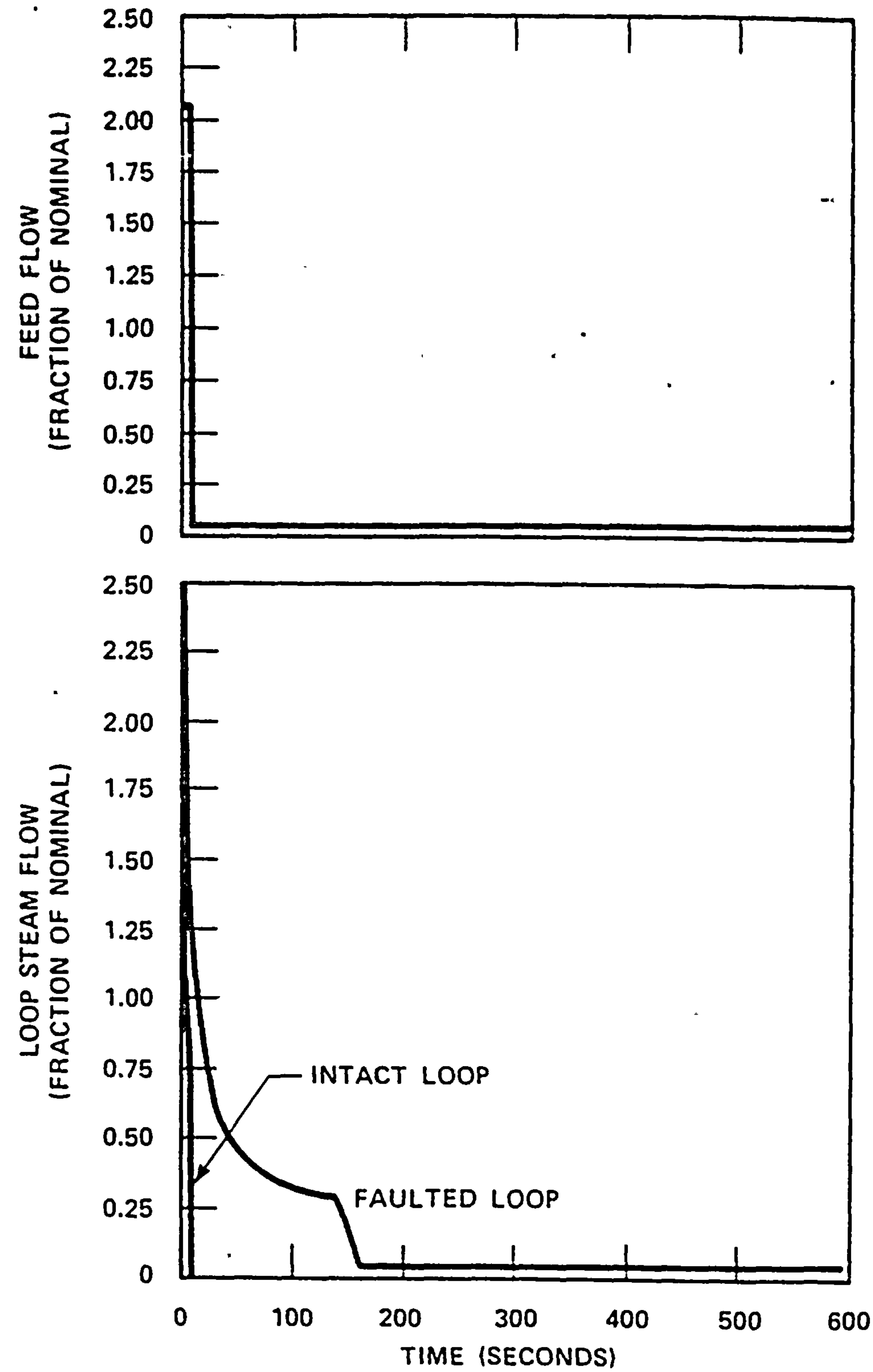
R. E. GINNA BAST CONCENTRATION REDUCTION STUDY	Figure 1 4.37 ft^2 Steamline Break One Loop in Service



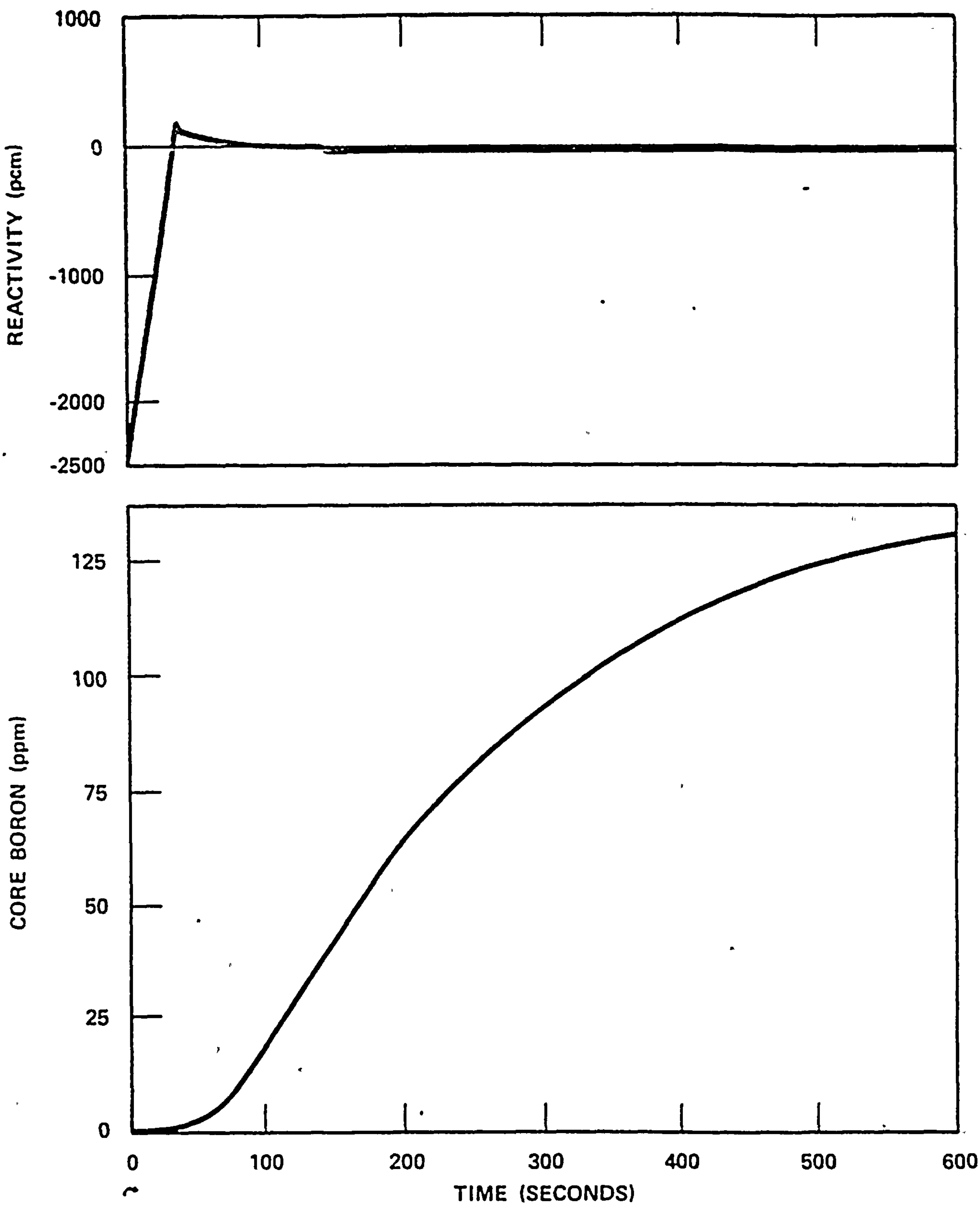
R. E. GINNA BAST CONCENTRATION REDUCTION STUDY	Figure 2 4.37 ft^2 Steamline Break One Loop in Service



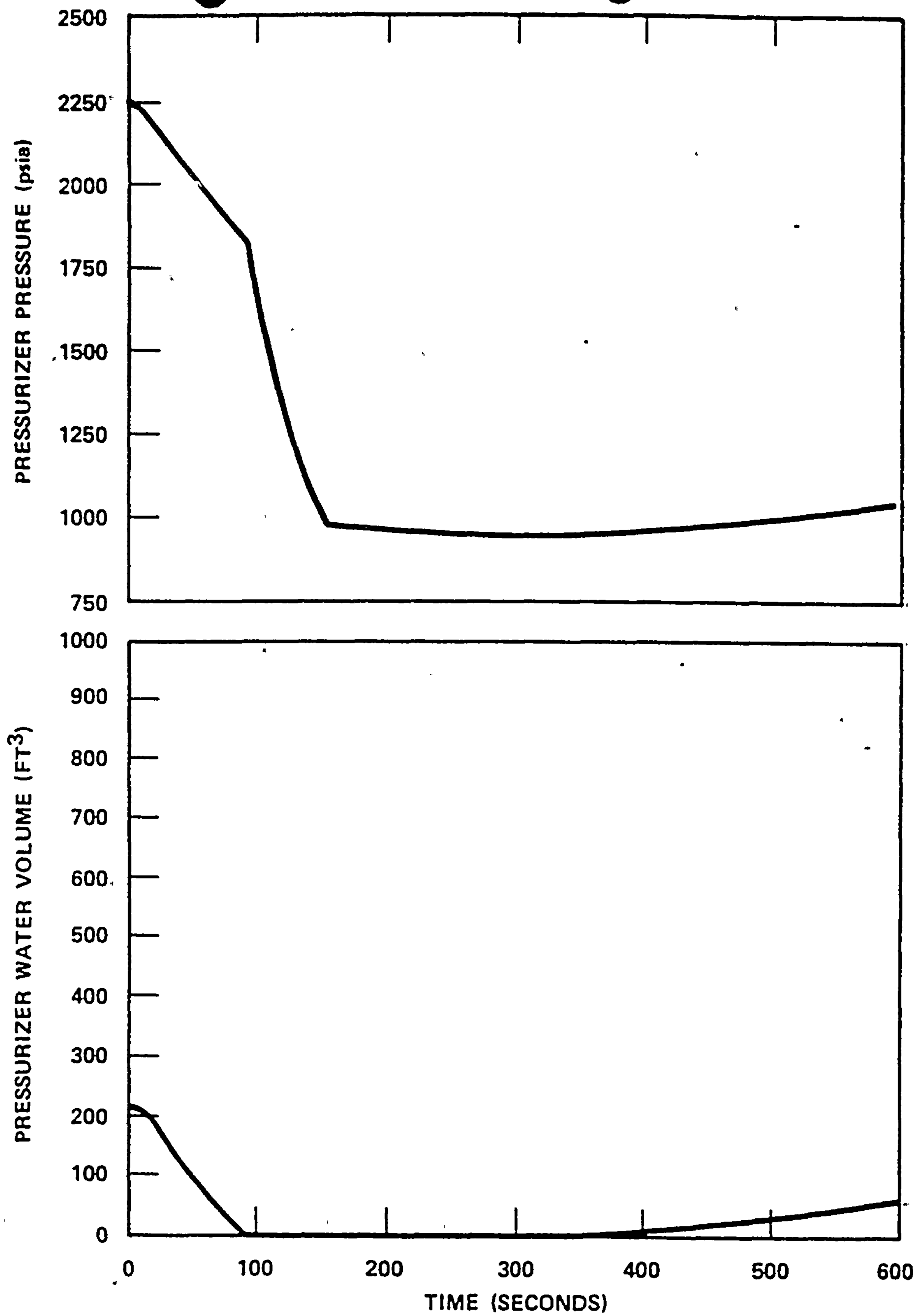
R. E. GINNA BAST CONCENTRATION REDUCTION STUDY	Figure 3 4.37 ft^2 Steamline Break One Loop in Service



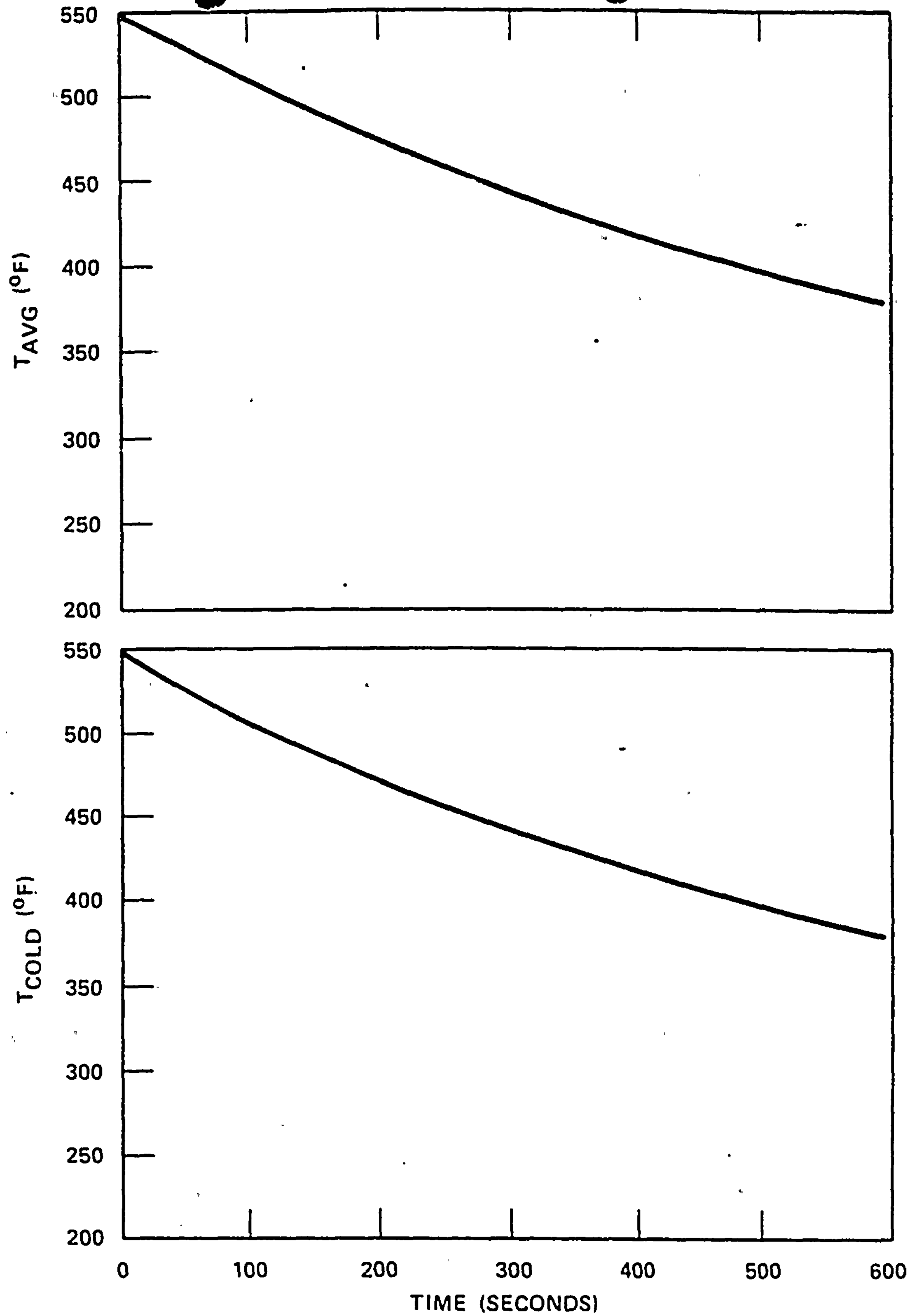
R. E. GINNA BAST CONCENTRATION REDUCTION STUDY	Figure 4 4.37 ft^2 Steamline Break One Loop in Service



R. E. GINNA BAST CONCENTRATION REDUCTION STUDY	Figure 5 4.37 ft^2 Steamline Break One Loop in Service

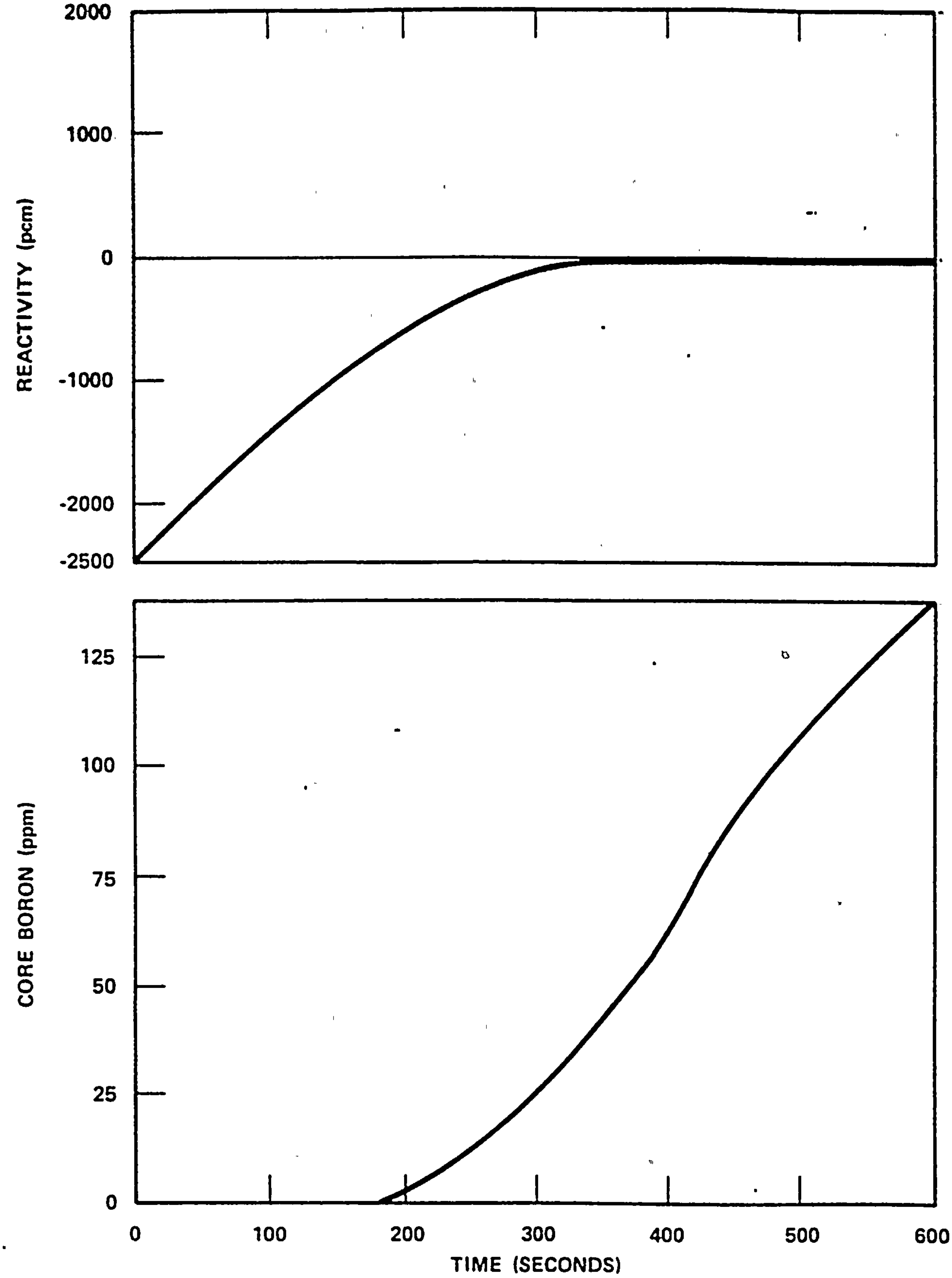


R. E. GINNA BAST CONCENTRATION REDUCTION STUDY	Figure 6 Failed Safety Valve One Loop in Service



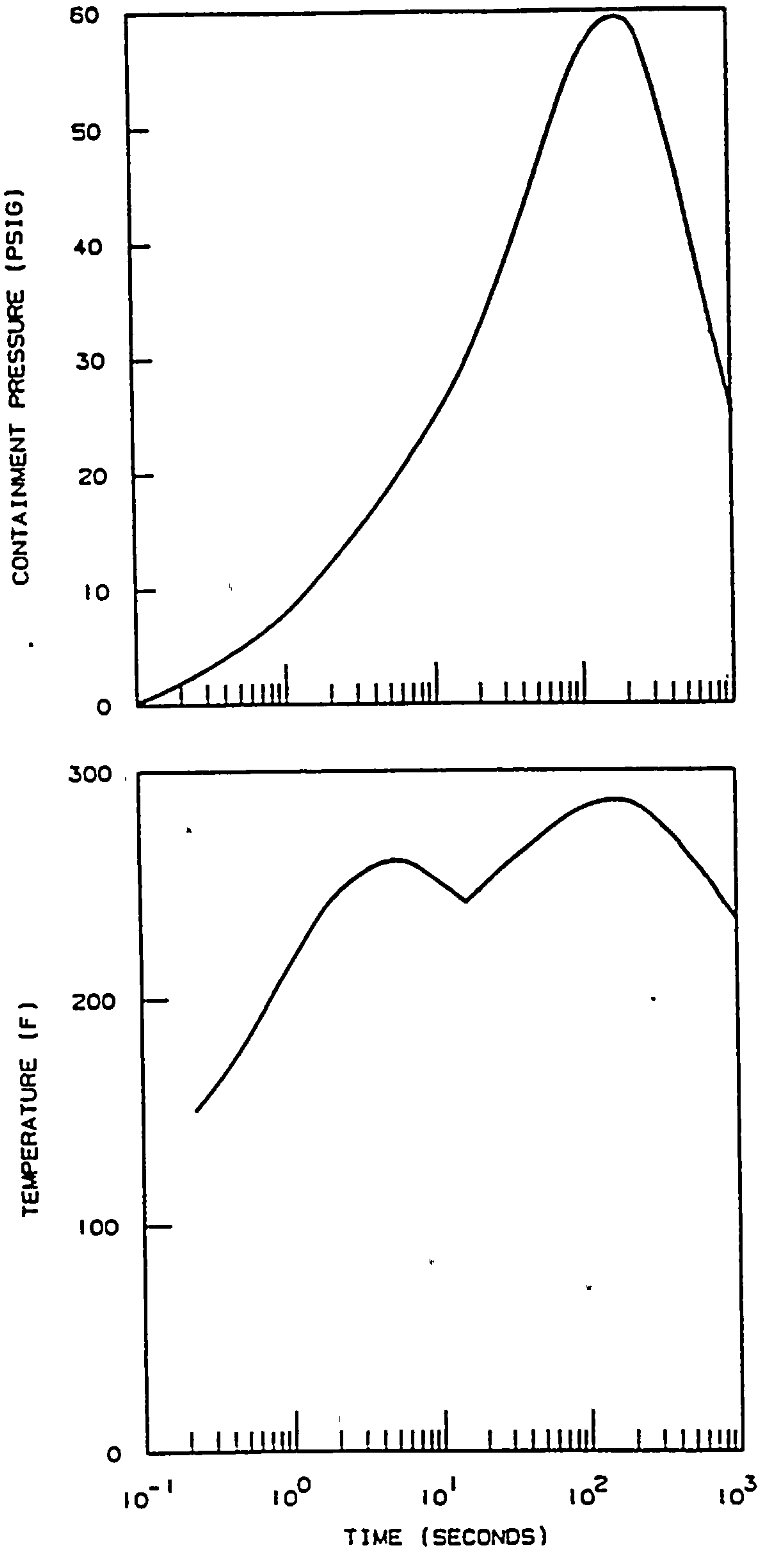
R. E. GINNA
BAST CONCENTRATION
REDUCTION STUDY

Figure 7
Failed Safety Valve One Loop in Service

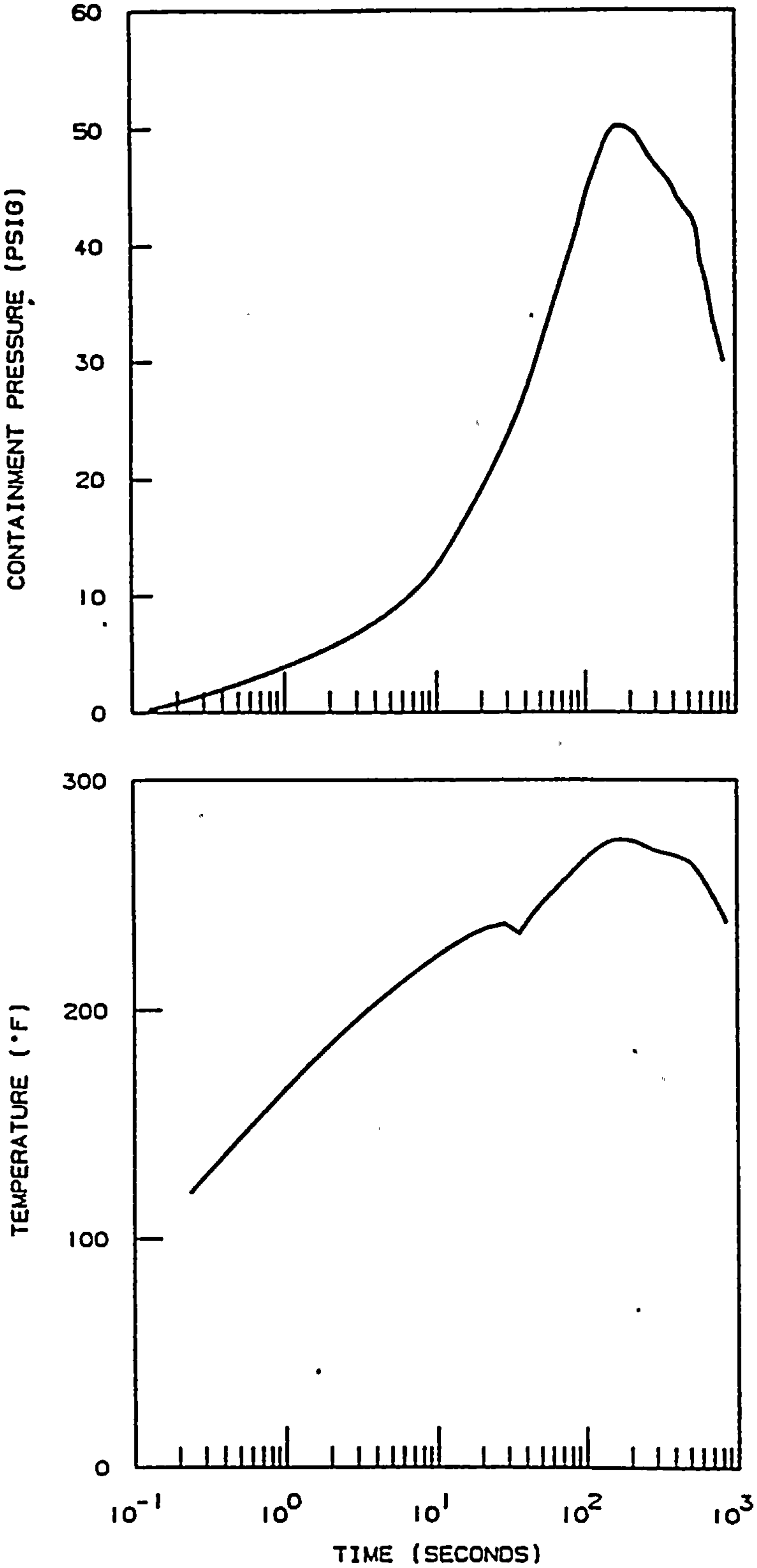


R. E. GINNA BAST CONCENTRATION REDUCTION STUDY	Figure 8 Failed Safety Valve One Loop in Service

12702.1

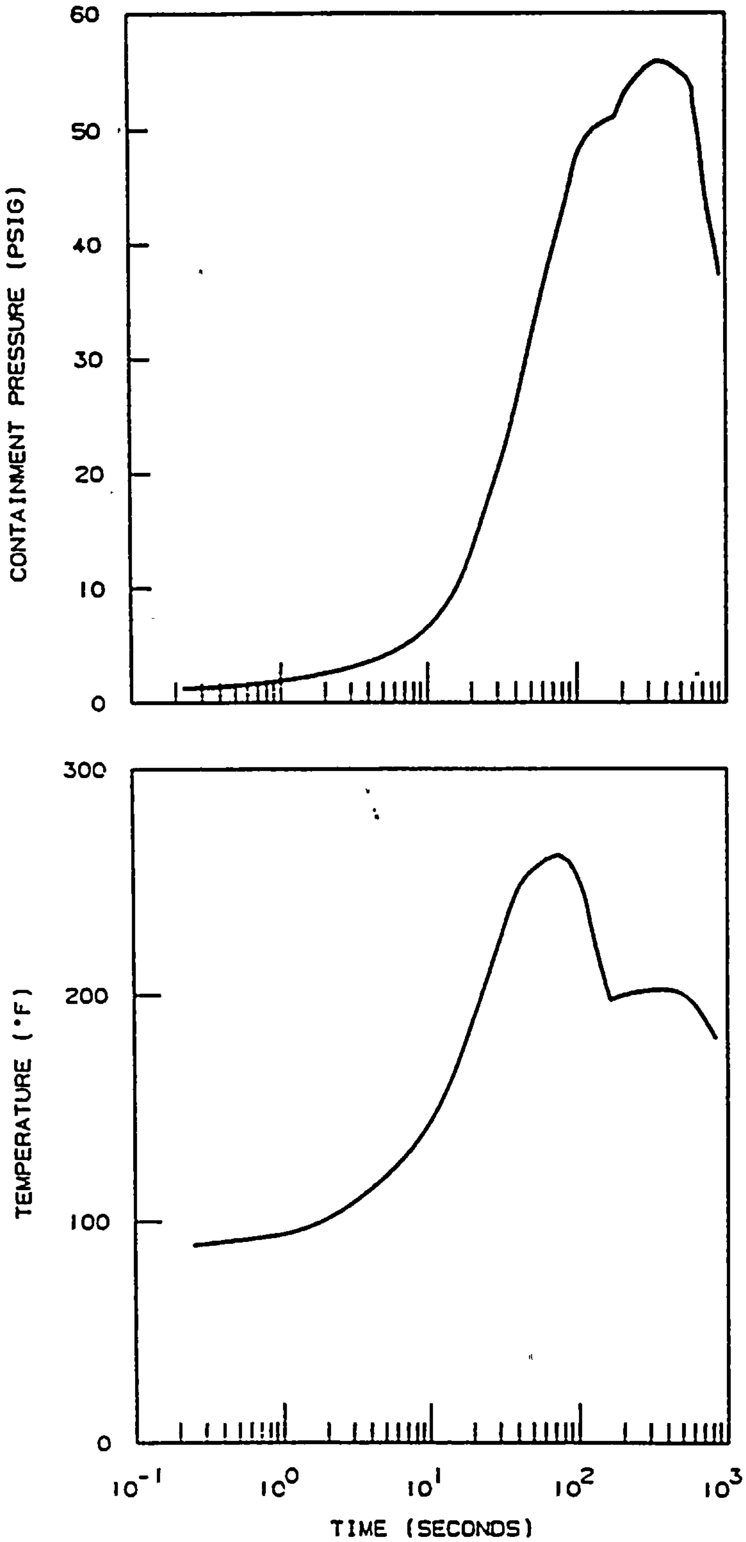


R. E. GINNA BAST CONCENTRATION REDUCTION STUDY	Figure 9 4.37 Ft^2 Per Steamline Break — 0% Power

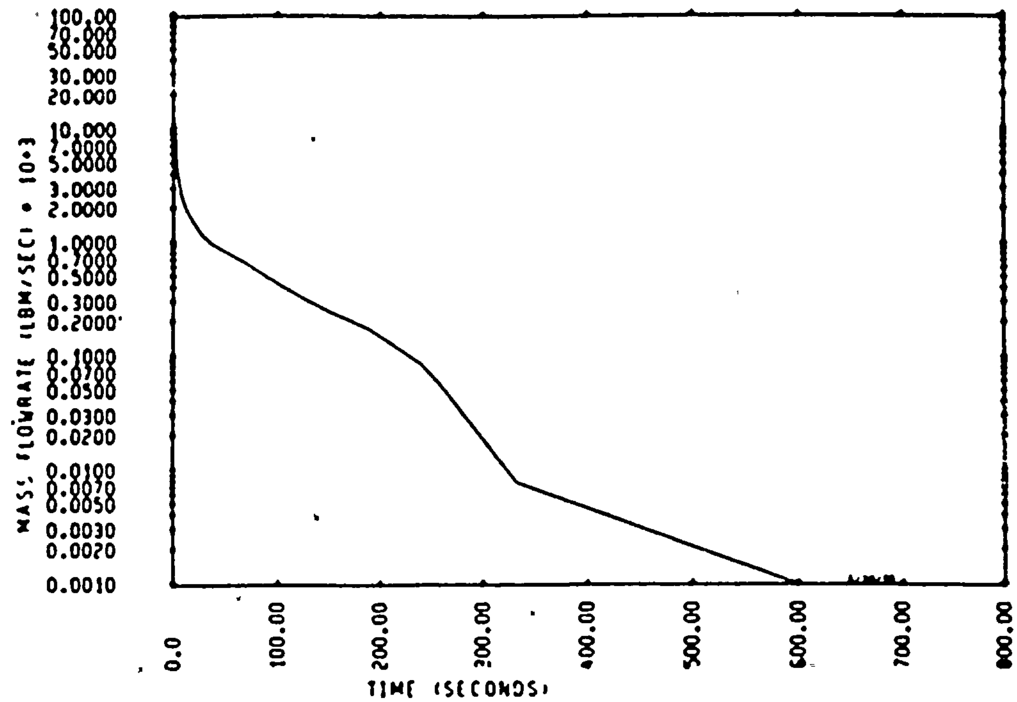


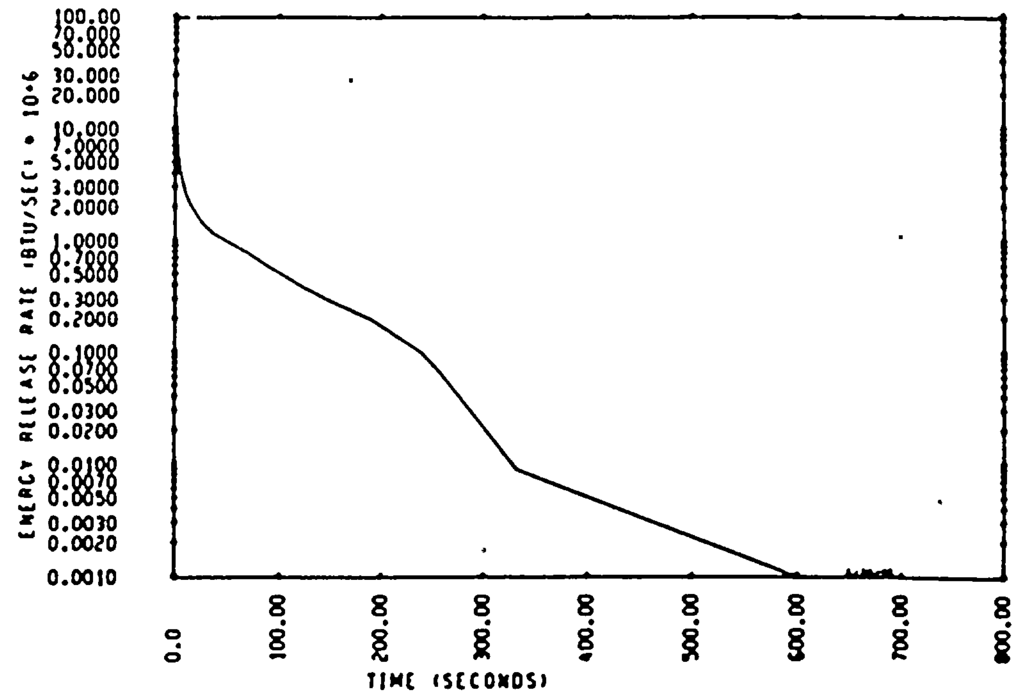
R. E. GINNA BAST CONCENTRATION REDUCTION STUDY	Figure 10 1.4 Ft^2 Per Steamline Break – 102% Power

12702.3



R. E. GINNA BAST CONCENTRATION REDUCTION STUDY	Figure 11 0.6 Ft^2 Split Steamline Break — 102% Power





R. E. GINNA BAST CONCENTRATION REDUCTION STUDY	Figure 12 4.37 Ft^2 Per Steambreak — 0% Power Mass and Energy Release Flow Rates