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ENCLOSURES:

ATWS Analysis for Turkey Point No. 3 & 4 ....

ACKNOWLEDGED

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PLANT NAME: Turkey Point #3 & 4

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FLORIDA POWER & LIGHT COMPANY

MAR 17 1975

March 13, 1975
L-75-129

Regulatory File Cy.

RECEIVED MAR 17 1975 U.S. ATOMIC ENERGY COMMISSION

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Case:

Re: Turkey Point Plant Units 3 & 4
Docket Nos. 50-250 & 50-251
ATWT Analysis

The AEC technical report WASH-1270 requests that all plants falling under Section IC, Category 1, of the Regulatory Staff ATWT position submit analyses of the consequences of ATWT events. Florida Power & Light Company, in compliance with this request, wishes to advise you that we are referencing the Westinghouse topical report, WCAP-8404, "Anticipated Transient Without Trip Analysis for Westinghouse PWR's With 44 Series Steam Generators", September, 1974, as the analysis applicable to Turkey Point Units 3 & 4. The method of analysis, models and computer codes as well as other pertinent information are presented in the Westinghouse topical report, WCAP-8330, "Westinghouse Anticipated Transients Without Trip Analysis", August, 1974, which we are also referencing as applicable to Turkey Point Units 3 & 4 in these areas. Appendix B of WCAP-8330 (post ATWT shutdown) demonstrates the capability of the Westinghouse NSSS to achieve a safe shutdown by operator action following the most limiting ATWT event. Appendices D and E demonstrate that peak containment pressure and radiological consequences of an ATWT are within acceptable limits.

We have reviewed the subject reports and have determined that the analyses and parameters listed in WCAP-8404 for the three loop plant represent the Turkey Point Units 3 & 4 with the following qualifications:

1. As shown in Table 1, the reactivity insertion rate for Turkey Point Units 3 & 4 can be as high as 0.46% $\Delta k/k$, while the analysis of the Rod Withdrawal at Power for

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a three loop plant in WCAP-8404 was performed for an insertion rate of 0.3% Δk/k. However, the Turkey Point Units 3 & 4 parameter is bound by a sensitivity study in WCAP-8404 which showed that even with a reactivity insertion rate of 0.5% Δk/k a satisfactory DNB ratio and an acceptable peak primary system pressure would be obtained. Additional sensitivity studies reported in WCAP-8404 show that other minor differences noted in Table 1, such as steam generator mass have a minimal effect on DNB margin or maximum reactor coolant system pressure.

2. The auxiliary feedwater pumps for Turkey Point Units 3 & 4 are turbine driven instead of motor driven. As a result, Turkey Point Units 3 & 4 auxiliary feedwater reaches full flow in just under 180 seconds (instead of the 60 seconds assumed for the three loop reference case in WCAP-8404). The auxiliary feedwater flow rate for Turkey Point may be as low as 600 gpm (instead of 1200 gpm assumed in WCAP-8404). As a result, the analysis for Loss of Load, Loss of Feedwater and Loss of AC Power (Station Blackout) presented in WCAP-8404 do not conform to Turkey Point Units 3 & 4 conditions as in each case the assumption is made that auxiliary feedwater flow of 1200 gpm will begin after 60 seconds. To assure a valid evaluation, the reactor vendor (Westinghouse) was asked to perform an additional analysis of the Loss of Feedwater, the most severe of these transients, using input parameters specific to Turkey Point Units 3 & 4 operating at 2300 MW_t and 2250 psia. This analysis is presented in the Appendix.

A comparison between the parameters used for the Turkey Point Units 3 & 4 Loss of Feedwater Analysis and those of the generic three loop plant analysis of WCAP-8404 is shown in Table 1. The table shows that the Turkey Point power operated relief valves have a larger steam flow capacity than the generic plant which compensates to some extent for the less favorable auxiliary feedwater characteristics. The result is a peak reactor coolant system pressure (including an 80 psi allowance for elevation and pump head) of 2991 psia for Turkey Point versus 2967 psia for the generic three loop plant analyzed in WCAP-8404 (p. 4-169). The conclusions for the Turkey Point Units as to the consequences of an ATWT are therefore identical to those for the generic study as summarized at the end of WCAP-8404. "The results of these studies show that in all reference cases Reactor Coolant System peak pressure does not exceed 3000 psia, the minimum DNB ratio is not less than 1.0, and containment peak pressure does not exceed design pressure. No impairment of the Reactor Coolant System boundary integrity is expected based on these peak pressures. Since the core thermal performance, the volume of reactor coolant and secondary fluid released, and the containment pressure transient are all less severe for these ATWT events than for design basis conditions, the radiological consequences of these postulated ATWT events are well within the guideline values set forth in 10 CFR Part 100."

In addition to analyses of the plant to demonstrate the consequences of an ATWT, WASH-1270 requests that a review of the reactor protection system be performed to assess the system susceptibility to common mode failure. These analyses have been previously performed for the relay logic protection system by Westinghouse and are documented in WCAP-7486, "An Evaluation of Anticiapted Operational Transients in Westinghouse PWR's", May, 1971.

In view of the results reported in WCAP-8404 and WCAP-7486 and the results of the plant specific analysis for the Loss of Feedwater ATWT for Turkey Point, and in recognition of the low likelihood of occurrence of these hypothetical ATWT events, we believe that no modifications of Turkey Point Units 3 & 4 are required to mitigate the consequences of an ATWT.

Very truly yours,

Robert E. Uhrig

Robert E. Uhrig
Vice President

REU:nch

cc: Mr. Jack R. Newman

TABLE 1

Parameter	WCAP-8404 ATWT Model	Re-Analysis of Turkey Point Units 3 & 4
Steam Generator Type	44 series	44 series
Number of Loops	3	3
Core Power (MW_t)	2300	2300
Nominal Pressurizer Pressure (psia)	2250	2250
Nominal Coolant Flow (gpm per loop)	89,500	88,500
Nominal Average Coolant Temperature (F)	575.5	575.5
Nominal Coolant No-load Temperature (F)	547	547
Total RCS Volume Including Pressurizer and Surge Line (ft^3)	9072	9343
Total Volume of Pressurizer and Surge Line (ft^3)	1328.6	1328.6
Number of Power Operated Relief Valves	2	2
Steam Capacity of Each Power Operated Relief Valve @ 2350 psia (lb/hr)	179,000	210,000
Number of Safety Valves	3	3
Steam Capacity of Each Safety Valve (lb/hr @ 2500 psia)	288,000	288,000
Best Estimate Rod Worth of Bank D at its Full Power Insertion Limit (%Δk/k)	0.3	0.46 (BOL-70%) 0.34 (70%-EOL)
Steam Generator Design Pressure (psia)	1100	1100
Steam Generator Nominal Steam Temperature (F)	523	516.1
Nominal Steam Flow (lb/sec)	2832	2798.7
Nominal Feedwater Temperature (F)	446.6	441.2
Nominal Fluid Mass in Steam Generator (lb)	238,500	236,340

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TABLE 1 (cont'd.)

Parameter	WCAP-8404 ATWT Model	Turkey Point Units 3 & 4
Auxiliary Feedwater Temperature (F)	130	<120
Auxiliary Feedwater Available (gal)	140,000	185,000
Capacity of Auxiliary Feedwater (gpm)	1200	600
Auxiliary Feedwater Initiation Time (sec)	60	180
Volume of Line Between Auxiliary Feedwater Connection on Feedline and Steam Generator Inlet, Total for all Loops (ft^3)	500	317

APPENDIX

Loss of Feedwater ATWT for Turkey Point Units 3&4

Loss of normal feedwater could result from a malfunction in the feedwater condensate system or its control system from such causes as simultaneous trip of both condensate pumps, simultaneous trip of both main feedwater pumps (or closure of their discharge valves), or simultaneous closure of all feedwater control valves. The vast majority of these cases would cause only a partial loss of feedwater flow. The most likely cause of a complete loss of feedwater would be a loss of station power (station blackout). Not withstanding the low probability of occurrence, a complete loss of normal feedwater is evaluated with the additional assumption that a non-mechanistic common mode failure prevents rods from dropping into the core.

The loss of main feedwater produces a large imbalance in the heat source/sink relationship. When feedwater flow to the steam generators is terminated, the secondary system can no longer remove all of the heat that is generated in the reactor core. This heat buildup in the primary system is indicated by rising Reactor Coolant System temperature and pressure, and by increasing pressurizer water level, which is due to the insurge of expanding reactor coolant. Water level in the steam generators drops as the remaining water in the secondary system, unreplenished by the main feedwater flow, is boiled off. When the steam generator water level falls to the point where the steam generator tubes are exposed and primary-to-secondary system heat transfer is reduced, the reactor coolant temperature and pressure begin to increase at a greater rate. This greater rate of primary system temperature and pressure increase is maintained as the pressurizer fills and releases water through the safety and relief valves. (The safety and relief valves have a smaller volumetric relief capacity for water than for steam.) Reactivity feedback, due to the high primary system temperature reduces core power. The system pressure begins to decrease and a steam space is again formed in the pressurizer.

The loss of feedwater ATWT involves a heat source/sink mismatch; therefore, the peak pressure attained in the primary system depends upon the

ability of the pressurizer safety and relief valves to release the reactor coolant volumetric insurge to the pressurizer. The volumetric relief capacities of these valves are reduced when the pressurizer fills and water is passed instead of steam. During a loss of feedwater ATWT, the heat source/sink mismatch causes the reactor coolant temperature and coolant expansion rate to increase and the core reactivity and power to drop. Reduction of the pressurizer safety and relief valve volumetric relief capacity (due to filling the pressurizer and relieving water) early in the transient when core power is still relatively high, will result in a higher peak Reactor Coolant System pressure than the peak pressure that would result from reduction of pressurizer relief and safety valve capacity later in the transient, when core power is lower.

For protection against the effects of a loss of feedwater, the reactor would be tripped when any of the following conditions are reached:

°Steam/feedwater flow mismatch (low feedwater flow) and low steam generator water level (40% mismatch and 25% of narrow span, respectively)

°Overtemperature Delta T reactor trip

°High pressurizer pressure (2400 psia)

°High pressurizer level (92% of span)

°Steam generator low-low water level (10% of narrow span)

°Low reactor coolant flow (90% of nominal)

Analysis of Effects and Consequences

The following assumptions were made in the analysis:

°Initial normal full power operation early in core life. Since

the negative temperature coefficient of reactivity reduces core power as the coolant temperature rises, and the temperature coefficient becomes more negative with core life, the ATWT loss of feed is less severe later in core life. The reactivity coefficients are the same as those used in WCAP-8404.*

°Normal operation of the following control systems:

1) Pressurizer pressure control, including heaters, spray, and both the power-operated and the spring-loaded valves.

2) Turbine governor valves in impulse pressure control prior to trip, and valve closure on turbine trip.

3) Steam dump to condenser at 40% of rated turbine flow following turbine trip.

°Turbine trip 30 seconds after loss of feed (this is after generation of a reactor and turbine trip signal on low steam generator water level in coincidence with steam/feed flow mismatch)

°No credit for automatic reactor trip

°No credit for automatic control rod insertion as reactor coolant temperature rises

°Main feedwater flow falls to zero in the first 4 seconds of the transient, with no main feed after that time.

°Auxiliary feedwater flow begins at 180 seconds, at a rate of 600 gpm.

°Auxiliary feedwater is injected into the feedwater pipe at a temperature of 120°F, 317 ft^3 upstream of the steam generator, such that the cooler water enters the steam generator after this volume is purged.

*WCAP-8404, "Anticipated Transients Without Trip Analysis for Westinghouse PWR's with 44 Series Steam Generators", September, 1974.

°Primary-to-secondary heat transfer area is reduced as the steam generator shell-side water inventory drops below the value necessary to wet the tubes.

°A list of the important plant parameters are given in Table 1.

Results

The peak pressure in the Reactor Coolant System was 2911 psia and occurred approximately 113 seconds after the termination of feedwater supply to the steam generators. The pressurizer reached a peak pressure of 2875 psia at the same time.

The chronology of events for this case is shown in Table 2 and transient plots are presented in Figures 1 through 12.

Figure 2 depicts the primary system mass flow rate as a fraction of nominal. The gradual drop in flow rate, before pump cavitation occurs, is due to coolant expansion (drop in density). The volumetric flow rate, however, is relatively constant before the pump is assumed to cavitate.

In addition to the automatic reactor trips, the operator may shutdown the reactor with emergency boration or safety injection.

Conclusions

The DNB ratio for the hot channel increases above its initial value during the transient. The peak Reactor Coolant System pressure (including a 80 psi allowance for elevation and pump head) is 2991 psia. Based on these results, no core damage or impairment of the Reactor Coolant System integrity is expected for the loss of feedwater ATWT.

TABLE 1

TURKEY POINT PARAMETERS FOR ATWT ANALYSIS

Parameters	Value
Core:	
Core power (MW_t)	2300
Core length (ft)	12
Number of assemblies	157
Reactor Coolant System:	
Total volume (ft^3) including pressurizer and surge line	9343
Nominal[a] pressure (psia)	2250
Nominal[a] flow (gpm)	265,500
Nominal[a] average temperature (°F)	575.45
No-load temperature (°F)	547
Nominal[a] reactor vessel inlet temperature (°F)	546.2
Nominal[a] reactor vessel outlet temperature (°F)	604.7
Pressurizer:	
Total volume of pressurizer and surge line (ft^3)	1328.6
Nominal[a] water volume (ft^3)	808
Heater capacity (kw)	1300
Maximum spray rate (lbs/sec)	63.1
Power-operated relief valve steam flow capacity (lbs/hr) (at 2350 psia)	2-210,000 ea.
Safety valve steam flow capacity (lbs/hr) (at 2500 psia)	3-288,000 ea.
Power-operated relief valve opening pressure (psia)	2350
Safety valve, start open-full open pressure (psia)	2515-2590
Secondary System:	
Steam generator (SG) type	44
SG design pressure (psia)	1100
Nominal[a] steam pressure (psia)	785
No-load steam pressure (psia)	1020
Nominal[a] steam temperature (°F)	516.1
Nominal[a] steam flow (lbs/sec)	933/SG

TABLE 1 (cont'd.)

Parameters	Value
Nominal[a] SG secondary side fluid mass (lbs)	78780/SG
Maximum steam moisture (%)	0.25
Nominal[a] feed enthalpy (Btu/lb)	419.3
Auxiliary Feed flow capacity (gpm)	600
Auxiliary feed purge volume (ft^3)	317
Auxiliary feed water available (gal)	185,000
Auxiliary feed enthalpy (Btu/lb)	100

Note:

[a]Nominal refers to value at rated full power

TABLE 2

SEQUENCE OF EVENTS FOR LOSS OF FEEDWATER WITHOUT A REACTOR TRIP

Event	Time (Sec)
Main Feedwater Supply to All Steam Generators is Terminated	0-4
Steam generator water level begins to fall	
Steam temperature and presssure begin to rise	
Reactor Coolant System temperature and pressure begin to increase	
Expanding reactor coolant surges into the pressurizer, causing level to rise	
Core power begins to drop	
Reactor/Turbine Trip Signal: Low Steam Generator Water Level and Steam/Feed Flow Mismatch	7
Reactor/Turbine Trip Signal: Low-Low Steam Generator Water Level	11
Turbine is Assumed to Trip	30
40% steam dump to condensers	
Reactor coolant temperature rises more steeply	
Core Power declines more rapidly	
Power-Operated Relief Valves on the Pressurizer Open and Release Steam	33
Pressurizer pressure >2350 psia rises very rapidly	
Reactor/Turbine Trip Signal: Overtemperature Delta T	41
Steam Generator Safety Valves Open and Hold Steam Pressure Constant	47
Total steam flow rises rapidly above the 40% being dumped to the condenser	
Core power decline begins to slow, as more heat is removed by increased steam flow	
Pressurizer Pressure Reaches a Peak of 2367 psia	43
Pressurizer Relief Valves Close	55
Total Steam Flow From the Steam Generators Reaches a Peak of 88% of Nominal Flow	66
Reactor Coolant System and pressurizer pressure drops	
Core power decreases very slowly	

TABLE 2 (cont'd.)

Event	Time (Sec)
Steam Generator Tubes are Effectively Uncovered and UA Begins to Decrease	68
Primary system pressure rises rapidly	
Steam flow and core power drops more rapidly	
Pressurizer Relief Valves Open	70
Pressurizer pressure levels off at2350 psia, as relief valves release steam	
Reactor/Turbine Trip Signal: High Pressurizer Water Volume	72.5
Rapid Rise in Pressurizer Pressure - Relief Valves Cannot Release Steam Fast Enough to Hold Pressure at 2350	73
Reactor/Turbine Trip Signal: High Pressure	80.4
Pressurizer Safety Valves Open	82
Pressurizer Fills With Water	80
No steam space remains in the pressurizer	
Relief valves release water	
Pressurizer pressure continues to rise rapidly	
Steam Generator Safety Valves Close	81
Total steam flow consists of steam dump to condenser	
Peak Reactor Coolant System Pressure is Reached (2911 psia)	113
Peak Pressurizer Pressure is Reached (2875 psia)	113
Reactor/Turbine Trip Overpower Delta T Trip	114
Pump is Assumed to Cavitate (Cold Leg Temperature is $\leq$6°F Below Saturation Temperature)	135
Reactor/Turbine Trip Signal: Low Reactor Coolant Flow	137

TABLE 2 (cont'd.)

Event	Time (Sec)
All Auxiliary Feedwater Pumps are Assumed to Start	180
Purging of the main feedwater lines begins; main feedwater remaining in the lines is pushed into the steam generators by the auxiliary feedwater (3% of nominal feedwater flow)	
Steam flow slowly decreases	
Core power slowly decreases	
Primary system pressure drops	
Steam Space Forms in Pressurizer, as Water Level Begins to Drop	
All pressurizer valves are closed	250-260
Purging of the main feedwater lines by auxiliary feedwater is completed - colder (auxiliary feedwater) water 100 BTU/lb enters the steam generators and heat transfer in the steam generator increases slightly	
Core power at 3% of nominal, flow at 3%	600
Primary system pressure is 2192 psia	

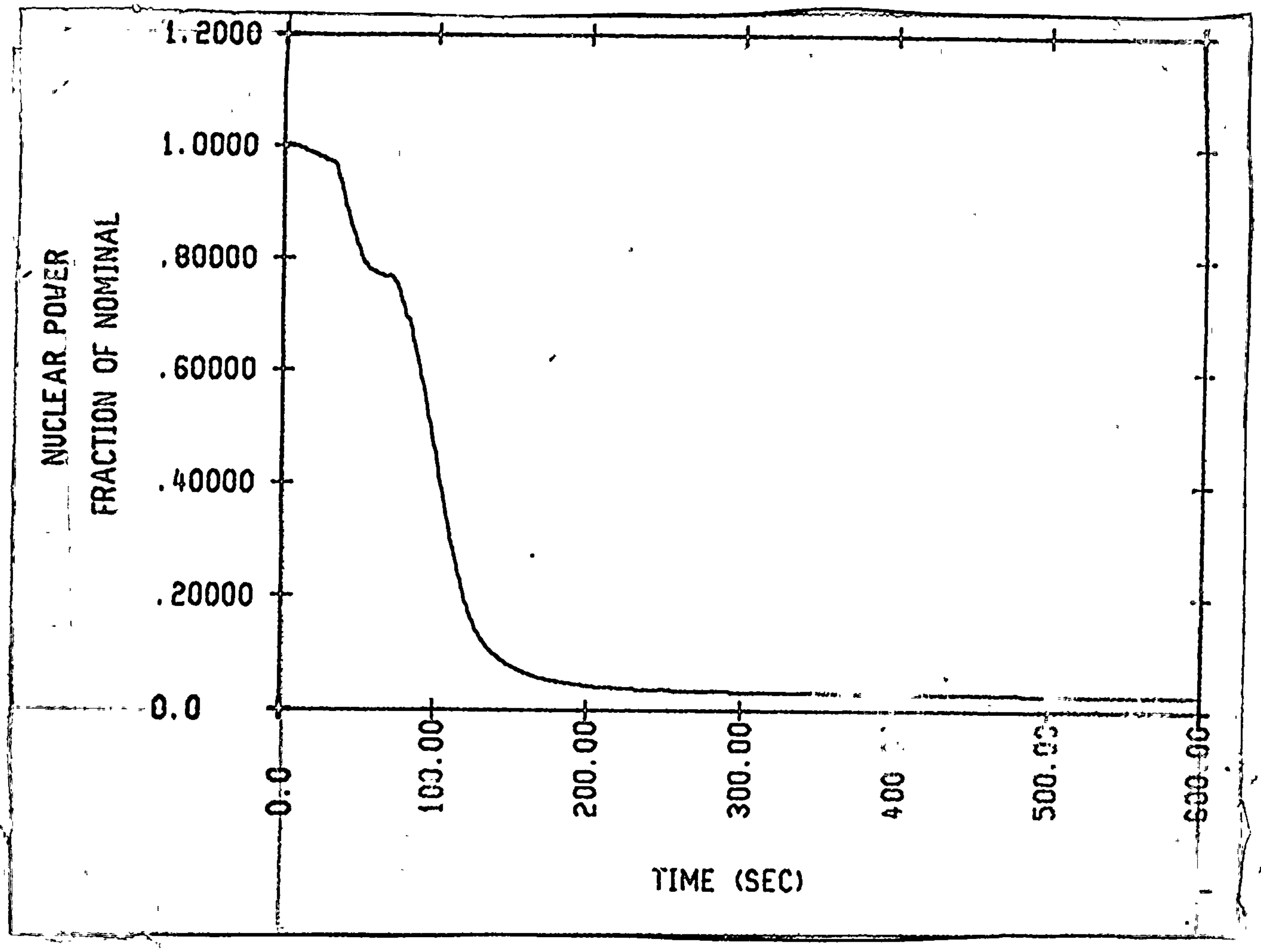

Figure 1 Nuclear Power Versus Time

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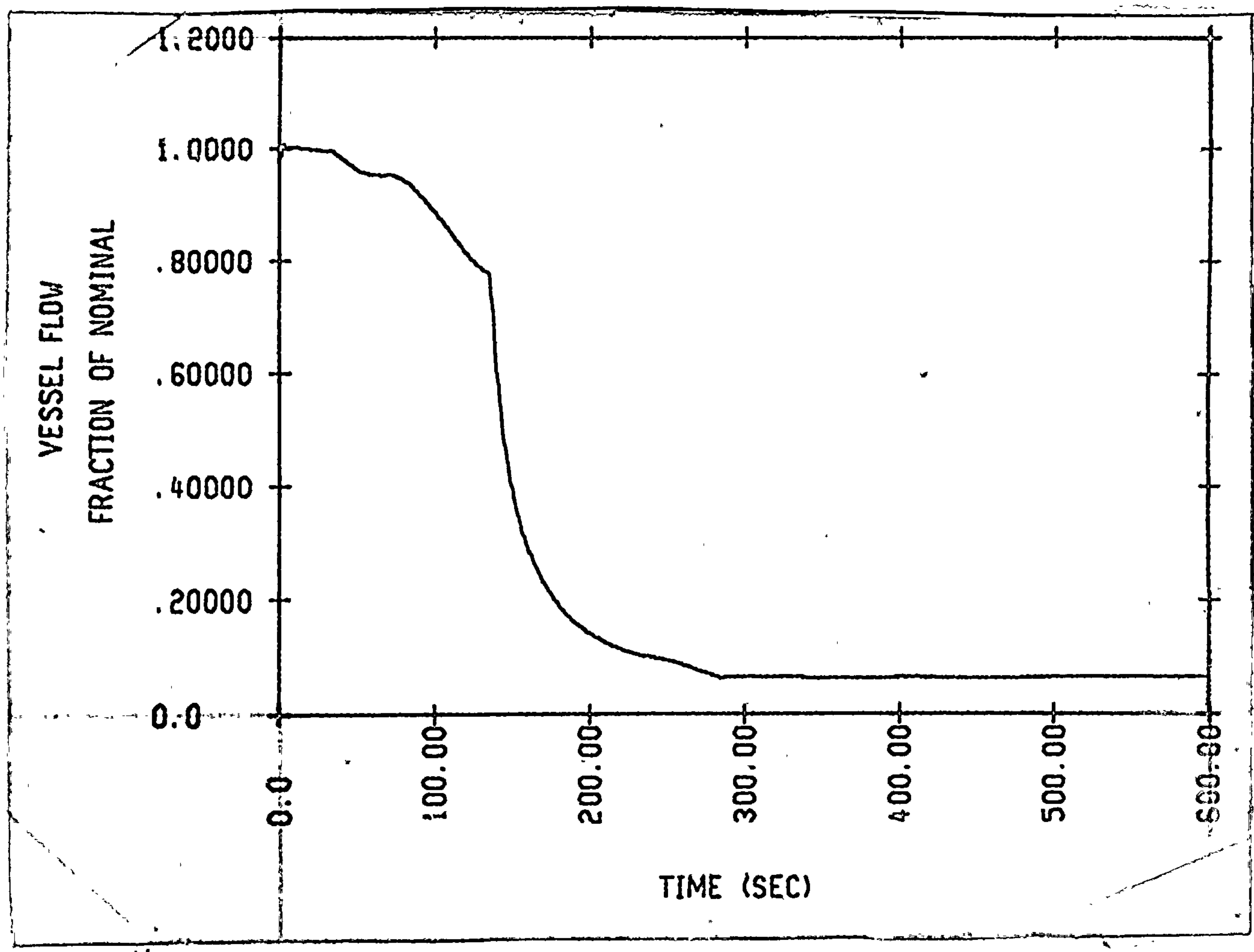

Figure 2 Vessel Flow Versus Time

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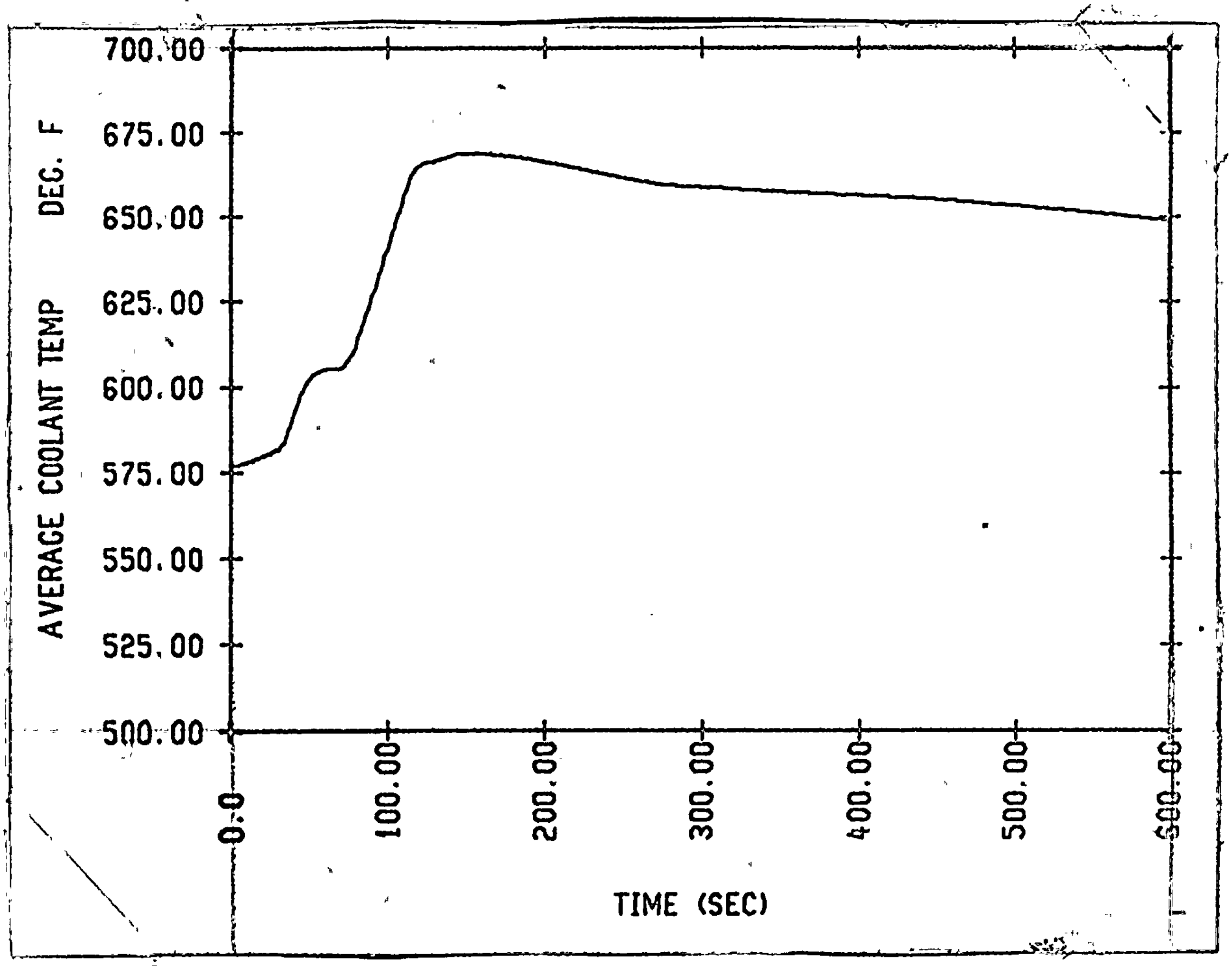

Figure 3 Average Temperature Versus Time

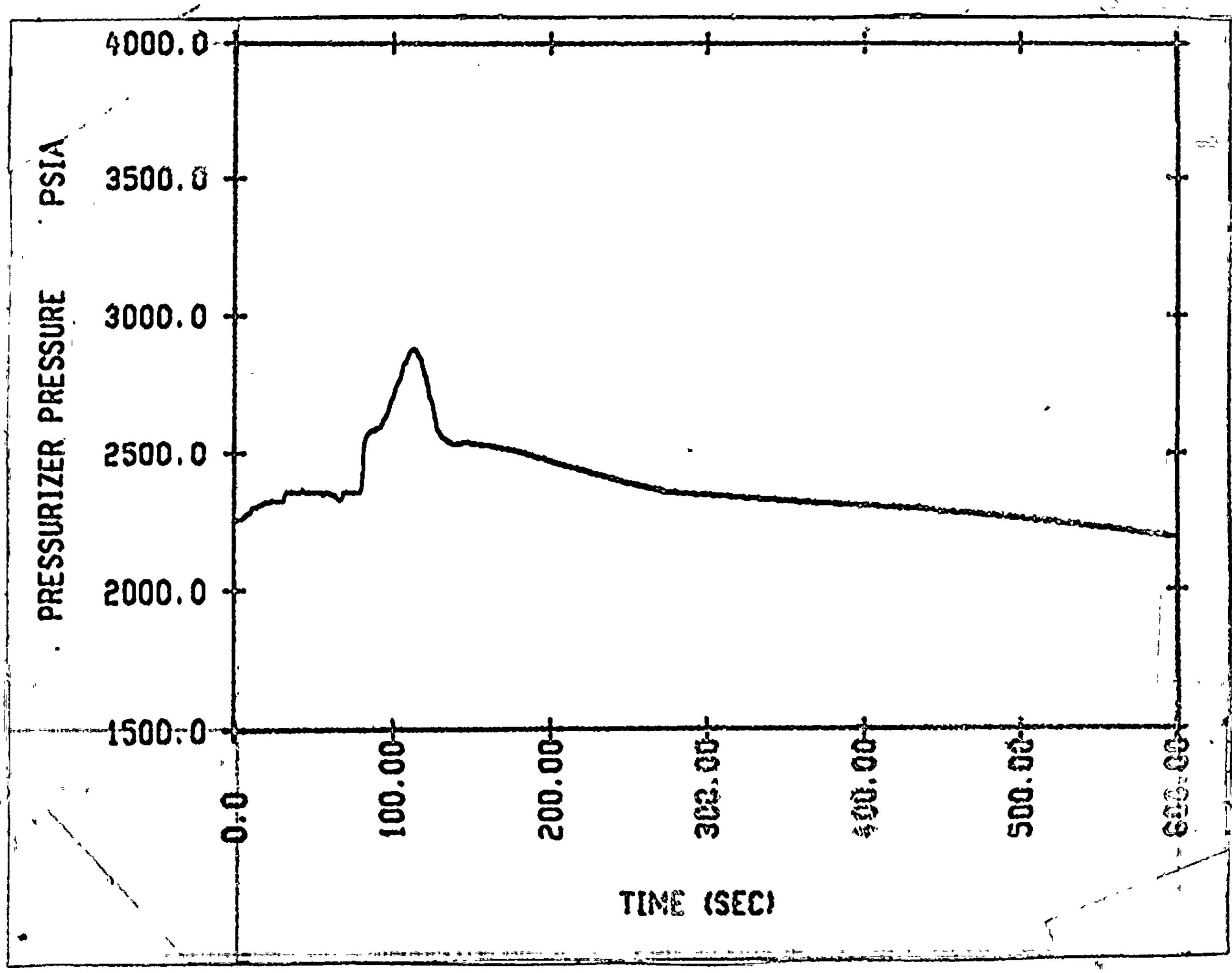

Figure 4 Pressurizer Pressure Versus Time

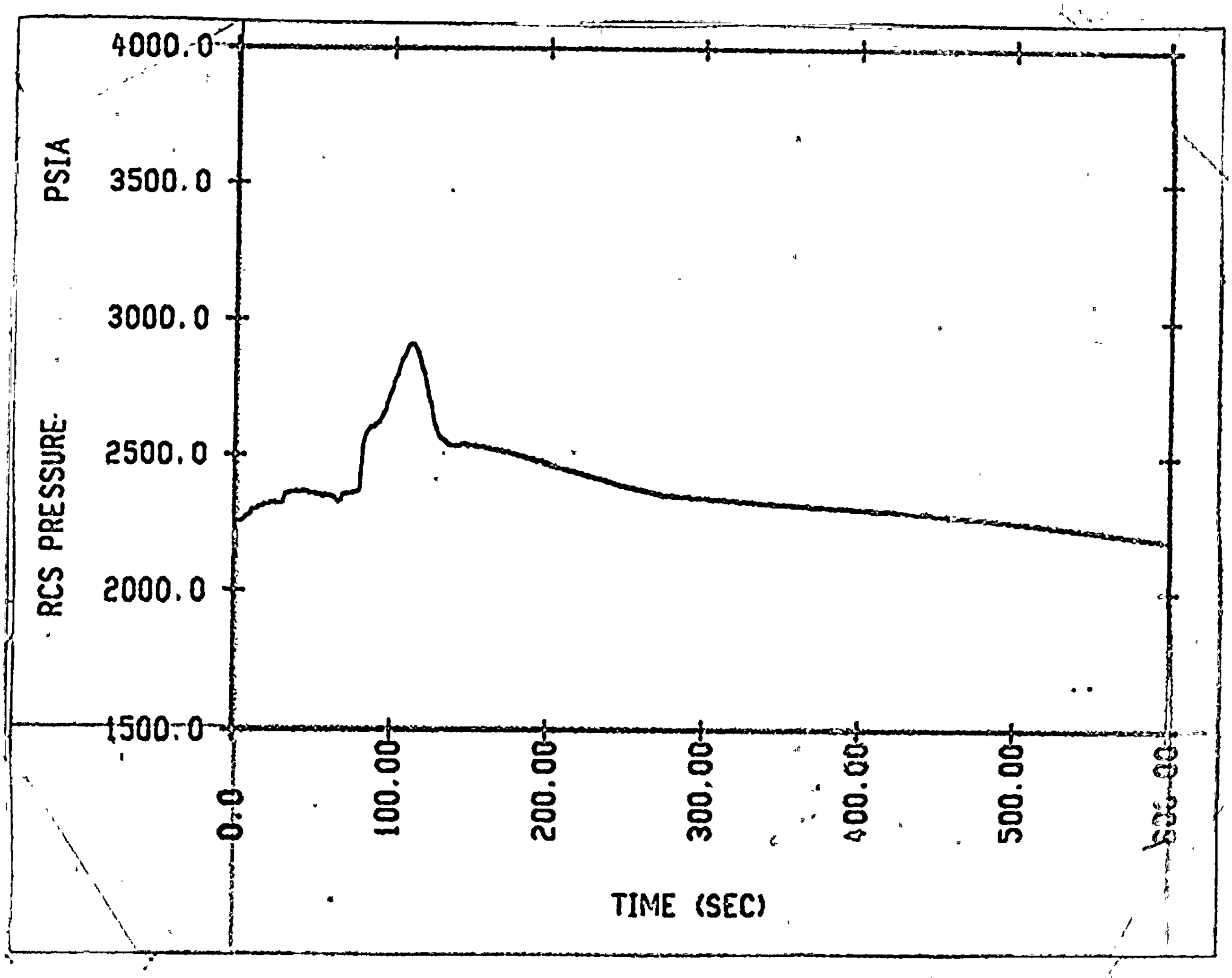

Figure 5 RCS Pressure Versus Time

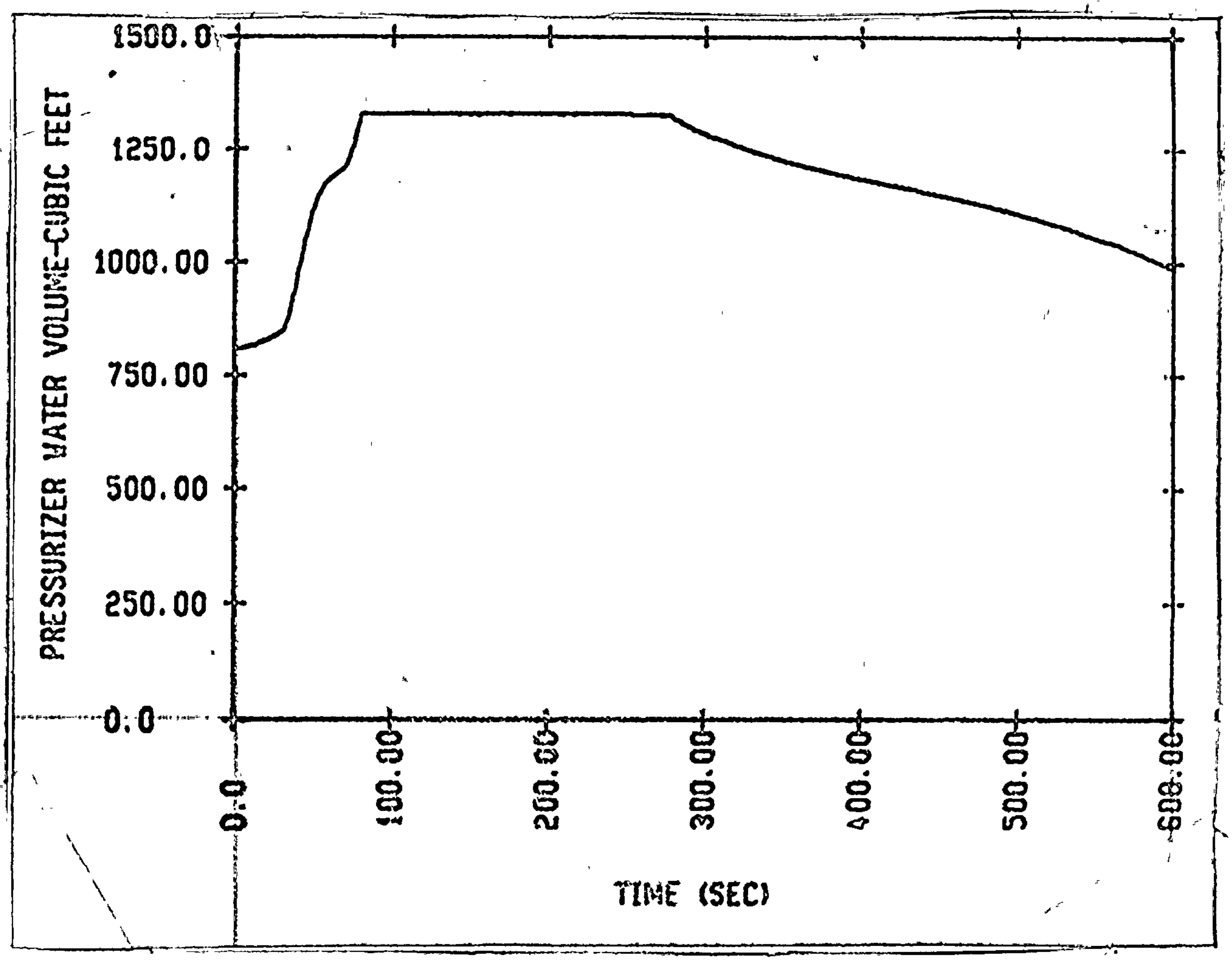

Figure 6 Pressurizer Water Volume Versus Time

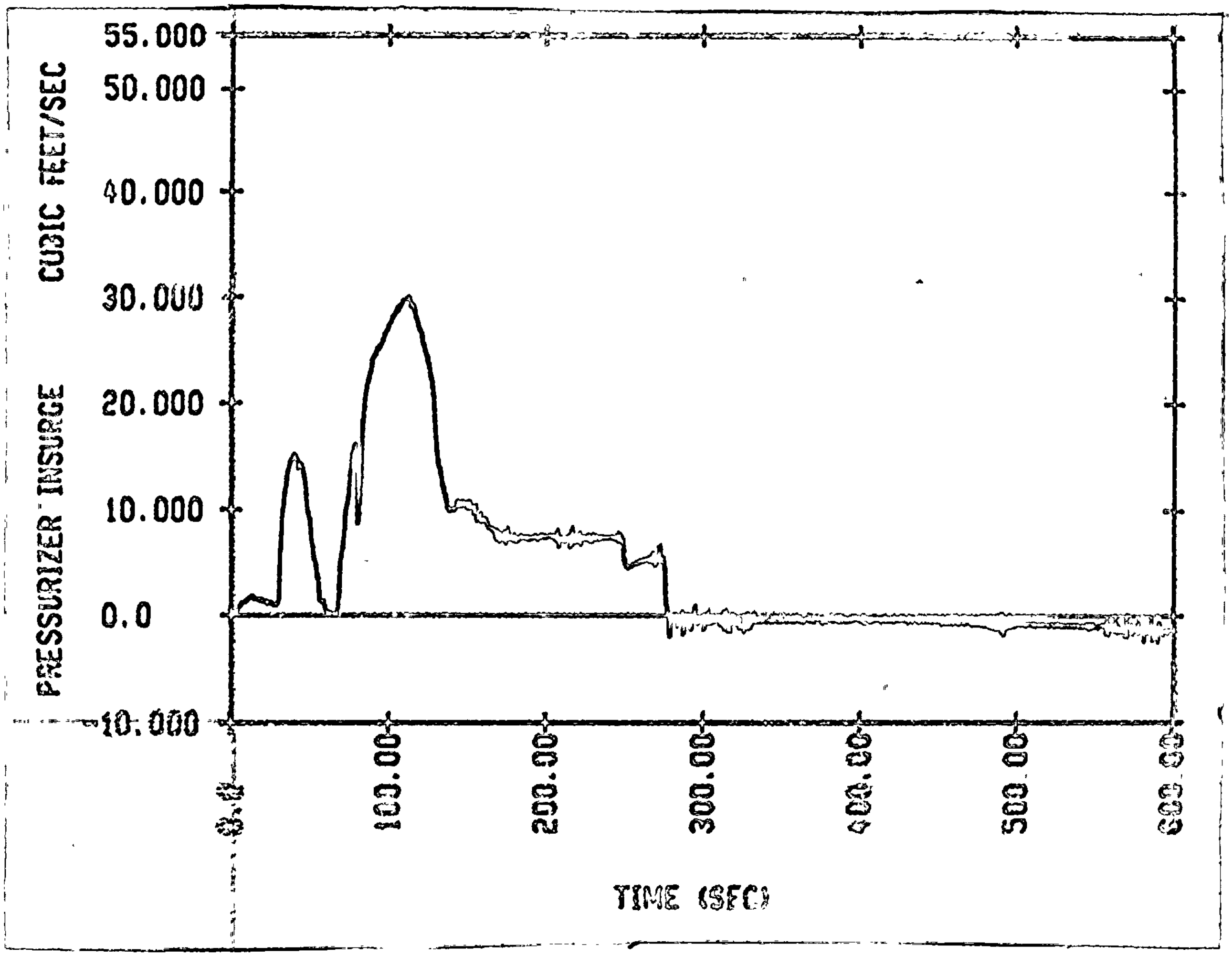

Figure 7 Pressurizer Insurge Versus Time

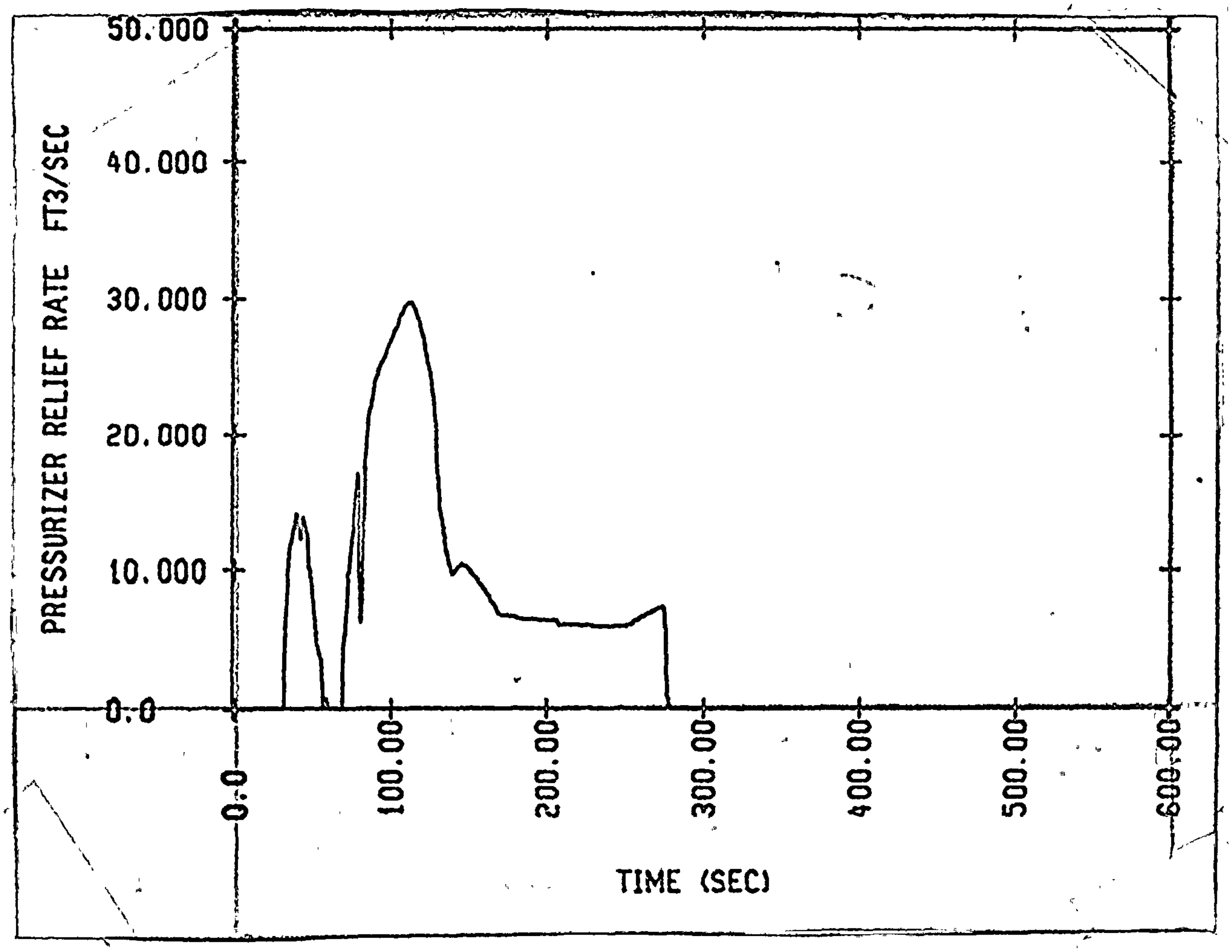

Figure 8 Pressurizer Relief Rate Versus Time

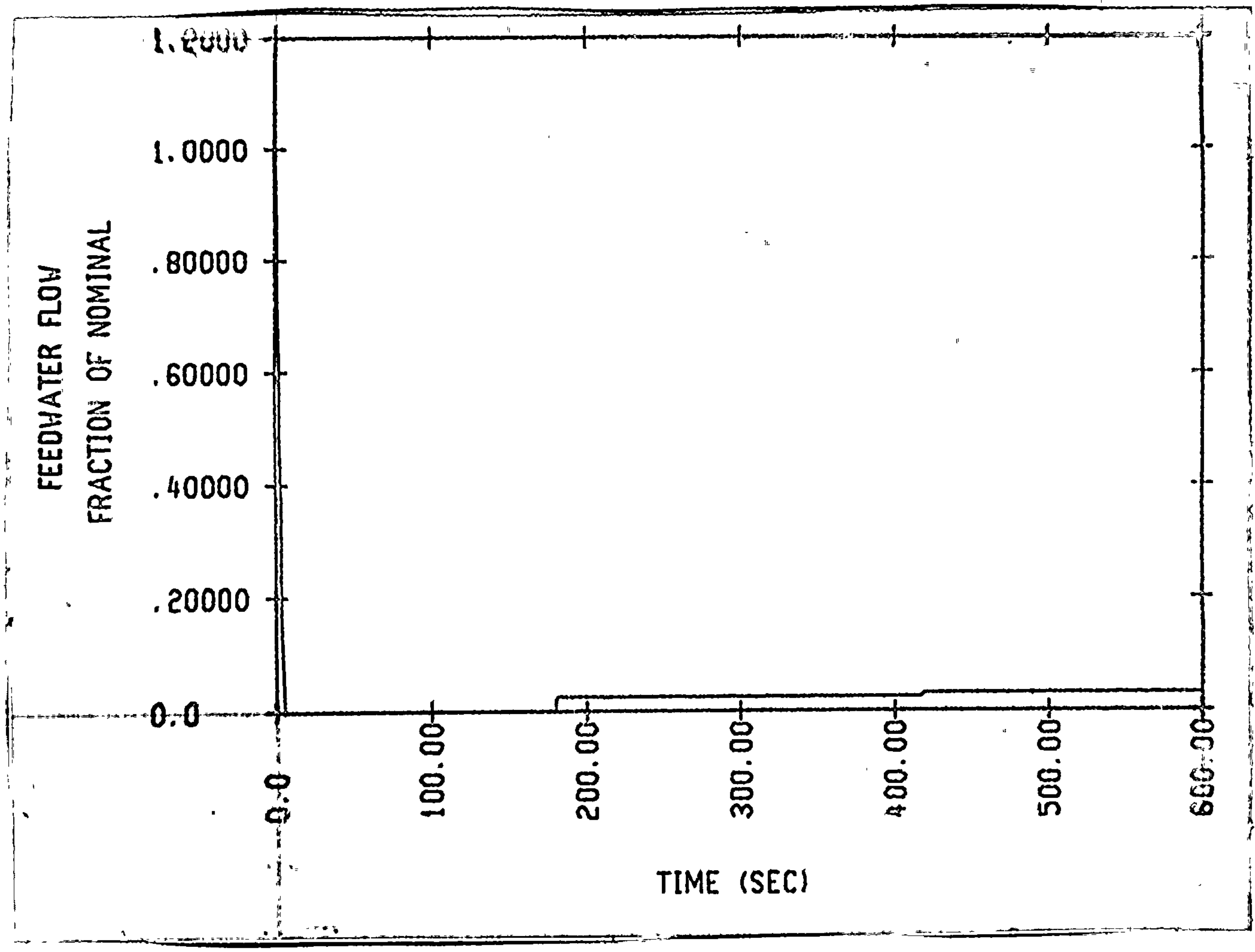

Figure 9 Feedwater Flow Versus Time

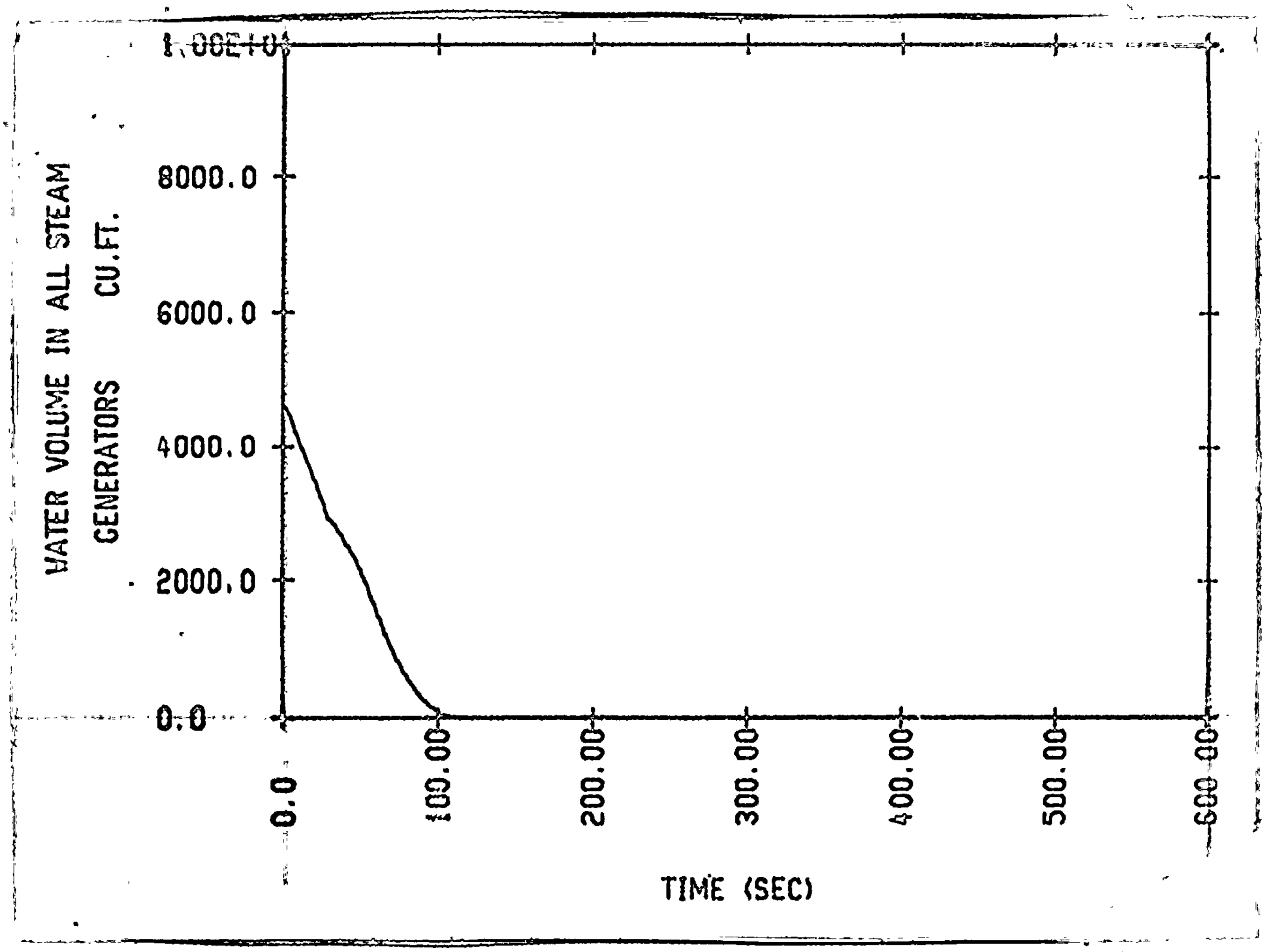

Figure 10 Water Volume in all Steam Generators Versus Time

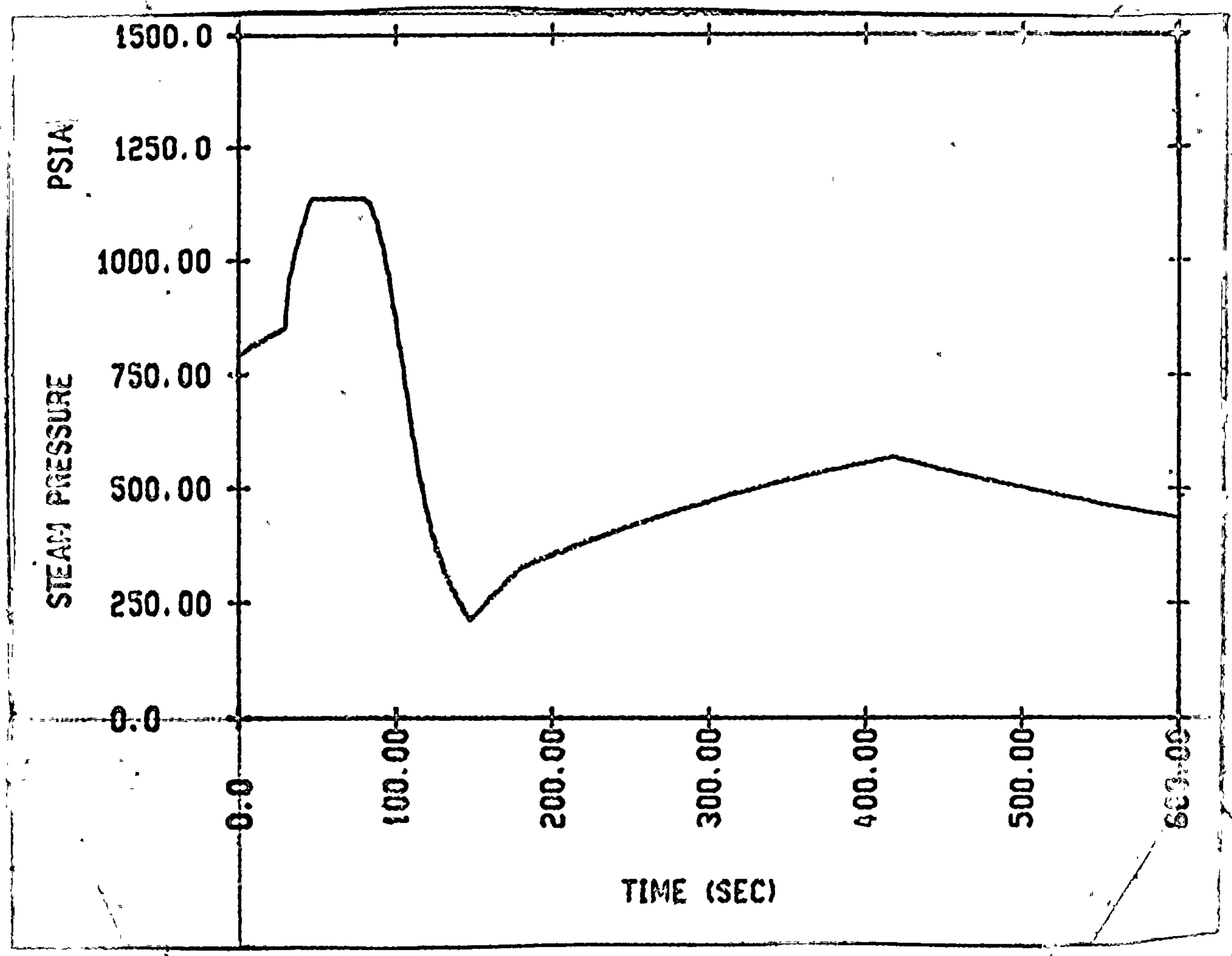

Figure 11 Steam Pressure Versus Time

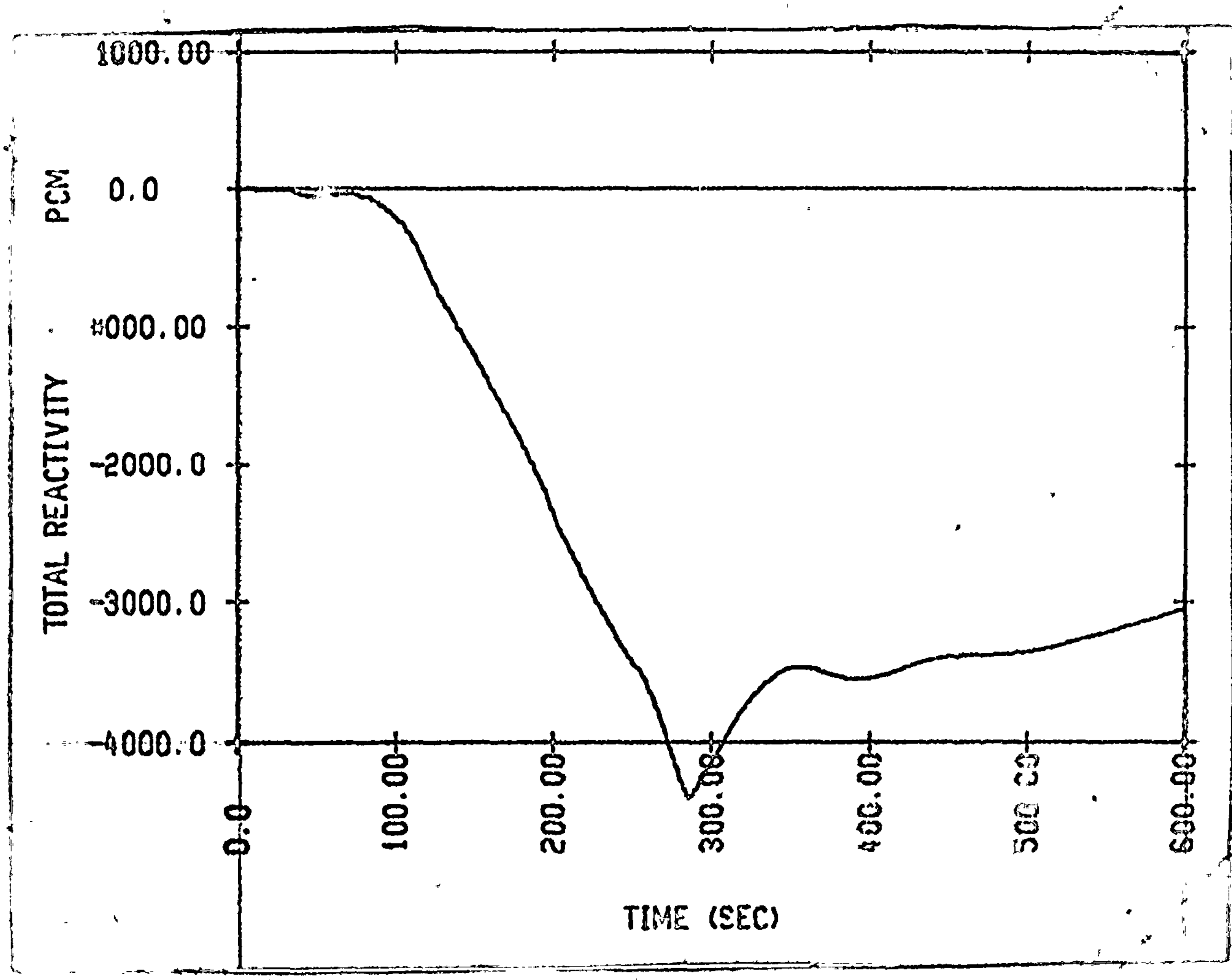

Figure 12 Total Reactivity Versus Time