CONNECTICUT YANKEE ATOMIC POWER COMPANY

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May 19, 1980

Docket No. 50-213

Director of Nuclear Reactor Regulation
Attn: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References: (1) D. G. Eisenhut letter to D. C. Switzer dated October 11, 1979.
(2) W. G. Counsil letter to D. G. Eisenhut dated December 4, 1979.
(3) D. M. Crutchfield letter to W. G. Counsil dated May 7, 1980.

Gentlemen:

Haddam Neck Plant
Auxiliary Feedwater Systems

In Reference (1), Connecticut Yankee Atomic Power Company (CYAPCO) was requested to respond to Enclosure 2 of that reference regarding a generic request for additional information on Auxiliary Feedwater System flow requirements. As indicated in Reference (2), and confirmed by Reference (3), CYAPCO estimated completion of this effort on May 15, 1980. In fulfillment of that request, the attached information is being docketed regarding the design basis transient and accident conditions for the Auxiliary Feedwater System at the Haddam Neck Plant. In establishing the Auxiliary Feedwater System flow requirements, the following conditions were evaluated:

(1) Loss of Main Feedwater (LMFW).
(2) LMFW with Loss of Offsite AC Power.
(3) LMFW with Loss of Onsite and Offsite AC Power.
(4) Plant Cooldown.
(5) Turbine Trip With and Without Bypass.
(6) Main Steam Isolation Valve Closure.
(7) Main Steamline Break.
(8) Small Break LOCA.

In accordance with the provisions of Enclosure 2 of Reference (1), it is emphasized that the main feedline break is not considered in this evaluation as it is not a design basis event for the Haddam Neck Plant. Therefore, CYAPCO's conclusion regarding the adequacy of the existing Auxiliary Feedwater

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System does not consider this postulated accident. The analytical results presented in the attachment for the feedline break are provided for informational purposes only. Please recognize that any Staff recommendations for modifying the Auxiliary Feedwater System as a result of the review of the feedline break analysis will be considered inappropriate.

Detailed responses to the specific requests of Enclosure 2 of Reference (1) are incorporated into the text of the attached document. Based upon these analyses, CYAPCO has concluded that the Auxiliary Feedwater System is adequately sized and designed to comply with the acceptance criteria identified in Section 1.4. Therefore, no modifications are contemplated as a result of completion of this effort.

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

W. G. Counsil

W. G. Counsil
Vice President

Attachment

By: W. F. Fee

W. F. Fee
Vice President

Connecticut Yankee

Basis for Auxiliary Feedwater System Flow Requirements

Introduction

Enclosure 2 of the October 11, 1979 letter from Mr. Eisenhut to Mr. D. C. Switzer (docket number 50-213) requested the Connecticut Yankee Atomic Power Company to provide Auxiliary Feedwater (AFW) System design basis information as applicable to the design basis transients and accident conditions for their nuclear facility. This letter contains the requested information.

The following plant transients and accident conditions have been considered in establishing AFW flow requirements:

- Loss of Main Feedwater (LMFW)
- LMFW with loss of offsite AC power
- LMFW with loss of onsite and offsite AC power
- Plant Cooldown
- Turbine trip with and without bypass
- Main steam isolation valve closure
- Main steam line break
- Small break LOCA

The feedwater line break accident is not included as part of the design basis for the Connecticut Yankee Atomic Power Plant (CY). However, we have included in this report the results of a detailed study of the AFW system performance in the event of this accident.

1. Discussion of Plant Transients Considered in AFW Design

In order to assess the performance of the auxiliary feedwater system, the adequacy of the minimum available flow during the loss of heat sink (LOHS) events listed above must be demonstrated. By "adequacy" we mean the ability of the available flow, assuming single failures and conservatisms as defined in section 3.0, to remove primary side heat to a degree that the plant acceptance criteria for the events are not violated.

In addition to LOHS type events, the adverse effects of the maximum deliverable AFW flow on the most severe RC overcooling events must be assessed.

1.1 Adequacy of Minimum AFW Flow

The following events have been addressed to assess the adequacy of minimum AFW flow:

1.1.1 LMFW - This event is the bounding case as far as this type of event is concerned. As a result, an analysis has been performed to determine adequacy of minimum AFW flow for this event. The results of this analysis are described in section 2.2.

1.1.2 LMFW with Loss of Offsite AC Power - In this case, the pumps have tripped and it is no longer necessary to remove pump heat through the steam generators. Therefore, the AFW flow requirement will be smaller for this case than for case 1.1.1. Results of a sensitivity study supporting this conclusion are included. Since the AFW system is completely independent of offsite AC power, this case is then clearly bounded by case 1.1.1.

1.1.3 LMFW with Loss of Onsite and Offsite AC Power - This is identical to case 1.1.2 as far as AFW flow requirements are concerned. Since the system is completely independent of both offsite and onsite AC power, it too is bounded by case 1.1.1.

1.1.4 Turbine Trip with and without Bypass - A turbine trip results in an immediate reactor scram. If bypass is not available, the steam generator pressure will rise to the safety valve setpoint and remain there. The same assumption was made in case 1.1.1, with the additional conservatism of having the steam generator level at the low level setpoint. For this case, we have assumed nominal initial steam generator level. Therefore, this case is bounded by case 1.1.1. If steam bypass is available, the steam generator will stabilize at about 910 psi instead of the 1000 psi safety valve setpoint. This means that AFW is being pumped against a lower head, and hence more flow is provided. Also this case is bounded by case 1.1.1.

1.1.5 Main Steam Isolation Valve (MSIV) Closure - Closure of any MSIV results in reactor trip. The steam generators will pressurize up to the safety valve setpoint. This accident produces the same effects of a turbine trip without bypass as far as the AFW system is concerned, and hence it is also bounded by case 1.1.1.

1.1.6 Small Break LOCA - For a certain spectrum of sizes of small break LOCAs, the steam generators will be required to remove that fraction of decay heat not being removed through the break itself. Hence, the AFW flow rates required will be less then that required for the loss of feedwater accident (case 1.1.1) where all the decay heat must be removed through the steam generators. Since no component of the AFW system is shared with the ECCS system (e.g., the diesels which power ECCS do not power AFW pumps), it is demonstrated that a small break LOCA is bounded by case 1.1.1 as far as the AFW system is concerned.

1.2 Acceptability of Max AFW Flow

The following events have been considered to determine the acceptability of the maximum AFW flow rate.

1.2.1 Steam Line Break - Run-out flow during a main steam line break will maximize the consequences of excessive AFW flow on plant response. As a result, an analysis has been performed to determine acceptability of the maximum AFW flow for this event. The results of this analysis are described in section 2.1.

1.2.2 Plant Cooldown - The cooldown resulting from delivery of the maximum possible amount of AFW to the steam generators (750 gpm at 900 psig) has been calculated to be less than 60°F in the first 10 minutes following a scram. Although this calculation is very conservative because it neglects the substantial contribution of decay heat, the results are within the acceptance criteria as defined in section 1.4.2.2. Plant cooldown due to spurious actuation of auxiliary feedwater has not been analyzed because the S.G. level control system will prevent an unacceptable cooldown. Failure of the control system will result in the trip discussed above.

1.3 Accident Not in the Design Basis

1.3.1 Feedline Break - This accident is not in the design basis for the AFW system at CY. However, an analysis was performed as requested and its results are included in this submittal (section 2.3) for informational purposes.

1.4 Acceptance Criteria

The criteria for determining if the AFW flow rate is acceptable for the events described in sections 1.1 and 1.2 are given below:

1.4.1 Adequacy of Minimum AFW Flow (section 1.1) - The AFW flow rate shall be considered adequate for events 1.1.1 through 1.1.5 providing:

1.4.1a - The pressurizer pressure corresponding to the PORV setpoint (2285 psia) is not reached as a result of low steam generator inventory.

1.4.1b - No DNB condition is experienced at the clad surface of any fuel rod in the core.

1.4.1c - Sufficient steam generator level remains to remove the primary side heat generated. The RETRAN model used calculates this point to be when the collapsed level in the steam generator secondary side falls below 10% of the height of the U-tubes.

For event 1.1.6 (small break LOCA) the criterion will be that sufficient steam generator liquid level will remain to remove that fraction of primary side heat generated which is not removed via the break, such that 10CFR50.46 limits are not exceeded.

1.4.2 Acceptability of Maximum AFW Flow (section 1.2) - Two separate criteria are to be considered here; one for the cooldown resulting from a steam side accident and one for the cooldown resulting from a shutdown.

1.4.2.1 Effect of maximum run-out flow on the limiting secondary steam release accident: The AFW flow rate shall be considered not to exceed the maximum permissible limit providing the reactor does not return to critical in the ten minute period following the accident.

1.4.2.2 Cooldown following normal reactor shutdown: The AFW flow rate shall be considered not to exceed the maximum permissible providing the primary side coolant does not cool down more than $100^{o}F$ in the ten minute period following the shutdown.

For the above cases, it is assumed that the operator would take action at ten minutes into the transient to control the AFW system which is causing the overcooling.

2. Analyzed Events

Sections 1.1 and 1.2 have reduced the number of transients to be included in the design basis to two limiting events. These events are the steam line break and the loss of feedwater. A summary of the method of analysis and the results obtained are included in this section. In addition, a description of the feedline break (which is not included in the design basis) is included for informational purposes.

2.1 Steam Line Break - The analysis of the steam line break accident with AFW available has previously been docketed (ref. 1). This analysis assumed a circumferential rupture of a 24 inch diameter steam line. A conservatively high value of AFW flow was calculated assuming all pumps are automatically initiated and delivering water in a run-out condition due to reduced back pressure at the broken steam generator. The worst pressure split between the steam generators (900 psia intact, 0 psia broken) was assumed in order to divert the maximum amount of AFW flow into the broken steam generator. It was assumed that the operator acted to isolate the AFW system from the break ten minutes into the transient. The results showed that no return to criticality occurred, even with the most reactive rod stuck in the withdrawn position. Hence, the AFW system met the acceptance criteria defined in section 1.4.2 for this transient.

2.2 Loss of Feedwater - The loss of feedwater accident has been determined to be the limiting design basis event in terms of minimum AFW flow required. The following is a discussion of the analysis performed to verify the adequacy of the existing AFW system.

2.2.1 Method of Analysis - The analysis was performed with the RETRAN1/MOD2 computer code. RETRAN was developed to describe the thermal-hydraulic behavior of light water reactors subjected to accidents and operational transients.

The CY system was modeled using 25 volumes, 36 junctions and 5 heat slabs. The nodalization diagram for this model is shown in Figure 1. Three loops are shown lumped together on the left side of this figure and the single remaining loop is shown on the right side. Each steam generator is modeled with two primary side volumes, three secondary side volumes (representing the U-tube region, the steam dome and the downcomer region) and two heat slabs. The nodalization of the lumped and single steam generators is shown in figures 2 and 3 respectively.

The reactor core is modeled as one volume and one heat slab. For all cases it is conservatively assumed that decay heat is 20% above the nominal values. The latent heat in the core is transferred into the primary coolant following scram. Any other metal heat absorbed by either the primary or secondary side is conservatively neglected. The kinetics data utilizes cycle 10 beginning of life values when the moderator temperature coefficient is the least negative. The most reactive rod is assumed to be stuck in the fully withdrawn position, thereby assuring the slowest rate of power decrease following scram.

RETRAN has a non-equilibrium pressurizer model which was used in this analysis. This model neglects any heat transfer between the liquid and vapor regions, thereby providing the maximum system pressure as the pressurizer refills due to primary side heatup.

It is assumed in this analysis that the reactor has been operating at 102% power (1861.5 MW) for an infinite time prior to initiation of the transient. The feedwater flow is assumed to be stopped (no flow coastdown) at time zero and the steam generator water level begins to drop from its normal water level (assumed at beginning of transient) to its low level setpoint (10% of narrow range indication). All S.G. levels are assumed to reach the setpoint simultaneously. At this point the reactor is scrammed on coincident low level and steam/feed mismatch. The steam generator blowdown lines are assumed to isolate. The secondary side pressure rises to the safety valve setpoint of 1000 psia and remains there throughout the rest of the transient. No credit is taken for the atmospheric dump or steam bypass which would reduce the steam generator pressure and substantially increase the AFW flow rate. All reactor coolant pumps are assumed to continue operating, thereby providing maximum pump heat to be transferred to the secondary side. The pressurizer heaters are assumed to remain operable, thereby increasing the primary side pressure as quickly as possible.

The AFW system at CY is shown in Figure 4. It is modeled as a fill junction into the downcomer region of each steam generator. It is assumed that only one of the two pumps is operating, as this is the limiting single failure. The system begins to pump 350 gpm of water (conservatively low) after a conserva' ely long 30 second startup time following the initiation signal. The temperature of this flow is assumed to be 120^{o}F. This is the high temperature alarm setpoint of the demineralized water storage tank (DWST) from which the AFW pumps take suction. The volume between the AFW inlet and the stear generator for the longest of the four feedlines is 1426 gallons. It was conservatively assumed that water at the normal feedwater enthalpy of 432 BTU/LB will enter the steam generator following AFW initiation until the colder AFW can sweep out the entire feedline volume from the injection point. This assumes that no heat is transferred from the feedlines to the ambient air following scram. After the feedlines are swept out, the AFW enthalpy changes to the 88 BTU/LB value corresponding to the 120^{o}F temperature.

2.2.2 <u>Case 1 - no AFW</u> - This case was run to determine the time the PORV would lift if no AFW was delivered to the steam generator. The resulting lift time is then the earliest possible time for this event to occur. The sequence of events is given in Table 1. The plots of pressurizer pressure, steam generator liquid mass and steam generator mass accumulation rate are shown in figures 5, 6 and 7. The results show a PORV lift time of 989.8 seconds. When this case was rerun without the 20% decay heat penalty, the results showed a lift time of 1185.8 seconds. If the reactor coolant pumps are assumed to trip at time 0, then PORV lift occurs at 1270.7 seconds.

2.2.3 Case 2 - AFW Initiated at 10 Minutes - This case represents manual initiation of AFW. The feedwater flow stops at time zero, the reactor scrams shortly thereafter and the secondary side inventory is depleted for 10 minutes. At this point, the operator is assumed to take manual action to start AFW flow. After the 30 second startup time, AFW flow begins to sweep out the feedlines. During this 978 second process, water is entering the steam generator at 432 BTU/LB. The enthalpy of the water into the steam generator is reduced to 88 BTU/LB once this sweeping process is completed (1608 seconds into the transient). The AFW system is now capable of removing primary heat generated and the primary side pressure begins to turn around.

The pressure peaks at 1614 seconds and 2254 psia, below the 2285 psia PORV setpoint. After this, the AFW flow turns the transient around. A sequence of events is given in table 2. Plots of important parameters are shown in figures 8, 9 and 10.

2.2.4 Case 3 - AFW Automatically Initiated - This case represents the automatic initiation of AFW which will be implemented during the 1980 refueling outage. The initiation setpoint is to be set at 45% of steam generator wide range level indication on two of the four steam generators. We have assumed a 10% of full scale error in the worst direction for all four steam generators. The water inventory was conservatively calculated to be 76,626 lb_m total in all four steam generators if they reach a 35% level simultaneously. This signal was used to initiate the AFW system.

The sequence of events is shown in table 3. The relevant plots are given in figures 11, 12 and 13. The results show that the secondary side is capable of removing decay heat at all times during the transient. The small pressure rise shown is a result of the pressurizer heaters trying to recover normal operating pressure.

2.3 Feedline Rupture: This accident is not in the design basis of the plant. The results are included for informational purposes.

2.3.1 Case 1 - Break Upstream of Feedwater Check Valves: This event appears as a loss of feedwater accident to the reactor system. Run-out flow to the break prevents any feedwater (either main or auxiliary) from entering the affected steam generators until the break is isolated. Check valves prevent blowdown of the steam generators. All the feedlines downstream of the check valves blow down until the feedwater regulating valves close after scram, after which the three intact lines are isolated from the break.

The model and assumptions used for this case are identical to those described in section 2.2, with the exception that there is an extra delay time before AFW delivery associated with refill of the blown down feedlines. Prior to reg. valve closure, all main feedwater in the lines upstream of the check valves will flow back through the reg. valves and the common header to the break. Thus, a length of 135 feet of pipe between the check valve and the reg. valve are empty and must be refilled before any AFW can enter the steam generator. The feedwater regulating valves are conservatively assumed to close after the liquid blowdown is completed and before there is a substantial pressure reduction. It is conservatively assumed that the AFW system is isolated at 10 minutes and, after a 30 second delay time, delivers 340 gpm of water against a 1000 psia pressure into the three intact feedlines. No feedwater enters the steam generators until the three 135 ft. sections of pipe (half the total length from reg. valve and steam generator) between the feedwater regulating valves and the check valves is refilled. This process requires 378 seconds (1008 seconds into transient). Following this, the colder AFW sweeps the hot main feedwater from the remaining section of feedline, which results in 432 BTU/LB water entering the steam generators. This sweeping process takes another 378 seconds, after which the 88 BTU/LB water enters the steam generators (1386 seconds after the transient begins). This last event is what turns the transient around.

The sequence of events is shown in table 4. Figures 14, 15 and 16 are the plots for this case. The results show that the pressurizer PORV lift occurs at 992 seconds and cycles for 404 seconds. The maximum pressurizer level during the accident is 61.8%.

2.3.2 Case 2 - Break Downstream of Feedwater Check Valves - This event would blow down the affected steam generator as well as the feedline system up to the check valves on the three intact lines. This type of break, however, is a steam break for feedring-type plants such as CY. Hence, a cooldown transient would result. This transient is bounded by the steam line break accident since the feedwater nozzle is smaller than the steam nozzle. Since this is a cooldown transient, the AFW system would not be required quickly and no analysis need be performed.

3.0 Capability of AFW System

This section is included to describe the capabilities of the AFW system at CY.

3.1 AFW Minimum Flow Rates - Table 5 gives the minimum flow rates that can be expected to reach the intact steam generators for various combinations of flow isolation and pressure. These values are determined assuming pump bearing cooling and seal leakage flow rates of 13 gpm at 1100 psig and a recirculation flow rate of 100 gpm at 1250 psig for each pump. Pump wear has been assumed to be negligible since these pumps are used only during plant startup and shutdown.

It has been determined that 318.2 gpm of AFW flow will be needed to remove decay heat and pump heat 900 seconds after shutdown (time less than PORV lift time with no AFW). This value assumes ANS+20 decay heat. As can be seen from table 5, two steam generators must be available to be able to receive this much flow. Note that if more realistic decay heat values were to be used and both pumps were assumed to start, then only one steam generator is required. This is consistent with the experience at the plant.

3.2 AFW Source Inventory - The primary source of water is the DWST which has a minimum capacity of 50,000 gallons by technical specifications. This is supplemented by the primary water storage tank (PWST), which has a technical specification minimum of 80,000 gallons. In addition to these sources, the recycle water storage tank (100,000 gallons) is normally available. The long term water sources are the well water pumps, which can provide 250 gpm each. These sources provide adequate water to feed the AFW system indefinitely.

Maximum conditions required to switch to the RHR system is 340 psig and 375^{o}F. A maximum of 68,100 gallons of AFW are required to cool down the reactor to 300^{o}F in the four hour period starting from the beginning of any transient. Hence, we do not need any more water than the combined volumes of the DWST and the PWST.

4.0 Conclusions

Based on the analyses presented here, we conclude that the auxiliary feedwater system for the Connecticut Yankee Nuclear Power Plant is adequately sized to meet the acceptance criteria defined in section 1.4 for the events in its design basis (section 1.1 and 1.2). The water inventory for this system is large enough to remove primary side heat until well past the point where the RHR system operation starts.

Reference

1) Letter from W. G. Counsil to D. L. Ziemann, docket No. 50-213, dated January 30, 1980.

Table 1

Loss of Feedwater, Case 1 (no aux feed)

Sequence of Events

Time (seconds)	Event
0	Feedwater lost
7.2	Reactor Scram
989.8	PORV Lift

Table 2

Loss of Feedwater, Case 2 (10 minute AFW initiation)

Sequence of Events

Time (seconds)	Event
0	Feedwater Lost
7.2	Reactor Scram
600	AFW Initiated
630	AFW Flow Begins
1608	AFW Sweeps Out Feedlines
1614	Peak Pressure = 2254 psia

Table 3

Loss of Feedwater, Case 3 (auto AFW initiation)

Sequence of Events

Time (seconds)	Event
0	Feedwater Lost
7.2	Reactor Scram
180	AFW Initiated
210	AFW Flow Begins
1188	AFW Sweeps Out Feedlines

Table 4

Feedline Rupture, Case 1

Sequence of Events

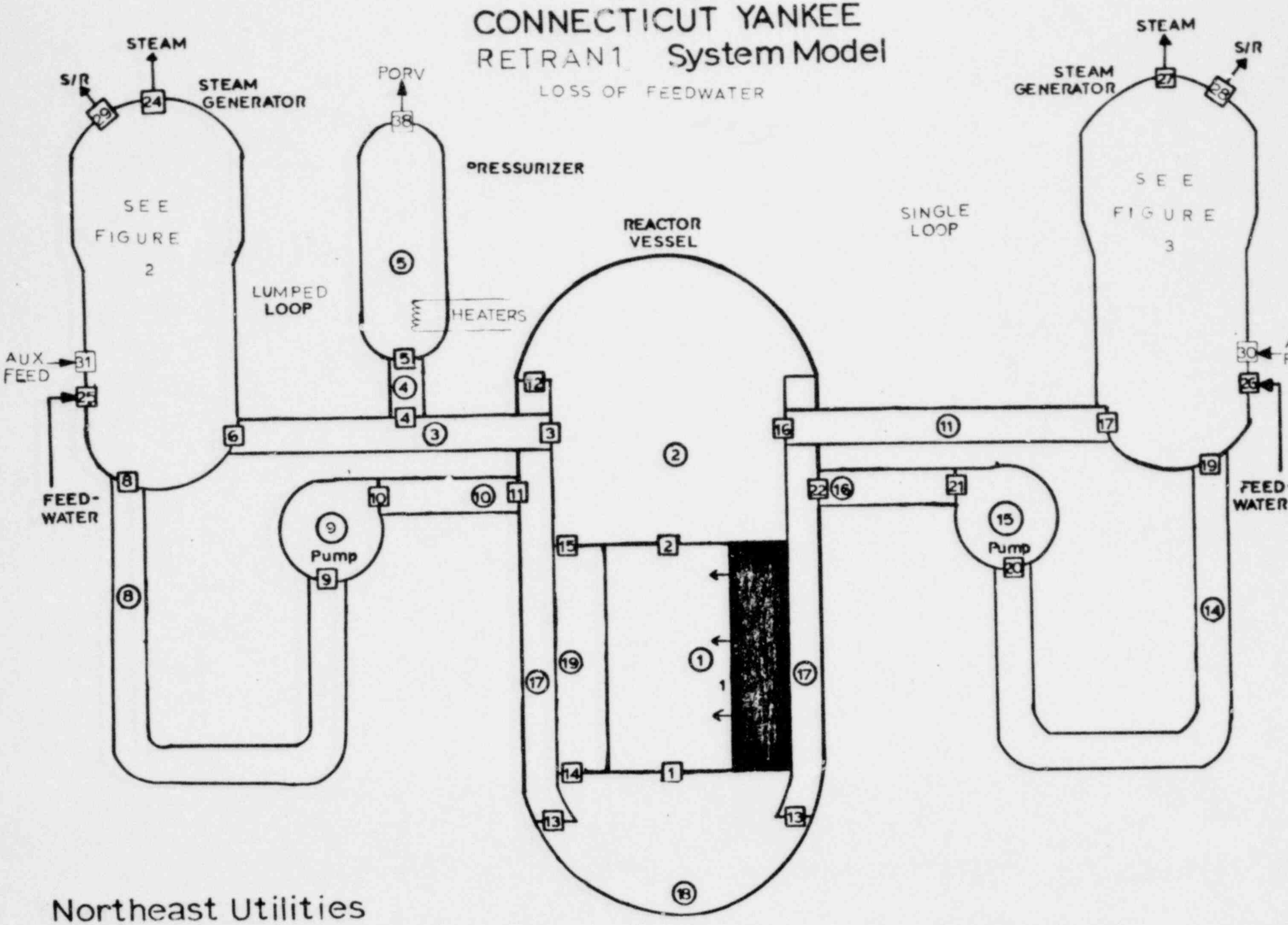
Time (seconds)	Event
0	Feedwater Lost
7.2	Reactor Scram
600	AFW Isolated from Break
630	AFW Flow into Feedlines Begins
992	PORV Lifts
1008	AFW Refills Feedlines
1386	AFW Sweeps out Feedlines
1396	Pressurizer Pressure Below PORV Setpoint

TABLE 5

Minimum Expected AFW Flow Rates

Condition	total flow to all intact steam generators (GPM)	
	1 pump	2 pumps
one ruptured SG and 3 intact SG's at 1000 psig	0	0
4 intact SG's at 1000 psig	350	545
3 intact SG's at 1000 psig with the other 1 isolated	340	515
2 intact SG's at 1000 psig with the other 2 isolated	320	450
1 intact SG at 1000 psig with the other 3 isolated	245	300

FIGURE 1
CONNECTICUT YANKEE
RETRAN1 System Model
LOSS OF FEEDWATER



Northeast Utilities

FIGURE 2

Steam Generator Nodalization

(3 Volume Secondary Side)

Lumped Steam Generator

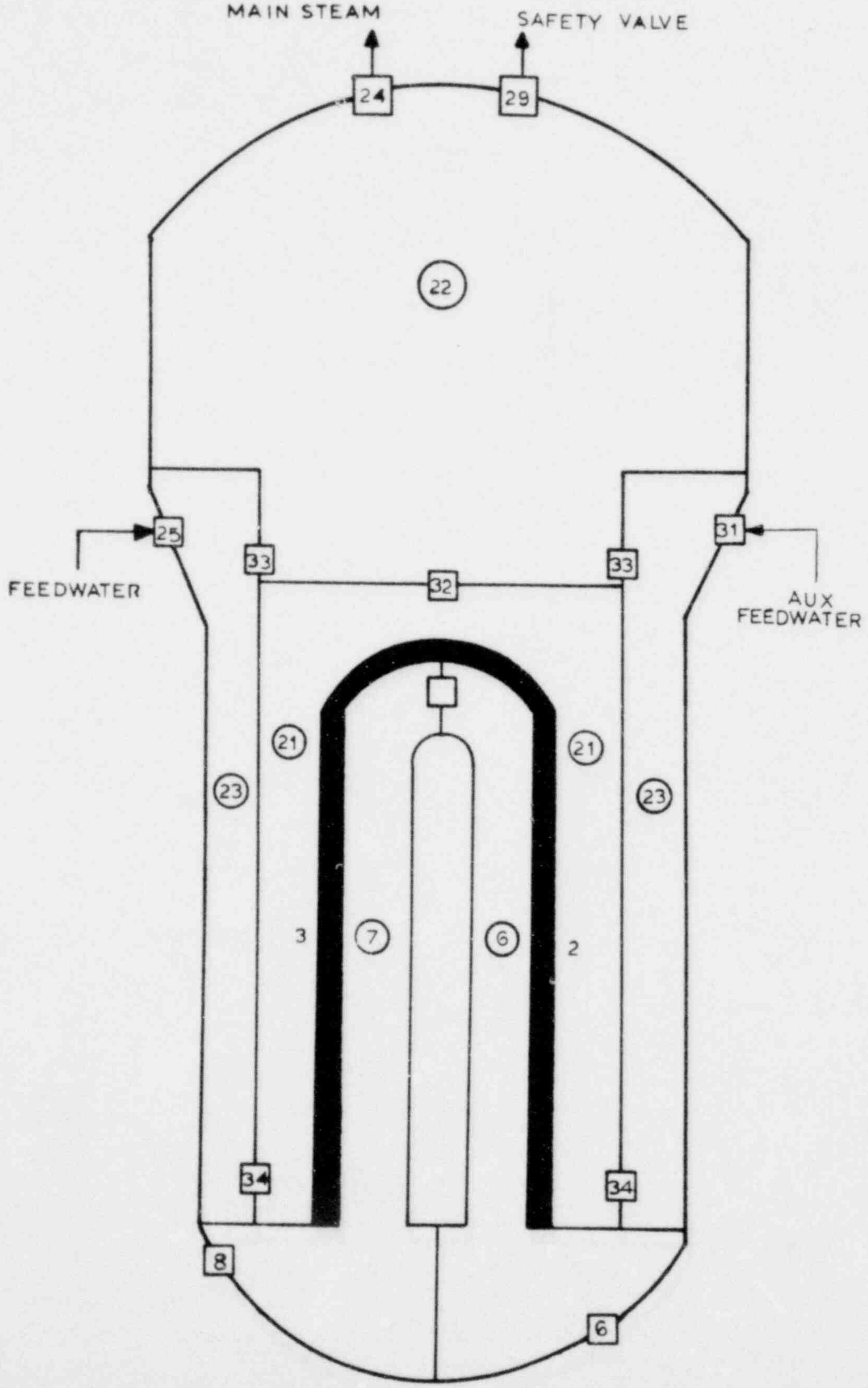
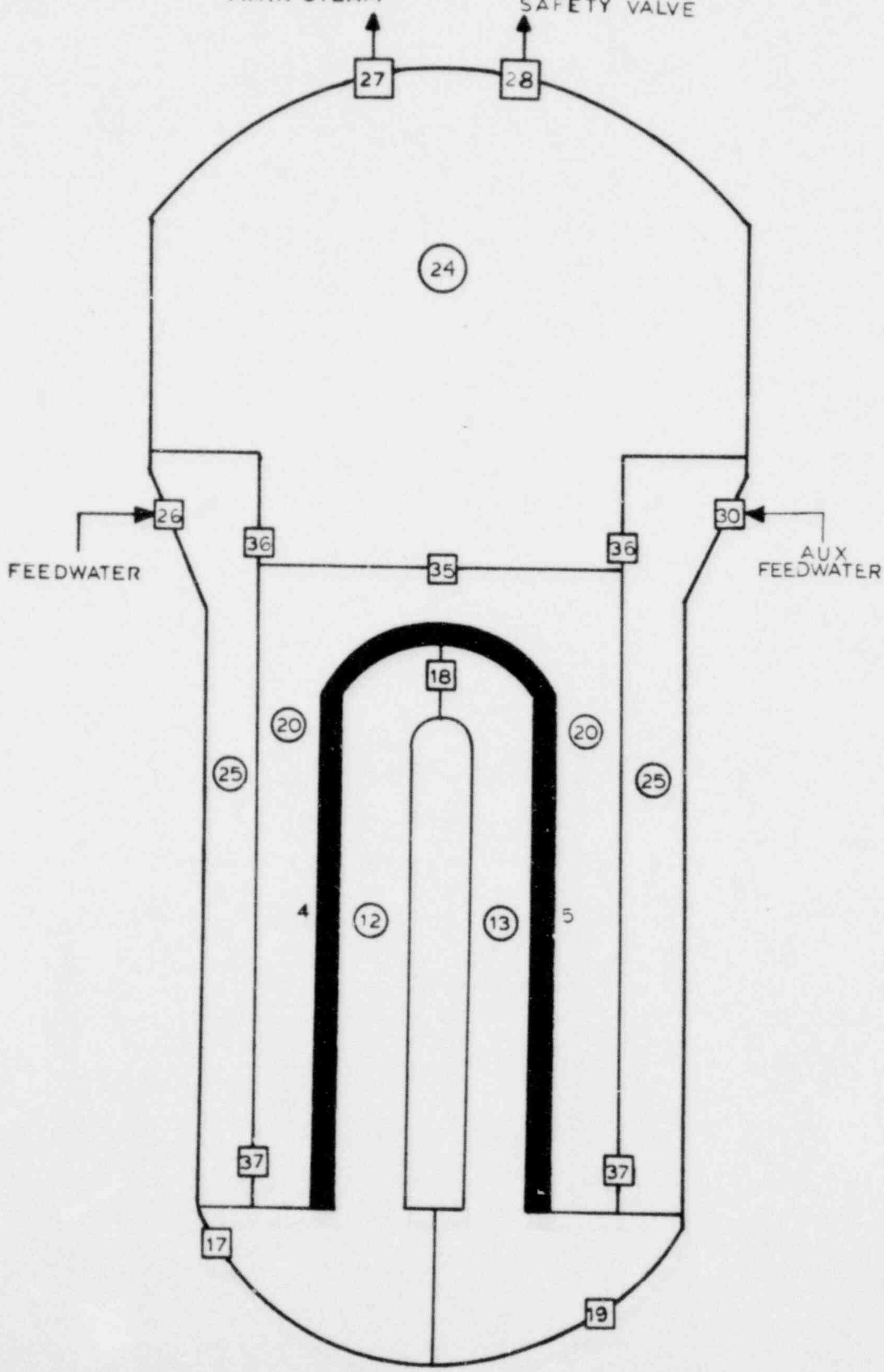


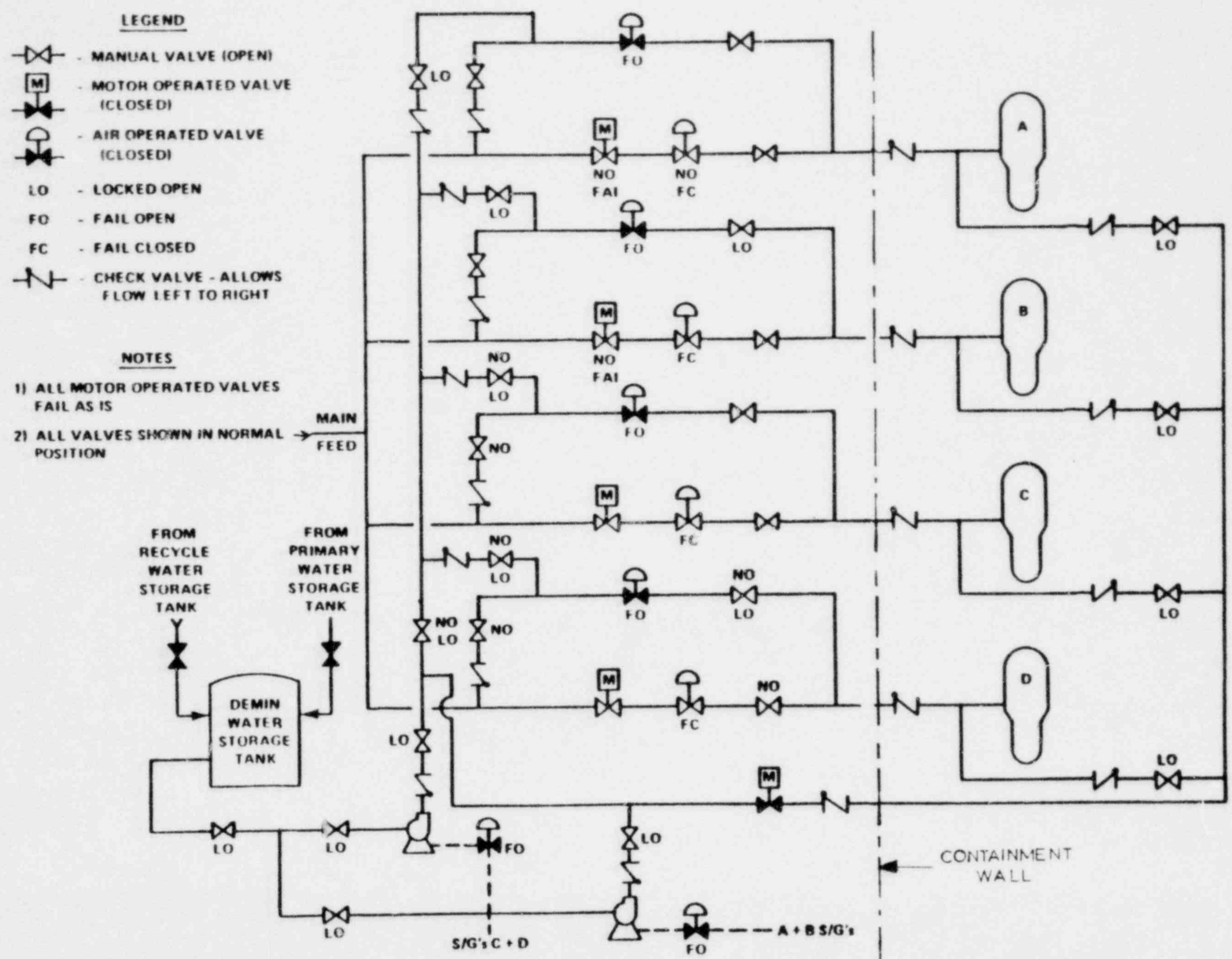
FIGURE 3

Steam Generator Nodalization

(3 Volume Secondary Side)

Single Steam Generator





Auxiliary Feedwater System
Haddam Neck
Figure 4

FIGURE 5

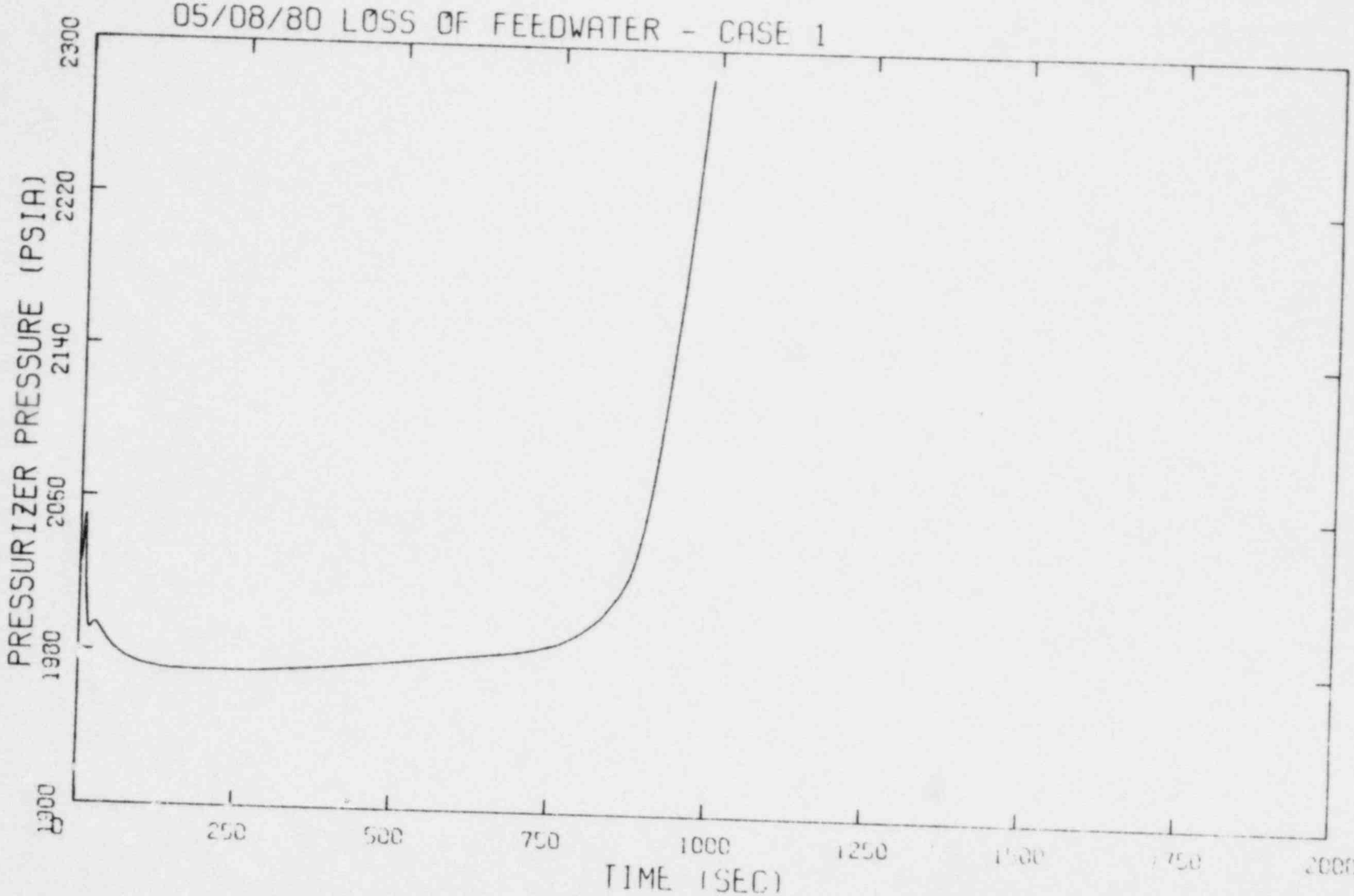
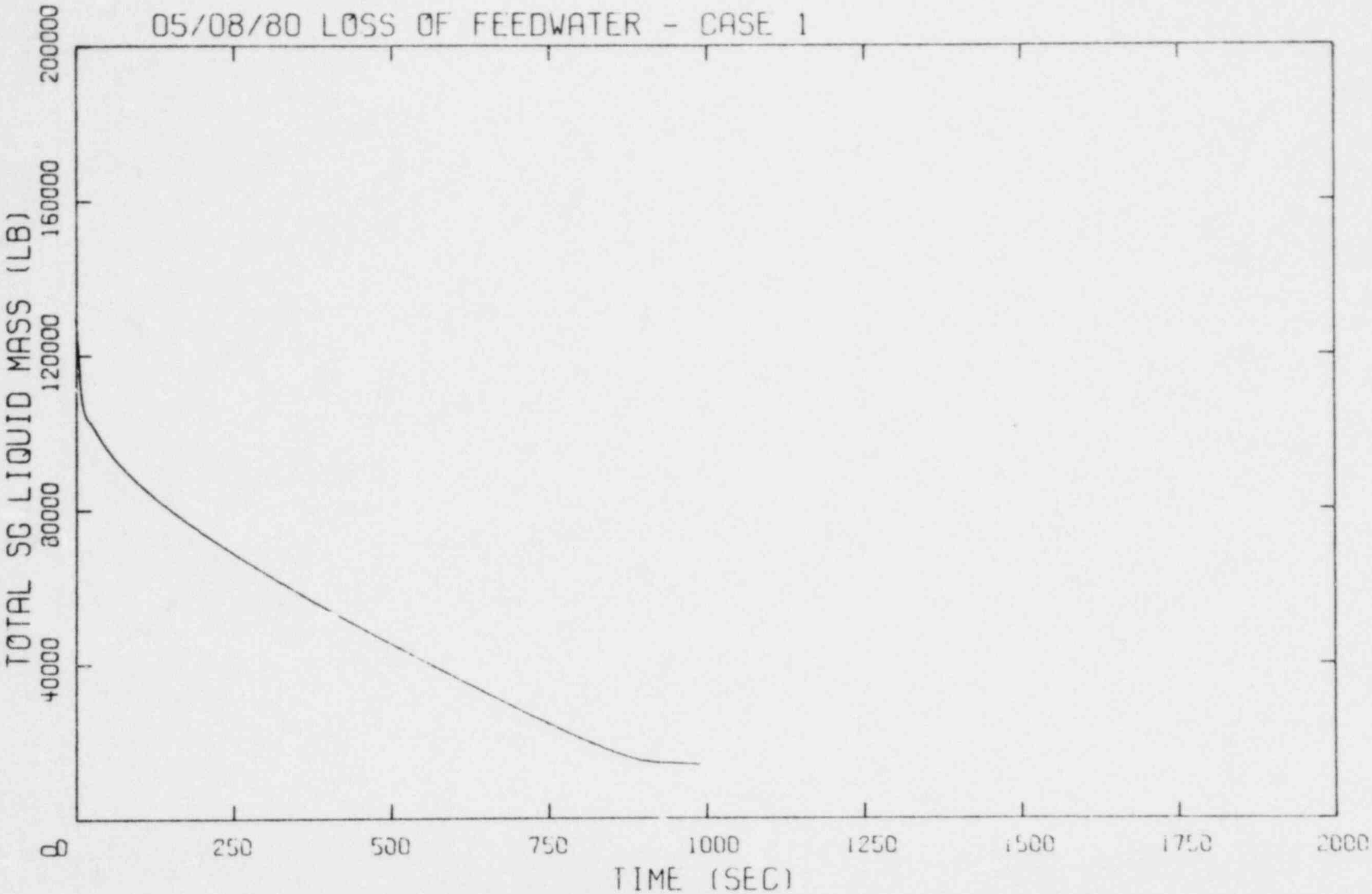

05/08/80 LOSS OF FEEDWATER - CASE 1
PRESSURIZER PRESSURE (PSIA)
2300
2220
2140
2050
1980
1900
0
250
500
750
1000
1250
1500
1750
2000
TIME (SEC)

FIGURE 6



05/08/80 LOSS OF FEEDWATER - CASE 1
TOTAL SG LIQUID MASS (LB)
200000
160000
120000
80000
40000
0
0
250
500
750
1000
1250
1500
1750
2000
TIME (SEC)

FIGURE 7

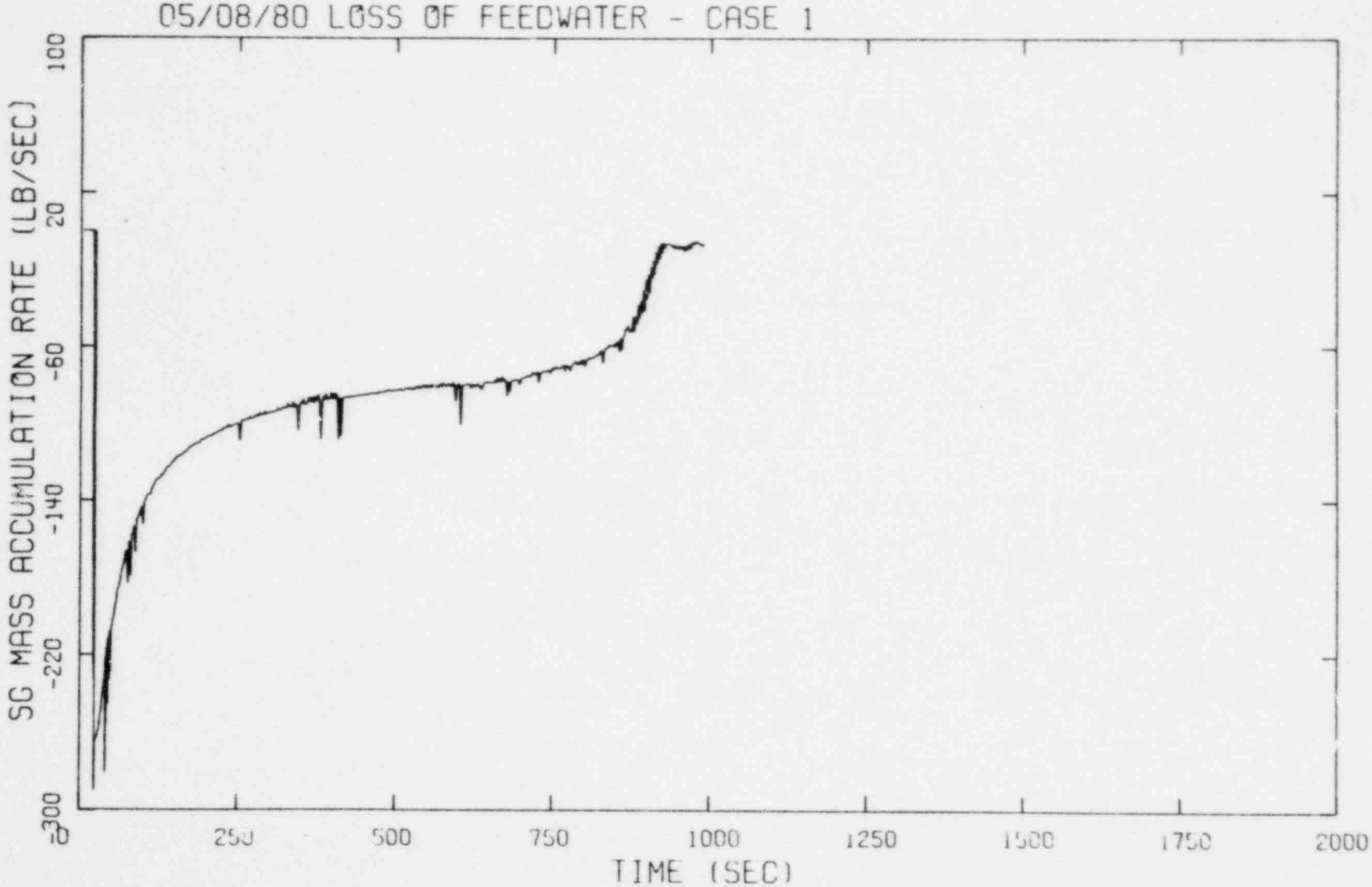

05/08/80 LOSS OF FEEDWATER - CASE 1
SG MASS ACCUMULATION RATE (LB/SEC)
100
20
-60
-140
-220
-300
0
250
500
750
1000
1250
1500
1750
2000
TIME (SEC)

FIGURE 8

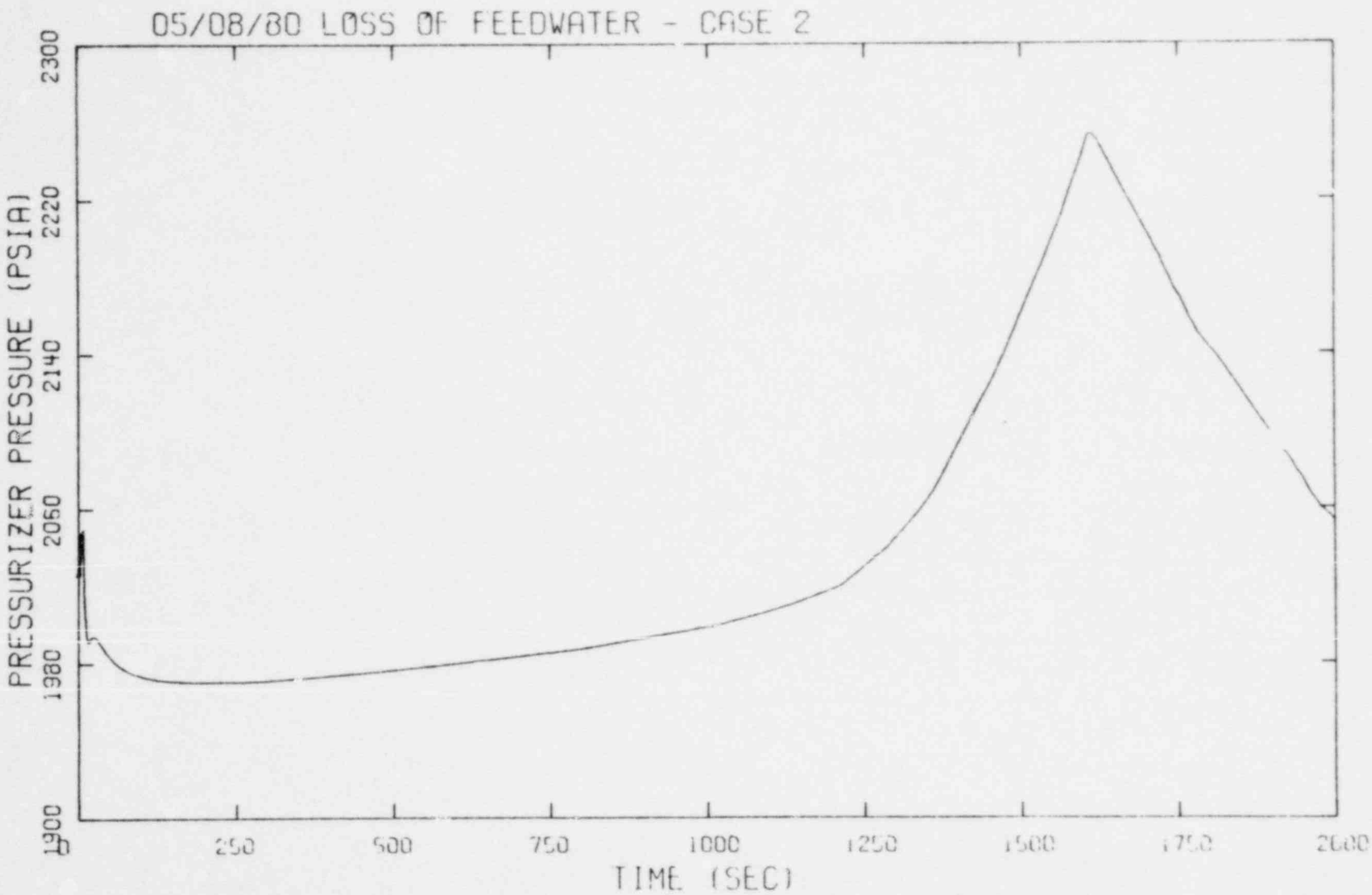

05/08/80 LOSS OF FEEDWATER - CASE 2
PRESSURIZER PRESSURE (PSIA)
2300
2220
2140
2060
1980
1900
0
250
500
750
1000
1250
1500
1750
2000
TIME (SEC)

FIGURE 9

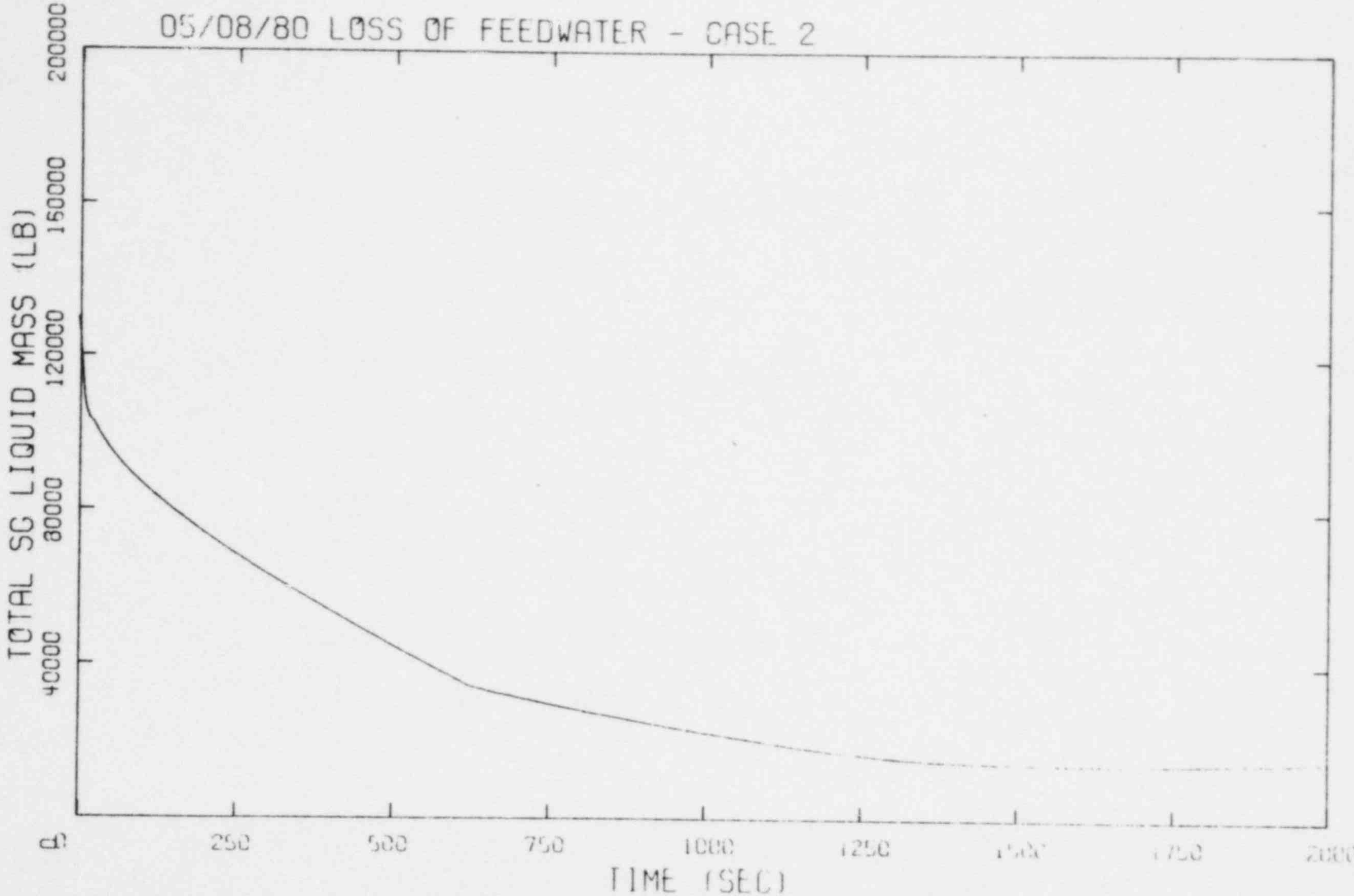

05/08/80 LOSS OF FEEDWATER - CASE 2
TOTAL SG LIQUID MASS (LB)
200000
160000
120000
80000
40000
0
250
500
750
1000
1250
2000
TIME (SEC)

FIGURE 10

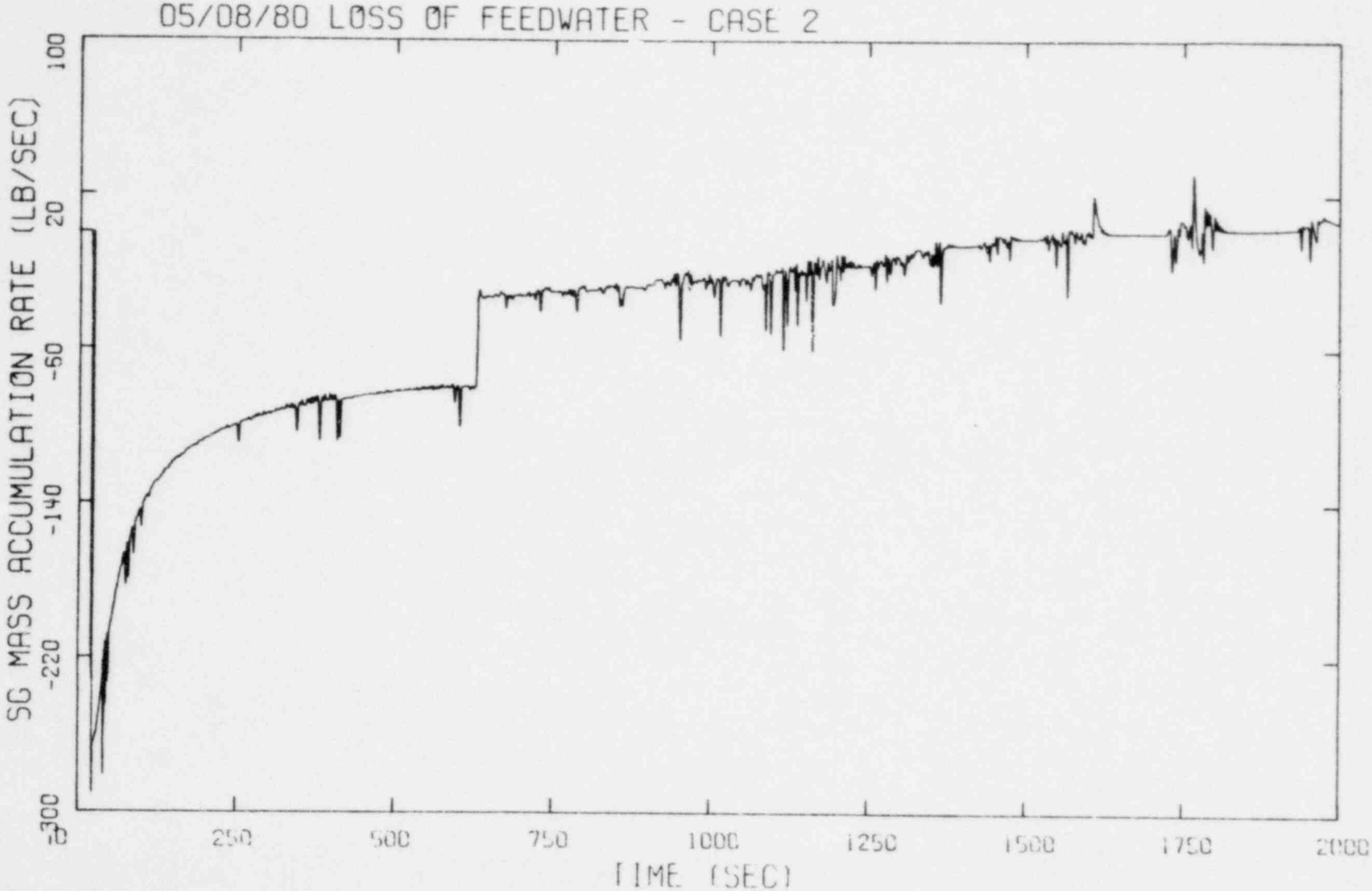
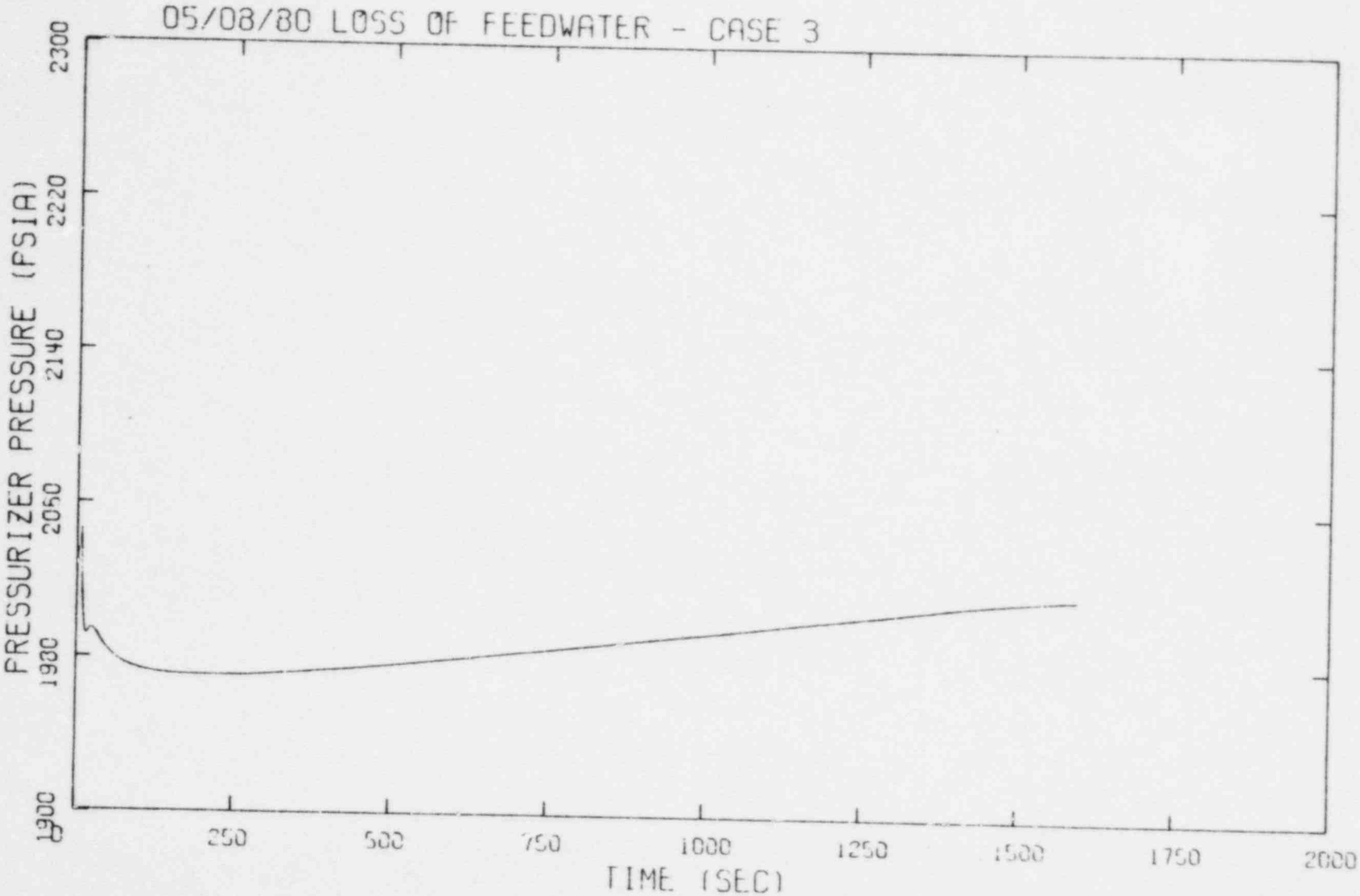
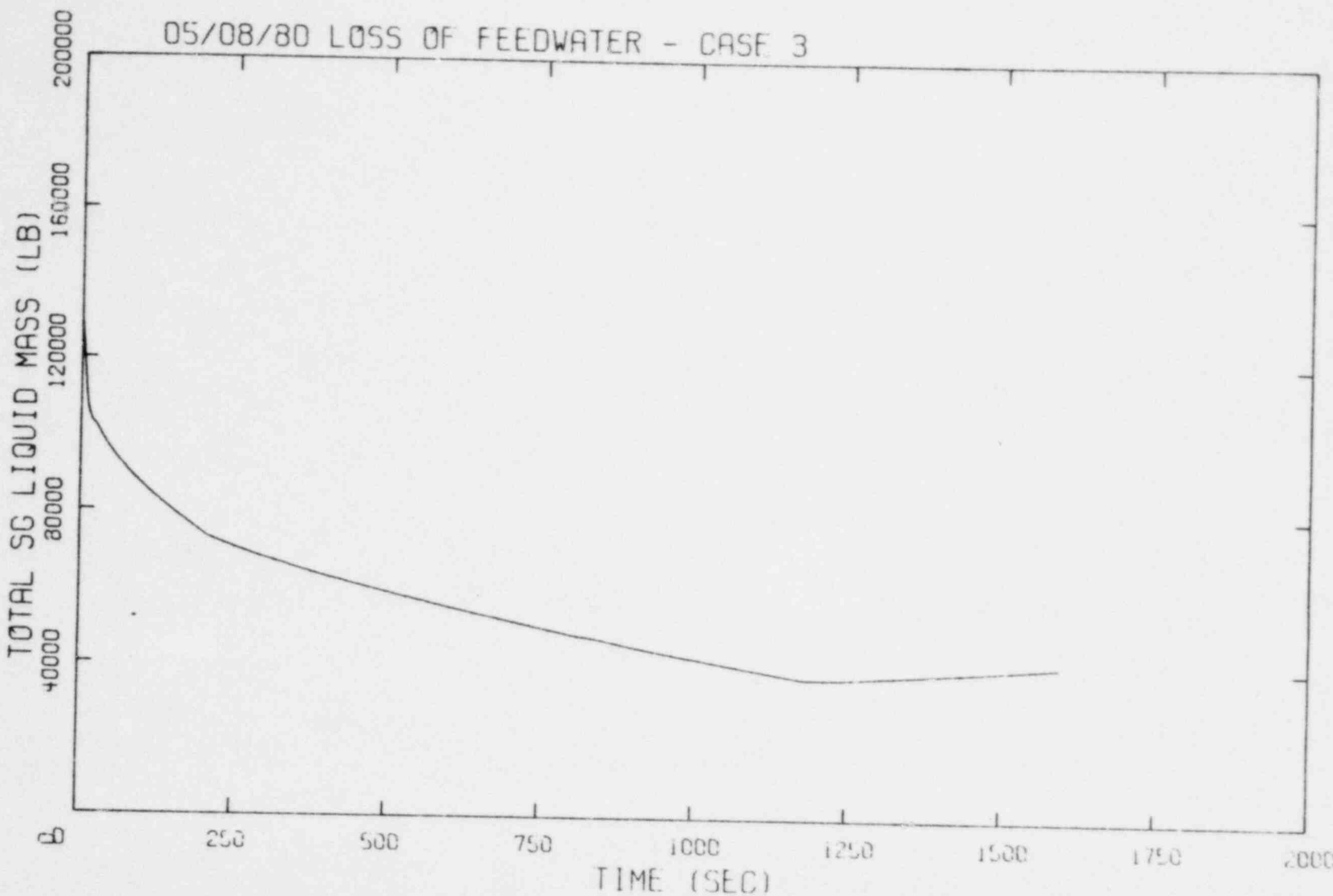

05/08/80 LOSS OF FEEDWATER - CASE 2
SG MASS ACCUMULATION RATE (LB/SEC)
100
20
-60
-140
-220
-300
0
250
500
750
1000
1250
1500
1750
2000
TIME (SEC)

FIGURE 11



05/08/80 LOSS OF FEEDWATER - CASE 3
PRESSURIZER PRESSURE (PSIA)
2300
2220
2140
2060
1980
1900
0
250
500
750
1000
1250
1500
1750
2000
TIME (SEC)

FIGURE 12



05/08/80 LOSS OF FEEDWATER - CASE 3
TOTAL SG LIQUID MASS (LB)
200000
160000
120000
80000
40000
0
250
500
750
1000
1250
1500
1750
2000
TIME (SEC)

FIGURE 13

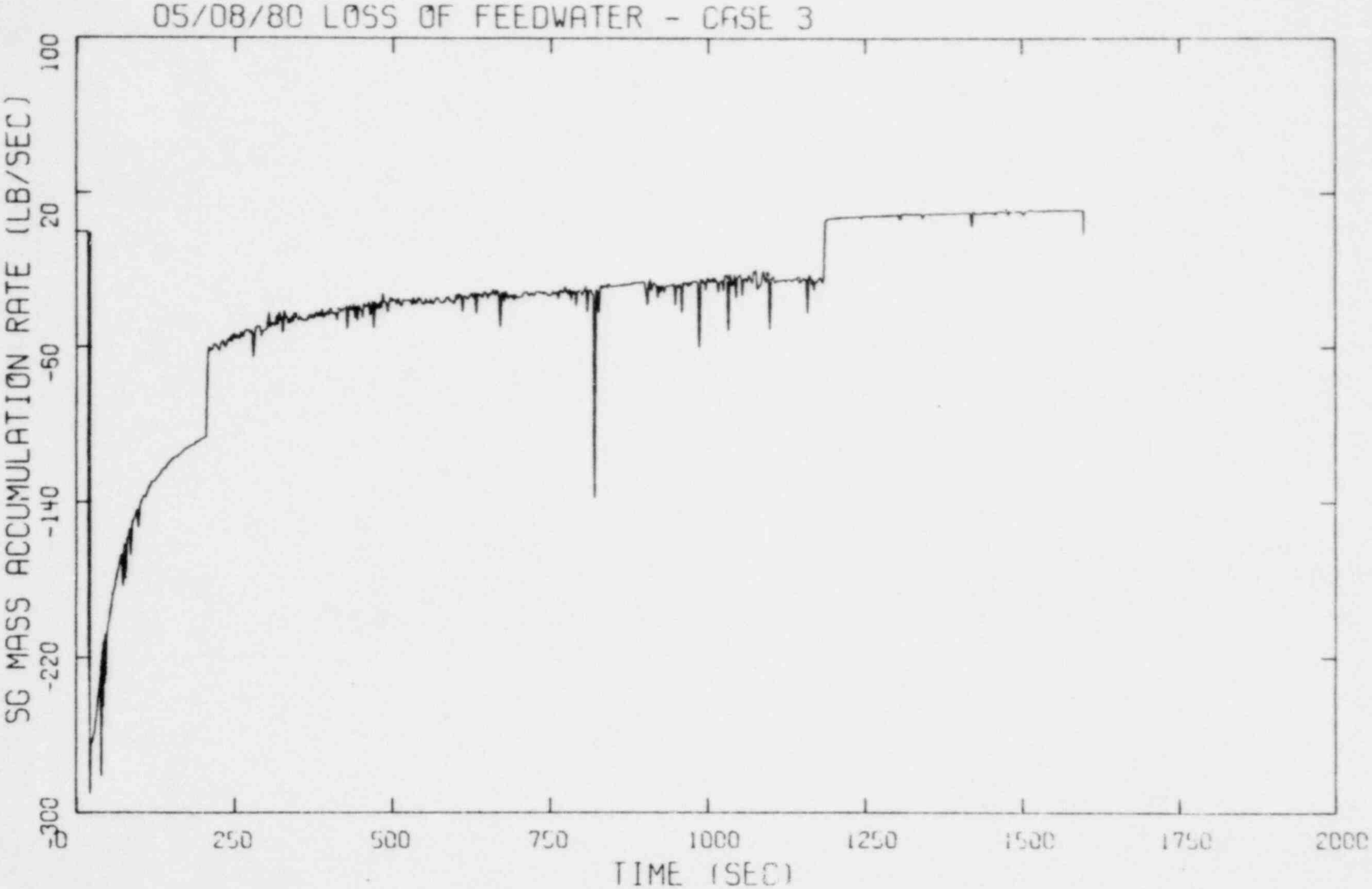
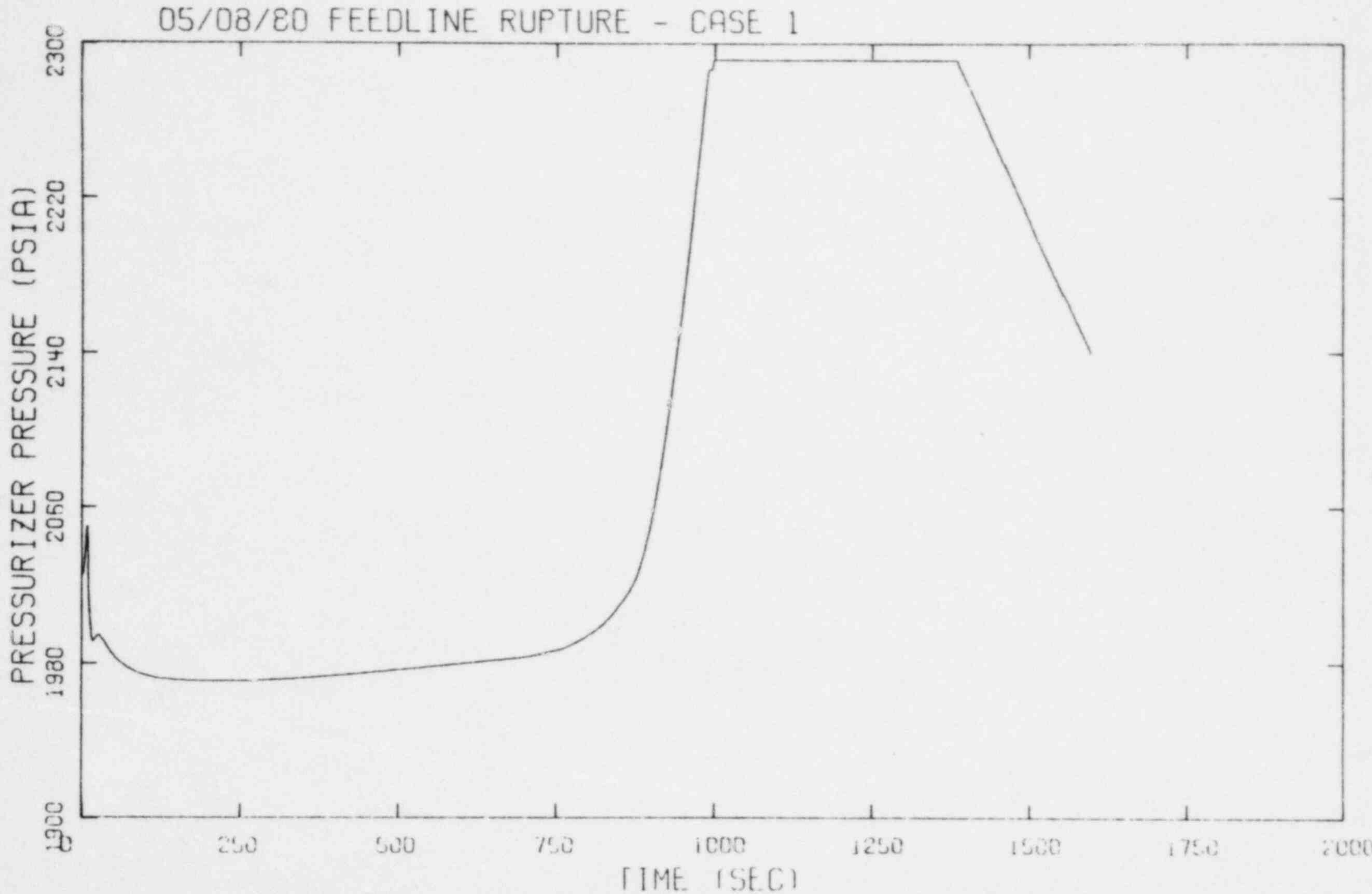

05/08/80 LOSS OF FEEDWATER - CASE 3
SG MASS ACCUMULATION RATE (LB/SEC)
100
20
-60
-140
-220
-300
0
250
500
750
1000
1250
1500
1750
2000
TIME (SEC)

FIGURE 14



05/08/80 FEEDLINE RUPTURE - CASE 1
PRESSURIZER PRESSURE (PSIA)
2300
2220
2140
2060
1980
1900
0
250
500
750
1000
1250
1500
1750
2000
TIME (SEC)

FIGURE 15

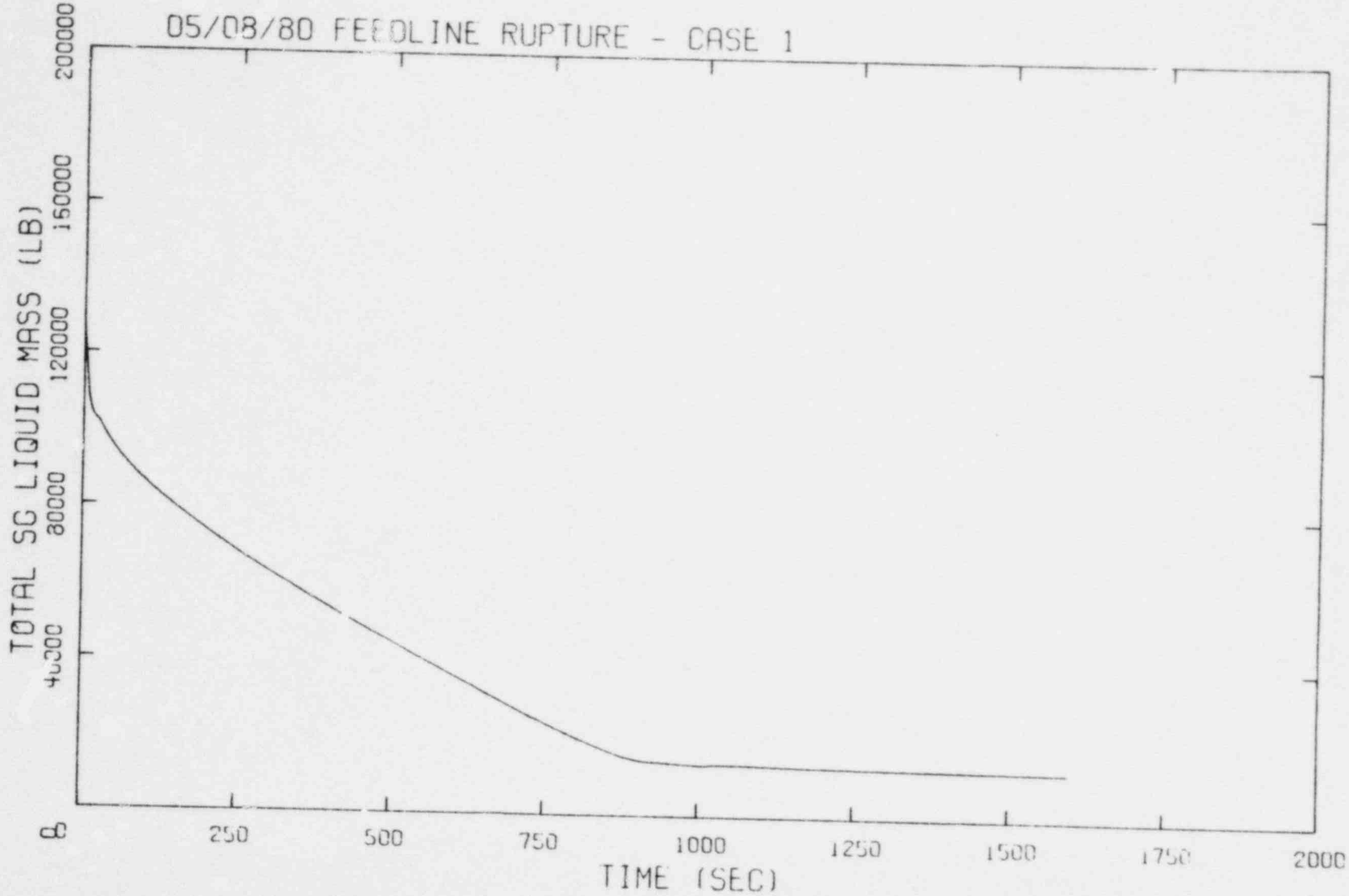
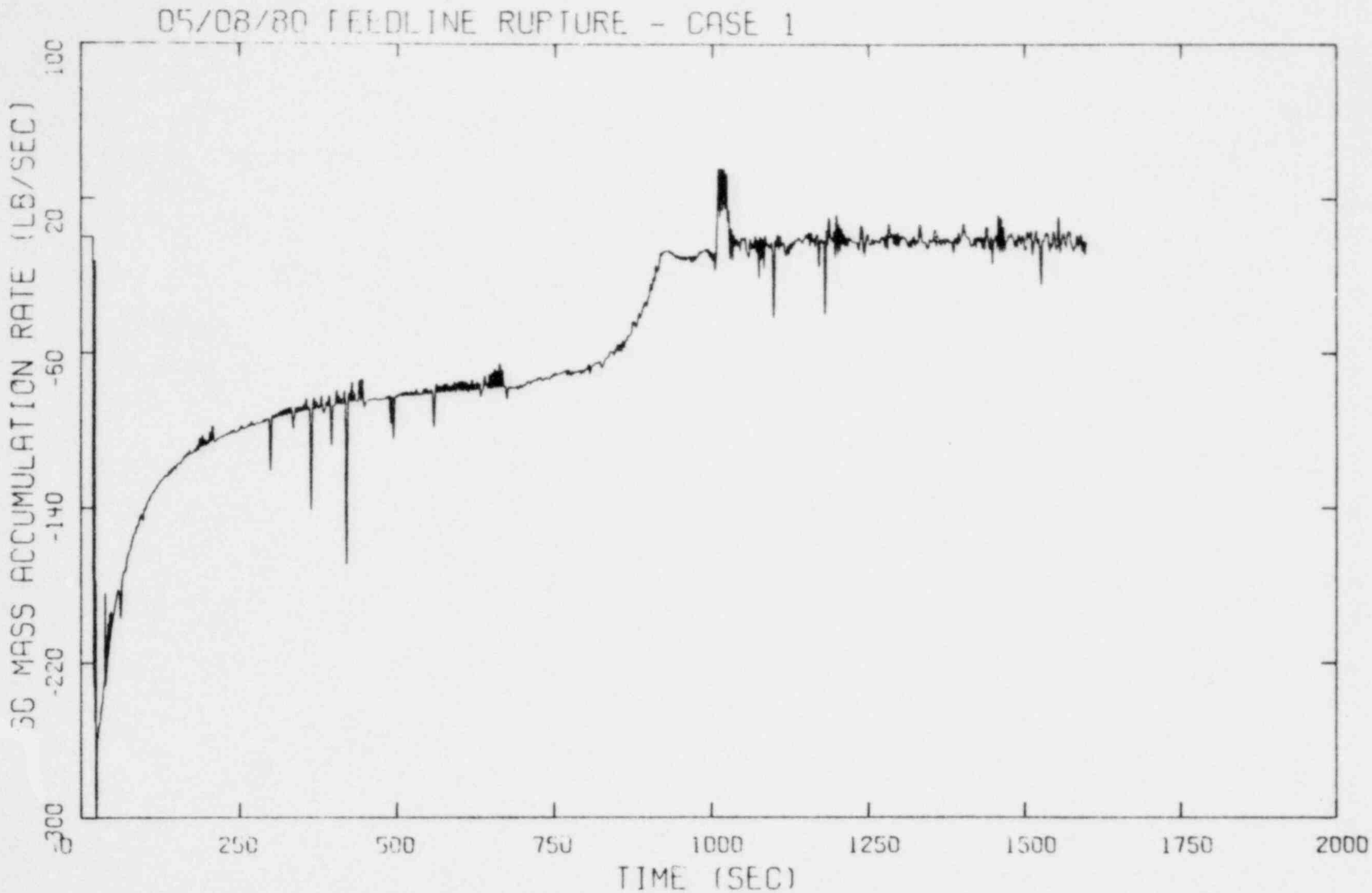

05/08/80 FEEDLINE RUPTURE - CASE 1
TOTAL SG LIQUID MASS (LB)
200000
160000
120000
80000
40000
0
250
500
750
1000
1250
1500
1750
2000
TIME (SEC)

FIGURE 16


05/08/80 FEEDLINE RUPTURE - CASE 1
SG MASS ACCUMULATION RATE (LB/SEC)
100
20
-60
-140
-220
-300
0
250
500
750
1000
1250
1500
1750
2000
TIME (SEC)