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RESIDUAL HEAT REMOVAL EVALUATIONS FOR HPCI FAILURE OR RHR SHUTDOWN COOLING FAILURE FOR THE MONTICELLO NUCLEAR GENERATING PLANT

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1. INTRODUCTION AND SUMMARY

This report presents the results of evaluations of the following transient events:

(1) Loss-of-Offsite Power and HPCI Failure

A loss-of-offsite power during normal operation, followed by reactor scram, and the assumed failure of the high pressure coolant injection (HPCI) system is assumed in this event. No credit is taken for the reactor core isolation cooling (RCIC) system. All other systems are assumed to operate normally.

The results of analyses are presented which show that the manual actuation of 3 automatic depressurization system (ADS) valves at 10 minutes after scram will depressurize the reactor vessel such that the low pressure coolant injection (LPCI) and low pressure core spray (LPCS) systems will be available to provide core cooling and maintain the peak cladding temperature (PCT) below 2200°F. The normal operation of the shutdown cooling (SDC) mode of the residual heat removal (RHR) system can then be used to cooldown the reactor and bring the reactor to a cold shutdown condition.

(2) Loss of Shutdown Cooling Mode of RHR During Normal Shutdown

A loss of the normal shutdown cooling (SDC) mode of the residual heat removal (RHR) system during a normal reactor shutdown is assumed in this event. The SDC mode is assumed to be unavailable at 24 hours after a normal reactor scram, when it would normally have been placed into operation as part of the normal reactor cooldown process.

The results of an evaluation are presented which show that the alternate shutdown cooling (ASDC) system, which employs the LPCS and Safety/Relief (S/R) valves to remove decay heat from the reactor, can be used to cooldown the reactor and bring the reactor to a cold shutdown condition.

2. TRANSIENT ANALYSIS

The analysis of the two transient events described in Section 1, and a description of the operation of the alternate shutdown cooling (ASDC) system, is presented in this section.

2.1 Loss-of-Offsite Power and HPCI Failure

2.1.1 Event Description

The reactor is assumed to be operating at 102% of rated power when a loss-of-offsite power occurs. A reactor scram is initiated at time zero followed by recirculation pump trip, closure of the main steam isolation valves (MSIV) in 2 seconds, and loss of all feedwater flow in 5 seconds. A failure of the high pressure coolant injection (HPCI) system is assumed. No credit is taken for the reactor core isolation cooling (RCIC) system. All other reactor systems are assumed to function normally. The automatic depressurization system (ADS) must be initiated manually to depressurize the reactor and allow LPCI and LPCS systems to be available for core cooling.

After manual ADS actuation, the LPCI and LPCS will restore the core coolant inventory and cool down the reactor water temperature to about 281°F. At this time the shutdown cooling (SDC) mode of the residual heat removal (RHR) system is placed in operation to cooldown the reactor and bring the reactor to a cold shutdown condition.

2.1.2 Analysis Results

Sensitivity studies were performed assuming manual ADS actuation at 10, 5 and 3 minutes after reactor scram to determine the uncovered core time as a function of ADS delay time. Also, at the time of reactor scram (time zero), 3, 2 and 1 ADS valve(s) were assumed to be actuated to determine the vessel depressurization rate, vessel water temperature cooldown rate, and the time for vessel pressure to reach 300 psig.

The initial plant conditions and input parameters are given in Table 1. The reactor pressure and the water level inside the shroud during the ADS depressurization cases is shown in Figures 1a through 4b. The key results are summarized in Table 2. A constant depressurization flow area throughout the transient was assumed in each ADS depressurization case. LPCS and LPCI flow was initiated to restore level when the pressure permissive was reached. The level outside of the shroud is flooded to the top of the vessel in about 10 to 15 minutes after the initiation of depressurization in each case. The water level given in Figures 1b, 2b, 3b and 4b is the level inside the shroud, which reaches a constant high level above the core after the internal shroud has been flooded. In an actual event, the operator would be expected to manually depressurize the reactor, and control both reactor pressure and level after the LPCS and LPCI pumps have restored the normal water level.

The transient results presented in Table 2 show that manually actuating 3 ADS valves at 10 minutes after scram will enable the LPCI and LPCS systems to be available to maintain a coolable core geometry. For this case, it is predicted that 1.1 feet of active fuel is uncovered for a short period of time. The peak clad temperature (PCT) is estimated to be less than 1200°F and a coolable core geometry is maintained.

If ADS valves are manually actuated at time zero, actuation of 2 or 3 ADS valves will enable LPCI and LPCS systems to be available early enough to provide adequate core cooling and a peak clad temperature of less than 1200°F. The estimates of peak clad temperature are based upon calculated core uncovery times. These core uncovery times have been compared with previous generic core heatup calculations to estimate the peak clad temperature.

2.1.3 Vessel Thermal Cycle Effect

During a loss-of-offsite power and HPCI failure event, systems for normal water level control (RCIC) would be available for which no credit was taken in the analysis. Therefore, this event could be classified as an emergency event. However, the following evaluations were performed to determine the effect of this event on the vessel thermal cycle design specification in the existing stress report.

The reactor vessel design specification (Reference 2) considered one rapid blowdown at a fluid temperature rate of 1000°F/hr. from 546°F to 370°F followed by a rate of 100°F/hr. from 370°F to 100°F. The reactor vessel stress analysis (Reference 3) considered all shutdowns as blowdowns for conservatism.

For the 3 ADS blowdown event, the temperature rate from 530°F to 400°F was found to be approximately 1175°F/hr. (see Figure 6). For the 2 ADS blowdown event, the temperature rate from 530°F to 400°F was found to be approximately

810^{o}F/hr. (see Figure 7). These SRV blowdowns to 400^{o}F are within the existing stress report analysis for normal and upset events even though these blowdowns can be considered emergency events which do not require such analysis for secondary thermal stresses.

Since these events do not involve any increase of primary stresses beyond the design basis, the subject rapid depressurization transients (Figures 1a, 2a, 3a, 4a, 6 and 7) are within the ASME acceptance stress limits for emergency conditions as previously analyzed for the Monticello pressure vessel.

2.2 Loss of Shutdown Cooling Mode of RHR During Normal Shutdown

2.2.1 Event Description

Following a normal reactor scram, the main condenser is used as a heat sink to remove heat and reduce the reactor water temperature to about 281^{o}F. The shutdown cooling mode (SDC) of the RHR system, and the reactor vessel head spray systems, are then normally placed into operation to cool the reactor water temperature to less than 212^{o}F. These systems also maintain the vessel temperature below 212^{o}F by removing the fission product decay heat absorbed by the reactor coolant. However, in this event it is assumed that the SDC mode of the RHR cannot be established.

Reactor cooldown is then performed by alternate shutdown cooling (ASDC), in which the low pressure core spray system is employed to continuously inject coolant into the vessel to remove the fission product decay heat. As illustrated in Figure 5, the heated reactor water is discharged through the safety/relief (S/R) valves to the suppression pool. Both of the residual heat removal (RHR) system heat exchangers systems are placed in operation to remove the heat from the suppression pool water.

2.2.2 Analysis Results

The required S/R valves discharge flow rate to achieve cold shutdown was evaluated as follows.

Twenty-four hours after scram, the reactor reaches a quasi-steady state. The fission product decay heat is conservatively assumed to be given by the May-Witt decay heat correlation.

$\dot{Q}_D$ = 0.713% of reactor initial power

$$\dot{Q}_D = 11{,}513 \text{ BTU/Sec} \quad (1)$$

This decay heat is dissipated into the suppression pool by flow through the S/R valves,

$$\dot{Q}_D = W_v \, C_p \, (T_v - T_p) \quad (2)$$

A schematic flow diagram of the alternate shutdown cooling system and the definition of the nomenclature is given in Figure 5.

The stored energy in the suppression pool is extracted by the two RHR heat exchangers which derived cooling water (service water) from the river intake,

$$\dot{Q}_{RHR} = K \, (T_p - T_{sw}) \quad (3)$$

The RHR heat exchangers are sized such that each of the two RHR loops has a heat removal capacity (K) of 199.6 (Btu/sec-°F). At quasi-steady state, all the decay heat discharged into the suppression pool is removed by the RHR heat exchangers. This is expressed mathematically as:

$$\dot{Q}_D = \dot{Q}_{RHR} \tag{4a}$$

or

$$W_v C_p (T_v - T_p) = K (T_p - T_{sw}) \tag{4b}$$

The suppression pool temperature and the flow rate through the S/R valves
is given by the following equations:

$$T_p = (\dot{Q}_D/K) + T_{sw} \tag{5}$$

$$W_v = (K/C_p) [(T_p - T_{sw})/(T_v - T_p)] \tag{6}$$

Equation (5) is obtained by substituting Equation (4a) into Equation (3), and Equation (6) is obtained by rearranging Equation (4b).

Since the service water temperature, T_{sw}, varies between Summer and Winter, the pool temperature is expected to vary accordingly. Assuming the Summer and Winter values listed in Table 1, the pool temperature and the required flow through the S/R valves was calculated. The results are given in Table 3. The required maximum flow through the S/R valves is about 205 lb/sec. This inventory is easily supplied by the core spray system, whose rated flow is 3020 gpm (~400 lb/sec) per loop.

The alternate shutdown cooling path, which employs the LPCS and S/RV flow, and the pool cooling mode of the RHR system will therefore allow reactor cooldown, and allows the reactor to be brought to a cold shutdown condition. The suppression pool temperature will also be maintained well within acceptable limits.

Alternate shutdown cooling has been approved on the GESSAR PDA and the Shoreham FSAR docket. Testing has also been conducted to verify acceptable performance of the S/RV's under the liquid flow conditions which result during alternate shutdown cooling.

3. CONCLUSIONS

The conclusions of this study are as follows:

(1) During the loss-of-offsite power and HPCI failure event, manually actuating 3 ADS valves within 10 minutes after scram, or manually actuating 2 ADS valves immediately following scram, will enable the LPCI and LPCS systems to be available to maintain fuel clad temperatures below the 2200°F limit, and bring the reactor to a cold shutdown condition.

(2) In the loss of shutdown cooling mode of RHR during normal shutdown event, the LPCS can be used to inject coolant into the vessel, flooding the entire vessel. The S/R valves should be manually opened to discharge the heated reactor water (carrying fission-product decay heat) back to the suppression pool. The required maximum flow through the S/R valves has been calculated to be 205 lb/sec, which is well within the capability of the LPCS. The alternate shutdown cooling path will therefore allow the reactor to be brought to a cold shutdown condition.

4. REFERENCES

1. Reactor Pressure Vessel Purchase Specification 21A1112, Attachment D, General Electric Company.

2. Monticello Reactor Pressure Vessel Stress Report, CB&I Contract No. 9-5624, General Electric Company Vendor Print File #1811-336-1.

TABLE 1

PLANT INITIAL CONDITIONS AND INPUT PARAMETERS

Parameter	Value
Initial Power (Mwt/%NBR)	1703/102
Vessel Pressure (psig)	1025
Turbine Pressure (psig)	965
Core Flow (lb/hr/%NBR)	$57.6 \times 10^6/100$
Steam Flow (lb/hr/%NBR)	$6.9 \times 10^6/102$
Feedwater Enthalpy (BTU/lb)	349
Initial Water Level (ft. above vessel zero)	40.5
Top of Active Fuel, (ft. above vessel zero)	29.29
Bottom of Active Fuel (ft. above vessel zero)	17.29
Rated LPCI Flow Rate at Reactor Pressure of 262 psig (gpm)	2,000 per pump (3 pumps)
Rated LPCS Flow Rate at Reactor Pressure of 130 psig (gpm)	3020 per system (2 systems)
Total No. of S/R Valves and Setpressure (#/psig)	8/1080 + 1%
Total No. of ADS Valves	3
S/R Valve Opening Delay (sec)	0.4
S/R Valve Closing Delay (sec)	1.5
S/R Valve Response Time (sec)	0.15

TABLE 1 (Cont'd)

PLANT INITIAL CONDITIONS AND INPUT PARAMETERS

Parameter	Value
Service Water Temperature, Summer (°F)	86
Service Water Temperature, Winter (°F)	37
RHR Pool Cooling Capacity, K (BTU/Sec-°F) (two loops)	199.6 per loop
Heat Capacity of Water, Cp (BTU/lb-°F)	1.0
May-Witt Decay Heat, 24 hours after scram (% initial power)	0.713
Reactor Vessel Temperature, 24 hours after scram (assumed), (°F)	200

TABLE 2

RESULTS OF LOSS OF OFFSITE POWER AND HPCI FAILURE EVENT FOR MONTICELLO

Transient Case	Time When Vessel Pressure Reached 300 psig (sec)	Reactor Water Level Inside Shroud* (ft)	Vessel Temp at Shroud* (°F)	Minimum Water Level Inside Shroud (ft above vessel 0)	Minimum Water Level Outside Shroud (ft above vessel 0)	Core Uncovered Time** (Sec)	Estimated Peak Clad. Temperature (°F)	Vessel Depress. Rate (psi/sec)
3 ADS with 10-min delay	860	26.4	431	25.9	24.9	57.3 (Lev. Loc. 5)	<1200	4.6
3 ADS with 5-min delay	585	28.2	429	27.4	25.6	5.8 (Lev. Loc. 5)	<1100	4.5
3 ADS with 3-min delay	471	28.7	429	28.1	26.2	2.0 (Lev. Loc. 5)	<1100	4.0
3 ADS at t = 0	335	30.1	428	29.4	27.2	0	-	3.9
2 ADS at t = 0	556	27.3	427	26.5	25.2	81.5 (Lev. Loc. 5)	<1200	2.8
1 ADS at t = 0	1410	20.5	427	18.6	23	457 (Lev. Loc. 3)	>2200	1.8

* Values at reactor vessel pressure of 300 psig

** The uncovered time is estimated by the SAFE computer code. SAFE is not the approved model to predict reflood time, but provides reasonable estimates. Level Location 5 is 1.1 ft below top of active fuel. Level location 3 is at mid-plane of the active fuel length.

TABLE 3

POOL TEMPERATURE AND REQUIRED FLOW THROUGH S/R VALVES
DURING ALTERNATE SHUTDOWN COOLING

Service Water Temperature (°F)	Number of RHR Heat Exchanger Loops	Maximum* Suppression Pool Temperature (°F)	Required Flow Through S/R Valves (lb/sec)
37 (Winter)	1	94.7	109.4
86 (Summer)	1	143.7	204.6
37 (Winter)	2	65.8	85.7
86 (Summer)	2	114.8	134.9

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* From Equation (5), based on decay heat of 0.713% at 24 hours after scram.

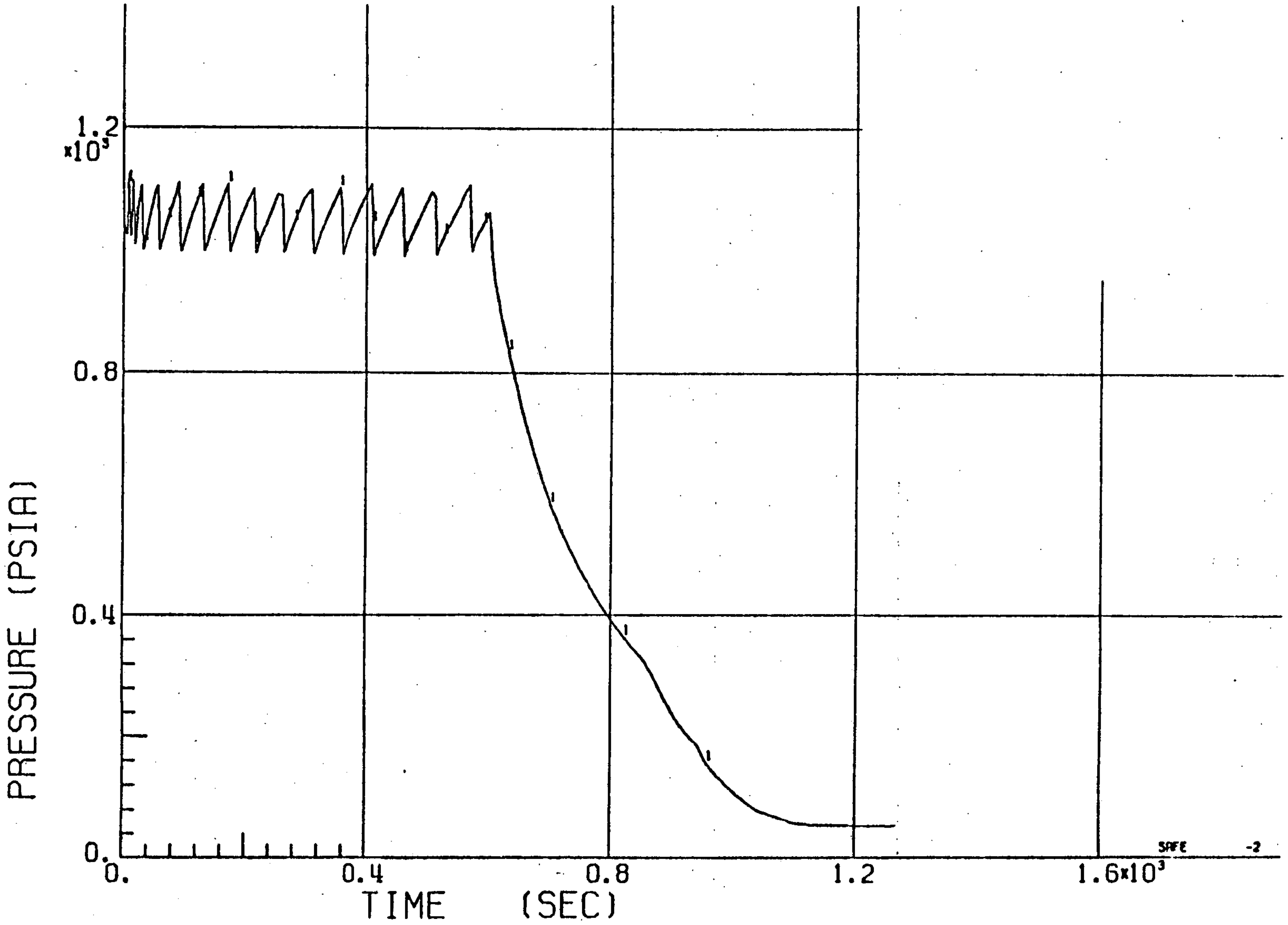


FIGURE 1a.

REACTOR PRESSURE VS. TIME, 3 ADS VALVES 10 MINUTES AFTER SCRAM

198C4

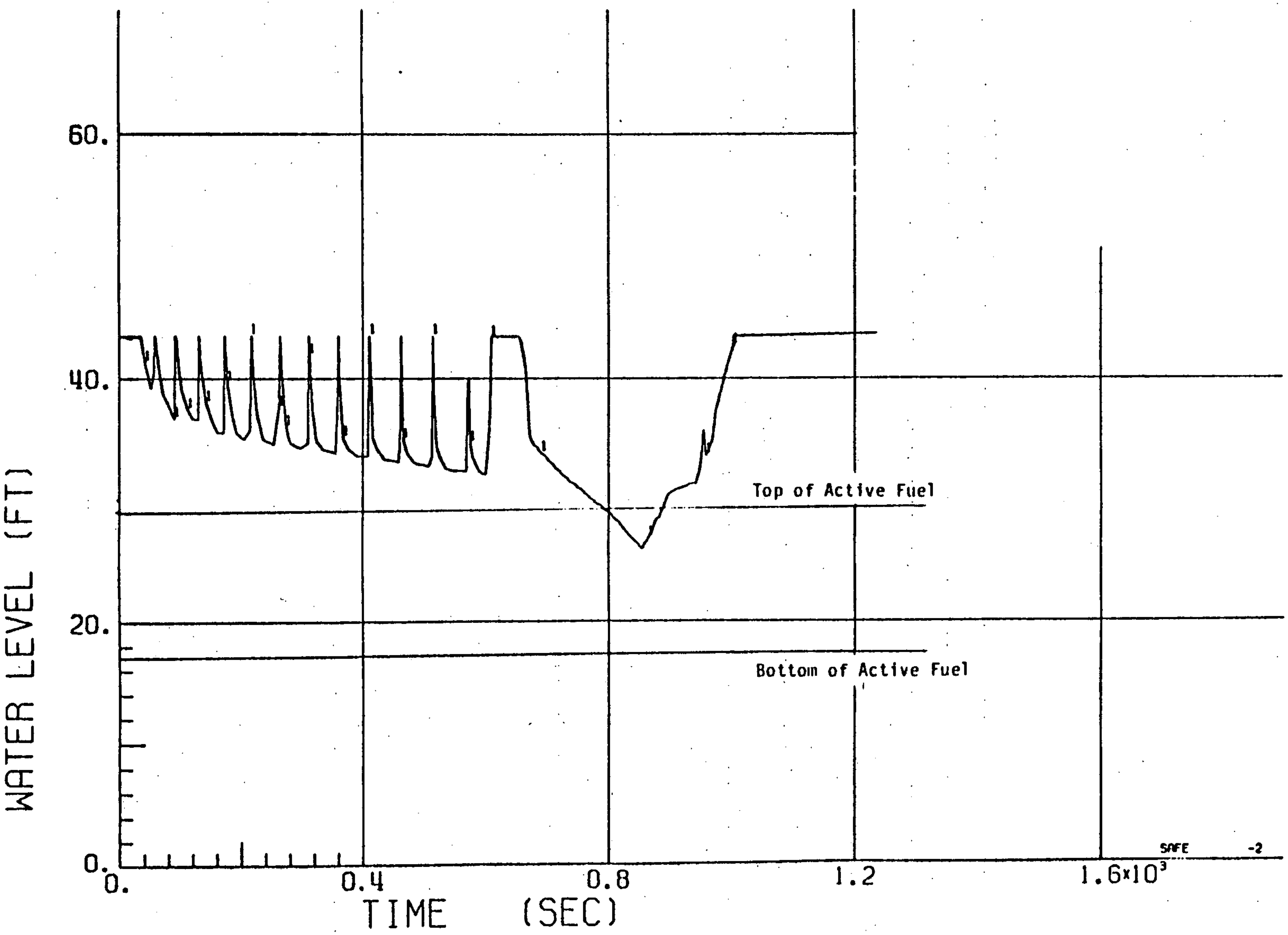


FIGURE 1b.

REACTOR WATER LEVEL VS. TIME, 3 ADS VALVES 10 MINUTES AFTER SCRAM

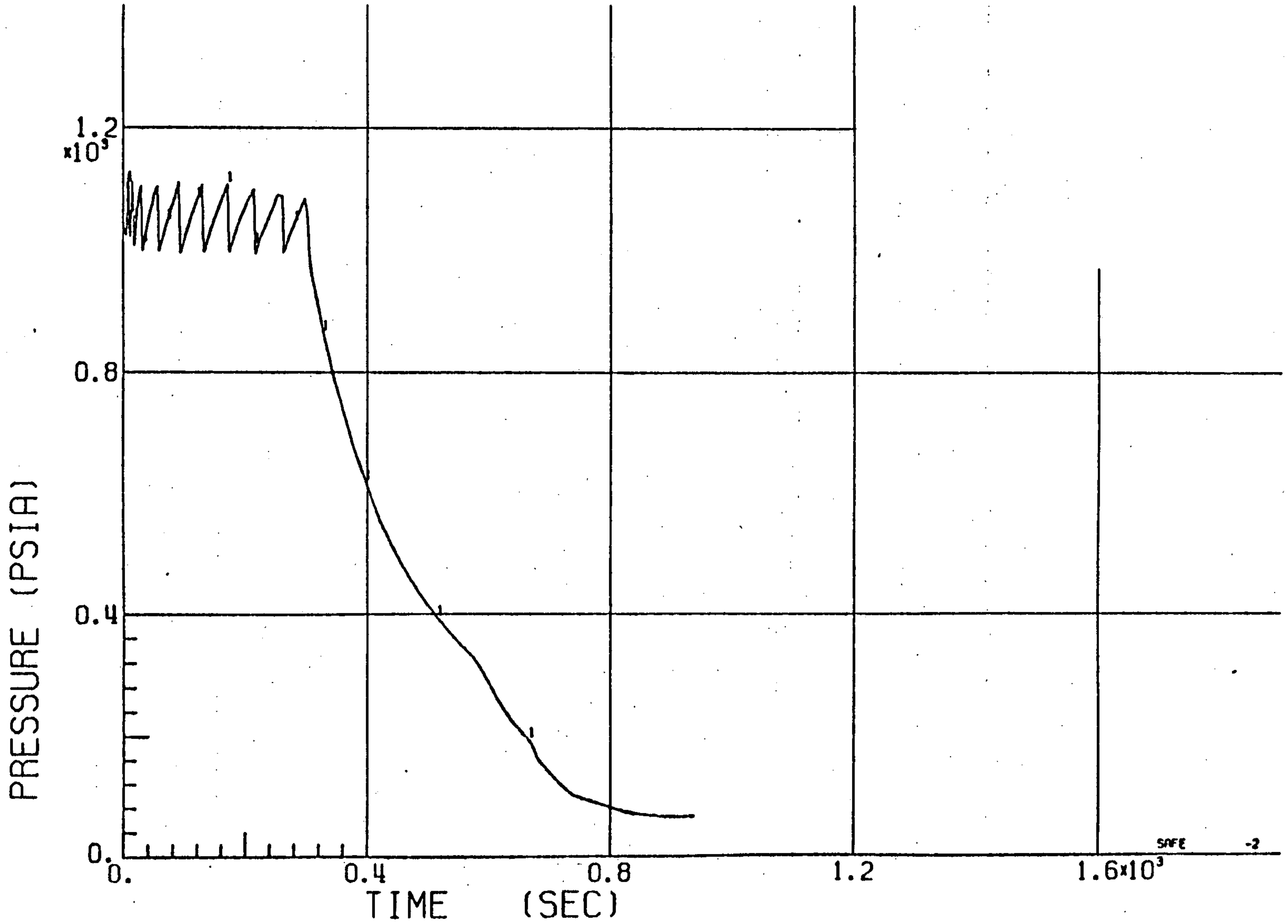


FIGURE 2a.

REACTOR PRESSURE VS. TIME, 3 ADS VALVES 5 MINUTES AFTER SCRAM

198C5

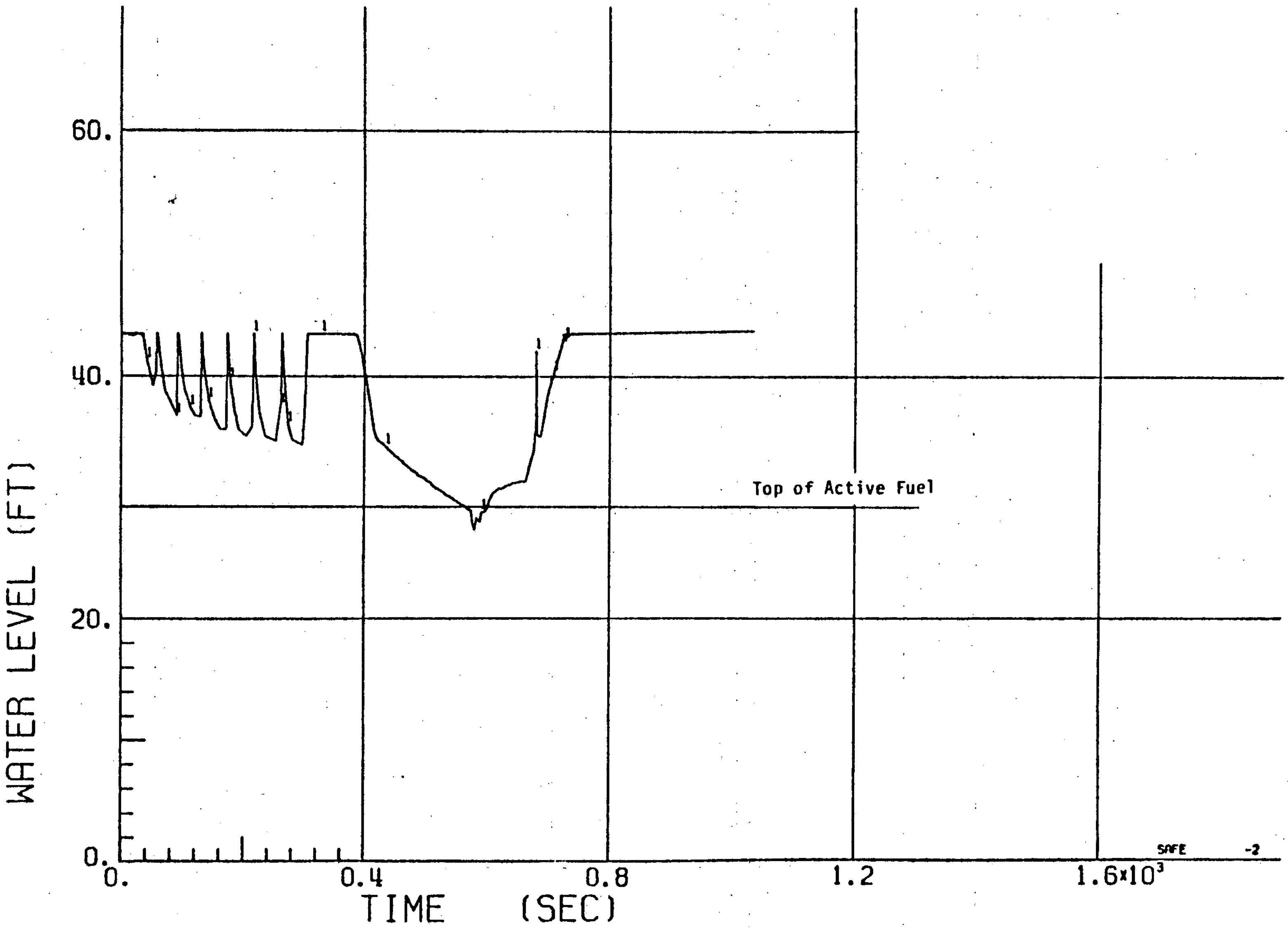


FIGURE 2b.

REACTOR WATER LEVEL VS. TIME, 3 ADS VALVES 5 MINUTES AFTER SCRAM

198C9

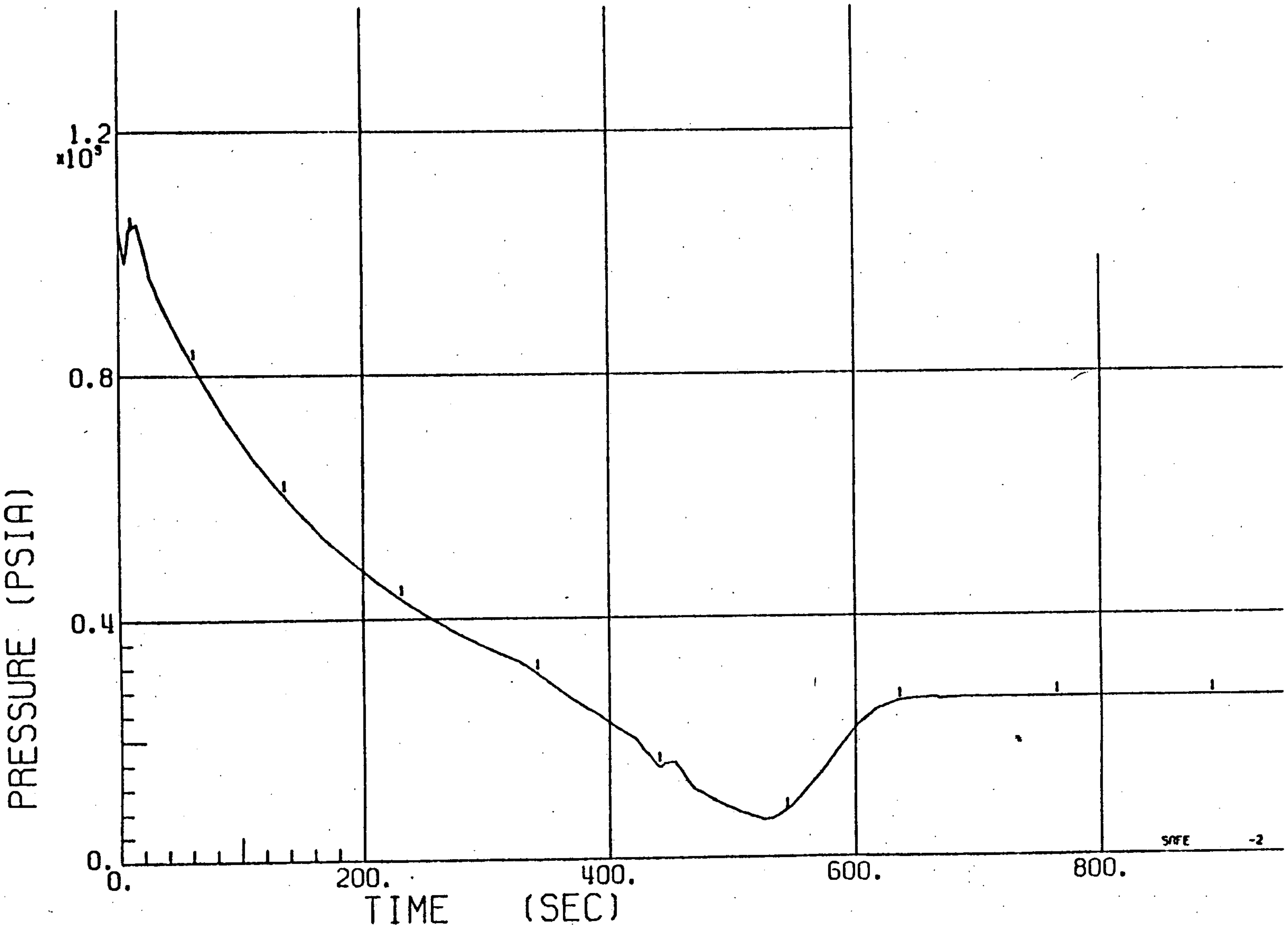


FIGURE 3a.

REACTOR PRESSURE VS. TIME, 3 ADS VALVES AT TIME OF SCRAM

198C6

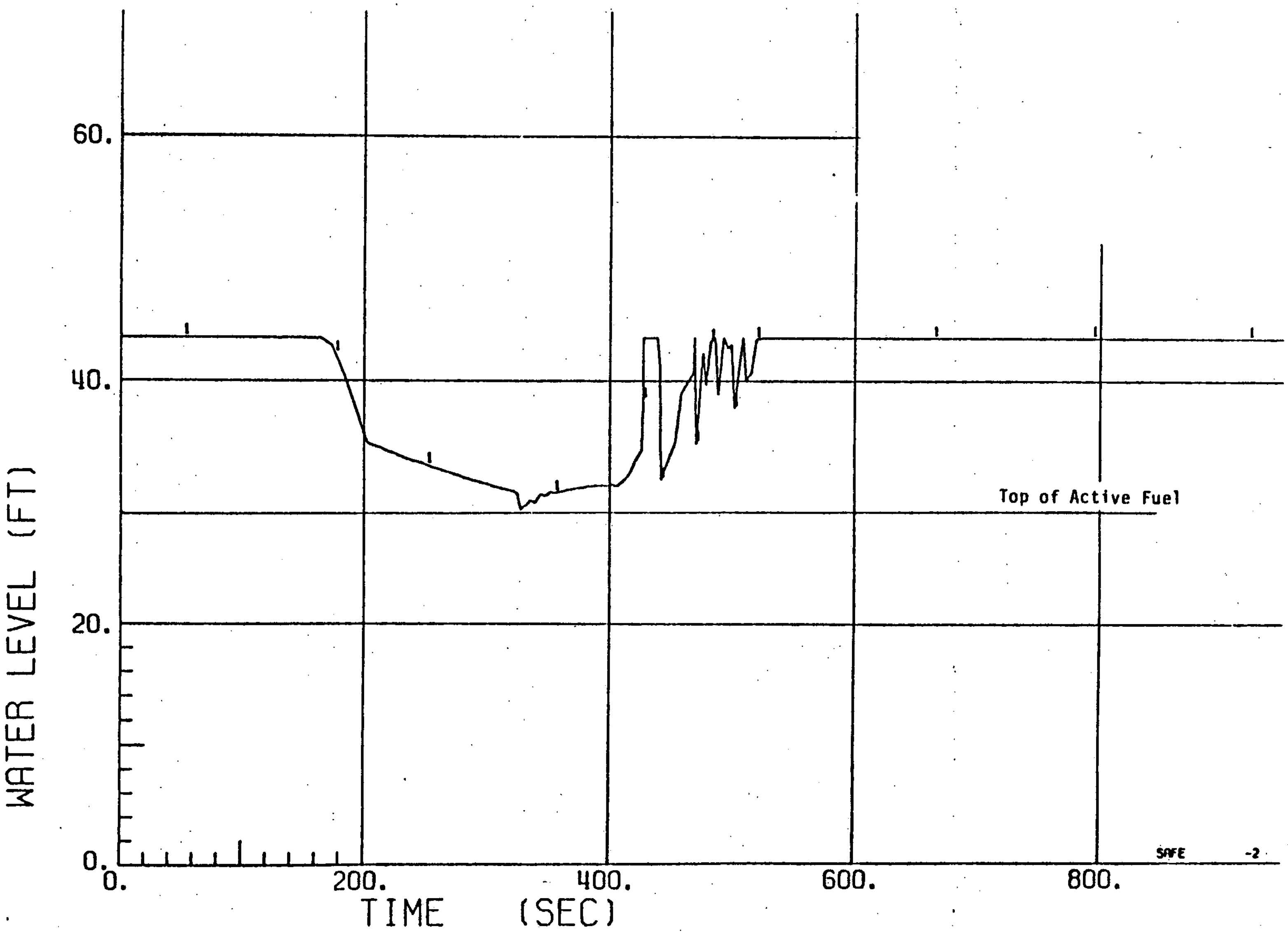


FIGURE 3b.

REACTOR WATER LEVEL VS. TIME, 3 ADS VALVES AT TIME OF SCRAM

198C10

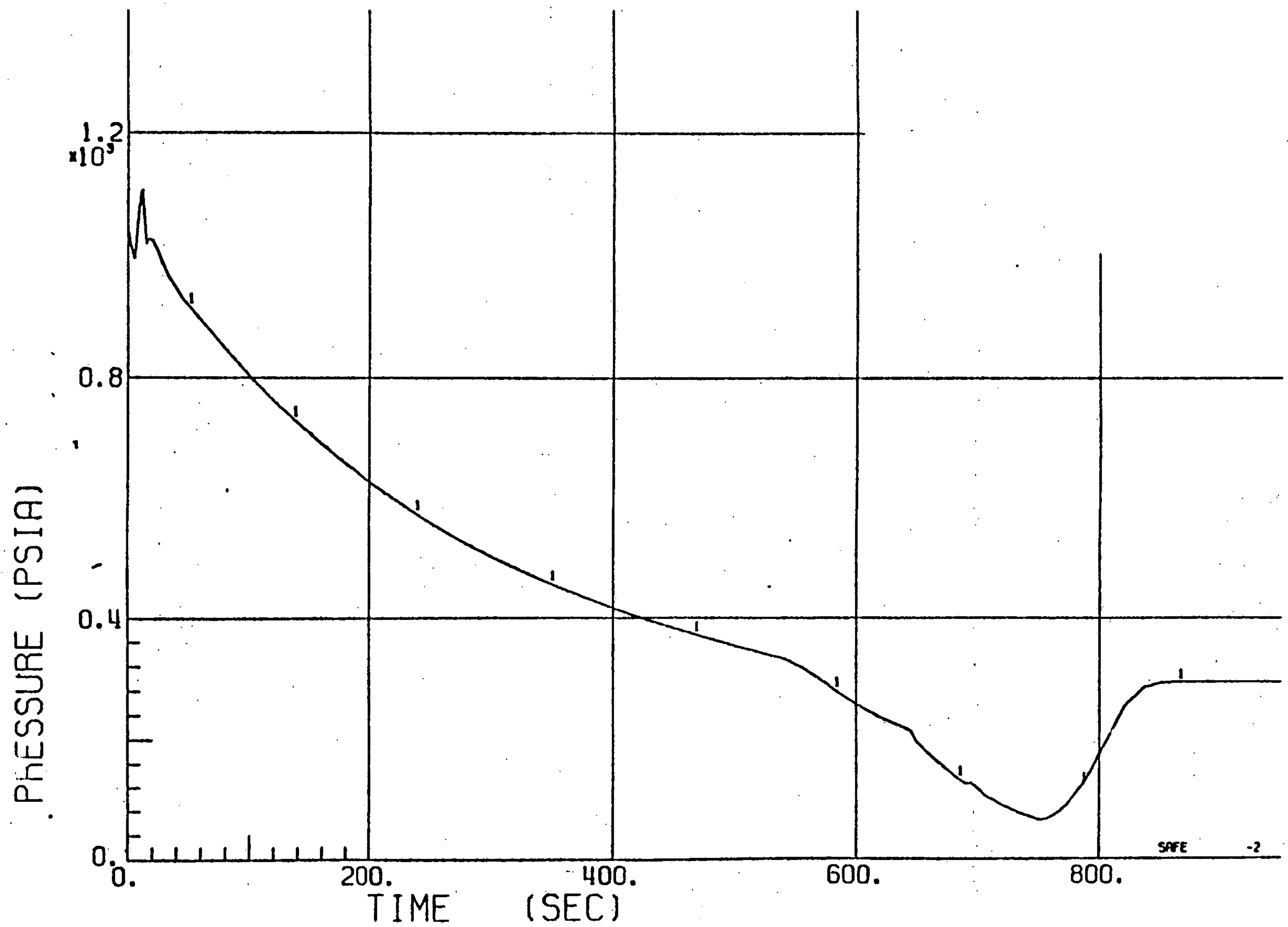


FIGURE 4a.

REACTOR PRESSURE VS. TIME, 2 ADS VALVES AT TIME OF SCRAM

198C7

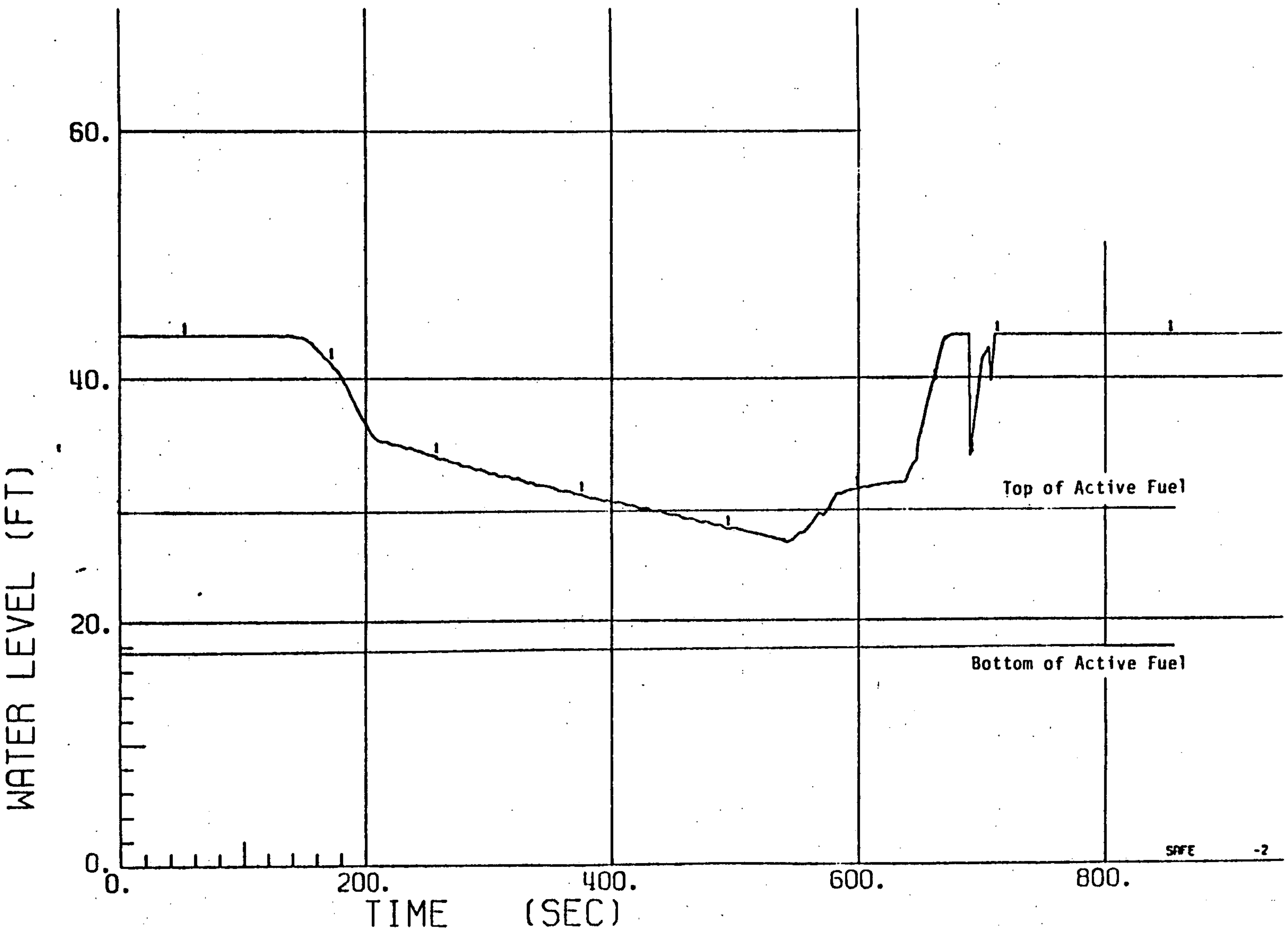
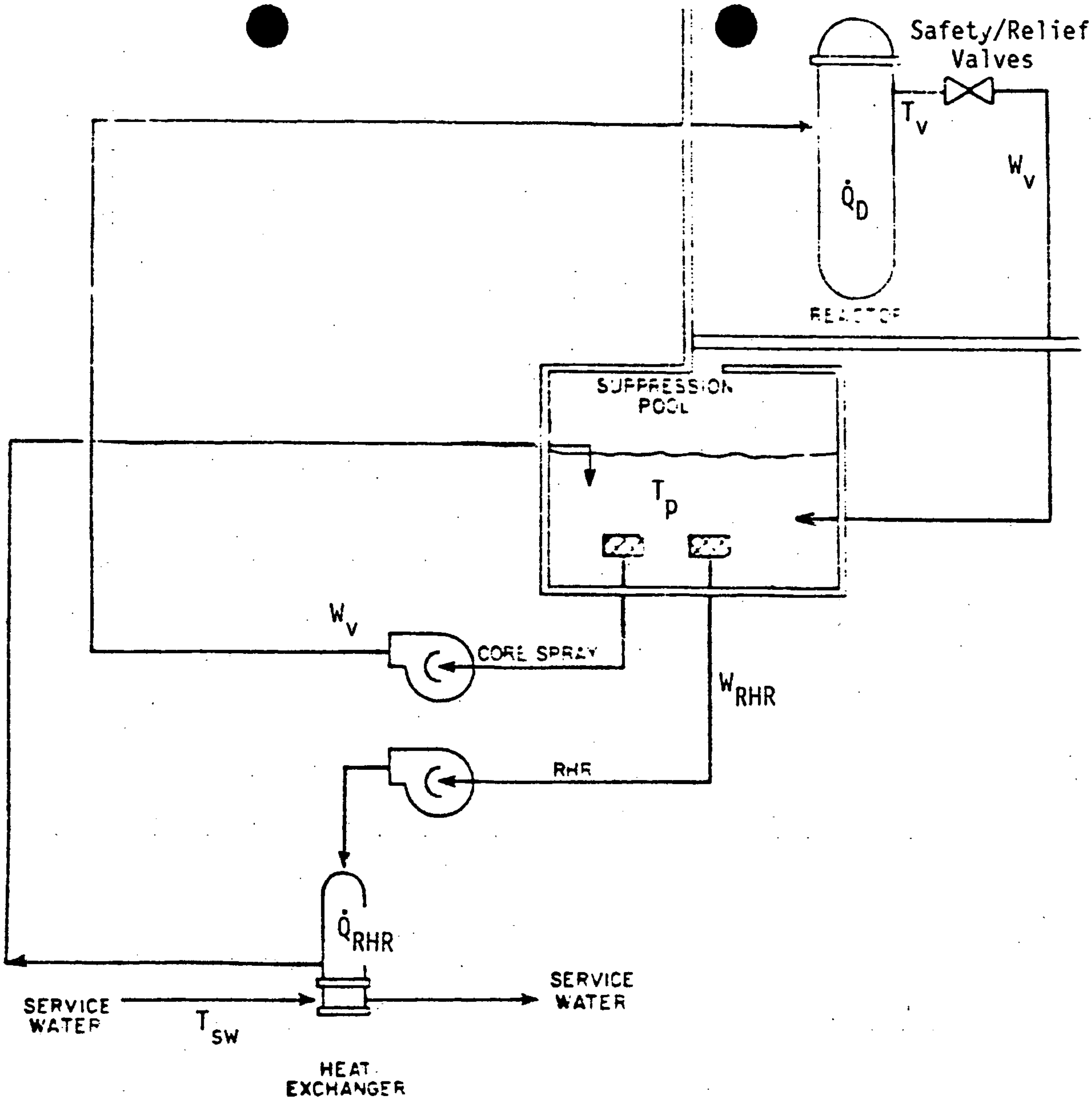


FIGURE 4b.

REACTOR WATER LEVEL VS. TIME, 2 ADS VALVES AT TIME OF SCRAM

198C11



T_{sw} = service water temperature (°F)

T_v = vessel water temperature (°F)

T_p = suppression pool water temperature (°F)

W_v = S/R flow rate (lb/sec)

K = RHR pool cooling capacity (BTU/Sec-°F)

$\dot{Q}_{RHR}$ = $K\,(T_p - T_{sw})$, Heat Exchanger Rate (BTU/sec)

$\dot{Q}_D$ = $W_v\,C_p\,(T_v - T_p)$, Fission Product-Decay Heat Rate (BTU/sec)

C_p = Heat capacity of Reactor Water (BTU/lb-°F)

FIGURE 5 SCHEMATIC FLOW DIAGRAM OF ALTERNATE SHUTDOWN COOLING SYSTEM

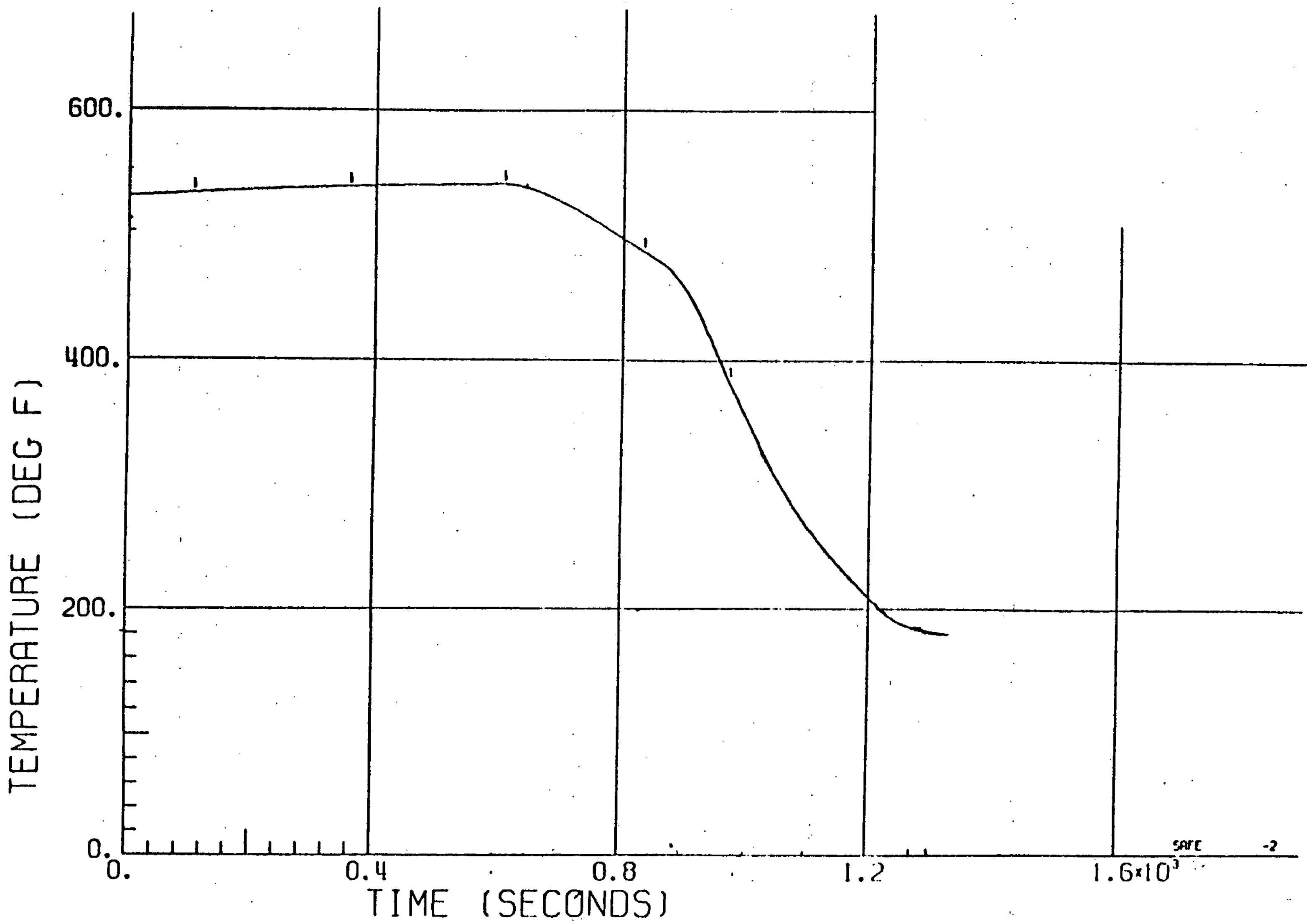


FIGURE 6

VESSEL LOWER PLENUM TEMPERATURE VS. TIME, 3 ADS VALVES 10 MINUTES AFTER SCRAM

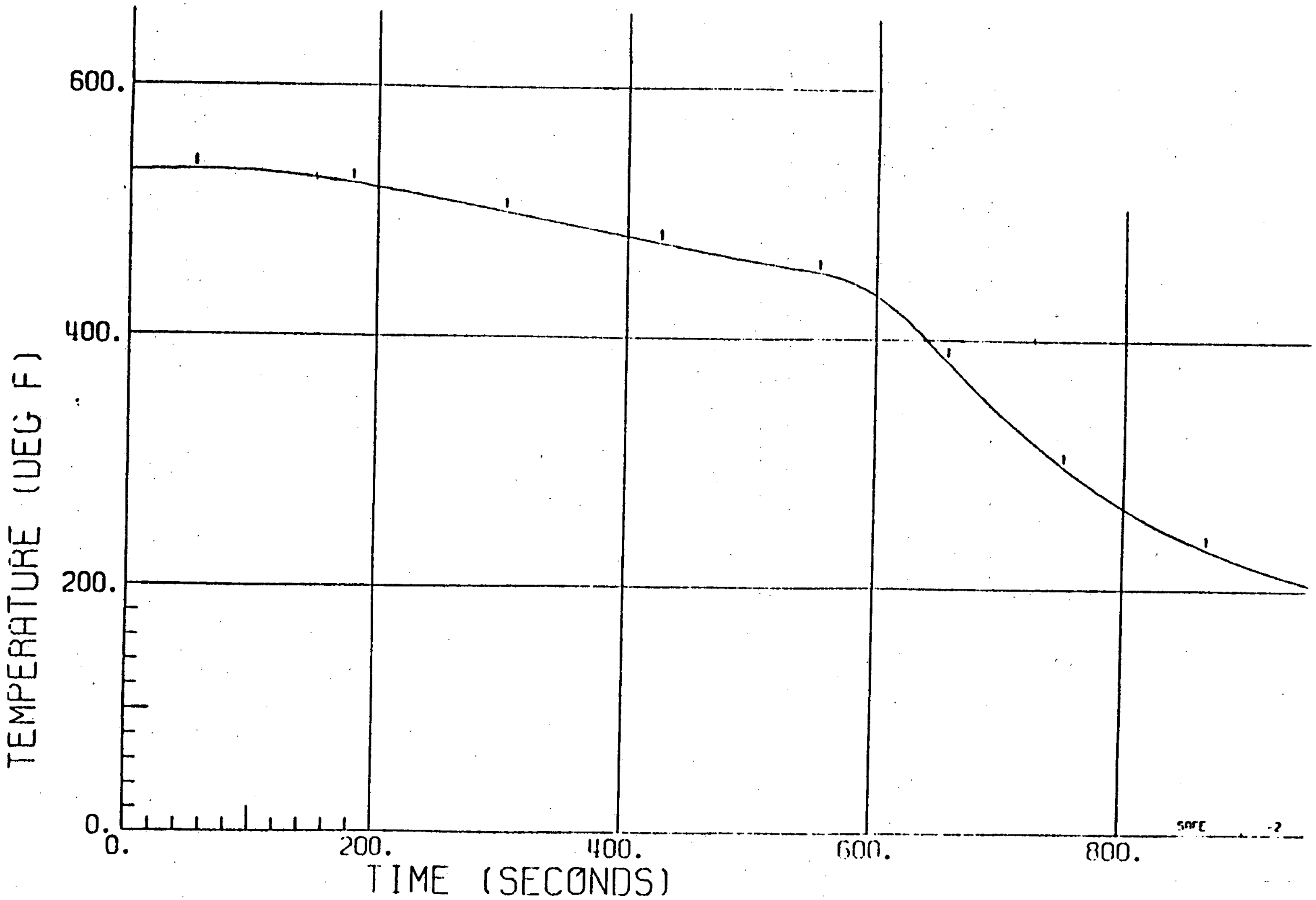


FIGURE 7

VESSEL LOWER PLENUM TEMPERATURE VS. TIME, 2 ADS VALVES AT TIME OF SCRAM