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CONF-851007--4

LA-UR -85-945

TITLE A TRAC-PF1/MOD-1 ANALYSIS OF LOSS-OF-FLOW TEST L9-4

LA-UR--85-945

TI85 010106

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SUBMITTED TO The Third International Topical Meeting on Reactor Thermal Hydraulics
Newport, RI
October 15-18, 1985

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FORM NO. 836 R4

A TRAC-PF1/MOD-1 ANALYSIS OF LOSS-OF-FLOW TEST L9-4*

by

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ABSTRACT

Los Alamos National Laboratory is developing the Transient Reactor Analysis Code (TRAC) to provide advanced best-estimate predictions of postulated accidents in pressurized water reactors (PWRs) and for many thermal-hydraulic experimental facilities.

As part of our independent assessment of code version TRAC-PF1/MOD1, we analyzed Loss-of-Fluid Test (LOFT) L9-4 and compared the test data to the calculated results. This was an anticipated-transient-without-scram test in which the pumps were tripped, the steam generator main feedwater discontinued, and the main steam-outlet valve closed. This data comparison is the first extensive test of TRAC's reactor-kinetics models. The comparisons show that TRAC can calculate the power generation within a nuclear reactor if the program is supplied with adequate reactor-kinetics input specifications. The data comparisons also indicate that TRAC calculated the thermal-hydraulic parameters within LOFT well with only minor discrepancies.

A number of models within TRAC-PF1/MOD1 were verified for the first time. They include the reactor-kinetics models, the trip-activated time-step controls, and the LOFT pump-coastdown calculations.

In general, the final input description is adequate to analyze the experiment. The calculations indicate the importance and difficulty of obtaining accurate and applicable reactor-kinetics input data. They also indicate the need to include the effects of xenon-poisoning buildup in the analysis.

*This work was funded by the US Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Division of Accident Evaluation.

1. INTRODUCTION

Los Alamos National Laboratory is developing the Transient Reactor Analysis Code (TRAC)[1] to provide advanced best-estimate predictions of postulated accidents in pressurized water reactors (PWRs) and for many thermal-hydraulic experimental facilities.

As part of our independent assessment of code version TRAC-PF1/MOD1, we analyzed Loss-of-Fluid Test (LOFT) L9-4 and compared the test data to the calculated results. This was an anticipated-transient-without-scram (ATWS) test in which the pumps were tripped, the steam-generator main feedwater discontinued, and the main steam-outlet valve closed. This data comparison is the first extensive test of TRAC's reactor-kinetics models. The comparisons show that TRAC can calculate the power generation within a nuclear reactor if the program is supplied with adequate reactor-kinetics input specifications. The data comparisons also indicate that TRAC calculated the thermal-hydraulic parameters within LOFT well with two minor exceptions: the velocity in the intact loop was consistently greater in the TRAC calculation although it generally fell within the data uncertainty; and because of the leakage behavior of the reflood-assist bypass valves (RABVs), the temperatures and flows in the broken loop were not calculated well.

A number of models within TRAC-PF1/MOD1 were verified for the first time. They include the reactor-kinetics models, the trip-activated time-step controls, and the LOFT pump-coastdown calculations.

In general, the final input description is adequate to analyze the experiment. The calculations indicate the importance and difficulty of obtaining accurate and applicable reactor-kinetics input data. They also indicate the need to include the effects of xenon-poisoning buildup in the analysis.

The details of the test, analysis, and data comparisons are presented in the following sections. There we:

- describe the test apparatus,
- describe the experiments and explain the experimental phenomena,
- describe the TRAC input, and
- present and discuss the data comparison.

2. FACILITY DESCRIPTION

The LOFT facility (shown in Fig. 1) is a 50 MW(t) PWR, detailed descriptions of which may be found in Refs. 2, 3, and 4. The description presented here will be limited to the particular configuration of the facility used for Test L9-4 and to specific details of the facility particularly important to this experiment.

The characteristics of the LOFT nuclear core were important in this test because one of the primary purposes of these data comparisons was to check the reactor-kinetics models in TRAC. The core consists of an array of 1300 1.67-m fuel pins: the core has a equivalent diameter of 0.61 m. From a reactor-kinetics standpoint, the LOFT core is not typical, in that it has a much higher radial-peaking factor than a normal PWR. The nonuniformity of fluxes complicated our obtaining applicable input specifications for TRAC's kinetics models and makes the results less accurate than might be obtained in a typical PWR analysis.

The ground rules for this test as detailed in the experimental operating procedures[3] were that the power-operated relief valve (PORV) was assumed to be inoperative, with excess primary-system pressure relieved through the pressurizer by a safety relief valve (SRV). The SRV was scaled to represent three Zion SRVs. The pressurizer spray also was inoperative. Most of the broken-loop hot leg was removed, and both the broken-loop hot and cold legs terminated at blind flanges. Water and steam could leave the primary system only through the SRV.

The RABVs are located between the broken-loop cold and hot legs. These valves are made to seal when there is a significant pressure difference across their seats. When the pressure difference is low or when the direction of the pressure difference changes, they leak in an unpredictable manner. This leakage made accurate calculation of their performance impossible within the context of TRAC.

No emergency core-cooling system (ECCS) water was added during this transient. A limited amount of water was added to the steam-generator secondary (SGS) from the auxiliary feedwater system. Steam was released from the SGS by the main-bypass valve, which was operator controlled to maintain the SGS pressure between 6.29 and 6.63 MPa per the experimental operating specification.[3]

3. TEST DESCRIPTION

The important events and phenomena during this test are listed in Table I and described below.

Before the transient began, the reactor was run for ~50 h. Therefore, near-equilibrium values of decay heat and xenon buildups could be expected.

The test began when the coolant pumps tripped, the main SGS feedwater terminated and the main SGS outlet valve began closing. At 10 s, an auxiliary SGS feedwater flow of 0.05 ℓ/s began, and the main steam-bypass valve was used to control the SGS pressure between 6.29 and 6.63 MPa.

During the early part of the transient, several events occurred more or less simultaneously. As the coolant pumps coasted down, the mass flow in the primary loop dropped. This decreased flow caused the temperature difference across the core and the average core-liquid temperature to increase. This increased average temperature caused the overall reactor multiplication constant k to fall below 1.0, which resulted in a decline in core power. The average core-liquid temperature continued to rise steeply until natural circulation stabilized the intact-loop flow. Stabilized flow and the falling power caused the average core-liquid temperature to decrease. A change in the slope of the core power also occurred at this time, even though the overall reactor multiplication constant k remained below 1.0.

Along with the decrease in reactor power came a decrease in the average temperature of the reactor fuel and a corresponding increase in the fuel-temperature reactivity. These trends continued until ~250 s into the transient when, because of the rise in the reactivities associated with fuel temperature and coolant temperature, the value of k increased to close to 1.0.

Other phenomena occurring during the first 250 s included an expansion of the fluid in the primary system resulting from the increase in temperature of the fluid in the core and hot leg. Water was forced into the pressurizer, and the system pressure increased until relieved by the SRV. When the core power dropped, a slow buildup of negative reactivity began, caused by the buildup of xenon within the fuel rods. At 37 s into the transient, the speed of the second pump fell to the point at which friction caused its rotor to stop.

At the beginning of the transient, the rate of water flow into the SGS was insufficient to make up for the steam vented, and the level of liquid in the SGS steadily decreased. Between 250 to 500 s, this decrease degraded heat transfer within the secondary to the point that the steam-generator-outlet temperature on the primary side began to increase.

The low flow rate during natural circulation resulted in a delay before the hotter water reached the core. Mixing with fluid in the lower plenum and also heat transfer to the vessel walls increased the delay and prevented a sharp increase in the core-fluid temperature. Eventually, the average temperature in the core rose as a result of dryout in the SGS, and the k again decreased, causing the power to decrease. As described before, a compensating effect occurred because of a decrease in both the fuel temperature and the temperature difference across the core, and a new core-power equilibrium was reached at about 1050 s.

The framework of this equilibrium is interesting in that the power generation in the reactor is just equal to the power removed in the rest of the system. Any difference between the two results in a change in average core-liquid temperature, and a new equilibrium is reached quickly. Also note that the negative reactivity introduced as a result of xenon buildup caused a compensating decrease in the average core-liquid temperature.

Finally, the decrease in natural-circulation driving potential that occurred when the SGS dried out resulted in stopping of the first pump at 732 s.

4. TRAC INPUT DESCRIPTION

A component diagram of the input description is presented in Figs. 2 and 3. The input used a total of 39 components consisting of 161 volumes. In general, four nodes were used to model heat transfer in the walls, except in the vessel, where 10 nodes were used. The modeling appears to provide sufficient detail, with the possible exception of the SGS, where more detailed modeling would result in a smoother change in heat transfer during the dryout.

The analysis was conducted using TRAC-PF1/MOD1, Version 11.6, with updates JMFX1 and JMFX2 included and update FXSBCOOL removed.

The input description was based on that for LOFT test L6-1 (Ref. 5). Changes included:

- modification of the TEE modeling in the vessel to give better steady-state pressure drops;
- changes to the trips and control blocks, reflecting differences in the initial conditions and test procedures;
- modified pump input to reflect the decoupling of the motor generator during pump coastdown (the option to do this was included in one of the updates added in this version of TRAC);
- boundary and initial conditions to reflect Test L9-4;
- the removal of a small ECCS flow, which was included in the L6-1 input to represent water injected through the pump bearings; and
- extensive changes in the reactor-kinetics input, which are discussed below.

That the power profiles in LOFT are more peaked than in a normal PWR makes it important that the reactor-kinetics input, and particularly the input quantity POWEXP, is correct. The parameter POWEXP is the exponent to which the cell values of the normalized power distribution are raised in calculating the average reactivity-feedback parameters. The correct exponent is dependent on the procedure used to obtain the values of the fuel-temperature reactivity coefficients (RCTF) and coolant-temperature reactivity coefficients (RCTC) input into TRAC.

If these coefficients were obtained from a source such as LEOPARD (Ref. 6), where the spatial flux distribution in the core was not included, the correct value for POWEXP would be approximately 2.0. This was true of the data used for the L9-4 analysis.

Even using a POWEXP of 2.0, there were two other inaccuracies in the L9-4 reactor-kinetics input. The effects of radial flux peaking were not included because of the use of a one-dimensional core. The effects of the 1/T dependency of reactivity on fuel temperatures also were not included in the averaging scheme used by TRAC.

The values used for RCTF in the L6-1 analysis were obtained from Ref. 7. They reflected a correction that accounted for the use of a POWEXP of 0.0. Because we used a POWEXP of 2.0, it was necessary to input the original calculated values given in Ref. 8.

The values of RCTC used in the L6-1 input were multiplied by a factor of 0.84 to reflect the recommendations given in Ref. 7.

The effects of xenon-poisoning buildup were included by using the programmed-reactivity option in TRAC. Input values obtained from INEL were confirmed using the the LEOPARD code with an input reflecting power-squared averaging of the thermal flux.

5. PRESENTATION AND DISCUSSION OF DATA COMPARISONS

Figures 4 through 14 present the calculated results with comparisons from the experiment. The designation in the box at the right of the curves indicates the LOFT transducer designation--for example, RE-T-77-2A2. The TRAC component type and number are listed at the bottom of the box--for example, CORE ID=37. The cell or cell boundary is given at the top of the box. In general, TRAC results are presented as a solid line, whereas experiment results are presented as a dashed line. In some figures, the ordinate axis is scaled to show more detail in the region of interest at the expense of initial conditions.

Most of the calculated results are within the error bounds of the test data, and the size of the error bounds exhibited by much of the data would seem to limit the significance of differences between the calculated results and the data. There is a consistency, however, between the different transducers that raises the significance of the differences in the comparisons. For example, the mass flow in the intact loop and the temperature difference across the core are consistent with the core power.

Figure 4 presents a comparison of calculated and measured reactor power. In general, the comparison between the two is good. The most significant difference is slightly lower calculated power between 40 and 600 s. Although the calculation is well within the error limits, the difference also could be caused by the lack of radial flux-peaking effects in the TRAC analysis.

Figures 5 through 7 present the calculated coolant-temperature reactivity, fuel-temperature reactivity, and the reactor multiplication constant k. If we examine Fig. 5, we can see the effects of the average coolant temperature on the reactor kinetics. Each phase in this figure may be linked to a change in the coolant temperature.

1. First, a rapid heating occurs as a result of the coastdown of the pumps.
2. Next, a cooldown occurs as a result of stabilization of intact-loop flow rates and a decrease in core power.
3. Another heating occurs as a result of the steam-generator dryout.
4. Finally, a gradual cooldown occurs as a result of the xenon-poisoning buildup.

The effect of core average coolant temperature throughout the transient is to decrease the total reactivity.

In Fig. 6, we see the effects of fuel temperature on reactivity. The effect of the core average fuel temperature is to increase the total reactivity throughout the transient. In Fig. 7, the calculated reactor multiplication constant k is shown. The reactor multiplication constant k has a value near 1.0 for only brief periods of time after the beginning of the transient and never reaches the highly negative values associated with a scram (the effect of the scram at 1507 s was not included in the programmed-reactivity table).

Figure 8 presents the intact-loop hot-leg liquid temperature. The TRAC calculations and the test data agree reasonably well. The fact that the TRAC results are generally lower than the test data indicates that the calculated mass flow in the intact loop is too high.

Figures 9 and 10 present liquid temperatures for the steam-generator outlet plenum and vessel lower plenum. For the first 200 s, the coastdown of the pumps has little effect on the steam-generator outlet temperature. Then, the dryout of the SGS causes the outlet temperature to increase sharply. Although this increase is comparatively smooth in the test data, it occurs in steps in the calculations because of the limited number of volumes used in the TRAC input to represent the SGS above the tube sheet. When we compare the steam-generator outlet plenum temperature to that in the vessel lower plenum (Fig. 10), we see that mixing in the lower plenum and heat transfer to the vessel walls has a significant moderating effect on the change in the temperature of the water entering the core. After 600 s, there is a consistent, small overprediction of these temperatures by TRAC. We attribute this to the overpredicted intact-loop mass flow, which results in an underprediction in the temperature difference across the core.

In Fig. 9, the decline in steam-generator outlet-plenum liquid temperature, which occurs in both the test data and the calculated results after 650 s, is caused by the xenon-poisoning buildup in the reactor. In an alternate analysis in which xenon poisoning was not included, this temperature remained high and relatively constant, causing incorrect calculations of the primary-system pressure.

In Fig. 11, the calculated and measured intact-loop hot-leg liquid velocities are compared. TRAC appears to overpredict this velocity consistently. This overprediction agrees with the comparison made of the temperatures across the core. Although the calculated intact-loop hot-leg flow also was high in Test L6-7 (Ref. 9), the inverse temperature profile in the downcomer seen in LOFT L6-7 is not present in Test L9-4 because Test L6-7 cooled the primary whereas this test heated the primary.

The broken-loop hot-leg temperature comparison is presented in Fig. 12. The test data undergo a very steep rise just after the start of the transient. We believe this measured temperature rise was caused by a large temporary leakage through the RABV when the direction of flow within it reversed. The TRAC input did not include provisions for this leakage and consequently the calculated result did not match the test result. The broken-loop cold-leg temperatures showed large differences that we believe also may be partially a result of the temporary leakage in the RABV. The location of the broken-loop hot- and cold-leg liquid temperature measurements in the LOFT system also affects these data comparisons. These measurements are located at the top of pipes terminated with blind flanges that are beyond the point where the RABV connects the hot and cold broken loops (see Fig. 1, measurement stations BL-1 and BL-2). Therefore, they are in regions of stagnant liquid and may be affected by thermal stratification of this liquid.

Figure 13 compares the calculated and measured primary-system pressures. On the whole, TRAC results compare reasonably well to the measured data. The differences evident after 850 s may be a result of differences in the heating and cooling of fluid in the broken loop.

Figure 14 presents the CPU time as a function of problem time. As a whole, TRAC runs the analysis at better than real time. The majority of the CPU time is used to match the rapid fluctuations that occur during SRV cycling. The calculation was run on a Cray-1 computer.

During the course of the analysis, many alternate input descriptions were used. The significant results of the alternate analyses were

- uncertainties in the values of RCTC and RCTF can lead to significant differences in the results, and
- the inclusion of xenon-poisoning buildup is necessary, especially for the calculation of primary-system pressure.

6. CONCLUSIONS

Analysis of the predicted results of LOFT L9-4 showed that TRAC-PF1/MOD1 can simulate adequately most of the phenomena associated with this ATWS experiment. Models within TRAC that were extensively verified for the first time included the reactor-kinetics models, the trip-activated time-step controls, and the LOFT pump-coastdown calculations.

In general, the final input description was adequate to analyze this experiment. The calculations indicated the importance and difficulty in obtaining adequate reactor-kinetics input data. They also indicated the need to include the effects of xenon poisoning in the analysis.

Finally, TRAC did an adequate job of handling the reactor kinetics during a ATWS experiment in the LOFT systems. Because of the flatter flux profiles, we would expect TRAC to do an even better job of handling the reactor kinetics in a typical reactor.

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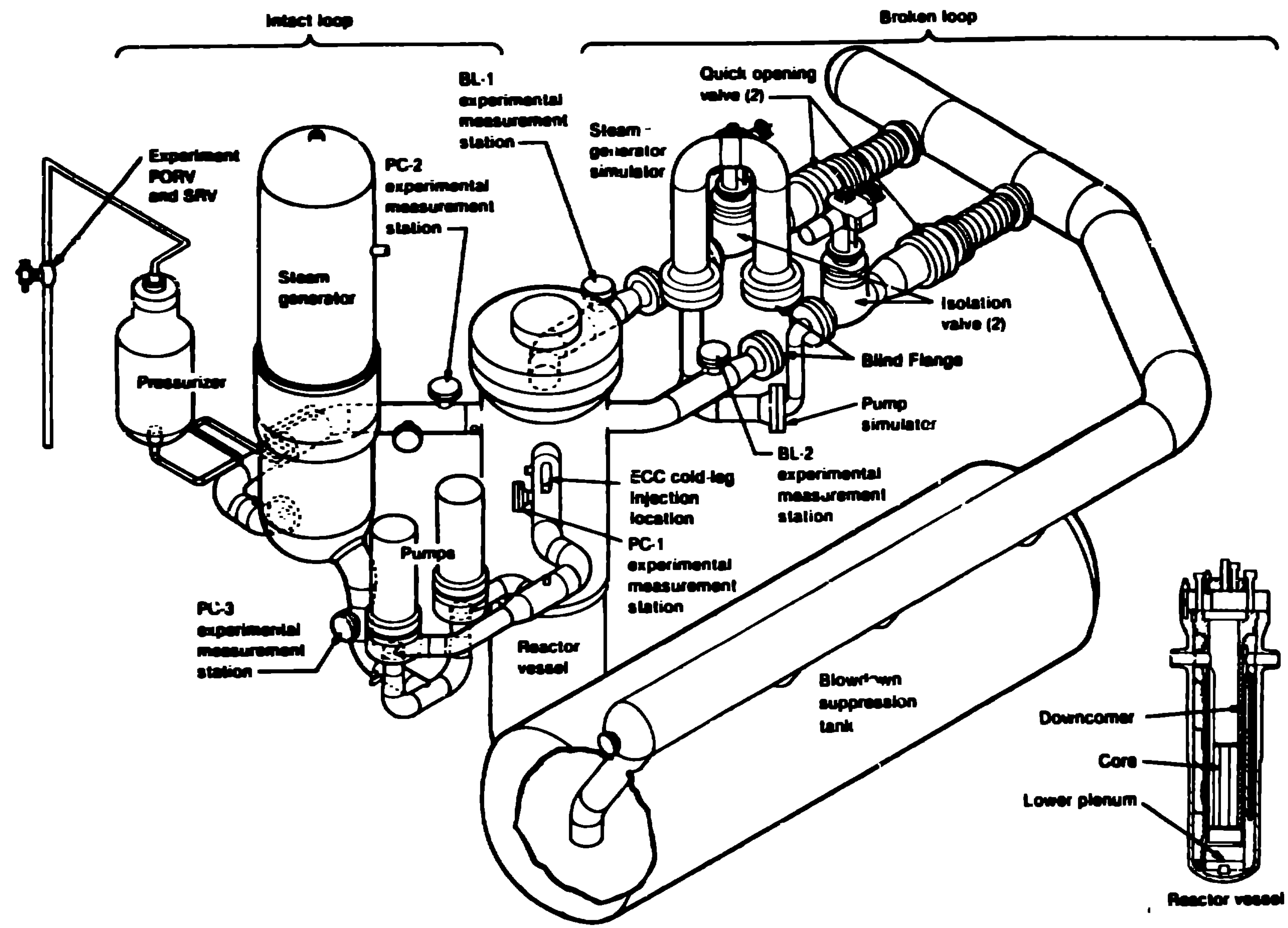
[9] J. K. Meier, "A TRAC-PF1 Analysis of Loss-of-Fluid Test L6-7/L9-2." Proceedings of the Joint ANS/ASME Meeting on Design Construction and Operation of Nuclear Power Plants, Portland, Oregon, August 5-8, 1984, ASME paper 84-NE-8.

TABLE I

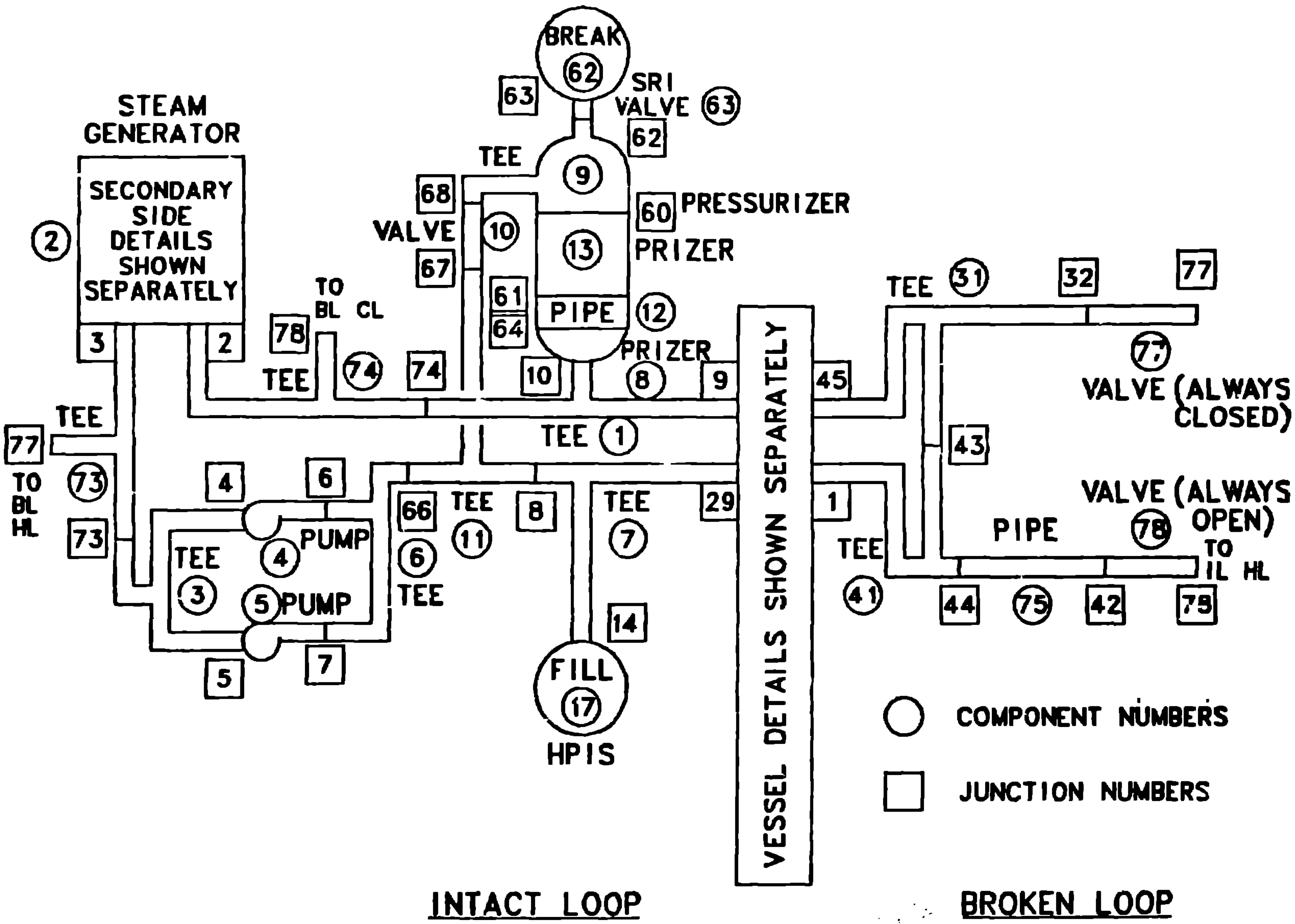
SEQUENCE OF EVENTS FOR EXPERIMENT L9-4

Event	Measured (s)	Calculated (s)
Primary-coolant pumps tripped	0.0	0.0
Main-feedwater pump tripped	0.15 ± 0.05	0.0
Auxiliary-feedwater flow initiated	10.8 ± 0.2	10.0
SRV started to open, cycling initiated	18.5 ± 0.1	22.1
Primary-coolant pump 2 impeller stopped	37.0 ± 1.0	30.0
SRV cycling ended	128.0 ± 0.3	50
SRV cycling reinitiated	328.0 ± 0.5	397
SGS liquid level reached bottom of indicated range (0.25 m above tube sheet)	458.0 ± 2.0	450
SRV cycling ended	580 ± 0.5	663
Primary-coolant pump 1 impeller stopped	732.0 ± 1.0	595.0
Reactor scrammed (experiment terminated)	1507.0 ± 0.5	1500.8

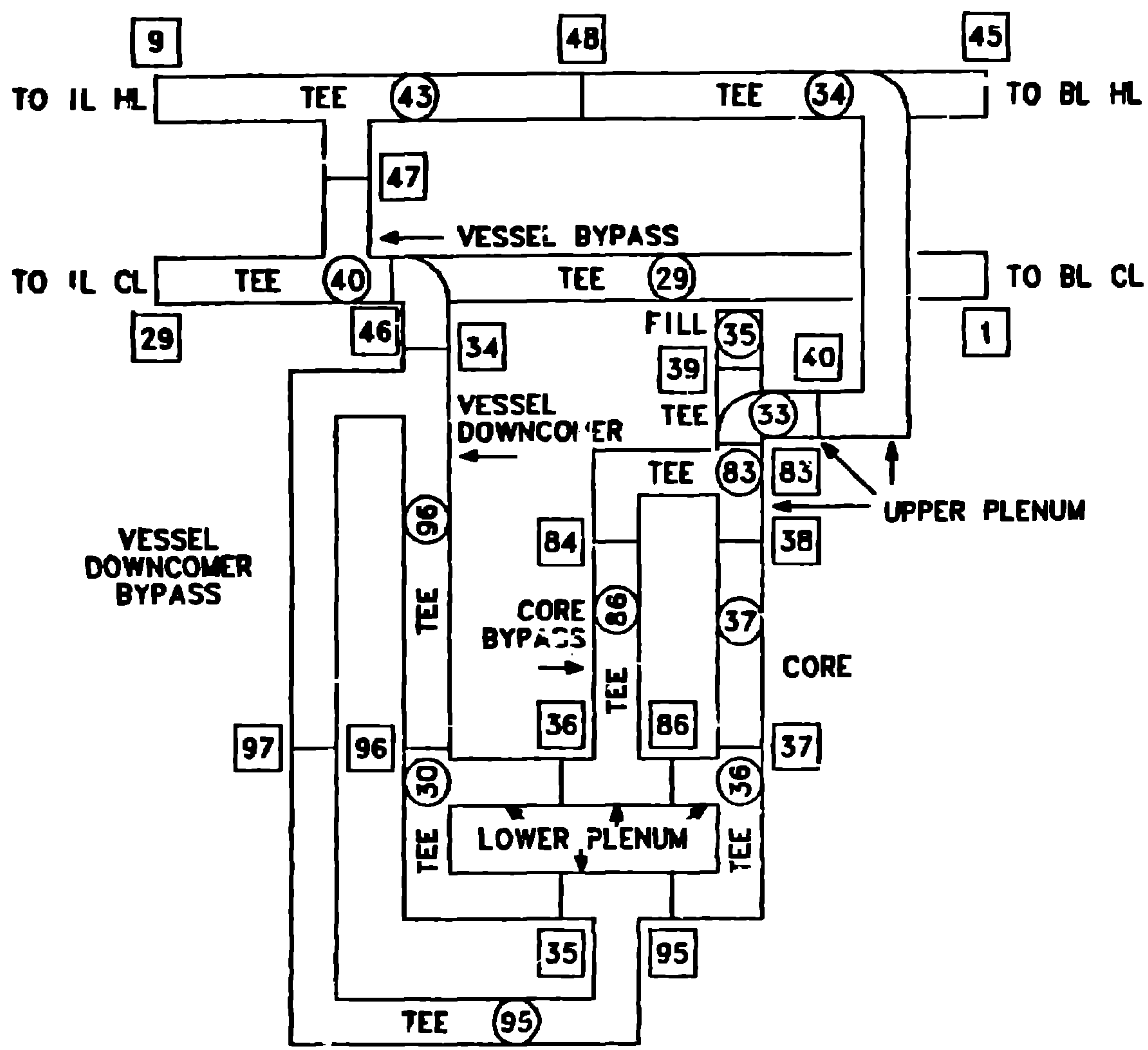
LIST OF FIGURES



Intact loop
Broken loop
Experiment PORV and SRV
Pressurizer
Steam generator
PC-2 experimental measurement station
BL-1 experimental measurement station
Steam generator simulator
Quick opening valve (2)
Isolation valve (2)
Blind Flange
Pump simulator
BL-2 experimental measurement station
ECC cold-leg injection location
PC-1 experimental measurement station
Pumps
PC-3 experimental measurement station
Reactor vessel
suppression tank
Downcomer
Core
Lower plenum
Reactor vessel



BREAK
62
63
SRI VALVE 63
62
STEAM GENERATOR
SECONDARY SIDE DETAILS SHOWN SEPARATELY
2
TEE
9
68
VALVE
10
60 PRESSURIZER
13
PRIZER
67
61
64
PIPE
12
TO BL CL
78
74
74
10
PRIZER
8
9
3
2
TEE
TEE 31
32
77
45
77
VALVE (ALWAYS CLOSED)
TEE
77
TEE 1
43
TO BL HL
73
73
4
6
66
TEE
11
8
TEE
7
29
1
VALVE (ALWAYS OPEN)
PIPE
78
TO IL HL
TEE
3
4 PUMP
6
TEE
5 PUMP
TEE
41
44
75
42
75
14
5
7
FILL
17
HPIS
VESSEL DETAILS SHOWN SEPARATELY
COMPONENT NUMBERS
JUNCTION NUMBERS
INTACT LOOP
BROKEN LOOP


9
48
45
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43
TEE
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TO BL HL
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VESSEL BYPASS
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TO BL CL
29
46
34
FILL
35
39
40
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VESSEL
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TEE
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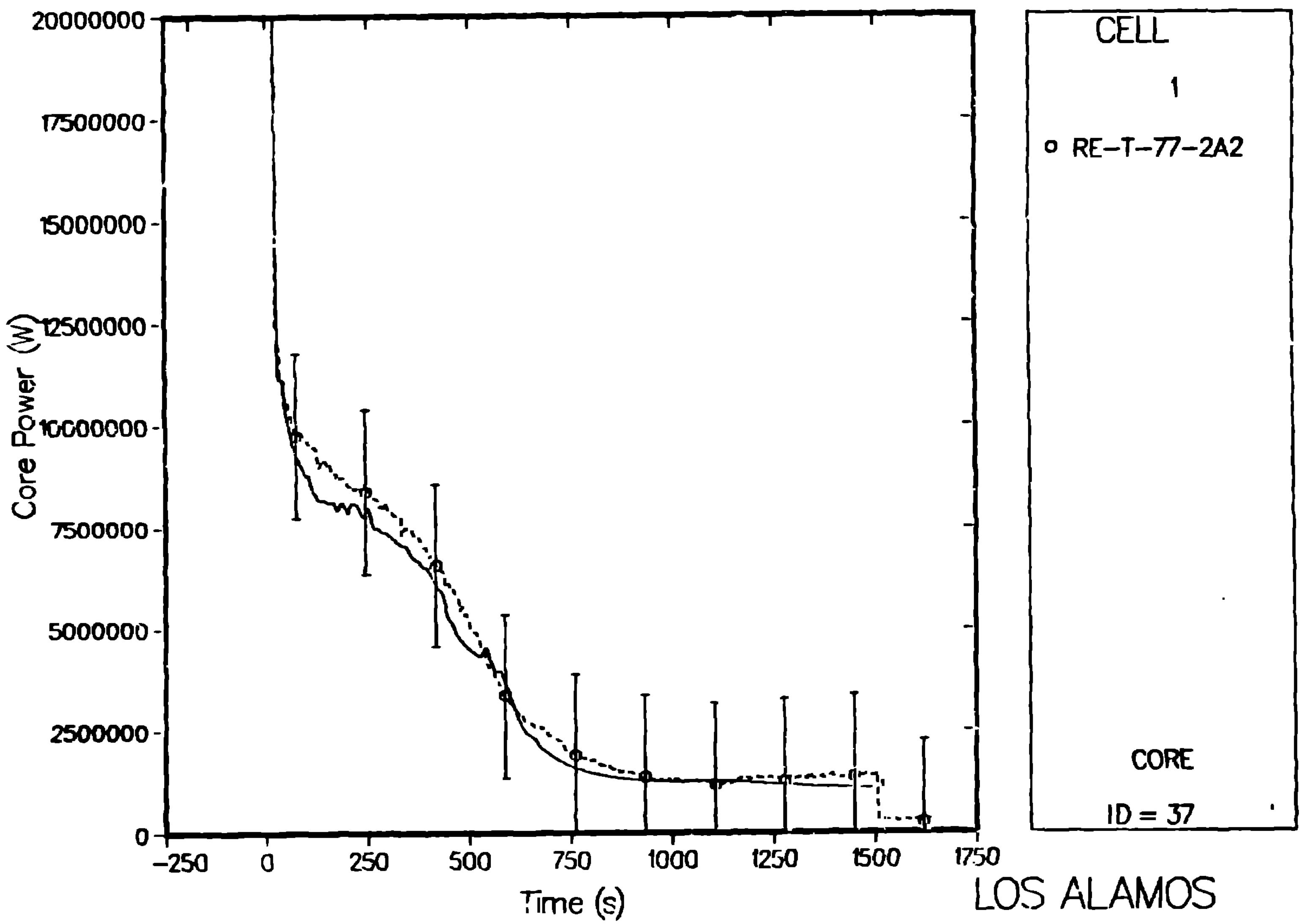
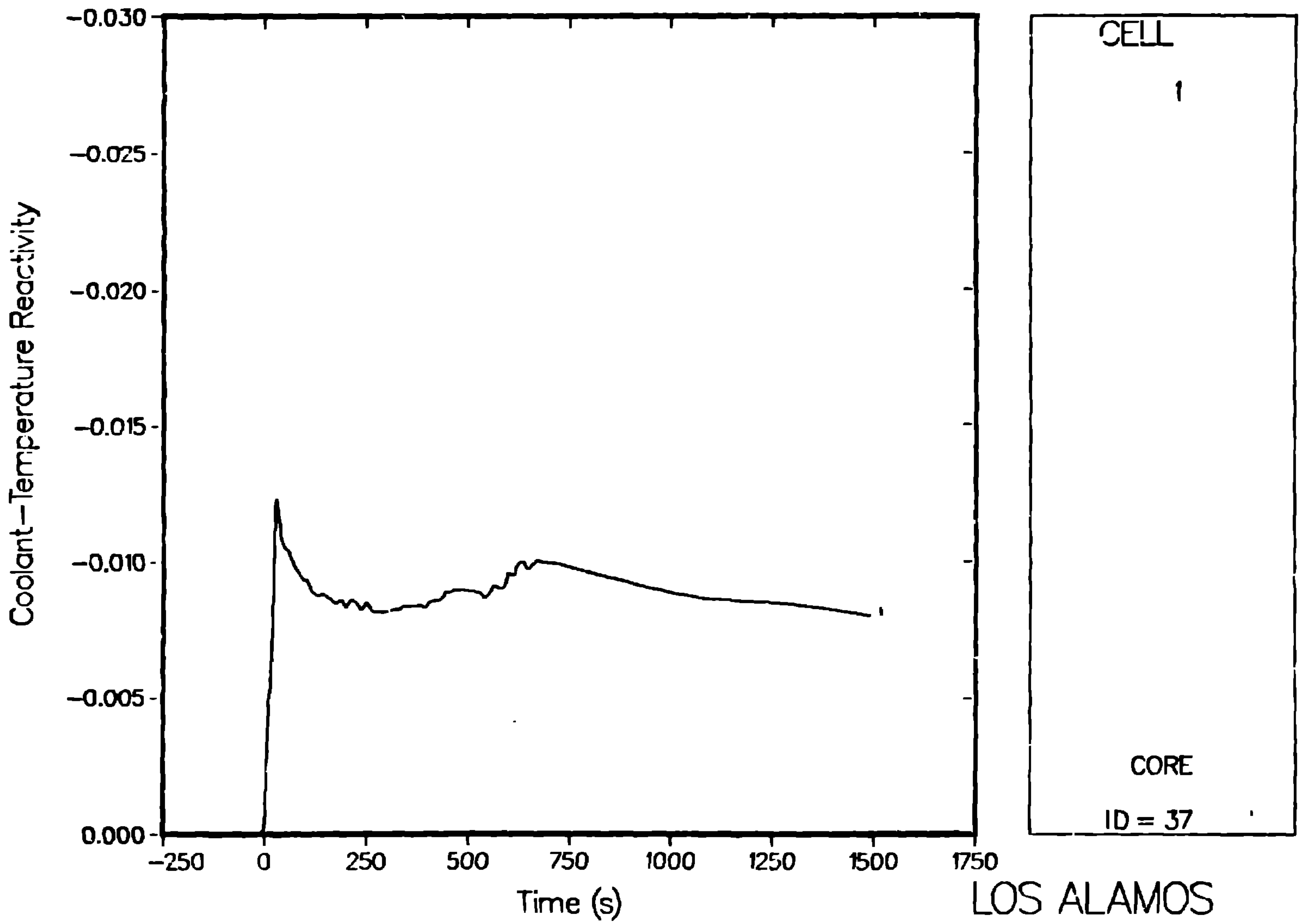


CELL
1
RE-T-77-2A2
CORE
ID = 37
Core Power (W)
20000000
17500000
15000000
12500000
10000000
7500000
5000000
2500000
0
-250
0
250
500
750
1000
1250
1500
1750
Time (s)

LOS ALAMOS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-0.025
-0.020
-0.015
-0.010
-0.005
0.000
Coolant-Temperature Reactivity
-250
0
250
500
750
1000
1250
1500
1750
Time (s)
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CORE
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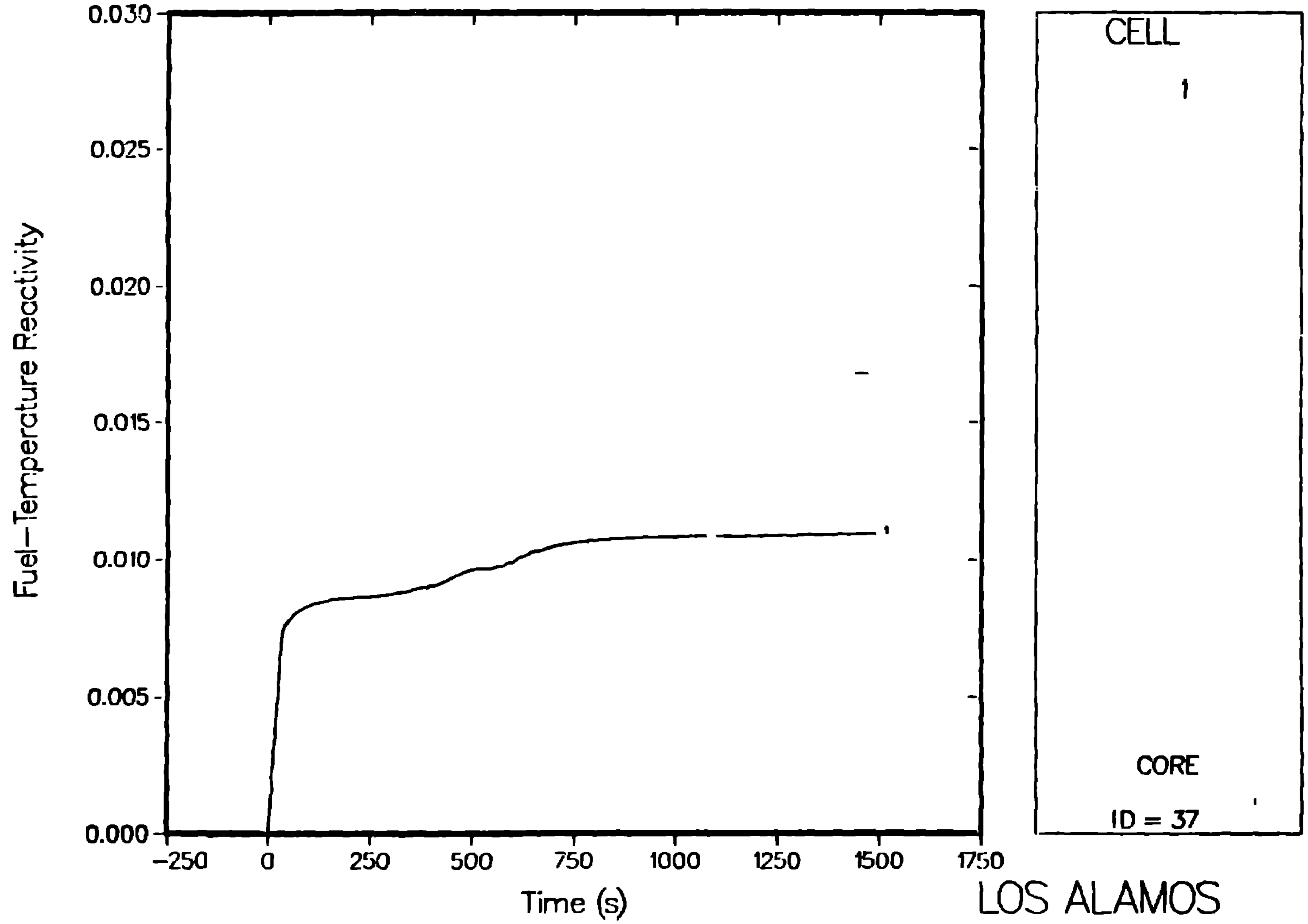
LOS ALAMOS



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0.025
0.020
0.015
0.010
0.005
0.000
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500
750
1000
1250
1500
1750
Time (s)
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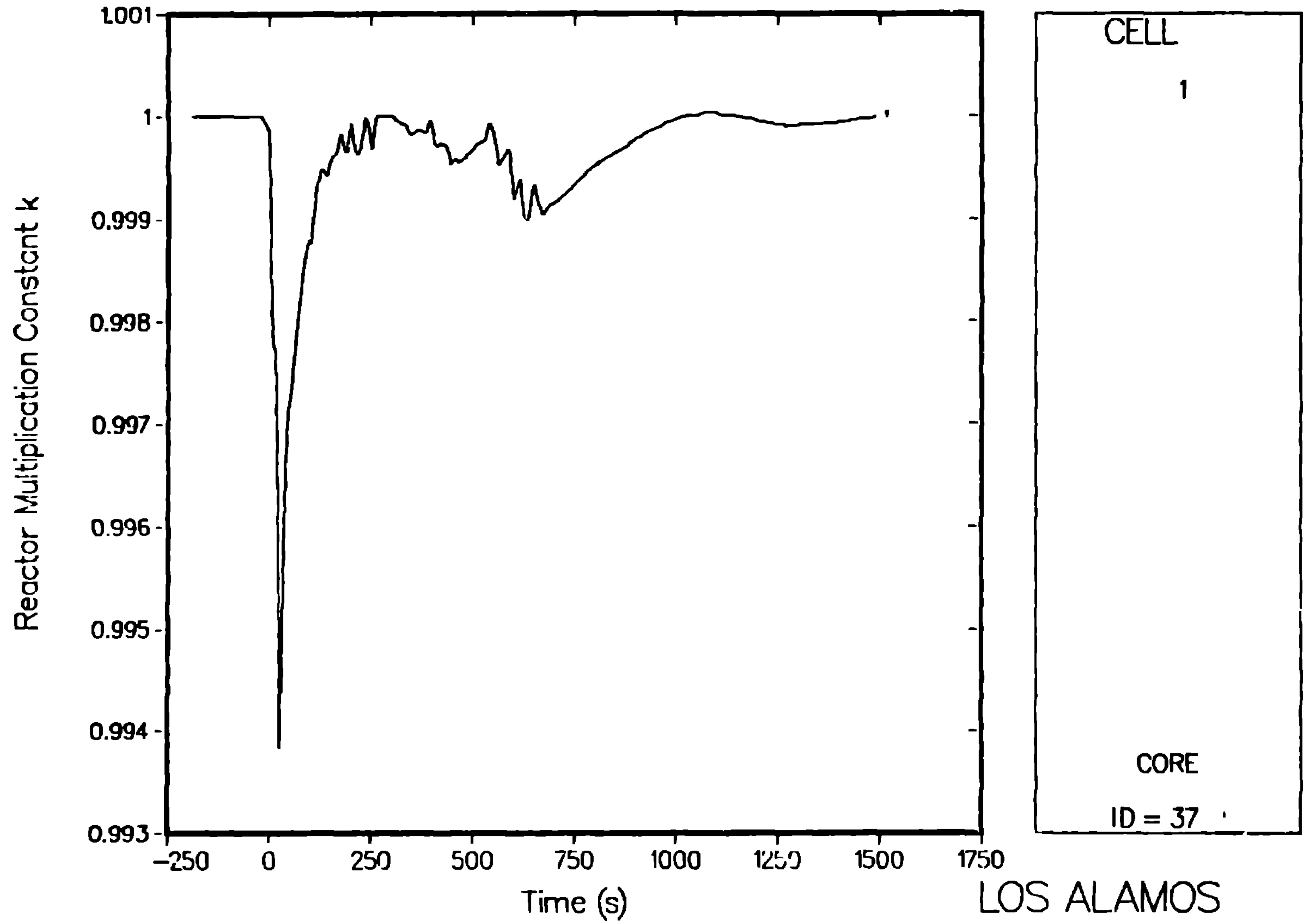
LOS ALAMOS



1.001
1
0.999
0.998
0.997
0.996
0.995
0.994
0.993
Reactor Multiplication Constant k
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0
250
500
750
1000
1250
1500
1750
Time (s)
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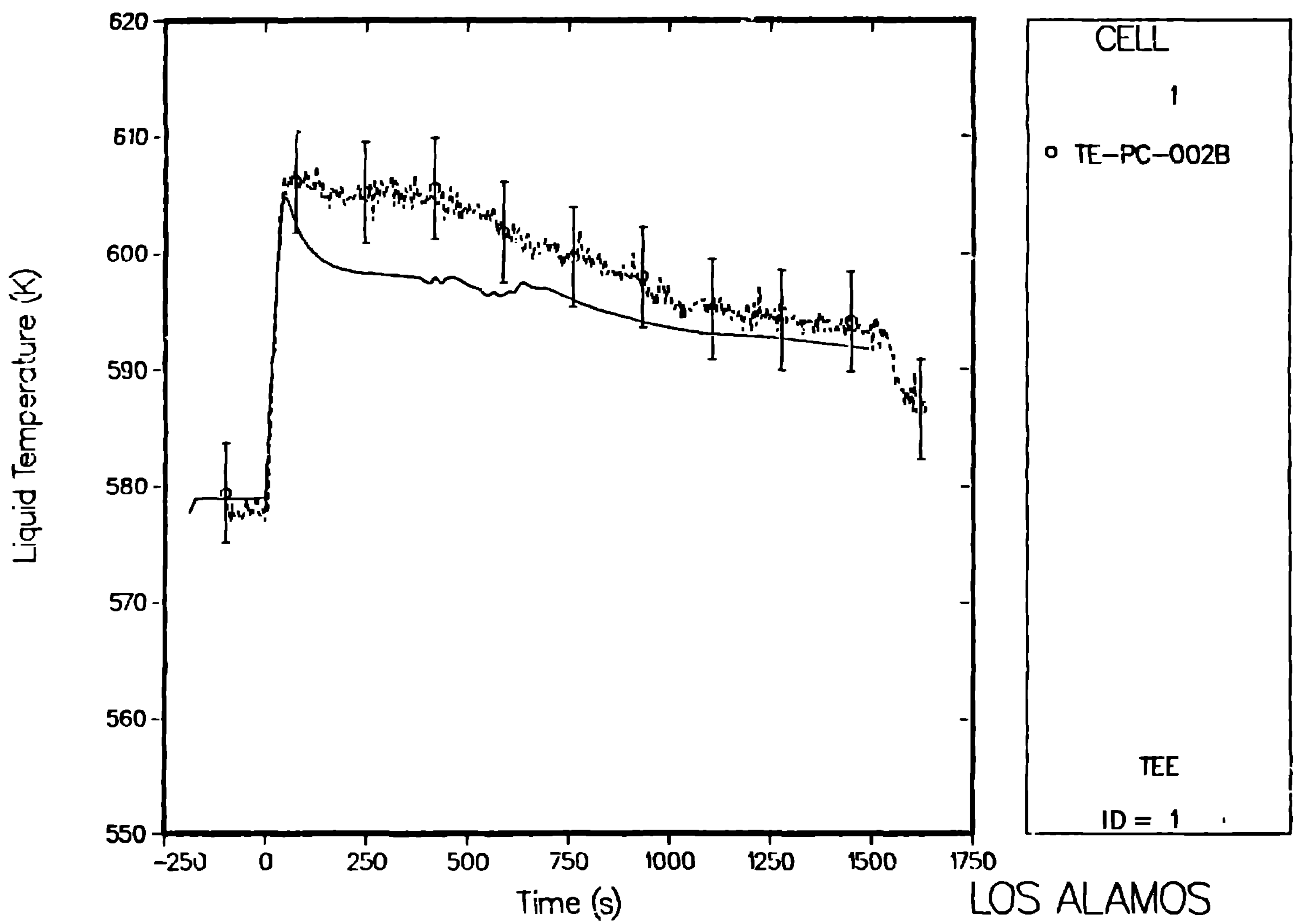
LOS ALAMOS



Liquid Temperature (K)
620
610
600
590
580
570
560
550
-250
0
250
500
750
1000
1250
1500
1750
Time (s)
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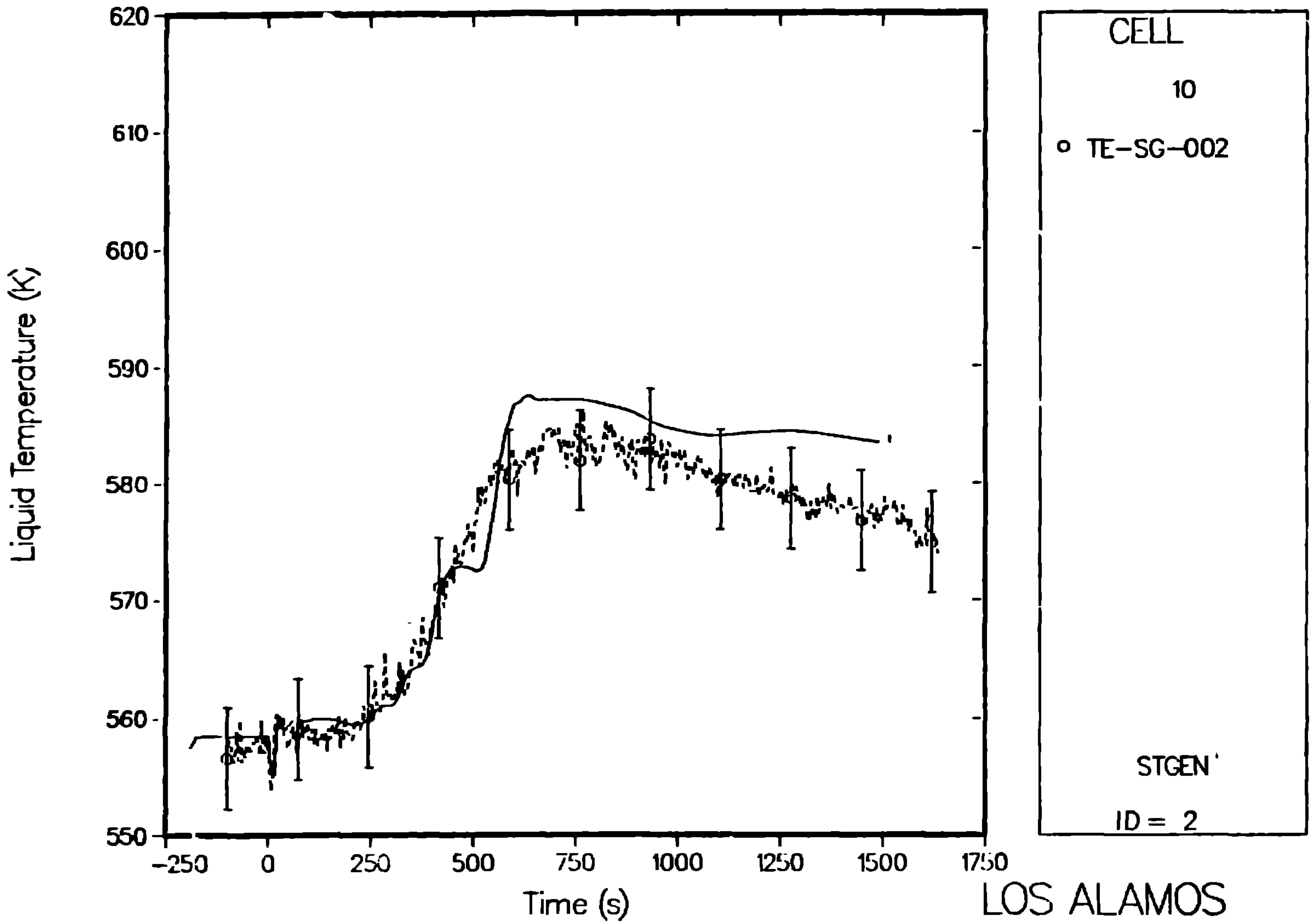
LOS ALAMOS



Liquid Temperature (K)
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610
600
590
580
570
560
550
-250
0
250
500
750
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1250
1500
1750
Time (s)
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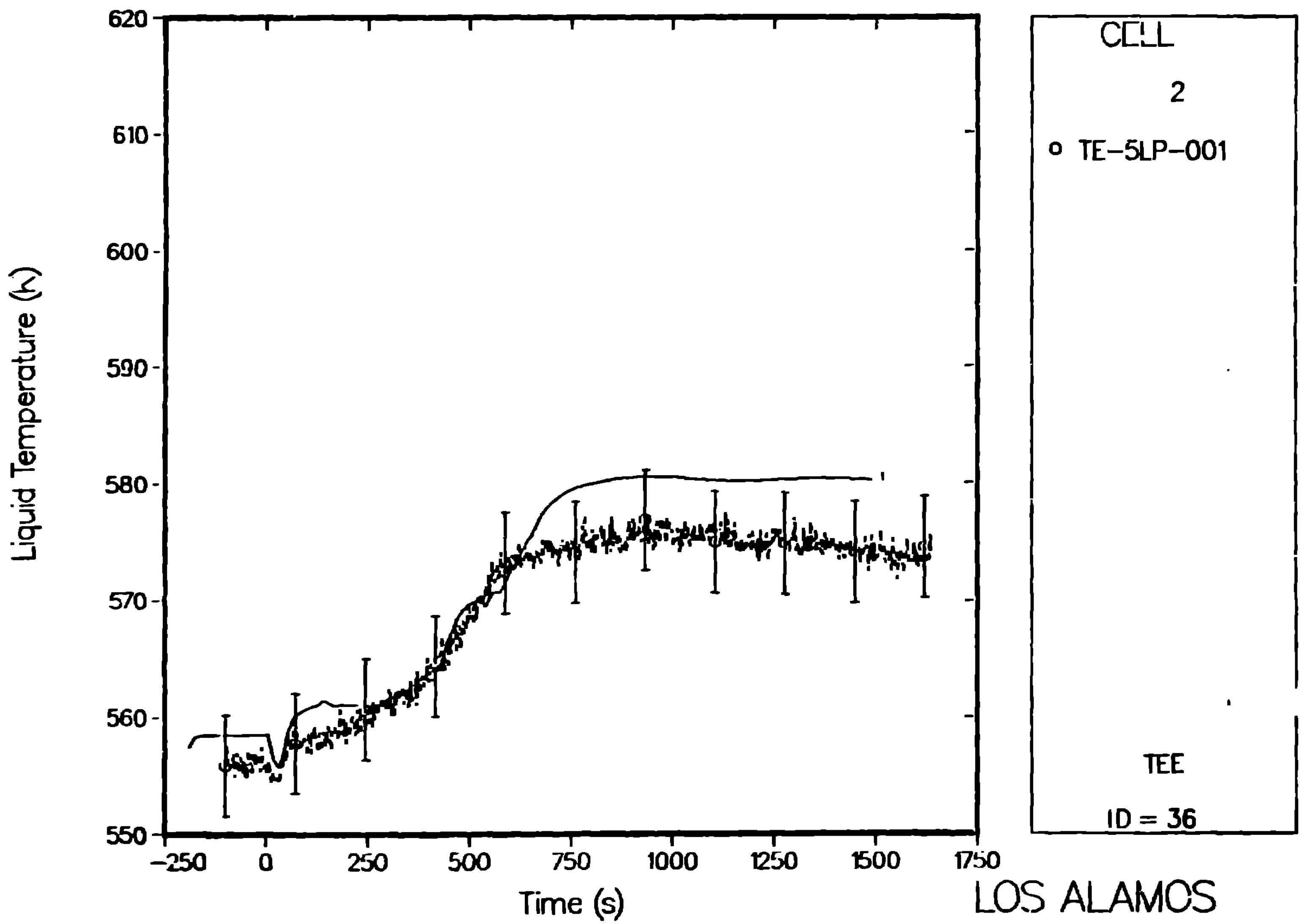
LOS ALAMOS



620
610
600
590
580
570
560
550
Liquid Temperature (K)
−250
0
250
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750
1000
1250
1500
1750
Time (s)
CELL
2
o TE−5LP−001
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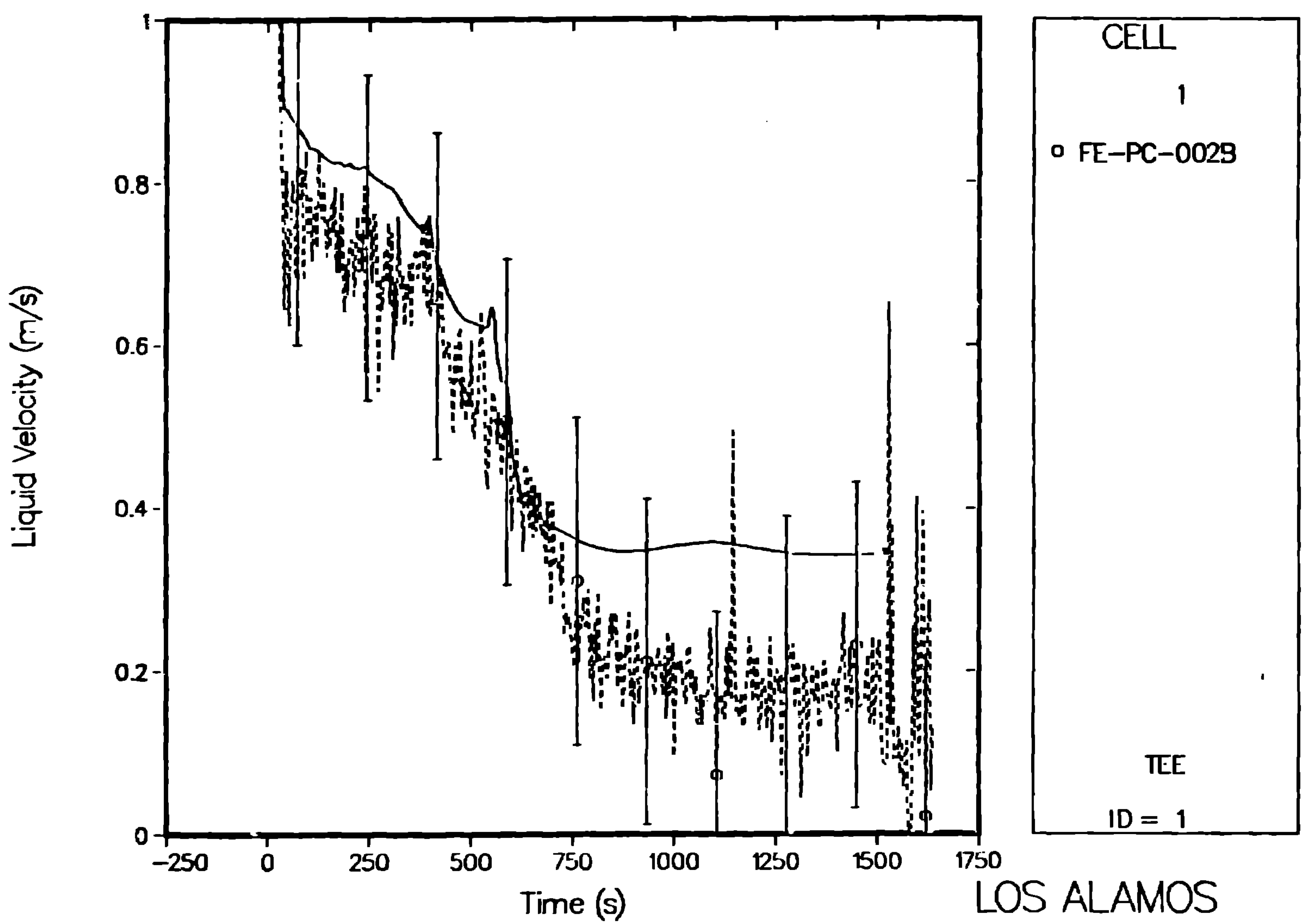
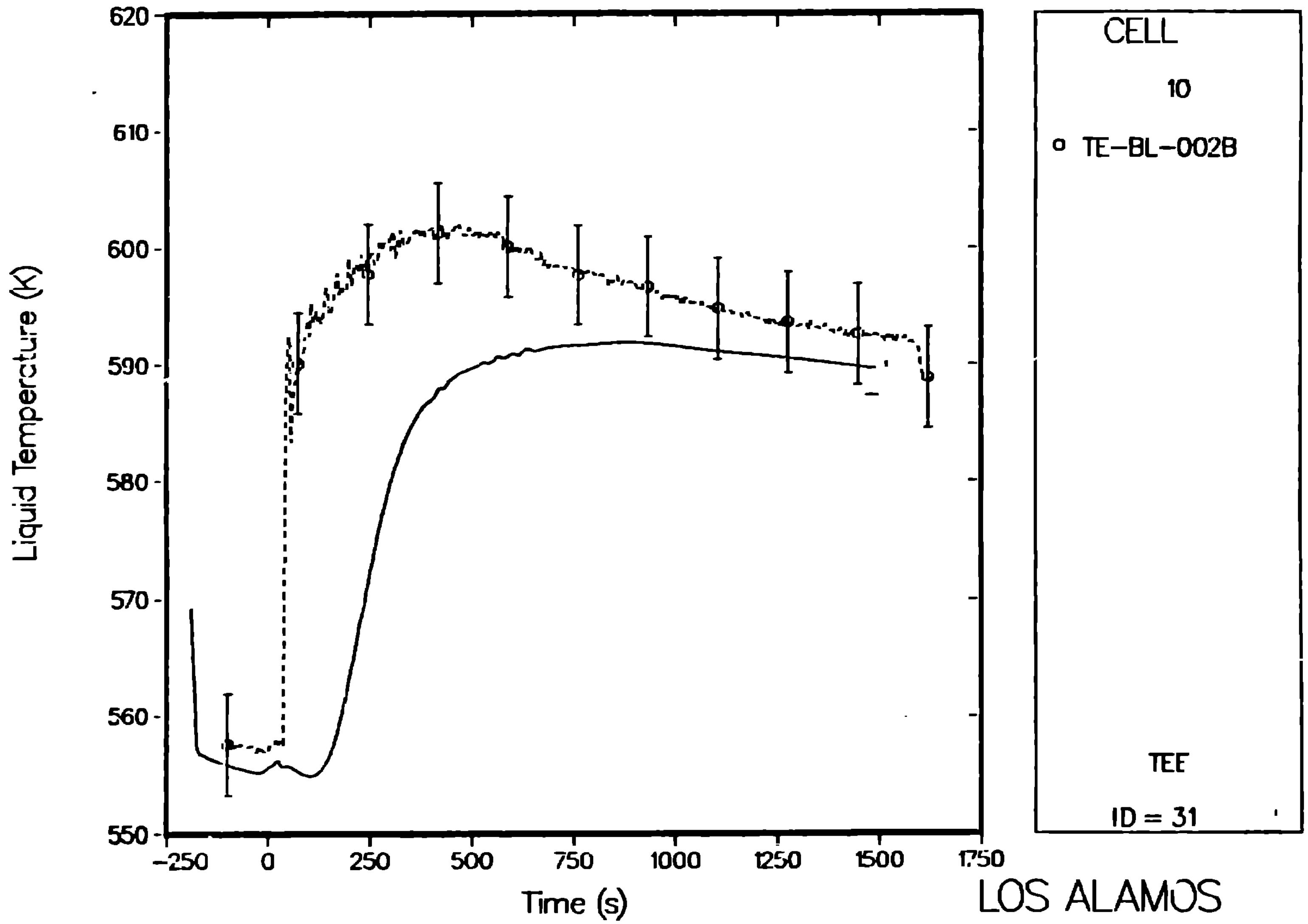
ID = 36
LOS ALAMOS



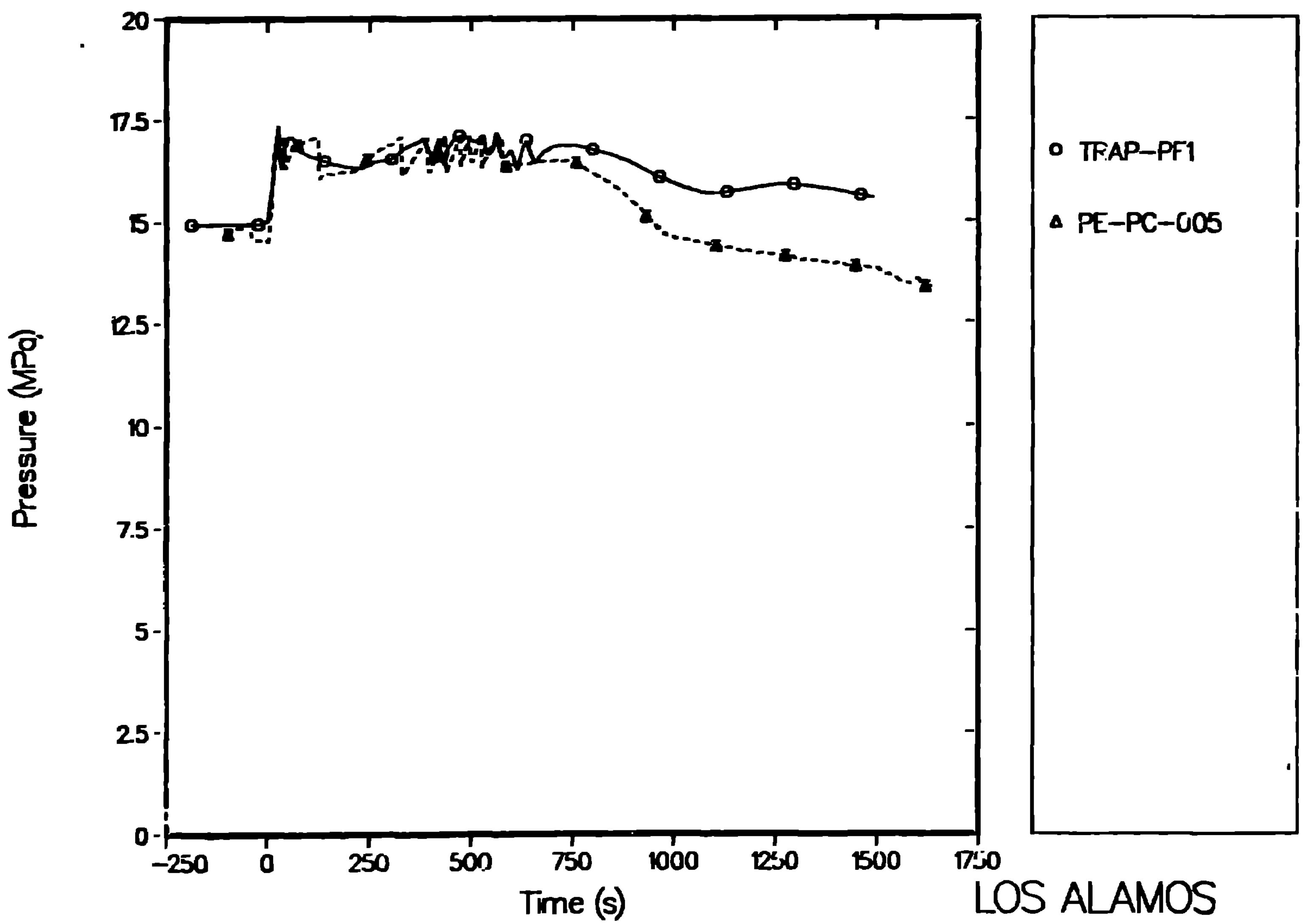
Liquid Velocity (m/s)
0
0.2
0.4
0.6
0.8
1
-250
0
250
500
750
1000
1250
1500
1750
Time (s)
CELL
1
FE-PC-002B
TEE
ID = 1

LOS ALAMOS



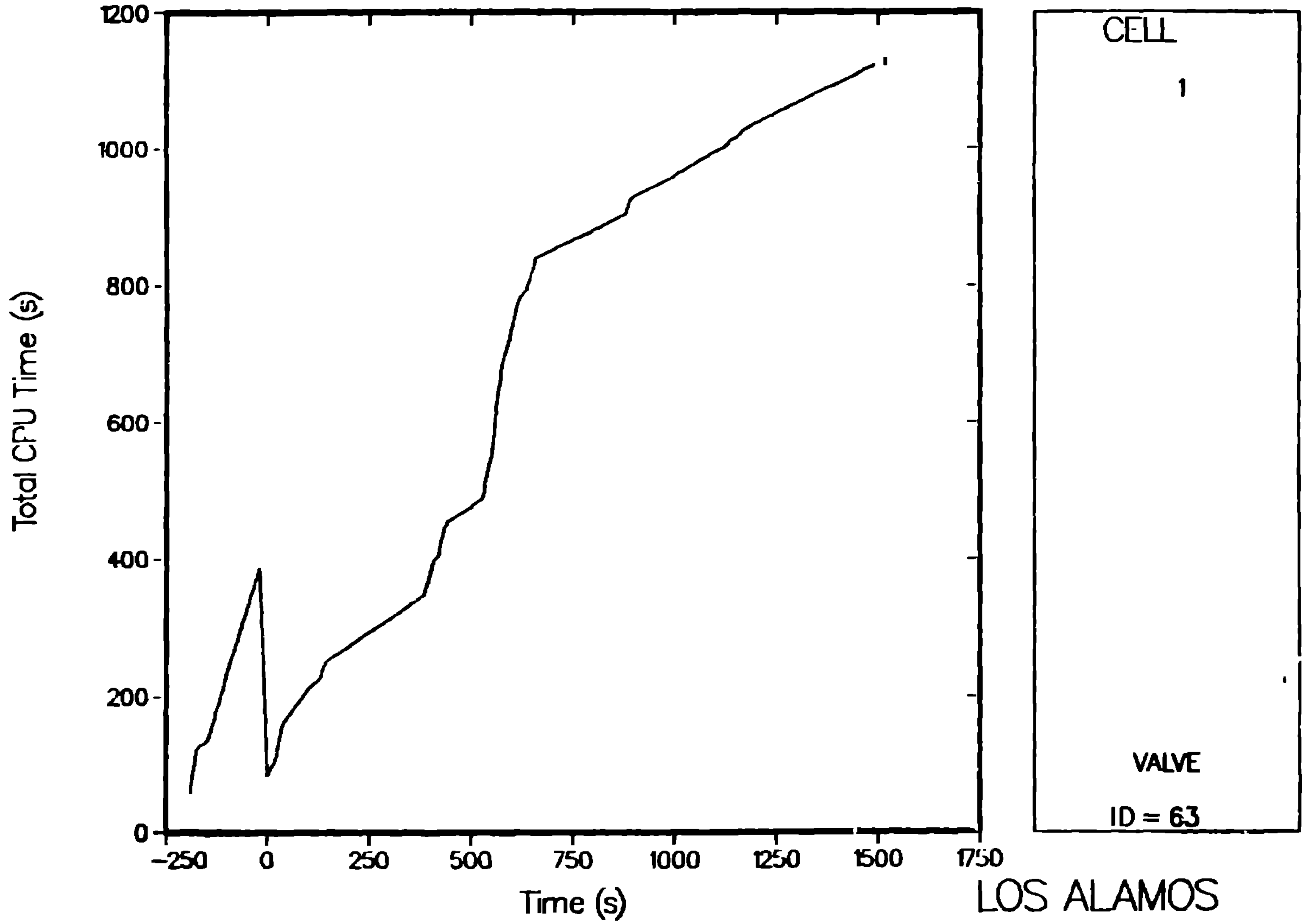
620
610
600
590
580
570
560
550
Liquid Temperature (K)
-250
0
250
500
750
1000
1250
1500
1750
Time (s)
CELL
10
TE-BL-002B
TEE
ID = 31

LOS ALAMOS



Pressure (MPa)
20
17.5
15
12.5
10
7.5
5
2.5
0
-250
0
250
500
750
1000
1250
1500
1750
Time (s)
TRAP-PF1
PE-PC-005

LOS ALAMOS



1200
1000
800
600
400
200
0
Total CPU Time (s)
−250
0
250
500
750
1000
1250
1500
1750
Time (s)
CELL
1
VALVE
ID = 63

LOS ALAMOS