NUREG/IA-0066
GD/PE-N/721

International Agreement Report

RELAP5/MOD2 Analysis of LOFT Experiment L9-4

Prepared by
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Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

April 1992

Prepared as part of
The Agreement on Research Participation and Technical Exchange
under the International Thermal-Hydraulic Code Assessment
and Application Program (ICAP)

Published by
U.S. Nuclear Regulatory Commission

9206150248 920430
PDR NUREG
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CENTRAL ELECTRICITY GENERATING BOARD
GENERATION DEVELOPMENT AND CONSTRUCTION DIVISION
PLANT ENGINEERING DEPARTMENT
NUCLEAR PLANT BRANCH

Title : RELAP5/MOD2 analysis of LOFT Experiment L9-4

Author : M.B.Keevill - Safety Technology Section

Approved by :

Head of Safety Technology Section

Summary : As part of a programme to validate RELAP5/MOD2 for use in the analysis of certain fault transients in the Sizewell B PWR, the code has been used to simulate experiment L9-4 carried out in the Loss-Of-Fluid Test (LOFT) facility. Experiment L9-4 simulated a Loss-Of-Offsite-Power Anticipated Transient Without Trip (LOOP ATWT) in which power is lost to the primary coolant pumps and main feed is lost to the steam generators but the control rods fail to insert in the reactor core.

RELAP5/MOD2 generally predicted the transient well, although there were some differences compared to the test data. These differences are largely due to the use of power and flow as boundary conditions and because of uncertainties in the power and flow experimental data. The most noticeable difference was that the steam generator was predicted to boil down too fast. This is believed to be partly due to errors in the RELAP5 interphase drag model. The RELAP5 calculation also showed the primary pressure to be very sensitive to the primary flow rate, making the exact simulation of primary side relief valve movements difficult to reproduce.

Date : December 1988

1. Introduction

The thermalhydraulics computer code RELAP5/MOD2 is being used by GDCD for calculation of certain small break loss-of-coolant accidents and pressurised fault transients in the Sizewell 'B' PWR. To help validate RELAP5/MOD2 for this application, the code has been used to simulate experiment L9-4 carried out in the Loss-Of-Fluid Test (LOFT) facility. Experiment L9-4 simulated a Loss-Of-Offsite-Power Anticipated Transient Without Trip (LOOP ATWT) in which power is lost to the primary coolant pumps and main feed is lost to the steam generators but the control rods fail to insert in the reactor core.

This report describes the RELAP5/MOD2 analysis of L9-4 in detail.

2. The Loss-of-Fluid Test (LOFT) facility

The LOFT integral test facility is a 50MW Pressurised Water Reactor (PWR) (Fig.1) designed to simulate the system responses of a commercial PWR during a loss-of-coolant accident and transients. Instrumentation is provided to measure detailed thermalhydraulic and nuclear conditions throughout the system in a transient. The LOFT facility consists of:-

1. A reactor vessel with a nuclear core.
2. An intact loop with an active steam generator, pressuriser and two primary coolant pumps connected in parallel.
3. A "broken" loop with a simulated pump, simulated steam generator and two quick-opening blowdown valve assemblies (the pump and steam generator simulators were disconnected for test L9-4).
4. A blowdown suppression system consisting of a header, suppression tank and a spray system.
5. An emergency core coolant (ECC) injection system consisting of two low-pressure injection system (LPIS) pumps, two high pressure injection system (HPIS) pumps and two accumulators.
6. A pressure relief line from the top of the pressuriser to the blowdown suppression tank containing the experiment power-operated relief valve (PORV) and safety relief valve (SRV) in parallel with the pressure relief line containing the plant PORV and SRVs. (The experiment PORV was inoperative in test L9-4 due to the assumed loss of offsite power.)

3. Description of test L9-4

Experiment L9-4, which was performed on September 24 1982, simulated a loss-of-offsite-power accident without reactor trip. The test is described in detail in Ref.1. A brief description is given below.

The experiment was initiated from typical commercial PWR operating conditions by tripping the primary coolant pumps and main feedwater pump, and by closing the steam generator main steam control valve. During the ensuing transient, the following operating conditions applied:-

- the pressuriser PORV was inoperative. This simulated behaviour in a loss of offsite power event in a full-size plant, when the PORV cannot operate due to loss of instrument air.

- the pressuriser spray was inoperative due to the primary coolant pump trip.
- auxiliary feedwater was initiated 10 seconds after the start of the transient. The delay simulated the start up time of diesel generators used for emergency power generation in commercial PWRs.
- steam generator secondary side pressure was manually controlled by steam bleed through the main steam bypass valve. This simulated the action of the steam generator safety relief valves in the commercial plant.

The sequence of events in the experiment is described briefly as follows:-

After the primary coolant pumps tripped, the decreasing primary flow resulted in a rapid core coolant temperature rise. This initial heatup caused the reactor power to decrease rapidly, principally due to the effect of moderator density feedback. Volumetric expansion of the primary coolant during the heatup caused the primary pressure to increase to the SRV setpoint by 18.5 seconds following which the SRV cycled four times to control the primary pressure. The transition from forced flow to natural circulation flow began during the pump coastdown and natural circulation was fully established by approximately 80 s after experiment initiation. By about 200s sufficient water had boiled off from the steam generator to significantly degrade heat transfer such that primary pressure again started to increase towards the SRV setpoint, causing further SRV cycling. By the fifth subsequent SRV cycle at 500s, the pressuriser had filled completely so that discharge through the SRV consisted entirely of single-phase liquid. The SRV cycled a further two times, after which the steam generator heat transfer and environmental heat losses were sufficient to remove the reduced core power and no further SRV cycling took place. At approximately 1000s primary pressure control was regained as a steam bubble was reformed in the pressuriser and the primary side depressurisation rate decreased. The core heat generation was now sufficiently small that it could be dissipated by the auxiliary feedwater flow into the steam generator. The experiment was terminated at 1507s by reactor trip.

4. RELAP5/MOD2 model of LOFT facility

The code version used for the analysis of experiment L9-4 was RELAP5/MOD2 Cycle 36.05 UK Version E03.

The input model was based on that used previously by GDCD for analysis of LOFT loss-of-feed fault experiment LP-FW-01 (Ref.3). A noding diagram is given in Fig.2. The following changes were made to the input deck used for the Ref.3 calculations:-

1. The pressuriser spray was disabled.
2. The pressuriser SRV was modelled as a trip valve so that it was either fully open or fully closed depending on the pressuriser pressure in relation to the setpoints. The SRV flow area was initially obtained from a separate RELAP5 calculation since the actual area of the experiment SRV was unknown. This calculation modelled only the SRV with inlet and outlet plena. The valve area was adjusted until the required flow rate was obtained, using the calibration data given in Ref.1. After preliminary RELAP5 transient calculations, some further changes were necessary to the modelling of the SRV. The area was adjusted slightly and the control logic was modified to ensure that the valve opened or closed fully when a pressure setpoint was violated. These changes were necessary to obtain the correct depressurisation rate when the valve was open.
3. Nodes representing lengths of pipe on the broken loop cold leg, which were blanked off for test L9-4, were deleted.
4. A steam generator auxiliary feedwater system was added.

5. The control system and trip data were modified to represent the experimental setpoints for L9-4. Ref.1 states that during the experiment the steam bypass valve was adjusted manually to control the steam generator pressure between specified limits. No further information describing the valve characteristics was provided so the bypass valve was modelled as a motor valve, with a valve change rate as given in the source deck. Refs 1.& 2. gave conflicting values for pressure limits within which the steam generator was controlled. Examination of the experimental data led to the values given in Ref.2. being adopted ie. 6.63 and 6.97 MPa.

6. Ref.2 states that the LOFT moderator reactivity feedback was typical of PWR end-of-life conditions. This implies a large, negative moderator temperature coefficient. However, there was no reactor physics data available for the LOFT facility nuclear core under end-of-life conditions so it was not possible to simulate reactivity feedback using the RELAP5/MOD2 point kinetics model. Because of this, the reactor power during the experiment was input as a table of power versus time. The data for the input table was obtained initially from the LOFT L9-4 experimental data tape which gave the core power as measured by the ex-core detectors. To this data was added the contribution from decay heat which was calculated from the known power history, as described in Appendix 1, using the ANS 1979 standard. The corresponding power curve is shown in Fig.3.

7. The primary coolant pump coastdown was modelled using an input table of pump velocity versus time, with data again obtained from the experimental data tape. The corresponding pump velocity curve is shown in Fig.4. This was done because preliminary RELAP5 transient calculations, with pump behaviour determined by homologous pump curves, showed the pumps to be coasting down too quickly after trip. The design of the pump drive system is unusual since the pump motor is electrically connected to a generator/flywheel which is, in turn, driven through a fluid coupling by a motor (the prime mover). This gives rise to uncertainty over the inertia of the system during coastdown and this is believed to be the reason for the fast coastdown calculated by RELAP5. Note that in the experiment pump 2 stopped at 37s but pump 1 continued to rotate, driven by natural circulation flow, before finally stopping at 732s. In the RELAP5 pump velocity table, both pumps were set to stop at 37s in order to give slightly improved agreement with the measured loop flow.

8. The steam generator separator fall-back junction loss coefficient was adjusted to prevent premature steam carry-under during the early part of the transient.

5. RELAP5/MOD2 Calculations

5.1 Initial conditions.

Prior to performing the transient calculation, a steady state calculation was performed in which the RELAP5 control logic was used to adjust the pump speed, feed and steam flows to obtain the correct initial values for primary mass flow, secondary pressure, steam generator level and cold leg temperature. In addition, a null transient calculation was performed to confirm that the steady state was fully converged. The initial conditions obtained at the end of the null transient are compared with the experiment initial conditions in Table 1.Agreement is seen to be satisfactory.

5.2 Transient calculations

5.2.1 Preliminary calculations

The first attempts at a transient calculation were unsuccessful, with some calculations ending in code failure. The main difficulty was that the calculation showed unusual 'spikes' in the calculated primary pressure. Reducing the maximum calculational timestep to 0.01s gave a slight improvement but spikes still remained. The pressure spikes were eventually found to be due to the code unexpectedly predicting dryout in one of the core nodes. Investigation revealed that one of the Biasi critical heat flux correlations (Biasi B) used by RELAP5 was being applied at pressures outside its range of validity (140 bar). At pressures above 162.5 bar, the correlation actually gave negative CHF predictions and this had occurred in the calculation of L9-4 when the correlation was selected by RELAP5, as the mass flux fell below a threshold value ($300 kg/m^2s$). This can be seen in Fig.5 which plots the CHF predicted by both Biasi A & B correlations for a mass flux of $300 kg/m^2s$.

A modified code version was produced by AEEW which linearized the calculated CHF from the value calculated at 140 bar down to zero at the critical pressure of 221 bar. This was found to remove the problem and subsequent calculations were performed using this code version. It should be noted however that a large discontinuity remains when the code switches between Biasi A and Biasi B correlations. Fig.6 compares the CHF given by the modified Biasi B correlation with that given by the BW-2 correlation (Ref.4). It is seen that the two models are in good agreement at pressures above 70 bar.

The preliminary calculations were also found to be sensitive to the calculational timestep. This sensitivity to timestep size was found to be due to numerical oscillations in the steam generator separator during the first few seconds of the transient. Some adjustment was made to the resistance of the liquid fall-back junction to remove these oscillations. When this was done the sensitivity to timestep was no longer present.

Having identified and resolved these various problems, a final RELAP5 calculation was performed. This is described below.

5.3 Comparison of final calculation with experimental results

The calculated primary and secondary pressures during the transient are shown in Fig.7,8 & 9 together with the corresponding experimental results. It can be seen that RELAP5 predicts the primary pressure well in the first 40s of the transient, although one extra SRV cycle is predicted due to a slight discrepancy in the primary to secondary heat transfer. The SRV cycle at 125s is not however predicted, indicating that too much heat is being taken out of the primary circuit.

Between 100 and 220 s the primary pressure stays fairly constant whilst steam generator boil down continues. The calculated steam generator liquid level is compared with the measured level in Fig.10. It can be seen that the calculated level falls more rapidly than in the experiment. (Note that two scales are used in Fig.10 to be consistent with LOFT terminology, where a level of 0.0m actually corresponds to a level 2.95m above the tubesheet.) The measured steam generator level suggests that in the experiment steam generator dryout occurred at about 600s, which is 230s later than predicted by RELAP5. (Note that after 500s the measured steam generator level appears to remain at 0.25m. This is a false reading because the lower pressure tapping for the level indicator is 0.25m above the tube sheet, so levels below this value are outside the measurement range).

From about 220s the calculation predicts that the primary heatup resumes again causing the onset of a second phase of SRV cycling at 296s. In the experiment the heatup resumed slightly earlier at approximately 200s with SRV cycling commencing at 330s, which suggests a more gradual heatup than predicted by RELAP5. The reason for this discrepancy is probably difference between the predicted and actual rates at which the heat sink is lost. Figs.11 & 12 compare the measured and predicted hot and cold leg temperatures. The over-rapid increase in calculated cold leg temperature shown in Fig.12 is again attributable to the calculation of too rapid a loss of heat sink between 300 and 600 s. The result of the above is that RELAP5 predicts a primary heat-up which, although starting later, is more rapid than in the experiment and larger in magnitude, so that SRV cycling recommences earlier in the calculation.

In the second phase of SRV cycling, the final SRV cycle is calculated to occur at 640s which compares quite well with 575s in the experiment, although RELAP5 predicted 25 SRV cycles during this phase compared to the 7 cycles which actually took place. This is again presumed due to discrepancy in the calculated primary to secondary heat transfer. Also, during this period, the calculation shows the secondary side pressure to be cycling between the high and low pressure setpoints. This is probably because the RELAP5 model of the steam bypass valve control is too coarse.

After 700s the core power is balanced by auxiliary feedwater to the SG boiling off, together with the steam generator environmental heat losses, and the primary pressure begins to fall. RELAP5 predicts this primary cooldown well. From about 700s onwards, the power level remains essentially constant at around 1.5MW, this level being determined, via reactivity feedbacks, principally by the heat removal capability of the supplied auxiliary feedwater. The calculation still shows the secondary side pressure to be oscillating between the setpoints during the primary cooldown, again because the control of the steam bypass valve is too coarse.

Pressuriser liquid levels for the calculation and the experiment are shown in Fig.13. Apart from the slightly different initial liquid levels, the initial pressuriser insurge during the heatup is well predicted. In the test, the pressuriser liquid level reached the top of the measuring range (1.8m) after 437s and did not fall back into the measuring range until 1125s. In the calculation the pressuriser filled at 375s, after which single-phase liquid was discharged through the SRV, until approximately 1200s when, because of the more rapid cooldown, volumetric contraction of the primary fluid caused a steam bubble to reform in the top of the pressuriser. The void fraction at the SRV is shown in Fig.14.

5.4 Computing time

The RELAP5 calculations were performed on the Harwell CRAY2 computer. The 1500s transient took 2578 s of CPU time to calculate. The maximum time step size was 0.05s for the first 700s of the transient and 0.1s for the remainder.

6. Discussion

6.1 General

In general RELAP5/MOD2 gave a reasonable prediction of the experiment, bearing in mind the lack of information available to describe some aspects of the experiment. The first 40 s of the transient containing the initial heatup and first phase of SRV cycling are well predicted. It is during this phase of an ATWT that the highest core powers occur and the primary pumps coastdown, giving most cause for concern regarding the possibility of Departure from Nucleate Boiling (DNB). The accuracy of this part of the calculation is therefore encouraging.

The remainder of the transient is less well predicted, although the main phenomena of natural circulation, secondary boil down, pressuriser insurge and outsurge and primary cooldown are still reasonably represented in terms of trends and timescales.

The most important factor governing the calculation of L9-4 is the primary to secondary heat transfer rate, particularly as the core power is input as a boundary condition with no reactivity feedback. In the early part of the transient prior to steam generator dryout, the steam generator heat transfer is dominated by the heat transfer coefficient on the primary side of the tubes, the heat transfer regime being convection to sub-cooled liquid under conditions of reducing primary flow. After the steam generator dries out, the primary to secondary heat transfer is dominated by the heat transfer coefficient on the secondary side of the tubes, the heat transfer regime being convection to single-phase superheated vapour. For both these regimes RELAP5 uses the Dittus-Boelter correlation to determine the heat transfer coefficient.

The most noticeable discrepancy between the calculation and the experiment is that RELAP5 calculates the steam generator to boil down too rapidly, giving rise to an early loss of heat sink. This then causes the rapid second phase of SRV cycling since too little heat is being removed from the primary side. The early loss of heat sink also keeps the primary coolant temperatures high for the remainder of the transient. Possible reasons for the error in the calculation were examined by performing sensitivity studies. These are described further below.

6.1 Sensitivity to initial steam generator inventory

In the base case calculation the initial steam generator secondary side inventory is 1946Kg with a steady state circulation ratio of 5.1. To determine whether the rapid boildown was caused by insufficient water in the steam generator secondary side, a sensitivity study was performed in which the circulation ratio was increased to 8 by adjusting junction loss coefficients in the steam generator secondary side. This had the effect of increasing the inventory by 8%. The resulting boildown is shown in Fig.15. It can be seen that although the level calculation is improved from 50 to 200s, significant differences still remain at the beginning and end of the boildown. It was also found that the revised inventory did not significantly improve the primary pressure calculation. It therefore seems that a very large circulation ratio would be needed to increase the initial steam generator inventory to the value needed to match the experimental boildown behaviour. Such an increase was not thought to be justifiable for the present calculation. It is possible that errors in the inventory calculation may have arisen because of errors in the void calculation in the riser region, resulting from errors in the RELAP5 interphase drag model. Systematic errors have previously been noted in the study in Ref.5.

6.2 Sensitivity to primary flow rate.

It has been stated earlier that a pump velocity versus time table was input to the calculation to specify the pump coastdown, using data from the experimental data tape, because of uncertainty over the pump inertia. It was found in initial calculations that after approximately 40s, the predicted primary pressure was considerably lower than that measured in the experiment (Fig.16). (Note that these are preliminary calculations which used a slightly different power curve.) This implied that heat removal from the primary circuit was too great. Examination of the fluid velocities in the hot and cold legs showed that, from 22s onwards, the calculated fluid velocity was too high, being of the order of 0.8 m/s compared with 0.6 m/s in the experiment (Fig.17). However, the uncertainty on the measured values is quoted in Ref.1 to be $\pm$ 0.46m/s. In the base case calculation the pump flow resistance was therefore increased, which had the effect of reducing the fluid velocity, although it was still within the measurement uncertainty band. This gave improved agreement between the calculated and measured fluid velocities (Fig.18). The primary pressure can be seen from Fig.16 to be considerably higher, with three SRV cycles now calculated within the first 50s. This demonstrates that the calculation is highly sensitive to the primary flow rate.

This sensitivity to primary flow arises because, as the primary flow rate falls, the primary to secondary heat transfer becomes dominated by the low heat transfer coefficient on the primary side of the steam generator tubes. This effect disappears after the steam generator dries out since the heat transfer coefficient on the secondary side of the tubes then becomes the dominant term. Use of the RELAP5 point kinetics model for the calculation might have reduced this sensitivity since thermalhydraulic feedback would have been introduced via the moderator temperature coefficient of reactivity.

6.3 Sensitivity to input power

The uncertainty in the core power measured by the ex-core flux detectors is quoted to be $\pm 2.0MW$ (Ref.1). The previous power history used to calculate the decay heat contribution was also measured using the ex-core detectors and is therefore subject to the same uncertainty. The uncertainty in the initial power level is quoted in Ref.1 to be $\pm 5.4MW$ and it is not clear how this power level was obtained. In view of these uncertainties it is likely that there is some inaccuracy in the input core power table and this may partly explain the rapid steam generator boildown.

6.4 Sensitivity to steam generator noding

The RELAP5 steam generator model consists of five nodes in the secondary side boiler region, four of which model the tube bundle. It can be seen from Fig.10 that the discrepancy between calculated and actual level increases with time, effectively in a series of steps. Each of these steps corresponds to dryout occurring in a RELAP5 node and this suggests that a better prediction of the level could perhaps be achieved by increasing the number of nodes at the bottom of the riser.

6.3 Shortage of experimental data

No reactor physics data was provided for end-of-life core conditions, necessitating the use of a power table in the present calculation. Since the core power in experiment L9-4 is controlled by reactivity feedback, the calculation would have been a good test of the RELAP5 point kinetics model and inclusion of point kinetics with feedback would also have rendered the calculation to be less sensitive to primary flow. The omission of the data from the Experimental Data report is therefore disappointing.

7. Conclusions

1. As part of a GDCD programme to validate RELAP5/MOD2 for future use, calculations have been performed to simulate LOFT experiment L9-4, a loss-of-offsite-power anticipated transient without trip in which the primary coolant pumps coastdown and main feed to the steam generators is lost.

2. The transient was generally well predicted by RELAP5/MOD2 particularly the initial primary heatup which is the phase of the transient where DNB is most likely to occur. The primary coolant system remained sub-cooled throughout the transient.

3. RELAP5 predicted the steam generator boildown to be significantly faster than in the experiment, which subsequently affected the remainder of the calculation. The reason for this is not clear at present but it is likely that inaccuracies in the input power and primary flow are the main contributory factors. Systematic errors in the calculation of the void fraction in the riser region may also have contributed to an underprediction of the initial steam generator mass inventory.

4. Due to lack of data, pump coastdown and reactor power had to be specified as boundary conditions. This caused the calculation to be very sensitive to the primary coolant flow rate, to the extent that changing the flow within the measurement uncertainty band had a large effect on primary pressure. It is believed that the inclusion of reactivity feedback modelling would have alleviated this sensitivity.

5. RELAP5/MOD2 was found in this study to be applying the Biasi critical heat flux correlation outside its range of validity, resulting in calculation of a negative critical heat flux at pressures above 162.5 bar. The coding has been modified in RELAP5/MOD2 Cycle 36.05 to correct this error.

8. References

1. 'Experiment data report for LOFT Anticipated Transient Without Scram Experiment L9-4' NUREG/CR2978 November 1982

2. 'Quick-look Report on LOFT Nuclear Experiment L9-4' EGG-LOFT-6071 October 1982

3. 'RELAP5/MOD2 analysis of OECD LOFT test LP-FW-01' PWR/HTWG/P(87)497 M.G.Croxford and C.Harwood May 1987

4. 'RETRAN-02 - A program for transient thermal-hydraulic analysis of complex fluid flow systems -Volume 1 Theory and numerics' NP-1850-CCMA Energy Incorporated November 1984

5. 'Assessment of interphase drag correlations in the RELAP5/MOD2 and TRAC-PF1/MOD1 thermalhydraulic codes' GD/PE-N/557 K.H.Ardron and A.J.Clare March 1987

Table 1 Initial Conditions

DESCRIPTION		MEASURED	CALCULATED
PRIMARY COOLANT SYSTEM			
Mass flow	kg/s	461.0	460.99
Hot leg pressure	MPa	14.81	14.799
Temperature across core	K	20.1	18.89
Cold leg temperature	K	556.9	558.78
Hot leg temperature	K	577.0	579.01
Boron concentration	ppm	679	--
REACTOR VESSEL			
Power level	MW	50.7	50.7
PRESSURISER			
Steam volume	m^3	0.36	0.48
Liquid volume	m^3	0.57	0.52
Liquid temperature	K	614.5	614.1
Pressure	MPa	14.81	14.76
Liquid level	m	1.07	0.981
STEAM GENERATOR			
Liquid level	m	3.12	3.12
Liquid temperature	K	538.2	531.3
Pressure	MPa	5.54	5.55
Mass flow	kg/s	27.4	26.59

Table 2 Chronology of events

EVENT	TIME(s) MEASURED	TIME(s) CALCULATED
Primary coolant pumps tripped	0.0	0.0
Main feedwater pump tripped	0.15	0.0
Main steam control valve started to close	1.75	1.75
Auxiliary feedwater initiated	10.8	10.0
Main steam control valve fully closed	13.5	13.1
Steam bypass valve started to open	15.5	15.5
SRV started to open (> 17.23 MPa in PZR)	18.5	19.0
Primary pump 2 stopped	37.0	37.0
SRV cycling ended	128.0	38.0
Significant decrease in primary to secondary heat transfer	209.0	275.0
SRV cycling reinititiated	328.0	296.0
PZR liquid level reached top of indicating range (1.8m)	437.0	345.0
SG liquid level reached bottom of indicating range	458.0	280.0
SRV cycling ended	580.0	636.0
Primary pump 1 stopped	732.0	37.0
Pressure control regained by PZR	985.0	1250.0
PZR level decreased into operating range	1125.0	> 1500.0
Reactor tripped	1507.0	1510.0

Appendix 1
Calculation of decay heat

According to the 1979 ANS Decay Heat Standard ANSI/ANS 5.1 1979 a power pulse at time s results in decay heat at time t :-

$$D_t = \frac{GP(s)}{Q} \sum_{i=1}^{i=23} \alpha_i e^{-\lambda_i(t-s)}$$

where α_i, λ_i are given in Table 7 of ANSI/ANS 5.1 1979.

For operation between times L and U, decay heat at time t is therefore given by :-

$$D_t = \frac{G}{Q} \int_{s=L}^{s=U} P(s) \sum_{i=1}^{i=23} \alpha_i e^{-\lambda_i(t-s)} ds$$

$$D_t = \frac{G}{Q} \sum_{i=1}^{i=23} \alpha_i e^{-\lambda_i t} \int_L^U P(s) e^{\lambda_i s} ds$$

If it is assumed that P(s) is made up of straight line segments, thus in general the functions to be evaluated are of the form :-

$$\int_L^U (A + Bs) e^{\lambda s} ds$$

This is :-

$$A \int e^{\lambda s} ds + B \int s e^{\lambda s} ds$$

$$= A[\frac{e^{\lambda s}}{\lambda}] + B[s \frac{e^{\lambda s}}{\lambda} - \int \frac{e^{\lambda s}}{\lambda} ds]$$

$$= A[\frac{e^{\lambda s}}{\lambda}] + B[s \frac{e^{\lambda s}}{\lambda} - \frac{e^{\lambda s}}{\lambda^2}]$$

$$= \frac{1}{\lambda^2} [A\lambda e^{\lambda s} + Bs\lambda e^{\lambda s} - Be^{\lambda s}]_L^U$$

$$= \frac{1}{\lambda^2} [((A + Bs)\lambda - B)e^{\lambda s}]_L^U$$

$$= \frac{1}{\lambda^2} [((A + BU)\lambda - B)e^{\lambda U} - ((A + BL)\lambda - B)e^{\lambda L}]$$

If we write :-

$$const(i) = \frac{G}{Q} \frac{\alpha_i}{\lambda_i^2} [((A + BU)\lambda_i - B)e^{\lambda_i U} - ((A + BL)\lambda_i - B)e^{\lambda_i L}]$$

the decay heat at time t is thus:-

$$D_t = \sum_{i=1}^{i=23} const(i) \times e^{-\lambda_i t}$$

This equation was used to calculate the decay power for the fission power history of the L9-4 experiment.

Note

Each interval to the left of t contributes as $\int_L^U$ and the interval containing t contributes as $\int_L^t$.

So if L < t, integral is $\int_L^{min(U,t)}$ and if L > t, there is no contribution.

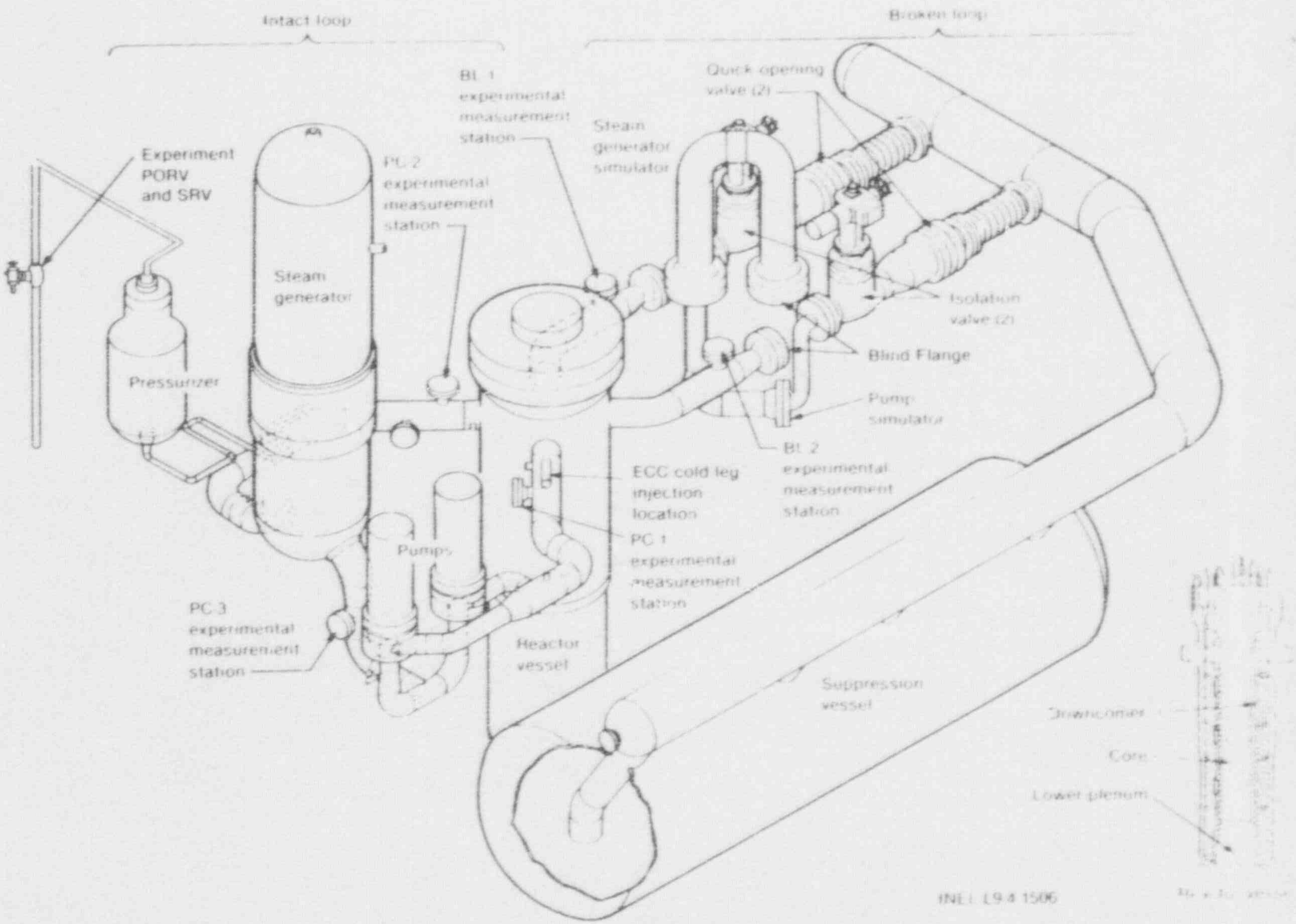

Figure 1. Axonometric projection of LOFT system.

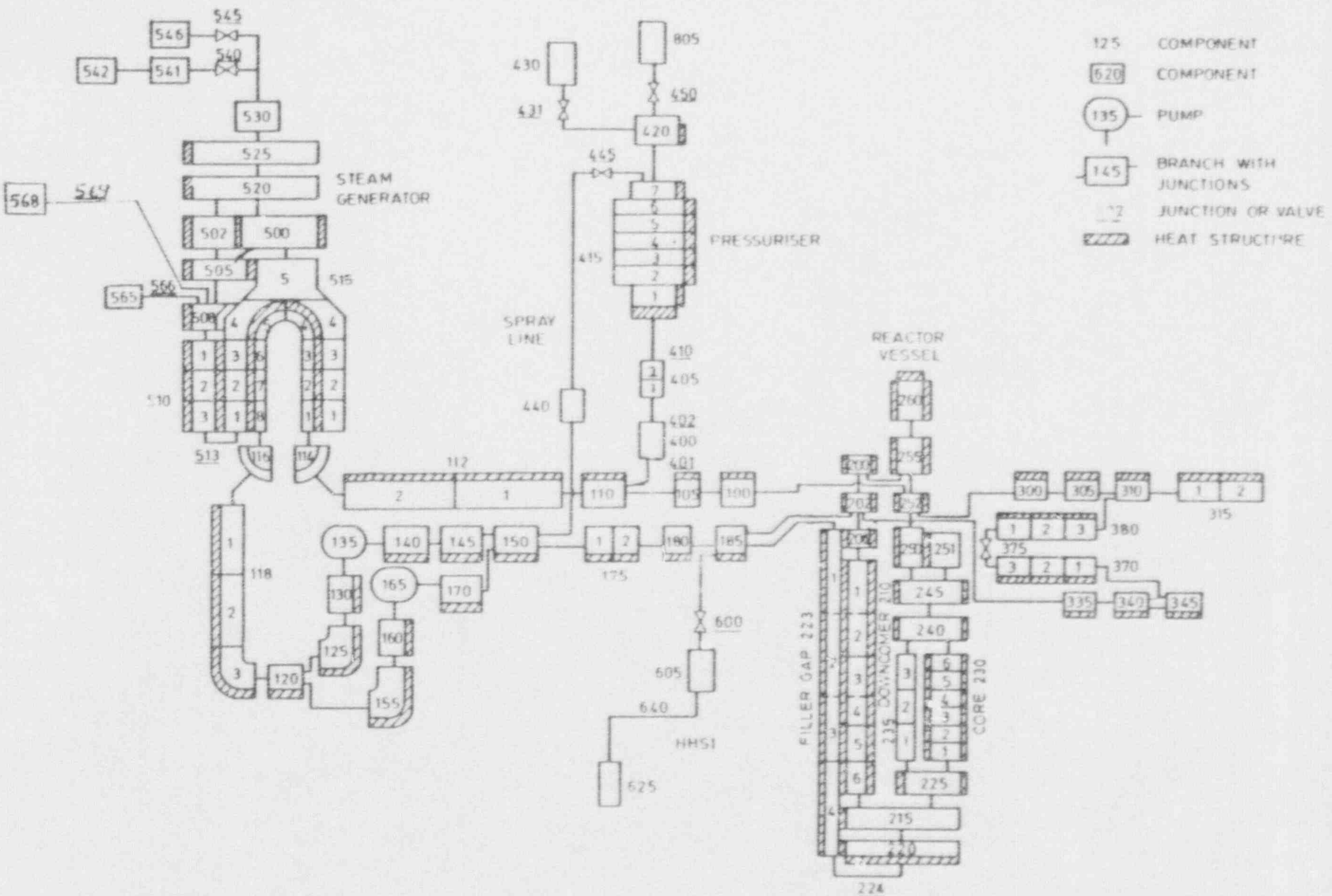

FIGURE 2 - RELAP 5/MOD 2 NODING DIAGRAM FOR CALCULATION OF LOFT TEST L9-4

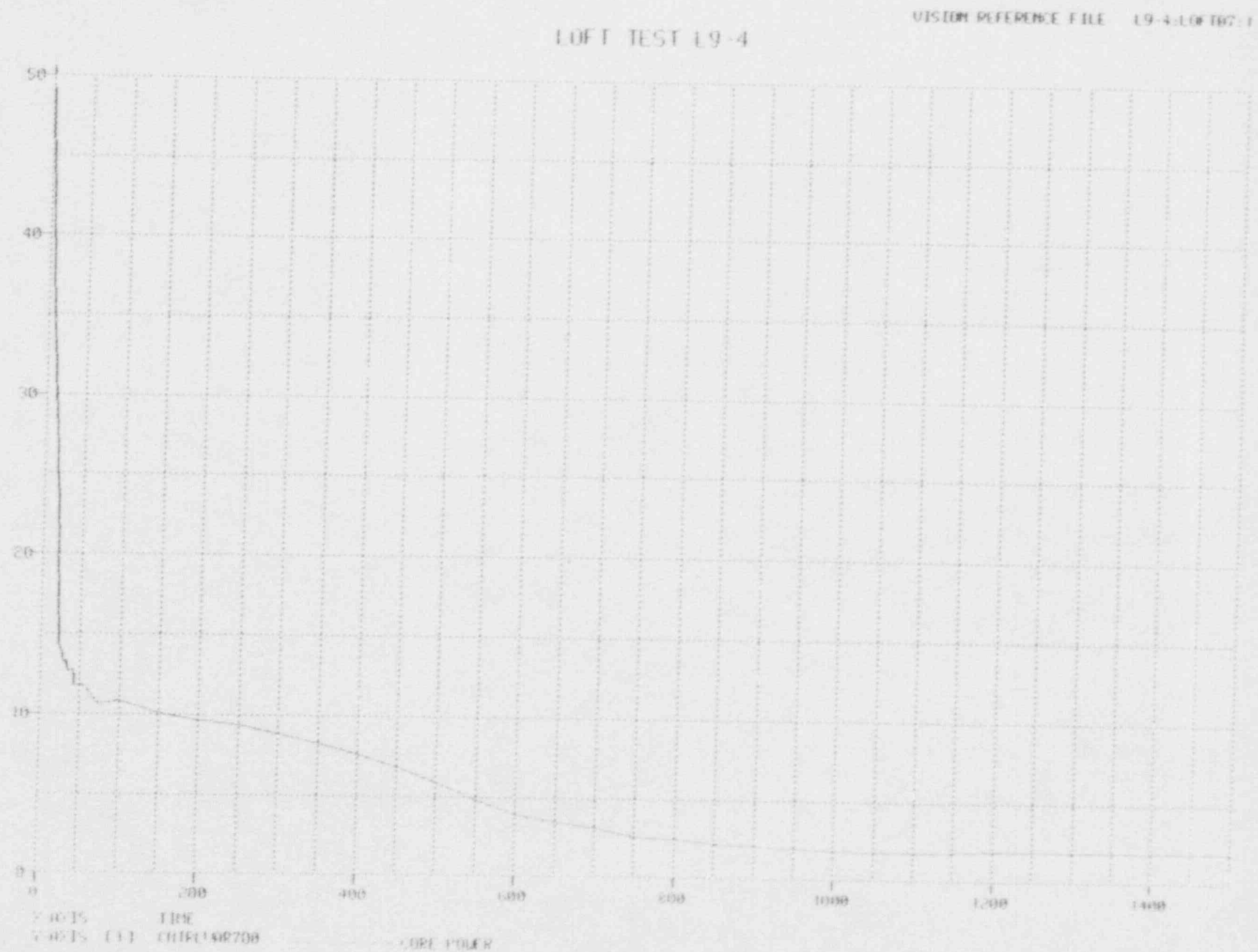

Fig. 3 Core power (MW)

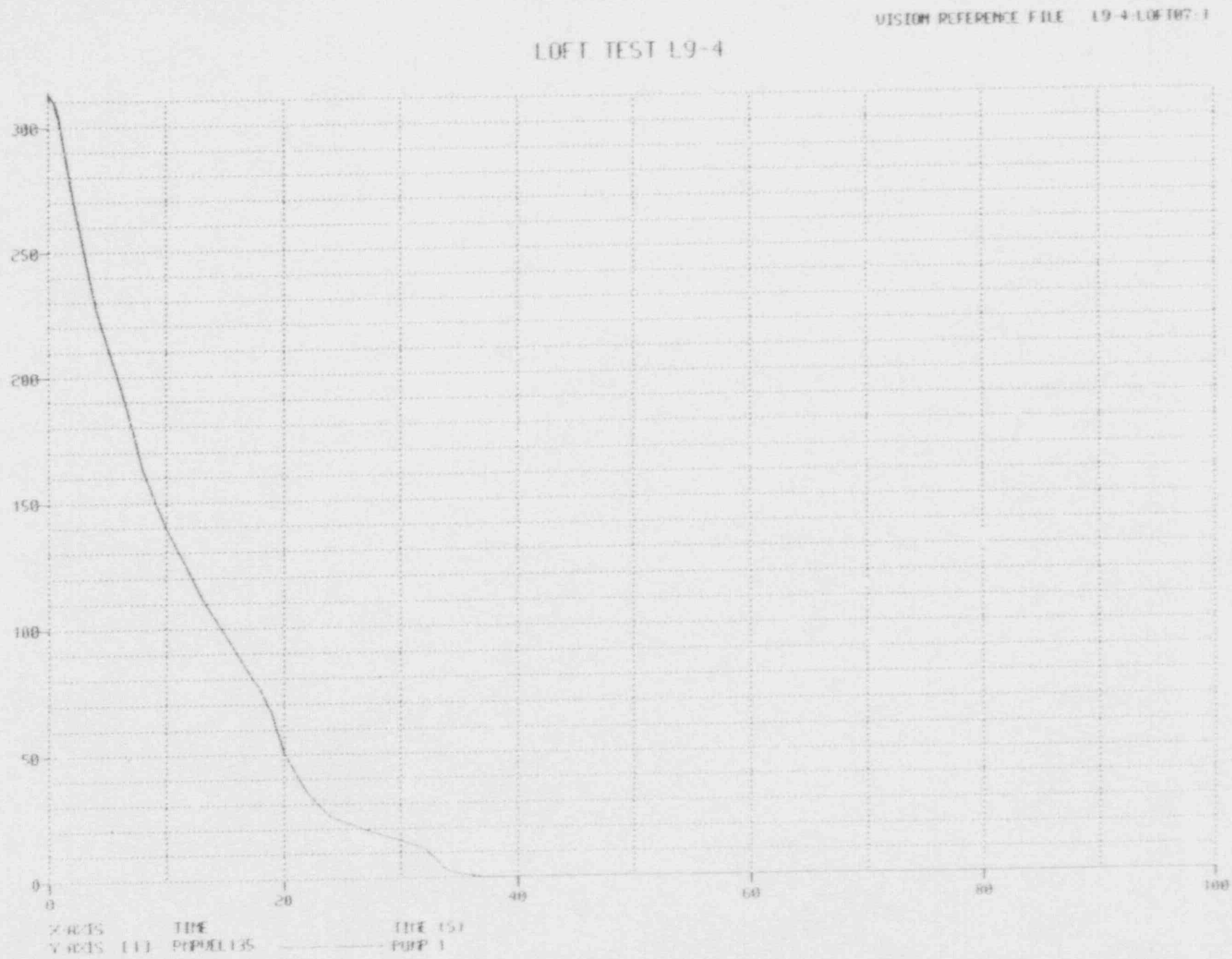

Fig. 4 Pump velocity (rad/s)

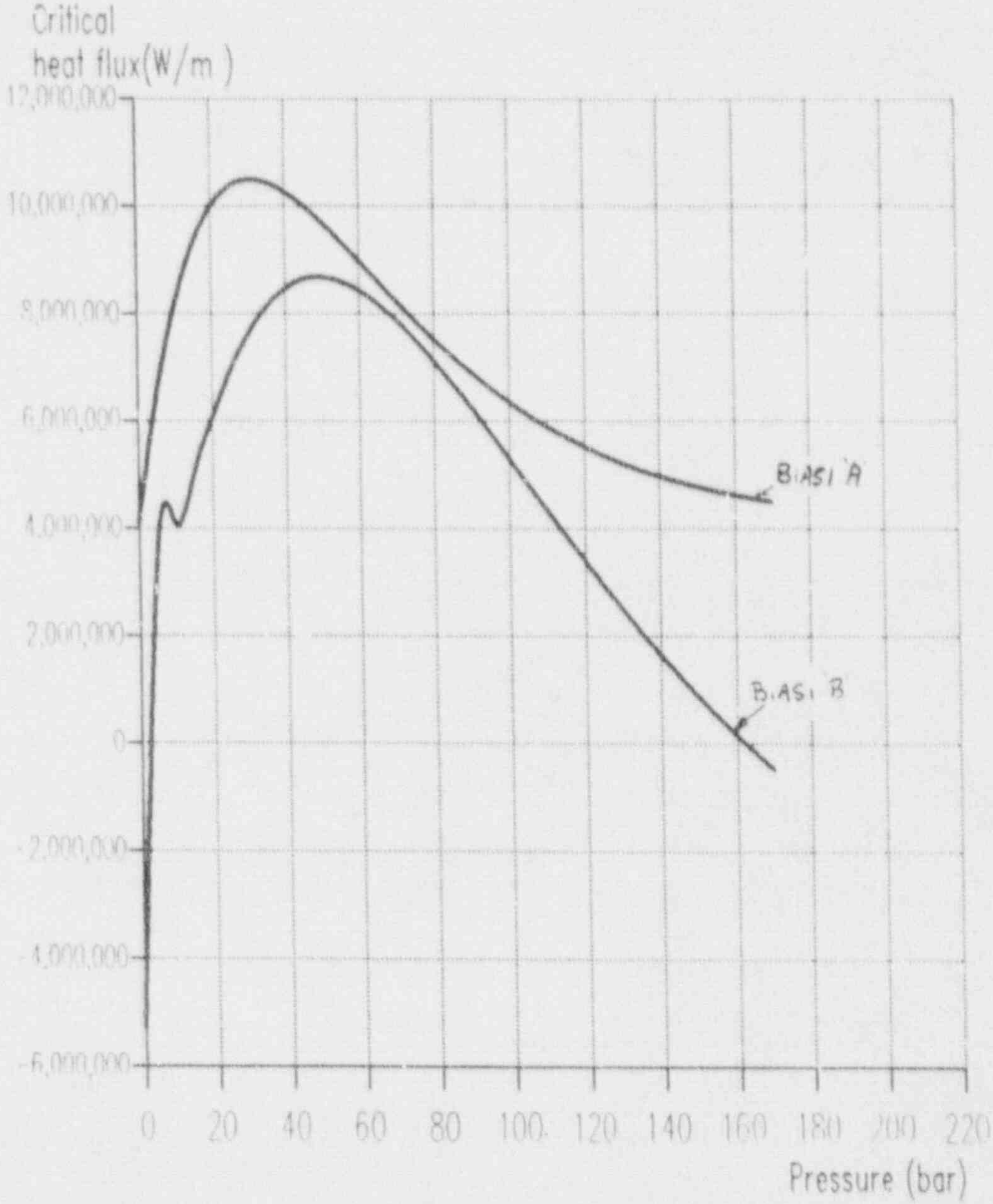

FIG. 5 BIASI CHF'S vs PRESSURE (G = 300 kg/m²s)

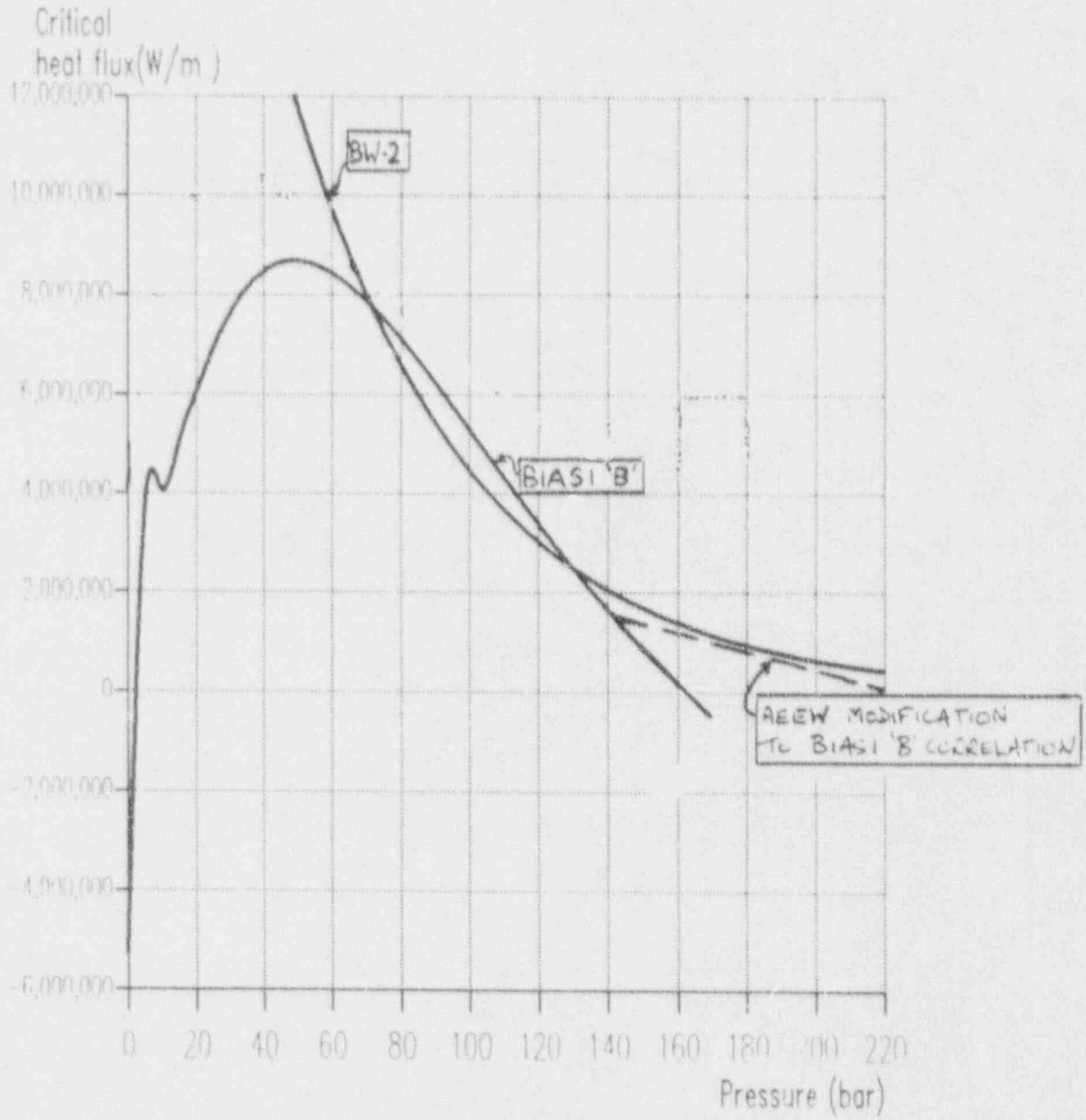

FIG 6 COMPARISON OF MODIFIED BIASI 'B' AND BW-2 CORRELATIONS

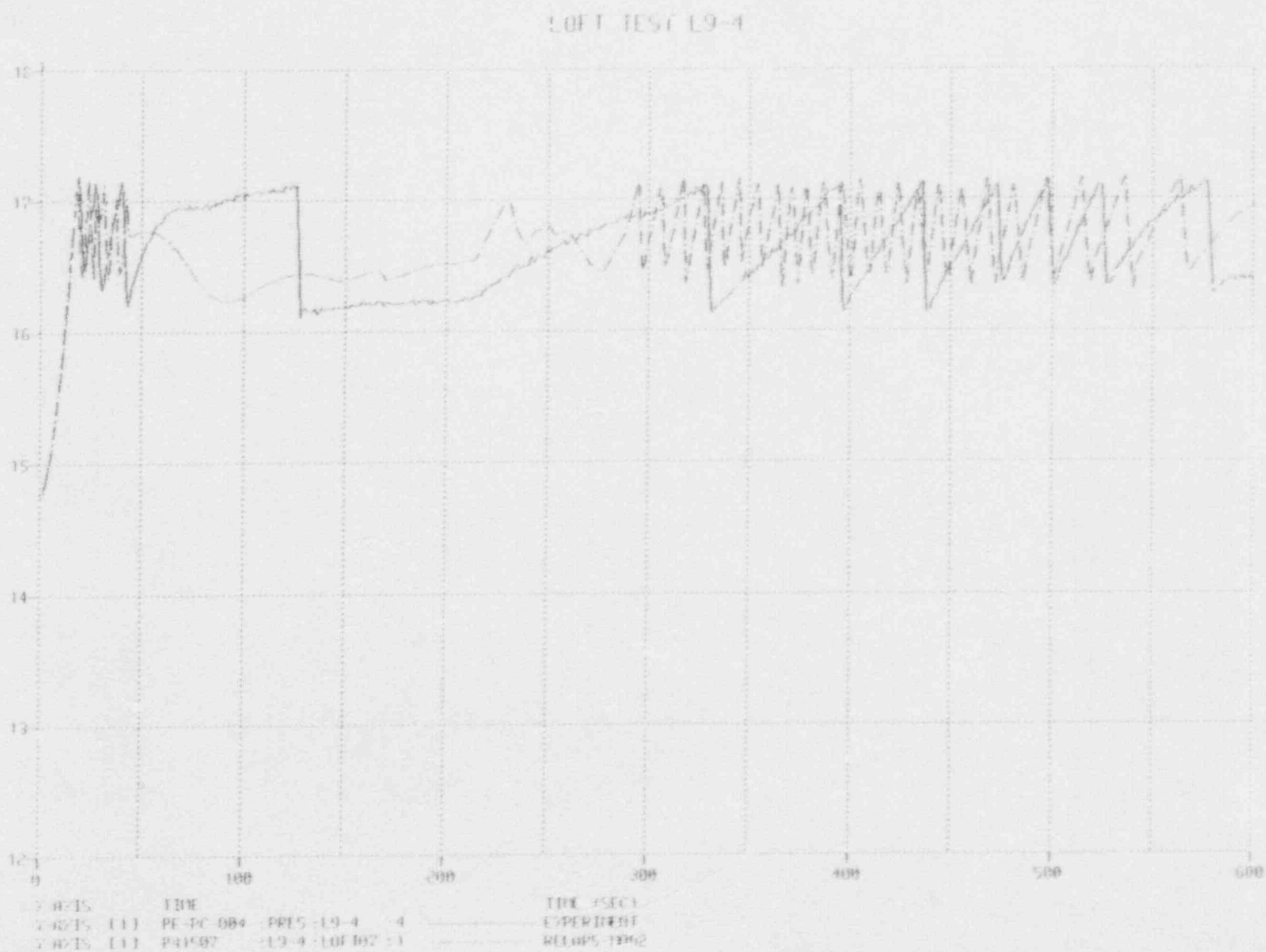

Fig.7 Pressurizer pressure (MPa) 0-600 seconds

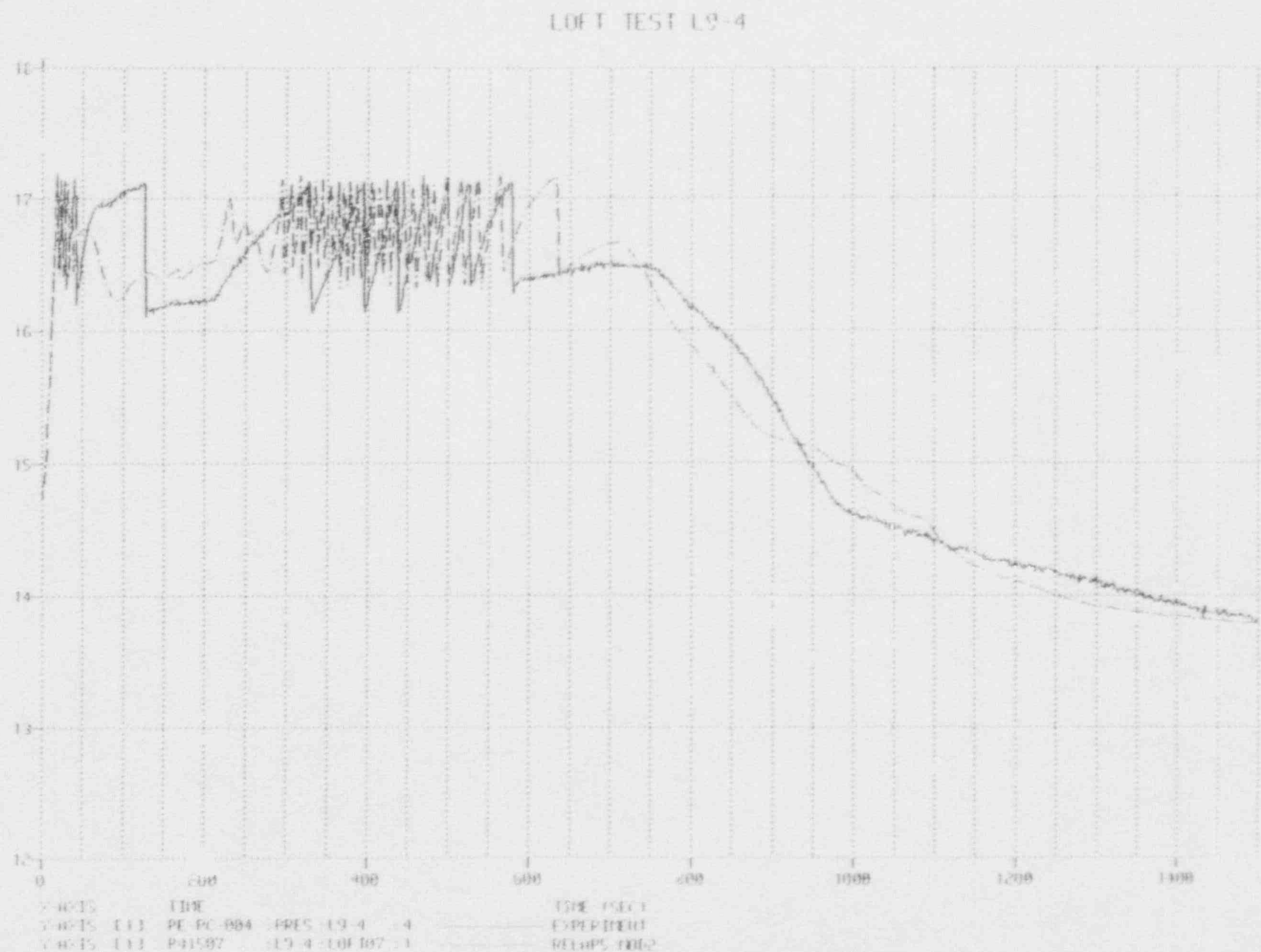

Fig 8 Pressurizer pressure (MPa) 0-1500 seconds

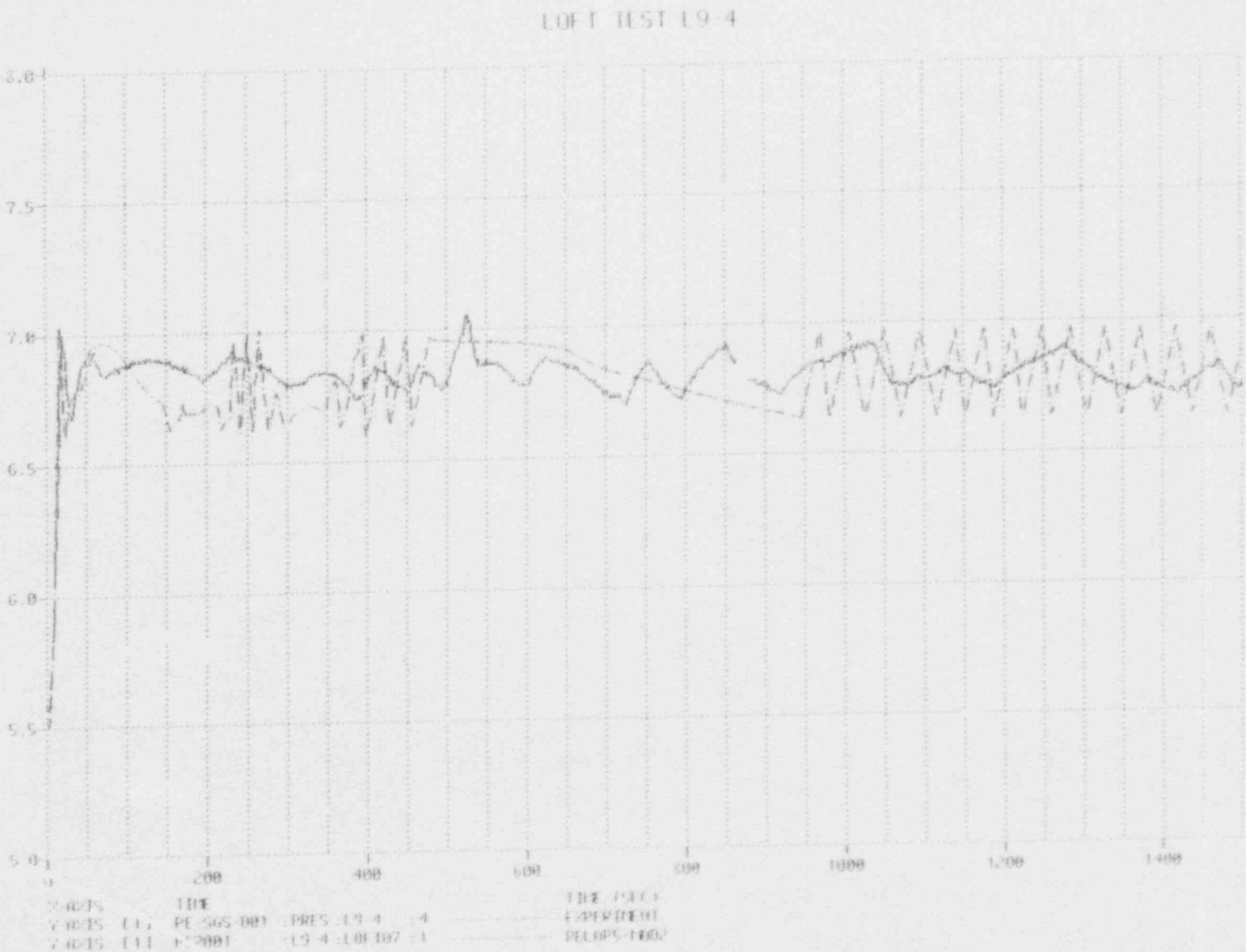

Fig.9 Steam generator dome pressure (MPa)

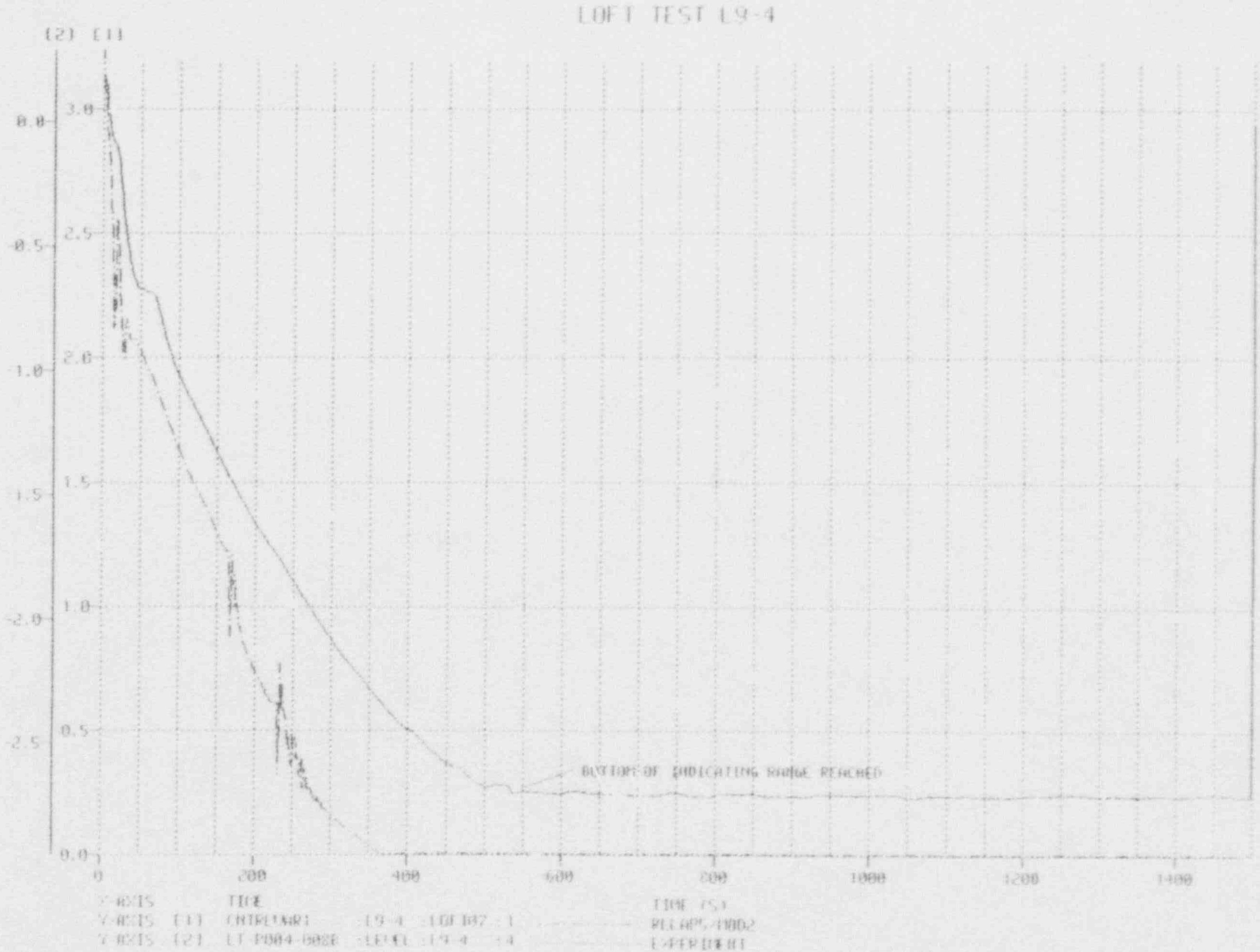

Fig.10 Steam generator liquid level (m)

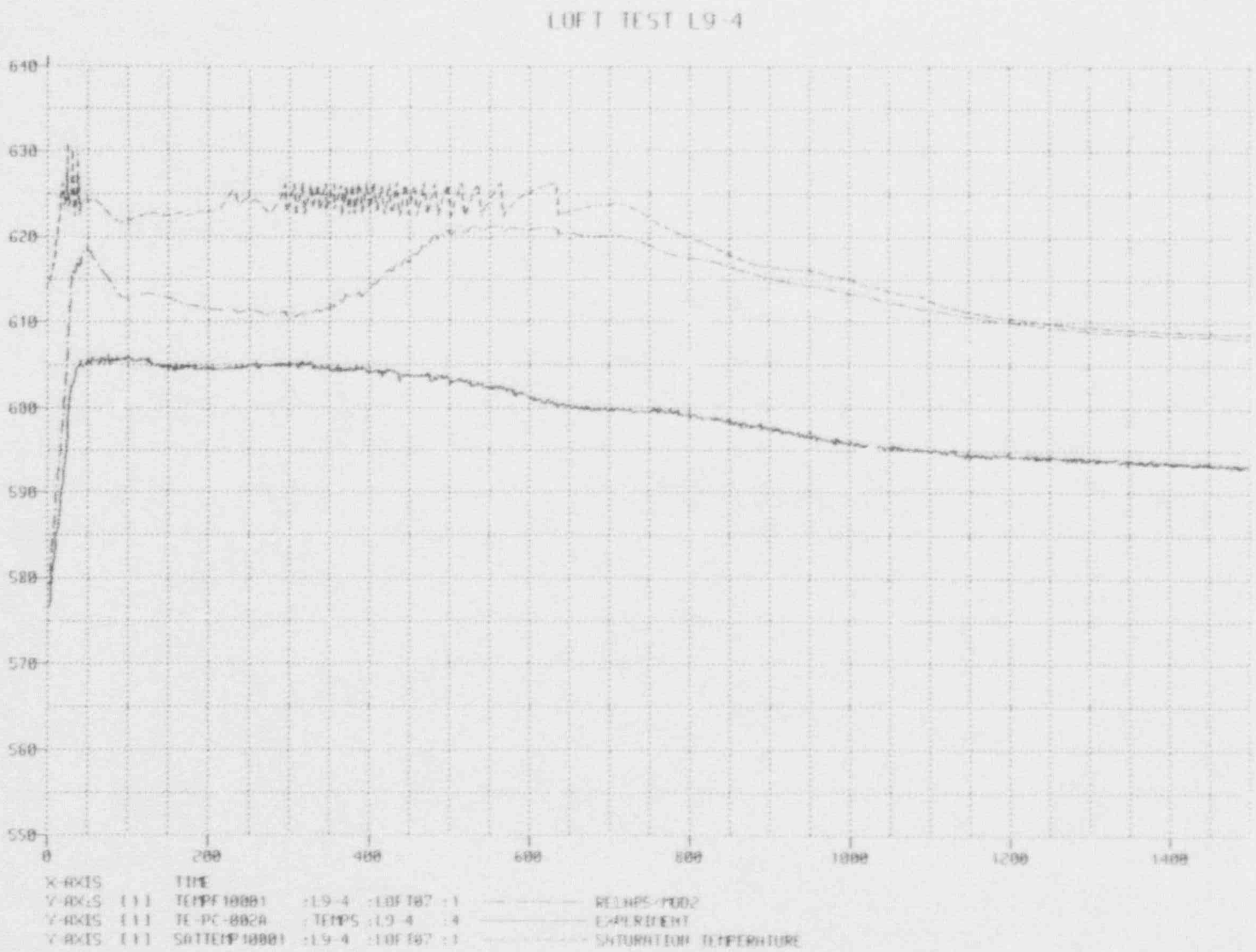

Fig.11 Hot leg temperatures (K)

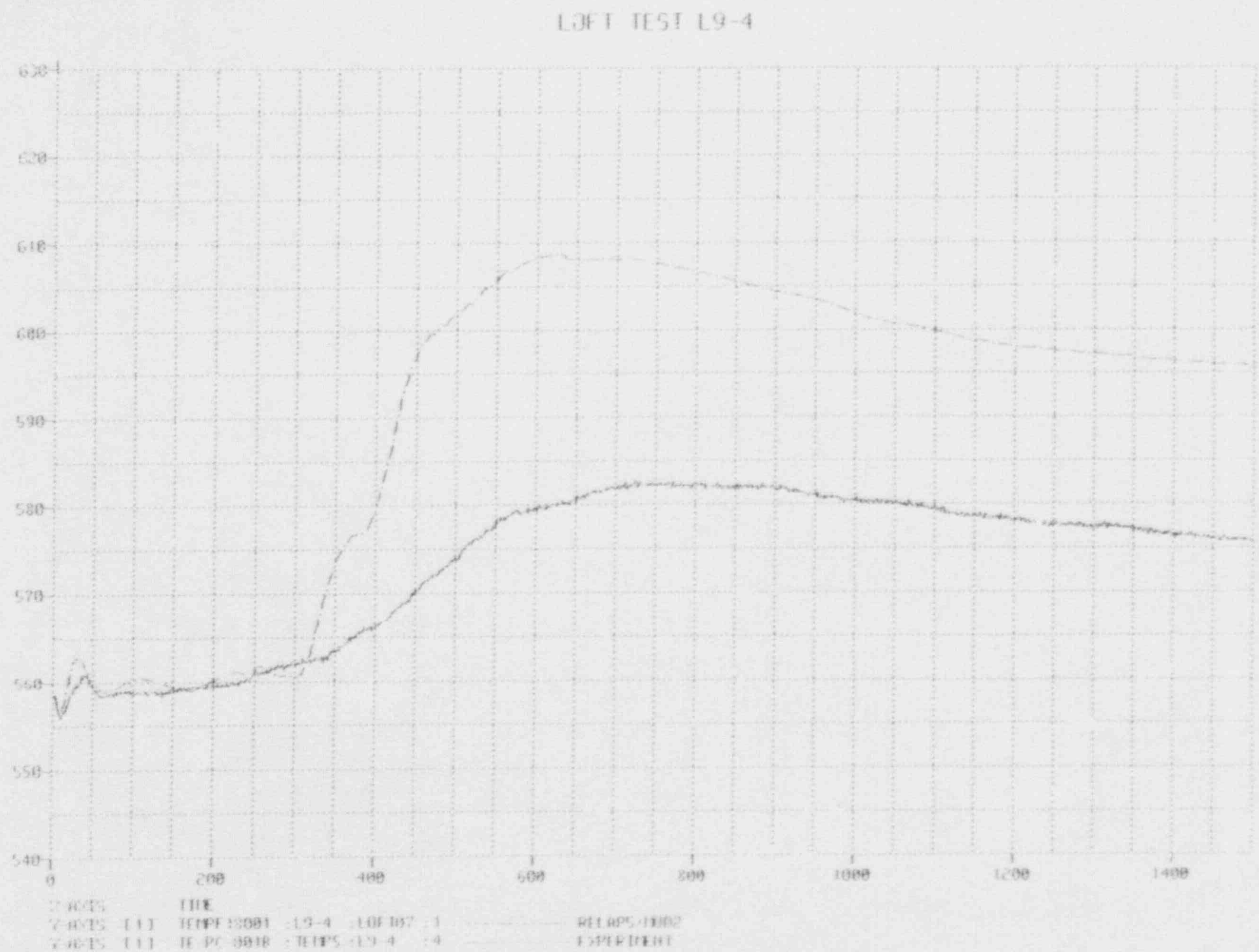

Fig. 12 Cold leg temperatures (K)

LOFT TEST L9-4

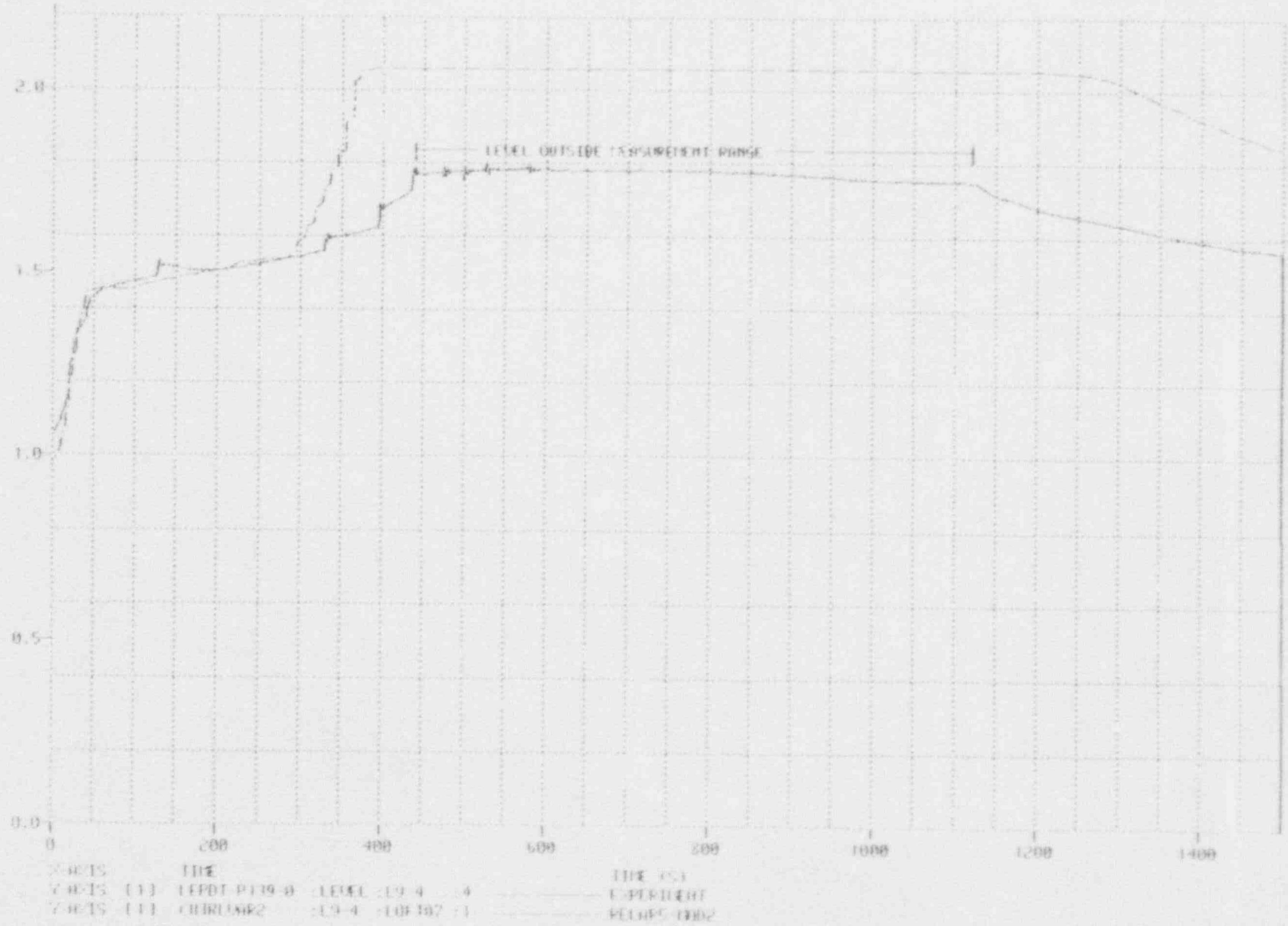

Fig. 13 Pressurizer liquid level (m)

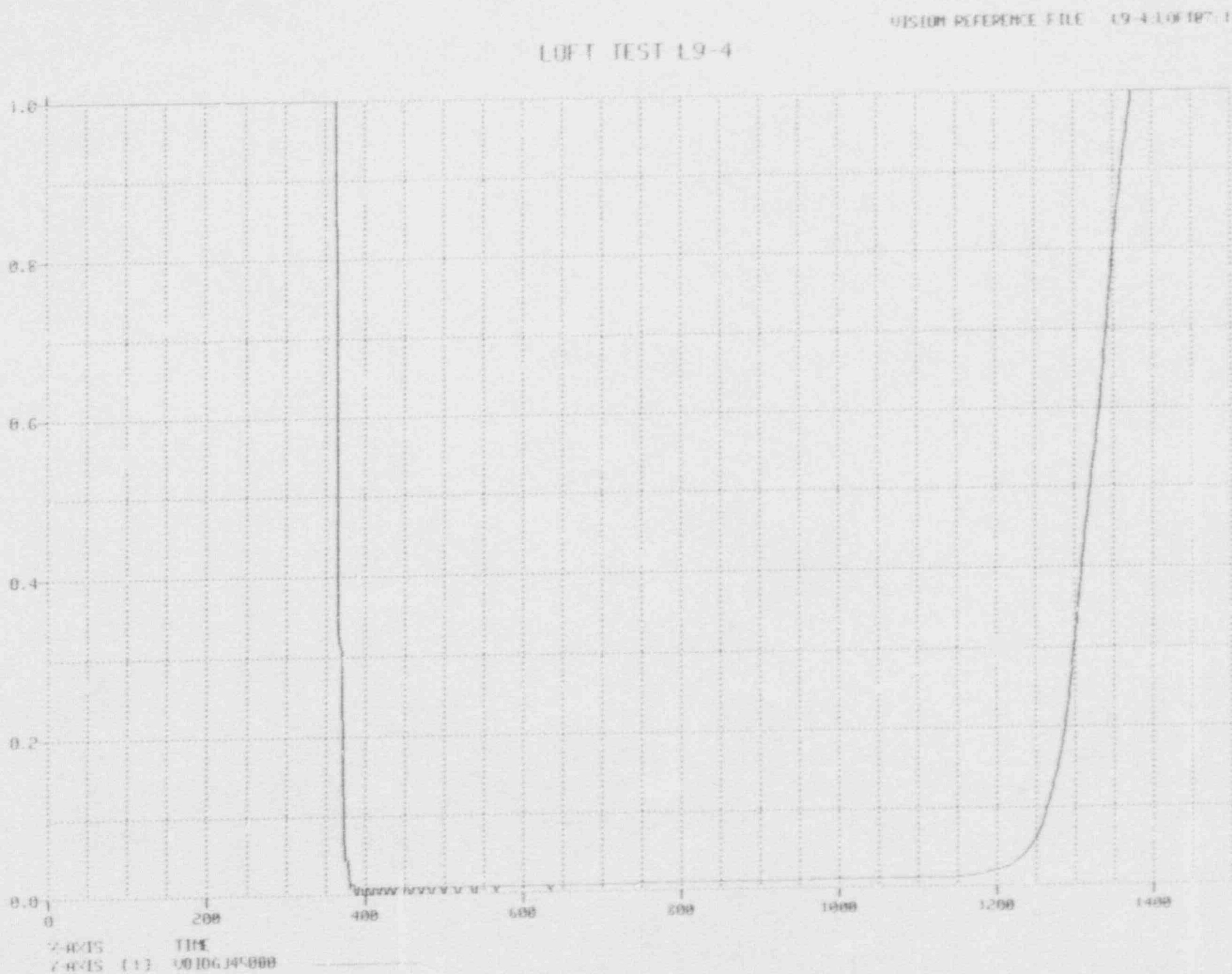

Fig.14 SRV Void Fraction

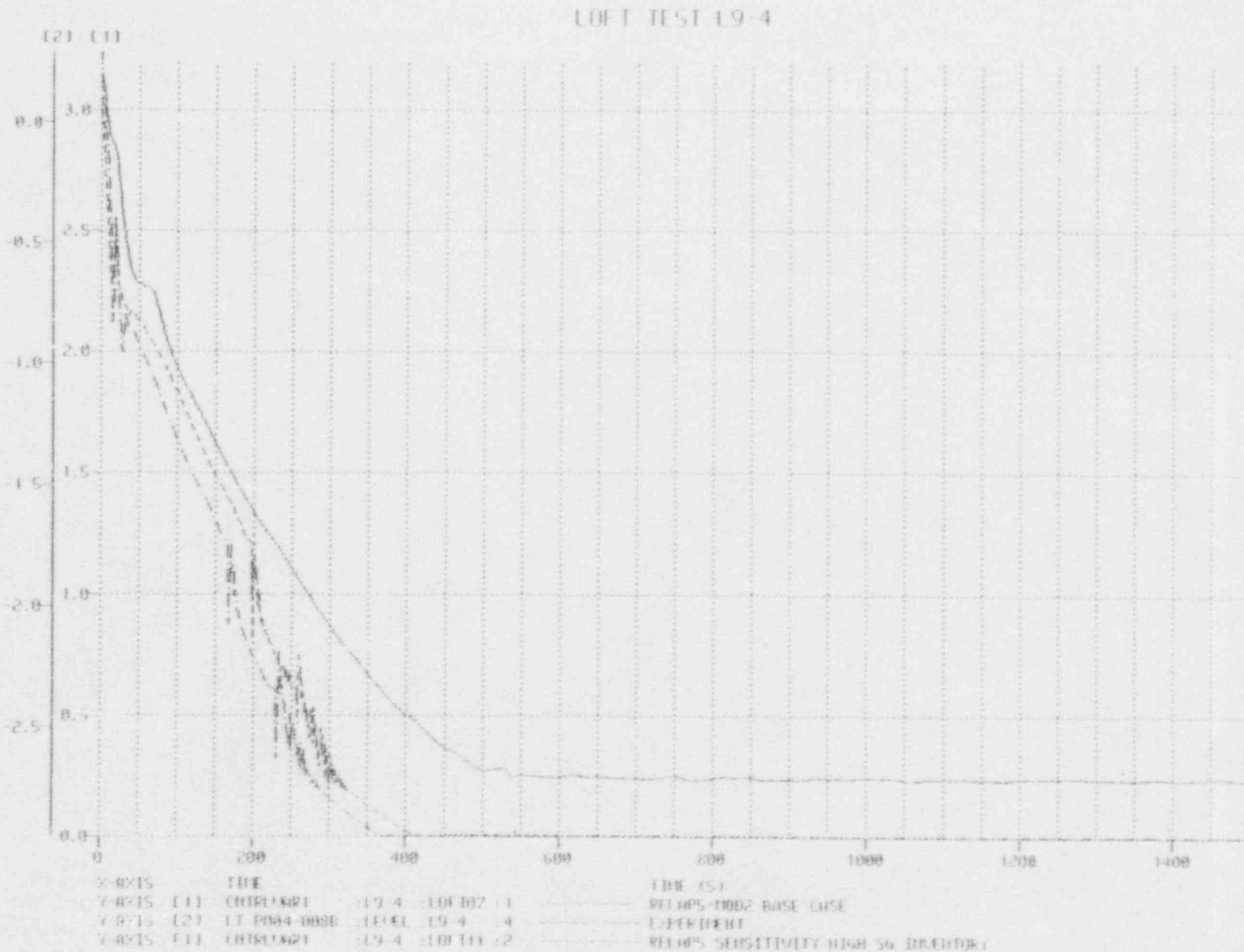

Fig. 15 Steam generator level (m): Sensitivity to SG inventory

LOFT TEST L9-4

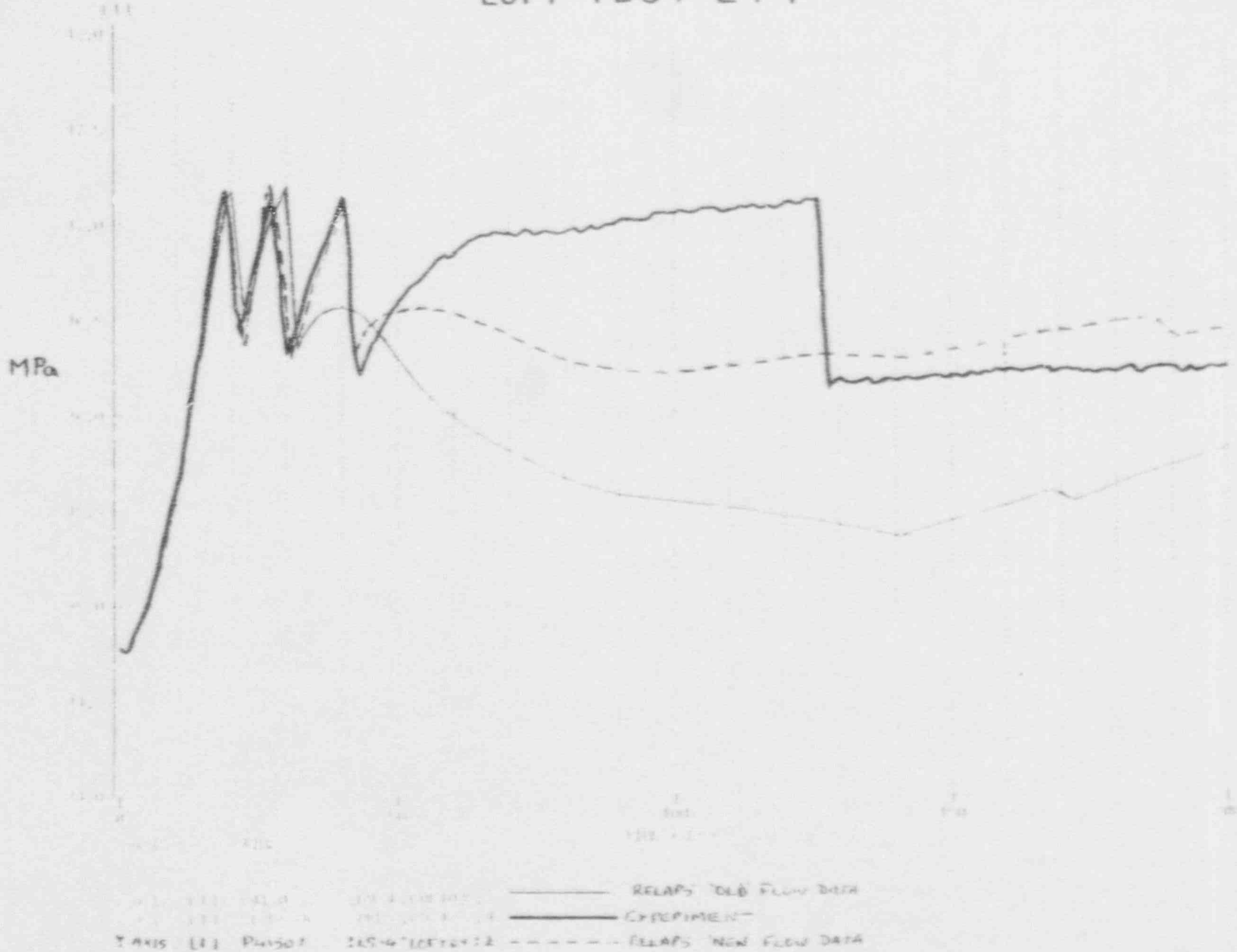

Fig 16 Pressurizer Pressure - Sensitivity to Flow

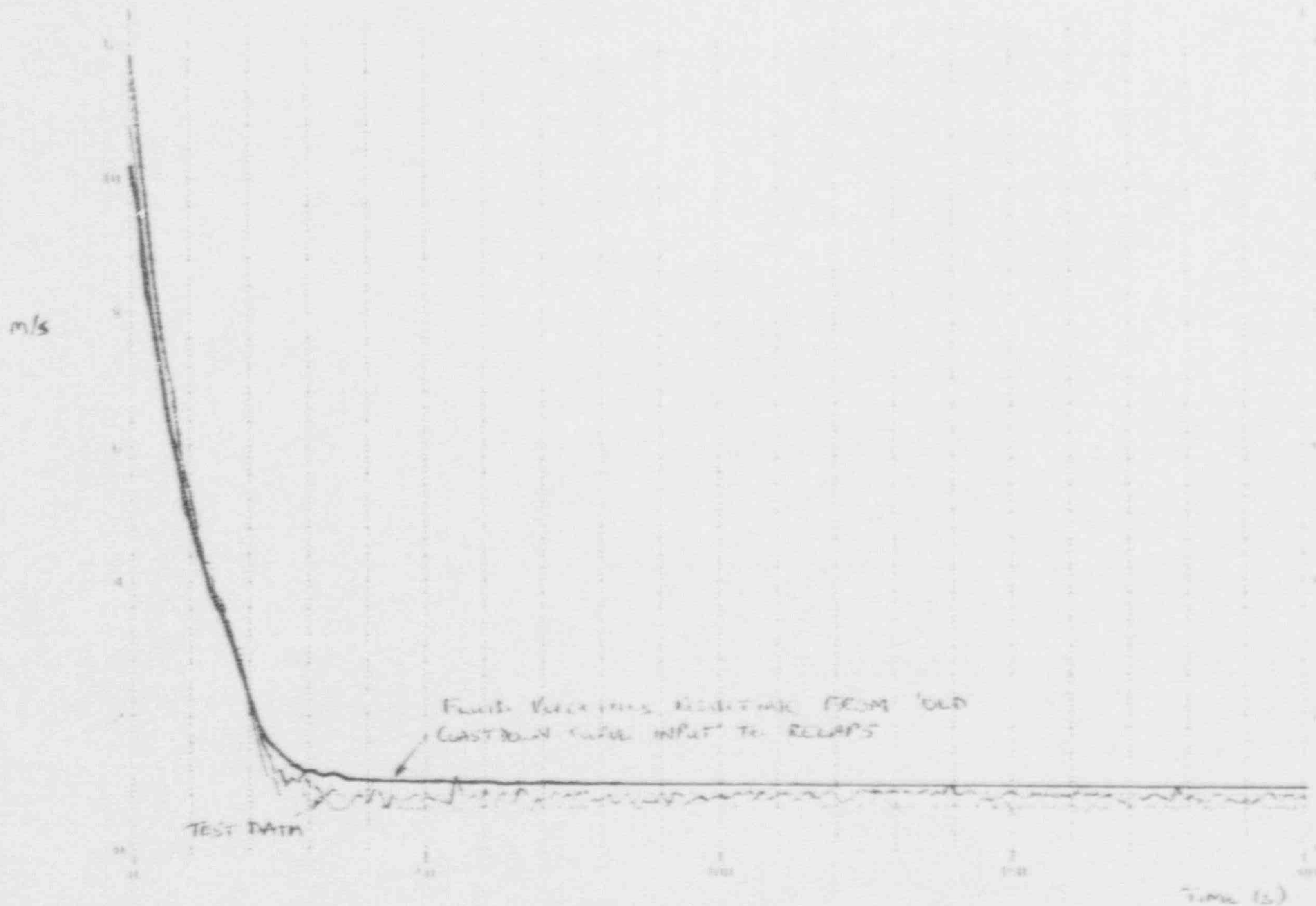

FIG. 17 COMPARISON OF RELAP5 FLUID VELOCITY WITH TEST DATA

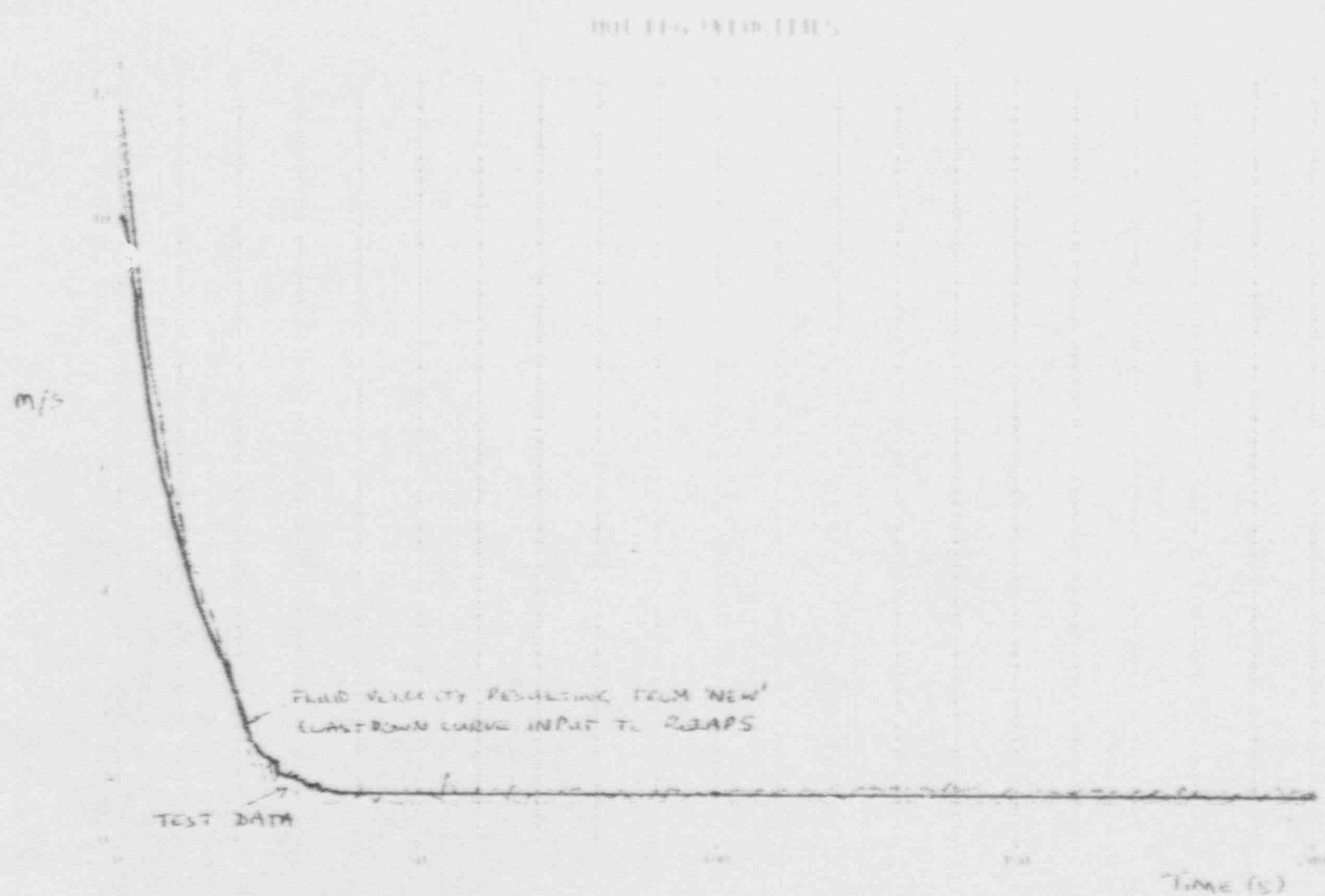

FIG 18 COMPARISON OF RELAP5 FLUID VELOCITY WITH TEST DATA

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BIBLIOGRAPHIC DATA SHEET

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1. REPORT NUMBER (Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/IA-0066
GD/PE-N/721

2. TITLE AND SUBTITLE

RELAP5/MOD2 Analysis of LOFT Experiment L9-4

3. DATE REPORT PUBLISHED

MONTH	YEAR
April	1992

4. FIN OR GRANT NUMBER

A4682

5. AUTHOR(S)

M.B. Keevill

6. TYPE OF REPORT

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION — NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

National Power Nuclear
Barnett Way
Barnwood, Gloucester GL4 7RS
United Kingdom

9. SPONSORING ORGANIZATION — NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

As part of a program to validate RELAP5/MOD2 for use in the analysis of certain fault transients in the Sizewell B PWR, the code has been used to simulate experiment L9-4 carried out in the Loss-Of-Fluid Test (LOFT) facility. Experiment L9-4 simulated a Loss-Of-Offsite-Power Anticipated Transient Without Trip (LOOP ATWT) in which power is lost to the primary coolant pumps and main feed is lost to the steam generators but the control rods fail to insert in the reactor core.

RELAP5/MOD2 generally predicted the transient well, although there were some differences compared to the test data. These differences are largely due to the use of power and flow as boundary conditions and because of uncertainties in the power and flow experimental data. The most noticeable difference was that the steam generator was predicted to boil down too fast. This is believed to be partly due to errors in the RELAP5 interphase drag model. The RELAP5 calculation also showed the primary pressure to be very sensitive to the primary flow rate, making the exact simulation of primary side relief valve movements difficult to reproduce.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

RELAP5/MOD2
LOFT Experiment L9-4
Loss-of-Offsite-Power Anticipated Transient Without Trip (Loop ATWT)
fault transients
Sizewell B PWR

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page)

unclassified

(This Report)

unclassified

15. NUMBER OF PAGES

16. PRICE

NRC FORM 335 (2-89)

THIS DOCUMENT WAS PRINTED USING RECYCLED PAPER

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NUCLEAR REGULATORY COMMISSION
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