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TRAC–PF1/MOD1 Post–Test Calculations of the OECD LOFT Experiment LP–SB–3

Prepared by
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Office of Nuclear Regulatory Research
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NOTICE

SUMMARY

Analysis of the small, cold leg break, OECD LOFT Experiment LP-SB-3 using the best-estimate computer code TRAC-PF1/MOD1 is presented.

Descriptions of the LOFT facility and the LP-SB-3 experiment are given and development of the TRAC-PF1/MOD1 input model is detailed. The calculations performed in achieving the steady state conditions, from which the experiment was initiated, and the specification of experimental boundary conditions are outlined.

Results of the TRAC-PF1/MOD1 calculation are found to be generally consistent with those reported, by members of the OECD LOFT Program Review Group, in the LP-SB-3 "Comparison Report". Overall trends with respect to pressure histories, minimum primary system mass inventory and accumulator behaviour are reasonably well reproduced by TRAC-PF1/MOD1. Prior to break uncovery, the break mass flow rate is slightly over-predicted by the TRAC critical flow model. (Subcooled and two-phase choked flow multipliers of 1.0 were used throughout the calculation). The most significant discrepancy is in the rate with which the fuel rod cladding temperature rises during the core uncovery phase of the transient. TRAC-PF1/MOD1, in common with other codes, significantly over-predicts the rate with which the core heats up. Contrary to experimental observations, conditions for reflux condensation are not predicted by TRAC-PF1/MOD1 during this part of the transient. This, together with the under-prediction of core density by TRAC's interphase drag model, is considered a potential contributor to the poorly predicted rate of core heat-up.

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1 INTRODUCTION

This paper describes a post-test calculation of the OECD LOFT small, cold leg break experiment LP-SB-3 using the "best-estimate" computer code TRAC-PF1/MOD1. Sections 2, 3 and 4 describe the LOFT facility, the LP-SB-3 experiment and the version of TRAC-PF1/MOD1 used, respectively. Development of the input model is detailed in Section 5 and the calculations performed in achieving the steady state conditions, from which the experiment was initiated, are outlined in Section 6. The experimental boundary conditions, and the way in which they are specified to the code, are defined in Section 7. Section 8 describes the transient calculation. The main conclusions from the analysis are summarised in Section 9.

2 THE LOSS OF FLUID TEST (LOFT) FACILITY

The Loss of Fluid Test (LOFT) facility, at the Idaho National Engineering Laboratory (INEL), is a 50 MW(t) Pressurised Water Reactor (PWR) system designed to simulate the major components and system responses of a commercial PWR during Loss-of-Coolant Accidents (LOCAs) or operational transient accidents. The experimental assembly is instrumented in order that system variables can be measured and recorded during transients. The facility is comprised of five major subsystems - the reactor vessel, the operating (intact) loop, the "broken" loop, the blowdown suppression system and the Emergency Core Cooling System (ECCS). The configuration of the major LOFT components, for experiment LP-SB-3, is shown in Figure 1.

The operating (intact) loop simulates three loops of a commercial four-loop PWR and contains a steam generator (of vertical, U-tube design), two primary coolant pumps (in parallel), a pressurizer, a venturi flowmeter and connecting piping. The break location for experiment LP-SB-3 was in the cold leg of the intact loop between the primary coolant pumps and the reactor vessel.

The broken loop consists of a hot leg and a cold leg connected to the reactor vessel and the blowdown suppression tank header. Each leg contains a Quick-Opening Blowdown Valve (QOBV), a recirculation line, an isolation valve and connecting piping. The recirculation lines provide a small flow from the broken loop to the intact loop and are used to maintain the broken loop fluid temperature at approximately the core inlet temperature prior to experiment initiation. During experiment LP-SB-3, the QOBVs and the isolation valves remained closed (because the break was in the operating loop). Broken loop spool pieces, with orifices to simulate the steam generator and pump hydraulic resistances, were installed for experiment LP-SB-3.

The LOFT reactor vessel has an annular downcomer, a lower plenum, lower core support plates, a nuclear core (containing 1300 fuel rods) and an upper plenum. The downcomer is connected to the cold legs of the operating and broken loops, and the upper plenum is connected to the hot legs of the operating and broken loops.

The LOFT ECCS consists of two accumulators, a High Pressure Injection System (HPIS) and a Low Pressure Injection System (LPIS). Each system is designed to inject scaled flows of emergency core coolant directly into the primary coolant system. The HPIS was not used during experiment LP-SB-3.

3 EXPERIMENT LP-SB-3

Experiment LP-SB-3 was conducted on 5 March 1984 in the LOFT facility at the Idaho National Engineering Laboratory. LP-SB-3 was the sixth in a series of experiments, sponsored by a consortium of countries under the auspices of the Organisation for Economic Cooperation and Development (OECD).

The LP-SB-3 experiment simulated a 4.66 cm (1.84 inch) equivalent diameter break in a cold leg pipe of a commercial PWR. The break line piping and details of the break nozzle (0.88 cm diameter) are shown in Figure 1. The transient was specifically designed to achieve conditions which would allow an assessment of:

(i) The mechanism for core heat transfer under slow coolant boil off conditions.

(ii) The effectiveness of operator initiated steam generator feed and bleed recovery.

(iii) The effectiveness of accumulator injection when a small pressure differential exists between the accumulator and the primary system.

The experiment was initiated, from conditions representative of those in a commercial PWR, by opening a valve in the Intact Loop Cold Leg (ILCL) break piping. The primary system depressurised rapidly until fluid saturation conditions were reached in the hot leg at ~ 100 seconds. This resulted in a decrease in the primary system depressurisation rate and, with void formation in the fluid, a reduction in the break mass flow rate. The continued operation of the primary coolant pumps homogenised the fluid and the primary system void fraction steadily increased as fluid was discharged out the break. The pumps were tripped after 1600 seconds when approximately 2800 kg of coolant inventory remained in the primary system. After pump trip, the fluid expelled from the break was predominantly steam. The HPIS was not used during the experiment and the mass inventory was allowed to decrease, uncovering the core.

As the core became increasingly void of fluid, the fuel cladding started to heat up. The break was isolated, to prevent further depressurisation of the primary system, when the cladding temperature reached 835K. When the maximum cladding temperature reached 988K, a steam generator feed and bleed procedure was initiated. The energy removal through the secondary system depressurised the primary system and the fuel cladding started to cool. Accumulator flow commenced when the primary system had depressurised to 2.79 MPa and the fuel cladding was gradually quenched, from the bottom, upwards. The LPIS, initiated at a primary system pressure of 1.03 MPa, quickly refilled the core region, thus terminating the experiment.

4 TRAC-PF1/MOD1

TRAC (Transient Reactor Analysis Code) is an advanced "best-estimate" computer code, developed at the Los Alamos National Laboratory, for analysing transients in thermal-hydraulic systems. Specifically, TRAC-PF1/MOD1 was developed for analysing postulated accidents in PWRs. The version of the code used for the calculations described in this paper was AEEW Version BØ2A which contains the LANL updates to TRAC-PF1/MOD1 Version 12.7.

5 TRAC-PF1/MOD1 INPUT MODEL FOR LP-SB-3

The development of a TRAC-PF1/MOD1 input model for analysis of LOFT small break experiments was based on a TRAC-PF1/MOD1 large break deck for the experiment LP-FP-1. The LP-FP-1 deck, developed at AEEW, originated from the LANL input deck for experiment L2-3. Additional published data on the LOFT facility (1, 2, 3) were employed, where necessary, in producing the small break deck.

The modifications, to the large break deck are described in References 4, 5, 6 and 7 and included:

(i) Replacement of the three-dimensional vessel with a one-dimensional representation.

(ii) Inclusion of the ILCL break.

(iii) Inclusion of the primary pump injection.

(iv) Renodalisation of the broken loop to provide a coarser representation.

(v) Renodalisation of the ECCS to provide a coarser representation.

(vi) Accurate repositioning of the ECCS injection point in the ILCL.

With the exception of those modifications specifically required to model the LP-SB-3 experiment, the input deck was identical to that used for the TRAC-PF1/MOD1 analyses of LP-SB-1 (7) and LP-SB-2 (8).

Figures 2, 3 and 4 show the nodalisation diagrams for the primary system, the reactor vessel and the steam generator secondary side, respectively.

A total of 41 components, 47 junctions and 182 cells were used in the model.

A microfiche listing of the TRAC-PF1/MOD1 input deck for LP-SB-3 is contained in Appendix I.

6 STEADY STATE CALCULATION

AEEW Version BØ2A of the TRAC-PF1/MOD1 code - which incorporates the updates contained in LANL Version 12.7 - was used for the steady state calculations. Steady state mode calculations were run for 410 seconds and, in order to determine system conditions during transient mode code operation, a short period (50 seconds) of transient mode "steady state" (ie with no break flow) was also run. In running the steady-state calculation, a total of 773 seconds of CPU time were used with an average time step size of 0.096 seconds.

The calculations were performed with control systems governing the behaviour of the steam generator secondary side steam and feedwater mass flow rates and the speeds of the primary coolant pumps.

The initial conditions predicted by TRAC-PF1/MOD1 for experiment LP-SB-3 are compared with the measured data in Table 1. The calculations produced stable initial conditions, within, or very close to, the quoted experimental uncertainties, for all significant parameters.

The magnitudes of the steady state pressure drops around the primary circuit, the environmental heat losses from the system and the core bypass flow rates, obtained from the TRAC-PF1/MOD1 calculations associated with analysis of the LOFT LP-SB-1 experiment, are reported (7) to be in reasonable agreement with the available data.

7 BOUNDARY CONDITIONS FOR TRANSIENT CALCULATION

7.1 Decay Heat Data

Following reactor scram, decay heat data were specified to the TRAC-PF1/MOD1 transient calculation by means of a "power versus time" table. Data contained in Reference 9 were used throughout the transient.

7.2 Primary Pump Injection

During experiment LP-SB-3, the primary coolant pump injection system was set up to deliver a total flow of 0.12 ls^{-1} to the primary coolant pumps (10). This was simulated, in the TRAC-PF1/MOD1 model, by using "FILL" components to supply the primary pump injection system with liquid at a constant rate of 1.63875×10^{-3} ms^{-1}. The flow areas of the pump injection pipes were 3.66131×10^{-2} m^2 which implied an injection rate of 6.0×10^{-5} m^3s^{-1}, or 0.06 ls^{-1}, to each pump. The primary coolant pump injection was inserted downstream of each pump.

7.3 Operational Setpoints

The operational setpoints (for reactor scram, main feedwater shut off, MSCV closure and opening, primary coolant pump trip, break valve closure, steam bypass valve opening, main feedwater initiation, accumulator injection initiation and LPIS initiation) measured during the experiment, and the way in which the setpoints were specified in the TRAC-PF1/MOD1 calculation, are given in Table 2.

8 TRANSIENT CALCULATION

8.1 Introduction

The transient calculation was restarted from the end of the transient mode steady state calculation by opening the VALVE component in the ILCL break piping. As for the steady state calculations, AEEW Version BØ2A of TRAC-PF1/MOD1 - which incorporates the code updates contained in LANL Version 12.7 - was used for the transient calculation.

In this Section, the TRAC-PF1/MOD1 predictions are compared with the experimental data and with the results presented - by members of the OECD LOFT Program Review Group - in the LP-SB-3 "Comparison Report" (11). For discussion purposes, results are considered separately for each of the three distinct phases of the transient ie mass depletion phase, core boil-off phase and core cool-down and recovery phase.

8.2 CPU Usage and Time Step Behaviour

6400 seconds of elapsed transient were calculated, requiring 4702 seconds of CRAY-XMP CPU time (see Figure 5). This corresponds to a CPU/real time ratio of 0.73. The user-specified minimum allowable time step throughout the calculation was limited to 1×10^{-5} seconds. At ~ 300 seconds, the code started to take very small time steps and it was found necessary to reduce the user-specified maximum allowable time step size from 0.5 seconds to 0.2 seconds for a short period, during which the calculation "recovered" (see Figure 6). The average time step for the calculation (problem time/total number of time steps) was 0.32 seconds.

8.3 Chronology of Events

A comparison of the measured and predicted timings of significant events is given in Table 3.

Experiment LP-SB-3 was initiated by opening the break valve in the ILCL break line. After 9.2 seconds, the primary system had depressurised to the reactor scram set point of 14.19 MPa. The steam generator control system responded to the reactor scram by isolating the feedwater flow to the steam generator and preventing the steam flow out of the steam generator. The main feedwater pump was shut off at 9.4 seconds and the feedwater was isolated by 10.8 seconds. The Main Steam Control Valve (MSCV) started to close at 9.5 seconds and took ~ 11 seconds to become fully closed. The timings of these initial events were predicted, by the TRAC-PF1/MOD1 calculation, to within ~ 2 seconds.

Following reactor scram, the secondary system pressure increased sharply due to the isolation of the feedwater and the MSCV closure. In order to keep the secondary system pressure between 6.35 MPa and 7.09 MPa, the MSCV cycled four times, during the experiment, between 87.5 seconds and 1030 seconds. TRAC-PF1/MOD1 predicted the MSCV to cycle only twice, at 27 seconds and at 124 seconds.

TRAC-PF1/MOD1 correctly predicted the time at which fluid saturation occurred in the primary system ie at ~ 100 seconds.

An abrupt degradation in the differential pressure across the primary coolant pumps was observed at 875 seconds, in the experiment. Pump degradation was predicted to occur between 700 seconds and 1100 seconds. In both the experiment and the calculation, the primary coolant pumps were tripped at 1600 seconds. This resulted in the break uncovering at 1612 seconds in the experiment and at ~ 1625 seconds in the calculation.

The start of core heat up (in the upper part of the core) caused by the slow boiling off of liquid was detected, in the experiment, at ~ 3800 seconds. This was predicted, by the TRAC-PF1/MOD1 calculation, to commence ~ 260 seconds later. When the highest indicated cladding temperature reached 835.2K, the break was isolated. This occurred, some 50 seconds later than predicted, at 4742 seconds.

The cladding temperature continued to rise until the highest indicated value reached 988K, at which time the secondary side feed and bleed operation was initiated by fully opening the main steam control bypass valve. This action, which occurred at 5415 seconds in the experiment and was predicted, by the TRAC-PF1/MOD1 calculation, to occur at 4964 seconds, caused a very rapid cooldown and depressurisation of the secondary system. The maximum fuel clad temperature was measured at 5422 seconds and predicted to occur at 4980 seconds.

In the experiment, the primary system pressure reached the accumulator setpoint (2.79 MPa) at 5588 seconds. In the TRAC-PF1/MOD1 calculation, accumulator injection was predicted to commence at 5308 seconds. The LPIS was initiated (at a primary system pressure of 1.03 MPa) at 6785 seconds in the experiment; the core quickly refilled and the experiment was terminated at 6845 seconds. The calculation was terminated at 6400 seconds when accumulator nitrogen was predicted to have entered the system.

A series of pictures showing the predicted void fraction distribution, the liquid and vapour velocities and the occurrences of stratified flow conditions, throughout the system, is presented in Appendix II.

8.4 Mass Depletion Phase

The mass depletion phase is defined as that part of the transient between break initiation (0 seconds) and pump trip (1600 seconds).

8.4.1 Primary and Secondary System Pressures

A comparison of the measured and predicted primary system pressure histories is shown in Figure 7. The initial, rapid subcooled depressurisation is well represented and the time at which fluid saturation conditions occur (~ 100 seconds) is also well-predicted. Following the end of subcooled depressurisation, and prior to pump trip, (100-1600 seconds) the overall primary system pressure history is well reproduced; detailed characteristics of the primary system pressure behaviour, resulting from the cycling of the secondary side MSCV, are not represented exactly, however.

Following isolation of the main feedwater and closure of the MSCV, the rate at which the secondary system pressurises is slightly over-predicted, as shown in Figure 8, and the cycling of the MSCV is not accurately reproduced. During the experiment, the MSCV cycled four times, between 87.5 seconds and 1030 seconds; it was predicted, by the TRAC-PF1/MOD1 calculation, to cycle only twice - at 27 seconds and at 124 seconds. It can be seen in Figure 8 that the setpoints at which the MSCV was opened and closed were not consistent throughout the experiment. Values of 7.09 MPa and 6.35 MPa (10), respectively, were specified to the TRAC-PF1/MOD1 calculation. A further factor, complicating simulation of this part of the transient, is the leakage rate through the "closed" MSCV which is not specified precisely in the experiment documentation. In the calculation, to account for steam leakage, the minimum MSCV flow area was restricted to 1.0% of its fully opened value. This corresponded to a leakage rate of ~ 0.12 kgs^{-1} at 7 MPa and is of the order implied by Reference 10. As shown in Figure 9, the overall trend in secondary system pressure history is reasonably well reproduced and, at the time at which the primary coolant pumps trip (1600 seconds), the predicted secondary side pressure is ~ 0.1 MPa greater than the measured value.

Calculations presented by members of the OECD LOFT Program Review Group also failed to reproduce precisely the experimental MSCV cycling behaviour and, in line with the TRAC-PF1/MOD1 calculation, slightly over-predicted the secondary system pressure just prior to pump trip (1600 seconds).

8.4.2 Break Mass Flow Rate, Break Line Density and Primary System Mass Inventory

A comparison of the measured and predicted break mass flow rates is shown in Figure 10. Although well-predicted for the first ~ 500 seconds of the transient, TRAC-PF1/MOD1 slightly over-predicts the break mass flow rate after this time, until the primary coolant pumps are tripped at 1600 seconds. Figure 11 indicates that the break line density is well reproduced during the mass depletion phase, suggesting that the slightly over-predicted break mass flow rate between ~ 500 seconds and 1600 seconds is a consequence of the TRAC critical flow model. Subcooled and two-phase choked flow multipliers of 1.0 were used throughout the calculation. (It appears that a multiplier of

~ 0.8 would have given the best agreement over the 500 seconds to 1600 seconds period.) The break mass flow rate predicted by TRAC-PF1/MOD1 (between ~ 500 seconds and 1600 seconds) is ~ 10% higher than that implied by the Homogeneous Equilibrium Model (12). Details are given in Appendix III. Since no ILCL stratification, prior to pump trip, is observed in the experiment, serious deficiencies in the TRAC-PF1/MOD1 branch offtake model (highlighted during analysis of the LP-SB-1 (7) and LP-SB-2 (8) experiments) are not manifested in this calculation.

Consistent with the over-predicted break mass flow rate, the rate at which the primary system mass inventory is predicted to decline is more rapid than that measured (see Figure 12). It is thought that the discrepancy between the measured and predicted initial primary system mass inventories, as shown in Figure 12, is caused by a difference in interpretation, between the experiment and the calculation, of what constitutes the "primary system". The predicted initial primary system mass inventory includes all system components except the secondary side of the steam generator component (component numbers 20, 21, 22 and 27), the steam line valve (component number 23) and the ECCS upstream of the valve in the accumulator line. A similar approach was adopted for the TRAC-PF1/MOD1 calculations of LP-SB-1 (7) and LP-SB-2 (8), in which the initial primary system mass inventories were over-predicted slightly (by ~ 150 kg). The difference between the initial mass inventory predicted for LP-SB-3 and those predicted for LP-SB-1 and LP-SB-2 results mainly from the inclusion, for LP-SB-3, of the broken loop steam generator and pump simulators and a larger part of the ECC line. The initial experimental mass inventory on the LP-SB-3 experiment data tape is only slightly (~ 100 kg) greater than that for LP-SB-1 and LP-SB-2 (13), despite the inclusion, in the LP-SB-3 experiment of the broken loop simulator section.

The break mass flow rates predicted by members of the OECD LOFT Program Review Group are, generally, in good agreement with the experimental data. These calculations used subcooled choked flow multipliers of 0.93 to 1.0 and two-phase choked flow multipliers of 0.7 to 0.81.

8.4.3 Primary System Densities

A comparison of the measured and predicted ILHL and ILCL densities are shown in Figures 13 and 14. The times at which TRAC-PF1/MOD1 predicts stratification to occur are also indicated. The ILHL density is very well-predicted by the TRAC-PF1/MOD1 calculation; the density in the ILCL is slightly over-predicted. Stratification is predicted in both legs when the primary coolant pumps are tripped at 1600 seconds.

The calculations performed by members of the OECD LOFT Program Review Group (11) consistently over-predicted the ILCL density and, to a lesser extent, the ILHL density.

8.4.4 Pumps Degradation

The measured and predicted differential pressures across the primary coolant pumps are shown in Figure 15. The differential pressure decreases steadily after fluid saturation conditions are reached due to void formation in the primary system. During the experiment, oscillations in the pump differential pressure - associated with a change from symmetric flow through both pumps to asymmetric flow through only one pump - were observed between ~ 700 and ~ 875 seconds (10). TRAC-PF1/MOD1, in common with other codes (11), was not able to reproduce this situation. The pump characteristics employed in the LP-SB-3 input deck were those reported for the TRAC-PF1/MOD1 calculation of LP-SB-2 (8 (calculation "A")). In the experiment, pump degradation was considered to have occurred at ~ 875 seconds; TRAC-PF1/MOD1 predicted this to occur between ~ 700 seconds and ~ 1100 seconds. The calculations performed by members of the OECD LOFT Program Review Group (11) tended to over-predict the time at which the pumps degraded (by up to ~ 700 seconds).

8.5 Core Boil-Off Phase

The core boil-off phase is defined as that part of the transient between pump trip (1600 seconds) and steam generator feed and bleed initiation (5415 seconds - experiment, 4964 seconds - calculation).

8.5.1 Primary and Secondary System Pressures

As shown in Figure 7, following primary coolant pump trip, the primary system pressure history is well-predicted prior to the time at which the break is isolated (4742 seconds - experiment, 4688 seconds - calculation). Figure 16 shows the measured and predicted differences between primary and secondary system pressures. Figure 17 shows the primary and secondary system pressure histories as predicted and as measured. Although the trends in primary and secondary system pressure histories are reasonably well reproduced by the TRAC-PF1/MOD1 calculation, the measured and predicted results (most noticeably for the secondary system) appear to diverge slightly after ~ 3500 seconds. The predicted difference between the primary and secondary system pressures, immediately prior to break isolation, is significantly larger than that measured. Following break isolation, the primary side pressure rises but is predicted to remain below that of the secondary system until the secondary side feed and bleed is initiated. This represents an important departure from the experiment in which, following break isolation and prior to feed and bleed initiation, there was a ~ 500 second period when the primary system pressure was above that of the secondary system during which reflux condensation could occur. This difference between the measured and predicted relative primary and secondary system pressures has important consequences for the rate at which the core heats up, as discussed in Section 8.5.4.

8.5.2 Break Mass Flow Rate and Primary System Mass Inventory

Following pump trip (at 1600 seconds) the break uncovers and, as shown in Figure 10, the break mass flow rate is well-predicted up until the time at which the break is isolated (4742 seconds - experiment, 4688 seconds - calculation).

As shown in Figure 12, the residual primary system mass inventory, predicted by TRAC-PF1/MOD1, at the time of pump trip (1600 seconds) is in good agreement with the experimental value of ~ 2800 kg. It should be recognised, however, that the agreement is, to some extent, fortuitous because the over-predicted break mass flow rate prior to this time and the over-predicted initial inventory act as compensating errors. The residual mass (~ 1500 kg) at the time at which the break is isolated is well-predicted and reflects the accurate reproduction of the break mass flow rate following pump trip. In line with the experimental procedure, primary pump injection was turned off in the calculation following pump trip; the primary system mass inventory therefore remained constant following break isolation until the accumulator flow was initiated.

8.5.3 Primary System Densities

Following pump trip, the liquid in the system falls to the lowest parts of the circuit, ie the reactor pressure vessel and the loop seals. As shown in Figure 13, the hot leg density rises immediately after the pumps are tripped due to liquid draining back from the "up-side" of the steam generator tubes. In the experiment, the instrument measuring the loop seal density (positioned in the piping leading from the steam generator outlet) becomes immersed in liquid following pump trip and remains so until the end of the transient. In the calculation, the presence of liquid is not predicted in this part of the circuit until ~ 3200 seconds (although liquid is present in the "pump side" of the loop seal prior to this time) - see Appendix II. Reasons for this discrepancy have not been fully investigated. However, factors which might adversely influence the predicted behaviour have been identified, ie incorrect pressure imbalance across the loop seal and insufficiently detailed nodalisation of the vertical sections of the loop seal. In general, the calculations submitted by members of the OECD LOFT Program Review Group (11) correctly predicted the loop seal density following pump trip (but over-predicted the density prior to pump trip).

8.5.4 Fuel Cladding Temperatures

The predicted peak fuel rod cladding temperatures (at each of the six levels implied by the TRAC-PF1/MOD1 core noding scheme) are compared with the experimental data (at the nearest corresponding elevation) in Figures 18-23. A fuel rod power peaking factor of 1.6 (as implied in an AEEW input deck for LOFT test L2-6) was assigned to the peak rated rod in the TRAC-PF1/MOD1 calculation. The following observations are made:

(i) The time at which the core starts to heat up is over-predicted by ~ 260 seconds (see Figure 23). The times at which the break is isolated (peak clad temperature of 835.2 K) and the steam generator feed and bleed is initiated (peak clad temperature of 988 K) are under-predicted by ~ 50 seconds and ~ 450 seconds, respectively.

(ii) The predicted rate at which the core heats up is greater than that measured. Agreement is considerably better at the top of the core (see Figure 23) than at lower elevations and agreement worsens following break isolation (see, for example, Figure 20).

(iii) As measured experimentally, no heat-up is predicted at the bottom of the core (see Figure 18). The heat-up at the 0.28 m elevation, observed in the experiment, was not predicted by the TRAC-PF1/MOD1 calculation in the cell representing elevations from 0.15 m to 0.42 m (see Figure 19).

The above observations are consistent with those presented, by members of the LOFT Program Review Group, in the LP-SB-3 "Comparison Report" (11). It is widely reported (11, 14, 15) that an important factor, contributing to the over-predicted rates of core heat-up, is the inability of the codes, using a 1-dimensional vessel representation, to correctly predict the distribution, throughout the core, of reflux condensation returning to the vessel. However, in the TRAC-PF1/MOD1 analysis, an important discrepancy exists between the measured and predicted differences in primary and secondary pressure. During the core heat-up phase in the calculation, the primary side pressure is below that of the secondary and it is unlikely, therefore, for the presence of reflux condensation to be predicted. It is suggested, therefore, that in the TRAC-PF1/MOD1 calculation, it is the lack of reflux condensation, rather than its poorly predicted distribution throughout the core, that might lead to the heat-up rate being over-predicted. This certainly appears to be the case following break isolation when, in the experiment, the primary side pressure becomes greater than that of the secondary and there is a significant reduction in the rate at which the core continues to heat up. In the calculation, the primary side pressure does not exceed that of the secondary until the steam generator feed and bleed is initiated and no reduction in the rate of heat up is predicted following break isolation.

Although not fully investigated, it is thought that the tendency of TRAC-PF1/MOD1 to under-predict the core density, during certain conditions, due to deficiencies in its interphase drag model (16) could result in the core drying out at a greater rate than that measured and might contribute to the rate of core heat-up being over-predicted.

8.6 Core Cool-Down and Recovery Phase

The core cool-down and recovery phase is defined as that part of the transient between steam generator feed and bleed initiation

(5415 seconds - experiment, 4964 seconds - calculation) and experiment termination (6845 seconds).

8.6.1 Primary and Secondary System Pressures

As shown in Figure 9, steam generator feed and bleed initiation results in a rapid decrease in secondary system pressure. The corresponding decrease in primary system pressure is shown in Figure 7. The rates at which the primary and secondary system pressures are predicted to decline are slightly under-predicted. Sensitivity of the pressure histories to the size assumed for the secondary side steam bypass valve, used during feed and bleed operation, and the consequent effect on the accumulator flow rate, is discussed in Section 8.6.3.

8.6.2 Fuel Cladding Temperatures

Following steam generator feed and bleed initiation, the core takes slightly longer to quench in the calculation then in the experiment. As shown in Figures 20, 21 and 22, the enhanced rate with which the core is predicted to cool, following the commencement of accumulator flow (~ 5300 seconds) is very well reproduced. However, in the calculation, as a result of the slightly under-predicted primary side depressurisation rate, there is a relatively longer period (approximately twice as long) between feed and bleed initiation and accumulator flow commencement (during which the rate of cooling is less rapid).

8.6.3 Accumulator Injection Flow

Accumulator flow is predicted to commence ~ 280 seconds earlier than in the experiment. As shown in Figure 24, its behaviour is reasonably well reproduced. The oscillatory accumulator flow, reported in other calculations (11), is not observed in the TRAC-PF1/MOD1 analysis. Previously reported peaks in the accumulator flow rate predicted by TRAC-PF1/MOD1 were found to be the result of incorrect mass flow rates being attributed, by TRAC, to "signal variables" used for plotting. (The problem arises when the void fraction in the bottom cell of the accumulator is $\gtrsim 0.0$).

At ~ 6400 seconds, accumulator nitrogen was predicted to have entered the system and the calculation was terminated. Finer nodalisation of the bottom of the accumulator (to ensure $\alpha = 0.0$ in the bottom cell) could have prevented this situation.

The size of the steam bypass valve used during the feed and bleed operation was not specified in the available documentation. The effect, on the secondary side depressurisation rate and the accumulator flow rate, of increasing the valve area by ~ 70% (from 2.46×10^{-4} m^2 to 4.093×10^{-4} m^2) is shown in Figures 25 and 26. The larger flow area produced a more rapid secondary side depressurisation and, consequently, a greater accumulator flow rate, in better agreement with the experimental data.

9 CONCLUSIONS

The main findings of the TRAC-PF1/MOD1 analysis of the OECD LOFT experiment LP-SB-3 were as follows:

(i) The calculation ran with a CPU to real time ratio of 0.73 and an average time step of 0.32 seconds.

(ii) Despite the break line density being well-predicted, TRAC-PF1/MOD1 slightly over-predicted the break mass flow rate prior to pump trip. This is thought to be a consequence of deficiencies in the TRAC critical flow model - the predicted break mass flow rate during this period was ~ 10% greater than that implied by the Homogeneous Equilibrium Model. In general, the calculations reported (by members of the OECD LOFT Program Review Group) in the LP-SB-3 "Comparison Report" correctly reproduced the experimental break mass flow rates by applying appropriate break flow multipliers (of $<$ 1.0). (Subcooled and two-phase choked flow multipliers of 1.0 were used throughout the TRAC-PF1/MOD1 calculation).

(iii) Overall trends with respect to primary and secondary system pressure histories were reasonably well-predicted. Differences between the experiment and the calculation in the relative magnitudes of the primary and secondary system pressures, however, resulted in conditions for reflux condensation not being predicted during the core heat-up phase of the transient. This was contrary to the experimental behaviour and was thought to contribute - particularly following break isolation - to the predicted rate of core heat-up being significantly greater than that observed experimentally. Although not fully investigated, under-prediction of the core density, due to deficiencies in the TRAC-PF1/MOD1 interphase drag model, has been identified as a potential contributor to the poorly predicted rate with which the core temperature rises. The calculations presented in the LP-SB-3 "Comparison Report" also significantly over-predicted the rate of core heat-up.

(iv) Minimum primary system mass inventory was reasonably well reproduced. As observed in the experiment, no uncovery of the bottom of the core was predicted by the TRAC-PF1/MOD1 calculation.

(v) The predicted time between steam generator feed and bleed initiation and the core being quenched was slightly longer than that observed experimentally. The main reason for this was thought to be a delay, relative to the experiment, in the time at which the accumulator flow was predicted to commence. This was caused by the predicted rate of depressurisation during feed and bleed operation being slower than that measured and was shown to be rectified when the flow area of the steam bypass valve used during the feed and bleed operation (not specified in the available documentation) was increased.

(vi) During the core recovery phase, the mass flow rate from the accumulator was reasonably well reproduced. Severe oscillations in the accumulator flow, reported in the calculations performed by members of the OECD LOFT Program Review Group, were not observed in the TRAC-PF1/MOD1 calculation.

(vii) Accumulator nitrogen was predicted to have entered the circuit at ~ 6400 seconds and the calculation was terminated at this time, ie before initiation of the Low Pressure Injection System. The bottom cell of the accumulator was partially full at this time and finer nodalisation of the bottom of the accumulator could have prevented the premature ingress of nitrogen.

The calculation reported should be considered a "Base Case", the results of which have not been exhaustively analysed. With the exception of the rate at which the rod cladding temperatures rose during the core uncovery phase of the transient, agreement with the experimental data was considered reasonable. A fuller investigation of factors influencing the rate with which the core heats up (including the effect of deficiencies in the TRAC-PF1/MOD1 interphase drag model) would be desirable.

10 ACKNOWLEDGEMENTS

The guidance of Dr C G Richards during the course of this work is gratefully acknowledged.

11 REFERENCES

1 REEDER D L, LOFT System and Test Description, July 1978, (TREE-1208, NUREG/CR-0247).

2 KEE K J, SCHALLY P J, WINTERS L, Base Input for LOFT RELAP5 Calculations, July 1980 (EGG-LOFT-5199).

3 GRUSH W H, TANAKA M, MARSILI P, Best-Estimate Prediction for the OECD LOFT Project Small Cold Leg Break Experiment LP-SB-3, November 1983 (OECD LOFT-T-3005).

4 ALLEN E J, Internal Document.

5 ALLEN E J, Internal Document.

6 NEILL A P, Internal Document.

7 ALLEN E J, TRAC-PF1/MOD1 Post-Test Calculations of the OECD LOFT Experiment LP-SB-1 (AEEW - R 2254).

8 PELAYO F, TRAC-PF1/MOD1 Post-Test Calculations of the OECD LOFT Experiment LP-SB-2, March 1987 (AEEW - R 2202).

9 McPHERSON G D, Decay Heat Data for OECD LOFT Experiments, October 1985 (Letter to OECD LOFT Programme Review Group Members).

10 ALEMBERTI A, PROTO G, Experiment Analysis and Summary Report on OECD LOFT Nuclear Experiment LP-SB-3 (DRAFT), December 1985 (T-20000-XX-140-020).

11 ALEMBERTI A, Comparison Report on OECD LOFT Experiment LP-SB-3, August 1986 (T-20000-XX-140-021).

12 HALL, CZAPARY, Tables of Homogeneous Equilibrium Flow Parameters for Water in SI Units, September 1980 (EGG-2056).

13 MODRO S M, FELDMAN E M, NEWMAN N and PETERS M D, Experiment Analysis and Summary Report on OECD LOFT Nuclear Experiments LP-SB-1 and LP-SB-2, May 1984 (OECD LOFT-T-3205).

14 HARWOOD C, BROWN G, RELAP5/MOD2 Calculation of OECD LOFT Test LP-SB-03, April 1986 (GD/PE-N/535).

15 GUNTAY S, RELAP5/MOD2 Assessment: OECD-LOFT Small Break Experiment LP-SB-3, Gemeinschaftsberict Nr 13, April 1986.

16 ARDRON K H, CLARE A J, Assessment of Interphase Drag Calculations in the RELAP5/MOD2 and TRAC-PF1/MOD1 Thermal-Hydraulic Codes, November 1986 (GD/PE-N/557 (Revised)).

17 Safety Code Development Group, Energy Division, LANL, TRAC-PF1/MOD1: An Advanced Best-Estimate Computer Program for Pressurised Water Reactor Thermal-Hydraulic Analysis, July 1986, (NUREG/CR-3858, LA-10157-MS).

TABLE 1

INITIAL CONDITIONS FOR EXPERIMENT LP-SB-3

	Measured (10)		Predicted
PRIMARY COOLANT SYSTEM			
Power Level (MW)	50.3	± 1.2	50.3
Primary Coolant Mass Flow Rate (kgs^{-1})	482.6	± 2.6	482.2
Pressurizer Pressure (MPa)	15.26	± 0.11	15.20
Pressurizer Liquid Level (m)	1.115	± 0.06	1.15
ILCL Temperature (K)	556.6	± 1	558.1
ILHL Temperature (K)	576.6	± 1	577.6
STEAM GENERATOR SECONDARY SIDE			
Liquid Level (m)	0.21	± 0.005	0.21
Pressure (MPa)	5.58	± 0.06	5.57
Liquid Temperature (K)	535.6	± 3.6	536.0
Mass Flow Rate (kgs^{-1})	26.67	± 0.77	26.40
ECCS - ACCUMULATOR A			
Pressure (MPa)	2.84	± 0.1	2.84
Temperature (K)	303.7	± 6	303.7
Liquid Level (m)	2.04	± 0.01	2.04

TABLE 2

OPERATIONAL SETPOINTS FOR EXPERIMENT LP-SB-3

Action	Measured During Experiment (10)		Specified to TRAC-PF1/MOD1 Calculation	
	Reference	Setpoint	Reference (component/ cell no)	Setpoint
Small Break Valve Opened	Time	0 secs	Time	0 secs
Reactor Scrammed	ILHL Pressure	14.19 MPa	ILHL Pressure (1/4)	14.19 MPa
Main Feedwater Shut Off	ILHL Pressure	14.19 MPa	ILHL Pressure (1/4)	14.19 MPa +0.2 secs
MSCV Begins to Close	Secondary System Pressure	6.35 MPa	Initially:- ILHL Pressure (1/4)	14.19 MPa +0.29 secs
			During MSCV Cycling:- Secondary Pressure (21/2)	6.35 MPa
MSCV Begins to Open	Secondary System Pressure	7.09 MPa	Secondary Pressure (21/2)	7.09 MPa
Primary Coolant Pumps Tripped	Primary Coolant System Mass Inventory Remaining	2800* kg	Time	1600 secs
Break Valve Closed	Maximum Cladding Temperature	835.2K	Maximum Cladding Temperature of Supplementary Rod	835.2K
Steam Bypass Valve Opened	Maximum Cladding Temperature	988K	Maximum Cladding Temperature of Supplementary Rod	988.0K
Main Feedwater Initiated	Maximum Cladding Temperature	988K	Maximum Cladding Temperature of Supplementary Rod	988.0K
Accumulator Injection Initiated	ILHL Pressure	2.79 MPa	ILHL Pressure (1/4)	2.79 MPa
LPIS Initiated	ILHL Pressure	1.03 MPa	ILHL Pressure (1/4)	1.03 MPa

* It had been intended to trip the primary coolant pumps when 2000 kg of primary mass inventory remained.

TABLE 3

CHRONOLOGY OF EVENTS FOR EXPERIMENT LP-SB-3

Event	Time After Experiment Initiation (seconds)	
	Measured (10)	Predicted
Small break valve opened	0.0	0.0
Reactor Scrammed	9.21 ± 0.01	11.22
Main feedwater pump shut off on scram signal signal	9.41 ± 0.01	11.42
MSCV started to close	9.5 ± 0.2	11.52
Main feedwater valve isolated	10.81 ± 0.01	12.22
MSCV closed	21 ± 0.2	24.7
Pressurizer level below indicating range	67 ± 3	84.4[1]
First time MSCV cycled opened	87.5 ± 0.2	26.7
Subcooled blowdown ended	98.5 ± 0.5	100
Pumps differential pressure degradation observed	875 ± 3	700 - 1100[2]
Last time MSCV cycled open	1030 ± 0.2	124.4
Primary coolant pumps tripped	1600 ± 2	1600
Break uncovered	1612 ± 5	1624
Primary coolant pumps decoupled from flywheels	1626 ± 1	1650
Start of core heat up	3800 ± 50	4060[3]
Break isolated	4742 ± 2	4688
Steam generator feed and bleed initiated	5415 ± 5	4964
Maximum cladding temperature reached	5422 ± 1	4980
Accumulator injection initiated	5588 ± 3	5308
Core quenched	5800 ± 50	5560[4]
LPIS initiated	6785 ± 2	-
Experiment terminated	6845 ± 2	-

[1] Defined as the time at which the liquid level in the pressurizer < 0.01 m.

[2] As implied by Figure 15.

[3] Defined as the time at which $T_{max\ peak\ rod} - T_{sat} > 2K$.

[4] Defined as the time at which $T_{max\ peak\ rod} - T_{sat} < 2K$.

APPENDIX I

MICROFICHE LISTING OF THE TRAC-PF1/MOD1 INPUT DECK

FOR LP-SB-3

The microfiche can be found on the inside of the back cover of this report.

APPENDIX II

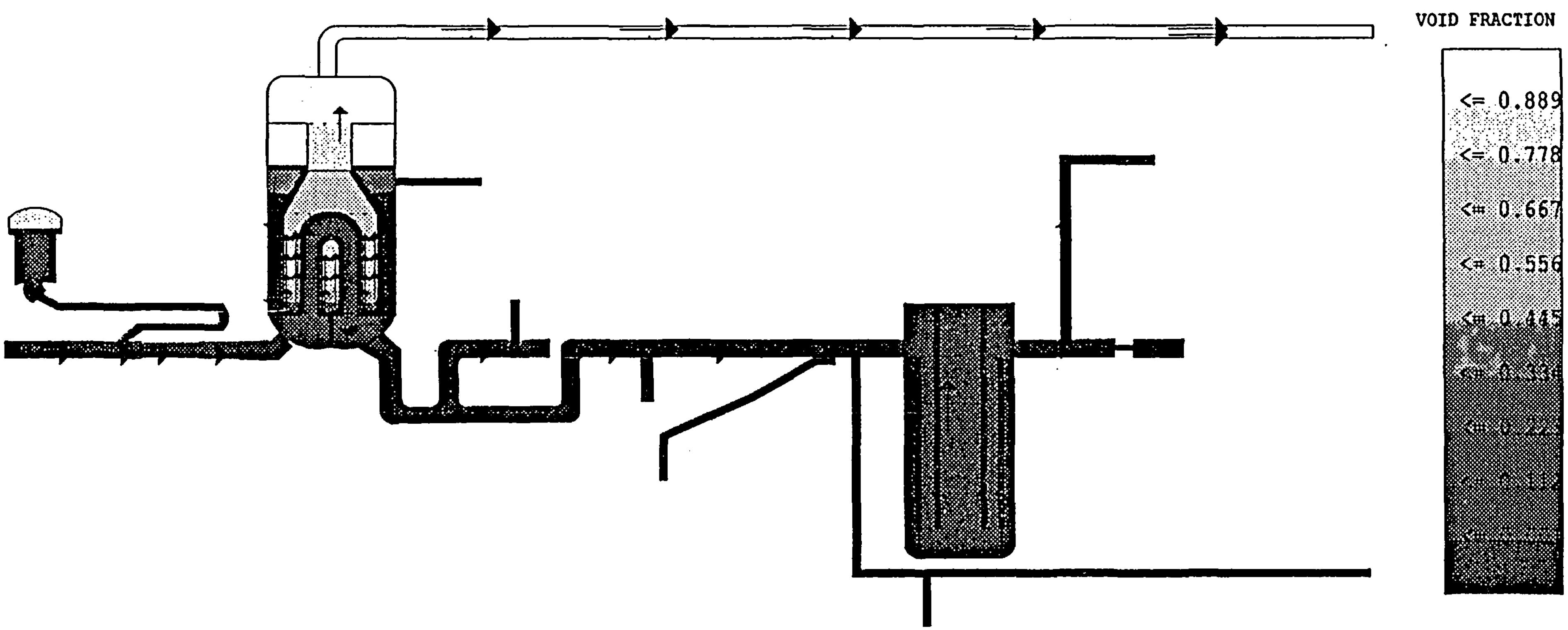
SERIES OF PICTURES SHOWING PREDICTED SYSTEM CONDITIONS THROUGHOUT THE LP-SB-3 TRANSIENT

The following series of pictures shows, at selected intervals, the predicted void fraction distribution, the liquid and vapour velocities and the occurrences of stratified flow conditions throughout the system. The nodalisation may be compared with that of Figures 2, 3 and 4.

S.M.A.R.T. SYSTEM MIMIC FOR ANALYSIS OF REACTOR TRANSIENTS

TITLE OF FRAME:- LOFT TEST LP-SB-3 TRAC CALCULATION

Ti 0.0
VLIQ
(10.0)
VVAP
(10.0)



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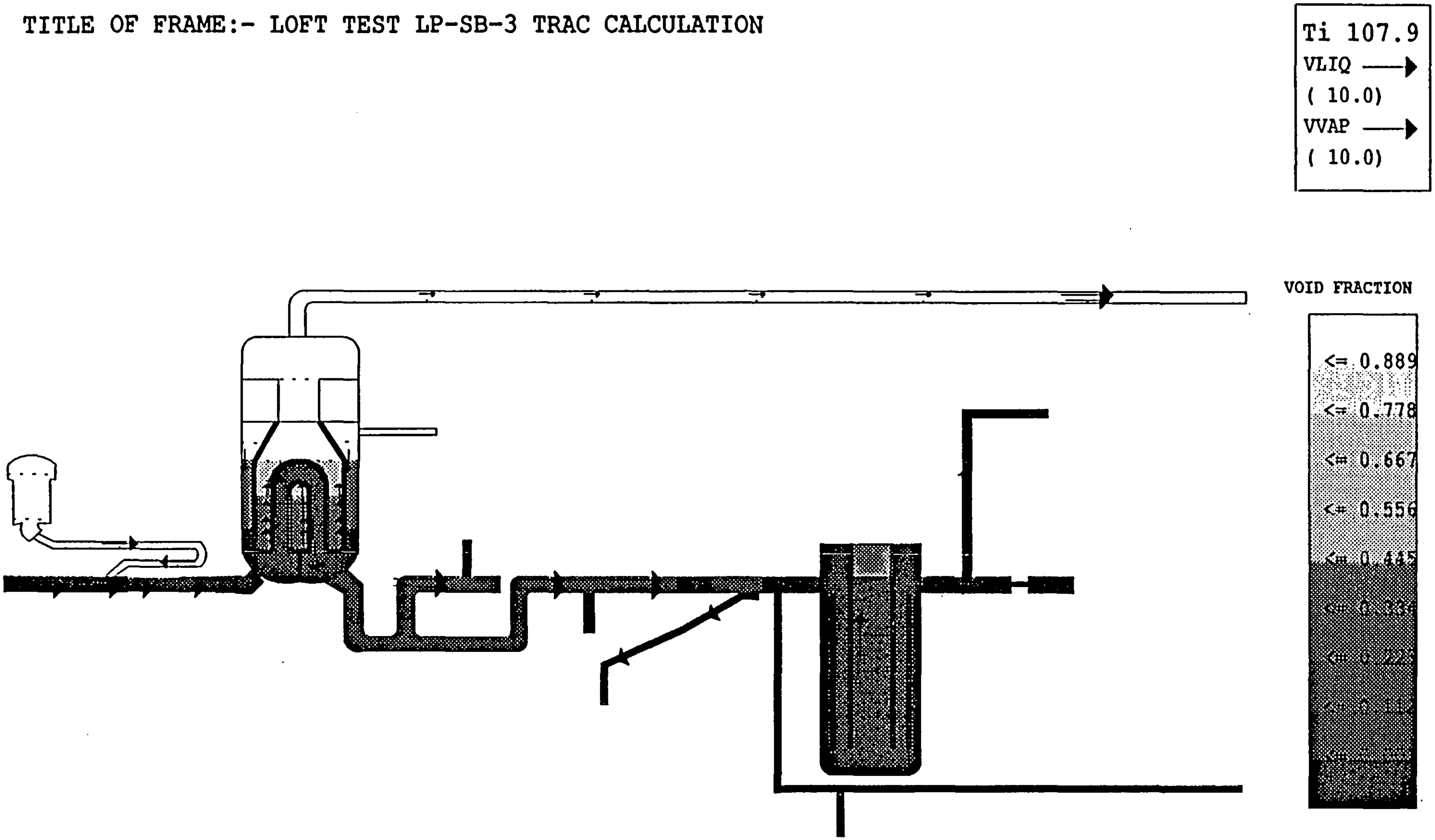
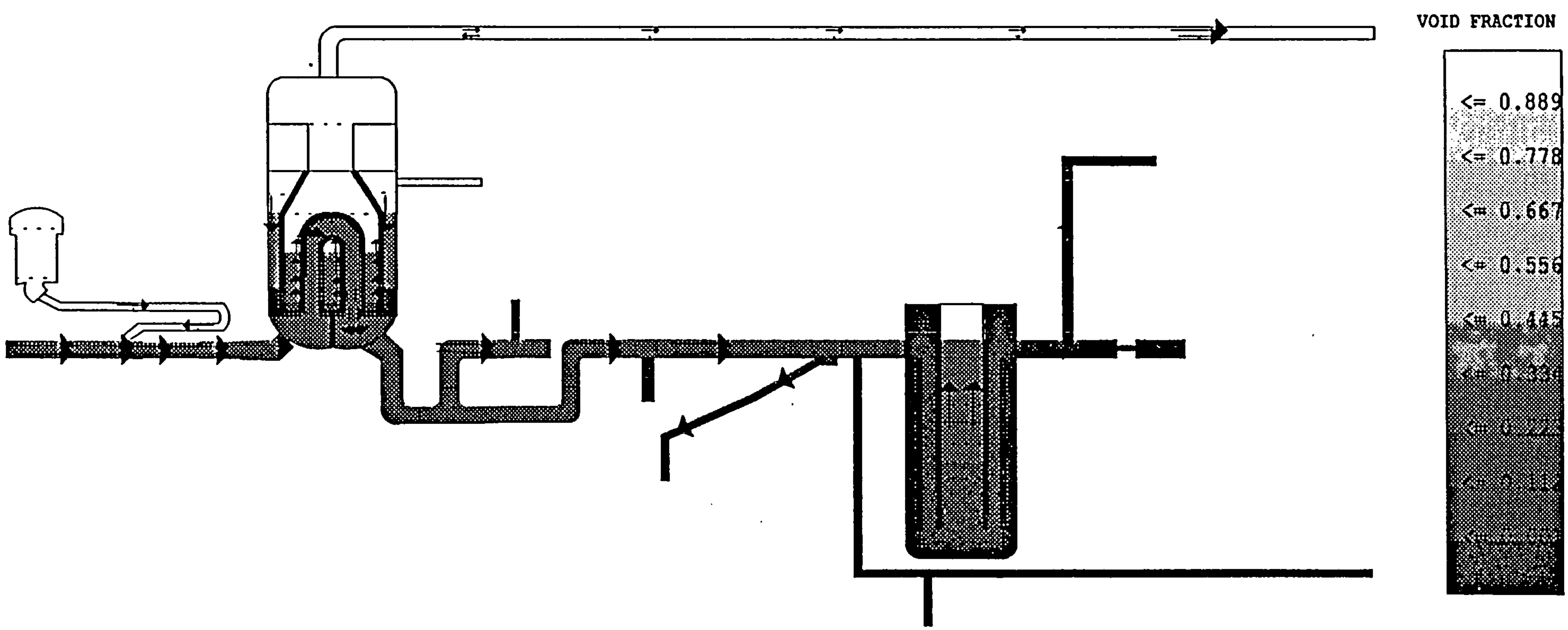
TITLE OF FRAME:- LOFT TEST LP-SB-3 TRAC CALCULATION



S.M.A.R.T. SYSTEM MIMIC FOR ANALYSIS OF REACTOR TRANSIENTS

TITLE OF FRAME:- LOFT TEST LP-SB-3 TRAC CALCULATION

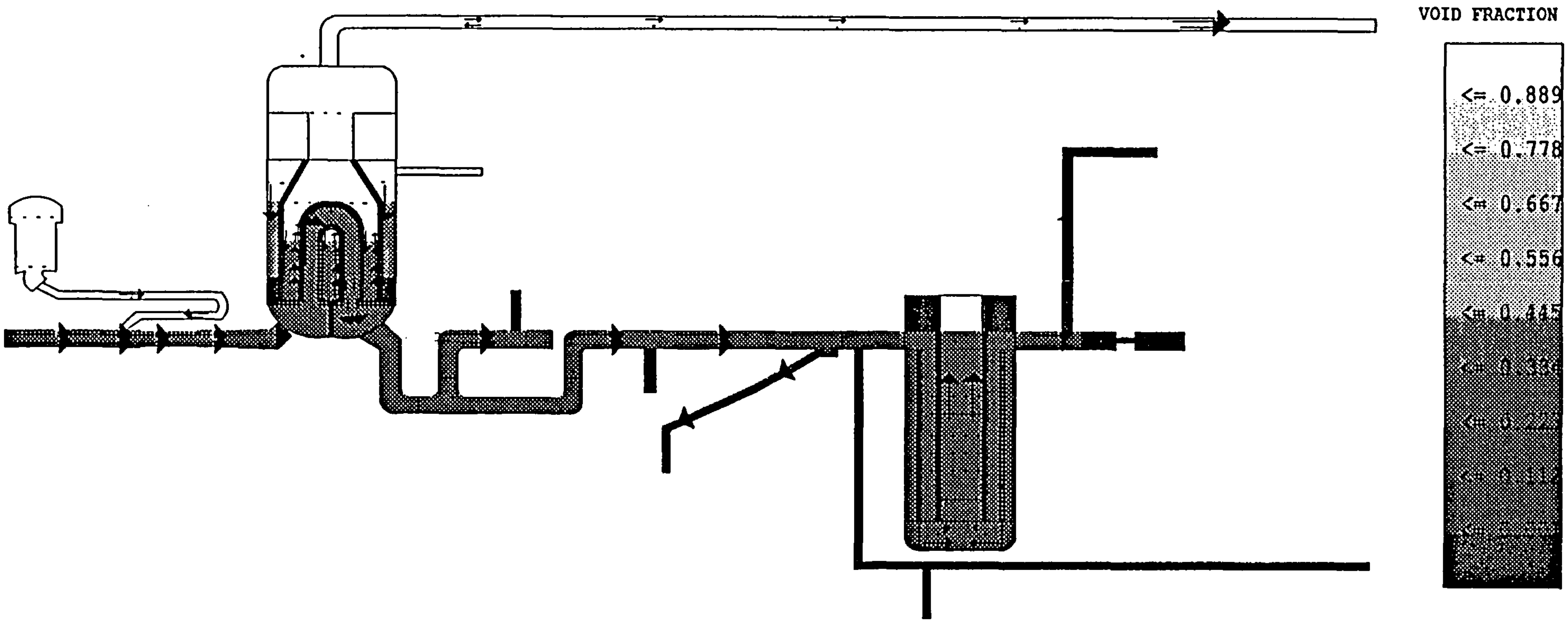
Ti 210.4
VLIQ →
(10.0)
VVAP →
(10.0)



S.M.A.R.T. SYSTEM MIMIC FOR ANALYSIS OF REACTOR TRANSIENTS

TITLE OF FRAME:- LOFT TEST LP-SB-3 TRAC CALCULATION

Ti 300.6
VLIQ →
(10.0)
VVAP →
(10.0)



S.M.A.R.T. SYSTEM MIMIC FOR ANALYSIS OF REACTOR TRANSIENTS

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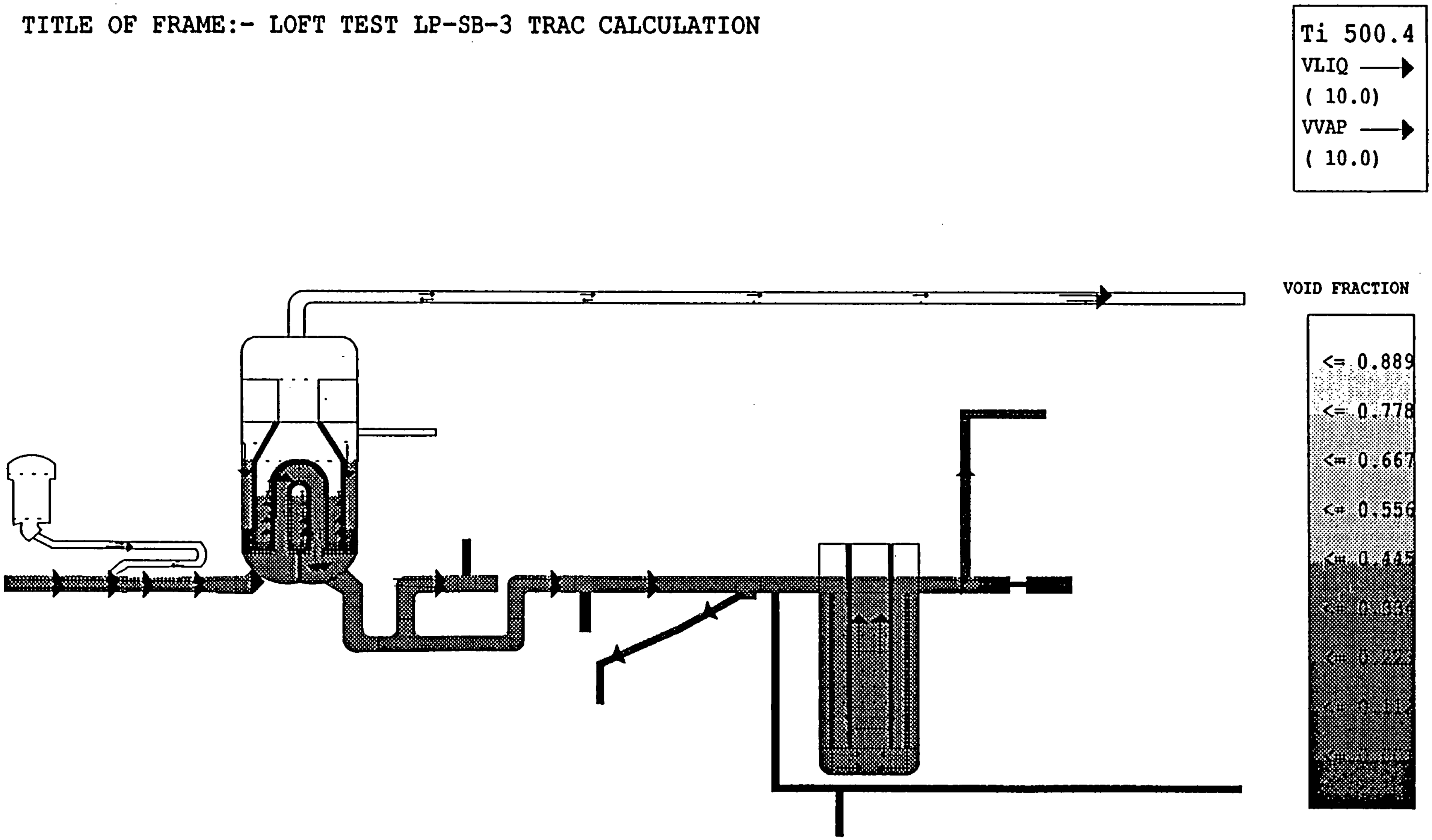

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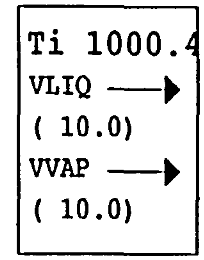
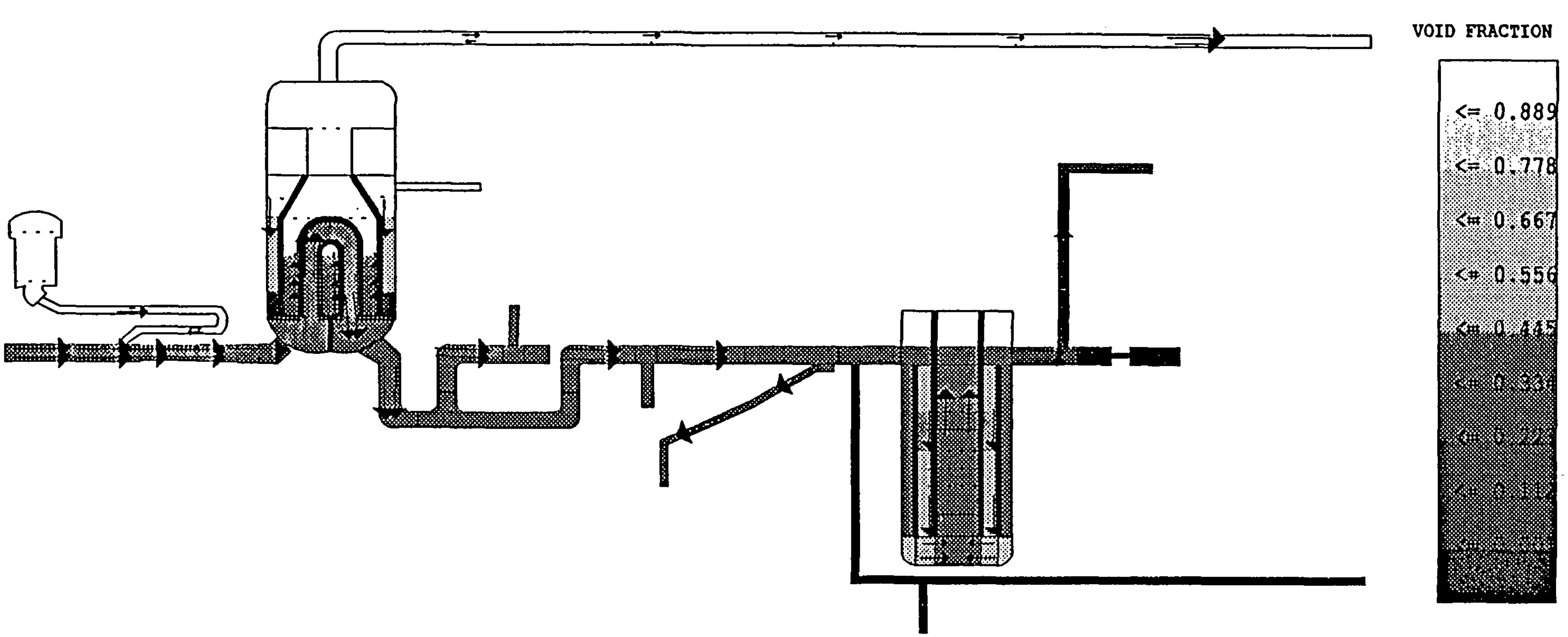
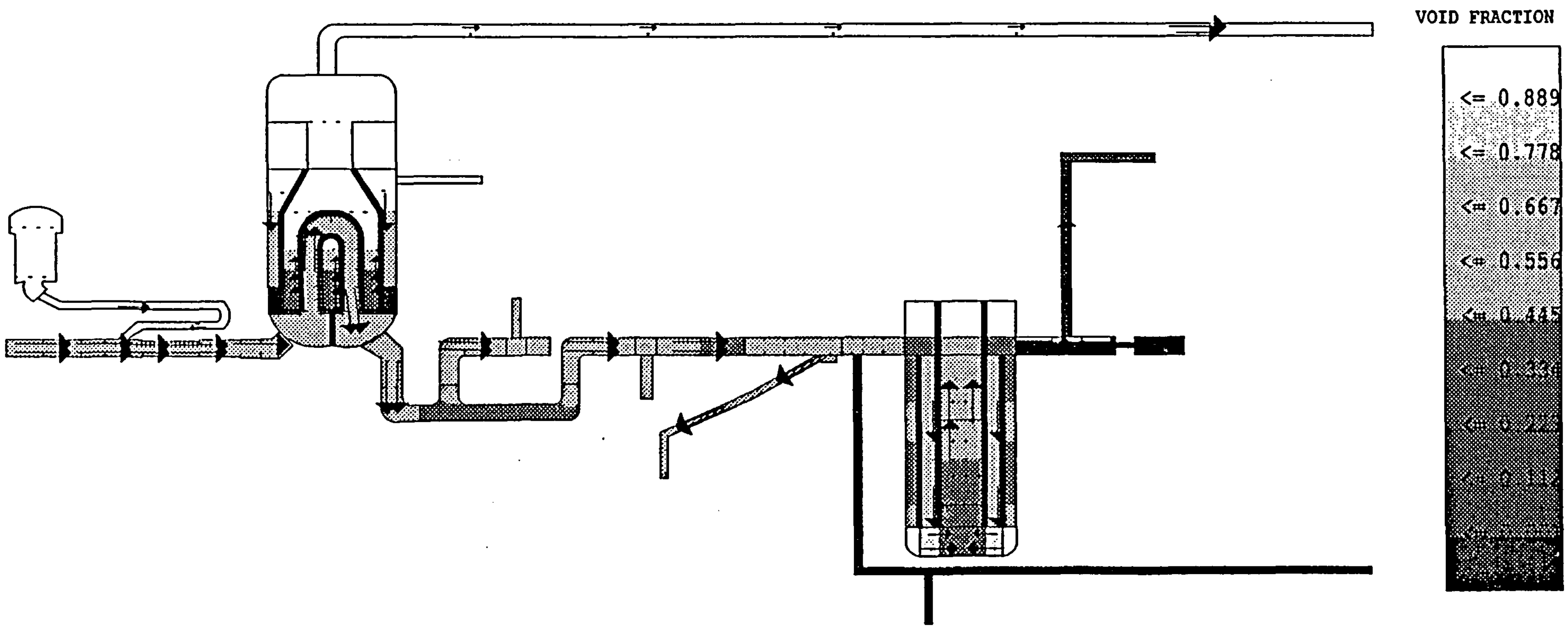




S.M.A.R.T. SYSTEM MIMIC FOR ANALYSIS OF REACTOR TRANSIENTS

TITLE OF FRAME:- LOFT TEST LP-SB-3 TRAC CALCULATION

Ti 1513.4
VLIQ
(10.0)
VVAP
(10.0)



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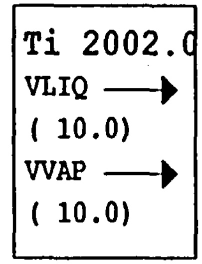
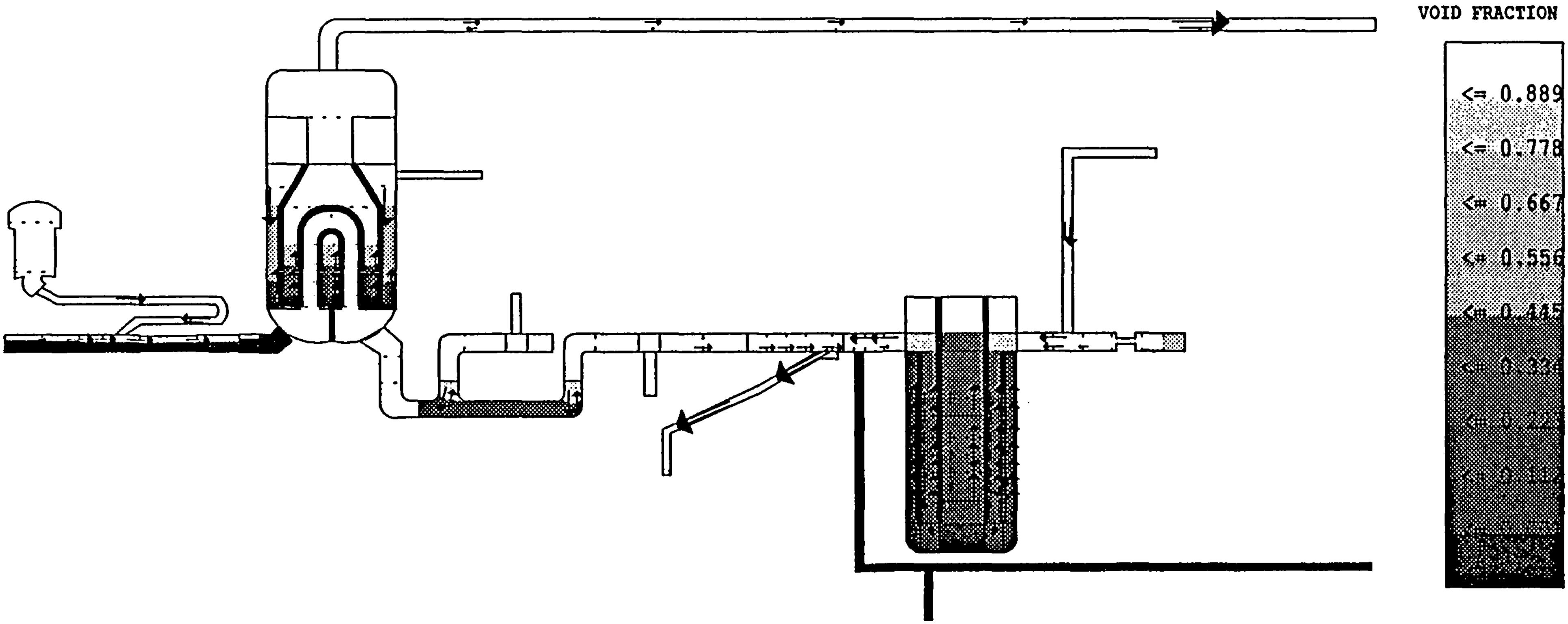
TITLE OF FRAME:- LOFT TEST LP-SB-3 TRAC CALCULATION

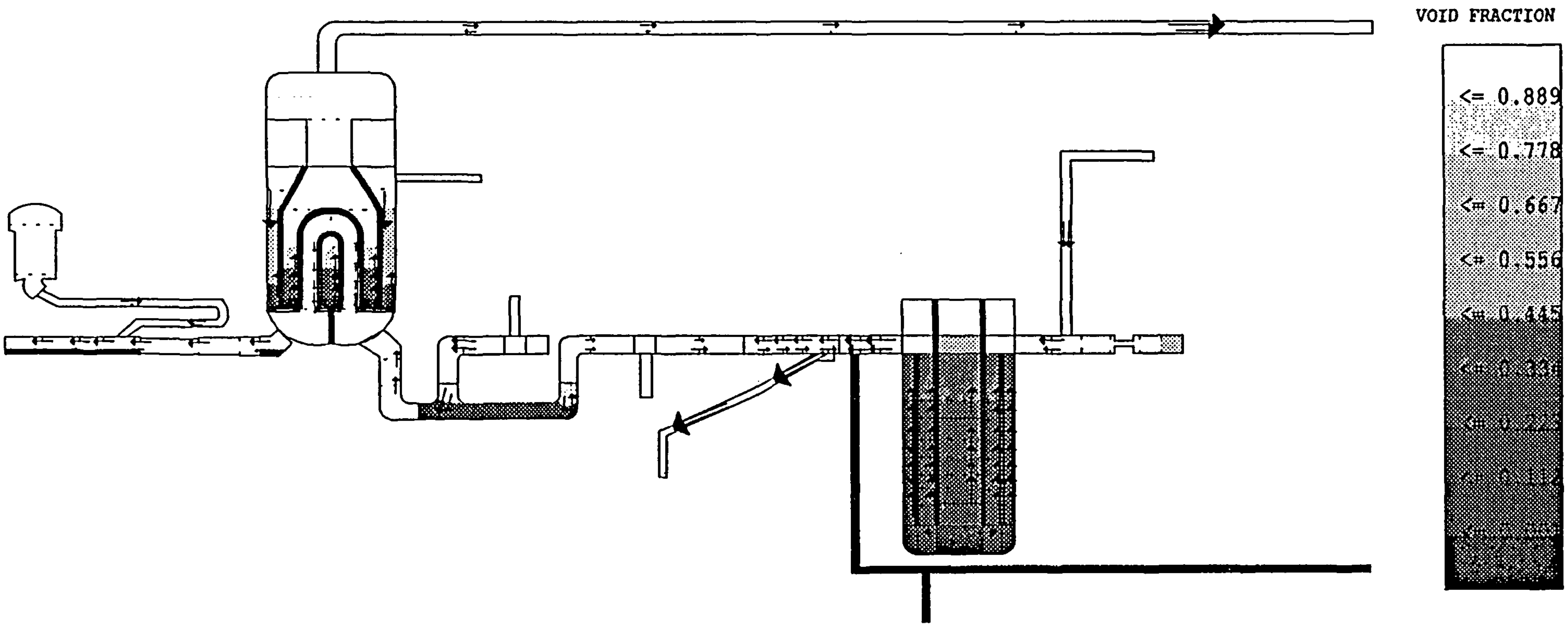




S.M.A.R.T. SYSTEM MIMIC FOR ANALYSIS OF REACTOR TRANSIENTS

TITLE OF FRAME:- LOFT TEST LP-SB-3 TRAC CALCULATION

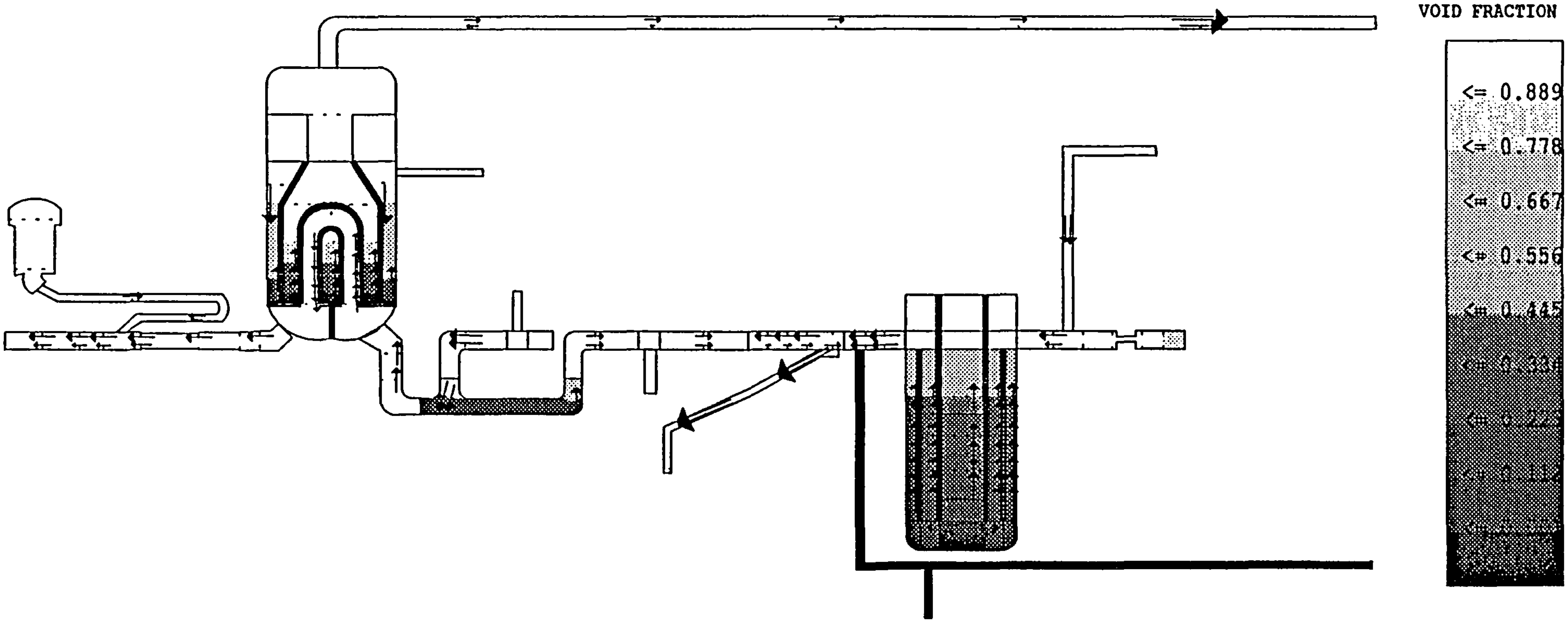
Ti 2514.8
VLIQ
(10.0)
VVAP
(10.0)



S.M.A.R.T. SYSTEM MIMIC FOR ANALYSIS OF REACTOR TRANSIENTS

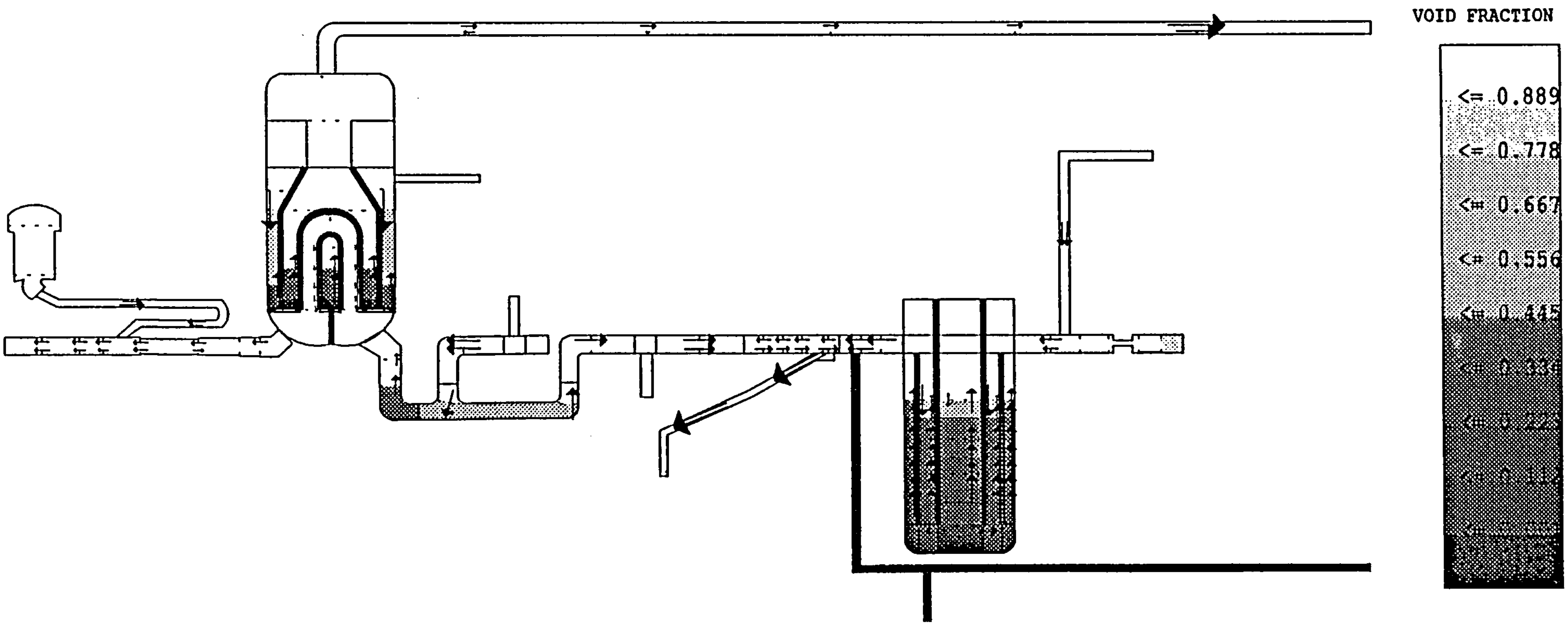
TITLE OF FRAME:- LOFT TEST LP-SB-3 TRAC CALCULATION

Ti 3024.8
VLIQ
(10.0)
VVAP
(10.0)



S.M.A.R.T. SYSTEM MIMIC FOR ANALYSIS OF REACTOR TRANSIENTS
TITLE OF FRAME:- LOFT TEST LP-SB-3 TRAC CALCULATION

Ti 3505.2
VLIQ
(10.0)
VVAP
(10.0)



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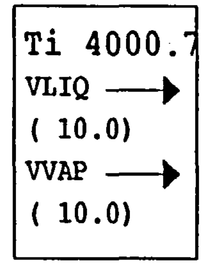
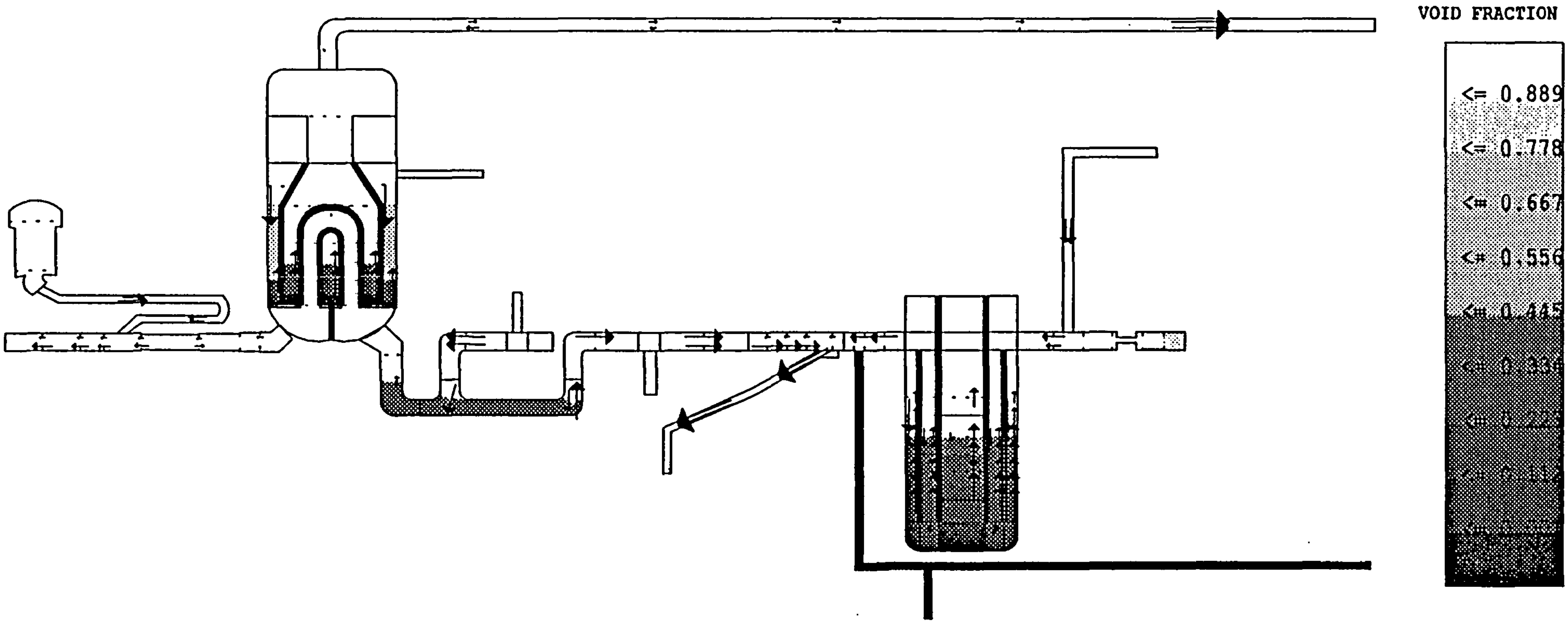




S.M.A.R.T. SYSTEM MIMIC FOR ANALYSIS OF REACTOR TRANSIENTS

TITLE OF FRAME:- LOFT TEST LP-SB-3 TRAC CALCULATION

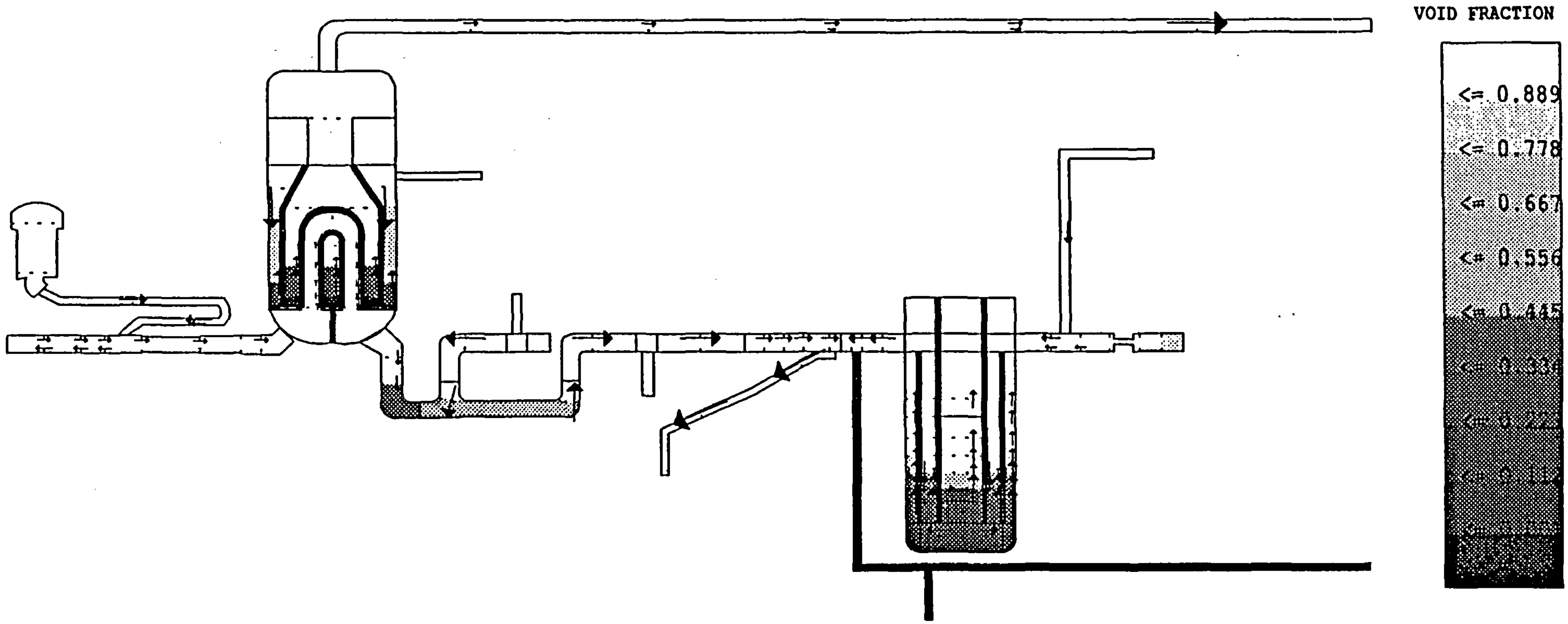
Ti 4515.0

VLIQ →

(10.0)

VVAP →

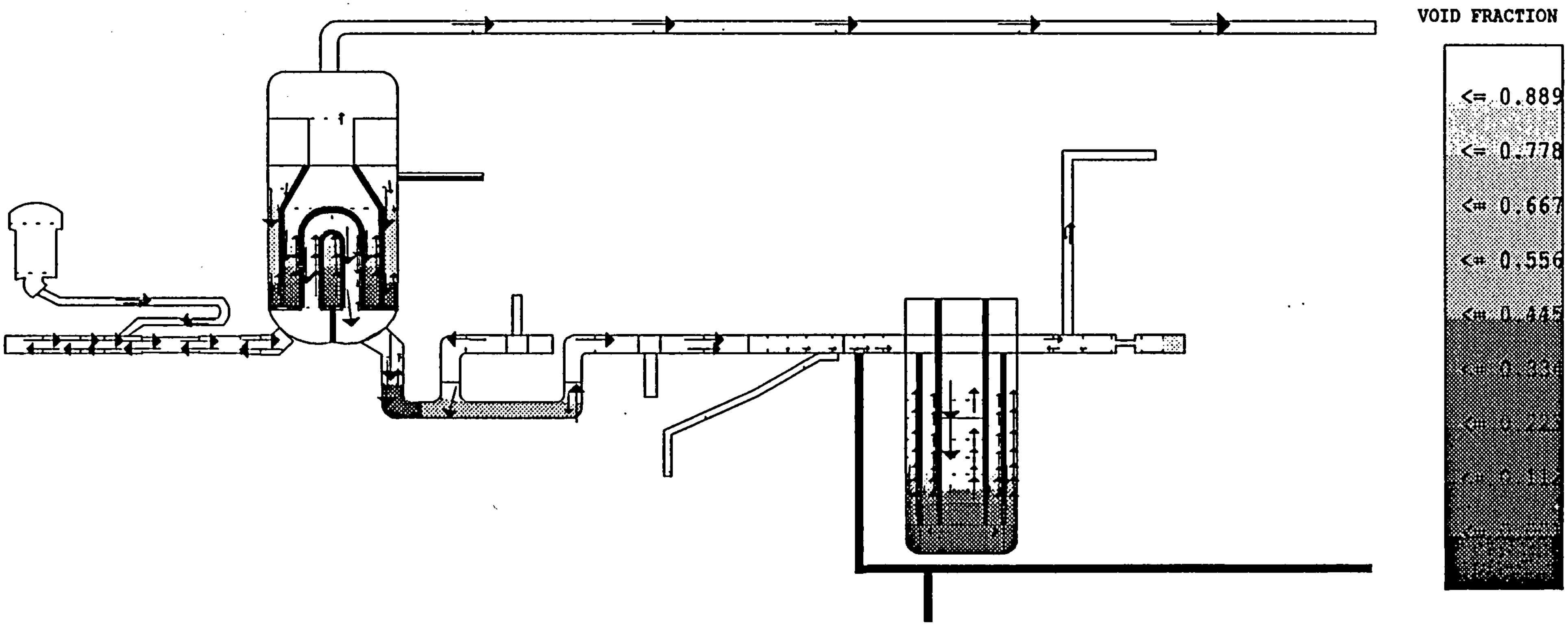
(10.0)



S.M.A.R.T. SYSTEM MIMIC FOR ANALYSIS OF REACTOR TRANSIENTS

TITLE OF FRAME:- LOFT TEST LP-SB-3 TRAC CALCULATION

Ti 5013.6
VLIQ ⟶
(10.0)
VVAP ⟶
(10.0)



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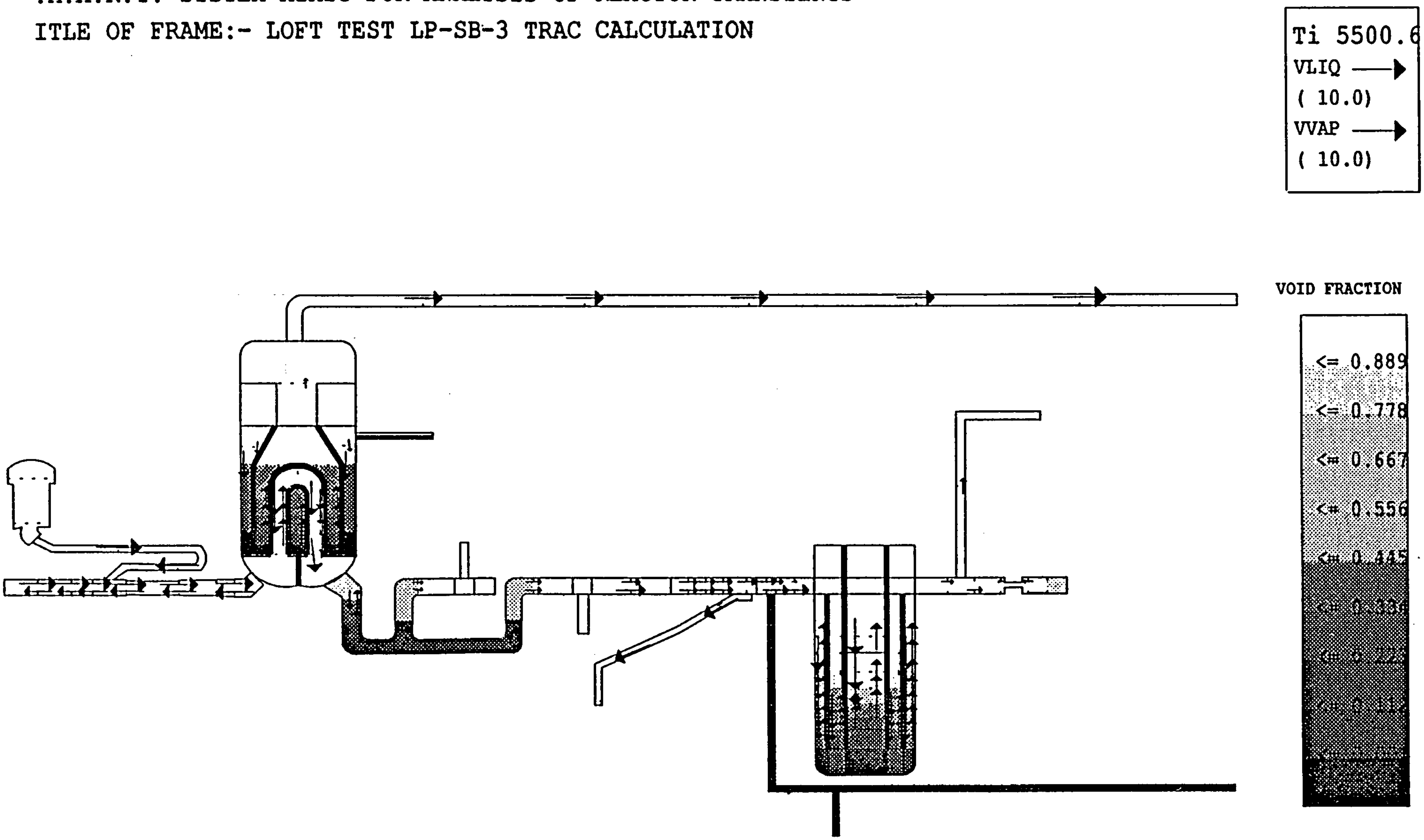

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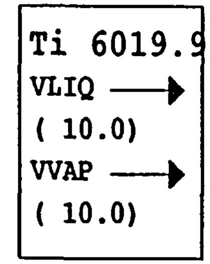

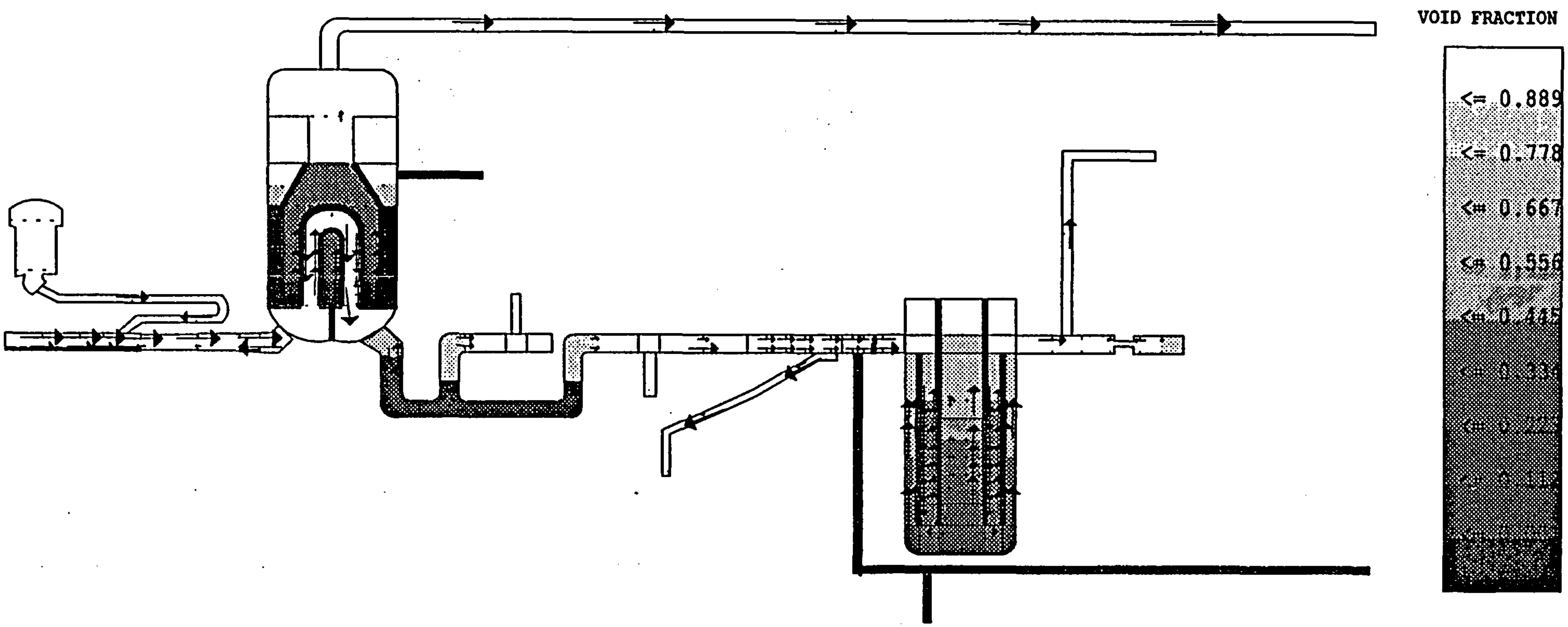

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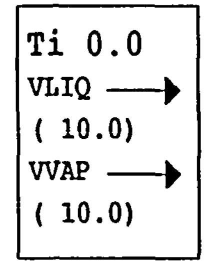

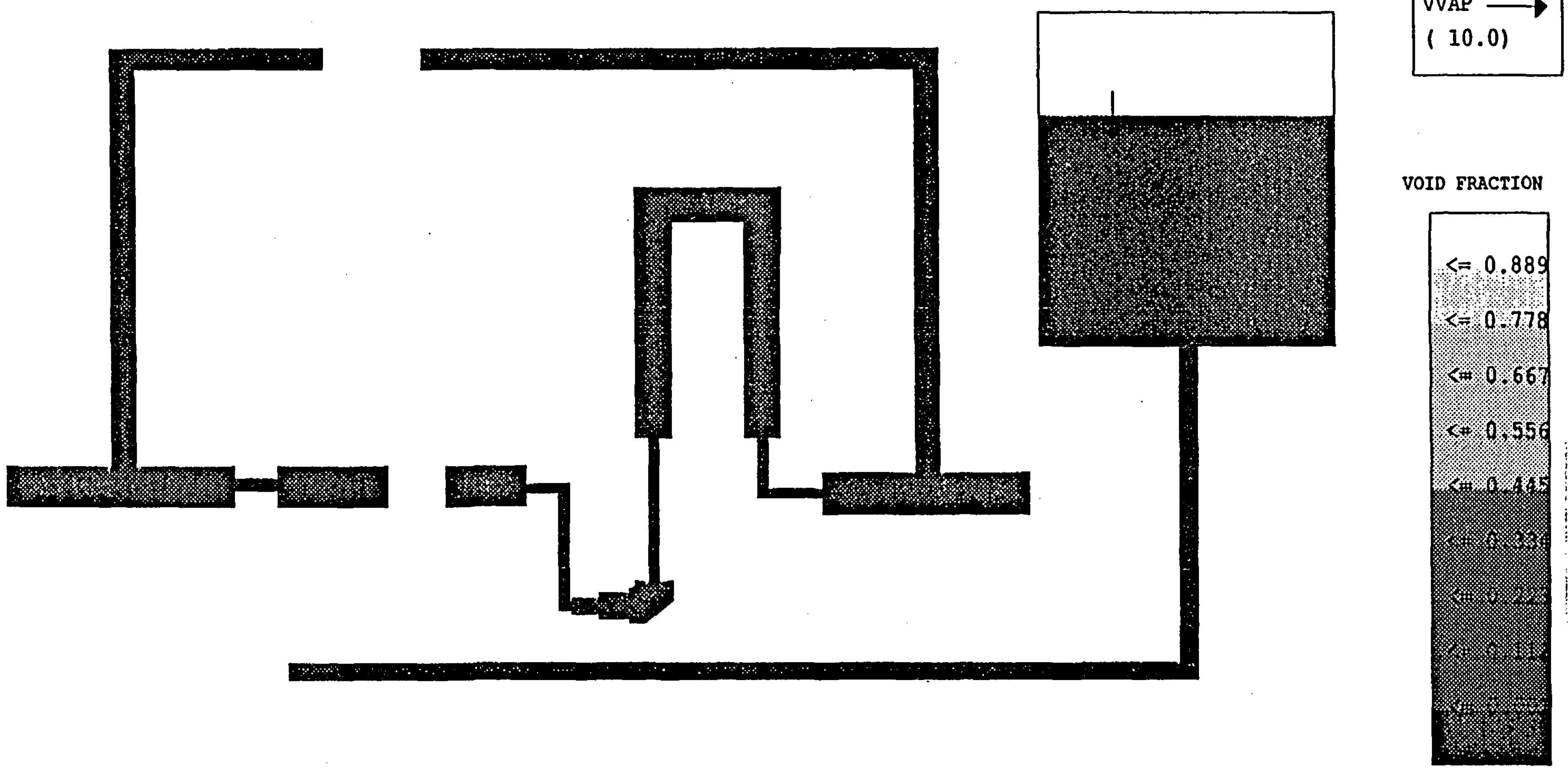

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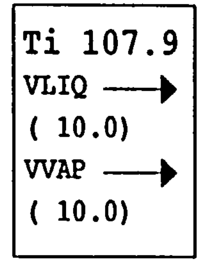

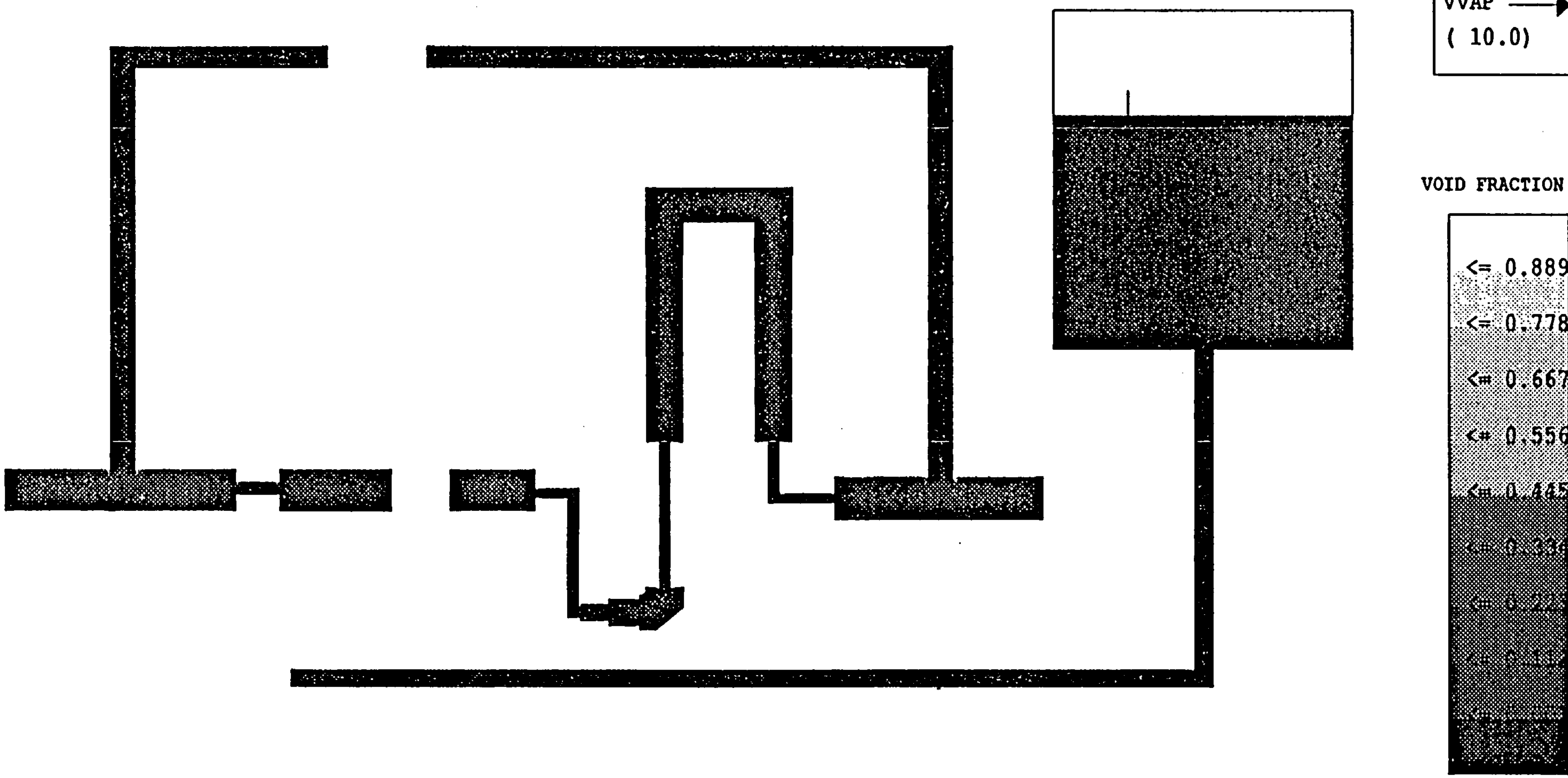

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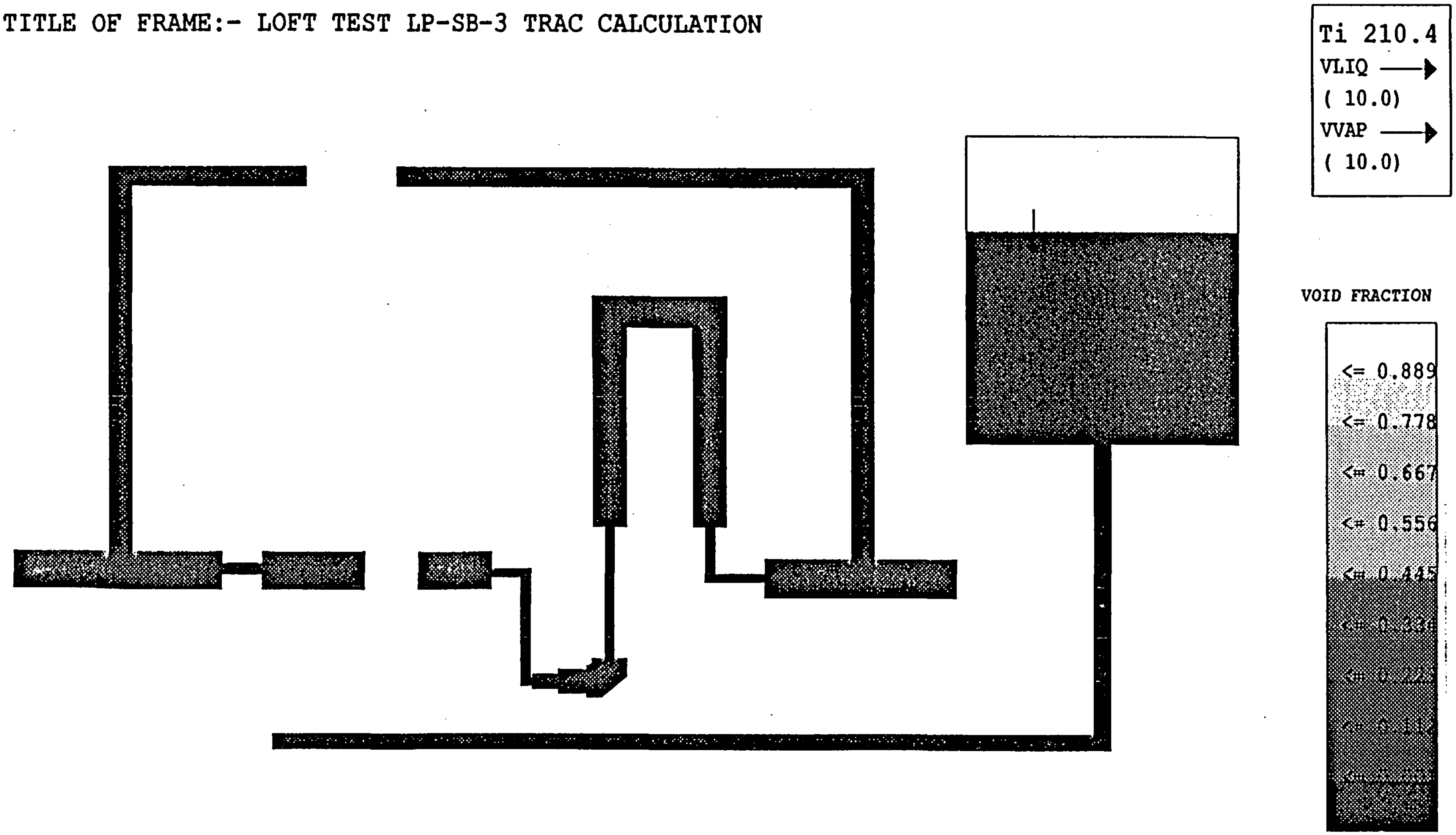

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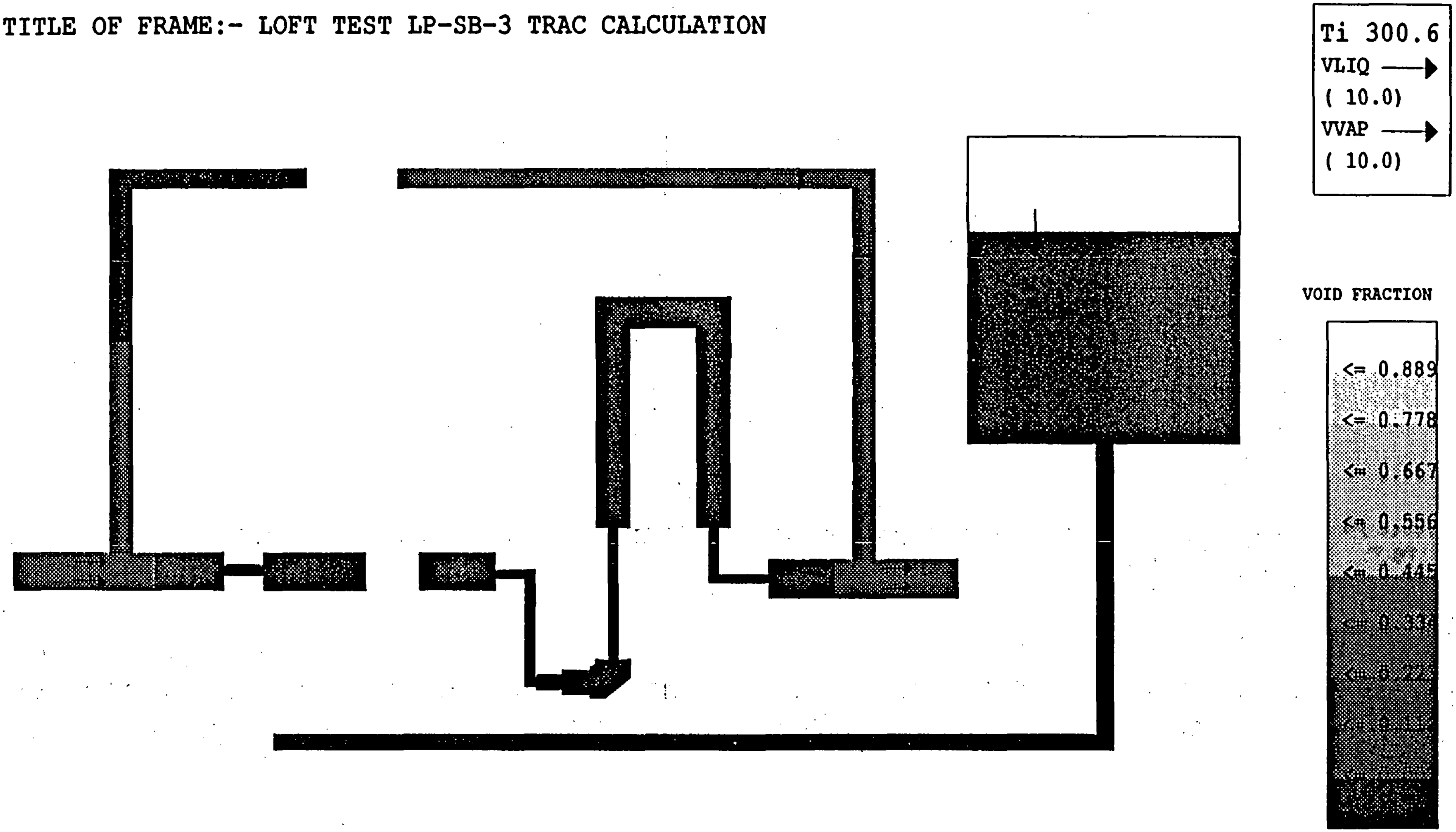

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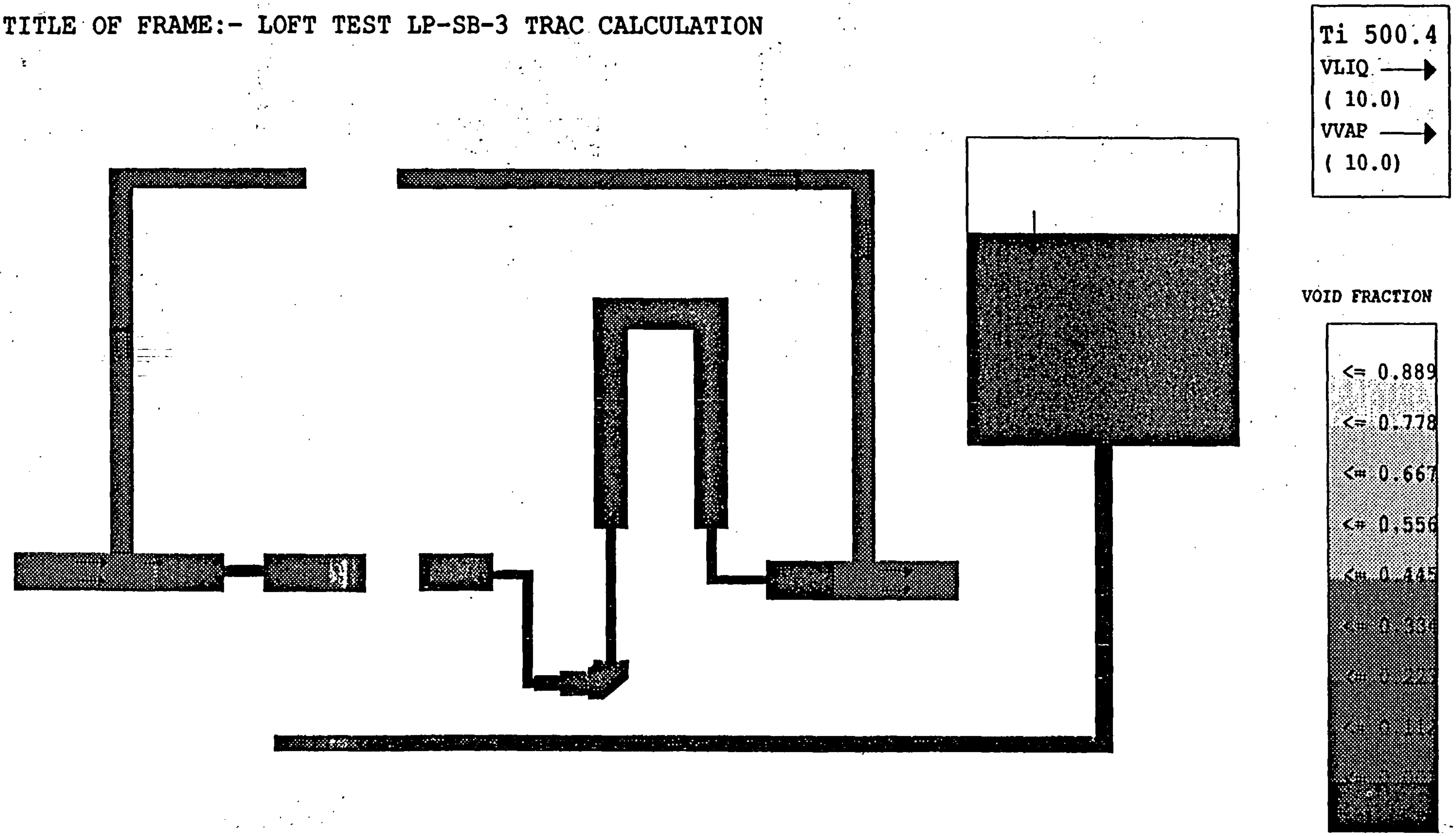

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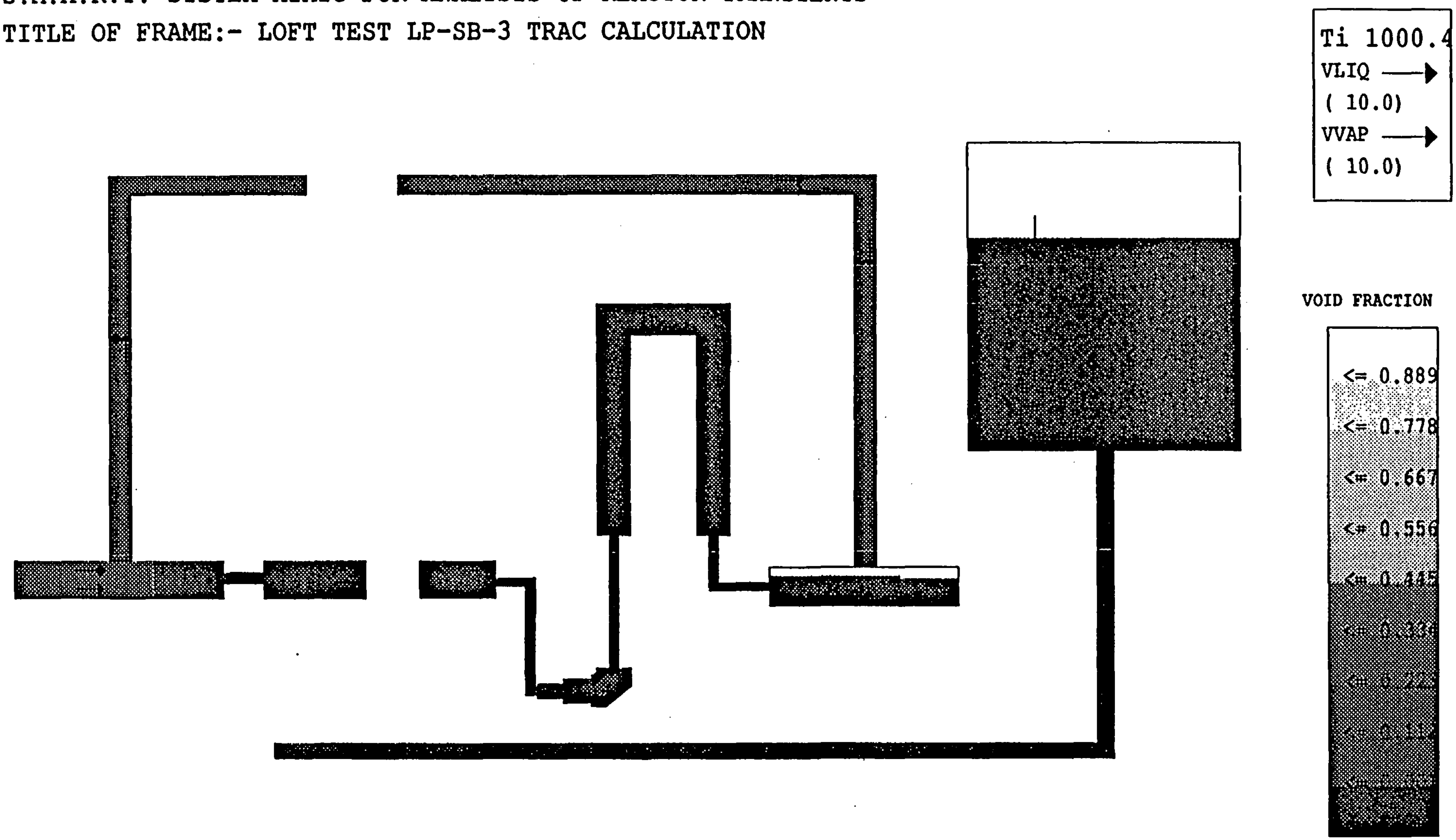

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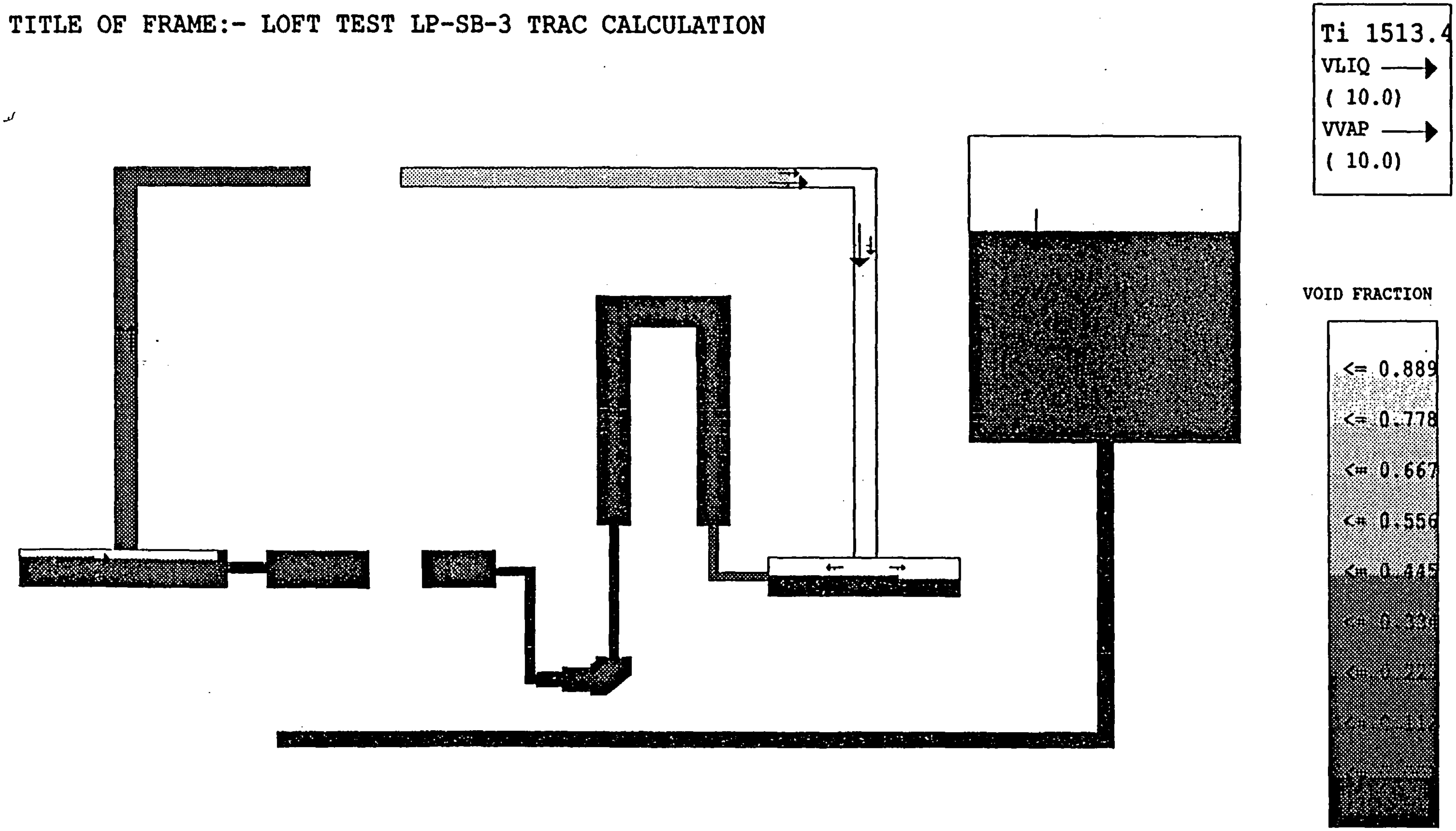

S.M.A.R.T. SYSTEM MIMIC FOR ANALYSIS OF REACTOR TRANSIENTS

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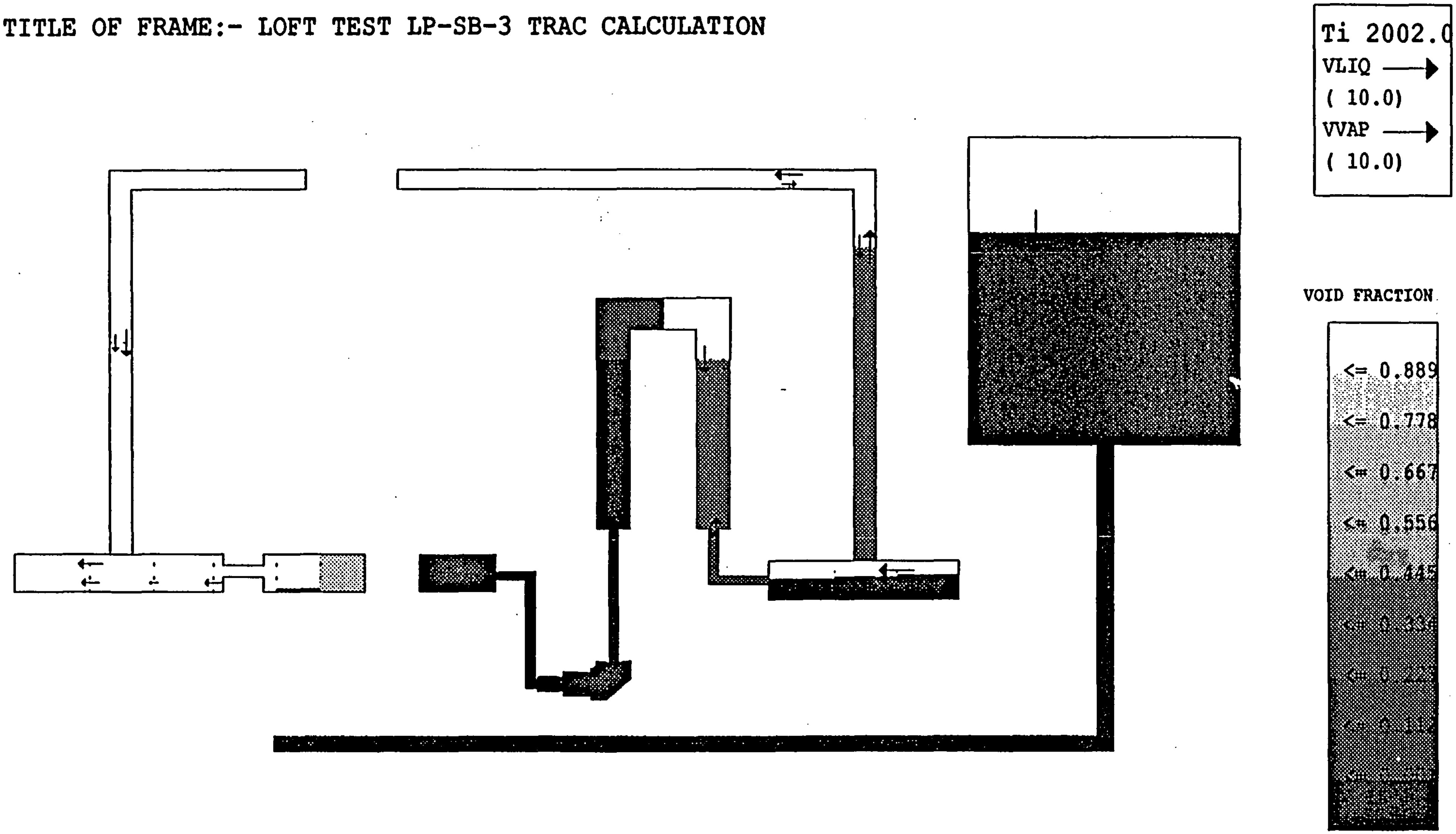

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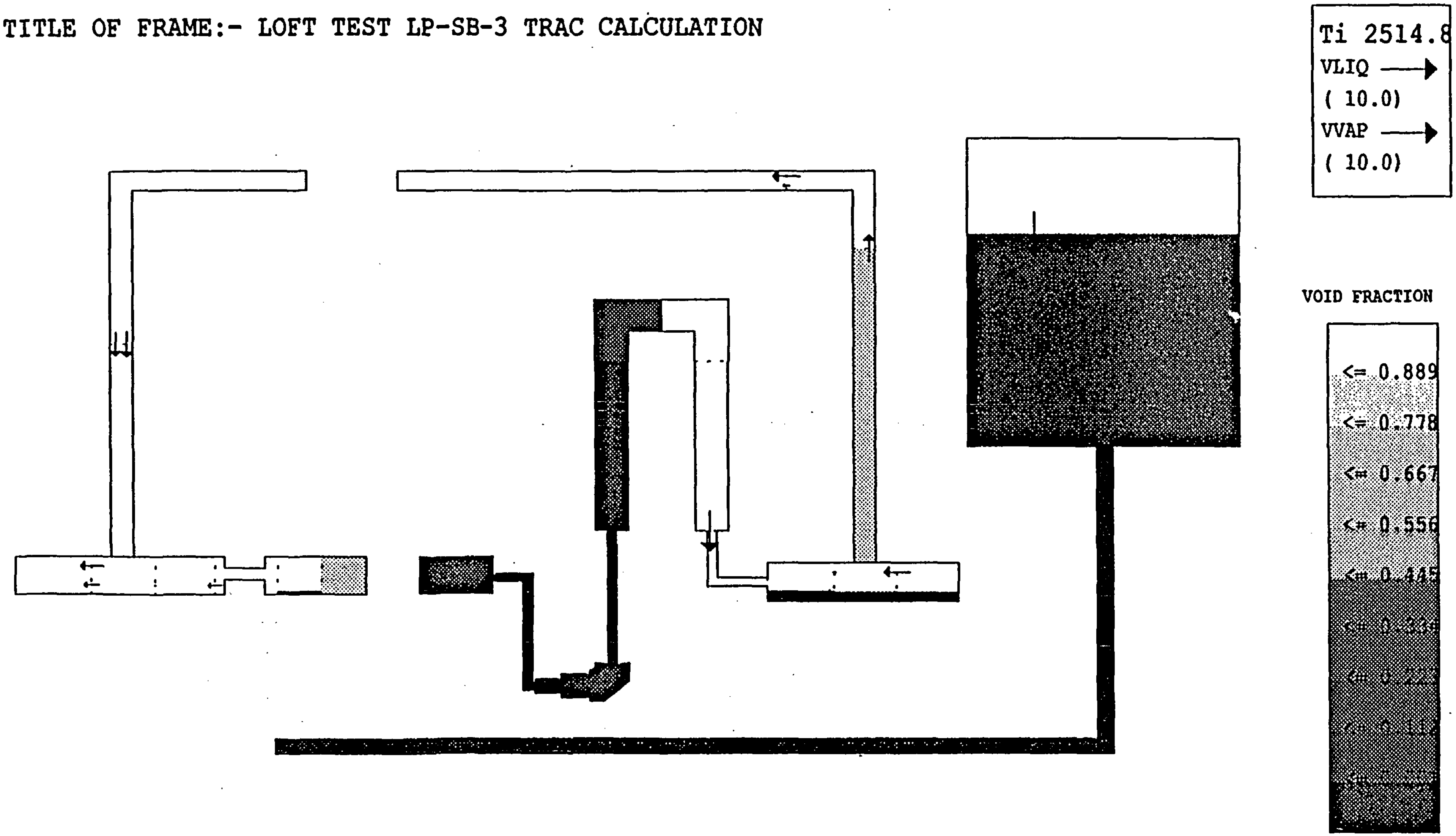

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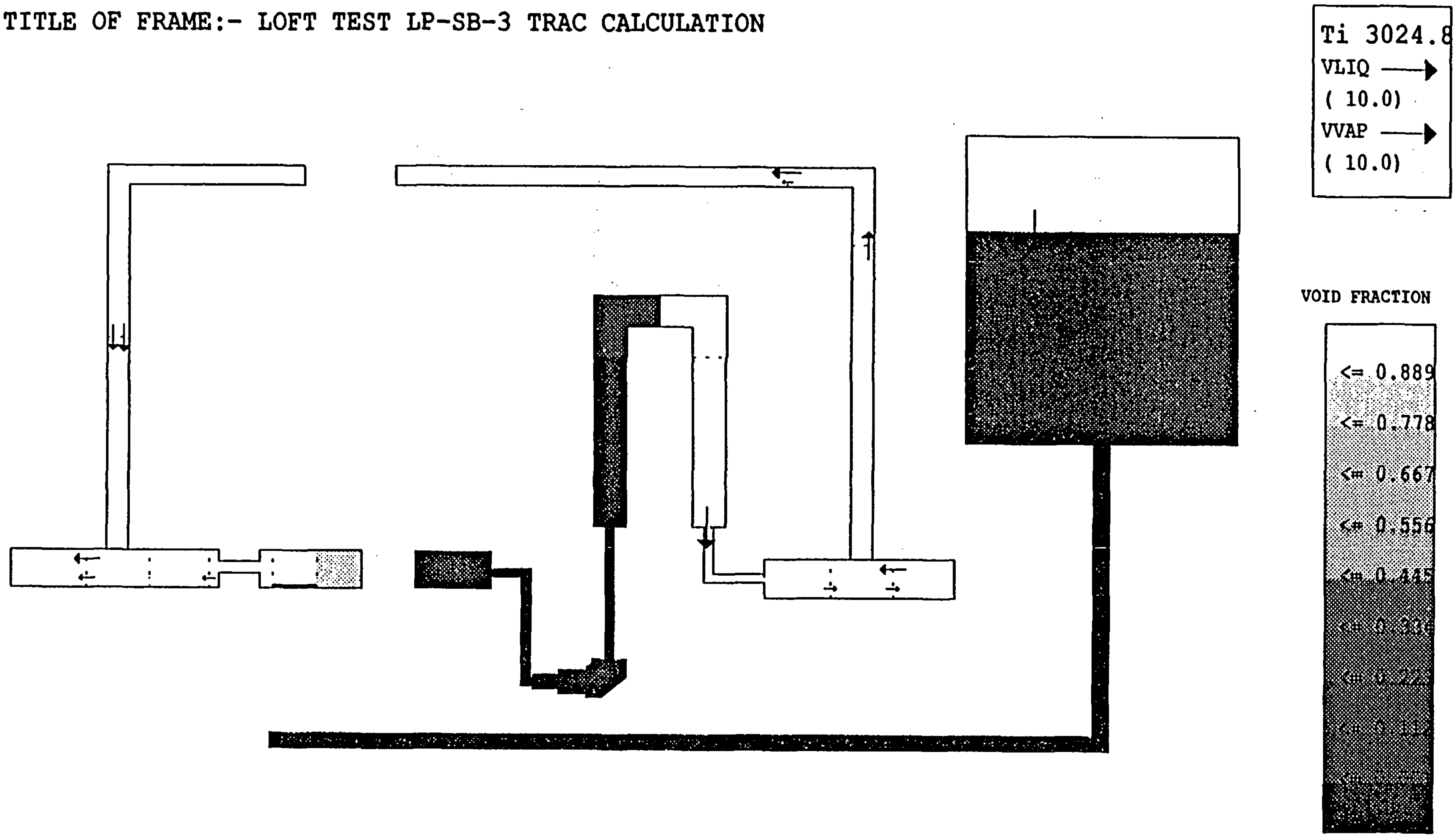

S.M.A.R.T. SYSTEM MIMIC FOR ANALYSIS OF REACTOR TRANSIENTS

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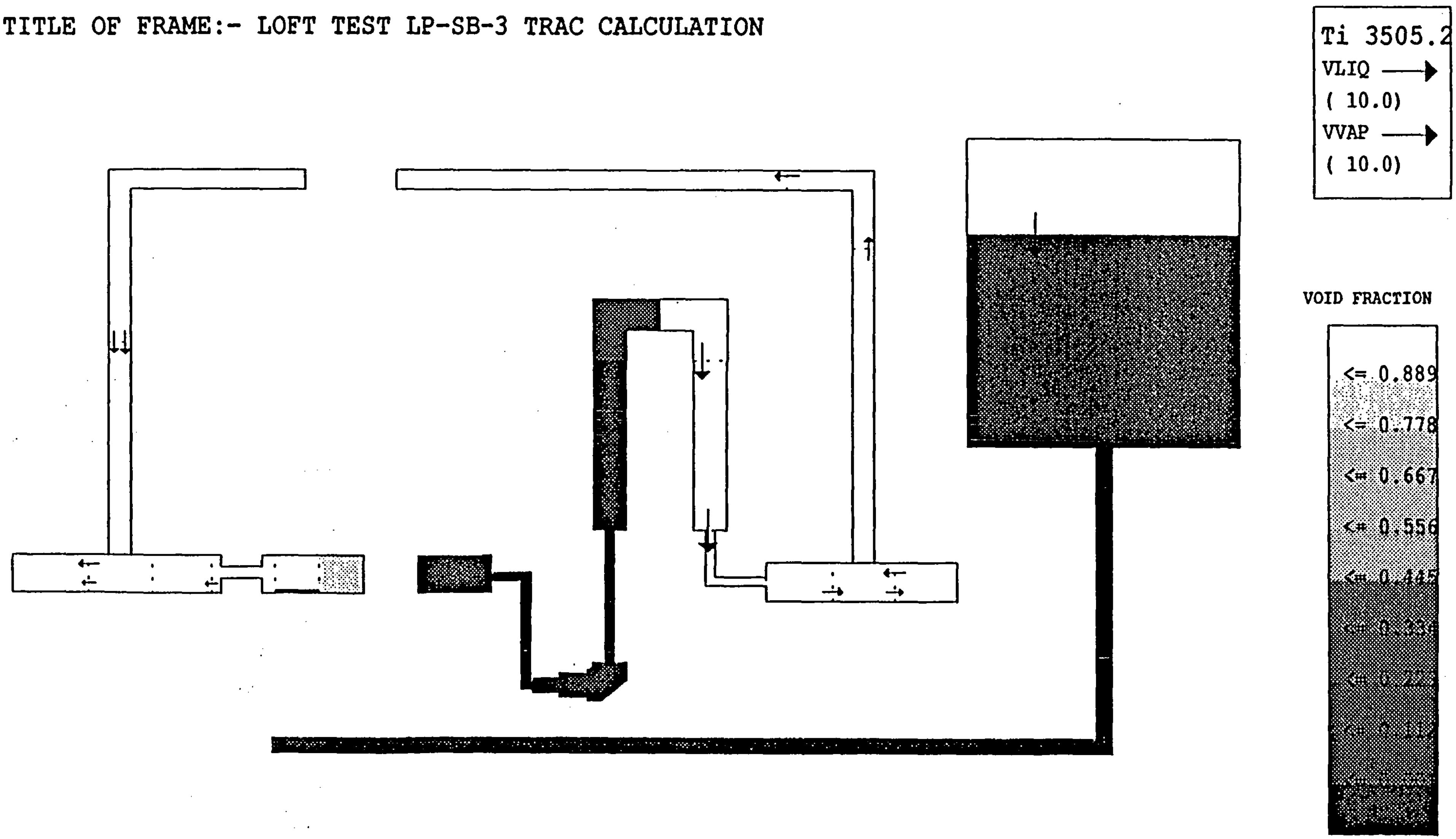
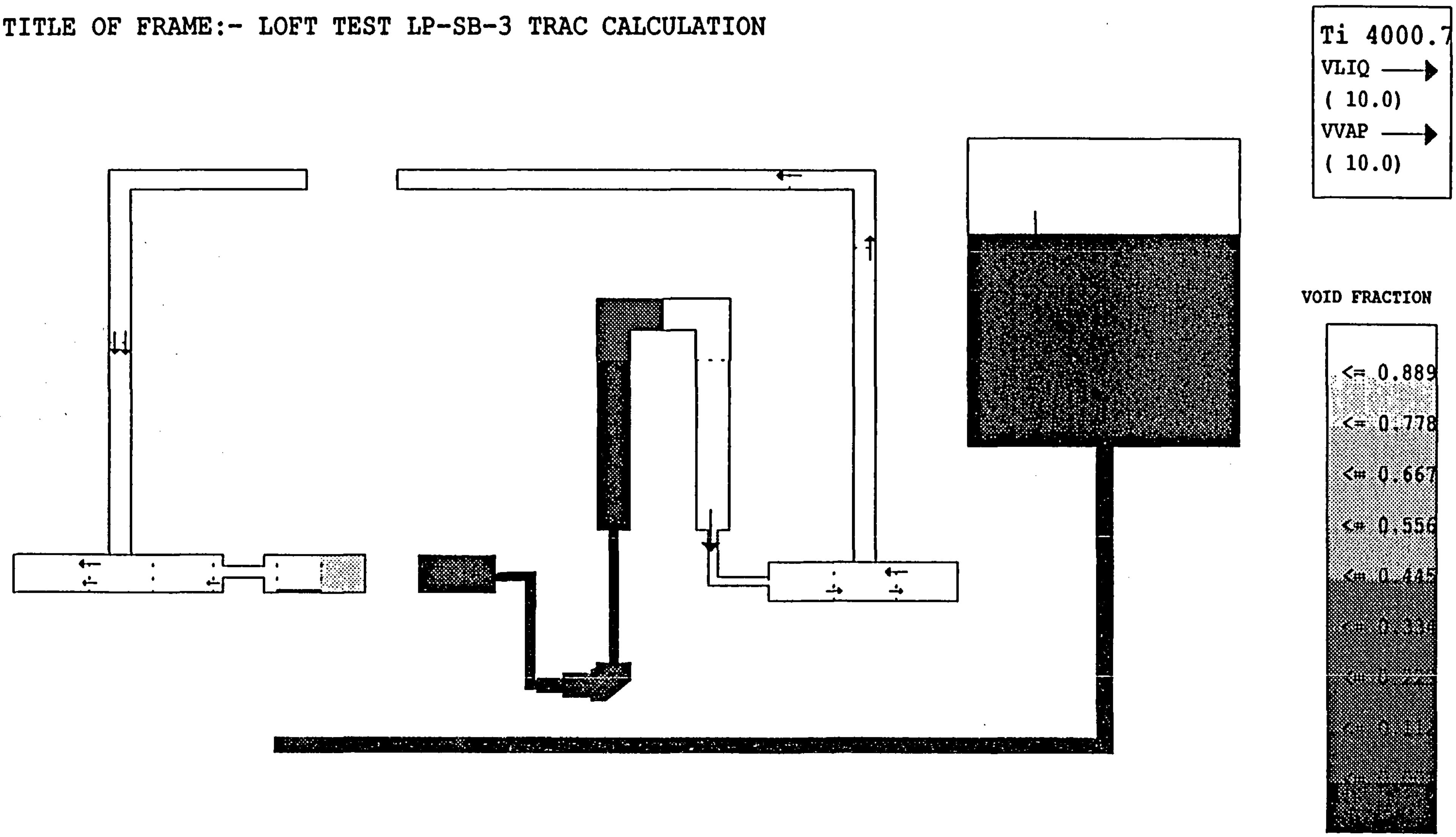


S.M.A.R.T. SYSTEM MIMIC FOR ANALYSIS OF REACTOR TRANSIENTS

TITLE OF FRAME:- LOFT TEST LP-SB-3 TRAC CALCULATION



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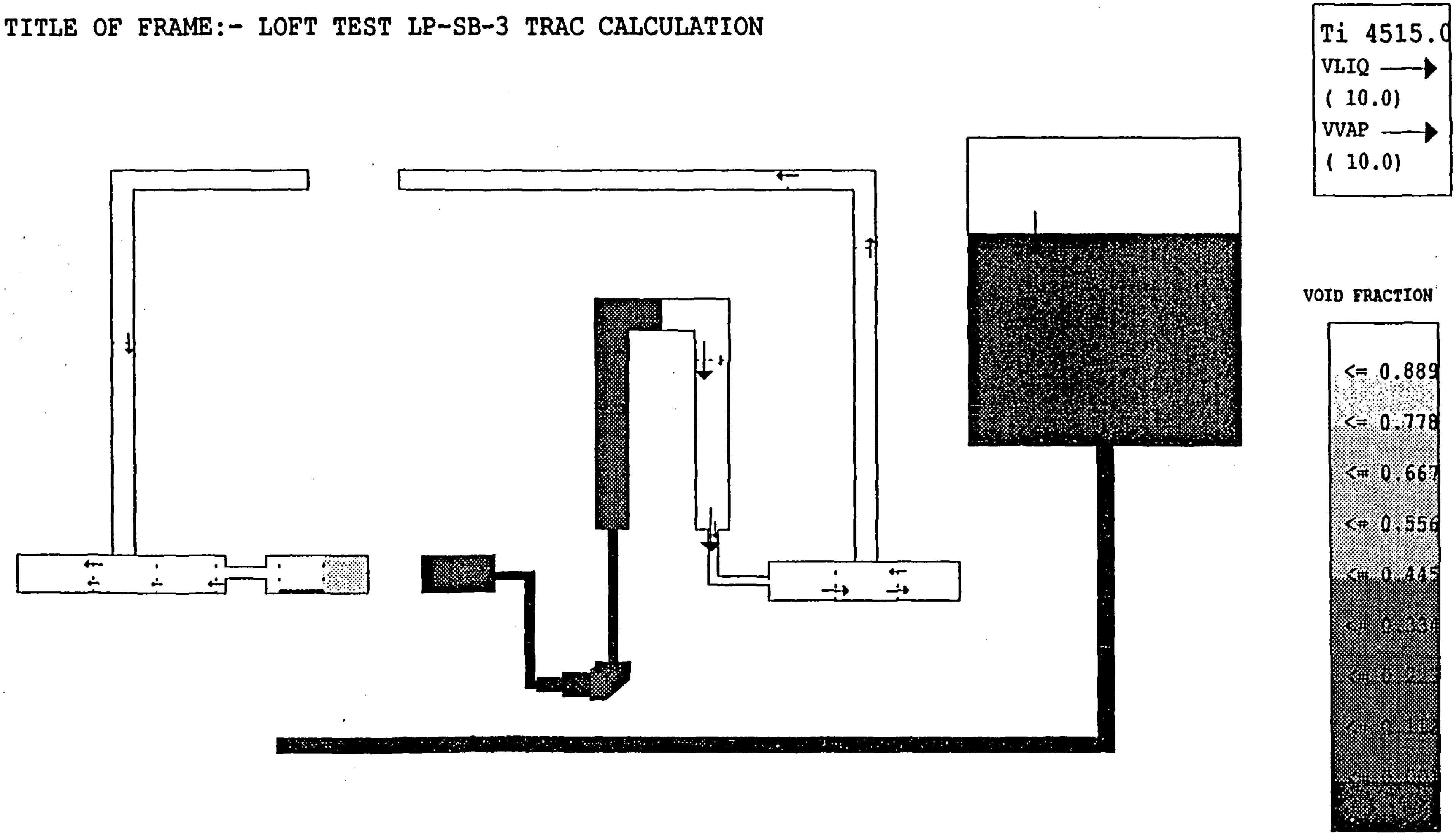
TITLE OF FRAME:- LOFT TEST LP-SB-3 TRAC CALCULATION



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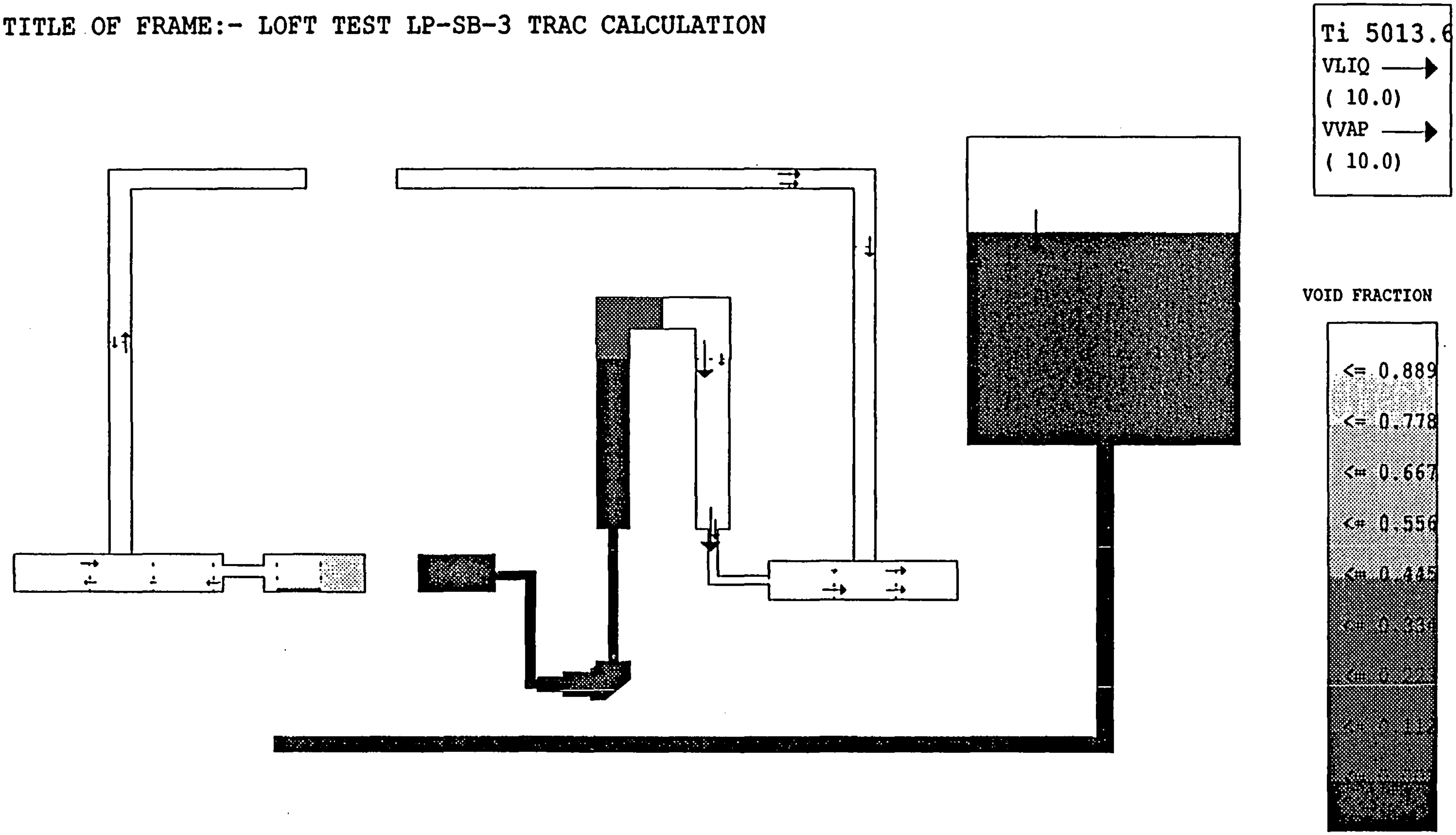
TITLE OF FRAME:- LOFT TEST LP-SB-3 TRAC CALCULATION



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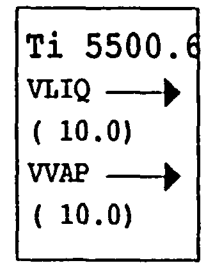

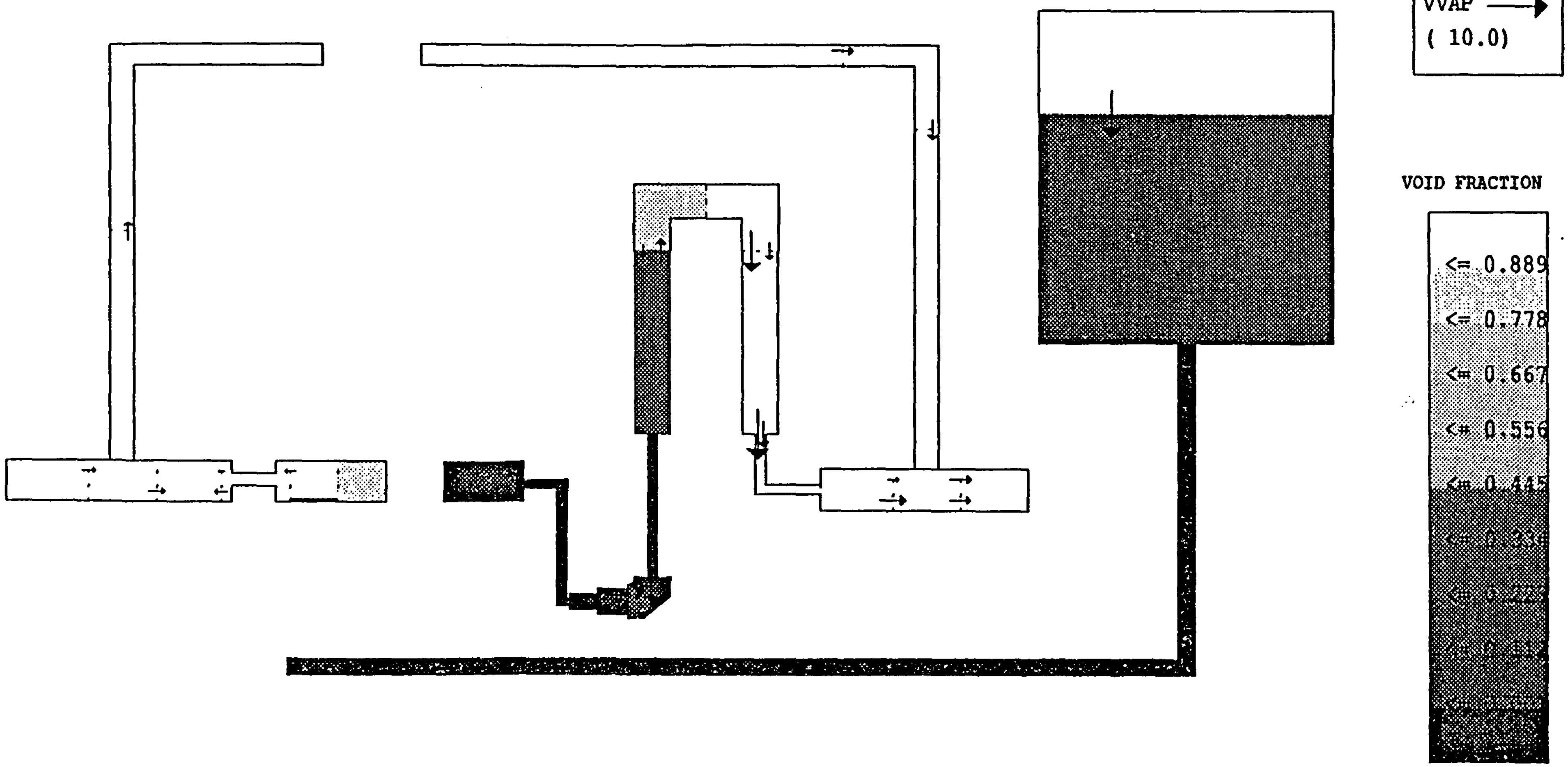

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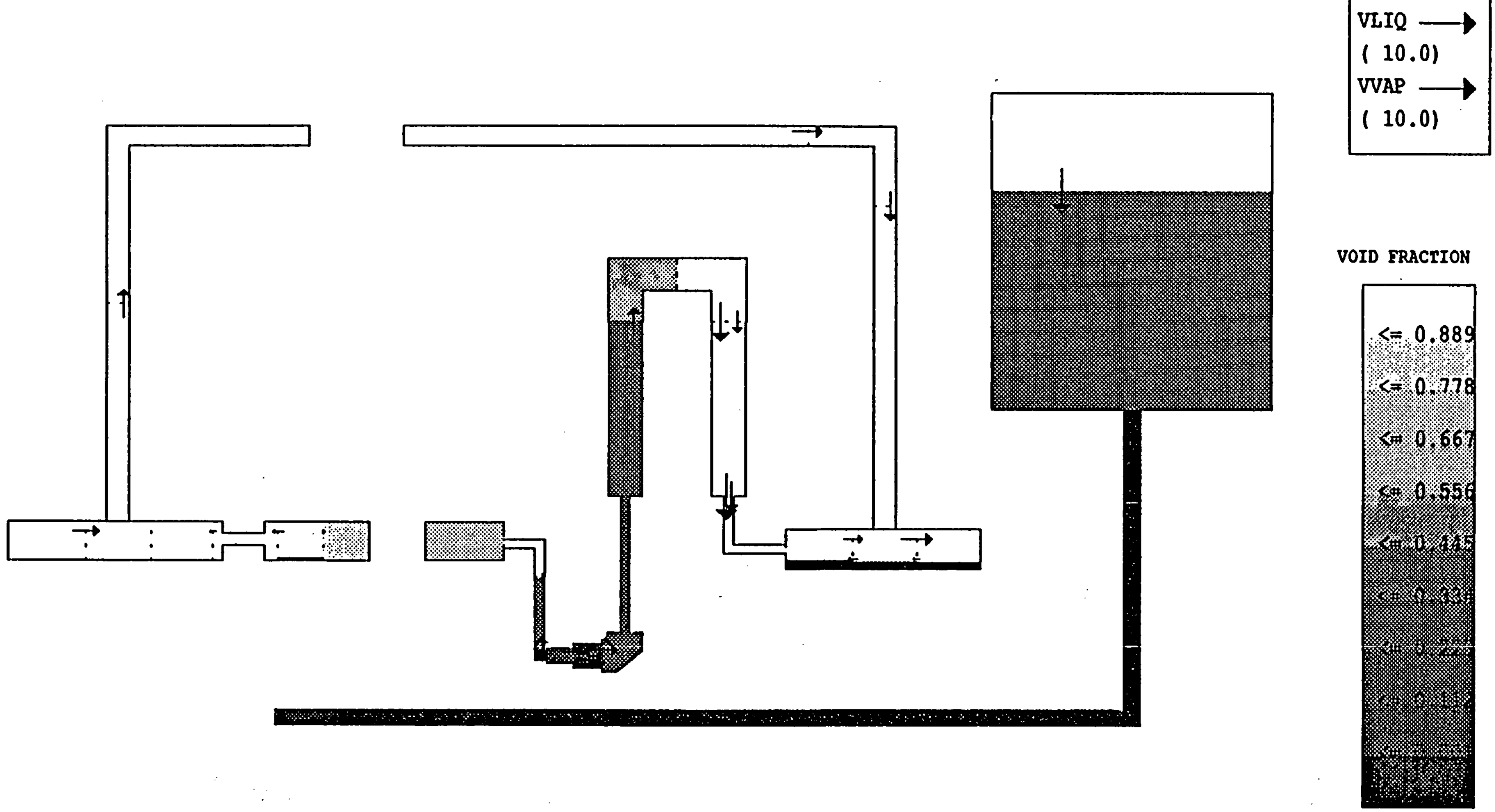
TITLE OF FRAME:- LOFT TEST LP-SB-3 TRAC CALCULATION



APPENDIX III

COMPARISON OF BREAK MASS FLOW RATE PREDICTED BY TRAC-PF1/MOD1 WITH THAT IMPLIED BY THE HOMOGENEOUS EQUILIBRIUM MODEL

As described in Section 8.4.2, although well-predicted for the first ~ 500 seconds of the transient, TRAC-PF1/MOD1 slightly over-predicted the break mass flow rate after this time, until the primary coolant pumps were tripped (and the break uncovered) at 1600 seconds. The fluid conditions in the last cell of the break line, at 1000, 1200 and 1400 seconds, are given in Table AIII.1. The predicted break mass flow rates at these times are compared with those implied by the Homogeneous Equilibrium Model (HEM) (12) in Table AIII.2. The TRAC-PF1/MOD1 predicted values are found to be ~ 10% higher than the HEM figures. Figure AIII.1 - taken from the TRAC Code Manual (17) - confirms that TRAC-PF1/MOD1 is expected to over-predict mass fluxes (under these conditions) when compared with the Homogeneous Equilibrium Model.

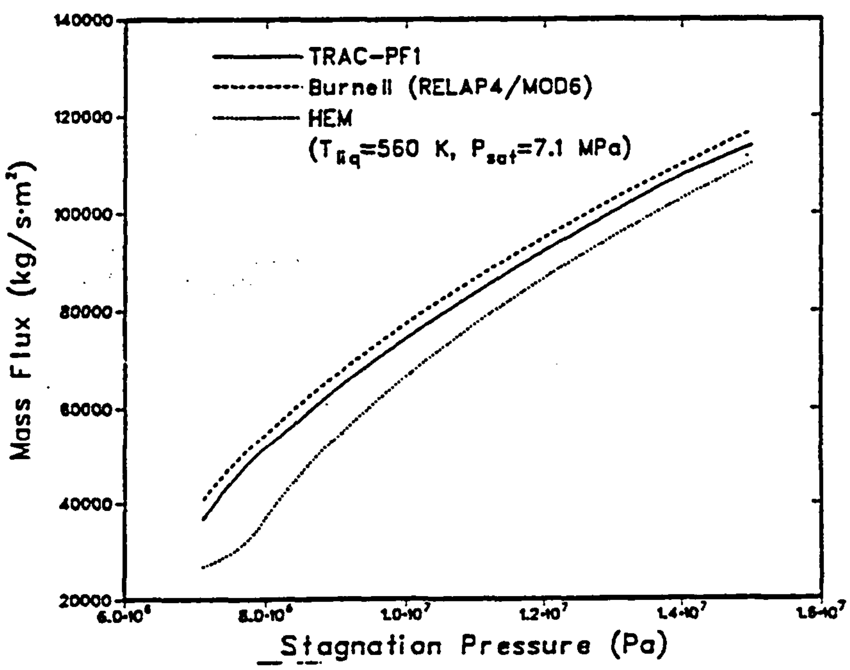


FIGURE AIII.1

TABLE AIII.1

FLUID CONDITIONS AT END OF BREAK LINE

Time (seconds)	Pressure (MPa)	Temperature (K)	Liquid Density (kgm^{-3})	Vapour Density (kgm^{-3})	Void Fraction	Liquid Velocity (ms^{-1})	Vapour Velocity (ms^{-1})	Average-Density (kgm^{-3})	Quality
1000	7.14616	560.4	738.5	37.37	0.3350	54.05	56.85	503.62	0.0261
1200	7.11658	560.1	739.0	37.20	0.4389	61.20	64.07	530.98	0.0396
1400	7.07897	559.8	739.7	36.98	0.4969	66.16	69.08	390.52	0.0490

TABLE AIII.2

Time (seconds)	Break Mass Flow Rate Predicted by TRAC-PF1/MOD1 (kgs^{-1})	Break Mass Flow Rate Implied by HEM (kgs^{-1})	% by which TRAC-PF1/MOD1 over-predicts break mass flow compared with HEM
1000	1.6578	1.4996	9.5
1200	1.6070	1.4386	10.5
1400	1.5748	1.4028	10.9

FIGURES

In Figures 7-26, the following line types are used:

———————— TRAC-PF1/MOD1 Calculation

\- - - - - - Experimental Data

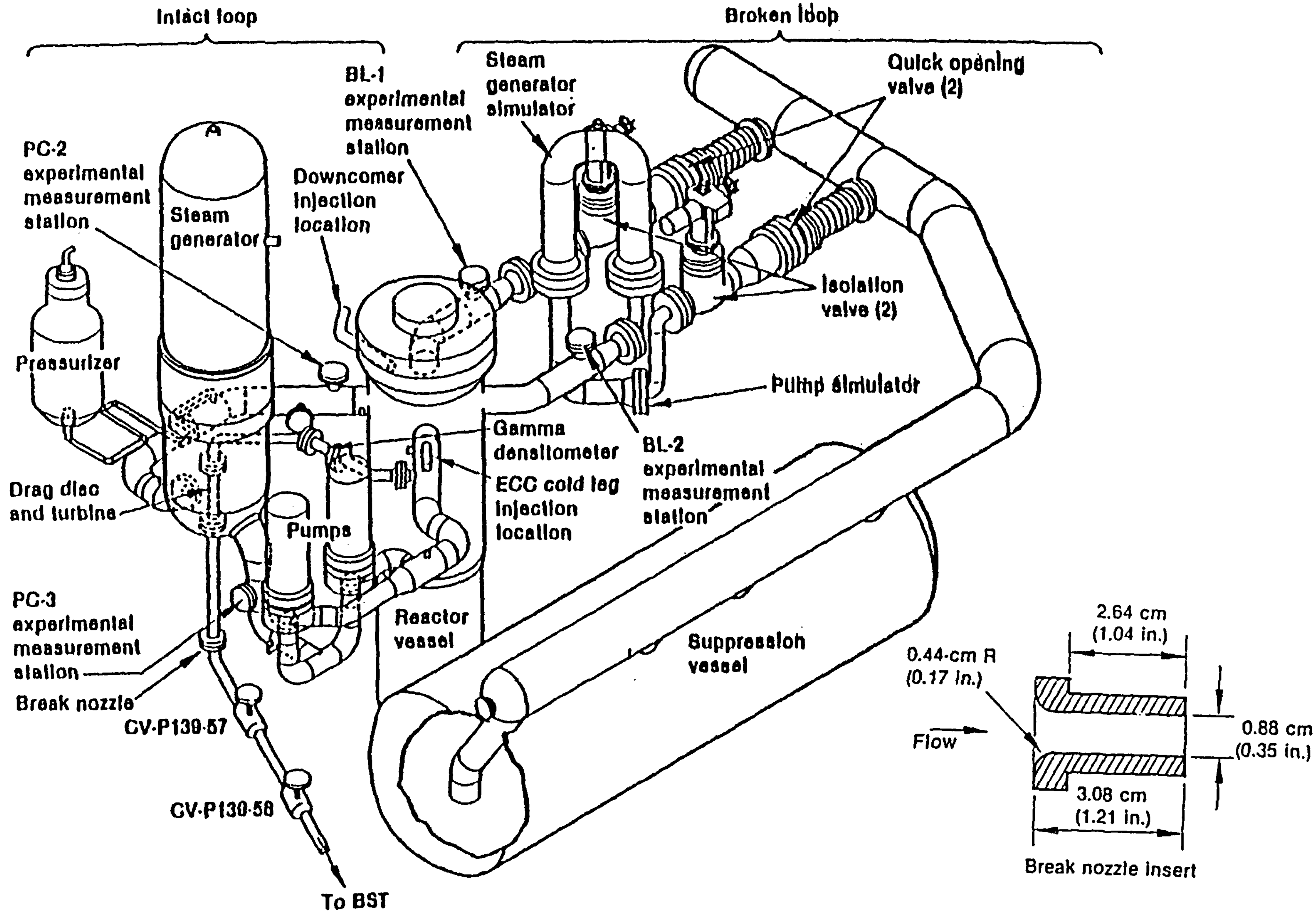


Figure 1 LOFT configuration for cold leg intact loop small break Experiment LP-SB-03

Component No	Component Name	Component No	Component Name
1	Intact Loop Hot Leg	63	Pump No 1 Injection Tee
8	Pressurizer	6	Pump Discharge
2	Steam Generator	71	ILCL, Break Line + Nozzle
3	Pump Suction	7	ILCL + ECC Line
4	Pump No 2	13	ECC Line
61	Pump No 2 Injection Tee	14	Accumulator Line Valve
5	Pump No 1	15	Accumulator

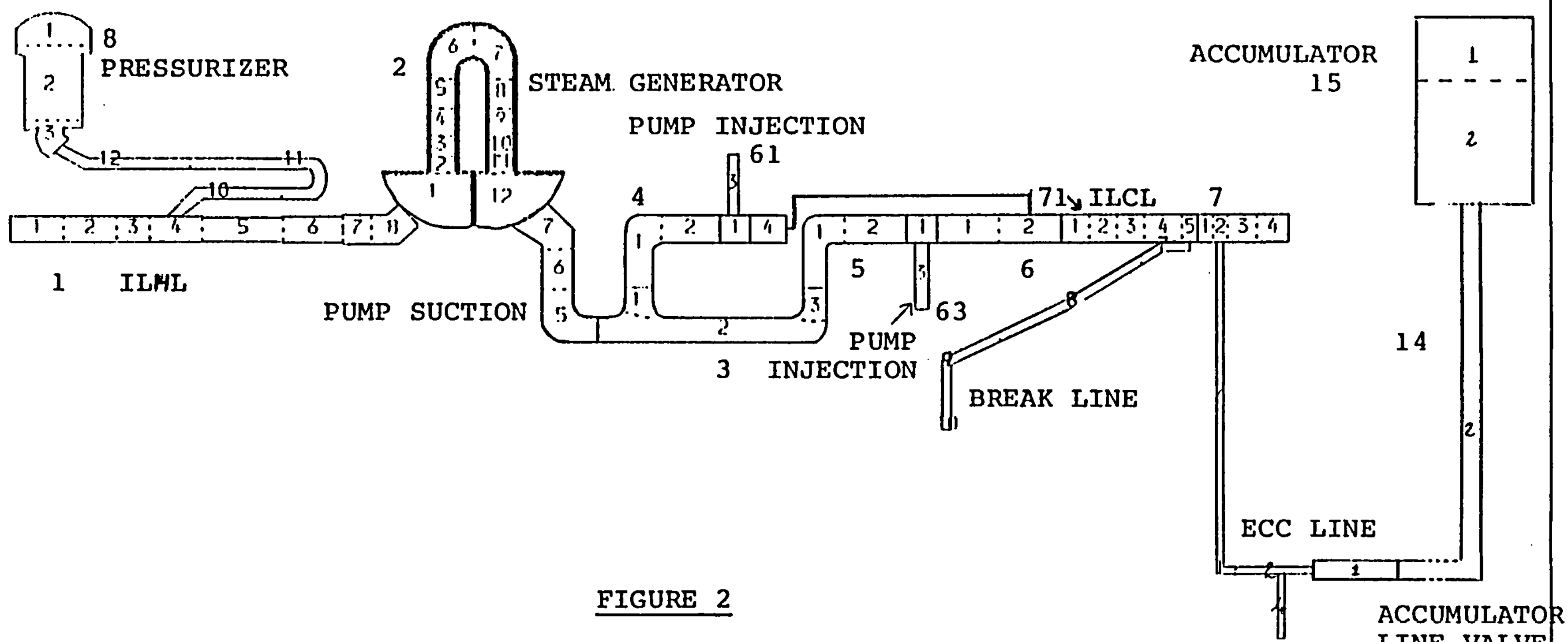


FIGURE 2

PRIMARY SYSTEM NODALISATION DIAGRAM FOR LP-SB-3

(a) INTACT LOOP

Component No	Component Name
31	Broken Loop Hot Leg
41	Broken Loop Cold Leg

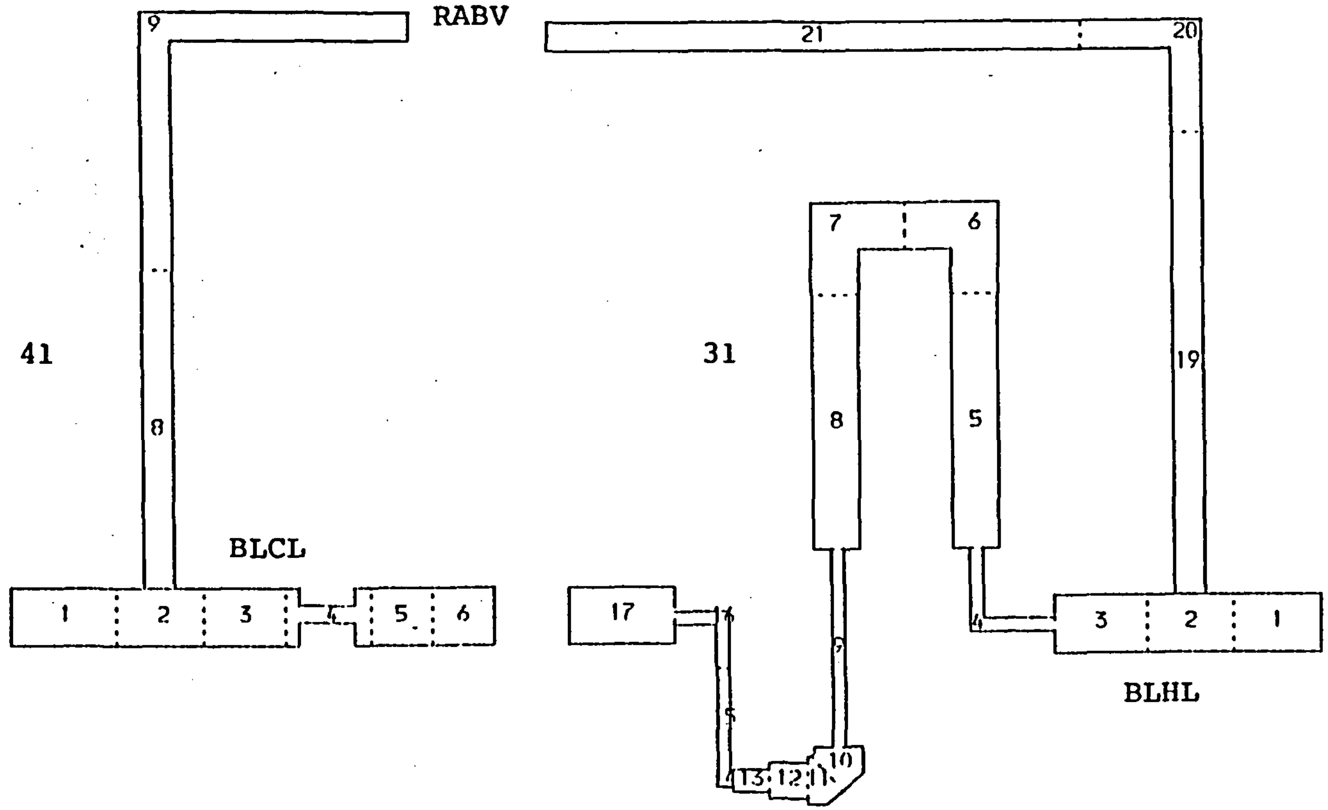
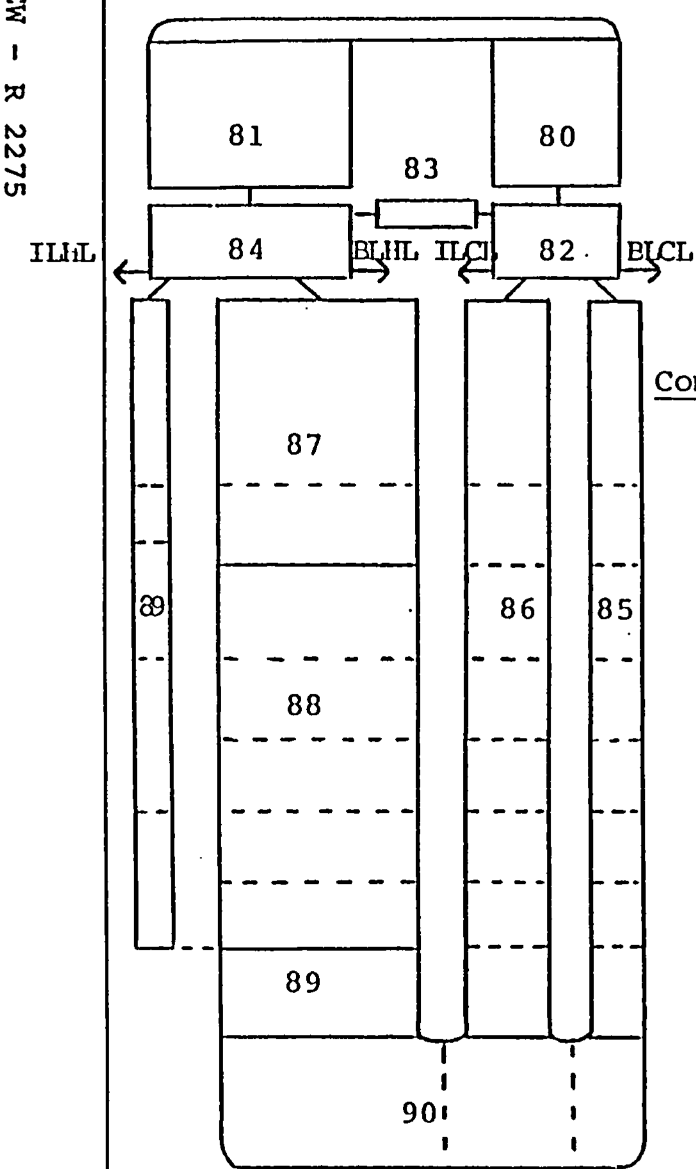


FIGURE 2

PRIMARY SYSTEM NODALISATION DIAGRAM FOR LP-SB-3

(b) BROKEN LOOP



Component No	Component Name
79	Bypass Path 5
80	Upper Plenum (D'comer)
81	Upper Plenum (core)
82	Plenum Component (D'comer)
83	Bypass Path 4
84	Plenum Component (core)
85	Downcomer Bypass
86	Downcomer
87	Pipe Above Core
88	Core
89	Lower Plenum and Core Bypass
90	Lower Plenum

Vessel Representation for "System Mimic"
(See Appendix II)

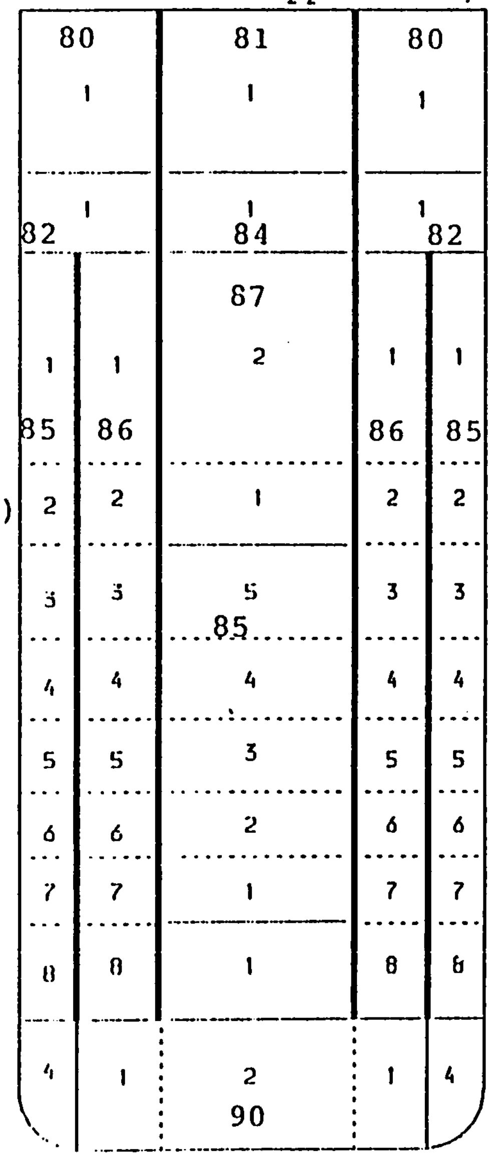
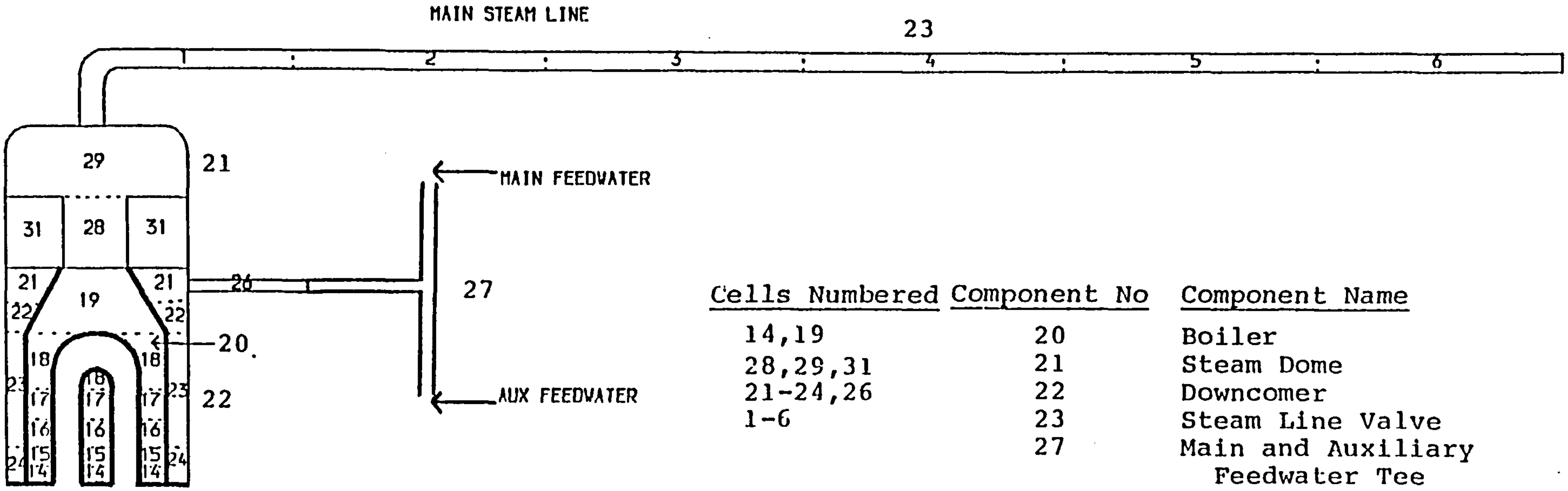


FIGURE 3

TRAC-PF1/MOD1 REACTOR VESSEL NODALISATION DIAGRAM FOR LP-SB-3



Cells Numbered	Component No	Component Name
14,19	20	Boiler
28,29,31	21	Steam Dome
21-24,26	22	Downcomer
1-6	23	Steam Line Valve
	27	Main and Auxiliary Feedwater Tee

FIGURE 4

TRAC-PF1/MOD1 STEAM GENERATOR SECONDARY SIDE NODALISATION DIAGRAM FOR LP-SB-3

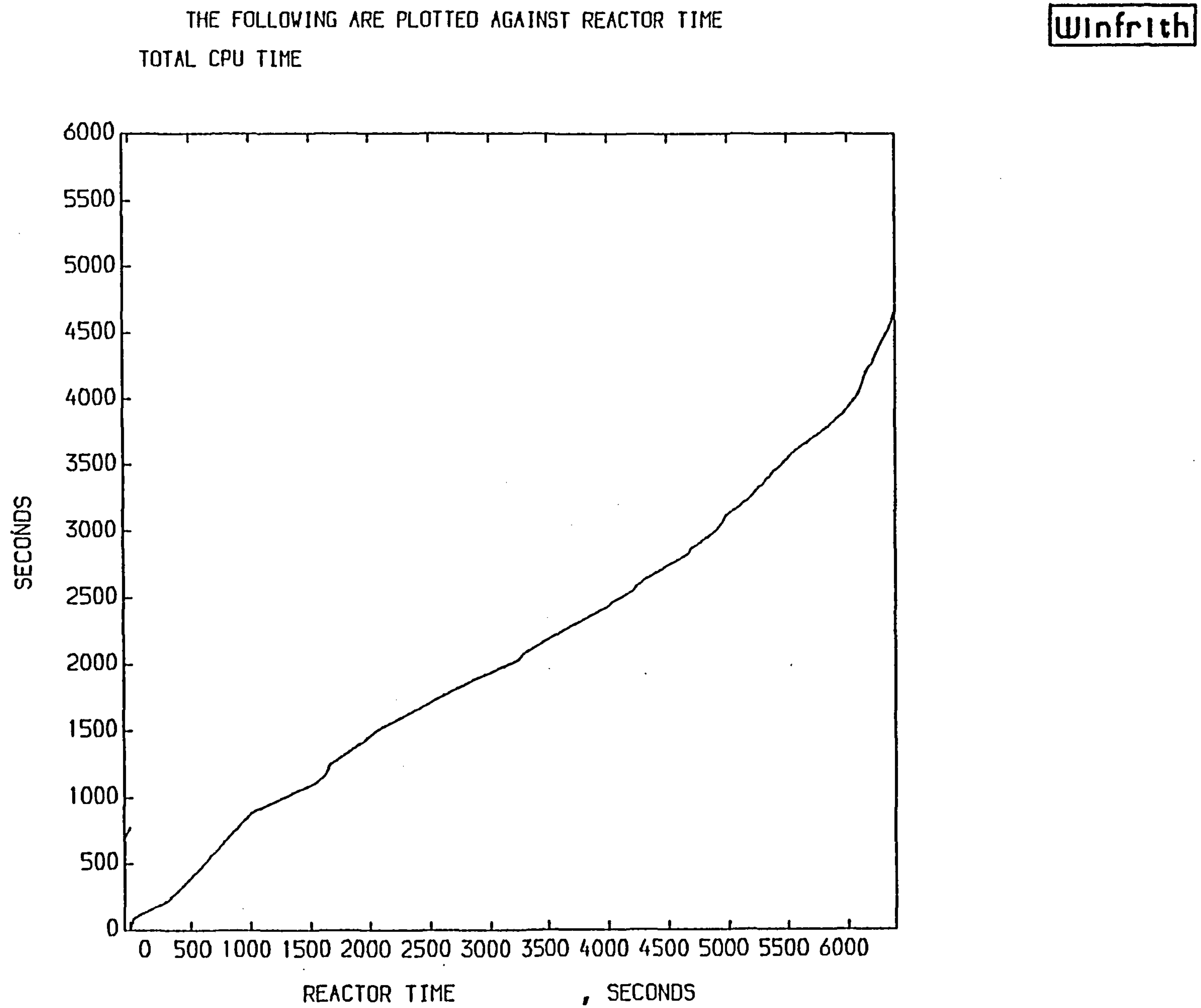


FIGURE 5

LP-SB-3 CALCULATION
CPU USAGE

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THE FOLLOWING ARE PLOTTED AGAINST REACTOR TIME

TIME STEP SIZE

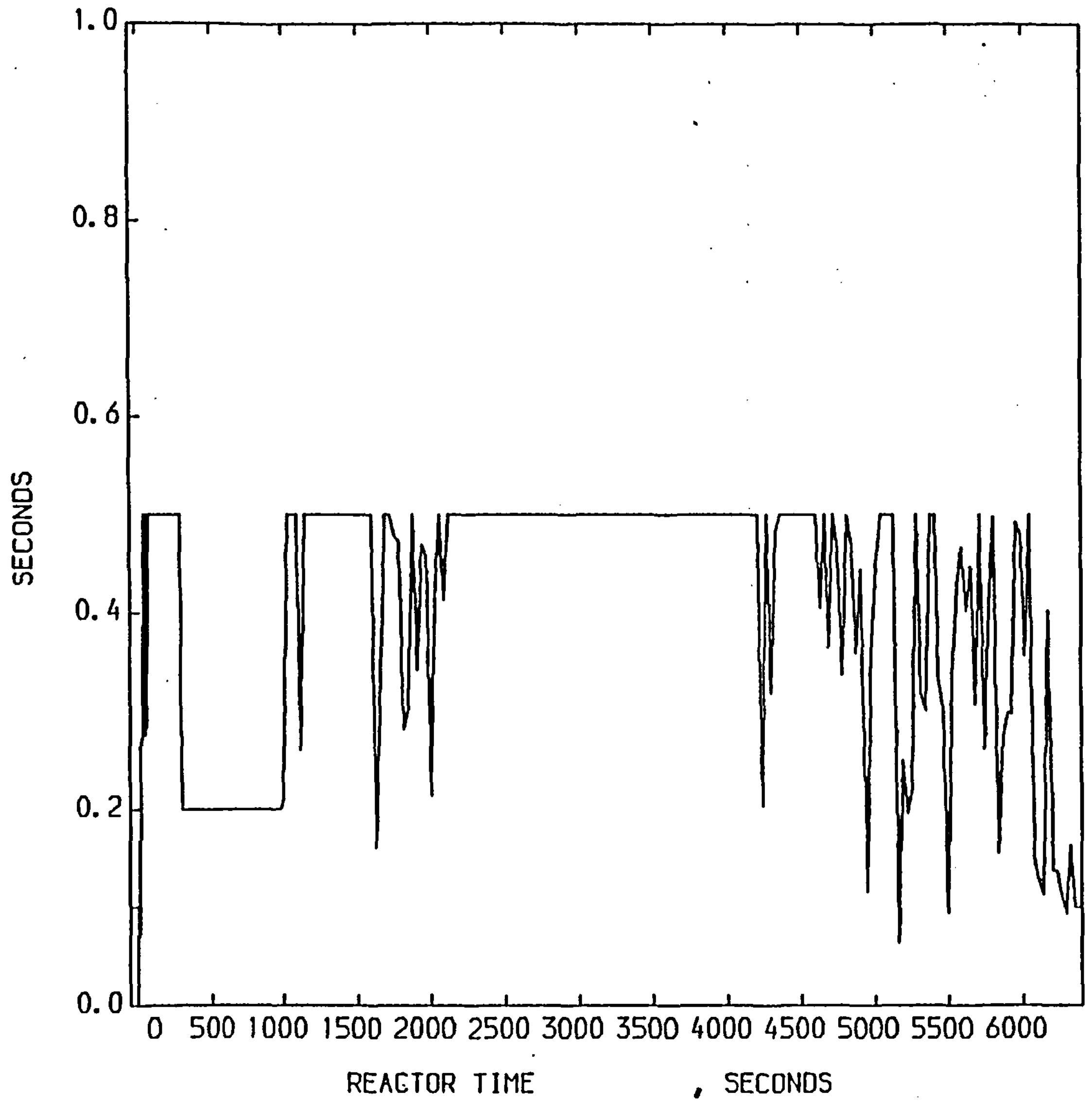

FIGURE 6

LP-SB-3 CALCULATION
TIME STEP SIZE

THE FOLLOWING ARE PLOTTED AGAINST TIME

PE-PC-002 ,PRESSURE

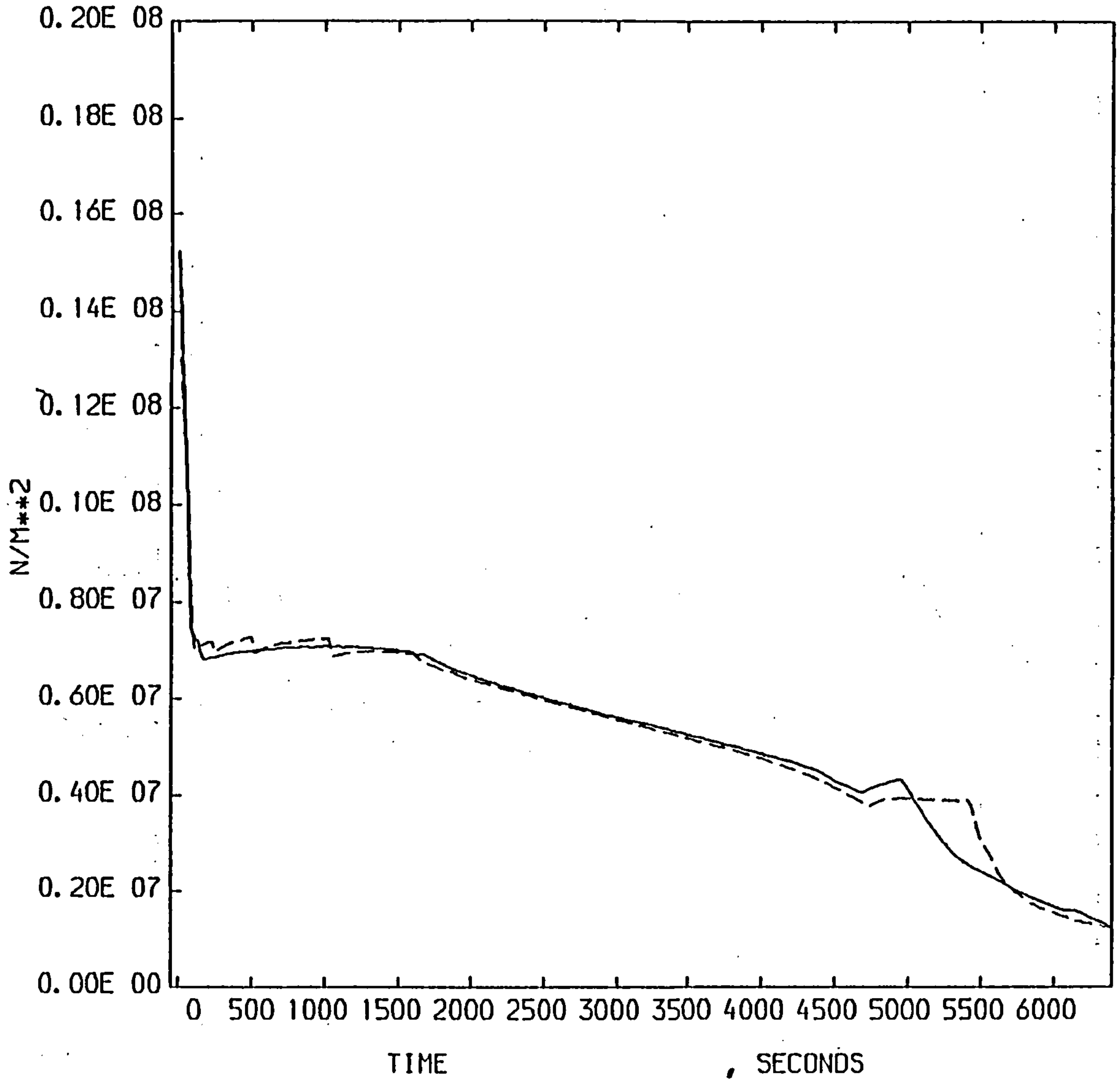

FIGURE 7

LP-SB-3 CALCULATION
PRIMARY SYSTEM PRESSURE

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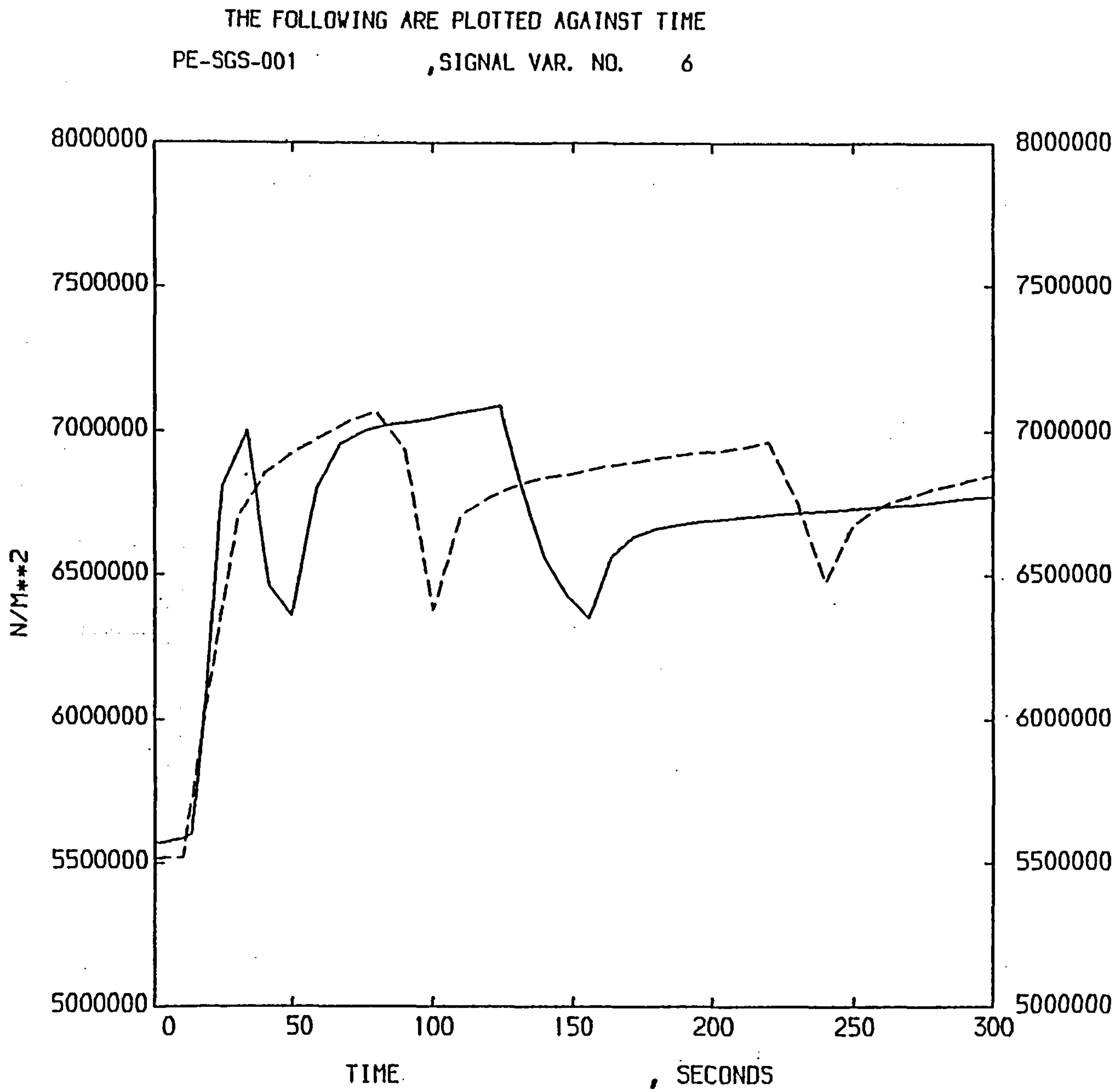

FIGURE 8

LP-SB-3 CALCULATION
SECONDARY SYSTEM PRESSURES (0-300 SECONDS)

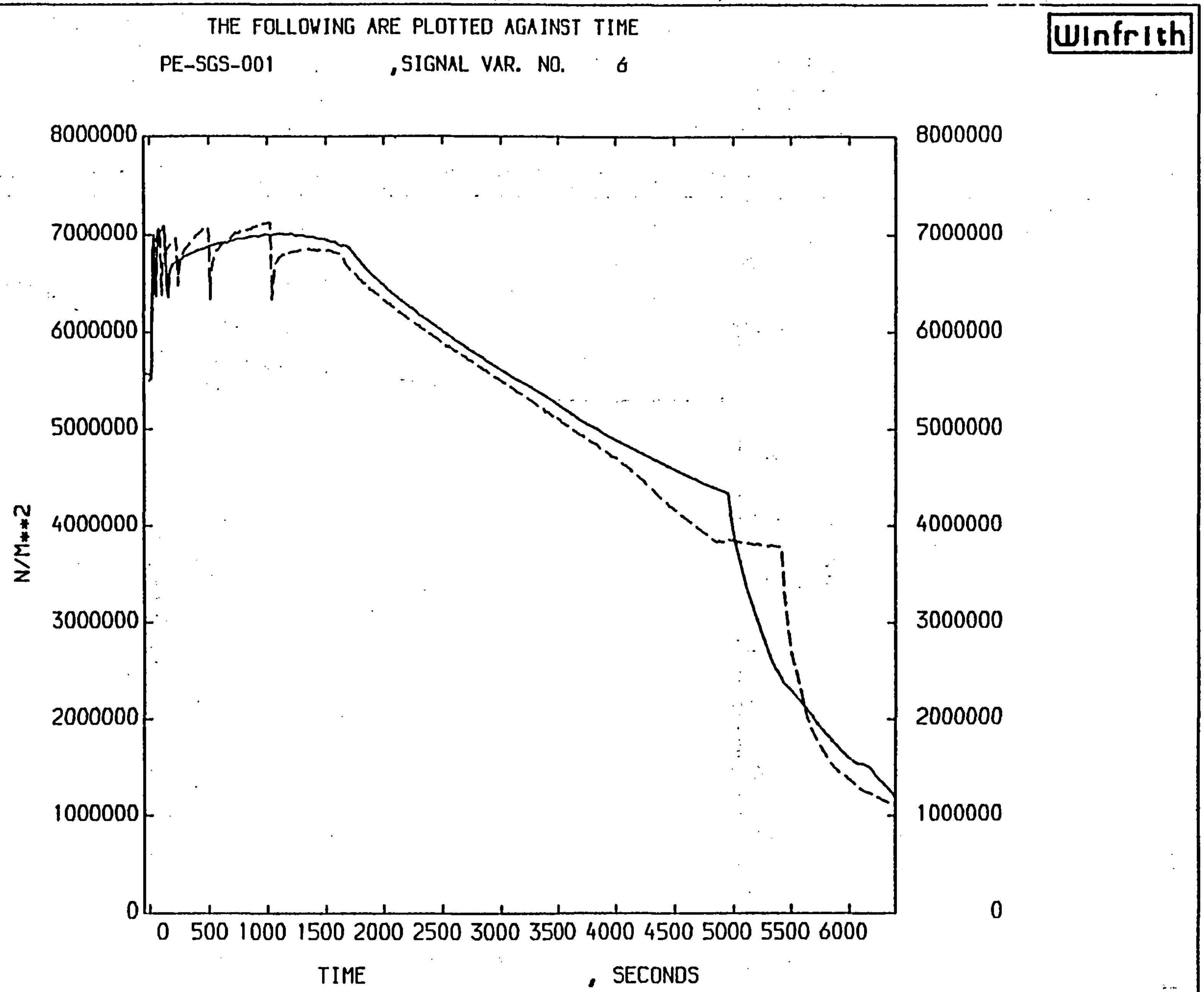

FIGURE 9

LP-SB-3 CALCULATION
SECONDARY SYSTEM PRESSURES (0-6400 SECONDS)

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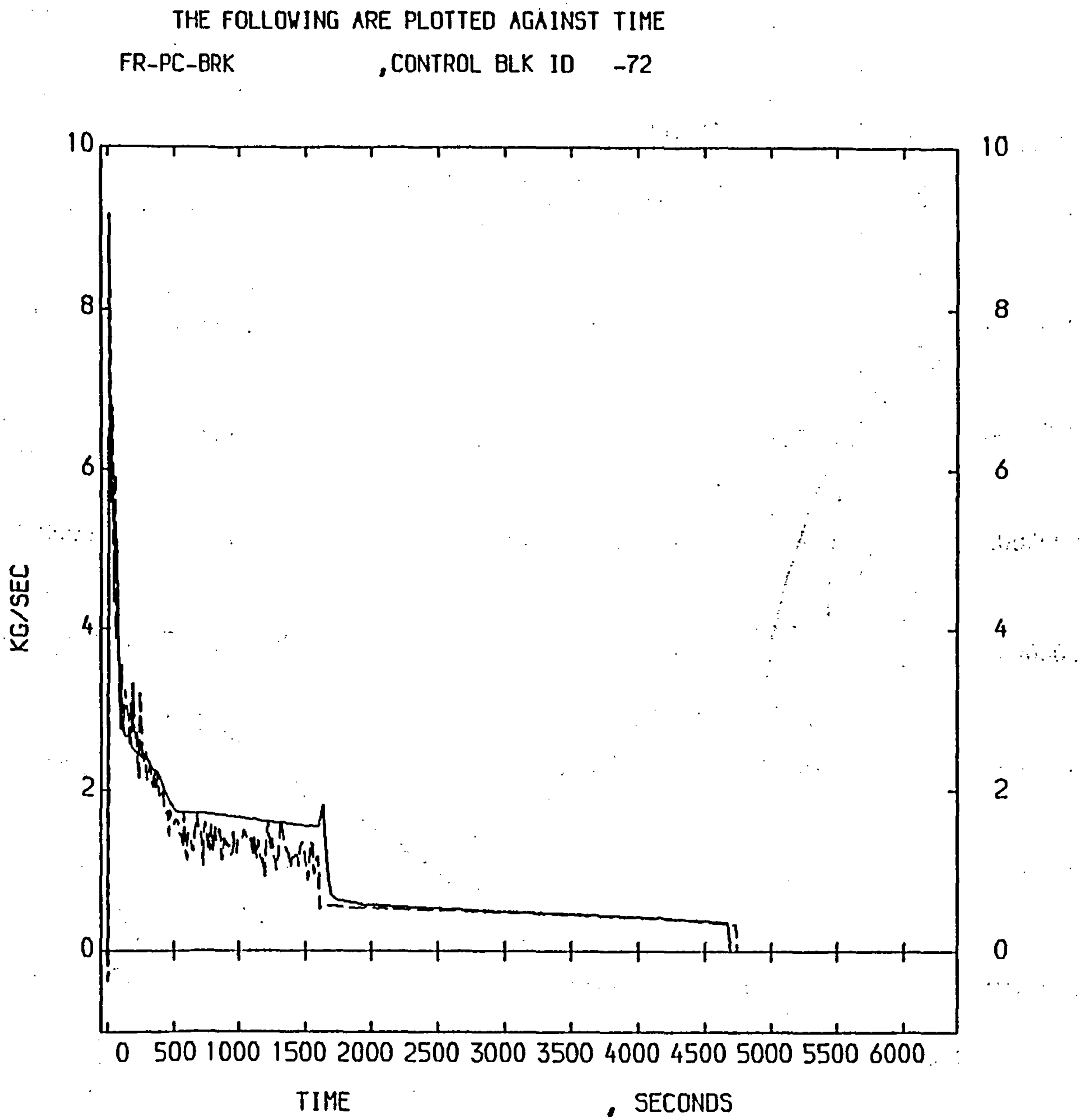

FIGURE 10

LP-SB-3 CALCULATION
BREAK MASS FLOW RATE

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THE FOLLOWING ARE PLOTTED AGAINST TIME

DE-PC-S05B ,MIXTURE DENSITY

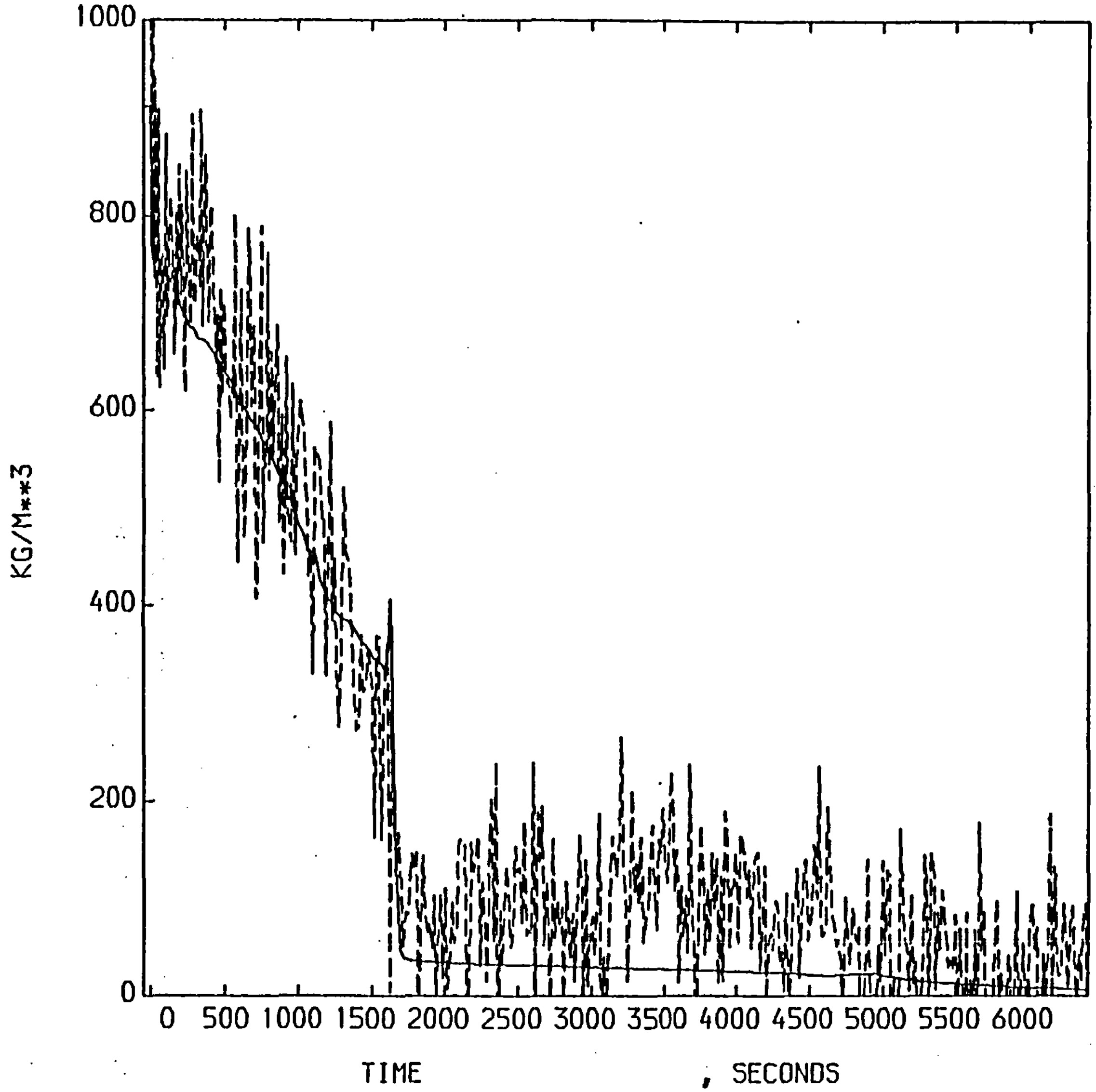

FIGURE 11

LP-SB-3 CALCULATION
BREAK LINE DENSITY

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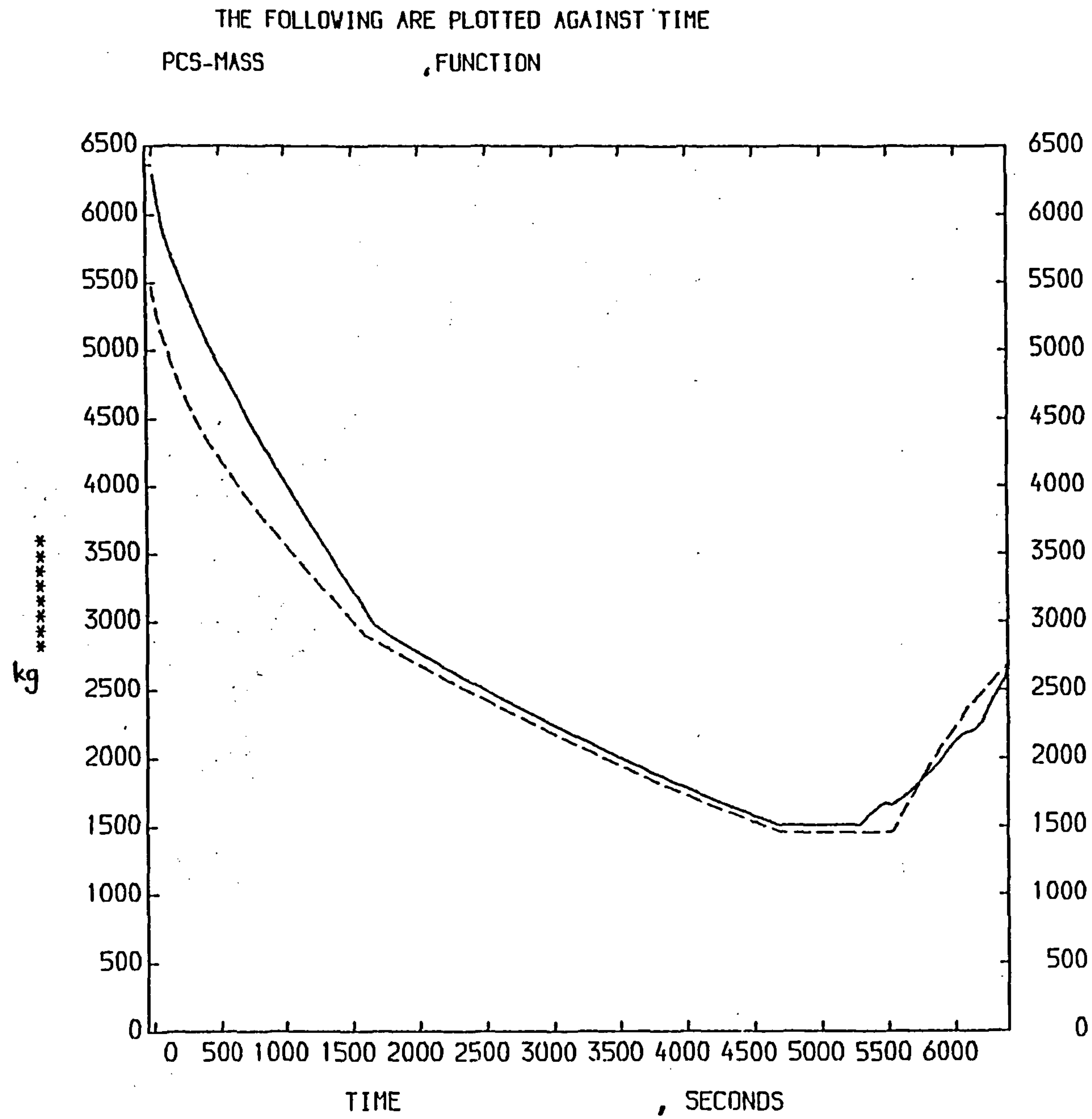

FIGURE 12

LP-SB-3 CALCULATION
PRIMARY SYSTEM MASS INVENTORY

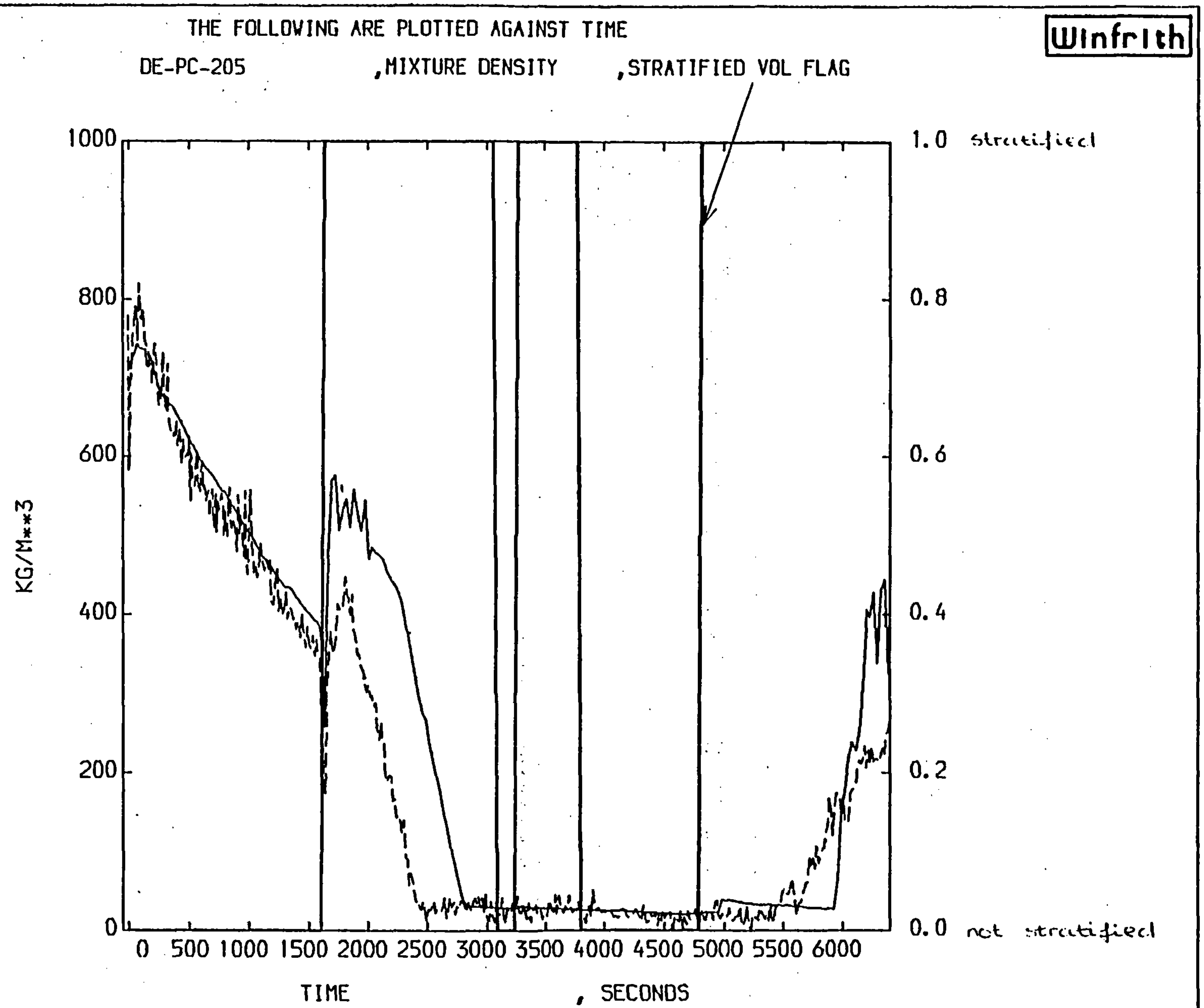

FIGURE 13

LP-SB-3 CALCULATION
ILHL DENSITY AND STRATIFICATION ONSET

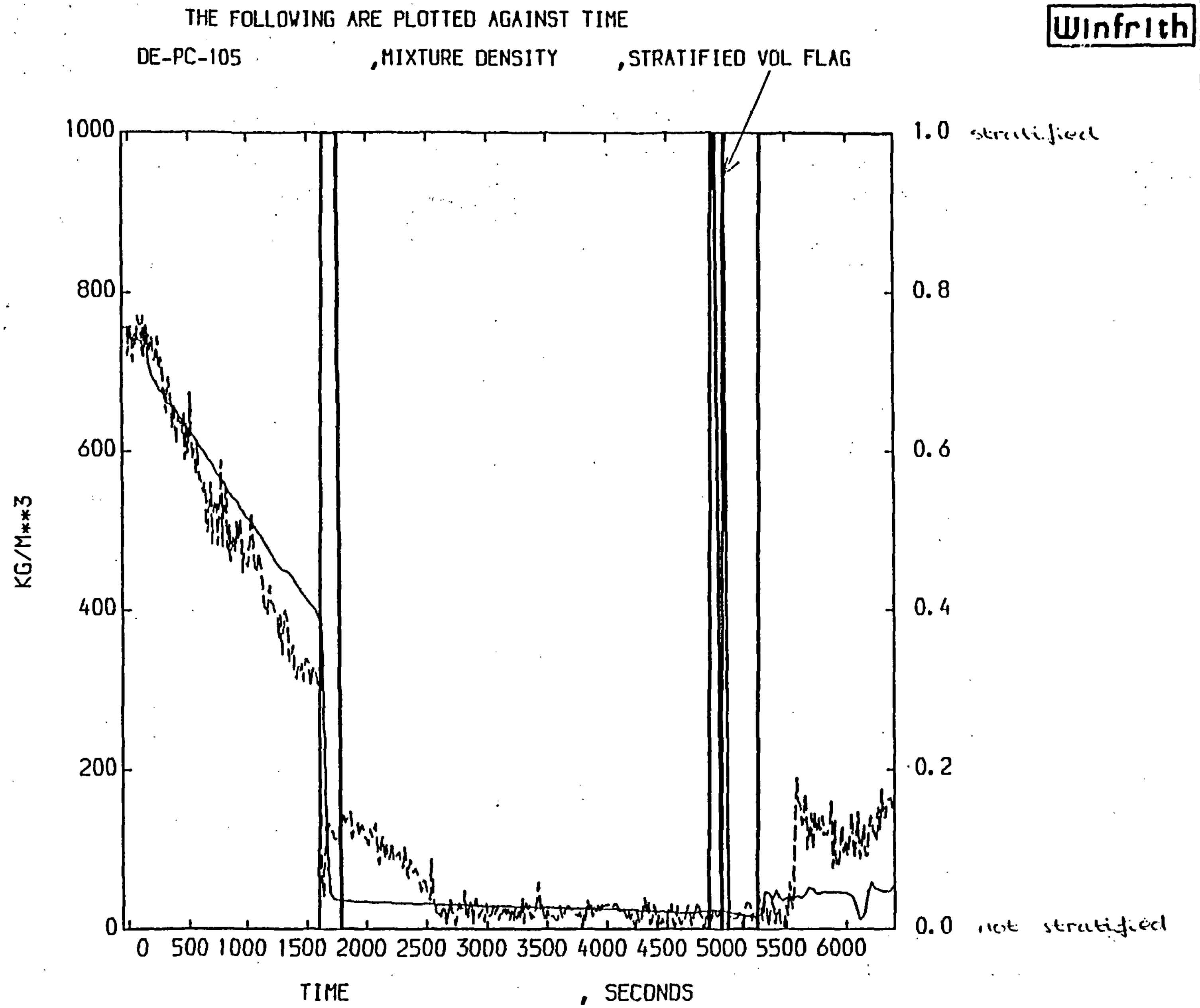

FIGURE 14

LP-SB-3 CALCULATION
ILCL DENSITY AND STRATIFICATION ONSET

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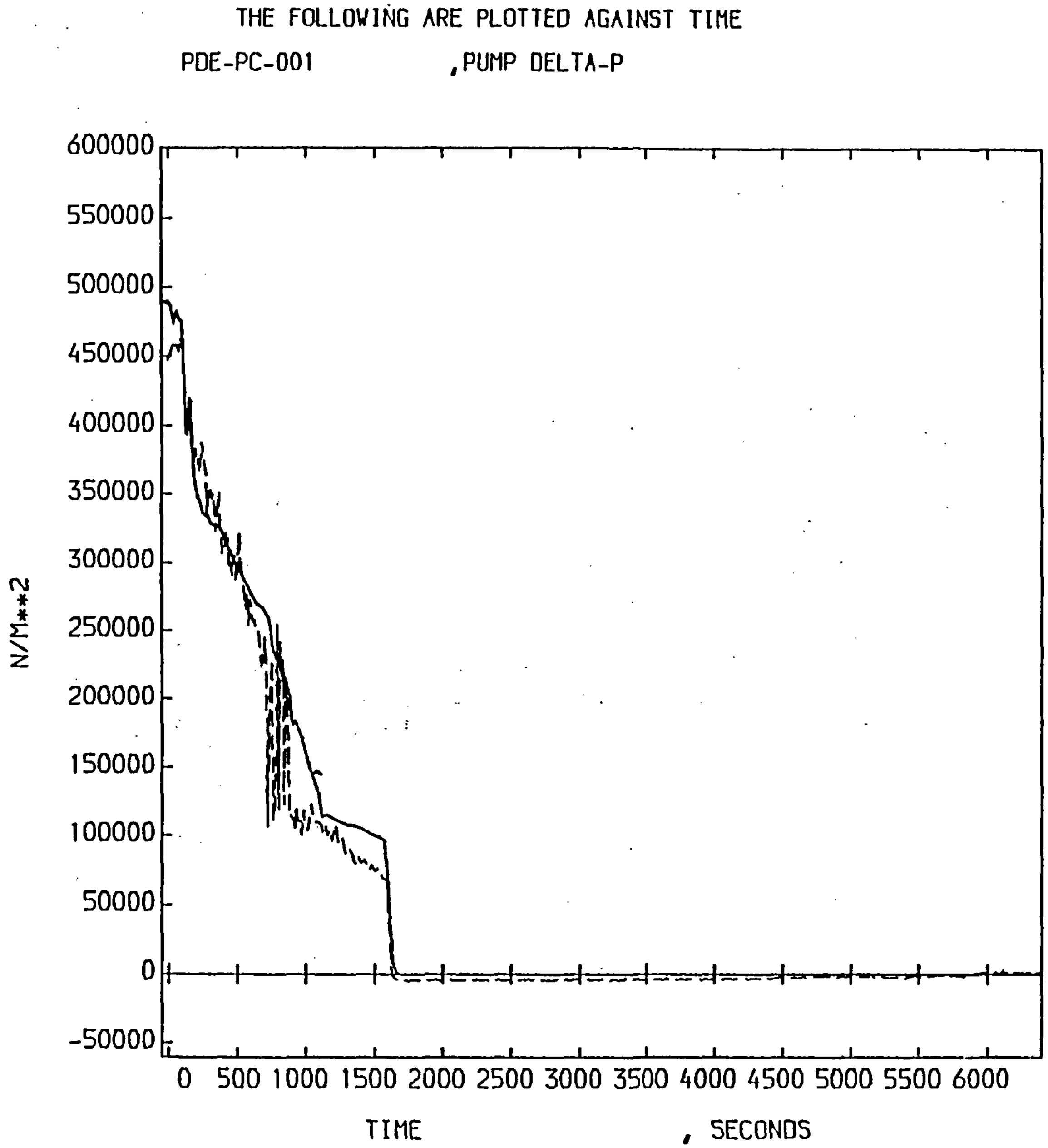

FIGURE 15

LP-SB-3 CALCULATION
PRIMARY COOLANT PUMPS DIFFERENTIAL PRESSURE

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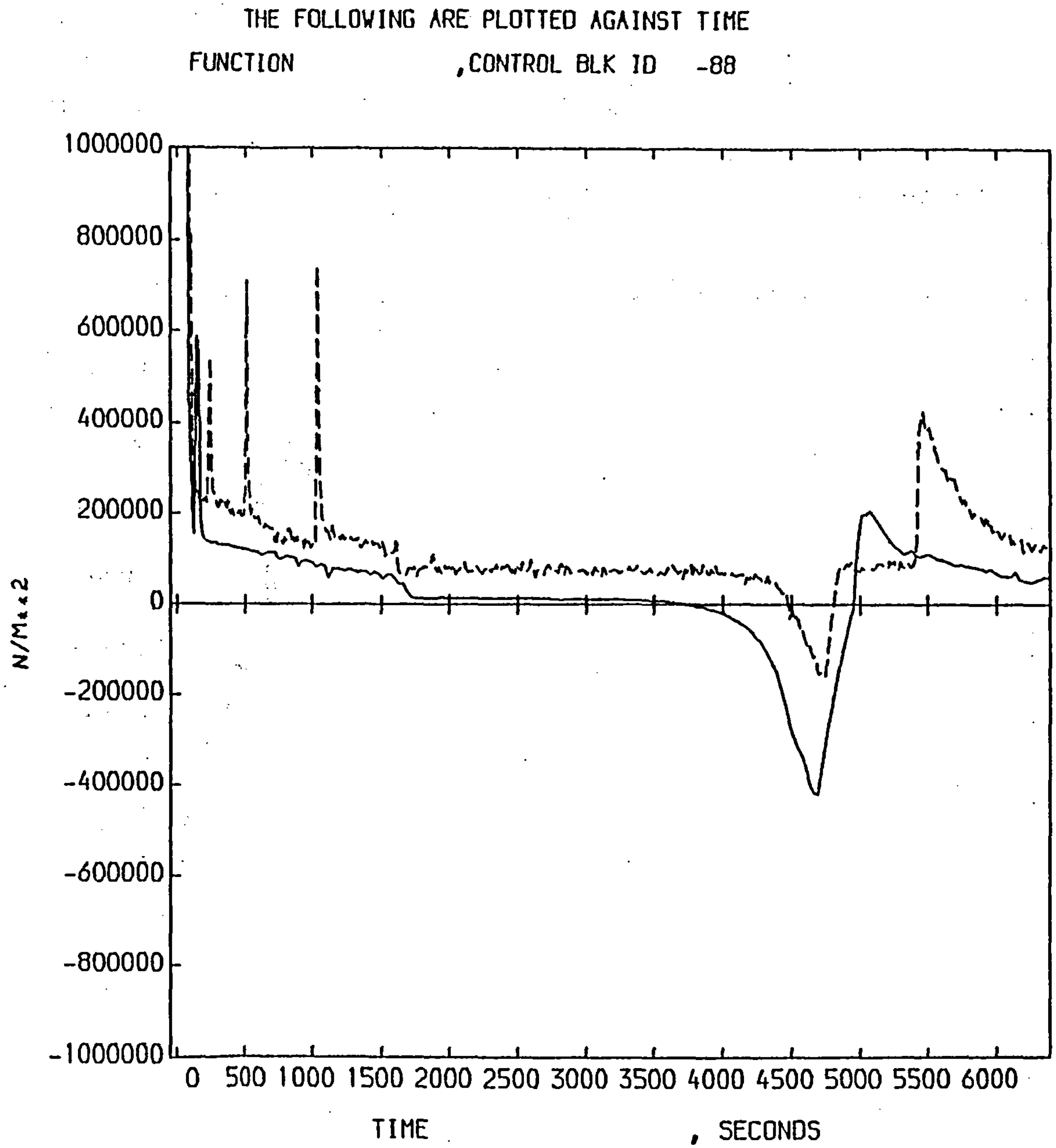

FIGURE 16

LP-SB-3 CALCULATION

PRIMARY SYSTEM - SECONDARY SYSTEM PRESSURE

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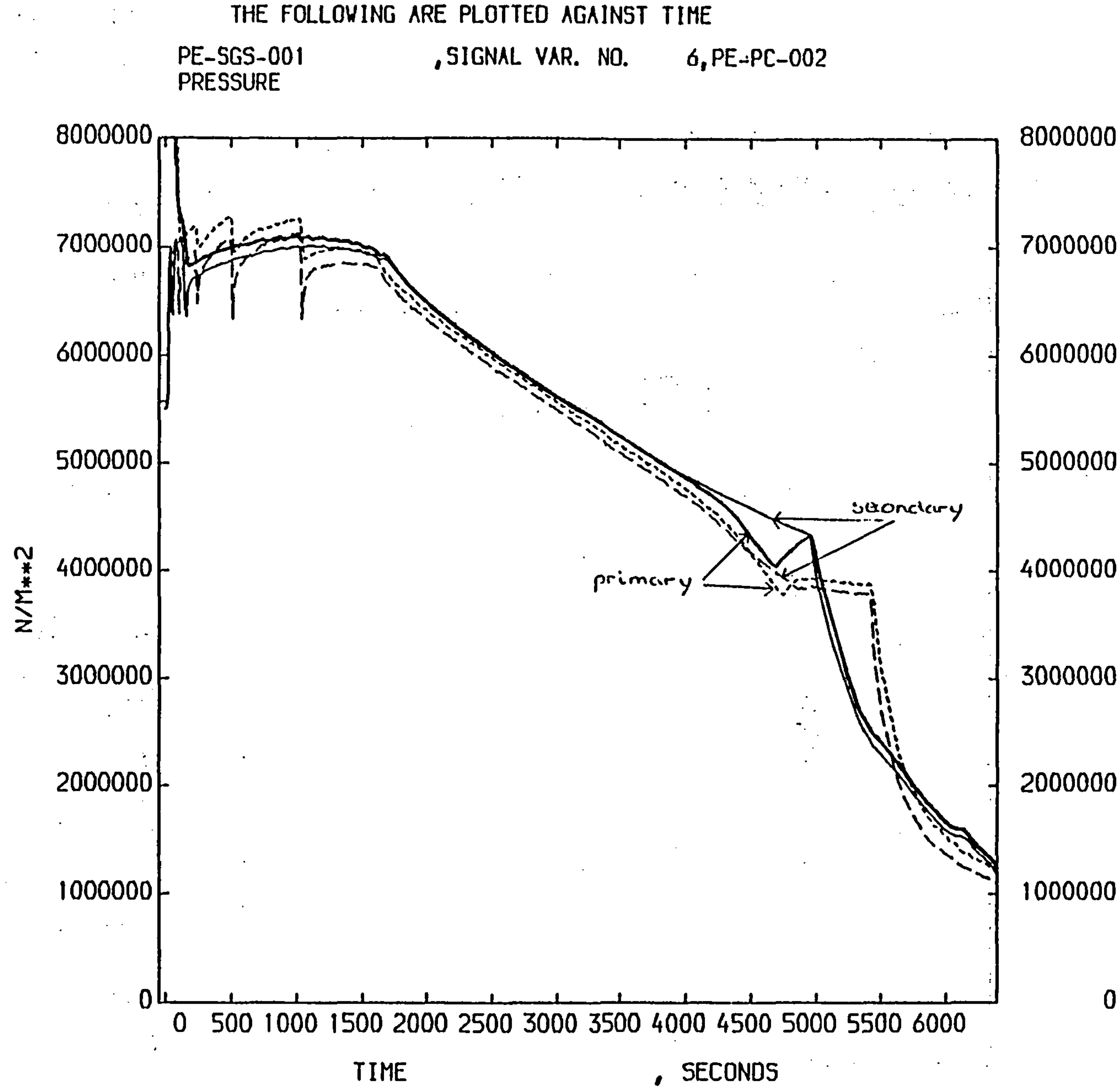

FIGURE 17

LP-SB-3 CALCULATION
PRIMARY AND SECONDARY SYSTEM PRESSURES

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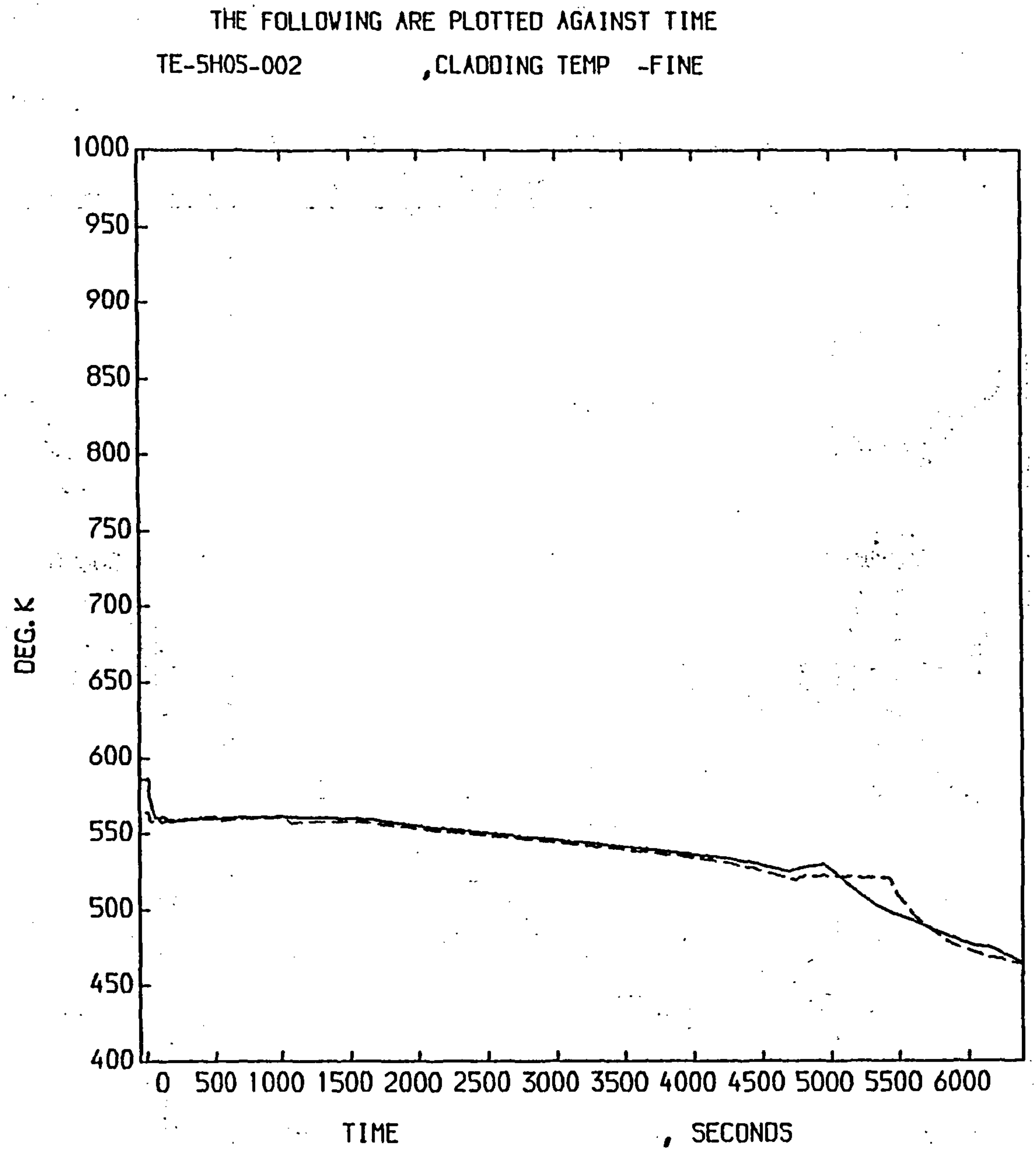

FIGURE 18

LP-SB-3 CALCULATION
PEAK CLAD TEMPERATURES (0.05M)

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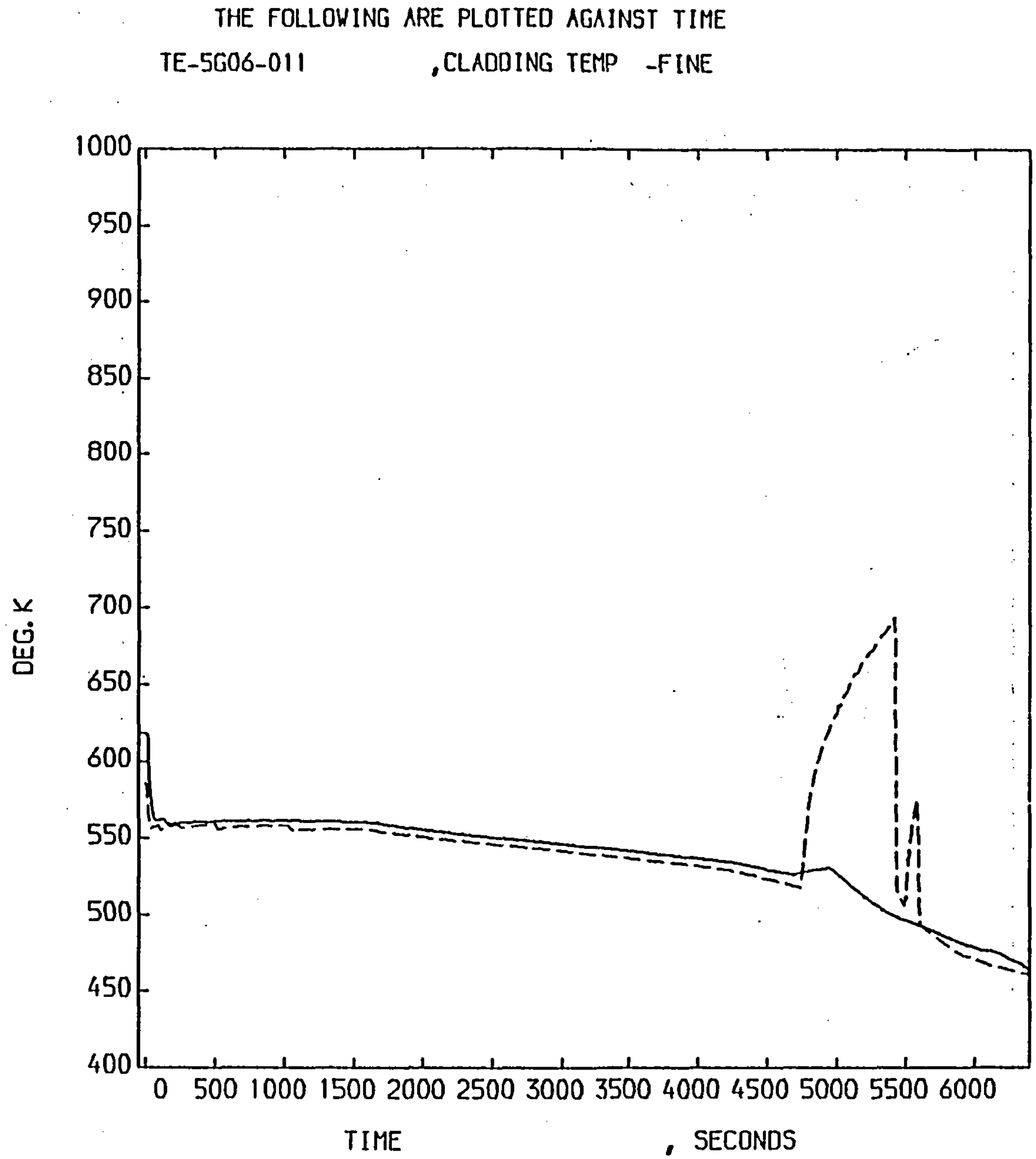

FIGURE 19

LP-SB-3 CALCULATION
PEAK CLAD TEMPERATURES (0.28M)

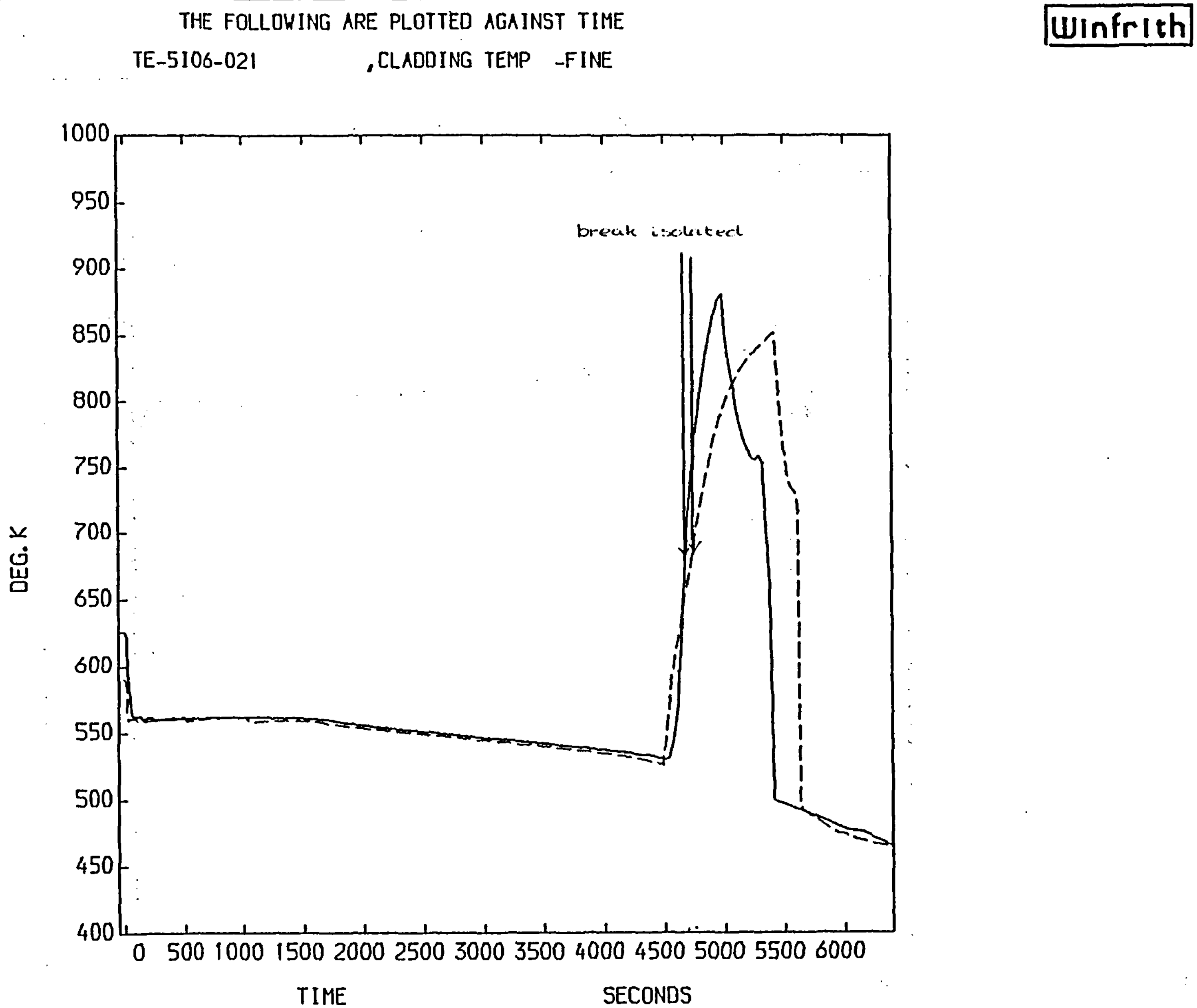

FIGURE 20

LP-SB-3 CALCULATION

PEAK CLAD TEMPERATURES (0.53M)

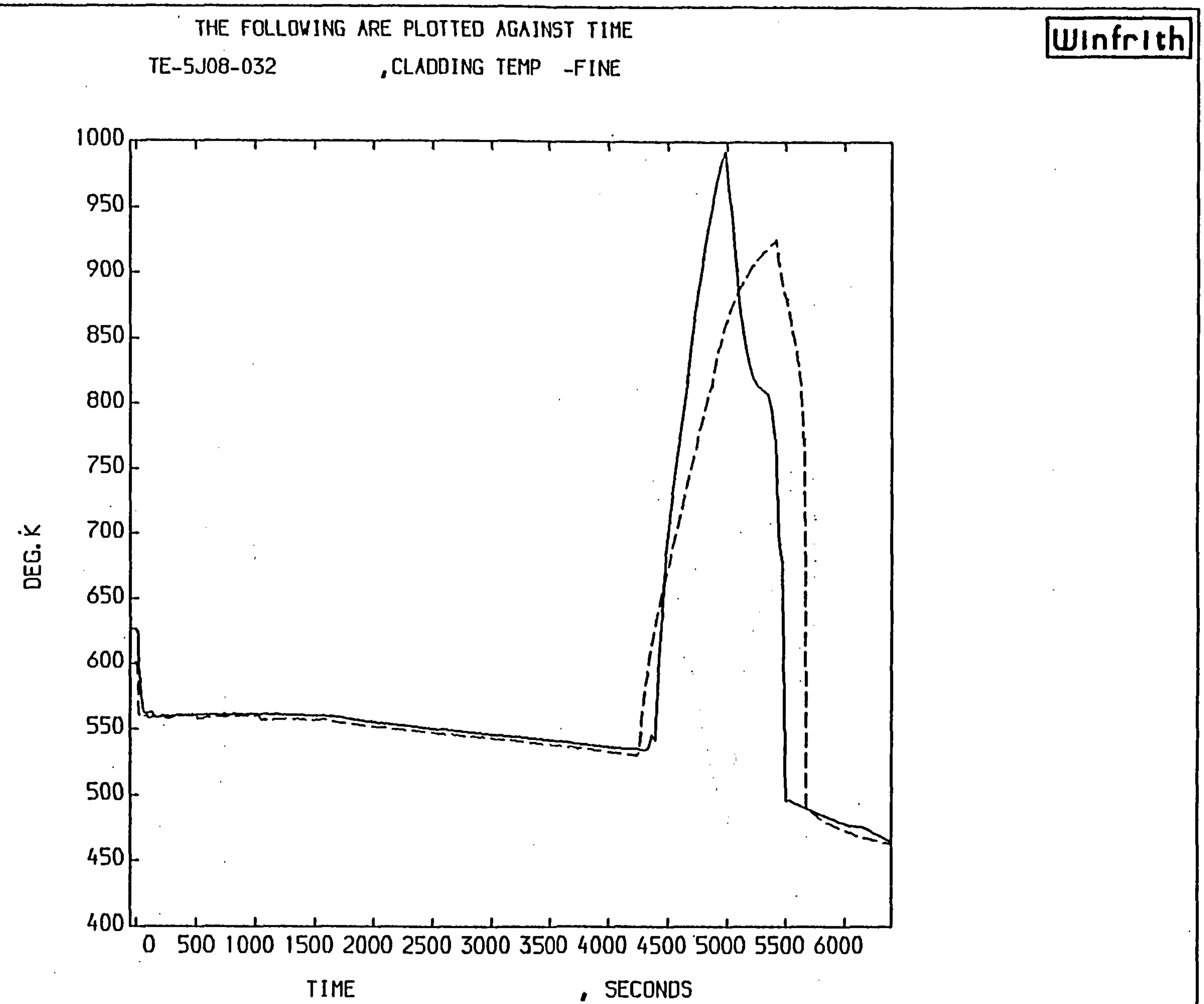

FIGURE 21

LP-SB-3 CALCULATION
PEAK CLAD TEMPERATURES (0.81M)

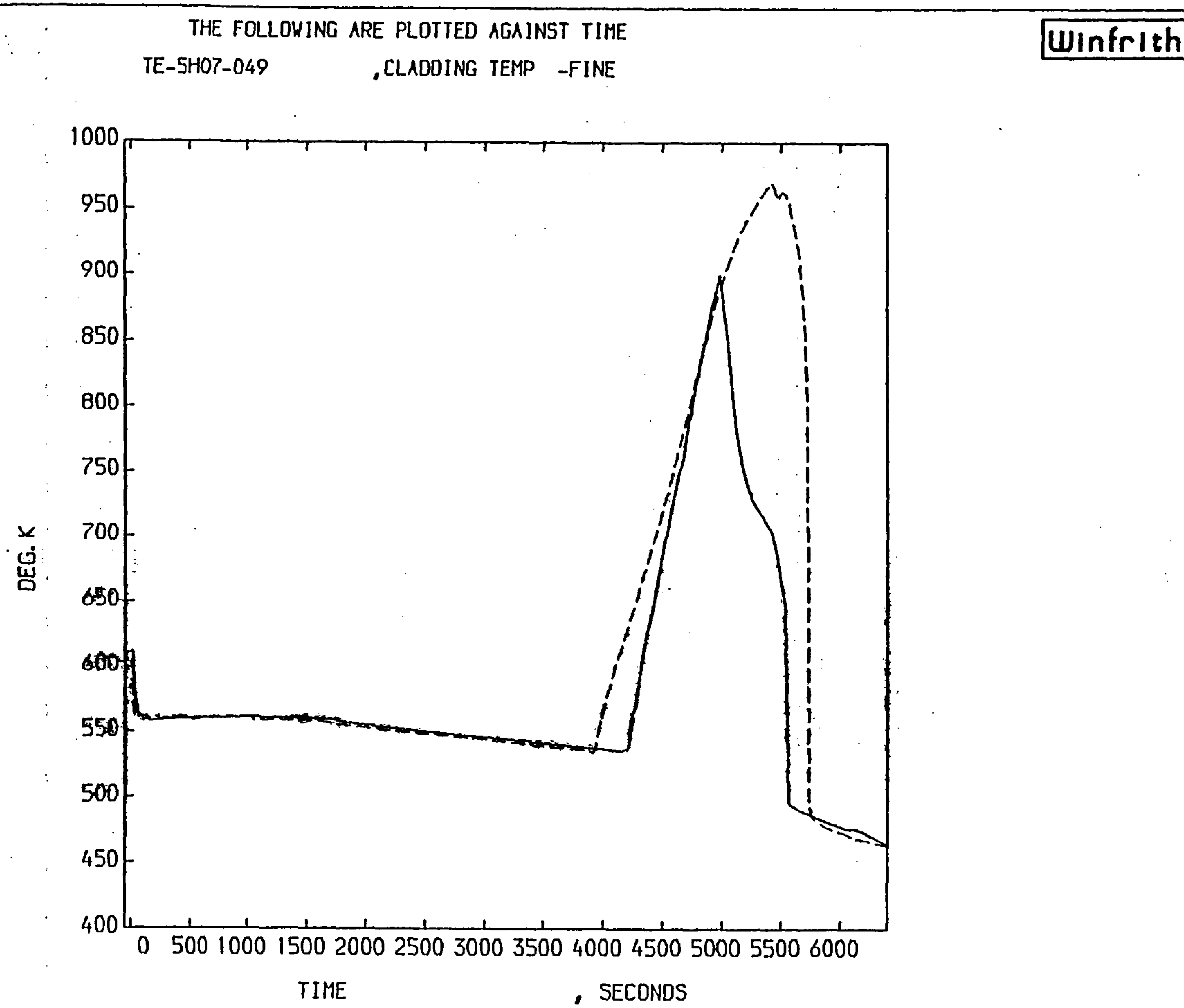

FIGURE 22

LP-SB-3 CALCULATION
PEAK CLAD TEMPERATURES (1.25M)

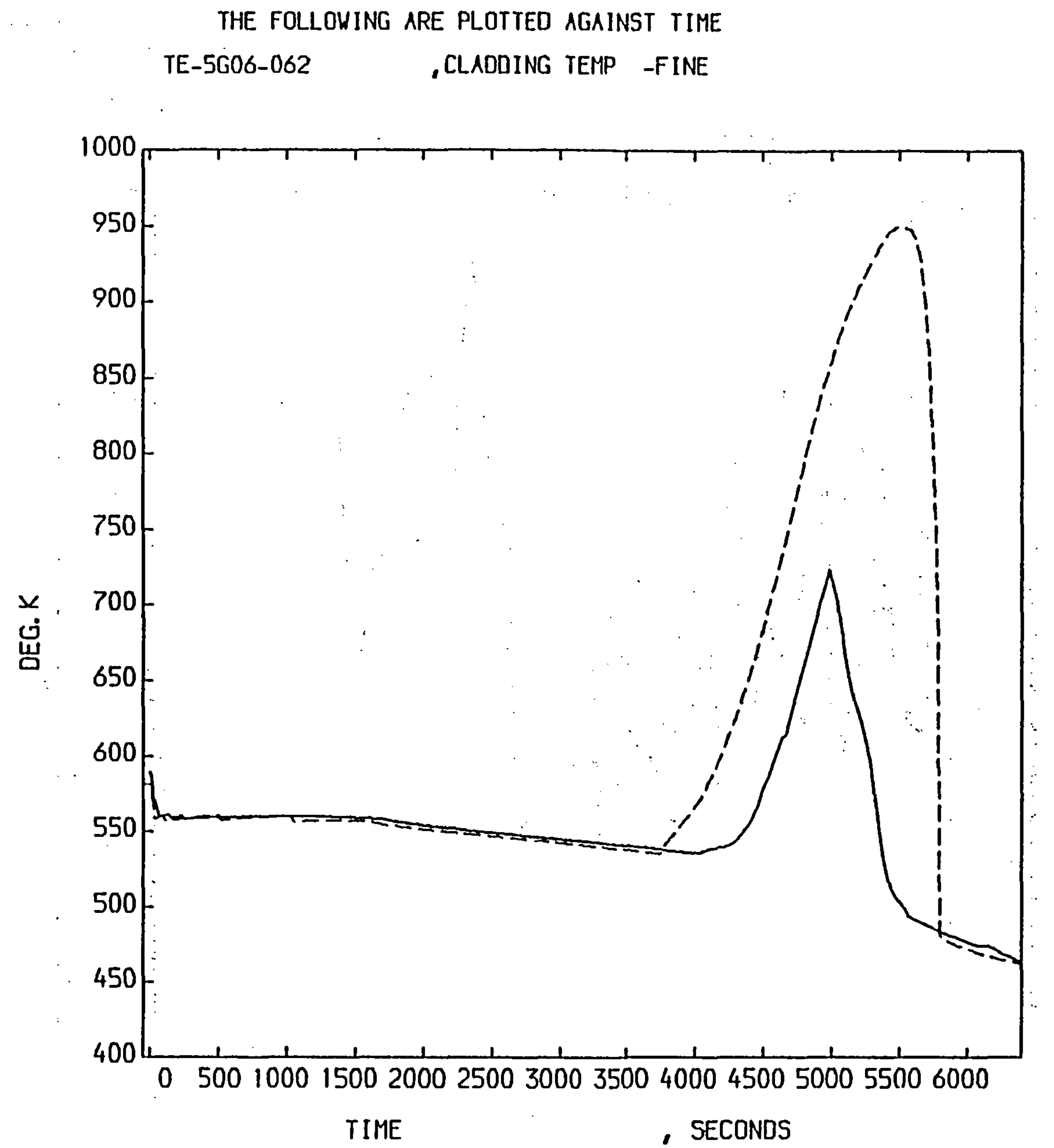

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FIGURE 23

LP-SB-3 CALCULATION
PEAK CLAD TEMPERATURES (1.57M)

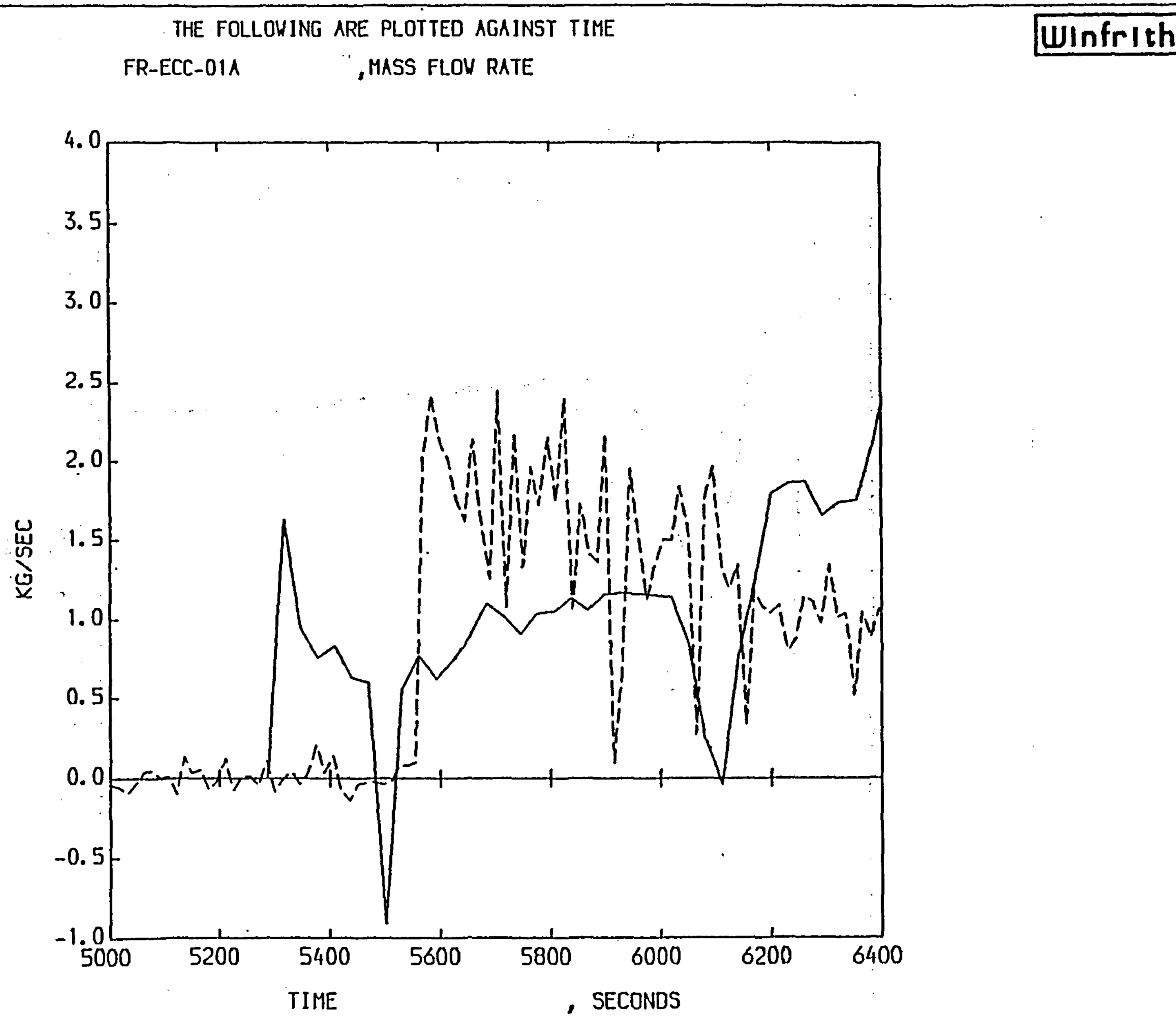

FIGURE 24

LP-SB-3 CALCULATION
ACCUMULATOR MASS FLOW

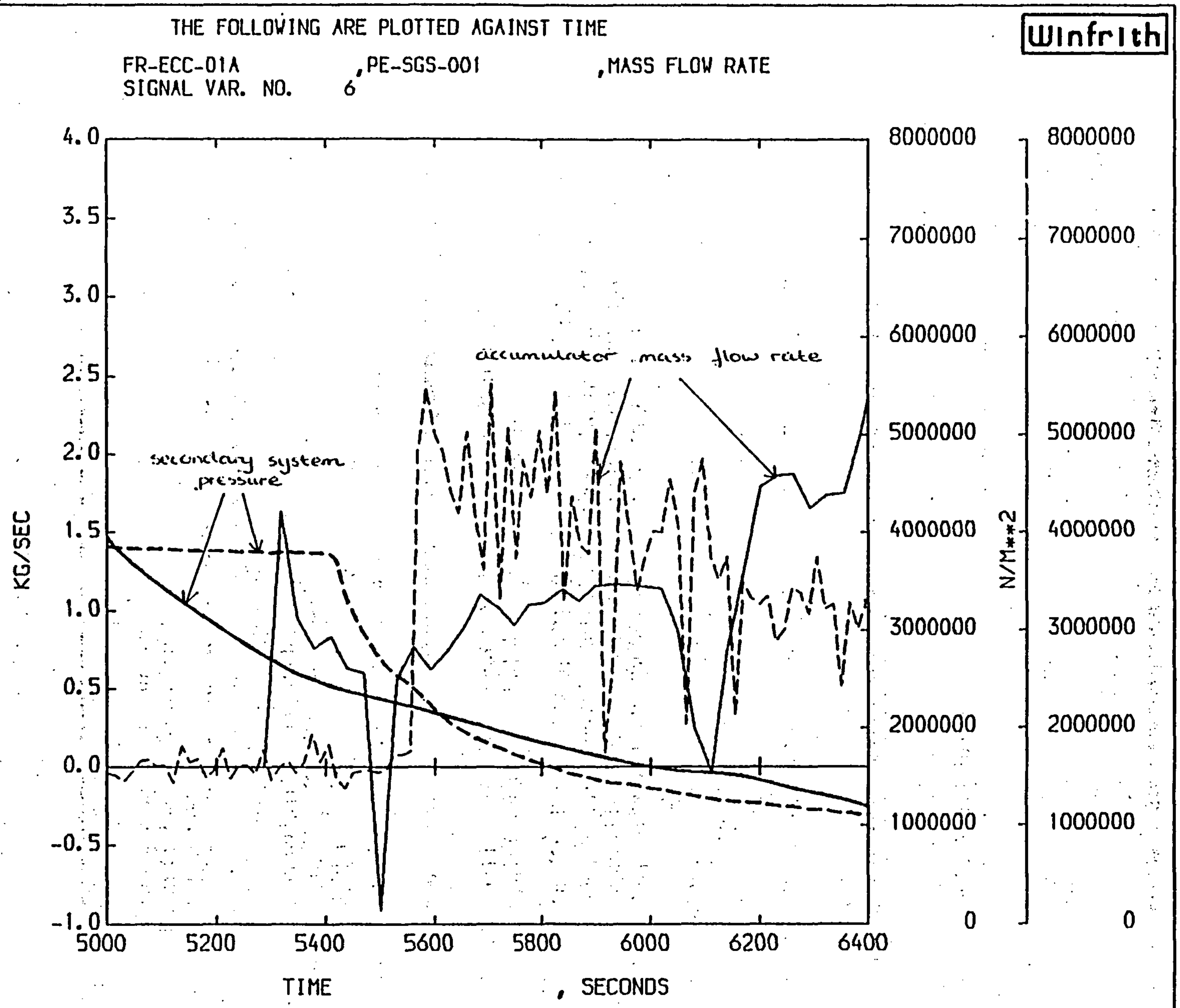

FIGURE 25

LP-SB-3 CALCULATION

ACCUMULATOR MASS FLOW RATE AND SECONDARY SYSTEM PRESSURE

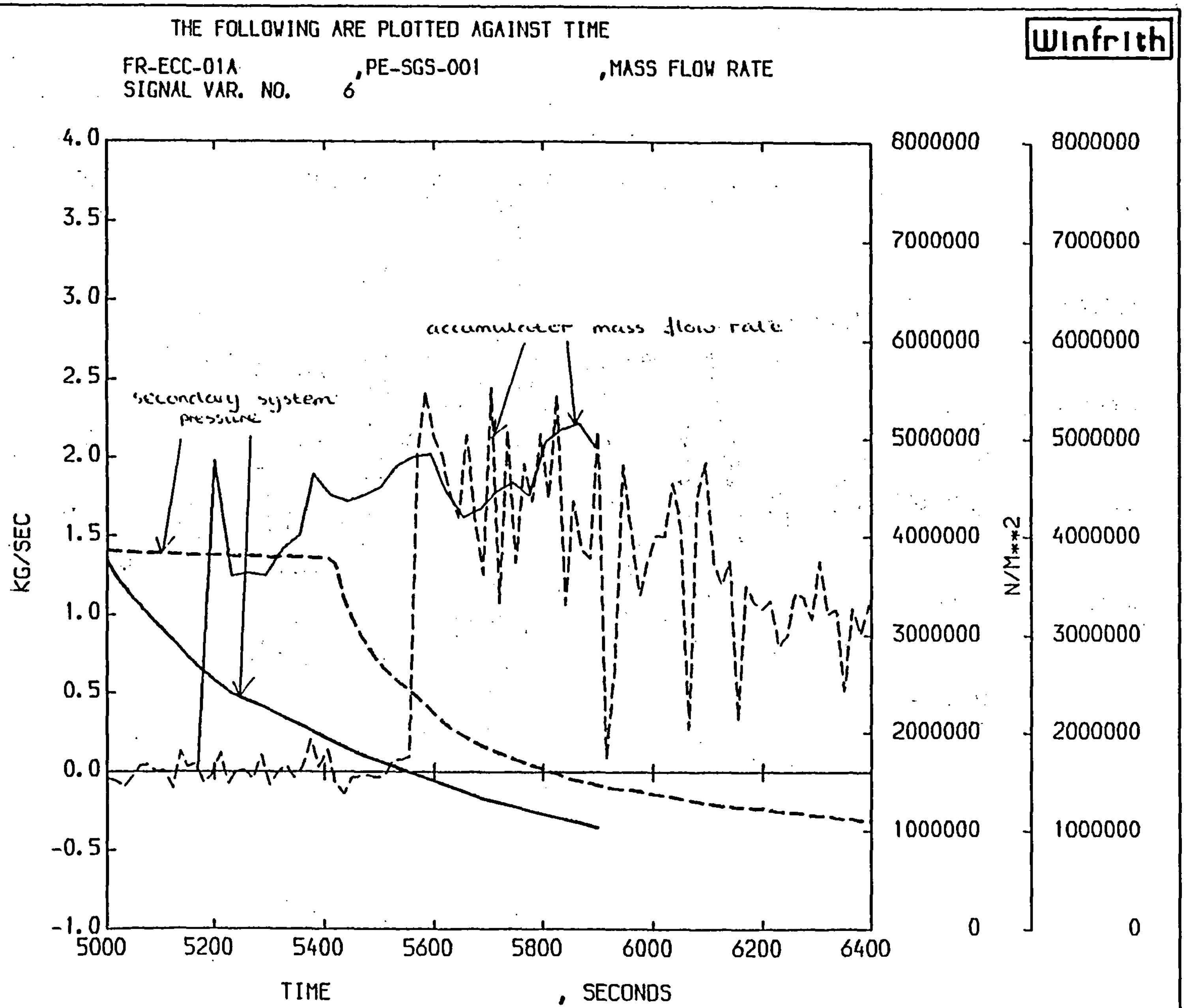

FIGURE 26

LP-SB-3 CALCULATION

ACCUM MASS FLOW RATE AND SEC PRES (ST BYPASS VALVE AREA INCREASED)

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Analysis of the small, cold leg break, OECD LOFT Experiment LP-SB-3 using the best-estimate computer code TRAC-PF1/MOD1 is presented.

Descriptions of the LOFT facility and the LP-SB-3 experiment are given and development of the TRAC-PF1/MOD1 input model is detailed. The caluclations performed in achieving the steady state conditions, from which the experiment was initiated, and the specification of experimental boundary conditions are outlined.

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