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CODE ASSESSMENT AND APPLICATIONS PROGRAM

PRETEST PREDICTION OF SEMISCALE TEST S-07-10B

By

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IDAHO NATIONAL ENGINEERING LABORATORY

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INTERIM REPORT

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ABSTRACT

A best estimate prediction of Semiscale Test S-07-10B was performed at INEL by EG&G Idaho as part of the RELAP4/MOD6 code assessment effort and as the Nuclear Regulatory Commission pretest calculation for the Small Break Experiment. The RELAP4/MOD6 Update 4 and the RELAP4/MOD7 computer codes were used to analyze Semiscale Test S-07-10B, a 10% communicative cold leg break experiment. The Semiscale Mod-3 system utilized an electrically heated simulated core operating at a power level of 1.94 MW. The initial system pressure and temperature in the upper plenum was 2276 psia and 604°F, respectively.

SUMMARY

A best estimate pretest analysis of Semiscale Test S-07-10B was performed at INEL by EG&G Idaho as the Nuclear Regulatory Commission Division of Systems Safety (NRC-DSS) pretest prediction for the Small Break Experiment (SBE).

The analysis was performed using the RELAP4/MOD6 Update 4 and RELAP4/MOD7 computer codes. The system nodalization is based on guidelines developed for the RELAP4/MOD6 code assessment effort at INEL. The analysis thus serves the dual purpose of being part of the assessment effort as well as the INEL/NRC SBE pretest prediction.

Semiscale Test S-07-10B, the experiment data base for SBE, was a 10% communicative cold leg break with initial conditions of 2276 psia and 604°F (upper plenum). ECC injection was limited to the intact loop cold leg. The power density was 9.18 kW/ft in the 9 high power rods. There were also 14 low power rods (5.65 kW/ft) and 2 unpowered rods locations in the simulated core.

The results of the analysis show good comparison between the RELAP4/MOD6 and RELAP4/MOD7 calculations. The predicted system response showed expected results. A comparison of the pretest prediction to data plus any posttest analyses will be presented in the SBE final report.

CONTENTS

FIGURES

TABLES

I. INTRODUCTION

The following report documents the INEL pretest analysis of Semiscale Test S-07-10B[1]. The analysis was performed at INEL by EG&G Idaho as part of the RELAP4/MOD6 code assessment effort. The analysis also constitutes the Nuclear Regulatory Commission (NRC) pretest prediction for the Small Break Experiment (SBE)[2] as part of the United States Standard Problem program. The Standard Problem program is a continuing effort by the NRC to evaluate the adequacy of participant computer codes for both best estimate and licensing calculations. The program utilizes a series of separate effects and integral experiments to help better define code capabilities and future code development efforts.

The RELAP4/MOD6[3] Update 4 computer code and an experimental version of the RELAP4/MOD7[4] code were used to perform the analysis of Test S-07-10B. A discussion of the computer code updates required for the analysis is presented in Section II. The system nodalization utilized for the study is described and the analytical and systemic modeling features used are discussed in Section III.

Semiscale Test S-07-10B, the experimental data base for the SBE, was an electrically heated core test performed to experimentally characterize the thermal-hydraulic behavior of the Mod-3 system during a small break loss of coolant accident. Test S-07-10B was conducted with a communicative cold leg break, the break area scaled to represent 10% of the area of a cold leg pipe in a pressurized water reactor. The simulated core consisted of 9 high power rods (9.18 kW/ft), 14 low power rods (5.65 kW/ft), an unpowered rod and a liquid level probe. The initial conditions and specified test parameters used for the SBE are presented in Section III.

The results of the analysis are presented in Section IV with a general discussion of the analysis.

II. COMPUTER CODE DESCRIPTION

The INEL/NRC SBE pretest analysis was peformed using an updated version of the RELAP4/MOD6 Update 4 computer code, stored at INEL under Configuration Control Numbers C0010006 (RELAP4/MOD6) and H002011B (steam tables). The code was updated to:

1. Self initialize the system pressure balance at problem initiation.

2. Correct known coding errors in Wilson bubble rise model.

3. Record calculated cladding temperature information corresponding to the physical location of the heater rod thermocouples (Figure 1). This allows reporting calculated cladding temperatures at the actual measurement location.

These update directives are stored with the RELAP4 input deck used for the SBE pretest analysis at INEL under Configuration Control Number H003584B.

An identical calculation was performed with an experimental version of the RELAP4/MOD7 computer code (identified internally as Version 87), stored at INEL under Configuration Control Numbers C0010007 (RELAP4/MOD7) and H009982B (Steam Tables). The code was updated to

1. Record calculated cladding temperature information corresponding to the physical location of the heater rod thermocouples (Figure 1). This allows reporting calculated cladding temperatures at the actual measurement location.

2. Remove time step control for zero flow crossings at junctions during countercurrent flow.

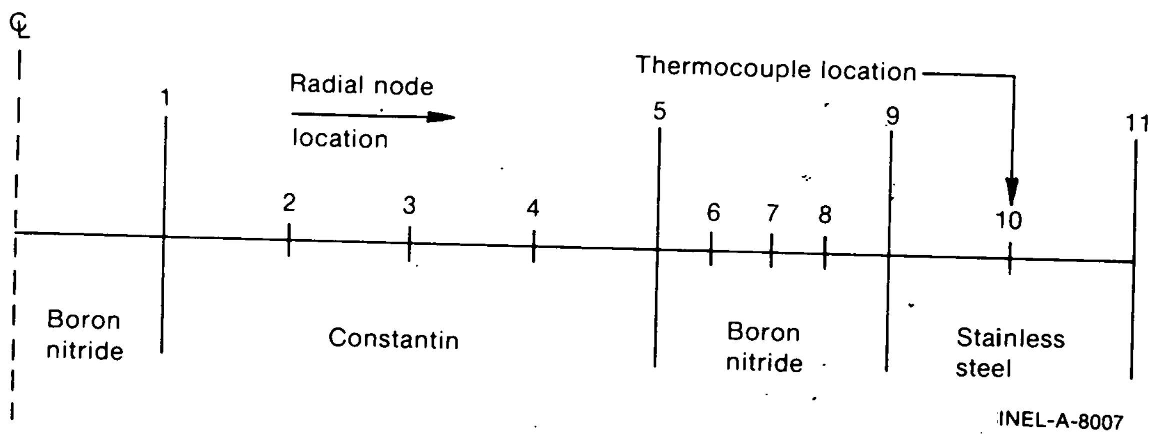

Figure 1 RELAP4 heater rod nodalization scheme.

These update directives are stored with the RELAP4 input deck at INEL under Configuration Control Number H004984B.

III. INPUT MODEL

The system nodalization used for the SBE pretest prediction was developed by the Semiscale Program for use with the RELAP4/MOD6 Update 4 computer code. The Semiscale Mod-3 test assembly is represented by 42 volumes, 56 junctions, and 50 heat slabs as shown in Figure 2 and described in Table I. The input model allows the analysis to be used as part of the assessment effort for RELAP4/MOD6 as well as the INEL/NRC pretest prediction for the SBE.

The analysis involved the use of numerous analytical modeling features contained in the RELAP4/MOD6 computer code. Comments on the major modeling options used (both analytical and systemic) are listed below.

1. MVMIX = 0 is used at all junctions, except the MVMIX = 3 is used at:

   JUN 2 (accumulator outlet)
   JUN 4 (pressurizer outlet)
   JUNS 31, 33 (support tube inlet and outlet)
   JUNS 32, 34, 37, 39, 40 (guide tube inlet and outlets)
   JUNS 47, 50, 54, 56 (steam generator secondary feedwater inlets)
   JUNS 52, 53 (LPIS, HPIS)

2. Vertical slip is used at all vertical junctions in the model except in the steam generator tubes. The junctions with slip are:

   JUN 36 (downcomer outlet)
   JUNS 35 (lower plenum)
   JUNS 1, 43, 44, 45, 46, 5 (core)
   JUNS 6, 7 (upper plenum)
   JUNS 31, 32, 33, 34 (support tubes and guide tubes)

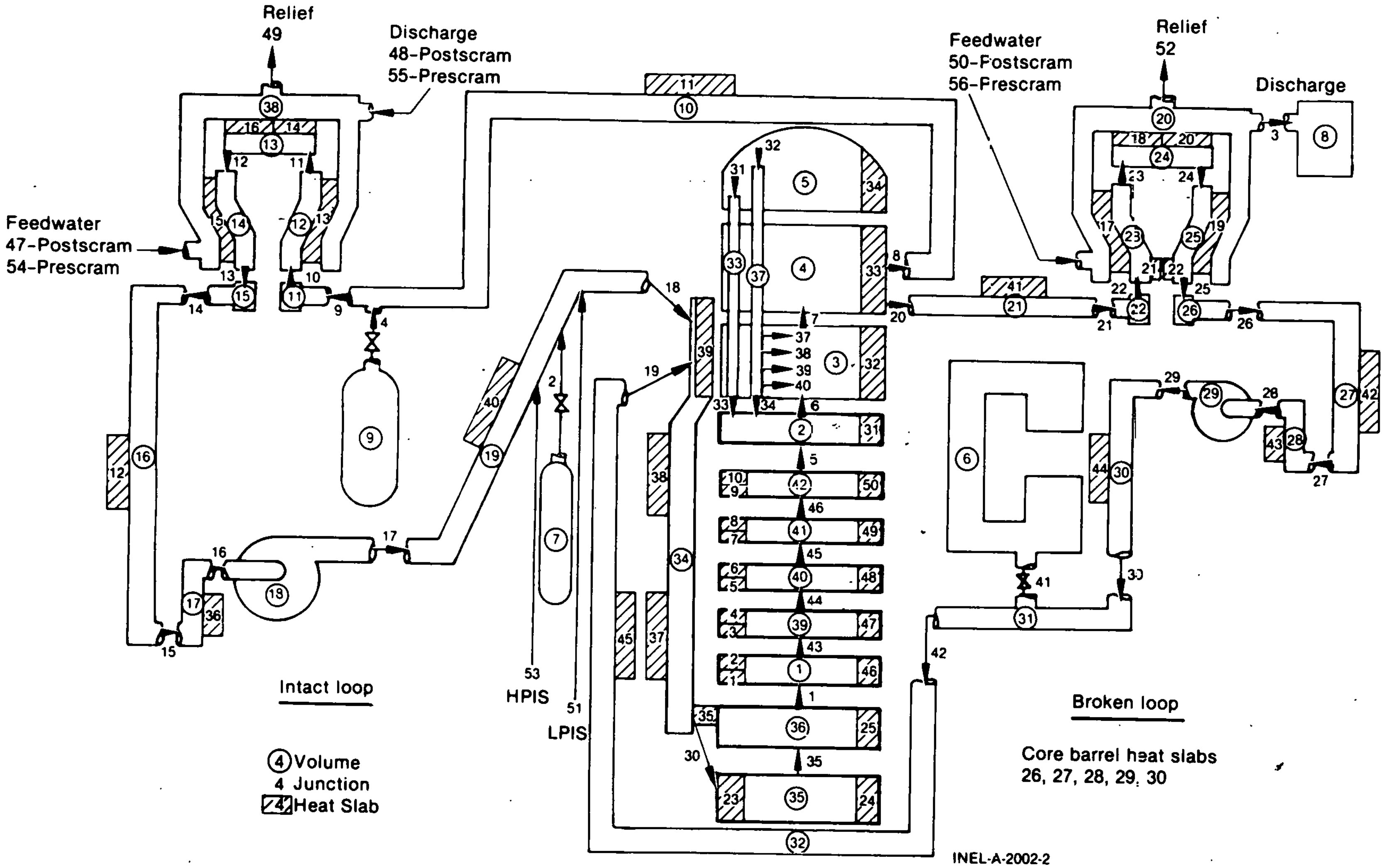

Figure 2 RELAP4/MOD6 nodalization for Test S-07-10B pretest prediction.

TABLE I

VOLUME AND HEAT SLAB DESCRIPTION FOR THE INEL/NRC SBE SYSTEM NODALIZATION

Volume No.	Description
10	Intact loop hot leg
11	Intact loop steam generator inlet plenum
12, 13, 14	Intact loop steam generator tube bundle
15	Intact loop steam generator outlet plenum
16	Intact loop pump suction - downflow
17	Intact loop pump suction - upflow
18	Intact loop pump
19	Intact loop cold leg
9	Pressurizer
7	Accumulator - intact loop
38	Intact loop steam generator secondary
21	Broken loop hot leg
22	Broken loop steam generator inlet plenum
23, 24, 25	Broken loop steam generator tube bundle
26	Broken loop steam generator outlet plenum
27	Broken loop pump suction - downflow
28	Broken loop pump suction - upflow
29	Broken loop pump
30	Broken loop pump discharge
31	Break assembly
32	Broken loop cold leg adjacent to the vessel
20	Broken loop steam generator secondary
6	Pressure suppression vessel
8	Atmospheric dump
34	Inlet annulus and downcomer

TABLE I (Continued)

VOLUME AND HEAT SLAB DESCRIPTION FOR THE INEL/NRC SBE SYSTEM NODALIZATION

Volume No.	Description
35	Lower plenum
36	Core mixer box
3	Mid-volume of the upper plenum
5	Upper head
33	Support tubes
37	Guide tubes
1, 39, 40, 41, 42	Core
4	Top volume of the upper plenum
2	Bottom volume at the upper plenum

Heat Slab No.	Description
1,2,3,4,5,6,7,8,9,10	High power rods
46,47,48,49,50	Low power rods
26,27,28,29,30	Core barrel
23, 24, 25	Lower plenum
31, 32, 33, 34	Upper plenum/head
37, 38, 39	Downcomer
11, 12, 36, 37	Intact loop piping
13, 14, 15, 16	Intact loop steam generator tubes
41, 42, 43, 44, 45	Broken loop piping
17, 18, 19, 20	Broken loop steam generator tubes
21, 22	Broken loop steam generator tube sheet

3. Wilson buble rise is used in the downcomer volume (VOL 34). Complete phase separation is modeled in the pressurizer (VOL 9) and the accumulator (VOL 7). The bubble rise model with constant VBUB is used in:

   VOL 5 (upper head)
   VOL 6 (suppression tank)
   VOL 16 (intact loop pump suction - downcomer)
   VOL 17 (intact loop pump suction - upflow)
   VOLS 20, 38 (steam generator secondaries)
   VOL 27 (broken loop pump suction - downflow)
   VOL 28 (broken loop pump suction - upflow)

4. The pressurizer surge line is lumped into the pressurizer volume.

5. Critical flow is modeled using the Henry-Fauske/Homogeneous Equilibrium Model option. A multiplier of 1.0 is applied to the subcooled and saturated flow with a transition quality of .02.

6. Core heater rods are modeled as follows:

Boron nitride	-	1 node
Constantan	-	4 nodes
Boron nitride	-	4 nodes
316 stainless steel	-	2 nodes

7. Valves will be included in the pressurizer line and accumulator line to shut off flow when the tanks are emptied.

8. Core heat transfer is calculed with the default and/or recommended options of RELAP4/MOD6 Update 4. These are (1) use of HTS2 heat transfer surface, (2) CHF calculated with recommended CHF correlations[a], (3) transition boiling calculated with the Tong Young transition boiling correlation, and (4) film boiling calculated with the Condie-Bengston III film boiling correlation.

9. The enthalpy transport model was used to initialize the calculation but was not used during the transient.

The input model was reviewed by a subcommittee of the EG&G Pretest Prediction Consistency Committee. A completed sign off sheet attesting to the acceptability of the model is contained in Appendix A.

A similar input model (shown in Figure 3) was used for the RELAP4/MOD7 calculation with an ECC mixing volume (Vol. 43) included for the nonequilibrium model and fill junction added to the broken loop steam generator secondary side as the steam discharge for self initialization. These new models and other changes in analytical modeling features are described below.

1. The new slip velocity model developed for RELAP4/MOD7 is utilized. The new model employs a flow regime dependent correlation which results in a more accurate value for interphase slip velocities.

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a The recommended correlations are the W-3 correlation for the subcooled regime, Hsu and Beckner's modified W-3 correlation for the saturated high flow regime and Smith and Griffith's modified Zuber for the saturated low flow regime.

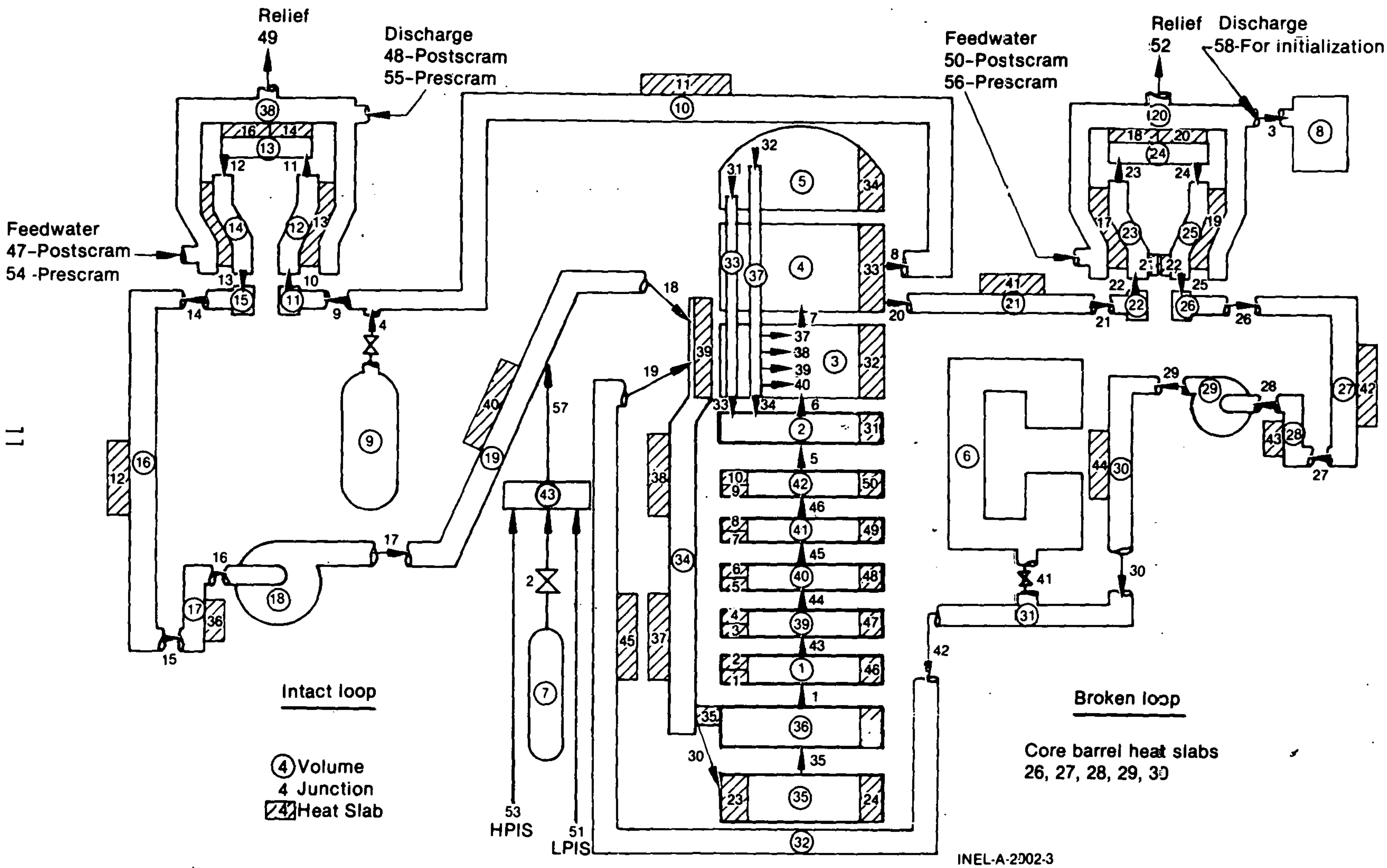

Figure 3 RELAP4/MOD7 nodalization for Test S-07-10B pretest prediction.

2. The nonequilibrium model developed for RELAP4/MOD7 is used. The model allows coexistance of subcooled emergency core cooling water and saturated primary system steam in a single volume. The model was applied to the intact loop cold leg and all reactor vessel volumes and is initiated 1 sec after the start of accumulator flow.

3. The RELAP4/MOD7 self-initialization routine was used to affect an initial system pressure and energy balance.

The analysis assumes initial steady state conditions in the Semiscale test facility based on the given initial conditions shown in Table II[1]. The initial primary pump speeds used for the calculation was higher than those specified for Test S-07-10B to facilitate an initial system balance. Also, the steam generator secondary side temperatures were changed slightly to achieve a system energy balance within $\pm 1\%$. The measured core power, pressure suppression tank pressure history, and pump speed histories provided as boundary conditions from Test S-07-10B are shown in Figures 4, 5, and 6 respectively.

The steam generator secondary side flow histories were also provided as boundary conditions with the exception of the broken loop steam generator discharge flow. The valve failed to close during the test so the discharge line is represented in the model as a dump to atmosphere. The line resistance is calculated based on the required initial flow rate for an energy balance and the known initial pressure drop across the line. The intact loop steam generator secondary side feedwater and discharge flowrates and the broken loop steam generator secondary side feedwater flowrate are shown in Figure 7.

TABLE II

INITIAL CONDITIONS FOR SEMISCALE TEST S-07-10B

System Parameter	Test S-07-10B	INEL SBE Calculation
Upper plenum pressure (psia)	2276.	2276.
Inlet fluid temperature (^{o}F)	541.	541.
Outlet fluid temperature (^{o}F)	604.	604.
Core power (kW)	1940.	1940.
Core flow rate (gpm)	202.7	202.7
Intact loop primary pump speed (rpm)	2215.	2392.
Broken loop primary pump speed (rpm)	13732.	15120.
Pressure suppression tank pressure (psia)	129.	129.
Pressurizer liquid mass (lbm)	22.9	22.9
Intact loop steam generator secondary side temperature (^{o}F)	516.8	530.4
Broken loop steam generator secondary side temperature (^{o}F)	524.2	530.9
Accumulator water volume (ft^3)	1.60	1.60
Accumulator gas volume (ft^3)	.88	.88
Accumulator pressure (psia)	397.4	397.4

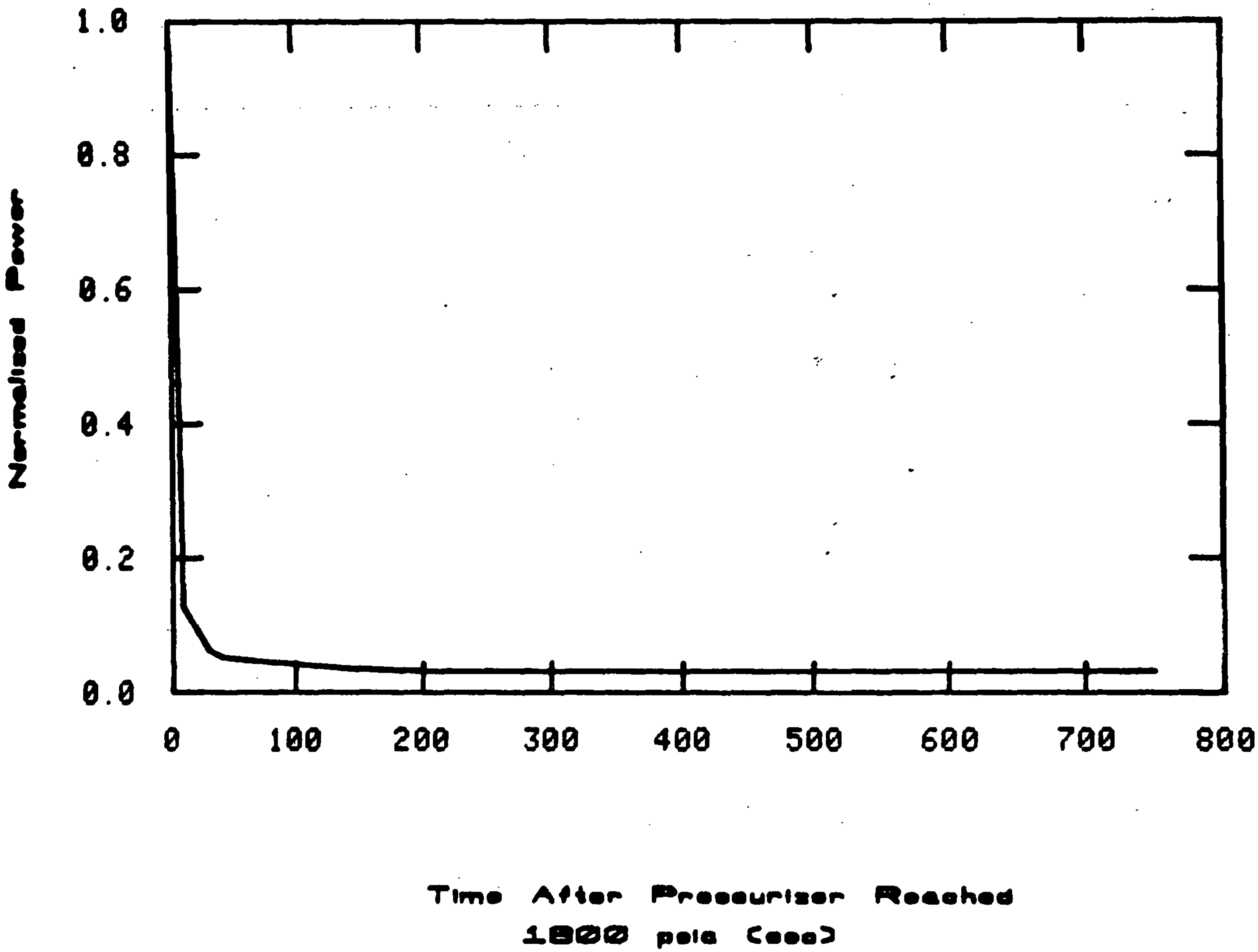

Figure 4 Core power history for Semiscale Test S-07-10B.

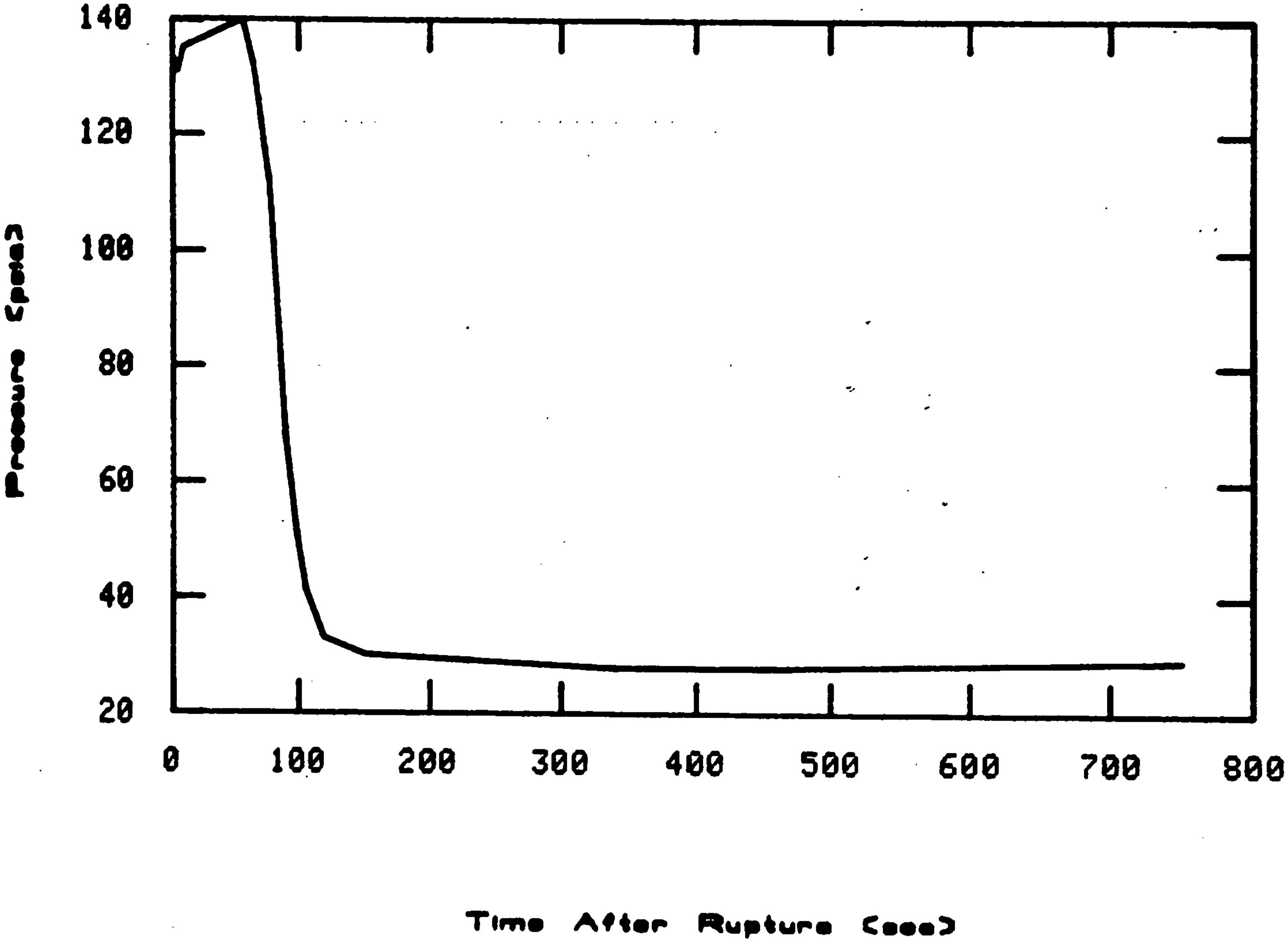

Figure 5 Pressure suppression tank pressure history for Semiscale Test S-07-10B.

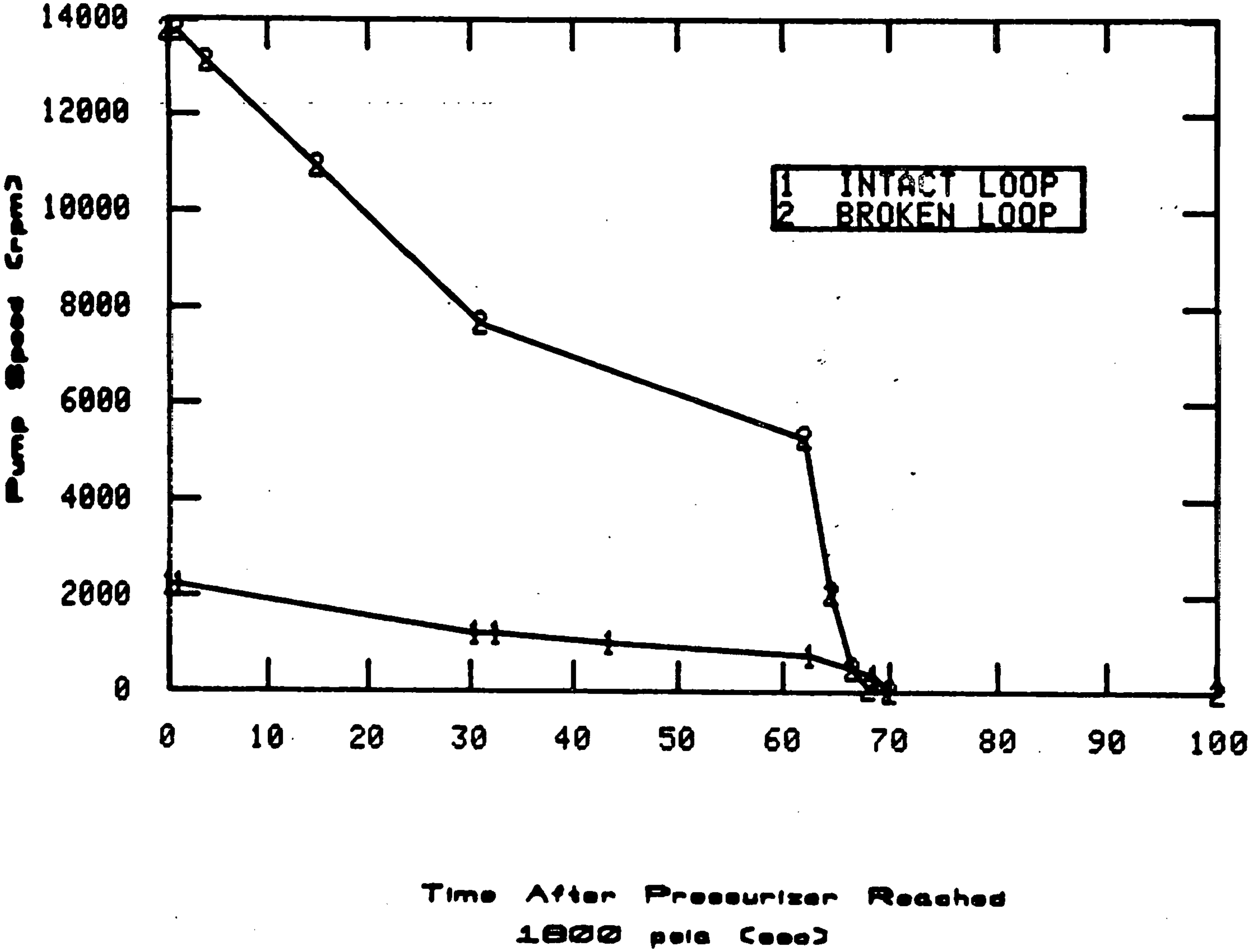

Figure 6 Primary pump speed histories for Semiscale Test S-07-10B.

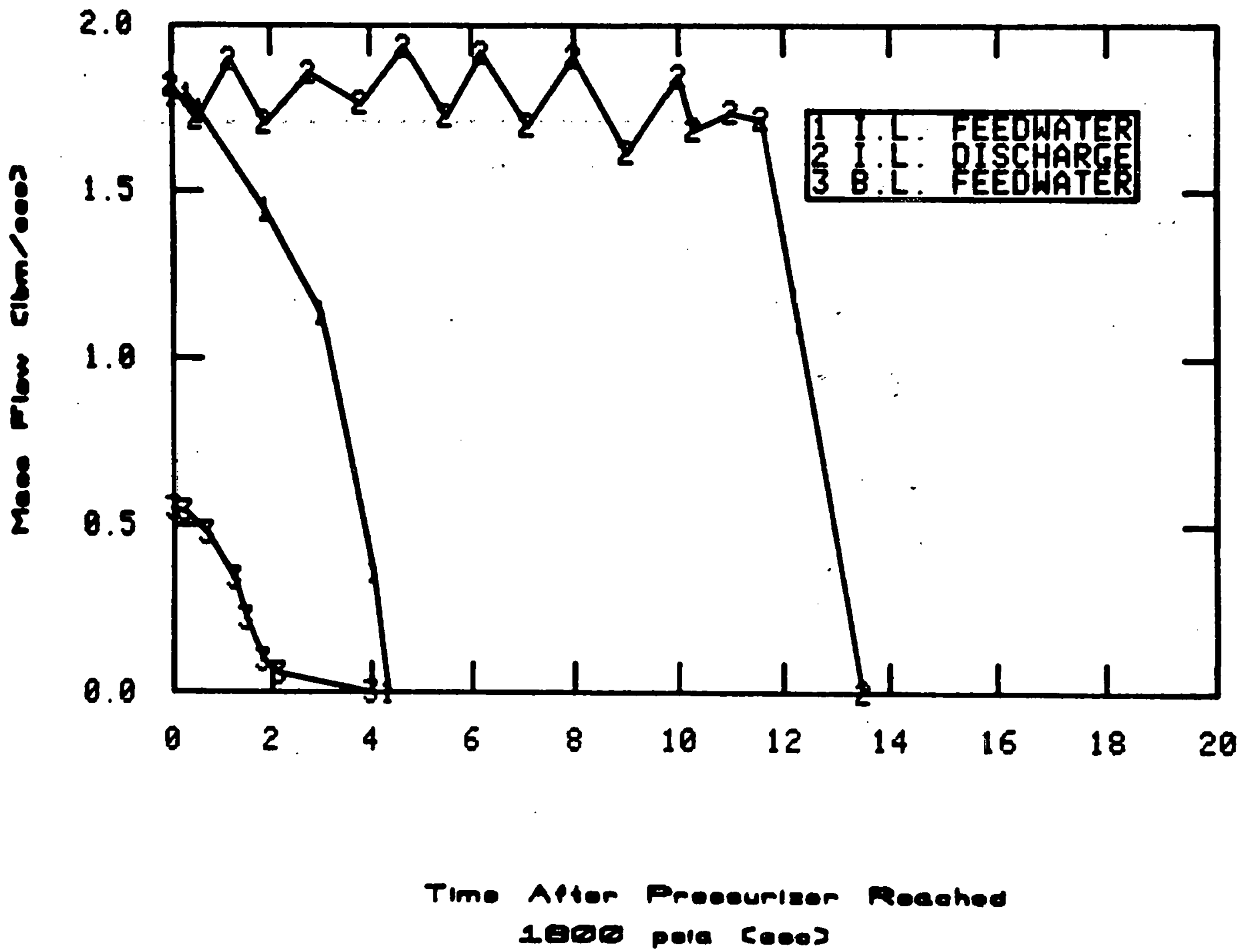

Figure 7 Steam generator secondary side flow histories.

IV. RESULTS

1. CALCULATIONAL SUMMARY

The RELAP4/MOD6 prediction of the SBE was run 450 sec transient time and required 4 hours of CDC 7600 time. The RELAP4/MOD7 calculation was terminated at 300 sec transient time and used 5.7 hours of CDC 7600 time. The MOD7 calculation execution time per time step was approximately the same as the MOD6 calculation with the MOD7 analysis requiring smaller time steps. This was not expected and the INEL Reference Code Development Branch is presently studying the problem in order to decrease computer time requirements.

2. SYSTEM RESPONSE

The predicted system pressure response is shown in Figure 8 for both calculations. The system depressurizes rapidly to 1650 psia (saturation), levels out during pressurizer delivery, and then rapidly depressurizes after the pressurizer empties at 10 sec. Core power, primary coolant pumps, and steam generator secondary feedwater and steam discharge flows begin coastdown at 7.3 sec (pressurizer pressure = 1800 psi). The depressurization rate decreases at 25 sec when the primary system pressure equalizes with the intact loop steam generator secondary side pressure.

At approximately 110 sec, the break volume becomes two phase and the system pressure decreases at a faster rate. Beyond 230 sec, the MOD7 calculation predicts a slightly higher system pressure. The initiation of accumulator flow (Figure 9) and HPIS flow at 267 sec in the MOD6 calculation produces a nonequilibrium condensation related

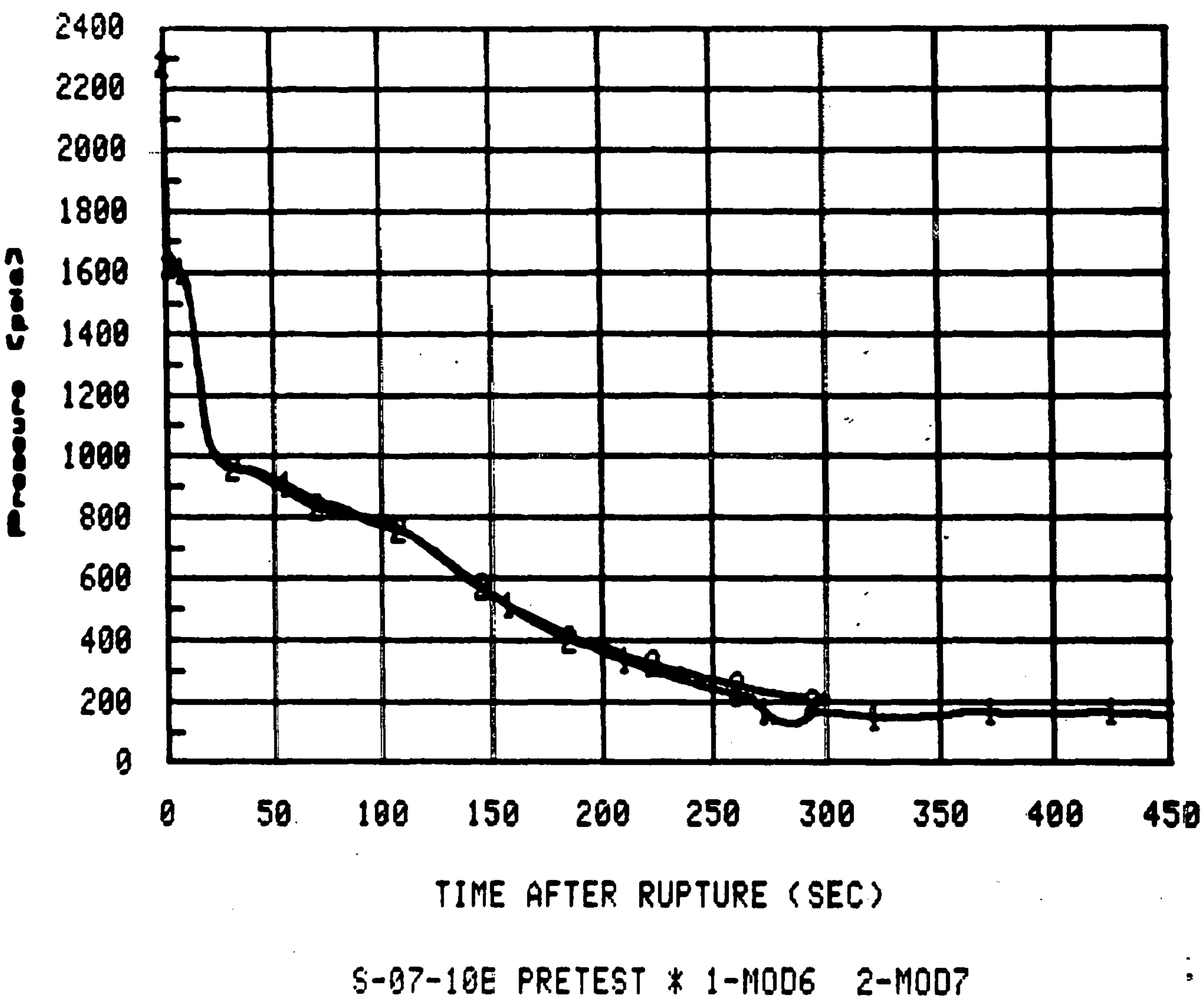

Figure 8 Upper plenum pressure.

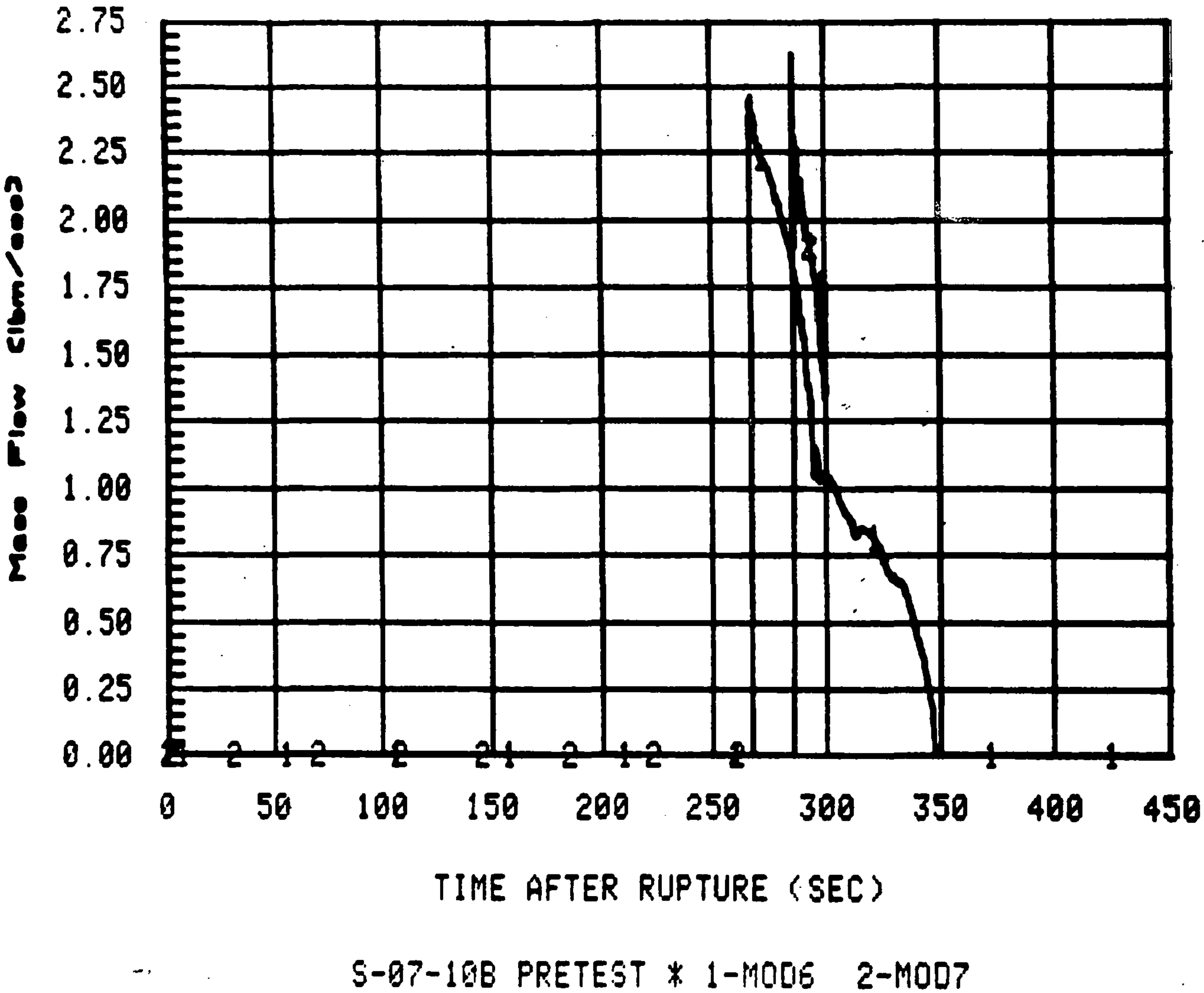

Figure 9 Intact loop accumulator flow.

pressure drop. The MOD7 calculation predicts the initiation of accumulator and HPIS flow at 287 sec but does not show the condensation related pressure drop since the nonequilibrium model is being used.

The high power heater rod surface temperatures are shown at selected elevations in Figures 10 and 11 for the MOD6 and MOD7 calculation, respectively. Both calculations show surface temperatures following the coolant saturation temperature out to approximately 110 sec. At this time, both calculations predict CHF at elevations above 96 in. However, the MOD7 calculation predicts higher slip velocities than the MOD6 calculation in the core at low void fractions. The MOD7 calculation thus predicts (in comparison to the MOD6 calculation) (1) a more rapid voiding of the core volumes, (2) a more rapid heatup of the rods, (3) an earlier CHF in the rest of the core, and (4) a slightly higher system pressure (observed previously) due to the earlier predictions of CHF. The rods are observed to rewet rapidly within a few seconds of the initiation of accumulator and HPIS flow.

The core inlet flow is shown in Figure 12 for the MOD6 calculation and Figure 13 for the MOD7 calculation. The core outlet flow for both calculations is shown in Figure 14. The predicted core inlet and outlet flow is observed to drop sharply when the primary system pressure reaches the steam generator secondary side pressure in the intact loop at 25 sec. The flows stagnate in both calculations at approximately 60 sec and oscillate slightly until 125 sec when the upper plenum mixture level reaches the hot leg elevations. The MOD6 calculation predicts stagnant flow at both inlet and outlet until initiation of accumulator and HPIS flow. The MOD7 calculation predicts the same stagnation in the outlet flow but predicts oscillatory core inlet flow. The oscillations result from the higher slip velocities in the core forcing the water down through the core resulting in water packing in the lower plenum volumes.

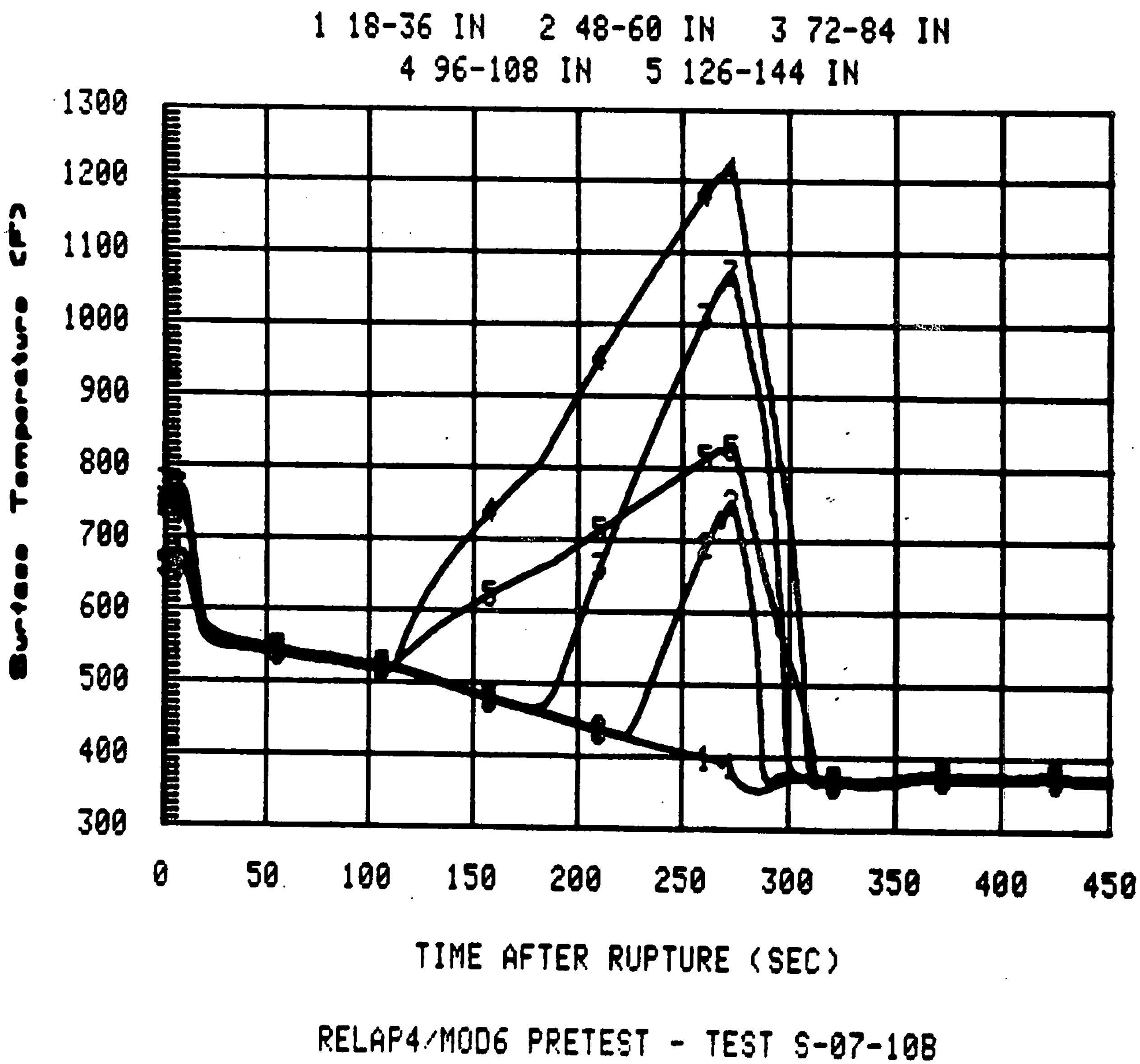

Figure 10 Heater rod surface temperatures - high power rods - RELAP4/MOD6 calculation.

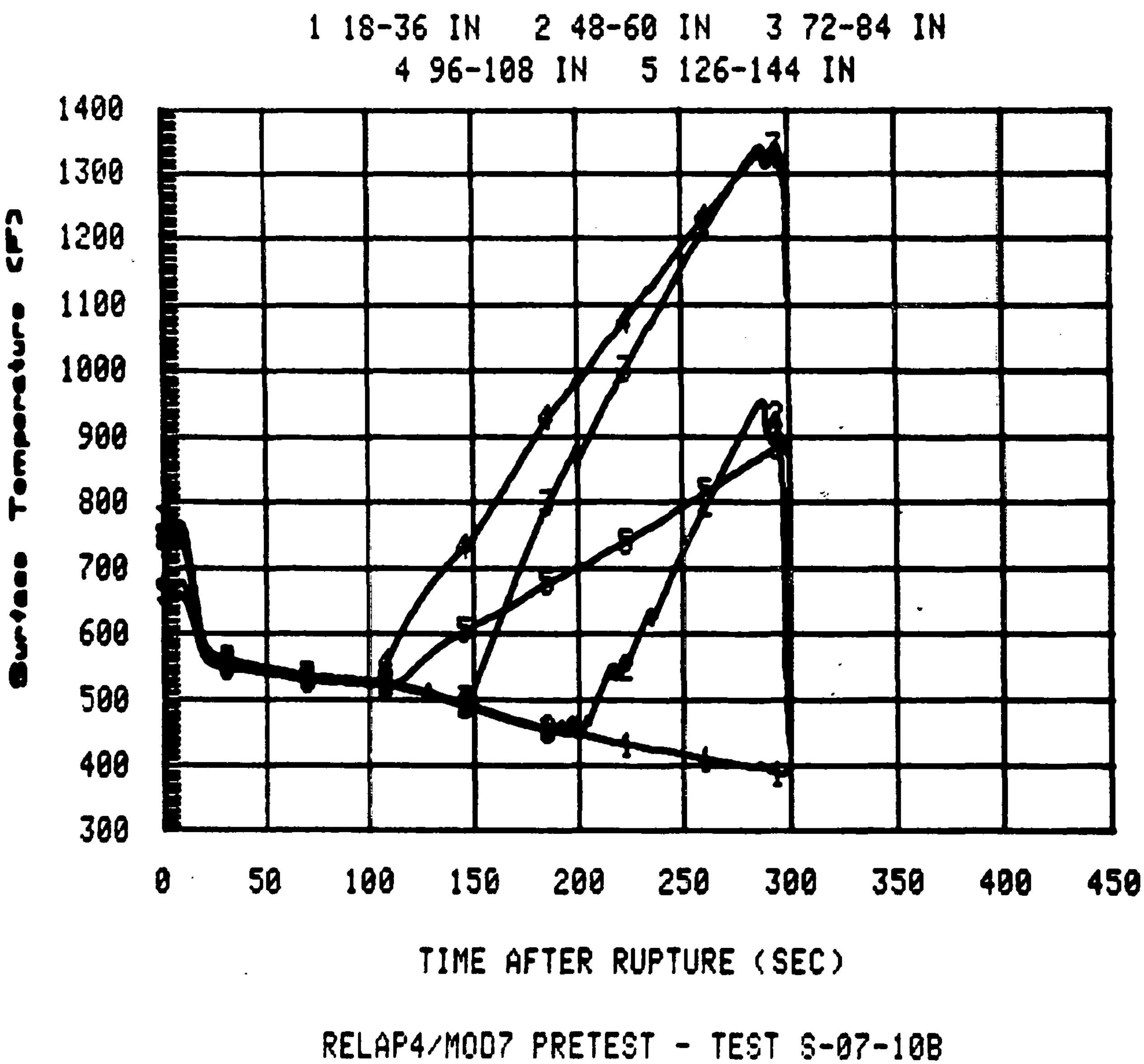

Figure 11 Heater rod surface temperatures - high power rods RELAP4/MOD7 calculation.

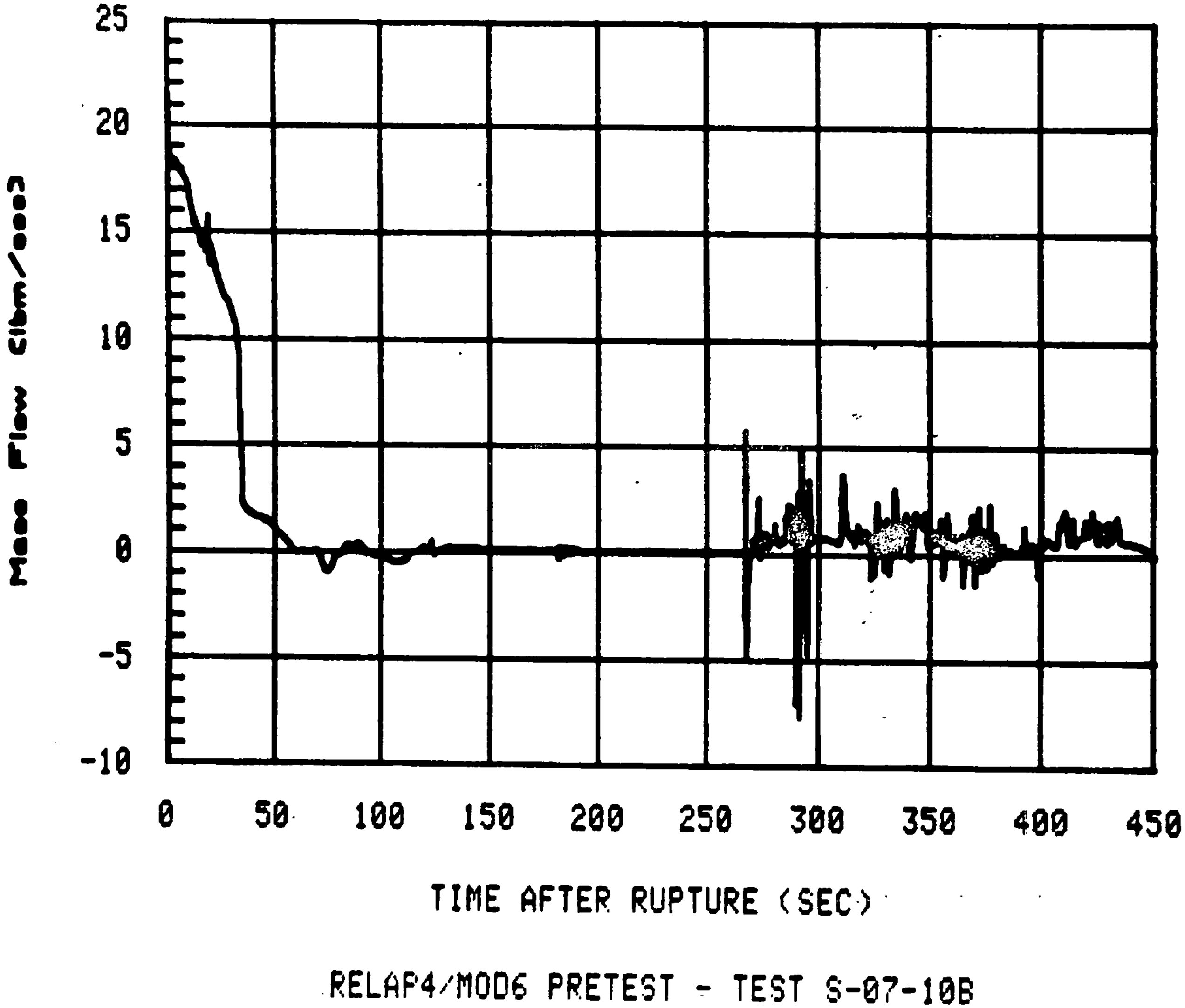

Figure 12 Core inlet flow - RELAP4/MOD6 calculation.

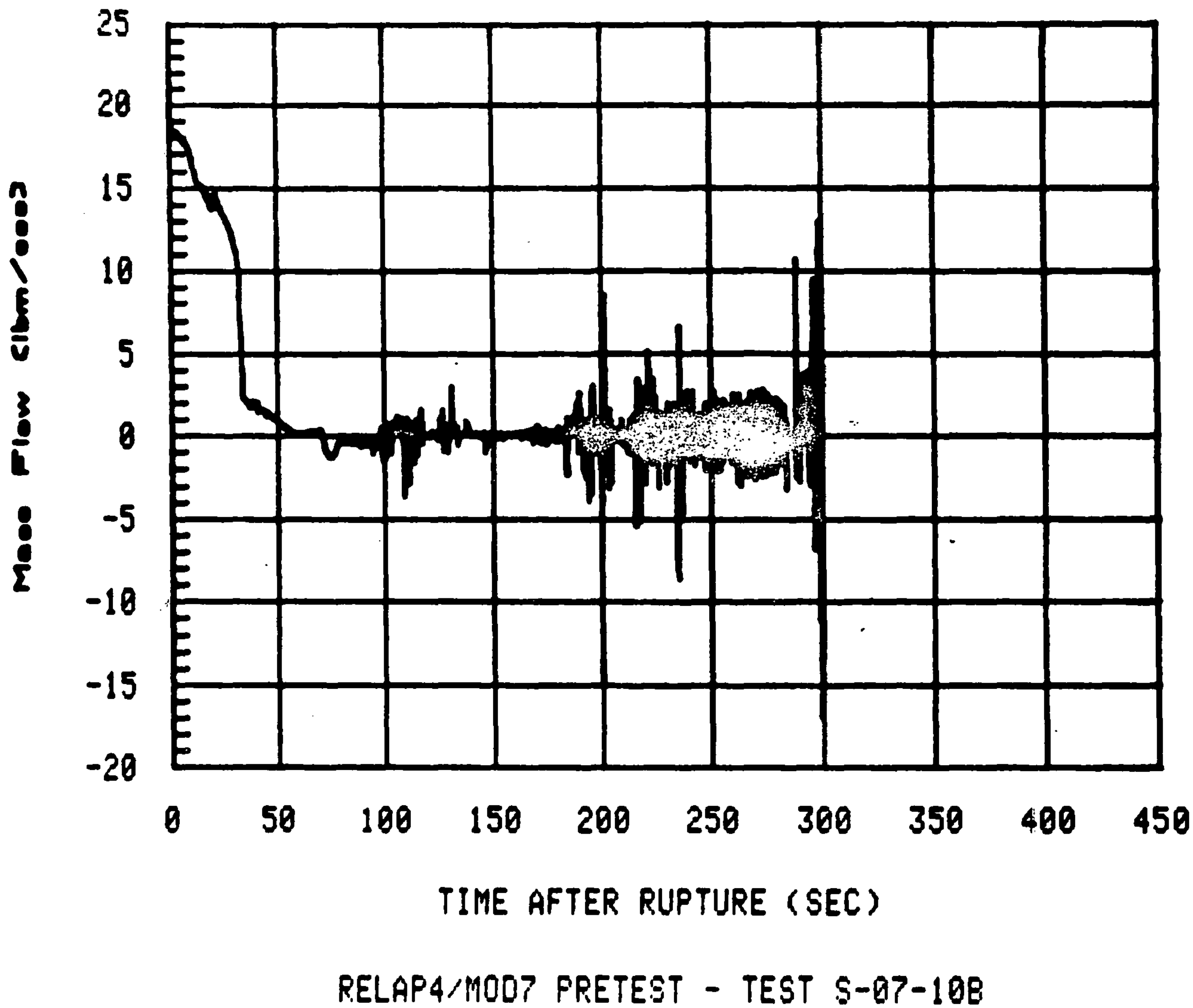

Figure 13 Core inlet flow - RELAP4/MOD7 calculation.

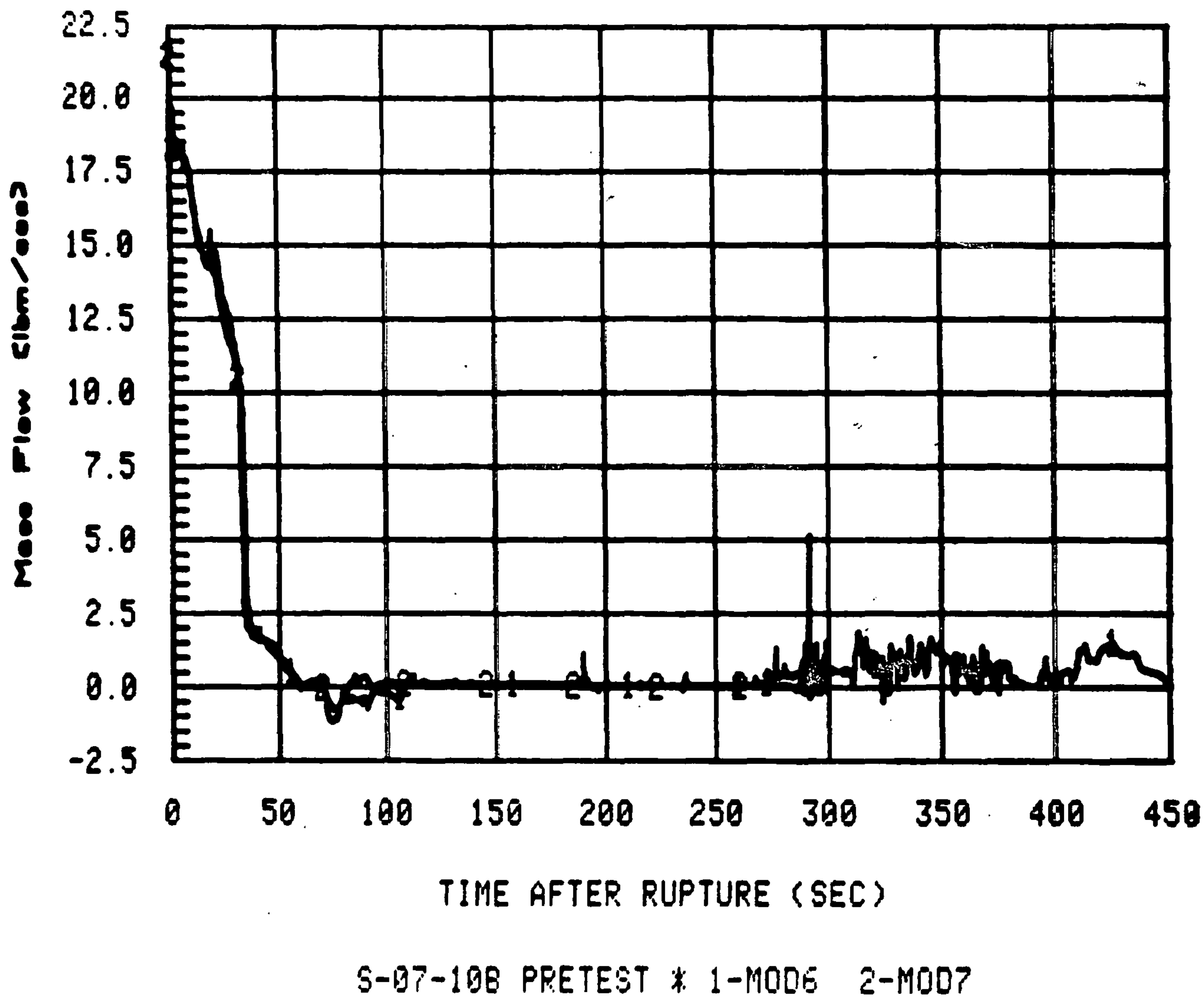

Figure 14 Core outlet flow.

The break flow is shown in Figure 15 for both calculations. The flow follows system pressure for the first 25 sec, increases until 50 sec and then drops again when the flow becomes two phase at 70 sec. The increase in break flow between 25 and 50 sec is related to the steam generator secondary behavior. At 25 sec, the system pressure is approximately equal to the intact loop steam generator secondary side pressure. The broken loop steam generator secondary pressure, however, is dropping due to the open steam discharge line valve and is below the primary system pressure. The hot leg flows then redistribute, with more flow going into the broken loop hot leg. Figure 16 illustrates this increase in flow in the broken loop and Figure 17 shows the rapid drop in intact loop hot leg flow at 25 sec. The increase in broken loop hot leg flow causes an increase in the break flow. The break flow follows system pressure until accumulator and HPIS initiation. The MOD6 calculation shows a sharp rise in break flow at 290 sec due to condensation effects in the break volume. The MOD7 calculation does not go out far enough to show this behavior.

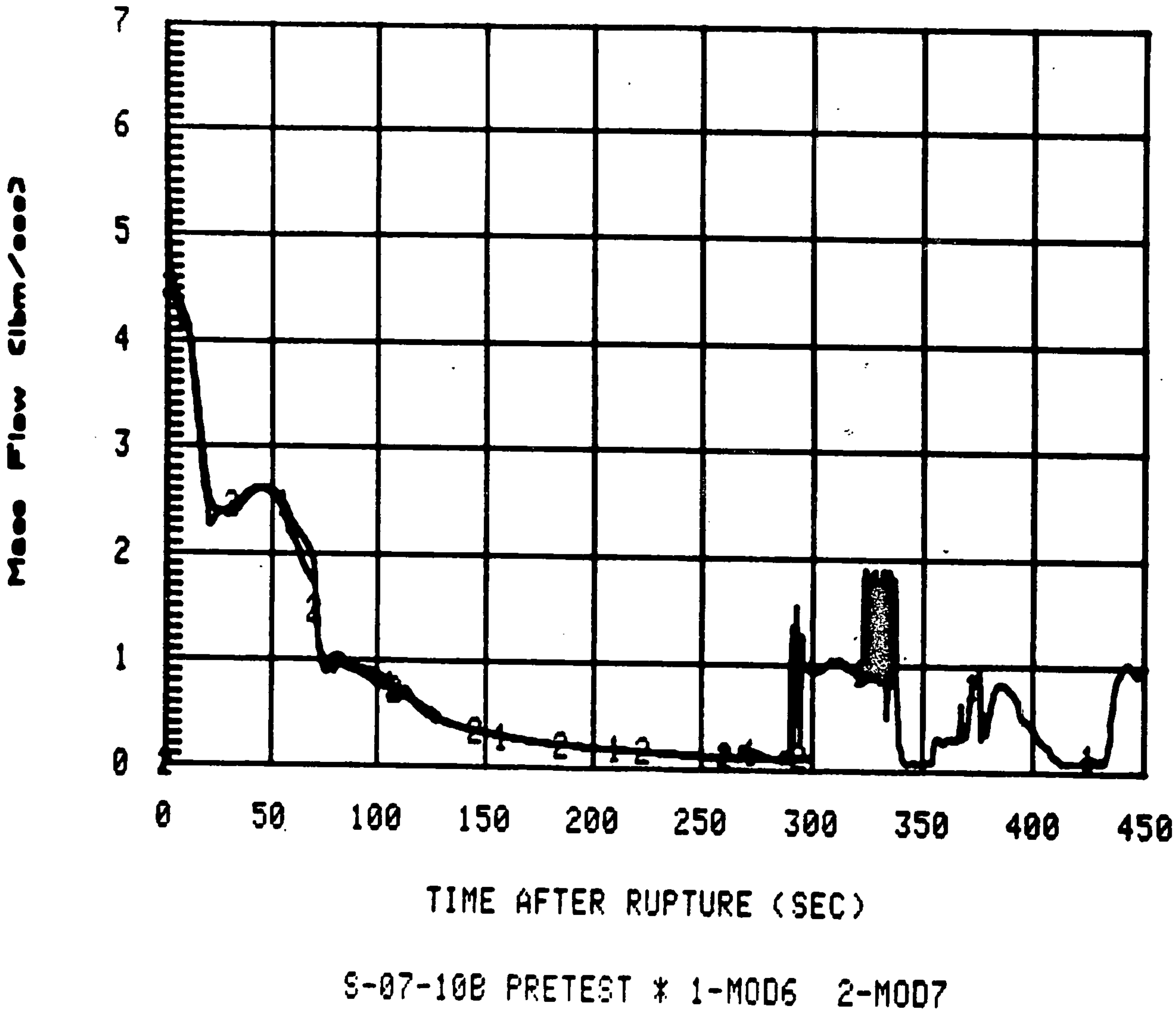

Figure 15 Break flow.

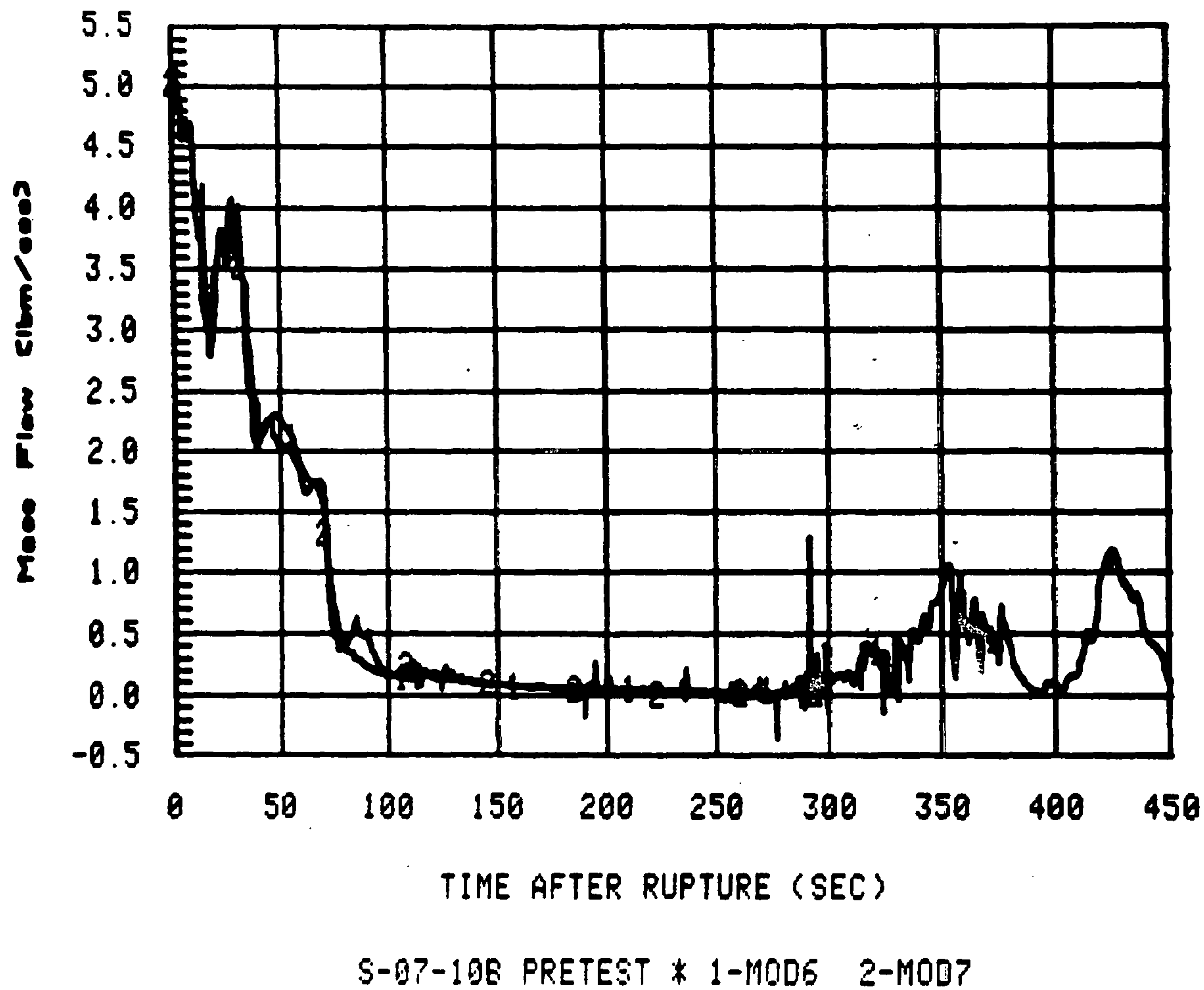

Figure 16 Broken loop hot leg flow.

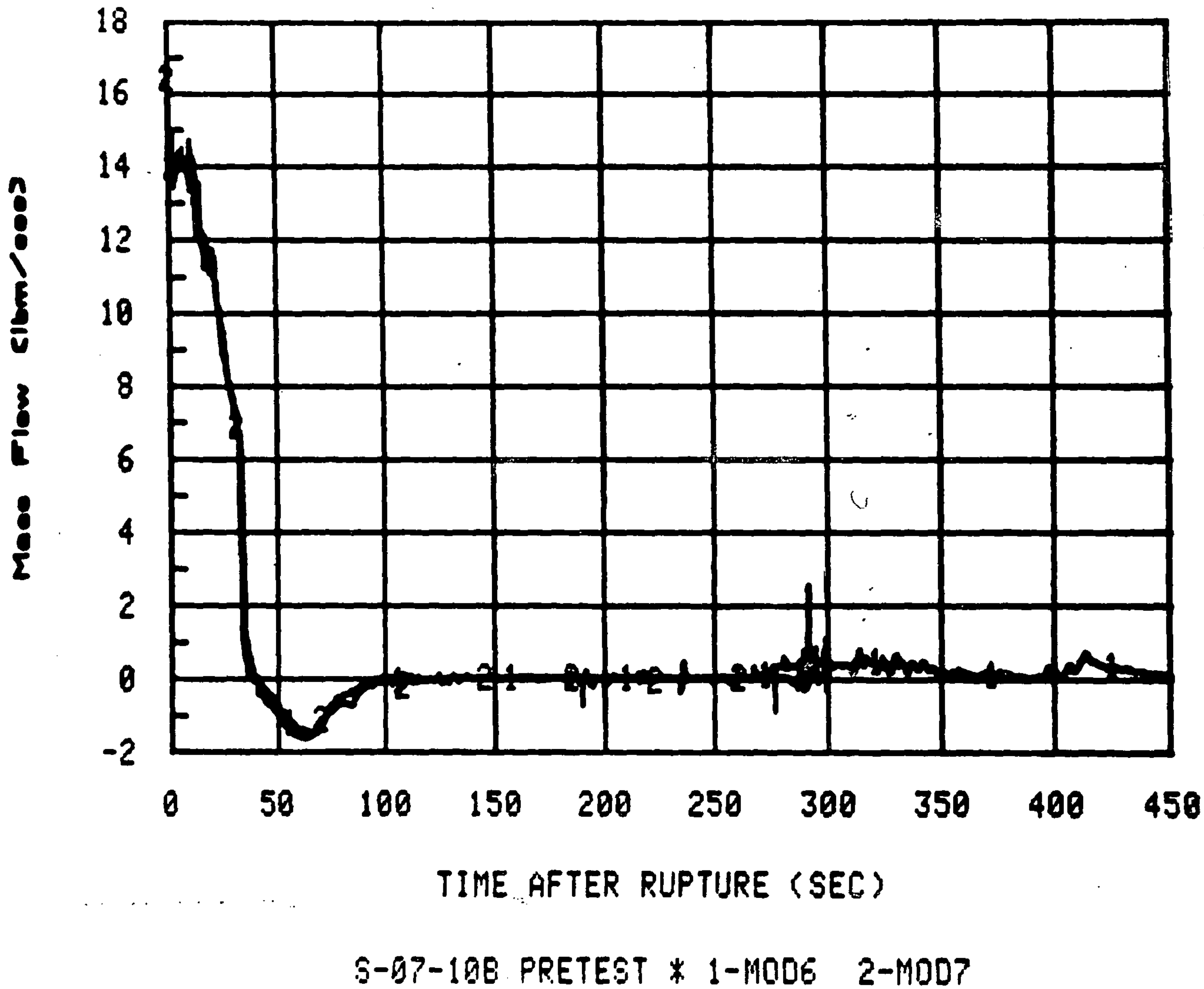

Figure 17 Intact loop hot leg flow

V. CONCLUSIONS

The analysis of Semiscale Test S-07-10B yielded expected results. Further analysis of the prediction will be performed when the data is released. Work will continue on the RELAP4/MOD7 code to identify areas where code running time can be decreased. The results of the MOD7 analysis do indicate good agreement with the MOD6 calculation.

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3. RELAP4/MOD6 - A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems - Users Manual, CDAP TR 003, January 1978.

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APPENDIX A

PRETEST PREDICTION CONSISTENCY SIGNOFF SHEET

The ECCS Applications and Analyses RELAP4/MOD6 blowdown model for the Small Break Experiment, Semiscale Test S-07-10B, has been reviewed by a subcommittee of the EG&G Pretest Prediction Consistency Committee. The subcommittee found the model to be acceptable within the recommended modeling guidelines.

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