EGG-SEMI-6395
September 1983

QUICK LOOK REPORT FOR SEMISCALE MOD-2B EXPERIMENT S-SG-1

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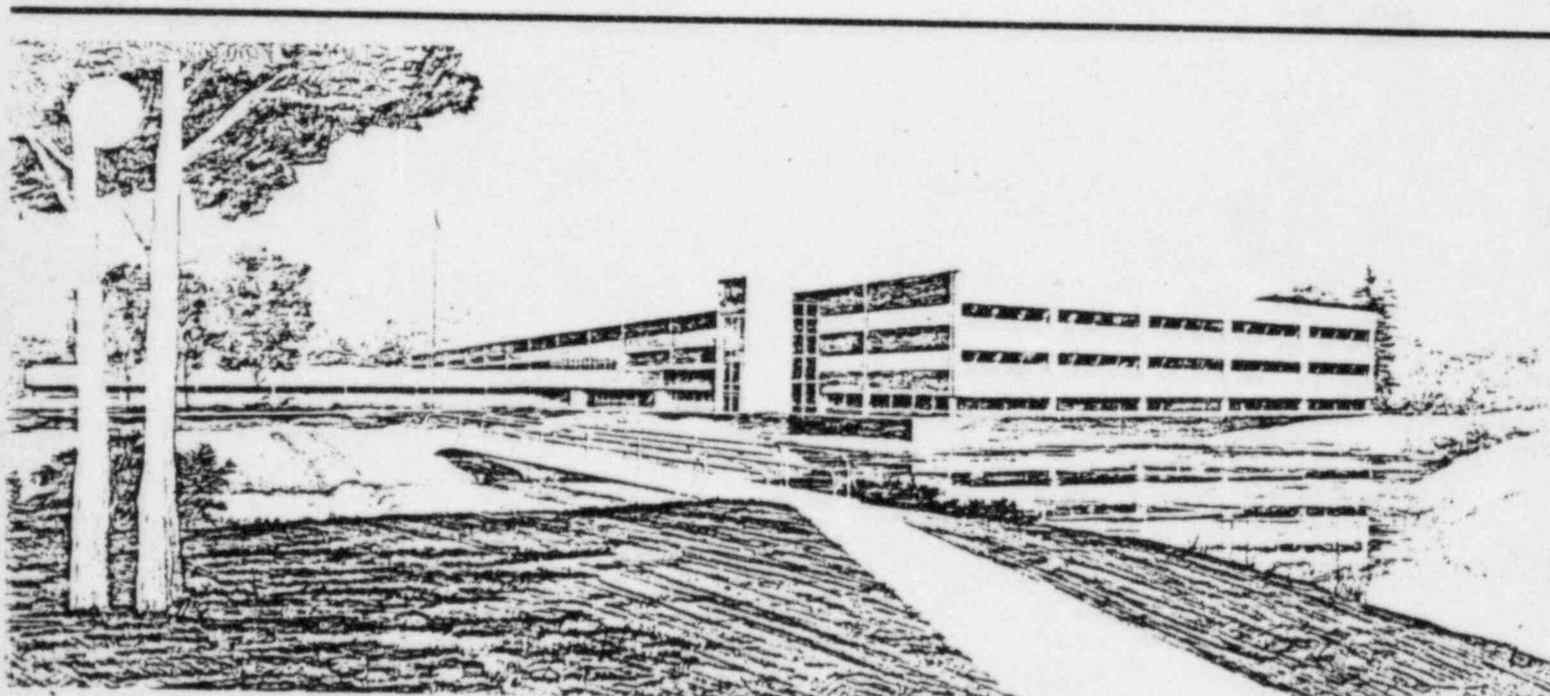

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Prepared for the
U.S. NUCLEAR REGULATORY COMMISSION
Under DOE Contract No. DE-AC07-76ID01570

EG&G Idaho

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EXPERIMENT S-SG-1

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ABSTRACT

Results of a preliminary analysis of the first test performed in the Semiscale Mod-2B Steam Generator Tube Rupture Series are presented. Test S-SG-1 simulated a pressurized water reactor accident initiated by a double-ended offset shear of one cold side steam generator tube. The transient included an initial 10-minute period during which only automatic plant protection system response to the initiating event occurred. This period was followed by an operator-induced, limited recovery procedure to establish an unaffected steam generator feed and steam condition and later, to terminate safety injection. The test results provide a measured evaluation of the effectiveness of secondary side steam and feed in Semiscale and the effect of the high pressure injection system operation on the Semiscale response to a single steam generator tube rupture.

SUMMARY

This report presents a preliminary analysis of the Semiscale Mod-2B Steam Generator Tube Rupture Series (SG) test S-SG-1. S-SG-1 is the first test of the SG Series to be conducted. The test series is designed to study not only the effect of the number of tubes ruptured (break size), but also the effect of limited operator responses to the accident following an initial 10-minute simulated identification period. Future tests will simulate 5- and 10-tube ruptures.

Test S-SG-1 simulates a pressurized water reactor transient initiated by a double-ended offset shear of one cold side steam generator tube. Data from this experiment will be examined to evaluate event signatures, event severities in Semiscale, and recovery procedures, with the principal objective of providing data to benchmark computer code calculations.

Test S-SG-1 was designed in two parts: (a) an initial 600 s period in which only automatically functioning plant protection systems were assumed to operate, followed by (b) an operator controlled recovery period including an unaffected loop steam generator feed and bleed and termination of safety injection.

The signature of a single tube rupture is characterized by a relatively rapid decrease of the primary coolant system pressure to a saturation condition in the hot legs as primary coolant system fluid flows through the broken tube into the affected loop steam generator secondary. Automatic protective actions that influence the pressure response during this early period are core scram and main steam isolation valve (MSIV) closure. Both are initiated by a low pressurizer pressure trip at 13.1 MPa (1900 psia). Main coolant pump trip, feedwater termination, auxiliary feedwater start, and safety injection start are all initiated on a safety injection signal at a pressurizer pressure of 12.5 MPa (1814 psia). Part of the pressure response during this early period is lifting of both the affected and unaffected loop steam generator relief valves (atmospheric dump valves) as primary-to-secondary heat transfer raises the pressure of

the secondaries after MSIV closure. Following the attainment of a saturation condition in the hot legs the primary and secondary system pressures remain fairly constant as safety injection (SI) fluid enters the primary, and break flow leaves the primary system to the affected loop steam generator secondary. Decay heat is removed by natural circulation.

The recovery procedure in S-SG-1 was not initiated until 600 s, so as to simulate a period necessary for operators to identify the tube rupture. Operator response at 600 s included latching open the unaffected loop atmospheric dump valve (ADV) in an attempt to depressurize the unaffected loop secondary and increase the loop heat sink. At 3000 s, SI flow was terminated to study the further effect of SI operation.

Increasing the heat sink by ADV operation in the unaffected loop did not cause a decrease in primary pressure. Core decay heat was removed by single phase natural circulation, but the overall loop natural circulation was not greatly affected by the increased heat sink. However, termination of safety injection had a large effect on the primary pressure. The SI had been pumping against a nearly full system at a higher rate than primary to affected loop secondary break flow. Upon termination of SI there was a net outflow of mass from the system and a resultant depressurization. The primary system pressure quickly dropped to the affected loop secondary pressure, and natural circulation in the loop, supported by the unaffected loop heat sink, continued to gradually reduce both pressures together.

Comparison between pretest RELAP5 computer calculations and experimental data for test S-SG-1 shows good qualitative agreement. Differences in initial conditions account for some of the differences between code and data; however, all trends and approximate magnitudes were closely predicted. To improve calculational capabilities it is recommended that a more detailed heat loss characterization of the pressurizer be performed and concurrently better instrument the pressurizer metal structures with thermocouples. These additional data would be of greatest benefit if the recommendations were implemented early in the test series.

Two tests were performed to satisfy the objectives of S-SG-1; S-SG-1A on August 3, 1983, and S-SG-1B on August 10, 1983. Both tests had almost identical initial conditions and boundary conditions. S-SG-1A had a more representative early response than S-SG-1B, but a critical instrument (pressurizer level) failed during the test. During S-SG-1B, the pressurizer level measurement performed properly but the initial affected loop steam generator coolant level was too high and affected the secondary side pressure response during the first 100 s of the test. Together S-SG-1A and S-SG-1B met the objectives of S-SG-1, and both will be used in this report to characterize the single tube rupture event.

CONTENTS

FIGURES

TABLES

1. INTRODUCTION

This report documents preliminary results from Semiscale Mod-2B test S-SG-1, the first experiment in the Semiscale Steam Generator Tube Rupture (SG) Test Series.[1] The test series includes experiments designed to investigate both tube rupture initiated transients and transients otherwise induced but concurrent with tube rupture. Data from these experiments will be examined to evaluate event signatures, event severities in Semiscale, and recovery procedures, with the objective of providing data to assess computer code calculations. Although inherent scaling distortions and facility limitations preclude interpreting the results of the SG Test Series as precise replications of pressurized water reactor response, the experiments are designed to provide thermal-hydraulic behavior that will be representative of PWR behavior. Subsequent references in this document to simulating a full-scale PWR address the design of the experiment rather than the quantitative results.

Test S-SG-1 simulated a pressurized water reactor transient initiated by a double-ended offset shear of one cold side steam generator tube. The test was designed in two parts, an initial 600 s period in which only automatically functioning plant protection systems were assumed to operate, and a subsequent operator controlled recovery period, which included steaming and feeding of the unaffected loop steam generator secondary and termination of safety injection (high pressure injection and charging pumps). Automatically occurring events included safety injection (SI), main steam isolation valve closure, core scram, main feedwater secured and auxiliary feedwater actuated, main coolant pumps tripped, and pressurizer heaters off. Recovery operations were initiated at 600 s after the occurrence of the break. (A time of 600 s is within the range of transient identification and response times that have occurred, or are expected to occur, in actual plant transients.) For S-SG-1 the recovery involved steaming the unaffected loop secondary through the atmospheric dump valve (ADV) and feeding with auxiliary feedwater. The ADV was latched open in an attempt to bring primary system pressure down by enhancing the heat sink. After a sufficient period of time to determine the effect of the unaffected loop secondary sink on primary system depressurization had elapsed, SI was

terminated. When the system pressure stabilized below the affected loop steam generator secondary ADV setpoint pressure, the test was terminated.

A preliminary analysis of test S-SG-1 is presented in the following sections. Section 2 describes the system configuration and test conduct. Section 3 presents results from test data analysis. Section 4 presents a comparison of test data to the RELAP5 pretest prediction, and Section 5 summarizes conclusions drawn from the preliminary analysis.

2. SYSTEM CONFIGURATION AND TEST CONDUCT

2.1 System Configuration

The Semiscale Mod-2B system configuration is illustrated in Figure 1. The system is scaled from a reference four-loop PWR system based on the core power ratio, 2(MW)/3411(MW).[2,3] Component elevations, dynamic pressure heads, and liquid distribution were maintained as similar as practical. The two-loop test configuration consisted of the vessel with a 25-rod electrically heated core with external downcomer, tube-and-shell steam generators and associated loop piping with circulation pumps. The affected loop (the loop in which the steam generator tube rupture occurs) is scaled to represent one loop of a four-loop PWR and the unaffected loop represents three loops of a four-loop PWR. The Semiscale Steam Generator Tube Rupture Experiment Operating Specification[1] gives more detail about the specific components.

Special modifications to the Semiscale Mod-2B system were incorporated to properly control boundary conditions for the steam generator tube rupture series. These included condensing systems and catch tanks to accurately measure system effluent from the steam generator secondaries, special effluent flow controls in the steam generator secondaries to give properly scaled steam relief flow rates, and the inclusion of a tube-rupture break assembly to simulate the primary to secondary flow path created by the tube rupture.

In both the unaffected and affected loop, a simulated power operated atmospheric dump valve (ADV) and a staged safety relief valve (SRV) system are situated on the main steam line. They represent scaled ADV and SRV flow capacities and operation.[3]

Figure 2 shows the orientation used in Semiscale to simulate ADV and SRV operation in both the affected and unaffected loop. The ADV orifice side can be used in the manual latched open mode or in the automatic pressure actuated mode. The SRV side is operated in the automatic pressure actuation mode only. The SRV orifice is designed to pass a scaled flow

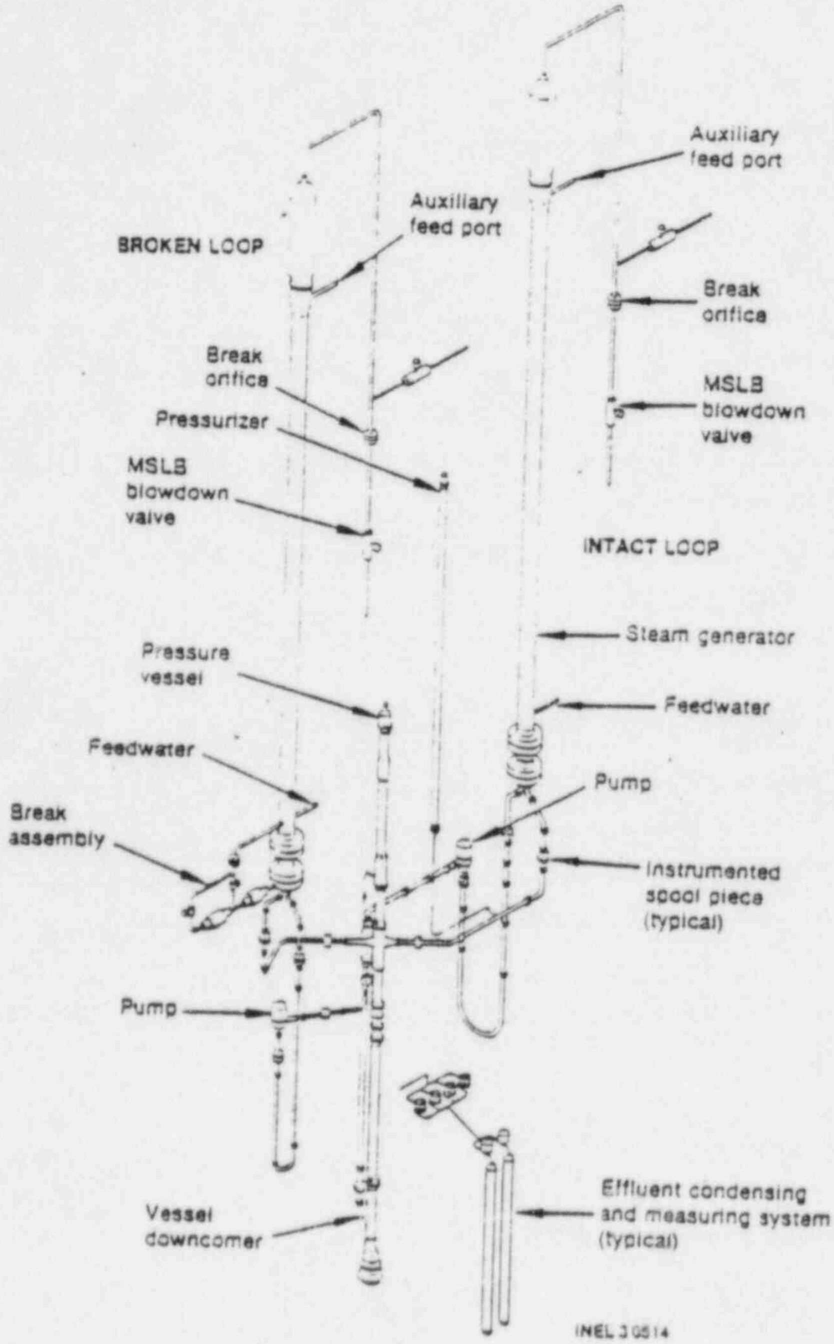

Figure 1. Semiscale Mod-2B system as configured for the SG test series.

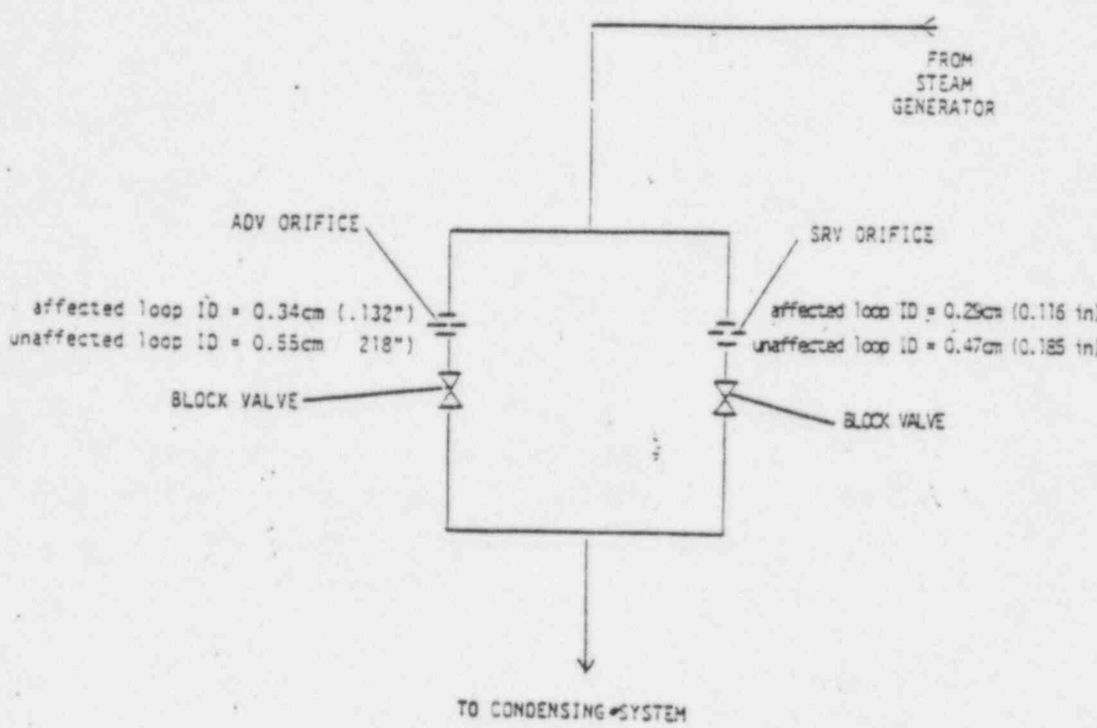

Figure 2. ADV and safety relief valve system.

corresponding to only the first stage of relief of the SRV in a PWR (PWR SRV's typically have 5 stages of relief). The ADV orifice is designed to pass scaled flow corresponding to ADV operation in a PWR. On a PWR, the pressure relief setpoint for the ADV stage is encountered first and the various multistaged SRV relief setpoints are encountered at higher pressures. In Semiscale the relief setpoint for the ADV side is 5.85 MPa (848 psia) in the affected loop and 6.55 MPa (949 psia) in the unaffected loop. The first stage SRV relief setpoint is 5.94 MPa (861 psia) in the affected loop and 6.74 MPa (977 psia) for the unaffected loop.[a] Figures 3 and 4 show flow rate versus pressure for the ADV stage and ADV stage plus first stage SRV operation for the affected and unaffected loops, respectively. The ADV stage can also be operated in the latched open manual mode during the recovery procedure with the SRV side block valve shut.

The tube rupture break assembly connects the primary coolant system with the secondary side in the vicinity of the affected loop steam generator tube sheet (see Figure 5). The break assembly can be connected to either the hot leg or cold leg side of the primary at the broken loop steam generator plenum, 57.1 cm (22.5 in) below the top of the tube sheet. The break assembly connects to the secondary at one location, 36.5 cm (14.4 in) above the top of the tube sheet on the cold leg side of the generator. For test S-SG-1 the break assembly was on the cold leg side of the primary. The break assembly consists of a break orifice and venturi flow meters to measure single phase break mass flow rate. The break orifice is a symetrical conical flow tube as depicted in Figure 6. The break orifice is interchangeable. Figure 6 shows the dimensions for a 1-, 5-, and 10-tube break orifice. Test S-SG-1 used the 1-tube break orifice with a 0.079 cm (0.0308 in) I.D. The flow tube was calibrated in single phase water and can be used to monitor a break mass flow rate.

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a. The ADV and SRV relief setpoints were set to different values for the two steam generators, and artifically low, to ensure ADV operation during the transient. The scaling of these relief setpoints is discussed in detail in Reference 1.

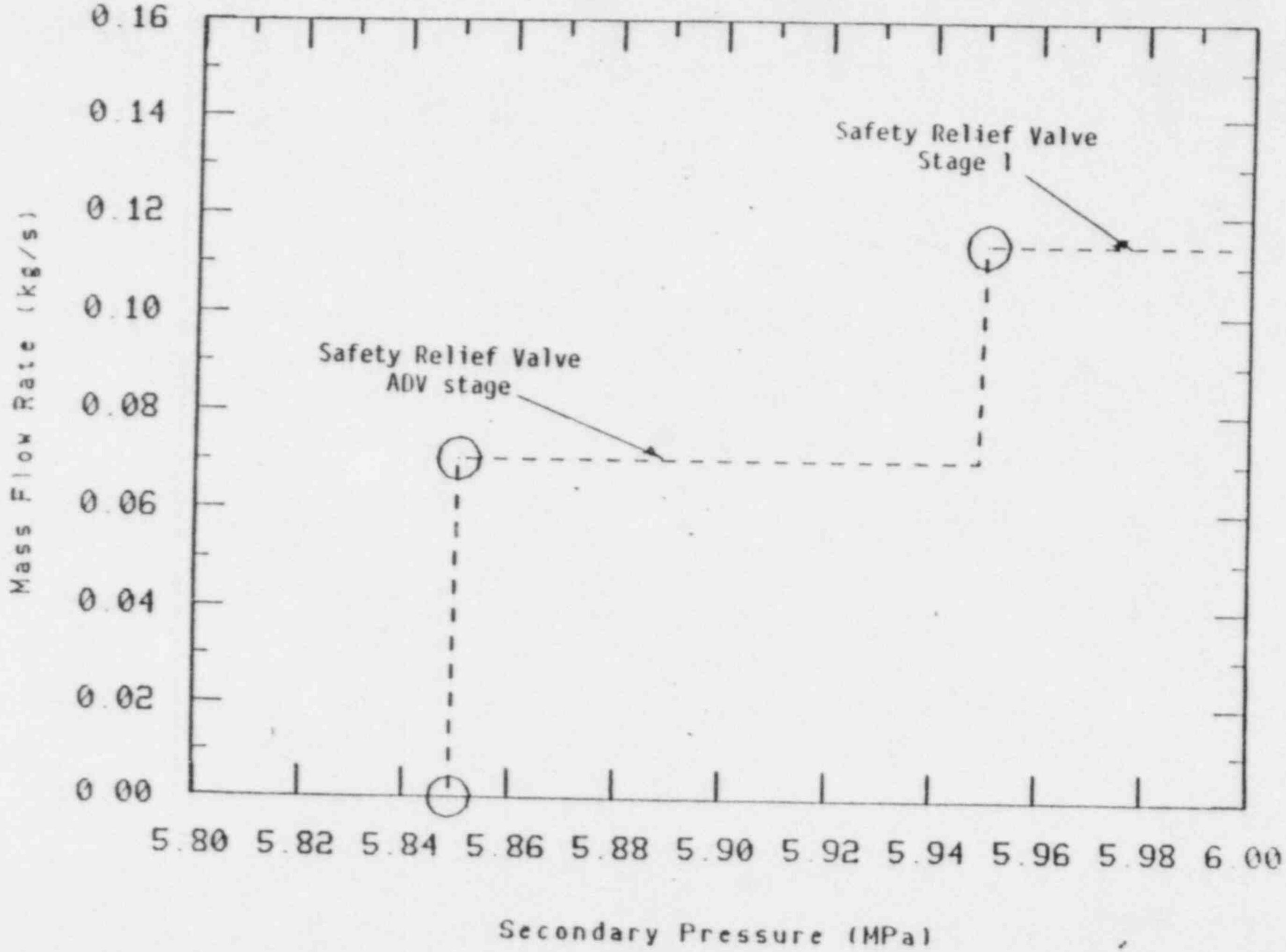


Figure 3. Broken loop steam generator safety relief valve operation.

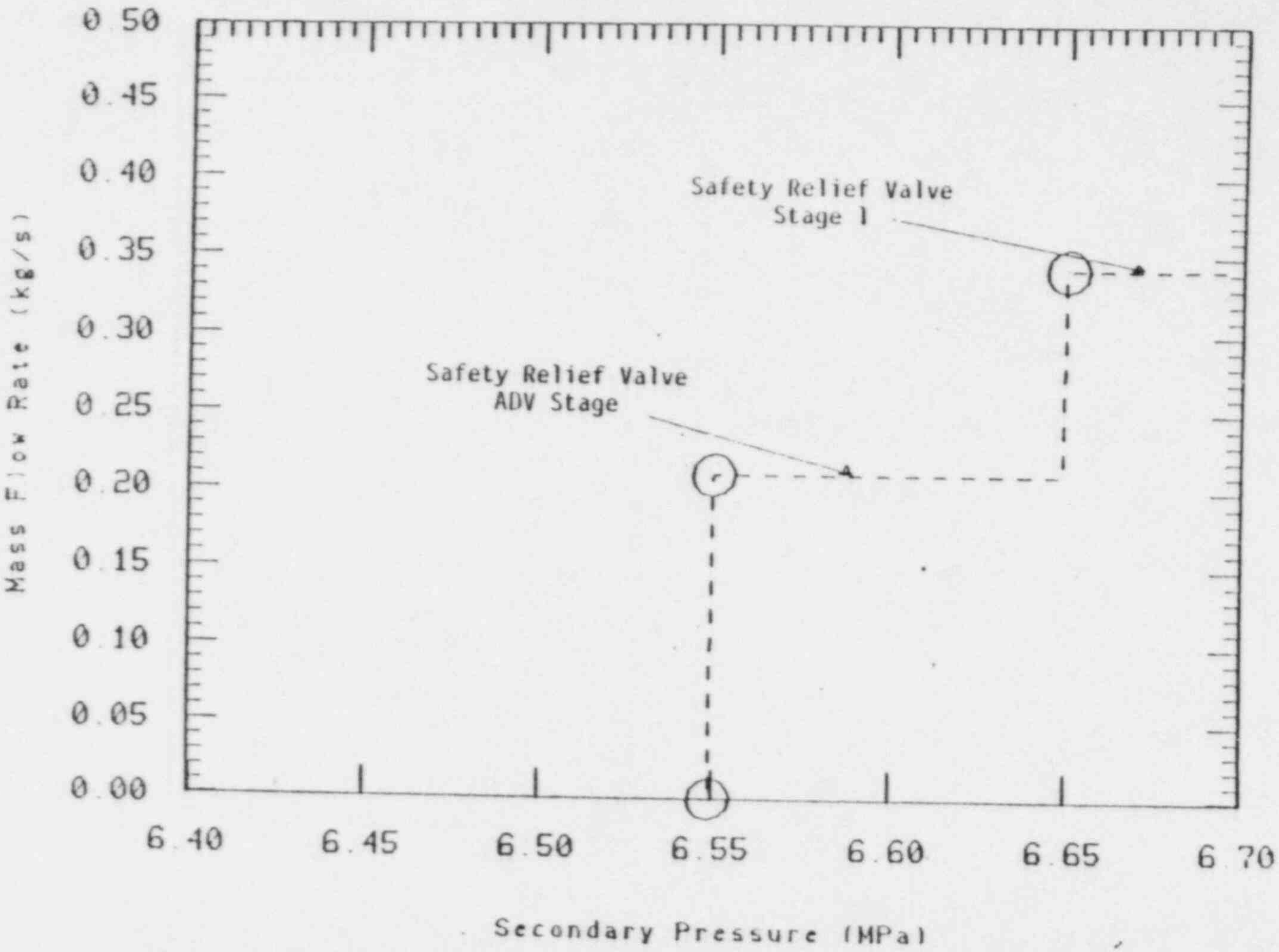


Figure 4. Intact loop steam generator safety relief valve operation.

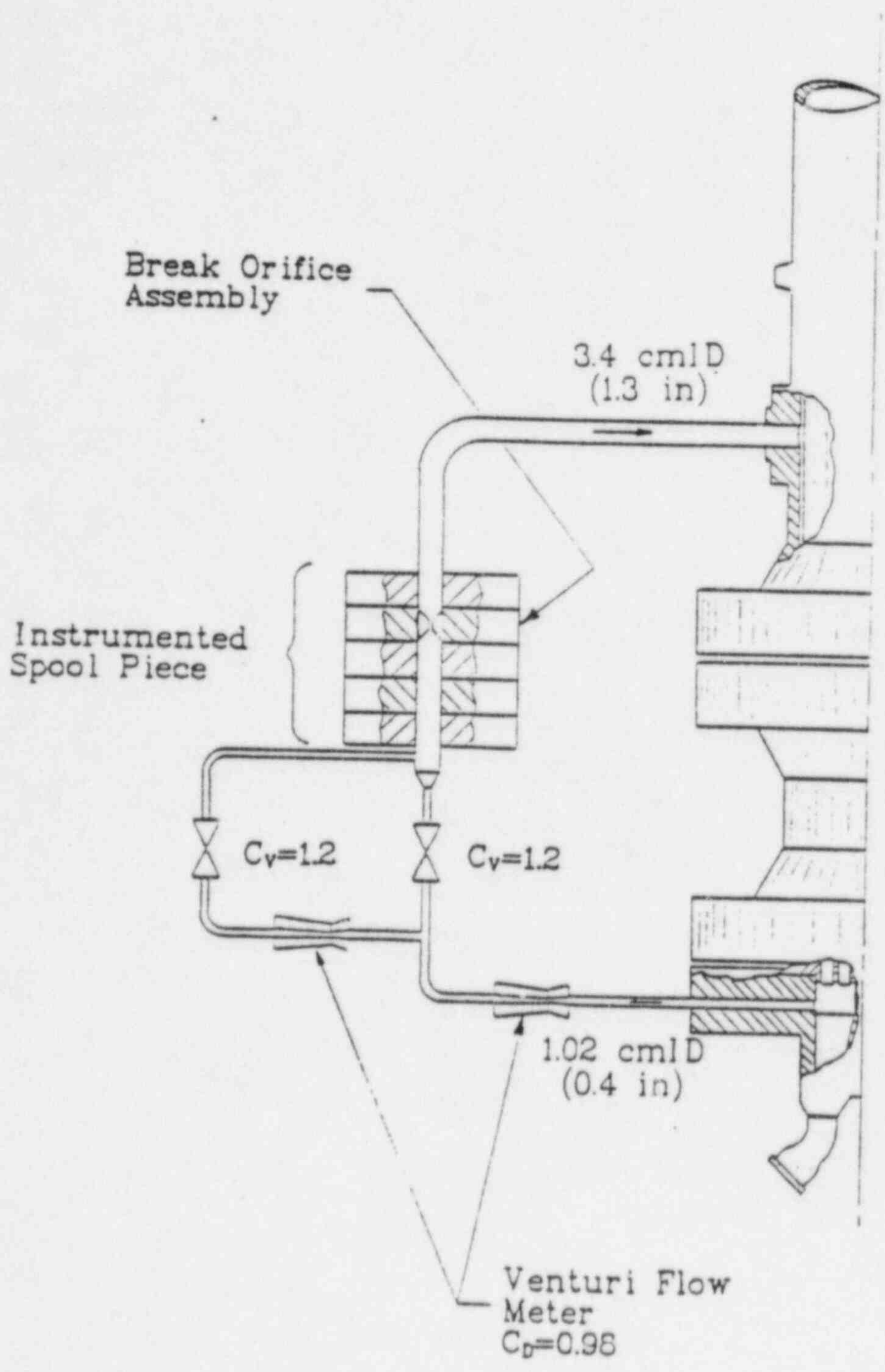
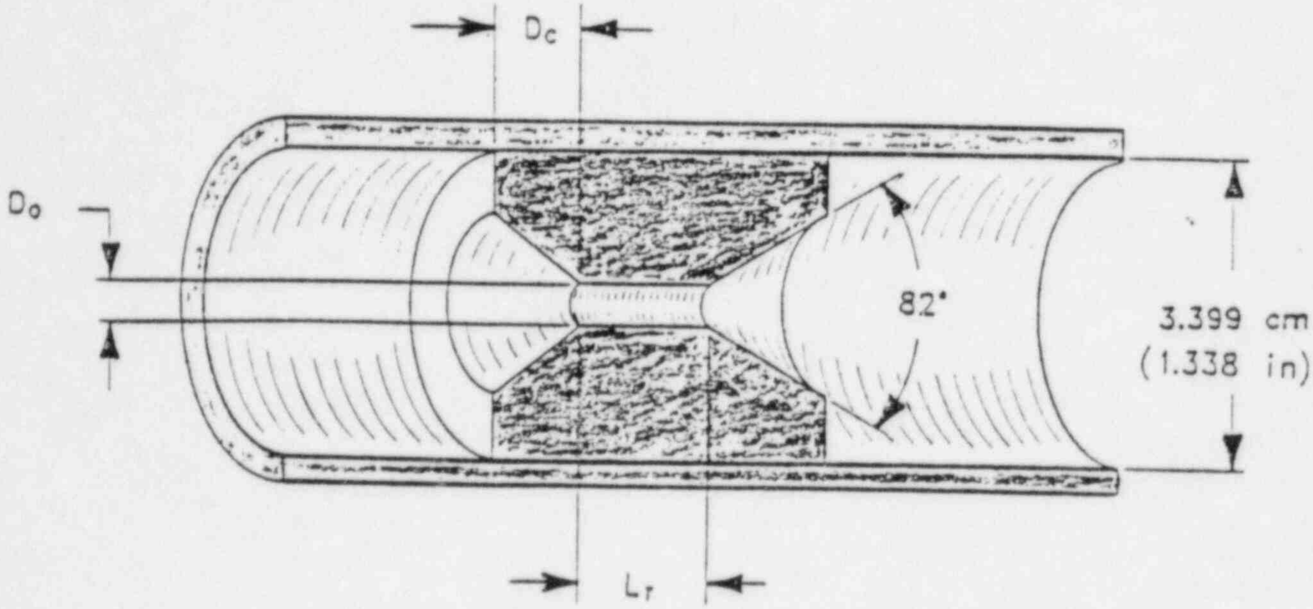


Figure 5. Semiscale tube-rupture break assembly.



TUBE RUPTURE	D_o		L_T		D_c	
	cm	in	cm	in	cm	in
1 TUBE	.079	.0308	.198	.078	1.473	.662
5 TUBE	.175	.0689	.439	.173	1.372	.709
10 TUBE	.249	.0975	.622	.245	1.270	.745

Figure 6. Semiscale conical break orifice.

Heat loss makeup in the Semiscale system is accomplished by using external heaters distributed fairly uniformly throughout the Semiscale system. These heaters are controlled by six separate power supplies including: vessel, hot legs, cold legs, unaffected loop pump suction, affected loop pump suction and pressurizer. The total power provided by these heaters is about 47 kW. An additional 20 kW of heat loss makeup was provided by augmenting core power throughout the transient. Control of the heaters is as follows: If the maximum allowable temperature level (900 K) is reached on the outside surface of the pipe insulation, external power to that component is reduced by half. If the temperature trip limit continues to be exceeded, power to that component is terminated.

2.2 Test Conduct

The system was filled with demineralized water and vented to ensure a liquid full system. Instrumentation was calibrated and zeroed as necessary. The system was heated to initial conditions using core power and forced flow with the primary coolant pumps running. Specified and measured initial conditions are listed in Table 1.

The test was initiated at t = 0 by opening a block valve in the break assembly allowing primary fluid to flow into the affected loop secondary. Table 2 contains a sequence of significant events for S-SG-1. The first 600 s involved automatically occurring events such as core scram, main steam isolation valve closure, auxiliary feedwater start and main feedwater stop, main coolant pump trip and HPIS/charging flow initiation. The initiating events for these actions were from low pressurizer pressure trip (13.1 MPa (1900 psia)) and SI signal (12.51 MPa (1814 psia)). The recovery procedure for S-SG-1 involved initiation of intact loop steam and feed and SI termination. SI includes both high pressure injection flow and charging pump flow as described in Reference 1. The recovery procedure started at 600 s, the simulated time required for operator identification of the tube rupture. Unaffected loop auxiliary feed was controlled by attempting to maintain the secondary water level between 800 and 1050 cm (315 and 413 in.). Affected loop auxiliary feed was terminated prior to 600 s on a 1050 cm (413 in.) level trip. The unaffected loop ADV was latched open in

TABLE 1. INITIAL CONDITIONS FOR S-SG-1B

	Specified	Measured
Primary Cold Leg Flow Rate (Nominal)		
Affected Loop	2.7 ·/s (43 gpm)	2.84 ·/s (450 gpm)
Unaffected Loop	8.1 ·/s (128 gpm)	8.44 ·/s (134 gpm)
Pressurizer Pressure	15.6 ± 0.14 (2250 ± 20 psig)	15.42 (2235 psia)
Pressurizer Liquid Volume	0.0102 ± 0.0008 m^3 (0.36 ± 0.028 ft^3)	0.0101 m^3 (0.358 ft^3)
Core Power	2.0 ± 0.01 MW	2000 kW
Loop to Loop Cold Leg Fluid Temperature Differential	2 K (3.6°F)	2 K (3.6°F)
Core Fluid Temperature Rise	37 ± 1.5 K (66.6 ± 3°F)	39 K (70.2°F)
Steam Generator Pressure		
Affected Loop	5.55 ± 0.07 MPa (793 ± 10 psig)	5.53 MPa (802 psia)
Unaffected Loop	5.55 ± 0.07 MPa (793 ± 10 psig)	5.48 MPa (795 psia)
Steam Generator Secondary Fluid Mass		
Affected Loop	100 ± 10 kg (220 ± 22 lbm)	188 kg[a] (414 lbm)
Intact Loop	100 ± 10 kg (220 ± 22 lbm)	107 kg[a] (235 lbm) (1117-51) 158 kg[a] (348 lbm) (1117-836)
Primary Leakage at t = 0	0.006 kg/s (0.0132 lbm/s)	0.000712 kg/s (0.00156 lbm/s)

a. These values were determined from data acquisition system levels following main steam isolation valve closure. Initial conditions were established using process indicated levels which have a high uncertainty in a steaming condition; however the specified process levels where achieved prior to test initiation.

TABLE 2. SEQUENCE OF SIGNIFICANT EVENTS FOR S-SG-1B

Event	Time(s) Condition
Steam generator tube rupture (block valve open)	t = 0
Pressurizer heaters off	t = 0
Low pressurizer pressure reactor trip	t = 91
Main steam isolation valves closed	t = 91
Safety injection signal	t = 96
HPIS/charging initiated	t = 96
Main feedwater secured	t = 96
Auxiliary feedwater actuated	t = 96
Reactor coolant pumps tripped	t = 96
Affected loop auxiliary feedwater off	t = 120
Latch open unaffected loop ADV	t = 600
Terminate SI	t = 3000
Primary pressure equals affected loop secondary pressure	t = 5000
Test Termination	t = 5000

an attempt to bring the primary pressure below the broken loop relief setpoint. SI was terminated at 3000 s to examine the influence of SI on system pressure. The test was terminated at 5000 s. External heater power for heat loss makeup remained on for the entire transient. Even though slight voiding occurred in the vessel upper head no external temperature limits were exceeded.

3. RESULTS

This section discusses the overall thermal-hydraulic response during test S-SG-1. The discussion is organized into two basic areas: the early response to automatically occurring events (0-600 s) and the recovery period involving operator actions (600-5000 s). Two experiments were performed to accomplish the objectives of S-SG-1, tests S-SG-1A and S-SG-1B. The discussion of early response is based on test S-SG-1A data and the recovery period response discussion is based on S-SG-1B data. Both tests had almost identical initial conditions and boundary conditions, but test S-SG-1A had a more representative signature response early in time due to a slightly lower initial fluid inventory in the affected loop steam generator. However, a critical instrument (pressurizer level) failed during the experiment. On S-SG-1B, the pressurizer level measurement performed properly but the initial affected loop steam generator level affected the early signature pressure response (t 100 s). The overall 5000 s transient response, however, was not affected. (Appendix A compares differences in boundary conditions between S-SG-1A and S-SG-1B.)

3.1 System Behavior--Tube Rupture Signature Early in Time

The occurrence of a tube rupture during normal operation in a PWR has a very distinctive signature response. This signature response is best examined by comparing primary and secondary pressures (Figure 7). The tube rupture (occurring in the affected loop steam generator) initiates the transient at time 0. Primary fluid originally at 15.6 MPa (2250 psia) flows into the affected loop steam generator originally at 5.5 MPa (797 psia). The loss of mass from the primary loop caused a fairly steady primary depressurization until about 48 s, at which time a marked increase in depressurization occurs. This increase in depressurization occurred about the time that the pressurizer liquid level had lowered to the axial location of the top of the pressurizer internal heaters. Above the heaters, there is a relatively large surface area for spontaneous nucleation, which tends to increase flashing and retard the depressurization. Below the heaters this free surface area decreases,

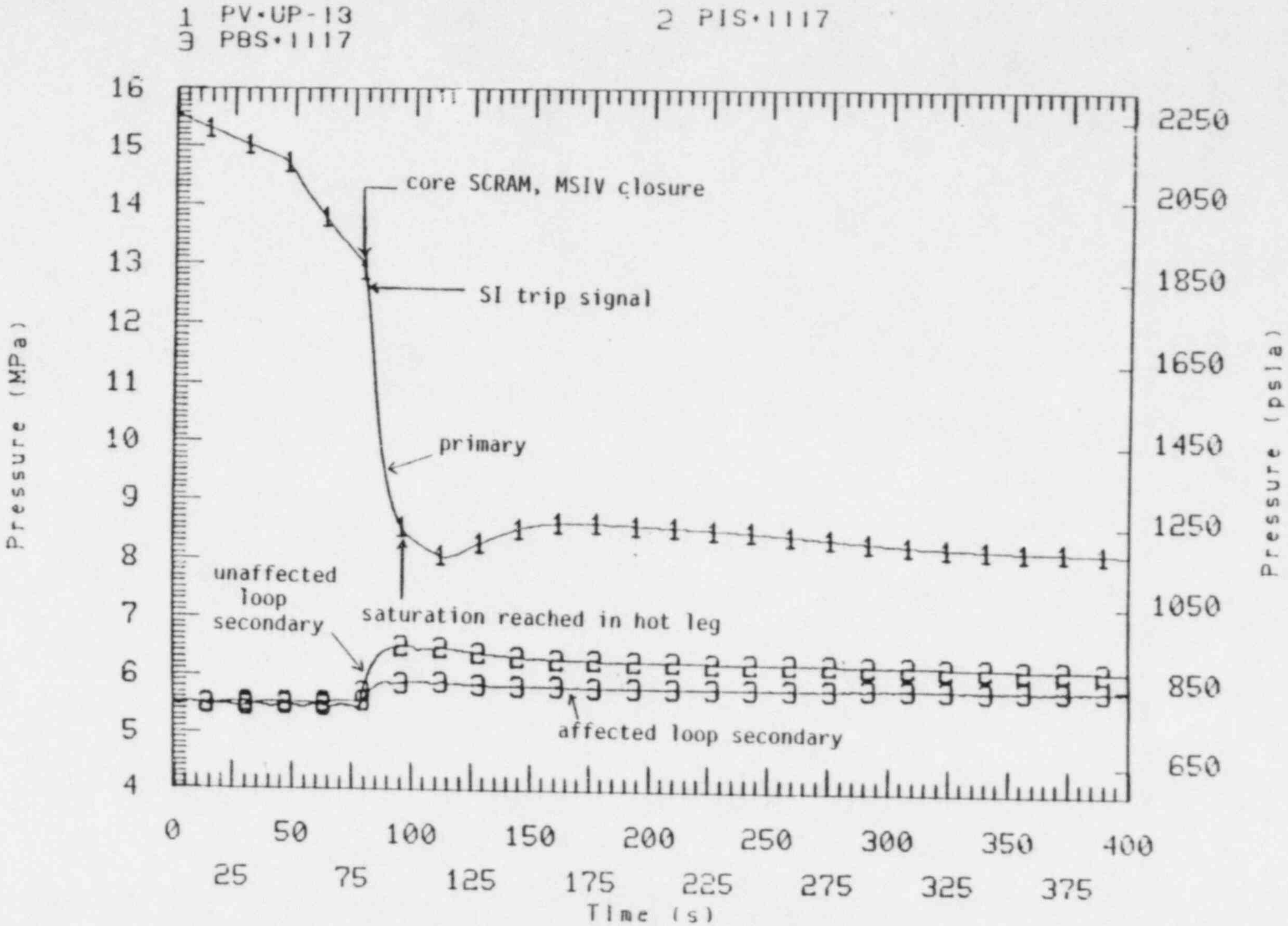


Figure 7. System pressure response signature for a 1 steam generator tube rupture cold side break.

possibly retarding flashing and increasing the depressurization rate. Further analysis of this phenomenon will be included in posttest studies. Following this interface change, the primary depressurization was fairly steady until the low pressurizer pressure setpoint of 13.1 MPa (1900 psia) was achieved (about 83 s (S-SG-1A)). Prior to achieving the low pressurizer pressure trip, both the affected and unaffected loop steam generator pressures remained fairly constant as core power was removed via normal secondary steaming conditions with the primary loop pumps running (see Figure 8). The energy addition from primary to secondary break flow was so small as to cause a negligible pressure rise during this period. However, at the low pressurizer pressure trip point, two prominent events occurred: the core power was scrammed to the ANS decay power curve and the main steam isolation valves were closed on the steam generator.

Upon MSIV closure the heat transfer to both the affected and unaffected loop steam generator secondaries caused a rapid pressurization of the secondaries. The secondary pressure in each loop rose until the ADV trip setpoints were achieved (6.55 MPa (950 psia) in the intact loop and 5.85 MPa (848 psia) in the broken loop). Figure 8 compares the affected and unaffected loop secondary pressures, showing that flow out the ADV was sufficient to prevent further pressure increases once the valve was opened.

Following the core scram at 13.1 MPa (1900 psia), the safety injection signal achieved at 12.51 MPa (Figure 7) (1814 psia) initiated: (a) terminating power to the primary coolant pumps, (b) starting SI flow, (c) terminating main feedwater and starting auxiliary feedwater to the secondaries. No major change in depressurization rate occurred from these events. Following pump trip and coastdown, the loop flow reduced to typical natural circulation values[4] as shown on Figure 9. Eventually the primary system depressurization was sufficient for the hot leg fluid to reach a saturation condition at about 100 s (Figure 10) which caused a major inflection point due to flashing in the system retarding the depressurization rate. For the remaining 600 s primary pressure remained above both secondary pressures causing a primary to affected loop secondary mass flow.

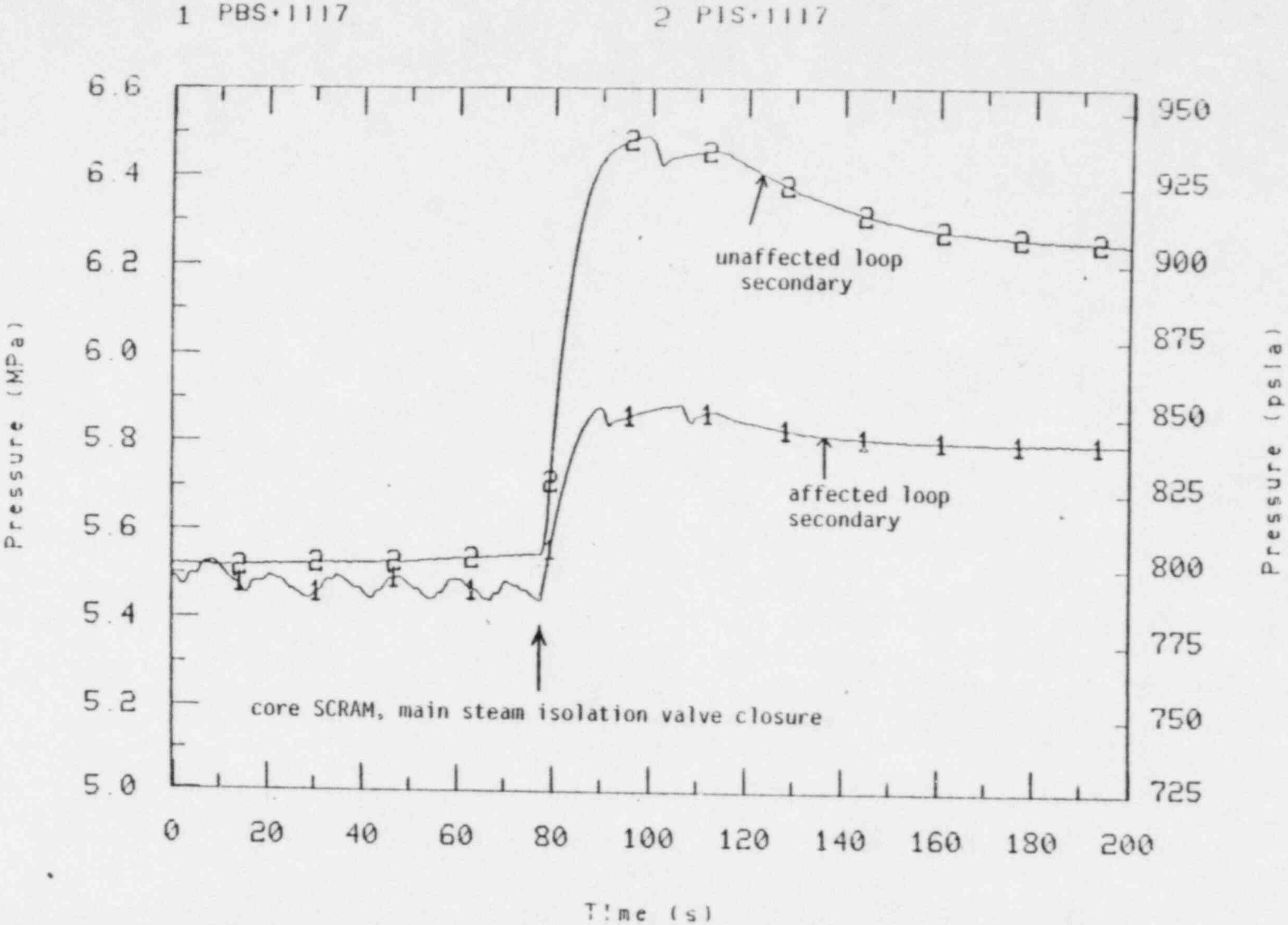


Figure 8. Effect of main steam isolation valve closure and core scram on affected and unaffected `oop secondary pressure for a 1 tube rupture.

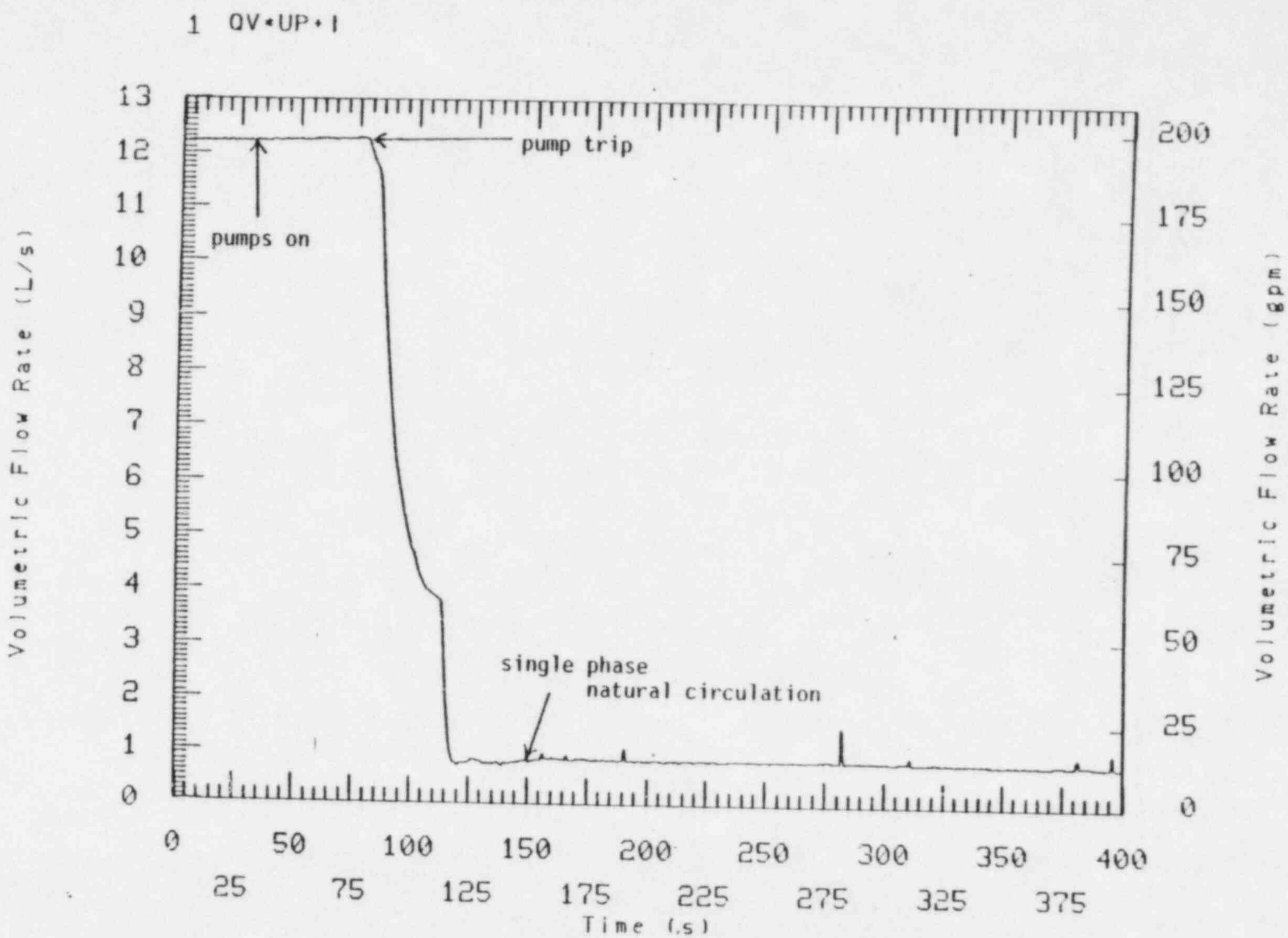


Figure 9. Change from pumped flow to single phase natural circulation flow upon pump trip.

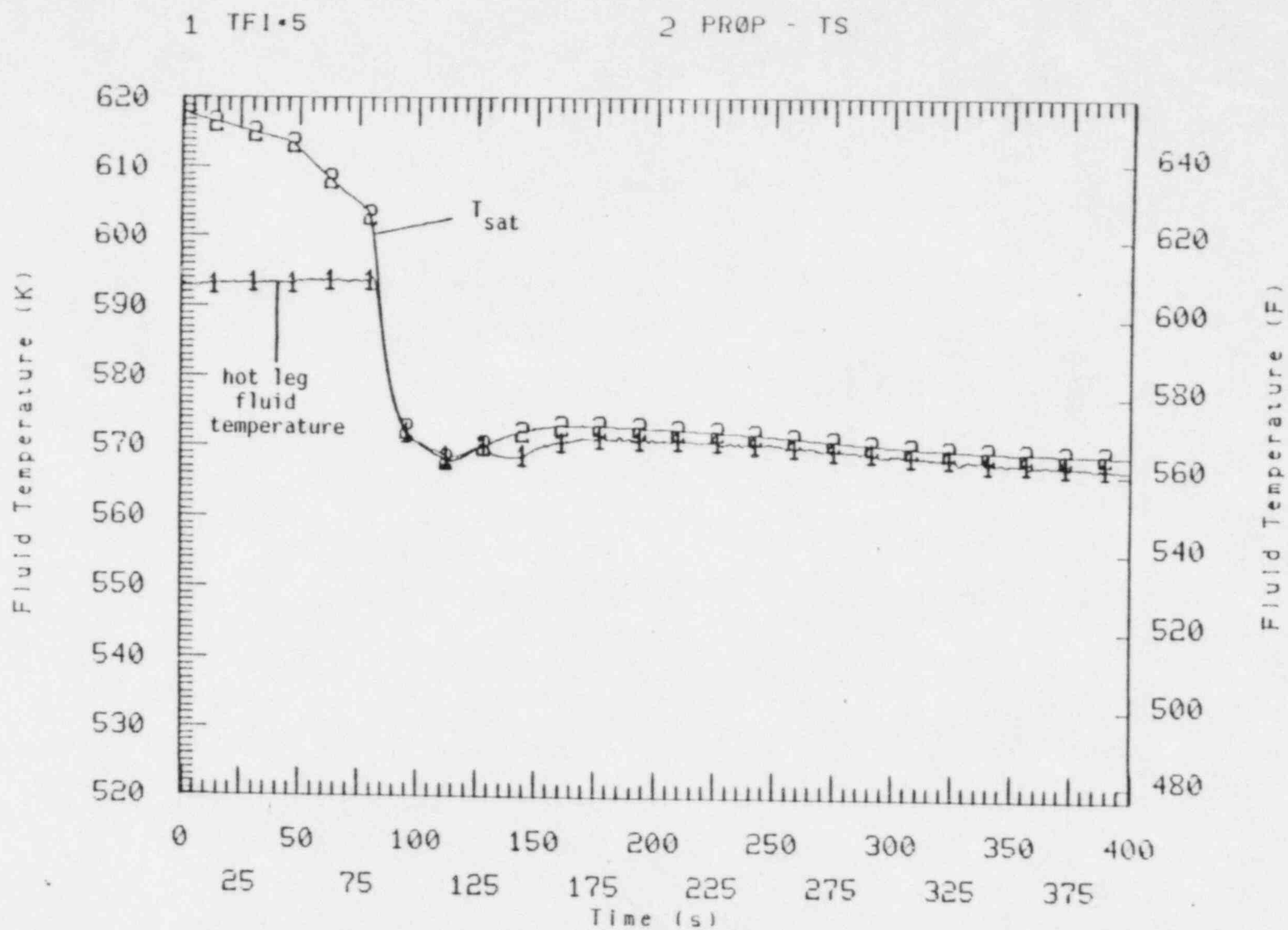


Figure 10. Comparison of hot leg fluid temperature and saturation temperature for single tube rupture cold leg side.

The primary to secondary break flow persisted throughout the initial period as shown on Figure 11. As long as break flow remained above total SI flow, primary system mass inventory depleted. Figure 12 shows the pressurizer collapsed liquid level depletion occurring in the initial 100 s.[a] The vessel upper head level remained fairly constant for the initial period as shown on Figure 13, but the measurement cannot be taken as a precise indication of level during this period. For the first 125 s the upper head level measurement was influenced by frictional pressure drop and velocity effects on the differential pressure. Once the loop pumps coasted down these flow effects were removed.

During the initial time period the steam generator collapsed liquid level was affected by ADV flow and auxiliary feed flow in the unaffected loop and break flow with ADV and auxiliary feed flow in the affected loop. Figure 14 shows the collapsed liquid level in both the unaffected and affected loop. The liquid level in both steam generators settled to a pool type condition following main steam isolation valve closure. Prior to main steam isolation valve closure the liquid level was affected by flow effects in the secondary near the differential pressure taps. During the initial time period, the affected loop steam generator had break flow into the secondary, auxiliary feedwater into the secondary and ADV flow out of the secondary as shown in Figure 15. The break flow was greater than either the ADV or auxiliary feedwater flow and dominated the mass balance during this period. Although the measurement of affected loop steam generator collapsed level indicated full on Figure 14 the net effect of break flow was to continue filling the steam generator secondary above the level of the upper pressure tap. Flow into the unaffected loop generator was from auxiliary feedwater and flow out was ADV flow as shown in Figure 16. The net effect was a slight filling of the unaffected loop steam generator during this initial period (see Figure 14).

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a. The level from S-SG-1B was substituted here for discussion purposes. The response for S-SG-1A and S-SG-1B are expected to be similar corresponding to the net loss of primary fluid to the secondary.

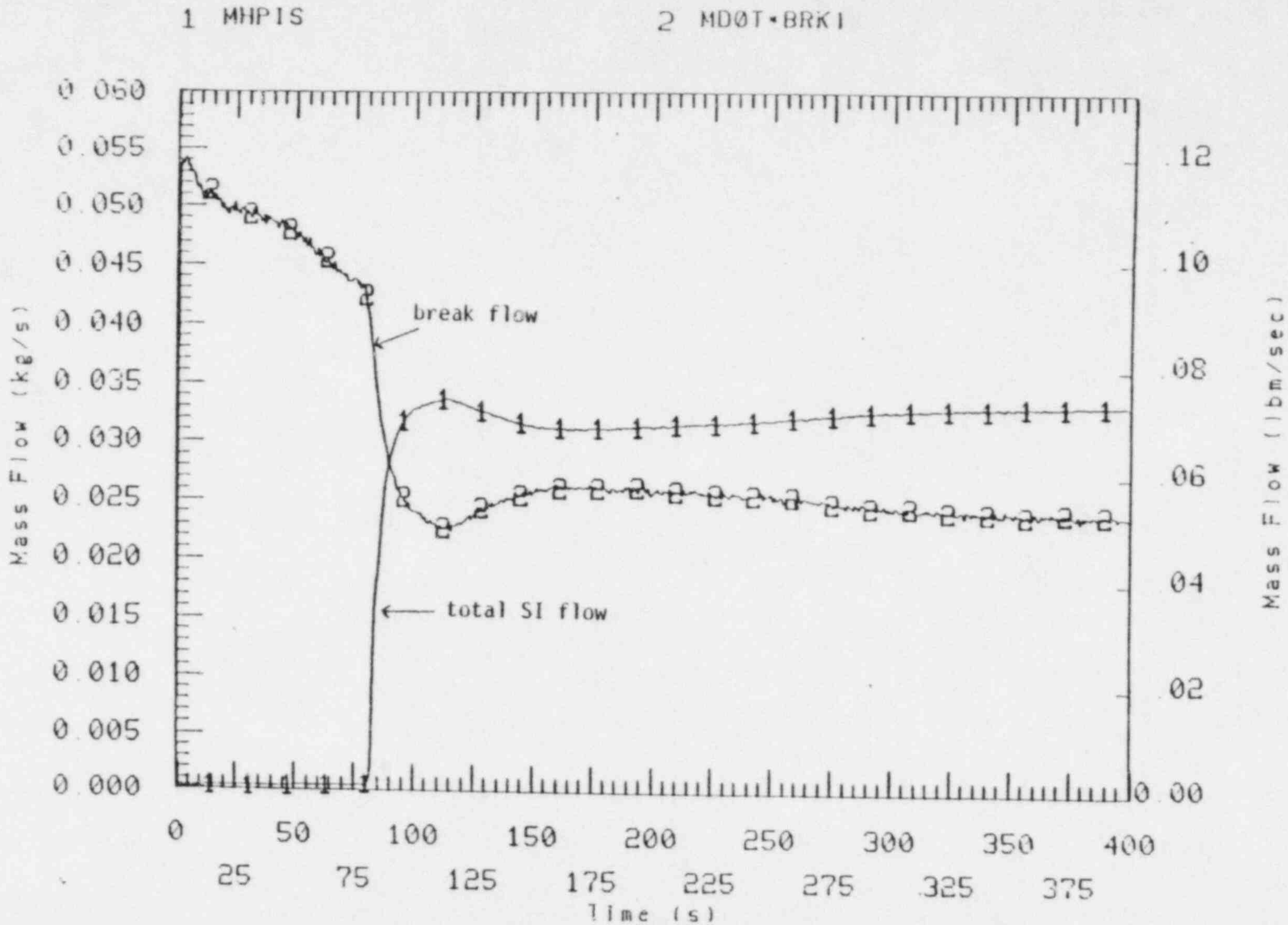


Figure 11. Comparison of tube rupture break flow and total SI flow.

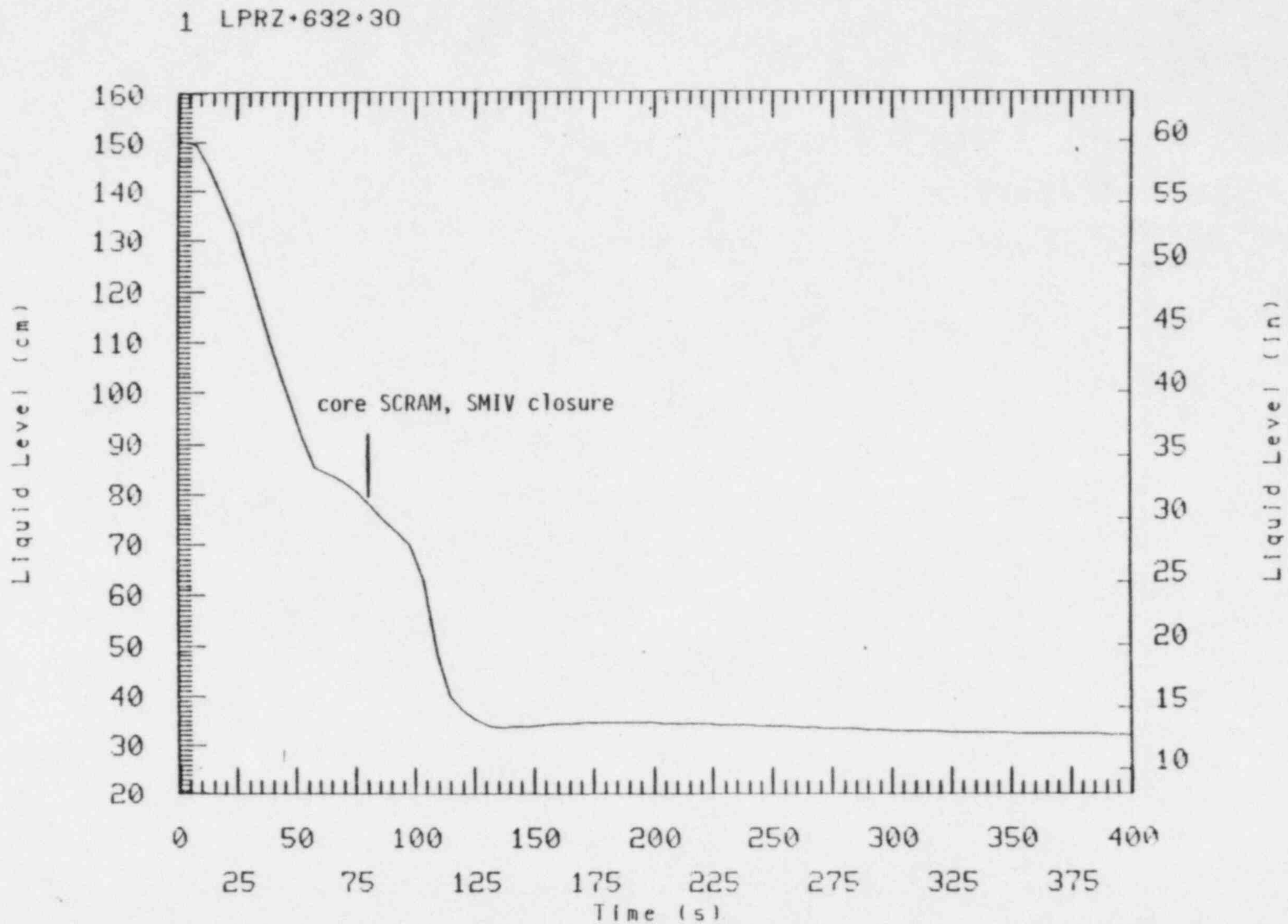


Figure 12. Pressurizer collapsed liquid level for Test S-SG-1B.

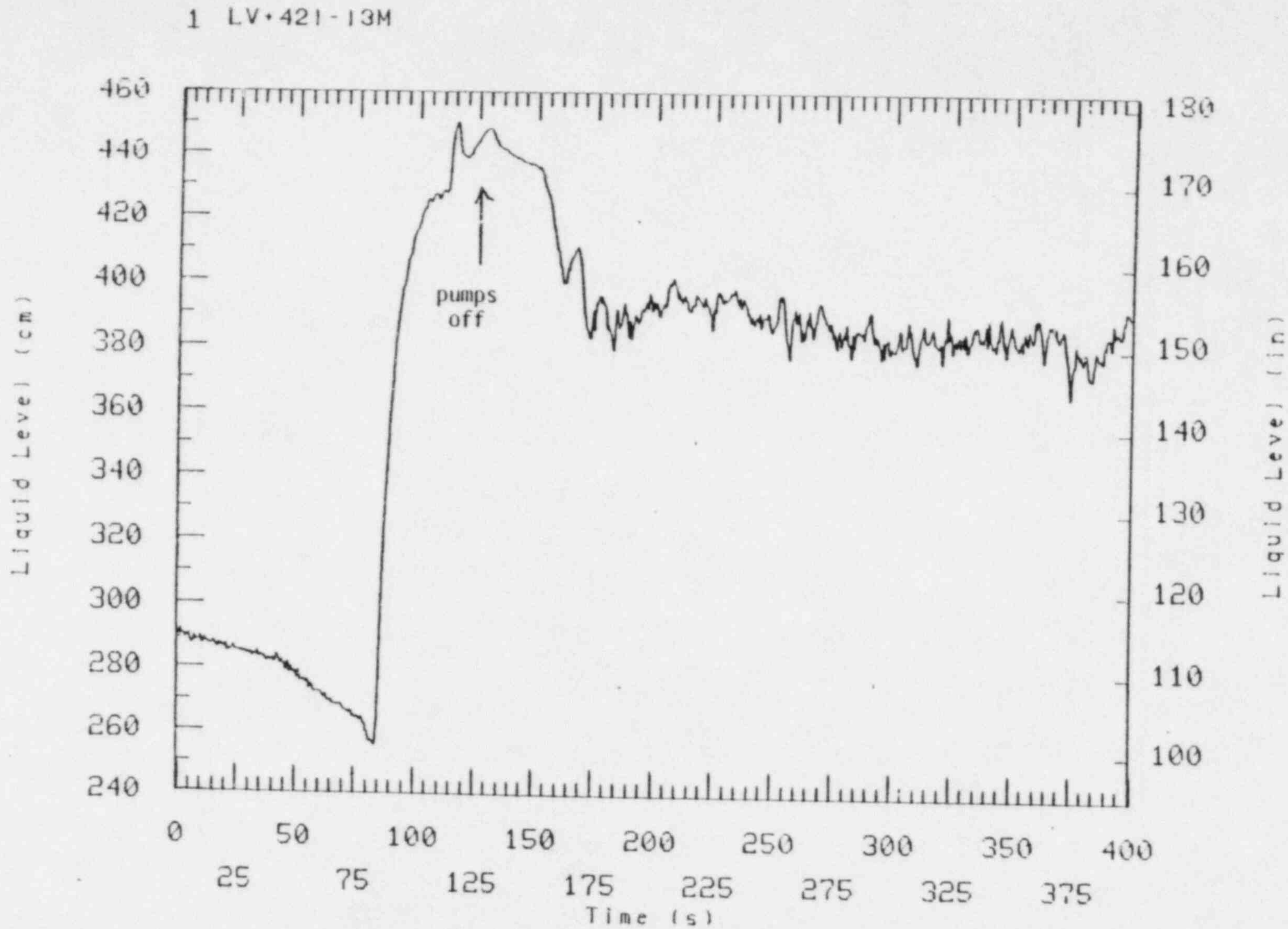


Figure 13. Vessel upper head collapsed liquid level.

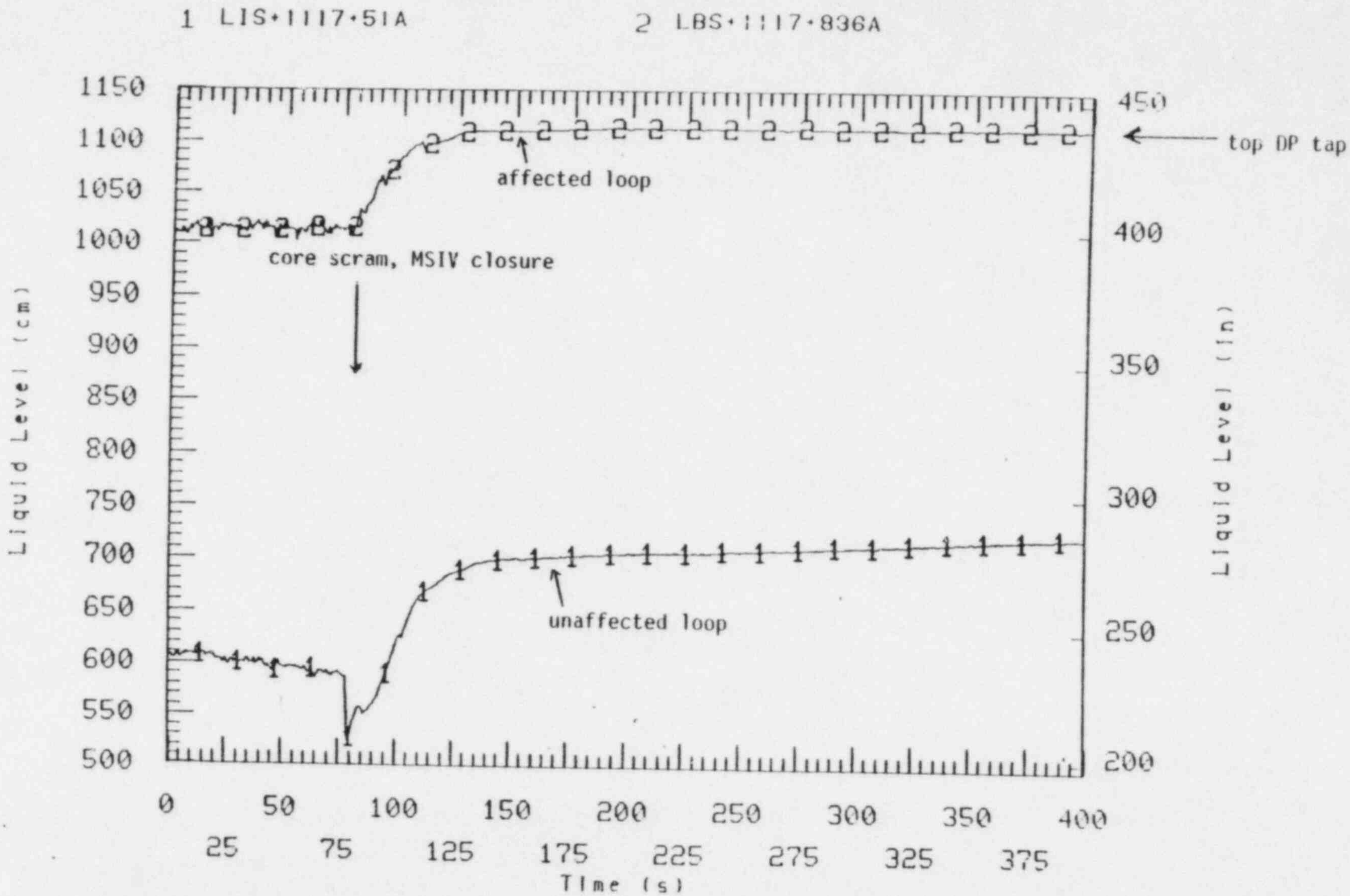
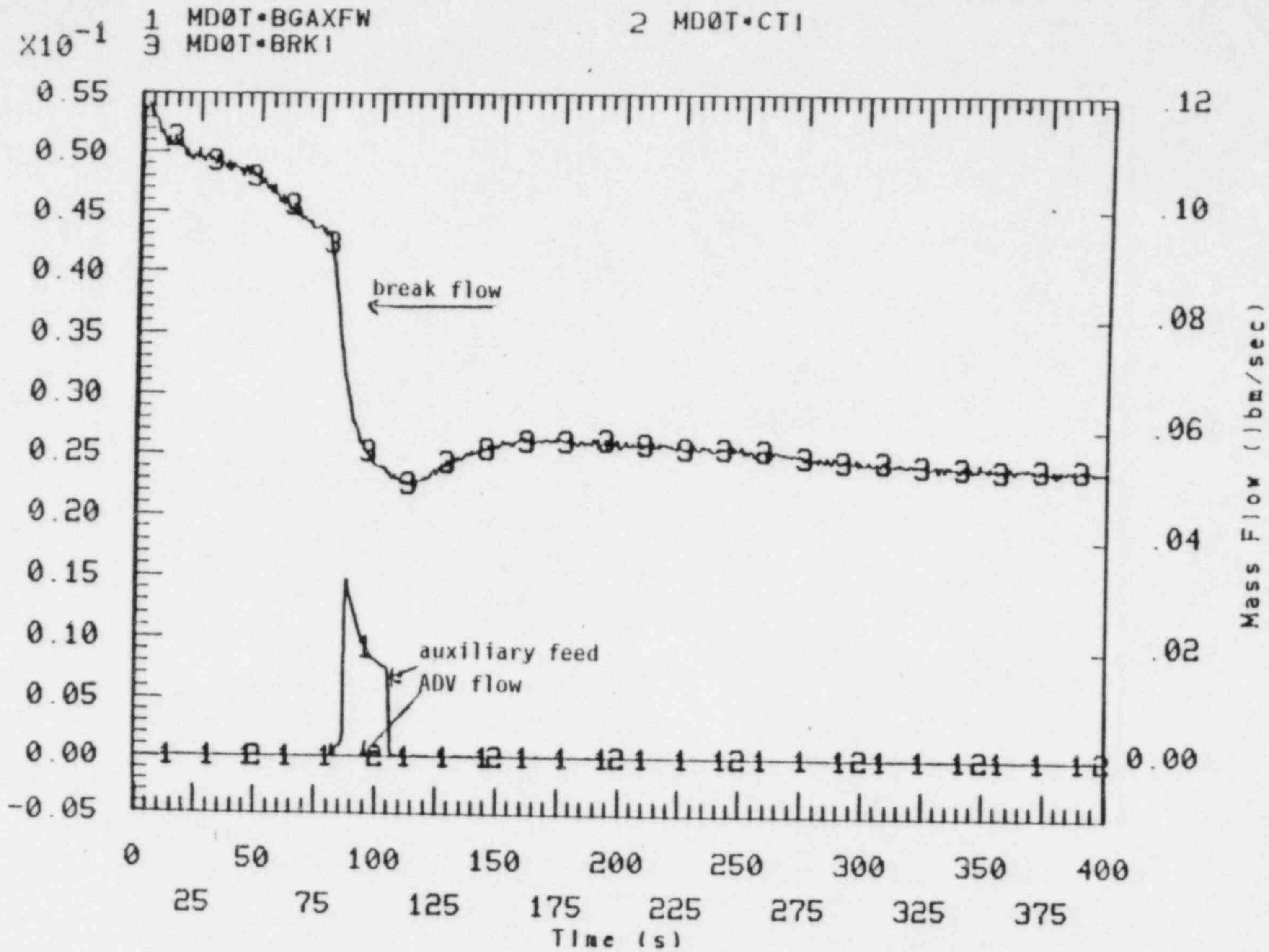


Figure 14. Comparison of affected and unaffected loop steam generator collapsed liquid level during a single tube rupture.



SG1A

Figure 15. Comparison of break flow, ADV flow and auxiliary feed flow in the affected loop steam generator during a single tube rupture.

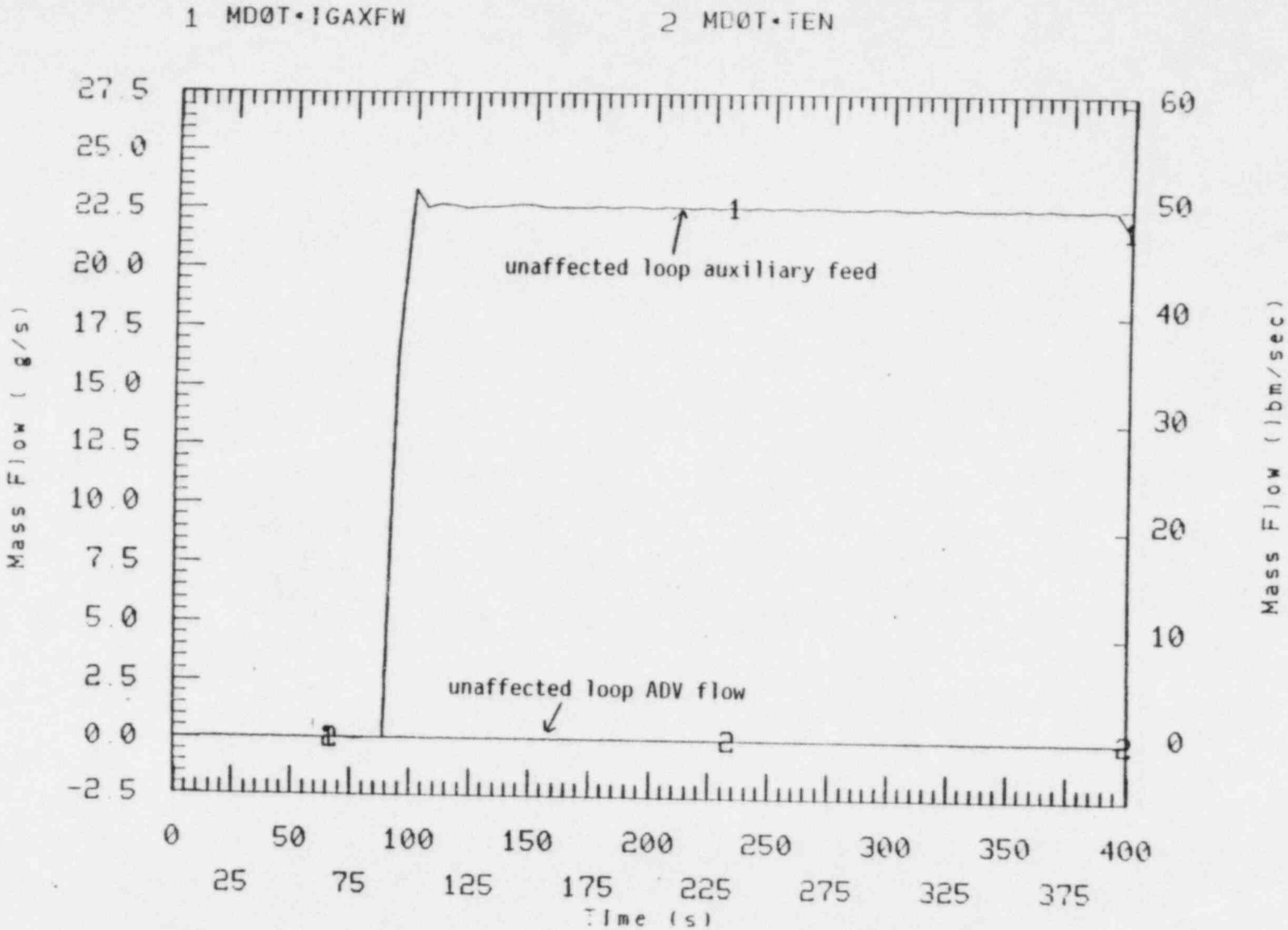


Figure 16. Comparison of ADV flow and auxiliary flow in the unaffected loop steam generator secondary during a surge tube rupture.

3.2 Tube Rupture Signature During the Recovery Phase

The recovery of the system for test S-SG-1 involved unaffected loop feed and bleed and termination of SI. Recovery in this sense meant reducing the primary pressure below the affected loop ADV setpoint and eventually reducing the pressure of the primary to equalize with the affected loop secondary pressure. This action in effect stopped the release of primary coolant to the environment via the ADV. Operator action at 600 s included latching open the unaffected loop ADV and maintaining the latched open condition as follows: If the collapsed secondary liquid level fell below 250 cm (98 in.), the ADV was shut. If the liquid level then increased due to auxiliary feed to 400 cm, the ADV was opened again. This operation was repeated to maintain the level between 250 and 400 cm (98 and 157 in.). At 3000 s the SI was shut off to examine the effect of SI on primary system pressure response.

Increasing the heat sink by ADV operation in the unaffected loop did not, by itself, decrease primary pressure. The overall primary and secondary system pressure response for test S-SG-1B is shown on Figure 17. The response for the first 600 s was essentially identical to S-SG-1A as discussed previously. At 600 s the unaffected loop steam generator ADV was latched open and the secondary pressure decreased as steam was released. This action created an increased heat sink for decay heat removal as shown on Figure 18, a comparison of the unaffected loop secondary and primary fluid temperatures. The unaffected loop ADV was in the latched open mode until about 3200 s at which time it was closed. The secondary level increased to about 400 cm (157 in.) above the tube sheet (about 4500 s), at which time the ADV was opened again. The primary pressure was only slightly affected by the ADV operation during this period. Core decay heat was removed by single phase natural circulation driven by overall loop density differences. Apparently the changes in heat sink conditions did not change the cold side density enough to affect the overall single phase natural circulation. In fact, the primary pressure started increasing at about 1500 s as HPIS flow pumped against a nearly full system. Referring to Figure 19 the pressurizer and upper head started filling at about the same time that the system pressure started increasing.

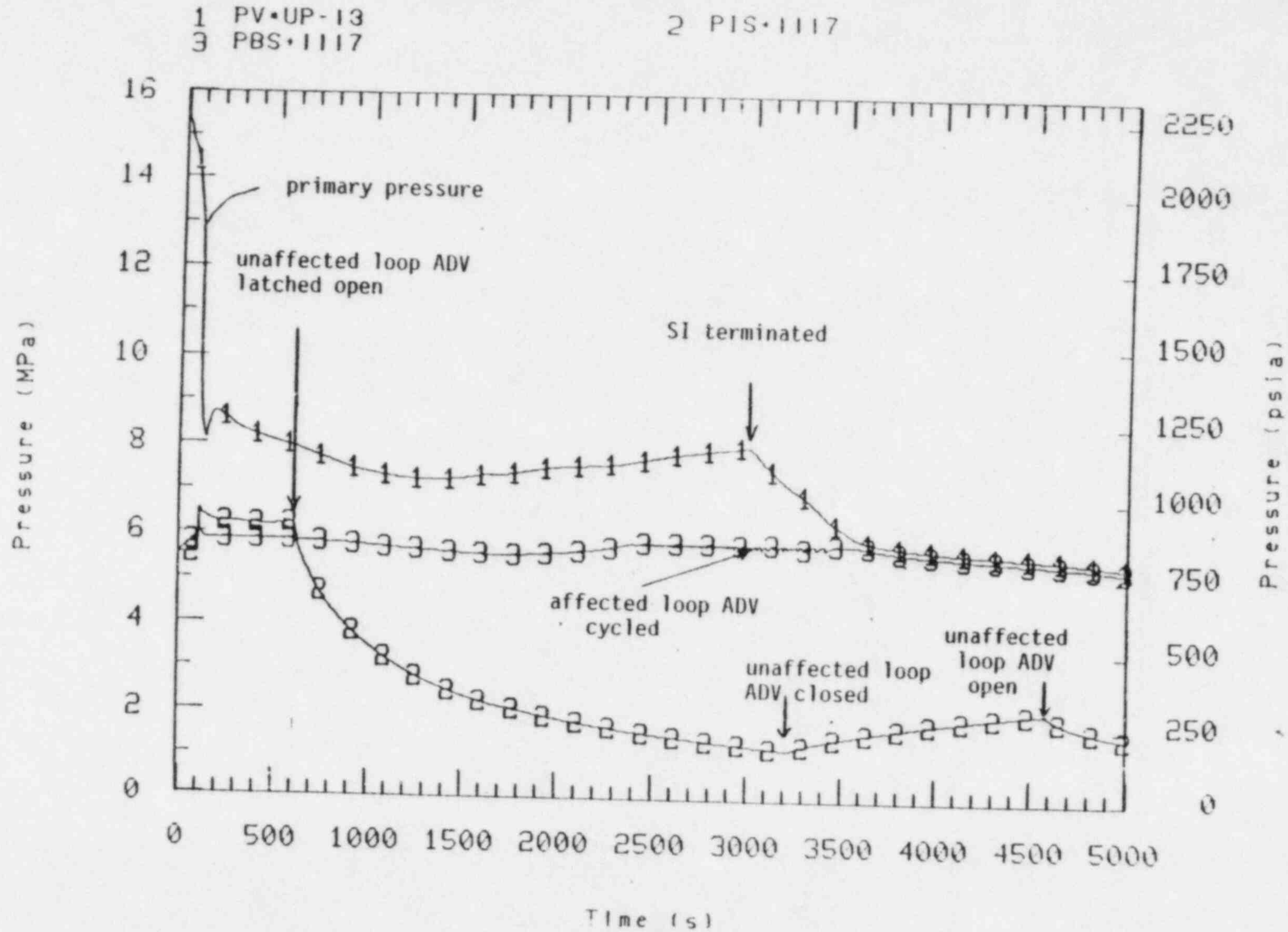


Figure 17. Comparison of primary and secondary pressure during a single tube rupture.

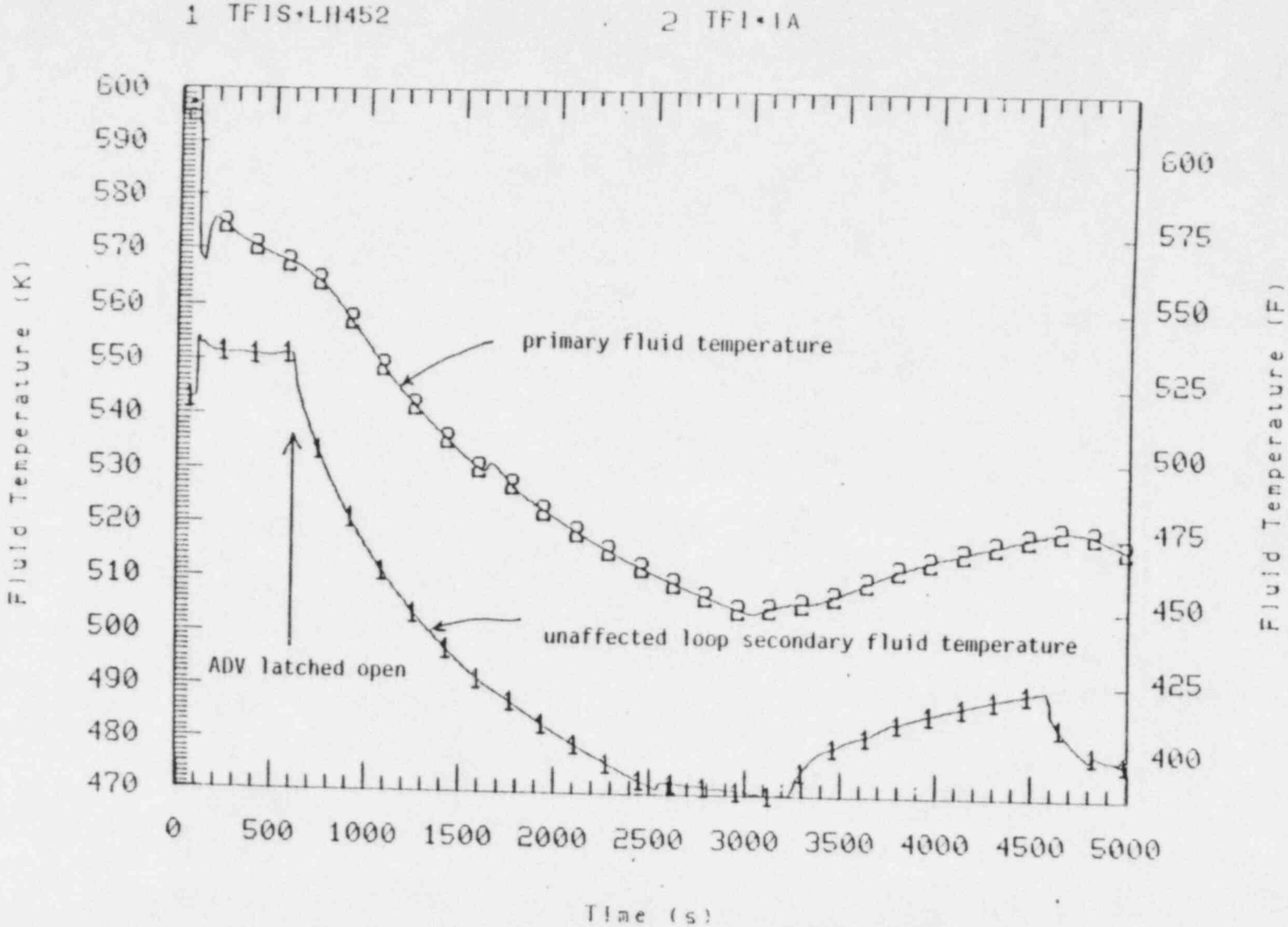


Figure 18. Comparison of primary and secondary fluid temperature.

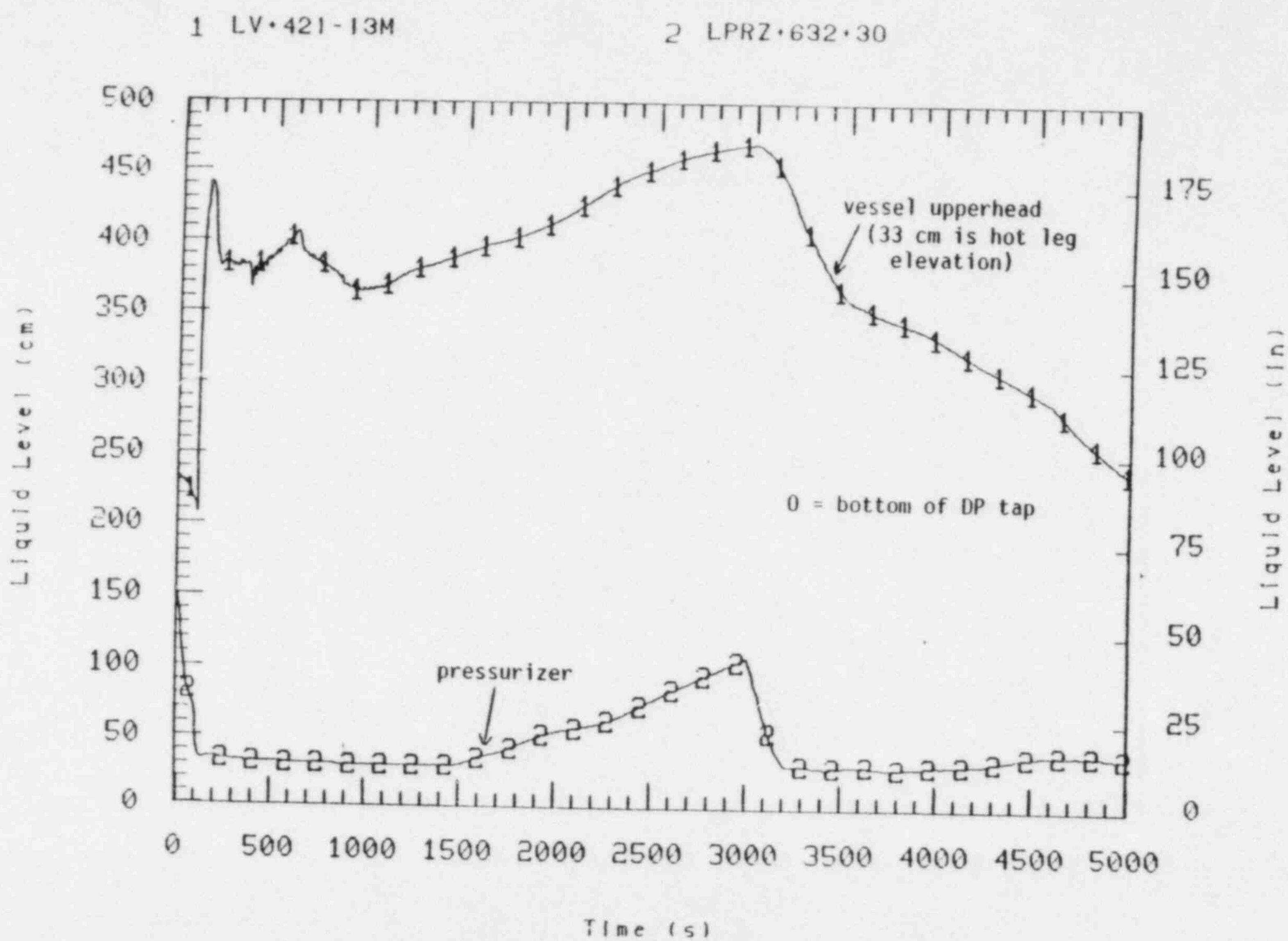


Figure 19. Pressurizer and vessel upper head liquid level for Test S-SG-1B.

The combined mass balance of SI flow and break flow was such that no significant voiding occurred in the vessel. During the first one hundred seconds, break flow depleted system mass; however, once SI was initiated SI flow was higher than break flow (Figure 20). This caused a net filling of the system during the period prior to SI termination (3000 s) as shown on Figure 19. At no time during the transient did the upper vessel level drop below the hot leg elevation.

The termination of SI had a large effect on the reduction of primary pressure as shown on Figure 17. Upon termination of SI, primary pressure reduced to the affected loop steam generator ADV setpoint (5.85 MPa (848 psia)) within 500 s, followed by a gradual reduction of both primary and secondary pressure. Prior to SI termination both primary and affected loop secondary pressure had been gradually increasing which eventually resulted in lifting of the affected loop ADV at about 3000 s. Cycling of the affected loop ADV persisted until about 3500 s. The gradual reduction in primary and affected loop secondary pressure was primarily caused by natural circulation in the unaffected loop. Following SI termination, the vessel upper head liquid level remained well above the hot legs (see Figure 19). Eventually the primary and secondary pressures nearly equilibrated which terminated the loss of mass from the system. This condition persisted until the end of the test at 5000 s as the primary and affected loop secondary pressures both gradually approached the unaffected loop saturation pressure.

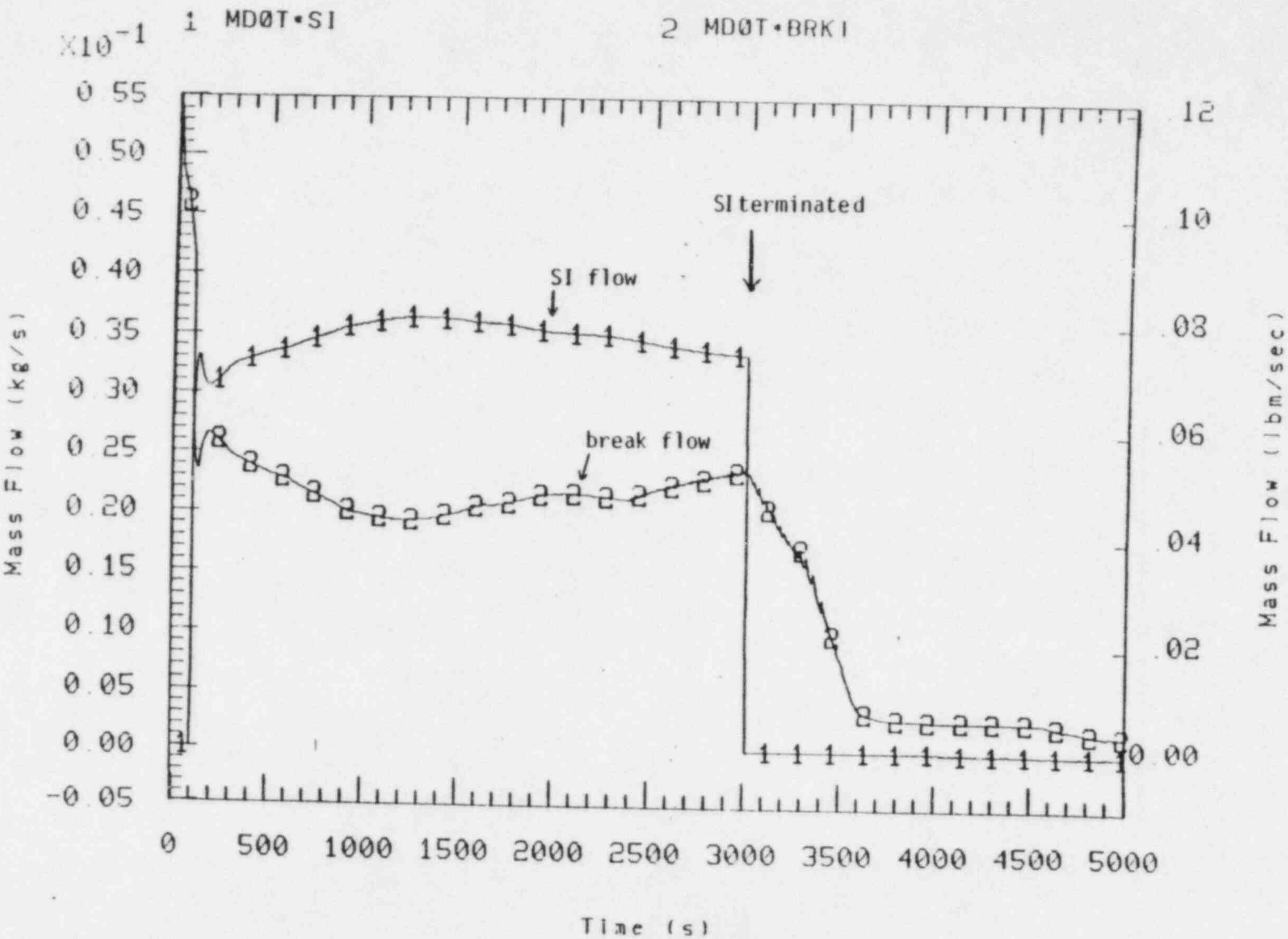


Figure 20. Comparison of tube rupture break flow and SI flow.

4. COMPARISON OF PRETEST CALCULATIONS TO DATA

This section compares the pretest calculation[5] to experimental data for Test S-SG-18. Tables 3 and 4 compare the calculated and experimental initial conditions and sequence of events respectively.

Upon steam generator tube rupture at time zero both code calculation and data exhibited a rapid depressurization of the primary coolant system (PCS) (see Figure 21). Good depressurization agreement was obtained for the first 70 s. The depressurization rate between 70 and 150 s was somewhat greater in the experiment than predicted as evidenced by the time of scram (92 s in the experiment compared to 136 s in the pretest calculation based on a low pressurizer pressure trip at 13.1 MPa (1900 psia)). It is believed that the higher measured depressurization rate is due to a decreased interphasic area due to the internal pressurizer heaters. The RELAP5 pressurizer model accounts for the area change due to the heaters but did not calculate the increased depressurization rate. This may be due to the fact that vapor generation is a homogeneous calculation for the entire volume. The SIS (12.5 MPa (1814 psia)) was obtained approximately 5 s after scram in both calculation and experiment. In general, good agreement between the measured and calculated PCS pressure was achieved through the first 1500 s of the transient.

For both code calculation and data the pressurizer liquid level[a] showed a rapid drop as primary fluid flowed from primary to the affected loop secondary. Figures 22 and 23 show the break flow rate and pressurizer liquid level comparisons, respectively. SI flow was sufficiently higher than break flow in both code calculation and experiment to cause a filling of the pressurizer starting at 1200 s in the calculation and 1500 s in the experiment. Figure 22 shows, however, that the RELAP5 calculated break flow as a function of pressure was consistently lower than that measured with the calibrated conical break orifice.

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a. The liquid levels discussed in this section are pooled liquid level rather than collapsed liquid level as used in the previous sections.

TABLE 3. INITIAL CONDITIONS FOR S-SG-18

	Calculated	Measured
Primary Cold Leg Flow Rate (Nominal)		
Affected Loop	2.61 L/s (41.4 gpm)	2.84 L/s (45.0 gpm)
Unaffected Loop	7.22 L/s (114 gpm)	8.44 L/s (134 gpm)
Pressurizer Pressure	15.59 MPa (2262 psia)	15.42 MPa (2236 psia)
Pressurizer Liquid Volume	.0106 m^3	0.0101 m^3 (0.358 ft^3)
Core Power	2.0 MW	2.0 MW
Loop to Loop Cold Leg Fluid Temperature Differential	1 K (1.8 F)	2 K (3.6 F)
Core Fluid Temperature Rise	36.9 K (66.4 F)	39 K (70.2 F)
Steam Generator Pressure		
Affected Loop	5.56 MPa (806 psia)	5.53 MPa (802 psia)
Unaffected Loop	5.54 MPa (804 psia)	5.48 MPa (784 psia)
Steam Generator Secondary Fluid Mass		
Affected Loop	99.3 kg (219 lbm)	188 kg[a] (415 lbm)
Intact Loop	100.2 kg (221 lbm)	107 kg[a] (236 lbm) (1117-51) 158 kg[a] (1117-836)
Primary Leakage at t = 0	0.003 kg/s (6.6E-3 lbm/s)	0.000712 kg/s (1.57E-3 lbm/s)

a. These values were determined from data acquisition system levels following main steam isolation valve closure. Initial conditions were established using process indicated levels which have a high uncertainty in a steam condition; however the specified process levels where achieved prior to test initiation.

TABLE 4. ACTUAL VS CALCULATED CHRONOLOGY OF EVENTS

Event	Actual(s)	Calculated
Steam generator tube rupture (block valve open)	0	0
Pressurizer heaters off	0	0
Low pressurizer pressure reactor trip	91	136
Main steam isolation valves closed	91	136
Safety injection signal	96	141
HPIS/charging initiated	96	141
Main feedwater secured	96	141
Auxiliary feedwater actuated	96	141
Reactor coolant pumps tripped	96	141
Recovery began	600	600
First closure of ILSG ADV	3200	1800
Terminate SI	3000	3000
Primary pressure equals affected loop secondary pressure	5000	5000
Test termination	5000	5000

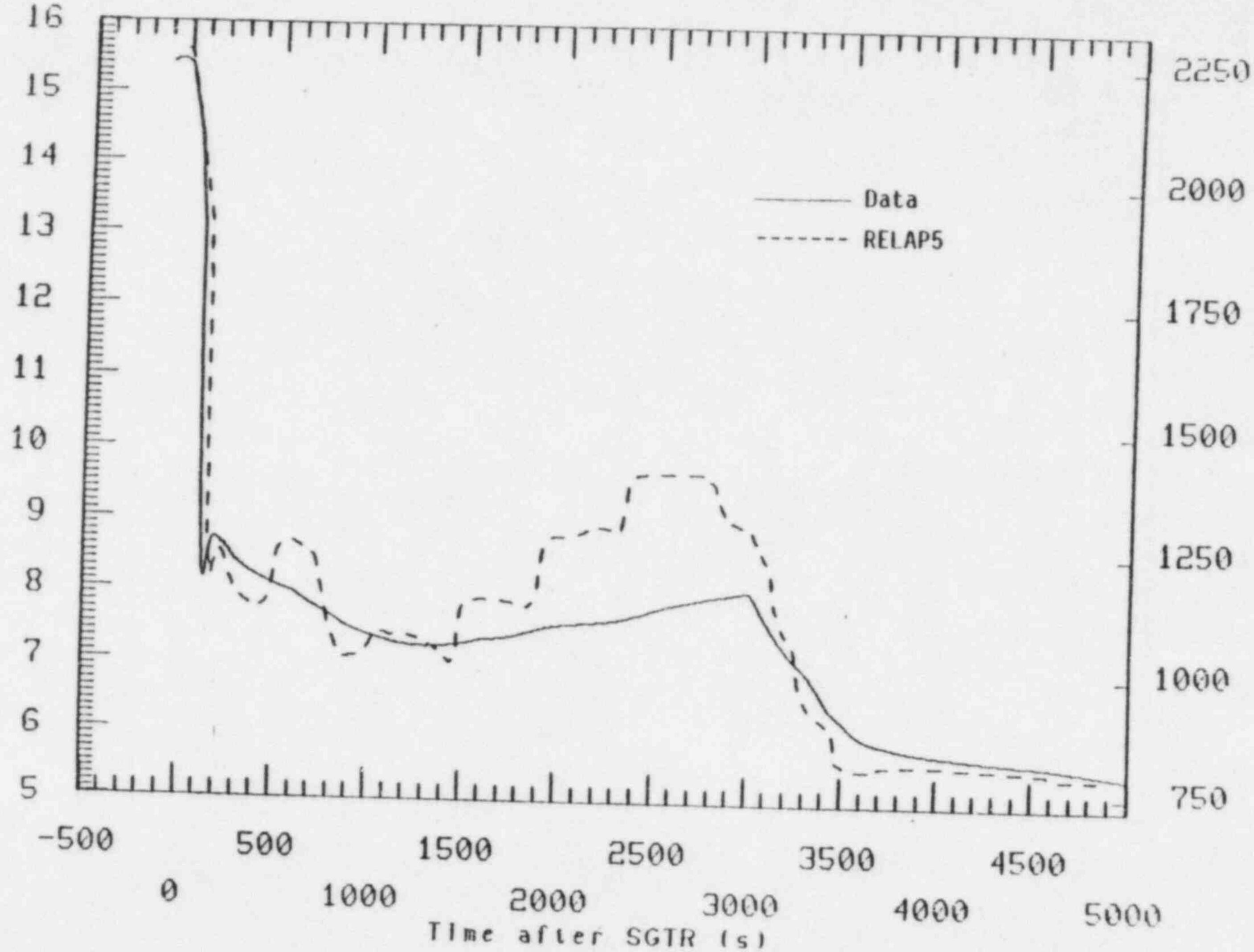

Figure 21. Comparison of primary pressures.

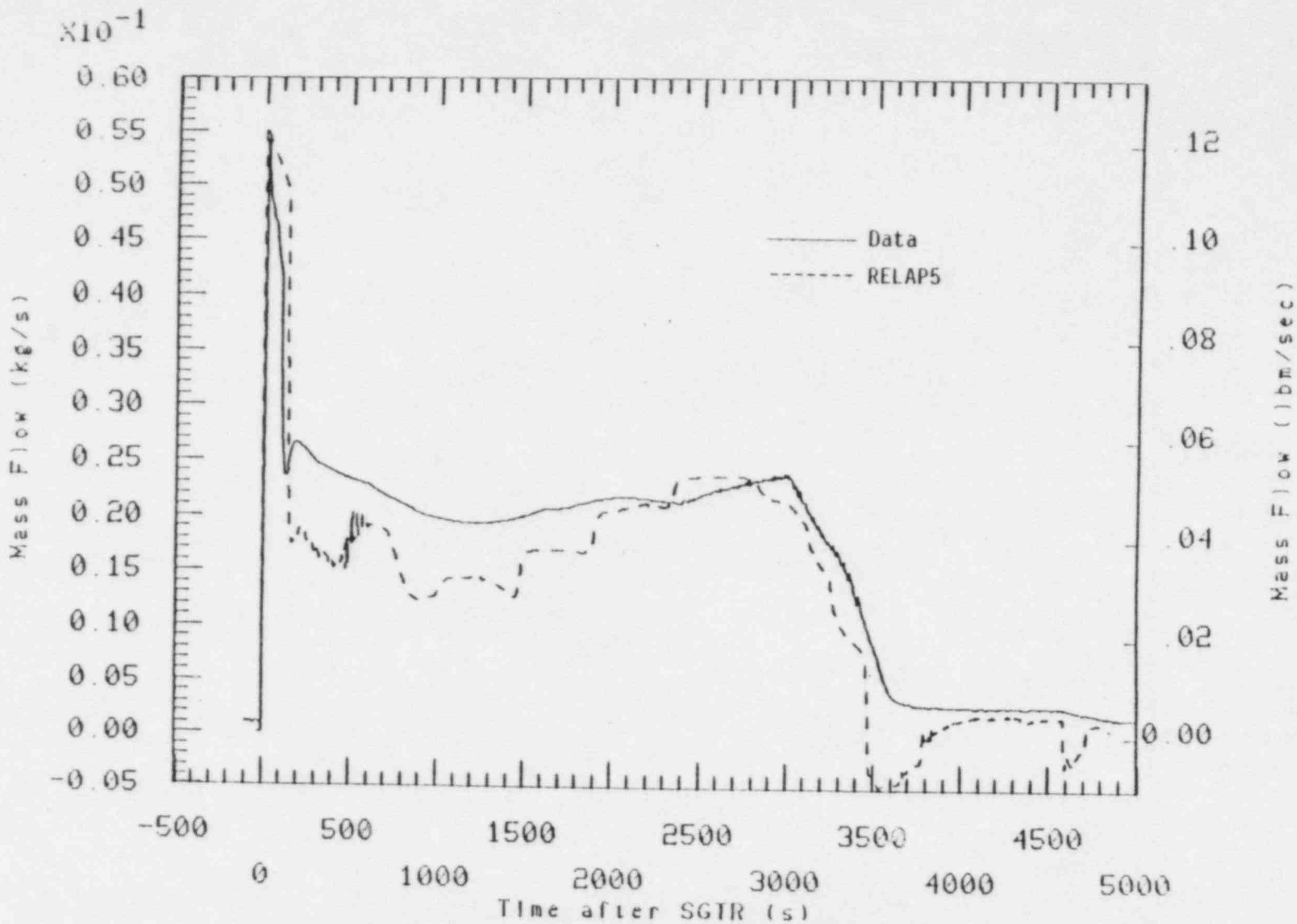

Figure 22. Break flow comparison.

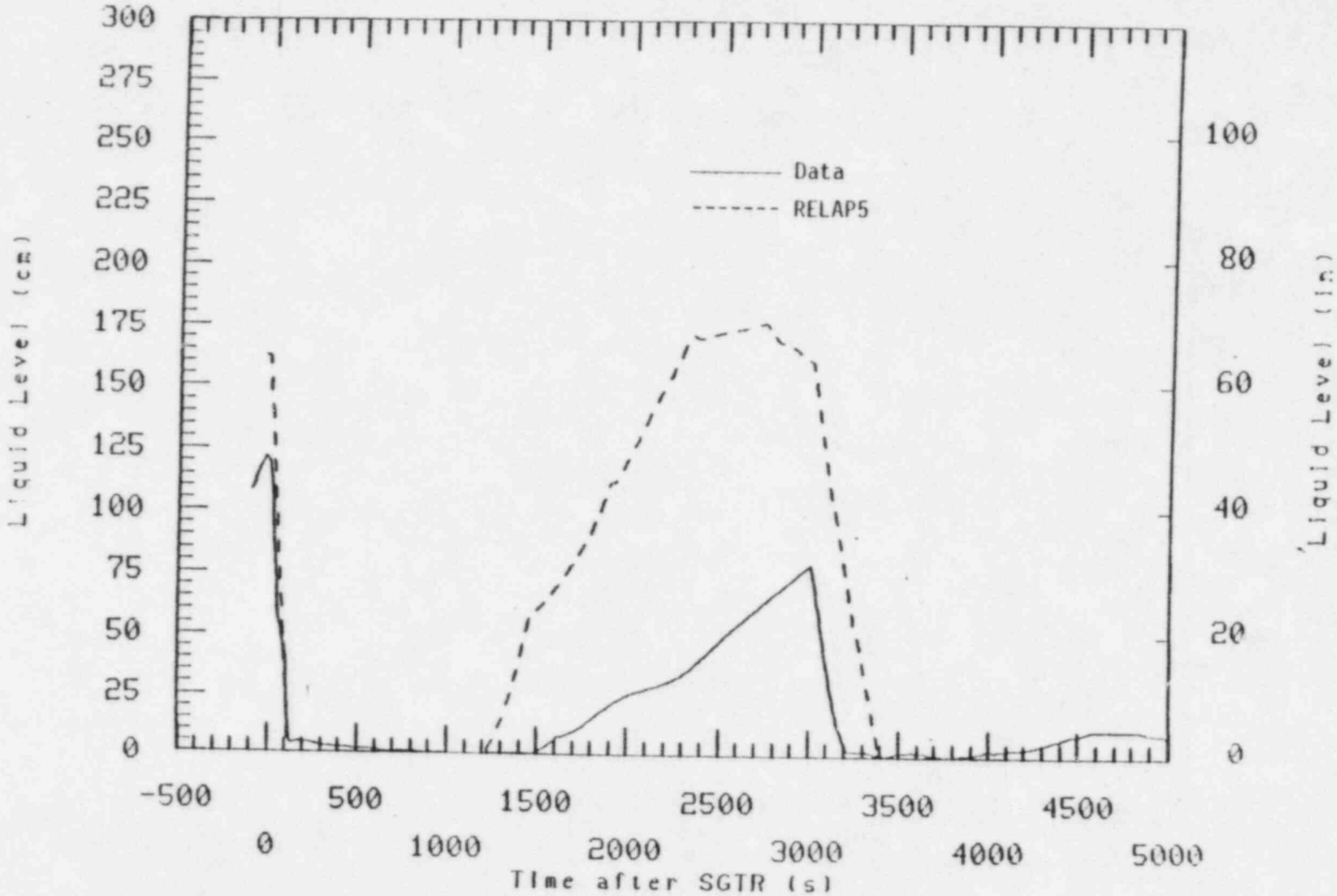

Figure 23. Comparison of pressurizer liquid level.

As the pressurizer began refilling during the 1200 to 1500 s time period, the calculated pressurizer pressure began to increase at a much higher rate than the measured pressure (see Figure 21). An increase in calculated pressure when a voided pressurizer is refilled is due to three phenomena: (a) compression of the vapor space as its volume is reduced by the rising water level, (b) vapor generation at the wall, and (c) vapor generation at the liquid/vapor interface. The third mechanism is considered insignificant relative to the others. The pressure increase due to compression of the vapor space should have been larger for the calculation than the experiment simply because the pressurizer was calculated to fill higher than actually occurred. The calculated steam temperature in the upper region of the pressurizer did not decrease as rapidly as measured during the test (Figure 24), indicating less wall heat loss computed by the code. This difference in steam temperature in the upper part of the pressurizer suggests that the pressurizer model is probably predicting environmental heat loss incorrectly. Also indicative of this deficiency is the amount of superheat present in the pressurizer steam space (see Figure 25). As the PCS pressure (hence the saturation temperature) decreased after 3000 s the calculated superheat increased due to heat transfer from the walls. Since the measured superheat is considerably less than calculated, it is concluded that the data wall temperatures cooled more quickly, leading to less vapor generation at the wall.

Figure 26 shows a comparison of the calculated and measured primary and secondary pressures, showing overall good agreement between calculation and data. Two prominent events occurred which affected these pressures: operation of unaffected loop ADV with feed to increase the heat sink at 600 s and termination of SI at 3000 s. Upon terminating SI in both experiment and calculation the pressurizer began to drain thus depressurizing the PCS. At approximately 3500 s the calculated primary and affected loop steam generator secondary pressures were essentially equal. In the experiment, these pressures were close to equilibration, however there was still a small break flow (0.002 kg/s) at the end of the test. The calculation had predicted two short periods of reverse flow. The test

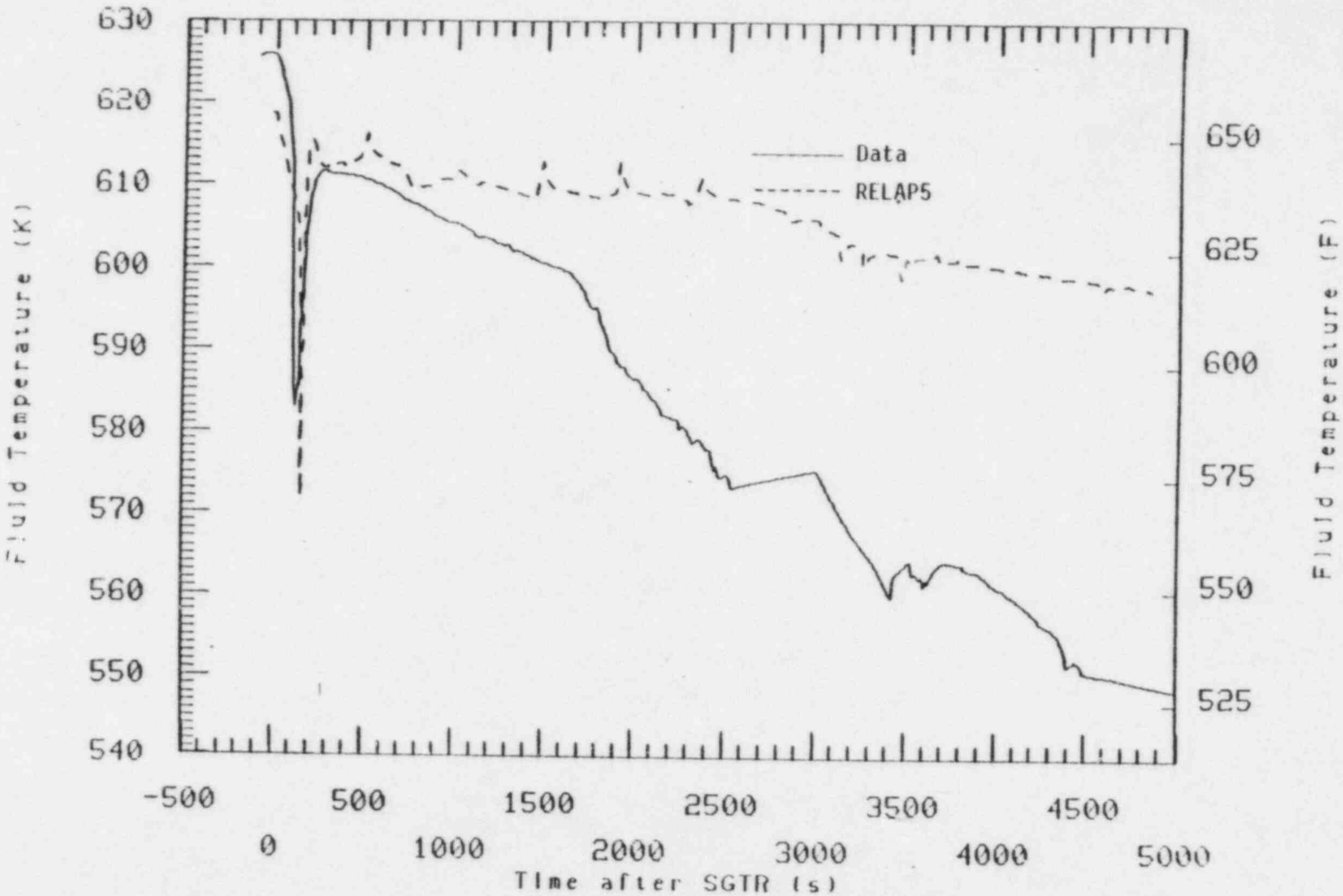

Figure 24. Upper pressurizer temperatures.

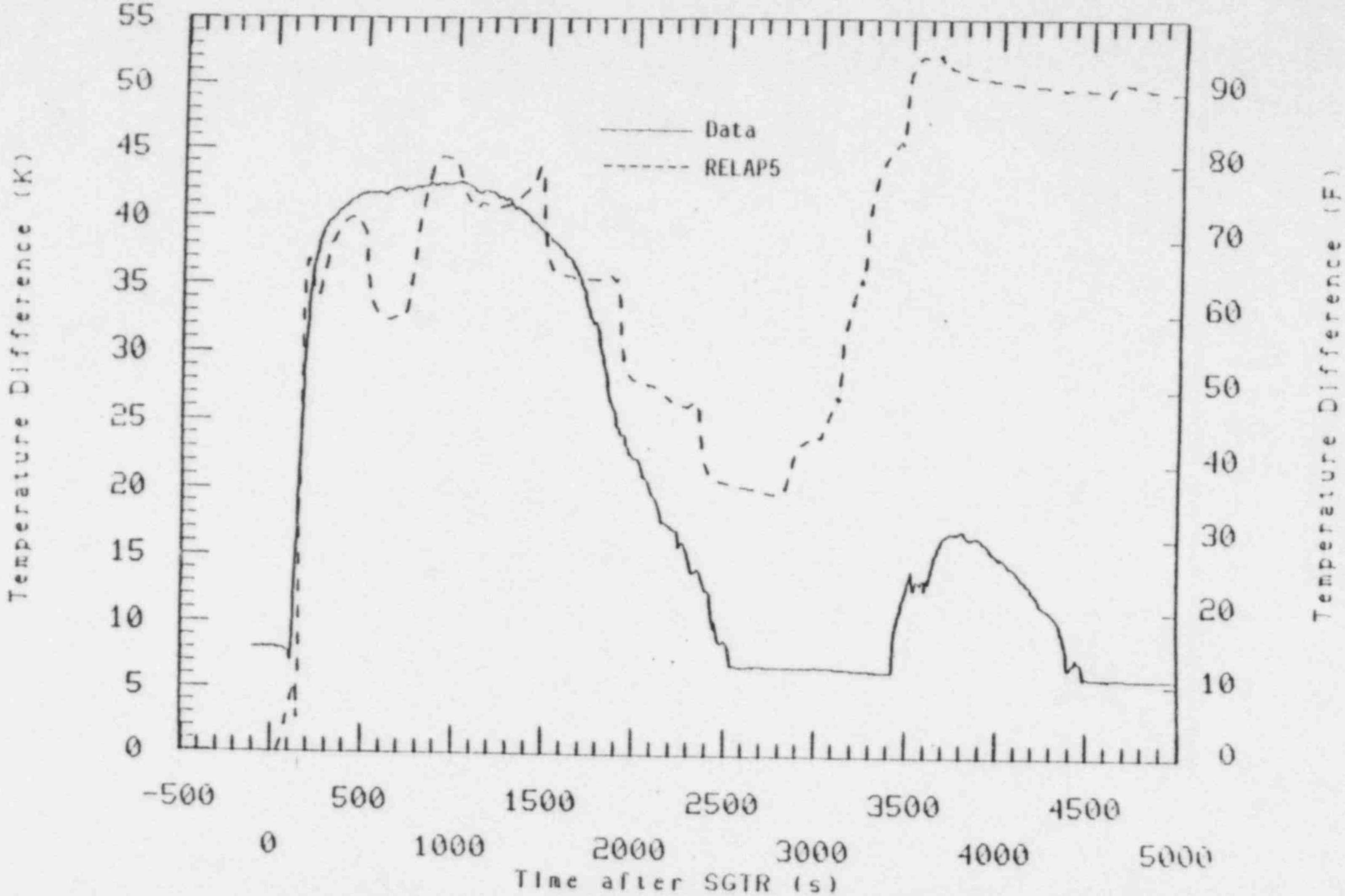

Figure 25. Pressurizer superheat comparison.

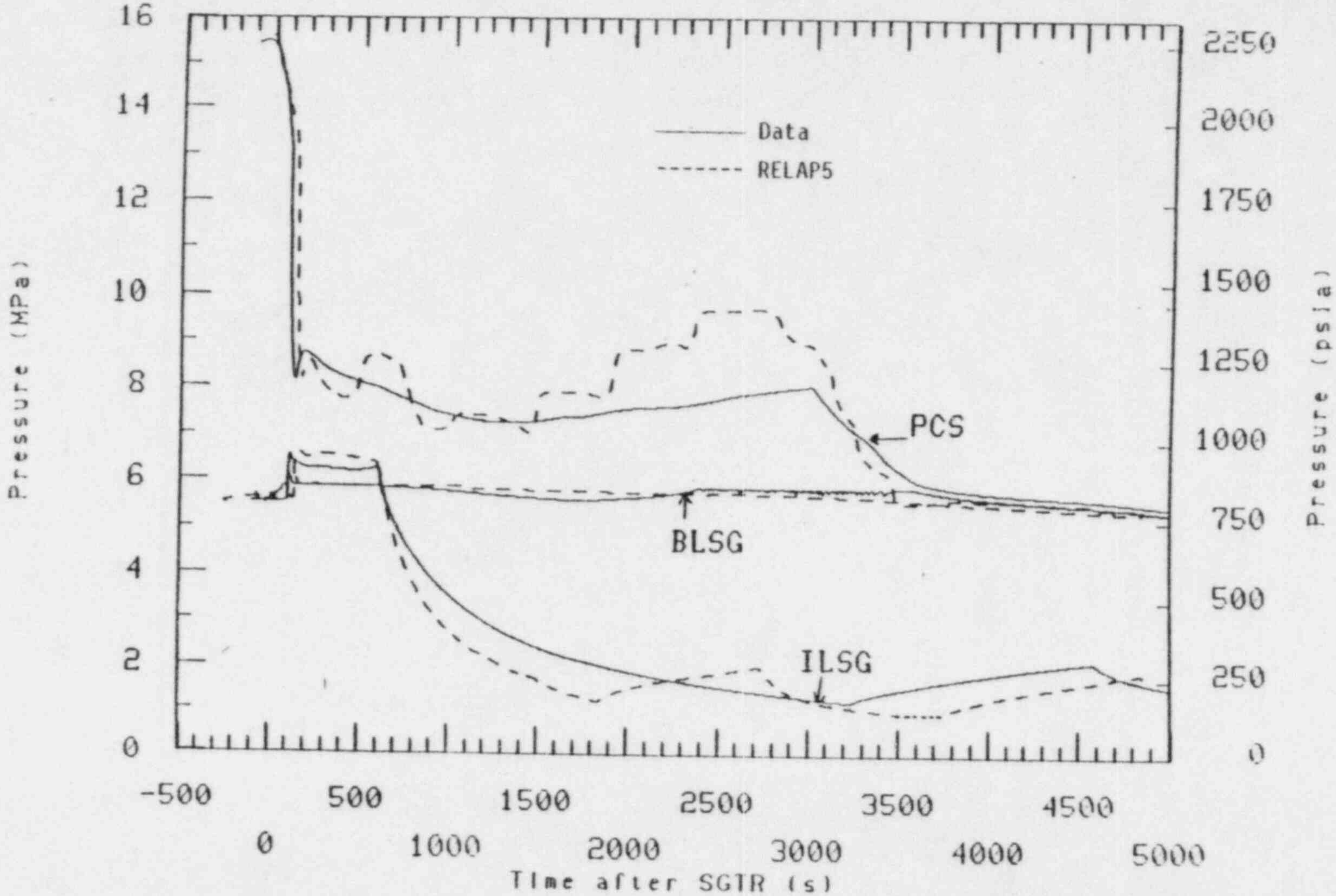

Figure 26. Comparison of primary and secondary pressures.

was terminated after it was shown that unaffected loop feed and steam could maintain the PCS at a pressure less than the affected loop relief valve setpoint.

The operation of the unaffected loop steam generator ADV can also be seen in Figure 26 commencing at 600 s. The first closing of the ADV was predicted to end at 1800 s, whereas in the experiment, the valve closed for the first time at about 3200 s. This difference was caused by a higher initial secondary coolant mass in the test. Figure 27 shows that the predicted and measured ADV flowrates from the unaffected loop generator agreed well for the entire test. The difference in total mass through the unaffected loop ADV is probably due to the initial mass discrepancy. The affected loop ADV total flow comparison shows an early difference which again is attributed to having too much mass in the affected loop secondary initially (hence lower quality flow through the valve). There also was a discharge from the affected loop steam generator during the 2500 to 3500 s period that can be seen on the expanded plot of affected loop secondary pressure, Figure 28. The affected loop secondary pressure increased to the relief valve setpoint during the 1700 to 2500 s period and remained there for approximately 1000 s. The PCS also underwent a repressurization during this time. It seems to have been reflected more in the affected steam generator secondary pressure than predicted. One possible explanation is that more heat loss from the affected loop steam generator was predicted than actually occurred. In addition, it is likely that the affected loop secondary was nearly water-solid during this stage of the experiment (see Figure 29 and note that the upper tap of the level DP is located at 1117 cm (440 in.)) and therefore could read no higher. The absolute top of the affected loop secondary is at 1174 cm (462 in.)).

Overall, good qualitative agreement was achieved between the pretest calculation and experimental data. Better knowledge of initial levels in the experiment in the steam generator secondaries and pressurizer heat loss characteristics might have resulted in better quantitative agreement; however, all trends and approximate magnitudes were closely predicted.

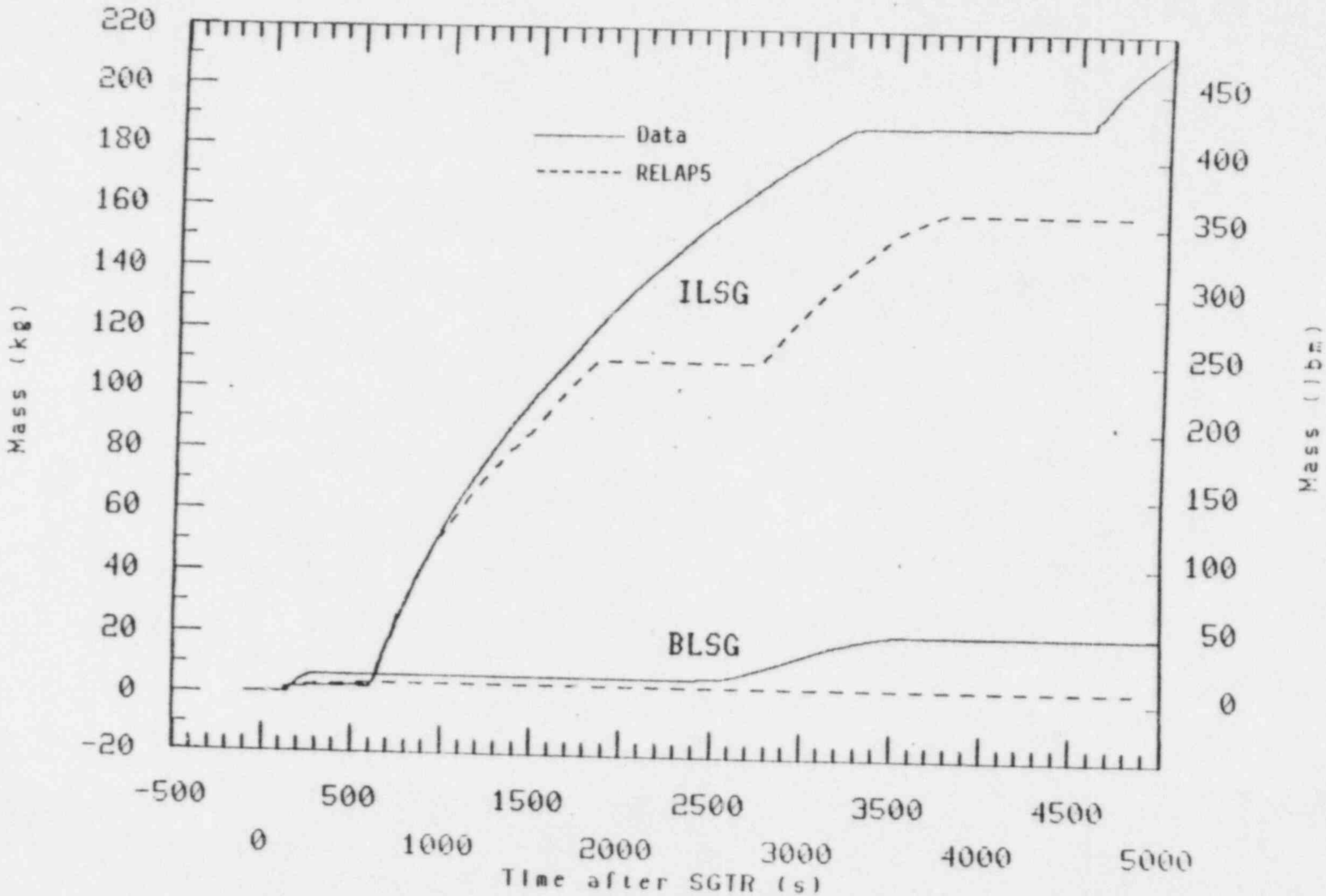

Figure 27. Total ADV/SRV flow comparisons.

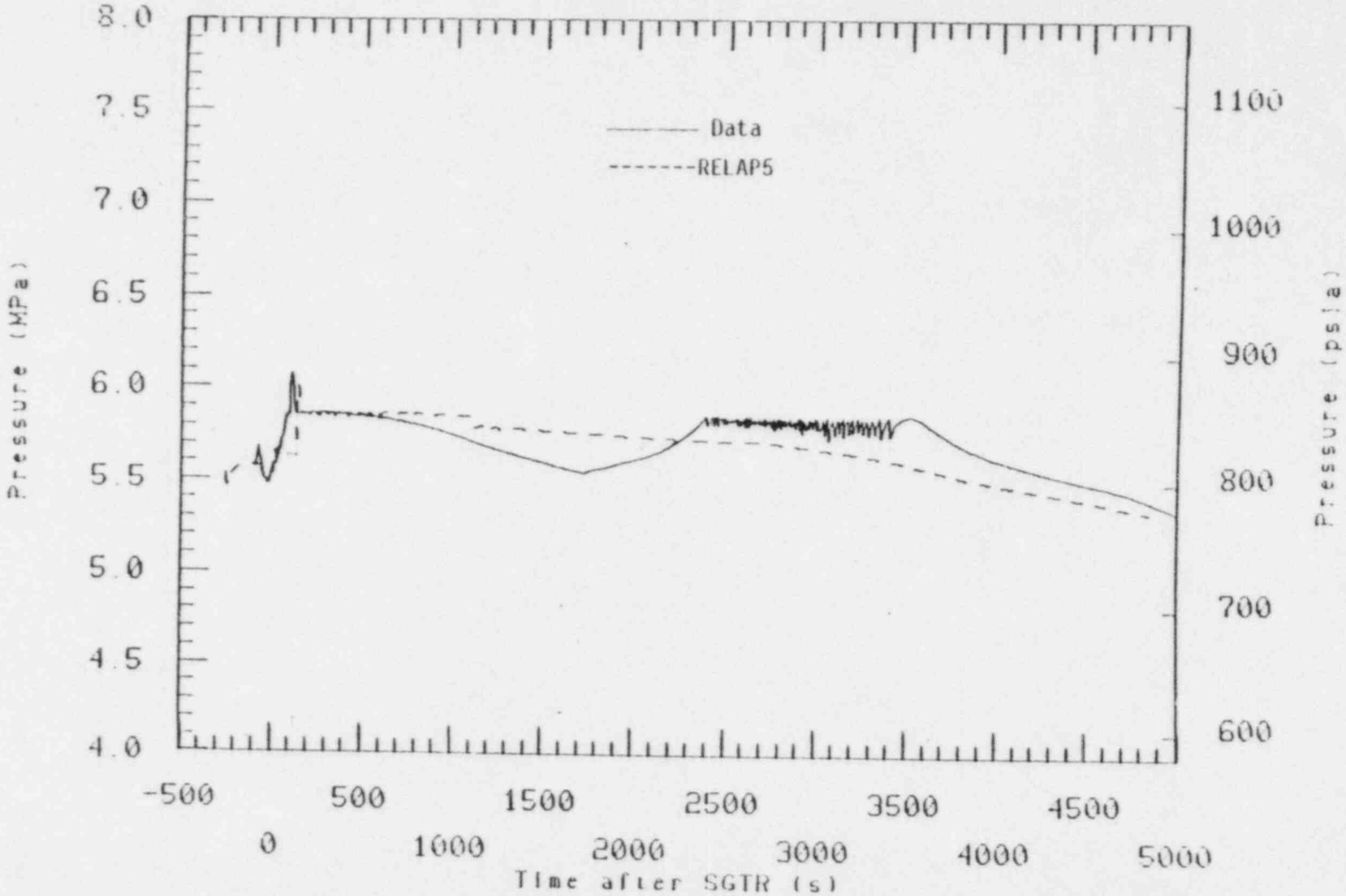

Figure 28. BLSG secondary pressure comparison.

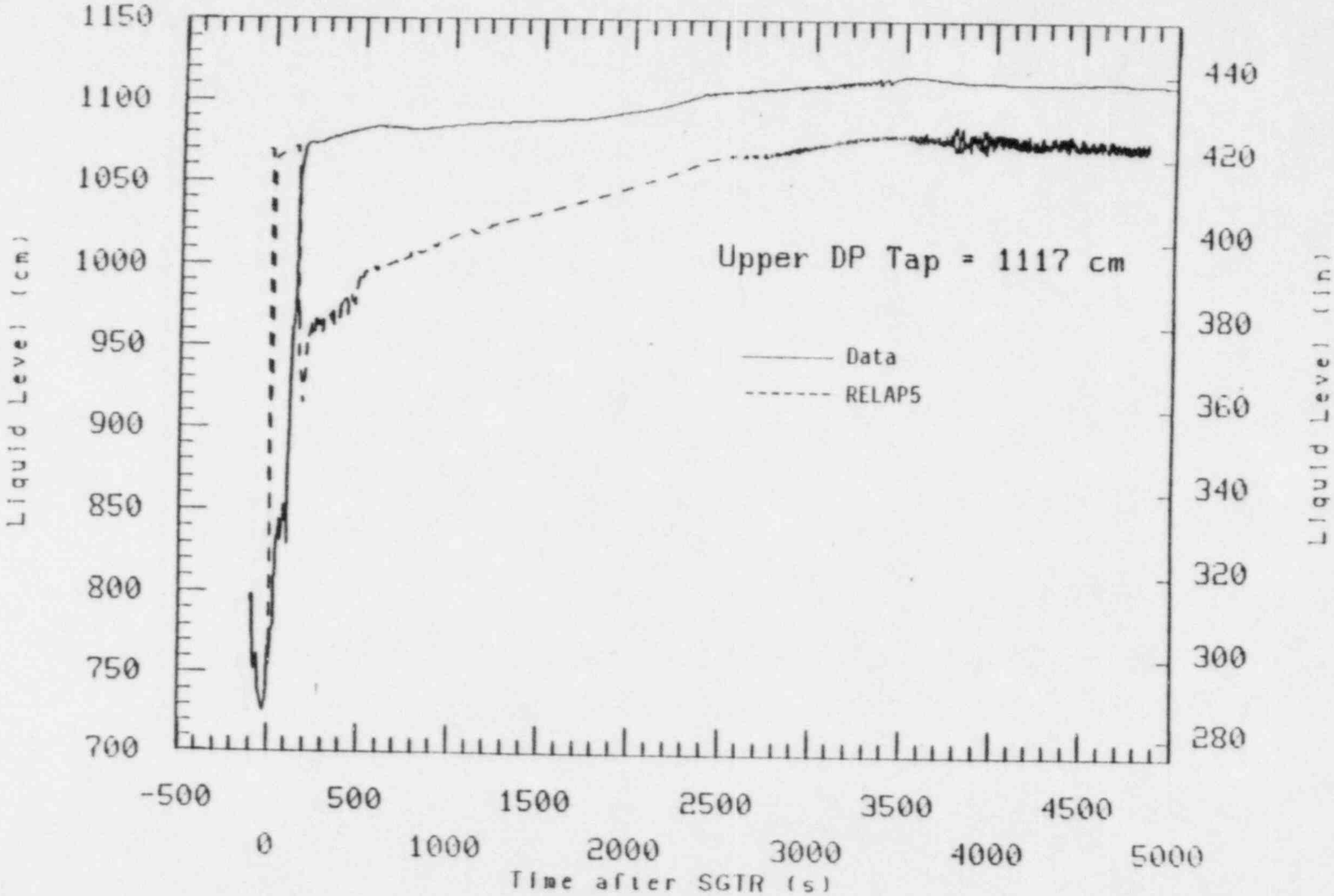

Figure 29. BLSG downcomer liquid level comparison.

5. CONCLUSIONS

1. Unaffected steam generator steam and feed and termination of safety injection flow was a sufficient operator response to recover the Semiscale facility from a simulated single steam generator cold side tube rupture.

   Recovery criteria included reducing the primary system pressure below the affected steam generator pressure relief setpoint and reducing the difference between primary and affected steam generator secondary pressure to a value low enough to terminate (or nearly terminate) break flow.

2. During a single steam generator tube rupture, unaffected steam generator steam and feed using ADV flow and auxiliary feedwater flow did not, by itself, reduce the Semiscale system pressure to below the affected steam generator ADV setpoint. During the period of steam and feed operation, the effect of SI overshadowed the steam and feed operation and caused a pressurization of the primary system.

   Termination of SI in combination with unaffected loop feed and bleed had a strong effect in reducing the Semiscale system pressure below affected loop secondary ADV relief setpoint.

3. The vessel collapsed liquid level remained above the hot legs during the entire single steam generator tube rupture transient.

4. RELAP5 pretest calculations agreed qualitatively with test S-SG-1 results. All trends and approximate magnitudes were closely predicted.

Two recommendations are proffered which could help RELAP5 correctly calculate the system pressure during this period of filling. Extensive pressurizer heat loss characterization testing is needed. This testing should develop a heat loss correlation both as a function of liquid level

and of thermal-hydraulic conditions. Second, more metal and fluid thermocouples are needed in the pressurizer for future tube rupture experiments to better characterize the phenomena.

6. REFERENCES

1. Martinez, L. J., Shimeck, D. J., Experiment Operating Specification for the Semiscale Mod-2B Steam Generator Tube Rupture Experiment Series, EGG-SEMI-6285, July 1983.

2. T. K. Larson, J. L. Anderson, and D. J. Shimeck, Scaling Criteria and an Assessment of Semiscale Mod-3 Scaling for Small Break Loss-of-Coolant Transients, EGG-SEMI-5121, March 1980.

3. Zion Nuclear Plant, Final Safety Analysis Report, Commonwealth Edison Company.

4. G. G. Loomis, Kunihisa Soda, Results of the Semiscale Mod-2A Natural Circulation Experiments, NUREG/CR-2335, EGG-2200, September 1982.

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APPENDIX A

COMPARISON OF S-SG-1A AND S-SG-1B

There were only two significant differences between tests S-SG-1A and S-SG-1B. The first was a slight difference in the affected loop steam generator secondary water level. The second was an unaffected loop leaky steam relief valve for S-SG-1A but not for S-SG-1B. Other differences in conditions between the tests were negligible. The initial level in the affected loop steam generator was above the upper differential pressure tap (1117 cm (440 in.)) for both S-SG-1A and S-SG-1B. This corresponds to a liquid mass inventory of at least 188 kg (413 lbm). The specified[a] level was: "as close to 100 kg (220 lbm) as possible." However, the process indicated initial level is not an accurate indication of steam generator inventory under steaming conditions and read low when initial conditions were set. This led to the secondary being overfilled prior to initiation of the transients. The S-SG-1A secondary level was apparently slightly less than S-SG-1B because the affected loop secondary pressure response did not show an immediate increase upon tube rupture as shown on Figure A-1. The evidence that the affected steam generator was nearly full in both tests is that large oscillations in differential pressure across the affected loop main steam line orifice occurred for both S-SG-1A and S-SG-1B as shown on Figure A-2. Large oscillations in differential pressure correspond to a chugging type flow regime of water and steam; therefore, the liquid level was near the top.

The only other difference between S-SG-1A and S-SG-1B was that S-SG-1A had a leaky unaffected loop safety relief valve (SRV). This effectively removed about half the auxiliary feedwater inventory being delivered to the unaffected loop steam generator secondary. Since the leak was into a catch tank the leak rate was measured and the net auxiliary feed rate boundary condition was known. There was no similar leak during S-SG-1B.

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a. Martinez, L. J., Shimeck, D. J., Experiment Operating Specification for the Semiscale Mod-2B Steam Generator Tube Rupture Experiment, EGG-SEMI-6285, July 1983.

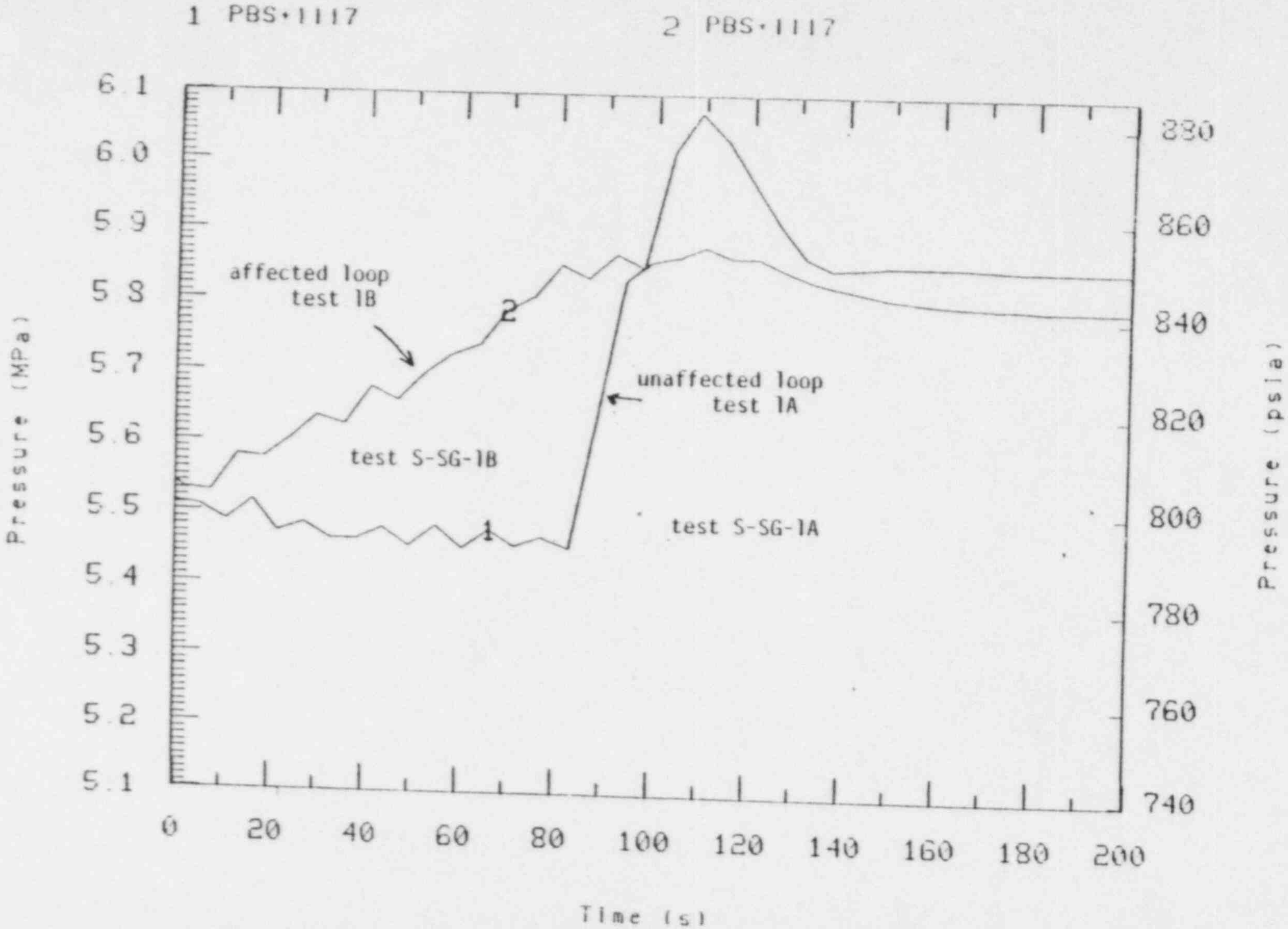

Figure A-1. Comparison of affected loop secondary pressure for S-SG-1A and S-SG-1B.

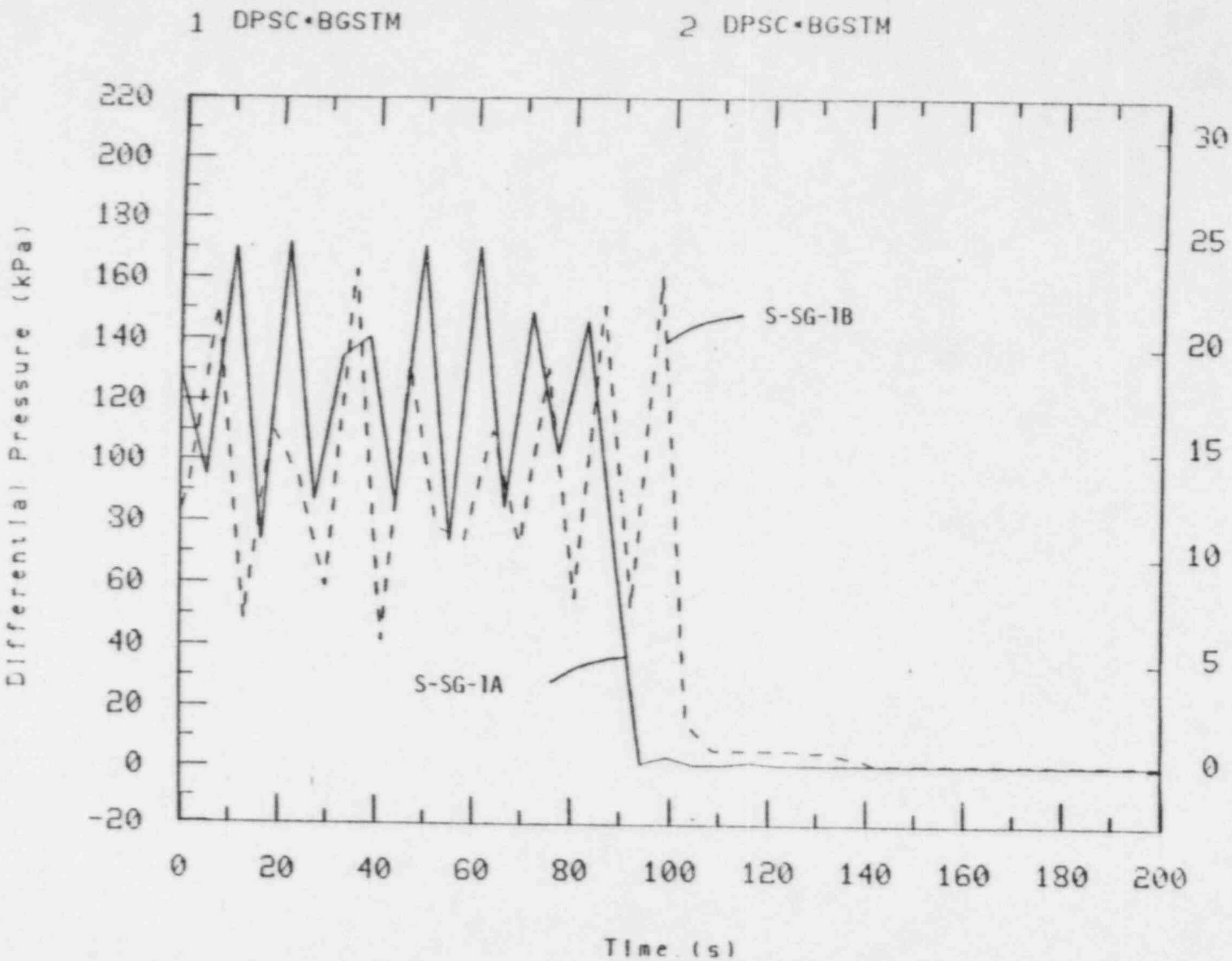

Figure A-2. Comparison of differential pressure across main steam line orifice for S-SG-1A and S-SG-1B.

In summary, test S-SG-1B had no SRV leaks and boundary conditions were as specified. Therefore, test S-SG-1B can be used for code assessment purposes. However, the early-in-time signature response (0-100 s) of S-SG-1B had an atypical affected steam generator response which is better represented by test S-SG-1A.

NRC FORM 335 (11-81) U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET**	1. REPORT NUMBER (Assigned by DDC) EGG-SEMI-6395
4. TITLE AND SUBTITLE QUICK LOOK REPORT FOR SEMISCALE MOD-2B EXPERIMENT S-SG-1	2. (Leave blank)
	3. RECIPIENT'S ACCESSION NO.
7. AUTHOR(S) G. G. Loomis, R. A. Shaw	5. DATE REPORT COMPLETED MONTH September YEAR 1983
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) EG&G Idaho, Inc. Idaho Falls, ID 83415	DATE REPORT ISSUED MONTH September YEAR 1983
	6. (Leave blank)
	8. (Leave blank)
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Division of Accident Evaluation Office of Nuclear Regulatory Research U.S. Nuclear Regulatory Commission Washington, DC 20555	10. PROJECT/TASK/WORK UNIT NO.
	11. FIN NO. A6038
13. TYPE OF REPORT Quick Look Report	PERIOD COVERED (Inclusive dates)
15. SUPPLEMENTARY NOTES	14. (Leave blank)

16. ABSTRACT (200 words or less)

Results of a preliminary analysis of the first test performed in the Semiscale Mod-2B Steam Generator Tube Rupture Series are presented. Test S-SG-1 simulated a pressurized water reactor accident initiated by a double-ended offset shear of one cold side steam generator tube. The transient included an initial 10-minute period during which only automatic plant protection system response to the initiating event occurred. This period was followed by an operator-induced, limited recovery procedure to establish an unaffected steam generator feed and steam condition and later, to terminate safety injection. The test results provide a measured evaluation of the effectiveness of secondary side steam and feed in Semiscale and the effect of the high pressure injection system operation on the Semiscale response to a single steam generator tube rupture.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT	19. SECURITY CLASS (This report)	21. NO. OF PAGES
Unlimited	Unclassified	
	20. SECURITY CLASS (This page) Unclassified	22. PRICE $

NRC FORM 335 (11-81)