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April 11, 1980

Mr. R. E. Tiller, Director
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TRANSMITTAL OF QUICK LOOK REPORT FOR SEMISCALE MOD-3 SMALL BREAK TESTS S-SB-P1, S-SB-P2, and S-SB-P7 - LPL-49-80

Dear Mr. Tiller:

Attached is the quick look report for Semiscale Tests S-SB-P1, S-SB-P2, and S-SB-P7, which were performed February 25, March 6 and 18, 1980, respectively. These tests were run to investigate the influence of pump operation on the overall behavior of the Semiscale system for a 2.5% communicative cold leg break. In Test S-SB-P1, the pumps were tripped early in the transient on a low pressure signal (corresponding to a system pressure of about 12.4 MPa). In Test S-SB-P7, the pumps were left running until a high system void fraction was reached. The pumps were then tripped at a system pressure of 3.08 MPa. Both of these tests were terminated at a system pressure of 1.42 MPa.

In Test S-SB-P2, the initial pump speeds were maintained throughout the transient, but the test was terminated at a system pressure of 4.14 MPa. Test S-SB-P2 was, therefore, identical to the initial portion of the transient for Test S-SB-P7. Data from this test was used to confirm the early system depressurization characteristics in Semiscale with a delayed pump trip.

Scaling distortions in Semiscale preclude a definitive statement about the potential effects of pump operation on the system behavior in a full-sized commercial pressurized water reactor. However, the data obtained from these experiments should be useful in determining the capability of computer codes to predict important phenomena which occurred in the tests. In this way Semiscale data can contribute to the evaluation of codes used to identify appropriate pump operation in the event of a small break loss-of-coolant accident in a full-sized plant.

Very truly yours,

L P Leach

L. P. Leach, Manager
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INTERIM REPORT

QUICK LOOK REPORT

FOR

SEMISCALE MOD-3 SMALL BREAK TESTS S-SB-P1, S-SB-P2, AND S-SB-P7

By

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April 1980

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The information contained in this summary report is preliminary and incomplete. Selected pertinent data are presented in order to draw preliminary conclusions and to expedite the reporting of research results.

ABSTRACT

The results of a preliminary analysis of the data from Semiscale small break Tests S-SB-P1, S-SB-P2, and S-SB-P7 are presented. These tests simulated 2.5% communicative cold leg breaks in a pressurized water reactor (PWR) system, and used initial and boundary conditions approximating those used in PWR audit calculations performed by the Code Assessment and Applications Program. The primary objective of these tests was to evaluate the effect of pump operation on overall Semiscale thermal-hydraulic behavior and system response characteristics. A preliminary analysis of the test data indicates that the pump operation affected break flow and total system mass inventory, resulting in core uncovery with an early pump trip, yet no core uncovery when the pump trip was delayed.

SUMMARY

This report presents the results of a preliminary analysis of the data from Semiscale small break Tests S-SB-P1, S-SB-P2, and S-SB-P7. The primary objective of these tests was to evaluate the effect of pump operation on overall Semiscale thermal-hydraulic behavior and system response characteristics. Of particular interest was the effect of pump operation on total system mass inventory and on system mass distribution. The initial conditions for Tests S-SB-P1, S-SB-P2, and S-SB-P7 were specified to be identical to those for Test S-SB-2, which simulated a 2.5% communicative break in the cold leg of a pressurized water reactor (PWR). The effect of pump operation was investigated by varying the time for tripping the primary coolant pumps. In Test S-SB-P1, the pumps were tripped at a system pressure of about 12.4 MPa, while in Test S-SB-P7, the pumps were tripped at a system pressure of 3.28 MPa. Both of these tests were terminated at a system pressure of 1.42 MPa. In Test S-SB-P2, the pumps were maintained at their initial speeds throughout the transient, but the test was terminated at a system pressure of 4.14 MPa.

Operation of the primary coolant pumps affected the fluid conditions that existed near the break and thus affected the break flow. The break flow in turn affected the system depressurization and primary system coolant inventory. The fluid temperatures at the break were higher in Test S-SB-P7 (delayed pump trip) than in Test S-SB-P1 (early pump trip), prior to 360 s. During this time, the fluid conditions at the break resulted in a higher break flow rate for the early pump trip test because of the fluid subcooling. During this first 360 s, more mass depletion occurred in Test S-SB-P1 than in Test S-SB-P7. At about 360 s, the intact loop seal voided in Test S-SB-P1, which had the early pump trip. The loop seal blowout cleared the liquid from both the intact and broken loop cold legs. In both tests the fluid had become two-phase near the break by 360 s, but more liquid remained in the cold legs in the delayed pump trip test than in the early pump trip test. This lower quality fluid near the break led to a slightly higher break flow for Test S-SB-P7 (delayed pump

trip) than for Test S-SB-P1 (early pump trip) after 360 s. The increased break flow was not enough to make up for the greater mass depletion in Test S-SB-P1 (early pump trip) prior to 360 s. Thus, the system mass remained greater in the delayed pump trip test than in the early pump trip test, throughout the transients. The mass in the vessel and core region also remained higher in the delayed pump trip test than in the early pump trip test. As a result, the core remained covered with adequate cooling in Test S-SB-P7 (delayed pump trip), but uncovered in Test S-SB-P1 (early pump trip), resulting in a cladding temperature excursion beginning at about 600 s. The temperatures turned over in Test S-SB-P7 at about 1050 s, when the vessel mass began increasing due to increasing high pressure injection system flow. The maximum clad temperature during this temperature excursion was 837 K.

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1. INTRODUCTION

This report presents the results of a preliminary analysis of the data from Semiscale small break Tests S-SB-P1, S-SB-P2 and S-SB-P7.[1, 2, 3] The small break test series is being conducted in the Semiscale Mod-3 facility to investigate the thermal-hydraulic phenomena resulting from a small break in the primary system piping, and to evaluate the ability of available computer codes to predict those phenomena.

The Semiscale small break test series is part of the overall water reactor safety research effort directed toward assessing and improving the analytical capability of computer codes to accurately predict the behavior of a pressurized water reactor (PWR) during a postulated small break loss-of-coolant accident (LOCA). This research effort also includes small break tests in the Loss-of-Fluid Test (LOFT) Facility and a series of computer code calculations performed by the Code Assessment and Applications Program as an audit of PWR small break calculations performed by reactor system vendors. The Semiscale small break test series has been designed to be compatible with LOFT small break tests and the PWR audit calculations.

The primary objective of Tests S-SB-P1, S-SB-P2, and S-SB-P7 is to determine the effect of pump operation on overall Semiscale thermal-hydraulic behavior and system response characteristics. Of particular interest is the effect of pump operation on total system mass inventory and on the distribution of system mass as a function of time during the slow system depressurization.

The data obtained from these tests will be used primarily to determine the capability of codes to predict important differences in system behavior caused by differences in pump operation. These tests should therefore provide a basis for assessing the reliability of code predictions used to define appropriate pump operation in the event of a small break LOCA in a full-sized nuclear plant.

The initial conditions for Tests S-SB-P1, S-SB-P2, and S-SB-P7 were specified to be identical to those for Test S-SB-2.[1] These conditions closely approximate those expected in a typical full-sized commercial PWR operating at full load conditions. The break size for these tests was 0.01613 cm^2 which is volume scaled to a 2.5% (11 cm^2) break in the cold leg of a full-sized PWR system. The break was located in the side of the cold leg piping between the pump discharge and the vessel, and was communicative in nature. To provide a direct measure of system mass lost out the break during the transients, the pressure suppression system was modified and a condenser system and catch tank (discussed later) were installed downstream of the break. The total effluent lost out the break during a given transient was determined by weighing the amount of liquid in the tank before and after each test.

Tests S-SB-P1 and S-SB-P7 were run to a system pressure of 1.42 MPa, which corresponds to the pressure at which accumulator injection is initiated in Combustion Engineering plants. In Test S-SB-P1, the intact and broken loop pumps were tripped early in the transient on a low pressure signal (corresponding to a system pressure of about 12.4 MPa). In Test S-SB-P7, the pumps were left running until a high system void fraction was reached. The pumps were then tripped at a system pressure of 3.08 MPa.

In Test S-SB-P2, the initial pump speeds were maintained for the duration of the test, but the experiment was terminated at a system pressure of 4.14 MPa. Test S-SB-P2 was therefore identical to the initial portion of the transient for Test S-SB-P7 where the pumps were tripped at a high system void fraction late in the transient. Since catch tank measurements were made at the termination of each test, a direct measure of total mass lost from the system at two points in the delayed pump trip transient was possible.

As discussed earlier, the combined results from Tests S-SB-P1, S-SB-P2 and S-SB-P7, which are discussed in a subsequent section of this report, will be used primarily as part of an experimental data base for the development and assessment of analytical techniques used to predict the

influence of pump operation on the behavior of a PWR that undergoes a small break LOCA. Because of potential scaling distortions inherent in the Semiscale system, the test results cannot be assumed to be typical of a PWR. The results can, however, be used as a means of identifying phenomena that may be important in a small break LOCA in a PWR system.

2. SYSTEM CONFIGURATION AND TEST OPERATION

2.1 System Configuration

An isometric drawing of the Semiscale Mod-3 system in its configuration for Tests S-SB-P1, S-SB-P2 and S-SB-P7 is shown in Figure 1. In this configuration, the elevations of most system components have been maintained the same as in a full-sized PWR. In particular, the elevation of the intact and broken loop pump suction legs relative to the cold leg centerline is similar to that in a PWR system.

The break nozzle configuration for Tests S-SB-P1, S-SB-P2 and S-SB-P7 is shown in detail in Figure 2. The break nozzle was volume-scaled to a 2.5% break in the cold leg of a PWR, and was modeled after the LOFT 2.5% break nozzle, including the length-to-diameter ratio. For these tests, the rupture discs downstream of the break orifice and instrument spool piece were replaced with a valve. This valve was opened to initiate blowdown and closed to terminate the test.

The pressure suppression tank for these tests was disconnected from the pressure suppression header, and the header was drained through a condenser system supplied with a small catch tank. Figure 3 is a schematic representation showing the condenser system and catch tank arrangement. The catch tank contents were weighed before and after each test. Because of the small volume of the catch tank (relative to the pressure suppression tank volume) this procedure provided a reasonably accurate measurement of the total break mass flow during the tests, and also provided a means of determining the amount of fluid remaining in the primary system at the termination of each test.

Figure 4 shows a plan view of the Mod-3 core including the location of the unpowered rods and the elevation of the cladding thermocouples used in Tests S-SB-P1, S-SB-P2, and S-SB-P7. The elevations are given in centimeters above the bottom of the heated length. The thermocouples are located approximately 0.95 mm beneath the cladding surface. The Mod-3 core

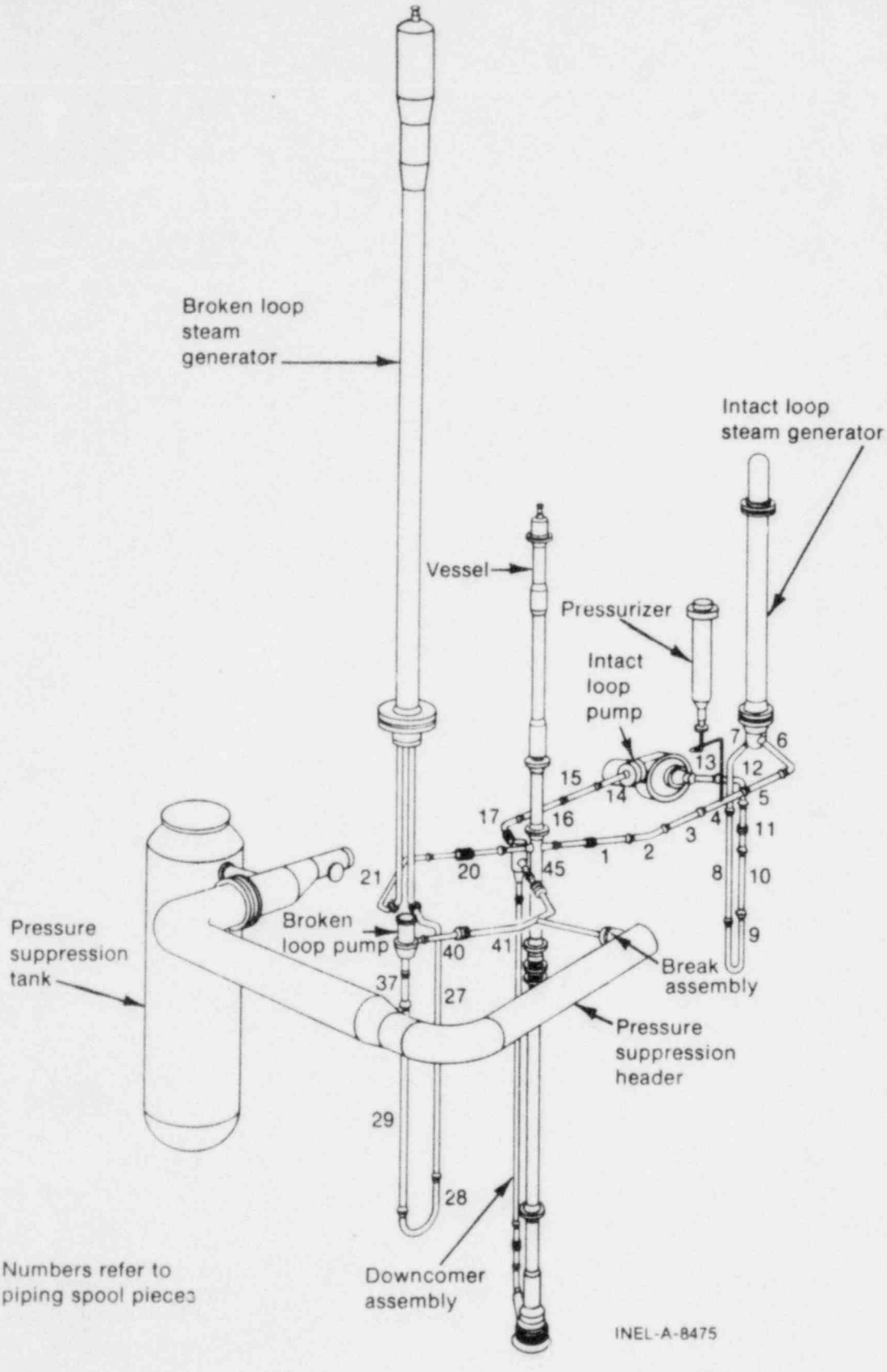


Figure 1. Semiscale Mod-3 system cold leg communicative small break configuration isometric.

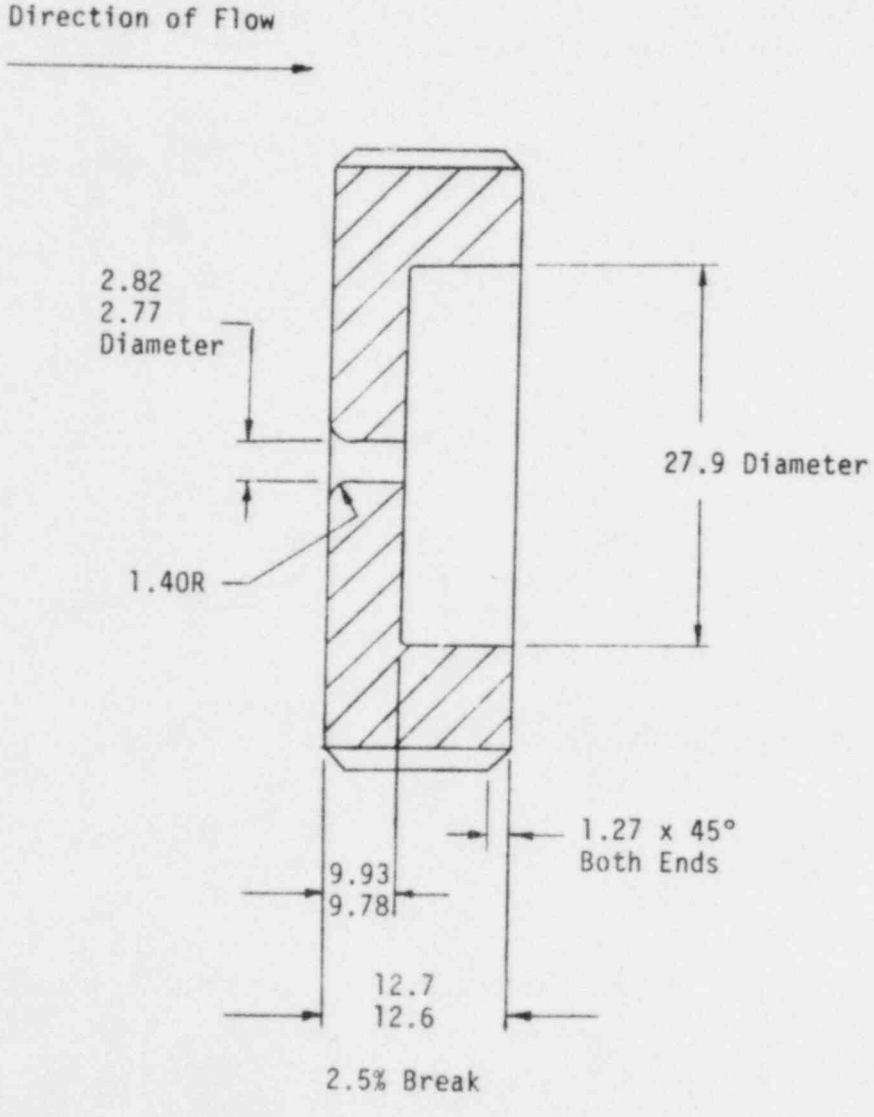
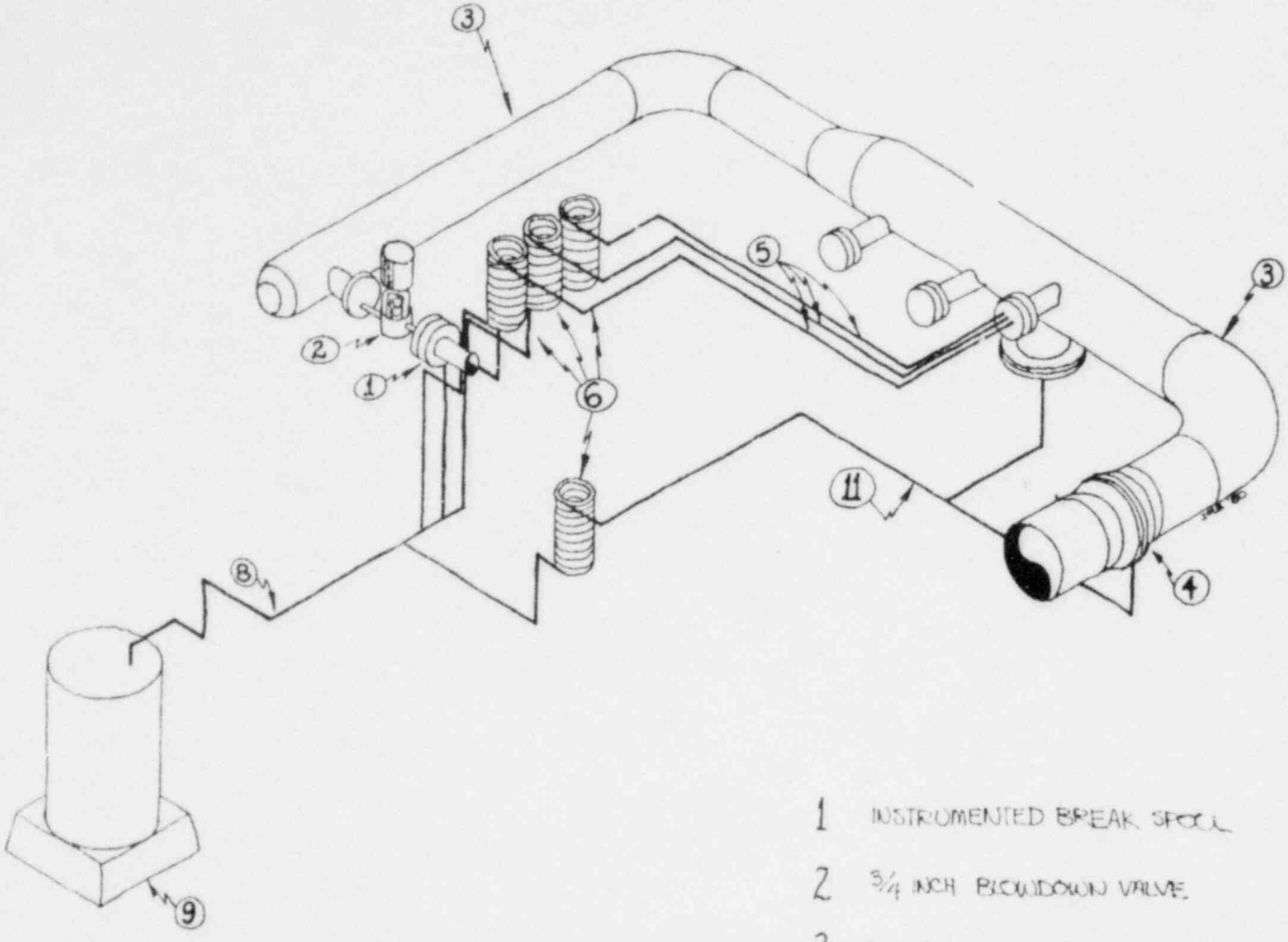


Figure 2. Semiscale 2.5% break nozzle configuration.



1 INSTRUMENTED BREAK SPOOL

2 3/4 INCH BLOWDOWN VALVE

3 BLOWDOWN HEADER

4 BLIND SEAL RING

5 STEAM LINES

6 CONDENSERS

SEMISCALE SYSTEM ULTRA-SMALL BREAK

CONDENSING SYSTEM ALTERNATE C

8 EFFLUENT MEASUREMENT LINE

9 WEIGH TANK

▲ SB-P1 THRU P4

PUMPS ON - PUMPS OFF TEST

11 DRAIN LINES

Figure 3. Condensing system configuration.

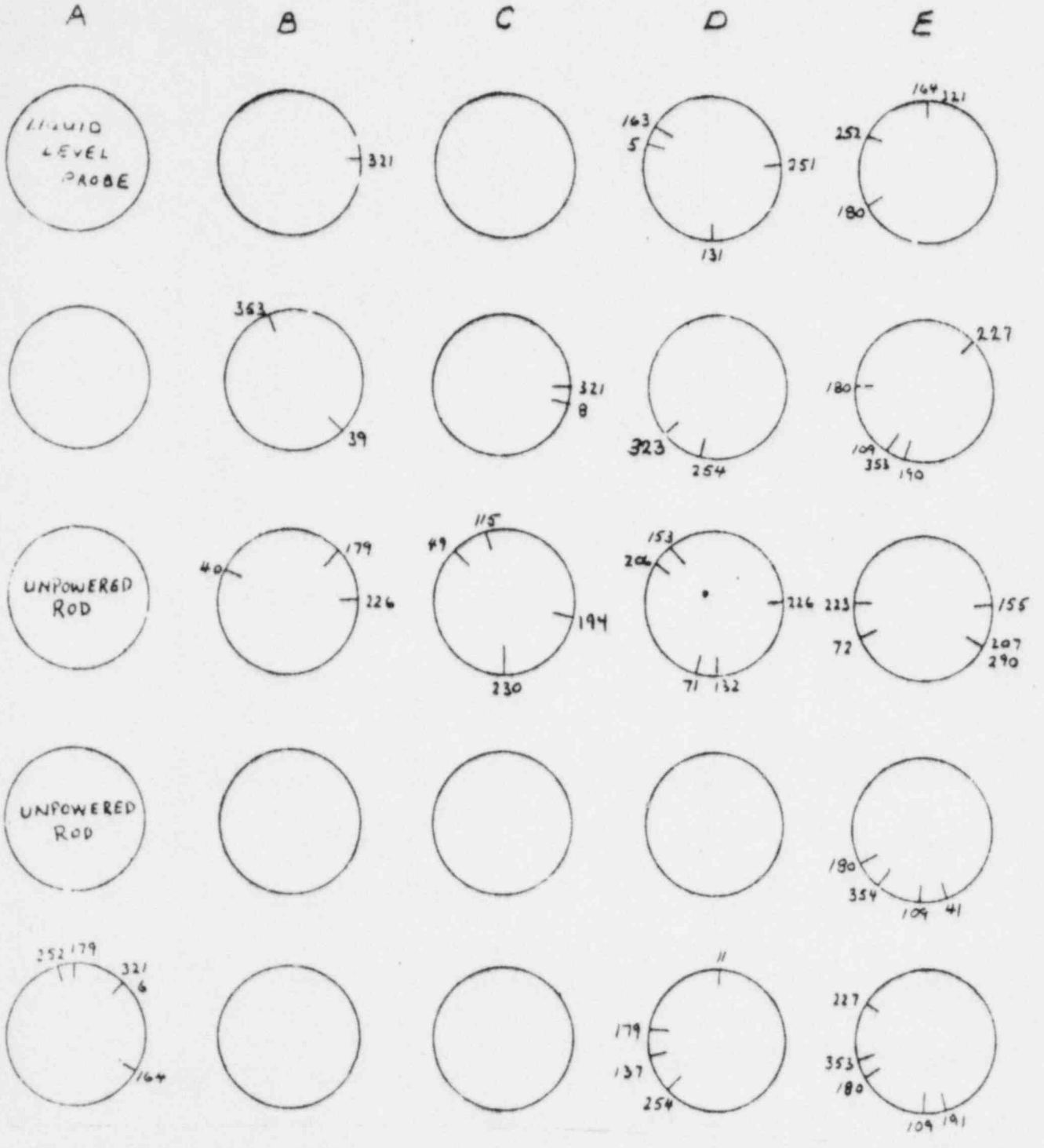


Figure 4. Locations of core rod thermocouples.

is scaled to a PWR, and has full-length (3.66 m) electric heater rods. The axial power profile for the Semiscale Mod-3 heater rods is shown in Figure 5. A flat radial power profile was used for the test. The locations of the in-core densitometers relative to the core axial profile are shown in Figure 6. More detailed information about the Mod-3 system may be found in the Semiscale Mod-3 System Design Description.[4]

2.2 Test Procedure and Conditions

Several hours prior to test initiation, the Semiscale system was filled with demineralized water and vented to ensure a liquid full system. Water in the steam generator feedwater tank was heated to the desired temperature and the required levels were established in the steam generator secondary sides. The instruments were calibrated and zeroed as required and the system was leak checked and hydro checked. The system was brought to initial conditions by a warmup procedure which included measurement of the system leak rate as a function of pressure. In each of the tests, the measured system leak rate was less than 1% of the nominal break flow rate at initial test conditions (15.5 MPa and 550 K) and decreased at lower system pressures. Leak rates of this magnitude should not significantly affect the system response during the tests.

After the system had equilibrated at the initial conditions, each test was initiated by opening the valve downstream of the break orifice. The sequence of events for each test following rupture is shown in Table 1. The test sequence was designed to simulate, as closely as possible, those used in the computer code calculations performed by the Code Assessment and Applications Program as an audit of PWR small break calculations performed by reactor system vendors. Figure 7 shows the core power decay used for Tests S-SB-P1, S-SB-P2, and S-SB-P7. The transient core power control was initiated 3.4 s after the pressurizer pressure reached 12.48 MPa. Core power was controlled to follow a post-trip power decay to the ANS decay heat standard +20%. The steam generator feedwater valves were closed about 5 s later. Ambient temperature high pressure injection flow into both the intact and broken loop (spool pieces 17 and 40) was begun about 28.4 s

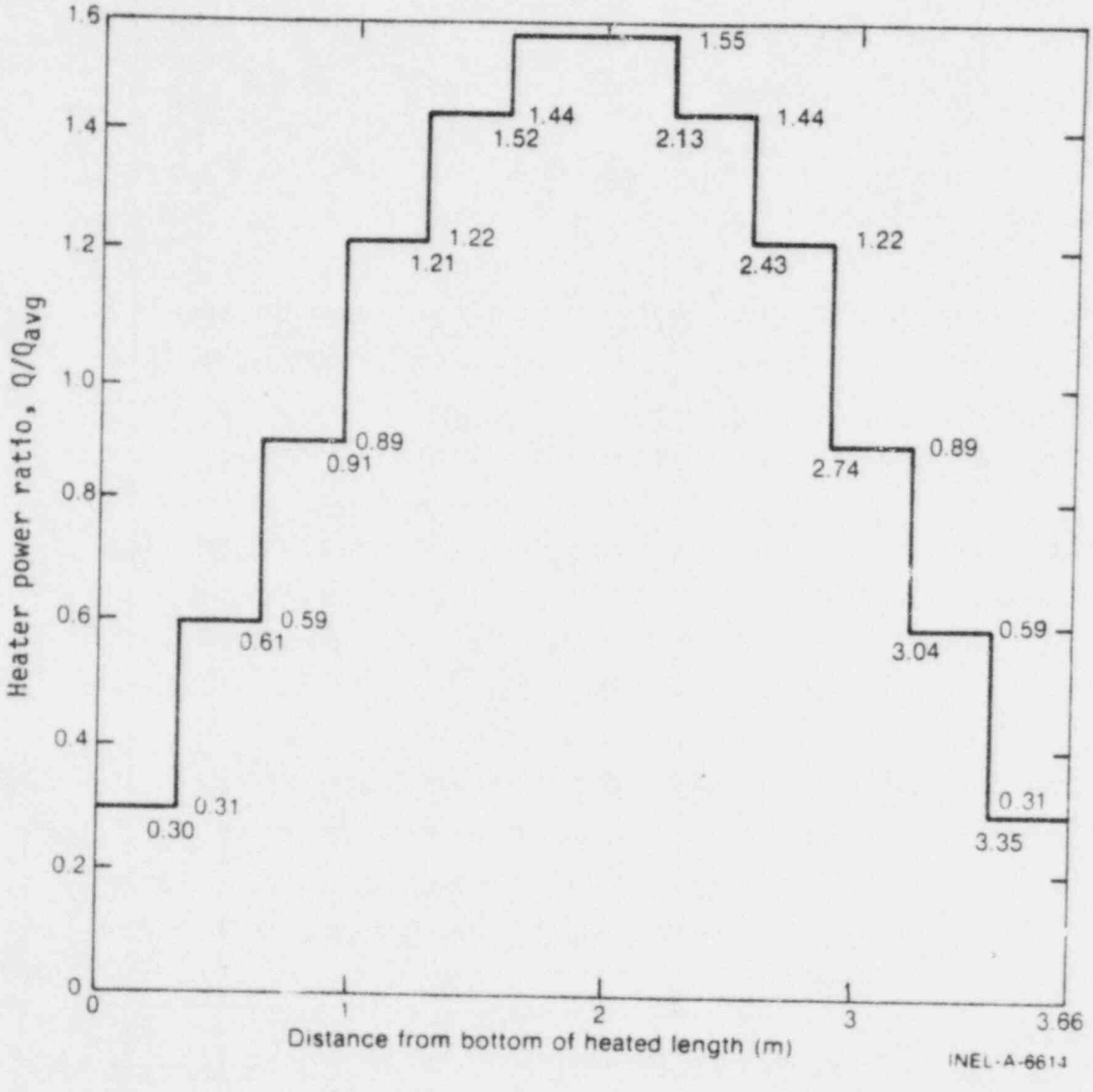


Figure 5. Semiscale Mod-3 heater rod axial power distribution.

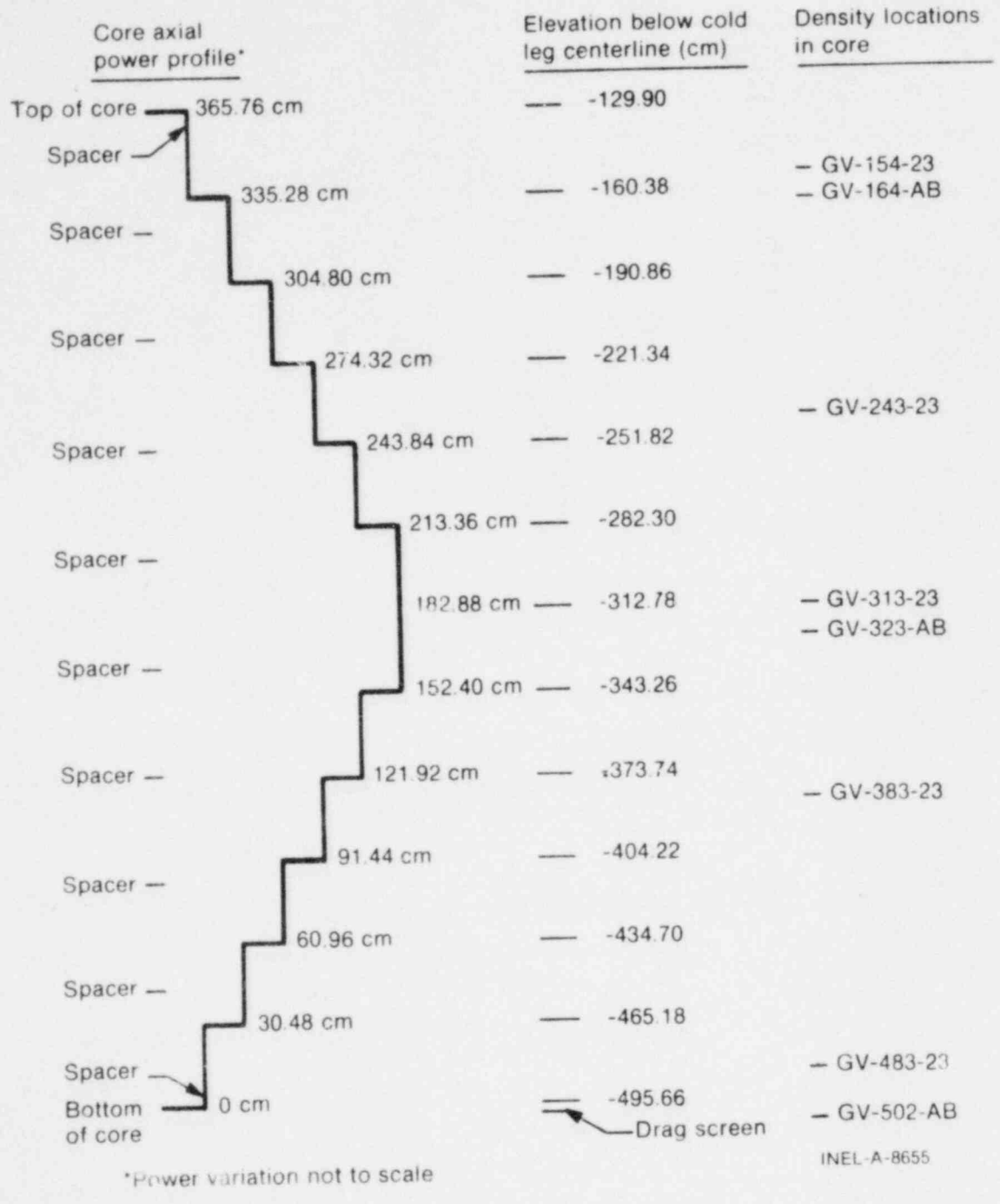


Figure 6. Axial power profile in relation to vessel instrumentation.

TABLE 1. SEQUENCE OF EVENTS

EVENT	SPECIFIED TIME(S)	ACTUAL TIME(S)		
		S-SB-P1	S-SB-P2	S-SB-P7
Rupture	0.0	0.0	0.0	0.0
Steam valves begin closing	0.0 s after pressurizer pressure reaches 12.48 MPa	16.8	16.8	16.8
Core power decay initiated	3.4 s after pressurizer pressure reaches 12.48 MPa	20.2	20.2	20.2
Pump coastdown begins				
S-SB-P1	0.0 s after SCRAM	20.2	N/A	N/A
S-SB-P7	0.0 s after pressurizer pressure reaches 3.28 MPa	N/A	N/A	1114
Steam generator feedwater valves begin closing	5.0 s after SCRAM	26.0	26.0	26.0
HPIS initiated	25.0 s after SCRAM	46.0	46.0	46.0

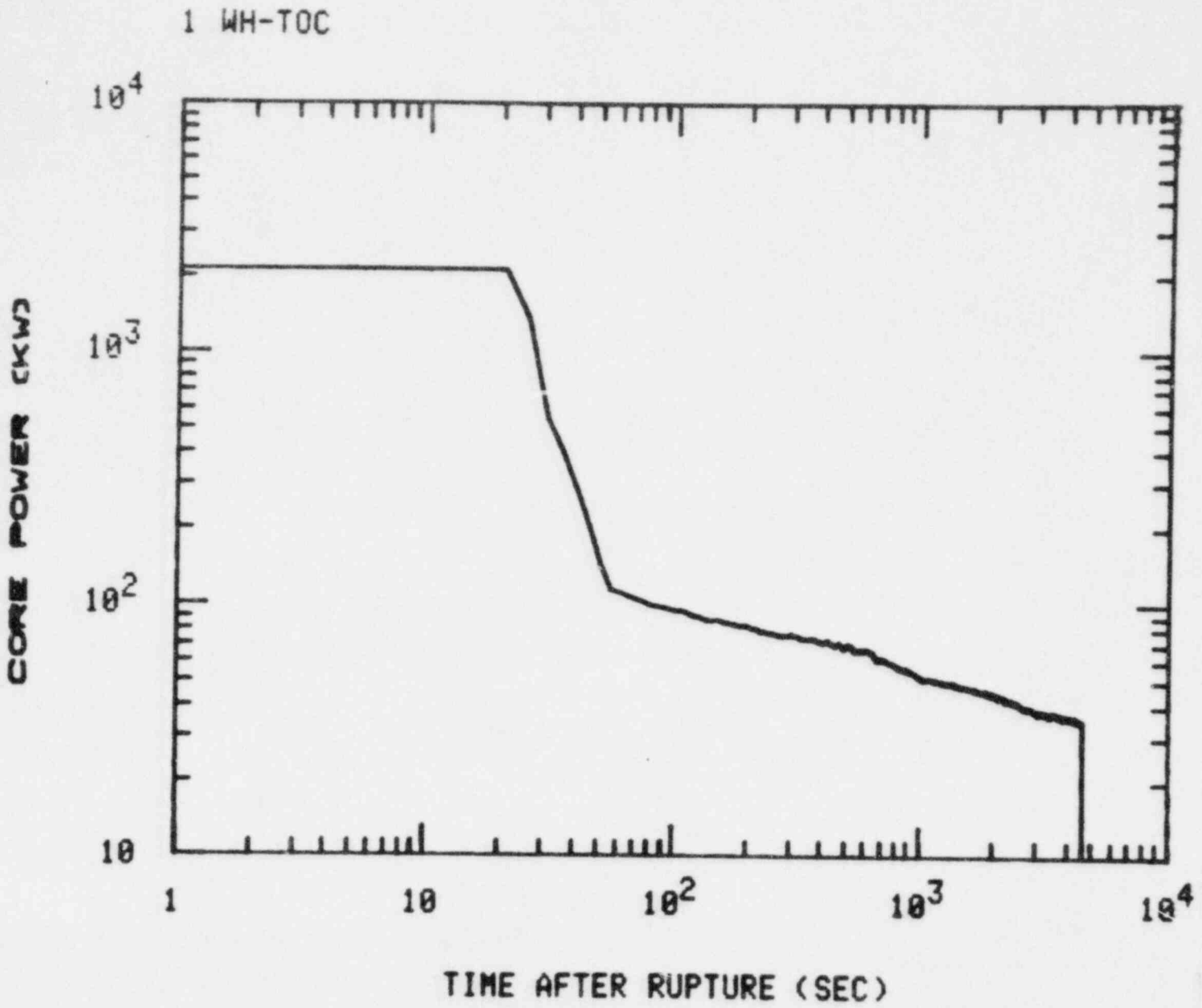


Figure 7. Core power decay.

after the pressure reached 12.48 MPa. Pumped emergency core coolant (ECC) (simulating high pressure injection) was controlled by the on-line computer to follow the flow versus pressure curves shown in Figure 8. The high pressure injection system (HPIS) flow was scaled to represent single train HPIS flow in a PWR. The system pressure was monitored by the computer and was used to calculate the required flow rate. The accumulators and low pressure injection systems were valved out of the primary system to prevent injection from these systems during the test, and to minimize uncertainties in the calculated system mass inventory.

The specified and actual test conditions for Tests S-SB-P1, S-SB-P2, and S-SB-P7 are listed in Table 2. In general, the actual conditions in each test showed reasonably good agreement with the specified values. Operating problems experienced during each of the tests also appeared to be minor. In Test S-SB-P1, however, some problems with the operation of the broken loop pump and with the intact loop steam generator did occur. In this test, the broken loop pump did not follow the pre-programmed coastdown curve. As a result, the pump remained at full speed for about 60 s into the coastdown, at which time the pump power was manually tripped, causing the pump speed to rapidly drop to zero at the prescribed time. Difficulties in controlling the auxiliary feedwater flow in this test also caused some atypical fluctuations in the intact loop steam generator secondary liquid level. However, neither of these operating problems appears to have significantly affected the overall system behavior in Test S-SB-P1. The only other problem observed in the three tests was a short delay in the time at which the controller initiated the pump coastdown in Test S-SB-P7. As a result, the pump coastdown was initiated at a system pressure of 3.08 MPa instead of the specified 3.28 MPa. Since the pump trip still occurred close to the maximum system void fraction in this test, the delay in the start of pump coastdown was not felt to have significantly affected the test results. No other significant operating problems were disclosed in the preliminary review of the data, and therefore, the three tests were judged to be acceptable.

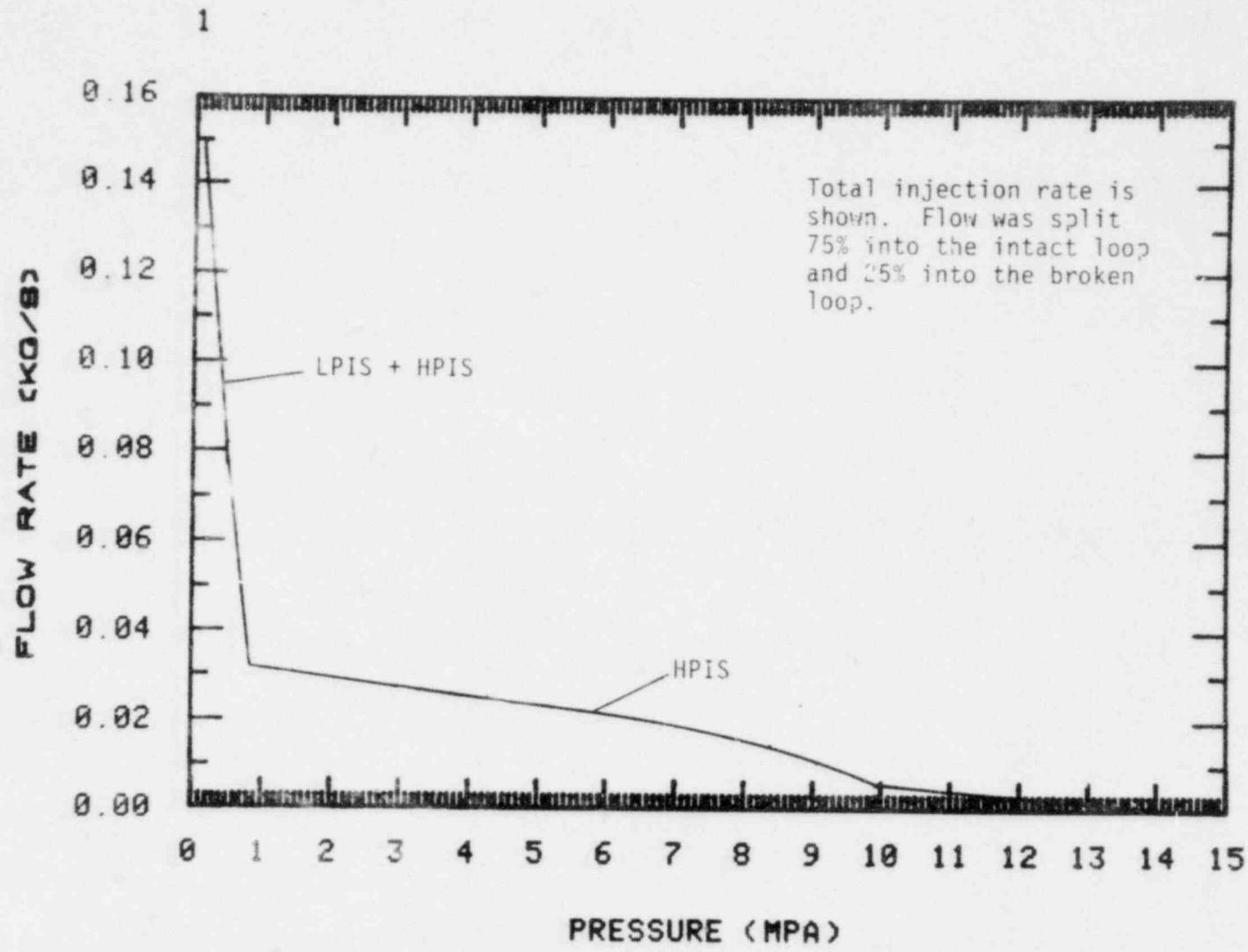


Figure 8. High pressure injection system control for intact and broken loop.

TABLE 2. INITIAL CONDITIONS AND OPERATIONAL REQUIREMENTS

CONDITIONS	SPECIFIED	MEASURED		
		S-SB-P1	S-SB-P2	S-SB-P7
Pressurizer pressure (MPa)	15.5	15.5	15.5	15.7
Hot leg fluid temperature (K)	583	585	583	584
Cold leg fluid temperature (K)	550	550	550	550
Core inlet flow rate (nominal-kg/s)	11.7	12.1	12.4	11.9
Total core power (MW)	1.97	1.96	1.97	1.97
Radial power profile	Flat	Flat	Flat	Flat
Steam generator secondary pressure (MPa)				
Intact loop	5.86	5.50	5.48	5.46
Broken loop	5.86	5.24	5.12	5.09
Steam generator feedwater temperature (K)	495	488	487	486
Auxiliary feedwater temperature	Ambient	Ambient	Ambient	Ambient
Steam generator liquid level				
Intact loop (DI-SG-LL)(KPad)	11	9	10	10
Broken loop (DB-SG-LL)(KPad)	33	35	29	28

The results of a preliminary analysis of data from Tests S-SB-P1, S-SB-P2, and S-SB-P7 are presented in the following section. The data from Test S-SB-P1 is used to describe the Semiscale system behavior when the pumps are tripped early in the transient. Since Test S-SB-P2 was specified to be identical to the initial portion of the transient for Test S-SB-P7, the combined results from Tests S-SB-P2 and S-SB-P7 are used to describe system behavior caused by a delayed pump trip at a high system void fraction. Included in this section are a discussion of overall system thermal-hydraulic response, an interpretation of the effect of pump operation on system mass inventory and distribution, and the resulting effect on core thermal response.

3. TEST RESULTS

A comparison of the upper plenum pressure response between the early and delayed pump trip tests is shown in Figure 9. The system fluid was subcooled in both tests prior to about 35 s, resulting in relatively rapid depressurizations that were nearly identical. At about 35 s, the fluid temperatures at the intact loop hot leg reached saturation in both tests, as shown in Figure 10. Once saturation occurred, flashing of fluid in the system caused the decrease in system depressurization rates for both tests shown in Figure 9. Circulation of fluid by the pumps caused the broken loop cold leg temperatures to be higher in Test S-SB-P7, with the delayed pump trip, than in Test S-SB-P1, with the early pump trip, between about 90 and 360 s (see Figure 11). As a result, although the fluid at the break remained subcooled in both tests, the enthalpy of the fluid exiting the break was higher in Test S-SB-P7, with the delayed pump trip, than in Test S-SB-P1, with the early pump trip. With more energy leaving the system through the break, Figure 9 shows the delayed pump trip resulted in a slightly faster system depressurization between about 90 and 360 s than occurred when the pumps were tripped early. In Test S-SB-P1, with the early pump trip, the downflow side of the pump suction began voiding at about 170 s, forming the loop seal. The liquid level reached the bottom of the pump suction at about 360 s, as shown in Figure 12, and the loop seal was blown out. The liquid was cleared from the cold legs when the pump suction voided, resulting in a two-phase mixture upstream of the break as shown in Figure 13. The higher quality fluid leaving the break at this time resulted in an increased depressurization rate at about 360 s into Test S-SB-P1 (early pump trip) as shown in Figure 9.

The primary system coolant inventories during Test S-SB-P1, with the early pump trip, and Test S-SB-P7, with the delayed pump trip, are shown in Figure 14. The coolant mass depletion was more rapid with the early pump trip than with the delayed pump trip during the first 360 s because of fluid conditions upstream of the break. As discussed previously, the fluid in the cold legs was more subcooled in Test S-SB-P1 (early pump trip) than

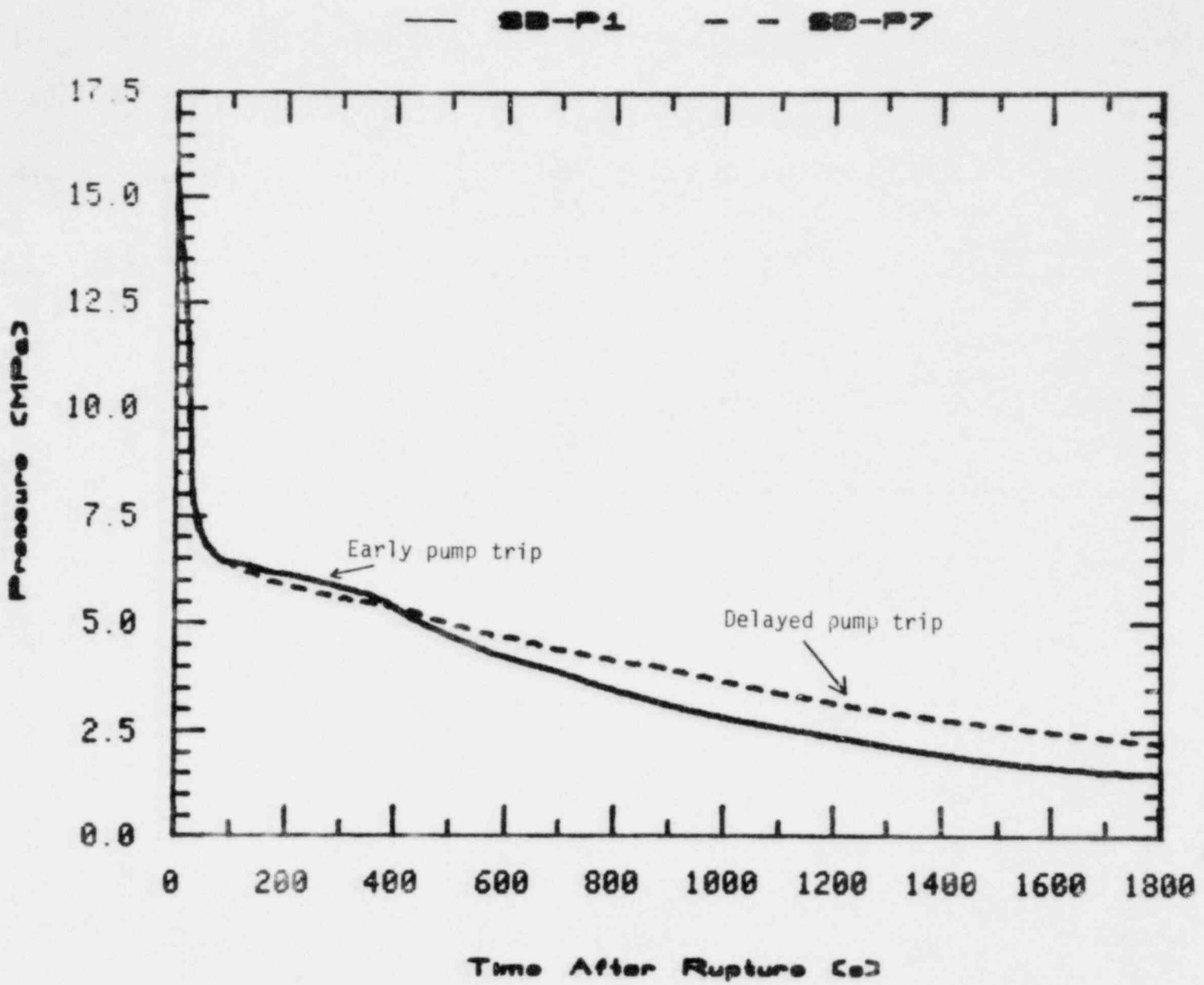


Figure 9. Upper plenum pressure response for Tests S-SB-P1 and S-SB-P7.

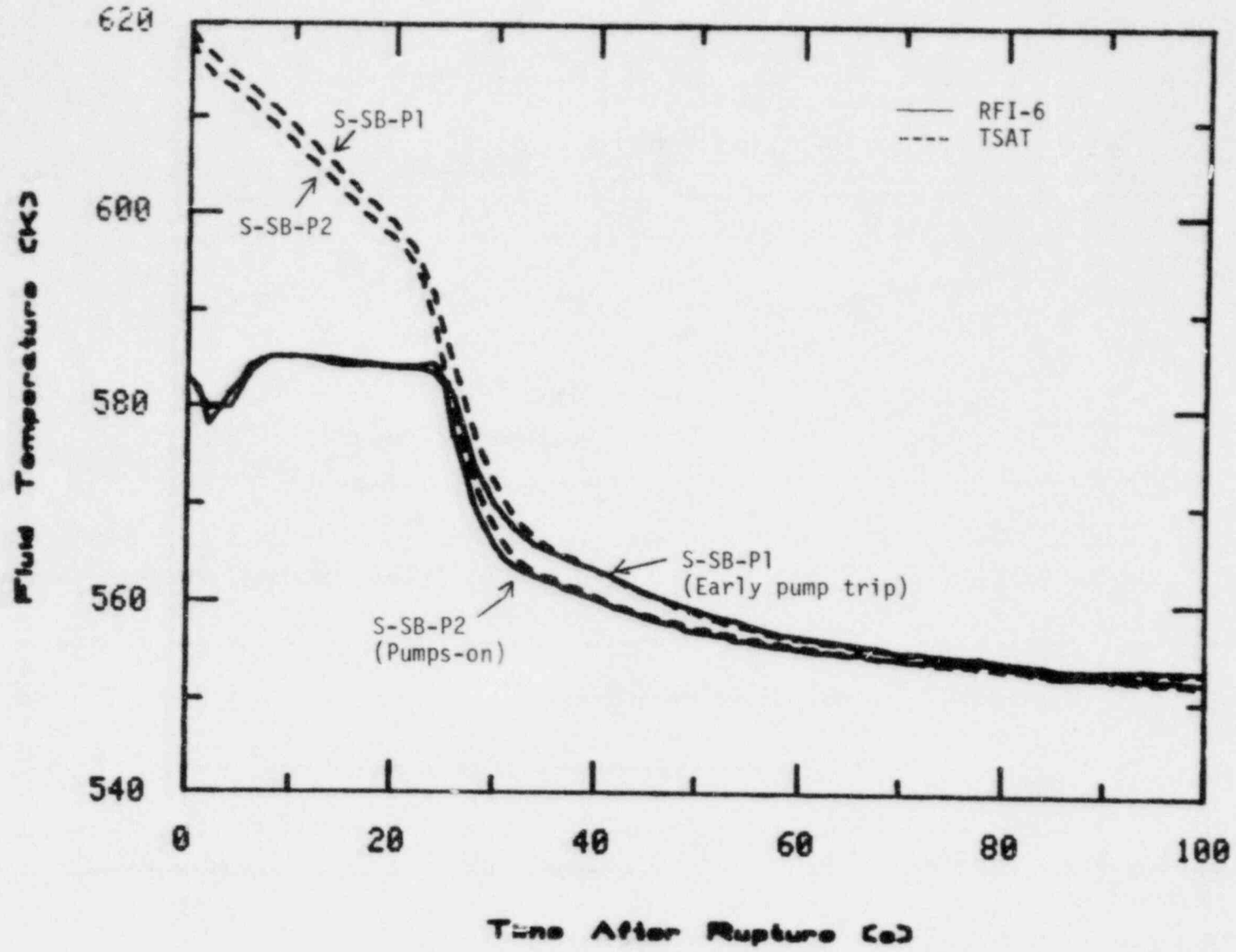


Figure 10. Comparison of hot leg fluid temperature with system saturation temperature for Tests S-SB-P1 and S-SB-P2.

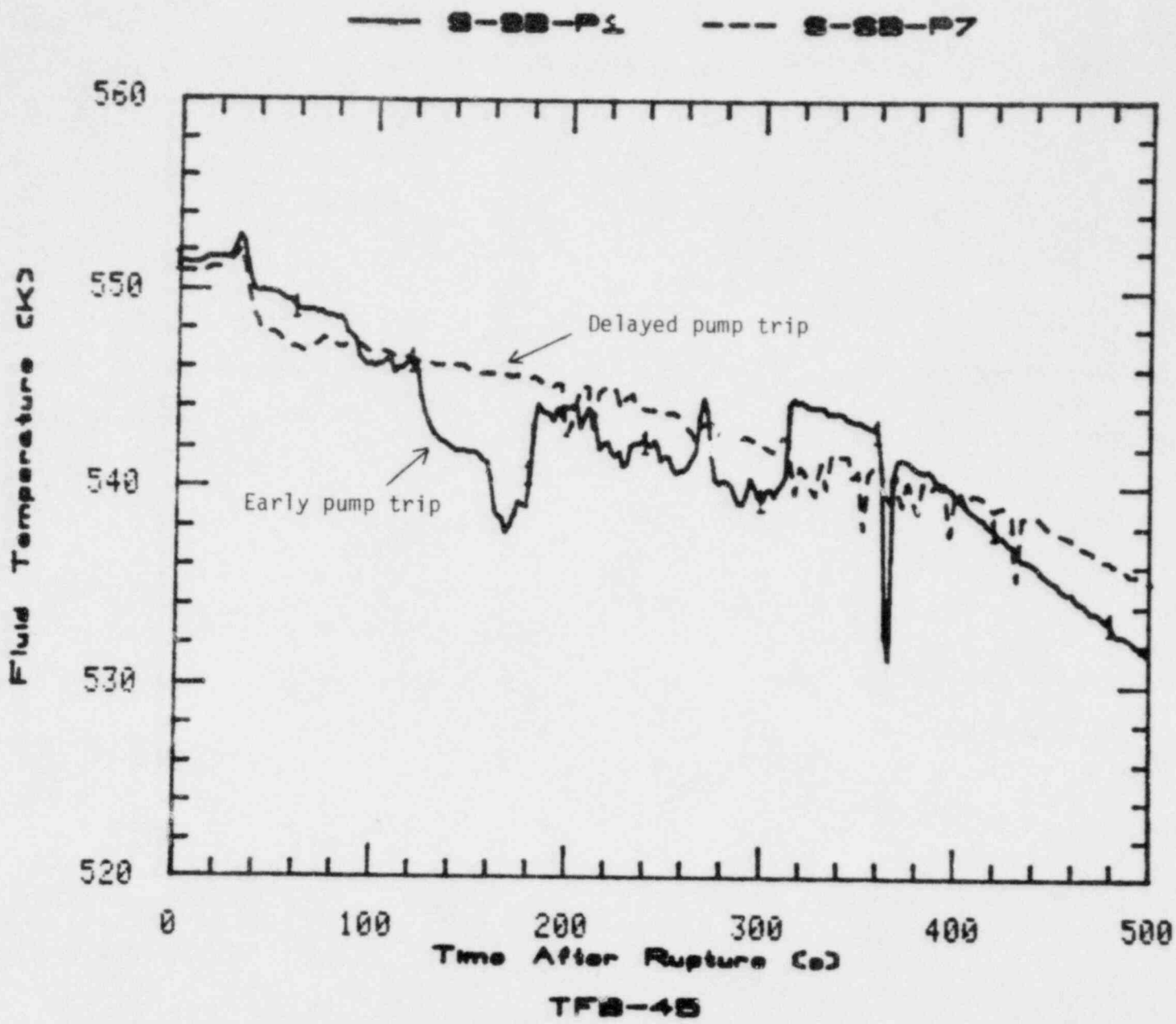


Figure 11. Comparison of fluid temperatures near the break for Tests S-SB-P1 and S-SB-P7.

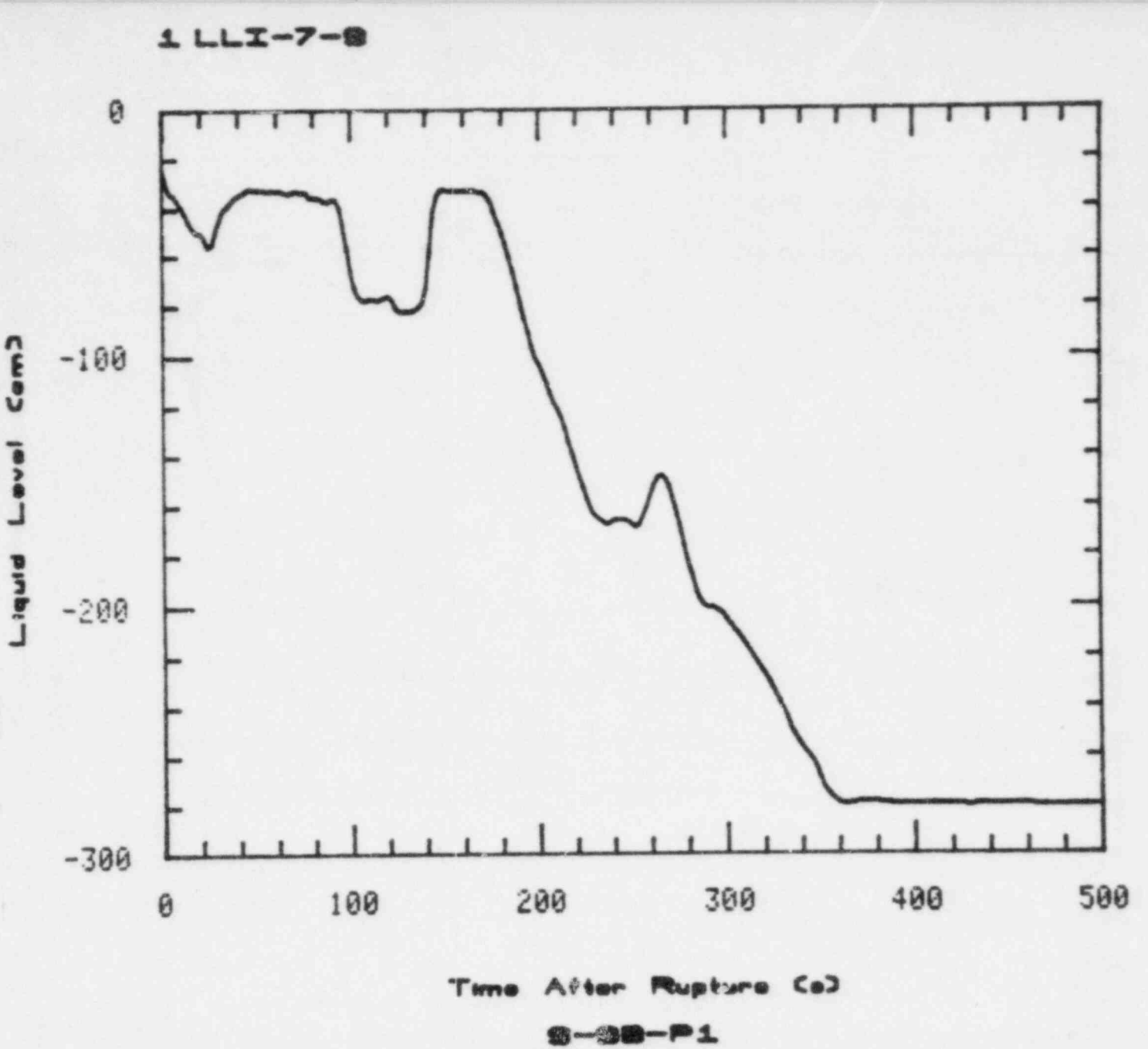


Figure 12. Collapsed liquid level in intact loop pump suction for Test S-SB-P1.

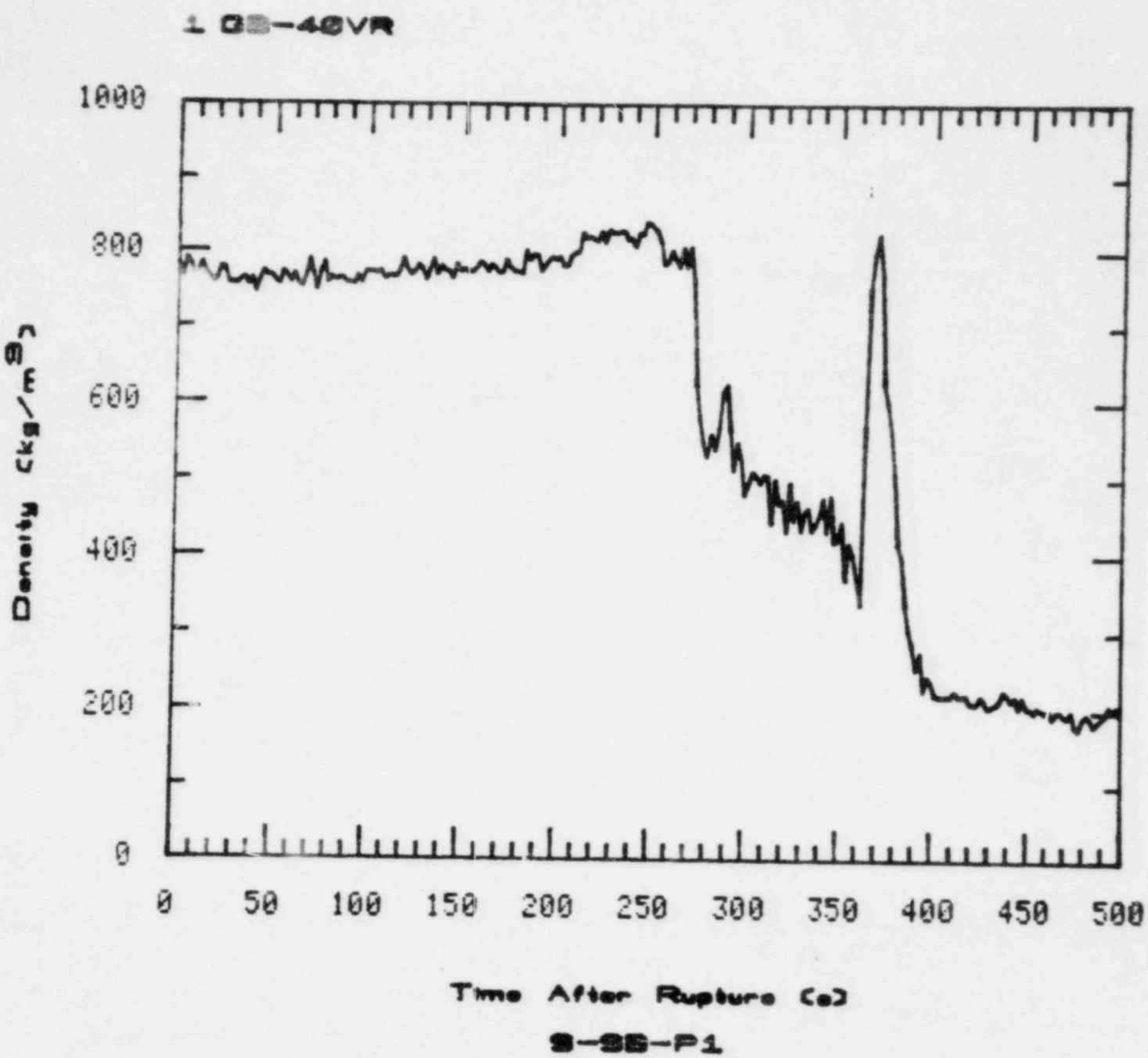


Figure 13. Broken loop cold leg density for Test S-SB-P1.

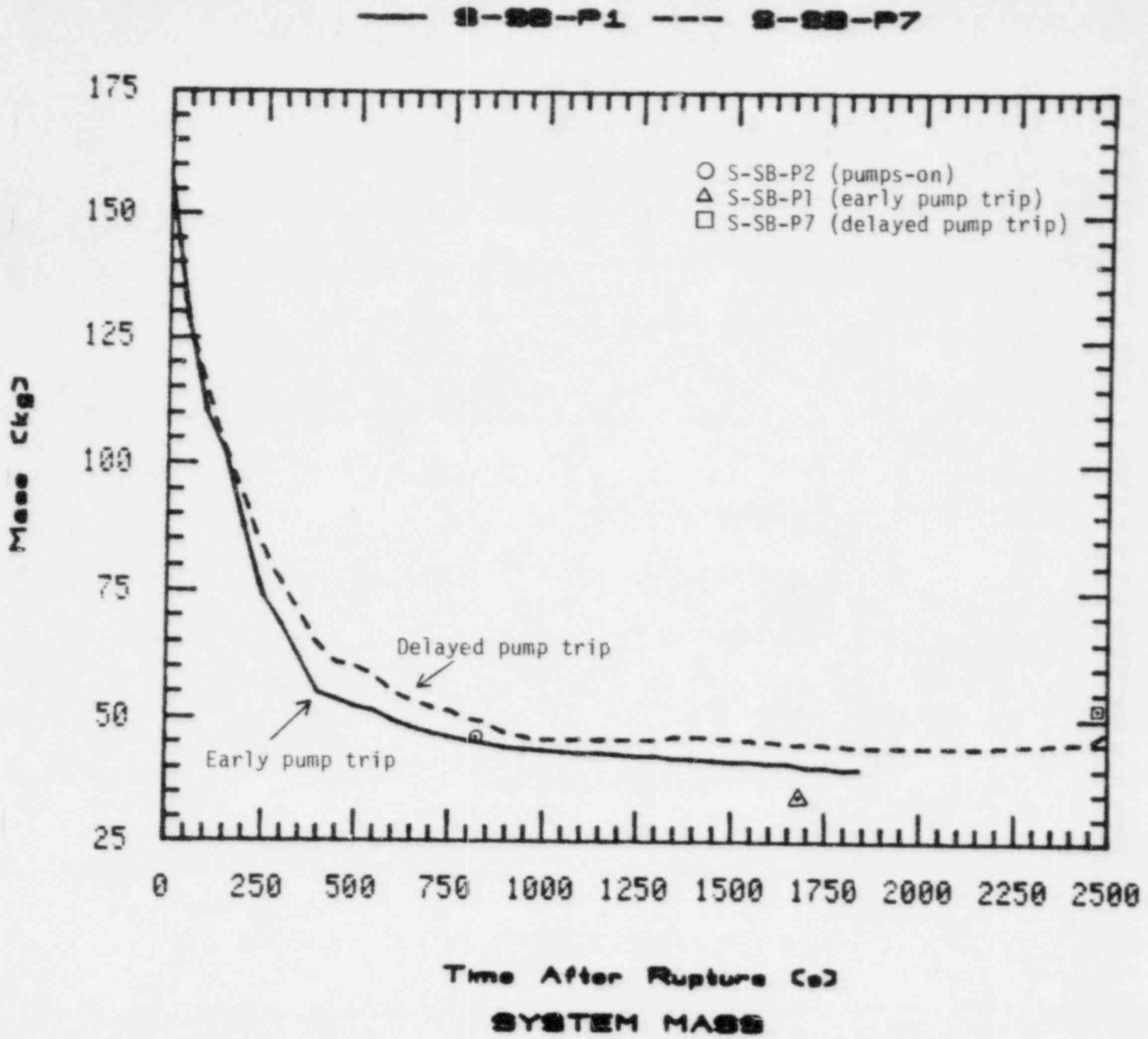


Figure 14. Comparison of total system mass for Tests S-SB-P1 and S-SB-P7.

in Test S-SB-P7 (delayed pump trip). This subcooling resulted in a higher break flow rate with the early pump trip than occurred when the pump trip was delayed, during the first 360 s of the tests. As discussed previously, the intact and broken loop cold legs were voided when the loop seal blew out in Test S-SB-P1 (early pump trip). At about 200 s into Test S-SB-P7 (delayed pump trip), the cold leg fluid reached saturation as shown in Figure 15, resulting in a decreased break flow rate. The pump heads degraded about this time and were thus unable to pump a steady flow of fluid. Rather, the pump suctions were alternately filled and voided, and slugs of liquid were pushed through the pumps. This slugging of fluid is indicated by the oscillations in the broken loop cold leg density shown in Figure 15. This slugging phenomena was also observed on video tapes taken during the test at the pump discharge. More liquid thus remained in the cold legs in the delayed pump trip test than in the early pump trip test, resulting in a slightly higher break flow rate for Test S-SB-P7 (delayed pump trip) after 360 s. As shown in Figure 14, the difference in primary system mass between the two tests began to decrease after 360 s. The relative increase in break flow in Test S-SB-P7 was not enough, however, to make up for the greater mass depletion in Test S-SB-P1 (early pump trip) prior to 360 s. Thus, the primary system mass was always greater in Test S-SB-P7 than in Test S-SB-P1.

The primary coolant system inventories shown in Figure 14 were calculated using fluid density and differential pressure measurements. The masses were calculated for individual regions of the systems, then summed to obtain the total primary coolant system mass. This procedure could produce some errors due to uncertainties in the interpretation of instrument readings. However, the system mass at the end of each test, calculated using the catch tank mass values as shown in Figure 14, indicate that although there may be errors in the calculated transient system mass, the qualitative information is correct.

Figure 16 shows the fraction of total primary coolant system mass contained in the vessel for Test S-SB-P1, with the early pump trip, and Test S-SB-P7, with the delayed pump trip. Early in Test S-SB-P7, the pumps

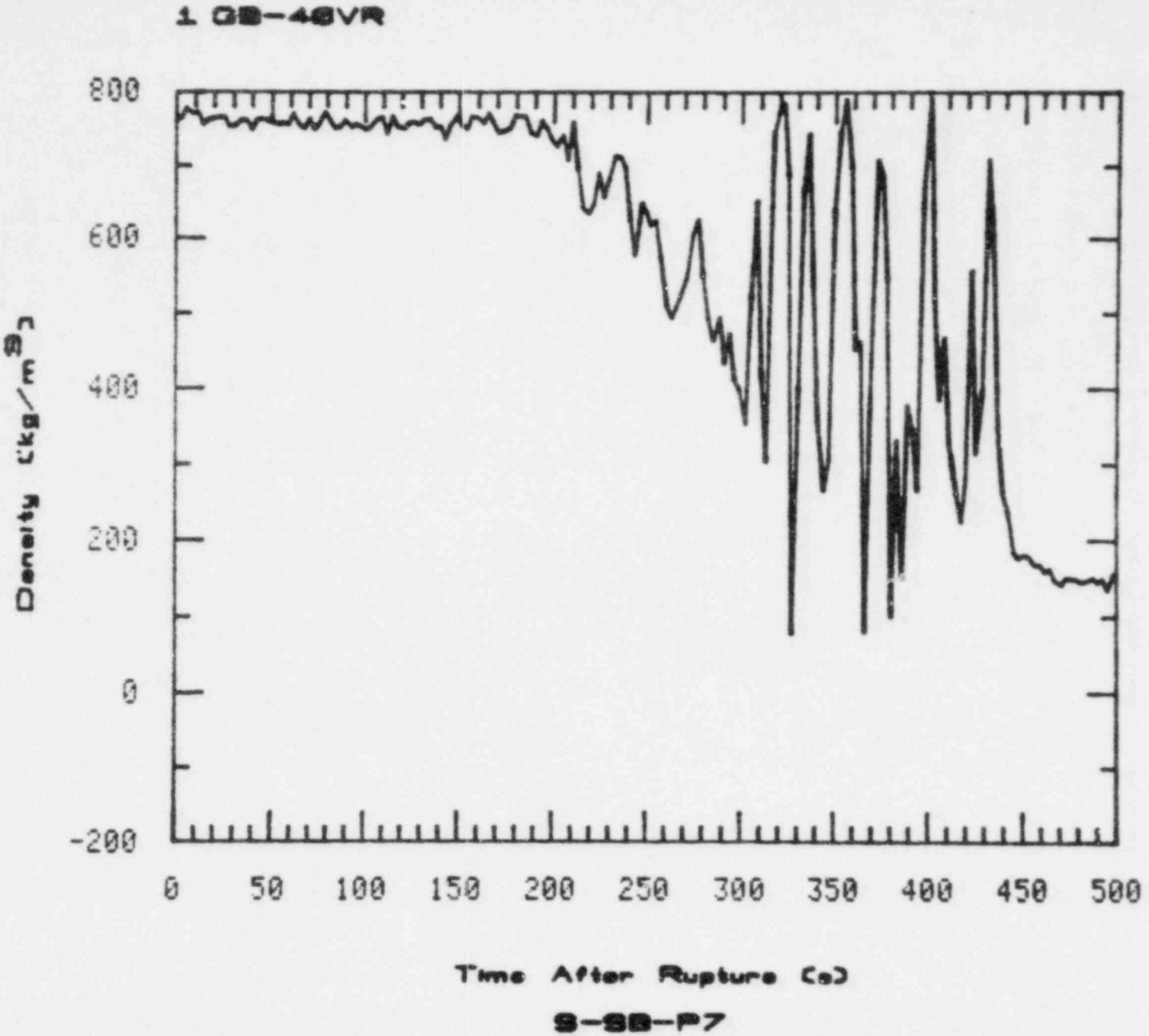


Figure 15. Broken loop cold leg density for Test S-SB-P7.

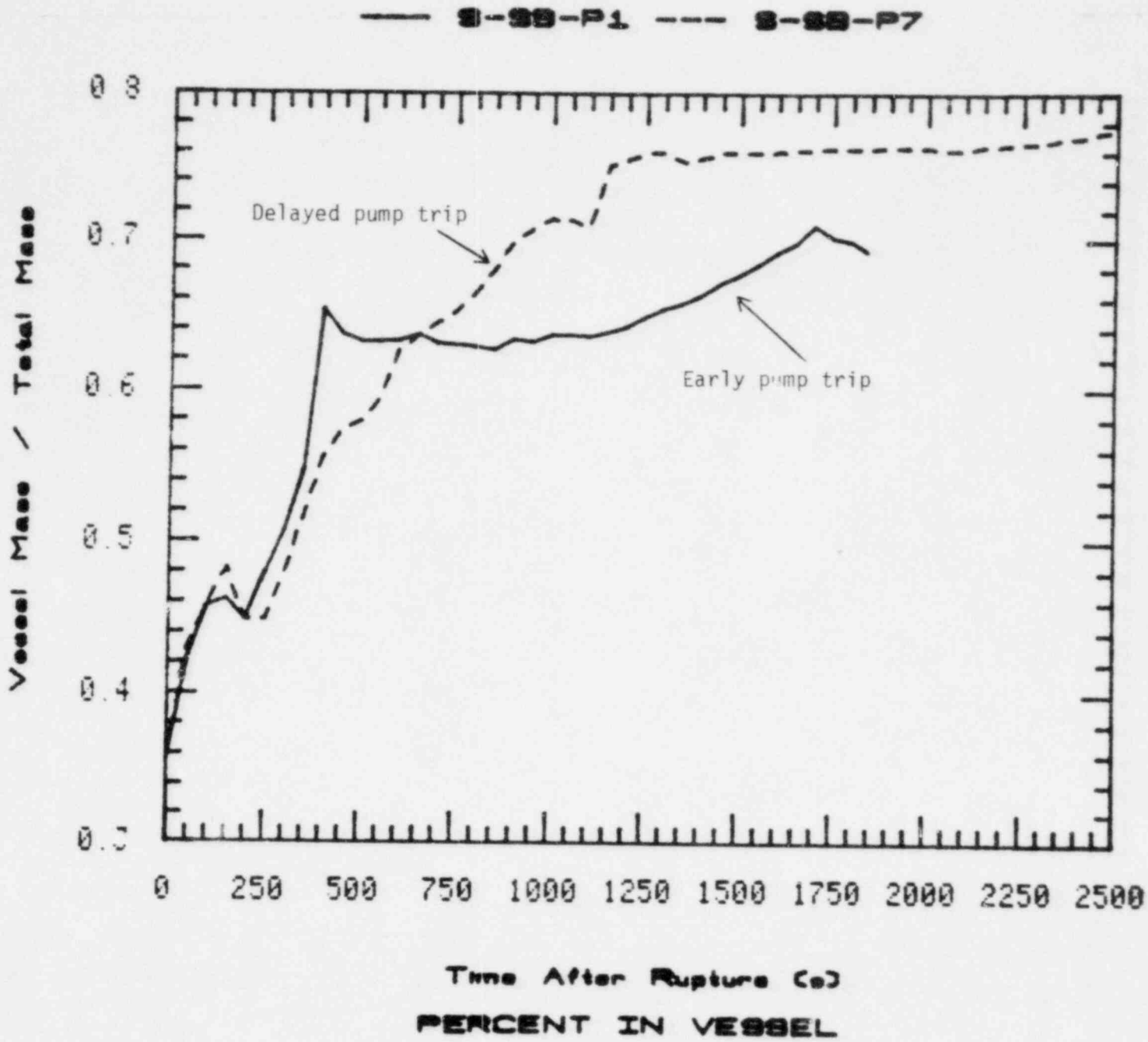


Figure 16. Comparison of the percent of total system mass in vessel for Tests S-SB-P1 and S-SB-P7.

circulated fluid throughout the system, such that the fraction of the system mass contained in the vessel was lower than in Test S-SB-P1 (early pump trip). However, operation of the pumps caused the downcomer level to depress in Test S-SB-P7, as shown in Figure 17. This pushed more of the vessel fluid into the core region, providing greater potential core cooling effectiveness. As the pump head degraded in Test S-SB-P7, less fluid was circulated through the loops and the vessel mass began increasing relative to the mass in the loops. By about 650 s, the fraction of fluid in the vessel became higher for the delayed pump trip test than for the early pump trip test. A significant amount of liquid also collected in the hot legs when the pumps were left running. When the pumps were tripped in Test S-SB-P7, at about 1140 s, the downcomer and core levels equilibrated, and a relatively large mass of fluid fell back into the vessel from the hot legs and steam generators. This additional mass delivered to the vessel region increased the potential for core cooling at the time the delayed pump trip occurred. The resultant vessel mass, shown in Figure 18, remained higher for the delayed pump trip test, than for the early pump trip test, throughout the transient.

The core remained covered throughout the delayed pump trip test, and therefore, adequate core cooling was provided. However, the vessel mass decreased enough in Test S-SB-P1 (early pump trip) such that the core uncovered at about 600 s, as indicated by the core mixture level shown in Figure 19. This resulted in the cladding temperature excursions shown in Figure 20. The core mixture level continued to decrease until about 1050 s, at which time almost half the core was uncovered. The core mixture level then began increasing as shown in Figure 19. This level increase provided enough core cooling to cause the turnover of cladding temperatures shown in Figure 20. The increase in core liquid level was due in part to increasing system mass due to the HPIS flow rate exceeding the break flow rate and in part to liquid fallback from the steam generators and hot legs.

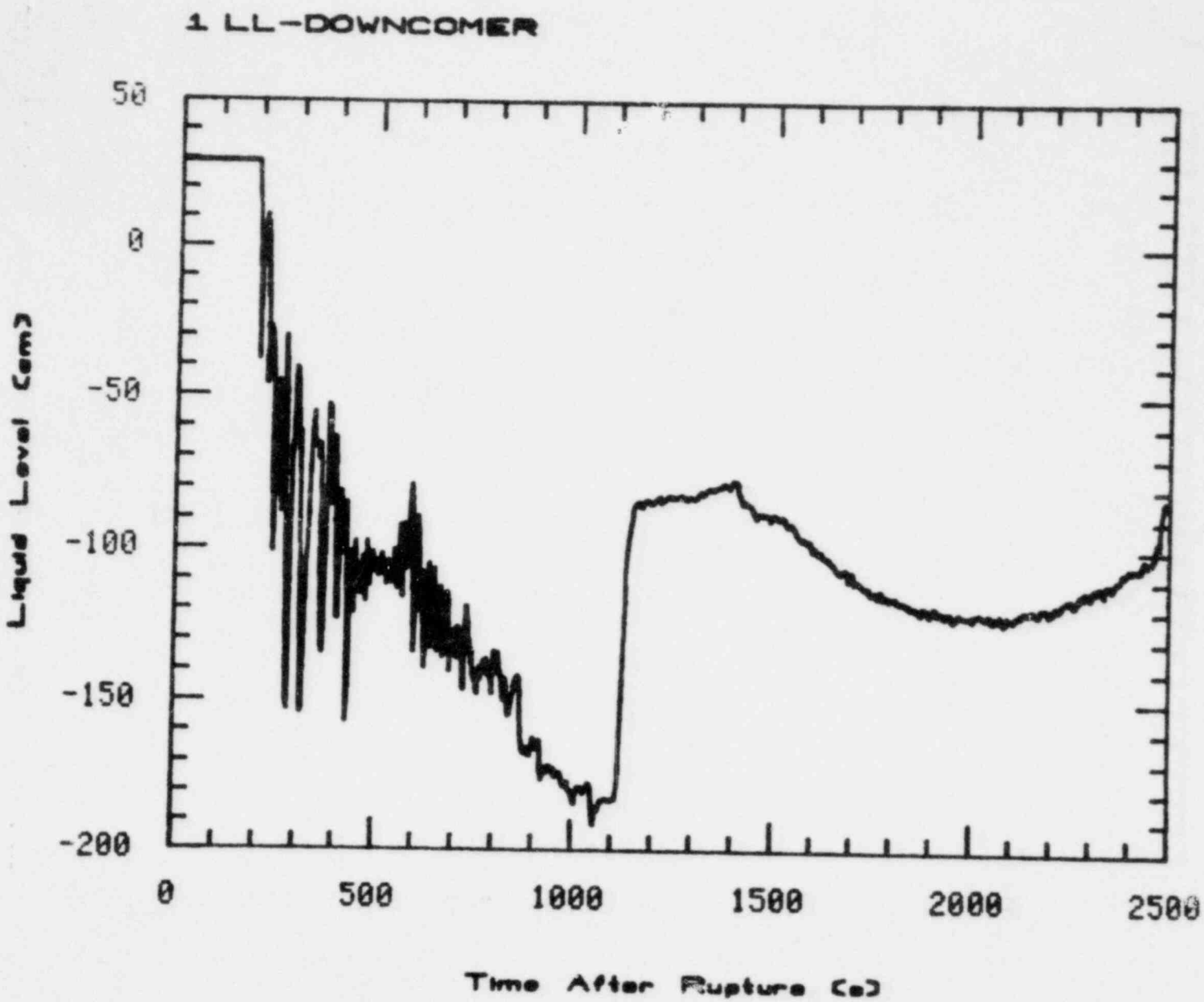


Figure 17. Downcomer collapsed liquid level for Test S-SB-P7.

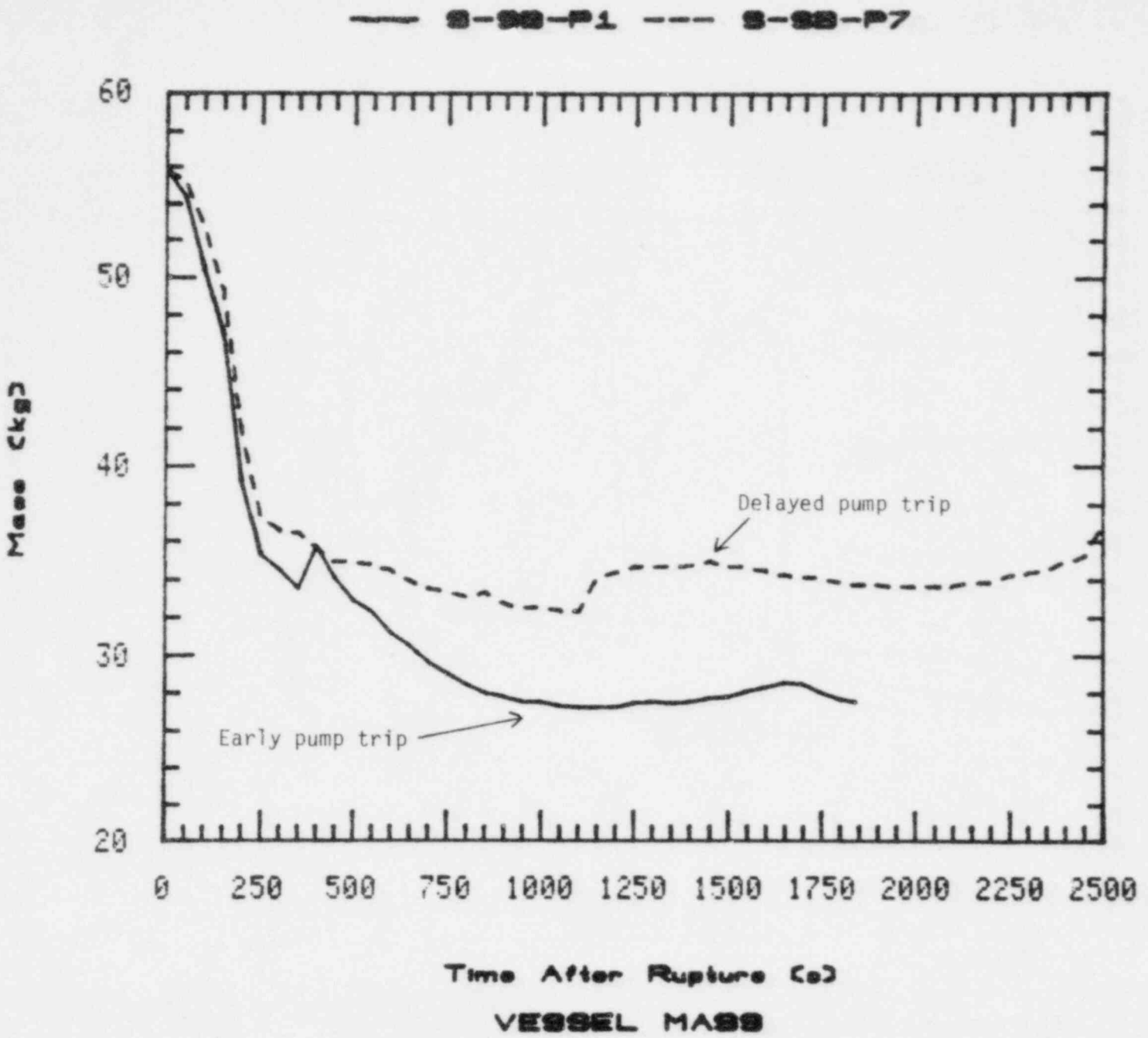


Figure 18. Comparison of vessel mass for Tests S-SB-P1 and S-SB-P7.

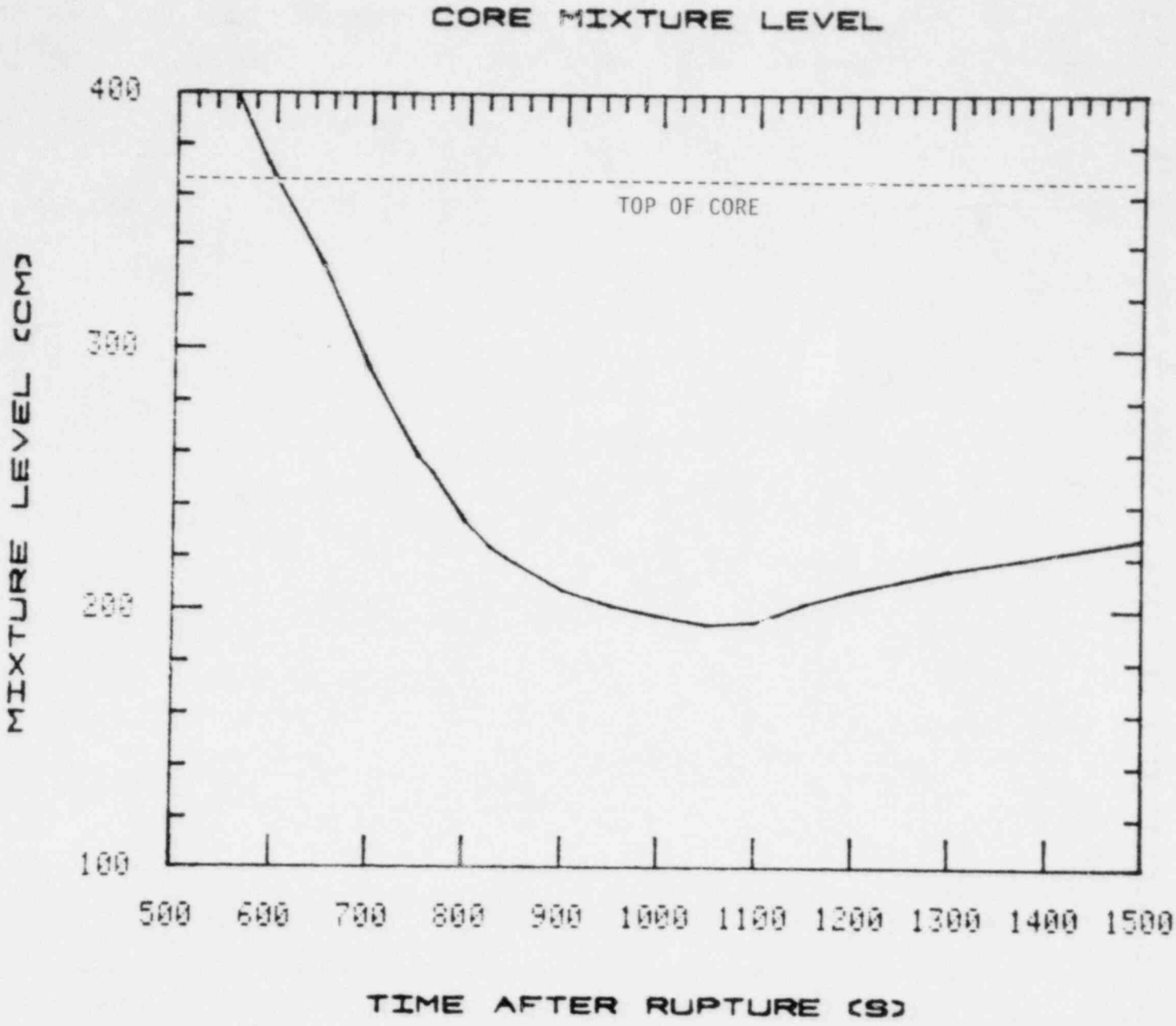


Figure 19. Core mixture level for Test S-SB-P1.

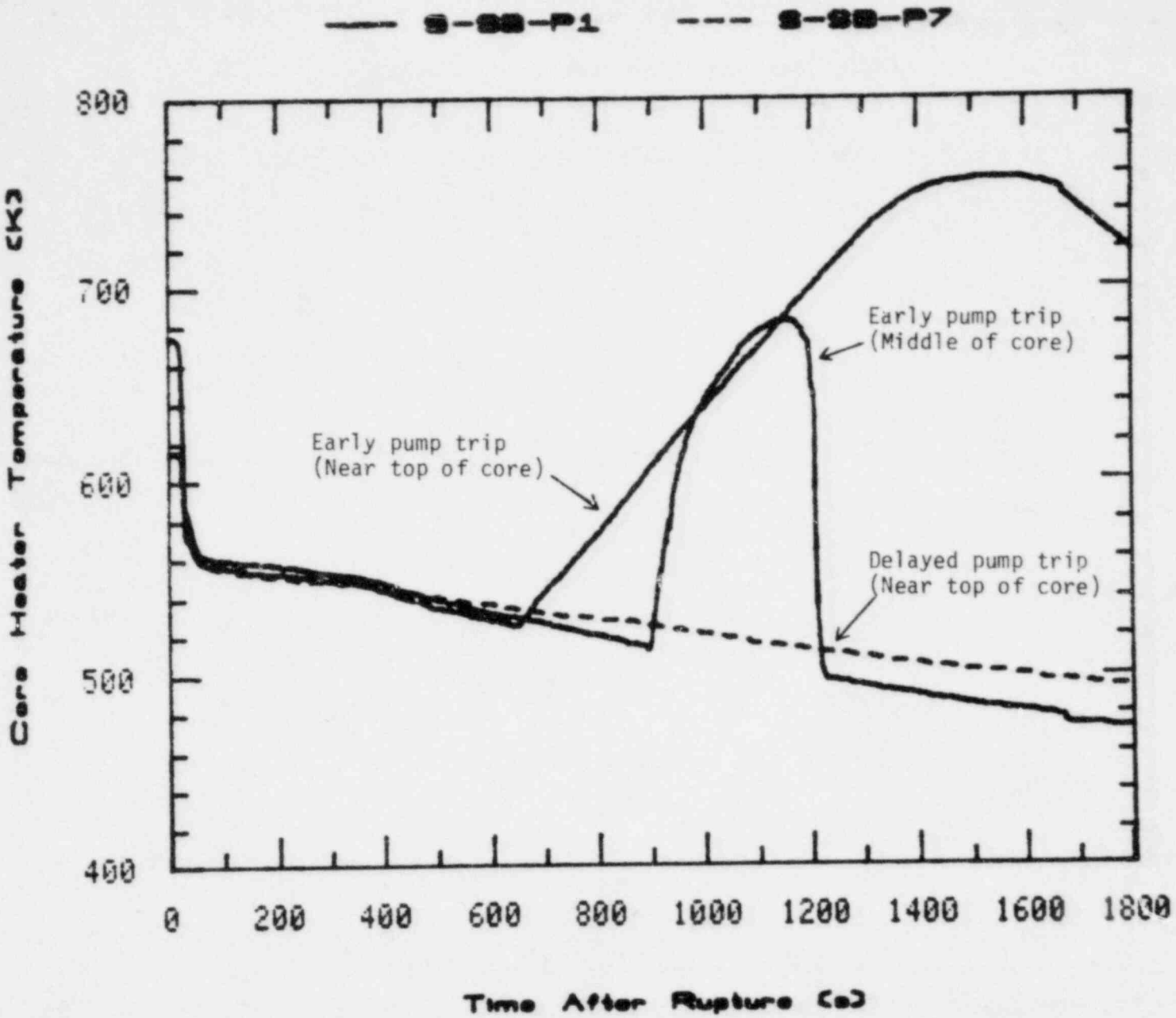


Figure 20. Comparison of cladding temperature response for Tests S-SB-P1 and S-SB-P7.

4. CONCLUSIONS

The results from Tests S-SB-P1, S-SB-P2 and S-SB-P7 indicate pump operation affected overall Semiscale system response. Pump operation influenced fluid conditions at the break location causing less subcooling at the break early in the transient when the pump trip was delayed. This smaller amount of subcooling resulted in lower break flows and less system mass loss during the early blowdown period than was observed in the early pump trip test. Since less mass was lost from the system with the pumps running, there was less potential for core uncovery during the delayed pump trip test than during the early pump trip test.

The delayed pump trip also influenced the system mass distribution. With the pump running for an extended period of time, a significant amount of mass was redistributed into the Semiscale loop piping. This resulted in a smaller percentage of total system mass in the core and vessel region with the delayed pump trip relative to that observed with an early pump trip, early in the experiment. However, because the total system mass was larger throughout the transient with the pumps left on, the lower percentage of system mass available for cooling the core was still greater than that available for cooling the core when the pumps were tripped early. As the pump head degraded in Test S-SB-P7, a higher percentage of the system mass settled into the vessel during the latter portion of the transient.

A final effect of the extended pump operation in Semiscale relates to the larger amount of entrained liquid that collected in the intact and broken loop hot legs when the pumps were running over that collected in the hot legs with an early pump trip. When the pumps were tripped late in the transient, this liquid tended to fall back into the vessel providing the potential for increased core cooling. Therefore, the potential for core uncovery in Semiscale following the delayed pump trip in Test S-SB-P7 was effectively precluded.

The results reported herein should not be construed as representative of what would occur in a full-sized PWR. The objective of these tests was to provide data for assessing the ability of codes to predict the influence of pump operation on overall Semiscale system behavior. Therefore, while scaling distortions in Semiscale may produce results atypical of those which might occur in a full sized PWR, the ability of codes to predict important phenomena occurring in Semiscale will provide insight into the ability of codes to predict similar phenomena in a full-sized commercial plant. In this way, Semiscale can contribute to the experimental data base used to evaluate the ability of computer codes to define appropriate pump operation in the event of a small break LOCA in a commercial PWR.

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2. G. W. Johnsen letter to R. E. Tiller, GWJ-8-80, Transmittal of Semiscale EOS Appendix for Small Break Tests S-SB-P1 and S-SB-P2, February 13, 1980.

3. L. P. Leach letter to R. E. Tiller, LPL-33-80, Experimental Operating Specification for Semiscale Small Break Test S-SB-P7, March 13, 1980.

4. System Design Description (SDD) for the Mod-3 Semiscale System, EG&G Idaho, Inc. (July 1978).