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QUICK LOOK REPORT FOR SEMISCALE MOD-2B TEST S-SG-7

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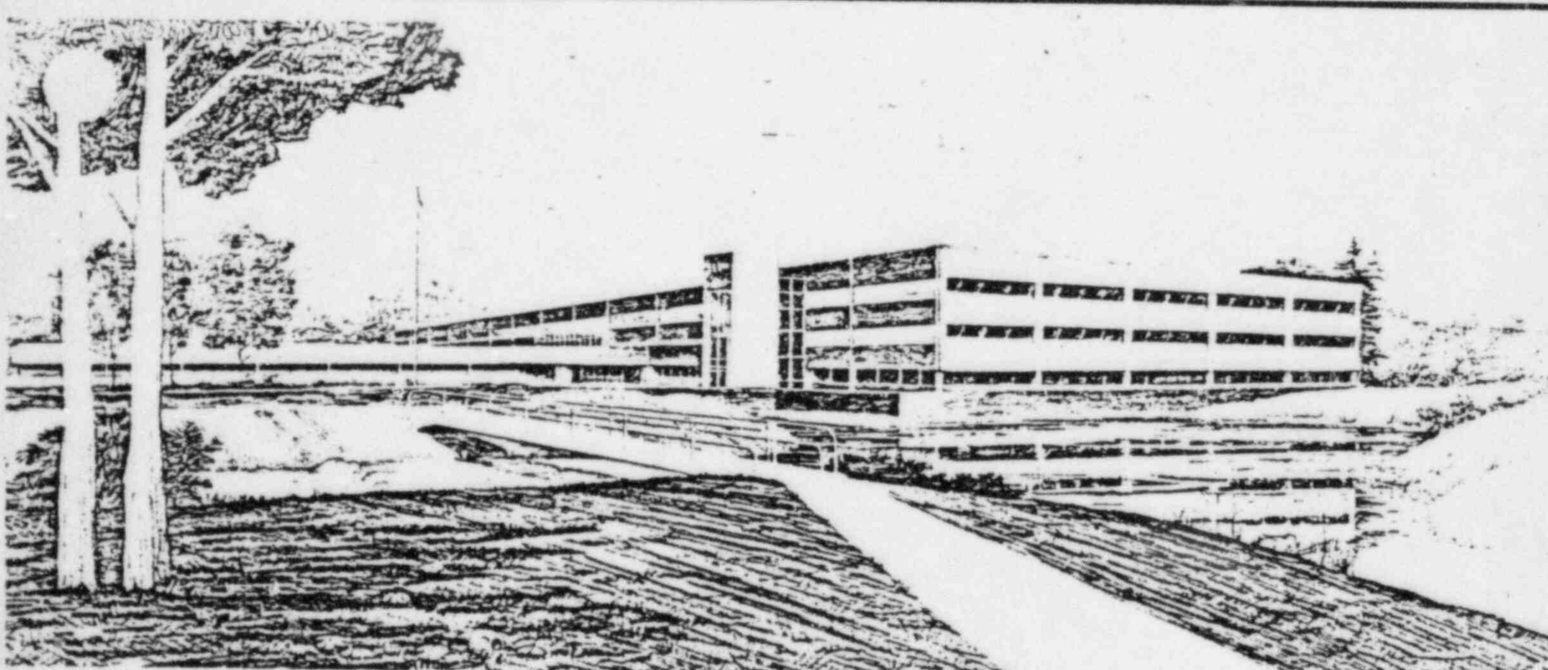

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TEST S-SG-7

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ABSTRACT

Results of a preliminary analysis of the fourth test performed in the Semiscale Mod-2B Steam Generator Tube Rupture Series are presented. Test S-SG-7 simulated a pressurized water reactor accident initiated by a double-ended offset shear of five cold side steam generator tubes. The transient included an initial 600 s period during which only automatic plant protection system response to the initiating event occurred. A total loss of both onsite and offsite power was assumed to occur at the SI injection signal. This effectively disabled the injection system and no SI was available for the entire transient. During the first 600 s the vessel collapsed liquid level was reduced to just below the top of the core but no core rod heat-up occurred. At 600 s, an operator induced recovery procedure was initiated which included unaffected loop secondary feed and steam (using ADV and auxiliary feed) to stabilize the primary pressure below the affected loop ADV setpoint. Following a period of stable primary pressure, the unaffected loop steam generator secondary filled and a new feed and steam operation commenced including ADV operation and auxiliary feed in an attempt to maintain the primary pressure below the affected loop secondary pressure. This operation caused a back flow from the affected loop generator to the primary which increased the vessel liquid inventory. The test results showed that without the use of SI, the unaffected loop steam generator feed and steam operation alone was sufficient to recover the Semiscale system from a simulated five-tube rupture without core rod heat up.

SUMMARY

This report presents a preliminary analysis of the Semiscale Mod-2B Steam Generator Tube Rupture Series (SG) Test S-SG-7. S-SG-7 is designed to study the effect of the number of tubes ruptured (break size), the location of the rupture (hot side or cold side of the steam generator) and the effect of limited operator responses to the accident following an initial 10-minute simulated identification period.

Test S-SG-7 simulated a pressurized water reactor transient initiated by a double-ended offset shear of five cold side steam generator tubes. 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 validate computer code capability.

Test S-SG-7 was designed to investigate the effect on system response of a loss of both offsite and onsite power at the safety injection trip signal. This resulted in a complete loss of safety injection during the entire transient; therefore, the effect of a loss of safety injection on overall system response is one of the main test objectives. Test S-SG-7 was designed in two parts: (a) an initial 600 s period in which only automatically functioning plant protection systems were assumed to operate and (b) an operator induced unaffected loop steam and feed to recover the plant.

The five-tube rupture signature was characterized by a relatively rapid decrease of the primary coolant system pressure to saturated conditions in the hot legs as primary coolant system fluid flowed through a break simulating five-tubes into the affected loop steam generator secondary. Automatic protective actions that influenced the pressure response during this early period were core scram and main steam isolation valve (MSIV) closure. Both were initiated by a low pressurizer pressure trip at 13.1 MPa (1888 psig). A loss of both offsite and onsite power was assumed to occur at the safety injection (SI) signal at a pressurizer pressure of 12.5 MPa (1803 psig). At the SI signal, main coolant pump

trip, feedwater termination, and auxiliary feedwater start were initiated. However, because of the loss of offsite and onsite power, safety injection did not occur nor was it on for the remainder of the transient. Part of the pressure response during this early period was a rapid increase in secondary pressure in both loops as primary-to-secondary heat transfer raised the pressure of the secondaries after MSIV closure. Accompanying this secondary pressure increase was a lifting of the atmospheric dump valve (ADV) in both the affected and unaffected steam generators. The five-tube break flow without SI was sufficient to leave the vessel collapsed liquid level near the top of the core at the end of 600 s (15 cm above the top). Following the attainment of saturation conditions in the hot legs, the primary and secondary system pressures remained fairly constant as flashing occurred in the primary and break flow entered the affected loop steam generator secondary. Core decay heat was removed by natural circulation.

At the conclusion of the 600 s operator diagnostic period the vessel collapsed liquid level was lower for 5-tube rupture transients without SI than with SI. Comparison of S-SG-7 (without SI) to S-SG-2 (with SI) shows that the vessel collapsed liquid level was just below the cold leg elevation for S-SG-2, and was near the top of the core for S-SG-7. On an overall system mass balance basis SI flow is only about 20% of break flow during the first 600 s. However, the overall mass inventory in the system was such that the vessel collapsed liquid level was significantly lower for the five tube rupture without SI (S-SG-7). On a long term basis without operator action, the effect of SI on core level would be significant.

During Test S-SG-7 a higher initial pressurizer liquid level was used relative to previous Semiscale 5-tube rupture experiments. The higher initial pressurizer collapsed liquid level resulted in a slower depressurization and a longer time to scram. However, the thermal-hydraulic state of the system during S-SG-7 after the first few hundred seconds was identical to other 5-tube rupture experiments with less initial pressurizer level. The extra pressurizer mass for S-SG-7 simply left the primary system via a longer period of high break flow. As a result, the system mass inventory after the first few hundred seconds was

essentially identical for S-SG-7 and other 5 tube rupture experiments with lower initial pressurizer mass (S-SG-2). The extra core power delivered to the primary fluid for S-SG-7 relative to other 5-tube rupture experiments (due to the longer time to scram) was dissipated in the longer secondary feed and steam period. Therefore a similar primary fluid temperature existed after a few hundred seconds for S-SG-7 and other 5-tube rupture experiments with lower initial pressurizer liquid inventory.

At 600 s an operator induced recovery procedure was initiated involving unaffected loop secondary feed and steam without the benefit of SI. The purpose of this recovery procedure was to first reduce the primary pressure below affected loop ADV setpoints and then to reduce the primary pressure below the affected loop secondary pressure. Unaffected loop secondary feed and steam was found to be a sufficient operator action to reduce primary pressure below affected loop ADV setpoints while at the same time being able to increase unaffected loop secondary level. Further, unaffected loop secondary steam and feed was able to reduce the primary pressure below the affected loop secondary pressure thus causing a back flow of affected loop secondary inventory into the primary. The backflow caused the vessel collapsed liquid level to increase from below the top of the core to the cold leg elevation while maintaining the unaffected loop secondary collapsed level above half full. At the start of each of the unaffected loop feed and steam operations, a momentary increase in flashing in the core and condensation potential in the unaffected loop steam generator occurred which apparently caused a decrease in vessel collapsed liquid level. However, the decrease in liquid level was small enough that no core heat-up occurred.

Overall good agreement was obtained between the data and pretest calculations. Timing of the vessel refill operation was not well predicted since the initial secondary mass in the unaffected steam generator was higher than specified.

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1. INTRODUCTION

This report documents preliminary results from Semiscale Mod-2B test S-SG-7, the fourth experiment performed 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 capability. 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 on simulation of a full-scale PWR address the design of the experiment rather than the quantitative results.

Test S-SG-7 simulated a pressurized water reactor transient initiated by a double-ended offset shear of five cold side steam generator tubes. The test basically simulated a five tube rupture transient accompanied by a loss of offsite and onsite power which disabled the safety injection throughout the transient. The test was designed in three parts, (a) an initial 600 s period in which only automatically functioning plant protection systems were assumed to operate, followed by (b) an operator induced secondary steam and feed of the unaffected loop steam generator (using ADV and auxiliary feed) in an attempt to maintain a stable primary pressure below the affected locp steam generator ADV setpoint thus precluding atmospheric discharge, and (c) an unaffected loop secondary feed and steam operation that maintained primary pressure below the affected loop secondary pressure. Automatically occurring events during the first 600 s included main steam isolation valve closure, termination of main feedwater, auxiliary feedwater initiation, and coastdown of the main coolant pumps. Safety injection was not used during the entire transient due tc the assumed loss of offsite and onsite power. Recovery operations were initiated at 600 s after the break occurred. (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.) The S-SG-7 recovery included unaffected loop generator feed and steam (steam through the ADV and feed from auxiliary feed) in an attempt to maintain the primary pressure at 5.72 ± 0.1 MPa (818 ± 15 psig). Maintaining the primary pressure at this pressure insured termination of atmospheric discharge through the affected loop ADV which was set at 5.85 MPa (836 psig). In the course of maintaining a stable primary pressure the unaffected loop steam generator secondary filled to the upper trip point of 1050 cm (413 in) (the total height of the steam generator is about 1170 cm (460 in)) where auxiliary feedwater is normally terminated. At this point a new unaffected loop feed and steam again using the ADV and auxiliary feed was initiated involving maintaining the primary pressure 0.137 ± 0.068 MPa (20 psia ± 10 psia) below the affected loop secondary pressure. This operation was designed to stimulate back flow through the break from the unaffected loop generator to the primary system. The reverse break flow was used to refill the depleted vessel inventory in the absence of SI. The test was terminated when the vessel collapsed liquid level was on a demonstrated filling trend above the heated length of the core.

A preliminary analysis of test S-SG-7 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 analysis, 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[a] 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 are incorporated to properly control and measure boundary conditions for the steam generator tube rupture series. These include condensing systems and catch tanks to accurately measure system mass flow rate from the steam generator secondaries, special effluent flow controls in the steam generator secondaries to give properly scaled steam relief flow rates, and a tube-rupture break assembly to simulate the primary to secondary flow path created by the tube rupture.

In both the unaffected and affected loops, 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] The SRV orifice is designed to pass a scaled flow 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

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a. For test S-SG-7 only 22 rods were utilized.

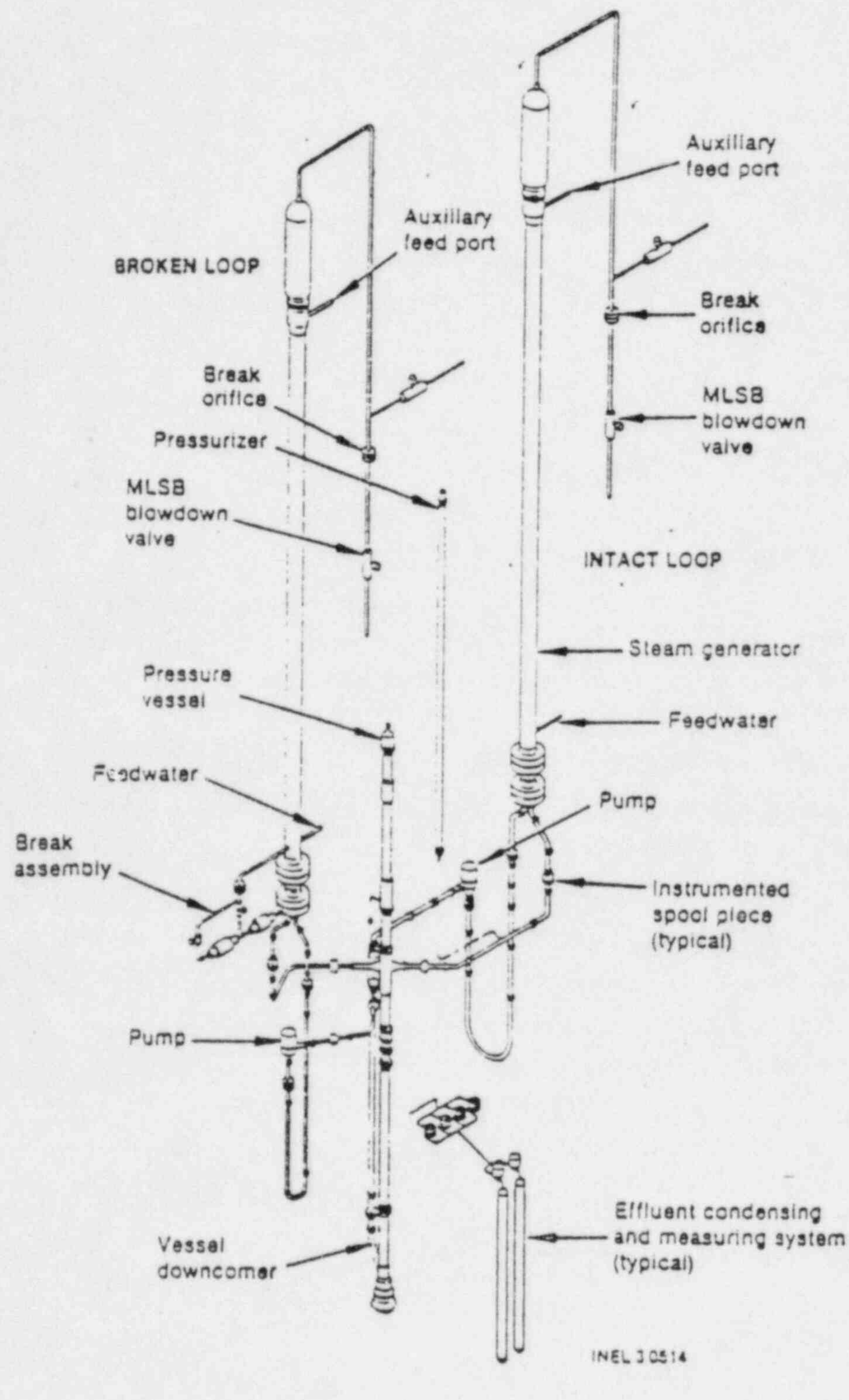

Figure 1. Semiscale Mod-2B system as configured for the SG test series.

designed to pass scaled flow corresponding to ADV operation in a PWR. In a PWR, the pressure relief setpoint for the ADV stage is encountered before the various multistaged SRV relief setpoints. Figure 2 shows the orientation used in Semiscale to simulate this operation in both the affected and unaffected loops. The parallel flow path arrangement allows ADV flow through the ADV block valve and orifice, and stage one SRV flow through the combination of both block valves and orifices. The block valves operate in an open or shut mode only, with the orifices controlling the flow rates. The ADV block valve opens automatically at the ADV pressure setpoint. If the pressure continues to rise after the ADV opens, the SRV block valve opens automatically at the SRV pressure setpoint. As the pressure decreases, the block valves close automatically, 69 kPa (10 psi) below their respective pressure setpoints. In Semiscale, the ADV relief setpoint is 5.85 MPa (836 psig) in the affected loop and 6.55 MPa (937 psig) in the unaffected loop. The first stage SRV relief setpoint is 5.94 MPa (849 psig) in the affected loop and 6.74 MPa (965 psig) in the unaffected loop.[a] Figures 3 and 4 show mass flow rate versus pressure for ADV and SRV operation for the affected and unaffected loops, respectively. The ADV can also be manually latched open during the recovery procedure with the SRV block valve shut.

The pressurizer PORV[b] provides a means of manually relieving primary system pressure from the top of the pressurizer. Semiscale uses a single valve with a flow control orifice to simulate the two PORV's of a full scale PWR. A 0.141 cm (0.055 in.) sharp edged orifice was sized to pass 0.03 kg/s (0.069 lb/s) at 16.2 MPa (2350 psia). The scaling criteria are presented in Appendix A of Reference 1. The pressurizer surge line hydraulic resistance was 1.8 x 10^9 m^{-4} for test S-SG-7. For previous tube rupture experiments (tests S-SG-1, S-SG-2, S-SG-5) the hydraulic

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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.

b. The PORV was not used on S-SG-7.

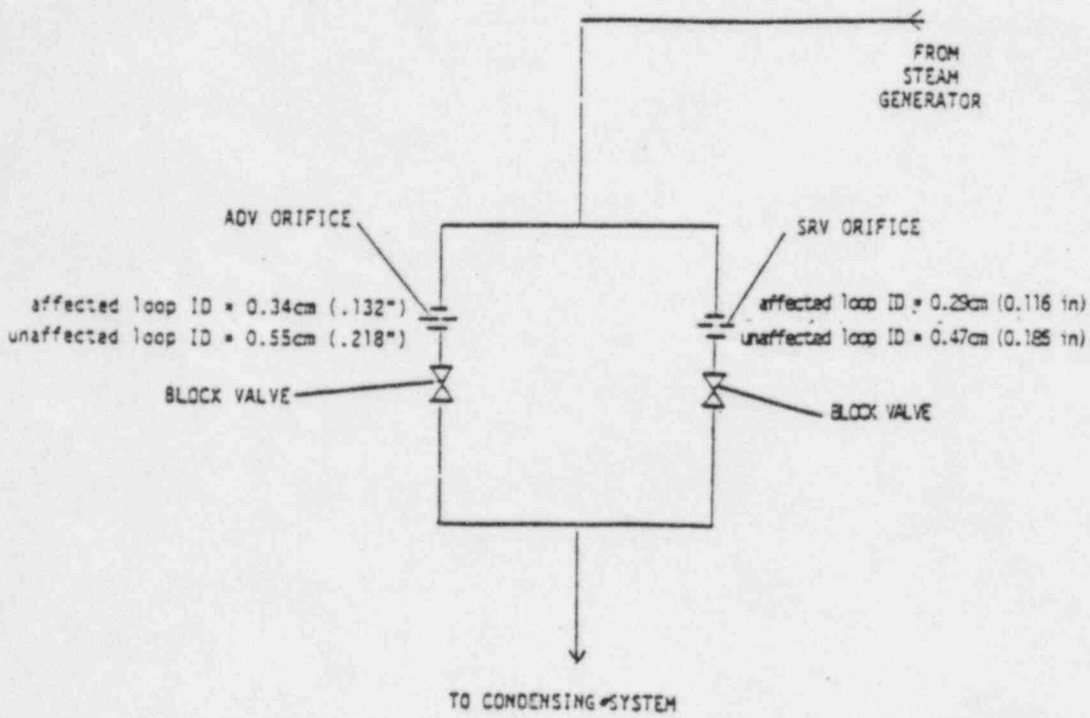


Figure 2. ADV and safety relief valve system.

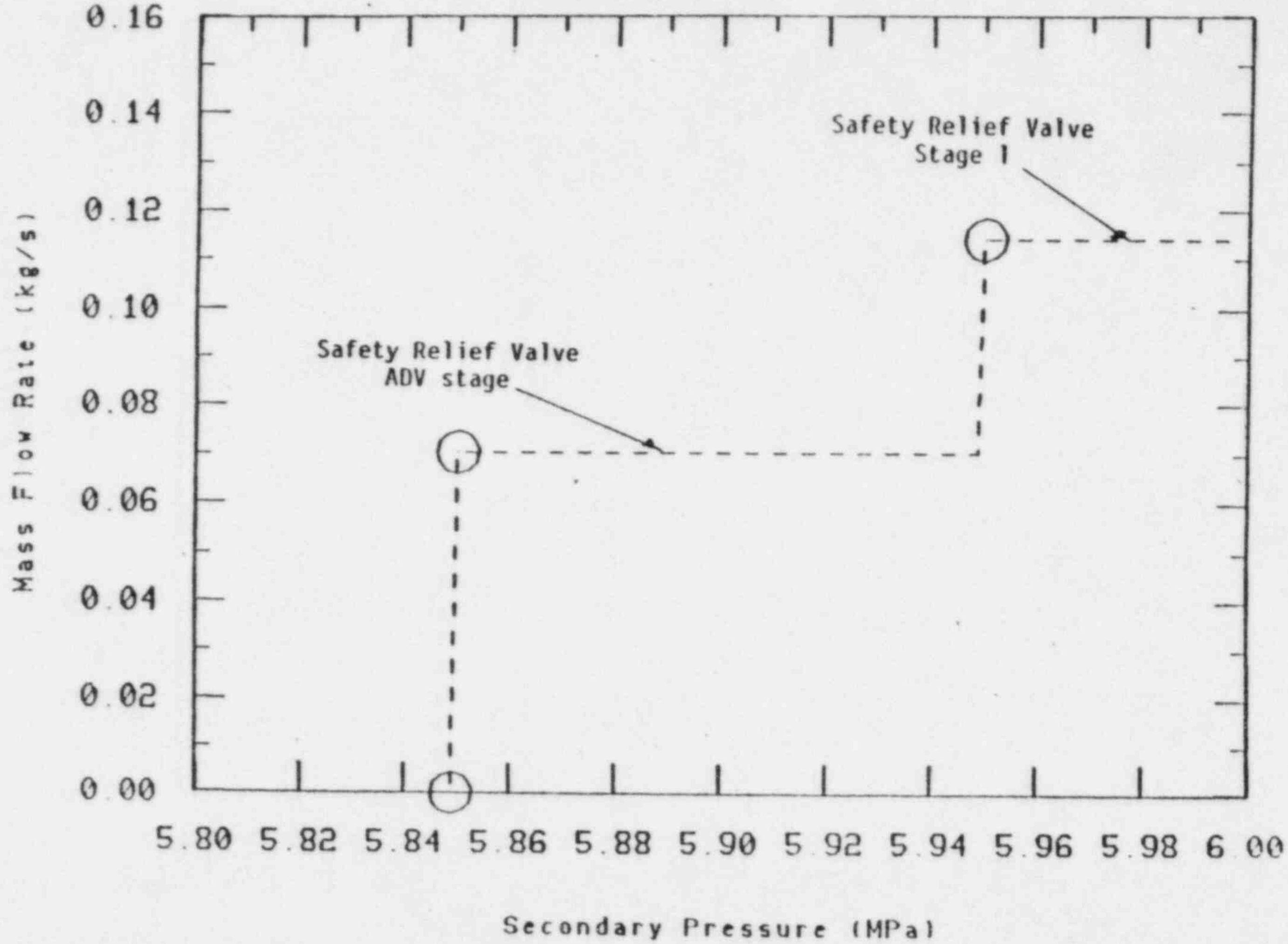


Figure 3. Broken loop steam generator safety relief valve operation.

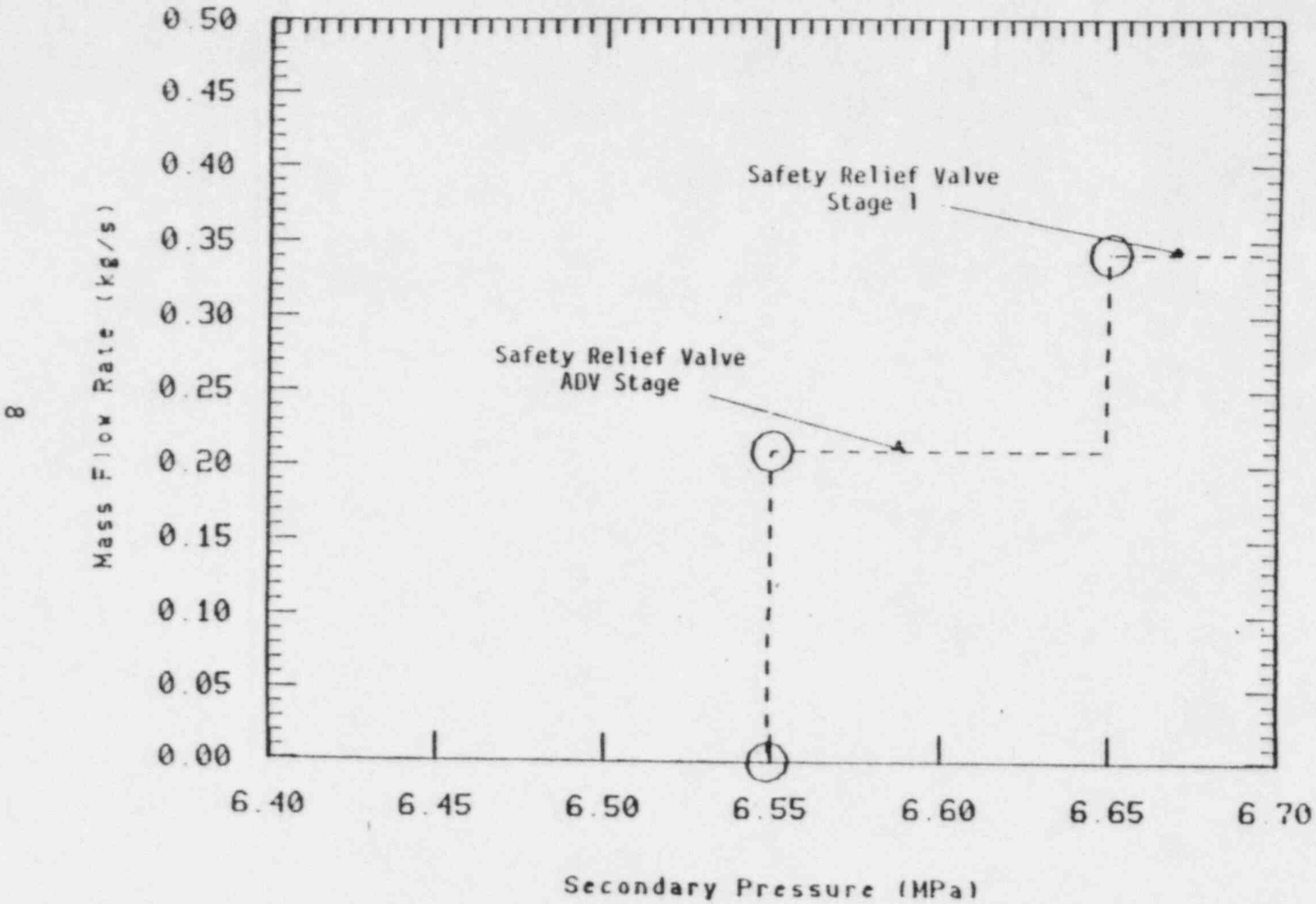


Figure 4. Intact loop steam generator safety relief valve operation.

resistance was $1.08 \times 10^{9}\ m^{-4}$. This difference in hydraulic resistance did not cause a change in phenomena because pressurizer drain time was controlled by break flow early in the transient.

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 (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-7, 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 an interchangeable symmetric conical flow tube as depicted in Figure 6. Figure 6 shows the dimensions for a 1-, 5-, and 10-tube break orifice. Test S-SG-7 used the 5-tube break orifice with a 0.175 cm (0.0689 in) ID. The flow tube was calibrated in single phase water and can be used to monitor break mass flow rate.

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 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 inside 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. Pressurizer external heaters were not used in S-SG-7. Power to the vessel upper head and upper plenum heater banks was terminated just before the transient initiation.

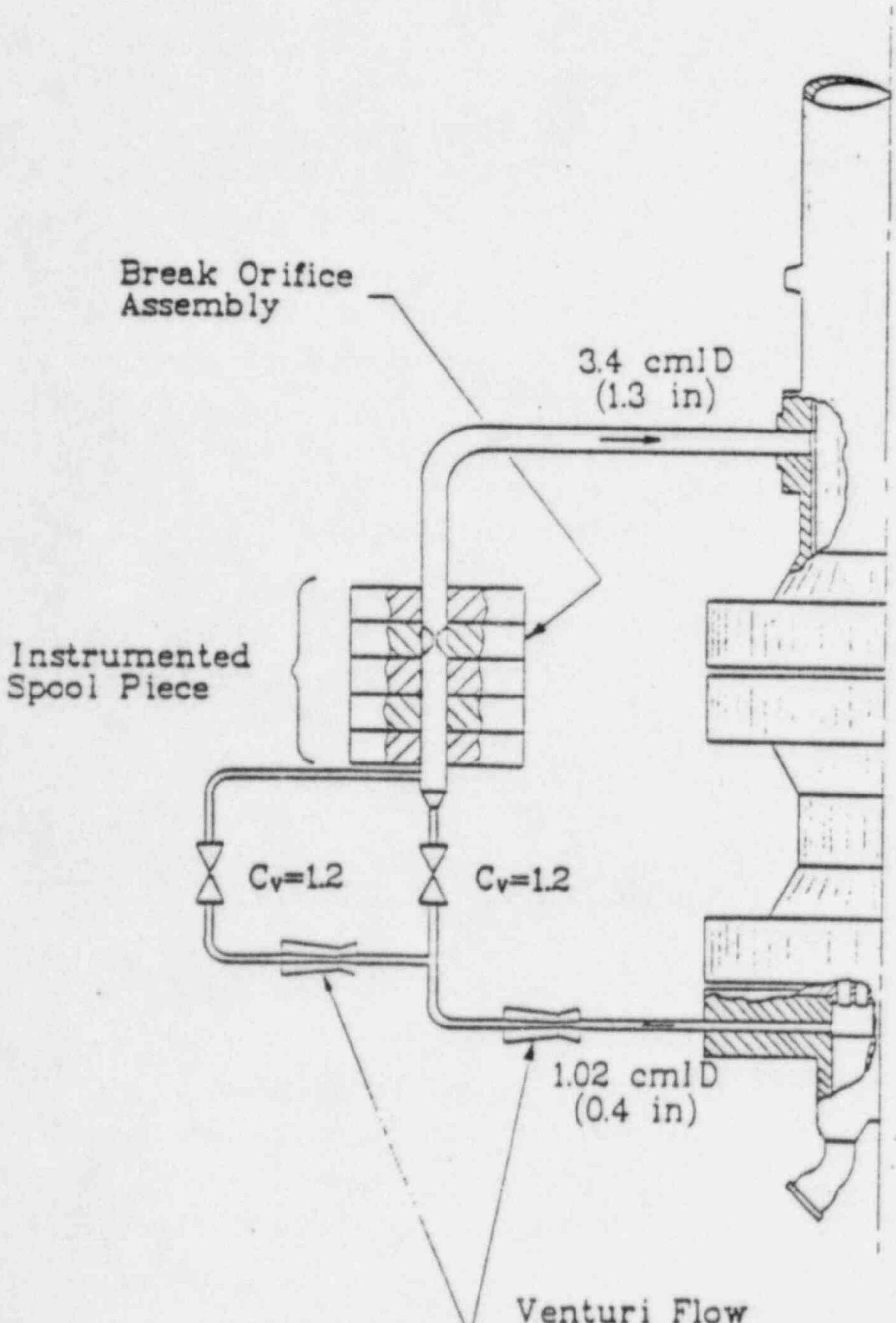
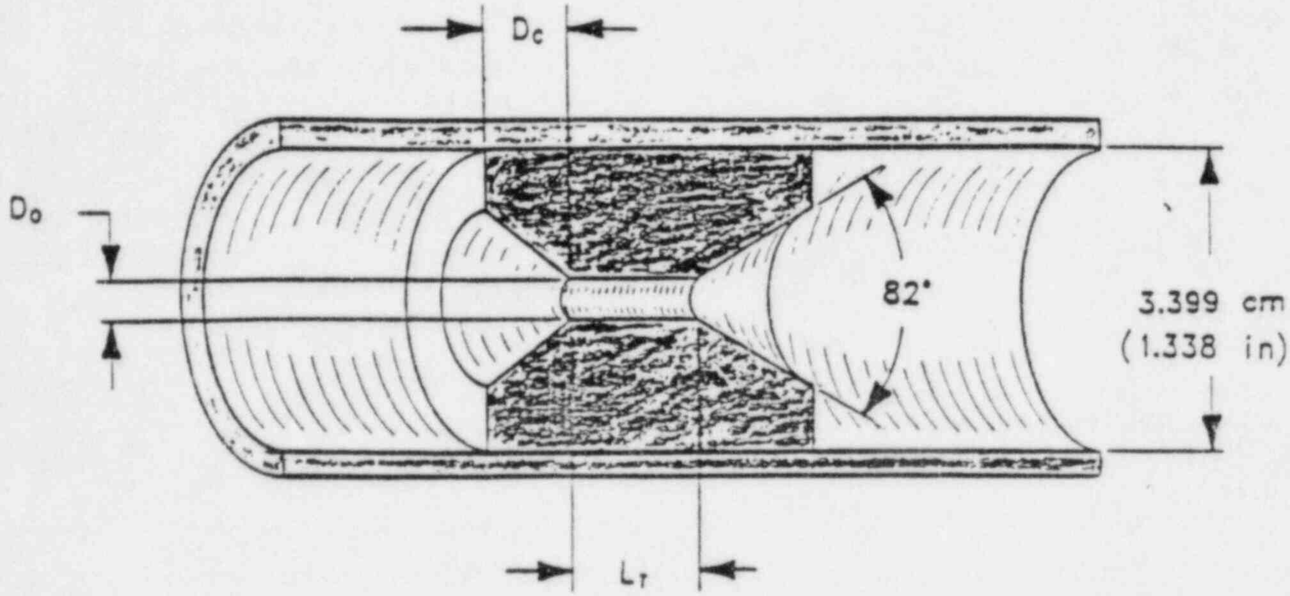


Figure 5. Semiscale tube-rupture break assembly.



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.

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-7. The first 600 s involved automatically occurring events such as core scram, main steam isolation valve closure, auxiliary feedwater start and main feedwater stop, and main coolant pump trip. The initiating events for these actions were a low pressurizer pressure trip (13.1 MPa (1888 psig)) and SI signal (12.51 MPa (1803 psig)). Loss of both onsite and offsite power was assumed to occur at the SI signal and initiated the main coolant pump trip and prevented use of the safety injection system. The safety injection system was not used for the entire transient. Power to the pressurizer external heaters and vessel upper head and upper plenum external heaters was terminated at t = 0.

The recovery procedure started at 600 s. This time period simulated the time required for operator identification of the tube rupture. Unaffected loop feed and steam using ADV operation and auxiliary feedwater began at 600 s in an attempt to maintain the primary pressure at 5.72 ± 0.1 MPa (818 ± 15 psig). The unaffected loop steam generator secondary liquid level was maintained between 800 cm and 1050 cm (355 and 413 in) during this operation. The 5.72 MPa (818 psig) primary system pressure was maintained until the unaffected loop generator secondary collapsed liquid level reached 1050 cm at which time the ADV was operated such that the primary pressure was maintained at 0.137 ± 0.068 MPa (20 ± 10 psia) below the affected loop secondary pressure. Maintaining the primary pressure below the secondary pressure caused a back flow of primary fluid resulting in a vessel refill. The unaffected loop auxiliary feedwater was

TABLE 1. INITIAL CONDITIONS FOR TEST S-SG-7

	Specified	Measured
Pressurizer pressure	15.6 ± 0.14 MPa (2250 ± 20 psig)	15.45 (2240 psig)
Pressurizer liquid volume	0.0102 ± 0.0008 m^3 (0.36 ± 0.028 ft^3)	0.0091 m^3 (0.32 ft^3)
Core power	2.0 ± 0.01 MW	1.99 MW
Loop to loop cold leg fluid temperature differential	2.0K (3.6F)	1.9K (3.4F)
Core fluid temperature rise	37 ± 1.5K (66.6 ± 3°F)	38.7K (69.6°F)
Steam generator pressure		
Affected loop	5.55 ± 0.07 MPa (793 ± 10 psig)	5.58 MPa (809 psig)
Unaffected loop	5.55 ± 0.07 MPa (793 ± 10 psig)	5.49 MPa (796 psig)
Steam generator secondary fluid mass[a]		
Affected loop[b]	100 + 40 - 20 kg (220 + 88 - 44 lbm)	109 kg (240 lbm)
Unaffected loop[c]	100 + 40 - 20 kg (220 + 88 - 44 lbm)	178 kg (392 lbm)
Primary leakage	<0.006 kg/s (<0.0132 lbm/s)	0.0029 kg/s (0.0063 lbm/s)

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

b. Measured with differential pressure cell LBS+1117+51.

c. Measured with differential pressure cell LIS+1117+51. Even though this parameter is 38 kg too high, phenomena occurring in the primary system was not effected. The steam generator primary tubes would have been covered even if the level had been within specification. The only effect on the experiment is the timing of secondary fill during the recovery procedures (see Section 3).

TABLE 2. SEQUENCE OF SIGNIFICANT EVENTS FOR TEST S-SG-7

Specified Criteria	Actual Time(s)	Event
0 s	0	Break flow initiated
P_{PRZ} = 13.1 MPa (1888 psig)	32	SCRAM
SCRAM	33	Core power on ANS decay
SCRAM	33	MSIV closure
P_{PRZ} = 12.5 MPa (1803 psig)	35	SIS
SIS	35.6	Main feedwater secured
SIS	35.6	Auxiliary feedwater initiated
SIS	36.9	Pumps off
600 s	600	Unaffected loop ADV operation started to maintain 5.72 ± 0.1 MPa (818 ± 15 psig) primary pressure
1050 cm (413 in) unaffected loop generator level achieved	2848	1050 cm (413 in) level in unaffected loop. ADV operated to maintain P_p 0.137 MPa (20 psig) less than affected loop secondary
Stable or increasing vessel level achieved	3750	Stable or increasing vessel level started
Stable or increasing vessel level maintained for 30 minutes	5500	Test terminated

operated to maintain the collapsed level between 800 cm and 1050 cm (314 in and 413 in). The test was terminated when the vessel collapsed liquid level was above the top of the core and in an increasing mode.

3. RESULTS

This section discusses the overall thermal-hydraulic response of the Semiscale system during test S-SG-7. The discussion is organized into two areas: (a) the initial response to automatically occurring events (0-600 s) including a comparison of S-SG-7 (no SI flow) and S-SG-2[4] (which included SI) and (b) the recovery period involving operator actions.

3.1 System Behavior--Tube Rupture Signature Early in Time (0 to 600 s)

The occurrence of a 5-tube rupture event during normal operation in a PWR has a very distinctive signature response, as shown in the comparison of primary and secondary pressure in Figure 7. The tube rupture (occurring in the affected loop steam generator) initiated the transient at 0 s. Primary fluid originally at 15.45 MPa (2240 psia) flowed through the conical flow tube break orifice into the affected loop steam generator originally at 5.6 MPa (803 psia). The loss of mass from the primary loop caused a fairly steady primary depressurization until about 32 s, at which time a marked increase in the depressurization rate occurred. At 32 s the low pressurizer pressure setpoint of 13.1 MPa (900 psia) was achieved which initiated two prominent events which greatly affected the depressurization rate: the core power was scrammed to the ANS decay power curve and the main steam isolation valves were closed on the steam generators. The large decrease in system pressure following core scram at 32 s was due to dropping core power and the primary liquid shrinking due to primary to secondary heat transfer. Upon MSIV closure, the heat transfer to both the affected and unaffected loop steam generator secondaries caused a rapid pressurization of the secondaries as shown in Figure 8. The secondary pressure in both generators briefly reached the ADV setpoints (6.55 MPa (938 psig) in the unaffected loop and 5.85 MPa (836 psig) in the affected loop).

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 (Figure 8). The energy addition to the

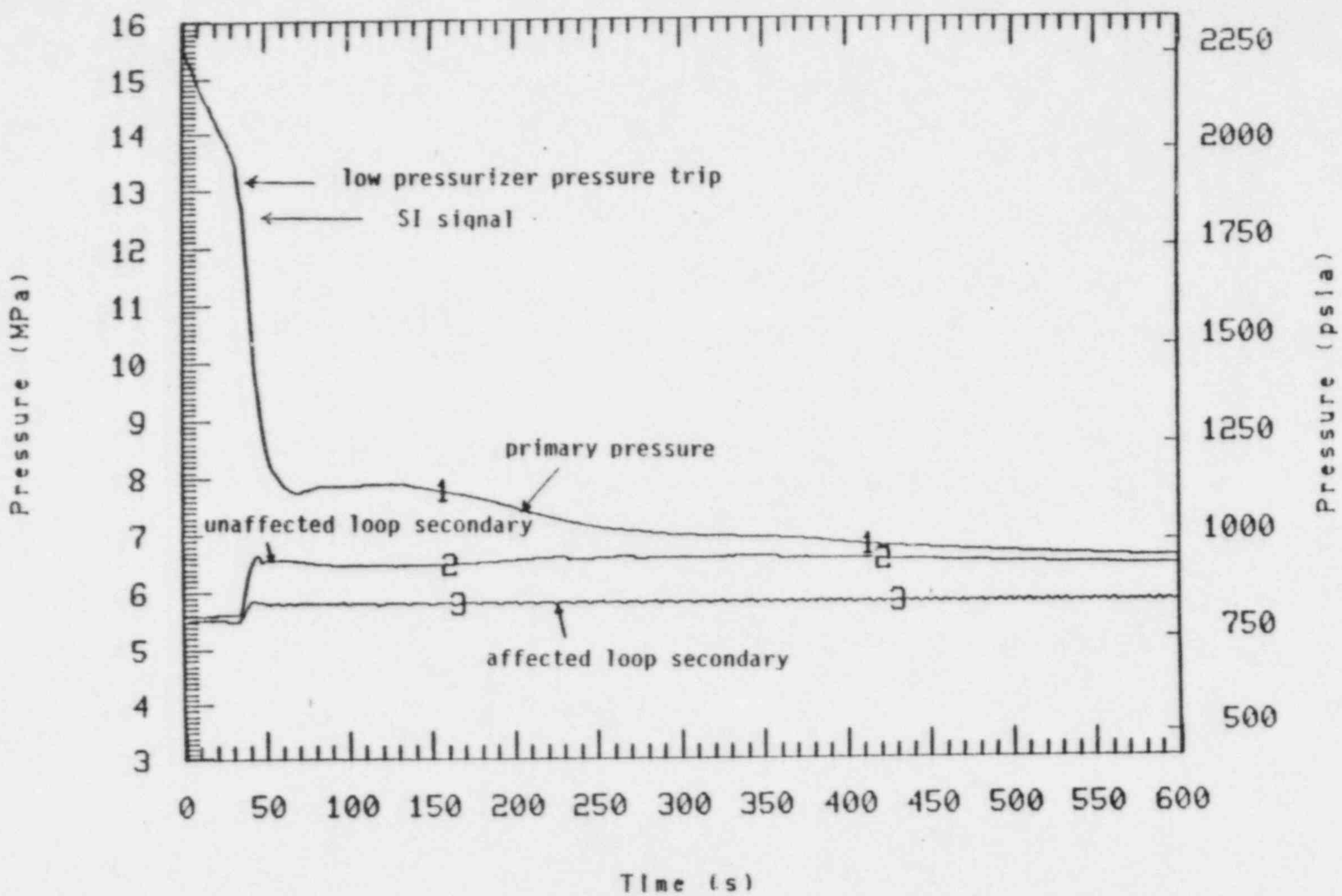


Figure 7. Comparison of primary and secondary pressure during a five tube rupture transient (S-SG-7).

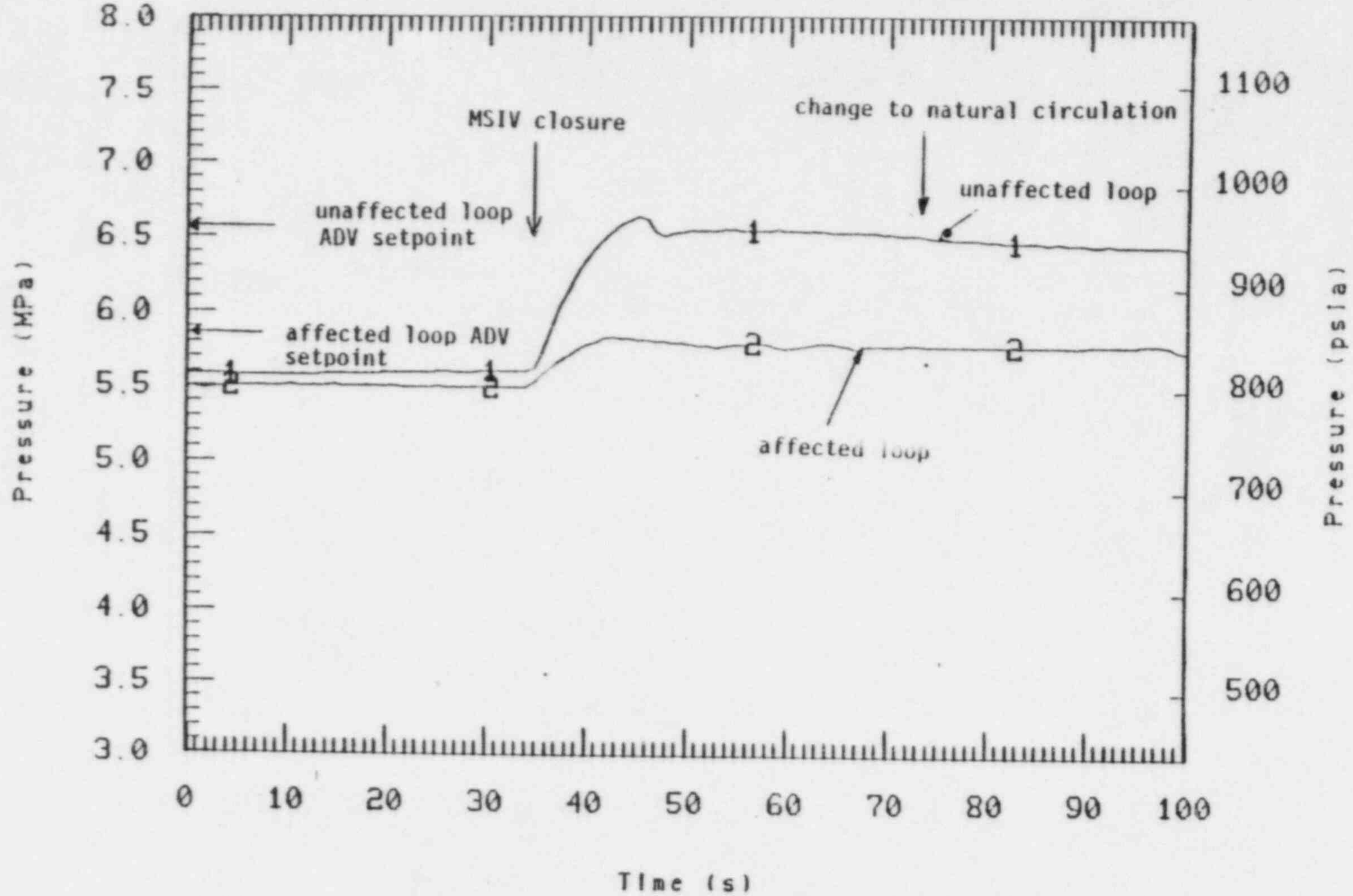


Figure 8. Comparison of affected and unaffected loop secondary pressure for a five tube rupture transient (S-SG-7).

affected loop secondary from break flow had a negligible effect on secondary pressure during this period because it was overpowered by the overall primary to secondary heat transfer.

The safety injection signal was achieved at 12.5 MPa (1803 psig) and initiated: (a) terminating power to the primary coolant pumps, and (b) terminating main feedwater and starting auxiliary feedwater to the secondaries. No major change in primary depressurization rate occurred from these events as their effects were overshadowed by the effect of core scram.

Following pump trip and coastdown, the loop flows reduced to typical natural circulation values[5] as shown on Figure 9. The termination of pump flow and subsequent transition to natural circulation resulted in reduced heat transfer from primary to secondary in the unaffected loop as evidenced by a slight decrease in unaffected loop secondary pressure (see Figure 8). Eventually, the primary system depressurization was sufficient for the hot leg fluid to reach a saturation condition at about 65 s (Figure 10). Flashing in the system then caused a major reduction in the depressurization rate (Figure 7). The primary pressure made a slight recovery between 75 and 100 s. This repressurization was mainly caused by superheated steam in the pressurizer (Figure 10) and reduced heat transfer to the unaffected loop secondary, aided by flashing in the reactor vessel.

Primary pressure remained above both secondary system pressures for the entire 600 s period (Figure 7) and caused a primary-to-affected loop secondary mass flow as shown in Figure 11. Figure 12 shows the pressurizer collapsed liquid level essentially depleted after 65 s. The effect of having no SI on the system mass inventory and vessel level during the initial 600 s period is discussed in the following section.

3.1.1 Effect of SI Flow on the Early Transient (0-600 s)

The loss of SI during a 5 tube rupture transient was found to have a significant effect on vessel collapsed liquid level at the end of the operator diagnostic period. However, on an overall system mass balance

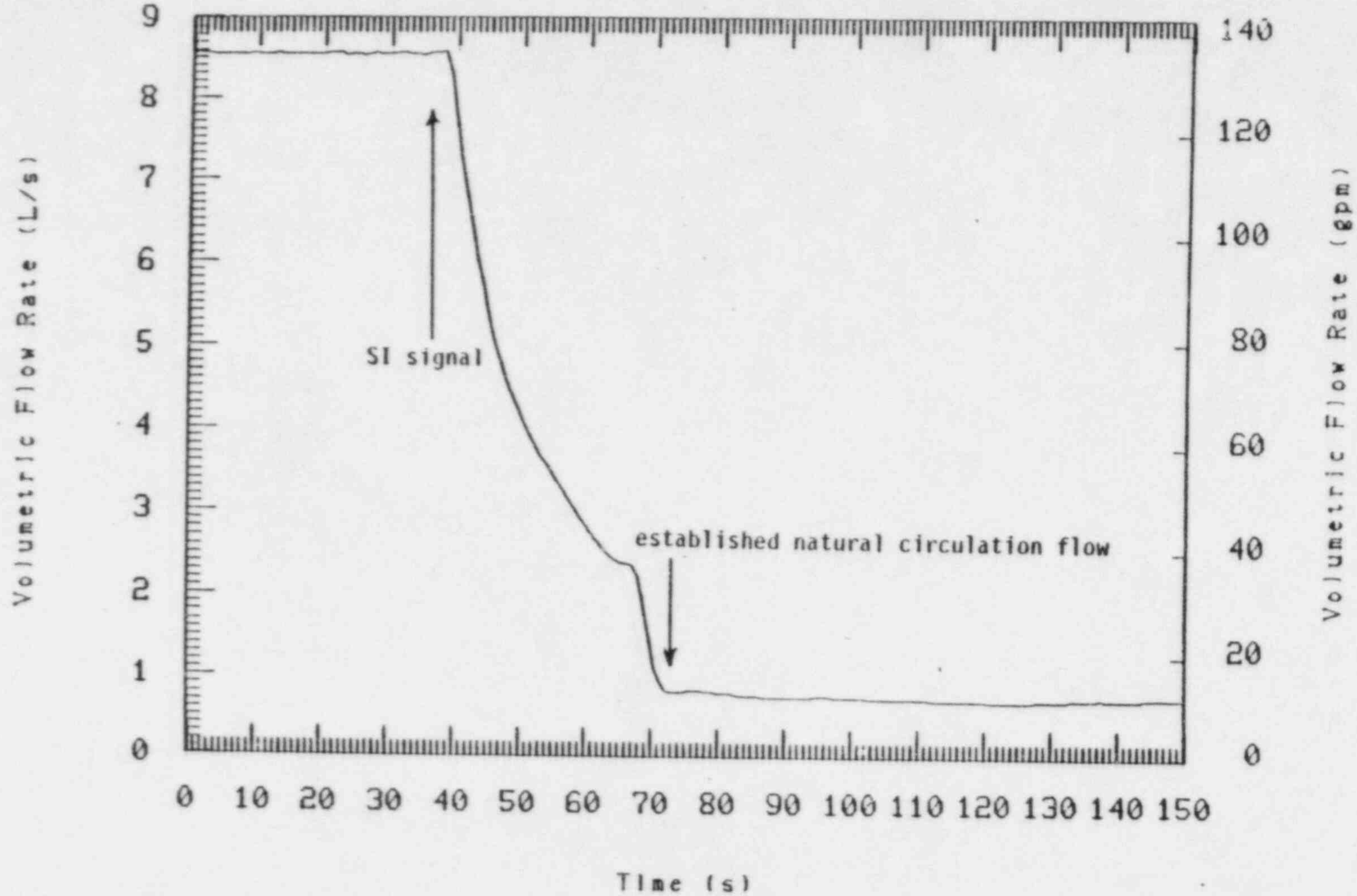


Figure 9. Cold leg volumetric flow in the unaffected loop during a five tube rupture transient (S-SG-7).

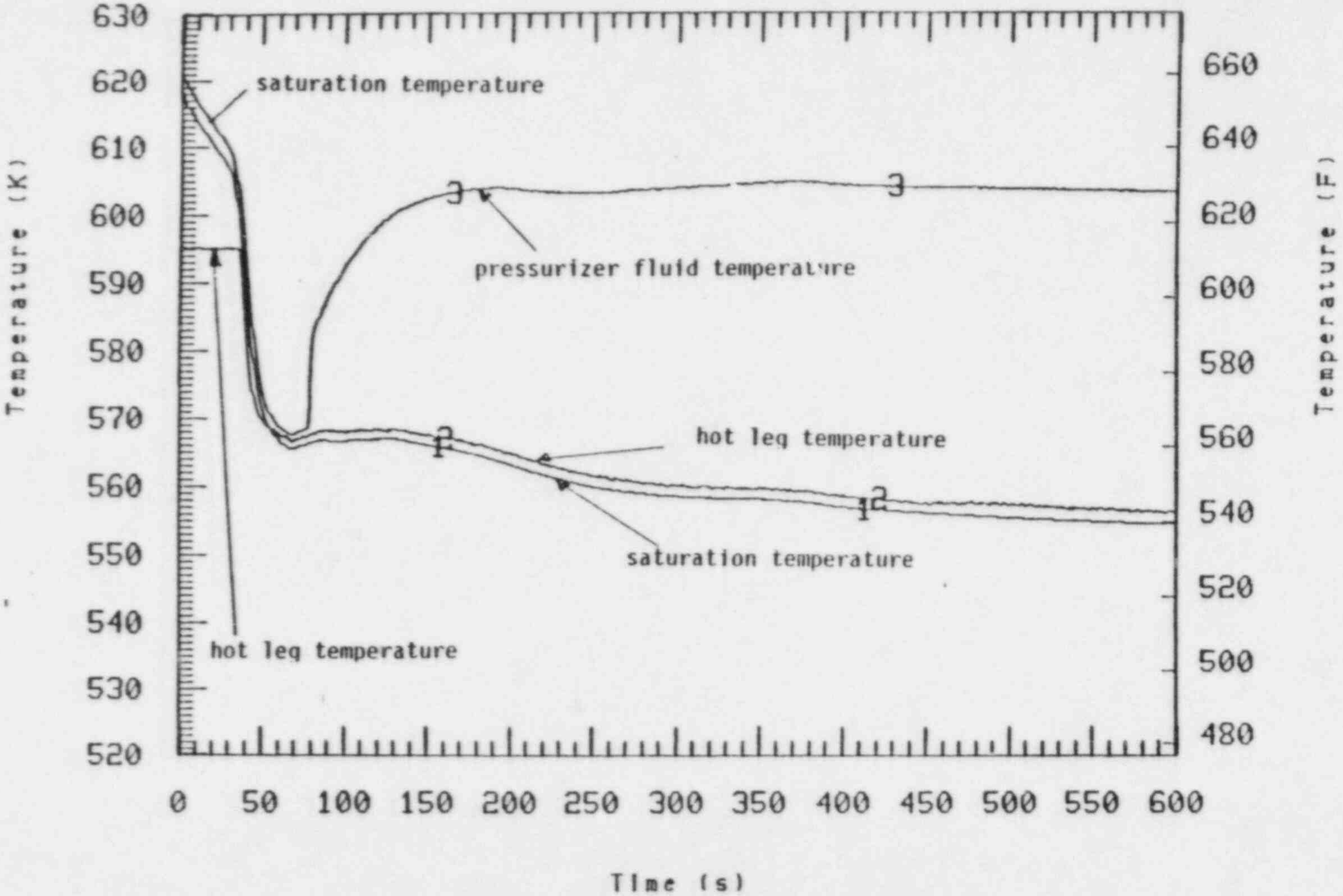


Figure 10. Comparison of unaffected loop hot leg fluid temperature, pressurizer fluid temperature, and saturation temperature during a five tube rupture transient.

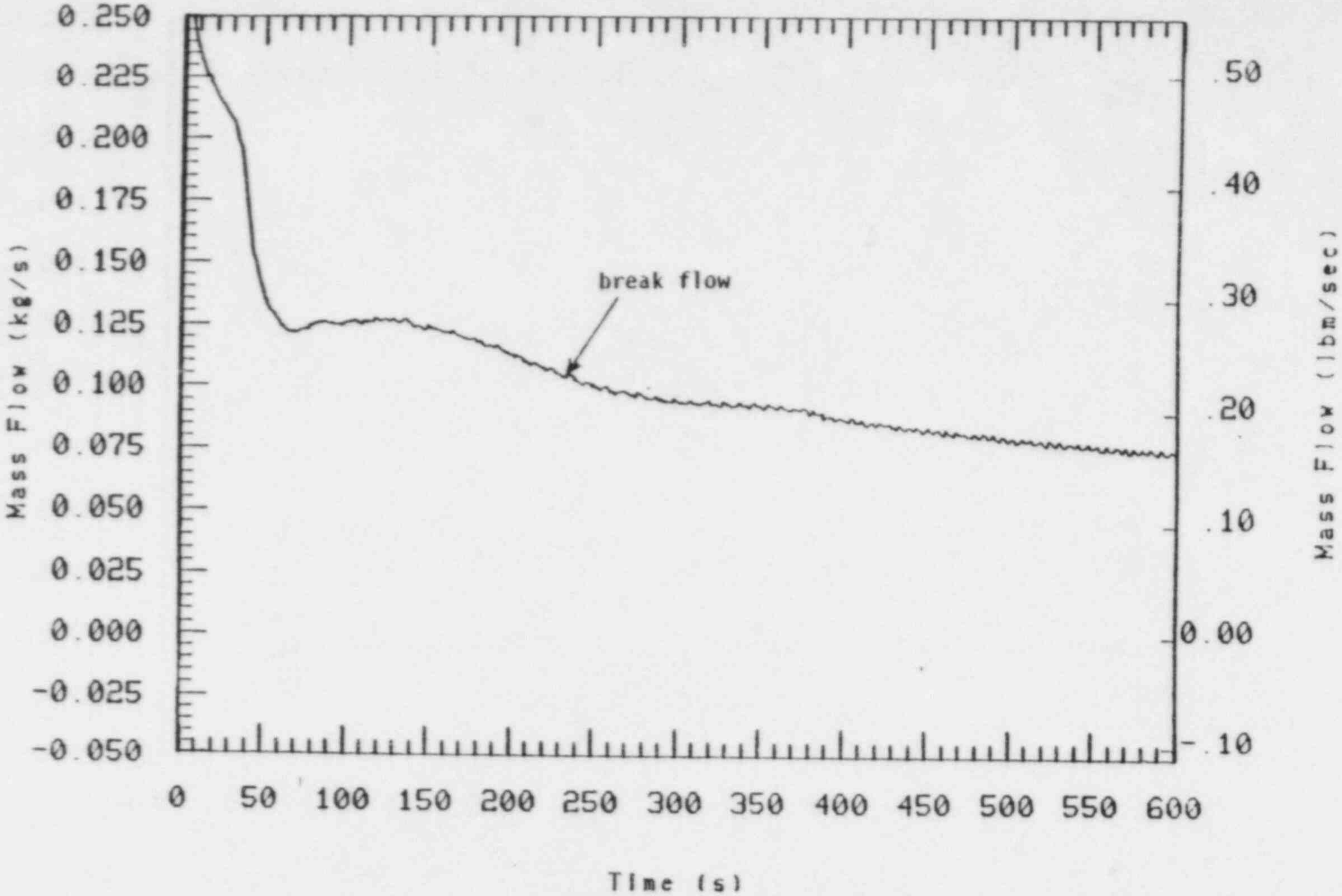


Figure 11. Break flow during a five tube rupture transient (S-SG-7).

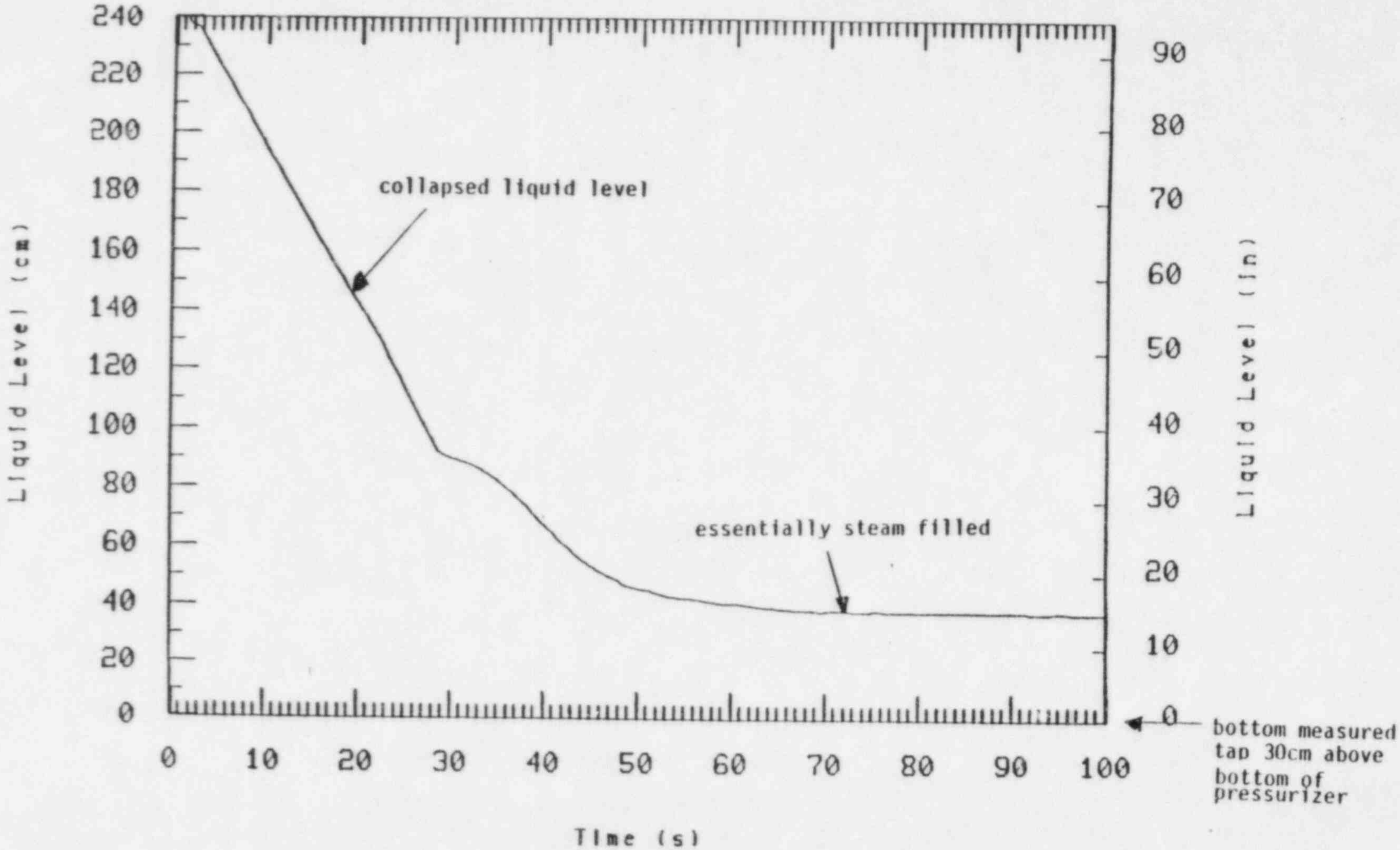


Figure 12. Pressurizer liquid level during a five tube rupture transient (S-SG-7).

basis SI flow was not large compared to the overall break flow. Comparison of the vessel collapsed liquid level for a 5-tube rupture transient where no SI was used (S-SG-7) and a 5-tube rupture transient where SI was used (S-SG-2) shows the vessel collapsed liquid level for S-SG-7 to be about 100 cm lower than for S-SG-2 at 600 s as shown in Figure 13. The vessel level for S-SG-2 (with SI) was near the cold leg and the vessel level for S-SG-7 (without SI) was near the top of the core. During the first 600 s only about 10 kg of SI was introduced to the system during S-SG-2 which compares to a total integrated break flow of 60 kg. Therefore SI was small enough relative to the break flow such that overall system mass inventory is relatively unaffected by SI early in time. However in S-SG-7 without SI, the vessel inventory was affected by break flow enough to cause a 100 cm reduction in vessel level at 600 s. Figure 14 compares break flow for S-SG-2 and S-SG-7 and SI flow for S-SG-2 showing that break flow dominates the system mass balance during the first 600 s. During the recovery phase (after 600 s) the lack of SI begins to have a pronounced effect on system mass inventory as SI dominates break flow during this period. Recovery of the plant without SI is discussed in Section 3.2.

3.1.2 Effect of Initial Pressurizer Liquid Level on the Early Depressurization

The initial pressurizer liquid level has a large effect on the primary system depressurization during the early portion of the transient and thus the timing of certain automatically occurring events. However, on an overall basis, differences in initial pressurizer liquid level do not cause differences in the long term system response such as primary pressure and stored energy in the primary. The pressurizer initial liquid level for test S-SG-7 was considerably higher than previous cold side 5-tube rupture experiments (S-SG-2 for example). Test S-SG-7 had an initial collapsed level in the pressurizer of about 240 cm above the measurement tap and test S-SG-2 had an initial collapsed level of about 140 cm above the tap (this corresponds to an initial liquid mass for S-SG-2 of 1.85 kg and 5.4 kg for S-SG-7). The level used for test S-SG-7 is the correct scaled value as specified in Reference 1. The initial pressurizer level used for S-SG-2 was low because a collapsed level was used as originally specified

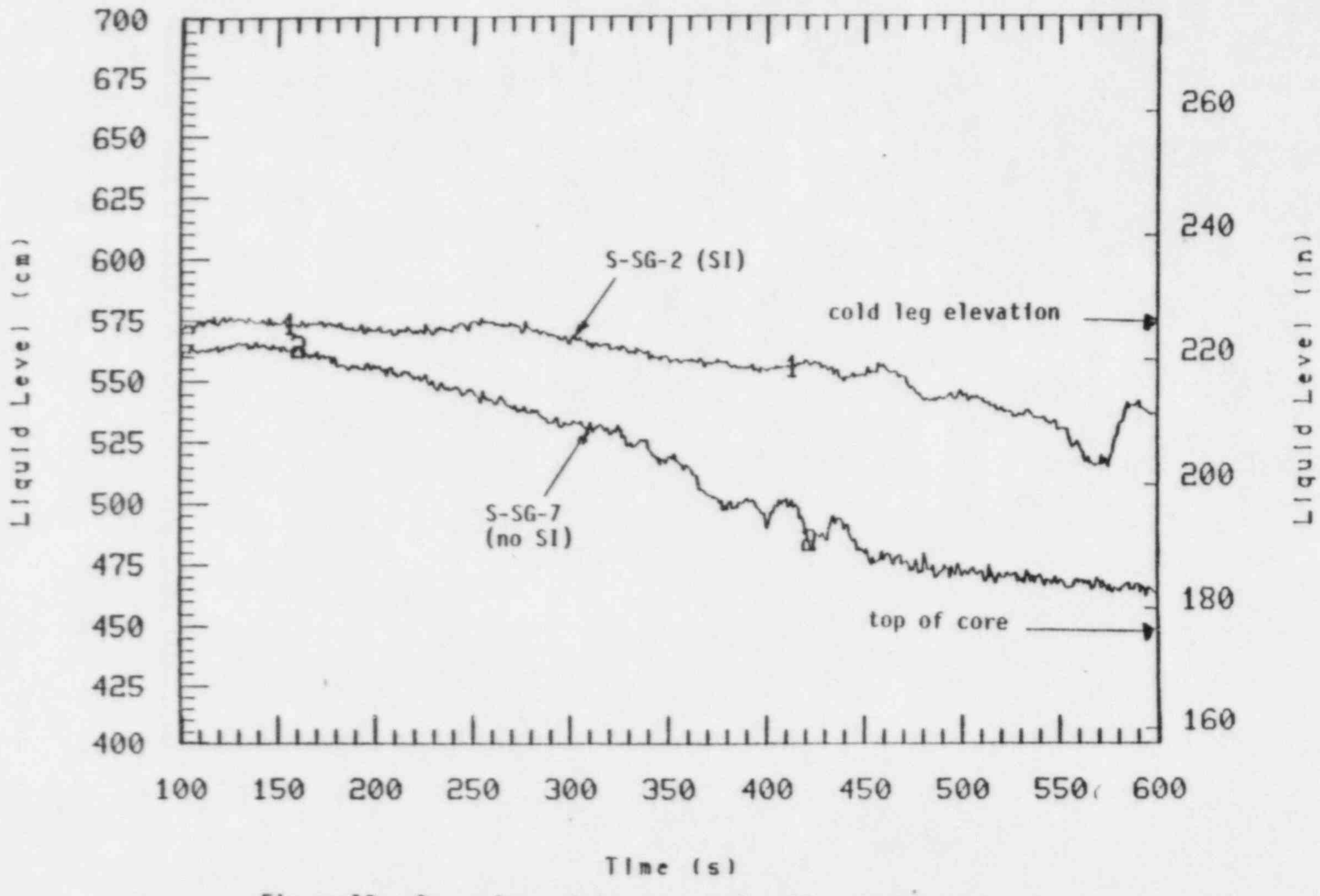


Figure 13. Comparison of lower vessel collapsed liquid level during two five tube rupture transients; one with SI (S-SG-2) and one without SI (S-SG-7)

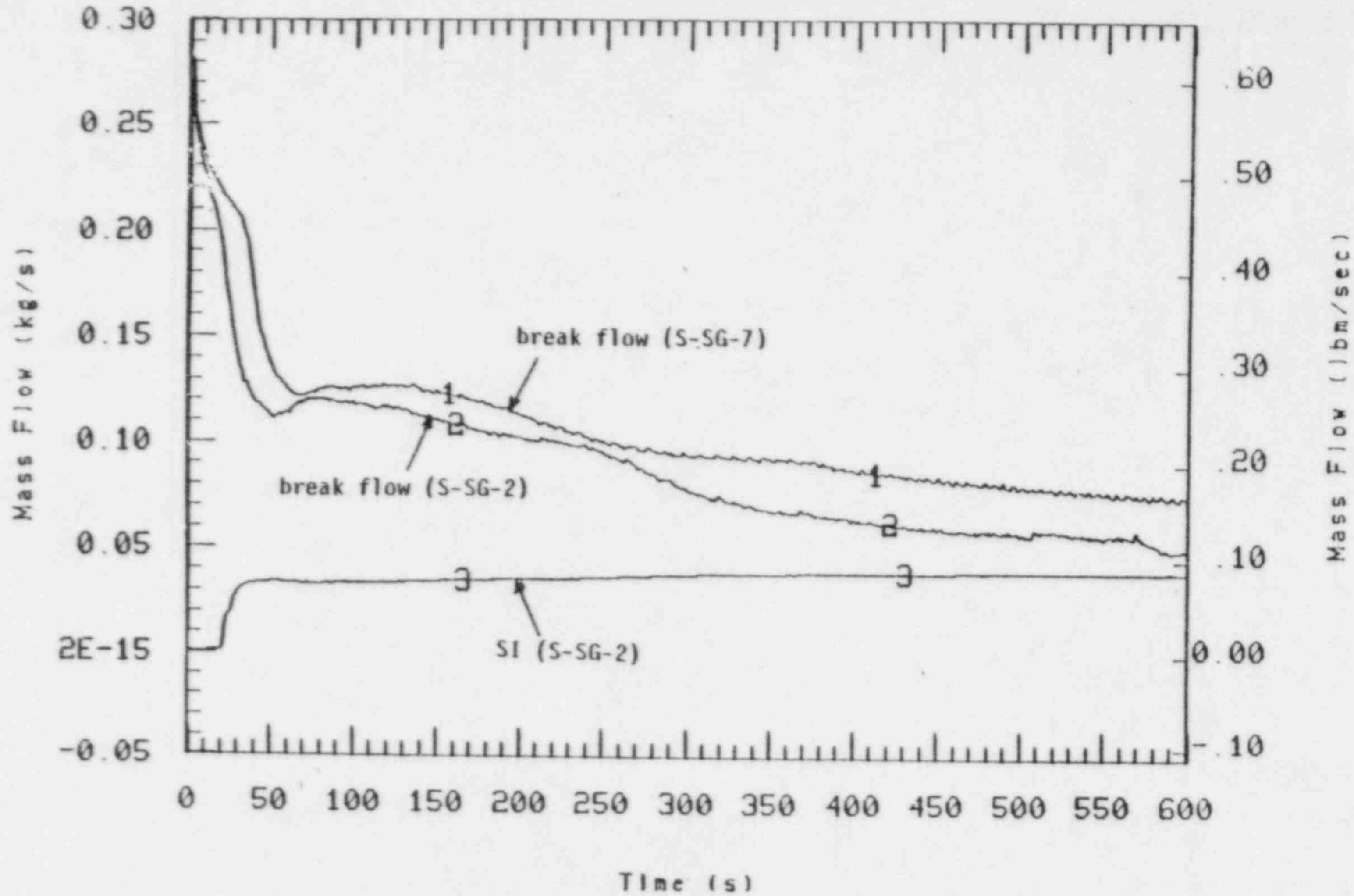


Figure 14. Comparison of break flow and SI flow during a five tube rupture with SI (S-SG-2) and a five tube rupture without SI (S-SG-7).

in Reference 1 rather than an interfacial level which was used for S-SG-7. Figure 15 compares the primary system pressure for test S-SG-2 and S-SG-7 showing a much slower depressurization time to the scram point (13.1 MPa (1888 psig) for test S-SG-7 (higher initial pressurizer level) than for S-SG-2 (lower initial pressurizer level). This corresponds directly to the drain time of the pressurizer as shown in Figure 16 which compares the collapsed level for S-SG-2 and S-SG-7. When the interfacial[a] area of the liquid in the pressurizer is high, as discussed in Reference 4, flashing retards the depressurization rate. However, when the pressurizer liquid level depletes to the pressurizer surge line, the interfacial area is drastically reduced thus reducing the flashing effect and a more rapid system depressurization results. With a higher initial collapsed height of liquid in test S-SG-7, the interface level remained higher for a longer period of time which caused a retarding effect on the depressurization. With a higher system pressure for S-SG-7 the break flow remained higher for a longer period of time as shown on Figure 17. However, the overall vessel level was about the same after 100 s as shown on Figure 18 which compares vessel upper head collapsed level for the two tests. On a preliminary basis without overall system mass balances, the additional mass in the pressurizer for S-SG-7 probably went out the break leaving the system mass inventory essentially the same.

The stored energy in the system fluid following pressurizer drain appears to be the same for S-SG-2 and S-SG-7 as demonstrated on Figure 19 which compares the saturation temperature in the hot leg to the hot leg fluid temperature. Following about 100 s into both transients the hot leg fluid temperatures were identical and indicated similar energy content in the system fluid even though 2 MW of core power was added to the primary for 20 s more in S-SG-7 than in S-SG-2. Even though core power was on 20 s longer, the feedwater termination and main steam isolation valve closure were also delayed 20 s for S-SG-7 relative to S-SG-2 and as a result, the

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a. Interfacial level is a "pooled" liquid level with steam above; collapsed level refers to all the fluid (both steam and liquid) between the measurement taps being treated as saturated liquid only.

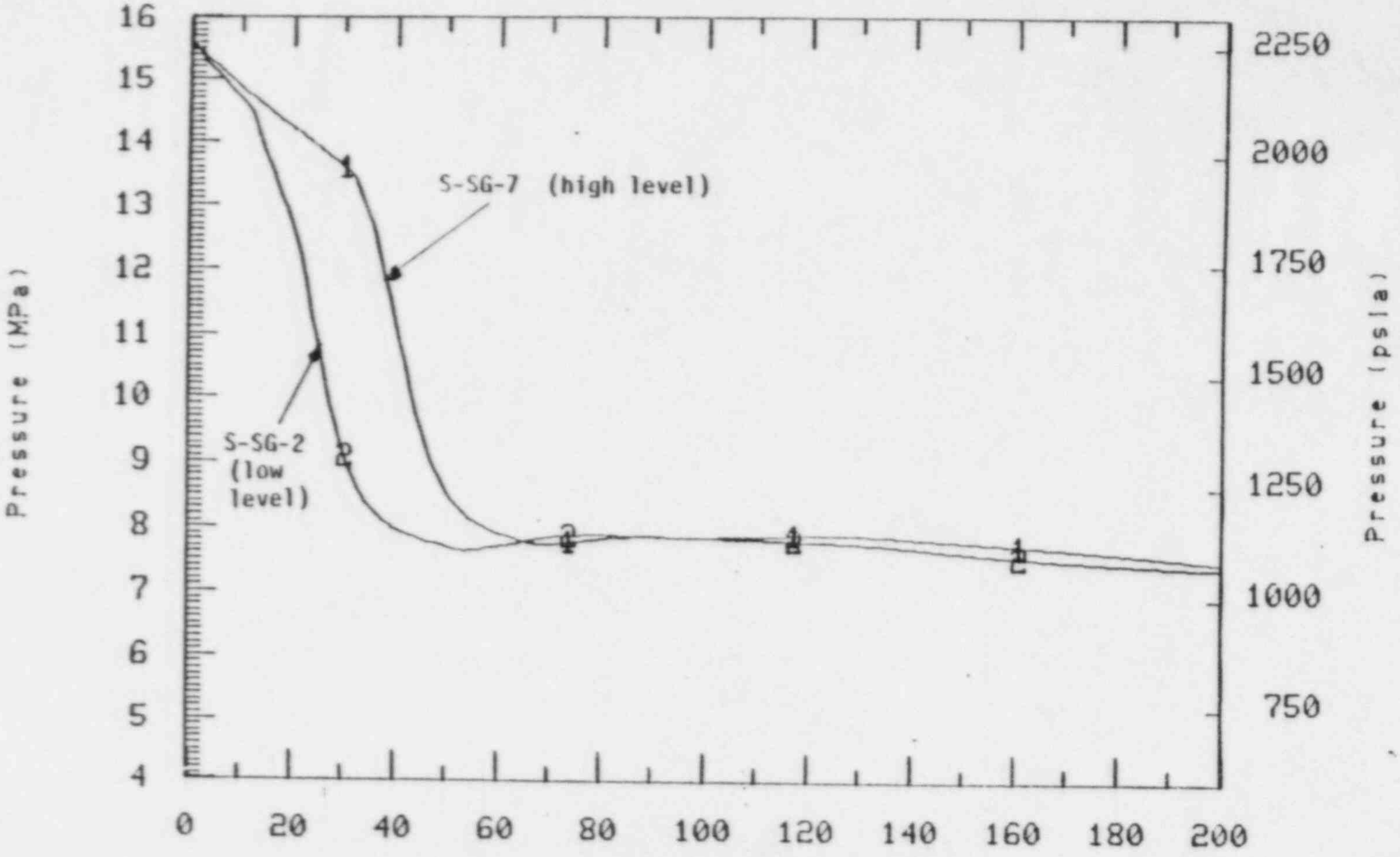

Figure 15. Comparison of system primary pressure response for two five tube rupture transients with different initial pressurizer levels (S-SG-2 and S-SG-7).

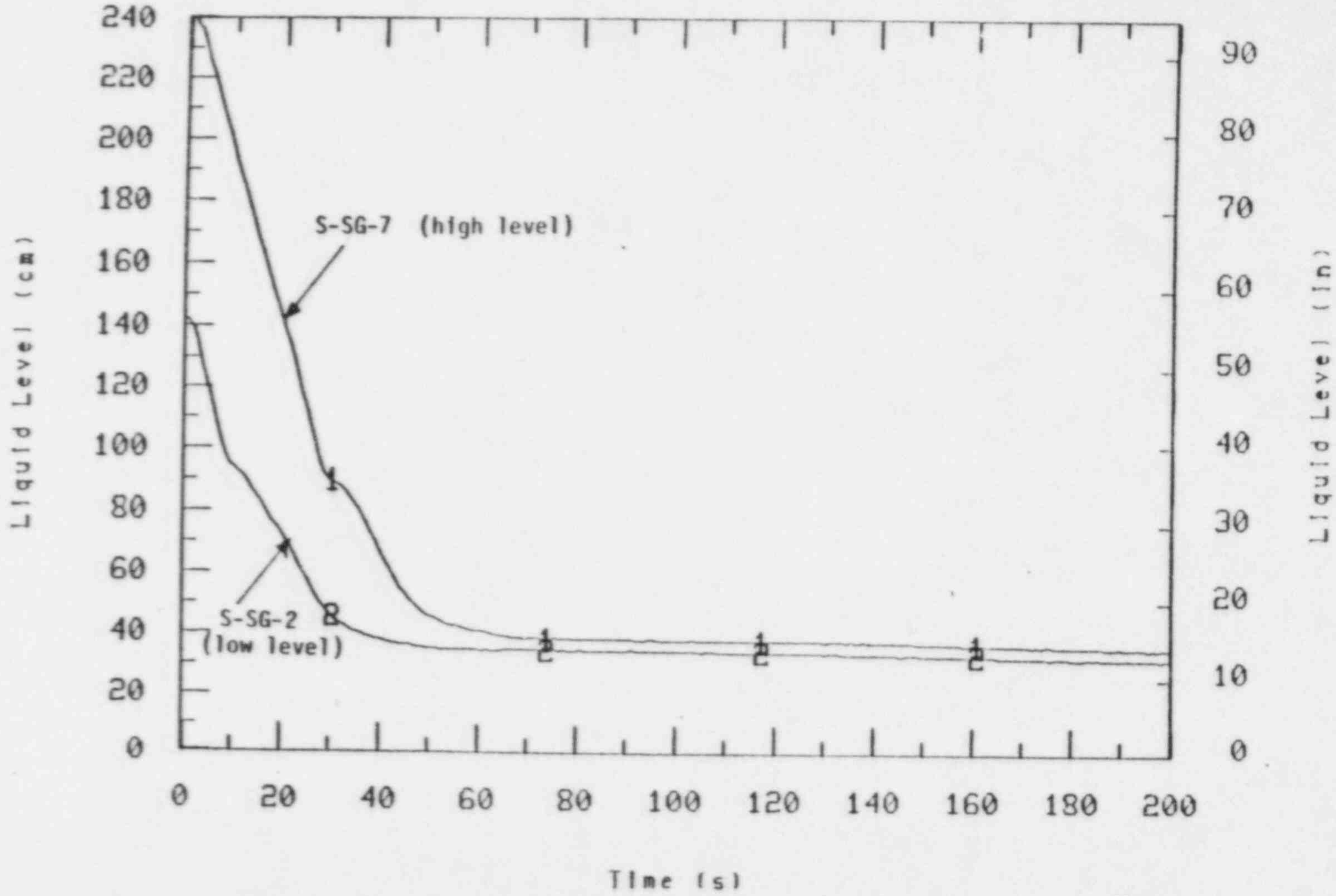

Figure 16. Comparison of pressurizer collapsed liquid level for two five tube rupture experiments with different initial pressurizer levels (S-SG-2 and S-SG-7).

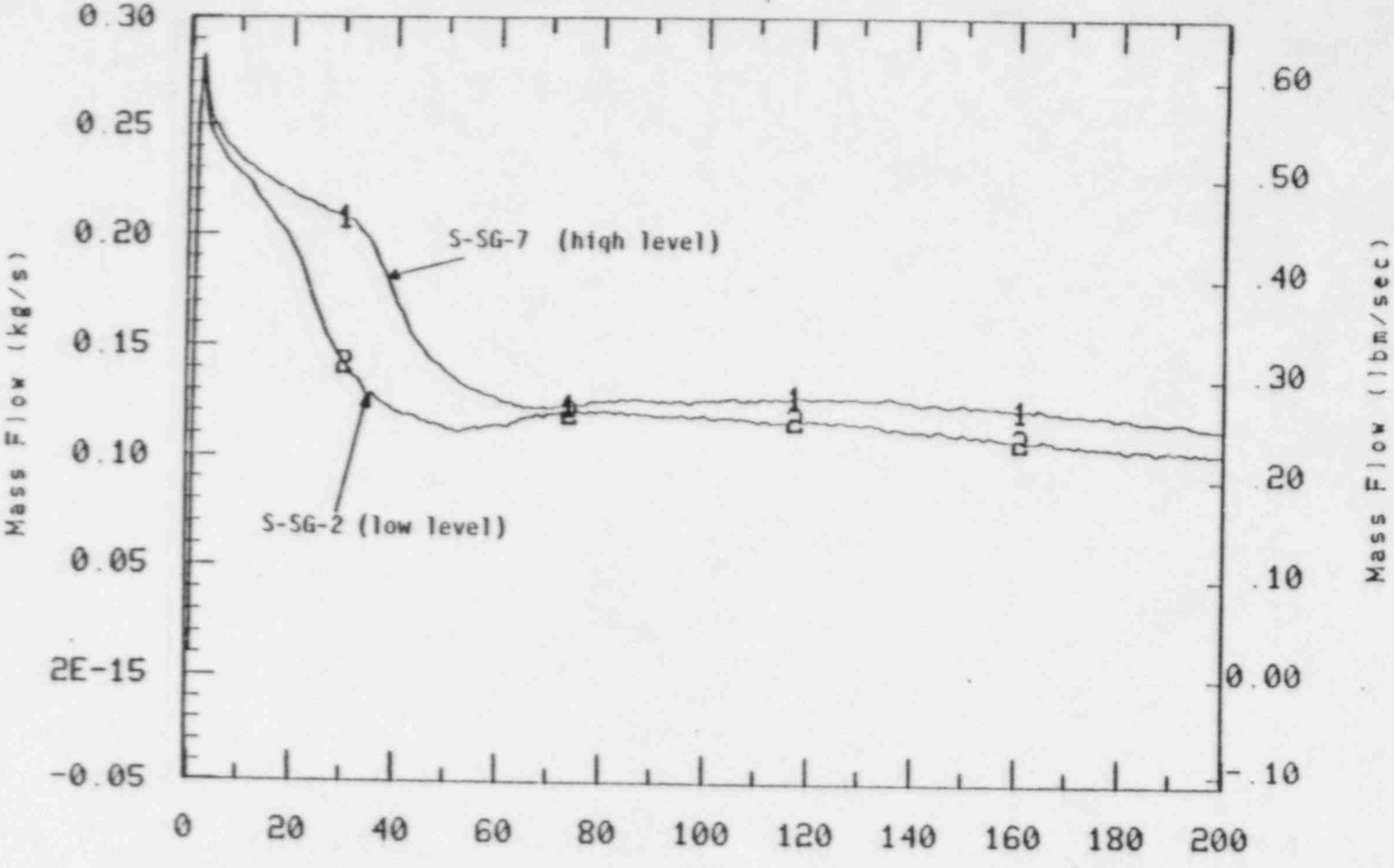

Figure 17. Comparison of break flow for two five tube rupture experiments with different initial pressurizer level (S-SG-2 and S-SG-7).

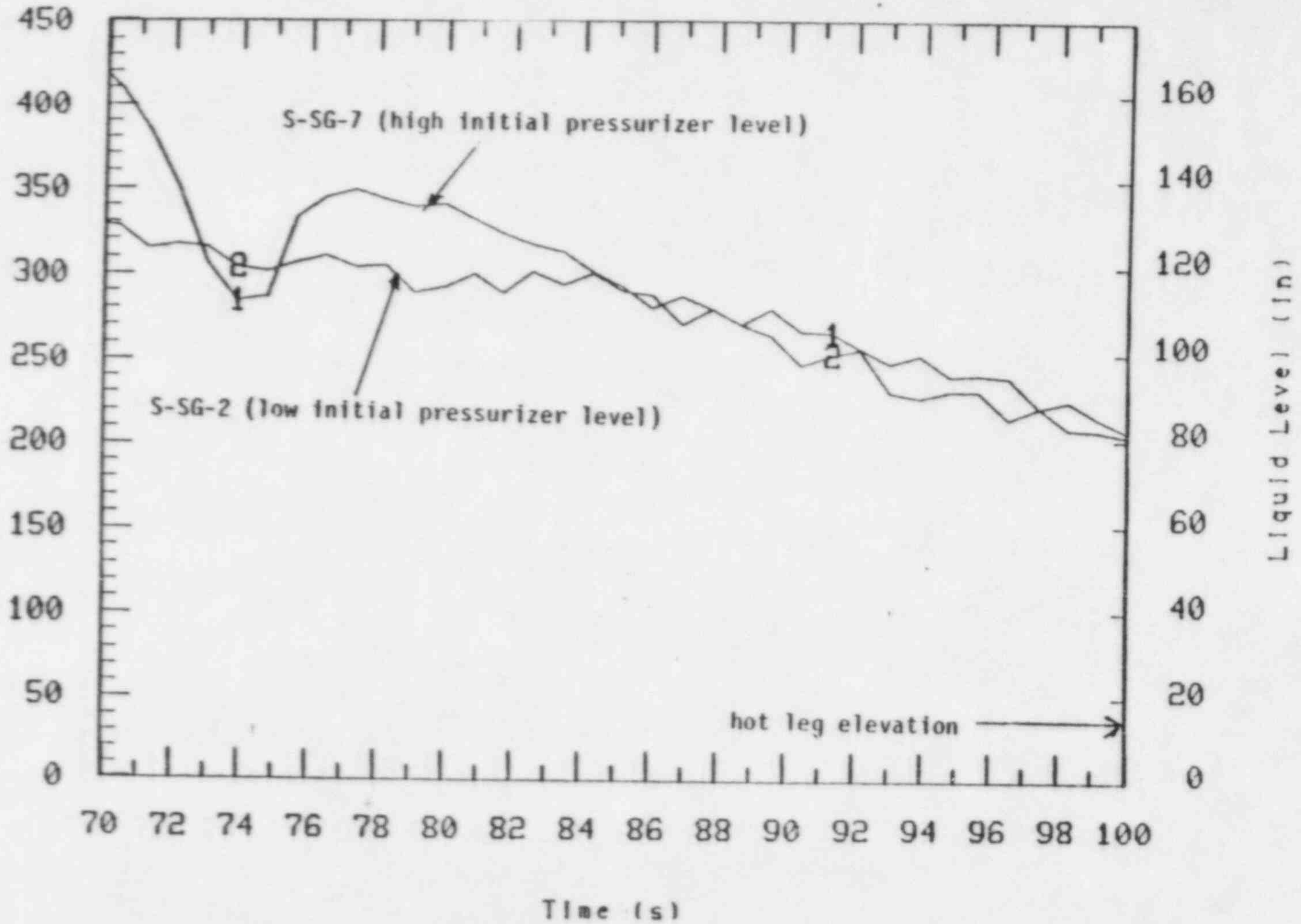

Figure 18. Comparison of vessel upper head collapsed liquid level during two five tube rupture transients with different initial pressure levels.

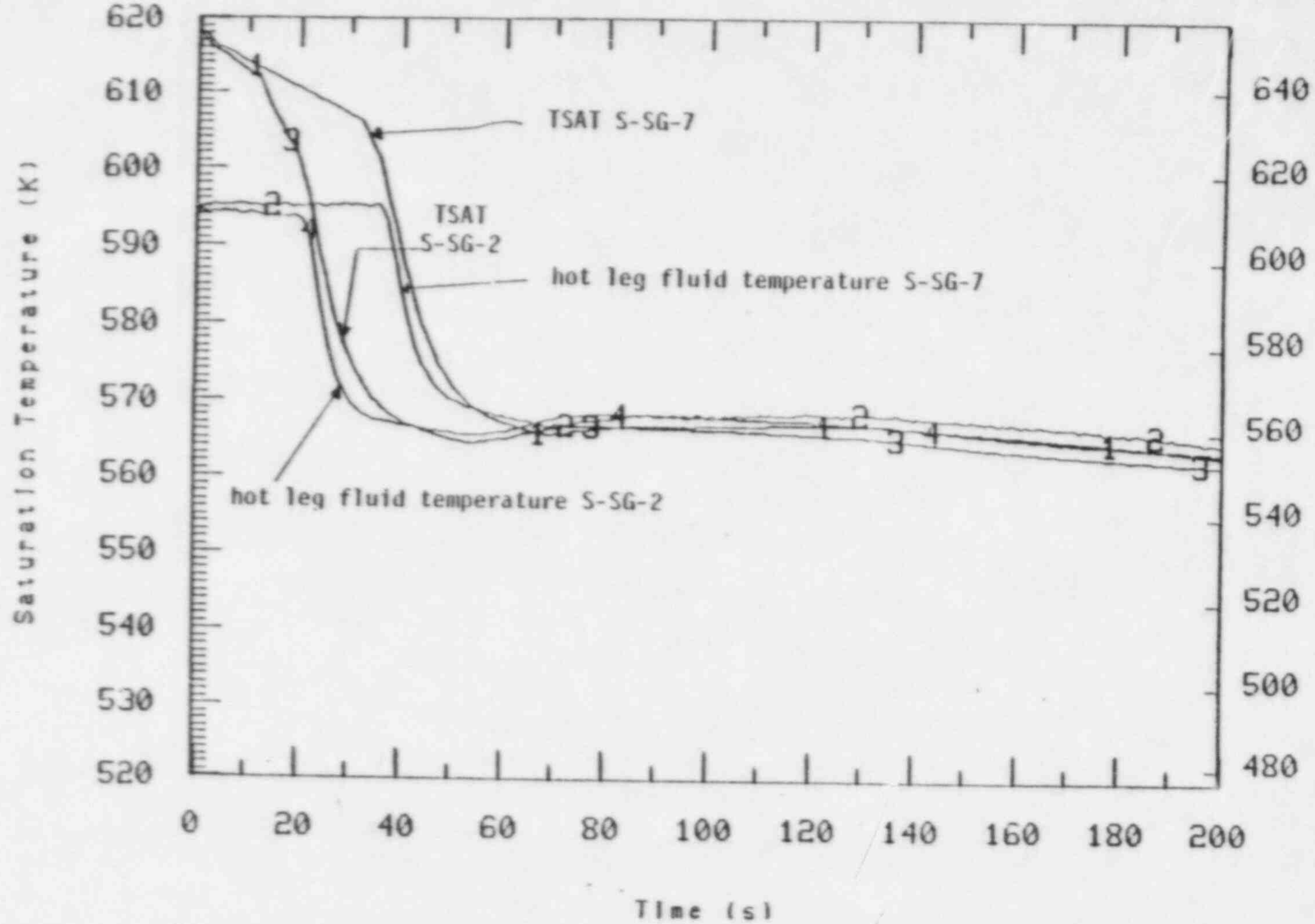

Figure 19. Comparison of hot leg fluid temperature and saturation temperatures for two five tube rupture transients with different initial pressurizer levels (S-SG-2 and S-SG-7).

extra energy delivered to the primary fluid during S-SG-7 was simply dissipated by continued normal steam and feed in the unaffected and affected loop secondaries prior to main steam isolation valve closure.

In summary, a higher initial pressurizer collapsed liquid level (increased liquid volume) resulted in a slower depressurization and a longer time to scram. However, the thermal-hydraulic state of the system after the first few hundred seconds was identical as the extra pressurizer mass for S-SG-7 simply left the primary system via the break flow. As a result, the system mass inventory after the first 200 s was essentially identical for S-SG-2 and S-SG-7. The extra core power delivered to the primary fluid for S-SG-7, compared to S-SG-2 was dissipated in the longer secondary steam and feed time prior to scram and resulted in similar primary fluid temperatures.

3.2 System Recovery in the Absence of SI

The Semiscale system was successfully recovered from a 5 tube rupture transient and put into a condition with first a stable and then increasing vessel liquid inventory without the use of SI. Recovery involved two separate operations: the first operation was unaffected loop steam generator feed and steam to maintain the primary pressure below the affected loop secondary ADV setpoint which effectively isolated the secondary from atmospheric discharge. The second operation involved using unaffected loop feed and steam to reduce the primary pressure below the affected loop secondary pressure which resulted in a backflow of affected loop secondary fluid into the primary system.

3.2.1 Use of Unaffected Loop Feed and Steam to Isolate the Affected Loop Generator

At 600 s an operator induced feed and steam action was initiated in the unaffected loop generator using ADV and auxiliary feed to reduce the primary pressure to 5.72 MPa (818 psig) (which is below the affected loop ADV setpoint of 5.85 MPa (848 psig)). This operator induced unaffected loop feed and steam action was successful in bringing the primary pressure

below the affected loop ADV setpoint as shown on Figure 20. Starting at 600 s the unaffected loop secondary pressure was rapidly reduced by opening the ADV. At about 800 s the ADV was closed because the primary pressure had achieved the desired value. Each rise and fall in unaffected loop secondary pressure shown in Figure 20 corresponds to an operator ADV operation to maintain the 5.72 ± 0.1 MPa (818 ± 15 psig) primary pressure. This operation was accomplished with a rising unaffected loop collapsed liquid level as shown in Figure 21. The auxiliary feedwater was operated to maintain the collapsed level between 800 and 1050 cm (314 and 413 in) and therefore was on throughout this period. The large depressions in collapsed secondary level shown on Figure 21 are due to velocity effects on the differential pressure measurement taps used to make the liquid level measurement. These effects occur whenever the ADV is opened. However the general filling trend of the unaffected loop secondary is clear.

Throughout the period when primary pressure was maintained at 5.72 MPa (818 psig) with unaffected loop secondary feed and steam (600-2848 s), the vessel mass inventory showed only a very slight downward trend as shown on Figure 22. In fact between 1800 and 2600 s the vessel level was relatively stable (about 20 cm below the top of the core). Immediately following ADV operation at 600 s there was a sudden marked decrease in vessel collapsed level that is attributed to flashing in the vessel and a general movement of fluid toward the increased condensation potential occurring in the unaffected loop secondary. Figure 20 shows a sudden drop in primary pressure occurring at 600 s due to the sudden increase in unaffected loop sink from the ADV operation. This sudden primary pressure decrease lead to flashing in the vessel as shown in Figure 23 which compares the various axial vessel density measurements. The density at all axial positions in the core simultaneously decreased as flashing occurred. Steam from the vessel rushed toward the increased condensation site in the unaffected loop secondary as shown on Figure 24 which shows the hot leg fluid volumetric flow increasing at 600 s. This same phenomena of flashing and movement of fluid toward the unaffected loop generator occurred to various degrees for every ADV operation as indicted on Figures 22-24. The decrease in vessel collapsed level accompanying the increase in secondary sink was not

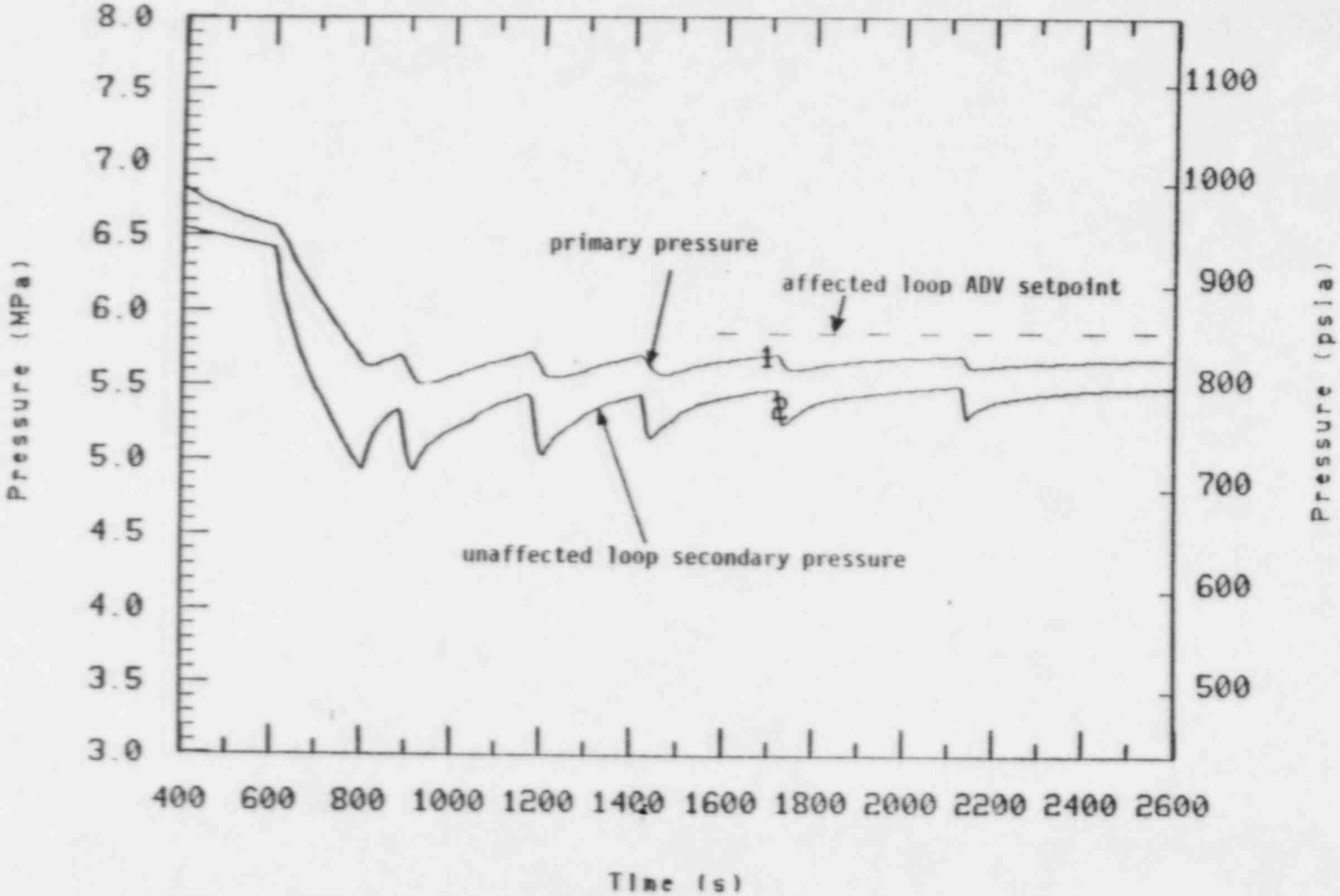

Figure 20. Comparison of primary and unaffected loop secondary pressure during secondary feed and steam recovery from a five tube rupture transient (S-SG-7).

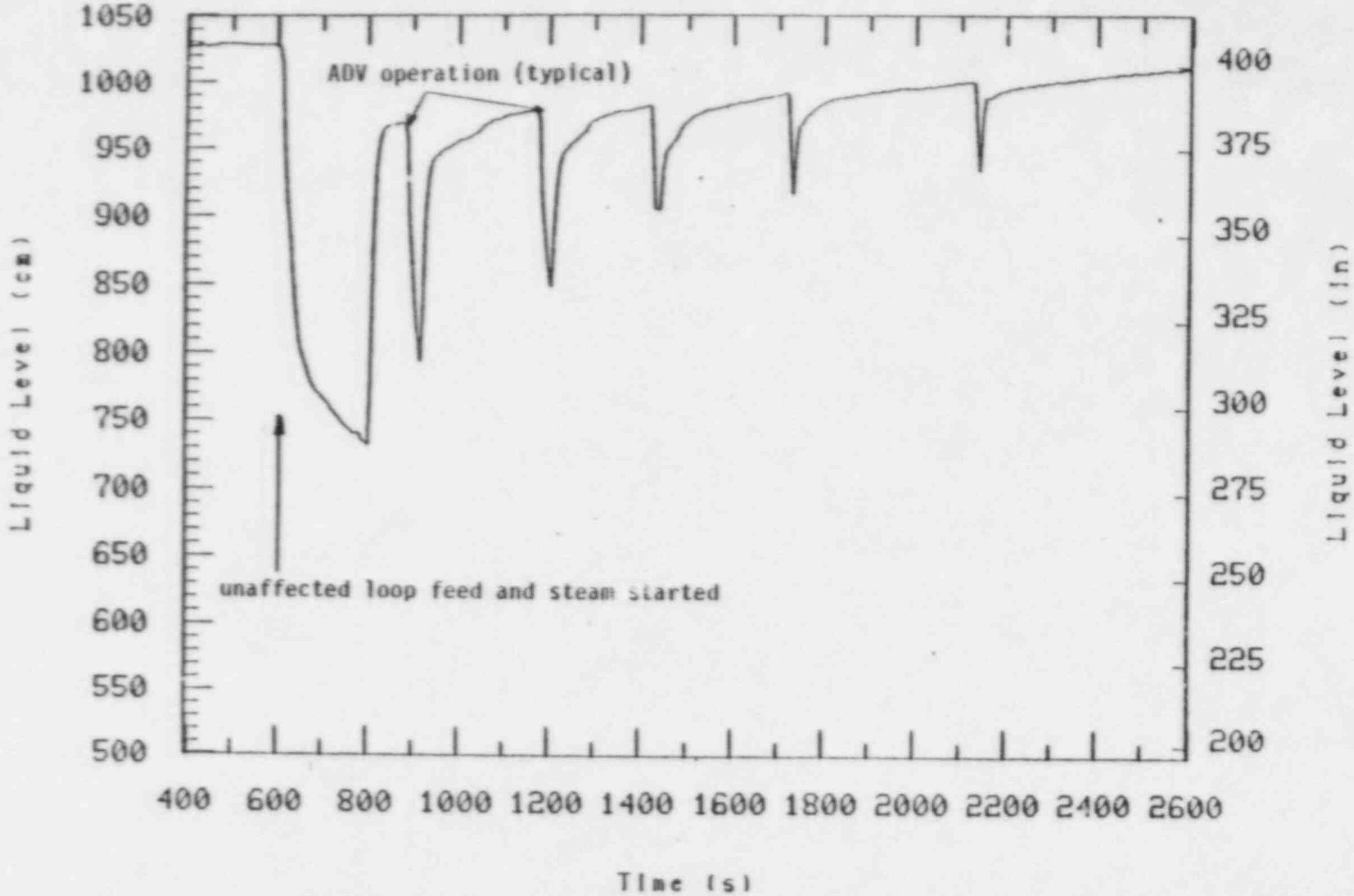

Figure 21. Unaffected loop secondary collapsed level during recovery of a five tube rupture transient (S-SG-7).

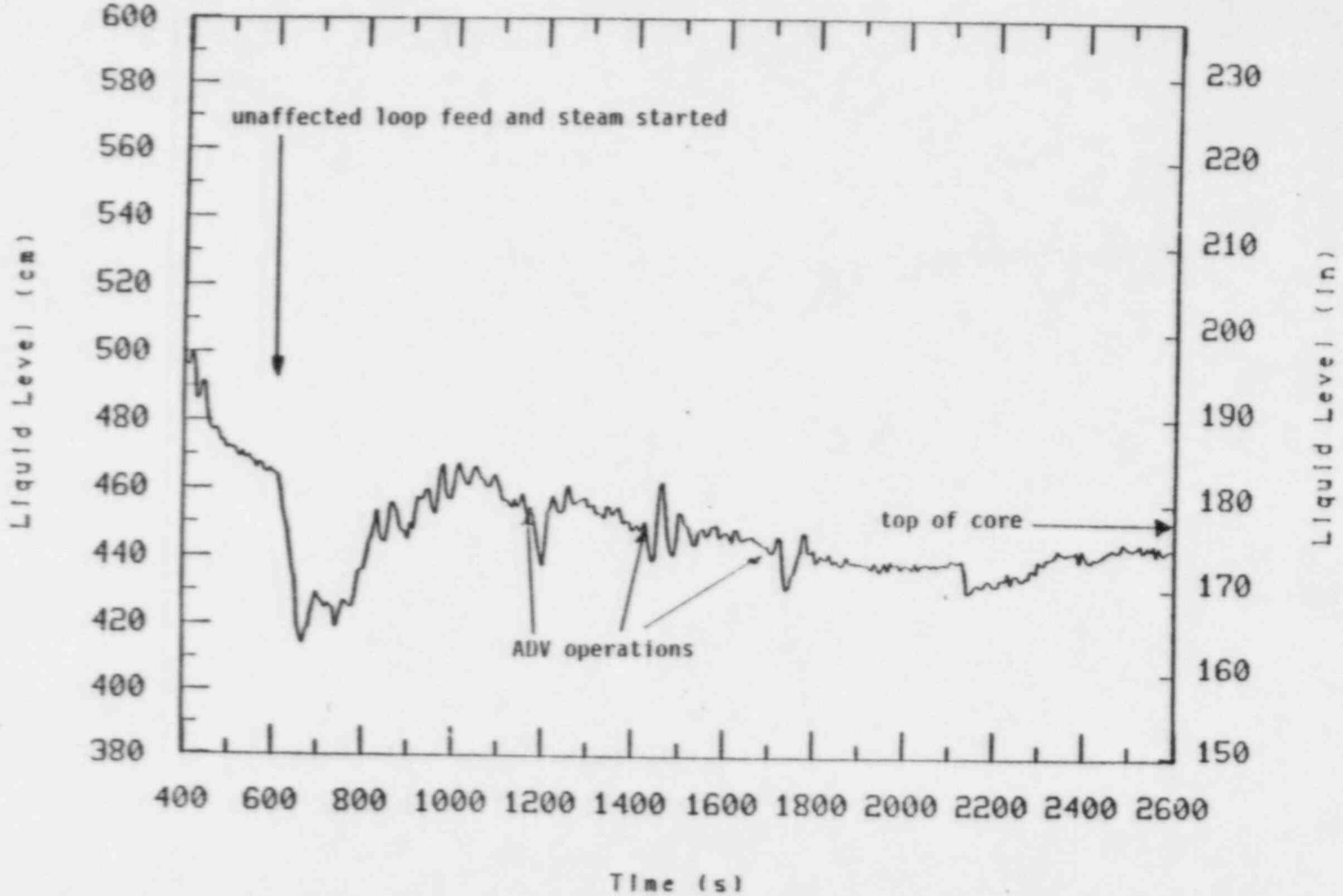

Figure 22. Lower vessel collapsed liquid level during recovery from a five tube rupture transient (S-SG-7).

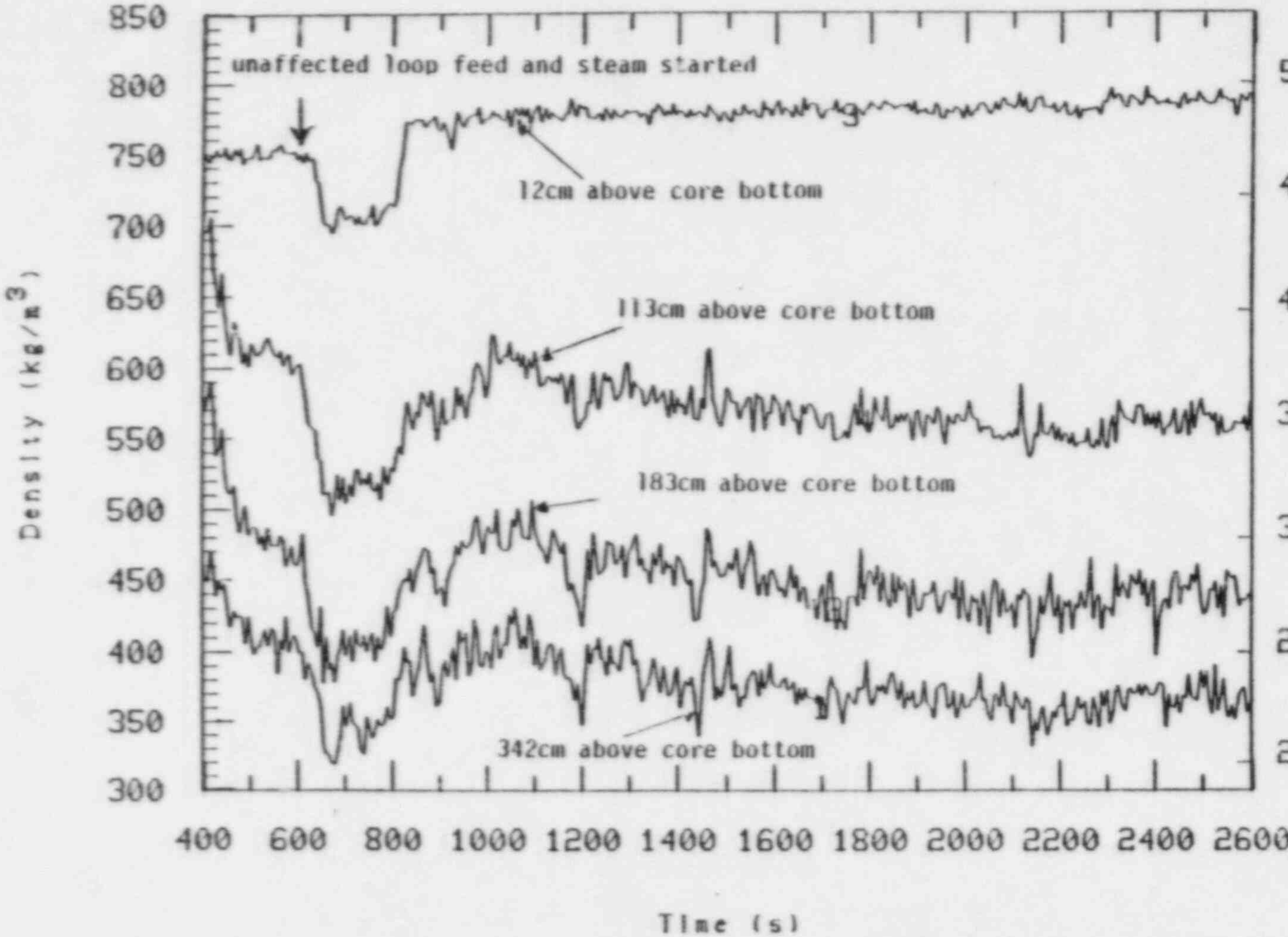

Figure 23. Comparison of axial density in the vessel core during recovery from a five tube rupture transient (S-SG-7).

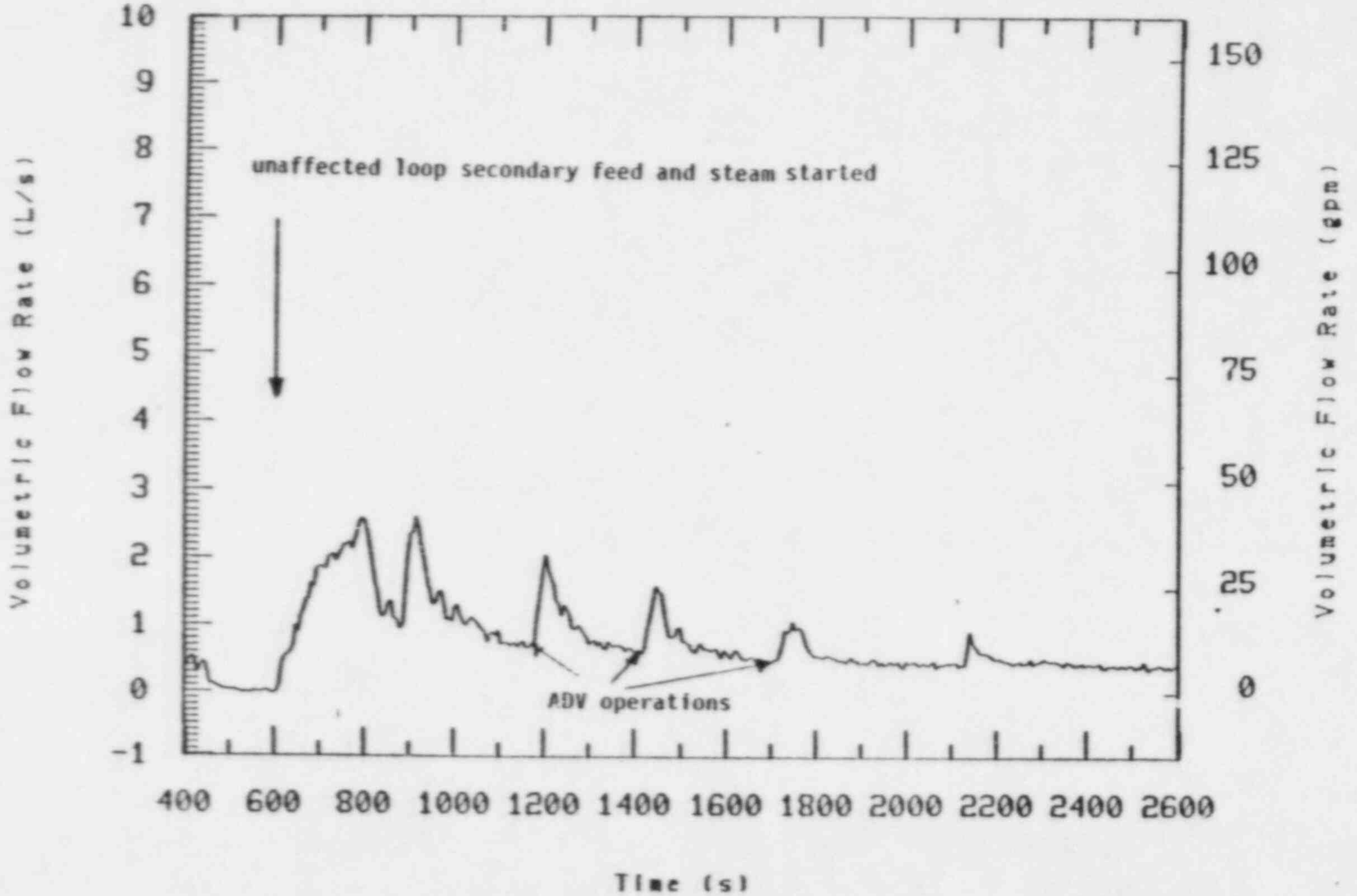

Figure 24. Hot leg fluid volumetric flow during recovery from a five tube rupture transient (S-SG-7).

sufficient to cause core rod heat-up as shown in Figure 25 which shows a heater rod temperature near the top of the core. The collapsed level in the core was sufficient to maintain a cooled core.

The break flow was positive for most of the time period between 600-2848 s as shown on Figure 26. This is because the differential pressure between primary and affected loop secondary was mostly positive as shown on Figure 27. The vessel level remained fairly stable during 1000 to 2840 s; therefore most of the positive break flow was supported by liquid in the affected loop pump suction as shown on Figure 28 which shows a generally decreasing suction level.

In summary, operator induced unaffected loop secondary feed and steam was a sufficient operation to isolate the affected loop steam generator from atmospheric discharge while maintaining a stable vessel collapsed liquid level (just below the top of the core). The operator induced feed and steam in the unaffected loop at 600 s caused a momentary increase in flashing in the core and condensation potential in the unaffected loop steam generator which caused a decrease in vessel collapsed liquid level. Each time the unaffected loop ADV was operated the same general phenomena occurred. However, the decrease in liquid level was small enough that no core rod heat-up occurred.

3.2.2 Use of Unaffected Loop Feed and Steam to Induce a Vessel Refill

Following the operator action to maintain the primary pressure below the affected loop ADV setpoint, a second operator action was initiated. When the unaffected loop secondary collapsed liquid level reached a predetermined value of 1050 cm (413 in),[a] unaffected loop feed and steam was commenced to maintain the primary pressure at 0.137 ± 0.068 MPa

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a. The collapsed level in the unaffected loop secondary was computed on line during the experiment without the benefit of corrections to the data such as pressure sensitivity; therefore the expected accuracy was ±20 cm (7.8 in) in controlling this level.

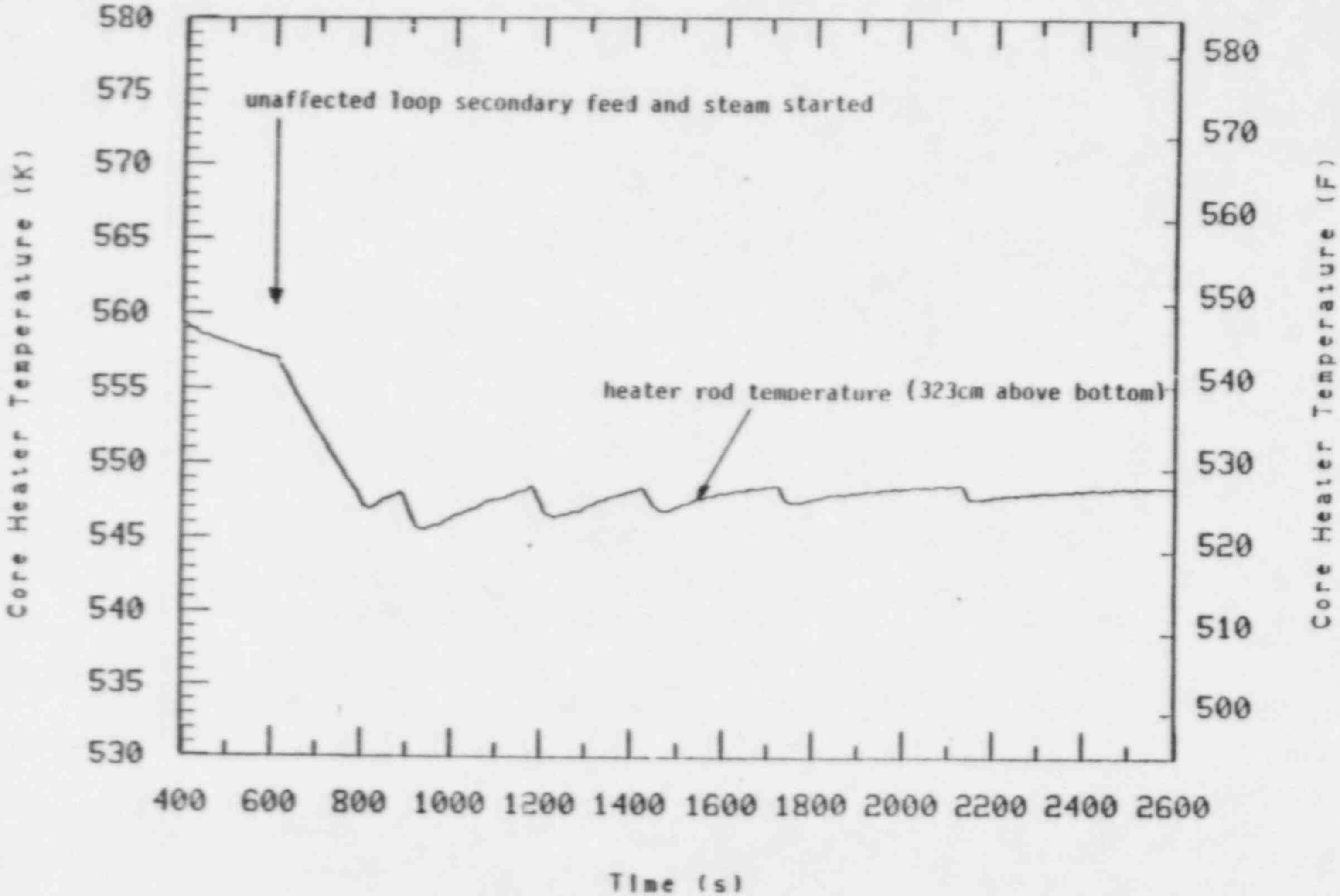

Figure 25. Upper core heater rod temperature during recovery from a five tube rupture transient (S-SG-7).

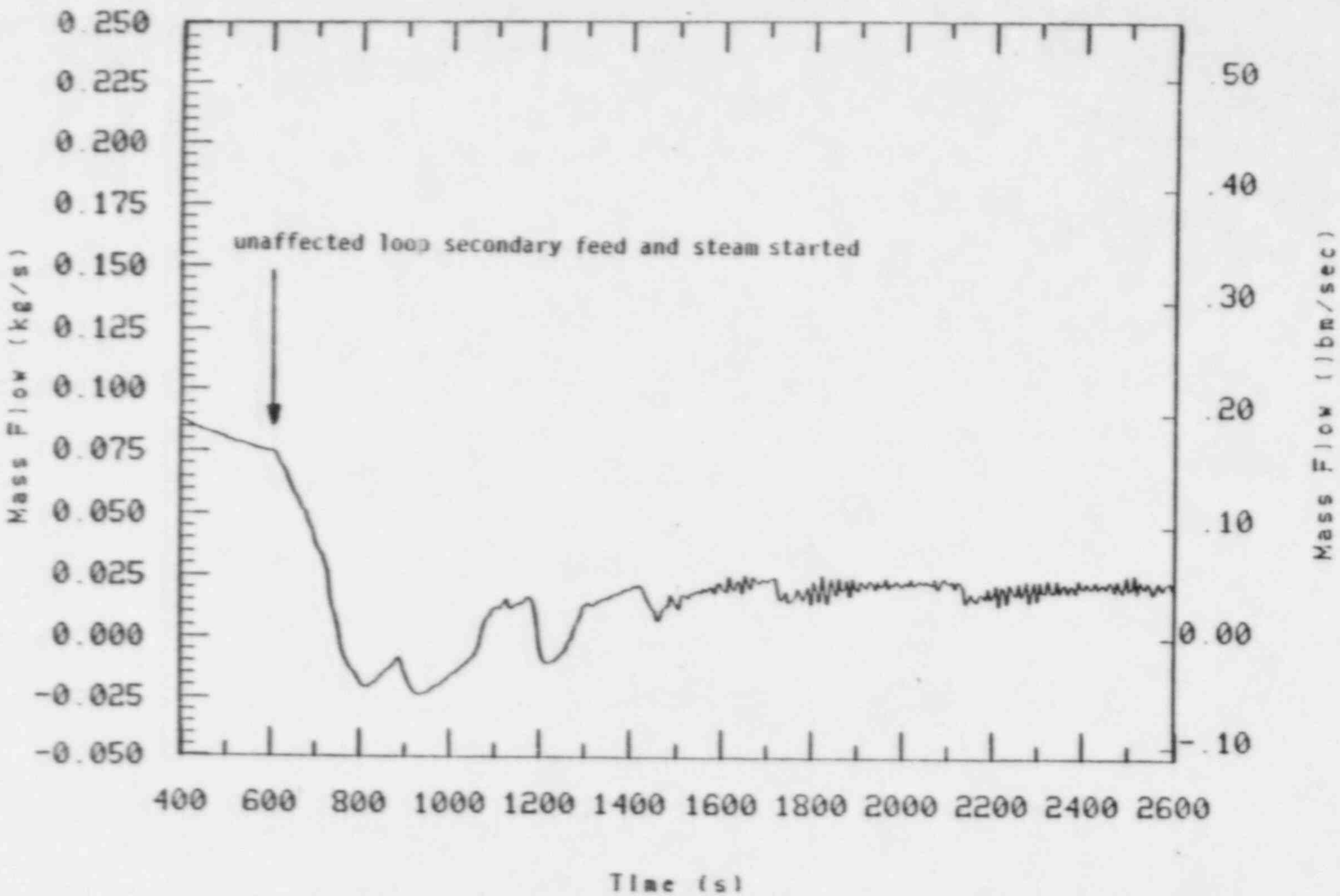

Figure 26. Break flow during recovery from a five tube rupture transient (S-SG-7).

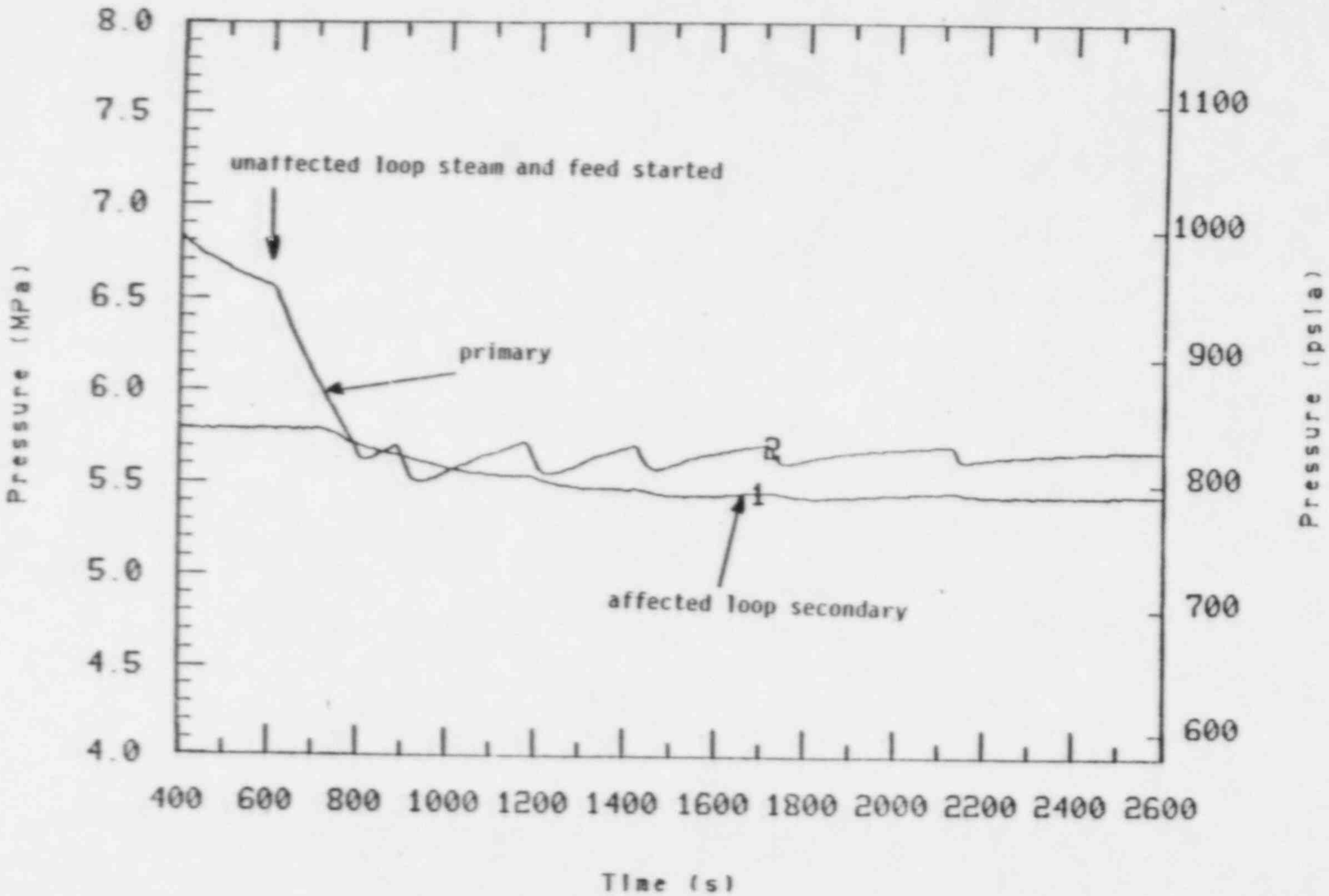

Figure 27. Comparison of primary pressure and affected loop secondary pressure during recovery of a five tube rupture transient (S-SG-7).

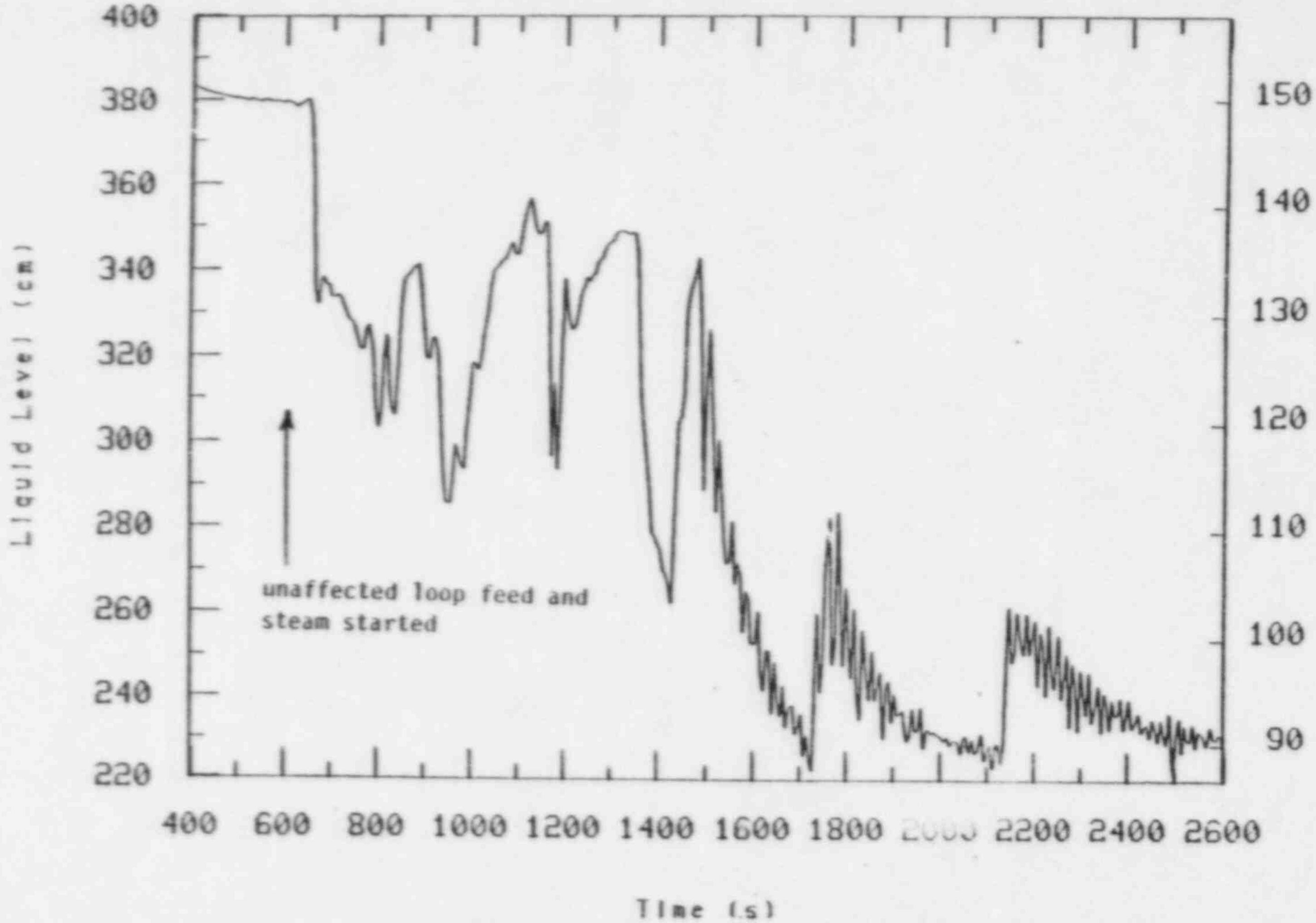

Figure 28. Affected loop pump suction collapsed liquid level during recovery from a five tube rupture transient (S-SG-7).

(20 ± 10 psia) below the affected loop secondary pressure. This operation was performed to stimulate back flow from the affected loop generator through the break to the primary system.

The use of unaffected loop steam and feed to stimulate back flow from the affected loop secondary was successful in inducing a vessel refill starting at about 2848 s. The unaffected loop feed and steam was initiated when the unaffected loop secondary reached the predetermined high level and resulted in a rapid decrease in primary pressure below the affected loop pressure as shown on Figure 29. The primary pressure was maintained by the operator below the affected loop secondary pressure thus inducing a back flow of affected loop secondary fluid into the primary system as shown on Figure 30. Figure 31 shows that the affected loop secondary collapsed liquid level decreased slightly due to the back flow and the unaffected loop level depleted throughout the time period due to a combination of steam (through the ADV) and feed (through auxiliary feedwater). The affected loop secondary to primary back flow induced a vessel refill as shown on Figure 32. In fact, the refill operation was sufficient to fill the vessel to near the vessel cold leg elevation which is 130 cm above the top of the core. Initially, the unaffected loop feed and steam operation caused a condensation induced vessel collapsed level depression as shown in Figure 32 starting at about 2850 s. The reasons for this vessel level depression are though to be similar to those discussed in Section 3.2.1 when the first unaffected loop feed and steam operation commenced at 600 s. Again, as with the 600 s vessel level depression, no core rod temperature heatup occurred as shown on Figure 33. As the back flow continued, the vessel level increased until the termination criteria was achieved at about 5500 s. The test was terminated when the vessel level showed a stable or increasing vessel collapsed level for 30 minutes. This increasing vessel collapsed level was accomplished without completely depleting the unaffected loop secondary level (see Figure 31) (at 5500 s the unaffected loop collapsed level indicates approximately half full).

In summary, an operator induced unaffected loop feed and steam operation was successful in refilling the vessel to near the vessel cold leg elevation. This refill was accomplished while maintaining the

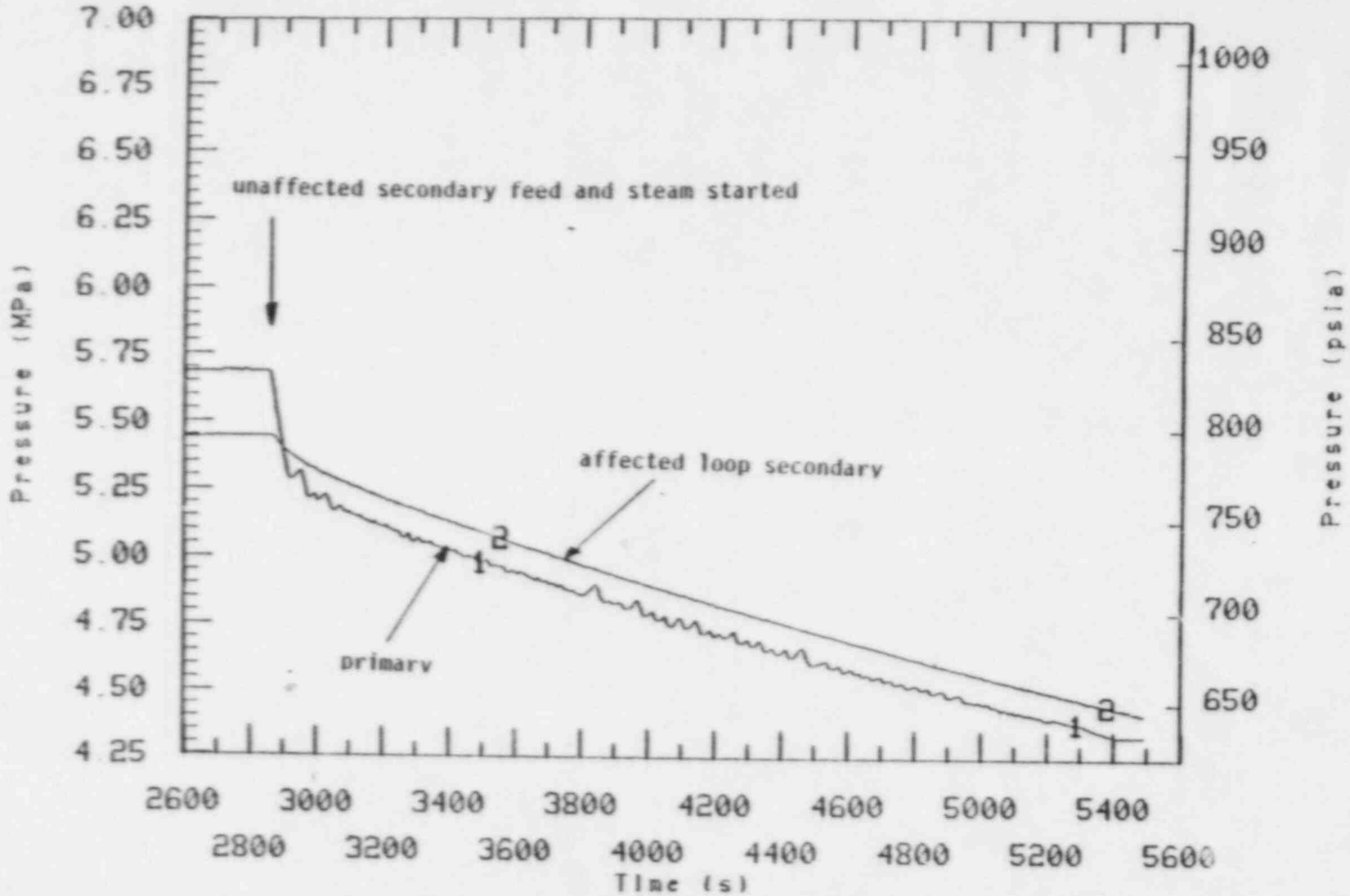

Figure 29. Comparison of primary and affected loop secondary pressure during recovery from a five tube rupture transient (S-SG-7).

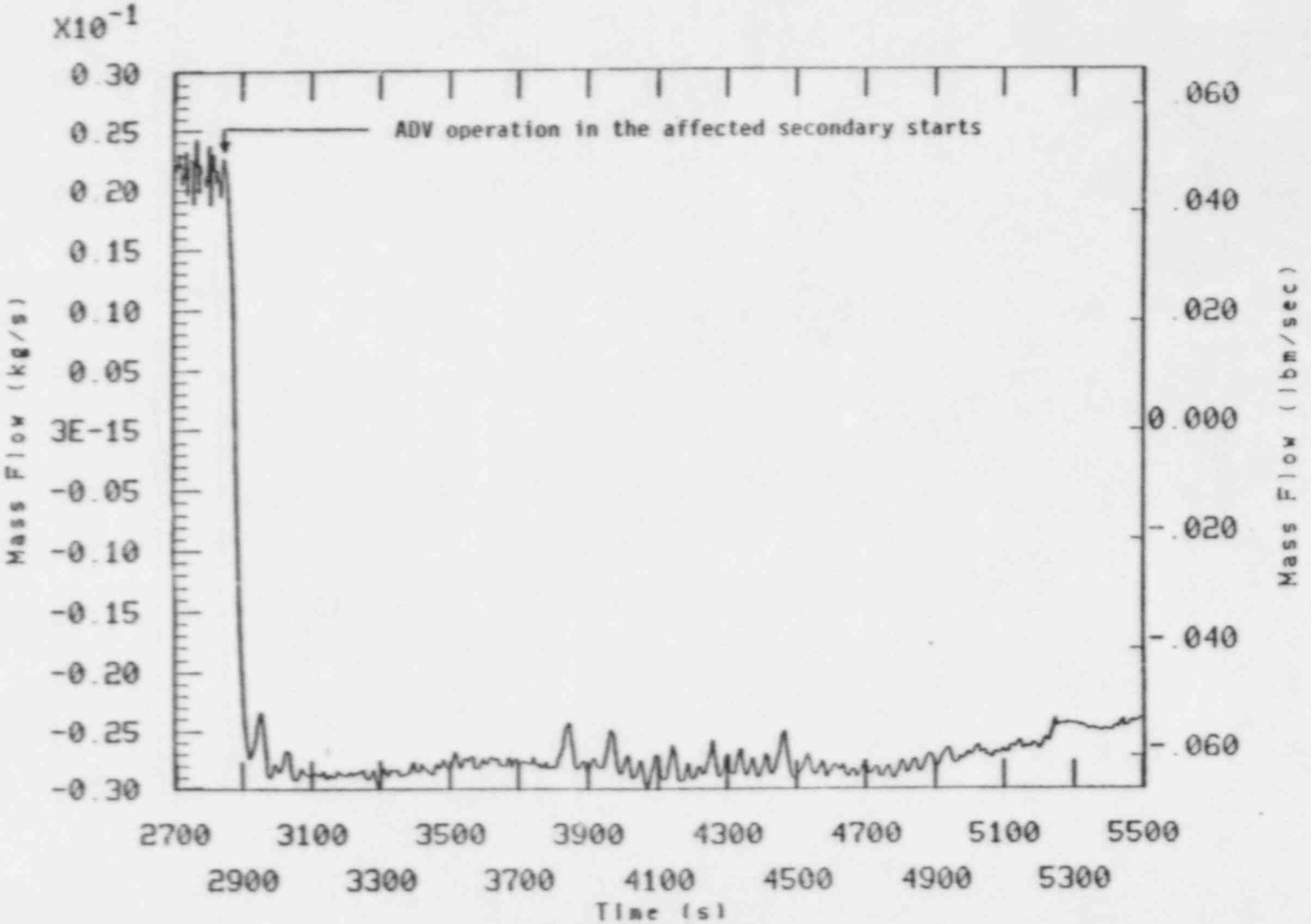

Figure 30. Break flow during recovery from a five tube rupture transient (S-SG-7).

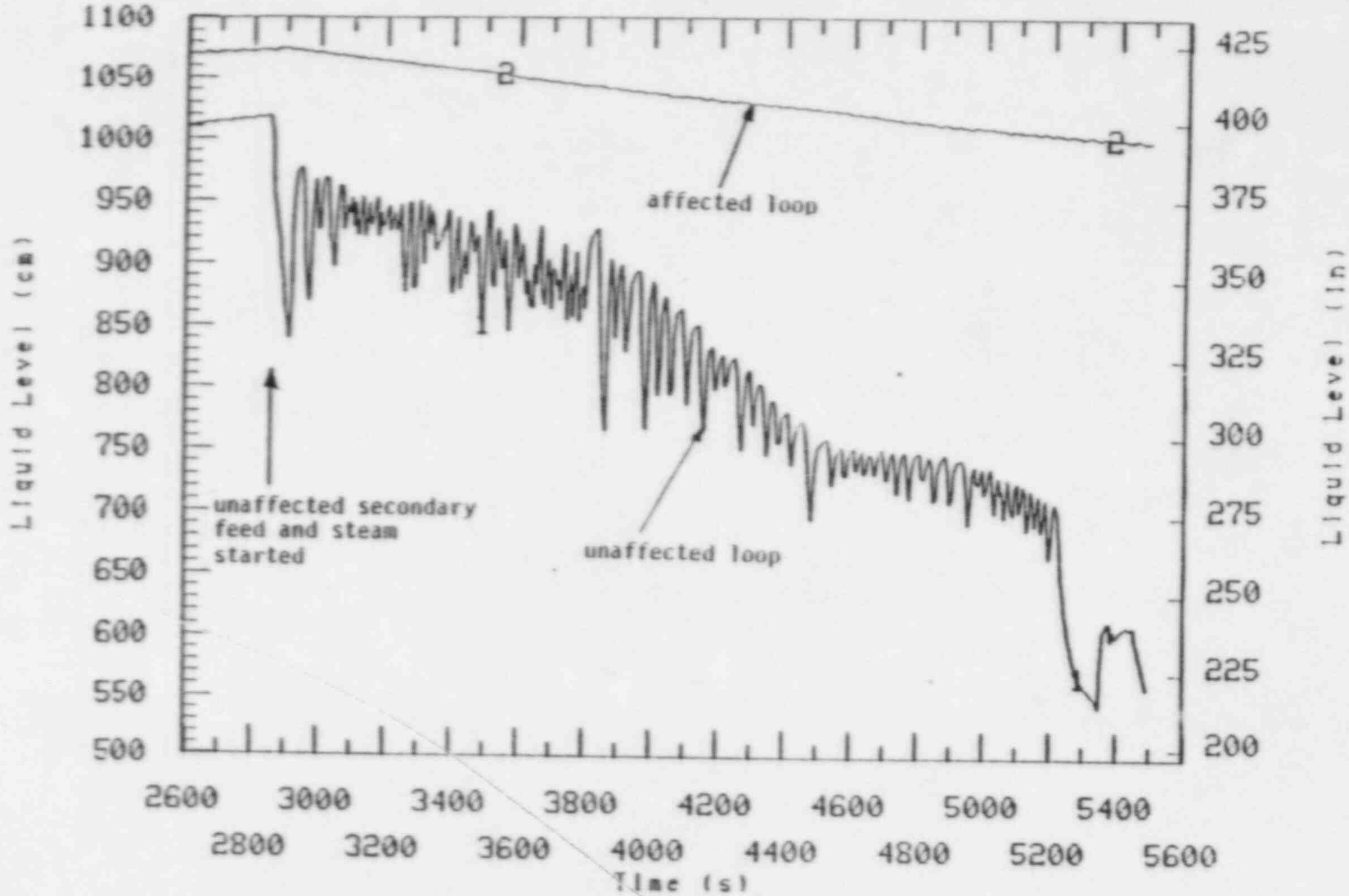

Figure 31. Comparison of affected and unaffected loop secondary level during recovery from a five tube rupture transient (S-SG-7).

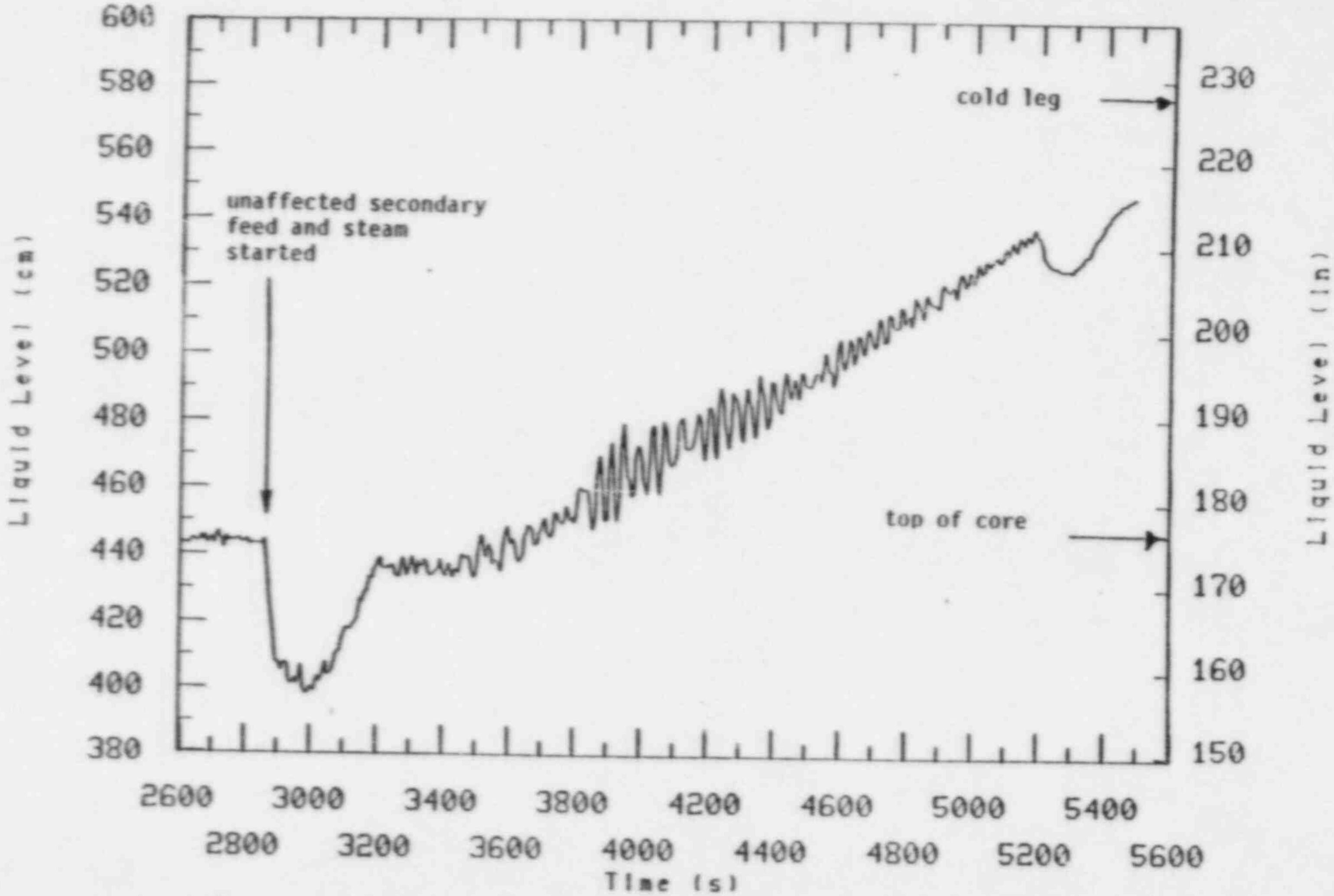

Figure 32. Lower vessel collapsed liquid level during recovery from a five tube rupture transient (S-SG-7).

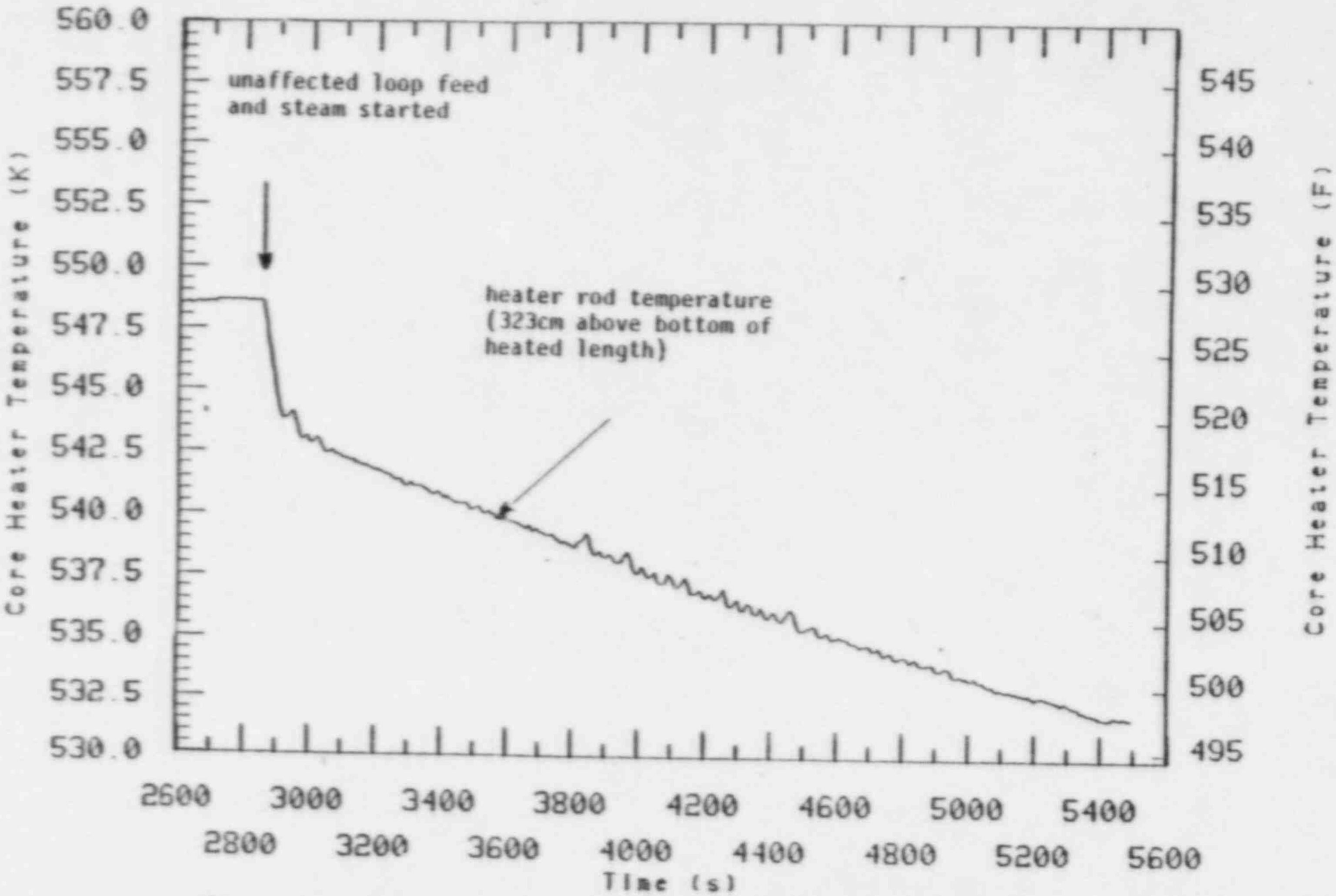

Figure 33. Upper core heater rod temperature during recovery from a five tube rupture transient (S-SG-7).

unaffected loop secondary collapsed level nearly half full. The initial operation to lower the primary pressure below the affected loop secondary pressure caused a momentary vessel level depletion. However the level depression was minor enough to preclude a core rod heatup.

4. COMPARISON TO THE PAD CALCULATION

This section compares the test results to the PAD calculation for Test S-SG-7. Included in this section is a comparison of the calculation and test results for the first 600 s of the experiment when only automatic plant responses were functioning, as well as the calculated and experimental results during the recovery phase. Table 3 compares the actual and calculated initial conditions and Table 4 compares the actual and calculated sequence of events.

4.1 Operator Diagnostic Period

The test was initiated by opening the break valve at 0 s. As shown in Figure 34, the measured primary coolant system depressurization rate during the first 30 s was slightly lower than predicted. One factor contributing to that discrepancy was the overprediction of break flowrate in the early portion of the transient ($t < 30$ s). It can be seen in Figure 35 that not only was the break flowrate overpredicted by as much as 16%, but that the calculated flowrate reached its peak value approximately two seconds sooner than actual. The slightly lower measured primary coolant system depressurization rate was the reason that scram, which had been predicted to occur at 27.3 s, was delayed until 32.0 s. (Scram occurred on a low pressurizer pressure signal at 13.1 MPa (1900 psia).)

The rate of primary coolant system depressurization increased after scram in both the calculation and the experiment due to the decreased power generation and decrease in vapor generation rate as the liquid/vapor interface fell from the pressurizer into the much smaller surge line (Figure 36). SIS, based on a pressurizer pressure of 12.51 MPa (1814 psia), was reached at 35.0 s compared to 30 s in the calculation. At SIS, the primary coolant pumps began to coastdown and the auxiliary feedwater system was enabled. HPIS was assumed to be unavailable. Between scram and the time that the pumps coasted to a stop--the pumps stopped at 65 and 70 s for the calculation and experiment respectively--good agreement was achieved between the calculated and measured primary coolant system depressurization rates.

TABLE 3. COMPARISON OF ACTUAL AND CALCULATED INITIAL CONDITIONS

Parameter	Calculated	Measured
Pressurizer pressure	15.49 MPa (2247 psia)	15.45 MPa (2240 psia)
Core temperature differential	37.0K (66.5°F)	38.7K (69.6°F)
Initial core power	2.0 MW	1.99 MW
Pressurizer liquid volume	0.0102 m^3 (0.36 ft^3)	0.0091 m^3 (0.32 ft^3)
SG secondary pressure		
Affected loop	5.53 MPa (802 psia)	5.58 MPa (809 psia)
Unaffected loop	5.53 MPa (802 psia)	5.49 MPa (796 psia)
SG secondary mass		
Unaffected loop	101.96 kg (224.8 lbm)	178 kg (293 lbm)
Affected loop	100.8 kg (222.2 lbm)	109 kg (240 lbm)

TABLE 4. COMPARISON OF CALCULATED AND ACTUAL SEQUENCE OF EVENTS

	Calculated Time (s)	Actual Time (s)
Transient initiation	0	0
SCRAM	27.3	32
SIS	30	35
Recovery initiated	600	600
1050 cm (34.4 ft) unaffected loop generator level	4650	2848
Termination	5100[a]	5500

a. The calculation was terminated when the vessel liquid level reached the top of the heated length.

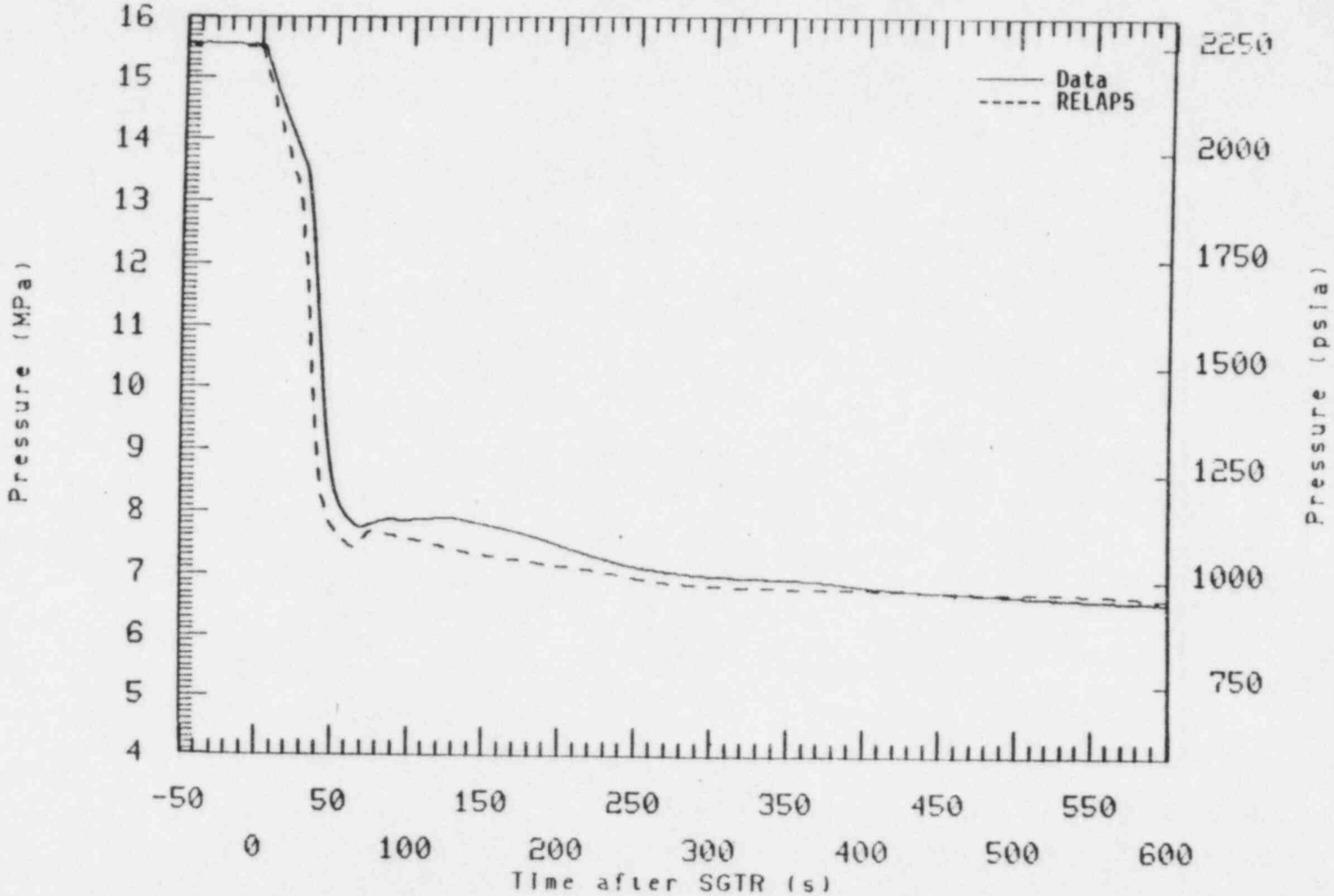

Figure 34. Pressurizer pressure comparison 0 - 600s.

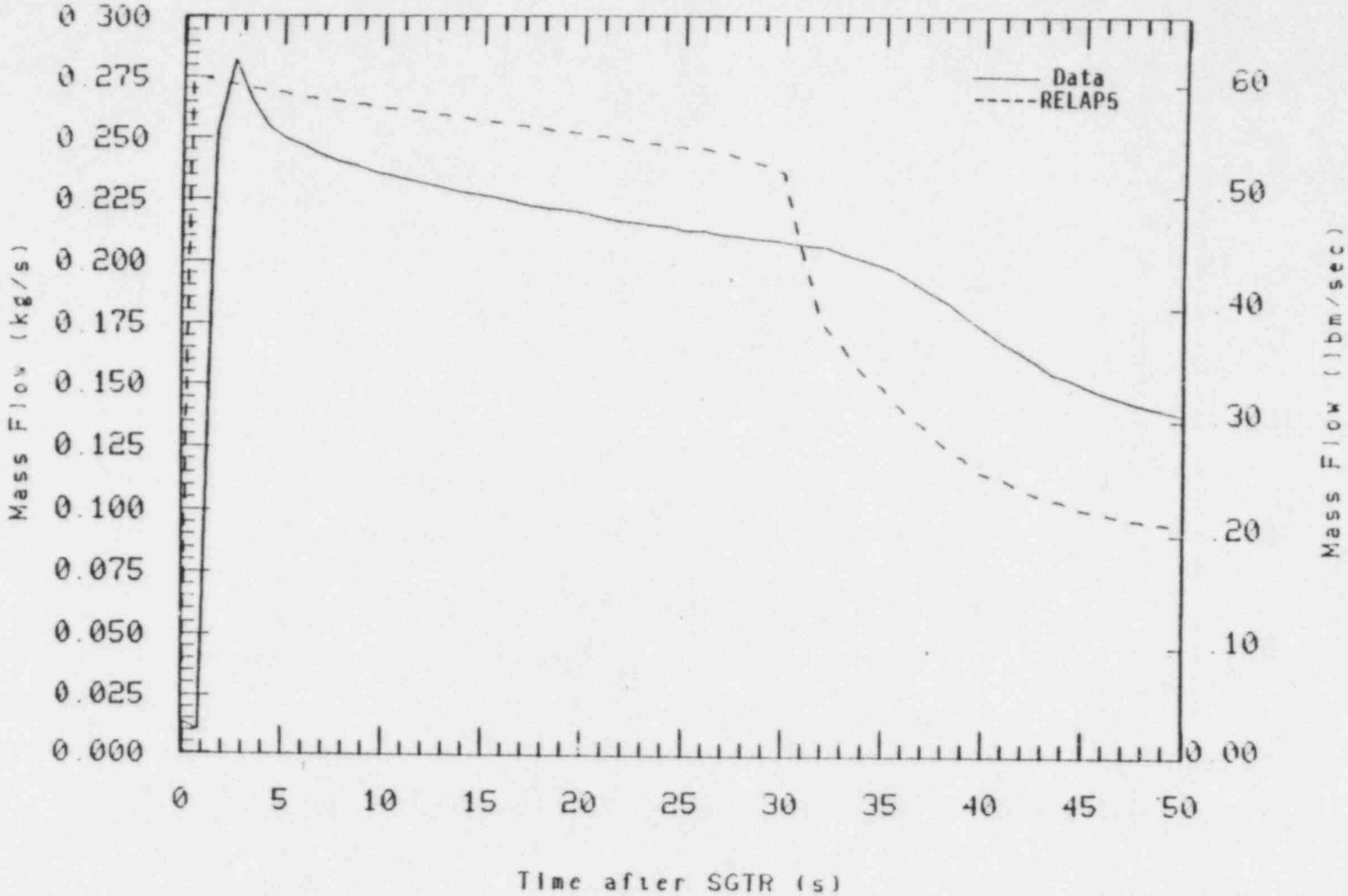

Figure 35. Break flowrate comparison 0 - 50s.

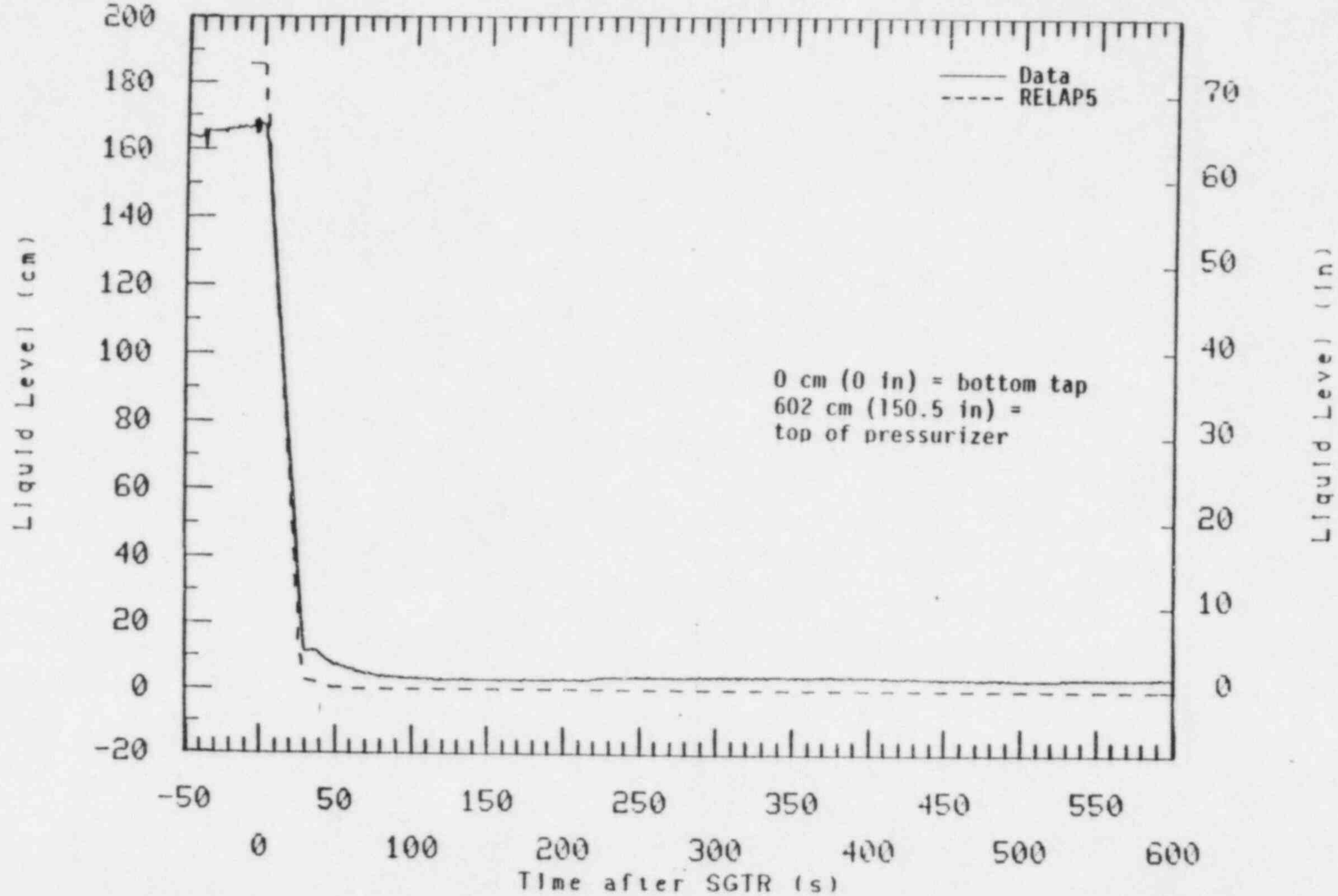

Figure 36. Pressurizer interface liquid level comparison 0 - 600s.

Following coastdown of the pumps, there was a small (0.2 MPa (29 psid)) increase in primary coolant system pressure as natural circulation was established. That increase was well predicted in magnitude but not in duration. The calculation indicated that the primary coolant system pressure would begin to decrease again approximately 20 s after pump coastdown was complete. However, about 55 s was actually required and a primary coolant system pressure divergence between the calculation and experiment occurred. That difference gradually narrowed until by 450 s the calculated and measured primary coolant system pressures crossed. Good agreement was maintained throughout the remainder of the 600 s diagnostic period.

Secondary pressure response is shown in Figure 37. The main steam valves were closed at scram which explains the approximate 5 s difference in predicted and measured secondary pressurization. Note that the pressure in the affected steam generator did begin to increase at 0 s due to the break; however the major increase occurred due to closure of the main steam valves. Also note that the predicted pre-scram pressure increase in the affected steam generator was greater than measured, again owing to the aforementioned primary coolant system depressurization rate variance.

Following closure of the main steam valves, both steam generators pressurized, as predicted, to their respective relief valve setpoints and remained there throughout the remainder of the 600 s diagnostic period. The calculation had predicted that the second stage of the ASG relief valve would open; however that did not occur. This is probably due to inadequate modeling of the ASG metal mass. It can also be seen that the ASG relief valve operated about 0.1 MPa (14.5 psid) low during the test.

Overall system response is best shown in the primary coolant system and secondary pressure overlay, Figure 38. As discussed earlier, good agreement was obtained throughout the first 600 s of the transient.

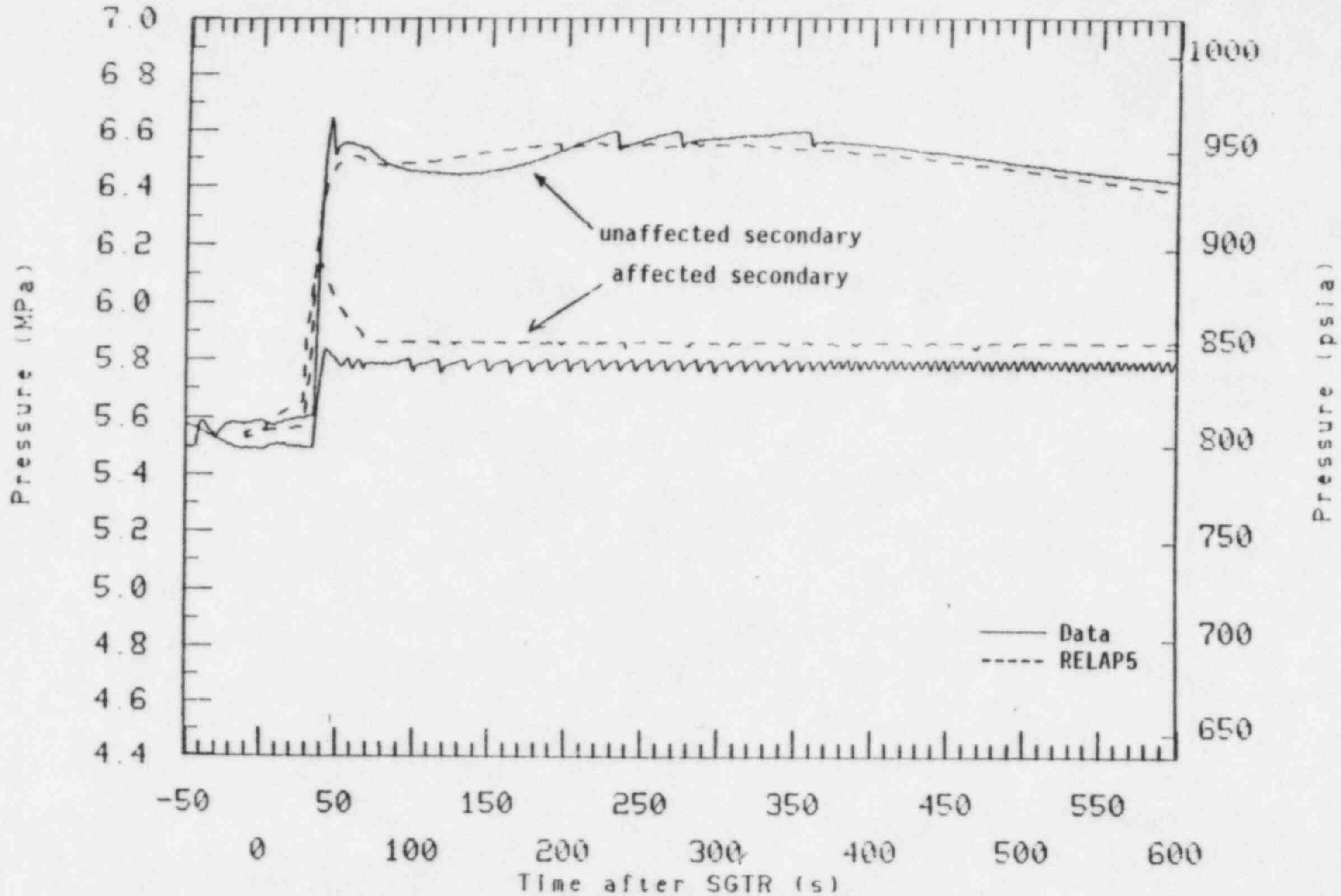

Figure 37. Secondary pressure comparison
0 - 600s.

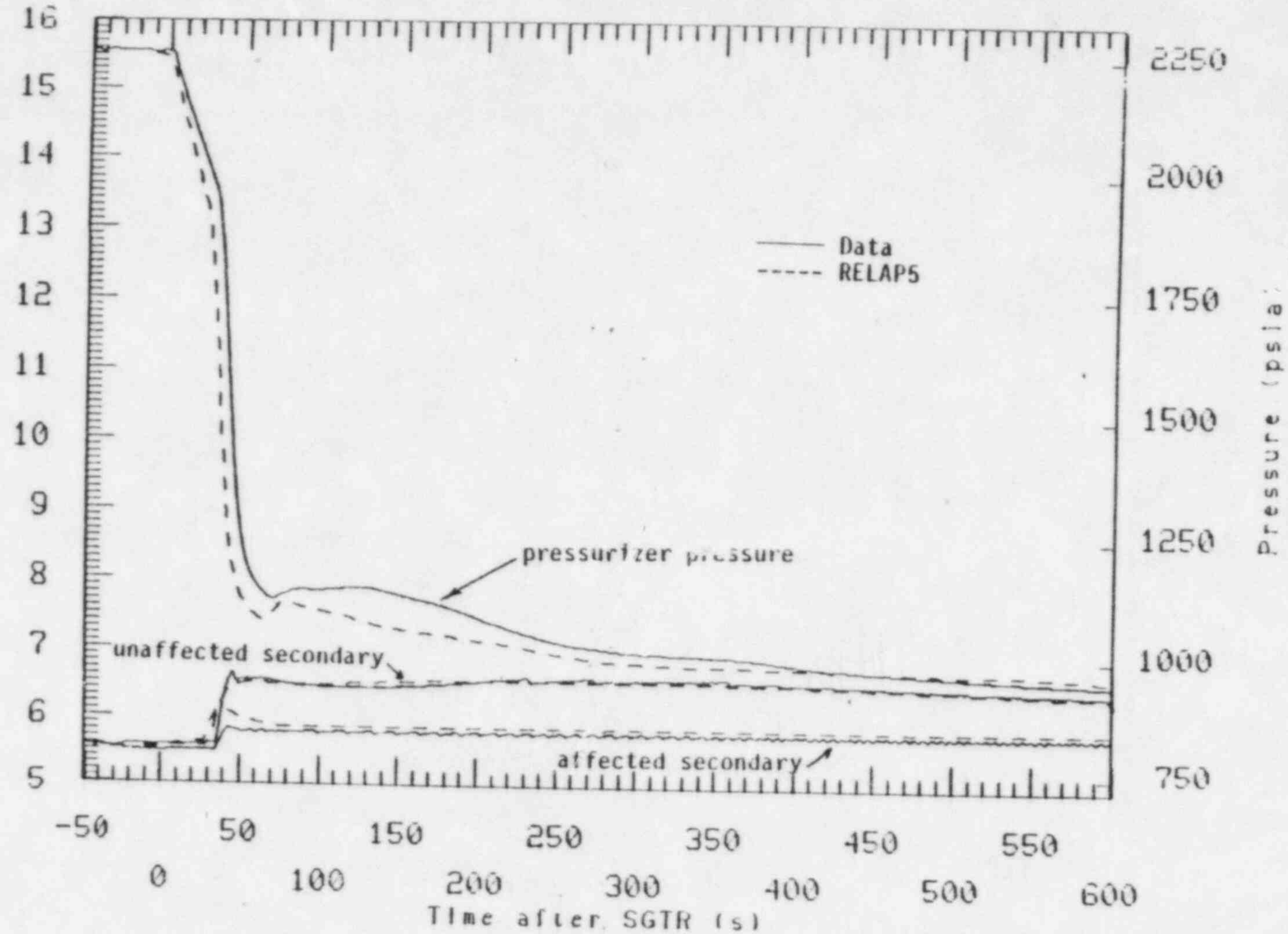

Figure 38. System response comparison 0 - 600s.

4.2 Plant Recovery Phase--600 s to Test Termination

Plant recovery in test S-SG-7 began at 600 s and was achieved by an unaffected loop secondary steam and feed operation. Beginning at 600 s, the unaffected loop secondary was depressurized by cycling the ADV. Auxiliary feedwater replenished the fluid expelled through the ADV.

The calculated and measured secondary pressures are compared in Figure 39. At 600 s, the unaffected loop steam generator ADV was opened and the secondary pressure in that steam generator fell to approximately 5.0 MPa (725.2 psia). The calculation had indicated that the minimum pressure at that time would be approximately 0.2 MPa (29 psid) higher and would occur about 100 s sooner. The oscillations in the unaffected loop secondary pressure were due to the manner in which the operator cycled the ADV, i.e., he used the entire specified operating band, whereas the calculation tried to maintain the pressure as close to the nominal control value as possible.

By approximately 700 s, the primary coolant system pressure (Figure 40) had fallen below the affected loop secondary pressure, and reverse break flow resulted (Figure 41). Continued cycling of the unaffected loop ADV was sufficient to depressurize both the primary coolant system and affected loop secondary to the 5.72 MPa (830 psia) goal. The steam and feed operation was adequate to maintain that pressure with relatively infrequent ADV cyclings throughout the remainder of that phase of the test. Since the ADV was cycled to maintain a specified primary coolant system (thus affected secondary) pressure, good calculational agreement was ensured; however good agreement was also achieved between the calculated and measured unaffected loop secondary pressures.

The final portion of the recovery was initiated when the liquid level in the unaffected loop steam generator reached 1050 cm (34.4 ft). The timing of that event is clearly related to the mass inventory in the particular steam generator. As discussed in Section 2, the initial secondary mass in the unaffected steam generator was substantially greater than the specified initial condition; thus, the time required to reach the

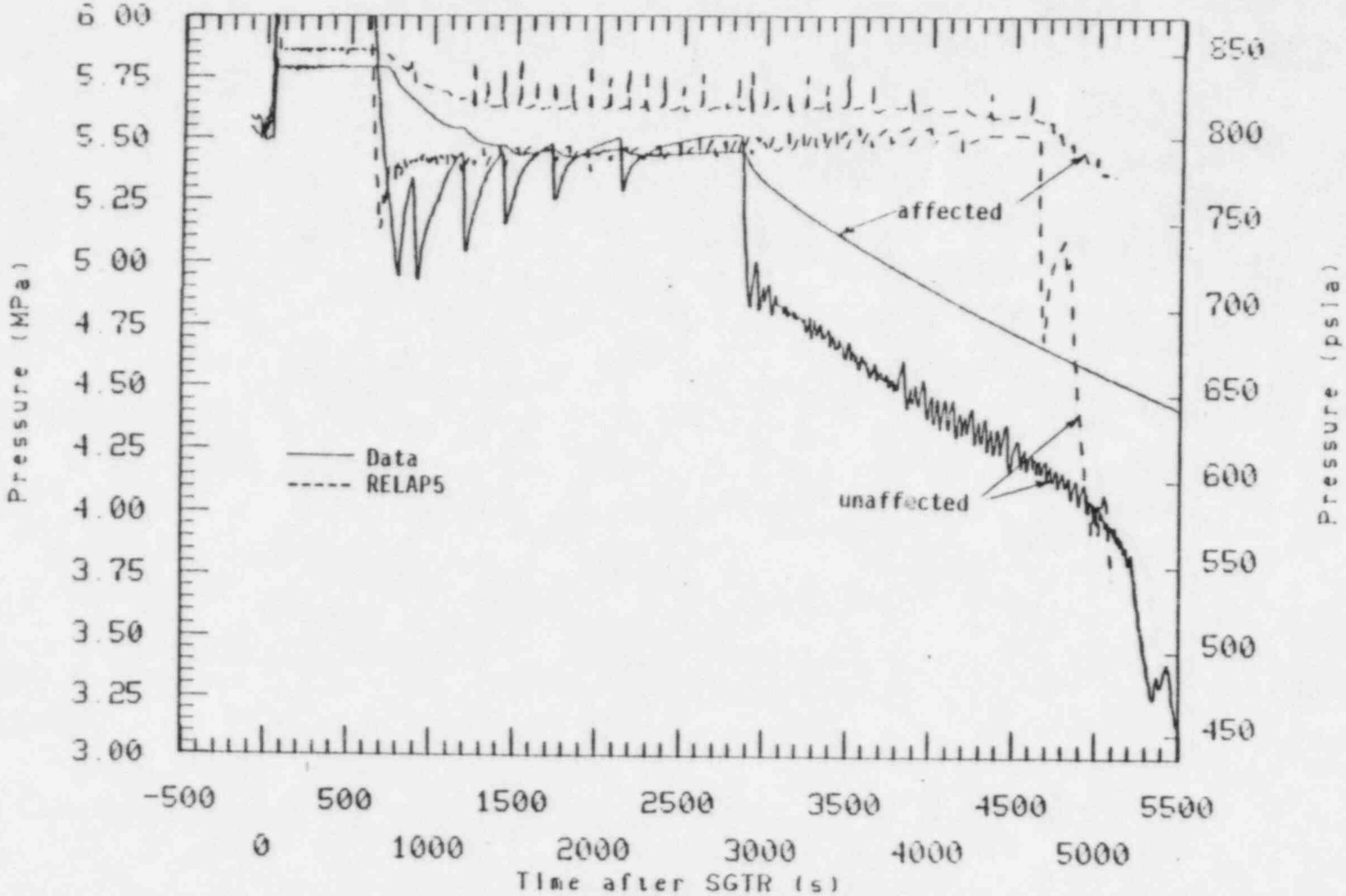

Figure 39. Secondary pressure comparison.

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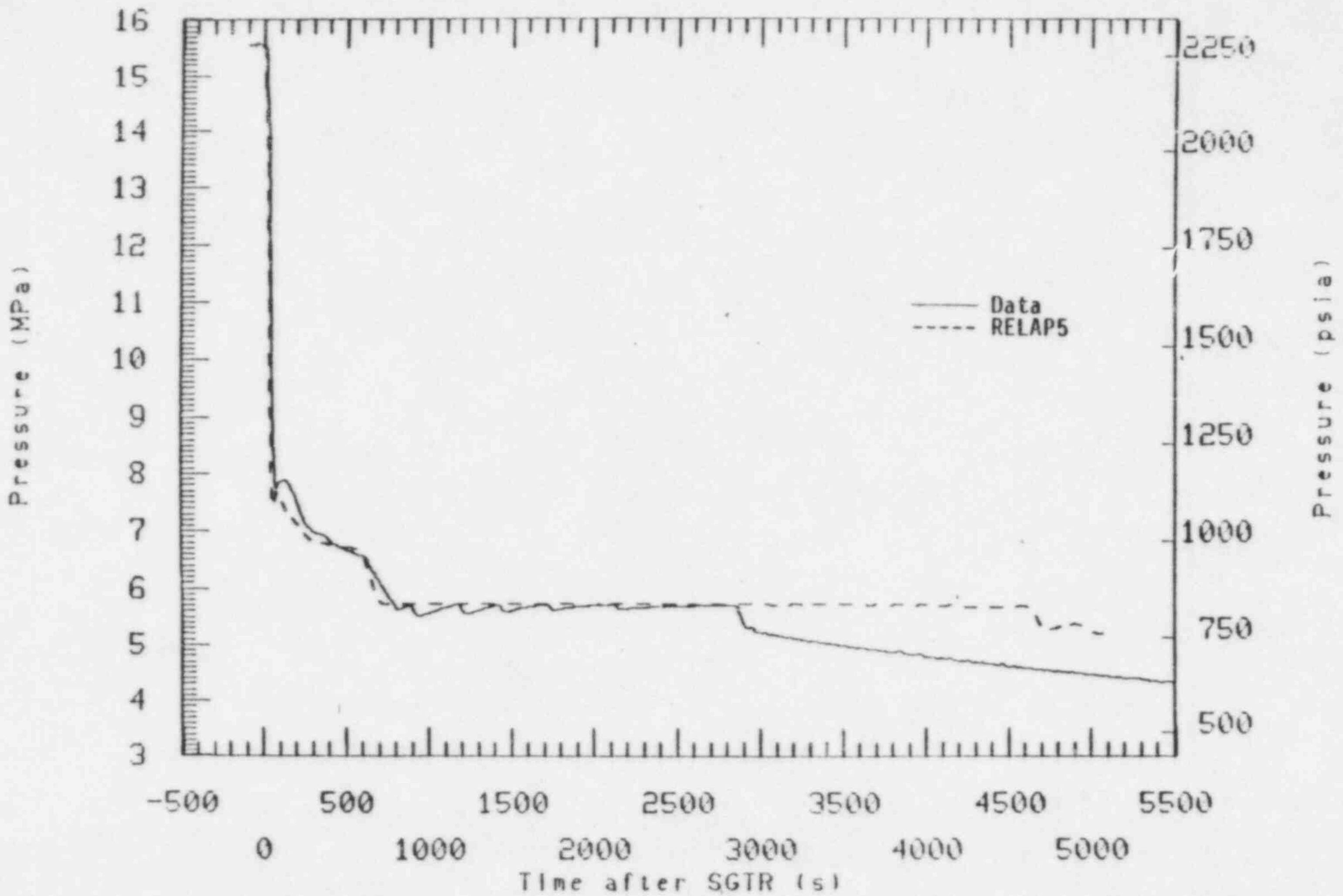

Figure 40. Primary pressure comparison.

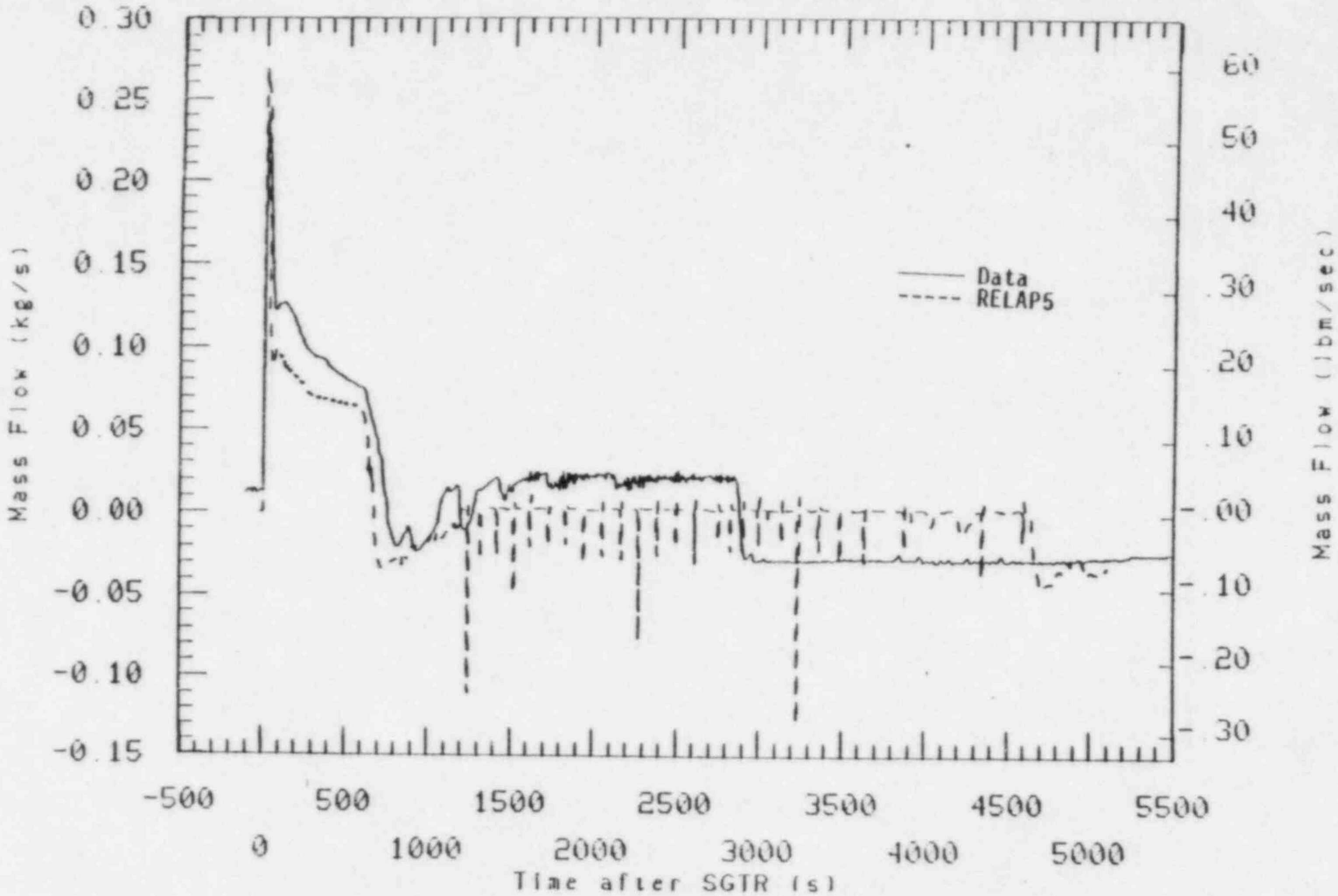

Figure 41. Break flowrate comparison.

1050 cm (34.4 ft) level in the test was greatly reduced. The initial mass discrepancy is illustrated in Figure 42. At initiation of the recovery operation, the measured unaffected loop steam generator liquid level was approximately 100 cm (3.28 ft) higher than in the calculation. Consequently, the 1050 cm (34.4 ft) liquid level was reached approximately 1800 s sooner in the experiment than predicted. Note that the code had predicted approximately 2000 s to fill the unaffected loop steam generator from 950 cm (31.2 ft) to 1050 cm (34.4 ft). Therefore, if the initial secondary mass had been close to the specification, good agreement in event timing would have been achieved.

The test plan was to refill the vessel through back flow from the affected steam generator when the unaffected steam generator liquid level reached the above discussed 1050 cm (34.4 ft) level. This was to be effected by cycling the unaffected loop ADV such that the primary coolant system pressure was maintained 0.14 MPa (20 psid) below the affected secondary. That procedure was implemented at 2848 s in the experiment as opposed to 4650 s in the calculation. The calculated core liquid level (Figure 43) was, at most times, approximately 90 cm (35 in) lower than measured during the test. The drop in core level seen during the test at initiation of the refill operation was not predicted by RELAP5. The calculation indicated that the heated length could be refilled at approximately 6 s/cm whereas approximately 10 s/cm was actually required. One reason for this was that the affected steam generator model was somewhat more responsive to the unaffected secondary ADV operation and the 0.14 MPa (20 psid) pressure difference was not well maintained in the calculation. The increased pressure differential resulted in increased reverse flow. Better control of the specified pressure difference in the calculation would have resulted in closer agreement with data. (Note in Figure 41 that the calculated reverse break flowrate was occasionally as much as 50% too high.)

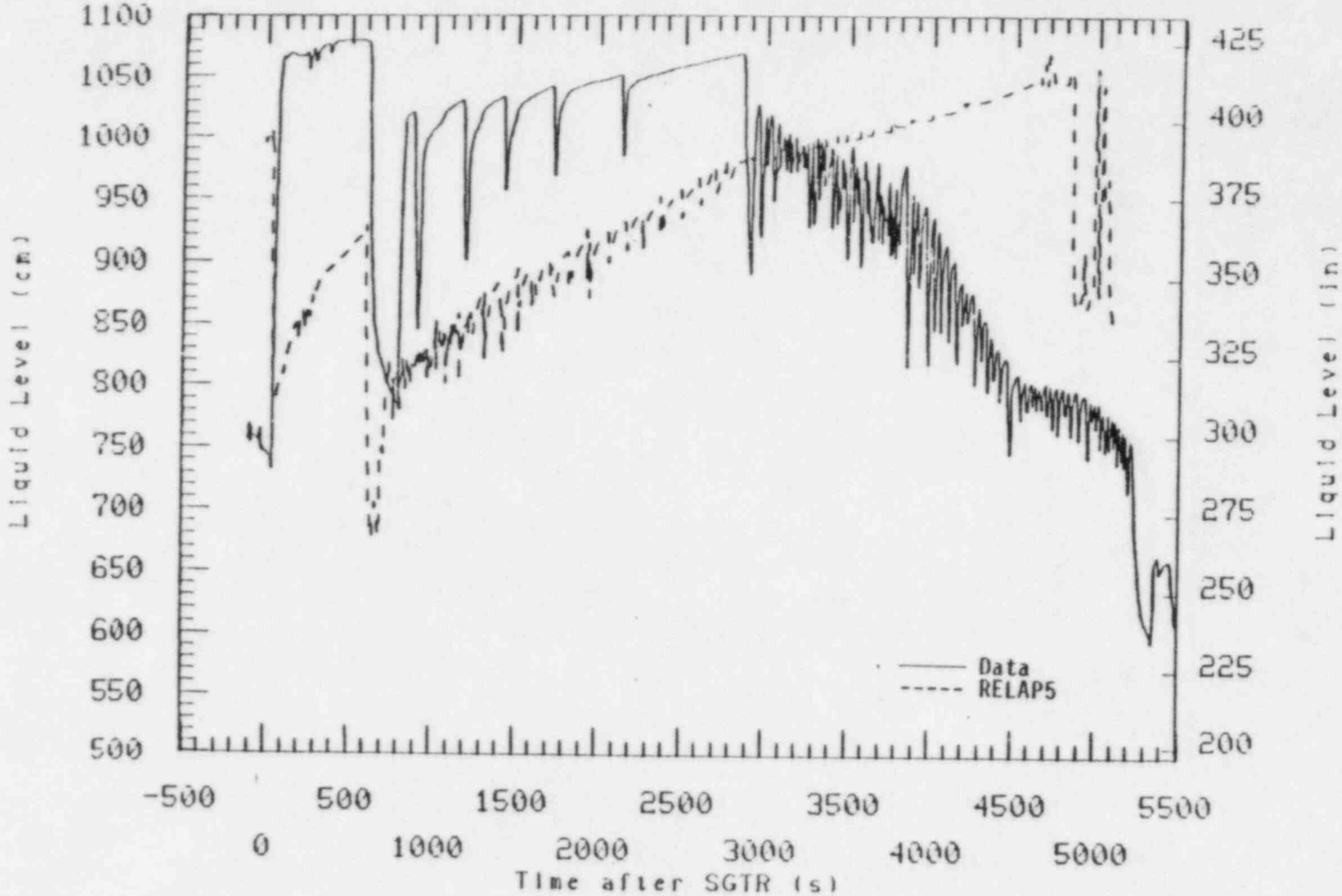

Figure 42. Unaffected steam generator secondary liquid level comparison.

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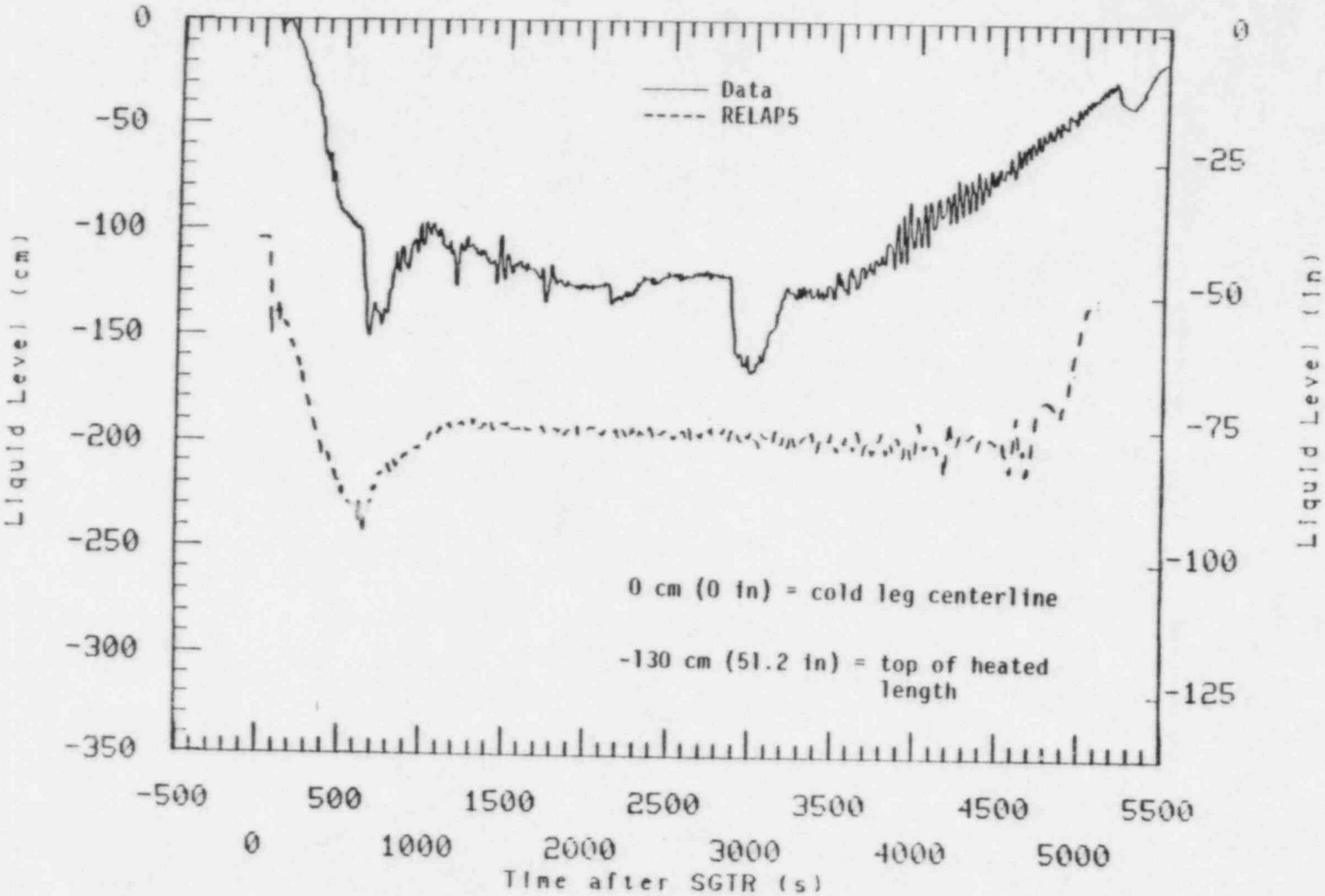

Figure 43. Core liquid level comparison.

5. CONCLUSIONS

Based on a preliminary analysis of Test S-SG-7 test results the following conclusions have been reached.

1. In Semiscale, more vessel liquid voiding occurs during the 5 tube rupture transients without SI than those transients with SI during the 600 s operator diagnostic period. On an overall system mass balance basis during the first 600 s, the addition of SI contributes a relatively small percentage of total system mass inventory and total integrated break flow; however, the vessel collapsed liquid level was 100 cm lower for the experiment without SI (S-SG-7). This difference was enough to cause the vessel collapsed level to be reduced from just below the cold leg for the experiment with SI to the top of the core for the experiment without SI. On a long term basis, without operator action, the effect of SI would be significant.

2. During a 5 tube rupture in Semiscale the pressurizer initial liquid level has a significant effect on the timing of automatically occurring events during the early depressurization period. However, the overall thermal hydraulic state of the system after the first few hundred seconds of the transient is unaffected by initial pressurizer level. With more initial pressurizer liquid level during S-SG-7 than S-SG-2, the time to scram was increased. The increase in scram time is attributed to a longer period with a high interfacial area in the pressurizer. This tends to promote flashing and retard depressurization. Even though the timing of scram is affected by initial pressurizer level, the overall system hydraulic condition of the two tests after the first few hundred seconds is identical. The additional mass in the pressurizer for S-SG-7 compared to S-SG-2 simply leaves the system during a sustained period of high break flow. The final vessel inventory for the two cases was similar.

3. During a 5 tube rupture transient in Semiscale without SI, unaffected loop secondary feed and steam is a sufficient operator action to reduce primary pressure below affected loop ADV setpoints and also to reduce primary pressure below affected loop secondary pressure. Unaffected loop secondary steam and feed reduced the primary pressure below affected loop ADV setpoints (isolated secondary) with an increasing unaffected loop secondary level. Further, unaffected loop secondary steam and feed was able to reduce the primary pressure below the affected loop secondary pressure thus causing a back flow of affected loop secondary inventory into the primary. The back flow caused the vessel collapsed liquid level to increase from below the top of the core to the cold leg elevation while maintaining the unaffected loop secondary collapsed level above half full.

4. The accident signature (0-600 s) was well predicted by RELAP5/MOD1.5. A higher than specified initial condition for the unaffected secondary mass inventory during the experiment resulted in the difference between the predicted and actual timing of the filling of the unaffected steam generator. The time to refill the vessel was poorly predicted, due to inadequate control of the specified break differential pressure in the calculation. The RELAP5 prediction was conservative with regard to core liquid level.

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16. ABSTRACT (200 words or less)
Results of a preliminary analysis of the fourth test performed in the Semiscale Mod-2B Steam Generator Tube Rupture Series are presented. Test S-SG-7 simulated a pressurized water reactor accident initiated by a double-ended offset shear of five cold side steam generator tubes. The transient included an initial 600 s period during which only automatic plant protection system response to the initiating event occurred. A total loss of both onsite and offsite power was assumed to occur at the SI injection signal. This effectively disabled the injection system and no SI was available for the entire transient. During the first 600 s the vessel collapsed liquid level was reduced to just below the top of the core but no core rod heat-up occurred. At 600 s, an operator induced recovery procedure was initiated which included unaffected loop secondary feed and steam (using ADV and auxiliary feed) to stabilize the primary pressure below the affected loop ADV setpoint. Following a period of stable primary pressure, the unaffected loop steam generator secondary filled and a new feed and steam operation commenced including ADV operation and auxiliary feed in an attempt to maintain the primary pressure below the affected loop secondary pressure. This operation caused a back flow from the affected loop generator to the primary which increased the vessel liquid inventory. The test results showed that without the use of SI, the unaffected loop steam generator feed and steam operation alone was sufficient to recovery the Semiscale system from a simulated five-tube rupture without a core rod heat up.

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