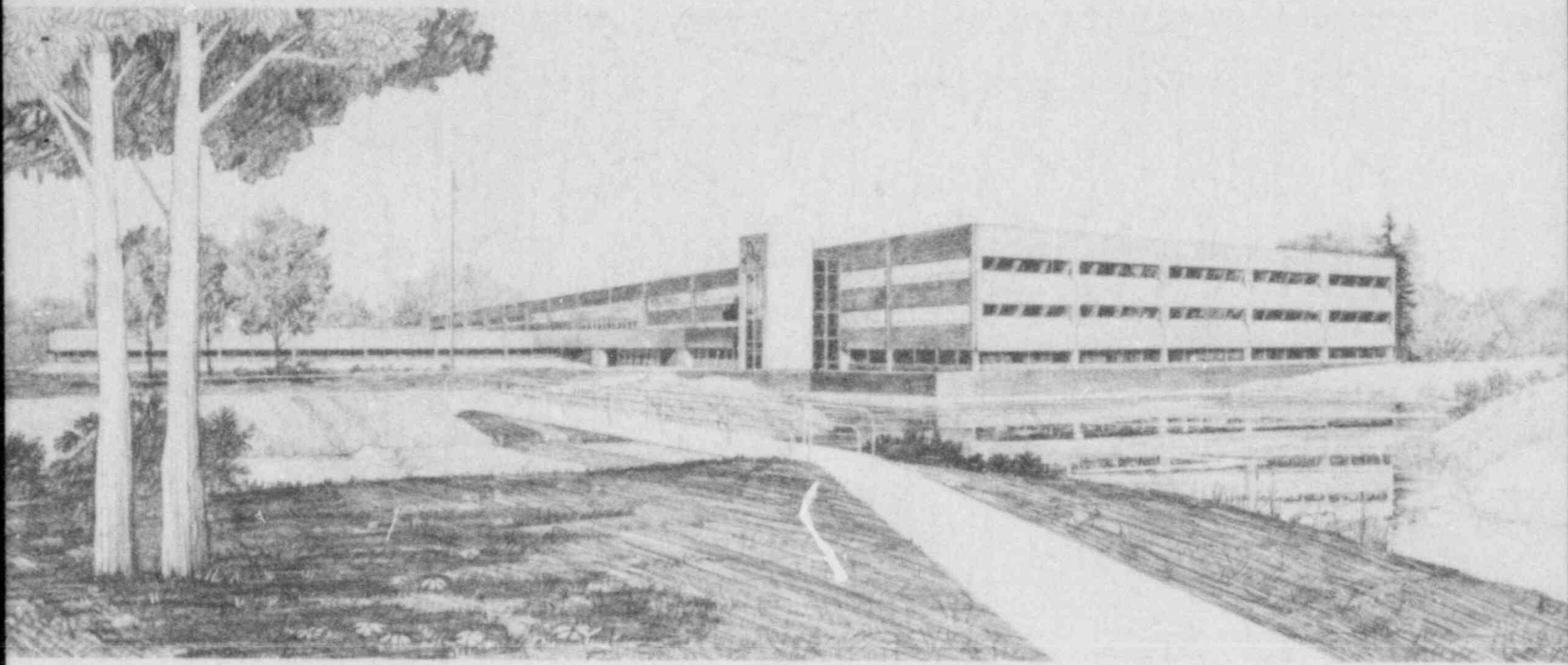
NUREG/CR-4073
EGG-2363



Idaho National Engineering Laboratory
Operated by the U.S. Department of Energy

Results of the Semiscale Mod-2B Steam Generator Tube Rupture Test Series

Guy G. Loomis

January 1985

Prepared for the
U.S. Nuclear Regulatory Commission
Under DOE Contract No. DE-AC07-76IDO1570

EG&G Idaho

Available from

GPO Sales Program
Division of Technical Information and Document Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

and

National Technical Information Service
Springfield, Virginia 22161

NUREG/CR-4073
EGG-2363
Distribution Category: R2

RESULTS OF THE SEMISCALE MOD-2B STEAM GENERATOR TUBE RUPTURE TEST SERIES

Guy G. Loomis

Published January 1985

EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under DOE Contract No. DE-AC07-76IDO1570
FIN No. A6038

ABSTRACT

A series of experiments was conducted in a scaled model of a pressurized water reactor (Semiscale Mod-2B) to investigate steam generator tube rupture system signature response and recovery techniques. The tube rupture was assumed to occur during normal full power operation [15.6 MPa (2262 psia) system pressure; 37 K (67°F) core differential temperature]. From the experimental results, the characteristic system signature responses for a wide range of number of tubes ruptured and rupture locations have been examined. In addition, recovery techniques requiring operator actions were examined. These recovery techniques included the use of pressurizer auxiliary spray and internal heaters, steam generator feed and steam, primary feed and bleed, and safety injection. The effectiveness of using these techniques for primary system pressure and subcooling control is discussed.

FIN No. A6038—Semiscale Program

SUMMARY

The Semiscale experimental program conducted by EG&G Idaho, Inc., is part of the overall research and development program sponsored by the U.S. Nuclear Regulatory Commission (NRC) through the Department of Energy (DOE) to evaluate the behavior of pressurized-water-reactor (PWR) systems during hypothesized accident sequences. Its primary objective is to obtain representative integral and separate-effects thermal-hydraulic response data to provide an experimental basis for analytical model development and assessment. This report presents the results obtained from the Semiscale Mod-2B steam generator tube rupture test series. The Mod-2B system is a small-scale, nonnuclear, experimental system in which nuclear heating is simulated by an electrically heated core. The system includes a vessel and two operating loops, both of which contain an active steam generator.

The steam generator tube rupture test series was performed in the Semiscale Mod-2B system at typical PWR system pressure and temperature [15.6 MPa (2262 psia) pressure; 37 K (67°F) core differential temperature]. The following summarizes important results from analysis of the steam generator tube rupture test series.

Tube rupture signature responses were investigated in the Semiscale Mod-2B system by establishing a primary-to-secondary system flow near the tube sheet in the affected loop generator. The signature response was investigated for a 600-s period following the tube rupture, assumed to be the initiating event. The 600-s period was thought to be a reasonable period for an operator to determine that a tube rupture had occurred and which generator had suffered the tube rupture. Only automatically occurring events transpired during this operator diagnostic period, including core scram, main steam isolation valve (MSIV) closure, safety injection (SI) initiation, main coolant pump trip, feedwater termination, and auxiliary feedwater initiation. Upon initiation of the tube rupture, the primary system depressurized as primary fluid flowed into the broken loop secondary system via the tube rupture break assembly. On an overall basis, the primary system depressurized with varying depressurization rates until saturation conditions were reached in the hot leg, at which time the depressurization slowed due to flashing. Major inflection points in the primary system depressurization were caused by the following occurrences. The saturated fluid in the pressurizer immediately began flashing as break flow initiated the primary system depressurization. When the interfacial fluid level in the pressurizer reached the surge line, there was an abrupt interfacial area reduction which caused a rapid decrease in the amount of flashing. This decrease in flashing resulted in an increase in the primary system depressurization rate until the low pressurizer pressure trip [13.1 MPa (1900 psia)] was reached, causing core scram and MSIV closure. Core scram caused a second increase in primary system depressurization as primary-to-secondary system heat transfer continued due to decay heat in the core. As a result of the MSIV closure and continued primary-to-secondary system heat transfer without steam relief, the secondary pressure in both loops rose to the atmospheric dump valve (ADV) setpoint and cycled several times during the first 600 s. At the end of 600 s, the system was in various modes of natural circulation supported by a heat source (core decay heat) and heat sink (steam generator secondary pressure lower than primary pressure). The various modes of natural circulation depended on the amount of system mass voiding, corresponding to the difference between tube rupture break flow and safety injection flow.

The signature response was found to be similar for one-, five-, and ten-tube ruptures; only the timing of events, the system mass inventory, and natural circulation modes were different. All three cases show increased depressurization to the saturation condition upon core scram. The core scram time was dependent on the early depressurization rate which was proportional to the tube rupture break size and break flow rate. For the single-tube rupture case, the mass inventory at the end of the operator diagnostic period was about 87%, which is typical of single-phase natural circulation inventories. At 600 s, the five-tube case had an inventory of about 60% and the ten-tube case had an inventory of 52%, both of which are more typical for the reflux-condensation mode of natural circulation. In all three cases, there was sufficient vessel inventory to preclude core rod heatup.

Tube rupture location, whether at the inlet or at the outlet of a tube, has essentially no effect on tube rupture signature response. Signature response measures, such as primary system pressure, break flow rate, and pressurizer liquid level, are identical for inlet and outlet breaks.

The initial pressurizer collapsed liquid level was found to have a large effect in the timing of certain automatically occurring events during the operator identification period. A higher initial pressurizer collapsed liquid level resulted in a slower primary system depressurization and a longer time to scram; however, the thermal-hydraulic state of the system after the first few hundred seconds was identical for different initial pressurizer levels. This was because the extra mass in the higher initial level case simply left the primary system via the break flow. As a result, the system mass inventory after the first few hundred seconds was essentially identical in all cases.

The primary system signature response for a main steam line break followed by a tube rupture was found to be similar to the signature response for a tube rupture without the complication of a main steam line break. The major difference was that the secondary system release to atmosphere is greater for the main steam line break case. Prior to a tube rupture, the main steam line break caused only a minor reduction in primary system pressure [15.5 to 14.2 MPa (2247 to 2059 psia)]. This reduction was caused by an increase in heat sink due to a secondary pressure reduction corresponding to the main steam line break. Once the tube rupture occurred, the primary system pressure decreased in a similar manner to the case where there was no compounding main steam line break. The primary system pressure decreased at various rates to the saturation condition in the hot leg, at which point flashing reduced the depressurization rate. A main inflection point in primary system depressurization occurred as the pressurizer liquid level lowered to the bottom of the pressurizer and the interfacial area for flashing changed. (This was similar to the normal tube rupture signature response without the complication of main steam line break.) The affected loop secondary system pressure decreased during the operator identification period because tube rupture break flow was less than main steam line break flow, causing a reduction in secondary liquid level. As a result, there was a considerably higher release of secondary system fluid to atmosphere than for a tube rupture event alone.

At the end of the operator diagnostic period, the primary system pressure was above the steam generator relief valve setpoints and subcooling in the hot leg was nonexistent; therefore, operator action was required to reduce system pressure below the relief valve setpoint and increase the loop subcooling. Recovery techniques were employed in the Semiscale experiments to first lower the primary system pressure below the broken loop ADV setpoint, thus isolating the secondary system from atmospheric release of inventory, and then increase loop subcooling. The operator techniques examined in the Semiscale experiments involved: an unaffected loop secondary feed and steam to increase loop subcooling and lower primary system pressure; primary feed and bleed, using safety injection (SI) and pressurizer power-operated relief valve (PORV) operation to control loop pressure; pressurizer auxiliary spray to control loop pressure and inventory; SI to control pressure and inventory; and pressurizer internal heaters to control primary system pressure and subcooling.

The Semiscale experimental results show that the effectiveness of pressure control and loop cooling due to unaffected loop feed and steam is dependent on the hydraulic state of the loop, which is dependent on the number of tubes ruptured and the natural circulation mode. For instance, a single-tube rupture leaves the system in single-phase natural circulation at the end of the operator diagnostic period, whereas the five- and ten-tube rupture cases with more system voiding leave the system in the reflux condenser mode. The feed and steam operation has a large effect on primary system pressure if the primary system is in a more voided state, such as occurs with a five-tube rupture event; however, for a single-tube rupture, the pressure decrease due to feed and steam is lower. For the single-tube rupture case, the increased steam generator heat sink increased primary-to-secondary system heat transfer by increasing the differential temperature across the tubes. The increased heat transfer caused a primary system fluid temperature reduction which increased shrinkage of fluid in the system. For the five- and ten-tube rupture cases, the initiation of unaffected loop feed and steam increases the condensation in the primary system tubes. The mass rate of condensation is proportional to the differential temperature across the tubes, and the system pressure is proportional to the mass rate of condensation; therefore, the increase in differential temperature caused by the feed and steam operation increased the depressurization rate. During unaffected loop feed and steam operation, the redistribution of primary system fluid was again dependent on the hydraulic condition in the loop at the onset of feed and steam. For the more voided five-tube case, feed and steam caused a pronounced mass redistribution. Feed and steam operation during the five-tube case caused a filling of the unaffected loop primary tubes. Condensed steam generated in the core and entrained fluid from other parts of the system were the primary sources of fluid filling the tubes. The rapid depressurization associated with the feed and bleed operation caused a flashing effect for fluid in the core; however, there was no rod heatup associated with this level swell. For the single-tube case, where single-phase natural circulation dominated, feed and steam initiation promoted essentially no change in mass distribution in the system, since no condensation-based, low-pressure areas existed.

PORV operation along with SI is effective in reducing primary system pressure below affected loop relief valve setpoints without core uncovery. Even though the core was not uncovered during the PORV operation, there was a significant system mass inventory redistribution and net overall system mass inventory reduction. Upon initiation of PORV operation, primary system fluid was transported to the pressurizer from other parts of the system and eventually filled the pressurizer. The primary source of the fluid filling the pressurizer was the vessel. The effectiveness of PORV operation for reducing primary system pressure decreased as the liquid level in the pressurizer increased. Once the pressurizer filled, an open PORV had only a small effect on primary pressure control. This is because the primary volume reduction due to PORV liquid flow is much less than the volume reduction from steam flow that occurred during early PORV operation.

Pressurizer auxiliary spray is effective in reducing primary system pressure, but the effectiveness is dependent on pressurizer wall and steam superheat removal. The introduction of cold auxiliary spray water into the pressurizer changed the fluid and metal temperatures from superheated to saturated. Once the fluid was saturated, further spray condensed the saturated steam. Initially, the removal of superheat caused a slight pressurization of the primary system due to evaporation of auxiliary spray. Once the superheat was removed, continued spray caused a pressure reduction due to condensation.

Pressurizer internal heaters are ineffective for increasing primary system pressure during a tube rupture. As long as SI was off, bubble formation in the pressurizer due to heater operation could not offset the fluid volume lost due to tube rupture break flow. The net result was no compression of the primary fluid and thus no net rise in primary pressure.

The use of SI in a nearly full system causes a compression of steam spaces and a primary system pressurization. The primary system pressurization due to SI increases the subcooling in the hot leg. Termination of SI during a tube rupture causes a lowering of primary system pressure because the continued break flow expands the voids in the system.

ACKNOWLEDGMENTS

Credit is due to W. A. Owca, who helped analyze much of the data during the steam generator tube rupture test series, and to Roy A. Shaw, who provided a technical review of the document. Special thanks to Carolee Kearns, who performed computer graphics on all the figures.

CONTENTS

FIGURES

TABLES

RESULTS OF THE SEMISCALE MOD-2B STEAM GENERATOR TUBE RUPTURE TEST SERIES

INTRODUCTION

The Semiscale experimental program conducted by EG&G Idaho, Inc., is part of the overall research and development program sponsored by the U.S. Nuclear Regulatory Commission (USNRC) through the Department of Energy (DOE) to evaluate the behavior of pressurized-water-reactor (PWR) systems during hypothesized accident sequences. Its primary objective is to obtain representative integral and separate effects thermal-hydraulic response data to provide an experimental basis for analytical model development and assessment. The Semiscale Mod-2B steam generator tube rupture test series was authorized and performed under this program.

Transients initiated by steam generator tube rupture and transients otherwise induced but concurrent with tube rupture are considered relatively probable during the normal life of a commercial PWR. The study of tube rupture transients is important because the tube rupture allows a primary-to-secondary system flow path which can eventually result in a secondary system release of radioactive fluid to the atmosphere. To mitigate a tube rupture transient, operators utilize emergency recovery procedures to isolate the affected generator from atmospheric release and then gradually increase primary system inventory and loop subcooling. The computer codes commonly used to calculate system response during a tube rupture transient have not been adequately verified against an integral data base involving tube rupture and operator recovery procedures;[1] therefore, a series of experiments was performed in the Semiscale Mod-2B system to provide an integral data base for code assessment and development.

The Semiscale Mod-2B system is a small-scale model of the primary and secondary system of a four-loop nuclear generating plant. One loop, the intact or unaffected loop, simulates three loops of a PWR; and another loop, the broken or affected loop, simulates the PWR loop in which the tube rupture occurs. Both loops have active pumps and U-tube-in-shell steam generators. There is a vessel with an electrically heated core and an external downcomer. The modified-volume scaling philosophy followed in the design of the Mod-2B system preserves most of the important first-order effects thought important for small break loss-of-coolant accidents (LOCA's).[2] Most notably, the 1:1 elevation scaling of the Semiscale system is an important criterion for preserving the factors influencing natural circulation behavior, which is a major heat rejection mechanism during a tube rupture.

This report presents the results of the Semiscale steam generator tube rupture experiments. Specific topics include: general system signature response to a tube rupture; influence of tube rupture location and number of tubes ruptured on system signature response; influence of initial pressurizer level on signature response; and the signature response of a main steam line break with concurrent tube rupture. In addition, operator recovery techniques were examined, including: the effectiveness of unaffected loop secondary feed and steam on pressure and subcooling control; the effectiveness of pressurizer power operated relief valve (PORV) operation on pressure control; the effectiveness of pressurizer auxiliary spray for inventory and pressure control; the effectiveness of pressurizer internal heaters for pressure control; and the effect of safety injection (SI) on pressure control.

SYSTEM DESCRIPTION, EXPERIMENT MATRIX, AND EXPERIMENT CONDUCT

System Description

The steam generator tube rupture test series was performed in the Semiscale Mod-2B test facility, which is a small-scale model of a four-loop, PWR power-generating plant (scaling factor 1/1705). The Mod-2B system incorporates the major components of a PWR, including steam generators, vessel, downcomer, pumps, pressurizer, and loop piping (as shown in Figure 1). One loop (unaffected) is scaled to simulate the three intact loops in a PWR, while the other (affected) simulates the single loop in which the tube rupture is postulated to occur. The Semiscale Mod-2B system utilizes an electrically heated core to represent a PWR nuclear core.

For all experiments, the vessel core consisted of a 5 x 5 array of internally heated electric rods, 21 to 23 of which were powered. The rods were geometrically similar to nuclear rods with a heated length of 3.66 m (144 in.) and an outside diameter of 1.072 cm (0.42 in.). All rods were powered equally.

In both the unaffected and affected loop secondaries, 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] Although SRV's in a PWR typically have five stages of relief, the SRV orifice is designed to pass a scaled flow corresponding to only the first stage of relief. The ADV orifice is designed to pass scaled flow corresponding to ADV operation in a PWR, where the pressure relief setpoint for the ADV stage is encountered before the various multistaged SRV relief setpoints. 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 (848 psia) in the affected loop and 6.55 MPa (950 psia) in the unaffected loop. The first stage SRV relief setpoint is 5.94 MPa (861 psia) in the affected loop and 6.74 MPa (977 psia) in the unaffected loop. These relief setpoints were artificially lowered to ensure ADV opening as expected on full-sized plants. The ADV can also be manually latched open during the recovery procedure with the SRV block valve shut. Appendix A contains a description of the scaling rationale for the relief setpoints used in Semiscale.

The tube rupture break assembly connected the primary coolant system with the secondary side in the vicinity of the steam generator tube sheet of the affected loop (Figure 2). The break assembly could be connected to either the hot leg or cold leg side of the primary system at the steam generator plenum of the affected loop, 57.1 cm (22.5 in.) below the top of the tube sheet. The break assembly was connected to the secondary system at one location, 36.5 cm (14.37 in.) above the top of the tube sheet on the cold leg side of the generator. The break assembly consisted of a break orifice and venturi flowmeters to measure single-phase break mass flow rate. The break orifice was an interchangeable symmetric conical flow tube, as depicted in Figure 3. Figure 3 also shows the dimensions for one-, five-, and ten-tube break orifices. The breaks were assumed to be double-ended offset shear breaks. The flow tube is calibrated in single-phase water and could be used to monitor break mass flow rate in both directions because of the symmetry of the flow tube.

The facility pressurizer PORV 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 PORVs of a full scale PWR. A 0.141-cm (0.055-in.), sharp-edged orifice was sized to pass 0.03 kg/s (0.066 lbm/s) at 16.2 MPa (2349 psia). Pressurizer internal heaters can be operated in the variable mode, backup mode, or warmup mode. The variable and backup mode total power is 2.35 kW, and the warmup mode total power is 13.3 kW.

Heat loss makeup in the Semiscale system is accomplished by using external heaters distributed uniformly throughout the system. These heaters are controlled by six separate power supplies, including the vessel, hot legs, cold legs, unaffected loop pump

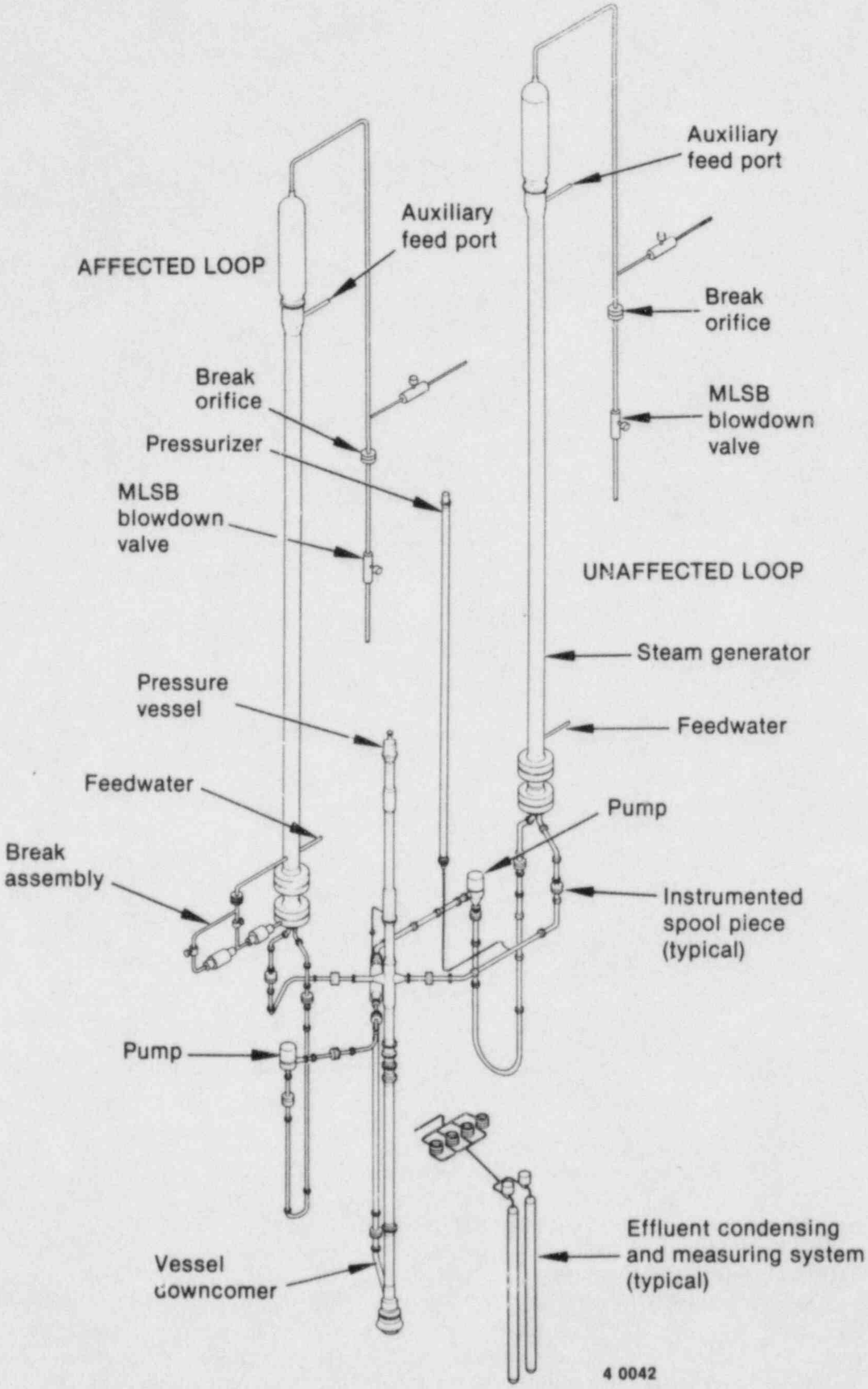


Figure 1. Semiscale Mod-2B system configuration for the steam generator test series.

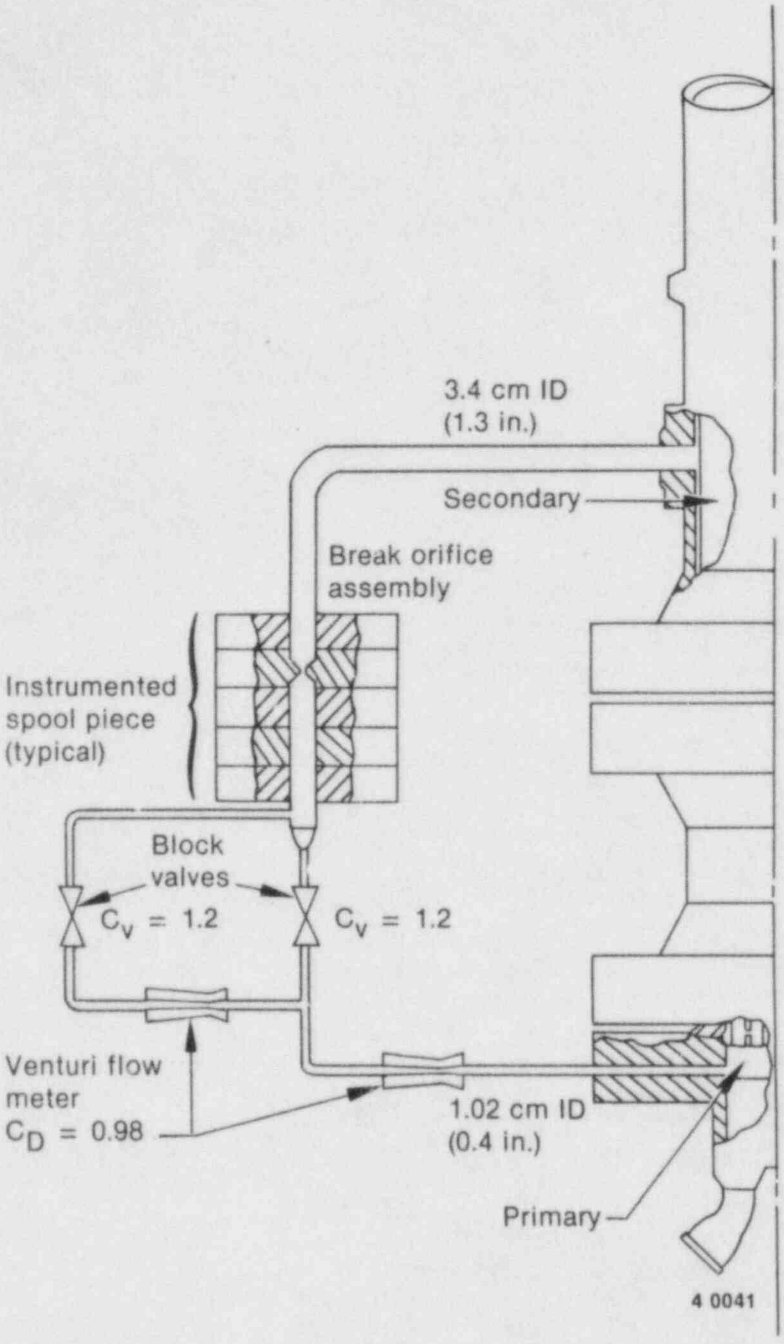
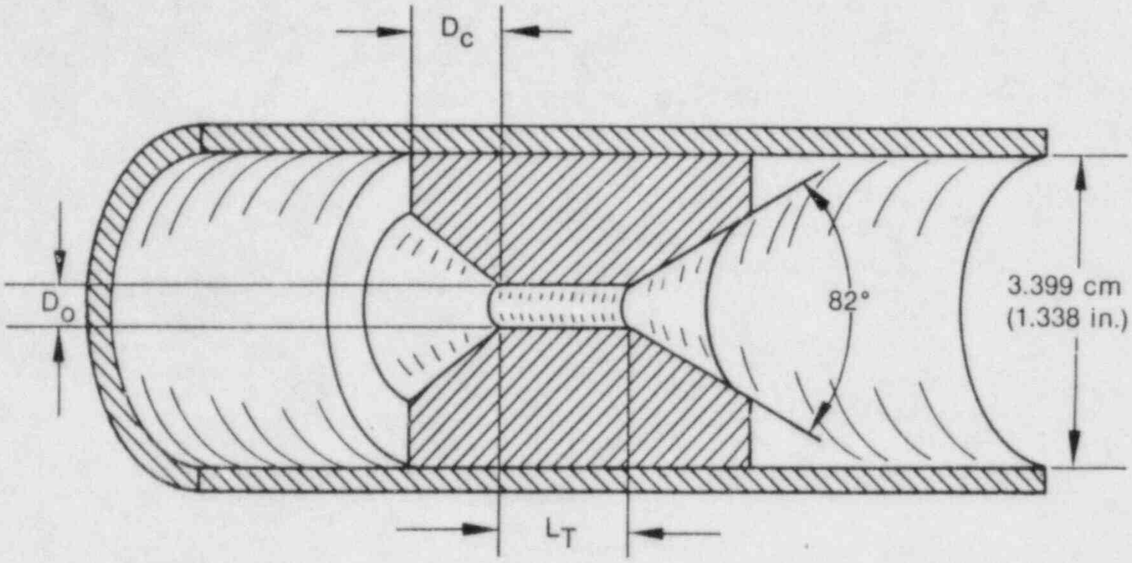


Figure 2. Semiscale Mod-2B tube rupture break assembly.

suction, affected loop pump suction, and pressurizer. The total power provided by these heaters is 47 kW. An additional 20 to 28 kW of heat loss makeup was provided by augmenting core power throughout the transient. The additional heat loss was added to make up vessel and pump seal cooling losses which were not compensated for by loop heaters. Control of the heaters was as follows: If the maximum allowable heater temperature [900 K (1160°F)] was reached on the inside surface of the pipe insulation, external power to that component was reduced by half. If the temperature trip limit continued to be exceeded, power to that component was terminated. The purpose for terminating power is to prevent damage to the heater element. A heater temperature of 900 K (1160°F) does not imply that the pipe wall or fluid temperature reached that value.



Tube rupture	D_O		L_T		D_C	
	cm	in.	cm	in.	cm	in.
1 tube	0.079	0.0308	0.198	0.078	1.473	0.662
5 tube	0.175	0.0689	0.439	0.173	1.372	0.709
10 tube	0.249	0.0975	0.622	0.245	1.270	0.745

4 0040

Figure 3. Semiscale Mod-2B conical flow break tube.

Conditions in the system were monitored by an extensive network of metal and fluid thermocouples and pressure and differential pressure transducers. Both steam-generator long and short tubes were extensively instrumented with both primary-side and secondary-side fluid thermocouples and several primary-side differential pressure transducers. Average fluid density was measured in the loops and vessel by X-ray and gamma densitometers. Volumetric flow was measured by turbine meters, and momentum flux was measured by drag screens. Special condensing systems and catch tanks were used to accurately measure system mass flow rate from the steam generator secondary relief valves and the PORV valve. For one of the experiments, a concurrent main steam line break used a special condensing system and catch tank to measure effluent.

Experiment Matrix

The steam generator tube rupture series consisted of nine experiments involving a variety of tube rupture locations, number of tubes ruptured, compounding failures, and recovery procedures. Table 1 summarizes the test matrix for the tube rupture series. The break area spectrum represented one, five, and ten tubes ruptured on both the hot and cold side of the affected loop steam generator. The breaks were assumed to be double-ended offset shear breaks near the tube sheet. The first eight experiments involved a tube rupture as the initiating event, and the ninth experiment involved a main steam line break with a compounding tube rupture. All nine experiments involved an early time period (usually 0 to 600 s) during which only automatically occurring events were functional followed by an

Table 1. Steam generator tube rupture test matrix

Experiment Test Number	Number of Tubes and Location	Initiating Event	Recovery Techniques	Comments
S-SG-1	1 (cold side)	SGTR	Unaffected loop feed and steam; termination of SI.	SI terminated on high vessel and pressurizer level to reduce primary pressure; recovery involved establishing pressure equilibrium between primary and affected loop secondary.
S-SG-2	5 (cold side)	SGTR	Unaffected loop feed and steam; PORV operation; termination of SI.	ADV used to control unaffected loop secondary depressurization at 2.76 MPa/h (400 psi/h); PORV used to control primary depressurization on same rate; recovery included reducing primary pressure to accumulator injection setpoint [4.22 MPa (610 psia)]
S-SG-3	10 (cold side)	SGTR	Auxiliary pressurizer spray, pressurizer internal heaters, SI, unaffected loop pump, unaffected loop secondary feed and steam.	Test scenario based on PWR (Zion) emergency operating procedures; recovery involved establishing a slow primary depressurization and a subcooled fluid condition in the loop.
S-SG-4	1 (cold side)	SGTR	Pressurizer auxiliary spray, unaffected loop secondary steam and feed, SI, pressurizer internal heaters.	Main coolant pump trip delayed until 600 s; SI cycled, pressurizer internal heaters cycled; recovery included establishing primary pressure control with pressurizer internal heaters below affected loop ADV setpoint.
S-SG-5	5 (hot side)	SGTR	Unaffected loop secondary steam and feed, pressurizer internal heaters, SI cycling.	Recovery involved early unaffected loop secondary feed and steam at t = scram + 60 s (82 s), SI cycling and pressurizer internal heaters powered to maintain primary pressure below affected loop ADV setpoint.
S-SG-6	5 (hot side)	SGTR	Unaffected loop secondary steam and feed, pressurizer auxiliary spray, SI.	Compounding failure was a stuck open affected loop ADV; recovery included reducing primary pressure to LPIS setpoint [1.38 MPa (200 psia)].
S-SG-7	5 (cold side)	SGTR	Unaffected loop secondary feed and steam.	Compounding failure was a complete loss of onsite and offsite power; recovery involved using unaffected loop feed and steam to reduce the primary pressure below the affected loop secondary thus causing a back flow through the break.
S-SG-8	1 (cold side)	SGTR	Unaffected loop secondary feed and steam and primary feed and bleed using PORV.	Compounding failure was a stuck open PORV; recovery included bringing the primary pressure below the accumulator setpoint pressure [4.22 MPa (612 psia)].
S-SG-9	1 (cold side)	MSLB	Unaffected loop feed and steam, SI, PORV operation.	Compounding failure was a main steam line break; main steam line break was the initiating event followed by tube rupture 60 s later; recovery involved establishing an increasing loop fluid subcooling and primary system inventory.

operator recovery period. This 600-s period was an assumed operator diagnostic period which was thought to be a reasonable time for the operators to identify which steam generator had suffered the tube rupture and to initiate a planned recovery. Some experiments assumed early operator diagnosis of the steam generator tube rupture (60 s after transient initiation) but still assumed 600 s for the identification of which generator had the tube rupture. Many of the experiments involved compounding failures, such as a stuck open affected loop ADV, complete loss of onsite and offsite power, stuck open PORV, and main steam line break. The main steam line break was the initiating event for the experiment, with a tube rupture caused by the main steam line break.

Operator recovery techniques included combinations of the following: unaffected loop secondary system feed and steam (using auxiliary feedwater and controlled steam release through the ADV); pressurizer auxiliary spray; pressurizer internal heaters; unaffected loop pump operation; and primary system feed and bleed, using SI and PORV operation. Test scenarios were developed based on emergency operating procedures for commercial nuclear PWR's undergoing a tube rupture.[4] Table 1 lists the recovery procedures used for the various experiments.

Since this report concentrates only on the phenomena encountered during the test series, it does not address each individual test in detail. The experimental test data for each of the nine tests in the series are available to the public on the NRC Division of Accident Evaluation (DAE) data bank. Semiscale facility configuration information for each test will be contained in the Semiscale configuration document (soon to be completed).

Experiment Conduct

As a general procedure prior to initiation of the transient, the system was filled with demineralized water and vented to ensure a liquid-filled system. The system was heated to initial conditions using core power and pressurized using pressurizer internal heaters to draw a steam bubble. The steam generator secondaries dissipated the core heat by steaming to atmosphere. Table 2a (metric) and Table 2b (English) contain a list of important initial conditions throughout the system. The Semiscale initial conditions were typical of PWR full-power operating hydraulic conditions in the primary and secondary systems.

Most transients were initiated at 0 s by opening the tube rupture break block valve (see Figure 2), allowing primary system fluid to flow into the affected loop secondary. The system depressurized to the low pressurizer trip pressure [13.1 MPa (1900 psia)], which initiated core scram and main steam isolation valve (MSIV) closure. Following scram, the core

Table 2a. Initial conditions for the steam generator tube rupture test series (metric)

Experiment Number	Primary System Pressure (MPa)	Pressurizer Liquid Volume (m^3)	Core Power (kW)	Loop to Loop Cold Leg Fluid Temperature Differential (K)	Core Fluid Temperature Rise (K)	Steam Generator Pressure (MPa)		Steam Generator Secondary Fluid Mass (kg)		Primary Leakage (kg/s)
						Affected Loop	Unaffected Loop	Affected Loop	Unaffected Loop	
S-SG-1	15.42	0.0034	2000	2.0	39.0	5.53	5.98	>188	107	0.000712
S-SG-2	15.50	0.0028	2010	0.5	38.5	5.55	5.42	118	118	0.004000
S-SG-3	15.45	0.0106	1990	0.1	37.5	5.50	5.52	93	88	0.0031
S-SG-4	15.56	0.0098	1990	1.4	35.9	5.52	5.43	83	88	0.0009
S-SG-5	15.47	0.0094	1990	0.2	38.35	5.62	5.47	94	105	0.003
S-SG-6	15.68	0.0097	1990	0.3	37.9	5.62	5.56	97	88	0.0033
S-SG-7	15.45	0.0091	1990	1.9	38.7	5.58	5.49	109	178	0.0029
S-SG-8	15.54	0.0105	1990	0.7	37.8	5.58	5.49	97	88	0.002
S-SG-9	15.65	0.0098	2000	1.5	37.45	5.60	5.58	93	95	0.0032

Table 2b. Initial conditions for the steam generator tube rupture test series (English)

Experiment Number	Primary System Pressure (psia)	Pressurizer Liquid Volume (ft^3)	Core Power (kW)	Loop to Loop Cold Leg Fluid Temperature Differential (°F)	Core Fluid Temperature Rise (°F)	Steam Generator Pressure (psia)		Steam Generator Secondary Fluid Mass (lbm)		Primary Leakage (lbm/s)
						Affected Loop	Unaffected Loop	Affected Loop	Unaffected Loop	
S-SG-1	2235	0.12	2000	3.6	70.2	802	795	414	235	0.00156
S-SG-2	2248	0.10	2010	0.9	69.3	805	786	260	260	0.0088
S-SG-3	2252	0.37	1990	0.18	67.5	797	800	204	194	0.0069
S-SG-4	2256	0.34	1990	2.5	64.5	800	787	183	194	0.002
S-SG-5	2242	0.33	1990	0.36	69.0	814	792	208	233	0.006
S-SG-6	2274	0.34	1990	0.54	68.2	815	866	213	194	0.0073
S-SG-7	2252	0.32	1990	3.4	69.6	809	796	240	392	0.0063
S-SG-8	2253	0.37	1990	1.3	68.0	809	796	214	194	0.004
S-SG-9	2269	0.35	2000	2.7	67.4	812	809	204	209	0.007

power was controlled to the ANS decay curve. As the primary system further depressurized to the SI pressure trip [12.51 MPa (1814 psia)], the following automatically occurring events transpired: SI was initiated; main feedwater was terminated and auxiliary feedwater was started; and main coolant pumps began a controlled coastdown. The experiment simulating a main steam line break was initiated at 0 s by opening block valves in the main steam line, followed 60 s later by opening the block valve in the tube rupture break assembly. All experiments involved a 600-s operator diagnostic period, during which time only automatically occurring events transpired, followed by a recovery period.

Recovery involved reducing primary pressure below the affected loop ADV setpoint pressure [5.85 MPa (848 psia) in Semiscale] to isolate secondary fluid release to atmosphere via the ADV and then establishing primary system pressure and inventory control. Recovery techniques started with the termination of auxiliary feedwater to the affected loop generator and then involved the following, either separately or in combination: unaffected loop generator feed and steam (using auxiliary feed and ADV steam); primary feed and bleed (using SI and pressurizer PORV operation); pressurizer auxiliary spray; pressurizer internal heaters; and unaffected loop pump operation.

EXPERIMENTAL RESULTS

This section presents results from the Semiscale steam generator tube rupture experiments during which both signature response and operator recovery techniques were investigated. The discussion on signature response includes: general signature response of a tube rupture transient; effect of the number of tubes ruptured on system signature response; effect of tube rupture location on system signature response; effect of pressurizer initial liquid inventory on signature response; and signature response of a main steam line break compounded by a tube rupture. Topics specific to recovery techniques include: the effectiveness of steam generator feed and steam in cooling fluid in the loop and lowering primary system pressure; the effectiveness of pressurizer spray for reducing primary system pressure: the effectiveness of pressurizer PORV operation for reducing primary system pressure; and the effectiveness of using SI and pressurizer internal heaters to increase fluid inventory and subcooling. In the discussion of these topics, phenomena such as condensation, flashing, natural circulation, primary-to-secondary system heat transfer, and overall mass and energy distributions are examined.

Signature Response for Tube Rupture Transients

Tube rupture signature response was investigated in the Semiscale Mod-2B system by establishing a primary-to-secondary system flow through a scaled conical flow break tube near the tube sheet of the affected loop generator. This section characterizes the primary and secondary system response during the early portion of the tube rupture transient (prior to recovery procedures) and explains the thermal-hydraulic driving mechanisms causing the behavior.

Signature response is also presented in Appendix B, using special primary system pressure versus hot leg fluid temperature plots for all experiments of the steam generator test series. These types of plots are commonly used as part of abnormal transient operating guidelines (ATOG) in PWR plants.

General Signature Response for a Tube Rupture Transient. The occurrence of a tube rupture in the Semiscale system during typical PWR operating conditions has a very distinctive signature response. The signature response to a tube rupture transient can be characterized by such parameters as primary and secondary system pressures, liquid levels, fluid flow rates, and temperatures. For discussion purposes, an experiment involving a single, cold-side tube rupture in the Semiscale system is used for this section. The signature response is discussed for a time period of 600 s, which was assumed to include only automatically occurring events without operator action.

The tube rupture, occurring at t = 0 s, caused a primary system depressurization and loss of primary mass to the affected loop secondary system. Figure 4 compares the primary and secondary system pressure early in the transient. Primary system fluid, originally at 15.54 MPa (2247 psia), flowed through the conical flow tube break orifice into the affected loop steam generator, which was initially at 5.58 MPa (809 psia). The loss of mass from the primary system caused a steady primary depressurization until the pressurizer emptied at about t = 134 s (Figure 5), at which time the primary system depressurization rate increased. The increase in primary system depressurization rate, corresponding to the interfacial liquid level[a] of the pressurizer reaching the bottom of the pressurizer and entering the surge line, is attributed to a change in the amount of free surface area and flashing of saturated pressurizer fluid. As long as the interfacial level was above the bottom of the pressurizer and not in the surge line, the interfacial surface area was high, promoting flashing which, in turn, retarded the primary system depressurization. When the interfacial liquid level reached the surge line (due to break flow), the interfacial surface area decreased, which retarded flashing and resulted in an increase in depressurization rate. Shortly after the pressurizer interfacial level cleared the bottom of the pressurizer,the low pressurizer pressure setpoint of 13.1 MPa (1900 psia) was achieved (t = 146 s), automatically causing core power scram to the ANS decay curve and MSIV closure on both steam generators.

a. Interfacial level is a "pooled" liquid level with saturated steam above and saturated liquid below and is determined using a differential pressure measurement.

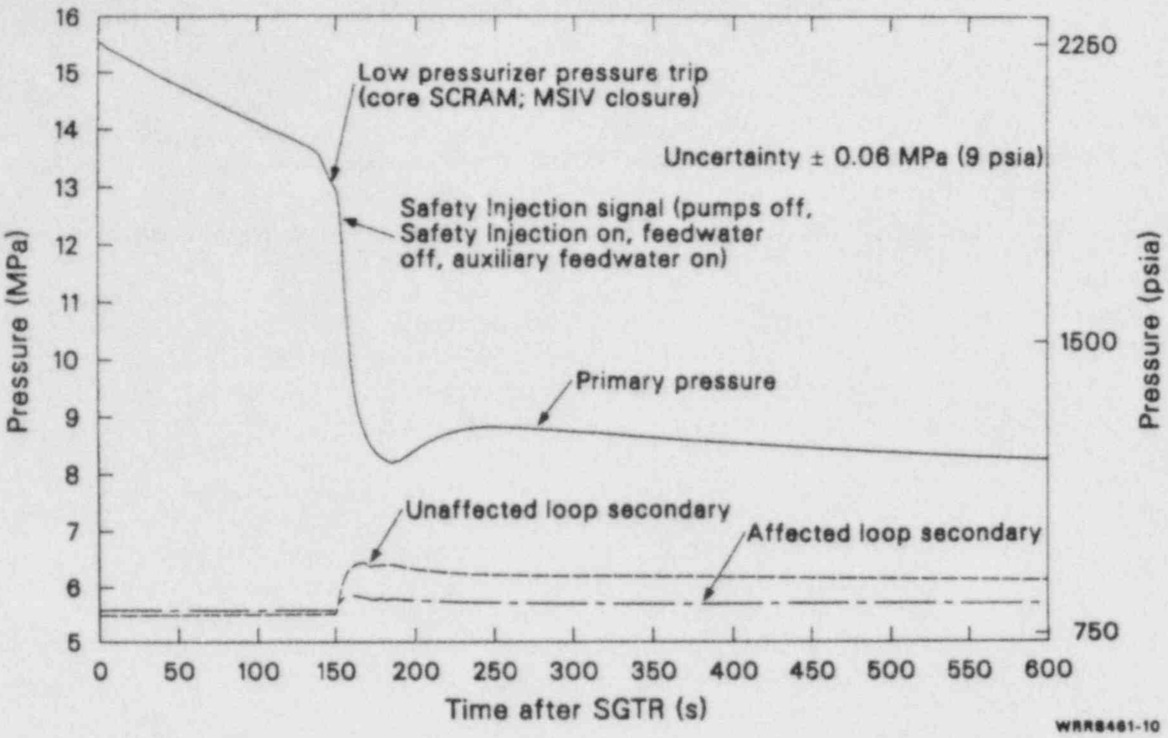


Figure 4. Primary and secondary system pressure during a cold-side, one-tube rupture transient.

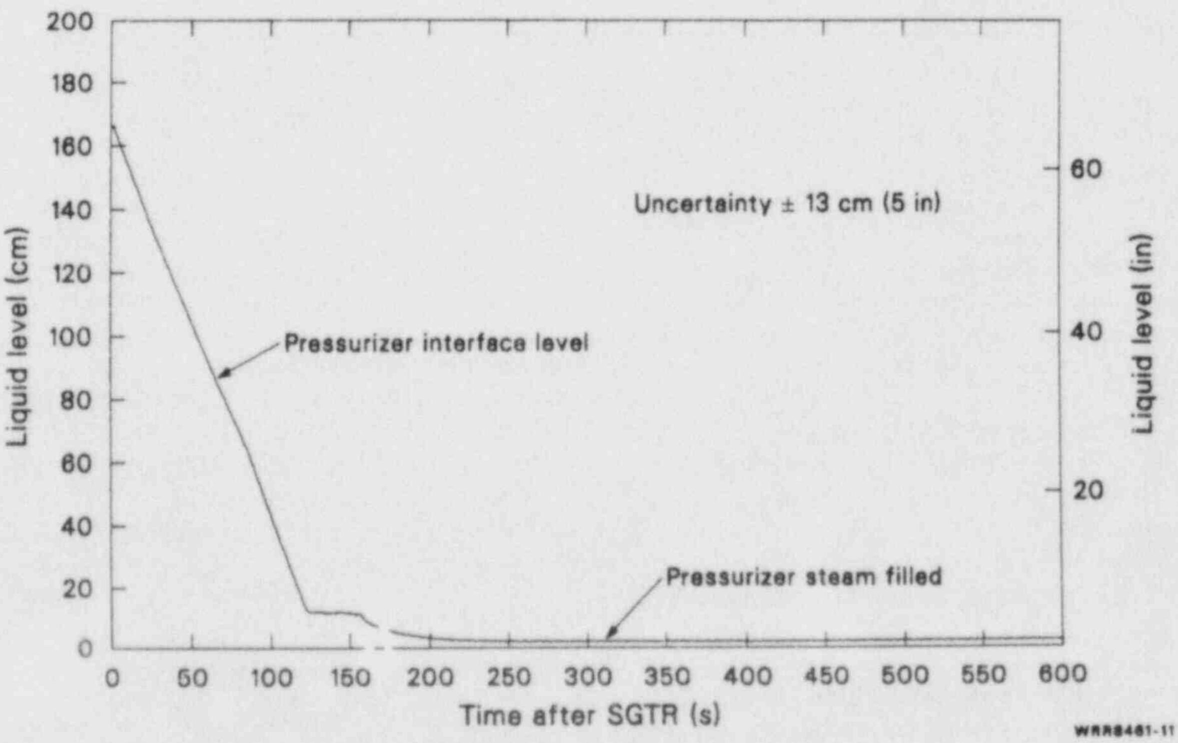


Figure 5. Pressurizer interfacial liquid level during a cold-side, one-tube rupture transient.

Upon MSIV closure, primary-to-secondary-system heat transfer in both the affected and unaffected loop steam generators caused a rapid pressurization of the secondary system, as shown in Figure 6. Prior to achieving the low pressurizer pressure trip, both the unaffected and affected loop steam generator secondary pressures remained fairly constant, as core power was removed via normal secondary steaming conditions through an open MSIV. The energy addition due to tube rupture break flow from the primary system to the secondary system caused essentially no rise in affected loop secondary pressure during this early period. This was because the energy removal during normal steaming was about six times the energy addition due to tube rupture break flow. Following MSIV closure, the pressure rose briefly to the ADV setpoint pressure in both generator secondaries. During the first 600 s, the ADV's cycled several times in each generator as primary-to-secondary heat transfer caused boiling in the secondary system. The inflow of auxiliary feedwater into the unaffected loop and combined break flow plus auxiliary feedwater into the affected loop compared to the mass expelled through the ADV operation, causing a slow filling trend in each generator, as shown in Figure 7.[a] At the end of 600 s, neither generator was full; however, the affected loop generator was within 75 cm (29.5 in.) of the top.

Following core scram, the primary system pressure showed an increase in depressurization rate as the primary fluid cooled due to primary-to-secondary-system heat transfer without full core power. The primary fluid cooling caused volumetric shrinkage, resulting in primary system depressurization. No major change in primary depressurization occurred when the primary pressure reached the SI signal [12.51 MPa (1814 psia)], which automatically initiated the following: termination of power to the primary coolant pumps; initiation of SI; and termination of main feedwater and start of auxiliary feedwater to the secondaries. The effects of the automatic SI events were overshadowed by the rapid reduction of core power and primary fluid shrinkage due to primary-to-secondary heat transfer. Eventually, the primary system depressurization was sufficient for the hot leg fluid to reach

a. Collapsed level refers to all the fluid (both steam and liquid) between the differential pressure measurement tap being treated as saturated liquid only.

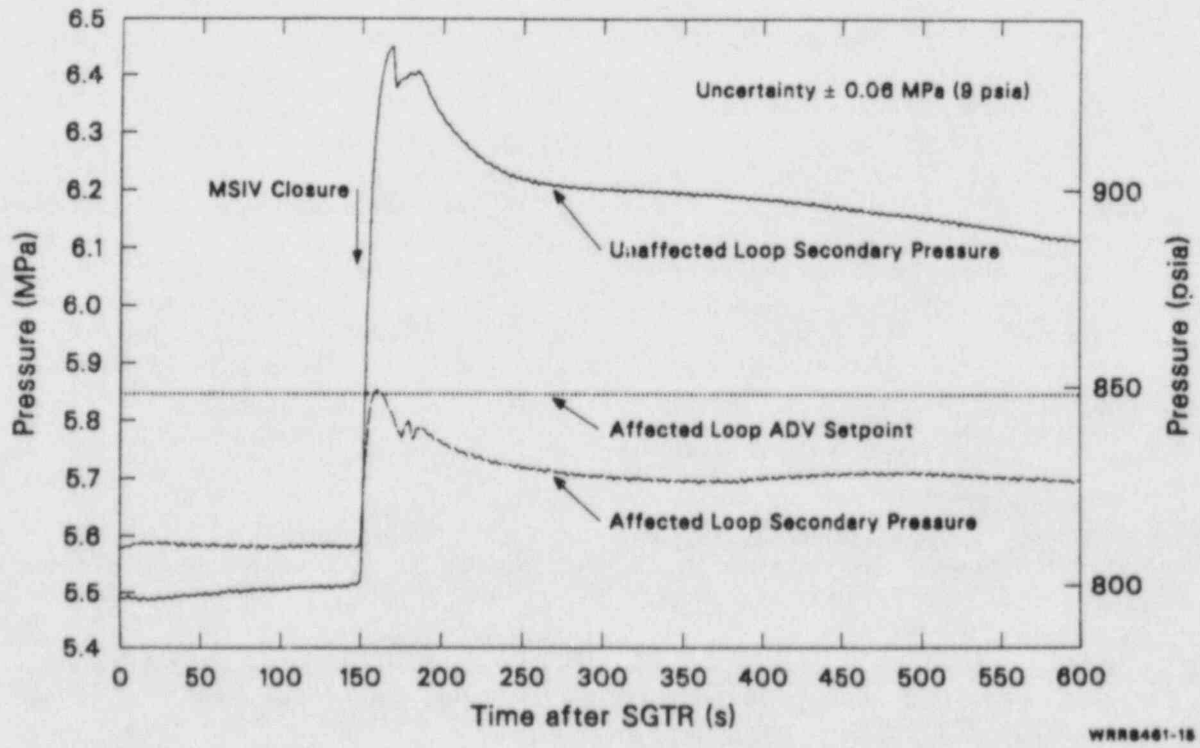


Figure 6. Unaffected and affected loop secondary pressure during a cold-side, one-tube rupture transient.

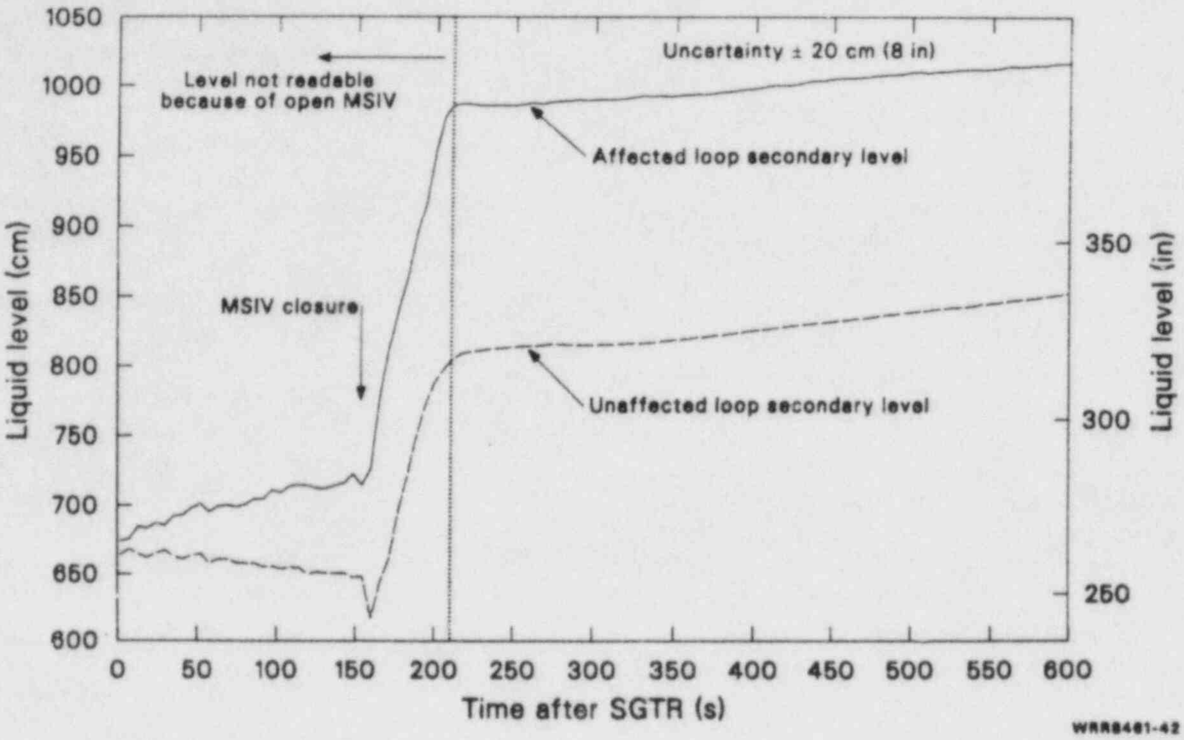


Figure 7. Unaffected and affected loop secondary collapsed liquid level during a cold-side, one-tube rupture transient.

saturation conditions at about t = 220 s (Figure 8). Flashing in the system then caused a major reduction in the depressurization rate. The primary system pressure made a slight recovery between 190 and 240 s. The repressurization was caused by a combination of superheated steam in the pressurizer, due to heat transfer from the pressurizer walls to pressurizer fluid (Figure 8), and the change from forced circulation to natural circulation heat transfer in the steam generators that occurred as the primary pumps coasted down (Figure 9). During the first 600 s of the single-tube rupture transient discussed here, the natural circulation mode was single-phase liquid natural circulation; and the magnitude of the flow rate is typical of single-phase results found previously in Semiscale separate-effects experiments.[5]

Following the slight primary system repressurization period (190 to 240 s), the primary pressure first stabilized, then followed a slow depressurization, but remained above the affected loop ADV setpoint for the entire initial 600-s period. This slow depressurization was supported by a combined energy balance, including SI flow, primary-to-secondary-system heat transfer, break flow, flashing in the hot leg, and system environmental heat loss.

During the operator identification period (the first 600 s), only minor system mass voiding occurred. Figure 10 compares the primary steam generator tube collapsed level and the vessel upper head collapsed level. The primary tubes remained essentially full, and the vessel upper head level was reduced to 375 cm (148 in.) above the cold leg. Because of the positive differential pressure between the primary and affected loop secondary, a positive break flow persisted throughout this early period; however, SI flow, once initiated, was slightly larger than break flow, as shown in Figure 11. The slight filling trend in vessel upper head level during the first 600 s, as shown in Figure 10, was caused by a slightly larger SI flow than break flow. During this vessel filling, the pressurizer remained steam-filled with no liquid filling trend (Figure 5).

In summary, during the single-tube rupture, after 600 s of only automatically occurring events, the system was in a single-phase natural circulation mode supported by a heat source (core decay heat) and a heat sink (lower steam generator secondary pressure than primary pressure). Primary system feed using SI was slightly higher than break flow into the affected loop generator secondary, resulting in a slight filling trend in the vessel upper head while the pressurizer remained empty and the steam

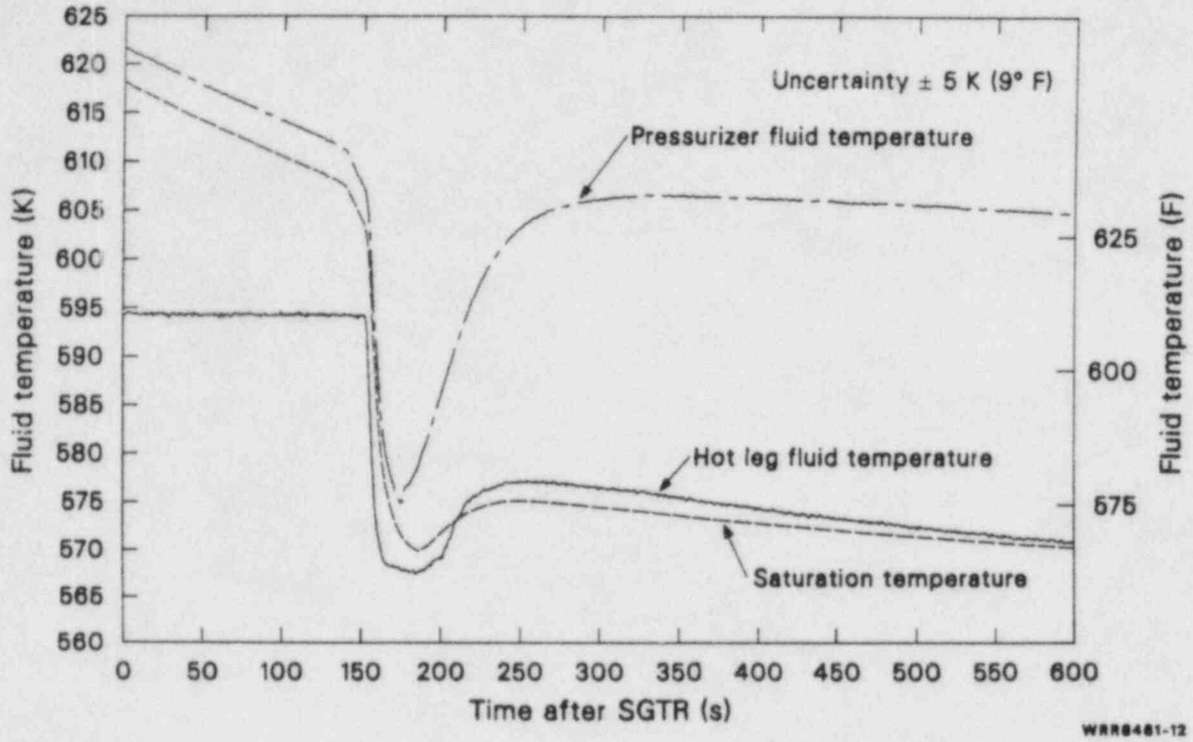


Figure 8. Pressurizer and hot leg fluid temperature and the saturation temperature during a cold-side, one-tube rupture transient.

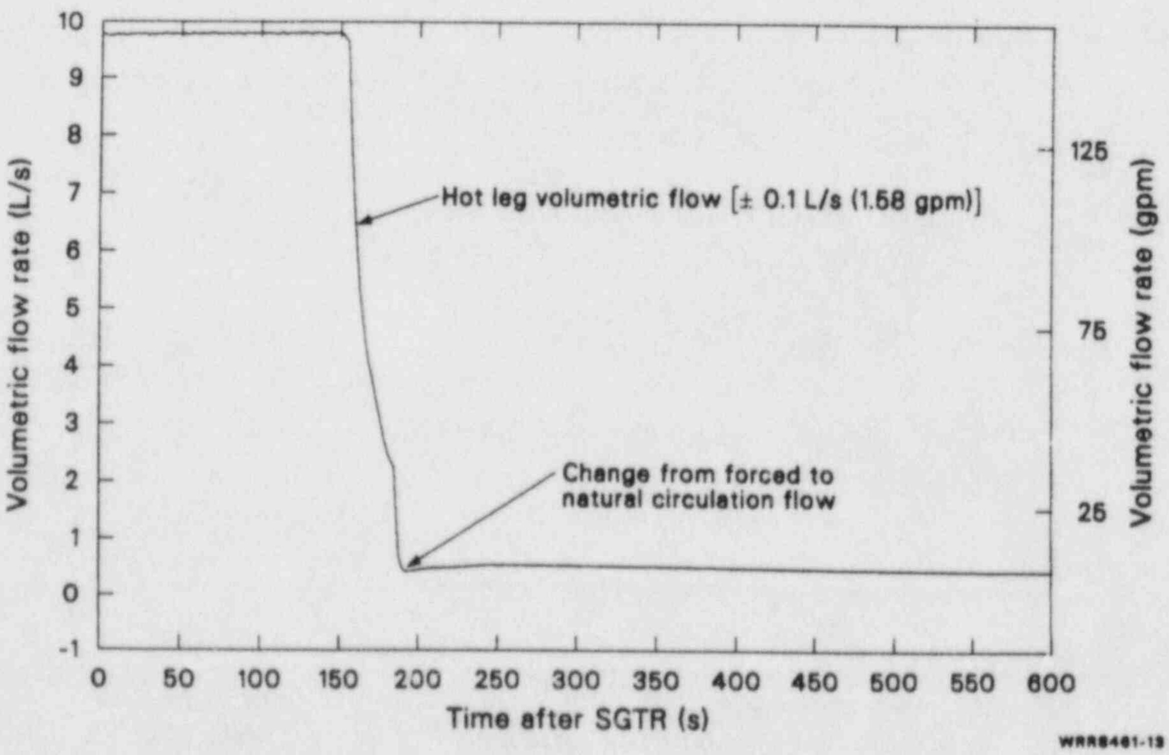


Figure 9. Hot leg volumetric flow during a cold-side, one-tube rupture transient.

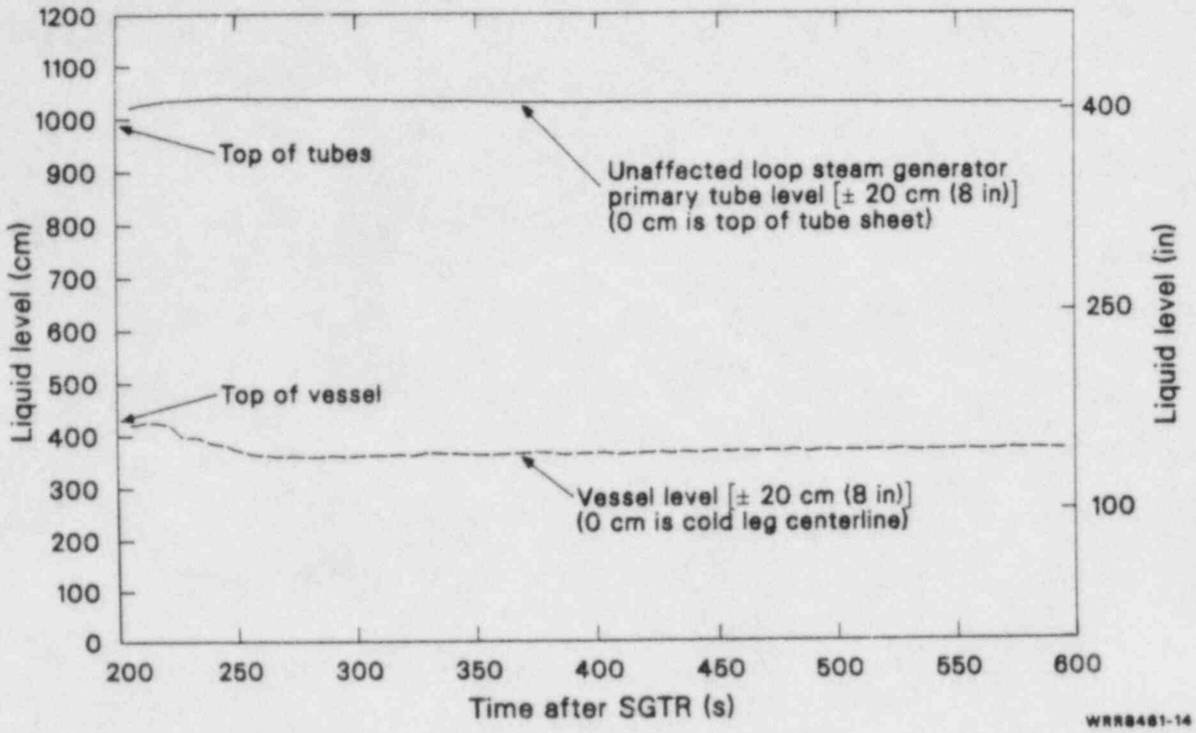


Figure 10. Collapsed liquid level for the unaffected loop steam generator primary tube and vessel upper head during a cold-side, one-tube rupture transient.

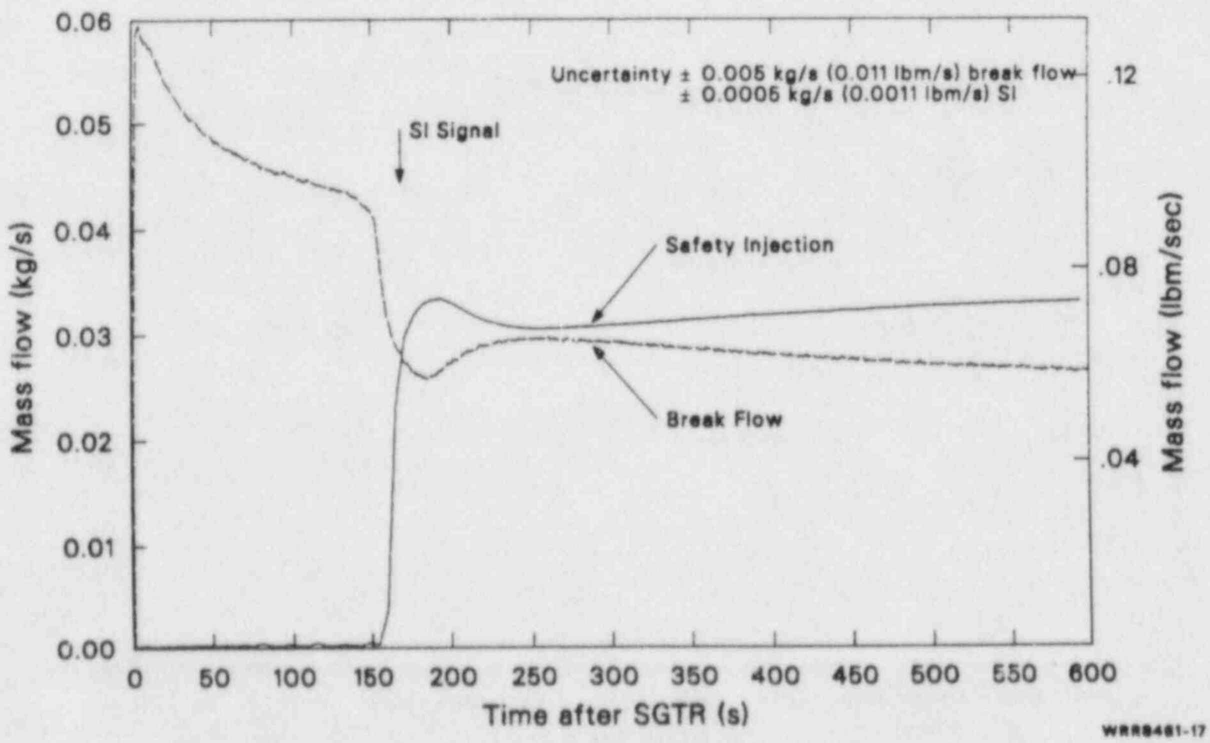


Figure 11. Tube rupture break flow and SI flow during a cold-side, one-tube rupture transient.

generator primary system tubes remained full. Following the rapid primary system depressurization associated with core scram, the primary system depressurized slowly and remained above the affected loop ADV setpoint pressure at the end of 600 s. The slow depressurization was the result of an overall energy balance, including fluid shrinkage, due to primary system environmental heat loss and primary-to-secondary heat transfer, combined with break flow competing against core decay heat, flashing, and SI flow.

The Effect of the Number of Tubes Ruptured on Signature Response. Variation of the throat area of the conical flow tube allowed simulation of multitube ruptures in the Semiscale facility. One-, five-, and ten-tube rupture experiments were performed to examine the effect of the number of tubes ruptured on the basic signature response.

The overall system response for a one-tube, five-tube, and ten-tube rupture is similar; however, the timing of events is quite different. Figure 12 shows the same rapid primary system depressurization to saturation conditions for a one-, five-, and ten-tube rupture. The most rapid depressurization period corresponds to core scram, as discussed previously; however, the initial depressurization rate (prior to core scram) was quite different for each case, resulting in a different low pressurizer pressure trip times and thus scram times (16.4 s for the ten-tube case, 32 s for the five-tube case, and 146 s for the one-tube case). The depressurization rate prior to scram increased proportionally with increasing number of tubes ruptured. The depressurization rate was 0.0128 MPa/s (1.85 psia/s) for one tube, 0.065 MPa/s (9.43 psia/s) for five tubes, and 0.120 MPa/s (17.4 psia/s) for ten tubes.

The attainment of repressurization following saturation in the primary system is more pronounced for the one-tube break than for either the five- or ten-tube breaks. For the larger number of tubes ruptured, there is a higher break flow (Figure 13) which dominates the pressurization effects, including pressurizer steam superheat and the change from forced to natural circulation in the loop. The similarity in response for the break spectrum was also seen in pressurizer collapsed liquid level (Figure 14) and the secondary pressurization in the steam generators (Figure 15), implying similar phenomena for the entire break spectrum.

The fundamental difference between the one-tube, five-tube, and ten-tube ruptures was the relationship between break flow and SI flow. The break

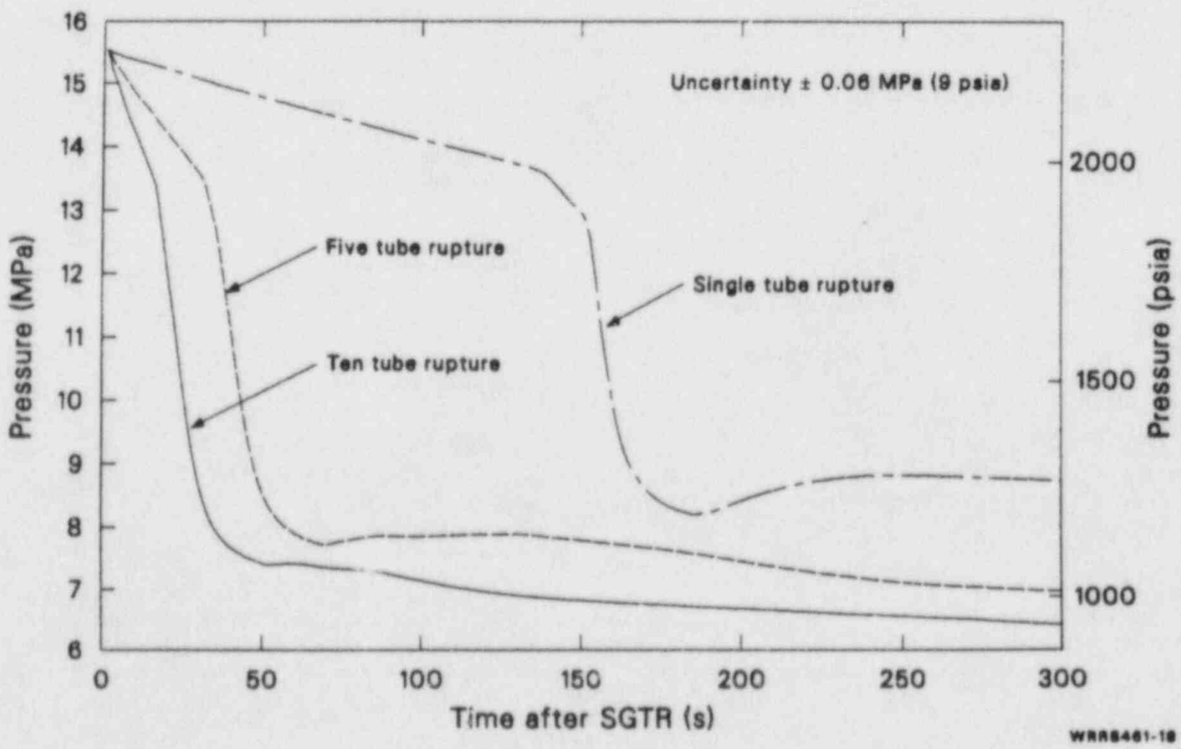


Figure 12. Primary system pressure response for a one-, a five-, and a ten-tube rupture transient.

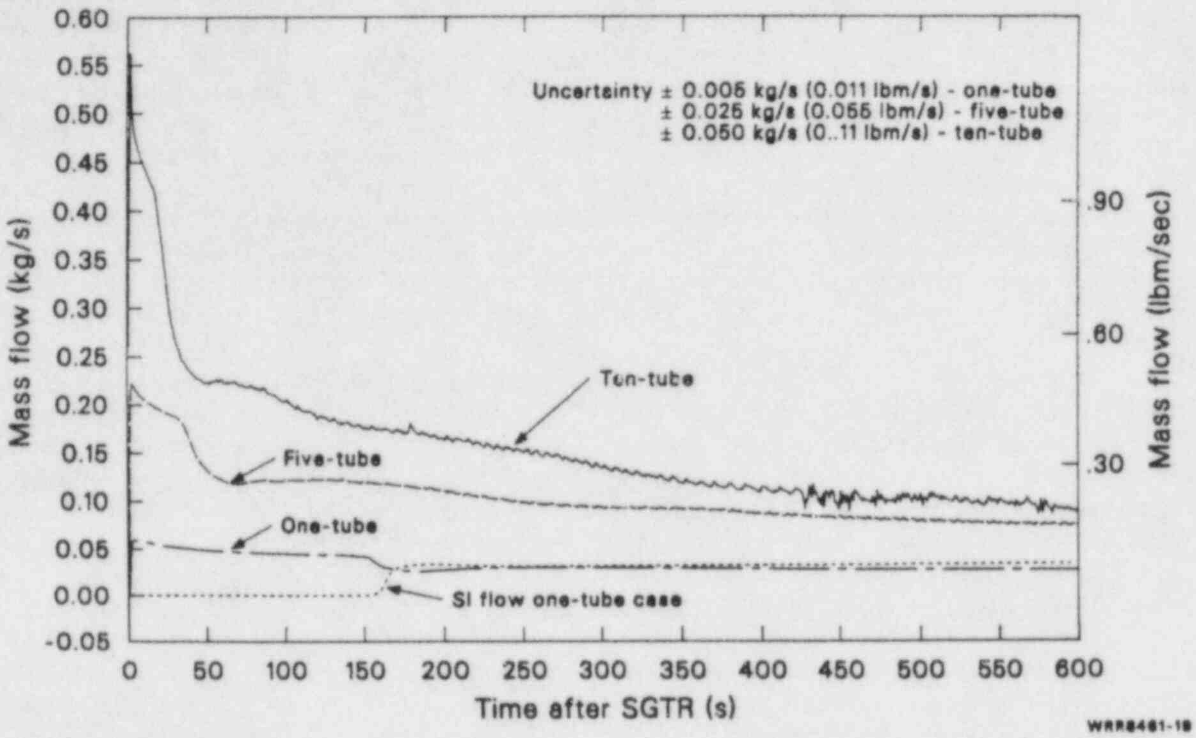


Figure 13. Break flow and SI flow for a one-, a five-, and a ten-tube rupture transient.

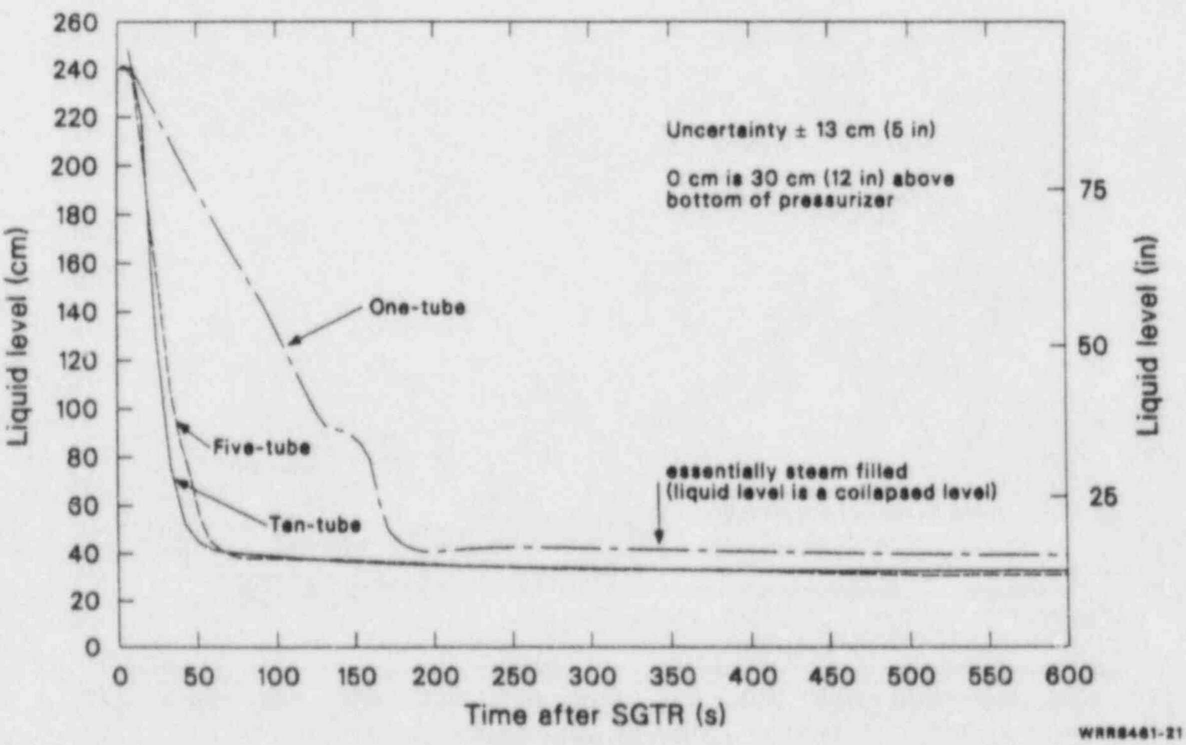


Figure 14. Pressurizer collapsed liquid level for a one-, a five-, and a ten-tube rupture transient.

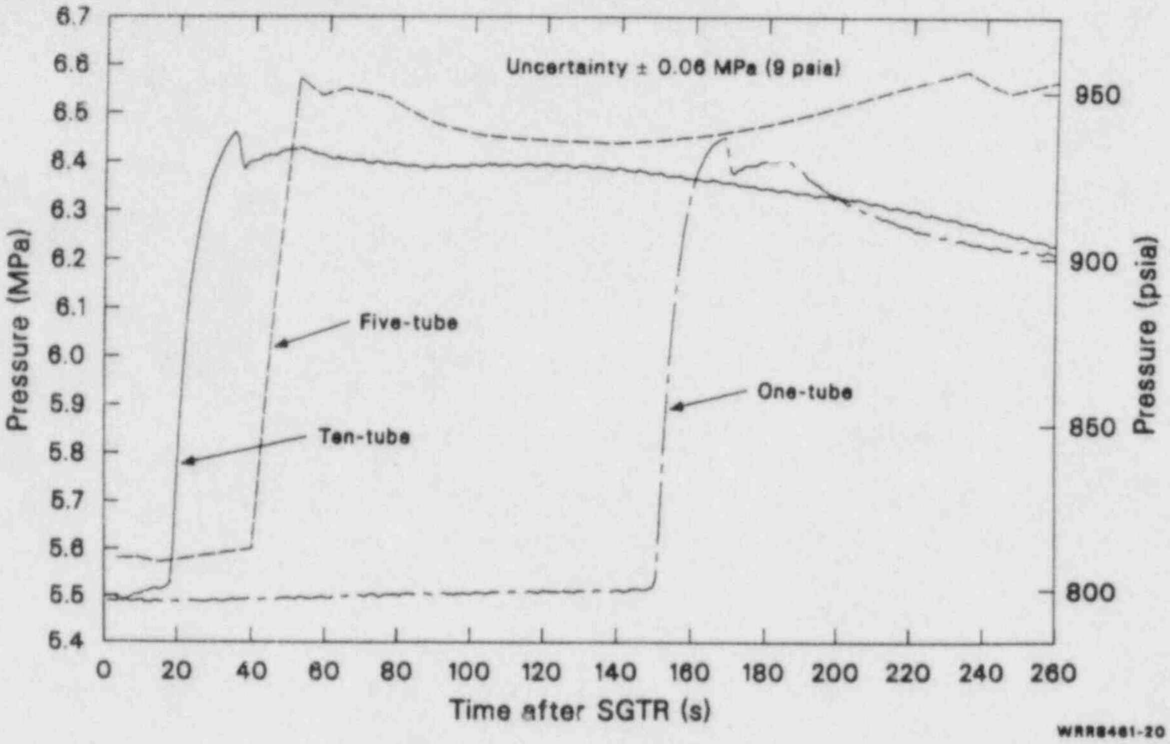


Figure 15. Unaffected loop secondary pressure for a one-, a five-, and a ten-tube rupture transient.

flow was much higher in relation to the SI flow for the five- and ten-tube ruptures (Figure 13), resulting in more extensive vessel voiding as shown in vessel collapsed level (Figure 16). This difference in system voiding resulted in considerably different system mass inventories (Figure 17), which led to different natural circulation modes in the loop at the end of 600 s. For the single-tube case, the inventory was about 87%, which is typical for a single-phase natural circulation approximation.[5] At t = 600 s, the five-tube case had an inventory of about 60%; for the ten-tube case, the inventory was 52%. These inventories for the five- and ten-tubes cases are typical of a reflux condensation mode of natural circulation.[a,5]

In summary, the thermal-hydraulic response for the entire break spectrum studied in Semiscale was similar except for the timing of events. All three experiments show increased depressurization to the saturation condition upon core scram. The core scram time was dependent on the early depressurization rate which was proportional to the tube rupture break size and break flow rate. With larger break flow for the five- and ten-tube ruptures, there was more system voiding, resulting in completely different modes of natural circulation at the end of the operator diagnostic period (first 600 s). For the five- and ten-tube rupture transients, a reflux-condensation mode predominated; and for the single-tube case, a single-phase mode existed.

The Effect of Tube Rupture Location on Signature Response. Two five-tube rupture transients initiated from identical initial and boundary conditions were performed in the Semiscale system, one with the break at the inlet and one with the break at the outlet of the steam generator. The object was to assess whether the difference in hydraulic conditions was sufficient to cause differences in signature response. The inlet and outlet represent the maximum difference in initial hydraulic conditions in the steam generator and therefore should produce the maximum difference in initial break flow and signature response.

The signature responses for cold side (inlet) and hot side (outlet) tube ruptures were found to be essentially identical during the early portion of five-tube rupture transients in Semiscale. Even though

a. During reflux, steam generated in the core travels to the steam generator, where it is condensed in both the up side and down side of the tubes. Steam that was condensed in the up side runs back into the core via the hot leg counter-current to the steam flow. Steam condensed in the down side travels as liquid to the pump suction and cold leg.

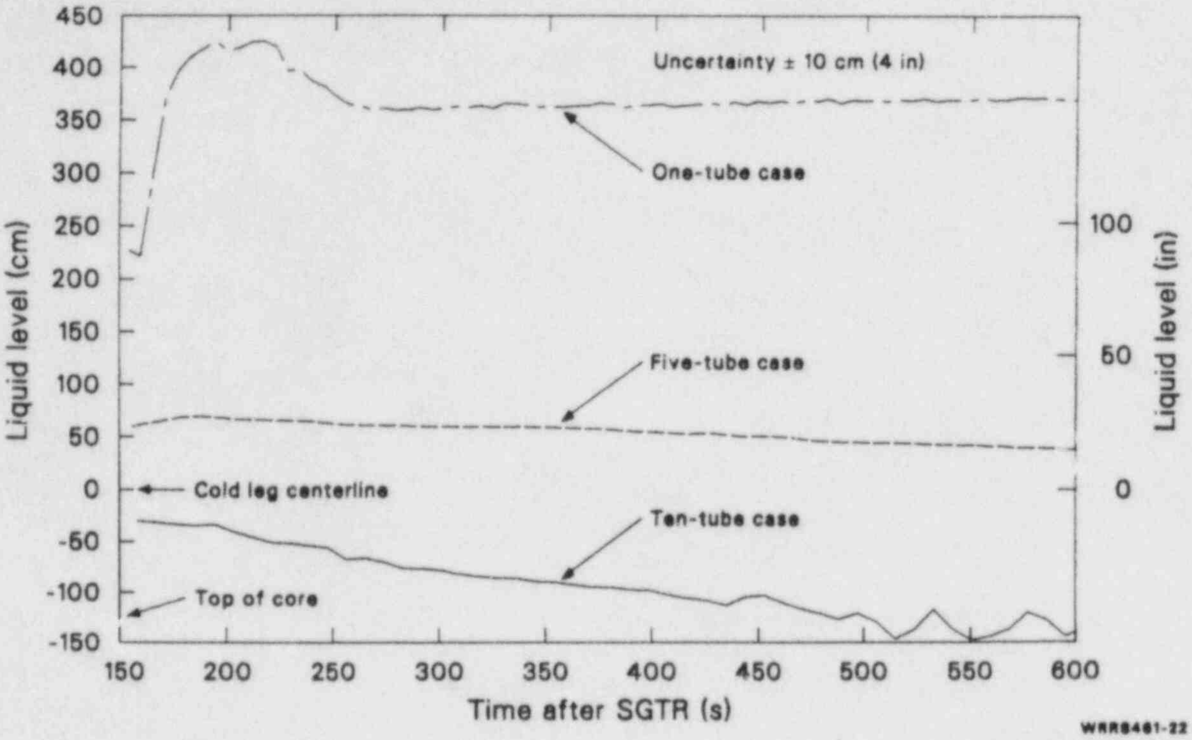


Figure 16. Vessel collapsed liquid level for a one-, a five-, and a ten-tube rupture transient.

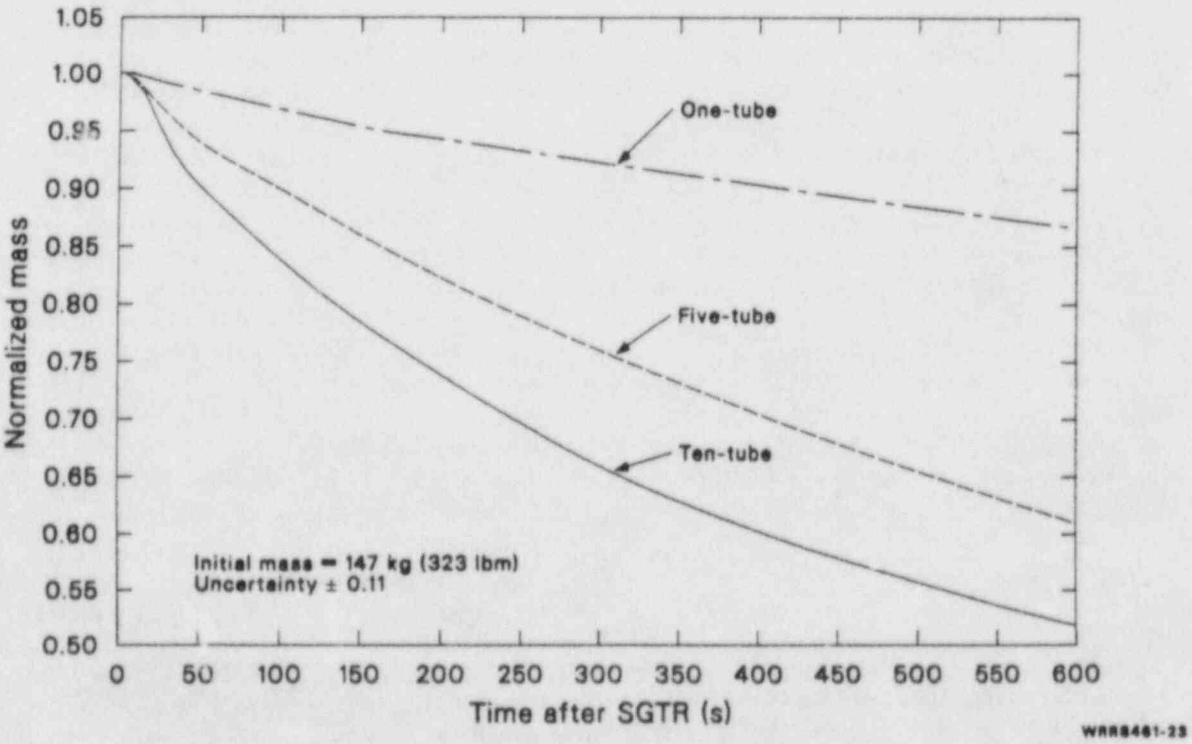


Figure 17. System mass inventory for a one-, a five-, and a ten-tube rupture transient.

the initial fluid density for the hot side tube rupture near the break was only about 89% of the density in the cold side [676 kg/m^3 (42 lbm/ft^3) for the hot side and 755 kg/m^3 (47 lbm/ft^3) for the cold side], the break flow was not greatly effected by tube rupture location, as shown in Figure 18. This is because the flow through the conical flow tube is proportional to the square root of the fluid density, which should result in about a 5% difference in flow rate (within the uncertainty of the measurement). Since the break flows were similar, the primary system depressurization rate (Figure 19) and pressurizer drain time (Figure 20) were also similar.

In summary, tube rupture location (whether at the inlet or the outlet of a tube) has essentially no effect on tube rupture signature response. Signature response parameters such as primary pressure, break flow rate, and pressurizer liquid level are essentially identical for inlet and outlet side breaks. Even though the hydraulic conditions (fluid densities) are different at the inlet and outlet, flow out of the break is proportional to the square root of the density and the effect is negligible.

The Effect of Pressurizer Initial Liquid Level on Signature Response. During normal operation in a PWR, the pressurizer is about half full of saturated liquid. Internal pressurizer heaters immersed in this liquid are used to maintain loop subcooling and pressure control. This is accomplished by establishing a steam bubble which pushes against the liquid-full system. It has been shown previously that a significant change in primary system depressurization results when the interfacial liquid level in the pressurizer is reduced due to break flow from the bottom of the pressurizer into the surge line. Therefore, differences in the initial pressurizer liquid level could have an effect on the early signature response of a tube rupture. To investigate this, two otherwise identical five-tube rupture transients were performed in the Semiscale Mod-2B facility. One experiment had an initial pressurizer liquid volume of about 42%, and the other had a 27% initial pressurizer liquid volume. The experiment with 42% initial pressurizer liquid volume is more representative of correct volume scaling to a PWR.

The initial pressurizer liquid level was found to have a large effect on primary system depressurization during the operator response period (0 to 600 s), thus affecting the timing of certain automatically occurring events. On an overall basis, differences in initial pressurizer liquid level do not

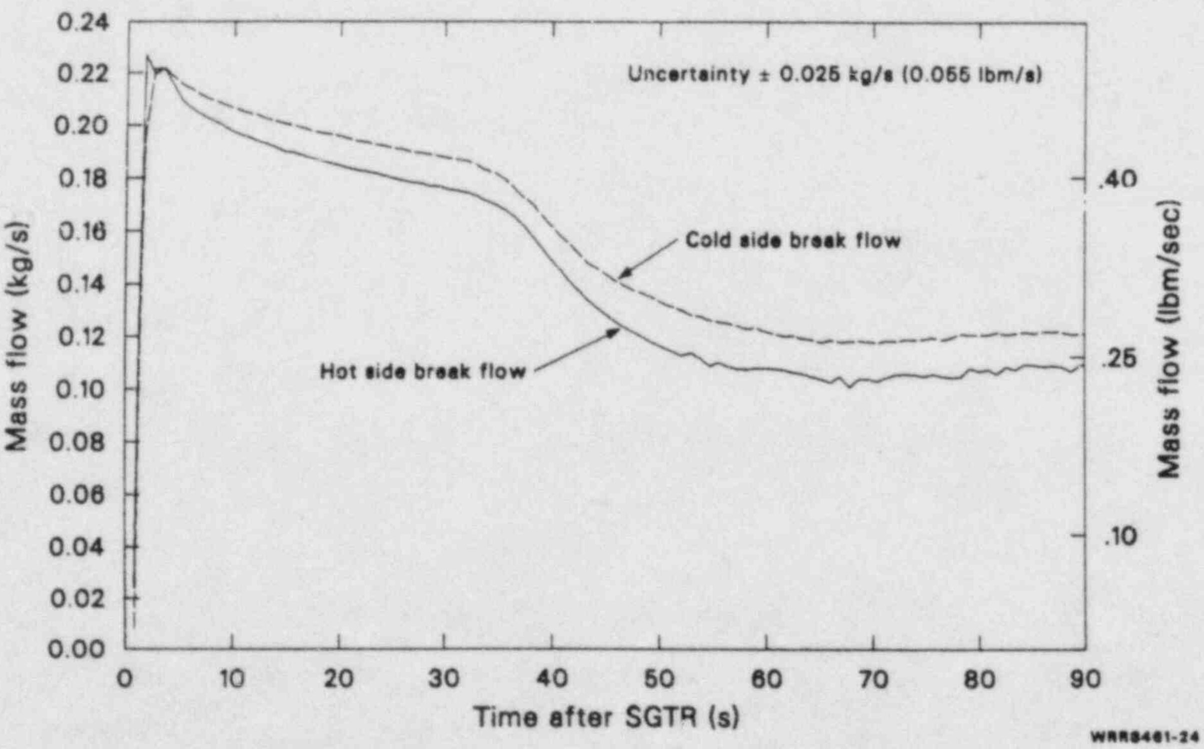


Figure 18. Break flow for a hot-side and a cold-side five-tube rupture transient.

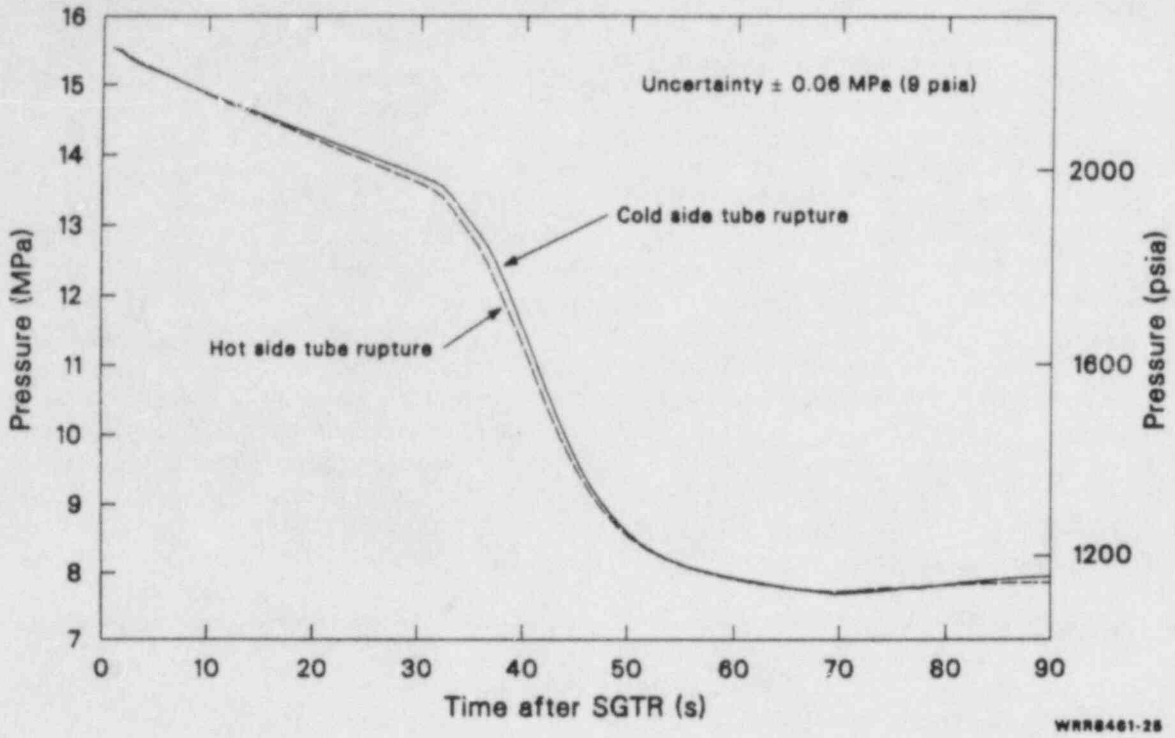


Figure 19. Primary system pressure for a hot-side and a cold-side five-tube rupture transient.

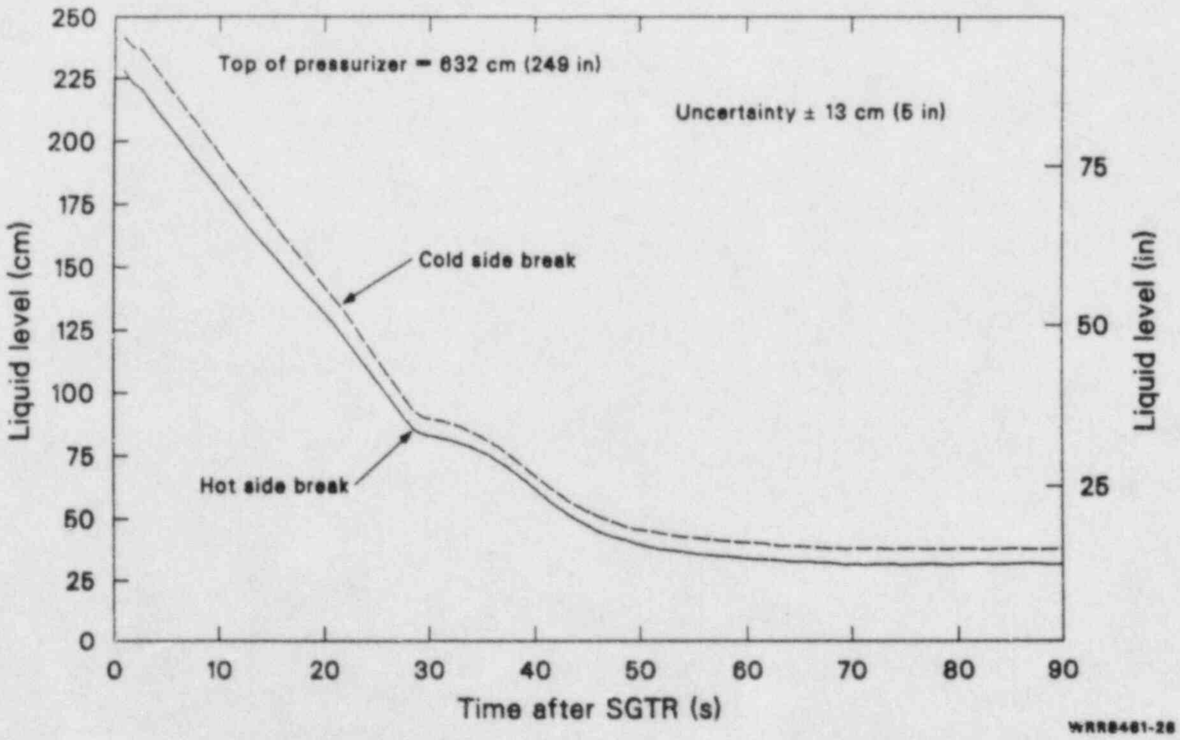


Figure 20. Pressurizer collapsed liquid level for a hot-side and a cold-side five-tube rupture transient.

cause differences in long-term system response characteristics, such as primary pressure and stored energy in the primary fluid. Figure 21 compares the primary system pressure for the five-tube rupture experiments with normal and low initial pressurizer liquid volumes. The overall trend of depressurization is identical, but the time to scram is considerably different (32 s for the normal pressurizer level case and 18 s for the low pressurizer level case). For the first 12 s, the primary system depressurization for the two experiments is nearly identical, as essentially identical break flow (Figure 22) removes mass from the primary system. At 12 s, the primary system depressurization rate for the low pressurizer initial liquid volume case shows a great increase, while the normal liquid volume case continued to depressurize at the same rate. Coincidently, at about 12 s, the interfacial level in the pressurizer for the low initial liquid volume case reached the surge line, resulting in a large change in interfacial surface area for flashing. With the higher initital liquid level in the pressurizer for the normal case, the pressurizer interface surface area for flashing remained higher for a longer period of time, which caused a retarding effect on depressurization. With the higher system pressure for the normal initial pressurizer level case, the break flow remained higher for a longer period of time (Figure 22); however, the overall vessel level was about the same after 100 s. Figure 23 compares the vessel upper head collapsed liquid level for the two experiments. The additional mass in the pressurizer for the normal case simply went out through the break, leaving the system mass inventory similar.

The stored energy in the system fluid following pressurizer drain appears to be the same for the two cases, as shown in Figure 24 which compares hot leg fluid temperature for both. Prior to scram, 2 MW of core power was added to the primary system for 20 s more in the normal level case than in the low level case. Even though core power was on 20 s longer for the normal case, the feedwater termination and MSIV closure were also delayed 20 s. As a result, the extra energy delivered to the primary fluid was simply dissipated by continued pretransient steam and feed in the unaffected and affected loop secondaries; i.e., after 100 s, the hot leg fluid temperatures are similar, indicating similar fluid energy content.

In summary, a higher initial pressurizer collapsed liquid level (increased liquid volume) resulted in a slower depressurization and a longer time to scram. The thermal-hydraulic state of the system after the first few hundred seconds was identical, as the extra

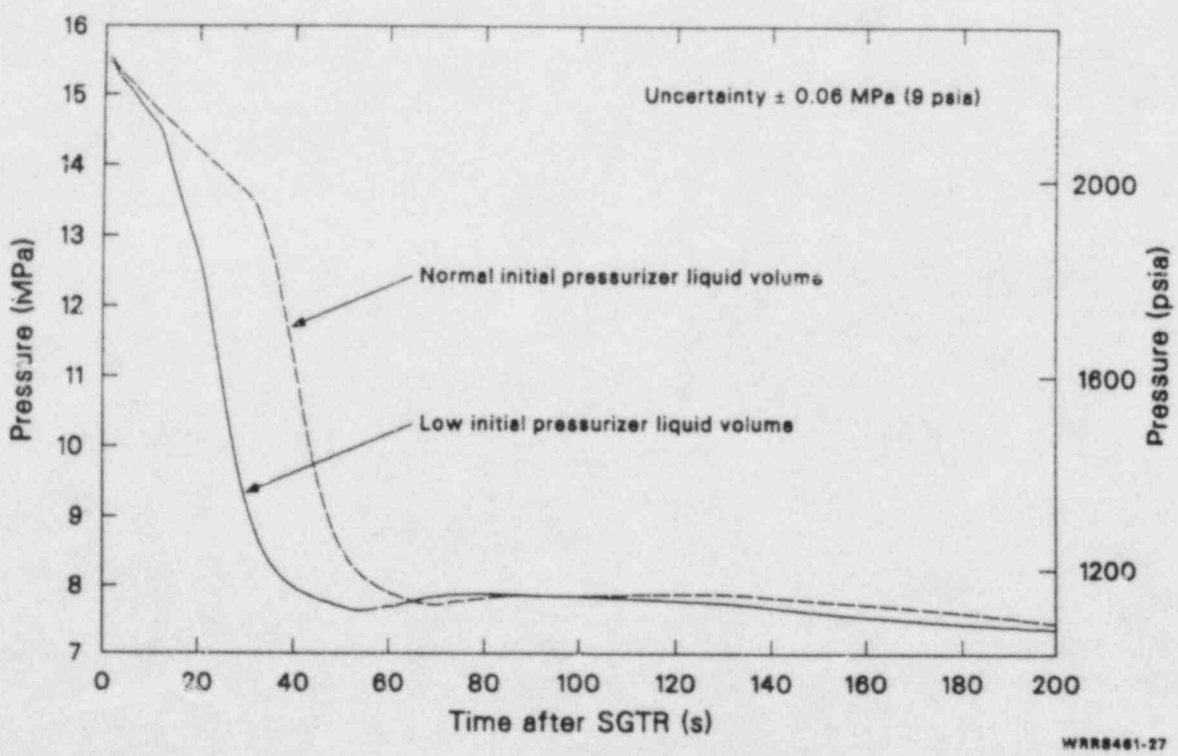
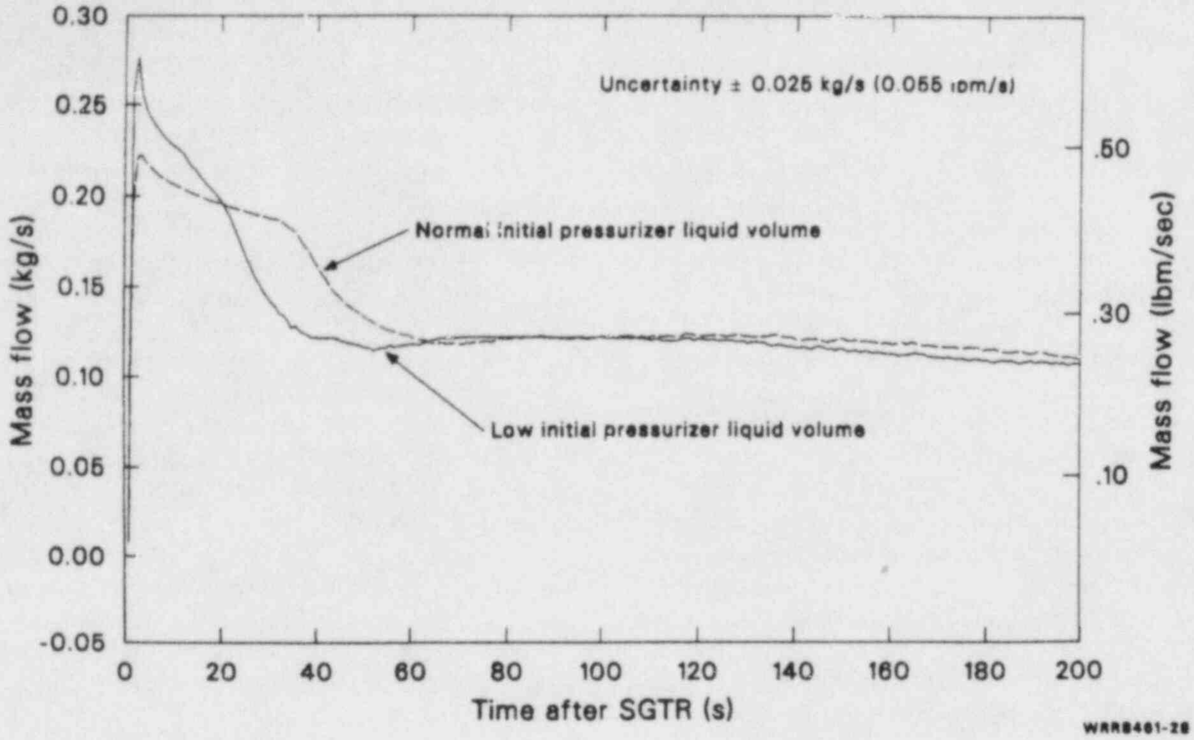
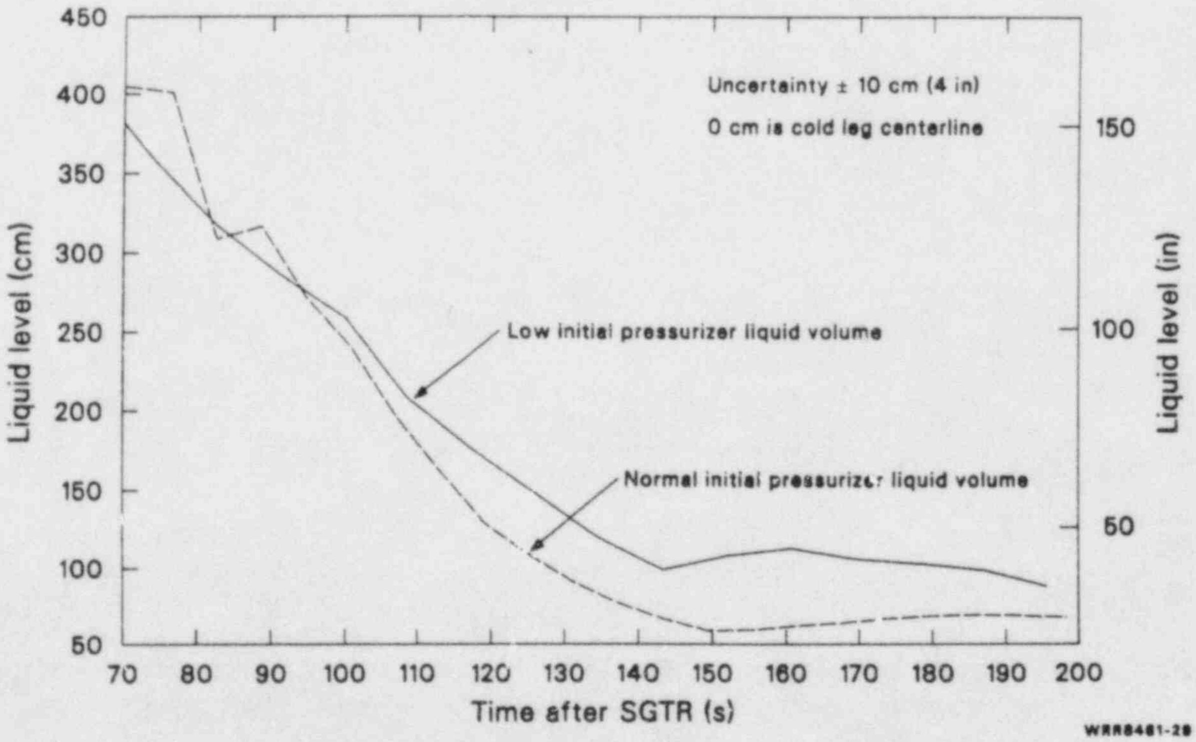


Figure 21. Primary system pressure response for two five-tube rupture transients with different initial pressurizer collapsed liquid levels.



WRR8461-28

Figure 22. Break flow for two five-tube rupture transients with different initial pressurizer collapsed liquid levels.



WRR8461-29

Figure 23. Vessel collapsed liquid level for two five-tube rupture transients with different initial pressurizer collapsed liquid levels.

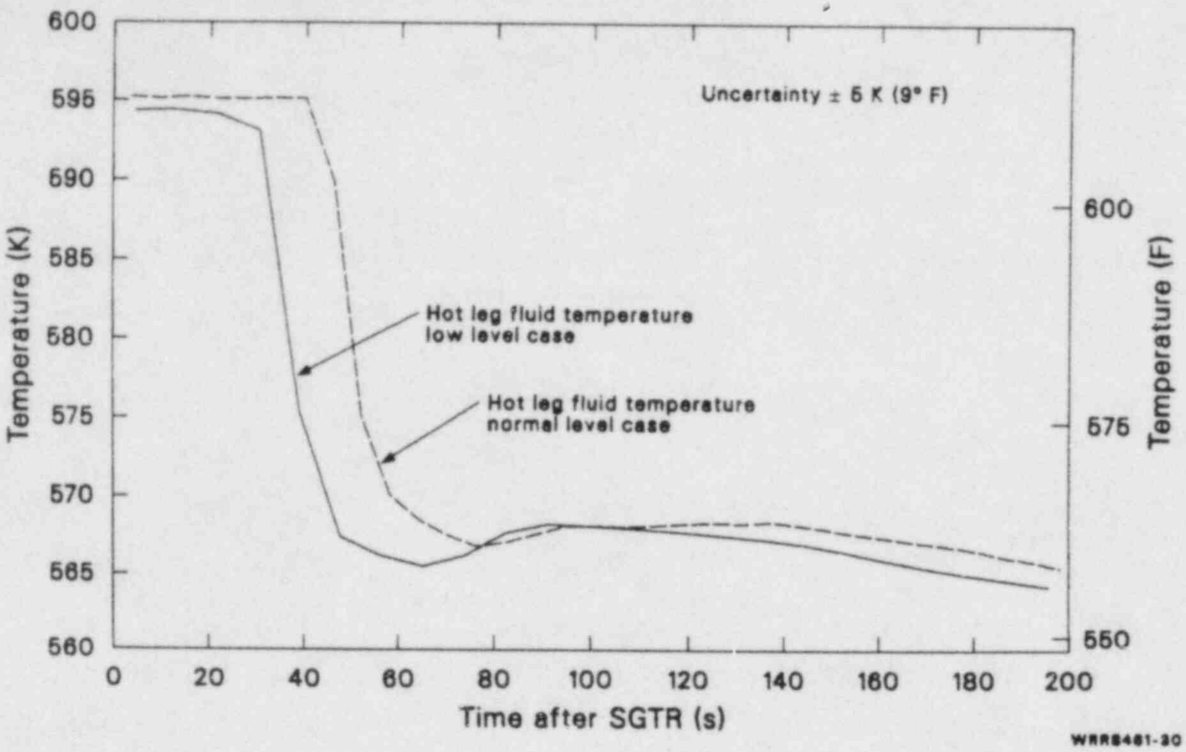


Figure 24. Hot leg fluid temperature for two five-tube rupture transients with different initial pressurizer collapsed liquid levels.

pressurizer mass for the normal case 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 the two cases. The extra core power delivered to the primary fluid for the normal level case, compared to the low level case, was dissipated in the longer secondary steam and feed time prior to scram and resulted in similar primary fluid temperatures.

Signature Response of a Main Steam Line Break with Concurrent Tube Rupture. In the event of a main steam line break in a nuclear power generating plant, it is possible to have a concurrent tube rupture due to the sudden increase in primary-to-secondary-system differential pressure. An experiment was performed in the Semiscale system with a main steam line break as the initiating event, followed 60 s later by a single-tube rupture. The signature response during the operator diagnostic period (0 to 600 s) is discussed in this section.

On an overall basis, the signature response for a main steam line break followed by a tube rupture is similar to the signature response for a steam generator tube rupture alone. Figure 25 compares the primary and secondary system pressures during the first 600 s of a main steam line break and tube rupture transient. At t = 0, the main steam line break occurs, causing a reduction in secondary pressure as steam flow is increased out the steam generator secondaries. Both the unaffected and affected loop generators show a decrease, because a common tie between the steam generator secondaries was assumed. The steam line break was assumed to occur downstream of the steam line flow restrictor in the affected loop steam line but also upstream of the MSIV. Prior to MSIV closure in the unaffected loop, fluid in the steam line communicates with the affected loop through a common header. Upon achieving a low secondary system pressure trip, the MSIV was closed in both loops, causing the pressurization of both secondaries (Figure 25). Following MSIV closure, no further fluid communication between the two generators existed. Following MSIV closure and concurrent core scram, the pressurization was not sustained in the affected loop because the steam line break continued to allow steam release. In addition, primary-to-secondary heat transfer was reduced because the core power was on the ANS decay curve. In the unaffected loop, the pressure slightly increased as primary-to-secondary heat transfer persisted with core power on decay heat with no steam release (MSIV closed and steam line break isolated). In general, the primary pressure response to the main steam line break was characterized by only a minor reduction in primary system pressure [from 15.6 to 14.2 MPa (2247 to 2059 psia)].

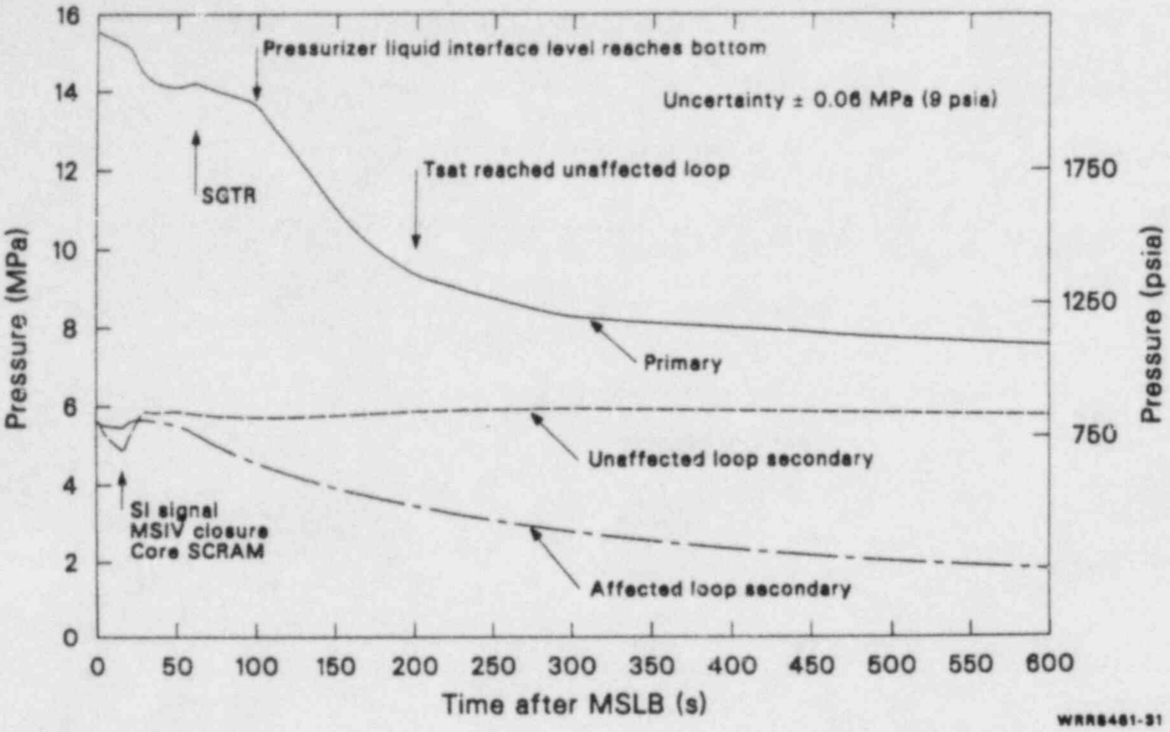


Figure 25. Primary and secondary system pressure during a main steam line break with concurrent one-tube rupture.

At t = 60 s, the tube rupture occurred, resulting in a depressurization of the primary system at various rates until saturation conditions were reached in the hot leg (Figure 26). Once hot leg saturation conditions were reached, flashing reduced the depressurization rate. A typical increase in primary depressurization occurred as the pressurizer interfacial liquid level entered the surge line, causing a reduction in flashing (see Figure 27). At 600 s, the primary system pressure was above the affected loop SRV setpoint; however, the affected loop secondary system pressure continued to drop as the main steam line break allowed a steam relief path to atmosphere.

The affected loop secondary system pressure decreased throughout the time period following the tube rupture (Figure 25). This was a direct result of tube rupture break flow being much smaller than the steam line break flow, as shown in Figure 28. As a result, the affected loop secondary system level[a] decreased throughout the time period 60 to 600 s, as shown in Figure 29, resulting in considerable atmospheric discharge of primary and secondary system fluid.

Break flow remained above SI flow until about 350 s, as shown in Figure 30. This resulted in a decrease in vessel upper head level (Figure 29). Break flow and SI flow were approximately equal[b] during the time period 350 to 600 s, resulting in a vessel upper head level of 260 cm (102 in.) above the cold leg at 600 s.

In summary, the primary system response for a concurrent main steam line break and tube rupture is similar to the signature response for the tube rupture event alone. Prior to the tube rupture, the main steam line break caused only a minor reduction in primary system pressure. Following the single-tube rupture break initiation at t = 60 s, the primary system pressure started a slow depressurization until the liquid interface level in the pressurizer entered the surge line. At that time, flashing in the

a. With the steam line break causing flow out the top of the affected loop generator, this measurement is for trend only. The level is probably within ± 100 cm (39 in.) of an actual collapsed level.

b. Figure 30 actually shows break flow being higher than SI flow. However, considering the uncertainty in the break flow measurement and a fairly level vessel upper head (Figure 29), break flow and SI flow are approximately equal.

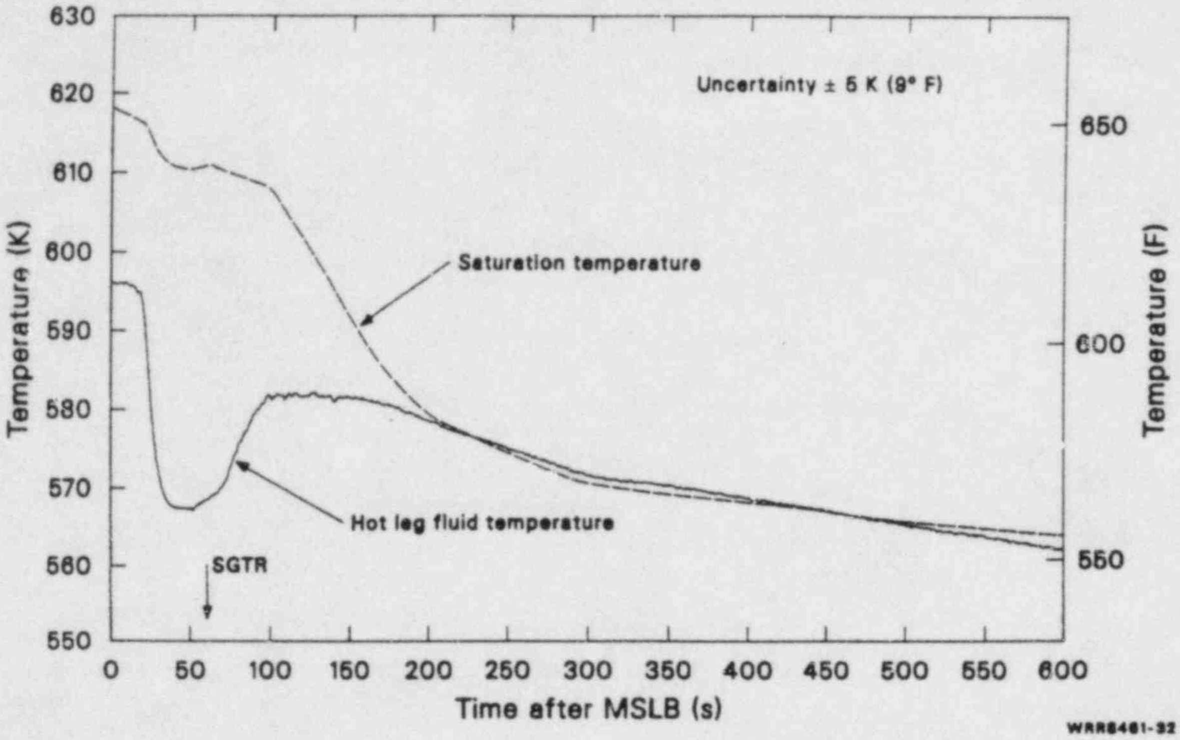


Figure 26. Hot leg fluid temperature and saturation temperature during a main steam line break with concurrent one-tube rupture.

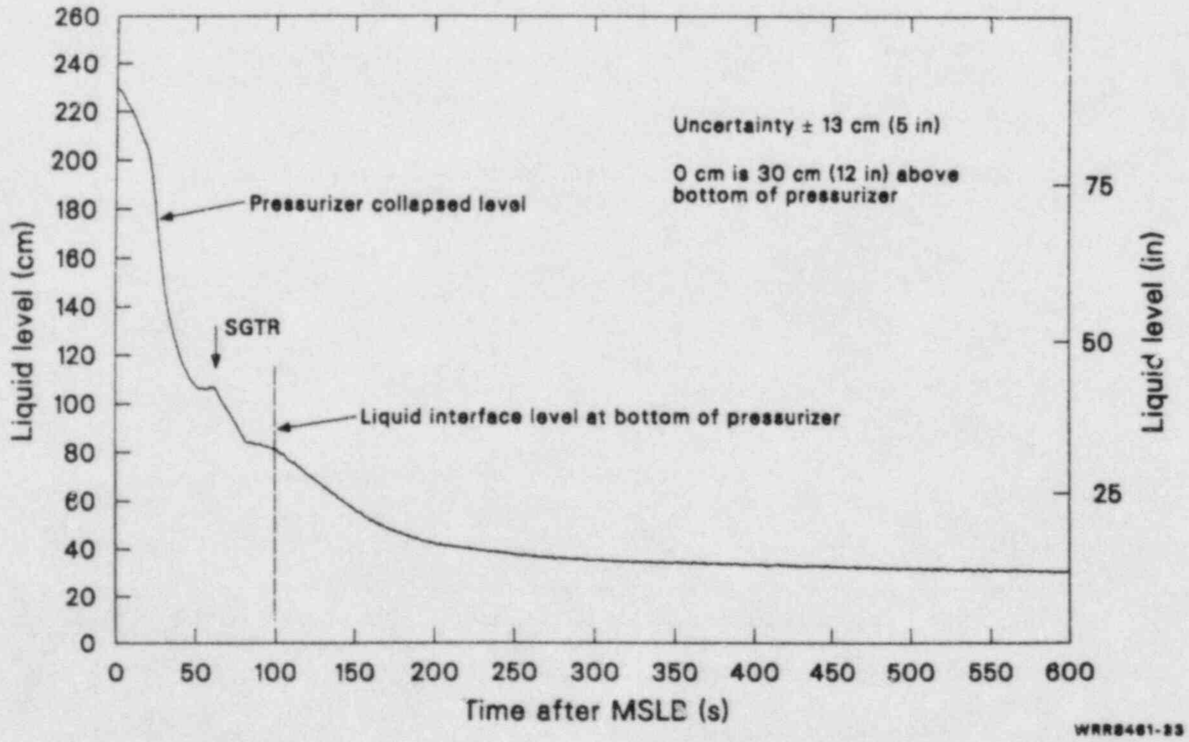


Figure 27. Pressurizer collapsed liquid level during a main steam line break with concurrent one-tube rupture.

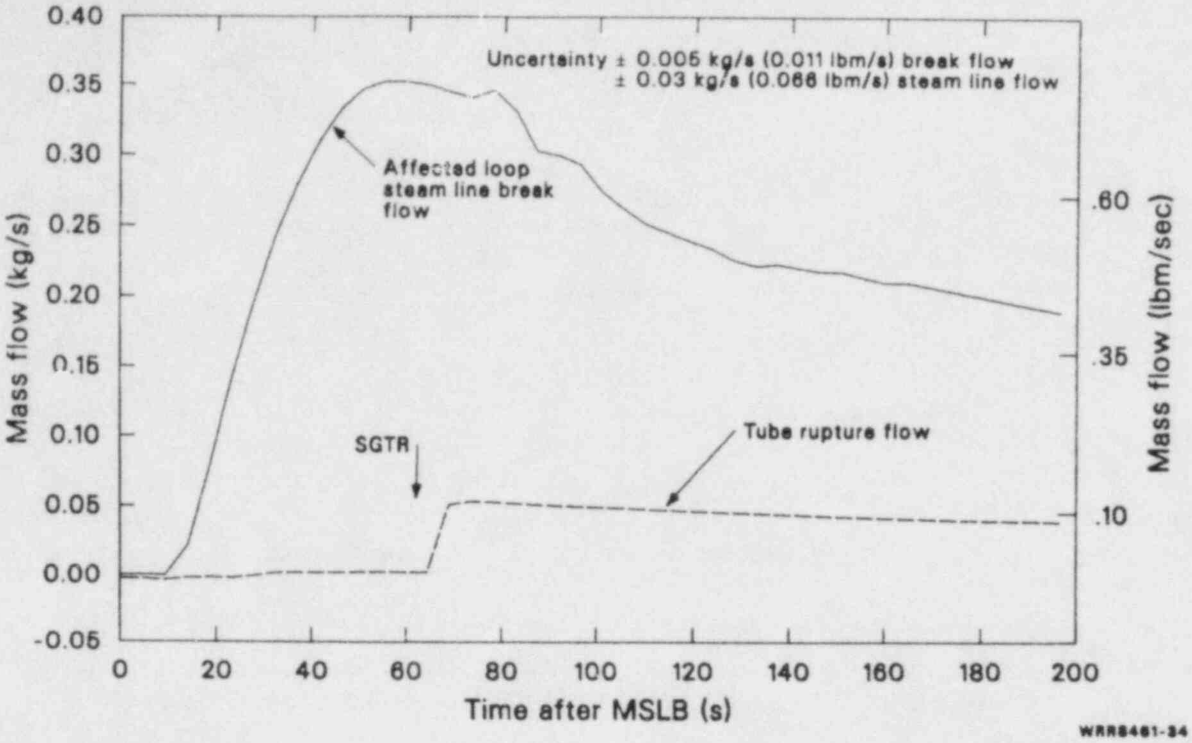


Figure 28. Affected loop steam line break flow and tube rupture flow during a main steam line break with concurrent one-tube rupture.

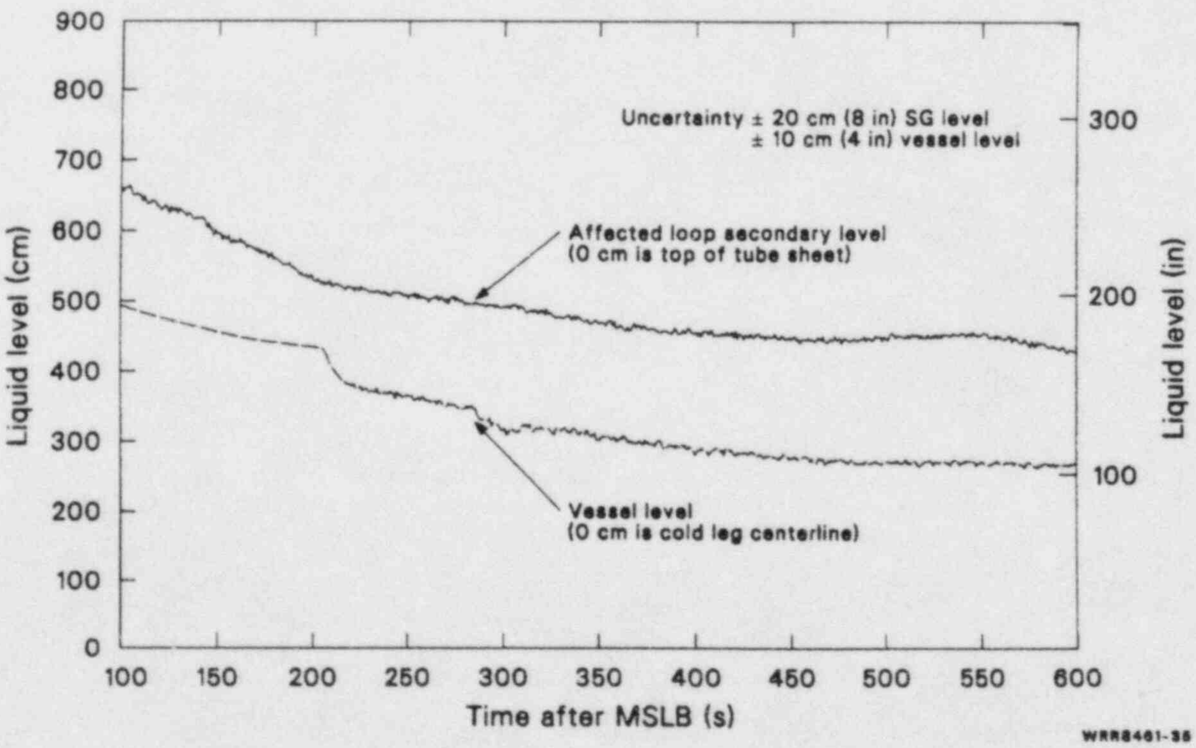


Figure 29. Vessel and affected loop secondary collapsed liquid level during a main steam line break with concurrent one-tube rupture.

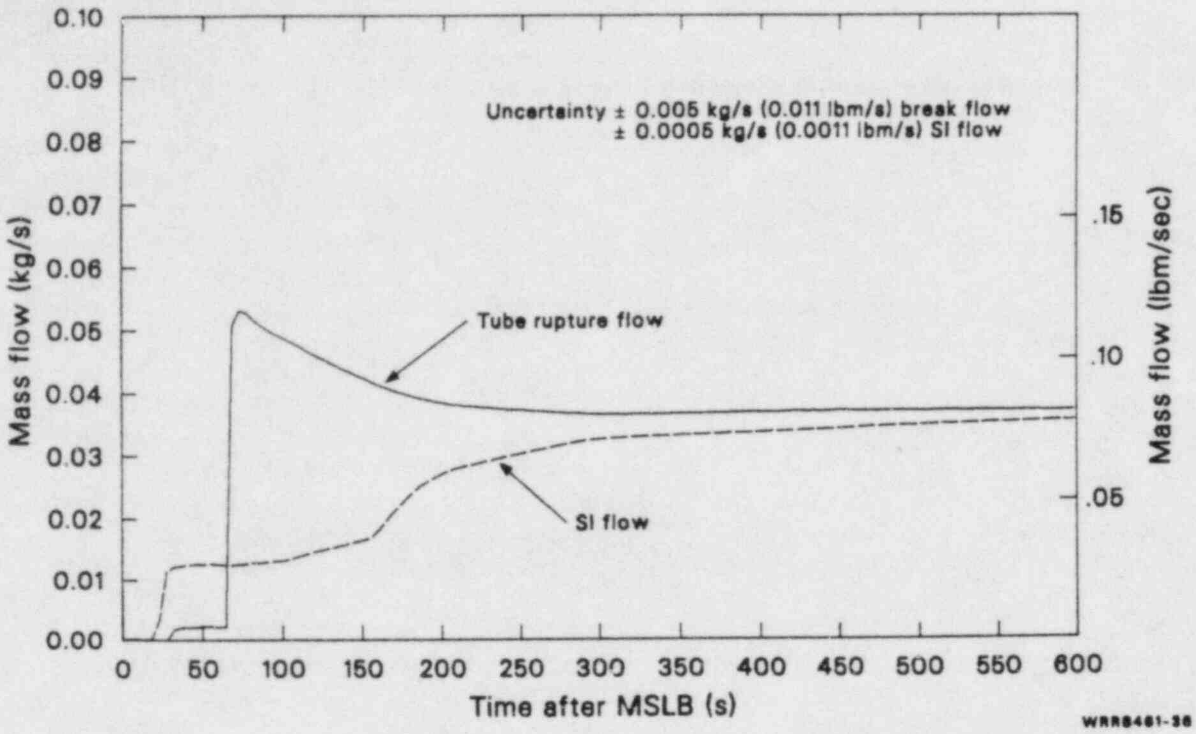


Figure 30. Tube rupture flow and SI flow during a main steam line break with concurrent one-tube rupture.

pressurizer was retarded, increasing the primary system depressurization rate. This relatively rapid primary depressurization persisted until saturation conditions were achieved in the hot legs, at which point flashing again retarded the depressurization rate. The main steam line break flow out of the affected loop generator was much higher than the single-tube rupture break flow into the affected loop generator, resulting in a decrease in affected loop secondary liquid level and a depressurization of the secondary system. This resulted in considerably more atmospheric discharge of primary and secondary system fluid than for the tube rupture event alone. The vessel upper head level remained about 260 cm (102 in.) above the cold leg at 600 s.

Recovery Techniques During a Tube Rupture

Following the operator diagnostic period, a PWR operator has a variety of means available to recover the plant from a tube rupture transient. Recovery requires first reducing the primary system pressure to below the affected loop generator ADV setpoint, thus terminating any atmospheric release of radioactive fluid, and then regaining control of both system fluid inventory and pressure. Operator action is dependent on other compounding failures, such as main steam line break, loss of onsite and offsite power, stuck open ADV in the affected loop generator, or a stuck open pressurizer PORV. Depending on the compounding failure, if any, the operator methods of recovery include: unaffected loop feed and steam, using ADV steam and auxiliary feedwater; primary system feed and bleed, using pressurizer PORV operation and safety injection; pressurizer auxiliary spray; pressurizer internal heaters; and SI operation. This section discusses the effectiveness of these operator-induced responses and the thermal-hydraulic phenomena governing these responses.

The Effectiveness of Unaffected Loop Feed and Steam on Pressure Control and Loop Fluid Cooling. Following the operator diagnostic period of a steam generator tube rupture transient, one of the operator options for loop pressure control and loop fluid cooling is to induce a feed and steam of the unaffected loop generators. The secondary feed is from auxiliary feedwater, and the steam is controlled by using the ADV's.

The Semiscale experimental results show that the effectiveness for pressure control and loop cooling due to feed and steam is dependent on the hydraulic state of the loop. As mentioned previously, the hydraulic state of the loop is dependent on the number of tubes ruptured, with the larger number of tubes ruptured producing a more voided system. With a more voided system, the core decay heat removal mechanism tends to be two-phase natural circulation or reflux condensation; while with a single-tube rupture, single-phase natural circulation occurs. What follows is a discussion of the role of the natural circulation mode on the effectiveness for primary pressure control and cooling of a feed and steam operation, and the redistribution of primary mass associated with the sudden increase in cooling in the steam generators.

The Role of the Natural Circulation Mode on the Effectiveness of Unaffected Loop Feed and Steam. The effectiveness of primary system pressure control by secondary feed and steam depends on the natural circulation mode in the loop. The natural circulation mode in the loop depends on system inventory,[5] which is a direct function of the number of tubes ruptured. Figure 31 compares the primary system pressure response following the onset of unaffected loop secondary feed and steam for both a single-tube rupture and a five-tube rupture. Both experiments show an increase in primary system depressurization rate following feed and steam initiation; however, the experiment with five tubes ruptured exhibited a much higher depressurization rate. This difference in depressurization rate is a direct result of different heat transfer mechanisms occurring in the steam generator at the onset of feed and steam. For the five-tube rupture case, the heat transfer/core heat rejection mechanism was reflux condensation; while for the single-tube rupture, the mode of heat rejection was single-phase natural circulation. Reference 5 describes reflux and single-phase natural circulation in the Semiscale system in terms of mass inventory and steam generator or core differential temperature. At the pressures encountered during the steam generator tube rupture [4 to 8 MPa (580 to 1160 psia)], reflux occurs at inventories below 65% with a nearly zero differential temperature (saturation conditions throughout the loop). Single-phase natural circulation occurs at system mass inventories of about 92% with differential temperatures dependent on core decay heat [30 K (54°F) at 60-kW core power]. Figure 32 compares system mass inventory for a one- and a five-tube rupture experiment, showing single-phase natural circulation type inventories for the one-tube case and reflux-type inventories for the five-tube case at the onset of feed and steam. Similarly, Figure 33 shows a nearly zero differential temperature for the five-tube case (indicating

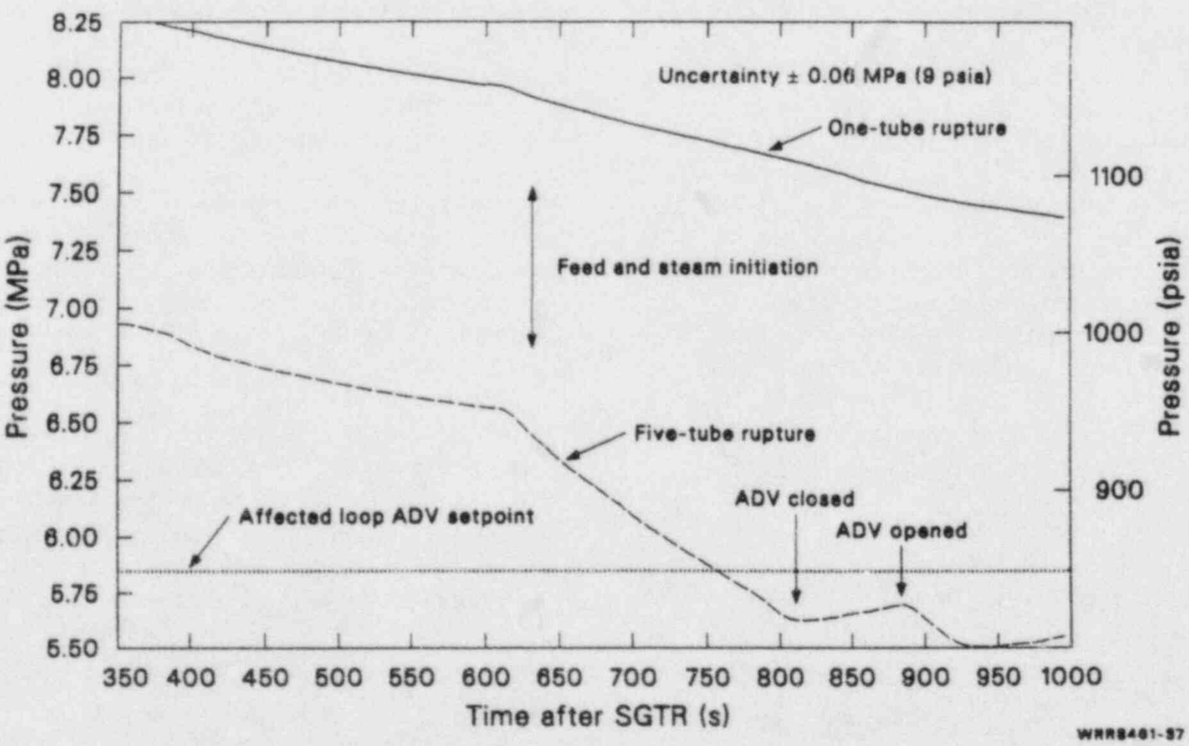


Figure 31. Primary system pressure response for unaffected loop feed and steam during recovery for a one- and a five-tube rupture transient.

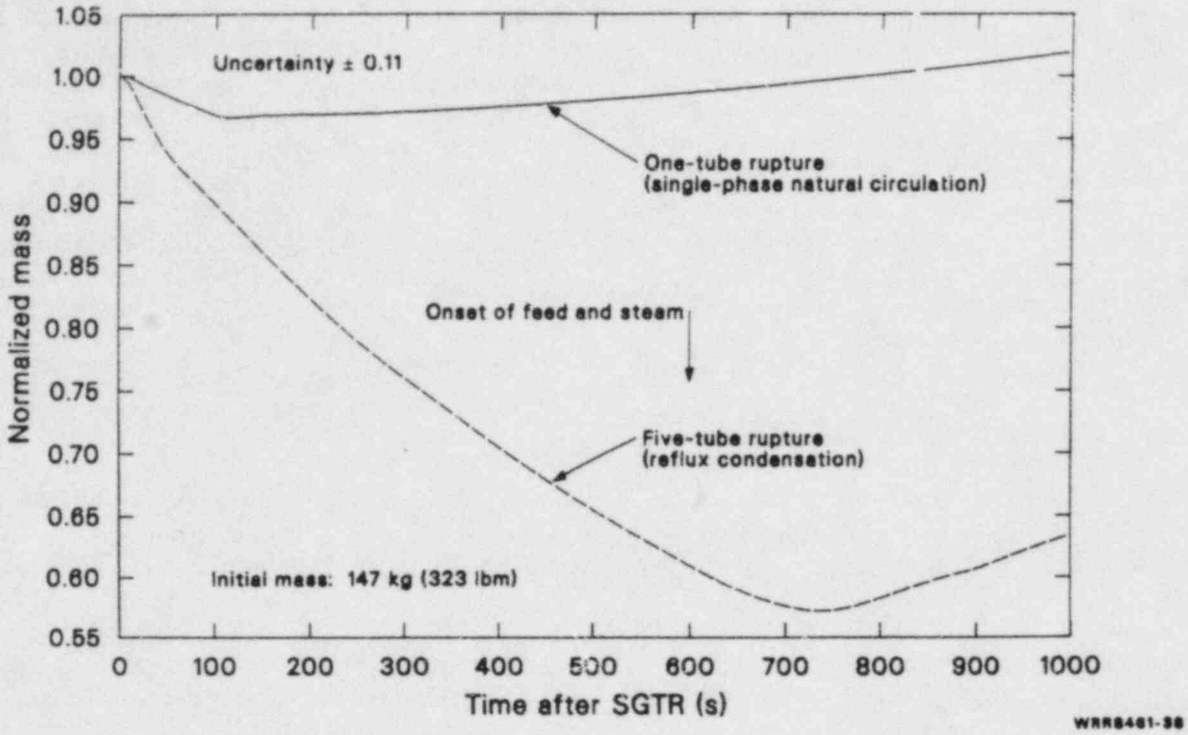


Figure 32. System mass inventory for unaffected loop feed and steam during recovery for a one- and a five-tube rupture transient.

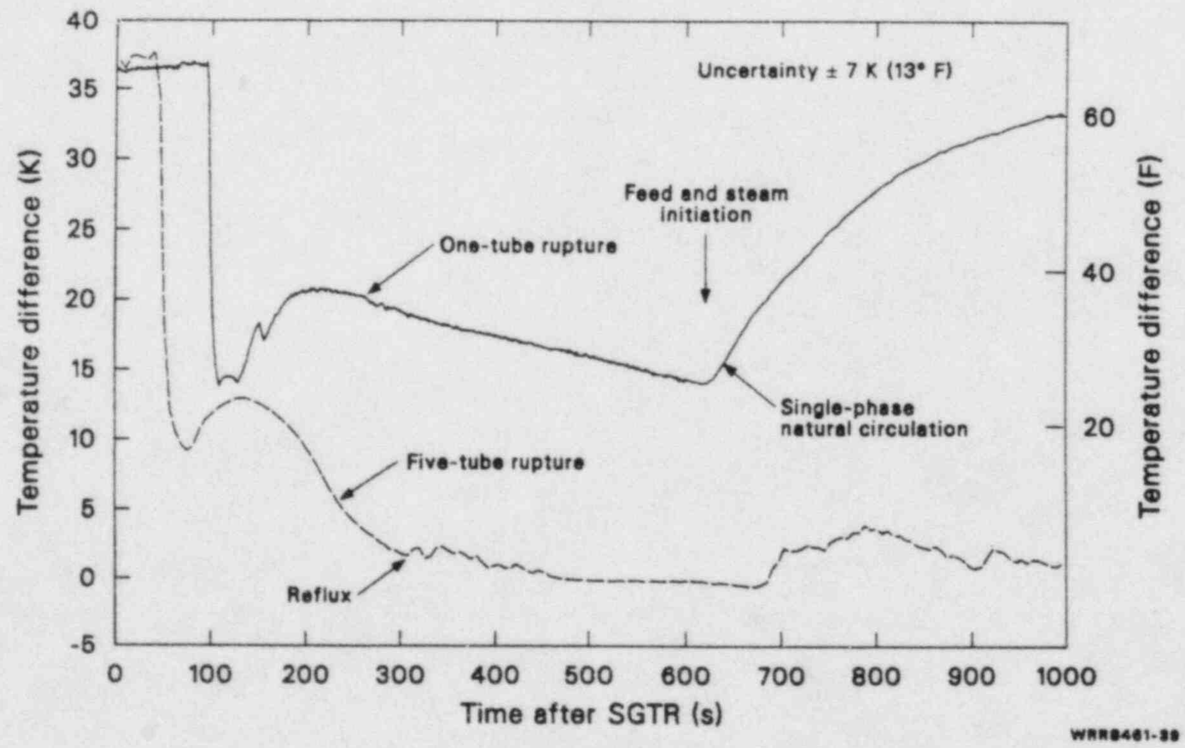


Figure 33. Primary fluid temperature differential across the unaffected loop steam generator for a one- and a five-tube rupture transient.

reflux) and about a 20 K (36°F) differential temperature for the one-tube case (indicating single-phase natural circulation).

For the single-tube case, the sudden increase in heat sink corresponding to feed and steam initiation caused an increase in primary-to-secondary-system single-phase natural circulation heat transfer by increasing the differential temperature across the primary tubes (as shown in Figure 34). This increase in primary-to-secondary-system heat transfer cooled the loop fluid, causing the increase in primary system depressurization shown in Figure 31. Figure 35 compares the cold leg fluid temperature with the saturation temperature for the single-tube case. The primary system loop fluid remained subcooled throughout the process, and the primary system fluid temperature decrease corresponding to feed and steam is prominent. This cooling of fluid caused an increase in fluid density (shrinkage), resulting in the slight depressurization rate increase observed on Figure 31.

For the five-tube case (reflux), the sudden increase in heat sink due to feed and steam initiation increased the condensation occurring in the primary system tubes. The mass rate of condensation is proportional to the differential temperature across the tube,[6] and the system pressure is proportional to the mass rate of condensation; therefore, any increase in differential temperature across the tube due to feed and steam of the secondary system increases the depressurization rate.

For the five-tube rupture case, the recovery scenario involved maintaining primary system pressure below the affected loop ADV setpoint by cycling the ADV. At t = 800 s, the ADV was closed and the heat sink diminished, causing an increase in primary system pressure (see Figure 31). At t = 900 s, the ADV was again opened, causing an increase in depressurization rate; therefore, feed and steam operation was effective in maintaining a primary system pressure band below the ADV setpoint by either increasing or decreasing condensation occurring in the primary system.

Primary System Mass Redistribution Due to Feed and Steam. The redistribution of primary system mass upon feed and steam initiation is also affected by the hydraulic condition in the loop. For the five-tube case (reflux), there is a pronounced mass redistribution upon feed and steam initiation; however, for the single-tube case (single-phase

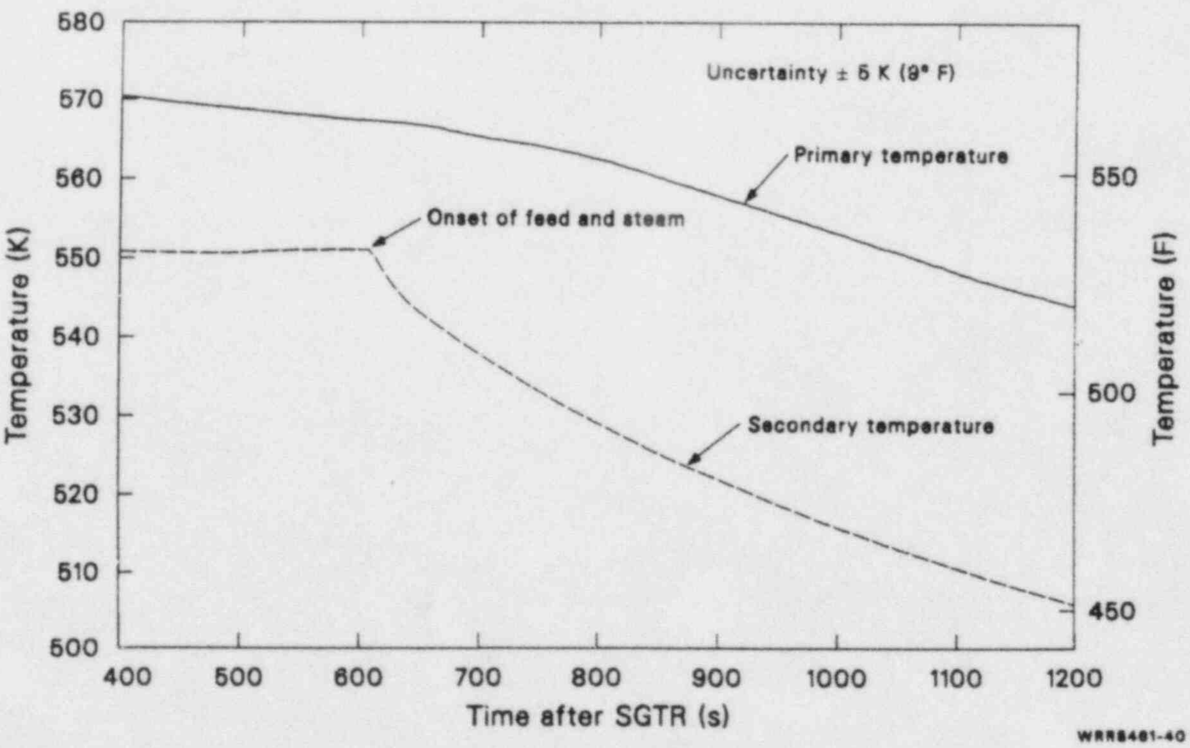


Figure 34. Unaffected loop primary and secondary fluid temperature during feed and steam for a one-tube rupture transient.

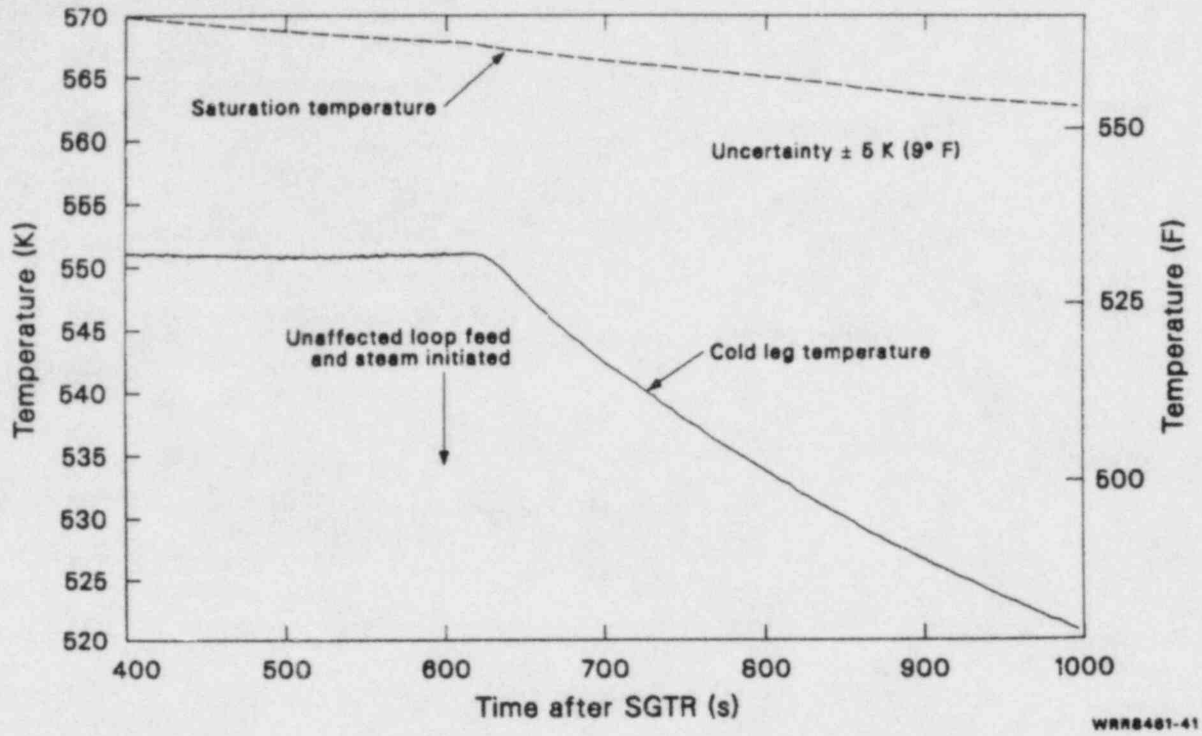


Figure 35. Unaffected loop cold leg fluid temperature during feed and steam for a one-tube rupture transient.

natural circulation), there was essentially no change in mass distribution. Figure 36 compares the hot collapsed vessel liquid level for the one-tube and five-tube cases. There is very little change in level for the one-tube case, but a significant depression in level for the five-tube case. Even though the vessel collapsed liquid level was reduced to below the top of the core, no core rod heatup occurred for the five-tube case. For the five-tube case, the depressurization in vessel level was caused by flashing of vessel fluid as the primary system pressure dropped. Figure 37, which presents the local axial density profile during the time period of feed and steam, shows that the decrease in vessel density was fairly uniform throughout the core as bubbles were formed by the flashing process.

For the five-tube case with a voided system, the local low pressure region created by the condensation process in the unaffected loop steam generator tubes caused a filling of the tubes and a redistribution of mass in other parts of the primary system loop. Figure 38 shows the primary side liquid level in a medium tube[a] of the unaffected loop generator, showing a filling trend immediately upon initiation of feed and steam. The source of the liquid level increase in the steam generator tubes came from condensed core steam and entrained liquid from other parts of the system. The core steam generation rate was about 0.041 kg/s (0.09 lbm/s), which translates to a tube fill rate of 5.4×10^{-5} m^3/s (0.0019 ft^3/s). The actual tube fill rate shown on Figure 38 was about 2.5×10^{-5} m^3/s (0.00088 ft^3/s). Extrapolating to six tubes gives a fill rate of 1.5×10^{-4} m^3/s (0.0052 ft^3/s), which is higher than the core steam rate can account for; therefore, the remainder is assumed to be entrained liquid from other parts of the loop.

Figure 39 presents the liquid level in the unaffected loop pump suction, affected loop pump suction, and affected loop steam generator primary tubes. All show an emptying trend upon unaffected loop feed and steam; therefore, the loop pump suctions and affected loop steam generator tubes were a source for some of the filling of the unaffected loop primary system tubes. The condensation process associated with the feed and steam operation caused a differential pressure between the primary tubes and the rest of the loop. This differential pressure caused a fluid flow that entrained liquid towards the condensation site in the tubes. Even

a. The unaffected loop steam generator contains six primary tubes—two long, two short, and two medium length tubes. The affected loop generator contains two long tubes.

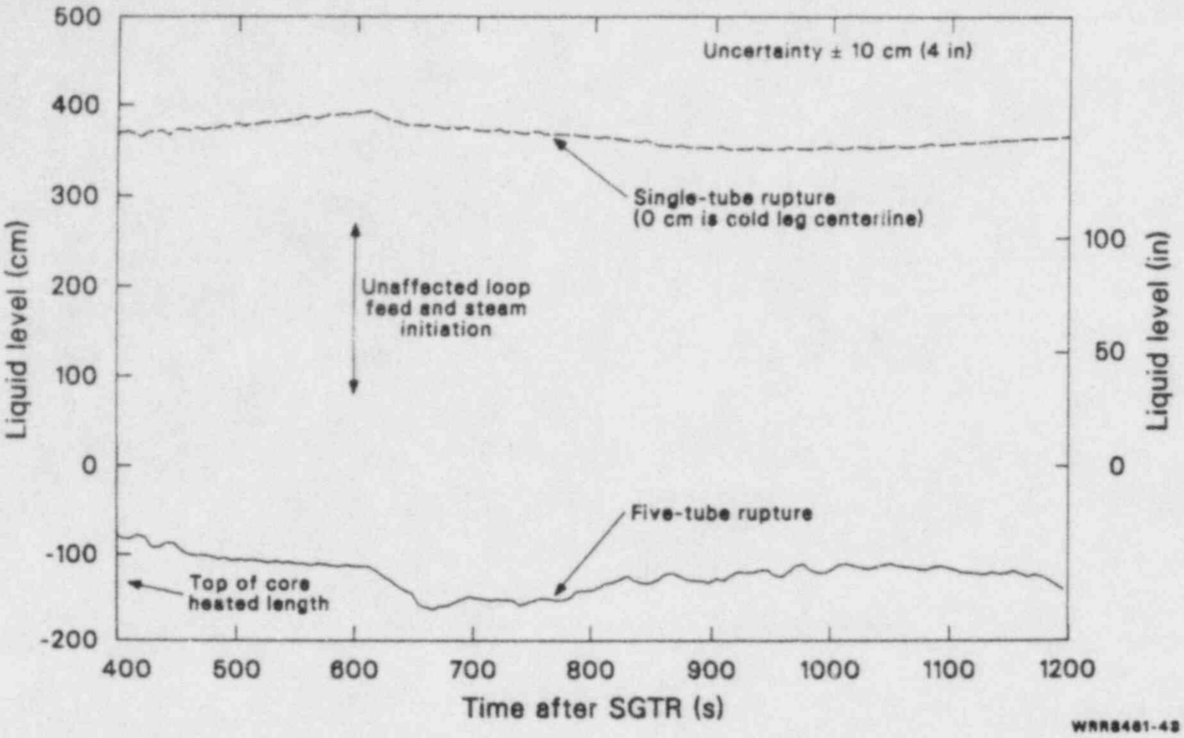


Figure 36. Vessel collapsed liquid level during feed and steam for a one- and a five-tube rupture transient.

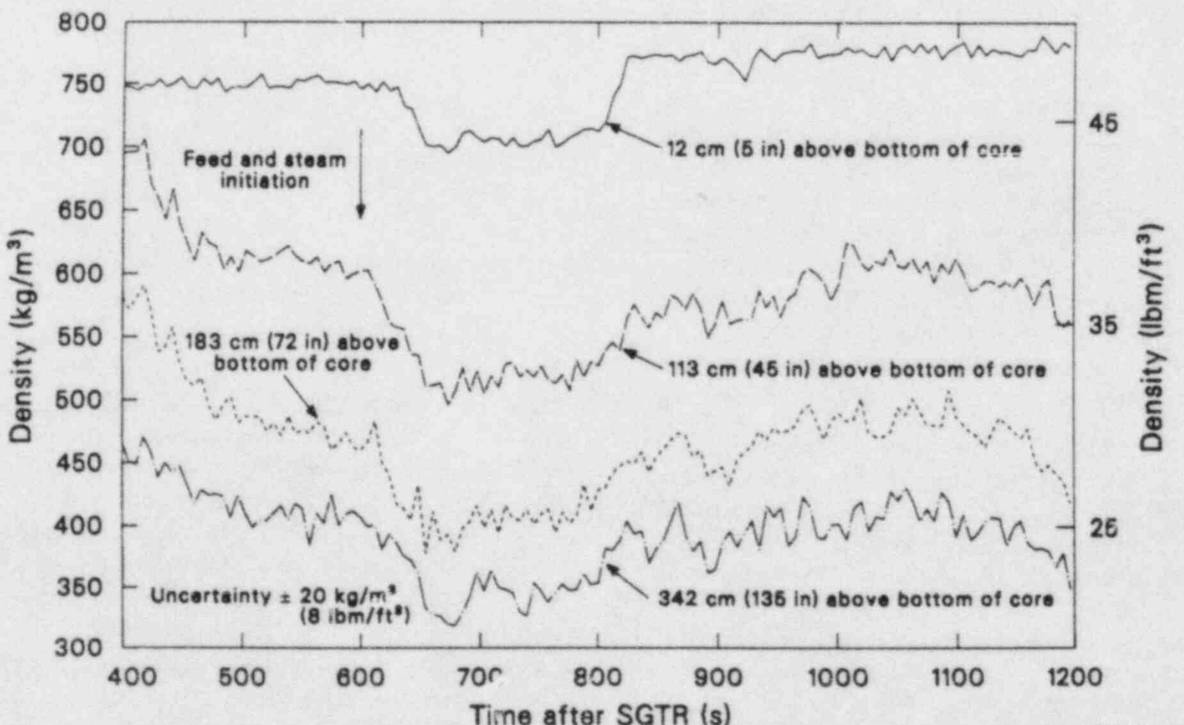


Figure 37. Axial variation in core fluid density during feed and steam for a five-tube rupture transient.

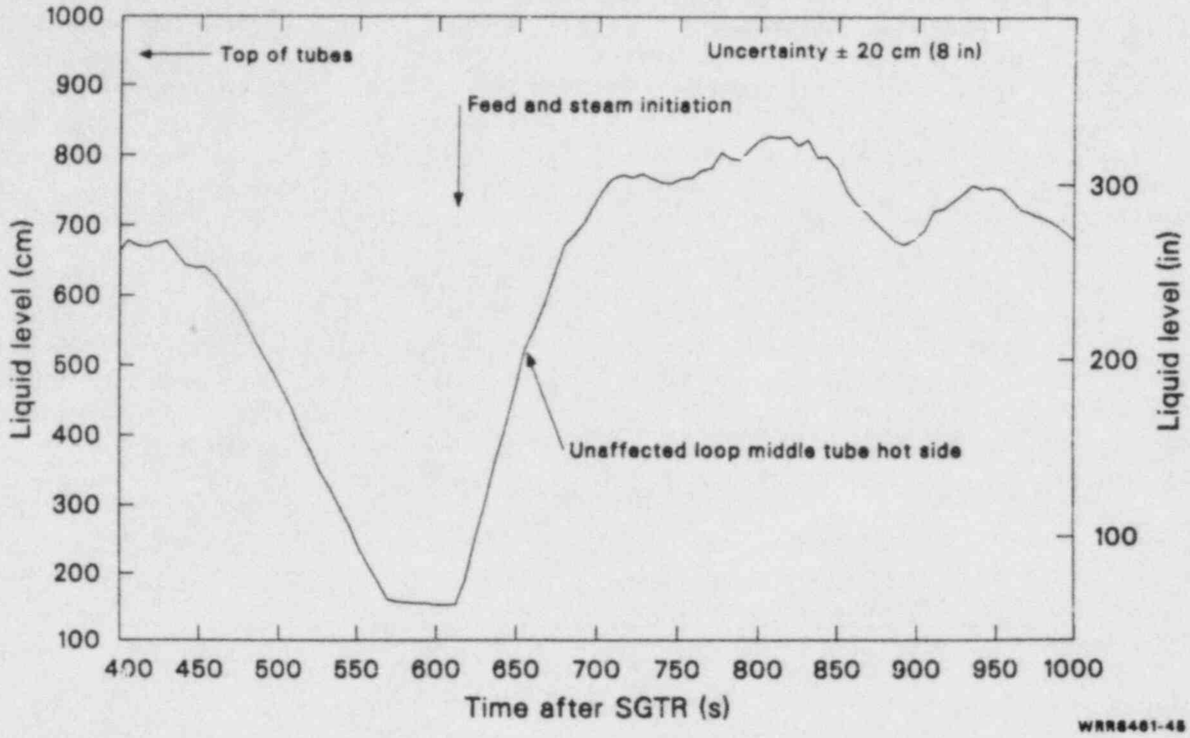


Figure 38. Unaffected loop primary tube collapsed liquid level during feed and steam for a five-tube rupture transient.

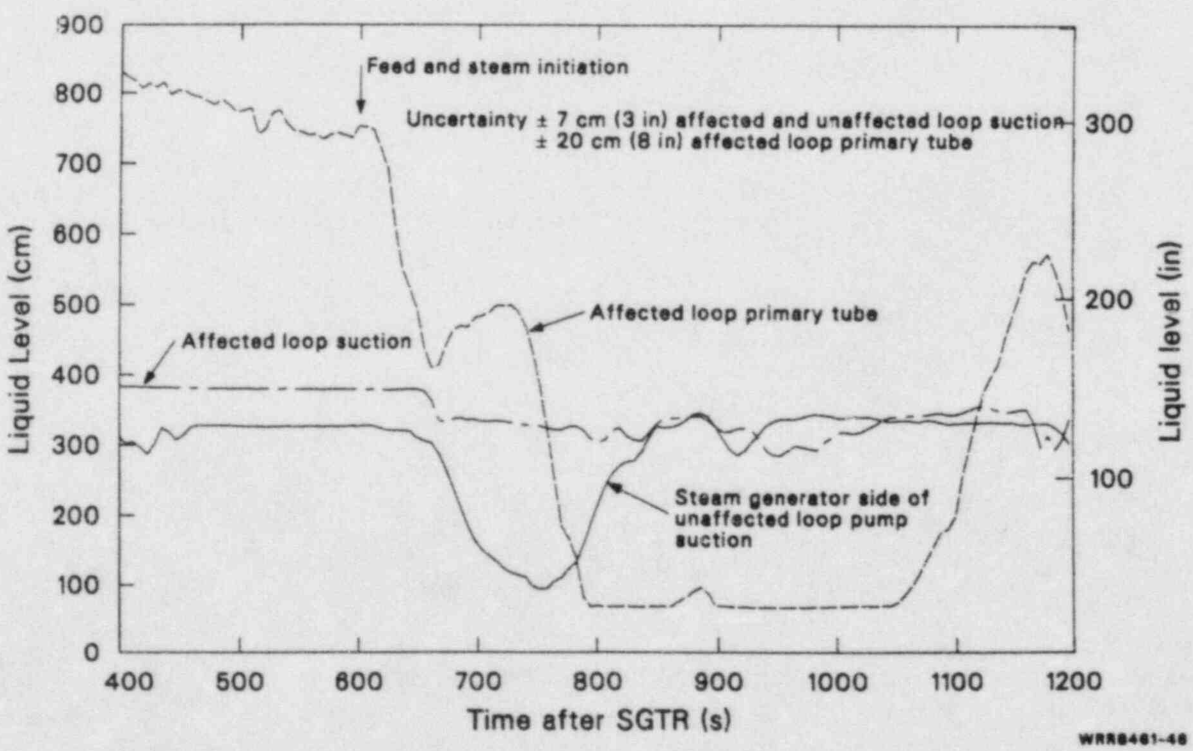


Figure 39. Collapsed liquid level for the unaffected loop pump suction, an affected loop primary steam generator tube, and the affected loop pump suction during feed and steam for a five-tube rupture transient.

though the medium primary tube became nearly full during the feed and steam operation (Figure 38), condensation continued as evidenced by a fairly uniform depressurization rate (Figure 31). Figure 38 shows the liquid level of only one of the six tubes in the unaffected loop steam generator; therefore, the tube liquid level depicted is not necessarily representative of all six steam generator tubes. The steam generator side of the unaffected loop pump suction showed a draining upon feed and steam initiation at $t = 600$ s; however, at about $t = 750$ s the suction filled again. This was probably due to a steam generator tube filling with condensation from the hot side and spilling over into the cold side of the tube (or actual flooding due to liquid entrainment) and then flowing down with gravity to the steam generator side of the pump suction. The rate of fill of the down flow side of the suction corresponds within 10% to the core steam mass generation rate, suggesting complete upflow or downflow side condensation. Sufficient data are not available to determine the exact mechanism. The original decrease in level in the steam generator side of the pump suction occurring at feed and steam initiation (Figure 39 at $t = 600$ s) is attributed to a combination of entrained liquid (as flashed steam rushed to the condensation site in the cold side of the tubes) and liquid that is pulled through the system toward the hot side of the tubes by condensation in the hot side of the tubes.

The Effectiveness of PORV Operation for Controlling Primary System Pressure (Primary Feed and Bleed). An operator has the option of opening the PORV to reduce or control primary system pressure[4] during a tube rupture transient. In a PWR, PORV operations are performed along with safety injection, resulting in primary system pressure control without significant mass reduction (primary feed and bleed). In actual plant experience at the GINNA nuclear power generating facility,[7] the PORV was cycled during a tube rupture transient, resulting in a significant primary system pressure reduction (as shown on Figure 40). The overall system data from the the GINNA accident is limited; therefore, PORV operations were performed during tube rupture simulations in the well-instrumented Semiscale facility to examine overall system response.

During a single-tube rupture transient in Semiscale, PORV operation (latched open) in conjunction with SI was effective in reducing primary system pressure below ADV setpoints without core uncovery. Even though the core was not uncovered during the PORV operation, there was a significant

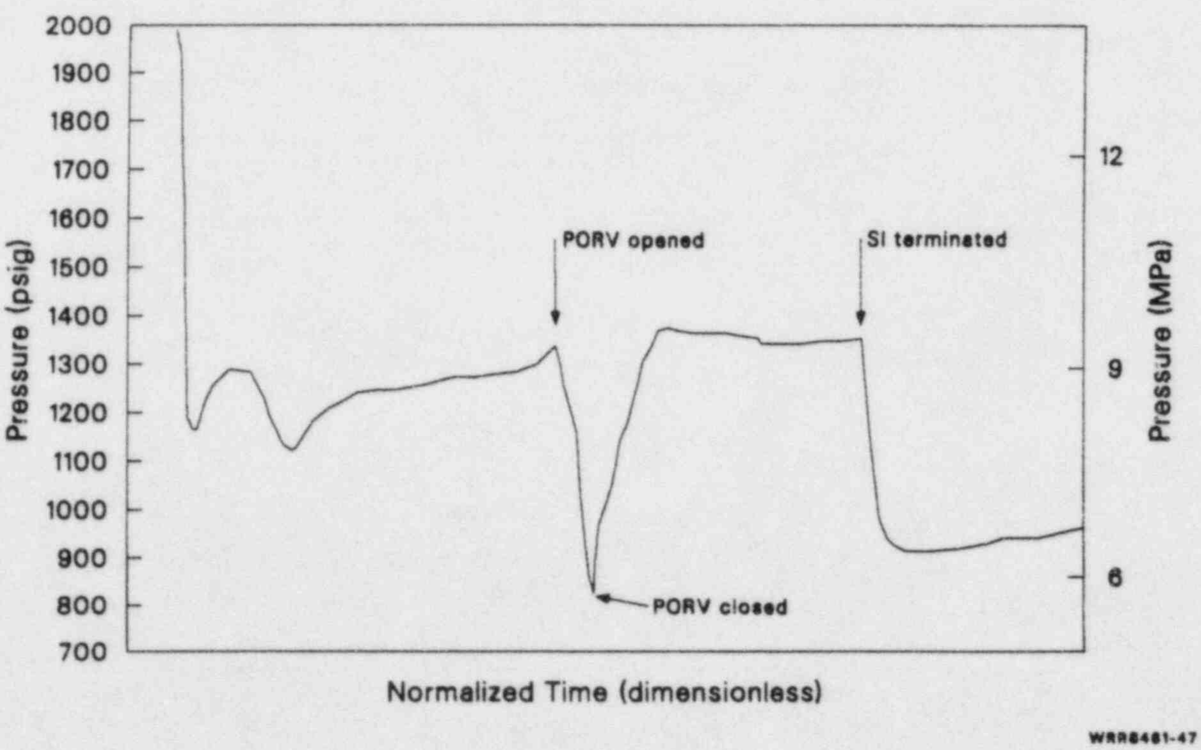


Figure 40. Primary system pressure response during PORV operation for the GINNA steam generator tube rupture transient.

system mass inventory redistribution and net overall system mass inventory reduction. Figure 41 presents an overlay of the primary system pressure with the pressurizer collapsed liquid level during a PORV operation. At t = 600 s, the pressurizer was full of steam, and opening the PORV caused a rapid primary system depressurization (similar to the GINNA response) as steam rushed out of the primary system through the PORV. The local low pressure created by the steam leaving the pressurizer via the PORV caused a flow of primary system fluid toward the pressurizer and eventual filling. Figure 42 shows that the differential pressure between the pressurizer and vessel upper head increased when the PORV opened, which was the driving potential for the liquid mass transport from the primary system into the pressurizer. Following the PORV opening, most of the liquid entering the pressurizer came from the vessel. Figure 43 compares the pressurizer collapsed liquid level and the vessel upper head level. Approximately 0.0148 m^3 (0.52 ft^3) of collapsed fluid left the vessel upper head between 600 and 800 s after the transient initiation and the increase in pressurizer collapsed level during this period corresponded to approximately 0.020 m^3 (0.706 ft^3). The remaining mass that filled the pressurizer came from loop piping and SI flow. As shown in Figure 44, the vessel level depleted to just above the core heated length as the pressurizer continued to fill beyond 800 s. The volume associated with the pressurizer level increase between 800 and 1100 s was about 0.014 m^3 (0.494 ft^3); since the vessel level depletion only accounted for about 0.0046 m^3 (0.162 ft^3), again, fluid in the loop piping and SI flow account for the deficit in pressurizer fill.

The PORV flow rate out of the system depended on the pressurizer level. The PORV flow varied from single-phase steam to a two-phase steam/water mix to single-phase water. Figure 45 compares the PORV flow rate to the pressurizer liquid level. As long as there was a steam space, the measured mass flow rate out of the PORV corresponded to that expected from single-phase steam calculations[8] [0.0116 kg/s (0.0255 lbm/s)].[a] As the liquid level in the pressurizer increased (1100 to 1600 s), the mass flow out of the PORV changed from single-phase steam to a two-phase mixture. Between 1600 and 2100 s, the PORV flow corresponded to the single-phase liquid calculation [~0.057 kg/s (0.125 lbm/s)]. When the PORV mass flow increased due to the change from single-phase steam to a two-phase mixture, the overall system mass inventory decreased significantly. Figure 46 compares the combined tube

a. The mass flow rate was calculated based on $\dot{m} = YAC\sqrt{2\Delta P\varrho}$ where Y = expansion factor; C = discharge coefficient; A = orifice area; ΔP = critical pressure drop; ϱ = density of steam.

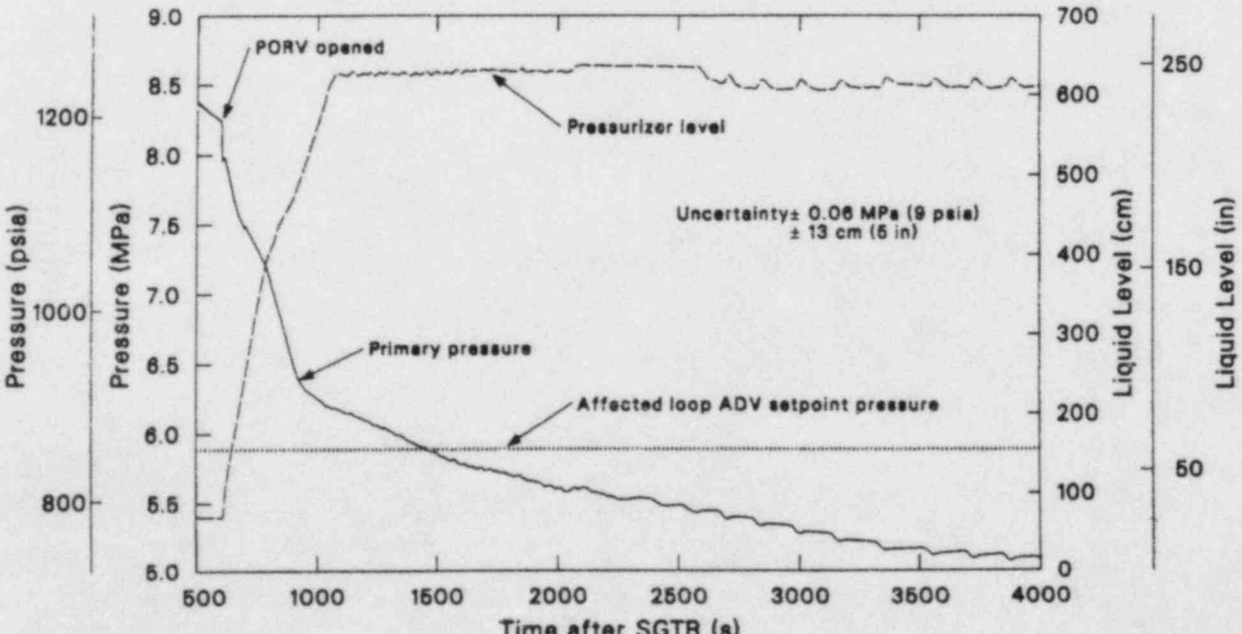


Figure 41. Primary system pressure and pressurizer collapsed liquid level during PORV operation for a single-tube rupture transient.

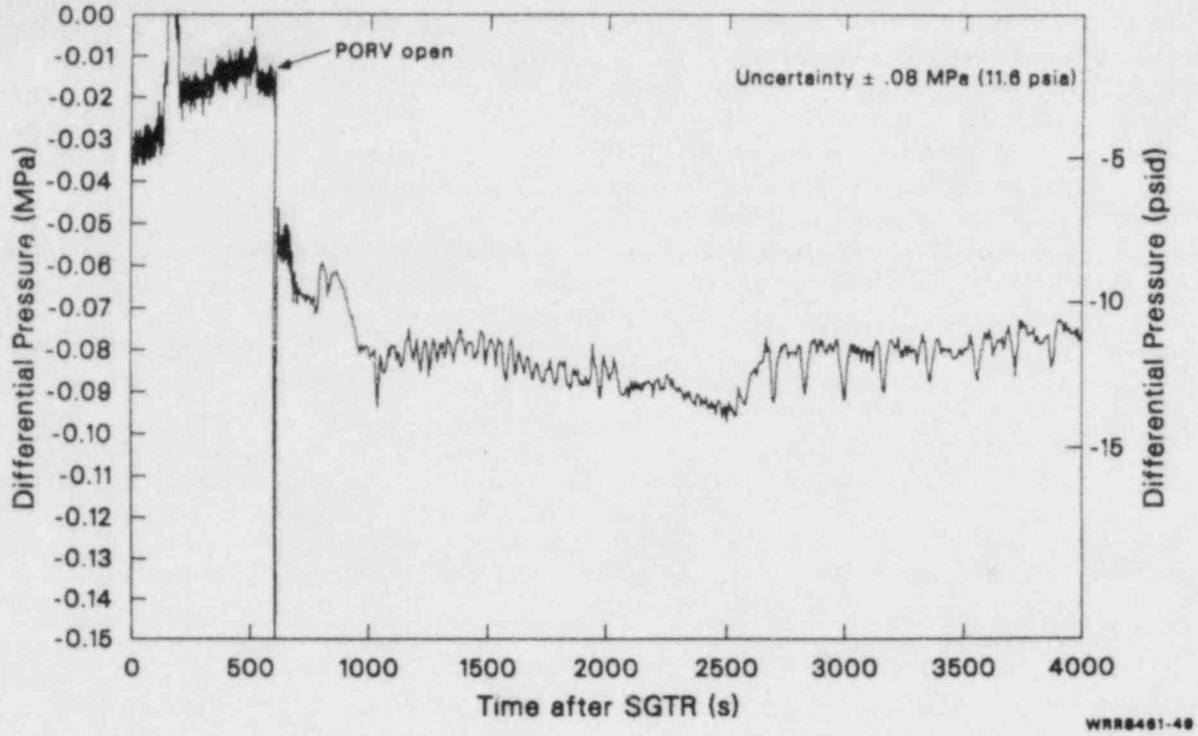


Figure 42. Differential pressure between the pressurizer and the vessel upper plenum during PORV operation for a single-tube rupture transient.

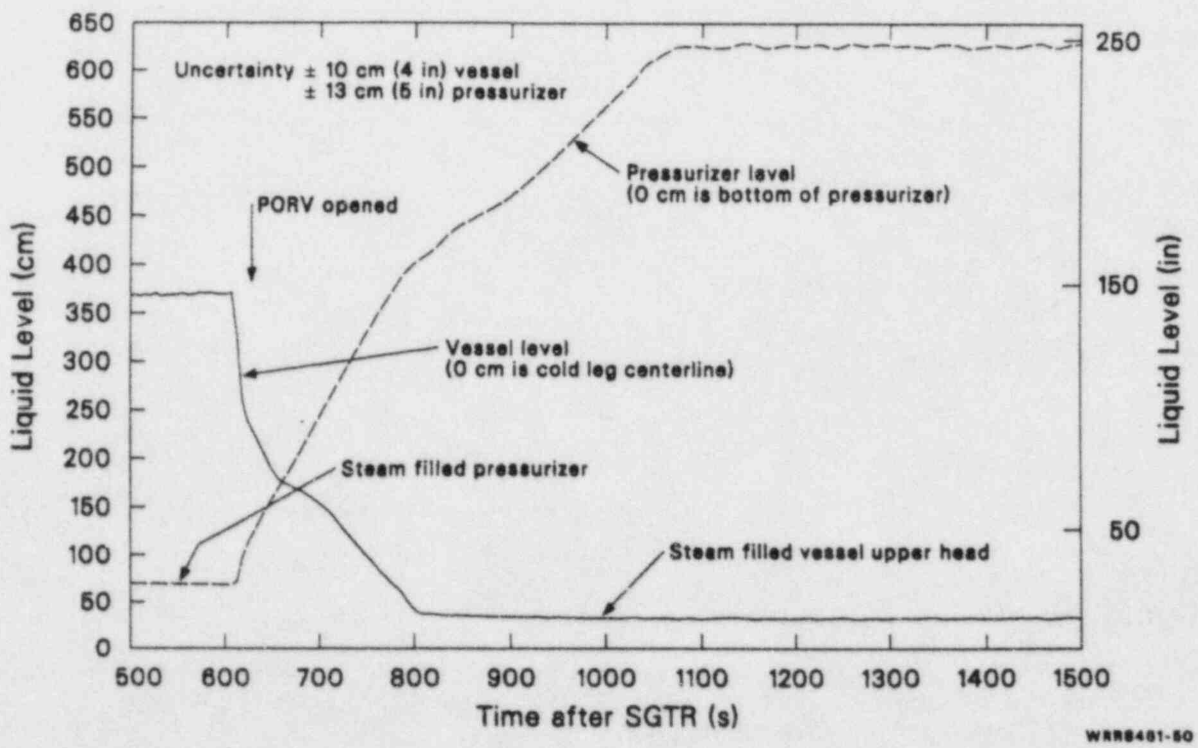


Figure 43. Collapsed liquid level in the pressurizer and vessel upper head during PORV operation for a one-tube rupture transient.

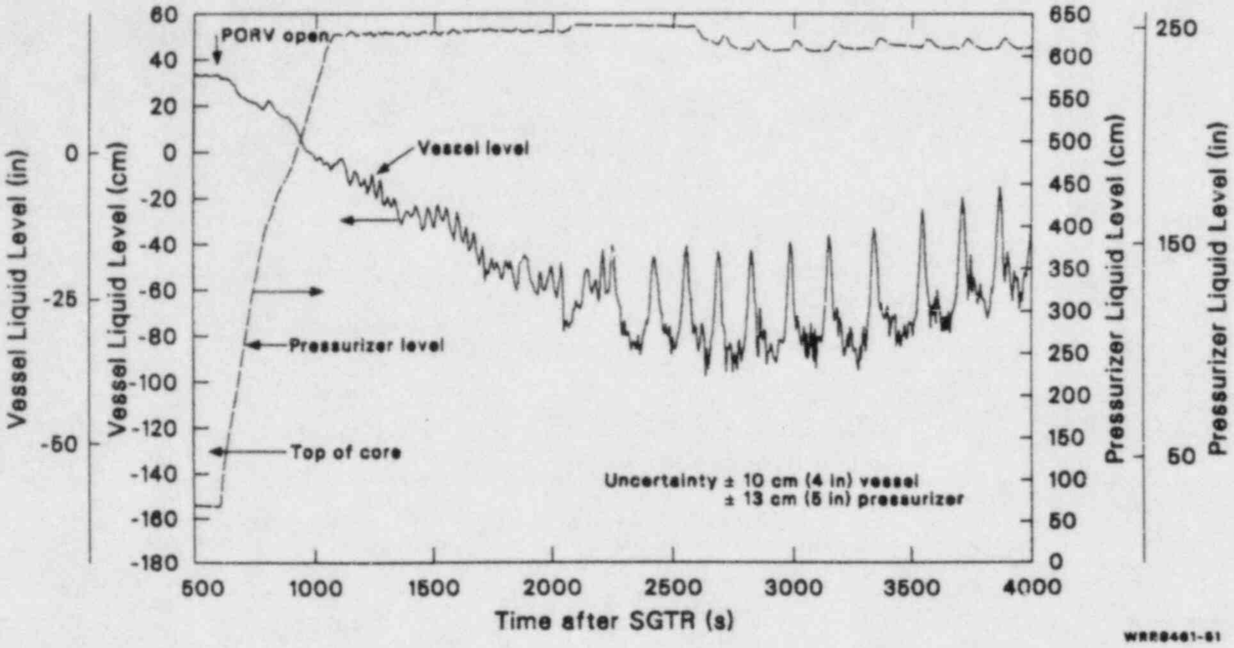


Figure 44. Collapsed liquid level in the pressurizer and lower vessel during PORV operation for a one-tube rupture transient.

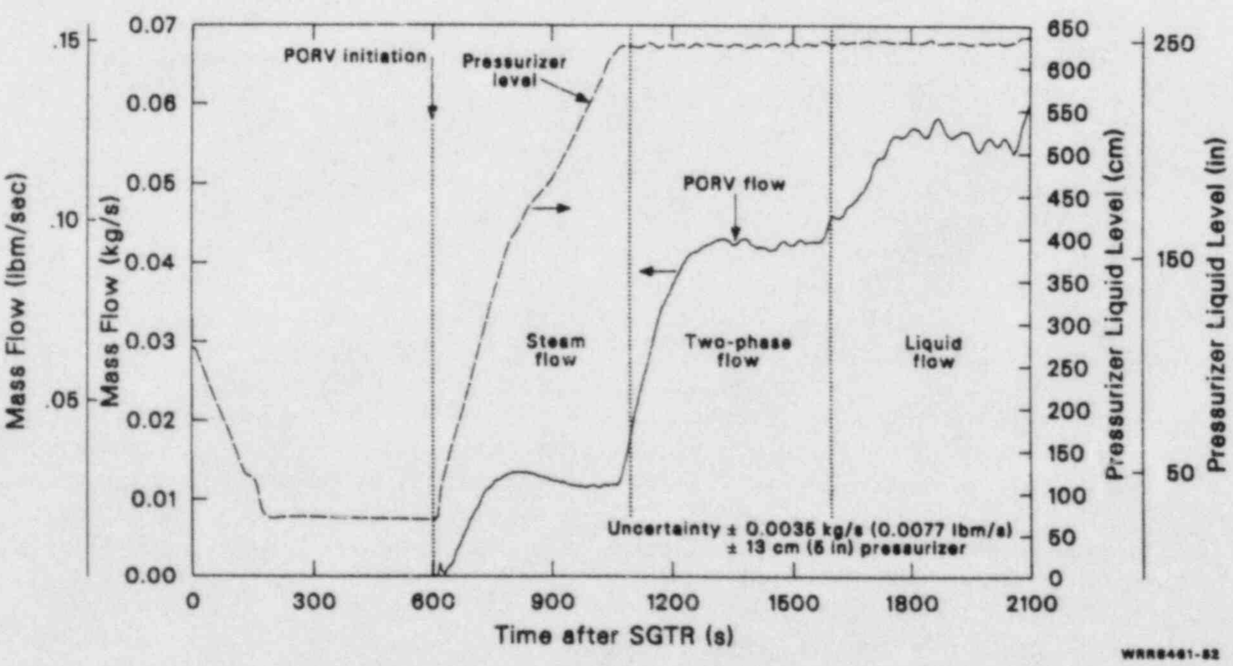


Figure 45. PORV flow and collapsed liquid level in the pressurizer during PORV operation for a one-tube rupture transient.

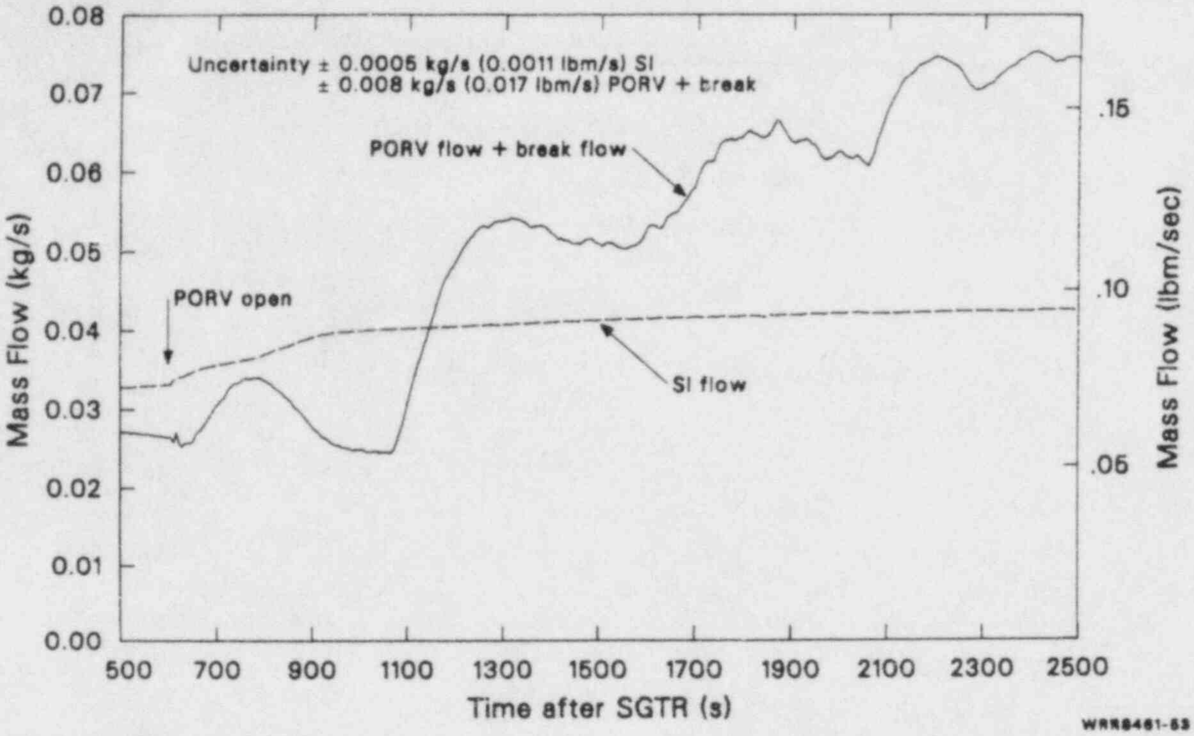


Figure 46. Combined break flow and PORV flow with SI flow during PORV operation for a one-tube rupture transient.

rupture break mass flow added to the PORV mass flow (outflow) and the SI flow (inflow). Shortly after 1100 s, the combined outflow exceeds the combined inflow, resulting in the decrease in loop inventory shown in Figure 47.

Once the pressurizer filled at about 1100 s, the open PORV had only a small effect on primary system pressure (Figure 41). The vessel upper head remained voided of liquid, and the pressurizer remained full of liquid. With two-phase flow and single-phase liquid flow out the PORV, there was only a small reduction in primary system volume compared to a large primary fluid volume reduction when the PORV flow was pure steam; therefore, there was a much slower depressurization once the pressurizer filled. Even though the depressurization rate decreased, the combined primary feed and bleed using PORV and SI was able to bring the primary system pressure below the affected loop ADV setpoint without uncovering the core.

The Effectiveness of Pressurizer Auxiliary Spray for Controlling Primary System Pressure. If normal pressurizer spray is unavailable (main coolant pumps are off) or the PORV cannot be used during a tube rupture, pressurizer auxiliary spray can be used to reduce primary system pressure.[4] This section discusses the effectiveness of auxiliary spray to reduce primary system pressure and the effect of spray initiation on system mass distribution.

In the Semiscale experiments, auxiliary pressurizer spray was effective in reducing primary system pressure only if the pressurizer liquid level was below the top of the pressurizer. Experiments involving auxiliary spray were performed with different liquid levels in the pressurizer. One such experiment allowed the pressurizer to fill completely. For this case, a comparison of pressure response and pressurizer liquid level shows that primary system depressurization is dependent on pressurizer liquid level. Figure 48 shows that as the pressurizer collapsed liquid level increased (steam space decreased) the depressurization rate due to spray decreased. A spray of fluid [about 300 K (80°F)][a] entered the pressurizer from the top, resulting in dropwise condensation of pressurizer steam. As the steam space diminished, more liquid entered the pressurizer pool without condensing steam. The above experiment was a one-tube rupture with a nearly full vessel upper head and

a. Although normal PWR pressurizer auxiliary spray temperature varies between 422 and 533 K (300 to 500°F), the Semiscale spray rate and temperature were chosen to give a desired primary system depressurization rate [0.0068 MPa/s (1 psia/s)]. This was determined by a separate effects test.

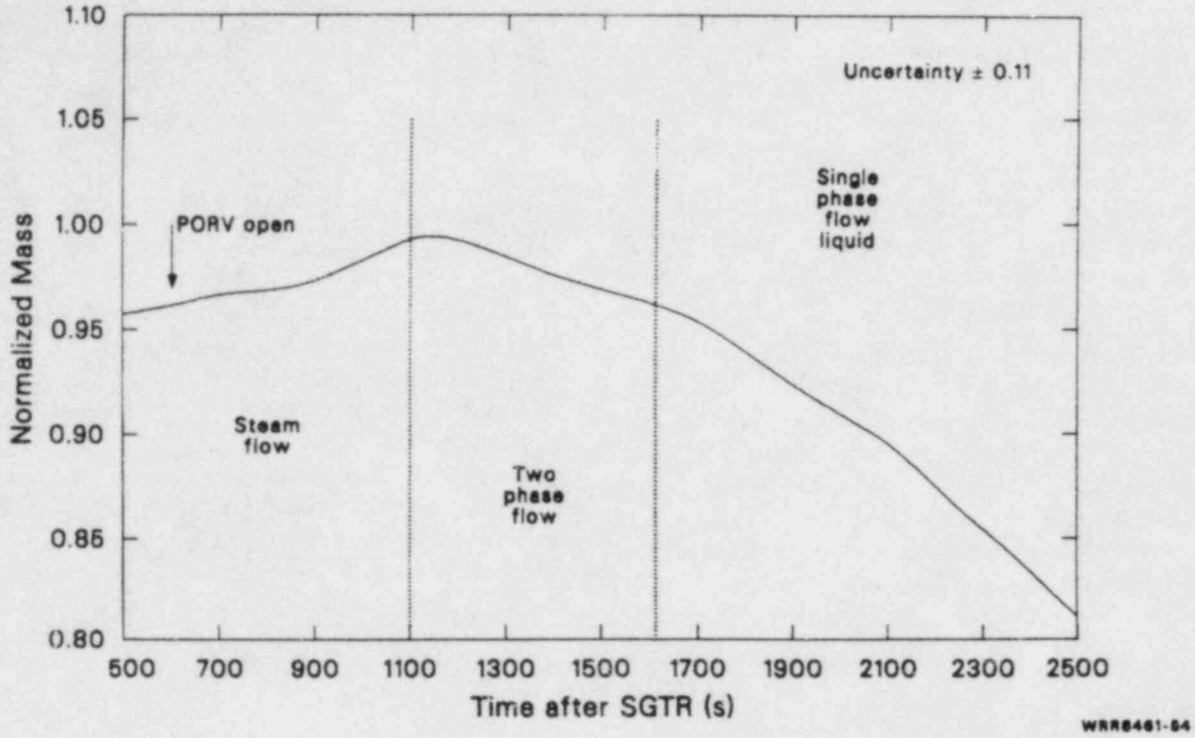


Figure 47. System mass inventory during PORV operation for a one-tube rupture transient.

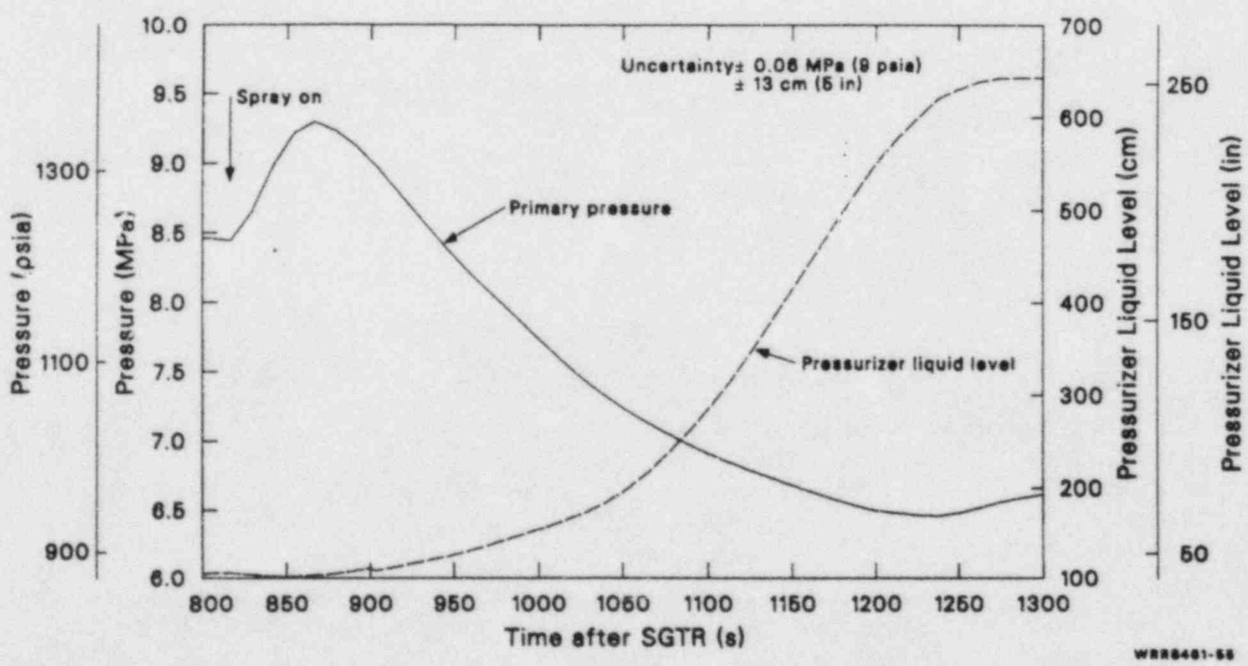


Figure 48. Primary system pressure and pressurizer collapsed liquid level during pressurizer auxiliary spray for a one-tube rupture transient.

included unaffected loop feed and steam. The feed and steam operation had only a minor effect on primary system depressurization for the one-tube case, and most of the depressurization was due to condensation from the spray operation. A more complete discussion of the phenomena involved during spray follows for a case where the collapsed liquid level was only allowed to fill about 50% of the pressurizer and unaffected loop secondary feed and steam were not used.

Introduction of cold [300 K (80°F)] pressurizer spray initially caused a primary system pressurization which was attributed to a change from superheated fluid and wall conditions to saturated conditions. Figure 49 compares the primary pressure response and pressurizer level and shows a slight pressurization upon introduction of cold spray. This pressurization is attributed to the superheated steam and superheated walls changing the subcooled spray to saturation, then evaporating the liquid.[a] The net result is steam production that has a pressurization effect in the primary system. Figure 50 shows the axial fluid temperature gradient in the pressurizer; note the change from superheated to saturated fluid condition (top to bottom). By 660 s, all of the fluid in the pressurizer is saturated, at which time the primary system pressure starts decreasing (Figure 49). The decrease in pressure starting at 660 s is attributed to condensation of saturated steam in the pressurizer. Figures 49 and 50 confirm that the pressurizer was essentially filled with saturated steam at 660 s; so the pressure decrease is due to the dropwise condensation of saturated steam tempered by the evaporation of liquid as spray comes in contact with the superheated pressurizer walls. Figure 51 shows the axial pressurizer wall temperature gradient (on the OD of the pressurizer). The superheated walls were not entirely quenched (to saturated conditions) until about 900 s. The quench pattern followed a top down quench.

A significant portion of the pressurizer liquid level increase shown on Figure 48 is attributed to pressurizer spray. Figure 52 compares the integrated pressurizer fill mass flow rate to the integrated auxiliary spray mass flow rate and shows that the spray contributed about half of the total pressurizer fluid mass during the initial spray operation. The other half of the mass came from other parts of the

a. The increase in primary pressure corresponding to spray initiation was more enhanced for the single-tube rupture case shown on Figure 49 [about 0.9 MPa (131 psia) rise for the single-tube rupture case and 0.1 MPa (14.5 psia) rise for the ten-tube rupture case]. This difference is related to the amount of voids in the primary system at the onset of spray. With more voids in the system for the ten-tube rupture case, evaporation of the liquid (steam generation) caused compression of a larger steam space than for the single-tube rupture case. Compression of a larger steam space causes a smaller pressurization, because primary pressure is approximately inversely proportional to volume (perfect gas).

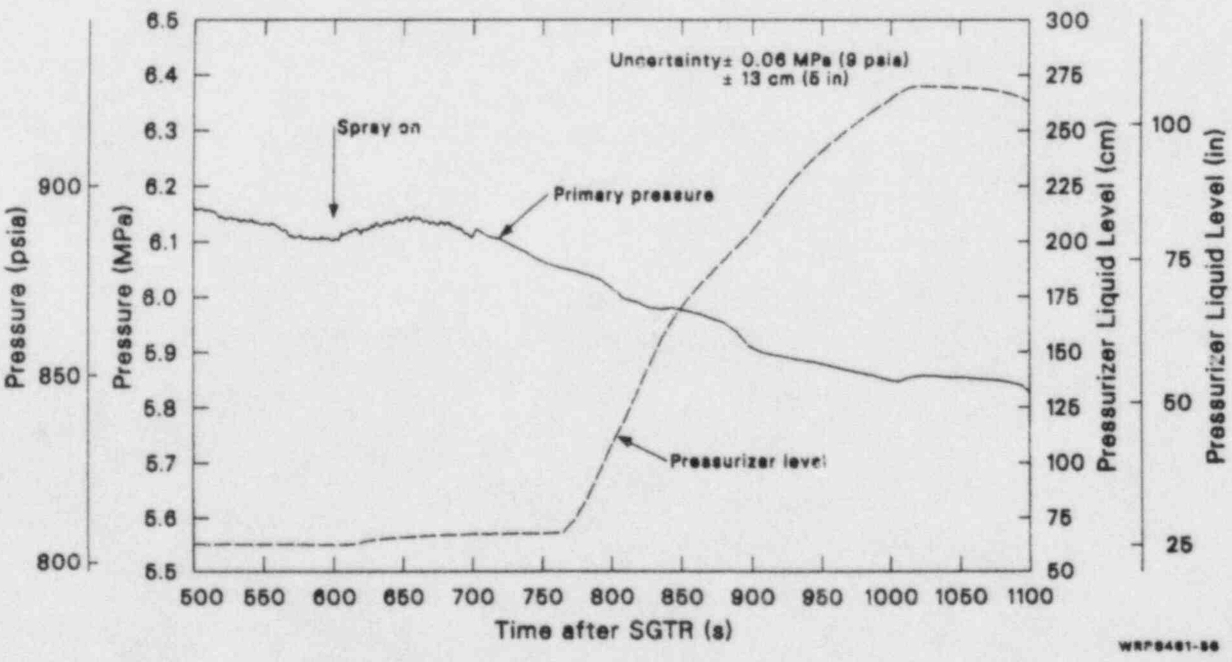


Figure 49. Primary system pressure and pressurizer collapsed liquid level during pressurizer auxiliary spray for a ten-tube rupture transient.

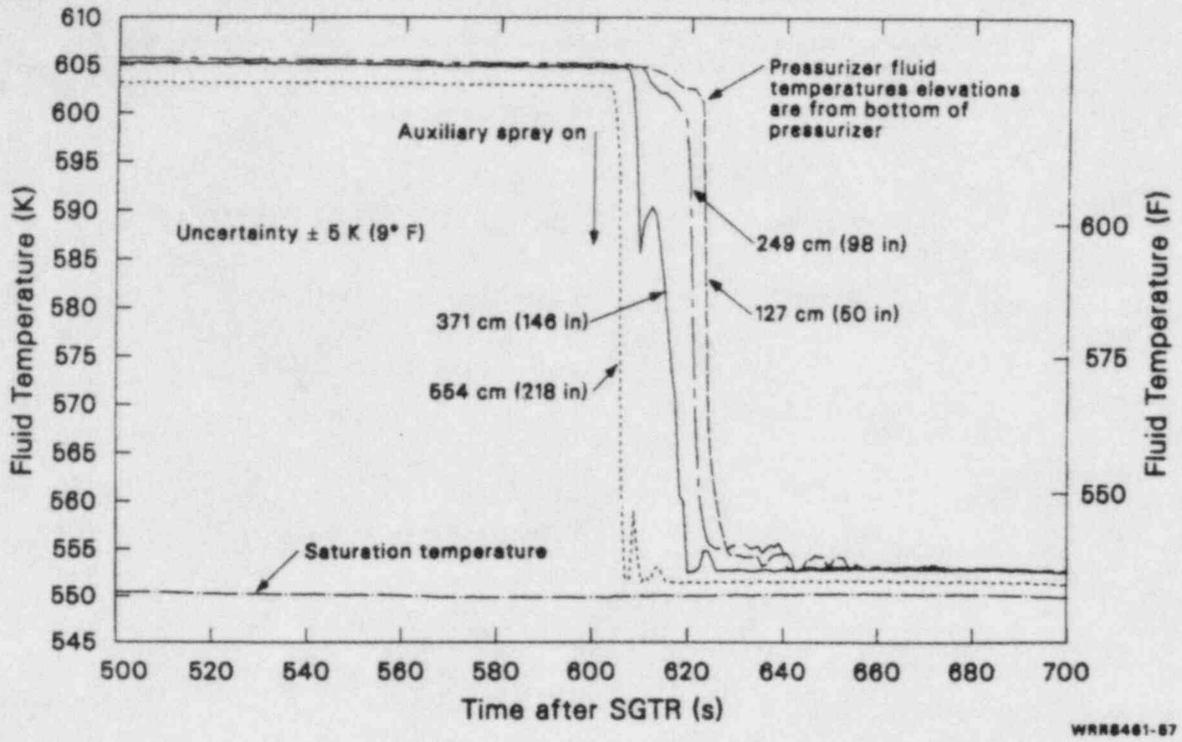


Figure 50. Axial variation in pressurizer fluid temperature with saturation temperature during pressurizer auxiliary spray for a ten-tube rupture transient.

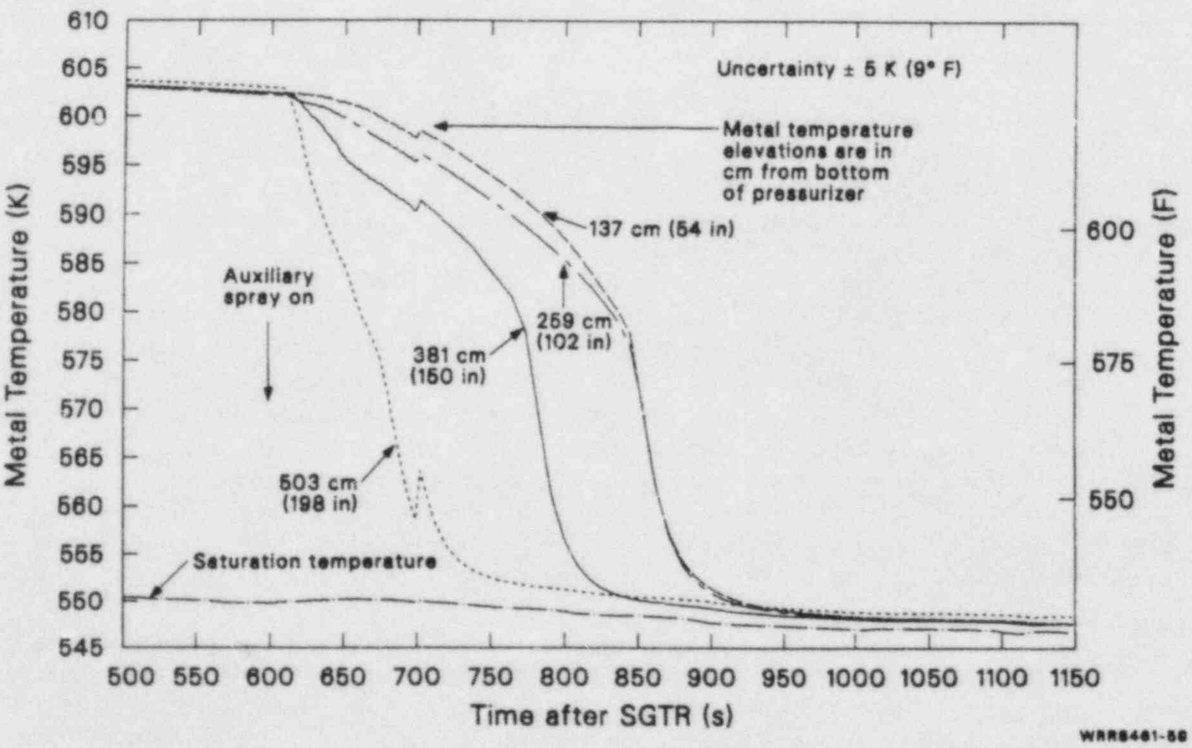


Figure 51. Axial variation in pressurizer metal temperature with saturation temperature during pressurizer auxiliary spray for a ten-tube rupture transient.

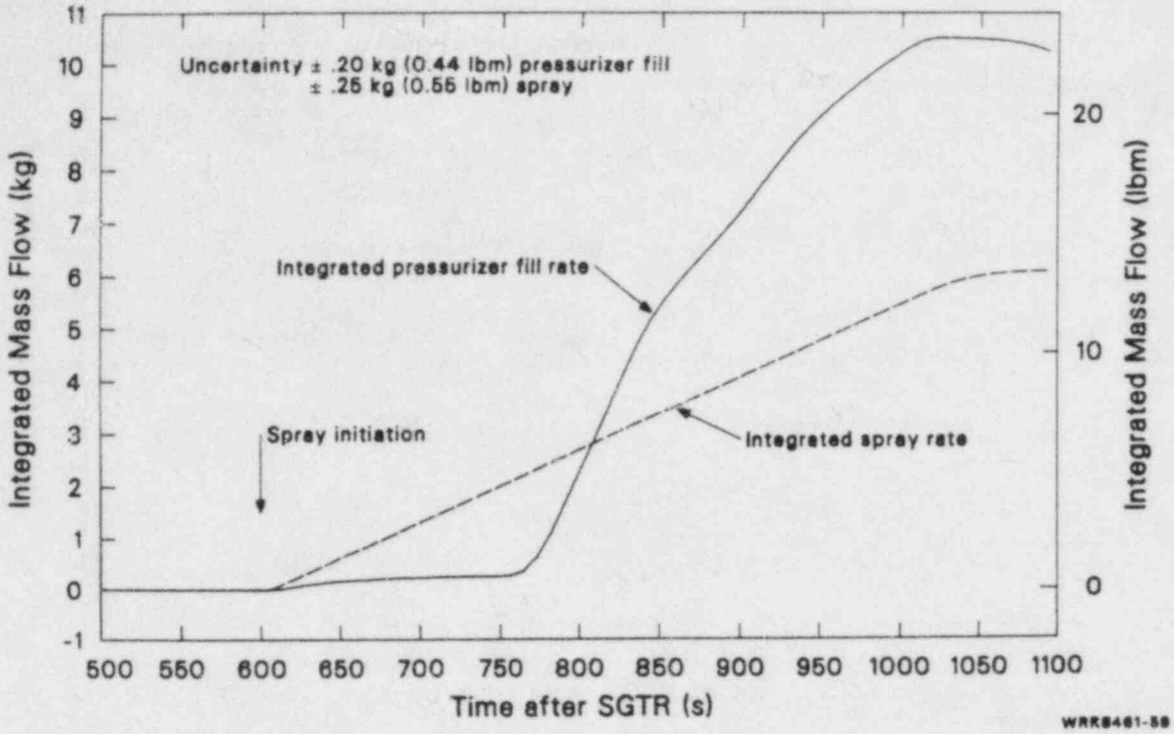


Figure 52. Integrated mass flow rate for the pressurizer and pressurizer auxiliary spray during a ten-tube rupture transient.

system as the condensation process created a low-pressure region in the pressurizer. The rapid increase in mass in the pressurizer at about 760 s corresponds approximately to the rapid decrease in liquid level in the unaffected loop pump suction on the steam generator side (Figure 53). There was essentially no change in vessel level during this spray period, as the collapsed level remained near the top of the core. In addition, liquid levels in other parts of the loop remained about the same during the spray period. The most probable mechanism for the mass transport from the steam generator side of the pump suction is flashing of liquid and the resulting mass redistribution. The flashing caused a net mass transport out of the steam generator side of the pump suction. Flashed steam with some entrained water rose toward the steam generator and to the condensation site in the pressurizer. Some of the water went to replenish liquid that had flashed in the pump side of the suction, since the pump side remained nearly full of liquid throughout the spray period.

The preceding discussion was for a ten-tube rupture when the system was in a considerably voided state at the start of spray. (Steam generator tubes, hot leg, pressurizer, and vessel upper head were all voided.) During another experiment involving a one-tube rupture and spray, there was a depression of the vessel upper head level due to the fluid flowing to the condensation site in the pressurizer; however, the vessel collapsed level remained above the cold leg during the spray operation.

The Effectiveness of Pressurizer Internal Heaters for Controlling Primary System Pressure. During PWR operations, pressurizer internal heaters are used to pressurize the primary system and thus subcool the loop fluid. This is accomplished by creating a steam bubble in the top of the pressurizer by boiling pressurizer liquid with the heaters. This steam bubble pushes against system liquid, thus pressurizing the system. During a tube rupture transient, depending on pressurizer liquid level, the internal heaters can be used to increase subcooling in the loop. This section examines the effectiveness of using pressurizer internal heaters to pressurize the system during a tube rupture.

The capability of the pressurizer heaters to increase primary system pressure and thus increase loop subcooling is dependent upon the relationship of break flow, SI flow, and the amount of steam generation in the pressurizer due to heater operation. When the primary system is intact (no break),

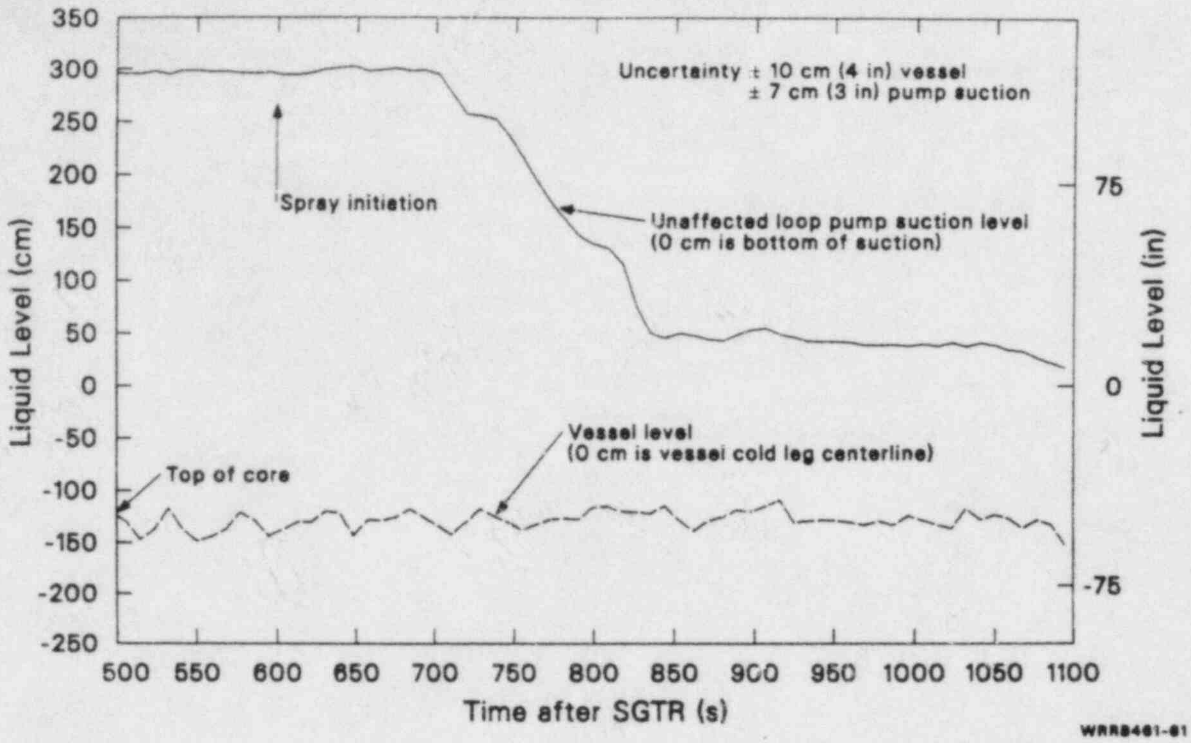


Figure 53. Collapsed liquid level in the unaffected loop pump suction and vessel during pressurizer auxiliary spray for a ten-tube rupture transient.

expansion of the steam bubble automatically results in a primary system pressure increase due to steam pushing against a solid system. With a break in the system, however, creation of steam by the pressurizer heaters has the tendency to push liquid out the break. Whether or not the steam generation in the pressurizer causes primary system pressurization is dependent on the relationship of break flow, SI flow, and system voids.

With the approximately correct heater capacity[a] in Semiscale, pressurizer internal heater operation was unable to cause pressurization of the primary system during a single-tube rupture. Figure 54 compares primary pressure and internal heater power during recovery from a single-tube rupture transient. There is no net increase in primary pressure throughout the time period of internal heater operation. Operational procedures caused the cycling in pressure as SI was alternatively turned on and off, depending on the pressurizer level shown on Figure 55. When SI was on, the steam space in the top of the pressurizer was compressed due to the SI flow and the pressure increased; conversely, when SI was terminated, the pressure decreased. The reason the pressure did not increase with the application of internal heater power during this period was because steam created in the pressurizer by internal heater power pushed fluid out the break. Figure 56 compares break flow and flow out of the surge line. When SI was terminated, fluid drained out of the pressurizer as break flow remained high. Expansion of the steam bubble in the pressurizer from boiling simply caused an outflow of fluid toward the break with no net compression of primary fluid. If the primary system and affected loop secondary system were completely coupled hydraulically (i.e., the affected loop secondary was full with no break flow), it is conceivable that pressurizer internal heater power could pressurize the primary system without SI.

In summary, in the presence of a tube rupture, pressurizer internal heaters were ineffective for increasing the Semiscale primary system pressure.

a. In the Semiscale experiments, the pressurizer total heater capacity is 2.35 kW, which is overscaled by a factor of 2.35 on a total system volume basis. However, pressurizer heat loss is estimated to be about 1 kW, which diminishes the effect of the overscaled heater capacity. Considering heat loss, the net overscaling is only about a factor of 1.42, which is exactly the amount the pressurizer volume is overscaled; therefore, the heater capacity is approximately correct considering heat loss and volume.

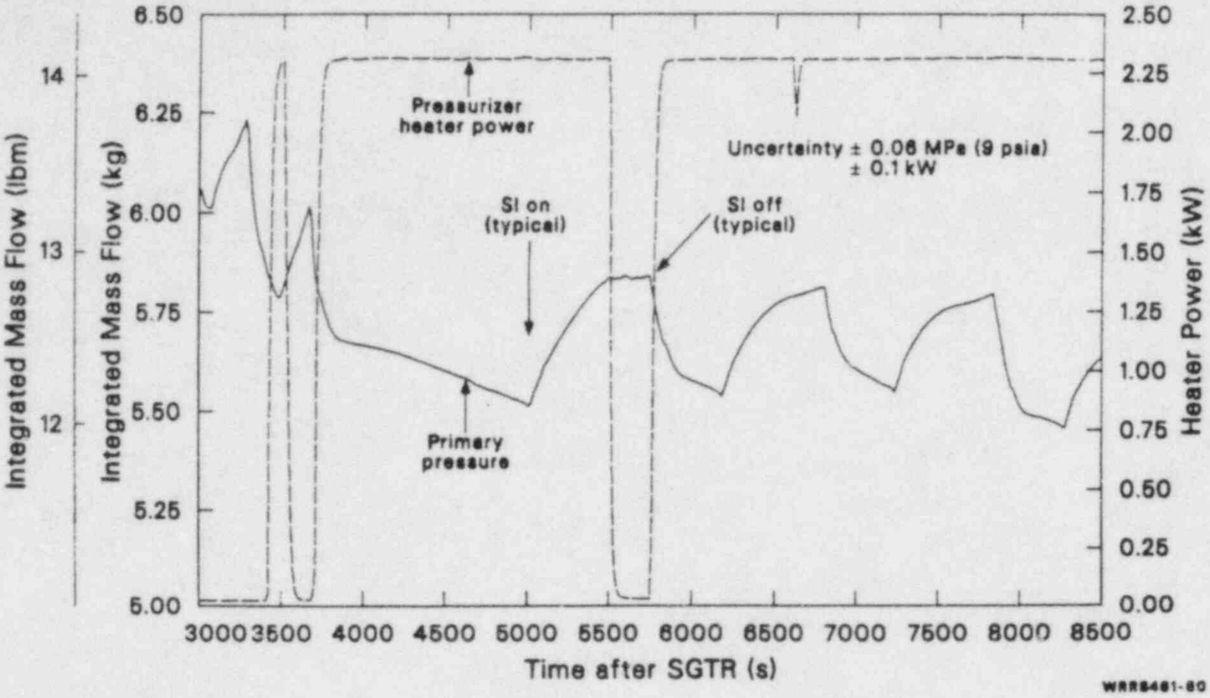


Figure 54. Pressurizer pressure and heater power during a cold-side, one-tube rupture transient.

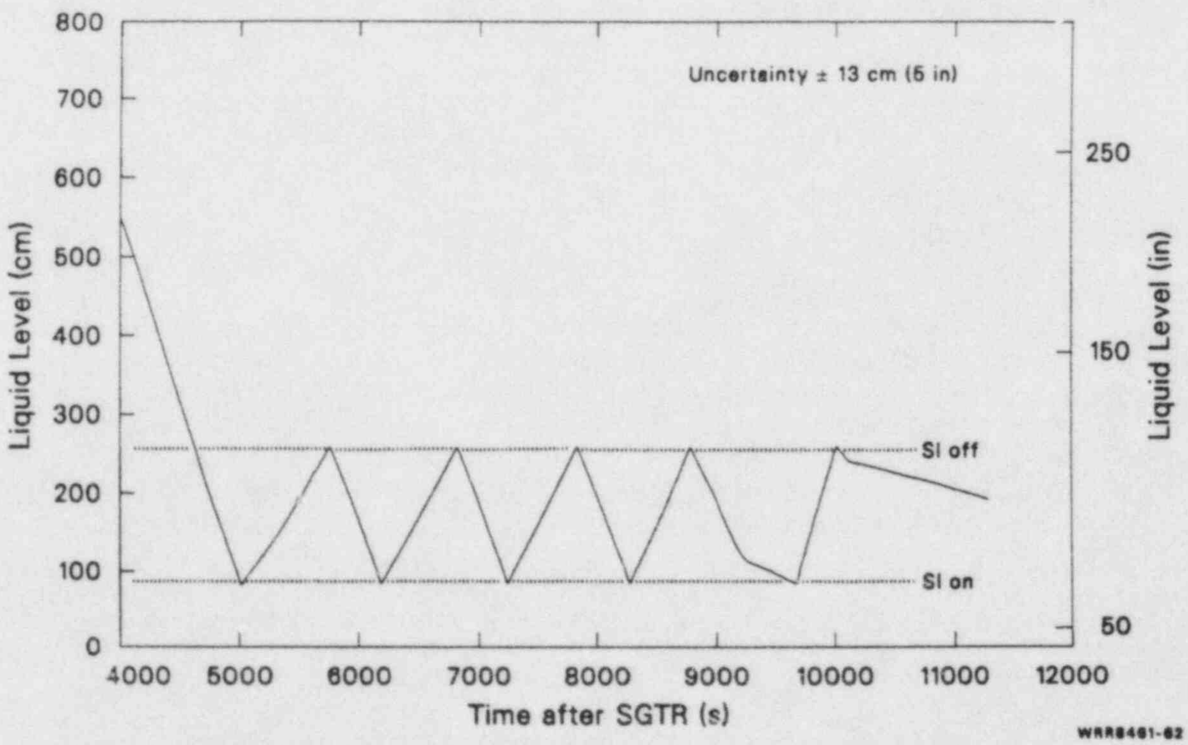


Figure 55. Pressurizer collapsed liquid level during pressurizer internal heater operation for a one-tube rupture transient.

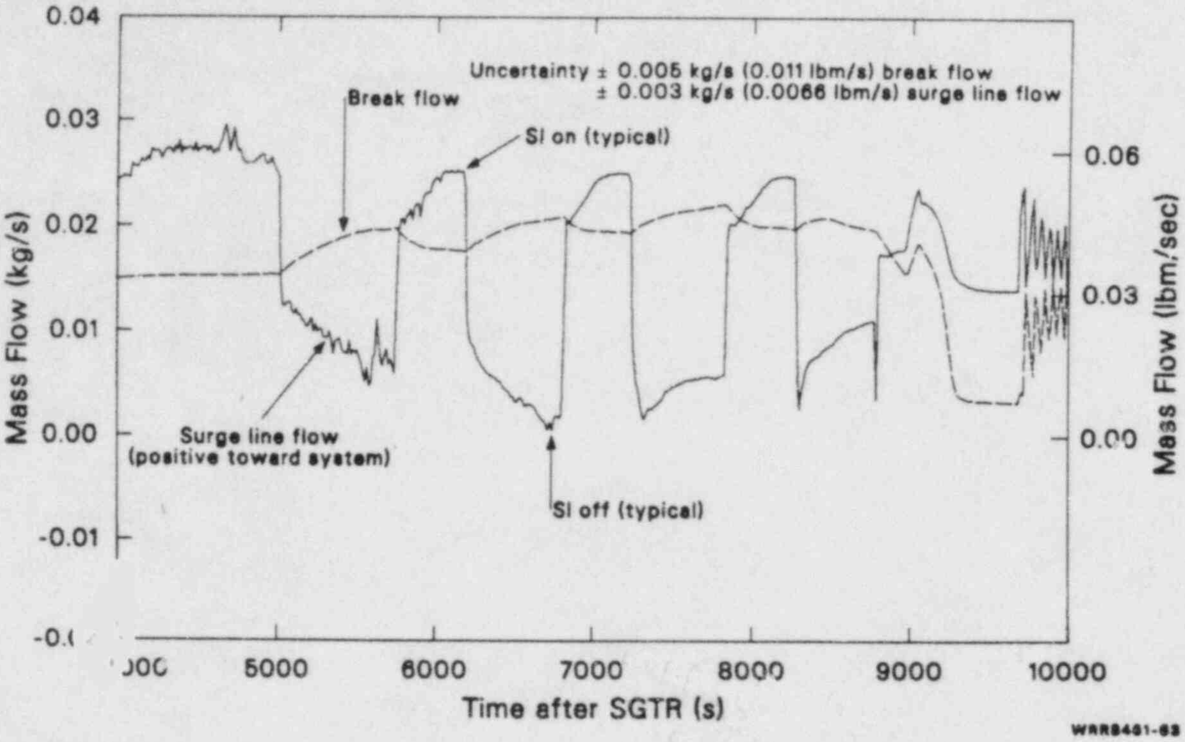


Figure 56. Break flow and pressurizer surge line flow during pressurizer internal heater operation for a cold-side, one-tube rupture transient.

When SI was off, bubble formation in the pressurizer due to heater operation could not offset the fluid volume lost in the break flow, resulting in no net primary system pressurization.

The Effectiveness of Safety Injection for Controlling Primary System Pressure. As part of the PWR emergency operating procedures, SI can be terminated to control primary system pressure depending on loop subcooling and pressurizer level.[4] During the GINNA transient,[7] SI was terminated, resulting in rapid primary system depressurization (as shown in Figure 57). In a similar manner, the Semiscale experiments used SI termination to reduce primary system pressure. This section discusses the system response upon SI termination during a single-tube rupture experiment in Semiscale.

During recovery from a single-tube rupture, termination of SI in conjunction with unaffected loop feed and steam was effective in first terminating an increase in primary system pressure and then causing primary system depressurization. This termination of SI and resulting primary depressurization was accomplished without core uncovery or core rod heatup. As discussed earlier, during a single-tube rupture the system mass inventory was such that at recovery initiation (600 s) the pressurizer was steam-filled and the vessel upper head was partially voided [collapsed level 75 cm (29.5 in.) below the top]. Feed and steam supported an increase in subcooling by cooling the loop fluid (see Figure 58); however, the primary pressure also increased during the period of unaffected loop feed and steam, as shown in Figure 59. This increase in primary pressure, which also increased the subcooling, was caused by void compression in the pressurizer and vessel upper head as SI flow was greater than break flow (Figure 60). With a higher SI flow than break flow, both the vessel and pressurizer collapsed levels increased (Figure 61), thus compressing the steam space in both regions. When SI was terminated at 3000 s, the combined rate of mass flow out of the pressurizer and vessel about equaled the break flow (Figure 60), allowing an increase in steam space in the pressurizer and vessel. Upon termination of SI, the primary pressure dropped as the voids expanded in the pressurizer and vessel (Figure 59). This primary depressurization was accomplished without the complications of flashing as the hot leg remained subcooled (Figure 58). Figure 59 indicates that the expansion of the voids upon SI termination followed a perfect gas assumption within 10%.

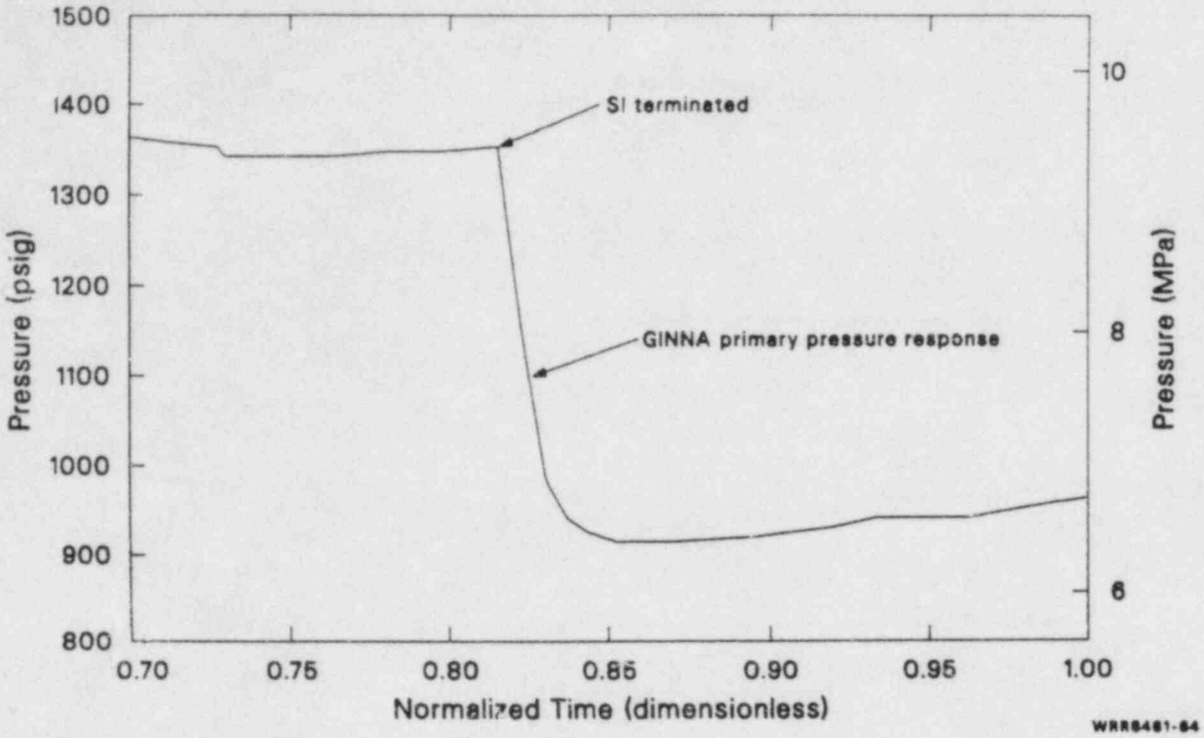


Figure 57. Primary system pressure response during the GINNA steam generator tube rupture transient SI operation.

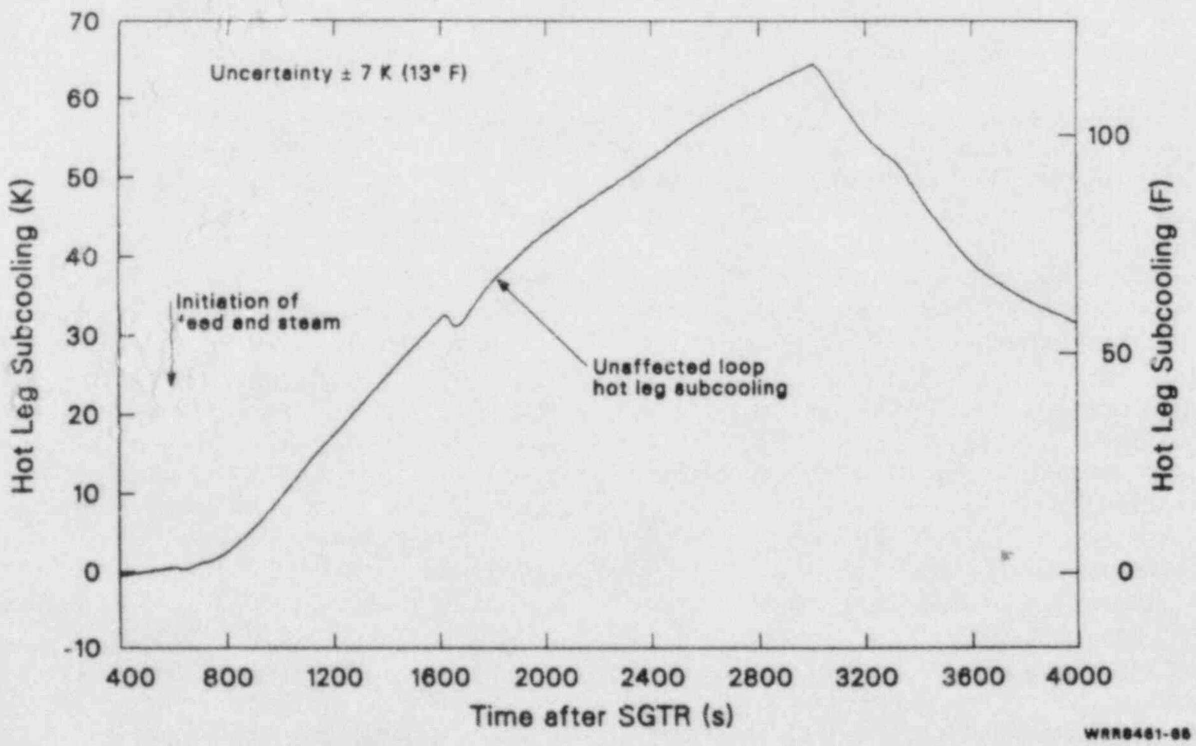


Figure 58. Hot leg fluid subcooling in the unaffected loop during steam and feed with SI for a one-tube rupture transient.

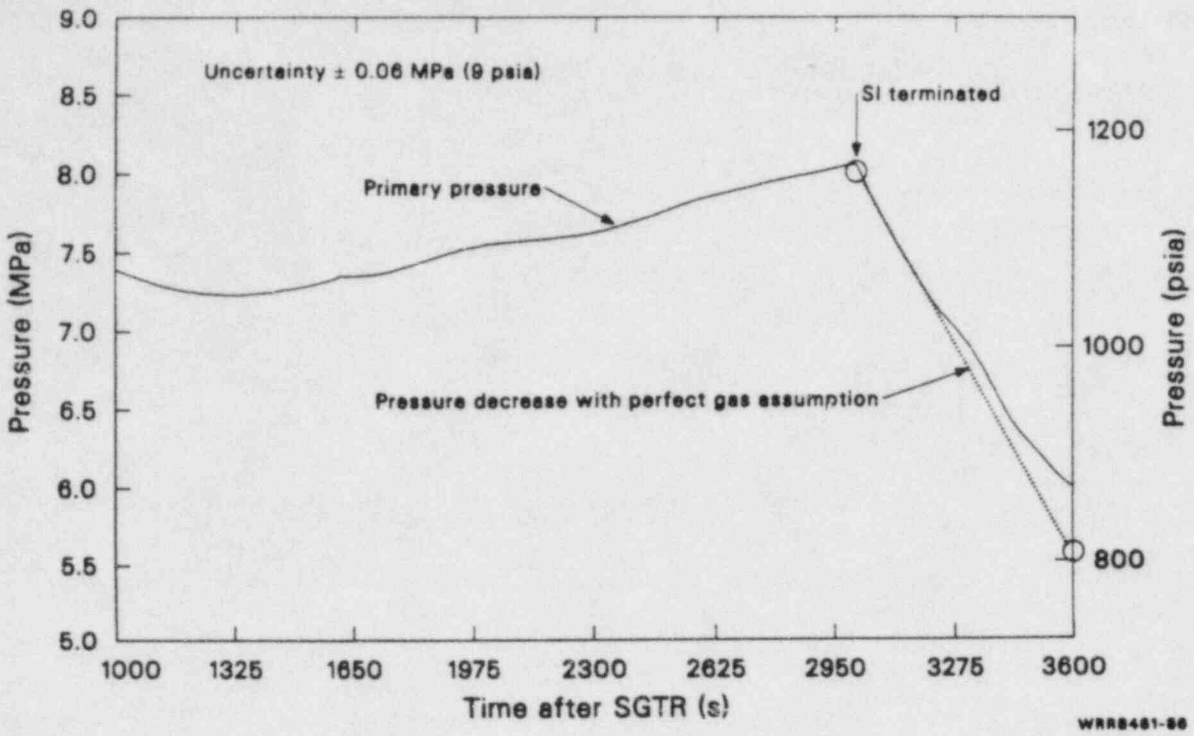


Figure 59. Primary system pressure during SI termination with perfect gas assumption for a one-tube rupture transient.

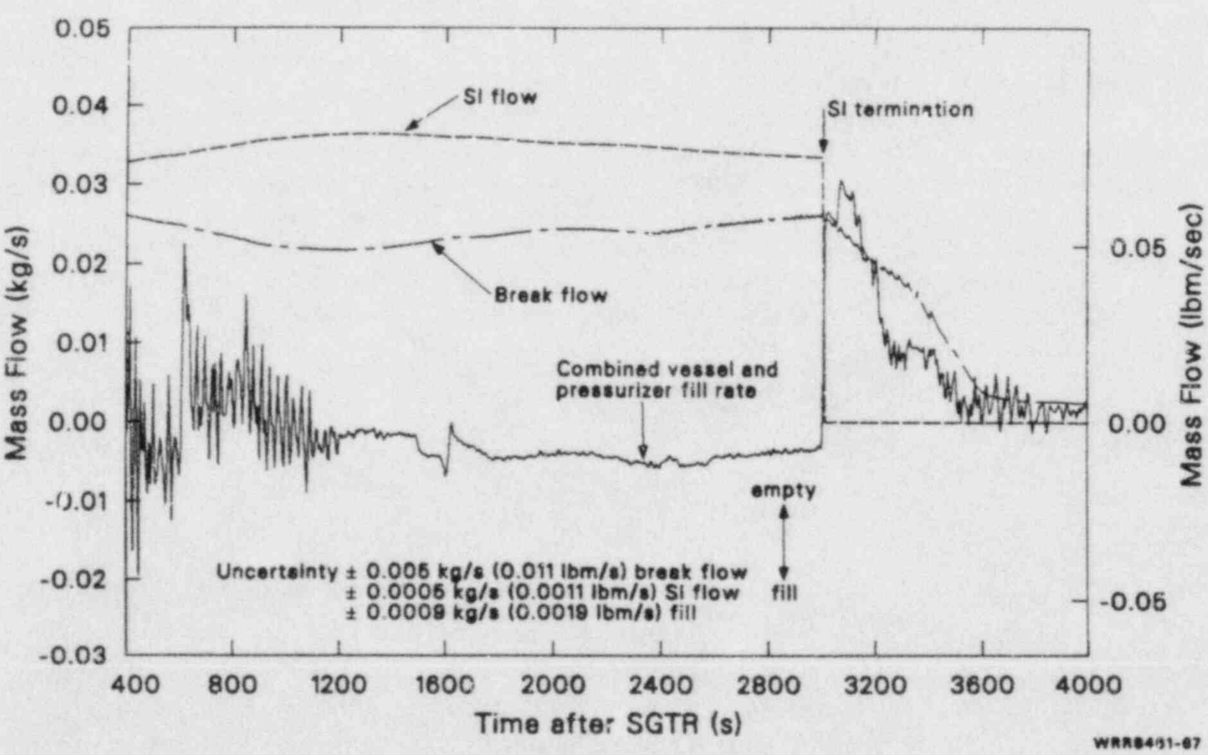


Figure 60. Break mass flow rate, SI mass flow rate, and the combined vessel and pressurizer fill rate during SI termination for a one-tube rupture transient.

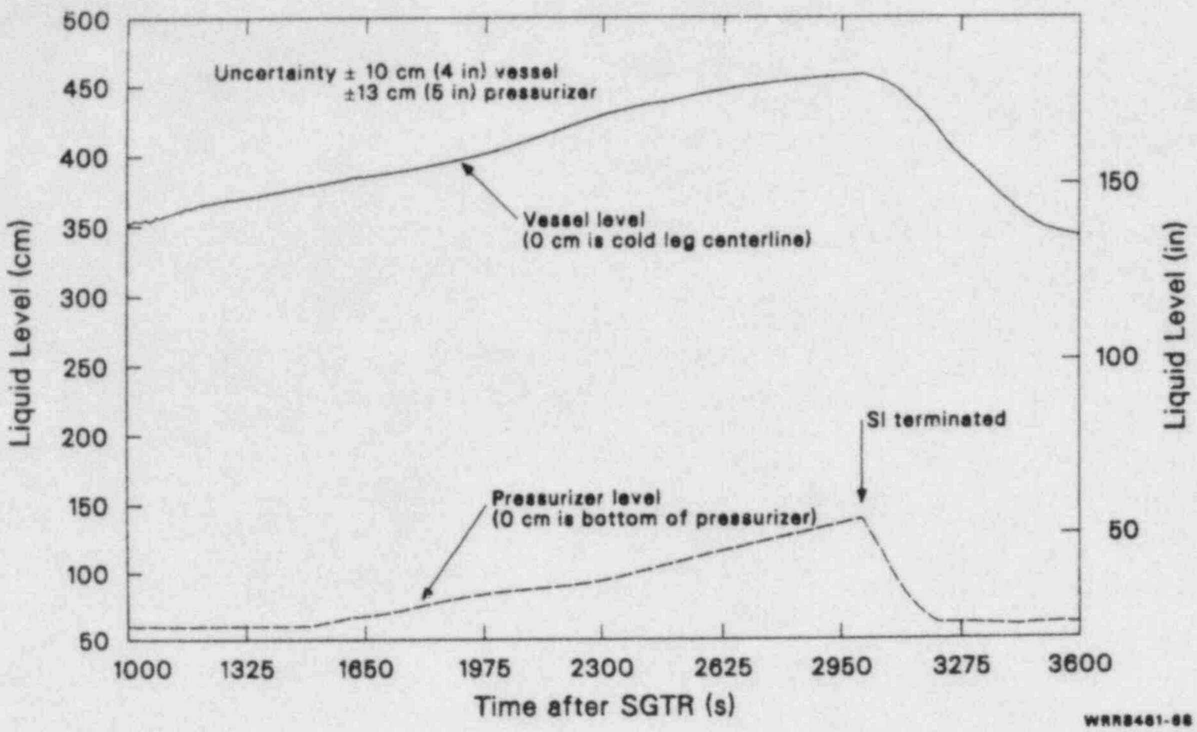


Figure 61. Collapsed liquid level in the pressurizer and vessel upper head during SI termination for a one-tube rupture transient.

The depressurization continued until the primary pressure reached the affected loop steam generator secondary pressure, effectively stopping break flow. With break flow stopped, the vessel liquid level remained above the cold leg elevation; therefore, the pressure reduction was accomplished without core uncovery or core heatup. Even though the pressure reduction was accomplished without core uncovery, the pressurizer was nearly empty of liquid, implying a lack of primary pressure control via pressurizer interval heaters. Pressure control can be accomplished, however, by cycling SI with continued feed and steam of the unaffected loop generator.

In summary, use of SI in a nearly full system (one tube break) resulted in primary system pressurization and an increase in hot leg subcooling. This increase in pressure and subcooling was due to compression of steam space in the pressurizer and vessel upper head while unaffected loop feed and steam aided in the subcooling of loop fluid. Once SI was terminated, the break mass flow rate about equaled the combined drain mass flow rate of the vessel upper head plus pressurizer, resulting in an expansion of steam space and a lowering of pressure. The pressure decrease was accomplished while maintaining sufficient vessel liquid inventory to preclude core uncovery and core rod heatup.

CONCLUSIONS

The following conclusions have been formulated, based on an analysis of the results of the Semiscale Mod-2B steam generator tube rupture test series.

1. The system signature response for a steam generator tube rupture is very distinctive for a wide range of tubes ruptured and for different rupture locations. The signature response as characterized by primary and secondary system pressures, flow, liquid levels, and temperatures is similar for tube ruptures on both the hot and cold sides of the steam generator and for a break spectrum including one, five, and ten tubes ruptured. In addition, the signature response for a main steam line break followed by a steam generator tube rupture is similar to the signature response when the tube rupture is the singular event. During a 600-s operator identification period, a tube rupture caused a rapid reduction in primary system pressure to saturation conditions, followed by a slow saturated blowdown as primary system fluid flowed to the affected loop secondary system through the tube rupture. Automatically occurring events that transpire during the 600 s period that affect the depressurization curve include: core scram, main steam isolation valve closure, safety injection initiation, feedwater termination and auxiliary feedwater initiation, and main coolant pump trip.

2. The pressurizer initial level at the time of tube rupture initiation has a large effect on the primary system depressurization and thus the timing of certain events. During the initial depressurization, flashing in the pressurizer retards primary system depressurization by steam formation. As long as there is an interfacial liquid level in the pressurizer, the flashing effect is enhanced; but, when the level drops to the surge line (a large area reduction), the flashing effect is diminished, resulting in an increase in depressurization. A higher initial pressurizer level thus leads to a slower primary system depressurization to the low pressure trip points. Even through the timing of certain trip points was different for different initial pressurizer levels, after a few hundred seconds the system mass inventory and hydraulic conditions were essentially the same.

3. The effectiveness of primary system pressure control during recovery from a tube rupture through use of unaffected loop secondary feed and steam is strongly dependent on the hydraulic state of the loop. If the loop is highly voided (in two-phase or reflux natural circulation), feed and steam induces condensation in the primary tubes of the unaffected loop which greatly reduces primary pressure; however, if the loop is nearly full of liquid (single-phase natural circulation), the effect of feed and steam on primary pressure is minimal. In this highly flooded situation, the feed and steam causes only a slight pressure reduction due to shrinkage of fluid caused by the increase in primary-to-secondary heat transfer.

4. During recovery from a tube rupture, PORV operation was effective in reducing primary system pressure; however, the effectiveness of the PORV operation is dependent on the pressurizer liquid level. When the level is low (below half-full), the flow out the PORV is mostly steam; this affords a large primary volume removal in a short time, thus causing a large primary pressure reduction. When the pressurizer is nearly full, the outflow is mostly liquid, resulting in a small volume reduction and, thus, a small pressure reduction.

5. During recovery from a tube rupture, pressurizer auxiliary spray is effective for reducing primary system pressure; however, the effectiveness is strongly dependent on pressurizer wall and steam superheat removal. The introduction of cold auxiliary spray water into the pressurizer changes the fluid and wall temperatures from superheated to saturated. This removal of superheat causes a slight primary pressurization due to evaporation of auxiliary spray. Once the superheat is removed, continued spray causes a pressure reduction due to condensation. The effectiveness of the spray operation for pressure reduction depends on pressurizer liquid level. The spray is only effective if the liquid level is below the top of the pressurizer.

6. Pressurizer internal heaters are ineffective for increasing primary system pressure during a tube rupture. Bubble formation in the pressurizer due to heater operation could not offset the fluid volume lost due to tube rupture break flow. As a result, there was no compression of primary fluid and thus no net rise in primary pressure. In order for pressurizer heaters to be effective for increasing primary pressure, tube rupture break flow would have to be zero.

7. During recovery from a tube rupture, control of SI can be used to either increase primary system pressure and loop subcooling or reduce primary pressure. Use of SI in a nearly full system causes a compression of steam spaces and a primary system pressurization. The primary pressurization due to SI increases the subcooling in the hot leg. Termination of SI can cause a lowering of primary pressure because of expansion of voids in the loop caused by continued tube rupture break flow.

REFERENCES

1. G. G. Loomis, "Steam Generator Tube Rupture in an Experimental Facility Scaled from a Pressurized Water Reactor," *5th International Meeting on Thermal Nuclear Reactor Safety, Karlsruhe West Germany, September, 1984.*

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

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

4. Emergency Operating Procedure for Zion Nuclear Power Plant (EOP-10).

5. G. G. Loomis and K. Soda, *Results of the Semiscale Mod-2A Natural Circulation Experiments*, NUREG/CR-2335, September, 1982.

6. J. G. Collier, *Convective Boiling and Condensation*, McGraw-Hill, New York, 1972.

7. T. T. Martin, *NRC Report on the January 25, 1982 Steam Generator Tube Rupture at R. E. Ginna Nuclear Power Plant*, NUREG-0909, April, 1982.

8. Crane Manual, "Flow of Fluids through Valves Fittings and Pipe," Crane Co., New York, New York, Technical Paper No. 410, 1982.

APPENDIX A
THE EFFECT OF SCALE ON SEMISCALE STEAM GENERATOR TUBE RUPTURE RESULTS

APPENDIX A
THE EFFECT OF SCALE ON SEMISCALE STEAM GENERATOR TUBE RUPTURE RESULTS

The Semiscale Mod-2B facility was designed to produce thermal-hydraulic responses similar to those expected in large scale PWRs. Certain scaling distortions, however, preclude the Semiscale simulations from being precise replications of PWR transients. This section discusses the effect of some of these scaling distortions on the steam generator tube rupture results. Even though the scaling distortions may preclude the use of Semiscale results as demonstration, the primary purpose of the Semiscale data base is for computer code assessment and verification.

Basically, the 1:1 elevation scaling for steam generators, vessel, and pump suctions allows the correct geometry for natural circulation phenomena to occur. Elevation is one of the most important scaling factors, because natural circulation is driven by gravity head differences in the loop. Because the main coolant pump is tripped during tube rupture transients, natural circulation is the heat transfer mechanism for core decay heat removal. Correct scaling of the steam generators and heat loss are also involved in correct natural circulation phenomena.

The metal-mass-to-liquid-volume ratios for both unaffected and affected loop steam generators are both about a factor of 8.9 overscaled. This scaling distortion is an unavoidable consequence of the small scale/high pressure system. The Semiscale piping requires a large metal mass to withstand the pressure but also requires a small liquid volume to maintain volume scaling. The PWR metal-mass-to-liquid-volume ratio at pretransient hot conditions is about 4.7. The Semiscale unaffected loop metal-mass-to liquid-volume ratio is 42.5, and the affected loop ratio is 45.5. This scaling distortion is not thought to affect natural circulation-type behavior because of the long-term nature of the natural circulation flow condition (on the order of hours). This time frame allows equalization of metal and fluid temperatures. However, distortions do exist in the short transients, such as during MSIV closure at low pressurizer pressure trip. Previous experiments have shown that the oversized metal mass in the Semiscale steam generators causes a lower pressurization upon MSIV closure and core scram. The metal mass represents a large heat sink and thus reduces the pressurization. The referenced PWR plant ADV setpoint is 7.22 MPa (1048 psia). To achieve a lifting condition of the Semiscale ADV valves during MSIV closure and core scram, two changes were specified: the initial operating secondary system pressure was increased, and the ADV relief valve setpoints were lowered.

The initial steady-state secondary system pressure was increased from a nominal value of 5.34 MPa (775 psia) (representative of many PWR's) to a higher value. Since several Westinghouse units (Salem Units 1 and 2, for example) operate at 5.55 MPa (805 psia), the Semiscale secondaries were operated at this value.

The ADV relief valve setpoints were lowered in the Semiscale system to provide a more representative lifting of secondary system relief valves upon MSIV closure. Previous Semiscale data indicate only a 1.2 and 0.5 MPa (174 and 72.5 psia) secondary system pressurization upon MSIV closure for the unaffected and affected loop respectively; therefore, the relief valves would never be challenged unless the setpoints were lowered. To ensure that the relief valves would be challenged during the steam generator test series, the ADV setpoints were lowered to no more than 1.20 MPa (174 psia) above the initial pressure of 5.55 MPa (805 psia) in the unaffected loop and 0.5 MPa (72.5 psia) above the initial value in the affected loop. To allow for some margin in the conditions, a 0.2-MPa (29 psia) allowance was used, thus making the unaffected loop setpoint 6.55 MPa (950 psia) and the affected loop setpoint 5.85 MPa (848 psia). The desired scaled flow through these valves was obtained by taking the desired PWR flow and dividing by the PWR/Semiscale thermal power ratio (1705.5). The flow of saturated steam in the unaffected loop at 6.55 MPa (950 psia) was 0.21 kg/s (10.46 lbm/s), and flow in the affected loop at 5.85 MPa (848 psia) was 0.07 kg/s (0.154 lbm/s).

Another scaling distortion which affected loop secondary system performance was the amount of initial liquid in the affected loop steam generator which can distort the secondary fill time during a tube rupture. Table A-1 summarizes the PWR volume,[A-1] correctly scaled volume, and actual

Table A-1. Initial steam generator secondary system volumes of a PWR and Semiscale

System	Liquid Volume (m^3/ft^3)	Steam Volume (m^3/ft^3)
PWR (Zion)	85.0/3000	81.0/286.0
Correctly scaled Semiscale		
Unaffected loop	0.149/5.26	0.14/4.94
Affected loop	0.049/1.73	0.047/1.66
Semiscale		
Unaffected loop	0.13/4.59	0.21/7.41
Affected loop	0.13/4.59	0.13/4.59

Semiscale values. The unaffected loop is approximately correct in scale on water volume and only about 1.5 overscaled on steam volume. The affected loop, however, is about 2.65 overscaled on both water and steam. The water volume overscaling was intentional to reduce steam volume distortion and still be able to operate in a stable manner. As a result of these scaling considerations, filling of the affected loop secondary system due to tube rupture break flow should take longer in Semiscale than in a PWR. This is important because filling of the secondary can result in water flow through the ADV valve, thus increasing the amount of radioactive release from a PWR. For that reason, the Semiscale results cannot be considered conservative; assuming the break flow is correct, the filling time should be longer in Semiscale than in a PWR. The affected loop steam volume is about 0.13 m^3 (4.59 ft^3) in Semiscale, and the correctly scaled value should be 0.047 m^3 (1.65 ft^3). Therefore, the Semiscale break flow should take about 2.7 times as long as a correctly scaled system to fill the affected loop generator.

REFERENCE

A-1. Zion Nuclear Generating Station System Description, Chapter 21.

APPENDIX B
STEAM GENERATOR TUBE RUPTURE SIGNATURE RESPONSE USING ABNORMAL TRANSIENT OPERATION GUIDELINES (ATOG)

APPENDIX B
STEAM GENERATOR TUBE RUPTURE SIGNATURE RESPONSE USING ABNORMAL TRANSIENT OPERATION GUIDELINES (ATOG)

As part of an accident signature response, steam generator tube rupture can be characterized on plots of primary system pressure versus hot leg fluid temperature. During both normal and abnormal operation, it is desirable to maintain a fluid subcooling margin in the hot leg of a PWR [usually 22 to 28 K (40 to 50°F)]. This appendix presents "Abnormal Transient Operating Guidelines" (ATOG) plots of primary pressure versus hot leg fluid temperature. Superimposed on these plots are the saturation line and a 22 K (40°F) subcooled line. The pressure/temperature data to the right of the saturation line imply superheated steam, and the data to the left of the line imply subcooled liquid. The desirable operating conditions in a PWR are to stay on the subcooled line or to the left of the subcooled line. Figures B-1 through B-9 present ATOG-type plots for the steam generator test series, Tests S-SG-1 through S-SG-9 respectively. Significant operator actions during recovery are indicated on these plots, such as unaffected loop feed and steam (F/S), primary feed and bleed (F/B), safety injection (SI), and pressurizer auxiliary spray. The effect of these actions on increasing or decreasing subcooling is obvious in these figures.

All the tube rupture experiments were initiated from a subcooled condition. In the flow of primary fluid to the secondary due to the tube rupture, the hot leg fluid eventually became saturated (Figures B-1 through B-8). In Experiment S-SG-9 (Figure B-9), the initiating event was a main steam line break (MSLB) followed by a tube rupture. The MSLB did not adversely affect loop subcooling; however, the steam generator tube rupture (SGTR) caused the change from subcooled to saturated condition. Recovery techniques generally increased hot leg subcooling; however, in several experiments this was not possible because of compounding failures. In experiments S-SG-6 and S-SG-7 (Figures B-6 and B-7 respectively), the hot leg fluid remained saturated despite recovery efforts. During S-SG-6, the compounding failure was a stuck open affected loop atmospheric dump valve; in S-SG-7, a complete onsite and offsite power loss was assumed, precluding the use of SI during recovery. During S-SG-8, F/S was delayed while F/B was commenced following a 600-s operator diagnostic period. The F/B operation reduced primary pressure, causing flashing in the hot leg; and the F/S operation, once started, was unable to promote subcooling of fluid in the hot leg.

The usefulness of these ATOG plots can be seen by reviewing the chronology of events for Test S-SG-5, as shown in Figure B-5. Starting from subcooled primary system fluid conditions [approximately 22 K (40°F)], the tube rupture event occurred, resulting in a rapid depressurization to saturation conditions. For this experiment, it was assumed that the operator identified that a tube rupture had occurred early (about the time the system fluid achieved saturation conditions). Following normal emergency procedures, feed and steam of the unaffected loop steam generator was initiated while SI and tube rupture break flow continued. Eventually, SI flow was greater than break flow, allowing a net positive influx of system mass which caused a compression of voids in the system. The operator would observe this on an ATOG plot as an increase in loop subcooling, as the void compression increased loop pressure but not temperature. Since the primary system loop and affected loop secondary were hydraulically coupled via the break and, further, since SI had increased primary system pressure, the affected loop ADV cycled several times, maintaining primary pressure at the affected loop ADV setpoint. Meanwhile, continued feed and steam in the intact loop increased primary fluid subcooling. To eliminate excessive affected loop ADV cycling and potential atmospheric release of secondary fluid, SI was terminated, thus removing the compressing effects on system voids. The primary system pressure then dropped, decreasing primary fluid subcooling, which remained above 22 K (40°F). Since primary system pressure was below the affected loop ADV setpoint, potential affected loop secondary fluid release to atmosphere was no longer a problem. An operator could plot progress during a transient on similar ATOG plots and immediately ascertain its effect on primary system pressure control and primary fluid subcooling.

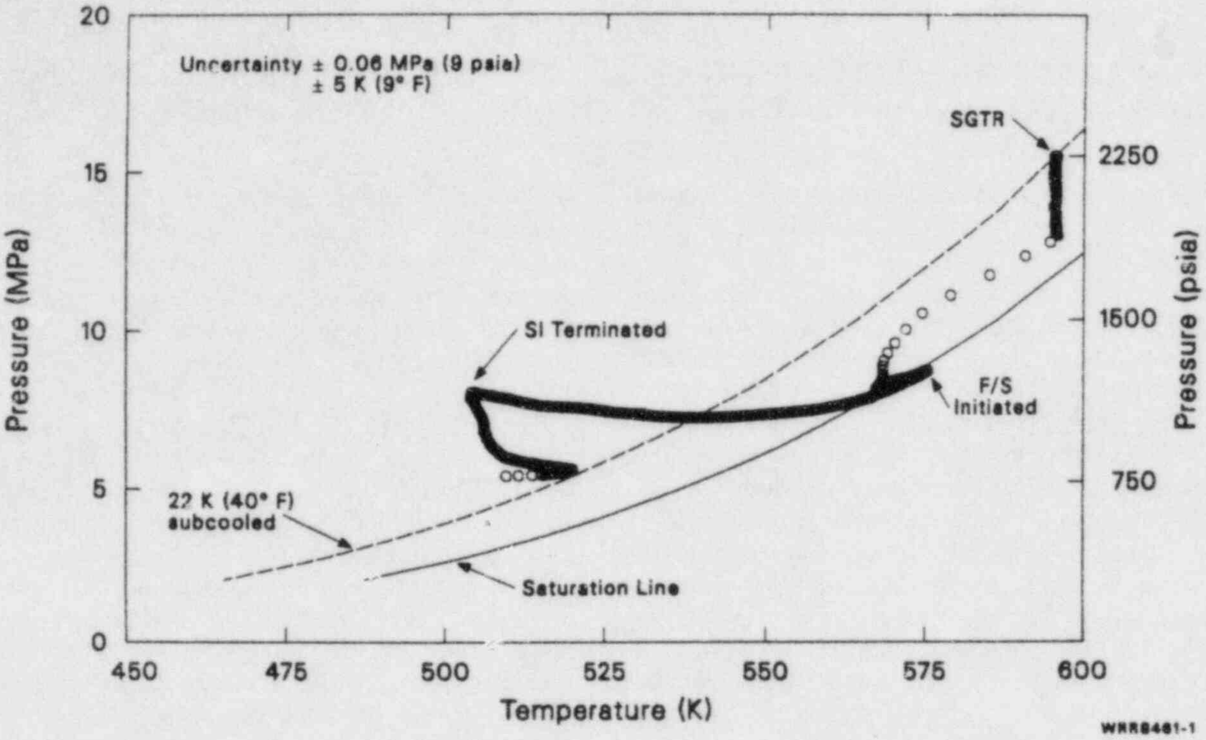


Figure B-1. ATOG plot for Semiscale Experiment S-SG-1 (one-tube rupture with feed and steam).

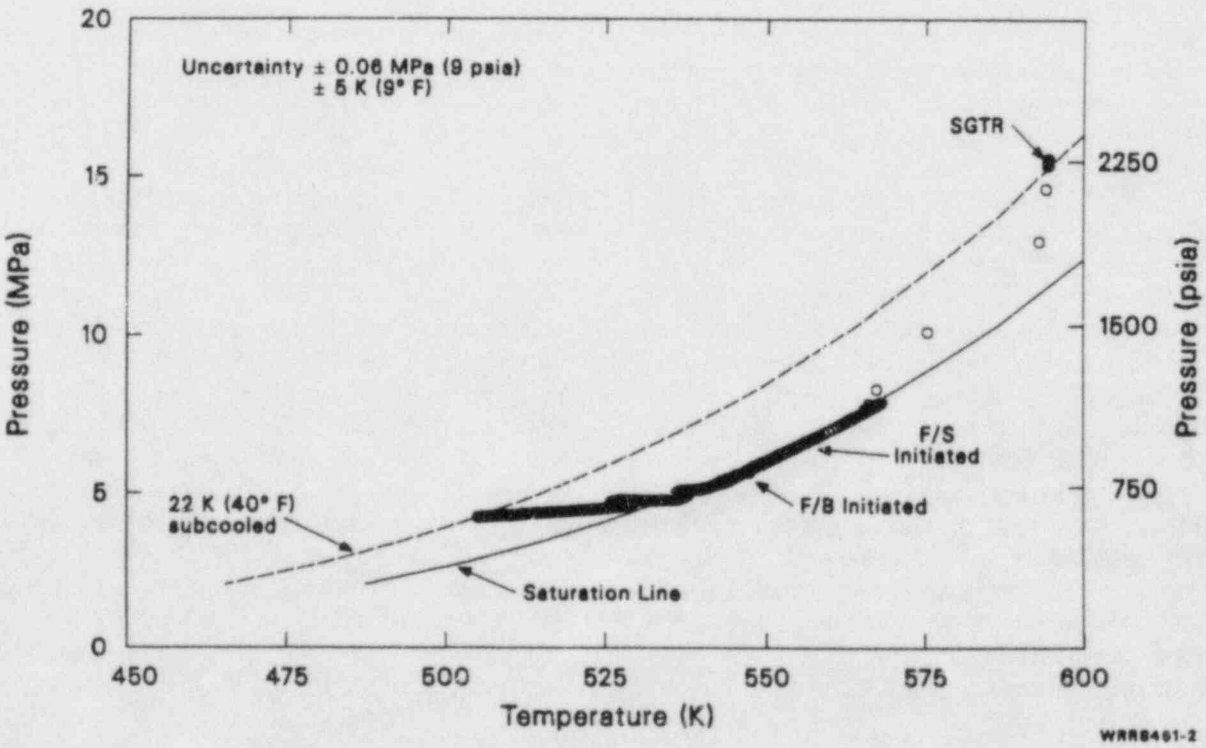


Figure B-2. ATOG plot for Semiscale Experiment S-SG-2 (five-tube rupture with feed and steam and feed and bleed).

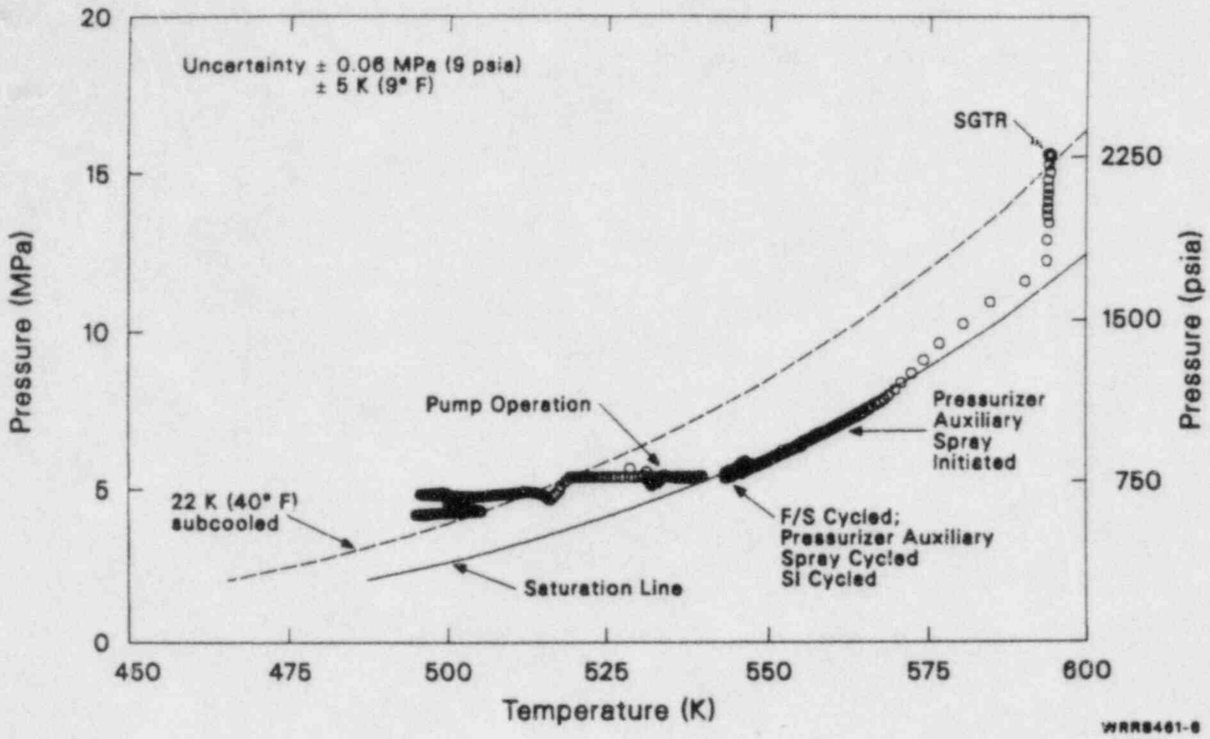


Figure B-3. ATOG plot for Semiscale Exreriment S-SG-3 (ten-tube rupture with combined recovery).

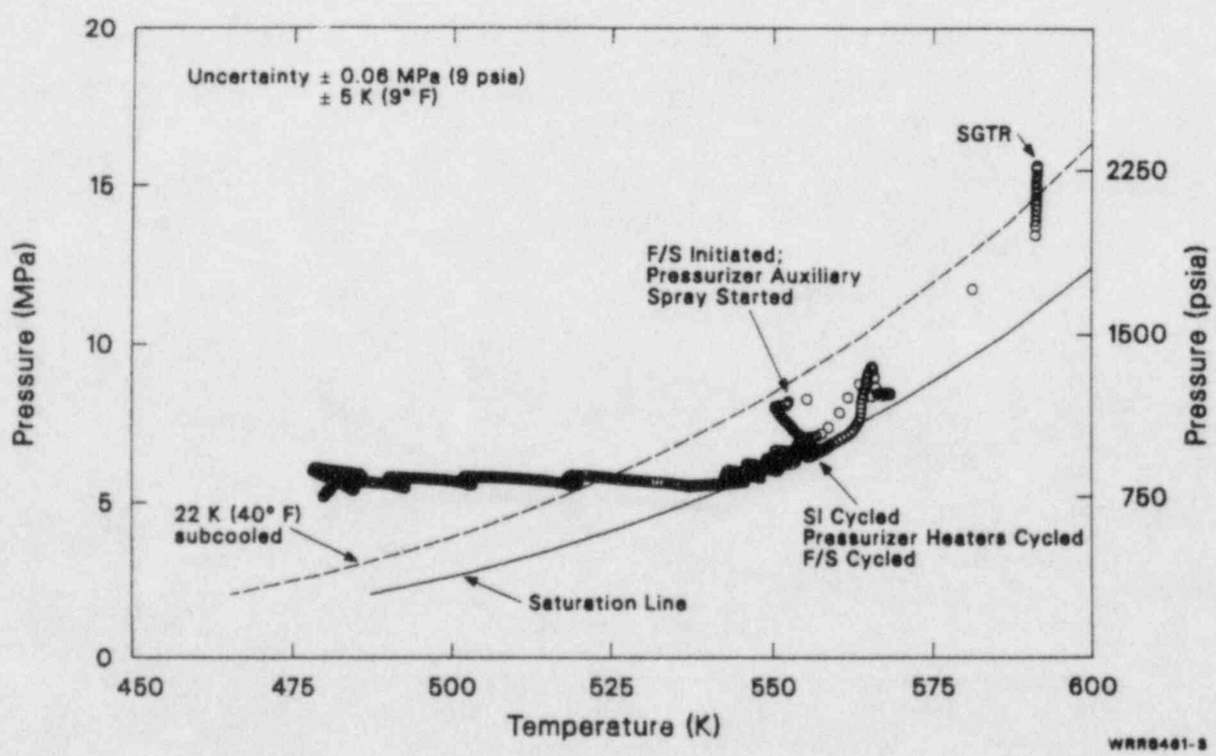


Figure B-4. ATOG plot for Semiscale Experiment S-SG-4 (one-tube rupture with delayed pump trip).

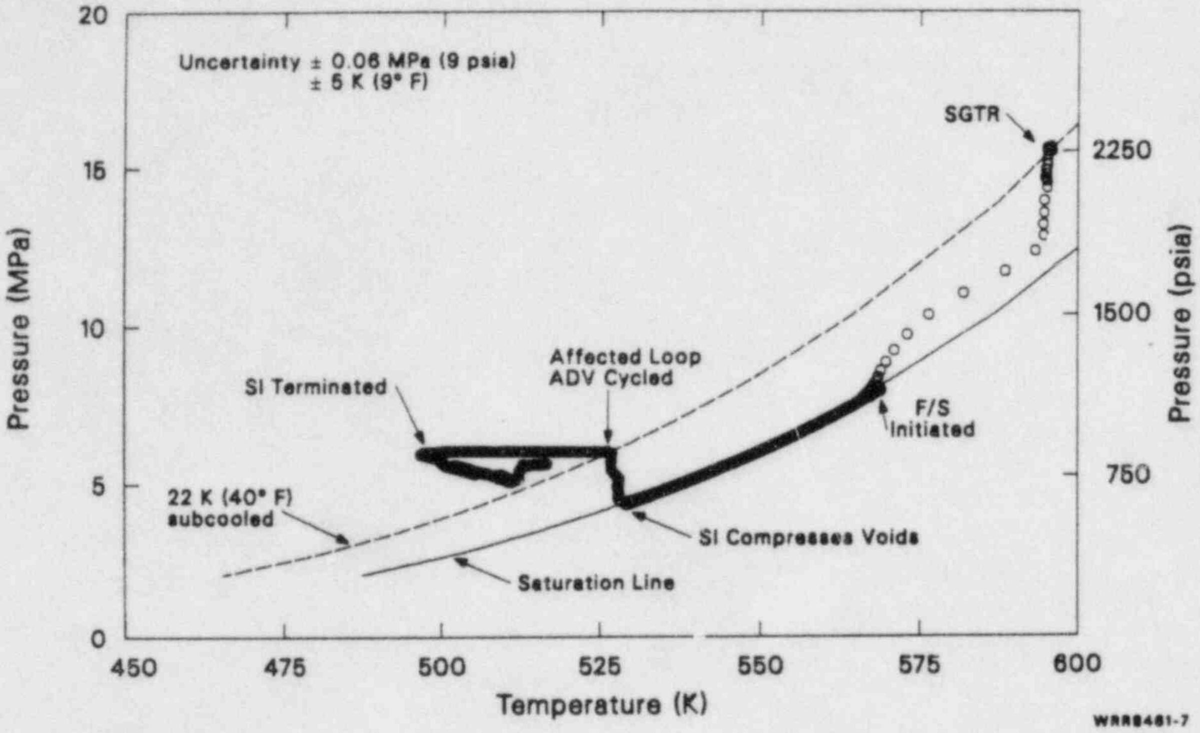


Figure B-5. ATOG plot for Semiscale Experiment S-SG-5 (five-tube rupture with early feed and steam).

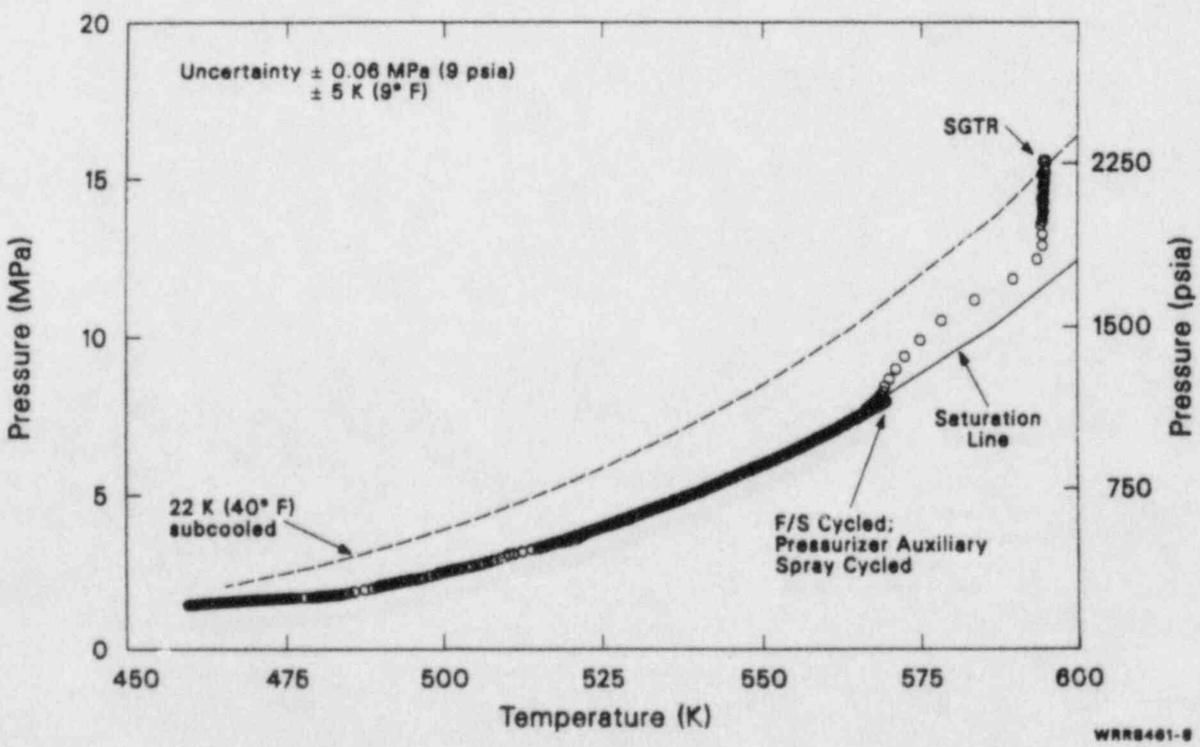


Figure B-6. ATOG plot for Semiscale Experiment S-SG-6 (five-tube rupture with stuck open affected loop ADV).

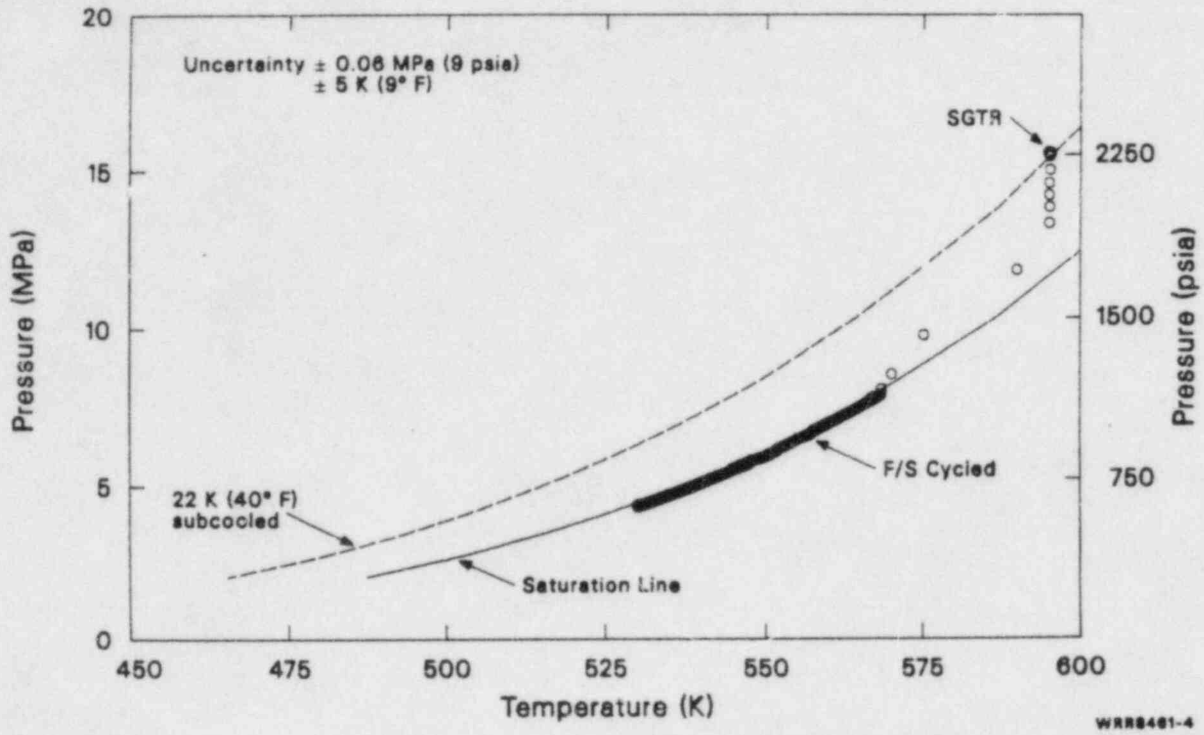


Figure B-7. ATOG plot for Semiscale Experiment S-SG-7 (five-tube rupture with power loss).

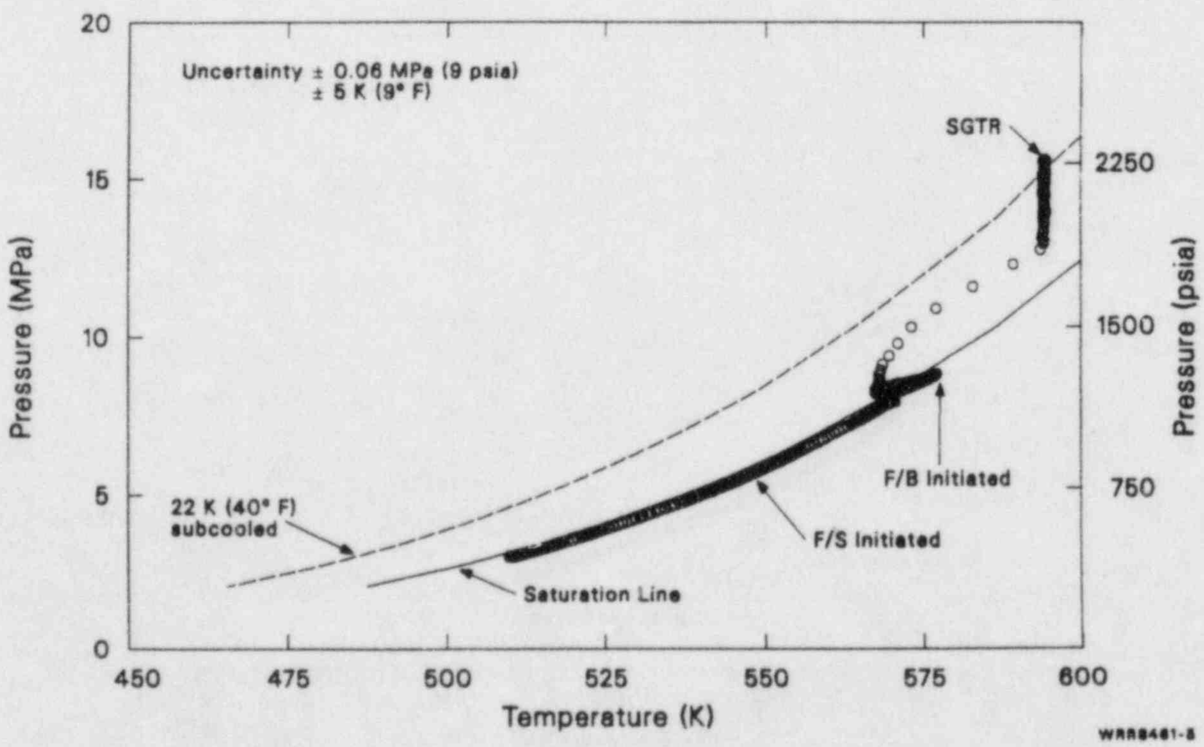


Figure B-8. ATOG plot for Semiscale Experiment S-SG-8 (one-tube rupture with feed and bleed and feed and steam).

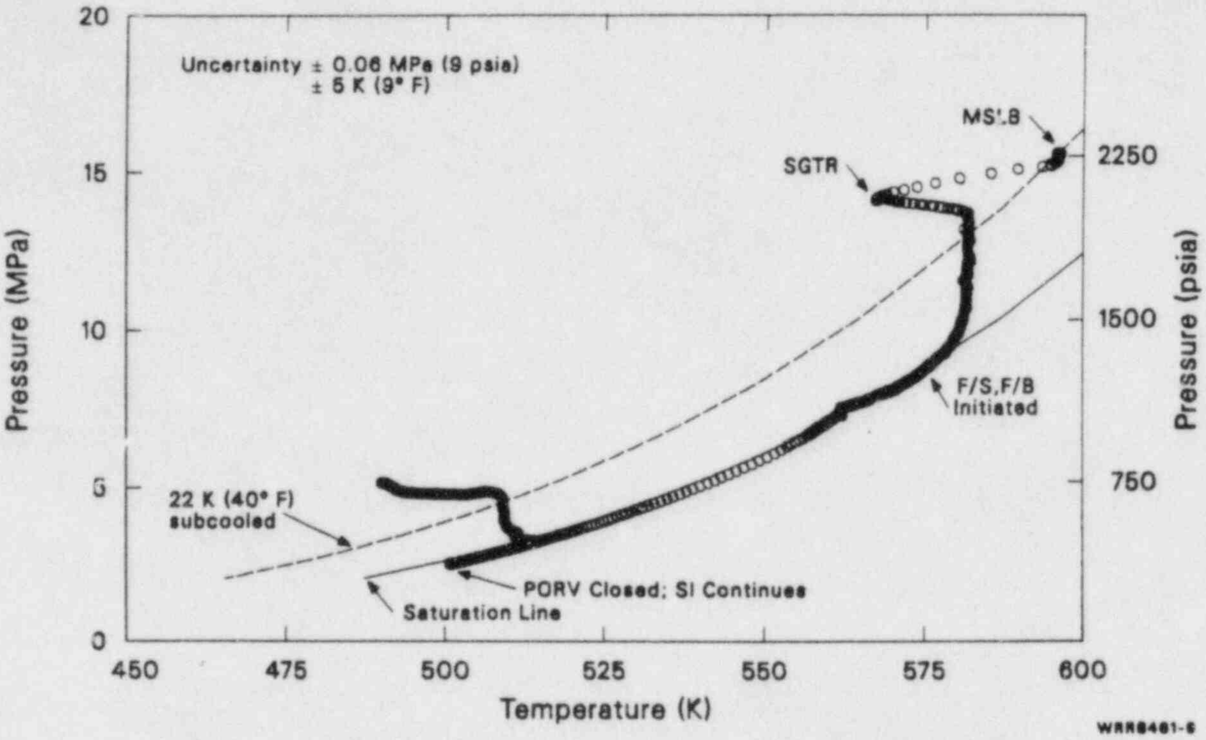


Figure B-9. ATOG plot for Semiscale Experiment S-SG-9 (main steam line break with concurrent one-tube rupture, including feed and steam, feed and bleed, and SI).

NRC FORM 335
(2-84)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)
NUREG/CR-4073

2. TITLE AND SUBTITLE
Results of the Semiscale Mod-2B Steam Generator Tube Rupture Test Series

3. LEAVE BLANK

4. DATE REPORT COMPLETED
MONTH: January | YEAR: 1985

5. AUTHOR(S)
Guy G. Loomis

6. DATE REPORT ISSUED
MONTH: January | YEAR: 1985

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER
A6038

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

11a. TYPE OF REPORT

b. PERIOD COVERED (Inclusive dates)
8/83 through 4/84

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

A series of experiments was conducted in a scaled model of a pressurized water reactor (Semiscale Mod-2B) to investigate steam generator tube rupture system signature response and recovery techniques. The tube rupture was assumed to occur during normal full power operation [15.6 MPa (2262 psia) system pressure; 37 K (67°F) core differential temperature]. From the experimental results, the characteristic system signature responses for a wide range of number of tubes ruptured and rupture locations have been examined. In addition, recovery techniques requiring operator actions were examined. These recovery techniques included the use of pressurizer auxiliary spray and internal heaters, steam generator feed and steam, primary feed and bleed, and safety injection. The effectiveness of using these techniques for primary system pressure and subcooling control is discussed.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT

16. SECURITY CLASSIFICATION
(This page) Unclassified
(This report) Unclassified

17. NUMBER OF PAGES
62

18. PRICE

120555078877 1 1ANIRZ
US NRC
ADM-DIV OF TIDC
POLICY & PUB MGT BR-PDR NUREG
W-501
WASHINGTON DC 20555

EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, Idaho 83415