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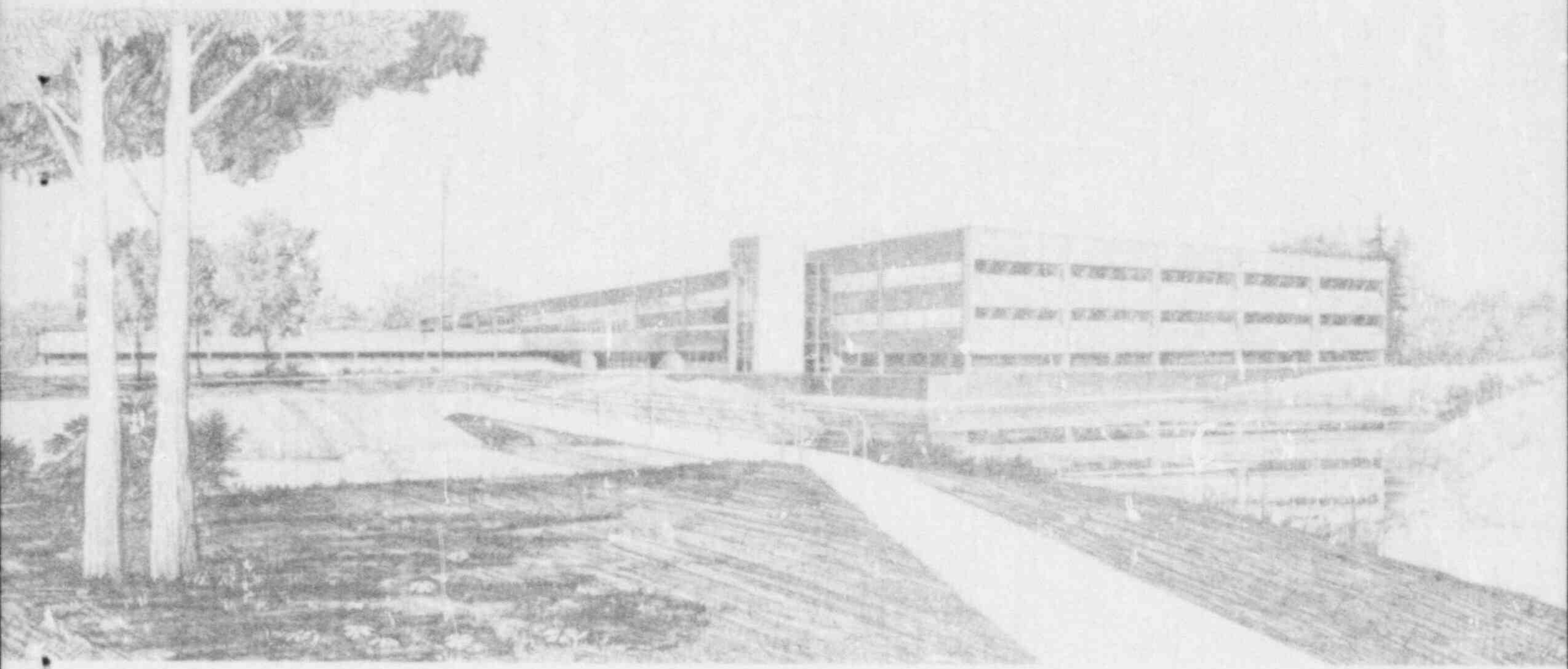

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Summary of Semiscale Small Break Loss-of-Coolant Accident Experiments (1979 to 1985)

Guy G. Loomis

September 1985

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ABSTRACT

Following the loss-of-coolant accident at TMI in 1979, a multitude of small break loss-of-coolant accident experiments were performed in the various Mods of the Semiscale facility at the Idaho National Engineering Laboratory (INEL). A summary of what experiments have been performed and a description of the various Semiscale Mods is given. The signature response of various kinds of small breaks are characterized. Small break loss-of-coolant accident issues addressed by Semiscale testing are discussed, including: effect of break location and break size, effect of core bypass flow, preferred primary coolant pump operation, effectiveness of upper head emergency core cooling injection, and recovery procedures. Phenomena of interest to small break loss-of-coolant accident analysis is presented including core uncovery heat transfer and natural circulation. Recommendations are given that can improve calculational capabilities for future small break testing.

FIN No. A6038—Semiscale Program

EXECUTIVE SUMMARY

The Semiscale experimental program conducted by EG&G Idaho, Inc., is part of the overall research and development program sponsored by the United States 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. This report presents a description of the extensive small break loss-of-coolant accident (SBLOCA) experimental data base available from Semiscale testing and summarizes pertinent results from these experiments.

Small break experiments were conducted in a series of Mods of the Semiscale facility including Mod-1, Mod-2A, Mod-2B, Mod-2C, and Mod-3. Basically, all of these Mods included a vessel with electrically heated core simulating a nuclear core and two loops (one loop simulates three unaffected loops and the other simulates the loop in which a small break is postulated to occur). All of the experiments are performed at high pressure/high temperature conditions [15.5 MPa (2250 psia), 595 K (611°F)] hot leg temperature with a core differential temperature of 37 K (67°F).

The water reactor research community shifted interests from large break loss-of-coolant accidents to SBLOCAs with the advent of the TMI-2 accident in March 1979. Experimental series were performed in the various Mods of Semiscale to investigate such topics as effect on transient severity of: break location and size, core bypass flow, preferred primary coolant pump operation, and upper head emergency core cooling injection. In addition, the effectiveness of recovery techniques, and the consequences of compounding failure during SBLOCAs is discussed. SBLOCA phenomena of interest to model development efforts includes core uncovery heat transfer and natural circulation flow. The following summarizes important results condensed from the multitude of small break experiments performed in the Semiscale facility.

The signature response for three different kinds of SBLOCA were investigated in the Semiscale facilities including pipe breaks, steam generator (SG) tube rupture, and stuck open pressurizer power operated relief valve (PORV) (TMI-2 type accident). The pipe break and steam generator tube rupture were found to have similar signature response, while all three SBLOCA types were accompanied by a net loss of primary mass inventory. The pressure signature response to both a steam generator tube rupture and pipe break show a continuous decrease in primary pressure upon break initiation with varying inflection points. Core scram to the ANS decay curve increased the primary depressurization rate dramatically as primary fluid cooled in the absence of full core power input but with continuing primary to secondary heat transfer (shrinkage of the primary fluid). The primary pressure dropped rapidly until fluid saturation conditions were achieved in the loop, at which time flashing of fluid resulted in a decreasing depressurization rate. Events that affect the pressure signature response for pipe breaks include pump suction seal clearing and break uncovery and accumulator injection.

As depressurization occurs during a small cold leg pipe break, the depletion of liquid in the system (because of a higher break flow than safety injection flow) follows a general top-down voiding. Because of the geometry of the pump suction piping, a seal of water becomes trapped in the pump suction forming a plug for steam flow from the vessel and hot legs to the break. A manometric balance of fluid heads develops that causes a depression of the liquid level in the pump suction and core. This manometric balance and core liquid level depression is aggravated by fluid heldup in the primary loop, most notably the primary U-tubes. The manometric balance is only in a quasi-steady state mode as a steam water interface is pushed down the downflow side of the pump suction and up the upflow side (pushed by an expanding steam bubble created in the core). Once the pump suction is cleared of liquid, a steam path to the break relieves the core steam being generated in the core and relaxes the liquid level depression in the core. In Semiscale, the intact loop seal clears of liquid first, sometimes followed by the broken loop seal for pipe breaks below 10%. For pipe breaks above 10% (intermediate breaks), the broken loop seal clears first. This trend is consistent with the 9 to 1 hydraulic resistance split between the broken loop and intact loop. Also, this seal clearing is related to the amount of core bypass flow between the vessel upper head and downcomer. For example, with larger bypass flows (on the order of 4%) there is enough steam relief in the bypass line such that the broken loop seal never clears because the steam relief path in the intact loop and bypass line is sufficient to relieve core steam generation.

The relationship of break flow and safety injection flow determines the relative severity (core liquid depletion) of a SBLOCA. The break flow is directly related to the break size. During the Semiscale experiments a 2.5, 5, and 10% break were performed and the 5% break produced the most severe core level depletion. The trend of the Semiscale data (break size versus minimum core liquid level) however suggests that 6 to 7% breaks might produce slightly lower vessel liquid levels.

Break location has a large effect on transient severity. SBLOCAs were performed in Semiscale in the hot leg, cold leg, and pump suction with the cold leg and pump suction leading to the maximum mass inventory reduction. Basically, with cold leg breaks and pump suction breaks, fluid at the break remains subcooled longer, thus increasing the break mass flow rate.

The amount of core bypass flow between the downcomer and upper head was found to have a large effect on transient severity. A larger bypass flow relieves more steam during the pump suction seal clearing period and alleviates the core level depression. Semiscale investigated a range of bypass flow between 0.9 and 4%, which is within the range of core bypass flows present in commercial PWRs plants (0.4 to 5%). For the lower bypass flow in Semiscale experiments, the vessel liquid level was depressed to the bottom of the core accompanied by core heater rod temperature excursions. For the high bypass flows in Semiscale experiments, the vessel liquid level depletion associated with pump suction seal formation was less severe resulting in no core heater rod temperature excursions.

The preferred operation of the primary pumps during a SBLOCA was investigated in Semiscale. Pump operation during SBLOCA affects the system depressurization rate and system mass inventory by influencing conditions upstream of the break. Evaluation of both Semiscale and LOFT results indicated that the pumps should be turned off to eliminate pumping of cold leg fluid to the break. With the pumps off a lower density fluid exists at the break, such that the break flow was increased resulting in a reduced mass inventory. Comparison of Semiscale and LOFT results show emergency core coolant (ECC) injection location was important to assessing the problem and further integral testing is required to fully understand this question.

The effectiveness of upper head injection (UHI) on transient response was investigated in Semiscale experiments. The advantages of upper head injection over normal cold leg injection were found to be minimal for a variety of break sizes (2.5, 5, and 10% breaks). The extra coolant mass injected during upper head injection experiments was almost exactly offset by an increased break flow discharge. Even though the system mass inventory was identical for experiments with and without UHI following injection, there was a slightly improved margin for core coolability.

Normal recovery procedures used by commercial PWR plants during a steam generator tube rupture were examined in the Semiscale experiments and found to be adequate to control primary pressure and loop subcooling. The following recovery procedures were found effective in controlling loop pressure and subcooling without significant core uncovery: primary feed and bleed using safety injection (SI) and pressurizer power operated relief valve operation (PORV); secondary feed and steam using auxiliary feed and atmospheric dump valve operation; termination of SI; and pressurizer auxiliary spray. Pressurizer internal heaters were found not to be effective as long as a break in the primary system persisted.

Compounding failures during a SBLOCA were examined in Semiscale testing. During a very small pipe break (0.4%), complete loss of charging and high pressure injection systems (HPIS) was assumed. For this case operator recovery included steam generator feed and steam and primary bleed to reduce the primary pressure to low pressure injection systems (LPIS) setpoints. This operator action was designed to reduce primary pressure to LPIS setpoints before break mass flow had significantly uncovered the core. On a relatively short time basis, accumulator injection mitigated an initial core heatup; however, following emptying of the accumulator tanks, there was a second core temperature excursion. During this second core temperature excursion, core power (electric) was manually tripped with a peak temperature of 945 K. At the time the core power was tripped, the primary pressure had been reduced to near the

LPIS pressure setpoint. During steam generator tube rupture the following compounding failures were examined: stuck open PORV, complete loss of on and offsite power, and main steam line break as an initiating event. For all these compounding failures in the Semiscale experiments, normal recovery techniques were adequate to preclude any core heatup.

During Semiscale experiments, phenomena occurred that are particularly interesting to model development, including the role of natural circulation as a heat rejection mode, core uncovery heat transfer, and accumulator flow oscillation. From these phenomena the following observations are offered: as a SBLOCA progresses the natural circulation mode changes from single-phase to two-phase to reflux condenser mode of heat rejection. All three of these modes of heat rejection are adequate to remove core decay heat. The core thermal behavior during a SBLOCA is governed by the fluid void distribution in the core. As long as a froth level exists at an elevation no core heat ups occur. Oscillations occur in the accumulator injection during a SBLOCA because of condensation effects in the injection location changing the overall accumulator to primary system differential pressure. Although the effect of these accumulator oscillations on core level were pronounced, the oscillatory injection did not detract from adequate cooling of the core.

Further, integral and separate effects Semiscale testing could improve knowledge of SBLOCA phenomena and also calculational capability. Separate effects experiments including parts of the Semiscale facility could include: two-phase pump testing, core interfacial drag, core thermal hydraulics experiments, and steam generator primary tubes interfacial drag-flooding-condensation effects. Integral testing could examine: lower vessel breaks, ultra small breaks with degraded emergency core cooling, and preferred pump operation during a SBLOCA.

CONTENTS

SUMMARY OF SEMISCALE SMALL BREAK LOSS-OF-COOLANT ACCIDENT EXPERIMENTS (1979 TO 1985)

INTRODUCTION

The Semiscale experimental program conducted by EG&G Idaho, Inc., is part of the overall research and development program sponsored by the United States 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.

Small break loss-of-coolant accidents are considered relatively probable during the normal operating life of a commercial PWR. In fact small breaks in the form of steam generator tube rupture, pump seal leaks, and stuck open pressurizer power operated relief valve (PORV) have already occurred. Additional anticipated small breaks include instrumentation lines and small pipe cracks associated with normal or abnormal operation.

The real safety issue associated with small breaks is the possibility of severe vessel liquid voiding before the primary pressure has decreased to safety injection setpoints. If the core liquid level is depressed or depleted to a low enough level, core rod heat up and possible fuel damage may result before safety injection initiates a reflood of the core. In large break LOCAs (>10% breaks),[a] the vessel liquid inventory quickly flashes and core heat up can start early in the transient. However, because of the accompanying rapid depressurization, both accumulator and low pressure injection systems (LPIS) can refill and reflood the vessel before significant core rod heat up occurs. Semiscale has provided an extensive data base on issues associated with small break accident analysis that has increased the understanding of SBLOCA phenomena. In addition, Semiscale has been instrumental in providing an integral data base for code development and assessment.

Over the years various modifications of the Semiscale System have been made and these facilities are referred to as Mod-1, Mod-2A, Mod-2B, Mod-2C, and Mod-3. These systems have all been integral scaled simulations of commercial PWR nuclear generating plants. The Semiscale facility has evolved over the years from primarily a large break simulator (200%-design basis accident) to a facility that examines a wide range of plant transients including small breaks with and without compounding failures. Because the need for small break data was so acute, many of the small breaks performed early in the Semiscale program involved systems that were designed specifically for large break and were simply retro-fitted to simulate small break phenomena. More recently, small break experiments performed by Semiscale involve system designs and measurement systems that were specifically designed for small break simulation. The Semiscale Mods all involved a two loop system including one loop that represented three unaffected loops in a commercial PWR and another loop in which the small break is simulated. The loops included active pumps and steam generators. The loops were connected to a simulated pressure vessel that contained an electrically heated core simulator and various other vessel internals.

This report contains a historical background to Semiscale SBLOCA experiments including a comprehensive list of what SBLOCA experiments have been performed in the Semiscale Mods. A discussion on the facility configurations for the various Semiscale Mods is also given. SBLOCAs have a distinctive signature response, which is discussed for a variety of types of small breaks. Important SBLOCA issues discussed in this topical

a. A 200% break equals a double-ended offset shear of the main coolant piping in one loop of a four-loop PWR. Small pipe breaks are assumed to be centerline tears or cracks in the main coolant piping.

include: accident severity as affected by break location, break size, system configuration, small break response with compounding failures, preferred primary coolant pump operation during a transient with small break symptoms, and relative merits of upper head safety injection. Also discussed in this topical are thermal-hydraulic phenomena associated with small break LOCAs that are important to computer models used to calculate SBLOCA response. Finally, recommendations are given for future data needs in the field of SLBOCA research.

HISTORICAL BACKGROUND/SUMMARY OF SEMISCALE SMALL BREAK EXPERIMENTS

The occurrence of the accident at TMI-2 in March 1979, completely redirected the water reactor research effort throughout the world. Prior to the TMI event, almost the entire focus of water reactor research had been on large break design basis LOCAs that are double-ended offset-shear guillotine breaks of the cold leg pipe. Table 1 summarizes the breadth of SBLOCA that have been performed in the Semiscale Mods. Semiscale had investigated SBLOCAs prior to the TMI accident.[1,2] The first Semiscale SBLOCA was accomplished in the Mod-1 facility in 1976 and was a 6% noncommunicative cold leg break. The Mod-1 system was a LOFT (Loss-of-Fluid Test) scaled facility with a short core [1.68 m (5.5 ft)], short intact loop steam generator (Type 1), and inactive components in the broken loop. Because of these scaling distortions, the Mod-1 facility was not particularly suited to performing SBLOCA experiments. Prior to the TMI-2 accident one additional 10% SBLOCA was performed in the Mod-3 system involving delayed ECC to investigate core uncovery heat transfer.

Following the accident at TMI-2 an extensive number of small break experiments were performed in the Mod-3, Mod-2A, Mod-2B, and Mod-2C systems. The purpose of these experiments were to examine the following topics: preferred main coolant pump operation during SBLOCAs, the effectiveness of upper head injection during SBLOCAs, the role of natural circulation during SBLOCAs, loss-of-onsite/offsite power with a small break, steam generator tube rupture (which is another form of SBLOCA), and, in general, characterize the signature response to a SBLOCA. Many of the latter experiments, especially those involving tube rupture, investigated commonly used recovery procedures during the primary small break. Two recent SBLOCA experiments called S-LH-1 and S-LH-2 was performed in the new Mod-2C facility, which is the state-of-the-art in SBLOCA experimental facilities.

Appendix A contains a summary of the applicability of Semiscale small break experiments for code development and assessment purposes. This appendix matches issues and the most applicable experiment to satisfy the issues. In addition, the adequacy of configuration documentation and data to satisfy the issue is assessed.

oop				
UHI	Steam Generator Operation Drained/Isolated	Pump Operation Intact Loop/Broken Loop	Core Heatup/Peak Rod Temperature (K)	Objective
N/A	Isolated	Pump coasted down	No/initial condition	First small break run in Semiscale. Provided data for standard problem evaluation
N/A	Isolated then blowdown	4400 s throttle intact loop pump to half flow; 6000 s terminate power	Yes/1050	Duplicate the TMI scenario in Semiscale
N/A	Isolated then blowdown	4400 s throttle intact loop pump to half flow; 6000 s terminate power	Yes/1050	Duplicate the TMI scenario in Semiscale
N/A	Isolated	Tripped 1 s after 12.41 MPa	Yes/1160	10% break with delayed ECC
N/A	Drained (broken loop only)	Tripped 1 s after 12.41 MPa	Yes/1070	10% break with delayed ECC and broken loop steam generator secondary blowdown
N/A	Isolated	Tripped at 12.4 MPa (early)	Yes/760	Influence of pump operation on small cold leg breaks (pumps tripped early)
N/A	Isolated	Did not trip	No/initial condition	Baseline for examining influence of pump operation during small cold leg breaks (pumps not tripped)
N/A	Isolated	Tripped at 12.4 MPa (early)	No/initial condition	Influence of pump operation on small hot leg breaks (pumps tripped early)
N/A	Isolated	Did not trip	No/initial condition	Baseline for examining the influence of pump operation during small hot leg breaks (pumps not tripped)
N/A	Isolated	Tripped at 3.08 MPa (delayed)	No/initial condition	Influence of pump operation on small cold leg breaks (pumps trip delayed)
N/A	Auxiliary feed; late feed and steam	Broken loop tripped 30 s after 12.48 MPa/intact loop tripped at 140 s after 12.48 MPa but only coasted down to 10% initial speed	No/initial condition	Small cold leg break with typical boundary conditions expected in a PWR
N/A	Auxiliary feed; late feed and steam	Intact loop tripped 140 s and broken loop tripped at 80 s after system pressure, 12.48 MPa	Yes/800	Same as S-SB-2 except core power augmented to makeup heat loss
N/A	Auxiliary feed; broken loop isolated then drained	Intact loop pump tripped at 0 s, broken loop pump tripped at 1.5 s	No/initial condition	LOFT L3-1 counterpart (2-1/2% cold leg break with system modifications)
N/A	Auxiliary feed; broken loop isolated then drained	Intact loop pump tripped at 0 s, broken loop pump tripped at 1.5 s	Yes/780	LOFT L3-1 counterpart with augmented core power to makeup heat loss
N/A	Isolated	Tripped 1 s after 12.41 MPa	Yes/660	Baseline for 10% UHI test
8.5	Isolated	Tripped 1 s after 12.41 MPa	Yes/630	10% UHI test

Table 1. Summary of Semiscale small break experiments

				ECC Activation Pressure Intact Loop/Broken L (MPa)		
Test Identifier[a]	Configuration	Break Size/Location	Heat Loss Makeup Technique	HPIS	AIS	LPIS
S-02-6[1,2]	Mod-1	6%/cold leg noncommunicative *single-ended*	None	N/A/N/A	4.14/4.14	1.03/1.03
S-TMI-3C[3]	Mod 3	0.2%/PORV	Augmented core power	Intermittent/intermittent	N/A/N/A	N/A/N/A
S-TMI-3I[3]	Mod-3	0.2%/PORV	Augmented core power	Intermittent/intermittent	N/A/N/A	N/A/N/A
S-07-10[4]	Mod-3	10%/cold leg	None	1.45/N/A	1.45/N/A	1.73/N/A
S-07-10D[5]	Mod-3	10%/cold leg	None	1.60/N/A	1.60/N/A	2.10/N/A
S-SB-P1[6,7]	Mod-3	2.5%/cold leg	None	13.2/13.2	N/A/N/A	N/A/N/A
S-SB-P2[6,7]	Mod-3	2.5%/cold leg	None	13.2/13.2	N/A/N/A	N/A/N/A
S-SB-P3[6,8,9]	Mod-3	2.5%/hot leg	None	13.2/13.2	N/A/N/A	N/A/N/A
S-SB-P4[6,8,9]	Mod-3	2.5%/hot leg	None	13.2/13.2	N/A/N/A	N/A/N/A
S-SB-P7[6,7]	Mod-3	2.5%/cold leg	None	13.2/13.2	N/A/N/A	N/A/N/A
S-SB-2[10,11]	Mod-3	2.5%/cold leg	None	12.48/12.48	4.23/4.25	0.89/0.89
S-SB-2A[12,10,13]	Mod-3	2.5%/cold leg	Augmented core power	12.48/12.48	4.54/4.16	0.89/0.89
S-SB-4[14,15]	Mod-3	2.5%/cold leg	None	12.4/N/A	3.8/N/A	0.88/N/A
S-SB-4A[14,15]	Mod-3	2.5%/cold leg	Augmented core power	12.4/N/A	4.3/N/A	0.88/N/A
S-UT-1[16,17]	Mod-2A	10%/cold leg	None	13.3/N/A	2.77/N/A	N/A/N/A
S-UT-2[16,18,19]	Mod-2A	10%/cold leg	None	13.5/N/A	2.98/N/A	1.15/N/A

Table 1. (continued)

Test Identifier[a]	Configuration	Break Size/Location	Heat Loss Makeup Technique	ECC Activation Pressure Intact Loop/Broken Loop (MPa) HPIS	AIS	LPIS
S-SG-4[40,47]	Mod-2B	1-tube/broken loop cold side	External heaters; core power augmented 20 kW	12.5/12.5	Not used	Not used
S-SG-5[41,47]	Mod-2B	5-tube/broken loop hot side	External heaters; core power augmented 20 kW	12.5/12.5	Not used	Not used
S-SG-6[42,47]	Mod-2B	5-tube/broken loop hot side	External heaters; core power augmented 20 kW	12.5/12.5	Not used	Not used
S-SG-7[43,47]	Mod-2B	5-tube/broken loop hot side	External heaters; core power augmented 20 kW	None	Not used	Not used
S-SG-8[44,47]	Mod-2B	1-tube/broken loop cold side	External heaters; core power augmented 20 kW	12.5/12.5	Not used	Not used
S-SG-9[45,47]	Mod-2B	1-tube/broken loop cold side	External heaters; core power augmented 20 kW	HPIS initiated on low secondary pressure (4.86 MPa)	Not used	Not used
S-LH-1[46]	Mod-2C	5% cold leg	External heaters; no core augmentation	25 s after 12.5 MPa achieved	4.2 MPa	Not used
S-LH-2[46]	Mod-2C	5% cold leg	External heaters; no core augmentation	25 s after 12.5 MPa achieved	4.2 MPa	Not used

a. Superscripts refer to reference number.

UHI	Steam Generator Operation Drained/Isolated	Pump Operation Intact Loop/Broken Loop	Core Heatup/Peak Rod Temperature (K)	Objective
N/A	Only relief valve operation for 600 s; feed and steam during recovery	Delayed; tripped at 602 s	No	Tube rupture with delayed pump trip
N/A	Early SG feed and steam; relief valve latched open 80 s	Tripped at 24.5 s on SI signal	No	First tube rupture on hot side of steam generator
N/A	Stuck open relief valve on broken loop; feed and steam on intact loop	Tripped at 38.5 s on SI signal	No	5-tube rupture with compounding failure of stuck open broken loop SG relief valve
N/A	Only relief valve for 600 s; intact loop feed and steam during recovery	Tripped at 37 s on SI signal and loss of power	No	5-tube rupture with compounding failure of complete loss of onsite and offsite power, no SI
N/A	Only relief valve for 600 s; later feed and steam	Tripped at 154.5 s on SI signal	No	1-tube rupture with compounding time of stuck open PORV
N/A	Only relief valve for 600 s; IL feed and steam during recovery	Tripped at 17 s on low secondary pressure	No	Main steamline break compounded by tube rupture
N/A	Isolated; steam relief allowed	Tripped at 2 s after 12.6 MPa primary pressure	Yes	Repeat of S-UT-8 hydraulic boundary combinations; 0.9% upper head to downcomer core bypass flow
N/A	Isolated; steam relief allowed	Tripped at 2 s after 12.6 MPa primary pressure	Yes	Repeat of S-LH-1; examined effect of core bypass flow; 3% core bypass flow allowed

oop/Broken Loop LPIS	UHI	Steam Generator Operation Drained/Isolated	Pump Operation Intact Loop/Broken Loop	Core Heatup/Peak Rod Temperature (K)	Objective
N/A/N/A	N/A	Isolated	Tripped 3.4 s after 12.8 MPa	No/initial condition	Baseline for loop heater performance
0.98/0.98	N/A	Isolated	Tripped 3.4 s after 12.8 MPa	No/initial condition	Baseline for 2-1/2% UHI test
0.98/0.98	8.7	Isolated	Tripped 3.4 s after 12.8 MPa	No/initial condition	2-1/2% UHI test
0.98/N/A	N/A	Isolated	Tripped 3.4 s after 12.8 MPa	Yes/660	Baseline for 5% UHI test
0.98/0.98	8.7	Isolated	Tripped 3.4 s after 12.8 MPa	Yes/570	5% UHI test; 4% core bypass flow
N/A/N/A	N/A	Isolated	Tripped 3.4 s after 12.8 MPa	Yes/825	Similar to Test S-UT-6 with lower upper head bypass flow(1.1%) and other upper head modifications
N/A/N/A	N/A	Isolated then feed and bleed	Pumps off entire test	Yes/950	Ultra small break without HPIS
N/A/N/A	N/A	Isolated then feed and bleed	Pump off entire test	No/initial condition	Ultra small break with HPIS
None	N/A	Secondary boil-off via relief valves	Tripped 2 s after transient initiation	Yes/825	Initiated by loss-of-offsite power; no ECC PORV primary bleed
None	N/A	Secondary boil-off later drained	Tripped 2 s after transient initiation	Yes/830	Initiated by loss-of-offsite power; with system recovery
1.2/1.2	N/A	Isolated primary core	Tripped 30 s after transient initiation	Yes/561	5% SBLOCA pump suction break
Not used	N/A	Secondary boil-off drained	Tripped at 0 s	Yes/1145	First station blackout test; no ECC
Not used	N/A	Boil-off/drained	Tripped at 0 s off at 60 s	Yes/1144	Station blackout; system heated steam in vessel upper plenum thermocouple housing caused blowdown
Not used	N/A	Only a relief valve operation for 600 s; unaffected loop feed and steam on level	Tripped at 96 s on SI signal	No	First steam generator tube rupture; limited recovery
Not used	N/A	Only relief valve operation for 600 s; unaffected loop feed and steam on level	Tripped at 22 s on SI signal	No	5-tube rupture with PORV operation
Not used	N/A	Only relief valve operation for 600 s; unaffected loop feed and steam on level	Tripped at 20 s on SI signal	No	10-tube rupture with complicated recovery involving pressurizer spray

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Table 1. (continued)

				ECC Activation Pressure Intact L (MPa)	
Test Identifier[a]	Configuration	Break Size/Location	Heat Loss Makeup Technique	HPIS	AIS
S-UT-3[20]	Mod-2A	2.5%/cold leg	None	12.8/N/A	4.2/4.2
S-UT-4[21,22]	Mod-2A	2.5%/cold leg	External loop heaters	12.8/N/A	2.8/2.8
S-UT-5[16,21,23]	Mod-2A	2.5%/cold leg	External loop heaters	12.9/N/A	2.95/2.9
S-UT-6[24,25]	Mod-2A	5%/cold leg	External loop heaters	12.6/N/A	2.8/2.8
S-UT-7[16,24,26]	Mod-2A	5%/cold leg	External loop heaters	12.5/12.6	2.85/2.9
S-UT-8[27,28]	Mod-2B	5%/cold leg	External loop heaters	7.6/N/A	4.14/4.0
S-NC-8B[29,30,31]	Mod-2A	0.4%/cold leg	External loop heaters	N/A/N/A	4.14/4.14
S-NC-9[29,31,32]	Mod-2A	0.4%/cold leg	External loop heaters	12.4/12.4	4.14/4.14
S-PL-2[33]	Mod-2B	Pressurizer PORV	External heaters; augmented core power 75 kW	None	None
S-PL-3[34]	Mod-2B	Pressurizer PORV	External heaters; augmented core power 55 kW	3.5/3.5	None
S-PL-4[35]	Mod-2B	5% pump suction	None	7.1/7.4	4.2/4.2
S-TR-1[36]	Mod-3	Pressurizer PORV	Core power augmented 80 kW	None	4.1/4.1 (not used)
S-TR-2[36]	Mod-3	Pressurizer PORV	Core power augmented 80 kW; reduced in 10 kW increments as vessel liquid level decreased	8.7 intact loop only	Not used
S-SG-1[37,47]	Mod-2B	1-tube/broken loop cold side	External heaters; core power augmented 20 kW	12.5/12.5	Not used
S-SG-2[38,47]	Mod-2B	5-tube/broken loop cold side	External heaters; core power augmented 20 kW	12.5/12.5	Not used
S-SG-3[39,47]	Mod-2B	10-tube/broken loop cold side	External heaters; core power augmented 20 kW	12.5/12.5	Not used

FACILITY DESCRIPTION

Five different versions of Semiscale were used to perform small break loss-of-coolant experiments including: Mod-1, Mod-3, Mod-2A, Mod-2B, and Mod-2C. Figures 1 through 5 give schematics of these five mods, which represent a steady improvement in control and measurement of boundary conditions, scaling, and measurement of phenomena. This section briefly describes the important characteristics of each of the Semiscale Mods. All of the Mods were basically two-loop, including an intact loop that simulated three unaffected loops of a four loop PWR and a broken loop that simulates the loop in which the small break loss-of-coolant accident occurred. Major differences between the Mods, are summarized in Table 2. A brief description of the various Mods follows.

MOD-1

The MOD-1 was designed to simulate the Loss-of-Fluid-Test (LOFT) facility that incorporated a 1.68 m (5.5 ft) core and inactive components in the broken loop. This modification was designed to investigate design basis accidents (200% double-ended offset shear breaks of the primary piping). Elevation scaling was practically nonexistent however, an attempt was made to represent correct

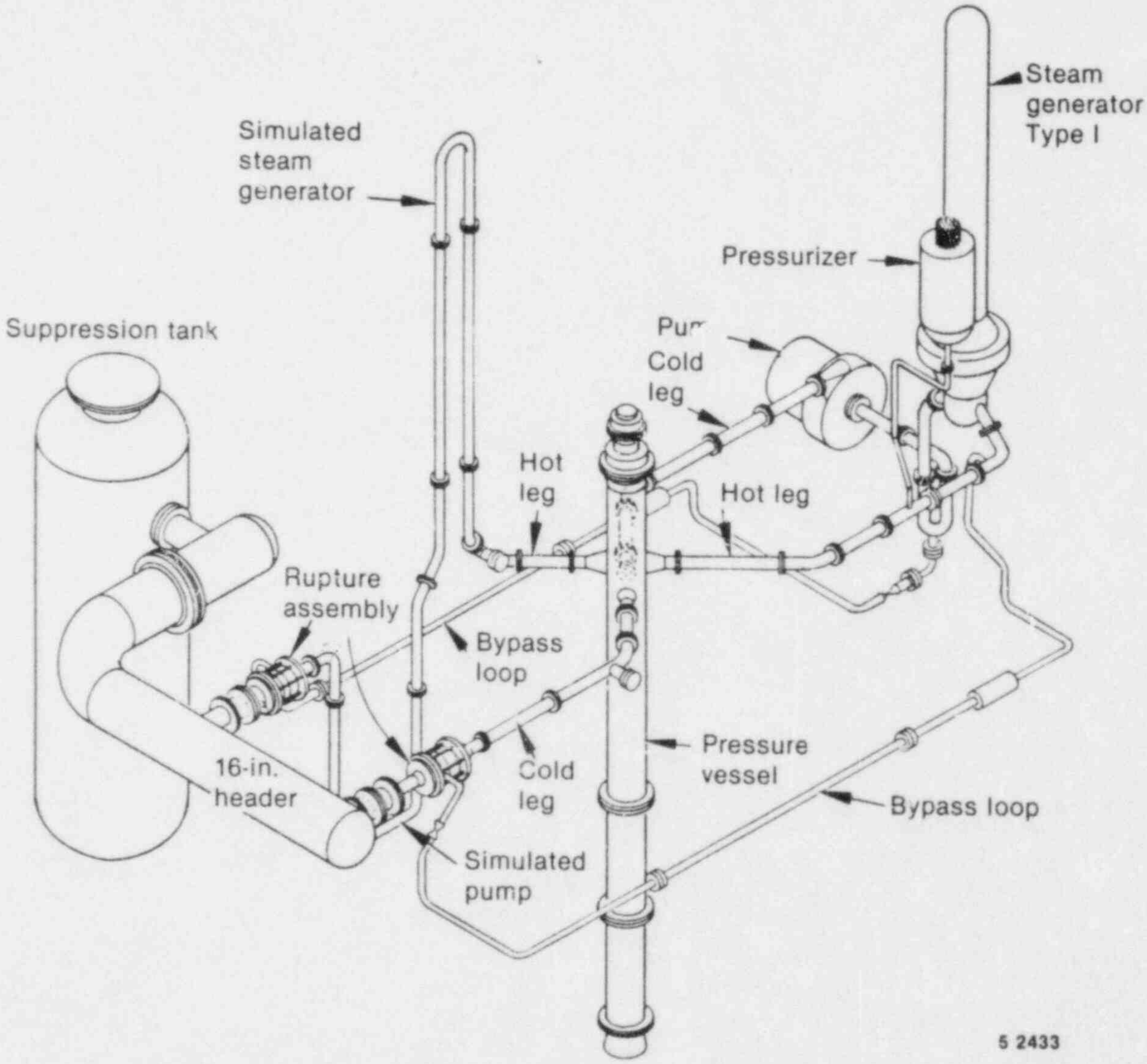

Figure 1. Semiscale Mod-1 system for cold leg break configuration.

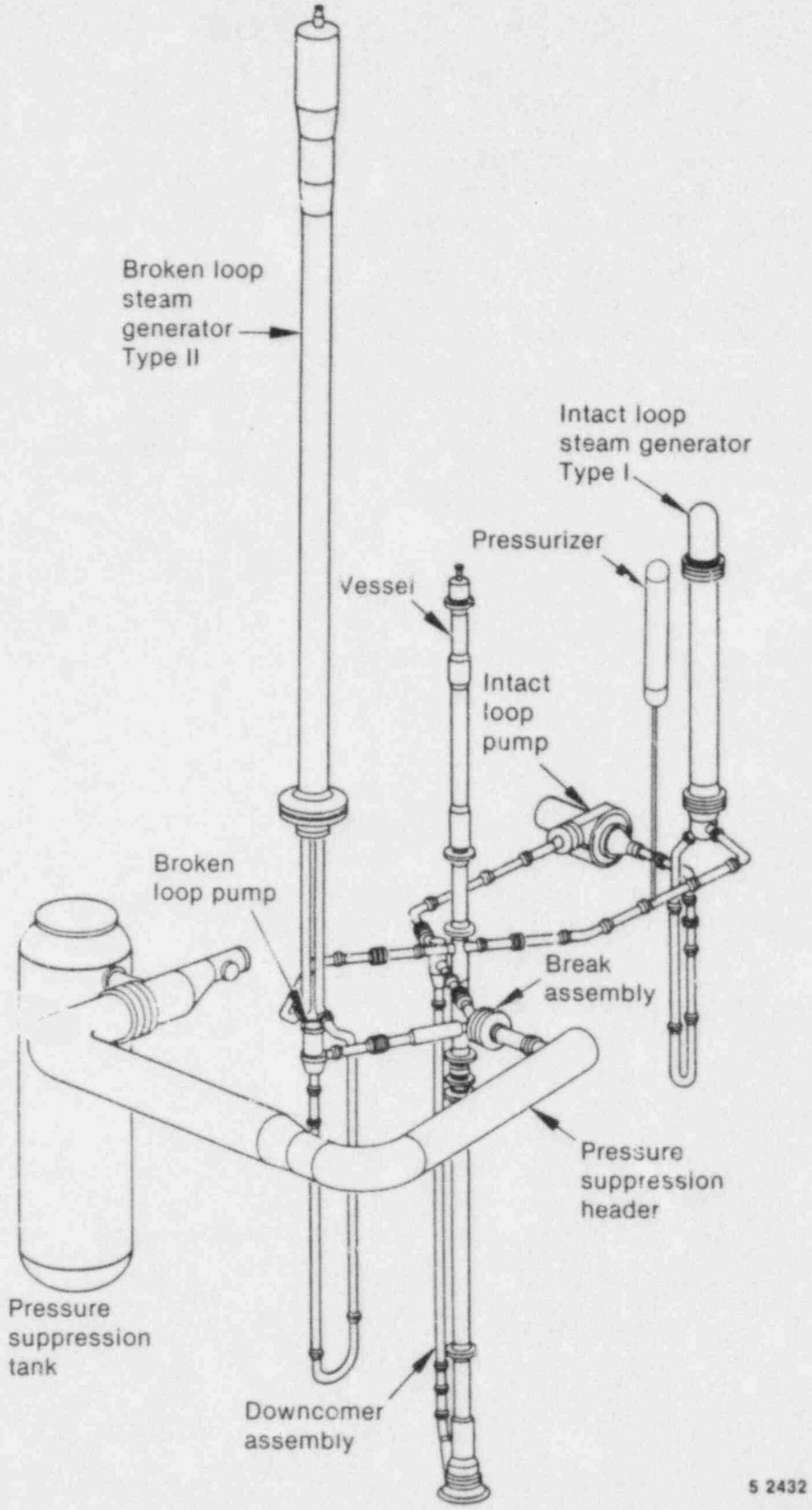

Figure 2. Semiscale Mod-3 system for cold leg break configuration.

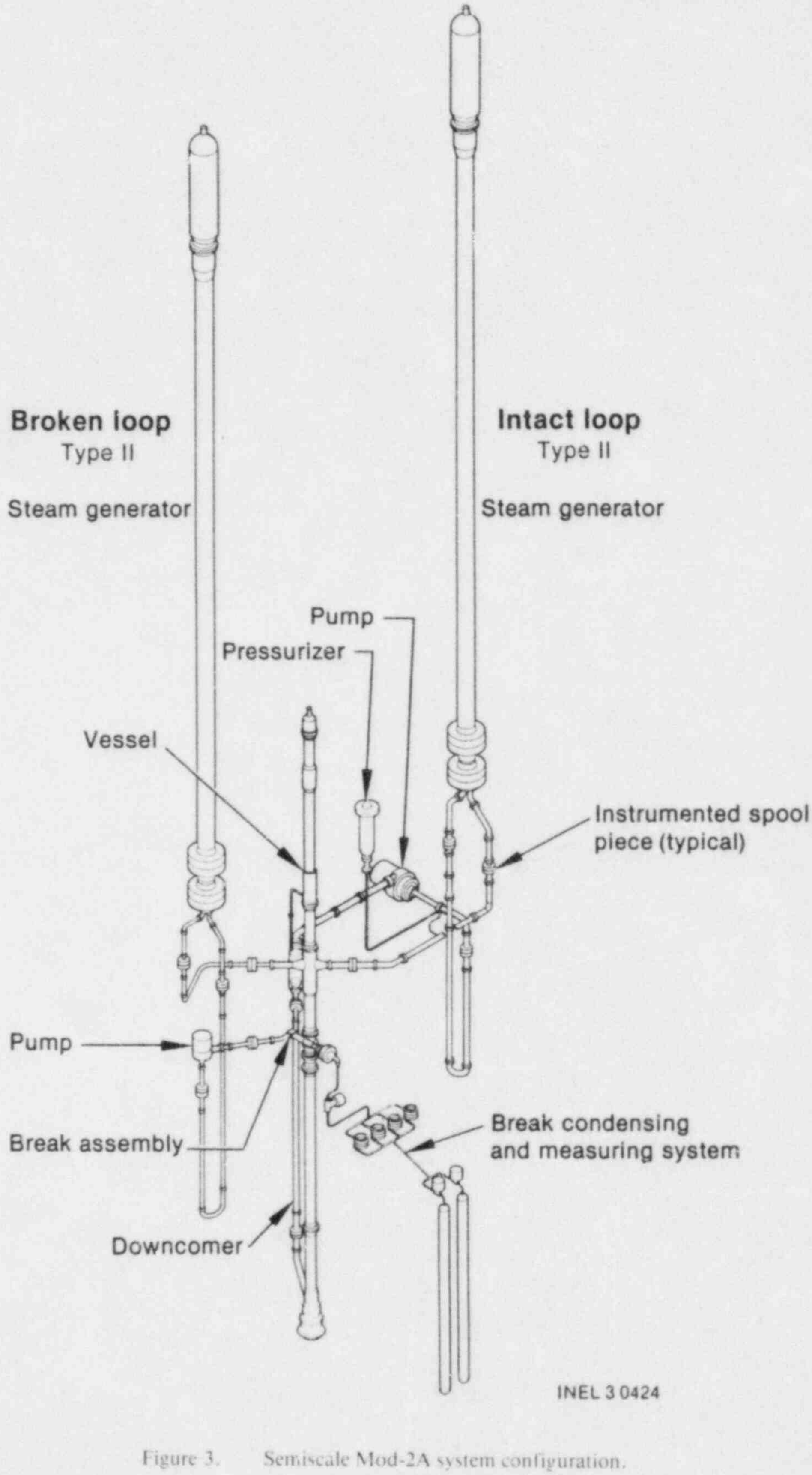

Figure 3. Semiscale Mod-2A system configuration.

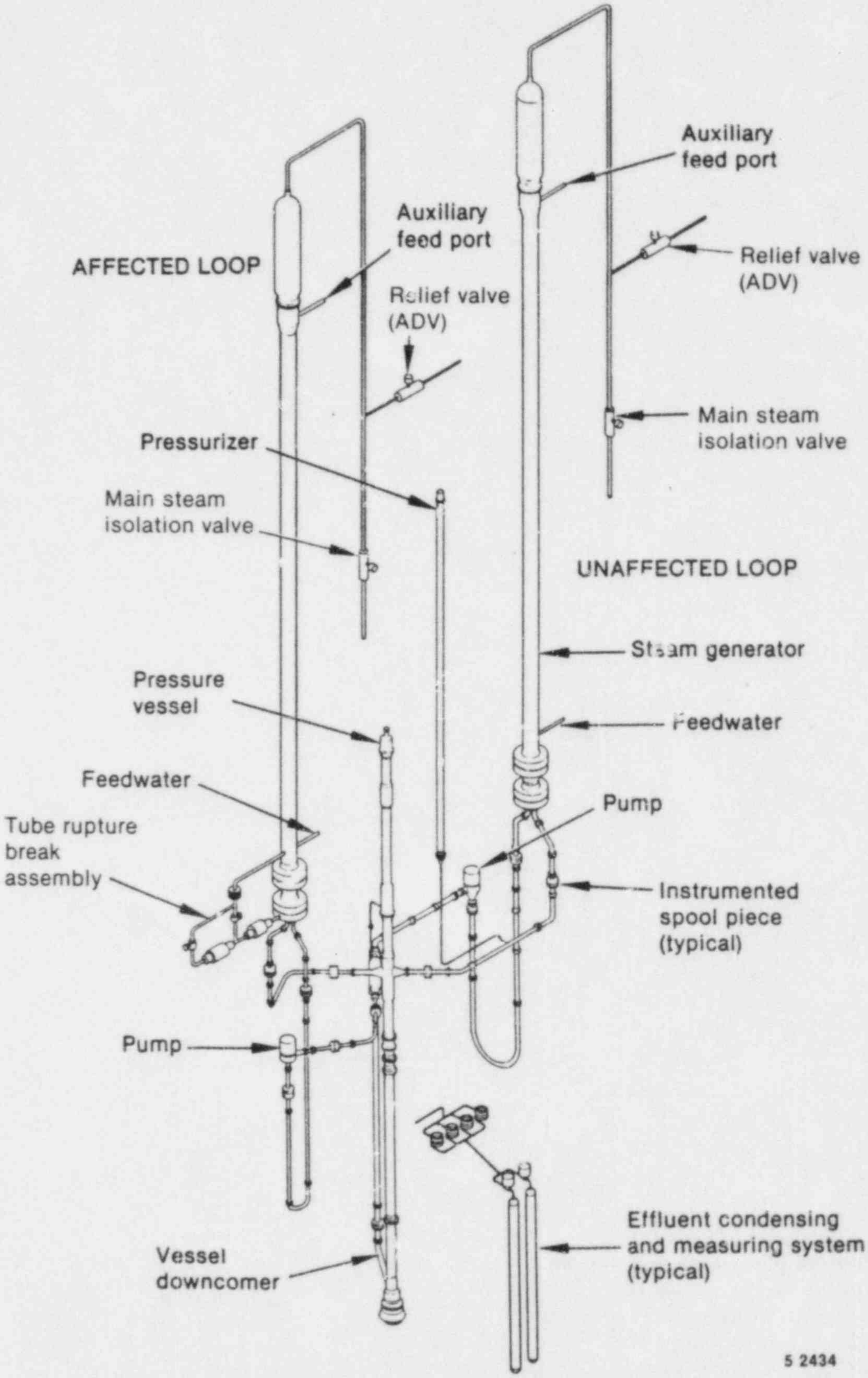

Figure 4. Semiscale Mod-2B system configuration.

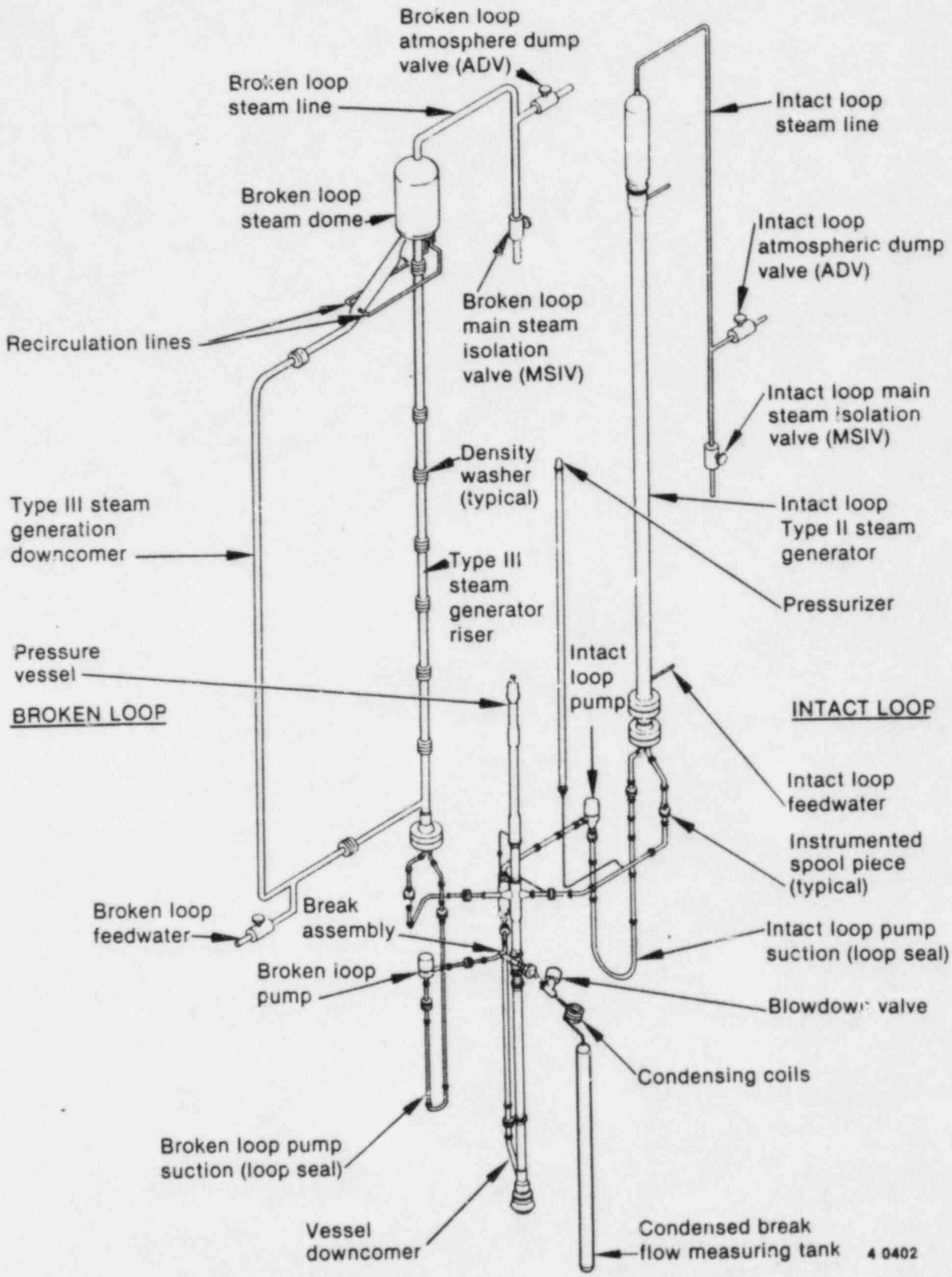

Figure 5. Semiscale Mod-2C system configuration.

Table 2. Comparison of Semiscale Mods

Facility Mod	Intact Loop Steam Generator	Broken Loop Steam Generator	Intact Loop Pump	Broken Loop Pump	Electrically Heated Core	
Mod-1	Type I; 54 to 5.13 m (16.85 ft tubes) (models LOFT facility)	None; resistance simulator	Lawrence pump	Orifice; locked rotor resistance only simulated	1.67 m (5.5 ft) active length core; rods extended out top of vessel; maximum power 1.6 MW; 40 rods with PWR pitch and size	Inta bro sch
Mod-3	Type I	Type II; 2 tubes 1:1 elevation scaling	Lawrence pump	High-speed vertical; bottom suction, side discharge locked rotor resistance nozzle at discharge	3.66 m (12 ft) length core; rods extended out top of vessel; maximum power 2.0 MW; 25 rods with PWR pitch and size	Inta bro sch
Mod-2A	Type II; 6 tubes 1:1 elevation scaling	Type II; 2 tubes 1:1 elevation scaling	Lawrence pump (removed for natural circulation tests)	High-speed vertical; bottom suction side discharge locked rotor resistance nozzle at discharge	3.66 m (12 ft) length core; rods extended out top of vessel; maximum power 2.0 MW; 25 rods with PWR pitch and size	Inta bro sch
Mod-2B	Type II; 6 tubes 1:1 elevation scaling; scaled relief valves on secondaries	Type II; 2 tubes 1:1 elevation scaling; scaled relief valves on secondary	High-speed vertical; bottom suction; side discharge; locked rotor resistance nozzle at discharge	Same as Mod-2A	Same as Mod-2A	Inta sch 3 in loop sch pipi
Mod-2C	Type II; 6 tubes; 1:1 elevation scaling	Type III; 2 tubes; external downcomer; elaborate separator; 1:1 scaling; extensive instrumentation; improved secondary volume scaling	Same as Mod-2B	Same as Mod-2B	Same as Mod-2B	Inta sch 1-1/ less

Piping	Heat Loss Makeup	Downcomer	Vessel Upper Head	Original Purpose of Facility
3 in. sch 160; 1-1/2 in. rbon steel	None	Internal annulus in vessel; no in-core densitometers	None	Large break LOCA
3 in. sch 160; 1-1/2 in. rbon steel	Augumented core power	External pipe with core/downcomer densitometers	Contains ECC injection port; simulated guide tube; two simulated support columns; support plate separates upper head from upper plenum	Large break LOCA
3 in. sch 160; 1-1/2 in. rbon steel	Augmented core power and external band heaters on loop piping only (last natural circulation test used vessel heaters also)	External pipe with core/downcomer densitometers	Same as Mod-3 variation in bypass line resistance using valve; upper plenum/upper head resistance varied	SBLOCA; natural circulation
has a 2-1/2 in. ions; mostly 0 pipes; broken a 1-1/2 in. ing. Most nless steel	Heater tape over both piping vessel; no band heaters on suctions	Same as Mod-2A	Basically same as Mod-3 bypass line used orifice; upper plenum/upper head resistance changed by plugging support columns	SBLOCA; power loss; anticipated transients without scram; steam generator tube rupture.
all 2-1/2 in. ken loop 160 all stain-	Same as Mod-2B	Same as Mod-2B	Same as Mod-2B; bypass line can be varied by changing orifice	Steam and feed line break and further SBLOCA analysis

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power to volume ratio and relative volume of components. Heat-loss make-up was not available nor needed because large break accidents happened fast [about 30 s depressurization from 15.5 MPa to 0.13 MPa (2250 to 20 psia)]. Only one noncommunicative (single-ended) 6% small break in the cold leg was performed in this facility.

MOD-3

The Mod-3 represented a vast improvement in elevation scaling and measurement capability. The system changed from one active loop (intact loop) used for Mod-1 to a true two loop system by inclusion of a new full length active steam generator and a new high specific speed active pump in the broken loop. The facility was designed as a blowdown-refill-reflood facility to examine large break phenomena. Prior to the accident at TMI-2 the facility had only been used for that purpose. The major improvement in measurement capability was made possible by the use of an external downcomer to the vessel that allowed use of *in-core* gamma densitometers to measure local fluid conditions. Both during and after the accident at TMI 2 the Mod-3 facility was used to answer specific small break safety questions such as whether to leave the pumps on or turn them off during a diagnosed small break. The Mod-3 facility had no external heat loss makeup system and used augmented core power for some of the experiments to makeup for the large heat losses from the atypically large surface area. The inclusion of a vessel upper head with a simulated guide tube and simulated support columns allowed simulation of small break LOCA experiments with upper head injection.

MOD-2A

The Mod-2A facility was the first Semiscale Mod to be designed to run specifically small break experiments. The inclusion of the Type II-full-length steam generator in the intact loop (Mod-3 used a short 4 m (13.1 ft) Type I steam generator in the intact loop) made possible almost complete 1:1 scaling of elevation that is critical to natural circulation type phenomena. For the first time, external band heaters were used on the loop piping to offset heat loss that is critical in a small-scale high pressure facility such as Semiscale (the heat loss is on the order of the core decay heat for much of the transient). A bypass line between the vessel upper head and downcomer inlet annulus contained an adjustable valve to set the core bypass flow rate.

MOD-2B

Improvements to the heat loss makeup technique and inclusion of a new high speed vertically oriented intact loop pump characterized the Mod-2B. The heat loss makeup technique changed from band heaters to heater tape with a fairly uniform coverage of the tape (this was not possible with the band heaters). The upper head to upper plenum flow path was changed by plugging instrument holes and drain holes that had been used on the support column during Mod-2A testing. The bypass line between the vessel upperhead to downcomer inlet annulus used fixed orifices to set a desired bypass flow.

MOD-2C

The MOD-2C represents the current state-of-the-art facility for small break and anticipated transient testing. Mod-2C includes a new Type III broken loop steam generator including correct 1:1 elevation scaling, an external downcomer and correctly volume scaled riser secondary volume. Using an external downcomer allows better measurement of steam generator secondary hydraulic conditions during transients.

As these various Mods evolved, there was a steady improvement in measurement techniques. Specifically, the early Mods used drag disks/screens to measure break flow; whereas the newer Mods use condensing systems and catch tanks. Improvements have been made in control and measurement of high pressure injection fluid into the system. As the Mods became more sophisticated, scaled relief valves were added to the steam generator secondaries also with condensing systems and catch tanks. The basic instrumentation in Semiscale is common to all the Mods including pressure cells, differential pressure cells to determine liquid level, loop and steam generator fluid and metal thermocouples, core heater rod cladding thermocouples, and x and γ-ray densitometers, turbine meters and drag disks to measure flow (at least single-phase initial conditions).

For nearly all the small break experiments, transients were initiated from full power operation with pumps running, core power on, the loop full of water and pressurized to 15.5 MPa (2250 psia) with pressure controlled by using a steam bubble in the pressurizer. Typical PWR core differential temperature were used [35 to 39 K (63 to 70°F)] with a nominal hot leg temperature of [595 K (610°F)]. Transients were initiated in several ways. Mod-1 used rupture disks and scaled blowdown nozzles. All other Mods used quick opening blowdown valves and blowdown nozzles. The breaks were mostly considered centerline pipe breaks as the nozzle was concentrically centered. Table 1 lists variations in this. The references used in Table 1 gives further details of these MODs and also lists the Mod associated with a given experiment.

SUMMARY OF RESULTS FROM SEMISCALE SMALL BREAK LOCA EXPERIMENTS (1979 to 1985)

This section summarizes important results condensed from all the small break LOCA experiments run in Semiscale. First, the signature responses for a variety of types of small breaks (centerline pipe break, PORV leak, or steam generator tube rupture) are discussed. Next, there is a discussion on small break issues that have been addressed in the Semiscale testing including: severity of small break accident relative to break size and break location; severity of accident relative to upper head to downcomer bypass flow; preferred operation of the primary coolant pumps during a small break; preferred use of upper head injection; and preferred recovery procedures during a small break with and without compounding failures. Finally, interesting phenomena that has been identified during the Semiscale small break experiments are discussed including: core uncovery heat transfer; natural circulation phenomena, and accumulator chattering phenomena.

General Signature Response to a Small Break Loss-of-Coolant Accident

The general signature response for three types of small break LOCA are discussed in this section. The three types of small break are pipe breaks, PORV stuck open (TMI type), and steam generator tube rupture.

Pipe Break. A great variety of centerline pipe break experiments were performed in the Semiscale system covering a large spectrum of break size, location, and operation scenario. However the system signature response was similar for all these experiments. The response for a typical 2.5%[21,22] small break and a 0.4%[30] small break is used for discussion proposes.

Following the initiation of a break, the primary system depressurization is continuous and represented by several definite inflection points in depressurization rate. Figure 6 shows the primary pressure response for a 2-1/2% centerline small pipe break. On an overall basis the major inflection points in depressurization are caused by achievement of saturation conditions for the loop fluid, pump suction liquid seal clearing and break uncovery, and introduction of accumulator flow. The causes of each of these inflection points are discussed below.

During a small break, as the primary system loses mass out the break, the loop fluid changes from subcooled conditions to saturation conditions. This change causes a large change in the loop depressurization rate because of the start of flashing whenever saturation conditions are achieved. Figure 7 shows the system depressurization for a 0.4% SBLOCA[a] (the 0.4% break is a very small break that accentuates the effect of reaching saturation conditions on depressurization). The first major inflection in pressure was caused by the reduction in core power associated with core scram to the ANS decay curve. The temperature rise across the core (due to full core power) was suddenly reduced upon scram resulting in a decrease in density of the subcooled loop fluid. The main result was a rapid increase in depressurization corresponding to the reduction in expansion of the subcooled fluid. Primary pressure continued to decrease rapidly as subcooled fluid flowed out the break until about 130 s, at which time fluid saturation conditions were achieved in the vessel upper plenum and hot leg. The depressurization rate decreased at this point because flashing of hot leg fluid caused the available fluid to occupy more volume, thus retarding the depressurization rate.

The next inflection point in depressurization rate occurs when the vessel liquid level reaches the hot leg. There is a temporary increase in depressurization rate due to the two-phase natural circulation flow rate increase and resulting increased condensation of steam in the steam generator. Natural circulation phenomena associated with small breaks will be discussed in a later section. This temporary increase in depressurization rate due to a two-phase flow increase is not sustained because the system pressure soon reaches the saturation pressure corresponding to the cold leg accompanied by additional flashing of liquid that again retards the

a. For the experiment shown in Figure 7 core power was scrammed at 105 s on a low pressure trip and HPIS was started about 120 s. The steam generators were isolated by 115 s and pumps were off at blowdown initiation.

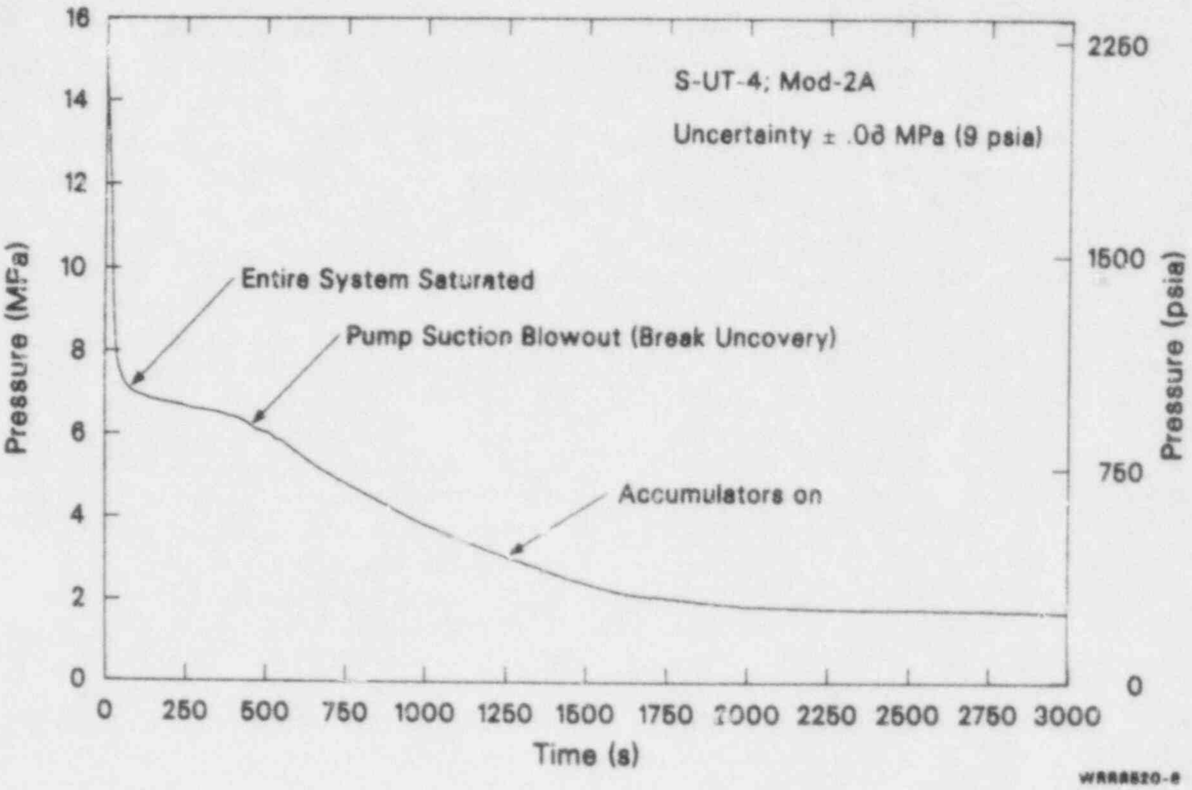


Figure 6. Typical small break loss-of-coolant accident pressure response (2-112% break).

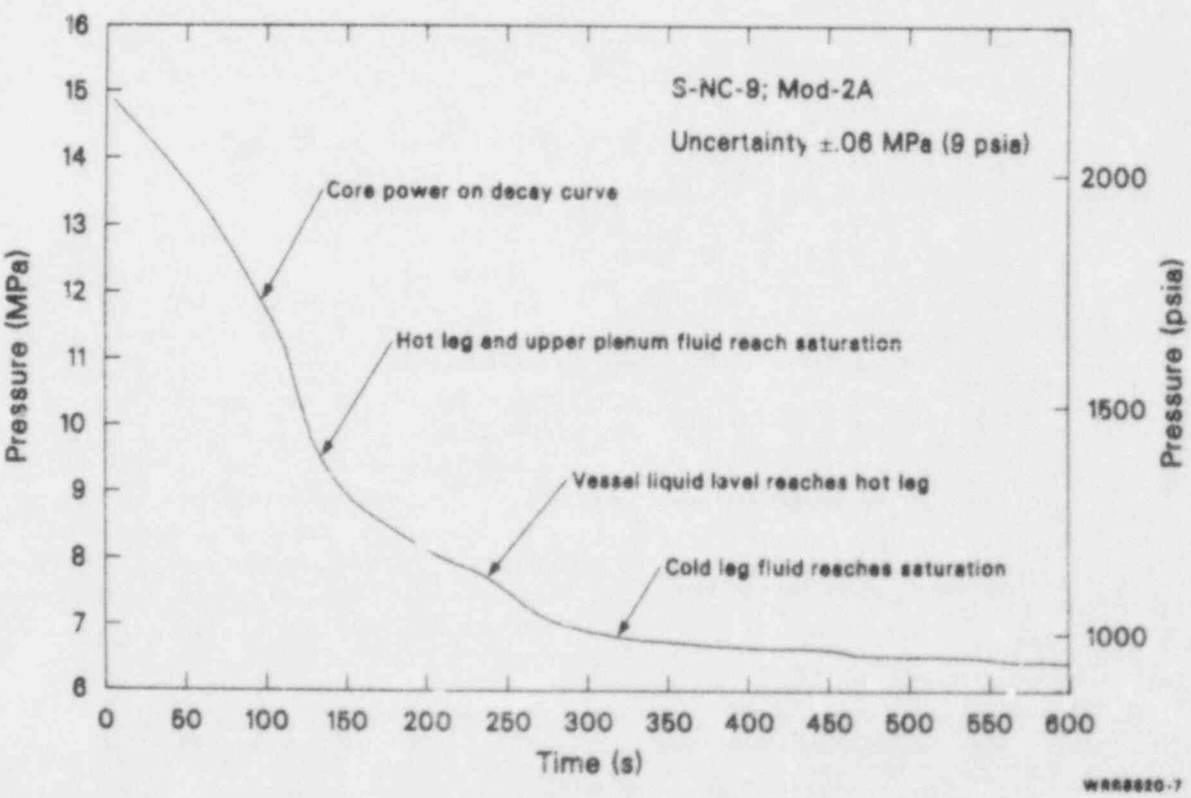


Figure 7. Small break pressure response early in time (0.4% break).

depressurization rate. Figure 8 compares the saturation temperature (based on primary pressure) to both the hot leg and cold leg fluid temperatures showing that the timing of saturation conditions in the various parts of the loop correspond with the major inflection points in pressure.

Referring back to Figure 6, the break uncovery can have a large effect on system depressurization rate during a small break LOCA. Break uncovery is related to clearing of the pump suction seals that is discussed later. Figure 6 (typical 2-1/2% small break experiment with normal ECC parameters) shows a large increase in depressurization corresponding to break uncovery. During a small pipe break experiment in Semiscale a centerline pipe tear or rupture was assumed. Thus, the centerline of the break nozzle was placed even with the centerline of the broken loop piping. Basically, the fluid in the broken loop behaves in a stratified manner during small breaks.[a] The fluid in the broken loop looks like a pool of liquid with steam on top. When the level of the pool of broken loop fluid reaches the level of the break nozzle, the break flow becomes almost single-phase steam that greatly enhances the depressurization rate. Visual observations also show that liquid phase can be entrained into the nozzle even after the stratified steam/water interface has passed the centerline of the nozzle.

Figure 6 also shows the effects of accumulator injection on system depressurization rate. Injection of accumulator fluid [beginning when the system pressure was 4.2 MPa (609 psia)] caused a vessel reflood of hot vessel structures and the core. The resultant steam generation tended to retard the depressurization process. The depressurization rate following termination of accumulator injection was fairly slow and steady governed by an energy and mass balance involving, break flow, core decay heat, HPIS, and primary to secondary heat transfer.

The slow steady depressurization period following accumulator injection is characterized as a period of boil-off of fluid in the core, two-phase flow out the break, and a counter balanced inflow of fluid from the HPIS systems. As long as HPIS flow is greater than break flow during this period, core

a. This observation is supported by visual data. The visual observations were made with an optical probe and video camera. The view was on the centerline of the break.

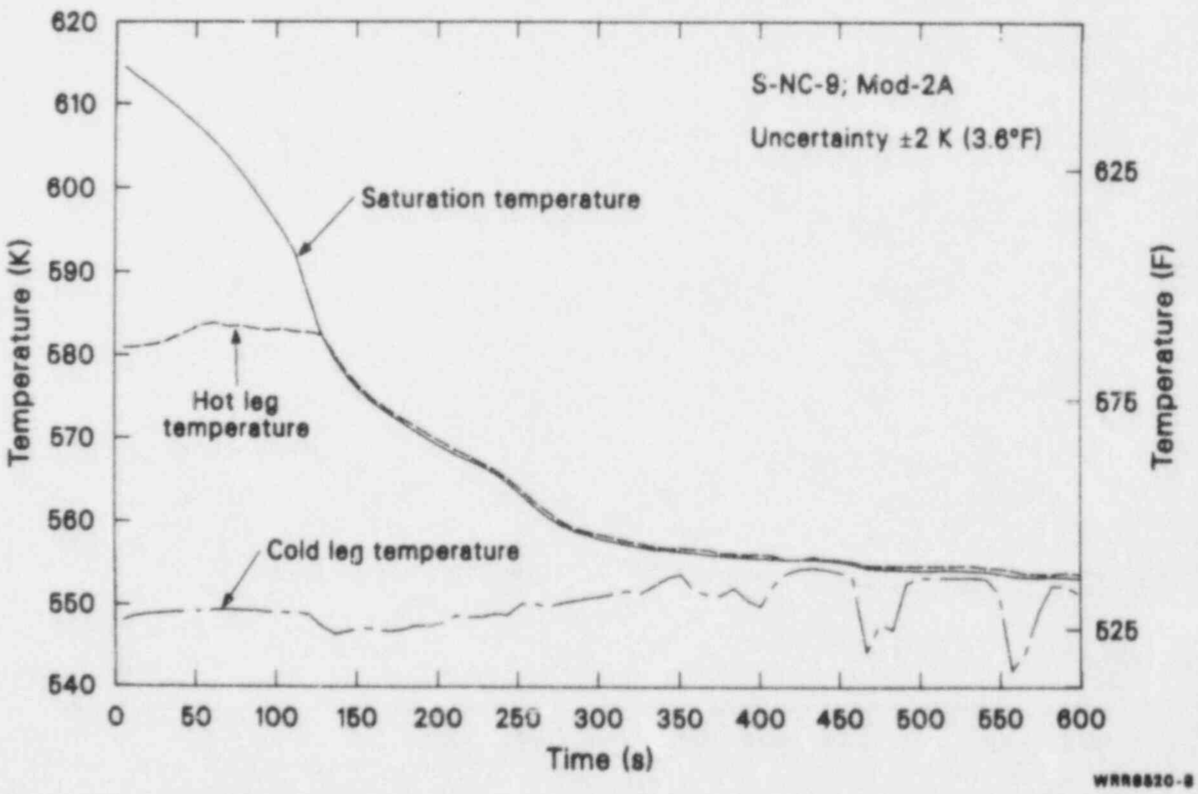


Figure 8. Comparison of hot leg and cold leg fluid temperature with saturation temperature (0.4% break).

uncovery and rod heatup did not occur prior to system pressure reaching LPIS set points. However in the absence of HPIS flow core rod uncovery and heatup can occur before system pressure reaches LPIS setpoints. Maintaining HPIS flow is critical to vessel liquid inventories during small break transients. For the small break example shown in Figure 6 the HPIS rate about equaled the break flowrate at about 1300 s, as shown in Figure 9. HPIS flow combined with accumulator flow at about the same time resulted in a general filling of the system, as shown in Figure 10. Also shown in Figure 10 is the general filling trend of the vessel and downcomer.

One of the most dramatic events occurring during small pipe break transients is the formation of liquid seals in the pump suctions of both loops. This seal formation and eventual clearing has a large effect on vessel inventory and core rod heatup. Accompanying the early system depressurization there is subcooled fluid everywhere in the system except the pressurizer. Early in time the pressurizer fluid flashes and the inventory in the pressurizer depletes. At the time the pressurizer fluid depletes, fluid in the hot leg and primary tubes of the steam generator becomes saturated and flashing also occurs there. Therefore, as depressurization continues the depletion of liquid in the system is stratified with a top down voiding. Because of the geometry of the pump suctions a seal of water becomes trapped in the suctions forming a plug for steam flow from the vessel around the loop to the break.[a] Since the core power is on decay heat, boiling of liquid in the core produces steam that has a pressurization effect. Because this pressurization cannot be relieved due to the pump suction seal formation (steam binding), both the vessel liquid level and the pump suction liquid level are manometrically depressed and in some cases can lead to momentary core rod heatup. The manometric balance of heads in the loop is in a quasi-steady state mode as more core steam generation expands against liquid heads in the loop. As a result, both the liquid level in the downflow side of the suction and vessel are depressed. Clearing of the intact loop suction of liquid can be envisioned as a steam/liquid interface traveling down the downflow side of the suction and up the upflow side in an orderly

a. This explanation applies only to cold leg breaks; steam venting would occur for a hot leg break as soon as the pathway from vessel to break was cleared.

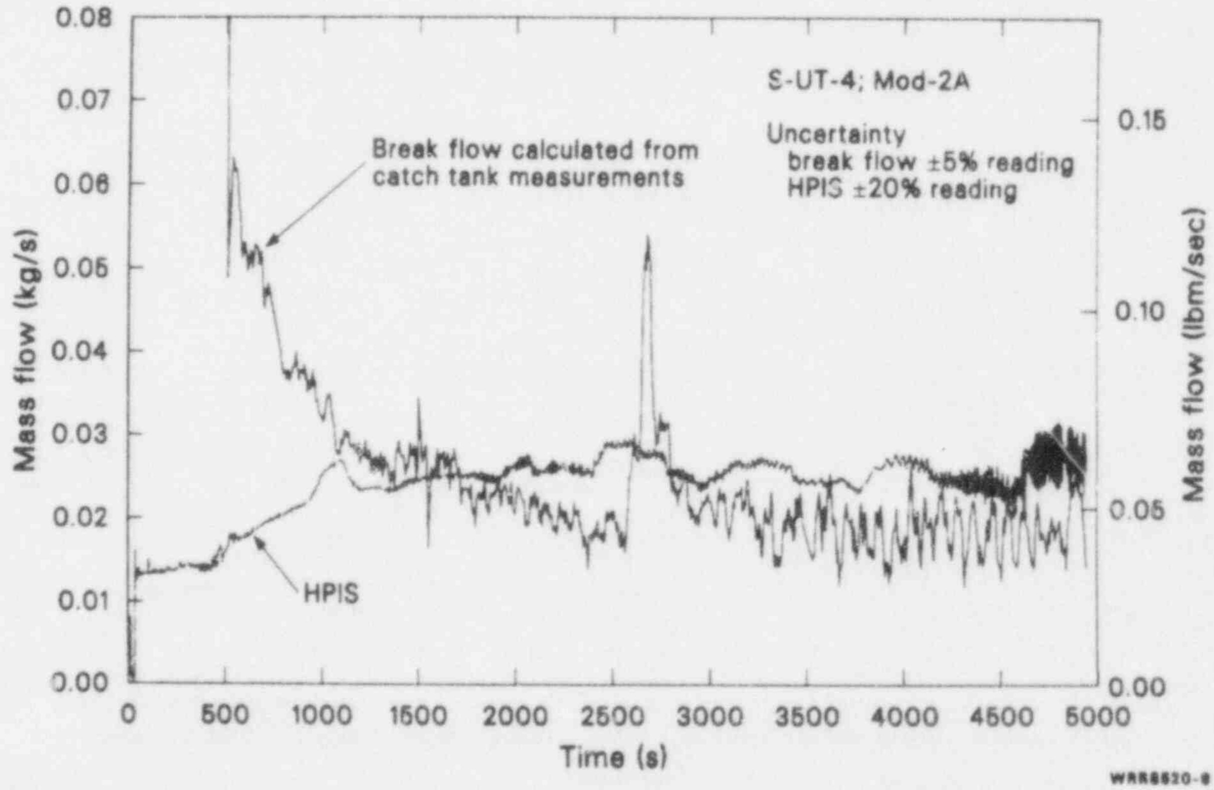


Figure 9. Comparison of break flow to HPIS flow (2-1/2% break).

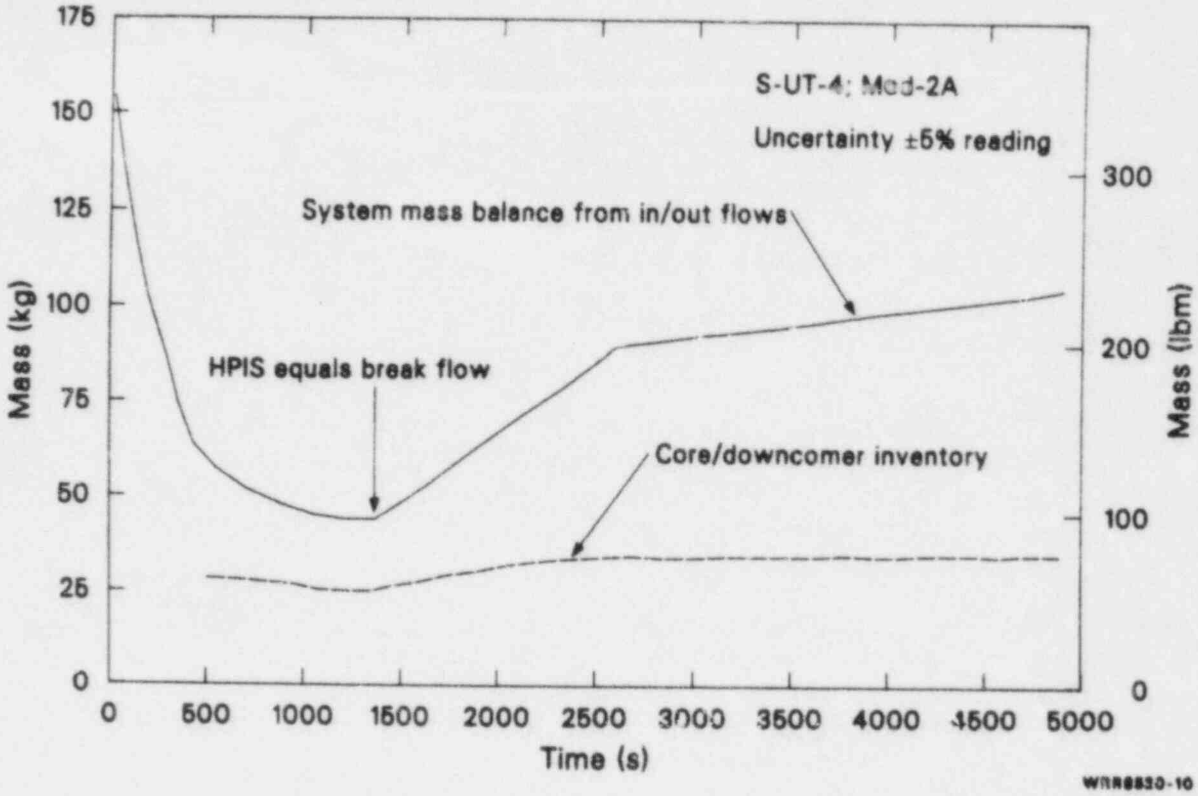


Figure 10. System mass balance compared to core/downcomer liquid inventory.

manner until cleared. The vessel level reaches a minimum value when the pump suction liquid level reaches the bottom of the suction, as shown in Figure 11. The vessel level continues to increase until clearing of the seal of liquid has been completed. Figure 12 is an illustration of system fluid distribution just prior to intact loop seal clearing and Figure 13 illustrates the fluid distribution after the intact loop seal clearing. As an example, (shown in Figure 13) the broken loop seal never did clear and some of the fluid from the intact loop and vessel downcomer migrated to the broken loop suction. The important result of the intact loop seal clearing was that there was a path for steam relief from the vessel upper plenum to the break through the cold leg of the intact loop that relieved the manometric balance of heads around the loop. As part of this head redistribution, the vessel liquid level increased. During this occurrence the core was usually recovered sufficiently to quench rod positions that had temperature excursions. For the break spectrum investigated in Semiscale (0.4% to 10%) the pump seal induced core rod temperature excursion was not sustained and seal clearing always lead to partial recovery of core liquid level. The severity of the vessel liquid level depletion during the seal formation is influenced by liquid heads in the primary U-tubes and more importantly the amount of bypass flow between the vessel upper head and downcomer, which is discussed later.

On many experiments the pump suction seal clearing and break uncovery (discussed under depressurization response) occurred at about the same time. As long as there was a suction seal, the pressurization because of core boiling manometrically pushed fluid up the downcomer and into the cold leg thus covering the break. Once the seal was removed the cold leg fluid was no longer held up and the cold leg fluid level relaxed such that the break uncovered. This lead to a more rapid depressurization as seen in Figure 6.

Two distinct types of vessel fluid depletions occurred during small breaks for a variety of reasons. Regardless of type of depletion, under normal ECC operation no sustained uncontrollable core heatup occurred for the break spectrum examined in Semiscale testing (0.4 to 10%).

The first vessel fluid depletion and core rod heatup occurs because of pump suction seal formation. Figure 14 shows core vessel collapsed level and rod temperature response during a typical 10%

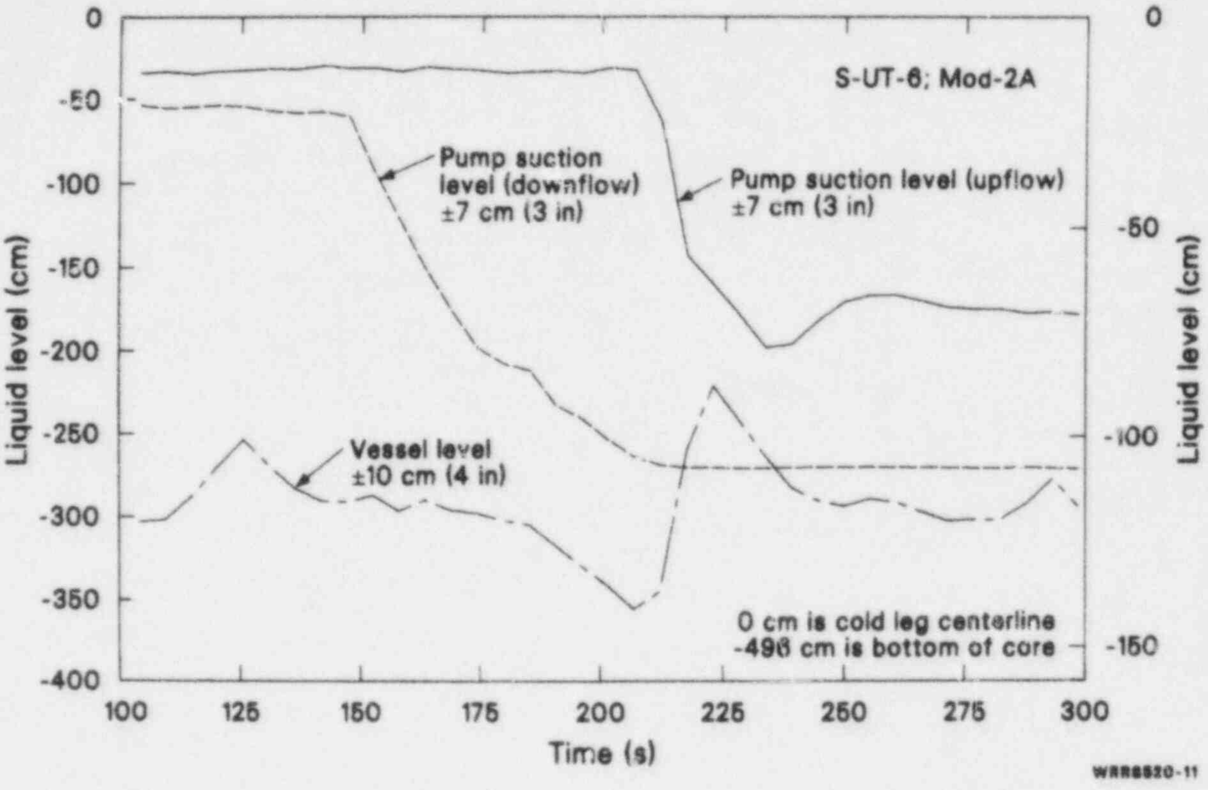


Figure 11. Comparison of intact loop suction liquid levels and vessel liquid level (5% break).

small break LOCA.[16,17] The seal formation depressed the vessel level and a minor core heatup occurred. However, when the pump suction seal cleared the entire core was again submerged in enough liquid to quench all rod heatups. Although pump suction seal formation and some vessel level depression occurred for the entire break spectrum encountered in Semiscale experiments with normal ECC, only the 5% break experiment exhibited sufficient vessel liquid level depression to cause rod heatup.

The second vessel fluid depletion and core heatup occurs during the period between pump seal clearing and accumulator injection. The system depressurization is slow during this period and if break flow is greater than HPIS there is a net outflow of vessel liquid that can lead to core rod heatup. In the Mod-2A system with 1:1 elevation scaling the only experiment with normal ECC parameters that exhibited sufficient core boil-off to cause rod heatup prior to accumulator injection was the 5% break experiment. However in the Mod-3 system with LOFT-scaled steam generators the 2.5% break experiment also exhibited some minor core rod heatup prior to accumulator injection. Figure 15 compares core collapsed liquid level and upper core thermocouple response for a 5% Mod-2A break case[24,25] showing the gradual boil-off of core collapsed liquid level and resulting core rod temperature excursion. Only minor heating occurred because accumulator activation quenched the core (the relationship between core level and core heatup is discussed later).

Another way core heatup can occur during small break transients is for the normal ECC delivery to be interrupted. For instance on one small break experiment (0.4%, break),[30] HPIS was assumed to be disabled. Liquid inventory was maintained sufficiently in the vessel to prevent uncontrolled core heatup through the depletion of accumulator tanks. After the depletion of accumulator tanks the relationship between depressurization rate and break flow was such that extensive[a] core uncovery occurred before the system LPIS setpoint pressure was reached. Core heatup and its relationship to vessel inventory will be discussed in detail in a later section.

a. Operational requirements are that electric core power was to be tripped when the highest rod temperature reached 1070 K (1466°F).

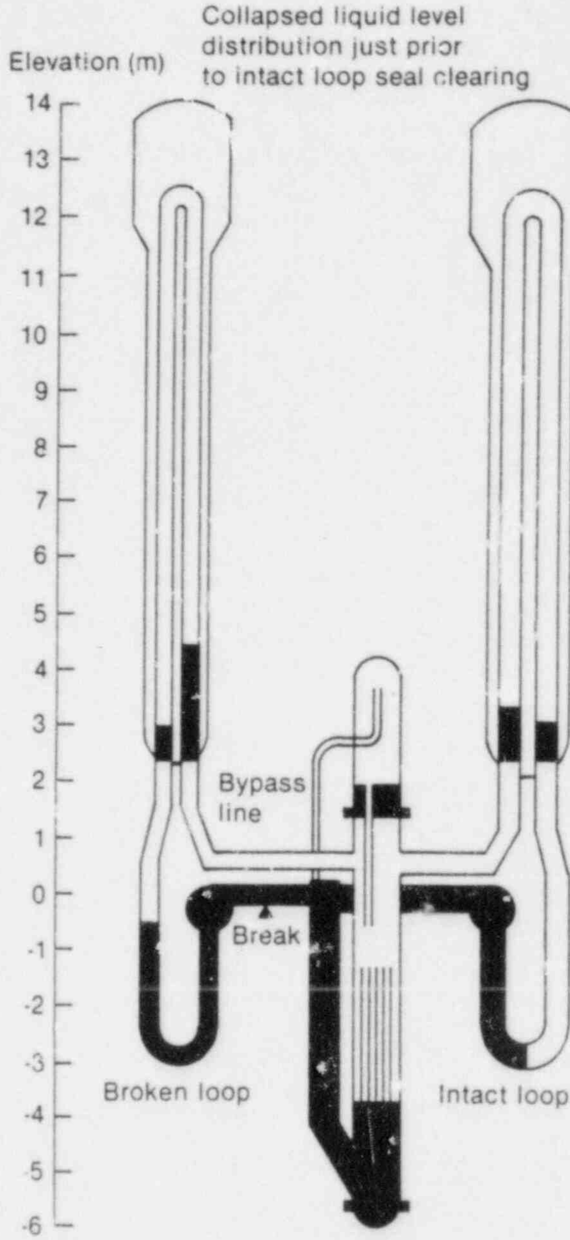


Figure 12. Illustration of the system liquid mass distribution just prior to intact loop seal clearing (5% break S-UT-8).

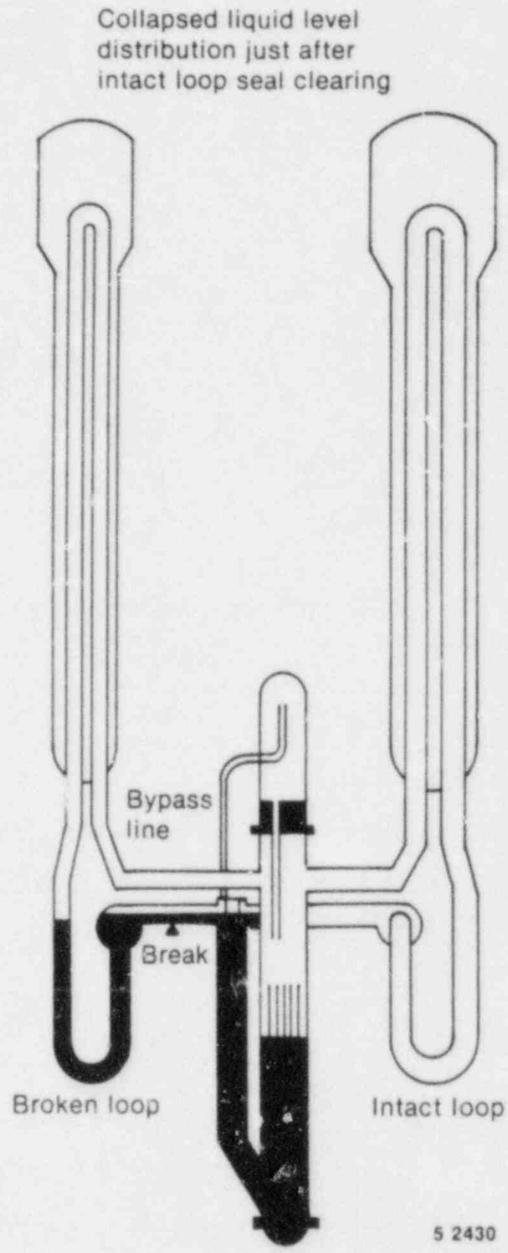


Figure 13. Illustration of the system liquid mass distribution just after intact loop seal clearing (5% break S-UT-8).

In summary, as long as normal ECC systems are activated during small break LOCAs for the break spectrum 0.4% to 10% no sustained core rod temperature excursions are expected.

TMI-2 Type Small Break. The accident at TMI-2 involved a primary system overpressurization and a stuck open pressurizer power operated relief valve (PORV). The small break part of the transient, then is the flow of primary fluid through the stuck open PORV. The transient at TMI-2 was initiated by a loss of the condensate feed pumps in the steam generator secondary feedwater system. The loss of these pumps induced the main feedwater pumps to trip, resulting in an electrical turbine trip and main steam isolation valve closure. Loss of turbine and feedwater activated the auxiliary feedwater system pumps; however, the auxiliary feedwater pump isolation valves were inadvertently closed and auxiliary feedwater was not available until eight min into the transient. With the steam generators isolated and full core power on, the primary fluid average

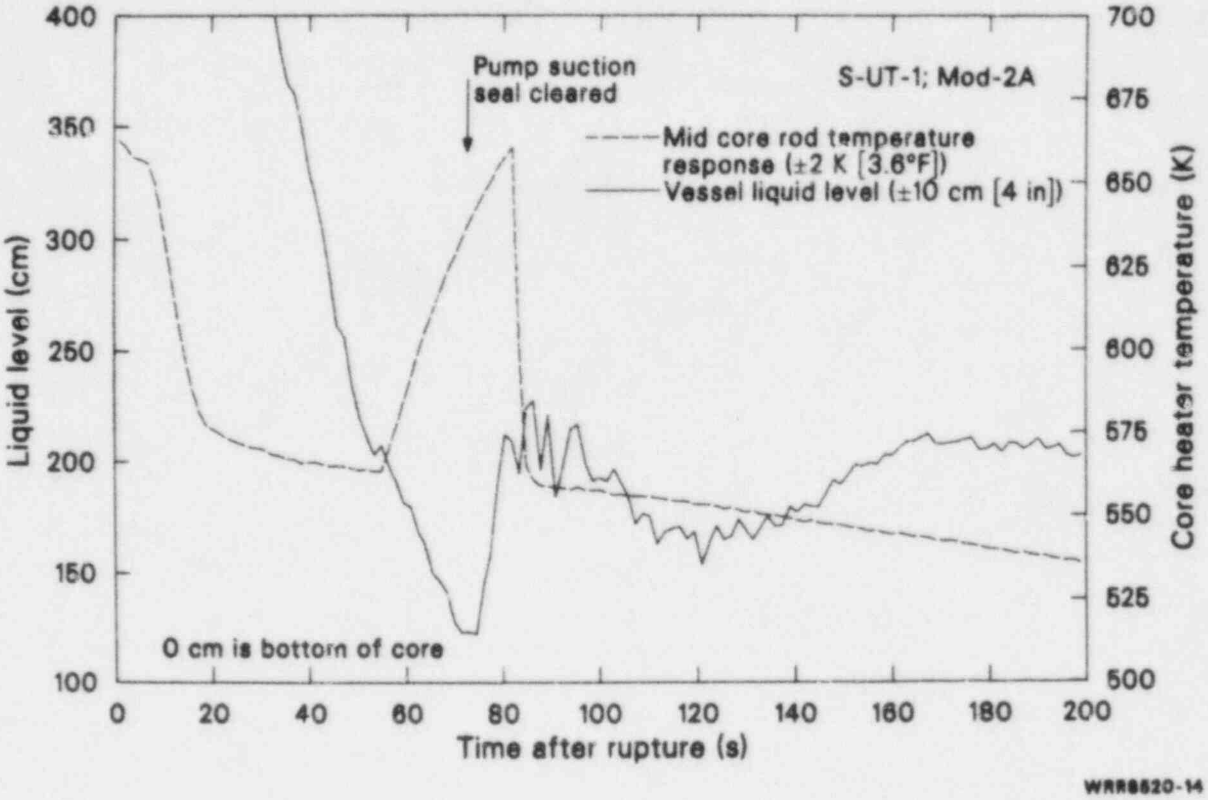


Figure 14. Comparison of vessel collapsed liquid level and midcore rod temperature response during pump suction seal formation and clearing (10% break).

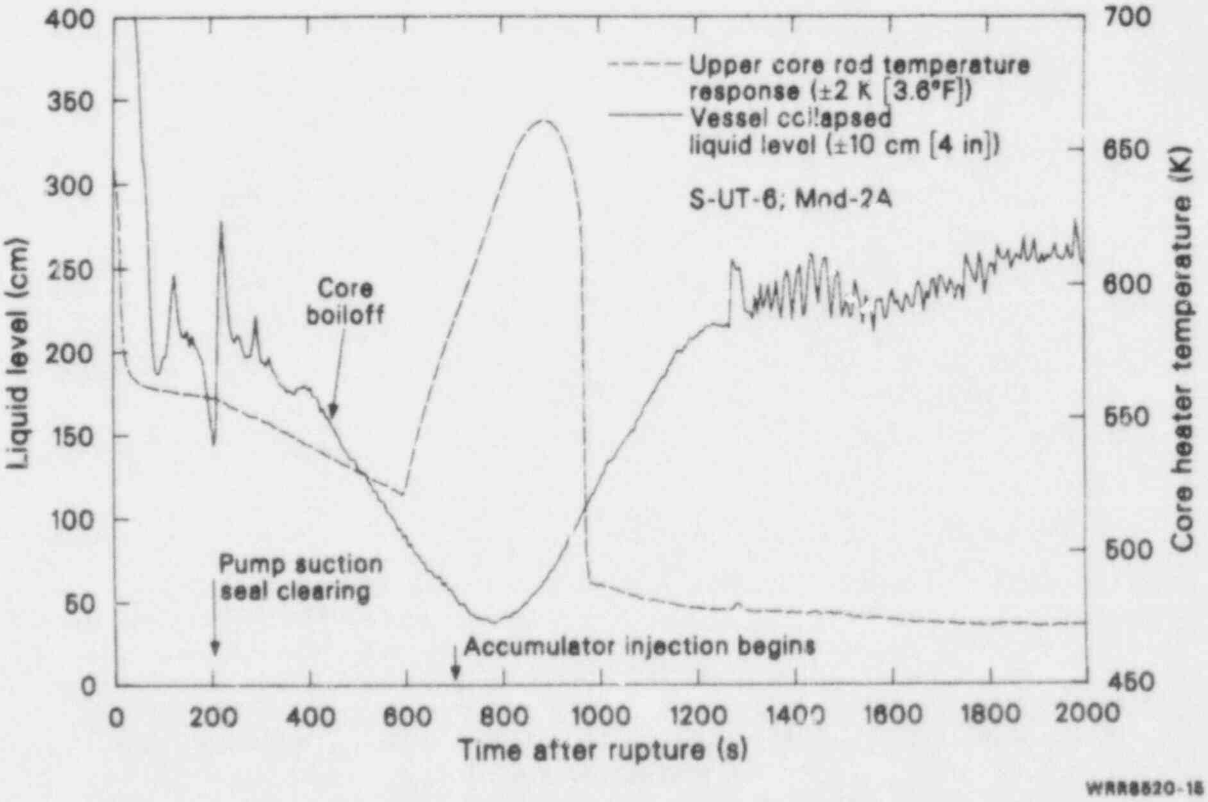


Figure 15. Comparison of vessel collapsed liquid level and upper core rod temperature response for the core boil-off period following pump suction seal clearing (5% break).

temperature increased, resulting in fluid expansion causing an overpressurization. The PORV opened at a high pressure trip 15.55 MPa (2255 psig), however, the primary pressure continued to increase because the rate of fluid expansion exceeded the PORV flow rate for liquid flow. The core power was automatically scrammed when the primary pressure reached 16.24 MPa (2355 psig). With core power off the primary pressure decreased as primary fluid flowed out the stuck open PORV. Eventually the primary pressure decreased enough to activate HPIS at a system pressure of 11.03 MPa (1500 psia), however, level swell due to core decay heat and HPIS flow caused a filling of the pressurizer and HPIS was terminated to avoid further overpressurization. The continued discharge of primary fluid eventually lead to loop pump head degradation and power to the main coolant pumps was terminated. This action coupled with a continued flow of primary fluid out the stuck open PORV resulted in core uncovery and core rod heat up.

The Semiscale TMI simulations[3] were performed in the MOD-3 using a best estimate of the TMI boundary conditions. The overall response of the Semiscale system to the TMI scenario agreed well with what TMI data was available. The following discussion is divided into two parts. The first part discusses the short term (0 to 60 s) response and the second part examines the long term response (0 to 7000 s) leading to core uncovery.

Figure 16 compares the TMI pressure response to the Semiscale response early in time showing similar trends. It was not clear from the TMI data whether the code safety valves on top of the pressurizer actually lifted, however, the Semiscale valves did lift. Following opening of the PORV the system pressure continued to increase as the steam generators were isolated at time 0 with continued core power. Only after the core scrammed on a high pressure trip did the primary pressure decrease. The rapid decrease in pressure occurring at about 21.5 s started before the code safety valve opened indicating loss of core power alone caused the decrease, however, the effect of closing the code safety valve is readily apparent as the depressurization rate greatly decreased. The pressurizer level increased as shown in Figure 17 due to primary level swell. The steam generator secondary pressure rose to relief valve set points [8.0 MPa (1160 psia) in the Semiscale Simulations] as core energy was transferred from the primary to secondary fluid as seen in Figure 18. Opening the relief valves caused an immediate reduction in secondary pressure even though core power continued at the initial value. The safety valve was reseated at about 27 s causing a slight pressurization of the secondary as the core power was now only on decay heat.

On a long term basis the major occurrences that effected the core uncovery were the termination of HPIS and termination of pump power. The primary pressure rapidly reduced to the loop saturation pressure accompanied by flashing, retarding the primary depressurization, as shown in Figure 19. Following attainment of saturation conditions the depressurization of the primary was slow and stayed between 6 and 7 MPa (875 and 1055 psia) for the remainder of the transient. The flow of fluid out the PORV caused a mass inventory redistribution in the primary. Figure 20 shows the filling of the pressurizer caused by suction of loop fluid created by the flow of fluid out the PORV. The pressurizer was full by 1700 s and remained full for the remainder of the transient. With a high pressurizer level, TMI-2 operators felt justified in turning off HPIS as they thought the system was liquid solid; however, voids were formed in the vessel (due to mass depletion from the PORV flow) that eventually lead to core uncovery. Figure 21 compares pressurizer level, vessel liquid level, and core heater rod temperature showing the ultimate effect of turning loop cooling pumps off in the Semiscale simulations. Within 600 s following pump trip, the core level begin boiling off (due to decay heat) eventually resulting in core rod heat up. As long as the primary cooling pumps were on, core power decay heat could be removed by way of convection to steam in the core, heat loss in the empty steam generator, and loop piping. When the pumps were tripped the only possible mode of core decay heat removal was reflux condensation; however, with the steam generator secondaries empty, reflux was ineffective resulting in a core boil-off and core rod heatup as shown in Figure 21.

The break mass flow rate out the PORV was large enough to cause vessel voiding; however compared to the entire primary mass inventory the break flow was small as shown in Figure 22. Between 2000 and 6000 s only 72 kg (158 lbm) out of a total initial mass of 150 kg (330 lbm) left the system; therefore had the steam generator secondary been active (feed and steam) reflux could have

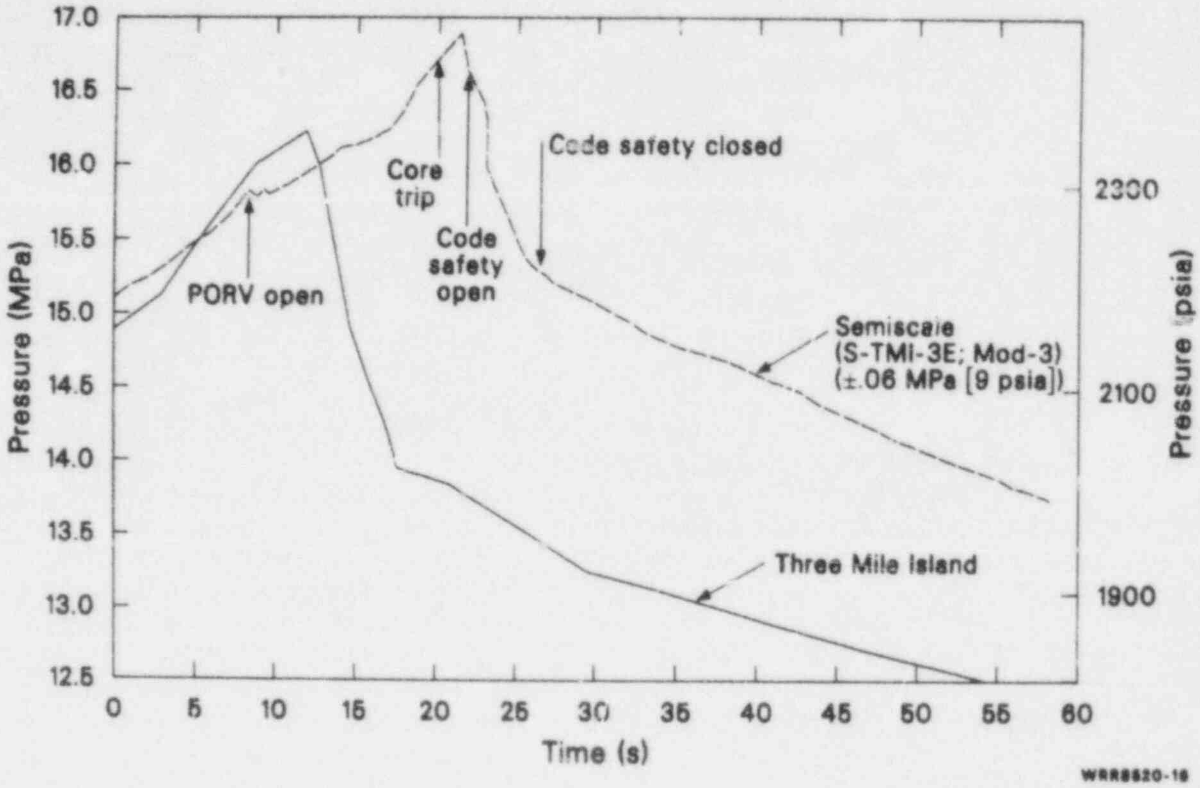


Figure 16. Comparison of primary pressure for Three Mile Island and the Semiscale simulation.

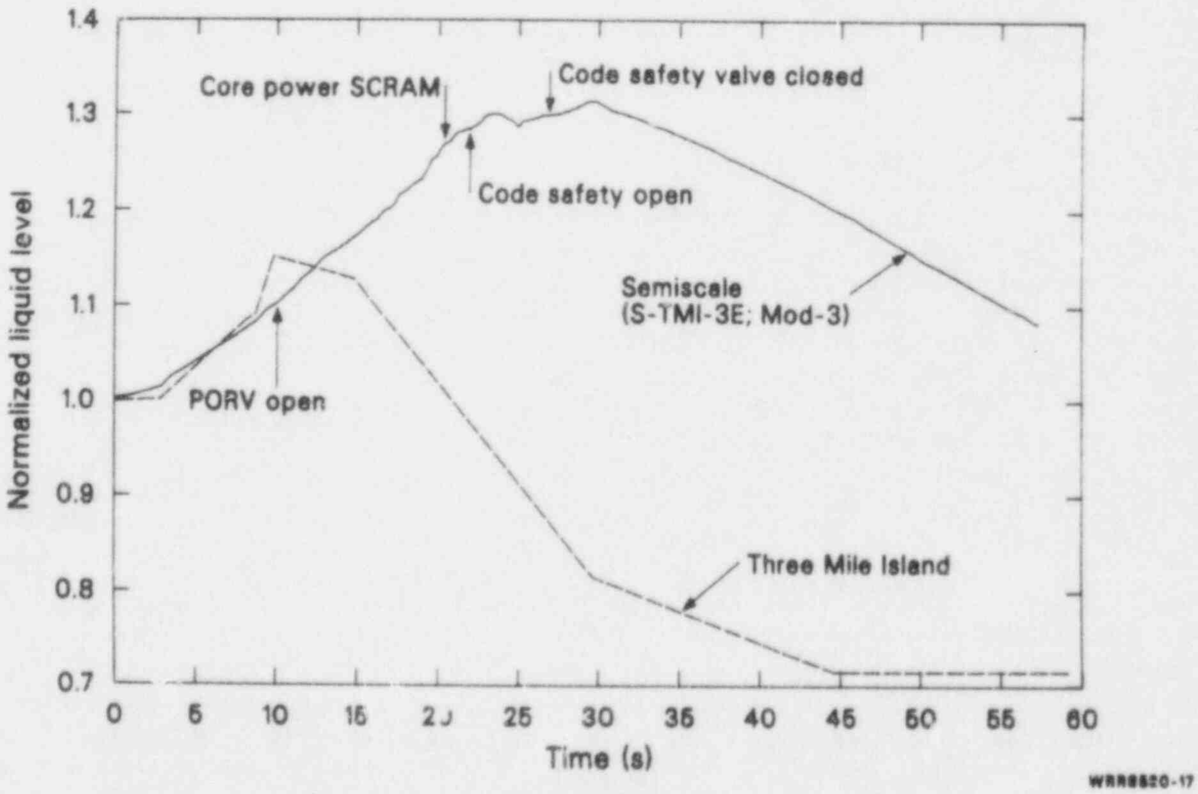
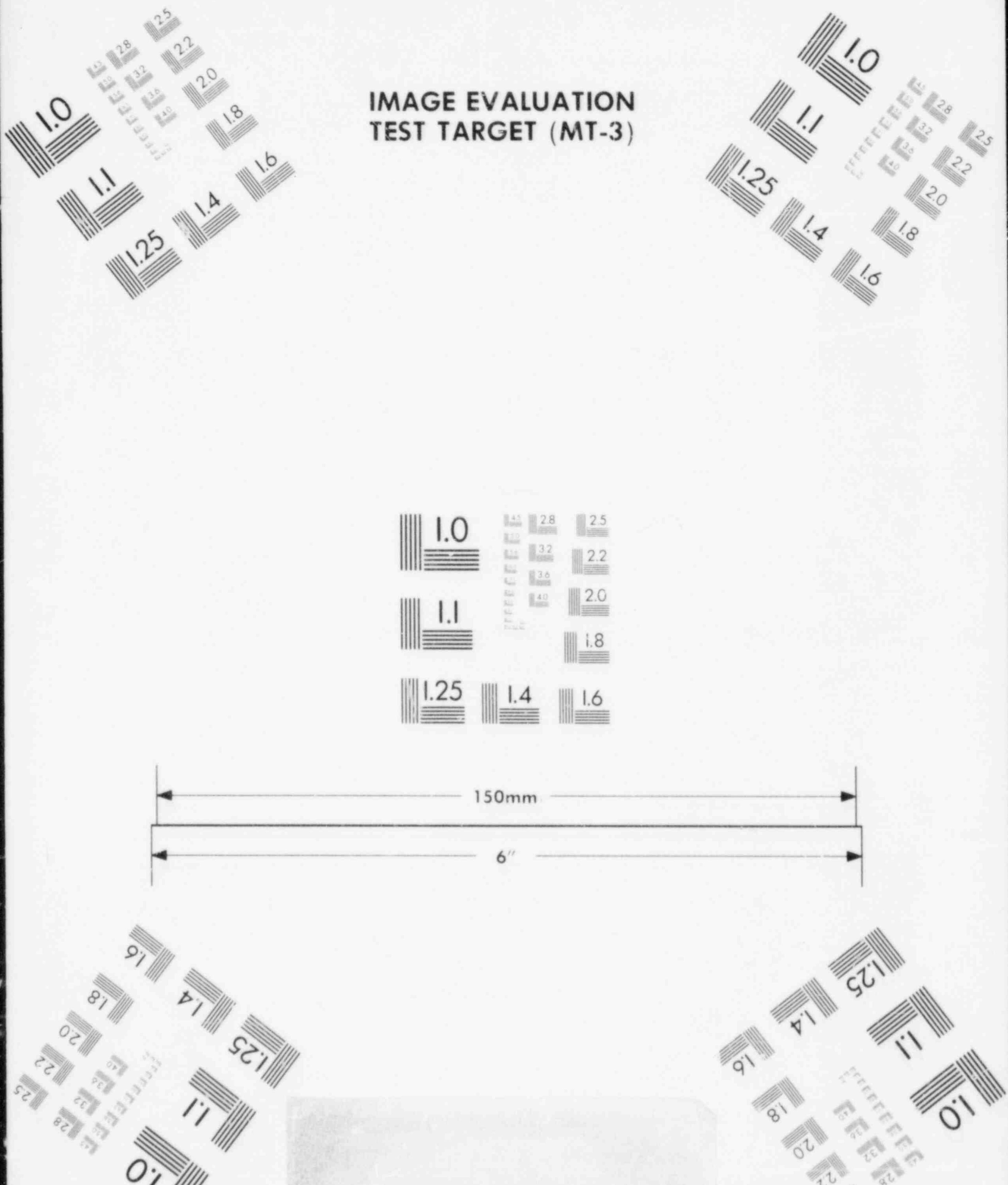


Figure 17. Comparison of normalized pressurizer liquid levels from Three Mile Island data and from Semiscale Test S-TMI-3E.

IMAGE EVALUATION TEST TARGET (MT-3)



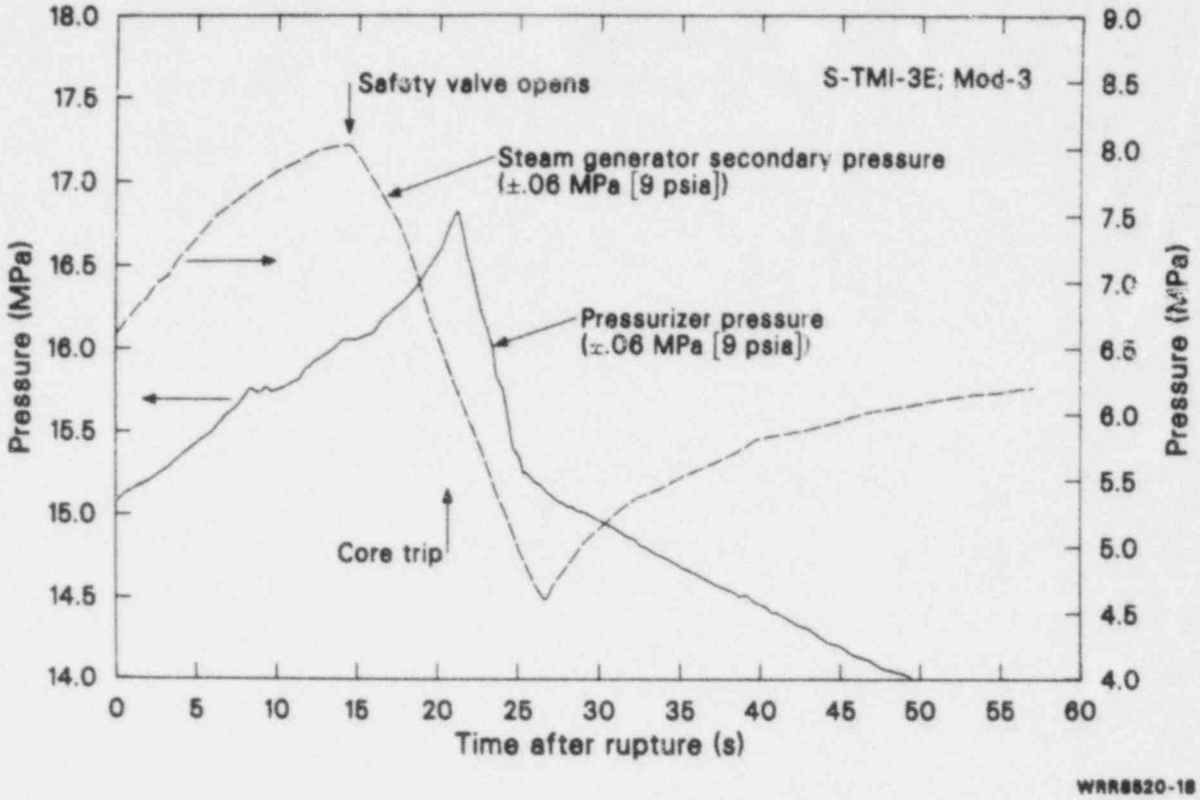


Figure 18. Comparison of system pressure and intact loop steam generator secondary pressure for Semiscale Test S-TMI-3E.

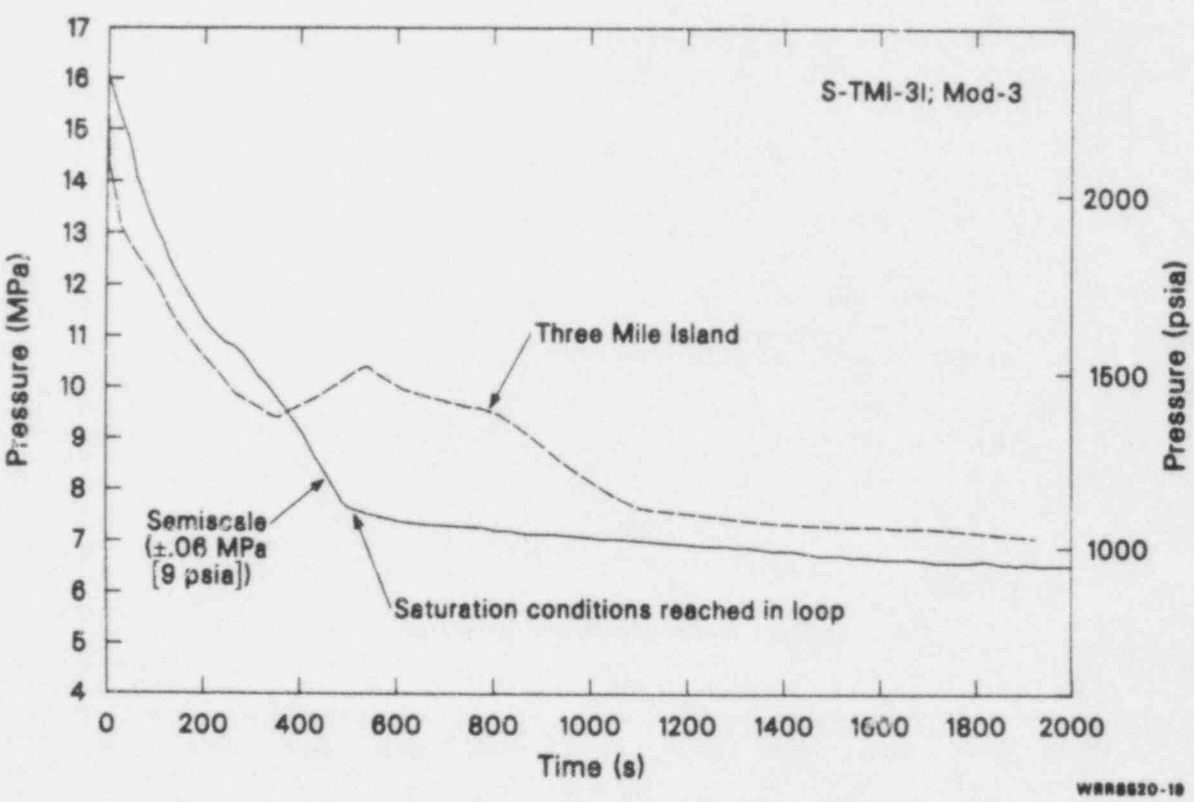


Figure 19. Comparison of system pressure for TMI transient and Semiscale simulation (Test S-TMI-3I).

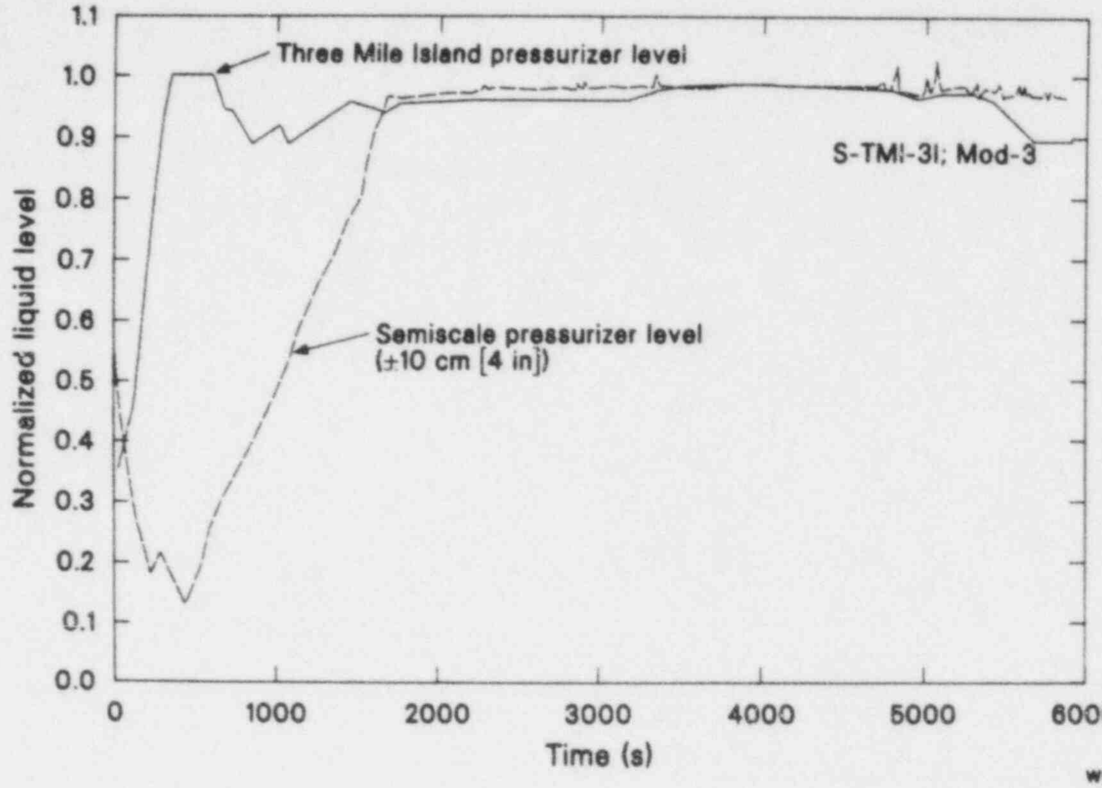


Figure 20. Comparison of normalized level in the pressurizer for TMI and the Semiscale simulation (Test 49.MI-3I).

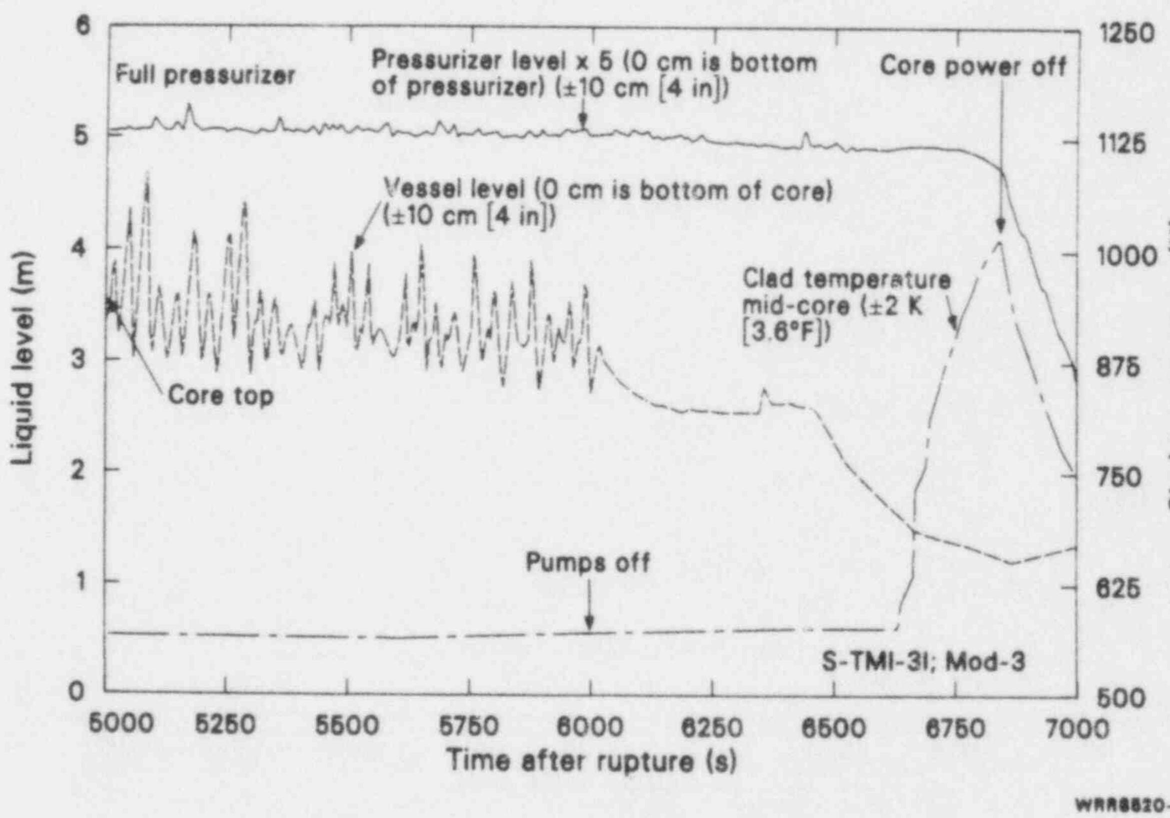


Figure 21. Comparison of pressurizer and core liquid level with rod cladding temperature for Semiscale Test S-TMI-3I.

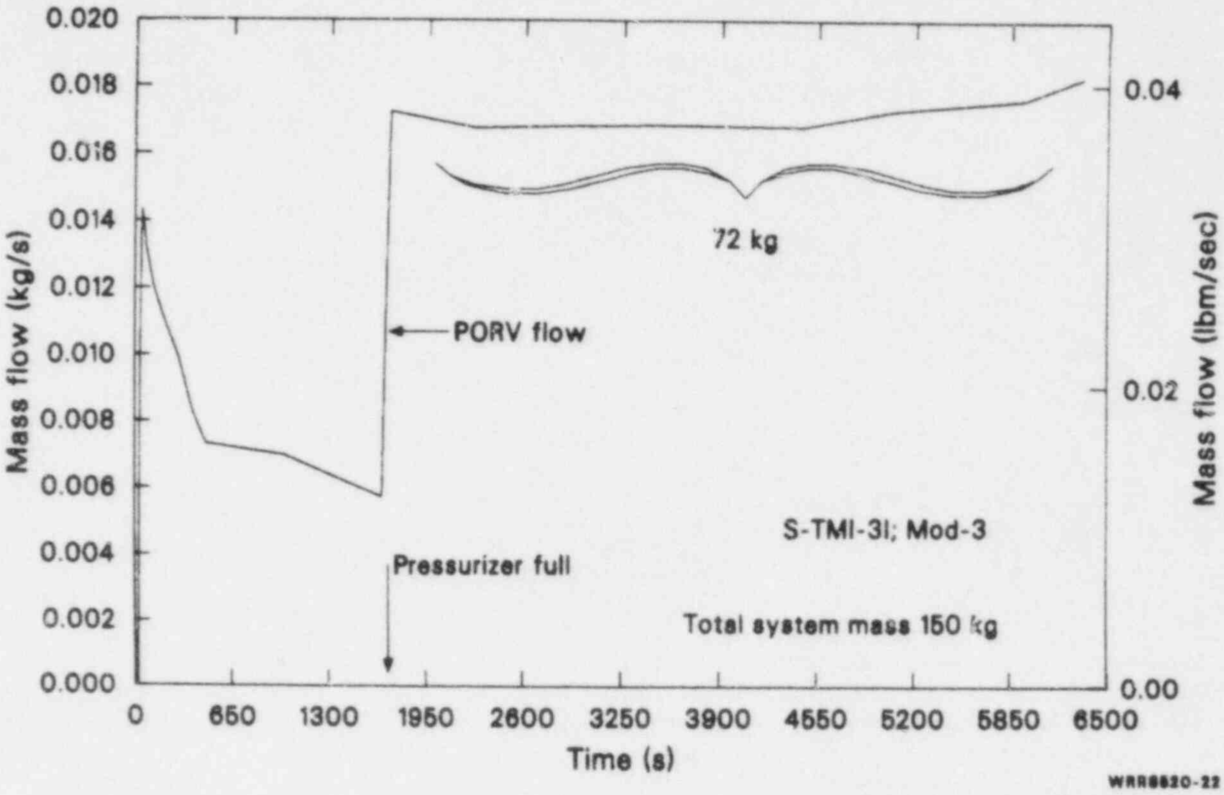


Figure 22. Estimated break flow for the Semiscale simulation (Test S-TMI-31).

provided long term core cooling for thousands of s[a] without the use of primary pumps. Figure 22 also shows a large increase in break mass flow rate when the pressurizer filled with liquid. This corresponds to the changes from two-phase flow to single-phase liquid flow.

A core heater rod temperature excursion accompanied the core boil-off. In the Semiscale experiments core power was terminated on a high temperature trip [1050 K (1430°F)]; however in the TMI actual accident considerably higher temperatures occurred. Based on the heat transfer coefficients in the core for the Semiscale simulations core cladding temperatures in excess of 2000 K (3140°F) could have occurred in TMI.

During the TMI accident, the first strong indication of severe liquid voiding in the core was evidenced by superheat in the hot legs. The Semiscale simulation also indicated superheat in the hot legs at about the same time as the TMI transient as shown in Figure 23. This indicates that the Semiscale correctly scaled such important parameters as PORV flow, system volume, and core decay heat.

In summary, a stuck open PORV such as occurred at TMI-2 is another kind of small break accident involving a decrease in system mass inventory. The flow of fluid through the PORV caused a decrease of loop mass inventory as loop fluid was pulled into the pressurizer. The pressurizer once filled remained filled for the entire transient even though other parts of the loop were voiding. HPIS was disabled to avoid a liquid full pressurizer and primary coolant pumps were turned off to avoid cavitation. These two operator induced events contributed to a core rod heat up with rod cladding temperatures estimated to be in excess of 2000 K (3140°F) based on Semiscale heat transfer results.

Steam Generator Tube Rupture. Steam generator tube rupture is another form of a small break, as the tube rupture allows a flow of primary fluid out of the system to the affected loop steam generator secondary. In many ways, the tube rupture signature response is similar to the pipe break response discussed earlier.

a. Reflux condensation occurs when the vessel collapsed liquid level is below the hot leg, and the steam generator tubes are steam filled. Steam generated in the core flows to the heat sink provided by the steam generator secondary where it is condensed and flows back to the core.

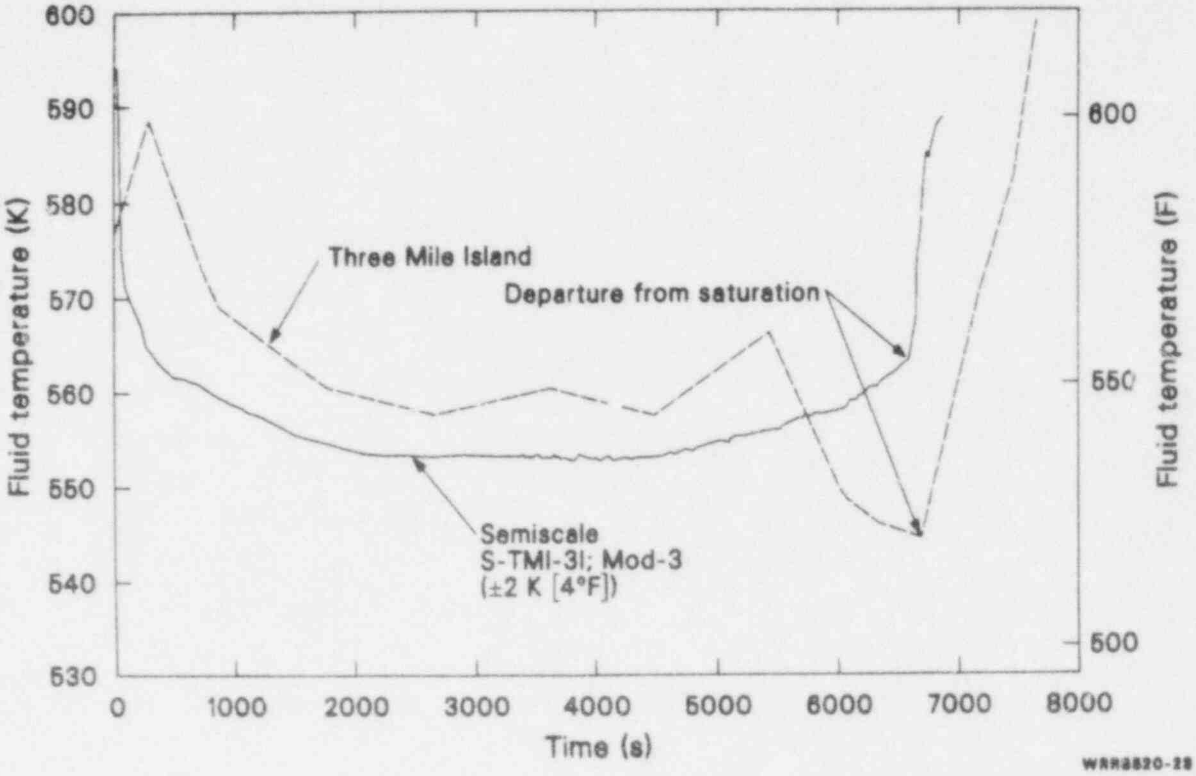


Figure 23. Comparison of hot leg fluid temperature for TMI and the Semiscale simulation (Test S-TMI-3I).

The occurrence of a tube rupture in the Semiscale Mod-2B[37-45] during typical PWR type operating conditions has a very distinctive signature response. The system signature response can be characterized by such parameters as primary and secondary system pressure, system liquid levels, fluid flow rates, and temperatures. The signature response is discussed for a time period of 600 s, which was assumed to include only automatically occurring events without operator action. A time of 600 s was chosen to be representative of the time required for an operator to identify the occurrence of a tube rupture transient. For discussion purposes, a single cold side tube rupture in the Semiscale system is used for this section.[44]

The tube rupture, occurring at 0 s, caused a primary system depressurization and loss of primary mass to the broken loop steam generator secondary system. Figure 24 compares the primary and secondary pressures early in the transient. Primary fluid, originally at 15.54 MPa (2253 psia) flowed through the conical flow tube break orifice into the broken loop steam generator secondary originally 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 25) at which time the primary depressurization rapidly increased. The increase in primary depressurization corresponded exactly in time to the interfacial liquid level of the pressurizer reaching the bottom of the pressurizer. When the pressurizer level reached the surge line connecting the pressurizer to the hot leg there was a large change in the amount of volume for 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 volume was high and promoted flashing, which in turn retarded the primary depressurization. When the interfacial liquid level depleted to the surge line (due to break flow), the volume of saturated liquid decreased which retarded flashing, resulting in an increase in depressurization. Shortly after the pressurizer interfacial level cleared the bottom of the pressurizer, the low pressurizer pressure set point of 13.1 MPa was achieved, automatically causing core power scram to the ANS decay curve and the main steam isolation valve (MSIV) closure on both steam generators.

Upon MSIV closure, primary to secondary heat transfer in both the broken and intact loop steam generators caused a rapid pressurization of the secondaries, as shown in Figure 24. Prior to achieving

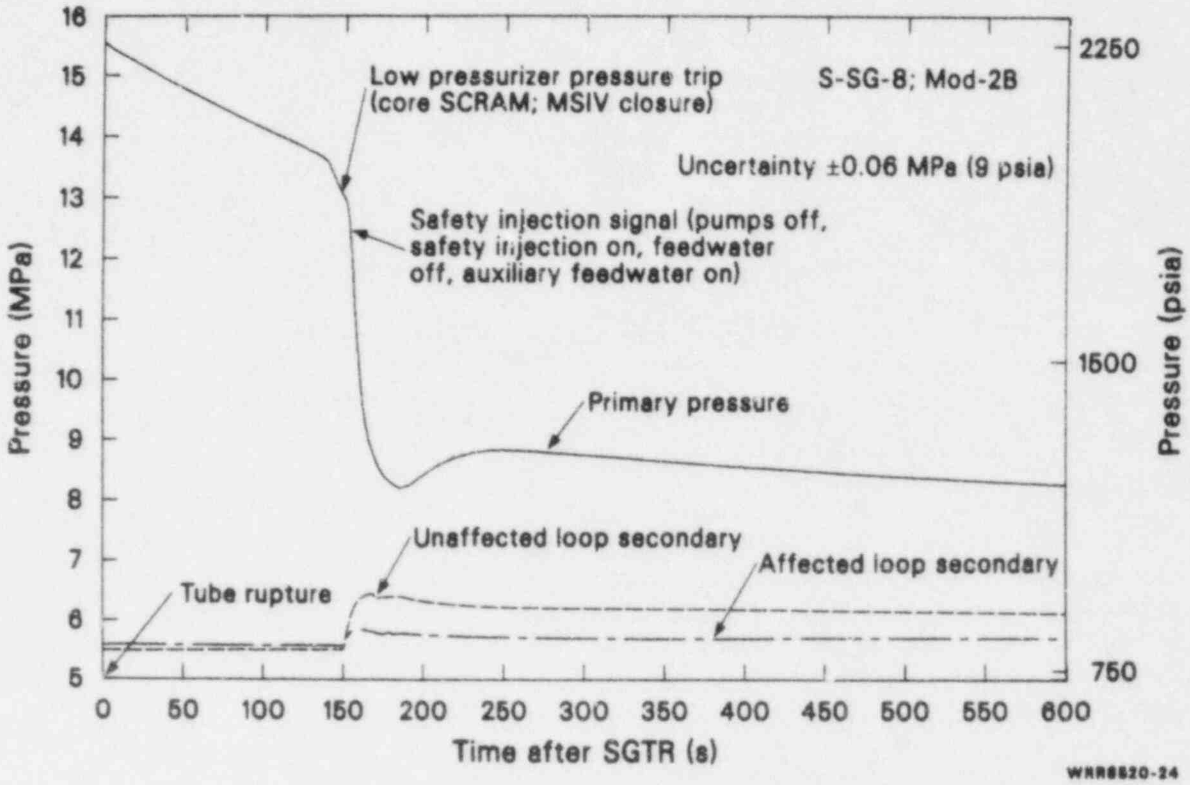


Figure 24. Comparison of primary and secondary pressure during a cold side, one-tube rupture transient.

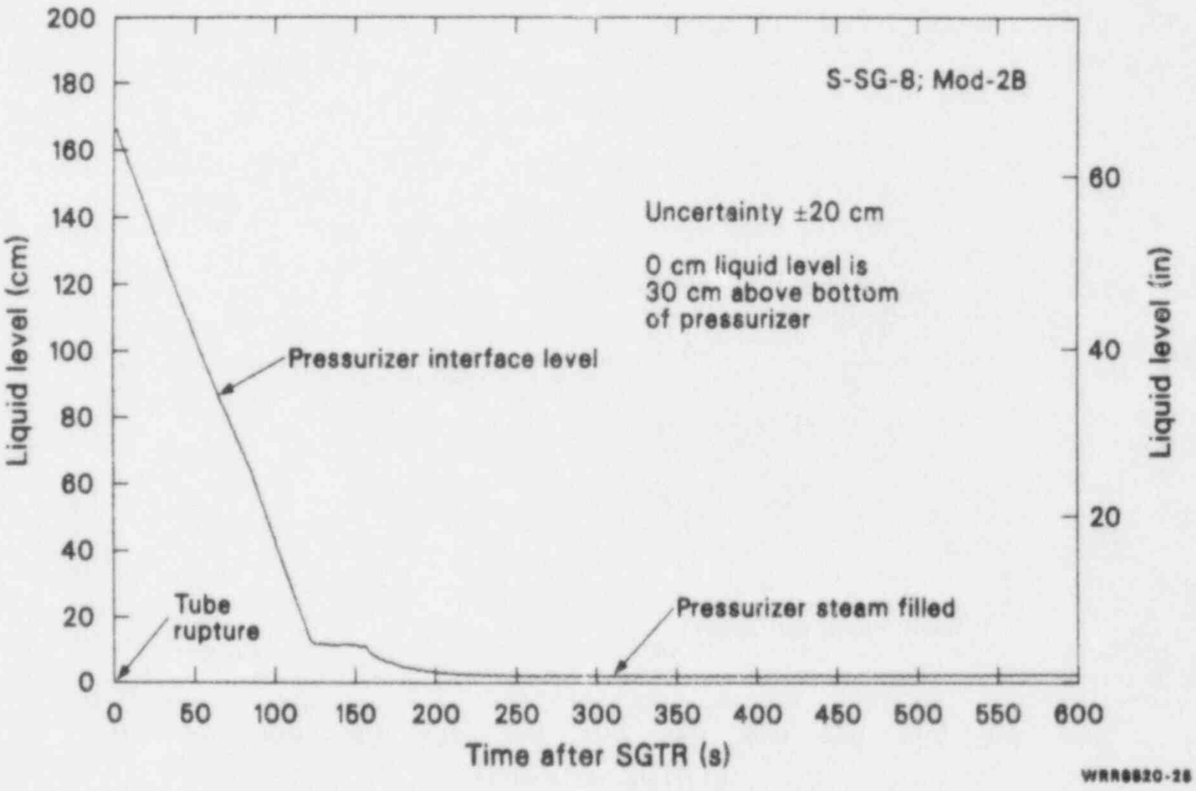


Figure 25. Pressurizer interfacial liquid level during a cold side, one-tube rupture transient.

the low pressurizer pressure trip, both the intact, and broken loop steam generator secondary pressure remained fairly constant as full core power was removed be way of normal secondary steaming conditions through a full open MSIV. The energy addition due to tube rupture break flow from the primary to broken loop secondary caused a negligible rise in broken loop secondary pressure prior to MSIV closure. Following MSIV closure the pressure rose quickly in both generator secondaries to the atmospheric dump valve (ADV) set point pressures[a] and the ADVs were cycled several times. The secondary pressure soon leveled out below the ADV set point as primary to secondary heat transfer was reduced due to a reduction in primary heat source after core scram.

Following core scram, the system primary pressure showed an increase in depressurization rate due to a shrinkage of the primary fluid caused by cool down (greater heat loss from primary to secondary than heat input from the core). No major change in primary depressurization occurred when the primary pressure reached the safety injection signal [12.51 MPa (1813 psia)] that automatically induced termination of power to the primary coolant pumps, initiation of safety injection, termination of main feedwater, and start up of auxiliary feedwater to the secondaries. The effects of the automatic safety injection events were overshadowed by the rapid reduction of core power and resulting primary fluid shrinkage due to primary-to-secondary heat transfer. Eventually, the primary system depressurization was sufficient for the hot leg fluid to reach saturation conditions at about 220 s, (Figure 26). Flashing in the system fluid then caused a major reduction in the depressurization rate. The primary pressure made a slight recovery between 190 and 240 s. This repressurization was caused by a combination of: superheated steam in the pressurizer due to heat transfer from the pressurizer walls to the pressurizer fluid (Figure 26), and the change from forced circulation to natural circulation heat transfer in the steam generators that occurs as the primary pumps coast down. Following pump coastdown, the core decay heat removal mechanism was single-phase natural circulation and the magnitude of the flow rate is typical of single-phase results found previously in Semiscale separate effects experiments. Following the slight primary repressurization period (190 to 240 s), the primary pressure first stabilized then followed a slow depressurization but remained above the broken loop ADV set point for the entire initial 600 s period. This slow depressurization was supported by a combined energy balance including safety injection flow, primary to secondary heat transfer, break flow, and primary and secondary wall heat loss.

During the first 600 s, only minor system mass voiding occurred, as shown in Figure 27, which compares a primary unaffected loop 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 from 421 cm to 375 cm (165 to 137 in.) above the cold leg. Because of the positive differential pressure between the primary and broken loop secondary, a positive break flow persisted throughout the early period; however, safety injection flow, once initiated, was slightly higher than break flow rate resulting in a slight filling trend in vessel upper head level during the first 600 s, as shown in Figure 27.

This basic signature response was found to be typical for one-, five-, and ten-tube ruptures; only the timing of events such as core scram, MSIV closure, and safety injection were different. In addition, the signature response was found to be essentially identical for hot side and cold side tube ruptures. The fundamental difference for the break spectrum studied was the relationship of safety injection and break flow. For the five- and ten-tube breaks, the vessel liquid inventory was considerably less than for the one-tube case because of a much higher break flow in relation to safety injection flow. At 600 s, the one-tube break had a system inventory of about 87%; the five-tube break had an inventory of 60%; and the 10-tube break had an inventory of 52%. Even though the vessel liquid collapsed level was reduced to the top of the core during the ten-tube rupture and within 15 cm of the top of the core for the five-tube rupture, no core rod heatup occurred.

SBLOCA Issues Addressed In Semiscale Experiments

Following the accident at TMI, a great number of issues arose relative to small break loss-of-coolant accidents. The following issues relative to

a. The ADV set point pressure is different for the intact loop and broken loop because of metal mass scaling differences between the two generators.

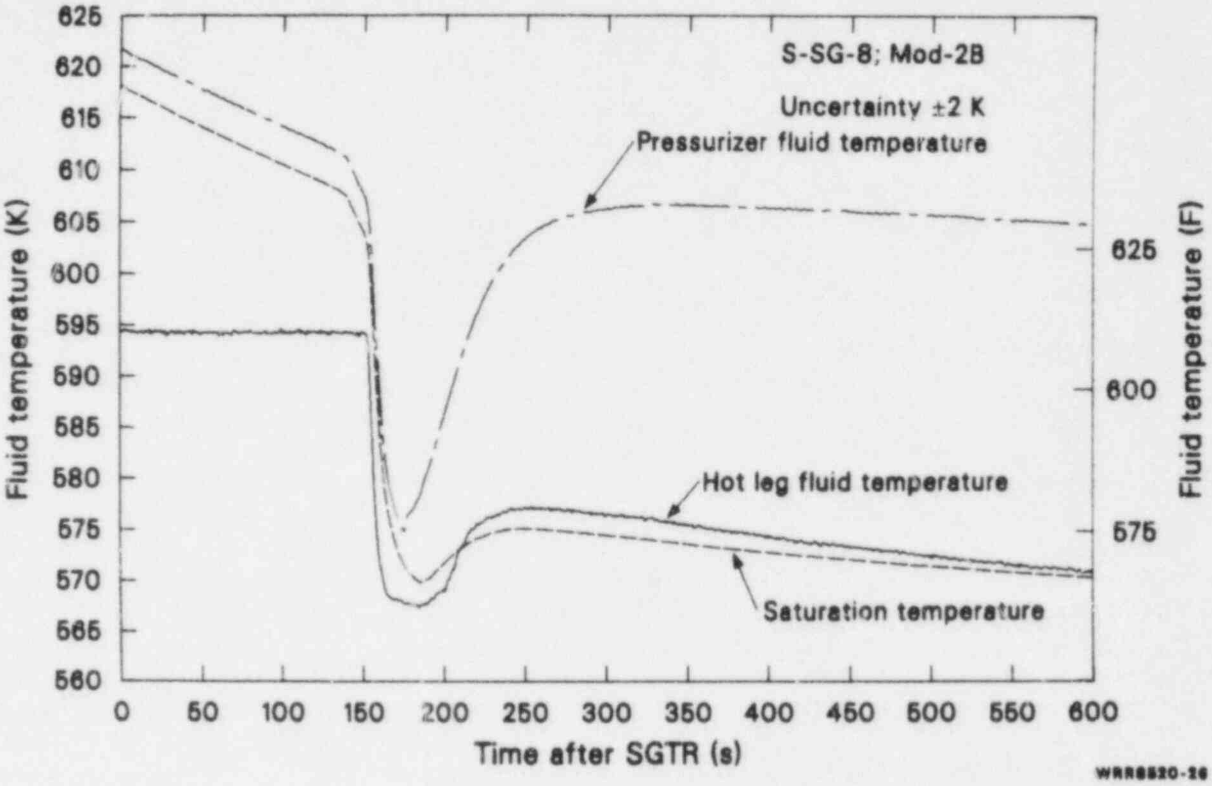


Figure 26. Comparison of fluid temperatures and saturation temperature for a cold side, one-tube rupture transient.

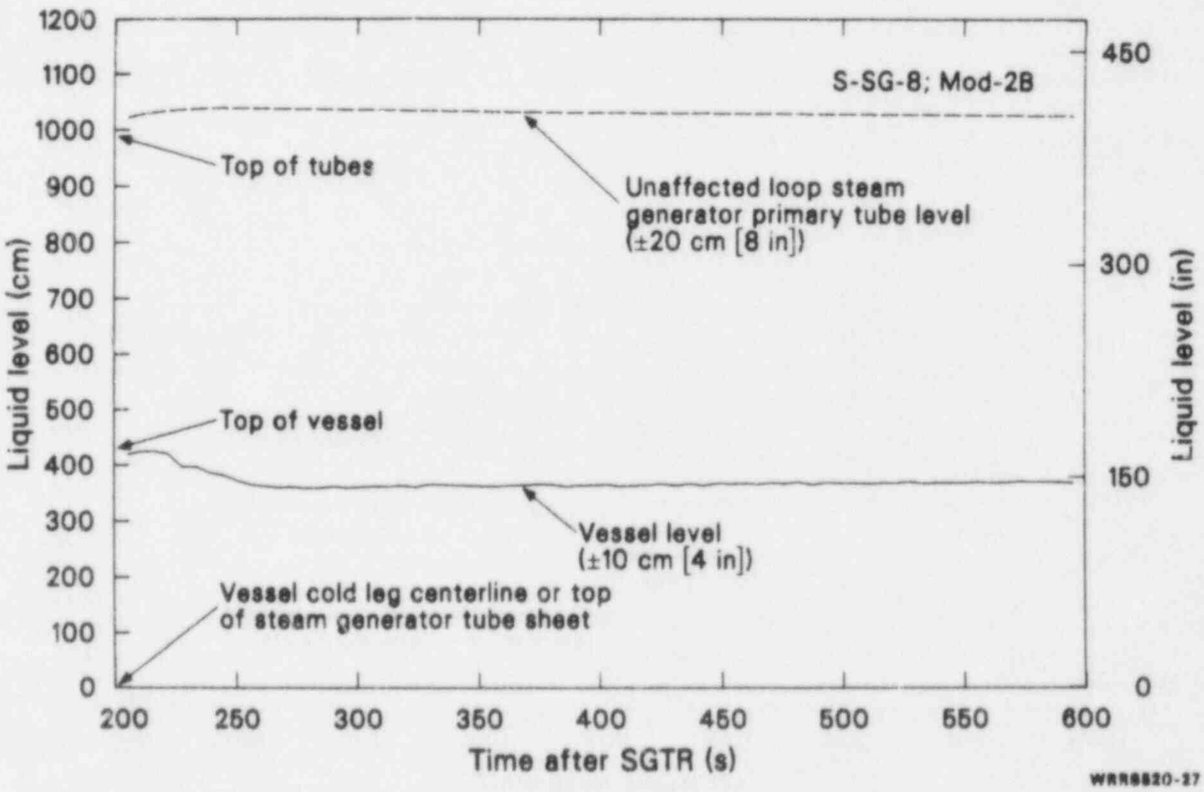


Figure 27. Comparison of collapsed liquid level for unaffected loop steam generator primary tube and vessel upper head during a cold side, one-tube rupture transient.

accident severity have been addressed by the Semiscale SBLOCA experiments: the effect of break size, the effect of break location, the effect of upper head to downcomer bypass flow, the preferred operation of primary coolant pumps during a SBLOCA, the relative merits of upperhead injection versus nonupperhead injection during SBLOCA, recovery procedures from a small break, and the effect of compounding failures during a small break.

Effect of Break Size (pipe breaks). The severity of a small break loss-of-coolant accident regardless of break size can be measured by examining the amount of liquid uncovery in the core associated with a small pipe break. The central problem is how much core liquid uncovery occurs prior to accumulator injection. Larger small breaks do allow more system fluid out of the break; however the time to depressurize the primary system to accumulator set points is quicker resulting in a faster core reflood. During very small breaks the break flow can be on the same order of the HPIS flow resulting in only a small net loss of mass inventory. There is a range of break size however, where the combination of break flow and HPIS flow result in core uncovery. This range was determined to be 2.5 to 10%; therefore, these breaks were examined in the bulk of Semiscale testing.

The severity of a SBLOCA is directly related to the depressurization rate. The sooner accumulator and LPIS pressure setpoints are reached the sooner a mass inventory increase can occur. The pressure response is directly related to break uncovery, which is coupled to pump suction seal clearing as discussed earlier. Figure 28 compares the depressurization rate for 2.5,[22] 5,[25] and 10%[17] small break LOCAs performed, in the Semicale Mod-2B. All experiments achieve saturation conditions within a few seconds of each other; however, major differences in time to accumulator setpoint occur largely because of different times for pump suction seal clearing (at 60, 200 and 425 s, respectively for the 10, 5, and 2.5% break). Removing the seals causes a break uncovery and faster rate of depressurization as primary steam rushes out of the loop through the break. HPIS flow was less than break flow for the examples shown in Figure 28; however, HPIS flow has a bigger effect in maintaining loop mass inventory for the 2.5 and 5% break than the 10% break. For instance, the HPIS about equaled break flow for the 2.5 and 5% break cases at 1300 and 1500 s, respectively while the 10% break flow was always higher than HPIS flow.

The relationship between HPIS flow, core boiloff, and break flow resulted in a break size of ~5% having the most severe core liquid level depression, as shown in Figure 29. For the 5% break size, a core temperature excursion [maximum core temperature 660 K (728°F)] occurred during core boiloff and was mitigated by the introduction of accumulator ECC flow, as shown in Figure 30. The shape of core liquid level versus a number of tube ruptures on Figure 29 suggests that a 6 to 7% break might produce slightly lower vessel collapsed levels.

Break Location (pipe breaks). Three basic locations for a pipe break were examined in the Semiscale experiments, cold leg (between the pump discharge and the vessel), hot leg between the steam generator and the vessel, and the pump suction.

Most small break transients experiments concentrate on cold leg breaks. This logically was based on the observation that the highest break flow occurs during the subcooled decompression and any phenomena lengthening the duration of the subcooled break flow should result in a maximum expulsion of system liquid inventory. During a cold leg break transient the cold leg fluid is initially nominally about 37 K (67°F) cooler than the hot leg. The break flow in the cold leg remains subcooled liquid until the cold leg saturates, which occurs a significant time period after the hot leg saturates. In the hot leg break, the hot leg is one of the first components to saturate thereby changing the break flow from a high density subcooled flow to a relatively low saturated flow thereby keeping more fluid in the system longer. Figure 31 compares the system mass inventory as a function of time for hot leg[7] and cold leg[8,9] breaks (2.5% break experiments) confirming that the hot leg breaks allow more mass to remain in the system. In addition, with a hot leg break the steam binding problem and resulting manometric core liquid level depression do not occur as the relief path from the core to the break is not blocked.

The response of a 5% pump suction break[35] was found to be very similar to the response of a 5% cold leg break. Such phenomena as depressurization rate (Figure 32), seal formation, and break uncovery are similar for pump suction breaks and cold leg breaks. Figure 33 compares the vessel collapsed liquid level showing almost identical liquid levels for pump suction breaks and cold leg breaks. The fluid density in the pump suction and cold leg are sufficiently similar to allow similar break mass flow.

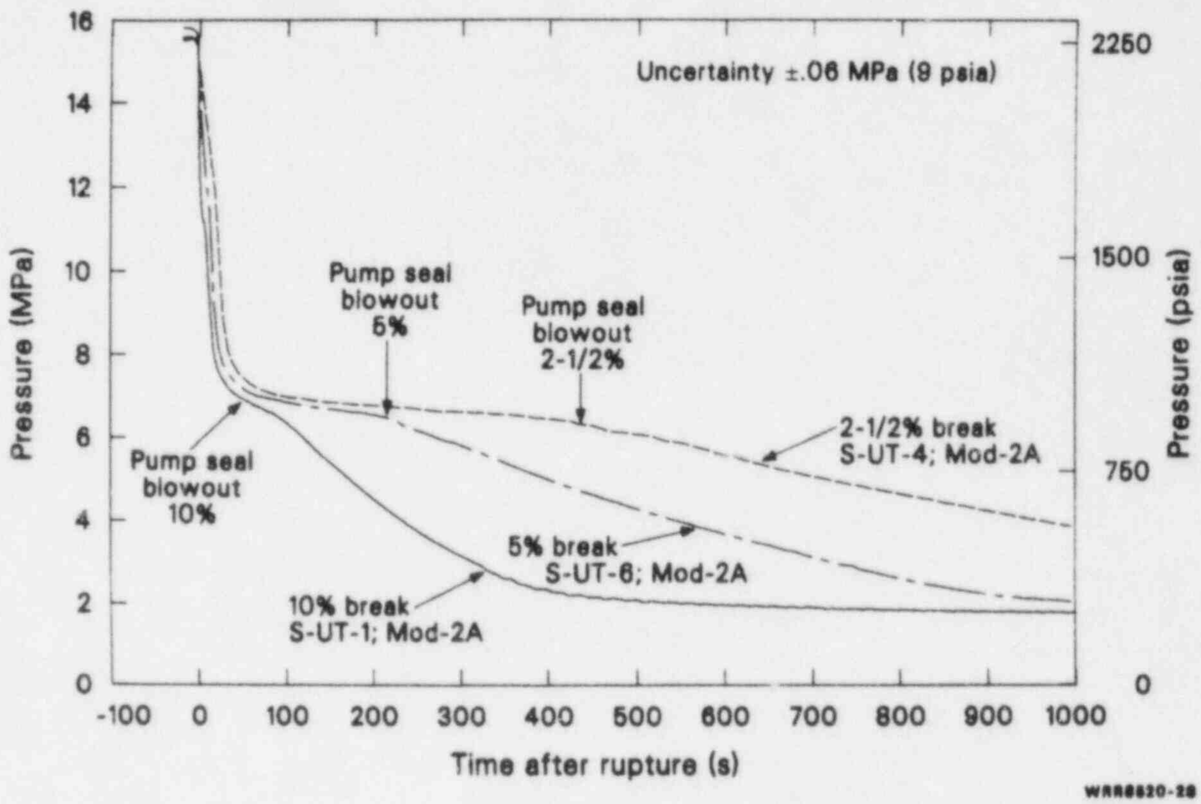


Figure 28. Primary pressure for 2-1/2, 5, and 10% cold leg breaks.

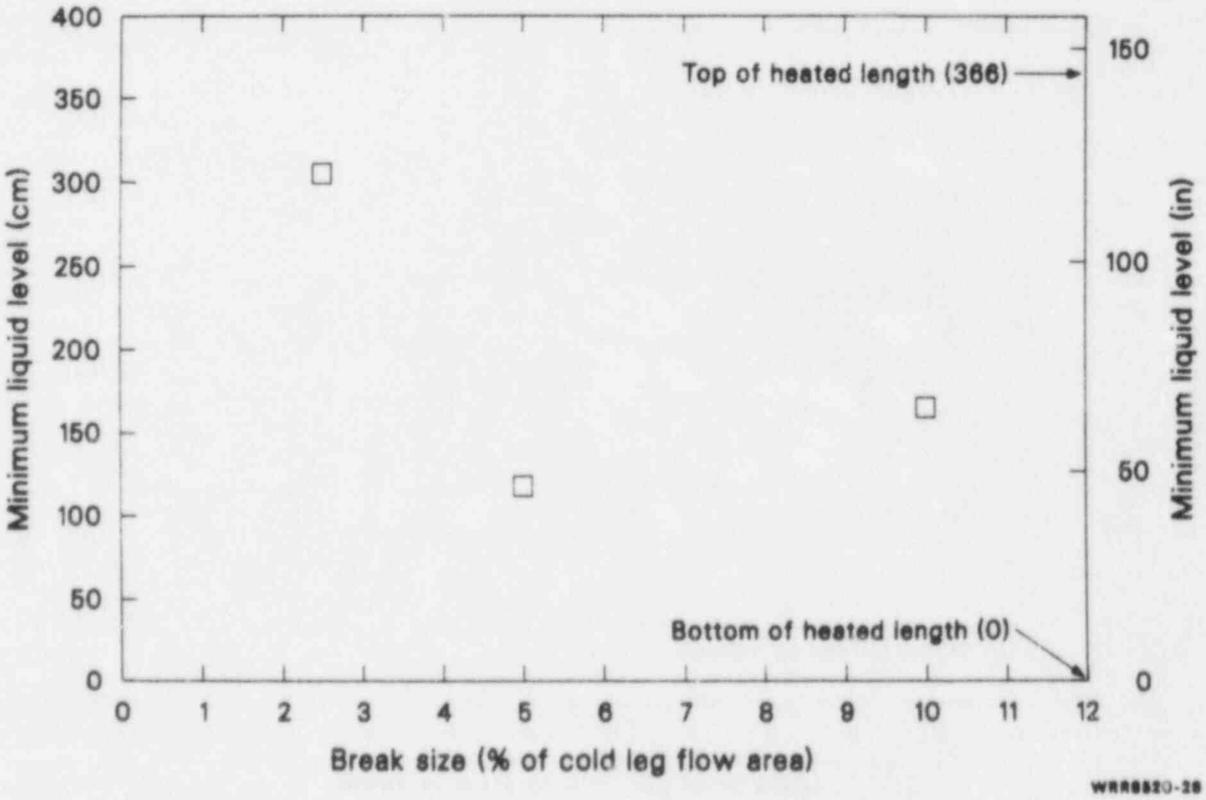


Figure 29. Small break LOCA severity as a function of break size.

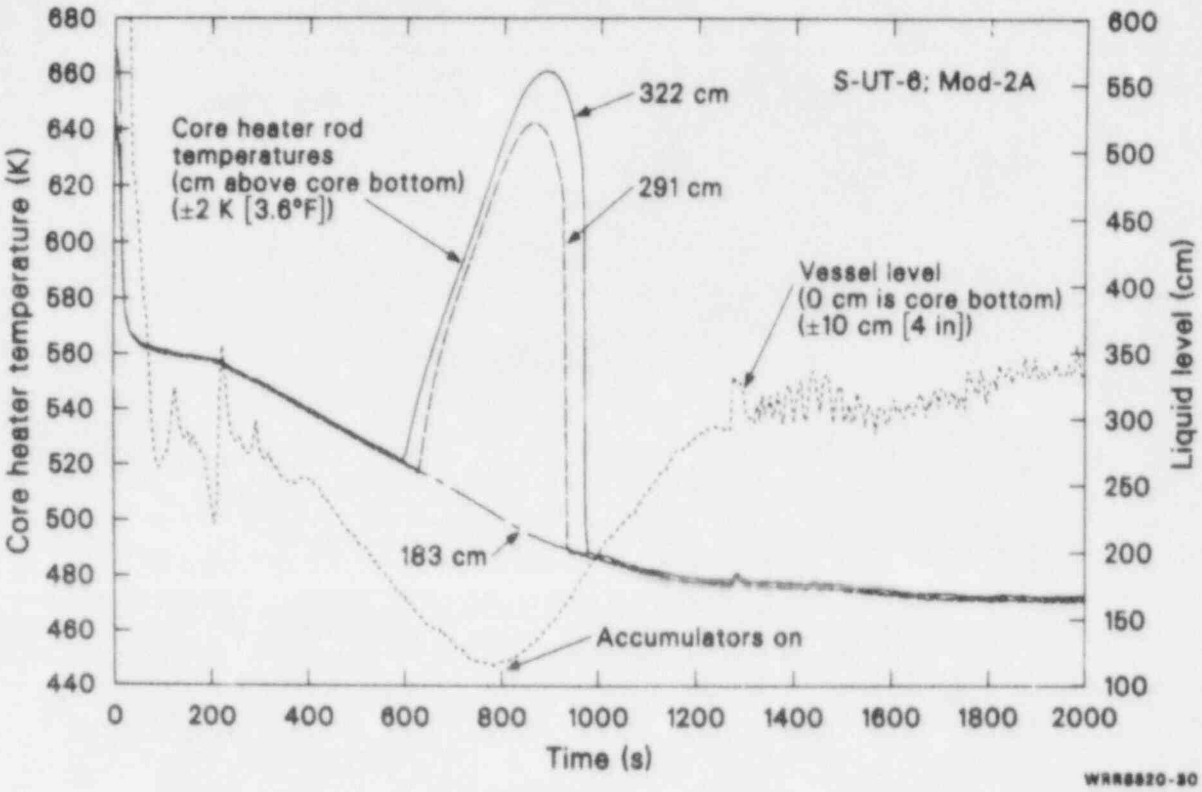


Figure 30. Heater rod cladding temperatures illustrating temperature excursion in upper part of the core (5% break).

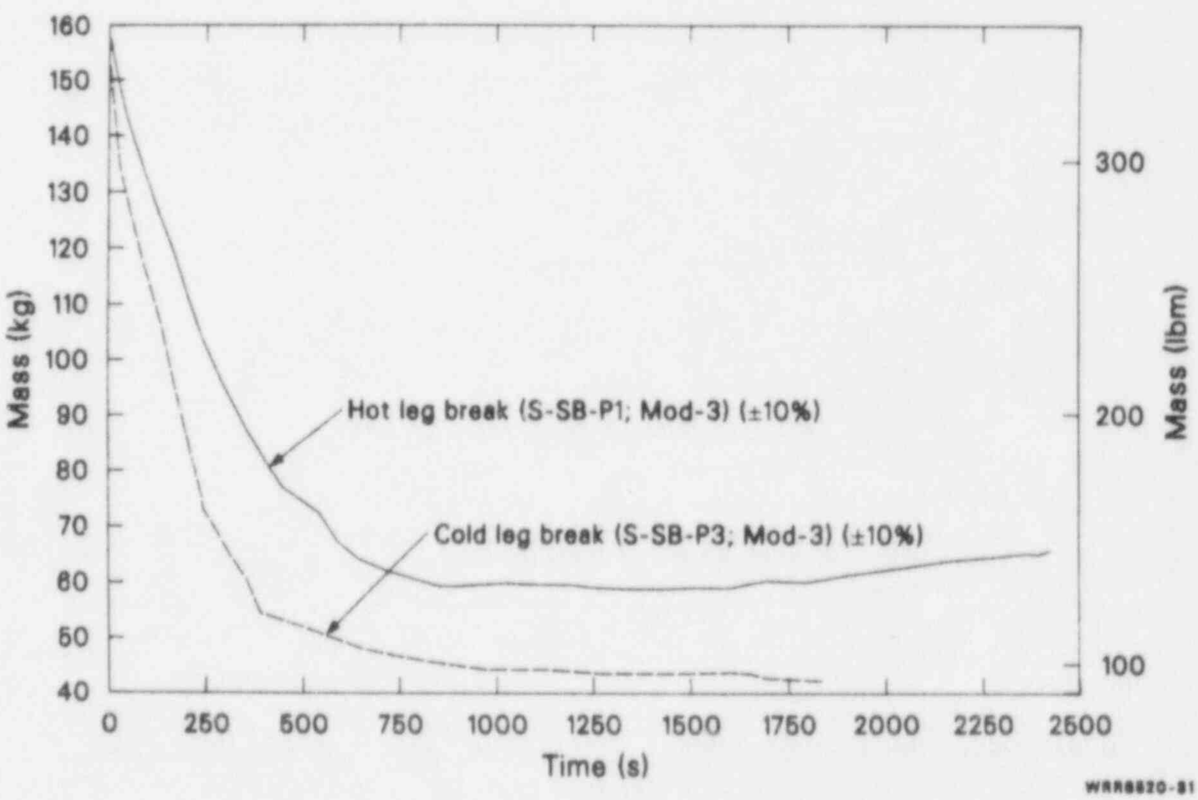


Figure 31. Comparison of system mass inventory for a 2-1/2% cold leg small break experiment and a 2-1/2% hot leg small break experiment.

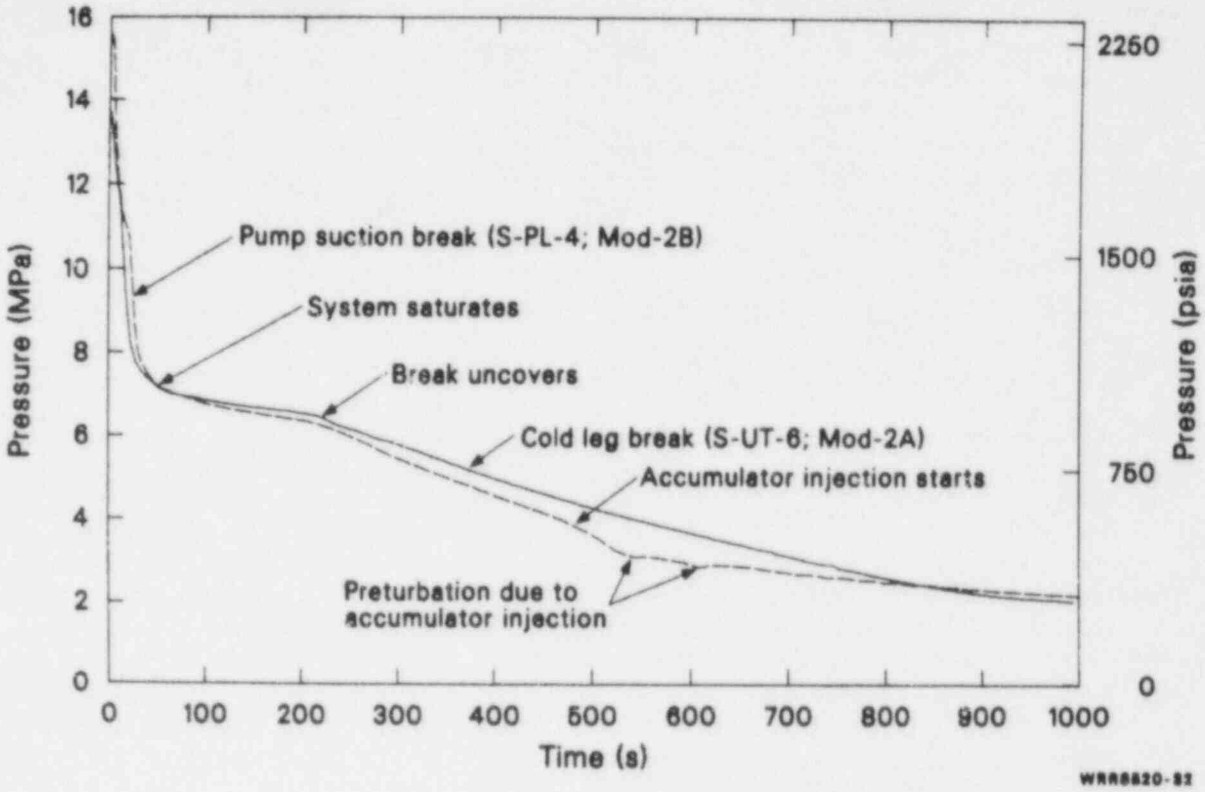


Figure 32. Comparison of primary pressure during pump suction and cold leg break.

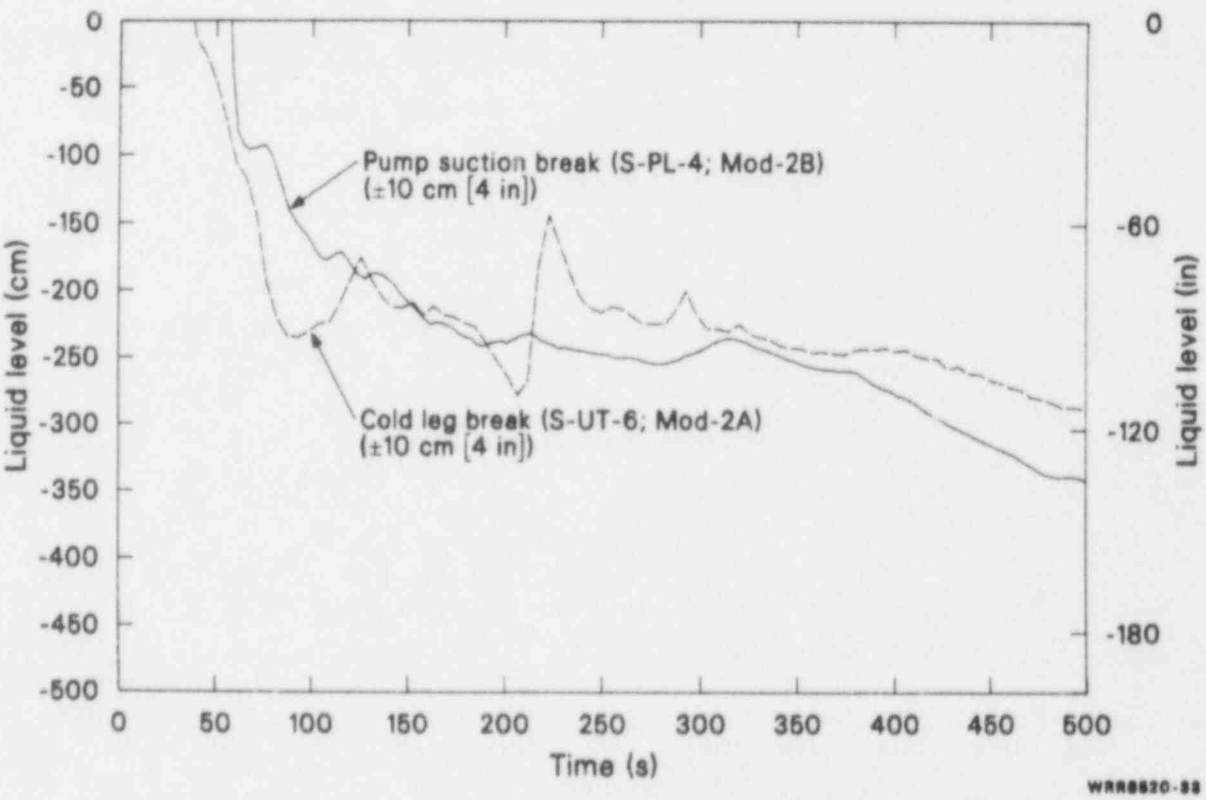


Figure 33. Comparison of vessel liquid levels during a pump suction break and a cold leg break.

Effect of Downcomer to Upper Head Bypass Flow. Commercial PWRs have an initial core bypass from the downcomer inlet annulus to the upper head varying between 0.4 and 4% of vessel core mass flow rate. During a small break transient the formation of liquid seals and resulting core level depression can be greatly affected by the amount of core bypass between the downcomer and vessel upper head. As discussed earlier, core decay heat produces steam in the core that has a pressurization effect in the vessel upperhead and hot legs because of the liquid seal in the pump suction. As the pressure builds up, both the downflow side of the pump suction and core liquid level are depressed. A facility with a higher bypass flow (lower hydraulic resistance between the upper head and downcomer inlet annulus) can minimize the core boiling induced pressurization effect because of an enhanced steam relief path to the break.

To address this issue, two 5% SBLOCAs were performed in the Semiscale Mod-2A with nearly identical boundary conditions but with different core bypass flows. One experiment had a core bypass flow of 4.0%[24,25] and the other 1.1%.[28] Both experiments showed the usual pump suction seal formation and core level depression. However, the core level depression was greatly enhanced for the lower bypass flow case as shown in Figure 34. For the low bypass flow case the severe core level depression lead to a core rod heat up that was mitigated by the loop seal clearing (indicated in Figure 34 as an increase in core liquid level). The vessel primary depressurization rate was not greatly affected by a different core level depression, as shown in Figure 35. Both experiments showed an increase in depressurization rate associated with the pump seal clearing and core uncovery as discussed earlier. The 1.1% core bypass case showed a faster core boil-off rate following the pump suction seal clearing (Figure 34). This is partly attributed to a slightly higher secondary pressure and thus less effective heat sink for the 1.1% bypass case (see Figure 36); nevertheless accumulator pressure setpoints were reached in both experiments resulting in a mitigation of core heatup, as shown in Figure 37.

Effect of Pump Operation on Small Break Phenomena. Pump operation during small break LOCAs affects the system depressurization rate and primary system coolant inventory by influencing conditions upstream of the break. Break experiments of 2-1/2% involving both pumps *on* and

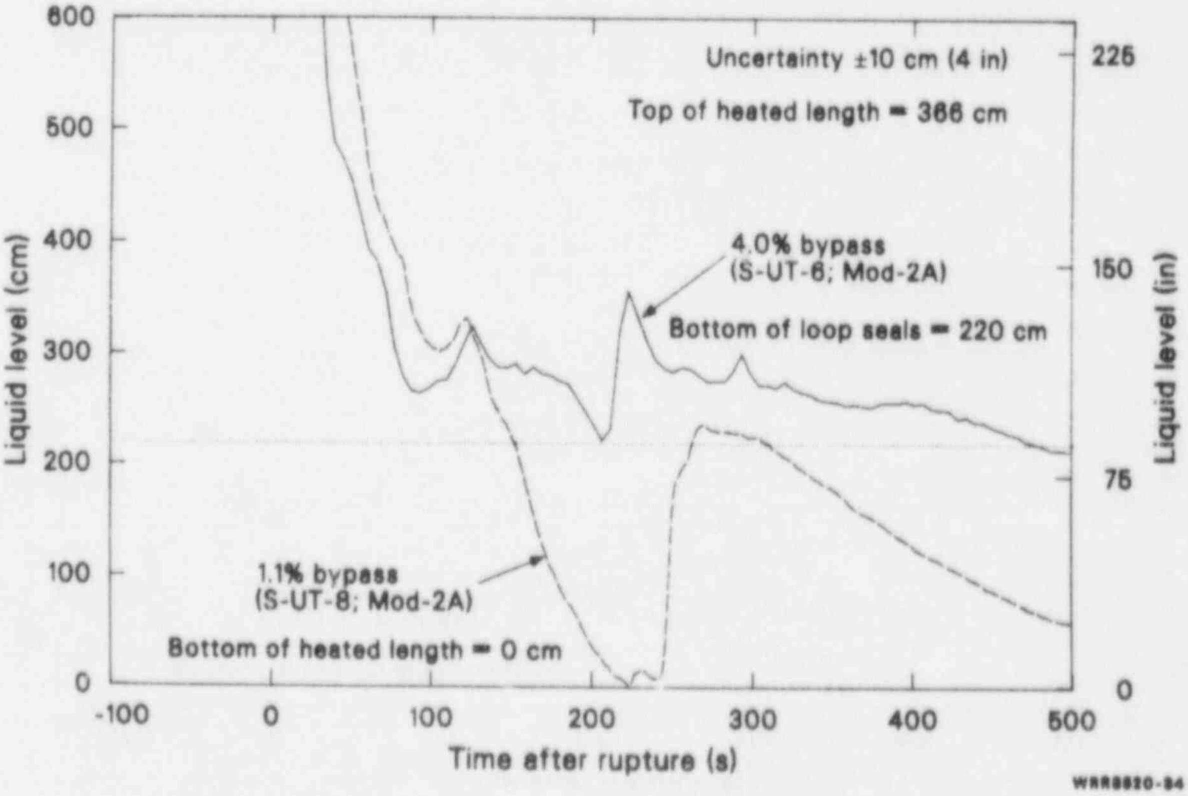


Figure 34. Vessel collapsed liquid level for a 4 and 1.1% core bypass flow area (5% cold leg break).

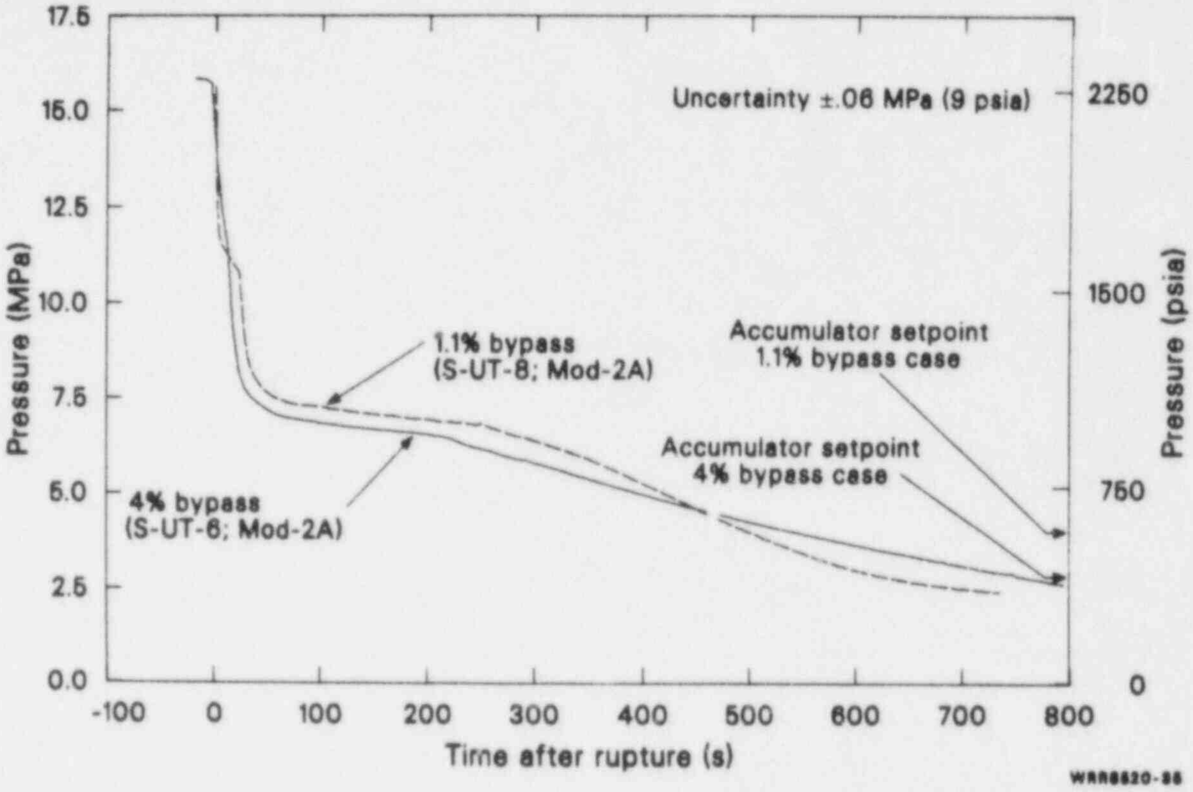


Figure 35. Vessel upper plenum pressure for a 4 and 1.1% core bypass flow (5% cold leg break).

pumps *off* were performed in both the Semiscale Mod-3[7] and the LOFT facility. For the pumps off case, the pump power was tripped at the low primary pressure trip [(12.48 MPa) (1809 psia)]. The combined results of these experiments showed that turning the primary pumps *off* at a low pressure trip tended to maximize the amount of primary coolant remaining in the system. As an additional combined result of Semiscale and LOFT experiments, injection location has as large an affect as pump operation.

During the LOFT small break experiment with the pumps *off*, stratification of fluid occurs in the system with steam at high points and liquid at low points. Consequently as the blowdown progresses, the cold leg voids in a stratified manner. Eventually the break uncovers and the break flow becomes mostly steam with little mass expulsion. However, when the primary pumps are left running in the LOFT system during a small break transient, the fluid in the cold leg tends to be on the average a higher density homogeneous mixture caused by the churning effect of the pump operation. This higher density fluid allows for a higher break flow and thus a more rapid depletion of system inventory. The LOFT experiments showed that more mass remained in the system during the transient with the pumps *off*.

Considering the whole small break transient, the trend found in the LOFT data for more mass retention for early pump trip was not repeated in the Semiscale simulations, as shown in Figure 38. Not only is the relationship of system mass retention for pumps *on* or pumps *off* not the same for Semiscale and LOFT but the magnitude is considerably different. First, the Semiscale system mass retention is relatively independent of pump operation. The Semiscale pump degradation in two-phase conditions is much higher than the LOFT pump (which more closely simulates a PWR head degradation). When the Semiscale pump is left running during the small break, the capability to impart energy to the fluid is so degraded that it behaves as if stopped. It would be expected then that the LOFT pumps *off* case would agree with either the Semiscale pumps *on* or *off* case; however, the comparison is also not close. Figure 38 shows that the magnitude of the amount of mass retained in the LOFT pumps *off* experiment was considerably greater then either of the Semiscale experiments.

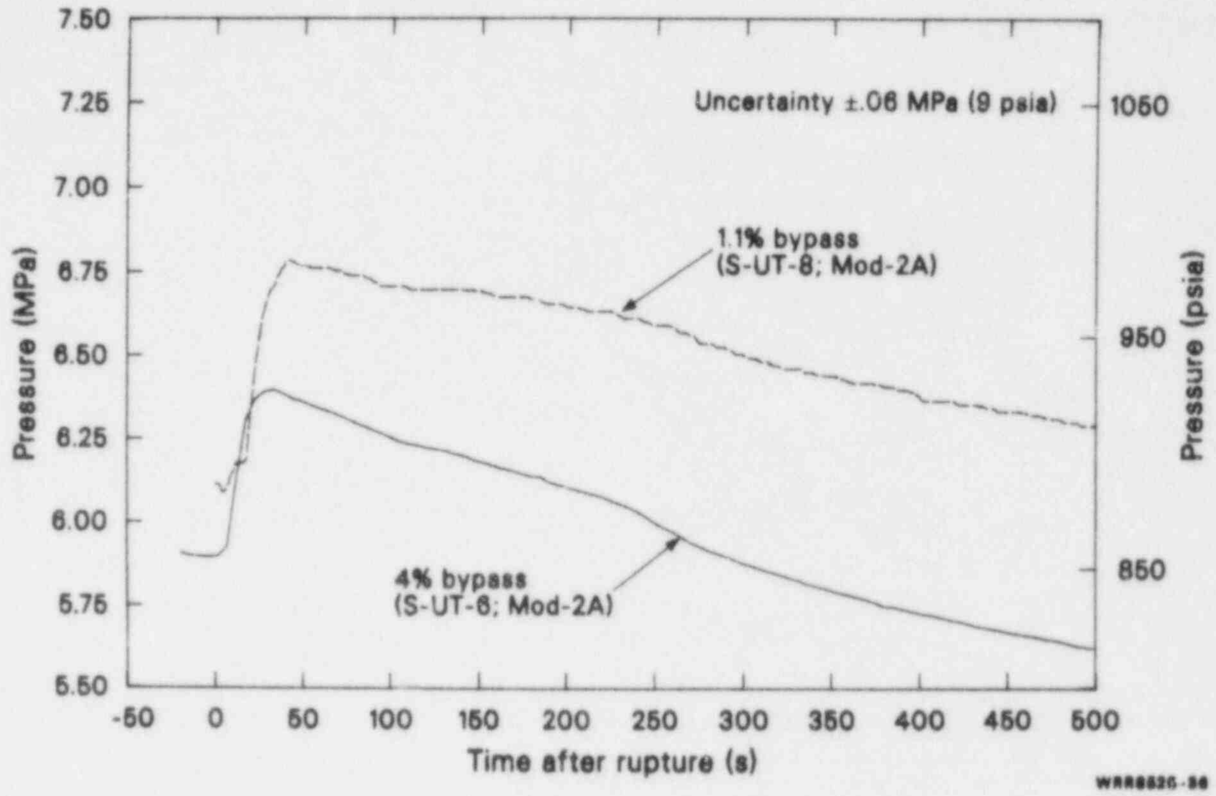


Figure 36. Broken loop steam generator secondary pressure for a 4 and 1.1% core bypass flow (5% cold leg break).

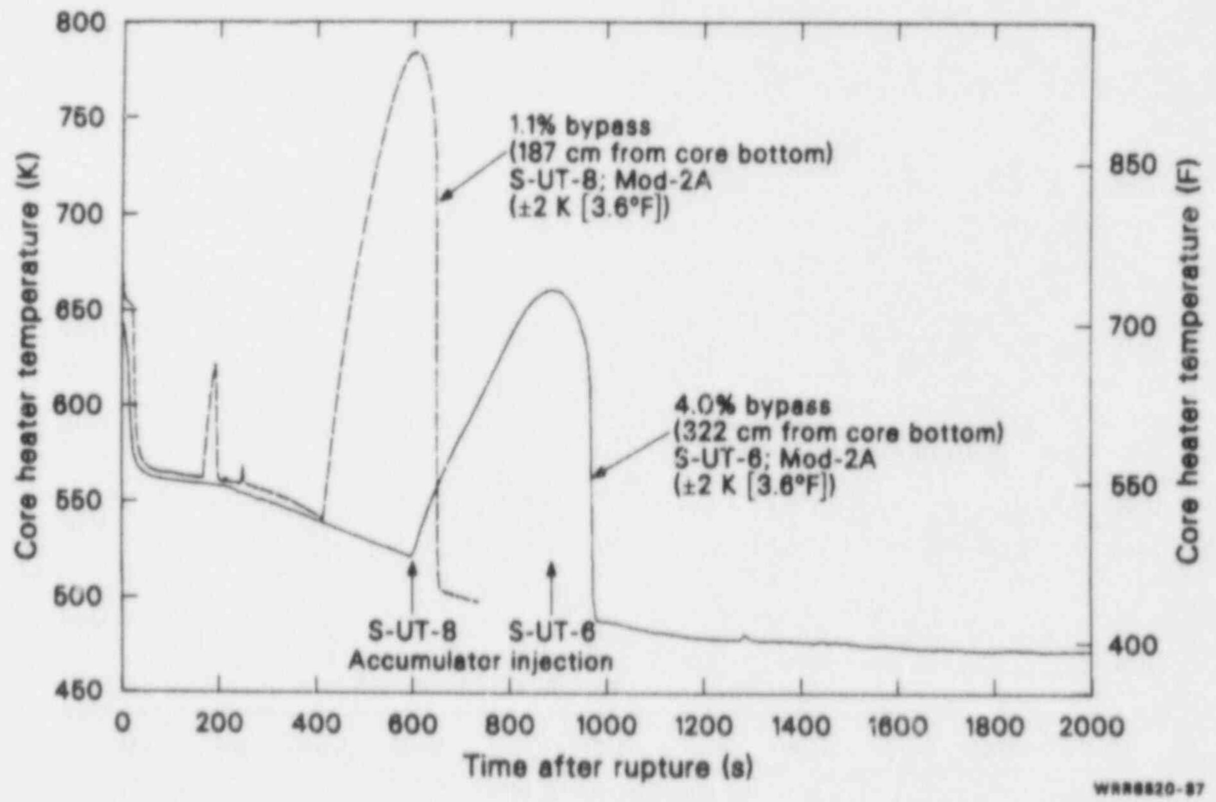


Figure 37. Core heater rod response at the midcore for the 1.1% core bypass and near the top of the core for the 4% core bypass (5% cold leg break).

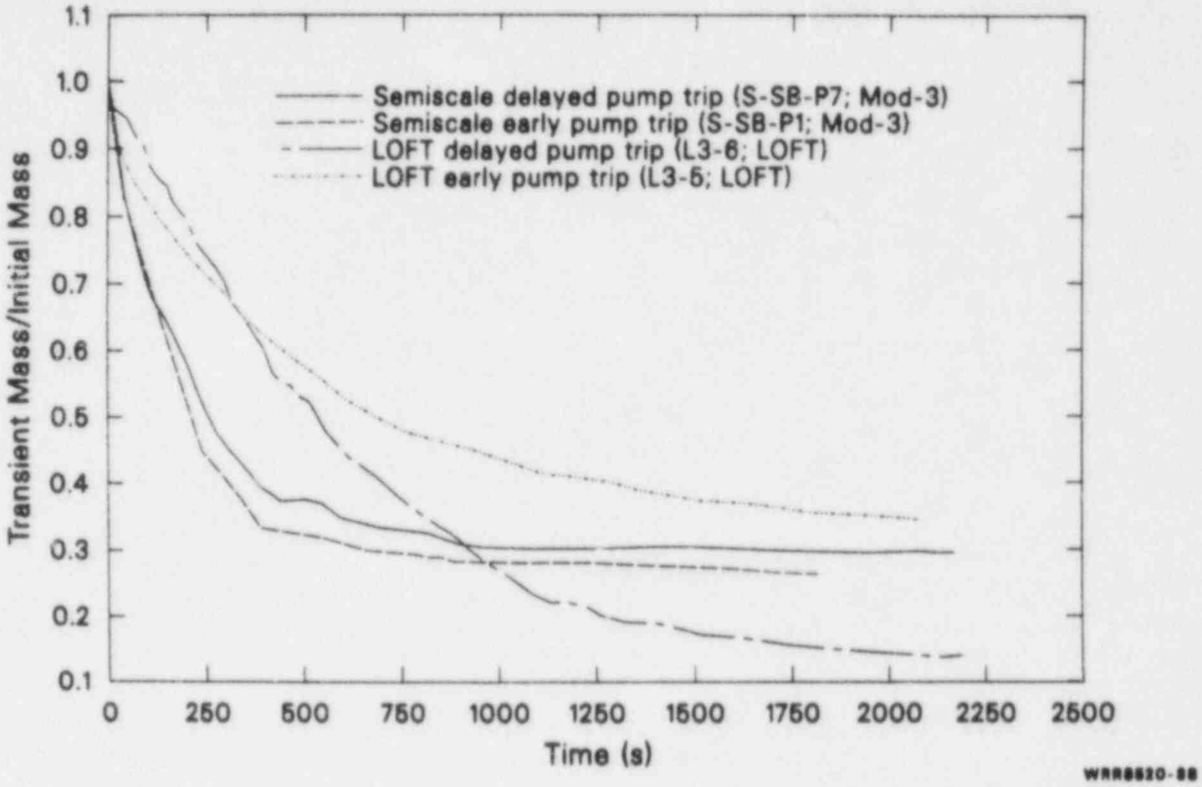


Figure 38. Comparison of LOFT and Semiscale normalized system mass during small breaks for cases with pumps on and with pumps off (2-1/2% breaks).

This difference had nothing to do with pump operation but rather with ECC injection location. In the Semiscale experiments ECC was injected in the cold leg just upstream of the break and in the LOFT experiment ECC was essentially injected into the downcomer. By injecting fluid just upstream of the break the fluid density at the break remained high resulting in higher break flow rates. For the LOFT experiment with pumps *off*, ECC fluid had little effect on fluid conditions just upstream of the break and consequently on break flow; therefore the system mass depletion was governed by a stratified removal of fluid. In Figure 38 the Semiscale pumps *off* case actually showed less mass retention than the pumps on case. This was because with the pumps *on*, cold ECC was mixed with system fluid and forced to the break location, thus presenting a higher density fluid at the break promoting a higher break flow. In short, the importance of ECC injection location in these Semiscale test results was greater than the effects of the pump operation.

In summary, the LOFT and Semiscale results regarding pump operation during a small break LOCA are contradicting but when combined help to understand which pump operation maximizes system mass inventory during the transient. The LOFT results clearly show that the pumps *off* case resulted in more mass retention; however, the tests involved downcomer ECC injection which minimized the effect of ECC on fluid conditions upstream of the break. The Semiscale experiments demonstrate the importance of ECC on upstream fluid conditions but due to a too rapid degradation in pump operation the energy transfer to the fluid due to the pump does not correctly model commercial PWR behavior.

Effectiveness of Upper Head Injection (UHI) During SBLOCA. A series of SBLOCA experiments was conducted in the Mod-2A to investigate the influence of upperhead ECC injection on transient response.[18] Several commercial PWRs employ this ECC technique that was primarily designed to offset large break transients; therefore, the Semiscale experiments are designed to investigate the effectiveness of UHI impact. For a variety of break sizes (2.5, 5, and 10%) it was found that the effect of UHI injection on the transient signature response was minimal. The extra coolant mass injected, during the UHI experiments was almost exactly offset by an increased break discharge. Basically UHI involves accumulator injection at 8.6 MPa (1247 psig) primary pressure with a total volume of water injected equal to approximately the vessel upperhead.

The overall depressurization signature was only slightly affected by the presence of UHI as shown on Figure 39. This was because all of the usual small break phenomena such as pump seal clearing and break uncovery occur also for tests involving UHI. For all break sizes, there was a slightly higher depressurization rate during the period of UHI injection. This higher rate resulted from the condensation of vapor by the cold accumulator water [(injected at 8.6 MPa) (1247 psia)]. The vessel upper head remained at a higher collapsed liquid level during the period of UHI injection as shown in Figure 40. The refilling of the upper head starting at about 220 s for the 5% break case is associated with pump seal clearing and rapid decrease in primary pressure. This caused an increase in the differential pressure between the upperhead accumulator tank and the vessel upperhead rapidly increasing the flow into the upperhead until accumulator depletion. Even though the upperhead drain characteristics were different for UHI and non-UHI experiments the overall system inventory was similar at any point in time. Break flow was increased by an amount equal to the UHI injected, resulting in similar mass inventories for the two cases. Figure 41 compares the integrated break mass flow showing an increased break flow for the UHI cases. Also shown in Figure 41 is the total mass of UHI that about equals the differential in integrated mass flow for UHI and non-UHI experiments. The reason for the increased break flow was a longer time to break uncovery and increased subcooling at the break. Figure 39 showed that break uncovery was delayed for the UHI experiments (increased depressurization associated with break uncovery). Figure 42 indicates that the vessel collapsed liquid levels are essentially identical following vessel accumulator injection for all break sizes. It is significant to point out that during the period of minimum core level for the worst case break (5% case), UHI caused an improved margin for core cooling, as shown in Figure 42 even though the overall mass in the system was identical as indicated in Figure 41.

Operator Recovery Procedures During SBLOCA. During a SBLOCA a PWR operator has a variety of means available to recover the plant. Operators can control primary pressure by using:

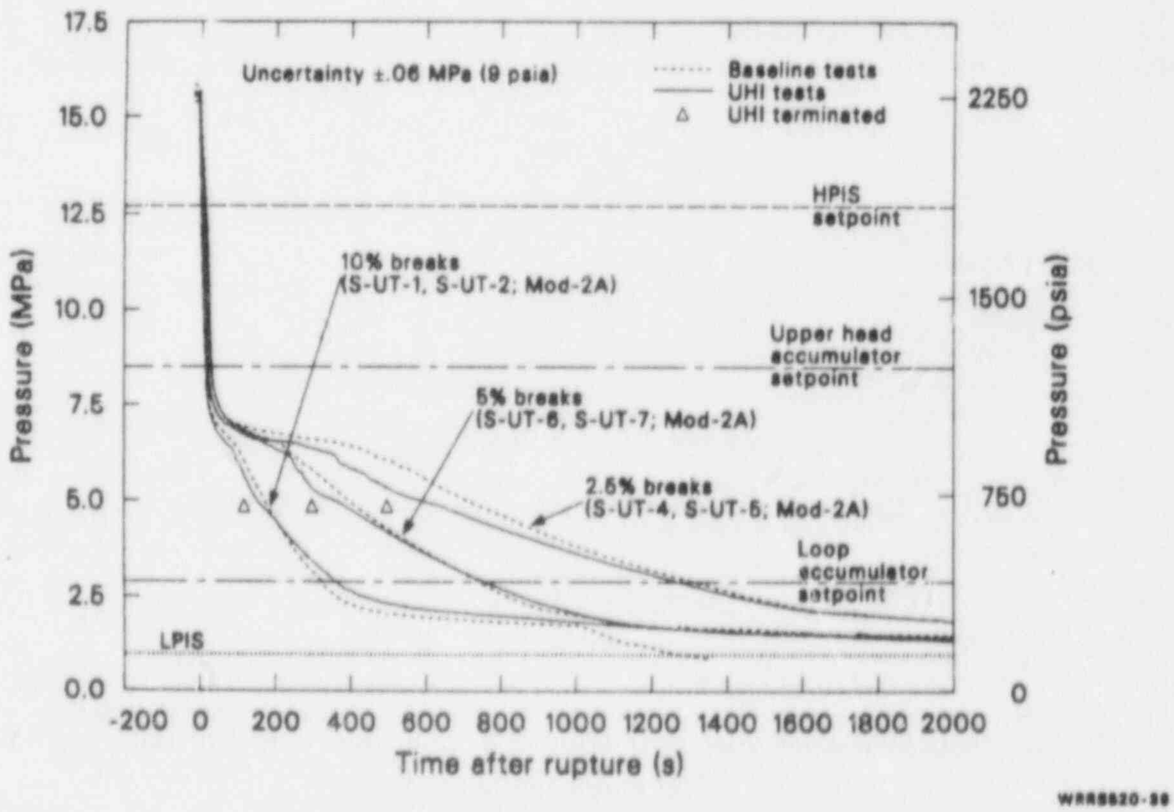


Figure 39. Primary pressure for 2-1/2, 5, and 10% breaks, with and without UHI.

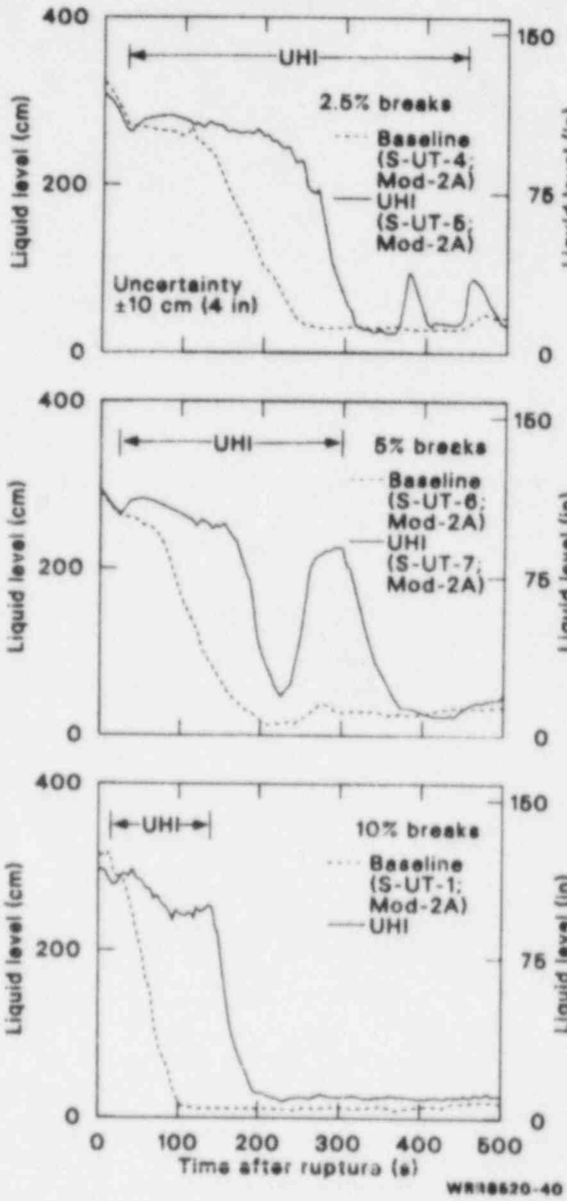


Figure 40. Collapsed liquid level in the vessel upper head for 2-1/2, 5, and 10% breaks, with and without UHI.

combinations of secondary feed and steam, manually operated HPIS (ECC injection), primary feed and bleed, pressurizer heaters, and pressurizer auxiliary spray.

During a loss of primary coolant, because of a cold leg break for the break spectrum 2.5 to 10%, normal automatic recovery procedures are found to be adequate. Low pressure injection (LPIS) set points can be reached without uncontrolled core uncovery. However, for ultra small breaks such as occur for PWR stuck open PORV or steam generator tube rupture (on the order of 0.4%), the primary pressure can remain well above ECC (accumulator, LPIS) pressure set points [6-7 MPa (870-1015 psia)], while the vessel inventory depletes due to an overall system mass redistribution (recall in the discussion of TMI signature response the pressurizer filled with liquid and the vessel was depleted). During these cases operator action is required to assure safe reactor response. Semiscale investigated the relative effectiveness of such recovery procedures during the steam generator tube rupture test series. Recovery included primary mass inventory control and loop fluid subcooling control. Normal recovery combinations suggested by typical United States PWR emergency operation procedures were followed. These included unaffected loop feed and steam, using atmospheric dump valve and auxiliary feedwater, primary system feed and bleed, using pressurizer PORV operation and safety injection; pressurizer auxiliary spray; pressurizer internal heaters; and SI operation. During the Semiscale tube rupture experiments conditions changed from subcooled conditions (prior to the break) to saturation conditions and then back to subcooled conditions as the operator used the various techniques to control mass inventory and subcooling. This type of behavior can be observed on plots called ATOG (Abnormal Transient Operation Guidelines) plots. As an example, Figure 43 shows a typical ATOG plot with the transition from subcooled conditions at full power, full flow conditions through the transient, back to subcooled conditions at natural circulation and pumps off.

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,[41] 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 that caused a compression of voids in the system. The operator would observe this on an ATOG plot (Figure 43) 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 by way of the break and, further, since SI had increased primary system pressure, the affected loop ADV cycled several times,

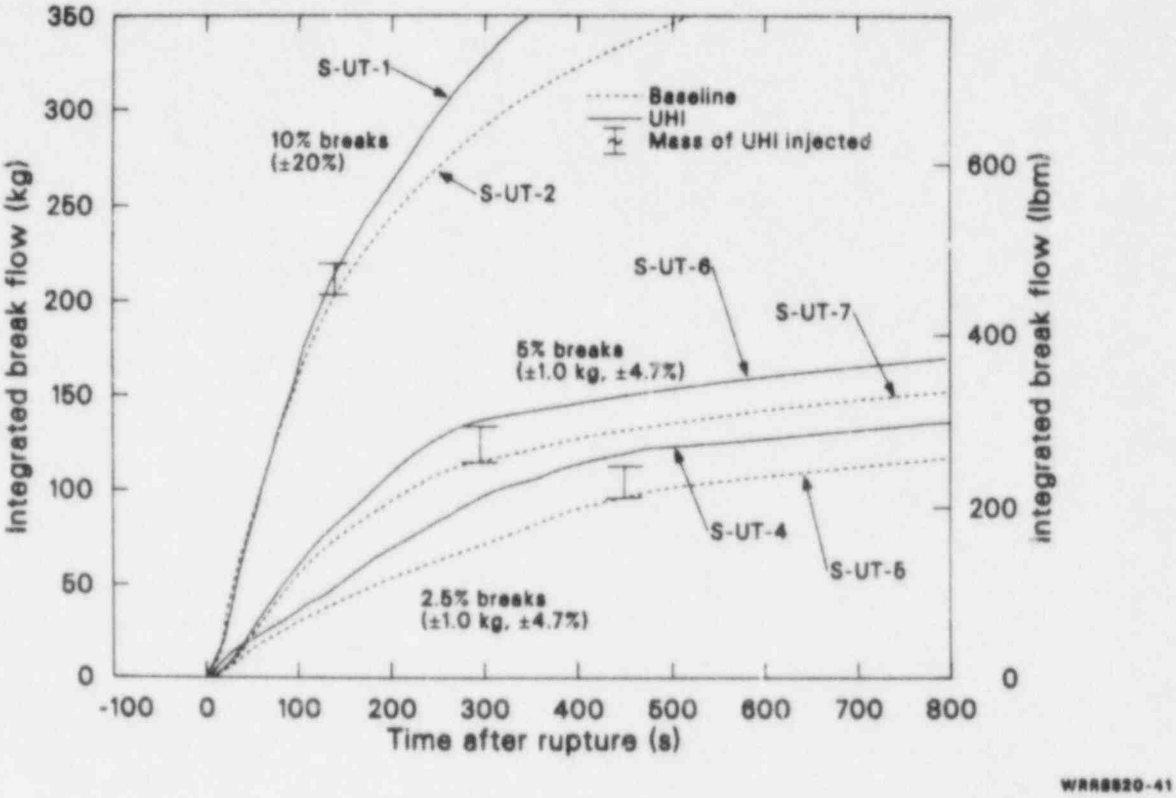


Figure 41. Integrated break mass flow for 2-1/2, 5, and 10% breaks, with and without UHI.

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.

The effectiveness of the various recovery techniques were assessed using the Semiscale Mod-2B tube rupture transients. The following discussion includes the effectiveness of steam generator feed and steam on pressure control and loop cooling, primary feed and bleed for inventory and pressure control, pressurizer auxiliary spray for pressure and inventory control, pressurizer internal heaters for pressure control and safety injection for inventory and pressure control.

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.[47] 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 rate of pressure decrease due to feed and steam is slower, as shown in Figure 44. 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

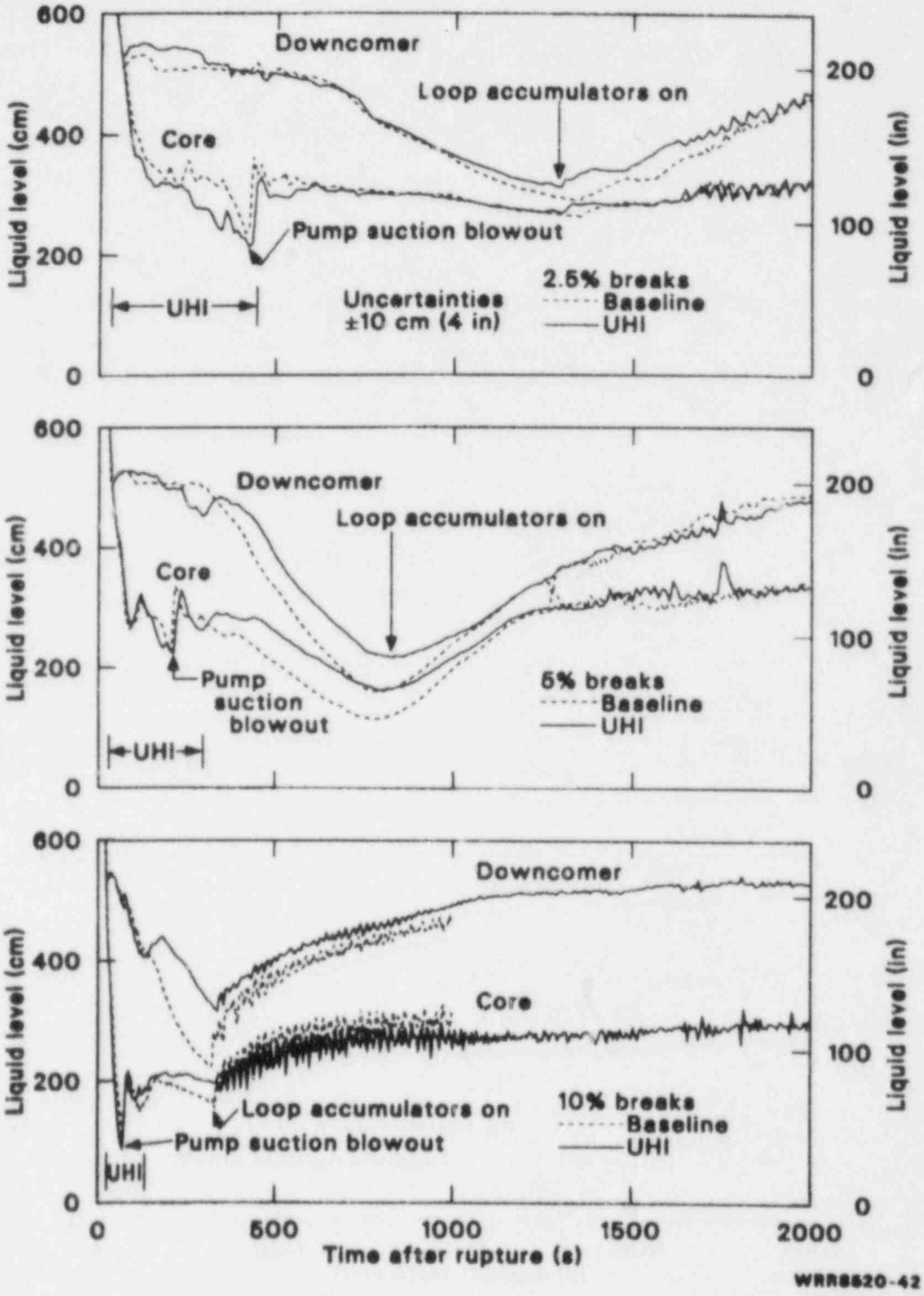


Figure 42. Collapsed liquid levels in the vessel (downcomer and core) for 2-1/2, 5, and 10% breaks, with and without UHI.

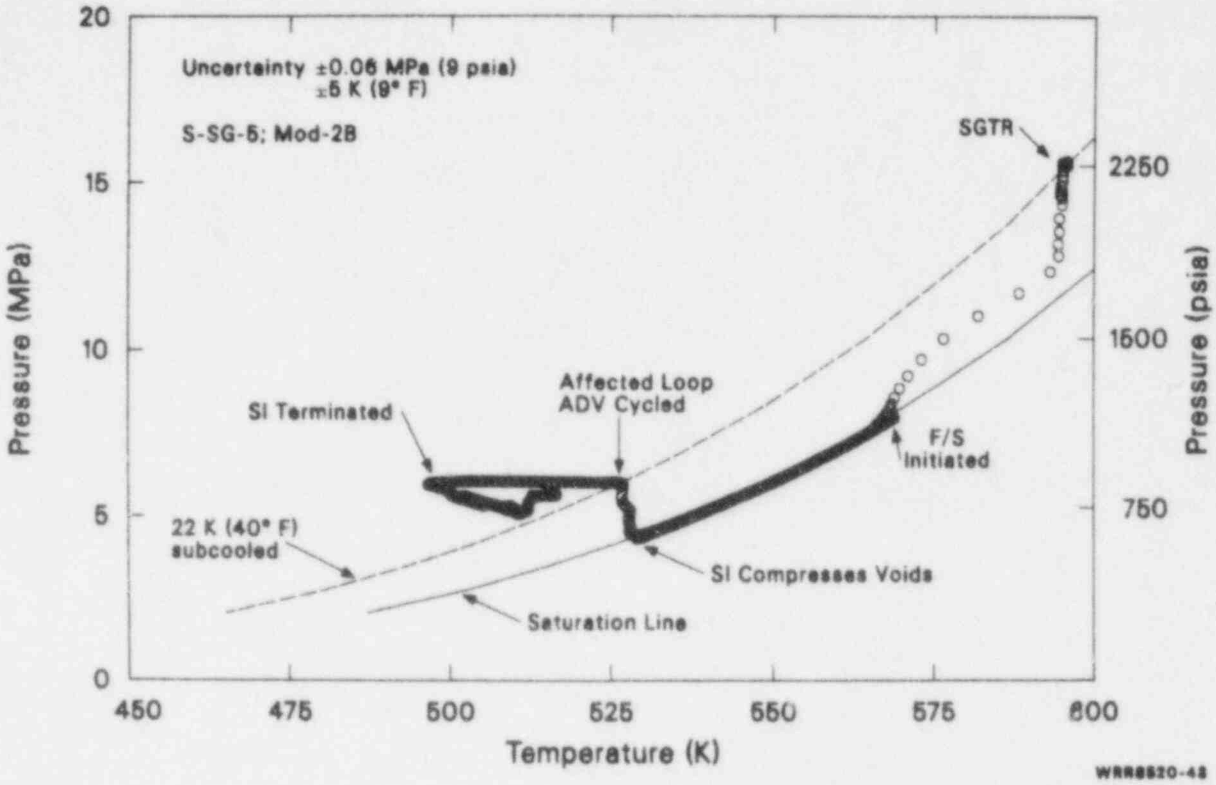


Figure 43. ATOG plot for a Semiscale five-tube rupture, with early feed and steam.

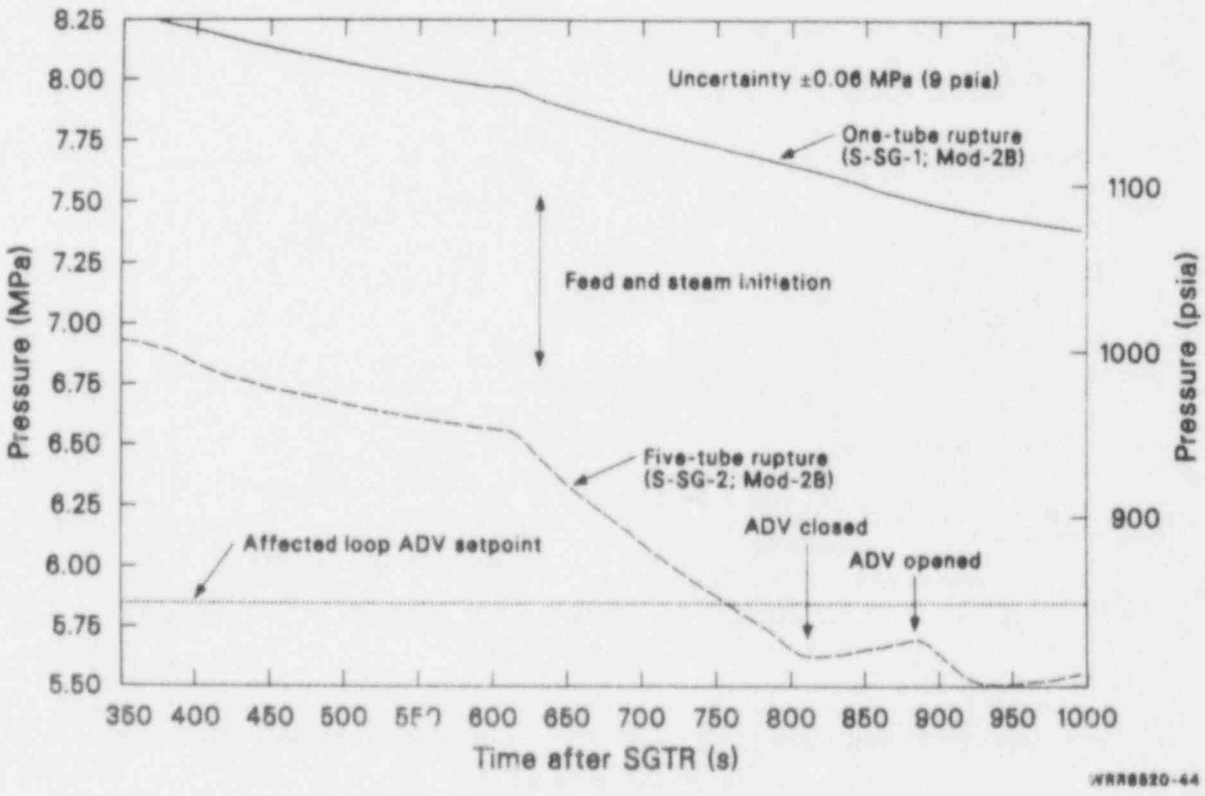


Figure 44. Primary system pressure response for unaffected loop feed and steam during recovery for a one- and five-tube rupture transient.

proportional to the mass rate of condensation; therefore, the increase in differential temperature caused by the feed and steam operation increased the depressurization rate.

PORV operation along with SI is effective in reducing primary system pressure below affected loop relief valve setpoints; however, there was a significant system mass inventory redistribution as shown on Figure 45.[47] 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. This is a similar response as was discussed for the TMI-2 stuck open PORV accident. The effectiveness of PORV operation for reducing primary system pressure decreased as the liquid level in the pressurizer increased. Once the pressurizer filled, open PORV had only a small effect on primary pressure control. This is because the primary fluid volume reduction due to PORV liquid flow is much less then the volume reduction from steam flow that occurred during early PORV operation.

Pressurizer auxiliary spray is effective in reducing primary pressure as long as the pressurizer maintains a steam space as shown in Figure 46.[47] As the pressurizer fills with condensed steam and the inflow of primary loop liquid mass, the effectiveness for pressure reduction is reduced. A droplet of auxiliary spray water is more likely to reach the pressurizer liquid pool without reaching saturation temperature by steam condensation if the pool level is higher. The initial spike in pressure upon spray initiation is attributed to steam and wall superheat as the subcooled spray drops evaporate upon contact with the fluid and walls. The evaporation of drops causes a pressure increase due to the volumetric increase involved. Once the superheat is removed continued spray causes an effective primary pressure reduction because the droplets remain liquid and condense saturated steam.

Pressurizer internal heaters are ineffective for increasing primary system pressure during a single-tube rupture in Semiscale, as shown in Figure 47.[47] 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.[47] The primary system pressurization due to SI increases the subcooling in the hot leg. Termination of SI during a tube rupture causes

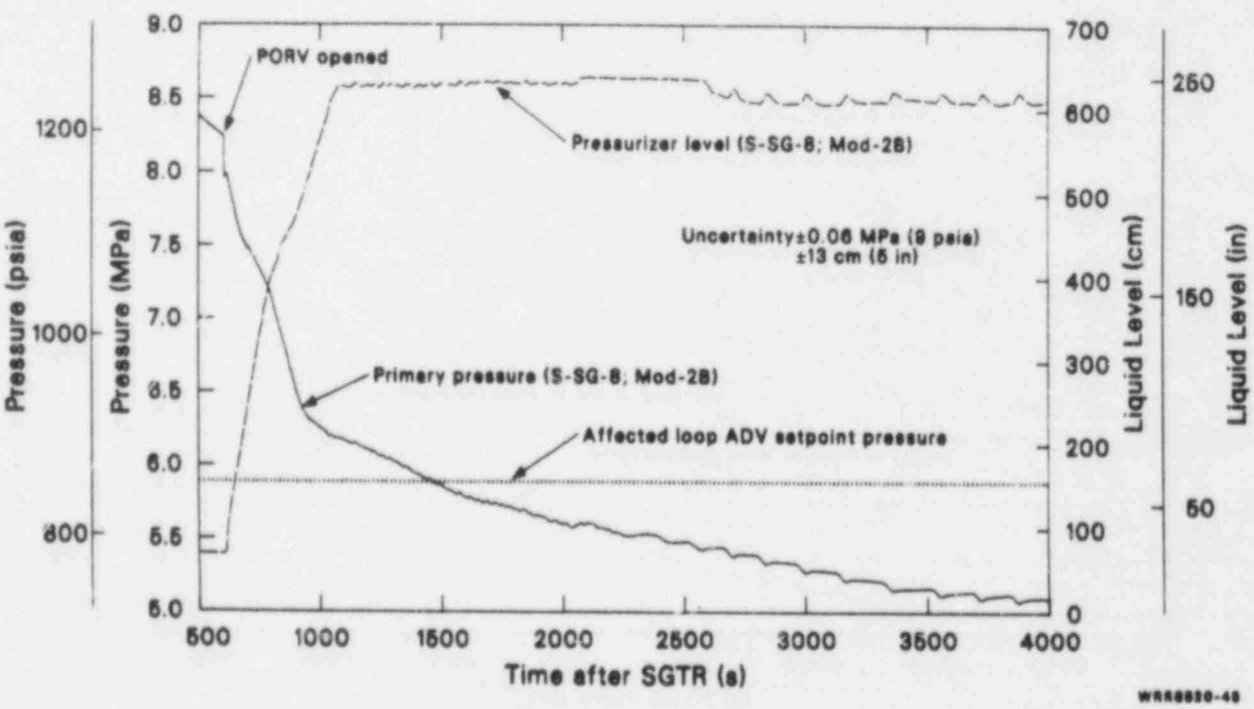


Figure 45. Primary system pressure and pressurizer collapsed liquid level during PORV operation for a single-tube rupture transient.

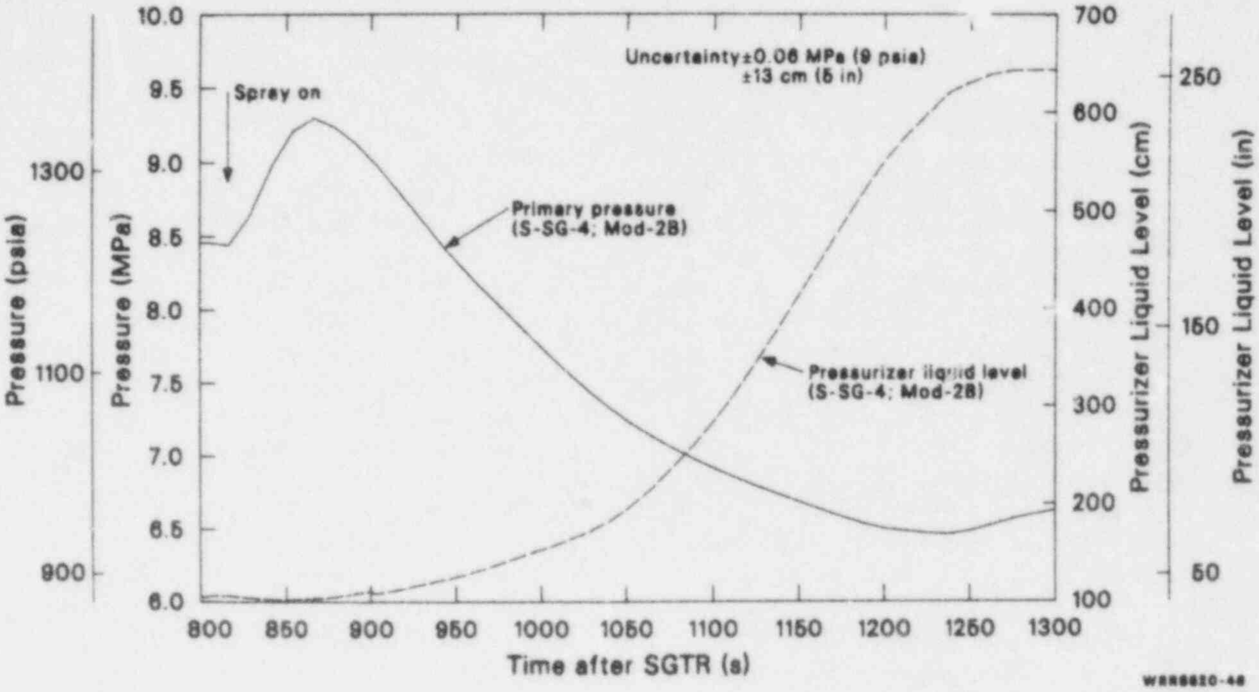


Figure 46. Primary system pressure and pressurizer collapsed liquid level during pressurizer auxiliary spray for a one-tube rupture transient.

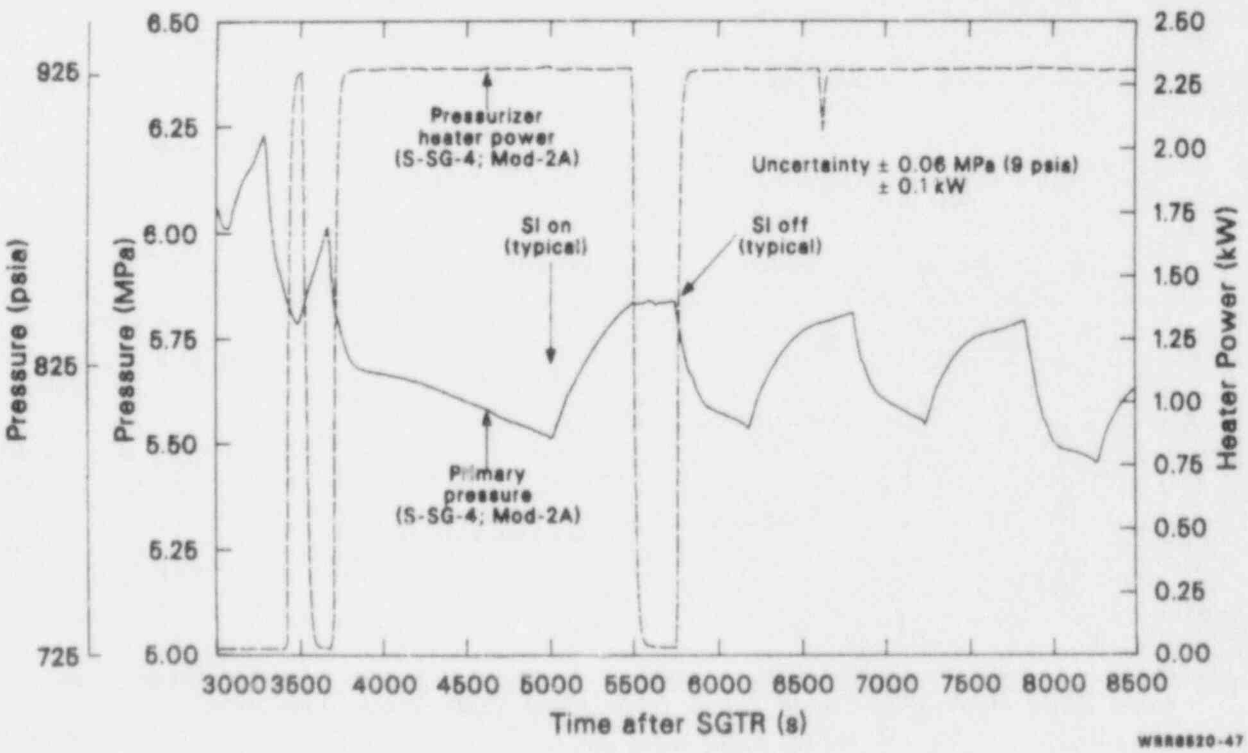


Figure 47. Pressurizer pressure and heater power during a cold side, one-tube rupture transient.

a lowering of primary system pressure because the continued break flow expands the voids in the system. Figure 48 shows that the pressure decrease accompanying the SI termination followed the perfect gas assumption.

Compounding Failures During Small Break. A great variety of compounding failures concurrent with a SBLOCA can be envisioned. Several examples of the compounding failures were examined in Semiscale testing. Already discussed was the TMI stuck open PORV, a failure that was the small break; however, this section discusses compounding failures that were investigated in Semiscale during a small break. Compounding failures for two types of small breaks were examined (pipe breaks and tube rupture breaks).

Compound Failure During a Pipe Break. Semiscale investigated a very small (0.4%) cold leg break with a complete loss of high pressure injection flow as the compounding failure. In this event, core liquid level boil-off and core heat up occurred due to decay heat before accumulator injection pressure setpoints were reached. For this extreme case (no HPIS), operator action was required to mitigate the core rod heat up as shown in Figure 49. The core rod heat up started at about 1200 s with the primary pressure well above the accumulator setpoints pressure of 4.2 MPa (609 psia). For this particular example the operators initiated a rapid unaffected loop feed and steam of the secondary that eventually brought the primary pressure to accumulator actuation pressure. Accumulator flow caused a partial increase of core liquid level as shown on Figure 50 and mitigated the first rod temperature excursion. At about 3500 s a second core heat up occurred due to continued core liquid boil-off. The primary depressurization rate at this time was essentially zero and accumulator pressure and primary pressure equalized resulting in zero accumulator flow and the core simply boiled off. Meanwhile, the steam generator secondaries boiled dry and the operators then used the PORV to stimulate a primary depressurization to induce further accumulator flow and eventually reach LPIS setpoints. The PORV flow decreased the primary pressure and stimulated a net positive flow of accumulator water into the primary that caused a vessel liquid filling trend and turnaround of core heater rod temperatures. Once the accumulators ran dry the core rod temperature again increased and the core power was manually tripped at about 945 K, about the time the LPIS pressure setpoint was reached.

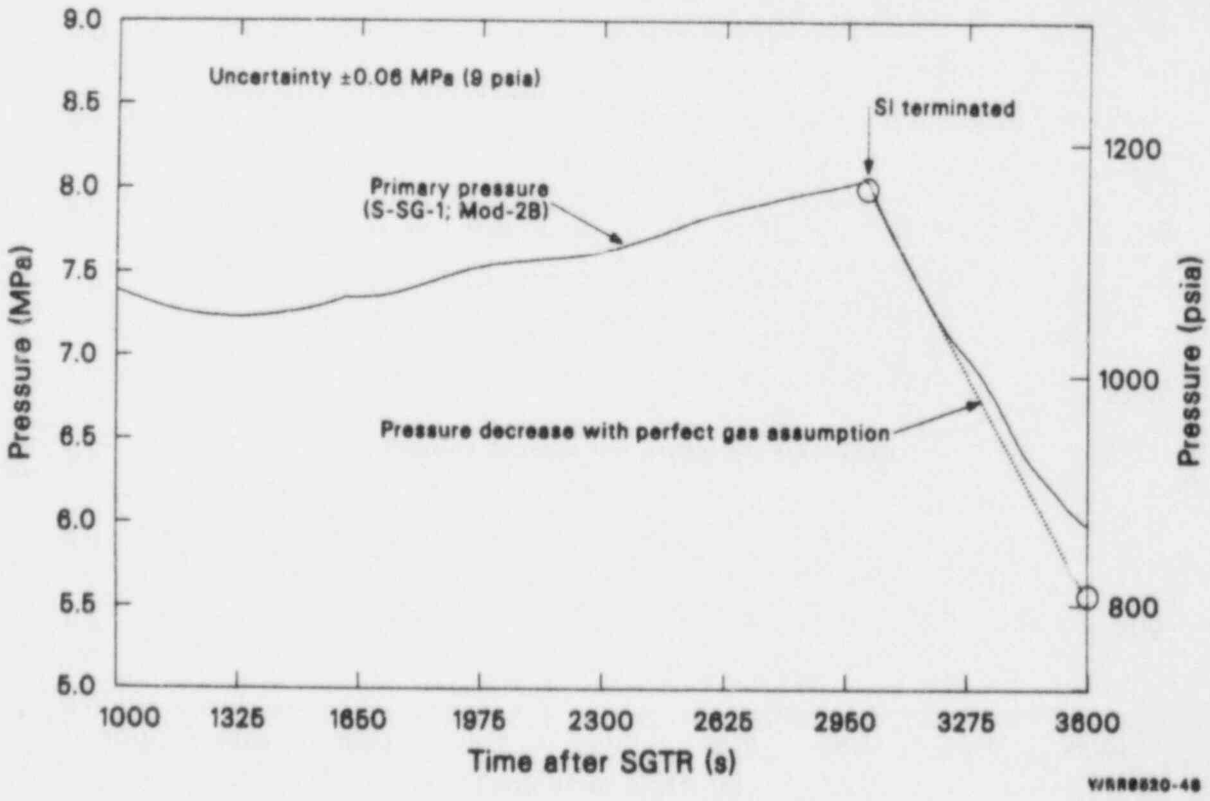


Figure 48. Primary system pressure during SI termination, with perfect gas assumption for a one-tube rupture transient.

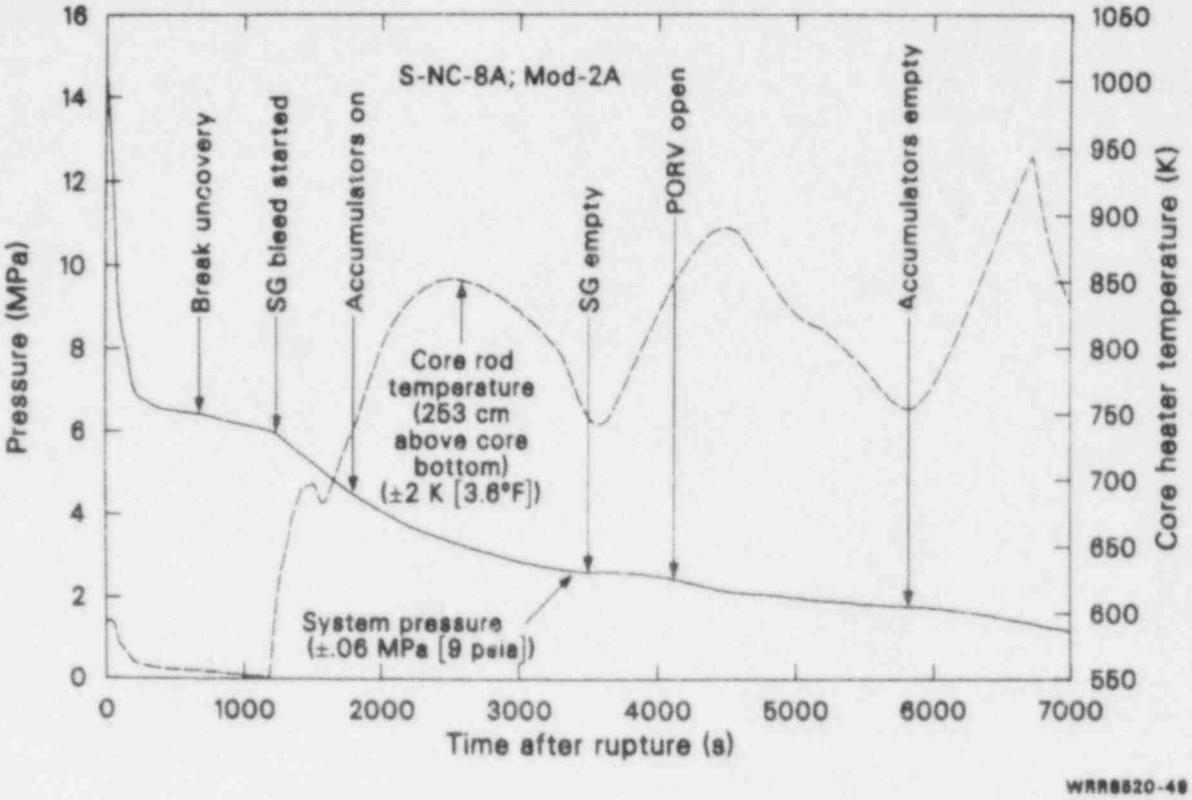


Figure 49. Comparison of system pressure and core rod temperature for 0.4% cold leg break.

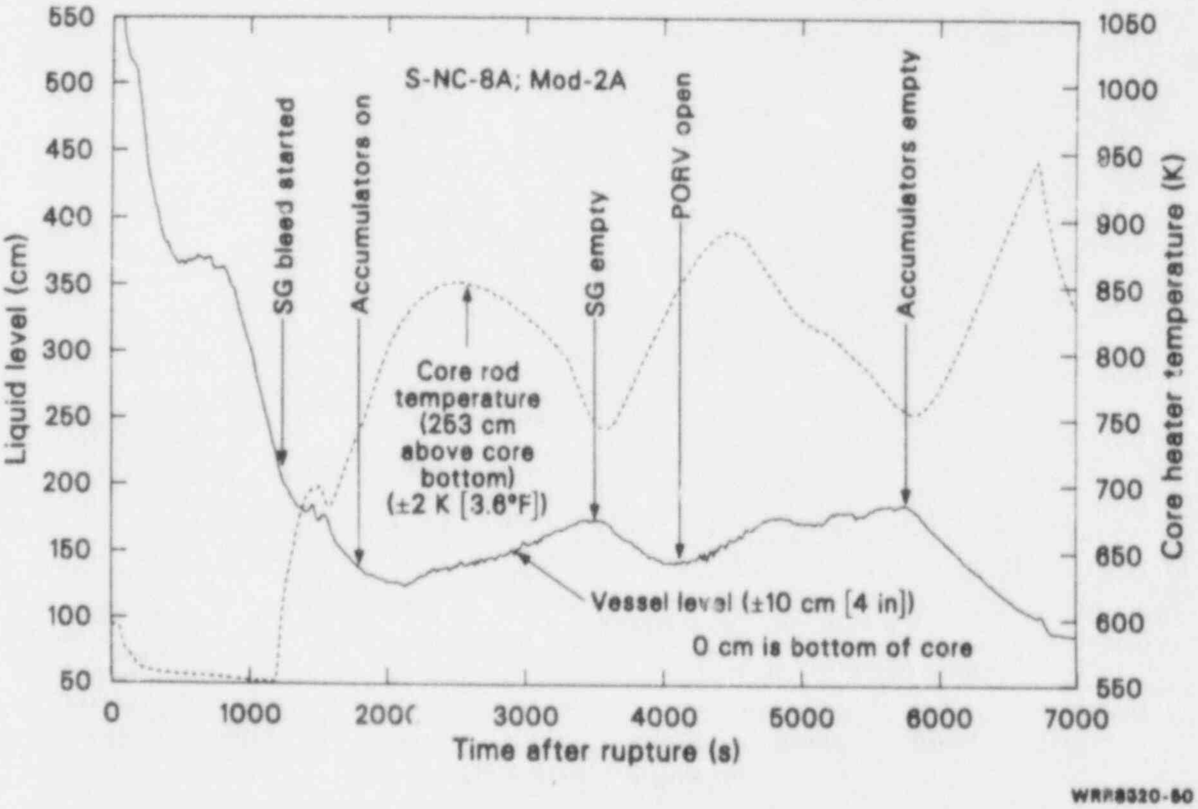


Figure 50. Comparison of vessel collapsed liquid level and core rod temperature for a 0.4% cold leg break.

Compounding Failures During Tube Rupture. Compounding failures examined during tube rupture included stuck open affected loop ADV,[42] stuck open PORV,[44] complete loss of on and offsite power (no HPIS),[43] and a main steam line break as an initiating event with a tube rupture.[45] For all the compounding failures examined in the Semiscale experiments, normal recovery procedures (as discussed in the previous section) were adequate to preclude any core heat up during the tube rupture. As an example, the complete loss of on and offsite power case is discussed, which is the worst case (maximum core uncovery). For this experiment (a 5 tube rupture) recovery procedures involved only unaffected loop feed and steam using ADV steam flow and auxiliary feed with a complete absence of SI (HPIS). Without some operator action to lower primary pressure below affected loop secondary relief valve setpoints, primary to secondary break flow would eventually fill the affected loop secondary and the pressure would equilibrate between primary and secondary. The problem is that the pressure would equilibrate above affected loop secondary relief valve setpoints thus indirectly opening the primary to atmosphere (because of the hydraulic coupling between primary and secondary). If SI were present other means could be used (such as PORV flow) to lower primary pressure below relief valve setpoints. Recovery procedures started at 600 s after tube rupture with an unaffected loop feed and steam to maintain a 5.72 MPa (829 psia) primary pressure (which is below the affected loop ADV pressure setpoint) that isolates the affected loop secondary from atmospheric release (see Figure 51). Next, a more intensive unaffected loop secondary feed and steam was started to reduce primary pressure below the affected loop pressure thus inducing a back flow of affected loop secondary fluid to the primary. The primary pressure decreases during this feed and steam operation because of increased condensation in the primary tubes and the secondary becomes a larger sink with feed and steam. Figure 52 shows the primary and affected loop secondary pressures showing the sudden decrease of primary pressure below the affected loop secondary as the feed and steam operation started. Since primary pressure was lower than affected loop secondary pressure a back flow of secondary fluid through the break to the primary occurred, as shown in Figure 53. The reverse break flow eventually caused a filling of the vessel level also shown in Figure 53. Meanwhile the primary

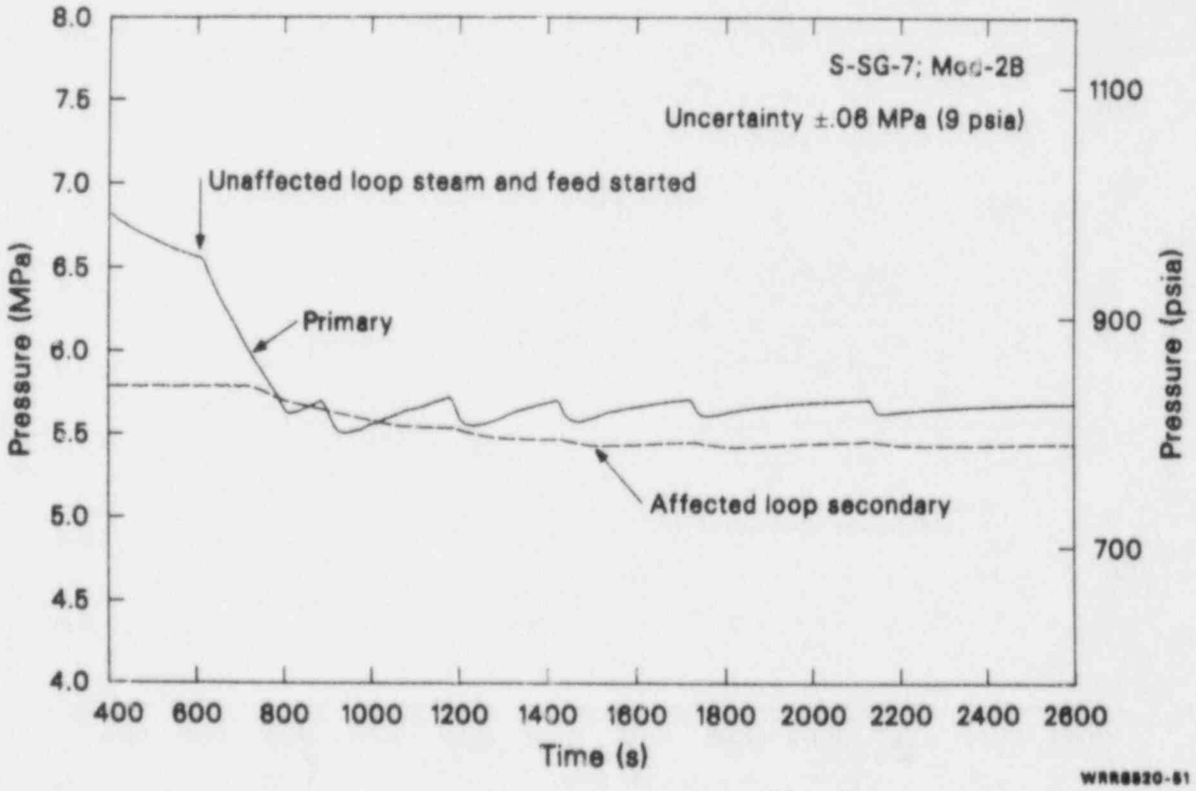


Figure 51. Comparison of primary pressure and affected loop secondary pressure during recovery of a five-tube rupture transient.

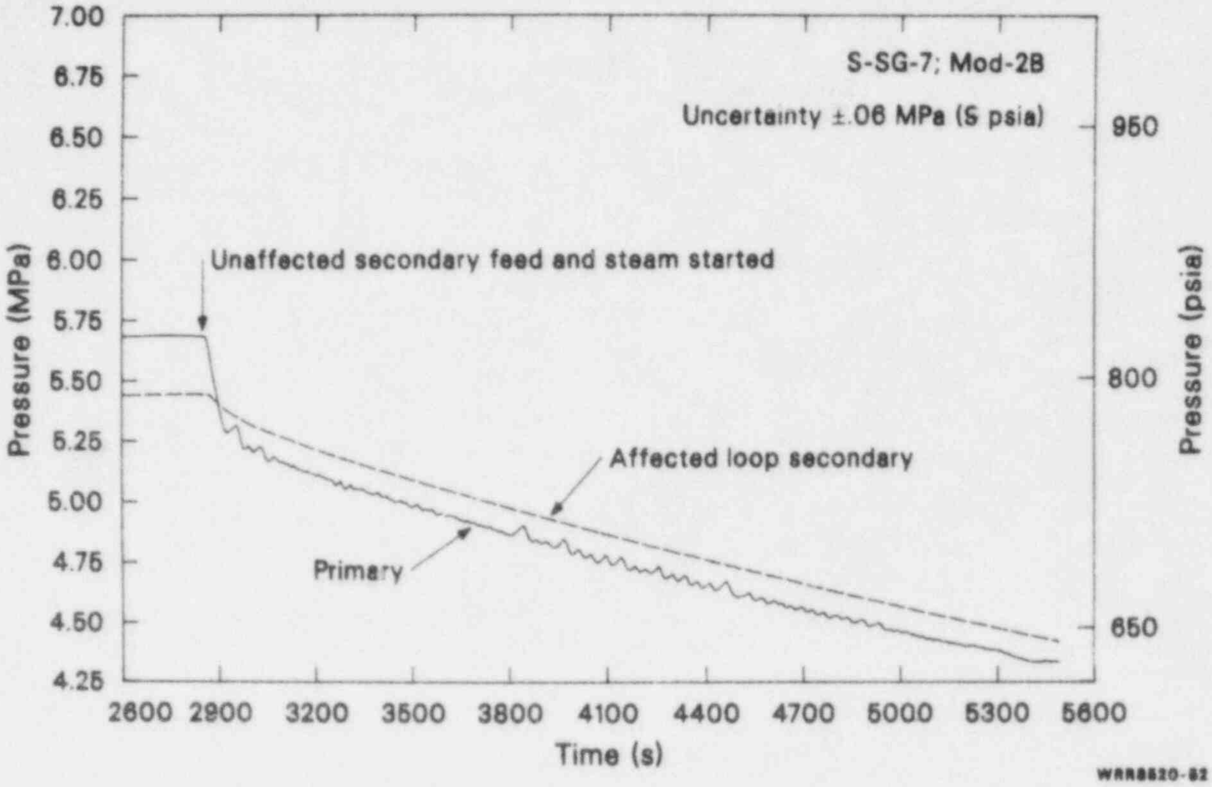


Figure 52. Comparison of primary and affected loop secondary pressure during recovery from a five-tube rupture transient with SI.

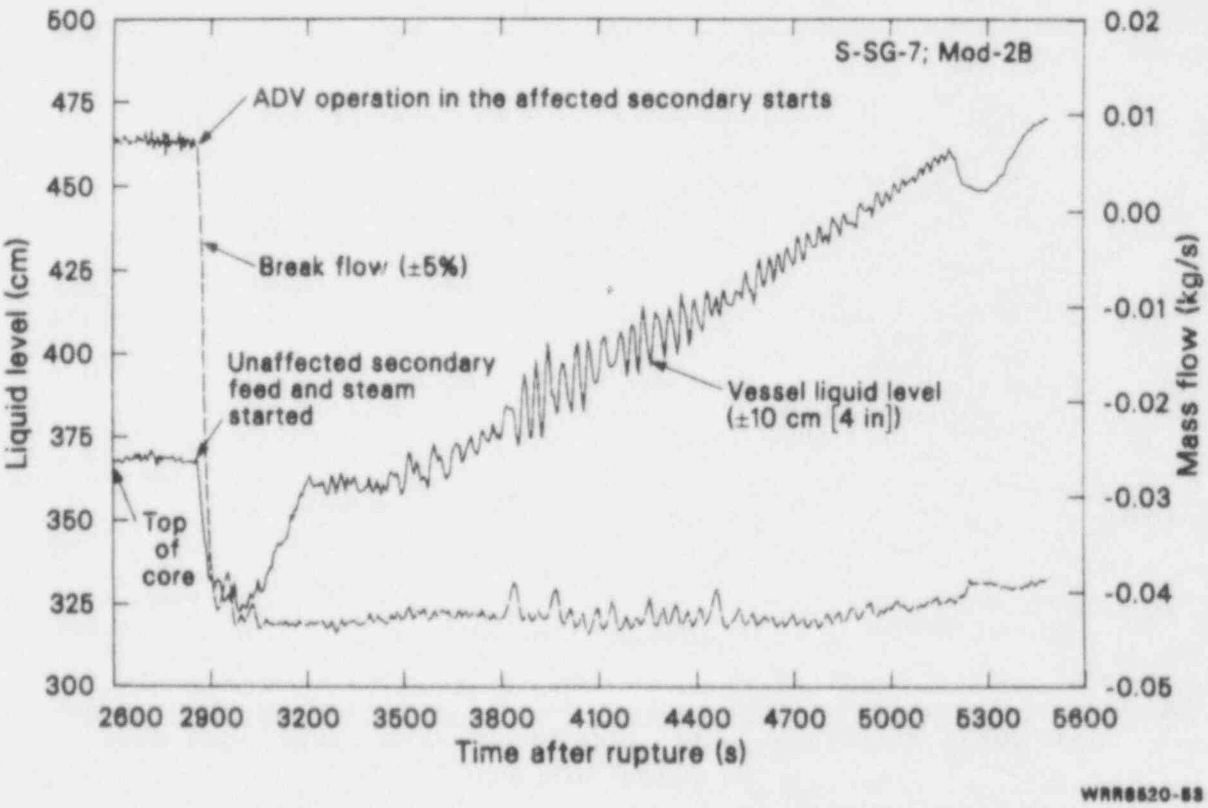


Figure 53. Break flow and lower vessel collapsed liquid level during recovery from a five-tube rupture transient without SI.

pressure continued to decrease to the accumulator setpoint pressure [4.24 MPa (615 psia)] without core uncovery or core rod heat up. In short, the compounding failure of a total loss of SI due to a loss of both on and offsite power was successfully mitigated by operator induced feed and steam of the unaffected loop generator.

Phenomena Associated With SBLOCA

Due to the unique nature of SBLOCAs several phenomena occurred presenting a challenge to state-of-the-art computer calculational ability. Phenomena of interest includes: the change from forced flow to natural circulation as the primary pumps coast down accompanied by a net loss of primary mass inventory, core uncovery heat transfer, and accumulator flow chattering.

Natural Circulation Phenomena During SBLOCA. Natural circulation is an important heat removal mechanism during a SBLOCA. Power to the primary pumps is terminated on a low pressurizer pressure trip and the pumps quickly coast down (on the order of 30 to 50 s) to zero speed. At the point that the primary pumps speed is zero, any core heat removal from the system is accomplished by natural circulation. Therefore correctly modeling this behavior is important to calculating the overall severity of a SBLOCA.

Separate effects steady state testing in the Semiscale facility indicated that the loop natural circulation mode was largely dependent on system mass inventory and only slightly dependent on secondary mass inventory. This trend was also observed during SBLOCAs in the Semiscale system. During a very small break experiment (0.4% cold leg pipe break)[30] all of the major modes of natural circulation were observed including single-phase, two-phase, and reflux as the system mass inventory decreased because of break flow. For this experiment no HPIS was used resulting in a continual decrease in system mass inventory. Figure 54 shows the characteristic primary pressure response associated with a 0.4% SBLOCA including a rapid depressurization associated with core scram and an overall decrease in depressurization associated with the entire system reaching saturation conditions. Due to the very small break a natural circulation phenomena not usually observed during a small break was the sudden increase in depressurization

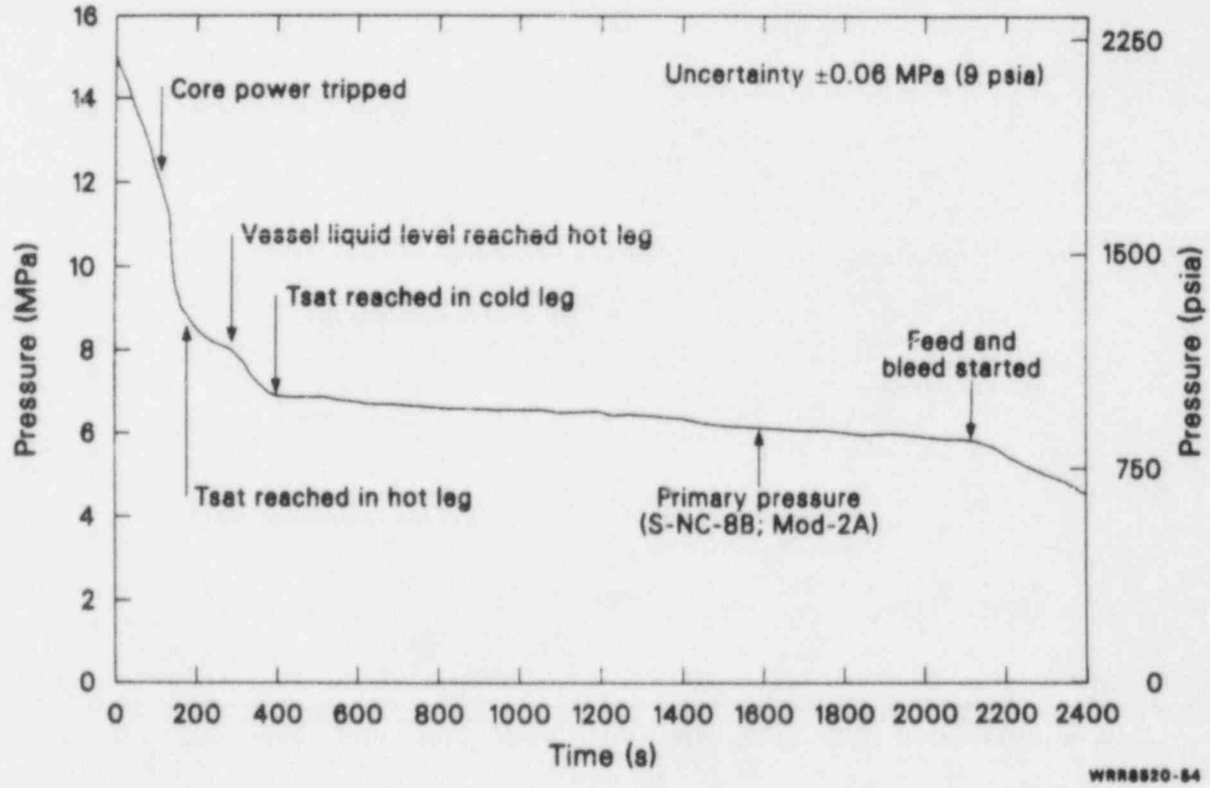


Figure 54. Primary system pressure for a 0.4% cold leg break.

rate (at 300 s) as the hot legs uncovered, as shown in Figure 54. This increase in depressurization corresponds in time with the change from single-phase natural circulation to two-phase natural circulation as evidenced by the downcomer flow increase, as shown in Figure 55. With two-phase natural circulation steam bubbles are condensed in the steam generator causing an increase in depressurization rate. The condensation process is supported by a lower secondary pressure than primary, as shown in Figure 56. Referring back to the downcomer flow (Figure 55) there was a temporary increase in downcomer flow starting at about 120 s. This corresponds to the point in time when the pressurizer empties of liquid. Steam from the pressurizer entered the hot leg where it mixed with the ongoing two-phase mixture inducing a temporary larger density gradient in the loop leading to a momentary increase in downcomer flow.

As more coolant left the system by way of the break, the density gradient between fluid in the upflow side of the steam generator, hot leg, core, and fluid in the downflow side of the steam generator, pump suction, cold leg, and downcomer increased, leading to increased primary loop flow. As the voiding became more pronounced the loop flow rate increased and eventually peaked. As further mass was expelled, the loop mass flow rate decreased due to voids coming over the top of the steam generator tubes, thus reducing the overall density gradient in the loop. Eventually, enough mass was expelled from the system that fluid in the intact-loop steam generator depleted (see Figure 57), leading to a reflux condition in the intact-loop steam generator. Reflux was visually observed[a] to begin at about 825 s in the intact loop, but did not occur in the broken-loop steam generator until much later (1900 s). The broken-loop natural circulation behavior (which represents the broken loop of a four-loop PWR) was generally decoupled from the intact-loop behavior (which represents three unbroken loops of a four-loop PWR), as shown in Figure 58, which compares mass flow rate in the intact loop, broken loop, and downcomer. The two-phase peaking in the intact loop occurred at about 390 s while the broken-loop peaking did not occur until about 650 s. The downcomer mass flow rate as a function of system mass inventory for the transient blowdown case and the

a. Visual observations were made with optical probes near the bottom of the steam generator tube sheet.

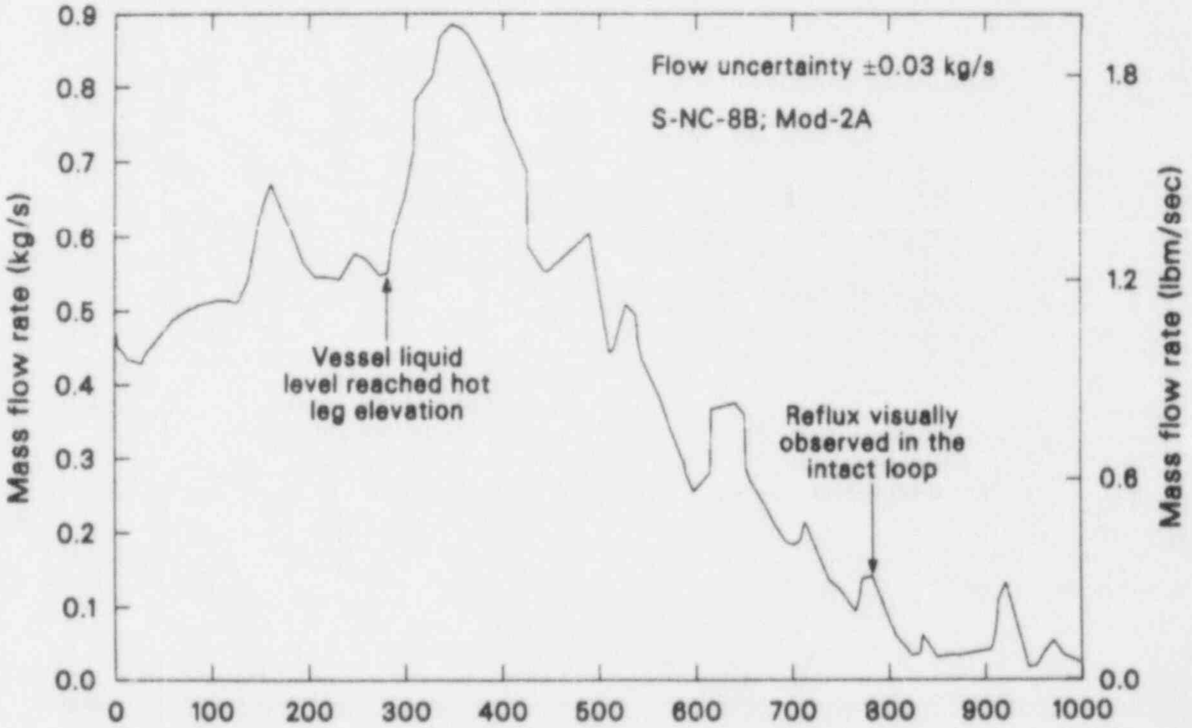


Figure 55. Downcomer mass flow rate during a small break transient (0.4% cold leg break).

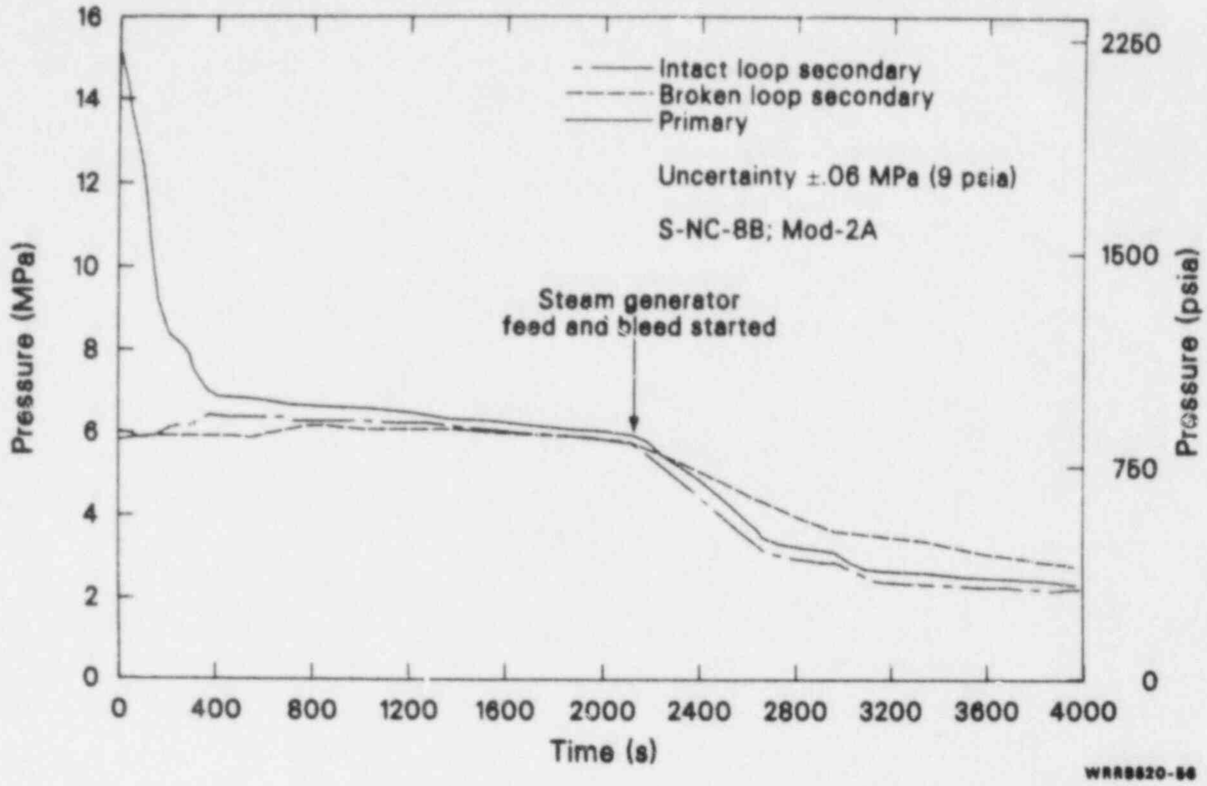


Figure 56. Comparison of primary steam pressure and intact loop and broken loop steam generator secondary pressures during a 0.4% small break transient.

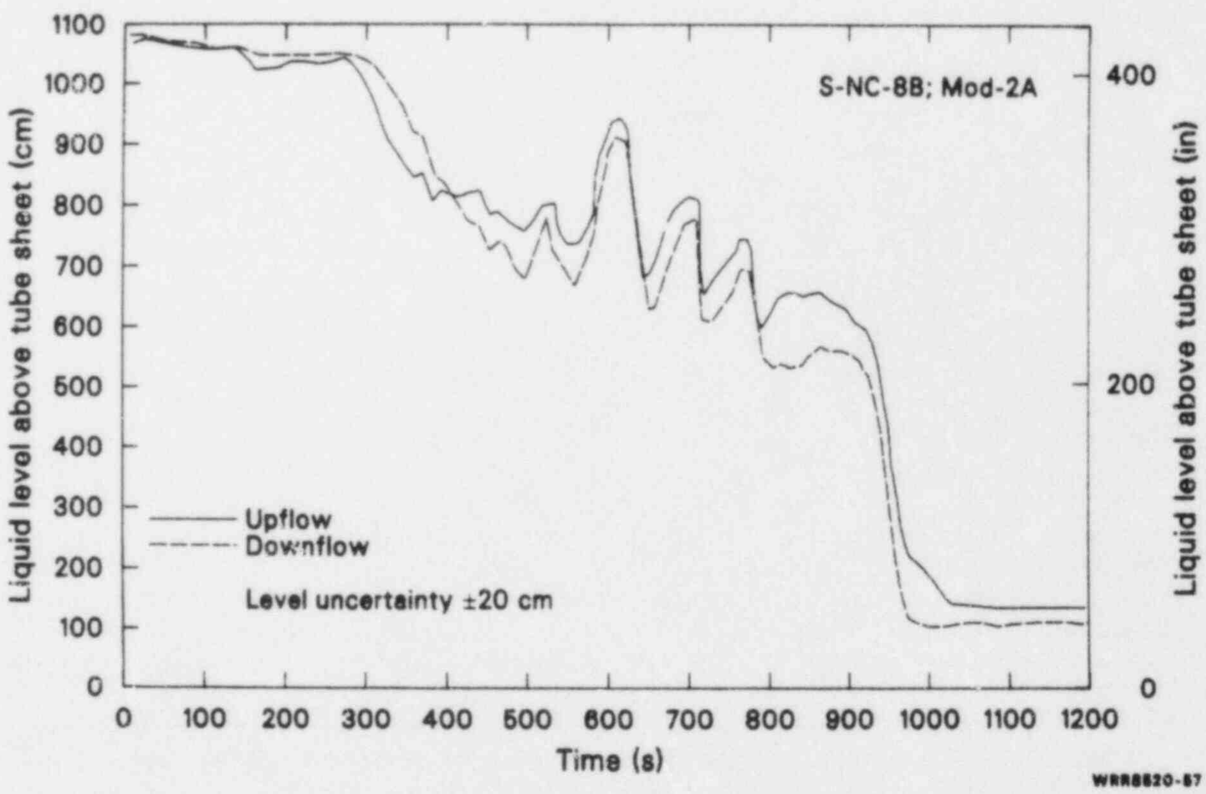


Figure 57. Comparison of upflow and downflow collapsed liquid level in the intact loop steam generator tubes during a small break transient (0.4% break).

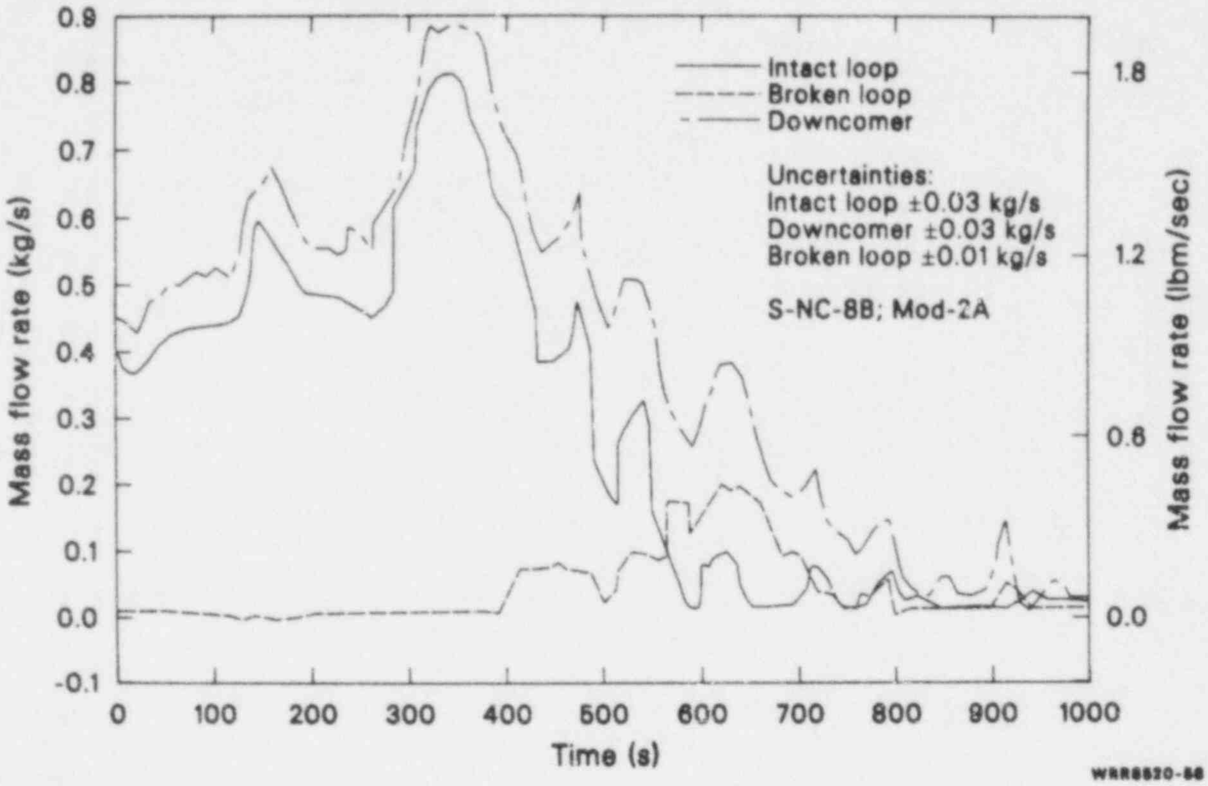


Figure 58. Comparison of downcomer, broken loop, and intact loop mass flow during a small break transient (0.4% break). Broken loop flow turbine is in dead band prior to 400 s.

steady state two loop case is similar, as shown in Figure 59. Besides the nature of the experiment (steady state versus transient), the main difference in these two systems was the inclusion of the pressurizer mass for the transient case. Inclusion of the pressurizer mass resulted in a departure from single-phase type values at a much lower system mass inventory because the hot leg uncovered at a lower percentage of total system mass inventory.

The main heat rejection mechanism in the system during the first 1000 s of the 0.4% small break transient was two-phase natural circulation. Although the feedwater line and steam line in the steam generator secondary were both closed at blowdown initiation, the secondary remained a heat sink for the first 2000 s of the transient. Figure 56 compares primary system pressure with secondary pressure, showing that the secondary pressure is lower than the primary for the first 2000 s of the transient.

At ~2000 s into the transient, mass depletion in the system due to break flow and a depression of core level caused by the formation of pump suction liquid seals caused a core uncovery, which prompted initiation of steam generator feed and bleed as a recovery operation. The steam generator feed and bleed increased the condensation potential in the system, which precipitated an increase in depressurization rate, as shown in Figure 54. This increased depressurization rate caused the accumulator pressure setpoint to be reached more quickly. Later, after accumulator flow had depleted (about 5000 s into the transient) reflux cooling in the intact loop (visually observed) supported by steam generator auxiliary feed and bleed maintained the core level and thus core coolability even in the absence of accumulator flow. The net mass flow rate from the break was small enough to maintain the core level within an interval where core rod temperature excursions did not occur. Eventually, (in the absence of emergency core coolant the core collapsed) liquid level would deplete and cause core rod temperature excursions. Core rod temperature excursions occurred at total system mass inventories of about 35%, which is a collapsed level in the vessel of about 275 cm (108 in.) above the bottom of the core.

Core Thermal-Hydraulic Response. Transient severity during a SBLOCA can be measured by the amount of liquid uncovery of the core and the amount of core rod heat up during the boil-off

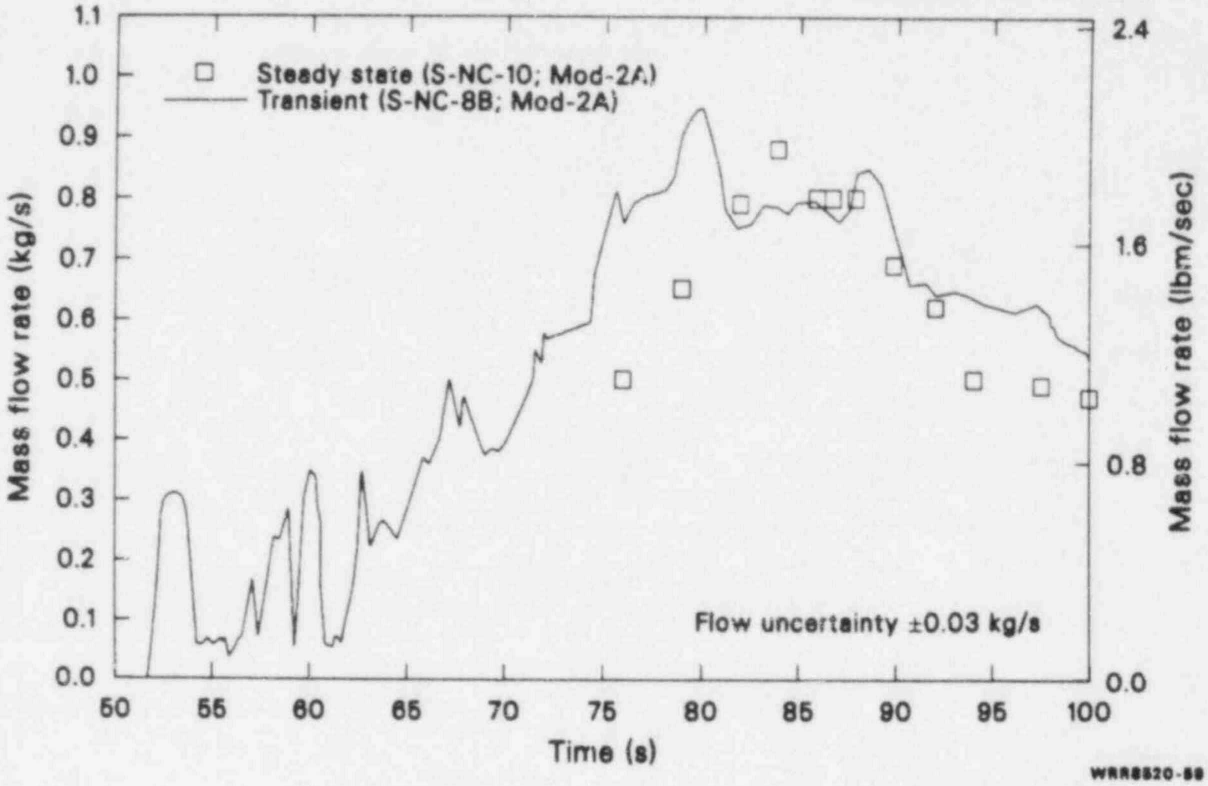


Figure 59. Comparison of downcomer mass flow rate as a function of the system mass inventory for steady state and transient experiments.

associated with the slow depressurization period. Any core heat ups induced by pump suction seal formation are not a good measure of transient severity as the pump suction liquid seal always clears and the rods are quickly rewetted. However, during the boil-off period, sustained core heat up occurs. A compilation of the range in collapsed liquid levels for which incipient core heat up occurs has been compiled for the cold leg small break spectrum studied in Semiscale, as shown in Figure 60. Heat up occurs (depending on break size) with collapsed levels between 150 to 250 cm from the bottom. The range in level for incipient core heatup is attributed to different blowdown dynamics causing different froth levels in the core. For a given break size, the difference between a given minimum collapsed level and the collapsed level corresponding to incipient dryout is a good qualitative measure of whether a dryout occurs, is imminent, or is not imminent. Collapsed level data from two types of experiments involving different accumulator setpoint pressures as shown on Figure 60. For both accumulator setpoints there is a a good margin for cooling at the lower and upper end of the spectrum (2.5 to 10% breaks), however for the 5% break case the margin is reduced. However, there is a distinct advantage to inject accumulator water at the higher pressure [4.1 MPa versus 2.8 MPa (595 versus 406 psia)]. By injecting earlier during the slow primary depressurization period the resulting early reflood can cause a core level increase precluding the collapsed level reaching the level for which incipient core heat up occurs.

The collapsed level doesn't present an exact physical representation of water availability in the core during the boil-off period as there is a distinct froth[a] level on top of a pool of liquid at the bottom of the core. Figure 61 compares the collapsed level and froth level (as determined from the in-core densitometers) for a TMI type scenario. As the core liquid boils off and the collapsed level decreases, the froth level and collapsed level approach each other about in the region of incipient core heatup shown on Figure 60. One reason the froth level collapses after dropping below the midplane of the core is the presence of subcooled water in the lower core does not support the boiling necessary to maintain a froth level. For core positions immersed in a saturated liquid, boiling produces steam that

a. The froth level was defined to be the point where the densitometers showed a rapid decrease in density to steam type valves.

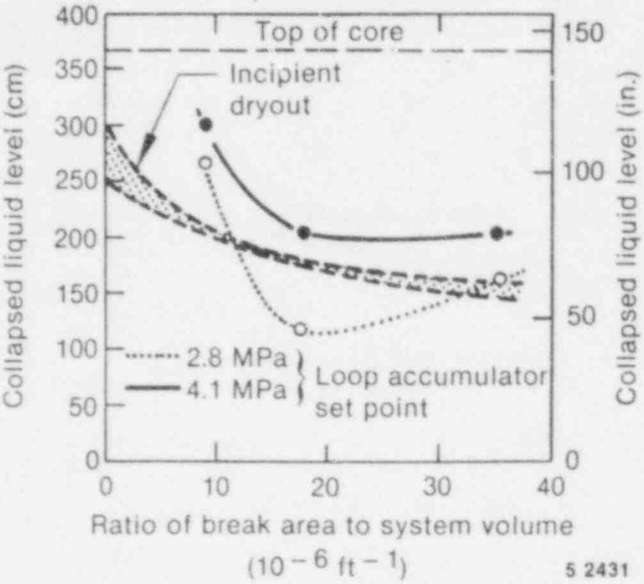


Figure 60. Estimated minimum core collapsed liquid levels for a 4.1 and 2.8 MPa loop accumulator pressure setpoint.

supports a froth level. However, this saturated region evidently boils off leaving the subcooled pool of water with a small froth level. Figure 62 compares the core heater rod temperature at the top, middle, and bottom of the core with densitometers near these locations. The densitometers show the rapid drop in density associated with the froth level passing a position and a definite fluid stratification in the core throughout the boil-off period. At 6550 s the collapsed level was about 200 cm above the bottom of the core (Figure 61), which is when the core heat up at that elevation occurred as shown in Figure 62. This collapsed level does not correspond with the band of incipient core heatup, as shown in Figure 60. Figure 60 refers to heatups for cold leg centerline breaks only and therefore, does not apply to the TMI case[a] (stuck open PORV), as shown in Figures 61 and 62. Therefore, boil-off dynamics appear to be affected by break location and Figure 60 applies only to cold leg centerline breaks.

Loop Accumulator Injection Behavior. During a SBLOCA the accumulator pressure is just above primary pressure throughout the time where accumulator tanks fluid empties into the primary system. Oscillation in both primary pressure,

a. The TMI simulation has a break area to system where ratio of $2.15 \times 10^{-6} m^{-1}$ (6.56×10^{-7} ft^{-1}) should have resulted in an incipient core heat up of between 250 and 300 cm, as shown in Figure 60.

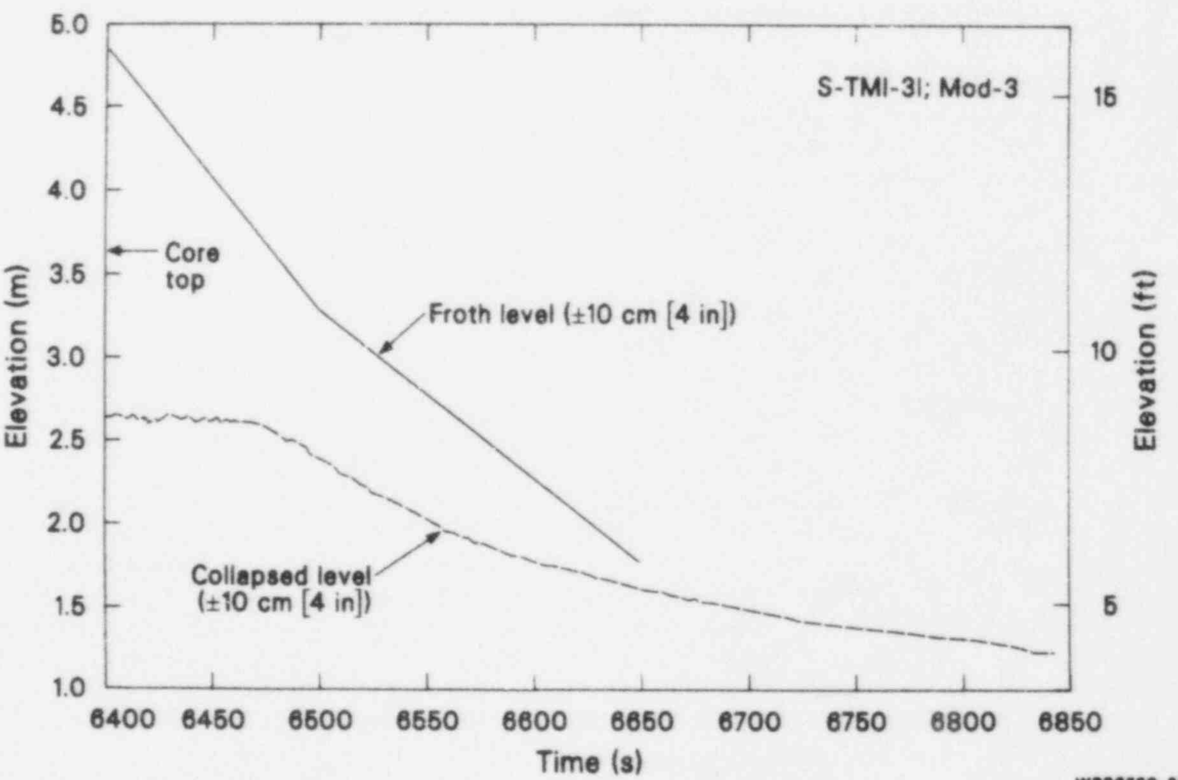


Figure 61. Comparison of collapsed level and froth level in the core for the Semiscale simulation (Test S-TMI-3I).

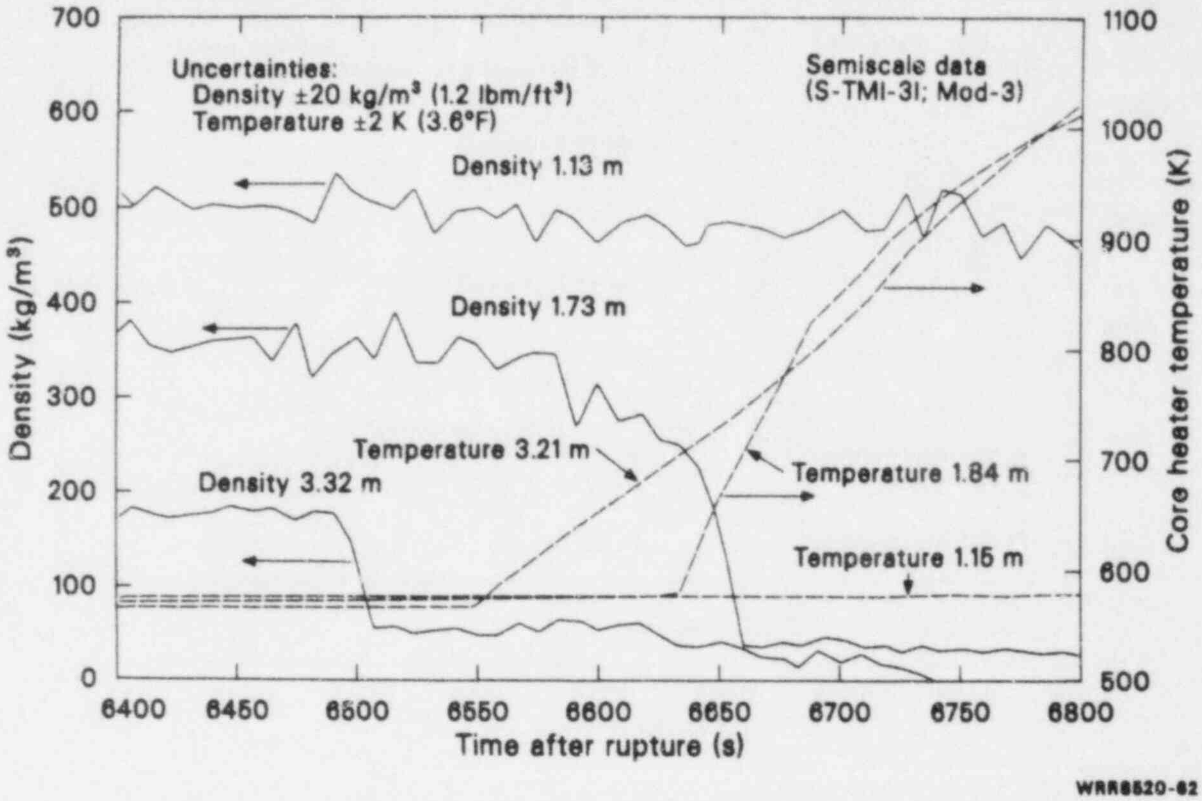


Figure 62. Comparison of in-core fluid density and core rod thermocouple response for the Semiscale simulation (S-TMI-3I).

accumulator pressure, and accumulator injection rate are observed during accumulator injection for a SBLOCA (Figure 63) that presents a challenge to computer code calculation.

When the accumulator begins injecting, the condensation effect of the cold water causes the primary pressure to drop, in turn increasing the injection rate. But the effect of injecting mass into the system and dropping the pressure is to increase the vapor generation rate and cause the primary pressure to plateau or, in some instances, even to rise. This in turn causes the pressure driven accumulator flow to stop and the vapor generation rate and the system pressure to decrease, starting a new cycle. This cyclic behavior was observed to occur for the duration of accumulator liquid injection. With decreasing break size, the magnitude of the oscillations decreased to a point at which the accumulators appeared to *float* on the system pressure. The oscillatory behavior of the core collapsed liquid level is prominent, as shown in Figure 64. Although the effect on core level was pronounced, the oscillatory injection did not detract from adequate cooling of the core.

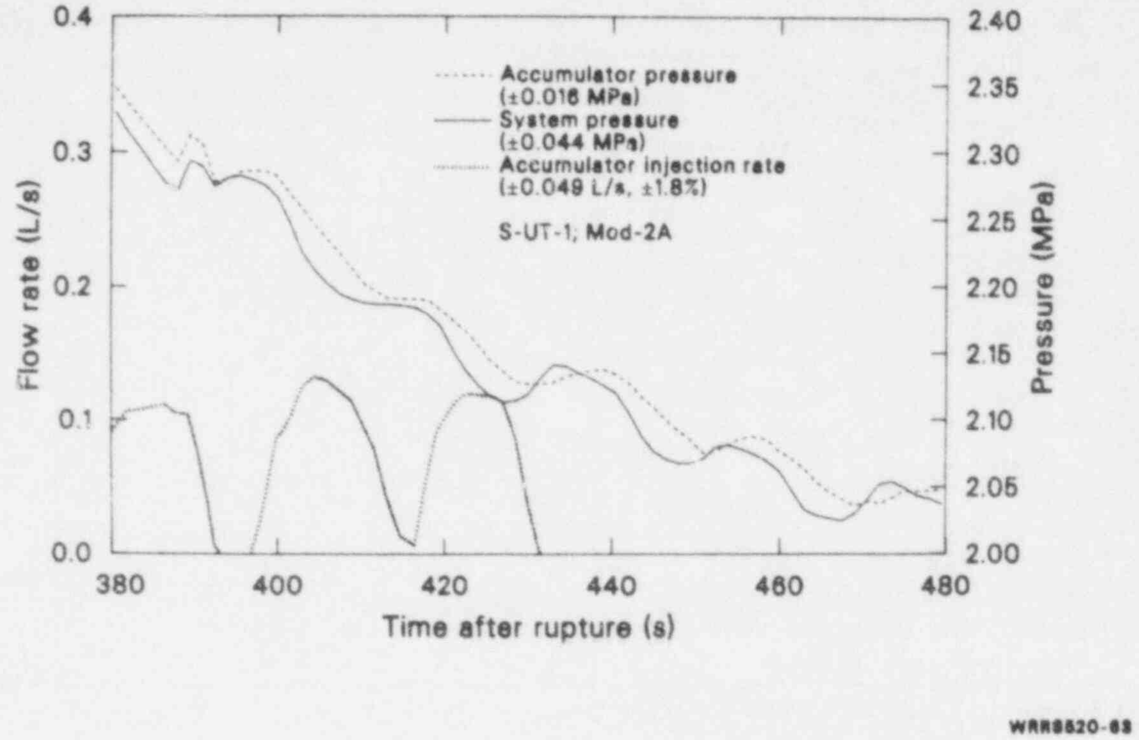


Figure 63. Loop accumulator injection oscillation driven by differential pressure behavior (Test S-UT-1).

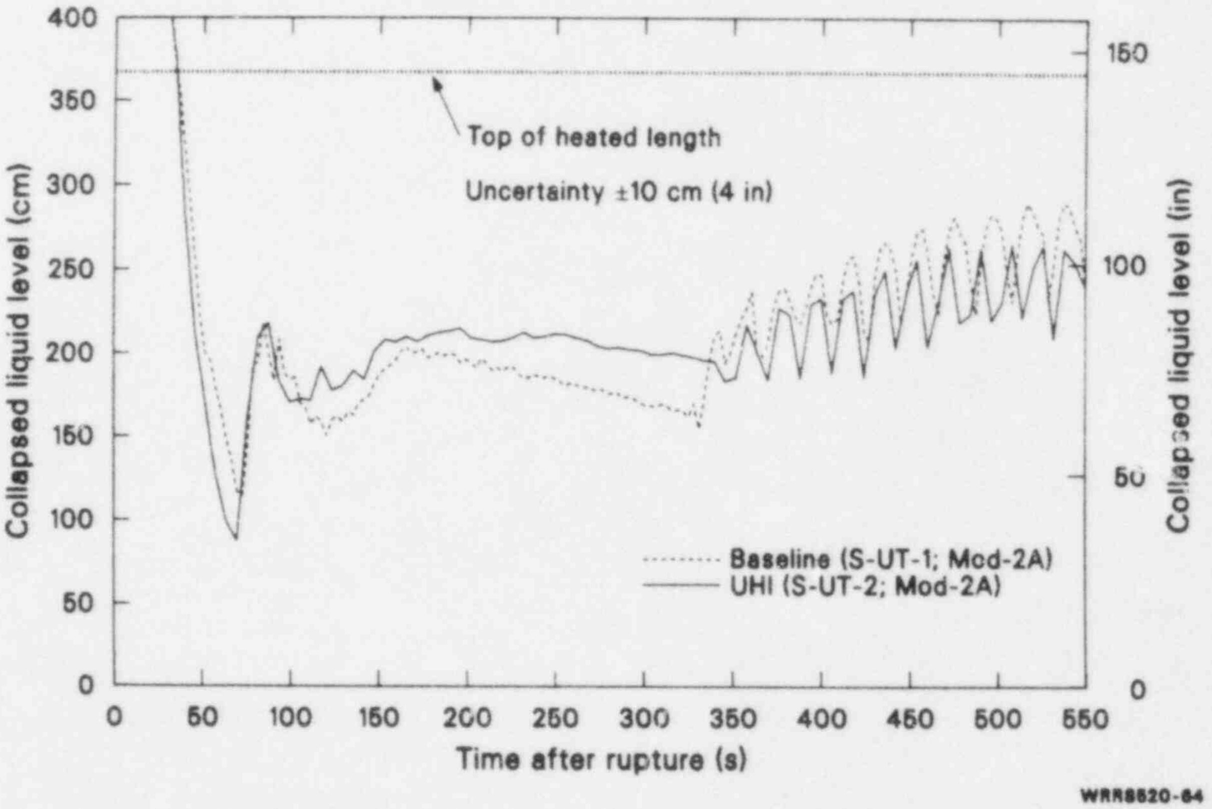


Figure 64. Core collapsed liquid level showing influence of loop accumulator injection oscillations (10% breaks).

RECOMMENDATIONS FOR FUTURE EXPERIMENTAL NEEDS

This section discusses issues in the field of small break loss-of-coolant research that with further testing could improve knowledge of SBLOCA phenomena and also calculational capability. This discussion is divided into two areas: recommended separate effects testing and recommended integral testing.

Recommended Separate Effects Testing

Improvements in the code that could greatly enhance calculational capability include two-phase pump models; interfacial drag models; and post-CHF heat transfer models.

Two-Phase Pump Testing. Current code calculations use a two-phase pump model that includes a two-phase multiplier and single-phase and two-phase difference curves. These models are based on data taken for a completely different pump (the Lawrence pump used in the intact loop for Mod-1, Mod-3, and Mod-2A); however, more recent data in Semiscale has been performed with new higher speed, high specific speed pumps. With these new pumps, the pump operation was shown in S-LH-1/S-LH-2 to have a strong effect on fluid behavior and can influence the *holdup* of liquid in the primary tubes. Therefore, to accurately model the Semiscale data at least one of either the intact loop or broken loop pump should be tested in two-phase conditions in at least the first quadrant[a]. Data from this testing could be factored into the existing pump model. With an improved two-phase pump degradation model, more credibility can be placed on other calculated parameters from the code.

Interfacial Drag/Core Thermal-Hydraulics. Test S-LH-1 and S-LH-2 showed a poorly predicted (by RELAP5) void distribution in the core during core liquid level depressions that precluded accurate calculation of core rod heat ups that were observed in the data.[46] Interfacial drag and core heat transfer are both involved in correctly calculating the void distribution in the core. Both useful interfacial drag and high pressure [6.9 MPa (1000 psia)] post-CHF heat transfer information could be obtained from separate effects boil-off experiments involving only the Semiscale vessel and core. Boundary conditions could be accurately measured at the entrance and exit to the vessel that could later be used to drive the codes while checking existing models.

Interfacial Drag-Flooding-Condensation Effects In Primary Tubes. Test S-LH-1 and S-LH-2 showed a strong relationship between liquid hold-up in the primary steam generator tubes and core liquid level depressions. This phenomena is thought to be closely related to interfacial drag, flooding in tubes, and condensation in the primary tubes. The codes are unable to exactly calculate this phenomena;[46] therefore a separate effects test could be performed involving only the vessel/core as a steam source and the broken loop steam generator as a sink. A reflux meter and the extensive array of differential pressure cells in the broken loop steam generator could give a good measure on boundary conditions for these flooding experiments.

Recommended Integral Testing

The following integral testing could give additional insight into SBLOCA safety related issues relative to PWR behavior as well as improve assessment efforts for the computer codes.

Preferred Pump Operation During a SBLOCA. The question of preferred pump operation during a SBLOCA was not adequately addressed in Semiscale and LOFT tests because (a) LOFT had poor elevation scaling and ECC injection location that affects break flow and; (b) Semiscale had poor control of boundary conditions (HPIS flow) during early testing in the Mod-3. Additional experiments involving delayed versus early pump trip could be performed in Semiscal to determine the preferred pump operation. As an example of where pump operation was desirable, was during the TMI accident. A continued pump operation would have significantly delayed the TMI core damage because forced convection to steam has a significantly higher heat transfer coefficient than natural convection to steam or simply an adiabatic case. For

a. Pump operation is defined on homologous curves divided into four quadrants depending on speed, flow, and head. The early blowdown phase of a SBLOCA usually lies in the first quadrant.

these experiments the scaled hydraulic torque could be matched as closely as possible with an estimated PWR hydraulic torque.

Lower Vessel Breaks. Lower vessel SBLOCA have never been investigated in the Semiscale system. Performing lower vessel breaks experiments would provide a new data base for code verification and a correct calculation of this phenomena would greatly enhance confidence in the codes. Since all the codes were developed for TMI or cold leg/hot leg SBLOCAs, a lower vessel SBLOCA would present a unique challenge to the existing codes.

Ultra Small Breaks With Degraded ECC. Semiscale has performed two ultra small breaks[30,32] but measurement of boundary conditions were only fair (see Table A-1, Item 9). These high risk accidents are relatively very probable; however, concurrent with degraded ECC is less likely, nevertheless, the consequence is very high (melted core). Using the state-of-the-art heat loss make-up techniques, scaling, and measurement systems, these new experiments could provide great insight into the mechanism involved during ultra small breaks with degraded ECC.

CONCLUSIONS

1. An extensive data base involving Semiscale Small Break Loss-of-Coolant simulations is available for code development and assessment use.

   To aid future assessment and development efforts, the Semiscale SBLOCA experiments have been put into historical perspective relative to the overall water reactor safety effort. The Semiscale simulations have been cataloged according to type of SBLOCA including a brief description of each of the experiments. In addition, the various Mods of the Semiscale system are described including Mod-1, Mod-3, Mod-2A, Mod-2B, and Mod-2C. Also cataloged are the recommended experiments to assess the various issues associated with SBLOCA including an estimate of data quality and quality of system description documentation for code assessment and development purposes.

2. Semiscale SBLOCA signature response characteristics for a variety of types of SBLOCAs stimulates thinking about large scale commercial PWR response during SBLOCAs.

   Experiments involving pipe breaks, steam generator tube rupture and TMI-2 type (stuck open PORV) characterize the signature response for SBLOCA.

3. Many SBLOCA issues examined in Semiscale testing give insight into SBLOCA severity for a large scale PWR response.

   Important SBLOCA issues addressed by Semiscale testing include: accident severity as affected by break location, break size, and system configuration; small break response with compounding failures; preferred primary coolant pump operation during a transient with small break symptoms; and, relative merits of upper head safety injection.

4. Future Semiscale integral and separate effects testing could improve understanding of phenomena and code calculational capability.

   Recommendations for future Semiscale separate effects testing include: two-phase pump testing; interfacial drag and core heat transfer work; and flooding work in the steam generator. Future integral Semiscale testing recommendations includes: preferred pump operation during SBLOCA; lower vessel breaks; and, ultra small breaks.

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APPENDIX A

APPLICABILITY OF SEMISCALE SMALL BREAK EXPERIMENTS FOR CODE DEVELOPMENT AND ASSESSMENT

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APPLICABILITY OF SEMISCALE SMALL BREAK EXPERIMENTS FOR CODE DEVELOPMENT AND ASSESSMENT

This appendix presents a table summarizing the applicability of Semiscale small break data for code development and assessment. Table A-1 gives a list of issues important to small break analysis along with the recommended Semiscale experiment or experiments that best satisfy that issue. Also listed is the adequacy of system documentation (used for modeling information) as well as the adequacy of the data to examine the issue. A rating of poor, fair, good or excellent has been assigned to each of the issues as to adequacy of data and location. An excellent rating of documentation means the system configuration is documented in a referencable document (suitable for model construction). A poor, fair, good rating for documentation means the information is available; however, such information can be obtained only through interaction with Semiscale personnel. The information is contained in internal documentation, operator logs, and test procedure. The poor, fair, or good rating was assigned based on how recent the testing and which system was used for the testing. An excellent rating for data means there is sufficient data to get both qualitative and quantitative information about an issue. An excellent data rating implies excellent control of initial and boundary conditions. A rating of good means that quantitative data is available with a higher uncertainty than the excellent rating. Also control of boundary and initial conditions is adequate for assessment purposes even though some boundary conditions of second order importance may be lacking. A fair rating for data implies missing boundary conditions and a high uncertainty on key initial conditions (e.g., steam generator secondary level); however, overall trends such as pressure and vessel level are adequately measured.

Most of the issues listed in Table A-1 apply only to code assessment involving overall integral system effects; however, some of the data is applicable for model development most notably in the field of core hydraulics/heat transfer (Item 4). Semiscale has in-core gamma densitometers that allow an estimate of channel average void fraction. Special care was given to placing, core rod thermocouples at the same axial and azimuthal orientation as the gamma densitometer beam. The advantage of using the Semiscal data for core heat transfer model development is the estimation of location void fraction offered by the gamma densitometers. The disadvantage of the Semiscale data for core heat transfer model development is the lack of superheated steam probes. Another area of promise for model development is the use of triplet thermocouples in the broken loop steam generator (Item 6). A triplet includes a matched set of primary fluid, wall, and secondary fluid thermocouples that can be used for primary to secondary heat transfer studies especially in the fluid of condensation heat transfer. Another area for possible model development application is in the field of countercurrent two-phase flow in vertical and horizontal tubes (Item 5). Semiscale has two and three beam gamma densitometers to determine flow regime as well as an optical probe at the steam generator inlet plenum. Although such parameters as slip are not determined the data gives qualitative support for models.

Table A-1. Applicability of Semiscale small break experiments

Issues	Recommended Experiment(s)	Adequacy of Configuration Documentation	Adequacy of Data
Overall SBLOCA blowdown data	S-LH-1	Excellent[A-1]	Excellent; two-phase flow data is inadequate after saturation conditions achieved in loop (40 s in cold leg); however, liquid level measurements were maximized especially in the steam generator primary tubes; excellent measurement of boundary conditions including break flow and HPIS injection.
Effect of downcomer to upperhead bypass flow on SBLOCA severity	S-LH-1; S-LH-2	Excellent[A-1]	Excellent (see 1 above); initial core bypass flow measured within 10%.
Manometric core liquid level depression (steam binding)	S-LH-1	Excellent[A-1]	Excellent (see 1); liquid levels in all components accurately measured.
Core thermal hydraulics (pre- and post-CHF and liquid distribution in the core during heat up	S-LH-1; S-LH-2	Excellent[A-1]	Excellent; core gamma densitometers were specially matched to core rod thermocouples; void distribution in the core was accurately measured. A good sample of both axial and azimuthal thermocouples were matched to gamma densitometer locations.
Countercurrent two-phase flow in vertical tubes and horizontal pipes (flow regime studies)	S-LH-1; S-LH-2	Excellent[A-1]	Good; optical probes gave a view of steam generator inlet plenum during SBLOCA; extensive level measurements in tubes; hot leg gamma densitometers and turbine meters also give evidence of countercurrent flow; densitometers give good insight into flow regime.
Condensation effects in primary tubes (primary to secondary heat transfer)	S-LH-1; S-LH-2	Excellent[A-1]	Good; matched/calibrated triplet thermocouples in steam generator tubes can give good primary to secondary heat transfer information.
Steam generator tube rupture signature response	S-SG-1 (one tube) S-SG-2 (five tube) S-SG-3 (ten tube)	Excellent[A-2]	Good-excellent; good control of all boundary conditions; extensive measurement of all effluent from the system including steam generator relief valve flow.
Recovery during tube rupture:		Excellent[A-2]	Good-excellent; excellent control and measurement of boundary conditions; especially safety injection, PORV flow and pressurizer auxiliary spray.
Primary feed and bleed	S-SG-2; S-SG-8		
Pressurizer internal heater operation	S-SG-3; S-SG-4		
Secondary feed and bleed	S-SG-2; S-SG-7; S-SG-5		
Pressurizer auxiliary spray	S-SG-3; S-SG-4		
Safety injection	S-SG-1		
Natural circulation during a SBLOCA (single-phase, two-phase; reflux)	S-NC-8B; S-NC-9	Fair[B]	Good-data includes optical view of both inlet and outlet plenums of the steam generator as well as special low flow measuring turbine meters; steam generators extensively instrumented with triplet thermocouples (primary/ secondary/ wall); single-phase data also available because system was heated using only single-phase natural circulation.

Table A-1. (continued)

Issues	Recommended Experiment(s)	Adequacy of Configuration Documentation	Adequacy of Data
Effect of upper head injection on severity 2.5% break 5.0% break 10.0% break	 S-UT-4; S-UT-5 S-UT-6; S-UT-7 S-UT-1; S-UT-2	Fair[a]	Fair-good; occasional inaccuracies in key boundary conditions such as HPIS flow; however overall effect on severity due to UHI compared to non-UHI can be assessed.
Effect of break size on severity 2.5% 5.0% 10.0%	 S-UT-4 S-UT-6 S-UT-1	Fair[a]	Good-important boundary conditions generally available; data sufficient to assess break size effect on accident severity especially fluid mass distribution during blowdown.
Effect on pump operation on severity	S-SB-P1; S-SB-P2; S-SB-P3; S-SB-P4; S-SB-P7	Poor[a]	Fair; break flow data poor; however overall pressure and mass distribution adequate; pump is small scale, low specific speed with poor two-phase degradation data.
Scaling issues (comparison of LOFT Test L3-1 provides comparison of scale)	S-SB-4; S-SB-4A	Poor[a]	Fair; break flow measurement poor; overall pressure and mass distribution good.
Multidimensional affects: loop-to-loop; within the steam generator tubes; core thermal-hydraulics	S-LH-1; S-LH-2	Excellent[A-1]	Good; steam generator tubes in broken loop extensively instrumented with differential pressure cells; core extensively instrumented both axially and azimuthally with core rod thermocouples.
ECC mixing and condensation	S-UT-1; S-UT-4; S-UT-6	Fair[a]	Good; temperature at ECC injection location limited to one thermocouple.
Break flow	S-LH-1 UT test series	Excellent[A-1] Fair[a]	Good-excellent; break flow measured accurately with condensing/catch tank system; however, staggered DP cells across break are lacking; good gamma densitometer information either side of the break allow flow regime estimation; during the UT test series, optical probes were used to obtain films of a centerline view of the break showing stratification and liquid phase being pulled into the break nozzle.

a. Adequate information on configuration is available but would require extensive interaction with Semiscale analysis and operation personnel using internal documentation. Mod-2A has fair to good documentation availability and Mod-3 has poor documentation availability.

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13. ABSTRACT (200 words or less)

Following the loss-of-coolant accident at TMI in 1979, a multitude of small break loss-of-coolant accident experiments were performed in the various Mods of the Semiscale facility at the Idaho National Engineering Laboratory (INEL). A summary of what experiments have been performed and a description of the various Semiscale Mods is given. The signature response of various kinds of small breaks are characterized. Small break loss-of-coolant accident issues addressed by Semiscale testing are discussed, including: effect of break location and break size, effect of core bypass flow, preferred primary coolant pump operation, effectiveness of upper head emergency core cooling injection, and recovery procedures. Phenomena of interest to small break loss-of-coolant accident analysis is presented including core uncovery heat transfer and natural circulation. Recommendations are given that can improve calculational capabilities for future small break testing.

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