CONF-801102--6

AN INVESTIGATION OF SMALL BREAK LOSS-OF-COOLANT PHENOMENA IN A SMALL SCALE NONNUCLEAR TEST FACILITY

MASTER

For Presentation at

1980 Winter Annual ASME Meeting
Chicago, Illinois
November 16-21, 1980

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Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research under DOE Contract No. DE-AC07-76ID01570.

Abstract

A small-scale nonnuclear integral test facility designed to simulate a pressurized water reactor (PWR) system was used to evaluate the effects of a small break loss-of-coolant accident (LOCA) on the system thermal-hydraulic response. The experiment approximated a 2.5% (11-cm diameter) communicative break in the cold leg of a PWR, and included initial conditions which were similar to conditions in a PWR operating at full power. The 2.5% break size ensured that the nominal break flow rate was greater than the high pressure injection system (HPIS) flow rate, thus providing the potential for a continuous system depressurization. The sequence of events was similar to that used in evaluation model analysis of small break loss-of-coolant accidents, and included simulated reactor scram and loss of offsite power. Comparisons of experimental data with computer code calculations are used to demonstrate the capabilities and limitations of integral system calculations used to predict phenomena which can be important in the assessment of a small break LOCA in a PWR.

εb

NOMENCLATURE

AIS accumulator injection system

ECCS emergency core cooling system

HPIS high pressure injection system

LOCA loss-of-coolant accident

LPIS low pressure injection system

PWR pressurized water reactor.

INTRODUCTION

A small break loss-of-coolant accident (LOCA) in a pressurized water reactor (PWR) is a more probable occurrence than the design basis type accident involving a double-ended offset shear of a primary coolant pipe[1]. In some instances, a small break LOCA could place stringent demands on the emergency core cooling system (ECCS) to maintain adequate cooling of the core. For example, a LOCA involving the rupture of a pipe of sufficient diameter to result in a nominal break flow rate somewhat greater than the high pressure injection system (HPIS) flow rate, would be characterized by a continuous, although prolonged, system depressurization during which a depletion of the primary system liquid inventory would occur. The possibility of a core temperature transient under such circumstances would depend on whether the vessel water-steam mixture level decreased below the top of the core significantly before the system depressurized to the accumulator injection system (AIS) setpoint pressure.

Experimental data which can be used to help evaluate the effects of a small break LOCA in a PWR have been limited to several experiments performed in the Semiscale Mod-1 and Mod-3 test facilities.[2,3] These experiments simulated piping ruptures with break flow areas equivalent to between 6 and 10% of the flow area of a single loop of the primary coolant system piping. Results of these experiments indicated that it was well within the capability of the ECCS system to ensure that the core remained covered during the transient, thus preventing a core temperature excursion. However, additional testing which extended the range of break sizes to smaller values was necessary to obtain a better understanding of the thermal-hydraulic phenomena resulting from small breaks in the primary coolant system piping.

The Semiscale Program, which is conducted by EG&G Idaho, Inc., for the United States Government, responded to the need for additional experimental data by performing a series of small break LOCA experiments in the Mod-3 test facility. The Semiscale Mod-3 system is a small-scale nonnuclear test facility which simulates the principal physical features of a 4-loop PWR, but is much smaller in volume. The Semiscale small break test series was conducted to investigate the thermal-hydraulic phenomena resulting from a small break in the primary system piping, and also to provide experimental data for evaluating the analytical capability of computer codes to accurately predict those phenomena.

This paper describes the results of a small break loss-of-coolant experiment[4] which simulated an 11-cm diameter communicative break in the cold leg of a PWR, with the break being located at the centerline of the cold leg between the pump discharge and the vessel. The flow area of the break for the experiment was equivalent to 2.5% of the flow area of a single cold leg. The 2.5% break size was chosen to ensure that the nominal break flow rate during the transient would be greater than the HPIS flow rate, thus providing the potential for a continuous system depressurization and depletion of the primary system liquid inventory. Initial fluid conditions, trip set points, and system geometry were specified to match as closely as possible typical PWR operating conditions.

The results from the Semiscale small break test, which are discussed in this paper, will be used primarily as part of an experimental data base for the development and assessment of analytical techniques used to predict the behavior of a PWR that undergoes a small break LOCA. Because of potential scaling distortions inherent in small-scale systems such as the Semiscale system, the test results cannot be assumed to be typical of a PWR and thus are not directly applicable to a full-size PWR system. The results can, however, be used as a means of identifying thermal-hydraulic phenomena that may be important in a small break LOCA in a PWR system.

SYSTEM DESCRIPTION

The Semiscale Mod-3 experimental facility is a small-scale system with components that represent the principal physical features of a full-size PWR system. The Mod-3 system, shown in Figure 1, consists of a pressure vessel with simulated reactor internals and an external downcomer assembly; an intact loop and broken loop with components that are representative of PWR components; a pressure suppression system with suppression tank and header; and a simulated ECC injection system with accumulators and injection pumps. The system is designed to operate at typical PWR pressure and temperature (i.e., 15.5 MPa and 594 K). The total primary system liquid volume is about 0.195 m^3, which is approximately 1/1700 the volume of a full-size PWR. The overall system scaling rationale is based on applying the Mod-3 versus PWR core thermal power ratio to the primary system volume. This rationale results in the same power-to-volume ratio in the Mod-3 system as exists in a PWR system. Additional design and scaling requirements are to preserve, as closely as possible, the full-scale elevations and volume distributions of the primary system flow circuit, and to maintain flow velocities sufficiently high to preserve representative flow regime behavior. Also, primary system flow resistances are scaled on a one-to-one basis with PWR components, producing full-scale pressure drops throughout the system.

The intact loop of the Semiscale Mod-3 system contains 75% of the total loop volume and is representative of three operating loops of a four-loop PWR system. The intact loop consists of primary coolant piping, a pressurizer, a tube-in-shell steam generator, and a circulating pump. The steam generator is designed to model the Loss-of-Fluid Test (LOFT) facility steam generator from volume scaling as well as component elevation considerations, and is not full length relative to a PWR steam generator.

The broken loop is a volume-scaled representation of one loop of a four-loop PWR system and contains 25% of the total loop volume. The broken loop includes primary coolant piping, a circulating pump, a piping rupture assembly, and a full-length tube-in-shell steam generator designed to model a PWR

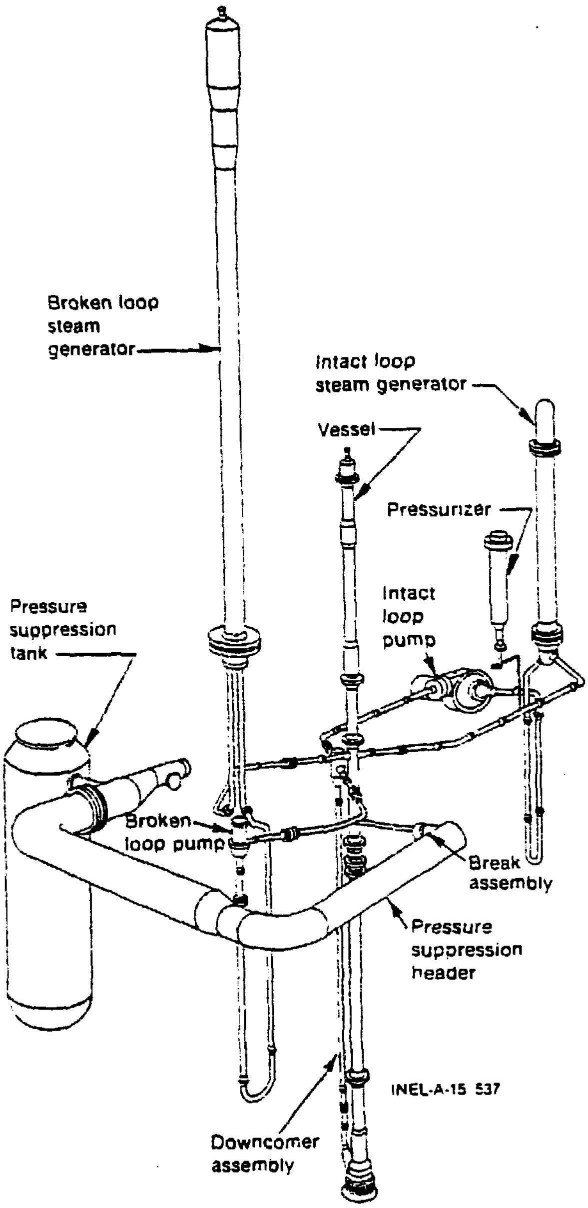

Figure 1. Semiscale Mod-3 cold leg break configuration - communicative.

U-tube steam generator. The piping rupture assembly contains diaphragm rupture discs and a break orifice to provide the desired break area. The break orifice configuration for the small break test discussed herein is shown in Figure 2. The break orifice diameter was sized sufficiently large that the nominal break flow rate throughout the early part of the experiment would be greater than the HPIS flow rate.

The core simulator in the Mod-3 system consists of a 5 by 5 matrix of electrically heated rods with typical PWR fuel rod diameter (1.07 cm). Each rod has a heated length of 3.66-m (which is identical to the length of a PWR nuclear fuel rod) and an axial

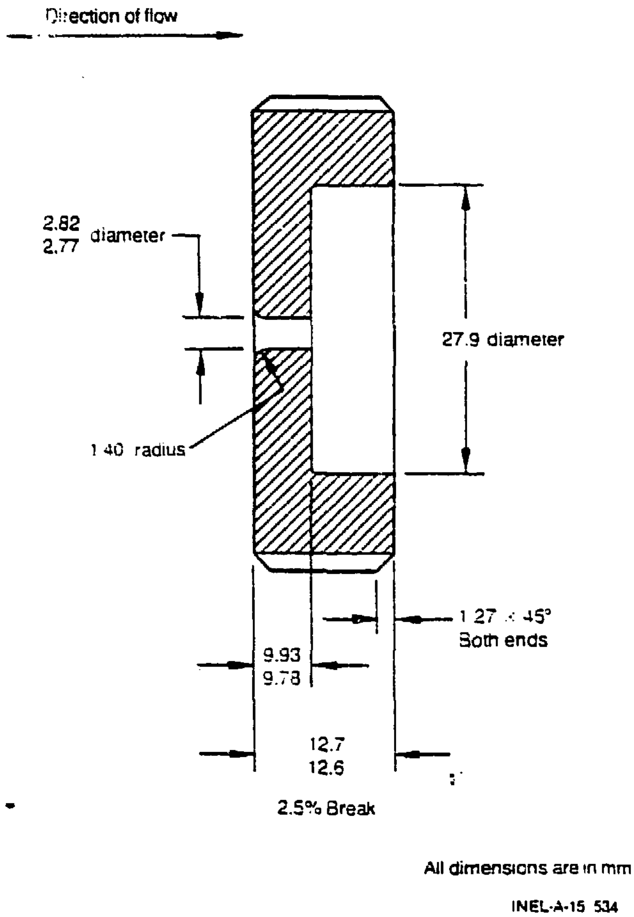

Figure 2. Semiscale 2.5% break nozzle configuration.

peaking factor of 1.58. The rods are positioned and held in the core with 10 grid spacers located on 40.01-cm centers along the length of the active zone of the core assembly. The grid spacers maintain the heater rods on a typical PWR pitch of 1.43-cm. The elevation of the core simulator relative to other portions of the pressure vessel is indicated in Figure 3.

The Mod-3 coolant injection system is designed to be representative of the emergency core cooling system (ECCS) of a typical PWR. The coolant injection system is composed of pressurized nitrogen-water accumulators, a low pressure pumped injection subsystem, and a high pressure pumped injection subsystem. Coolant is injected into both the intact loop and broken loop.

The pressure suppression system employed in the Mod-3 system is designed to simulate the back pressure created by the containment building in a PWR system. The system consists of a header section and a pressure tank with an internal downcomer. The blowdown effluent is directed into the pressure tank through the header section. The pressure tank is maintained partially full of subcooled water and the downcomer pipe projects below the water surface to accommodate the blowdown effluent. Pressure in the header section is maintained within a specified operating range by injecting steam from an auxiliary steam supply system.

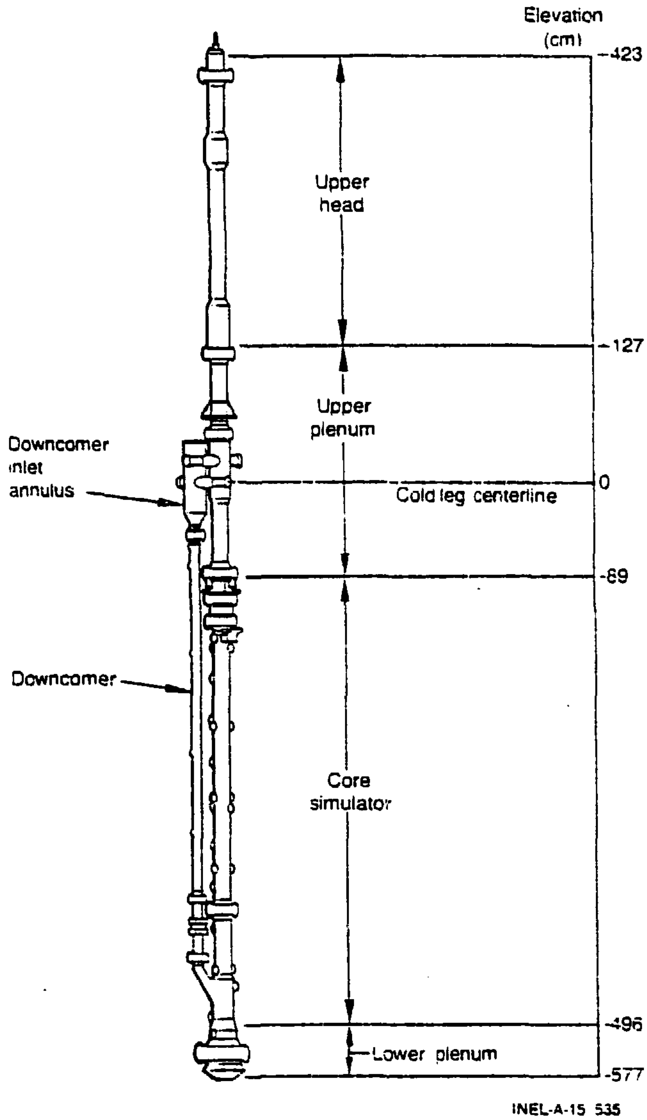

Figure 3. Semiscale Mode-3 vessel configuration.

TEST PROCEDURE AND OPERATION CONDITIONS

For the 2.5% break experiment, the system was brought to initial conditions that were representative of the conditions in a PWR operating at full power. After the system had equilibrated at the initial conditions, the test was initiated by rupturing discs downstream of the break orifice which allowed the primary system fluid to blow down through the break orifice. The sequence of events following rupture was similar to the sequence used in evaluation model analysis of small break loss-of-coolant accidents, and included a simulated loss of offsite power. The core power decay (American Nuclear Society power decay curve + 20%)[5] and coastdown of the intact and broken loop pumps were initiated on a low pressure trip and occurred 3.4 s after the pressurizer pressure reached 12.48 MPa (i.e., at about 20.9 s after rupture). The intact and broken loop steam generator feedwater valves were closed at about 25.9 s after rupture, and auxiliary feedwater flow did not begin until 63.4 s later. High pressure injection flow began at 45.9 s after rupture, and the flow rate was controlled by an on-line computer to follow a predetermined flow versus pressure curve. Accumulator flow began once the system pressure decreased below 4.14 MPa. At 3950 s after rupture, steam generator secondary feed-and-bleed was initiated to determine the effect on the primary system depressurization rate. The test was terminated at about 4600 s, after allowing the low pressure injection system (LPIS) coolant to be injected for sufficient time to ensure that primary system liquid inventory recovery was underway.

DISCUSSION OF RESULTS

The response of the Mod-3 system to the 2.5% cold leg break simulation was dominated by the fact that the nominal break flow rate was greater than the HPIS flow rate for much of the early part of the test (i.e., before accumulator flow began). As a result, the primary system depressurization was continuous during this period, and the system liquid inventory exhibited a gradual decrease as liquid was depleted through the break. The liquid inventory depletion caused the steam/water mixture level in the vessel to drop slightly below the top of the core just prior to the initiation of accumulator flow. However, no core temperature excursion was observed. Once accumulator injection began, the rate of addition of liquid to the primary system was sufficient to ensure that the core remained covered, and the vessel liquid inventory increased.

In the following sections, the overall system thermal-hydraulic response during the experiment is discussed, and the factors that influenced the system behavior are identified. Included is a comparison of test data with results obtained from calculations using the RELAP4/MOD7[1] computer code which is a thermal-hydraulics code used to predict the transient response of water cooled reactor systems or thermal-hydraulic test facilities such as the Semiscale Mod-3 system.

System Depressurization

For the 2.5% break experiment conducted in the Mod-3 system, the initiation of coolant injection from the different components of the emergency core cooling system was controlled by the system pressure (as is the case in a PWR system). Since the break flow rate for the experiment was greater than the HPIS flow rate, the severity of the core thermal response was dependent on whether or not the depletion of the primary system liquid inventory led to uncovering of the core significantly before the AIS setpoint pressure was reached. The rate of system depressurization and the factors which influenced the depressurization rate were therefore important in determining the eventual outcome of the test.

The upper plenum pressure response for the test is shown in Figure 4. Immediately following rupture, the system fluid was subcooled, and the depressurization was relatively rapid. At about 40 s, the system pressure decreased to the saturation pressure of the upper plenum and hot leg fluid. The resulting flashing of the hot fluid caused the decrease in the rate of system depressurization shown in Figure 4. The system then continued to depressurize at a fairly constant rate until about 850 s after rupture. The change in depressurization rate at this time was caused by an increasing system liquid inventory brought about by accumulator liquid injection (initiated at about 660 s after rupture). The resulting decrease in primary system volume occupied by steam tended to decrease the rate at which depressurization occurred. The relatively slow system depressurization after 850 s continued until about 3950 s after rupture, at which time bleeding of steam from the steam generator secondary sides was initiated to bring the primary system down to the LPIS initiation pressure. LPIS flow began at about 4300 s after rupture.

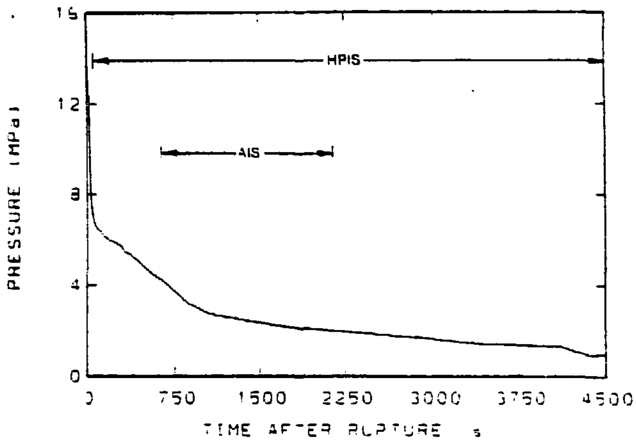

Figure 4. Upper plenum pressure response.

The primary system depressurization response during the test was affected by structural heat losses to the environment to a considerably greater degree than would be expected to occur in a PWR. For comparison, Table 1 presents total structural heat losses for the Semiscale Mod-3 system and for a PWR, both operating at 100% power. As indicated in Table 1, structural heat losses in the Mod-3 system are significantly larger (relative to the core power) than in a PWR, especially for low core power conditions. The sources of structural heat losses in the Mod-3 system include excessive component surface area to fluid volume ratios (relative to a PWR), a large number of instrumentation penetrations through piping and component insulation, and cooling of system instrumentation. The surface area to fluid volume ratios of components in the Mod-3 system are generally about nine times larger than in a PWR, thus providing a greater potential for heat transfer to the environment. Also, the cooling fluid required for system instrumentation represents a substantial heat sink for the primary system fluid.

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[1] The Idaho National Engineering Laboratory historical code configuration control number for the RELAP4/MOD7 computer program used for the calculations discussed herein is H007184B.

TABLE 1. TOTAL SYSTEM HEAT LOSSES

	PWR	Semiscale
Heat losses at 100% operating power (kW)	1500 - 2000[1]	80 - 117[2]
Percent of 100% power	0.054 - 0.071	4.0 - 5.85
Percent of 4% power	1.34 - 1.79	100.0 - 146.2

[1] Values for Combustion Engineering System 80.

[2] Experimentally obtained values for the Mod-3 system.

The combined structural heat losses in the Mod-3 system during a slow transient can be of the same order of magnitude as heat addition to the primary system fluid from the core heater rods. Figure 5 compares the core power for the 2.5% break experiment with the results of a calculation of the total structural heat loss to the environment. The calculation was performed using the RELAP4/MOD7 computer code. The model for the calculation was adjusted to have a heat loss of 80 kW at initial conditions, which is an experimentally determined value of the steady-state full power heat loss in the Mod-3 system. As indicated in Figure 5, structural heat transfer to the environment in the Mod-3 system must be considered equally as important as core heat transfer when evaluating the system depressurization response.

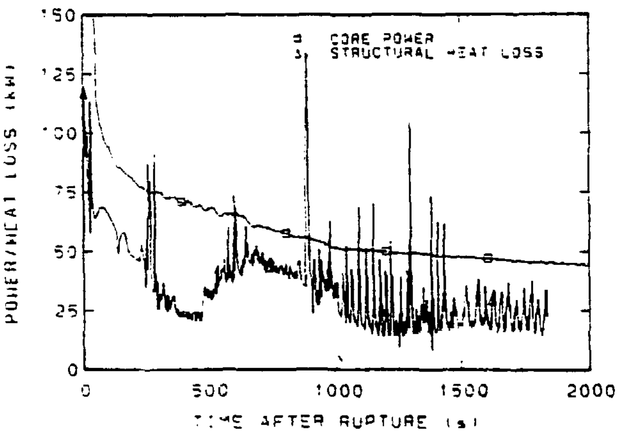

Figure 5. System structural heat loss relative to core power.

In order to investigate the sensitivity of the system depressurization to the structural heat losses discussed above, two RELAP4/MOD7 calculations were performed, both of which simulated the 2.5% cold leg break experiment. The first calculation modeled heat losses to the environment equivalent to 80 kW at initial conditions, while the second calculation assumed no heat losses to the environment. The system depressurization response obtained from the calculations is presented in Figure 6. Included for

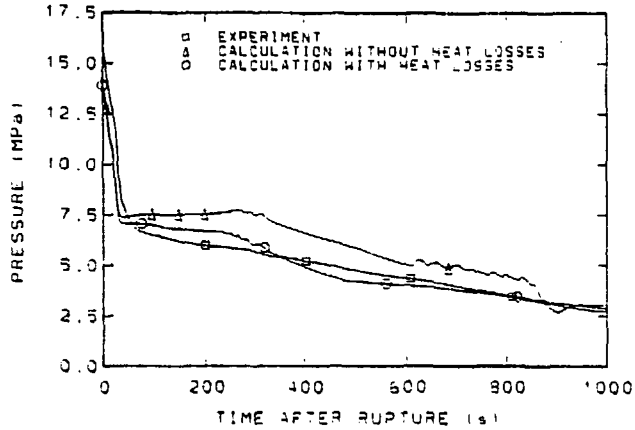

Figure 6. Comparison of the system pressure response to RELAP4 calculations with and without system heat losses.

comparison in the figure is the system pressure obtained from the test. As indicated in Figure 6, the calculation assuming structural heat losses compares very favorably to the measured system pressure. On the other hand, the calculation assuming no heat losses (which is more representative of a PWR) remained considerably above the calculation with structural heat losses (as well as above the measured system pressure). These results indicate that structural heat transfer to the environment in the Mod-3 system tends to hasten system depressurization, and causes early initiation of the AIS flow relative to what may be expected in a PWR. The early initiation of the AIS flow would, in turn, lessen the severity of uncovering the core in the Mod-3 system in that liquid recovery of the core would occur earlier than might otherwise be expected. Thus, when attempting to evaluate PWR response to a small break LOCA in terms of the Mod-3 test results, the excessive structural heat loss and resulting effects on system depressurization must be taken into consideration.

Loop Response

The loop hydraulic response for the 2.5% break experiment was characterized by an initial gradual depletion of the liquid inventory in the steam generator and hot leg of both the intact and broken loops, followed by a rapid blowout of the liquid remaining in the intact loop pump suction and cold leg. Immediately following rupture, loss of primary system fluid out the break was made up by flow of pressurizer fluid into the intact loop hot leg, and the entire system remained liquid solid. Once the pressurizer fluid was depleted and the primary system depressurized to the saturation pressure of the hot leg fluid (which occurred by about 40 s after rupture), fluid in the steam generator and hot leg of both loops began to flash to steam. A gradual draining of both steam generators then occurred, until by about 140 s after rupture, the steam generator side of the pump suction leg in both loops began to void. The liquid in the pump suction legs at this time is referred to as the loop seal in that it prevents steam in the vessel and hot legs from flowing toward the break. The existence of the loop seal had the tendency to depress the mixture level in the vessel, as steam generated in the core region maintained a somewhat higher pressure in the upper plenum and hot legs than existed near the break location.

As the system depressurized further, the voiding occurring in the downflow sections of both the intact and broken loop pump suction legs continued, and voiding also began (at about 180 s after rupture) in both the intact and broken loop cold legs. By about 270 s after rupture, the liquid level in the intact loop pump suction downflow reached the bottom of the pump suction leg, and the loop seal began to blow out. The system liquid distribution at this time is illustrated in Figure 7. The net effect of the blowout of the intact loop seal was to clear liquid remaining in the horizontal sections of both the intact and broken loop cold legs, and also to force some liquid down the downcomer and into the core region. The system liquid distribution immediately following the blowout of the intact loop seal is shown in Figure 8. Once the intact loop seal blowout occurred, the pressure differential between the hot and cold legs equalized and there was no further driving potential to blow out the loop seal in the broken loop. As a result, the broken loop pump suction leg remained partially filled with liquid for the duration of the test, thus limiting steam flow in the hot leg of the broken loop.

As discussed previously, AIS flow into both the intact and broken loop cold legs was initiated at about 660 s after rupture. During the injection period, the accumulator pressures closely followed the system pressure, and the nitrogen gas expansion for this slow depressurization was nearly isothermal. Because of the small differential pressure between

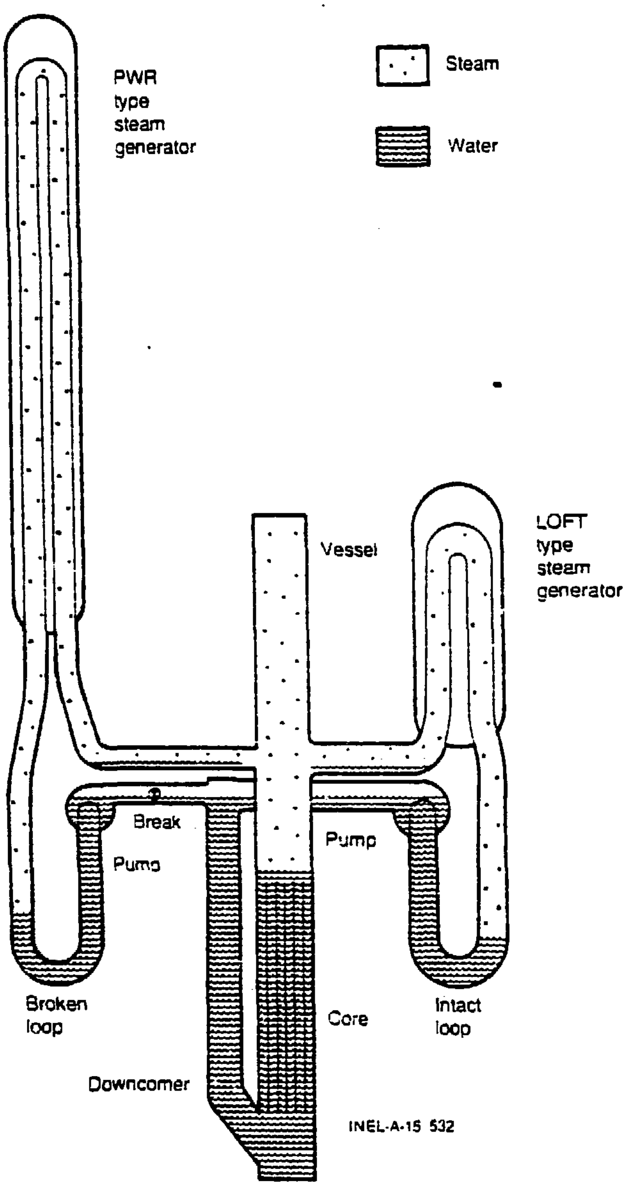

Figure 7. Illustration of the system mass distribution just prior to the loop seal blowout.

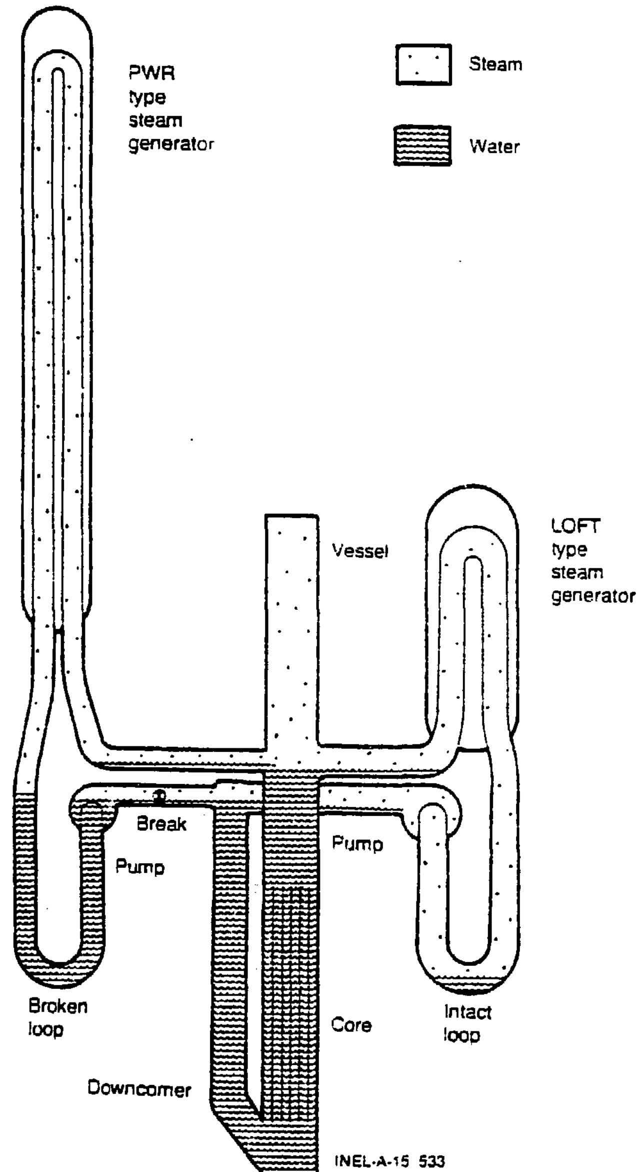

Figure 8. Illustration of the system mass distribution following the loop seal blowout.

the accumulators and the system, the AIS flow rate into the system was small. However, the total ECC injection rate (AIS plus HPIS) was greater than the break flow rate, and the system liquid inventory increased. Nevertheless, although most of the intact loop accumulator liquid passed into the downcomer and thus was available for cooling the core, much of the broken loop accumulator liquid appears to have remained in the pump suction region or to have exited the system through the break.

Break Response

The mass flow rate out the break was calculated by combining a momentum flux measurement taken downstream of the break orifice with a density obtained by assuming an isenthalpic expansion (to the downstream pressure) of the fluid upstream of the break orifice. As indicated in Figure 9, the break flow dropped off immediately following rupture as the system depressurized at a relatively rapid rate.

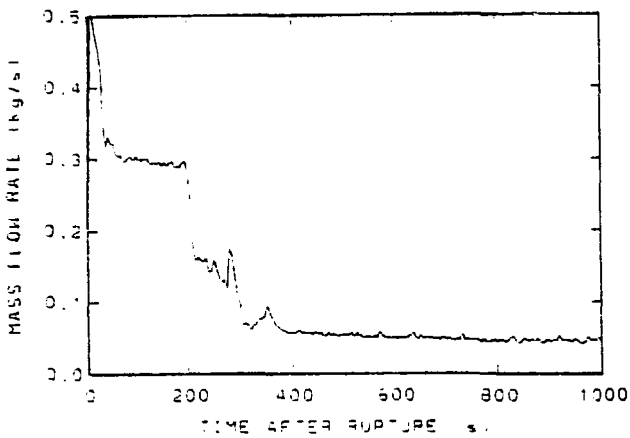

Figure 9. Break flow response.

Once saturation occurred (at about 40 s after rupture), the system depressurized at a much slower rate (Figure 4), and the break flow rate leveled off. The decrease in break flow between about 200 and 300 s was due to an intermittent uncovering of the break orifice as the liquid level in the loop decreased. After 300 s, flow through the break was predominantly steam. The break flow rate dropped below the HPIS flow rate once the system depressurized to about 3.10 MPa, i.e., by about 900 s after rupture.

Core Thermal-Hydraulic Response

Depletion of the liquid in the vessel, and in particular in the core region, began once the system pressure dropped to the saturation pressure of the upper core and upper plenum fluid. Figure 10 shows the fluid densities at several elevations from the bottom to the top of the core. The onset of bulk boiling is indicated by the decrease in fluid densities in the upper portion of the core beginning at about 40 s after rupture. Boiling of the core fluid at this time was enhanced by the continued primary system depressurization combined with the essentially constant level of the core decay power. As a result, the overall core void fraction increased steadily as indicated by the decrease in fluid densities shown in Figure 10.

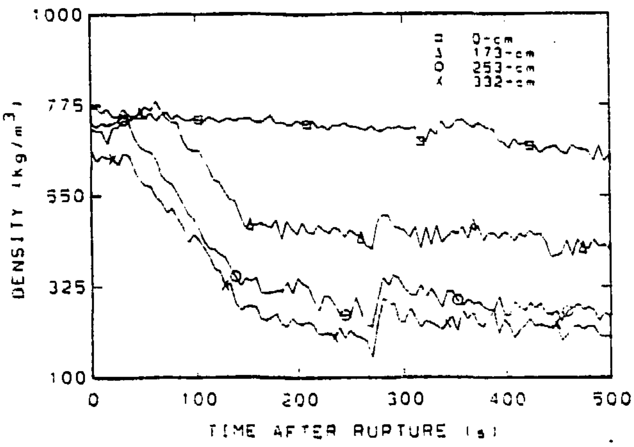

Figure 10. Comparison of fluid densities in the core. (Elevations shown are above bottom of heated core.)

Blowout of the loop seal (discussed previously) had a considerable effect on the core hydraulic response although it had essentially no effect on the core thermal behavior. The slight increase in the core fluid densities shown in Figure 10 at about 270 s after rupture is a result of additional liquid being forced into the downcomer and core as the loop seal blowout occurred. The net increase in the core liquid level at this time was equivalent to about 20% of the liquid available in the core just prior to the loop seal blowout. However, since the core was liquid covered both before and after the blowout occurred, no noticeable change in the core heater rod cladding temperatures was observed.

Following the blowout of the loop seal, the vessel liquid inventory continued to decrease as a gradual boiloff of the liquid in the core region occurred. Figure 11 shows the downcomer and core

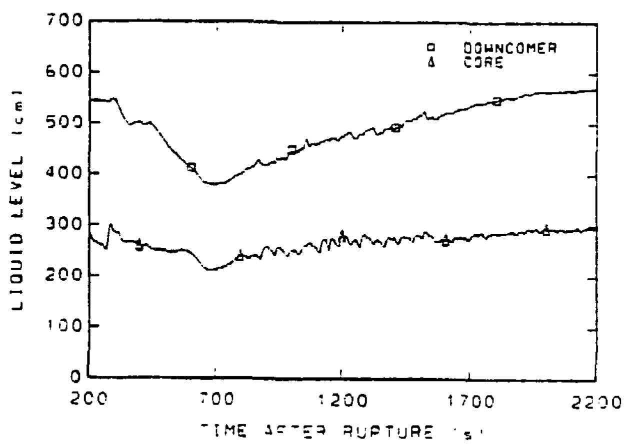

Figure 11. Downcomer and core collapsed liquid level. (Elevations shown are above bottom of heated core.)

collapsed liquid levels as calculated from differential pressure measurements, and indicates the depletion of the vessel liquid inventory between about 300 and 700 s after rupture. The collapsed liquid levels shown in Figure 11 are representative of what the liquid level would be assuming the fluid contained no voids. The actual liquid/steam mixture level in the core was greater than the collapsed level shown in the figure, and for most of the test (with the exception of a brief period around 630 s) was above the top of the core.

By about 630 s after rupture, the vessel liquid inventory had decreased sufficiently causing the liquid/steam mixture level to drop slightly below the top of the core. The uncovering of the top of the core at this time is indicated in Figure 12 which shows the fluid densities in the upper elevations of the core. The density at the 342-cm elevation is representative of a steam environment, while the

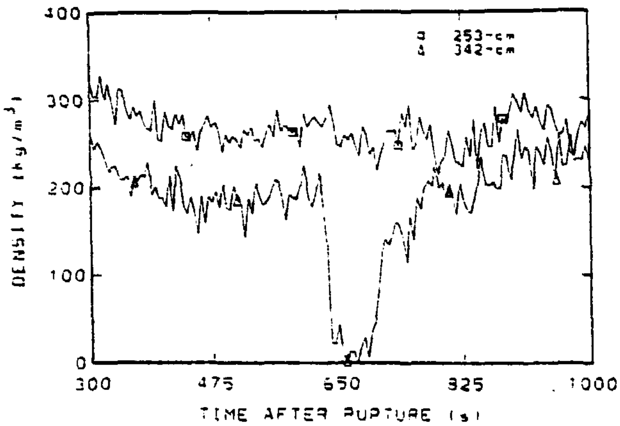

Figure 12. Fluid densities in the upper portion of the core. (Elevations shown are above bottom of heated core.)

density at the 253-cm elevation is representative of a two-phase mixture. The heater rod cladding temperatures at the very top of the core exhibited a small increase as dryout occurred. However, rod cladding temperatures throughout the rest of the core (being in a two-phase environment) remained a few degrees above the fluid saturation temperature. Figure 13 compares the cladding temperatures at various elevations in the core with the fluid saturation temperature. The slight temperature increase which occurred at the 354-cm elevation is typical of the response observed at several locations in the uppermost regions of the core.

By about 660 s after rupture, the system had depressurized to the AIS setpoint pressure, and the resulting ECC injection caused the mixture level to increase above the top of the core (Figure 12) cooling the few heater rod locations that exhibited temperature increases. After this time the gradual increase in liquid inventory in the vessel (Figure 11) resulting primarily from the accumulator injection, prevented any further temperature excursions in the core. The core remained liquid covered throughout the remainder of the test.

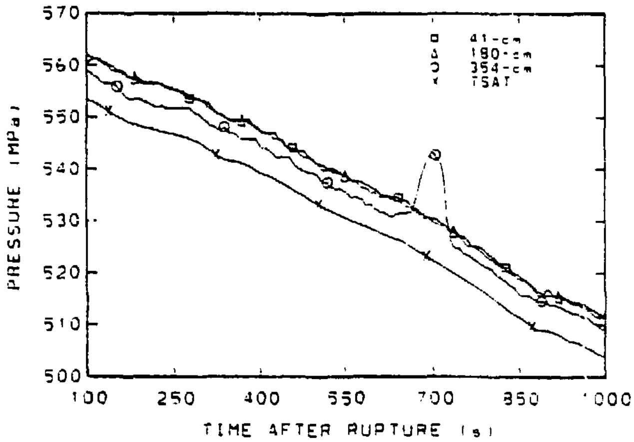

Figure 13. Comparison of cladding temperatures at the 41-, 190-, 180-, and 354- cm elevation in the core with the fluid saturation temperature. (Elevation shown are above bottom of heated core.)

Calculated System Response

This section presents a brief description of the model used in the RELAP4/MOD7 calculation for the experiment and compares the test data with results obtained from calculations. The model nodalization used in the RELAP4/MOD7 calculations to represent the Mod-3 system is shown in Figure 14. The model included 38 control volumes and 54 junctions. Table 2 provides a physical description of each control volume. The flow junctions used in the model connect the individual control volumes, as well as connecting fill volumes to the control volumes. Fill junctions were used to represent the HPIS, LPIS, and the intact and broken loop steam generator secondary water supply. A total of 50 heat slabs were used to represent heat conducting solids in contact with the coolant in the core, downcomer, steam generators, vessel, and piping. Also, saturated fluid was used in the accumulator, rather than the ambient temperature accumulator fluid used in the test. The use of saturated accumulator fluid was necessitated for the calculation because ambient temperature fluid caused an excessively rapid system depressurization once AIS flow was initiated. The increased rate of system depressurization resulted from complete mixing of the subcooled ECC liquid with the saturated primary fluid at the accumulator injection points. This complete mixing is an inherent limitation to the RELAP4 code, and is not representative of the actual response observed in the test.

The more significant code analytical options employed in the calculation are as follows: The Henry-Fauske and homogeneous equilibrium (HEM) critical flow models were used for the subcooled and two-phase break flow regimes, respectively. A break flow multiplier of 1.0 was used during both the subcooled regime and the saturated regime. Vertical slip was used in the model at all downcomer, core, and support and guide tube junctions. To be consistent with the use of slip in the core, the bubble rise model was not used in either the upper or lower

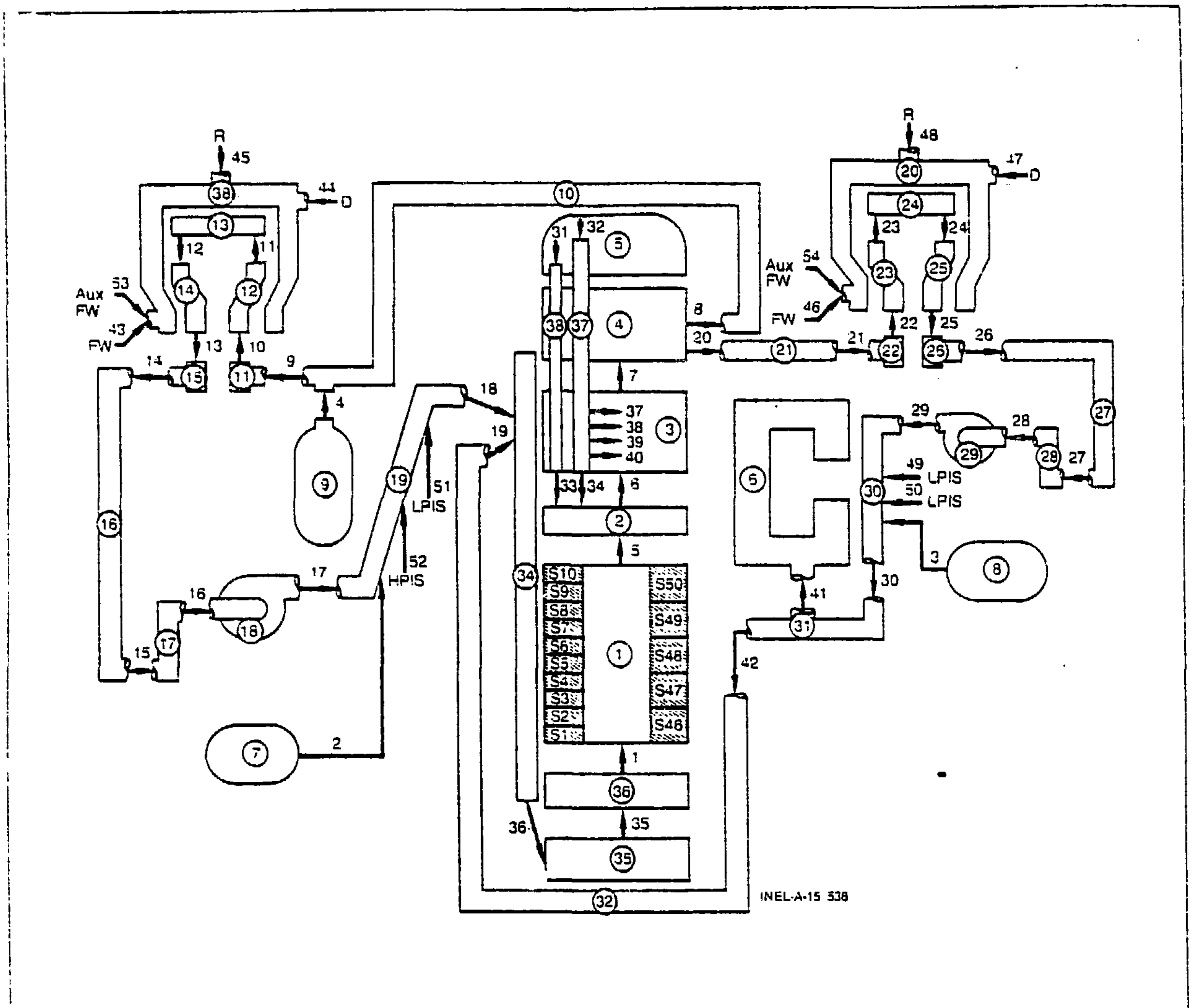

Figure 14. RELAP4/MOD7 nodalization diagram for the Mod 3 system.

plenums. The standard bubble rise model was used in the downcomer, core, upper head, pressurizer, pump suctions, and steam generator secondaries.

Overall, the system thermal-hydraulic response was well predicted using the RELAP4/MOD7 code. The calculated system pressure (shown in Figure 6) exhibited essentially the same trends as the test data, and generally accurately predicted the magnitude of the system pressure. The good prediction of the pressure response, in turn, is dependent on accurately predicting the overall system hydraulic behavior. Figure 15 compares the actual system mass distribution as a function of time with the distribution obtained from the calculation. Included in Figure 15 are the mass inventories in the intact loop, the broken loop, and the vessel (including the downcomer, core, lower and upper plenums, and upper head). The actual liquid inventories in each portion of the system were obtained from differential pressure measurements (to determine liquid levels), as well as fluid density measurements, and are thus accurate to only about 10%. As indicated in Figure 15, however, the calculated mass inventories

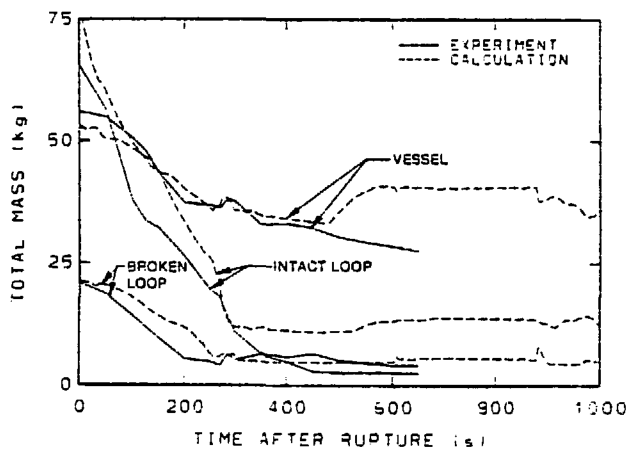

Figure 15. Measured versus calculated system mass distribution.

TABLE 2. RELAP4/MOD7 MODEL CONTROL VOLUME DESCRIPTION

Control Volume	Description
1	Core
2	Bottom of the upper plenum
3	Mid-volume of the upper plenum
4	Top volume of the upper plenum
5	Upper head
6	Pressure suppression vessel
7	Accumulator - intact loop
8	Accumulator - broken loop
9	Pressurizer
10	Intact loop hot leg
11	Intact loop steam generator inlet plenum
12, 13, 14	Intact loop steam generator tube bundle
15	Intact loop steam generator outlet plenum
16	Intact loop pump suction - downflow
17	Intact loop pump suction - upflow
18	Intact loop pump
19	Intact loop cold leg
20	Broken loop steam generator secondary
21	Broken loop hot leg
22	Broken loop steam generator inlet plenum
23, 24, 25	Broken loop steam generator tube bundle
26	Broken loop steam generator outlet plenum
27	Broken loop pump suction - downflow
28	Broken loop pump suction - upflow
29	Broken loop pump
30	Broken loop pump discharge
31	Break assembly
32	Broken loop cold leg
33	Support tubes
34	Inlet annulus and downcomer
35	Lower plenum
36	Core mixer box
37	Guide tube
38	Intact loop steam generator secondary

compare favorably with the observed system response. In particular, the depletion of the intact loop liquid inventory in the calculation led to a blowout of the loop seal at about the same time as observed in the test. Figure 16 compares the fluid density measured at the inlet of the intact loop pump with the density obtained from the calculation, and indicates that the loop seal voided only about 20 s earlier in the calculation than in the test. As a result, the calculated intact loop flow rate exhibited the same trends as the test data as shown in Figure 17 which compares the measured intact loop volumetric flow rate with that obtained from the calculation.

The calculation of the liquid level in the core region was not as favorable as the prediction of the remainder of the system response, although the calculated cladding temperature response was not adversely affected. A comparison of the collapsed liquid level in the core with the collapsed level obtained from the calculation is shown in Figure 18.

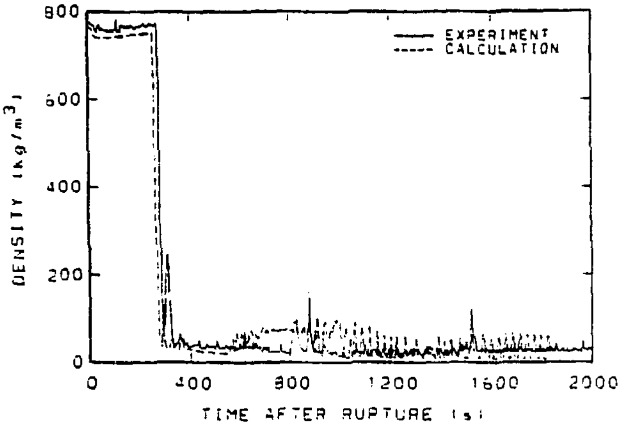

Figure 16. Comparison of the fluid density near the intact loop pump inlet with the density obtained from the RELAP4 calculation.

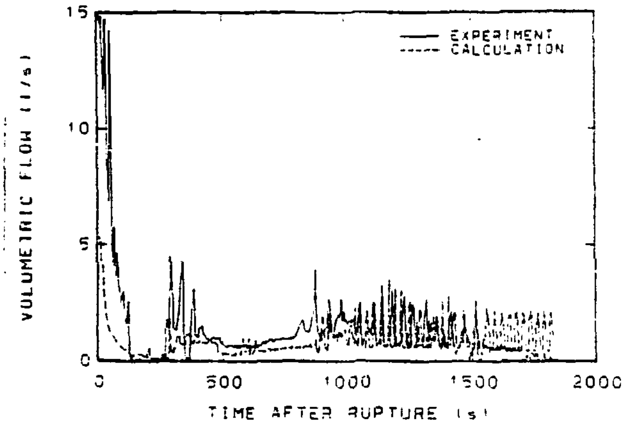

Figure 17. Comparison of the intact loop cold leg volumetric flow with the volumetric flow obtained from the RELAP4 calculation.

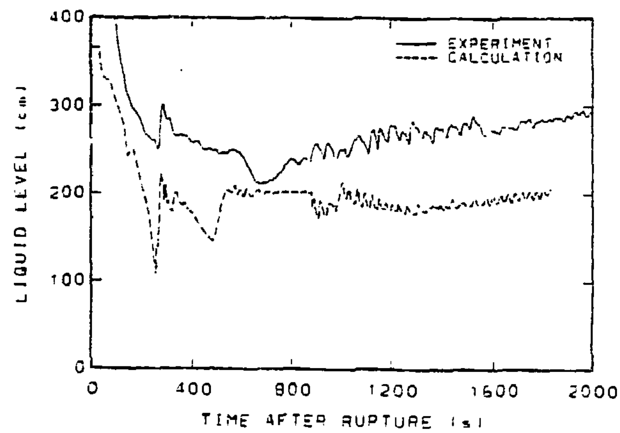

Figure 18. Comparison of the core collapsed liquid level with the level obtained from the RELAP4 calculation. (Liquid levels shown are above bottom of heated core.)

The calculation predicted more voiding of the core than was observed during the test, and also exhibited a brief dryout near the top of the core prior to the blowout of the loop seal. As a result, the calculation indicated a small temperature increase at this time as shown in Figure 19, which compares the measured versus calculated temperature response at the very top of the core. As indicated in Figure 19, test data did not exhibit the temperature increase prior to the loop seal blowout, and in fact the vessel mixture level was above the top of the heated length of the core during this time period. Also, the calculation did not predict the minor temperature increase observed at some locations in the uppermost regions of the core in the test just prior

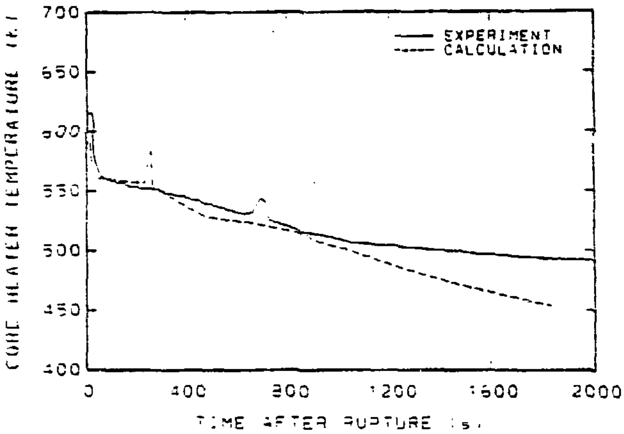

Figure 19. Comparison of a typical cladding temperature response near the top of the core with the temperature obtained from the RELAP4 calculation.

to accumulator injection. Although the collapsed liquid level for the calculation was below that observed in the test just prior to accumulator injection, the calculated mixture level was higher than the measured mixture level. As a result, no temperature excursion occurred in the calculation during this time period.

CONCLUSIONS

Based on the results of the 2.5% cold leg break experiment conducted in the Semiscale Mod-3 system, the following general conclusions have been made relative to the effects of the break on the system thermal-hydraulic response. Although the 2.5% break size was sufficiently large to provide the potential for uncovering the core, no significant uncovering of the core was observed during the transient (although the vessel mixture level did drop somewhat below the top of the core for a brief period just prior to AIS initiation). Also, both during the period of AIS flow and following termination of AIS flow, the rate of addition of ECC liquid to the primary system was sufficient to ensure that the core remained covered and that the vessel liquid inventory increased.

In relating the experimental results to the possible PWR response to a similar small break LOCA, the effects on the system depressurization of the excessive structural heat losses in the Mod-3 system must be taken into consideration. Structural heat losses in the Mod-3 system tend to hasten system depressurization relative to what may be expected in a PWR. Thus, the time required for a PWR system to depressurize to the AIS setpoint pressure may be somewhat greater than was observed experimentally, and more depletion of the primary system liquid inventory could occur.

In general, calculations of the system thermal-hydraulic behavior performed with the RELAP4/MOD7 computer code predicted the test results well. The calculated loop hydraulics as well as the overall primary system liquid distribution during the transient were highly representative of the experimental results. In addition, although the calculation indicated a somewhat lower vessel mixture level than was observed experimentally, the calculated core thermal response was similar to the measured core thermal behavior.

REFERENCES

1 Reactor Safety Study-An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, United States Nuclear Regulatory Commission, WASH-1400 (NUREG-75/014), October 1975.

2 Cartmill, C. E., "Analysis of Standard Problem Six (Semiscale Test S-02-6) Data," EG&G Idaho, Inc., TREE-NUREG-1050, August 1977.

3 Sackett, K. E., "Experiment Data Report for Semiscale Mod-3 Small Break Test S-07-10 (Baseline Test Series)," NUREG/CR-1456, EGG-2035, May 1980.

4 Miyasaki, D. H., "Experiment Data Report for Semiscale Mod-3 Small Break Test Series (Test S-SB-2 and S-SB-2A)," NUREG/CR-1459, EGG-2038, June 1980.

5 American Nuclear Society Standards, Decay Energy Release Following Shutdown of Uranium- Fueled Reactors, ANS-5, October 1971.