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Author(s):

James P. Adams

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G. D. McPherson, Chief, LOFT Research Branch
Division of Reactor Safety Research, USNRC

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

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QUICK-LOOK REPORT ON LOFT NUCLEAR EXPERIMENT L9-1/L3-3

By

J. P. Adams

Approved:

Charles W. Solbrig

C. W. Solbrig, Manager
LOFT Program Office

Charles W Solbrig / for

L. P. Leach, Manager
LOFT Department

ABSTRACT

Loss-of-Fluid Test (LOFT) Experiment L9-1/L3-3, which was completed on April 15, 1981, consisted of two parts designated L9-1 and L3-3 that were completed sequentially. L9-1 simulated a multiple failure transient initiated from a loss-of-feedwater accident (LOFA) and had as its main objective, measurement of the effectiveness of a scaled power-operated relief valve (PORV) to control primary system pressure with a dry steam generator secondary. L3-3 simulated two recovery modes from a LOFA. The first mode consisted of turning off the primary coolant pumps and latching open the PORV to depressurize the primary system to the emergency core cooling system (ECCS) injection setpoint. The second mode consisted of shutting the PORV and refilling the steam generator to restore the secondary heat sink and remove decay heat through the secondary. ECCS injection was inhibited throughout L9-1 and L3-3 to better assess the effectiveness of the independent recovery modes and was used only after termination of L3-3 to recover the plant. In addition, a slow core uncovery (L8-1A) was attempted. This attempt was unsuccessful due to insufficient primary system mass depletion. Data from Experiment L9-1/L3-3 and from the preexperiment prediction calculated using RELAP5, an advanced one-dimensional system analysis code based on a nonhomogeneous, nonequilibrium, hydrodynamic model, are compared in this report. In L9-1, the PORV was successful in controlling primary system pressure with a dry steam generator secondary. In L3-3, each recovery method was effective in bringing the plant to a safe condition.

NRC FIN No. A6048 - LOFT Experimental Program

SUMMARY

Loss-of-Fluid Test (LOFT) Experiment L9-1/L3-3, which was completed on April 15, 1981, consisted of two parts designated L9-1 and L3-3 that were completed sequentially. L9-1 simulated a multiple failure transient initiated from a loss-of-feedwater accident (LOFA) and had as its main objective, measurement of the effectiveness of a scaled power-operated relief valve (PORV) to control primary system pressure with a dry steam generator secondary.

L3-3 simulated two recovery modes from a LOFA. The first mode consisted of turning off the primary coolant pumps and latching open the PORV to depressurize the primary system to the emergency core cooling system (ECCS) injection setpoint. The second mode consisted of shutting the PORV and refilling the steam generator to restore the secondary heat sink and remove decay heat through the secondary. ECCS injection was inhibited throughout L9-1 and L3-3 to better assess the effectiveness of the independent recovery modes and was used only after termination of L3-3 to recover the plant.

Data from Experiment L9-1/L3-3 and from the preexperiment prediction calculated using RELAP5, an advanced one-dimensional system analysis code based on a nonhomogeneous, nonequilibrium, hydrodynamic model, are compared in this report.

The experiment was initiated from operating conditions representative of a commercial pressurized water reactor (PWR). All feedwater was shut off to initiate the experiment and primary system temperature started to increase as the steam generator boiled dry. The reactor scrammed on high hot leg pressure at 65.4 s. Pressurizer spray controlled pressure until it was manually shut off at 1246 s (20 min, 46 s). The PORV then effectively controlled pressure by cycling open approximately 5% of the time until 3270 s (54 min, 30 s), when it was latched open, terminating L9-1 and initiating L3-3.

The primary pressure decreased to the saturation pressure of 12.3 MPa (1784 psia). This pressure was below the high-pressure injection system setpoint of 13.2 MPa (1915 psia) by 3329 s (55 min, 29 s) which would have initiated ECCS injection had it not been purposely locked out. The primary system pressure continued to decrease until the PORV was shut at 4850 s (1 h, 21 min) which terminated the first recovery mode simulation. The second recovery mode simulation was initiated at 5115 s (1 h, 25 min) by refilling the steam generator secondary. Operator-controlled secondary feed and bleed was initiated at 6712 s (1 h, 52 min) and the experiment was terminated at 9517 s (2 h, 39 min) prior to initiating ECCS injection.

A prediction of Experiment L9-1/L3-3 was made with the RELAP5 computer code. While the trends of the experiment were correctly predicted, magnitudes were significantly different due, at least in part, to differences in pressurizer spray effectiveness and primary system heat sinks.

Pressurizer spray was more effective in controlling primary system pressure than predicted. Spray prior to reactor scram delayed the time of scram to 65.4 s, approximately 9 s later than predicted which, together with a lower than predicted steam generator liquid level, resulted in less liquid mass in the steam generator secondary at reactor scram.

A combination of more effective than modeled pressurizer spray and larger than expected primary system heat sink reduced the primary system heatup rate to less than predicted. This resulted in a less integrated cycling time (open 5% of the time versus approximately 22% predicted) for the PORV in L9-1 and less total primary system mass depletion (24% depletion versus 35% predicted) throughout Experiment L9-1/L3-3 than was predicted. Most of this mass depletion discrepancy was encountered during L3-3.

Steam leakage through the main steam control valve reduced secondary pressure during the experiment but is not considered to have significantly affected primary system response.

Refilling the steam generator caused a shrink of the primary system fluid. This shrink resulted in the pressurizer completely draining and the reactor vessel mixture level decreasing approximately 0.2 m.

ACKNOWLEDGMENTS

A Quick Look Report is, due to the scope of the document and the short schedule for issuance, of necessity a team effort. This report is no exception. To list all personnel who assisted in the analysis for and preparation of this report would be impossible and yet, at the risk of error by ommission, I would like to give particular acknowledgment to the following: T. L. DeYoung, A. E. Sanchez, and T. L. Shrum on who I relied extensively for preparation of Sections 1, 2, and 5 and both appendixes; members of the Scientific Computing Division for preparation of the data figures; members of the LOFT Measurements Division for data preparation as well as many helpful discussions on the analysis results; and members of the LOFT Program Division, especially V. T. Berta and J. H. Linebarger for extensive discussions on both technical content and style. Finally, I wish to acknowledge G. Hammer and J. Isom for assistance in text preparation.

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DEFINITIONS

Flow reversal - the inception of negative flow in a system component or at a particular location in the system.

Flow rereversal - the reinception of positive flow in system piping, in a component, or at a particular location in the system.

Forced circulation - circulation (flow) caused by the pumps in the loop.

Loop circulation - positive loop flow which proceeds from the heat source (the core) to the heat sink (the steam generator) and then returns to the heat source.

Natural circulation - circulation (flow) caused by density gradients, induced by heat generation in the core and sustained by concomitant heat removal.

Positive flow - flow in the direction that occurs during normal operation in piping, a component, or a loop.

Pump seal - the U-shaped piping on the inlet side of the primary coolant pumps.

Reflux flow - condensation in steam generator primary tubes with concomitant fallback of condensed liquid film into the intact loop hot leg and reactor vessel upper plenum.

Subcooled blowdown - the period during a loss-of-coolant transient when subcooled fluid is leaving the system through the break and system fluid is saturated only in the pressurizer and downstream of the break.

Mass flux-induced flow - flow in the loops induced by mass influx or efflux (for example, break- or ECCS-induced flow).

Subcooled break flow - the period during a loss-of-coolant transient when subcooled fluid is leaving the system from at least one location.

Subcooled break flow - the period during a loss-of-coolant transient when subcooled fluid is leaving the system from at least one location.

Mass flux-induced flow - flow in the loops by mass influx or efflux (for example, break- or ECCS-induced flow).

Submeter (or subcooling meter) - the calculated value, from measured parameters, of the fluid subcooling in the reactor vessel upper plenum. Positive values indicate the fluid is subcooled.

Time zero - time of predefined transient initiating event (for LOFAs, defined as first indication of decrease in feed flow).

QUICK-LOOK REPORT ON LOFT NUCLEAR EXPERIMENT L9-1/L3-3

1. INTRODUCTION

Experiment L9-1/L3-3 was successfully completed on April 15, 1981, in the Loss-of-Fluid Test (LOFT) facility. This experiment consisted of two parts, L9-1 and L3-3. The first experiment, L9-1, addressed the issue of safely controlling primary coolant pressure without recourse to the emergency core coolant system (ECCS) after a loss-of-feedwater accident (LOFA), while the second, L3-3, stressed two methods of plant recovery from this accident. In addition, a slow core uncovery (L8-1A) was attempted. This attempt was unsuccessful due to insufficient primary system mass depletion. L9-1 was a multiple failure transient; specifically, a LOFA with no auxiliary feedwater injection and with delayed scram. L3-3 simulated an operator-controlled recovery from the L9-1 LOFA by shutting off the primary coolant pumps and depressurizing the primary system through the power-operated relief valve (PORV). An alternate recovery method was then tested by shutting the PORV, restoring the steam generator heat sink, and feeding and bleeding the secondary. The ECCS was inhibited throughout both L9-1 and L3-3 in order to study the two recovery modes and was only used for final primary system refill after termination of Experiment L3-3.

The main objectives of L9-1 and L3-3 were to evaluate the system response to a LOFA simulation with multiple failures and to evaluate the effectiveness of two methods of recovery from this transient. The ability of computer codes to calculate system thermal and hydraulic response to these conditions was also to be evaluated. The detailed experiment objectives required to satisfy the programmatic objectives for each part of Experiment L9-1/L3-3 are listed in Appendix A.

This report presents a preliminary examination of the LOFT plant performance during Experiments L9-1/L3-3 and a comparison of selected data from the RELAP5 preexperiment calculation. The LOFT integral test facility[1] is a scale model of a large pressurized water reactor (PWR).

The intent of the facility was to model the nuclear, thermal-hydraulic phenomena which would occur in a large PWR during a loss-of-coolant accident (LOCA). In general, components used in LOFT are similar in design to those of a large PWR and, because of this and the scaling philosophy employed, a LOFT loss-of-coolant experiment (LOCE) is expected to closely resemble a large PWR LOCA. The RELAP5 computer code[2] used for the preexperiment prediction is an advanced, one-dimensional, system analysis code based on a nonhomogeneous, nonequilibrium, hydrodynamic model. It features top-down structural design, with the significant programming elements coupled in modular fashion, and includes those thermal-hydraulic and mechanical models required to describe the processes that occur during blowdown of a large PWR.

This report contains an evaluation of plant performance for Experiment L9-1/L3-3 in Section 2, including a summary of specified and measured initial conditions in Table 1 and a listing of identifiable significant events in Table 2. Section 3 presents a summary of Experiment L9-1/L3-3 results, followed in Section 4 by conclusions reached from the preliminary examination of results reported in Section 3. Data plots are presented in Section 5 to support the experiment chronology in Section 2 and the discussion of results in Section 3. The plots presented also include comparisons of measured and predicted data. The predicted data were calculated by EG&G Idaho, Inc.[3] using the RELAP5[a] computer code. The LOFT system geometry is shown in Appendix B. Additional details of the LOFT system are presented in Reference 1.

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a. The version of RELAP5 used was RELAP5/MOD"1" Cycle 6, a production version of the RELAP5/MOD"1" code which is filed under Idaho National Engineering Laboratory Configuration Control Number F00181.

TABLE 1. INITIAL CONDITIONS FOR EXPERIMENT L9-1/L3-3

Parameter	Specified Value[a]	Measured Value
Primary Coolant System		
Mass flow rate (kg/s)	478.8 ± 6.3	479.1 ± 2.6
(x10^6 lbm/hr)	3.8 ± 0.05	3.80 ± 0.02
Hot leg pressure (MPa)	14.95 ± 0.1	14.90 ± 0.10
(psia)	2168 ± 15.0	2161 ± 19.5
Cold leg temperature (K)	556.8 ± 1.1	558.9 ± 1.3[b]
(°F)	542.5 ±	546.0 ± 2.3
Boron concentration (ppm)	As required to maintain temperature	631 ± 5
Hot leg temperature (K)	--	578.2 ± 1.8
(°F)		580.8 ± 3.2
Reactor		
Power level (MW)	50 ± 1	49.6 ± 0.9
Maximum linear heat generation rate (kW/m)	--	50.8 ± 3.6
(kW/ft)		15.5 ± 1.1
Control rod position (above full-in position) (m)	1.37 ± 0.01	1.38 ± 0.01
(in.)	54.0 ± 0.4	54.3 ± 0.4
Steam Generator Secondary Side		
Water level (m)[c]	0.25 ± 0.05	0.14 ± 0.08
(in.)	10.0 ± 2.0	5.6 ± 0.2
Water temperature (K)	--	545.0 ± 0.8
(°F)		521.0 ± 1.4
Pressure (MPa)	--	5.67 ± 0.08
(psia)		822.4 ± 11.6
Mass flow rate (kg/s)	--	27.0 ± 1.0
(lbm/s)		59.5 ± 2.2
Broken Loop		
Hot leg temperature (K)	556.9 ± 16.7	563.3 ± 2.6
(°F)	542.5 ± 30.0	553.9 ± 4.7

TABLE 1. (continued)

Parameter	Specified Value[a]	Measured Value
Broken Loop (continued)		
Cold leg temperature (K)	556.9 ± 16.7	557.6 ± 2.6
(°F)	542.5 ± 30.0	543.7 ± 4.7
Suppression Tank		
Liquid level (m)	1.27 ± 0.05	1.34 ± 0.10[b]
(in.)	50.0 ± 2.0	52.8 ± 3.9
Gas volume (m^3)	--	53.8 ± 2.8
(ft^3)		1900 ± 99
Water temperature (K)	d	358.0 ± 0.6
(°F)	d	184.4 ± 1.1
Pressure (gas space) (MPa)	d	0.136 ± 0.004
(psia)	d	19.7 ± 0.6
Pressurizer		
Steam volume (m^3)	--	0.43 ± 0.05
(ft^3)		15.2 ± 1.8
Liquid volume (m^3)	--	0.50 ± 0.05
(ft^3)		17.3 ± 1.8
Water temperature (K)	--	614.9 ± 1.3
(°F)		646.8 ± 2.3
Pressure (MPa)	--	14.93 ± 0.25
(psia)		2165 ± 36
Liquid level (m)	1.02 ± 0.05	0.92 ± 0.10[b]
(in.)	40.0 ± 2.0	36.3 ± 3.9

a. Listed values are as specified in the Experiment Operating Specification (EOS).[4] If no value is listed, that parameter is not specified by the EOS.

b. These values are out of the ranges specified by Reference 4.

c. Steam generator liquid level is referenced to zero power level, 2.95 m (9.68 ft) above the top of the tube sheet.

d. Values specified consistant with EOS requirements.

TABLE 2. EXPERIMENT L9-1/L3-3 CHRONOLOGY OF EVENTS

Event[b]	Time After Experiment Initiation[a] (s) Measured Data	Prediction Data
L9-1		
Main feedwater off	0.0	0.0
Pressurizer spray activated (15.32 MPa, 2221.98 psia, in pressurizer)	30.0 ± 0.1	24.0
Reactor scram (15.67 MPa, 2272.74 psia, in intact loop hot leg)	65.4 $^{+0.2}_{-0.0}$	56.5
Steam generator main steam control valve closed	77.2 $^{+0.0}_{-0.2}$	132
Steam generator liquid level reached bottom of indicating range (0.25 m, 0.83 ft, above tube sheet)	190 $^{+10}_{-20}$	67.0
Pressurizer spray valve cycling initiated (15.32 MPa, 2221.98 psia, in pressurizer)	208.9 ± 0.1	358.0
Pressurizer liquid level reached top of indicating range (1.83 m, 6.00 ft, above bottom)	1089.7 ± 29.9	896
Pressurizer spray valve cycling ended	1246.0 ± 0.1	1640.0
Experiment PORV cycling initiated (16.18 MPa, 2346.71 psia, in pressurizer)	1467.9 ± 0.1	968.0
L3-3		
Experiment PORV latched open	3269.9 ± 0.1	1628.0
Primary coolant pumps tripped off	3284.8 $^{+0.2}_{-0.0}$	1628.0
Primary coolant pump coastdown completed	3304.2 $^{+0.8}_{-0.0}$	1643.0

TABLE 2. (continued)

Event[b]	Time After Experiment Initiation[a] (s) Measured Data	Prediction Data
Upper plenum fluid reached saturation pressure	3329.4 ± 0.2	1628.0
Experiment PORV closed	4849.7 ± 0.1	3430.0
Steam generator secondary refill initiated	5114.6 + 0.2 - 0.0	3430.0
Natural circulation initiated	5205 + 10 - 5	---
Steam generator secondary refill completed	5746.4 + 0.2 - 0.0	4554.0
Pressurizer liquid level reached bottom of indicating range (0.06 m, 0.20 ft, above bottom)	5915 ± 5	3594
Steam generator secondary feed and bleed initiated	6712.2 + 0.2 - 0.0	5460.0
Experiment completed (secondary feed and bleed ended)[c]	9517.4 + 0.0 - 0.2	---

a. Experiment initiation defined as when main feedwater flow started to decrease.

b. Pertinent setpoint and level values are enclosed in parentheses.

c. Experiment was terminated just prior to ECCS injection initiation.

2. PLANT EVALUATION

An evaluation of plant performance is presented. The discussion summarizes the initial experimental conditions, the identifiable significant events, and the instrumentation performance for Experiment L9-1/L3-3.

Since Experiment L9-1 simulated a multiple failure transient which challenged the PORV, a 1.32×10^{-2} kg/s/MW (104.95 lbm/hr/MW) steam-scaled experiment PORV, geometrically similar to large PWR PORVs, was installed downstream of an instrumented spool piece in parallel with the plant PORV. This relief capacity corresponded to the minimum relief capacity in a generic Westinghouse PWR design with an equivalent flow area of 1.69×10^{-3} m^2 (0.0182 ft^2), and resulted in an experiment PORV flow area of 2.48×10^{-5} m^2 (0.0002668 ft^2).

2.1 Initial Experimental Conditions

A summary of the specified and measured system conditions immediately prior to Experiment L9-1/L3-3 is given in Table 1. All of the initial conditions were within specified limits except intact loop cold leg temperature (0.18% high), steam generator liquid level (1.75% low), pressurizer liquid level (3.43% high), and suppression tank liquid level (6% low). These out-of-specification values did not adversely affect the experiment results.

2.2 Chronology of Events

Identifiable significant events for Experiment L9-1/L3-3 are listed in Table 2, where their times of occurrence are compared with the times predicted by the RELAP5 calculation. An annotated primary-system/secondary-system depressurization history is shown in Figure 1.

The main feedwater pump was tripped off and the time when its flow started to decrease was designated as L9-1 initiation, time zero. Rising primary system pressure tripped pressurizer spray on at 30 s, followed by reactor scram on high hot leg pressure at 65 s (1 min, 5 s). Primary coolant shrink was essentially halted with steam generator secondary steam control valve closure at 77 s (1 min, 17 s). This coincided with the time at which the steam generator secondary liquid level had dropped to 21% of its initial level. Primary system pressure again increased to the pressurizer spray setpoint at 209 s (3 min, 29 s) and was controlled by pressurizer spray until that was manually terminated at 1246 s (20 min, 46 s). Primary system pressure then increased further to the experiment PORV setpoint, which started to cycle at 1468 s (24 min, 28 s). The experiment PORV was allowed to cycle and control primary system pressure for 1800 s, until it was manually latched open at 3270 s (54 min, 30 s), the end of L9-1 and the beginning of L3-3.

After latching the experiment PORV open, primary system depressurization was immediate and rapid (0.10 MPa/s, 14.7 psia/s). The upper plenum reached saturated conditions within 60 s, and the experiment PORV was closed at 4850 s (1 h, 21 min) to end the first LOFA recovery mode. Steam generator secondary refill was then initiated at 5115 s (1 h, 25 min), to begin the second LOFA recovery mode, and was terminated at 5746 s (1 h, 36 min). Secondary feed and bleed was started 966 s (16 min, 6 s) later and finally ended at 9517 s (2 h, 39 min), just prior to ECCS injection and final plant recovery.

2.3 Instrumentation Performance

The instrumentation used for Experiment L9-1/L3-3 was the same instrumentation used for Experiment L3-6/L8-1[5,6] with some additions and deletions. The additional instrumentation provided measurement of density, temperature, pressure, momentum flux, and mass flow upstream of the experiment PORV; fluid velocity and temperature in the intact loop and secondary coolant system; momentum flux in the intact loop; and differential pressures in both intact and broken loops. In addition to these changes, the Experiment L3-6/L8-1 break line instrumentation was deleted for this experiment.

During the experiment, several parameters were monitored on cathode ray tubes in the control room, visitor display room, and technical support center in real time to determine the thermal and hydraulic status of the core. The monitored parameters included:

1. Liquid level--both in the upper plenum and in the core

2. Fluid and cladding temperatures--both in the upper plenum and in the core.

There were 722 instruments recorded for evaluation of the experimental results. Of the number examined at this time, 97% performed satisfactorily.

3. RESULTS FROM EXPERIMENT L9-1/L3-3

The preliminary analysis presented in this section is based on data processed and available within approximately the first 2 weeks following the conduct of Experiment L9-1/L3-3. In certain instances, this analysis reflects the current lack of confirmatory data or analysis. Analysis of the data will continue in order to further refine these preliminary results and conclusions. The discussion in the following subsections treats the phenomena of each part of the experiment separately.

3.1 Experiment L9-1 Phenomena and Comparison with Prediction

The significant phenomena measured during L9-1 are discussed in this section. In addition, comparison is made between the measured data and preexperiment predictions.[3]

3.1.1 Experiment L9-1 Phenomena

Phenomena observed during L9-1 include:

1. Steam generator liquid inventory depleted rapidly, reaching a level of 0.64 m (2.1 ft) above the top of the tube sheet at reactor scram (65.4 s). It then continued to decrease under the influence of core decay heat, reaching the bottom of the indicating range, which corresponded to 0.25 m (0.82 ft) above the top of the tube sheet by 190 s (3 min, 10 s). Decreasing secondary system pressure due to steam valve leakage and increasing fluid temperature throughout L9-1 are evidence that the secondary side had dried out (that is, there was no liquid present).

2. After the effects of reactor scram on the transient had ended and throughout the rest of L9-1, the combined core decay heat and primary coolant pump power continued to exceed primary system coolant heat losses. This heat imbalance caused the primary system coolant temperature to continually increase which resulted in a coolant swell and insurge into the pressurizer (Figure 2). Pressurizer spray was effective in maintaining primary system pressure until it was manually shut off at 1246 s (20 min, 46 s), allowing the pressure to increase to the PORV setpoint at 1468 s (24 min, 28 s) (Figure 1).

3. The PORV was able to control the primary system pressure throughout the remainder of L9-1 by cycling open approximately 5% of the time and emitting 175 kg of mass by 3270 s (54 min, 30 s) (see Figure 1).

3.1.2 Comparison with Prediction

The preexperiment prediction correctly predicted that the PORV would control primary system pressure with a dry steam generator secondary. The predicted experimental time scale was sensitive to variations or differences between actual and predicted time-dependent boundary conditions, as noted in Reference 3. Significant differences between predicted and actual system behavior caused by these variations are listed as follows:

1. The steam generator liquid level at reactor scram was predicted to be 0.9 m versus approximately 0.6 m in the experiment. One reason for this difference was that the pressurizer spray was predicted to be much less effective in depressurizing the primary system than it actually was, as can be seen in Figure 3. The effect of the spray delayed the increase of primary system pressure to the scram setpoint, delaying the scram to 65.4 s, or

8.9 s later than predicted and, together with the low initial steam generator liquid level, resulted in more mass depletion from the steam generator secondary.

2. Figure 4 shows the overall primary system hot leg temperature overlayed with the prediction. The actual heatup rate was much slower than predicted. A comparison of heat source strength and fluid heatup rate in the experiment indicates that there was approximately a 1-MW heat sink during L9-1, nearly twice as much as was modeled. The location of this heat sink is currently not known.

3. The pressurizer spray controlled primary system pressure after reactor scram and was manually shut off after operating 1037 s (17 min, 17 s) so that the PORV would be challenged. In the prediction, primary system coolant swell eventually filled the pressurizer and challenged the PORV without requiring that the spray be shut off. This difference was apparently a direct result of the larger than predicted heat losses mentioned above.

4. As mentioned in Section 3.1.1, there was an apparent small steam leak in the steam generator secondary. Similar leakage has been detected in previous experiments and results from incomplete seating of the main steam control valve. A RELAP5 study wherein the measured secondary system pressure was input to the calculation showed no effect on the predicted primary system response in L9-1.

5. During the PORV cycling, approximately 175 kg of mass left the primary system, as opposed to 300 kg predicted to leave (Figure 5). This difference was due to less primary coolant swell than predicted which, in turn, was affected by the larger than expected primary system heat losses.

3.2 Experiment L3-3 Phenomena and Comparison with Prediction

After the PORV had cycled for approximately 1800 s (30 min), L9-1 was terminated and L3-3 was initiated by latching open the PORV and turning off the primary coolant pumps. This section discusses the observed phenomena and compares these with the prediction.

3.2.1 Experiment L3-3 Phenomena

Phenomena observed during L3-3 include:

1. When the PORV was latched open, the primary system pressure quickly decreased to saturation pressure of 12.3 MPa (1784 psia), as seen in Figure 1. This pressure was below the ECCS injection setpoint of 13.2 MPa (1915 psia) and demonstrated the ability of recovering from a LOFA with a dry steam generator secondary. PORV response to a wide range of upstream conditions (10.5 to 16 MPa, 1523 to 2321 psia, pressure and 50 to 700 kg/m^3, 3.1 to 44 lbm/ft^3, fluid density) was measured (Figures 1 and 6).

2. Primary system mass depletion continued until 4850 s (1 h, 21 min), when the PORV was shut, terminating the first LOFA recovery mode simulation. The second recovery mode simulation was initiated by refilling the steam generator secondary. The combination of these two operator actions caused a redistribution of mass as the pressurizer quickly drained (Figure 2), and the resulting primary coolant temperature decrease caused a coolant level shrink. The mixture level shrink in the reactor vessel was indicated by thermocouples and liquid level conductivity probes which are sensitive to coolant quality. The data for these instruments located in the reactor vessel just above the hot leg nozzles indicate that the level decreased approximately 0.2 m.

3. Primary system pressure decreased rapidly in response to the reestablishment of the steam generator heat sink. Natural circulation started up at approximately 5205 s (1 h, 27 min). Operator-controlled steam generator feed and bleed operations were initiated at 6712 s (1 h, 52 min), and the depressurization rate increased (Figure 1). Thus, reestablishment of the steam generator heat sink was also shown to be an effective means of recovery from a LOFA.

3.2.2 Comparison with Prediction

Again, the system response was, in general, correctly predicted, including depressurization to the ECCS injection setpoint during the first recovery mode simulation and primary system depressurization and natural circulation caused by steam generator refill and feed and bleed during the second recovery mode simulation.

Significant differences between the experiment data and prediction caused by differences in the time-dependent boundary conditions include:

1. There was much less primary system repressurization subsequent to latching open the PORV than predicted (Figure 7). This difference can be explained, at least in part, by the greater than modeled heat sink discussed in Section 3.1.2. Prior to the experiment, this predicted repressurization was believed to be related to the steam generation rate in the primary system, as calculated by the code, and may not be physical. The modeling dependency of this repressurization is being investigated.

2. The total primary system mass depletion was approximately 1475 kg (3245 lbm), as opposed to 1900 kg (4180 lbm) which was predicted (Figure 5). This is believed to be due to the difference between actual and predicted upstream fluid conditions which were, in turn, influenced by primary coolant swell and primary system heat losses.

4. CONCLUSIONS

The conduct of LOFT Experiment L9-1/L3-3 and the data cquired concerning integral system response to the experiment are considered to have met the objectives as defined by the Experiment Operating Specification[4] and discussed in Section 3. Conclusions based on the preliminary analysis and experiment assessment are as follows:

1. For L9-1:

   a. The PORV effectively controlled primary system pressure with the steam generator dry.

   b. Pressurizer spray was more effective than predicted in mitigating primary system pressurization as steam generator heat transfer degraded. This contributed to a lower than predicted steam generator liquid level at reactor scram.

   c. The primary system heat losses were higher than predicted which resulted in a slower heatup rate and a lower total cycle time for the PORV than predicted.

   d. There was steam leakage from the steam generator which caused the secondary pressure to decrease during the transient. This did not significantly affect primary system response.

   e. There was less mass depletion during L9-1 than predicted due to more primary system heat losses. It is concluded that the additional heat losses encountered in the experiment are due to a mechanism not previously identified. Efforts are continuing to identify the heat loss mechanism.

2. For L3-3:

   a. Each LOFA recovery mode simulation was able to adequately bring the plant to a safe condition.

   b. Significantly less mass left the primary system than was predicted. The difference is directly attributed to the difference in fluid conditions upstream of the PORV.

   c. The reactor vessel mixture level decreased approximately 0.2 m as a result of primary coolant shrink due to steam generator refill.

5. DATA PRESENTATION

This section presents selected, preliminary data from Experiment L9-1/L3-3. Experimental data are overlayed with data from the pretest prediction made using the RELAP5 computer code. A listing of the data plots is presented in Table 3. Table 4 gives the nomenclature system used in instrumentation identification. A complete list of the LOFT instrumentation and data acquisition requirements for the experiment is given in Reference 4.

The maximum (2σ) uncertainties in the report data are:

1. Temperature - ± 3 K ($\pm 6^{\circ}$F)
2. Pressure - ± 0.21 MPa (± 30 psi)
3. Density - ± 0.043 Mg/m^3 (± 3 lb/ft^3)
4. Mass inventory - ± 250 kg (± 550 lbm).

TABLE 3. LIST OF DATA PLOTS

Figure	Title	Measurement Identification	Page
1.	Comparison of pressure in primary and secondary systems	PE-PC-5 PE-SGS-1	20
2.	Pressurizer liquid level	LEPDE-P139-006	20
3.	Comparison of primary system pressure with prediction	PE-PC-5	21
4.	Comparison of hot leg fluid temperature with prediction	TE-PC-2B	21
5.	Primary system mass inventory compared with prediction		22
6.	Fluid density upstream of PORV	DE-PC-S03	22
7.	Primary system pressure compared with prediction	PE-PC-5	23

TABLE 4. NOMENCLATURE FOR LOFT INSTRUMENTATION

Designations for the Different Types of Transducers[a]

TE	-	Temperature element	FE	-	Coolant flow transducer
PE	-	Pressure transducer	DE	-	Densitometer
PdE	-	Differential pressure transducer	DiE	-	Displacement transducer
			ME	-	Momentum flux transducer
LE	-	Coolant level transducer	FT	-	Flow rate transducer

Designations for the Different Systems, Except the Nuclear Core

PE	-	Primary coolant intact loop	LP	-	Lower plenum
BL	-	Broken loop	ST	-	Downcomer stalk
RV	-	Reactor vessel	P120	-	ECCS
SV	-	Suppression tank	P128	-	Primary coolant addition and control
UP	-	Upper plenum			

Designations for Nuclear Core Instrumentation

Transducer location (inches from bottom of fuel rod)

Fuel assembly row

Fuel assembly column

Fuel assembly number

Transducer type

TE-3B11-28

a. Includes only instruments discussed in this report.

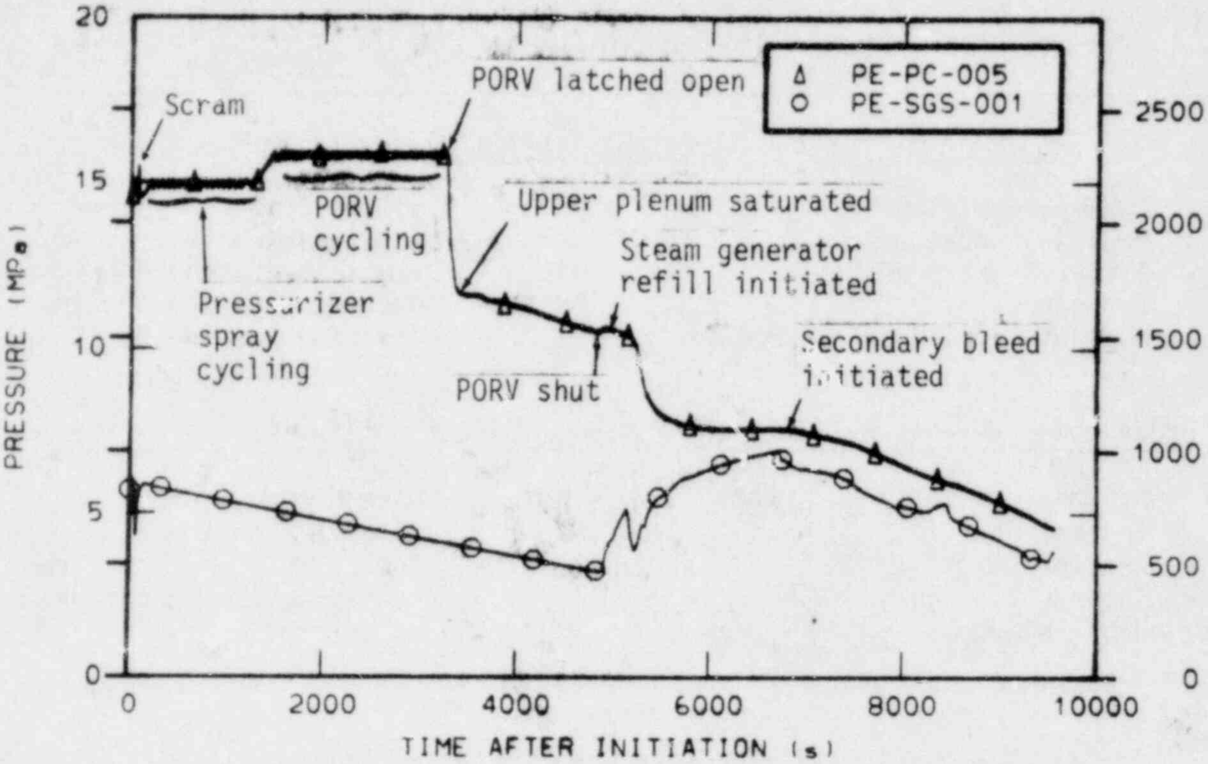

Figure 1. Comparison of pressure in primary and secondary systems.

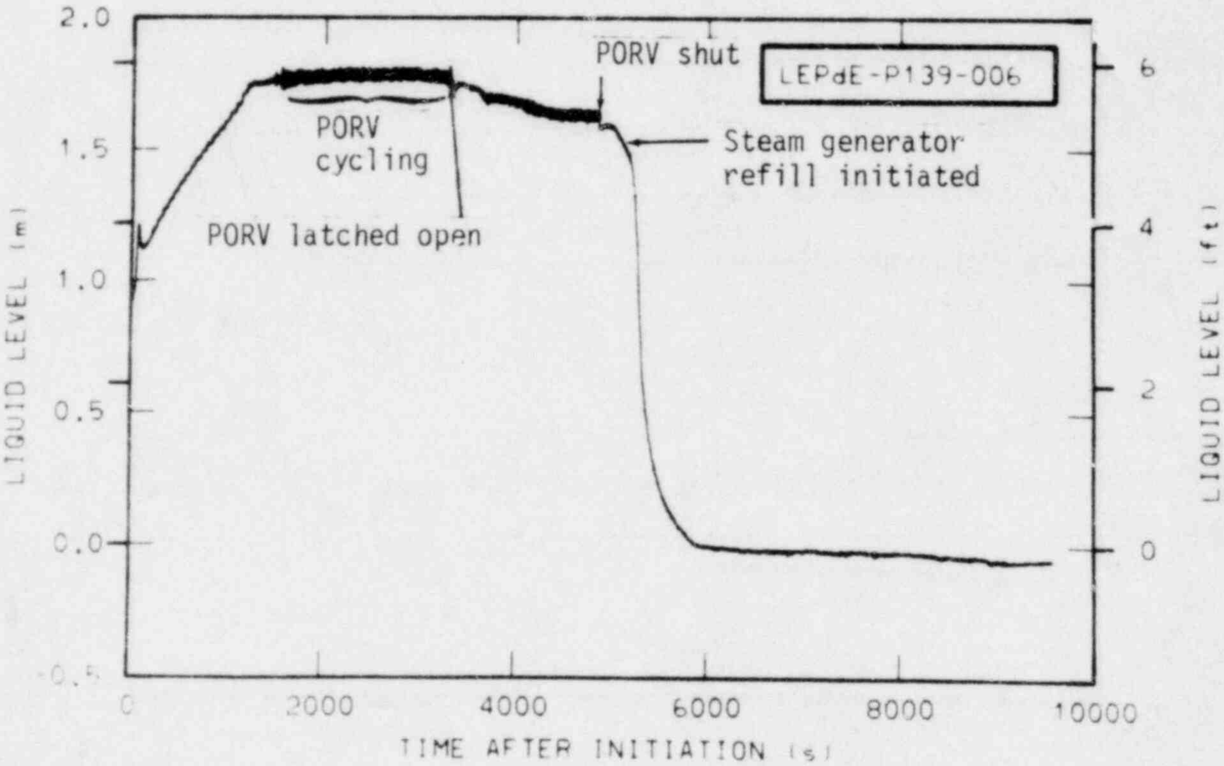

Figure 2. Pressurizer liquid level.

IMAGE EVALUATION TEST TARGET (MT-3)

6″

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

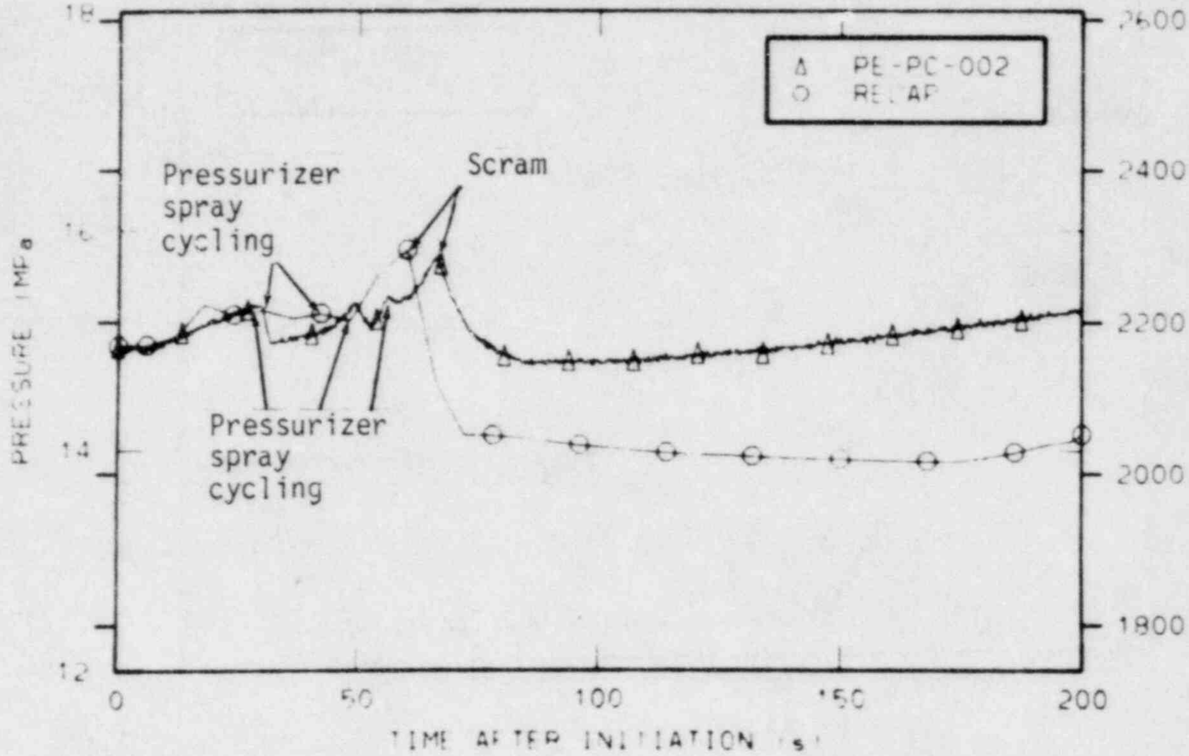

Figure 3. Comparison of primary system pressure with prediction.

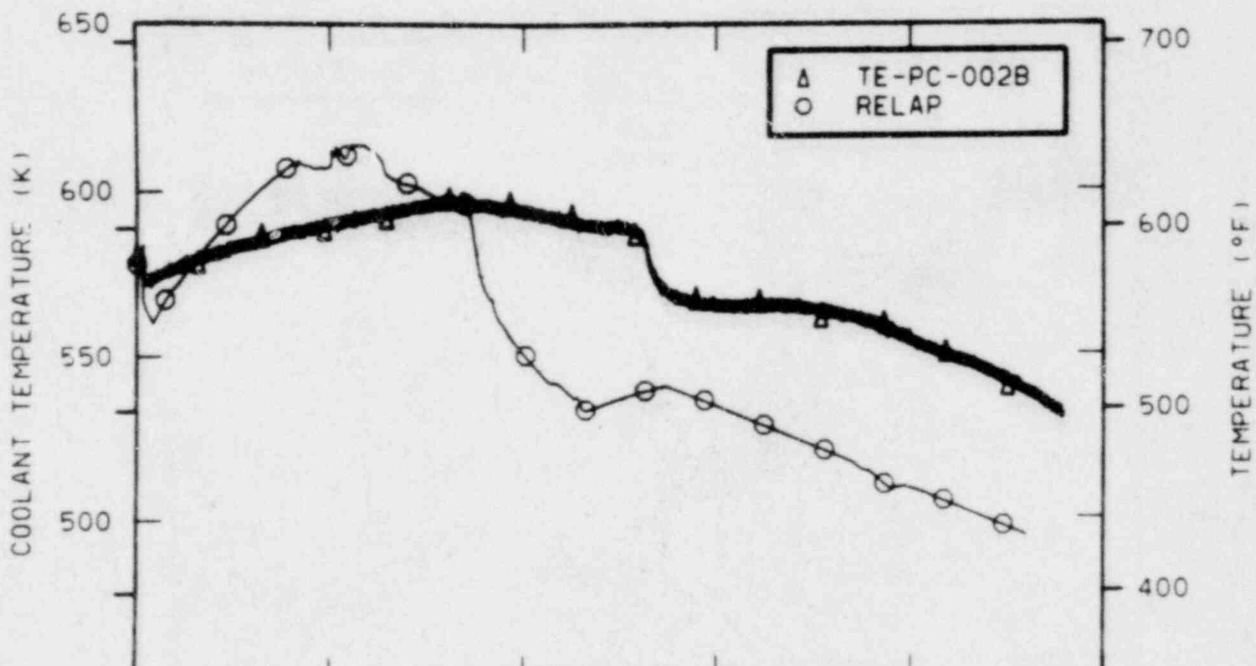

Figure 4. Comparison of hot leg fluid temperature with prediction.

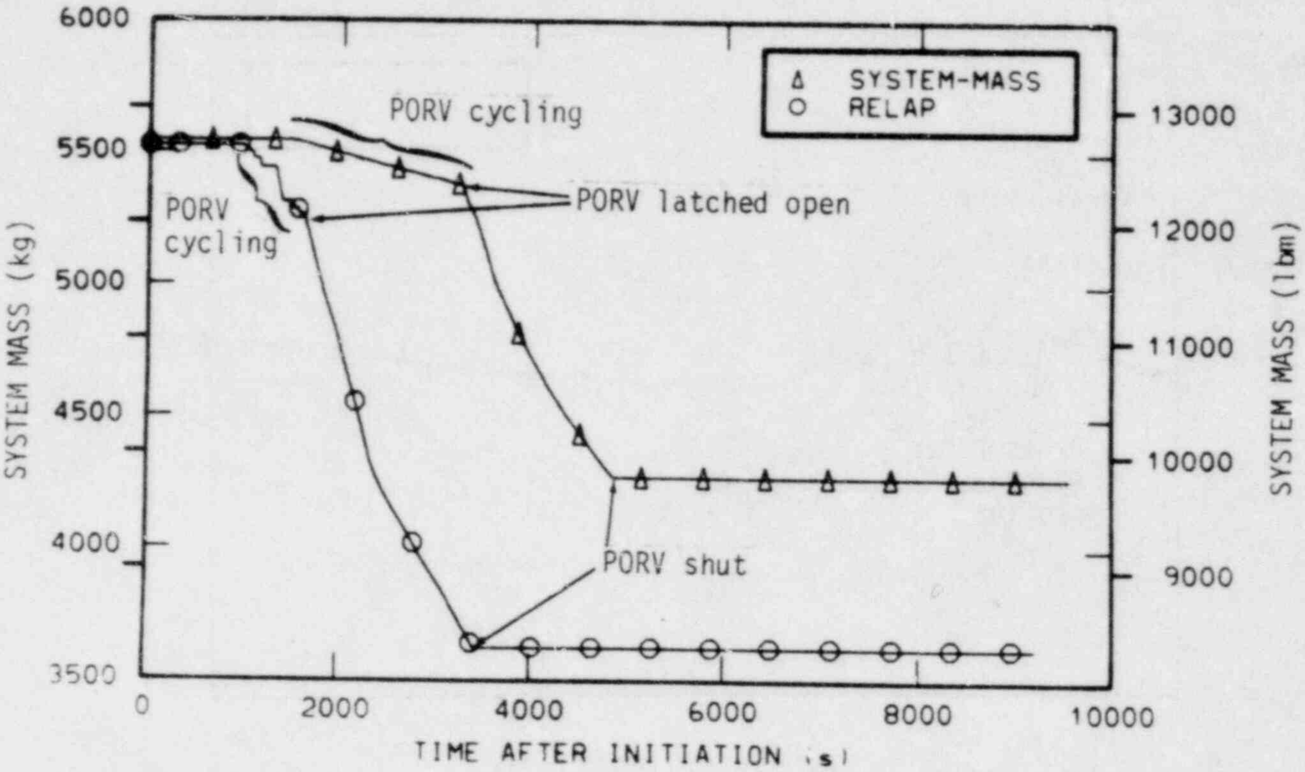

Figure 5. Primary system mass inventory compared with prediction.

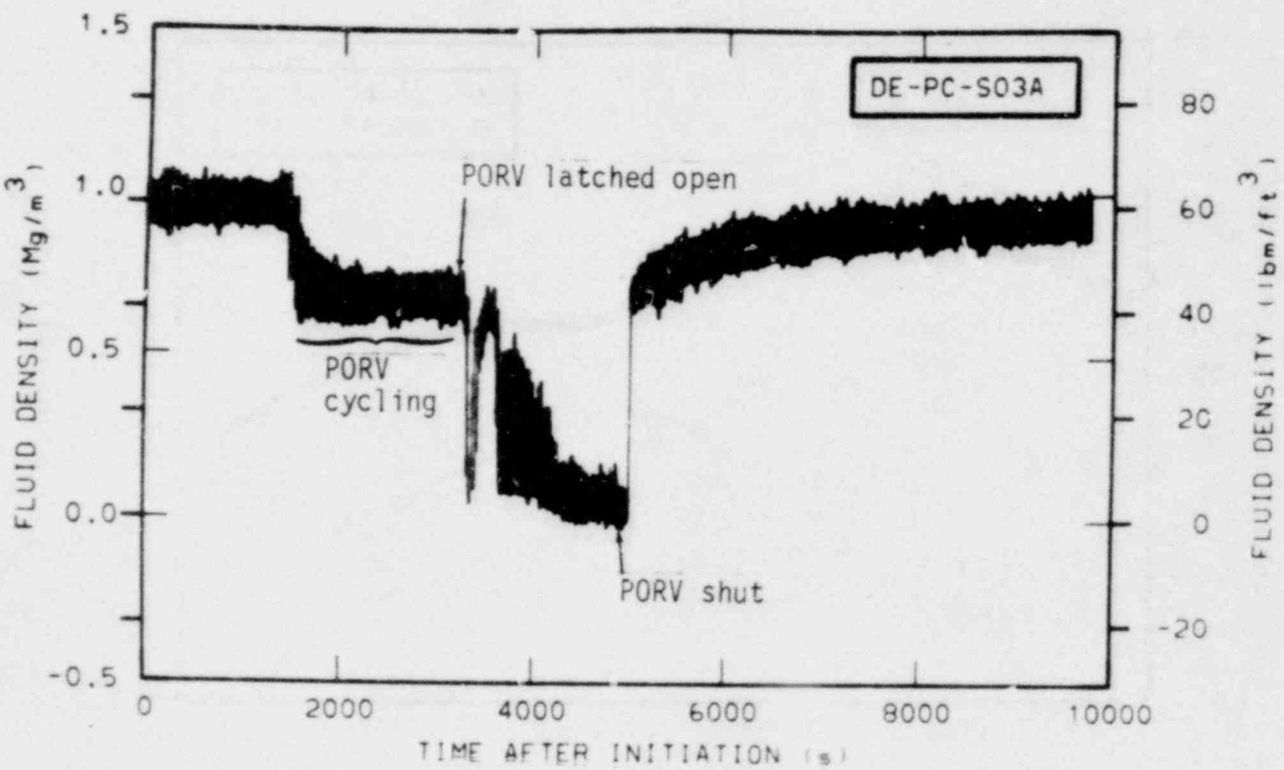

Figure 6. Fluid density upstream of PORV.

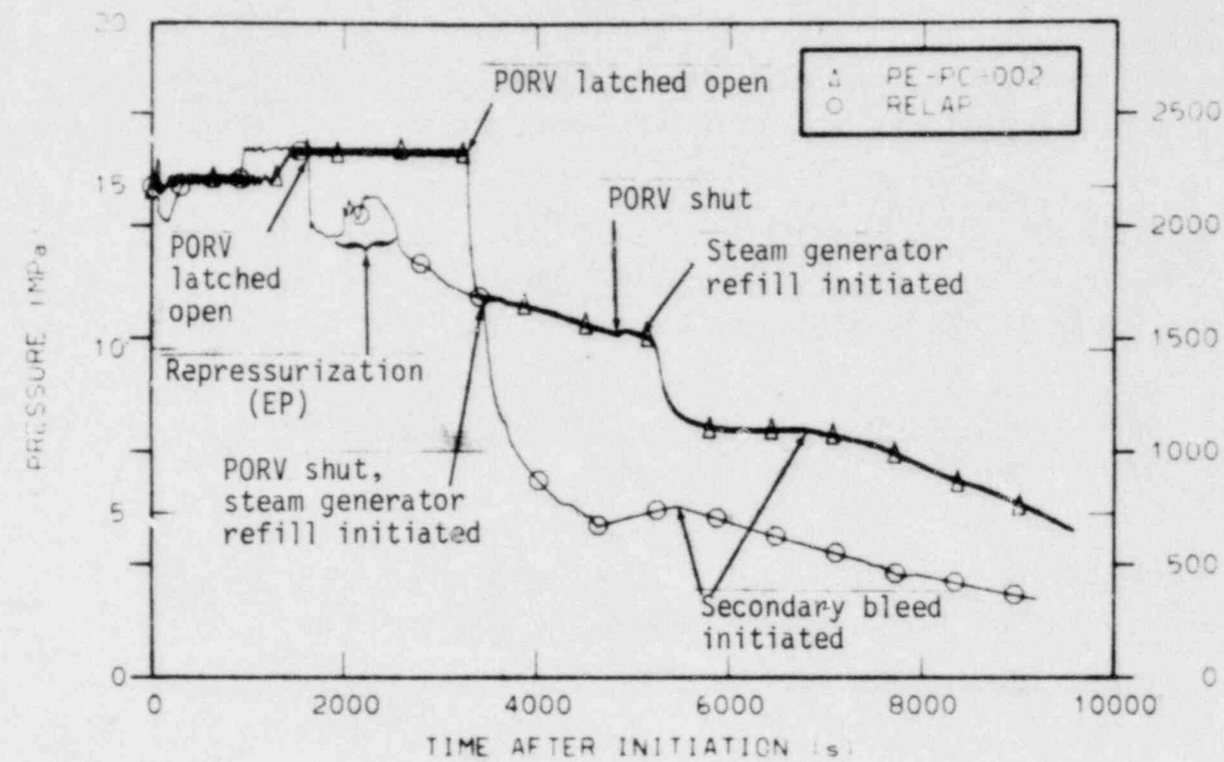

Figure 7. Primary system pressure compared with prediction.

6. REFERENCES

1. D. L. Reeder, LOFT System and Test Description (5.5 ft Nuclear Core 1 LOCEs), NUREG/CR-0247, TREE-1208, July 1978.

2. V. H. Ransom et al., RELAP5/MOD1 Code Manual, NUREG/CR-1826, EGG-2070, November 1980.

3. G. A. Taylor, R. J. Beelman, S. R. Behling, Best Estimate Prediction for LOFT Nuclear Experiment L9-1/L3-3, EGG-LOFT-5413, April 1981.

4. R. J. Beelman, LOFT Experiment Operating Specification, Anticipated Transients with Multiple Failures, Test Series L9, Nuclear Test L9-1/L3-3/L8-1A, EGG-LOFT-5334, April 1981.

5. G. E. McCreery, Quick-Look Report on LOFT Nuclear Experiment L3-6/L8-1, EGG-LOFT-5318, December 1980.

6. P. D. Bayless and J. M. Carpenter, Experiment Data Report for LOFT Nuclear Small Break Experiment L3-6 and Severe Core Transient Experiment L8-1, NUREG/CR-1868, January 1981.

APPENDIX A

LOFT EXPERIMENT L9-1/L3-3 OBJECTIVES

APPENDIX A

LOFT EXPERIMENT L9-1/L3-3 OBJECTIVES

The detailed experiment objectives required to satisfy the programmatic objectives for each part of Experiment L9-1/L3-3 are as follows:

1. For L9-1:

   a. To evaluate uncertainties in predicted primary and secondary thermal-hydraulic response associated with steam generator dryout during delayed scram.

   b. To evaluate the adequacy of the power-operated relief valve (PORV) to provide overpressure protection in a loss-of-feedwater accident (LOFA).

2. For L3-3:

   a. To investigate uncertainties in system response during a PORV imposed small break with loss of secondary heat sink.

   b. To assess the adequacy of modeling assumptions which are used in small break performance predictions such as those identified in NUREG-0623.[A-1]

   c. To assess the effectiveness of steam generator refill on LOFAs following reestablishment of auxiliary feedwater availability.

d. To assess the relative magnitude of the change in reactor vessel mixture level as a result of primary coolant system shrink during steam generator refill.

e. To contribute to the NRC relief and safety valve testing program by providing experimental data on PORV performance characteristics over a range of PORV inlet fluid conditions.

REFERENCE

A-1. B. W. Sheron, Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors, NUREG-0623, November 1979.

APPENDIX B

LOFT SYSTEM GEOMETRY

APPENDIX B

LOFT SYSTEM GEOMETRY

The Loss-of-Fluid Test (LOFT) system geometry is shown in Figure B-1. An experiment power-operated relief valve (PORV) was installed in parallel with the plant PORV and vented to the blowdown suppression tank as shown in Figure B-2. Figure B-3 shows the LOFT steam generator geometry and instrument locations. Figure B-4 shows the LOFT pressurizer with operating levels, volumes, and instrumentation.

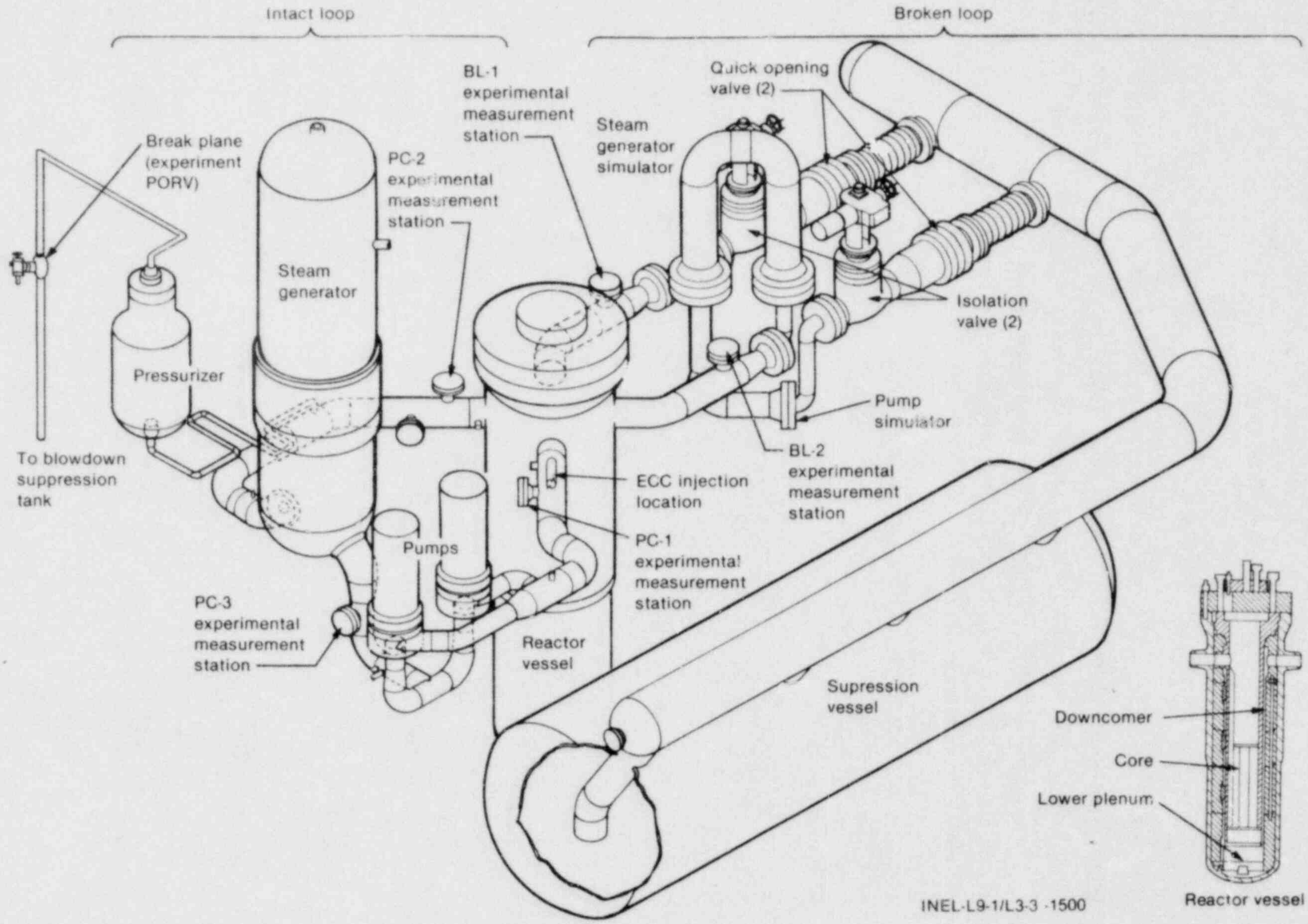

Figure B-1. Axonometric projection of LOFT system.

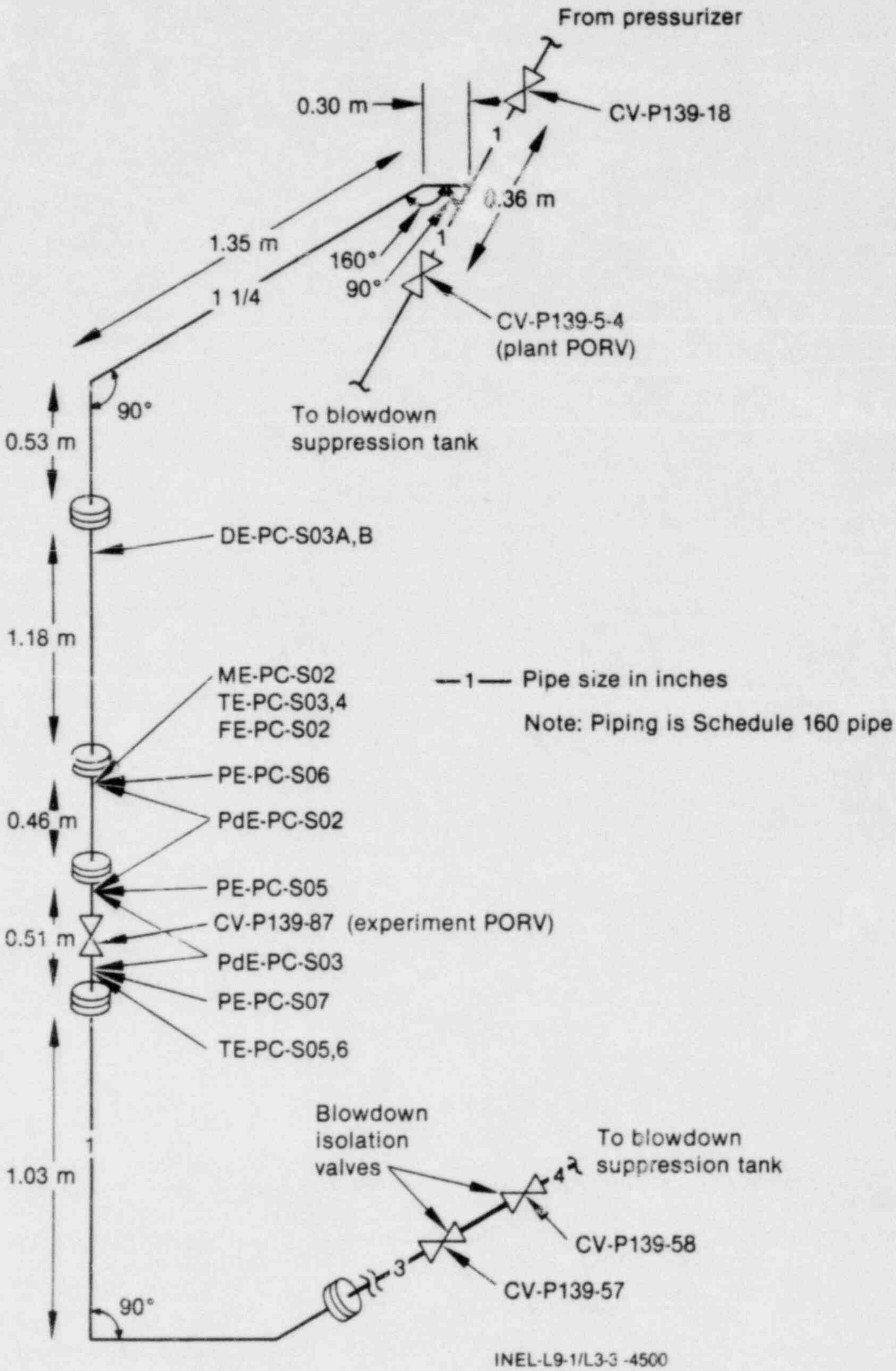

Figure B-2. LOFT PORV spool piece configuration.

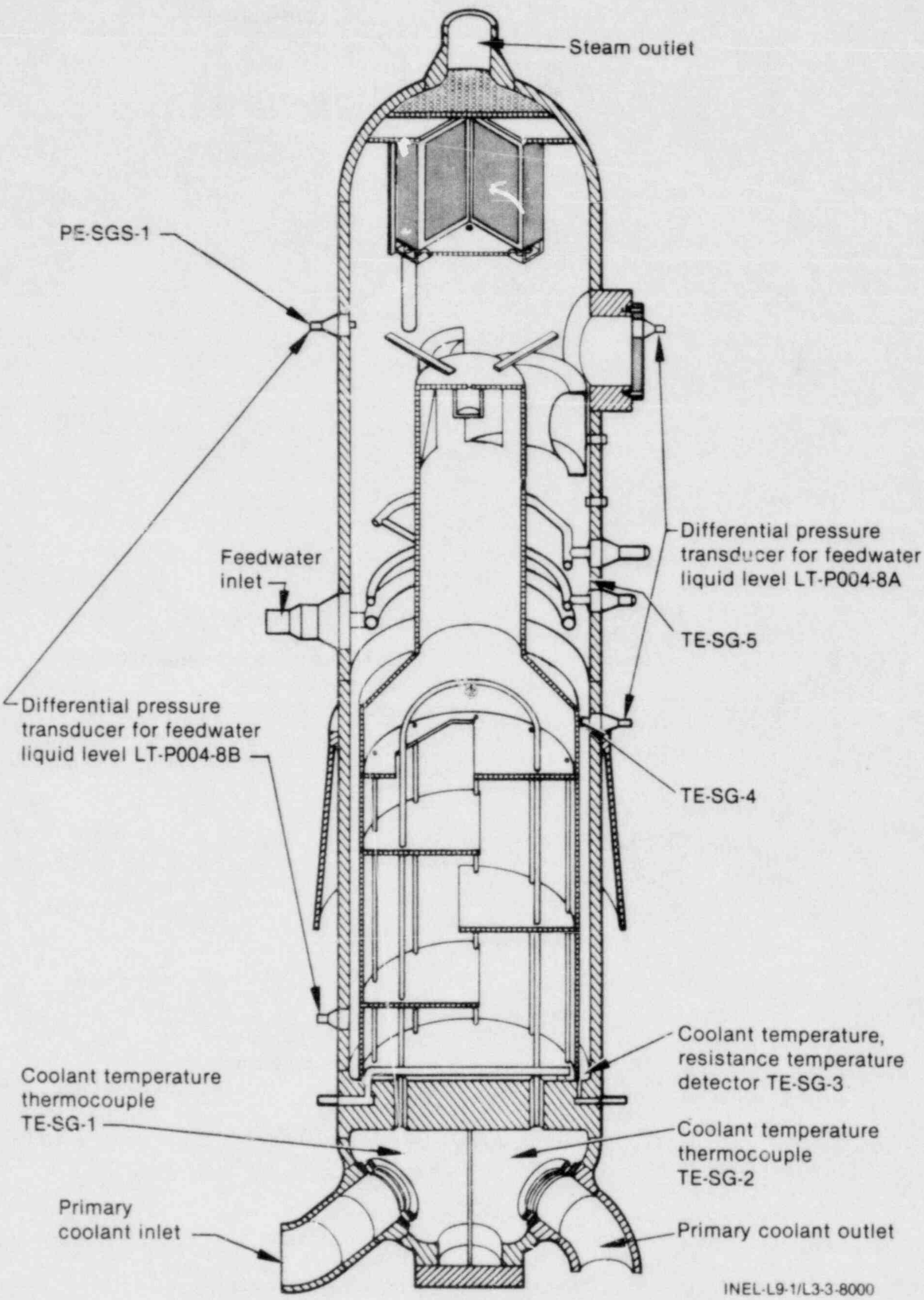

Figure B-3. LOFT steam generator and instrumentation.

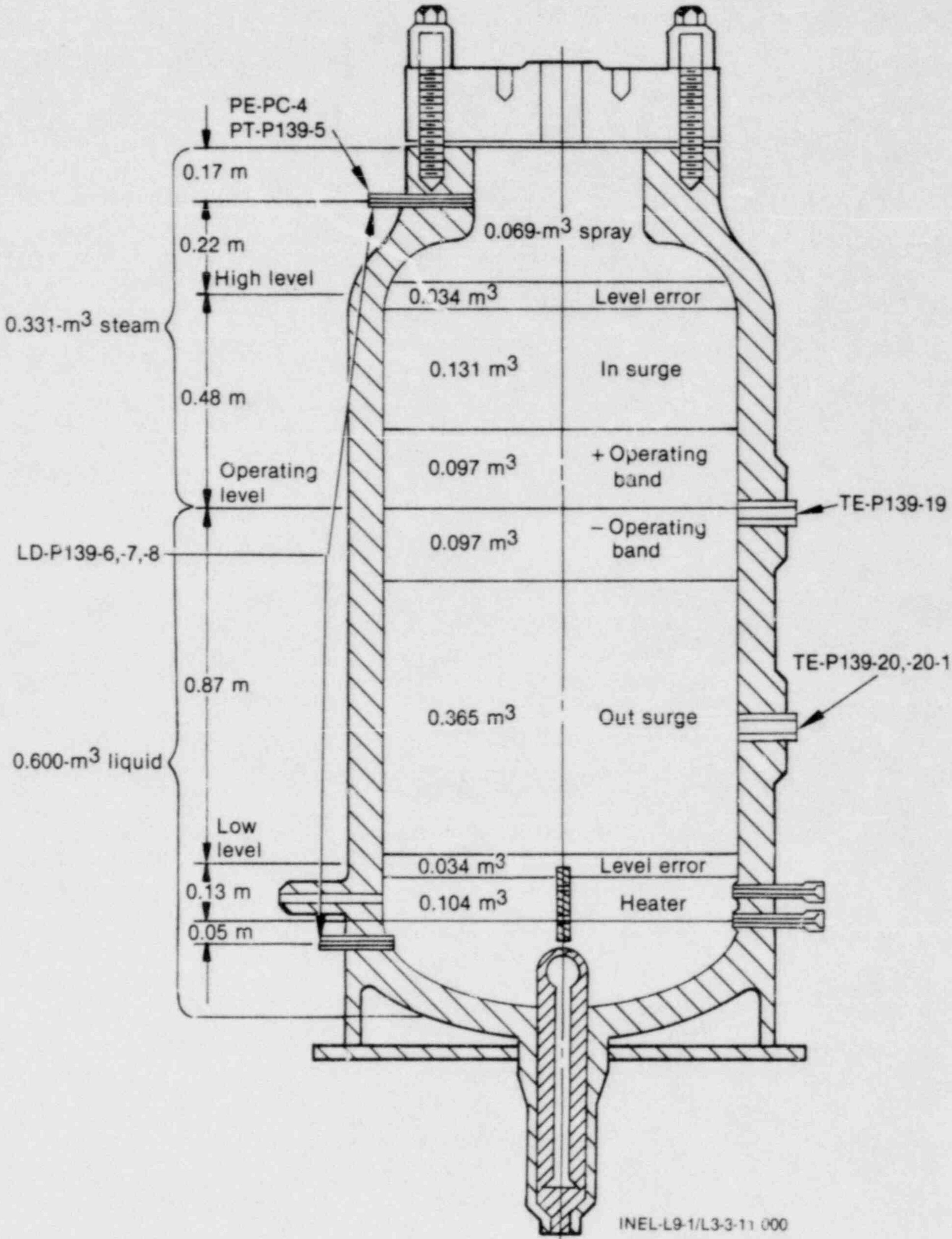

Figure B-4. LOFT pressurizer and instrumentation.