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QUICK-LOOK REPORT ON LOFT NUCLEAR EXPERIMENT L3-2

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MASTER

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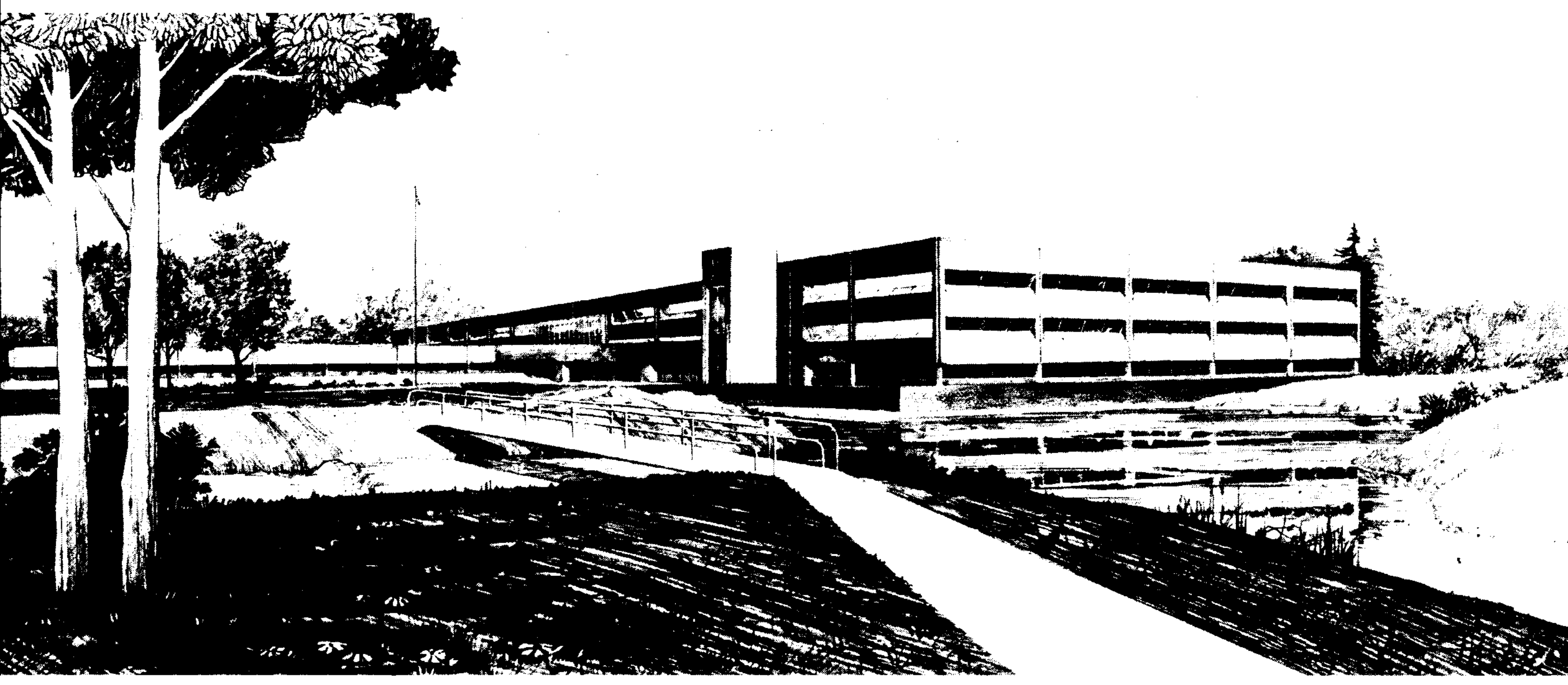
Idaho Operations Office • Idaho National Engineering Laboratory



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QUICK LOOK REPORT ON LOFT NUCLEAR EXPERIMENT L3-2

Approved:

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LOFT

The information contained in this summary report is preliminary and incomplete. Selected pertinent data are presented in order to draw preliminary conclusions and to expedite the reporting of research results.

ABSTRACT

Loss-of-Coolant Experiment (LOCE) L3-2, the third experiment in the Loss-of-Fluid Test (LOFT) Small Break Series L3 scheduled for performance in the LOFT facility, was successfully completed on February 7, 1980. LOCE L3-2 simulated a single-ended offset shear break of a small (1-in.-diameter) pipe connected to the cold leg of a four-loop large pressurized water reactor. After experiment initiation, the primary system depressurization rate stabilized, decreasing system pressure to a point where flow from the high-pressure injection system approximated the break flow. Operator intervention, as planned, was used to reduce system pressure so that the final stage of plant cooldown could be accomplished with the purification system. The reactor system was brought to a cold shutdown condition 6.5 hours after experiment initiation.

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DEFINITIONS

Flow reversal - the inception of negative flow in system piping, in a 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 loop circulation - loop 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 loop circulation - loop 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.

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.

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

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.

SUMMARY

The preliminary evaluation has been completed of the results from the nuclear Loss-of-Coolant Experiment (LOCE) L3-2, which was successfully completed on February 7, 1980, in the Loss-of-Fluid Test (LOFT) facility. LOCE L3-2 is the third experiment in the LOFT Small Break Series L3 and simulated a single-ended offset shear break of a small (1-in.-diameter) pipe in a large pressurized water reactor (PWR).

Prior to the break, the nuclear core was operating at a steady state maximum linear heat generation rate of 52.2 ± 3.7 kW/m. Other significant initial conditions for LOCE L3-2 were: system pressure, 14.85 ± 0.04 MPa; core outlet temperature, 575.8 ± 3 K; and intact loop flow rate, 481.5 ± 6.3 kg/s. At 13 s after experiment initiation, the reactor scrammed on a low system pressure signal. The intact loop primary coolant pumps were tripped after the reactor scrammed which initiated pump coast-down. Pump coast-down was followed by the inception of natural loop circulation. As planned, operator intervention was used later in the transient to decrease the primary system pressure. Once plant conditions allowed operator control of system pressure, the system was cooled down by the purification system and the experiment terminated.

Core thermal response was benign, due primarily to the core remaining covered during the transient. Core thermocouple response either followed saturation temperature or was below saturation temperature throughout the transient. No fuel rod damage occurred.

The steam generator was an effective heat sink for the system throughout the experiment as indicated by the trends in system pressure and the effectiveness of the operator-induced steam bleeding. Natural loop circulation was measurable and effective early and late in the experiment. Natural loop circulation could not be measured between these times. There is evidence that another cooling mode may have occurred when natural loop circulation could not be measured.

Computer predictions of LOCE L3-2 transient response were made by EG&G Idaho, Inc., using RELAP5 and RELAP4 and by Los Alamos Scientific Laboratory using TRAC-P1A to calculate system performance. These calculations are compared with experimental data in this report; however, the RELAP4 and TRAC-P1A calculations terminated prior to the time of operator intervention during the experiment. Preliminary analysis of the data from LOCE L3-2 indicates that the dominant phenomena, in the proper time sequence, were predicted except for the large mass loss from the system early in the experiment. The reason for this discrepancy is not known, but is being investigated.

The data supplied by the instrumentation and data acquisition system allowed experiment objectives to be met. Of the 595 instruments recorded, 573 (96%) were estimated to have operated successfully.

LOCE L3-2 provided experimental data on hydraulic behavior during the blowdown and plant recovery phases of a postulated loss-of-coolant accident in a large nuclear PWR. The intensive analysis of LOCE L3-2 data currently underway (a) will result in additional understanding of loss-of-coolant accidents and (b) together with results from other Nuclear Regulatory Commission experimental programs, will contribute to the data base required for development and assessment of analytical models for licensing commercial PWRs.

QUICK-LOOK REPORT ON LOFT NUCLEAR EXPERIMENT L3-2

1. INTRODUCTION

The Loss-of-Fluid Test (LOFT) facility[1] is a 50 MW(t) volumetrically scaled pressurized water reactor (PWR) system designed to study the response of the engineered safety features (ESF) in commercial PWR systems during a postulated loss-of-coolant accident (LOCA). With recognition of the differences in commercial PWR designs and inherent distortions in reduced scale systems, the design objective for the LOFT facility was to produce the significant thermal-hydraulic phenomena that would occur in commercial PWR systems in the same sequence and with approximately the same time frames and magnitudes. The objectives of the LOFT experimental program are

1. To provide data required to evaluate the adequacy and improve the analytical methods currently used to predict the response of large PWRs to postulated accident conditions, the performance of ESFs with particular emphasis on emergency core cooling systems (ECCS), and the quantitative margins of safety inherent in the performance of the ESF.

2. To identify and investigate any unexpected event(s) or threshold(s) in the response of either the plant or the ESF and develop analytical techniques that adequately describe and account for such unexpected behavior(s).

3. To evaluate and develop methods to prepare for, operate during, and recover systems and plant from reactor accident conditions.

4. To identify and investigate methods by which the safety of nuclear reactors can be enhanced, with emphasis on the interaction of the operator with the plant.

Loss-of-Coolant Experiment (LOCE) L3-2, the third experiment in the LOFT Small Break Series L3[a] scheduled for performance in the LOFT facility, was successfully completed on February 7, 1980. LOCE L3-2 simulated a single-ended offset shear of a small (1-in.-diameter) pipe connected to the cold leg of a four-loop large PWR. The LOFT system geometry is shown in Figure 1, and a representation of the core configuration illustrating the instrumentation and position designations is shown in Figures 2 and 3, respectively. Additional details of the core and fuel modules are given in Reference 1. The small break orifice geometry unique to the L3 series LOCEs is shown in Figure 4.

Experiment Series L3 was designed to be consistent with the LOFT experimental program objectives by providing experimental data to assist in answering questions delineated in the Experiment Operating Specification (EOS).[6] The questions, as stated in the EOS, are:

"(1) How does the primary coolant system respond during a small break when,

(a) break flow is greater than HPIS? (L3-1),

(b) break flow is the same order of magnitude as the HPIS as system pressure stabilizes later in the transient? (L3-2),

(c) the steam generator is decoupled (system repressurization)? (L3-3),

(d) the break is in the pressurizer system? (L3-4).

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a. The first experiment in the L3 series was LOCE L3-0, a nonnuclear experiment which used the pressurizer power-operated relief valve as the break orifice. LOCE L3-0 was conducted on May 31, 1979, and was reported by References 2 and 3. The second experiment in the L3 series was LOCE L3-1, a nuclear experiment which simulated a single-ended offset shear in a small (4-in.-diameter) pipe connected to the cold leg of a four-loop large PWR. LOCE L3-1 was conducted on November 20, 1979, and was reported by References 4 and 5.

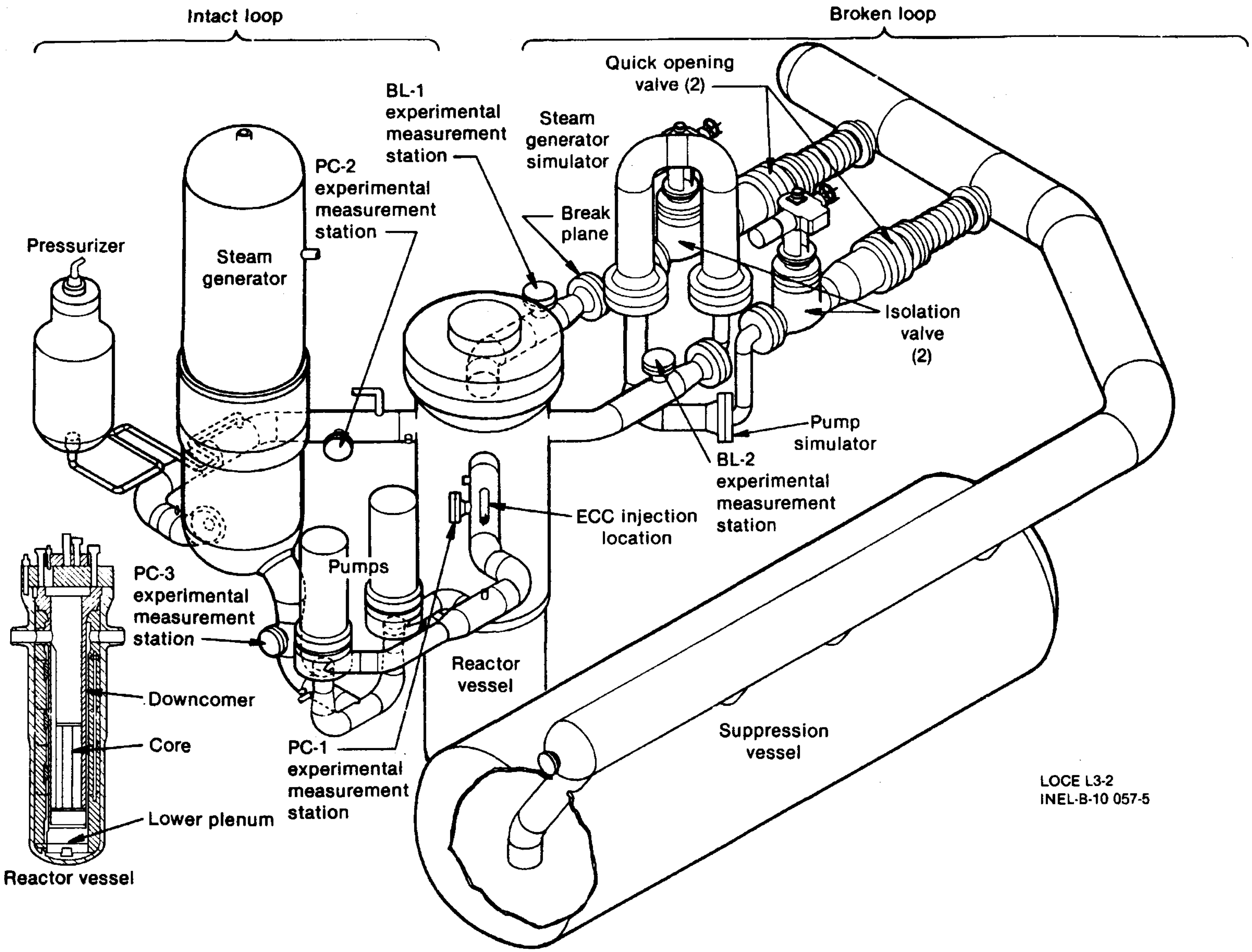

Figure 1. Axonometric projection of LOFT system.

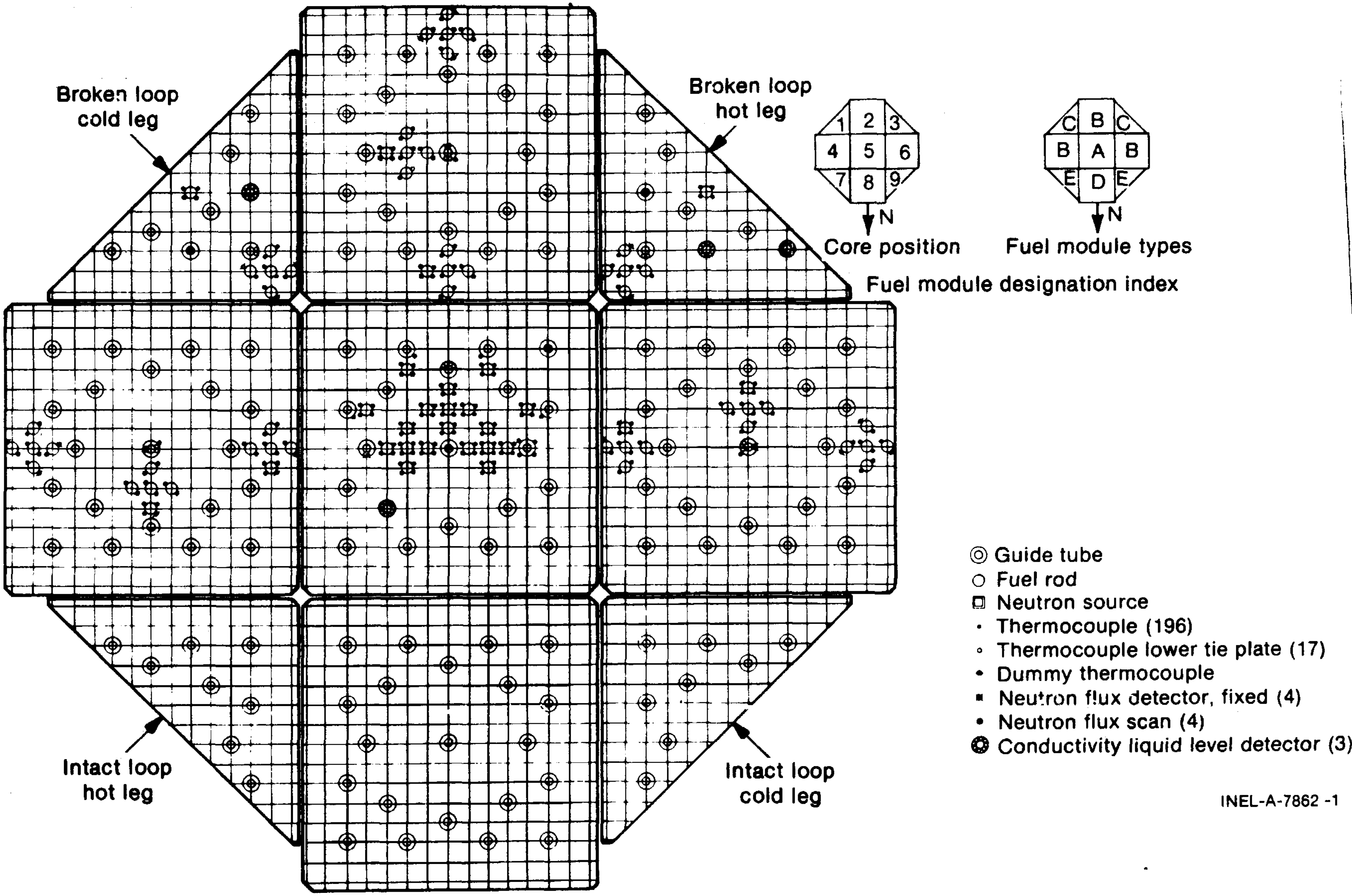

Figure 2. LOFT core configuration and instrumentation.

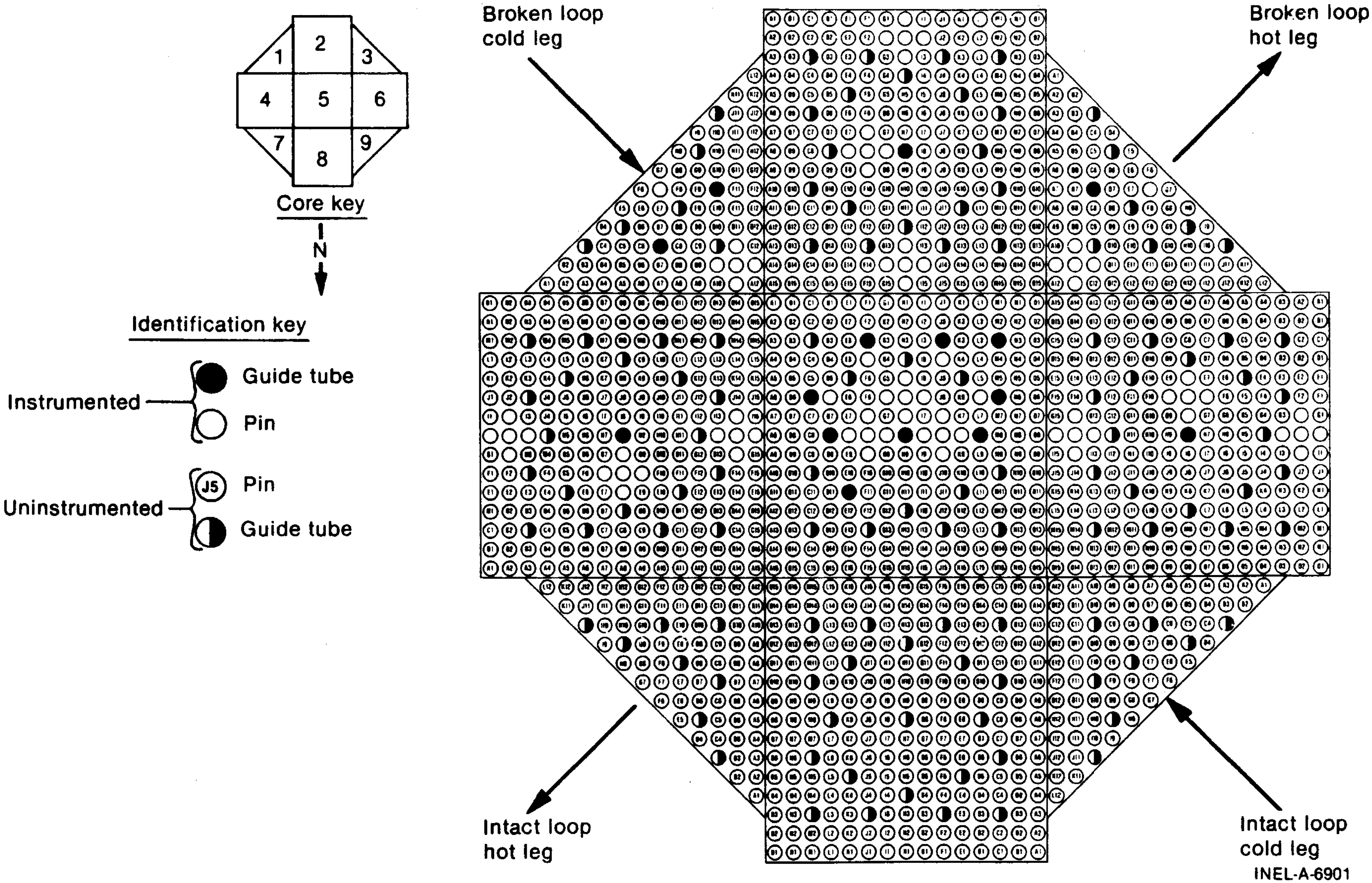

Figure 3. LOFT core map showing position designations.

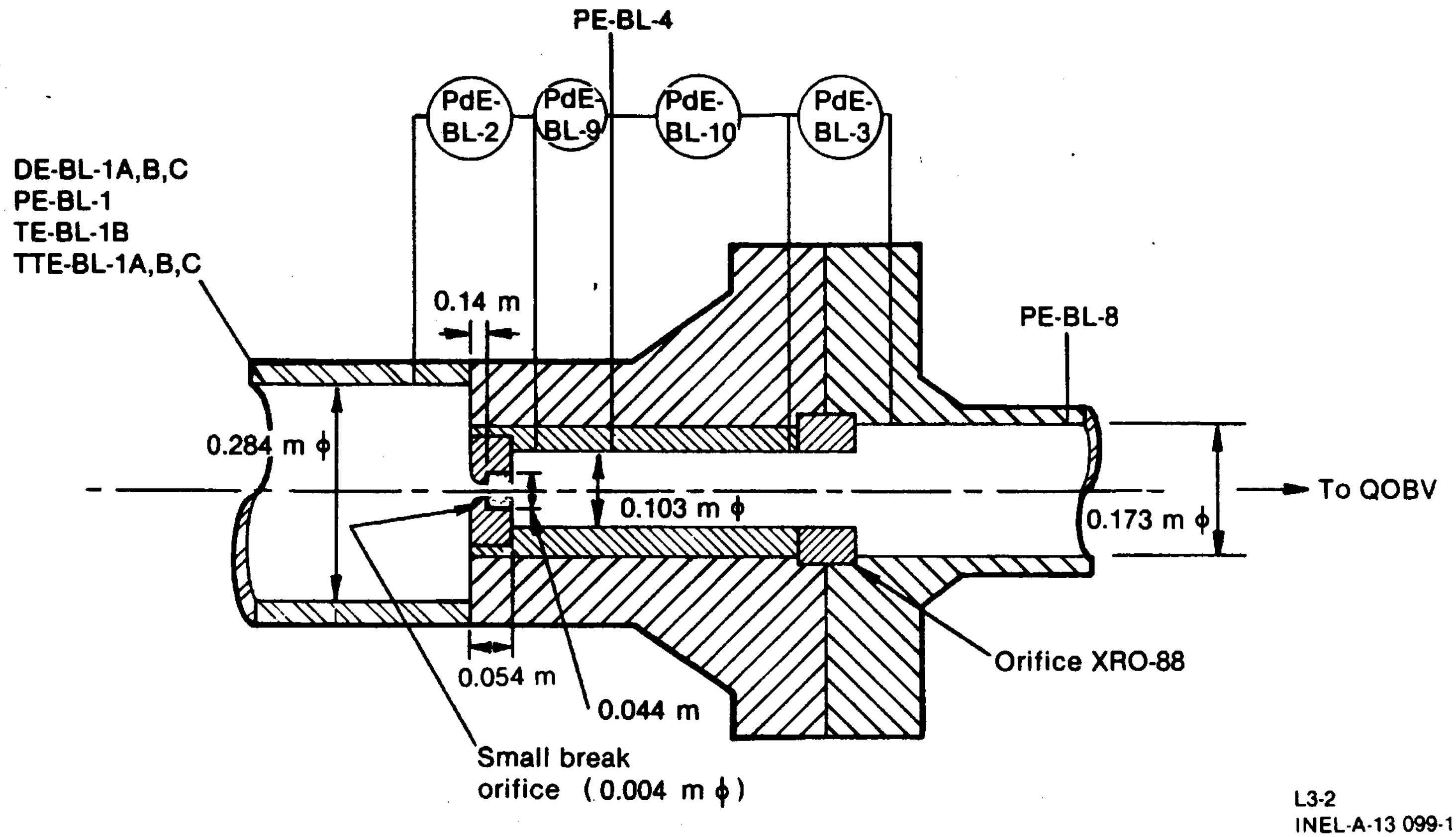

Figure 4. Small break orifice configuration.

(2) How many of the major systems, such as, LPIS, accumulator, HPIS, steam generator, etc. are needed to prevent core damage during a small break, and are these same systems required for other break sizes or locations?

(3) How effectively do the ECC systems perform during the consequent pressure transients for these types of depressurizations?

(4) Does primary coolant pump operation during a small LOCA affect system void fraction, and what are the effects of void fraction on the small LOCA transient (L3-5 and L3-6)?

(5) What kind of recovery procedures should be used in the event of a small break LOCA?

(6) Are there key times in the transient where operator action is required to protect the core?

(7) Are there operator/equipment actions that must not occur?

(8) Given a small break occurrence of unknown size or location, are there any operator actions that are dependent on the break unknowns which would aid plant recovery in one case and impede plant recovery for another case?

(9) Are typical commercial reactor process instruments capable of providing accurate information on plant conditions during a transient? Specifically;

 (a) Which instruments furnish relevant data and which do not?

 (b) Can the operator use information from typical process instruments to estimate the break size and location?

(10) Are there improvements that can be made to typical commercial pressurized water reactor instrumentation to monitor a small LOCA? Are there any additional measurements that should be provided?

(11) Are there improvements that can be made in commercial plant design to improve the safety of the plant?"

Questions 1 and 4 are experiment specific questions, which can be answered from the results of one or two experiments. The remaining nine are general questions, which require more than the results of one or two experiments to answer.

LOCE L3-2 provided data to answer experiment specific question (1.b) and will contribute to answering the general questions, but not the remaining experiment specific questions (1.a, 1.c, 1.d, and 4).

This report presents a preliminary examination of plant performance (Section 2), followed by a summary of the results from LOFT LOCE L3-2 (Section 3). Section 4 presents 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 4. The data plots presented include comparisons of LOCE L3-2 data with (a) LOCE L3-2 pretest calculations[7] made by EG&G Idaho, Inc., using the RELAP4[8,a] and RELAP5[9,b] computer codes and (b) LOCE L3-2 pretest calculations made by Los Alamos Scientific Laboratory

a. The experimental RELAP4 code used was RELAP4/MODG, Version 92, (experimental version of RELAP4/MOD7), Idaho National Engineering Laboratory Configuration Control Number H00718B. The new object deck, which includes changes to correct known coding errors and to incorporate the LOFT steam valve control logic into the code, was RLP4G92LFT04, Idaho National Engineering Laboratory Configuration Control Number H01168IB.

b. The version of the code used was RELAP5/MOD"0". The source deck and update input data deck are stored under Idaho National Engineering Laboratory Configuration Control Numbers H005785B and H005985B, respectively.

(LASL)[10] using the TRAC-P1A[11] computer code. The predictions of primary system pressure, break mass flow, and pressurizer liquid level during the blowdown phase of the transient from RELAP4, RELAP5, and TRAC-P1A are compared with the measured data.

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 LOCE L3-2. Data plots showing results of the evaluation are provided in Figures 5 through 26 in Section 5.

2.1 Initial Experimental Conditions

A summary of the specified and measured system conditions immediately prior to LOCE L3-2 blowdown initiation is given in Table 1. The measured average initial temperature of the primary coolant was 566.8 ± 3 K. The range of cladding temperatures was 558.0 ± 3 to 614.8 ± 3 K. The initial mass flow rate in the primary coolant loop was 481.5 ± 6.3 kg/s, and pressurizer pressure was 14.85 ± 0.04 MPa. The initial power level of 49.0 ± 1.0 MW yielded a maximum linear heat generation rate (MLHGR) of 52.2 ± 3.7 kW/m. All of the initial conditions were within specified limits.

2.2 Chronology of Events

Identifiable significant events that occurred during LOCE L3-2 are listed in Table 2, where they are compared with times predicted by the RELAP4 and RELAP5 calculations. At 13 s into the transient, reactor scram was initiated by a low pressure signal in the primary system hot leg (Figure 6). After the reactor scrammed, the intact loop primary coolant pumps were tripped and started to coast down.

TABLE 1. INITIAL CONDITIONS FOR LOCE L3-2

Parameter	EOS Specified Value[6,a]	Measured Value
Primary Coolant System		
Mass flow rate (kg/s)	478.8 ± 8.8	481.5 ± 6.3
Hot leg pressure (MPa)	14.95 ± 0.34	14.85 ± 0.04
Cold leg temperature (K)	556.8 ± 2.2	557.8 ± 3
Hot leg temperature (K)	As required	575.8 ± 0.5
Boron concentration (ppm)	--	747 ± 15
Reactor Vessel		
Power level (MW)	50.0 ± 2	49.0 ± 1.0
Maximum linear heat generation rate (kW/m)	--	52.2 ± 3.7
Control rod position (meters above full-in position)	1.372 ± 0.013	1.372 ± 0.010
Broken Loop		
Hot leg fluid temperature (K)	556.8 ± 13.9	556.9 ± 5.0
Cold leg fluid temperature (K)	556.8 ± 13.9	561.9 ± 5.0
Steam Generator Secondary Side		
Water level (m)[b,c]	0.25 ± 0.05	--
Water temperature (K)	--	543.1 ± 1.4
Pressure (MPa)	--	5.51 ± 0.11
Mass flow rate (kg/s)	--	27.3 ± 0.4
ECCS Accumulator A		
Gas volume (m^3)	--	1.22 ± 0.03

TABLE 1. (continued)

Parameter	EOS Specified Value[6,a]	Measured Value
Liquid Level (m)	1.85 ± 0.05	1.84 ± 0.01
Standpipe position (m)[d]	0.47 ± 0.03	0.48 ± 0.01
Pressure (MPa)	4.22 ± 0.17	4.38 ± 0.06
Temperature (K)	305.4 ± 5.6	307.5 ± 0.7
Boron concentration (ppm)	3000	3396 ± 15
Suppression Tank		
Liquid level (m)	1.27 ± 0.05	1.28 ± 0.06
Gas volume (m^3)	--	55.5 ± 1.9
Downcomer submergence (m)[e]	--	0.42 ± 0.06
Water temperature (K)[f]	--	363.2 ± 2.7
Pressure (gas space) (MPa)[f]	--	0.136 ± 0.008
Pressurizer		
Steam volume (m^3)	--	0.29 ± 0.05
Water volume (m^3)	--	0.67 ± 0.05
Water temperature (K)	As required to establish pressure	614.8 ± 0.3
Pressure (MPa)	14.95 ± 0.34	14.85 ± 0.04
Level (m)	1.13 ± 0.18	1.20 ± 0.02
HPIS		
Initiation pressure (MPa)	13.16 ± 0.19	13.07 ± 0.24
Initial flow (l/s)	0.32 ± 0.13	0.38 ± 0.02

TABLE 1. (continued)

Parameter	EOS Specified Value[6,a]	Measured Value
LPIS[g]		
Initiation pressure (MPa)	1.60 ± 0.19	1.59 ± 0.04

a. If no value is listed, that parameter is not specified by the EOS.

b. The water level is defined as 0.0 at 2.95 m above the top of the tube sheet.

c. Ambiguous initial readings. Absolute value cannot be determined.

d. The standpipe position is defined as 0 at 0.3175 m above the bottom of the accumulator.

e. Based on average submergence of four downcomers.

f. Suppression tank pressure and water temperature ranges specified in the EOS.

g. LPIS - low-pressure injection system.

TABLE 2. CHRONOLOGY OF EVENTS - EXPERIMENTAL DATA VERSUS PRETEST PREDICTIONS

	Time After LOCE Initiation (s)		
Event	LOCE L3-2 Data	RELAP5[a] Prediction	RELAP4[a] Prediction
Reactor scrammed	12.91 ± 0.10	94.0	45.8
Control rods reached bottom	14.98 ± 0.10	Not calculated	47.8
Primary coolant pumps tripped	16.90 ± 0.10	94.0	47.8
HPIS initiated	33.84 ± 0.10	127.0	88.0
Primary coolant pumps coast-down completed	35.0 ± 1.0	Not calculated	~60
First indication in core of natural loop circulation	36.0 ± 2.0	Not calculated	Not calculated
Secondary coolant system auxiliary feed pump started (initial steam generator fill)	114.0 ± 1.0	154.0	112.6
Pressurizer emptied	136.0 ± 7.0	400.0	359.0
Upper plenum fluid reached saturation temperature (end of subcooled blowdown)	180.0 ± 1.0	450.0	440.0
End of subcooled break flow[b]	650 to 800	--	--
Secondary coolant system auxiliary feed pumps tripped (terminated initial steam generator fill)	1 878.0 ± 1.0	1 954.0	1 913.0
Secondary coolant system steam bleed initiated	4 118.0 ± 1.0	3 600.0	--
HPIS flow ≥ break flow	4 200.0 ± 10	4 350.0	--
Accumulator injection initiated	5 029.0 ± 4.0	7 200.0	--
Primary system fluid becomes subcooled	8 200 ± 50	--	--
Purification system cooldown initiated[c]	12 300 ± 60	--	--

TABLE 2. (continued)

Event	Time After LOCE Initiation (s)		
	LOCE L3-2 Data	RELAP5[a] Prediction	RELAP4[a] Prediction
LPIS injection initiated	21 418 $\pm$ 5	--	--
Experiment completed[d]	23 350 $\pm$ 100	--	--

a. RELAP4 calculation terminated at 3600 s, RELAP5 at 7800 s.

b. Subcooled break flow continued throughout the transient in RELAP4 and RELAP5 calculations.

c. From experiment log.

d. End of experiment is defined as T_{system} <366.5 K.

Just before the pump coast-down was completed, the high-pressure injection system (HPIS) started injecting coolant into the intact loop cold leg. Forced loop circulation then ended as the pumps coasted down, and natural loop circulation followed at 36 s driven by the residual stored thermal and fission product energies in the core.

The pressurizer emptied at 136 s, followed by fluid saturation in the upper plenum at 180 s (Figure 7). Fluid saturation in the upper plenum increased the velocity of the fluid exiting the core and in the intact loop hot leg (Figures 8 and 9).

Figures 10 and 11 show a measurable positive temperature differential across the core and the steam generator until sometime between 600 and 1000 s. After that time, and until 8500 s, a negative temperature gradient existed from inlet to outlet on the primary side of the steam generator (Figure 11). Positive core flow continued through this period (Figure 8).

The pressure difference between the primary and secondary system decreased to about 0.10 MPa by 2000 s (Figure 12). Correspondingly the primary system pressure plateaued at 2000 s and then started to decrease, again, by 2500 s. At 2500 s, a negative temperature gradient of 1 or 2 K had developed from inlet to outlet on the steam generator primary side.

At 4118 s (Figure 5), operator-initiated bleeding of secondary steam, as planned, increased the depressurization rates of both the primary and secondary systems (Figure 12). At about the same time, the net depletion of system mass inventory stopped when HPIS flow matched break mass flow (Figure 13), as confirmed by the density in the broken loop cold leg (Figure 14).

The liquid level did not go below the bottom of the reactor vessel nozzles in the cold legs and was higher in the hot legs (Figures 14 and 15). Therefore, the core was covered throughout the experiment and remained cool as confirmed by the fuel cladding temperature and upper plenum fluid temperatures (Figures 16 and 18).

Accumulator injection pressure was reached at 5029 s, as the steam bleeding operation continued to be effective in reducing system pressure. At 8200 s, the fluid in the reactor vessel became subcooled (Figure 17). Purification system cooldown was initiated at 12 300 s, and cold shutdown temperature reached 366.5 K at 23 350 s, ending the experiment. Low-pressure injection system (LPIS) pressure was reached at 21 418 s. LPIS influence on the experiment was negligible (Figure 5).

2.3 Instrumentation Performance

The instrumentation used for LOCE L3-2 was essentially the same instrumentation used for LOCE L3-1[4]. For LOCE L3-2, low range (0 to 5 kPa) differential pressure transducers were added to the emergency core coolant (ECC) pitot tube rakes.

Of the 595 instruments operable prior to and recorded for LOCE L3-2, it is estimated that 573 (96%) performed satisfactorily. The pulsed neutron activation (PNA) flowmeter in the intact loop hot leg provided 12 data points of flow velocity during the experiment. However, the measurement technique is still under developement and the data should be interpreted with that in mind.

3. EXPERIMENTAL RESULTS FROM LOCE L3-2

The preliminary analysis presented in this section is based on data processed and available within the first week following the conduct of LOCE L3-2 and, in certain instances, reflects the current lack of confirmatory data or analysis. Analysis of the LOCE L3-2 data will continue in order to further support the preliminary results and conclusions.

3.1 Discussion of Phenomena and Comparison with Predictions

Flow out of the steam generator exhibited different behavior than was calculated prior to the experiment. This and other unanticipated phenomena, as well as some which were anticipated to occur, are discussed in this section.

3.1.1 Break Flow

Figure 19 compares the break flow during the experiment with the RELAP4, RELAP5, and TRAC-P1A results. Measured break flow exceeded the calculated flows, particularly during the early portion of the transient. Therefore, more mass exited the system early in the transient than was predicted by the codes.

The comparisons in Figures 20, 21, and 22 are consistent with the difference between measured and calculated break flows early in the transient. The codes underpredicted the system depressurization rate causing the predicted reactor scram to occur late as shown in Table 2. The pressurizer was calculated to empty between 359 and 400 s; whereas, during the experiment the pressurizer emptied at 136 s. Both of these differences indicate more system mass loss early in the transient than was calculated.

System break flow during the experiment was calculated from suppression tank liquid level measurements. The result was confirmed, early in the transient, by calculating system mass flow from pressurizer liquid level data. The pressurizer calculation accounted for system fluid swell and the change in pressurized liquid level due to flashing as the system pressure decreased (Figure 23). The result was confirmed later in the transient by calculations using calibration data on the LOCE L3-2 break orifice (Figure 24).

The cause of the excess mass flow during the experiment is being investigated. The two candidate causes, currently being considered, are (a) flow from the system, other than the break orifice, into the suppression tank and (b) miscalculation of break flow early in the transient.

3.1.2 Natural Loop Circulation and Core Cooling

As shown in Section 2.2, measured natural loop circulation continued until sometime between 1000 and 2000 s. Both the fluid temperature rise across the core and temperature drop from inlet to outlet in the steam generator primary side had become 1 K or less. The pressure difference between the primary and secondary systems was about 0.1 MPa, which is less than the uncertainty in the measurement.

Positive core outlet flow continued (was measured until 8500 s), the core remained cool, and the steam generator continued to be effective as a heat sink even though there was no measurable natural loop circulation between 2000 and 8500 s. At 2000 s, a negative temperature difference between the steam generator inlet and outlet started to develop, possibly indicating the onset of another cooling mode in the steam generator. From that time until late in the transient (8500 s), the negative temperature difference increased. Either natural circulation continued, but could not be measured, or another mode of cooling, such as reflux, occurred. The negative temperature differential across the steam generator may be evidence of reflux cooling.

Measurable natural loop circulation occurred again late in the experiment, starting at 8500 s as evidenced by core and steam generator differential temperature (Figures 10 and 11). By that time the fluid in the system had become subcooled. Fluid velocities were very small and could not be detected at the core outlet (Figure 8).

The combination of measurable natural loop circulation and the cooling mode between 2000 and 8500 s was sufficient to keep the system from repressurizing and to retain effective primary to secondary system thermal communication throughout the experiment, Thus, the operator-initiated steam bleeding at 4118 s was effective in reducing primary system pressure.

3.1.3 Unanticipated Phenomena and/or Events

As noted in Section 3.1.1, the initial break mass flow was larger than anticipated. The discussion and possible causes are found in Section 3.1.1.

The reflood-assist-bypass (RAB) line, between the broken loop hot and cold legs, allowed a small amount of leakage flow from the hot leg to the cold leg. In effect LOCE L3-2 had a communicative break. Consideration is being given to removal of the RAB prior to the next small break experiment.

Steam leaked from the secondary side of the steam generator, presumably at either the main steam value or its bypass. The amount of leakage calculated between 2500 and 3500 s was 2.3 kg/min, based on the change in steam generator liquid level.

Calculations after the experiment, which included the leak, showed no change in primary system pressure for the first 500 s of the transient. However, the secondary pressure was increasing but stayed below the main steam control valve high pressure relief setpoint. Calculations without the leak showed much higher secondary system pressurization rates and multiple violations of the high pressure relief setpoint. Secondary system pressure did not reach the high pressure relief setpoint during the experiment.

The pressurizer was expected to fill late in the experiment, but not at the rate which occurred. At 21 394 s, the pressurizer started to fill at a rate of 110 mm/s which reduced to 30 mm/s before filling stopped at 21 422 s with the level at 1.4 m (it did not completely fill at that time). The relatively rapid insurge of fluid appears to be the result of condensing superheated steam in the pressurizer. In the process system, pressure was decreased sufficiently to initiate the LPIS.

3.2 Experiment Objectives

Results from LOCE L3-2 which address the questions listed in Section 1 are discussed in this section. The first question is experiment specific, and LOCE L3-2 provided sufficient information to provide an answer to this question. The remaining questions are general questions. General questions cannot be answered completely by data from a single experiment. However, the information derived from LOCE L3-2, based on a preliminary assessment of the data, is presented.

The answers to questions (8), (9.b), (10), and (11) from Section 1 are beyond the scope of this document, and they are not addressed. Discussions of each the remaining questions follow:

3.2.1 "How does the primary coolant system respond during a small break when break flow is the same order of magnitude as the HPIS as system pressure stabilizes later in the transient?"

The previous sections of the report coupled with the conclusions in Section 4 provide an answer to this question.

3.2.2 "How effectively do the ECC systems perform during the consequent pressure transients for this type of depressurization?" (The type mentioned in Section 3.2.1.)

The core remained covered and cool throughout the transient. The HPIS was effective in stopping the system mass inventory depletion. The accumulator hastened vessel refill and did not appear to retard system depressurization. The LPIS was initiated after the pressurizer filled, which was well after the purification system was in control of system cooldown. Thus, the LPIS did not play a significant role during the experiment.

3.2.3 "How many of the major systems, such as, LPIS, accumulator, HPIS, steam generator, etc. are needed to prevent core damage during a small break, and are these same systems required for other break sizes or locations?"

Both the HPIS and steam generator were used and effective in preventing core damage during this experiment. The accumulator and LPIS probably were not needed. Whether or not the HPIS was necessary is a matter of further investigation and possible future testing. The question is, "Would the combination of steam generator cooling coupled with operator-initiated steam feed-and-bleed, starting earlier in the transient, have been sufficient to prevent core damage?"

3.2.4 "What kind of recovery procedures should be used in the event of a small break LOCA?"

Operator-initiated feed-and-bleed was effective in reducing primary system pressure. Consideration should be given to optimizing the time this operation is initiated. Earlier initiation may reduce the severity of the transient.

3.2.5 "Are there key times in the transient where operator action is required to protect the core?"

The only operator action taken during LOCE L3-2 was secondary steam bleeding. At the time this action took place, system pressure depressurization rate was decreasing. 82 s after the action, HPIS flow stopped further net system mass depletion. It's not apparent from this experiment that any operator action was required to protect the core.

3.2.6 "Are there operator/equipment actions that must not occur?"

Part of the answer to this question correlates with the answer for Section 3.2.3. The operator should not take any action which would jeopardize the operation of both the steam generator and the HPIS. Whether or not the loss of either would cause damage to the core, should be studied and addressed in future testing.

3.2.7 "Are typical commercial pressurized water reactor process instruments capable of providing accurate information on plant conditions during a transient? Specifically, which instruments furnish relevant data and which do not?"

Wide range pressure instruments are needed to monitor system pressure. Commercial PWR instruments appear adequate for this purpose (Figure 25).

The submeter (Figure 26) provides no information on system liquid level when the fluid in the system is saturated. The inherent error (± 5 K) in the submeter makes accurate determination of the loss or reestablishment of subcooled conditions in the system difficult. During LOCE L3-2 the operator depended on hand calculation using readings from other instruments to determine the time subcooling was reestablished rather than using the indication from the submeter.

4. CONCLUSIONS

The conduct of LOFT LOCE L3-2 and the experimental data acquired concerning integral systems phenomena associated with a loss of coolant are considered to have met the objectives as defined by the EOS[6] and discussed in Section 3. Conclusions based on the preliminary analyses and experiment assessment are as follows:

1. The core remained covered during the entire transient. No fuel rod damage resulted.

2. The steam generator was an effective heat sink throughout the experiment even though natural loop circulation could not be measured for 6500 s, starting about 2000 s into the transient.

3. Another cooling mode may have occurred in the steam generator during the period natural loop circulation could not be measured.

4. Measurable natural loop circulation was reestablished as the vessel refilled and the system fluid became subcooled.

5. Secondary steam bleeding was effective in reducing primary system pressure.

6. HPIS flow equaled or exceeded break flow about the time secondary system steam bleeding was initiated.

7. The mass leaving the system early in the transient was significantly greater than anticipated.

8. Computer calculations predicted the dominant phenomena, in the proper time sequence, except for the large mass flow from the system early in the experiment.

5. DATA PRESENTATION

This section presents selected, preliminary data from LOCE L3-2. LOCE L3-2 data are overlayed with data from LOCE L3-2 pretest calculations using the RELAP4, RELAP5, and TRAC-P1A computer codes. 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 LOCE L3-2 is given in Reference 6.

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

1. Temperature - $\pm$ 3 K
2. Pressure - $\pm$ 0.21 MPa
3. Density - $\pm$ 0.043 Mg/m^3
4. Mass flow rate - $\pm$ 10% (integrated uncertainty)
5. Submeter - $\pm$ 5 K.

TABLE 3. LIST OF DATA PLOTS

TABLE 3. (continued)

Figure	Title	Measurement Identification	Page
19	Comparison of broken loop cold leg mas_ flow with predictions		35
20	Comparison of system pressure with predictions from 0 to 1000 s	PE-PC-6	35
21	Comparison of pressurizer liquid level with predictions	LT-P139-6	36
22	Comparison of system pressure with predictions from 0 to 8000 s	PE-PC-6	56
23	Comparison of broken loop cold leg mass flow calculated from suppression tank and pressurizer liquid level data		37
24	Comparison of broken loop cold leg mass flow calculated from suppression tank liquid level data and break orifice calibration test		37
25	Comparison of reactor vessel upper plenum pressure measured with process and experiment instrumentation	PT-P139-2 PE-1UP-1A	38
26	Submeter output		38

TABLE 4. NOMENCLATURE FOR LOFT INSTRUMENTATION

<u>Designations for the Different Types of Transducers:</u>[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

<u>Designations for the Different Systems, Except the Nuclear Core:</u>

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

<u>Designations for Nuclear Core Instrumentation:</u>

Transducer location (inches from bottom of fuel rod)

Fuel assembly row

Fuel assembly column

Fuel assembly number

Transducer type

TE-3B11-28

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a. Includes only instruments discussed in this report.

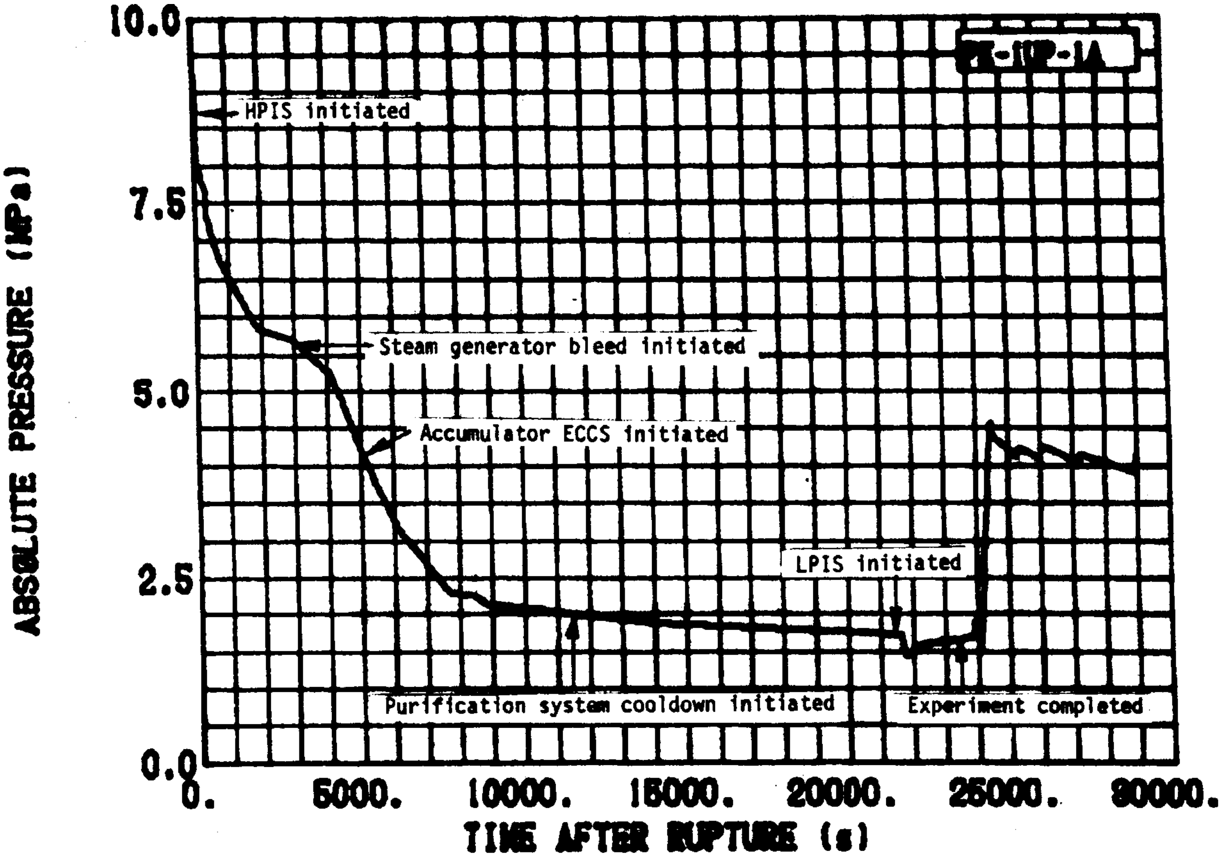

Figure 5. Pressure in reactor vessel upper plenum.

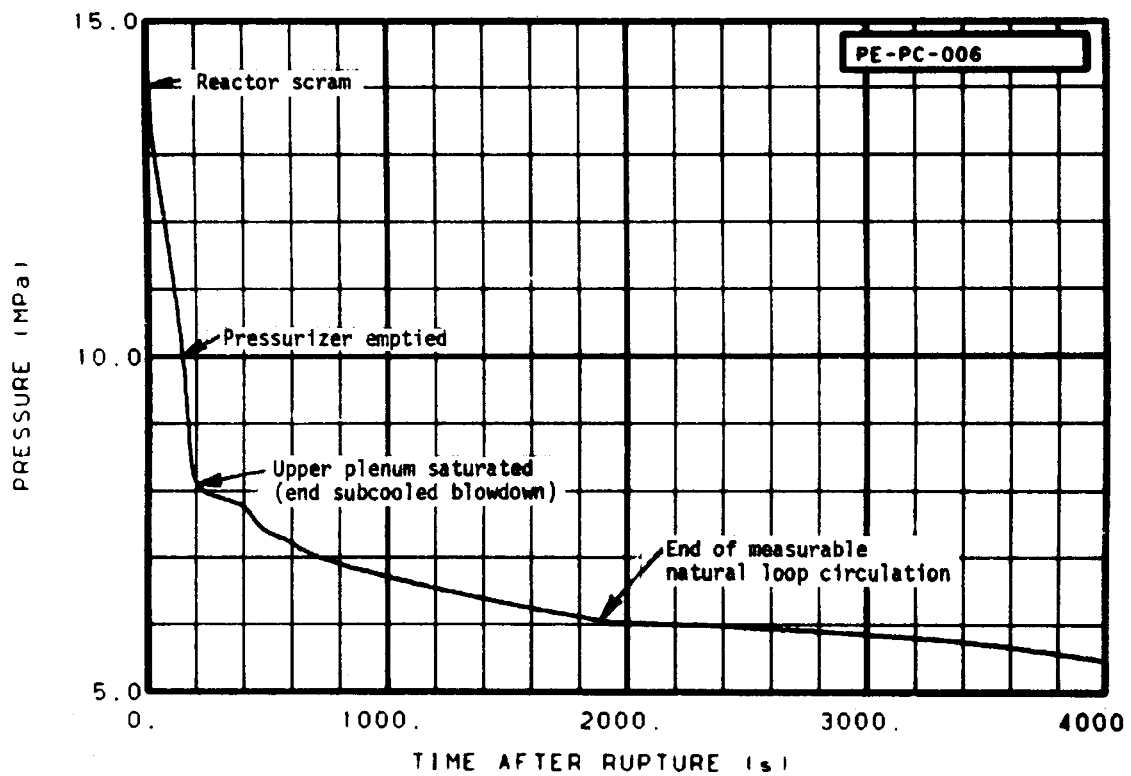

Figure 6. Pressure in primary system intact loop from 0 to 4000 s.

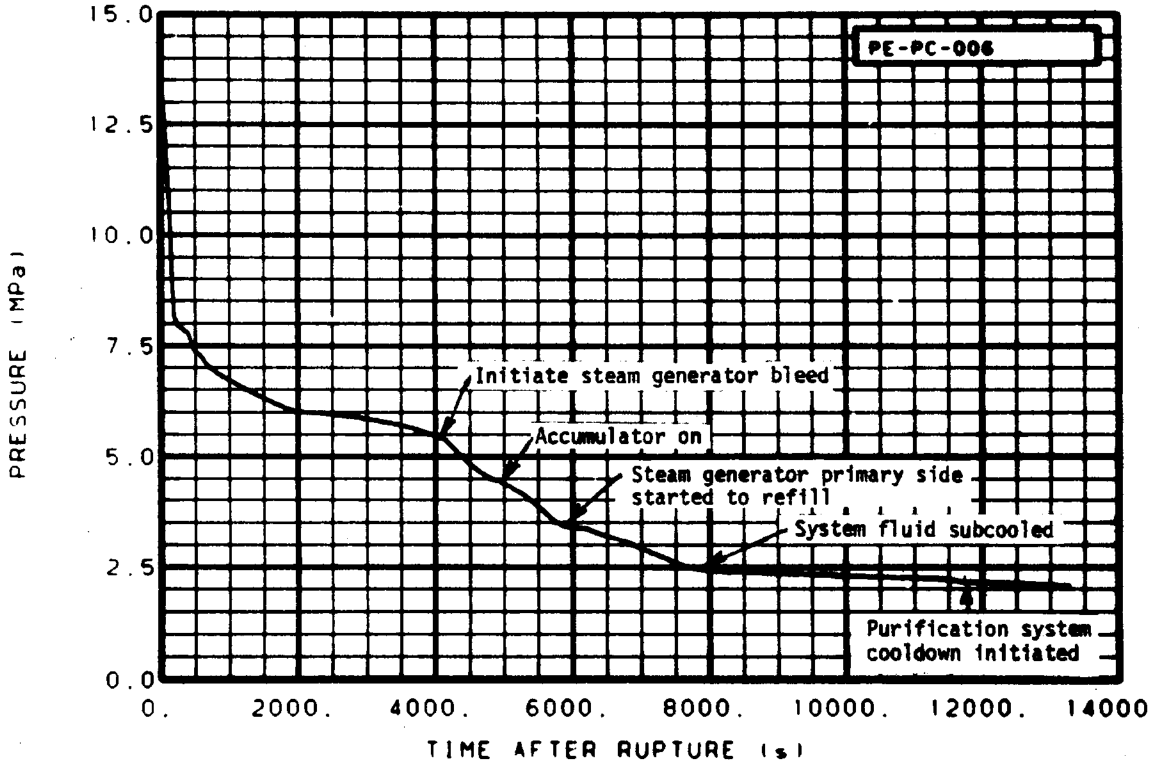

Figure 7. Pressure in primary system intact loop from 0 to 14 000 s.

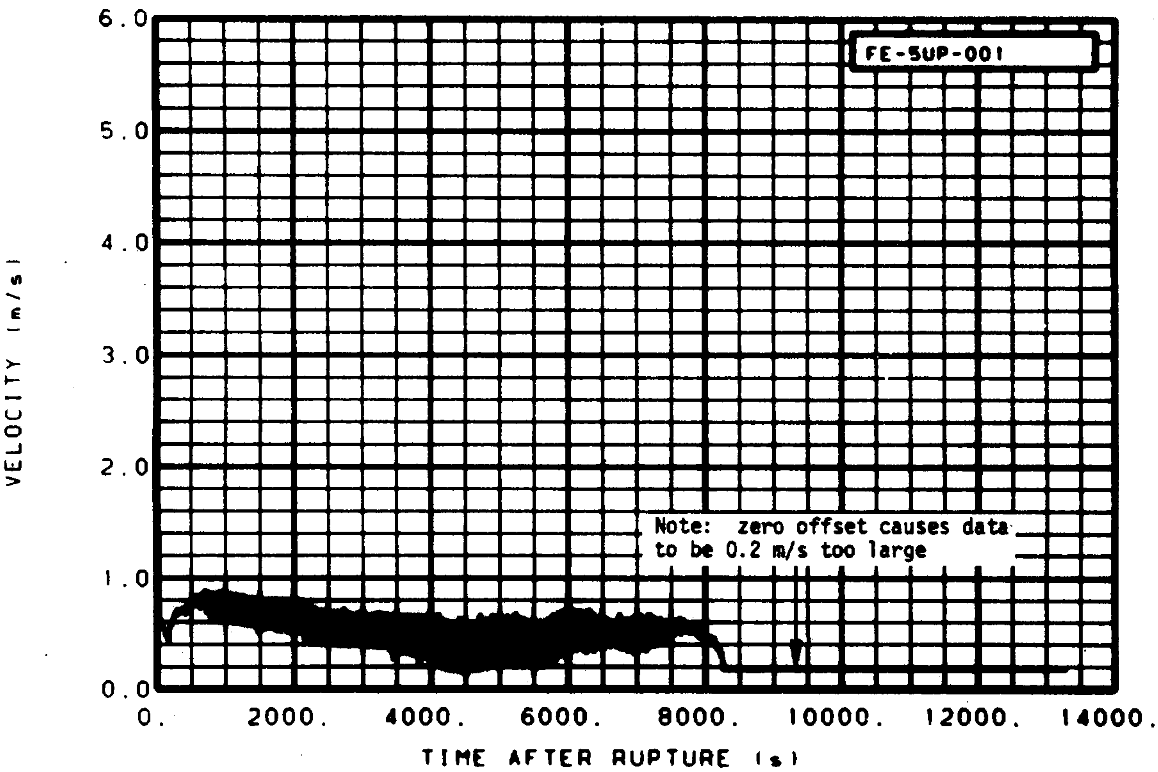

Figure 8. Fluid velocity above center fuel module.

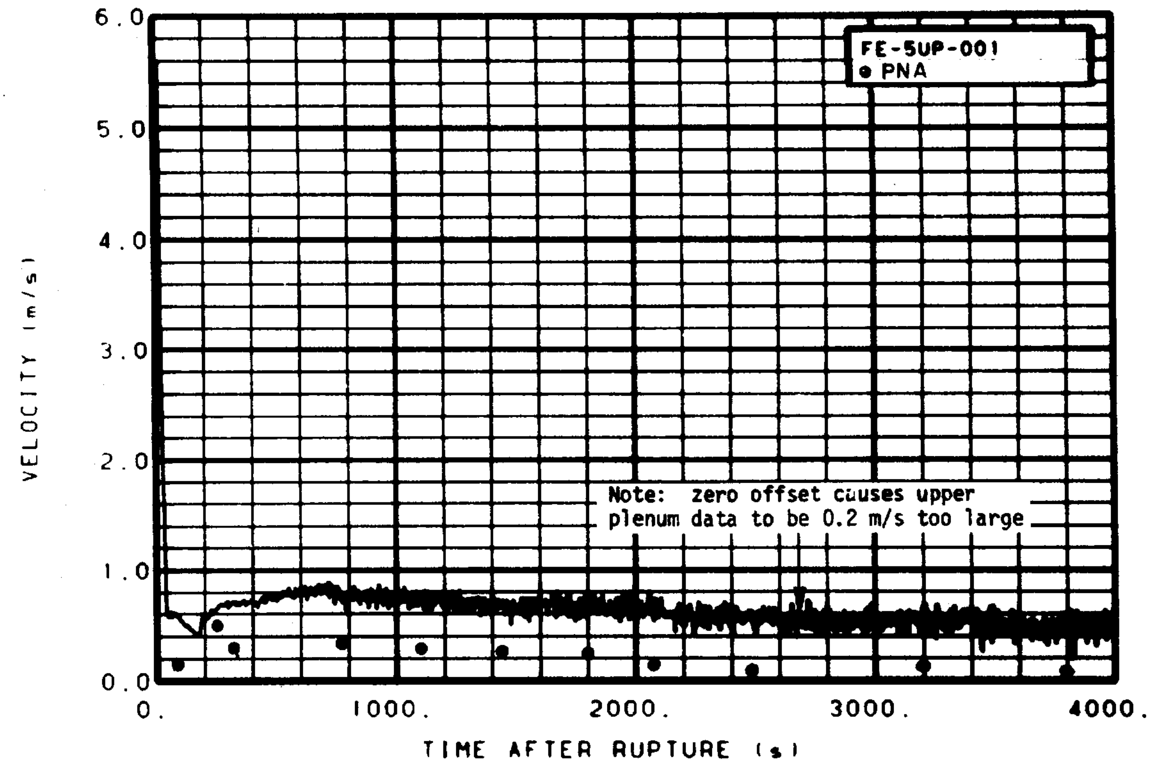

Figure 9. Comparison of fluid velocity above center fuel module and in the intact loop hot leg.

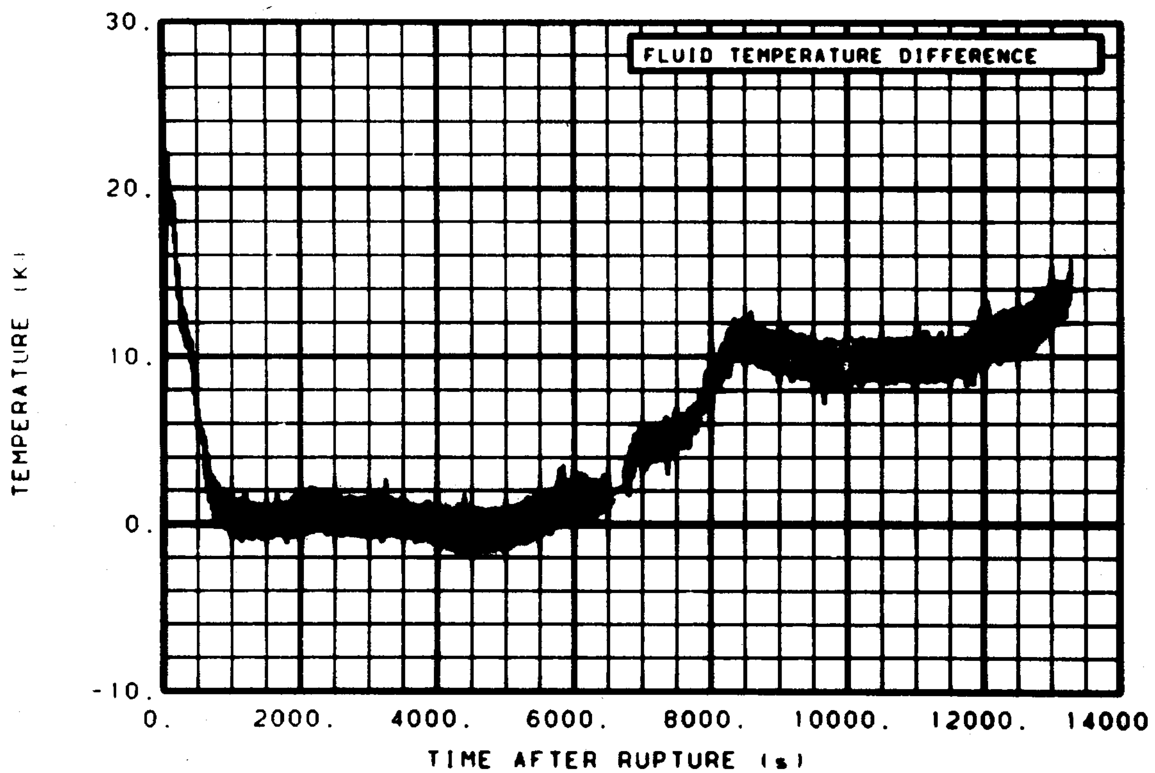

Figure 10. Fluid temperature difference across the center fuel module.

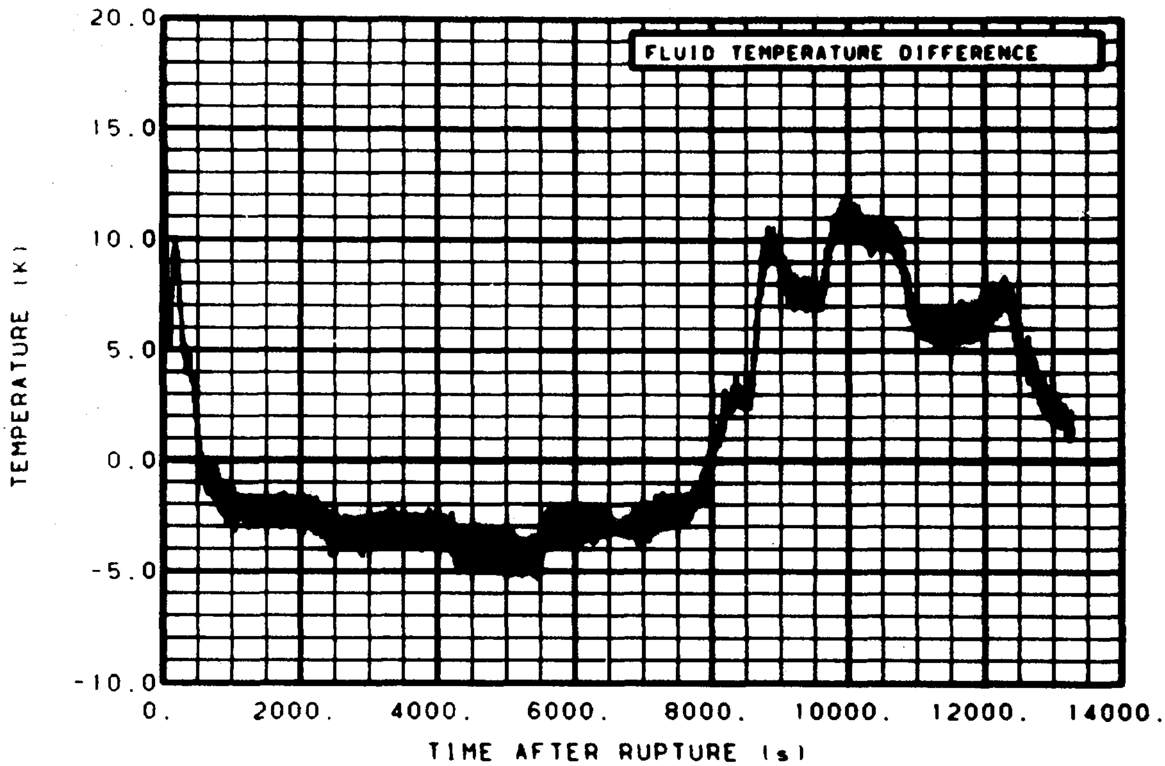

Figure 11. Fluid temperature difference in the steam generator between the primary system inlet plenum and the secondary system downcomer.

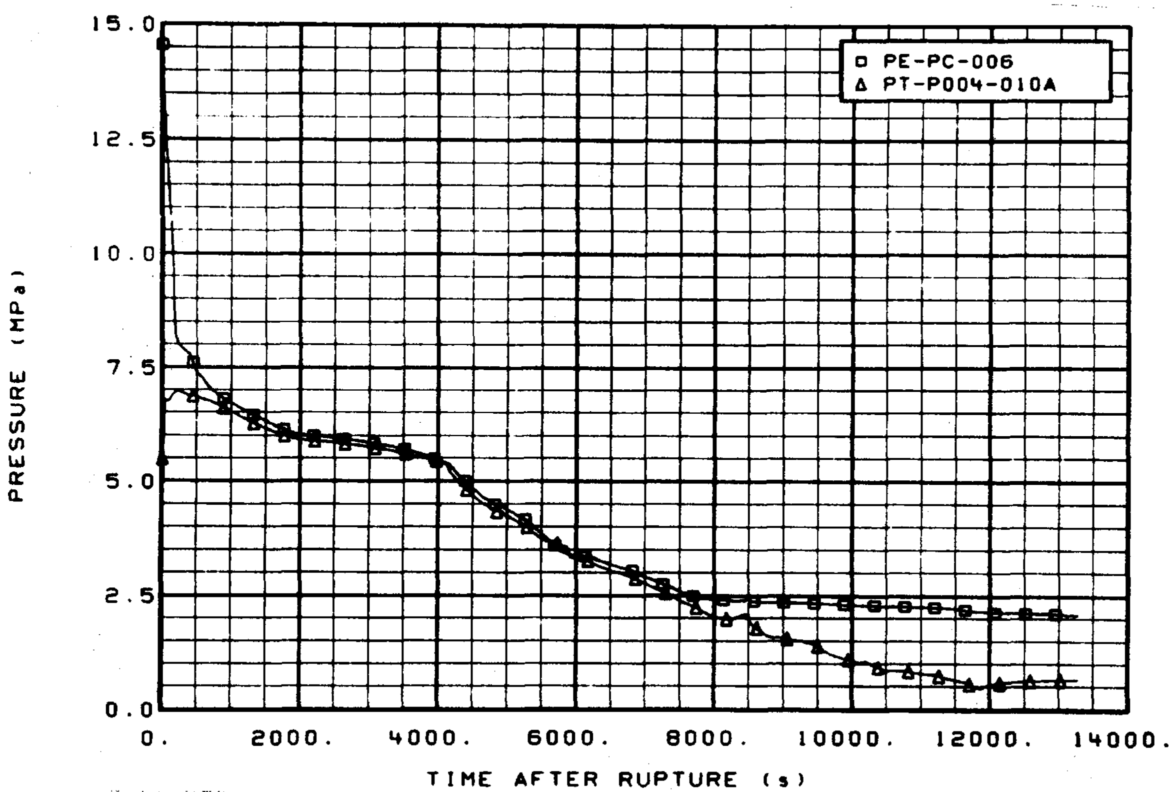

Figure 12. Comparison of primary and secondary system pressures.

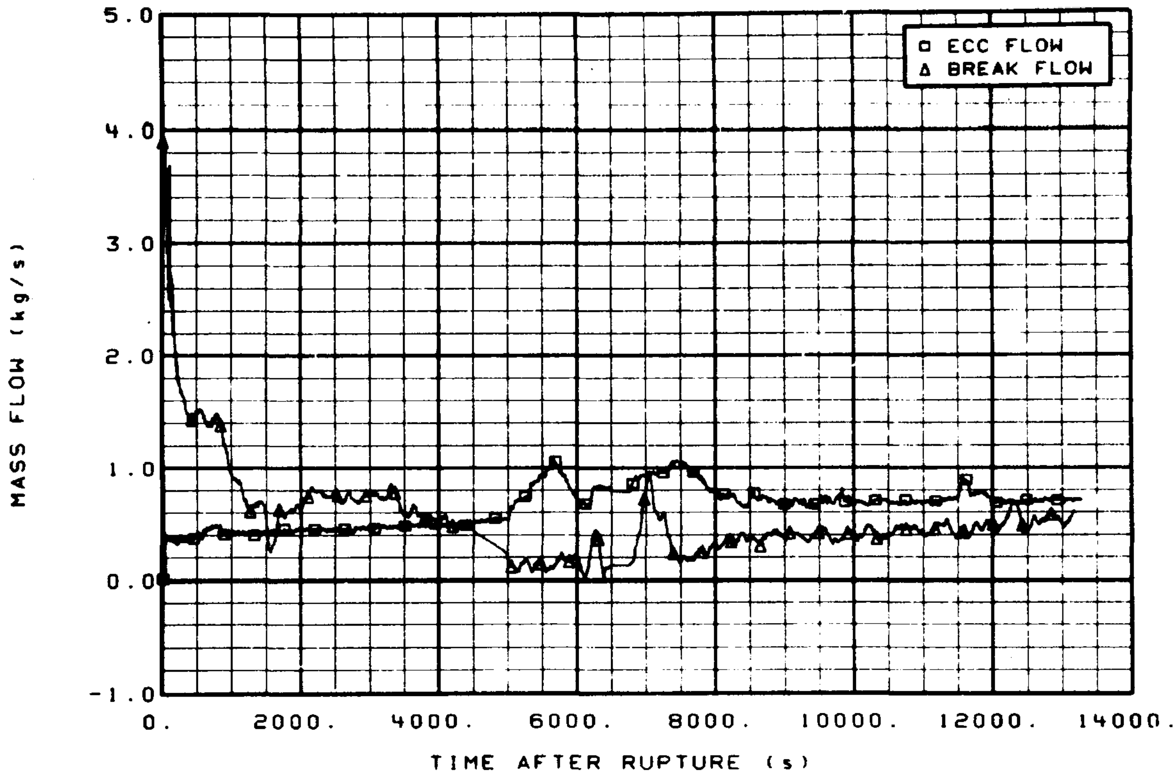

Figure 13. Comparison of break flow and ECCS flow.

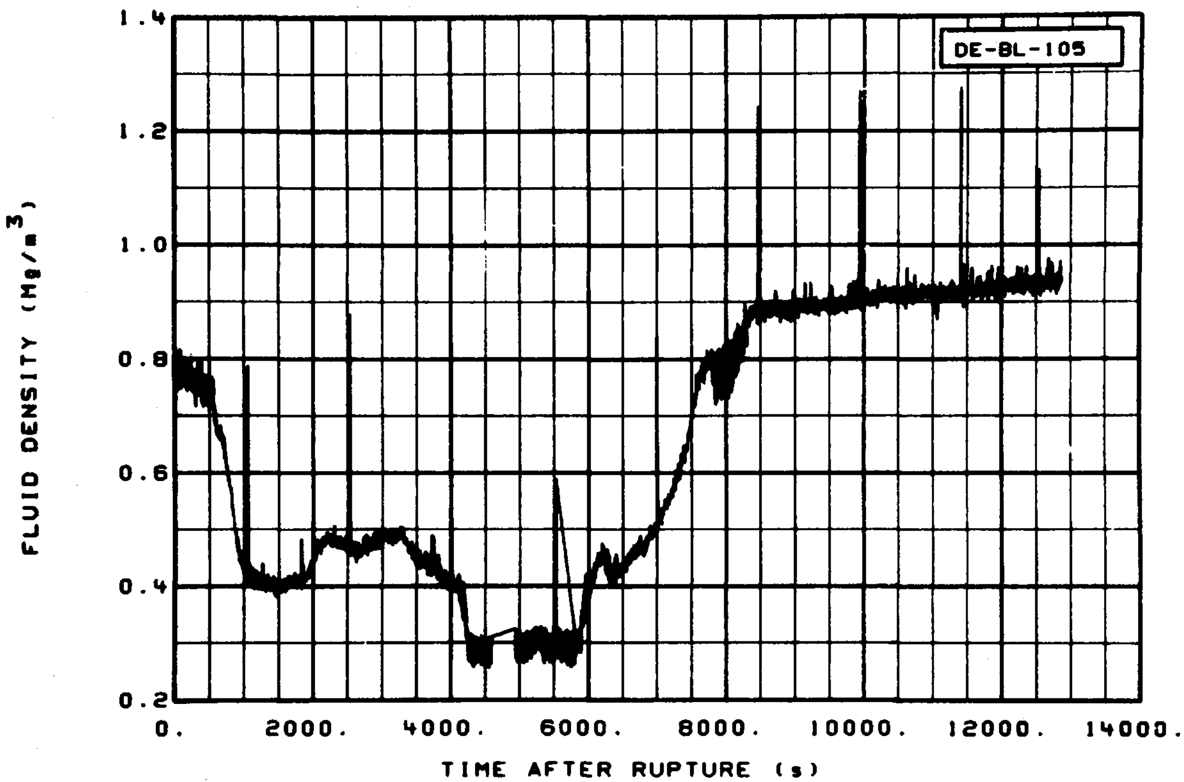

Figure 14. Average fluid density in the broken loop cold leg.

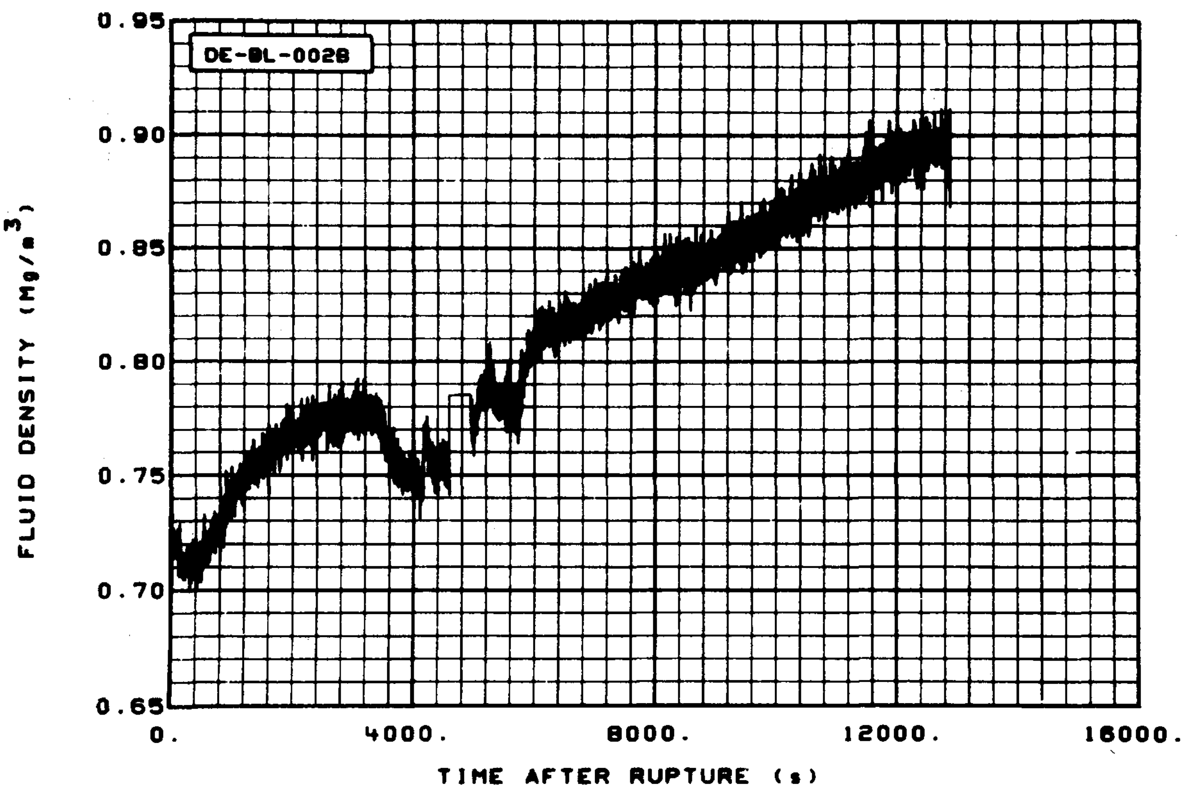

Figure 15. Fluid density in the broken loop hot leg.

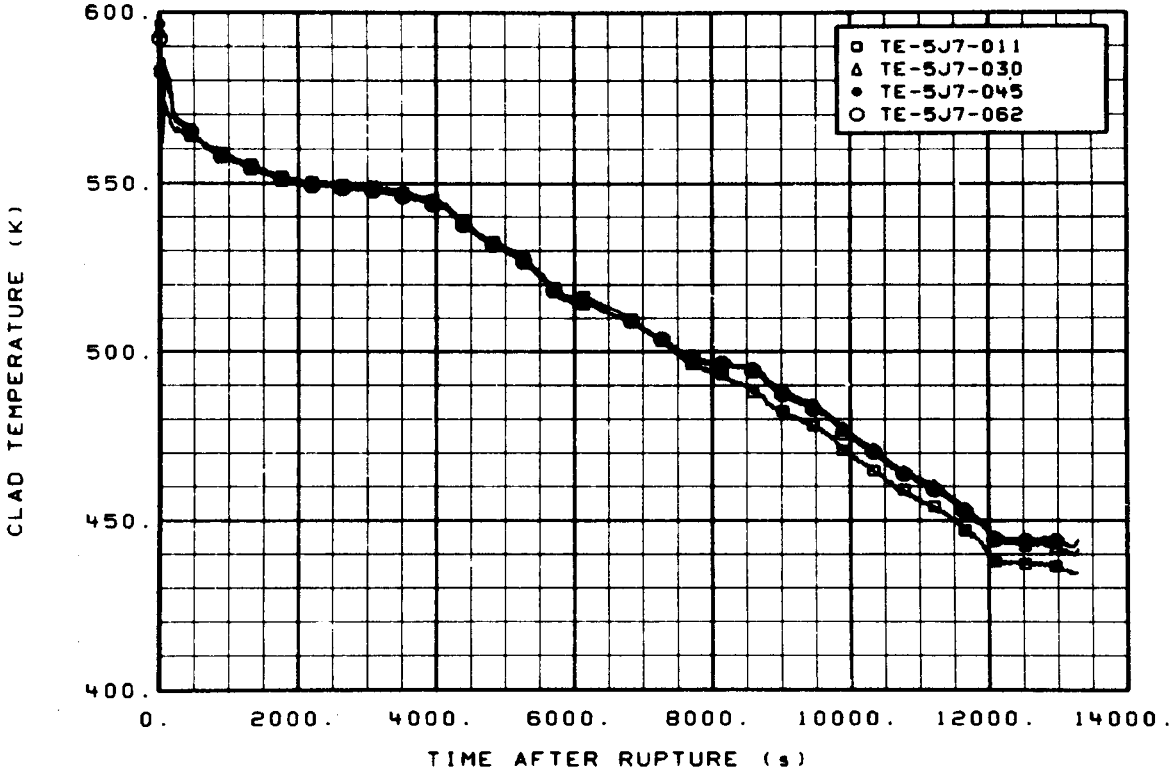

Figure 16. Fuel cladding thermocouple temperatures in the center fuel module.

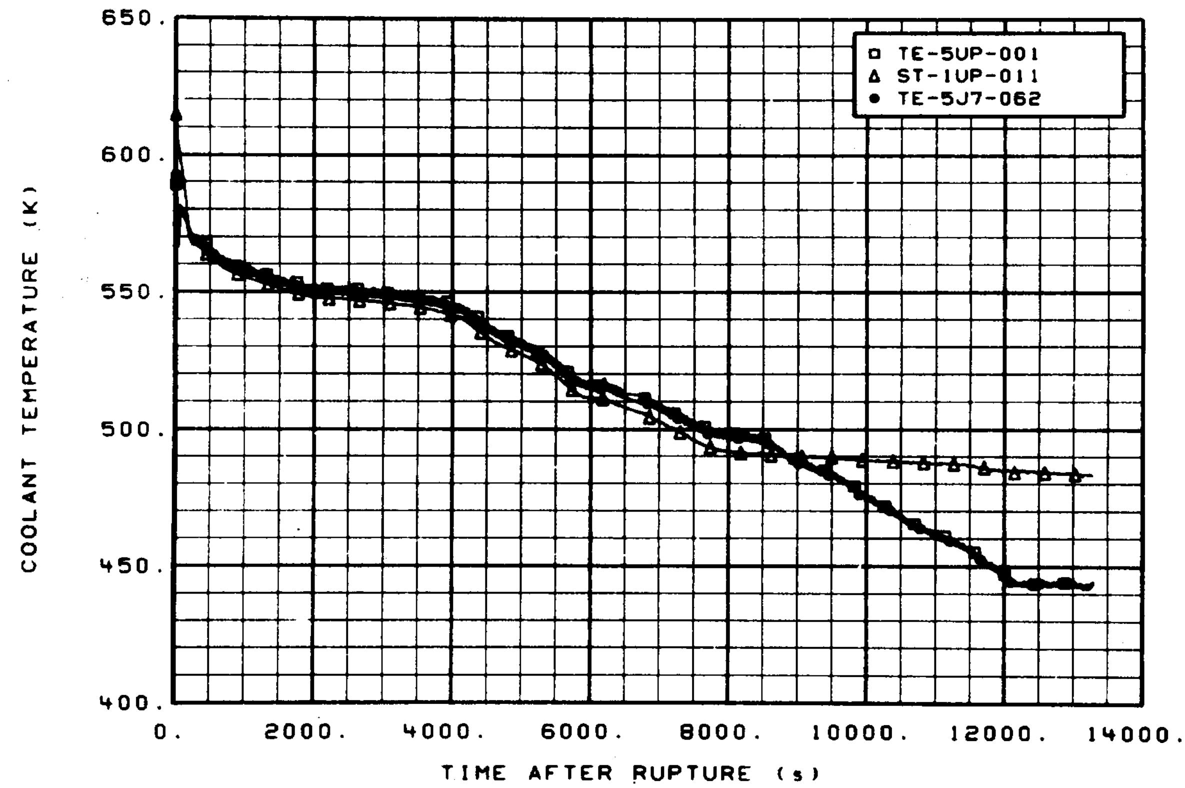

Figure 17. Comparison of upper plenum fluid, fuel cladding, and fluid saturation temperatures.

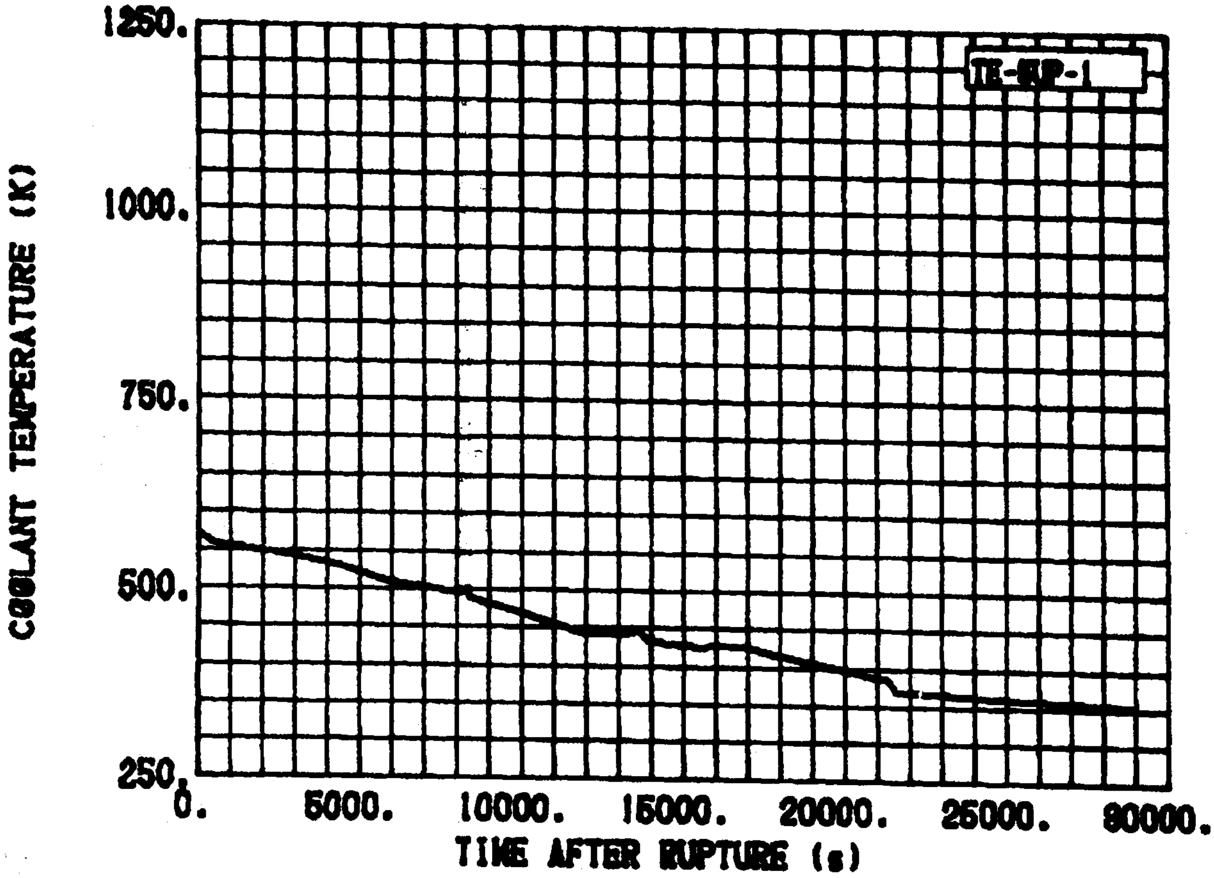

Figure 18. Fluid temperature in the reactor vessel upper plenum.

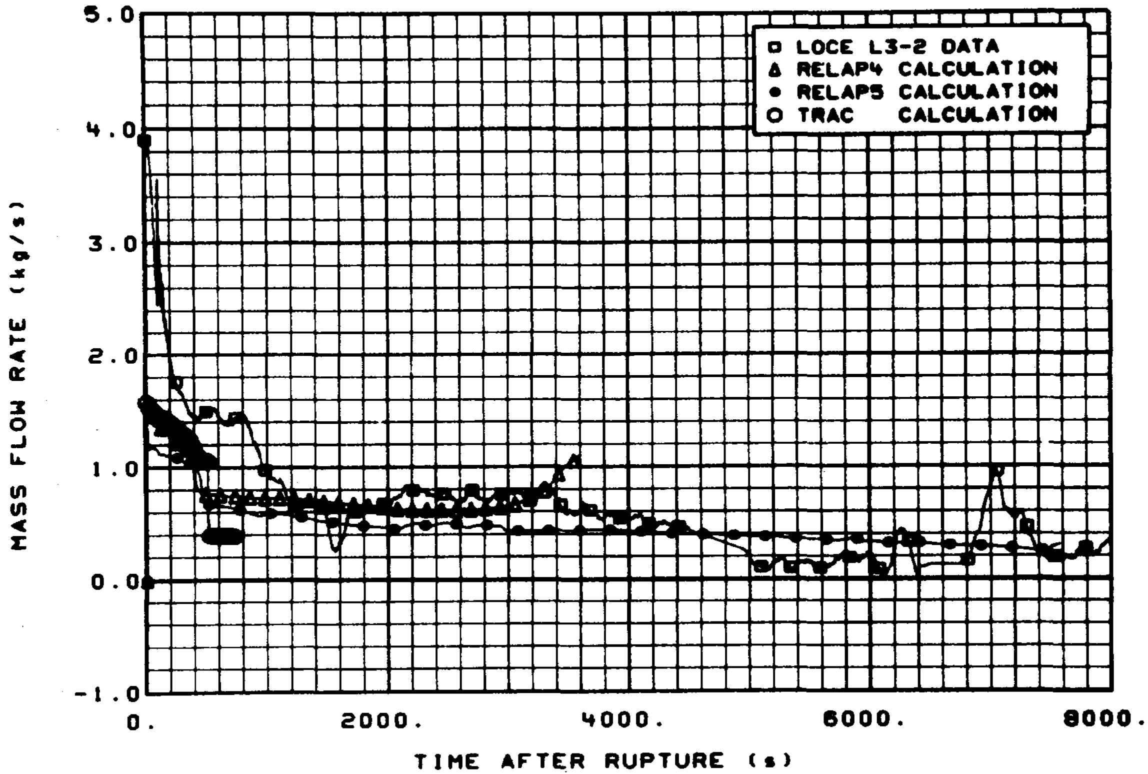

Figure 19. Comparison of broken loop cold leg mass flow with predictions.

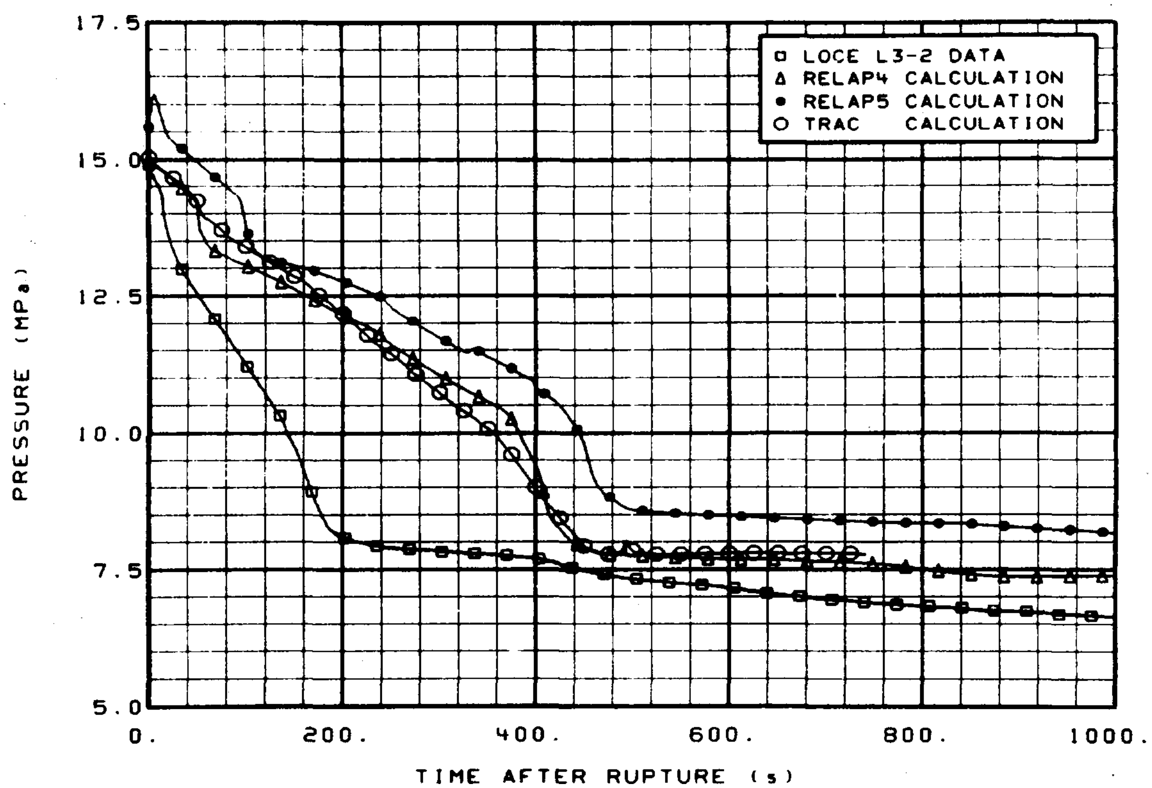

Figure 20. Comparison of system pressure with predictions from 0 to 1000 s.

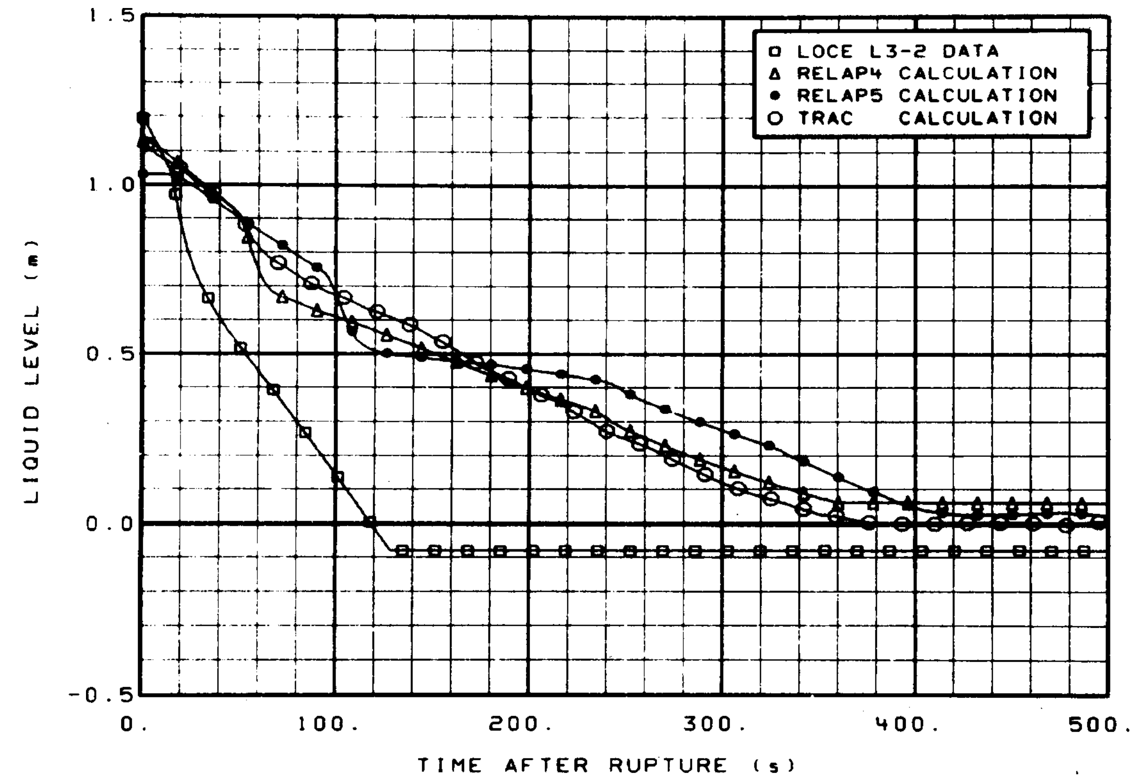

Figure 21. Comparison of pressurizer liquid level with predictions.

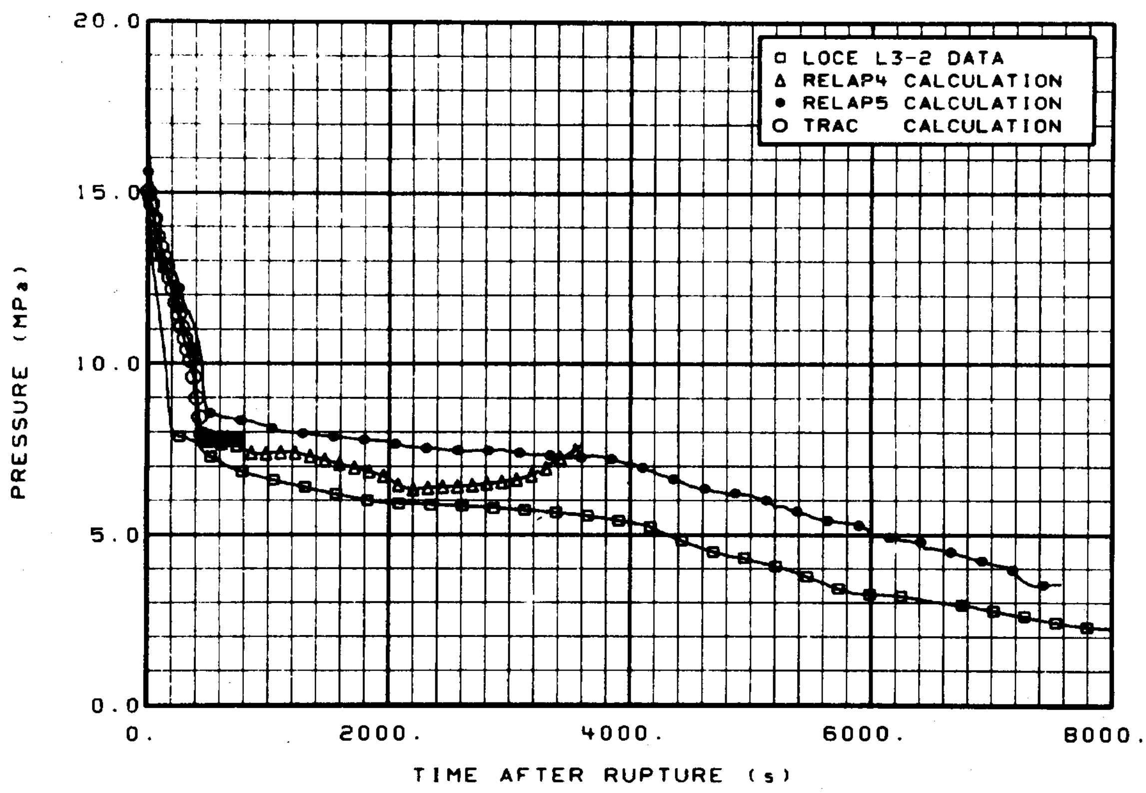

Figure 22. Comparison of system pressure with predictions from 0 to 8000 s.

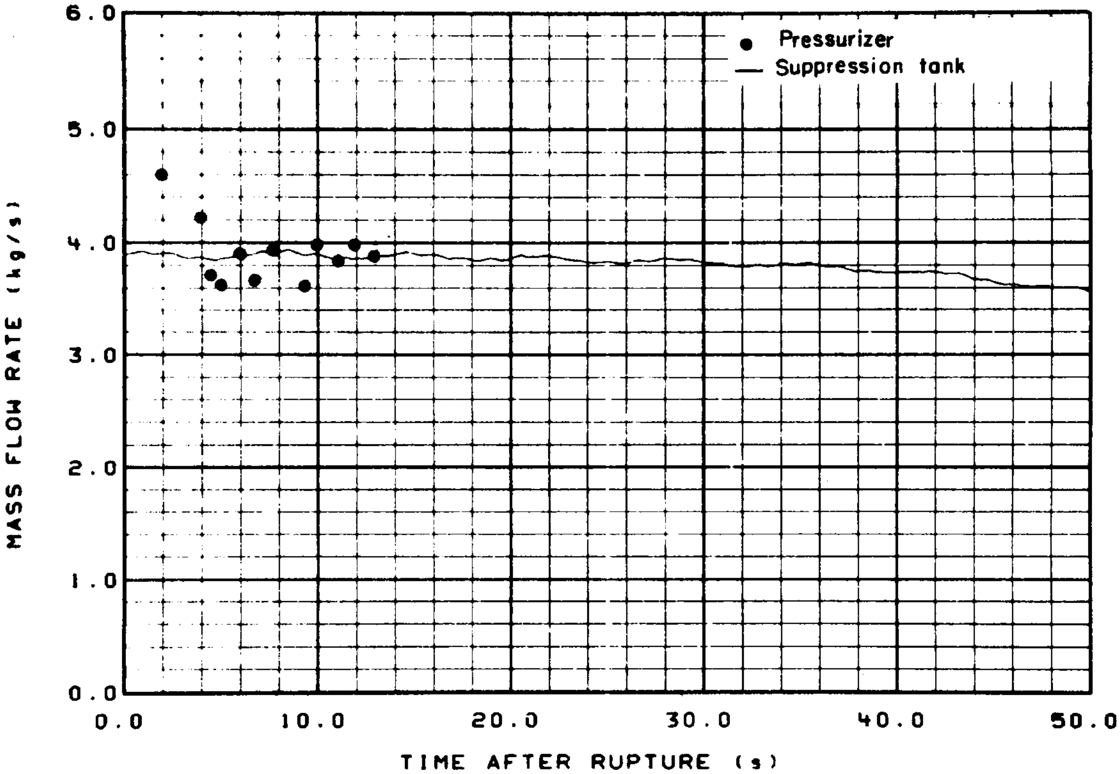

Figure 23. Comparison of broken loop cold leg mass flow calculated from suppression tank and pressurizer liquid level data.

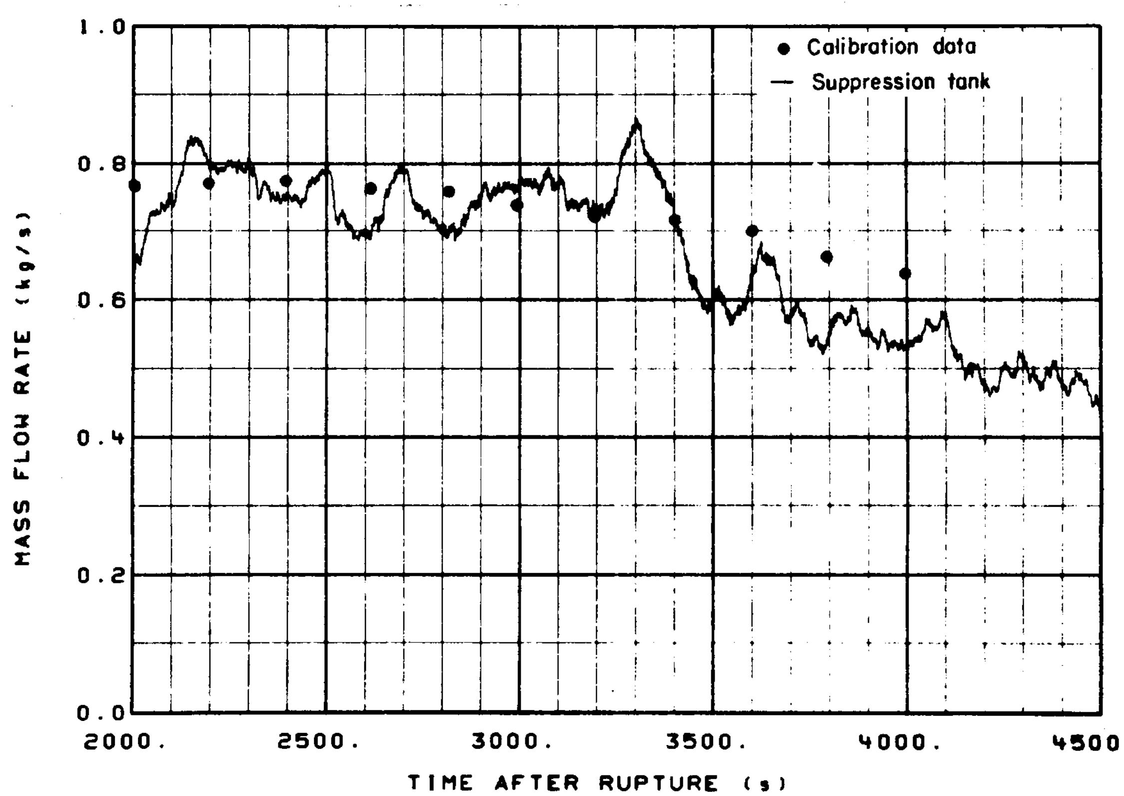

Figure 24. Comparison of broken loop cold leg mass flow calculated from suppression tank liquid level data and break orifice calibration test.

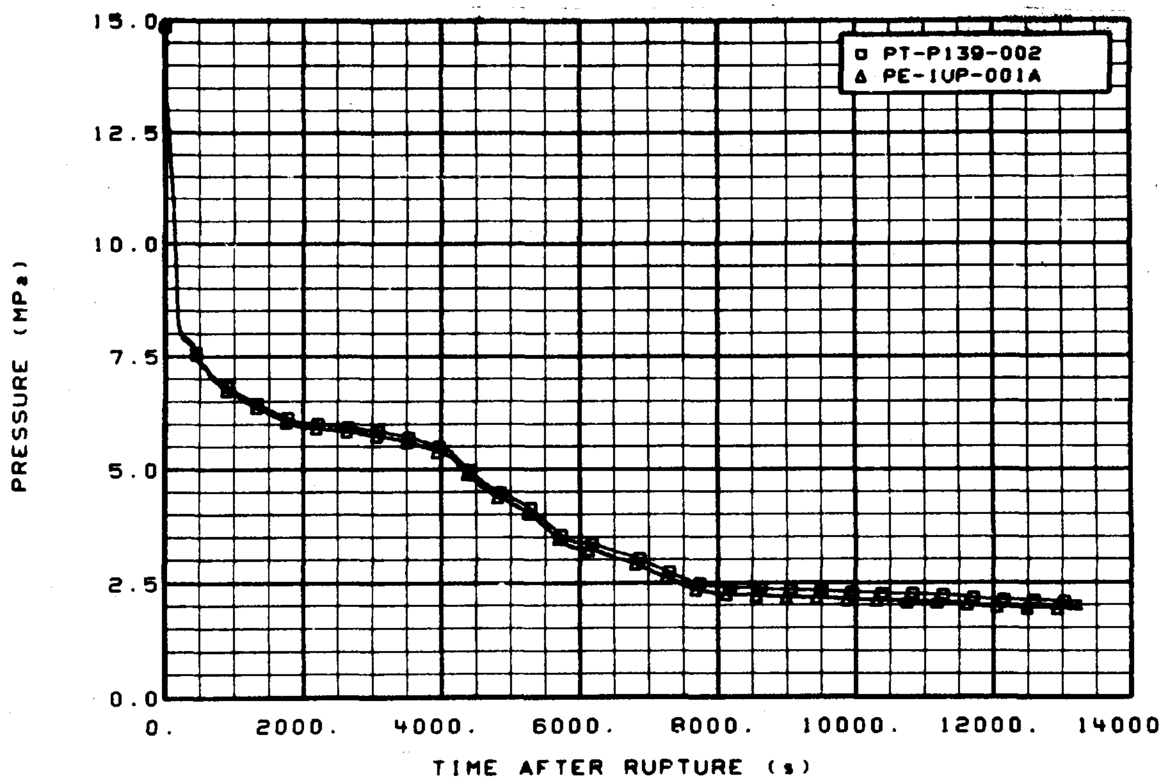

Figure 25. Comparison of reactor vessel upper plenum pressure measured with process and experiment instrumentation.

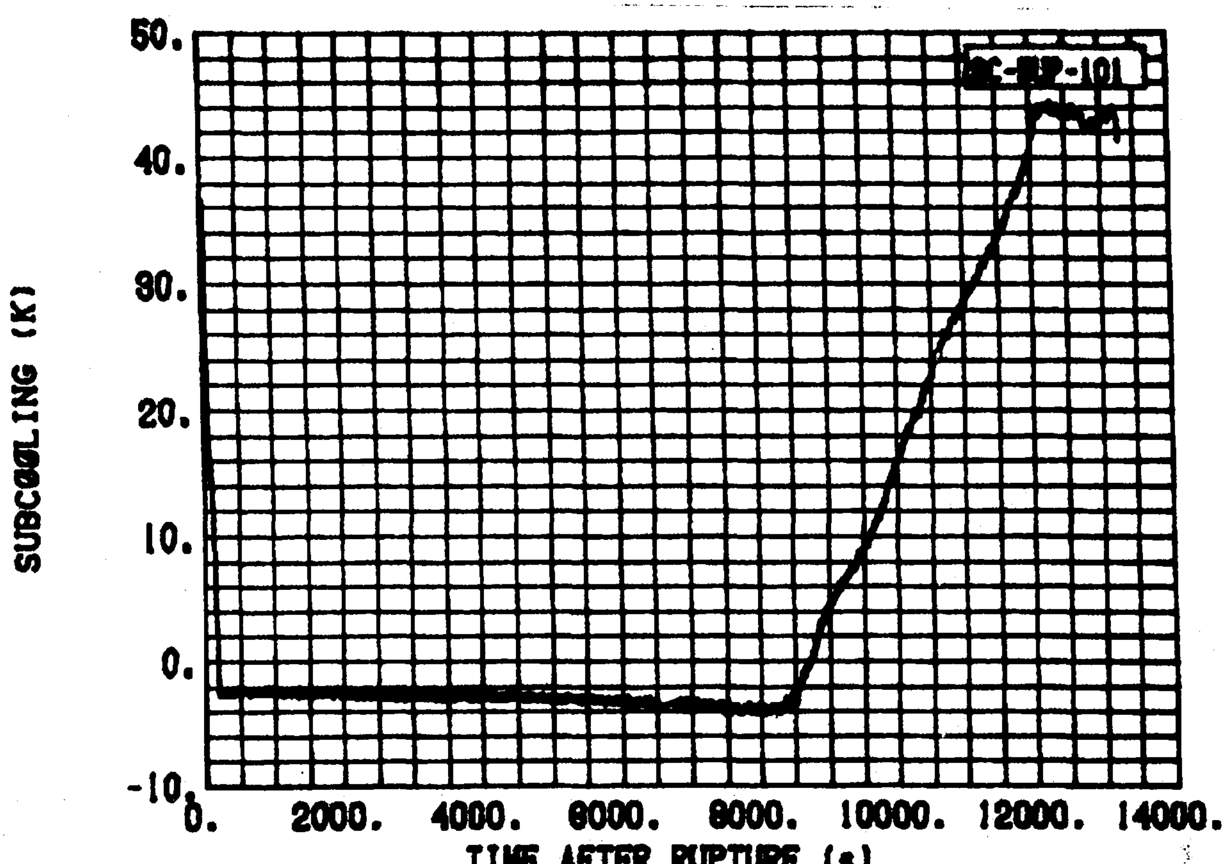

Figure 26. Submeter output.

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