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# QUICK-LOOK REPORT ON LOFT NUCLEAR EXPERIMENT L3-7

Glen E. McCreery

## U.S. Department of Energy
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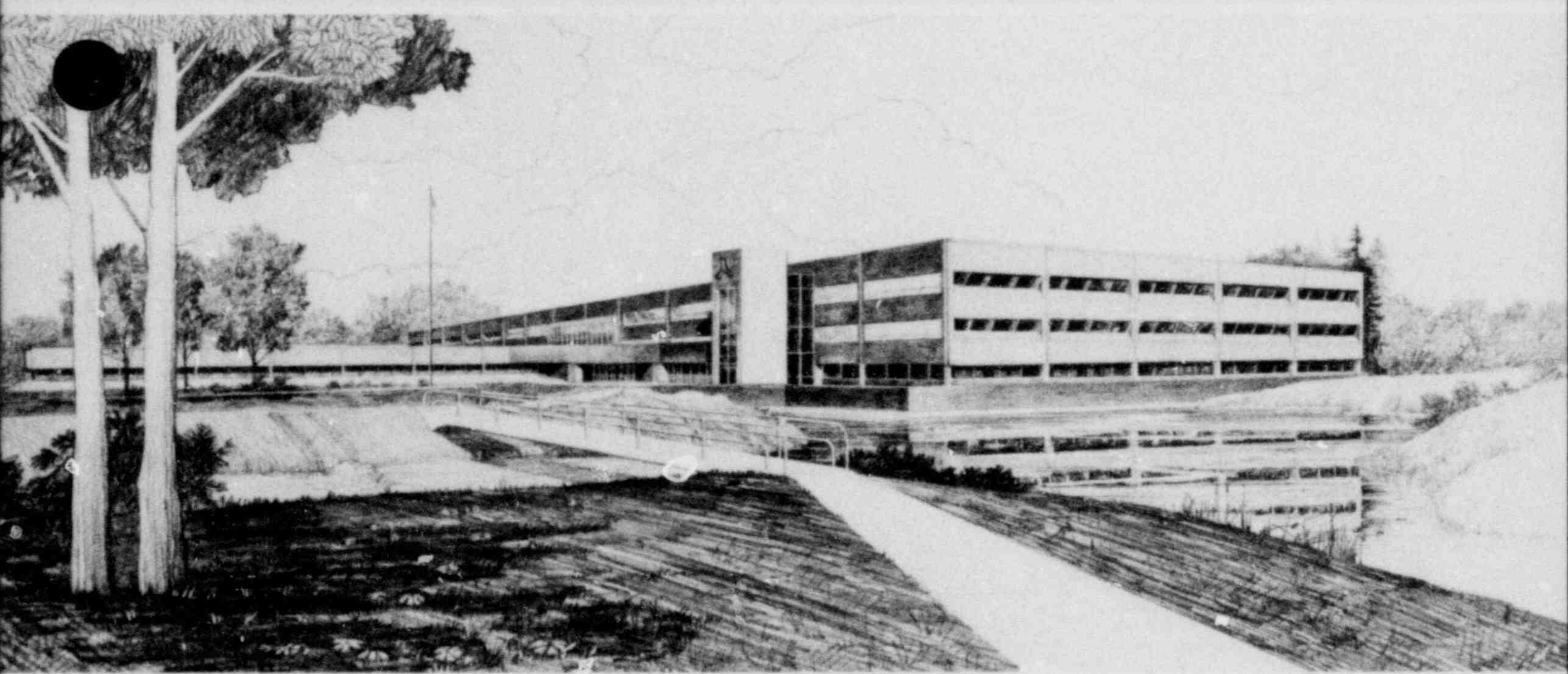

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G. D. McPherson, Chief, LOFT Research Branch,

Division of Reactor Safety Research, USNRC



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

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INTERIM REPORT

# QUICK-LOOK REPORT ON LOFT NUCLEAR EXPERIMENT L3-7

By:
G. E. McCreery

Approved:

Charles W. Solbrig
C. W. Solbrig
LOFT Program Division

N C Kaufman
N. C. Kaufman, Director
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) L3-7, the fourth experiment in the Loss-of-Fluid Test (LOFT) Small Break Test Series L3 scheduled for performance in the LOFT facility, was successfully completed on June 20, 1980. LOCE L3-7 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 pressure continuously decreased to the point that high-pressure injection system (HPIS) flow was about 70% of the break flow. At 1805 s after experiment initiation, the HPIS was turned off to allow system fluid inventory to decrease more rapidly. At 3576 s, steam generator secondary fluid feed and bleed was initiated to induce a more rapid depressurization of the primary system. The HPIS was again turned on at 5974 s, at which time it exceeded break flow, and the break isolation valve was closed at 7302 s. Steam generator heat transfer was effective in removing decay heat energy not removed by the break, throughout the experiment. The reactor system was brought to a cold shutdown condition 29 500 s after experiment initiation.

Natural circulation was maintained throughout the large majority of the transient, and the concomitant heat transfer in the steam generator was effective in removing decay heat. However, the reflux flow mode was not established, although the intact loop hot leg voided and condensation occurred in the steam generator primary tubes. Computer code calculations predicted the dominant phenomena of the transient well, except for system depressurization during the period from 1200 to 3600 s, and system repressurization after the break was isolated.

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FIGURES



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## DEFINITIONS

Flow reversal - the inception of negative flow in a system component or at a particular location in the system. Positive flow is defined as the normal flow direction at steady state operation.

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.

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

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) L3-7, which was successfully completed on June 20, 1980, in the Loss-of-Fluid Test (LOFT) facility. LOCE L3-7 is the fourth experiment in the LOFT Small Break Test Series L3 and simulated a single-ended offset shear break of a small (1-in.-diameter) pipe attached to a cold leg of a large pressurized water reactor (PWR).

The primary objectives of LOCE L3-7 were to establish a break flow approximately equal to high-pressure injection system (HPIS) flow when the primary pressure was in the range of 6.9 MPa, to establish conditions for steam generator reflux cooling, to isolate the break and recover the plant to cold shutdown, and to analyze the data obtained to investigate associated phenomena. The test initial conditions and sequence of events were consistent with the objectives.

Prior to the break, the nuclear core was operating at a steady state maximum linear heat generation rate of 52.8 ± 3.7 kW/m. Other significant initial conditions for LOCE L3-7 were: system pressure, 14.95 ± 0.34 MPa; core outlet temperature, 576.1 ± 0.5 K; and intact loop flow rate, 478.8 ± 8.8 kg/s. At 36 s after the break occurred, the reactor scrammed on a low system pressure signal. Within 10 s after scram verification, the pumps were manually tripped and coasted down. Pump coastdown was followed by the inception of natural loop circulation. Between 1800 s (30 min) and 5974 s (1 hr 40 min), HPIS was turned off to hasten the loss of fluid inventory to establish the conditions considered favorable for reflux flow in the primary loop. Starting at 3600 s (1 hr), operator-controlled steam bleeding, by opening the main steam bypass valve early and the main steam valve later in the transient, and feeding, using both the auxiliary and main feedwater systems, were used to decrease primary system pressure.

Later in the experiment, 7200 s (2 hr), the quick-opening blowdown valve (QOBV) was closed, which isolated the break. System mass depletion stopped and all decay heat energy, not lost to the environment, was removed

by the steam generator. Primary system pressure gradually increased, causing the fluid in the system to become subcooled. Subsequently, the purification system was used to bring the reactor to a cold shutdown condition, and the experiment terminated.

The steam generator was an effective heat sink throughout the experiment. Both convection and condensation heat transfer modes were observed in the steam generator during the time the system fluid was saturated [382 to 7915 s (2 hr 12 min)]. During that time interval, the system did not repressurize, although system pressure did plateau after the HPIS was turned off and steam generator feeding was stopped at 1800 s (30 min). At 3600 s (1 hr), initiation of steam generator feeding and bleeding reestablished system depressurization, hastening progress of the system towards cold shutdown.

Natural loop circulation was effective in transporting energy from the core to the steam generator from pump coastdown at 61 s to initiation of purification system cooling at 18 180 s (5 hr). Reflux flow, that is, return flow from the steam generator through the intact loop hot leg to the core, did not appear to occur. This conclusion is based on flow measurements in the intact loop hot leg and steam generator inlet and outlet primary system fluid temperature measurements.

When the break was isolated at 7305 s (2 hr), the system gradually started to repressurize, natural circulation continued, and the core continued to cool. As system pressure rose, the fluid subcooled, retarding natural circulation velocities.

The gradual increase in system pressure, after break isolation, was induced by the increase in system fluid inventory driven by the HPIS. Superheated steam in both the pressurizer and vessel upper head was compressed and gradually condensed. This nonequilibrium process caused the pressurizer to start refilling, at 8200 s (2 hr 33 min), before the vessel head was refilled. At 12 000 s (3 hr 20 min), operator-initiated primary system bleeding through the power-operated relief valve was required to control primary system pressure.

Preexperiment computer calculations of the LOCE L3-7 transient were made by EG&G Idaho, Inc., using RELAP4 and RELAP5 and by Los Alamos Scientific Laboratory using TRAC-PIA. The RELAP4 and TRAC calculations terminated at 1800 s (30 min) and 3600 s (1 hr), respectively. The RELAP5 calculation terminated at 11 000 s (3 hr), when the system was predicted to have refilled. All three codes adequately predicted early (0 to 1200 s) system depressurization but overpredicted the time of pressurizer emptying. TRAC predicted the system would slightly repressurize between 1800 s (30 min) and 3600 s (1 hr), when the data showed no increase. RELAP5 predicted the system pressure trends, except for overpredicting system pressure between 1800 s (30 min) and 3600 s (1 hr) and not predicting the gradual system pressure rise after the break was isolated. The overprediction of system pressure between 1800 and 3600 s was probably due to weepage in the secondary system main steam stop valve that occurred during the experiment, but was not included in the calculation.

QUICK-LOOK REPORT ON LOFT NUCLEAR EXPERIMENT L3-7

## 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 engineering 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 scaled 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) L3-7, the fourth experiment in the LOFT Small Break Test Series L3 scheduled for performance in the LOFT facility, was successfully completed on June 20, 1980. The experiments in Test Series L3 that have been completed are summarized in Table 1. A summary of the specified and measured system conditions immediately prior to LOCE L3-7 blowdown initiation is given in Table 2. Identifiable significant events for LOCE L3-7 are listed in Table 3. LOCE L3-7 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 and core configuration are shown in Appendix A. Additional details of the LOFT system are presented in Reference 1.

The primary objectives of LOCE L3-7 were to impose a break flow equal to high-pressure injection system (HPIS) flow at an intermediate pressure during the transient, to establish conditions for steam generator reflux cooling, to isolate the break and recover the plant to cold shutdown, and to analyze the data obtained to investigate associated phenomena.

Completion of LOCE L3-7 will provide additional insight into small break LOCA behavior and assist in answering the following questions:

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

2. Can the secondary coolant system effectively remove heat from the primary coolant system when the primary coolant system liquid level has dropped low enough to void the primary side of the steam generator?

3. What are the effects of turning off the HPIS injection flow later on in the transient?

4. How effectively do the major systems, such as, low-pressure injection system (LPIS), accumulator, HPIS, steam generator,

TABLE 1. LOFT TEST SERIES L3 EXPERIMENTS PERFORMED TO DATE

| Experiment | Reference | Date Completed | Power Level (MW) | Core $\Delta T(K)$ | Description |
|---|---|---|---|---|---|
| L3-0 | 2, 3 | 05-31-79 | 0 | 0 | Small break PORV, nonnuclear |
| L3-1 | 4, 5 | 11-20-79 | 50 | 35 | Small break in cold leg, break flow greater than or equal to HPIS flow.[a] |
| L3-2 | 6 | 02-06-80 | 50 | 35 | Small break in cold leg, HPIS flow greater than or equal to saturated break flow.[a] |
| L3-7 | 7 | 06-20-80 | 50 | 35 | Small break in cold leg, HPIS flow greater than or equal to saturated break flow.[a] |

a. Primary flow at 78.8 kg/s ($3.8 \times 10^6$ lbm/hr).

TABLE 2. INITIAL CONDITIONS FOR LOCE L3-2

| Parameter | Specified Value[7,a] | Measured Value |
|---|---|---|
| Primary Coolant System | | |
| Mass flow rate (kg/s) | 478.8 ± 8.8 | 481.3 ± 6.3 |
| Hot leg pressure (MPa) | 14.95 ± 0.34 | 14.90 ± 0.25 |
| Cold leg temperature (K) | 556.8 ± 2.2 | 556.4 ± 3 |
| Hot leg temperature (K) | -- | 576.1 ± 0.5 |
| Boron concentration (ppm) | As required. | 726 ± 5 |
| Reactor Vessel | | |
| Power level (MW) | 50.0 ± 2 | 48.7 ± 1 |
| Maximum linear heat generation rate (kW/m) | -- | 52.8 ± 3.7 |
| Control rod position (metres above full-in position) | 1.372 ± 0.013 | 1.373 ± 0.010 |
| Broken Loop | | |
| Cold leg fluid temperature (K) | 556.8 ± 13.9 | 557.7 ± 2.5 |
| Hot leg fluid temperature (K) | 556.8 ± 13.9 | 561.4 ± 2.5 |
| Steam Generator Secondary Side | | |
| Water level (m)[b,c] | 0.25 ± 0.05 | 0.25 ± 0.06 |
| Water temperature (K) | -- | 544.0 ± 0.2 |
| Pressure (MPa) | -- | 5.58 ± 0.012 |
| Mass flow rate (kg/s) | -- | 28.0 ± 0.4 |
| ECCS Accumulator A | | |
| Gas volume ($m^3$) | -- | 1.19 ± 0.03 |
| Liquid level (m) | 1.85 ± 0.5 | 1.85 ± 0.01 |
| Standpipe position (m)[d] | 0.79 ± 0.03 | 0.79 ± 0.03 |
| Pressure (MPa) | 4.22 ± 0.17 | 4.31 ± 0.06 |
| Temperature (K) | 305.4 ± 5.6 | 306.6 ± 0.7 |
| Boron concentration (ppm) | 3000 | 3405 ± 5 |
| Suppression Tank | | |
| Liquid level (m) | 1.27 ± 0.05 | 1.46 ± 0.03 |
| Gas volume ($m^3$) | -- | 49.8 ± 1.5 |
| Downcomer submergence (m)[e] | -- | 0.60 ± 0.06 |
| Water temperature (K)[f] | -- | 363.8 ± 2.7 |
| Pressure (gas space) (MPa)[f] | -- | 0.124 ± 0.008 |

TABLE 2. (continued)

| Parameter | Specified Value[7,a] | Measured Value |
|---|---|---|
| Pressurizer | | |
| Steam volume ($m^3$) | -- | $0.30 \pm 0.05$ |
| Water volume ($m^3$) | -- | $0.63 \pm 0.05$ |
| Water temperature (K) | -- | $615.0 \pm 0.3$ |
| Pressure (MPa) | As required to establish pressure. | $14.90 \pm 0.04$ |
| Level (m) | $1.13 \pm 0.18$ | $1.10 \pm 0.02$ |
| HPIS | | |
| Initiation pressure (MPa) | $13.16 \pm 0.19$ | $13.35 \pm 0.24$ |
| Initial flow (L/s) | $0.32 \pm 0.13$ | $0.32 \pm 0.02$ |
| LPIS | | |
| Initiation pressure (MPa) | $1.60 \pm 0.19$ | -- |

a. If no value is listed, that parameter is not specified by the Experiment Operating Specification (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.

TABLE 3. CHRONOLOGY OF EVENTS - EXPERIMENTAL DATA VERSUS PRETEST PREDICTIONS

| | Time After LOCE Initiation (s) | | |
|---|---|---|---|
| Event | LOCE L3-7 Data | RELAP5[a] Prediction | RELAP4[a] Prediction |
| 1. Reactor scrammed | 36.0 ± 0.1 | 34.0 | 45.8 |
| 2. Control rods reached bottom | 38.0 ± 0.1 | Not calculated. | 47.8 |
| 3. Primary coolant pumps tripped | 39.3 ± 0.5 | 34.0 | 47.8 |
| 4. HPIS initiated | 65.6 ± 0.1 | 123.0 | 88.0 |
| 5. Primary coolant pumps coastdown completed | 56.2 ± 0.1 | Not calculated. | 60 |
| 6. First indication in core of natural loop circulation | 60.8 ± 0.5 | Not calculated. | Not calculated. |
| 7. Secondary coolant system auxiliary feed pump started (initial steam generator fill) | 75.0 ± 3.0 | 94.0 | 112.6 |
| 8. Pressurizer emptied | 264.0 ± 7.0 | 400.0 | 359.0 |
| 9. Upper plenum fluid reached saturation temperature (end of subcooled blowdown) | 382.0 ± 6.0 | 450.0 | -- |
| 10. Intact loop hot leg voiding initiated | 1 037 ± 10 | -- | -- |
| 11. HPIS turned off | 1 805.3 ± 0.1 | 1 800.0 | -- |
| 12. Secondary coolant system auxiliary feed pumps tripped (terminated initial steam generator fill) | 1 800 ± 5 | 1 800 | |
| 13. Secondary coolant system steam bleed initiated | 3 576 ± 1 | 3 600.0 | -- |
| 14. HPIS turned on | 5 974.2 ± 0.1 | 5 400.0 | -- |
| 15. HPIS flow ≥ break flow[b] | 6 000 ± 50 | -- | -- |
| 16. Accumulator injection initiated | 6 028 ± 5 | 7 200.0 | -- |

TABLE 3. (continued)

| Event | Time After LOCE Initiation (s) LOCE L3-7 Data | RELAP5[a] Prediction | RELAP4[a] Prediction |
|---|---|---|---|
| 17. QOBV[c] isolation valve closed | 7 302 ± 0.1 | 7 200 | -- |
| 18. Primary system fluid becomes subcooled | 7 915 ± 20 | 7 500 | -- |
| 19. Primary system venting initiated | 12 047 ± 10 | -- | -- |
| 20. Purification system cooldown initiated[d] | 18 180 ± 60 | -- | -- |
| 21. Primary system venting stopped | 18 600 ± 100 | -- | -- |
| 22. LPIS injection initiated[e] | -- | -- | -- |
| 23. Experiment completed[f] | 29 500 ± 100 | -- | -- |

a. RELAP4 calculation terminated at 1800 s, RELAP5 at 1200 s.

b. Using the best currently available value for break flow, HPIS was turned off prior to exceeding break flow at approximately 1800 s. When HPIS was again turned on at approximately 6000 s, it exceeded break flow.

c. QOBV--quick-opening blowdown valve.

d. From experiment log.

e. LPIS did not initiate since primary pressure exceeded 2.8 MPa throughout the transient.

f. End of experiment is defined as $T_{system}$ = 366.5 K.

etc.. perform to prevent core damage? Do any of these systems appear not to be needed for this particular break size and/or location?

5. What kind of recovery procedures should be used in the event of a small break LOCA and, in particular, which recovery heat transfer mode is most appropriate?

6. Are there key times in the transient when operator action is required to protect the core?

7. From an analysis point of view, are there operator/equipment actions that must not occur?

8. Given a small break occurrence of unknown size and location, are there operator actions that are dependent on the break unknowns that would aid plant recovery in one case and impede plant recovery in 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?

   (c) Can the instruments be arranged in the control room in a mannner that would aid in diagnosing and following the transient?

10. Are there any additional measurements that should be provided in the control room? Are there improvements that can be made to typical commercial reactor instrumentation to monitor a small LOCA?

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

12. Are there data processing techniques and data display systems which will augment operator capabi' ies to diagnose plant status and respond to off-normal conditions.

The primary objectives of LOCE L3-7 were met by the experiment. The 12 questions are general and require the results of more than one or two experiments to answer. However, data from LOCE L3-7 will contribute to answering the general questions.

This report presents a preliminary examination of plant performance (Section 2), followed by a summary of the results from LOFT LOCE L3-7 (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 L3-7 data with LOCE L3-7 pretest calculations[8] made by EG&G Idaho, Inc., using the RELAP5[9,a] computer code.

In addition, the data are compared with pretest calculations for the first 1800 s made by EG&G Idaho, Inc., for LOCE L3-2 using the RELAP4[10,b]

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a. 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.

b. 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 stea- valve control logic into the code, was RLP4G92LFT04, Idaho National E neering Laboratory Configuration Control Number H01168IB.

computer code and with pretest calculations for the first 3600 s made by Los Alamos Scientific Laboratory (LASL)[11] using the TRAC-P1A[12] 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 L3-7. Data plots showing results of the evaluation are provided in Figures 1 through 20 in Section 5.

## 2.1 Initial Experimental Conditions

A summary of the specified and measured system conditions immediately prior to LOCE L3-7 blowdown initiation is given in Table 2. The measured average initial temperature of the cold leg primary coolant was 556.4 ± 3 K. The range of cladding temperatures was 562 ± 3 to 613.5 ± 3 K. The initial mass flow rate in the primary coolant loop was 478.8 ± 8.8 kg/s, and pressurizer pressure was 14.95 ± 0.34 MPa. The initial power level of 50.0 ± 1.0 MW yielded a maximum linear heat generation rate (MLHGR) of 52.8 ± 3.7 kW/m. All of the initial conditions were within specified limits, except suppression tank liquid level, which did not affect meeting experiment objectives.

## 2.2 Chronology of Events

Identifiable significant events for LOCE L3-7 are listed in Table 3, where their times of occurrence are compared with the times predicted by the RELAP4 and RELAP5 calculations. At 36 s into the transient, reactor scram was initiated by a low pressure signal in the primary system hot leg (Figure 2). After the reactor scrammed, the intact loop primary coolant pumps were tripped and started to coast down.

Just before the pump coastdown was completed, the 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 61 s, driven by the residual stored thermal and fission product energies in the core, and sustained by heat transfer in the steam generator.

The pressurizer emptied at 264 s, followed by fluid saturation in the upper plenum at 382 s (Figure 1). Fluid saturation in the upper plenum increased the velocity of the fluid exiting the core and in the intact loop hot leg (Figure 3). The intact loop hot leg began voiding at 300 s (Figure 11). The HPIS flow was shut off at 1805 s (Figure 10). This time corresponds with significant break uncovery when the broken loop cold leg density is less than about 0.4 (Figure 20).

The temperature difference between the primary and secondary systems decreased to about 1.5 K by 2500 s (Figure 7). The small temperature difference is indicative of the high heat transfer rate due to condensation in the steam generator primary tubes. Natural circulation continued through this period (Figure 3), and until after purification system cooldown was initiated at 18 180 s (from calculations described in Section 3.1.1).

At 3576 s (Figure 1), operator-initiated bleeding of secondary steam, as planned, increased the depressurization rates of both the primary and secondary systems (Figure 8). At 5974 s, the net depletion of system mass inventory stopped when HPIS flow was turned on and equaled or exceeded break flow, as confirmed by the decreased intact loop hot leg voiding (Figure 4).

The core was covered throughout the experiment and remained cool as confirmed by the fuel cladding temperature and upper plenum fluid temperatures (Figures 12 and 13).

Accumulator injection pressure was reached at 6028 s, as the steam bleeding operation continued to be effective in reducing system pressure.

At 7305 s, the QOBV isolation valve was closed and HPIS flow continued. Natural circulation velocity exhibited a significant decrease at this time (Figure 4) due to transition to single-phase flow. Decay heat removal continued after 7800 s as is confirmed by decreasing upper plenum fluid temperature (Figure 15).

At 7915 s, the fluid in the reactor vessel became subcooled (Figure 15). At 12 000 s, primary system steam bleeding through the power-operated relief valve (PORV) was initiated to reduce system pressure (Figure 1), and was effective. Purification system cooldown was initiated at 18 180 s, and cold shutdown temperature reached 366 K at 29 350 s, ending the experiment. The LPIS was not used because the initiation pressure was not reached.

### 2.3 Instrumentation Performance

The instrumentation used for LOCE L3-7 was essentially the same instrumentation used for LOCE L3-2, with a few additions. These include momentum flux and velocity in the intact loop hot leg, steam dome pressure, and liquid levels in the suppression tank and in Accumulator A.

Of the 633 instruments operable prior to and recorded for LOCE L3-7, it is estimated that 97% performed satisfactorily. The pulsed neutron activation (PNA) flowmeter provided 27 data points during the experiment (Figure 3) which agree closely with the data from the flow turbine in the intact loop hot leg.

## 3. EXPERIMENTAL RESULTS FROM LOCE L3-7

The preliminary analysis presented in this section is based on data processed and available within the first week following the conduct of LOCE L3-7 and, in certain instances, reflects the current lack of confirmatory data or analysis. Analysis of the LOCE L3-7 data will continue in order to further support the preliminary results and conclusions.

### 3.1 Discussion of Phenomena and Comparison with Predictions

Conditions were established in LOCE L3-7 for natural circulation in the primary loop. An objective of LOCE L3-7 was to establish conditions for reflux flow. The conditions thought to be necessary for reflux flow were established; however, the flow mode was not observed. These phenomena, comparison with computer code predictions, and phenomena which were unanticipated are discussed in this section.

#### 3.1.1 Natural Circulation in the Primary Loop

Measurable natural circulation in the primary loop continued from 61 s into the transient until after the purification system started at 18 180 s. Single-phase natural circulation was fully established within about 35 s starting at 61 s (Figure 14), where the temperature rise indicates the fluid slowing to the natural circulation velocity. Single-phase natural circulation continued until about 375 s, at which time upper plenum temperature reached saturation (Figure 14) and two-phase natural circulation was initiated. The transition at this time from single-phase to two-phase natural circulation is characterized by a smooth increase in core exit velocity (Figure 3).

Single-phase natural circulation was reestablished after the break was isolated at 7200 s, as indicated by core and steam generator differential temperature (Figures 5 and 6). By that time, the fluid in the system had become subcooled. Fluid velocities were smaller than during two-phase natural circulation and could not be detected by core outlet or intact loop turbine meters, but are shown by the PNA (Figure 3). Core velocity, calculated from decay power and temperature differential, was approximately 0.05 m/s from 7800 s to 15 000 s.

Condensation heat transfer in the steam generator occurred during two-phase natural circulation during the period from 1200 to 7200 s, as indicated by the small primary-to-secondary temperature differential (Figure 9). Forced- and free-convection heat transfer occurred during single-phase and two-phase natural circulation.

The combination of natural loop circulation and steam generator heat transfer was sufficient to remove decay heat throughout the experiment. The fuel rod temperature increase during the brief period from 7500 to 7800 s (Figure 13) is due to establishing the core temperature differential necessary to drive single-phase natural circulation.

### 3.1.2 Reflux Flow Mode

The reflux flow mode did not occur, although condensation heat transfer did occur in the steam generator, and the intact loop partially voided during the period of condensation. Reflux is characterized by counter-current flow of condensed liquid in the steam generator to the hot leg. Both the lower intact loop hot leg turbine meter and the PNA velocity measurement (Figure 3) indicated positive flow rather than counter-current flow. Thus, if a condensate film falls down the steam generator tubes to the inlet plenum, it mixes with the two-phase fluid which has a dominant positive velocity through the hot leg and steam generator.

Another characteristic of reflux flow is higher quality fluid exiting the steam generator than entering, due to condensed liquid falling back on the upside to the hot leg and stratifying. The high quality is detected by a measured outlet plenum fluid temperature that is larger than the inlet temperature. The higher measured outlet temperature is due to radiation from the hot plenum walls through optically thin steam to the thermocouple. The outlet temperature was less than or equal to the inlet temperature during this experiment (Figure 6).

The major reason why the reflux flow mode did not occur is probably that the intact loop hot leg did not void sufficiently.

### 3.1.3 Unanticipated Phenomena and/or Events

Superheated steam and noncondensible gas in the system and possibly pressurizer hot wall heat transfer caused the system pressure to increase to a higher level than calculated (Figure 4) after the break was isolated at 7305 s. Also, the upper plenum and fuel cladding temperatures (Figure 13) increased during the period from 7305 to about 7800 s, while the lower plenum temperature continued to decrease (Figure 15). After 7800 s, when single-phase natural circulation was established, the hot leg temperature continued to decrease at about the same rate as before the break was isolated. The pressure and temperature increases are indicative of decreased steam generator heat transfer due to decreased flow rate (Figure 3) and cessation of condensation heat transfer, as indicated by an increased temperature difference across the steam generator (Figure 9).

After 780[illegible] [illegible]m pressure continued to increase, although the steam generator was [illegible]ective in removing decay heat. The increased pressure due to flow from the HPIS compressing the noncondensibles or steam in the vessel head caused the upper plenum liquid to become subcooled at 7915 s.

### 3.1.4 Comparison With Computer Code Predictions

Computer code predictions[8] of LOCE L3-7 were made by EG&G Idaho, Inc., using RELAP5[9] and RELAP4.[10] In addition, computer code predictions[11] were made by Los Alamos Scientific Laboratory using TRAC-PIA.[12] The RELAP4 prediction was originally made for LOCE L3-2.[6] Since the initial and boundary conditions for LOCE L3-7 and LOCE L3-2 were identical during the first 1800 s of the transient. These code predictions are compared with LOCE L3-7 during this initial period.

Comparisons of code predictions with data for break mass flow rate, pressure, intact loop flow rate, and pressurizer level, are shown in Figures 16 through 19. In general, the comparisons show good agreement between the predictions and the measured data.

The RELAP5 system pressure was accurately predicted, except for system pressure during the period from 1200 to 3600 s, and after 7200 s. The RELAP5 calculations predicted higher system pressure than is shown by the measured data during the period from 1200 to 3600 s with oscillations due to secondary steam releases at the high-pressure relief setpoint. The actual pressure was lower and the releases did not occur because of steam control valve weepage, and possibly also because of reflood assist bypass (RAB) bypass leakage, and lower calculated break mass flow than in the measured data during this period.

The RELAP5 calculations predicted a lower mass flow rate than was shown by the measured data during the initial period before the HPIS flow was terminated at 1800 s (Figure 16). The mass flow data past 1800 s is not accurate and no direct comparison can be made with the calculations.

As expected, steam weeped from the secondary side of the steam generator, presumably at either the main steam valve or its bypass. The amount of weepage calculated to occur between 1800 and 3500 s was 0.2 kg/s, based on the change in steam generator liquid level.

The RELAP5 calculations predicted a leveling of of system pressure after the break was isolated at 7200 s, until it showed a rapid increase when the system went solid. The actual pressure gradually increased, because HPIS flow compressed the remaining superheated steam and noncondensible gas in the vessel head and pressurizer, as explained in Section 3.1.3. The code probably did not predict repressurization due to not modeling noncondensible gas in the system and pressurizer wall heat transfer.

The TRAC-P1A calculations were carried out for the first 3600 s of the transient. A preliminary comparison of the calculations with the measured data shows generally good agreement, except that the system pressure is predicted to increase after the HPIS flow was stopped at 1800 s (Figure 19).

Because the computer codes predicted the break mass flow rate well for the initial 1200 s of the transient, and because the flow rate was significantly lower than in LOCE L3-2, with the same initial and boundary conditions during this period, it is concluded that LOCE L3-2 exhibited a leak in the valve in the broken loop cold leg warmup line. The leak was repaired before LOCE L3-7.

## 3.2 Experiment Objectives

Results from LOCE L3-7 that address the questions listed in Section 1 are discussed in this section. The first four questions are experiment specific, and LOCE L3-7 provided sufficient information to provide an answer to these questions. The remaining questions are general questions. General questions cannot be answered completely by data from a single experiment. However, the information derived from LOCE L3-7, based on a preliminary assessment of the data, is presented.

The answers to Questions 9b, 9c, 10, 11, and 12 from Section 1 are beyond the scope of this document, and they are not addressed. Questions 1 through 3 are addressed in the previous section of the report. Questions 6, 7, 8, and 9a are discussed in Reference 6. Discussions of Questions 4 and 5 follow:

1. Question 4--How effectively do the major systems, such as LPIS, accumulator, HPIS, steam generator, etc., perform to prevent core damage? Do any of these systems appear not to be needed for this particular break size and/or location?

   Although the HPIS was turned off between 1800 to 5913 s, it was effective in maintaining the system mass inventory significantly above the core during the two time periods that it was turned on. The accumulator had little effect on vessel refill and system pressurization. The LPIS was not initiated because system pressure did not lower to initiation pressure.

2. Question 5--What kind of recovery procedures should be used in the event of a small break LOCA and, in particular, which recovery heat transfer mode is most appropriate?

   In this experiment, the break was isolated while the system was saturated. The isolation caused the system to become subcooled, the pressure to increase, and the fuel rods to continue to cool by natural circulation driven convection heat transfer in the steam generator. Operator intervention was required to bleed mass in the primary system to reduce pressure so that the purification system could be initiated to bring the system to a cold shutdown condition.

## 4. CONCLUSIONS

The conduct of LOFT LOCE L3-7 and the experimental data acquired concerning integral system phenomena associated with a loss of coolant are considered to have met the objectives as defined by the Experiment Operating Specification[7] 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. Natural circulation was initiated within 61 s after the break occurred. Natural circulation was initially single phase and transitioned to two phase at about 375 s. After the break was closed (7200 s), the system became subcooled (79 s) and returned to single-phase natural circulation. Natural circulation continued until after the purification system was started at 18 180 s.

3. The steam generator was an effective heat sink throughout the experiment, both when the break assisted in removing energy from the system and when the break was isolated.

4. Convection heat transfer occurred in the steam generator during single-phase natural convection flow. Condensation in the steam generator during two-phase natural circulation occurred during the period from 1200 to 7200 s and provided a high heat transfer rate.

5. The reflux flow mode was not measured although condensation occurred in the steam generator and the intact loop hot leg partially voided.

6. HPIS flow was approximately 70% of break flow until the HPIS was turned off at 1800 s. It equaled or exceeded break flow after it was turned on at 5974 s.

7. Single-phase natural loop circulation was reestablished when the break was isolated at 7200 s. The loop velocity exhibited a significant decrease at that time, but did not cease. Single-phase natural circulation velocity is characteristically lower than two-phase natural circulation.

8. System recovery continued after the QOBV isolation valve was closed, although the primary system pressure increased. Steam generator heat transfer was the sole means of effective decay heat removal after that time until primary system steam bleeding was initiated at 12 000 s.

9. Computer calculations predicted the dominant phenomena, in the proper time sequence, except for (a) system depressurization during the period from 1200 to 3600 s due to expected steam control valve weepage and possibly to RAB bypass flow and higher break mass flow than calculated and (b) gradual system repressurization after the break was isolated, probably due to superheated steam and noncondensible gases in the system.

10. The high break mass flow rate exhibited in LOCE L3-2 was due to a valve leak in the cold leg broken loop warmup line. This is confirmed by the significantly longer break mass flow rate in LOCE L3-7 during the initial 1800 s. The initial and boundary conditions during the first 1800 s after break initiation were identical in the two experiments. The valve was repaired prior to LOCE L3-7.

# 5. DATA PRESENTATION

This section presents selected, preliminary data from LOCE L3-7. LOCE L3-2 data are overlayed with data from LOCE L3-7 pretest calculations using the RELAP4, RELAP5, and TRAC-P1A computer codes. A listing of the data plots is presented in Table 4. Table 5 gives the nomenclature system used in instrumentation identification. A complete list of the LOFT instrumentation and data acquisition requirements for LOCE L3-7 is given in Reference 7.

The maximum ($2\sigma$) uncertainties in the report 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 |
| 6. | Differential temperature (TE-SG-1)-(TE-SG-2) | - | $\pm 0.5$ K. |

TABLE 4. LIST OF DATA PLOTS

TABLE 4. (continued)

TABLE 5. 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:

| | | | |
|---|---|---|---|
| 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 |

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.

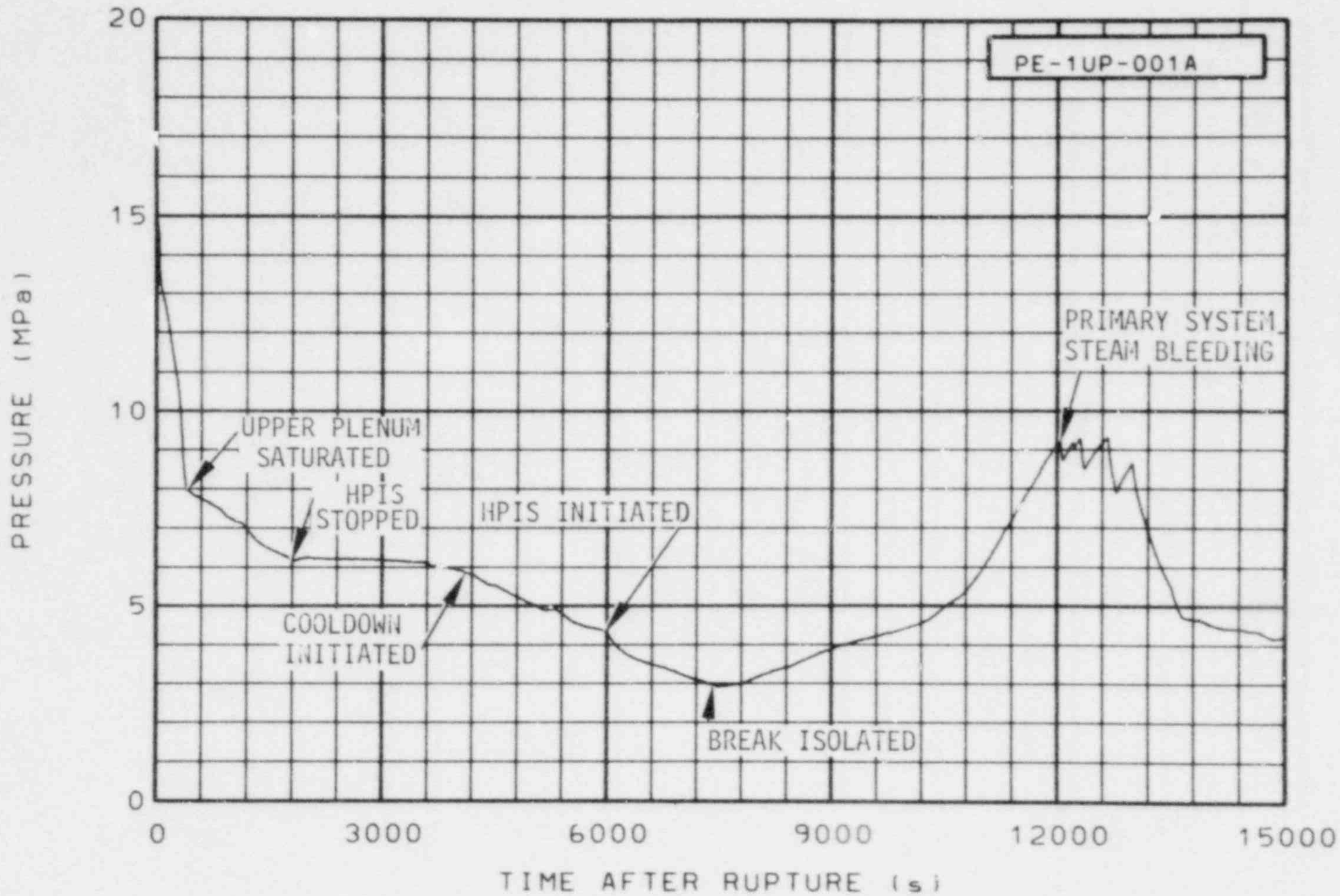


Figure 1. Pressure in reactor vessel upper plenum.

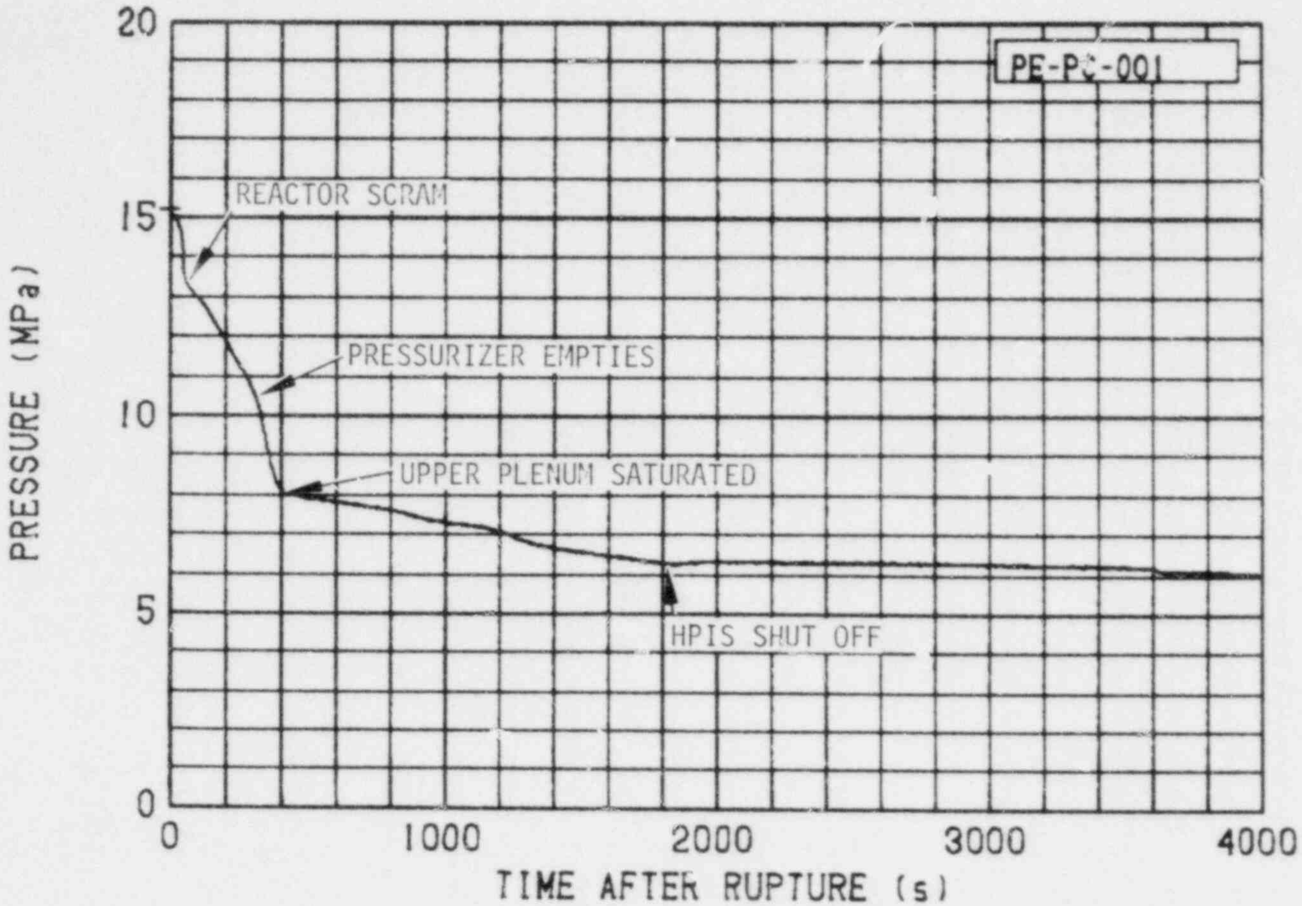


Figure 2. Pressure in primary system intact loop from 0 to 4000 s.

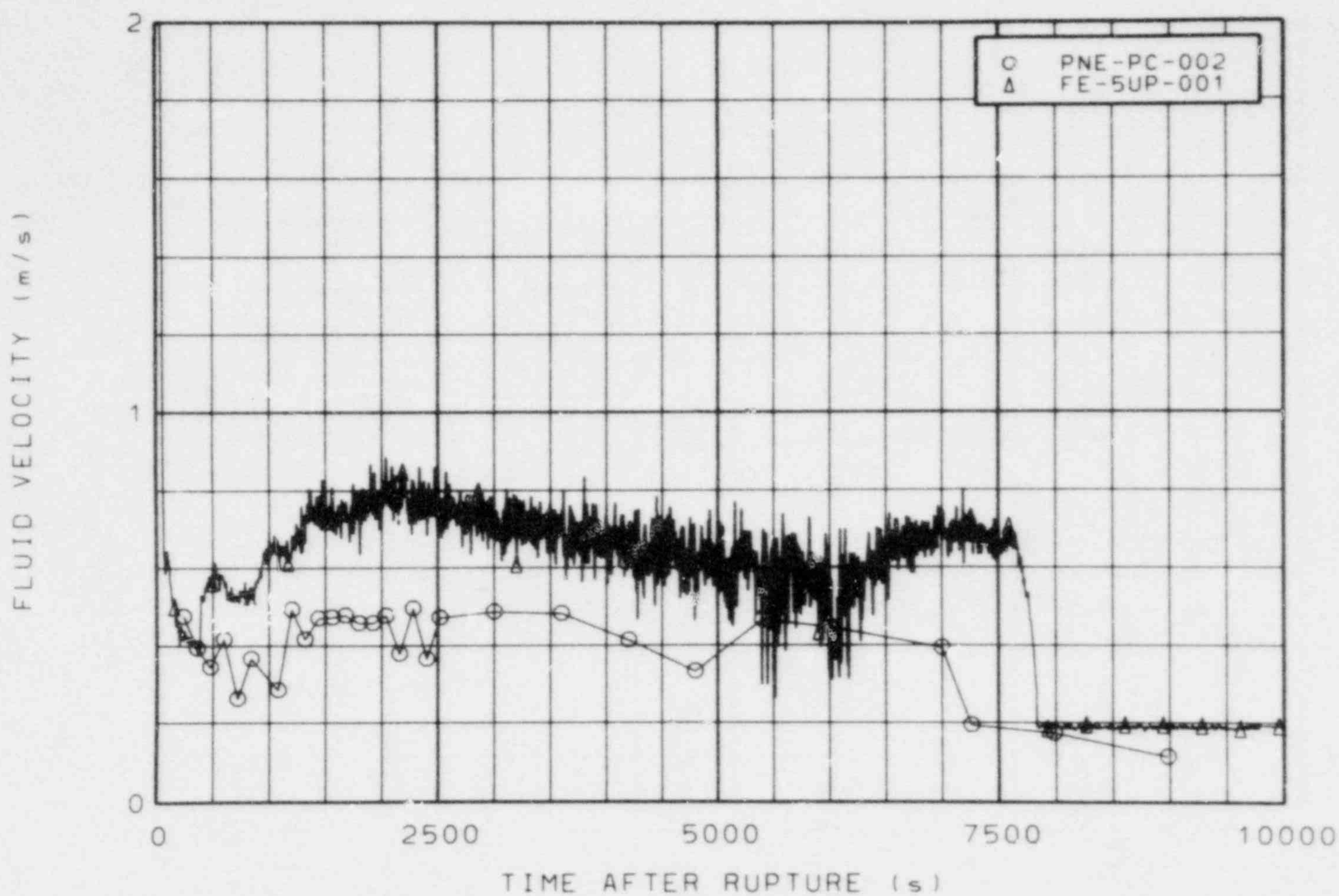


Figure 3. Comparison of fluid velocity above center fuel module and in the intact loop hot leg.

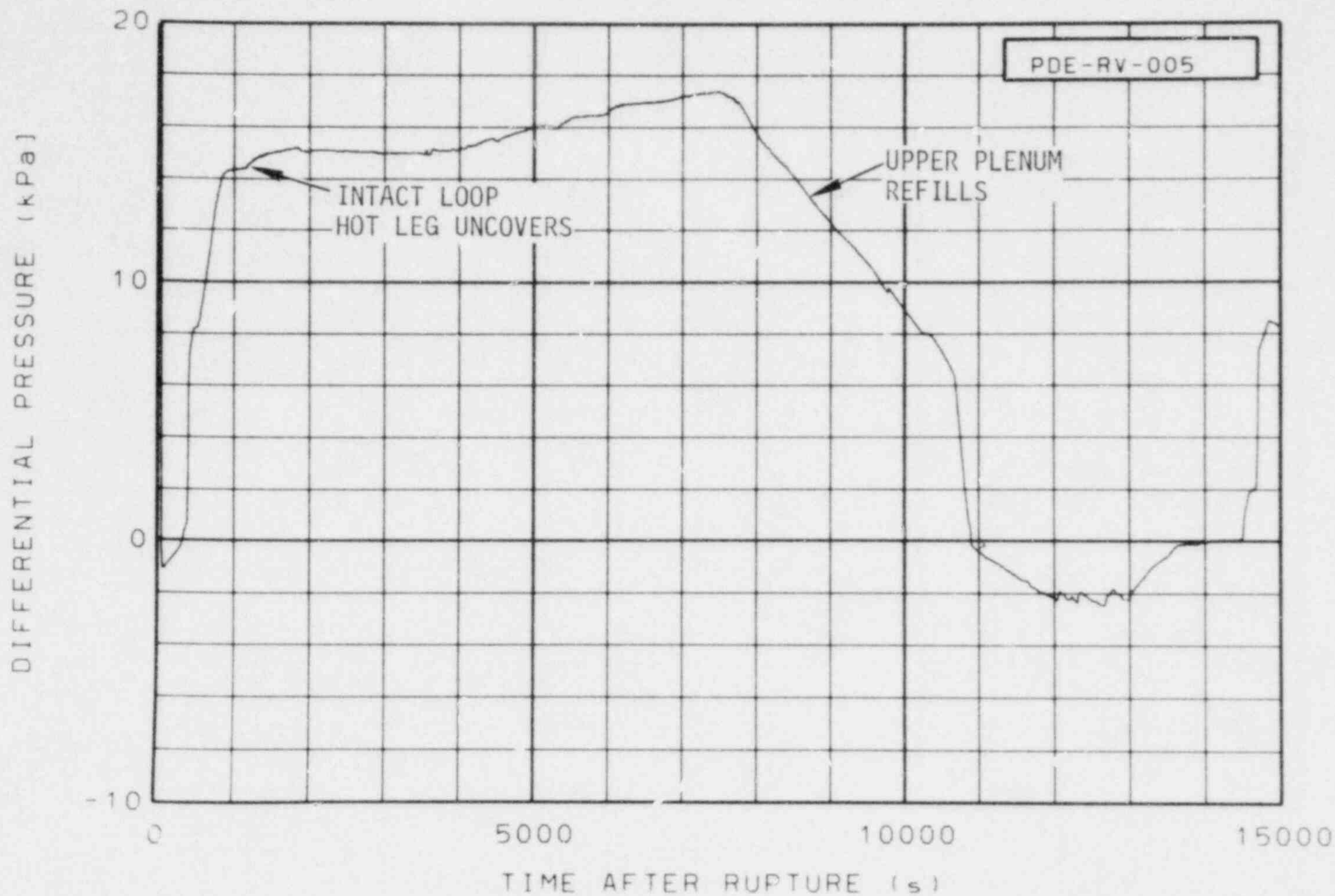


Figure 4. Pressure differential, intact loop hot leg to upper plenum.

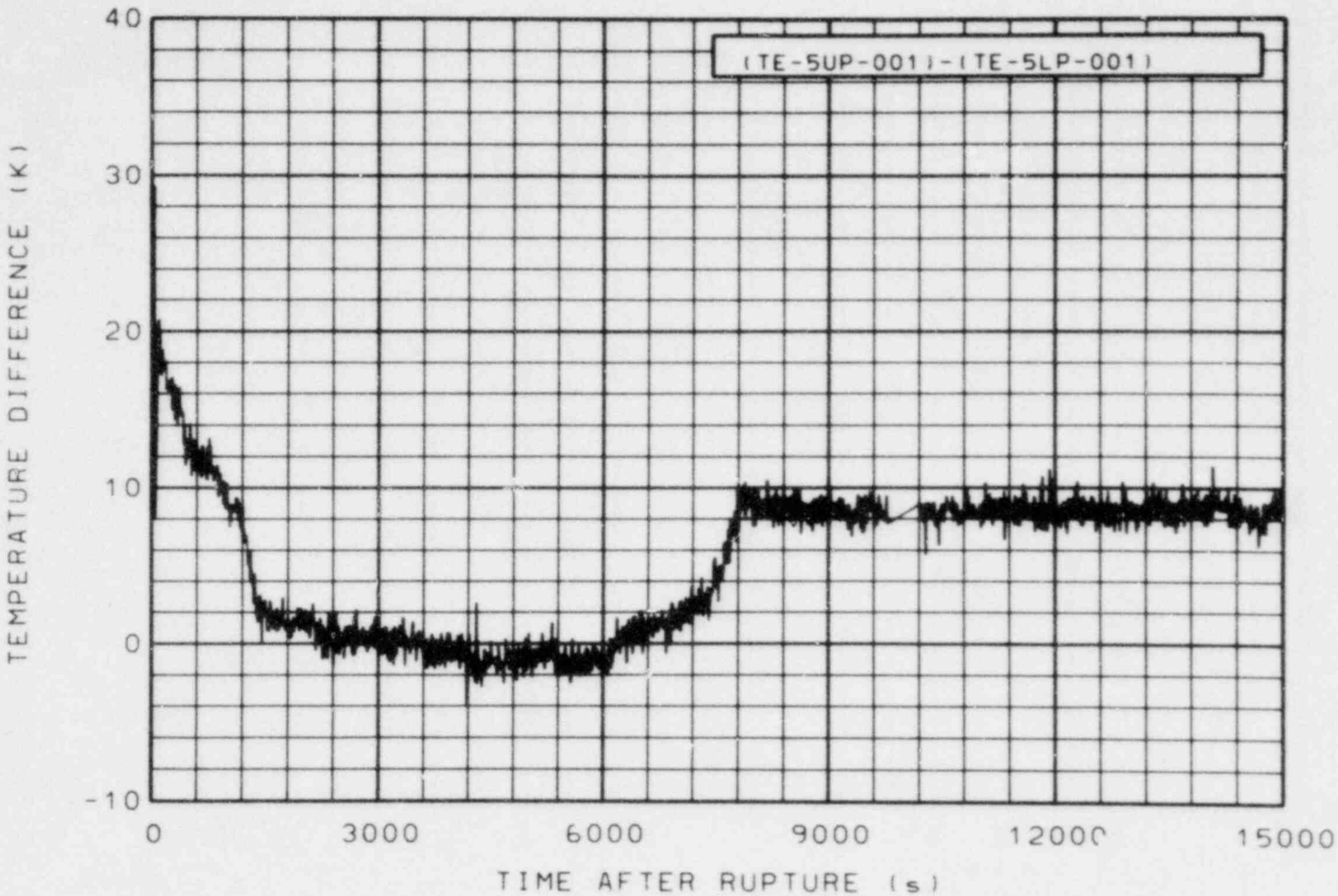


Figure 5. Fluid temperature difference across the center fuel module.

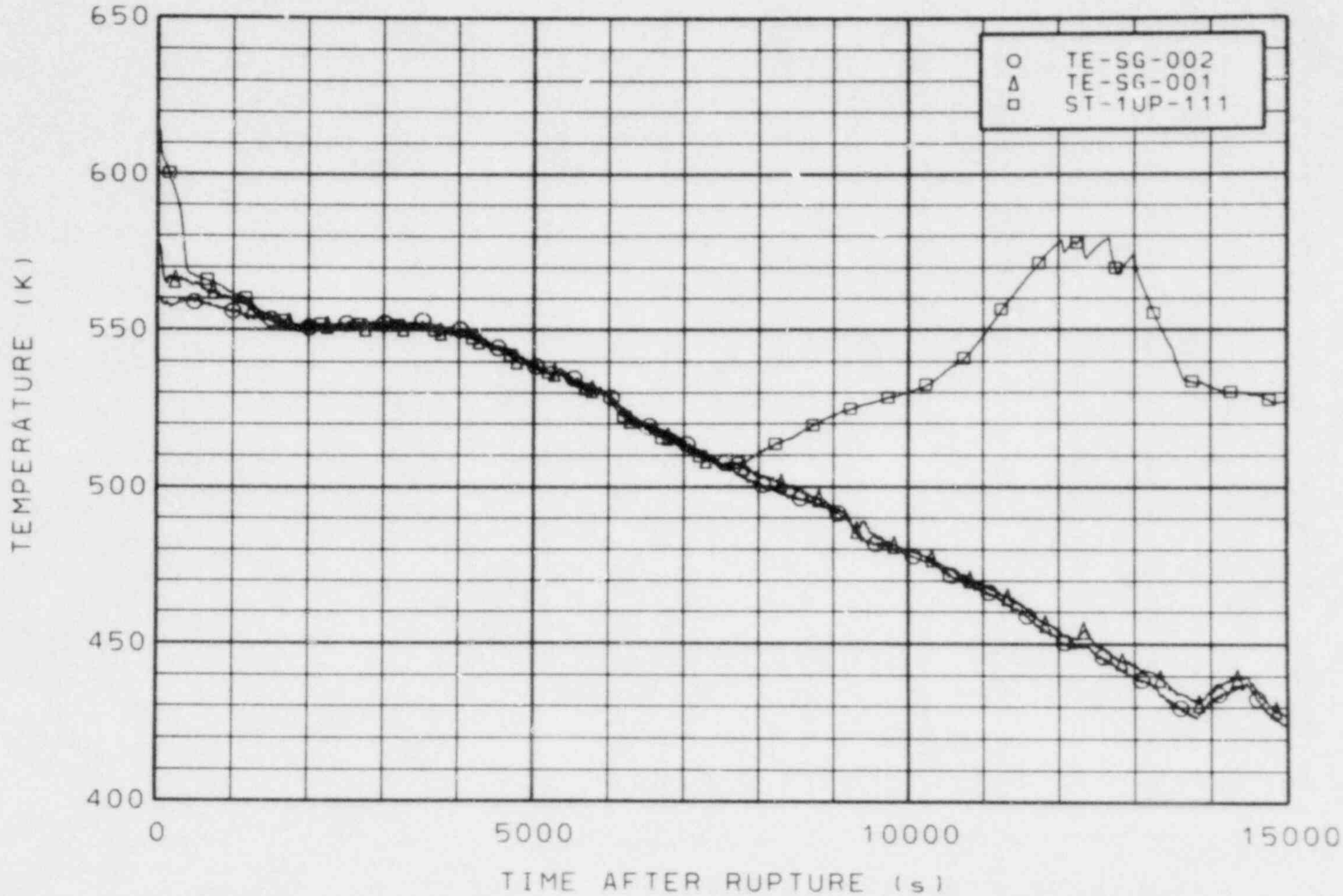


Figure 6. Comparison of steam generator primary inlet temperature, outlet temperature, and saturation temperature.

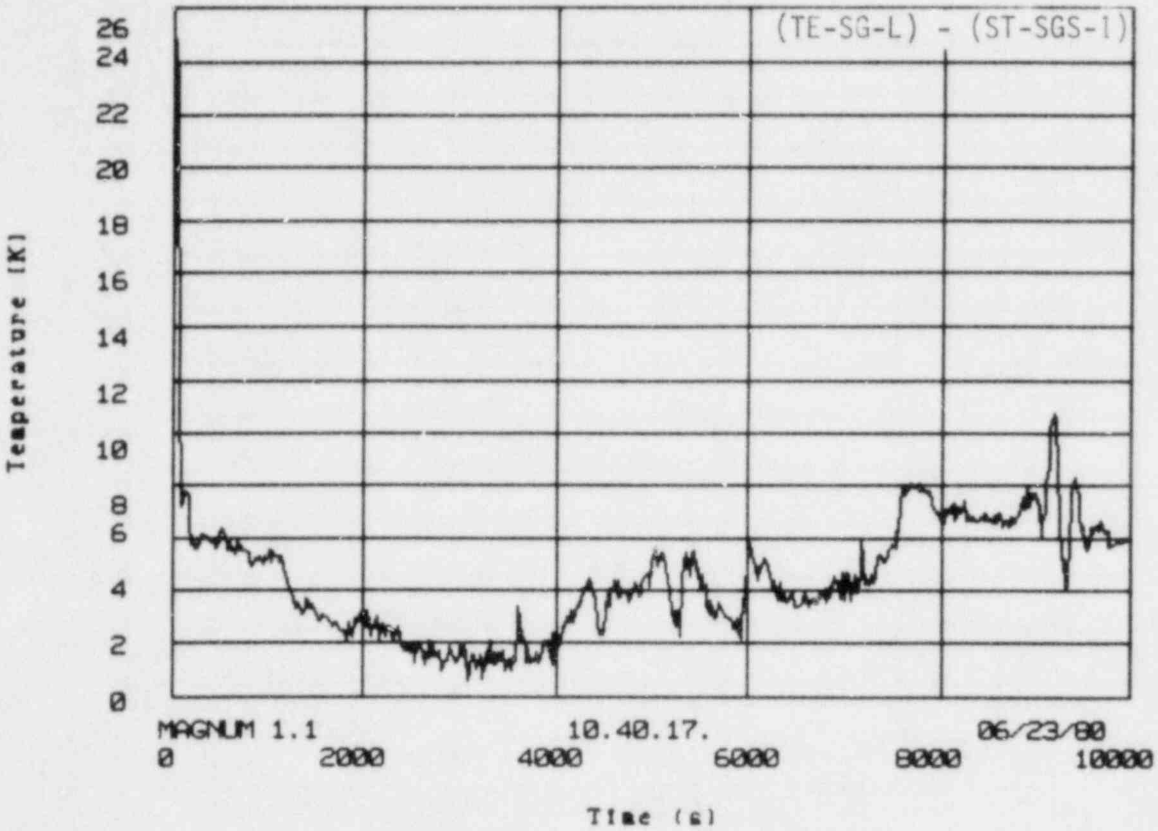


Figure 7. Fluid temperature difference in the steam generator between the primary system inlet plenum and the secondary system downcomer.

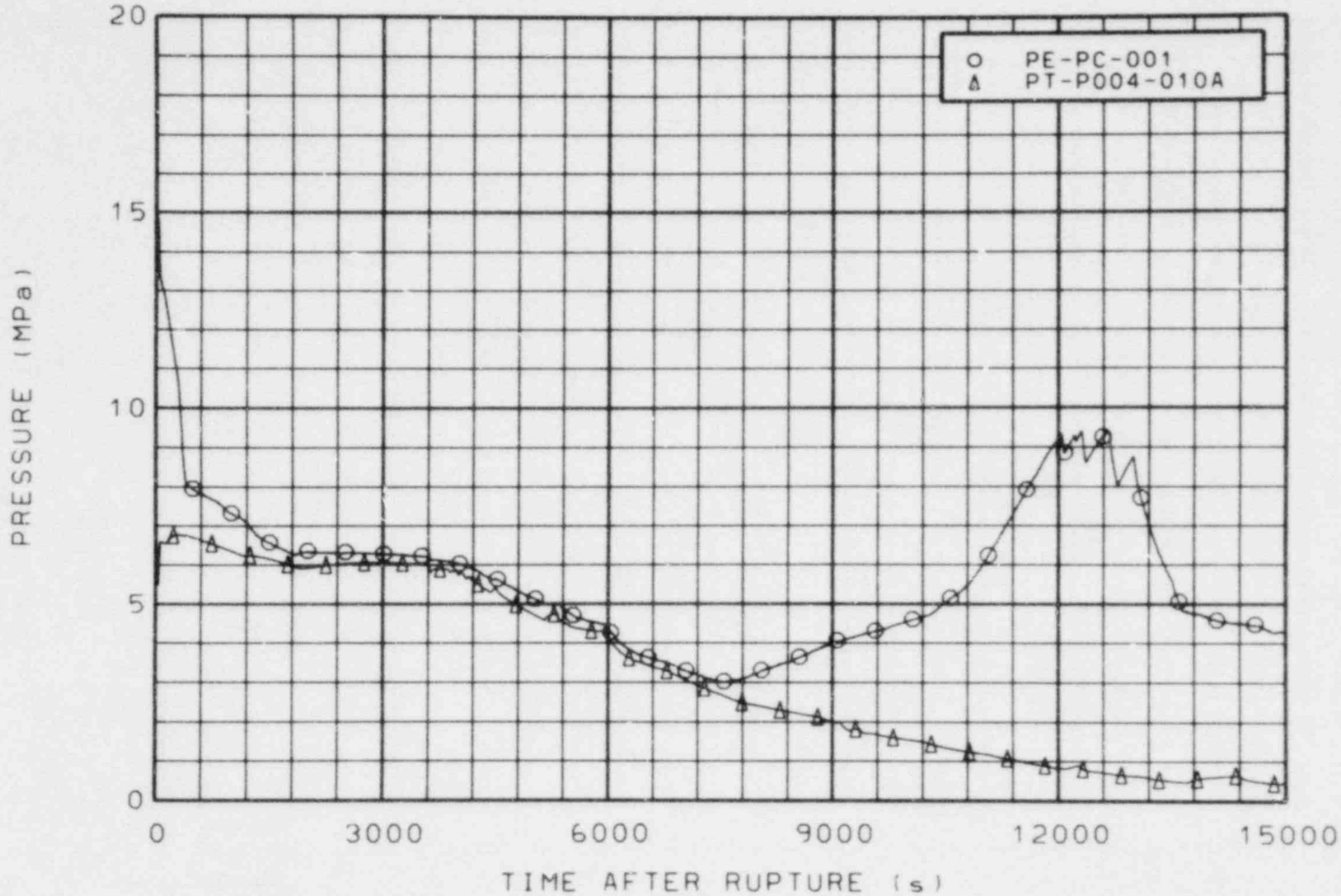


Figure 8. Comparison of primary and secondary system pressures.

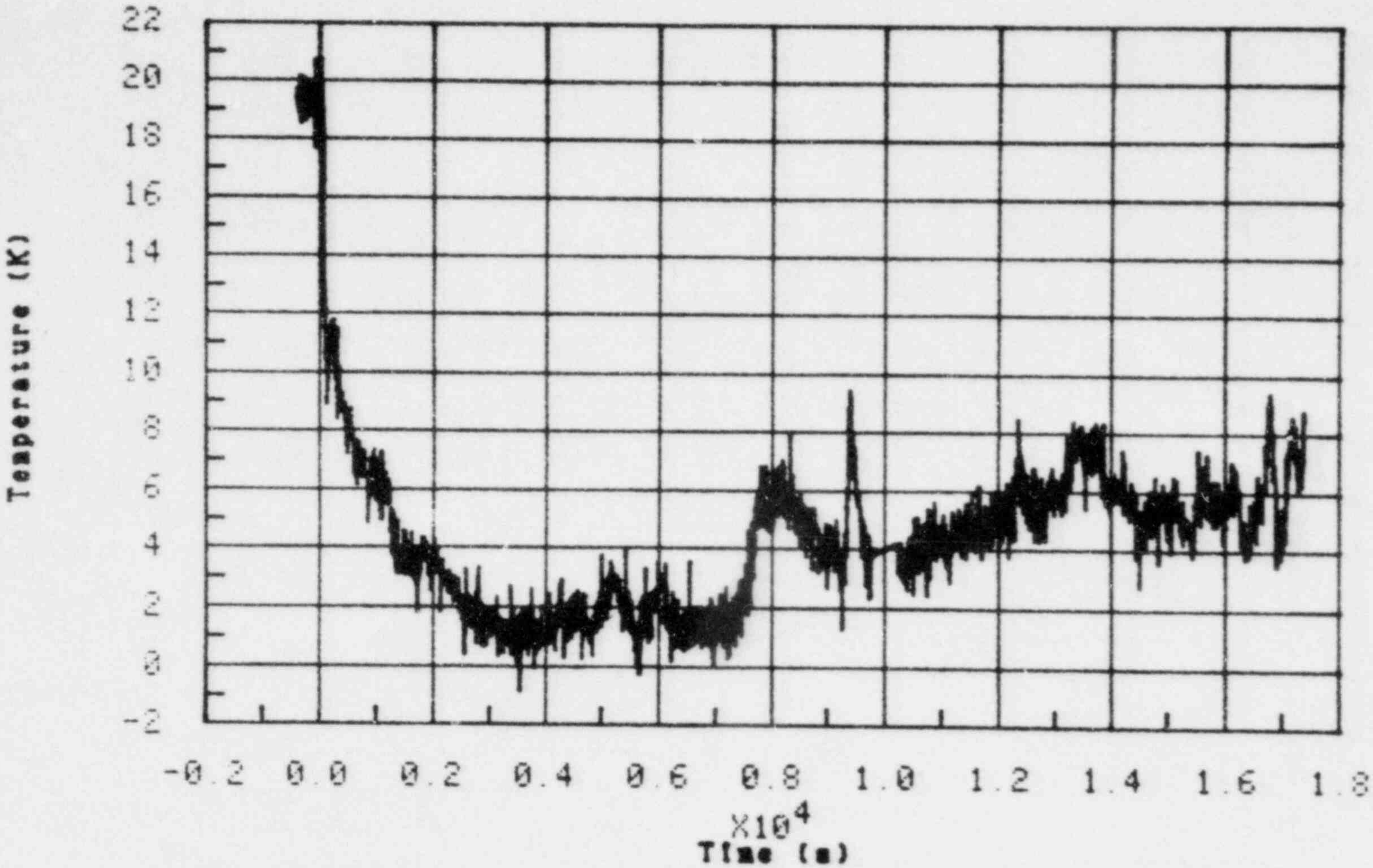


Figure 9. Temperature difference, steam generator primary minus steam generator secondary.

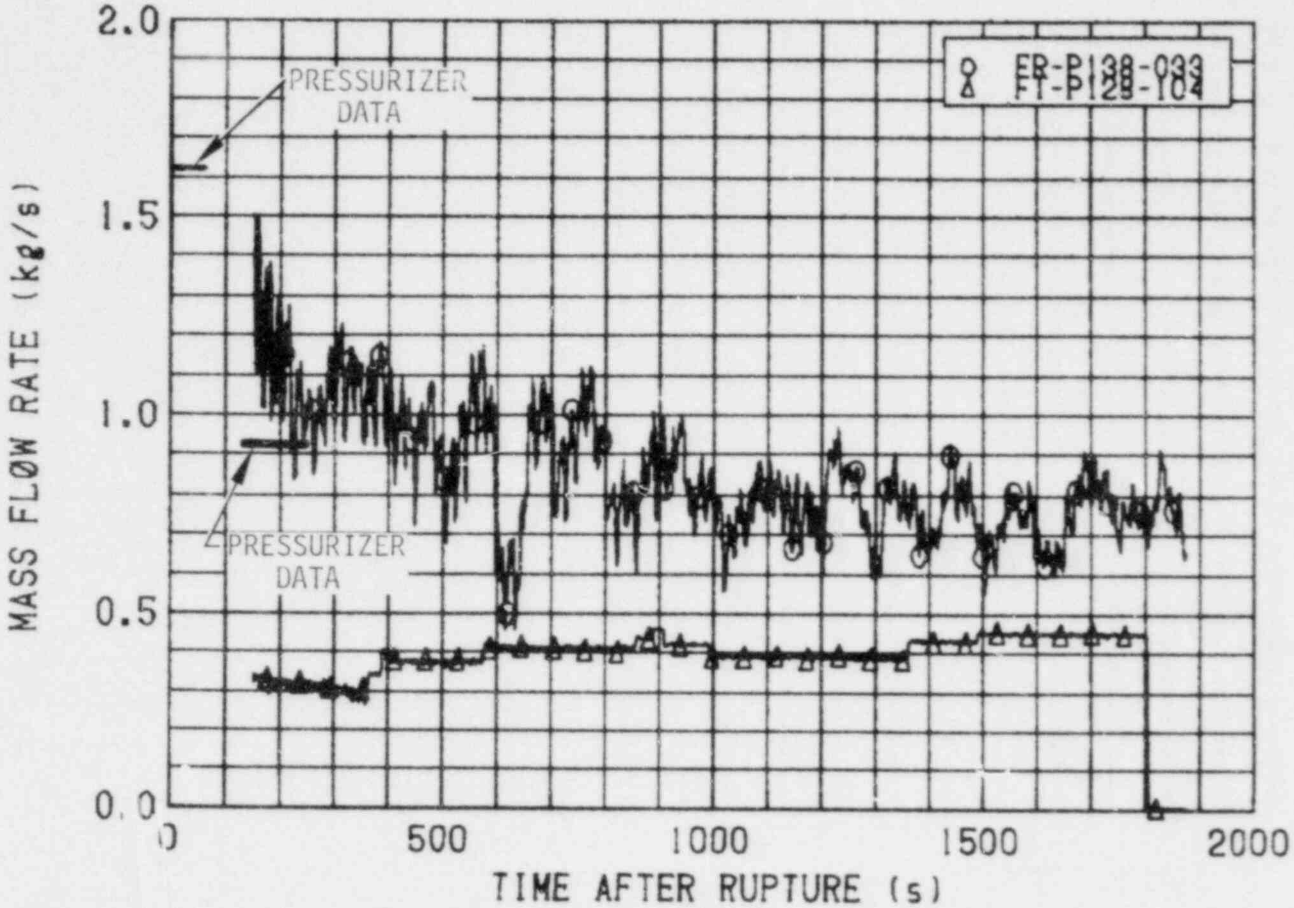


Figure 10. Comparison of break flow and ECCS flow.

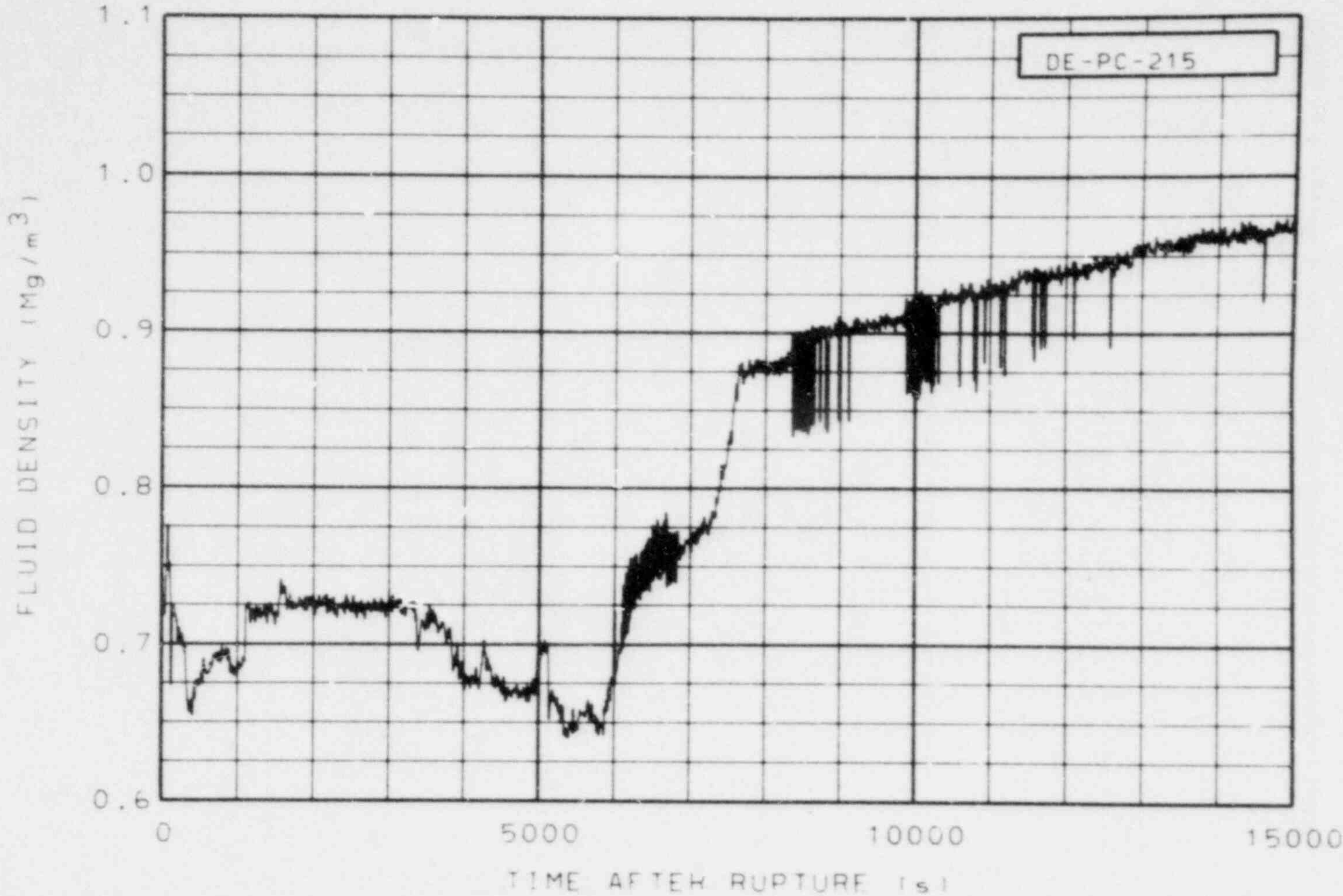


Figure 11. Fluid density in the intact loop hot leg.

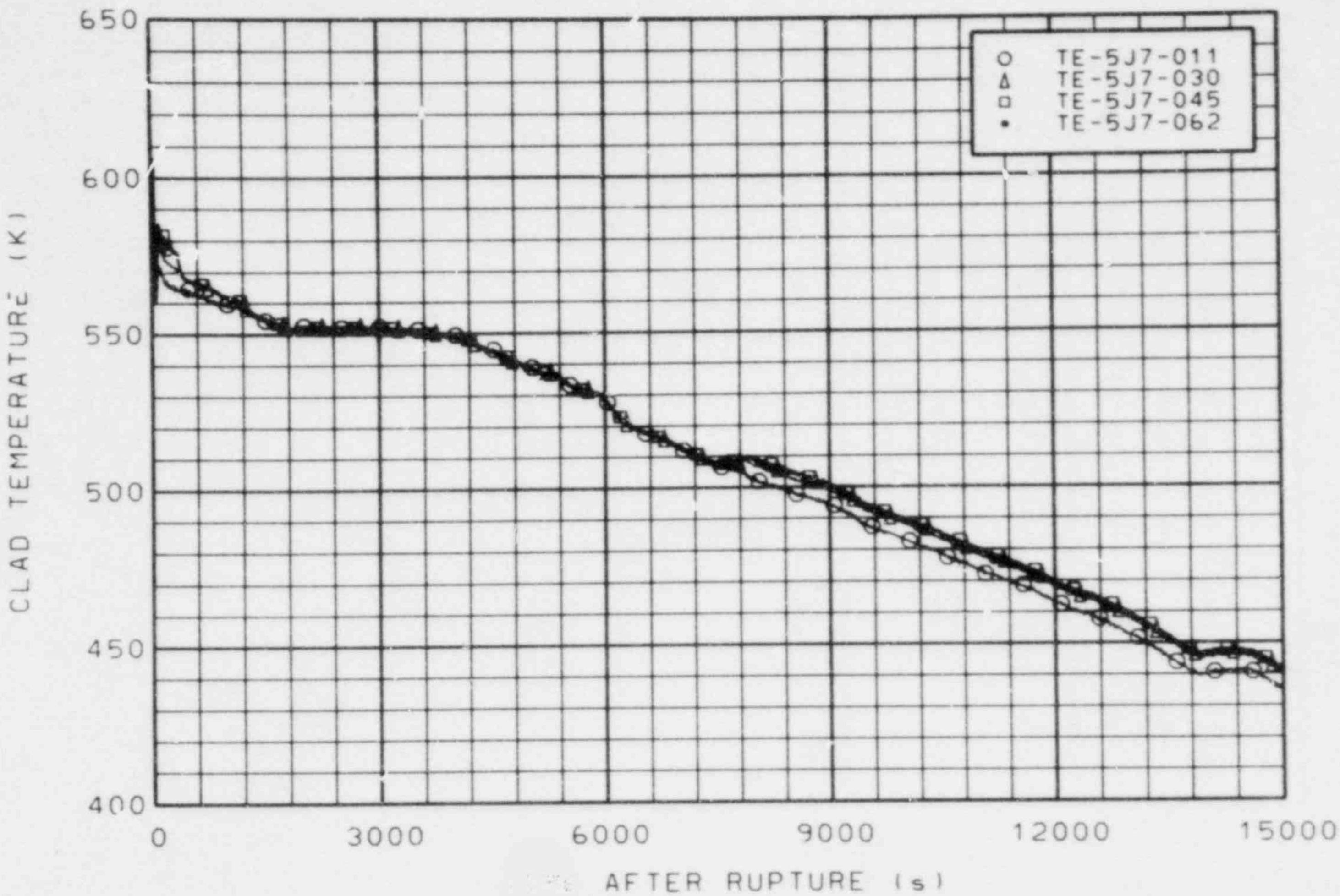


Figure 12. Fuel cladding thermocouple temperatures in the center fuel module.

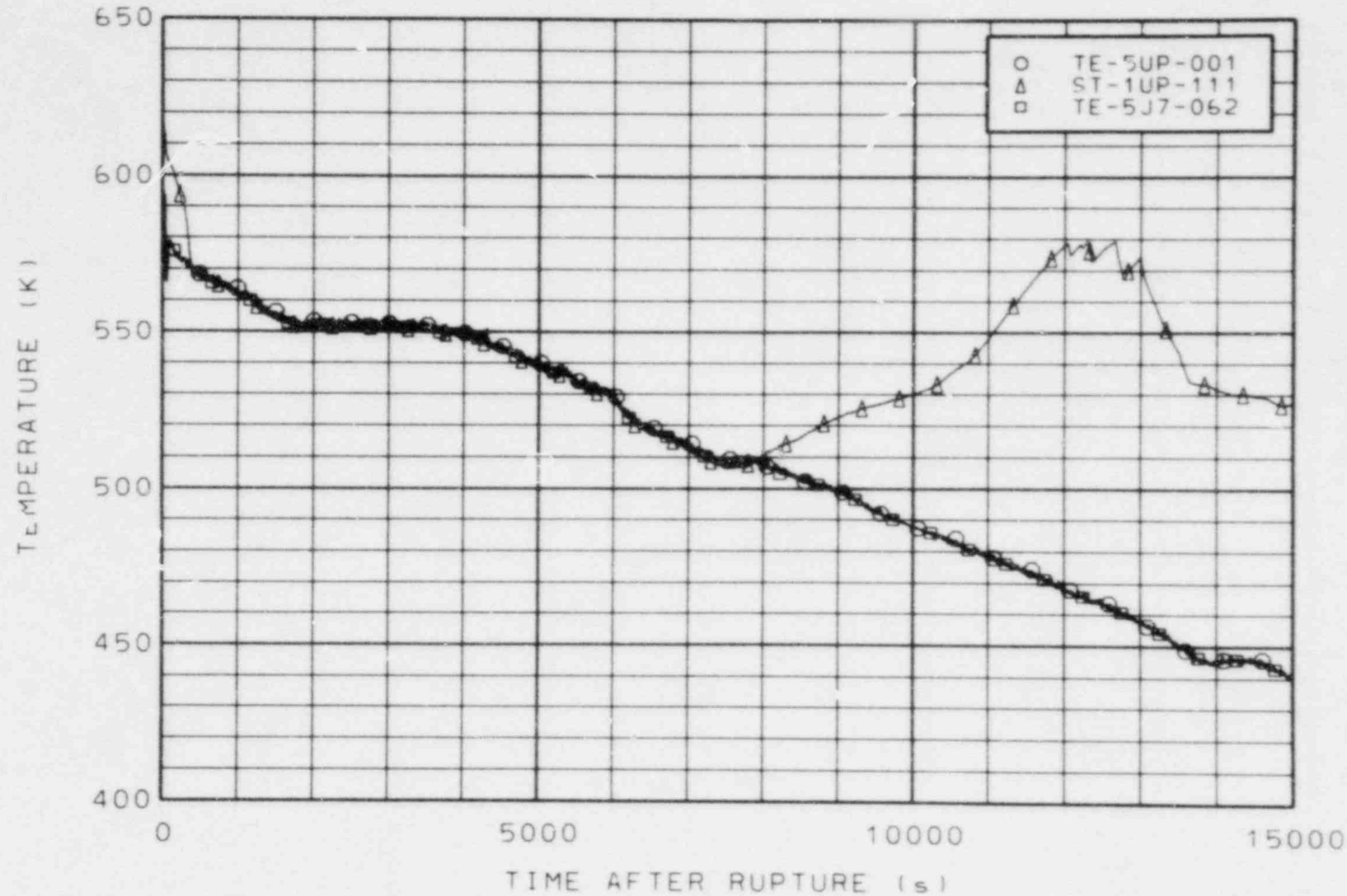


Figure 13. Comparison of upper plenum fluid, fuel cladding, and fluid saturation temperatures.

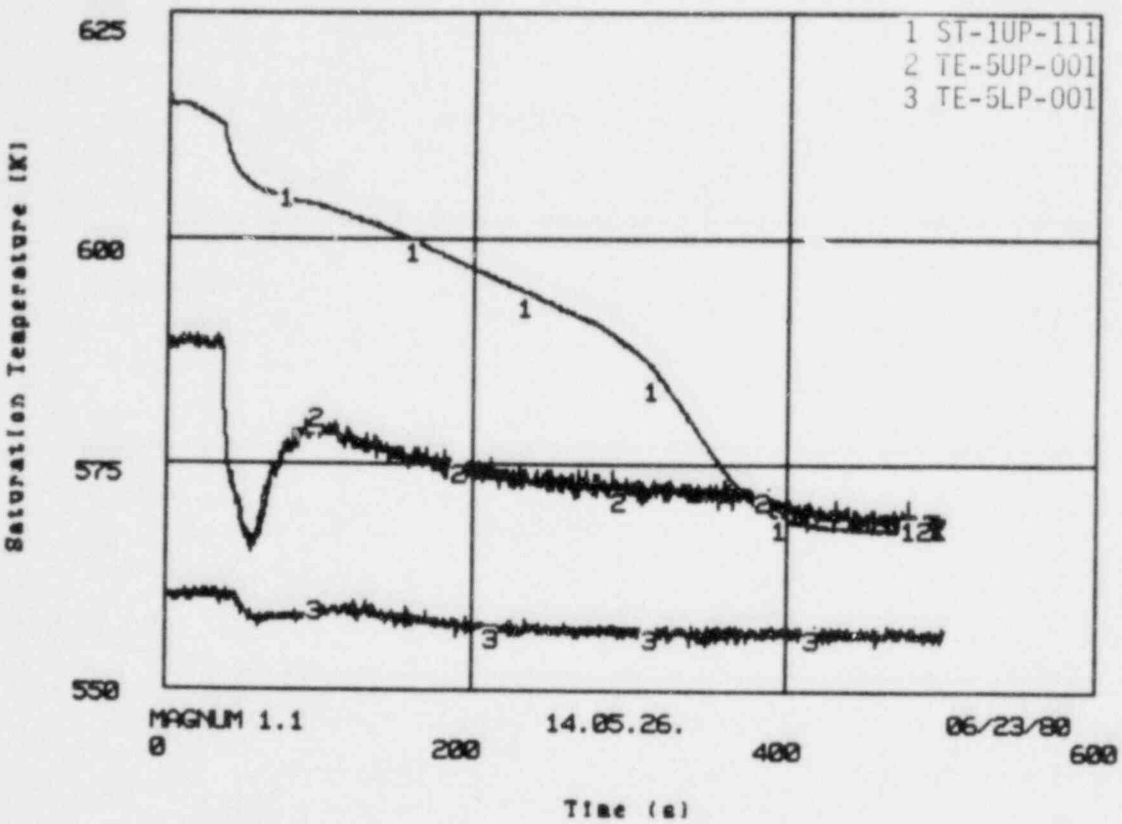


Figure 14. Comparison of upper plenum fluid, lower plenum fluid, and fluid saturation temperatures from 0 to 600 s.

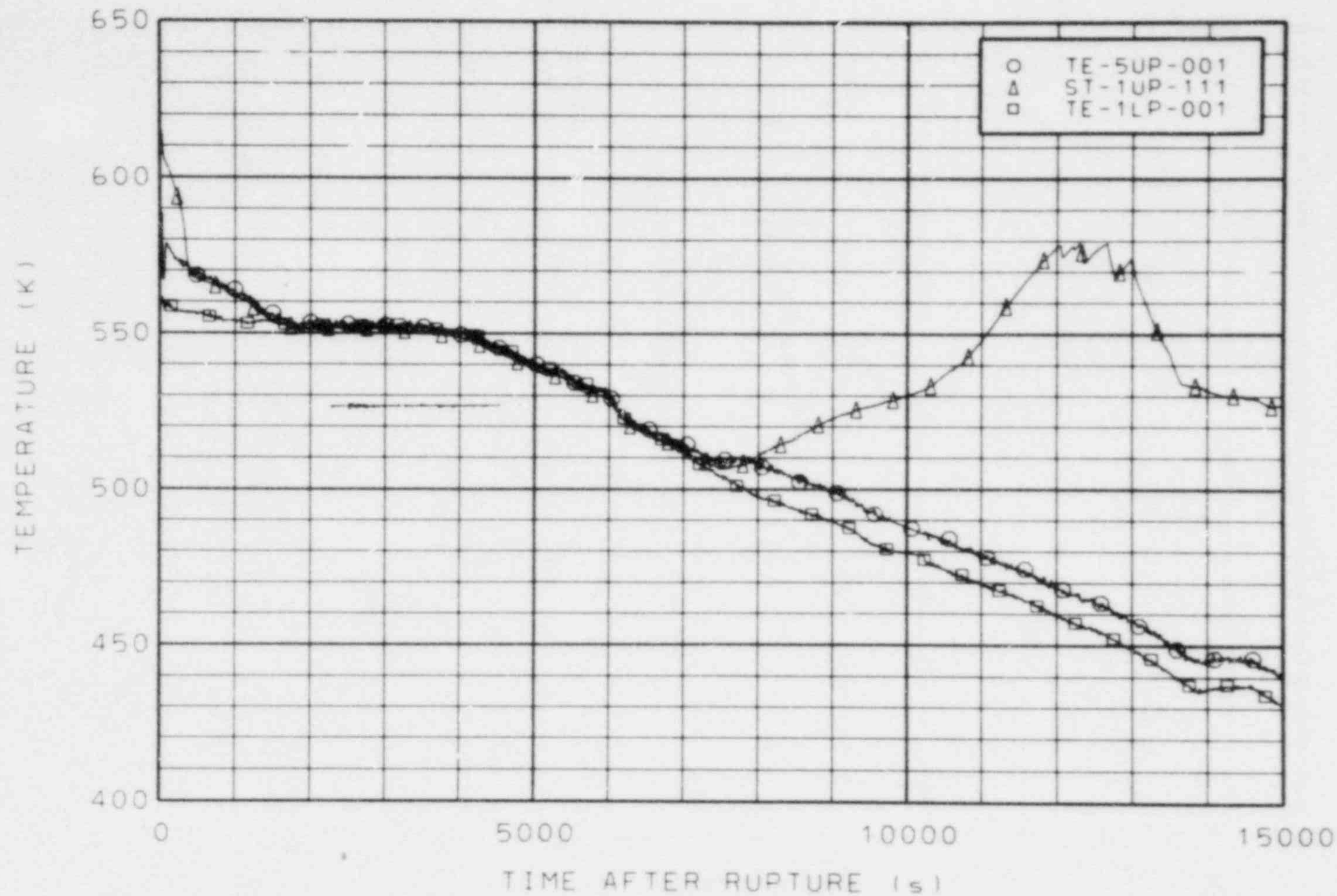


Figure 15. Comparison of upper plenum fluid, lower plenum fluid, and fluid saturation temperatures from 0 to 18 000 s.

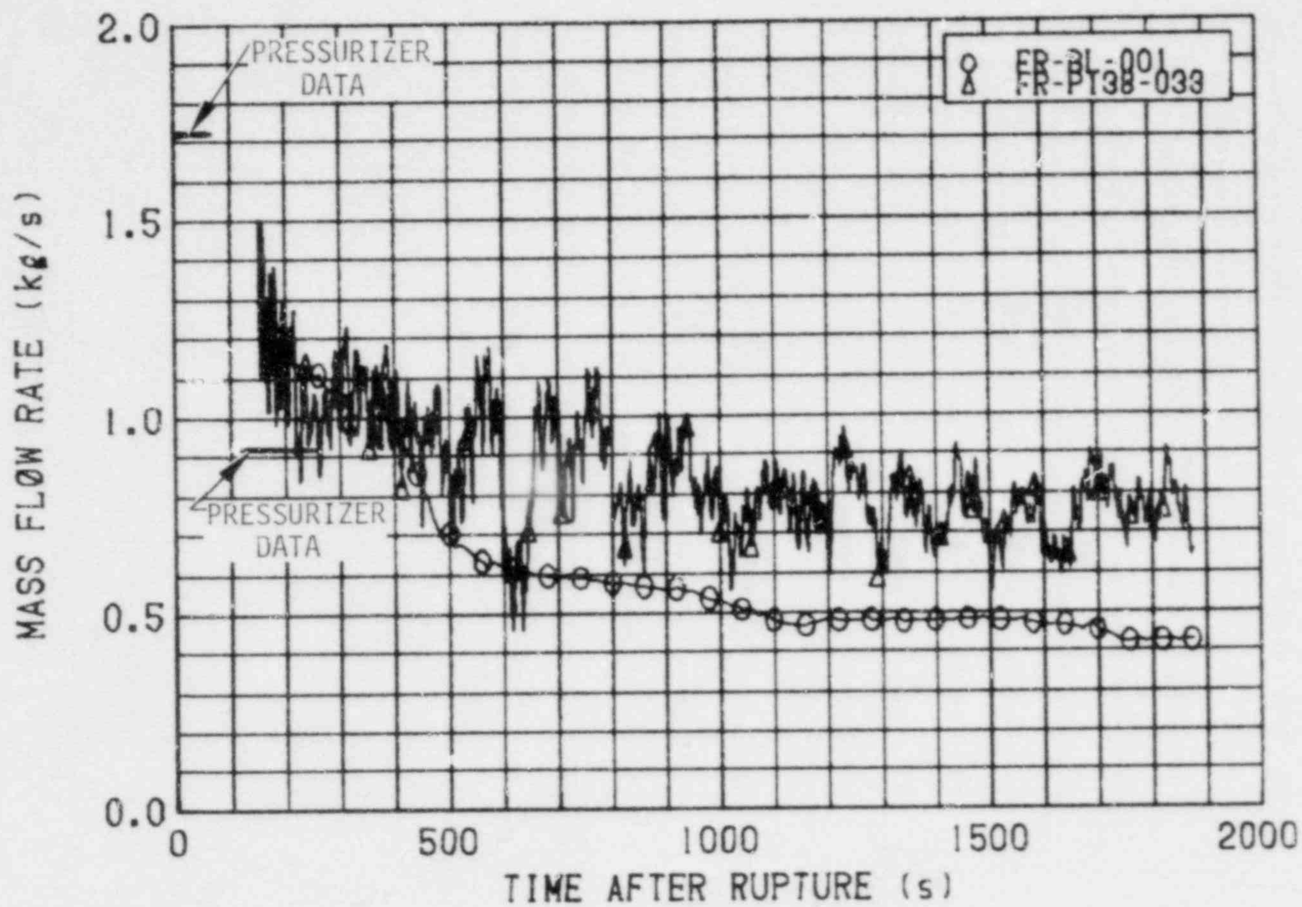


Figure 16. Comparison of broken loop cold leg mass flow with predictions.

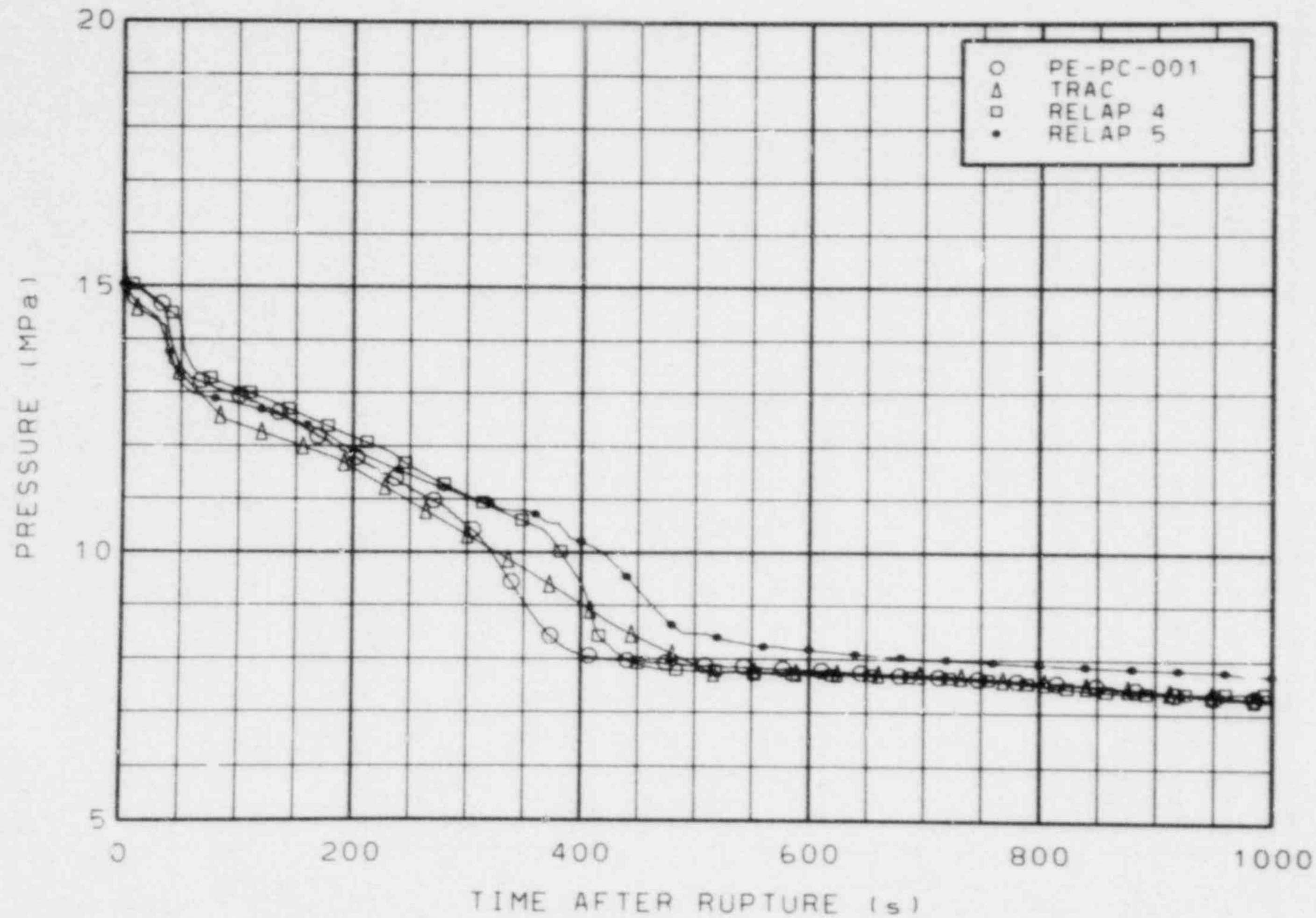


Figure 17. Comparison of system pressure with predictions from 0 to 1000 s.

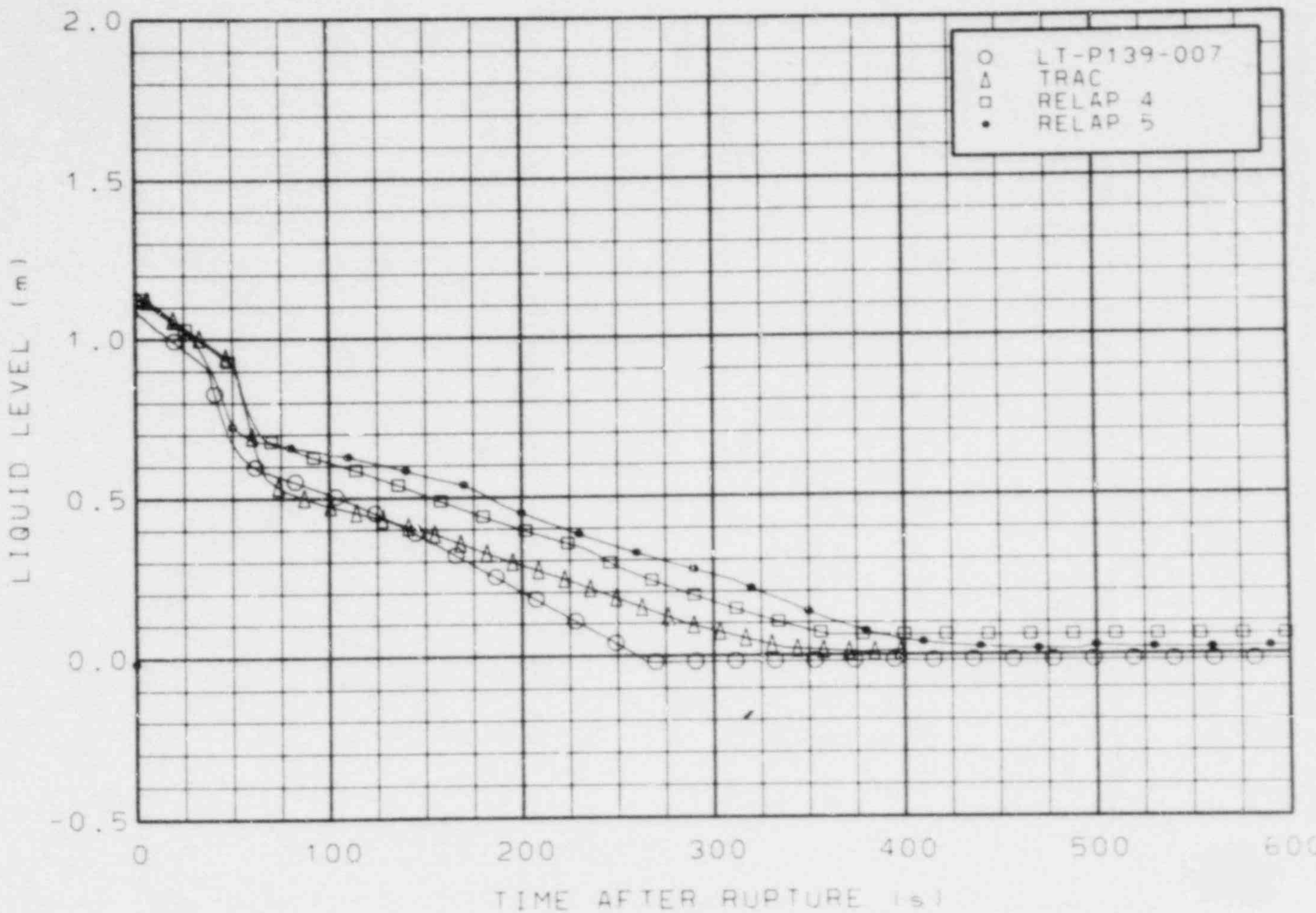


Figure 18. Comparison of pressurizer liquid level with predictions.

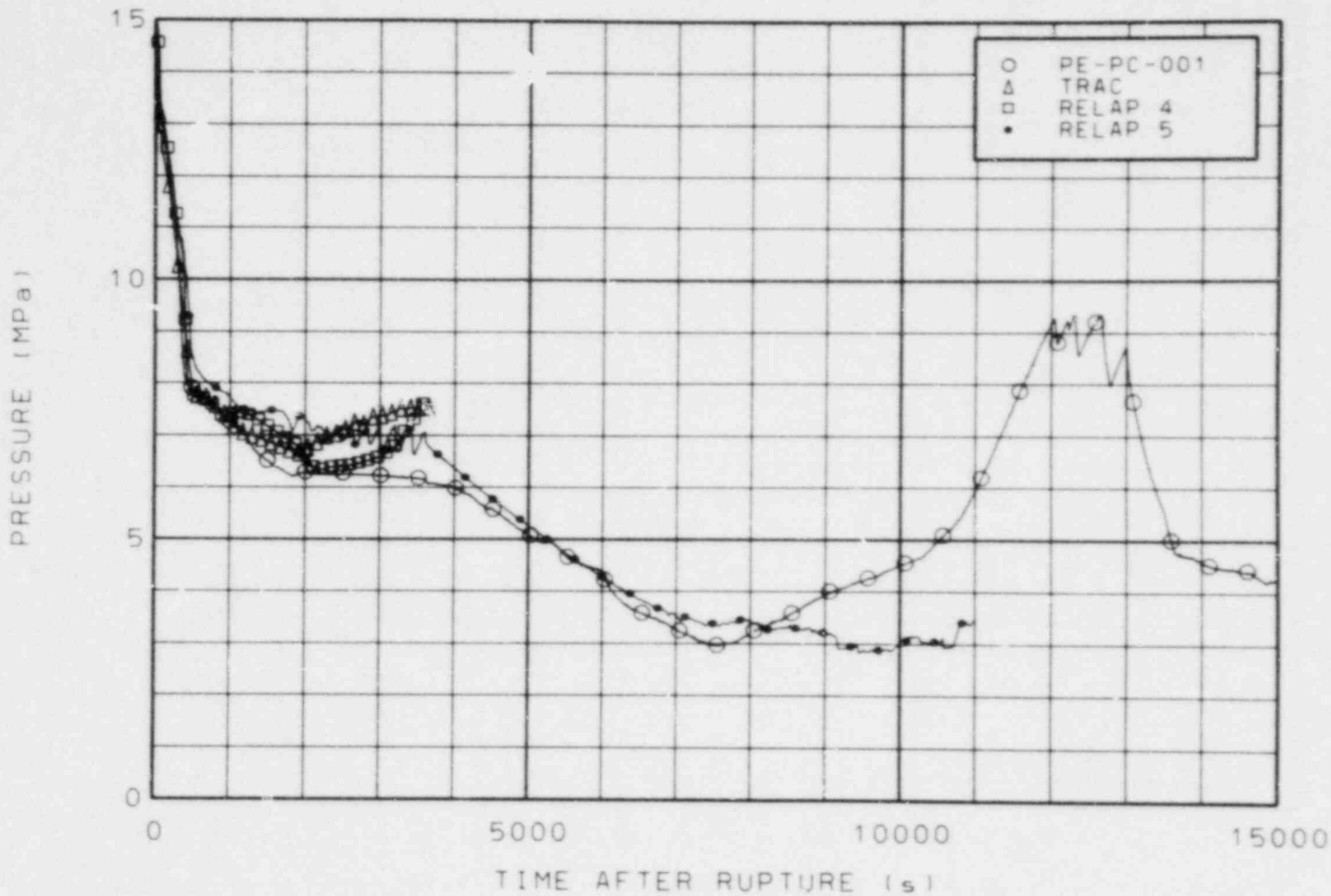


Figure 19. Comparison of system pressure with predictions from 0 to 15 000 s.

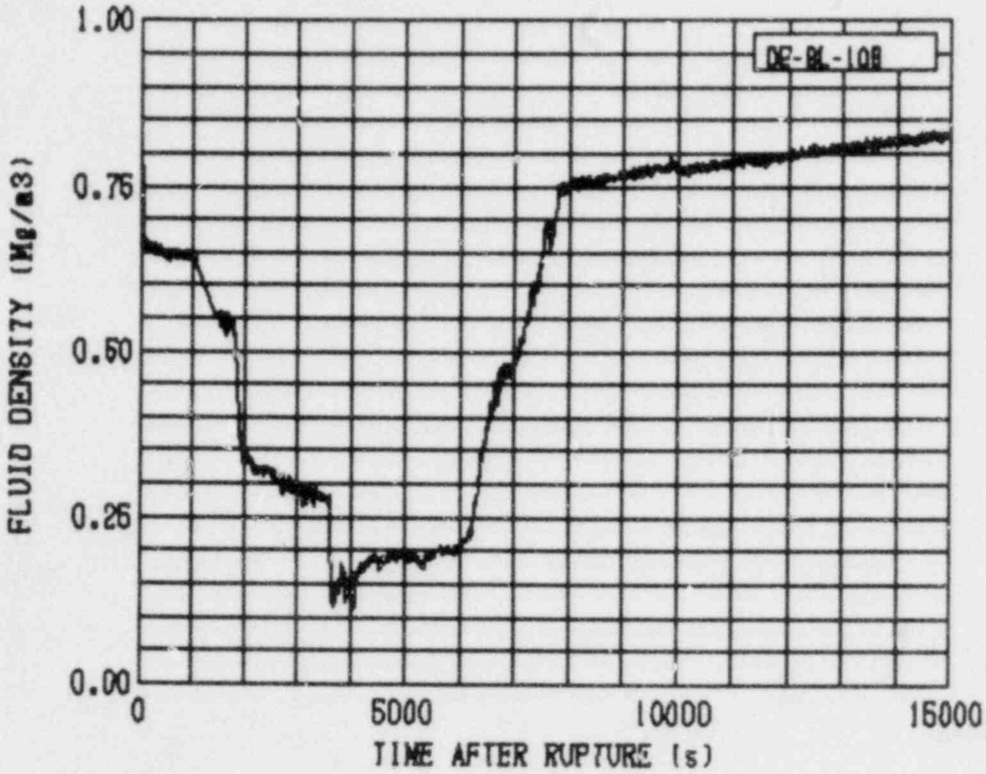


Figure 20. Broken loop cold leg average density.

## 6. REFERENCES


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# APPENDIX A

# LOFT SYSTEM GEOMETRY AND CORE CONFIGURATION

## APPENDIX A

## LOFT SYSTEM GEOMETRY AND CORE CONFIGURATION

The LOFT system geometry is shown in Figure A-1, and a representation of the core configuration illustrating the instrumentation and position designations is shown in Figures A-2 and A-3, respectively. The small break orifice geometry unique to the Test Series L3 LOCEs is shown in Figure A-4. Figure A-5 shows the LOFT steam generator geometry and instrument locations.

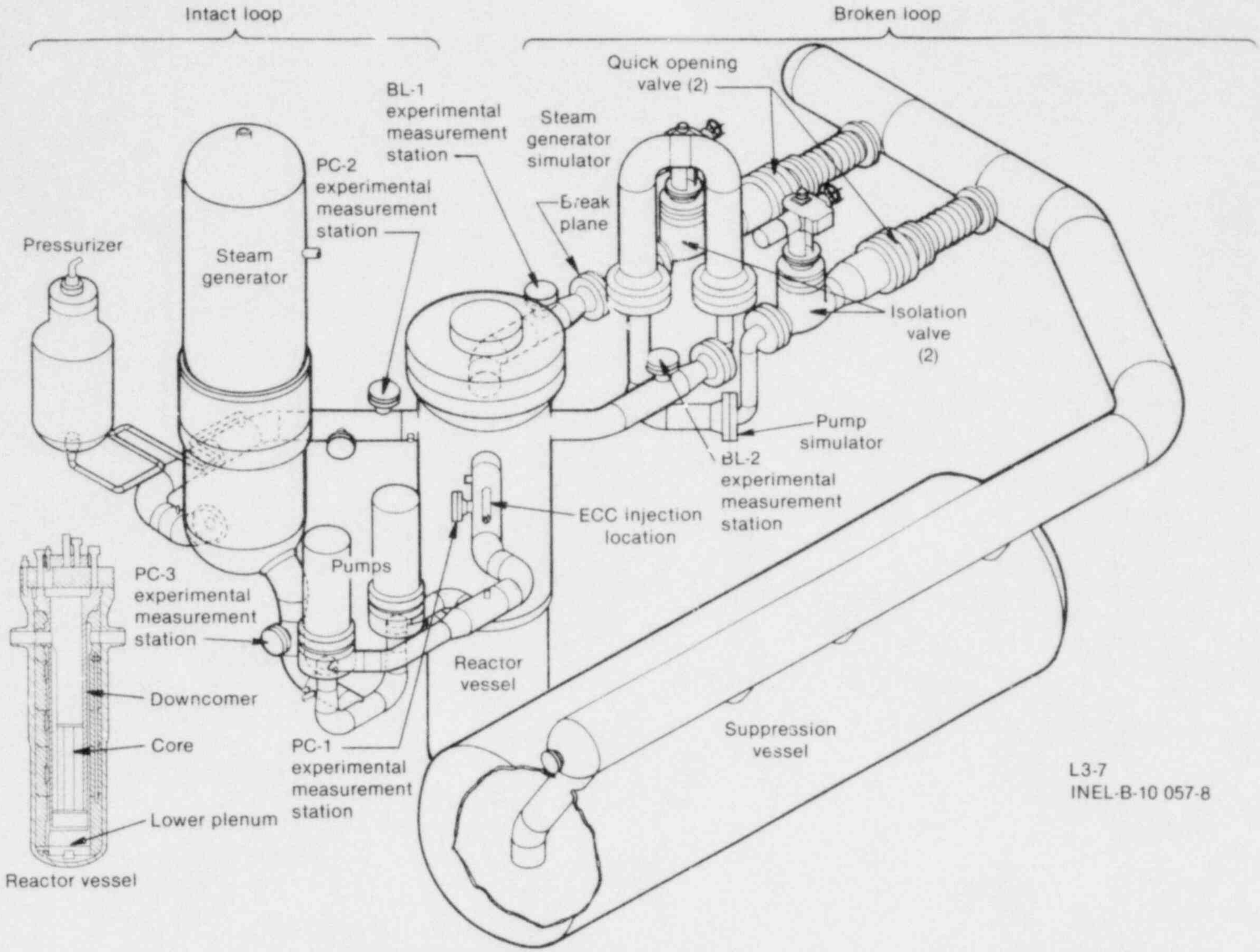


Figure A-1 Axonometric projection of LOFT system.

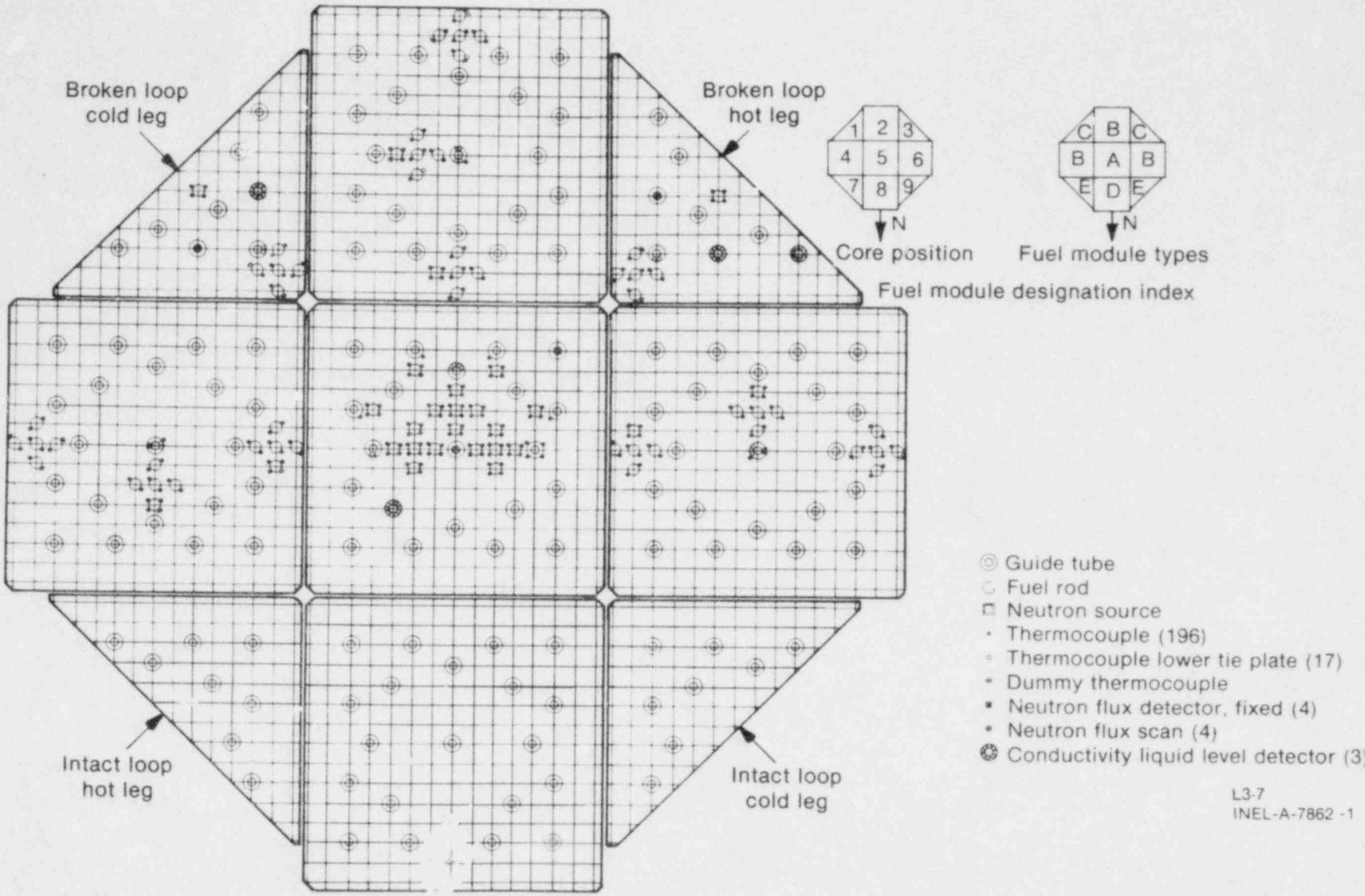


Figure A-2. LOFT core configuration and instrumentation.

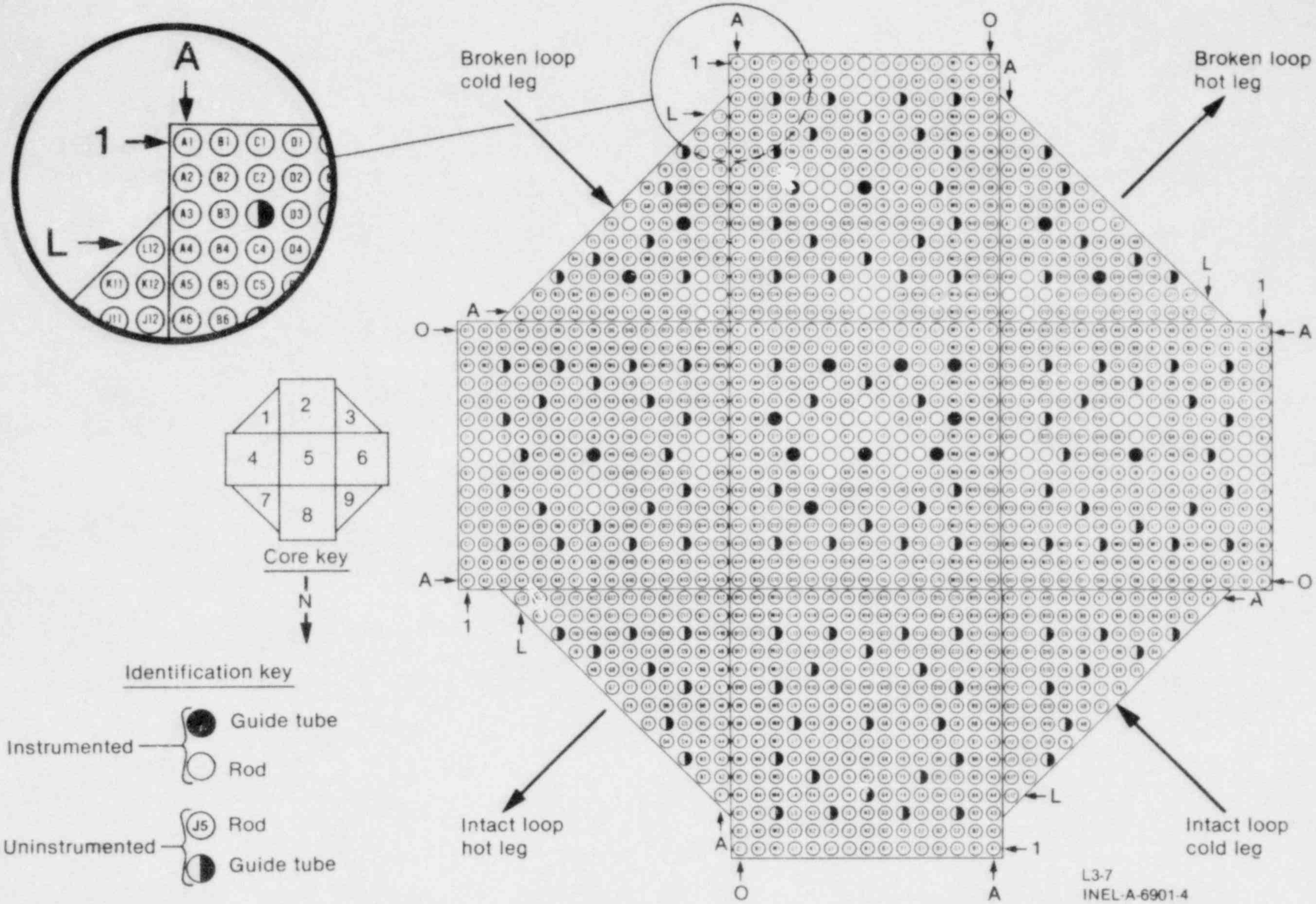


Figure A-3. LOFT core map showing position designations.

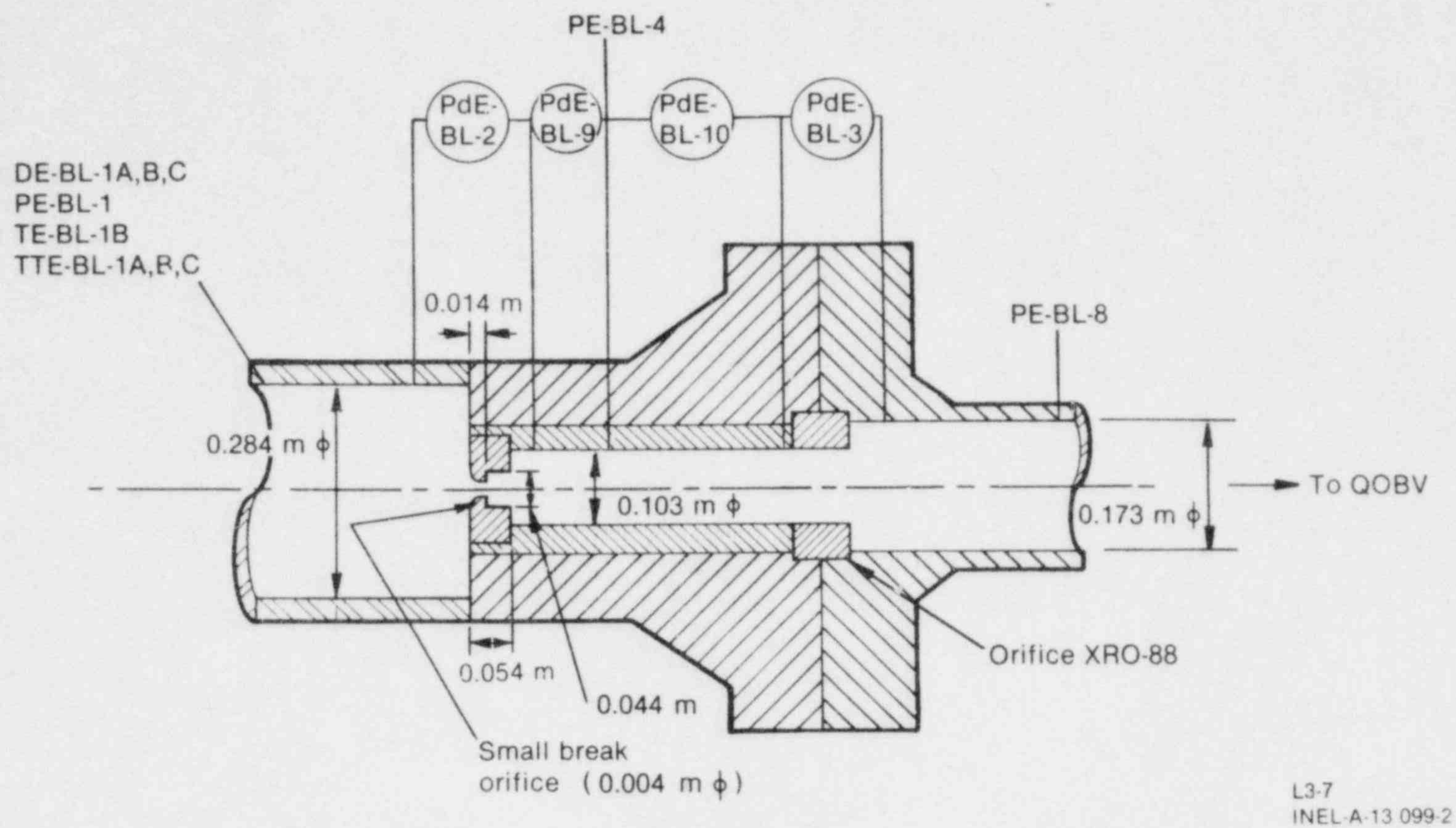


Figure A-4. LOFT small break orifice configuration.

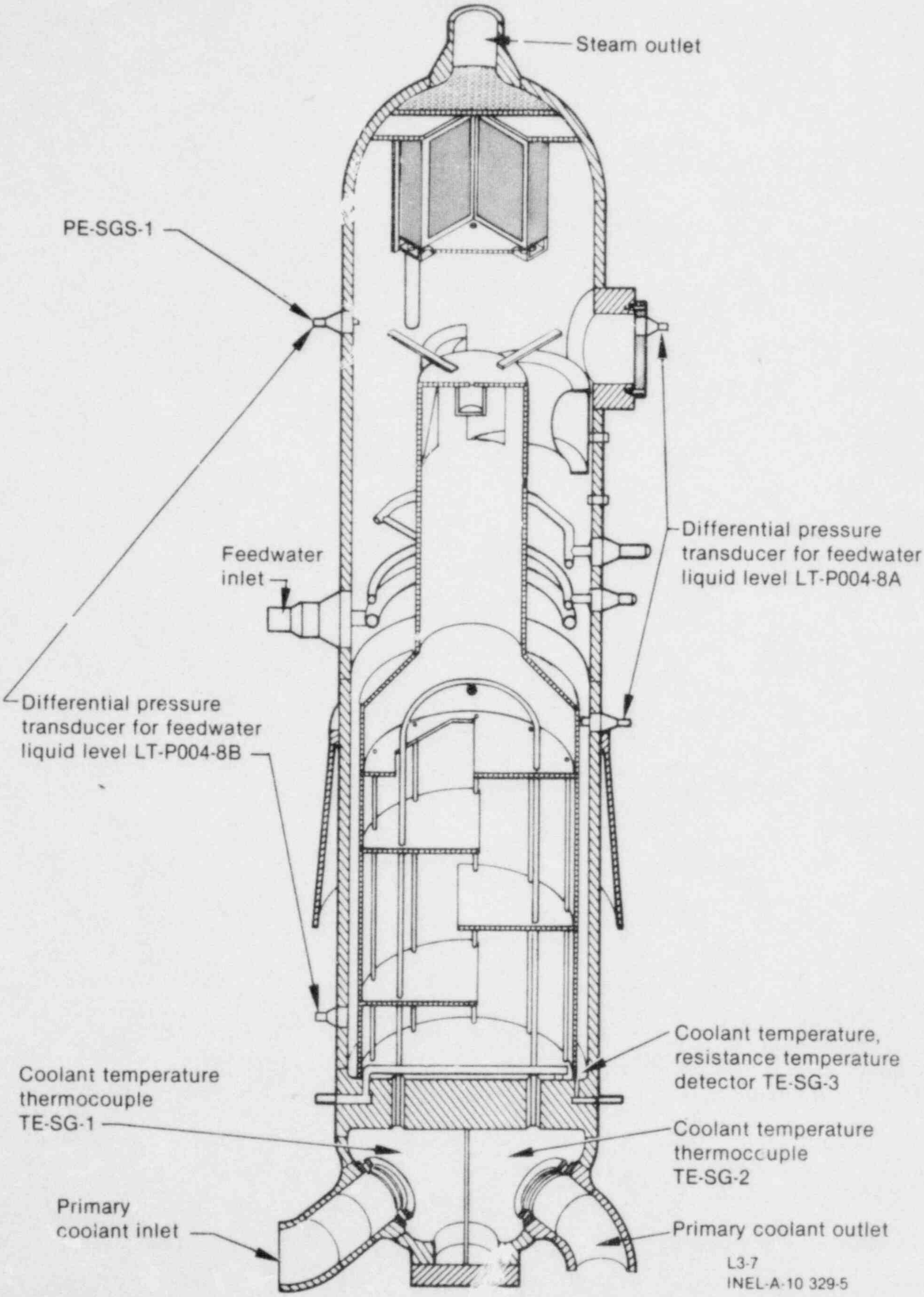


Figure A-5. LOFT steam generator and instrumentation.