PDR

EGG-LOFT-5921
Project No. P 394

June 1982

QUICK-LOOK REPORT ON LOFT NUCLEAR EXPERIMENT L2-5

RC Research and/or Technical Assistance Rept

James P. Adams

U.S. Department of Energy
Idaho Operations Office • Idaho National Engineering Laboratory

This is an informal report intended for use as a preliminary or working document

Prepared for the
U.S. Nuclear Regulatory Commission
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A6048

EG&G Idaho

EG&G Idaho Inc

FORM EG&G 398
(Rev. 03 82)

INTERIM REPORT

Accession No. ______________

Report No. EGG-LOFT-5921

Contract Program or Project Title:

LOFT Program

Subject of this Document:

Quick-Look Report on LOFT Nuclear Experiment L2-5

Type of Document:

Preliminary Analysis

Author(s):

James P. Adams

Date of Document:

June 1982

Responsible NRC Individual and NRC Office or Division:

G. D. McPherson, Chief, Integral Systems Section, Experimental Programs Branch, USNRC.

> This document was prepared primarily for preliminary or internal use. It has not received full review and approval. Since there may be substantive changes, this document should not be considered final.

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

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C.
Under DOE Contract No. DE-AC07-76ID01570
NRC FIN No. A6048

INTERIM REPORT

EGG-LOFT-5921

QUICK-LOOK REPORT ON LOFT NUCLEAR EXPERIMENT L2-5

By:

J. P. Adams

Approved:

D. J. Hanson, Manager
LOFT Program Division

L. P. Leach, Manager
LOFT Department

ABSTRACT

Experiment L2-5, simulating a guillotine (complete offset) shear of a main coolant pipe in a four-loop commercial pressurized water reactor (PWR) operating at nominal conditions, was completed on June 16, 1982, in the Loss-of-Fluid Test (LOFT) PWR. Assumptions of loss of offsite power and atypical primary coolant pump coastdown were incorporated into the simulation in an attempt to create core flow stagnation which would prevent the early core-wide quench phenomena observed in earlier accident simulations. The Experiment L2-5 accident simulation was successful in preventing the early core-wide quench phenomena. The fuel cladding reached a maximum temperature of 1077 K (1479°F) at 28.5 s into the transient. The emergency core cooling system (ECCS) began operation at 17 s into the transient and reflooded the core and quenched the fuel cladding by 65 s. The Experiment L2-5 simulation showed that the ECCS, as scaled in LOFT, returned the reactor coolant system to a stable condition in a situation wherein the peak cladding temperature occurred during the reflood phase of the transient. The Experiment L2-5 simulation also showed that this highly improbable scenario of accident events leading to the worst case large break condition (core flow stagnation), although causing a more severe core thermal transient than that resulting from a main coolant pipe shear only, did not result in fuel rod rupture. During plant recovery, utilization of core exit and upper plenum temperature measurements was shown to be inadequate as a means of reactor vessel liquid level control.

NRC FIN No. A6048--LOFT Experimental Program

SUMMARY

Loss-of-Fluid Test (LOFT) Experiment L2-5, which was completed on June 16, 1982, simulated a guillotine (complete offset) rupture of an inlet pipe in a pressurized water reactor (PWR) and was the third loss-of-coolant experiment (LOCE) in the LOFT Power Ascension Experiment Series L2. Experiment L2-5 differed principally from the previous large-break experiments, Experiments L2-2 and L2-3, in that the primary coolant pumps were turned off and were decoupled from their flywheels within 1 s after rupture initiation in Experiment L2-5 (resulting in an atypically fast coastdown), whereas pump operation continued throughout Experiments L2-2 and L2-3.

Experiment L2-5 was initiated from nominal PWR operating conditions by opening the two quick-opening blowdown valves. Within the first 1 s, primary coolant pumps were tripped, the system depressurized to upper plenum fluid saturation conditions, and the core cladding temperature started to deviate from saturation. The early bottom-up quench that occurred in Experiment L2-3 did not occur in Experiment L2-5, as intended by experiment design. A top-down quench which affected the top half of the central fuel assembly occurred between 12 to 23 s in the transient. The peripheral fuel assembly temperature remained saturated, except for some isolated short intervals in some measurements, until approximately 23 s when they departed from saturation. Accumulator injection initiated at 17 s, refilled the lower plenum by 31 s, and, combined with high- and low-pressure injection, reflooded the core by 55 s. The core was fully quenched by 65 s.

Core thermal behavior can be characterized by three different cladding temperature scenarios, each occurring in a different region of the core:

1. An immediate and sustained deviation from saturation lasting until final core quench in the lower half of the central fuel assembly

2. An immediate deviation from saturation followed by a top-down quench and a second temperature excursion in the top half of the central fuel assembly and the high-power locations in the peripheral fuel assemblies

3. A delayed (20 s) temperature excursion in other regions of the peripheral fuel assemblies.

Conclusions from this preliminary analysis of Experiment L2-5 are as follows:

1. Primary coolant system hydraulics were consistant with the core thermal behavior and were similar to the Experiment L2-3 hydraulics. Hydraulic differences between Experiments L2-3 and L2-5 were consistant with the difference in primary coolant pump operations.

2. Cladding surface thermocouples had only a minor effect on the core thermal behavior. This is consistant with previously published data from separate effects tests.

3. No fuel rod rupture occurred during Experiment L2-5, as determined from postexperiment chemical analysis of the primary coolant. However, the possibility of cladding deformation cannot be ruled out at this time.

4. Preexperiment calculations of Experiment L2-5 agreed very well with the measured peak cladding temperature and primary system pressure. Differences between prediction and data include the top-down quench (not calculated) and core reflood rate (calculated low).

5. The Experiment L2-5 simulation also showed that this highly improbable scenario of accident events leading to the worst case large-break condition (core flow stagnation), although causing a more severe core thermal transient than that resulting from a main coolant pipe shear only, did not result in fuel rod rupture.

6. During plant recovery, utilization of core exit and upper plenum temperature measurements was shown to be inadequate as a means for reactor vessel liquid level control.

ACKNOWLEDGMENTS

I would like to acknowledge the following for their assistance in this report: S. Behling, P. Demmie, D. Batt, and V. Berta for technical input as well as review of the text; A. Sanchez-Pope for preparation of Tables 2 and 3; S. Thompson and B. Marlow for preparation of the Figures; and G. Hammer for editing and report layout.

CONTENTS

FIGURES

TABLES

QUICK-LOOK REPORT ON LOFT NUCLEAR EXPERIMENT L2-5

1. INTRODUCTION

Experiment L2-5 was successfully completed on June 16, 1982, in the Loss-of-Fluid Test (LOFT) facility. This experiment simulated a guillotine (complete offset) rupture of an inlet pipe in a pressurized water reactor (PWR) and was the third experiment in the LOFT Power Ascension Experiment Series L2. Experiment L2-5 differed from the previous large-break Loss-of-Coolant Experiments (LOCEs) L2-2[1,2] and L2-3[3,4] in that the primary coolant pumps were turned off within 1 s of experiment initiation and were simultaneously decoupled from their external flywheels. In Experiments L2-2 and L2-3, primary coolant pump operation was continued throughout. Figure 1 shows the pump speed for Experiments L2-5 and L2-3. Table 1 summarizes the three large-break experiments. The atypical primary coolant pump operation in Experiment L2-5 [approximating a loss-of-coolant accident (LOCA) simultaneous with a loss of site power and atypically fast pump coastdown] was specified[5] in an attempt to cause early flow stagnation in the core and preclude the early bottom-up core-wide rewet which occurred in Experiments L2-2 and L2-3.

Experiment L2-5 was initiated, after operating the reactor at 36.0 MW for 40 effective full power hours (EFPH) to build up a fission decay product inventory, by opening the two quick-opening blowdown valves (QOBVs). The primary coolant pumps were turned off within 1 s. High-pressure injection system (HPIS) and low-pressure injection system (LPIS) injections were delayed to 24 and 37 s, respectively, to simulate the delay expected for a PWR emergency diesel to begin delivering power (in response to the loss of site power). For the purpose of this report, the experiment was completed when core reflood and fuel cladding quench were achieved.

The programmatic objectives for Experiment L2-5 were to:

1. Provide experimental data to demonstrate that Appendix K[6] assumptions result in a conservative prediction of peak cladding

TABLE 1. LOFT TEST SERIES L2 EXPERIMENTS PERFORMED TO DATE

Experiment	Date Completed	Power Level MW	MLHGR[a] kW/m	MLHGR[a] kW/ft	Description
L2-2	12/9/78	25	26	8	Large-break experiment initiated from conditions representative of lower PWR power conditions (~60% normal) with pumps running throughout.
L2-3	5/12/79	37	39	12	Large-break experiment initiated from conditions representative of a PWR operating at normal power with pumps running throughout.
L2-5	6/16/82	36	40	12	Large-break experiment initiated from conditions representative of a PWR operating at normal power and coincident with loss of site power and atypical pump coastdown.

a. MLHGR--maximum linear heat generation rate.

temperature, even if core hydraulic conditions were to occur in a commercial reactor which precluded early return to nucleate boiling (rewet).

2. Provide data to confirm that results from earlier LOFT large-break experiments were not being significantly affected by external cladding thermocouples.

To support the programmatic objectives, the following were defined as the specific objectives for Experiment L2-5:[5]

1. To determine if early core rewet occurs following a scaled LOFT 200% double-ended cold leg break with immediate primary coolant pump trip.

2. To provide data on core thermal response which can be used to evaluate computer code predictions and to compare with acceptance criteria in 10CFR50.46.[6]

3. To determine system behavior and core thermal response during the reflood portion of a double-ended cold leg break experiment.

4. To evaluate cladding surface thermocouple effects during blowdown and reflood by comparing the responses of LOFT fuel bundle instrumentation.

An evaluation of plant performance for Experiment L2-5 is presented in Section 2, including a summary of specified and measured initial conditions in Table 2 and a chronological listing of identifiable significant events in Table 3. Section 3 presents a summary of Experiment L2-5 results and Section 4 contains conclusions based on these results. Data plots are presented in Section 5 to support and clarify the experiment chronology of events in Section 2 and the results and conclusions discussed in Sections 3 and 4. Also included are comparisons of measured data with preexperiment calculations performed by EG&G Idaho, Inc.,[7] using the RELAP5[a] computer code.[8] The LOFT system geometry for Experiment L2-5 is shown in Appendix A.

---

a. The analysis was performed using RELAP5/MOD1 Cycle 17, a production version of RELAP5/MOD1 which is filed under Idaho National Engineering Laboratory Computer Code Configuration Management Archival Number F00708.

2. PLANT EVALUATION

The initial conditions, identifiable significant events, and experimental measurements for Experiment L2-5 are summarized in this section.

2.1 Initial Conditions

A summary of the specified[5] and measured system conditions immediately prior to Experiment L2-5 is given in Table 2. All initial conditions were within the limits specified by Reference 5 except core power (1% low), core outlet temperature (<1% low) and differential temperature (2% low), blowdown suppression tank (BST) liquid level (1% high), and BST pressure (15% high). The out-of-specification values of core power, core outlet and differential temperatures, and BST liquid level are not expected to have affected the results of the experiment, since the deviations from specification are small. The BST pressure was much higher than specified, but should not have adversely affected the transient during the blowdown phase since break flow was choked during this time. During the reflood phase, the higher BST pressure may have had an effect, although it is considered to be small, since the difference between specified and measured pressure is small compared to primary coolant system pressure during reflood. The conclusion that these out-of-specification conditions did not adversely affect the experiment results will, however, have to be verified in the postexperiment analysis. The out-of-specification conditions were of short duration and began with the opening of the broken loop isolation valves within 1 min of experiment initiation.

2.2 Chronology of Events

Table 3 contains a list of identifiable events measured in Experiment L2-5 and compares the measured times with those predicted[7] and with those measured in Experiment L2-3. An annotated primary system pressure curve is shown in Figure 2. The experiment was initiated by opening the

TABLE 2. INITIAL CONDITIONS FOR EXPERIMENT L2-5

Parameter	Specified Value[a]	Measured Value
Primary Coolant System		
Core ΔT (K)	35.8 ± 2	33.1 ± 4.3[b]
(°F)	64.5 ± 2	59.6 ± 7.7
Hot leg pressure (MPa)	14.95 ± 0.1	14.94 ± 0.06
(psia)	2168 ± 15.0	2167 ± 9
Core outlet temperature (K)	592 ± 2	589.7 ± 1.6[b]
(°F)	605 ± 4	601.8 ± 2.9
Cold leg temperature (K)	--	556.6 ± 4.0
(°F)		542.2 ± 7.2
Mass flow rate (kg/s)	--	192.4 ± 7.8
(lbm/hr x 10^6)		1.53 ± 0.06
Boron concentration (ppm)	As required to maintain reactor critical	668 ± 15 ppm
Reactor Vessel		
Power level (MW)	37.5 ± 1.0	36.0 ± 1.2[b]
Maximum linear heat generation rate (kW/m)	--	40.0 ± 3.0
(kW/ft)		12.2 ± 0.9
Control rod position (above full-in position) (m)	1.37 ± 0.01	1.38 ± 0.01
(in.)	54.0 ± 2.0	54.3 ± 0.4
Steam Generator Secondary Side		
Water level (m)	0.19 ± 0.05	0.23 ± 0.06
(in.)	7.5 ± 2.0	9.16 ± 2.4
Water temperature (K)	--	542.4 ± 4.1
(°F)	--	516.1 ± 7.4
Pressure (MPa)	--	5.85 ± 0.06
(psia)	--	848 ± 8.7
Mass flow rate (kg/s)	--	20.5 ± 1.4
(lbm/s)	--	45.2 ± 3.1
Saturation temperature (K)	--	547.1 ± 0.6
(°F)	--	525.1 ± 1.1

TABLE 2. (continued)

Parameter	Specified Value[a]	Measured Value
Pressurizer		
Liquid volume (m^3)	--	0.61 ± 0.02
(ft^3)	--	21.5 ± 0.7
Steam volume (m^3)	--	0.32 ± 0.02
(ft^3)	--	11.3 ± 0.3
Liquid level (m)	1.13 ± 0.18	1.14 ± 0.03
(in.)	44.5 ± 7.0	44.8 ± 1.2
Saturation temperature (K)	--	615.0 ± 0.3
(°F)	--	647.3 ± 0.54
Broken Loop		
Cold leg temperature (K)	As close as practical to	554.3 ± 4.2
(°F)	intact loop temperature	538.3 ± 7.6
Hot leg temperature (K)	As close as practical to	563 ± 4
(°F)	intact loop temperature	552 ± 7
Suppression Tank		
Liquid level (m)	1.27 + 0.127 - 0	1.41 ± 0.06[b]
(in.)	50.0 + 5 - 0	55.5 ± 2.36
Gas volume (m^3)	--	51.7 ± 2.1
(ft^3)	--	1825 ± 74
Water temperature (K)	356 ± 5	358.4 ± 3.0
(°F)	181 ± 9	185.4 ± 5.4
Pressure (gas space) (MPa)	0.08 ± 0.005	0.097 ± 0.007[b]
(psia)	12.5 ± 0.7	14.1 ± 1.01
Boron concentration (ppm)	>3000	3687 ± 15
ECC Accumulator A		
Gas volume (m^3)	--	0.91 ± 0.01
(ft^3)	--	32.1 ± 0.3
Liquid volume (m^3)	--	2.85 ± 0.01
(ft^3)	--	100.6 ± 0.3
Pressure (MPa)	4.2 ± 0.2	4.29 ± 0.06
(psia)	612 ± 25	622 ± 7

TABLE 2. (continued)

Parameter	Specified Value[a]	Measured Value
ECC Accumulator A (continued)		
Liquid temperature (K)	306 ± 3	303 ± 3
(°F)	90 ± 5	86.4 ± 5
Liquid level (m)	2.045 ± 0.025	2.04 ± 0.01
(in.)	80.5 ± 1.0	80.3 ± 0.4
Standpipe position (m)	1.245 ± 0.025	1.24 ± 0.03
(in.)	49 ± 1	49 ± 1
High-Pressure Injection System		
Flow rate (cm^3/s)	760 ± 63	740 ± 20
(gpm)	12 ± 1	11.7 ± 0.3
Liquid temperature (K)	303 ± 3	302 ± 1
(°F)	85 ± 5	84.2 ± 1.8
Low-Pressure Injection System		
Liquid temperature (K)	303 ± 3	302 ± 1
(°F)	85 ± 5	83.8 ± 1.8
Flow rate (ℓ/s)	--	5.5 ± 0.5
(gpm)	--	73 ± 7
Borated Water Storage Tank		
Liquid volume (m^3)	>83.3	92.7 ± 0.2
(ft^3)	>2940	3274 ± 7
Liquid temperature (K)	303 ± 3	302 ± 1
(°F)	85 ± 5	83.9 ± 1.8

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

b. These values are outside the band specified by the EOS. The out-of-specification conditions are considered to not have adversely affected the experimental results.

TABLE 3. CHRONOLOGY OF EVENTS FOR EXPERIMENT L2-5

Event	Time after Experiment Initiation (s)		
	Prediction	Experiment L2-5	Experiment L2-3
Experiment initiated[a]	0	0	0
End of subcooled blowdown	<0.2	0.043 ± 0.01	0.05
Reactor scrammed	0.103	0.24 ± 0.2	0.103
Cladding temperatures initially deviate from saturation	0.8-1.0	0.91 ± 0.2	0.96
Primary coolant pump disconnected from flywheel	1.0	0.94 ± 0.5	--[b]
End of subcooled break flow (cold leg)	2.1	3.4 ± 0.5	3.0
Top-down quench initiated	--[c]	12.1 ± 1.0	16
Pressurizer emptied	13.8	16.3 ± 2	14
Accumulator injection initiated	14.3	17.3 ± 0.7	17
End of top-down quench	--[c]	22.7 ± 1	22
HPIS injection initiated	22	24.0 ± 1	14
Maximum cladding temperature reached	6.6	28.5 ± 0.5	4.95
Lower plenum refill complete	45	31.2 ± 1[d]	35
LPIS injection initiated	35	37.0 ± 0.5	29
Accumulator empty	46.8	49.4 ± 1	49
Core reflood completed	--[c]	55.3 ± 1.5[d]	55
Core cladding quenched	78	65 ± 2	61
BST maximum temperature reached	70.0	72.5 ± 1	70

TABLE 3. (continued)

Event	Time after Experiment Initiation (s)		
	Prediction	Experiment L2-5	Experiment L2-3
LPIS injection manually stopped	--[c]	107.1 ± 0.7	--[e]
HPIS injection manually stopped	--[c]	144 ± 2	--[e]
HPIS injection reinitiated	--[c]	274 ± 2	--[e]
LPIS injection reinitiated	--[c]	347 ± 2	--[e]

a. Experiment initiation defined to be when the QOBVs are opened.

b. Primary coolant pumps were operated by intent throughout Experiment L2-3.

c. Not calculated.

d. Based on liquid level conductivity probe data which define liquid to cover the void fraction range from 0 to 0.2.

e. Event did not occur during Experiment L2-3.

two QOBVs. The primary coolant pumps were turned off and the primary coolant system depressurized to saturation, both within 1 s. The core cladding temperatures in the central fuel assembly departed from saturation within 2 s. A top-down partial quench of the central fuel assembly began at 12 s. Accumulator injection began at 17 s. The maximum cladding temperature of 1077 K (1479°F) was reached at 28.5 s, just prior to the completion of lower plenum refill at 31 s. LPIS flow initiated at 37 s, and combined with accumulator flow to reflood the core by 55 s.

2.3 Instrumentation Performance

The instrumentation used for Experiment L2-5 was similar to that used for Experiments L2-2 and L2-3 with the following changes:

1. The instrumented central fuel assembly was replaced. The new fuel assembly had fuel surface and centerline thermocouples, embedded cladding thermocouples, an axial string of self-powered neutron detectors, core inlet drag disc-turbine transducers, and core inlet and exit ultrasonic densitometers in addition to the instrumentation which existed in the previous fuel assembly.

2. Break flow was measured in the broken loop using a two-range drag disk assembly in conjunction with a gamma densitometer. In Experiments L2-2 and L2-3, break flow was measured using a single-range drag disk, a turbine meter, and a densitometer.

During the experiment, several parameters were monitored in real time on cathode ray tubes in the control room, visitor display room, and technical support center to determine the thermal and hydraulic state of the plant. The monitor systems include:

1. Safety parameter display system (SPDS) (15 primary and secondary system parameters): The primary coolant system parameters monitored by the SPDS were displayed and available for diagnosis prior to initiation of the transient if desired. All aspects of this portion of the SPDS worked as expected except for periodic display problems. There were some calibration difficulties with secondary system parameters.

2. Automated data qualification (ADQ) (a subset of cladding temperatures): The ADQ display did not activate until 1-1/2 min after experiment initiation. When the display was activated, however, it functioned properly.

There were 616 instruments recorded for evaluation of the experimental results. Of the number examined at this time, 95% performed satisfactorily.

3. RESULTS FROM EXPERIMENT L2-5

The preliminary analysis presented in this section is based on data processed and available within the first 2 weeks following Experiment L2-5. In certain instances, the results discussed reflect the degree of incompleteness of the analysis this soon after the experiment. Analysis of the data will continue, and complete analysis results of the experiment will be reprrted in future documents.

3.1 Core Thermal Response

The core thermal response during Experiment L2-5 varied as a function of location within the core. This response can best be characterized by examining three core regions:

1. The lower half of the central fuel assembly

2. The upper half of the central fuel assembly plus the high-power regions in the peripheral fuel assemblies (that is, those portions of the peripheral fuel assemblies adjacent to the central fuel assembly and that are near the peak power elevation of 0.66 m, 26 in.)

3. The lower-power regions of the peripheral fuel assemblies.

Cladding temperatures in the first region departed from saturation within the first 2 s after experiment initiation. Cladding temperatures initially rose quickly in response to degraded cooling, reached a plateau within 10 s, and then remained at approximately those levels for an additional 20 s (see Figure 3). The maximum measured cladding temperature of 1077 K (1479°F) occurred during this time. At approximately 30 s, a gradual cooling trend initiated as emergency core cooling system (ECCS) water filled the lower plenum. The gradual cooling trend continued until all fuel rods were quenched by 65 s. This thermal behavior differed from that which occurred during Experiment L2-3 in that the early rewet in Experiment L2-3 did not occur in Experiment L2-5 (see Figure 3). Figure 3 compares maximum cladding temperatures which occurred at different elevations. The apparent difference in quench resulted from this difference in elevation.

A markedly different thermal response was measured in the second region. As shown in Figure 4, the cladding temperatures were similar to those in the first region for nearly 15 s. At this time, there was a top-down quench, which lasted for up to 5 s, followed by a second cladding temperature excursion with a generally lower peak value. Final quench occurred during core reflood as in the first region. The top-down quench in this region is similar to the top-down quench which occurred somewhat later in Experiment L2-3, as shown in Figure 4.

In the third region there was a general cooling which was sufficient to maintain the cladding quenched until after 20 s, as shown in Figure 5. Some cladding temperatures in this region departed from saturation momentarily as shown in Figure 6, but followed the general trend, and underwent a sustained departure from saturation only after 20 s. The thermal behavior in this region was very similar to that which occurred in Experiment L2-3, as shown in Figure 5.

In summary, the core thermal behavior in Experiment L2-5 was very similar to that which occurred in Experiment L2-3 except for those phenomena which can be uniquely linked to differences in primary coolant pump operation during the experiments; that is, the early bottom-up core-wide rewet which occurred in Experiment L2-3 but not in Experiment L2-5, and the top-down quench which occurred earlier in Experiment L2-5 than in L2-3. Cladding temperature time histories are shown as a function of axial and radial core locations in Figures 7 and 8, respectively.

The effect of the cladding thermocouples on the thermal behavior of the core during Experiment L2-5 was small. Figure 9 is a comparison of fuel centerline temperatures in fuel rods with and without cladding thermocouples. As shown in the figure, the fuel rods with cladding thermocouples quenched approximately 5 s prior to the rods with no cladding thermocouples.

Figure 10 compares five fuel pellet offset temperatures from rods with and without cladding thermocouples. The times at which these fuel rods quenched differed by less than 5 s due to the presence or absence of cladding thermocouples. Measurements from the axial motion detectors also support this conclusion. The transitory temperature decrease for one measurement noted on the figure occurred simultaneously with a momentary cladding quench on the same fuel rod down to the peak power axial location of 0.66 m (26 in). In summary, the effects of cladding thermocouples on core thermal response during Experiment L2-5 was small, and this is consistent with previously reported results from other experiments.[9]

A fluid sample was taken from the BST subsequent to Experiment L2-5, and analyzed for fission products. The results indicate that no fuel cladding rupture occurred. Figure 11 shows measured fuel rod cladding temperature plotted versus pressure differential across the cladding. Differential pressure is the difference between fuel rod internal pressure, calculated using FRAP-T6[a,10] and measured cladding temperatures, and measured reactor vessel pressure. Also included, for reference, are fuel rod cladding rupture data from separate effects testing.[11] As shown in the figure, a possibility exists that some cladding deformation occurred. Therefore, it is concluded that, although no fuel rod rupture occurred during Experiment L2-5, some cladding deformation may have occurred. This conclusion will be confirmed when the fuel module is subsequently removed and visually inspected.

3.2 Hydraulic Response

Figure 12 shows the upper plenum pressure response from Experiments L2-5 and L2-3. As shown, the pressure responses for these two experiments are very similar. This similarity in pressure response

---

a. The version of the FRAP-T6 computer code used is filed under Idaho National Engineering Laboratory Computer Code Configuration Management Archival Number F00900.

resulted in similar times for initiation of accumulator injection, as shown in Table 3. Thus, pump operation had little effect on primary system depressurization.

The core fluid density variations during Experiment L2-5 can be inferred from measurements taken by seven in-core self-powered neutron detectors (SPNDs). Figure 13 shows the response of one SPND in a peripheral fuel assembly. As shown, the coolant density increased suddenly in this fuel assembly at approximately 10 s. This density increase was also measured by all other peripheral SPNDs. In addition, smaller density increases were measured at this time in the central fuel assembly. At approximately 15 s, the density decreased, which was a precurser to the general cladding temperature excursion at approximately 20 s. Starting at approximately 40 s, the SPNDs measured a cyclic core fluid density which has been described elsewhere as gravity reflood oscillations.[12] These oscillations were also measured during Experiment L2-3.

Additional hydraulic information is shown in Figures 14, 15, and 16, which show the core liquid level in the central assembly, a peripheral fuel assembly, and the downcomer and lower plenum. As shown, refill of the lower plenum started at 22 s and was completed by 31 s. In this and other liquid level measurements made using conductivity probes, liquid is considered to be present at a given elevation when the void fraction is less than 0.20, and is indicated on the figures by an "X". In Experiment L2-5, reflood of the core was initiated at 37 s and was completed by 55 s, a reflood rate of 0.093 m/s (3.7 in./s). This, again, is very close to the times and rate measured in Experiment L2-3 (see Table 3). During reflood, preliminary examination of hot leg densities in the broken and intact loops indicates very high void fractions. The upper head could be causing significant deentrainment.

In summary, the primary coolant hydraulics were consistent with the core thermal response. These hydraulics in Experiment L2-5 were similar to those measured during Experiment L2-3 with the exception of the core-wide bottom-up density surge in Experiment L2-3 (wherein the primary coolant pump operation was different) which did not occur during Experiment L2-5.

3.3 Comparison of Experimental Data with Experiment Prediction

In this section, the measured experimental data are compared with preexperiment calculations,[7] including core thermal and primary coolant system hydraulic response. Event time comparisons are shown in Table 3.

Figure 17 shows a comparison of measured and calculated cladding temperatures for the hottest core location. The calculated maximum peak cladding temperature was 1082 K (1488°F) compared to the measured maximum peak cladding temperature of 1077 K (1479°F), or a difference of 5 K (9°F). As also shown, both measured and calculated temperatures exhibited a cooling trend prior to quench due to ECCS injection. The calculated time of cladding quench was also very close (within 3 s) of measured. However, the top-down quench, which was measured in parts of the core, was not calculated. Also, the delayed (beyond 20 s) departure from saturation (measured in the peripheral fuel assemblies) was not calculated. The calculation of the maximum peak temperature is the most important calculation from a fuel rod damage viewpoint, and in this regard the calculation is good though the peak was calculated to occur 22 s earlier than in the measured data.

Figure 18 shows the measured and calculated upper plenum pressure. The calculated pressure was within 1 MPa of the measured pressure. The calculated pressure, lower than measured, resulted in a calculated time for accumulator injection of 14 s, compared to 17 s measured.

The calculated reflood rate was approximately 0.04 m/s (1.5 in./s), compared to the measured reflood rate of 0.083 m/s (3.7 in./s). This difference, which may be caused by incorrect calculation of emergency core coolant (ECC) bypass and/or entrainment, will be investigated during postexperiment analysis.

In summary, the preexperiment calculation of Experiment L2-5 correctly predicted the peak cladding temperature and primary system pressure. The analysis of the differences between calculated and measured system hydraulic phenomena and the effects on core thermal response, as discussed above, is preliminary.

3.4 Postexperiment L2-5 Plant Recovery

Subsequent to the core reflood and quench described in Sections 3.1 and 3.2, the ECCS flow was terminated and an attempt was made to control the reactor vessel liquid level below the nozzles by cycling the HPIS and LPIS flows to maintain liquid inventory. The ECCS flow cycling was based on operator observation of coolant thermocouples in the upper plenum. This was done to provide operating experience in preparation for the recovery phase of the next large break experiment, Experiment L2-6, where the reactor vessel liquid level after core reflood is to be controlled below the nozzles in order to minimize the transport of fission products. This procedure was not included as part of the Experiment L2-5 programmatic objectives and, therefore, was not included in the preexperiment calculations.

During this recovery phase, the liquid level dropped into the core region starting at approximately 190 s and continued to drop, due to a decay heat-induced boil-off, at a rate of approximately 0.03 m/s (1 in./s). HPIS and LPIS flows were reinitiated at 274 and 347 s, respectively, in an attempt to reflood the core and turn the temperature excursion around. The combined flows reflooded the core at an average rate of 0.03 m/s (1 in./s), and the core was fully quenched by 430 s (see Figure 19). The temperature information observed by reactor operators for liquid level control and also coolant temperature information at the core exit are shown in Figure 20. Comparing this information with that in Figure 19, clearly shows that these temperature measurements are not adequate for liquid level control and, hence, maintaining adequate core cooling.

4. CONCLUSIONS

The conduct of Experiment L2-5 and the data acquired concerning integral system response to the experiment are considered to have met the specific objectives as defined by Reference 5 and listed in Section 1.

Cladding thermal response varied within the core with three general scenarios: (a) immediate and sustained departure from saturation which occurred in the lower half of the central fuel assembly, (b) immediate departure from saturation followed by top-down quench (14 to 17 s), and a second departure from saturation occurring in the upper half of the central fuel assembly and high-power regions of the peripheral fuel assemblies, and (c) delayed departure (20 s) from saturation in the rest of the peripheral fuel assemblies.

The effect of the external cladding thermocouples on the core thermal behavior was small and was consistant with previously reported results from separate effects tests. No evidence of fuel rupture was detected, though some cladding deformation may have occurred. This will be confirmed by visual inspection after the fuel assembly is removed.

The measured primary coolant hydraulics were consistant with the core thermal behavior and were similar to the hydraulics measured during Experiment L2-3 excepting those directly related to primary coolant pump operation. Preexperiment calculations correctly predicted the peak cladding temperature and primary system pressure during Experiment L2-5. Additional efforts will be required to correctly calculate detailed core thermal behavior and core reflood.

The Experiment L2-5 simulation also showed that this highly improbable scenario of accident events leading to the worst case large-break conditions (core flow stagnation), although causing a more severe core transient than that resulting from a main coolant pipe shear only, did not result in fuel rod rupture.

During plant recovery, utilization of core exit and upper plenum temperature measurements was shown to be inadequate as a means of reactor vessel liquid level control.

5. DATA PRESENTATION

This section presents selected preliminary data from Experiment L2-5. Experimental data are overlayed with results from the preexperiment calculations made using the RELAP5/MOD1 computer code.[7] 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 the experiment is given in Reference 5.

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

Fluid temperature	-	±4 K (±7°F)
Fuel centerline temperature	-	±79 K (±142°F)
Fuel pellet offset temperature	-	±18 K (±32°F)
Pressure	-	±0.26 MPa (±21 psi)
Liquid level	-	±0.17 m (±0.56 ft)
Reactor power	-	±2 MW.

TABLE 4. LIST OF DATA PLOTS

TABLE 4. (continued)

TABLE 5. NOMENCLATURE FOR LOFT INSTRUMENTATION

Designations for the Different Types of Transducers

TE	-	Temperature element	FE	-	Coolant flow transducer
TF	-	Fuel pellet offset temperature element	TC	-	Fuel centerline temperature element
PE	-	Pressure transducer	DE	-	Densitometer
PdE	-	Differential pressure transducer	ME	-	Momentum flux transducer
LE	-	Coolant level transducer	FT	-	Flow rate transducer
			ST	-	Saturation temperature based on pressure

Designations for the Different Systems, Except the Nuclear Core

PE	-	Primary coolant intact loop	LP	-	Lower plenum
BL	-	Broken loop	ST	-	Downcomer stalk
RV	-	Reactor vessel	P120	-	Emergency core coolant system
SV	-	Suppression tank			
UP	-	Upper plenum	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.

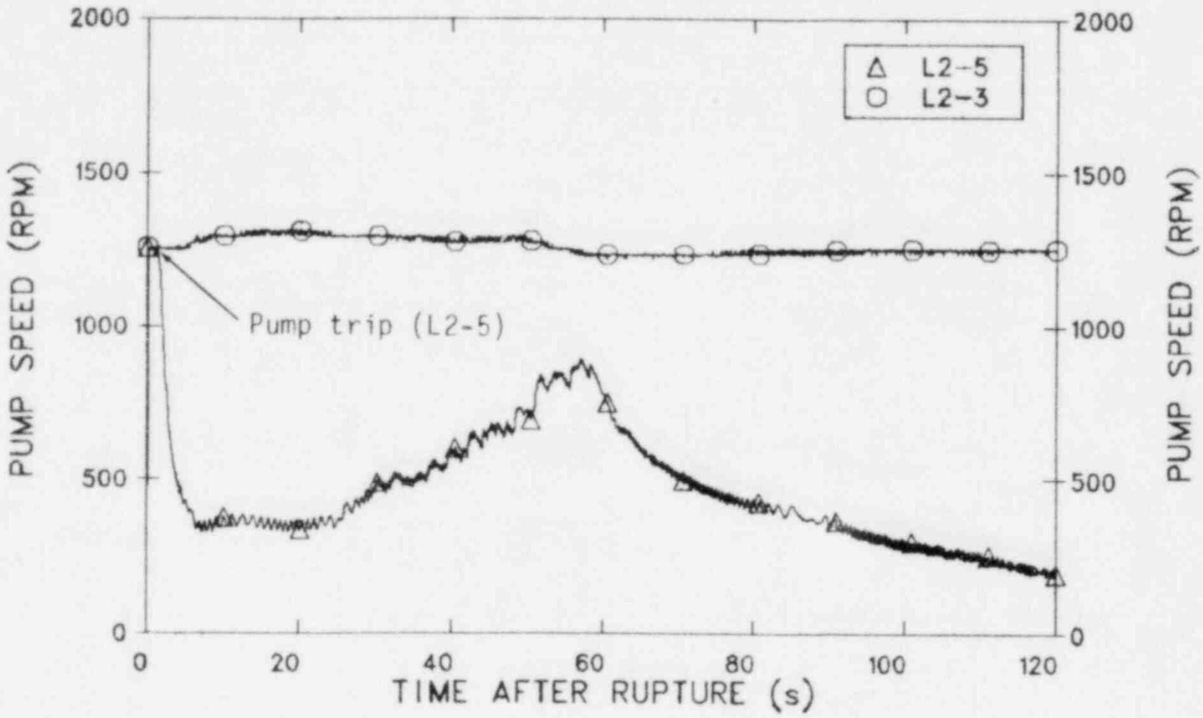

Figure 1. Comparison of primary coolant pump frequencies for Experiments L2-5 and L2-3.

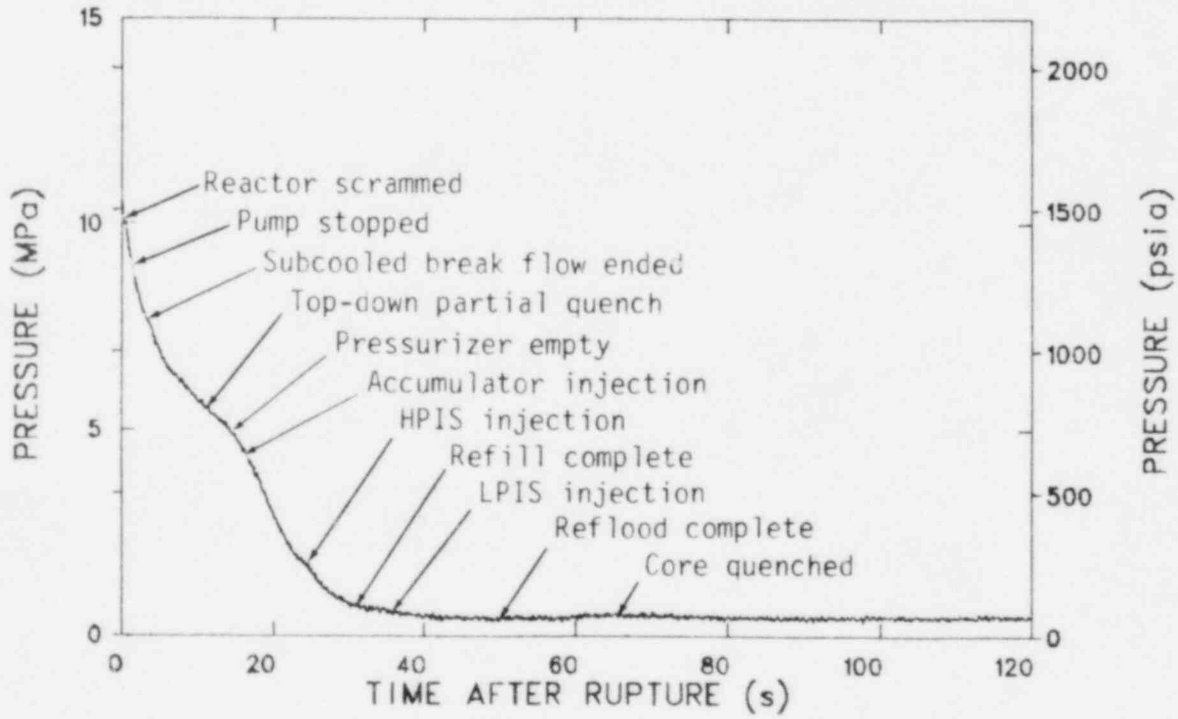

Figure 2. Response of primary system pressure during Experiment L2-5.

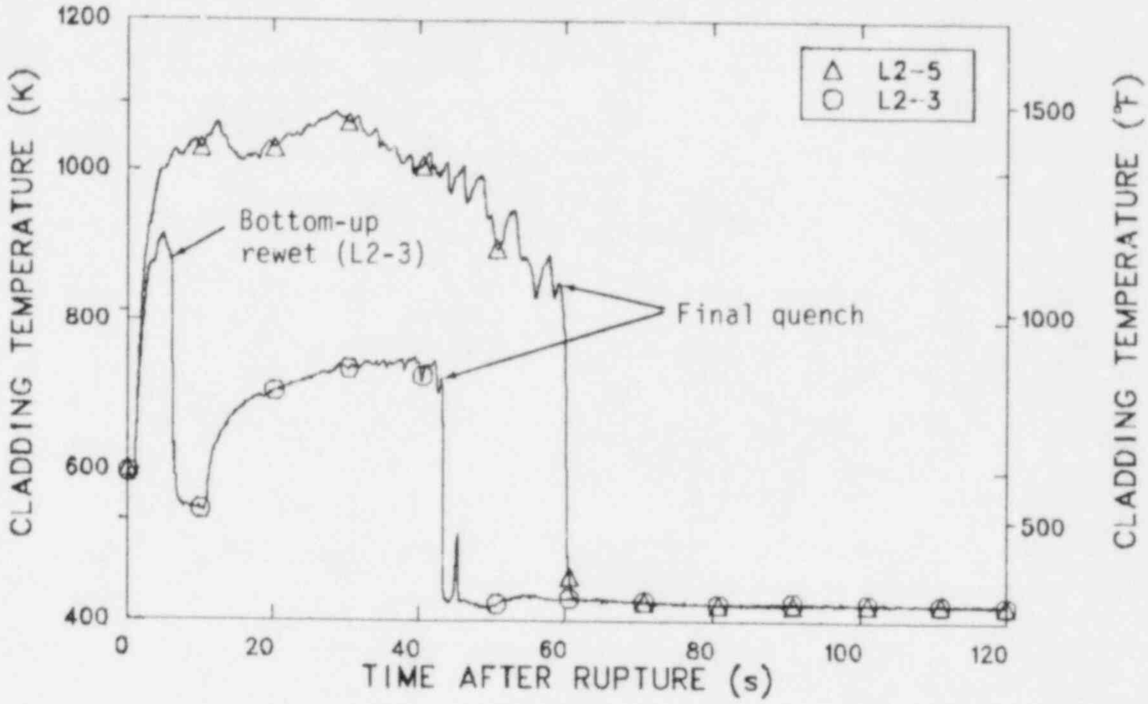

Figure 3. Comparison of maximum cladding temperatures for Experiments L2-5 and L2-3.

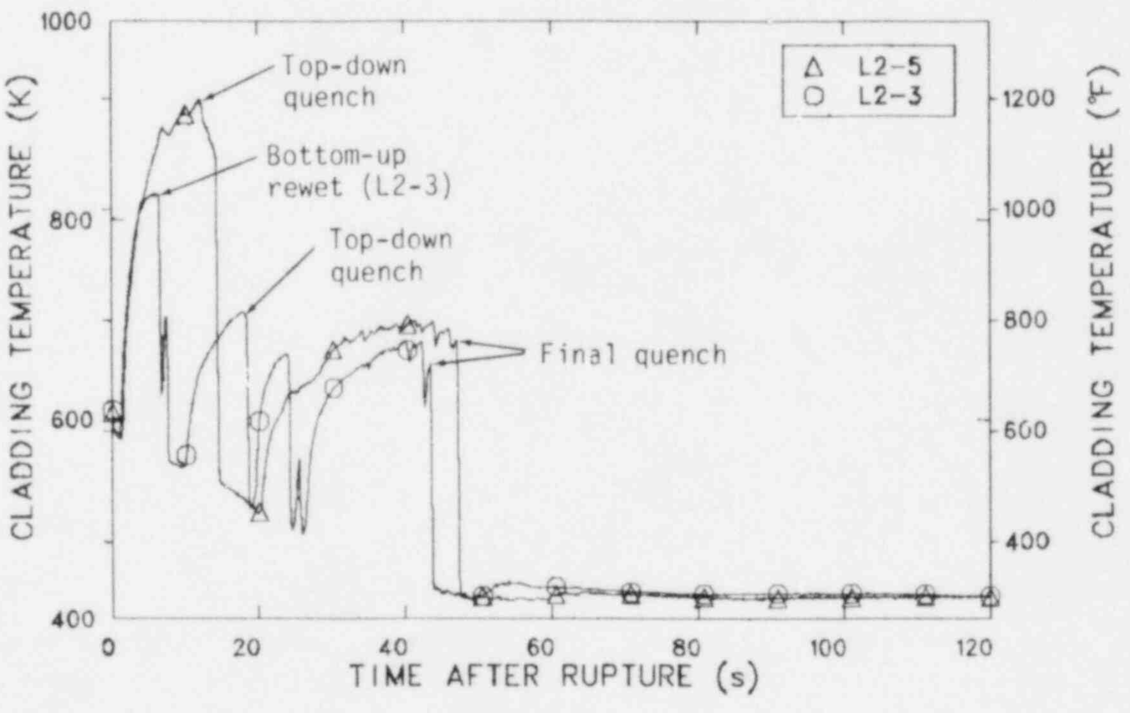

Figure 4. Comparison of cladding temperatures in the upper half of the central fuel assembly for Experiments L2-5 and L2-3.

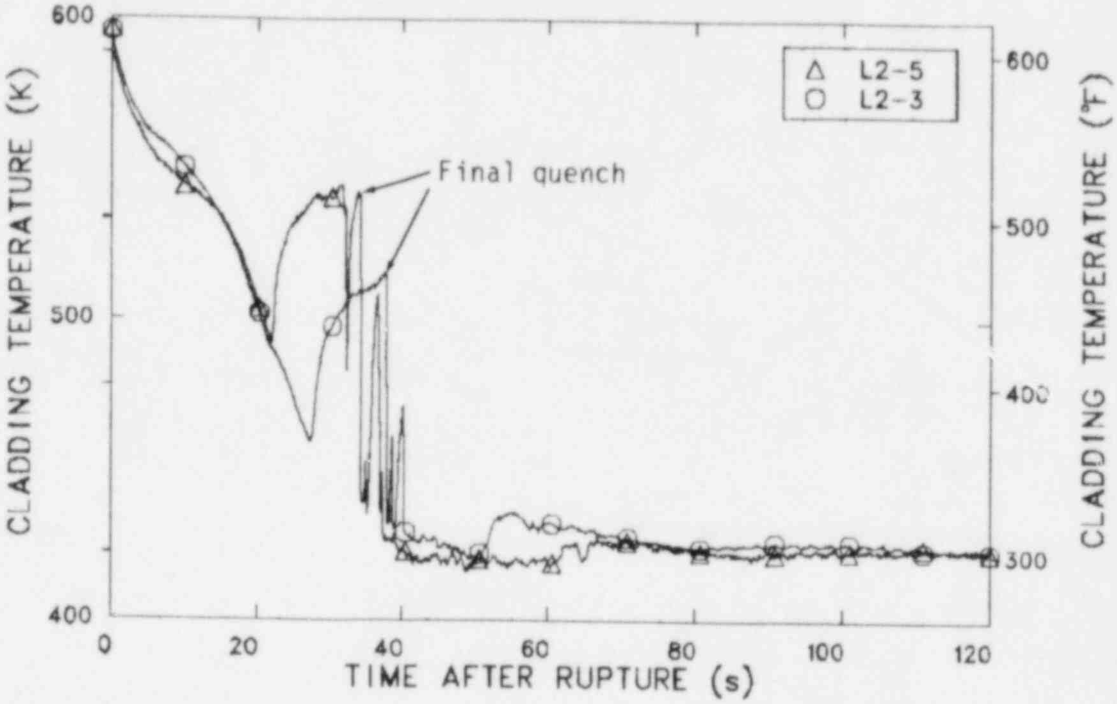

Figure 5. Comparison of cladding temperatures in a peripheral fuel assembly for Experiments L2-5 and L2-3.

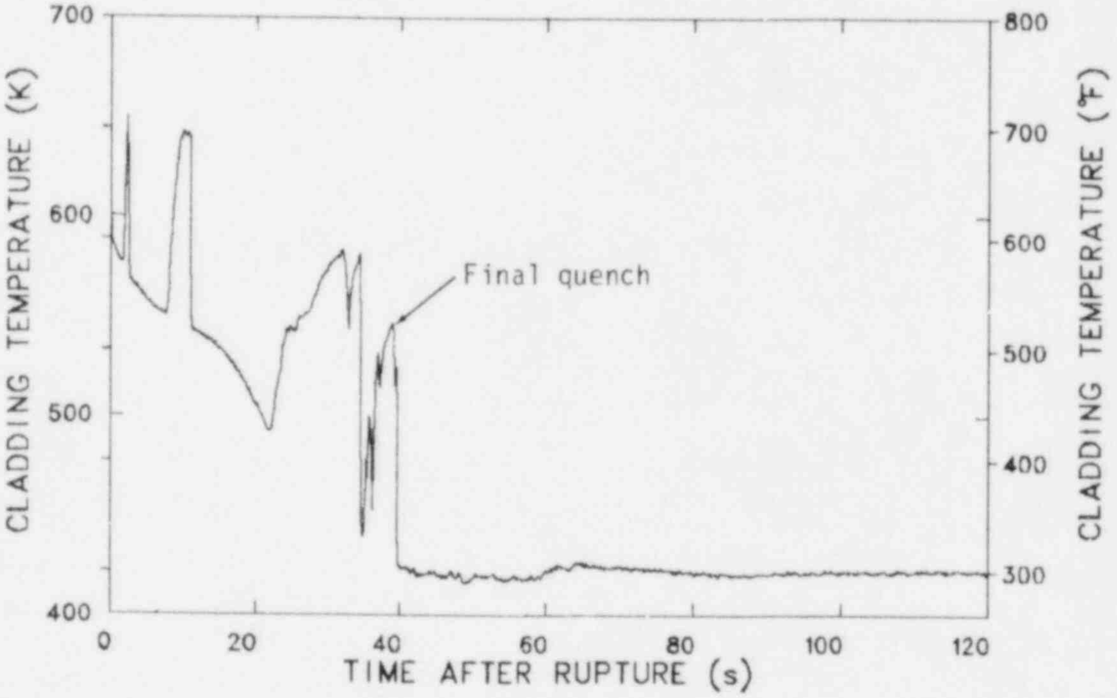

Figure 6. Cladding temperature in a peripheral fuel assembly for Experiment L2-5.

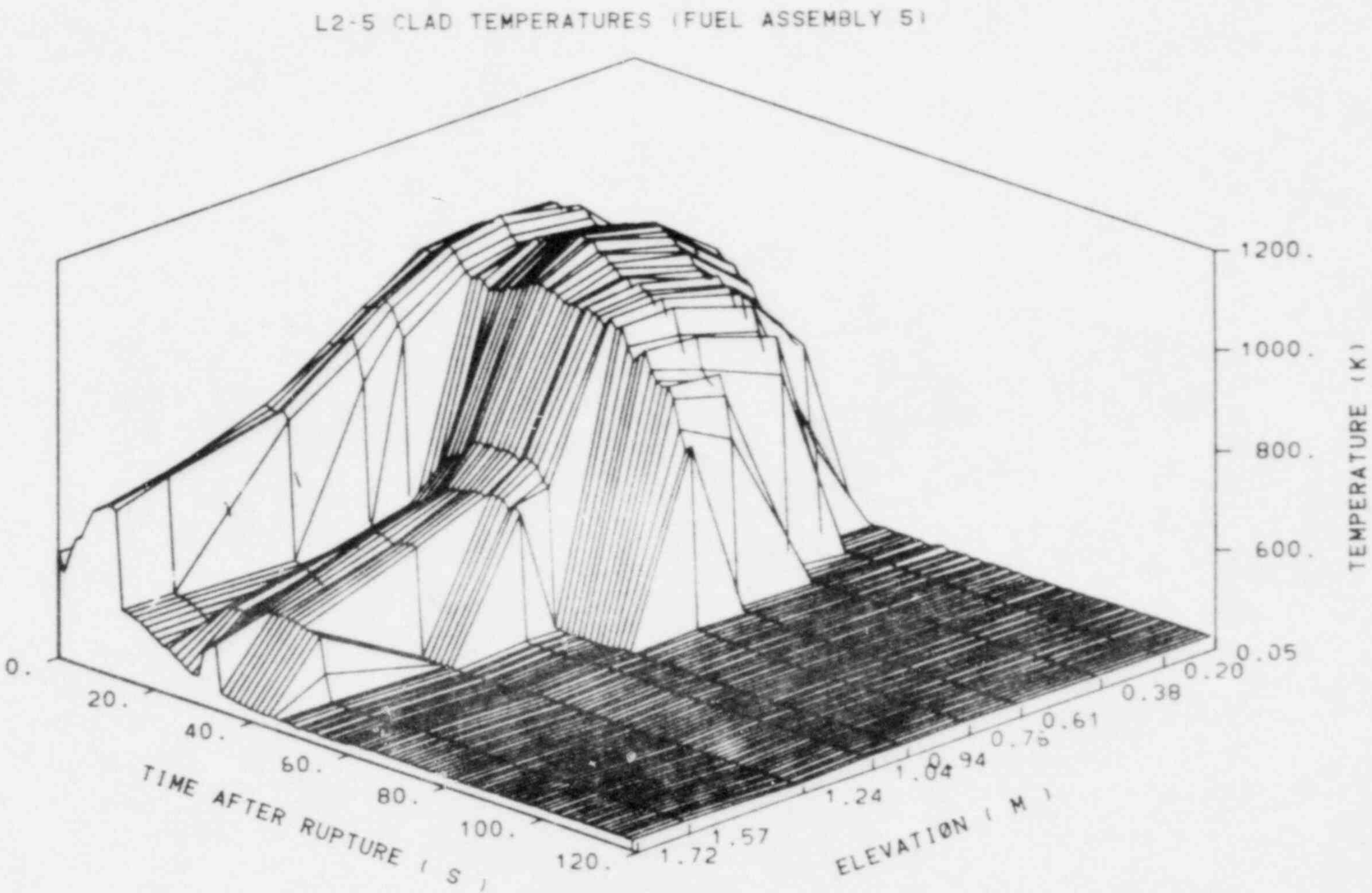

Figure 7. Axial three-dimensional plot of cladding temperature in the central fuel assembly for Experiment L2-5.

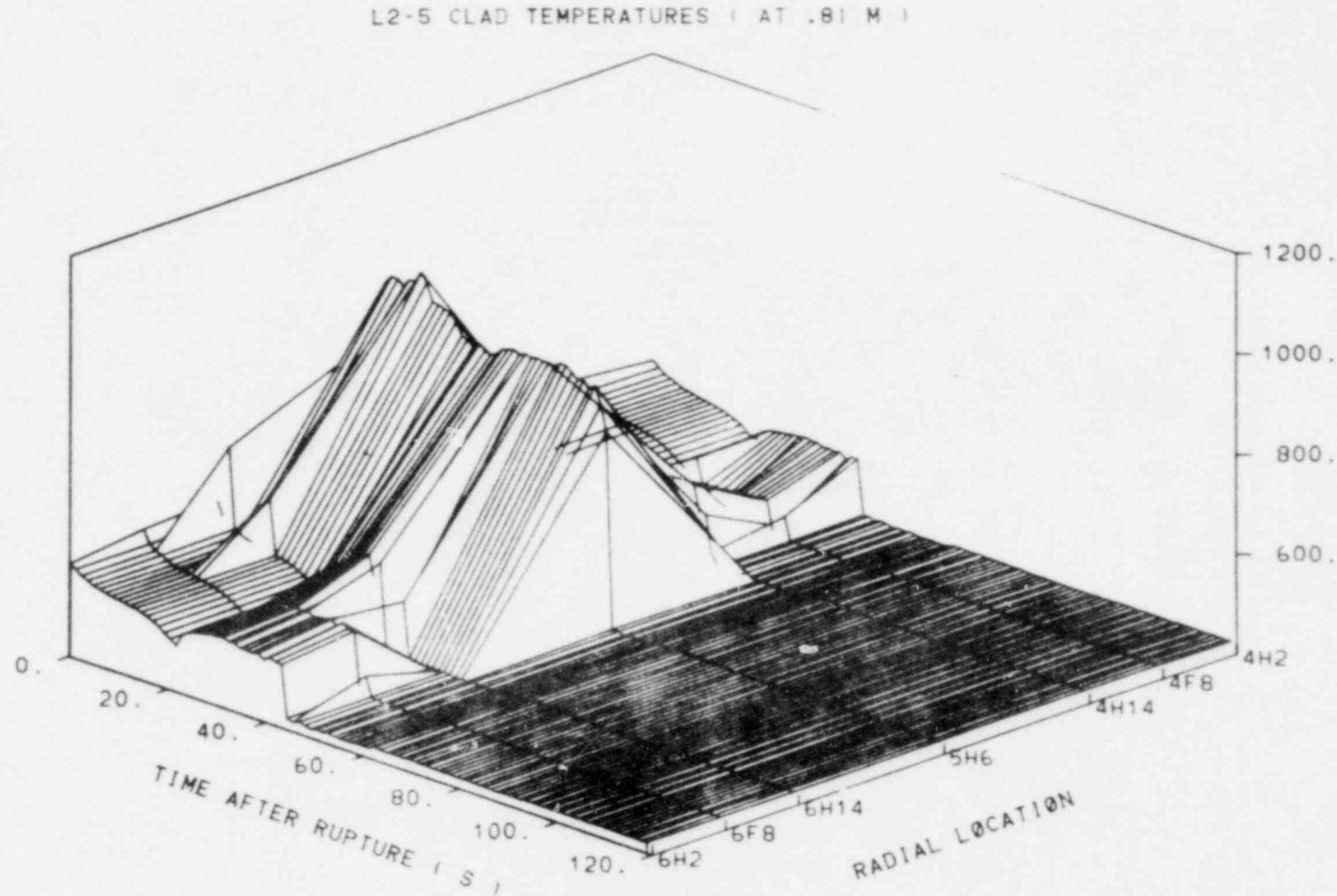

Figure 8. Radial three-dimensional plot of cladding temperature at the 0.81-m (32-in.) elevation for Experiment L2-5.

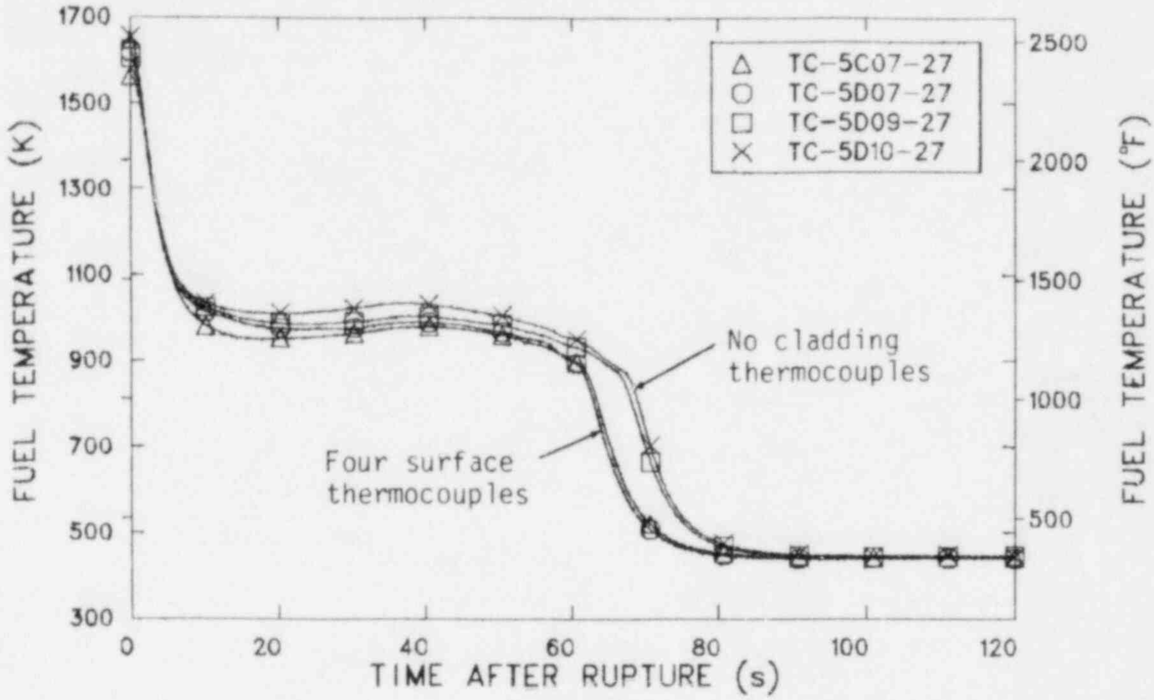

Figure 9. Comparison of fuel centerline temperatures on fuel rods with and without surface cladding thermocouples for Experiment L2-5.

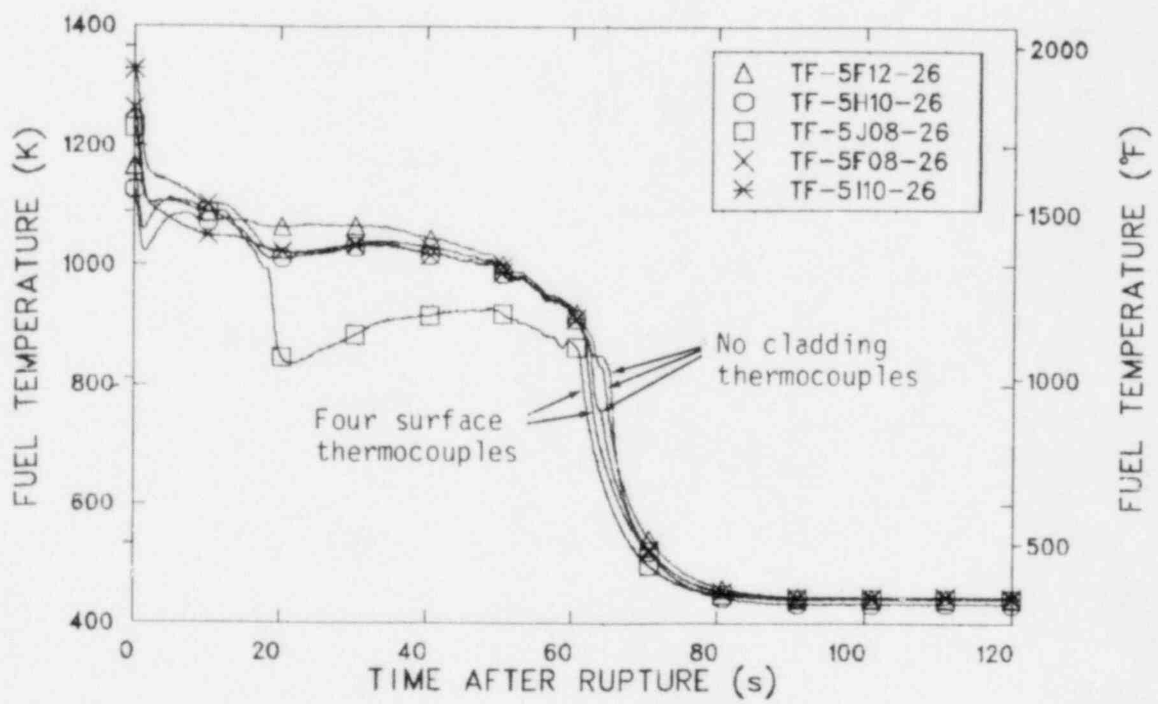

Figure 10. Comparison of fuel surface temperatures on fuel rods with and without surface cladding thermocouples for Experiment L2-5.

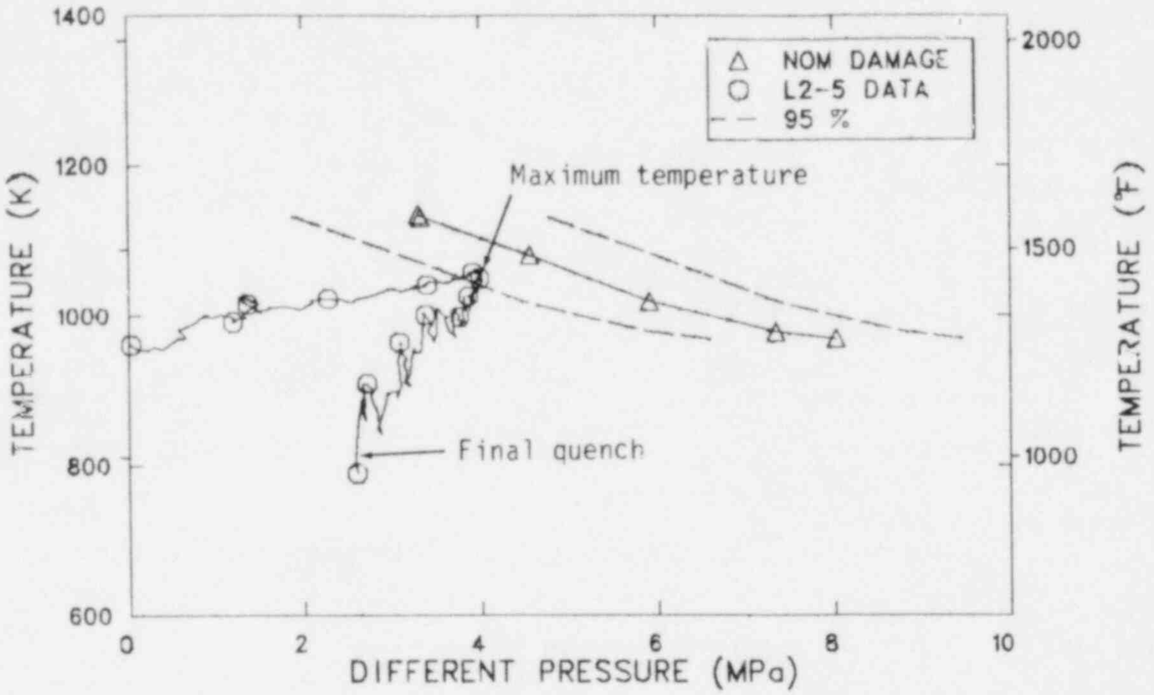

Figure 11. Maximum cladding temperature versus cladding differential pressure compared with fuel rod damage data.

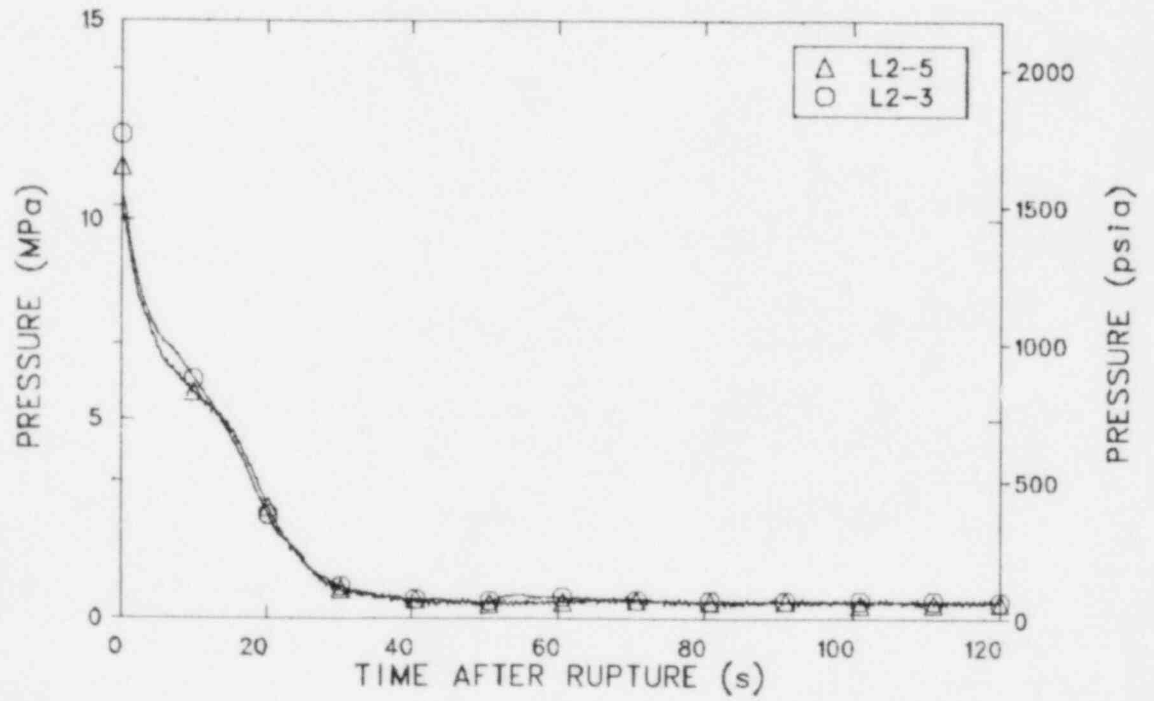

Figure 12. Comparison of upper plenum pressure for Experiments L2-5 and L2-3.

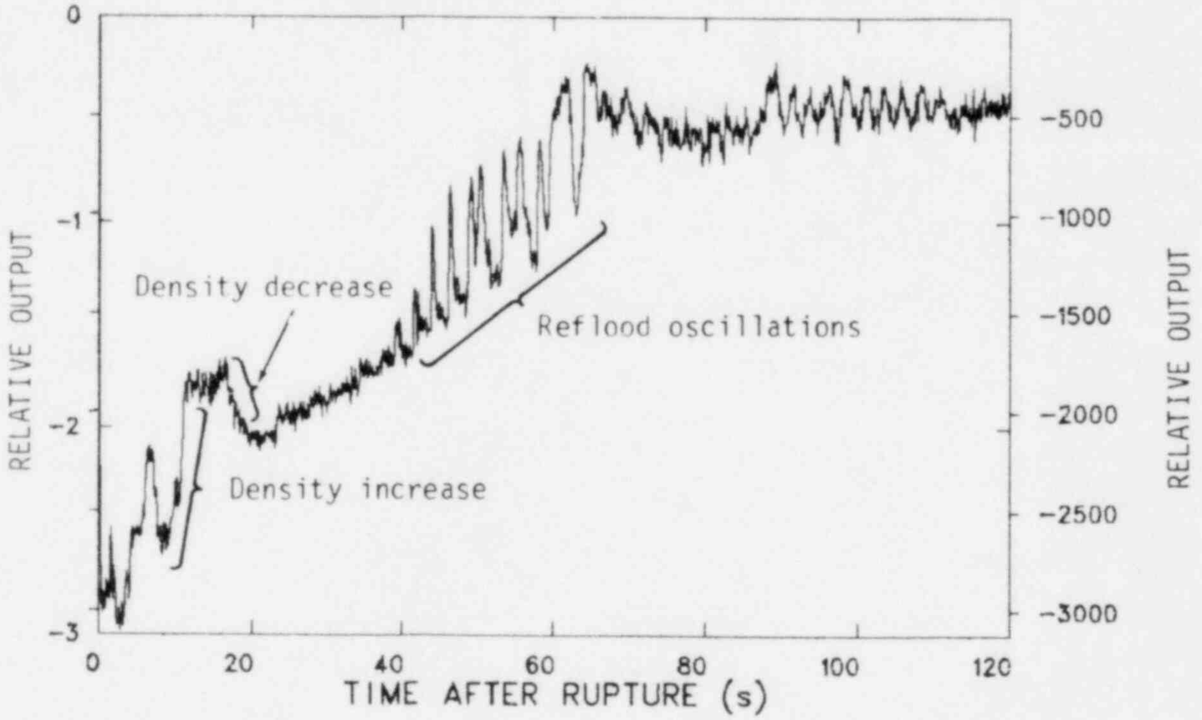

Figure 13. Response of SPND in a peripheral fuel assembly for Experiment L2-5.

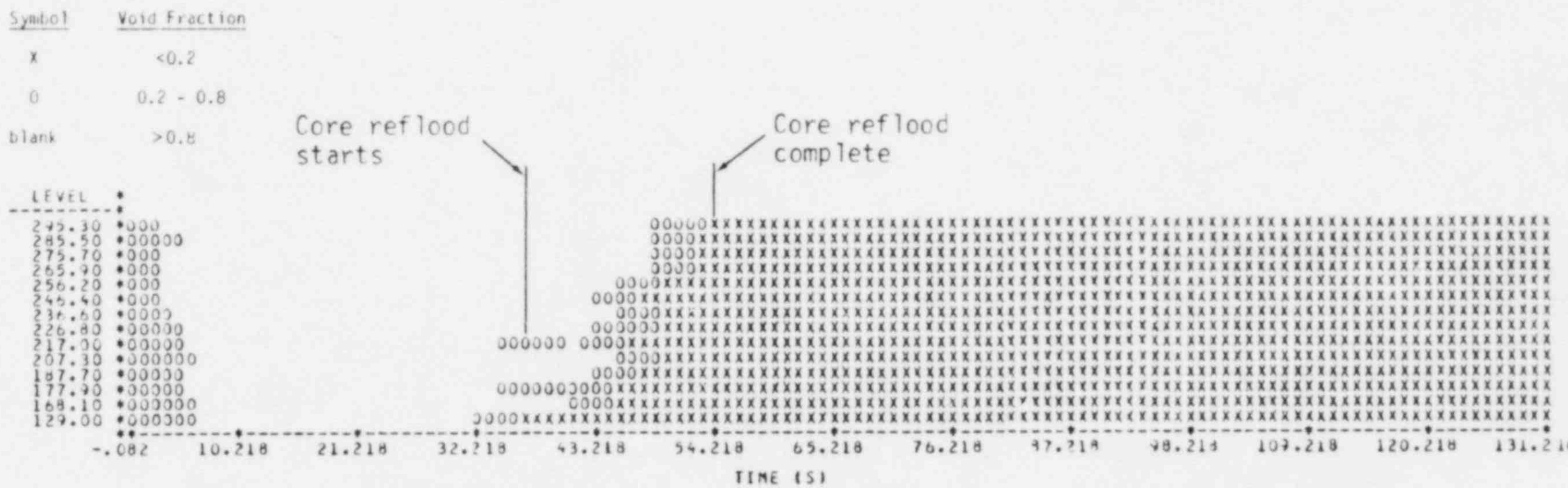

Figure 14. Liquid level in the central fuel assembly for Experiment L2-5.

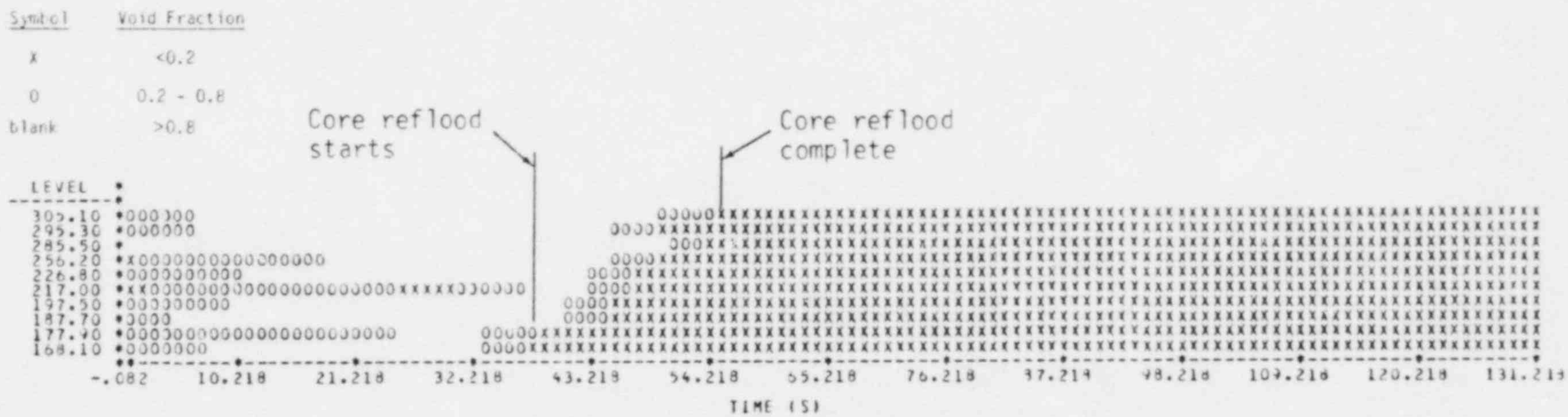

Figure 15. Liquid level in a peripheral fuel assembly for Experiment L2-5.

Figure 16. Liquid level in the downcomer and lower plenum for Experiment L2-5.

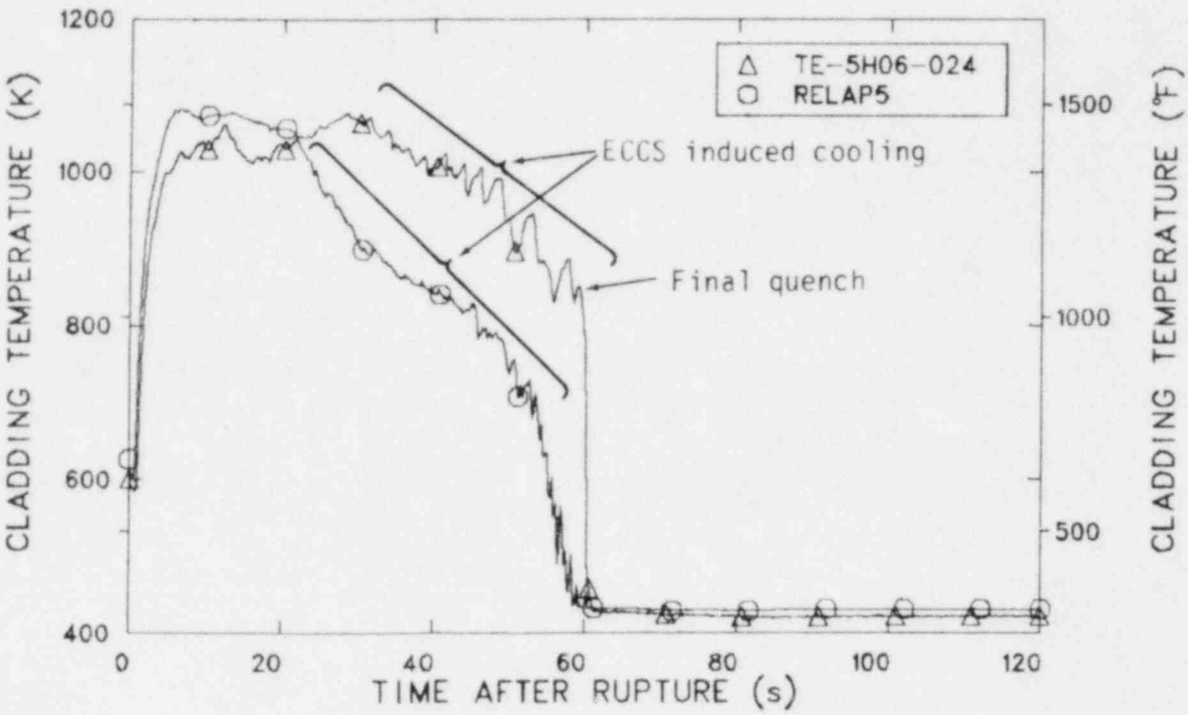

Figure 17. Comparison of maximum cladding temperature with prediction for Experiment L2-5.

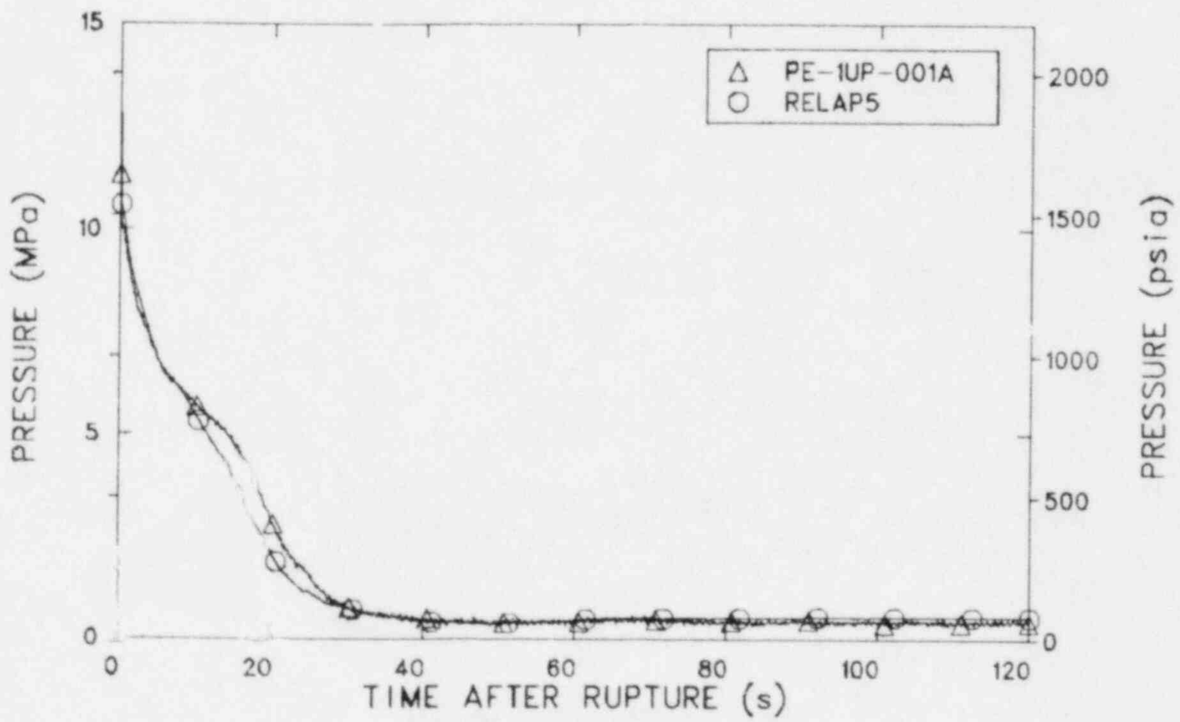

Figure 18. Comparison of upper plenum pressure with prediction for Fxperiment L2-5.

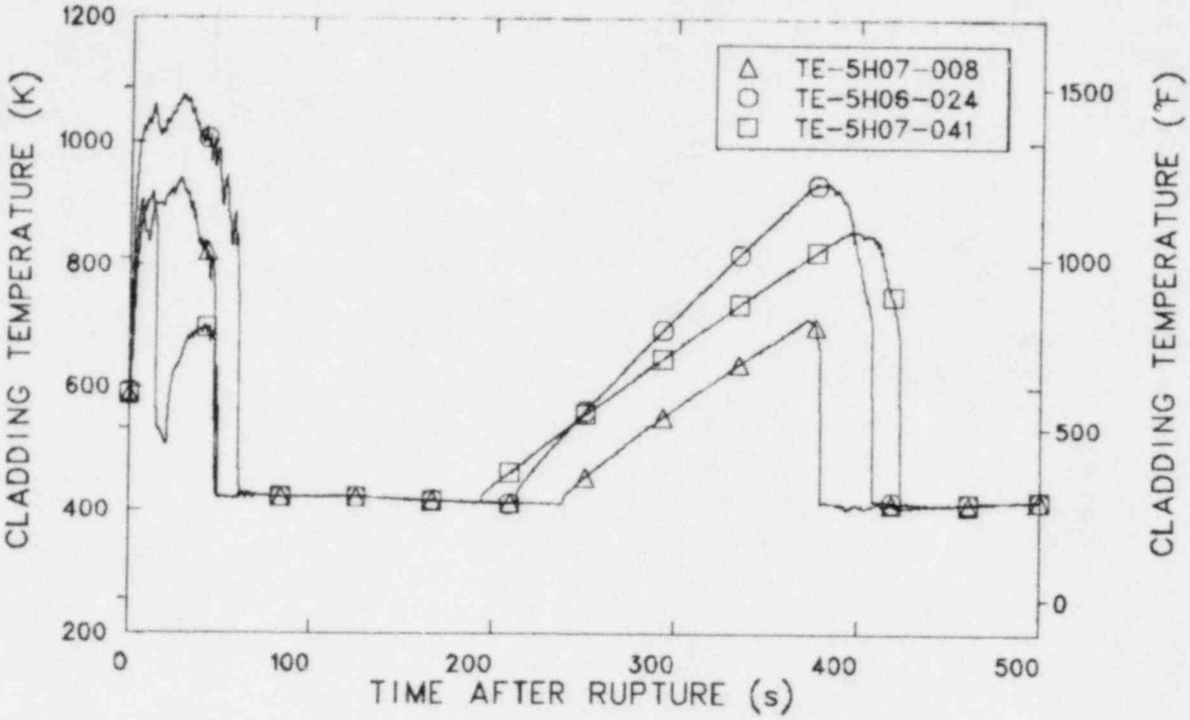

Figure 19. Comparison of cladding temperatures at different elevations during second core heatup for Experiment L2-5.

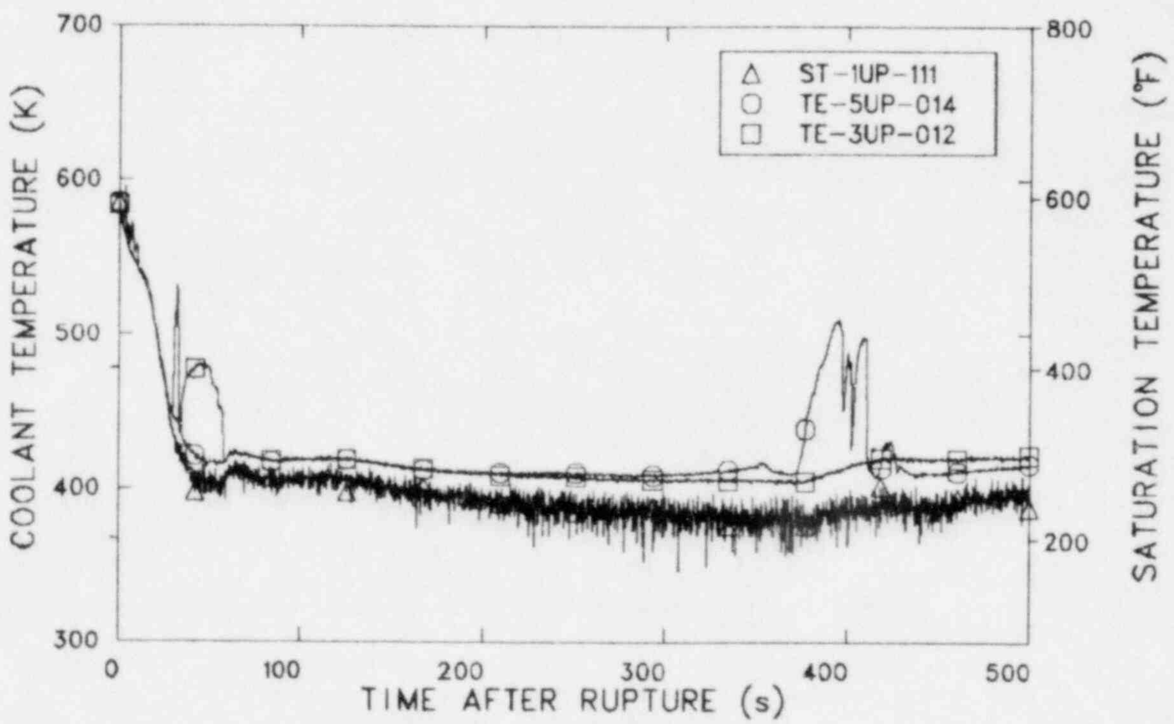

Figure 20. Comparison of core exit and upper plenum fluid temperatures with saturation temperature for Experiment L2-5.

6. REFERENCES

1. D. L. Batt, Quick-Look Report on LOFT Nuclear Experiment L2-2, LOFT-TR-103, December 1978.

2. M. L. McCormick-Barger, Experiment Data Report for LOFT Power Ascension Test L2-2, NUREG/CR-0492, TREE-1322, February 1978.

3. D. L. Reeder, Quick-Look Report on LOFT Nuclear Experiment L2-3, QLR-22-3, May 1979.

4. P. G. Prassinos, B. M. Galusha, D. B. Engleman, Experiment Data Report for LOFT Power Ascension Experiment L2-3, NUREG/CR-0792, TREE-1326, July 1979.

5. R. S. Semken, LOFT Experiment Operating Specification, LOFT Power Ascension Experiment Series L2, Nuclear Experiment L2-5, EGG-LOFT-5696, December 1981.

6. U.S. Atomic Energy Commission, Code of Federal Regulations Title 10, Atomic Energy, Part 50, "Licensing of Production and Utilization Facilities," Appendix K, "ECCS Evaluation Models," Docket No. RM-50-1, January 1976.

7. P. N. Demmie, T. H. Chen, S. R. Behling, Best Estimate Prediction for LOFT Nuclear Experiment L2-5, EGG-LOFT-5869, May 1982.

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

9. E. L. Tolman, "Blowdown Quench Characteristics of Nuclear and Electric Rods--Influence of Cladding Surface Thermocouples", Ninth Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, October 26-30, 1981, NUREG/CP-0024.

10. L. J. Siefken et al., FRAP-T6: A Computer Code for the Transient Analysis of Oxide Fuel Rods, NUREG/CR-2148, EGG-2104, May 1981.

11. C. S. Olsen, Burst Strength of Zircaloy Tubing with Embedded Thermocouples, EG&G Idaho Report, LO-14-82-081, May 1982.

12. Y. L. Cheung and P. Griffith, Gravity Reflood Oscillations in a Pressurized Water Reactor, WCAP-8238, Westinghouse Electric Corporation, February 1980.

APPENDIX A

LOFT SYSTEM GEOMETRY

APPENDIX A

LOFT SYSTEM GEOMETRY

The Loss-of-Fluid Test (LOFT) system geometry is shown in Figure A-1. Figure A-2 shows the LOFT pressurizer and instrumentation. Figure A-3 shows the location of all LOFT cladding surface thermocouples, and Figure A-4 is a map of the central fuel assembly showing the location of all instrumentation.

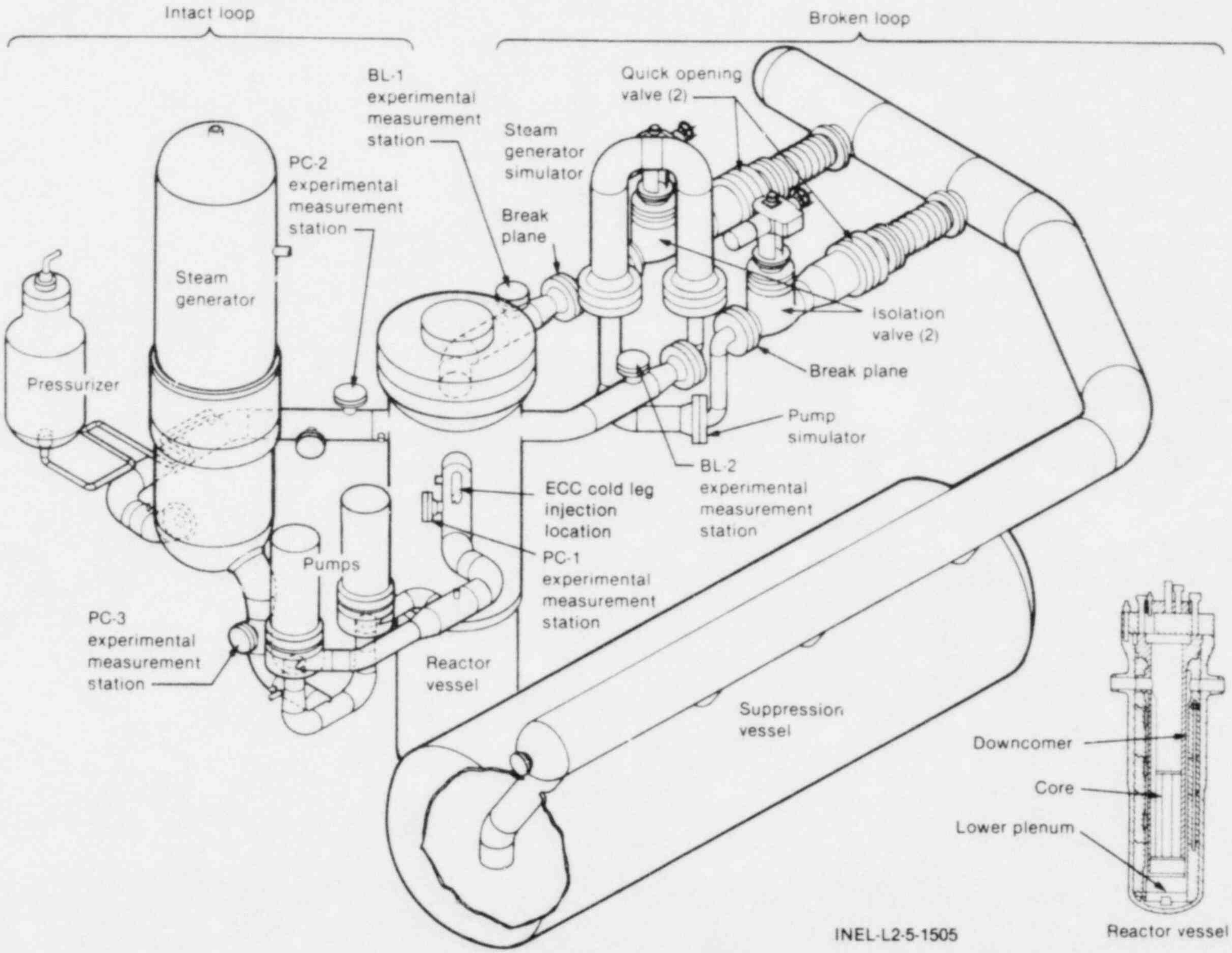

Figure A-1. Axonometric projection of LOFT system.

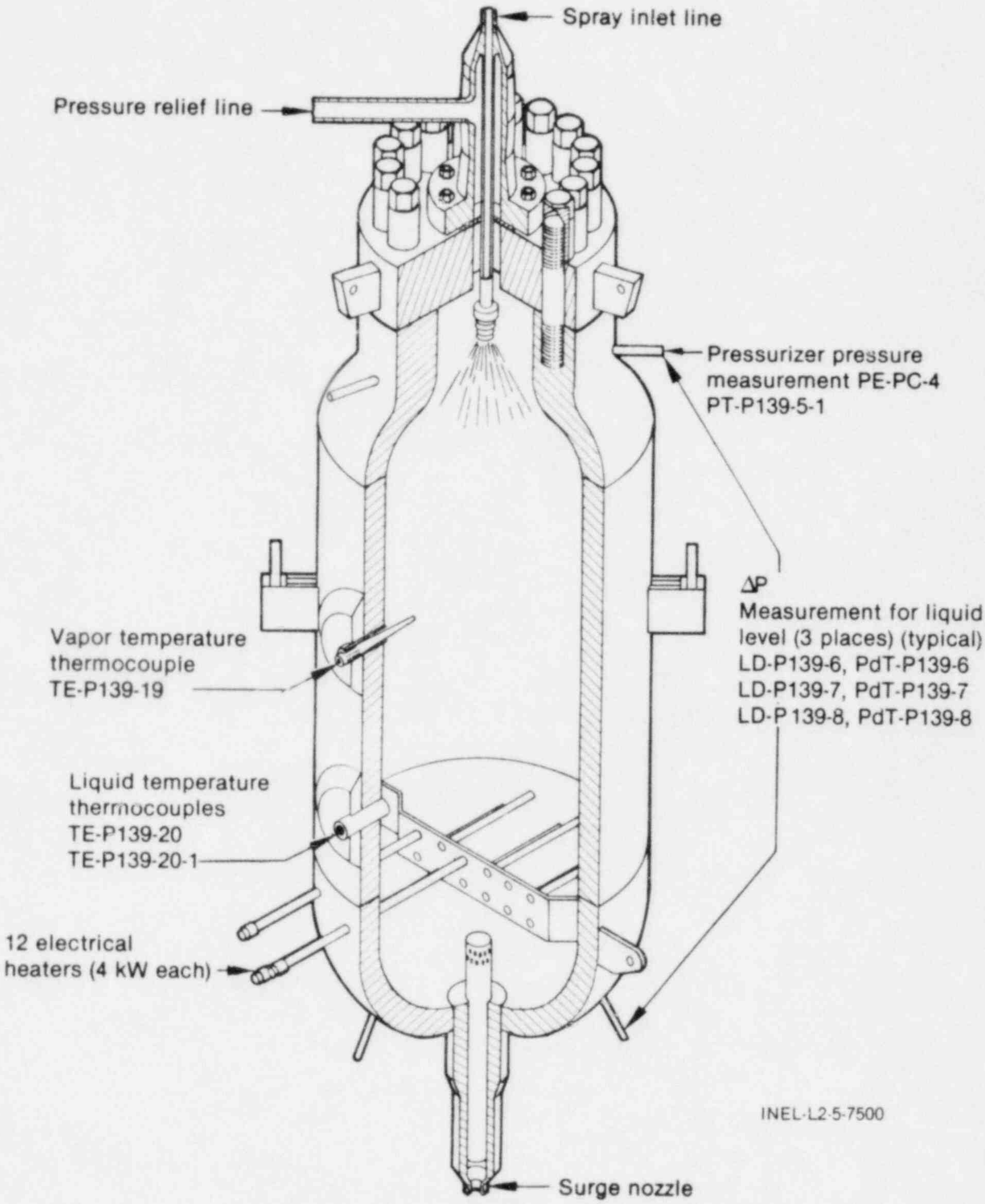

Figure A-2. LOFT pressurizer and instrumentation.

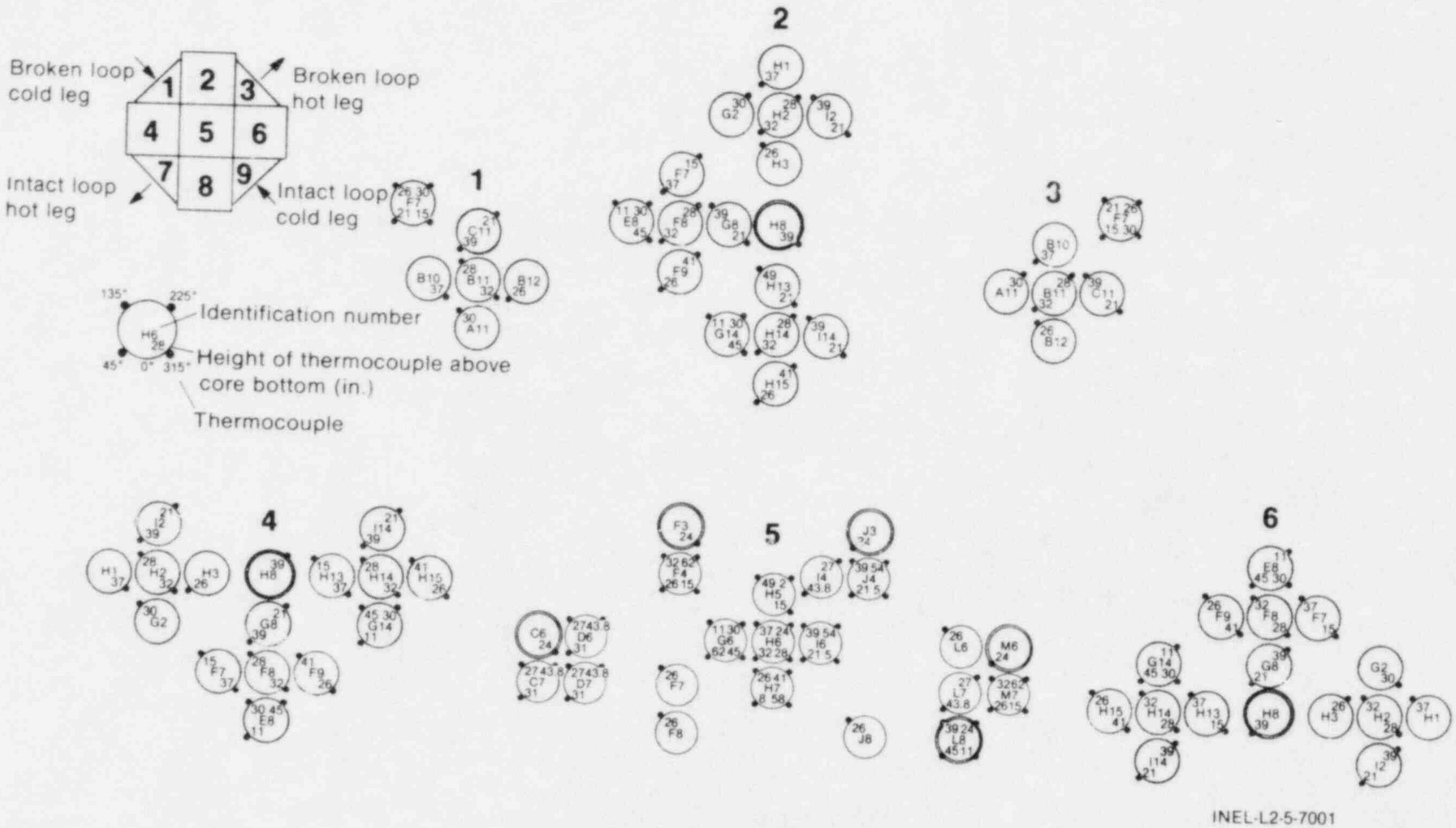

Figure A-3. LOFT fuel rod cladding thermocouples.

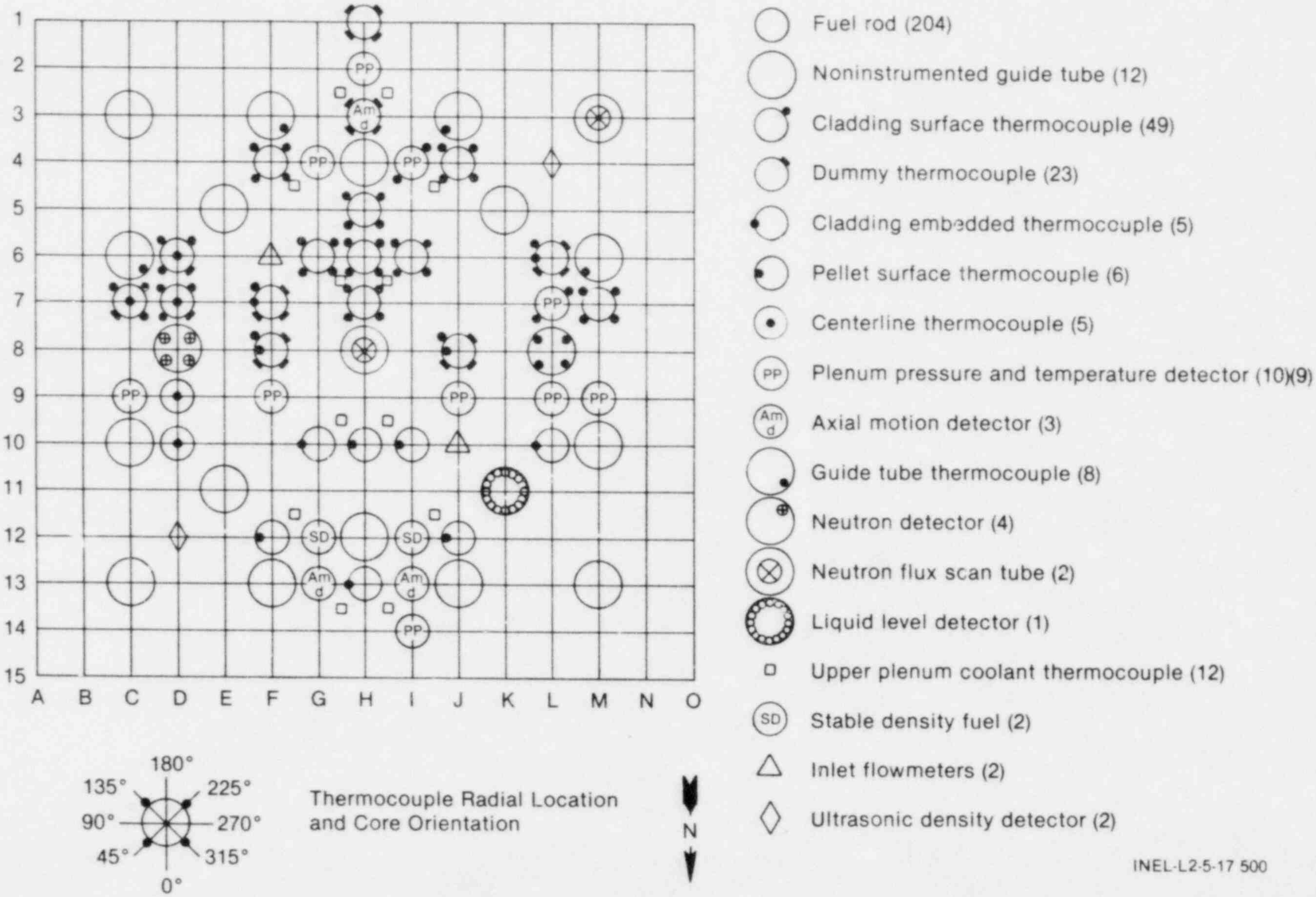

Figure A-4. LOFT central fuel assembly instrumentation.