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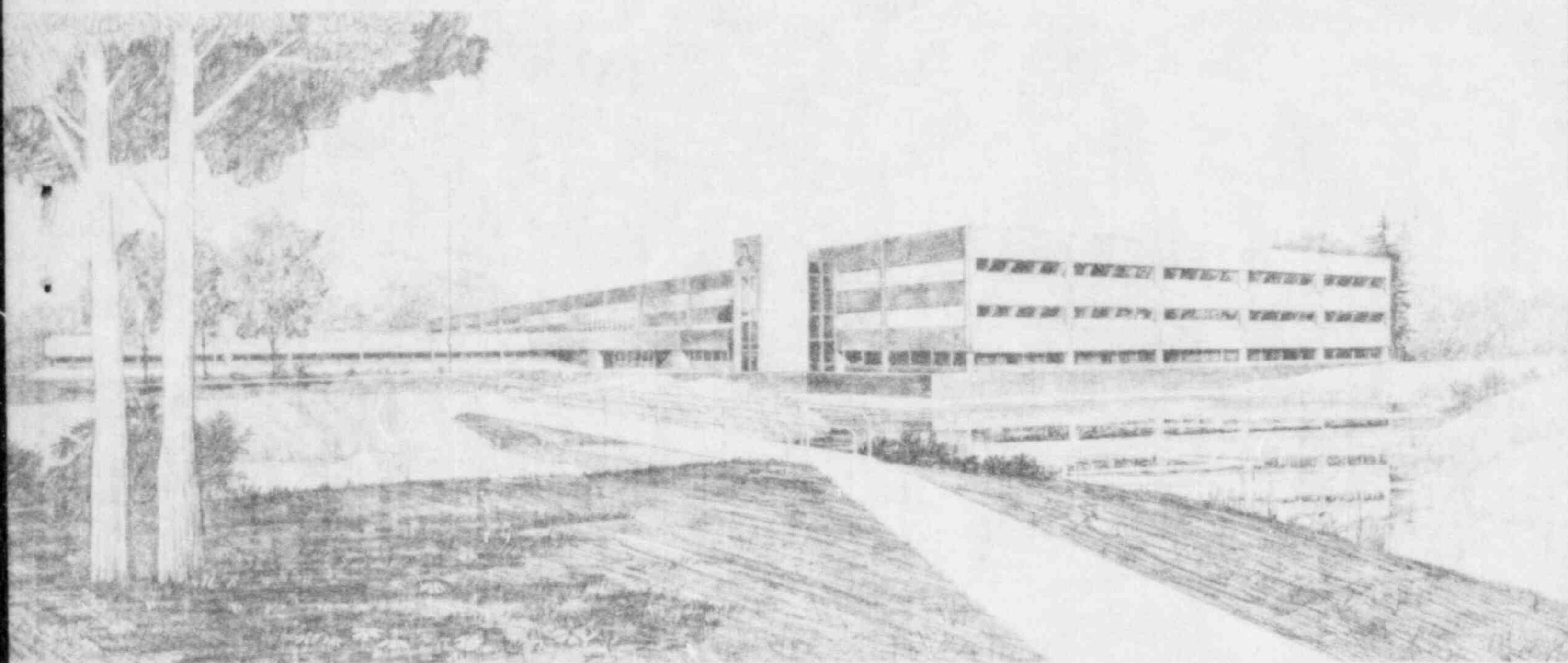

Idaho National Engineering Laboratory
Operated by the U.S. Department of Energy

Summary of the Nuclear Regulatory Commission's LOFT Program Research Findings

Charles L. Nalezny

April 1985

Prepared for the
U.S. Nuclear Regulatory Commission
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SUMMARY OF THE NUCLEAR REGULATORY COMMISSION'S LOFT PROGRAM RESEARCH FINDINGS

Charles L. Nalezny

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Idaho Falls, Idaho 83415

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FIN No. A6048

ABSTRACT

This document is a summary of the main research results of the Loss-of-Fluid Test (LOFT) Program relative to code assessment, code development, licensing, rulemaking, safety technology, and reactor operations. The LOFT facility is a 50 MW(t) pressurized water reactor (PWR) system with instruments that measure and provide data on the system thermal-hydraulic and nuclear conditions. The transient response of the LOFT system to accident events is similar to large [~1000 MW(e)] commercial PWRs. The main objectives of the LOFT Experimental Program were to qualify the engineered safety systems used in commercial PWRs and to verify the computer codes used in safety analyses.

The LOFT Program contributed to the improvement of computer codes used to predict the response of commercial PWRs, demonstrated the adequacy of engineered safety systems, and contributed to improved understanding of PWR accident phenomena, particularly those associated with the evaluation models in Appendix K to 10 CFR 50 (the "ECCS rule").

NRC FIN No. A6048—LOFT Experimental Program

EXECUTIVE SUMMARY

In six years of operation, the Loss-of-Fluid Test (LOFT) Experimental Program has produced a large volume of data. The results of individual experiments or experiment series have been documented in technical reports, research information letters, journal articles, and other summary reports. This document presents a concise summary of the main research results of the LOFT Program related to thermal-hydraulic modeling, licensing (Appendix K to 10 CFR 50), safety issues, safety technology, rulemaking, and accident management.

The LOFT facility is a 50 MW(t) pressurized water reactor (PWR) system that was designed to simulate the major components and system responses of a commercial, four-loop PWR [~1000 MW(e)] during a loss-of-coolant accident (LOCA). The volume of the LOFT facility is ~1/50th of a commercial PWR. The LOFT Experimental Program was conducted by EG&G Idaho, Inc., at the Idaho National Engineering Laboratory (INEL). The program was sponsored by the United States Nuclear Regulatory Commission (USNRC) as part of their Water Reactor Safety Research Program and was administered by the United States Department of Energy (DOE).

Since 1976, 43 nonnuclear and nuclear experiments were performed in the LOFT facility. The first 10 experiments were nonnuclear "miniblowdowns" performed to check out the system, especially the blowdown suppression system. The next six experiments were nonnuclear, large-break LOCA simulations. The nonnuclear large-break experiments were designed to (a) explore subcooled and saturated blowdown phenomena, (b) provide an information base from which nuclear core and steam generator heat transfer effects could be established during nuclear experiments, (c) determine the effects of emergency core coolant (ECC) injection into the primary system, and (d) establish the history of ECC delivery into the lower plenum.

The remaining 27 experiments consisted of the following:

1. Three were nuclear large-break (200% double-ended cold leg break) experiments. The main objectives of the nuclear large-break experiments were to:

    a. Provide a comparison of nonnuclear and nuclear experimental data to determine and separate the effects of nuclear heat on blowdown phenomena

    b. Provide data on thermal-hydraulic and fuel behavior

    c. Use the data to evaluate and verify computer models and nuclear safety codes.

2. Nine were nonnuclear and nuclear small-break experiments which were designed to:

    a. Determine the important thermal-hydraulic, operational, and neutronic phenomena during a variety of small-break LOCAs

    b. Evaluate the effectiveness of plant recovery methods, diagnostic procedures, and emergency core cooling systems (ECCSs) during slow depressurization transients.

3. Two experiments were nuclear intermediate-break experiments intended to:

    a. Determine the effectiveness of degraded ECCSs during an intermediate-size-break LOCA

    b. Determine and understand the core cooling and system hydraulic behavior for a break size that may include characteristics from both small and large breaks.

4. Thirteen were anticipated transients and anticipated transients with multiple failures. The objectives of the anticipated transient experiments were to

    a. Evaluate plant and control system performance during anticipated transients such as loss of steam load and loss of feedwater

    b. Provide data to assist in analyzing the relationship between the behavior of

LOFT and the behavior of a commercial PWR during operational transients

c. Continue development and testing of the operational diagnostic and display system (ODDS).

The objective of the anticipated transient with multiple failure experiments was to study multiple-failure scenarios such as an anticipated transient without scram (ATWS) initiated by a loss-of-offsite power.

The LOFT program contributed to the development and assessment of thermal-hydraulic models used in computer codes such as RELAP4 and RELAP5, which are used to simulate accidents in nuclear plants and to improved modeling techniques used to represent components and systems in plants. Examples of important thermal-hydraulic models include installation of a postcritical heat flux (post-CHF) heat transfer correlation (Biasi-Zuber) that was needed to predict the rewet that occurred early in the blowdown during LOFT large-break Experiments L2-2 and L2-3. An example of an improved modeling technique is the development of a split-downcomer model which was developed to more directly approximate the azimuthel asymmetry observed in downcomer flow during large-break experiments.

Results from the large-break experiments were used to evaluate the conservative nature of several features of the ECCS evaluation models contained in Appendix K to 10 CFR 50. Specific features discussed are the discharge model, end of blowdown, CHF heat transfer, and post-CHF heat transfer. The discharge model that produced the best agreement with LOFT data was the Henry-Fauske model for subcooled conditions and the homogenous equilibrium model (HEM) for saturated conditions with a break-flow multiplier of 0.848 for both. The Henry-Fauske/HEM combination produced better agreement with LOFT results than the Moody model which is specified in Appendix K. The conservativeness of the assumption requiring bypass of all ECC until the end of blowdown was demonstrated during the nonnuclear and nuclear large-break experiments. During these experiments, ECC was able to reach the lower plenum during the blowdown. The experiments showed that as much as 80% of the ECC reached the lower plenum. Finally, the assumption prohibiting the return to nucleate boiling under all conditions before the reflood phase was shown to be invalid under the conditions of Experiments L2-2 and L2-3. During Experiments L2-2 and L2-3, the core was quenched during the early part of the blowdown by a flow reversal.

An assessment was made of the impact of LOFT results on two unresolved safety issues: shutdown decay heat removal requirements and station blackout. Small-break Experiment L3-7 and the recovery phase of Experiment L9-1/L3-3 (an ATWS initiated by a loss of feedwater) verified that shutdown decay heat could be transferred from the primary system to the steam generator by single-phase and two-phase natural circulation. A station blackout is defined as a loss-of-offsite power followed by a failure to start the emergency diesels. In the event of a station blackout, the capability to cool the core depends on systems which do not depend on alternating current power, such as a steam-driven auxiliary feedwater system. LOFT Experiment L9-4 simulated an ATWS combined with a station blackout. The LOFT auxiliary feedwater system, which has the scaled capacity of the Zion-1 steam-driven auxiliary feedwater system, was able to remove decay heat and cool the core after the initial pressure surge was maintained within acceptable safety levels by the safety relief valve (SRV).

LOFT research results contributed data that could help resolve four issues related to the accident at Three Mile Island—Unit 2 (TMI-2): human-machine interface, interruption of ECCS after a LOCA, instrumentation to follow the course of an accident, and revised small-break LOCA methods to show compliance with Appendix K to 10 CFR 50. The human-machine interface problem was addressed by the augmented operator capability (AOC) program. The principal results of the LOFT AOC program was the development of computer generated color cathode ray tube (CRT) displays that show the functional relationship among a number of related parameters. Typical displays show the limits of primary system temperature based on reactor power and the current operating point, and the history of primary system temperature versus pressure. The temperature-pressure display also shows the normal operating range, the saturation line, and the steam region. During LOFT Experiments L3-5/L3-5A and L2-5, the ECCS was interrupted. During L3-5/L3-5A, the ECCS was interrupted intentionally to achieve specific program objectives. During L2-5, the ECCS was terminated as planned, but had to be

reinitiated after the core boiled down and the core heated up. The results showed that following a large break (Experiment L2-5) ECC injection must not be interrupted for an extended period or the core would be uncovered and would heatup excessively, while a longer interruption of ECC could be tolerated after a small break (Experiment L3-5/L3-5A) if alternate means of decay heat removal were available. The LOFT experiments demonstrated the overall adequacy of the nuclear plant instrumentation requirements specified in NRC Regulatory Guide 1.97. Revised small-break LOCA methods to show compliance with Appendix K deals with documentation requirements for the ECCS evaluation models. LOFT data are presented which can be used to qualify vendor condensation heat transfer models, and natural circulation models.

The LOFT Experimental Program made a significant contribution to four areas of safety technology: fuel prepressurization, liquid level detectors, thermocouples (TCs) (fuel cladding and core exit), and reactor coolant pump current monitoring. Data from the LOFT experiments indicate that unpressurized fuel may perform as well as prepressurized fuel if the fuel pellets are designed and manufactured properly. Specific fuel pellet design features that are important are: chamfered, length to width ratio, density, and pore size. LOFT uses conductivity probes and differential pressure transducers to measure liquid level. In addition, self-powered neutron detectors (SPNDs) have been shown to respond to changes in liquid density. Conductivity probes respond best when the liquid-vapor interface is clean. Droplets and froth tend to wet the probes and give false readings. Differential pressure transducers can be used to measure liquid level if density changes in the reference leg and vessel are compensated for. The important contribution of fuel cladding TCs to the LOFT Program is discussed in detail. Fuel cladding TCs provided the most direct and reliable information on core conditions during experiments involving core uncovery. Under certain experiment conditions, however, the LOFT core exit TCs did not provide reliable information on conditions in the core during core uncovery experiments. During LOFT small-break Experiment L3-6, primary coolant pump monitoring provided a reliable measure of primary system coolant inventory. A display of cold leg temperature versus primary coolant pump current was developed and shown to be useful for differentiating between an overcooling transient and a small-break LOCA.

A considerable amount of LOFT data has been accumulated relative to rulemaking, specifically the ATWS rule and the ECCS rule. Two LOFT experiments addressed the ATWS issue: Experiment L9-3, which was an ATWS initiated by a loss of feedwater, and Experiment L9-4, which was an ATWS initiated by a loss-of-offsite power. The LOFT ATWS experiments demonstrated that a properly sized SRV could relieve the initial pressure surge caused by the power mismatch between the reactor and the steam generator and that the steam generator or power-operated relief valve (PORV) could remove decay heat after the reactor is shut down. The LOFT data were also used to evaluate the two-phase approach proposed by the NRC in 1978 for updating the ECCS rule. LOFT data supports basing Phase 1 on more realistic models of blowdown phenomena such as discharge, end of blowdown, and return to nucleate boiling.

During the LOFT small-break and anticipated transient experiment series, several accident management procedures were evaluated. The techniques were: steam generator feed and bleed, primary system feed and bleed, and operation of the primary coolant pumps. The steam generator was used successfully as a heat sink during 12 experiments that varied from small-break LOCEs to an ATWS initiated by a loss of feedwater. LOFT Experiments L3-5 and L3-6 were used to investigate the optimum time to trip the primary coolant pumps during a small-break LOCA. The results of the LOFT study were that the pumps should be tripped early in the transient, and as discussed previously, pump current could be used as a measure of the optimum time for tripping the pumps. Pump current has been shown to correlate well with void fraction which is a measure of primary system coolant inventory.

There are a number of recommendations for future applications of LOFT data. The two areas where LOFT data could be used very effectively are accident diagnosis and licensing with best estimate models. Accident diagnoses techniques and procedures can be qualified using data from LOFT experiments. Using actual data that corresponds to measurements available in commercial PWRs will allow realistic assessments of the diagnostic techniques and will indicate where more information is required to improve diagnostic capability. Analytical codes and models are at the stage of development where they could replace the present evaluation models used in licensing. LOFT results could be used to develop uncertainty factors and safety factors that would provide the same or greater level of safety to the public.

ACKNOWLEDGMENTS

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CONTENTS

NOMENCLATURE

Acronyms

AEC	Atomic Energy Commission (United States)
AFW	Auxiliary feedwater
AOC	Augmented operator capability
ATWS	Anticipated transient without scram
BST	Blowdown suppression tank
BWR	Boiling water reactor
BWST	Borated water storage tank
CE	Combustion Engineering
CRT	Cathode ray tube
CHF	Critical heat flux
DHR	Decay heat removal
DNB	Departure from nucleate boiling
DOE	Department of Energy (United States)
DTT	Drag disk-turbine transducer
ECC	Emergency core coolant
ECCS	Emergency core cooling system
EM	Evaluation model
ESF	Engineered safety feature
FRD	Federal Republic of Germany
HEM	Homogeneous equilibrium model
HPIS	High-pressure injection system
INEL	Idaho National Engineering Laboratory
JAERI	Japan Atomic Energy Research Institute
KWU	Kraftwerkunion
L/D	Length to diameter (ratio)
LLT	Liquid level transducer
LOBI	Loop Blowdown Investigation
LOCA	Loss-of-coolant accident
LOCE	Loss-of-coolant experiment
LOFA	Loss-of-feedwater accident
LOFT	Loss-of-Fluid Test
LPIS	Low-pressure injection system
LSTF	Large-Scale Test Facility
LVDT	Linear variable displacement transducer
MLHGR	Maximum linear heat generation rate
NRC	Nuclear Regulatory Commission (United States)
NSSS	Nuclear steam supply system
PCI	Pellet-cladding interaction
PKL	Primarkeislauf
PNA	Pulsed neutron activation
PORV	Power-operated relief valve
PST	Pressure suppression tank
PWR	Pressurized water reactor
QOBV	Quick-opening blowdown valve
RDT	Reactor Development and Technology
RHR	Residual heat removal
RNB	Return to nucleate boiling
ROSA	Rig of Safety Assessment
RTD	Resistance temperature detector

SDHR	Shutdown decay heat removal
SIG	Special Inquiry Group
SPND	Self-powered neutron detector
SRV	Safety relief valve
STEP	Safety Test Engineering Program (INEL)
TAN	Test Area North (INEL)
TMI-2	Three Mile Island—Unit 2
TSC	Technical Support Center
USI	Unresolved safety issue
WCL	Westinghouse Canada Limited

Conversion Factors for SI[a] and U.S. Customary Units

From	To	Multiply By
K	°F	1.8 K - 459.67
kg/s-m^2	lb/ft^2-h	737.4
kg/s	lb/s	2.2046
kW/m	kW/ft	0.3048
m	ft	3.2808
MPa	psi	145.05
W/m^2	Btu/ft^2-h	0.3169
L/s	gpm	15.85

a. SI—System of International Units.

SUMMARY OF THE NUCLEAR REGULATORY COMMISSION'S LOFT PROGRAM RESEARCH FINDINGS

1. INTRODUCTION

The Loss-of-Fluid Test (LOFT) Program discussed in this document was one of several water reactor research experimental programs conducted by EG&G Idaho, Inc., at the Idaho National Engineering Laboratory (INEL). The program was sponsored by the United States Nuclear Regulatory Commission (USNRC) as part of their water reactor safety research and was administered by the United States Department of Energy (DOE). The LOFT facility is an experimental system containing a nuclear pressurized water reactor (PWR) that was designed to simulate the major components and system responses of a commercial PWR during loss-of-coolant accidents (LOCAs) and during anticipated transients caused by abnormal PWR operations.

The LOFT experimental program has produced a large amount of data investigating PWR behavior during a variety of simulated accident conditions. These data have been documented in numerous technical and topical reports, at conferences and in the subsequent proceedings publications, and in scientific and technical journals. This document includes a concise presentation of the most important LOFT research results (experimental and analytical) and a discussion of how the results relate to resolution of USNRC regulatory concerns for PWRs. Specifically, it discusses how LOFT results relate to PWR LOCA analyses for (a) thermal-hydraulic modeling, (b) Appendix K to 10 CFR 50, (c) safety issues, (d) Three Mile Island, (e) safety technology, (f) rulemaking, and (g) accident management procedures.

Section 2 presents a history of the LOFT Program, a facility description, and an outline of major experiments. Next, the LOFT scaling rationale and the relationship of LOFT to a commercial PWR are discussed in Section 3. This is followed in Section 4 by a discussion of the impact of LOFT results on thermal-hydraulic modeling. Section 5 presents LOFT data that confirm the conservativeness of the acceptance criteria for emergency core cooling systems (ECCSs) (10 CFR 50.46) and the Appendix K evaluation criteria for ECCS models. Section 6 presents LOFT results that relate to unresolved and resolved safety issues. Section 7 deals with issues related to the small-break LOCA at Three Mile Island-Unit 2 (TMI-2). This section also presents results important to the resolution of four generic issues, and two TMI Action Plan items. Section 8 deals with the impact of LOFT on such safety technology items as fuel prepressurization, thermocouples, liquid level detectors, and pump current monitoring. Section 9 discusses the use of LOFT results in the USNRC rulemaking process. This is followed in Section 10 by a summary of accident management procedures that have been used in the various LOFT experiments and a discussion of their effectiveness. Section 11 lists a number of future applications for LOFT results. Section 12 is an overall summary of the major LOFT accomplishments and conclusions.

2. HISTORY AND SCOPE OF THE LOFT PROGRAM

The initial planning for an experimental program to investigate maximum credible hazards in PWRs began in March 1962, when the United States Atomic Energy Commission (AEC), Division of Reactor Development, initiated the Safety Test Engineering Program (STEP) as part of their Nuclear Safety Program. The STEP tests included kinetic tests and engineering field tests. The engineering field tests were to investigate the maximum credible hazards, the most important of which were identified as loss-of-coolant accidents (LOCAs). Initial studies resulted in a proposed test program designated as the Loss-of-Fluid Test (LOFT) Program to investigate LOCAs in PWRs.

A detailed history of the early years of LOFT and details regarding the initial LOFT proposal are presented in References 2-1 and 2-2, respectively. The following give a brief history of the LOFT Program, describe the LOFT facility and compare it with other experimental facilities where PWR accident behavior can be studied, and discuss the experimental program completed in LOFT.

2.1 History

The LOFT Program, as initially conceived, was to investigate the consequences of a LOCA in a commercial PWR due to the rupture of a primary coolant pipe. The entire experimental program was to be limited to a few nonnuclear (with a core simulator) experiments simulating ruptures in reactor primary coolant pipes and one nuclear (with an operating nuclear core) experiment simulating a rupture in a reactor primary coolant pipe that would result in meltdown of the reactor core (uncontrolled LOCA). The objectives of this program were to:[2-3]

1. Conduct experiments which would demonstrate the behavior of the reactor system and plant equipment with respect to safety under simulated accident conditions

2. Provide quantitative information which would assist in more accurately defining the hazards that might be associated with the LOCA.

Nearly 4 years of design and 3 years of construction were subject to considerable modification when the basic mission of the LOFT Program was redirected by the AEC in May 1967.[2-4] Since the initiation of the LOFT Program, changes of emphasis in the nuclear industry indicated that a test of engineered safeguards designed to prevent loss of coolant would be more valuable than the data collected in a LOCA simulation. The revised program objectives were to:[2-5]

1. Verify the capability of analytical methods to predict the LOCA response of large power reactors, the performance of engineered safety systems, and the margins of safety inherent in their performance

2. Identify any unexpected events or thresholds not then accounted for in the analysis of plant response or in the design of engineered safety systems.

The LOFT Program, thus reoriented, was to evaluate core cooling systems in a series of at least four integral tests.[a]

Early in 1969, the second major modification was made to the LOFT design. As stated in Reference 2-7, the program objectives were revised to require an integral test system which would:

1. Provide data required to evaluate the adequacy of and to improve the analytical methods currently used to predict:

    a. The LOCA response of large PWRs

    b. The performance of engineered safety features (ESF) with particular emphasis on ECCSs

    c. The quantitative margins of safety inherent in the performance of the ESF.

2. Identify and investigate any unexpected event(s) or threshold(s) in the response of either the plant or the ESF and develop

a. The term "integral" defines an experiment combining the nuclear, thermal, hydraulic, and structural processes occurring during a LOCA, and differentiates it from the single- and dual-effects, nonnuclear, small-scale, thermal-hydraulic experiments conducted for loss-of-coolant analysis.[2-6]

analytical techniques that adequately describe and account for the unexpected behavior(s).

3. Provide experience in the application of Division of Reactor Development and Technology (RDT) standards and other standards and codes generally applicable to large PWRs by their use and evaluation by the LOFT Program.[2-6] (Note: This objective was satisfied during design and construction of the LOFT facility and did not apply to the experimental program.)

The new design was required, therefore, to model, as nearly as possible, a "typical current generation large PWR primary coolant system, reactor system, and ECCS."[2-6]

In September 1971, the containment vessel construction was completed. The containment vessel was successfully leak tested 1 year later. Another major milestone was the installation of the test assembly in the LOFT containment vessel in 1973. Construction was complete in mid-1975, and nonnuclear testing began in 1976.

The nonnuclear loss-of-coolant experiment (LOCE) series was completed in April 1978, and the first nuclear large-break LOCE was completed in December 1978. In March 1979, a small-break LOCA occurred in the TMI-2 nuclear power plant. This event changed the emphasis for LOCA analysis from large breaks to small breaks. A second large-break LOCE was conducted in the LOFT facility on May 12, 1979, followed by the first small-break LOCE on May 31, 1979. Final changes made to the program resulted in the following program objectives which were to:

1. Provide data required to evaluate the adequacy of, and to 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 coolant systems (ECCSs), and the quantitative margins of safety inherent in the performance of the ESFs.

2. Identify and investigate any unexpected events(s) or threshold(s) in the response of either the plant or the ESFs and develop analytical techniques that adequately describe and account for the unexpected behavior(s)

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

4. Identify and investigate methods by which reactor safety can be enhanced, with emphasis on the interaction of the operator with the plant.

The LOFT Experimental Program is outlined in more detail in Section 2.3.

2.2 Experimental Facilities

This section describes the LOFT facility, and then briefly describes other experimental facilities where PWR accident behavior can be studied to indicate how LOFT operating characteristics and size compare with other experimental facilities and commercial PWRs. The experimental facilities described are the Semiscale facility located at the INEL, the Large-Scale Test Facility (LSTF) in Japan, the Loop Blowdown Investigation (LOBI) facility in Italy, and the Primarkreislauf (PKL) test facility in Germany. The operating characteristics for LOFT and the other facilities are then compared with those of two commercial PWRs, Trojan and Biblis.

2.2.1 LOFT Facility. The LOFT integral experiment facility is a scale model of a commercial PWR. The purpose of the facility is to model the nuclear, and thermal-hydraulic phenomena expected to occur in a commercial PWR during a LOCA. In general, coolant volumes and flow areas in LOFT were scaled using the ratio of the LOFT core [50 MW(t)] to a commercial PWR core [3000 MW(t)]. Also, components used in LOFT are similar in design to those of a commercial PWR. Because of the scaling and component design, the LOFT LOCEs closely model the large-break LOCAs hypothesized for commercial PWRs. The following is a brief description of the LOFT facility. A detailed system description is presented in Reference 2-8.

2.2.1.1 General Arrangement. The LOFT facility is located at the INEL, Test Area North (TAN). The facility includes a containment building (vessel), service building, control and equipment building, and a large storage building.

The LOFT containment building is divided into a gas-tight test chamber and basement. The test chamber is an above-grade 21.34-m (inside diameter) cylindrical vessel with a hemispherical head. Personnel access to the test chamber and basement is via double-door air locks. The air locks and the test chamber access door have closure seals to maintain the 0.361-MPa design pressure of the containment. The test chamber houses:

1. The experiment assembly

2. The 45-Mg overhead crane

3. Support equipment.

The basement houses:

1. Boric acid tanks

2. Emergency core coolant (ECC) low-pressure injection system (LPIS) pumps

3. Primary component cooling system heat exchangers

4. Other support equipment.

The service building (Building TAN-650) and the control and equipment building (Building TAN-630) are located immediately adjacent to the containment building. These buildings are connected to a large storage building (Building TAN-629), originally designed as a large shielded aircraft hangar for General Electric Company's Aircraft Nuclear Propulsion Program, which was terminated in 1961.

The LOFT integral experiment facility is shown in Figure 2-1. The major components of the experiment assembly, which includes the reactor and primary coolant systems, are mounted on a double-width railroad dolly by a special frame. The major components in the reactor, primary coolant, and blowdown suppression systems for a cold leg break configuration are shown in Figure 2-2.

The LOFT experimental system is discussed in more detail in the following section. The LOFT Experimental Program is outlined in Section 2.3; experimental results are presented in Section 3.

2.2.1.2 LOFT Experimental System. The LOFT experimental system consists of the systems used for the conduct of LOFT experiments as follows:

1. Reactor system

2. Primary coolant system

3. Blowdown suppression system

4. Emergency core cooling system

5. Secondary coolant system.

These systems are instrumented to provide continuous monitoring of the nuclear, thermal, hydraulic, and structural processes occurring during the LOFT experiments. The major system parameters of the LOFT system are given in Table 2-1.

2.2.1.2.1 Reactor System—The reactor system shown in Figure 2-3 consists of

1. Reactor vessel and head

2. Core support barrel

3. Upper and lower core support structures

4. Flow skirt

5. Reactor vessel fillers

6. 1.68-m long nuclear core (core simulator for nonnuclear experiments).

2.2.1.2.2 Primary Coolant System—The primary coolant system consists of two coolant loops connected to the reactor system. The loops are designated the intact loop and the broken loop. The primary system (a) removes heat from the reactor system during normal, upset, emergency, and faulted conditions; (b) provides and maintains a primary coolant and fission product envelope; (c) provides the capability of producing a coolant environment and time behavior during an experiment which will be similar to that found during a commercial PWR accident; (d) provides the capability of removing residual heat from the reactor system during normal, upset, emergency, and faulted conditions; and (e) simulates the three unbroken loops (intact loop) and one broken loop (broken loop) of a four-loop PWR during accident transients.

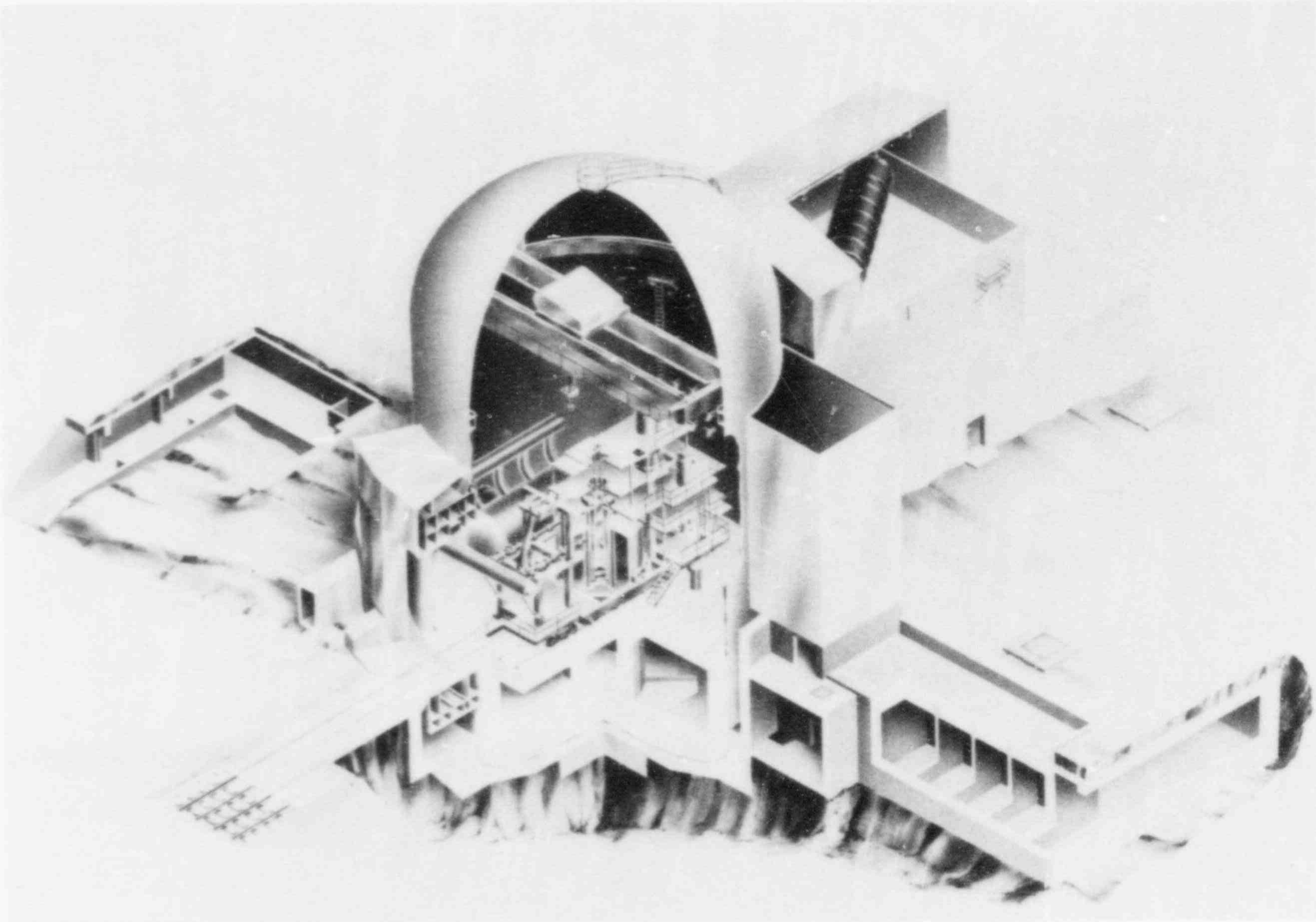

Figure 2-1. LOFT integral experiment facility.

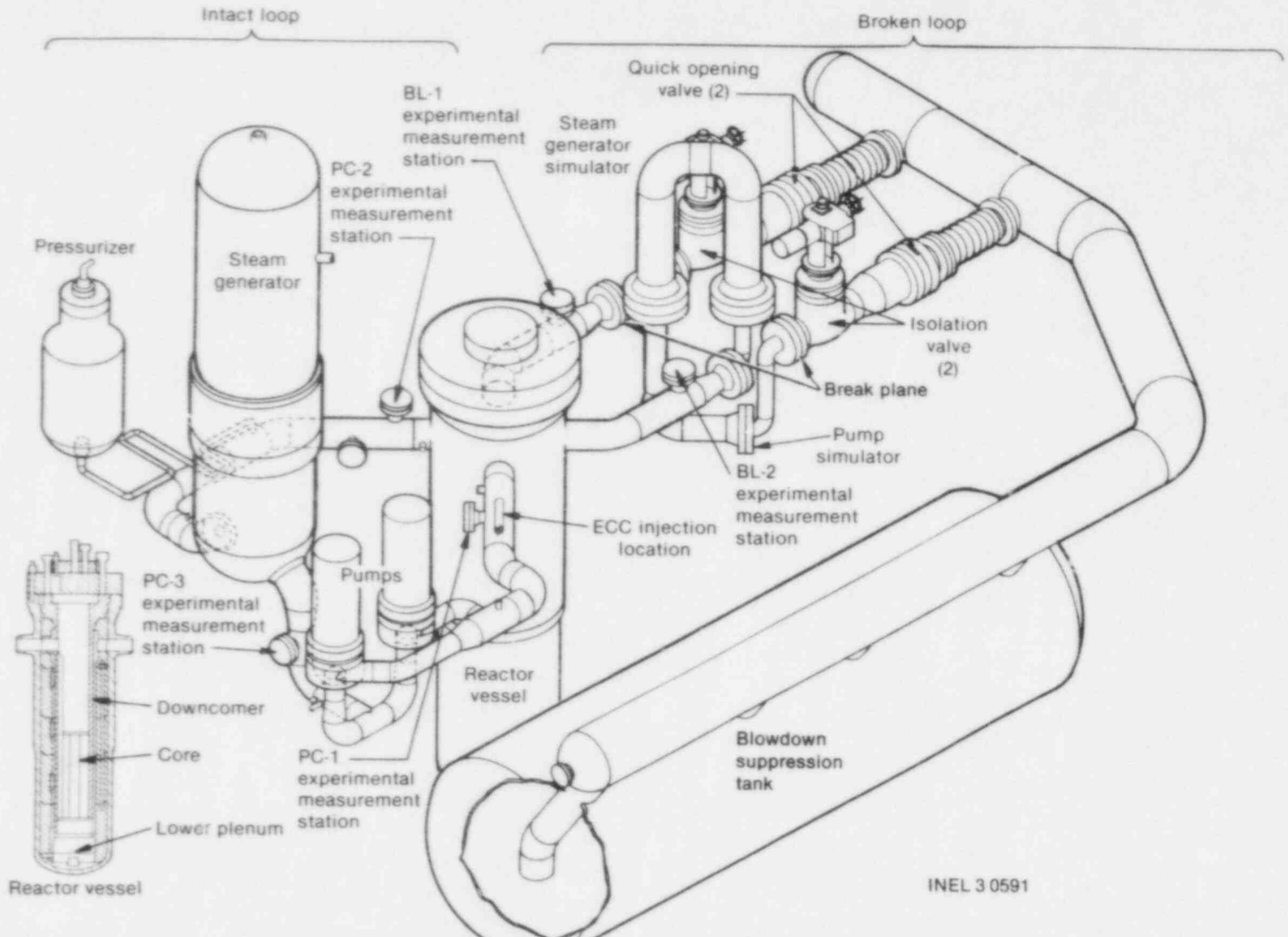

Figure 2-2. Axonometric projection of LOFT system configuration for a cold leg break.

Table 2-1. Major LOFT system parameters

Parameter	Value
Core	
Number of fuel rods	1300
Fueled length	1.68 m
Fuel rod diameter	10.7 mm
Fuel enrichment	4.05 wt%
UO_2 density	93% of theoretical
Core power	50 MW(t)
Primary Coolant System	
Operating pressure	15.5 MPa
Maximum hot leg temperature	575 K
Maximum coolant flow rate	479 kg/s

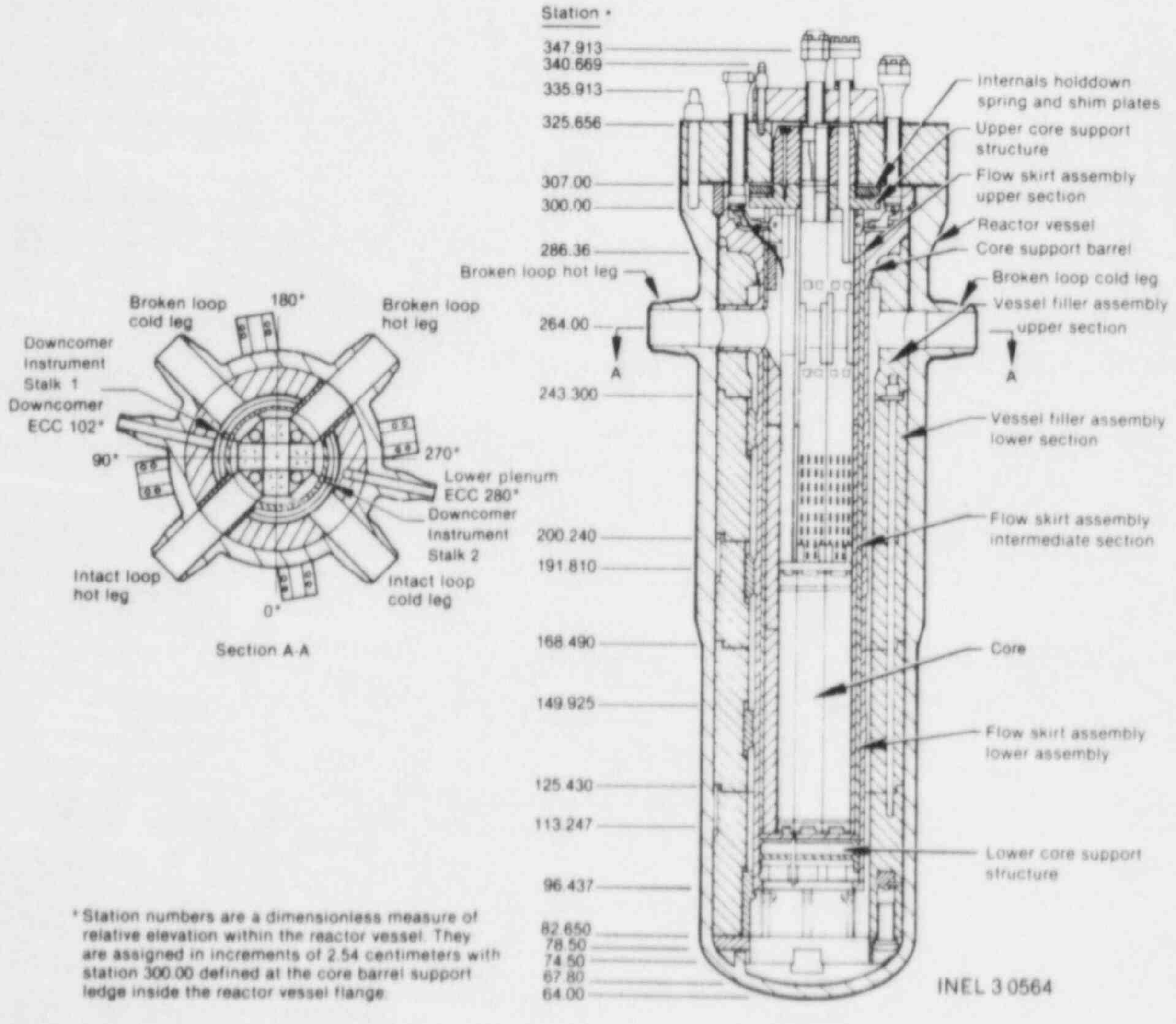

Figure 2-3. Reactor system vessel and internals arrangement.

The principal components contained in the intact loop are:

1. Steam generator

2. Primary coolant pumps

3. Pressurizer

4. Primary coolant venturi (flow measuring device)

5. Intact loop piping.

The broken loop simulates the broken loop of a four-loop PWR during a LOCA. The loop is not closed in the LOFT system, as its hot and cold legs end in quick-opening blowdown valves (QOBVs) that discharge to the blowdown suppression system. To maintain the dead-ended legs of the loop at specified hot and cold leg temperatures, two 25.4-mm warmup lines are provided.

The principal components of the broken loop are:

1. System piping

2. Steam generator simulator

3. Pump simulator

4. QOBVs and upstream isolation valves

5. Reflood assist bypass system

6. Break simulating orifices.

2.2.1.2.3 Blowdown Suppression System – The blowdown suppression system was designed to simulate the containment back pressure response of commercial PWRs during LOCAs and contain the blowdown effluent. This system also greatly simplifies recovery from a LOCE and requalification of the system for succeeding experiments (that is, the LOFT containment is not contaminated during the LOCE).

During the LOFT LOCEs, the primary coolant will blow out of the broken loop, through the blowdown suppression tank header and downcomers, and into a condensing pool (borated water) in the suppression tank. The four downcomers extend into the suppression tank and direct the blowdown effluent (a two-phase water-steam mixture) vertically below the condensing pool surface.

The major components of the blowdown suppression system are:

1. Blowdown suppression header

2. Blowdown suppression tank

3. Blowdown suppression tank spray system.

2.2.1.2.4 Emergency Core Cooling System – The LOFT ECCS provides plant protection and an experimental simulation of commercial PWR ECCSs. The plant protection functions are to provide (a) required core cooling over the full range of break sizes and locations and (b) the capability for long-term core cooling in conjunction with the purification system.

The experimental functions of the LOFT ECCS are to (a) simulate the response of a commercial PWR ECCS during a LOCA, (b) allow for investigation of the margin available in commercial PWR ESFs, and (c) provide flexibility for investigation beyond the capability of commercial PWRs.

The LOFT ECCS is arranged in two separate groups of equipment. Each group contains three subsystems: the high-pressure injection system (HPIS), the accumulator system, and the low-pressure injection system (LPIS). The groups can act singly or simultaneously. In general, the groups use separate piping, but in some instances, it is possible to valve them into the same piping. During the LOFT LOCEs, normally only one group is used. The other group is maintained in the standby mode and is under the control of the plant protection system.

2.2.1.2.5 Secondary Coolant System – The secondary coolant system (a) removes the heat transferred to the secondary system in the steam generator to the environment, (b) controls reactor power during power operation and influences the primary coolant and reactor systems in a manner similar to the secondary system of a commercial PWR, and (c) removes decay heat under normal conditions.

The main loop of the secondary coolant system consists of the shell side of the steam generator, the air-cooled condenser, the condensate receiver, the condensate subcooler, the main feedwater pump, and the main steam control and feedwater flow control valves.

Steam flows from the shell side of the steam generator through the main steam control valve to the air-cooled condenser. Condensate leaving the condenser passes through a receiver that acts as a surge volume for the system. Condensate leaving the receiver passes through a subcooler and then through the main feedwater pump. The main feedwater pump discharge has a recirculation line that allows a small amount of feedwater to return to the subcooler. The feedwater passes through the feedwater control valve to the feed ring in the steam generator.

2.2.1.3 Measurements. The LOFT experimental measurements are summarized in this section. The number of experimental and process measurements and their locations on the LOFT experimental system are identified in Table 2-2.

Table 2-2. LOFT system measurements summary[a]

Parameter	Measurement Location	Number of Experimental Measurements	Number of Process Measurements[b]	Total Number of Measurements[c]
Coolant temperature	Downcomer	14	—	14
	Intact loop piping	6	5	11
	Broken loop piping	7	4	11
	Suppression system	12	8	20
	Steam generator	2	1	3
	Core and plenums	54	—	54
	Secondary coolant system	3	1	4
	ECCS	—	2	2
	Pressurizer	—	2	2
Coolant level	Downcomer	1	—	1
	Lower plenum	1	—	1
	Core	3	—	3
	Upper plenum	1	—	1
	Suppression system	2	2	4
	Pressurizer	—	3	3
	Steam generator (secondary side)	—	4	4
	Condensate receiver	—	2	2
	Borated water storage tank	—	2	2
	Accumulators	1	2	3
Coolant velocity, momentum, and flow direction [drag disk-turbine transducers (DTTs)]	Lower plenum	2	—	2
	Upper plenum	3	—	3
	Intact loop	6	—	6
	Broken loop (momentum flux only)	12	—	12
Coolant density	Intact loop	9	—	9
	Broken loop	6	—	6
Coolant pressure	Upper plenum	4	—	4
	Downcomer	10	—	10
	Intact loop	7	4	11
	Broken loop	6	—	6
	Suppression tank	15	8	23
	Secondary coolant system	1	3	4
	ECCS	—	9	9
Differential pressure	Reactor vessel	8	—	8
	Intact loop	11	4	15
	Blowdown system	8	—	8
Pump speed	Primary coolant pumps	2	—	2
Displacement	Center fuel assembly fuel rods	3	—	3
Coolant flow	Primary coolant system	—	3	3
	Pressure reduction and decontamination system	—	1	1
	Suppression system	—	4	4
	Secondary coolant system	—	3	3
	ECCS	—	8	8

Table 2-2. (continued)

Parameter	Measurement Location	Number of Experimental Measurements	Number of Process Measurements[b]	Total Number of Measurements[c]
Neutron flux	Core	8	—	8
	Shield tank	—	19	19
Metal temperatures	Fuel cladding (external)	185	—	185
	Fuel cladding (internal)	5	—	5
	Fuel module support structures	10	—	10
	Reactor vessel	14	—	14
	Guide tubes	11	—	11
Fuel Rod Internal Pressure	Fuel	10	—	10
Fuel Pellet Temperature	Centerline	5	—	5
	Imbedded	5	—	5

a. This measurements summary includes process instruments recorded on the data acquisition and visual display system (DAVDS).

b. Additional process measurements are:

(1) Primary system Motor Generators A and B frequencies

(2) Primary coolant pump voltage

(3) Primary coolant pump current

(4) Control rod limit switches

(5) Valve position.

c. Total number of instruments is 578.

Experimental instruments in the LOFT core area monitor neutron flux, temperature, coolant level, and fuel rod displacement. Thermocouples measure fuel centerline, fuel rod cladding, and core support tube temperatures. Conductivity probes measure core coolant level. Self-powered neutron detectors (SPNDs) measure core neutron flux levels, and linear variable displacement transducers (LVDTs) are used to measure displacement of the center fuel assembly.

The reactor vessel plenums above and below the reactor core and the reactor vessel downcomer have experimental instrumentation. Thermocouples measure coolant temperatures in the upper and lower end boxes of selected fuel assemblies and in the upper plenum and downcomer of the reactor vessel. Strain gauge pressure transducers measure primary coolant pressures above and below the reactor core and in the reactor vessel downcomer. Conductivity probes measure water level in the reactor vessel downcomer, upper plenum, and lower plenum. Drag disk-turbine transducers (DTTs) measure coolant momentum flux, velocity, and flow direction both above and below the core and in the reactor vessel downcomer.

Temperature, pressure, differential pressure, velocity, density, and momentum flux are measured in the intact loop piping, in the reactor vessel nozzles, and between the steam generator and primary coolant pump inlet. Additional temperature measurements are made in the inlet and outlet plenums of the steam generator. The temperature and pressure transducers are standard commercial units. Velocity and momentum flux measurements are made with DTT rakes, and density is measured with a gamma densitometer. Eddy current displacement transducers measure primary coolant pump speed.

Temperature, pressure, differential pressure, density, and momentum flux are measured in the broken loop. In addition, temperature, pressure,

and liquid level are measured in the blowdown suppression tank, and pressure and differential pressure are measured at selected locations in the broken loop piping.

Temperature, pressure, and level of the secondary system coolant are measured in the steam generator.

Standard commercial strain gauges and accelerometers monitor strain and acceleration of selected structural components.

Table 2-3 presents a summary of the nomenclature used to designate LOFT experimental measurements.

2.2.2 Other Experiment Facilities. Other experiment facilities available or planned to study accident behavior of PWRs or PWR fuel rods are Semiscale, LOBI, PKL, and LSTF. This section provides a short description of these facilities, and their capabilities relative to the LOFT facility.

Table 2-3. Nomenclature for LOFT instrumentation

Designations for the Different Types of Transducers

TE	-	Temperature element	AE	-	Accelerometer
TT	-	Temperature transmitter	RPE	-	Pump speed element
PE	-	Pressure element	DE	-	Densitometer
PT	-	Absolute pressure transmitter	DIE	-	Displacement element
PdE	-	Differential pressure element	ME	-	Momentum flux detector
LE	-	Coolant level element	NE	-	Neutron detector
LT	-	Coolant level transmitter	PNE	-	Pulse neutron detector
FE	-	Coolant flow element			
FT	-	Coolant flow transmitter			
PC	-	Primary coolant intact loop	UP	-	Upper plenum
BL	-	Broken loop	LP	-	Lower plenum
SG	-	Steam generator	1ST	-	Downcomer stalk 1
SGS	-	Steam generator secondary	2ST	-	Downcomer stalk 2
RV	-	Reactor vessel			
SV	-	Suppression tank			

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

2.2.2.1 Semiscale Test System. The Semiscale test system (the Mod-2A system), located at the INEL, is a 1/1500th scale two-loop representation of a full-size commercial four-loop PWR system. The Semiscale Mod-2A system can simulate small, intermediate-size, and large breaks anywhere in a PWR system (the minimum size break in Semiscale is equivalent to a 4-in. diameter break in a commercial PWR). It can also simulate a wide range of PWR operational transients.

The Mod-2A system consists of a pressure vessel with simulated reactor internals and an external downcomer; an intact loop with piping sections, pressurizer, pump, and full-length steam generator; a broken loop with piping sections, pump, full-length steam generator, and pipe rupture assembly; a pressure suppression system with header and suppression tank; a steam supply system; and a coolant injection system with injection pumps, accumulators, and delivery piping.

The reactor core is simulated by an electrically-heated core containing 25 heater rods. Total core power is 2 MW. All major components and subcomponents are scaled one-to-one in elevation to a four-loop PWR. A guard heater system and extensive thermal insulation control atypical heat losses.

2.2.2.2 LOBI Test Facility. The LOBI test facility, located in Ispra, Italy, and discussed in Reference 2-9, is designed as an approximately 1/700 by volume scale version of a four-loop 1300 MW(e) PWR primary coolant system. It consists of two primary coolant loops connected to the reactor pressure vessel. Both experimental loops are active loops, each containing a circulation pump and a steam generator. LOBI has an electrically heated rod bundle consisting of 64 heater rods. Total core power is 5.3 MW. The test facility is designed for an operating pressure and temperature of 16.0 MPa and 598 K, respectively. Pipe ruptures of various sizes, ranging from double-ended large breaks to single-ended small leaks, may be simulated at three different locations within the broken loop (hot leg, cold leg, or loop seal). Other possible rupture locations are in the lower plenum of the reactor pressure vessel model and in steam generator U-tubes.

The ECCS consists of intermediate pressure (up to 6.0 MPa) injection pumps or accumulators, and can provide ECC water for both cold leg and combined cold and hot leg injection into both experimental loops.

The design of the experimental primary loops and the individual components ensured that the following parameters are maintained as close as possible to the value of the reference plant: (a) the ratio of power to volume, (b) the ratio of the volumes of various components and pipework sections to each other, (c) the ratio of rupture size to primary coolant system volume, and (d) the single-phase, steady state pressure drop and temperature distribution along the flow paths. The height and relative elevations of components are scaled 1:1, thus preserving gravitational heads.

2.2.2.3 PKL Test Facility. The PKL test facility, located in the Federal Republic of Germany (FRD), represents a typical Kraftwerkunion (KWU) 1300 MW(e) four-loop PWR on a scale of 1:134 by volume. It was designed to simulate the behavior of the entire primary coolant system during the refill and reflood phase of a LOCA. In view of the importance of the driving gravity forces during reflood, as well as for natural circulation, all elevations correspond to actual reactor dimensions.

The test facility is designed for a maximum operating pressure of 3.5 MPa. The test bundle simulating the core consists of 340 electrically heated rods with a total core power of 1.5 MW. The facility includes three primary coolant loops (one of double capacity simulating two loops) which contain steam generators whose secondary sides can be cooled down. The PKL test facility is described in more detail in Reference 2-10.

2.2.2.4 Large-Scale Test Facility. The LSTF is operated by the Japan Atomic Energy Research Institute (JAERI) for the Rig of Safety Assessment (ROSA) IV Program for small-break experiments and analysis. The primary objectives of the ROSA-IV Program are to develop a computer code based upon a two-velocity, two-temperature flow model and to assess the code by using the integral test data. Integral tests will be performed in the transient LSTF.

The facility is a 1:48 scale by volume version of a 3400 MW(t) PWR. Pressure, temperature, flow rate, pressure loss, and two-phase flow characteristics in a PWR can be simulated. The height and elevation of each component is full scale. The facility has two loops simulating the intact and broken loops. The core is simulated by 1000 full-height electric heater rods. The maximum core power is 10 MW, which is satisfactory for studying the

coolability of the core under small-break or natural circulation conditions where the fission product decay heat is the principal heat source.

2.2.2.5 Comparison of Facilities. A number of parameters for two commercial PWRs (Trojan and Biblis) and five experimental facilities (of which LOFT is the largest and also the only nuclear facility) are compared in Table 2-4. LOFT is the only integral nuclear PWR experimental facility and is also the largest integral PWR experimental facility in the world. Of the nonnuclear experimental facilities, LSTF has the largest volume and Semiscale has the smallest.

The main difference between LOFT and a commercial PWR are:

1. LOFT has one operating (intact) loop and a blowdown (broken) loop with a pump and a steam generator simulator, while commercial PWRs have four operating loops

2. The LOFT core is 1.68 m long, while commercial PWRs have core lengths between 3.66 and 3.9 m

3. LOFT has a short steam generator relative to commercial PWRs.

The scaling rationale for LOFT and Semiscale and the effects of LOFT scaling on experimental results are discussed in Section 3.

Table 2-4. Comparison of commercial PWR and integral experiment facilities

Parameter	Trojan	Biblis	LOFT	Semiscale Mod-2A	LOBI (50-mm downcomer)	PKL (Core II)	LSTF
Volume (m^3)	347	437	7.7	0.21	0.82	2.93	7.23
Trojan volume/volume	1	0.79	45	1 652	423	119	48
Power [MW(t)]	3 400	3 752	50	2.2	5.3	1.5	10
Maximum linear heat generation rate (kW/m)	50 to 62	55.7	62.3	~50	55.7	Decay heat	Decay heat
Trojan power/power	1	0.91	62	1 545	642	227	340
Number of operating loops	4	4	1	2	2	3	2
Core type	Nuclear	Nuclear	Nuclear	Electrical[a]	Electrical[b]	Electrical[a]	Electrical
Number of fuel rods	39 400	49 400	1 300	25	64	340	1 060
Core length (m)	3.66	3.9	1.68	3.66	3.9	3.9	3.66
Core diameter (m)	3.40	3.8	0.61	0.10	0.133	0.31	NA[c]
Pump specific speed (rpm-gpm$^{1/2}$/ft$^{3/4}$)	5 400	6 749	3 338	1 680	1 509	NA	NA
HPIS available	Yes	Yes	Yes	Yes	Yes	No	Yes
Accumulator available	Yes	Yes	Yes	Yes	Yes	No	Yes
LPIS available	Yes	Yes	Yes	Yes	No	Yes	NA[a]
Design pressure (MPa)	17.2	17.2	17.2	17.2	17.2	4.0	17.2
Number of operating steam generators	4	4	1	2	2	3	2
Steam generator height	Full	Full	Short	Full	Full	Full	Full

a. Center heated—MgO.

b. Skin heated.

c. NA—not applicable.

2.3 LOFT Experimental Program

The LOFT Experimental Program included a series of nonnuclear LOCEs incorporating a core hydraulic simulator, followed by several series of nuclear experiments with a nuclear core producing power. Prior to the nonnuclear LOCEs, a series of mini-blowdown experiments was performed.

The mini-blowdown experiments (Series LO) was a series of 10 experiments conducted between January 27 and February 17, 1976, to (a) qualify components and operational procedures for simultaneous opening of the QOBVs, (b) verify the ability of the blowdown suppression system to withstand the structural loading caused by the injection of primary coolant system fluid during blowdown, and (c) provide experimental data for determining the maximum permissible blowdown suppression tank downcomer submergence level for nonnuclear LOCEs based on the design load limits of the suppression tank. The mini-blowdown experiments consisted of opening either one or both QOBVs to allow the volumes (0.0481 m^3 in each leg) between the QOBVs and the upstream isolation valves to flow into the blowdown suppression tank. Table 2-5 summarizes the experiments in the mini-blowdown series, and gives the number of QOBVs, the downcomer submergence level, and the fluid temperature for each experiment.

The nonnuclear LOCEs (Series L1) were large-break blowdown experiments and were performed to: (a) determine that the equipment/systems function properly, (b) demonstrate that the entire experiment facility can withstand the structural loads of blowdown, (c) determine that the blowdown experiment procedures are adequate, (d) provide experience to operators prior to nuclear experiments, (e) obtain isothermal LOCE data for comparison with similar data from other experimental programs, and (f) experimentally determine thermal-hydraulic system behavior prior to nuclear blowdown. The nonnuclear LOCEs are summarized in Table 2-6.

Table 2-5. Mini-blowdown experiments

Experiment	Number of QOBVs	Downcomer Submergence (cm)	QOBV Fluid Temperatures (K)	
			QOBV-1	QOBV-15
LO-2	2	13.5	506.5	503.8
LO-3	1	28.7	501.5	504.5
LO-3A	2	28.1	506.4	502.1
LO-3B	1	26.7	506.9	505.3
LO-3C	2	29.9	507.4	507.1
LO-4	2	38.1	523.2	523.4
LO-5	2	56.2	523.6	524.4
LO-8	1	40.1	505.7	—
LO-9	1	21.9	518.5	510.8
LO-10	2	27.9	523.9	518.1

Table 2-6. Nonnuclear experiments

Experiment	Completed	Description	Primary Coolant System Temperature (K)	Primary Coolant System Pressure (MPa)	Results
L1-1	03/04/76	Hot leg break with a scaled area equal to 0.5 of PWR pipe area.	555	9.21	Blowdown loads proved small; system and instrumentation checked out. (Semiscale LOFT-counterpart Test S-01-1[a,2-12] provided additional data.)
L1-2	05/10/76	Cold leg break with a scaled area equal to PWR full pipe area.	555	15.65	Minimal downcomer hot wall effect shown by delayed ECC injection. (Semiscale LOFT-counterpart Test S-01-2[2-13] provided additional data.)
L1-3	06/28/76	Cold leg break with a scaled area equal to PWR full pipe area.	555	15.65	No accumulator injection, provided a baseline for LOCEs L1-3A and L1-4. (Semiscale LOFT-counterpart Test S-01-3[2-14] provided additional data.)
L1-3A	07/15/76	Cold leg break with a scaled area equal to PWR full pipe area.	556	15.56	ECCS not swept out the break. Base case obtained for quantification of ECC bypass for cold leg injection.
L1-4	05/03/77	Cold leg break with a scaled area equal to PWR full pipe area. Designated International Standard Problem 5 and United States Standard Problem 7.	552	15.75	Amount of core bypass with ECC injection into intact loop cold leg was evaluated and compared to LOCE L1-3A reactor vessel water inventory. (Semiscale LOFT-counterpart Test S-01-4A[2-15] provided additional data.)
L1-5	04/25/78	Cold leg break with a scaled area equal to PWR full pipe area.[b]	555	15.55	Behaviors of nuclear core and core simulator were similar, but saturation pressures were slightly different. (Semiscale LOFT-counterpart Test S-01-6[2-16] provided additional data.)

a. LOFT-counterpart experiments performed in the Semiscale facility[2-11] at the INEL provided data for comparison to study scaling effects on LOCE performance and, since they were performed prior to the LOFT experiments, provided assurance that the experiments would not damage the LOFT system.

b. Nuclear core was installed, but not producing heat.

The nuclear experiments (Series L2, L3, L5, L6, L8, and L9) provided the principal source of data to satisfy the LOFT Program objectives. For these experiments, the LOFT system simulated operation of a commercial PWR to provide data on thermal-hydraulic behavior during large-, intermediate-size-, and small-break LOCEs; anticipated transients; transients with core uncovery; and transients with multiple failures. The nuclear experiments are summarized in the order they were performed in Table 2-7.

Table 2-7. Nuclear experiments

Experiment	Date Completed	Description	Reactor Power (MW)	Primary Coolant System Temperature (K)	Primary Coolant System Pressure (MPa)	Primary Coolant System Flow Rate (kg/s)	Results
L2-2	12/09/78	Cold leg break with a scaled area equal to PWR full pipe area.	24.88	569 (average)	15.6	194.2	Experienced early rewet and much lower fuel cladding temperature than expected. Peak cladding temperature occurred during blowdown rather than during reflood. (Semiscale LOFT-counterpart Test S-06-2[2-17] provided additional data.)
L2-3	05/12/79	Cold leg break with a scaled area equal to PWR full pipe area. Designated United States Standard Problem 10.	36.7	577	15.1	199.8	Confirmed behavior of LOCE L2-2 (lower cladding temperature) at nominal PWR operating conditions. (Semiscale LOFT-counterpart Tests S-06-3[2-18] and S-06-6[2-19] provided additional data.)
L3-0	05/31/79	Small break simulated in pressurizer power-operated relief valve (PORV) similar to that at TMI-2[a]. After the experiment was completed, the NRC requested the data be withheld from analysis while blind prediction calculations were performed.	—[b]	558.2 (average)	14.74	201	Confirmed possibility for a stuck-open PORV to result in a liquid full pressurizer with simultaneous voiding in the reactor vessel. (Semiscale Tests S-TMI-3C and S-TMI-3I[2-20] provided additional data.)
L3-1	11/20/79	Cold leg single-ended break with a scaled area equal to the flow area of a 4-in. Schedule 160 pipe connected in the cold leg of a commercial PWR. Designated International Standard Problem 9 and NRC Licensing Requirement Standard Problem.	48.9	564 (average)	15.02	484	Break flow was sufficient to remove decay heat; no operator action required. (Semiscale LOFT-counterpart Tests S-SB-4 and S-SB-4A[2-21] provided additional data.)
L3-2	02/07/80	Cold leg single-ended break with a scaled area equal to the flow area of a pipe with a 1-in. inside diameter connected to the cold leg of a commercial PWR.	49.0	566 (average)	14.85	481.5	Steam generator was an effective heat sink. Natural circulation modes were stable and reversible.
L6-5	05/29/80	Anticipated transient experiment involving a loss of feedwater to the steam generator secondary.	36.7	561 (average)	14.79	479.7	Demonstrated the need to account for near equilibrium conditions in the pressurizer in computer codes.
L3-7	06/20/80	Cold leg single-ended break with a scaled area equal to the flow area of a pipe with a 1-in. inside diameter connected to the cold leg of a commercial PWR.	48.7	566 (average)	14.90	481.3	Similar to LOCE L3-2 but without bypass leakage. Steam generator feed and bleed was effective in removing core decay heat even with pumps shut down.

Table 2-7. (continued)

Experiment	Date Completed	Description	Reactor Power (MW)	Primary Coolant System Temperature (K)	Primary Coolant System Pressure (MPa)	Primary Coolant System Flow Rate (kg/s)	Results
L3-5/L3-5A	09/29/80	Cold leg single-ended break with a scaled area equal to the flow area of a 4-in. Schedule 160 pipe connected in the cold leg of a commercial PWR and with primary coolant pump shutoff.	49.0	567 (average)	14.86	476.4	LOCE L3-5 was used as a baseline for LOCE L3-6 to compare the effect of pump operation on system mass depletions and was an initial attempt to monitor reactor vessel liquid level using vessel thermocouples. LOCE L3-5A demonstrated reestablishment of natural circulation (single and two phase) following reestablishment of steam generator as a heat sink. (Semiscale LOFT-counterpart Tests S-SB-P1[2-22] and S-SB-P3[2-23] provided additional data.)
L6-2	10/07/80	Anticipated transient involving a loss of forced coolant flow in all primary coolant loops.	37.2	561 (average)	14.78	465.9	Pressurizer heaters proved effective in controlling primary system pressure, which decreased in this event as a result of primary coolant volume shrinkage following reactor scram.
L6-1	10/08/80	Anticipated transient experiment involving a loss of steam load in all steam generators.	36.9	560 (average)	14.78	478.5	Pressurizer heater and spray systems proved effective in controlling primary system pressure increases and resulting coolant insurges into the pressurizer in this event.
L6-3	10/09/80	Anticipated transient experiment involving an excessive load increase in all steam generators.	36.9	560 (average)	14.87	479.3	The pressurizer heaters and the high-pressure injection system (HPIS) proved effective in controlling the pressure in a primary system cooldown resulting from an excessive steam demand.
L3-6/L8-1	12/10/80	Cold leg single-ended break with a scaled area equal to the flow area of a 4-in. Schedule 160 pipe connected in the cold leg of a commercial PWR and with primary coolant pumps left on (L3-6), and with partial core uncovery (L8-1). Designated International Standard Problem 11.	50.0	568 (average)	14.87	483.3	LOCE L3-6 was used for comparison with LOCE L3-5 to measure the effect of pump operation onprimary system mass depletion and mass distribution. The results demonstrated the desirability of early pump shutdown after a small break LOCE. Experiment L8-1 provided data on core heatup in highly voided conditions. (Semiscale LOFT-counterpart Tests S-SB-P2[2-22] and S-SB-P4[2-23] provided additional data.)

Table 2-7. (continued)

Experiment	Date Completed	Description	Reactor Power (MW)	Primary Coolant System Temperature (K)	Primary Coolant System Pressure (MPa)	Primary Coolant System Flow Rate (kg/s)	Results
L9-1/L3-3	04/15/81	Multiple failure transient experiment initiated from a loss of feedwater accident (L9-1), and plant recovery using two independent modes (L3-3).	49.6	569 (average)	14.90	479.1	Experiment L9-1 demonstrated the capability of the PORV to control primary system pressure with a dry steam generator. LOCE L3-3 demonstrated two independent methods to recover from a LOCA: 1. Latching open PORV without steam generator 2. Recovery using steam generator as a heat sink.
L6-7/L9-2	07/31/81	Anticipated transient experiment simulating a turbine trip from full power followed by a secondary side-controlled rapid cooldown with primary coolant pumps operating (L6-7), and a continued rapid cooldown with pumps tripped (L9-2).	49.0	567 (average)	14.75	483.7	Experiment L6-7 simulated an overcooling transient similar to that which occurred in Arkansas Nuclear One—Unit 2 (ANO-2), provided data base to qualify computer codes for licensing calculations, and demonstrated use of pump current to diagnose the accident. During Experiment L9-2, natural circulation flow was not affected by voids in the reactor vessel and hot leg or by thermal stratification in the upper plenum.
L5-1	09/24/81	Cold leg single-ended break with a scaled area equal to 11.2-in. inside diameter break in PWR emergency core cooling system (ECCS) injection pipe.	45.9	565 (average)	14.93	308.2	Ninty-five percent of the core uncovered slowly, and temperature increased (up to 3 K/s). Accumulator injection started temperature to decrease in 30 s.
L8-2	10/12/81	Same as for Experiment L5-1 except the accumulator injection was withheld to determine the effect on fuel cladding temperature caused by restarting the primary coolant pumps.	46.0	566 (average)	14.86	311.1	Pump restart not effective in cooling core during high void conditions.
L9-3	04/07/82	Anticipated transient without scram (ATWS) experiment simulating a loss of feed-water followed by an operator-controlled recovery through the PORV and cooldown by initiation of secondary feedwater injection.	48.7	567 (average)	15.01	467.6	Experiment L9-3 was the first ATWS performed in a PWR. The reactor shut down due to moderator temperature feedback. PORV and safety relief valve (SRV) flows combined were sufficient to control primary coolant system pressure and to depressurize the system.
L6-6A L6-6B	04/19/82 04/21/82	Each experiment was an anticipated transient experiment simulating a boron dilution accident during refueling operation with residual heat removal.	— —	302 298	0.289 0.285	—[c] —[c]	Experiments L6-6A and L6-6B showed that the perfect mixing model is valid and that the effective mixing volume, consisting of only the reactor vessel volume up to the top of the nozzles and thae recirculation line, is the minimum volume.

Table 2-7. (continued)

Experiment	Date Completed	Description	Reactor Power (MW)	Primary Coolant System Temperature (K)	Primary Coolant System Pressure (MPa)	Primary Coolant System Flow Rate (kg/s)	Results
L2-5	06/16/82	Cold leg break with a scaled area equal to PWR full pipearea with primary coolant pumps turned off and decoupled from their external flywheels within 1 s after experiment initiation.	36.0	573 (average)	14.94	192.4	The ECCS prevented core damage even under the worse case accident scenario (that is, a 200% offset shear of a cold leg with loss of offsite power plus atypically fast pump coastdown).
L6-8	08/26 through 08/31/82	A series of six anticipated transient experiments: L6-8B-1 and L6-8B-2 simulated uncontrolled control rod withdrawal transients; L6-8C-1 simulated a recovery from steam generator tube rupture with primary coolant pumps on, L6-8C-2 simulated a recovery from steam generator tube rupture with primary coolant pumps off, and L6-8C-3 simulated a recovery from a small break with primary system void formation; and L6-8D simulated a primary system natural circulation cooldown with void formation.	—	—	—	—	The thermal and hydraulic response of the plant during rod withdrawal simulations L6-8B-1 and L6-8B-2 were as expected based on preexperiment calculations. The scram on high pressure in L6-8B-1 occurred earlier than predicted, and it is believed this difference was caused by nonequilibrium conditions in the pressurizer which were not adequately calculated. Doppler dominated the L6-8B-2 negative reactivity feedback as predicted. In L6-8B-2, the reactor scrammed due to a high peak power setpoint which is not typical in most commercial PWRs. The continued operation of the primary coolant pumps during L6-8C-1 and L6-8C-3 provided the operator with information on primary coolant system mass inventory which would otherwise have been lacking due to loss of pressurizer liquid level and hot leg subcooling. During the later portion of L6-8C-2, there was an indication of the presence of a persistent steam bubble outside the pressurizer. This steam bubble caused the pressurizer liquid level to increase even through primary coolant system charging via HPIS flow was reduced. The PORV was unintentionally open during most of the L6-8D transient. However, the experiment provided data on natural circulation cooling in the presence of voiding.
L6-8B-1			37.6	564 (average)	14.9	482	
L6-8B-2			37.3	565 (average)	14.8	475	
L6-8C-1			Shutdown	561 (average)	15.5	476.5	
L6-8C-2			Shutdown	561 (average)	15.5	475	
L6-8C-3			Shutdown	560 (average)	15.6	482	
L6-8D			Shutdown	532 (average)	7.17	484	

Table 2-7. (continued)

Experiment	Date Completed	Description	Reactor Power (MW)	Primary Coolant System Temperature (K)	Pressure (MPa)	Flow Rate (kg/s)	Results
L9-4	09/24/82	ATWS experiment simulating a loss-of-offsite power without reactor scram.	50.7	556.9 (cold leg)	14.84	463.0	For high pressure, temperature, and power conditions, natural circulation was effective in an ATWS situation as long as a heat sink was available. There was sufficient relief capacity in the scaled SRV to control the pressure transient in the loss-of-offsite-power ATWS. Turning the primary pumps off in a loss-of-offsite-power ATWS caused a more rapid decrease in reactor power (in an ATWS condition) than occurred in a loss-of-feedwater-induced ATWS when the primary pumps were left on.

a. In March 1979, a small-break loss-of-coolant accident (LOCA) occurred in the Three Mile Island—Unit 2 (TMI-2) power reactor. The LOCA was initiated by a loss-of-feedwater to the secondary system which caused the PORV to open and then stick open.

b. Nuclear core was not producing heat.

c. Primary coolant system recirculation flow was 4.77 L/s for L6-6A and 9.51 L/s for L6-6B.

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3. LOFT RESULTS RELATIVE TO COMMERCIAL PWR BEHAVIOR

This section summarizes the scaling criteria used for LOFT and discusses the compromises made as a result of practical considerations. A short discussion of Semiscale scaling is also presented to illustrate the relationship between a commercial PWR and LOFT as follows: The scale relationship between LOFT and Semiscale is similar to the scale relationship between a commercial PWR and LOFT. Comparisons between LOFT and Semiscale results were used to verify LOFT scaling. The remainder of the section describes the methodology developed to qualify the RELAP4 and RELAP5 computer codes to predict the behavior of a commercial PWR under corresponding accident or transient conditions. Two examples are included to illustrate the methodology and to demonstrate the degree to which LOFT simulates a commercial PWR.

3.1 LOFT Scaling

Three scaling approaches were considered as a basis for the LOFT system: linear scaling, classical dimensionless numbers, and volumetric scaling.[3-1] Linear scaling, which maintains length-to-diameter ratios, would be most appropriate during the very short subcooled blowdown. Linear scaling, however, results in a large reduction of time scale and distorts the energy distribution processes during saturated blowdown which is of primary concern during a PWR LOCA. Also, the application of dimensionless numbers to complex, transient, two-phase flow situations was not understood well enough to use them as the main scaling criteria for LOFT.

Volumetric scaling was the best suited for application in LOFT. This means that, to the extent practicable, each component and pipe volume and the total primary system volume in the LOFT system are proportional to their respective volumes in a commercial PWR. The constant of proportionality is the ratio of the two systems' powers. That is,

$$\frac{\text{MW(t) LOFT}}{\text{MW(t) CPWR}} = \frac{\text{Volume LOFT}}{\text{Volume CPWR}}$$

where CPWR is a commercial PWR.

Volume scaling ensures that the same relative amount of fluid is available for energy exchange in LOFT as in the commercial PWR. By controlling system break size, it also allows preservation of the time scale. Thus, similar thermal-hydraulic phenomena occur in both systems in the same time scale.

Table 3-1 shows the distribution of volume in LOFT and a commercial PWR. The comparison is good except for the total volume in LOFT which is 38% larger than the "ideal volumetrically scaled" volume. The main volume differences are in: (a) the lower plenum, (b) the downcomer, (c) intact loop exclusive of the steam generator, and (d) the reflood assist bypass system. The volumes in these four areas were purposely changed from "ideal" to improve the LOCE performance and/or system design.

The lower plenum volume in LOFT was increased by 0.23 m^3 over the ideal volume because ideal volumetric scaling resulted in a distance from the bottom of the core barrel to the bottom of the lower plenum of only 29.2 cm. This distance was considered unrealistically shallow and prone to cause sweep-out of the lower plenum water during reverse core flow. Therefore, this distance was increased to 73.7 cm to provide a more realistic lower plenum depth and depth-to-diameter ratio.

The downcomer gap width sizing requirements include not only volume but also metal surface area, flow area, pressure drop, and flooding velocity. These considerations lead to a width sizing requirement ranging from 1.91 to 5.08 cm. The 5.08-cm downcomer gap width was selected for LOFT on the basis of providing ECC flow down the downcomer that is similar to that of a commercial PWR. While the increase of 0.36 m^3 results in more relative volume in the downcomer than is typical of a commercial PWR, it is less than 5% of total system volume. Further, it results in a metal surface area-to-volume ratio in the downcomer which is more typical of a commercial PWR.

The intact and broken loop piping diameters were made larger than "ideal" for several reasons. Larger diameter pipe (14 in. versus 10 in.) provides a practical piping arrangement, within the allowable system space envelope, that meets structural stress requirements. Furthermore, the larger pipe has a more favorable surface-to-volume ratio from a structural heat input viewpoint. In the primary

Table 3-1. Comparison of LOFT and commercial PWR primary coolant system volume distribution

Item	CPWR Volume (m^3)	CPWR Percent of Total	LOFT Percent of Total	LOFT Volume (m^3)	Ideal LOFT[a] Volume (m^3)
Reactor Vessel					
Outlet plenum	55.48	15.95	11.7	0.90	0.89
Core and bypass	26.05	7.50	4.0	0.31	0.42
Lower plenum	29.74	8.58	9.22	0.71	0.48
Downcomer and inlet annulus	20.42	5.89	8.91	0.69	0.33
Subtotal		37.95	33.91		
Intact Loop[b]					
Hot leg pipe	6.71	1.94	4.88	0.38	0.11
Cold leg pipe	7.22	2.08	5.00	0.39	0.12
Pump suction pipe	10.71	3.09	4.41	0.34	0.17
Steam generator	91.50	26.40	18.2	1.40	1.48
Pump	6.71	1.96	2.57	0.04	0.11
Subtotal		35.47	35.06		
Broken loop					
Hot leg to break	2.24	0.65	2.27	0.18	0.04
Cold leg to break	5.98	1.72	2.60	0.20	0.10
Steam generator	30.50	8.80	7.12	0.55	0.49
Pump	2.27	0.65	0.84	0.07	0.04
Additional volume, part of outlet plenum	NA[c]	NA	NA	0.20	NA
Additional volume, part of inlet plenum	NA	NA	NA	0.21	NA
Subtotal		11.17	10.56		
Pressurizer	50.98	14.70	12.50	0.96	0.82
Total	346.64	100	100	7.70	5.58

a. $\text{Ideal LOFT Volume} = (\text{CPWR Volume}) \times \dfrac{\text{LOFT Core Power of 55 MW(t)}}{\text{CPWR Core Power of 3411 MW(t)}}$

b. Values for CPWR for three loops combined.

c. NA—not applicable.

system, the value of the metal surface area-to-volume ratio together with the metal and coolant temperatures determines the heat transfer per unit of coolant and, hence, the ECC temperature, its effectiveness in the core, and flow regimes throughout the primary system. A reduced scale model such as LOFT will have a larger surface area-to-volume ratio than a commercial PWR. Table 3-2 lists the surface areas and surface area-to-volume ratios for LOFT and a commercial PWR.

The pressure distribution in the intact and broken loops of LOFT are reasonable in comparison to that in the commercial PWR. For the same fluid properties, scaled flow areas, and scaled mass flow rates, $\Delta P_{LOFT} = \Delta P_{PWR}$ if $K_{LOFT} = K_{PWR}$, where K is the resistance factor. This type of scaling was used in determining the LOFT broken loop resistances given in Table 3-3 for a cold leg break. Note that the pump resistance (the most significant component) is five times higher than that of the steam generator. Outlet break comparisons give similar results. Also the pump and steam generator in the broken loop are simulated resistances. The effect of nonoperating components in the broken loop was assessed and is not considered significant to the overall behavior.[3-2]

Table 3-2. Comparison of surface areas and surface area-to-volume ratios

Item	LOFT	Commercial PWR
Core surface area (m^2)	88.6	5170
Intact loop steam generator tubes (m^2)	328.1	12756[a]
Structure area (m^2)	145.6	6313[b]
Total (m^2)	562.3	24239
$\frac{\text{Core surface area}}{\text{Actual PCS}^c\text{ volume}}$ (m^{-1})	11.5[d]	14.9
$\frac{\text{Steam generator tube area}}{\text{Actual PCS volume}}$ (m^{-1})	42.5	36.7[a]
$\frac{\text{Structure surface area}}{\text{Actual PCS Volume}}$ (m^{-1})	18.6	18.2
$\frac{\text{Total primary surface area}}{\text{Actual PCS volume}}$ (m^{-1})	72.8	69.9

a. Three intact loop steam generators only.

b. By difference, commercial PWR value includes 4534 m^2 for broken loop steam generator.

c. PCS—primary coolant system.

d. If the "LOFT ideal volume" of 5.58 m^3 is used, this value is 15.9.

Table 3-3. Comparison of flow resistances in LOFT and commercial PWR broken loops for a cold leg break

Item	K Factor Commercial PWR	K Factor LOFT
1. Broken loop cold leg	0.043	0.071
2. Hot leg to steam generator	0.087	0.12
3. Steam generator	3.5 to 4.00	3.87
4. Steam generator to pump	0.48	0.14
5. Pump locked rotor resistance	18.0	18.0
6. Pump to break	0.071	0.76

Pump inertia in the LOFT intact loop is made variable through the use of a flywheel-driven generator electrically connected to the pump motors. Thus, the coastdown-flow-rate/unit-speed transient under a loss-of-power condition can be adjusted to the required value.

Selection of core length was a key design decision of LOFT and it is discussed in detail in Reference 3-2. The 1.68-m long core provides reasonable axial and radial power profiles, it is large enough diametrically (0.61 m) to allow extensive instrumentation and allow radial flow effects, and it has fuel assemblies which are geometric, full-scaled replicas of their commercial PWR counterparts (except for length) which preserves the commercial PWR fuel thermal time constant in LOFT. The initial core ΔT is maintained typical of the commercial PWR core ΔT. Although the LOFT core is scaled on a volume/power basis, the flow area is about twice, and the core length half their respective ideal values. Thus, the core resistance is about half the ideal value. Reference 3-2 showed that preservation of initial core ΔT is more important than preservation of typical core hydraulic resistance.

It had been postulated that asymmetrical flow effects may be important during the LOCA, particularly relative to flooding of the downcomer. There are four inlet nozzles on the 14-m downcomer circumference of a commercial PWR. One-fourth the PWR downcomer circumference is about 3.5 m, which compares reasonably well with one-half the LOFT downcomer circumference of 2.6 m. Thus, the downcomer circumference per intact loop nozzle in LOFT is close to the commercial PWR value. On this basis, one might expect about the same degree of flow asymmetry in LOFT, with a single operating loop, as in a four-loop commercial PWR.

3.2 Semiscale Scaling

Originally, Semiscale was part of the LOFT Program. The Semiscale Mod-1 facility was a reduced-scale version of the LOFT facility and was scaled volumetrically to LOFT in the same way that LOFT was scaled to a commercial PWR (Semiscale Mod-1 was approximately 1/30th the volume of LOFT).[3-3,3-4] Semiscale has electrically heated rods to simulate nuclear fuel rods. As with LOFT, a number of flow areas and volume distortions were unavoidable and are discussed in detail in Reference 3-4.

The similarity of scaling approach for both the LOFT and Semiscale Mod-1 designs and the closely coordinated test programs were intended to provide:

1. An experimental basis for assessing the applicability and limitations of the scaling concepts used for both facilities

2. An experimental base needed to evaluate the capability of analytical models to predict the LOCA behavior of two systems with significantly different size components.

Since it is very unlikely that large- or small-break data from a commercial size PWR will ever be available, LOFT and Semiscale are the best sources of data to confirm that scale effects and the distortions caused by design considerations do not affect the major LOCA phenomena and that scaling is well enough understood to allow the results from LOFT and Semiscale to be used to verify codes used to predict the behavior of commercial PWRs.

The results of a number of LOFT and Semiscale Mod-1 counterpart tests[a] have been compared. The results are very similar and indicate that volume scaling does preserve time, and that the LOCEs in both systems are controlled by the same phenomena which are independent of size.

System pressures versus time for LOFT Experiment L1-4, Semiscale LOFT-counterpart Test S-01-4A, and the RELAP4/MOD5 computer code[3-5] prediction for Experiment L1-4 are shown in Figure 3-1. The agreement was very good until ECC water injection began. The instantaneous mixing assumption in the code resulted in an underprediction of pressure after 25 s.

Fluid densities in the broken loop cold leg for LOFT Experiment L1-4, Semiscale LOFT-counterpart Test S-01-04A, and the RELAP4/MOD5 prediction for Experiment L1-4 are shown in Figure 3-2. The agreement up to ~30 s is fairly good for all three curves, but, after that time, the Semiscale data show much more fluid in the pipe. The difference in behavior between Semiscale and LOFT is due to the "hot-wall" effect in the Semiscale downcomer. Owing to the relatively thin (0.0127 m) annulus of the Semiscale reactor vessel downcomer, the injected ECC water heats up rapidly and boils. The resultant steam flow up the downcomer carries a portion of the injected ECC flow up the downcomer and out the broken loop cold leg. Although some bypassing also occurs in LOFT, there is not as much, as indicated in Figure 3-2.

The close agreement among LOFT and Semiscale counterpart tests and computer code prediction data

a. These tests are identified in Tables 2-6 and 2-7 of Section 2.

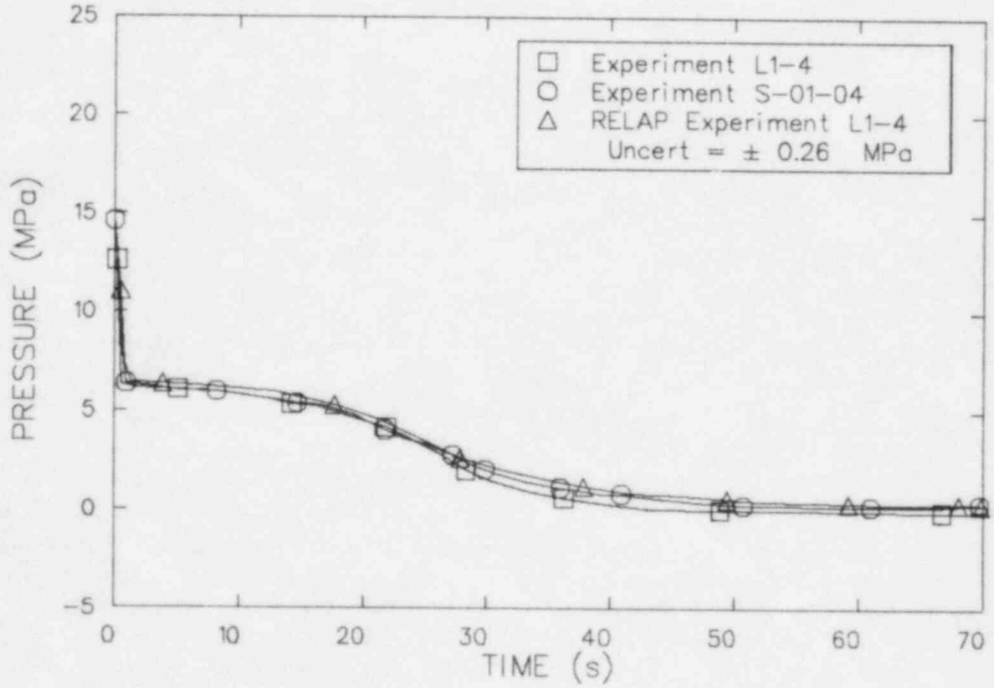

Figure 3-1. Pressure in primary system for LOFT Experiment L1-4 (measured and predicted) and Semiscale Test S-01-4 (measured).

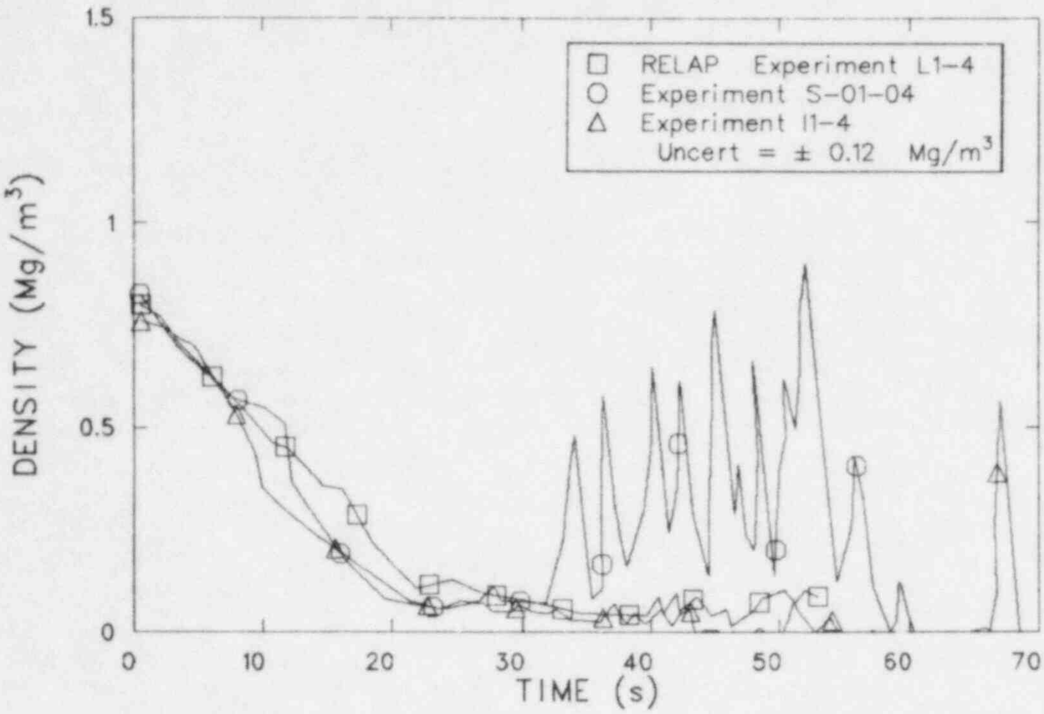

Figure 3-2. Fluid density in broken loop cold leg for LOFT Experment L1-4 (measured and predicted) and Semiscale Experiment S-01-4 (measured).

strongly indicates that LOFT and Semiscale can be used to study the behavior of commercial PWRs under accident and transient conditions, and that LOFT and Semiscale can be used to qualify RELAP or any other systems codes to predict the behavior of a commercial PWR under accident conditions.

3.3 Application of LOFT Results to Commercial PWRs

One of the most important applications of LOFT research results has been to verify the capability of the RELAP4 and RELAP5 computer codes to predict the behavior of commercial PWRs. This has been done by developing a methodology for referencing commercial PWR transient calculations to LOFT experimental data.

The steps in the methodology for using a LOFT transient as a basis for calculating a transient from the same initiating event in a commercial PWR are diagrammed in Figure 3-3. Three elements are employed:

1. The calculational mechanism
2. The operational package
3. The system model upon which the calculational mechanism operates within the constraints of the operational package.

The calculation mechanism is usually a computer code, such as RELAP4/MOD7, which contains a collection of physics, correlations, and numerical solution techniques for reactor transient analysis. The operational package is that part of the code input deck covering input quantities such as operational setpoints, correlation selections, and multipliers and other constant values. The system model is the geometric representation of the plant that is constructed using the computer code modeling features.

The three elements are applied first to a LOFT transient. The calculation is compared to the measured transient and then revised until the agreement is considered to be the best possible. The differences between calculated and measured LOFT response are a measure of the minimum uncertainties that can be expected in a commercial plant calculation. The three elements are then applied to the commercial PWR. Changes to the operational package and system model of LOFT are made based only on operational and geometric differences between LOFT and the commercial PWR. The

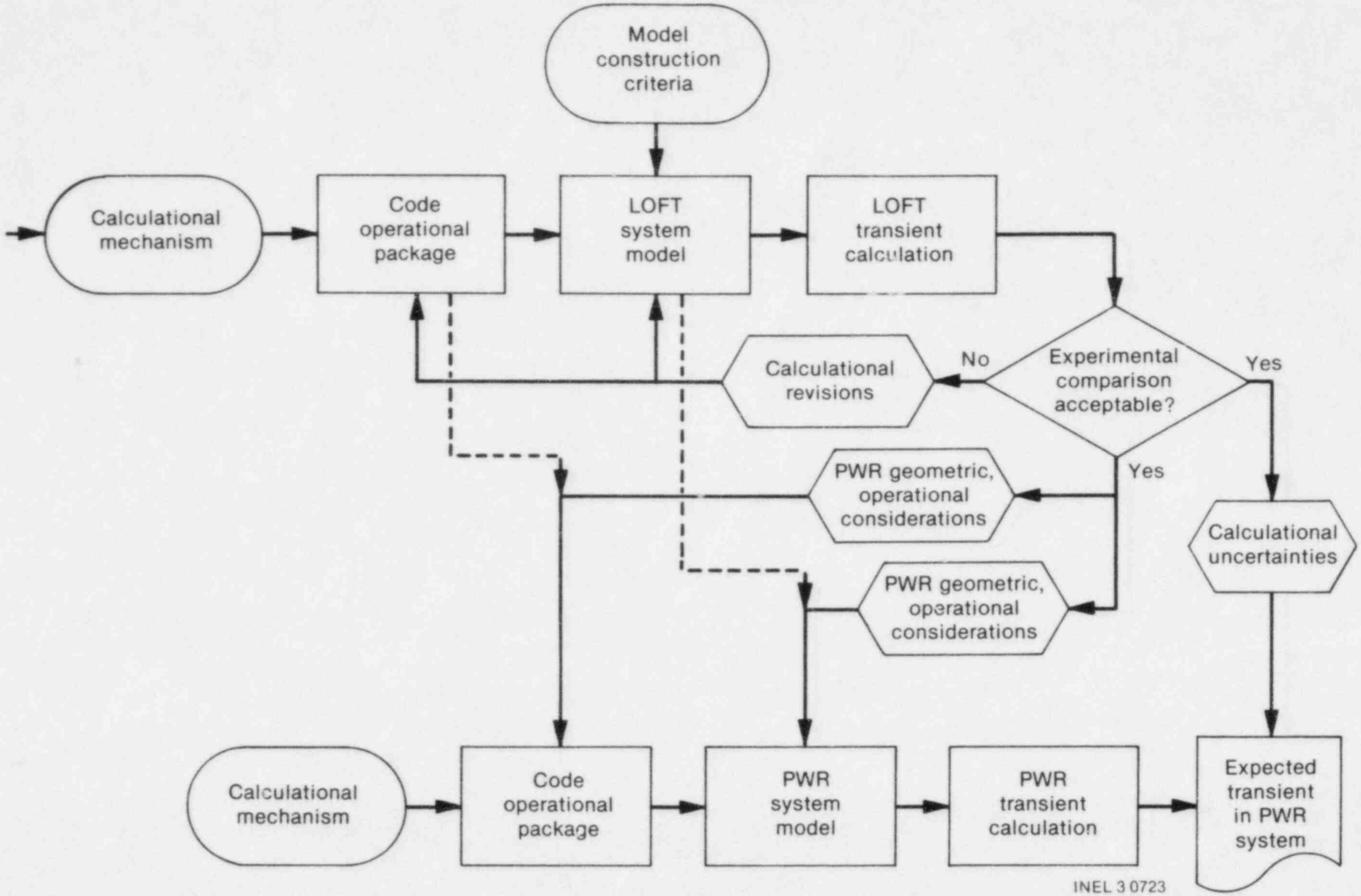

Figure 3-3. Methodology using LOFT experimental data to calculate commercial PWR transients.

PWR calculation that results is, as a consequence, traceable to the LOFT transient for an evaluation of the soundness of the calculation and a measure of the minimum uncertainty associated with the calculation for the commercial PWR.

The next step involves analyzing the operational and geometric differences between LOFT and the commercial PWR to determine the origins of the principal differences in the transients in the two systems. The analysis is done by imposing on the commercial system the corresponding actual operational and geometric parameters of the LOFT system, either singly or in combination, to make the commercial PWR transient more like the transient in LOFT. In this way, the manner in which the LOFT transient transforms to follow the commercial PWR transient can be understood.

The methodology was applied in a preliminary fashion to the Zion-1 commercial PWR in the calculation of an hypothesized transient resulting from a 200% (double-ended) cold leg break. The LOFT reference experiments were Experiments L2-2 and L2-3,[3-6,3-7] 200% cold leg break LOCEs, and the calculational mechanism was RELAP4/MOD6.[3-8] The results of these studies are presented in Reference 3-9. Data from the Zion-1 calculations and measured and calculated data from LOFT experiments are compared in the following paragraphs.

Pressure in the intact loop hot leg for LOFT Experiment L2-2[3-6] is shown in Figure 3-4. The initial core power for Experiment L2-2 was 25 MW [a maximum linear heat generation rate (MLHGR) of 26.4 kW/m]. RELAP4 calculations for Experiment L2-2 and a 200% cold leg break in Zion-1 are overlayed in Figure 3-4 for comparison (References 3-7 and 3-9). The Zion-1 calculations were made with initial core power of 70%, which is comparable to 25 MW in LOFT. The pressure history for LOFT and Zion-1 are very similar.

Fuel cladding temperature for LOFT Experiment L2-3[3-10] is shown in Figure 3-5. The initial core power for Experiment L2-3 was 36 MW (an MLHGR of 39 kW/m). RELAP4 calculations for Experiment L2-3 and a 200% cold leg break in Zion-1 are overlayed in Figure 3-5 for comparison.[3-9] The Zion-1 calculations were made with initial core power of 110%, which is comparable to 38 MW in LOFT. The fuel cladding temperatures in Zion-1 were not predicted to be as high in LOFT; however, the temperature trend appears similar.

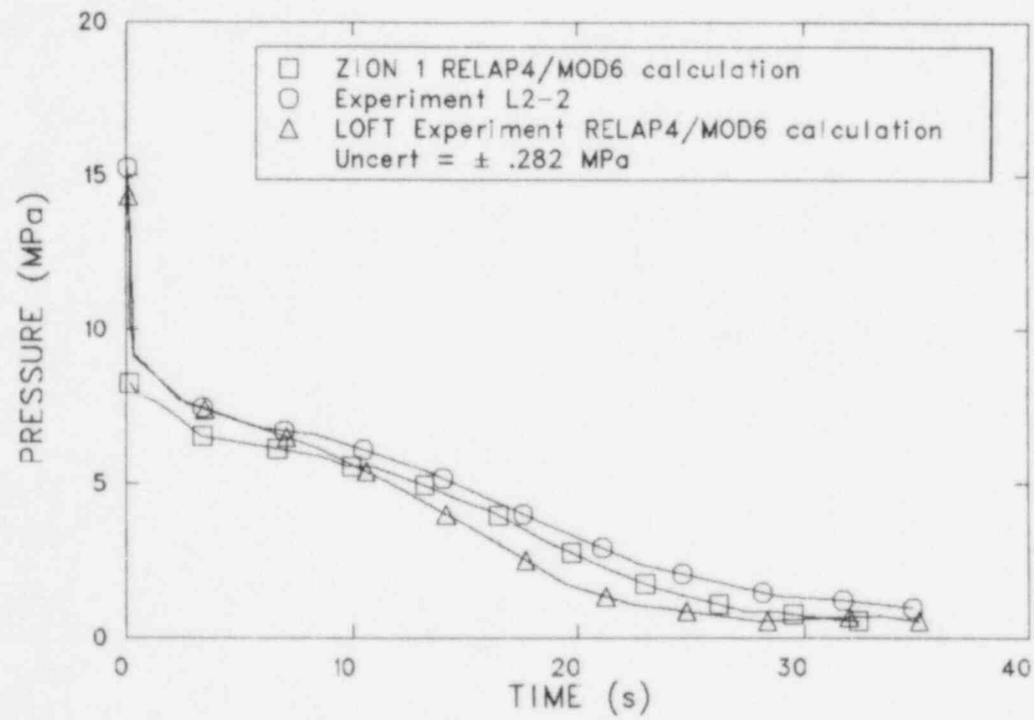

Figure 3-4. Pressure in intact loop hot leg for LOFT Experiment L2-2 (measured and predicted) and Zion-1 at 70% power (calculated).

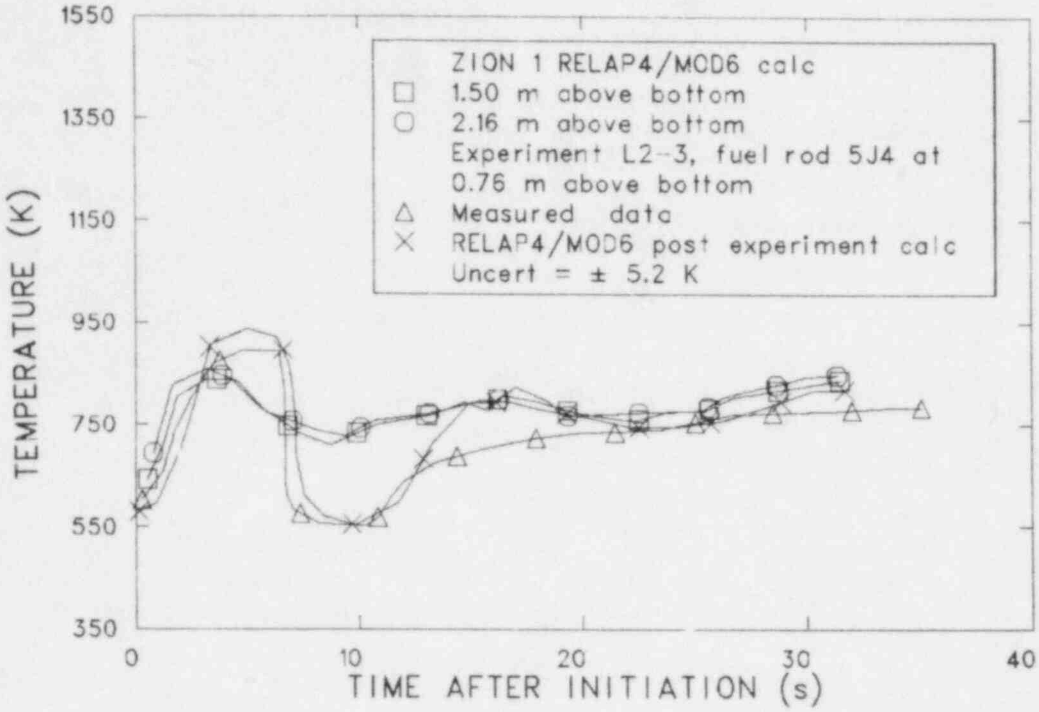

Figure 3-5. Fuel cladding temperature for LOFT Experiment L2-3 (measured and calculated) and Zion-1 at 110% power (calculated).

The results of this study for the large-break LOCAs indicate that despite some differences in core power distribution and pressurizer scaling, Zion-1 would behave qualitatively very much like LOFT during a large-break LOCA.

The methodology was also applied to the prediction of the response of Zion-1 to a 1-in. diameter cold leg break. The LOFT reference experiment was Experiment L3-7.[3-11] Measured pressure and RELAP4 postexperiment calculations of pressure for Experiment L3-7 and corresponding RELAP4 calculated pressure for Zion-1[a] are overlayed for comparison in Figure 3-6. The relatively large differences between LOFT behavior and calculated Zion-1 behavior is due to differences in trip setpoints, decay heat, pressurizer, core bypass, and energy transport to the steam generator. The Zion-1 calculation was repeated with LOFT trip setpoints and with decay heat, pressurizer, core bypass, and energy transport to the steam generator all scaled from LOFT. The recalculated pressure history for Zion-1 fell within the LOFT data band, as shown in Figure 3-7.

The most important result of these studies was to provide a measure of the minimum uncertainty of the accuracy of computed parameters for a large-break transient in Zion-1. Referring to Figure 3-4, a minimum uncertainty of approximately ±1.5 MPa could be placed on the RELAP4/MOD6 pressure predictions. Referring to Figure 3-5, a minimum uncertainty of ±100 K could be placed on the RELAP4/MOD6 fuel cladding temperature predictions.

a. The Zion-1 calculations are unpublished results.

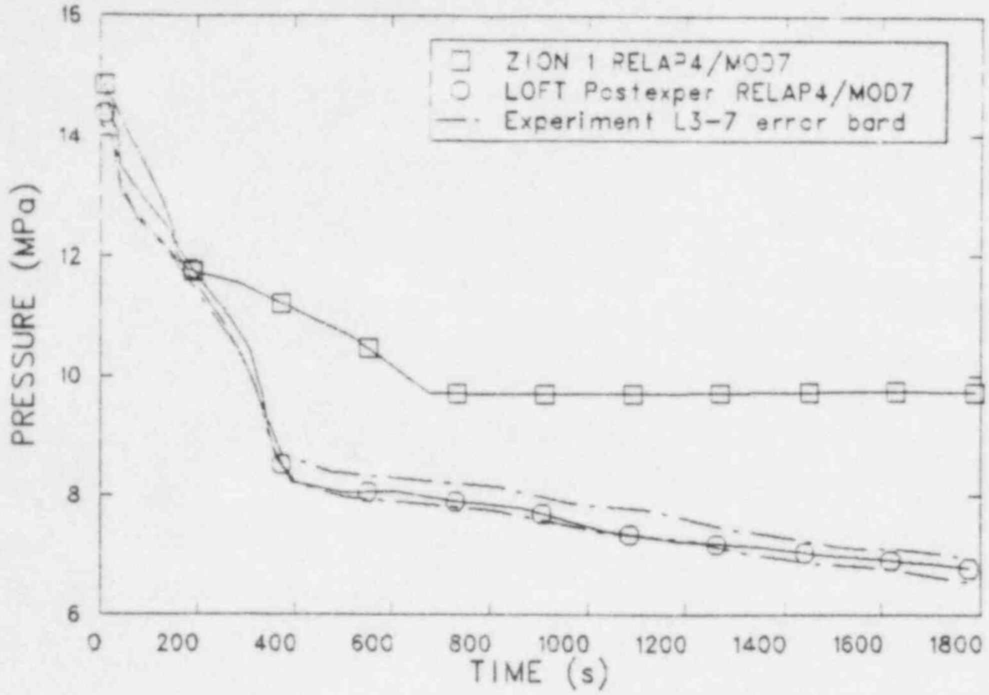

Figure 3-6. Pressure in the primary system for LOFT Experiment L3-7 (measured and calculated) and for a 1-in. diameter break in Zion-1 (calculated).

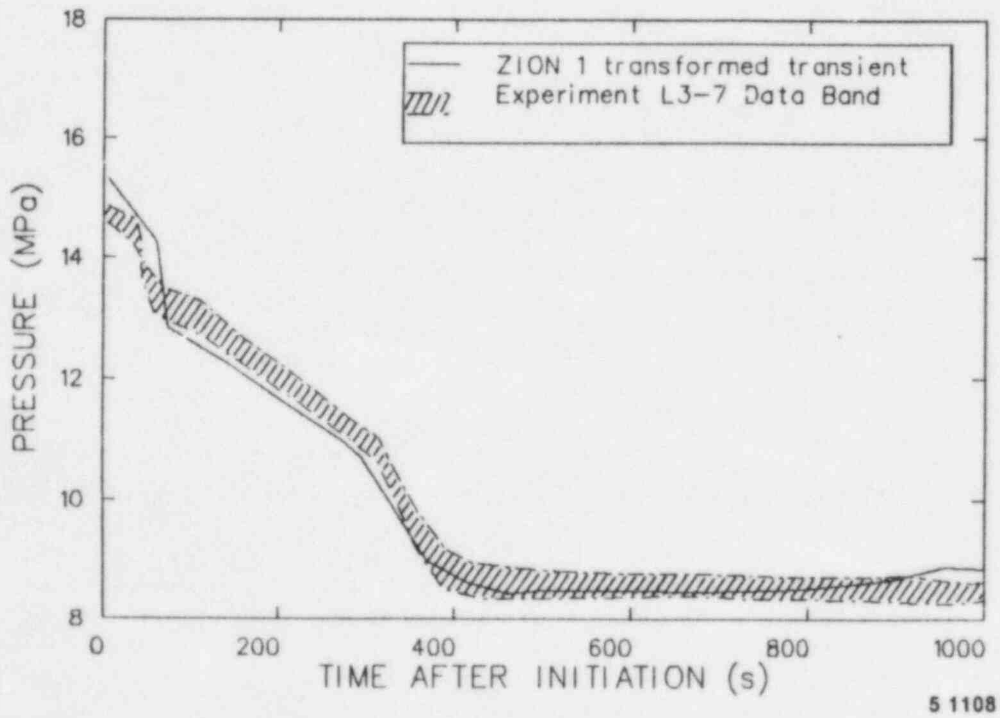

Figure 3-7. Pressure in primary system for LOFT Experiment L3-7 (measured and calculated) and for a 1-in. diameter break in Zion-1 using LOFT trip setpoints power-scaled break, and core bypass ratio.

3.4 References

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4. IMPACT OF LOFT RESULTS ON THERMAL-HYDRAULIC MODELING

In the nuclear industry, computers are used for design, safety analysis, and procedure development, as well as to drive reactor simulators used to train operators. In all these applications, accuracy is important, and best estimate models are used or referenced. Best estimate models use the most accurate code models and the best modeling techniques to predict the most probable results.

A number of thermal-hydraulic phenomena need to be properly modeled to produce accurate results. The discussion in this section focuses primarily on the experience with the RELAP4 and RELAP5 computer codes, the codes primarily used at the INEL by EG&G Idaho, Inc., to make preexperiment predictions for the LOFT experiments.

Two types of modeling are discussed: phenomenological and component. Phenomenological modeling deals primarily with the equations and correlations used to describe phenomena. One exception to this is pump modeling, which deals with correlations that describe the performance characteristics of a component.

Component modeling usually refers to the input model specified by the analyst using the modeling capabilities of the code. Component models consist of the nodalization plus specific phenomenological models necessary to describe the characteristics of the component.

The discussions of the various modeling topics in this section include a description of the phenomena or component being modeled, a description of the model, and a representative example of analytical results compared to experimental data.

4.1 Phenomenological Models

The phenomenological modeling considered in this section includes modeling of break flow, heat transfer, dynamic loading, and pumps.

4.1.1 Break Flow Modeling. Break flow modeling is very important in best estimate calculations. For large breaks, accurate break flow calculations are needed to accurately predict thermal-hydraulic phenomena in the reactor core, and to accurately predict core heatup. For small breaks, accurate break flow calculations are necessary to predict if and when core uncovery will occur.

The importance of accurate break flow modeling was demonstrated during LOFT Experiments L2-2[4-1] and L2-3.[4-2] Accurate predictions of the subcooled and saturated break flow and the transition from subcooled to saturated break flow were important factors in predicting the flow reversal through the core that quenched the fuel rods.

RELAP4/MOD6 has several critical flow models available. However, for large breaks, the break flow model combination that produces the best agreement with LOFT data is the Henry-Fauske[4-3] (subcooled) and homogeneous equilibrium (HEM) (saturated) critical flow models, with a break flow multiplier of 0.840 on both, with a transition from the Henry-Fauske model to the HEM model between 0 and 0.25% quality. Figure 4-1 illustrates the generally good correlation obtained with the Henry-Fauske/HEM combination and the broken loop cold leg flow for LOFT Experiment L2-2.

It should be noted that the LOFT large-break experiments were performed with a single set of nozzles. For a cold leg break, the cold leg nozzle diameter is 10.955 cm, has a 10% taper at the inlet, and is approximately 40.6 cm long. The hot leg nozzle diameter is 10.320 cm, has a 10% taper at the inlet, and is also approximately 40.6 cm long.

Hall compared the models in RELAP4/MOD6 with experimental data from Semiscale and the General Electric Company in Reference 4-4. The results of Hall's study also indicated that the Henry-Fauske/HEM combination critical flow model produced the best overall results. The data for Hall's study were obtained from flow tubes and nozzles with the following geometry:

1. Converging-diverging nozzles with throat diameters from 0.43 to 2.45 cm
2. Tubes with a 1.27 cm diameter and from 0 to 63.5 cm long
3. An orifice with a 1.27 cm diameter.

This study showed that break geometry is a very important parameter. As the length-to-diameter (L/D) ratio of the type of nozzles used for the

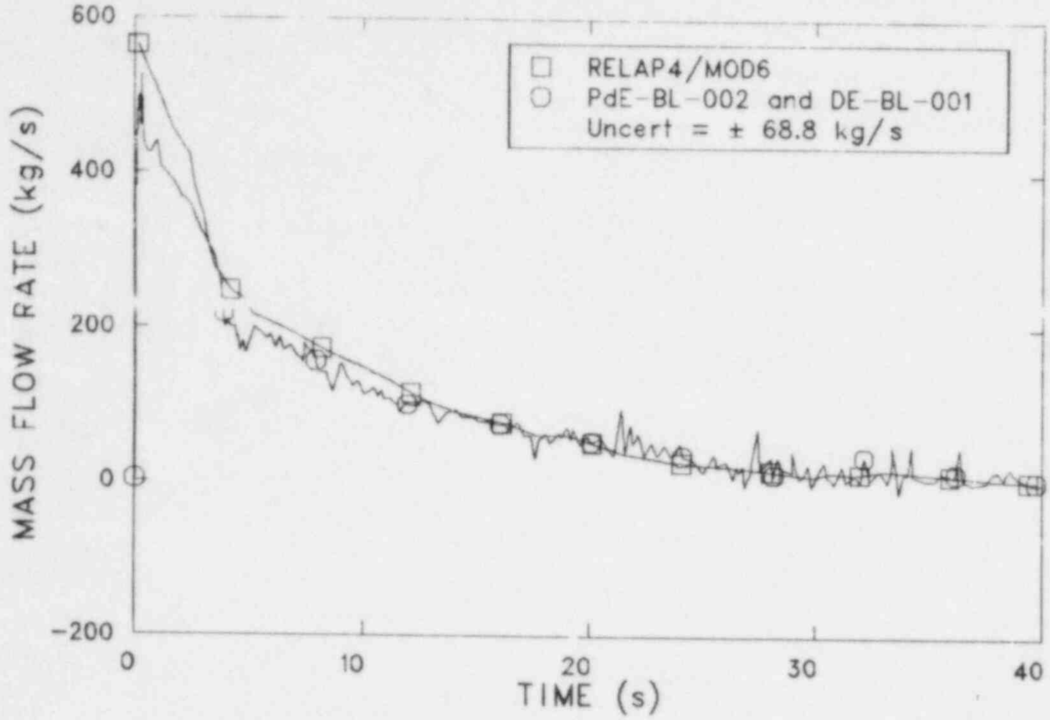


Figure 4-1. Measured and predicted mass flow rate in broken loop for LOFT Experiment L2-2.

LOFT large-break experiments increases, the break flow decreases rapidly until the L/D is in the 5-to-10 range, and then decreases very slowly as the L/D ratio increases to 50 or more. In the subcooled regime, the break flow through a very short nozzle (L/D $\sim$ 0) is more than three times the break flow through a long nozzle (L/D = 50). In the saturated regime, the nozzle L/D ratio also has a large effect on break flow. Hall's work demonstrated the importance of considering the L/D ratio of a nozzle when making break flow predictions. This is particularly important since the same size break in different parts of a system can produce significantly different flows and make comparisons of accident consequences invalid.

Conditions upstream of a large break tend to be more nearly homogeneous (single-phase liquid, dispersed bubble, mist, etc.). Near the end of a large-break blowdown, when ECC water reaches the break, the flow regime becomes slug flow. This is apparent from fluid density, momentum flux measurements upstream of the break, and from break flows computed from measurements. The RELAP4 and RELAP5 codes tend to homogenize, or smooth, the flow. The computed mixture tends to be more uniform (homogeneous or stratified), and the break flow is computed from the average conditions. The effect of flow regime on break flow has not been investigated, but could be significant.

Despite the possibility of slug flow near the end of a large break LOCA, computing critical flow for small breaks is more difficult than it is for large breaks. Conditions upstream of a small break can be more variable over the whole pressure range. Stratified flow has been found to have a particularly important effect on break flow. The critical flow models used in the RELAP4 codes, MOD0 through MOD7, were developed for large break transients, and did not contain a stratified flow model. An attempt was made to compensate for this modeling deficiency. Specifically, the control volume, or cell, immediately upstream of the break was split into two cells of equal fluid volume. The first new cell maintained the full pipe geometry of the old cell. The second new cell, located next to the break orifice, had a cell height equal to the orifice diameter. The junction between the two cells had the same diameter and flow area as the orifice, but the critical flow model was not activated. Other cell and junction input parameters were chosen so as to minimize the pressure difference between the two new cells. With phase separation in the upstream cell and a RELAP4 model called the junction smoothing model at the intervening junction, the

fluid entering the downstream cell was representative of the fluid at the orifice elevation rather than the pipe average fluid. The critical flow was then calculated based on the average fluid conditions in the downstream cell. This technique provided an improved critical flow calculation prior to orifice uncovery. However, the complex phenomena which occur at the entrance of a partially covered orifice could not be modeled. Break flow multipliers derived from separate effects testing were applied to the critical flow at the break orifice. A value of 1.13 was applied to the Henry-Fauske model, and 1.8 was applied to the HEM. These coefficients were obtained empirically from separate effects testing. The default transition quality of 2% was used. Figure 4-2 shows predicted and measured mass flow for LOFT Experiment L3-7. The prediction is with the RELAP4/MOD7 stratified flow model.

The RELAP5/MOD1 computer code[4-5] uses a choked flow model developed by Ransom and Trapp.[4-6] The RELAP5 choked flow criteria for two-phase flow is based on thermal equilibrium (equilibrium interphase mass transfer) and allows interphase slip in the throat and stratified or homogeneous flow upstream of the break plane. Subcooled choked flow is computed using the Bernoulli equation and the differences between the upstream pressure and the vaporization pressure at the throat. The vaporization pressure at the throat is obtained from the Alamgir-Lienhard-Jones correlation (References 4-7, 4-8, and 4-9).

As with the RELAP4 break flow models, break flow coefficients are needed to obtain good agreement between RELAP5 and LOFT experiment results. Figures 4-3 and 4-4 show a comparison of postexperiment calculations with data from LOFT large-break Experiment L2-3.[4-10] For the cold leg break, a subcooled coefficient of 0.98, and a saturated coefficient of 0.84 was used. For the hot leg, a coefficient of 1.0 was used for the subcooled and saturated coefficients. The hot leg coefficient of 1.0 was necessary because of the flow control effect caused by the pump and steam generator simulators in the hot leg piping.

RELAP5/MOD1 break flow predictions for small breaks are not as good as the corresponding large-break predictions. Figure 4-5 shows a postexperiment calculation compared with measured data for LOFT small-break Experiment L3-6.[4-11] A nozzle discharge coefficient of 0.84 for the subcooled and saturated critical flow regimes was applied to the model in order to compensate for the

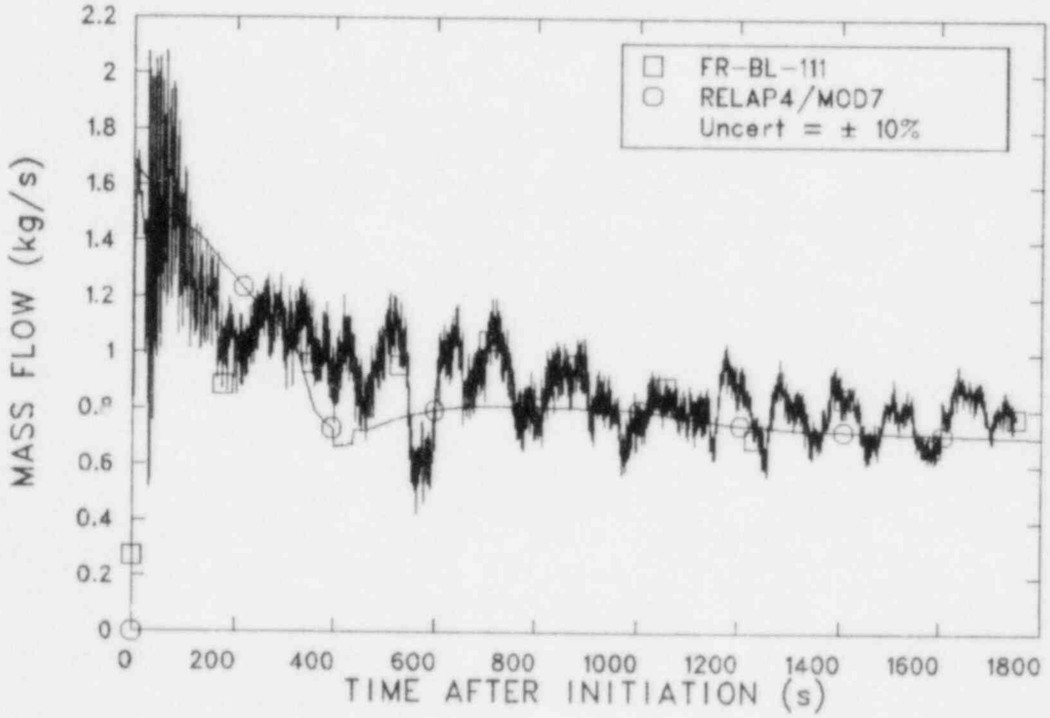


Figure 4-2. Measured and predicted mass flow rate in broken loop for LOFT Experiment L3-7 (prediction is with RELAP4/MOD7 stratified flow model).

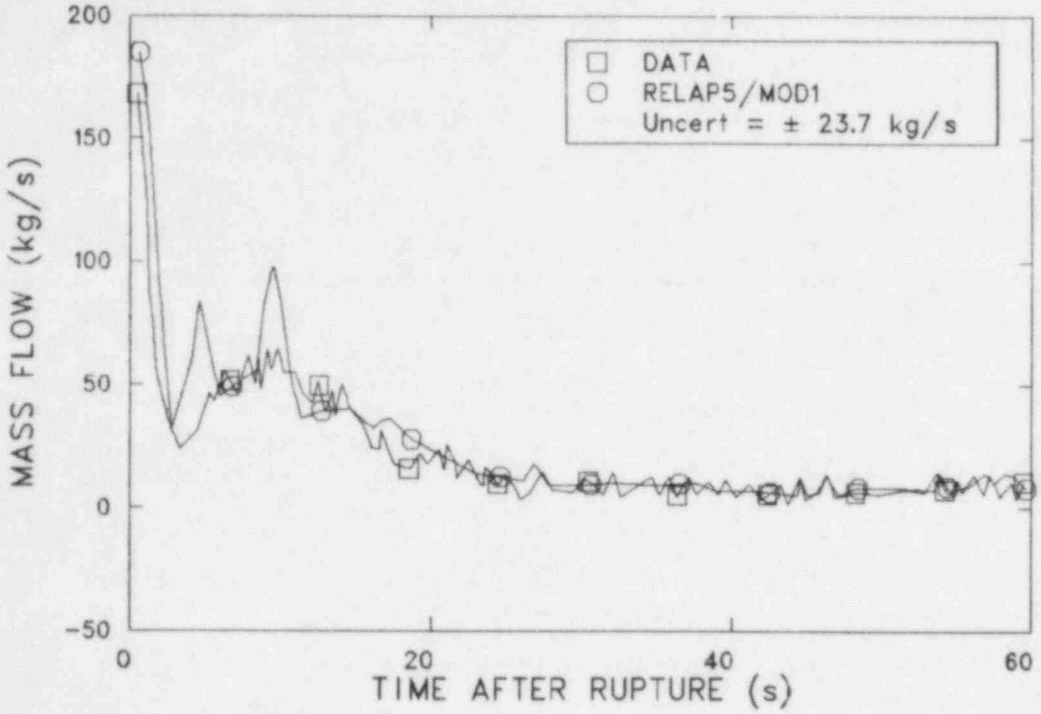


Figure 4-3. Measured and calculated mass flow rate in broken loop hot leg for LOFT Experiment L2-3.

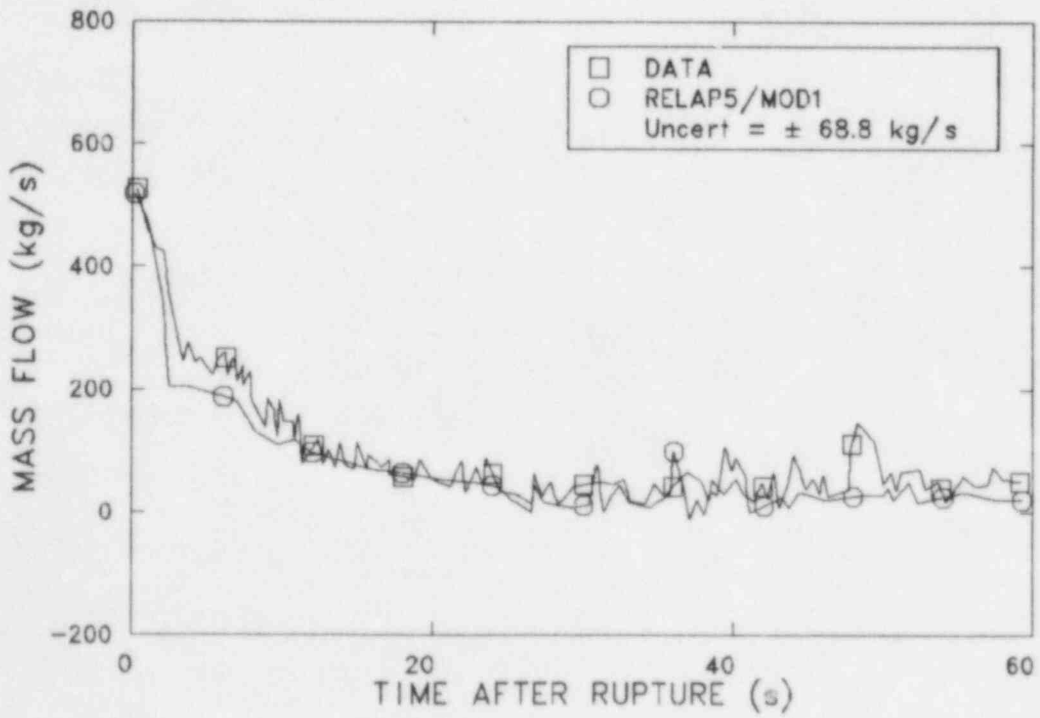


Figure 4-4. Measured and calculated mass flow rate in broken loop cold leg for LOFT Experiment L2-3.

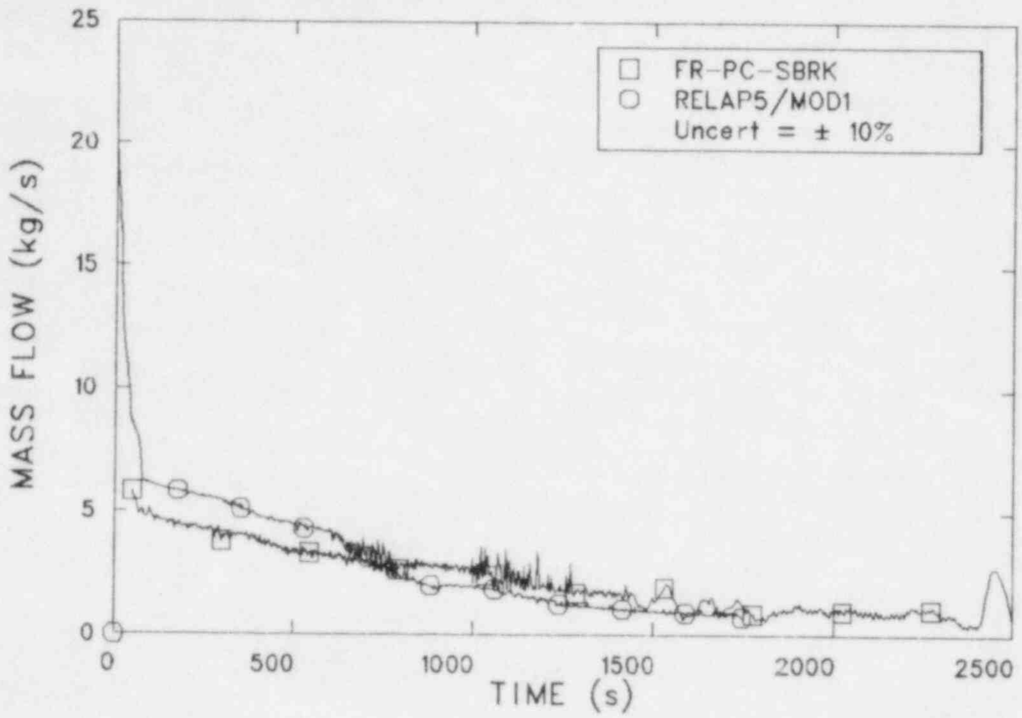


Figure 4-5. Measured and calculated break mass flow rate for LOFT Experiment L3-6.

actual nozzle characteristics. This value provided the best calculation of break flow and pressure response when the upstream densities were properly calculated.

A study reported in Reference 4-12 showed that the empirical break flow multipliers currently used with one-dimensional critical flow models are a function of multidimensional effects and the L/D ratio of the nozzle. The multidimensional analysis of the Semiscale Henry nozzle reported in Reference 4-12 showed that theoretically, a break flow multiplier of 0.833 was necessary to correct a one-dimensional critical flow analysis for multidimensional effects. This compares well with the empirical multiplier of 0.84.

All of the one-dimensional break flow models are based on the equilibrium assumption (thermal equilibrium between phases) or the frozen assumption (no vapor production during depressurization). The RELAP5/MOD1 subcooled choked flow model is also based on thermal equilibrium, but includes the effect of rapid depressurization on the pressure at which flashing occurs. The effect of nonequilibrium vapor production was discussed in Reference 4-13, and is shown to be important during the early portion of a blowdown when subcooled water enters the nozzle.

4.1.2 Heat Transfer Modeling. Accurate predictions of the core thermal response during large-break LOCAs require precise calculations of core heat transfer. Postcritical heat flux (post-CHF) heat transfer is particularly important since departure from nucleate boiling (DNB) can occur very early in severe accidents. The preexperiment analyses of LOFT large-break LOCEs L2-2 and L2-3 did not predict a quench of the core which occurred between 5 and 10 s after experiment initiation.[4-1,4-2] For Experiment L2-2, the problem with the calculation was believed to be related primarily to the break flow calculations. Further analysis showed that the calculation of heat transfer was also incorrect.

The postexperiment analysis of the large-break Experiment L2-3[4-2] showed that the heat transfer packages in RELAP4 and RELAP5 (which included modified Condie-Bengston and Groeneveld film boiling correlations) could not predict the quenching of the core observed during low-flow, low-quality, and high-pressure (~7 MPa) flow conditions experienced during the experiment, because these conditions were outside the range of data that they were developed from. A review of heat transfer data and additional heat transfer research has led to development and use of improved film boiling heat transfer correlations and concomitant improvements in predictive capability.

The inability to predict the LOFT rewet behavior led to an evaluation of many of the correlations used in the heat transfer portions of the computer codes. A comparison of the minimum film boiling temperature (ΔT_{min}) versus X (quality) curve data from a test conducted by General Electric Company[4-14] and data generated by the HTS2 routine of RELAP4/MOD6 is shown in Figure 4-6. Figure 4-6 shows that the RELAP4/MOD6 heat transfer correlation is in poor agreement with the experiment data and will never predict rewetting when the local equilibrium quality is greater than 50% at 6.87 MPa. The reason is that Hsu-Beckner's CHF correlation[4-15] used by RELAP4/MOD6 was developed by correlating the Semiscale high-flow CHF data and is not suitable for LOFT mid-flow cases (mass flux of 100 to 600 kg/s-m^2). For the Experiment L2-3 postexperiment analysis,[4-2] the Biasi CHF correlation[4-16] was installed into RELAP4/MOD6 which covered the mass flux ranges from 100 to 6000 kg/s-m^2. Figure 4-7 presents a similar comparison to that shown in Figure 4-6. The Biasi CHF correlation is in better agreement with the data. With the Biasi CHF correlation, RELAP4/MOD6 can calculate rewetting for quality higher than 50%.

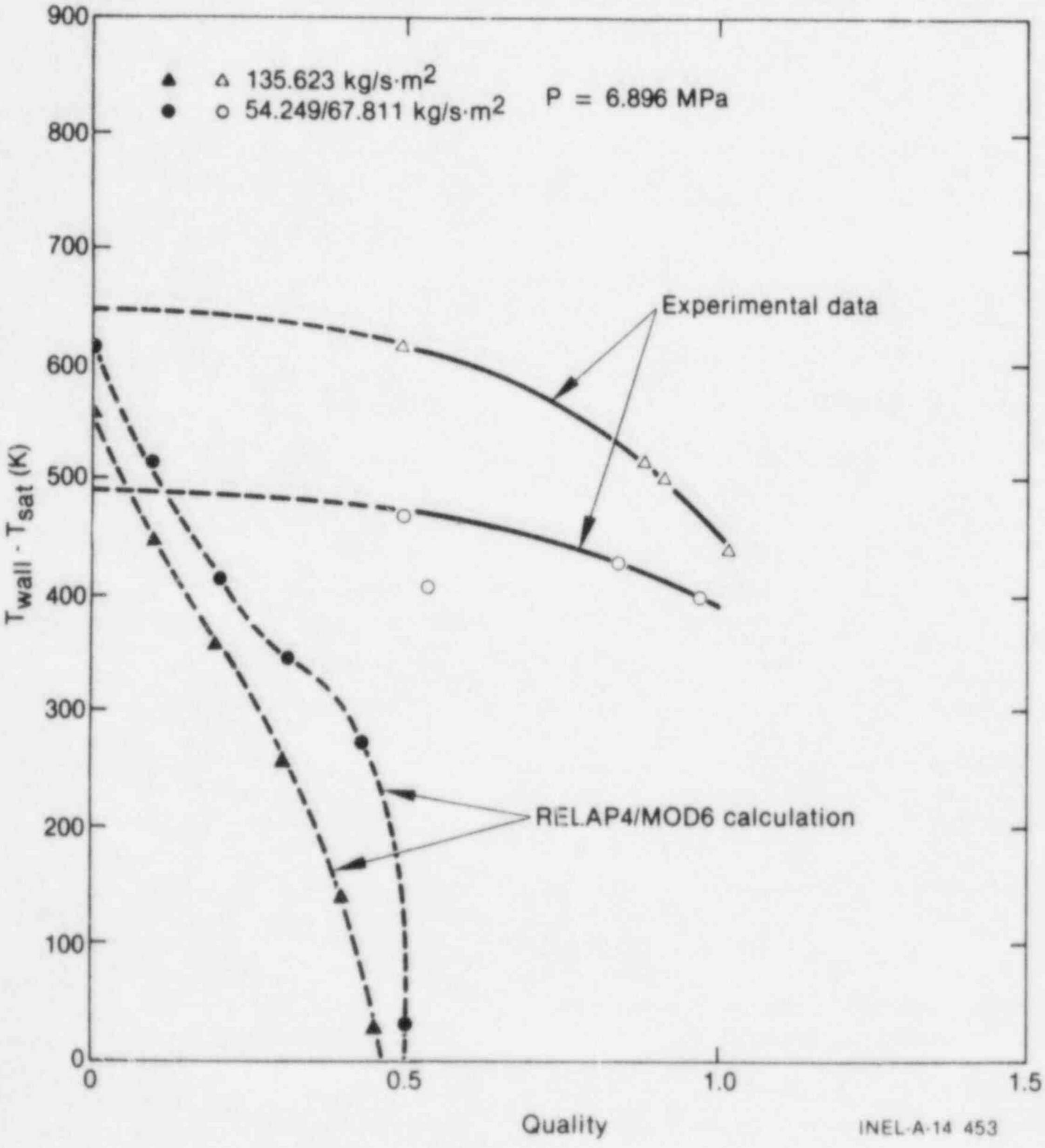


Figure 4-6. Minimum film boiling temperature calculated by RELAP4/MOD6 compared with experimental data from the General Electric Company test.

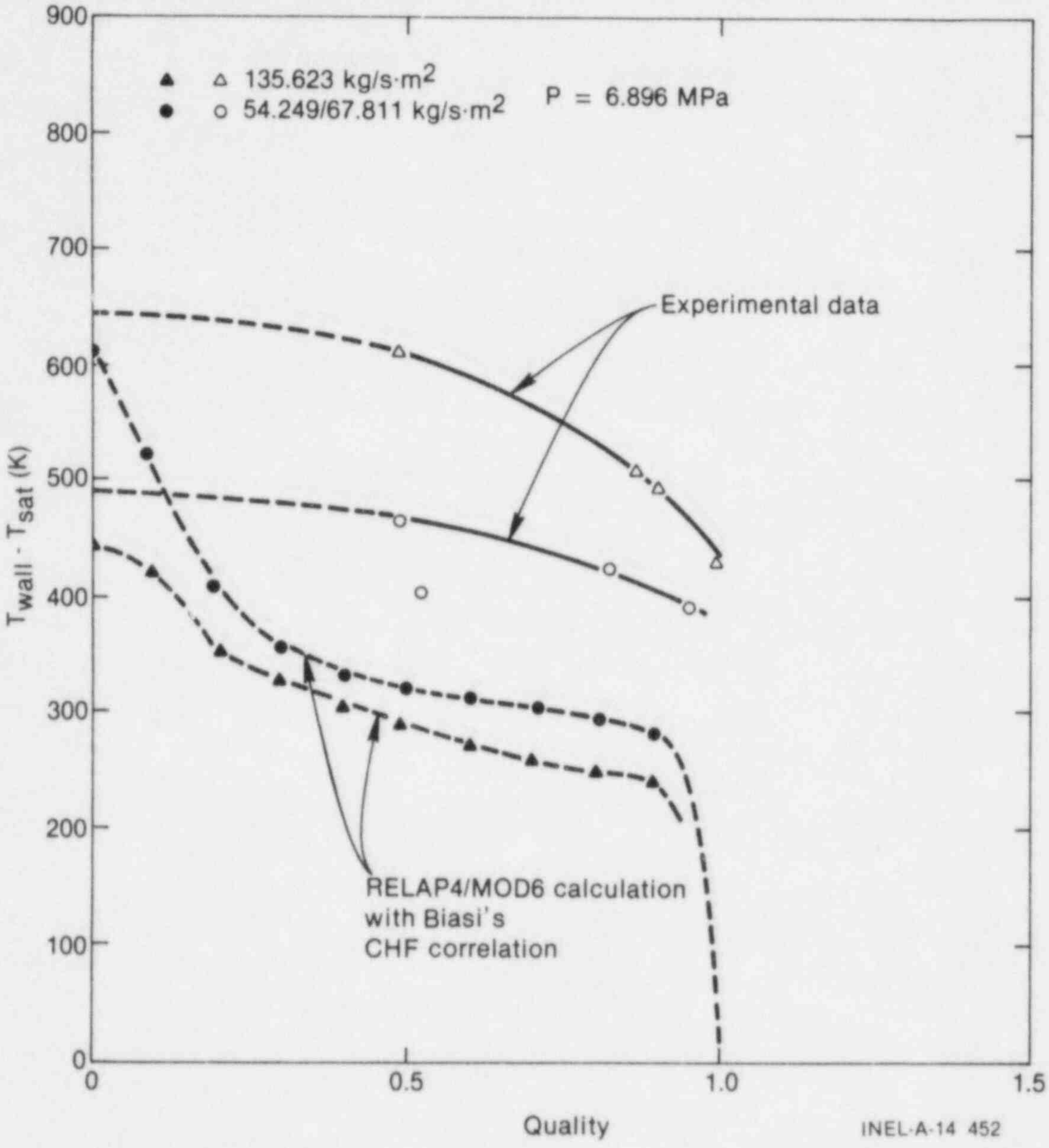


Figure 4-7. Minimum film boiling temperature calculated by RELAP4/MOD6 with the Biasi CHF correlation compared with experimental data from the General Electric Company.

During the preexperiment analysis for Experiment L2-5,[4-17] the Biasi-Zuber CHF correlation was installed in the RELAP5/MOD1 computer code. The Biasi-Zuber CHF correlation is based on the Biasi[4-16] and the modified Zuber[4-18,4-19] CHF correlations. Adding the Zuber CHF correlation extended the range of the Biasi from 100 to 6000 kg/s-m^2 to -667 to 6000 kg/s-m^2. The results of Experiment L2-3 were recalculated in Figures 4-8 and 4-9 using RELAP5/MOD1 with the Biasi-Zuber CHF correlation. The comparisons show that RELAP5 with the Biasi-Zuber CHF correlation predicts the core rewet very well, but predicts a lower maximum cladding temperature than measured.

LOFT Experiment L2-5 was a large-break LOCE that differed from large-break LOCEs L2-2 and L2-3, in that the primary coolant pumps were turned off within 1 s and simultaneously decoupled from their external flywheels to provide an atypically fast pump coastdown. The experiment was designed to cause early flow stagnation in the core to prevent the core-wide rewet that occurred

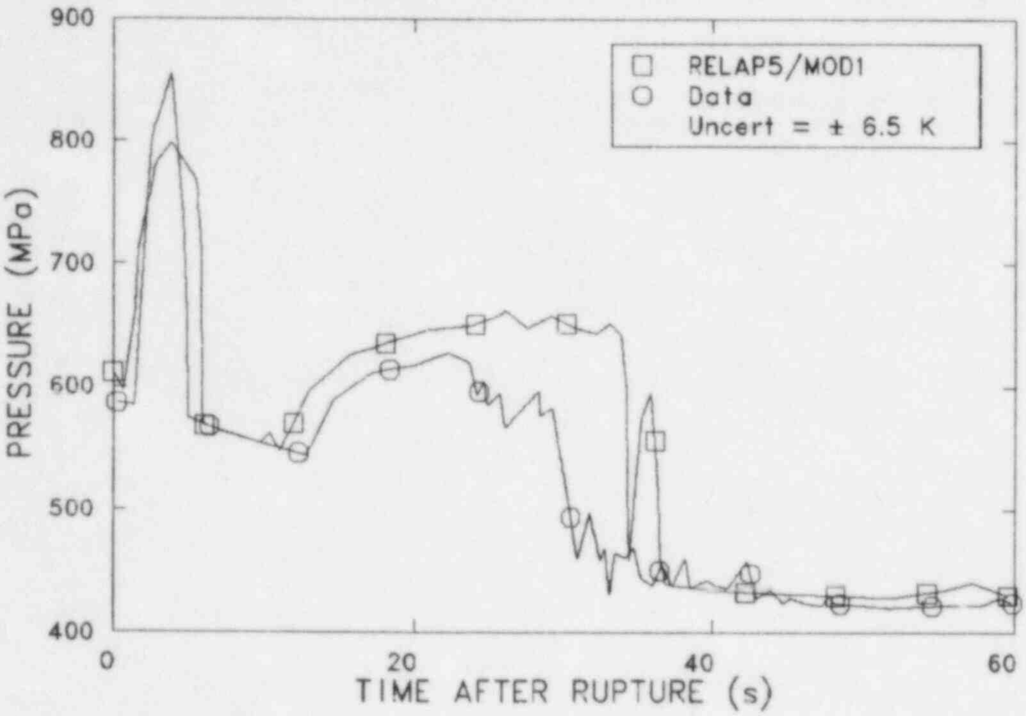


Figure 4-8. Measured and calculated fuel cladding temperature for high-powered fuel rods at bottom of core for LOFT Experiment L2-3.

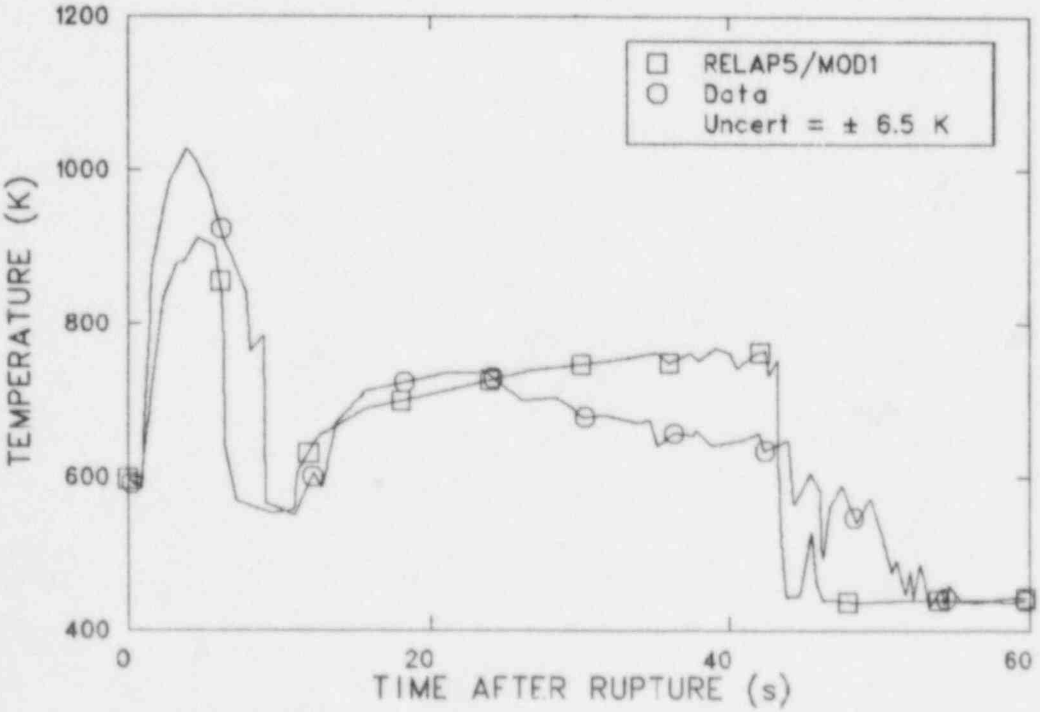


Figure 4-9. Measured and calculated fuel cladding temperature for high-powered fuel rods at hot spot of core for LOFT Experiment L2-3.

in Experiments L2-2 and L2-3. Figure 4-10 shows a comparison of maximum cladding temperature with the RELAP5/MOD1 predictions for Experiment L2-5. The figure shows that the cladding temperatures were predicted very well (within 80 K) for the first 30 s. Beyond 30 s, the predicted cladding temperatures were lower than the measured temperatures (on the order of 200 K), and the first quench was predicted to occur about 10 s earlier than actually measured.

4.1.3 Dynamic Loading Modeling. Data obtained from the LOFT system checkout tests (mini-blowdown Experiment Series L0) and the first four nonnuclear blowdown experiments (Experiment Series L1) can be used to evaluate the capability to predict dynamic loads on blowdown suppression systems. During a large-break LOCE, the LOFT blowdown suppression system is subjected to large transient loads as fluid discharged from the primary system is collected in a header (dry well) and is condensed under water in the blowdown suppression tank (wet well).

The transient loads in the blowdown suppression tank are caused by three mechansims:

1. Loads associated with clearing the vents of noncondensable gas due to increased pressure and flow in the dry well
2. Loads on internal structures and tank surfaces due to impact of high-velocity water
3. Loads due to sudden collapse of bubbles in the wet well and possible fluid structure interaction.

At the time that the LOFT Experimental Program was to be initiated, conservative licensing calculations[4-20] predicted the loads on the LOFT blowdown suppression tank would exceed the design limits of the tank (predicted loads were $\sim$ 18 x 10^6 N in the down direction, $\sim$ 13 x 10^6 N in the up direction, and 8.3 x 10^6 N longitudinally; the design limits of the tank are 13 x 10^6 N in the down direction and 4.5 x 10^6 in the up direction). A more realistic analysis in Reference 4-21 indicated that the suppression tank loads would be close to the design limits, but no experimental data were available to verify the analysis. The latter analysis studied the response of a pressure suppression tank (PST) partially filled with water into which air and steam were discharged. This analysis treated a single downcomer venting into a vertical cylindrical tank with a hemispherical bottom, see Figure 4-11. The noncondensable gas initially in the header was assumed to be completely discharged before the steam began to flow into the PST. Since the water in the PST was nearly incompressible and started from rest, a potential flow solution was applied.

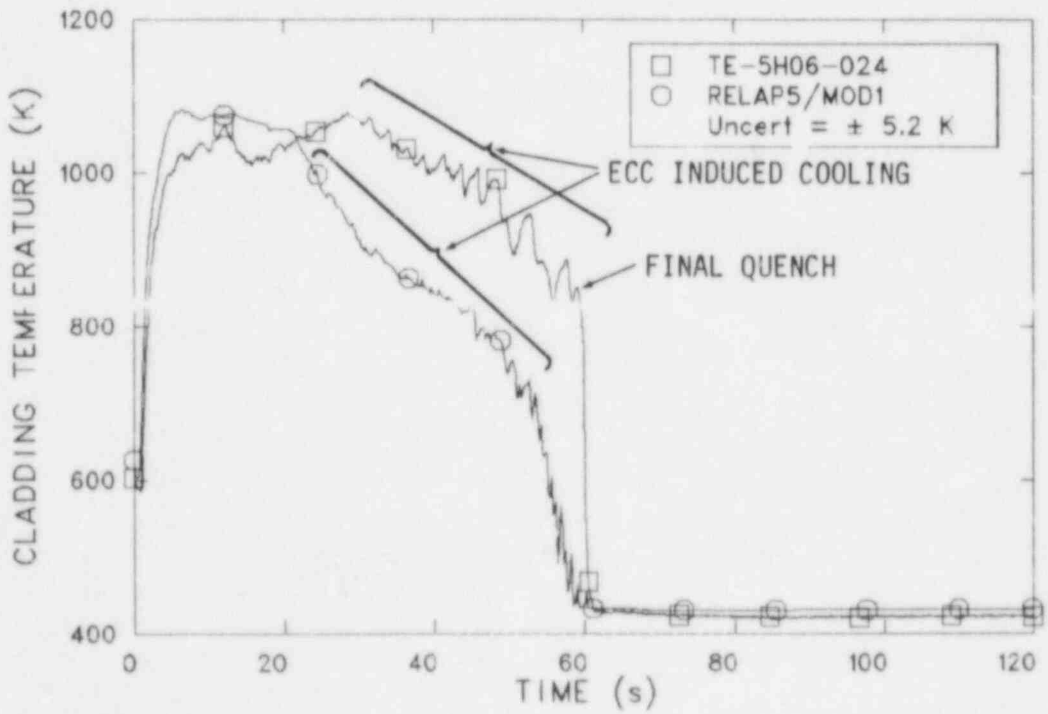


Figure 4-10. Measured and predicted maximum fuel cladding temperature for LOFT Experiment L2-5.

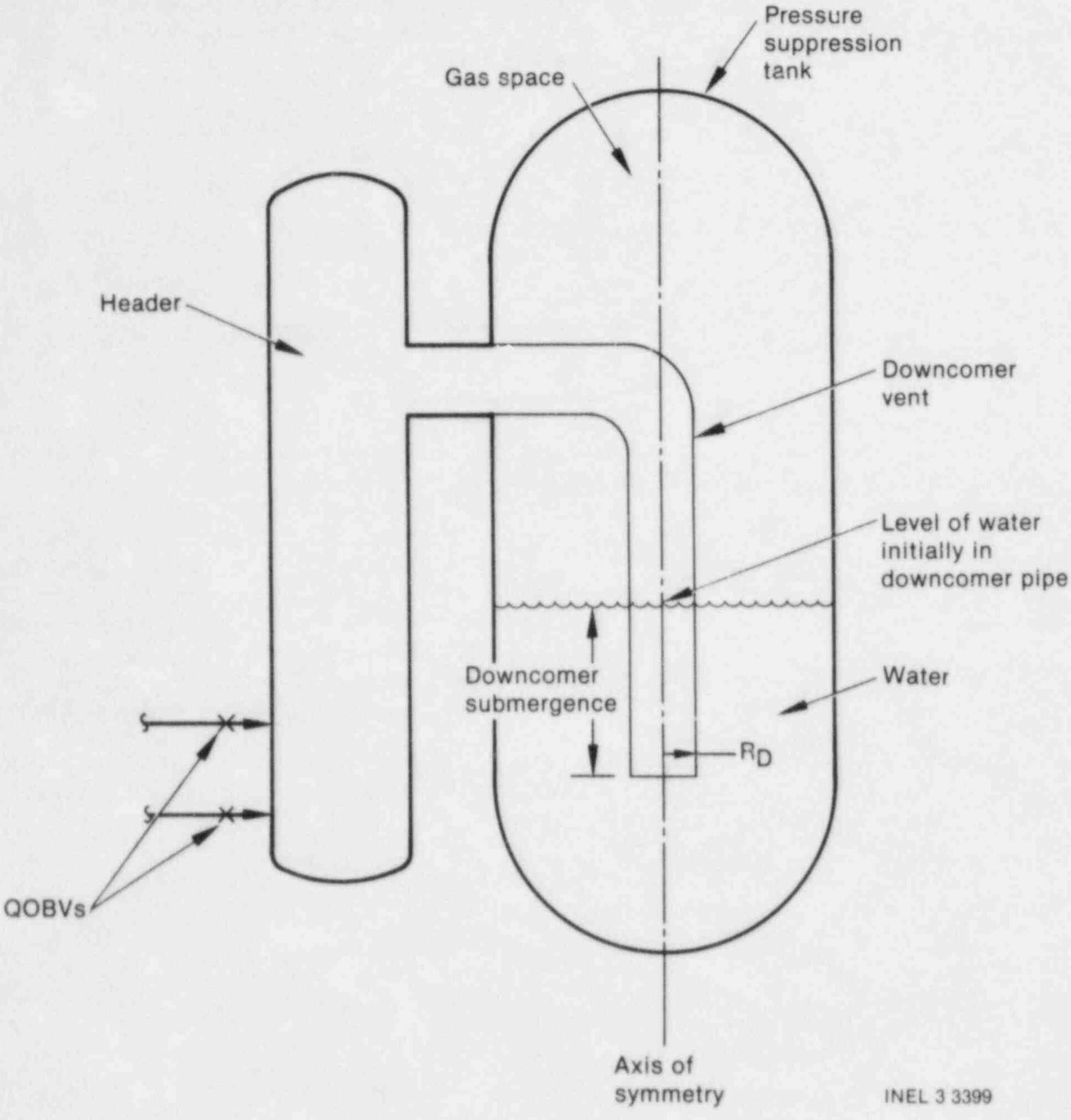


Figure 4-11. Schematic diagram of idealized pressure suppression system.

The analysis was divided into two phases: clearing water from the downcomer vent (vent clearing) and the formation and growth of a bubble at the bottom of the vent. The results of the analysis were reasonably valid until the bubble broke the water surface or until the effects of steam condensation became important. The driving pressure in the header was obtained from the output of a calculation using the RELAP computer code.

Results of the analysis were compared with experimental results from the Semiscale Program. The Semiscale experimental results were particularly good because the geometry of the Semiscale PST was very similar to the analytical approximation. Experimental and predicted pressure directly below the Semiscale downcomer vent are shown in Figure 4-12 for two Semiscale tests. The predictions were in close agreement with the experimental data.

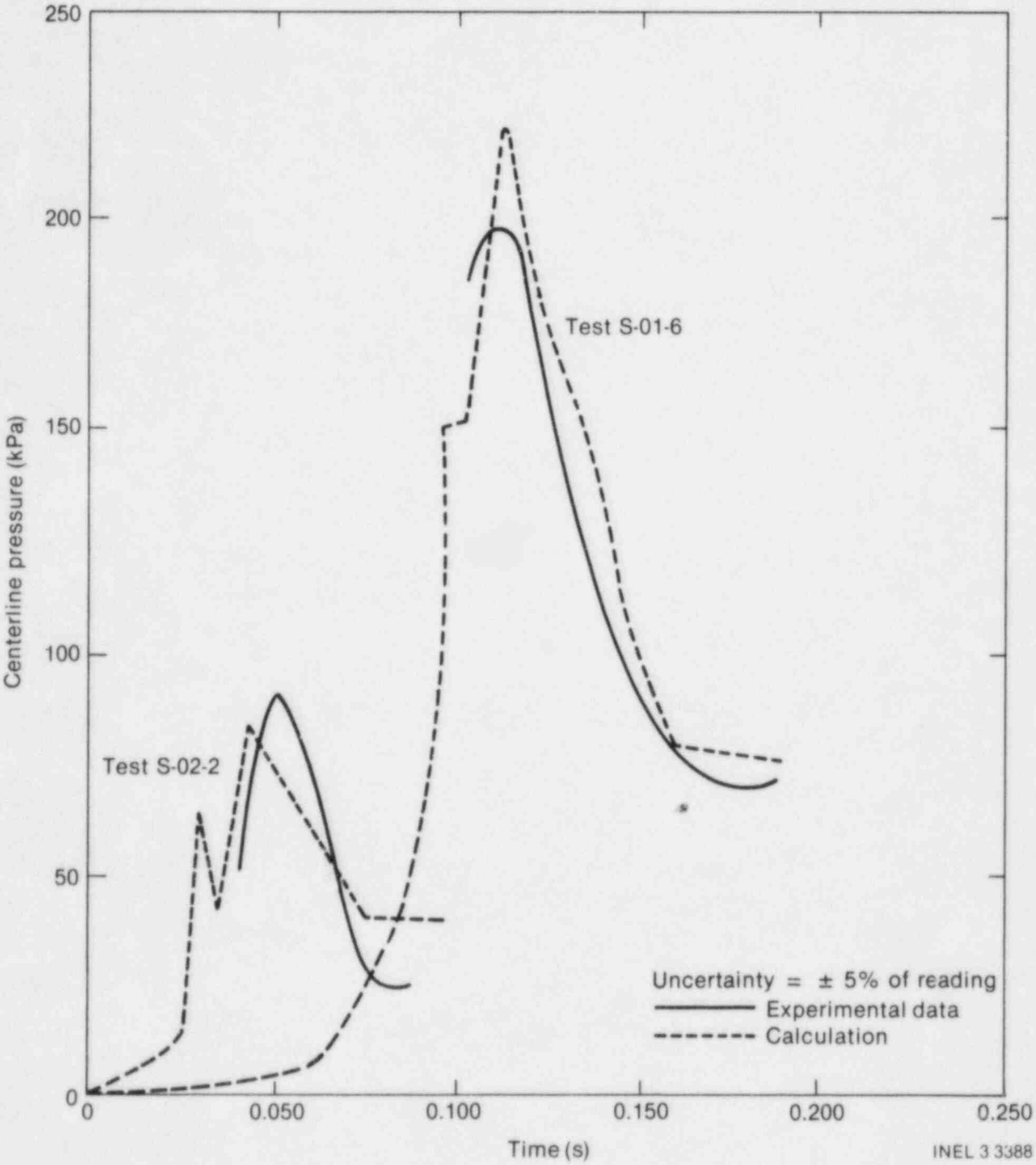


Figure 4-12. Measured and calculated pressure near bottom of pressure suppression tank for Semiscale Tests S-01-6 and S-02-2.

The corresponding load predictions for Semiscale were also excellent, being within 18% of measured values.

The same analysis was used to predict pressure and vertical loads on the LOFT blowdown suppression tank (BST). The LOFT BST shown in Figure 2-2 is very different from the simple idealized pressure suppression system. The tank is a horizontal cylinder with four vertical downcomer vents. In order to perform the analysis, several approximations had to be made. First, the BST was divided into four equal volumes, one for each downcomer vent. Then it was assumed that each BST volume would behave like an equivalent cylindrical BST. Finally it was assumed that the pressure and flow history in each downcomer vent would be identical and exactly in phase.

A series of 10 mini-blowdown experiments (Experiment Series LO) was conducted in LOFT to provide experimental data to verify the analysis reported in Reference 4-21 to determine the maximum permissible suppression tank downcomer submergence level for LOFT nonnuclear LOCE L1-1. The experiments consisted of venting the volumes between the QOBVs and the upstream isolation valves into the BST. The experiments and experimental results are reported in detail in Reference 4-22.

The results were mixed, the pressure predictions directly beneath the downcomer vents agreed well with the experimental data, but the load measurements were generally lower than predicted. A summary of the LOFT and Semiscale predictions along with the associated experimental data is presented in Table 4-1. The difference between predicted and analytical results for LOFT were due to the difference in geometry and unsynchronized venting of water through the downcomer vents which resulted in lower loads on the BST; however, the analysis was considered valid for predicting trends. The analysis and experimental data from the mini-blowdown experiments were used to establish a downcomer vent submergence level of 30 cm for LOFT nonnuclear LOCE L1-1.

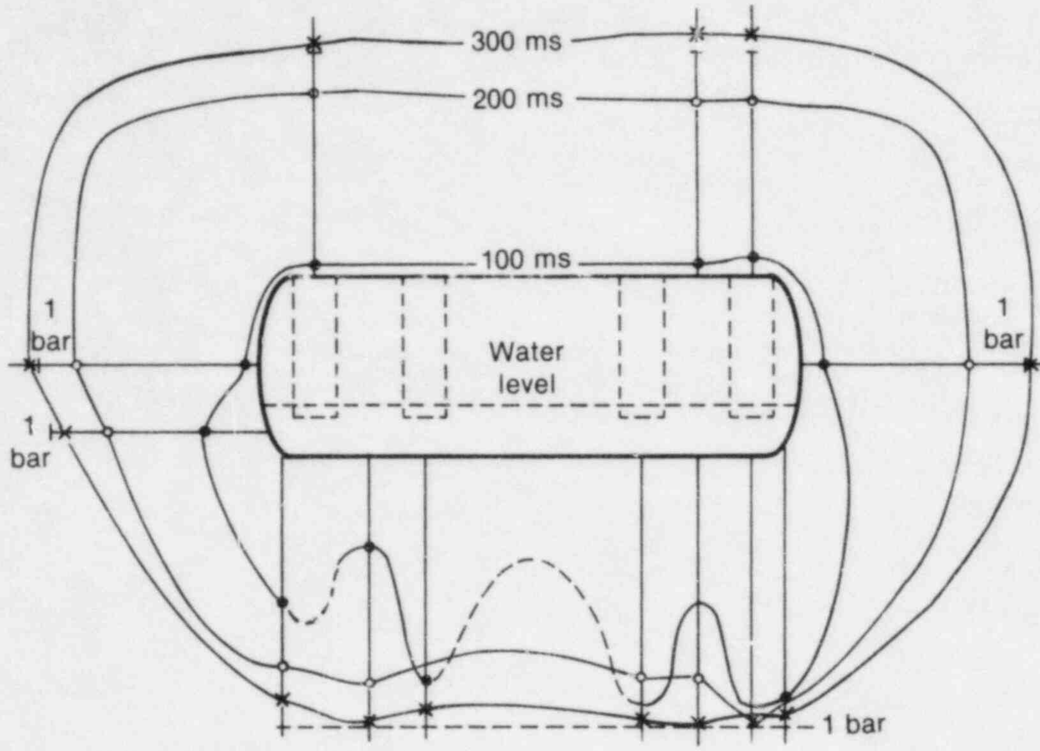
BST loads were also measured during the first four LOFT nonnuclear LOCEs (Experiment Series L1). The measured loads were all less than predicted, and well within design limits. Figure 4-13

Table 4-1. Comparison of LOFT mini-blowdown and Semiscale test results with analysis

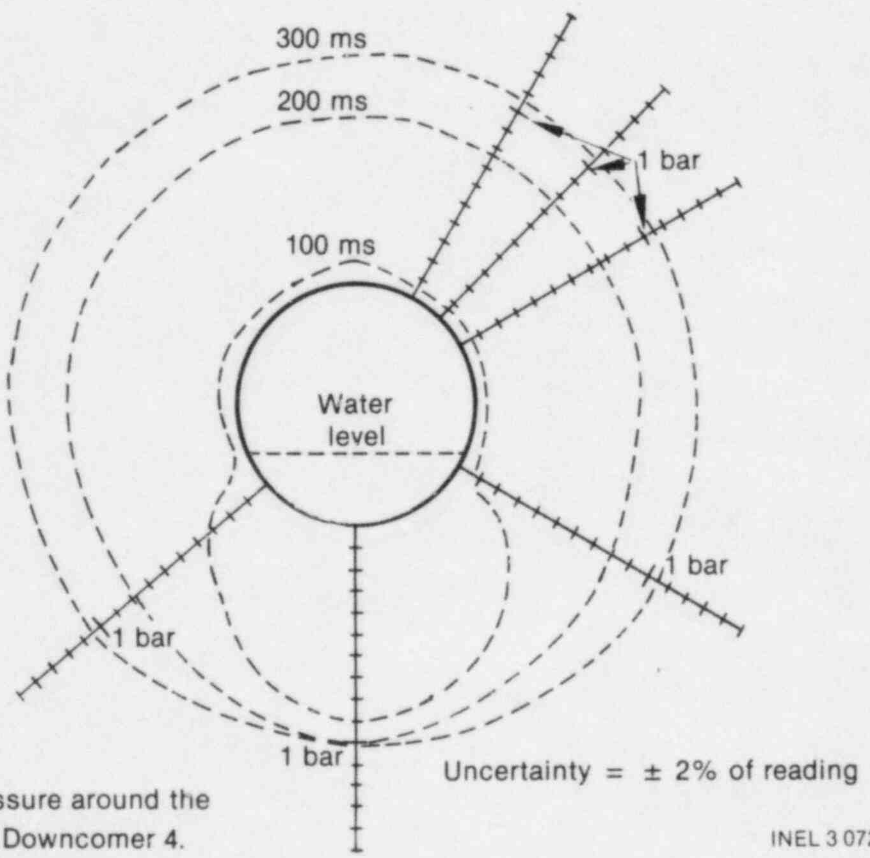
Experiment	Downcomer Submergence (cm)	Depth (cm)	Maximum Download (N_f) Data	Maximum Download (N_f) Calculated	Maximum Upload (N_f) Data	Maximum Upload (N_f) Calculated	Volume Above Pool Per Downcomer (m^3)
Semiscale[a]							
S-02-2	42.2	58.5	57 800	48 900	NA	19 100	2.16
S-01-6	75.7	166.1	146 900	124 [illegible]	111 200	97 900	1.32
LOFT[b]							
LO-3	30.5	116.7	498 400	801 000	[illegible]69 100	1 010 100	14.1
LO-3A	30.5	116.7	427 200	801 000	244 700	1 010 100	14.1
LO-3B	30.6	116.7	400 500	801 000	298 100	1 010 100	14.1
LO-8	40.6	126.8	618 500	84[illegible]	369 300	1 054 600	13.3
LO-2	15.3	[illegible]1.5	409 400	1 [illegible]	124 600	1 450 700	15.3
LO-3C	30.5	116.7	774 300	1 619 800	391 600	1 748 800	14.1
LO-4	40.6	126.8	1 815 600	2 162 700	1 085 800	2 269 500	13.3
LO-5	58.4	144.8	1 904 600	2 380 700	907 800	2 500 900	11.9
LO-9	25.4	111.6	360 500	543 400	173 500	1 139 200	14.4
LO-10	25.4	111.6	1 014 600	1 860 100	422 700	1 984 700	14.4

a. The Semiscale suppression tank is 3.41 m long and 1.03 m in diameter. The downcomer is 20.3 cm in diameter, and the total volume in the header and downcomer pipe is 0.59 m^3.

b. The LOFT suppression tank is 10.2 m long and 3.35 m in diameter. The downcomer is 61.0 cm in diameter, and the total volume in the header and downcomer pipe is 22.2 m^3.



a. Pressure around the tank.



b. Pressure around the tank at Downcomer 4.

Figure 4-13. Pressure profile in the blowdown suppression tank for LOFT Experiment L1-1.

illustrates the pressure distribution around the wet well for Experiment L1-1 at 100, 200, and 300 ms from the start of the transient and were plotted as increasing from the wet well outward. The pressure distributions were only approximate, however, due to the number of available pressure transducers. Figure 4-13a shows the distribution along the wet well. Early in the transient, the pressure was seen to be highest directly under the downcomers, with a significant pressure reduction with distance from each vent exit. Higher pressures were noted under Downcomer 4 at the closed end of the header. The pressure in the wet well gas space was equal throughout the gas space at any single time. At early times (100 ms), the pressure along the bottom of the tank was larger than in the gas space, which resulted in a net download on the wet well. At later times (200 ms) the pressure in the gas space was larger, which resulted in a net upload on the wet well.

Figure 4-13b illustrates the circumferential distribution of pressure at Downcomer 4. Behavior similar to that illustrated by Figure 4-13a is noted. Pressure distributions circumferentially around the cylindrical wet well are symmetrical and produce no net side loading.

The major pressure increase (initial bubble growth) under all four downcomers does not occur at the same time. This can be seen in Figure 4-14, which shows pressure beneath Downcomers 1 and 4 during Experiment L1-3A. This phase difference in the tank loading is a major contribution to the relatively small loads. LOFT loads are much less than calculated by licensing-type assumptions and are less than calculated by codes specially developed based on uniform behavior in each downcomer. When performing calculations for suppression tanks, the codes should account for nonuniform venting behavior to properly calculate loads.

4.1.4 Pump Modeling. Pump behavior has been demonstrated to have a strong influence on PWR behavior during certain types of accidents. Pump characteristic curves are empirically developed by pump manufacturers and uniquely define head and torque response of a pump as functions of volumetric flow and pump speed. Typically, four-quadrant pump curves are generated for single-phase conditions. The four-quadrant curves can be converted to simple homologous curves where head-and-torque ratios (actual value to rated value) are input as functions of pump speed and volumetric flow rates.

Homologous pump curves for single-phase conditions are readily available. However, data relative to the response of pumps under two-phase conditions are very scarce.

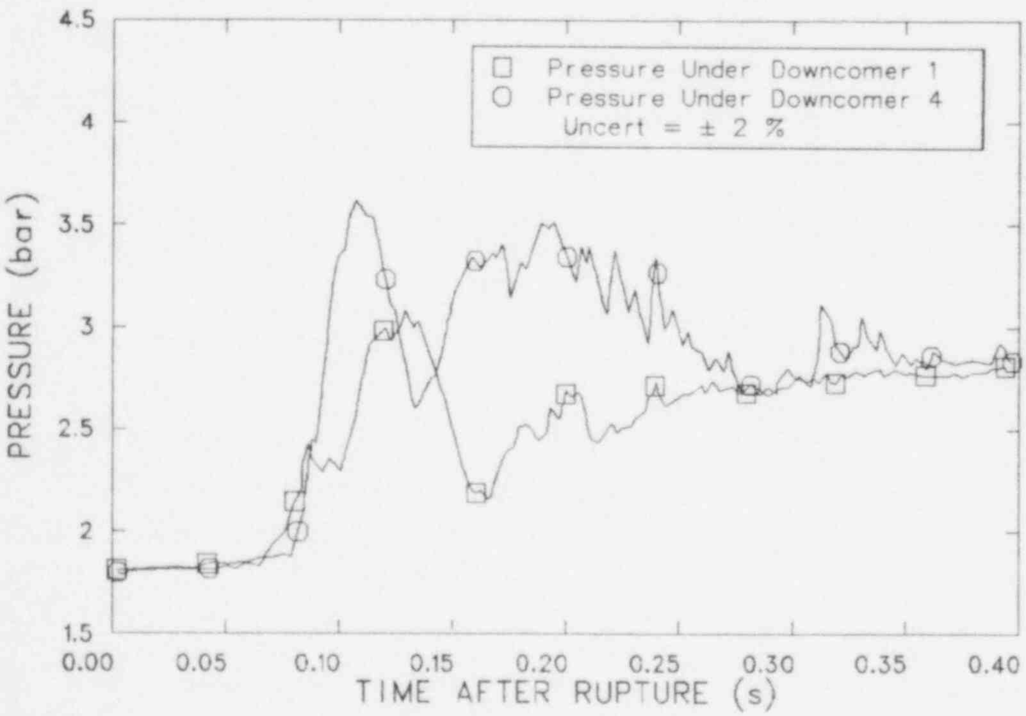


Figure 4-14. Pressure variation between Downcomers 1 and 4 for LOFT Experiment L1-3A.

The RELAP5/MOD1 pump model allows users to provide their own homologous curves. The pump model allows the user the option of accounting for cavitation or two-phase degradation effects on pump response. The user must supply a separate set of homologous, two-phase curves for head and torque which are in the form of difference curves. A default system is available to allow the analyst to use two-phase curves developed from the 1-1/2 loop model Semiscale pump and Westinghouse Canada Limited (WCL) experiments.[4-5] The assumptions inherent in use of the RELAP5/MOD1 two-phase flow pump curves are:

1. The two-phase curves determined empirically for the normal operating region of the pump are also valid as interpolating factors in all other operating regions.

2. The relationship of the two-phase to the single-phase behavior of the Semiscale pump is applicable to large reactor pumps. This assumes an independent pump specific speed for the pump model of two-phase flow.

Differences between the preexperiment predictions and measured data for LOFT Experiment L3-6[4-23] were due to using the LOFT single-phase homologous pump curves and the default two-phase curves in RELAP5. An extensive analysis of the two-phase pump data for Experiment L3-6 was performed and reported in Reference 4-24. The results of the analysis indicated that the two-phase pump curves in RELAP5/MOD1 were incorrect for the LOFT pumps. Figure 4-15 is a normalized plot of head versus void fraction for a LOFT pump, the Semiscale pump, and a scaled Combustion Engineering (CE) pump.

The difference between the Semiscale and LOFT pumps is significant. The minimum normalized head for the LOFT pump is 0.4, while the Semiscale pump develops virtually zero head for void fractions between 0.3 and 0.6.

From the Experiment L3-6 data, a new first quadrant two-phase pump curve was developed for the LOFT pump, and Experiment L3-6 was reanalyzed. Significant improvement in calculating mass inventory can be seen in Figures 4-16 and 4-17 as a result of using more appropriate two-phase pump curves in the postexperiment analysis.[4-25]

4.2 Component Models

The component modeling considered in this section includes modeling of the downcomer, pressurizer, accumulator, and steam generator.

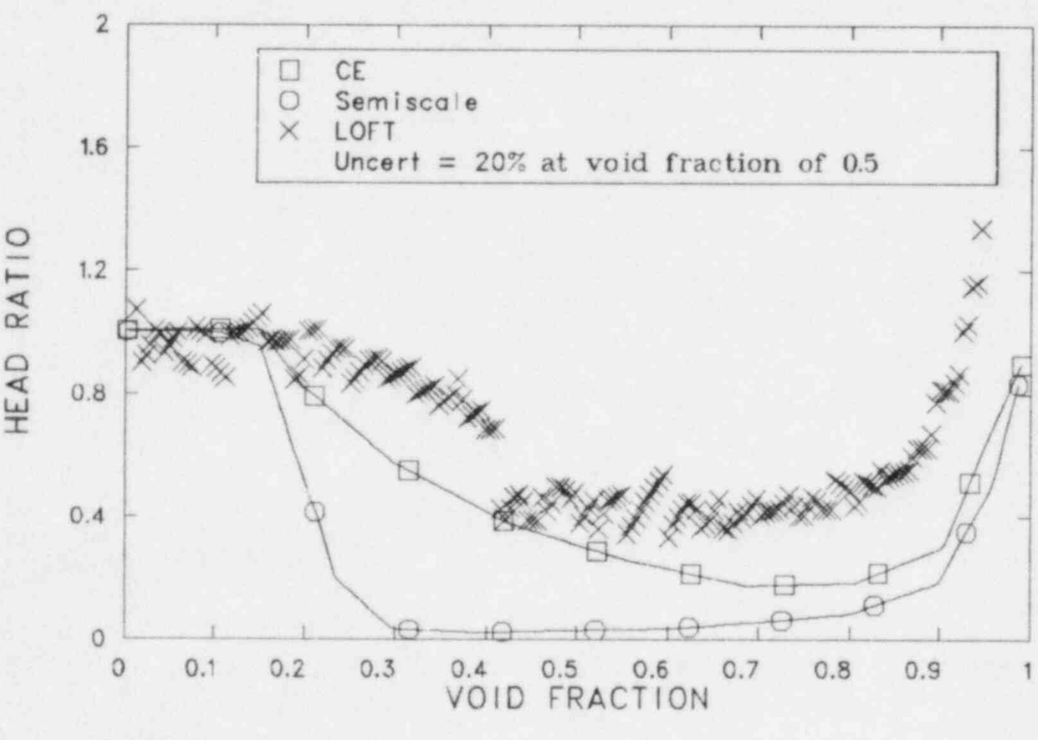


Figure 4-15. Pump head ratio versus void fraction for LOFT, Semiscale, and scaled Combustion Engineering pumps.

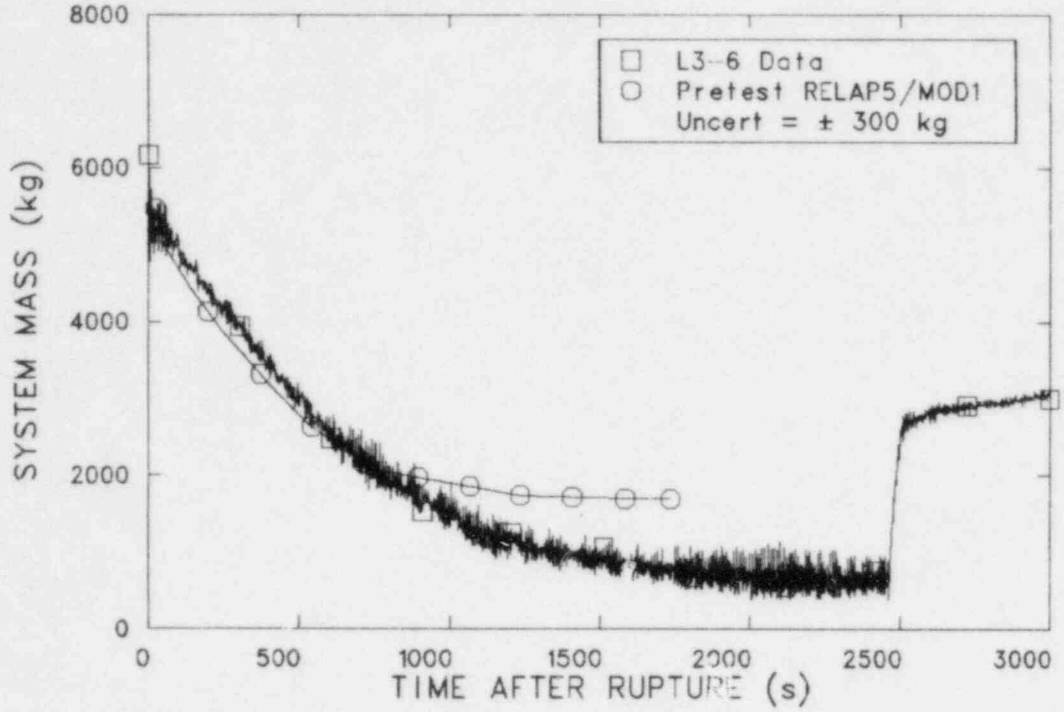


Figure 4-16. Measured and predicted system mass inventory for LOFT Experiment L3-6.

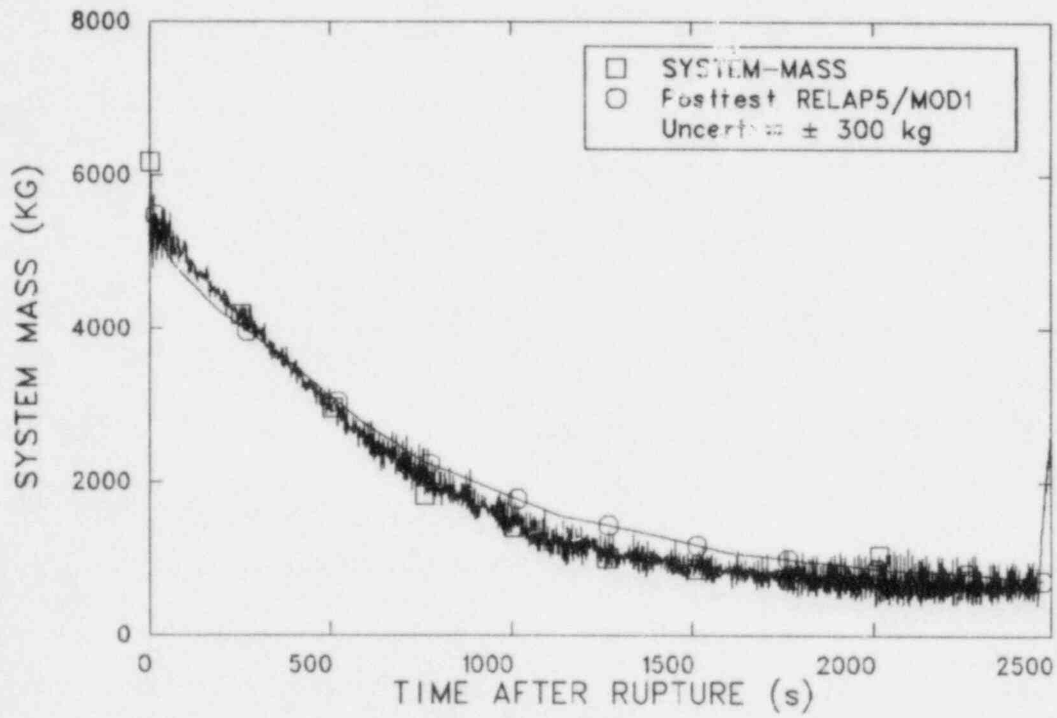


Figure 4-17. Measured and calculated system mass inventory for LOFT Experiment L3-6 (calculation was with improved LOFT two-phase pump curve).

4.2.1 Downcomer Modeling. Modeling the reactor vessel downcomer region has been an area of major interest in LOCA analysis because of the importance of flow in the downcomer during reflood. The conventional nodalization scheme for the upper annulus and downcomer is a series of two or more vertically stacked volumes. The ability, however, of this one-dimensional model to calculate the behavior of the fluid flow in the downcomer, which is basically two or three dimensional, was questioned since azimuthal differences in downcomer behavior had been evident in data from the first LOFT large-break experiment (Experiment L1-1).[4-26]

The azimuthal difference can be seen very clearly in data from LOCE L1-4, a 200% cold leg break with cold leg ECC injection.[4-27] Figures 4-18 and 4-19 show the momentum flux in the reactor vessel downcomer instrument stalks located on opposite sides of the downcomer. Note that there was very little difference in momentum flux in the stalks until ~28 s after rupture. At that time, ECC flow, which was initiated at 23 s after rupture, began to penetrate the intact loop side of the downcomer, causing the positive flow spikes at ME-2ST-1, while negative flow (upflow) dominated at ME-1ST-1 on the broken loop side for another 15 s.

A RELAP4/MOD5 calculation was performed which demonstrated the inability of the code to adequately calculate downcomer performance using the conventional "single-downcomer" nodalization. The failure to adequately model downcomer performance is evident in Figure 4-20, which gives the reactor vessel liquid mass calculated from experimental data and from a single-downcomer RELAP4 calculation. While the experimental data showed significant ECC accumulation in the reactor vessel between 23 and 42 s, the single-downcomer calculation showed no accumulation. In this calculation, the RELAP4 vertical slip option was used to simulate countercurrent flow between the inlet annulus and downcomer. In the case of modeling LOFT Experiment L1-4, blowdown and condensation effects apparently dominated the pressure calculations so that vertical slip did not significantly affect the mass flows.

The "split-downcomer" nodalization was devised as a means of allowing the code to more directly approximate the azimuthal asymmetry observed in downcomer performance. In this nodalization, the upper annulus and downcomer were modeled as two parallel flow paths, as shown in Figure 4-21. Form loss coefficients for the cross-flow junctions were based solely on the physical geometry of the region. The success of this modeling scheme in

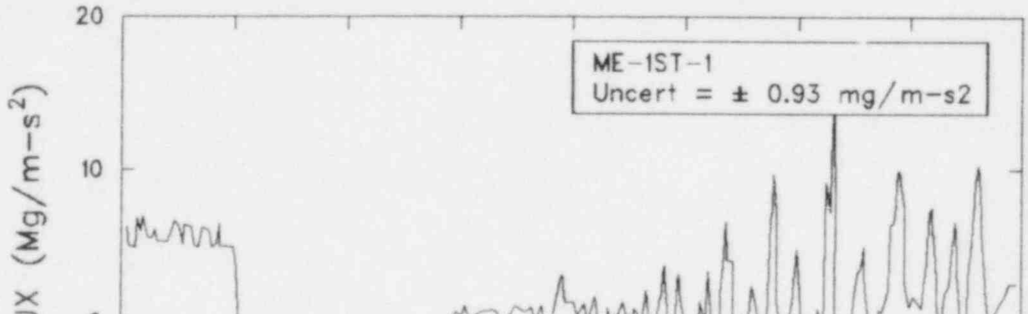


Figure 4-18. Momentum flux in reactor vessel downcomer instrument stalk under broken loop cold leg for LOFT Experiment L1-4.

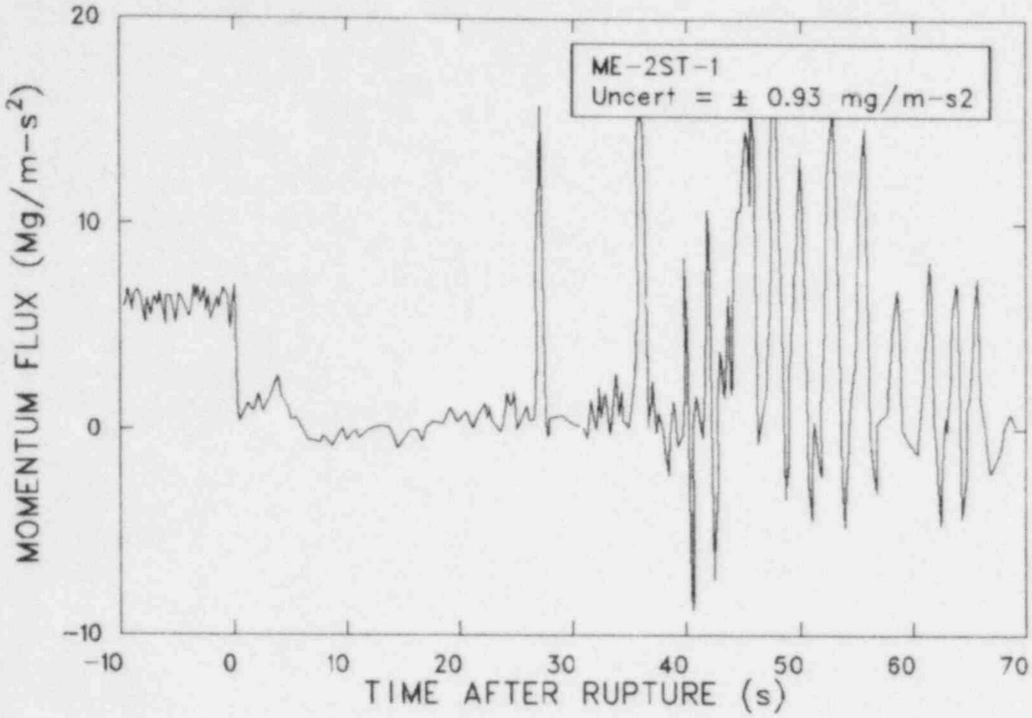


Figure 4-19. Momentum flux in reactor vessel downcomer instrument stalk under intact loop cold leg for LOFT Experiment L1-4.

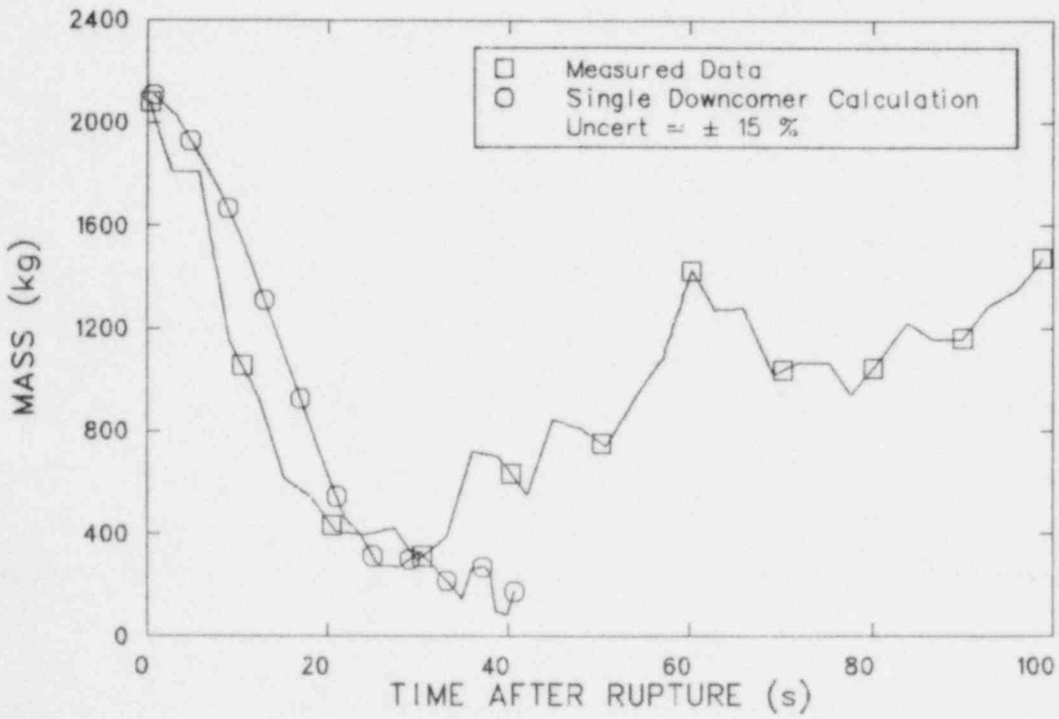
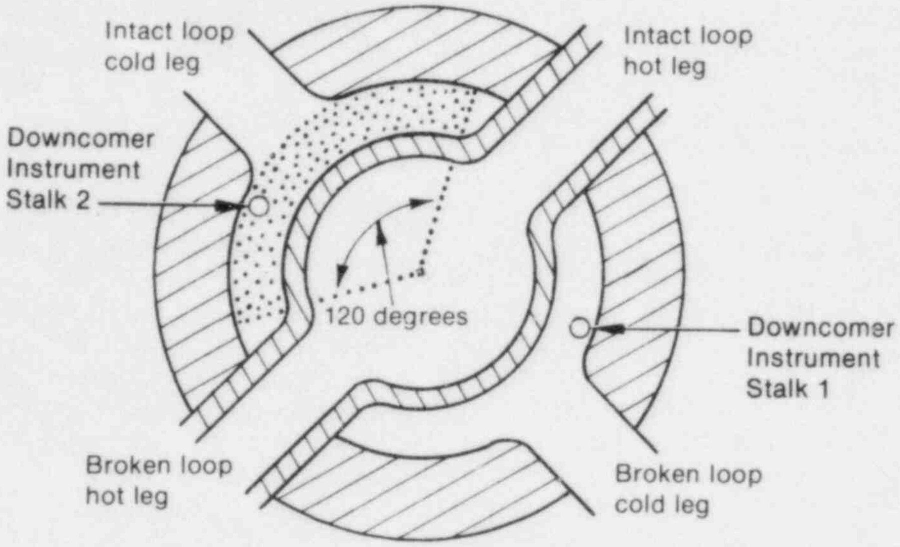
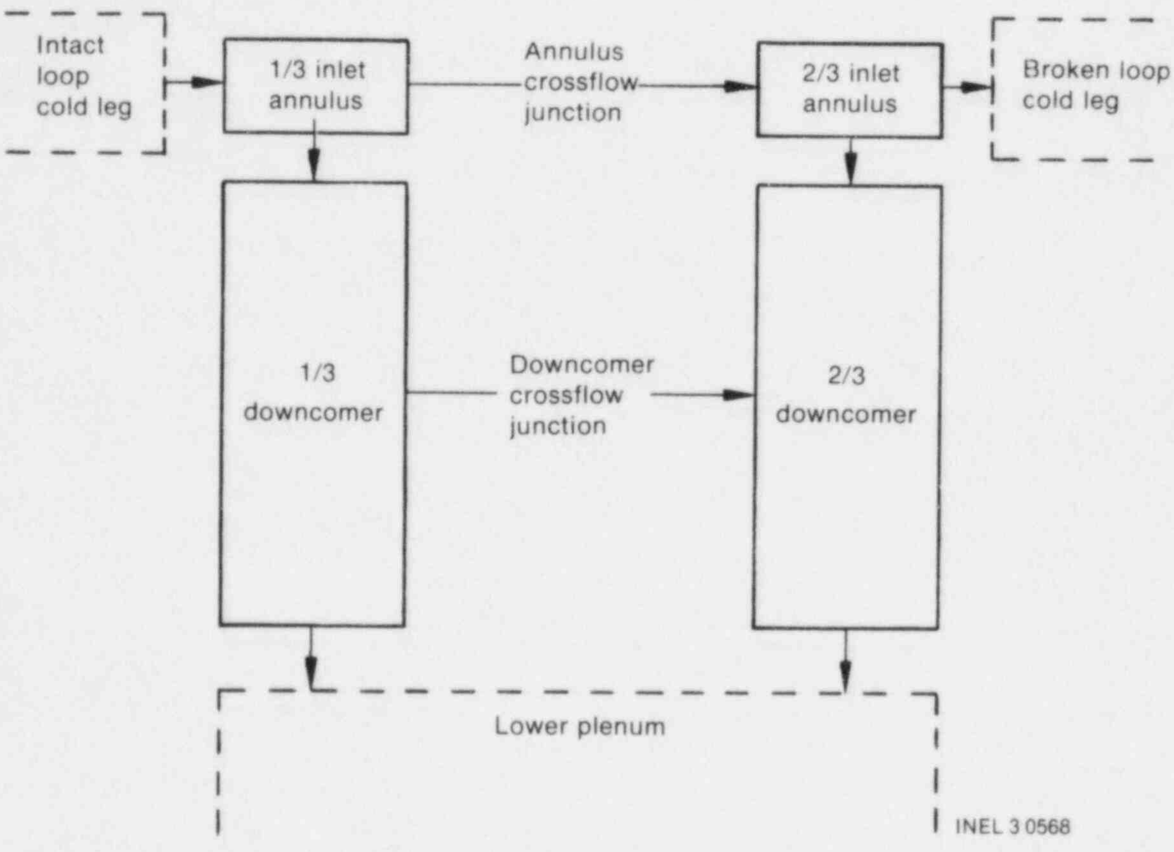


Figure 4-20. Measured and calculated reactor vessel liquid mass for LOFT Experiment L1-4 (calculation was with RELAP4 single-downcomer model).



a. Reactor vessel section at nozzle centerline showing 1/3 section under intact loop cold leg



b. Downcomer nodalization

Figure 4-21. RELAP4 split-downcomer model used for LOFT Experiment L1-4 calculations.

calculating overall downcomer performance can be seen in Figure 4-22, which shows measured data and pre- and postexperiment RELAP4/MOD6 calculations of the reactor vessel liquid mass for Experiment L1-4. The plot shows that the reactor mass inventory was predicted well with very little difference between the pre- and postexperiment calculations.

It is difficult to model the complex multidimensional phenomena that occur in a downcomer with a one-dimensional code. The split-downcomer nodalization is an innovative application of the RELAP4 computer code that produces the best results within the limits of the code.

4.2.2 Pressurizer Modeling. During the course of the LOFT Experimental Program, experimental and analytical results from several experiments indicated that the pressurizer exerts a strong influence on system behavior. During the postexperiment analysis of LOFT Experiment L1-3A,[4-28] it appeared that the pressurizer in the RELAP4/MOD5 preexperiment prediction run tended to empty too fast.[4-29] This was apparent from comparisons between the RELAP4/MOD5 predicted and experimentally measured pressurizer pressure and liquid level. In the preexperiment analyses, the pressurizer and surge line were modeled as two volumes, with a single junction connecting the pressurizer to the surge line, and a single junction connecting the surge line to the intact loop hot leg piping.

Parametric studies showed that modeling the surge line with more than one control volume had only a slight effect on the pressurizer outlet flow rate, and the computer running time increased considerably. The effect of Fanning and single-phase form-loss coefficients on calculated pressurizer discharge flow rates and the effect of the critical flow contraction coefficient were also investigated. The result of these investigations led to a new RELAP4/MOD5 pressurizer model, which had the following changes to the surge line:

1. A small increase in the single-phase form loss in the surge line nozzle

2. An increase in the form loss in the pressurizer surge line was made to account for pipe bends

3. An increase in the form loss was made to account for the difference in Fanning losses between smooth and rough piping

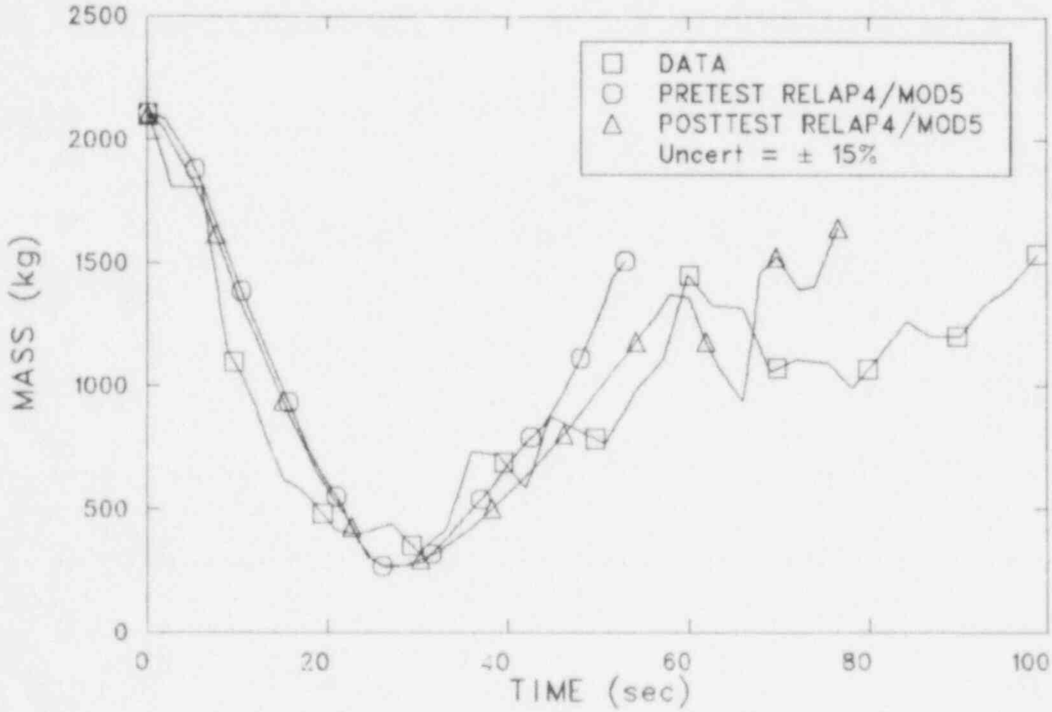


Figure 4-22. Measured, preexperiment prediction, and postexperiment calculation of reactor vessel liquid mass for LOFT Experiment L1-4 (prediction and calculation were with RELAP4 split-downcomer model).

4. Separate two-phase multipliers were applied to the Fanning and form losses discussed above to account for two-phase effects[4-30]

5. A contraction coefficient of 0.75 was applied to the pressurizer surge line outlet junction to account for the effect of the final bend on the critical flow rate

6. A 30% reduction in the bubble velocity in the bubble rise model in the pressurizer was implemented to account for the lower buoyancy effect due to the higher pressures in the pressurizer

7. A slight increase was made in the elevation of the junction between the pressurizer and the pressurizer surge line.

After the new pressurizer model was developed and incorporated into the RELAP4/MOD5 model of the LOFT system, a calculation was made to show the effects of the pressurizer model changes. Figure 4-23 shows a comparison among pre- and postexperiment calculations and measurement of pressure in the intact loop. The differences in the primary coolant system pressures in the RELAP4 runs were an unexpected result. Careful analysis of the RELAP4 outputs revealed that the primary system pressures at the end of subcooled blowdown (~0.2 s) tended to be controlled by the saturation pressure of the control volume in the primary system which has the highest temperature. In the two RELAP4/MOD5 calculations, this proved to be the volume in the intact loop hot leg into which the pressurizer discharged. By reducing the pressurizer discharge flow of high enthalpy fluid, the energy input into the intact loop was decreased and the temperature in the intact loop was lowered. The postexperiment analysis agreed better with the measured data.

A comparison among pre- and postexperiment calculations and measurement of pressurizer liquid level is shown in Figure 4-24. The effect of changing the junction elevation of the pressurizer outlet junction can be seen in the figure, which shows a discontinuity in the pressurizer liquid level versus time curve at ~0.1 s.

A simple model of the pressurizer was found to be adequate for the LOFT large-break experiments (one volume for the pressurizer and one volume for the surge line). For small-break experiments, several volumes are required to model the pressurizer and

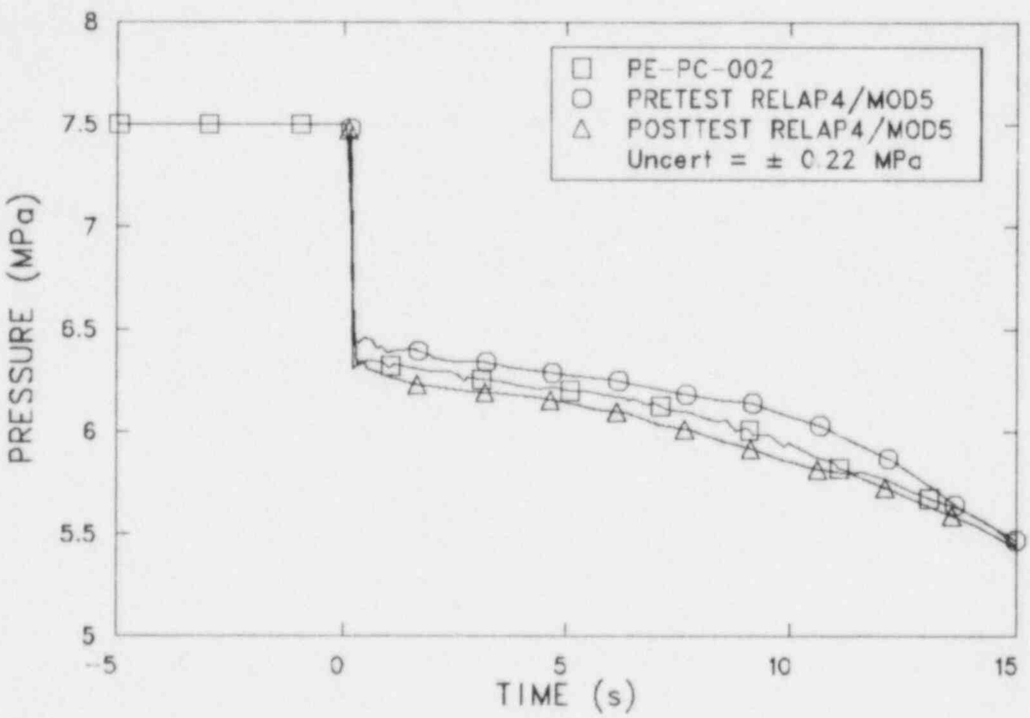


Figure 4-23. Measured, preexperiment prediction, and postexperiment calculation of pressure in intact loop for LOFT Experiment L1-3A.

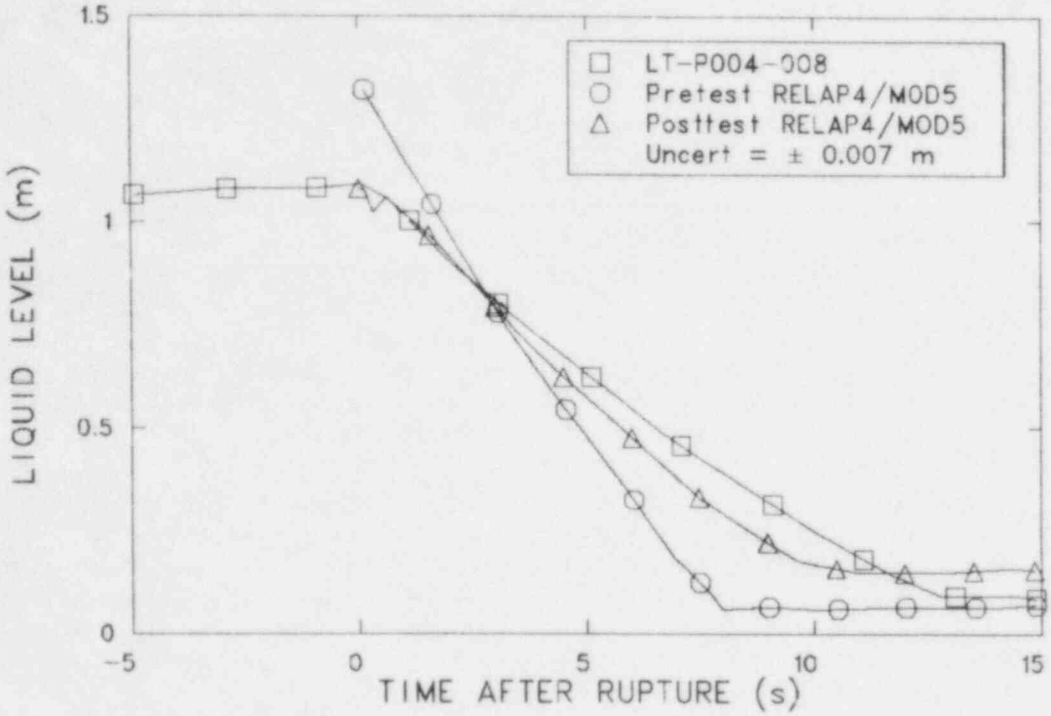


Figure 4-24. Measured, preexperiment prediction, and postexperiment calculation of liquid level in pressurizer for LOFT Experiment L1-3A.

surge line. The preexperiment RELAP5/MOD"0" analysis of LOFT Experiment L3-7 required a pressurizer model with six volumes to model the pressurizer behavior.[4-31] LOFT results have also shown that for operational transients, where the pressurizer is an active system component, the pressurizer must be modeled in even greater detail.

The postexperiment evaluation of Experiment L9-3, an anticipated transient without scram (ATWS) initiated by a loss of feedwater,[4-32,4-33] indicated that nonequilibrium conditions in the pressurizer are important and that the physical models in RELAP5/MOD1 did not adequately calculate the nonequilibrium effects. Two types of nonequilibrium behavior were evident in the pressurizer during Experiment L9-3: thermal nonequilibrium between the vapor and liquid when the vapor was compressed, and hydraulic nonequilibrium (liquid distribution) which occurred when the pressurizer sprays were turned on.

The thermal nonequilibrium behavior at the liquid-vapor interface, when the system pressure increased and the vapor was compressed, was poorly predicted for Experiment L9-3, because the input model had large control volumes in the central portion of the pressurizer. The assumption of complete liquid and vapor mixing in the large volumes containing the interface caused excessive condensation and pressure reduction. This effect can be seen in Figure 4-25, which shows measured and predicted pressure in the primary system for Experiment L9-3.[4-32] Early in the transient (between 10 and 30 s), the predicted pressure did not increase as much as the actual measured pressure increased. Later in the transient (between 50 and 150 s) the predictions were higher than measured because of higher-than-predicted heat transfer to the steam generator. For the Experiment L9-4 prediction,[4-33] the pressurizer was nodalized more finely (15 control volumes compared with 8 control volumes for the Experiment L9-3 prediction) to minimize the condensation effects.

Some improvement in predicting pressure response can be seen in Figure 4-26, which shows measured and predicted primary pressure for Experiment L9-4.[4-34] The improvement is particularly evident in the first 20 s. Between 50 and 450 s, the predicted and measured pressure cycled between the same limits as the safety relief valve (SRV) cycled opened and closed. The difference between predicted and measured pressure after 450 s was due to primary-to-secondary heat transfer.

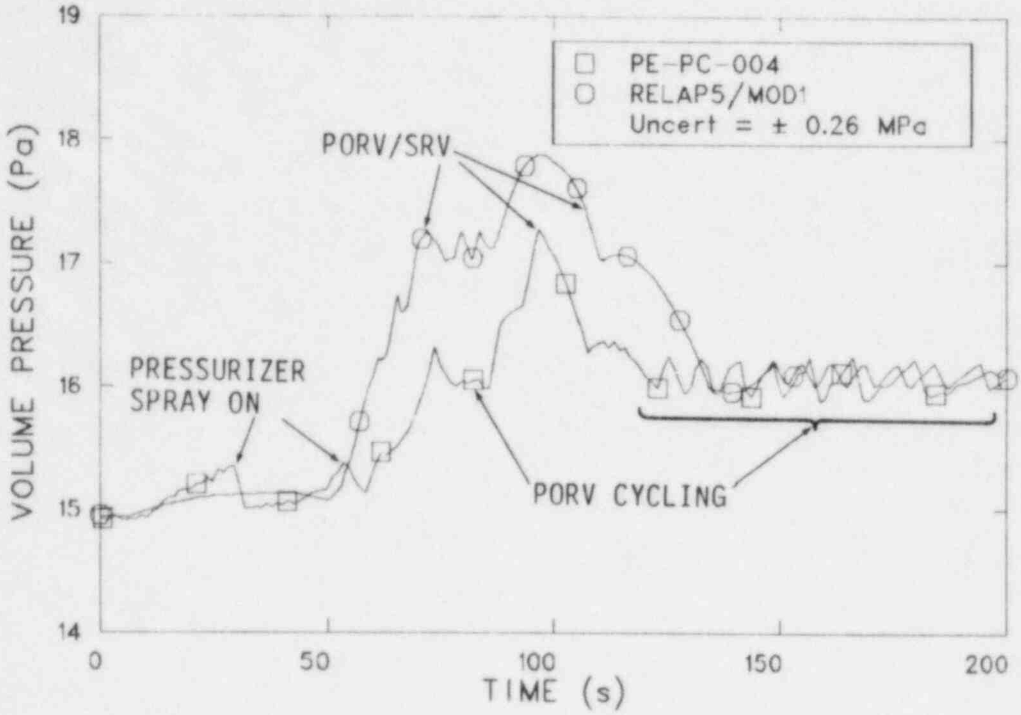


Figure 4-25. Measured and predicted pressure in primary system for LOFT Experiment L9-3.

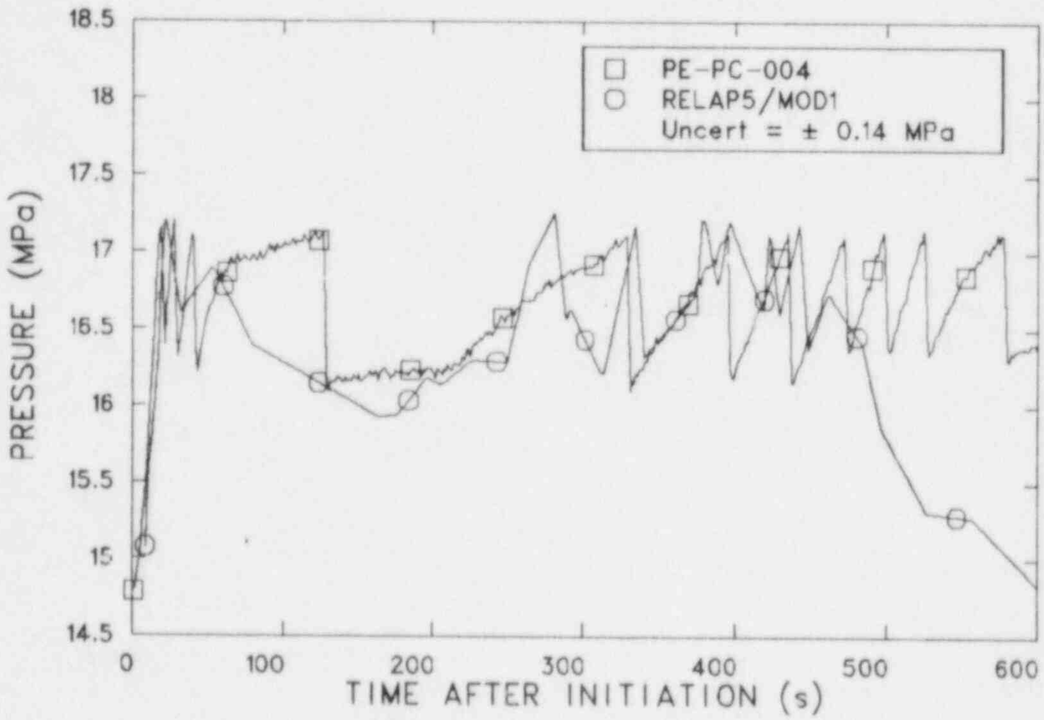


Figure 4-26. Measured and predicted pressure in primary system for first 600 s in LOFT Experiment L9-4.

The operation of the pressurizer spray was not calculated well for Experiment L9-3 (hydraulic nonequilibrium). Liquid was calculated to remain in the top pressurizer control volume while the lower volumes contained vapor. This holdup caused an error in the calculated PORV/SRV flow. This can be clearly seen in Figure 4-27, which shows measured and predicted PORV and SRV mass flow during Experiment L9-3.

The problem of hydraulic nonequilibrium is more difficult to solve than thermal nonequilibrium. It cannot be overcome by finer nodalization. In fact, finer nodalization may make it worse. The stratification models used for the pressurizer require improvement.

4.2.3 Accumulator Modeling. With RELAP4, the accumulator was modeled as a volume filled with water and nitrogen. A value of k, the polytropic gas constant, had to be input into the code. Calculations showed that the flow rate from the accumulator was sensitive to k. RELAP4 calculations for the LOFT system showed a reduction in accumulator flow rate of ∿15% when k is increased from 1.0 (isothermal) to 1.4 (isentropic). The gas constant is dependent on the volumetric expansion rate and the surface-to-volume ratio of the gas. For LOFT, k was found to be between 1.05 and 1.3 for the range of experiments performed. Typically, a value of 1.05 is used for small-break experiments, and 1.3 is used for large-break experiments. These represent good average values for k during experiments. Slightly better results would be obtained if the value of k was varied as a function of accumulator flow.

The LOFT and Semiscale experimental results indicate that the thermodynamic behavior of PWR accumulators will be nearly isentropic for all size LOCAs (that is, $k \simeq 1.4$). Because of the larger size of a commercial PWR accumulator, the surface-to-volume ratio is smaller than it is in LOFT and Semiscale. Therefore, heat transfer effects are also much smaller. See Reference 4-35 for details of this study.

To overcome the difficulties associated with the polytropic gas constant in RELAP4, an accumulator model has been installed in RELAP5/MOD1.[4-5] This model features mechanistic relationships for heat transfer from the tank wall and water surface, condensation in the vapor dome, and vaporization from the water surface to the dome. The accumulator model consists of a hydrodynamic model and a heat transfer model. The latter includes the vaporization and condensation effects.

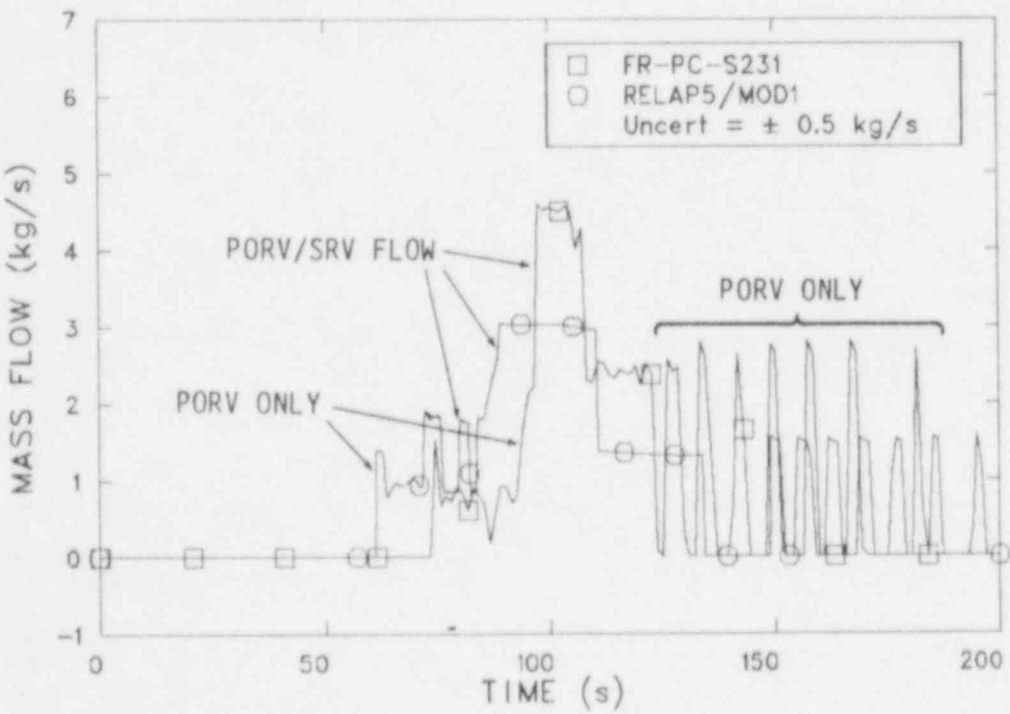


Figure 4-27. Measured and predicted mass flow through PORV and combined PORV and SRV for LOFT Experiment L9-3.

The accumulator is modeled as a lumped-parameter component. The basic features of the model are:

1. Heat transfer from the accumulator walls is modeled using natural convection correlations.

2. The nitrogen is modeled as an ideal gas with constant specific heat. The energy released as a result of vapor condensation is transferred to the nitrogen.

3. Because of the high heat capacity and large mass of water below the liquid-gas interface, the water is assumed to remain at its initial temperature.

The accumulator model has been tested in RELAP5 and gives good agreement with experimental data from LOFT small-break Experiment L3-1[4-36] and a Westinghouse UHI accumulator performance test.[4-37] The comparison to the experimental data in both experiments is of the form of pressure versus vapor volume. The LOFT data used for comparison are liquid level and pressure history; the Westinghouse data used for comparison are integrated mass flow and pressure history.

LOFT Experiment L3-1 was a small-break experiment that provided a slow transient for the accumulator model checkout. Figure 4-28 shows a comparison among measured and calculated pressure versus vapor volume for Experiment L3-1. The plot includes a calculated isothermal curve (uppermost curve) and a calculated isentropic curve (lowermost curve). In the early part of the blowdown (high pressure and low vapor volume) the expansion of the nitrogen gas was essentially isentropic. As the gas expanded in the accumulator dome, its temperature decreased, causing a temperature difference between the gas and its surroundings. The surroundings then lost heat to the gas, causing the gas expansion to behave in a more isothermal manner.

The Westinghouse UHI accumulator was the second experiment simulated. This accumulator was tested (in place) by allowing it to discharge into the atmosphere (pressure vessel with cover off). This performance test resembled a large-break-type experiment and provided a fast accumulator transient for verifying the accumulator model. This accumulator was simulated by using the measured data and performance information contained in Reference 4-37 as input. Figure 4-29 shows the results of the comparison among the calculated results and the experimental data. The calculated

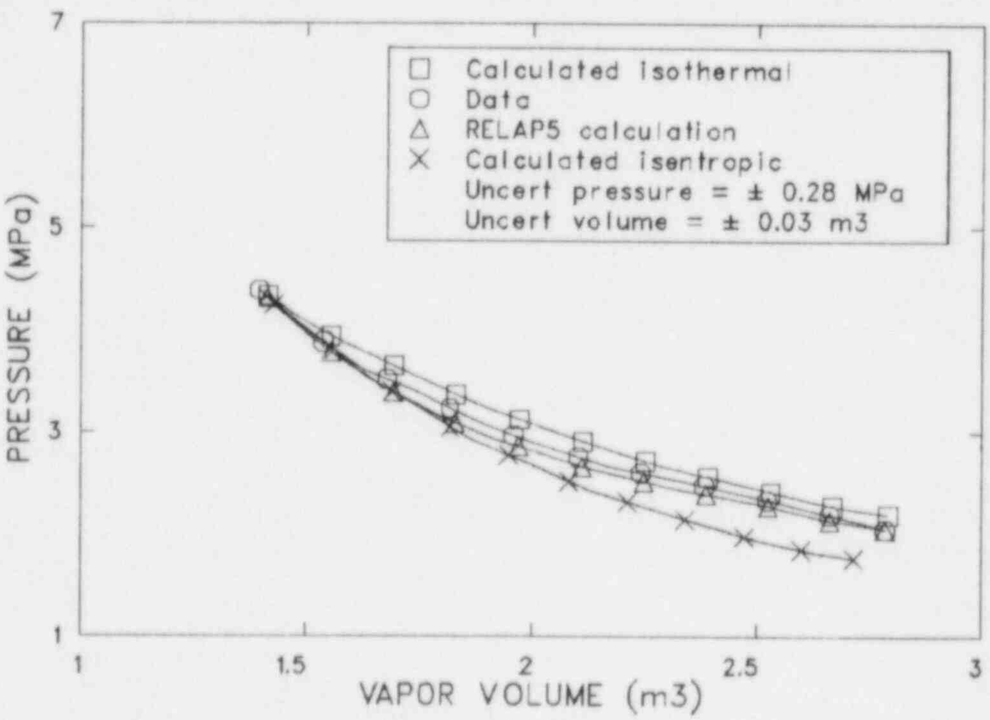


Figure 4-28. Measured and calculated pressure versus vapor volume in the accumulator for LOFT Experiment L3-1.

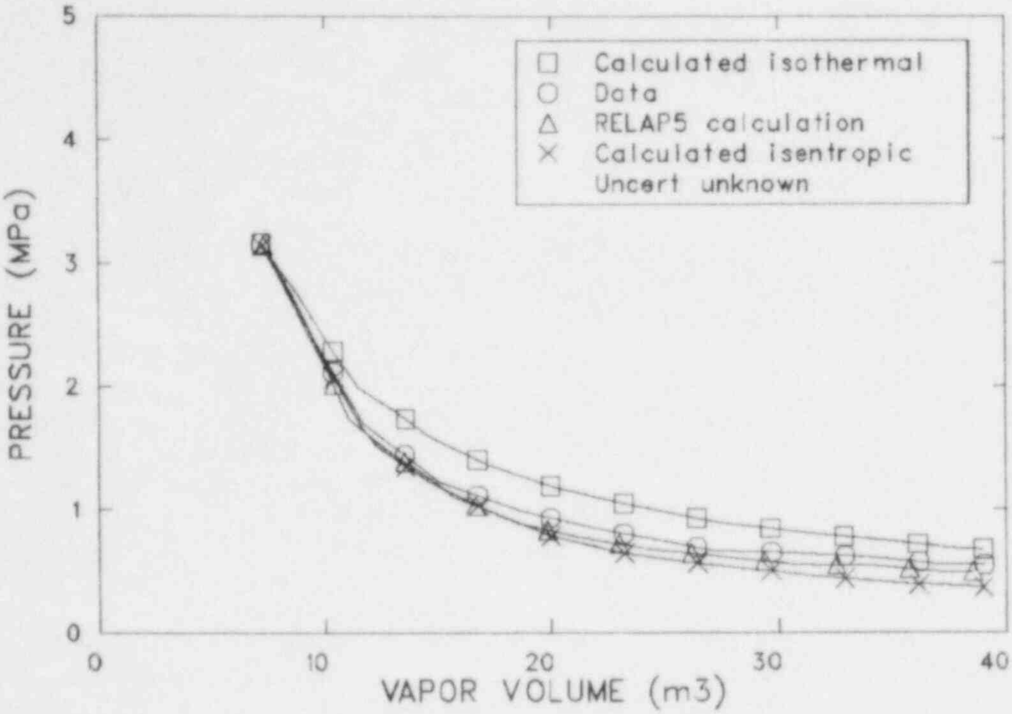


Figure 4-29. Measured and calculated pressure versus vapor volume in the accumulator for a Westinghouse accumulator performance test.

isothermal (uppermost curve) and isentropic (lowermost curve) curves are included for reference. The results calculated using RELAP5 are in good agreement with the measured data.

4.2.4 Steam Generator Modeling. The importance of steam generator modeling has become increasingly apparent as the LOFT Experimental Program has progressed, and the role of the steam generator has become more important in the transient. The steam generator models used for the large-break experiments were relatively simple. The RELAP4 model for Experiment L2-3 consisted of five volumes for the primary side and one volume for the secondary side.[4-38] The simple model was adequate for the large breaks where the steam generator was shutdown and played a relatively passive role in the experiment.

During the small-break experiments, the steam generator played a more active role. Consequently, the steam generator models became more detailed. The RELAP5 steam generator model for Experiment L9-4 consisted of 10 volumes to represent the primary side and 13 volumes to represent the secondary side. Heat structures were included to model the U-tubes and structural elements of the steam generator.[4-33] In addition to the more detailed model, a condensation heat transfer correlation was added to the RELAP5/MOD1 code to improve the calculation of steam generator secondary pressure during the early portion of a transient.[4-25]

Despite more detailed models, and an additional heat transfer correlation, the results of experiments still indicate that primary-to-secondary heat transfer in the steam generator is not computed correctly, particularly during experiments involving loss of feedwater. Figure 4-30 shows predicted and measured primary system pressure during LOFT Experiment L9-4 (a loss-of-offsite-power experiment that resulted in a loss of feedwater).[4-34] The generally poor prediction beyond 450 s was due to poor prediction of primary-to-secondary heat transfer.[4-39]

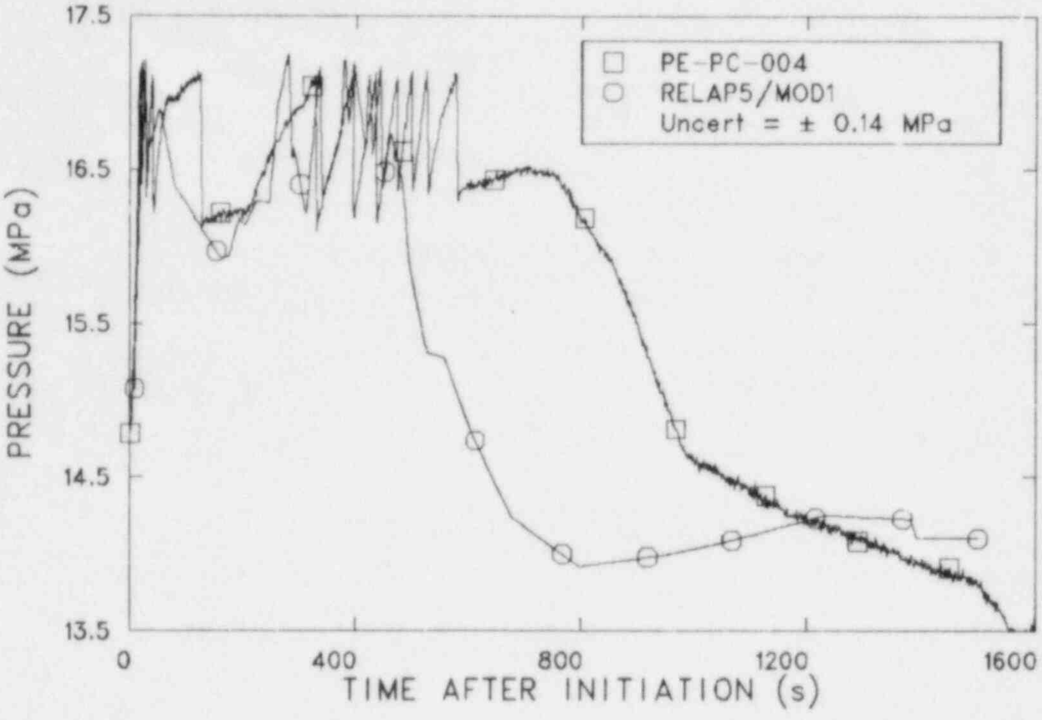


Figure 4-30. Measured and predicted pressure in primary system for LOFT Experiment L9-4.

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5. IMPACT ON 10 CFR 50.46 AND APPENDIX K

One of the main missions of the LOFT Program was to evaluate the performance of ECCSs to aid in determining the margin of safety in the acceptance criteria and the conservativeness in the Appendix K evaluation criteria for ECCS models. This section presents the results from the LOFT large-, intermediate-, and small-break experiment series as they relate to 10 CFR 50, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors,"[5-1] and 10 CFR 50, Appendix K, "ECCS Evaluation Models."[5-2]

5.1 Large-Break Experiments

In order for a nuclear plant to be licensed to operate by the USNRC, an analysis of a LOCA must be performed using an evaluation model (EM) that contains certain required and acceptable features described in Appendix K to 10 CFR 50. The analysis must demonstrate that the fuel cladding will not exceed 1477 K.

The LOFT large-break experiments (nonnuclear Experiment Series L1 and nuclear Experiment Series L2) were designed to provide thermal-hydraulic and fuel behavior data during double-ended (200%) cold leg break experiments at various reactor power and ECCS conditions. The nonnuclear large-break experiments (Series L1) are summarized in Table 2-6. The nuclear large-break experiments (Series L2) are summarized in Table 2-7, which summarizes all the nuclear experiments.

Preexperiment calculations were made for LOFT Experiment L2-3[5-3] conducted from initial conditions comparable to a commercial PWR at 100% power. Figure 5-1 shows that results obtained using the EM are conservative. The maximum cladding temperature was 900 K, while the EM prediction was 1465 K. The results showed that even the best estimate model was conservative. The results of the large-break experiments demonstrated the conservativeness of the following features of the ECCS evaluation models:

1. Discharge model
2. End of blowdown

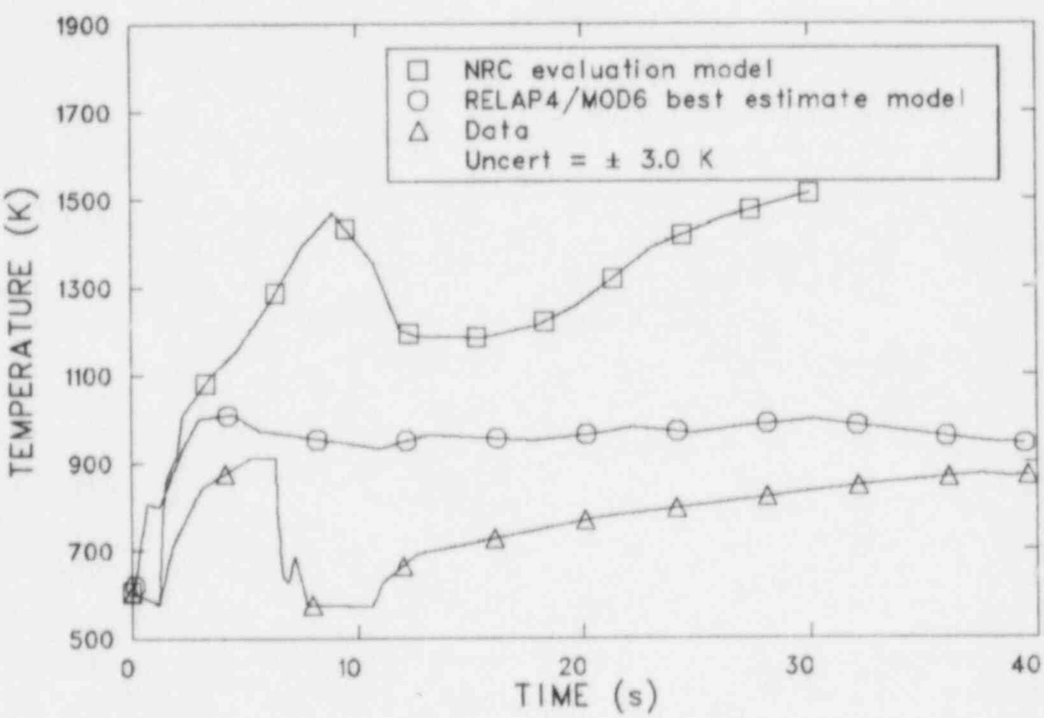

Figure 5-1. Measured and predicted maximum fuel cladding temperature for LOFT Experiment L2-3 (predictions were with RELAP4/MOD6 best estimate model and an NRC evaluation model).

3. CHF heat transfer

4. Post-CHF heat transfer.

5.1.1 Discharge Model. Appendix K states that, "for all times after the discharging fluid has been calculated to be two-phase, the discharge rate shall be calculated by use of the Moody model.[5-4] The calculations shall be conducted with at least three values of a discharge coefficient between the values of 0.6 and 1.0."

Since Appendix K was first issued, a large amount of experimental data has been generated in LOFT and compared with various discharge models to determine the most accurate discharge model. Calculations for the first large-break experiment in LOFT, nonnuclear Experiment L1-1, used the Henry-Fauske model for the subcooled water blowdown and the Moody model with a break flow multiplier of 0.6 for the saturated portion of the blowdown.[5-5] The results were considered satisfactory since the overall characteristics of the blowdown were predicted reasonably well and the uncertainty of the mass flow measurement was relatively large (up to ±25% according to a study reported in Reference 5-6).

LOFT Experiment L2-2, the first large-break nuclear experiment, demonstrated the importance of accurately calculating break flow.[5-7] The core quench that occurred during the first 6 s of the blowdown was a result of a flow reversal through the core. Predicting the flow reversal required more accurate prediction of break flow, particularly during the transition from subcooled to saturated conditions. The discharge model that produced the best agreement with LOFT data was the break flow multiplier of 0.848 on the Henry-Fauske (subcooled) and the homogeneous equilibrium (saturated) critical flow models, with a transition from the Henry-Fauske model to the homogeneous equilibrium model between 0 and 0.25% quality. Figure 5-2 illustrates the excellent correlation obtained with the Henry-Fauske and the HEM models with the 0.25% transition quality.

Using the Moody break flow model over a range of break sizes with a range of break flow multipliers gives some indication of plant behavior. Using the Henry-Fauske/HEM combination over a range of break sizes with a range of break flow multipliers would provide a more realistic indication of plant behavior, and would allow somewhat better judgments to be made regarding plant licensing.

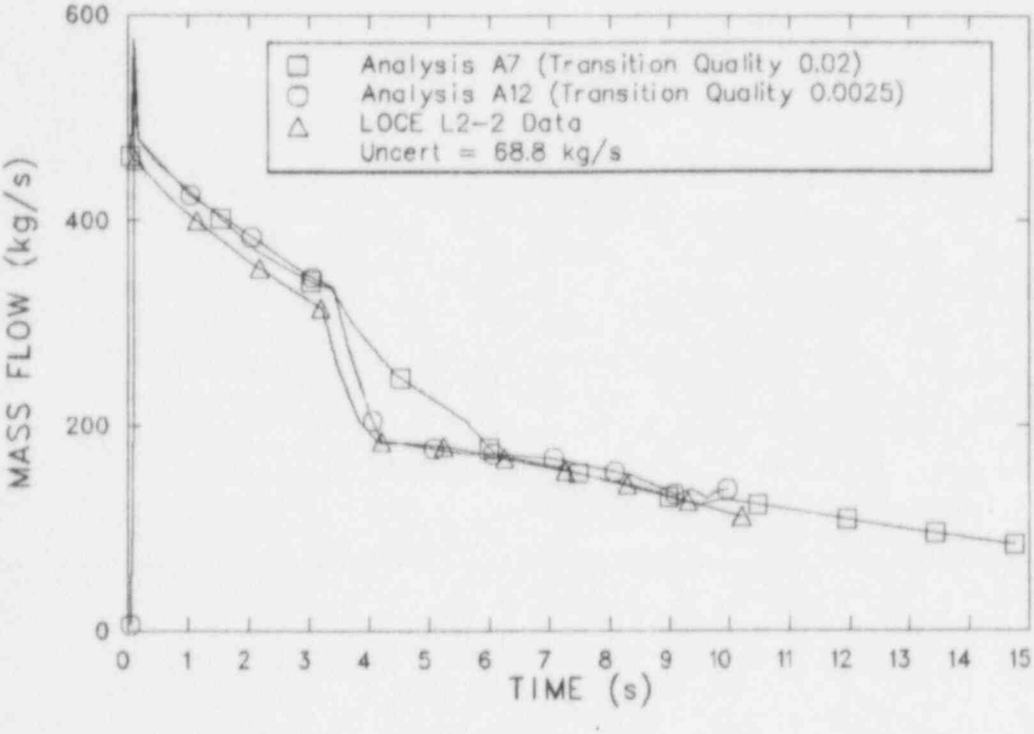

Figure 5-2. Measured and calculated mass flow rate in broken loop cold leg for LOFT Experiment L2-2 (calculations showed sensitivity of critical flow transition quality).

A more detailed discussion of the discharge model is presented in Section 4.1.1.

5.1.2 End of Blowdown. Appendix K, Section I.C.1.c states that, "for postulated breaks, all ECC water injected into the inlet lines or the reactor vessel during the bypass period shall be subtracted from the reactor vessel calculated inventory. The end of the bypassing period shall be defined by calculation of when downward flow is predicted to occur in the downcomer, or prediction of a threshold for droplet entrainment in the upward velocity or a suitable combination of analysis and experimental data."

LOFT Experiments L1-3A and L1-4, nonnuclear large-break experiments, provided specific data with which to evaluate the conservativeness of the Appendix K end-of-blowdown criteria. For Experiment L1-4,[5-8] ECC water was injected into the intact loop cold leg. The lower plenum refill rate for this experiment was compared with that for Experiment L1-3A,[5-9] which had the ECC water injected directly into the lower plenum. These data, shown in Figure 5-3, determined the amount of ECC water bypass, which was a primary objective for Experiment L1-4.[5-10] Figure 5-3, which shows liquid fraction in the reactor vessel for Experiments L1-3A and L1-4, indicates that ~45% of the ECC water injected between 22 and 65 s after rupture during L1-4 was either bypassed or stored in the intact loop piping. The figure shows that ECC water continued to drain into the reactor vessel from the intact loop piping after the ECC accumulator was empty. By 100 s, over 80% of the ECC water had reached the lower plenum.

Using the Appendix K evaluation model criteria, all of the ECC water injected before the end of blowdown (~50 s) would be subtracted from the reactor vessel inventory. This would amount to ~1100 kg, or 50% of the vessel volume (that is, at ~50 s after blowdown initiation, the vessel liquid volume would be considered to be totally empty).

The modeling of the reactor vessel downcomer region had been an area of major interest in LOCA analysis during the nonnuclear experiment series (Series L1). The conventional nodalization scheme for the upper annulus and downcomer was a series of two or more vertically stacked volumes. The ability of this one-dimensional model, however, to calculate the behavior of the fluid flow in the downcomer, which is basically two or three dimensional, was seriously questioned early in the pre-Experiment L1-4 analysis effort, and led to development of the "split-downcomer" model which is discussed in more detail in Section 4.2.1.

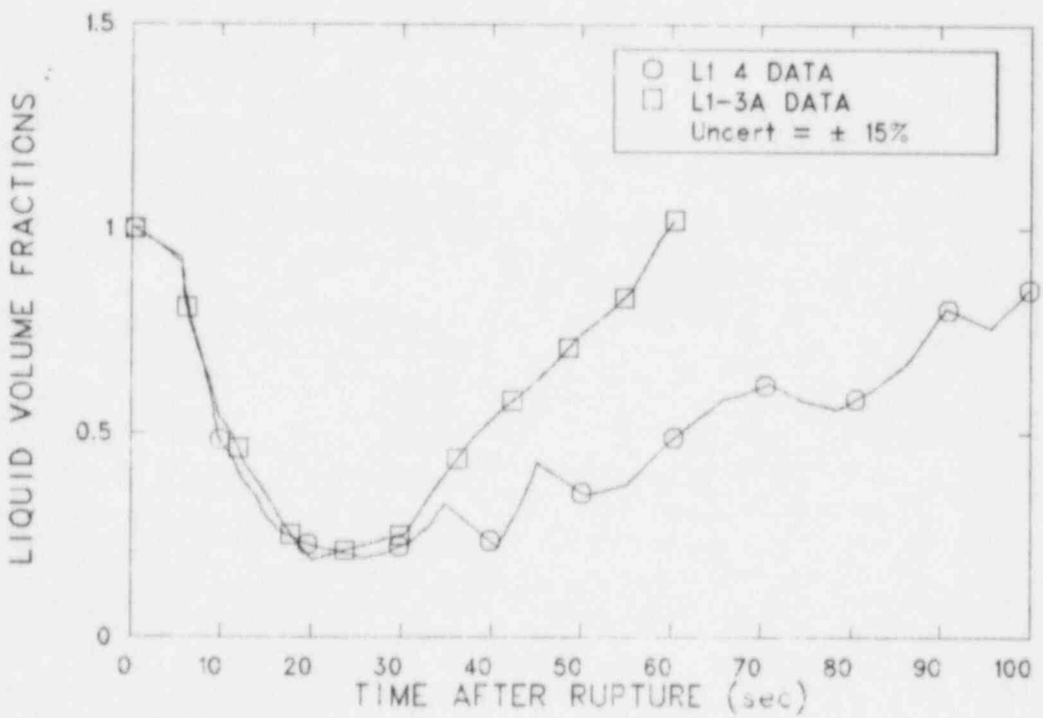

Figure 5-3. Liquid volume fraction in reactor vessel for LOFT Experiment L1-3A and L1-4.

The LOFT experimental data and the innovative modeling technique provided improved understanding of the complex phenomena that occur in the downcomer during ECC injection. The results (a) demonstrate that the Appendix K end-of-blowdown criteria are conservative and (b) provide a basis for assessing the conservativeness of the criteria or for changing it.

5.1.3 CHF Heat Transfer. Appendix K states that, "correlations developed from appropriate steady state and transient state experimental data are acceptable for use in predicting CHF during LOCA transients." The rule goes on to say that, "after CHF is predicted at an axial fuel rod location during blowdown, the calculation shall not use nucleate boiling heat transfer correlations at the CHF location subsequently during the blowdown, even if the calculated local fluid and surface conditions would apparently justify the reestablishment of nucleate boiling. Return to nucleate boiling (RNB) shall be permitted in the calculation, however, when justified by the calculated local fluid and surface conditions during the reflood portion of the LOCA."

The conservativeness of not permitting the use of nucleate boiling heat transfer correlations in the calculation from CHF until the reflood phase was demonstrated very clearly by LOFT Experiments L2-2, L2-3, and L2-5, nuclear large-break experiments (see References 5-11 through 5-23). Experiments L2-2 and L2-3 were performed with the primary coolant pumps running for the entire experiment. During both experiments, the core was quenched during the blowdown phase. During Experiment L2-5, the pumps were tripped at 0.94 s and atypical primary coolant pump coastdown was incorporated to create core flow stagnation conditions to prevent the early core quench.

The analytical predictions for Experiment L2-2[5-12] (conducted from an initial core power level of 25 MW) indicated that the maximum fuel rod cladding temperature would reach 900 to 1000 K and that the nuclear core would be cooled by ECC water 70 to 90 s after blowdown initiation. Instead, the maximum cladding temperature was limited to 789 K by an unpredicted, core-wide rewet that started at 5.5 s and ended at 8 s. The rewet started at the bottom of the core and progressed upward to the top of the core. The core remained rewet until 12 s. Subsequent fuel rod cladding temperatures did not exceed 680 K, and the cladding was fully cooled by ECC water 45 s after experiment initiation. Fuel rod temperature behavior is shown in Figure 5-4, which shows predicted and measured fuel cladding temperatures for the center fuel module.

The early rewet was caused by the resumption of positive core flow (that is, flow from bottom-to-top through the core) and the subsequent introduction of high-density fluid into the core from the intact loop cold leg. Flow behavior in the core is illustrated in Figure 5-5. Saturation of the upper and lower plenum fluid decoupled the core inlet and outlet flows from the broken loop hot and cold leg flows, respectively. The coolant pumps in the intact loop then were free to reestablish positive core flow by 2.5 s. Saturated choking occurred in the broken loop cold leg starting at 3.4 s, 1.7 s prior to the inception of saturated fluid conditions in the intact loop cold leg. This condition caused the fluid inventory in the reactor vessel to increase for almost 2 s, starting at 4.0 s. The added, high-density fluid in the reactor vessel flowed from the lower plenum through the core, rewetting the fuel rod surfaces, starting at 5.5 s.

The measured cladding temperature response of Experiment L2-3 (conducted from an initial core power level of 36.7 MW) was similar to that observed in Experiment L2-2, as shown in Figure 5-6. The observed differences in fuel cladding temperature response are attributable to the difference in core power for the two experiments and are: (a) the peak fuel cladding temperature was higher for Experiment L2-3 than for Experiment L2-2, 914 and 789 K, respectively, and (b) the pattern of the final rewet, or reflood, in Experiment L2-3 was bottom, top, then middle; whereas, the rewet pattern for Experiment L2-2 was bottom, middle, then top.

The revised predictions for Experiment L2-3,[5-16] based on a computer model developed using the results of Experiment L2-2,[5-15] were very good for system hydraulics, but did not predict the rewet that occurred at 8 s after blowdown initiation. A post-experiment analysis reported in Reference 5-19 was performed to determine the reason for the disparity between the predicted and measured fuel rod cladding temperature responses in the LOFT core. The analysis indicated that the post-CHF heat transfer calculations were incorrect. Post-CHF heat transfer modeling is discussed in more detail in Section 4.1.2.

Experiment L2-5 combined a 200% cold leg break experiment with a loss of site power.[5-23] The

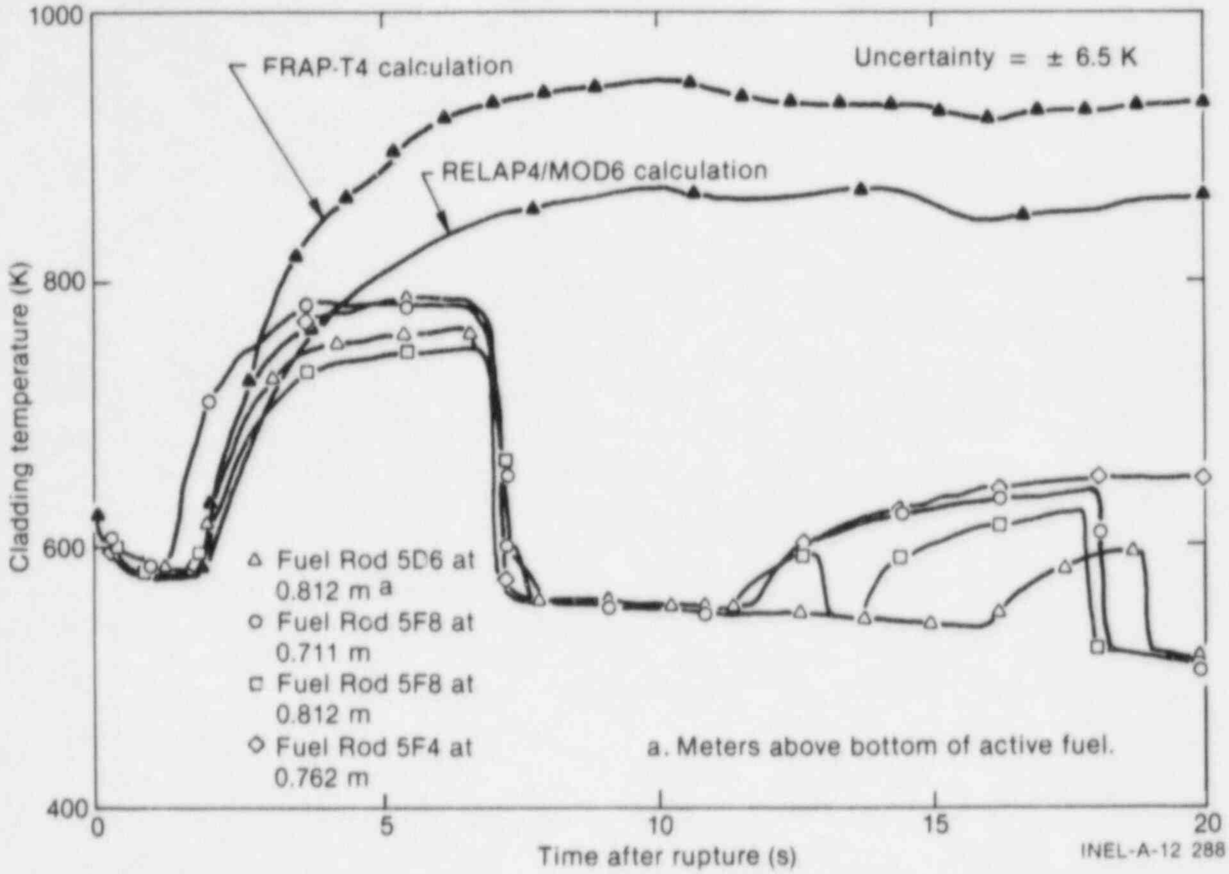

Figure 5-4. Measured and predicted fuel cladding temperature near midplane of center fule module for LOFT Experiment L2-2.

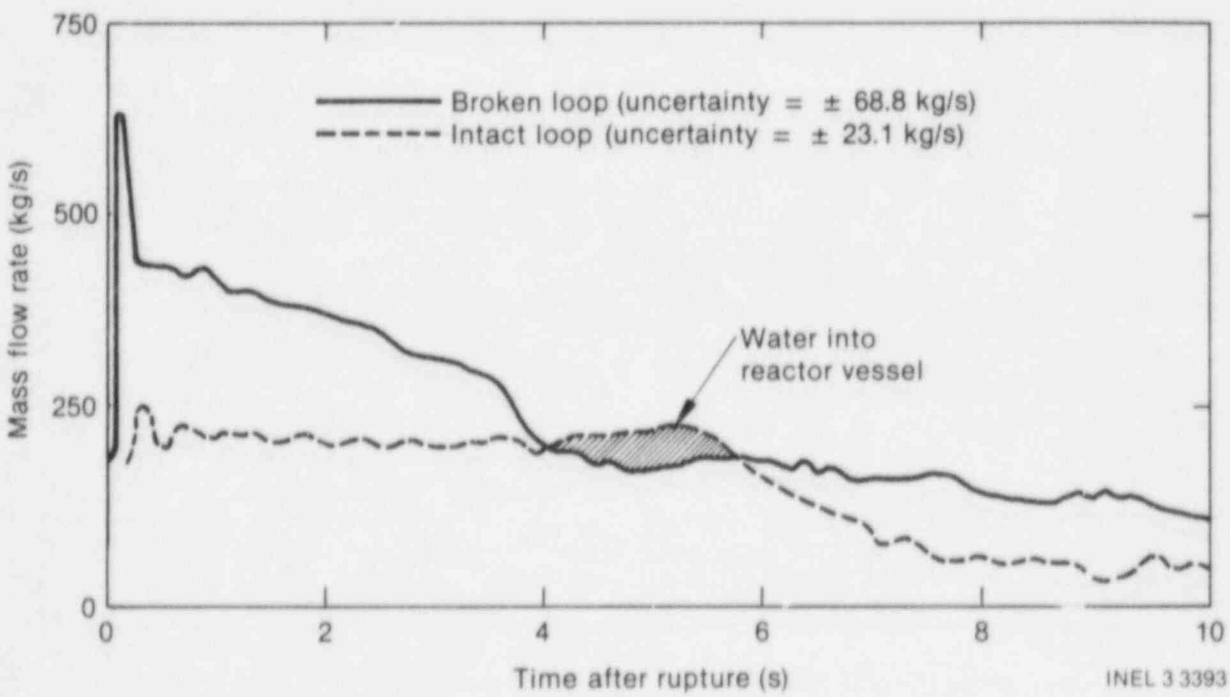

Figure 5-5. Mass flow rate in cold legs of intact and broken loops for LOFT Experiment L2-2.

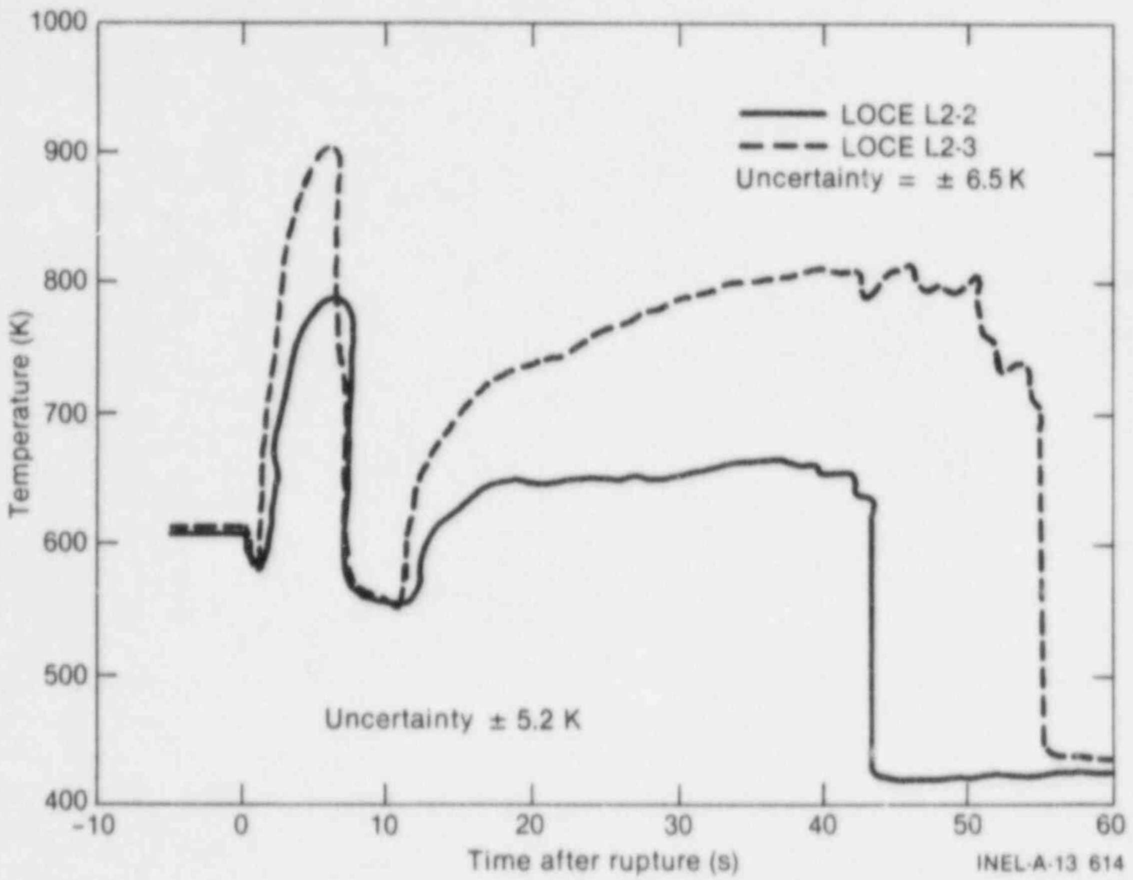

Figure 5-6. Maximum fuel cladding temperature for LOFT Experiments L2-2 and L2-3.

experiment was initiated from a core power level of 36 MW. Following the reactor scram, the operators tripped the primary coolant pumps at 0.94 ± 0.01 s. The pumps were disconnected from their flywheels to produce an atypical pump coastdown which would create core flow stagnation conditions.

Cladding temperatures in the lower half of the central fuel assembly 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 5-7. The maximum measured cladding temperature of 1077 K occurred during this time. At ~30 s, a gradual cooling trend initiated as ECC water filled the lower plenum. The gradual cooling trend continued until all fuel rods were quenched by 65 s. As shown in Figure 5-7, 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.

The thermal response measured in the upper half of the central fuel assembly was markedly different from that measured in the lower half. As shown in Figure 5-8, the cladding temperatures were similar to those in the lower half (Figure 5-7) for nearly 15 s. Then there was a top-down quench lasting up to 5 s which was followed by a second cladding temperature excursion with a generally lower peak value. Final quench in the upper half occurred during core reflood, as in the lower half. The top-down quench in Experiment L2-5 was similar to the top-down quench which occurred somewhat later in Experiment L2-3, as shown in Figure 5-8.

The main conclusion of the LOFT large-break experiments relative to RNB is that RNB will occur very early in the blowdown, and can only be prevented by a total and prolonged stagnant flow condition in the core. Following Experiments L2-2 and L2-3, a detailed study was performed using the RELAP5 computer code to determine the primary coolant pump effects on core thermal response during large-break LOCA transients in the Zion nuclear power plant.[5-24] The study concluded that the only

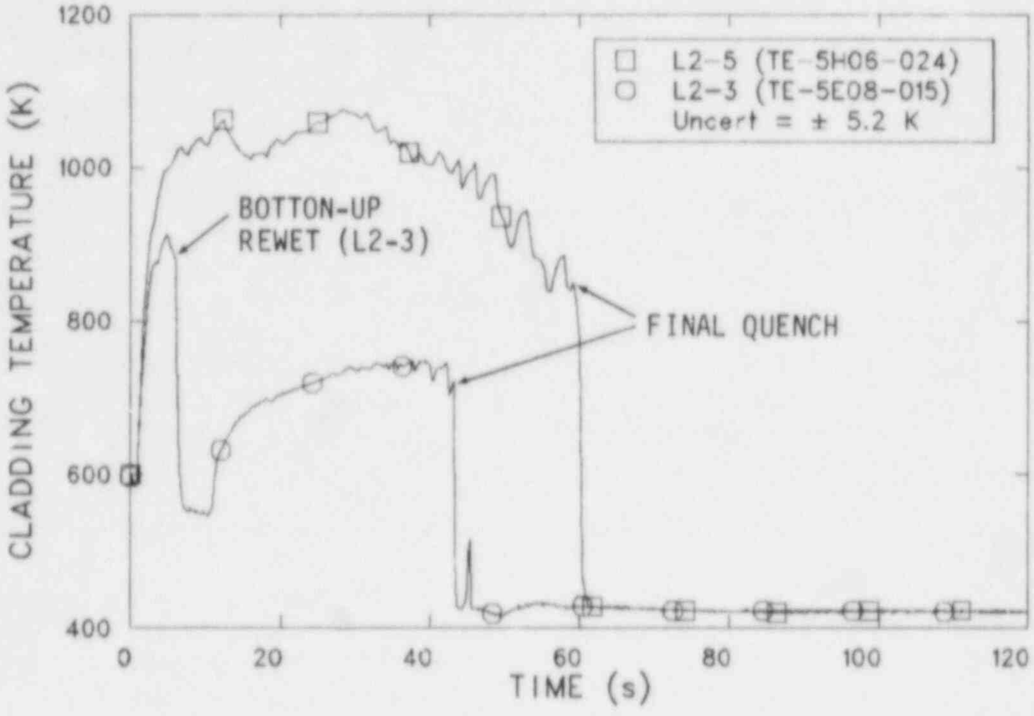

Figure 5-7. Maximum fuel cladding temperature in lower half of fuel module for LOFT Experiments L2-3 and L2-5.

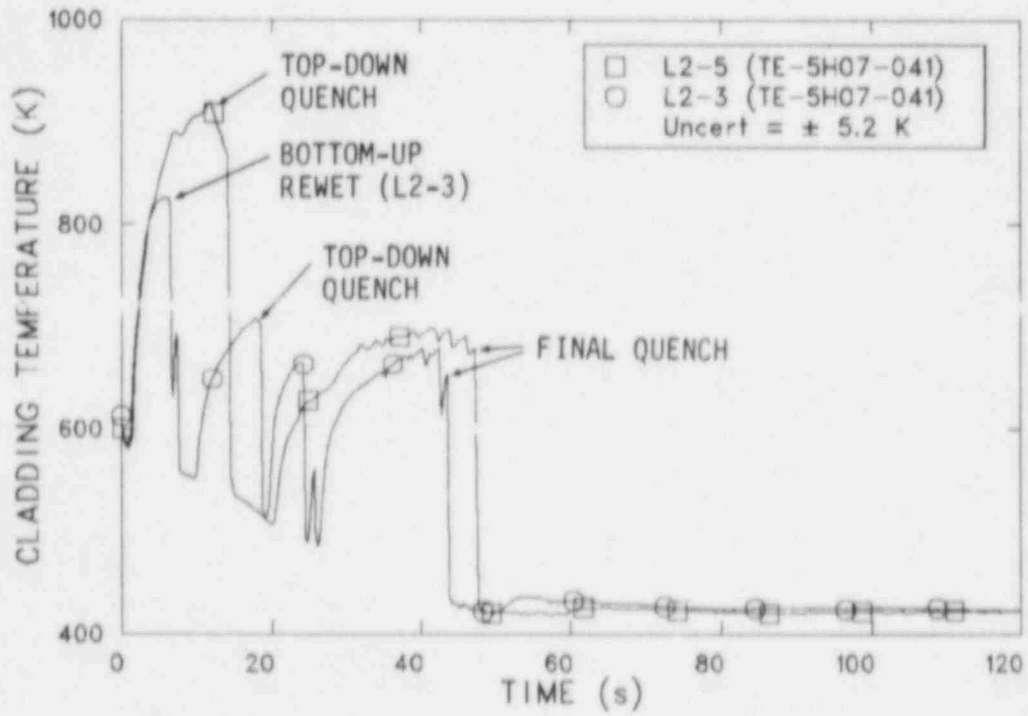

Figure 5-8. Fuel cladding temperature in upper half of center fuel module for LOFT Experiments L2-3 and L2-5.

conditions which would prevent an early core quench after a 200% cold leg break in the Zion nuclear plant were a loss of site power and a broken intact loop pump shaft occurring at the same time as the cold leg pipe break.

5.1.4 Post-CHF Heat Transfer. Appendix K, Section I.C.5 states that, "the following correlations are acceptable in the post-CHF regimes:

1. Groeneveld flow film boiling correlation[5-25]

2. Dougall-Rohsenow flow film boiling correlation[5-26]

3. Westinghouse steady state transition boiling correlation[5-27]

4. McDonough, Milich, and King transition boiling correlation."[5-28]

Appendix K prohibits the use of the Groeneveld correlation near its low-pressure singularity and the first term of the Westinghouse correlation in the post-CHF regimes. In addition, the rule states that, "transition heat transfer shall not be reapplied following CHF in a LOCA transient even if fuel rod cladding superheat returns below 422 K, except during the reflood portion of the LOCA when justified by calculated local fluid and surface conditions."

LOFT Experiment L2-3 demonstrated that the Groeneveld correlation could not predict the core quench that occurred early in the blowdown. A review of the post-CHF correlations in RELAP4/MOD6 showed that rewet could never be predicted for the mid-flow conditions (100 to 600 kg/s·m^2) encountered during Experiment L2-3. As a result of the review, the Biasi CHF correlation[5-29] was installed in RELAP4/MOD6 to cover mid-flow conditions. See Section 4.1.2 for additional discussion of post-CHF heat transfer modeling.

The postexperiment analysis[5-3] used the same RELAP4/MOD6 model of the LOFT system, break flow multipliers (0.848), and transition quality (0.0025) as the revised experiment prediction,[5-16] but used the Biasi CHF correlation and the measured initial experiment conditions instead of the Condie-Bengston correlation used in the revised experiment prediction.

Typical fuel rod cladding surface temperature response calculated in the postexperiment analysis is compared with the experimental data in Figures 5-9 and 5-10. A core-wide rewet was predicted in the postexperiment analysis calculation.

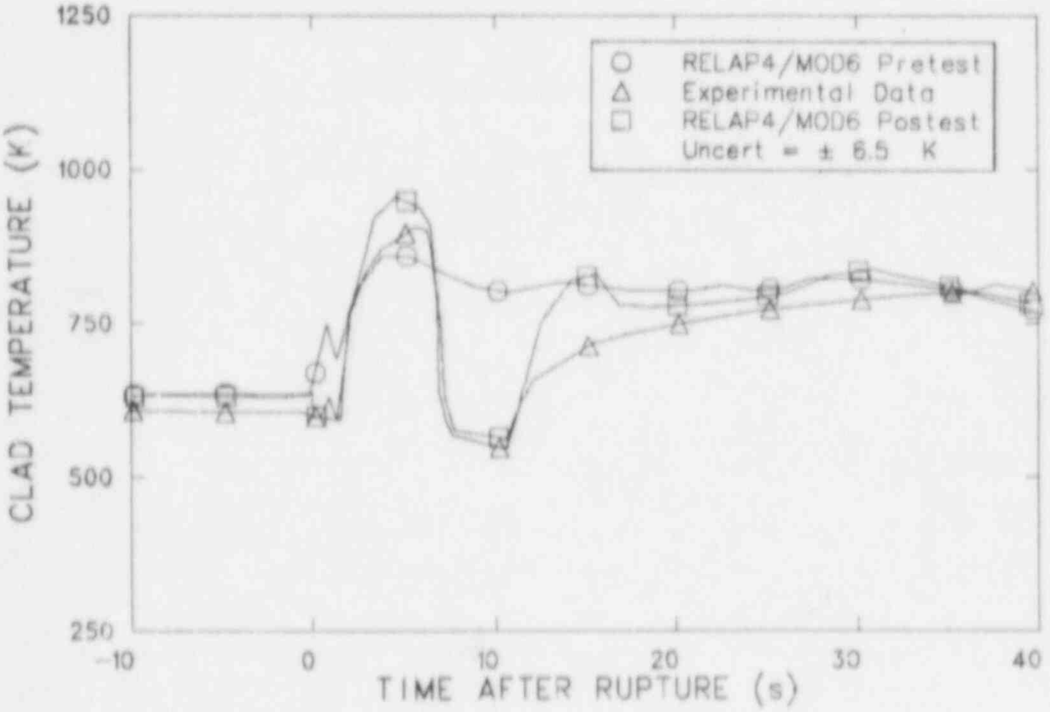

Figure 5-9. Measured, preexperiment prediction, and postexperiment calculation of fuel cladding temperature on Fuel Rod 5F4 at 0.762 m above bottom of core in center fuel module for LOFT Experiment L2-3.

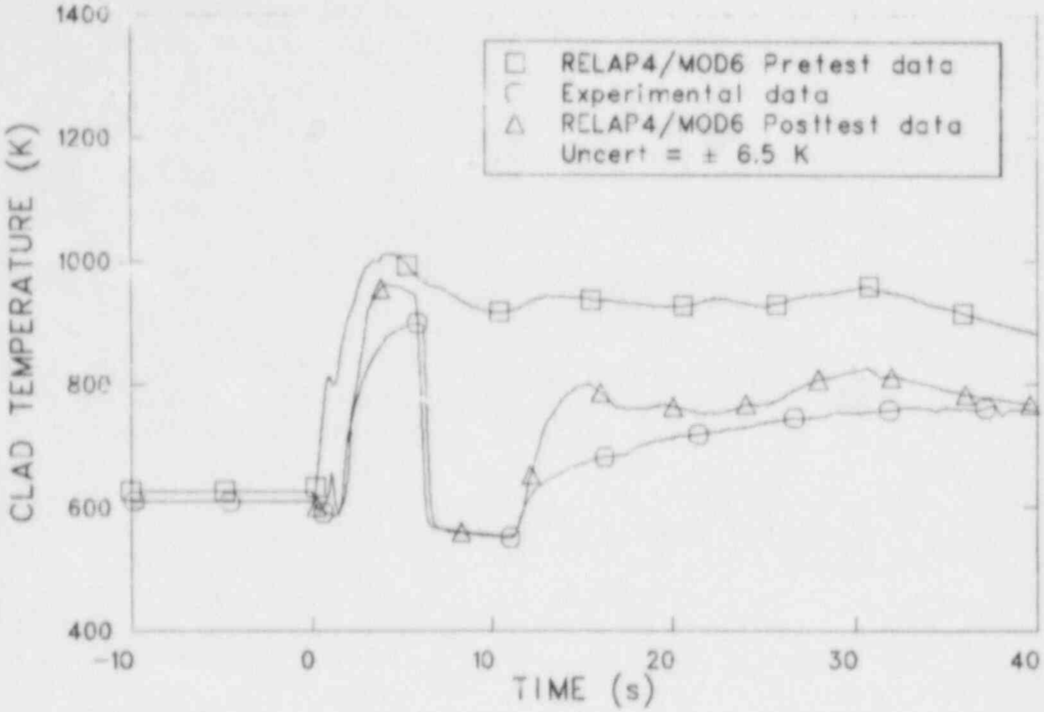

Figure 5-10. Measured, preexperiment prediction, and postexperiment calculation of fuel cladding temperature on Fuel Rod 5J4 at 0.533 m above bottom of core in center fuel module for LOFT Experiment L2-3.

The fuel rod cladding surface temperature calculated in the postexperiment analysis was in reasonably good agreement with the experimental data. The peak cladding surface temperature in the postexperiment calculations, however, was about 70 K higher than was observed in the experimental data. During the first 10 s of the Experiment L2-3 blowdown, the CHF time calculated in the postexperiment analysis occurred about 0.5 s later than in the experimental data, and for the lower and upper thirds of the fuel rod, the rewet time was calculated to occur about 0.5 to 1.0 s earlier in the postexperiment calculation than in the experimental data. For the upper third of the fuel rod, the experimental data showed that the fuel rod experienced a multiple rewetting. This multiple rewetting phenomenon was calculated in the postexperiment analysis, but was not predicted in the preexperiment prediction. Similarly, the postexperiment calculation for the second rewetting for the upper third of the fuel rod was about 3.0 to 3.5 s earlier than was shown in the experimental data, and the second CHF time was about 0.5 s later in the calculation than in the experimental data. These discrepancies resulted from the dryout void fraction used in RELAP4/MOD6 and some of the fine hydraulic structures not predicted in the postexperiment calculation.

The LOFT large-break experiment and analytical results provided valuable data regarding heat transfer phenomena that can occur during an early core quench following a large-break LOCA. The results demonstrated that accurate heat transfer correlations are necessary to predict the accident consequences accurately.

5.2 Small-Break Experiments

LOFT Experiment Series L3 provided data regarding the performance of ECCSs during a variety of small-break LOCEs, plus long-term cooling required to prevent core damage. A short summary of the small-break experiments is presented in Table 2-7. More detail regarding the experiments can be obtained from References 5-30 through 5-53.

This section discusses long-term cooling requirements specified in 10 CFR 50.46 and documentation requirements for the ECCS evaluation models.

5.2.1 Long-Term Cooling Requirements. Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," of 10 CFR 50[5-1] states that, "after any operation of the ECCS, the core

temperature shall be maintained at an acceptably low level, and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core."

After a large- or intermediate-break LOCE in LOFT, the primary system pressure decreases rapidly to the ECCS accumulator pressure setpoint (less than 4.14 MPa). When the system pressure drops below the accumulator setpoint, a large volume of borated water is released which quenches the core. The quench removes the stored energy in the fuel and in the primary cooling system structural components.

Following the quench by the accumulator, decay heat is removed by the ECCS LPIS and the low-pressure heat exchangers. The LPIS consists of a borated water storage tank (BWST), two high-capacity pumps, and valves and piping to inject borated water into any combination of the following: reactor vessel inlet piping, outlet piping, upper and lower plenums, or downcomer. The BWST will supply coolant; however, the LPIS pumps also have the capability to take suction from the broken loop hot leg, BST, or the decontamination sump for decay heat removal and for long-term cooling. Heat exchangers in the pump discharges are provided as an alternate decay heat removal path. The LPIS may be decontaminated by pumping decontamination solution from the BST through the system and discharging to the low-level radioactive waste system.

The scenario for small breaks is quite different from large breaks. Following a small-break LOCA, the primary system pressure remains above the accumulator and LPIS initiation setpoints for an extended period of time (the pressure decay curve for Experiment L3-7 is shown in Figure 5-11). The major objective following a small-break LOCA, therefore, is to remove sufficient stored energy from the fuel and primary coolant system structural components to reduce the primary system pressure to the accumulator and LPIS pressure setpoints so that long-term cooling can be activated.

LOFT Experiment L3-7 simulated a 1-in. diameter break in the cold leg of a commercial PWR (see References 5-51, 5-52, and 5-53). During Experiment L3-7, the steam generator was used to remove decay heat, and to accelerate the cooldown and depressurization of the LOFT system. The primary coolant pumps were turned off during the experiment, and single- and two-phase natural circulation was used throughout the experiment to transfer heat to the steam generator, which was maintained as a heat sink.

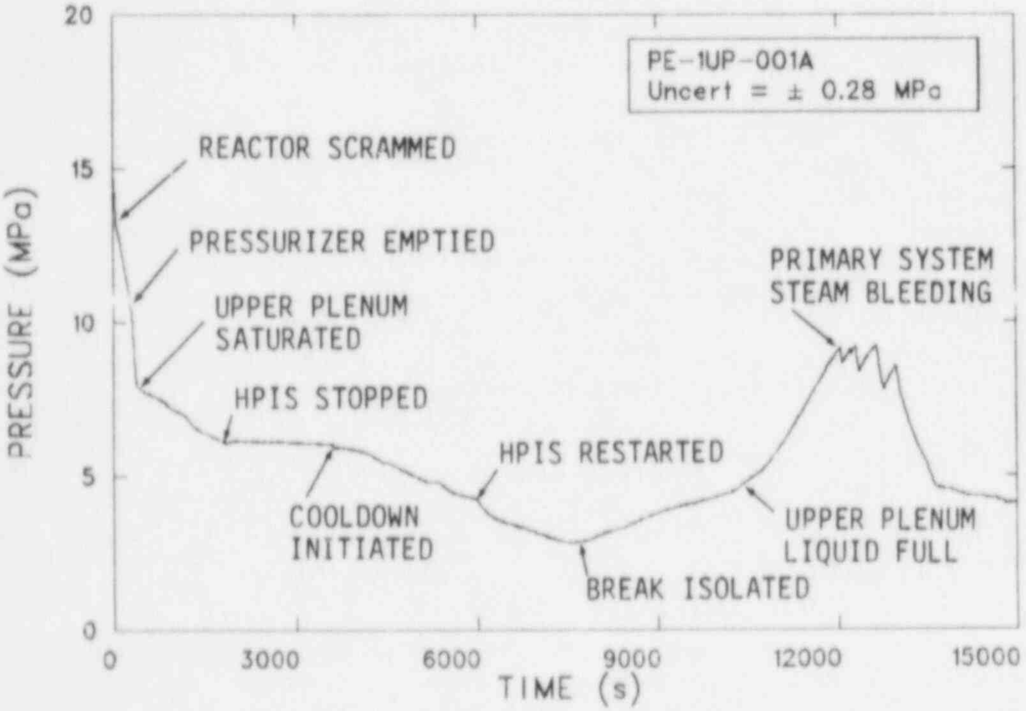

Figure 5-11. Pressure in reactor vessel upper plenum for LOFT Experiment L3-7.

An annotated depressurization curve for the primary system is shown in Figure 5-11. At 36 s after the break occurred, the reactor scrammed on a low system pressure signal. Within 10 s after scram verification, the primary coolant pumps had been manually tripped and had coasted down. Pump coastdown was followed by the inception of natural loop circulation. Between 1800 and 5974 s, the HPIS was turned off to hasten the loss of fluid inventory. Starting at 3600 s, operator-controlled steam bleeding (opening the main steam bypass valve early and the main steam valve later in the transient) and steam generator feeding (using both the auxiliary and main feedwater systems) were used to decrease primary system pressure. Steam generator secondary feed and bleed maintained an effective heat sink throughout the experiment.

At 7302 s after experiment initiation, the blowdown isolation valve 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 due to continued coolant input from the HPIS and decay heat input, 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 was terminated.

Conditions were established in Experiment L3-7 for natural circulation in the primary system. Measurable natural circulation continued from 61 s into the transient, until after the purification system recirculation started at 18,180 s. Single-phase natural circulation started at 61 s, and continued until about 375 s, at which time the upper plenum temperature reached saturation and the transition to two-phase natural circulation began. Two-phase natural circulation was fully established by about 1550 s, when the core temperature differential approached zero.

Single-phase natural circulation was reestablished after the break was isolated at 7200 s. By that time, the fluid in the system had become subcooled. Fluid velocities in the system were lower than during two-phase natural circulation. Core fluid velocity, calculated from decay power and temperature differential, was ~0.05 m/s from 7800 to 15,000 s.

The combination of natural loop circulation and steam generator heat transfer was sufficient to remove decay heat throughout the experiment, including the period of system repressurization beginning at 7915 s. Measured steam generator inlet-to-outlet and primary-to-secondary temperature differences confirmed that the primary-to-secondary heat transfer rates were high. The liquid level in the steam generator secondary side was maintained throughout the experiment, see Figure 5-12.

The parameter of primary interest during Experiment L3-7 was fuel cladding temperature. Cladding temperatures at four locations in the center fuel bundle are shown in Figure 5-13. Coolant temperature in the upper end box of Fuel Assembly 5 is shown in Figure 5-14. Figures 5-13 and 5-14 show that the fuel cladding temperature was very close to the coolant temperature throughout the experiment. This means that the core was never uncovered and that the rate of energy removal through the break and through the steam generator was sufficient to keep the fuel temperature from increasing. In fact, the fuel cladding temperature decreased.

The important result of Experiment L3-7 was that the LOFT primary coolant system temperature and pressure could be controlled during a small-break LOCA by the steam generator and its auxiliary systems until the setpoints were reached to initiate the safety grade decay heat removal systems and establish long-term cooling. This experiment demonstrated the importance of the steam generator in accident recovery procedures.

5.2.2 Documentation Requirements for the ECCS Evaluation Models. Section II.4 of Appendix K to 10 CFR 50 states that, "to the extent practicable, predictions of the evaluation model, or portions thereof, shall be compared with applicable experimental information."

After the accident at Three Mile Island—Unit 2 (TMI-2), the USNRC issued a clarification to the documentation requirements for small breaks (Requirement II.K.3.30, "Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K").[5-54] In the clarification, vendors were directed to provide experimental verification of condensation heat transfer rates, and single- and two-phase natural circulation models. In addition, vendors were directed to provide additional systems verification of their small-break LOCA models by providing predictions of Semiscale Test S-07-10B and LOFT Experiment L3-1.

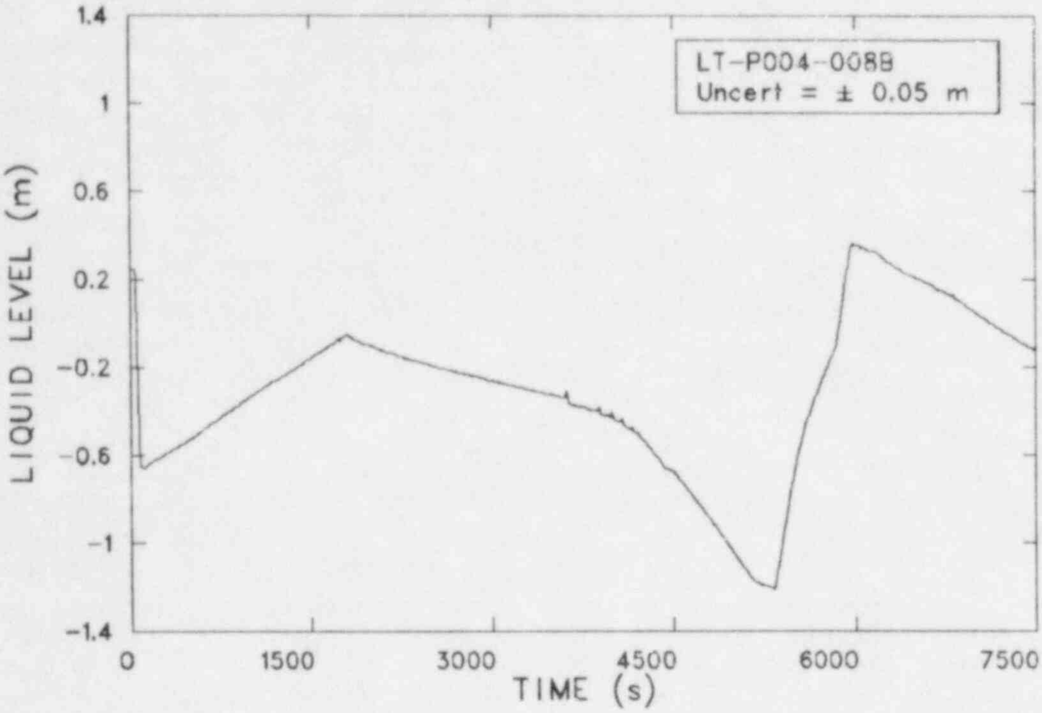

Figure 5-12. Liquid level in steam generator secondary side for LOFT Experiment L3-7.

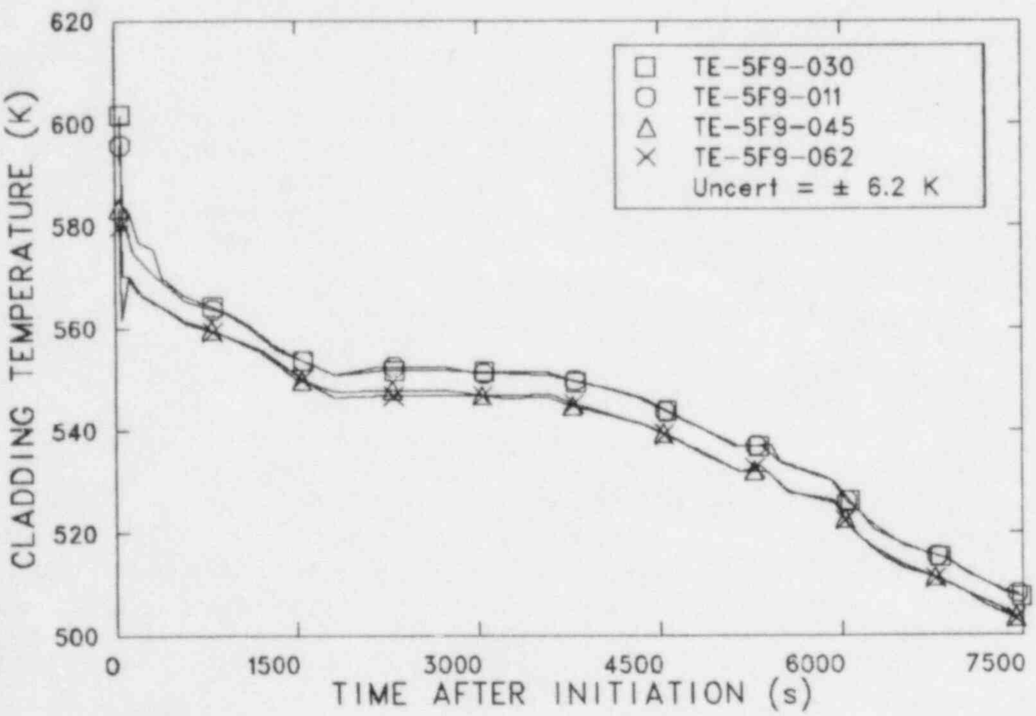

Figure 5-13. Fuel cladding temperature on Fuel Rod 5F9 at 0.28, 0.76, 1.14, and 1.57 m above bottom of core in center fuel module for LOFT Experiment L3-7.

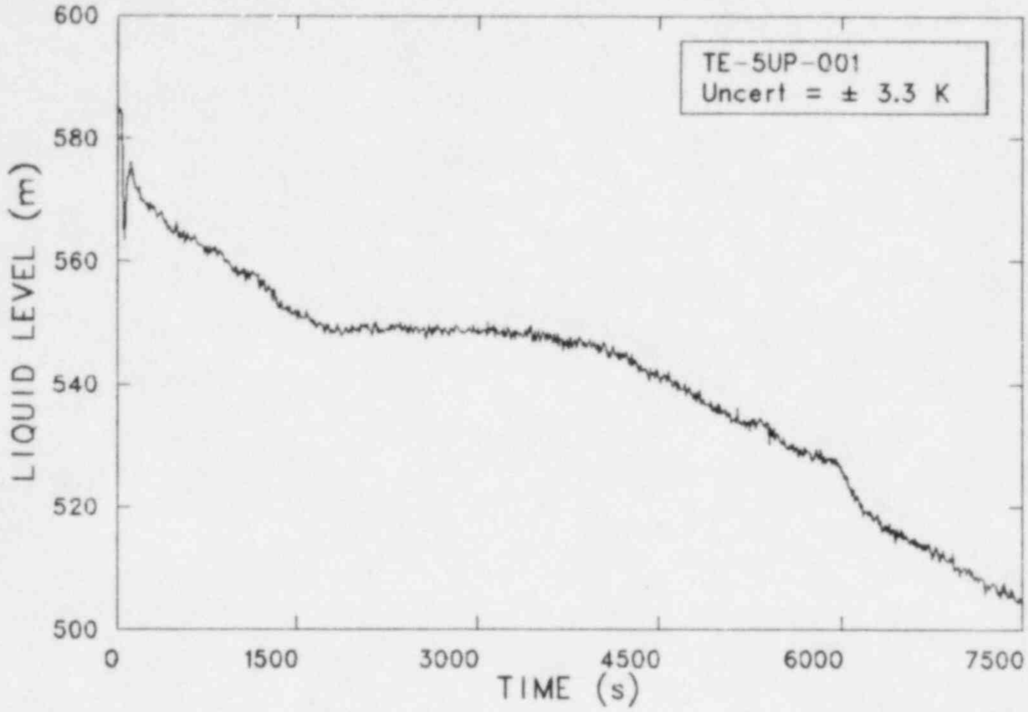

Figure 5-14. Coolant temperature in reactor vessel at upper end box of center fuel module for LOFT Experiment L3-7.

Since Reference 5-54 was issued, LOFT Small-Break Experiment Series L3 has been completed, and more data are now available on condensation heat transfer, single- and two-phase natural circulation, and system performance during small-break LOCAs.

5.2.2.1 Condensation Heat Transfer. LOFT Experiment L3-7 (see References 5-42, 5-51, and 5-53) provided data that could be used to qualify a condensation heat transfer model. During a significant portion of the experiment, two-phase natural circulation was used to transfer heat to the steam generator. The heat transfer modes in the steam generator were convection and condensation of steam within the fluid.

5.2.2.2 Single- and Two-Phase Natural Circulation. Single- and two-phase natural circulation was observed during several LOFT experiments. The small-break experiments that are most suitable for quilifying natural circulation models are LOFT Experiments L3-5/L3-5A[5-46] and L3-7.[5-53]

LOCE L3-5/L3-5A consisted of two parts. The first part, L3-5, simulated a 4-in. (2.5%) diameter pipe break in a commercial PWR and was one of two experiments required to investigate the effects of primary coolant pump operation during a small-break LOCA. In L3-5, the primary coolant pumps were tripped. The second part of the experiment, L3-5A, demonstrated reestablishment of the steam genrator as a heat sink and the reestablishment of two-phase natural circulation. Pressure versus time for the primary and secondary systems are temperature difference across the core are shown in Figure 5-15 and 5-16, respectively, for Experiment L3-5/L3-5A.

Experiment 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 commercial PWR. The prime objective of the experiment was to impose a break flow equal to HPIS flow at an intermediate (~6.9-MPa) pressure during the transient, to establish conditions for steam generator cooling, and to isolate the break and recover the plant to cold shutdown.

An annotated plot of pressure in the primary and secondary systems during Experiment L3-7 is shown in Figure 5-17. 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. Starting at 3600 s, operator-controlled steam bleeding (by opening the main steam bypass

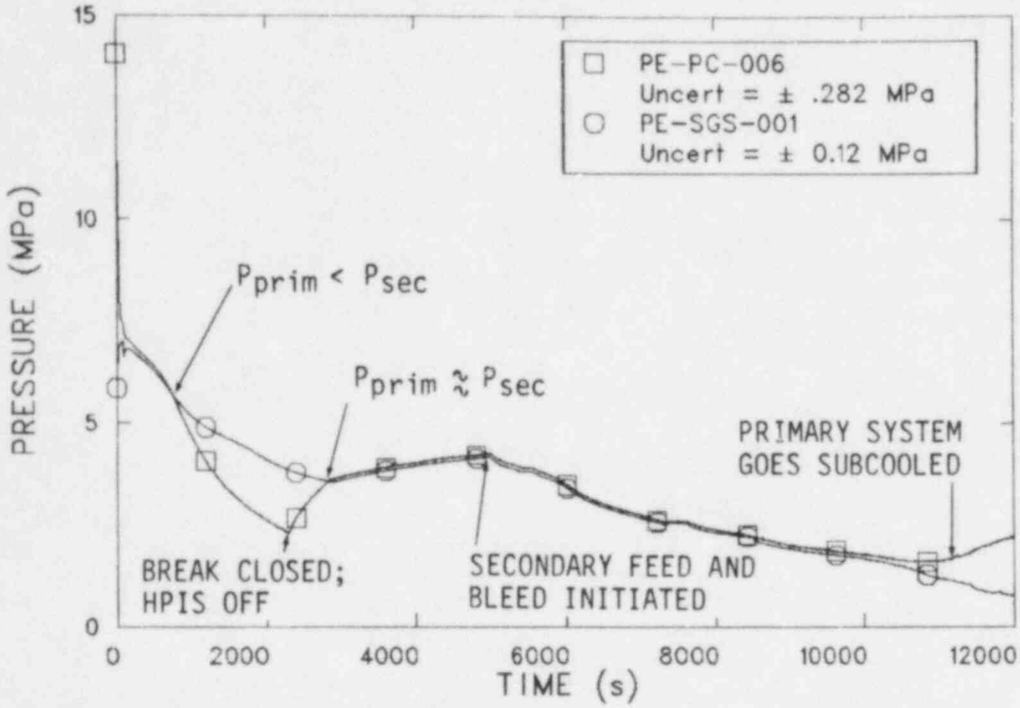

Figure 5-15. Pressure in primary system intact loop and in steam generator for LOFT Experiment L3-5/L3-5A.

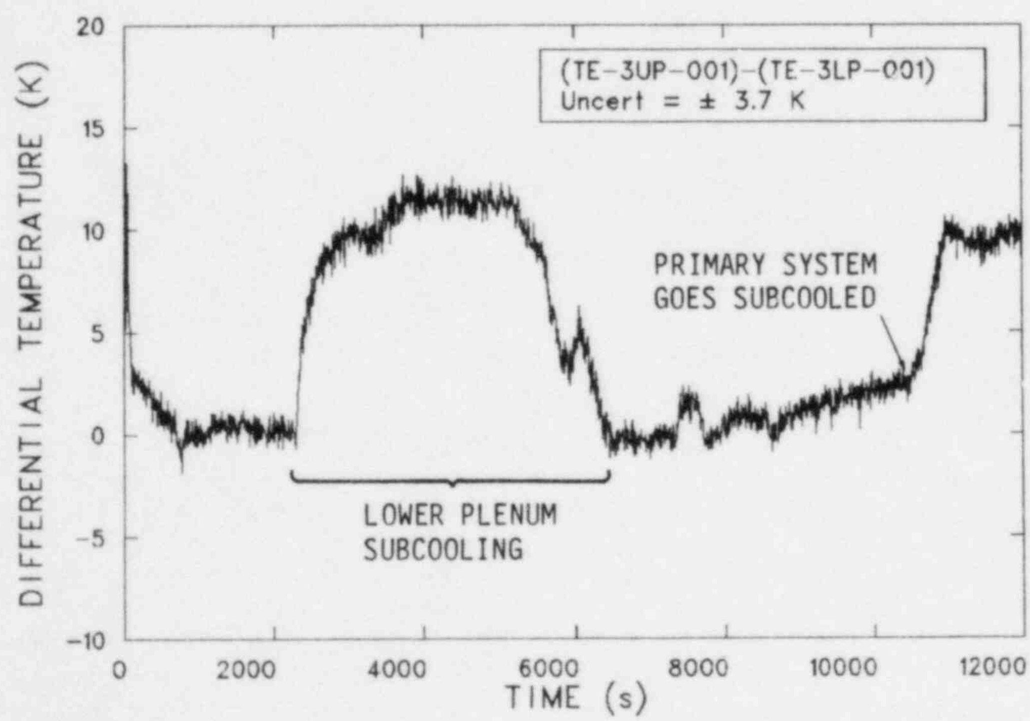

Figure 5-16. Temperature difference across the core for LOFT Experiment L3-5/L3-5A.

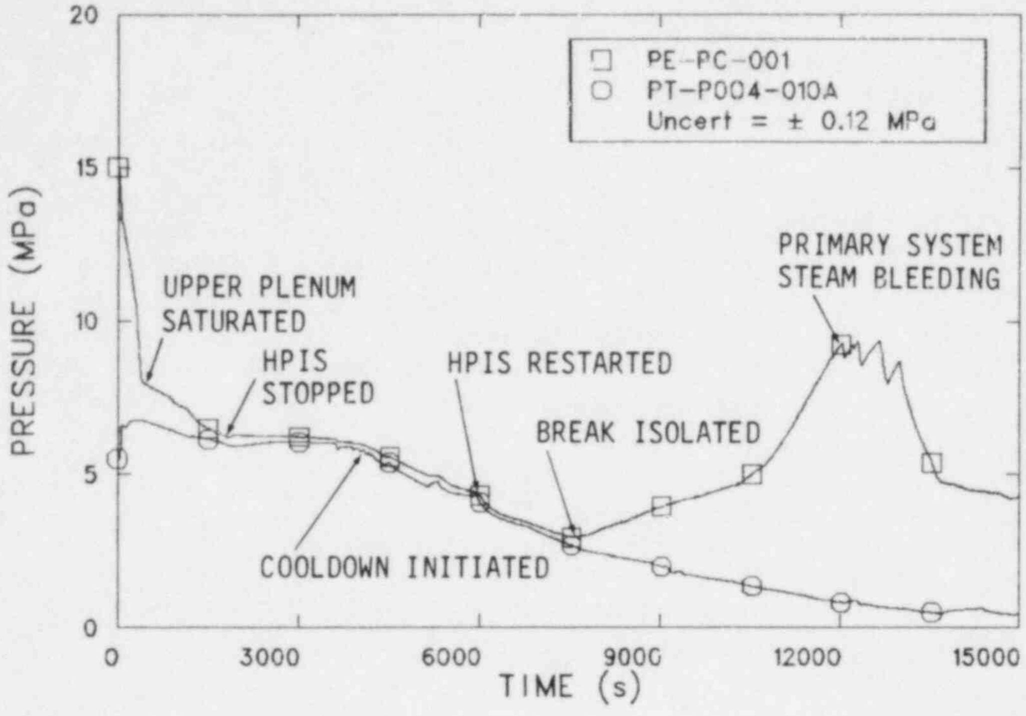

Figure 5-17. Pressure in primary and secondary systems for LOFT Experiment L3-7.

valve early and the main steam valve later in the transient) and feeding (using both the auxillary and main feedwater systems) were used to decrease primary system pressure.

At 7200 s, the break was isolated. System mass depletion stopped and all decay heat energy, not lost to the environment, was removed by the steam generator. Due to HPIS injection, 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 was 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). 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 recirculation cooling at 18 180 s.

5.2.2.3 System Performance. LOFT Small-Break Experiment Series L3 provided information about system performance during small-break transients. In addition to overall performance, the experiments provided specific information about: (a) pressurizer behavior when the PORV is open, and (b) system behavior with primary coolant pumps on and off.

5.2.2.3.1 Pressurizer Behavior with PORV Open— LOFT Experiment L3-0 simulated a small break at the top of the LOFT pressurizer by opening the PORV (see References 5-30, 5-31, and 5-32). The reactor was shut down for the experiment, and the ECCS (including the HPIS, LPIS, and accumulators) was not allowed to actuate until the transient was terminated. Power to the primary coolant pumps was tripped at transient initiation, and the pumps were allowed to coast down.

Experiment L3-0 was initiated by opening the PORV which allowed fluid from the top of the pressurizer vessel to blow down through pressurizer system piping to the BST. The transient was slow and remained under manual control until it was terminated when the primary system pressure had dropped to 3.53 ± 0.2 MPa at 2460 s after initiation. System depressurization to 6.8 MPa was rapid until primary system saturation occurred at 48 s; a gradual pressure reduction continued until the transient was terminated.

System hydraulics were characterized by pressurizer blowdown to primary saturation pressure, followed by a refill of the pressurizer to the top of its indicating range by 84 s due to vapor generation in the primary system. The pressurizer continued to indicate "liquid full" until ~1450 s, when the liquid level slowly dropped back into the indicating range, see Figures 5-18 and 5-19.

The main conclusions of Experiment L3-0 were:

1. During a LOCA caused by a stuck open PORV, the pressurizer starts filling when the primary system fluid saturates some place other than in the pressurizer itself. The pressurizer fills and may remain full even though steam exists in the hot leg piping and in the top of the reactor vessel. This will give a false indication of system coolant inventory.

2. During a blowdown, the indicated pressurizer liquid level will be higher than the actual level unless the level indication is compensated for fluid temperature.

Additional information on pressurizer behavior with the PORV/SRV cycling open and closed to control primary pressure during ATWS simulations was obtained from LOFT Experiments L9-3 and L9-4, which are discussed in Section 4.2.2.

5.2.2.3.2 System Behavior with Pumps On and Off— LOFT Experiments L3-5 and L3-6 were conducted to investigate the effect of pump operation on system behavior (see Reference 5-43 and 5-50). For Experiment L3-5, the primary coolant pumps were tripped within 1 s after break initiation, and had coasted down by 18 s. For Experiment L3-6, the primary coolant pumps continued to operate at full speed until they were tripped at 2371 s, when the reactor primary coolant system pressure had decreased to 2.27 MPa. Continued operation of the primary coolant pumps in LOFT produced a significant difference in system mass inventory relative to the early pump trip case. A comparison of the computed primary coolant system mass inventory in Figure 5-20 shows that the minimum coolant inventory reached in Experiment L3-5 was 1820 ± 50 kg at 2100 s,[a] whereas the minimum in Experiment L3-6 was 679 ± 50 kg at 2400 s. Minimum coolant mass was ~33% of initial mass for the early pump trip case, as compared to ~12% for the delayed pump trip case. The difference in coolant depletion is directly attributable to the effect pump

a. Total LOFT primary coolant system water inventory at operating conditions is ~5500 kg.

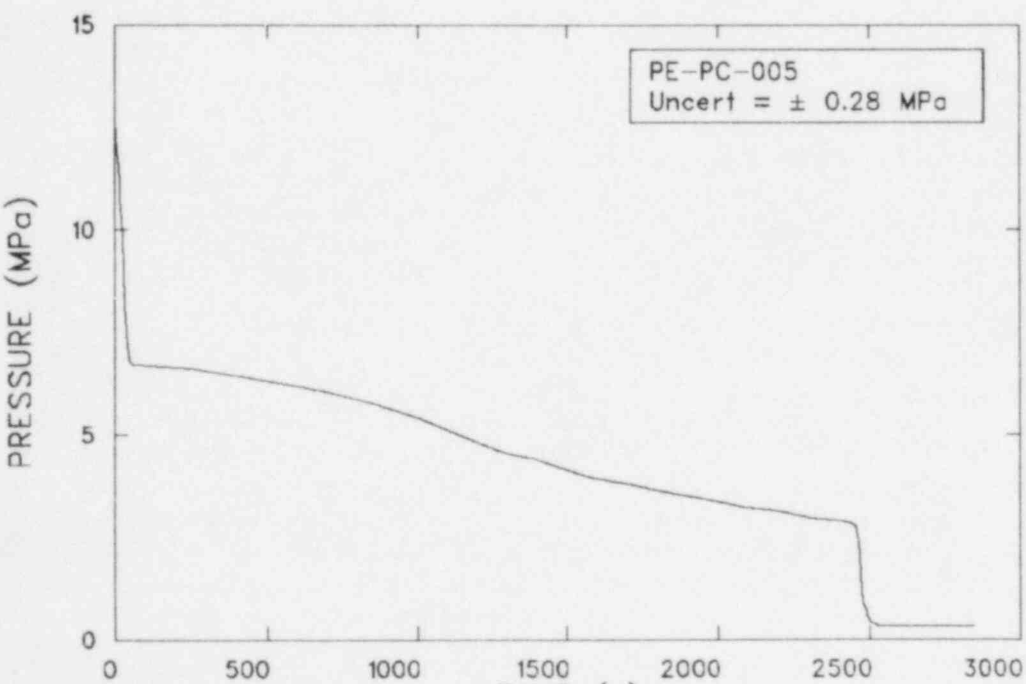

Figure 5-18. Pressure in primary system for LOFT Experiment L3-0.

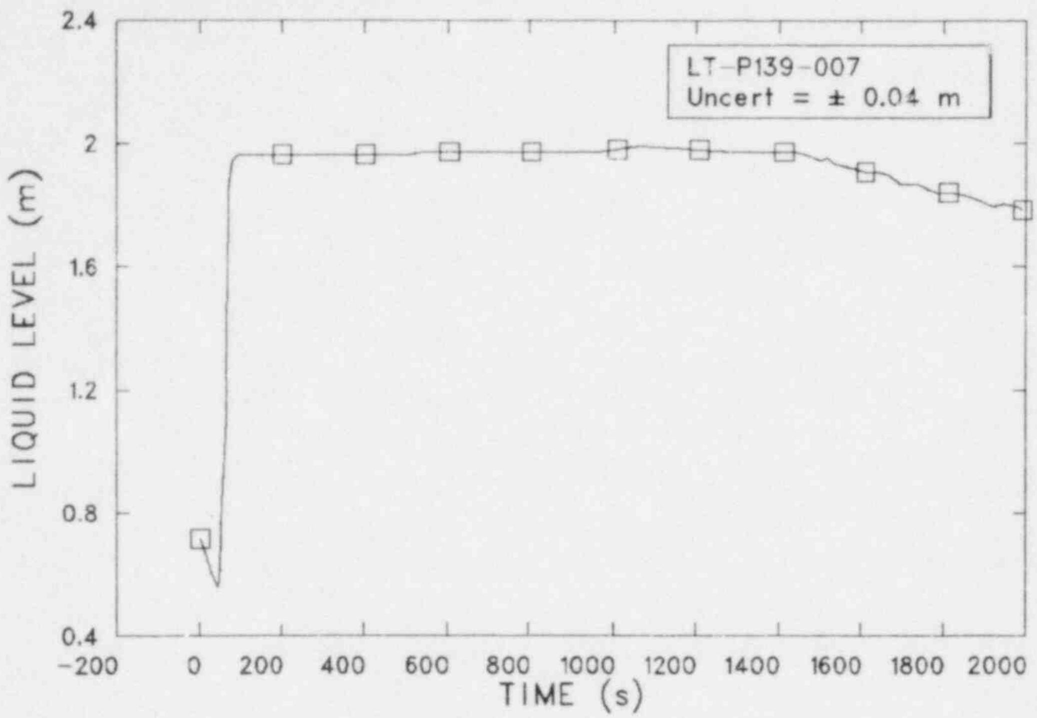

Figure 5-19. Liquid level in pressurizer for LOFT Experiment L3-0.

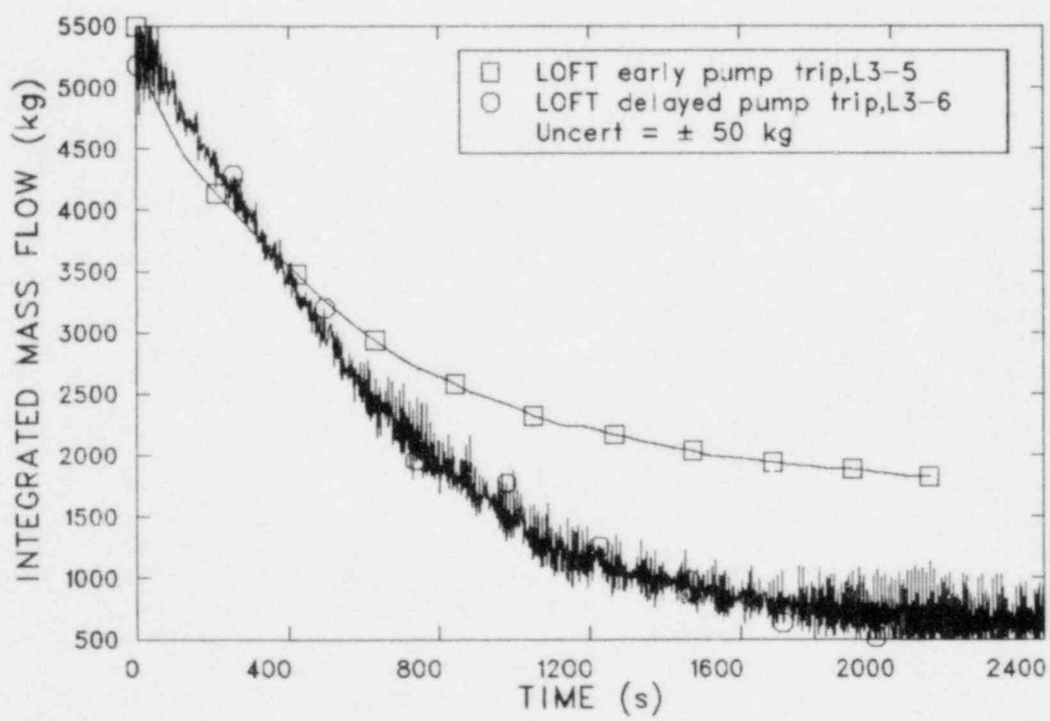

Figure 5-20. Mass inventory in primary system for LOFT Experiments L3-5 (early pump trip) and L3-6 (delayed pump trip).

operation had on the rate of discharge through the break.

The mass flow rate at the break was significantly higher for most of the transient with the pumps operating than with them tripped (see Figure 5-21) because of the delivery of higher density fluid to the break location, as shown in Figures 5-22 and 5-23. The exception to this occurred during the first 135 s in the blowdowns, when the total system mass depleted more rapidly in Experiment L3-5 (see Figure 5-21). The absence of pumped flow in this experiment caused the fluid in the intact loop cold leg to remain subcooled longer and to a greater extent than in Experiment L3-6. Pump operation in Experiment L3-6 caused greater homogenization of loop coolant temperatures. The greater degree of subcooling in Experiment L3-5 led to a slightly higher break mass flow during this period.

With the pumps tripped in Experiment L3-5, primary coolant mass depletion was characterized by draining of liquid from the higher elevations downward, and eventual stratification of liquid and vapor in the intact loop cold leg. When the water level in the intact loop cold leg dropped to the elevation of the tee connection to the break orifice, the density of fluid delivered to the break showed a marked decrease, indicating the passage of a greater proportion of steam. The break orifice being located at the end of a tee connection to the main coolant pipe accentuated the separation of liquid and steam, since the steam component negotiates the 90-degree turn into the tee more easily than the liquid.

In Experiment L3-6, the pumps continued to deliver a homogeneous mixture of steam and liquid to the intact loop cold leg throughout the period they were operated. Consequently, source fluid at the tee connection was of higher density than was the case in Experiment L3-5.

The primary coolant pumps were operated at a constant speed of 3250 rpm until 2371 s in Experiment L3-6. At ~30 s, the pumps began to cavitate with little change in inlet and outlet coolant density. Performance of the pumps throughout the transient was smooth, with no indication of slugging as would be indicated by fluctuations in pump current, differential pressure, and coolant density. The gradually decreasing density of coolant passing through the pump was attended by a corresponding drop in pump motor power. Figure 5-24 shows that the correlation of these two parameters produces a fairly uniform relationship.

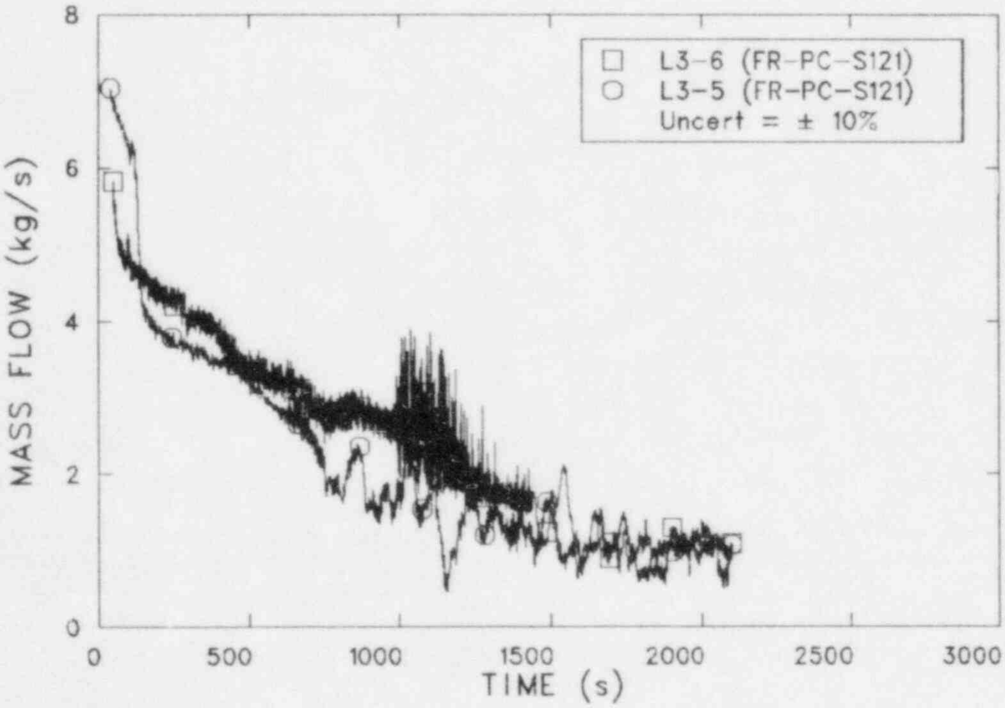

Figure 5-21. Break mass flow rate for LOFT Experiments L3-5 (early pump trip) and L3-6 (delayed pump trip).

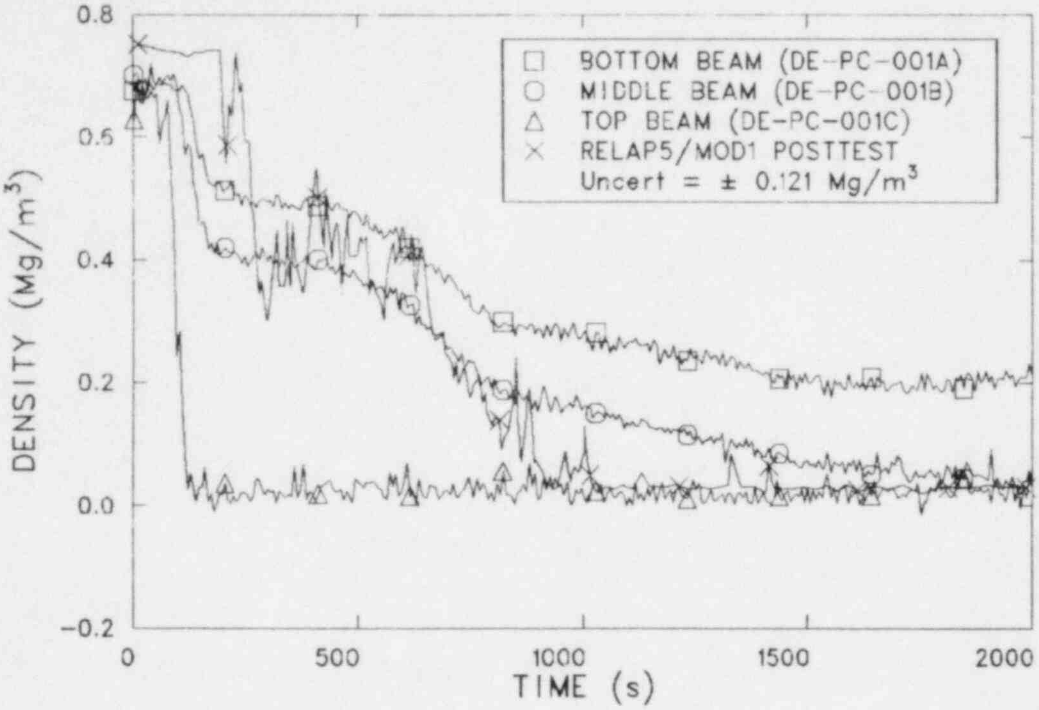

Figure 5-22. Fluid density in intact loop cold leg for LOFT Experiment L3-5 with early pump trip (chordal densities).

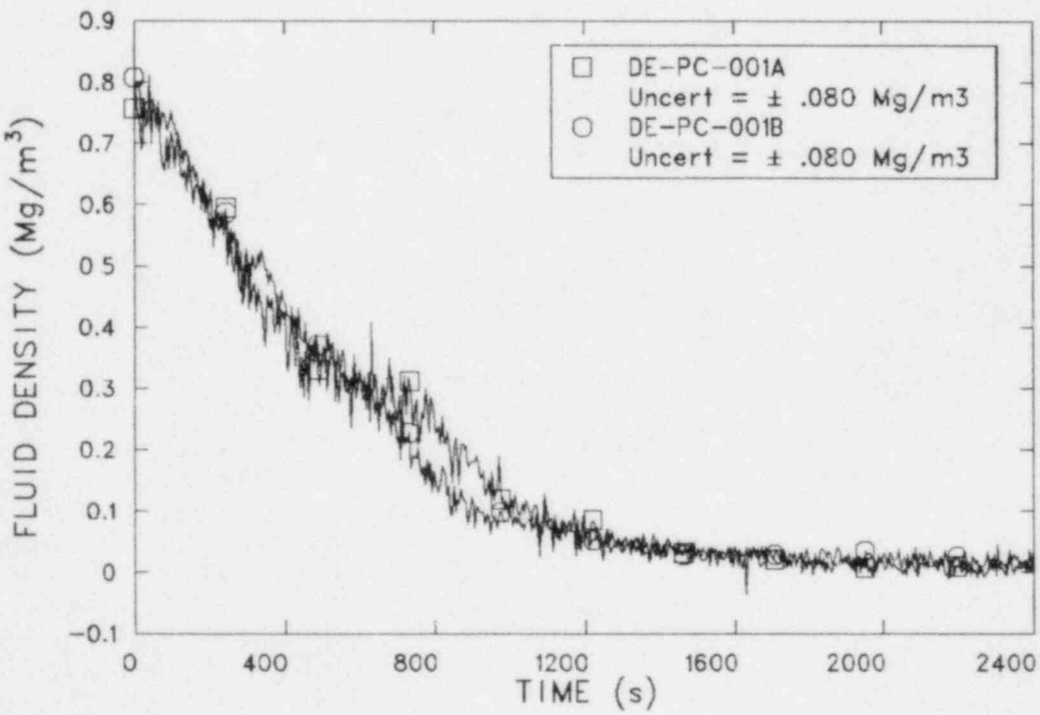

Figure 5-23. Fluid density in intact loop cold leg for LOFT Experiment L3-6 with delayed pump trip (chordal densities).

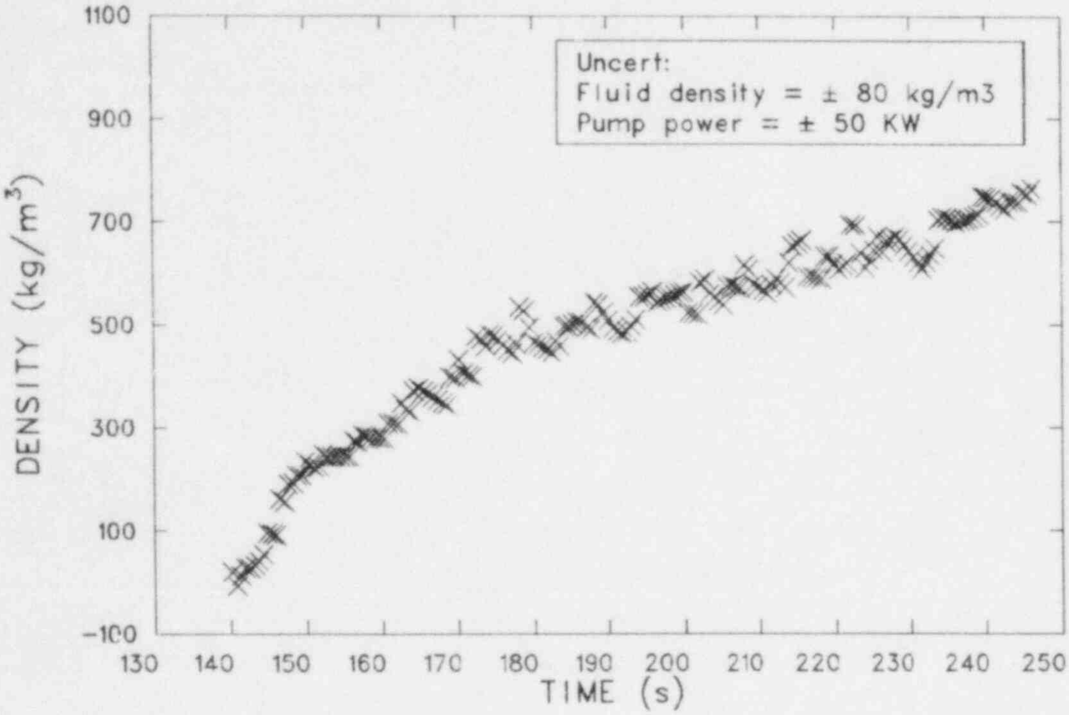

Figure 5-24. Pump inlet fluid density versus power for LOFT Experiment L3-6 with delayed pump trip.

Experiment L3-5 was terminated at 2400 s, when the break was isolated and secondary feed and bleed operations began. At this point, the coolant level in the reactor vessel was determined to be ~0.6 m above the top of the core. In Experiment L3-6, the primary coolant pumps were intentionally tripped off at 2371 s, and coasted down within 90 s. Fuel rod cladding temperatures remained near the system saturation temperature while the pumps were operating, indicating satisfactory forced cooling was occurring even into a dispersed-droplet or annular-flow-regime period which occurred just prior to pump trip.

Within 25 s after pump trip in Experiment L3-6 (that is after the experiment was terminated), the coolant velocity in the reactor vessel was insufficient to entrain liquid droplets up through the core. The liquid level collapsed to below the lower core elevation, exposing the core to a pure steam environment. As the liquid film on the fuel rods dried out, temperature excursions began. The rate of fuel rod cladding temperature increase at the maximum power elevation (0.74 m) was ~2.5 K/s, which is close to a computed adiabatic heatup rate of 2.8 to 3.3 K/s.

The onset of core temperature excursions in Experiment L3-6 was rapidly followed by operator-initiated accumulator injection, which began ~53 s later. A maximum recorded fuel rod cladding temperature of 637 K occurred at 2466 s.

Clearly, for the accident scenario investigated in LOFT, early pump trip produced a much less severe transient in terms of primary coolant depletion and, consequently, core coolability. However, care should be taken to not extrapolate the LOFT results directly to a commercial PWR. Differences in break location and system geometry could produce significantly different results.

5.3 Intermediate-Break Experiments

After extensive investigation of large- and small-break LOCAs, the behavior of a commercial PWR after the rupture of an ECC injection line or other primary coolant system penetration lines remained a concern because of the potential of degraded ECC performance, and because of the uncertainty in modeling flow regimes transitional between stratified and homogeneous. LOFT Experiments L5-1 and L8-2 (see References 5-55, 5-56, and 5-57) were performed to investigate ECCS performance and thermal-hydraulic response during intermediate-break LOCEs. These experiments are summarized in Table 2-7.

Experiment L5-1 was an intermediate-break experiment designed to simulate the rupture of a single 28.45 cm (11.2-in.) inside diameter accumulator line in a commercial PWR. All three components of the plant protection system (HPIS, accumulator, and LPIS) were used to bring the plant to a safe shutdown. Experiment L8-2 was a repeat of Experiment L5-1 up to the time of accumulator injection. Accumulator injection was delayed in Experiment L8-2 to evaluate the LOFT system thermal-hydraulic response following a significant primary system coolant inventory loss and a primary coolant pump restart. Annotated depressurization curves for the primary and secondary systems during Experiment L5-1 are shown in Figure 5-25.

5.3.1 ECCS Performance After Intermediate Break. As the primary coolant system mass inventory decreased, the reactor vessel liquid level had dropped from the upper plenum into the core by 108 s. The liquid level was radially uniform, as evidenced by the thermocouple measurements at the same elevation in the various fuel assemblies which showed departure from saturation at the same time as shown in Figure 5-26. The cladding heatup rate in the hottest region in the core was 2.7 K/s, or 57% of adiabatic. The reactor vessel liquid level continued to drop, uniformly, at a rate of ~0.02 m/s until the core was ~95% uncovered at 186 s, when the accumulator injection initiated. The primary coolant system mass inventory had decreased to 16% of its original value at the time accumulator injection began as shown in Figure 5-27. The fuel cladding started quench ~2.5 s later, which corresponds to the delay time observed in previous experiments.

During the Experiment L5-1 reflood, the peak cladding temperature reached a maximum of 715 K at the 0.99-m elevation, ~0.33 m above the peak power elevation. The reflood continued in a radially uniform manner at an average reflood rate of 0.055 m/s until the core was completely reflooded at 214 s.

After the core reflood was started, the primary system pressure started to increase because of steam generation from the ECC. By 225 s, primary system pressure exceeded the accumulator injection pressure, and this injection ceased. It did not restart until 308 s, when the primary system pressure again decreased to below accumulator injection pressure. The reflood of the core was completed with the injection of ~28% of the scaled accumulator liquid volume.

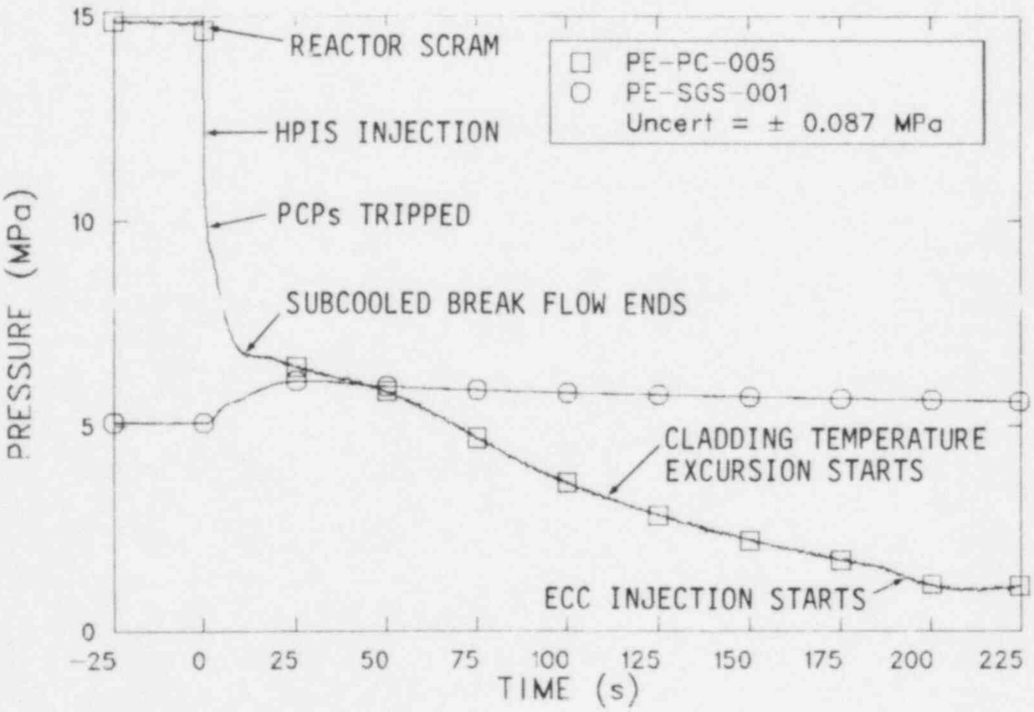

Figure 5-25. Pressure in primary and secondary systems for LOFT Experiment L5-1.

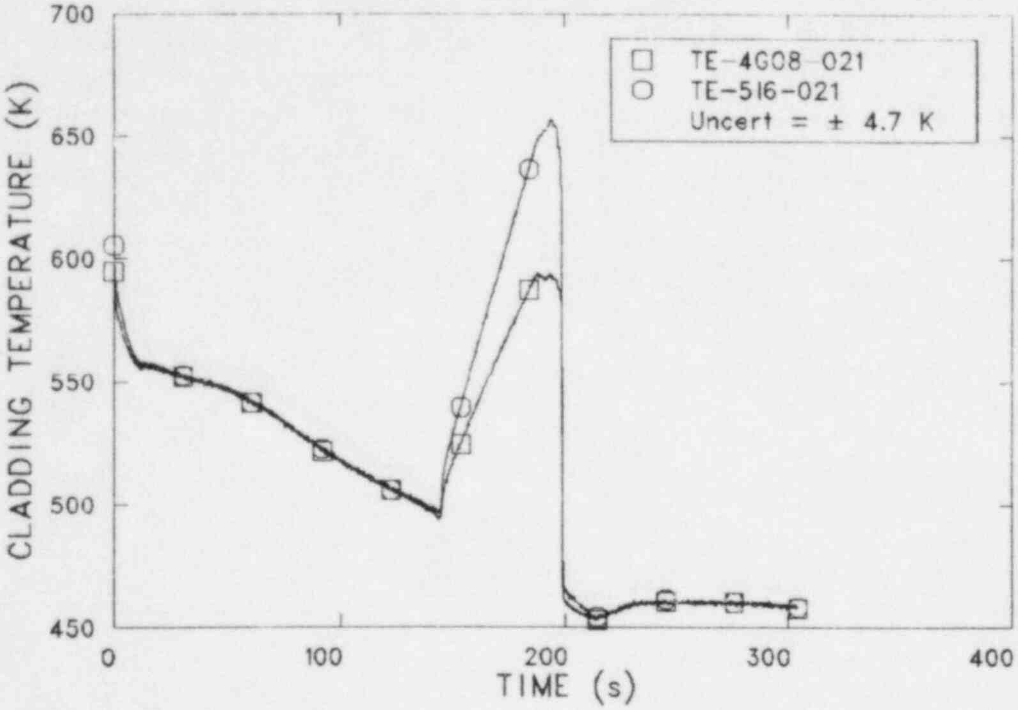

Figure 5-26. Fuel cladding temperature in center fuel module and a peripheral fuel module for LOFT Experiment L5-1.

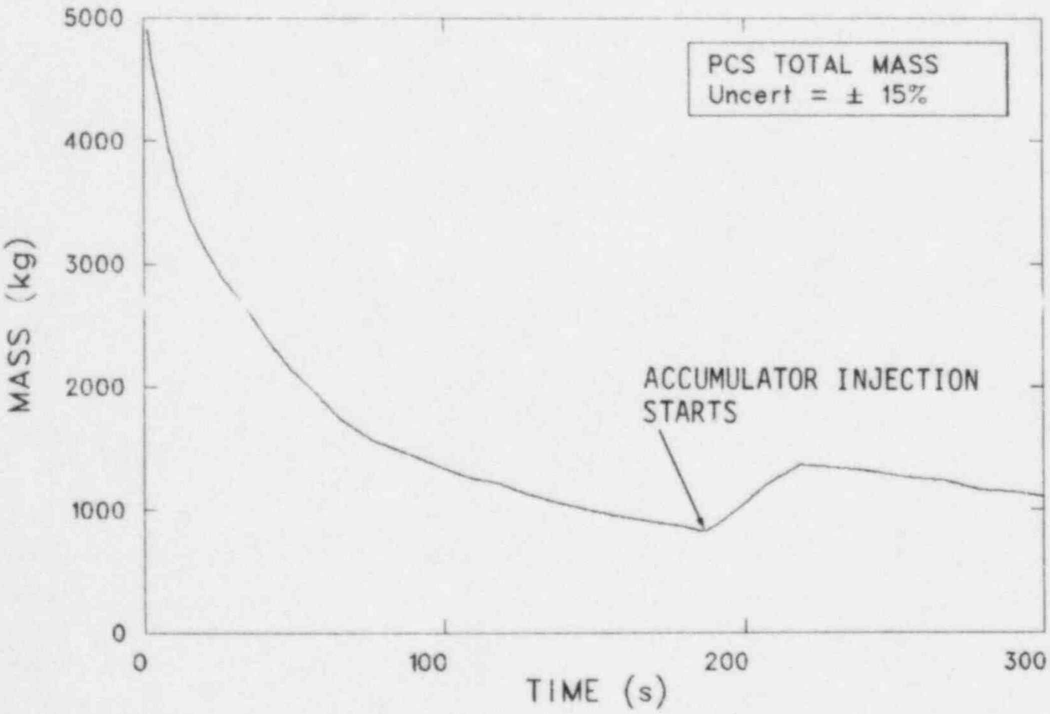

Figure 5-27. Mass inventory in primary system for LOFT Experiment L5-1.

The results of Experiment L5-1 demonstrated that ECCS injection was sufficient to recover the system during an intermediate-size break (28.45-cm inside diameter) and prevent an excessive core thermal excursion with accumulator injection starting at a primary system pressure of 1.66 MPa.

Experiment L8-2 was a repeat of Experiment L5-1 up to the time of accumulator injection. In Experiment L8-2, accumulator injection was withheld in order to determine the effect of restarting the primary coolant pumps on the fuel cladding temperature. Accumulator injection was manually initiated on a predetermined value of fuel cladding temperature. The initial conditions were very nearly the same for both experiments.

Of primary interest in Experiment L8-2 was the primary coolant pump behavior and the effect on cladding temperature after the pumps were restarted. At the initiation of pump restart, the pump frequency quickly surged to ~56 Hz and then slowed down to between 18.3 and 28.3 Hz. The initial surge caused a partial quench of the fuel cladding in the lower 0.13 m of the core (see Figure 5-28) and minor cooling up to 0.99 m. After the initial surge, the fuel cladding, which had been partially quenched, again dried out and the thermal excursion continued. The pumps continued to pump steam through the core, and did slow down the heatup rate by as much as 25% in the lower half of the core; however, the pumps were unable to reverse the thermal excursion in any part of the core except during the initial surge.

Accumulator 'A' injection was manually initiated at the predetermined cladding temperature of 950 K. The cladding temperature increased at a heatup rate of 2.8 K/s, or 60% adiabatic, to a second predetermined value of 978 K before the reflood process halted the cladding temperature rise. At this temperature, HPIS 'B' and Accumulator 'B' injections were added. The reflood process quenched the hot region of fuel cladding during this time, and limited the cladding temperature to a maximum of 987 K.

The calculated and measured fuel cladding surface temperatures near the core hot spot are compared in Figure 5-29. The comparison is good until the primary coolant pumps were restarted. The measured data show a minor influence starting at 235 s, when the pumps were restarted; whereas, the prediction shows that the heatup rate decreased more.

The fuel cladding quench and core reflood results of these intermediate-break experiments did not

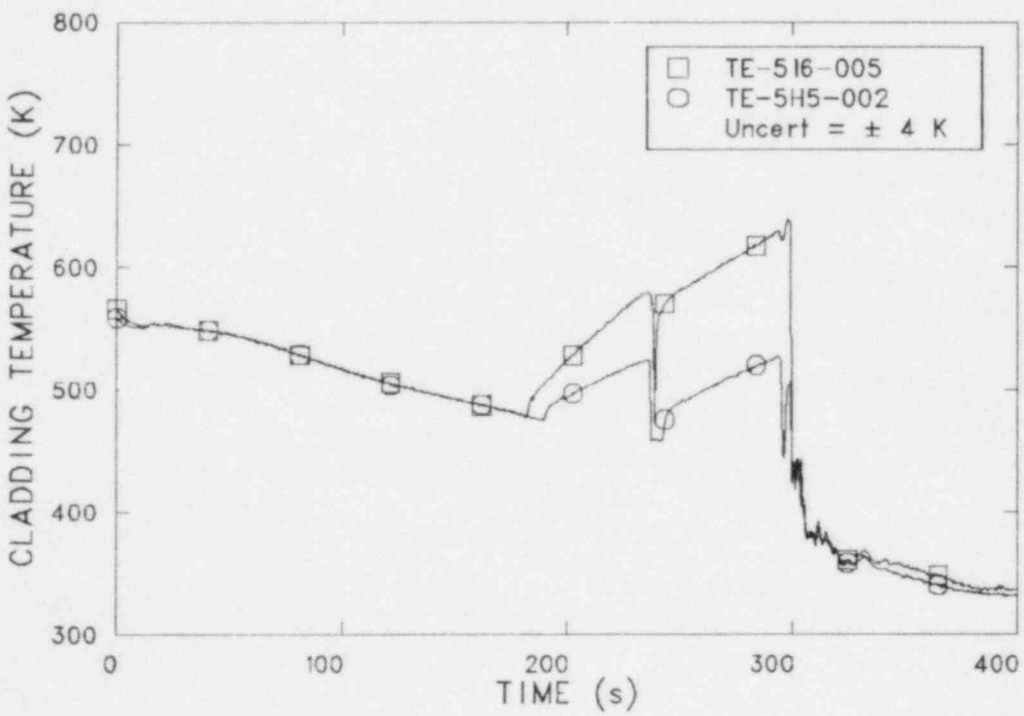

Figure 5-28. Fuel cladding temperature in center fuel module for LOFT Experiment L8-2.

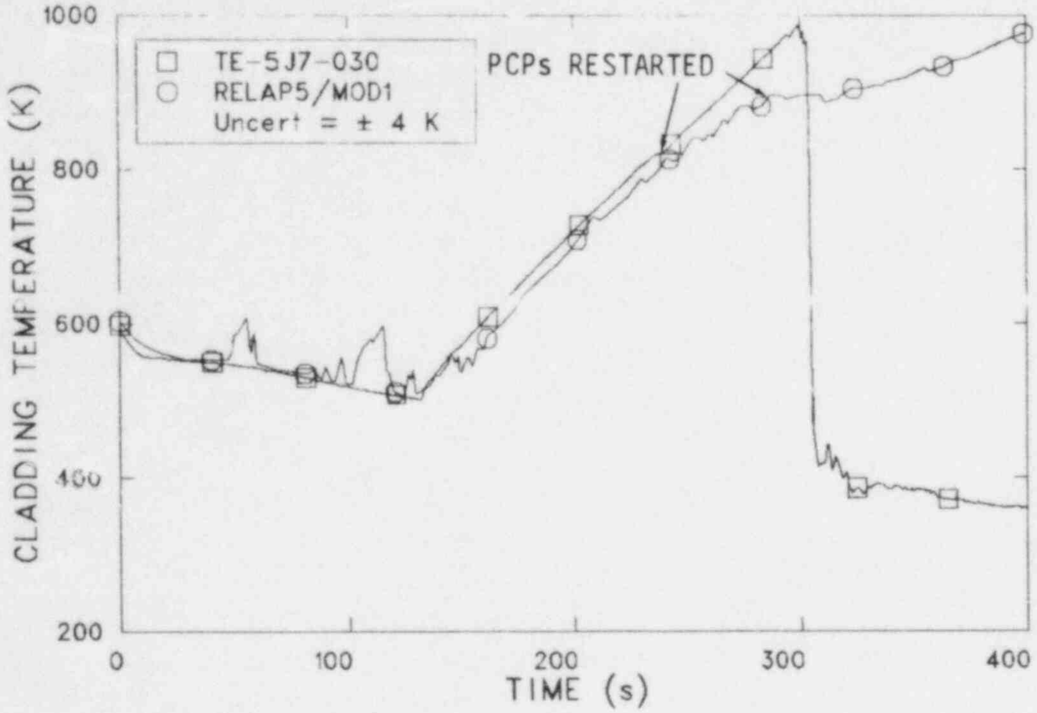

Figure 5-29. Measured and predicted maximum fuel cladding temperature for LOFT Experiment L8-2.

reveal the occurrence of any phenomena that would adversely affect the design of the ECCSs, even to the use of a low-pressure setpoint (1.66 MPa) for the accumulator. Since this conclusion also applies for the largest pipe break possible in commercial PWRs and since these intermediate-break transients exhibited some of the characteristics of the large break transient, these data support the position that the ECCS design currently employed in commercial PWRs is adequate for "automatic" PWR recovery from pipe breaks within this size range.

5.3.2 Intermediate-Break Flow Modeling. The response of Experiment L5-1 was dominated by the break flow because most of the system energy was removed through the break. The break nozzle designed for Experiment L5-1 has a smooth elliptical entrance and a small L/D ratio, which is different from nozzles used for previous LOFT small-break experiments. Pre- and postexperiment analyses of Experiments L5-1 and L8-2 were performed with RELAP5/MOD1.

During the preexperiment analysis phase, a sensitivity study was run on the discharge coefficient for subcooled and saturated discharge because no calibration data were available for deriving the discharge coefficients. Coefficients of 0.7, 0.84, and 1.0 were used for subcooled discharge, and 0.84 and 1.0 were used for the saturated discharge. Without specific data, it was believed that the range of coefficients spanned the current values.

In the postexperiment analysis reported in Reference 5-57, the best agreement was obtained with a subcooled discharge coefficient of 0.98 and a saturated discharge coefficient of 0.84. The results are shown in Figure 5-30.

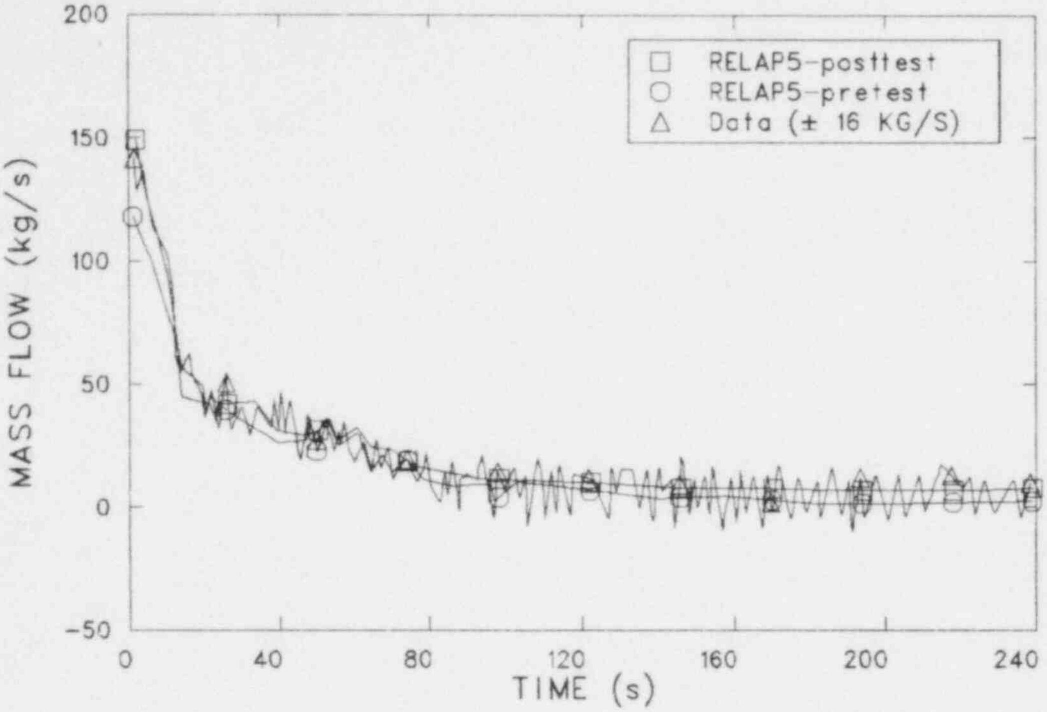

Figure 5-30. Measured, preexperiment prediction, and postexperiment calculation of break mass flow rate for LOFT Experiment L5-1.

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6. IMPACT ON SAFETY ISSUES

The reactor licensing process is based upon conservative design and development practices. The adequacy of the licensing process is continuously questioned. In most cases, the questions are readily and fully answered. Occasionally, however, there are questions or concerns which the USNRC does not consider to be immediate safety concerns, but which require additional consideration for resolution. These items are called unsolved safety issues (USIs).

A total of 26 USIs have been identified.[6-1] Technical resolution is complete for 11 USIs', however, complete resolution, which requires implementation, is generally an ongoing task. Table 6-1 presents a short summary of LOFT data that relates the USIs.

LOFT has provided a significant amount of data to three USIs: shutdown decay heat removal (SDHR) requirements, station blackout, and ATWS. In addition, LOFT Experiment L1-5 provided data regarding asymmetric blowdown loads on reactor primary coolant systems, and the mini-blowdown experiments (Series L0) and the nonnuclear large-break experiments (Series L1) provided experimental and analytical results that are applicable to several USIs that deal with hydrodynamic loads on a BST (Mark I Long-Term Program and SRV Pool Dynamic Loads).

This section presents the LOFT results relevant to SDHR requirements and station blackout. A discussion of ATWS is presented in Section 9, Rulemaking. Section 4 presents the results of the mini-blowdown and nonnuclear large-break experiments that are relevant to the Mark I Long-Term Program and SRV Pool Dynamic Loads.

The data from Experiment L1-5 regarding asymmetric blowdown loads on reactor primary coolant systems has not been analyzed in detail because the measured displacement of the center fuel module (see Reference 6-2) was much less than predicted in the preexperiment analysis reported in Reference 6-3.

Table 6-1. Summary of LOFT information relevant to unresolved safety issues

Unresolved Safety Issue	Description	Status	Relevant LOFT Information
Water hammer	Shock waves in the primary or secondary system generated by rapid valve motion, rapid pump startup, rapid condensation of steam pockets, etc. Damage from water hammer to pipe, pipe supports and valves has been reported.	Unresolved	None
Asymmetric blowdown loads on reactor primary coolant systems	The concern is that loads generated by a pipe break may damage the reactor vessel supports, primary system piping, or reactor internals.	Technical resolution complete	LOFT Experiment L1-5 provided data regarding decompression loads on the reactor core.
Steam generator tube integrity	Steam generator tube integrity can be degraded by corrosion, cracking, denting, and vibration-induced fatigue cracks. The concern is the ability of degraded steam generator tubes to withstand normal operating and transient conditions.	Unresolved	None
Mark I Short-Term Program	Preliminary evaluation of suppression pool hydrodynamic loads associated with LOCAs and operation of SRVs.	Technical resolution complete	See Mark I Long-Term Program.

Table 6-1. (continued)

Unresolved Safety Issue	Description	Status	Relevant LOFT Information
Mark I Long-Term Program	Determine suppression pool hydrodynamic loads associated with LOCAs and operation of SRVs. The program involves analytical and experimental studies.	Unresolved	The LOFT experiments did not directly provide data for resolving the Mark I containment safety problem. However, the LOFT BST is similar to a Mark I suppression tank, and the suppression tank data obtained during LOFT Experiment Series L0 (mini-blowdown experiments) and Series L1 (non-nuclear large-break experiments) may be applicable and can be used to evaluate or verify analytical techniques.
Mark II containment pool dynamic loads	Determine the adequacy of the Mark II containment to withstand LOCA blowdown loads.	Technical resolution complete	Same as Mark I Long-Term Program.
Anticipated transient without scram (ATWS)	An ATWS is an anticipated transient during which the reactor fails to scram. The concern is that an ATWS could lead to a core melt and release of radioactive fission products.	Technical resolution complete. These proposed rules have been issued for comment. Final rule to be issued in 1983	LOFT Experiments L9-3 and L9-4 were ATWS experiments. These experiments provide data that can be used in the resolution of the ATWS issue.
Boiling water reactor (BWR) feedwater nozzle cracking	Cracks in feedwater nozzle in BWRs have been caused by stress corrosion at nozzle welds.	Technical resolution complete	None
Reactor vessel materials toughness	Concern is long-term fracture toughness of reactor vessel material subjected to neutron indication.	Unresolved	None
Steam generator and reactor coolant pump supports	ASM AS72, used in support of steam generators and pumps, has been found to lack adequate toughness at 70°F. Fracture toughness of steam generator and coolant pump supports of all PWRs will be reassessed.	Unresolved	None
Systems interaction	The purpose of this task is to identify where the present design analysis and review procedures may not adequately account for potential adverse systems interactions, and to recommend regulatory action to correct deficiencies in the procedures.	Unresolved	None
Qualification of Class 1E safety related equipment	NRC staff has worked with nuclear industry to develop standards for equipment qualification and documentation which will ensure the high level of equipment reliability.	Technical resolution complete	None
Reactor vessel pressure transient protection	The purpose of this task was to provide an overpressure protection system that would be in use when the plant is in a cold shutdown condition.	Technical resolution complete	None

Table 6-1. (continued)

Unresolved Safety Issue	Description	Status	Relevant LOFT Information
Residual heat removal (RHR) requirements	The purpose of this task was to resolve differences between the NRC and Westinghouse Electric Corporation regarding a proposed revision to the standard review plan dealing with the RHR system. The revision would have required plants to have the capability to go from hot to cold shutdown without offsite power. Westinghouse contended that the revision would have significant impact on the chemical volume control system and interface requirements on balance of plant design.	Technical resolution complete	None
Control of heavy loads near spent fuel	Overhead handling systems (cranes) are used to lift heavy objects in the vicinity of spent fuel in PWRs and BWRs. If a heavy object, e.g., a spent fuel shipping cask or shielding block, were to fall or tip onto spent fuel in the storage pool or the reactor core and damage the fuel, there could be a release of radioactivity to the environment and a potential for radiation overexposure to inplant personnel. If many fuel assemblies were damaged and the damaged fuel contained a large amount of undecayed fission products, radiation releases to the environment could exceed 10 CFR Part 100 guidelines.	Technical resolution complete	None
Determination of SRV pool dynamic loads and temperature limits for BWR containments	Operation of BWR primary system pressure relief valves can result in hydrodynamic loads on the suppression pool retaining structures or structures located within the pool. These loads result from: 1. Initial vent cleaning of a relief valve pipe 2. Steam quenching due to locally high pool temperature.	Unresolved	Data from Experiment Series L0 (mini-blowdown experiments)provide data on the hydrodynamic pressures and loads generated by vent cleaning.
Seismic design criteria	The seismic design process required by the NRC includes: defining the magnitude of an earthquake for a site, determining free-field motion at the site, determining the motion of site structure and equipment, and fully comparing seismic loads with allowable loads. While this seismic design sequence includes many conservative factors, certain aspects of the sequence may not be conservative for all plant sites. The objective of this program is to investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites, to investigate alternate approaches to parts of design sequence, to quantify the overall conservatism of the design sequence, and to modify the NRC criteria in the Standard Review Plan if changes are found to be justified.	Unresolved, a draft report is being reveiwed	None
Pipe cracks in boiling water reactors	Leaks and cracks in the heat affected zones of welds that join austenitic stainless steel piping and associated components in BWRs have been detected. All the cracks were attributed to intergranular stress corrosion cracking due to high local stress, sensitization of material, and high oxygen content in the water.	Technical resolution complete	None

Table 6-1. (continued)

Unresolved Safety Issue	Description	Status	Relevant LOFT Information
Containment emergency sump	Following a LOCA in a PWR, water flowing from the break in the primary system would collect on the containment floor. During the injection mode, water for core cooling and containment spray is drawn from a large supply tank. When the water reaches a low level in the tank, pumps are realigned to draw from the containment. This is called the recirculation mode wherein water is drawn from the containment floor or sump and pumped to the primary system or containment spray headers. This program addresses the safety issue of adequate sump or suppression pool function in the recirculation mode. It is the objective of this program to develop improved criteria for design, testing, and evaluation which will provide better assurance that emergency sumps will function to satisfy system requirements. The principal concerns are: (a) Break-initiated debris from the insulation used on piping and components inside the containment could cause blockage of the sump or otherwise adversely affect the operation of the pumps, spray nozzles, and valves of the safety systems. (b) Adverse flow conditions that affect hydraulic performance of the sump have been encountered requiring design and procedural modifications to eliminate them; these conditions, air entrainment, cavitation, and vortex formation, are aggravated by blockage.	Unresolved	None
Station blackout	The loss of ac power from the offsite and onsite sources is referred to as a station blackout. In the event of a station blackout, the capability to cool the reactor core would be dependent on the availability of systems which do not require ac power supplies, and on the ability to restore ac power in a timely manner. The concern is that the occurrence of a station blackout may be a relatively high probability event and that the consequences of this event may be unacceptable, for example, severe core damage may result.	Unresolved	LOFT Experiment L9-4 was an ATWS initiated by a loss of offsite power. The experiment simulated a station blackout combined with a failure to scram the reactor.
Shutdown decay heat removal requirements	The NRC staff feels that providing an alternative means of decay heat removal (DHR) could substantially increase the plant's capability to deal with a broader spectrum of transients and accidents and potentially could, therefore, significantly reduce the overall risk to the public. This task will investigate alternative means of DHR in PWR plants, including but not limited to using existing equipment where possible. The approach taken to this problem comprises the following subtasks: 1. Development of criteria to judge acceptability of DHR systems in existing and future plants 2. Development of means for improvement of DHR systems 3. Assessment of existing plants to identify those in which DHR systems require improvement 4. Development of a plan for implementing new requirements, if any, for DHR systems required to meet the acceptance criteria above.	Unresolved	LOFT Experiments L3-2, L3-7, and L9-1/L3-3 provided data on several methods of removing shutdown decay heat.

Table 6-1. (continued)

Unresolved Safety Issue	Description	Status	Relevant LOFT Information
Seismic qualification of equipment inoperating plants	The seismic qualification of the equipment in operating plants will be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this USI is to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria for new plants. This guidance will concern eqiupment required to safely shut down as well as equipment whose function is not required for safe shutdown, but whose failure could result in adverse conditions which migl impair shutdown functions.	Unresolved	None
Safety implications of control systems	This issue concerns the potential for accidents or transients being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration and would be in addition to any control system failure that may have initiated the event. The purpose of this USI is to define generic criteria that may be used for plant-specific reviews. A specific subtask of this issue will be to study the steam generator overfill transient in PWRs and the reactor overfill transient in BWRs to determine and define the need for preventive and/or mitigating design measures to accommodate this transient.	Unresolved	None
Hydrogen control measures and effects of hydrogen burns on safety equipment	Postulated reactor accidents which result in a degraded or melted core can result in generation and release to the containment of large quantities of hydrogen. The hydrogen is formed from the reaction of the zirconium fuel cladding with steam at high temperatures and/or by radiolysis of water. This USI will investigate means to predict the quantity and release rate of hydrogen following degraded core accidents and various means to cope with large releases to the containment such as inerting of the containment or controlled burning. The potential effects of proposed hydrogen burns on safety related equipment will be investigated.	Unresolved	None
Pressurized thermal shock	Neutron irradiation of reactor pressure vessel weld and plate materials decreases the fracture toughness of the materials. The fracture toughness sensitivity to radiation-induced change is increased by the presence of certain materials such as copper Decreased fracture toughness makes it more likely that, if a severe overcooling event occurs followed by or concurrent with high vessel pressure, and if a small crack is present on the vessel's inner surface, that crack could grow to a size that threatens vessel integrity.	Unresolved	None

6.1 Shutdown Decay Heat Removal Requirements

Following a LOCA or transient, such as a loss of offsite power or turbine trip, the reactor normally scrams. Decay heat continues to be generated, decreasing exponentially with time, and must be removed along with the stored heat in the reactor fuel, structural components, and primary coolant system to bring the plant to cold shutdown, The transition from reactor trip to "hot shutdown," excluding the initial reflood phase in a severe LOCA as discussed in Reference 6-4, is defined as the shutdown decay heat removal (SDHR) phase. Studies such as WASH-1400[6-5] have shown that, in general, for PWRs the major contributor to the probability of severe damage to the fuel stems from failures to retain cooling in the SDHR phase.

Subtask 2.1 of the Task Action Plan[6-4] is concerned with phenomenological studies. Part I of the subtask concerns alternate means of SDHR from PWRs that appear to be technically feasible, but which have not yet been formally approved, even for emergency situations, in either existing or future plants. The SDHR procedures being considered are:

1. Transfer of heat from the reactor core to the steam generators by two-phase natural circulation. Two-phase natural circulation is a condition where coolant, heated in the reactor core, forms a two-phase mixture which rises and flows through the hot leg piping to the steam generator where it is cooled and condensed. The cooler, denser coolant then flows back to the reactor through the cold leg piping.
2. Operation of the steam generating units as reflux condensers, as an alternative to true natural circulation. When a U-tube steam generator operates as a reflux condenser, a two-phase mixture flows up the inlet leg of the U-tubes and condenses before reaching the top of the U-tubes. The condensate then flows back down the U-tubes and returns to the reactor via the hot leg.
3. The use of a high-pressure residual heat removal (RHR) system which could, in an emergency, be brought into use before stable "hot shutdown" conditions have been established.
4. Application of the feed and bleed concept.
5. Operation of a shutdown PWR with limited boiling in the core.

Natural circulation (Procedure 1) was investigated during LOFT Experiments L3-7 and L9-1/L3-3, steam generator feed and bleed (Procedure 4) was investigated during Experiment L3-7, and reflux cooling (Procedure 2) was inferred during Experiment L3-2.

6.1.1 Transfer of Heat from the Reactor Core to the Steam Generators by Natural Circulation. Natural circulation was investigated in several LOFT experiments. During Experiment L3-7, which simulated a 1-in. diameter break in the cold leg of a commercial PWR,[6-6,6-7] natural circulation was used throughout the experiment to remove shutdown decay heat. The experiment results demonstrated that the transition between single-phase and two-phase natural circulation and between two-phase and single-phase natural circulation was smooth, and both single- and two-phase natural circulation were effective means of SDHR. LOFT Experiment L9-1/L3-3, which simulated an ATWS initiated by a loss of feedwater,[6-8] demonstrated that two-phase natural circulation can be restarted and used effectively to cool the plant after a loss-of-feedwater-initiated ATWS.

The primary objectives for Experiment L3-7 were to impose a break flow equal to HPIS flow at an intermediate pressure (~6.9 MPa) during the transient, to isolate the break, to cool the plant to establish cold shutdown, and to analyze the data obtained to investigate associated phenomena such as:

1. 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 a U-tube steam generator?
2. 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?
3. 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?

An annotated depressurization curve for the primary system during Experiment L3-7 is shown Figure 6-1. At 36 s after the break was initiated, the reactor scrammed on a low system pressure signal. At 39 s, the primary coolant pumps had been manually tripped. Pump coastdown was complete by 56.2 s. The first indication of natural circulation occurred at 613 s. Between 1800 and 5974 s, the HPIS was turned off. Starting at 3600 s, operator-controlled steam bleeding (opening the main steam bypass valve early and the main steam valve later in the transient) and steam generator feeding (using both the auxiliary and main feedwater systems) were used to decrease primary system pressure.

Conditions were established in Experiment L3-7 for natural circulation in the primary loop. Measurable natural circulation continued from 61 s into the transient, until after the purification system recirculation started at 18 180 s. Single-phase natural circulation was fully established by 96 s and continued until about 375 s, at which time the upper plenum temperature reached saturation and the transition to two-phase natural circulation began. Two-phase natural circulation was fully established by about 1500 s, when the core temperature differential approached zero, see Figure 6-2. The transition from single-phase to two-phase flow during Experiment L3-7 started at 375 s and ended at 1500 s.

Single-phase natural circulation was reestablished after the break was isolated at 7200 s, as indicated by core differential temperature shown in Figure 6-2. By that time, the fluid in the system had become subcooled. Fluid velocities were lower than during two-phase natural circulation. Fluid velocity in the core, calculated from decay power and temperature differential, was ~0.05 m/s from 7800 to 15 000 s.

During single-phase natural circulation, convective heat transfer dominated. During two-phase natural circulation, the dominant heat transfer mode was condensation of steam within the steam generator. The combination of natural loop circulation and steam generator heat transfer was sufficient to remove decay heat throughout the experiment, including the period of system repressurization beginning at 7915 s shown in Figure 6-1. Measured steam generator inlet-to-outlet and primary-to-secondary temperature differences confirmed that the primary-to-secondary heat transfer rates were high during the entire experiment.

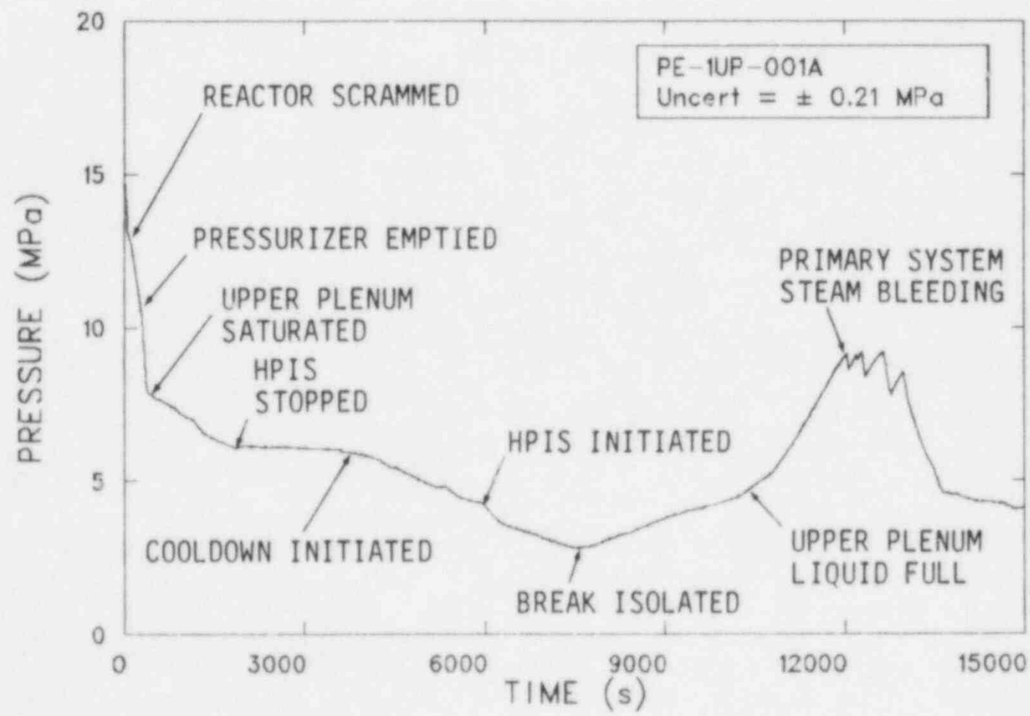

Figure 6-1. Pressure in reactor vessel upper plenum for LOFT Experiment L3-7.

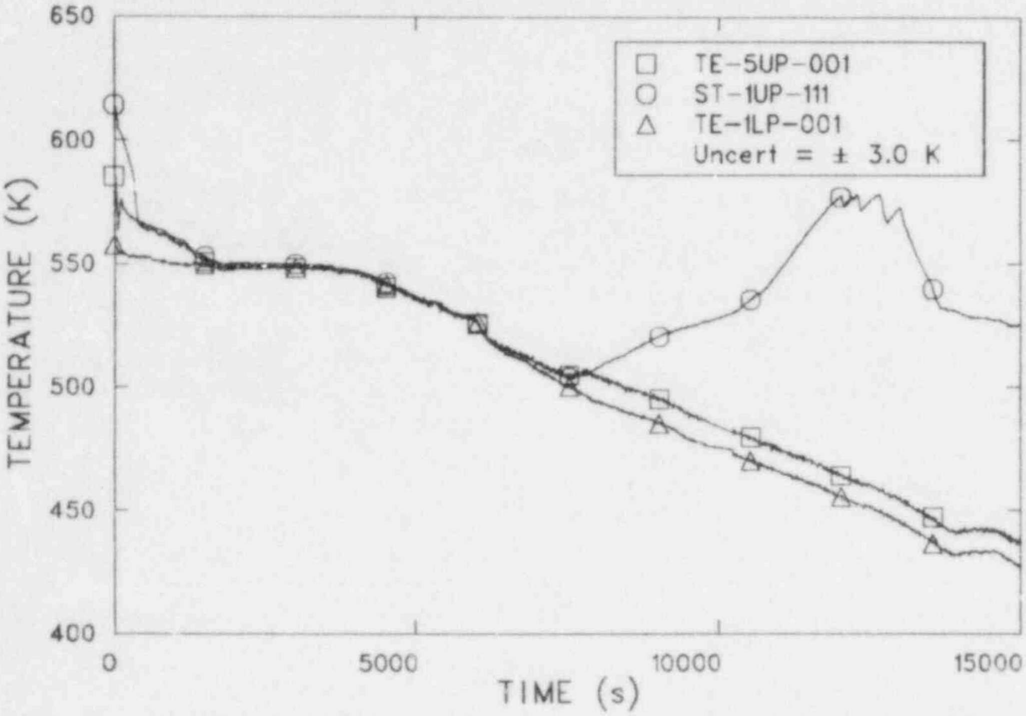

Figure 6-2. Fluid temperature in reactor vessel upper and lower plenums and fluid saturation temperature for LOFT Experiment L3-7.

Postexperiment analyses were performed using the RELAP4/MOD7 and RELAP5/MOD1 computer codes to calculate Experiment L3-7 results.[6-7] Calculations of the intact loop hot leg velocity for Experiment L3-7 are compared with the pulse neutron activation (PNA) data in Figure 6-3. Both codes predicted a positive flow through the hot leg to the steam generator. During single-phase natural circulation, the predicted velocities agree with the PNA data. As two-phase natural circulation was established, both the liquid and vapor velocities predicted by RELAP5 generally exceed the PNA data values.

LOFT Experiment L9-1/L3-3, simulated a multiple failure transient initiated from a loss-of-feedwater accident (LOFA). L3-3 simulated two recovery modes from a LOFA (L9-1). The first recovery mode consisted of turning off the primary coolant pumps and latching open the PORV to depressurize the primary system to the ECCS injection setpoint. The second mode consisted of refilling the steam generator to restore the secondary heat sink and removing decay heat through the secondary. Figure 6-4 illustrates the effectiveness of reestablishing two-phase natural circulation.

Primary system pressure decreased rapidly after reestablishment of the steam generator as a heat sink. Natural circulation started at ~5205 s. Operator-controlled steam generator feed and bleed operations were initiated at 6712 s, and the depressurization rate increased again.

LOFT Experiments L3-7 and L9-1/L3-3 demonstrated that:

1. Single- and two-phase natural circulation are effective and stable modes of decay heat removal in LOFT which contains an inverted U-tube steam generator. The experiments demonstrated that the transition between single- and two-phase natural circulation is stable, and that natural circulation occurs whenever a heat source and heat sink (steam generator) exist with sufficient coolant inventory, provided no flow blockage is present.

2. The results of the postexperiment analyses indicated that both RELAP4 and RELAP5 were able to predict the characteristics of LOCEs and transients during which two-phase natural circulation occurred.

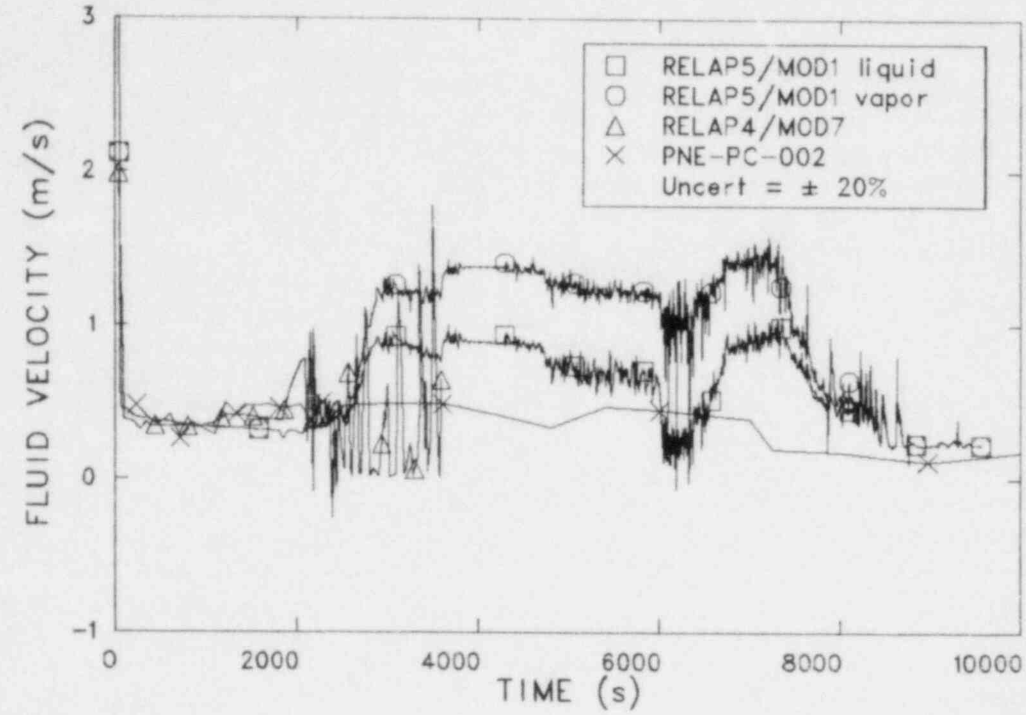

Figure 6-3. Measured and calculated fluid velocity in intact loop hot leg for LOFT Experiment L3-7 (measurements were made using PNA).

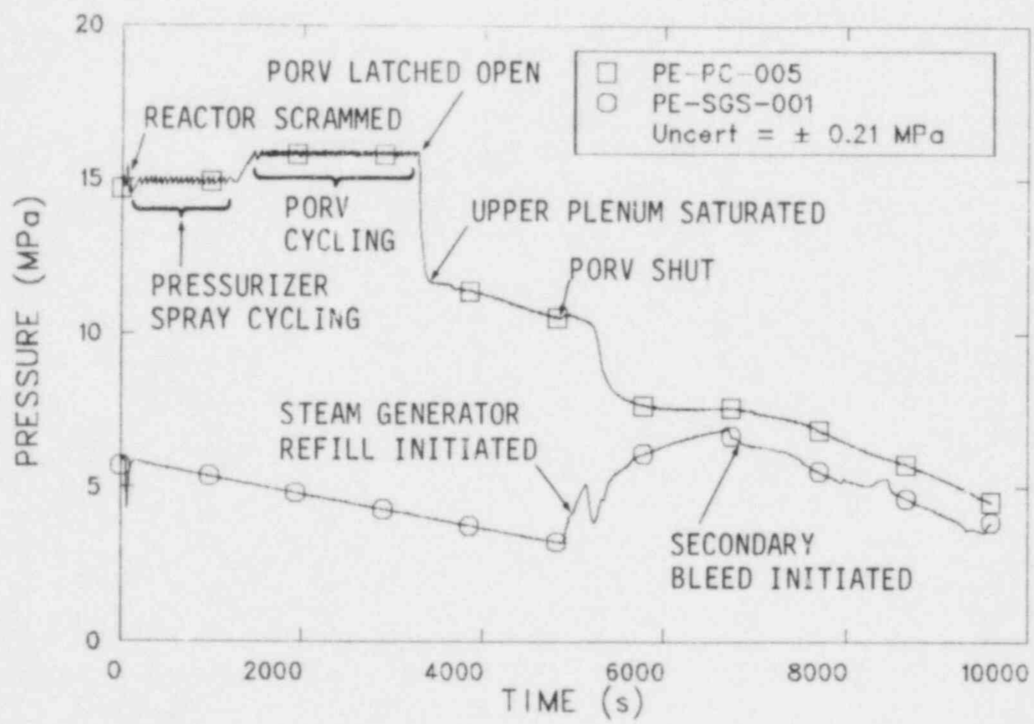

Figure 6-4. Pressure in primary and secondary systems for LOFT Experiment L9-1/L3-3.

6.1.2 Demonstration of the Feed and Bleed Concept. Feed and bleed can be used in the steam generator to maintain it as a heat sink or it can be used in the primary system by bleeding steam and water off through the PORV (for commercial PWRs which have PORVs in the pressurizer) and by feeding water into the primary system with the HPIS to maintain sufficient inventory to cover the core.

LOFT Experiment L3-7 was a good example of the use of steam generator feed and bleed (see References 6-6 and 6-7). By maintaining the steam generator as a heat sink during the small-break LOCE, the system was cooled and depressurized more rapidly. This resulted in less coolant being lost from the system which reduced the likelihood of core uncovery.

The first recovery mode of Experiment L9-1/L3-3[6-8] (also discussed in Section 6.1.1) demonstrated the capability of the PORV to control pressure by cycling and to reduce pressure when it is latched open. Figure 6-4 shows primary system pressure during the time the PORV was cycling and when it was latched open to reduce primary system pressure. Figure 6-5 shows measured and predicted primary system coolant inventory. The slope of the measured mass inventory curve when the PORV was cycling is ~0.1 kg/s and 0.7 kg/s when the PORV was latched open. These slopes represent the average mass flow through the PORV, and are within the scaled HPIS capacity for LOFT which is 0.76 kg/s. Had the HPIS been used while the PORV was cycling or was latched open, the system mass would have been maintained.

The amount of coolant mass bled off through the PORV was more in the preexperiment RELAP5 prediction than was measured during the experiment. The difference was due to higher environmental heat losses from the primary system than assumed.

A recent report[6-9] presented a theoretical analysis of the capabilities and limitations of primary system feed and bleed in PWRs. Data from experiments performed in the Semiscale facility and from RELAP5 computer code calculations were used in this analysis. Figure 6-6 shows the parameters considered for determining the feasibility of primary system feed and bleed operation and indicates the possibility of an operating band. The governing parameters which determine this operating band are decay heat level, HPIS flow rate, and PORV flow rate (and enthalpy flow rate). Except for the core decay heat level, the remaining parameters are functions of primary system pressure.

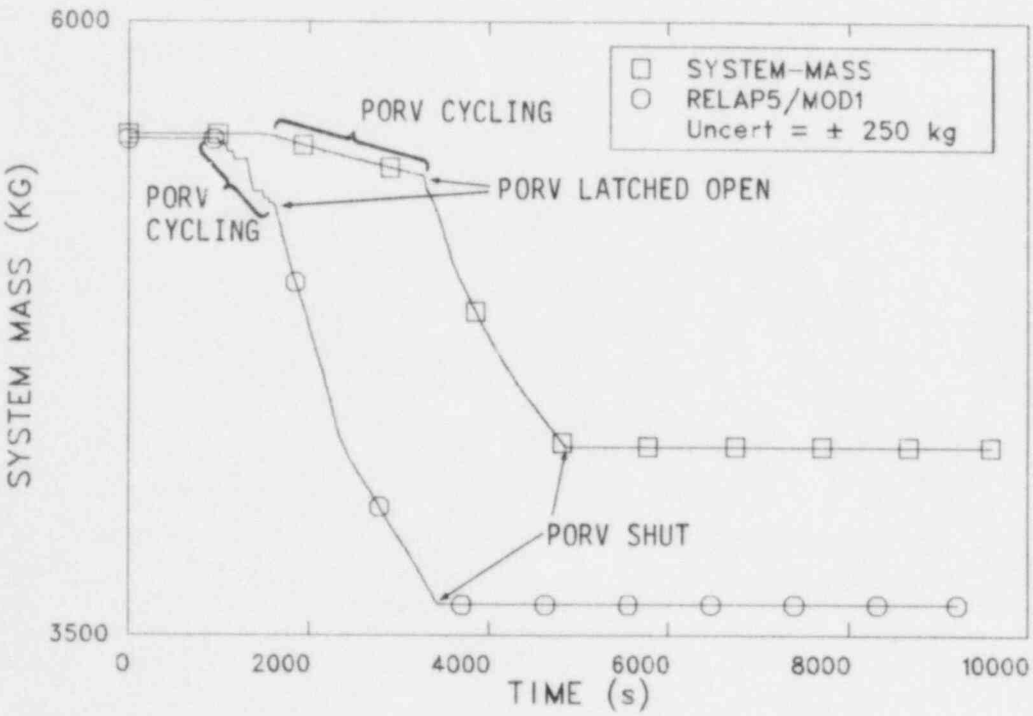

Figure 6-5. Measured and predicted mass inventory in primary system for LOFT Experiment L9-1/L3-3.

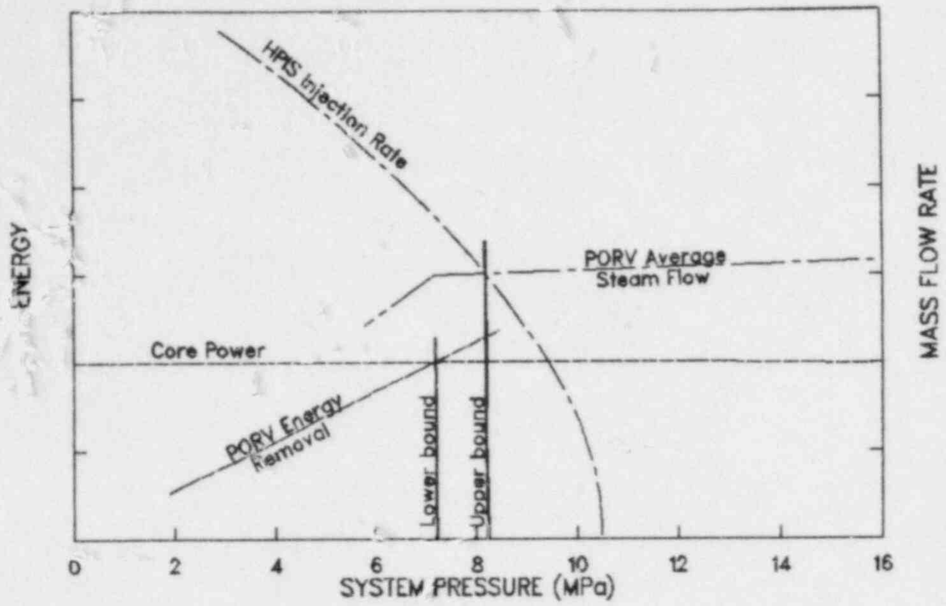

Figure 6-6. Simplified, theoretical operating map for primary feed and bleed.

The lower bound of the operating band represents the minimum pressure at which the PORV can pass enough steam (with the coolant replaced by ambient temperature water) to remove sufficient energy from the system. Steady state operation below this pressure without additional energy removal paths is not possible. Operation at a pressure above the lower bound may be accomplished by cycling the PORV open and closed. The upper pressure bound to the steady state operating band is defined by a balance between the PORV average coolant (steam) removal rate and the HPIS coolant injection rate.

Semiscale Tests S-SR-1 and -2 were conducted in the Semiscale Mod-2A system to evaluate system behavior during primary system feed and bleed operations. The boundary conditions of HPIS injection rate were scaled from Westinghouse plants with either "high head" (injection capacity up to the safety valve setpoint) or the "low head" (pump deadhead at typically 10.3 MPa) HPIS pumps. The PORV discharge capacity was scaled close to the value for a commercial PWR, but was slightly larger.

The results of the Semiscale tests indicated that as long as steam was bled off through the PORV, the HPIS could maintain primary system inventory. However, when the steam bubble in the pressurizer collapsed, and the pressurizer went solid, the mass flow through the PORV exceeded the HPIS capacity.

6.1.3 Operation of the Steam Generator Units as Reflux Condensers, as an Alternative to True Natural Circulation. During LOFT Experiment L3-2, reflux cooling was inferred from temperature measurements made in the steam generator (see References 6-7 and 6-10). Between 500 and 8000 s a negative temperature gradient developed in the primary system coolant between the steam generator inlet and outlet, see Figure 6-7. This negative temperature difference indicated that a mode of cooling, other than natural loop circulation occurred. The other mode was probably reflux cooling. The negative temperature gradient occurs when condensation takes place in the inlet side of the steam generator tubes. The fluid in the steam generator outlet increases in quality which causes the fluid temperature measurement to sense wall radiation heating. This contention is supported by the results of the separate effects tests[6-11] conducted in the PKL facility in Germany. The PKL test results showed that reflux flow occurred when the system was voided sufficiently. Those results strongly suggest that U-tube steam generators of a PWR could operate as reflux condensers to remove shutdown decay heat.

6.2 Station Blackout

Alternating current (ac) electrical power for most PWR safety systems is normally supplied by redundant offsite sources. Should offsite power sources

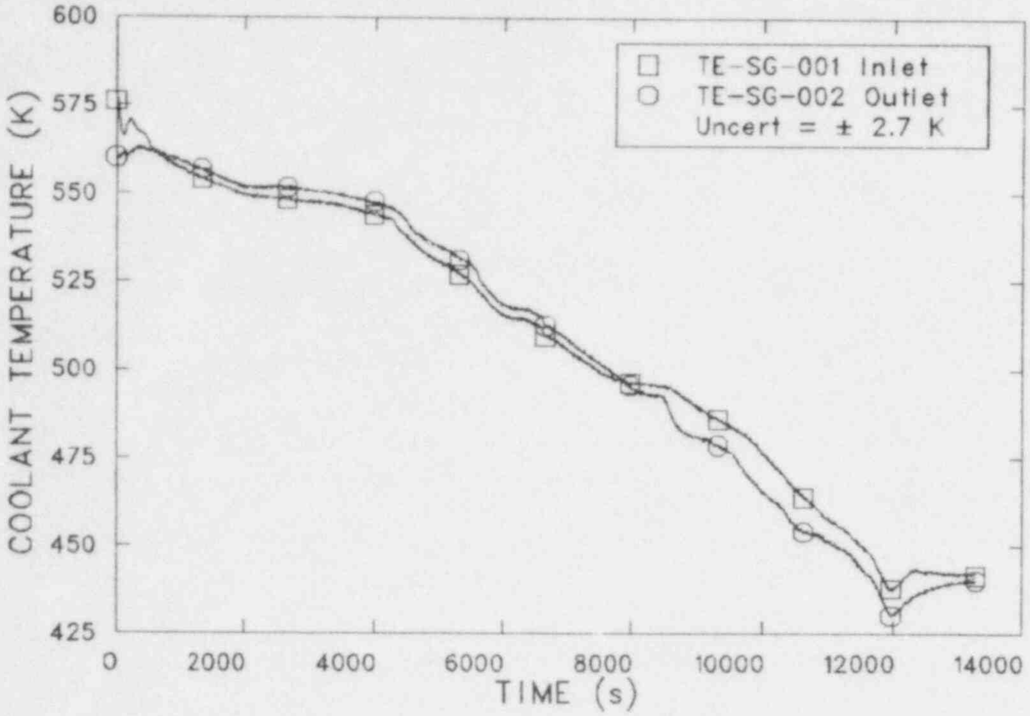

Figure 6-7. Fluid temperature in steam generator inlet and outlet plenum for LOFT Experiment L3-2.

fail, onsite sources (usually redundant diesel generators) are available to power the safety systems. In the event that these onsite sources also fail, most safety functions are lost and the PWR enters a station blackout condition. To reach a station blackout condition requires a loss of offsite power followed by multiple onsite power failures (failure to start and load diesel generators). In the event of a station blackout, the capability to cool the reactor core would be dependent on the availability of systems that do not require ac power supplies and on the ability to restore ac power in a timely manner. The most important system that does not require ac power and that contributes to core cooling is the turbine-driven feedwater pump(s).

Loss of offsite power calculations[6-12] using the RELAP4/MOD7 computer code[6-13] were performed on three commercial PWRs: Zion-1, a Westinghouse plant; Three Mile Island—Unit 2 (TMI-2), a Babcock and Wilcox (B&W) plant; and Calvert Cliffs-1, a Combustion Engineering (CE) plant. Results of the study are presented in Table 6-2 with the following conclusions:

1. A loss of offsite power followed by failure of all diesel generators and steam turbine-driven AFW leads to a core uncovery

2. After loss of offsite power and failure of all diesel generators, a core uncovery can be avoided if the steam turbine-driven AFW pump or a diesel generator are started within a time specific to the plant.

Results from LOFT Experiment L9-4 support the second conclusion that a core uncovery can be avoided if the turbine-driven feedwater pump operates during a station blackout.

LOFT Experiment L9-4 was performed to simulate an ATWS initiate by a loss of offsite power in a PWR. However, the experiment was a station blackout combined with a failure to scram. The experiment is described in detail in Reference 6-14.

Experiment L9-4 was initiated by tripping the primary coolant pumps and the secondary system main feedwater pump, and by closing the main steam control valve. Initial conditions for the LOFT PWR were typical of a commercial PWR. During the transient, the PORV, and pressurizer spray were assumed to be inoperative due to loss of offsite power. Only the SRV was operative. Auxiliary feedwater was started 10 s after transient initiation which could simulate startup of the turbine-driven feedwater in commercial PWRs. The moderator

Table 6-2. Results of calculations with failures of diesel generators and/or turbine-driven auxiliary feedwater, primary leakage at technical specification limit

Plant	No Turbine AFW and No Diesel Generators: Time Steam Generators Dry Out (s)	Time Loop Natural Circulation Ends (s)	Time Core Uncovery Begins (s)	Latest Time at which Turbine AFW may be Started to Avoid Core Uncovery[a] (s)	Latest Time at which Diesel Generator may be Started to Avoid Core Uncovery[b] (s)
Westinghouse Zion-1	2940	4740	5800	4240	4840
B&W TMI-2	712	2270	2715	2115	2355
CE Calvert Cliffs-1	2900	5635	6200	5200	—[c]

a. Assumes full turbine-driven AFW capability is delivered from one pump (Westinghouse and B&W) and two pumps (CE).

b. Assumes one diesel generator is started and used to power one motor-driven AFW pump plus one centrifugal charging pump.

c. Calvert Cliffs plant does not have motor-driven AFW pumps. One of the two turbine-driven AFW pumps must be started or feed and bleed of the primary system initiated to avoid core uncovery.

reactivity feedback was typical of PWR core end-of-life conditions. The experiment was specified to end at 1500 s with the insertion of the control rods.

Predictions for the experiment were made using the RELAP5/MOD1 computer code[6-15] and are reported in Reference 6-16. A preliminary evaluation of the experiment results plus a comparison between predictions and measured data are presented in Reference 6-14.

A comparison among predicted and measured hot and cold leg temperatures is shown in Figure 6-8. The rate of the initial hot leg temperature increase was predicted very well. The cold leg temperature trend was predicted well during the initial 300 s, although there was an offset in magnitude. The heat sink effectiveness of the steam generator degraded at approximately the same time in the prediction as was observed in the experiment. As was shown by the rapid increase in the predicted cold leg temperature, however, the predicted loss of heat sink was much more abrupt than observed in the experiment. Beyond 700 s, both hot and cold leg temperatures were predicted to remain constant, while the measured data showed a slow cooldown. The primary system energy balance calculation in the prediction did not predict this cooldown. A comparison between measured and predicted intact loop hot leg fluid velocities during natural circulation is shown in Figure 6-9.

The difference between predicted and measured reactor power shown in Figure 6-10 was caused by the differences between calculated and measured primary coolant pump behavior which resulted in lower measured flow during the first 350 s (see Figure 6-9). The lower flow caused a slightly higher

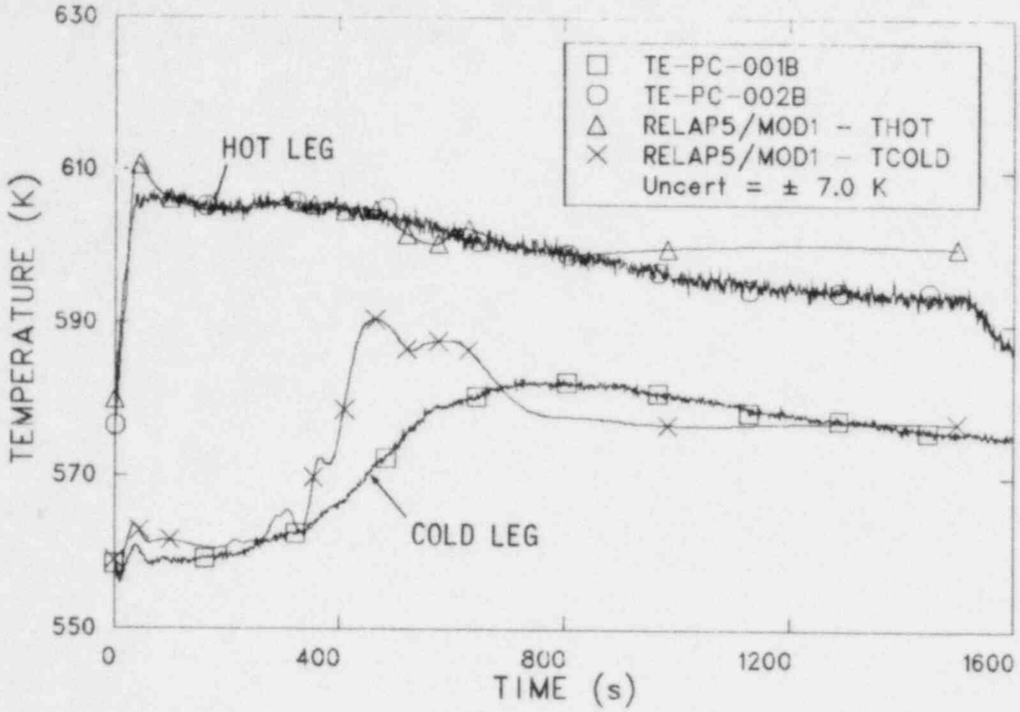

Figure 6-8. Measured and calculated temperature in intact loop hot and cold legs for LOFT Experiment L9-4.

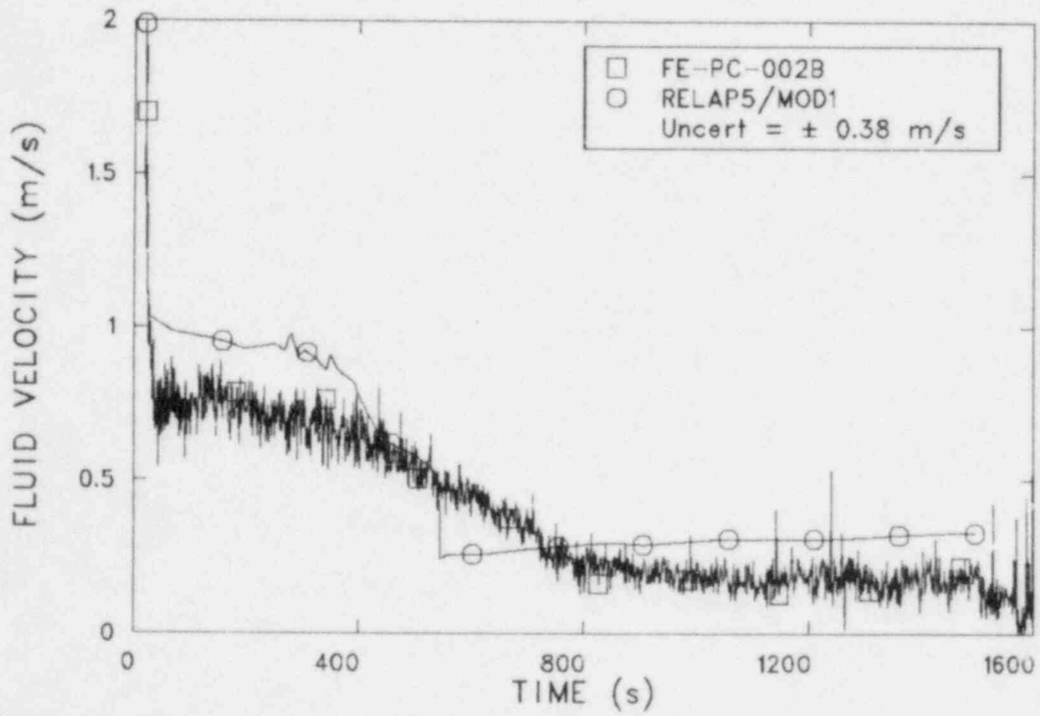

Figure 6-9. Measured and calculated fluid velocity in intact loop hot leg for LOFT Experiment L9-4.

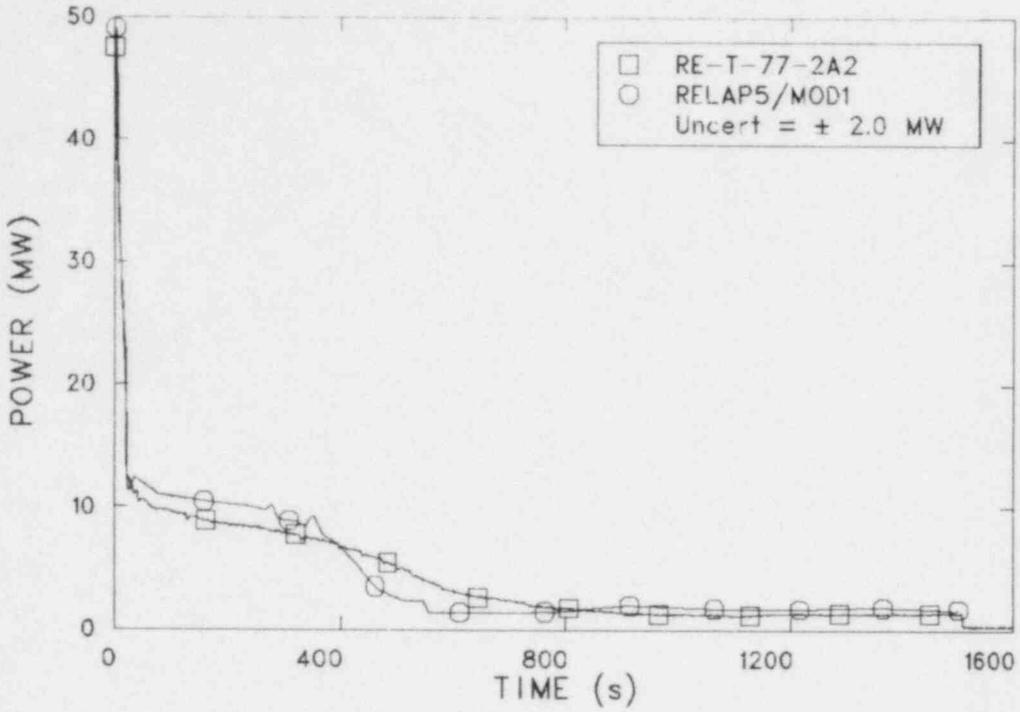

Figure 6-10. Measured and calculated reactor power for LOFT Experiment L9-4.

core coolant temperature, resulting in an increased negative moderator temperature reactivity and subsequent lower power. These differences were anticipated in the experiment prediction sensitivity calculations discussed in Reference 6-16.

The difference between the predicted and measured reactor power after 350 s was caused by the abrupt change in the predicted cold leg temperature. Comparisons of predicted and measured reactor power response as a function of moderator temperature during this time showed that the moderator density feedback was predicted accurately. After 1000 s, both the predicted and measured reactor power stabilized, although at different values.

Comparisons of the predicted and measured pressurizer liquid level and pressure are shown in Figures 6-11 and 6-12, respectively. Two rapid surges into the pressurizer were predicted and measured. The first during the initial heatup following pump trip, and the second when the heat sink degraded. The initial pressure increase to the SRV opening setpoint was predicted very well. The calculated period of cycling, however, was shorter than measured. The second period of SRV cycling began and ended earlier in the calculation because the predicted heat sink degradation was more abrupt than observed in the experiment. The relative smoothness of the measured heat sink degradation was responsible for both the gradual pressurization to the SRV opening setpoint when the degradation began, and the gradual depressurization after the last SRV cycle. Later in the transient, both the predicted pressurizer liquid level and pressure stabilized, rather than decreased, due to an apparent difference between calculated and actual primary system energy balance.

The results of LOFT Experiment L9-4 demonstrated that even after a very severe transient, a station blackout followed by a failure to scram, a single turbine-driven AFW pump could remove decay heat after the SRV controlled the initial pressure surge.

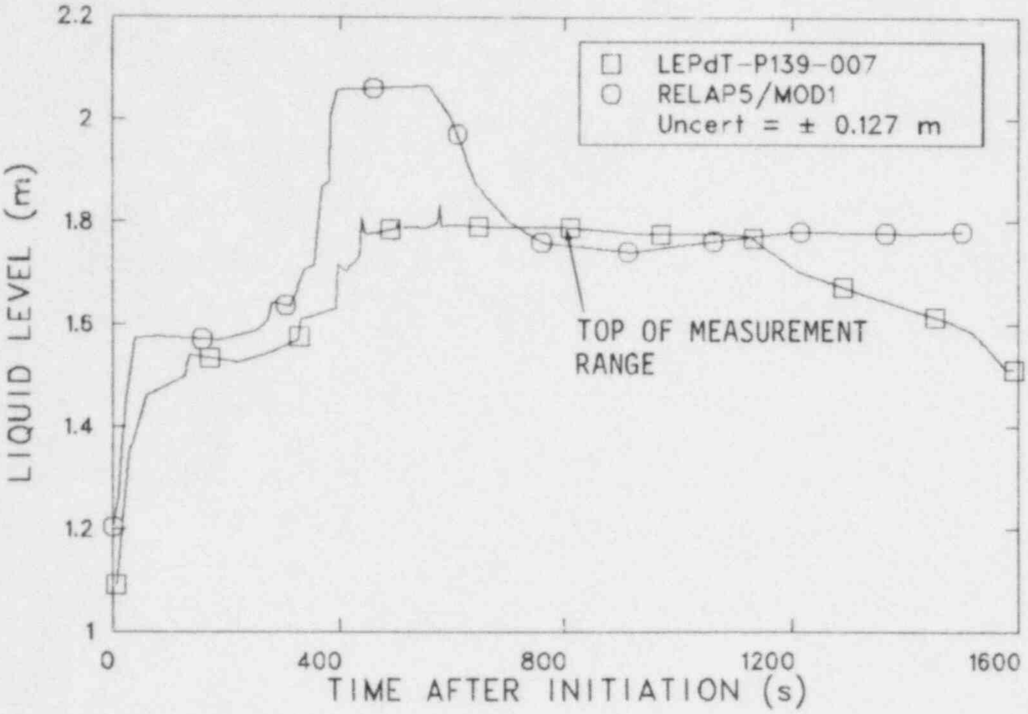

Figure 6-11. Measured and calculated liquid level in pressurizer for LOFT Experiment L9-4.

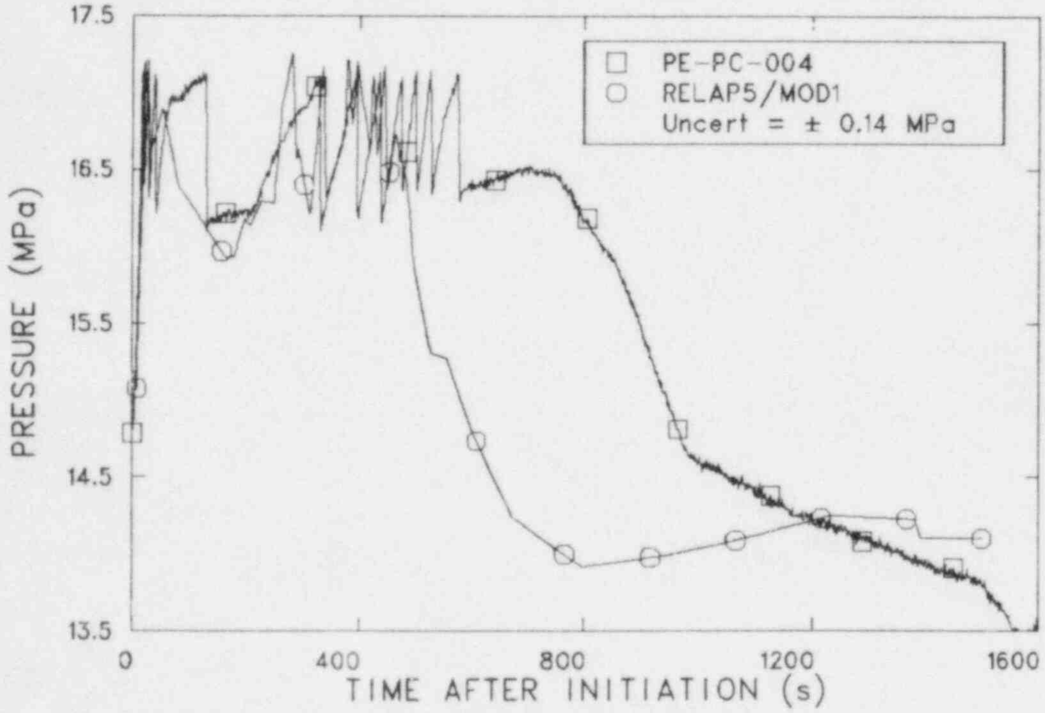

Figure 6-12. Measured and calculated pressure in pressurizer for LOFT Experiment L9-4.

6.3 References

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7. ISSUES RELATED TO THE THREE MILE ISLAND ACCIDENT

Following the accident at Three Mile Island—Unit 2 (TMI-2), the USNRC conducted an intensive investigation into the causes of the accident that identified a number of generic safety issues[7-1] and issued an Action Plan[7-2] aimed at preventing any reoccurrence of the TMI-2 accident at other nuclear plants. Results from a number of LOFT experiments have supplied data which aid in addressing both generic issues and items specific to the NRC TMI Action Plan.

7.1 Generic Issues

The TMI Special Inquiry Group (SIG) considered a number of generic issues related to the accident at TMI. Several of the issues have not been categorized as generic by the NRC because they are all of considerable importance to reactor safety. Four of these issues have been addressed in the LOFT Program:

1. Human-machine interfaces
2. Interruption of ECCS after a LOCA
3. Instrumentation to follow the course of an accident
4. Revised small-break LOCA, methods to show compliance with 10 CFR 50, Appendix K.

A brief discussion of each of these issues is presented in the following subsections.

7.1.1 Human-Machine Interfaces. Investigations following the accident at TMI indicated that, under emergency conditions, the operators were unable to assimilate the amount of data they were suddenly receiving. This posed the problem of how to get better information to an operator without overwhelming him in the process.

As part of the LOFT Augmented Operator Capability (AOC) Program, a basic computer-driven graphics display system was installed for trial use during LOFT experiments. Evaluations based on the experience at LOFT affirmed the potential benefits of computer-based graphics displays for commercial power reactor applications.

The principal results of the LOFT AOC Program were:

1. A computer which generates color cathode ray tube (CRT) displays using LOFT plant data has been installed and has been operated successfully during nine LOFT experiments. This work provides experience on which to base judgments in areas such as computer reliability, redundancy, architecture, and capability.

2. Several display concepts have been developed and used at LOFT. These include conventional displays such as a primary coolant system mimic, a safety parameter display, and an inferred reactor vessel liquid level display. The display development work has provided experience in the use of a powerful new method for reactor data display and analysis.

3. Several studies have been completed. First, a review of computer hardware which is highly reliable even in severe environments has been conducted and potential applications of such hardware identified. Second, a study of display techniques which might be appropriate to the multidimensional problem of a safety parameter display was conducted. Finally, CRT display design guidelines and color and symbology standardization recommendations have been published.

The significance to the NRC of the LOFT AOC Program results lies in two areas. First, the AOC Program experience confirms that computer-based diagnostic graphics have the potential of improving the operational safety of nuclear power plants. Second, AOC experience can help ensure that regulatory actions are well targeted in the area of improved reactor data presentation in the control room and emergency response facilities.

Table 7-1 and the accompanying Figures 7-1 and 7-2 describe the diagnostic graphics developed for LOFT Experiment L3-6/L8-1,[7-3,7-4] and used in subsequent experiments.

7.1.2 Interruption by ECCS After a LOCA. This issue arises in conjunction with the generic issue,

Table 7-1. Diagnostic graphics available for LOFT Experiment L3-6/L8-1

Diagnostic Graphic Display Type	Graphic Description	Recommendations for Application
Saturation temperature deviation display	A bar chart (Figure 7-1) which horizontally shows the deviation from saturation of the temperature of thermocouples distributed axially through the center module and upper plenum. This display provides operators and engineers with the reactor vessel level information derived from the trend of thermocouple readings as referenced to the saturation temperature of the primary coolant.	The use of structural thermocouples in the upper half of this diagnostic graphic is applicable whenever thermocouples within the reactor vessel are available.
PCS display	A mimic diagram (Figure 7-2e) of the primary coolant system (PCS) showing its major components. PCS parameters are displayed digitally at appropriate locations on the diagram.	
SCS display	A mimic diagram of the secondary coolant system (SCS) showing its major components. SCS parameters are displayed digitally at appropriate locations on the diagram.	Basic graphics that should be available in the technical support center (TSC).
ECCS display	A mimic diagram of the emergency core cooling system (ECCS) showing its major components. ECCS parameters are displayed digitally at appropriate locations on the display.	
CORTEMP display	A display of the fuel cladding temperature at all the thermocouple locations on the selected fuel rod. A joystick is used to select the desired fuel rod. The temperatures are displayed as a horizontal cladding temperature bar versus thermocouple elevation.	These graphics support safe conduct of LOCA experiments at LOFT and are not applicable to commercial reactors.
CLADTEMP display	CLADTEMP is composed of two diagrams on one display: a column chart showing the radial temperature profile of core outlet temperatures just above the core, and a bar chart showing the axial fuel cladding temperature distribution in the center fuel module. The temperature range automatically changes when temperatures reach the upper temperature scale limit.	
T-PW display	An operating map (Figure 7-2c) which shows the technical specification limits of primary system temperature based on reactor power. The current operating point is displayed. The trend of the operating point is displayed as a "track" of previous operating points.	Directly applicable to the TSC and as supplements to the minimum requirements of NUREG-0696 for the Safety Parameter Display System.
F-PW display	An operating map (Figure 7-2d) which shows the technical specification limits of primary system flow rate based on reactor power. The current operating point and trend are displayed as in the T-PW display.	

Table 7-1. (continued)

Diagnostic Graphic Display Type	Graphic Description	Recommendations for Application
P-T display	An operating map (Figure 7-2f) showing the current operating point and trend of temperature plotted versus presure. The map shows the normal operating region, saturation line, and steam region and provides the operator with a real-time presentation of the approach to saturation during a LOCA.	Directly applicable to the TSC and as supplements to the minimum requirements of NUREG-0696 for the Safety Parameter Display System.
MPT display	A real-time active operating map (Figure 7-2g) showing the pressure-temperature relationships which must be maintained during plant heatup and pressurization to protect the nuclear steam supply system (NSSS) from over-pressurization and the primary coolant pumps from cavitation. Current operating point and trend are displayed.	
HU display	A plot of primary coolant system temperature versus time providing a plot of current value and trend of temperature change to technical specification cooldown rate limits.	
HISTORY display	A parameter-versus-time plot (trend display) (Figure 7-2b) which provides the capability of simultaneously plotting up to 7 parameters against time. The operator chooses the parameter to be plotted and the parameter scale to be used with a default time scale of 30 minutes.	
System coolant inventory (SCI)	An active functional diagram (Figure 7-2h) of the ECCS providing information to the operator relative to flows, location, and reservoir of decay-heat-removal cooling water following a LOCA.	A diagnositc graph useful to both the Main Control Room (MCR) and TSC crews.
Safety parameter display (SPD)	A deviation bar chart (Figure 7-2a) of the LOFT safety parameter set which provides the reactor operator with an indication of the state-of-the-plant heat transfer cycles. The variables are compared in real-time to their expected value normalized to reactor power and provide the reactor operator with rapid indication of degrading plant conditions.	An example of information required by NUREG-0696[7-3] for SPDs
Liquid level transducer (LLT) display	A deviation bar chart showing the output of the conductivity liquid level probes which detect, by changes in conductivity, the presence of water at various levels in the reactor core region and upper plenum. This display complements the information presented by the SPD display.	A reactor vessel water level display graphic, applicable where point detectors are used, e.g., heated thermocouples.

Table 7-1. (continued)

Diagnostic Graphic Display Type	Graphic Description	Recommendations for Application
Shutdown safety parameter (SSP) display	A mimic display of the reactor vessel with vital temperature, pressure, and flow information provided for the operator. Te nperature indication become red if they exceed the saturation temperature for the existing plant pressure. The temperature displayed reflect heat removal by an ECCS injection as appropriate. A feature exists, referred to as ENG display, which provides the capability to recall data and play back stored data on any of the various displayes. This is used to enhance the operator's capability to analyze the information on different displays by enabling replay of the data and observance on the desired displays.	A diagnostic graph useful to both main control room and (MCR) TSC crews.

"Loss of Offsite Power Subsequent to Manual Safety Injection Reset Following a LOCA," (from Refe ence 7-5). The issue is also discussed in the TMI Report to the Commissioners,[7-1] because the consequences are potentially similar to the TMI-2 accident in that a core uncovery could occur with TMI-2 consequences.

The consequences of an interruption of ECCS following a LOCA will depend upon the size of the break, when the interruption of ECCS occurs, how long it lasts, what systems are available (operationally) to arrest the progress of the accident, and what operator actions are taken.

The issue has not been directly addressed in the LOFT Program; however, during LOFT Experiments L3-5/L3-5A and L2-5, the ECCS was interrupted and the results provided data to assess system response following an interruption of ECCS injection.

During small-break LOCE L3-5/L3-5A,[7-6] the break was isolated at 2309 s and the HPIS injection was interrupted from 2311 s until 5880 s. The reactor vessel liquid level experienced two minimums: the first during L3-5 at 2125 s, when the liquid level was 0.6 $\pm$ 0.4 m above the core and the second during L3-5A at 5880 s, when the liquid dropped to 0.31 $\pm$ 0.15 m above the core. After the ECCS was turned off, the secondary coolant system auxiliary feedwater pumps were tripped to allow the primary and secondary systems to equilibrate. Next the steam generator was reestablished as a heat sink (secondary feed and bleed) to evaluate the use of steam generator feed and bleed as a plant recovery technique.

A comparison of pressures in the primary and secondary systems for Experiment L3-5/L3-5A is shown in Figure 7-3. As soon as the break was isolated, and the HPIS shut off, the primary system quickly heated up and repressurized until the primary system pressure equalled the secondary system pressure. Decay heat then heated and pressurized both systems until secondary system feed and bleed was started.

If secondary system feed and bleed had not been used to recover the plant, both systems would have gradually repressurized. First the steam generators would have boiled dry, then the primary system would have heated up and repressurized to the PORV or SRV open setpoint, and eventually the primary system coolant would have boiled off and uncovered the core.

In summary, results from Experiment L3-5/L3-5A showed that:

1. If the ECCS is interrupted relatively late in a transient, the system will heat up

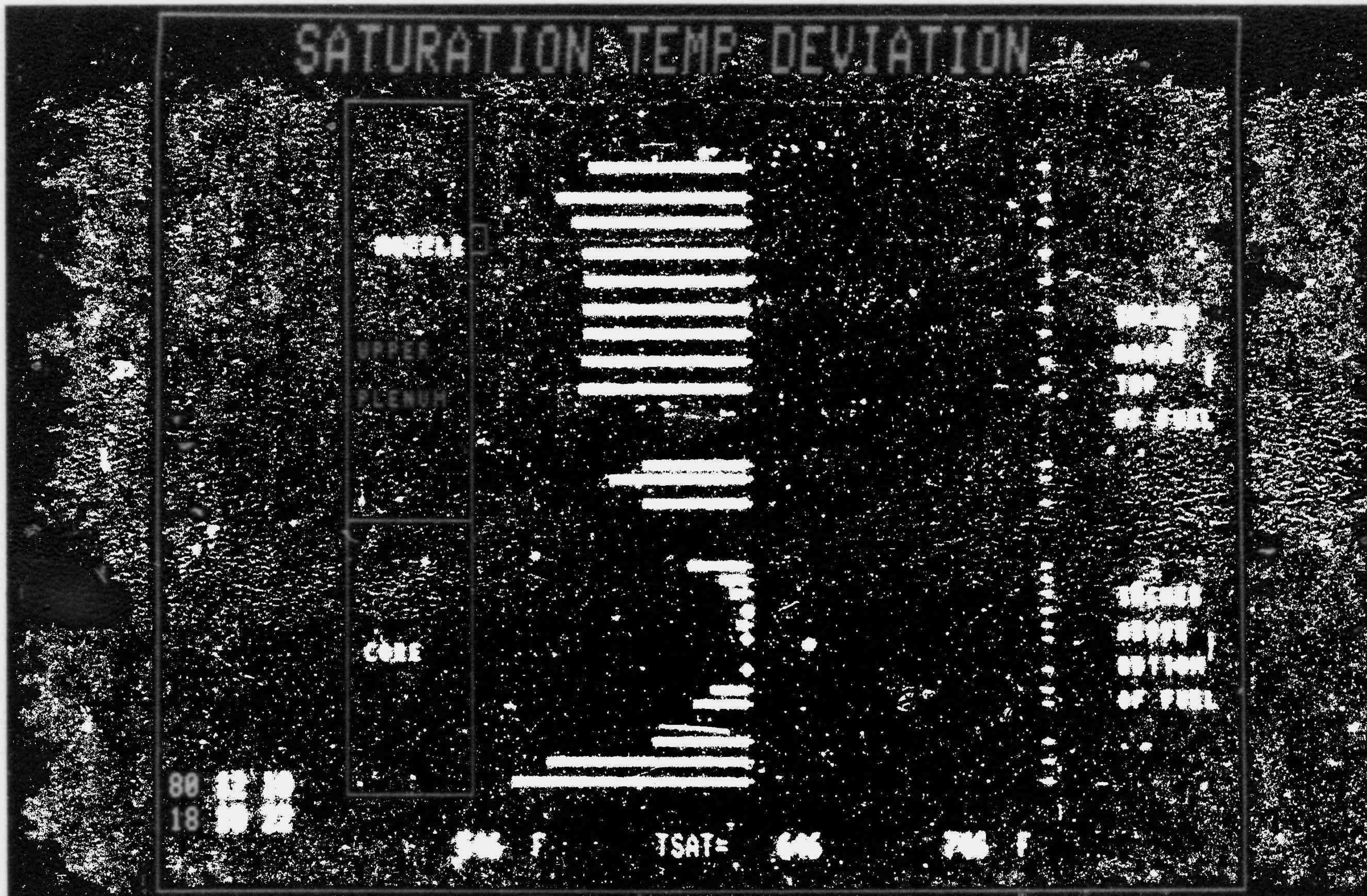

Figure 7-1. Example of display showing saturation temperature deviation.

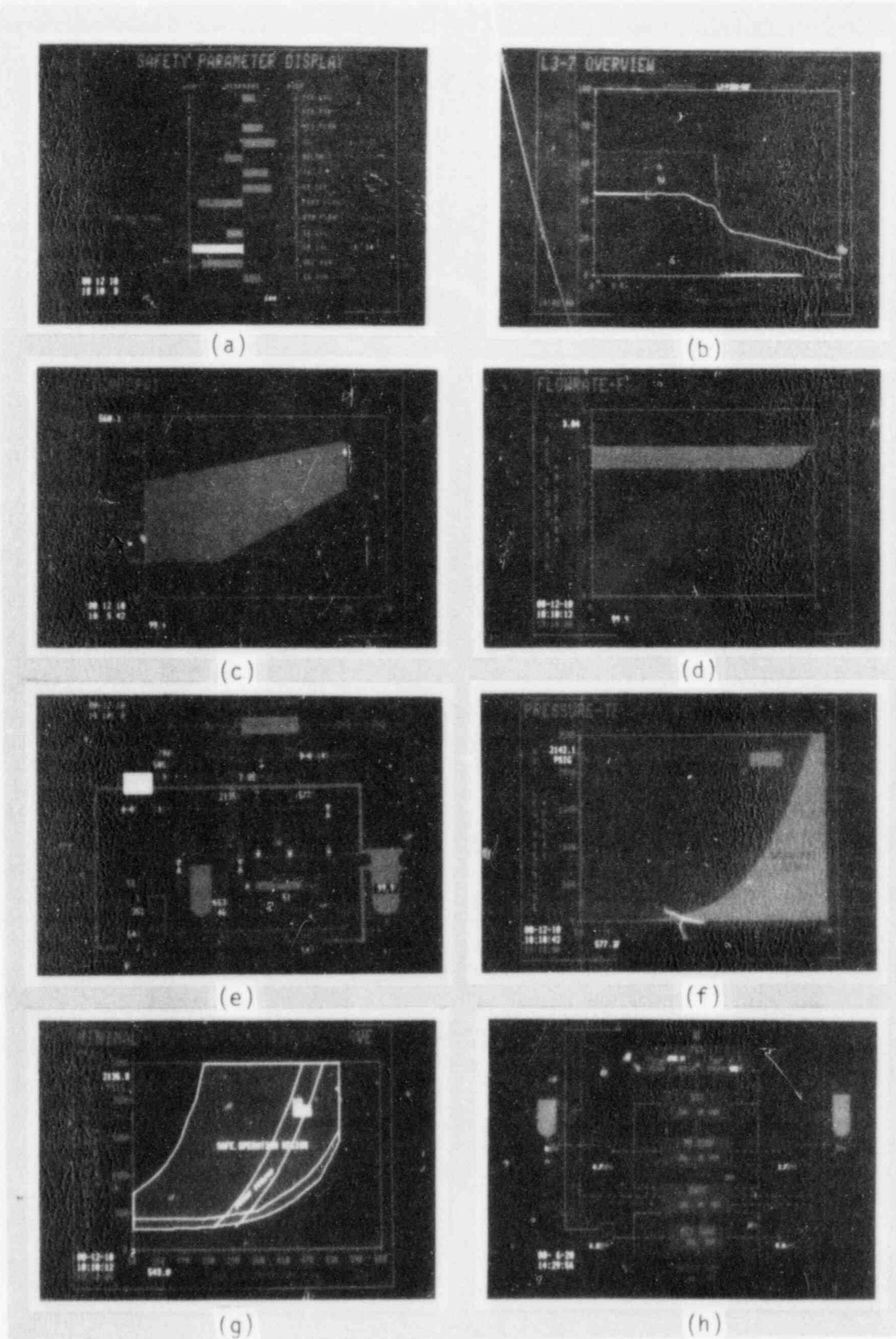

Figure 7-2. Example of diagnostic graphics display from LOFT human-machine interface program.

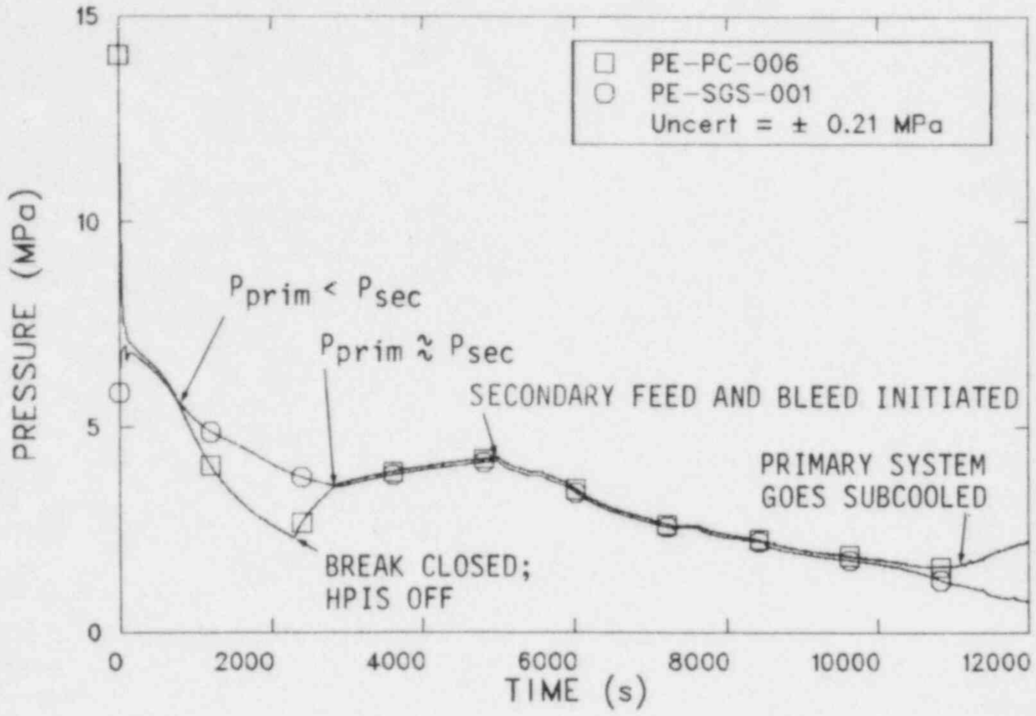

Figure 7-3. Pressure in primary system intact loop and steam generator for LOFT Experiment L3-5/L3-5A.

relatively slowly which will allow the operator time to take action.

2. Other systems (such as the stream generator) can be used to arrest the progress of a transient or at least delay the consequences until ECCS injection can be restarted.

The second LOFT experiment with delayed ECC was Experiment L2-5,[7-7] which simulated a double-ended 200% cold leg break with a simultaneous loss of site power and failure, by shearing, of a main coolant pump shaft in a commercial PWR. The initial core reflood and quench filled the vessel to the nozzle elevation, the ECCS was interrupted (LPIS injection was terminated at 107 s and HPIS was terminated at 144 s when the operators determined that the core was quenched and cooled), and the liquid in the core boiled-off rapidly and uncovered the entire core. HPIS was reinitiated at 274 s, and LPIS was reinitiated at 347 s to prevent the core from heating up excessively. Cladding temperatures at three core elevations are shown in Figure 7-4. The figure shows how quickly the core was uncovered. The core uncovery that occurred after termination of ECCS injection was not detected at the time it occurred by the upper plenum TCs which were being used to monitor liquid level in the core. The core uncovery became apparent when the cladding TCs indicated the core heatup (see Figure 8-18).

Experiment L2-5 demonstrated that if the ECCS is interrupted early in a large-break LOCA, a core uncovery can occur very rapidly, and ECCS flow must be reinitiated quickly to prevent core damage.

7.1.3 Instrumentation to Follow the Course of an Accident. The issue of needing instrumentation to follow the course of an accident was identified by the Technical Steering Activities Committee as Generic Task A-34, "Instrumentation for Monitoring Radiation and Process Variables During Accidents," and resolved with the publication of Regulatory Guide 1.97.[7-8] The purpos uch instrumentation is to ensure that appropriate parameters are monitored during an accident so that operators will have sufficient information available to mitigate its consequences.

The LOFT instrumentation in the primary coolant loops, reactor vessel, and steam generator were examined in light of Regulatory Guide 1.97 to determine the degree to which LOFT complies with current regulations, and where LOFT results indicate that additional instrumentation would be helpful in following the course of an accident.

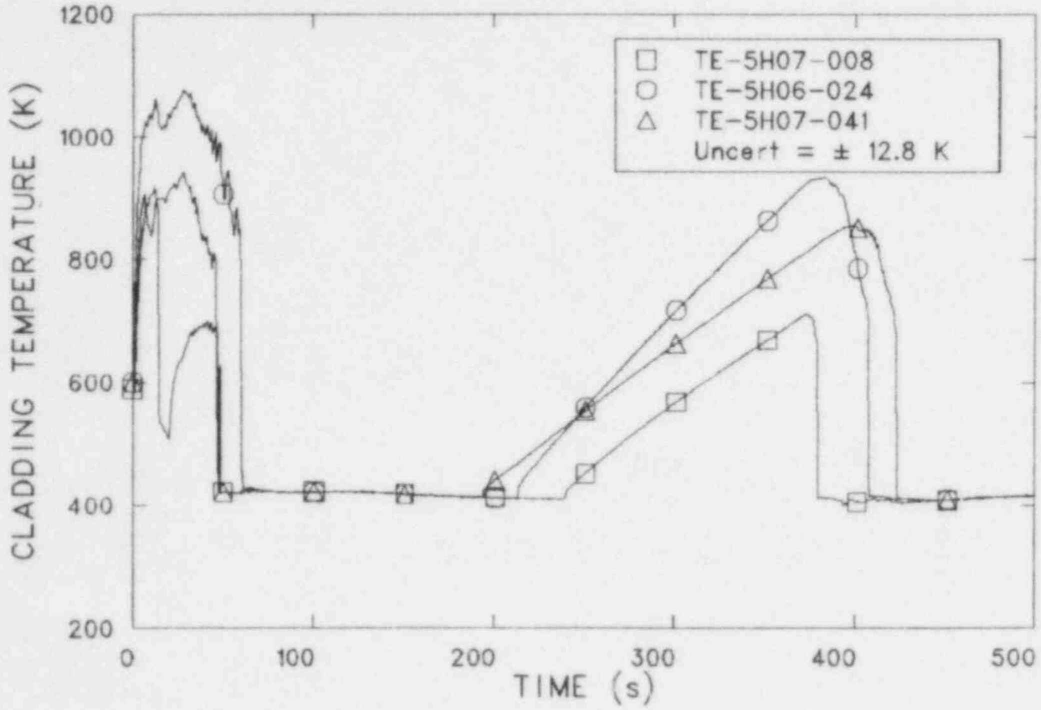

Figure 7-4. Fuel cladding temperature at three elevations in the core during second core heatup for LOFT Experiment L2-5.

LOFT's process instruments are representative of commercial PWR instruments. The process instruments in the LOFT primary and secondary coolant systems meet or exceed the requirements established by Regulatory Guide 1.97 (see Table 7-2), with the following exceptions: Regulatory Guide 1.97 requires slightly greater ranges for core exit, hot leg, and cold leg temperatures.

Instrument comparisons were made using data from small breaks, large breaks, and anticipated transients. Specifically, data were examined from the following experiments:

1. Small breaks simulating 1- and 4-in. diameter breaks in commercial PWRs, including two experiments that addressed the pumps on/off issue
2. Large (200% double-ended) cold leg breaks
3. Loss-of-steam load, loss of reactor coolant system flow, excessive load increase, and loss-of-feedwater incidents.

For all the LOFT experiments analyzed, both process and experimental instruments have adequate ranges to cover the extreme conditions that occurred at their locations. For every narrow-range process instrument, there was a wide-range process instrument in the same location that sufficiently covered off-normal conditions. The evaluation is discussed, by measurement types, as follows.

7.1.3.1 Temperature Measurements in Loop Piping. Process and experimental temperature sensors in the hot and cold legs were compared. The (process) resistance thermometer detectors (RTDs) in the thermal wells responded slower than nearby (experimental) thermocouples. In response to sharp temperature change, the delay times for the RTDs with respect to the thermocouples were:

1. No more than 5 s in the hot leg for either large breaks, small breaks, or anticipated transients
2. Approximately 5 s and no more than 10 s in the cold leg for large breaks
3. No more than 5 s in the cold leg for anticipated transients.

During an accident, rapid temperature changes can occur. Typically, the RTDs lagged the thermocouples. The advantage of the faster response time of a thermocouple in direct contact with the fluid is outweighed by the disadvantage of the potential

Table 7-2. Range comparison of LOFT experimental and process instruments with corresponding instruments required by Regulatory Guide 1.97

Location and Measurement Type	LOFT Experimental		LOFT Process		Regulatory Guide 1.97 Range	Comments
	Measurement	Range	Measurement	Range		
Primary Reactor Coolant Loop						
Temperature						
Hot leg	TE-PC-2A TE-PC-2B TE-PC-2C	300 - 2300°F 300 - 2300°F 300 - 2300°F	TT-P139-28-1 TT-P139-32-1 TT-P139-33-1 TT-P139-34-1 TE-P139-32 TE-P139-33 TE-P139-34	50 - 650°F 50 - 650°F 50 - 650°F 50 - 650°F 500 - 650°F 500 - 650°F 500 - 650°F	50 - 750°F	—
Steam generator inlet	TE-SG-1	300 - 2300°F	None	—	None	—
Steam generator outlet	TE-SG-2 TE-PC-3B1 TE-PC-3B2 TE-PC-3B3	300 - 2300°F 300 - 2300°F 300 - 2300°F 300 - 2300°F	None	—	None	—
Cold leg down-stream of pump	TE-PC-1A TE-PC-1B TE-PC-1C TE-PC-4 TE-PC-5 TE-PC-6 TE-PC-7 TE-PC-8 TE-PC-9 TE-PC-10 TE-PC-11	300 - 2300°F 300 - 2300°F 300 - 2300°F 32 - 2300°F 32 - 2300°F 32 - 2300°F 32 - 2300°F 32 - 2300°F 32 - 2300°F 32 - 2300°F 32 - 2300°F	TT-P139-28-2 TT-P139-29	500 - 650°F 50 - 650°F	50 - 750°F	—
Pressure						
Hot leg	PE-PC-2	0 - 3000 psig	PT-P139-2 PT-P139-3 PT-P139-4	0 - 3000 psig 0 - 3000 psig 0 - 3000 psig	0 - 3000 psig (0 - 4000 psig for CE plants)	No location is specified for coolant pressure in Regulatory Guide 1.97.
Pressurizer	PE-PC-4	0 - 3000 psig	PT-P139-5	1500 - 2500 psig	—	—
Steam generator outlet	PE-PC-5 PE-PC-6	0 - 2500 psig 0 - 2500 psig	—	—	—	—
Cold leg	PE-PC-1	0 - 3000 psig	—	—	—	—
Mass flow rate	(Refer to momentum flux and velocity)	—	FT-P139-27-1 FT-P139-27-2 FT-P139-27-3	0 - 5 x 10^6 lbm/hr 0 - 5 x 10^6 lbm/hr 0 - 5 x 10^6 lbm/hr	None	LOFT's ranges are equivalent to 0 - 130% design flow.

Table 7-2. (continued)

Location and Measurement Type	LOFT Experimental Measurement	LOFT Experimental Range	LOFT Process Measurement	LOFT Process Range	Regulatory Guide 1.97 Range	Comments
Primary Reactor Coolant Loop (continued)						
Momentum flux						
Hot leg	ME-PC-2A ME-PC-2B ME-PC-2C	0.8 - 14 Klbm/ft-s^2 0.8 - 14 kblm/ft-s^2 0.8 - 14 klbm/ft-s^2	None	—	None	—
Cold leg	ME-PC-1A ME-PC-1B ME-PC-1C	2 - 50 klbm/ft-s^2 2 - 50 klbm/ft-s^2 2 - 50 klbm/ft-s^2	—	—	—	—
Velocity						
Hot leg	FE-PC-2A FE-PC-2B FE-PC-2C	1.5 - 30 ft/s 1.5 - 30 ft/s 1.5 - 30 ft/s	—	—	—	For LOFT large-break experiments, the instrument range was 2 - 50 ft/s.
Cold leg	FE-PC-1A FE-PC-1B FE-PC-1C	2 - 50 ft/s 2 - 50 ft/s 2 - 50 ft/s	—	—	—	Instruments were used for LOFT large-break experiments only.
Density						
Hot leg	DE-PC-2A DE-PC-2B DE-PC-2C	0.5 - 62.4 lbm/ft^3 0.5 - 62.4 lbm/ft^3 0.5 - 62.4 lbm/ft^3	None	—	None	—
Steam generator outlet	DE-PC-3A DE-PC-3B DE-PC-3C	0.5 - 62.4 lbm/ft^3 0.5 - 62.4 lbm/ft^3 0.5 - 62.4 lbm/ft^3	—	—	—	—
Cold leg	DE-PC-1A DE-PC-1B DE-PC-1C	0.5 - 62.4 lbm/ft^3 0.5 - 62.4 lbm/ft^3 0.5 - 62.4 lbm/ft^3	—	—	—	—
Subcooling	SC-5UP-102	200°F subcooling to 35°F superheat (core exit)	None	—	200°F subcooling to 35°F superheat	Continuous display (hot leg subcooling is available following an experiment using LOFT's experimental data).
Pump parameters						
Pump speed	RPE-PC-1 RPE-PC-2	0 - 4500 rpm 0 - 4500 rpm	—	—	—	—

Table 7-2. (continued)

Location and Measurement Type	LOFT Experimental Measurement	LOFT Experimental Range	LOFT Process Measurement	LOFT Process Range	Regulatory Guide 1.97 Range	Comments
Primary Reactor Coolant Loop (continued)						
Pump parameters (continued)						
Pump frequency	—	—	PCP-1-F PCP-2-F	0 - 75 Hz 0 - 75 Hz	—	—
Pump current	—	—	PCP-1-IRMS PCP-2-IRMS PCP-1-IAC PCP-2-IAC	0 - 1000 A 0 - 1000 A 0 - 1000 A 0 - 1000 A	—	No range was specified in Regulatory Guide 1.97.
Pump power	—	—	PCP-1-P PCP-2-P	0 - 1000 kW 0 - 1000 kW	—	—
Pump voltage	—	—	PCP-1-VRMS PCP-2-VRMS PCP-1-VAC PCP-2-VAC	0 - 600 V 0 - 600 V 0 - 600 V 0 - 600 V	—	—
Pump head	PdE-PC-1	±100 psid	None	—	None	—
Reactor Vessel						
Temperature						
Core exit	300 - 2300°F	—	None	—	200 - 2300°F (200 - 1650°F for operating plants)	There are 43 LOFT experimental measurement transducers.
Core inlet	—	100 - 1300°F	—	—	—	17 thermocouples.
Downcomer	—	0 - 2300°F	—	—	—	28 thermocouples.
Cladding	—	300 - 2300°F 300 - 2800°F	—	—	—	175 thermocouples. 10 thermocouples.
Guide tube	—	300 - 2300°F	—	—	—	11 thermocouples.
Pressure						
Core exit	PE-1UP-1A PE-1UP-1A1	0 - 3000 psig 0 - 3000 psig	None	—	None	—
Downcomer	PE-1ST-1A PE-1ST-1B PE-1ST-3A PE-1ST-3B PE-2ST-1A PE-2ST-1B	0 - 3000 psig 0 - 200 psig 0 - 3000 psig 0 - 200 psig 0 - 3000 psig 0 - 200 psig	—	—	—	—

Table 7-2. (continued)

Location and Measurement Type	LOFT Experimental Measurement	LOFT Experimental Range	LOFT Process Measurement	LOFT Process Range	Regulatory Guide 1.97 Range	Comments
Reactor Vessel (continued)						
Momentum flux						
Core exit	ME-1UP-1 ME-3UP-1 ME-5UP-1	0.2 - 3.5 klbm/ft-s^2 0.2 - 3.5 klbm/ft-s^2 0.2 - 3.5 klbm/ft-s^2	None	—	None	—
Down-comer	ME-1ST-1 ME-2ST-1	0.2 - 3.5 klbm/ft-s^2 0.2 - 3.5 klbm/ft-s^2	—	—	—	—
Velocity						
Core exit	FE-5UP-1	1.5 - 30 ft/s	None	—	None	—
Liquid level	LE-1ST	8 in. above reactor vessel bottom to 85 in. above the top of the core.	—	—	Bottom of core to top of vessel	In LOFT, the upper measurement is at the reactor vessel inlet nozzles.
	LE-1F10 LE-3F10 LE-5E11	Bottom to top of core	—	—	—	In LOFT, the measurement is 1 in. above core bottom to 13 in. above core top.
	LE-3UP	36 in. to 92 in. above the top of the core.	—	—	—	—
Steam Generator Secondary						
Temperature	TE-SG-3	0 - 600°F	None	—	None	—
Pressure	PE-SGS-1	atm to 1000 psia	PT-P4-10A	0 - 1200 psig	atm to 20% above the lowest safety valve setting	—

for leakage introduced by using an intrusive instrument. Therefore, no recommendation is made to exchange the current thermal well temperature measurements for intrusive temperature measurements.

7.1.3.2 Pressure Measurements. For most experiments examined, all pressure transducers in the primary coolant system, regardless of location, read within the uncertainty band of any one transducer with two exceptions: (a) in the large-break experiments, during subcooled blowdown, a maximum pressure difference between the hot and cold legs of ~1 MPa was observed for 0.2 s and, (b) in the small-break experiment that simulated a stuck open pressurizer PORV, pressurizer pressure dropped below primary coolant system pressure for ~40 s while fluid flowed into the pressurizer. In both cases, the pressure deviations were small and of short duration and thus, had no bearing on operator actions. Because pressures throughout the system

were almost identical, no additional pressure measurements are needed.

7.1.3.3 Liquid Level Measurements in Steam Generator and Pressurizer. LOFT used differential pressure transducers to measure liquid level in the steam generator and the pressurizer. Transducers with internal legs and transducers with external legs were used. Difficulties was encountered with each type. During transients, pressure and temperature continuously changed. Thus, density in the active leg changed. This made the level readings inaccurate. With extreme temperature and pressure changes, the liquid in the reference leg reached saturation and flashed, which caused the liquid level in the reference leg to change. This produced gross errors in the magnitude of the level reading. However, the time that abrupt changes occur and the direction of the changes could be obtained from the level readings.

No additional instruments are recommended; however, for differential pressure liquid levels currently installed, density compensation is recommended along with some means of preventing the liquid in the legs from flashing.

7.1.3.4 Flow Measurements. The process flow measurements are only valid during single-phase liquid flow and are not capable of indicating flow direction. Experimental flow instruments in LOFT provide additional information during two-phase flow and single-phase steam flow, and can indicate flow direction. The principal flow measuring instruments in LOFT consist of drag disks or screens (measuring momentum flux) and turbine transducers (measuring velocity).

An important type of flow that LOFT experimental flow measurements did not consistently measure was natural circulation. The difference in temperature measurements across the core and/or steam generator indicated natural circulation as long as the system fluid remained subcooled. However, once saturation pressure was reached in the system, neither of the temperature differences adequately indicated natural circulation conditions. Therefore, there could be extended periods of time during certain accidents when the operator has no knowledge of the presence or absence of natural circulation flow if only temperature difference methods are relied upon.

Commercial PWR operators need to know flow data during transient conditions; however, these measurements are impossible with current technology. The drag disks and turbine meters used in LOFT have short lifetimes in the primary coolant system even during normal operating conditions.

7.1.3.5 Degraded Core Cooling Measurements. During LOFT transients, the first indication that degraded core cooling was imminent was supplied by the primary coolant loop instruments. The most direct indication of loop coolant voiding was the decrease in density recorded by the densitometers in the hot and cold legs. When the primary coolant pumps were operating, the differential pressure across the pumps also provided an indication of liquid inventory because pump differential pressure degraded as the system voided as shown in Figure 7-5, which shows differential pressure across the pump and cold leg density for LOFT Experiment L3-6, a transient in which the pumps were operating. In transients in which the primary coolant pumps were not operating, the LOFT reactor vessel liquid level measurements (consisting of conductivity probes) indicated liquid level within the vessel. However, conductivity probes can give false readings when the pumps are running. During LOFT Experiment L3-6, a two-phase mixture, rather than definable liquid level, existed in the reactor vessel. LOFT reactor vessel liquid level measurements indicated that the vessel was full of liquid when in actuality, as soon as the pumps were turned off, the liquid level collapsed to below the bottom of the core. During Experiment L3-6, primary coolant loop density measurements provided a more consistent measure of system mass depletion than the reactor vessel liquid level measurements.

Also, during Experiment L3-6, once the loops were totally voided, both process and experimental temperature sensors in the loops experienced hot wall effects. That is, thermal radiation from the hot pipe walls caused the thermocouples to falsely register superheat conditions. As the liquid level dropped into the reactor vessel, the upper plenum thermocouples and the core exit thermocouples, respectively, began to experience hot wall effects. When the liquid level dropped into the core, the best indication of degraded core cooling was the dramatic temperature rise sensed by the fuel cladding thermocouples. The guide tube thermocouples also sensed a temperature increase, but experienced time delays and diminished magnitudes when compared to the cladding thermocouples. The subcooled meter and the hot leg, cold leg, and core exit thermocouples, because they are farther from the core

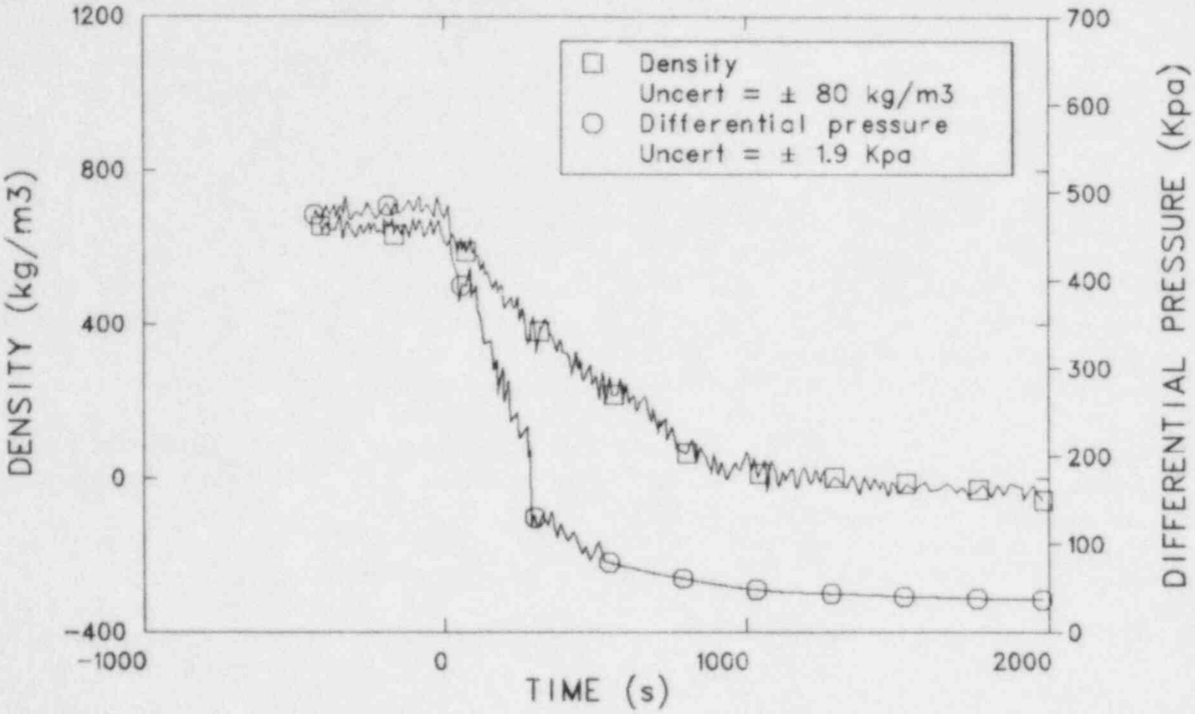

Figure 7-5. Fluid density in intact loop cold leg and differential pressure across the primary coolant pumps for LOFT Experiment L3-6.

than the guide tube and cladding thermocouples, did not immediately sense the temperature excursion in the core. Figure 7-6 shows cladding, guide tube, and hot leg temperature behavior during degraded core cooling and subsequent core quench.

The process instruments (subcooled meter, hot leg, cold leg, and core exit thermocouples) did not provide sufficient information on degraded core cooling. Therefore, cladding or guide tube temperature measurements are recommended for this purpose. Cladding temperatures would provide the best data, but installation problems may dictate the use of guide tube thermocouples instead. Measurement of fluid density in both the hot and cold legs is also recommended to provide a measure of voiding.

Based on this analysis, the following potential improvements were identified:

1. Installation of cladding temperature sensors (300 to 2800°F), or as an alternative temperature sensors mounted in guide tubes (300 to 2300°F) to alert operators of core heatup in the event of core uncovery. Temperature sensors should have response times of 75 ms or shorter.

2. Installation of density measurement instrumentation (10 to 1000 kg/m^3) in each of the primary coolant loops to indicate system void and warn operators of degraded core cooling.

3. Provision for density compensation for vessel differential pressure liquid level measurements to account for temperature and pressure variations, and provide a method of dealing with flashing to ensure proper level readings during transients.

7.1.4 Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K. This issue is addressed in detail in Section 5.2.3.

7.2 TMI Action Plan Requirements

Two items in the final recommendations of the NRC TMI Action Plan[7-2] relate to experiments that have been conducted in the LOFT facility. The two items are closely related and are discussed together:

1. Instrumentation to verify natural circulation

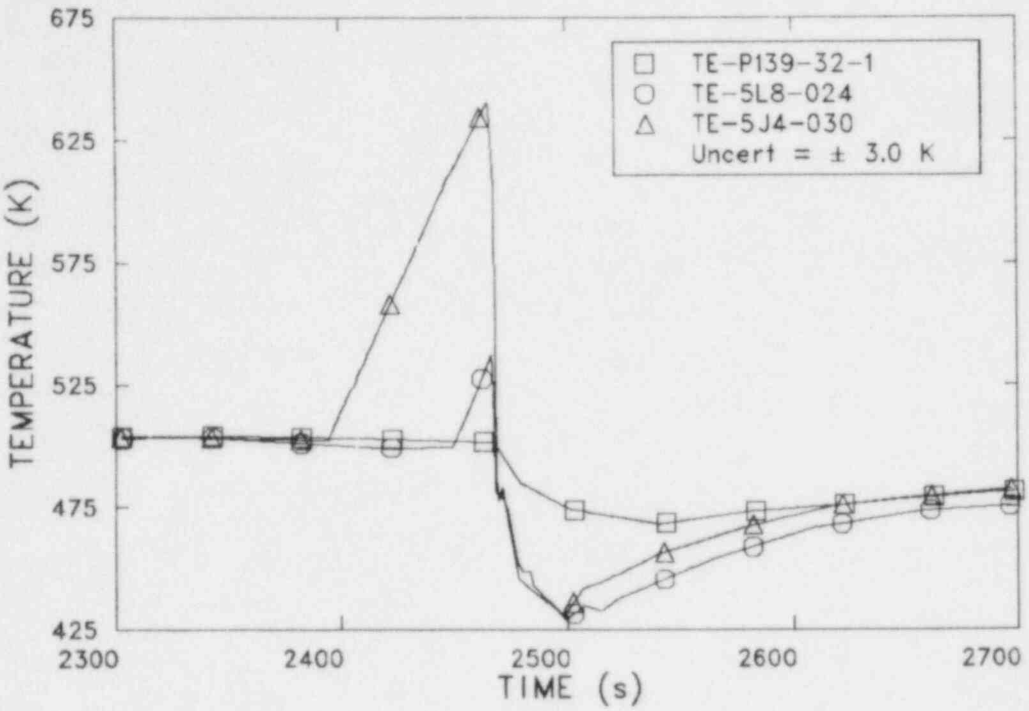

Figure 7-6. Temperature in intact loop hot leg, Fuel Rod 5L8 guide tube, and Fuel Rod 5JA cladding during degraded core cooling and subsequent core quench for LOFT Experiment L3-6/L8-1.

2. Experimental verification of two-phase natural circulation.

LOFT uses four measurements to detect single-phase or two-phase natural circulation: differential temperature across the core, differential temperature across the steam generator, flowmeter (turbine) output from above the core, and flowmeter (turbine and PNA) output from the primary loop piping.

LOFT Experiment L3-7 simulated the break of a small (1 in. diameter) pipe attached to a cold leg of a commercial PWR. During the experiment, single-phase and two-phase natural circulation transferred heat from the primary system to the steam generator. Experiment L3-7 provided good experimental data to identify single-phase and two-phase natural circulation as discussed in the following paragraphs.

Fluid temperatures in the upper plenum, lower plenum, and corresponding saturation temperature are compared in Figure 7-7. Fluid temperatures in the steam generator primary inlet, outlet, and corresponding saturation temperature are compared in Figure 7-8. Fluid velocities above the center fuel module and in the intact loop hot leg are shown in Figure 7-9.

Single-phase natural circulation is easily detected by differential temperature across the core and steam generator.

Two-phase natural circulation cannot be detected by differential temperatures because the system was saturated and no temperature difference was indicated across the core and steam generator. When the primary system was saturated, the fluid velocities in the primary system were higher than when it was subcooled. Fluid velocities in the intact loop hot leg were detected with PNA, and above the center fuel module by a turbine meter.

During single-phase natural circulation, fluid velocities are generally low and require techniques such as PNA, or sensitive turbine meters. Two-phase natural circulation, however, can only be detected by fluid velocity. During two-phase natural circulation, fluid velocities are higher than during single-phase natural circulation, and turbines and PNA both work well. PNA, however, only works with the liquid fraction of the two-phase flow. If the void fraction becomes too high, the technique cannot be used.

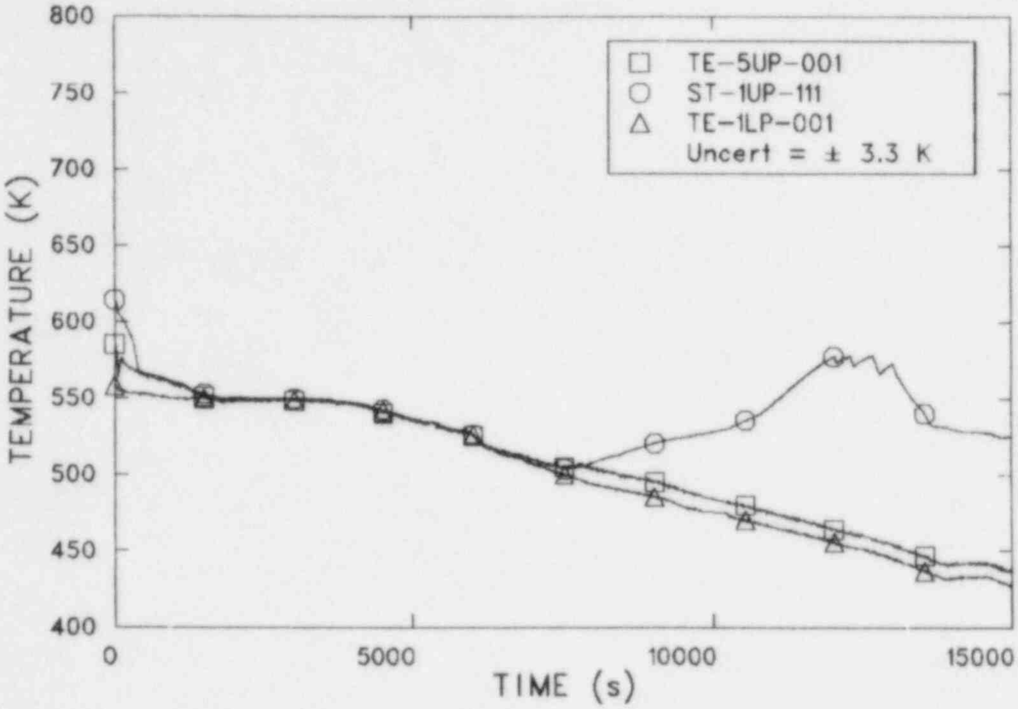

Figure 7-7. Fluid temperature in reactor vessel upper and lower plenums and fluid saturation temperature for LOFT Experiment L3-7.

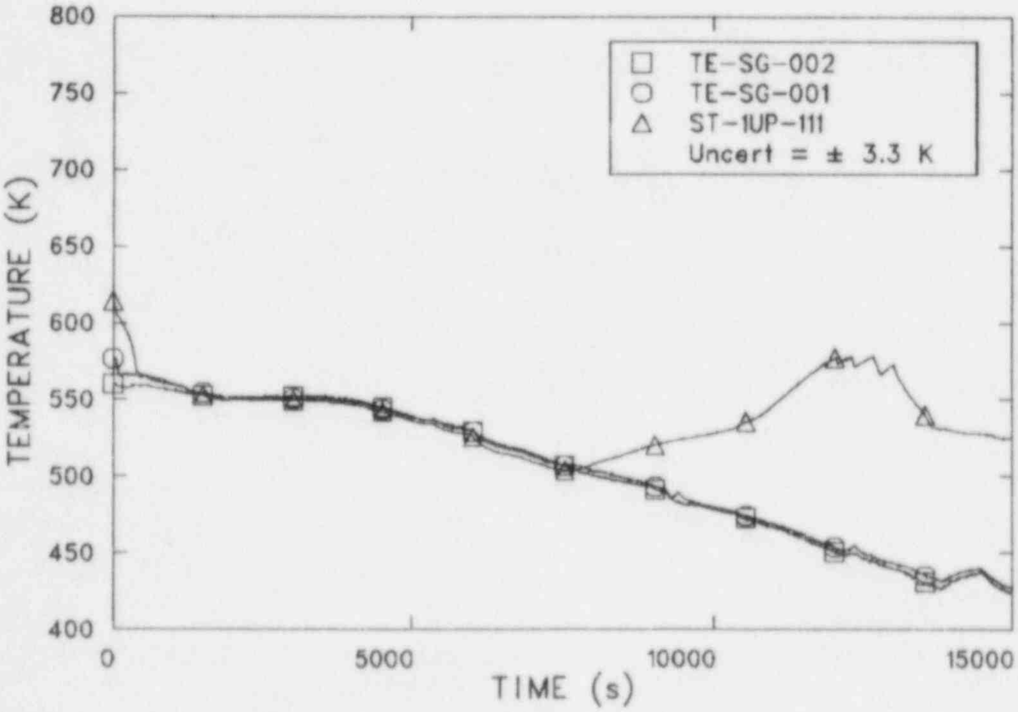

Figure 7-8. Fluid temperature in steam generator inlet and outlet plenums and saturation temperature for LOFT Experiment L3-7.

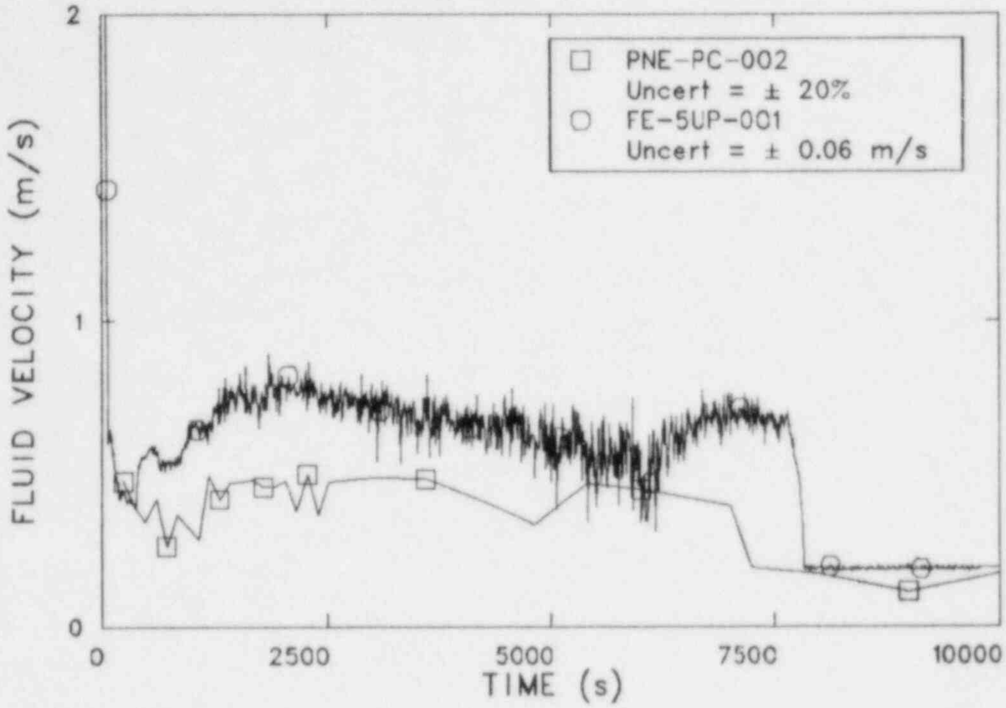

Figure 7-9. Fluid velocity above center fuel module and in intact loop hot leg for LOFT Experiment L3-7 (measurements were made with pulse neutron activation and turbine meter).

7.3 References

7-1. Nuclear Regulatory Commission Special Inquiry Group, *Three-Mile Island, A Report to the Commissioners*, 2, 1, April 5, 1979.

7-2. *NRC Action Plan Developed as a Result of the TMI-2 Accident*, NUREG-0660, May 1980.

7-3. G. E. McCreery, *Quick-Look Report on LOFT Nuclear Experiment L3-6/L8-1*, EGG-LOFT-5318, Rev. 1, July 1981.

7-4. P. D. Bayless, *Experiment Results Report for LOFT Nuclear Experiments L3-5, L3-6, and L8-1*, EGG-LOFT-5471, July 1981.

7-5. NRC, "Staff Discussion of 15 Technical Issues Listed in Attachment to November 3, 1976 Memo from Director, NRR to NRR Staff," NUREG-0138, Issue 4, November 1976.

7-6. J. P. Adams, *Quick-Look Report on LOFT Nuclear Experiment L3-5/L3-5A*, EGG-LOFT-5242, October 1980.

7-7. J. P. Adams, *Quick-Look Report on LOFT Nuclear Experiment L2-5*, EGG-LOFT-5921, June 1982.

7-8. United States Nuclear Regulatory Commission, *Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident*, Regulatory Guide 1.97, Revision 2, December 1980.

8. SAFETY TECHNOLOGY

Safety technology encompasses design features, instrumentation, and devices of an innovative nature that enhances or improves the safety of a nuclear reactor. The LOFT Program has made important contributions to safety technology in the following:

1. Fuel prepressurization

2. Liquid level detectors

3. Thermocouples (fuel cladding and core exit)

4. Reactor coolant pump current monitoring.

8.1 Fuel Prepressurization

Fuel rod pressurization was initially instituted to prevent fuel rod cladding flattening and intrusion into pellet stack discontinuities (gaps) caused by densifying fuel pellets. Subsequent improvements in the manufacture of fuel pellets eliminated the densification problems, but pressurization was continued for the purpose of minimizing pellet-cladding interaction (PCI).

Experience on the LOFT Program[8-1] and data from the Zorita Research and Development Program[8-2] both indicate that unpressurized fuel may perform as well as prepressurized fuel if the fuel rods are designed and manufactured properly. In addition, fuel rod burst/coolant-channel-blockage research indicates that unpressurized fuel balloons less under severe LOCA conditions, thus improving core coolability and plant recovery. The following is a summary of the LOFT, Zorita, and fuel rod burst/coolant-channel-blockage research results.

8.1.1 LOFT Results. Fuel rods with atmospheric prepressurization have been successfully used in the LOFT reactor to study PWR behavior over a wide range of accident conditions including LOCAs, anticipated transients, and ATWS. To date, LOFT has accumulated a peak fuel burnup of 3300 MWd/MTU and 27 simulated accidents which have resulted in nine core uncoveries and three occurrences of fast power ramping of the fuel. During this time, the LOFT fuel has never experienced a fuel rod break or rupture even with the adverse effects of densifying fuel. The fuel used in the unpressurized assemblies in LOFT was manufactured during June and July of 1972. Archive UO_2 samples were resintered for 14 and 24 h at 1973 K in order to ascertain the extent of densification with the old fuel. The average densification is 3.5% and results in a one sided upper tolerance limit of 4.2%. According to the NRC classification guidelines, the LOFT fuel is unstable.

The LOFT core utilization since 1977 has included three fuel cycles discussed in Table 8-1. The LOFT fuel[8-3] has been exposed to the test sequence summarized in the following paragraphs.

The unpressurized A1 center fuel module was used for the initial nuclear experiments and was replaced after exposure to three large-break LOCEs, accumulation of 987 MWd/MTU peak fuel burnup, and before a full-power (52 kW/m peak) preconditioning status had been achieved. Poolside examination of the center fuel bundle determined that it had sustained no fuel damage and was reusable even though some of the cladding had exceeded the temperature threshold for recrystallization during LOCE L2-3.[8-4]

The A2 center fuel module was replaced after exposure to two intermediate-break LOCEs, six small-break LOCEs, and five operational-transient experiments, one of which (turbine load increase) caused a rapid power increase from 39 to 45 kW/m in 10 s at the peak-power zone. During this time there was an accumulation of 2222 MWd/MTU peak fuel burnup and achievement of a full-power preconditioned status. The rapid power increase was sustained after achievement of the full-power preconditioned status. The final experiment for this fuel module caused some of the cladding to exceed the recrystallization temperature. It is known that no fuel rods are leaking, but further information is unavailable since no poolside examination was performed. The fuel module was replaced to improve the core measurement capabilities and is judged to be in a reusable condition.

The F1 center fuel module has prepressurized fuel and has been exposed to one large-break LOCE, three small-break LOCEs, and four operational transients of which two (control rod withdrawals) caused rapid power increases (47 to 53 kW/m in 7 s and 47 to 56 kW/m in 58 s at the core peak-power zone). It has accumulated 765 MWd/MTU peak

Table 8-1. LOFT fuel utilization

	Fuel Modules		
Cycle	Peripherals (8)	Center	Fuel Loading Data
1	Core 1	A1[a]	September 1977
2	Core 1	A2[a]	July 1979
3	Core 1	F1[b]	January 1982

a. Nominal fuel density of 92.5%, all fuel rods prepressurized to atmospheric pressure.

b. Nominal fuel density of 92.5%, fuel in two rods stabilized with pore formers, and fuel rods prepressurized to 2.41 MPa with outside row at atmospheric pressure.

fuel burnup and achieved a full-power preconditioned status. The rapid power increases were sustained before a full-power preconditioned status had been achieved, with some of the cladding in a recrystallized condition and with analysis indicating that up to 4% swelling of the hottest fuel rod's cladding may have occurred in LOCE L2-5.

The peripheral fuel modules have been exposed to all of the experiments since September 1977 and have accumulated 3317 MWd/MTU peak fuel rod burnup. Some of the cladding has exceeded recrystallization temperatures during the experiments. The fuel rods have been maintained in a full-power preconditioned status since February 1980. Visual inspections (July 1979 and January 1982) of the peripheral fuel module surfaces exposed by removal of the center fuel module have shown conditions which are judged to be normal for PWR fuel rods in service. The peak-power zone of the peripheral fuel is ~83% (43.3 kW/m at full power) of the center fuel bundle peak-power zone.

8.1.2 Zorita Results. The Zorita Research and Development Program[8-4] provided information on the performance of Zircaloy-4 clad UO_2 fuel rods, similar to the LOFT fuel rods, in up to five reactor cycles to average fuel rod burnups up to 58 200 MWd/MTU. Over 160 fuel rods were tested with a principal design variable being the internal pressurization (helium and air at atmospheric pressure or helium at 3.45 MPa). Both fuel rod types performed satisfactorily to high burnup except during a power transient when a higher percentage (38.5 versus 35.0) of the pressurized fuel rods failed by PCI.

The Zorita Program did not include the following features used to counteract PCI effects:

1. Shorter-chamfered-end fuel pellets

2. Controlled fuel pellet temperature to minimize gaseous fission product release from the fuel pellets.

The conclusions from the Zorita Program relative to fuel prepressurization are that cladding creepdown is less for pressurized rods than for identical unpressurized rods at all burnup levels. PCI is less for the pressurized fuel rod design; however, prepressurization above atmospheric pressure had no significant effect on rod growth, fission gas release, or PCI-related cladding defects.

8.1.3 Fuel Rod Burst/Coolant-Channel-Blockage Research Results. United Kingdom researchers reported[8-5] development of severe coplanar axially-extended flow channel obstructions under postulated worst case (isothermal) LOCA conditions which were much greater than would be predicted by the United States licensing models.[8-6] Further experimental work[8-7] using increasingly larger (up to 4 by 4) bundles of fuel rods has shown a trend of increasingly larger flow blockage also greater than would be predicted by the licensing models for fuel rod swelling and flow channel blockage. In addition, LOCA testing in LOFT,

which approached the temperatures needed for fuel rod swelling, has shown a trend for uniform temperatures in large regions of the fuel modules[8-8] which is a condition needed to develop large axially-extended fuel rod deformations and flow channel blockages.

The fuel rod swelling that will precede fuel rod burst also will expand the envelope of the fuel assembly so that fuel assemblies will be wedged together, significantly increasing the time (a) to remove the damaged fuel, and (b) of no revenue production from the power generating facility.

The above potential for aggravating the consequences of a LOCA by constricting the coolant channels and mechanically wedging the fuel assemblies together is judged to remain an area of uncertainty requiring additional confirmatory research for resolution. The above mentioned aggravating events caused by fuel rod swelling can be reduced and or eliminated, depending on the severity of the LOCA, by not prepressuring the fuel rods. Potentials for cost savings are:

1. Lower post-LOCA facility clean up costs because of lower fission product release due to lower fuel temperature

2. Earlier return of facility to revenue production

3. Curtailment of fuel rod swelling and burst and flow channel blockage research.

8.2 Liquid Level Detectors

LOFT uses two types of instruments to indicate liquid level. These instruments are liquid level transducers (LLTs) (conductivity probes) and differential pressure transducers. Another instrument, the self-powered neutron detector (SPND), has been investigated for its use as a liquid level detector. This section provides a short description of each instrument, a summary of the LOFT experience with each, and a discussion of their limitations.

8.2.1 Liquid Level Transducers. An LLT system comprises a conductivity-sensitive transducer, electronic signal conditioning equipment, and data recording equipment, see Figure 8-1. Each transducer consists of a support tube; up to 19 equally spaced, independent, conductivity-sensitivity electrodes; and a common ground plane (refer to Figure 8-2). For each transducer, the electrodes are spaced equally over the expected water level. The presence or absence of liquid is determined by the voltage across the media between an electrode and the ground plane. The impedance between the electrode and the ground plane varies from 0.25 to 1.5 kΩ with a boric acid, lithium hydroxide, water mixture, and is >5 kΩ in steam.

The design of each transducer differs, depending on where the transducer is located. The differences in design are in the spacing of the electrodes, the types of ports (either slotted or circular), and the presence of and/or type of splash shielding.

Splash shields are installed on the transducers, in locations where there is a large amount of turbulent liquid such as the downcomer and lower plenum, to prevent splashing of the primary coolant that could cause erroneous liquid level measurements. No shields are included for LLTs installed in the core and upper plenum because the liquid level in these areas is less turbulent. Additional design and installation details plus measurement uncertainties are presented in Reference 8-9.

Typically, LLT output is presented in the form of a bubble plot that indicates the presence or absence of liquid at the conductivity probe elevations as a function of time. Figure 8-3 is a bubble plot from the LLT in Fuel Assembly 1 during LOFT Experiment L2-3,[8-10] a large cold leg break. The bubble plot shows the rapid system voiding and the reflood of the core. During LOFT Experiment L3-6,[8-11] a small-break experiment performed with the pumps running, however, the LLTs failed to indicate the correct voiding of the reactor vessel because the electrodes were kept wet by the circulating two-phase mixture. The LLTs indicated a much lower void fraction and, consequently, a higher liquid inventory than actually existed in the system. When the pumps were turned off, the liquid level collapsed much lower than had been planned and uncovered the core. The LOFT experimental results showed that the LLTs work best with a clean liquid-vapor interface. Experience indicates that the transition from wet to dry can lag the actual passage of a liquid level by a substantial amount of time.

8.2.2 Differential Pressure Transducers. LOFT uses differential pressure measurements to determine liquid levels in the pressurizer, accumulators, and the secondary side of the stream generator.

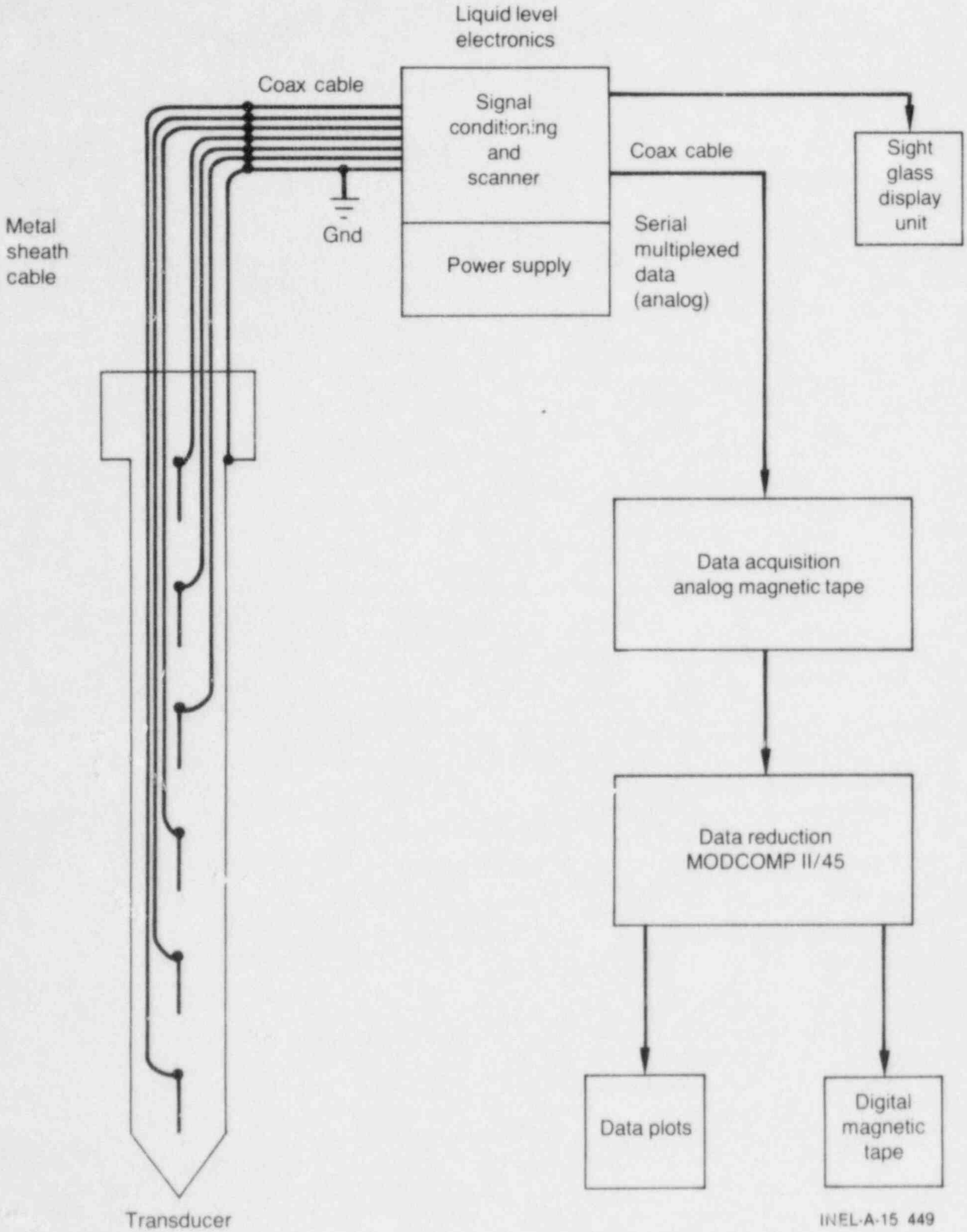

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Figure 8-1. Liquid level transducer system.

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Figure 8-2. Typical liquid level transducer.

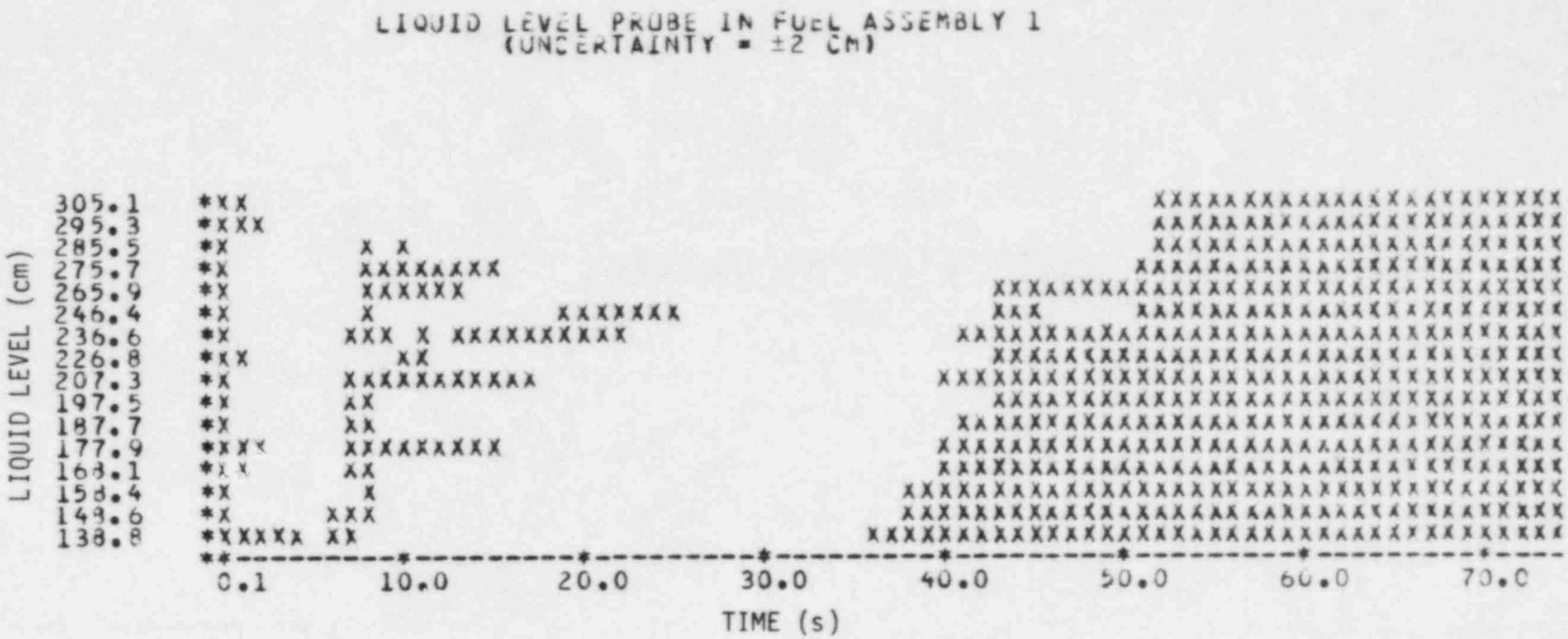

Figure 8-3. Bubble plot showing liquid level in reactor vessel at Fuel Assembly 1 for LOFT Experiment L2-3.

Figure 8-4 illustrates the typical installation of a differential pressure transducer for liquid level measurement. For applications where the temperature and pressure do not vary significantly, good liquid level measurements are possible. However, liquid level measurements under transient conditions (blowdown) are very difficult. The problems associated with determining liquid levels with differential pressure measurements during transients is discussed in Reference 8-12.

Typical problems identified for differential pressure measurements in LOFT experiments include:

1. Liquid "flashing" in the reference leg during changes in pressure
2. Density changes of liquid in the vessel during transients which must be accounted for properly indicate the height of the liquid
3. In a reactor vessel with complex internal structures and flow barriers, densities can vary from region to region with time, making accurate determination of liquid level more complex.

Careful design and installation of differential pressure transducers are necessary to obtain

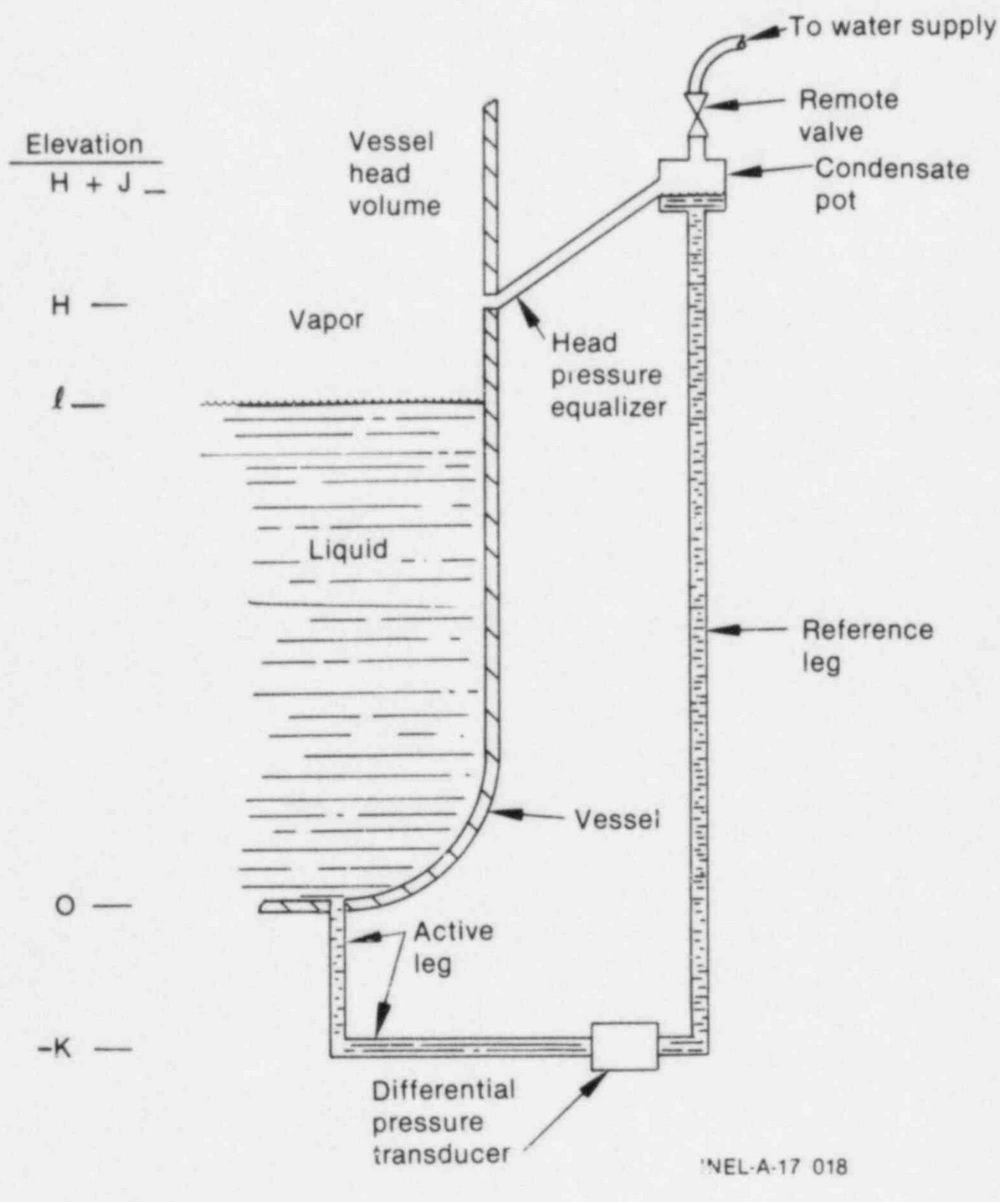

Figure 8-4. Typical piping configuration for a transient level instrument.

satisfactory liquid level measurements. For transient measurements, calibration and temperature/density compensation is necessary to prevent excessive errors.

Figure 8-5 shows liquid level in the pressurizer computed from differential pressure measurements and computed by RELAP4 and RELAP5 computer codes for LOFT Experiment L3-0,[8-13] which simulated a small break at the top of the LOFT pressurizer by opening the PORV. The figure shows the large correction that had to be made to correct for the large density change in the pressurizer during the transient.

8.2.3 Self-Powered Neutron Detectors. SPNDs are not normally used to measure liquid level in LOFT; however, their use as liquid level detectors has been evaluated. The LOFT SPNDs have cobalt emitters which exhibit sensitivity to water density that has been correlated with independently measured moderator density variation.[8-14] The sensitivity of SPNDs to water density variation indicates that an array of SPNDs could be used to detect liquid level in a PWR during accident conditions.

The LOFT SPNDs, shown schematically in Figure 8-6, consist of a 0.23-m long by 0.19-cm diameter cobalt emitter inside an Inconel collector, which also forms the sheath. The emitter is separated from the sheath by aluminum oxide insulation. Figure 8-7 is an illustration of a LOFT core guide tube, cut away to show how the SPND is installed.

There are two leads from the detector assembly: a signal lead connected to the emitter and a cable compensation lead. The signals from the two lead wires are independently amplified by low-input bias ($< 10^{-14}$ A) current-to-voltage amplifiers. The compensation voltage output is subtracted from the main signal output, and this difference is again amplified and input into the computerized data collection system.

LOFT SPND characteristics and the response of SPNDs during LOFT LOCEs are reported in Reference 8-14. Moderator density variations affect neutron and gamma flux which in turn is reflected in SPND output. The effect of decreasing the core water density (increasing the void fraction) on neutron flux is twofold: first, decreased water density results in negative core reactivity insertion due to the negative void coefficient of reactivity. This reduces the neutron multiplication and, hence, the neutron flux itself. Second, as the density decreases and approaches that of steam, the moderating efficiency drops, resulting in a reduction in the ratio of thermal to total neutron flux, this effect combines with the first to reduce the SPND neutron current.

The effect of decreasing core water density on the gamma flux is, to first order, simply that of decreased absorption or attenuation resulting in an increased gamma flux and SPND gamma current. Since the SPND gamma current is opposite in polarity to the neutron current, this change (increase) in current behaves like a decrease in neutron current. Thus, the combined effect of reduced SPND neutron current and increased gamma current is decreasing (more negative) SPND current with decreasing water density. When the water density increases in the core (that is, during quench or reflood), the SPND current increases (that is, becomes more positive).

The response of the LOFT SPNDs during LOCE L2-3 is shown in Figures 8-8 and 8-9. The SPND output in Figure 8-8 clearly shows the reactor scram. In addition, the initial core-wide quench that occurred at 6.5 s produced a measurable increase in SPND output which correlated well with the large drop in fuel rod cladding temperatures which occurred when the liquid level in the core rose and quenched the rod, see Figure 8-9.

The results of these preliminary results are encouraging and indicate that SPNDs could be used to monitor liquid level in PWRs and detect voiding in the core. However, a great deal of testing, design, and verification testing are required before a reliable SPND liquid level system is perfected.

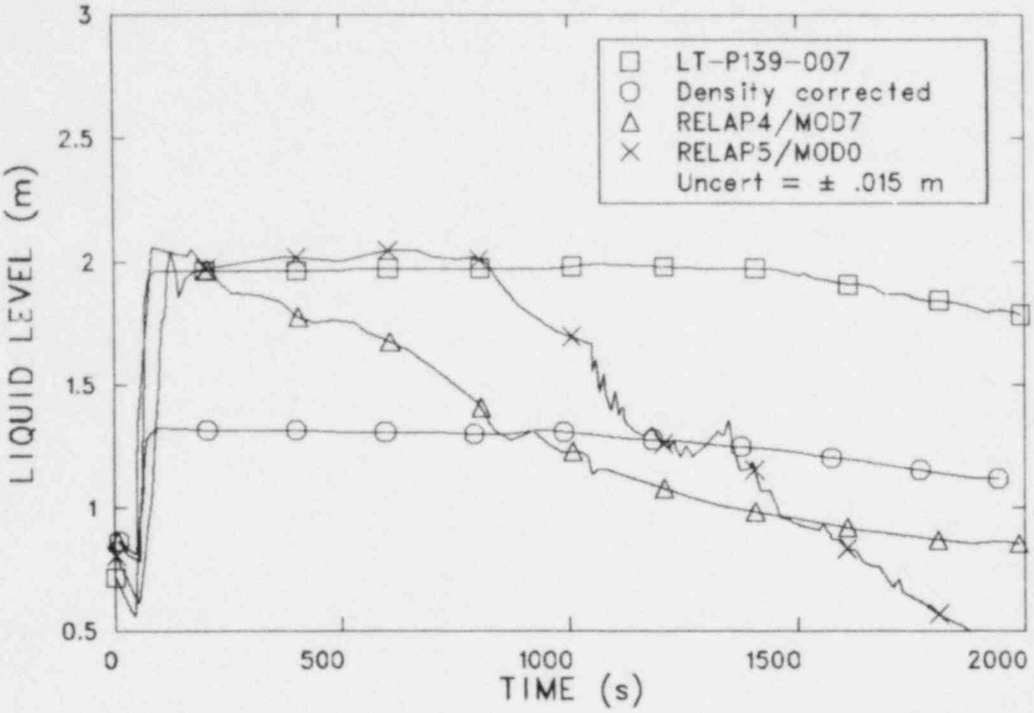

Figure 8-5. Measured and calculated liquid level in pressurizer for LOFT Experiment L3-0, showing the effect of transient conditions on liquid level computed from differential pressure measurements.

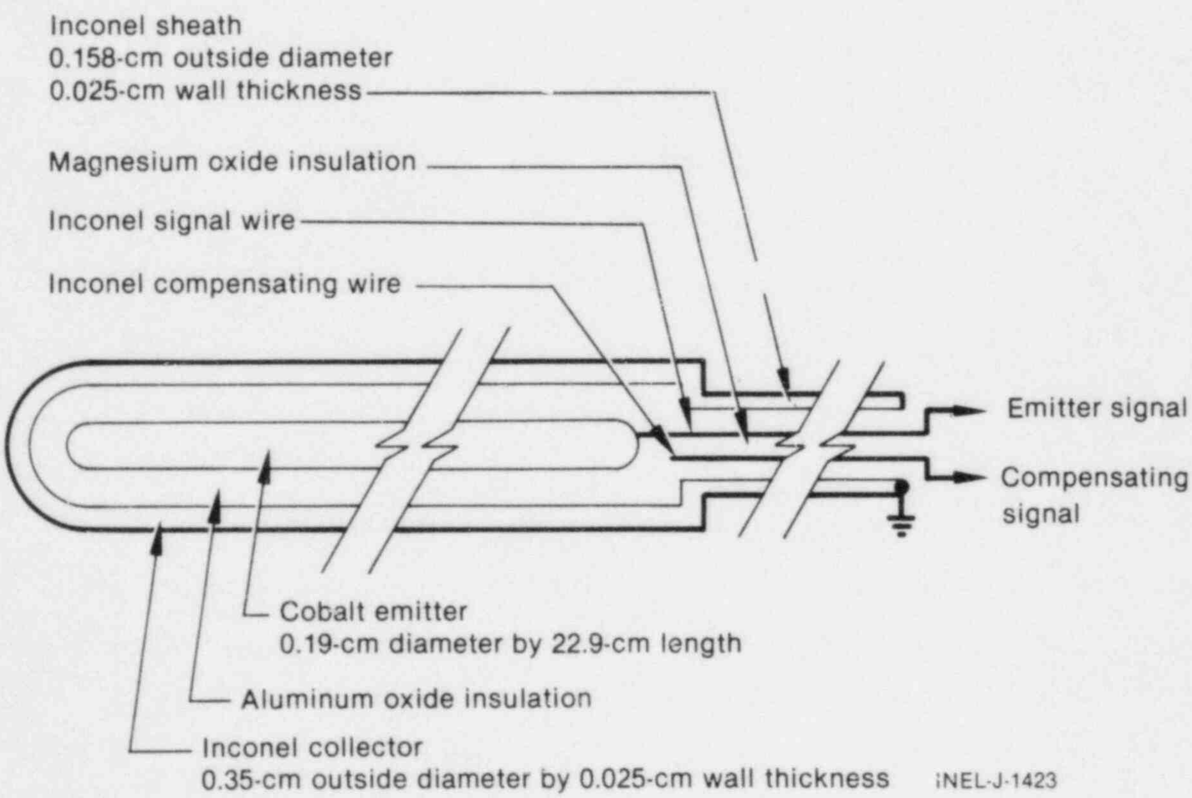

Figure 8-6. Schematic diagram of LOFT SPND.

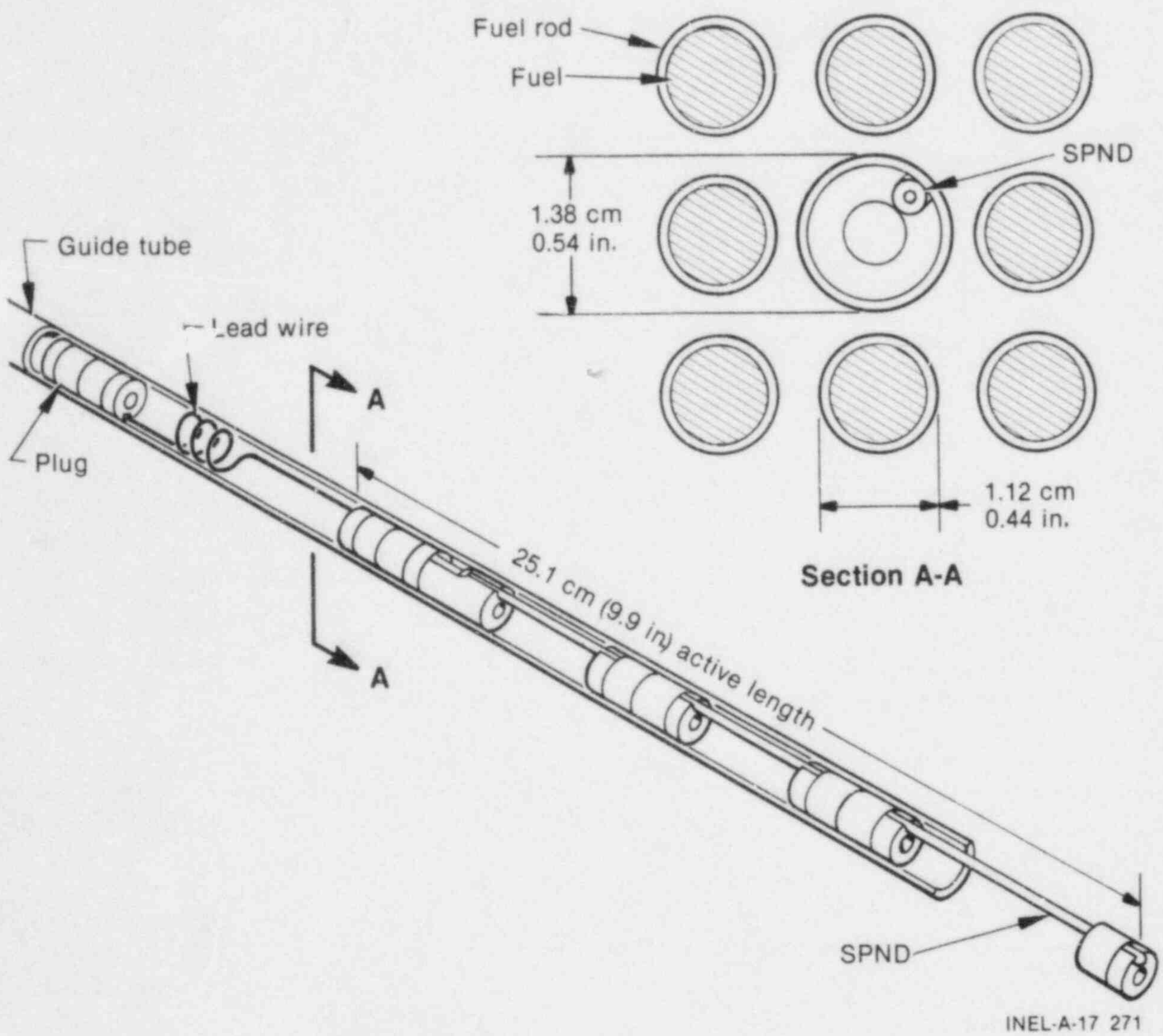

Figure 8-7. LOFT SPND guide tube assembly.

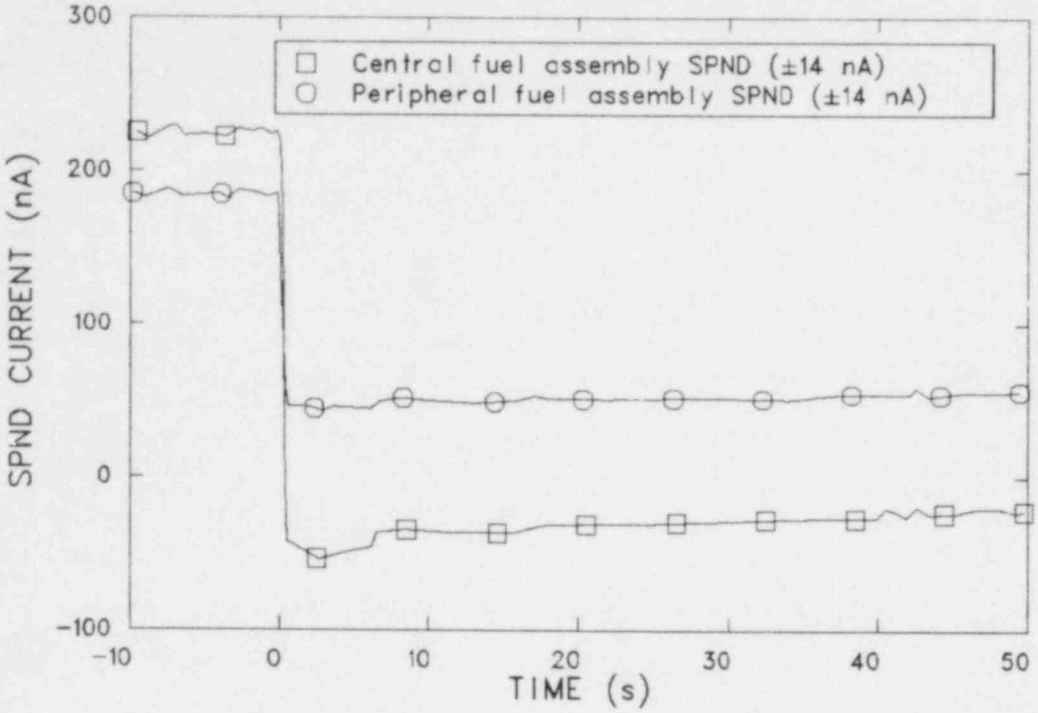

Figure 8-8. Output from central and peripheral SPNDs for LOFT Experiment L2-3.

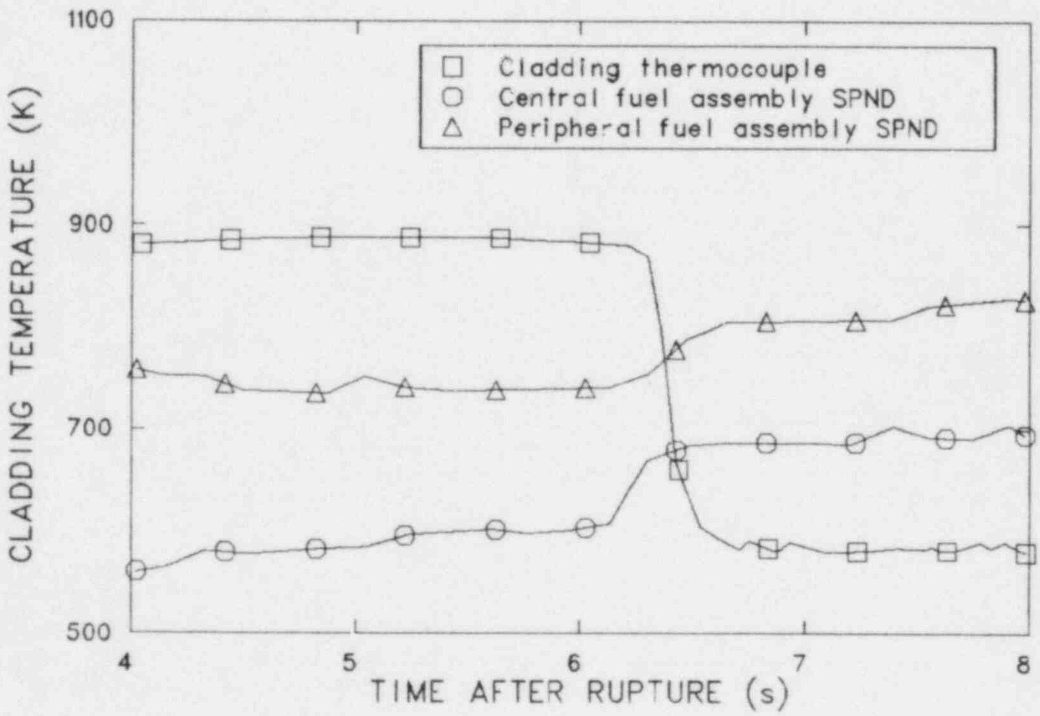

Figure 8-9. Output from central and peripheral SPNDs and an adjacent fuel cladding thermocouple during quench for LOFT Experiment L2-3.

8.3 Thermocouples

Thermocouples (TCs) have been used extensively in the LOFT facility, and in general represent a well established technology for measuring metal or fluid temperatures. Two LOFT applications of TCs are particularly important to safety technology and are discussed in this section: fuel cladding TCs and core outlet TCs.

8.3.1 Fuel Cladding Thermocouples. Fuel cladding TCs have been used extensively in LOFT to monitor real-time behavior of LOFT fuel under a variety of accident conditions. The LOFT core contains 186 external fuel cladding TCs attached to 76 fuel rods located throughout the core, as shown in Figure 8-10. The LOFT cladding TCs provided the most direct information regarding the condition of the fuel during LOFT experiments.

The thermal responses of the peak power rods are of most interest; hence, these rods are the most heavily instrumented in LOFT. These peak-power rods are contained in the center fuel module. Figure 8-11 shows the center module cross section emphasizing the TC locations on the instrumented

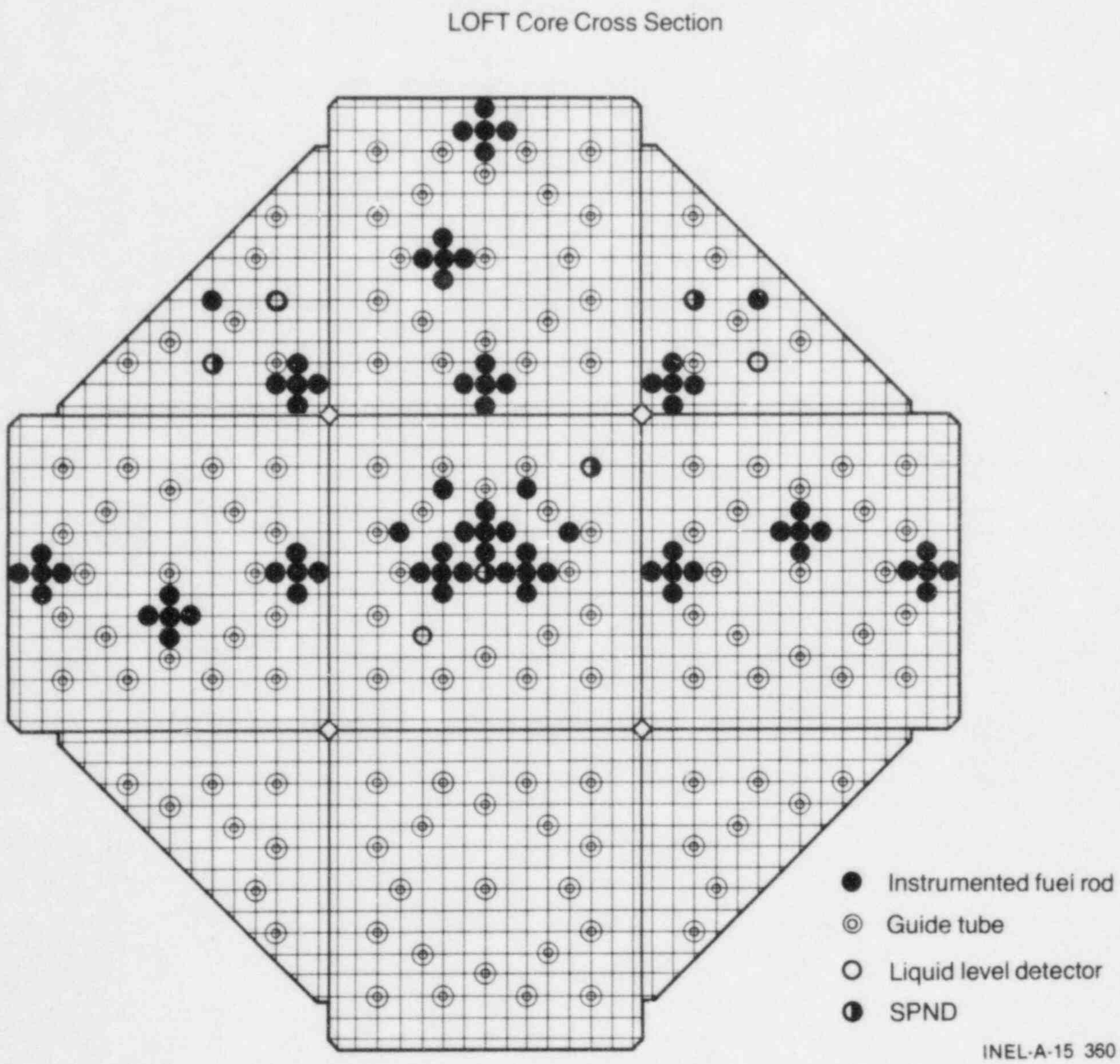

Figure 8-10. LOFT core configuration showing locations of instrumented fuel rods.

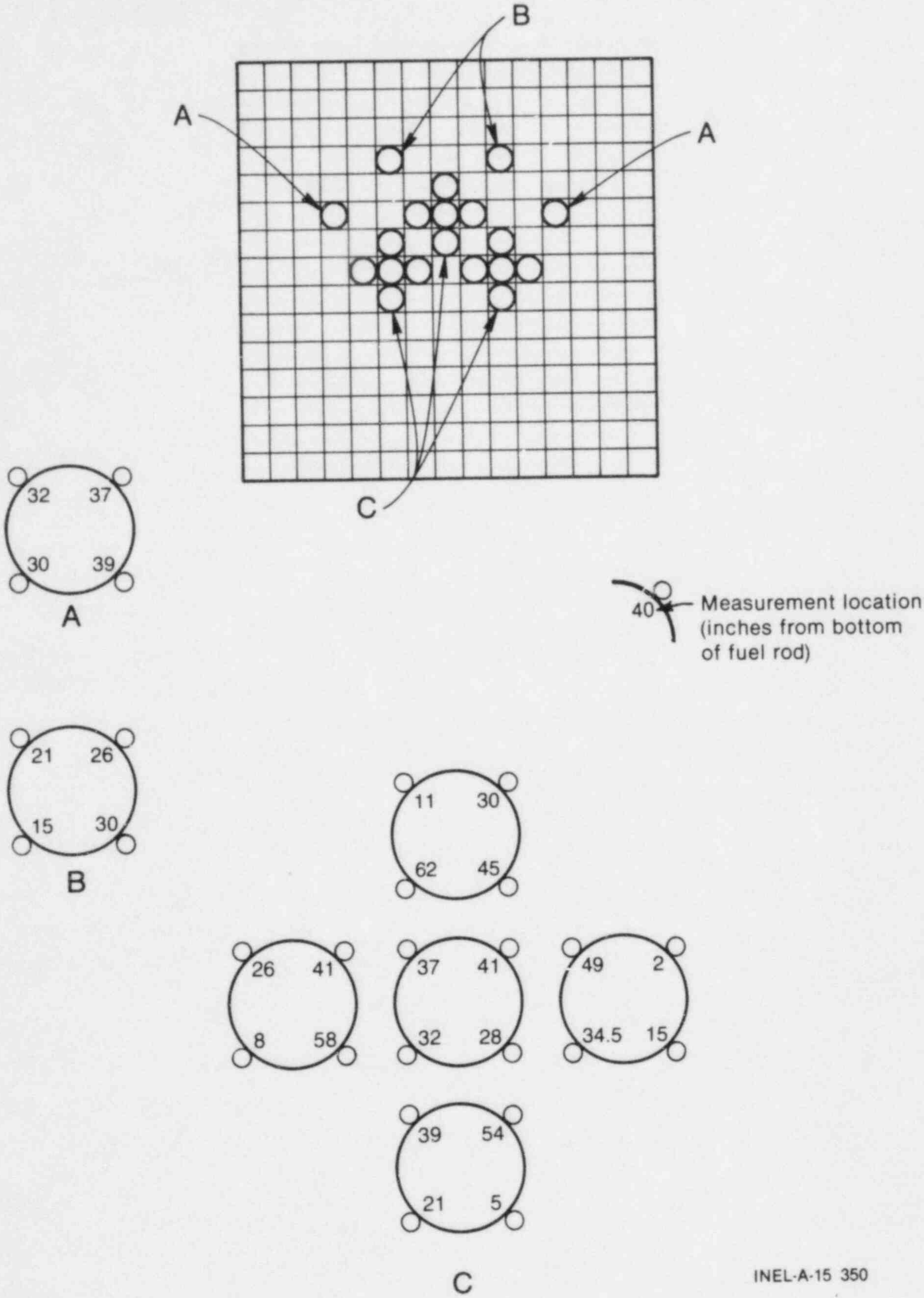

Figure 8-11. Instrumented fuel rods in LOFT center fuel module.

fuel rods. Three groups, or clusters, of five adjacent rods near the core center, are identically instrumented with TCs ranging in axial elevation from 5 to 162.7 cm. The center rod in each cluster represents the hot-test rods in the core that are completely surrounded by other fuel rods. The four single rods farther out from the core center each contain four TCs ranging axially from 38 to 99 cm. These four rods are the peak power rods in the core and have greater powers (2 to 7%) than rods in the five rod instrumented clusters.

The LOFT method of attaching a TC to the cladding is shown schematically in Figure 8-12. The 1.17-mm outside diameter titanium TC sheath is laser welded to the cladding surface at ~2.5-cm intervals. To reduce axial rod bowing effects from nonsymmetric thermal response, dummy thermocouple segments extend from upper level TCs to the lowest axial TC position on each rod. The thermal junction is flattened to 0.67 mm and is shown schematically in Figure 8-13.

The capability of fuel cladding TCs to provide accurate cladding temperature measurements has been studied extensively in support of the LOFT Program. The studies were concerned primarily with the effect of the TCs on the accuracy of the measurements since it was postulated that the TCs and their leads acted as fins which cooled the fuel rods during LOFT experiments. The results of the studies are summarized in References 8-15, 8-16, and 8-17. Reference 8-15 describes experiments performed with external, internal, and embedded TCs on electrical heater rods designed to simulate nuclear fuel rods. The experiments showed that external cladding TCs indicated quenching 2 s before the cladding quenched during rapid flooding experiments (1 to 2 m/s), and had negligible effects on thermal-hydraulic conditions in the core. The results of the studies described in Reference 8-16 and 8-17 showed that external cladding TCs measure temperatures to within 20 K during blowdown and to within 40 K during reflood.

Examples of TC output during several LOFT experiments are presented in earlier sections of this document, for example, Figures 5-9 and 5-10 show cladding temperature during large-break Experiment L2-3.

Another example of TC output is presented in Figure 8-14, which shows cladding temperature and saturation temperature during LOFT Experiment L8-1.[8-18] Experiment L8-1 was a severe core transient which followed Experiment L3-6, a small-break experiment performed with the reactor coolant pumps on. Experiment L8-1 was initiated by turning the pumps off. The pumps were turned off at 2371 s and coasted down within 87 s. Within 25 s of pump trip, the fuel cladding temperature excursion began because the pump-driven flow was no longer sufficient to entrain liquid in the core and this liquid collapsed below the lower core elevation. The cladding TCs provided the first indication that the fuel was uncovered. The operators initiated HPIS and accumulator injection within 68 s and quenched the core. The rapid core dryout and temperature excursion of Experiment L8-1 was not predicted because the preexperiment calculations

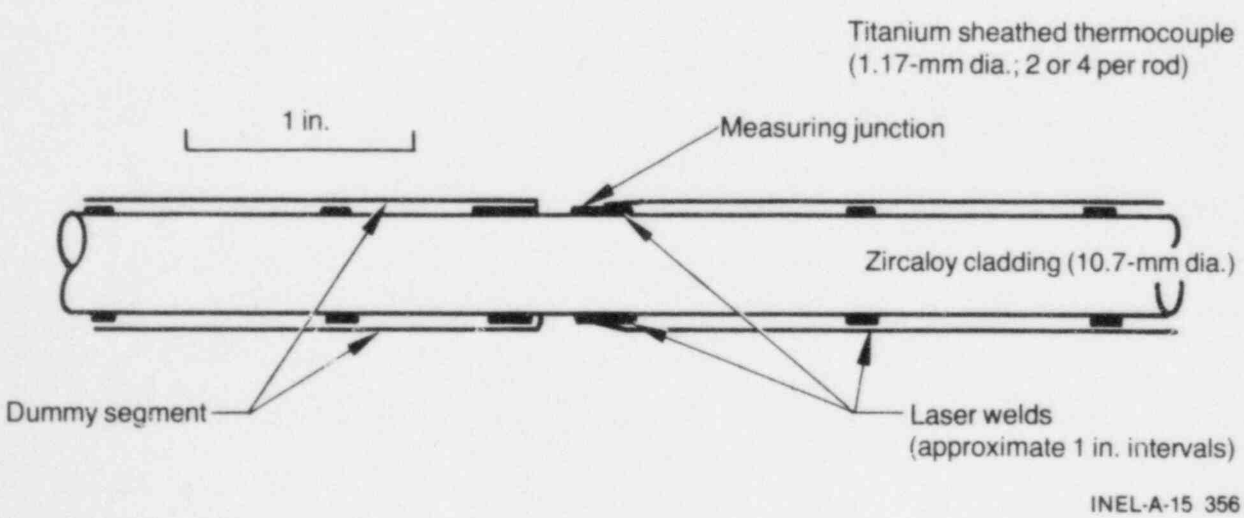

Figure 8-12. Illustration showing LOFT method of attaching thermocouples to cladding.

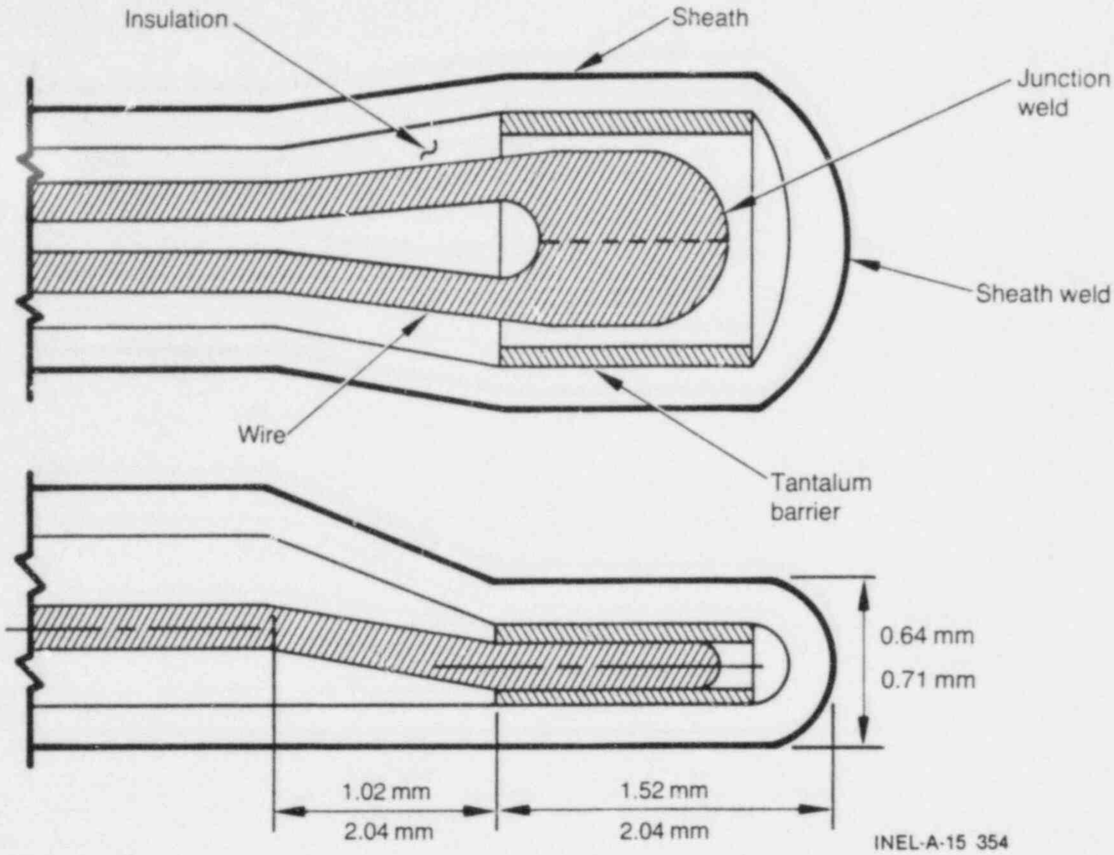

Figure 8-13. LOFT cladding thermocouple tip geometry.

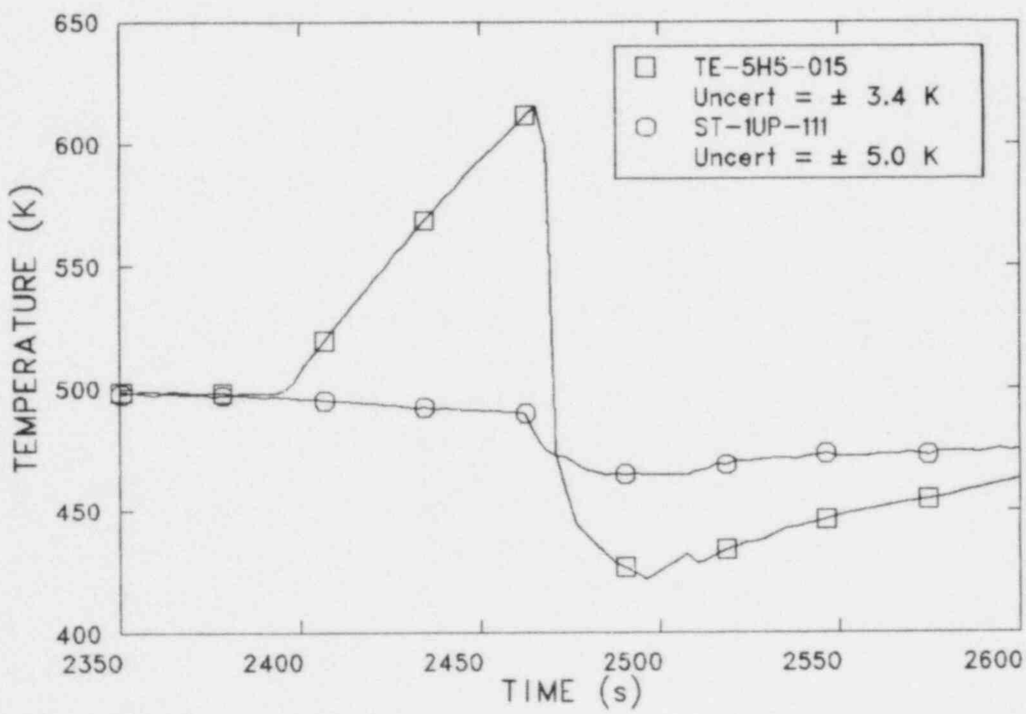

Figure 8-14. Fuel cladding and saturation temperatures in reactor vessel for LOFT Experiment L3-6/L8-1.

predicted the core would be liquid full at the end of the Experiment L3-6 transient. Indications of liquid level within the core responded sufficiently slow that the temperature rise was terminated before they showed significant voiding.

8.3.2 Core Exit Upper Plenum Thermocouples. PWR vendors are currently proposing the use of the core exit TCs as an indication of when the core reaches superheat and, therefore, inadequate core cooling conditions. Several LOFT accident simulations have provided data regarding the performance of core exit TCs that can be used to evaluate the ability of the TCs to indicate inadequate core cooling conditions.

There are 21 Type K (Chrome Alumel) TCs located in the LOFT upper end boxes as shown in Figure 8-15. The TCs protrude horizontally into the flow path. Core exit TCs in commercial PWRs are similar to those in LOFT.

LOFT Experiments L3-6/L8-1, L5-1, and L2-5 provided data on the response of core exit TCs from small-, intermediate-, and large-break LOCAs. The three experiments are examples of different scenarios leading to an inadequate core cooling condition which showed similar results with respect to the temperature differences between the core exit and the core cladding temperature.

LOFT Experiment L3-6/L8-1 was a small-break accident simulation with delayed pump trip (see Reference 8-18), that is, the primary pumps were tripped 2371 s after the reactor had scrammed. By

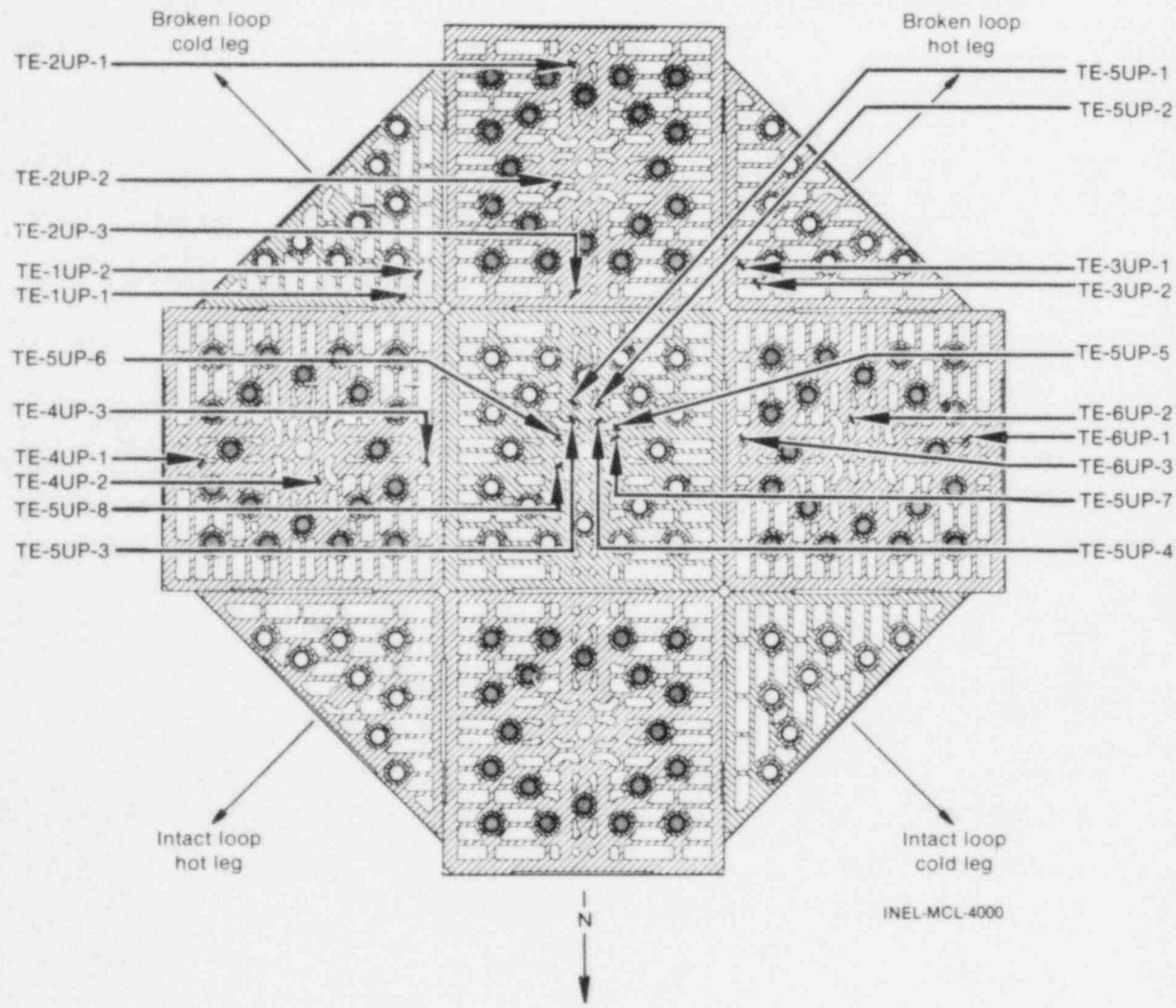

Figure 8-15. Thermocouple locations in LOFT core upper end boxes.

the time the pumps had stopped, decay heat generation had decreased to ~607 kW, corresponding to slightly more than 1% of full power. At this time, the vessel (upper plenum) fluid temperature was ~499 K and the pressure was about 2.61 MPa. Within 25 s after the pumps tripped, a cladding temperature excursion occurred, indicating an inadequate core cooling condition (see Figure 8-14). The core exit TC temperature remained unchanged for a period of about 51 s; however, during this period, the core cladding temperature displayed a significant increase at a rate of about 2.1 K/s. After this period, the core exit temperature began increasing at a much slower rate (0.44 K/s) for ~12 s until the core was quenched (see Figure 8-16). Even though the core experienced a heatup to over 620 K and was quenched, the upper plenum TCs did not provide indication of the event.

Fuel cladding, core exit, and upper plenum temperatures during LOFT intermediate-break Experiment L5-1[8-19] are shown in Figure 8-17. At ~108 s, with a decay heat generation of 1270 kW, corresponding to almost 3% of full power, and after the reactor scrammed and primary pumps tripped, the core cladding temperatures began to show an inadequate core cooling condition. At 115.6 s, the core cladding temperature began to increase at a rate of 2.6 K/s. At this time, the system upper plenum pressure was 3.34 MPa, and the fluid temperature was 508 K. The core exit TC temperature showed no significant response for about 19 s, after which a time reflood was initiated and it began to increase at 1.1 K/s and continued increasing at this rate for the duration that the inadequate core cooling condition existed.

Fuel cladding, core exit, upper plenum, and saturation temperatures for LOFT large-break Experiment L2-5[8-20] are shown in Figure 8-18. As discussed in Section 5.3.1, at ~212 s after the reactor scrammed and after subsequent reflood and core quench, and inadequate core cooling condition occurred with decay heat generation at 890 kW, corresponding to almost 2% of full power. At that time, the system pressure was about 0.39 MPa, the fluid temperature was about 410 K, and the core cladding temperature began to increase at a rate of 3.1 K/s. As the core cladding began to increase in temperature, the core exit TC temperature showed no significant change for about 160 s. After this period of no response, core reflood was initiated and the associated steam flow caused the exit

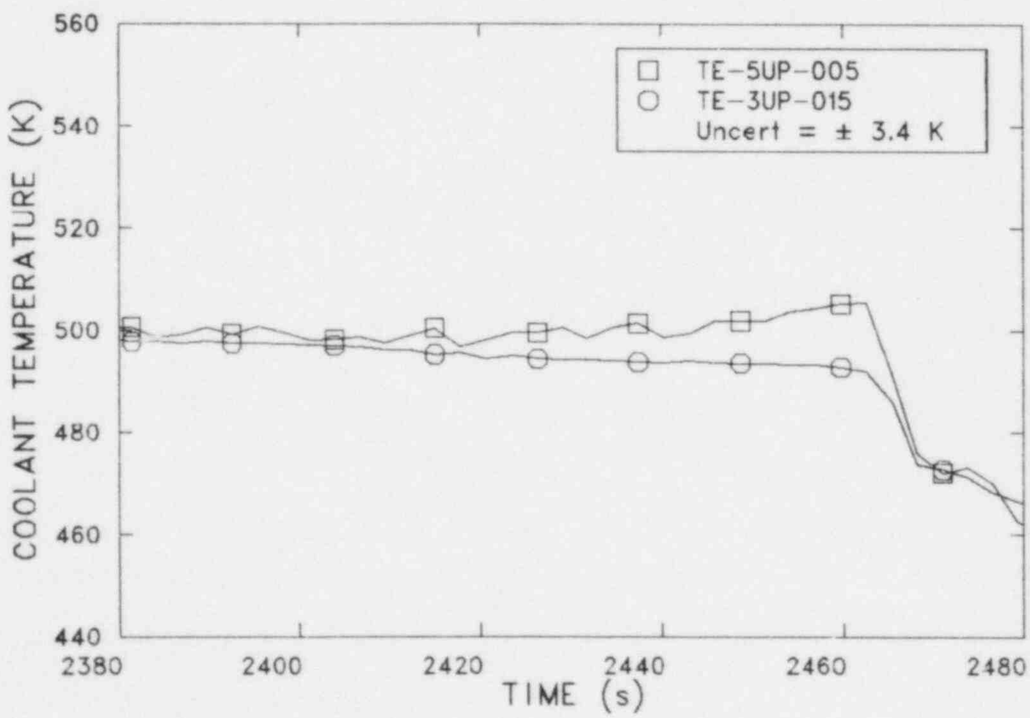

Figure 8-16. Coolant temperature at core exit for LOFT Experiment L3-6/L8-1.

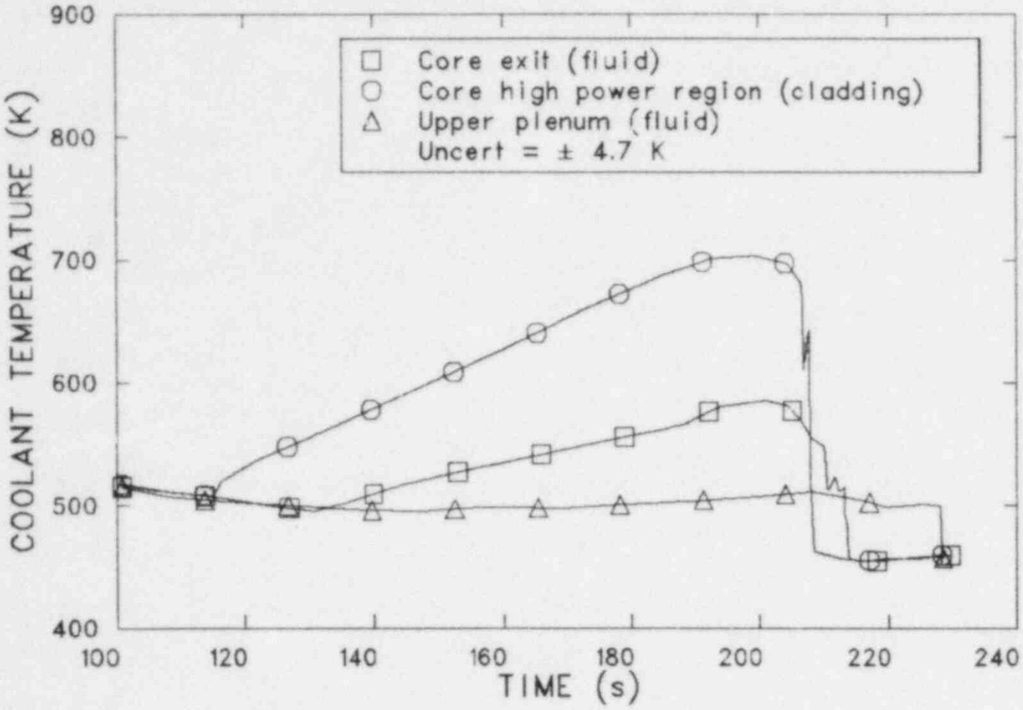

Figure 8-17. Fuel cladding, core exit, and upper plenum temperatures for LOFT Experiment L5-1.

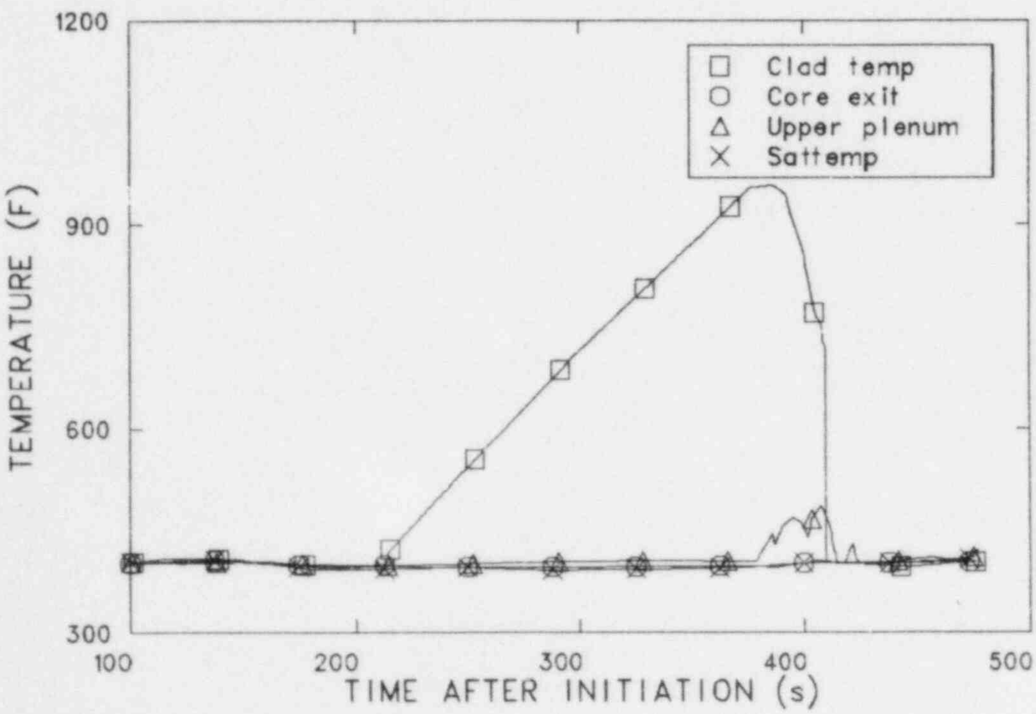

Figure 8-18. Fuel cladding, core exit, upper plenum, and saturation temperatures for LOFT Experiment L2-5.

temperature to show a sharp increase at an approximate rate of 3.2 K/s for about 33 s and then begin to fall off as additional core coolant was introduced.

The three LOFT experiments simulating accidents with distinct core recovery scenarios showed the following important results:

1. A significant time delay (19 to 160 s) existed in all cases between the start of a core uncovery (as determined by a core cladding temperature excursion) and any appreciable response from the core exit TCs.

2. Once core exit TCs responded, the rate of temperature increase was significantly lower than that of the core cladding TCs. Differences in the rate of increase between core exit and core cladding temperatures ranged from ~0.44 K/s for a small break (Experiment L3-6/L8-1) to 3.2 K/s for a large break (Experiment L2-5). In the large-break simulation, however, the rate was sustained for only a brief period.

8.4 Primary Coolant Pump Current Monitoring

Use of data from several LOFT experiments indicate that pump current is a very useful parameter for diagnosing small-break LOCAs and operational transients. During a LOCA, the pressurizer liquid level does not provide enough information to monitor primary coolant system inventory. Pressurizer liquid level can give ambiguous readings, as evidenced by the TMI accident, or read zero for extended periods. Based on the LOFT experience, pump current can be correlated with void fraction in the case of a small break or fluid density in the case of an overcooling transient.

During LOFT small-break Experiment L3-6,[8-18,8-21] pump current correlated well with density, as shown in Figure 8-19. During LOFT Experiment L6-7/L9-2,[8-22] an overcooling transient, there was a measurable increase in current as the primary coolant temperature dropped and fluid density increased. Figures 8-20 and 8-21 show coolant temperature and pump current, respectively, during

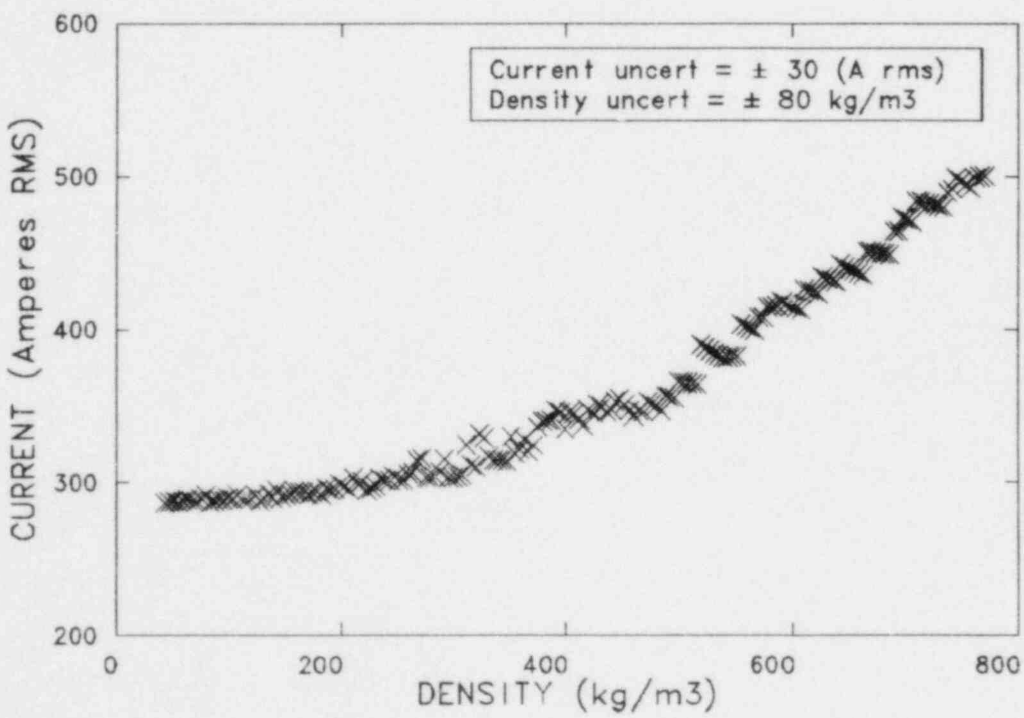

Figure 8-19. Pump current versus average coolant density for LOFT Experiment L3-6.

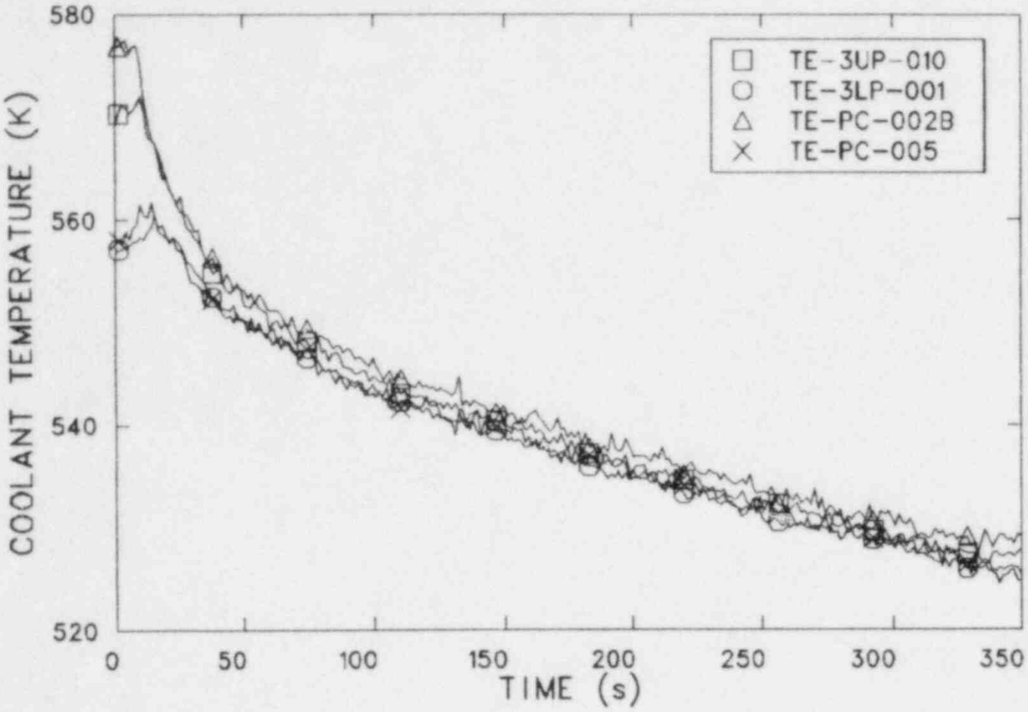

Figure 8-20. Temperature in reactor vessel upper and lower plenums and in intact loop hot and cold legs for LOFT Experiment L6-7.

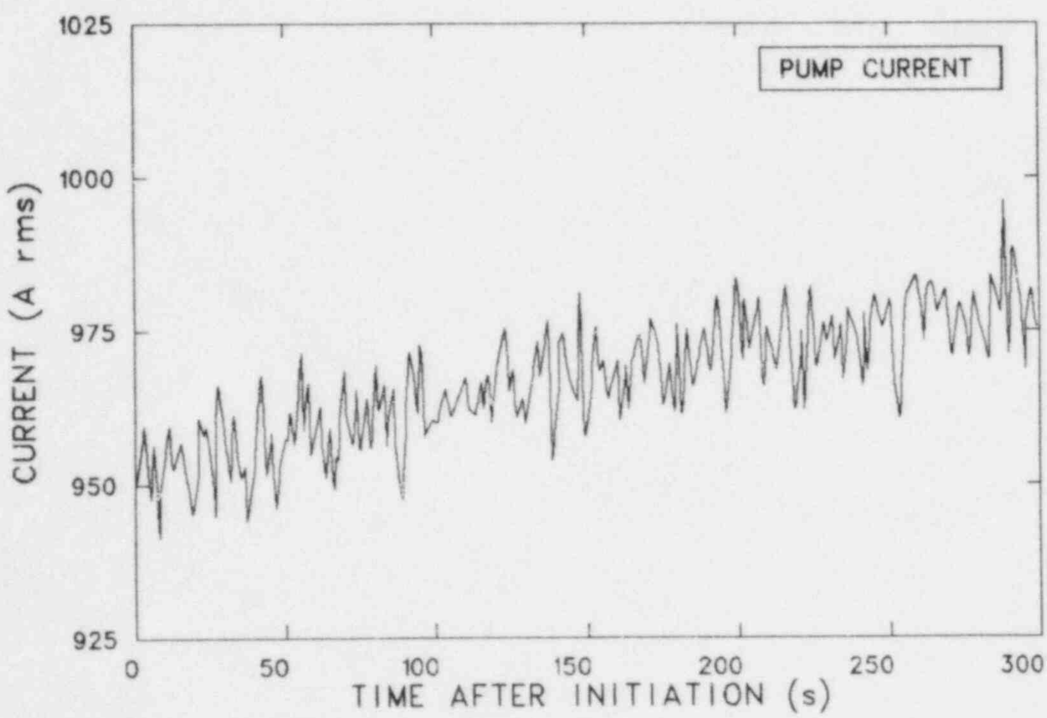

Figure 8-21. Primary coolant pump current for LOFT Experiment L6-7.

Experiment L6-7. During the first 300 s, as the coolant temperature dropped from ∿555 to 527 K, the average coolant density at the pump inlet increased from 748 to 794 kg/m^3, the pressure decreased from 14.75 to 10.05 MPa, and the pump currents increase from ∿950 to 975 A. A simultaneous pressure and temperature decrease in the primary system could be interpreted as a primary system leak. A corresponding increase in pump current, however, could only be caused by an increase in coolant density which is an indication of an overcooling transient.

Two recent analyses address the use of pump current (or power) monitoring as a diagnostic tool for accident management. The analysis reported in Reference 8-23 develops and verifies equations for pump power during two-phase flow conditions, and discusses a pump power display that operators could use to distinguish between a small-break LOCA and plant operational transients. An analysis reported in Reference 8-24 recommends turning off the primary coolant pumps during a small-break LOCA when pump current reaches a predetermined value in order to minimize coolant inventory loss.

8.5 References

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9. RULEMAKING

LOFT research results provided data for two areas of NRC rulemaking: Anticipated transient without scram (ATWS) and Appendix K to 10 CFR 50, ECCS Evaluation Models. The ECCS evaluation models and relevant LOFT results are discussed in detail in Section 5.

9.1 Anticipated Transient Without Scram

An ATWS is defined as any of a number of anticipated operating transients during which the reactor does not scram as designed. Concern regarding ATWS events has been a subject of extensive and continuing study by the NRC staff. The significance of ATWS for reactor safety is that some ATWS events could result in melting of the reactor fuel and the release of a large amount of radioactive fission products. This section summarizes proposed ATWS rules and presents relevant LOFT results that can be used by NRC in preparing their final ATWS rule.

9.1.1 Summary of Proposed ATWS Rules. Three proposed rules for the reduction of risk from ATWS events in commercial PWRs have been published in the Federal Register. The first is a petition for rulemaking filed by 20 utilities,[9-1] the second and third are alternate ATWS rules proposed by the NRC.[9-2]

9.1.1.1 ATWS Rule Proposed by the Utilities. The utilities proposed that for light-water nuclear power reactors for which a construction permit application was filed as of the effective date of the rule, the ATWS rule should be as follows:

1. For boiling water reactors manufactured by the General Electric Company, provide: (a) a means to initiate a reactor scram upon receipt of a signal indicative of an ATWS event; (b) an independent, redundant and diverse electrical means to initiate a reactor scram upon receipt of a signal indicative of an ATWS event; and (c) a scram discharge volume system designed and installed such that it will have sufficient capacity to receive water exhausted by a full reactor scram.

2. For pressurized water reactors manufactured by Combustion Engineering Inc. and the Babcock & Wilcox Company, provide: (a) an alternate means to shut down the reactor that is diverse from and redundant to the electrical portion of the reactor protection system up to but not including the trip breakers, and (b) an automatic initiation of auxiliary feedwater independent of the reactor protection system.

3. For pressurized water reactors manufactured by the Westinghouse Electric Corporation, provide automatic initiation of turbine trip and auxiliary feedwater independent of the reactor protection system.

The utilities proposed that an ATWS rule for new plants be postponed until the degraded core rulemaking was completed, because the issues are related.

9.1.1.2 First ATWS Rule Proposed by the NRC. The first NRC-proposed ATWS rule includes the following provisions:

1. Increase the reliability of the reactor trip portion of the reactor protection system by the addition of supplementary protection systems that would be independent and diverse.

2. Improve the capability to mitigate ATWS events by providing actuation circuitry that is separate from the reactor protection system for such systems as the primary system relief valves, turbine trip, and auxiliary feedwater in PWRs.

3. Increase the capability to mitigate ATWS events by increasing the pressure relief capacity in the reactor coolant system.

9.1.1.3 Second ATWS Rule Proposed by the NRC (the Hendrie Rule). Each light-water-cooled commercial power reactor licensee shall establish and maintain a reliability assurance program for functions associated with the prevention and mitigation of an ATWS employing state-of-the-art methods and procedures to identify vulnerabilities to failure. Each licensee is responsible for the implementation of improvements to reduce ATWS risk. Defense in depth must be maintained by operating commercial power reactors only in modes that afford an

opportunity to learn from experience with ATWS events without severe radioactivity releases. Specific acceptance criteria are delineated as follows:

1. An initial reliability assurance program that includes:

   a. An analysis and classification of the principal determinants of the radiological severity of each class of ATWS accident sequences

   b. Training of operators in the diagnosis and prognosis of the several ATWS accident sequences

   c. Analysis of hypothetical errors in test and maintenance procedures for systems where reliability is important to ATWS prevention or mitigation

   d. An analysis of the blindspots in the experience base with systems important to ATWS prevention or mitigation through which reliability defects might escape detection for considerable periods of time

   e. An analysis of the suceptibility of the plant to common cause failure.

2. A continuing reliability assurance program that includes the following:

   a. Configuration control for design procedures and technical specifications

   b. Procedures for updating portions of the initial reliability assurance program

   c. An experience feedback system to review data from similar plants.

3. Design and operation for ATWS tolerance as follows:

   a. For boiling water reactors, this includes providing equipment to automatically trip the reactor coolant recirculation pumps and to deliver liquid reactivity poison to bring the reactor to hot shutdown. In addition, a reliable scram discharge volume system shall be provided.

   c. For pressurized water reactors, this includes providing for the prompt automatic startup of the auxiliary feedwater system under circumstances indicative of a transient entailing loss of main feedwater and a failure to scram, ensuring that the instruments necessary for diagnosis into recovery are not disabled, and ensuring that the HPIS valves function after the limiting ATWS transient.

4. Each commercial power reactor licensee shall prepare, submit for reviews and approval, and implement Limiting Conditions of Operation that proscribe operation in, and mandate expeditious retreat from, operation under conditions that compromise the ATWS tolerance of the plant. Limiting Conditions of Operation should also minimize operation under conditions in which the ATWS tolerance of the plant would be severely tested by a limiting ATWS event. Consideration of the prevailing plant parameters as well as equipment operability is appropriate in the Limiting Conditions of Operation.

5. The ATWS tolerance of a plant is inadequate if any of the more limiting transients, followed by a total failure of the scram system, result in any one of the following:

   a. Containment pressure or temperature above the design values

   b. Loss of coolable geometry in the core

   c. Releases of radioactive material that may realistically cause any offsite prompt fatalities or serious offsite property damage.

9.1.1.4 Final ATWS Rule. The final ATWS rule was issued by the NRC on June 19, 1984.

(a) Applicability. The requirements of this section apply to all commercial light-water-cooled nuclear power plants.

(b) Definition. For purposes of this section, "Anticipated Transient Without Scram" (ATWS) means an anticipated operational occurrence as defined in Appendix A of

this part followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion 20 of Apppendix A of this part.

(c) Requirements

(1) Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system.

(2) Each pressurized water reactor manufactured by Combustion Engineering or by Babcock and Wilcox must have a diverse scram system from the sensor output to interruption of power to the control rods. This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system (from sensor output to interruption of power to the control rods).

(3) Each boiling water reactor must have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI system must have redundant scram air header exhaust valves. The ARI must be designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

(4) Each boiling water reactor must have a standby liquid control system (SLCS) with a minimum flow capacity and boron content equivalent in control capacity to 86 gallons per minute of 13 weight percent sodium pentaborate solution. The SLCS and its injection lication must be designed to perform its function in a reliable manner. The SLCS initiation must be automatic and must be designed to perofrm its function in a reliable manner for plants granted a construction permit after [insert the effective date of this ammendment], and for plants granted a construction permit prior to [insert the effective date of this amendment] that have already been designed and built to include this feature.

(5) Each boiling water reactor must have equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner.

(6) Information sufficient to demonstrate to the Commission the adequacy of items in paragraphs (c)(1) through (c)(5) of this section shall be submitted to the Director, Office of Nuclear Reactor Regulation.

(d) Implementation. By 180 days after the issuance of the QA guidance for non-safety related components each licensee shall develop and submit to the Director of the Office of Nuclear Reactor Regulation a proposed schedule for meeting the requirements of paragraphs (c)(1) through (c)(5) of this section. Each shall include an explanation of the schedule along with a justification if the schedule calls for final implementation later than the second refueling outage after [insert the effective date of the amendment], or the date of issuance of a license authorizing operation about 5 percent of full power. A final schedule shall then be mutually agreed upon by the Commission and licensee.

9.1.2 Summary of Relevant LOFT Data. Two ATWS events were simulated in the LOFT facility that provided data regarding system performance and an assessment of the capabilities of the RELAP5 computer code to predict system performance during an ATWS event. The first ATWS event was induced by a loss of feedwater (Experiment L9-3) and the second ATWS event was induced by a loss of offsite power (Experiment L9-4).

9.1.2.1 ATWS Event Induced by Loss of Feedwater. Experiment L9-3 simulated an ATWS induced by a loss of feedwater. The experiment extended through the peak pressure region of the transient (∿10 min) and the recovery of the plant from the ATWS.

The principal objectives for Experiment L9-3 were to:[9-3]

1. Provide experimental data for benchmarking PWR vendors' ATWS computer codes as required by the NRC proposed ATWS rule
2. Evaluate alternate methods of achieving long-term shutdown (without the insertion of control rods) following an ATWS event, to address concerns defined in the proposed NRC staff rule
3. Determine the transient reactor power by using available neutron flux instrumentation and measured core thermal-hydraulic parameters to assess the applicability of the point kinetics model used in predicting transient reactor power
4. Determine the steam generator secondary dryout behavior and its effect on the primary system response characteristics
5. Determine the two-phase and subcooled flow characteristics of the experimental pressurizer PORV and SRV at high pressures ($\geq$17 MPa).

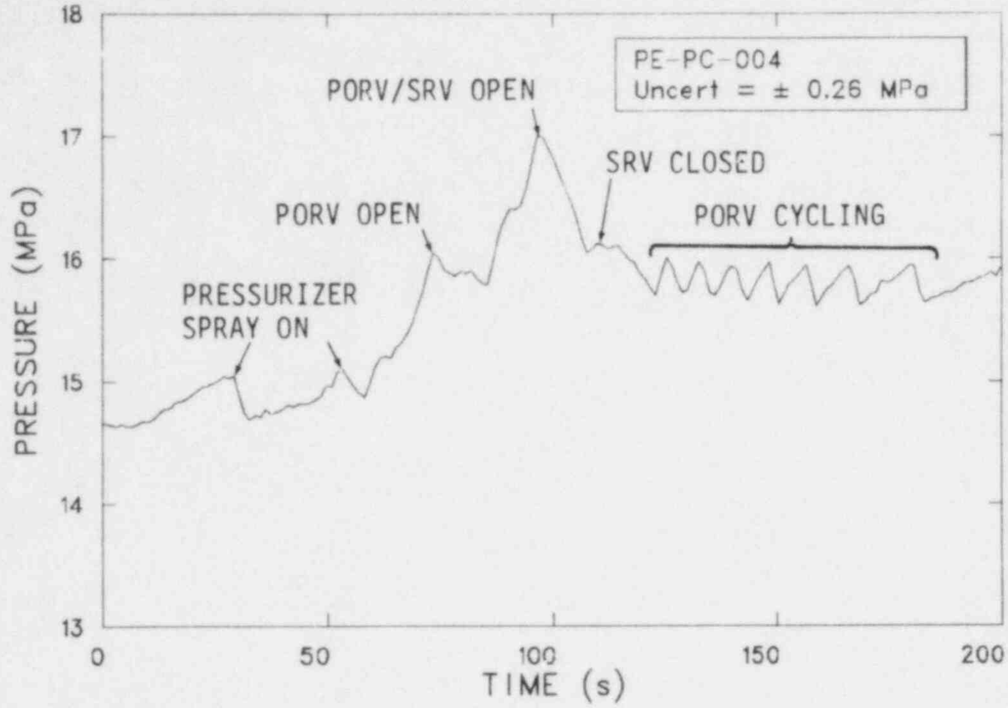
For experiment L9-3, a two-position actuator relief valve was installed on the pressurizer to simulate a scaled PORV and a scaled SRV. In the first position (designated as "PORV"), the valve was sized to relieve 0.66 kg/s of saturated steam at an upstream pressure of 16.2 MPa. This relief capacity was scaled to the minimum PORV capacity of a generic Westinghouse PWR.[9-4] In order to provide benchmark data on the most severe pressure transient resulting from an ATWS for code assessment, it was desirable to approach a maximum pressure of 18.6 MPa for the LOFT system. To do this, the relief capacity in the second position (designated "PORV/SRV") was sized to obtain a flow of 1.52 kg/s saturated steam at an upstream pressure of 17.2 MPa. This flow represented flow through the combined PORV/SRV. Details on system configuration and experimental procedures for Experiment L9-3 are given in Reference 9-5.

Experiment L9-3 was initiated by turning off the main feedwater pump. The steam generator steam control valve was closed manually at 67.3 $\pm$ 0.2 s. The experiment PORV opened at 73.8 $\pm$ 0.2 s, and the experiment SRV opened at 96.8 $\pm$ 0.2 s.

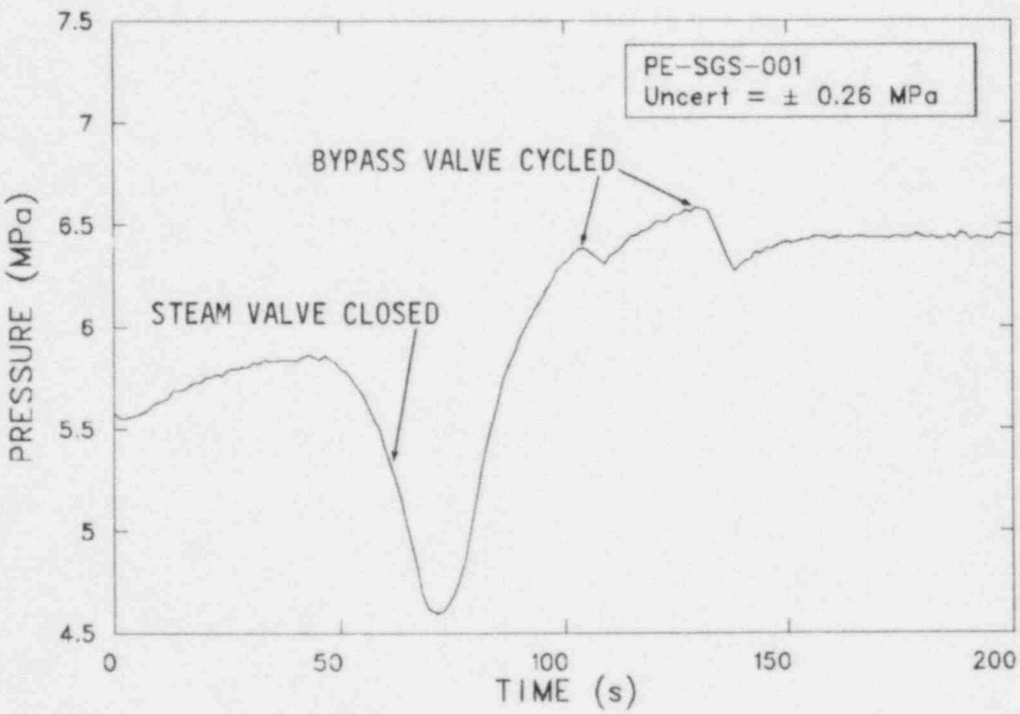
At ∿600 s, the recovery procedure was initiated by starting HPIS injection, starting the steam generator auxiliary feedwater and makeup pumps, and opening the experiment PORV. A supply (984 $\pm$ 38 L) of 7000 $\pm$ 200-ppm boron was available to the HPIS pump, which injected it to the downcomer at a rate of 0.38 $\pm$ 0.06 L/s. The steam generator feed was manually controlled to keep the primary system temperature between 583 and 594 K. The main steam bypass valve was opened as necessary to keep the steam generator pressure between 6.38 and 6.63 MPa. The experiment PORV was cycled as necessary to keep the primary system pressure between 15.0 and 15.6 MPa. Throughout the experiment, the control rods remained withdrawn and the primary coolant pumps were running.

Upon loss of feedwater to the steam generator, degradation of primary-to-secondary heat transfer began to occur. The primary coolant began a very slow heatup, and the resulting liquid swell increased the primary system pressure until the pressurizer spray setpoint was reached, as shown in Figure 9-1. The pressurizer spray effectively controlled pressure for the first 60 s, while the degradation in primary-to-secondary heat transfer was small.

At about 60 s, the liquid level in the secondary side had decreased such that the steam generator heat transfer was no longer sufficient to remove reactor power, and the primary coolant temperature started to increase rapidly, as shown in Figure 9-2. Due to moderator temperature feedback, the reactor power started to decrease rapidly, as shown in Figure 9-3. Between 50 to 100 s, the primary coolant temperature increased from 569 to 590 K under an average energy imbalance (heat source minus heat sink) of nearly 13 MW (26% of initial power). The resultant liquid swell increased the primary system pressure so rapidly that it could not be controlled by the PORV. The PORV cycled once with steam flow, then stayed open. AT ∿96 s, the pressure reached the SRV setpoint and the valve opened, more than doubling the mass flow rate, see Figure 9-4. By this time, the reactor power had decreased to ∿15 MW and the relief capacity of the combined PORV/SRV flow of 4.5 kg/s was sufficient to rapidly decrease system pressure (see Figure 9-1). By the time the SRV closed (at 107 s),



a. Pressure in primary system.



b. Pressure in secondary system.

Figure 9-1. Pressure in primary and secondary systems during first 200 s of LOFT Experiment L9-3.

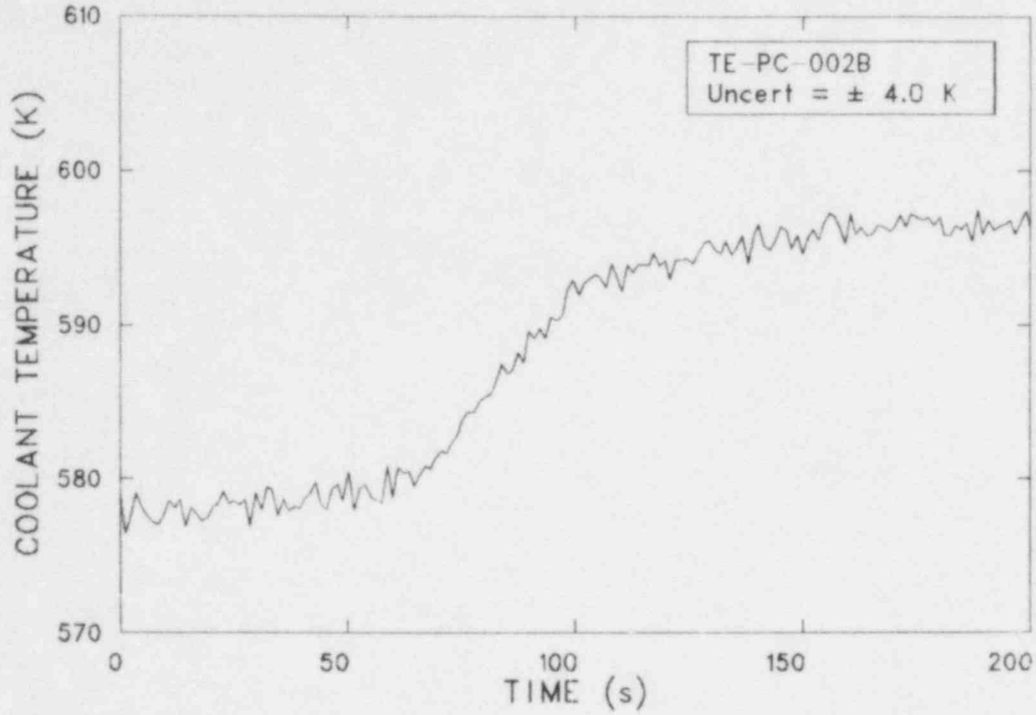


Figure 9-2. Temperature in intact loop hot leg for LOFT Experiment L9-3.

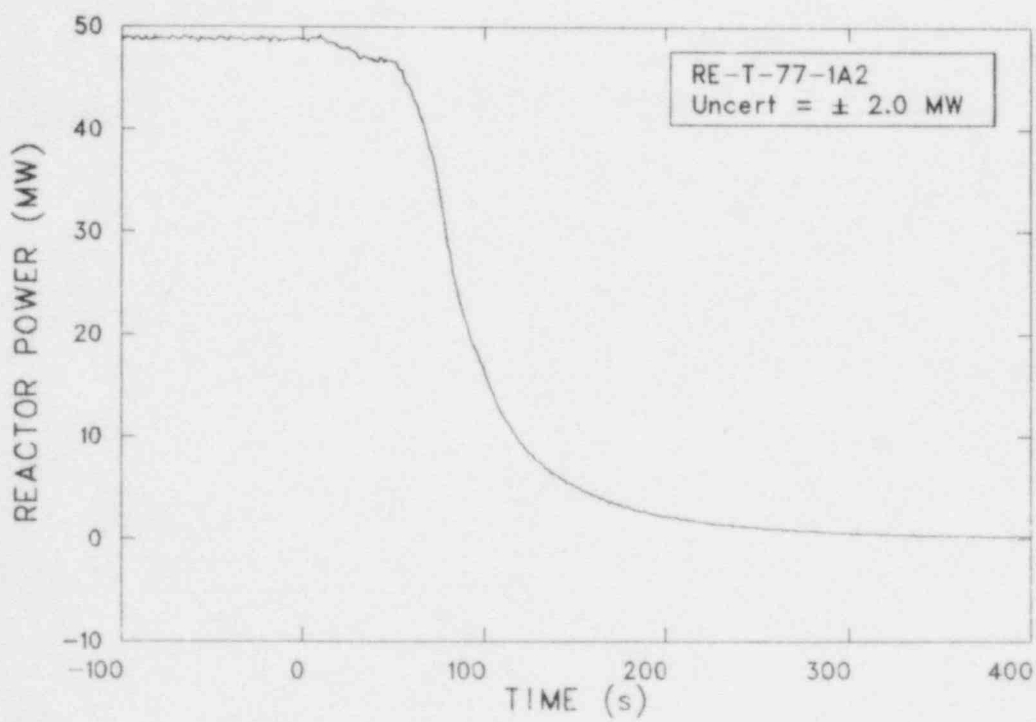


Figure 9-3. Reactor power for LOFT Experiment L9-3.

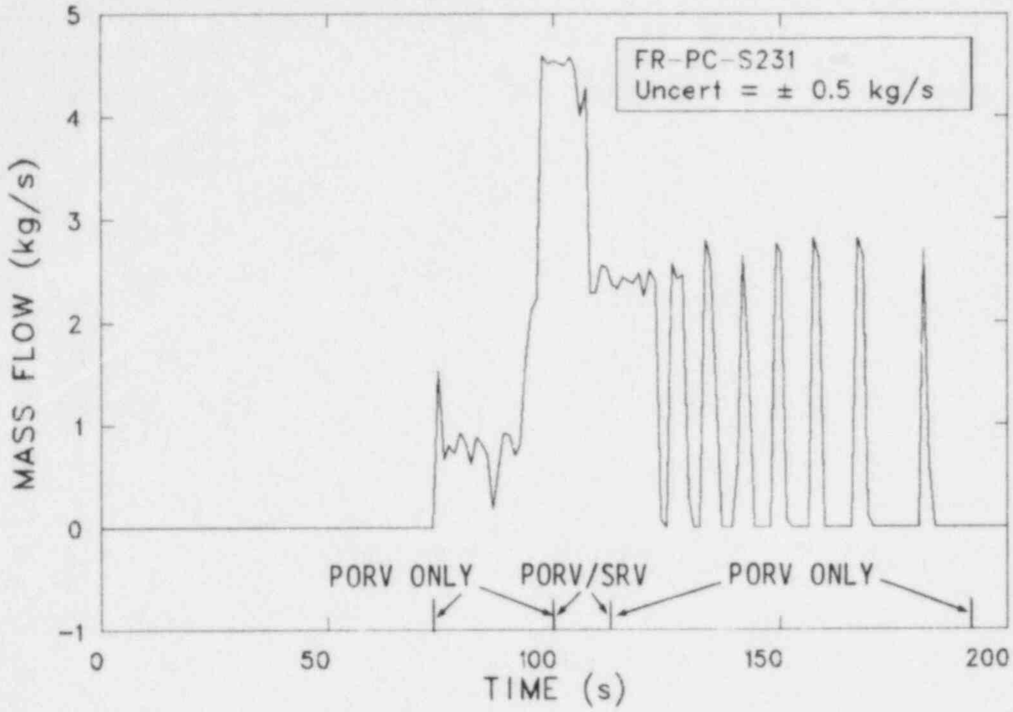


Figure 9-4. Mass flow out the PORV and combined PORV/SRV for LOFT Experiment L9-3.

the power had decreased further to ~13 MW, and the primary coolant swell was balanced by the PORV liquid flow alone.

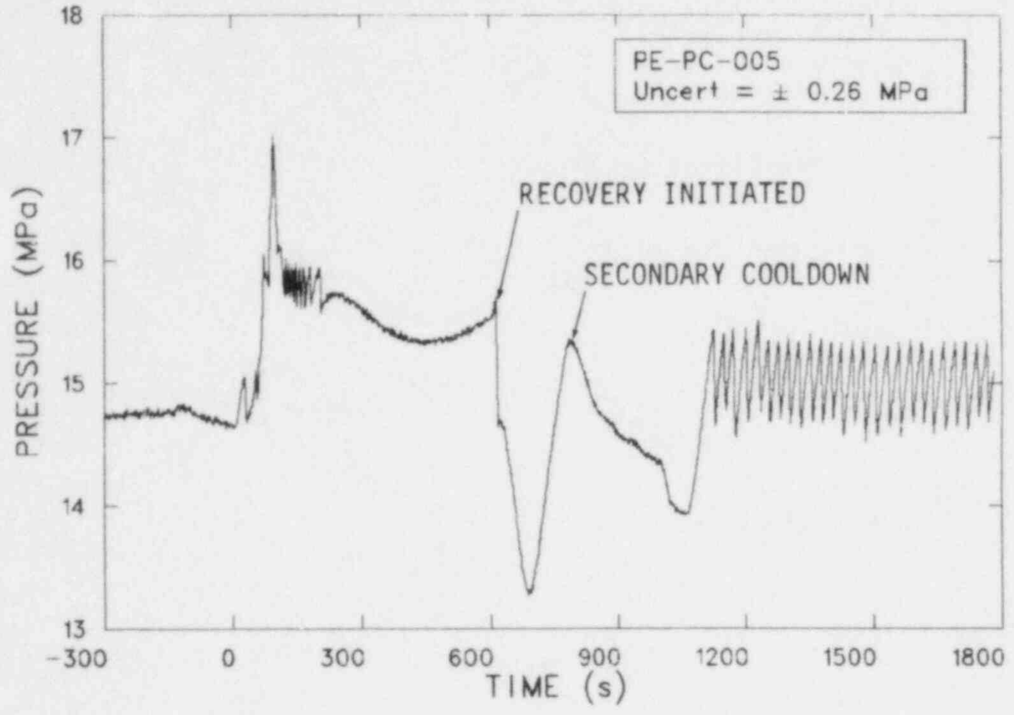
The LOFT system stabilized with a gradual primary system heatup. Pressure was controlled by cycling the PORV, until the operator-controlled recovery procedure was initiated at ~600 s. The decrease in reactor power because of the increasing coolant temperatures throughout Experiment L9-3 was as predicted, based on previous physics testing, and the reactor was shut down by 180 s and remained shut down through the remainder of the transient.

At ~600 s, the reactor operators initiated an HPIS flow of a highly concentrated boron (~7200 ppm) solution. This injection was sufficient to keep the reactor subcritical, with the control rods still at the steady state system operating positions during the subsequent primary system cooldown and depressurization.

As soon as HPIS flow was verified, injection of water into the steam generator was initiated to cool down the primary system, and the experiment PORV was latched open to depressurize the primary system. This resulted in a slight primary coolant cooldown and a rapid depressurization, as seen in Figure 9-5. The PORV was then shut and secondary feedwater flow was terminated, resulting in a partial pressure recovery as the primary coolant heated up again.

At ~770 s, an operator-controlled cooldown by way of secondary feed and bleed was initiated. This continued until the primary coolant temperature reached 583 K at 1090 s. Subsequent to this, HPIS injection continued and the PORV was manually cycled to control primary system pressure until experiment termination at 1960 s. The control rods were inserted at 1990 s.

The qualitative plant behavior during Experiment L9-3 was, in general, correctly predicted (see preexperiment RELAP5 predictions in Reference 9-6). There were, however, large differences between measured and calculated magnitudes, especially primary system pressure. Figure 9-6 shows a comparison between predicted and measured primary system pressure. As shown, during the first 50 s the actual pressure rise exceeded the prediction. In the prediction, the pressurizer spray did not initiate until after 50 s, when significant primary-to-secondary heat transfer degradation had occurred. Since the predicted primary coolant temperature rise was correct during this time, as shown in Figure 9-7, the difference in pressure response is attributed to a nonequilibrium condition (superheating of the steam bubble) in the pressurizer that was not calculated.



a. Pressure in primary system.

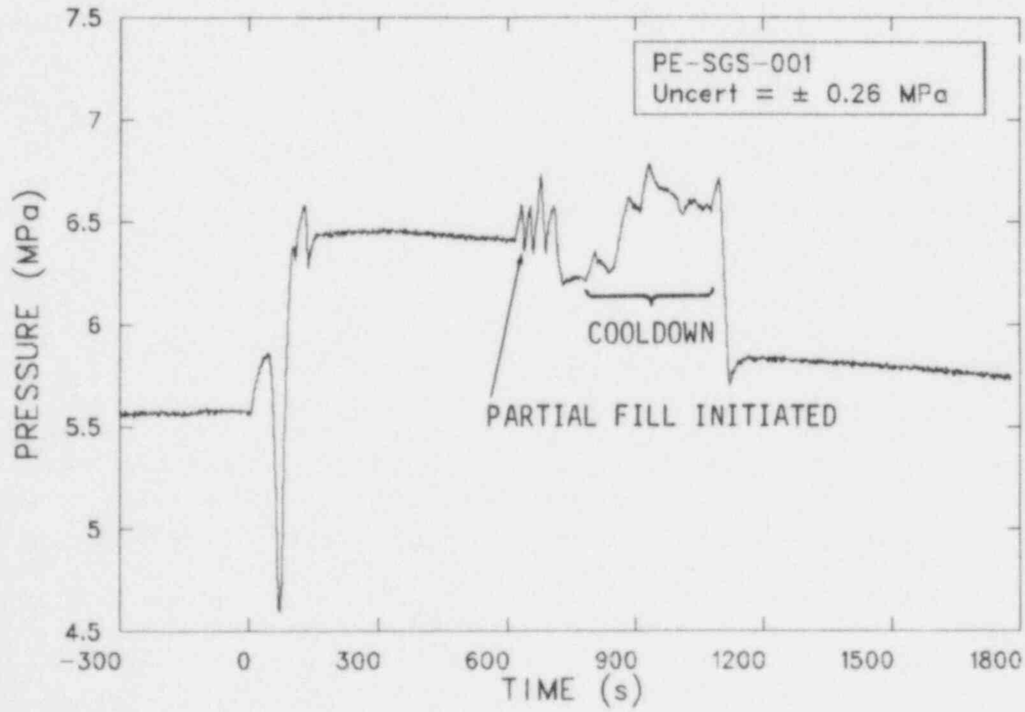


Figure 9-5. Pressure in primary and secondary systems for LOFT Experiment L9-3.

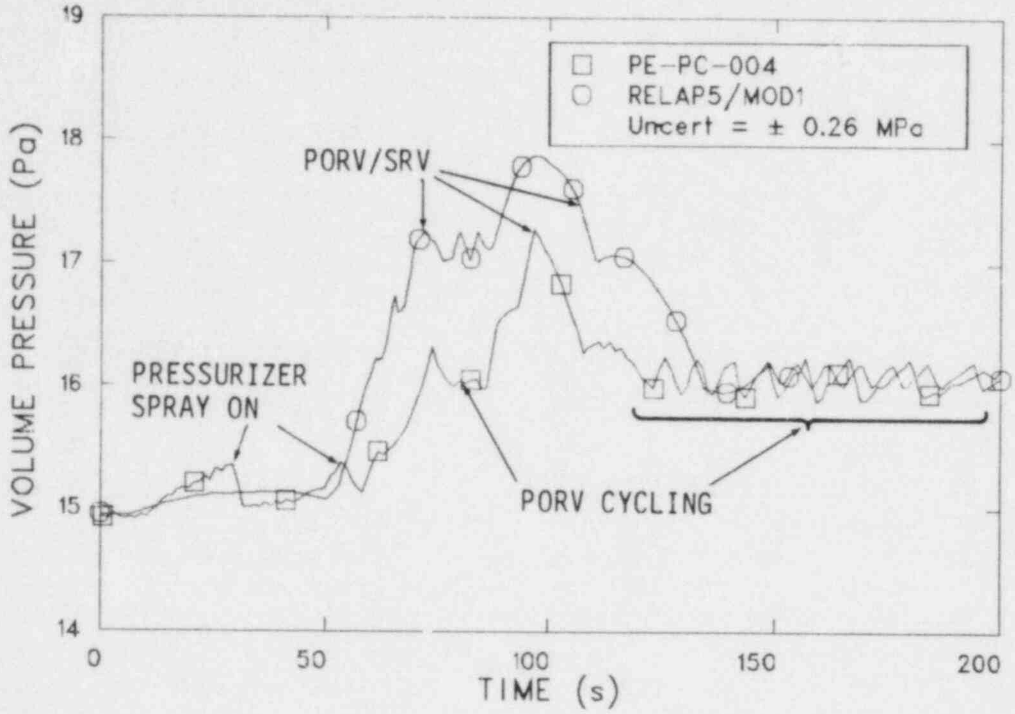


Figure 9-6. Measured and predicted pressure in primary system for LOFT Experiment L9-3.

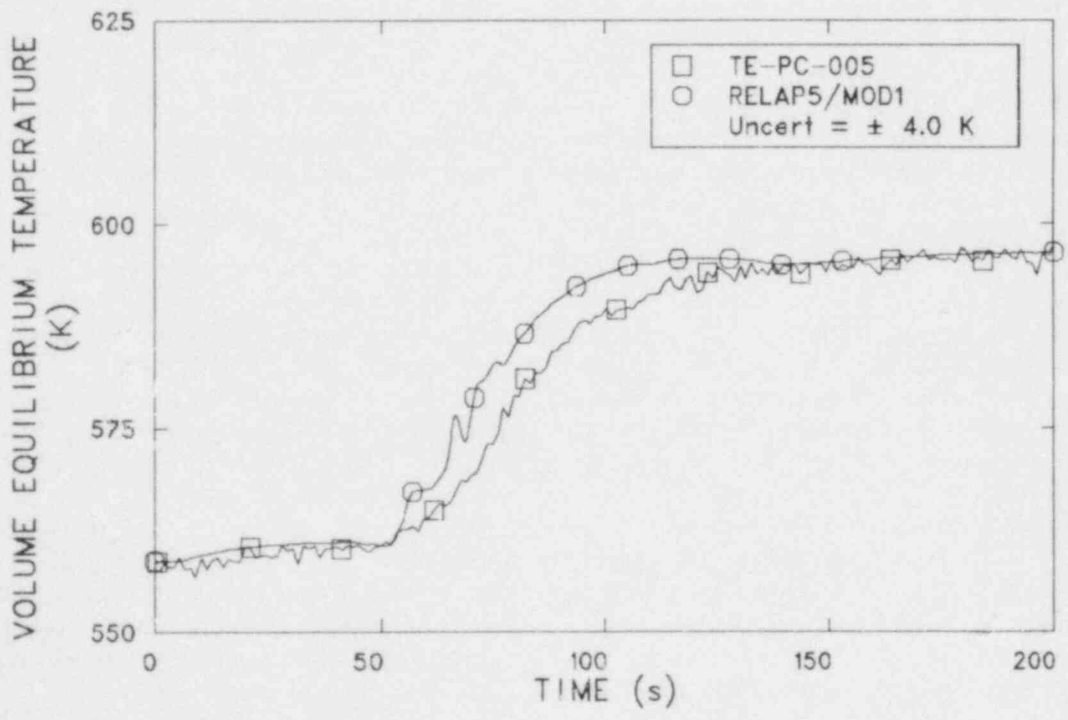


Figure 9-7. Measured and predicted temperature in intact loop cold leg for LOFT Experiment L9-3.

In the 50- to 100-s time interval (refer to Figure 9-3), the moderator temperature coefficient resulted in an average measured power reduction of -1.61 MW/K. The calculated response was -1.58 MW/K, which is only 2% different than the measured value.

During the rapid primary coolant temperature rise between 50 and 100 s, the measured primary coolant temperature and pressure were lower than predicted. These differences are attributed to the steam generator primary-to-secondary heat transfer which was larger than calculated. The larger-than-calculated heat transfer may in part be due to the main steam isolation valve being shut earlier than specified which resulted in a larger secondary coolant inventory. Also, the pressure response difference is attributed to the difference between measured and calculated flow out the PORV and combined PORV/SRV. The PORV and SRV flow rates were specified as 0.66 kg/s of steam at 16.2 MPa in the PORV position and 1.52 kg/s of steam at 17.2 MPa in the combined PORV/SRV position. The corresponding flow area inputs to RELAP5 were 2.214×10^{-5} m^2 and 4.718×10^{-5} m^2, respectively. However, the measured flow rates in Experiment L9-3 were 0.80 kg/s of steam and 2.44 kg/s of liquid at 16 MPa in the PORV position and 4.46 kg/s of liquid at 17 MPa in the combined PORV/SRV position.

9.1.2.2 ATWS Event Induced by Loss of Offsite Power

Experiment L9-4 simulated an ATWS initiated by a loss of offsite power. The transient was initiated by tripping the primary coolant pump and the secondary system main feedwater pump and by closing the main stream control valve. During the transient, the PORV was assumed to be inoperative due to the loss of offsite power. Only the SRV was operative. Auxiliary feedwater was started 10 s after transient initiation to simulate the startup of the diesel generator in a commercial PWR (it should be noted that the auxiliary feedwater could have simulated the startup of the steam turbine-driven feedwater pumps). The major objectives for Experiment L9-4 were to:[9-7]

1. Provide experimental data for benchmarking PWR vendors' ATWS computer codes as required by the NRC proposed ATWS rule

2. Evaluate alternate methods of achieving long-term shutdown (without the insertion of control rods) following an ATWS event, to address concerns defined in the proposed NRC staff rule

3. Determine the effect of primary coolant pump operation on initial system response and peak pressure by comparing results from Experiment L9-4 (pumps tripped) with results from Experiment L9-3 (pumps running)

4. Provide data for analysis of the effect of natural circulation cooling capability under high-power conditions

5. Provide data to evaluate the capabilities of the computer codes to predict the fluid conditions (temperature, pressure, and quality) in both the primary and the secondary systems and to evaluate the adequacy of point kinetics assumptions used in prediction of reactor power levels

6. Evaluate the adequacy of the proposed procedure for a rapid and safe recovery of the LOFT plant without insertion of the control rods.

For Experiment L9-4, the SRV relief capacity was sized to obtain a flow of 1.52 kg/s saturated steam at an upstream pressure of 17.2 MPa. This flow was scaled to represent the flow through three SRVs in a four-loop commercial PWR. The PORV was assumed to have failed closed due to loss of instrument air. Details on system configuration and experimental configuration for Experiment L9-4 are given in Reference 9-8.

After the primary coolant pumps tripped, the decreasing primary system flow resulted in a rapid core coolant temperature increase. This initial heatup caused the reactor power to decrease rapidly under the influence of moderator density feedback mechanisms. The transition from forced flow to natural circulation flow began during the pump coastdown, and natural circulation flow was fully established by ~80 s after experiment initiation. As the primary-to-secondary heat transfer degraded, due to loss of flow, the volumetric expansion of the primary coolant resulted in a system pressurization to the SRV opening setpoint (17.2 MPa), by 18.5 s, and the SRV cycled four times to control pressure. By ~200 s, steam generator boiloff had caused sufficient degradation of heat transfer such that system

pressure again began to increase, and the SRV opening setpoint was reached again. The pressurizer liquid level increased to the top of its indicating range and by the fourth subsequent SRV cycle at 500 s, the flow out the SRV was composed of single-phase liquid. The SRV continued cycling an additional four times, after which the steam generator heat transfer and environmental losses were sufficient to remove the reduced core power and no further SRV cycling occurred. By ~1000 s, primary system pressure control via the pressurizer was regained and the depressurization rate decreased. Subsequently, the pressurizer liquid level decreased into the indicating range. Plant operation continued with pressure being controlled by the pressurizer. The core heat generation was sufficiently small that it could be dissipated by the auxiliary feedwater flow in the secondary system. At 1507 s, Experiment L9-4 was terminated by a reactor scram.

After the primary coolant pumps were tripped, the mass flow rate in the primary system decreased, and a transition from forced convection flow to natural circulation flow began. The loss of primary system flow caused a rapid increase in hot leg temperature, as shown in Figure 9-8. This resulted in the rapid decrease in power shown in Figure 9-9 at ~2 s, which was due to moderator temperature reactivity feedback. The volumetric expansion of the primary coolant resulted in an increasing system pressure until the SRV opening setpoint (see Figure 9-10) was reached (~19 s) and the SRV cycled to control pressure. By ~200 s, the heat transfer from the primary to the secondary had degraded sufficiently to cause the primary system pressure to increase, and the SRV cycled again to control pressure. By ~600 s, the steam generator heat transfer and environmental heat losses were sufficient to remove the reduced core heat generation, and no further SRV cyclings occurred.

The depressurization rate decrease that occurred at ~1000 s (see Figure 9-10) was a result of bubble reformation in the pressurizer. The system stabilized with the pressure being controlled by the pressurizer and heat generation from the core dissipated by the auxiliary feedwater flow in the secondary system. This stabilized operation continued until the reactor was scrammed at 1507 s, concluding the experiment.

Power to the main feedwater pump was terminated at time zero coincident with the primary coolant pump trip. At 1.9 s, the main steam control valve started to close, and required 11.3 s to close completely. At 10.8 s, an auxiliary feedwater flow of 0.5 L/s was established. The steam generator secondary side pressure was manually controlled between 6.63 and 6.97 MPa by steam bleed

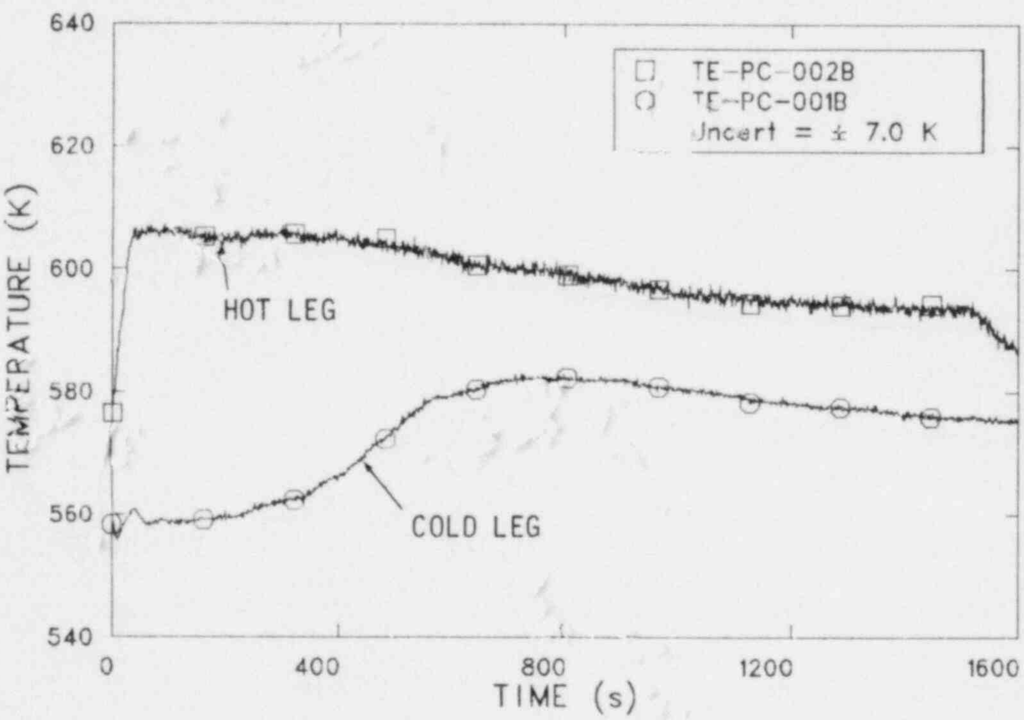


Figure 9-8. Temperature in intact loop hot and cold legs for LOFT Experiment L9-4.

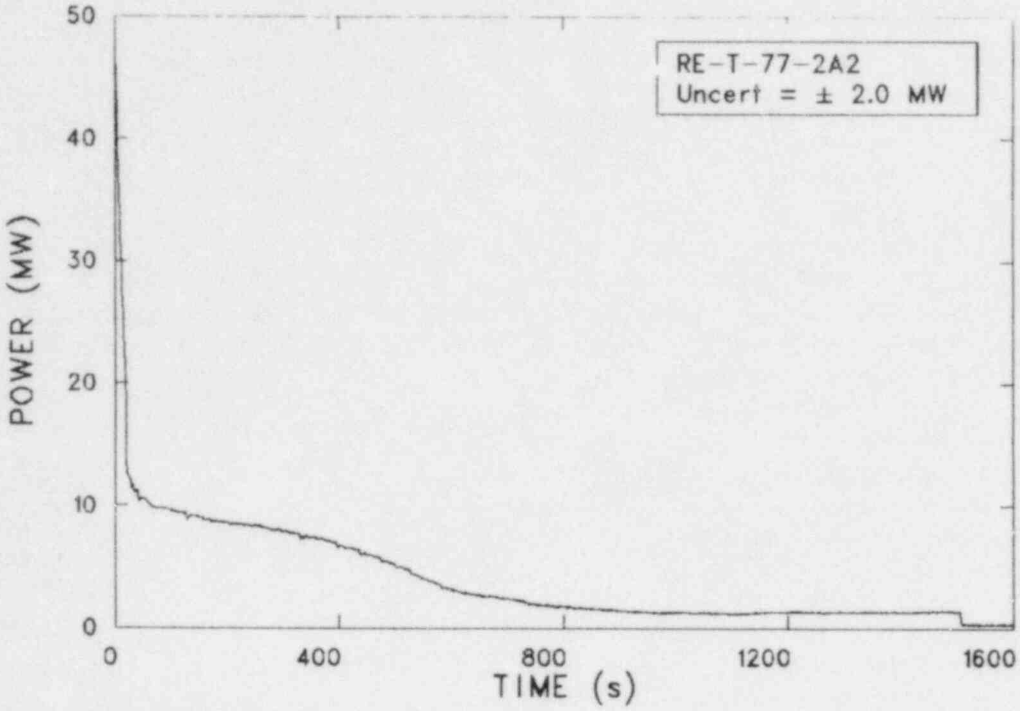


Figure 9-9. Reactor power for LOFT Experiment L9-4.

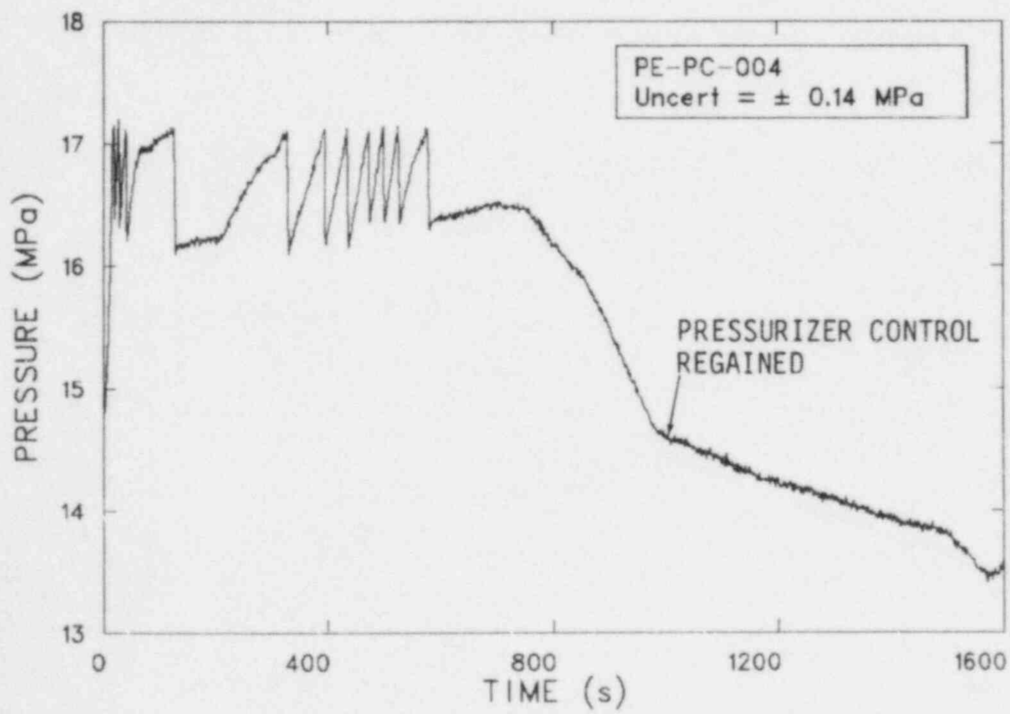


Figure 9-10. Pressure in pressurizer for LOFT Experiment L9-4.

through the main steam bypass valve, see Figure 9-11.

The collapsed liquid level measured in the steam generator downcomer decreased to the elevation of the top of the tubes at ~50 s, as shown in Figure 9-12. As the liquid level decreased, the primary-to-secondary heat transfer began to degrade such that at ~200 s, the primary system pressure increased rapidly. By 500 s, the steam generator liquid level had decreased to below the bottom of the indicating range, or 0.25 m above the top of the tube sheet. The auxiliary flow kept the steam generator from completely drying out, and provided a heat sink to maintain natural circulation throughout the transient.

The predicted plant behavior[9-9] agreed qualitatively with the measured data (see Reference 9-10 for an assessment of the preexperiment prediction). Figure 9-13 shows a comparison between predicted and measured hot and cold leg temperatures. The rate of the initial hot leg temperature increase was predicted well. There was a 5-K overprediction at 40 s, followed by good agreement between the predicted and measured data until ~700 s.

The cold leg temperature trend was predicted well qualitatively during the initial 300 s, although there was an offset in magnitude. The heat sink effectiveness of the steam generator degraded at approximately the same time in the prediction as was observed in the experiment. As was shown by the rapid increase in the predicted cold leg temperature, however the predicted loss of heat sink was much more abrupt than observed in the experiment. Beyond 700 s, both hot and cold leg temperatures were predicted to remain constant, while the measured data showed a slow cooldown. The primary system energy balance calculation in the prediction did not predict this cooldown. The intact loop hot leg fluid velocity during natural circulation was calculated reasonably well by RELAP5, as shown in Figure 9-14.

The difference between predicted and measured reactor power shown in Figure 9-15 was caused by the differences between calculated and measured primary coolant pump behavior which resulted in lower measured flow during the first 350 s, see Figure 9-14. The lower flow caused a slightly higher core coolant temperature, resulting in an increased negative moderator temperature reactivity and subsequent lower power. These differences were

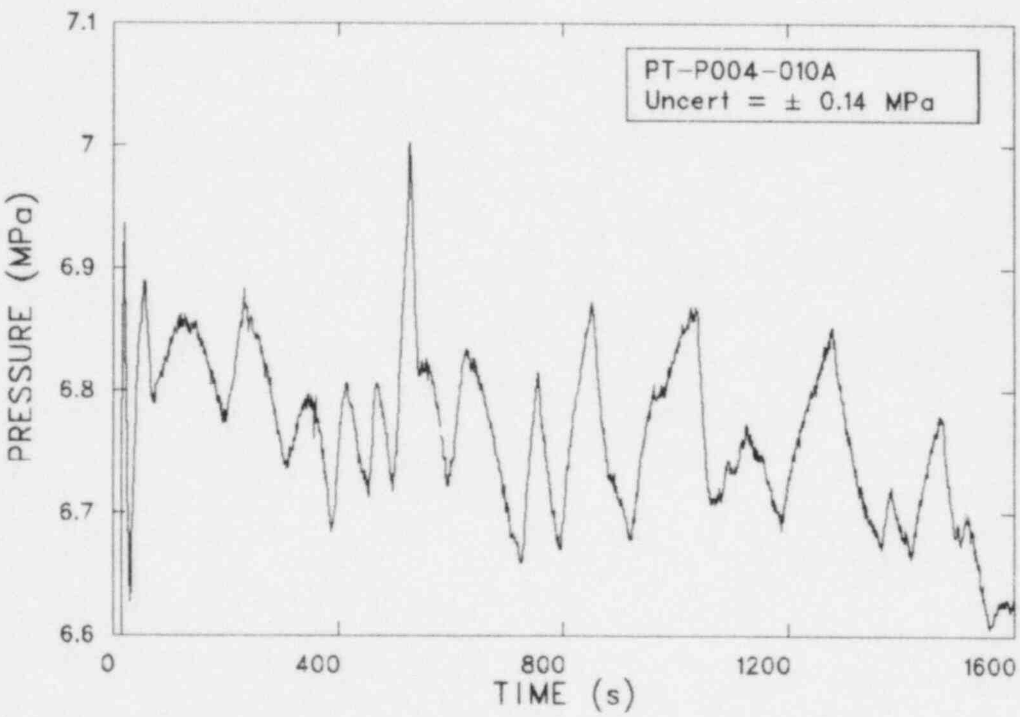


Figure 9-11. Pressure in steam generator secondary side for LOFT Experiment L9-4.

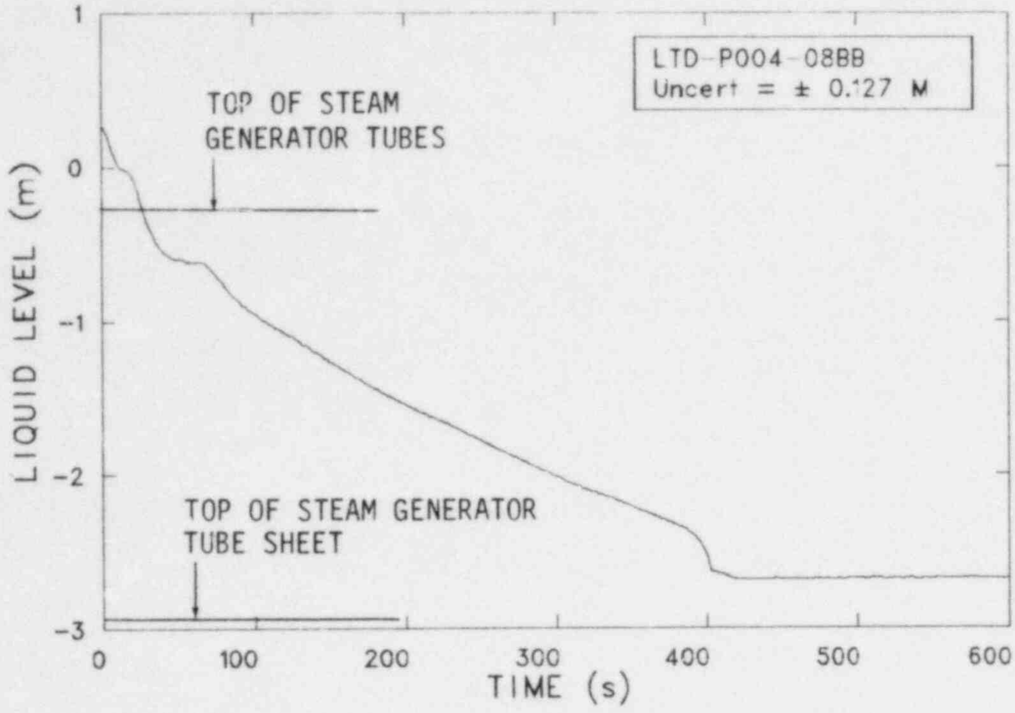


Figure 9-12. Liquid level in steam generator for LOFT Experiment L9-4.

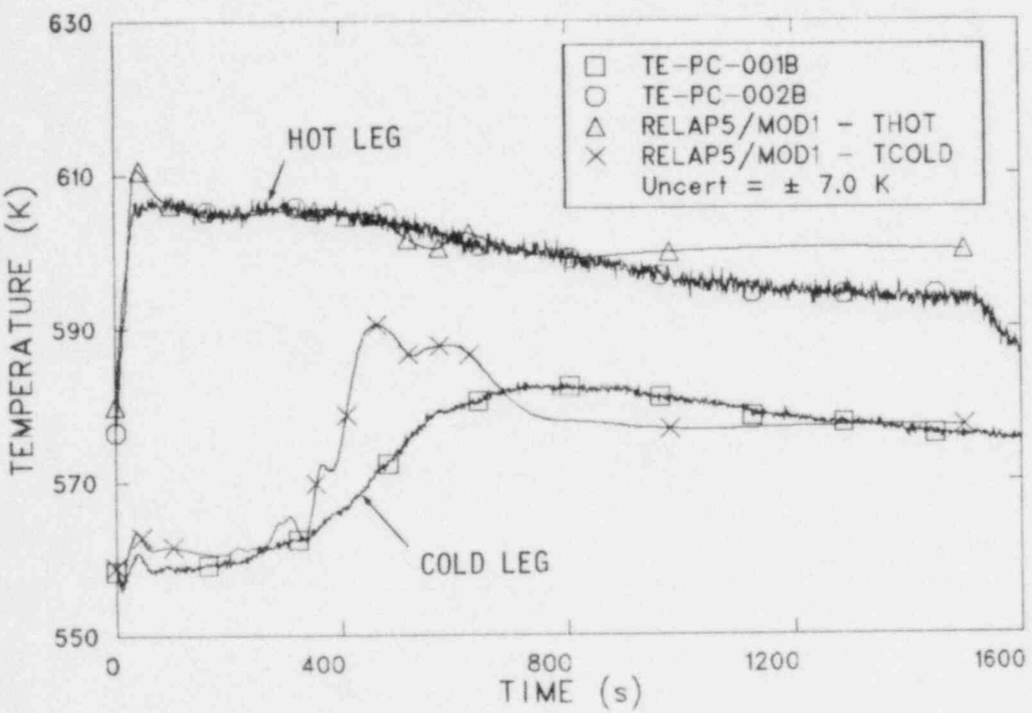
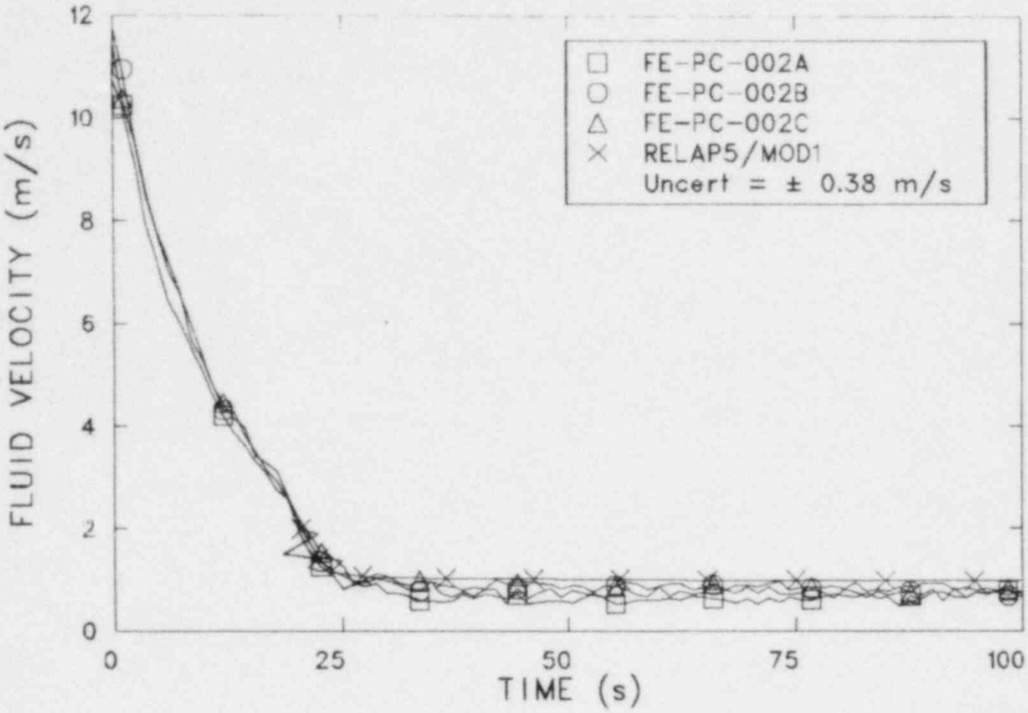
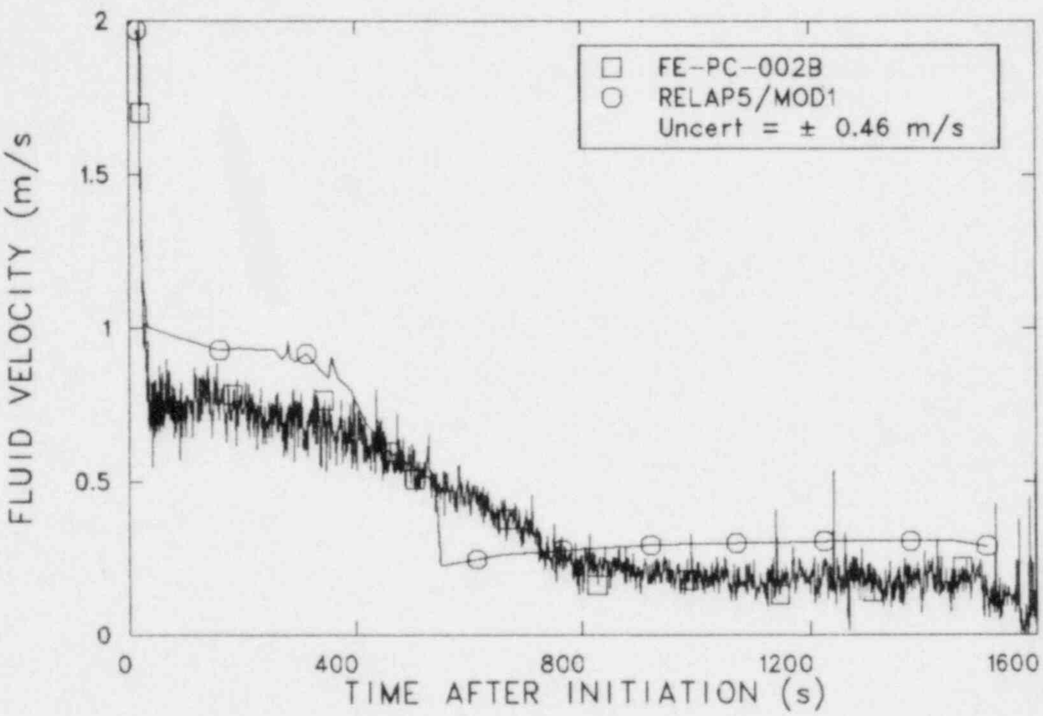


Figure 9-13. Measured and calculated temperature in intact loop hot and cold legs for LOFT Experiment L9-4.



a. Expanded time scale covering first 100 s.



b. Time scale covering duration of experiment.

Figure 9-14. Fluid velocity in intact loop hot leg for LOFT Experiment L9-4.

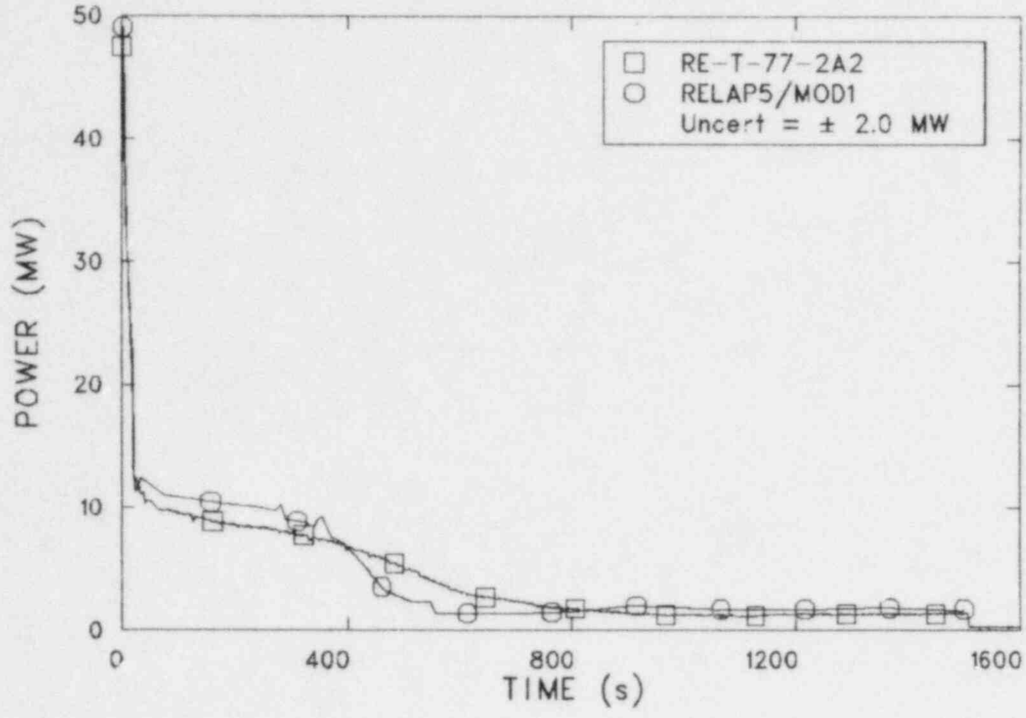


Figure 9-15. Measured and calculated reactor power for LOFT Experiment L9-4.

anticipated in the experiment prediction sensitivity calculations.

The difference between the predicted and measured power after 350 s was caused by the abrupt change in the predicted cold leg temperature. Comparisons of predicted and measured reactor power response as a function of moderator temperature during this time showed that the moderator density feedback was predicted accurately. After 1000 s, both the predicted and measured reactor power stabilized, although at different values.

An objective for Experiment L9-4 was to provide data for assessment of the effects of primary pump operation on primary system response to an ATWS by comparing the results with those of Experiment L9-3. Comparisons of the boundary conditions for these experiments are summarized in Table 9-1.

The most obvious effect of pump trip was the more rapid increase in primary coolant temperature in Experiment L9-4, as compared to Experiment L9-3 (see Figure 9-16), which was caused by the reduction in steam generator primary side heat transfer coefficient and increased residence time of the primary fluid in the core. The rapid temperature increase introduced negative reactivity into the core, causing the reactor power to decrease more rapidly in Experiment L9-4 than in Experiment L9-3, as shown in Figure 9-17. Also, this more rapid temperature increase caused a more rapid pressure increase in Experiment L9-4 (see Figure 9-18) and an earlier challenge to the SRV.

Although there were many differences in primary coolant system response to the two transients, there were two similarities of interest:

1. In both cases, the scaled SRV capacity was sufficient to control primary coolant pressure (see Figure 9-18)

2. As the steam generator liquid mass boiled off, the primary-tosecondary heat transfer degraded. When this heat transfer was no longer able to remove the heat being generated by the core, the pressure increase accelerated. This occurred in Experiment L9-4 at 210 s, with the reactor power equal to 7 MW. In Experiment L9-3, this occurred at 55 s, and the corresponding reactor power was 46 MW. Despite the differences in time and reactor power, the

Table 9-1 Comparison of boundary conditions for Experiments L9-4 and L9-3

Boundary Condition	Experiment L9-4	Experiment L9-3
Secondary feed availability	Loss of main feed, auxiliary feed available	Loss of all feed, including auxiliary feed
Primary coolant pump operation	Pumps tripped at initiation	Pump operation continued throughout
Primary relief valves	PORV assumed not available, only SRV used	Both PORV and SRV available
Secondary liquid level	Level below indicating range at 370 s	Level below indicating range at 360 s

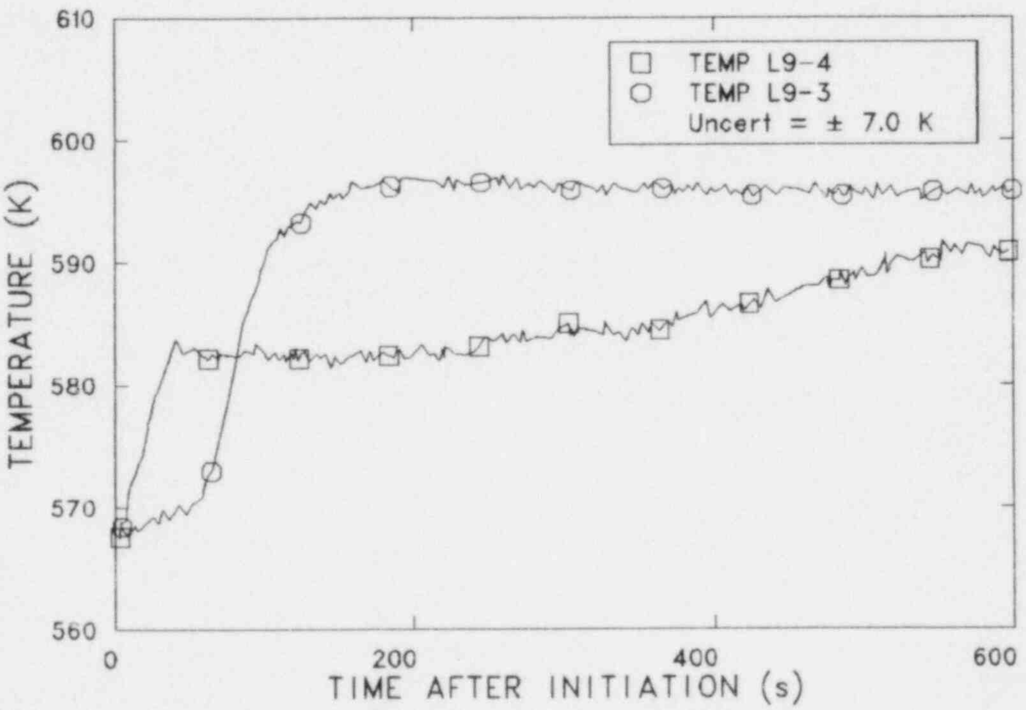


Figure 9-16. Average temperature in primary system for LOFT Experiments L9-3 and L9-4.

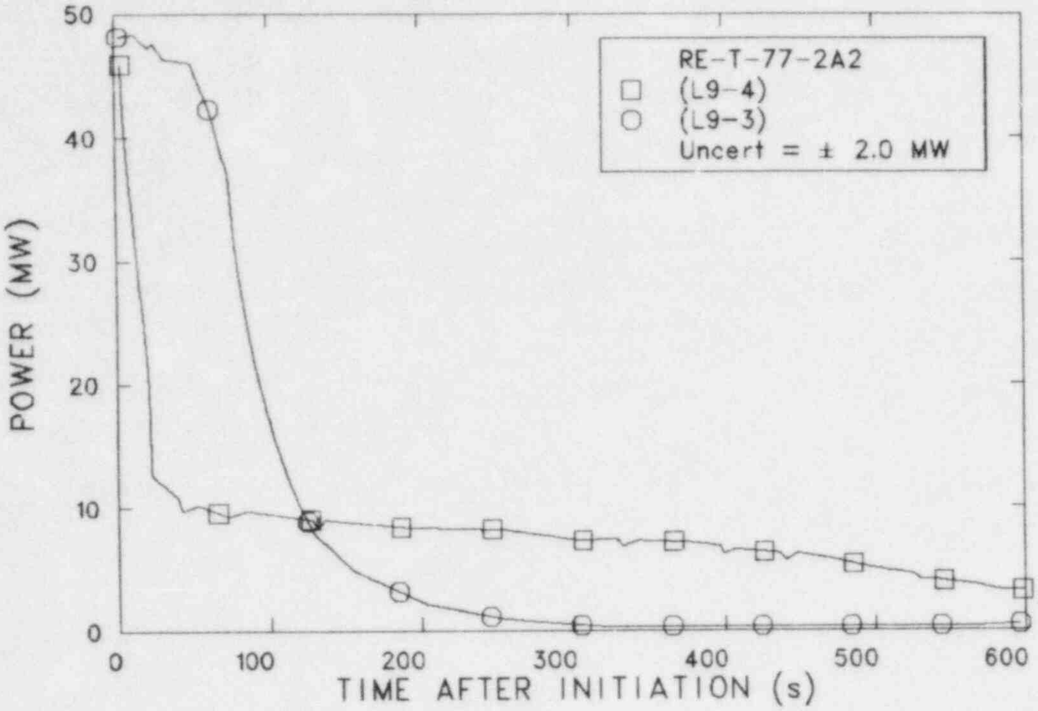


Figure 9-17. Reactor power for LOFT Experiments L9-3 and L9-4.

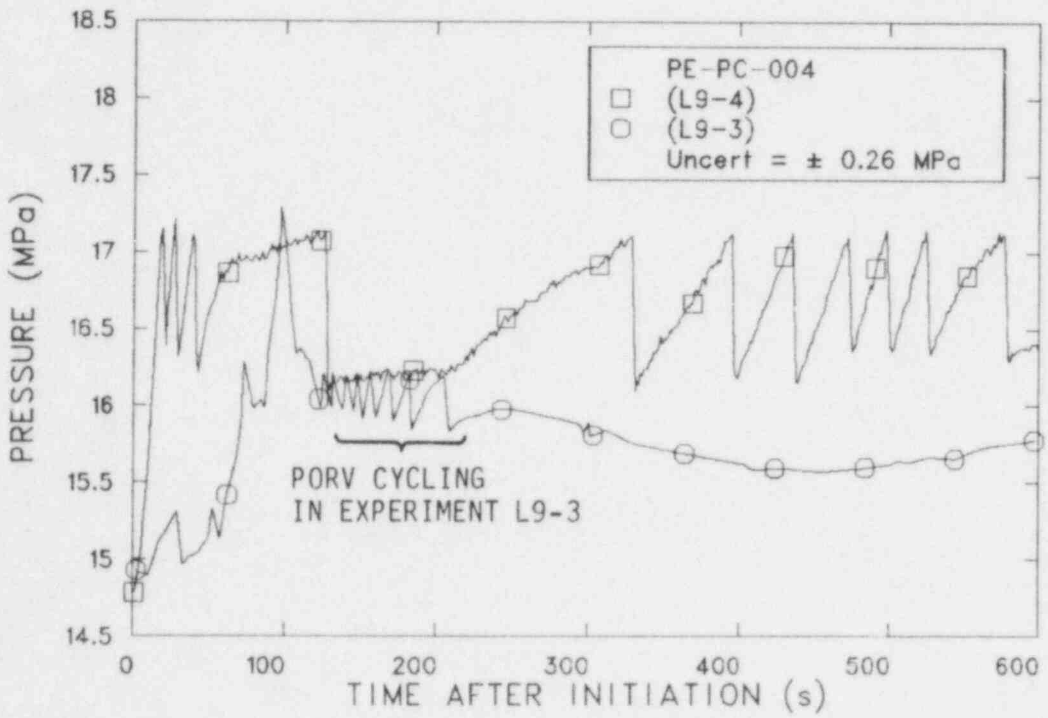


Figure 9-18. Pressure in pressurizer for LOFT Experiments L9-3 and L9-4.

steam generator liquid level at which this occurred was approximately the same for both experiments as shown in Figure 9-19.

9.1.3 Major Accomplishments and Conclusions of LOFT ATWS Experiments. Two experiments (Experiments L9-3 and 9-4) were successfully conducted in the LOFT system that simulated ATWS events and demonstrated plant recovery techniques.

Experiment L9-3 was the first ATWS experiment performed in the LOFT system. There have been no reported ATWS events in commercial PWRs. The results of Experiment L9-3 showed that an ATWS initiated from a loss of feedwater can be controlled by properly sized automatic safety systems. System recovery can be accomplished with recovery procedures involving PORV cycling, secondary system feed and bleed, and boron injection as follows.

1. Most of the transient behavior was qualitatively predicted correctly. However, the calculated magnitudes of thermal and hydraulic parameters, especially primary system pressure, were different than measured. Reasons for these differences were the larger than calculated PORV/SRV mass flow and primary-to-secondary heat transfer and nonequilibrium conditions in the pressurizer which were not calculated by the RELAP5/MOD1 computer code.

2. The reactor shut down, as expected, due to moderator temperature feedback. The quantitative response of the reactor power to moderator temperature was consistent with previous data and relatively well calculated.

3. By the time the pressure increased to the SRV setpoint, the primary coolant heat imbalance had decreased such that the combined PORV/SRV flow was sufficient to control primary system pressure and to rapidly depressurize the system.

4. The operator-controlled recovery procedure was successful. HPIS injection of highly borated water shut down the reactor and maintained it in a shutdown condition.

Experiment L9-4 demonstrated that:

1. The RELAP5/MOD1 computer code predicted the transient thermal-hydraulic and reactor kinetics phenomena reasonably well.

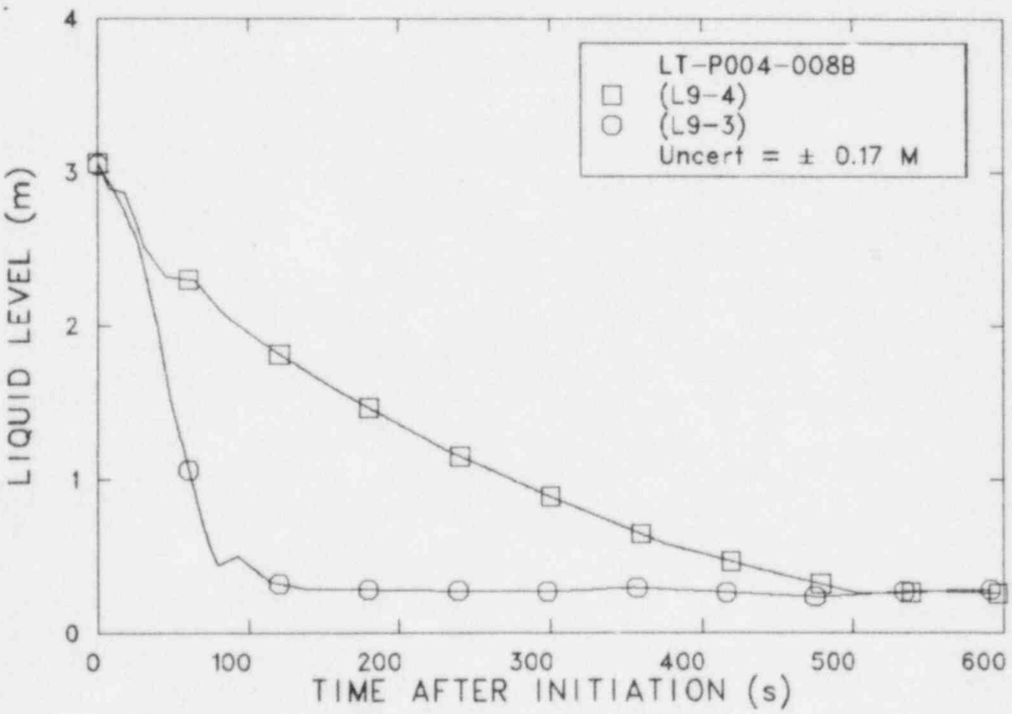


Figure 9-19. Liquid levels in steam generator for LOFT Experiments L9-3 and L9-4.

2. For high pressure, temperature, and power conditions, natural circulation was effective in removing fission and decay heat generated in an ATWS situation as long as a heat sink was available.

3. There was sufficient relief capacity in the scaled SRV to control the pressure transient in the loss-of-offsite-power ATWS.

4. The primary-pumps-off condition in a loss-of-offsite-power ATWS caused a more rapid decrease in reactor power (in an ATWS condition) than in a loss-of-feedwater-induced ATWS when the primary pumps were left on.

Relative to the three proposed ATWS rules, data from the two LOFT ATWS simulations:

1. Provided data regarding the amount of pressure relief capacity required to mitigate ATWS events.

2. Provided data to qualify codes to simulate ATWS events in commercial PWRs similar to the two LOFT experiments. The plant simulations can be used to train operators in the diagnosis and prognosis of ATWS events.

9.2 Appendix K to 10 CFR 50

On December 6, 1978, the NRC published an advance notice of proposed rulemaking[9-11] calling for a two-phase approach to the revision of 10 CFR 50. The first step would have been to make procedural changes and minor model changes without reducing the conservatism contained in Appendix K. The second phase would have made further technical changes based on research results and operating experience. Twenty-five comments were received by the closing date for comments, February 25, 1979. The major comments were that the proposed ECCS rule should be based on more realistic models and that the Phase 1 effort should be expanded to include additional research information.

Staff activity on the ECCS rulemaking was severely curtailed as a result of the accident at Three Mile Island—Unit 2 (TMI-2). Since July 1981, interest in ECCS rulemaking has been renewed.

The two-phase approach proposed in December 6, 1978, is still valid, and a stronger case can be made for basing the ECCS rule on more realistic models and expanding the Phase 1 effort to include additional research information that is well documented.

LOFT research results support changes to three features of the Appendix K ECCS evaluation models. These are the discharge model, the end of blowdown criteria, and the assumptions regarding return to nucleate boiling.

9.2.1 Discharge Model. At the time that Appendix K was issued, the major safety concern was the large-break LOCA. In addition, the discharge model was based on the analytical and experimental data available at the time. Since 1974, the understanding of break flow phenomena has increased significantly; a summary of the LOFT results related to break flow modeling is presented in Section 4.1.1, and a discussion of the conservatism of the Appendix K discharge model is presented in Sections 5.1.1, 5.2.2, and 5.3.2.

Future changes to the Appendix K discharge model should consider the results of the LOFT results and recent analytical model improvements. Specifically:

1. More flexibility should be allowed or encouraged with regard to break flow models or break flow computational techniques, which are supported by acceptable experimental data.

2. The importance of upstream condition on break flow is particularly important for small-break LOCAs. In formulating a new break flow model rule, the NRC should consider requiring that upstream conditions be taken into account in the calculations. The type of conditions that are important are: flow regime, void fraction, fluid velocity, break location, and break orientation.

9.2.2 End of Blowdown. The conservative assumptions regarding the bypass of all ECC coolant before the end of blowdown was based on a poor understanding of the phenomena. A discussion of the phenomena is presented in Section 5.1.2. Experiments in LOFT have shown that during a large-break LOCE, most of the coolant reached the lower plenum. The experimental results verify RELAP5 calculations for LOFT (see Section 4.2.1).

Future changes to Appendix K should consider deleting the present end-of-bypass criteria as such, and allowing the use of best estimate calculations of bypass phenomena which are supported by acceptable experimental data.

9.2.3 Return to Nucleate Boiling. The conservative assumption regarding RNB before reflood was based on the limited experimental data available before 1974. A summary of LOFT results regarding return to nucleate boiling is presented in Section 4.1.2.

In light of the current understanding of the phenomena, and increased post-CHF data base, consideration should be given to deleting the RNB restriction and allowing the use of best estimate calculations using post-CHF heat transfer correlations that are supported by acceptable experimental data.

9.3 References

9-1. "Electric Utilities Petition," *Federal Register, 45*, 215, Docket No. 50-29, November 4, 1980.

9-2. Advanced Notice of Proposed Rulemaking, "Standards for the Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants," *Federal Register, 46*, 226, November 24, 1981.

9-3. J. P. Adams, *Quick-Look Report on LOFT Nuclear Experiment L9-3*, EGG-LOFT-5848, April 1982.

9-4. W. Tauche, *Loss-of-Feedwater Induced Loss of Coolant Accident Analysis Report*, WCAP-9744, May 1980.

9-5. P. D. Bayless and J. M. Divine, *Experiment Data Report for LOFT Anticipated Transient Without Scram Experiment L9-3*, NUREG/CR-2717, EGG-2195, May 1982.

9-6. W. H. Grush and Y. Koizumi, *Best Estimate Prediction for LOFT Nuclear ATWS Experiment L9-3*, EGG-LOFT-5873, March 1982.

9-7. S. Silverman, *LOFT Experiment Operating Specification, Anticipated Transient without Scram Experiment, Nuclear Test L9-4*, EGG-LOFT-5897, September 1982.

9-8. D. L. Batt, J. M. Divine, K. L. McKenna, *Experiment Data Report for LOFT Anticipated Transient Without Scram Experiment L9-4*, NUREG/CR-2978, EGG-2227, November 1982.

9-9. Y. Koizumi and W. H. Grush, *Best Estimate Prediction for LOFT Nuclear ATWS Experiment L9-4*, EGG-LOFT-6023, September 1982.

9-10. A. E. Sanchez-Pope, *Quick-Look Report on LOFT Nuclear Experiment L9-4*, EGG-LOFT-6071, October 1982.

9-11. Advanced Notice of Proposed Rulemaking, "Acceptance criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Plants," *Federal Register, 43*, 235, December 6, 1978.

10. ACCIDENT MANAGEMENT

Accident management encompasses the automatic and manual operational procedures used to respond to an accident in a plant to arrest its progress and to bring the plant to a safe shutdown.

Commercial PWR safety systems are designed to operate automatically in the case of a large-break LOCA. Typically, the primary is scrammed, the reactor coolant pumps are shut off, the steam generator feedwater train and the main steam control valves are shut off, and the auxiliary steam generator feedwater pumps are started; the HPIS then begins injecting coolant, and finally the accumulators and LPIS begin injecting coolant into the system. In the case of a large break, the blowdown is over in less than 30 s, and the core is reflooded within 2 min.

It would be impossible for operators to respond to a large-break accident in the time required. Fortunately, there are only a few well defined actions required, and these lend themselves to automatic initiation. The results of Large-Break Experiment Series L2 conducted in the LOFT Program demonstrated that automatic operation of the engineered safety features inherent in commercial PWR designs is satisfactory.

A small-break LOCA differs dramatically from a large-break LOCA in the magnitude and timing of the system responses that are necessary. A discussion of the characteristics of and responses to a small-break LOCA are presented in Reference 10-1.

Unlike a large-break LOCA, a small-break LOCA requires many minutes to develop trends that, if unchecked, lead to sustained core uncovery and heatup. The actual onset of sustained core heatup, if unchecked, would not occur for tens of minutes to several hours depending upon break size.

All of the ECCSs are designed to operate automatically during a small-break LOCA, just as they do during a large-break LOCA. In addition, if the automatic responses of the ECCSs are not adequate, the operators can take independent actions to stop or mitigate the event.

As part of LOFT Small-Break Experiment Series L3, Anticipated Transient Experiment Series L6, and Multiple Failure Experiment Series L9, a significant effort was devoted to evaluating accident management procedures that could be used to stop or mitigate the consequences of small-break LOCAs or operational transients. The procedures include primary coolant pump operation, feed and bleed of the steam generator, and feed and bleed of the primary system.

10.1 Primary Coolant Pump Operation

Primary coolant pump operation has been shown to have an important effect on system response during small-break LOCAs and anticipated transients. LOFT Experiments L3-5 and L3-6 were designed to investigate the effect of pump operation on system behavior during a small-break LOCA. Details of these results are presented in Section 5.2. The experiments demonstrated that early in the transient, operating the pumps conserves system coolant by reducing break flow because a two-phase mixture is maintained upstream of the break rather than single-phase subcooled coolant which would be present at the break if the pumps were off. The experiments further showed that once the system coolant inventory is reduced so that the liquid level with pumps off is at the level of the main coolant system piping, break flow is reduced (and system coolant inventory conserved) if the pumps are turned off. Break flow is less if steam is upstream of the break rather than a two-phase mixture.

The appropriate time to turn the primary coolant pumps off after a small-break LOCA can be determined by monitoring pump current, as discussed in Section 8.4.

LOFT Experiments L9-3 and L9-4 showed the effect of pump operation during an ATWS. Experiment L9-3[10-2] was an ATWS initiated by a loss of feedwater, during which the pumps were kept running. Experiment L9-4[10-3] was an ATWS initiated by a loss of offsite power. The loss-of-offsite power tripped the pumps at the beginning of Experiment L9-4. A comparison of the boundary conditions for Experiments L9-3 and L9-4 is presented in Table 9-1.

The most obvious effect of pump trip was the more rapid increase in primary coolant temperature in Experiment L9-4 as compared to Experiment L9-3 (refer to Figure 9-16), which was caused

by the reduction in steam generator primary side heat transfer coefficient and increased residence time of the primary fluid in the core. The rapid temperature increase introduced negative reactivity into the core, causing the reactor power to decrease more rapidly in Experiment L9-4 than in Experiment L9-3 (refer to Figure 9-17). This more rapid temperature increase caused a more rapid pressure increase during Experiment L9-4 and an earlier challenge to the SRV (refer to Figure 9-18).

10.2 Steam Generator Feed and Bleed

During 12 small-break LOCEs and anticipated transient experiments, the steam generator has been shown to be very important in plant recovery. A discussion of steam generator feed and bleed is presented in Section 6.1.2. Table 10-1 presents a summary of the experiments during which the steam generator has been used in the recovery phase.

The results of the LOFT experiments showed that the steam generator was used effectively for accident management for every type of small-break LOFT LOCE and for most operational transients and anticipated transients with multiple failures. During a small-break LOCA or transient, the main function of the steam generator is to remove decay heat which would otherwise heat up and pressurize the primary system and increase the mass flow through the break or through the relief valves. Any action that removes decay heat reduces the loss of coolant and prevents or delays a core uncovery. Experiments L3-2, L3-7, and L9-1/L3-3 demonstrated that the steam generator can be reestablished as a heat sink to aid in accident recovery.

10.3 Primary System Feed and Bleed

During LOFT Experiments L6-8C-2, L6-8D,[10-12] feed and bleed of the primary system, and during

Table 10-1. Summary of experiments involving use of steam generator in accident recovery

Experiment	Experiment Description	Use of Steam Generator	References
L3-2	0.16% broken loop cold leg break LOCE, pumps tripped at start of experiment.	Auxiliary feedwater was initiated at 114 s and terminated at 1878 s. Steam bleed initiated at 4118 s. Auxiliary feedwater was used as necessary during steam bleeding to maintain steam generator as a heat sink.	10-4, 10-5
L3-5	2.5% intact loop cold leg break LOCE, pumps tripped at start of experiment.	Auxiliary feedwater was initiated at 63 s and terminated at 1800 s. Steam bleeding was initiated at 5011 s, and main feedwater was initiated at 7650 s.	10-6, 10-7
L3-6	2.5% intact loop cold leg break LOCE, pumps tripped at 2371 s.	Auxiliary feedwater initiated at 73 s and terminated at 1856 s.	10-7, 10-8
L3-7	0.16% broken loop cold leg break LOCE, pumps tripped at start of experiment.	Auxiliary feedwater initiated at 75 s and terminated at 1800 s. Steam bleed initiated at 3603 s, and auxiliary feedwater used as needed for steam bleeding.	10-5, 10-9

Table 10-1. (continued)

Experiment	Experiment Description	Use of Steam Generator	References
L6-1	Anticipated transient—loss of steam load.	Pressure in primary system was controlled by main steam control valve (MSCV) opening as needed.	10-10
L6-5	Anticipated Transient—loss of feedwater.	Pressure was controlled by MSCV opening as needed. Steam generator refill was started at 955 s.	10-11
L6-8/C-1	Small-break LOCE. Single steam generator tube rupture. Pumps operated continuously.	Feed and bleed of the steam generator in combination with small-break flow was used to depressurize the primary system.	10-12
L6-8/C-2	Same as small-break LOCE L6-8C-1, except pressurizer liquid level was higher.	Feed and bleed of the steam generator plus feed and bleed of the primary system was used to depressurize the primary system.	10-12
L6-8D	Natural circulation cooldown.	Steam generator was maintained as a heat sink by feed and bleed. Primary system feed and bleed was used to maintain pressurizer liquid level.	10-12
L9-1/L3-3	Anticipated transient with multiple failures. L9-1 was a loss-of-feedwater transient; L3-3 simulated two recovery procedures.	One recovery procedure for L3-3 consisted of refilling the steam generator and restoring it as a heat sink. The second recovery procedure consisted of latching the PORV open to depressurize the primary system.	10-13
L9-3	Anticipated transient without scram (ATWS) initiated by a loss of feedwater.	Primary recovery procedure was operation of the PORV/SRV. At 770 s operator-controlled feed and bleed of the steam generator was used to cool the primary system.	10-2

Table 10-1. (continued)

Experiment	Experiment Description	Use of Steam Generator	References
L9-4	ATWS initiated by loss of off-site power	The SRV prevented the primary system pressure from becoming excessive. Auxiliary feedwater flow and manual control of steam generator secondary side pressure was used to maintain the steam generator as a heat sink to remove decay heat.	10-3

Experiment L9-1/L3-3,[10-13] bleed of the primary system, were used to mitigate the consequences of an accident and to help depressurize the system.

Experiment L6-8C-2 simulated the rupture of a single steam generator tube by opening the primary system letdown valve. For the first 200 s, cooldown of the primary system was by steam generator feed and bleed. After 200 s, the PORV was latched open and HPIS flow was initiated. Figure 10-1 shows the primary system pressure behavior during Experiment L6-8C-2. Figure 10-2 shows the response of the pressurizer liquid level during the transient. Liquid level was restored 16 s after the HPIS and PORV flows were initiated.

LOFT Experiment L9-1/L3-3 consisted of two parts: L9-1, which addressed the issue of controlling primary system pressure without ECCS after a loss-of-feedwater accident (LOFA); L3-3 evaluated two methods of plant recovery. The first recovery method consisted of shutting off the primary coolant pumps and depressurizing the system through the PORV.

When the PORV was latched open, at 3270 s, the primary system pressure quickly decreased to a saturation pressure of 12.3 MPa, as shown in Figure 10-3. This pressure was below the ECCS injection setpoint of 13.2 MPa and demonstrated the ability of the system to recover from a LOFA with a dry steam generator secondary. The primary system pressure continued to decrease until the PORV was shut at 4850 s.

10.4 Conclusions

LOFT experiments provided data to evaluate procedures for recovering PWRs from small-break LOCAs, operational transients, and anticipated transients with multiple failures. The following conclusions can be made from the LOFT results:

1. Tripping the main coolant pumps early, following a small-break LOCA conserves coolant inventory in the primary coolant system and delays the onset of core uncovery. The appropriate time for tripping the pumps is when the collapsed liquid level reaches the reactor vessel inlet/outlet nozzles.

2. Steam generator feed and bleed can be used to remove decay heat following small-break LOCAs, operational transients, and anticipated transients with multiple failures. In the case of a small break, removing decay heat with the steam generator reduces the amount of coolant lost, thereby preventing or delaying a core uncovery. In the case of transients that result in an increase in system pressure that challenges the relief valves, the steam generator can remove decay heat and reduce the amount of coolant lost through the relief valves.

3. The LOFT Program demonstrated that the steam generator could be reestablished as a heat sink even after it has boiled dry.

4. Primary system feed and bleed or primary system bleed have been used in the LOFT Program to depressurize the primary system.

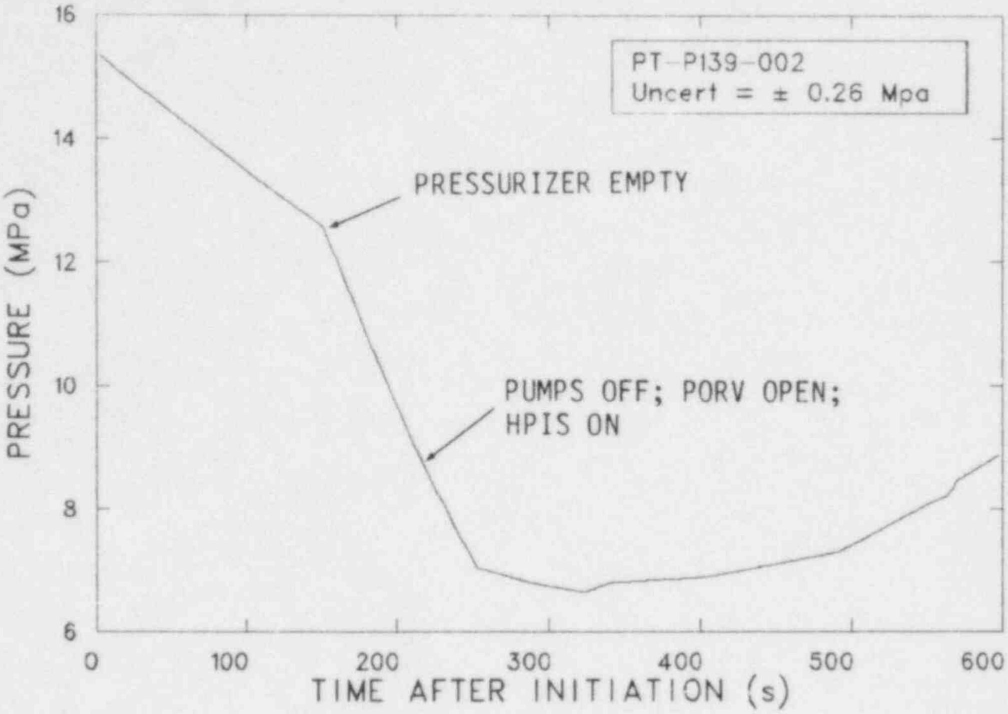


Figure 10-1. Pressure in primary system for LOFT Experiment L6-8C-2.

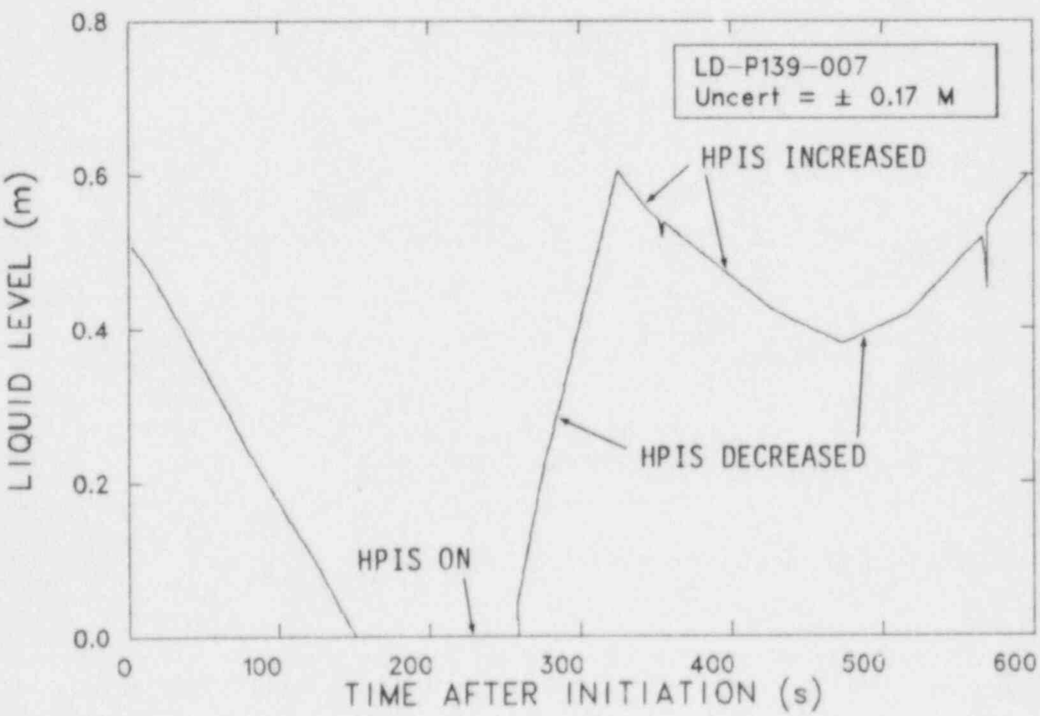


Figure 10-2. Liquid level in pressurizer for LOFT Experiment L6-8C-2.

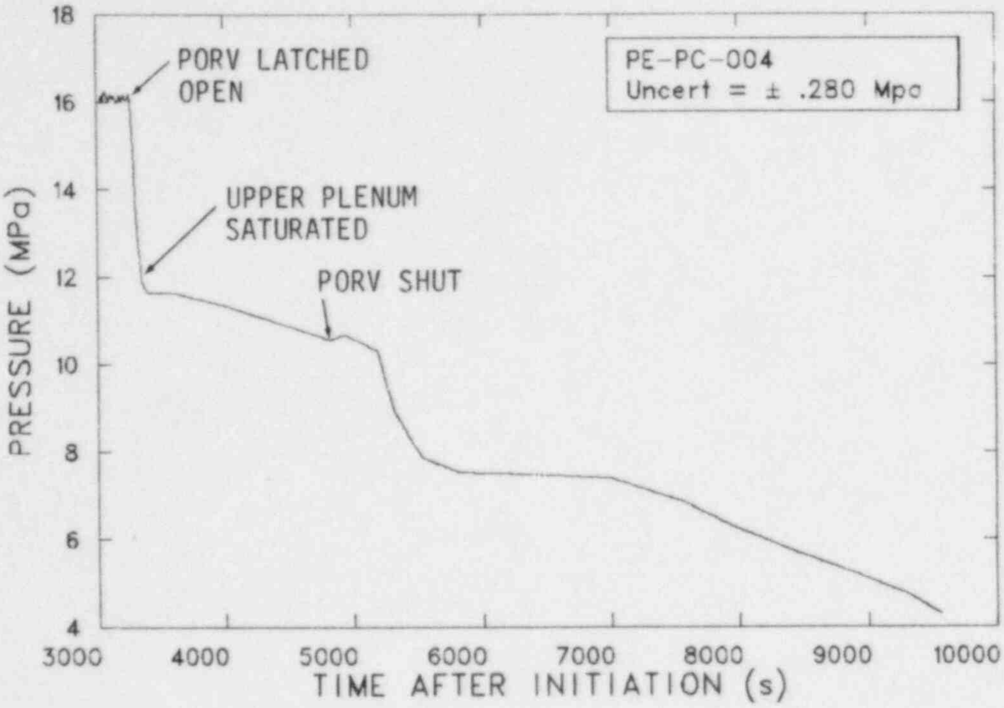


Figure 10-3. Pressure in pressurizer for LOFT Experiment L3-3.

10.5 References

10-1. *Mitigation of Small-Break LOCAs in Pressurized Water Reactor Systems*, NSAC-2, March 1982.

10-2. J. P. Adams, *Quick-Look Report on LOFT Nuclear Experiment L9-3*, EGG-LOFT-5848, April 1982.

10-3. A. E. Sanchez-Pope, *Quick-Look Report on LOFT Nuclear Experiment L9-4*, EGG-LOFT-6071, October 1982.

10-4. J. H. Linebarger, *Quick-Look Report on LOFT Nuclear Experiment L3-2*, EGG-LOFT-5104, February 1980.

10-5. W. H. Grush and G. E. McCreery, *Posttest Analysis of Loss-of-Fluid Tests L3-2 and L3-7*, EGG-LOFT-5632, October 1981.

10-6. J. P. Adams, *Quick-Look Report on LOFT Nuclear Experiment L3-5/L3-5A*, EGG-LOFT-5242, October 1980.

10-7. K. G. Condie et al., *Four-Inch Equivalent Break Loss-of-Coolant Experiments: Posttest Analysis of LOFT Experiments L3-1, L3-5 (Pumps off), and L3-6 (Pumps on)*, EGG-LOFT-5480, October 1981.

10-8. G. E. McCreery, *Quick-Look Report on LOFT Nuclear Experiment L3-6/L8-1*, EGG-LOFT-5318, Rev. 1, July 1981.

10-9. G. E. McCreery, *Quick-Look Report on LOFT Nuclear Experiment L3-7*, EGG-LOFT-5192, June 1980.

10-10. D. L. Reeder, *Quick-Look Report on LOFT Nuclear Experiments L6-1, L6-2, and L6-3*, EGG-LOFT-5270, October 1980.

10-11. D. L. Reeder, *Quick-Look Report on LOFT Nuclear Experiment L6-5*, EGG-LOFT-5165, June 1980.

10-12. J. P. Adams, *Quick-Look Report on LOFT Nuclear Experiment Series L6-8*, EGG-LOFT-6031, September 1982.

10-13. J. P. Adams, *Quick-Look Report on LOFT Nuclear Experiment L9-1/L3-3*, EGG-LOFT-5430, April 1981.

11. POTENTIAL FUTURE APPLICATIONS OF LOFT RESULTS

A large amount of information has been accumulated by the LOFT Experimental Program. Much of the information consists of experimental data from over 600 instruments, preexperiment predictions, and postexperiment analyses. The information covers a wide variety of LOCEs and transients ranging from large-break LOCEs to anticipated transients with multiple failures. Additional documentation exists which consists of facility descriptions, RELAP model descriptions, and data reports for each experiment that contain measurement lists and measurement uncertainties. LOFT data will continue to be valuable in such applications as nuclear safety code development and assessment. However, the data and results have great value in other areas of reactor safety research that require data from accident simulations in a large integral system. Specific examples are: accident diagnosis, licensing with best estimate models, probabilistic risk assessment, and development of fast running codes to drive training simulators.

11.1 Accident Diagnosis

To date, the most important accident diagnosis technique developed through use of LOFT data is a display of primary coolant pump current versus cold leg temperature. As long as the pump is running, an operator using this technique can differentiate between a small-break accident and an over-cooling transient, as discussed in Section 8.4.

Other more complex displays are possible which will aid in accident diagnosis. One involves the rate of change of system pressure with time plotted against the rate of change of primary system energy (that is, core heat input plus pump heat input minus heat transferred to the steam generator minus environmental heat losses).

This display would require a large number of measurements and an online computer to compute parameters. Accidents or transients, however, would readily be identified by the signature of their state point on the display. This display could be programmed to identify a large- or small-break LOCA (rate of change of pressure), an ATWS (power mismatch plus rate of change of pressure), overcooling transient (power mismatch plus rate of change of pressure), or most other transients. In addition, the effectiveness of corrective actions could be monitored, particularly the removal of decay heat by the steam generator. LOFT data would be very valuable in developing a display such as the one described above. Data from a large number of different accident and transient scenarios is available to verify the algorithms and the typical accident signatures.

11.2 Licensing with Best Estimate Models

The objective of the 1973 ECCS rule was to provide a conservative factor of safety for reactor design. The rule was very appropriate when it was issued, because of the many uncertainties regarding phenomena and analytical capabilities. Due in part to the contribution of the LOFT Experimental Program, many if not all of the uncertainties have been substantially reduced, and are bounded by analytical and experimental data (see Section 5).

One of the most important potential uses for LOFT data would be in developing criteria for licensing with best estimate models. LOFT data could be used to develop maximum allowable system temperatures and pressures, as well as maximum allowable fuel cladding temperatures. These limits could be established as conservative design limits minus a reasonable uncertainty factor which could be established as a function of the capability of a vendor or license applicant's analytical code capability.

11.3 Probabilistic Risk Assessment

The risk associated with nuclear power is defined as the product of the probability of an accident that results in a release of radioactivity and the consequences of the accident (that is, number of deaths, number of injuries, and amount of property damage).

The risk assessments of the Reactor Safety Study[11-1] were based on conservative estimates of probability and consequences. Since the Reactor Safety Study was issued, 40 accidents were simulated in the LOFT facility including the only ATWS events thus far studied experimentally. The

LOFT ATWS results are the only data available to qualify codes to predict the consequences of an ATWS in a commercial PWR.

11.4 Operator Training

The most convenient way to train operators is with training simulators. Because training simulators must perform in real time, however, they must be driven by either analog computers or digital computers using simple plant models and/or simplified equations. Realistic simulations of severe transients or accidents can be achieved with advanced computer codes such as RELAP or TRAC, which must be run on large mainframe computers, and typically, the codes run much slower than real time. A RELAP5 Engineering Simulator has been assembled for evaluating operator guidelines for commercial PWRs.[11-2] The engineering simulator consists of the RELAP5 code, an input deck describing a particular plant, color graphics software, a color graphics terminal, and a mainframe computer. The engineering simulator allows transients to be simulated in an interactive mode with graphics displays of the control room instrumentation.

The engineering simulator could be used effectively for operator training. By using the traceability methodology discussed in Section 3.3, the RELAP5 code could be programmed to simulate any/or all of the LOFT accident or transient events in a commercial PWR. The graphics displays would then illustrate plant conditions and instrument readout during an event. In addition, operator intervention could be simulated to train operators to respond properly.

11.5 References

11-1. United States Nuclear Regulatory Commission, *Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants*, WASH-1400, NUREG-75/0141, October 1975.

11-2. D. G. Hall, private communication, EG&G Idaho, Inc., June 7, 1983.

12. OVERALL PROGRAM ACCOMPLISHMENTS AND CONCLUSIONS

The NRC-LOFT Program was a unique, integral PWR Experimental Program. The main purposes of the LOFT experiments were to qualify the engineered safety features and to verify the computer codes used in safety analyses. In accomplishing these objectives, LOFT has compiled a significant list of accomplishments. Some of the major LOFT accomplishments are:

1. The LOFT experiments demonstrated the adequacy of the ECCSs under a wide range of conditions, including degraded ECCS performance.

2. The LOFT program contributed to the improvement of computer codes used to predict the response of commercial PWRs to LOCAs and to verify their adequacy over a wide range of accidents.

3. The experiment results contributed to improved understanding of LOCA phenomena, particularly those associated with the acceptance criteria for ECCS models in Appendix K of 10 CFR 50, as follows:

   a. Discharge models

   b. End of blowdown

   c. Post-CHF heat transfer

   d. Conservatism of evaluation models.

4. The experiment results provided valuable information regarding system behavior following a small-break LOCA, specifically: stability of natural circulation, and effectiveness of alternate means of heat removal.

5. The experiments demonstrated that the plant control systems and operator actions could deal effectively with a wide range of operational transients, such as secondary load rejection, loss of forced primary coolant flow, and loss of feedwater.

6. The experiments showed that an anticipated transient without scram initiated from a loss of feedwater (the limiting ATWS event) could be controlled by properly sized automatic safety systems.

The main conclusions of the LOFT program are:

1. The ECCS is effective in cooling the reactor core and preventing fuel damage after a LOCA. The ECCS remains effective in situations in which initiation delay and/or partial degradation of the ECCS occurs.

2. Current plant recovery systems and methods are effective in bringing the plant to safe shutdown conditions from anticipated transients and LOCAs.

3. Natural circulation cooling occurs in the primary coolant system whenever the steam generators are in a heat sink mode (able to transfer heat from the primary system to the secondary system). The primary coolant can be either liquid or a mixture of liquid and steam.

4. The electrical power to operate the primary coolant system pumps is sensitive to water density and can thus be used to identify the kind of accident that has occurred, that is, loss of coolant or anticipated transient. In conjunction with other plant measurements, the primary pump power can be used to identify the kind of anticipated transient that has occurred. Leaving the primary pumps on until accident identification is made, in those transients where operator intervention is possible or practical, does not adversely affect the course of the transient.

APPENDIX A

WESTINGHOUSE AND EPRI–REVIEW COMMENTS

APPENDIX A

WESTINGHOUSE AND EPRI—REVIEW COMMENTS

The following section presents comments received from N. Lee of Westinghouse Electric Corp. and R. Breen of the Electric Power Research Institute regarding the contents of the final version of this report.

WESTINGHOUSE COMMENTS

In general the following points should be made:

- Provide facts. Generalization/speculation based on little data should be avoided.
- Avoid extending the LOFT results to PWR without mentioning scaling issues.

In view of the above comments, we suggest that the Sections 8.1.3 and 9.2 be deleted.

ABSTRACT

Considering the scale principles, the statement "the transient response of the LOFT system to accident events is similar to large [$\sim$1000 MW(e)] commercial PWRs" should be reworded.

EXECUTIVE SUMMARY

This section should contain statements about scaling and caution readers about possible differences between LOFT facility transients and PWR transients.

p. iii — A rewording of the first sentence of the second paragraph is necessary to reflect possible differences in scaling.

p. v — Did you have enough burnup to claim that data from the LOFT experiments indicates that unpressurized fuel performs as well as prepressurized fuel?

p. v — You need to point out oversized valves were used for LOFT steam generator feed and bleed, primary system feed and bleed, and operation of the primary coolant pumps.

Chapter 2, HISTORY AND SCOPE OF THE LOFT PROGRAM

p. 2-2 — Provide a reference for the final changes made to the LOFT program.

p. 2-2 — It has been pointed out by many critics that LOFT LOCES do not closely model the large-break LOCAs hypothesized for commercial PWRs.

p. 2-7 — A short description of the principal components of the broken loop would be helpful to readers.

p. 2-17 — Did L3-3 demonstrate or *examine* two independent recovery methods?

Chapter 3, LOFT RESULTS RELATIVE TO COMMERCIAL PWR BEHAVIOR

p. 3-4 — The discussion regarding LOFT scaling, flow area and flow resistance is confusing.

p. 3-5 The last paragraph in Section 3-2 is subjective and the logic is not clear. This is not a necessary statement here.

p. 3-6 The statement "The differences between calculated and measured LOFT response are the minimum uncertainties that can be expected in a commercial plant calculation" should be explained.

p. 3-8 Why do you present the ZION results with different conditions anyway?

p. 3-9 The uncertainty should be RELAP's ability to predict LOFT—*not* RELAP's ability to predict a PWR transient. It would not be straight forward for a PWR due to the scaling.

Chapter 4, IMPACT OF LOFT RESULTS ON THERMAL–HYDRAULIC MODELING

p. 4-1 The discussion should not be limited to RELAP 4 and RELAP 5. At least TRAC discussions should be provided since TRAC is the NRC's best estimate code.

p. 4-20 Discuss TRAC model results relative to the split-down comer nodalization of the RELAP code.

Chapter 5, IMPACT ON 10 CFR 50.46 AND APPENDIX K

p. 5-4 A RELAP4 or 5 or TRAC calculation with a rewet model should be included to show that this model is needed to match data.

p. 5-13 The second and third paragraphs in Section 5.2.2.2 paragraphs contain statements that overlap with the ones on p. 5-10 and 5-11. Need to condense!

p. 5-24 The low-pressure setpoint (1.66 MPa) should be pointed out in the test introduction. Add "possibly" between data and support in the eighth line.

Chapter 6, IMPACT ON SAFETY ISSUES

p. 6-1 Add the statement "But it must be noted that the LOFT results should be interpreted with discretion considering the differences between the LOFT facility and commercial PWR's" at the end of the third paragraph.

p. 6-8 Item 1 overextends the results.

Chapter 7, ISSUES RELATED TO THE THREE MILE ISLAND ACCIDENT

p. 7-4 The statement in the sixth paragraph of Section 7.1.2 is speculation. It is not based on observation.

p. 7-12 No recommendations are necessary for this report.

p. 7-14 No suggestions. Provide facts only found from LOFT tests.

Chapter 8, SAFETY TECHNOLOGY

p. 8-1 Change "enhances or improves" to "have a potential to enhance or improve" in the third line.

p. 8-2 It must pointed out that the fact that data base from the Zorita program is too small to get a general conclusion.

p. 8-2 Section 8.1.3 should be deleted! This is based on very limited LOFT results which were overextended to recommendations.

p. 8-15 The performance of core exit T/Cs can be affected by the installation method, the upper plenum hardware and the fact that the LOFT core is only 5.5 ft long.

p. 8-15 The fact that the core exit T/C measurement did not work in LOFT should not be taken to mean that it would not work in a PWR!

Chapter 9, RULEMAKING

p. 9-20 Omit Section 9.2, Appendix K to 10 CFR 50. The three items are presented in previous sections. You do not need to make recommendations.

Chapter 10, ACCIDENT MANAGEMENT

p. 10-1 Delete the second paragraph in Section 10-1.

p. 10-2 Delete the next to last sentence in the second paragraph of Section 10.2.

p. 10-4 Change "can" to "could" in the first sentence in Item 2.

Chapter 11, POTENTIAL FUTURE APPLICATIONS OF LOFT RESULTS

p. 11-1 . . the data and results *may* have great value in other areas . .

Chapter 12, OVERALL PROGRAM ACCOMPLISHMENTS AND CONCLUSIONS

p. 12-1 To qualify the engineered safety features was not one of main purposes.

Add "for the LOFT facility" to the end of Item 1.

Add "for the LOFT system" to the end of Item 5.

The main conclusions of this report are subjective and the listing of the accomplishments 1-6 is enough for the purpose of this report.

EPRI COMMENTS

I. SUMMARY

In general, too much benefit to the industry and NRC is claimed for LOFT. Examples:

p. iv. The LOFT research results could *not*, by themselves resolve the issues of the human-machine interface, interruption of ECCS after a LOCA, instrumentation to follow the course of an

accident or revised small break LOCA analysis methods, although in various ways the LOFT results may help in the resolution or confirm conclusions made by the industry before the LOFT data were taken.

p. v. It is not apparent that any contribution was made on fuel pressurization.

Some contributions do seem to have been made; for example:

p. v. The unreliability of core exit thermocouples.

p. v. Support of more realistic ways of analyzing ATWS.

A main concern about the report is that there may be a tendency to assume that, because a particular behavior was observed in a LOFT experiment, that same behavior would be observed in a plant. It needs to be emphasized that LOFT and other such experiments are not capable of accurately simulating a complete, full-sized reactor. The most that such experiments can do is to provide an experimental base with which to compare models or codes which can be applied to plants. The better the simulation, the better the comparison, but it will still be a simulation and will be, in some respects, unlike the prototype. The effects of the differences must be evaluated for each application.

In particular, physical phenomena occurring or identified in LOFT may not be relevant to a formal, if stylized, safety assessment. Somewhat extravagant claims (overall) for LOFT results are made throughout the lead summary:

p. iv. "contributed tc development of models" is correct: however, the split-downcomer model leads to *unrealistic* calculated azimuthal velocities and flows.

p. iv. How is the multiplier value of 0.848 for H-F and HEM critical flow to be used to evaluate the conservative nature of Appendix K? The value in LOFT specific, and PWRs have different pipe diameters and *no blowdown nozzles* of the type used in LOFT.

p. iv. Is not the "core quenched. . . by a flow reversal" phenomena applicable to the LOFT cases studied only, and not relevant to a PWR safety analysis? (Distinguish here between physical phenomena in LOFT, and applications to safety assessment.)

p. iv. The AOC program and its results appear applicable only to LOFT and has not, to my knowledge, been adopted for a commercial PWR. (P, T state displays were independently developed by GPU and others.)

p. v. How did LOFT experiments demonstrate "the *overall adequacy* (emphasis added) of requirements. . . in Reg. Guide 1.97?" This appears inappropriate.

p. v. "Fuel pressurization" has been well studied by the industry: given the LOFT low burnup, short core, with atypical enrichment, and 15 x 15 rod size, how are these results applicable to current PWRs?

p. v. The liquid level and instrumentation discussion, both here and in the text, is very misleading. The virtues and drawbacks of conductivity probes, SPNDs, ΔP readings and outlet T/Cs are largely independent of LOFT results and experiments. (viz TMI, Semiscale, ORNL, and CE results.)

p. v. The LOFT ATWS tests are very useful and invaluable benchmarks for analysis, but that is all. The results are very sensitive to the MTC and the FDC, which need to be well characterized for LOFT and any plant.

p. v. Data on accident management is particular and applicable only to LOFT.

II Chapter 2, History and Scope of the LOFT Program

p. 2-12 The value of LOFT as a *nuclear* facility only must be stressed: being the *biggest* is not as relevant given the still large differences from LPWRs.

III. Chapter 3, LOFT Results Relative to Commercial PWRs

p. 3-1 The LOFT/Semiscale scaling is useful but *not* conclusive, since the extrapolation to PWR is still large.

p. 3-1, 3-2 Volume scaling alone is not sufficient given, for example, L/D effects, core length, diameter effects on flooding and flow regimes, which are not discussed. *The distortions introduced can only be resolved by analysis.* (The tuning over the years of RELAP to both Semiscale and LOFT data invalidates many of the conclusions made regarding scaling analysis and the adequacy of the RELAP results, see p. 3.7 and Figure 3-3.)

p. 3-8, 3-9 This is very revealing: the LPWR must be "transformed" (i.e., rescaled) to act like LOFT (and vice versa)—the differences that are shown in Figures 3-5 and 3-6 are *real scaling differences* not artifacts. If ZION were LOFT one would of course obtain Figure 3-7: the real point is it is *not* LOFT, and LOFT is *not* ZION.

IV. Chapter 4, Impact of LOFT on Thermohydraulic Modeling

p. 4-1 The nozzles used in LOFT render the extrapolation to LPWR incorrect.

p. 4-3 The playing with coefficients (1.13, 1.8, 0.98, 0.84, 1.0 for H-F, HEM, subcooled and saturated) illustrates a major *deficiency* in the methods for predictive purposes.

p. 4-5 Fig 4-10 Comparison to temperature alone is misleading: *the slope is the most relevant* (equivalent to heat transfer). These are different by up to a factor of 2.

p. 4-20, 4-21 What is the justification for items 1-7, and what relevance have they to LPWRs?

p. 4-22, 4-24 Nonequilibrium pressurizer models for LPWRs have been known to be necessary for years (viz RETRAN and MMS).

p. 4-24 As a physicist, I am appalled to be asked to believe for RELAP4 that the "gas constant" [isentropic exponent (8)] for a gas such as nitrogen changes with "expansion rate," "surface-to-volume ratio of the gas" and "as a function of accumulator flow." This is abject nonsense and is a pure modeling deficiency.

How does the RELAP5 model compare to say L9-3 and small break data? What should I use for an LPWR?!

p. 4-26 What should an LPWR use?

V. Chapter 5, Impact on 10 CFR 50.46 and Appendix K

The impact is certainly not clear from this chapter. Perhaps the impact would be more apparent if the author would rewrite the appropriate sections of 10 CFR 50.46 and Appendix K italicizing the changes based on LOFT and providing appropriate references. I don't find it particularly revealing to read on page 5-9 "The results demonstrated that accurate heat transfer correlations are necessary to predict the accident consequences accurately."

p. 5-1	The key question is the extrapolation of LOFT to LPWR: one could argue that the 565 K difference quoted is specifically to account for differences in core size, break flow, decay heat, etc. *for an LPWR*.
p. 5-1	How can a "best estimate" model be conservative?—it is either "wrong" scientifically or *not* a "best estimate" of any value!
p. 5-2	Comment on multipliers applies.
p. 5-3	Where is the "downward flow" comparison and the calculation of drop entrainment for LOFT?
p. 5-4	Core rewet, or RNB, may be due to the flows and short core applicable to LOFT for the break sizes tested in LOFT. Obviously RNB can occur—what are the *necessary physical conditions* or criteria for both LOFT and a PWR?
p. 5-9	Given the problems of RELAP4, how can one utilize the core heat transfer calculations? The statement that "accurate heat transfer correlations are necessary" is generally true but inappropriate as a conclusion from the work described.
p. 5-9 et seq.	How does the fact that LOFT has only *one* active steam generator affect the conclusions? Obviously, loop-to-loop effects and instabilities cannot be observed or quantified in LOFT.
p. 5-16	Conclusion 1 has been verified by plant data (viz TMI-2 and the Crystal River cooldown). Hence the requirements for inventory measurements and no ECC throttling for LPWRs.

VI. Chapter 6, Impact on Safety Issues

p. 6-1	Impact on USI's. The extent to which the LOFT *systems* are typical of LPWRs must be examined e.g., aux feed and steam generator secondary side, and one loop versus two to four loops.
p. 6-8	What is the basis for conclusion 2—it does not seem to be discussed.
p. 6-11	The "reflux condenser" section needs expansion and verification.

VII. Chapter 7, Issues Related to the TMI-2 Accident

p. 7-1	Since LOFT is not an OTSG plant, nor scaled to one, a discussion of why LOFT is relevant is necessary!
p. 7-1	How did the LOFT AOC "confirm. . . improving the operational safety of nuclear power plants." It is not shown in or by this report!

p. 7-1 The interruption must be accord with EPG's. Were the experiments conducted in this manner? (And hence relevant to LPWRs.)

p. 7-14 Why recommend "cladding or guide tube temperatures" for degraded core cooling information? Is this for LPWRs? Density measurements are also impractical in LPWRs!

p. 7-14 Natural circulation in an LPWR can be determined from the leg ΔTs.

VIII. Chapter 8, Safety Technology

There may be some value for large PWRs in the LOFT work on reactor coolant pump current under conditions of two phase flow, but the report should state how the LOFT findings apply to the large pumps.

Are they really suggesting *unpressurized fuel* for LPWRs on the basis of LOFT limited tests?!

The *channel blockage* results have not been shown in LOFT: there is no experiment with burst/blockage yet performed in LOFT!

The *liquid level detectors* are not (as I understand it) typical for LPWRs. *SPND effects* were shown at TMI and have been extensively analyzed—this is not even mentioned (also why the industry does not use them).

The *LOFT cladding T/Cs and outlet T/C* results have been discussed in numerous meetings. The results appear particular to LOFT.

Pump current monitoring is potentially useful: but the implementation in LPWRs is fraught with calibration and other questions. There is a report from a LOFT review meeting on this (which is ignored) which contains industry reactions.

IX. Chapter 9, Rulemaking

The ATWS tests 9-3 and 9-4 are useful benchmarks. RELAP5 does not predict all features of the tests "reasonably well" (p. 9-19). For example, where are the comparisons for Figures 9-19 and 9-19?

Emphasis should be placed on the value of the *data*.

The Appendix K implications have been discussed above.

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13. ABSTRACT (200 words or less)

This document is a summary of the main research results of the Loss-of-Fluid Test (LOFT) Program relative to code assessment, code development, licensing, rulemaking, safety technology, and reactor operations. The LOFT facility is a 50 MW(t) pressurized water reactor (PWR) system with instruments that measure and provide data on the system thermal-hydraulic and nuclear conditions. The transient response of the LOFT system to accident events is similar to large [~1000 MW(e)] commercial PWRs. The main objectives of the LOFT Experimental Program were to qualify the engineered safety systems used in commercial PWRs and to verify the computer codes used in safety analyses.

The LOFT Program contributed to the improvement of computer codes used to predict the response of commercial PWRs, demonstrated the adequacy of engineered safety systems, and contributed to improved understanding of PWR accident phenomena, particularly those associated with the evaluation models in Appendix K to 10 CFR 50 (the "ECCS rule").

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