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EGG-LOFT-5896
NE L9 Series
EDD L9-4
September 16, 1982

LOFT EXPERIMENT DEFINITION DOCUMENT L9-4
ANTICIPATED TRANSIENT WITHOUT SCRAM EXPERIMENT

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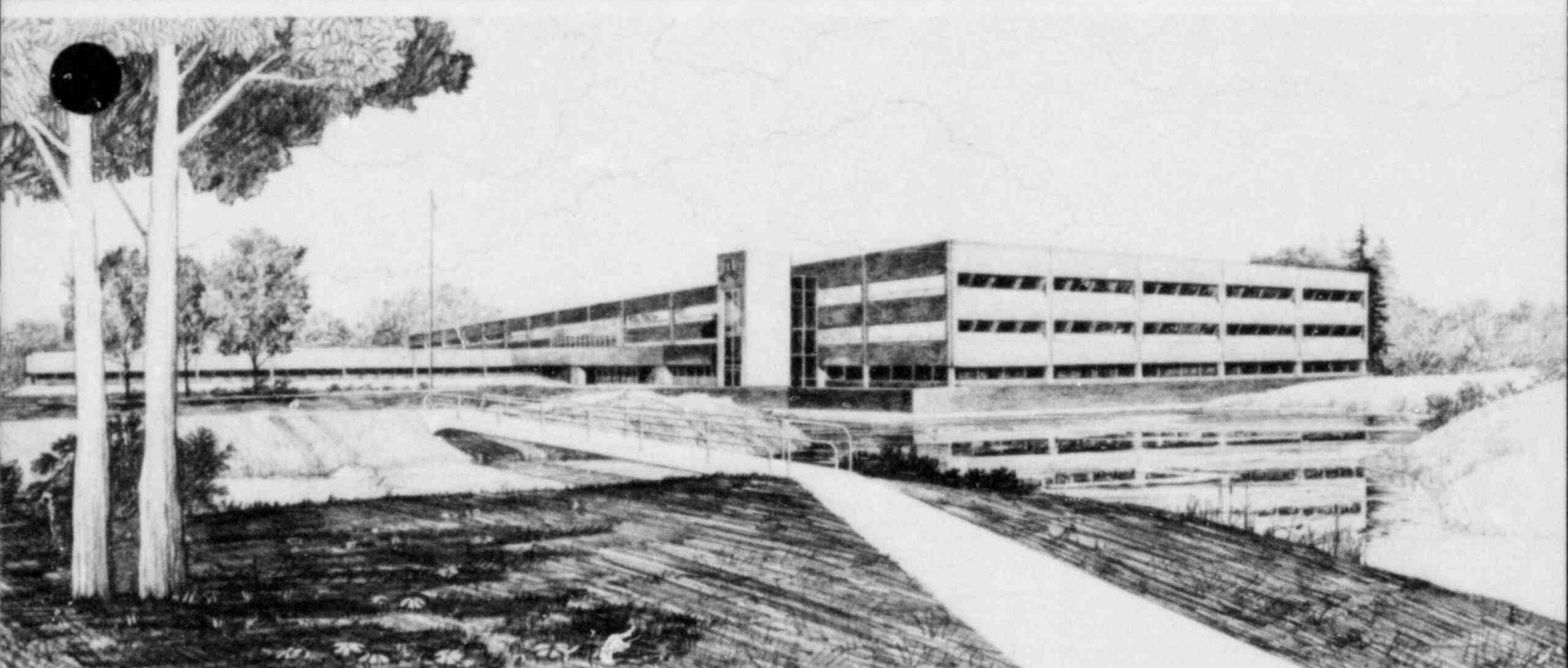

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Prepared for the
U.S. NUCLEAR REGULATORY COMMISSION
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A6048

EG&G Idaho

EG&G Idaho, Inc

FORM EG&G-398
(Rev. 11-81)

INTERIM REPORT

Accession No. ______

Report No. EGG-LOFT-5896
NE L9 Series
EDD L9-4

Contract Program or Project Title:
LOFT Program Division

Subject of this Document:

LOFT Experiment Definition Document L9-4
Anticipated Transient Without Scram Experiment

Type of Document:

LOFT Experiment Definition Document (EDD)

Author(s):

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Date of Document:

September 16, 1982

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Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C.
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LOFT EXPERIMENT DEFINITION DOCUMENT
L9-4 ATWS EXPERIMENT

Reviewed:	L. J. Hanson by EAH — Manager, Loft Program Division	9/14/82 — Date
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Approved:	L P Leach — Director, LOFT	9/14/82 — Date
Authorized for Release:	Hughes 9/14/82 — Configuration Document Control and Services	

DOE/ID Approval Letter
From: J. E. Solecki
To: L. P. Leach
Date: September 16, 1982

FOREWORD

This document is intended to provide, at an early stage, definition of test objectives, configuration, initial conditions, measurement requirements, preliminary analysis, and scenario for the L9-4 ATWS test. In addition, a discussion of special conditions and requirements to meet test objectives is provided. The information provided in this document should be used to initiate the Experiment Prediction (EP) and Experiment Safety Analysis (ESA) and to initiate planning of instrument and data acquisition requirements and system configuration modifications. An Experiment Operating Specification (EOS) will be forthcoming to finalize the special test requirements.

CONTENTS

FIGURES

ABBREVIATIONS

ATWS	Anticipated Transient Without Scram
BOL	Beginning of Life
DAVDS	Data Acquisition and Visual Display System
EOL	End of Life
EOP	Experiment Operating Procedure
EOS	Experiment Operating Specification
ESA	Experiment Safety Analysis
LEPD	LOFT Experiment Program Document
LFD	LOFT Facility Division
LOFT	Loss-of-Fluid Test (Facility)
m	Mass Flow
MFP	Main Feedwater Pump
MLHGR	Maximum Linear Heat Generation Rate
MSV	Main Steam Valve
NE	Nuclear Experiment
NRC	Nuclear Regulatory Commission
ODDS	Operation Diagnostic and Display System
PCCS	Primary Component Cooling System
PCP	Primary Coolant Pump
PCS	Primary Coolant System
PPS	Plant Protection System
PWR	Pressurized Water Reactor
QOBV	Quick-Opening Blowdown Valve
SCS	Secondary Coolant System
SDD	System Design Description
SG	Steam Generator

AT_{av}	Average of T_h and T_c
T_h	Hot leg primary coolant bulk temperature, reactor vessel exit bulk temperature
T_c	Cold leg primary coolant bulk temperature, reactor vessel inlet bulk temperature
TBS	To Be Supplied
T_{sat}	Saturation Temperature
V	Volume
<u>W</u>	Westinghouse
$\Delta\rho$	Reactivity Change

LOFT EXPERIMENT DEFINITION DOCUMENT L9-4 ANTICIPATED TRANSIENT WITHOUT SCRAM EXPERIMENT

1. INTRODUCTION

Anticipated transients without scram (ATWS) have been the subject of discussions and analyses within the nuclear industry since early 1969, and have been designated an unresolved safety issue by the Nuclear Regulatory Commission (TAP A-9).[1] The significance of ATWS in the evaluation of 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. The potential extent and probability of serious consequences resulting from an ATWS is detailed in Reference 2.

Since the Nuclear Regulatory Commission (NRC) considers the risk associated with ATWS events sufficient to justify their consideration,[2,3] LOFT Test L9-4 has been developed to gain a better understanding of system response characteristics for a postulated ATWS, and to determine the ability of existing analytical techniques to predict these response characteristics.

The judgement by the NRC as to whether nuclear power plants meet the standard of safety (i.e., severe radiological consequences to the public) required depends primarily on two factors: (1) the reliability of current reactor scram systems (ATWS prevention) and (2) the capability of existing reactor designs to mitigate the consequences of ATWS events (ATWS mitigation). The capability for mitigation is amenable to analysis and testing. In the case of scram system reliability, only limited information can be obtained from statistical analysis of operating experience and from studies of common mode failures.

In evaluating ATWS events the NRC denotes[1] ten initiating events for PWRs, which are expected to occur one or more times during the life of a nuclear power plant. These events are classified into four categories:

(1) reactivity related accidents, (2) degradation of reactor heat transfer, (3) degradation of reactor heat sink, and (4) primary system depressurization. For any initiating event in these categories, the NRC requires ATWS analysis to be performed using acceptable computer codes. The objective of the methodology is to use computer models that realistically predict the course of the ATWS event sequences being analyzed. The appropriateness and validity of the calculations are required to be supported by experimental evidence. The L9-4 experiment encompasses portions of all four NRC categories.

The planning of the L9-4 experiment is based on NRC's regulatory position defined in References 2, 3, and 4, and in Chapter 15 of Regulatory Guide 1.70. The L9-4 experiment is intended to simulate the important physical phenomena following a loss-of-offsite power and failure-to-scram transient. The NRC system transient codes which are used in ATWS analyses will be assessed against this test data. These computer codes are intended to realistically calculate the behavior of the plant following an ATWS event. Since conservatism in PWR relief valve flow areas and heat transfer degradation are inherent in the codes due to lack of test data, the L9-4 transient will serve to provide the answers to questions arising from Category 2 and 3 transients in regards to a mismatch between power generation and heat removal capability.

A loss-of-offsite power event has a probability of occurrence of 0.27% per plant year. This probability is a best-estimate frequency based on actual PWR power plant operation.[5]

A loss-of-offsite power can be caused by problems within the generating station or by problems external to the generating station. Although highly unlikely a turbine trip could be postulated to cause a loss-of-offsite power by creating a grid voltage and/or frequency disturbance large enough to upset the entire grid. Turbine trips are usually caused by problems with the station turbine generator. The station turbine generators are protected from damage by automatically tripping (shutting off) the turbine when parameters such as bearing temperature, vibration, or lubricating oil pressure exceed safe limits.

The more probable causes of a loss-of-offsite power are voltage/ frequency disturbances on the grid itself. These grid disturbances can be caused by upset conditions at other generating stations or by any number of events which effect the grid transmission lines. Lightning, tornadoes, floods, wind and ice storms have been known to cause a loss-of-offsite power. Car or airplane crashes into transmission lines or line supports and sabotage could also be postulated to cause a loss-of-offsite power.

The specific justifications for the L9-4 test are as follows:

1. Although an actual loss-of-offsite power with a failure to scram is a very unlikely event in a PWR, it encompasses other ATWS events which are more probable. These events include a loss of forced flow, a turbine trip, and loss of condensor vacuum. All of these events are anticipated transients which must be considered in safety analysis reports.

2. This transient will represent a significant challenge to existing computer codes. In addition to having to predict peak pressure and reactivity feedback, the codes will have to predict natural circulation under ATWS conditions, and a more diverse basis for code evaluation will result. Also, the transient represents the most demanding test for validating point kinetics approximations used in the calculation of transient reactor power.

2. TEST OBJECTIVES

To address issues relating to the consequences of a postulated ATWS, the following major programmatic objective has been defined for the LOFT ATWS experiment. This objective is:

1. Provide experimental data for benchmarking PWR vendor's ATWS computer codes as required by the NRC proposed ATWS rule (USNRC SECY-80-409).

To support the above programmatic objective, several test specific objectives have been identified. The test specific objectives for Test L9-4 are:

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

2. To provide data for analysis of the effect of natural circulation cooling capability under high power conditions.

3. To 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.

3. EXPERIMENT DESCRIPTION

The following scenario describes the Zion station response to an extended loss-of-offsite power. The loss-of-offsite power will be assumed to occur as the initiating event. All protective and control systems are assumed to operate as designated. Upon the loss-of-offsite power, there is an immediate main generator trip which trips the turbines. The reactor fails to scram due to mechanical distortions of either the control rods and/or the reactor core. Plant auxiliary power is also lost when the main generator and the preferred power source (offsite power) are unavailable. At this point, since the electrical generator will supply some power (due to inertia), all operating pumps will coast (within 40 seconds) to a stop. This includes:

a. The steam generator main feedwater pumps;
b. The circulating water pumps which provide cooling for the condenser;
c. The condensate pumps;
d. The component cooling water pumps;
e. The service water pumps which provide cooling for various heat exchangers; and,
f. The charging pumps.

In addition to the pump coastdowns, air compressors (i.e., instrument air) will also be disabled in a large plant with the result that the pneumatically operated valves cannot function properly. Therefore, the power operated relief valve will not be available for long-term plant pressure control.

At this point, only diesel power is left to supply power to some components in the reactor system. A startup signal will be sent to the diesel generators which provide emergency power for the engineered safety features and other equipment on the essential bus network, the most important of these being the emergency core cooling systems, the auxiliary feedwater system, and the component cooling system. The diesel loading sequence is designed to be completed within 30 seconds.

The system status for Zion at approximately 30 seconds into the transient would include the following:

1. Reactor fission power is greater than zero;
2. Primary coolant pumps are coasting down;
3. Main steam generator feedwater flow is isolated;
4. Steam generator steam flow is isolated;
5. Auxiliary feedwater is being delivered to the steam generators by the turbine driven auxiliary feed pump.

Once the transient is initiated, there will be an immediate rise in the primary system temperature and pressure due to a decrease in the steam generator heat sink efficiency. As the pumps coastdown, the temperature gradient across the core will increase. Due to the rise in coolant temperature, a significant amount of negative reactivity will be contributed by the moderator temperature coefficient, and the reactor power will decrease with the increase in the core fluid temperature. As the power decreases, the fuel temperature decreases, resulting in a positive reactivity insertion due to the fuel Doppler feedback. This retards the reactor power decay. After the pumps coastdown, natural circulation will be established in the primary coolant system. The ATWS at this point is reduced to a problem of whether heat addition from the core can be balanced by steam generator heat removal in a natural circulation mode.

At this juncture, the main feedwater to the steam generator has stopped, auxiliary feedwater is being delivered, and the main steam flow paths to the turbines are blocked. The pressure will rise in the steam generator secondary side and the relief valves[5] will start opening from 1050 psia to 1100 psia. Once natural circulation is achieved after the pumps coastdown, the steam generator liquid inventory will continue to be reduced despite auxiliary feedwater until steam generator dryout.

Since instrument air is lost and the primary pumps have coasted down, the pressurizer sprays do not initiate. Thus, the system continues to increase in temperature and pressure and the coolant density decreases. This causes some reactivity feedback and the reactor power decreases, but not enough to keep the system pressure from increasing. Eventually, the system pressure would rise to the pressurizer power operated relief valve (PORV) setpoint (16.2 MPa, 2350 psia). Since the PC¬V would open at this point, system pressure would decrease. However, since the PORV would be operating on an accumulator air system (i.e., instrument air is lost), after approximately 12 cycles (i.e., Zion), the PORV would fail in the closed position. The pressure would then continue to rise until the code safety valves opened (17.235 MPa, 2500 psia) which would mitigate the pressure increase.

The L9-4 test will be initiated from similar operating conditions and plant configuration simulating a typical W four loop plant (i.e., Zion) should it undergo an ATWS. It is expected that this test will display the same trends as expected in a PWR, but the magnitudes and response times will be slightly different because of scaling considerations.

Prior to the initiating event, the reactor and all support systems are assumed to be in a normal configuration. The reactor is assumed to be operating at 100% of rated power. All control systems are assumed to be in the automatic mode of operation and all key parameters are assumed to be within technical specification limits.

Loss-of-offsite power for short durations have been shown[6] to not have any significant effect on a nuclear power plant. Therefore, the scenario postulated in this study will be assumed to be the result of an extended loss-of-offsite power caused by upset conditions on the transmission lines. The sudden loss-of-offsite power will be assumed to cause a turbine trip.

3.1 Preliminary Analysis

Preliminary analysis for the L9-4 transient was accomplished with the RELAP5-CY15 version of the code. The RELAP5[7] program is a comprehensive code that predicts the interrelated effects of core neutronics, system thermal-hydraulics, and system component interactions.

Calculations for both the LOFT and Zion facilities were accomplished to plan the L9-4 transient. To simulate the L9-4 transient, the following major modifications were required to the base ATWS models for both LOFT and Zion:

1. Preventing a reactor scram.
2. Tripping the primary coolant pumps at time zero.
3. Steam generator modifications:
    a. Auxiliary feedwater system modeled
    b. Loss of main feedwater modeled at time zero
    c. Loss of main steam flow path at time zero
    d. Relief valves modeled.
4. Pressurizer modifications:
    a. Loss of pressurizer spray system
    b. Plant PORV valve characteristics modified to fail closed immediately (LOFT only)
    c. Safety valves modeled.

With the initiation of the transient, the PCPs for both systems begin to coast down, immediately inducing a decrease in the PCS flow. Within 3 minutes, the code predicts natural circulation is achieved in both systems, for the remainder of the transient.

Figure 1 illustrates the RELAP5 prediction of the LOFT and Zion steam generator liquid level behavior during the transient. As shown, the code predicts a complete dryout in both systems at approximately 650 seconds. As the steam generator heat sink degrades during this time, the temperature gradient across the core first increases rapidly until natural circulation is established. This results in an initial rapid increase in core outlet temperature as predicted by RELAP5 in Figure 2 for both plants. The peak coolant temperatures (642°F, 612 K) for both plants are achieved early in the transient, and both decrease only slightly from the maximum for the remainder of the transient. Figure 3 illustrates the

RELAP5 prediction of the system coolant temperature downstream of the steam generator at the core inlet. Since the LOFT steam generator heat removal capability is not totally degraded for the first 325 seconds of the transient, the core inlet temperatures remain relatively constant. However, slightly past this point, the steam generator heat transfer has degraded to the point where only a fraction of the core input power can be dissipated. Therefore, the fluid temperatures rise considerably and maintain these higher temperatures the remainder of the transient. The Zion calculations indicate total loss of heat sink at 850 seconds. The corresponding response in system pressure for the two plants is illustrated in Figure 4. In the Zion plant the system pressure climbs rapidly to the pressurizer PORV setpoints (2250 psia, 16.21 MPa). The resulting mass loss from the system results in a pressure decrease. As the steam generator dries out, the system pressure increases to the relief valve setpoint (2500 psia, 17.24 MPa). Since the PORV will not be used in the L9-4 experiment, it is not accounted for in the LOFT calculation. Thus, the system pressure increases immediately to the relief valve setpoint. As shown in Figure 5, the safety relief valve flow mitigates several LOFT pressure excursions. These pressure increases are also attributed to the decrease in steam generator secondary liquid level and subsequent loss of heat removal capability. The pressurizer liquid level rise for the two plants is shown in Figure 6.

The effect of the transient on reactor power is illustrated in Figure 7. The reactor power decreases sharply for both plants to 16% of total power within 35 seconds and maintains this level until total loss of the steam generator heat sink. The LOFT power then decreases to 3% of total power at 500 seconds, which is maintained for the remainder of the transient. Zion power decreases to 2% of total power at 850 seconds for the remainder of the transient. Core power is calculated from the reactor kinetics equations in RELAP5. The driving function for these equations is the reactivity. The primary contribution to the reactivity for this transient is attributed to changes in the fuel temperature (Doppler feedback), water temperature and density perturbations (moderator feedback). Figure 8 illustrates the LOFT and Zion reactivity predictions for this transient. As shown in Figure 8 the LOFT predicted reactivity decreases rapidly to -1.55 dollars at 31 seconds. This decrease in total reactivity is attributed to the large negative reactivity insertion induced by the increase in moderator temperature as shown in Figure 9. As shown, initially the large positive Doppler contribution due to the decrease in the average fuel temperatures cannot compensate for the larger moderator effect, resulting in a power decrease. Past this time, the Doppler contribution balances the moderator feedback resulting in a minimal negative reactivity insertion to 325 seconds. At 325 seconds the effect of the loss of heat sink results in an increase in moderator temperature and a larger negative reactivity. Gradually past this point, the LOFT reactivity decreases to minimum values the remainder of the transient.

3.2 System Configuration

The LOFT system configuration is shown in Figure 10. During the L9-4 transient, the inactive broken loop coolant has negligible effects on the system response. For this reason and consistent with plant modification constraints, the steam generator and primary coolant pump simulators in the hot leg broken loop should be isolated or removed from the system.

The system configuration should also include:

1. The reflood assist bypass valves CV-P-138-70 and CV-P-138-71 shall remain closed during the test.

2. A reactor scram should be prevented during the transient.

3. The pressurizer cycling and backup heaters should be inoperative during the transient.

4. The experimental PORV and safety valves shall be simulated by a single valve, CV-P-139-87, with a double actuator such that the first position corresponds to the PORV and the second position corresponds to the safety valve. To simulate the loss of PORV relief capability the PORV setpoints should be the same as the

safety valve: Open - 2500 psia, close - 2425 psia. The valve stem settings used for the L9-3 test should be used for the L9-4 test. These settings will yield scaled PWR relief flows.

5. The plant PORV isolation valve CP-P-139-18 should be open and the plant PORV CV-P-139-5-4 should be inactivated for the test.

6. The plant safety valves CV-P139-200 and CV-P139-201 lifting setpoints should be 2800 psia.

7. The main steam isolation valve CV-P4-11 shall be open. The main steam control valve CV-P4-10 shall start to close at the initiation of the transient, and be closed in 13 seconds.

8. The main feedwater shall be shut off at the initiation of the test. The auxiliary feedwater shall be operative in accordance with the requirements in Section 6.

9. The steam generator relief valves will not be used for the L9-4 test. Normally the setpoints are 1100 psia for RV-136 and 1120 psia for RV-137. Parametric analysis with RELAP5 indicated the relief valves to open continuously throughout the transient. However, the relief valves discharge to a collection tank which vents directly to containment. To alleviate this problem, the steam bypass flow will be utilized to prevent any flow out of the relief valves. The operators should monitor the secondary side pressure and prevent it from exceeding 1000 psia by throttling the valve (CV-P4-90). Secondary pressure should be maintained in the range of 950 to 1000 psia for the transient. If the bypass valve is not sufficient to maintain the range specified, the steam flow control valve (CV-P4-10) should also be used.

10. ECC system should be inhibited above a system pressure of 1800 psig during the loss-of-offsite power transient.

4. INITIAL CONDITIONS

The LOFT L9-4 test initial conditions have been designed to approximate commercial PWR operating conditions and at the same time to be consistent with the safety analysis of the LOFT plant itself without compromising experiment objectives.

Prior to the test, the necessary hardware configurations will be established and the plant heated to normal operating temperature using pump heat. The reactor will then be brought critical and power raised to the required power level and held there until the initial conditions noted below are established and stabilized. The reactor will then be held at this power level for a specified time to ensure the desired decay heat buildup. The test will then be initiated by performing the specified initiating events. The RSS should be inhibited to prevent a reactor scram during in the test.

The L9-4 experiment initial conditions are:

1.	Reactor power	50^{+0}_{-1} MW
2.	Cold leg temperature	544 ± 2°F (557 K)
3.	Core differential temperature	38 ± 2°F (21 K)
4.	Pressurizer pressure	2169 ± 15 psia (15 ± 0.1 MPa)
5.	Pressurizer level	46^{+0}_{-2} inches

The L9-4 experiment initial conditions (continued):

6.	PCS flow	As required[c]
7.	Steam generator level	10 ± 2 inches[a]
8.	SG secondary conditions	As required[c]
9.	Control rod positions	54 ± 0.5 inches[b]
10.	Boron concentration	As required[c] (~725 ppm)

a. The steam generator water level is defined as 0.0 at 116 inches above the top of the tube sheet.

b. Above full in position.

c. As required to obtain primary conditions.

5. MEASUREMENT REQUIREMENTS

Details of existing LOFT instrumentation may be found in Reference 8. Prior to heatup, instrument calibrations and checks will be performed. Anomalous measurements will be identified and corrections made.

Data will be recorded from approximately 700 instruments during the test. The measurements listed below will provide data from key points throughout the primary and secondary system and necessary associated systems. Data will be recorded until test termination. After the test DAVDS calibrations and performance checks will be performed per standard LOFT practice

The data will be collected and formally reviewed by the Data Integrity Review Committee (DIRC). Selected data will be published in the Quick Look Report (QLR) and Experiment Data Report (EDR).

The following measurements are considered adequate to characterize the transient.

Density	Intact Loop Cold Leg Intact Loop Hot Leg Upstream of Experimental PORV
Velocity	Cold Leg Hot Leg Core Inlet Core Outlet
Mass Flow Rate	Pressurizer Experimental PORV and Safety Valve Intact Loop Hot and Cold Legs Feedwater Main Steam Auxilliary Feedwater

Pressures	Pressurizer
	Intact Loop Hot Leg
	Intact Loop Cold Leg
	Upper Plenum
	Lower Plenum
	Downcomer
	Feedwater
	Steam Generator Dome
Differential Pressures	Steam Generator PCS
	Across Primary Coolant Pumps
	From Pump to RV Inlet
	Across Core
	From RV to Pressurizer
	From Pressurizer to SG
	From SG to Pumps
	From Intact Loop Hot Leg to Top of RV
	Surge Line
Temperatures	Cladding
	Core Coolant
	Downcomer
	Upper Plenum Coolant
	Upper Plenum Metal
	Lower Plenum Coolant
	Steam Generator
	Reactor Vessel
	Feedwater
	Intact Loop Hot Leg
	Intact Loop Cold Leg
	Pressurizer
Liquid Levels	Pressurizer
	Steam Generator

Power/Reactivity	Control Rod Position
	Core Power
	Core Reactivity
	Neutron Flux
Momentum Flux	Hot Leg
	Downcomer
	Core
Miscellaneous	Feedwater Flow Control
	Valve Position
	PCP Pump Speed
	Experimental PORV Position
	Steam Flow Control Valve Position

6. SEQUENCE OF EVENTS

1. Operate at 50 MW prior to the transient to establish a decay heat level ≥850 kW at 1000 seconds into the transient. Bring the reactor to the initial conditions specified in Section 4.

2. Initiate the test by tripping the PSMG motor breakers which will trip the PCPs (PCP-1, PCP-2). Also trip the MFP, and close valves CV-P4-8 (feedwater flow control valve) and CV-P4-91 (feedwater flow control valve bypass valve).

3. Prevent the plant from scramming during the transient. Inhibit ECC injection above a system pressure of 1800 psig during the test.

4. Initiate the closing of CV-P4-10 (main steam control valve) at test initiation.

5. The auxiliary feedwater flow will be initiated at 10 seconds into the transient at 8 gpm.

6. Initiate recovery at 1500 seconds by inserting control rods and recovery the plant in accordance with the Experiment Operating Procedure.

7. DISCUSSION

In PWRs, the limiting transient with respect to ATWS is a complete interruption in the delivery of feedwater to the steam generators at full power. Should the scram fail to shut the reactor down, the continued power generation and the declining heat removal, as the secondary coolant boils away, causes a rapid increase in reactor coolant system pressure. The severity of this pressure excursion is a sensitive function of the moderator temperature coefficient, the capacity of the relief valves attached to the reactor coolant system, and the speed with which the auxiliary feedwater flow is initiated.

The L9-4 transient has been designed to address NRC concerns stemming from analyses on PWRs which experience a loss-of-offsite power and failure to scram transient. Analysis indicates that in this postulated accident the loss of primary flow and degraded secondary heat removal capability in conjunction with the failure to scram results in a self-regulating system. The distinguishing characteristic of this scenario is the dissipation of the core power by the reactivity feedback effects and the degraded heat sink capability. The various considerations leading to the development of the L9-4 scenario together with special operating conditions and scaling compromises are discussed in subsequent paragraphs.

7.1 Primary Coolant System Flow Rate

A nominal steady-state PCS flow rate of approximately 3.52×10^6 lb/hr will be used for Test L9-4. This is higher than the scaled LPWR flow rate to conform with safety analysis requirements (m_{core}/V = 4.35 lbm/ft^3-sec LOFT, 3.23 lbm/ft^3-sec Zion). This will only affect the variables of interest (core ΔT, T_{av}, T_H) for the first minute of the transient, and preliminary analysis has shown that this difference does not influence overall system behavior following pump coastdown, and thus will not compromise test objectives.

7.2 Primary Coolant System Inlet Temperature

A typical commercial PWR (Zion) has a steady-state inlet temperature of 530°F and a ΔT of 65°F. Since this ΔT cannot be matched at the required high flow rate, the inlet temperature was chosen to give a T_{av} of 563°F (544°F + 38/2) which corresponds closely to the Zion average core temperature. This will aid in assessing the typicality of the moderator coefficient between the two plants.

7.3 Primary Coolant System Pressure

A nominal steady-state PCS pressure of 2169 psia will be used for Test L9-4. This has no effect on the test output as the transient pressure is governed by the decay heat, fission power, and heat sink capability.

7.4 Initial Power Level

The initial condition of maximum design power (50 MW) was chosen for Test L9-4 in order to maximize LOFT's power to volume ratio, and make it more typical of a PWR. Since the reactor power is drastically reduced in the L9-4 transient due to reactivity considerations, the fact that maximum linear heat generation rates are untypical will only slightly affect the Doppler contribution to the reactivity.

7.5 Control Rod Position

The control rod position of 54 inches withdrawn provides axial peaking factors representative of typical PWRs (lower one-third of core at BOL), and is consistent with rod heights of previous tests.

7.6 Pressurizer Level

During the L9-4 ATWS, maximum PCS pressures will occur as the pressurizer completely fills with liquid water from the expansion of the primary coolant. At this point relief flow from the pressurizer code test safety valves will decrease system pressure. The time required to achieve these maximum pressures is a function of the initial pressurizer liquid level.

The LOFT pressurizer level (46 inches) has been set to establish liquid volume which results in a ratio of pressurizer enthalpy to total PCS enthalpy equivalent to that in a commercial PWR (Trojan). The ratio of pressurizer steam volume to total PCS fluid volume for LOFT is also consistent with the Trojan PWR.

7.7 Pressurizer Sprays

Pressurizer sprays will be disabled during the initial portion of Test L9-4 in keeping with the accident scenario.

7.8 Pressurizer PORV and Relief Valve Setpoints

As discussed in Section 3, for a loss of offsite power the PORVs would fail closed after approximately 12 cycles during the transient. For the L9-4 transient, the PORV is assumed to have failed closed immediately. Therefore, the test valve CV-P-139-87 will model only the capacity of the relief valves in a large PWR. The test relief valve for the L9-4 experiment will lift at 2500 psia and reseat at 2425 psia consistent with code safety valve setpoints in most commercial plants. The same valve stem positions used on the L9-3 test should be used for the L9-4 tests.

7.9 Broken Loop Modifications

The broken loop will be isolated from the total experiment configuration for Test L9-4. By doing this for the test a more accurate calculation of the system shrink and swell can be made.

8. LOFT/PWR SCALING COMPARISON

The atypicalities that exist between LOFT and its prototype PWR are well documented.[9] LOFT was designed to scale significant features of a four loop PWR and reproducibly simulate typical system transient responses to a large break LOCA. Scaling criteria for various LOFT systems for Test L9-4 are discussed below.

8.1 Steam Generator

After steam generator dryout, practically all the heat generated in the reactor core will be dissipated in the primary coolant, raising its temperature. The larger the initial steam generator inventory, the longer will be the dryout time and consequently the lower the power level at the time of dryout due to moderator feedback effects. As illustrated in Figure 1, the LOFT steam generator secondary side inventory was depleted in the approximate time frame as the Zion calculation. However, the power prediction illustrated in Figure 7 indicates complete loss of heat sink occurs at 325 seconds for LOFT and 850 seconds for Zion.

The auxiliary feedwater flow will be initiated at ten seconds into the transient. This would be the fastest time for auxil[illegible] feedwater to be available for this type of transient in a commercial PWR. Nominally, anywhere from 420 to 900 gpm would be available in a large commercial plant (Zion) for auxiliary feedwater flow. Scaling (by power 1/65) this number to LOFT would result in a flow of approximately 6.5 to 14 gpm. Therefore, the minimum (8 gpm) auxiliary feedwater flow for LOFT was chosen.

8.2 Comparison of LOFT/PWR Heat Source/Sinks

The size of the LOFT facility and the scaling of LOFT components relative to a PWR may result in atypical results during the L9-4 test due to the effect of the various heat sources or sinks and their relative magnitudes. The areas of interest in LOFT for Test L9-4 include: a higher than scaled structural surface-area-to-fluid volume ratio which places less demand on energy removal systems; and, a higher relative structural and piping heat capacity than a PWR which means the LOFT structural heat sink will be greater in the L9-4 transient than a PWR, which contributes to the higher environmental heat losses. However, since energy removal out of the scaled test code safety valve and to the steam generator during the test are larger than the other heat sinks and sources, the effect of the higher than scaled surface area to volume ratio in LOFT will have a minimal effect on system response.

8.3 Comparison of Power-to-Volume Ratios

The coolant "volume" is defined as the volume of coolant that is heated up during the L9-4 transient. The effective volume of LOFT is 250.4 ft^3 (excludes ~26 ft^3 for the broken loop piping that is isolated for the experiment and the pressurizer steam volume). The Westinghouse Zion plant has a volume of 11390 ft^3 (excludes 1170 ft^3 for the pressurizer steam volume). The power to volume (P/V) ratios of LOFT and Zion respectively are 0.2 MW/ft^3 and 0.285 MW/ft^3.

The P/V ratio affects an ATWS transient by perturbations in the moderator reactivity feedback. As the coolant temperature rises, the moderator coefficient adds negative reactivity to the system causing a decrease in reactor power. The higher the P/V ratio the greater the temperature rise rate. At the same time the moderator feedback is faster which alleviates this difference. However, since the coolant expansion coefficient β,

where

$$\beta = \frac{1}{V}\frac{dV}{dT},$$

gets larger with higher temperature, this tends to result in faster increases in system pressure for larger P/V ratios.

8.4 Moderator Temperature Coefficient

The moderator in a PWR is a dilute solution of boric acid (~725 ppm boron for Test L9-4) in water. The boron in the solution is a strong neutron absorber and reduces the excess positive reactivity (chemical shimming) to allow criticality while the water molecules slow down high energy fission neutrons to thermal energies for further fission. As the moderator temperature increases, the moderating capability is reduced. This causes more neutron leakage from the reactor core and less neutron fissions. The reduction in boron density, on the other hand, decreases the neutron absorption rate. The net effect is a negative reactivity insertion.

A comparison of the LOFT and Zion (EOL) moderator reactivity curves is illustrated in Figure 11. For the L9-4 transient, normalized densities will be in the range of 1.0 to 0.9. The LOFT moderator temperature coefficient is more negative than the Zion coefficient over this range. Thus, it will give a faster and larger power decrease for the same rise in temperature.

8.5 Doppler Temperature Coefficient

The Doppler coefficient models the increased resonance absorption of neutrons in ^{238}U with increasing temperature. For the L9-4 transient, since the increase in moderator temperature decreases the reactor power, the net effect of the Doppler coefficient is to add positive reactivity to the core, thus regulating the system. A comparison of the LOFT and Zion Doppler reactivity curves is illustrated in Figure 12. The LOFT Doppler contribution will therefore be larger and faster than the Zion contribution for the same change in volume averaged fuel temperature, for the greater portion of the transient. The competing effects of Doppler and moderator serve to regulate the system reactivity, and thus power, faster in LOFT than Zion.

8.6 Primary Relief Capacity

The total PCS relief capacity in most commercial PWRs consists of relief flow from the PORVs and the code safety valves installed on the pressurizer. For example, the two Zion PORVs are sized to relieve 420,000 lbm/hr of saturated steam at 2350 psia while the three Zion safety relief valves are sized to relieve 1,260,000 lbm/hr of saturated steam at 2500 psia. The LOFT test relief valve capacities are power scaled (1 to 65) to a large PWR (Zion). The L9-4 test will utilize the L9-3 test relief valve configuration. The LOFT test relief valve is currently sized to relieve 4.46 kg/s of subcooled water at 2500 psia, or approximately 2.3 kg/s of saturated steam.

Preliminary analysis indicates the volumetric relief flow rate will be high enough to keep system pressure from increasing due to the primary coolant expansion rate.

8.7 Auxiliary Feedwater

In a loss-of-offsite power scenario, once the steam generator main feedwater pumps are tripped, the auxiliary feedwater will be available to deliver low temperature feedwater to the steam generators. As discussed previously, the LOFT auxiliary feedwater flow for Test L9-4 was power scaled, and will be 8 gpm during the test.

9. REFERENCES

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9. L. J. Ybarrondo, et al., "Examination of LOFT Scaling," ASME Paper 74-WA/HT-53, November 1974.

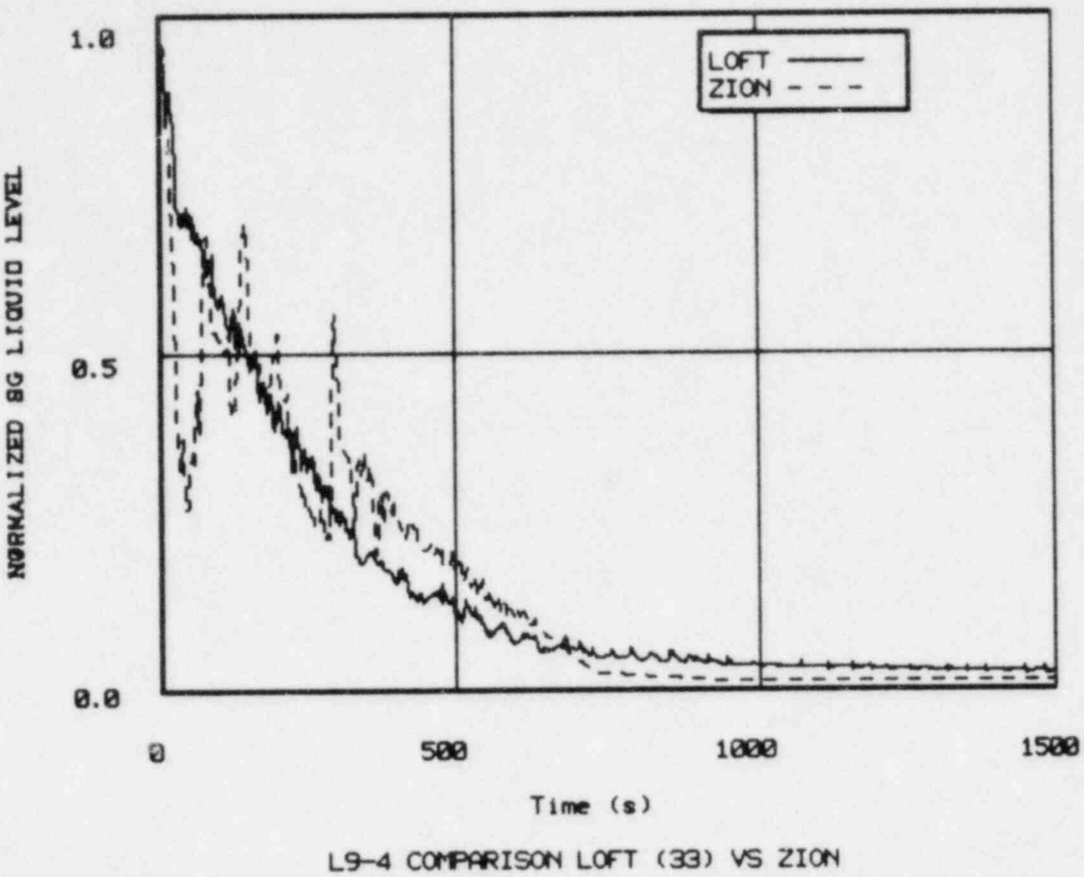

Figure 1. LOFT and Zion RELAP5 Steam Generator Liquid Level Predictions

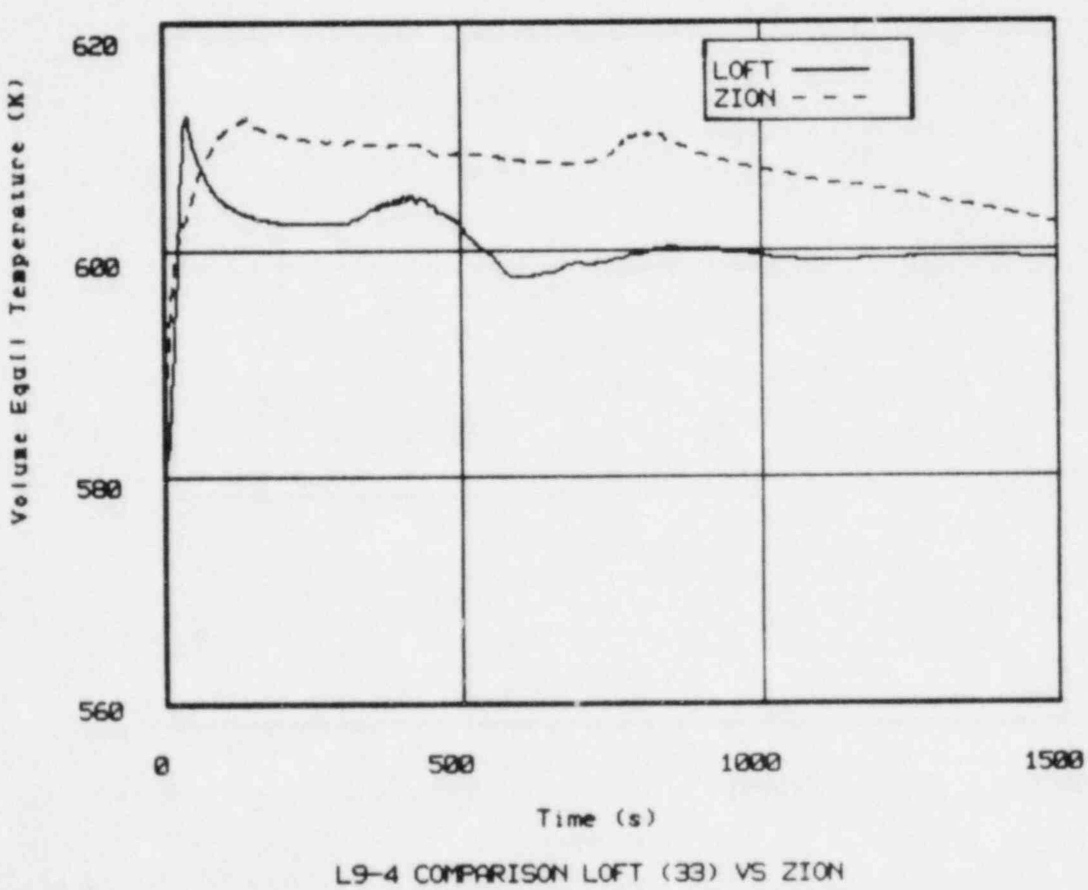

Figure 2. Core Outlet temperature RELAP5 predictions for LOFT and Zion

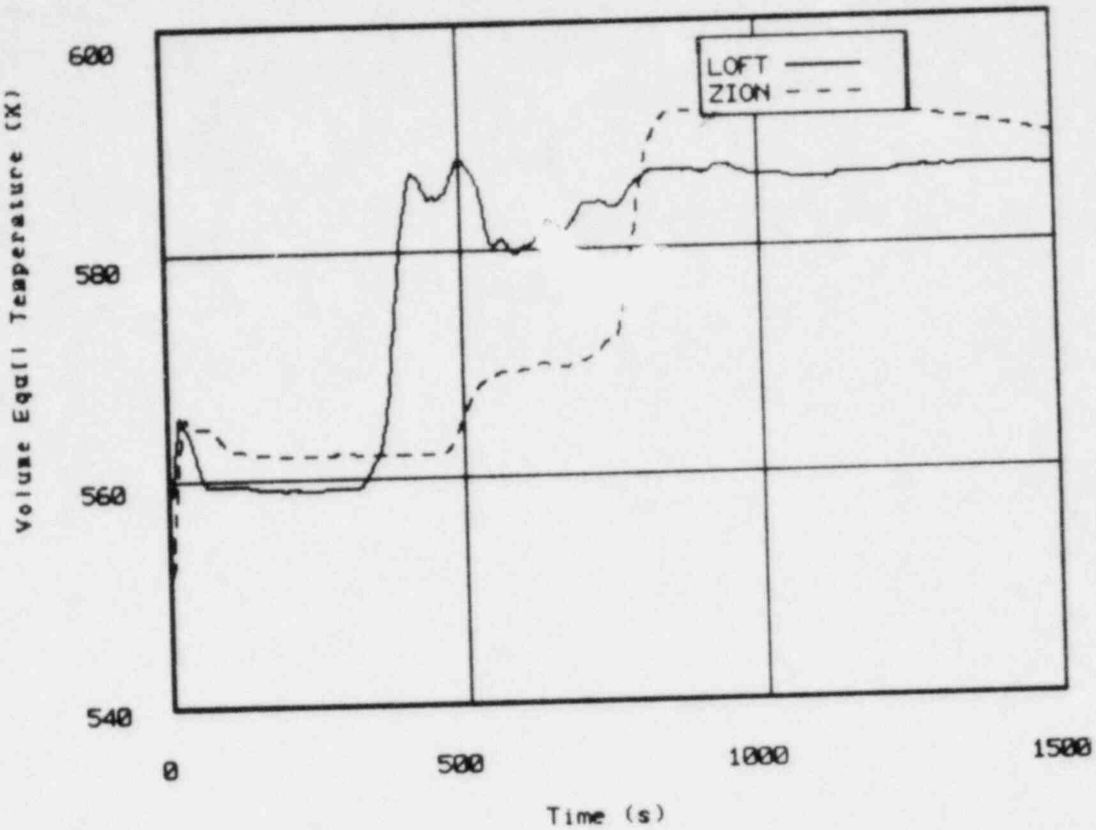

Figure 3. Core Inlet Temperature RELAP5 predictions for LOFT and Zion

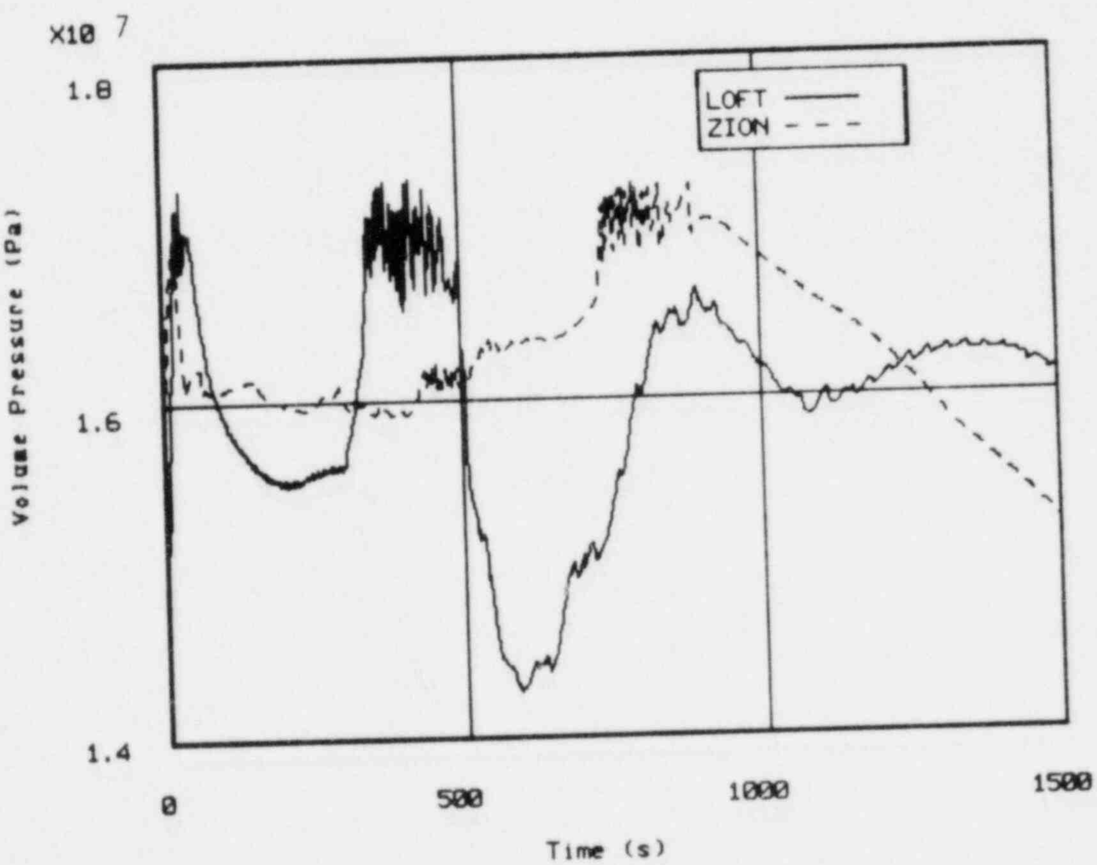

Figure 4. RELAP5 prediction of pressurizer pressure response for LOFT and Zion

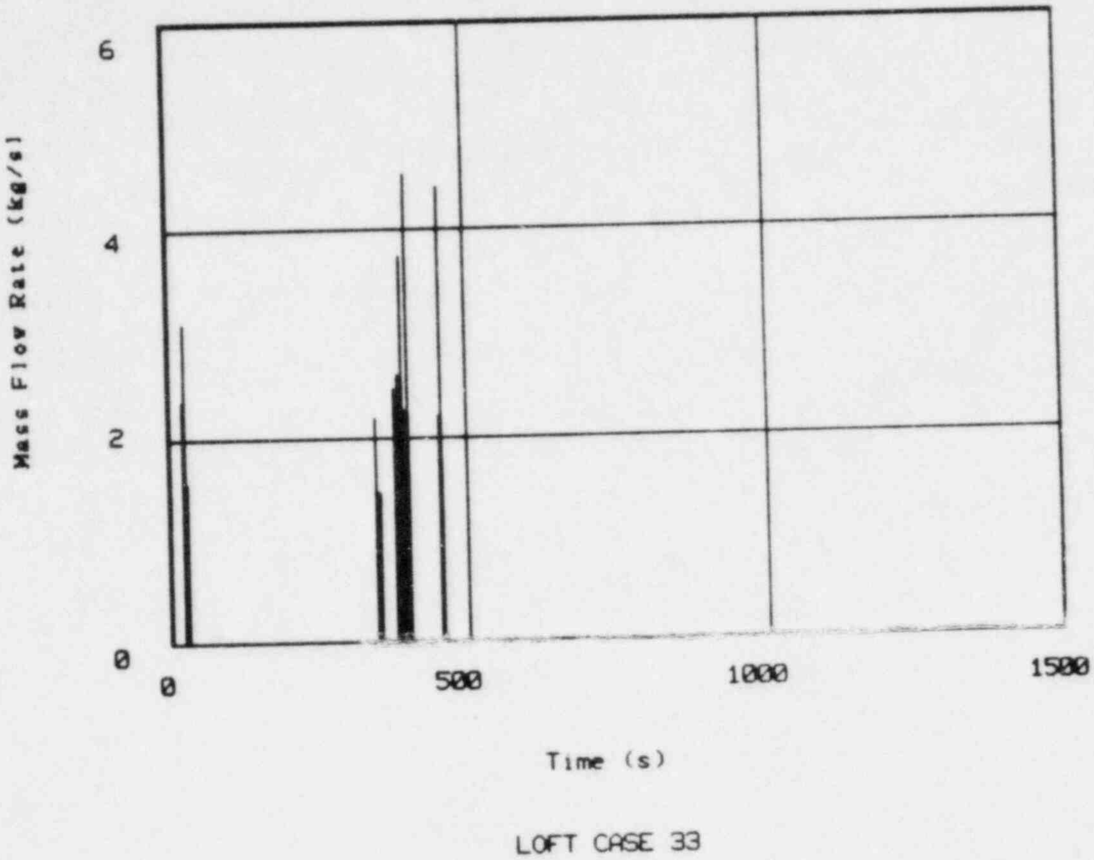

Figure 5. RELAP5 prediction of LOFT pressurizer safety relief valve flow rate.

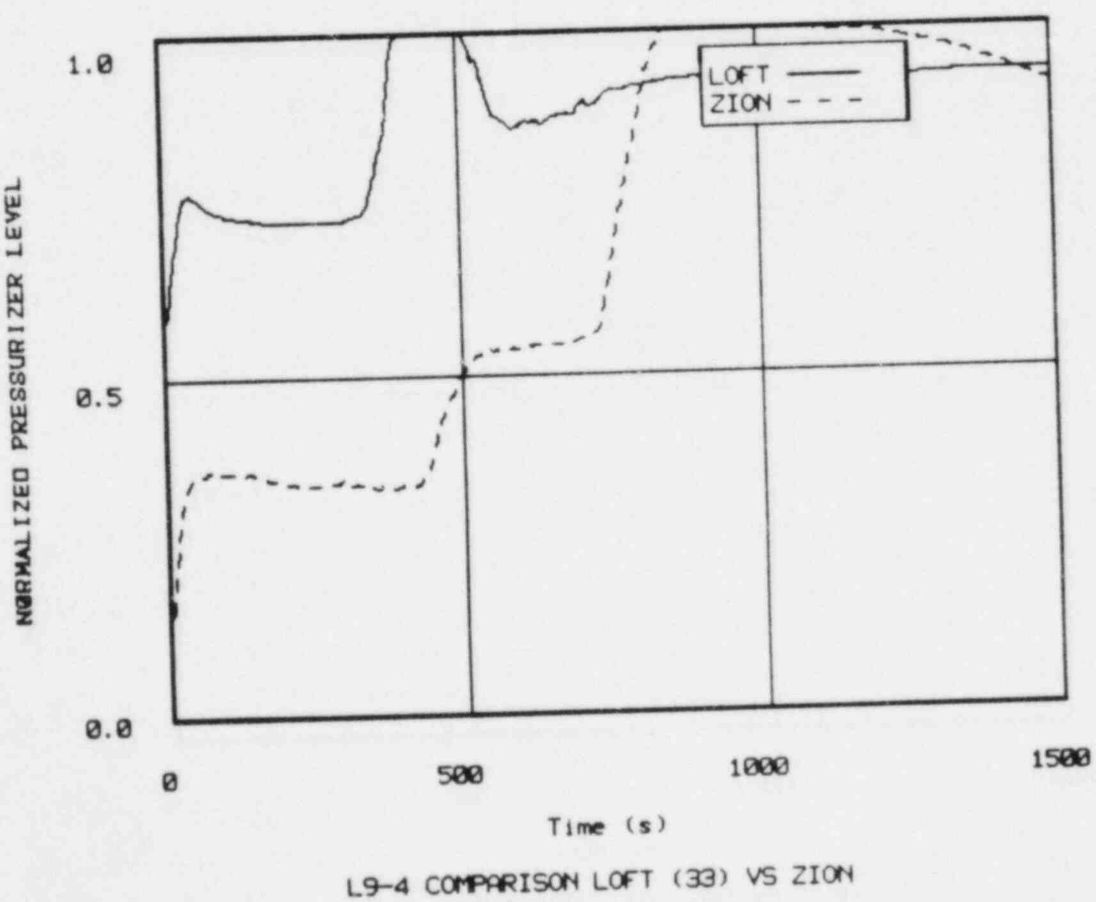

Figure 6. RELAP5 prediction of normalized pressurizer level for LOFT and Zion.

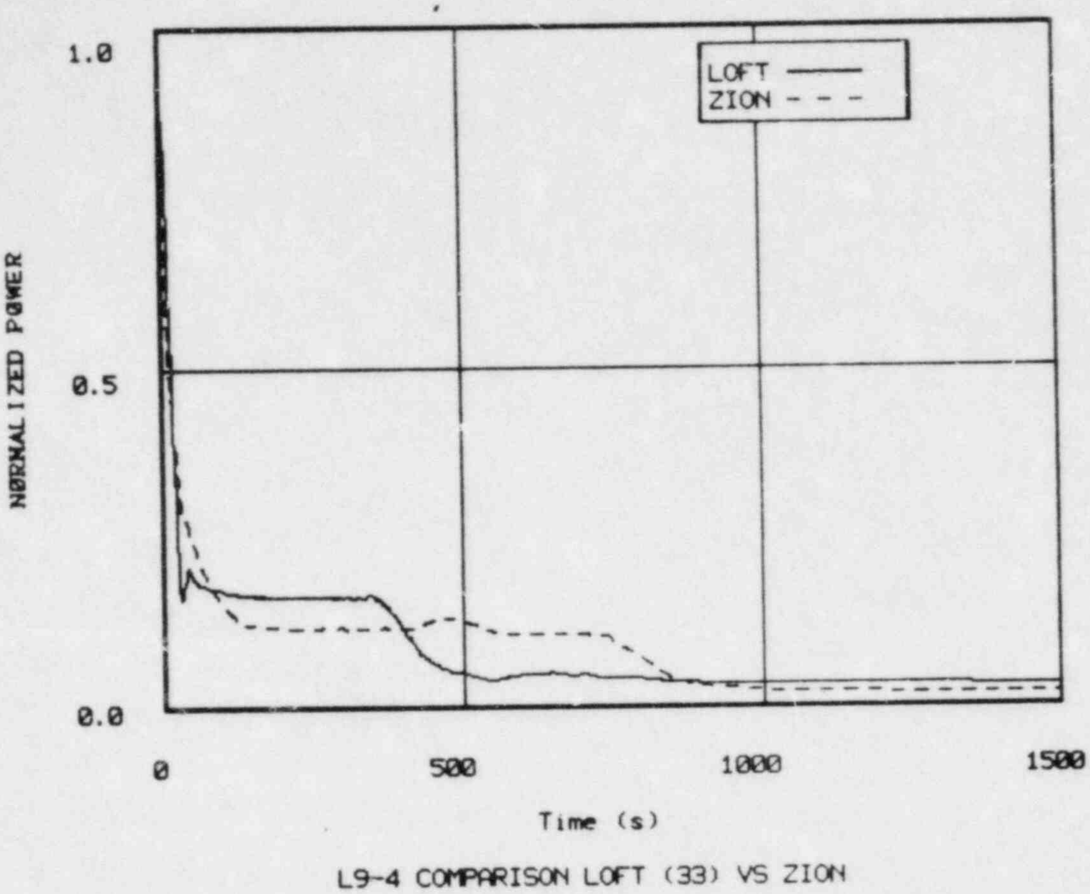

Figure 7. RELAP5 prediction of Reactor Power for LOFT and Zion.

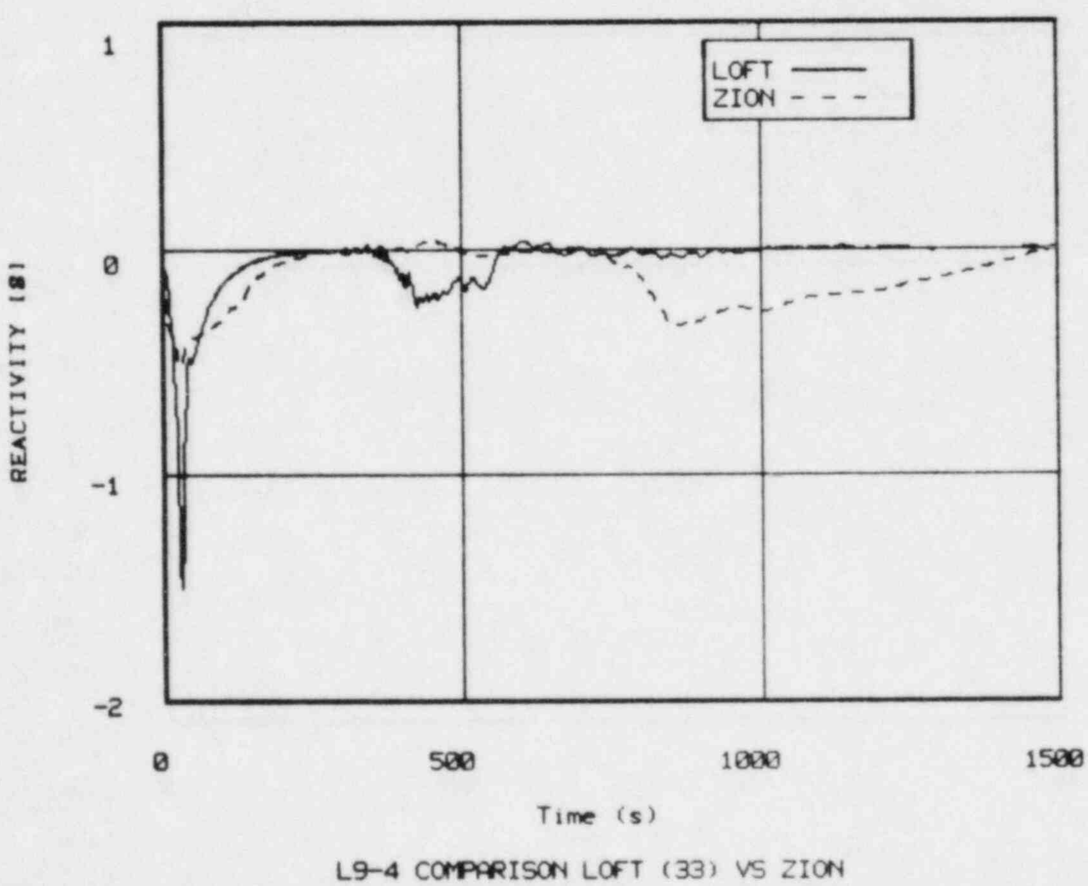

Figure 8. RELAP5 Reactivity Prediction for LOFT and Zion.

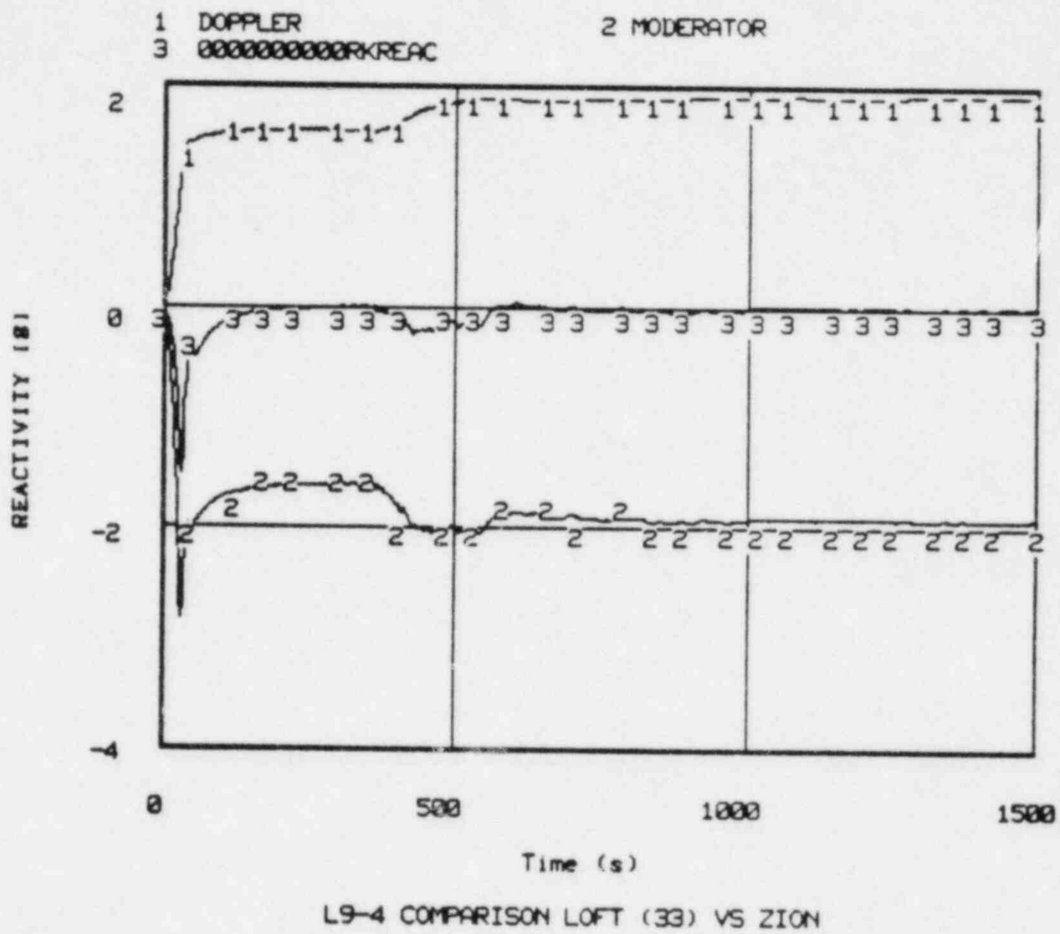

Figure 9. LOFT RELAP5 Reactivity Predictions.

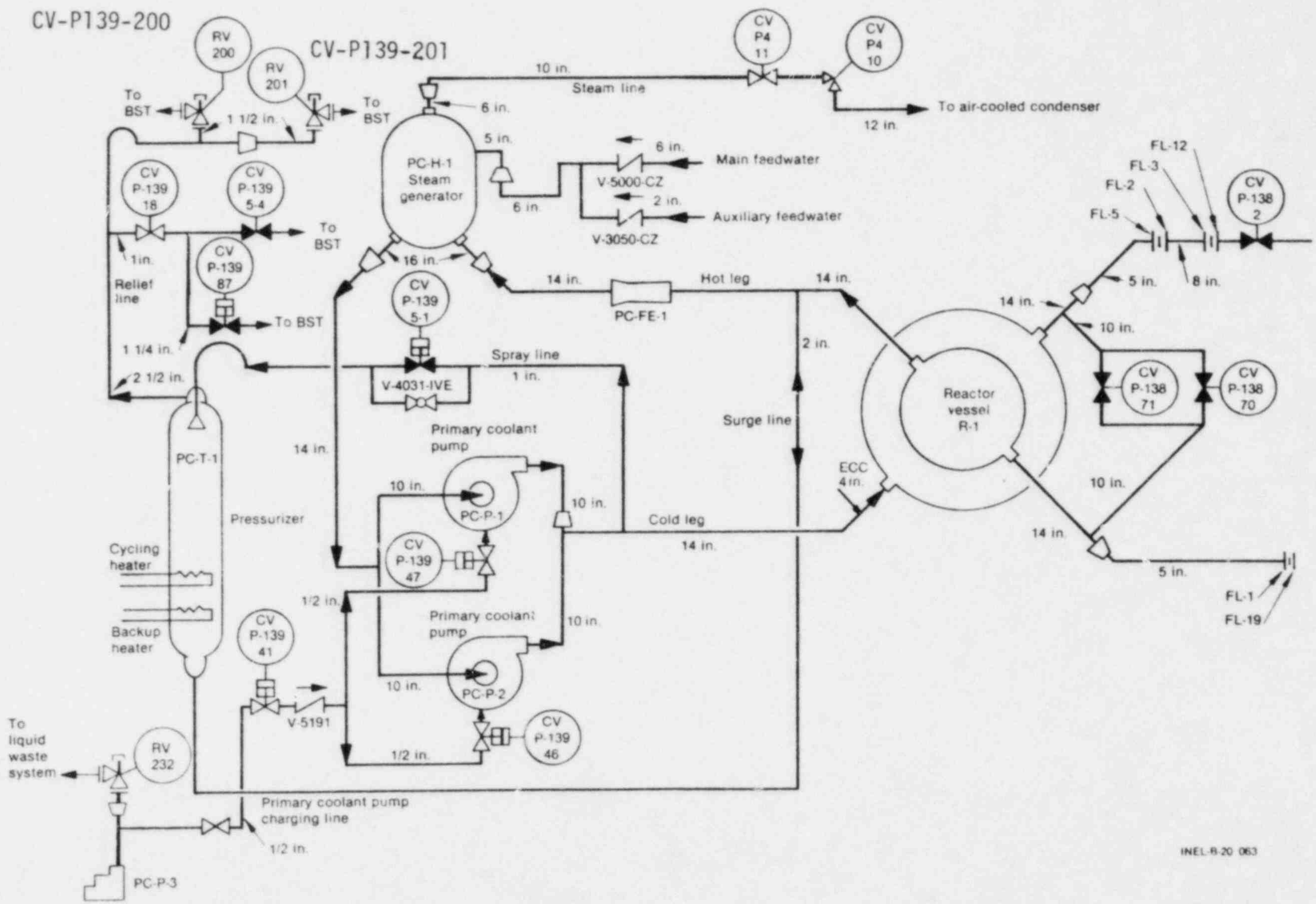

Figure 10. LOFT System Configuration

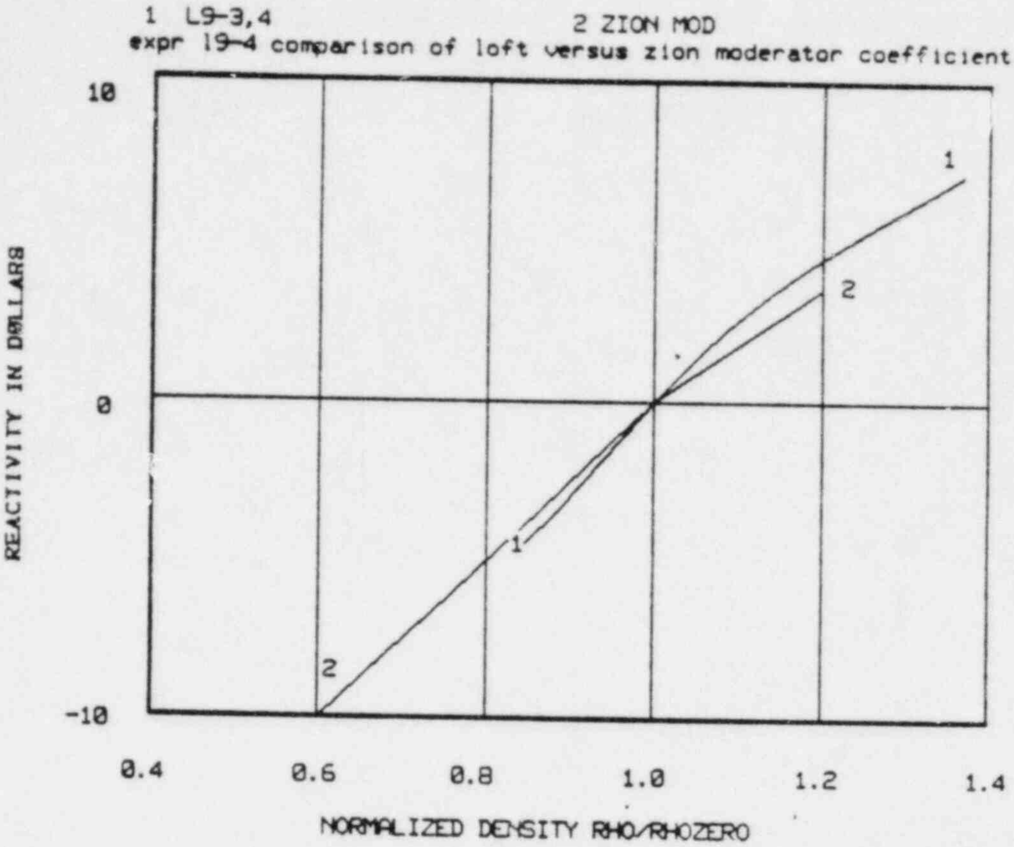

Figure 11. Comparison of LOFT and Zion Moderator Reactivity Curves

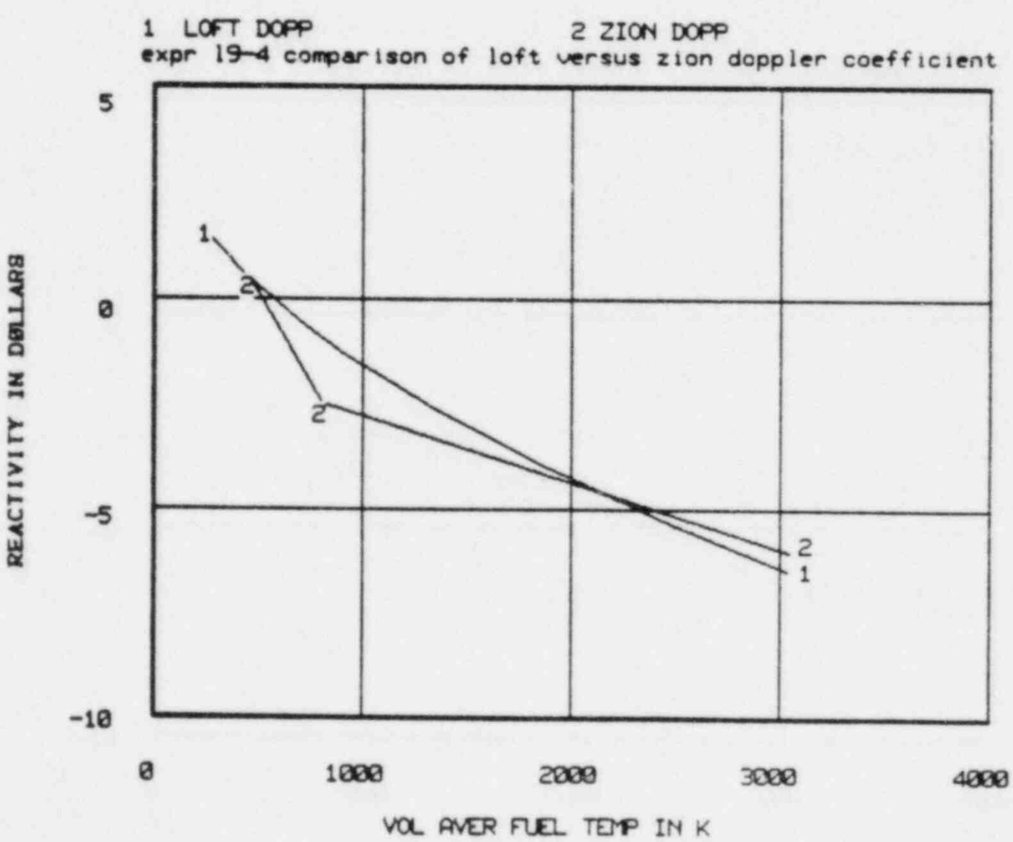

Figure 12, Comparison of LOFT and Zion Doppler Reactivity Curves