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LOFT EXPERIMENT DEFINITION DOCUMENT (EDD)
ANTICIPATED TRANSIENTS WITH MULTIPLE FAILURES
TEST SERIES L9
NUCLEAR TEST L9-1/L3-3/L8-1A

NRC Research and/or Technical Assistance Report

R. J. Beelman

U.S. Department of Energy
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INTERIM REPORT

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R. J. Beelman

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Responsible NRC individual and NRC Office or Division:

G. D. McPherson, Chief, LOFT Research Branch, Division of Reactor Safety Research, USNRC

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EG&G Idaho, Inc.
Idaho Falls, Idaho 83401

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LOFT EXPERIMENT DEFINITION DOCUMENT

ANTICIPATED TRANSIENTS WITH MULTIPLE FAILURES

TEST SERIES L9

NUCLEAR TEST L9-1/L3-3/L8-1A

Reviewed: Charles W. Solbrig
Manager, LOFT Program Division

E.L. Johnson by telecon 1/11/81
Manager, LOFT Facility Division

D J Hanson
Manager, LOFT Measurements Division

T.F. Pointer by telecon 1/11/81.
Manager, LOFT Technical Support Division

Approved:
Manager, LOFT Department

Authorized for Release:
Configuration Document Control and Services

DOE/ID Approval Letter
From: J. E. Solecki
To: L. P. Leach
Date: April 9, 1981

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FOREWORD

This document is intended to provide, at an early stage, definition of test objectives, configuration, initial conditions, measurement requirements, and scenario for the L9-1/L3-3 Loss-of-Fluid Test (LOFT) Experiment. 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.

ABBREVIATIONS

ATWS	Anticipated Transient Without Scram
BWST	Borated Water Storage Tank
BST	Blowdown Suppression Tank
CHF	Critical Heat Flux
DAVDS	Data Acquisition and Visual Display System
DBR	LOFT Design Basis Report
DOE	Department of Energy
ECC(S)	Emergency Core Cooling (System)
EOP	Experiment Operating Procedure
EOS	Experiment Operating Specification
ESA	Experiment Safety Analysis
ESF	Engineered Safety Feature
HEM	Homogeneous Equilibrium Model
HPIS	High-Pressure Injection System
JEG	Joint Experiment Group
LECS	LOFT Experiment Control System
LEPD	LOFT Experimental Program Document
LFD	LOFT Facility Division
LOCA	Loss-of-Coolant Accident
LOCE	Loss-of-Coolant Experiment
LOFA	Loss-of-Feedwater Accident
LOFT	Loss-of-Fluid Test (Facility)
LOFW-LOCA	Loss-of-Feedwater Induced Loss-of-Coolant Accident
LPIS	Low Pressure Injection System

LPWR	Large Pressurized Water Reactor
MFP	Main Feedwater Pump
MLHGR	Maximum Linear Heat Generation Rate
MTA	Mobile Test Assembly
NE	Nuclear Experiment
NRC	Nuclear Regulatory Commission
ODDS	Operational Diagnostic and Display System
P&ID	Piping and Instrument Diagram
PCCS	Primary Component Cooling System
PCP	Primary Coolant Pump
PCS	Primary Coolant System
PLS^2	Plant Log and Surveillance System
PNA	Pulsed Neutron Activation
POM	Plant Operating Manual
PORV	Power Operated Relief Valve
PPS	Plant Protection System
PSMG	Primary System Motor-Generator (Sets)
QOBV	Quick-Opening Blowdown Valve
SCS	Secondary Coolant System
SDD	System Design Description
SG	Steam Generator
SPGR	Specific Gravity
T_{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
TIP	Traversing In-core Probe
W	Westinghouse
$\Delta\rho$	Percent reactivity change = $\frac{K_2 - K_1}{K_1 K_2} \times 100$

where

K_1 = effective multiplication constant at the initial condition

K_2 = effective multiplication constant at the final conditions

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1. INTRODUCTION

In the event that a large pressurized water reactor (LPWR) experiences a loss of main feedwater and auxiliary feedwater is unavailable, an overpressurization of the primary coolant system (PCS) results as steam generator (SG) secondary heat removal capability degrades. No automatic emergency core cooling (ECC) actuation signal is initiated because PCS pressure remains above the ECC initiation setpoint. Furthermore, PCS pressure remains above the shutoff head of low head-high pressure injection system (HPIS) pumps and retards significantly the injection rate of safety grade injection HPIS pumps. A small break loss-of-coolant accident (LOCA) will result as the power operated relief valves (PORVs) open to relieve PCS pressure. If no action is taken to mitigate the loss of heat sink event, the PORVs will cycle repeatedly thereby removing decay heat and reducing PCS mass inventory until the core is uncovered. This transient may prove to be more severe than the normally analyzed design basis accidents.

Test L9-1 is designed to simulate a loss-of-feedwater induced loss-of-coolant accident (LOFW-LOCA) through the PORV. Code calculations indicate that without timely operator intervention engineered safety features (ESF) alone are insufficient to prevent core damage in the absence of auxiliary feedwater availability. The initiating event for Test L9-1 will be a manual trip of the main feedwater pump (MFP).

Test L3-3 is designed to simulate a typical LPWR LOFW-LOCA recovery scenario with the exception that emergency core cooling actuation is inhibited. Analysis[1] indicates that core damage is averted by holding open the POPVs within a specified maximum time after SG dryout. The initiating event for Test L3-3 is positioning the test PORV control switch to "open". The test PORV capacity may be insufficient to prevent PCS repressurization after holding open the test PORV and allowing the PCS to depressurize to saturation. Therefore, the secondary heat sink will be restored in Test L3-3 by refilling the SG.

L8-1A is intended to explore core thermal response under degraded core cooling conditions. The combination of operator actions described above is expected to reduce PCS pressure such that ECC accumulator injection can effectively recover the plant. However, in order to attempt an independent slow core uncovery divorced from the L9-1/L3-3 scenario, ECC accumulators will be valved out commencing with holding open the test PORV. The initiating event for test L8-1A will be 30°F superheat as evidenced by core cladding thermocouples. During the core uncovery primary coolant pumps (PCPs) will be restarted prior to final PCS recovery to assess their ability to reestablish flow in high void conditions and, subsequently, to assess the effectiveness of two-phase high quality forced convection on core cooling. The ECC high pressure injection system (HPIS) and low pressure injection system (LPIS) will be inhibited throughout the combined test scenario until final PCS recovery.

2. TEST OBJECTIVES

L9-1:

The major programmatic objective of Test L9-1 is to investigate a multiple failure scenario potentially more severe than design basis analysis. The test specific objectives are as follows.

1. To evaluate uncertainties in predicted primary and secondary thermal-hydraulic response associated with SG dryout during delayed scram.

2. To evaluate the adequacy of the PORV to provide overpressure protection in a loss of feedwater accident (LOFA).

L3-3:

Of the L3 Series objectives documented in the LOFT Experimental Program Document (LEPD), those series objectives addressed by L3-3 are:

1. To determine the important plant thermal, hydraulic, operational, and neutronic phenomena during a variety of small break loss-of-coolant accident (LOCA) tests in the LOFT facility. Identify and explain unexpected behavior.

2. To evaluate the effectiveness of current plant recovery methods to handle a small LOCA.

3. To determine the effectiveness of typical LPWR process instruments in providing accurate information on transient plant conditions.

4. To provide data to develop and test and Operational Diagnostic and Display System (ODDS) by operation of the system during each test.

The major programmatic objective of Test L3-3 is to evaluate the effectiveness of the PORV in mitigating the consequences of LOFAs. The test specific objectives are as follows:

1. To investigate uncertainties in system response during PORV imposed small break with loss of secondary heat sink.

2. To assess uncertainties in small break performance predictions identified in NUREG 0623[7].

3. To assess the effectiveness of steam generator refill on LOFAs following reestablishment of auxiliary feedwater availability.

4. To assess the relative magnitude of the change in reactor vessel mixture level as a result of PCS shrink during SG refill.

5. To contribute to the NRC relief and safety valve testing program by providing experimental data on PORV performance characteristics over a range of PORV inlet fluid conditions.

L8-1A:

The major programmatic objective of Test L8-1A is to investigate core thermal response under degraded core cooling conditions. The test specific objectives are as follows:

1. To determine core boil off thermal response under degraded core cooling conditions, while ensuring core reusability by maintaining core cladding temperature below 1500°F.

2. To determine PCP restart effectiveness on core cooling in high void conditions.

3. SYSTEM CONFIGURATION

The MTA will be reconfigured to install a scaled LPWR-typical PORV in parallel with the existing PORV. An equivalent flow area of 2.668×10^{-4} ft^2 (2.479×10^{-5} m^2) will provide 104.95 lbm/hr/MW of steam relief capacity at 2335 psig. ECCS injection will be into the normal automatic ECC injection points.

4. INITIAL CONDITIONS

Initial core power level will be 50 MW. PCS pressure will be 2171 psia (14.97 MPa). PCS flow will be 3.8×10^6 lbm/hr (478.8 KG/sec) at a cold leg temperature of 542.5°F (556.8 K). Pressurizer liquid level will be 40 inches (1.016 m). Steam generator level will be 126 inches (3.2 m) above the tube sheet.

5. MEASUREMENT REQUIREMENTS

The following measurements are considered adequate to characterize the transient.

Densities	Upstream of Test PORV
	Intact Loop Hot Leg
	Intact Loop Cold Leg
	Broken Loop Hot Leg
	Broken Loop Cold Leg
Mass Flow Rates	Upstream of Test PORV
	Intact Loop Hot Leg
	HPIS
	LPIS
	Feedwater
	ECC Accumulator
	Main Steam
Pressures	Upstream & Downstream of Test PORV
	Pressurizer
	Intact Loop Hot Leg
	ECC Accumulator
	Upper Plenum
	Feedwater
	Steam Generator

Temperatures	Upstream & Downstream of Test PORV
	Pressurizer (Liquid and Vapor Space)
	Intact Loop Hot Leg
	Intact Loop Cold Leg
	Feedwater
	Steam Generator (Primary and Secondary)
	Cladding Thermocouples
	Reactor Vessel (Upper and Lower Plenums)
Differential Pressures	Primary Coolant Pumps
	Test PORV
Liquid Levels	Steam Generator
	Blowdown Suppression Tank
	Pressurizer
	Accumumulator
	Reactor Vessel
Power/Reactivity	Power Range Nuclear Instruments
	Intermediate Range Nuclear Instruments
	Transient Reactivity Meters

6. SEQUENCE OF EVENTS

Pretest.1 Operate at 50 MW for a duration sufficient to establish a decay heat level not less than that corresponding to a minimum of 448 KW at 4000 seconds after shutdown.

Pretest.2 Inhibit HPIS and LPIS on the LOCE control panel.

L9-1.1 Initiate the test by tripping the MFP.

L9-1.2 Allow the plant to scram under automatic PPS control on High Hot Leg Pressure or High Hot Leg Temperature.

L9-1.3 Upon scram initiation allow CV-P4-10 (steam control valve) to operate automatically.

L9-1.4 Allow pressurizer sprays and test PORV to cycle automatically. Deenergize pressurizer heaters prior to holding test PORV open.

L3-3.1 Hold open test PORV at a hot leg temperature of $\sim$636°F such that the resulting depressurization to saturation ($\sim$ 2000 psig) will result in a PCS pressure greater than the ECC initiation setpoint (1896 psig). Then trip PCPs and allow them to coast down on their flywheels. PCP injection will not be initiated during the coastdown. Valve out ECC accumulators upon holding open the test PORV.

L3-3.2 Close the test PORV and initiate steam generator refill to avert core uncovery above ECC accumulator pressure and allow the PCS to depressurize to approximately 300 psig. Then reopen the test PORV.

L8-1A.1 Observe slow core uncovery as evidenced by 30°F superheat on any two core cladding thermocouples.

L8-1A.2 Allow core inventory to deplete until mixture level recedes to approximately 20 inches above core bottom as evidenced by 30°F superheat on any two core cladding thermocouples within 20 $\pm$ 4 inches or until the highest reading core cladding thermocouple on either PLSS or LECS indicates 1000°F.

L8-1A.3 Initiate PCP injection, close the test PORV and restart PCPs within 10 seconds of satisfying the requirements of L8-1A.2 above. Continue PCP speed increase until core cooling is indicated by decreasing cladding temperatures or until a PCP limit is attained.

L8-1A.4 When core thermal trends have steadied initiate plant recovery in accordance with ESA.

7. DISCUSSION

Test L9-1 is designed primarily to address concerns stemming from LOFW-LOCA analyses on LPWRs with minimum PORV capacity. Such analyses indicate that in LOFA induced loss of secondary heat removal capability, operator action is required to hold open PORVs within a specified maximum time after SG dryout in order to avert a slow core uncovery accident. The distinguishing characteristic of this scenario is the magnitude of the resulting PCS inventory depletion without subsequent depressurization to ECC injection. Test L3-3 is initiated by a typical recovery scenario from a postulated LOFW-LOCA. The test PORV will be held open allowing the PCS to depressurize to saturation. The timing will be such that the resulting PCS pressure will be above ECC initiation setpoint. The various considerations leading to the development of the L9-1/L3-3 scenario together with special operating conditions and scaling comparisons are discussed in subsequent paragraphs as outlined below:

1. Break Type, Size, and Location

2. Delayed Scram Criteria

3. PORV Setpoints

4. Surge Line Configuration

5. LOFT/LPWR Scaling Comparison

6. LOFT/LPWR Heat Sinks/Sources Comparison.

7.1 Break Type, Size and Location

The Test L9-1 simulates a PORV induced LOCA. A 104.95 lbm/hr/MW scaled PORV, geometrically similar to LPWR PORVs, will be installed downstream of an instrumented spoolpiece. This valve size corresponds to an equivalent LPWR PORV flow area of 0.0182 $ft.^2$

7.2 Delayed Scram Criteria

At test initiation the MFP will be tripped and steam generator level will be steamed down to reduce steam generator level to a minimum before scram. The plant will scram on a high hot leg pressure or high hot leg temperature trip due to loss of steam generator secondary side heat transfer area.

The corresponding LPWR transient would trip on a low steam generator water level signal. However, due to LOFT's lower than scaled power to volume ratio equalling approximately two-thirds LPWR ratios, and LOFT's lower than equilibrium decay heat level, it becomes necessary to deplete steam generator inventory without reducing PCS enthalpy in order to preserve the basic pressurizer insurge characteristic of this transient. Although other means of depleting steam generator inventory were investigated, none were feasible. Therefore, the steam generator will be steamed down with the reactor at power.

7.3 PORV Setpoints

The LOFT test PORV setpoints will be set to agree with LPWR PORV setpoints; namely, lift at 2335 psig, reseat at 2315 psig. These setpoints will allow easier correlation of break flow data to LPWR conditions. The LOFT plant PORV (CV-P139-5-4) setpoints remain unchanged; namely, lift at 2410 psig, reseat at 2390 psig.

Since the corresponding LPWR transient would initially trip on steam generator low level, the PORV would not actuate until steam generator dryout subsequent to reactor shutdown. The intent of not allowing L9-1 to trip on steam generator level is to advance the onset of steam generator dryout with the reactor at power as discussed in Section 7.2. Therefore, the relationship of the PORV setpoint to the high pressure scram setpoint is not as critical as the typicality of the test PORV setpoint pressures. Hence, the LOFT test PORV setpoint will not be reset below the high pressure trip setpoint for Test L9-1.

7.4 Surge Line Configuration

The LOFT pressurizer surge line attaches vertically at the hot leg centerline while that in the corresponding Westinghouse (W) LPWR attaches horizontally at pipe midplane. The net effect of this atypicality is that phase separation effects at the surge line will occur earlier in LOFT than in W LPWRs. The LOFT transient, therefore, would contribute a higher quality fluid to the surge line than would the LPWR during corresponding periods after the hot leg begins to void.

7.5 LOFT/LPWR Scaling Comparison

The scaling criteria for the various LOFT systems for Test L9-1 are as follows:

7.5.1 Break Size

The break size for L9-1 is not inherently scaled to LOFT geometry. Rather, a relief capacity of 104.95 lbm/hr/MW was used corresponding to minimum relief capacity in a generic Westinghouse (W) LPWR design. Data from WCAP 9744[1] indicates a W minimum PORV capacity of 358,000 lbm/hr at 2335 psig and an equivalent flow area of 0.0182 ft.[2]

Since,

$\dot{m}_c = G_c A$, and $G_c = G(x,P)$, where

$\dot{m}_c$ = critical mass flow rate

G_c = critical mass flux

A = flow area

x = quality

P = pressure,

and assuming identical upstream conditions and a test PORV relief setting of 2335 psig, then

$$G_c = \frac{\dot{m}}{A} = 1.967 \times 10^7 \text{ lbm/hr/ft}^2.$$

Therefore, applying the scaling factor of 104.95 lbm/hr/MW,

$$A_{scaled} = \frac{(104.95)\ (50)}{1.967 \times 10^7} = 2.668 \times 10^{-4} \text{ ft}^2.$$

7.5.2 Primary Coolant System Flowrate

The initial PCS flowrate of 3.8×10^6 lbm/hr is higher than the scaled LPWR flowrate to conform with safety analysis requirements. Preliminary analysis has shown that this difference does not significantly influence overall system behavior following break initiation.

7.5.3 Initial Power Level

The initial condition of maximum design power (50 MW) was chosen for Test L9-1 in order to maximize decay heat. LOFT's decay heat will be lower than scaled LPWR decay heat for two reasons. First, LOFT irradiation time is substantially less than representative LPWR irradiation times; therefore, fission product concentrations will be lower. Second, LOFT's power-to-volume ratio is low compared to an LPWR. Volume scaling of the power level would result in a required LOFT power of 73.44 MW. For a given enrichment, a lower than scaled core thermal power yields decreased core thermal neutron fluxes resulting in reduced burnups and lower fission product concentrations. The only way to minimize these atypicalities was to attain the highest allowable power prior to test initiation.

In addition, LOFT's environmental heat losses are proportionately higher than the corresponding LPWR scaled heat losses thereby requiring maximum decay heat. Nevertheless, LOFT environmental losses will be greater than fifty percent of available decay heat after 3600 seconds.

7.5.4 Pressurizer Level

The LOFT normal pressurizer level is set to establish a liquid volume which results in a ratio of pressurizer fluid enthalpy-to-total PCS fluid enthalpy for LOFT which is equivalent to the pressurizer-to-total PCS enthalpy ratio for an LPWR. In order to accurately account for pressurizer insurge response, pressurizer insurge volume should be power-to-volume scaled. This would require an initial vapor volume of 13.5 ft^3. The normal LOFT pressurizer level operating band allows a vapor volume of 11.5 ± 3.5 ft^3. Therefore, initial pressurizer level low in the operating band is sufficient.

7.5.5 Pressurizer Spray

The LOFT pressurizer volumetric spray flow rate of 20 gpm results in a volumeric spray flow rate-to-pressurizer vapor volume ratio which is fifty percent greater than LPWR ratios. LOFT's larger than scaled spray flow increases spray effectiveness thus reducing dependence on the test PORV. This effect is somewhat offset by LOFT's higher spray temperature.

7.5.6 Inlet Temperature

LPWR audit calculations[3,4,5,6] used an inlet temperature of 530°F and a ΔT of 59°F. Since this ΔT cannot be matched in LOFT at the required high flowrate, the inlet temperature was chosen to give a T_{AVE} which matched the audit calculations.

LOFT $\Delta T \cong 34°F$ (at 3.8×10^6 lbm/hr), and in the Audit Calculations, $T_{AVE} = 530 + \frac{59}{2}$ or 559.5°F.

Therefore, $T_{INLET\ LOFT} = 559.5 - \frac{34}{2} = 542.5°F$.

7.5.7 Control Rod Position

The control rod position of 54 inches withdrawn provides peaking factors representative of typical LPWRs and is consistent with rod heights of previous tests.

7.5.8 Secondary Coolant System

Secondary heat transfer will be minimized during Test L9-1. The following considerations affect the typicality of steam generator heat transfer effects during SG refill in L3-3 and necessitate delayed scram criteria for L9-1.

1. The LOFT steam generator tubes are too short to be prototypical, but the tube heat transfer area has been volume scaled to an LPWR.

2. The LOFT secondary coolant volume is approximately 30% larger than the scaled LPWR volume.

3. One LOFT steam generator represents four LPWR steam generators.

LOFT's larger than scaled secondary coolant volume and smaller than scaled power-to-volume ratio necessitate depleting SG inventory as much as possible before scram to conserve PCS energy.

7.6 Comparison of LOFT/LPWR Relative Heat Sources/Sinks

There is a potential that the size of the LOFT facility and the scaling of the LOFT components in relation to an LPWR will result in nonprototypical results during an anticipated transient/small break loss-of-coolant accident test due to the effect of the various heat sources or sinks in LOFT and their relative magnitudes. A complete study of these effects has not been performed; however, selected heat sources and sinks and their relative effect on results are discussed in this section.

7.6.1 Decay Heat

LOFT's lower than scaled decay heat and higher than scaled structural surface area-to-fluid volume ratio combine to produce a transient which is less severe thermally and places less demand on alternate means of energy removal; i.e., less dependence on the test PORV.

7.6.2 Energy Loss From Break

The L9-1 test PORV flow area was scaled to be representative of a particular LPWR PORV capacity-to-core thermal power design ratio. Since LOFT's break flow-to-decay heat ratio will be higher than scaled due to

LOFT's lower than scaled decay heat ratio and larger volume-to-power ratio, a smaller fraction of PCS energy will be removed per unit mass of break flow than in an LPWR.

7.6.3 Heat Loss or Gain From the LOFT Piping and Structurals

LOFT has a higher relative structural and piping heat capacity than an LPWR. This higher capacity will mean that the LOFT system heat source to the primary from piping and structurals will be greater in the long term transients than for an LPWR. This will have the effect of compensating to some degree for the lower LOFT decay heat and higher environmental losses.

7.6.4 Heat Source From Primary Coolant Pumps

The LOFT primary coolant pumps will be running during the test L9-1 blowdown. Heat will be imparted to the primary coolant during this phase of the experiment, but it will be negligible when compared to the core decay heat. Initially, the power to the primary coolant from the pumps will be about 75 kW compared to an initial decay heat of about 4000 kW. Primary coolant pumps will not be running during Test L3-3.

REFERENCES

1. W. Tauche, Loss of Feedwater Induced Loss-of-Coolant Accident Analysis Report, WCAP-9744, Westinghouse Electric Corporation, May 1980.

2. H. J. Welland, "L3 Series Test Requirements," LDR 111-7000, EG&G Idaho, Inc., December 1979.

3. C. A. Dobbe, et al., Audit Calculations for Westinghouse PWR Small Cold Leg Breaks, EGG-CAAP-TR-054 (August 1979).

4. C. D. Fletcher, Additional Audit Calculations for Westinghouse PWR Small Breaks, EGG-CAAP-5052 (November 1979).

5. C. B. Davis, Audit Calculations of Small Cold Leg Breaks in a Combustion Engineering Pressurized Water Reactor, EGG-CAAP-TR-055 (September 1979).

6. C. A. Dobbe, Babcock and Wilcox Audit Calculations - Small Break, EGG-CAAP-5061 (November 1979).

7. B. W. Sheron, Generic Assessment of Delayed Reactor Coolant PUmp Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors, NUREG-0623, November 1979.

8. Task II.D, Reactor Coolant System Relief and Safety Valves, NRC Action Plan Developed As A Result of the TMI-2 Accident, NUREG-0660, May 1980.

9. Rowsome letter to L. S. Tong, Accident Sequences for Experimental Investigation, October 1979.

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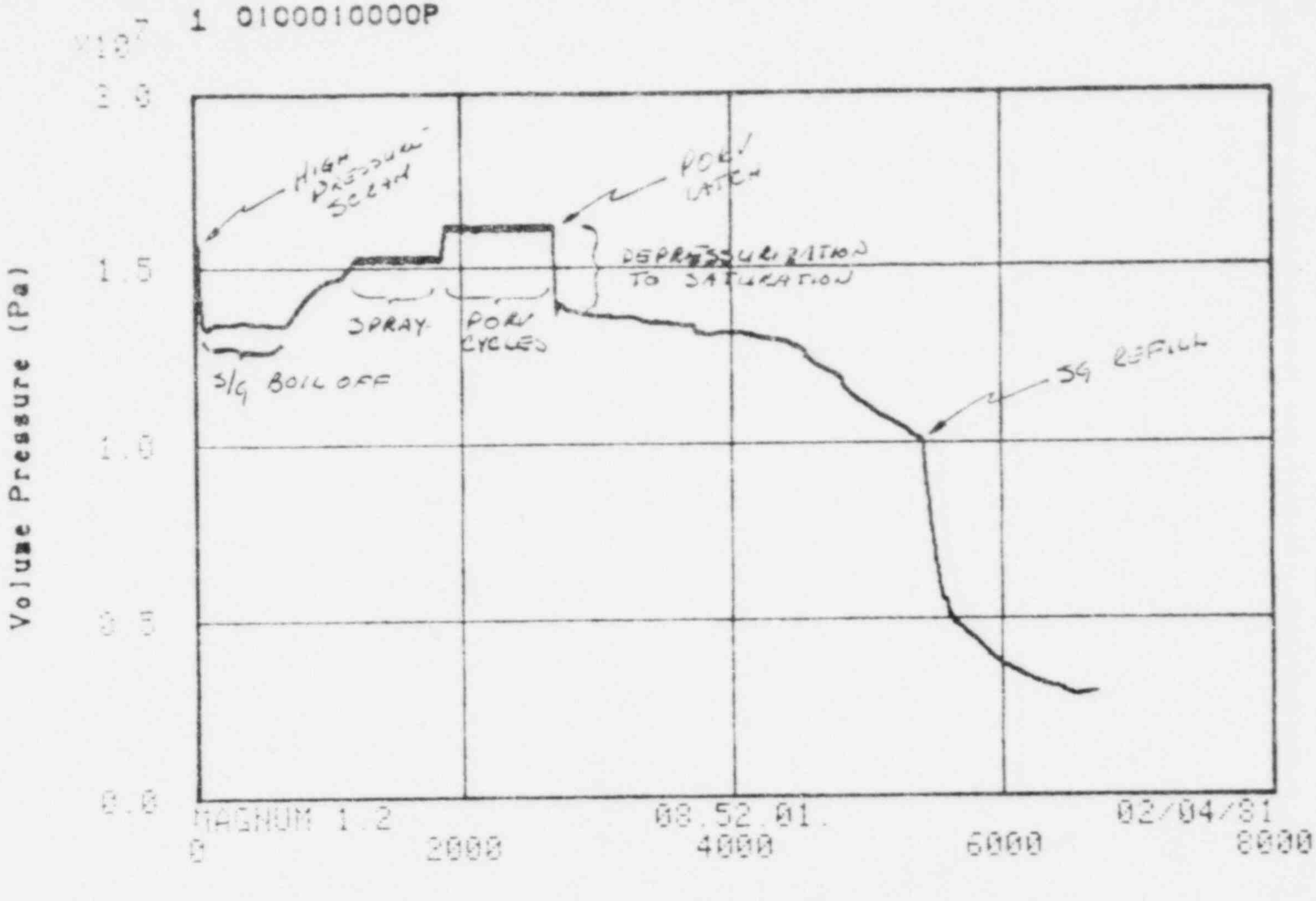

R5X180 LS-1 PREDICTION
PCS PRESSURE
PUMP TRIP AT 810 KELVIN
FIGURE 1

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EDD

TEST PORV INSTRUMENTATION/CONFIGURATION

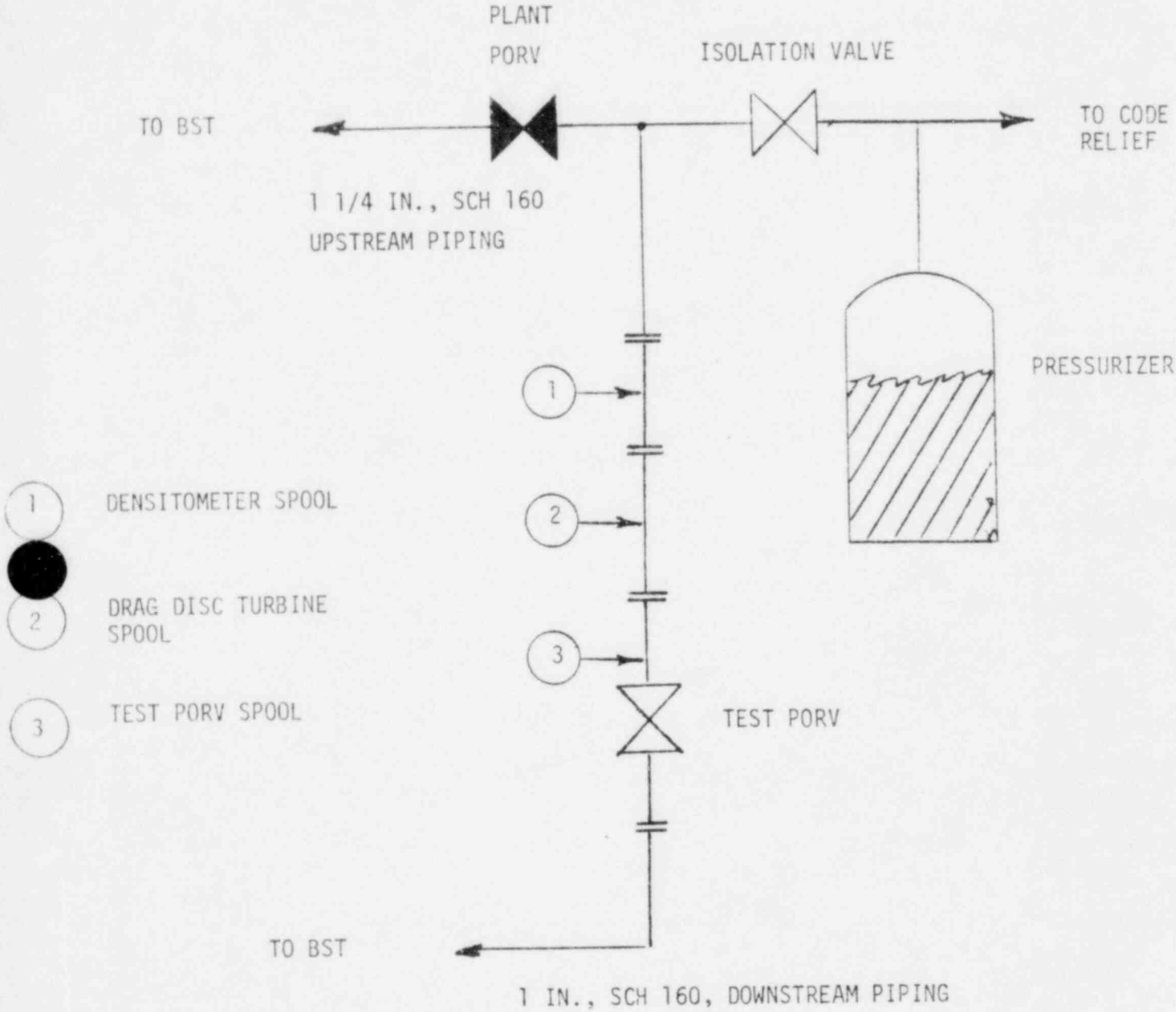

FIGURE 2