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**ANALYSIS OF STANDARD PROBLEM SIX
(SEMISCALE TEST S-02-6)
DATA**

CHARLES E. CARTMILL

MASTER

August 1977

EG&G Idaho, Inc.

IDAHO NATIONAL ENGINEERING LABORATORY

ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

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Approved:

D.J. Olson, Manager
Semiscale Program

J. O. Zane, Deputy Director
Water Reactor Research

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by

Charles E. Cartmill

EG&G IDAHO, INC.

August 1977

PREPARED FOR THE
U.S. NUCLEAR REGULATORY COMMISSION
AND
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION
IDAHO OPERATIONS OFFICE
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ABSTRACT

Test S-02-6 of the Semiscale Mod-1 blowdown heat transfer test series was conducted to supply data for the U.S. Nuclear Regulatory Commission Standard Problem Six. To determine the credibility of the data and thus establish the validity of Standard Problem Six, an analysis of the results of Test S-02-6 was performed and is presented. This analysis consisted of investigations of system hydraulic and core thermal data. The credibility of the system hydraulic data was investigated through comparisons of the data with data and calculations from related sources (Test S-02-4) and, when necessary, through assessment of physical events. The credibility of the core thermal data was based on a thorough analysis of physical events. The results of these investigations substantiate the validity of Test S-02-6 data.

SUMMARY

The Semiscale Mod-1 Program was designed to investigate the behavior of a pressurized water reactor system (PWR) model during an hypothesized loss-of-coolant accident (LOCA). The Mod-1 system is a scaled down model of a PWR that simulates the principal physical features of a nuclear plant, and is designed such that all facets of the accident may be investigated. Nuclear heating is simulated in the tests by a core composed of an array of electrically heated rods. The core is contained in a pressure vessel which also includes a downcomer, lower plenum, and upper plenum. The system piping is arranged in a 1-1/2-loop configuration with the broken loop containing passive simulators for the steam generator and pump.

Test S-02-6 of the Semiscale Mod-1 blowdown heat transfer test series was conducted to supply thermal-hydraulic data concerning the fluid phenomena that occur as a result of a small break. Although, due to the not totally representative single-ended break geometry, the phenomena occurring in the broken loop are not considered typical of those that might occur in a PWR, evaluation of the thermal-hydraulic response of the Semiscale system can be valuable in characterizing of the most significant thermal-hydraulic phenomena likely to occur in the primary coolant system of a PWR during the slow depressurization (blowdown) and emergency core cooling (ECC) phases of a small break LOCA. The data are to be used for the U.S. Nuclear Regulatory Commission (NRC) Standard Problem Six. The NRC standard problem tests are tests conducted to establish an experimental data base for the purpose of comparison with unbiased analytical calculations supplied by the standard problem program participants. The standard problems provide a means of evaluating the adequacy of system models used to predict the overall thermal-hydraulic response of a particular system during a blowdown transient. The results of Semiscale Test S-02-6 were withheld from the standard problem program participants until their analytical predictions had been completed.

The Semiscale Mod-1 system was arranged in a cold leg break configuration for Test S-02-6. The procedure followed in the approach to the test was: (a) the system was brought up to pressure (2265 psig), desired flow rate, and temperature of about 542°F; (b) core power was increased in successive steps to about 1.6 MW; (c) core power and core flow rate were adjusted to attain the desired core differential temperature of about 67°F and the system was allowed to equilibrate; and (d) after the desired initial steady state conditions had been attained, the system was subjected to decompression by introducing a leak in the broken loop cold leg. As the test proceeded, emergency core coolant (ECC) was injected into the intact and broken loop cold legs.

Test S-02-6 represented a simulated small (6%) single-ended break which resulted in a much longer subcooled blowdown period (63 sec compared with about 3 sec for a 200% break), and a much slower depressurization (system pressure of 600 psig occurring at 290 sec instead of at 17 sec for a 200% break) of the system. The period of time from the end of subcooled flow (63 sec) to the time of emergency core coolant injection (290 sec)

was uneventful because all significant dynamic effects were completed by the time the subcooled blowdown was complete and because the core thermal response was such that the temperature of all of the rods was below 600°F at the beginning of saturated blowdown.

A pump suction seal formed during blowdown and the resulting phenomena relating to the pump seal blowout that occurred in the pump suction leg piping were as expected, that is, the seal remained until the core liquid level dropped below the lower elevation of the pump suction leg at which time the seal blew out.

Core temperatures above the initial operating temperature did not occur during Test S-02-6, and emergency core coolant injection was sufficient to cool the core and keep it covered during the latter portion of the test. In the Semiscale system the small break did not result in a rod thermal behavior as severe as observed in the large break tests and thus the small break experiment represents a less severe loss-of-coolant experiment.

CONTENTS

ACKNOWLEDGMENTS	ii
ABSTRACT	iii
SUMMARY	iv
I. INTRODUCTION	1
II. EXPERIMENT DESCRIPTION	3
1. SYSTEM HARDWARE DESCRIPTION	3
2. POWER AND POWER CONTROL	4
3. TEST CONDITIONS AND PROCEDURES	9
4. INSTRUMENTATION	11
III. OVERALL THERMAL-HYDRAULIC BEHAVIOR DURING BLOWDOWN	13
1. HYDRAULIC PHENOMENA OCCURRING AT THE BREAK	13
2. FLUID BEHAVIOR GOVERNED BY PUMP SUCTION PHENOMENA	16
3. CORE THERMAL-HYDRAULIC PHENOMENA	24
4. HYDRAULIC PHENOMENA OCCURRING IN THE VESSEL FOLLOWING INITIATION OF ECC	27
IV. CONCLUSIONS	31
V. REFERENCES	32
APPENDIX A – STEAM GENERATOR RESPONSE	33
APPENDIX B – PIPING HEAT TRANSFER	39

FIGURES

1.	Semiscale configuration for Test S-02-6	4
2.	Semiscale Mod-1 electric heater rod	5
3.	Cross section of vessel with core	6
4.	Vessel cross section and core layout	7
5.	Semiscale Mod-1 axial power profile	8
6.	Core transient power – Test S-02-6	9
7.	Semiscale Mod-1 heated core configuration showing locations of heater rod thermocouples	11
8.	Comparison of mass flow rates for small break Test S-02-6 and 200% break Test S-02-4	14
9.	Comparison of pressures at the break for small break Test S-02-6 and 200% break Test S-02-4	15
10.	Mass flow rates at the core inlet and at intact and broken loop cold legs – S-02-6	15
11.	Fluid temperatures at the break versus saturation temperature – Test S-02-6	16
12.	Illustration of pump suction seal – Test S-02-6	17
13.	Fluid densities at the intact loop hot leg – Test S-02-6	18
14.	Fluid temperatures on each side of the steam generator – Test S-02-6	19
15.	Mass flow rates at the core barrel inlet and intact loop cold leg – Test S-02-6	19
16.	Collapsed liquid levels in the downcomer and core from differential pressure measurements – Test S-02-6	20
17.	Illustration of system hydraulic condition at 40 sec – Test S-02-6	21

18.	Illustration of system hydraulic condition at 50 sec – Test S-02-6	22
19.	Illustration of system hydraulic condition at 60 sec – Test S-02-6	23
20.	Illustration of system hydraulic condition at 100 sec – Test S-02-6	24
21.	Core temperature response (Rod F6) – Test S-02-6	25
22.	Semiscale Mod-1 heated core configuration showing radial locations where rods experience different temperature response	26
23.	Fluid density in system piping – Test S-02-6	27
24.	Intact loop cold leg fluid temperatures – Test S-02-6	28
25.	Upper plenum system pressure – Test S-02-6	29
26.	Intact loop cold leg volumetric flow rate – Test S-02-6	29
27.	Intact loop hot leg and pump inlet volumetric flow rates – Test S-02-6	30
A-1.	Intact loop steam generator configuration	36
A-2.	Steam generator secondary side fluid temperatures – Test S-02-6	37
A-3.	Steam generator secondary side metal temperatures – Test S-02-6	37
A-4.	Steam generator primary and secondary side fluid temperatures – Test S-02-6	38
B-1.	Intact and broken loop cold leg surface heat fluxes – Test S-02-6	42
B-2.	Vertical and horizontal fluid densities in instrument Spool 15 – Test S-02-6	42

TABLE

I.	Description of Blowdown Heat Transfer Test S-02-6, U.S. Nuclear Regulatory Commission Standard Problem Six	10
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ANALYSIS OF STANDARD PROBLEM SIX

(SEMISCALE TEST S-02-6) DATA

I. INTRODUCTION

The Semiscale Mod-1 experimental program conducted by EG&G Idaho, Inc. is part of the overall U.S. Nuclear Regulatory Commission (NRC) and Energy Research and Development Administration-sponsored research and development program to investigate the behavior of the pressurized water reactor (PWR) system during an hypothesized loss-of-coolant accident (LOCA). The Semiscale Mod-1 program is intended to provide transient thermal-hydraulic data from a simulated LOCA using a small scale experimental nonnuclear system. The Semiscale Mod-1 program is a major contributor of experimental data that provide a means of evaluating the adequacy of the overall system analytical models as well as the models of the individual system components. Selected experimental data produced by this program will also be used to aid other Energy Research and Development Administration-Nuclear Regulatory Commission-sponsored experimental programs, such as the Loss-of-Fluid Test (LOFT) program^[1], in optimizing test series, selecting test parameters, and evaluating test results.

The Semiscale Mod-1 tests are performed with an experimental system which simulates the principal physical features of a nuclear plant but which is smaller in volume. Nuclear heating is simulated in the tests by a core composed of an array of electrically heated rods. The core is contained in a pressure vessel which also includes a downcomer, lower plenum, and upper plenum. The Semiscale system piping is arranged such that the intact loop represents three loops of a four-loop nuclear plant, and the broken loop represents the fourth loop. In the present configuration the intact loop contains an active steam generator and pump, and the broken loop contains passive simulators for the steam generator and pump.

Test S-02-6 was a small (6%) single-ended break test which was conducted as part of the blowdown heat transfer test series (Test Series 2 of the Semiscale Mod-1 Program). The purpose of Test S-02-6 was to supply thermal-hydraulic information concerning the fluid phenomena that occur as a result of a small break. Although, due to the not totally representative single-ended break geometry, the phenomena occurring in the broken loop are not considered typical of those that might occur in a PWR, evaluation of the thermal-hydraulic response of the Semiscale system can be valuable in characterizing of the most significant thermal-hydraulic phenomena likely to occur in the primary coolant system of a PWR during the slow depressurization (blowdown) and emergency core cooling (ECC) phases of a small break LOCA. Test S-02-6 served as the Nuclear Regulatory Commission

Standard Problem Number Six^[a]. The specific objectives of this small break test were to determine (a) the general system thermal-hydraulic behavior during the slow depressurization that occurs as a result of the small break, (b) the general core temperature response, and whether the immersed core becomes uncovered during a small break, and (c) the effectiveness of the emergency core cooling (ECC) systems for a small break condition.

Analyses of the data from other tests performed in the blowdown heat transfer series are presented in References 2 through 6. This report presents the results of analysis performed to evaluate the observed core and loop piping thermal-hydraulic response of the Semiscale Mod-1 system. Discussions concerning the general system thermal-hydraulic behavior occurring during blowdown are presented. Specific fluid phenomena occurring at the break, within the pump suction leg, and in the core region, are also presented. Core and loop response calculations that were obtained through computer code models are used to explain experimental data. Section II of this report contains a description of the Semiscale Mod-1 system and the test plan followed for the small break Test S-02-6. The results of the analysis of the blowdown data are presented in Section III. Section III contains a discussion of the overall thermal-hydraulic behavior during blowdown, with emphasis placed on the break phenomena. The significant conclusions concerning the results from the small break test are presented in Section IV.

[a] The objective of the standard problem program is to provide a mechanism to systematically increase the participants' understanding and knowledge of the computational techniques used to analyze the performance of nuclear power reactors. Facilities such as Semiscale provide the experimental data used in assessing and verifying the computational techniques.

II. EXPERIMENT DESCRIPTION

The Semiscale Mod-1 experimental system and test plan for Test S-02-6 of the blowdown heat transfer test series are described. The topics discussed include the overall system hardware, power and power control, test conditions and procedures, and instrumentation.

1. SYSTEM HARDWARE DESCRIPTION

The Semiscale Mod-1 experimental apparatus is a small scale model of the Loss-of-Fluid Test (LOFT) system^[1] which is in turn a 55 MW(t) test reactor model of a four-loop pressurized water reactor. The Semiscale system, which has a liquid volume of approximately 7.8 ft³, consists of a pressure vessel and internals; an intact loop consisting of a pressurizer, steam generator, and coolant pump; and a broken loop with hydraulic resistances that simulate an inactive steam generator and a locked rotor pump.

The Semiscale Mod-1 system was modified from its normal configuration in order to provide a test which would represent a 6% break in a four-loop PWR for the small cold leg break and is shown in its modified configuration in Figure 1. These modifications included the addition of a longer intact loop pump suction leg and the deletion of the broken loop hot leg; the vessel was capped off at the hot leg outlet. The elevation of the low point of the intact loop pump suction leg (normally 41 in. below the cold leg centerline) was lowered to about 100 in. below the cold leg centerline. This lower elevation allows the percentage of core length between the top of the core and the bottom of the pump suction elevation to be essentially the same in the Semiscale system as it is in a PWR. The pump suction leg was lowered so that the percent of the core uncovered in Semiscale would more closely represent that percentage of the core uncovered in a PWR during a postulated small break accident. The broken loop hot leg was removed to better represent a single-ended break, and to eliminate the additional complexity of data analysis that would have resulted had the broken loop hot leg been left in the system. A complete description of the unmodified Semiscale system is given in Reference 7.

Emergency core coolant was injected during the small break experiment by three separate ECC systems: high pressure injection system (HPIS), low pressure injection system (LPIS), and gas pressurized water accumulator system.

Fig. 1 Semiscale configuration for Test S-02-6.

2. POWER AND POWER CONTROL

The core simulator in the Semiscale system consisted of 40 electrically heated rods of typical PWR fuel rod diameter (0.422 in.) and axial power peaking factor (1.58). Figure 2 illustrates the general construction of a Semiscale heater rod. The overall length of the rod is 207 in., and the bottom of the 5.5-foot heated section is about 143 in. below the vessel cold leg centerline as shown in Figure 3. The rods extend from the bottom of the heated section to the upper plenum and pass out through the vessel upper head.

The 40 rods are positioned and held in the core with 10 grid spacers which maintain the heaters on a typical PWR pitch (0.563 in.). Figure 4 shows a plan view of the Semiscale vessel and core simulator assembly. A heat shield assembly composed of 0.018-in.-thick stainless steel surrounds the rod matrix and reduces the core flow area to 7.39 in.². The four centrally located rods (the high power rods) can be powered at a level which is different

Fig. 2 Semiscale Mod-1 electric heater rod.

Fig. 3 Cross section of vessel with core.

Fig. 4 Vessel cross section and core layout.

than the remaining 36 rods (the low power rods). This method of powering the heater rods allows the simulation of radial peaking similar to that which exists in a nuclear core. Test S-02-6 had a radially peaked power profile; the center four rods had an initial peak power density of 14.25 kW/ft and the surrounding powered rods had a peak power density of 11.54 kW/ft providing a total core power of 1.56 MW. The axial power profile for the as-constructed rods is illustrated in Figure 5.

Figure 6 shows the transient electrical core power used during Test S-02-6. This electrical power was determined by analytically matching the surface heat flux calculated for an electrical rod to the surface heat flux calculated for a nuclear rod, assuming that both rods were subjected to the same transient boundary conditions as described in Reference 2.

Fig. 5 Semiscale Mod-I axial power profile.

Fig. 6 Core transient power - Test S-02-6.

3. TEST CONDITIONS AND PROCEDURES

Table I presents an overview of the test conditions and other pertinent information relating to the blowdown heat transfer Test S-02-6.

The basic procedure followed in the approach to rupture initiation in Test S-02-6 was as follows: (a) the system conditions were established at a pressure of 2265 psig, flow rate of 148 gpm, and a temperature of 542°F; (b) the core power was increased to the desired value in successive steps; (c) the core power and flow rate were adjusted to achieve a core differential temperature of 67°F and the system was allowed to equilibrate. After the initial steady state conditions had been achieved, blowdown was initiated by introducing a leak in the system piping by means of a rupture assembly.

TABLE I

DESCRIPTION OF BLOWDOWN HEAT TRANSFER TEST S-02-6,
U.S. NUCLEAR REGULATORY COMMISSION STANDARD PROBLEM SIX

Variable	Test S-02-6
Break size (%)	6%
Break location	Cold leg
System pressure (psig)	2265
Core flow (gpm) ^[a]	148
Core power (MW) ^[a]	1.56
Core inlet temperature (°F)	542
Core ΔT (°F) ^[a]	67
Heated rods in core	37 ^[h.c]
Peak power density on high power rods (kW/ft)	14.25
Peak power density on low power rods (kW/ft)	11.54
ECC injection	
Intact loop	
Accumulator	
Location	Cold leg
Injection pressure (psig)	600
Low pressure injection system	
Location	Cold leg
Injection pressure (psig)	150
Broken loop	
Accumulator	
Location	Cold leg
Injection pressure (psig)	600
Low pressure injection system	
Location	Cold leg
Injection pressure (psig)	150

[a] Rated conditions for Semiscale are: core flow - 165 gpm; core power 1.6 MW; core Δt - 66°F.

[b] Liquid level probe replaced one rod.

[c] Two rods unpowered.

4. INSTRUMENTATION

A description and listing of the instrumentation used in the Semiscale system for Test S-02-6 is included in Reference 8. Of particular concern to standard problem participants are the instrumentation measurements taken within the core region, specifically with respect to the heater rods. Figure 7 illustrates the heater rod matrix and cladding thermocouple locations for the core rod temperature measurements. The heater rods are identified within the matrix by reference to the row of letters across the top and the column of numbers

Fig. 7 Semiscale Mod-1 heated core configuration showing locations of heater rod thermocouples.

down the side of the matrix. Similarly, the thermocouples are identified by the rod that they are on and by their elevation above the bottom of the heated length of the core. The thermocouple on Rod D5 at the 29-in. elevation is thus referred to as TH-D5-29 where TH refers to a core heater cladding temperature, D5 refers to the rod upon which the thermocouple is located, and 29 gives the thermocouple elevation in inches above the core bottom. (The arrows in Figure 7 indicate thermocouple azimuthal locations.)

Core measurements taken in addition to the heater rod cladding temperatures included core voltage, power, current, fluid temperature, inlet flow, and inlet density. Electrical current was measured using a shunt circuit technique, and voltage was obtained from a digital voltmeter. The voltage and current measurements were processed through a Hall-effect multiplier to obtain the total core power. Core inlet flow rate and fluid density measurements were provided by instrumentation consisting of a turbine flowmeter, drag disc flowmeter, and gamma densitometer located below the bottom of the heater rods. Core fluid temperatures were measured with thermocouples attached to the grid spacers located along the heated length of the rods.

Measurement of test variables other than within the vessel was accomplished by detectors located within the intact and broken loop piping. Detectors were provided for measuring fluid and material temperatures, pressure, differential pressure, volumetric flow rate, velocity, momentum flux, fluid density, and torque, power, and speed of the pump. Five flow measuring devices were used in the broken loop cold leg for measuring the broken loop break flow rate.

III. OVERALL THERMAL-HYDRAULIC BEHAVIOR DURING BLOWDOWN

The results of the data analysis are presented in this section. Analysis related to phenomena which most strongly influence the overall thermal-hydraulic behavior of the system are discussed. Included within the discussion is a description of the general sequence of events occurring within the system as a result of the slow depressurization environment. This section also treats specific thermal-hydraulic phenomena occurring at the break, within the intact loop pump suction leg, and within the core region. Also included is a discussion of the hydraulic phenomena occurring within the vessel following initiation of ECC relative to the ECC effectiveness for the small break test. The influence of the steam generator and piping heat transfer on fluid behavior during Test S-02-6 is mentioned within this section, but not discussed. A detailed discussion of these topics is included within the appendixes.

In order to put the small break (6%) test in proper perspective with respect to severity of loss-of-coolant accident experiments, the results of Test S-02-6 have been compared in selected cases with results of Test S-02-4^[9] which is a 200% cold leg break test in the blowdown heat transfer test series. Test S-02-4 had essentially the same initial test conditions and ECC system requirements as Test S-02-6, thus providing a basis for comparison of results with those results from the small break test.

The blowdown process for the Semiscale single-ended small break (6%) was strongly influenced by the break flow rates, the pump behavior, and the pump inlet fluid phenomena caused by the depth of the pump suction leg below the cold leg centerline. The break flow rates were relatively small during the subcooled and saturated portions of Test S-02-6 which resulted in a slow depressurization of the system. The slow depressurization caused the fluid within the vessel and within the intact and broken loop cold legs to remain subcooled for 63 sec after break initiation. The pump, which was coasting down during this extended period of subcooled depressurization, delivered high density fluid to both the core and cold leg break which resulted in a positive flow through the core and relatively good core cooling. The pump coasted down to a stop before saturated flow occurred at the pump inlet. Once the pump stopped, a subcooled liquid plug formed in the pump suction leg. This plug remained until the core liquid level dropped below the lower elevation of the pump suction leg at which time the water plug was blown out. The core partially uncovered during this phase of the test, although a sufficient amount of fluid remained in the core to maintain the rods at temperatures well below the initial temperature values. The accumulator ECC injection was initiated at 600 psig and sufficient water was delivered to fill the core and keep it covered for the remainder of the test.

1. HYDRAULIC PHENOMENA OCCURRING AT THE BREAK

The hydraulic phenomena, occurring at the simulated break, control the system depressurization rate and directly influence much of the hydraulic behavior occurring throughout the system, specifically within the core region. The small break area (6%)

employed in Test S-02-6 caused the initial break flow rate to be relatively low in comparison with results of other tests (200% breaks) in the blowdown heat transfer series. Figure 8 compares the break flow rate for the small break with that for a 200% break test. The low

Fig. 8 Comparison of mass flow rates for small break Test S-02-6 and 200% break Test S-02-4.

(2.5 lbm/sec), almost constant, break flow rate for Test S-02-6 results in a much slower depressurization of the system as shown in Figure 9 which caused the subcooled period of flow to be longer (63 sec compared with about 3 sec for a 200% break test). This longer subcooled period of break flow combined with the small break flow rates significantly affected the hydraulic response of the rest of the system.

The fluid hydraulics occurring within the Semiscale intact loop during the first 40 sec of the small break test evidenced that sufficient subcooled fluid had been supplied to meet the demand of the break. When the intact loop pump coasted down to the point where the flow in the intact loop cold leg could not supply the total demand of the break, the flow direction in the core reversed to make up the difference between what was demanded at the break and what was being supplied from the intact loop. The fact that the break was being supplied by fluid from both the intact loop and core regions is illustrated in Figure 10 which indicates that the flow rates in the intact loop cold leg and within the core are essentially the same value as that in the broken loop immediately prior to 40 sec, but become lower than the break flow rate (an amount equivalent to that coming from the core) subsequent to 40 sec. The size of the break, resulting in the slow depressurization rate, affected the time at which the transition between subcooled and saturated fluid occurred at the break. Figure 11 shows the fluid temperature at the break versus the saturation temperature corresponding to

Fig. 9 Comparison of pressures at the break for small break Test S-02-6 and 200% break Test S-02-4.

Fig. 10 Mass flow rates at the core inlet and at intact and broken loop cold legs - Test S-02-6.

Fig. 11 Fluid temperatures at the break versus saturation temperature – Test S-02-6.

the measured pressure. Once saturation conditions existed at the break, the time period between 63 sec and the time at which ECC fluid entered the system (290 sec) was essentially uneventful. That is, all significant dynamic effects of the blowdown were over by 63 sec, and as a result of sustained positive flow through the core, the core thermal response was such that the temperature of all of the rods was below 600°F by this time.

2. FLUID BEHAVIOR GOVERNED BY PUMP SUCTION PHENOMENA

Making the pump suction leg longer to better represent PWR geometry for Test S-02-6 resulted in an intact loop and core fluid behavior within the Semiscale system which was typical of that expected in a PWR under small break conditions. The additional length of the pump suction leg resulted in the proper establishment of a water seal which formed in the pump suction pipe during and following the pump coastdown. As illustrated in Figure 12, water in the pump suction formed a seal which blocked the flow of steam between the intact loop hot and cold legs during the blowdown period. With essentially no intact loop flow, the collapsed water level in the core decreased at the same rate as the water level in the pump suction leg. As illustrated, the level in the pump suction leg could decrease well below the elevation of the cold leg piping which remained full of water. Once the water

Fig. 12 Illustration of pump suction seal – Test S-02-6.

level reached the bottom of the pump suction pipe, the water seal would be expected to blow out and the pressure would equilibrate which would cause the collapsed water levels in the core and downcomer to equilibrate to the same height.

The effect of a water seal forming at the pump suction was observed during Test S-02-6. This seal in the intact loop was characterized by a high density fluid at the pump inlet, and low density fluid within the hot leg of the intact loop. The density measured at the intact loop hot leg during the first 40 sec, shown in Figure 13, indicated large steam

Fig. 13 Fluid densities at the intact loop hot leg – Test S-02-6.

content which was caused by the generation of steam both from liquid flashing in the hot leg and from liquid boiling within the heated core. Heat transfer from the intact loop hot leg fluid to the steam generator secondary side, along with the subcooled water within the primary piping, was sufficient to condense steam flowing from the hot leg to the cold leg during the first 40 sec of blowdown. The occurrence of this phenomenon is illustrated by fluid temperatures on both sides of the steam generator (TSU-6 and TFU-8). These temperatures are compared with the saturation temperature in Figure 14. Just after 40 sec, a flow stagnation occurred just upstream of the stopped pump.

As the pump stopped (about 40 sec after rupture) the flow reversed direction at the core barrel inlet as shown in Figure 15, and the core collapsed liquid level (calculated from a differential pressure measurement across the core) decreased well below the level of the liquid in the downcomer and intact loop cold leg as indicated in Figure 16. Figures 17 through 20 illustrate what occurred to the pump seal between 40 and 100 sec after rupture. The system water levels represented in the figures were obtained from differential pressure measurements and are only representative of the true water levels. The differences in level that exist (where the same levels would be expected) are due to pressure drops between the two points and to the method of obtaining the collapsed liquid levels. Prior to 40 sec the depressurization of the system was sufficiently slow that stratification occurred in the upper plenum, and the upper plenum drained much like a reservoir would that has a leak somewhere lower in its system. A calculation of the upper plenum level using an integration of the break mass flow provided a very good comparison with the level shown in Figure 17.

Fig. 14 Fluid temperatures on each side of the steam generator - Test S-02-6.

Fig. 13 Mass flow rates at the core barrel inlet and intact loop cold leg - Test S-02-6.

As illustrated in Figure 17, at 40 sec after rupture the pump suction leg had started to drain below the cold leg centerline even though the density measurement indicated the cold leg was full. By 50 sec after rupture (Figure 18) the pump suction leg collapsed liquid level was 65 in. below the cold leg centerline, and the core liquid level was well below the top of the heated core. A stagnation condition existed at this time in the pump suction leg just

Fig. 16 Collapsed liquid levels in the downcomer and core from differential pressure measurements – Test S-02-6.

upstream of the pump, caused by the pump having stopped. However, positive flow through the pump was reestablished so that just after 60 sec following rupture the pump seal blew out and a direct hydraulic coupling existed between the intact loop and the core. Figure 19 illustrates the fluid conditions in the pump suction leg just prior to the seal blowing out. By 100 sec after rupture the system was hydraulically balanced as illustrated in Figure 20.

Fig. 17 Illustration of system hydraulic condition at 40 sec - Test S-02-6.

Fig. 18 Illustration of system hydraulic condition at 50 sec - Test S-02-6.

Fig. 19 Illustration of system hydraulic condition at 60 sec – Test S-02-6.

Fig. 20 Illustration of system hydraulic condition at 100 sec - Test S-02-6.

3. CORE THERMAL-HYDRAULIC PHENOMENA

The small break size combined with the single broken loop configuration caused a core hydraulic response that was different than that of previous Mod-1 tests. The core flow remained in the positive direction for the first 38 sec. of blowdown because the intact loop cold leg flow was sufficient to supply the break demand for flow. The positive core flow was

enhanced by the fact that the pump head did not completely degrade until 40 sec after rupture. Once the pump stopped, the intact loop cold leg flow could not completely supply the demand at the break, and a reversed core flow (up the downcomer) resulted, thus complementing the intact loop flow to the break as shown in Figure 15.

The fluid behavior within the core during the early blowdown period provided excellent cooling throughout the blowdown period as illustrated in Figure 21. The only exception to a continual cooling of the heater rods during this test occurred after 40 sec.

Fig. 21 Core temperature response (Rod F6) - Test S-02-6.

About 50 sec after rupture the core collapsed liquid level decreased below the top of the heated core. At this time a sharp, but minor, increase in some rod temperatures was observed. For the one heater rod shown, the bottom of the rod experienced a temperature increase typical of a boiling crisis whereas the top of the rod did not. This type of heater rod response is not indicative of a decreasing liquid level, but suggests that an axial quality gradient existed in the core. At the top of the core where the power generation rate was low, the quality was sufficiently low to maintain good cooling. In the center of the core where the power was high the quality of the mixture led to the observed departure from nucleate boiling (DNB).

A radial variation occurred in the temperature response of the rods which experienced a boiling crisis. As shown in Figure 22, only rods opposite the intact loop hot leg experienced a rapid temperature rise at 50 sec. Since incore density and flow measurements were not possible, the cause of this asymmetric core behavior can only be hypothesized. After the pump stopped, coolant in the intact loop hot leg was observed to flow back into

Fig. 22 Semiscale Mod-1 heated core configuration showing radial locations where rods experience different temperature response.

core. Without strong dynamic mixing, the coolant entering the upper plenum from the hot leg had a preferred path down one side of the core barrel while lighter steam flowed up the other side. This flow pattern would be expected for free convection. Rods on the intact loop hot leg side of the core would be subjected to the influx of a lower quality mixture than those on the opposite side of the core. As expected, only those rods opposite the intact loop hot leg experienced a boiling crisis whereas rods on the other side of the core did not.

At about 60 sec after rupture, water in the cold leg partially passed out the break and also dropped into the downcomer as seen from a decrease in both broken and intact loop densities in Figure 23. With the loss of mass in the broken loop cold leg, the break uncovered and the break flow changed from subcooled to saturated flow. Fluid in the downcomer flowed into the core causing the downcomer mixture level to drop and the core

Fig. 23 Fluid density in system piping – Test S-02-6.

collapsed mixture level to rise (as interpreted from measured differential pressure data in Figure 16). The resulting upward coolant flow through the core (Figure 15) caused those heater rods that experienced a boiling crisis to rewet.

From the time saturated conditions existed at the break until the accumulators injected water, the core collapsed mixture level decreased, but the core was maintained in a continual cooling condition. When the accumulators injected water, the core collapsed mixture level rose well above the heated region of the core.

Core temperatures greater than the initial operating temperatures did not occur during Test S-02-6, and the emergency core coolant was sufficient to cool the core and keep it covered during the latter portion of the test. The conclusion is that in the Semiscale Mod-1 system the small break did not result in a rod thermal behavior as severe as observed in the large break tests and thus the small break results in a less severe loss-of-coolant experiment.

4. HYDRAULIC PHENOMENA OCCURRING IN THE VESSEL FOLLOWING INITIATION OF ECC

The effectiveness of ECC injection systems during blowdown was determined on the basis of the core thermal behavior. From the time the pump suction seal blew out (60 sec)

until the accumulators started water injection (300 sec at a system pressure of 600 psia) the core contained sufficient coolant to remain cool as shown by the heater rod temperature response. When the accumulators began injecting water, some condensation of steam was observed downstream from the injection point as illustrated in Figure 24 by the decrease in fluid temperature below the saturation temperature. The lower fluid temperature increased the amount of energy being removed from the piping walls, but not appreciably, as discussed in Appendix B. The condensation of steam caused by the ECC injection resulted in a small decrease in system pressure as shown in Figure 25. The local decrease in pressure that resulted from steam condensation and the increased steam generation that resulted from a rise in the core mixture level increased the volumetric flow within the intact loop as shown in Figures 26 and 27. The injection rate was low (2 to 3 gpm) but even this low flow rate raised the collapsed liquid levels in the downcomer and core as shown in Figure 16. These processes resulted in excellent core cooling which illustrates the effectiveness of the ECC systems under small break conditions.

Fig. 24 Intact loop cold leg fluid temperatures - Test S-02-6.

Fig. 25 Upper plenum system pressure - Test S-02-6.

Fig. 26 Intact loop cold leg volumetric flow rate - Test S-02-6.

Fig. 27 Intact loop hot leg and pump inlet volumetric flow rates -- Test S-02-6.

IV. CONCLUSIONS

Test S-02-6 was performed to supply data for NRC Standard Problem Six. The analyses and investigations of these data were performed to validate the data for such use. The following is a summary of the conclusions reached from the analyses of the data obtained during the Semiscale Mod-1 Test S-02-6.

The simulated single-ended small break (6%) of Test S-02-6 resulted in a much longer subcooled blowdown period (63 sec compared with about 3 sec for a 200% break), and a much slower depressurization (system pressure of 600 occurring at 290 sec instead of 17 sec for a 200% break) of the system.

The hydraulic process occurring within the Semiscale system during subcooled blowdown (0 to 63 sec) was one of sufficient subcooled fluid being supplied to meet the demand at the break. When, due to pump degradation, the intact loop cold leg could not supply the total demand, the flow direction in the core reversed in order that the difference between what was demanded and that being supplied by the intact loop could be met.

The period of time from the end of subcooled flow (63 sec) to the time of emergency core coolant injection (290 sec) was uneventful because all significant dynamic events were completed by the time the subcooled blowdown was complete and because the core thermal response was such that the temperature of all of the rods was below 600°F at the beginning of saturated blowdown.

A pump suction seal formed during blowdown and the resulting phenomena relating to the pump seal blowout that occurred in the pump suction leg piping were as expected, that is, the seal remained until the core liquid level dropped below the lower elevation of the pump suction leg at which time the seal blew out.

The hydraulic behavior within the core during the blowdown period provided the means for excellent core cooling throughout the blowdown period. Although the pump stopped and a pump suction seal occurred, the resulting severity with respect to peak cladding temperature was minor.

Core temperatures above the initial operating temperatures did not occur during Test S-02-6, and the injected emergency core coolant was sufficient to cool the core and keep it covered during the latter portion of the test. The small break did not result in a rod thermal behavior as severe as observed in the large break tests and, therefore, the small break test represents a less severe loss-of-coolant experiment.

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APPENDIX A

STEAM GENERATOR RESPONSE

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APPENDIX A

STEAM GENERATOR RESPONSE

The effect of the steam generator heat transfer on the system blowdown behavior for the small break is of concern due to the length of time required for a complete blowdown. This section presents the results of an analysis of the data pertaining to the response of the steam generator during blowdown for Test S-02-6.

The steam generator was instrumented with four metal and four secondary fluid thermocouples paired to provide an indication of the heat transfer to the primary fluid during blowdown. The steam generator configuration and instrumentation locations are shown in Figure A-1.

The temperature histories measured by the secondary material and secondary fluid thermocouples provide a mechanism by which to analyze the steam generator thermal performance. The measured secondary fluid temperatures (TFU-SG1, and TFU-SG4) and the secondary material temperatures (TMU-SG1, TMU-SG2, and TMU-SG4) obtained during Test S-02-6 are shown in Figures A-2 and A-3, respectively. Calculations were performed to evaluate the heat transfer from the steam generator during Test S-02-6. The calculation entailed a simple energy balance using the steam generator tube bundle metal and secondary side fluid that is surrounding the tubes as the control volume. The change in the external steam generator structure energy state and external heat losses were neglected. An estimate of the secondary side fluid volume that was capable of transferring a significant amount of heat to the tubes was assumed sufficiently large such that the calculated heat transfer was regarded as a maximum probable value. The results indicate an average energy transfer of 7374 Btu from the secondary to the primary side between 72 and 600 sec after rupture. However, a similar energy balance between the time of rupture and 72 sec indicates the primary side fluid transferred 6800 Btu of energy to the secondary side. These results imply a net energy exchange to the primary side of only 574 Btu which represents about 2% of the energy initially stored in the primary system fluid. The small net energy exchange is a consequence of the length of time (72 sec) that the primary fluid transferred energy to the secondary side due to the slow depressurization of the system. The primary to secondary energy exchange is illustrated in Figure A-4, by the fact that TSU-6 and TFU-8 are higher in temperature than TFU-SG1 and TFU-SG4 until 72 sec. Prior to 72 sec the energy exchange process from the primary side to the secondary side was complicated by the venting of steam by a relief valve in the secondary side of the steam generator. Due to venting, the turnaround point (energy transfer from secondary side to primary side instead of from primary side to secondary side) shown to occur at 40 sec in Figure A-2 did not actually occur until 72 sec, as indicated in Figure A-4. The decrease in temperature of the secondary side fluid between 40 and 72 sec while the primary side was still transferring energy to the secondary side implies that the secondary side was losing energy by venting during this period.

INEL-A-2325

Fig. A-1 Intact loop steam generator configuration.

Fig. A-2 Steam generator secondary side fluid temperatures -- Test S-02-6.

Fig. A-3 Steam generator secondary side metal temperatures -- Test S-02-6.

Fig. A-4 Steam generator primary and secondary side fluid temperatures - Test S-02-6.

Because of simultaneous energy exchange into and out of the secondary side, determination of the steam generator heat transfer is almost impossible, especially with the limited instrumentation used on the secondary side. The energy exchange values quoted earlier are, therefore, gross estimates and should be viewed in that perspective. Nevertheless, the preceding analysis indicates that a relatively small amount of net energy was transferred from the steam generator secondary to the primary fluid for the small break compared with the overall initial amount of internal energy contained in the primary fluid prior to rupture.

APPENDIX B
PIPING HEAT TRANSFER

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APPENDIX B

PIPING HEAT TRANSFER

Piping heat transfer can affect the quality, velocity, and temperature of the primary fluid during the Semiscale Mod-1 blowdown transient. Piping heat transfer during Test S-02-6 has been evaluated to determine its dependence on azimuthal position in the pipe section during a slow depressurization of the system.

The piping wall heat fluxes were determined for the small break test from the observed temperature histories provided by thermocouples installed within the material of the piping walls in both the intact and broken loops. The inverse heat conduction problem was solved through use of a computer program, INVERT^[a]. The measured pipe metal temperatures were input into INVERT to deduce the heat transfer rates and the pipe inner wall temperatures as functions of time. The heat transfer rates and calculated inner wall temperatures were then used in conjunction with the measured fluid temperatures to deduce average heat transfer coefficients existing at given locations as functions of time. An error analysis of the INVERT method including experimental errors as applied to data similar to the Semiscale Mod-1 data is presented in Reference B-1.

Figure B-1 shows the surface heat flux values calculated for the cold legs of the intact and broken loops. The curves include heat fluxes calculated from thermocouples located at the top and bottom of Spool Pieces 15 and 20. The surface heat flux is seen to be negative for the first 40 sec after rupture (prior to the pump stopping) indicating the fluid temperature is greater than the pipe wall temperature. The fluid temperature following rupture rises due to the blowdown mixing processes of the superheated fluid coming from the intact loop hot leg. The intact loop cold leg fluid temperature continues to rise until the pressure due to depressurization causes saturation conditions to exist. The stratification existing at the vessel inlet, shown in Figure B-2, from 120 sec to the time of ECC injection (290 sec) causes the surface heat flux at the bottom of the Spool Piece 15 to be about three times as large as that at the top of the pipe. The time of ECC injection is clearly illustrated in Figure B-1 as the surface heat flux rises sharply at 290 sec due to the subcooled ECC injected. A similar but less severe rise occurs at 340 sec at Spool Piece 20 as the ECC fluid reaches that location on its way to the break.

Integrating the heat flux over the time of the test (0 to 600 sec) results in an energy level for the intact loop cold leg that is about 10% of the prerule fluid energy. The surface heat flux for the small break at Station 15 was compared with the heat flux calculated at the same location for a large break^[B-2]. Comparison of the surface heat flux values led to the conclusion that the small break results in more piping heat transfer than does the large break, and, therefore, results in a larger percentage of the total energy being lost out the break in the Semiscale system.

[a] For the purpose of historical configuration control, INVERT is referenced as program number II00086IB.

B-1 Intact and broken loop cold leg surface heat fluxes – Test S-02-6.

B-2 Vertical and horizontal fluid densities in instrument Spool 15 – Test S-02-6.

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