304
1-6-77
NRC-2
40 cyp
Plus
Cathell

Dr. # 1035

TREE-NUREG-1077
for U.S. Nuclear Regulatory Commission

THERMAL-HYDRAULIC ANALYSIS OF SEMISCALE MOD-1 INTEGRAL BLOWDOWN-REFLOOD TESTS (BASELINE ECC TEST SERIES)

JAMES M. COZZUOL

March 1977

MASTER

EG&G Idaho, Inc.

IDAHO NATIONAL ENGINEERING LABORATORY

ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

IDAHO OPERATIONS OFFICE UNDER CONTRACT EY-76-C-07-1570

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

Printed in the United States of America
Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22161
Price: Printed Copy \$5.00; Microfiche \$3.00

"The NRC will make available data tapes and operational computer codes on research programs dealing with postulated loss-of-coolant accidents in light water reactors. Persons requesting this information must reimburse the NRC contractors for their expenses in preparing copies of the data tapes and the operational computer codes. Requests should be submitted to the Research Applications Branch, Office of Nuclear Regulatory Research, Nuclear Regulatory Commission, Washington, D.C. 20555."

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the Energy Research and Development Administration, nor the Nuclear Regulatory Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

TREE-NUREG-1077

THERMAL-HYDRAULIC ANALYSIS OF SEMISCALE MOD-1

INTEGRAL BLOWDOWN-REFLOOD TESTS

(BASELINE ECC TEST SERIES)

Approved:

D.J. Olson, Manager
Semiscale Program

J. O. Zane, Deputy Director
Water Reactor Research

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Energy Research and Development Administration, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

TREE-NUREG-1077

Distributed Under Category:
NRC-2
Water Reactor Safety Research
Systems Engineering

**THERMAL-HYDRAULIC ANALYSIS OF SEMISCALE MOD-1
INTEGRAL BLOWDOWN-REFLOOD TESTS
(BASELINE ECC TEST SERIES)**

by

James M. Cozzoul

EG&G IDAHO, INC.

March 1977

PREPARED FOR THE
U. S. NUCLEAR REGULATORY COMMISSION
AND
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION
IDAHO OPERATIONS OFFICE
UNDER CONTRACT NO. EY-76-C-07-1570

ACKNOWLEDGMENTS

The author wishes to express his thanks to the personnel of the Fluids Laboratory Division for their efforts in conducting the tests and gathering the data; to the personnel of the Test Integration and Coordination Section for verification and documentation of the data; to the personnel of the Semiscale Data Processing Section for providing data processing support; and to Evelyn L. Pierson for her typing and proofreading efforts.

ABSTRACT

Selected experiment data from tests performed during the Semiscale Mod-1 baseline emergency core cooling test series (Series 4) are analyzed. The analysis is confined primarily to data that have been analyzed with respect to the significant phenomena that occur in the Mod-1 system during the blowdown, refill, and reflood phases of an integral blowdown-reflood test. Analyses concerned with the blowdown phase include an evaluation of the effect of unpowered rods on the departure from nucleate boiling (DNB) behavior of adjacent rods during blowdown. The analysis concerned specifically with the refill and reflood phases of the tests includes evaluations of the downcomer countercurrent flow and hot wall phenomena, the effect of accumulator nitrogen injection on the overall system response, and the downcomer mass depletion phenomena. The corresponding effect of each of these phenomena on the core thermal response is also evaluated.

SUMMARY

The Semiscale Mod-1 experiment 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 a hypothesized loss-of-coolant accident (LOCA) using a small scale experimental nonnuclear system. The Semiscale Mod-1 tests are performed with an experimental system which simulates the principal physical features of a nuclear plant but which is much smaller in volume. Nuclear heating is simulated in the tests by an electrically heated core contained in a pressure vessel. The pressure vessel also includes a downcomer, a lower plenum, and an upper plenum. The system piping is arranged in a 1-1/2 loop configuration with the intact loop containing an active steam generator and pump, and with the broken loop containing passive simulators for the steam generator and pump.

The Semiscale Mod-1 baseline emergency core cooling (ECC) test series was the first group of tests in the Semiscale program designed to study the integral blowdown-reflood response of the system with an electrically heated core. The test series consisted of six blowdown-reflood tests, each of which simulated a 200% offset shear break of the cold leg piping. The tests were conducted under similar operating conditions with the exception that ECC injection rates were varied. Once the desired initial conditions were established and the system reached equilibrium, the tests were initiated by breaking rupture disks downstream of the break nozzles on the vessel inlet and outlet sides of the broken loop. ECC injection into both the intact and broken loop cold legs was accomplished for each test in the series with accumulator, low pressure injection, and high pressure injection systems. In addition, ECC was injected directly into the lower plenum for two of the tests using an accumulator injection system.

The thermal-hydraulic phenomena that occurred in the Semiscale Mod-1 system during the integral blowdown-reflood tests of the baseline ECC test series have been investigated. Selected data from the baseline ECC tests have been analyzed to evaluate the thermal-hydraulic response of the core and the system during the blowdown, refill, and reflood phases of the tests. Since many of the results from the current investigation are similar to results obtained from previous investigations, the discussions presented in this report are limited to results that are significantly different from those observed previously, or to results that will provide an increased understanding of previously analyzed phenomena.

The thermal response of the Semiscale Mod-1 core during the blowdown phase of Tests S-04-5 and S-04-6 of the baseline ECC test series was evaluated to determine the effect of an unpowered rod on the DNB behavior of adjacent rods. Tests S-04-5 and S-04-6 were compared because for Test S-04-5 all 40 core heater rods were powered whereas for Test S-04-6 only 36 of the 40 core heater rods were powered. Results from these tests indicate that unpowering a rod in the Semiscale core shifts the DNB behavior of adjacent rods from early DNB to delayed DNB or early DNB with rewet, or from early DNB with rewet to

delayed DNB. The shift in DNB behavior of rods adjacent to an unpowered rod results in a higher rate of removal of stored energy during the first several seconds of blowdown than occurs if the neighboring rod is powered. Thus, a reduction of the maximum rod cladding temperatures during blowdown of as great as 350°F can occur on rods adjacent to an unpowered rod. The effect of the unpowered rod, however, appears to be limited to those rods directly adjacent to the unpowered rod.

The delivery of ECC fluid to the lower plenum in the Semiscale system is delayed by the downcomer countercurrent flow and hot wall effects. The downcomer countercurrent flow problem occurs between the time ECC injection into the intact loop begins (about 17 seconds after rupture) and the time the system pressure approaches the pressure of the suppression system (about 43 seconds after rupture). The countercurrent steam flow during this time is sufficient to prevent penetration of ECC below the upper fifth of the downcomer. Once the system pressure approaches the suppression system pressure, the countercurrent flow is reduced to a rate which is low enough to allow further penetration of ECC into the downcomer. However, as a result of the hot wall phenomena, an additional delay of about 11 seconds occurs before ECC penetrates to the lower plenum. The hot wall delay is a result of steam generation in the downcomer caused by ECC coming in contact with the hot downcomer walls. The additional steam being generated on the walls, combined with the existing steam flow, restricts further ECC penetration of the downcomer until sufficient energy has been removed from the walls to allow the downcomer fluid head to overcome the buoyant forces of the steam being generated.

The degree of superheat in the steam within the core during refill was found to influence the downcomer countercurrent flow response, and thus the time at which ECC penetrated to the lower plenum, by affecting the system pressure response. A high degree of superheat in the core steam causes the system pressure to remain at a somewhat higher value than would occur if a lesser degree of superheat existed in the core. As a result, the downcomer countercurrent flow remains high longer, and a correspondingly longer delay in penetration of ECC to the lower plenum occurs.

The injection of accumulator nitrogen into the intact loop cold leg, once the liquid supply is depleted, can result in a rapid increase in the system pressure which is significantly higher than could occur due solely to the addition of the nitrogen to the system volume. The increase in system pressure following the nitrogen injection is attributed primarily to the effect of the nitrogen injection on the intact loop and core hydraulic response. In the intact loop, the injection of nitrogen slows the rate of steam flow through the loop, which reduces the overall rate at which steam is removed from the core. As a result, the system pressure begins to increase. However, the pressure increased in several tests to a significantly higher value than could be attributed to the reduction of steam flow in the intact loop alone. A major portion of these large pressure increases is caused by a rapid increase in the rate of steam generation in the core when the accumulator nitrogen flow begins. The injection of nitrogen into the system causes a surge in the core liquid level which rapidly brings additional coolant fluid into contact with the hot heater rods and core support structure. The magnitude of the core liquid level surge, and thus the magnitude of the system pressure increase, is dependent on the rate of nitrogen injection into the system. The

system pressure continues to increase as a result of steam generation in the core until water is blown clear of the vessel inlet side break nozzle, thus providing an additional path for removal of steam from the system. Once water clears the break nozzle, the system pressure begins to decrease. However, because of a choked flow condition at the vessel inlet side break nozzle, the system pressure remains at a considerably higher value during most of reflood than does the suppression system pressure.

In the Semiscale system, once the accumulator liquid flow ceases, heat transfer from the downcomer walls causes fluid in the downcomer to boil. The boiling of the fluid forces liquid from the downcomer thus reducing the effective downcomer head. As a result, the liquid level in the core begins to drop causing an eventual increase in the rod cladding temperatures. The downcomer does not begin to refill again until a partial regeneration of steam in the downcomer insulator gaps occurs thus reestablishing the insulator effectiveness and allowing the low pressure injection system flow to refill the downcomer. Once the downcomer begins to refill, the core cladding temperatures again begin to decrease until quenching finally occurs.

An evaluation of the surface heat flux per unit volume of downcomer fluid during the period when the mass depletion occurs in the Semiscale downcomer indicates that the Semiscale downcomer heat flux is a factor of between 5 and 12 times higher than is required for typicality with a PWR. The high heat fluxes are a result of penetration of ECC liquid into the downcomer filler insulator and core barrel insulator gaps which significantly increases the rate of heat transfer across the gaps.

CONTENTS

ACKNOWLEDGMENTS	ii
ABSTRACT	iii
SUMMARY	iv
I. INTRODUCTION	1
II. EXPERIMENT DESCRIPTION	3
1. SYSTEM HARDWARE DESCRIPTION	3
2. TEST PLAN	11
III. TEST RESULTS	15
1. CORE THERMAL RESPONSE EARLY IN BLOWDOWN	15
2. DOWNCOMER RESPONSE DURING REFILL	24
2.1 Downcomer Countercurrent Flow and Hot Wall Effects	24
2.2 Effect of Core Superheat on Downcomer Countercurrent Flow	30
3. SYSTEM AND CORE RESPONSE DURING REFLOOD	33
3.1 Core Thermal Response During Reflood	33
3.2 Effects of Accumulator Nitrogen Injection	37
3.3 Downcomer Mass Depletion Phenomena	50
IV. CONCLUSIONS	61
1. CORE THERMAL RESPONSE EARLY IN BLOWDOWN	61
2. DOWNCOMER RESPONSE DURING REFILL	61
3. SYSTEM AND CORE RESPONSE DURING REFLOOD	62
V. REFERENCES	64

FIGURES

1.	Semiscale Mod-1 system cold leg break configuration — isometric	3
2.	Semiscale Mod-1 electric heater rod	5
3.	Semiscale Mod-1 vessel with core	6
4.	Vessel cross section and core layout	7
5.	Semiscale Mod-1 heater core — plan view	8
6.	Semiscale Mod-1 heater rod axial power distribution	9
7.	Core power control for the baseline ECC tests	10
8.	Rod cladding temperature response at 25-inch elevation for Tests S-04-5 and S-04-6	22
9.	Rod cladding temperature response at 14-inch elevation for Tests S-04-5 and S-04-6	22
10.	Rod cladding temperature response at 29-inch elevation for Tests S-04-5 and S-04-6	23
11.	Lower plenum fluid density for Test S-04-6	25
12.	Comparison of the system pressure, pressure suppression system pressure, downcomer fluid velocity for Test S-04-6	26
13.	Calculated average fluid densities in the downcomer between the 9- and 26-, 26- and 55-, and 55- and 110-inch elevations for Test S-04-6	26
14.	Comparison of measured downcomer fluid temperature at the 4-inch elevation with the fluid saturation temperature for Test S-04-6	27
15.	Comparison of measured downcomer fluid temperature at the 35-inch elevation with the fluid saturation temperature for Test S-04-6	28
16.	Comparison of measured downcomer fluid temperature at the 70-inch elevation with the fluid saturation temperature for Test S-04-6	28
17.	Comparison of measured downcomer fluid temperature at the 115-inch elevation with the fluid saturation temperature for Test S-04-6	29

18.	Comparison of measured downcomer fluid temperature at the 156-inch elevation with the fluid saturation temperature for Test S-04-6	29
19.	Fluid temperatures at the core inlet for Tests S-04-5 and S-04-6	31
20.	Comparison of system pressure, pressure suppression system pressure, and downcomer fluid velocity for Test S-04-5	31
21.	Downcomer fluid temperature at the 35-inch elevation for Test S-04-5 and Test S-04-6	32
22.	Core inlet fluid densities for Tests S-04-5 and S-04-6	32
23.	Rod cladding temperature response on Rod E4 for Test S-04-6	34
24.	Rod cladding temperature response on Rod G4 for Test S-04-6	34
25.	Rod cladding temperature response on Rod D6 at the 25-inch elevation for Tests S-04-5 and S-04-6	36
26.	Rod cladding temperature response on Rod D8 at the 26-inch elevation for Tests S-04-5 and S-04-6	36
27.	Average rod cladding temperatures as a function of elevation at the initiation of reflood for Tests S-04-5 and S-04-6	37
28.	Quench times as a function of elevation in the core for Test S-04-5	38
29.	Quench times as a function of elevation in the core for Test S-04-6	39
30.	Accumulator nitrogen integrated mass flow rate for Tests S-04-4, S-04-5, and S-04-6	40
31.	System pressure response for Tests S-04-4, S-04-5 and S-04-6 showing effect of accumulator nitrogen injection	41
32.	Comparison of the system pressure with the volumetric flow rate upstream of the ECC injection location in the intact loop for Test S-04-6	42
33.	Comparison of measured fluid temperature in intact loop cold leg with the fluid saturation temperature for Test S-04-6	42
34.	Core collapsed liquid level for Test S-04-5	45
35.	Comparison of the system pressure with the volumetric flow rate upstream of the ECC injection location in the intact loop for Test S-29-1	46

36.	Rod cladding temperature response on Rod F2 at the 7-inch elevation for Tests S-04-4, S-04-5, and S-04-6	47
37.	Rod cladding temperature response on Rod E3 at the 20-inch elevation for Tests S-04-4, S-04-5, and S-04-6	47
38.	Rod cladding temperature response on Rod E4 at the 28-inch elevation for Tests S-04-4, S-04-5, and S-04-6	48
39.	Core collapsed liquid level for Test S-04-4	48
40.	Volumetric flow in the vessel outlet side of the broken loop for Tests S-04-5 and S-04-6	49
41.	Comparison of the system pressure with the fluid density near the break on the vessel inlet side for Test S-04-6	50
42.	Downcomer collapsed liquid level for Test S-04-6	51
43.	Semiscale Mod-1 downcomer insulator configuration	52
44.	Metal and fluid temperatures on the vessel side of the downcomer at the 35-inch elevation for Test S-04-6	53
45.	Metal and fluid temperatures on the vessel side of the downcomer at the 115-inch elevation for Test S-04-6	53
46.	Metal and fluid temperatures on the vessel side of the downcomer at the 156-inch elevation for Test S-04-6	54
47.	Metal and fluid temperatures on the core barrel side of the downcomer at the 70-inch elevation for Test S-04-6	54
48.	Metal and fluid temperatures on the core barrel side of the downcomer at the 115-inch elevation for Test S-04-6	55
49.	Surface heat fluxes from the core barrel side of the downcomer for Semiscale and a PWR	57
50.	Surface heat fluxes from the vessel side of the downcomer for Semiscale and a PWR	58

TABLES

I.	Baseline ECC Series Test Conditions	12
II.	Time After Rupture at Which DNB Occurs	17
III.	Comparison of DNB Response for Test S-04-5 and S-04-6 on Rods Directly Adjacent to an Unpower Rod for Test S-04-6	20
IV.	Comparison of DNB Response for Tests S-04-5 and S-04-6 on Rods that Are not Directly Adjacent to an Unpowered Rod for Test S-04-6	21
V.	Relative Steam Volumetric Flow Rates at the Core Exit for Test S-04-6	44
VI.	Conditions for Heat Conduction Calculations	59

THERMAL-HYDRAULIC ANALYSIS OF SEMISCALE MOD-1

INTEGRAL BLOWDOWN-REFLOOD TESTS

(BASELINE ECC TEST SERIES)

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 a 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 and 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 much 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, a lower plenum, and an upper plenum. The system piping is arranged in a 1-1/2 loop configuration with the intact loop containing an active steam generator and pump, and with the broken loop containing passive simulators for the steam generator and pump.

The baseline emergency core cooling (ECC) test series was the fourth group of tests conducted in the Semiscale Mod-1 experimental program, and was the first group of tests designed to study the integral blowdown-reflood response of the Semiscale system with power applied to the electrically heated core. The primary objectives of the baseline ECC test series were to determine the sensitivity of system conditions at the initiation of reflood to changes in ECC injection scaling, and to determine correct programming of ECC flows to provide end of refill conditions which are typical of PWR end of refill conditions. A thorough evaluation of the Series 4 data relating to these objectives is currently underway.

This report presents an interpretation of the significant phenomena that occur in the Semiscale Mod-1 system during an integral blowdown-reflood test with an electrically heated core. The results provided in this report are limited principally to Tests S-04-5 and

S-04-6, as the phenomena that occur during these two tests are typical of the behavior observed in the other baseline ECC tests. Results from the other tests are included only when the differences in system response can be used to provide additional understanding of the phenomena being studied. Also, since many of the phenomena that occurred during this test series were similar to the phenomena that occurred during previous test programs that have been evaluated and reported, the coverage provided by this report is limited to information where either the results are significantly different from those previously presented or the data from the present test series provide new information that will add to the understanding of the previously analyzed phenomena.

Section II of this report presents a description of the Semiscale Mod-1 system and the test plan for the baseline ECC test series. The results from the analysis of data from the system and core thermal-hydraulic response are presented in Section III, and Section IV presents the more significant conclusions arising from the analysis.

II. EXPERIMENT DESCRIPTION

The Semiscale Mod-1 experimental system and test plan for tests of the baseline ECC test series are described. The topics discussed include the overall system configuration and test plan.

1. SYSTEM HARDWARE DESCRIPTION

The Semiscale Mod-1 test apparatus, shown in Figure 1, is a high pressure system consisting of a pressure vessel with simulated reactor internals (downcomer, lower plenum,

Fig. 1 Semiscale Mod-1 system cold leg break configuration – isometric.

core region, and upper plenum); an intact loop with a steam generator, pump, and pressurizer; a broken loop with rupture diaphragm assemblies, simulated steam generator, and simulated pump; a pressure suppression system with suppression tank and header; and a simulated ECC injection system with accumulators and injection pumps. The system water volume is approximately 7.8 ft³. Detailed descriptions of the system components, including volumes and flow resistances, and of the measurement and data acquisition systems are contained in Reference 2.

The intact loop of the Semiscale system is essentially a 1/30 volume scale model of three loops of a commercial PWR and consists of primary coolant piping, a steam generator, a pressurizer, and a circulating pump. The piping is primarily 3-inch Schedule 160 pipe. The steam generator is a tube-in-shell heat exchanger.

The broken loop is a volume-scaled representation of one loop of a four-loop PWR and consists of an inactive pump and steam generator simulators, two discharge nozzles, and two rupture assemblies which provide a simulated double-ended offset shear. The associated piping is primarily 2-inch Schedule 160 stainless steel. The simulated steam generator and simulated pump consist of piping with orifices to achieve the desired hydraulic resistances. The rupture assemblies contain a converging-diverging blowdown nozzle (to provide the desired break area) and two diaphragm rupture disks.

The pressure suppression system consists of a 91.7-ft³ pressure tank which is used to simulate the backpressure created by a containment building in a PWR system. The tank is maintained partially full of subcooled water and a downcomer pipe projects below the water surface to accommodate the blowdown effluent. A 16-inch header connects the pressure suppression tank to the primary coolant system.

The ECC injection system includes three types of injection subsystems: the compressed nitrogen-water accumulator subsystems, and the low pressure injection and high pressure injection pump subsystems. The ECC injection system interfaces with the vessel lower plenum and the intact loop and broken loop cold legs.

Measurements taken in the Semiscale system include volumetric flow rates, material temperatures, fluid temperatures, pressure drops, pressure, and fluid densities. Thermocouples were used for the material and fluid temperatures. The differential pressure drop measurements were made with calibrated differential pressure transducers. The volumetric flow rates were obtained with full flow turbine flowmeters, and the fluid densities were obtained with both vertical and horizontal single beam gamma densitometers. The total expected error in each type of measurement is identified in Reference 2.

The core simulator in the Semiscale system consists of 40 electrically heated rods of typical PWR fuel rod diameter (0.422 inch) and axial power peaking (1.58). Figure 2 illustrates the general construction of a Semiscale heater rod. The overall length of the rod is about 207 inches, and the bottom of the 5.5-foot heated section is about 143 inches 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.

Fig. 2 Semiscale Mod-1 electric heater rod.

Fig. 3 Semiscale Mod-1 vessel with core.

The rod heating element is constructed of constantan wire (55% copper, 45% nickel), coiled with a varying pitch, and sized to develop a specified power. Compacted boron nitride surrounds the element and insulates it from a composite sheath. The composite sheath (Figure 2) was manufactured from 316L stainless steel. The inner sheath was creased along the total rod length (convexly) at four locations spaced azimuthally around the rod circumference so as to accept four 0.025-inch-diameter thermocouple assemblies. The outer sheath was positioned over the inner sheath after installation of the thermocouple, and the composite assembly was redrawn. The thermocouples provide rod cladding temperature measurements at four different axial locations along the 5.5-foot heated length of the heater rods.

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 inch). Figure 4 shows a plan view of the Semiscale vessel and core simulator assembly, and Figure 5 illustrates the heater rod matrix and cladding thermocouple locations. The heater rods are located within the matrix by reference to the row of letters across the top and the column of numbers down the side of the matrix. Similarly, the thermocouples are located by the rod that they are on and by

Fig. 4 Vessel cross section and core layout.

Fig. 5 Semiscale Mod-1 heater core – plan view.

their elevation above the bottom of the heated length of the core. The thermocouple on Rod D5 at the 29-inch elevation is thus referred to as TH-D5-29 where TH means 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 5 indicate thermocouple azimuthal locations.) A heat shield assembly, composed of 0.018-inch 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) individually produce 66.23 kW at 173 Vdc and the remaining 36 rods (the low power rods) each produce 36.9 kW at 173 Vdc. The as-constructed axial power profile for the rods is illustrated in Figure 6. These profiles allow the simulation of radial peaking that exists in nuclear cores. A sufficient factor of safety was built into the rods, however, to allow for the implementation of a flat radial power profile in the core simulator if desired.

Fig. 6 Semiscale Mod-1 heater rod axial power distribution.

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 disk 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.

Figure 7 shows the transient electrical core power control used for the six baseline ECC tests^[a]. The electrical power control was defined to allow simulation of the thermal response of a nuclear rod with the electrical rods, and was determined by analytically matching the surface heat flux calculated for a nuclear rod, assuming that both rods were subjected to the same transient boundary conditions (obtained from test data). A detailed description of the technique used to determine the core power control is contained in Reference 3. The core power decay curve after 20 seconds is based on the American Nuclear Society (ANS) standard power decay curve^[4].

Fig. 7 Core power control for the baseline ECC tests.

[a] The core power decay curve used for Test S-04-4 was slightly modified to duplicate the decay curve used for Test S-02-9 of the blowdown heat transfer test series (Test Series 2).

2. TEST PLAN

Table I summarizes the pertinent information relating to the integral blowdown-reflood tests of the baseline ECC test series. The test series included six tests that were conducted under similar operating conditions, with the exception that the ECC injection rates were varied. The data obtained from these tests are presented in References 5 through 9. Tests S-04-1 through S-04-4 were conducted primarily to determine the sensitivity of system conditions at the initiation of reflood to changes in ECC injection scaling. Test S-04-1 was conducted with the ECC injection parameters system volume scaled. The ECC injection parameters for Test S-04-2 and S-04-3 were downcomer volume scaled. In addition, Test S-04-3 included ECC injection into the vessel lower plenum to eliminate the delay of ECC to the lower plenum caused by the Semiscale downcomer hot wall phenomena. Test S-04-4 was essentially a repeat of Test S-04-3 with the exception of the reduced lower plenum volume. The lower plenum volume for Test S-04-4 was only 56% of the lower plenum volume for Test S-04-1 through S-04-3.

Tests S-04-5 and S-04-6 were added to the test series after an evaluation of the earlier tests indicated that a test with a volumetrically scaled accumulator injection and a core area scaled low pressure injection system would be more suitable for the baseline integral blowdown-reflood test. In addition, Tests S-04-5 and S-04-6 were specifically designed to investigate the effect of unpowered rods on the departure from nucleate boiling (DNB) behavior of adjacent rods. To this end, all 40 rods were powered for Test S-04-5 whereas Rods C3, F3, D5 and F6 were left unpowered for Test S-04-6 to simulate the control rod guide tubes and instrument thimbles in a typical PWR fuel assembly. The lower plenum volume for Tests S-04-5 and S-04-6 was the same reduced lower plenum volume used for Test S-04-4.

With the exception of Test S-04-6, all tests in this series were conducted with a flat radial power profile. The power density at the high power zone of the heater rods was about 11.5 kW/ft. For Test S-04-6, Rods D4, E4, and E5 were operated at a 5% higher peak power density (12.1 kW/ft) than the remaining 33 rods to simulate the radial power profile near a control rod guide tube in a PWR fuel assembly. The total power supplied to the core varied for each test between 1.44 and 1.60 MW depending on the number of powered rods.

The test sequence for the integral blowdown-reflood experiments was essentially the same for all tests. Warmup to the initial test conditions was accomplished with the core heater rods. The system fluid was circulated through the intact loop and vessel at a rate sufficient to establish the desired core differential temperature. Heatup of the broken loop piping was accomplished with bypass lines which served to allow temporary circulation through the broken loop. Once the initial conditions were established and the system had reached equilibrium, the tests were initiated by breaking rupture disks in both the vessel inlet side and vessel outlet side of the broken loop. ECC injection began when the system pressure at the injection location fell to a preset value. The ECC injection rates themselves were controlled by adjusting the hydraulic resistance of the injection line prior to each test.

TABLE I

BASELINE ECC SERIES TEST CONDITIONS

	Test S-04-1	Test S-04-2	Test S-04-3	Test S-04-4	Test S-04-5	Test S-04-6
Core power (MW)	1.53	1.53	1.56	1.55	1.60	1.44
Intact loop cold leg fluid temperature ($^{\circ}$ F)	542	542	543	541	541	543
Hot leg to cold leg temperature differential ($^{\circ}$ F)	67	67	66	68	68	67
Steam generator feedwater temperature ($^{\circ}$ F)	440	440	440	440	440	440
Steam generator liquid level (from bottom of tube sheet) (in.)	116	116	116	116	116	116
Pressurizer pressure (psia)	2263	2263	2258	2263	2267	2252
Pressurizer liquid level (from bottom of hemispherical section) (in.)	23.5	23.5	23.5	23.5	25.3	24.9
Intact loop cold leg flow rate (gpm)	143	145	150	143	158	141
Pressure suppression system pressure (psia)	36.9	36.8	37.8	33.3	35.0	35.0

TABLE I (continued)

	Test S-04-1	Test S-04-2	Test S-04-3	Test S-04-4	Test S-04-5	Test S-04-6
Number of core heater rods powered	38	38	39	37	40	36
Intact loop accumulator						
Liquid volume (ft ³)	2.55	4.61	3.57	3.57	3.10	2.83
Injection rate (gpm)	25.4	42.8	42.2	42.0	23.0	23.0
Actuation pressure (psia)	545	604	620	650	608	607
Intact loop LPIS						
Actuation pressure (psia)	208	196	186	150	212	212
Injection rate (gpm)	2.7	5.0	5.9	6.0	5.2	4.7
Broken loop accumulator						
Liquid volume (ft ²)		1.23	1.12	1.12	0.58	0.58
Injection rate (gpm)	8.0	14.9	NA	14.8	9.0	7.8
Actuation pressure (psia)	425	435	587	550	612	612
Broken loop LPIS						
Actuation pressure (psia)	57	40	68	162	117	212
Injection rate (gpm)	1.0	1.7	1.4	1.8	1.2	1.2

TABLE I (continued)

	<u>Test S-04-1</u>	<u>Test S-04-2</u>	<u>Test S-04-3</u>	<u>Test S-04-4</u>	<u>Test S-04-5</u>	<u>Test S-04-6</u>
Lower plenum accumulator	none	none			none	none
Injection start time (sec)			38.0	39.0		
Injection stop time (sec)			52.5	53.0		
Liquid volume (ft ³)			1.48	1.48		
Injection rate (gpm)			45.8	47.7		

III. TEST RESULTS

The thermal-hydraulic response of the Semiscale Mod-1 system during integral blowdown-reflood tests has been evaluated. Since the results of parts of the analysis are similar to those presented previously^[3, 10], the analysis presented herein is confined to data which are significantly different from previous results or to data which will provide an increased understanding of previously analyzed phenomena. The results of the evaluation are discussed in the following sections in terms of phenomena associated with the blowdown, refill, and reflood phases of the tests^[a]. The initial section concerns the effect of unpowered rods on the DNB behavior of adjacent rods. The second section discusses the phenomena which occur in the vessel downcomer during the refill portion of a test and includes an analysis of the downcomer countercurrent flow and hot wall phenomena, as well as an analysis of the effect of core superheat on the downcomer countercurrent flow. The final section includes discussions concerning the core thermal response during reflood, the effect of accumulator nitrogen injection on the overall system response, and the downcomer mass depletion phenomenon. The discussions in these three sections present results primarily from Tests S-04-5 and S-04-6 as the behavior observed in these tests is typical of the response of other tests in the baseline ECC series.

1. CORE THERMAL RESPONSES EARLY IN BLOWDOWN

The thermal response of the Semiscale Mod-1 electrically heated core during blowdown was investigated extensively in the blowdown heat transfer test series (Series 2). The results of this investigation are included in Reference 3 which contains evaluations of: (a) the effect of core flow direction and magnitude on the heater rod thermal response; (b) the effects of initial operating conditions on the rod cladding temperature behavior during the transient; (c) the radial and axial patterns of the departure from nucleate boiling and rewetting characteristics of the rod surface; and (d) the repeatability of the core thermal response data. In addition, the reference includes comparisons of the core thermal response data with the response calculated with the RELAP4 code^[11].

Since the core thermal response during blowdown for tests in the baseline ECC test series is similar to the response observed in the blowdown heat transfer test series, results of an evaluation similar to that contained in Reference 3 are not presented in this report. However, results from the early tests of the baseline ECC test series indicated that rods adjacent to an unpowered rod (such as a failed rod or liquid level probe) exhibited

[a] For purposes of this report, the blowdown phase of a test is defined as the time between rupture and the initiation of ECC injection into the intact loop. The refill phase extends from the initiation of ECC injection until ECC fluid reaches the bottom of the heated length of the core. The reflood phase is defined as the time during which actual core reflooding takes place.

an increased tendency toward rewet and delayed DNB during the blowdown transient. Thus Tests S-04-5 and S-04-6 of the present test series were designed specifically to allow an evaluation of the effect of unpowered rods on the DNB behavior of adjacent rods in the Semiscale core, and to provide a means of qualitatively evaluating the effect of control rod guide tubes and instrument thimbles on the DNB behavior in a PWR fuel assembly. The results obtained from these evaluations are presented herein.

To test the premise that unpowered rods in the Semiscale core (or control rod guide tubes in a PWR) affect the DNB behavior of adjacent rods, Test S-04-5 was conducted with all 40 rods powered, whereas Test S-04-6 was conducted with four of the 40 rods unpowered (Rods C3, F3, D5, and F6). Figure 5 shows the locations of the four unpowered rods for Test S-04-6. In the Semiscale Mod-1 core geometry it is not possible to represent, rod for rod, the same configuration of fuel rods and control rod thimbles that exist in a section of a PWR fuel assembly. Thus the configuration of unpowered rods for Test S-04-6 was chosen to provide half core symmetry, and to allow one of the unpowered rods (Rod D5) to be adjacent to the high power rods (Rods D4, E4, and E5). All powered rods in Tests S-04-5 and S-04-6 were operated at an initial peak power density of 11.5 kW/ft, with the exception that for Test S-04-6, the three high power rods were operated at an initial peak power density of 12.1 kW/ft (5% peaked). The increased peak power on these three rods represents the increased power of rods that are adjacent to a control rod guide tube in a PWR fuel assembly. All initial conditions were essentially the same for both tests, with the exception of the differences in rod power and the corresponding differences in the core inlet flow required to establish the core differential temperature. An evaluation of the rod cladding temperatures for Tests S-04-5 and S-04-6 indicates that a one-to-one correlation exists between changes in DNB behavior and the location of the unpowered rods. Table II summarizes the times at which DNB occurs at each thermocouple location for Tests S-04-5 and S-04-6, and indicates the thermocouple locations which exhibit rewet shortly following DNB. Individual rod behavior that is different between the two tests, in most instances, is a direct result of the positions of the unpowered rods in Test S-04-6. The table shows that for Test S-04-5 most of the rod cladding temperatures between the 0- and 26-inch elevations^[a] indicate early DNB, whereas between the 26- and 33-inch elevations a combination of early DNB, early DNB with rewet, and delayed DNB type responses occur^[b]. For Test S-04-6, however, the cladding temperatures between the 0- and 26-inch elevations indicate a combination of early DNB, early DNB with rewet, and delayed DNB, and most of the cladding temperatures between the 26- and 33-inch elevations indicate delayed DNB. (The cladding temperature response during blowdown is similar for both tests above the 33-inch elevation). For Test S-04-6, 28 of the 34 rod thermocouple locations within the 0- and 33-inch elevations on rods adjacent to unpowered rods exhibit a change from early DNB to early DNB with rewet or delayed DNB, or from early DNB with rewet to delayed DNB

[a] Elevations are measured from the bottom of the heated length of the rod.

[b] For purposes of this report, early DNB is defined as a DNB which occurs prior to 1 second after rupture. Delayed DNB is defined as a DNB which occurs between 1 and 4 seconds after rupture.

TABLE II
TIME AFTER RUPTURE AT WHICH DNB OCCURS

<u>Thermocouple</u>	<u>Time (sec)</u>	
	<u>Test S-04-5</u>	<u>Test S-04-6</u>
TH-E3-05	0.160	0.160 ^[a]
TH-C7-07 ^[b]	0.765	0.817
TH-F2-07	0.160	0.160 ^[a]
TH-A4-09 ^[b]	0.817	0.748
TH-E4-09	0.991	0.830
TH-E6-09	0.887	3.287
TH-G3-13	0.696	0.939 ^[a]
TH-D2-14	0.800	0.696 ^[a]
TH-D4-14	0.678 ^[a]	3.061
TH-D6-14	0.713	3.182
TH-E8-14 ^[b]	0.626	0.609
TH-F4-14	0.765	3.322
TH-G5-14	0.730	0.678 ^[a]
TH-C7-15 ^[b]	0.835 ^[a]	0.678
TH-D7-20 ^[b]	0.696	0.678
TH-E2-20	0.626	0.678 ^[a]
TH-E3-20	0.713	3.200
TH-F5-20	0.626	3.269
TH-D1-21 ^[b]	0.540	0.539
TH-E5-21	0.630	0.870 ^[a]
TH-F2-22	0.626	3.165
TH-E3-24	0.783 ^[a]	3.182
TH-G5-24	0.678	2.974
TH-D6-25	0.640	3.078
TH-E5-25	0.643	2.974
TH-F2-25	0.643	0.835 ^[a]
TH-C4-26	0.835 ^[a]	3.356
TH-D8-26 ^[b]	0.609	0.539
TH-F5-26	0.835 ^[a]	3.113

TABLE II (continued)

Thermocouple	Time (sec)	
	Test S-04-5	Test S-04-6
TH-B5-28 ^[b]	0.852 ^[a]	3.235
TH-C5-28	0.830 ^[a]	3.391
TH-E4-28	0.765 ^[a]	3.182
TH-E4-28	0.956 ^[a]	3.443
TH-E6-28	0.852 ^[a]	3.304
TH-F4-28	0.835 ^[a]	3.252
TH-A4-29 ^[b]	0.730 ^[a]	0.609 ^[a]
TH-A5-29 ^[b]	0.574	0.556
TH-B6-29 ^[b]	0.591	0.487
TH-D3-29	3.426	3.495
TH-D4-29	3.582	3.461
TH-E8-29 ^[b]	0.696 ^[a]	0.556 ^[a]
TH-G4-29	0.800 ^[a]	2.869
TH-H5-32 ^[b]	0.835 ^[a]	3.148 ^[a]
TH-B3-33	0.556 ^[a]	3.287
TH-B5-33 ^[b]	3.269	3.304
TH-E1-33 ^[b]	0.783 ^[a]	3.148
TH-E2-33	3.548 ^[a]	3.635 ^[a]
TH-F5-33	3.480	3.356
TH-G4-33	3.548	3.130
TH-E6-37	3.165 ^[a]	3.669
TH-C2-38	3.550 ^[a]	3.043 ^[a]
TH-G4-38	3.252 ^[a]	3.269
TH-D3-39	3.374 ^[a]	3.548 ^[a]
TH-A5-44 ^[b]	3.509 ^[a]	3.495 ^[a]
TH-F4-44	3.390 ^[a]	3.180 ^[a]
TH-E7-45	3.391 ^[a]	3.600 ^[a]
TH-E8-45 ^[b]	3.687 ^[a]	3.440 ^[a]
TH-C4-53	[c]	[c]
TH-C6-53	[c]	[c]

TABLE II (continued)

Thermocouple	Time (sec)	
	Test S-04-5	Test S-04-6
TH-F5-54	[c]	[c]
TH-E4-55	[c]	[c]
TH-D8-57 ^[b]	[c]	[c]
TH-D2-61	[c]	[c]

[a] The temperature measurement indicates the rod rewet shortly after DNB occurs.

[b] The thermocouple is on a heater rod that is not directly adjacent to an unpowered rod.

[c] The temperature measurement indicates that DNB does not occur, but that the cladding temperature follows the fluid saturation temperature.

when compared with Test S-04-5. In addition, of the six temperature measurements which did not indicate a change in DNB behavior, five exhibit delayed DNB for both tests. However, on rods which were not adjacent to unpowered rods in Test S-04-6, only 3 of the 15 thermocouple locations within the 0- and 33-inch elevations exhibit a shift in DNB behavior toward delayed DNB when compared with Test S-04-5. For reference purposes, Table III compares the type of DNB behavior as a function of the rod power step for the two tests for those rods which were directly adjacent to an unpowered rod in Test S-04-6, and Table IV provides a similar comparison for those rods which were not directly adjacent to an unpowered rod in Test S-04-6. As shown in these tables the effect of the unpowered rod on DNB behavior is primarily limited to rods directly adjacent to the unpowered rods.

In general, results presented show that an unpowered rod in the Semiscale core results in a strong tendency to shift the DNB behavior of adjacent rods toward early DNB with rewet and delayed DNB. The shift in DNB behavior from early DNB to early DNB with rewet or delayed DNB, in turn, results in a significant improvement in the energy removed from the rods during the blowdown, and thus the maximum rod temperatures achieved are considerably reduced.

Figures 8 through 10 compare the different types of DNB behavior observed in Tests S-04-5 and S-04-6, and show the significantly lower rod cladding temperatures that result when a shift in the DNB behavior from early DNB to delayed DNB (Figure 8), or from early DNB to early DNB with rewet (Figure 9) occurs. Figure 10 compares the rod cladding

TABLE III

COMPARISON OF DNB RESPONSE FOR TEST S-04-5 AND S-04-6
ON RODS DIRECTLY ADJACENT TO AN UNPOWER ROD FOR TEST S-04-6

Power Step Elevation (inches)	Number of Thermocouple Locations which Indicate Early DNB		Number of Thermocouple Locations which Indicate Early DNB with Rewet		Number of Thermocouple Locations which Indicate Delayed DNB	
	Test S-04-5	Test S-04-6	Test S-04-5	Test S-04-6	Test S-04-5	Test S-04-6
0-5	1(100%) ^[a]	0(0%)	0(0%)	1(100%)	0(0%)	0(0%)
6-10	3(100%)	1(33%)	0(0%)	1(33%)	0(0%)	1(33%)
11-15	5(83%)	0(0%)	1(17%)	3(50%)	0(0%)	3(50%)
16-20	3(100%)	0(0%)	0(0%)	1(33%)	0(0%)	2(67%)
21-30	6(35%)	0(0%)	9(53%)	2(12%)	2(12%)	15(88%)
31-35	0(0%)	0(0%)	1(25%)	0(0%)	3(75%)	4(100%)
36-40	0(0%)	0(0%)	0(0%)	0(0%)	4(100%)	4(100%)
41-47	0(0%)	0(0%)	0(0%)	0(0%)	2(100%)	2(100%)
48-54 ^[b]	0(0%)	0(0%)	0(0%)	0(0%)	0(0%)	0(0%)
55-66 ^[b]	0(0%)	0(0%)	0(0%)	0(0%)	0(0%)	0(0%)

[a] Numbers in parenthesis are percentages of thermocouple measurements on a given power step which show the indicated DNB behavior.

[b] Rod temperatures follow fluid saturation temperature.

TABLE IV
 COMPARISON OF DNB RESPONSE FOR TESTS S-04-5 AND S-04-6
 ON RODS THAT ARE NOT DIRECTLY ADJACENT TO AN UNPOWERED ROD FOR TEST S-04-6

Power Step Elevation (inches)	Number of Thermocouple Locations which Indicate Early DNB		Number of Thermocouple Locations which Indicate Early DNB with Rewet		Number of Thermocouple Locations which Indicate Delayed DNB	
	Test S-04-5	Test S-04-6	Test S-04-5	Test S-04-6	Test S-04-5	Test S-04-6
0-5	[a]					
6-10	2(100%) ^[b]	2(100%)	0(0%)	0(0%)	0(0%)	0(0%)
11-15	1(50%)	2(100%)	1(50%)	0(0%)	0(0%)	0(0%)
16-20	1(100%)	1(100%)	0(0%)	0(0%)	0(0%)	0(0%)
21-30	4(57%)	4(57%)	3(43%)	2(29%)	0(0%)	1(14%)
31-35	0(0%)	0(0%)	2(50%)	0(0%)	2(50%)	4(100%)
36-40						
41-47	0(0%)	0(0%)	0(0%)	0(0%)	2(100%)	2(100%)
48-54 ^[c]	0(0%)	0(0%)	0(0%)	0(0%)	0(0%)	0(0%)
55-66 ^[c]	0(0%)	0(0%)	0(0%)	0(0%)	0(0%)	0(0%)

[a] No thermocouples were available on this power step on rods not adjacent to unpowered rods.

[b] Numbers in parenthesis are percentages of thermocouple measurements on a given power step which show the indicated DNB behavior.

[c] Rod temperatures follow fluid saturation temperature during blowdown.

Fig. 8 Rod cladding temperature response at 25-inch elevation for Tests S-04-5 and S-04-6.

Fig. 9 Rod cladding temperature response at 14-inch elevation for Tests S-04-5 and S-04-6.

Fig. 10 Rod cladding temperature response at 29-inch elevation for Tests S-04-5 and S-04-6.

temperature response on a rod which was not adjacent to an unpowered rod and shows the similarity in the DNB behavior. The differences in rod cladding temperatures during blowdown for Tests S-04-5 and S-04-6 caused by the shifts in DNB response continue through the refill and reflood phases of the tests. The effect of the rewet and delayed DNB behavior on the core thermal response during reflood is discussed in Section 3.1.

Since a PWR fuel assembly has numerous control rod guide tubes, the results from Semiscale Tests S-04-5 and S-04-6 indicate that similar delays in DNB and improved fuel rod cooling during blowdown may be observed in the PWR fuel assembly. However, although the presence of unpowered rods in the Semiscale core results in a significant increase in the time to DNB on adjacent rods, and thus improves energy removal from the core early in the blowdown transient, the differences in DNB response between Tests S-04-5 and S-04-6 cannot be used directly to quantitatively evaluate the effect of control rod thimbles and instrument tubes on the DNB time in a PWR fuel assembly. In a PWR fuel assembly the control rod thimble diameter is significantly larger than the diameter of a fuel rod (or a Semiscale heater rod). As a result, in the Semiscale core the subchannel adjacent to an unpowered rod contains about 15% more volume for fluid than the subchannel adjacent to a control rod thimble in a PWR fuel assembly.

In addition, results of a series of blowdown tests performed at Oak Ridge National Laboratory using a 49-rod electrically heated core (four rods unpowered) with 12-foot heater rods, indicate that the shift in DNB behavior caused by an unpowered rod is a function of the power density of the rods adjacent to the unpowered rod. The power densities at the rod high power zone for the Oak Ridge tests were varied between

approximately 11 and 17 kW/ft. The tests conducted at the higher rod power densities exhibited a substantially smaller shift in the time to DNB on the rods adjacent to an unpowered rod than was observed for tests with the lower rod power densities.

Since DNB behavior is a complex function of both fluid conditions adjacent to the rod and the rod power density at rupture, differences in these variables can strongly influence the time at which DNB occurs. Thus, further analysis, making use of such 3-D codes as CØBRA^[12], is required to adequately evaluate the effects of such variables as subchannel fluid volume and rod power on DNB behavior before direct quantitative comparison between the Semiscale DNB response and that expected in a PWR can be made.

2. DOWNCOMER RESPONSE DURING REFILL

The phenomena that occur in the Semiscale downcomer during refill have been investigated extensively in several previous Semiscale test programs including the Mod-1 isothermal tests^[10]. An evaluation of the data taken during these previous Semiscale tests showed that delivery of ECC fluid to the lower plenum and core region in the Semiscale system is delayed by two different phenomena. These phenomena include the downcomer countercurrent flow and the effect of downcomer hot walls. During the period when the pressure in the core is significantly higher than the pressure suppression system pressure, the countercurrent steam flow in the downcomer is sufficiently high to restrict the penetration of ECC fluid into the downcomer. In addition, once ECC fluid begins to penetrate the downcomer, heat transfer from the downcomer walls generates additional steam which combines with the existing steam flow and further restricts ECC penetration.

Since the baseline ECC tests were the first integral blowdown-reflood tests with an electrically heated core, the test results have been evaluated to determine the extent to which the downcomer countercurrent flow and hot wall phenomena delay the delivery of ECC fluid to the lower plenum under test conditions that are more typical of expected PWR conditions than existed during previous Semiscale tests. In addition, the effect of core steam superheat on the downcomer countercurrent flow response has been evaluated.

2.1 Downcomer Countercurrent Flow and Hot Wall Effects

An analysis of data from Test S-04-6 indicates that a considerable delay in delivery of ECC to the lower plenum occurs in the Semiscale Mod-1 system as a result of the downcomer countercurrent flow and hot wall effects. The lower plenum density measurement shown in Figure 11 indicates that significant amounts of ECC fluid do not reach the lower plenum until about 54 seconds after rupture, which is a delay of 37 seconds from the time ECC accumulator flow began. Analysis of test data shows that the first 26 seconds of the 37-second delay can be attributed primarily to the downcomer countercurrent flow, whereas the remaining 11 seconds can be attributed to the hot wall effect. The 11-second hot wall delay time for Test S-04-6 compares with a hot wall delay

Fig. 11 Lower plenum fluid density for Test S-04-6.

time of about 12 seconds^[10] obtained during Test S-01-2 of the isothermal test series. The ECC injection into the intact loop cold leg for Test S-01-2 was not initiated until the downcomer countercurrent flow had dropped to essentially zero, so that the 12-second delay in delivery of ECC to the lower plenum can be attributed entirely to the hot wall effect.

The portion of the delay in delivery of ECC to the lower plenum attributable solely to the countercurrent flow effect and the portion attributable to the hot wall effect for Test S-04-6 were determined by comparing various system pressures with the downcomer fluid velocity, and by comparing the average mass per unit volume at several elevations in the downcomer obtained from downcomer differential pressure measurements. Figure 12 compares the system pressure, the pressure suppression system pressure, and the downcomer fluid velocity for Test S-04-6. As shown in the figure the average velocity of the steam in the downcomer remains at a fairly high negative value (upward flow) until the system pressure approaches the suppression system pressure at about 43 seconds. (The large oscillations in the downcomer velocity prior to this time are caused by local pressure oscillations near the ECC injection location in the cold leg and in the upper section of the vessel downcomer, resulting from condensation effects). Once the suppression system pressure is reached (at about 43 seconds) the steam velocity in the downcomer becomes small, and penetration of ECC fluid below the 26-inch elevation in the downcomer occurs. Figure 13 compares the calculated average densities (obtained from differential pressure measurements) in the downcomer between the 9- and 26-, 26- and 55-, and 55- and 110-inch elevations, and shows

Fig. 12 Comparison of the system pressure, pressure suppression system pressure, and downcomer fluid velocity for Test S-04-6.

Fig. 13 Calculated average fluid densities in the downcomer between the 9- and 26-, 26- and 55-, and 55- and 110-inch elevations for Test S-04-6.

that substantial penetration of ECC into the downcomer does not occur until about 43 seconds. Considerable amounts of liquid are stored in the downcomer above the 26-inch elevation prior to this time as indicated by the high average density between the 9- and 26-inch elevation.

The hot wall portion of the refill phase of Test S-04-6 occurs between about 43 and 54 seconds. Figure 13 shows that during this time the average density in the downcomer first increases due to the penetration of subcooled ECC, and then decreases considerably as heat transfer from the downcomer walls causes a rapid increase in the rate of steam generation. The increased steam generation rate results in a substantial increase in the steam velocity up the downcomer (as is indicated in Figure 12 between about 50 and 53 seconds) which in turn essentially stops further ECC penetration of the downcomer. Once sufficient energy has been removed from the downcomer walls, the steam generation rate is reduced to a point where the downcomer liquid head is large enough to overcome the steam buoyant force and penetration of ECC fluid into the lower plenum occurs.

The downcomer fluid temperatures provide an additional means of monitoring the downcomer response during the hot wall phase of a test. Figures 14 through 18 compare the downcomer fluid temperatures at the 4-, 35-, 70-, 115-, and 156-inch elevations^[a] for Test S-04-6. The initial penetration of ECC fluid into the downcomer, after the system pressure

Fig. 14 Comparison of measured downcomer fluid temperature at the 4-inch elevation with the fluid saturation temperature for Test S-04-6.

[a] Downcomer elevations are given as distances below the cold leg centerline.

Fig. 15 Comparison of measured downcomer fluid temperature at the 35-inch elevation with the fluid saturation temperature for Test S-04-6.

Fig. 16 Comparison of measured downcomer fluid temperature at the 70-inch elevation with the fluid saturation temperature for Test S-04-6.

Fig. 17 Comparison of measured downcomer fluid temperature at the 115-inch elevation with the fluid saturation temperature for Test S-04-6.

Fig. 18 Comparison of measured downcomer fluid temperature at the 156-inch elevation with the fluid saturation temperature for Test S-04-6.

falls to the pressure of the suppression system, results in subcooled ECC reaching to at least the 35-inch elevation. However, the downcomer hot wall effect restricts penetration of ECC much below that elevation. The fluid temperatures at the 70-inch elevation and below do not indicate the presence of subcooled ECC, and when compared with the small increase in average density between the 55- and 110-inch elevations shown in Figure 13, indicate that very little ECC actually penetrates below the 55-inch elevation during the hot wall period. Once the hot wall effect is overcome, however, ECC penetration of the downcomer occurs very quickly. The fluid temperature responses at the 35- and 156-inch elevations shown in Figures 15 and 18 indicate that about two seconds is required for ECC to reach the lower plenum once the hot wall effect is overcome.

2.2 Effect of Core Superheat on Downcomer Countercurrent Flow

The degree of superheat in the steam within the core during refill was found to influence the downcomer countercurrent flow response (and thus the time at which core reflood began) by affecting the system depressurization rate. For tests in which the steam flowing from the core had a low degree of superheat during refill, the system pressure decreased at a somewhat faster rate than for tests with a higher degree of superheat. The faster rate of depressurization, in turn, resulted in a shorter period during which the countercurrent flow effectively restricted ECC penetration of the downcomer.

The effect of core steam superheat on the countercurrent flow response was evaluated by comparing results from Tests S-04-5 and S-04-6. The improved energy removal from the core during blowdown for Test S-04-6 (as discussed in Section 1) caused the average core cladding temperatures to be considerably lower than for Test S-04-5. The lower cladding temperatures for Test S-04-6, combined with the fact that there were four fewer rods powered, resulted in a significantly lower degree of superheat in the core steam during the time interval between 19 and 50 seconds. Figure 19 shows the core inlet fluid temperatures for the two tests and illustrates the differences in core steam superheat.

A comparison of the system pressure, pressure suppression system pressure, and downcomer fluid velocity for Test S-04-5, shown in Figure 20, with the same variables for Test S-04-6 (shown in Figure 12) illustrates the changes in depressurization rate and downcomer fluid velocity caused by the differences in core steam superheat. For Test S-04-5 the system pressure does not fall to the pressure suppression system pressure until about 47 seconds (as compared with 43 seconds for Test S-04-6) because of the higher core steam superheat. As a result, penetration of ECC fluid into the downcomer was delayed for Test S-04-5 by about four seconds. The delay in ECC penetration of the downcomer is illustrated in Figure 21 which compares the downcomer fluid temperatures at the 35-inch elevation below the cold leg centerline for the two tests. The figure indicates that subcooled ECC first reached the 35-inch elevation at about 43 seconds for Test S-04-6 as compared with about 47 seconds for Test S-04-5. These times correspond to the times at which the system pressure approaches the pressure suppression system pressure for the two tests.

The overall effect of the higher core superheat is to increase the time at which core reflood commences. Figure 22 compares the core inlet fluid densities for Tests S-04-5 and

Fig. 19 Fluid temperatures at the core inlet for Tests S-04-5 and S-04-6.

Fig. 20 Comparison of system pressure, pressure suppression system pressure, and downcomer fluid velocity for Test S-04-5.

Fig. 21 Downcomer fluid temperature at the 35-inch elevation for Test S-04-5 and Test S-04-6.

Fig. 22 Core inlet fluid densities for Tests S-04-5 and S-04-6.

S-04-6 and indicates that core reflood begins at about 62 and 58 seconds, respectively. The 4-second longer delay in the start of reflood for Test S-04-5 is directly attributable to the changes in the downcomer countercurrent flow caused by the higher degree of core superheat.

Since core superheat affects the downcomer countercurrent flow, refill or reflood codes must calculate the degree of core superheat to adequately model the phenomena occurring in the downcomer and to determine the time at which core reflood begins.

3. SYSTEM AND CORE RESPONSE DURING REFLOOD

The phenomena that occur in the Semiscale system during the core reflood phase of a blowdown-reflood test have been investigated. The areas covered in the investigation include: (a) the core thermal response during reflood, (b) the effect of accumulator nitrogen injection on the overall system response, and (c) the downcomer mass depletion phenomenon and its effect on the core thermal response.

3.1 Core Thermal Response During Reflood

The core thermal response is strongly affected by several events which occur in the system during the reflood phase of a test. These events include the injection of accumulator nitrogen into the intact loop cold leg after the liquid supply is depleted as well as a downcomer mass depletion process which occurs when the downcomer fluid temperature is raised to the saturation temperature. In addition, the type of DNB response (for example, early versus delayed DNB) experienced by the rods in the core during the blowdown phase of a test can strongly influence the thermal response of the core during reflood.

The effects of nitrogen injection and downcomer mass depletion on the core thermal response are illustrated by comparing the rod cladding temperatures at various elevations in the core. Figures 23 and 24 show the available cladding temperatures on Rods E4 and G4 for Test S-04-6. These temperatures are typical of the response observed for most rods in the core. Just prior to injection of accumulator nitrogen all the cladding temperatures, with the exception of the 9-inch elevation temperature, are increasing. Immediately following nitrogen injection, however, the cladding temperatures turn over and begin to decrease. The decrease in cladding temperature is a result of an increase in the steam generation rate in the lower part of the core and a corresponding increase in the steam flow rate through the core, which is caused by a surge in the core liquid level when the nitrogen injection begins. The nitrogen injection into the cold leg, in effect, speeds up the core reflood rate. A discussion of the effect of nitrogen injection on the core hydraulic response as well as on the overall system response is presented in Section 3.2. The somewhat earlier turnover and decrease in the cladding temperature at the 9-inch elevation is a result of the initiation of core reflood, and is representative of the temperature response near the bottom of the core shortly following the first penetration of ECC into the core.

Fig. 23. Rod cladding temperature response on Rod E4 for Test S-04-6.

Fig. 24 Rod cladding temperature response on Rod G4 for Test S-04-6.

The increase in rod cladding temperatures that occur shortly after 100 seconds at the higher elevations in the core is an effect of a downcomer mass depletion process. The mass depletion begins once the downcomer fluid is brought to saturation temperature and boiling occurs. The reduction in effective downcomer head, which results from the expansion of the downcomer fluid due to boiling, causes a drop in the core liquid level, which in turn causes a reduction in the rate at which steam is being generated in the core. The reduction in steam generation reduces the steam flow through the core and the rod cladding temperatures begin to increase. The temperatures continue to increase until sufficient energy has been removed from the downcomer walls to allow the low pressure injection system flow to begin refilling the downcomer. Once the downcomer begins to refill, steam generation in the core increases and the cladding temperatures once again begin to decrease. A discussion of the Semiscale downcomer mass depletion process is presented in Section 3.3.

The effect of DNB behavior during blowdown on the core thermal response during reflood is illustrated by comparing results from Tests S-04-5 and S-04-6. The improved energy removal from the core during blowdown for Test S-04-6 (as discussed in Section 1.1) results in significantly lower rod cladding temperatures during reflood than occur in Test S-04-5. Those rods that are adjacent to unpowered rods for Test S-04-6 exhibit the largest differences in cladding temperatures when compared with rods in Test S-04-5. The cladding temperatures in the high power zone on Rod D6^[a] for the two tests are compared in Figure 25 and the temperatures on Rod D8^[b] near the same elevation are compared in Figure 26. On Rod D6 an average of between 150 and 200°F difference exists between the two tests during reflood. The difference in temperature can be attributed primarily to the delayed DNB that occurs in Test S-04-6 which removes considerably more of the stored energy of the rod than in Test S-04-5 for which an early DNB occurs. For Rod D8, however, the DNB response is the same (early DNB) for both tests. Thus, the rod cladding temperatures are similar during the early part of blowdown for both tests. Significant differences in the rod temperature do not occur until the later part of blowdown and early part of reflood when the considerably lower average core temperatures for Test S-04-6 result in better cooling characteristics in the core as compared to Test S-04-5. Figure 27 compares the average cladding temperatures as a function of elevation in the core at the time reflood is initiated for Tests S-04-5 and S-04-6. The figure shows that the average temperatures for Test S-04-6 are between 40 and 150°F lower over most of the core than for Test S-04-5.

The lower average rod cladding temperatures during the reflood phase of Test S-04-6 result in considerably shorter times to quench than occur in Test S-04-5. Figures 28 and 29 show the quench time versus elevation in the core for Tests S-04-5 and S-04-6, respectively. Each point, with the exception of the points at 297 seconds, represent the quench time for a single thermocouple location. The points at 297 seconds, however, are an indication that quenching did not occur prior to 300 seconds for a given thermocouple location, and can be representative of more than one thermocouple location. For Test S-04-6 the quench time as

[a] Rod D6 is directly adjacent to an unpowered rod (Rod D5) for Test S-04-6.

[b] Rod D8 is not adjacent to an unpowered rod for Test S-04-6.

Fig. 25 Rod cladding temperature response on Rod D6 at the 25-inch elevation for Tests S-04-5 and S-04-6.

Fig. 26 Rod cladding temperature response on Rod D8 at the 26-inch elevation for Tests S-04-5 and S-04-6.

Fig. 27 Average rod cladding temperatures as a function of elevation at the initiation of reflood for Tests S-04-5 and S-04-6.

a function of elevation in the core occurs anywhere from 15 to 150 seconds earlier than for Test S-04-5, with the largest differences occurring between the 20- and 45-inch elevations.

3.2 Effects of Accumulator Nitrogen Injection

The effect on overall system response of accumulator nitrogen injection into the intact loop cold leg is demonstrated through analysis and comparison of results primarily from Tests S-04-4, S-04-5, and S-04-6. The magnitudes of the accumulator nitrogen flow rates for these three tests were sufficiently different to cause substantial differences in system behavior. For comparison purposes, the integrated accumulator nitrogen mass flow rates for the three tests are presented in Figure 30. The time axis in the figure has been shifted so that the total nitrogen mass injected into the system after a given period of time can be obtained directly for each test. Zero time corresponds to the time at which the accumulator nitrogen flow began for each test, or about 53 seconds for Test S-04-5, 78 seconds for Test S-04-5, and 69 seconds for Test S-04-6.

Fig. 28 Quench times as a function of elevation in the core for Test S-04-5.

Fig. 29 Quench times as a function of elevation in the core for Test S-04-6.

Fig. 30 Accumulator nitrogen integrated mass flow rate for Tests S-04-4, S-04-5, and S-04-6.

Although the system pressure response prior to the initiation of accumulator nitrogen injection into the intact loop cold leg is essentially the same for Tests S-04-4, S-04-5, and S-04-6, considerable differences in the system pressure are observed following the initiation of nitrogen flow. Figure 31 compares the pressure response in the intact loop cold leg for the three tests. As shown in the figure, the system pressures in Tests S-04-4 and S-04-6 increase rapidly after initiation of accumulator nitrogen flow, whereas for Test S-04-5 only a gradual increase in the system pressure occurs. The differences in system pressurization among the tests, however, cannot be attributed solely to the addition of the nitrogen to the system volume. Calculations of the maximum system pressure increase that could be caused by the injection of nitrogen into the system were performed for each test. Results of the calculations indicate that the nitrogen injection by itself is not sufficient to cause the pressure increases observed.

The system pressure increase which occurs after initiation of accumulator nitrogen flow is attributed primarily to the influence of the nitrogen injection on the intact loop and core hydraulics. In the intact loop, the injection of nitrogen effectively stops the rapid condensation of steam that occurs when large quantities of subcooled accumulator liquid

Fig. 31 System pressure response for Tests S-04-4, S-04-5, and S-04-6 showing effect of accumulator nitrogen injection.

are present. As a result, steam flow in the intact loop is greatly reduced, which in effect restricts one of the paths for removal of steam from the core area causing an increase in the system pressure. In the core itself, when the accumulator nitrogen flow begins, a surge in the core liquid level occurs which can lead to a rapid increase in the steam generation rate and a corresponding increase in the system pressure. The increase in steam generation in the core is dependent on the liquid inventory in the core region when the nitrogen flow begins, as well as on the rate at which nitrogen is injected.

3.2.1 Intact Loop and Core Hydraulic Response Following Nitrogen Injection. The effect of accumulator nitrogen injection on the intact loop response was determined by evaluating the loop hydraulics both prior to and after the nitrogen injection occurs. Figure 32 compares the volumetric flow rate of the steam upstream of the ECC injection location with the system pressure response for Test S-04-6. Prior to nitrogen injection, the volumetric flow rate of the steam is very high (above 1200 gpm). The high steam flow rate can be attributed to the fact that the presence of subcooled accumulator liquid causes complete condensation of the steam, which in turn results in a low pressure area near the steam-liquid interface that maintains the high steam flow rate. The fact that the steam is completely condensed when it comes in contact with the subcooled ECC is illustrated in Figure 33 which compares the fluid temperature just downstream of the ECC injection location with the fluid saturation temperature for Test S-04-6. The figure shows that prior to nitrogen injection the fluid temperature is about 100 degrees subcooled. Once the accumulator liquid flow ceases and the nitrogen flow begins, however, the interface between

Fig. 32 Comparison of the system pressure with the volumetric flow rate upstream of the ECC injection location in the intact loop for Test S-04-6.

Fig. 33 Comparison of measured fluid temperature in intact loop cold leg with the fluid saturation temperature for Test S-04-6.

the superheated steam and subcooled fluid is essentially eliminated. As a result, the condensation of steam in the loop is greatly reduced, and steam flow in the loop drops off rapidly as shown in Figure 32.

The reduction of steam flow in the intact loop in turn reduces the rate at which steam is being removed from the core. Table V compares the total steam flow rate from the core exit at 69 seconds (time at which nitrogen flow begins for Test S-04-6) with the steam flow rates at half second intervals, until the peak in system pressure occurs at about 73 seconds. At 70 seconds the total steam flow exiting the core slightly increases. The increase in steam flow corresponds to a surge in the core liquid level caused by the nitrogen injection. Between 70.5 and 72 seconds, the rapid condensation of steam in the intact loop cold leg has been greatly reduced, and the total steam flow exiting the core drops by about 30% which causes a significant increase in the system pressure. Between 72 and 73 seconds, however, the system pressure continues to increase even though the total steam flow exiting the core and the steam flow in the intact loop are also on the increase. This pressure increase (attributed to a rapid increase in the steam generation rate in the core) is caused by the surge in core liquid level which occurs shortly after the nitrogen flow begins. The effect of the core liquid level surge on the system pressure is discussed in following paragraphs.

The effect of nitrogen injection on the core hydraulic behavior is evaluated by comparing changes in core liquid levels that occur when nitrogen injection begins. The injection of nitrogen into the system causes a surge in the system liquid which can result in the addition of considerable fluid to the core volume. Figure 34 shows the collapsed liquid level in the core for Test S-04-5 obtained from a lower plenum-to-upper plenum differential pressure measurement. The figure indicates a substantial increase in the core liquid level immediately following nitrogen injection. The oscillations in the core collapsed liquid level shown in the figure are a result of steam generation in the core. The relative magnitude of the oscillations is indicative of the magnitude of the rate at which steam is being generated in the core, with large oscillations corresponding to a high rate of steam generation.

Similar increases in the core collapsed liquid level to those shown in Figure 34 are observed for other tests in the baseline ECC test series when nitrogen injection begins. The relative magnitudes of the increases in core liquid level, however, are considerably different for each test and depend mainly on the rate of nitrogen injection. In tests with a relatively high rate of nitrogen injection (Tests S-04-2, S-04-3, and S-04-4) the maximum increases in core liquid level and most violent liquid level oscillations are observed. For these tests, the initial surge in the core liquid level is sufficient to cover almost the entire heated length of the core. As a result, the steam generation rate increases very rapidly and a corresponding rapid increase in the system pressure occurs. In tests with a comparatively low rate of nitrogen injection, the core liquid level surge is relatively small (for example, Tests S-04-1, S-04-5, and S-04-6) and a correspondingly smaller increase in the system pressure occurs.

Although the relative magnitudes of the pressure increases due separately to the reduction of steam flow in the intact loop and to the increase in steam generation in the core for the baseline ECC tests cannot be accurately determined due to the coupled nature of the phenomena; results from Test S-29-1^[13] can be used to quantify the increase in

TABLE V

RELATIVE STEAM VOLUMETRIC FLOW RATES AT THE CORE EXIT FOR TEST S-04-6

Time (sec)	69.0	69.5	70.0	70.5	71.0	71.5	72.0	72.5	73.0
Intact Loop Hot Leg Volumetric Flow Rate (gpm)	900	900	950	800	750	600	450	900	1000
Broken Loop Hot Leg Volumetric Flow Rate (gpm)	280	280	290	300	310	340	390	460	500
System Pressure (psia)	44	44	45	47	48	55	65	83	96
Total Steam Flow Exiting Core (gpm)	1180	1180	1240	1100	1060	940	840	1360	1500
% of Total Steam Flow at 69 Seconds	100	100	105	93	90	80	71	115	127

Fig. 34 Core collapsed liquid level for Test S-04-5.

system pressure that can occur due solely to the reduction of steam flow in the intact loop. In Test S-29-1 penetration of ECC to the lower plenum does not occur until nitrogen injection begins. As a result, core reflood does not begin until well after the initiation of nitrogen injection so that an increase in the steam generation rate in the core can be eliminated as a factor for causing an increase in the system pressure. As in the baseline ECC tests, the accumulator nitrogen flow into the system for Test S-29-1 is not sufficient by itself to cause the pressure increase observed. Figure 35 compares the system pressure increase with the intact loop volumetric flow upstream of the accumulator injection location, and shows that the pressure increases rapidly from about 50 psia to about 90 psia as the flow in the intact loop falls towards zero. The 40-psi pressure increase that occurs can be attributed directly to the effect of stopping the intact loop steam flow. However, the pressure increases resulting from the decrease in steam flow in the intact loop for Tests S-04-1 through S-04-6 are expected to be considerably less than the 40-psi increase observed in Test S-29-1, primarily because in Test S-29-1 the vessel inlet side break nozzle area was half that for the baseline ECC tests (0.00131 ft^2 compared with 0.00262 ft^2). As a result, the system pressure could increase to a considerably higher value for Test S-29-1 because of the increased restriction to flow at the break nozzle.

3.2.2 Effect of Core Steam Generation on the System Pressure Following Nitrogen Injection. The differences in the magnitude of the pressure increases observed in Tests S-04-4, S-04-5, and S-04-6 (Figure 31) are attributed primarily to the differences in the steam generation rates in the core, which in turn correspond to the core liquid level response when nitrogen injection begins. The core cladding temperatures for these tests indicate the

Fig. 35 Comparison of the system pressure with the volumetric flow rate upstream of the ECC injection location in the intact loop for Test S-29-1.

magnitude of the increase in steam generation in the core when nitrogen flow begins. Figures 36, 37, and 38 compare the rod cladding temperatures on Rod F2 at the 7-inch elevation, on Rod E3 at the 20-inch elevation and on Rod E4 at the 28-inch elevation for the three tests. These cladding temperatures are typical of the measurements obtained at or near the three elevations. As shown in these figures, the maximum increase in steam generation rate is indicated in Test S-04-4. The cladding temperatures for Test S-04-4 show that shortly after nitrogen injection begins, the 7-inch and 20-inch elevations quench whereas the 28-inch elevation indicates a significant increase in the heat transfer from the rod surface. Each of these phenomena is indicative of a rapid increase in the rate of steam generation, and correspond to the substantially larger core liquid level swell than occurred for either Test S-04-5 or Test S-04-6. The core collapsed liquid level for Test S-04-4 is shown in Figure 39 and can be compared with the collapsed liquid level for Test S-04-5 shown in Figure 34.

The rod cladding temperatures for Tests S-04-5 and S-04-6 show significantly smaller decreases when nitrogen injection begins, and do not show the lower core rod surface quenching which occurs in Test S-04-4. However, the pressure increase for Test S-04-6 is about 30 psi larger than for Test S-04-5, even though the nitrogen injection rates were essentially the same for both tests. The differences in pressure response for these two tests, as in the other tests in the series, are directly attributable to differences in the rate of steam generation in the core. Thus, a substantial difference in the core hydraulic behavior following nitrogen injection must have occurred between Tests S-04-5 and S-04-6. Since the lower plenum-to-upper plenum differential pressure measurement failed for Test S-04-6

Fig. 36 Rod cladding temperature response on Rod F2 at the 7-inch elevation for Tests S-04-4, S-04-5, and S-04-6.

Fig. 37 Rod cladding temperature response on Rod E3 at the 20-inch elevation for Tests S-04-4, S-04-5, and S-04-6.

Fig. 38 Rod cladding temperature response on Rod E4 at the 28-inch elevation for Tests S-04-4, S-04-5, and S-04-6.

Fig. 39 Core collapsed liquid level for Test S-04-4.

(thus preventing determination of the core collapsed liquid level), the rod cladding temperatures and steam flow in the broken loop hot leg are used to illustrate the differences in the steam generation rate in the core for the two tests. Figure 40 compares the volumetric flow rates in the broken loop hot leg for the tests and shows a substantially larger increase in the volumetric flow rate following nitrogen injection for Tests S-04-6 than for Test S-04-5. This difference in flow rate is consistent with somewhat larger rod cladding temperature drops for Test S-04-6 of about 20 to 40°F as compared with about 5 to 10°F for Test S-04-5 (shown in Figures 36 through 38) and is also consistent with the premise of a higher rate of steam generation for Test S-04-6.

3.2.3 Long Term System Pressure Response Following Nitrogen Injection. The long term system pressure following nitrogen injection for most baseline ECC tests remains at a considerably higher level than just prior to the nitrogen injection. Figure 41 compares the system pressure response for Test S-04-6 with the fluid density near the break on the inlet side of the vessel. As shown in the figure, once the fluid plug in the inlet side of the broken loop is blown from the system, the pressure begins to drop. However, the system pressure falls to a value that is about 30 psi higher than it was prior to nitrogen injection and then again it begins to increase as the reflood progresses.

The primary reasons for the pressure remaining high for most of the reflood phase of Test S-04-6 (as well as other baseline ECC tests) are first, the core reflooding and the corresponding high rate of steam generation provide the source of high pressure, and second,

Fig. 40 Volumetric flow in the vessel outlet side of the broken loop for Tests S-04-5 and S-04-6.

Fig. 41 Comparison of the system pressure with the fluid density near the break on the vessel inlet side for Test S-04-6.

the flow at the vessel inlet side break nozzle is apparently choked during most of the reflood phase. A calculation based on the measured pressures, temperatures, and fluid densities, and using the homogenous equilibrium break flow model^[11], was performed to determine the choked flow rate corresponding to the measured system conditions. The measured break flow rate was then compared to the calculated choked flow rate, and results indicate that choked flow exists during most of the reflood phase of the test. The results of the calculation are, however, subject to considerable uncertainty, because the measured pressures and fluid densities used in the calculation were obtained near the lower limits of the measurement ranges, and are thus subject to considerable instrument error.

After about 100 seconds, the system pressure once again begins to decrease as shown in Figure 41. The decrease in pressure at this time is caused by a reduction of the effective downcomer head which in turn causes a drop in the core liquid level and a corresponding reduction in the rate of steam generation. The mechanism for the downcomer head reduction is discussed in Section 3.3.

3.3 Downcomer Mass Depletion Phenomena

Baseline ECC tests exhibit a downcomer mass depletion phenomena which occurs during a portion of the reflood phase of each test. The downcomer mass depletion occurs only after the intact loop accumulator liquid flow ceases and is a result of heat transfer from the downcomer walls which raises the fluid temperature to the saturation temperature. The resulting boiling of the fluid forces liquid from the downcomer and reduces the effective

downcomer head. Figure 42 illustrates the downcomer collapsed liquid level obtained from a differential pressure measurement for Test S-04-6. As shown in the figure, the collapsed liquid level begins to decrease rapidly after about 90 seconds. This time corresponds directly to the time at which the fluid in the downcomer reaches saturation (Figures 14 through 18). The decrease in downcomer head, in turn, results in a drop in the core liquid level (Figures 34 and 39) which causes the turnaround and eventual increase in core cladding temperatures discussed in Section 3.1. After about 125 seconds, the collapsed liquid level begins to increase once again under the influence of the low pressure injection system flow.

Fig. 42 Downcomer collapsed liquid level for Test S-04-6.

To determine whether the downcomer mass depletion phenomena observed in the baseline ECC tests is unique to the Semiscale system, or whether it is also representative of large system behavior, an evaluation of the heat transfer in the Semiscale downcomer has been performed. Since the Semiscale downcomer surface area to volume ratio is about 24 times larger than the same ratio for a PWR, the heat flux obtained in the Semiscale system during refill and reflood must be reduced by a factor of 24 to provide a total heat transfer per unit volume of downcomer fluid that is comparable to that of a PWR. To this end, the Semiscale Mod-1 downcomer design incorporates a steam gap insulator on both the vessel and core barrel sides of the downcomer. Figure 43 shows a schematic of the downcomer insulator arrangement in the Semiscale Mod-1 configuration. For reference purposes the relative locations of the metal and fluid thermocouples are shown^[a].

- [a] All metal temperatures in the Semiscale downcomer with the exception of downcomer filler insulator (TIV-FO) thermocouples, are located 1/16 of an inch below the metal surface. The downcomer filler insulator thermocouples are welded to the surface of the downcomer filler insulator.

Fig. 43 Semiscale Mod-1 downcomer insulator configuration.

To effectively reduce the downcomer heat transfer per unit volume of fluid to a value typical of that of a PWR, the Semiscale downcomer insulator gaps must remain filled with steam during the entire reflood portion of a test. However, numerous penetrations exist in both the vessel filler insulator and in the core barrel insulator which can allow penetration of ECC fluid into the steam gaps once the downcomer becomes filled with liquid. These penetrations include instrument penetrations in the vessel filler insulator, as well as penetrations due to fitting tolerances between the filler insulator and core barrel insulator and their corresponding support structures.

Comparisons of the metal and fluid temperatures at various elevations in the downcomer for Test S-04-6 indicate that prior to the time the downcomer becomes filled with ECC, the steam gaps function as expected. Figures 44 through 46 compare the available metal and fluid temperature measurements at the 35-, 115-, and 156-inch elevations below the cold leg centerline on the vessel side of the downcomer for Test S-04-6 and Figures 47 and 48 compare similar temperature measurements at the 70- and 115-inch elevation on the core barrel side for the same test. On both the vessel side and core barrel

Fig. 44 Metal and fluid temperatures on the vessel side of the downcomer at the 35-inch elevation for Test S-04-6.

Fig. 45 Metal and fluid temperatures on the vessel side of the downcomer at the 115-inch elevation for Test S-04-6.

Fig. 46 Metal and fluid temperatures on the vessel side of the downcomer at the 156-inch elevation for Test S-04-6.

Fig. 47 Metal and fluid temperatures on the core barrel side of the downcomer at the 70-inch elevation for Test S-04-6.

Fig. 48 Metal and fluid temperatures on the core barrel side of the downcomer at the 115-inch elevation for Test S-04-6.

side, with the exception of the 156-inch elevation measurements^[a], all the downcomer filler temperatures (TMV-FI) and the core barrel temperatures (TMV-CO) remain high and fairly constant until about 50 seconds. These measurements provide an indication that until 50 seconds, steam exists in the gap and the insulators are working as expected. However, once subcooled ECC fluid penetrates into the downcomer, the downcomer filler temperatures (TMV-FI) (with the exception of the 115-inch elevation measurement) and the core barrel temperatures (TMV-CO) begin to drop rapidly as shown in the figures. The rapid drop in these temperatures can be attributed to a considerable increase in the heat transfer across the insulator gaps, and can occur only if a significant amount of ECC penetrates into the gaps. The ECC fluid apparently penetrates the insulator gaps as a result of a reduction of the gap pressure below the downcomer pressure caused by condensation of the gap steam. The steam in the insulator gaps condenses because of rapid cooling of the vessel filler insulator and core barrel insulator metal caused by the subcooled ECC fluid in the downcomer. The metal temperatures in the downcomer indicate that between about 50 and 80 seconds the insulator gaps (especially on the core barrel side) fail to function as required because of liquid in the gaps.

[a] The filler insulator temperature (TMV-FI) at the 156-inch elevation tends to follow the fluid saturation temperature instead of remaining at a relatively high value as observed at other elevations. This behavior indicates that fluid exists in the gap at the 156-inch elevation during the entire blowdown and refill sequence. Since the downcomer insulator gap opens into the lower plenum to allow the gap fluid to blowdown after rupture, a path for fluid to reenter the gap exists.

After the downcomer fluid becomes saturated (at about 85 seconds for Test S-04-6), the heat transfer across the downcomer filler insulator and core barrel insulator is reduced sufficiently to allow partial vaporization of liquid in the insulator gaps, thus reestablishing the insulator effectiveness. The temperatures of the downcomer filler metal (TMV-FI) at the 35- and 115-inch elevations (Figures 44 and 48) indicate that between about 85 and 125 seconds steam once again exists in the insulator gaps. As a result, after 125 seconds the heat transfer across the insulator gaps is no longer sufficient to cause boiling of the downcomer fluid, and the downcomer begins to refill under the influence of the low pressure injection system flow (Figure 42).

To determine the magnitude of the change in downcomer surface heat flux which occurs when liquid is present in the insulator gaps, separate calculations of the heat flux were made with both liquid and steam represented in the gap. The calculations were performed with a 1-D program that solves the heat conduction equations by a finite difference approximation procedure. The program allows the use of time dependent heat transfer coefficients and fluid temperatures. The lack of adequate thermocouple measurements on the vessel filler and core barrel insulators^[a] prevents an accurate determination of the actual heat flux for these surfaces on the basis of the measured metal temperatures. Thus, the calculations performed were based on the measured downcomer fluid temperatures. McAdams nucleate boiling heat transfer correlation^[14] was used for obtaining the heat transfer coefficient for both the downcomer surfaces.

Results of the calculations are presented in Figures 49 and 50. The figures illustrate the surface heat fluxes on both the vessel and core barrel sides of the downcomer for the cases with liquid and with steam in the gaps. For comparison purposes results from calculations for the heat flux from a PWR core barrel and vessel are included. The various system parameters and fluid temperature assumptions used in the calculations are listed in Table VI.

As shown in the figures, addition of liquid to the insulator gap can result in an increase in the surface heat flux by a factor of between 10 and 15 for both the vessel side and core barrel side downcomer surfaces. Thus, with liquid in the insulator gaps the surface heat fluxes in the Semiscale downcomer between 50 and 80 seconds would be only a factor of between 2 and 5 lower than the expected heat flux in a PWR, whereas a factor of at least 24 lower is required for typicality. Since for most of the time between 50 and 80 seconds the temperature measurements obtained in the Semiscale downcomer indicate liquid in the insulator gaps, the downcomer surface heat flux to unit volume of downcomer liquid is not typical of that expected in a PWR. As a result, the mass depletion process in the Semiscale downcomer, which is a direct function of the rate at which heat is transferred from the downcomer walls to the fluid, is a much more severe process in the Semiscale geometry than would be expected in a PWR.

[a] The thermocouple TIV-FO, although on the vessel filler insulator, cannot be used for a heat flux determination once liquid penetrates the filler gap because the thermocouple follows the liquid temperature rather than the metal temperature.

Fig. 49 Surface heat fluxes from the core barrel side of the downcomer for Semiscale and a PWR.

Fig. 50 Surface heat fluxes from the vessel side of the downcomer for Semiscale and a PWR.

TABLE VI
CONDITIONS FOR HEAT CONDUCTION CALCULATIONS

Item	Material	Thickness (inches)	Boundary Conditions (Btu/hr ⁰ F-ft ²)
PWR Vessel	Carbon Steel	8.5	$h = 0.074 (T_w - T_f)^{2.86}$ on Downcomer Side, Outside Insulated
PWR Core ^[a] Barrel	Stainless Steel	2.25	$h = 0.074 (T_w - T_f)^{2.86}$ Both Sides
Semiscale Vessel	Carbon Steel	1.083	Outside Insulated
Vessel Filler	Stainless Steel	0.723	
Insulator Gap		0.050	
Downcomer Wall	Stainless Steel	0.125	$h = 0.074 (T_w - T_f)^{2.86}$
Semiscale Core Barrel ^[b]	Beryllium Copper	0.615	$h = 0.074 (T_w - T_f)^{2.86}$
Insulator Gap		0.050	Both Sides
Downcomer Wall	Stainless Steel	0.125	

Initial temperature at steady state = 550°F for all cases

Downcomer fluid temperature - $T_{df} = (550 - 8.3t)^\circ\text{F}$ for $t \leq 30$ seconds.

$T_{df} = 300^\circ\text{F}$ for $t > 30$ seconds.

[a] PWR core fluid temperature same as downcomer fluid temperature.

[b] Semiscale core fluid temperature

$T_{cf} = 550^\circ\text{F}$ for $t \leq 60$ seconds

$T_{cf} = [550 - 13.3(t-60)]$ for $60 < t \leq 75$ seconds

$T_{cf} = 350^\circ\text{F}$ for $t > 75$ seconds.

To determine the approximate time after refill at which the downcomer fluid in a PWR might become saturated, a calculation was performed on the basis of surface heat fluxes for the PWR core barrel and vessel downcomer surfaces shown in Figures 49 and 50. The assumption was made that once the PWR downcomer becomes filled no additional subcooled fluid is allowed to penetrate the downcomer. In addition, only heat transfer from the downcomer walls was considered in determining the heatup rate of the downcomer fluid. The heat flux used in the calculation is a time average (between 35 and 275 seconds) of both the vessel and core barrel side downcomer surface heat fluxes. An average initial

downcomer fluid subcooling of 90°F was assumed. The 90°F subcooling is based on the average fluid temperature in the Semiscale downcomer immediately following downcomer refill (Figures 14 through 18):

On the basis of the preceding assumptions, about 375 seconds would be required to raise the downcomer fluid temperature in a PWR to saturation (at a containment pressure of 35 psia). The 375 seconds for the PWR case compares with about 35 seconds to raise the downcomer fluid temperature in the Semiscale system to saturation once downcomer refill occurs.

IV. CONCLUSIONS

The thermal-hydraulic response of the Semiscale Mod-1 system during the baseline ECC test series has been investigated to determine the effect of unpowered rods on the DNB behavior of adjacent rods, and to determine the effects of the downcomer countercurrent flow and hot wall phenomena, the accumulator nitrogen injection, and the downcomer mass depletion process on the system and core thermal-hydraulic response. The results from these tests have led to an increased understanding of the phenomena that occur in the Semiscale system during integral blowdown-reflood tests with an electrically heated core.

The following is a summary of the conclusions reached from the analyses of the data obtained during the Semiscale Mod-1 baseline ECC test series.

1. CORE THERMAL RESPONSE EARLY IN BLOWDOWN

The effect of unpowering a rod in the Semiscale core is to shift the DNB behavior of adjacent rods from early DNB to early DNB with rewet or delayed DNB, or from early DNB with rewet to delayed DNB. The DNB behavior of rods adjacent to an unpowered rod results in a considerably higher rate of removal of the rod stored energy during the early stage of blowdown than would occur if the neighboring rod was not unpowered. A reduction of the maximum rod cladding temperatures obtained during blowdown of as high as 350°F can occur on rods adjacent to an unpowered rod.

2. DOWNCOMER RESPONSE DURING REFILL

The delivery of ECC fluid to the lower plenum during the Semiscale Mod-1 baseline ECC tests is delayed by the downcomer countercurrent flow and hot wall effects. A total delay of about 37 seconds occurs between the time the intact loop accumulator liquid injection begins and the time ECC penetrates to the vessel lower plenum. The first 26 seconds of the delay can be attributed primarily to the downcomer countercurrent flow phenomena and the remaining 11 seconds can be attributed to the Semiscale hot wall effect. The delays observed during the baseline ECC tests are consistent with those obtained from previous blowdown-reflood tests in the Semiscale system.

The degree of superheat in the steam within the core during refill was found to influence the downcomer countercurrent flow response. A high degree of superheat in the core steam causes the system to depressurize at a somewhat slower rate than would occur if a lesser degree of superheat existed in the core steam. The slower depressurization rate, in turn, results in a longer period of time during which the countercurrent flow effectively restricts ECC penetration of the downcomer.

3. SYSTEM AND CORE RESPONSE DURING REFLOOD

The overall system response, as well as the core thermal-hydraulic response, is affected by several events which occur in the Semiscale Mod-1 system during reflood. These events include the injection of accumulator nitrogen into the intact loop cold leg once the liquid supply is depleted and a downcomer mass depletion process which occurs once the downcomer fluid temperature is raised to the saturation temperature. In addition, the DNB response of the core during blowdown strongly influences the rod cladding temperatures during reflood and thus affects the rate at which rod quenching occurs.

The injection of accumulator nitrogen into the intact loop cold leg, once the liquid supply is depleted, can cause a rapid increase in the system pressure. However, the increase in system pressure cannot be attributed solely to the addition of the nitrogen to the system volume, but can be attributed primarily to the effect of the nitrogen injection on the intact loop and core hydraulics. In the intact loop, the injection of nitrogen effectively stops the rapid condensation of steam which occurs prior to nitrogen injection, thus greatly reducing the steam flow rate through the intact loop. As a result, the rate at which steam is removed from the core is reduced and an increase in the system pressure occurs. In the core itself, when the nitrogen flow begins, a surge in the core liquid level occurs which can lead to a rapid increase in the steam generation rate and a corresponding increase in the system pressure. The system pressure continues to increase as a result of steam generation in the core until water is blown clear of the vessel inlet side of the broken loop. Once water clears the break the system pressure begins to decrease. However, the pressure remains at a significantly higher value than prior to the injection of accumulator nitrogen for most of the reflood portion of the test. This higher pressure can apparently be attributed to choked flow at the vessel inlet side break nozzle combined with the high rate of steam generation in the core.

In the Semiscale system, downcomer mass depletion occurs during reflood as a result of heat transfer from the downcomer walls which causes the fluid to boil. The boiling of the fluid forces liquid from the downcomer thus reducing the effective downcomer head. The decrease in downcomer head, in turn, causes a reduction of the core liquid level and an eventual increase in the rod cladding temperatures. The downcomer head does not begin to increase again until a partial regeneration of steam in the downcomer insulator gaps reduces the heat transfer across the gaps, thus allowing the low pressure injection system flow to begin to refill the downcomer.

An evaluation of the downcomer thermal response during the period when the mass depletion occurs indicates that heat transfer per unit volume of downcomer fluid from the downcomer walls in the Semiscale system is much higher (by a factor of between 5 and 12) than would occur in a typical PWR system. Thus, the downcomer mass depletion phenomenon observed in the Semiscale system is thought to be nontypical of large system behavior.

The DNB response of the core during blowdown significantly affects the rate at which stored energy is removed from the rods and thus influences the temperatures of the rods during reflood. For tests in which delayed DNB or early DNB with rewet predominates, average rod cladding temperatures at the initiation of reflood are considerably lower than for tests in which early DNB predominates. As a result, rod quenching occurs earlier in the reflood sequence.

V. REFERENCES

1. *Quarterly Technical Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, January – March 1975*, ANCR-1254 (September 1975).
2. E. M. Feldman and D. J. Olson, *Semiscale Mod-1 Program and System Description for the Blowdown Heat Transfer Tests (Series 2)*, ANCR-1230 (August 1975).
3. T. K. Larson, *Core Thermal Response During Semiscale Mod-1 Blowdown Heat Transfer Tests*, ANCR-NUREG-1285 (June 1976).
4. Proposed ANS Standard, *Decay Energy Release Following Shutdown of Uranium-Fueled Thermal Reactors*, ANS-51 (October 1971).
5. H. S. Crapo, B. L. Collins, K. E. Sackett, *Experiment Data Report for Semiscale Mod-1 Test S-04-1 (Baseline ECC Test)*, ANCR-NUREG-1331 (September 1976).
6. H. S. Crapo, B. L. Collins, K. E. Sackett, *Experiment Data Report for Semiscale Mod-1 Test S-04-2 (Baseline ECC Test)*, ANCR-NUREG-1331 (September 1976).
7. H. S. Crapo, B. L. Collins, K. E. Sackett, *Experiment Data Report for Semiscale Mod-1 Test S-04-3 (Baseline ECC Test)*, TREE-NUREG-1002 (October 1976).
8. H. S. Crapo, B. L. Collins, K. E. Sackett, *Experiment Data Report for Semiscale Mod-1 Test S-04-4 (Baseline ECC Test)*, TREE-NUREG-1003 (October 1976).
9. H. S. Crapo, B. L. Collins, K. E. Sackett, *Experiment Data Report for Semiscale Mod-1 Tests S-04-5 and S-04-6 (Baseline ECC Tests)*, TREE-NUREG-1045 (January 1977).
10. D. J. Hanson, R. T. French, C. J. Shaffer, *An Evaluation of ECC-Related Phenomena During Semiscale Mod-1 Isothermal Tests*, ANCR-1239 (November 1975).
11. K. V. Moore and W. H. Rettig, *RELAP4 – A Computer Program for Transient Thermal-Hydraulic Analysis*, ANCR-1127, Rev. 1 (March 1975).
12. C. L. Wheeler, C. W. Steward, R. J. Cena, D. S. Rowe, A. M. Sutey, *CØBRA-IV-I: An Interim Version of CØBRA for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores*, BNWL-1962 (March 1976).
13. H. S. Crapo, M. F. Jensen, K. E. Sackett, *Experiment Data Report for Semiscale Mod-1 Test S-29-1 (Integral Test With Asymmetrical Break)*, ANCR-NUREG-1327 (July 1976).
14. W. H. McAdams, *Heat Transmission*, 3rd edition, New York: McGraw-Hill Book Company, Inc., 1954.

DISTRIBUTION RECORD FOR TREE-NUREG-1077

Internal Distribution

- 1 - Chicago Patent Group - ERDA
9800 South Cass Avenue
Argonne, Illinois 60439
- 2-4 - A. T. Morphew
Classification and Technical Information Officer
Idaho Operations Office - ERDA
Idaho Falls, ID 83401
- 5 - R. J. Beers, ID
- 6 - P. E. Litteneker, ID
- 7 - R. E. Tiller, ID
- 8 - R. E. Wood, ID
- 9 - H. P. Pearson, Supervisor
Technical Information
- 10-19 - INEL Technical Library
- 20-39 - Authors
- 40-94 - Special Internal

External Distribution

- 95-96 - Saul Levine, Director
Office of Nuclear Regulatory Research, NRC
Washington, D.C. 20555
- 97-401 - Distribution under NRC-2, Water Reactor Safety Research
Systems Engineering