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ANALYSIS OF THE THERMAL-HYDRAULIC BEHAVIOR RESULTING IN EARLY CRITICAL HEAT FLUX AND EVALUATION OF CHF CORRELATIONS FOR THE SEMISCALE CORE

DALE M. SNIDER

March 1977

MASTER

EG&G Idaho, Inc.

IDAHO NATIONAL ENGINEERING LABORATORY

ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

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by

Dale M. Snider

EG&G IDAHO, INC.

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ABSTRACT

This report presents analysis of the core thermal-hydraulics associated with early critical heat flux (CHF) in the Semiscale core during the loss-of-coolant accident (LOCA) resulting from a large cold leg pipe rupture. The thermal-hydraulic behavior was solved using COBRA-IV-I with boundary conditions calculated from measured data. Analysis indicated a flow stagnation occurred in the heated core shortly after rupture and lasted until 700 ms after rupture at which time fluid above the stagnation region reversed direction and coolant flowed from the upper plenum down through the core. During the time when a flow stagnation region existed, the core rapidly voided from both ends and this time of rapid emptying corresponded to the time period of measured critical heat fluxes. Analysis indicated the quality was from 30 to 70% at the time of CHF suggesting that the mechanism for CHF was dryout of the heater rod surface. The influx of lower quality coolant into the top of the heated core after complete flow reversal reduced the quality in the heated core and produced the observed rewets at about 1.2 s in parts of the core. An evaluation to determine the capability of seven CHF correlations to calculate the time of critical heat flux in Semiscale indicated that those correlations which depended strongly on quality did well in calculating the time of CHF. Those correlations which had the core entrance enthalpy and a length term as independent variables did not adequately calculate the time of CHF.

SUMMARY

Analysis was performed to determine the hydraulics associated with CHF in the Semiscale core during a large cold leg break induced loss-of-coolant accident. The core hydraulics were solved using the COBRA-IV-I computer program with boundary conditions calculated from measured data. The hydraulics were then used to evaluate the capability of seven CHF correlations to calculate the time of CHF in Semiscale.

The core response of Test S-02-9 from the blowdown heat transfer series was analyzed in depth in this study. This test simulated a large cold leg break accident from 100% power (1.56 MW) and 15.53 MN/m² initial pressure. In this test (and other tests in the blowdown heat transfer series), early CHF was observed to happen within 700 ms after rupture and as early as 480 ms after rupture. Some locations which experienced CHF would exhibit a rewet within 2 s after rupture.

The Semiscale core was modeled as a single channel from the core barrel inlet to the upper plenum. The boundary conditions used in the COBRA-IV-I calculation included (a) system pressure, (b) pressure difference between the core bottom and core top, (c) enthalpy at the core top or bottom when flow was into the core and (d) rate of energy transfer between core and coolant. These boundary conditions were determined from measured data.

Analysis indicated the large cold leg break produced a rapid core flow reversal which extended to the upper part of the heated core where a flow stagnation occurred for about 600 ms. The flow stagnation caused the core to empty from both the top and bottom and with the high core heat transfer rate, the core was rapidly depleted of water. The period during which the core was quickly emptying corresponded to the time of measured critical heat fluxes (500 to 650 ms after rupture). The range of calculated quality for the period of CHF was from 30 to 70% which suggested the mechanism of CHF was dryout of the rod surface.

At 700 ms after rupture, fluid above the stagnation region was calculated to reverse flow direction, and coolant flowed from the upper plenum down through the core. The influx of a lower enthalpy mixture reduced the quality in the heated core and led to the measured rewets starting at about 1.2 s after rupture.

Seven CHF correlations were evaluated to determine their capability to calculate the time of CHF in the Semiscale core. Those correlations which depended strongly on quality (General Electric Company, MacBeth, Biasi, and Babcock & Wilcox Company B&W-2 correlations) did well in calculating the time of CHF. The General Electric Company correlation which is a simple linear relation based on quality best calculated the time of CHF for Semiscale. Correlations which expressed the core entrance enthalpy and a length term from the entrance to the point of CHF as independent variables (Westinghouse W-3 and Barnett correlations) did not adequately calculate CHF for Semiscale. The Loss-of-Fluid Test CHF correlation did not calculate a CHF for Semiscale.

NOMENCLATURE

A	flow area
D_e	hydraulic diameter
D_h	heated diameter
D_r	rod diameter
G	mass flux
h	enthalpy
h_f	enthalpy of fluid
h_{fg}	heat of vaporization
h_i	inlet enthalpy
h_{sat}	saturated fluid enthalpy
p	pressure
q	heat flux
q_{CHF}	critical heat flux
x	quality
z	axial distance
ρ_f	density of fluid
ρ_g	density of gas

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ANALYSIS OF THE THERMAL-HYDRAULIC BEHAVIOR RESULTING IN EARLY CRITICAL HEAT FLUX AND EVALUATION OF CHF CORRELATIONS FOR THE SEMISCALE CORE

I. INTRODUCTION

The Semiscale Mod-1 experimental program conducted at the Idaho National Engineering Laboratory by EG&G Idaho, Inc., is part of the overall U.S. Nuclear Regulatory Commission and Energy Research and Development Administration sponsored research and development program to investigate the behavior of a 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)[1, a] program, in optimizing test series, selecting test parameter, 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 test by a core composed of an array of electrically heated rods. The core is contained in a pressure vessel which also includes a downcomer, lower plenum, and upper plenum. The system piping is arranged in a 1-1/2-loop configuration with the intact loop containing an active steam generator and pump, and with the broken loop containing passive simulators for the steam generator and pump.

A series of blowdown heat transfer tests was conducted in the Semiscale Mod-1 program to obtain information required to evaluate the heat transfer characteristics of the Semiscale Mod-1 core and to obtain the information necessary for evaluating the analytical models currently used to calculate core flow and heat transfer coefficients.

Behavior observed in the blowdown heat transfer test series which required detailed analysis was the time that a boiling crisis occurred. In the double-ended offset shear cold leg break tests, critical heat flux (CHF) was generally observed to occur in parts of the core within 800 ms after rupture and in some cases as early as 480 ms after rupture[2]. In many cases, those locations which experienced CHF were subsequently rewet within 1.5 s after rupture. The double-ended hot leg break test and the small cold leg break test of the blowdown heat transfer test series did not exhibit the early times to CHF.

[a] Loss-of-Fluid Test Facility — a 55-MW test reactor located at the Idaho National Engineering Laboratory.

The blowdown heat transfer test series, which was the second series conducted in the Semiscale Mod-1 system, consisted of one test initiated by a double-ended offset shear hot leg break, eight tests initiated by a double-ended offset shear cold leg break, and one test initiated by a single-ended "small" cold leg break. Two of these tests were conducted as standard problem tests^[a] for the U.S. Nuclear Regulatory Commission. Uninterpreted data for the blowdown heat transfer tests are available in References 3 through 11. A general discussion of the system behavior during the blowdown heat transfer series is given in Reference 12 and a general discussion of the core thermal-hydraulic response is given in Reference 2.

This report presents results from an analytical study to determine the thermal-hydraulic phenomena associated with the early CHF, and to evaluate the calculational capability of common critical heat flux correlations for use with respect to the Semiscale core. In this study hydraulic conditions were calculated with the COBRA-IV-I^[13] core thermal-hydraulic code using boundary conditions and core energy transfer rates determined from measured data. A single channel model for the Semiscale core was used in this study. Details of the COBRA-IV-I model, the boundary conditions, and the results of the COBRA-IV-I calculation are presented.

The critical heat flux correlations used in blowdown analysis have been derived using data from experiments at close to steady state conditions. By evaluating the capability of common CHF correlations to calculate a boiling crisis in the Semiscale core, a broader understanding is obtained for using these correlations in other systems during blowdown conditions. The capability of the correlations was evaluated by comparing the rod surface heat flux at the time of the boiling crisis with the critical heat flux calculated from the CHF correlations. The hydraulic conditions used in the CHF correlations for these calculations were derived from the COBRA-IV-I calculations. The rod surface heat flux was calculated from measured test data using an inversion technique described by Beck^[14].

Section II of this report contains a description of the Semiscale Mod-1 system and the test plan followed for the blowdown heat transfer test series. The COBRA-IV-I model, a description of the boundary conditions used in COBRA-IV-I, the COBRA-IV-I solution scheme, and a comparison between the COBRA-IV-I calculations and the measured hydraulic data are presented in Section III. Section IV contains a discussion of the calculated core hydraulic behavior, and a discussion of the core thermal response before and after CHF. Section V contains a discussion of the capability of the critical heat flux correlations evaluated in this study. Conclusions from this study are given in Section VI. The appendixes contain input conditions to the COBRA-IV-I program, results from a COBRA-IV-I sensitivity study, and descriptions of the critical heat flux correlations used in this study.

[a] A Nuclear Regulatory Commission standard problem test is a test conducted to establish an experimental data base for the purpose of comparison with unbiased analytical calculations supplied by the standard problem program participants.

II. EXPERIMENT DESCRIPTION

The Semiscale Mod-1 experimental system and test procedure for the blowdown heat transfer test series are described in the following subsections. The topics discussed include the overall system configuration, and the test conditions and procedures.

1. SYSTEM DESCRIPTION

The Semiscale Mod-1 experimental apparatus shown in Figure 1 is a small-scale model of LOFT which is in turn a 1-1/2-loop model of a four-loop PWR. The Semiscale system, like the LOFT system, is used to obtain transient thermal-hydraulic data under simulated LOCA conditions. The Semiscale system, which has a liquid volume of approximately 0.221 m^3 (7.8 ft^3), consists of a pressure vessel and internals; an intact loop consisting of a pressurizer, steam generator, and coolant pump; and a broken loop with hydraulic resistances that simulate an inactive steam generator and a locked rotor pump. The intact loop of the Semiscale system models three loops of a typical four-loop PWR. The broken loop simulates the fourth PWR loop. A detailed description of the system configuration can be found in Reference 15.

1.1 Mod-1 Core and Heater Rod

The Mod-1 core in the Semiscale system consists of 40 electrically heated rods of typical PWR fuel rod diameter [1.072 cm (0.422 in.)]. Figure 2 illustrates the general construction of a Semiscale heater rod. The rod heating element is constructed of constantan wire (55% copper, 45% nickel) coiled with a varying pitch, and sized to give a specified power. Compacted boron nitride surrounds and insulates the element from a composite sheath. Four Chromel-Alumel thermocouples swaged between inner and outer claddings provide rod cladding temperature measurements at different axial locations along the 168-cm (5.5-ft) heated length of the heater rods. The overall length of the rod is about 525.8 cm (207 in.) and the bottom of the heated length is about 363 cm (143 in.) below the vessel cold leg center line as shown in Figure 3. The rods extend from the bottom of the heated section to the upper plenum and pass out through the vessel upper head. The axial power distribution is illustrated in Figure 4. The 40 rods are positioned and held in the core with ten grid spacers which maintain the heaters on a typical PWR pitch [1.43 cm (0.563 in.)].

1.2 Core Measurements

Throughout this report, the heat flux data are referenced to the location of the thermocouple from which the heat flux was derived. Figure 5 shows the elevation and orientation of thermocouples with respect to the heater rod matrix. The thermocouples 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, and by their elevation above the bottom of the heated length of the core. The thermocouple on Rod D5 at 74 cm (29 in.) from the bottom of the heated core is thus referred to as TH-D5-29 where TH implies a core heater cladding temperature, D5 refers to the rod in which the thermocouple is located, and 29 gives the thermocouple elevation in inches above the core bottom.

Fig. 2 Semiscale Mod-1 electric heater rod.

Fig. 3 Active heater rod axial location relative to cold leg centerline.

Fig. 4 Semiscale Mod-1 axial power profile.

gamma attenuation device. The flow measurements were obtained from both a turbine flowmeter and a drag disk flowmeter used in conjunction with the fluid density measurement. Fluid temperature was measured using a Chromel-Alumel thermocouple or platinum resistance bulb temperature detector. A strain gage diaphragm transducer connected between the upper and lower plenums by water filled sense lines was used to measure the upper-to-lower-plenum differential pressure. A Hall-effect multiplier device measured total core power; a digital voltmeter measured core voltage; and a shunt circuit was used to obtain core current.

Radial Locations Viewed from the Top

EG&G-A-1190

Fig. 5 Thermocouple locations in Semiscale Mod-1 heated core.

2. TEST CONDITIONS AND PROCEDURE

Experimental data from Test S-02-9 were used for analysis in this study. Test S-02-9 was a simulated offset shear (200%) cold leg break test. This test had a flat radial power profile (some Semiscale tests had a center-peaked radial power profile) which could better be modeled by a single-channel hydraulic solution. For this test the core contained two failed heater rods, 37 powered heater rods, and one liquid level probe. Figure 5 showed the location of the two failed heaters and the liquid level probe.

In preparation for Test S-02-9, the system was filled with treated demineralized water and vented to assure a liquid full system. The system was heated to initial conditions in 56 K (100°F) temperature intervals during warmup to evaluate the operability of the measurement system. Prior to blowdown, the initial conditions were held constant for 10 to 20 minutes in order to establish system equilibrium. The rupture assemblies were then actuated to produce a simulated double-ended pipe break. Digitized data were recorded at approximately 17-ms intervals. A more detailed description of the operating procedure for Test S-02-9 is given in Reference 11. Test S-02-9 initial conditions are given in Table I.

TABLE I
TEST S-02-9 INITIAL CONDITIONS

Break	200% cold leg break. (A 200% break is a full size double-ended offset shear)	
Core power	1.56 MW	
Core inlet temperature	557 K	(542°F)
Core temperature rise	38 K	(68°F)
System pressure	15.53 MN/m ²	(2253 psia)
Peak power density	38.9 kW/m	(11.84 kW/ft)

III. COBRA-IV-I CALCULATION

The COBRA-IV-I^[a] model for the Semiscale core, a description of the COBRA-IV-I boundary conditions, the method of solution, and the incore calculated variables are presented in this section. Any reference to axial position is relative to the bottom of the heated core.

1. COBRA-IV-I MODEL

The Semiscale Mod-I core was modeled as a single channel extending from the core barrel inlet to the upper plenum as illustrated in Figure 6. The model contained 35 volumes with each volume having an axial height of 12.7 cm (5 in.). The flow area contraction between the core barrel inlet and heated length and the expansion at the top of the heated length were included within the model. The grid spacers, the mixer plate in the core barrel inlet housing, and the drag disk and turbine flowmeter flow resistances were included in the model as constant resistance factors. Axial frictional pressure losses were determined on the basis of homogeneous flow theory. A listing of a sample input to COBRA-IV-I is given in Appendix A.

[a] The code configuration control number is H000621B.

Fig. 6 COBRA-IV-I single-channel model of Semiscale Mod-1 core.

2. COBRA-IV-I BOUNDARY CONDITIONS

The boundary conditions used in the COBRA-IV-I calculations included (a) system pressure, (b) enthalpy at the core bottom or top when flow was into the core, (c) differential pressure between the core bottom and the core top, and (d) rate of energy transfer between core and coolant. These boundary conditions were calculated from measured data. Enthalpy at the core bottom was determined from the measured fluid density and system pressure. Enthalpy at the core top was determined from system pressure and fluid densities measured near the vessel in the intact and broken loop hot legs. The differential pressure between the upper and lower plenums was corrected for 38-cm (15-in.) head of mixture which existed between the pressure measurement tap in the lower plenum and the core barrel inlet. The specified energy transfer from the core heater rods to the coolant was obtained from an inverse heat conduction solution^[a] using measured heater rod temperature data. Appendix A gives the boundary conditions in graphical form.

3. COBRA-IV-I SOLUTION

COBRA-IV-I provides for the option to use an implicit or explicit scheme for solution to the applicable conservation equations. The explicit scheme was designed for fast transient analysis. In this study, the implicit method of solution was used to obtain steady state conditions and the explicit method was used for the transient solution. The calculation interval was 0.5 ms for the first 240-ms period and 1 ms thereafter. The COBRA-IV-I calculated variables were written onto magnetic tape at 20-ms intervals. Results presented in this report are taken from data on the tape and, therefore, are limited to a 20-ms resolution.

4. COMPARISON OF COBRA-IV-I CALCULATED VARIABLES AND EXPERIMENT DATA

COBRA-IV-I calculated density and mass flux were compared with the measured data. Because of the complex vessel and core geometry of the Semiscale system, neither flow nor thermodynamic state variables were measured in the central region of the core. Therefore, calculated variables were compared with measured data for the core barrel location at the bottom of the heated core. The mass flux calculated by COBRA-IV-I for the elevation of the drag disk is compared with the measured mass flux in Figure 7. The measured mass fluxes were obtained from both the drag disk flowmeter and the turbine flowmeter data combined with the measured density data. The calculated mass flux compares well with the flux from the drag disk. Between 100 and 300 ms, the drag disk reached its mechanical limit and during this period, no data were available for comparison with the calculation. The

[a] INVERT Code – code configuration control number is H000861B.

Fig. 7 Comparison of core inlet mass flux calculated by COBRA-IV-I with mass flux determined from measured data obtained from Test S-02-9.

turbine flowmeter had a slower response time than the drag disk and did not measure the fast flow reversal accurately. Beyond 800 ms after rupture, however, the turbine flowmeter provided data in good agreement with that from the drag disk flowmeter. Figure 8 compares the calculated and measured fluid densities. The density also is shown to be calculated quite well by COBRA-IV-I.

The good agreement between experiment data and the results produced by COBRA-IV-I lends confidence that COBRA-IV-I calculated the interior behavior for the Semiscale core correctly.

Fig. 8 Comparison of density calculated by COBRA-IV-I with measured density from Test S-02-9.

IV. DISCUSSION OF HEATER TEMPERATURE RESPONSE AND COBRA CALCULATED HYDRAULICS

This section discusses the heater rod response early in blowdown, the calculated core hydraulic behavior, and the core thermal behavior calculated using COBRA-IV-1, the thermal-hydraulics associated with CHF and rewet, and radial variations in heater rod temperature response at an axial level.

1. CORE HEAT TRANSFER PRIOR TO CHF

Prior to rupture, heat transfer was primarily by convection. The Dittus-Boelter^[16] forced convection correlation gives a heat transfer coefficient of $18 \text{ kW/m}^2\text{-K}$ ($3168 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$) for initial conditions in the peak power step which compares well with $19.5 \text{ kW/m}^2\text{-K}$ ($3440 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$) calculated on the basis of the actual rod power, rod temperature, and fluid temperature measurements. Immediately after rupture, the rod surface heat flux increased 45% above its initial value, the surface temperature decreased, and the fluid temperature increased as shown in Figure 9. The decrease in the difference between the rod surface temperature and the bulk fluid temperature resulted from an increase in nucleate boiling heat transfer. The increase in boiling was caused by the vapor pressure difference between the wall and the bulk fluid increasing as the system initially depressurized from 15.5 MN/m^2 (2253 psia) to about 11.0 MN/m^2 (1600 psia) in the first 50 ms of blowdown. After the initial rise in heat flux, the rod stored energy was reduced and the heat flux began to gradually decrease. However, at the time of boiling crisis (630 ms) surface heat fluxes of the heater rods were still larger in magnitude than the initial steady state heat fluxes.

Fig. 9 Heat flux, surface temperature, and fluid temperature response during Test S-02-9.

2. CALCULATED CORE HYDRAULIC BEHAVIOR

Pertinent COBRA-IV-I results are discussed in order to obtain an overview and understanding of the Semiscale Mod-1 core hydraulics during the early part of blowdown. Appendix A contains more details about the COBRA-IV-I calculated hydraulic variables. Semiscale core hydraulics are reflected in the calculated quality and flow rate. The quality indicates the instantaneous amount of water available for cooling the core and provides a measure of the voiding rate within the core. The flow rate indicates the direction of flow and how the core redistributes or loses coolant.

At rupture, the low flow resistance path to the cold leg break produced a fast flow reversal at the bottom of the core (Figure 7). This reversal was calculated, as shown in

Figure 10, to extend to the upper part of the heated core where a flow stagnation occurred. With the flow stagnation in the heated core, coolant flowed out the top and bottom of the core. This emptying of the core from both ends and a high core heat transfer rate led to a rapid and large magnitude increase in fluid quality as shown in Figure 11. After the flow reversal, coolant was reheated as it passed back down through the core. This reheating produced a higher quality below the stagnation region [below 97 cm (38 in.)] than above the stagnation region as shown in Figure 11.

At 700 ms after rupture, fluid above the stagnation region reversed flow direction, and coolant flowed from the upper plenum down through the core. This complete reversal of flow in the upper portion of the core brought a lower quality mixture into the heated section. The influx of the lower enthalpy fluid caused the quality values to decrease resulting in turnover of the quality curve as seen in Figure 11. From 1.6 s to the end of the calculation (2 s) both the mass flux and quality indicate that a quasi-steady condition existed. The mass flux became axially uniform within the core and the quality became steady at an axial level.

Fig. 10 COBRA-IV I calculated mass flux at axial core levels.

Fig. 11 COBRA-IV-I calculated quality at axial core levels.

3. THERMAL-HYDRAULICS ASSOCIATED WITH CHF AND REWET

In this section the relation between quality and CHF as well as mass flux and CHF are discussed for Test S-02-9. The COBRA-IV-I solution was used to provide the hydraulic variables associated with CHF and rewet.

Figure 12 shows the calculated quality relative to the mean heat flux at the core peak power zone. The mean heat flux was determined using 22 flux values calculated from measured heater rod temperature data within the 53- to 79-cm (21- to 31-in.) interval and an inverse conduction solution. A normal distribution was assumed for the instantaneous fluxes. Curves representing one standard deviation above and below the mean heat flux are also shown in the figure. The variation in heat flux up to the time of CHF (450 to 630 ms) was small, but became larger after CHF because of the occurrence of some rewetting. Figure 12 shows that the heat flux rapidly decreased during the time of rising quality (500 to 650 ms). This decrease was indicative of CHF. The slope of the quality curve is steep, and the quality ranges from 30 to 70% during the 150-ms period in which the critical heat fluxes occurred. This range in quality heat flux resulted from dryout at the rod surface as opposed to a steam film forming between the rod and a low quality mixture.

The mass flux has been found to be an important variable influencing CHF under quasi-steady conditions, but the mass flux had little obvious influence on early CHF in

Fig. 12 Quality calculated by COBRA-IV-I and the mean heat flux at the peak power step (53 to 79 cm).

Semiscale, especially in the upper section of core where the analysis showed a prolonged (600 ms) flow stagnation (Figure 10). This lack of mixing and reduced convection did not lead to a CHF as might be expected, indicating that low flow rates or prolonged stagnation does not necessarily produce CHF in Semiscale.

CHF occurred primarily in the center and bottom of the core during Test S-02-9 and the other large cold leg break tests in the blowdown heat transfer series^[2]. The analysis indicated that the axial distribution of CHF was from a higher quality in the center and bottom of the core than in the upper core region. The quality shown in Figure 11 rose at a faster rate and reached a larger magnitude below the core center than above the center during the time of CHF (500 to 650 ms). CHF strongly depends on quality^[17] and, therefore, CHF would be expected to occur in the core center and bottom of the core where the quality was high rather than in the upper part of the core where the quality was lower.

In the Semiscale core during Test S-02-9 and during the other cold leg break blowdown tests, core locations which experienced CHF often experienced a rewet shortly after the CHF. Analysis indicated that the rewets resulted from the influx of coolant into the top of the heated core. The rewets were observed to occur predominately in the upper part of the core as illustrated in Figure 13. Also some axial zones on the heaters experienced rewet and were bounded above and below by zones which had no rewets. From the analysis and measured data, this observed axial behavior in rewet was found to result from the axial quality distribution and the axial variation in heater rod surface temperature after CHF. The influence of the axial quality gradient alone can be observed from examining the rewet behavior over the peak power step. In this axial interval from 53 to 79 cm (21 to 31 in.), the

Fig. 13 Axial distribution of rewets.

surface temperatures after CHF and up until the time of rewet (0.6 to 1.3 s) were fairly uniform over the section as shown in Figure 14. The quality, however, was large at the bottom of the step and decreased in the upward core direction as shown in Figure 15. This axial variation in quality produced a large number of rewets at the top of the peak power step and no rewets at the bottom of the step.

The variation of both quality and surface temperature in the lower part of the core led to the zone of partial rewets separated by zones where no rewets were measured. The 28-

Fig. 14 Heater rod surface temperatures at the peak power step.

Fig. 15 COBRA-IV-I calculated quality gradient 1.18 s after rupture.

41-cm (11- to 16-in.) axial core section was a region that had partial rewets, and this section was bounded by regions above and below it by zones where no rewets occurred. This zone of rewets had a higher quality than the power zone above it (41 cm to 53 cm) as shown in Figure 16, but a lower heater rod surface temperature, as indicated by the surface temperature at the 36-cm (14-in.) elevation (TH-F4-14) being lower than that at the 51-cm (20-in.) elevation (TH-D7-20) as shown in Figure 16. It also had a lower quality than the step below it (15 cm to 28 cm) but a higher heater rod surface temperature as shown in Figure 16. The preceding quality-temperature relationship indicates that the axial distribution of rewet was from a trade-off between the magnitude of the heated rod surface temperature and the axial magnitude of fluid quality.

Fig. 16 Heater rod surface temperatures in the lower part of the heated core.

4. RADIAL VARIATIONS IN HEATER ROD TEMPERATURE RESPONSE AT AN AXIAL LEVEL

Variations in heater rod temperature response at an axial level were observed in the Semiscale core. Some rods experienced a boiling crisis at an axial level whereas others did not^[2]. The occurrence of rewetting of heater rods after CHF also varied from rod to rod at an axial level. In Test S-02-9, three heater rods in the 40-rod core bundle were unpowered. These unheated rods and the fast transient hydraulics seen in the core during blowdown could have led to the observed radial variation in heater rod temperature response at an axial level. The results presented in this report from the single channel model could not provide the information to evaluate the hydraulic differences between individual channels about the heater rods.

V. CAPABILITY OF CHF CORRELATIONS DURING BLOWDOWN

This section presents data and analysis to determine the capability of existing CHF correlations to predict a boiling crisis during blowdown. Critical heat flux equations have for the most part been generated from steady state data, and therefore, are not necessarily accurate for transient conditions except for specific cases in which they are shown to provide a good calculation of the CHF. This evaluation of CHF correlations provides a basis for using, or in some cases not using, the steady state critical heat flux correlations during transient blowdown conditions. Collier^[17], and Tong^[18], have provided views of the boiling crisis and CHF correlations; thus a review of existing CHF correlations is not given here. In this report the following seven CHF correlations are used to calculate the critical heat flux within the Semiscale core during blowdown conditions. The correlations evaluated in this study were the MacBeth^[19], Biasi^[20], Barnett^[21], Babcock & Wilcox Company B&W-2^[22], Westinghouse Electric Corporation W-3^[18], LOFT, and General Electric Company^[23] correlations. Appendix B provides a description of these correlations and the range in their independent variables for which they are considered to be accurate.

The approach in this section is to calculate the critical heat flux at each transient point in time using thermodynamic and hydrodynamic conditions calculated by COBRA-IV-I. Equations which contained a length variable and an inlet enthalpy (the Westinghouse Electric Corporation W-3 and the Barnett correlations), are not appropriate during a transient. These equations were modified for transient conditions. Appendix B describes the modifications to the correlations for use in the transient situation.

Figure 17 compares the calculated critical heat flux from correlations with the mean heat flux from measured data at the peak power step. The mean heat flux, as earlier discussed, was calculated from 22 fluxes obtained from thermocouple data within the peak power zone. A boiling crisis was measured to occur when the surface heat flux rapidly decreased. A boiling crisis was calculated to exist only if the calculated CHF was below the measured heat flux. The time at which the calculated critical heat flux and measured heat flux curves intersect was the calculated CHF time.

The General Electric Company (GE) correlation is shown in Figure 17 to intersect the mean heat flux as the mean heat flux begins to decrease, therefore providing an excellent calculation of the time of CHF. The good agreement between the GE correlation and data is expected from the algebraic form of the equation and the large and rapid increase in quality during the period of CHF (500 to 600 ms). The GE-CHF equation is a simple linear function of quality (Appendix B), and during the period of a large rise in quality, the calculated critical heat flux decreased below the mean heat flux. As shown in Figure 17, the MacBeth, B&W-2, and Biasi correlations which also strongly depend on quality calculated close to the same time for CHF and this time was 150 ms later than the mean critical heat flux time (500 ms). The Westinghouse W-3 correlation modified for transient conditions did not calculate a CHF. The calculated critical heat flux from the W-3 correlation reached a minimum value at about 350 ms and then increased to values outside the bounds of the

Fig. 17 Calculated critical heat fluxes compared with the mean heat flux at the peak power step (53 to 79 cm).

graph. The large calculated CHF after 350 ms was the result of an exponential expression in the correlation rapidly increasing as quality increased above 20%. The LOFT correlation did not calculate that a critical heat flux would occur, and the Barnett correlation, modified for transient conditions, calculated critical heat flux values which were below the measured heat flux even prior to rupture indicating CHF would occur at steady state conditions.

The results from the correlations are compared with the mean heat flux in the lower and upper parts of the core in Figures 18 and 19, respectively. The lower region is from 28 to 41 cm (11 to 16 in.), and the mean heat flux value in this region was calculated from seven fluxes. The upper part of the core shown in Figure 19 is between the 79- and 91-cm (31- to 36-in.) level, and the mean heat flux was calculated from eight fluxes. In the lower part of the core, the GE correlation again calculates the critical heat flux well. The mean critical heat flux occurred at about the same time as that for the peak power step (560 ms), and the quality as seen in Figure 11 rose at about the same rate as that in the peak power step. Therefore, the GE correlation which did well at the peak power location would be expected to do well in the lower part of the core. In the upper section of the core where few critical heat fluxes occurred, the GE correlation calculated that a boiling crisis would occur (Figure 19). A CHF was calculated because the calculated quality was about the same as in the center and lower parts of the core. The GE correlation using COBRA-IV-I calculated hydraulics provided a conservative calculation of the critical heat flux in the upper part of the core.

The results for the lower section of core from the MacBeth, Biasi, and B&W-2 correlations were similar to those for the peak power level. These correlations calculated

Fig. 18 Calculated critical heat fluxes compared with the mean heat flux in the lower part of the core (28 to 41 cm).

Fig. 19 Calculated critical heat fluxes compared with the mean heat flux in the upper part of the core (79 to 91 cm).

close to the same time for a boiling crisis which was about 100 ms after the mean heat flux began to decrease. In the upper part of the core, these three correlations calculated that a CHF also would have happened at this core elevation. The curve for the Westinghouse W-3 correlation remained larger than the mean heat flux curve at the 28- to 41-cm level (Figure 18); thus calculating no critical heat flux. Above the peak power step where few rods experienced CHF, the W-3 correlation calculated that no boiling crisis would happen. The LOFT correlation did not calculate that a critical heat flux would be reached in the lower part of the core (Figure 18), but did calculate that a CHF would occur above the center of the core (Figure 19). The Barnett correlation calculated critical heat flux values which were below the mean heat flux curve in both the upper and lower core before and after rupture as shown in Figures 18 and 19.

Variations in the COBRA-IV-I thermal-hydraulic behavior (specifically fluid quality and mass flux) used in the correlations had minor effects on the calculated time of critical heat flux. In Appendix A, the range in thermodynamic and hydraulic variables calculated by COBRA-IV-I from application of the estimated bounds in uncertainty in boundary conditions is given. Results given in the appendix show a variation existed in both quality and mass flux, with the most pronounced variation produced from the applied range in energy transfer rate from core to coolant. Figure 20 shows the GE correlation calculated critical heat flux values using the COBRA-IV-I hydraulics which were calculated using the range in core heat transfer rate. From the figure, the CHF curves are observed to be relatively close to the same value from 500 to 700 ms which encompassed the time period of CHF. The reason the GE correlation calculated close to the same CHF curves was a result of the correlation being strongly dependent on quality and a result of the rapid and large rise in the calculated quality during the time of CHF.

Fig. 20 Variation in CHF from the General Electric correlation using hydraulics calculated from COBRA-IV-I and the bounds in core heat transfer.

In this section, the time of critical heat flux was shown to be calculated well using an existing CHF correlation. This capability to calculate CHF resulted primarily from the fact that many of the correlations are strongly dependent on fluid quality, and that during the time of CHF, the quality rose quickly and to a large value. The large quality reduced the calculated critical heat flux to below the measured heat flux value at about the time of the measured boiling crisis.

VI. CONCLUSIONS

A better understanding of the Semiscale core hydraulic behavior during the early part of a blowdown transient was obtained, and insight into the thermal-hydraulic behavior leading to critical heat flux within the Semiscale core during a 200% cold leg break simulation was gained. Specific conclusions were:

- (1) The flow was calculated to stagnate in the heated core and, with the high heat flux from the Semiscale heaters, a rapid and large increase in quality resulted. The fast increase in measured heater rod temperatures which was indicative of a boiling crisis corresponded to this period when the calculated quality quickly rose. Therefore, the conclusion reached from the analysis is that the quality increase was the primary variable responsible for the boiling crisis in Semiscale.
- (2) The calculated void fraction was very high during the time interval when CHF occurred which suggests that dryout was the mechanism producing the boiling crisis.
- (3) From shortly after rupture to 700 ms after rupture, the flow was calculated to stagnate in the upper part of the heated core. After 700 ms, fluid above the stagnation point reversed flow direction and coolant flowed from the upper plenum down through the core. The influx of a low quality mixture into the heater core after 700 ms was responsible for the observed rewets.
- (4) Those steady state critical heat flux correlations which have a strong dependence on quality did well in calculating CHF. The simple General Electric Company correlation provided best results for the Semiscale core.
- (5) The excellent calculation capability of the General Electric Company correlation and the good capability of the MacBeth, Biasi, and B&W-2 correlations for the Semiscale core indicates that these steady state correlations are adequate for blowdown analysis in Semiscale, and that no new CHF calculation methods are needed for the Semiscale core.
- (6) The steady state correlations which needed modification in order to be applicable to blowdown with reverse flow were not able to provide an adequate calculation of CHF in the Semiscale core.

- (7) The observed radial and azimuthal temperature variations at the same elevation for different heater rods indicate that local subchannel behavior existed. The single channel COBRA-IV-I model which provided information about the average core behavior at axial locations did not provide the information for evaluating subchannel differences. A three-dimensional hydraulic analysis for the Semiscale core is needed for evaluating the subchannel behavior.
- (8) The purpose of the COBRA-IV-I analysis was to calculate the thermal-hydraulics in the Semiscale core where there were few measurements. More incore measurements are required to reduce the uncertainty in core hydraulics and to provide local data for evaluating the accuracy of the core hydraulic calculations. At this time, parts of the Semiscale system are being redesigned, and with a modified core more extensive incore measurements are expected to be available.

VII. 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. T. K. Larson, *Core Thermal Response During Semiscale Mod-1 Blowdown Heat Transfer Tests*, ANCR-1285 (June 1976).
3. H. S. Crapo, M. F. Jensen, K. E. Sackett, *Experimental Data Report for Semiscale Mod-1 Test S-02-1 (Blowdown Heat Transfer Test)*, ANCR-1231 (July 1975).
4. H. S. Crapo, M. F. Jensen, K. E. Sackett, *Experiment Data Report for Semiscale Mod-1 Test S-02-2 (Blowdown Heat Transfer Test)*, ANCR-1232 (August 1975).
5. H. S. Crapo, M. F. Jensen, K. E. Sackett, *Experiment Data Report for Semiscale Mod-1 Test S-02-3 (Blowdown Heat Transfer Test)*, ANCR-1233 (September 1975).
6. H. S. Crapo, M. F. Jensen, K. E. Sackett, *Experiment Data Report for Semiscale Mod-1 Test S-02-4 (Blowdown Heat Transfer Test)*, ANCR-1234 (November 1975).
7. H. S. Crapo, M. F. Jensen, K. E. Sackett, *Experiment Data Report for Semiscale Mod-1 Test S-02-5 (Blowdown Heat Transfer Test)*, ANCR-1235 (January 1976).
8. H. S. Crapo and K. E. Sackett, *Experiment Data Report for Semiscale Mod-1 Test S-02-8 (Blowdown Heat Transfer Test)*, ANCR-NUREG-1238 (August 1976).
9. H. S. Crapo, M. F. Jensen, K. E. Sackett, *Experiment Data Report for Semiscale Mod-1 Test S-02-7 (Blowdown Heat Transfer Test)*, ANCR-1237 (November 1975).
10. B. L. Collins, H. S. Crapo, and K. E. Sackett, *Experiment Data Report for Semiscale Mod-1 Test S-02-6 (Blowdown Heat Transfer Test)*, TREE-NUREG-1037 (January 1977).
11. H. S. Crapo, M. F. Jensen, K. E. Sackett, *Experiment Data Report for Semiscale Mod-1 Tests S-02-9 and S-02-9A (Blowdown Heat Transfer Tests)*, ANCR-1236 (January 1976).
12. J. M. Cozzuol, *Thermal-Hydraulic Analysis of the Semiscale Mod-1 Blowdown Heat Transfer Test Series*, ANCR-1287 (June 1976).
13. C. L. Wheeler et al, *COBRA-IV-I: An Interim Version of COBRA for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores*, BNWL-1962 (March 1976).

14. J. V. Beck, "Nonlinear Estimation Applied to the Nonlinear Inverse Heat Conduction Problem", *International Journal of Heat and Mass Transfer*, 13 (1970) pp 703-716.
15. E. M. Feldman and D. J. Olson, *Semiscale Mod-1 Program and System Description for the Blowdown Heat Transfer Tests (Test Series 2)*, ANCR-1230 (July, 1975).
16. M. Jakob, *Heat Transfer, 1*, New York: Wiley & Sons (1957).
17. J. G. Collier, *Convective Boiling and Condensation*, Maidenhead, Berkshire, UK: McGraw-Hill Book Company Limited (1972).
18. L. S. Tong, *Boiling Crisis and Critical Heat Flux*, TID-25887 (1972).
19. R. V. MacBeth, *Burnout Analysis Part 3. The Low Velocity Burnout Regime*, AEEW-R 222 (1963).
20. L. Biasi et al, "Studies on Burnout," Part 3, *Energia Nucleare*, 14, S (1967) pp 530-536.
21. P. G. Barnett, *A Correlation of Burnout Data for Uniformly Heated Annuli and Its Use for Predicting Burnout in Uniformly Heated Rod Bundles*, AEEW-R 463 (1966).
22. J. S. Gellerstedt et al, "Two Phase Flow and Heat Transfer in Rod Bundles," *American Society of Mechanical Engineers Conference, Los Angeles, California, November 18, 1969*, pp 63-71.
23. G. M. Roy, "Getting More out of BWR's," *Nucleonics*, 24, 11 (1966).

APPENDIX A
COBRA-IV-I BOUNDARY CONDITIONS

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APPENDIX A

COBRA-IV-I BOUNDARY CONDITIONS

In this appendix boundary conditions, a sample of data input, and results from COBRA-IV-I calculations using the range of uncertainties in boundary conditions are given.

1. BOUNDARY CONDITIONS

The boundary conditions specified for the COBRA-IV-I calculations were:

- (1) System pressure
- (2) Differential pressure across the core
- (3) Enthalpy at core bottom or top if flow was into the core
- (4) Core to coolant energy transfer rate.

The system pressure was measured in the upper plenum and this pressure is shown in Figure A-1. The measured differential pressure was from the lower plenum to the upper plenum. In order to obtain the pressure loss boundary condition from the core barrel inlet to the upper plenum, the head of mixture from the core barrel inlet to the pressure tap in the lower plenum was subtracted from the measured differential pressure. The differential pressure across the core is shown in Figure A-2. The enthalpy at the core inlet was determined from the measured system pressure and fluid density and is shown in Figure A-3. The enthalpy at the top of the core was determined from measured system pressure and fluid densities in the broken and unbroken hot legs adjacent to the vessel and this enthalpy was a constant 1.46×10^6 J/kg for the 2 s analyzed. Energy transfer from core to coolant was determined from measured heater rod temperature data and an inverse conduction solution. The method of inversion was that described by Beck[A-1]. The energy transfer rate was calculated for each axial power step, and these calculations were used directly as input into COBRA-IV-I.

Fig. A-1 Pressure boundary condition used in COBRA-IV-I.

Fig. A-2 Differential pressure across the core used as boundary conditions in COBRA-IV-I.

Fig. A-3 Inlet enthalpy boundary conditions used in COBRA-IV-I.

2. SAMPLE COBRA-IV-I INPUT

The input to COBRA-IV-I for the best estimate calculation is given in Table A-I. A description of operation procedures for COBRA-IV-I is given by Wheeler et al[A-2].

TABLE A-I

COBRA-IV-I SAMPLE INPUT DATA

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 12345678901234567890123456789012345678901234567890123456789012345678901234567890

2000

1 S-02-9 DP DRIVER BASE CASE

1	20	1							
200.381.8	0.01839	2.2873	355.5	1198.3	0.345	0.3852	0.002099		
300.417.4	0.01889	1.5427	394.0	1202.9	0.313	0.3760	0.001800		
440.454.0	0.01950	1.0554	434.8	1204.8	0.283	0.3658	0.001500		
520.471.1	0.01982	0.89137	454.2	1204.5	0.273	0.3604	0.001388		
600.486.2	0.02013	0.76975	471.7	1203.7	0.264	0.3550	0.001277		
680.499.9	0.02043	0.67581	487.7	1202.3	0.255	0.3494	0.001177		
760.512.3	0.02072	0.60097	502.7	1200.4	0.248	0.3435	0.001088		
880.529.3	0.02116	0.51333	523.4	1197.0	0.240	0.3350	0.000966		
1000.544.6	0.02159	0.44596	542.4	1192.9	0.232	0.3275	0.000866		
1120.558.5	0.02203	0.39244	560.5	1188.2	0.225	0.3198	0.000766		
1200.567.2	0.02232	0.36245	571.9	1184.8	0.221	0.3140	0.000700		
1360.583.3	0.02292	0.31256	593.5	1177.3	0.214	0.3036	0.000600		
1400.587.1	0.02307	0.30178	598.8	1175.3	0.213	0.3012	0.000577		
1560.601.5	0.02370	0.26384	619.2	1166.8	0.207	0.2900	0.000478		
1680.611.5	0.02420	0.23978	634.0	1159.8	0.203	0.2820	0.000411		
1800.621.0	0.02472	0.21861	648.5	1152.3	0.199	0.2738	0.000366		
2000.635.8	0.02565	0.18831	672.1	1138.3	0.192	0.2603	0.000288		
2200.649.5	0.02669	0.16272	695.5	1122.2	0.184	0.2465	0.000211		
2320.657.2	0.02739	0.14916	709.5	1111.4	0.179	0.2370	0.000177		
2480.666.9	0.02845	0.13266	729.1	1095.5	0.172	0.2240	0.000133		

2

.184 -.2

3 2

0. 1. 1. 1.

4 1 1

17.39065.8549.05

5 1 8

0.000.1195.1200.1194.1429.5199.52001.000

1

1.5981.5981.5981.5981. 1. 1.42 1.42

7 2 10 4

.1086 1.1371 2.1485 3.2512 3.3486 3.4342 3

.5314 4.6457 3.7657 3.8857 3

19.5

19.

1.56

13.41

8 1 1

115.61 1. 11.

9 75 1 8

175.2.000 1.0 .01 .05 1. 1. 1.

35 4000 30 30

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100

0. .0005.24 .0005.241 .0010 .40 .00101.4 .001 2. .001

5. .005 10.5 .005

10

.02

11 11 7 21 9

1350. 536. .6724 1. 628.

0. 1.674.05 1.219.1 1.152.2 1.152.53 1.1361. 1.086

2. 1.0023. .91875. .81857. .778410. .6665

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 12345678901234567890123456789012345678901234567890123456789012345678901234567890

TABLE A-I (continued)

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1234567890123456789012345678901234567890123456789012345678901234567890

0.	1.	.1	1.	.2	1.07	.28	1.15	.35	1.15	.8	1.67
10.	2.										
0.	9.04	.06	-4.46	.1	-4.76	.16	-2.86	.2	-3.86	.27	-1.26
.32	-.66	.37	1.34	.42	-.26	.56	2.4	.71	-2.36	.86	.74
.92	.44	1.08	1.74	1.27	-.06	1.35	-.46	1.45	-.06	1.6	.46
1.7	-.06	1.86	-.26	2.	.44						
0.	1.	2.6	1.	4.3	1.16	6.	1.14	76.4	1.59	7.8	1.59
8.4	.955	9.8	.955	10.	.955						

00000000111111112222222233333333333344444444555555555566666666667777777778
1234567890123456789012345678901234567890123456789012345678901234567890

3. CALCULATED INCORE VARIABLES FROM UNCERTAINTIES IN BOUNDARY CONDITIONS

The range of uncertainties in boundary conditions was obtained from Reference A-3, and from experience in evaluating Semiscale data. COBRA-IV-I solutions were made with a variation in one of the boundary conditions. The other boundary conditions remained the best estimate condition. The base condition quality and mass flux were then compared with those from the solution using uncertainties in a boundary condition.

3.1 Uncertainty in Pressure

The best estimate pressure and the range in uncertainty in pressure are shown in Figure A-1. Figures A-4 and A-5 show the calculated mass fluxes and qualities, respectively, at the peak power level. From these figures the uncertainties in pressure are seen to produce only minor variations in incore calculated parameters.

3.2 Uncertainty in Differential Pressure

The uncertainties in the differential pressure across the core are shown in Figure A-2. Figure A-6 shows the calculated mass flux and Figure A-7 shows the calculated quality using these pressure losses. When the variation in differential pressure boundary condition is supplied to COBRA-IV-I, an erroneous pressure loss value was used in the input table at 1.25 s. This input value was three orders of magnitude too large and resulted in an abnormal termination of the COBRA-IV-I calculation. Because the input error was after the time of CHF, the problem was not resolved using the correct value. Up to the point at which the solution terminated, no significant variation was produced in the incore calculated variables. This period encompassed the time of a boiling crisis.

Fig. A-4 Variation in the COBRA-IV-I calculated mass flux at the 64-cm (25-in.) elevation from the uncertainty in the system pressure boundary condition.

Fig. A-5 Variation in the COBRA-IV-I calculated quality at the 64-cm (25-in.) elevation from the uncertainty in the system pressure boundary condition.

Fig. A-6 Variation in COBRA-IV-I calculated mass flux from the uncertainty in differential pressure boundary condition.

Fig. A-7 Variation in COBRA-IV-I calculated quality from the uncertainty in differential pressure boundary condition.

3.3 Uncertainty in Core to Coolant Energy Transfer Rate

The range in uncertainty in heat flux was $\pm 7\%$. This value was based primarily on the uncertainty in knowing the core power. The calculated hydraulics were found to be most sensitive to this variable. The best estimate heat fluxes at axial levels are shown in Figure A-8. Figures A-9 and A-10 compare the calculated mass fluxes and qualities at the peak power level from the uncertainties in heat flux. As can be seen considerable differences in the quality resulted.

Fig. A-8 Heat transfer rates at axial elevations in the core.

Fig. A-9 Variation in COBRA-IV-I calculated mass flux from the uncertainty in core heat transfer rate.

Fig. A-10 Variation in COBRA-IV-I calculated quality from the uncertainty in core heat transfer rate.

4. REFERENCES

- A-1. J. V. Beck, "Nonlinear Estimation Applied to the Nonlinear Inverse Heat Conduction Problem," *International Journal of Heat and Mass Transfer*, 13 (1970) pp 703-716.
- A-2. C. L. Wheeler et al, *COBRA-IV-I: An Interim Version of COBRA for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores*, BNWL-1962 (March 1976).
- A-3. H. S. Crapo and K. E. Sackett, *Experiment Data Report for Semiscale Mod-1 Test S-02-8 (Blowdown Heat Transfer Test)*, ANCR-NUREG-1238 (August 1976).

APPENDIX B

DESCRIPTION OF CHF CORRELATIONS

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APPENDIX B

DESCRIPTION OF CHF CORRELATIONS

The algebraic form of the equations evaluated in this study are given. For each correlation, the range in independent variables for which the correlation is considered accurate is given when possible. The nomenclature used is given at the front of this report.

1. BABCOCK & WILCOX COMPANY B&W-2 CORRELATION [B-1]

This correlation is evaluated in English units. The following range in independent variables is given for this equation [B-2]:

diameter	0.2 to 0.5 in.
length	72 in.
pressure	2000 to 2400 psia
mass flux	0.75×10^6 to 4×10^6 lbm/hr-ft ²
quality	-0.03 to 0.20 (based on enthalpy).

The form of the equation is

$$q_{CHF} = 10^6 (1.155 - 0.407 D_e) \left\{ 0.37 \times 10^8 \right. \\ \left. (0.591 \times 10^{-6} G) [0.83 + 0.685(0.001p-2)] \right. \\ \left. - 0.1521 G x_{fg} \right\} / \left\{ 12.71 (3.054 - G/10^6) [0.712 + 0.2073(0.001p-2)] \right\} \quad (B-1)$$

2. BARNETT CORRELATION [B-3, B-4]

This correlation is evaluated in English units and the following range in independent variables is given for this correlation:

length	24 to 108 in.
pressure	600 to 1400 psia

mass flux	0.14×10^6 to 6.2×10^6 lbm/hr-ft ²
$h(z)-h_i$	0 to 412 Btu/lbm.

The equation is

$$q_{CHF} = 10^6 \frac{A(h_{fg}/649) + B(h_f - h_i)}{C + z} \quad (B-2)$$

where

$$\begin{aligned} A &= 67.45 D_h^{0.68} (G \times 10^{-6})^{0.192} \{1 - 0.744 \exp [-6.512 D_e (G \times 10^{-6})]\} \\ B &= 0.2587 D_h^{1.261} (G \times 10^{-6})^{0.817} \\ C &= 185. D_e^{1.415} (G \times 10^{-6})^{0.212} \\ D_e &= \left[D_r (D_r + D_h) \right]^{1/2} - D_r. \end{aligned} \quad (B-3)$$

The Barnett correlation as given in References B-3 and B-4 is not applicable for transient conditions. The specification of the inlet enthalpy and distance from inlet to location of CHF is not defined for reverse flow and for the case where the flow stagnates. The correlation was changed as subsequently described to be applicable for the transient analysis.

A constant heat flux (q_{CHF}) is assumed to be transferred to the coolant over the short distance z away from the location of the critical heat flux. A steady state balance along the length z gives

$$Q = D_r z q N_{rods} = GA (h - h_i) \quad (B-4)$$

where N_{rods} is the number of heated rods in the core, h is the enthalpy at the point of CHF and h_i is the enthalpy at a distance z away from the critical heat flux point. Solving for h_i gives

$$h_i = h - \frac{D_r z q N_{rods}}{GA} \quad (B-5)$$

$$K = \frac{D_r N_{rods}}{A} \quad (B-6)$$

$$h_i = h - Kzq/G. \quad (B-7)$$

In Equation (B-7), the h_i term is equivalent to the term h_i in the Barnett equation [Equation (B-2)]. Substituting Equation (B-7) into Equation (B-2) gives

$$q_{CHF} = \frac{(Ah_{fg}/649) + B(h_f - h)}{(C+z)10^{-6} - \frac{BKz}{G}} \quad (B-8)$$

As the length z approaches zero, Equation (B-8) reduces to the local condition equation for the critical heat flux.

$$q_{CHF} = \frac{(Ah_{fg}/649) + B(h_f - h)}{C} 10^6. \quad (B-9)$$

This equation can be expressed as a linear relation to quality

$$q_{CHF} = (A - B \times h_{fg}) 10^6. \quad (B-10)$$

Equation (B-10) is the form of the Barnett correlation used in this study.

3. BIASI CORRELATION [B-5]

This correlation is evaluated in cgs units and is valid over the following range of variables:

diameter	0.3 to 3.75 cm
length	20 to 600 cm
pressure	2.7 to 140 bar
mass flux	10 to 600 g/cm ² -s
quality	1/(1 + ρ_f/ρ_g) to 1.

The critical heat flux is given as the higher of the two values from the following equations. For the low quality region

$$q_{CHF} = \frac{1.883 \times 10^3}{D_e^n G^{1/6}} \left[\frac{f(P)}{G^{1/6}} - x \right]. \quad (B-11)$$

For the high quality region

$$q_{CHF} = \frac{1.78 \times 10^3 s(p)}{D_e^n G^{0.6}} (1 - x) \quad (B-12)$$

where

$$n = 0.4 \text{ for } D_e \geq 1 \text{ cm}$$

$$n = 0.6 \text{ for } D_e \leq 1 \text{ cm}$$

$$f(p) = 0.7249 + 0.099p \exp(-0.032p)$$

$$s(p) = 1.155 + 0.149p \exp(-0.019p) + \frac{0.889p}{10+p^2}$$

4. GENERAL ELECTRIC COMPANY CORRELATION [B-6]

The range of independent variables for the General Electric Company correlation as given by Tong[B-2] are

length less than or equal to 108 in.

pressure 1000 psia

mass flux $1. \times 10^6$ lbm/hr-ft².

The correlation is given in English units and the form of the correlation is

$$q_{CHF} = 10^6 (0.84 - x)$$

$$\text{for } G < 0.5 \times 10^6 \text{ lbm/hr-ft}^2 \quad (B-13)$$

$$q_{CHF} = 10^6 (0.8 - x)$$

$$\text{for } G \geq 0.5 \times 10^6 \text{ lbm/hr-ft}^2. \quad (B-14)$$

5. LOFT CORRELATION

The LOFT CHF correlation[a] used in this study was generated using CHF data for the following range in conditions:

length 5.5 ft

pressure 2000 to 2300 psia

mass flux 0.95×10^6 to 2.0×10^6 lbm/hr-ft²

quality -0.16 to 0.2 (based on enthalpy).

The correlation is evaluated using English units and the equation is

$$q_{CHF} = 0.12947 G + 710 p - 0.23018 p^2 - 2.9563 G x + 0.001004 G p x. \quad (B-15)$$

[a] Since this study was performed, the LOFT correlation has been modified to cover pressures from 1000 to 2000 psia.

6. MACBETH CORRELATION [B-7, B-8, B-9, B-10]

The correlation is given in English units and was generated from tube data. The range of independent variables for the low flow regime part of the correlation is

pressure	any system pressure
mass flux	0.1×10^6 to 0.75×10^6 lbm/hr-ft ²
quality	0 to 1.

The form of the equation for the low flow regime is

$$q = \frac{h_{fg}}{135} (G \times 10^{-6})^{0.5} (1 - x) (10^6). \quad (B-16)$$

The range in independent variables for the high flow regime is

pressure	560 to 2000 psia
mass flux	greater than 0.7 lbm/ft ² -hr.

The form of the equation for the high flow regime is

$$q = \frac{[A - 0.25 D_e (G \times 10^{-6}) h_{fg} x]}{B} 10^6 \quad (B-17)$$

where

$$A = Y_0 D_e Y_1 (G \times 10^{-6}) Y_2 [1 + Y_3 D_e + Y_4 (G \times 10^{-6}) + Y_5 D_e (G \times 10^{-6})]$$

$$B = Y_6 D_e Y_7 (G \times 10^{-6}) Y_8 [1 + Y_9 D_e + Y_{10} (G \times 10^{-6}) + Y_{11} D_e (G \times 10^{-6})].$$

The constants Y_1 through Y_{11} are given in Table B-I.

7. WESTINGHOUSE ELECTRIC CORPORATION W-3 CORRELATION [B-2]

The W-3 correlation is evaluated in English units. The range of independent variables is

diameter	0.2 to 0.7 in.
length	10 to 144 in.
pressure	1000 to 2400 psia

TABLE B-I
CONSTANTS FOR MACBETH CORRELATION^[B-10]

Constant	System Pressure (psia)					
	560	1000	1250	1550	1800	2000
Y_0	237	114	93.3	58.0	194	65.5
Y_1	1.20	0.811	1.10	0.834	2.09	1.19
Y_2	0.425	0.221	0.575	0.224	0.593	0.376
Y_3	-0.940	-0.128	0	-0.0336	-0.597	-0.577
Y_4	-0.0324	0.0274	0	0.0755	-0.131	0.220
Y_5	0.111	-0.0667	0	-0.296	-0.0482	-0.373
Y_6	19.3	127	88.5	48.3	231	17.1
Y_7	0.959	1.32	1.46	0.823	1.93	1.18
Y_8	0.831	0.411	1.00	0.121	0.612	-0.456
Y_9	2.61	-0.274	0	0	-0.575	-1.53
Y_{10}	-0.0578	-0.0397	0	0	-0.255	2.75
Y_{11}	0.124	-0.0221	0	0	0.110	2.24

mass flux 1×10^6 to 5×10^6 lbm/hr-ft²

quality less than 0.15.

The equation is

$$q_{CHF} = \left\{ [2.02 - 0.430 (0.001 p)] + [0.172 - 0.0001 p] \right. \\ \left. \exp [18.2x - 0.00413xp] \right\} (1.16 - 0.87x) \\ [(0.148 - 1.6x + 0.173x|x|) (G/10^6) + 1.04] \\ [0.266 + 0.836 \exp (-3.15 D_e)] \\ [0.826 + 0.0008 (h_{sat} - h_i)] 10^6.$$

(B-18)

The W-3 correlation similar to the Barnett correlation contains an inlet enthalpy, and therefore, this correlation is not applicable to transient conditions. The correlation was modified in a similar manner as the Barnett correlation. The variable h_i given in Equation (B-7) was substituted into Equation (B-18) and the critical heat flux was solved for with the length variable from the inlet to point of critical heat flux approaching zero. The resulting equation is the same as Equation (B-18) except h_i is replaced by the local enthalpy.

8. REFERENCES

- B-1. J. S. Gellerstedt et al, "Two Phase Flow and Heat Transfer in Rod Bundles," *American Society of Mechanical Engineers Conference, Los Angeles, California, November 18, 1969.* pp 63-71.
- B-2. L. S. Tong, *Boiling Crisis and Critical Heat Flux*, TID-25887 (1972).
- B-3. P. G. Barnett, *A Correlation of Burnout Data for Uniformly Heated Annuli and Its Use for Predicting Burnout in Uniformly Heated Rod Bundles*, AEEW-R 463 (1966).
- B-4. P. G. Barnett, *A Comparison of the Accuracy of Some Correlations for Burnout in Annuli and Rod Bundles*, AEEW-R 558 (1968).
- B-5. L. Biasi et al, "Studies on Burnout," Part 3, *Energia Nucleare*, 14, S (1967) pp 530-536.
- B-6. G. M. Roy, "Getting More out of BWR's," *Nucleonics*, 24, 11 (1966).
- B-7. R. V. MacBeth, *Burnout Analysis Part 3. The Low Velocity Burnout Regime*, AEEW-R 222 (1963).
- B-8. B. Thompson and R. V. MacBeth, *Boiling Water Heat Transfer – Burnout in Uniformly Heated Round Tubes: A Compilation of World Data with Accurate Correlations*, AEEW-R 356 (1964).
- B-9. D. H. Lee, *An Experimental Investigation of Forced Convection Burnout in High Pressure Water, Part 4. Large Diameter Tubes at About 1600 psia*, AEEW-R 479 (1966).
- B-10. J. G. Collier, *Convective Boiling and Condensation*, Maidenhead, Berkshire, UK: McGraw Hill Book Company Limited (1972).

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