EGG-CDAP-5207

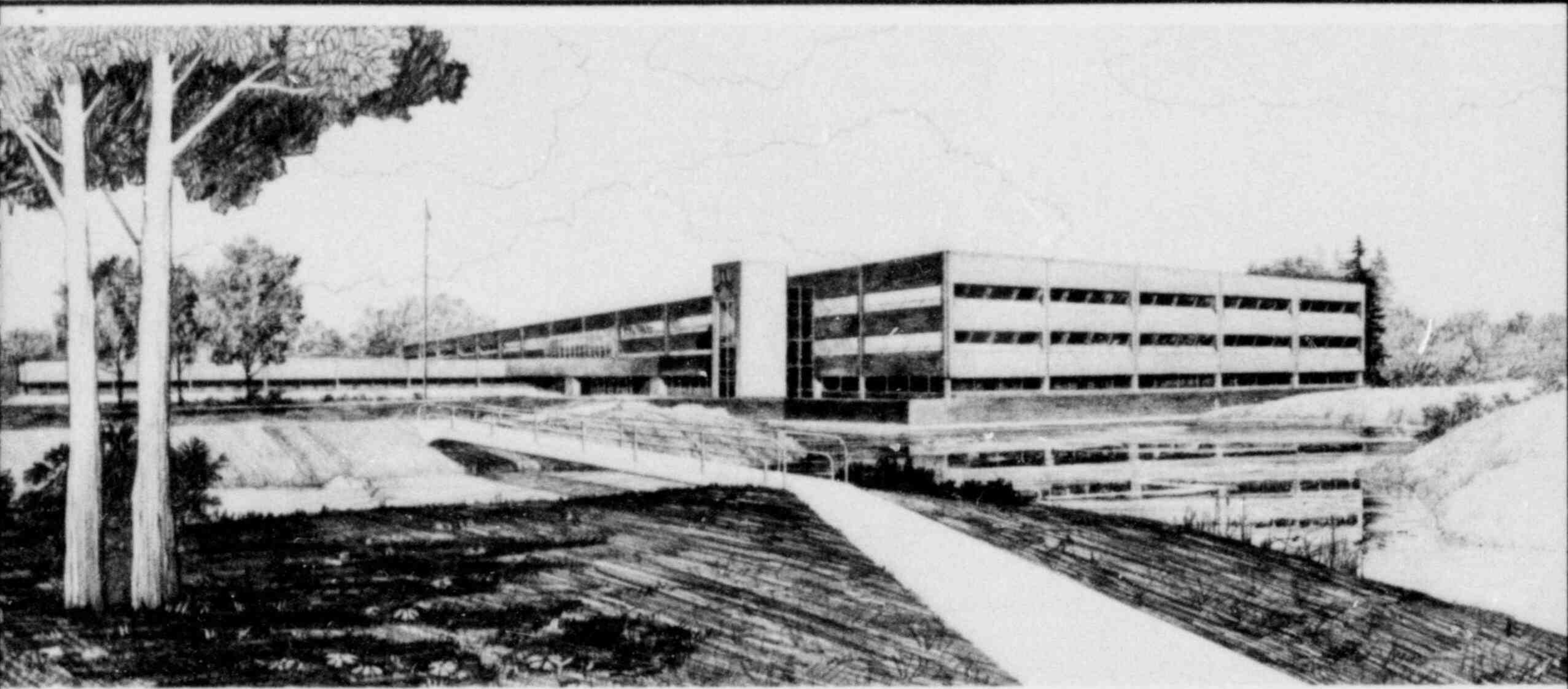
July 1980

SUMMARY LETTER ON CURRENT RESEARCH AND PREDICTIVE ABILITY ON TRANSIENT CHF

C. C. Tsai

U.S. Department of Energy

Idaho Operations Office • Idaho National Engineering Laboratory



This is an informal report intended for use as a preliminary or working document

Prepared for the
U.S. Nuclear Regulatory Commission
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A6278

EG&G Idaho

EG&G Idaho, Inc.

FORM EG&G-398
(Rev. 11-79)

INTERIM REPORT

Accession No. ____________

Report No. EGG-CDAP-5207

Contract Program or Project Title: Heat Transfer Correlation Development and Assessment

Subject of this Document: Summary Letter on Current Research and Predictive Ability on Transient CHF

Type of Document: Technical

Author(s): C. C. Tsai

Date of Document: July 1980

Responsible NRC Individual and NRC Office or Division: Y. Y. Hsu
NRC-RSR

This document was prepared primarily for preliminary or internal use. It has not received full review and approval. Since there may be substantive changes, this document should not be considered final.

EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C.
Under DOE Contract No. DE-AC07-76ID01570
NRC FIN No. A6278

INTERIM REPORT

CONTENTS

SUMMARY LETTER ON CURRENT RESEARCH AND PREDICTIVE ABILITY ON TRANSIENT CHF

THE ROLE OF CHF PREDICTION IN REACTOR SAFETY ANALYSIS

The large break loss-of-coolant accident (LOCA) in a light water reactor is highly unlikely, however, the design basis for current water reactor systems includes the consideration of its occurrence. Safety features such as emergency core cooling systems (ECCS) are installed to provide sufficient core cooling to prevent massive core damage and fission gas release. The criterion for acceptance of ECCS performance is that the maximum cladding temperature reached during a postulated LOCA be less than 1478 K. For a large break LOCA, reactor shutdown immediately follows as a consequence of moderator voiding. As early as 0.1 seconds after shutdown, the core power is reduced to less than 10% of initial value. Therefore, because of high temperature and heat capacity of the fuel pellet, the heat to be removed from the fuel rods during the early blowdown phase stems mainly from the stored energy in the fuel rods at the onset of the accident. On the other hand, the core coolant inventory, which strongly affects the heat removal from the fuel rods, decreases until ECCS injection and core refill. The time at which effective cooling of the cladding surface ceases (or the boiling crisis) after blowdown, is a relative measure of the amount of the stored energy removed. Therefore, this time strongly influences the peak cladding temperature during the accident. The boiling crisis in forced convective boiling is characterized by a temperature excursion of the heated wall. The maximum heat flux just before the boiling crisis is called the critical heat flux (CHF). The two boiling crisis mechanisms generally identified are the subcooled or low-quality crisis, characterized by a transition from nucleate boiling to film boiling in the bubbly flow regime, and the high quality dryout of the liquid film adhering to the heated surface in the annular flow regime. The time at which CHF occurs, after the transient initiation, is called time-to-CHF and, as has been explained, is an important parameter in LOCA analysis.

As mentioned, one of the CHF mechanisms is the transition from nucleate boiling to film boiling. Transition boiling correlations are generally keyed to CHF, therefore, the CHF value affects post-CHF heat transfer and consequently the temperature of the heated wall. The wall temperature in turn affects the surface rewetting behavior.

EXPERIMENTAL STUDIES ON BLOWDOWN THERMAL-HYDRAULIC RESPONSE AND CORE HEAT TRANSFER

To understand the system response which strongly affects the core heat transfer during a reactor blowdown, studies have been performed in small-scale test facilities. Some of the facilities used recently in the study of the core thermal response to a hypothetical LOCA are Semiscale (at INEL), Thermal-Hydraulic Test Facility (THTF, at ORNL), Two Loop Test Apparatus (TLTA, at GE San Jose), and the Freon Test Facility (at ANL). These test programs are briefly described below.

Semiscale Test Program[1]

The Semiscale Mod-1 1-1/2 loop system is a 1/50-scale model of the LOFT facility which is about 1/50 the size of current PWR power plants. The Mod-1 core is a 40 rod bundle, and the heated length of each rod is 167 cm with 10 power steps simulating the axial power profile in the LOFT reactor core. Among the test series completed and of interest for this study are

(1) Blowdown Heat Transfer Tests (Test Series 2)

(2) Special Test Series (Test Series 29)

(3) LOFT Counterpart Tests (Test Series 6)

(4) Baseline ECC Tests (Test Series 4)

The Blowdown Heat Transfer Test Series consisted of ten tests in which eight were double-ended 200% cold-leg break tests. All the 200% cold-leg break tests exhibited early CHF in the lower part of the core within one second and delayed CHF in the upper part.[2,3] In the 200% hot-leg break test, delayed CHF was observed at close to 24 seconds into the transient at mid-core (high power zone).[4] The Special Test Series consisted of three tests.[5,6,7] The first test studied the effects of asymmetrical break configuration with 50% inlet - 150% outlet area ratio on core thermal response. The core flow reversal was significantly smaller than that in the 200% cold-leg break tests. However, the core thermal response was not appreciably different in the early transient. In this test, the core flow returned to the upward direction at about six seconds into the transient, and post-CHF heat transfer increased significantly. The second and third tests were conducted at a lower system pressure at about 12.2 MPa to study the effect of system pressure on CHF. One of these tests had a flat core radial power profile and the other had a peaked radial power profile. The test results indicated that better core cooling during blowdown occurred at reduced system pressure. The LOFT counterpart tests were conducted to provide information on the expected severity of LOFT nuclear tests. In the last test,[8] the radial core power distribution was similar to that in a LOFT rod bundle. During this test, three thermocouples mounted on high-power rods exhibited early CHF within 1 second into the transient, but were quickly rewet within 4 seconds into the transient. The delayed CHF was observed at the middle and lower portions of the core at about 3.5 to 4 seconds, depending on rod power. The Baseline ECC Test Series was conducted to study the integral blowdown-reflood response of the system. One of the tests (S-04-6) was conducted to study the effect of an unpowered rod on DNB behavior of adjacent rods.[9] The results indicated that an unpowered rod delayed the CHF of adjacent rods. As a result of delayed CHF, the peak cladding temperature was up to 195 K lower.

THTF Program[10]

The Thermal-Hydraulic Test Facility (THTF) is a small-scale experimental system which simulates a PWR. THTF contains a full length, electrically heated rod bundle consisting of 49 rods. Each rod has a stepped, chopped cosine power profile. Depressurization is accomplished by simultaneous breaks at the inlet and outlet to the test section. Some of the tests and important findings of interest to this study are briefly described below.

Only the last three tests in the first test series had sufficient power to produce early CHF.[11] Test 104 had a 50% inlet - 50% outlet break area. The core was maintained at full power until 2 seconds after break initiation and then was reduced to zero power. CHF occurred at about 0.4 seconds into the transient in the middle and lower core. Test 103 was identical to Test 104 except the break was 60% outlet - 40% inlet. Early CHF was observed within 0.4 to 0.6 seconds after break initiation. Test 105 was identical to Test 103 except that starting at 2 seconds into the transient the core power was exponentially decayed from full power with a 0.45 second time constant until 6 seconds into the transient, when the power was reduced to zero. The time-to-CHF was 0.4 to 0.6 seconds as in Test 103; however, the maximum cladding temperature was 35 K higher. The occurrence and propagation of CHF were little affected by the power decay.

Test Series II,[12] consisting of 8 tests, was conducted to investigate the effects on the transient thermal-hydraulic behavior in THTF of (1) outlet fluid subcooling, (2) bundle power, and (3) the presence of inactive rods. The steady state outlet subcooling in THTF was varied by adjusting valve settings to vary the mass flow rate through the test section. The more opening of flow control valves not only increased the period but also delayed the timing of low core flow with high quality which occurred early in the transient. As a result, CHF occurred later and more of the bundle experienced CHF. This is contrary to what might be expected. If all other factors remain the same, one might expect less CHF as a result of increased subcooling. The results from this test series also indicated that a 25% increase in core power had no significant effect on THTF hydraulic response, but its direct effect on rod surface temperature is more important. Increased rod power results in earlier CHF and more of the bundle experiencing DNB. One of the tests was conducted with two failed rods in addition to the two inactive rods present in Test Series II. In this test, the effects of inactive rods on DNB behavior was evidently shown by the thermocouples at 226 cm from the bottom of the heated length (near the top of the high power step). In this test, some of the rods adjacent to inactive rods experienced CHF at this elevation; however, all the rods which did not experience CHF were adjacent to inactive rods. In a similar test in the first test series without inactive rods, all rods experienced CHF at this elevation. However, the inactive rod effect was not evident in the lower portion of the core, where the high enthalpy of the fluid and mixing may obscure the inactive rod effect.

TLTA Program[13]

The General Electric, Two Loop Test Facility was designed to simulate BWR system characteristics as closely as possible on a real time basis. The test section used in the Blowdown Heat Transfer Test Series is a small-scale model simulating a BWR/4 vessel and containing a 7 x 7 rod bundle. The 7 x 7 rod bundle later was replaced by an 8 x 8 bundle in the Blowdown/Emergency Core Cooling (BD/ECC) Test Series. The configuration of the test section was also modified step-by-step to eventually simulate a BWR/6 system.

The Blowdown Heat Transfer Test Series[14] consisting of 12 tests was conducted to investigate: (1) the time-to-CHF, (2) the hydrodynamics of lower plenum flashing and its influence on bundle thermal response, and (3) post-CHF and lower plenum flashing heat transfer. In the TLTA blowdown tests, the timing of the key hydraulic events such as jet pump suction uncovery, recirculation suction line uncovery, and lower plenum flashing strongly affected the core thermal response. The elevations of the jet pump suction, the recirculation line suction, the jet pump discharge, and the core inlet, therefore, all affected the core thermal response, including CHF. The important findings during the Blowdown Heat Transfer Tests were: (1) The system response was insensitive to bundle power variations from 3 to 6.5 MW. (2) CHF generally occurred after the lower plenum flashed. However, in peak power (6.09 MW) and overpower (6.5 MW) tests CHF occurred due to exceeding critical power while the mixture level remained above the bundle. (3) Heat transfer to single phase steam, which was generated beneath the mixture level both by flashing and by heat transfer from the bundle was the predominant cooling mechanism in the post-lower plenum flashing period. (4) Two-phase fluid within the bundle was maintained due to countercurrent flow limiting at the core inlet. Steam updraft from the lower plenum prevented the complete draining of liquid from the core region. (5) Within the range of this study, a larger break size produced more steam updraft which in turn prevented the liquid from falling back until later during the transient and delayed CHF.

The BD/ECC Test Series[15] was conducted to extend the information obtained from the Blowdown Heat Transfer Test Series. The scaling reference BWR was the BWR/6, 624 bundle plant. The BD/ECC Test Series used several test section configurations, because of test bundle change and scaling basis change, permitting in a logical progression, a comparison of data obtained in the previous 7 x 7 blowdown heat transfer tests with those of the 8 x 8 BD/ECC tests. The first phase of the BD/ECC program is the 8 x 8 Blowdown Heat Transfer. There was no ECC operation and the change of the scaling basis from BWR/4 to BWR/6 was stepwise. The first two tests in this phase were conducted to compare the average (4.5 MW) and peak (6.09 MW) power 8 x 8 bundle heat transfer data with corresponding 7 x 7 bundle data for the same BWR/4 TLTA configuration. These tests showed that substitution of an 8 x 8 bundle for the 7 x 7 bundle resulted in essentially the same TLTA system response. Both the 7 x 7 and 8 x 8 peak power tests experienced initial CHF at approximately the same time. The peak cladding temperatures in the 8 x 8 bundle test were substantially lower due to lower surface heat flux. Unlike the 7 x 7 bundle, the 8 x 8 bundle did not encounter any sustained dryout in the peak power test. The rest of the tests were conducted with modified TLTA configurations. First the break area, core, lower plenum, guide tubes, separation region, and downcomer were scaled to the BWR/6 on volume scale to form the "hybrid" BWR/6 system. Finally, the jet pump configuration was also scaled to BWR/6. The configuration changes, particularly the changes in break area, the extension of jet pump discharge into the lower plenum and below the core inlet, and the raising of the jet pump suction inlet all affected the timing of the key hydraulic events mentioned and, subsequently, the core thermal response.

ANL Freon Test Facility[16]

A small-scale apparatus simulating a PWR was constructed at the Argonne National Laboratory (ANL) to investigate transient CHF under a wide range of test conductions. Three test sections having symmetric stepped, uniform, or inlet-skewed axial heat flux profiles were constructed. The tests conducted included flow decay transient and blowdown with flow reversal.

The first series of flow transient tests was conducted with the uniform heat flux test section with 4%/sec linear flow decay and constant power. CHF was first detected nearest to the exit at more than 10 seconds, which corresponded to a very low inlet flow rate. The second test series consisted of inlet flow blockage tests conducted in the outlet-skewed heat flux test section (the reverse of inlet-skewed heat flux test section). In most cases, the inlet velocity decayed to zero in less than 0.5 seconds and pressure decreased only moderately during the transients. In general, CHF occurred first at the second to the last heat flux step at times in the range from 1.4 to 4.2 seconds. Exit break test was conducted one each with the symmetric step heated test section and the inlet skewed heat flux test section. CHF was first observed nearest the exit and propagated upstream. A number of blowdowns were conducted in the uniformly heated test section. One of the tests was an inlet break test where CHF was not observed until

8 seconds into blowdown. Then it was observed to propagate from the bottom to the top of the test section. The other tests were double-ended break tests with very early CHF. In general CHF was observed to propagate from top to bottom.

COMPARISON OF COMPUTED CODE PREDICTION AND EXPERIMENTALLY MEASURED TIME-TO-CHF

A number of comparisons of computer code predictions with core blowdown data has been made as part of blowdown heat transfer research. The following are some of the typical results.

Davis made a number of comparisons of RELAP4/MOD6[17] core component calculations with core blowdown data in Semiscale and THTF.[18] Thermal boundary conditions were not specified, instead the heat transfer package in RELAP4/MOD6 was used. The appropriate measured data were used for hydraulic boundary conditions. The recommended CHF correlations in RELAP4/MOD6 are the W-3 correlations[19] for the subcooled regime, the Hsu-Beckner correlation[20] for the saturated high-flow regime, and the Griffith's modified Zuber correlation[21] for the saturated low-flow regime. In this study, the CHF in the middle and upper core for Semiscale Test S-06-5 was predicted 2 to 5 seconds too early. Consequently, peak cladding temperatures were overpredicted by as much as 300 K. Similar results were obtained for Semiscale Tests S-04-5, S-04-6, and S-06-6. Substitution of the GE transient correlation[22] for the Hsu-Beckner correlation improved the prediction for the upper core but resulted in unrealistic rewets in the lower core. For THTF Test 105, CHF in the upper core was calculated at 0.5 seconds but sustained CHF was not observed until 2 seconds into the transient. Davis also used similar procedures to calculate the core response in THTF Test 177, which was designated the INEL code verification test.[23] Despite uncertainties in the measured vertical outlet mass flow rate, which was used as boundary condition in the core model, the time-to-CHF was well predicted throughout the core. Since the calculated flow at the midsection of the core was near stagnation at the time of CHF, the CHF could be expected to be DNB type and to be predicted by the modified Zuber correlation. A sensitivity-study found that the stagnation time was not significantly affected by the input flow uncertainty. Besides, the rapid power increase which strongly affected the time-to-CHF between 4 to 5 seconds into the transient was flow independent. As a result, CHF was well predicted.

Snider calculated local conditions in the Semiscale core for Test S-02-9 using the COBRA-IV-I[24] with measured hydraulic boundary conditions.[25] The thermal boundary condition was heat flux calculated with an inverse heat conduction code from thermocouple data. Seven correlations, including MacBeth,[26] Biasi,[27] Barnett,[28] B&W-2,[29] W-3, LOFT,[30] and GE correlations were used with the instantaneous local conditions to predict CHF onset. The GE correlation was found to have the closet prediction and MacBeth, Biasi and B&W-2 were considered to be adequate. Dryout was considered to be the mechanism causing CHF.

Dallman used RELAP4/MOD6 to calculate the TLTA system response during Tests 6006 and 6007.[31] The standard RELAP4/MOD6 heat transfer package was used. The W-3, Hsu-Beckner, and modified Zuber DNB option was used. There was generally good agreement between calculated and measured hydraulic events. However, the calculations predicted CHF too early and fewer rod surface rewets, and subsequently higher peak cladding temperature. An abnormality in the predicted CHF distribution along the heated length was also observed. At 3.05 m elevation, CHF was observed to occur within 1.5 seconds into the transient, however, it was not predicted until 7 seconds into the transient. For other elevations, CHF was predicted to occur earlier than measured.

Leung at ANL developed a computer program, CODA, based on homogeneous equilibrium model, neglecting sonic effects.[16] Eight CHF correlations were included in the CODA program as CHF options. The eight correlations are the Bowning,[32] Biasi, CISE,[33] B&W-2, Condie Mod7,[34] GE, Hsu-Beckner, and Griffith's modified Zuber correlations. Test results from

Semiscale, THTF, PBF LOC-11 test, Columbia University Blowdown Heat Transfer Facility, and ANL Freon Facility were used to test the predictive ability of each CHF correlation. It was concluded that the round tube correlations, like the Bowring, Biasi, and CISE correlations, could be applied to rod bundles, and particularly predicted early ($\leq$1 second) CHF well. The B&W-2 correlation obtained from mostly low quality rod-bundle data above 13.8 MPa was inadequate for blowdown analysis. The Condie Mod7 correlation was found to perform marginally. The GE correlation performed adequately, except for Semiscale data. The Hsu-Beckner and Griffith's modified Zuber correlation generally gave very conservative predictions. That is, more CHF was predicted and the predicted CHF was generally earlier than measured. However, if a mass velocity criterion, for instance an upper limit of 100 kg/m^{2}sec and a lower limit of -240 kg/m^{2}sec, was imposed, the modified Zuber correlation predicted delayed CHF well.

CONCLUSIONS AND RECOMMENDATIONS

(1) It has been demonstrated in various test facilities that the time-to-CHF strongly affects the peak cladding temperature during a hypothetical LOCA accident and therefore is an important parameter in reactor safety analysis.

(2) Within the range of interest, the effect of the core power level on the system hydraulic behavior, which in turn affects core thermal response, is not significant in comparison with the direct effect of core power on core thermal response.

(3) As demonstrated by the TLTA Tests, CHF in the heater core could occur during blowdown as a result of dryout or transition from nucleate boiling to film boiling, depending on the core power level at the onset of blowdown.

(4) In general, the system response during blowdown tests was qualitatively well understood. However, quantitative predictions, particularly of core thermal response, was somewhat limited. Inaccuracy in the calculated local hydraulic conditions and the limited predictive capability of various CHF correlations caused the difficulties in predicting the time-to-CHF and peak cladding temperature during the transients. If the local hydraulic conditions in a particular test were accurately known through measurements or calculations, a meaningful comparison of the predictive abilities of various CHF correlations could be made.

(5) Since it is difficult, if not impossible, to accurately measure local conditions in the core, the local conditions have to be calculated. The error in the calculated local conditions in the core can be reduced by utilizing the measured data near the core, like the core inlet flow measurement in the Semiscale tests.

(6) A core calculation model supplied with measured boundary conditions is ideal for testing core heat transfer analysis schemes, since lengthy system hydraulic computation can be avoided, besides reduced errors in the calculated local conditions in the core.

(7) The CHF correlations widely used in reactor safety analysis do not perform well throughout the entire range of blowdown conditions. In general, the performance of the more widely used CHF correlations can be characterized as follows:

Griffith's Modified Zuber Correlation	Capable of predicting delayed CHF caused by more tranquil flow transient (DNB type CHF). Performs well under low flow and countercurrent flow condition. Suggested mass flow range for its application is $-240\ kg/m^2scc \lesssim G \lesssim 100\ kg/m^2sec$

CISE and Biasi	Both are round tube correlations. Predict early CHF (dryout type CHF) in rod-bundle tests rather well. However, failed in many cases to predict delayed CHF. More suitable for predicting CHF under fast flow transient throughout the pressure range of interest to reactor safety analysis.
Bowring	A round tube correlation. Similar to CISE and Biasi correlations in predicting rod-bundle tests but not as well.
Hsu-Beckner	Capable of prediting delayed CHF. However, in most cases the predicted CHF is somewhat earlier than measured.
Condie Mod7	Generally, the predicted CHF is substantially earlier than measured. However, the general trend of CHF propagation along heated length was often well predicted. The correlation also tends to predict delayed CHF better. Some modifications may improve its performance.
B&W-2	The correlation was obtained from low quality data at high pressure. The suggested range of application is; the quality $x \lesssim 0.2$ and the pressure greater

	than 13.8 MPa. The working range is limited for blowdown analysis.
GE	Generally perform adequately in predicting early CHF (≤1.5 sec), but less well in predicting delayed CHF, particularly for Semiscale tests. However, note that this correlation was intended to describe the lower limit of available data.

Since there is no single CHF correlation that performs well throughout the entire range of blowdown conditions, the combination of two or more correlations could be used. This has been done in RELAP4 and TRAC computer codes. At present, the combination of the Griffith's modified Zuber correlation for low flow and the CISE or Biasi correlation for high flow, as has been done in TRAC-P1A computer code, is a good choice for best estimate calculations.

REFERENCES

1. L. J. Ball, et al., Semiscale Program Description, TREE-NUREG-1210 (1978).

2. J. M. Cuzzuol, Thermal-Hydraulic Analysis of the Semiscale Mod-1 Blowdown Heat Transfer Test Series, ANCR-NUREG-1287 (1976).

3. T. K. Larson, Core Thermal Response During Semiscale Mod-1 Blowdown Heat Transfer Tests, ANCR-NUREG-1285 (1976).

4. H. S. Crapo, M. F. Jensen, and K. E. Sackett, Experimental Data Report for Semiscale Mod-1 Test S-02-1 (Blowdown Heat Transfer Test), ANCR-1231 (1975).

5. H. S. Crapo, M. F. Jensen, and K. E. Sackett, Experimental Data Report for Semiscale Mod-1 Test S-29-1 (Integral Test with Asymmetrical Break), ANCR-NUREG-1327 (1976).

6. H. S. Crapo and K. E. Sackett, Experimental Data Report for Semiscale Mod-1 Test S-29-2 (Integral Test from Reduced Initial Pressure), ANCR-NUREG-1328 (1976).

7. H. S. Crapo, B. L. Collins, and K. E. Sackett, Experimental Data Report for Semiscale Mod-1 Test S-29-3 (Integral Test from Reduced Initial Pressure), ANCR-NUREG-1329 (1976).

8. V. Esparza and K. E. Sackett, Experimental Data Report for Semiscale Mod-1 Test S-06-6 (LOFT Counterpart Test), TREE-NUREG-1126 (1977).

9. J. M. Cozzuol, Thermal-Hydraulic Analysis of Semiscale Mod-1 Integral Blowdown-Reflood Tests (Baseline ECC Test Series), TREE-NUREG-1077.

10. Project Description ORNL PWR Blowcown Heat Transfer Separate-Effects Program - Thermal-Hydraulic Test Facility (THTF), NUREG/CR-0104.

11. W. G. Craddick et al., PWR Blowdown Heat Transfer Separate-Effects Program Data Evaluation Report - System Response for Thermal-Hydraulic Test Facility Test Series 100, ORNL/NUREG-19 (1977).

12. C. B. Mullins et al., PWR Blowdown Heat Transfer Separate - Effects Program Data Evaluation Report - THTF Test Series II, NUREG/CR-0539 (1979).

13. R. J. Muzzy, Preliminary BWR Blowdown/Emergency Core Cooling Program Plan, GEAP-21255 (1976).

14. R. Muralidharan, BWR Blowdown Heat Transfer Final Report, GEAP-21214 (1976).

15. W. S. Hwang, et at., BWR Blowdown/Emergency Core Cooling Program 64-Rod Bundle Blowdown Heat Transfer (8 x 8 BDHT) Final Report, GEAP-23977 (1978).

16. J. C. Leung, Transient Critical Heat Flux and Blowdown Heat Transfer Studies, Thesis, Northwestern University (1980).

17. RELAP4/MOD6 A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactor and Related Systems User's Manual, EG&G Report, CDAP-TR-003 (1978).

18. C. B. Davis, Comparison of RELAP4/MOD6 with Core Blowdown Data, EG&G Report, CVAP-TR-78-012 (1978).

19. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," J. Nucl. Energy, 6, 241 (1967).

20. Y. Y. Hsu and W. D. Beckner, "A Correlation for the Onset of Transient CHF," Cited in L. S. Tong and G. L. Bennet, "NRC Water Reactor Safety Research Program," Nuclear Safety, 18, 1 (1977).

21. P. Griffith, C. T. Avedisian and J. F. Walkush," Countercurrent Flow Critical Heat Flux," Presented at National Heat Flux Conference, San Francisco (1975).

22. G. M. Roy, "Getting More out of BWR's," Nucleonics, 24, 11 (1966).

23. C. B. Davis, Comparison of RELAP4/MOD6 to THTF Test 177, EG&G Report, CAAP-TR-049 (1979).

24. C. L. Wheeler et al., COBRA-IV-I: An Interim Version of COBRA for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Element and Core, BNWL-1962 (1976).

25. D. M. Snider, Analysis of the Thermal-Hydraulic Behavior Resulting in Early Critical Heat Flux and Evaluation of CHF Correlations for the Semiscale Core, TREE-NUREG-1073 (1977).

26. R. V. MacBeth, Burnout Analysis Part 3, The Low Velocity Burnout Regime, AEEW-R222 (1963).

27. L. Biasi et al., "Studies on Burnout" Part 3, Energia Nuclear, 14, S (1967).

28. P. G. Barnett, "A Correlation of Burnout Data for Uniformly Heated Annuli and its Use for Predicting Burnout in Uniformly Heated Rod Bundles," ASME Conference, Los Angeles, California (November 18, 1969).

29. J. S. Gellersted, et al., "Two Phase Flow and Heat Transfer in Rod Bundles," ASME Conference, Los Angeles, California (November 18, 1969).

30. S. A. Ende and R. C. Gottula, Evaluation and Results of LOFT Steady-State Departure from Nucleate Boiling Tests, TREE-NUREG-1043 (1977).

31. R. J. Dallman, RELAP4/MOD6 Data Comparison of BWR-BD/ECC Tests 6006 and 6007, EG&G Report, EGG-CAAP-5037 (1979).

32. R. W. Bowring, A Simple but Accurate Round Tube, Uniform Heat Flux Dryout Correlation Over the Pressure Range 0.7 - 17 MN/m^2, AEEW-R789 (1972).

33. S. Berlolettie, et al., "Heat Transfer Crisis with Steam-Water Mixture," Energia Nucleare, 12, 121 (1965).

34. K. G. Condie, et al., Development of the MOD7 CHF Correlation, EG&G Report, PN-181-78 (1978).