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EFFECT OF GEOMETRY AND OTHER PARAMETERS ON
BOTTOM FLOODING HEAT TRANSFER ASSOCIATED
WITH NUCLEAR FUEL BUNDLE SIMULATORS

By

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Idaho Falls, Idaho — 83401

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[a] Now Aerojet Nuclear Company.

ABSTRACT

Full-length emergency cooling heat transfer (FLECHT) experiments have been performed on boiling water reactor (BWR) and pressurized water reactor (PWR) fuel bundle simulators. These experiments were performed in the BWR-FLECHT and PWR-FLECHT Tests Projects conducted by Idaho Nuclear Corporation^[a] as part of the water reactor safety program of the U. S. Atomic Energy Commission. These bundle simulators, using resistance heaters electrically powered to simulate nuclear decay heat, were employed in typical reactor geometries to obtain experimental data over a wide range of test parameters. These data can be used to evaluate analytical methods used for calculating the heat transfer capabilities of reactor emergency cooling systems under simulated loss-of-coolant accident conditions. One type of emergency cooling system common to both BWR and PWR reactors which the FLECHT projects investigated is the bottom flooding system.

This report, which is based on a master's thesis for the University of Idaho, develops the relationship between the heat transfer results of bottom flooding tests from the two FLECHT projects. The results of 11 BWR-FLECHT and 25 PWR-FLECHT bottom flooding tests are presented in the form of transient heat transfer coefficients at different axial and radial locations. Selected data from thermocouples embedded in the cladding and from which these coefficients were calculated are also included. These 36 tests investigated heat transfer over the following ranges of parameters:

Peak Power	0.69 to 1.40 kW/ft
Initial Bundle Temperature	800 to 2200°F
Cold Flooding Rate	1 to 10 in./sec
Inlet Coolant Subcooling	25 to 295°F
Chamber Pressure	15 to 330 psi

The BWR-FLECHT experiments primarily studied the effect of pressure on bottom flooding. The PWR-FLECHT project investigated variation of all of these parameters.

This report qualitatively compares the effects of variation of the operating parameters on the localized heat transfer from the heater rod cladding. Representative tests are compared using the known effects of parametric variation to

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normalize differences in bundle operating conditions. This comparison is thus reduced to the effect of lattice geometry, that of a BWR versus that of a PWR, on the heat transfer during bottom flooding. This work shows that, although the diameter and spacing of the BWR simulator rods were greater than those of the PWR simulator rod, heat transfer coefficients are approximately the same for similar operating conditions.

SUMMARY

Full-length emergency cooling heat transfer (FLECHT) experiments have been performed on boiling water reactor (BWR) and pressurized water reactor (PWR) fuel bundle simulators. These experiments were performed by General Electric Company in the BWR-FLECHT Test Project and by Westinghouse Electric Corporation in the PWR-FLECHT Test Project, and were directed by Idaho Nuclear Corporation^[a] as part of the water reactor safety program of the U. S. Atomic Energy Commission. These bundle simulators, using resistance heaters electrically powered to simulate nuclear decay heat, were employed in typical reactor geometries to obtain experimental data over a wide range of test parameters. These data can be used to evaluate analytical methods used for calculating the heat transfer capabilities of reactor emergency cooling systems under simulated loss-of-coolant accident (LOCA) conditions. One type of emergency cooling system common to both BWR and PWR reactors which the FLECHT projects investigated is the bottom flooding system. Thirty-six tests conducted with bottom flooding of stainless steel-clad heater rod bundles are reviewed here. These 36 tests investigated heat transfer over the following ranges of parameters:

Peak power	0.69 to 1.40 kW/ft
Initial bundle temperature	800 to 2200°F
Cold flooding rate	1 to 10 in./sec
Inlet coolant subcooling	25 to 295°F
Chamber pressure	15 to 330 psi

The BWR-FLECHT experiments primarily examined effects of pressure, whereas the PWR-FLECHT tests examined all five variables.

The objective of this report is to develop a relationship between the bottom flooding information from the PWR-FLECHT project and that from the BWR-FLECHT project. The development of such a relationship would make the findings of one project useful in the other.

The specific objective of this report is to investigate the effect of variations in operating and design parameters, especially variations in geometry, on the

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overall heat transfer associated with the BWR and PWR nuclear bundle simulators operating under emergency core cooling (ECC) conditions. The basic difference between the two projects lies in the difference in lattice geometry of the simulated fuel rods. BWR-FLECHT rods are larger in diameter than PWR-FLECHT rods and the rod spacing is greater in BWR-FLECHT.

Calculations of the local heat transfer coefficients from data measured in this project were accomplished by the DATAR computer code developed by Westinghouse Electric Corporation. The code performs a transient conduction calculation based on the known temperature of the inside diameter of the cladding in PWR-FLECHT and the middle of the cladding in BWR-FLECHT. The DATAR code calculates the rod surface temperature, surface heat flux, and surface heat transfer coefficient as functions of time, by using the known temperature, the known electrical power generation rate at the thermocouple, and the temperature-dependent material properties of the rod at the thermocouple and by assuming the coolant is at the saturation temperature. The code solves a one-dimensional energy balance and determines the total heat transfer coefficient (the sum of radiation and convection heat transfer coefficients) as though the heat transfer were all due to convection. Axial conduction has been shown to be insignificant.

The calculated coefficients from the two FLECHT projects are compared. Included is a discussion of the variations in heat transfer resulting from variations in the test parameters of bundle power, cladding temperature at test initiation, cold flooding rate, chamber pressure, and inlet coolant subcooling. Radial and axial effects are described. The heat transfer coefficient of the BWR-FLECHT project is correlated with that of the PWR-FLECHT project at the six-foot elevation of the hottest rods.

To simplify comparison from test to test, a single thermocouple location was selected for each bundle. The midplane (six-foot elevation) on the hottest rod (Rod 17) represents the location of highest temperature in the BWR-FLECHT bundle at transient initiation and is thus the point of greatest interest in studies of ECC system effectiveness and behavior. Similarly, in the PWR-FLECHT 7 x 7 array, this location is the six-foot elevation on inner Rod 3D and in the 10 x 10 array it is the six-foot elevation on inner Rod 5F.

The following general effects of parametric variation were the same for both BWR-FLECHT and PWR-FLECHT. Increasing power increased quench time with a decrease in heat transfer coefficient at all times. Initial temperature variation had little effect on either quench time or heat transfer coefficient behavior. An increase in the flooding rate increased the heat transfer coefficient and shortened the quench time. An increase in pressure increased the heat transfer coefficient and decreased the quench time within the maximum pressure limit. Additional subcooling increased the heat transfer coefficient and decreased quench time.

PWR-FLECHT Test 5730 and BWR-FLECHT Test SS4-20 are compared on the basis of heat transfer coefficient. These tests performed under similar conditions with the primary exception of peak power which was 0.74 kW/ft for the BWR-FLECHT bundle and 1.24 kW/ft for the PWR-FLECHT bundle. To equate this power difference, the heat transfer coefficients for the hot rod were plotted on a dimensionless time base. Comparison of the heat transfer coefficients at the six-foot elevation shows that both bundles exhibited the same characteristics with the PWR-FLECHT data consistently higher than the BWR-FLECHT values. These results indicate that the PWR-FLECHT time-averaged heat transfer coefficient is 1.33 times higher at the six-foot elevation than the BWR-FLECHT time-averaged coefficient when compared on a dimensionless time base.

The liquid front in the PWR-FLECHT tests exhibited the characteristic of an initially high quench rate not evident in the BWR-FLECHT test data. PWR-FLECHT Test 5730 produced initial quench velocities during the first few seconds of the transients that approximated the cold flooding rates. Near the two-foot elevation, where the rod surface temperature reached the Liedenfrost temperature, film boiling commenced and the rate of quench front progression up the bundle was reduced to a rate dependent on the test parameters. For the BWR-FLECHT Test SS4-20, the film boiling regime existed from the very bottom of the heater section of the rod because of the high temperature (800°F) at the zero elevation. The quench velocity was immediately established at a rate dictated by the bundle power. Thus, shortly after test initiation, Test 5730 experienced wetting up the rod to a greater height than did Test SS4-20. In general, the wetting front was two feet higher in PWR-FLECHT. The greater

coolant depth in PWR-FLECHT also resulted in the higher elevations in PWR-FLECHT being closer to the quench front than in BWR-FLECHT. These higher elevations thus experienced higher heat transfer due to the lower void fraction nearer the front.

An interesting phenomenon is evidenced by the temperature profiles of Tests 5730 and SS4-20. As the coolant entered the bundle and cooled the lower end, the point of maximum temperature (hot spot) began to shift upward indicating that the cooling was less efficient at the higher elevations (midplane and above). The effect of hot-spot shift was more pronounced in the PWR-FLECHT results. Although the midplane was not the hottest spot throughout the entire transient in either project, the use of data from the six-foot elevation covers the initial and highest temperature segment of each test.

An estimate of the Reynolds number was made to determine whether turbulent flow was established in the two-phase mixture in the bundles. No local velocity, density, and flow rate measurements were made in the FLECHT tests; therefore, overall data were used to determine the bulk Reynolds number. All tests were found to be well into fully developed turbulent flow.

Many two-phase flow correlations relating heat transfer for turbulent fluid flow in passages are based on the Dittus-Boelter equation for heating a fluid:

$$Nu = \frac{h D_h}{K} = 0.023 Re^{0.8} Pr^{0.4}$$

where

Nu = Nusselt number

h = heat transfer coefficient

D_h = hydraulic diameter

K = coefficient of thermal conductivity

Re = Reynolds number

Pr = Prandtl number.

This equation is rearranged and the heat transfer coefficients are defined for the BWR-FLECHT and PWR-FLECHT bundles. A ratio of these heat transfer coefficients is then made. Cancellation of equal terms and substitution of

known values into the ratio result in a correlation of heat transfer coefficients that is dependent on bundle hydraulic diameter, lattice size, and fluid velocity. The subscripts B for BWR-FLECHT and P for PWR-FLECHT are used to represent the respective FLECHT terms in the ratio of coefficients:

$$\frac{h_{PWR}}{h_{BWR}} = \frac{C_{1P}}{C_{1B}} \left(\frac{D_{hB}}{D_{hP}} \right)^{0.2} \left[\frac{(V_f - V_q)_P}{(V_f - V_q)_B} \right]^{0.8}$$

where

- C_1 = lattice constant
- D_h = hydraulic diameter
- V_f = cold flooding rate
- V_q = quench front velocity.

The hydraulic diameters and lattice constants are derived from the bundle geometry. The cold flooding rate is a measured quantity and the quench front velocity is derived from the slopes of the quench elevation versus time curves derived from FLECHT data. When data are lacking, if the flooding rates are equal, the quench velocities can be assumed equal.

The use of the correlation relating PWR-FLECHT and BWR-FLECHT heat transfer coefficients generally underpredicts the heat transfer coefficient ratio somewhat. This underprediction is believed to be the result of the low axial power peaking factor in the lower end of the PWR-FLECHT bundles which allowed the quench front to rise higher in these bundles.

The theoretical correlation can be used to predict time-averaged heat transfer coefficients for BWR-FLECHT by use of PWR-FLECHT heat transfer coefficients and test data, or alternately, PWR-FLECHT coefficients can be predicted from BWR-FLECHT data. The correlation accounts for flooding rate differences, but chamber pressure and coolant subcooling which exert strong influences are not accounted for and, therefore, the correlation is valid only for tests at similar pressure and subcooling. The initial bundle temperature and peak power have no appreciable effect on heat transfer and can vary over the range investigated.

Development of a relation of quench velocity to pressure and coolant subcooling is suggested to make the correlation readily adaptable for predicting the heat transfer coefficient of bundles other than those of FLECHT geometry.

CONTENTS

ACKNOWLEDGMENTS	11
ABSTRACT	111
SUMMARY	v
GLOSSARY	xxii
I. INTRODUCTION	1
II. DESCRIPTION OF BWR AND PWR NUCLEAR PLANTS	5
III. RELATED EXPERIMENTS	9
1. GENERAL ELECTRIC COMPANY EXPERIMENTS	9
2. WESTINGHOUSE ELECTRIC CORPORATION EXPERIMENTS. . .	12
3. PHILLIPS PETROLEUM COMPANY AND IDAHO NUCLEAR CORPORATION EXPERIMENTS	16
IV. DESCRIPTION OF FLECHT EXPERIMENTS	25
1. BUNDLE ASSEMBLY	25
1.1 BWR-FLECHT	25
1.2 PWR-FLECHT	30
2. TEST FACILITY	34
2.1 BWR-FLECHT	34
2.2 PWR-FLECHT	38
3. TEST PROCEDURE	42
3.1 BWR-FLECHT	42
3.2 PWR-FLECHT	43
4. TEST CONDITIONS	44
V. QUALIFICATION OF DATA AND SUPPORTING EVALUATION	47
1. DATA DEFINITION AND REDUCTION	47
2. EFFECT OF RADIAL POSITION	51
2.1 BWR-FLECHT	51
2.2 PWR-FLECHT	53
3. EFFECT OF AXIAL POSITION	57

3.1 BWR-FLECHT.....	57
3.2 PWR-FLECHT.....	60
4. EFFECTS OF PARAMETRIC VARIATION.....	62
4.1 Power Variation.....	62
4.2 Test Initiation Temperature Variation.....	64
4.3 Cold Flooding Rate Variation.....	66
4.4 Pressure Variation.....	69
4.5 Subcooling Variation.....	78
VI. DISCUSSION OF RESULTS.....	82
1. COMPARISON OF HEAT TRANSFER DATA FOR BWR-FLECHT AND PWR-FLECHT.....	82
2. AXIAL POWER PROFILE EFFECTS.....	87
3. QUENCH VELOCITY COMPARISONS.....	93
VII. CORRELATION OF BWR-FLECHT AND PWR-FLECHT HEAT TRANSFER.....	95
VIII. CONCLUSIONS AND RECOMMENDATIONS.....	101
1. CONCLUSIONS.....	101
2. RECOMMENDATIONS.....	102
IX. REFERENCES.....	103
APPENDIX A -- PLOTS OF BWR-FLECHT TEMPERATURE AND HEAT TRANSFER COEFFICIENTS VERSUS TIME FOR THE HOTTEST ROD.....	105
APPENDIX B -- PLOTS OF PWR-FLECHT HEAT TRANSFER COEFFICIENT VERSUS TIME FOR THE HOTTEST ROD.....	119
APPENDIX C -- PLOTS OF BWR-FLECHT AND PWR-FLECHT HEAT TRANSFER COEFFICIENT VARIATIONS WITH RESPECT TO VARIOUS MAJOR PARAMETERS.....	147
APPENDIX D -- PLOTS OF BWR-FLECHT AND PWR-FLECHT QUENCH ELEVATION VERSUS TIME.....	177

FIGURES

1. BWR reactor assembly	6
2. PWR reactor assembly	7
3. PWR-FLECHT heat transfer coefficient versus dimensionless time	14
4. Schematic of view through port of SECHT III vessel	18
5. Sequence of development of liquid entrainment, SECHT III	19
6. Schematic of flow and heat transfer regimes, FHUST	21
7. Flow and heat transfer regimes, FHUST	22
8. Typical cladding temperature response, FHUST	23
9. Heat transfer coefficients for various inlet flooding rates, FHUST	24
10. Heater axial power profiles, BWR- and PWR-FLECHT	26
11. Bundle and test chamber cross-section, BWR-FLECHT	28
12. BWR-FLECHT test bundle	29
13. Schematic of heater rod, BWR-FLECHT	30
14. 10 x 10 PWR-FLECHT test section	31
15. 7 x 7 PWR-FLECHT test section	32
16. Schematic of heater rod, PWR-FLECHT	33
17. BWR-FLECHT test facility -- schematic	35
18. BWR-FLECHT test section and superimposed BWR reactor geometry	36
19. BWR-FLECHT power decay	37
20. PWR-FLECHT test facility -- schematic	39
21. PWR test chamber and bundle -- elevation	40
22. PWR-FLECHT power decay	41
23. Typical FLECHT surface temperature and heat transfer coefficient . .	48
24. BWR-FLECHT Test SS4-20 coolant temperature and pressure . . .	49

25. Radial differences in temperature and heat transfer coefficient at the six-foot elevation -- BWR-FLECHT Test SS4-10	52
26. Radial differences in temperature and heat transfer coefficient at the six-foot elevation -- BWR-FLECHT Test SS4-19	53
27. Radial differences in temperature and heat transfer coefficient at the six-foot elevation of the 7 x 7 PWR-FLECHT array -- Test 6553	55
28. Radial differences in temperature and heat transfer coefficient at the six-foot elevation of the 10 x 10 PWR-FLECHT array -- Test 4225	56
29. Effect of bundle size on heat transfer coefficient, PWR-FLECHT, six-foot elevation on the hot rod	57
30. Axial differences in temperature and heat transfer coefficient -- BWR-FLECHT Test SS4-19	58
31. Effect of water droplet entrainment on quenching at the upper elevations -- BWR-FLECHT Test SS4-11X	59
32. Effect of elevation on temperature and heat transfer coefficient -- PWR-FLECHT Test 4225	61
33. Effect of peak power on the heat transfer coefficient for the six-foot elevation on the hottest rod -- PWR-FLECHT	63
34. Variation of quench rate with power -- PWR-FLECHT, six-foot elevation on the hottest rod	64
35. Variation of heat transfer coefficient with power, dimensionless time base -- PWR-FLECHT, six-foot elevation on the hottest rod	65
36. Variation of heat transfer coefficient with power, dimensionless time base -- PWR-FLECHT, six-foot elevation on the hottest rod .	66
37. Variation of heat transfer coefficient with initial temperature, dimensionless time base -- PWR-FLECHT, six-foot elevation <u>on the hottest rod</u>	67
38. Variation of quench rate with initial temperature -- PWR-FLECHT, six-foot elevation on the hottest rod	68
39. Variation of heat transfer coefficient with cold flooding rate -- PWR-FLECHT, six-foot elevation on the hottest rod	69
40. Variation of heat transfer coefficient with cold flooding rate, dimensionless time base -- PWR-FLECHT, six-foot elevation on the hottest rod	70
41. Quench rate dependence on flooding rate -- PWR-FLECHT, six-foot elevation on the hottest rod	71

42. Effect on pressure on surface temperature and heat transfer coefficient -- BWR-FLECHT, six-foot elevation on the hottest rod	72
43. Effect of pressure on heat transfer coefficient -- PWR-FLECHT, six-foot elevation on the hottest rod	73
44. Variation of quench rate with pressure -- BWR-FLECHT, six-foot elevation on the hottest rod	74
45. Variation of heat transfer coefficient with pressure, dimensionless time base -- BWR-FLECHT, six-foot elevation on the hottest rod	75
46. Effect of subcooling on heat transfer coefficient -- PWR-FLECHT, six-foot elevation on the hottest rod	79
47. Variation of heat transfer coefficient with subcooling, dimensionless time base -- PWR-FLECHT, six-foot elevation on the hottest rod	80
48. Variation of quench rate with subcooling -- PWR-FLECHT, six-foot elevation on the hottest rod	81
49. Comparison of PWR-FLECHT and BWR-FLECHT heat transfer coefficients at the four-foot elevation on the hottest rods	83
50. Comparison of PWR-FLECHT and BWR-FLECHT heat transfer coefficients at the six-foot elevation on the hottest rods	84
51. Comparison of PWR-FLECHT and BWR-FLECHT heat transfer coefficients at the eight-foot elevation on the hottest rods	85
52. BWR-FLECHT axial power and temperature profiles at time of test initiation	88
53. PWR-FLECHT axial power and temperature profiles at time of test initiation	89
54. Axial temperature profiles for various dimensionless time, t^* , steps -- BWR-FLECHT Test SS4-20, Rod 17	91
55. Axial temperature profiles for various dimensionless time, t^* , steps -- PWR-FLECHT Test 5730	92
56. Elevation of the quench front progression versus time after flooding	93
A-1. Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-10	107
A-2. Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-11X	108

A-3. Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-12	109
A-4. Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-13X	110
A-5. Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-14R	111
A-6. Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-15Z	112
A-7. Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-16X	113
A-8. Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-17	114
A-9. Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-18	115
A-10. Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-19	116
A-11. Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-20	117
B-1. Variation of heat transfer coefficient with elevation -- PWR- FLECHT Test 0105	121
B-2. Variation of heat transfer coefficient with elevation -- PWR- FLECHT Test 0307	122
B-3. Variation of heat transfer coefficient with elevation -- PWR- FLECHT Test 0408	123
B-4. Variation of heat transfer coefficient with elevation -- PWR- FLECHT Test 0509	124
B-5. Variation of heat transfer coefficient with elevation -- PWR- FLECHT Test 0711	125
B-6. Variation of heat transfer coefficient with elevation -- PWR- FLECHT Test 0913	126
B-7. Variation of heat transfer coefficient with elevation -- PWR- FLECHT Test 1002	127
B-8. Variation of heat transfer coefficient with elevation -- PWR- FLECHT Test 1314	128
B-9. Variation of heat transfer coefficient with elevation -- PWR- FLECHT Test 1417	129

B-10. Variation of heat transfer coefficient with elevation -- PWR-FLECHT Test 1615	130
B-11. Variation of heat transfer coefficient with elevation -- PWR-FLECHT Test 1720	131
B-12. Variation of heat transfer coefficient with elevation -- PWR-FLECHT Test 2322	132
B-13. Variation of heat transfer coefficient with elevation -- PWR-FLECHT Test 4027	133
B-14. Variation of heat transfer coefficient with elevation -- PWR-FLECHT Test 4129	134
B-15. Variation of heat transfer coefficient with elevation -- PWR-FLECHT Test 4225	135
B-16. Variation of heat transfer coefficient with elevation -- PWR-FLECHT Test 4526	136
B-17. Variation of heat transfer coefficient with elevation -- PWR-FLECHT Test 4628	137
B-18. Variation of heat transfer coefficient with elevation -- PWR-FLECHT Test 5123	138
B-19. Variation of heat transfer coefficient with elevation -- PWR-FLECHT Test 5730	139
B-20. Variation of heat transfer coefficient with elevation -- PWR-FLECHT Test 6155	140
B-21. Variation of heat transfer coefficient with elevation -- PWR-FLECHT Test 6256	141
B-22. Variation of heat transfer coefficient with elevation -- PWR-FLECHT Test 6351	142
B-23. Variation of heat transfer coefficient with elevation -- PWR-FLECHT Test 6553	143
B-24. Variation of heat transfer coefficient with elevation -- PWR-FLECHT Test 6948	144
B-25. Variation of heat transfer coefficient with elevation -- PWR-FLECHT Test 7158	145
C-1. Variation of heat transfer coefficient with peak power for flooding rates of 2 and 6 in./sec -- PWR-FLECHT Rod 5F, six-foot elevation	149

C-2. Variation of heat transfer coefficient with peak power, dimensionless time base -- PWR-FLECHT Tests 4225, 4129, and 4027 at a flooding rate of 2 in./sec	150
C-3. Variation of heat transfer coefficient with peak power, dimensionless time base -- PWR-FLECHT Tests 4526, 5730, and 4628 at a flooding rate of 6 in./sec	151
C-4. Variation of heat transfer coefficient with initial temperature for flooding rates of 1 and 6 in./sec -- PWR-FLECHT, Rod 3D, six-foot elevation	152
C-5. Variation of heat transfer coefficient with initial temperature, dimensionless time base -- PWR-FLECHT Tests 6948, 6351, and 6553 at a flooding rate of 1 in./sec	153
C-6. Variation of heat transfer coefficient with initial temperature, dimensionless time base -- PWR-FLECHT Tests 0105, 1002, and 6155 at a flooding rate of 6 in./sec	154
C-7. Variation of heat transfer coefficient with initial temperature for a flooding rate of 10 in./sec -- PWR-FLECHT Tests 0307, 0408, and 0509, Rod 3D, six-foot elevation	155
C-8. Variation of heat transfer coefficient with initial temperature, dimensionless time base -- PWR-FLECHT Tests 0307, 0408, and 0509 at a flooding rate of 10 in./sec	156
C-9. Variation of heat transfer coefficient with flooding rate -- PWR-FLECHT Tests 6948, 4225, 1002, and 0408, six-foot elevation on the hottest rods	157
C-10. Variation of heat transfer coefficient with flooding rate, dimensionless time base -- PWR-FLECHT Tests 6948, 4225, 1002, and 0408	158
C-11. Variation of heat transfer coefficient with flooding rate -- PWR-FLECHT Tests 2322, 0711, and 0913, Rod 3D, six-foot elevation	159
C-12. Variation of heat transfer coefficient with flooding rate, dimensionless time base -- PWR-FLECHT Tests 2322, 0711, and 0913	160
C-13. Variation of heat transfer coefficient with flooding rate -- PWR-FLECHT Tests 7158, 6256, and 6155, Rod 3D, six-foot elevation	161
C-14. Variation of heat transfer coefficient with flooding rate, dimensionless time base -- PWR-FLECHT Tests 7158, 6256, and 6155	162

C-15. Variation of surface temperature and heat transfer coefficient with flooding rate -- BWR-FLECHT Tests SS4-19 and SS4-20, Rod 17, six-foot elevation	163
C-16. Variation of heat transfer coefficient with flooding rate, dimensionless time base -- BWR-FLECHT Tests SS4-19 and SS4-20	164
C-17. Variation of surface temperature and heat transfer coefficient with pressure -- BWR-FLECHT Tests SS4-10, -11X, -12, -14R, -16X, -17, and -18; Rod 17, six-foot elevation	165
C-18. Variation of heat transfer coefficient with pressure, dimensionless time base -- BWR-FLECHT Tests SS4-10, -11X, -12, and -14R.....	166
C-19. Variation of heat transfer coefficient with pressure, dimensionless time base -- BWR-FLECHT Tests SS4-16X, -17, and -18.	167
C-20. Variation of heat transfer coefficient with pressure -- PWR-FLECHT Tests 0711, 1615, 1002, and 1417; Rod 3D, six-foot elevation	168
C-21. Variation of heat transfer coefficient with pressure, dimensionless time base -- PWR-FLECHT Tests 0711, 1615, 1002, and 1417 ...	169
C-22. Variation of heat transfer coefficient with subcooling -- PWR-FLECHT Tests 1720, 1002 and 1314; Rod 3D, six-foot elevation. . .	170
C-23. Variation of heat transfer coefficient with pressure, dimensionless time base -- PWR-FLECHT Tests 1720, 1002, and 1314	171
C-24. Variation of heat transfer coefficient with pressure -- PWR-FLECHT Tests 5123 and 4225; Rod 5F, six-foot elevation.	172
C-25. Variation of heat transfer coefficient with pressure, dimensionless time base -- PWR-FLECHT Tests 5123 and 4225	173
C-26. Variation of surface temperature and heat transfer coefficient with subcooling -- BWR-FLECHT Tests SS4-13X, -14R, and -15Z; Rod 17, six-foot elevation	174
C-27. Variation of heat transfer coefficient with subcooling, dimensionless time base -- BWR-FLECHT Tests SS4-13X, -14R, and -15Z	175
D-1. Elevation of quench front with respect to time after flooding initiation -- BWR-FLECHT Test SS4-10	179
D-2. Elevation of quench front with respect to time after flooding initiation -- BWR-FLECHT Test SS4-11X	180
D-3. Elevation of quench front with respect of time after flooding initiation -- BWR-FLECHT Test SS4-12	181

D-4. Elevation of quench front with respect to time after flooding initiation -- BWR-FLECHT Test SS4-13X	182
D-5. Elevation of quench front with respect to time after flooding initiation -- BWR-FLECHT Test SS4-14R	183
D-6. Elevation of quench front with respect to time after flooding initiation -- BWR-FLECHT Test SS4-15Z	184
D-7. Elevation of quench front with respect to time after flooding initiation -- BWR-FLECHT Test SS4-16X	185
D-8. Elevation of quench front with respect to time after flooding initiation -- BWR-FLECHT Test SS4-17	186
D-9. Elevation of quench front with respect to time after flooding initiation -- BWR-FLECHT Test SS4-18	187
D-10. Elevation of quench front with respect to time after flooding initiation -- BWR-FLECHT Test SS4-19	188
D-11. Elevation of quench front with respect to time after flooding initiation -- BWR-FLECHT Test SS4-20	189
D-12. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 0105	190
D-13. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 0307	190
D-14. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 0408	191
D-15. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 0509	191
D-16. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 0711	192
D-17. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 0913	192
D-18. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 1002	193
D-19. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 1314	193
D-20. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 1417	194
D-21. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 1615	194

D-22. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 1720	195
D-23. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 2322	195
D-24. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 4027	196
D-25. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 4129	196
D-26. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 4228	197
D-27. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 4526	197
D-28. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 4628	198
D-29. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 5123	198
D-30. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 5730	199
D-31. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 6155	199
D-32. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 6256	200
D-33. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 6351	200
D-34. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 6553	201
D-35. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 6948	201
D-36. Elevation of quench front with respect to time after flooding initiation -- PWR-FLECHT Test 7158	202

TABLES

I. Liquid Droplet Entrainment	18
II. Comparison of Physical Features of BWR- and PWR-FLECHT Bundles.	27
III. FLECHT Planned Nominal Test Parameters	44
IV. FLECHT Data Summary, Bottom Flooding BWR-FLECHT and PWR-FLECHT -- Six-Foot Elevation on Hottest Rod	45
V. FLECHT Parameters, Bottom Flooding BWR-FLECHT and PWR-FLECHT	46
VI. Heat Transfer Coefficient Ratios at Four- and Six-Foot Elevations . .	99

GLOSSARY

Acronyms

BWR = Boiling Water Reactor
ECC = Emergency Core Cooling
FHUST = Fuel Heatup Simulation Tests
FLECHT = Full-Length Emergency Cooling Heat Transfer
LOCA = Loss-of-Coolant Accident
LOFT = Loss-of-Fluid Test
PWR = Pressurized Water Reactor
SECHT = Scoping Emergency Cooling Heat Transfer

Definitions

Bottom flooding -- the injection of ECC water into a bundle by filling from below.
Bundle -- an assembly of electrically heated rods which are maintained by reactor hardware in a 7 x 7 or 10 x 10 square array.
Bundle power -- the electrical power applied to the test bundle to represent fission product decay energy which is about 6% of the total power of a similarly sized nuclear bundle.
Cold flooding rate -- the rate at which ECC coolant at inlet conditions would rise in the cold bundle.
Coolant subcooling -- the amount of subcooling of the coolant below saturation temperature at test chamber pressure.
Dimensionless time -- real time divided by the time duration from start of transient to cladding quench at the elevation of interest.
Initiation temperature -- the highest temperature in the bundle at start of flooding.
Power decay -- the bundle power decayed to simulate the rate of decrease of fission product decay energy.
Peak power -- the initial maximum power in the highest powered rod in the bundle representing the maximum nuclear power caused by fission product decay in a nuclear bundle.
Power peaking, axial -- the axial profile of power generated by the electric heaters to simulate nuclear power distribution. Expressed as the ratio of power per foot at a point to the average power per foot output in the rod.
Quench -- the rapid reduction of cladding temperature to coolant saturation temperature.

Quench front -- the change from film boiling to nucleate boiling which results in the cladding temperature dropping to the saturation temperature.

Quench time -- the time duration from flooding initiation to cladding quench.

Quench velocity -- the rate at which the quench front moves up the bundle or rod as defined by the rapid drop in temperature.

EFFECT OF GEOMETRY AND OTHER PARAMETERS ON BOTTOM FLOODING HEAT TRANSFER ASSOCIATED WITH NUCLEAR FUEL BUNDLE SIMULATORS

I. INTRODUCTION

The number of nuclear power plants being designed, constructed, and put on line in the United States is greatly increasing. This increase and the growing incentive to locate nuclear power plants in or near population centers have resulted in an increase in the number and complexity of engineered safety systems employed to assure public safety in the unlikely event of an accident. Correspondingly, a greater need has arisen for a high degree of assurance that these safety systems will perform their intended function.

Presently, two principal types of nuclear reactors are in use as power producers in the United States. These are the boiling water reactors (BWR) and the pressurized water reactors (PWR). Analysis of postulated accidents for both BWR and PWR plants shows that the most serious of the postulated accidents would be one in which coolant becomes unavailable to the core. This loss-of-coolant accident (LOCA) is postulated to result from a primary coolant pipe rupture followed by a rapid depressurization, or blowdown, of the reactor system during which the coolant is ejected from the system.

After loss of coolant, redistribution of the stored energy in the fuel rods plus the heat from fission product decay within the fuel causes the cladding of the fuel rods to heat. If no method of core cooling were available, the resultant core thermal transient could result in extensive damage to the cladding, breach of the primary and containment systems, and release of radioactive fission products to the surrounding area. But, engineered safety systems in the form of emergency core cooling (ECC) systems are designed to actuate with the depressurization and provide cooling to arrest cladding heatup before significant core damage takes place.

In the BWR, the ECC systems that provide protection in the event of an LOCA are the top core spray and bottom flooding systems with the driving forces for the fluids of both systems furnished by centrifugal pumps. Each of these systems is designed to introduce sufficient cooling independently to halt the thermal transient. The startup time for these systems is about 30

seconds following actuation caused by a drop in reactor vessel pressure. In the PWR, the ECC systems are bottom flooding systems comprised of a high-pressure-injection centrifugal pump, an accumulator, and a low-pressure-injection centrifugal pump. These systems, which are activated by a low pressure signal, require 10 to 25 seconds to attain design flow. The nominal core flooding rates for both the PWR and BWR systems vary from about 1 to 3 in./sec depending on the manufacturer and the plant design.

At the present time, the evaluation of the ECC systems being designed or constructed for large reactor plants is based primarily on analysis. Experimental data for development and evaluation of the analysis are provided by a series of projects. A number of these projects, funded by the Atomic Energy Commission, are directed and conducted by Idaho Nuclear Corporation^[a] as related water reactor safety programs. These programs include the Scoping Emergency Cooling Heat Transfer (SECHT) Tests Project^[1-5], the Fuel Heatup Simulation Tests (FHUST) Program^[6,7], the Pressurized Water Reactor - Full Length Emergency Cooling Heat Transfer (PWR-FLECHT) Tests Project^[8-12], the Boiling Water Reactor - Full Length Emergency Cooling Heat Transfer (BWR-FLECHT) Tests Project^[13-17], the Semiscale Blowdown and Emergency Core Cooling (ECC) Project^[18], and the Loss-of-Fluid Test (LOFT) Program^[19].

Significant testing has been completed in these projects to date. The Semiscale Blowdown and Emergency Core Cooling (ECC) Project and the LOFT program remain active, whereas the SECHT, FHUST, and FLECHT projects are essentially completed. The semiscale tests, until recently, have emphasized aspects associated with depressurization of a system without emergency core coolant injection. Some tests with emergency core cooling have now been conducted but the results obtained are not directly applicable to the scope of this document. The information obtained from the small-scale experiments and the supporting analytical efforts are being used both for predictive analyses for LOFT, which is currently being constructed, and for continuing development of safety analyses of large power reactors. Testing in the FHUST program, the SECHT Tests Project, the BWR-FLECHT Tests Project (stainless steel-clad

[a] Now Aerojet Nuclear Company

bundles SS-1, -2, and -4 and Zircaloy-clad bundles Zr-1 through Zr-5)^[a], and the PWR-FLECHT Tests Project (Groups I, II, and III) is complete.

The FHUST, SECHT, and FLECHT bundle experiments were conducted with electrical resistance heater rods simulating reactor fuel rods. The rods were arranged in bundles varying in cross section from 37 rods in a modified square array in SECHT through 7 x 7 square arrays in FHUST and BWR-FLECHT to 7 x 7, and 10 x 10 square arrays in PWR-FLECHT. The rods used were of full-scale reactor diameter and spacing. Inconel cladding was used in FHUST, stainless steel or Zircaloy-2 cladding in SECHT, stainless steel or Zircaloy-2 cladding in BWR-FLECHT, and stainless steel or Zircaloy-4 cladding in PWR-FLECHT. Length varied from three feet in FHUST, three and six feet in SECHT, and 12 feet throughout PWR-FLECHT and BWR-FLECHT. The bundles were installed in test chambers. The outer surface of the chamber was insulated to prevent heat loss. Cooling water could be injected from the top or bottom in BWR-FLECHT or from the bottom in PWR-FLECHT. The test chambers also contained provisions for pressurization, draining, visual observation, instrumentation, and electrical and coolant flow control.

The objective of this report is to develop the relationship between the bottom flooding information from the PWR-FLECHT project and that obtained from the BWR-FLECHT project. The development of such a relationship would make the findings of one project useful in the other. Thus, the number of parameters investigated by future test projects could be reduced and could ultimately lead to the application of results from either project to the evaluation of the analysis for the LOCA of both PWR and BWR reactors.

The specific objective of this report is to investigate the effect of variations in operating and design parameters, especially variations in geometry, on the overall heat transfer associated with the BWR and PWR nuclear bundle simulators operating under emergency core cooling conditions. Because both FLECHT projects deal with the core conditions that are expected to exist at the end of blowdown, following reactor depressurization, the operating conditions, if not identical, are in the same ranges for both projects. The parameters that correspond are the chamber temperature and pressure, coolant flooding rate and temperature, rod cladding initial temperature, and peak power. The basic difference between the two projects lies in the difference in lattice geometry,

[a] Zr-1, Zr-2, Zr-3 ... identifies the test series; all BWR-FLECHT Zircaloy bundles employed heater rods clad with Zircaloy-2.

in other words in the differences in diameter and pitch of the simulated fuel rods. BWR-FLECHT rods are larger in diameter than PWR-FLECHT rods and the rod spacing is greater in BWR-FLECHT. Both FLECHT projects were conducted with stainless steel and Zircaloy cladding. However, since the zirconium-water reaction tends to mask the effects of variations in geometry which are of primary concern in this report, only results from heater rods clad with stainless steel are evaluated.

Bottom ECC flooding, which is the mode of ECC injection common to both reactor types, is investigated in this report; the top spray mode of the BWR's is not studied in this work. The area of interest in this report is that portion of the cooling cycle between the time ECC begins entering the bundle and the time the bundle is completely quenched.

The major source of data was thermocouples mounted inside the heater cladding. Data from these thermocouples were used in conjunction with the measured variables of rod electrical power and the bulk saturation temperature at test chamber pressure to calculate overall heat transfer coefficients at the rod surfaces. The calculated coefficients from the two FLECHT projects are compared. Included is a discussion of the variations in heat transfer resulting from variations in the test parameters of peak power, cladding temperature at test initiation, cold flooding rate, chamber pressure, and inlet coolant subcooling. Radial and axial effects are described. The heat transfer of the BWR-FLECHT project is correlated with that of the PWR-FLECHT project at the four- and six-foot elevations of the hottest rods. The following section contains a brief description of the BWR and PWR nuclear plants after which the two FLECHT projects were modeled.

II. DESCRIPTION OF BWR AND PWR NUCLEAR PLANTS

The BWR core is located in a large pool of slightly subcooled flowing water maintained at about 1000 psi by the surrounding pressure vessel. A typical BWR reactor assembly is shown in Figure 1^[20]. The core heat transmitted to the coolant causes some of the liquid to vaporize such that a two-phase mixture rises from the pool. This mixture passes through steam separators and dryers to the steam dome above the core and then out the vessel to the turbines by way of the main steam lines. The steam gives up its energy by driving the turbines. The condensate is pressurized, reheated, and returned to the reactor vessel as feedwater. The subcooled liquid is circulated down into the inlet plenum below the core by jet pumps operating as part of the recirculation system in the downcomer between the vessel and core shroud. As coolant is vaporized by the core heat, more coolant is circulated into the core inlet to replace that vaporized. The recirculation system operates by drawing a portion of the subcooled feedwater out of the downcomer, pressurizing it, and reinjecting it into jet pumps located in the downcomer. These jet pumps draw more subcooled liquid through the venturi surrounding each injector, thus creating the required circulation. The core power is directly dependent on plant power requirements.

The PWR core is located in a pressure vessel which maintains the coolant at about 2200 psi. Figure 2 shows a typical PWR reactor assembly^[21]. Unlike the BWR, the PWR primary system operates completely within the subcooled region. The coolant is circulated by pumps into the vessel above the core and through the downcomer around the core to the core lower inlet plenum. From here the water is forced upward into the core where it removes the heat from fission and passes out into the exit plenum. The high enthalpy coolant then leaves the reactor vessel through one to four hot-leg pipes, depending on plant design and manufacturer, and is circulated into the heat exchangers. The exchangers (steam generators) provide the heat transfer area to extract the primary system coolant energy and produce steam in a lower pressure (1000 psi) secondary system. The secondary systems provide steam to drive the turbines and return feedwater makeup to the steam generators. After leaving the steam generators, the primary coolant enters the pumps and is returned to the reactor vessel inlet plenum. To maintain subcooled water throughout the PWR system, a compressible volume in the form of a pressurizer is provided to allow for expansion on heatup.

Fig. 1 BWR reactor assembly^[20].

Fig. 2 PWR reactor assembly [21].

This compressible volume is connected to the primary system with a surge line and provides a steam bubble maintained with electric heaters.

III. RELATED EXPERIMENTS

Although emergency core cooling (ECC) systems of the bottom flooding type have been utilized from the beginning in large power reactors, relatively little quantitative information had been developed for these systems in the area of heat transfer until the FLECHT projects began producing data. The flow regimes had been identified, but little was known of the actual heat transfer processes. A number of scoping experiments and demonstrations tests have been run by AEC contractors and reactor vendors using fuel rod simulators in various arrays. All of the fuel rod simulators have electrically powered resistance heaters mounted vertically in test chambers.

1. GENERAL ELECTRIC COMPANY EXPERIMENTS

General Electric Company, which began BWR ECC evaluation work in 1960, described their initial flooding work as follows^[14]:

"During 1965 an intensive investigation of emergency spray cooling was carried out. The investigation covered spray distribution to rods in a core mockup assembly, single-rod spray cooling and flooding, and 36-rod tests of rod temperature during loss-of-coolant accident with various initial channel powers and spray cooling flow rates. These tests were run with uniform heat flux heaters. An analysis employing a simplified model, but accounting for both radiation and convection, gave values for convection heat transfer that permitted calculation of surface temperatures within 50°F of experimental results....

".... In addition to the work done at General Electric, and discussed above, emergency cooling studies have been carried out by the British^[4] and Japanese^[6]. Results of this previous work were used in planning and interpreting the results of the current test series."

"[4] G. L. Shires, A. R. Pickering, and P. T. Blacker, Film Cooling of Vertical Fuel Rods, AEEW-R343 (1964).

"[6] H. Uchida, A. Oyama, and V. Togo, Evaluation of Post-Incident Cooling System of Light Water Power Reactors, A/Conf. 28, p 436 (May 1964)."

The tests described by Schraub and Leonard^[14] were conducted with full-length stainless steel-clad, Nichrome-filament heaters which produced a chopped-cosine power distribution. These heaters were aligned, through use

of product line hardware, in a 7 x 7 array with square flow channels. ECC flooding rates varied from 2 to 10 in./sec, maximum cladding initial temperatures ranged from 1000 to 1600°F, and bundle maximum power varied from 100 to 325 kW. Schraub and Leonard observed gross behavior of the coolant in the bundle and classified it into three flow patterns corresponding to finite time periods: (a) quiescent filling, (b) intermittent volcanic, and (c) steady volcanic. The first pattern that occurred as the flooding began produced a small amount of steam. The second pattern was characterized by gushes of steam and water lasting several seconds followed by several seconds of quiet, then another gush. This pattern lasted from 1 to 4 minutes with the gushes diminishing in violence with time. The duration of this pattern was observed to be dependent on power level and flooding rate. Fluctuations that corresponded to the rate at which steam and water gushed were detected in the coolant head in the bundle. The third pattern was characterized by a steady effusion of steam and water from the top of the bundle. Although the bundle water level increased, no oscillations were observed. This pattern prevailed until quenching occurred. Schraub and Leonard related maximum cladding temperatures to the varied parameters and noted that the hottest spot in the bundle moved upward from the midplane (six feet) to the upper end as cooldown progressed. No heat transfer data were developed in the referenced report.

J. D. Duncan and J. W. Leonard reported a second series of BWR bottom flooding tests^[15] using Bundle SS-1 in the FLECHT project. This series was an extension of that described by Schraub and Leonard and used a full-scale electrically heated 49-rod stainless steel-clad bundle in the same test assembly. The maximum bundle power ranged from 240 to 390 kW and test power was varied to produce a decay rate and chopped-cosine axial profile representative of that in a power reactor. Flooding rates varied from 0.63 to 6.05 in./sec and initial cladding temperatures ranged from 1300 to 2150°F. Because of the high operating temperature, the cladding developed an oxide coating of between two and three mils thickness and some rod bowing was observed. Film heat transfer coefficients were calculated for Rod 17. Although the method of calculation was not stated the following assumptions were used: the energy storage rate in the selected rod was zero, the radiation sink surrounding the selected rod was at a temperature equal to the average temperature recorded for adjacent rods, and the convection sink was at a temperature of 212°F.

These coefficients were calculated at the bundle hot spot over the period of highest temperature (Rod 17, six-foot elevation, 0 to 2.5 minutes) and represent local coefficients. The coefficients thus calculated by Duncan and Leonard ranged from 5 to 15 Btu/hr-ft²-°F and generally increased with flooding rate. The investigators stated that the coefficients were almost entirely the result of radiation to the entrained water from the rod surfaces. They performed additional calculations using the following equation for radiation from the rod to the coolant with a view factor of 0.5 for the entrained liquid:

$$\frac{A_1 \sigma (T_1^4 - T_2^4)}{\frac{1}{\epsilon_1} + \frac{1}{\epsilon_2} - 1} = h_{\text{rad}} A_1 (T_1 - T_2)$$

where

- q = net radiation exchange between rod and coolant
- h_{rad} = "radiation" heat transfer coefficient
- A_1 = rod surface area
- σ = Stefan-Boltzman constant
- T_1 = absolute temperature of rod surface
- T_2 = absolute temperature of coolant
- ϵ_1 = emissivity of rod surface
- ϵ_2 = emissivity of coolant.

The cladding surface temperature as measured and the saturation temperature of the coolant (212°F) along with the cladding emissivity (0.6) and water emissivity (0.96) were employed in this equation to obtain heat transfer coefficients ranging between 7 and 13 Btu/hr-ft²-°F. These values for the heat transfer coefficients agree well with those obtained from the first method of calculation. Duncan and Leonard noted that these coefficients are average values over the period examined and were considered qualitative only. The data acquisition system used during these tests recorded data points every 45 seconds for a particular thermocouple. This recording interval for a transient situation did not provide the resolution necessary for calculating continuous heat transfer coefficients.

The final stainless steel bundle tests of the BWR-FLECHT project on Bundle SS-4 were performed with pressure as the primary variable^[16]. A number of these tests were with bottom flooding coolant injection. The data from these tests are used as the basis for the comparison in this report.

General Electric Company has not conducted a test series employing only bottom flooding with Zircaloy cladding; the Zr-5 BWR-FLECHT bundle utilized both top spray and bottom flooding. The final report on this project has been published^[17].

2. WESTINGHOUSE ELECTRIC CORPORATION EXPERIMENTS

Westinghouse Electric Corporation has been a major contributor to the FLECHT program and has published three reports describing the results of this work^[9-11]. The referenced studies were conducted using full-scale electrically heated 49- and 100-rod stainless steel- and Zircaloy-clad bundles mounted in square flow housings with product line grid spacing. Initial peak power ranged from 0.6 to 1.40 kW/ft-of-rod. A chopped-cosine axial power profile was used and the total bundle power was decreased to simulate reactor decay power. Other operating parameters and their ranges include: pressures from 15 to 90 psia, constant flooding rates from 0.6 to 17.6 in./sec, inlet subcooling from 17 to 162°F, and initial cladding temperatures from 801 to 2212°F. Some tests were also run with a varying flooding rate. Several tests were conducted in which coolant at the top of the bundle was allowed to return to the bundle to enhance bundle heat transfer. Other tests were performed with simulated flow blockage to investigate the cooling problems caused by cladding ballooning. Heat transfer coefficients were calculated using the temperature data for the rod cladding, the coolant saturation temperature, and the actual electrical power profile as input to the computer code DATAR^[9]. The coefficients for selected tests are used as the basis for the comparison in this report. The summarized conclusions of this series of tests are as follows^[9-11].

“For times right after flooding, the heat transfer coefficient increased with increasing initial clad temperature due to increased vapor generation.

"The heat transfer coefficient increased significantly with increasing flooding rate due to increasing vapor generation rate and liquid entrainment.

"During the early stages of flooding, heat generation has little effect on heat transfer coefficient. As flooding progresses, however, the increasing heat generation decreases the heat transfer coefficient significantly for a 2 in./sec flooding rate. There is only little effect for 6 in./sec flooding rate....

".... As pressure increases for constant subcooling, the heat transfer coefficient increases because the velocity required for liquid entrainment decreases and the two-phase mixture properties change with pressure.

"At the high flooding rates (6 in./sec or greater)...., the initial heat transfer coefficient at higher elevations (6 feet or above) increases with increasing distance from the bottom of the heated length. This effect is reversed at later times. At the extremely low flooding rate of 2 in./sec, the heat transfer coefficient at any given time decreases with increased elevation. This could be due to the lack of water droplets at higher elevations.

".... The current data indicate that at a flooding rate of 1 in./sec carryover represents approximately 10 percent of the coolant supplied to the flow housing throughout the run. This amount of carryover is expected to increase with flooding rate."

Empirical correlations have been developed by Westinghouse Electric Corporation that are based on the experimental results of these PWR-FLECHT tests^[11]. These correlations for the heat transfer coefficient can be divided into three periods approximating the different regimes observed in these bundles. Figure 3 illustrates the portion of the total curve to which each period applies. The following equations are considered valid over the following ranges of parameters:

Flooding Rate (V_{in}) ^[a]	1 to 10 in./sec
System Pressure (P) ^[a]	15 to 90 psia
Inlet Coolant Subcooling (ΔT_{sub})	16 to 189°F
Initial Cladding Temperature (T_{init})	1200 to 2200°F
Peak Power (Q'_{max})	0.69 to 1.40 kW/ft
Elevation (Z)	4 to 8 ft
Flow Blockage (B)	0 to 75%

[a] The correlation has since been found to overpredict the heat transfer coefficient at low pressures and low flooding rates.

Fig. 3 PWR-FLECHT heat transfer coefficient versus dimensionless time.

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Period I. $0 < t/t_{q6ft} < (t/t_{q6ft})_2$

$$h = h_1 + [h_{12} - h_1] [(1 - e^{-x}) - 0.9x e^{-x^2}] [1 - 1583 e^{-0.0625V_{in}^2} u e^{-u} \exp \{-(0.588Z - 3.824)^2\}]$$

where

t is the time after flooding and t_{q6ft} is the quench time at the 6-foot elevation

$$\begin{aligned} x &= 17.6 [1 + 4.37 e^{-0.0166\Delta T_{sub}}] [1 - \exp \{-(0.00075 + 0.0000272 (V_{in} - 8)^2) (T_{init} - 650)\}] (t/t_{q6ft}) \\ (t/t_{q6ft})_2 &= 0.62 [(1 - e^{-0.192Z}) - 0.115Z e^{-0.0368Z^2}] \\ h_{12} &= h_1 + [35.7 + (22 - 0.00303Z^{4.1}) (1 - e^{-0.0333P} - 0.3P e^{-0.011P^2})] [1 - e^{-0.2V_{in}}] + 17 [1 - e^{-V_{in}/0.5}] [1 - e^{-B/25}] \\ u &= 9 (t/t_{q6ft})^2 / (t/t_{q6ft})^2 \\ h_1 &= 4 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F} - \text{a mean value of experimental results.} \end{aligned}$$

Period II. $(t/t_{q6ft})_2 < t/t_{q6ft} < (t/t_{q6ft})_3$

$$h = h_2 + b_1 [y^2 + b_2 (y^2 - b_3 y^3)]$$

where

$$\begin{aligned} y &= (t/t_{q6ft}) - (t/t_{q6ft})_2 \\ h_2 &= h_1 + [h_{12} - h_1] [(1 - e^{-x_2}) - 0.9 x_2 e^{-x_2^2}] \\ (t/t_{q6ft})_3 &= 1.55 [(1 - e^{-0.205Z}) - 0.154 Z e^{-0.0421Z^2}] \\ x_2 &= 17.6 [1 + 4.37 e^{-0.0166\Delta T_{sub}}] [1 - \exp \{-(0.00075 + 0.0000272 (V_{in} - 8)^2) (T_{init} - 650)\}] (t/t_{q6ft})_2 \\ b_1 &= [682 - 650 (1 - e^{4-Z})] [1 - \exp \{0.95 (1 - 0.0488Z) V_{in}\}] [1 - \exp \{-0.0238\Delta T_{sub}\}] [0.0696 + 0.304 e^{-B/25}] \end{aligned}$$

$$b_2 = 0.5Z [1 - e^{-2(Z-3.5)}] [1.33 (1 - e^{-0.0227P}) - 1]$$

$$-2.9 [1 - e^{-V_{in}/2.5}] [1 - e^{-B/25}]$$

$$b_3 = 2.55 (Z - 3.7)^2 e^{3.7 - Z}.$$

Period III. $(t/t_{q6ft})_3 < t/t_{q6ft} < (t/t_{q6ft})_4$

$$h = h_3 + C [t/t_{q6ft} - (t/t_{q6ft})_3]$$

where $(t/t_{q6ft})_1 = 0.38$ for four-foot elevation
 $= 0.9$ for six-foot elevation
 $= 1.2$ for eight-foot elevation

$$C = 420 [1 - e^{-0.00625b_1}]$$

$$h_3 = h_2 + b_1 [y_3^2 + b_2 (y_3^2 - b_3 y_3^3)]$$

$$y_3 = (t/t_{q6ft})_3 - (t/t_{q6ft})_2$$

The quench time at the six-foot elevation is given as:

$$t_{q6ft} = 87.54 [e^{-0.107\Delta T_{sub}} (1 - e^{-0.667V_{in}}) + 3.44 V_{in}^{-1.205}$$

$$-16.75 e^{-2.5V_{in}}] [1 + 0.0000588 T_{init}] [1 + 0.91$$

$$e^{-0.0345P}] Q'_{max}$$

3. PHILLIPS PETROLEUM COMPANY AND IDAHO NUCLEAR CORPORATION EXPERIMENTS

Emergency core cooling experimentation involving bottom flooding began at the National Reactor Testing Station under the direction of Phillips Petroleum Company in 1967 and has since been continued by Idaho Nuclear Corporation. The initial project was the Scoping Emergency Cooling Heat Transfer (SECHT) Tests Project^[1] which was performed in four segments. All bundles were made up of a modified square array containing 37 electrically powered fuel rod simulators. A cylindrical pressure chamber surrounded the bundle and served as a flow guide.

Results of SECHT I and II using three-foot-long stainless steel-clad heaters have been reported^[2,5]. Power to the heaters was maintained at a uniform surface heat flux of 14,300 Btu/hr-ft² both axially and radially. The cold flooding rate was varied from 0.9 to 3.6 in./sec and the inlet water temperature was varied from 100 to 158°F. Both constant and varying heads were used for the coolant flow. The nontypical geometry and thermal-hydraulic conditions along with numerous random heater failures caused the findings of these tests to be qualitative only. By use of thermocouple readings and high speed photography, observations of the coolant flow phenomena were made. These studies confirmed that two-phase flow was the predominant flow regime and that the bundle was cooled by radiation to the water droplets and convection in the steam.

SECHT III tests employed six-foot-long stainless steel-clad heaters^[3]. Uniform power was maintained in all rods. The parameters varied were: bundle surface heat flux from 0 to 15,000 Btu/hr-ft², initial cladding temperature from 600 to 2000°F, coolant inlet temperature from 100 to 200°F, and cold flooding rate from 0.45 to 7.2 in./sec. A marginal cooling test at a flooding rate of 0.5 in./sec started at 1600°F with 15,000 Btu/hr-ft² heat flux resulted in a quasi-steady state condition lasting 184 seconds after the temperature rise was arrested. A calculation based on the assumption that only simulated decay energy and no stored energy was being removed from the rod resulted in a film heat transfer coefficient of 8.9 Btu/hr-ft²-°F. This coefficient is in general agreement with the results of General Electric Company. SECHT III investigators noted that the temperature turnaround exhibited by thermocouples at all elevations in the bundle occurred almost simultaneously during any particular test. They believed that relatively uniform heat transfer occurred throughout the entire bundle at the time of temperature turnaround and was due to a significant increase in entrainment. Figures 4 and 5, along with the droplet measurements and velocities listed in Table I, confirm the conclusions of these investigators. Figure 4 is a schematic of the view through the lower viewing port of the SECHT vessel. The photographic sequence of Figure 5 shows development of liquid entrainment. The first view of Figure 5 occurred about 4.5 seconds after water had reached the bottom of the heated section. Up to that time, only single-phase vapor had passed the port. This first view shows several tiny droplets and as time progresses increasing amounts of liquid are entrained. The total time elapsed from View 1 through View 6

TABLE I

LIQUID DROPLET ENTRAINMENT

$T_i = 1400^\circ\text{F}$, Flooding Rate = 3.6 in./sec, Coolant Temperature = 200°F .

Identification in Figure 5	Droplet Size		Droplet Upward Velocity (ft/sec)
	X-Dimension (in.)	Y-Dimension (in.)	
A	0.115	0.089	5.21
B	0.214	0.150	8.6
C	0.111	0.096	7.5
D	0.374	0.286	5.0

Fig. 4 Schematic of view through port of SECHT III vessel.

Fig. 5 Sequence of development of liquid entrainment, SECHT III.

was 664 milliseconds. Further conclusions on the heat transfer characteristics of SECHT were not made because of both the nontypical geometry and the thermal-hydraulic conditions (as with SECHT I and II).

In SECHT IV a Zircaloy-clad bundle was tested to provide results for comparing the differences between stainless steel and Zircaloy cladding^[4]. This series confirmed results of previous SECHT tests and is not discussed further in this report.

A scoping test using a nine-rod cluster and an experiment employing a full 49-rod test array were conducted in the FHUST test series^[7]. The nine-rod cluster, using stainless steel cladding, was mounted in a Vycor glass cylinder and heated at an unspecified power. Cooling water was introduced into the bottom of the array and allowed to escape at the upper end of the glass cylinder. Figures 6 and 7 illustrate the flow and heat transfer regimes observed. The five flow regimes are clearly shown in Figure 8. The following observations were reported.

"The result of the testing with the nine-rod assembly was a series of high speed movies (up to 1000 frames/second) and still photographs in which the heat transfer and fluid flow regimes, considered to be similar to the regimes developed during emergency core cooling in a large PWR, were identified....

"As the cooling water entered the bottom of the array, a region of film boiling was established. This region was characterized by a steam annulus between heated surface and the continuous water phase [Figure 7a]. As the surface temperature was reduced to below the Leidenfrost point, the heater surface became wetted and nucleate boiling was established. This wetting began at the bottom of the rod and progressed upward [Figure 7a]. As the surface area exposed to the incoming water increased, the steam velocity became sufficiently high for liquid droplet entrainment by the steam to take place. Figures [7b] and [7c] show the flow regimes at different stages of the entrainment development.... The flow regime in the upper portion of the core following the beginning of entrainment was characterized by liquid droplets dispersed in the flowing steam.

"The interfaces between the various flow regimes were observed to progress upward with time. As a flow regime interface passed a thermocouple location, a well defined change in temperature transient was observed on the recorded temperature curve [as shown in Figure 8]."

The 49-rod FHUST experiment consisted of Inconel-clad resistance heater rods with a three-foot uniformly heated length. The bundle was surrounded by a close fitting stainless steel canister which served to direct the coolant flow. Power was maintained constant throughout each test but was varied from 0 to 1400 watts/rod through the test project. Initial cladding temperatures ranged from 1000 to 1800°F and the flooding rate varied from 0.53 to 8.5 in./sec at temperatures from 75 to 200°F at atmospheric pressure. The temperature-time-data from thermocouples on the surface of the rod and in the coolant were input to the CINDA computer program along with the rod power

Fig. 6 Schematic of flow and heat transfer regimes, FHUST.

readings and rod thermal properties to obtain heat transfer coefficients. Examples of the coefficients thus calculated are shown in Figure 9. The dispersed flow regime of Figure 9 corresponds to the film boiling regime of Figure 8. The local heat transfer in the dispersed or two-phase flow was found to be highly dependent on inlet water flow rate and coolant temperature [7].

(a) 69-3509

(b) 69-3508

(c) 69-3510

Fig. 7 Flow and heat transfer regimes, FHUST.

Fig. 8 Typical cladding temperature response, FHUST.

Fig. 9 Heat transfer coefficients for various inlet flooding rates, FHUST.

IV. DESCRIPTION OF FLECHT EXPERIMENTS

The FLECHT experiments were conducted with full-scale, 12-foot-long simulators of individual core segments. These bundles were constructed using the dimensions and geometry of the respective BWR and PWR product line assemblies. The BWR-FLECHT project used 49-rod bundles and the PWR-FLECHT project used 49- and 100-rod bundles. These bundles were tested in facilities capable of simulating the reactor ECC conditions of: initial cladding temperature, vessel pressure, coolant flooding rate, coolant inlet temperature, heater rod power generation with both axial and radial peaking profiles, and power decay with time. Maximum bundle power was six percent of reactor beginning-of-life power. The test chambers also provided simulation of in-core radiation heat transfer.

1. BUNDLE ASSEMBLY

The bundles are comprised of the heater rods, spacer grids, tie plates, canisters, control rod thimbles, and other structural hardware. The instrumentation within the assemblies is also considered a part of the bundle. Hardware for BWR-FLECHT top spray and for PWR-FLECHT flow blockage experiments is not discussed. The axial power profiles of the FLECHT experiments are shown in Figure 10. Since the electrical resistivities of the heater coil materials are known functions of temperature the axial power distributions in the bundles remained known throughout the temperature excursions. Table II presents a physical comparison of the BWR-FLECHT and PWR-FLECHT bundles.

1.1 BWR-FLECHT

The BWR-FLECHT test bundle, Bundle SS-4, shown in Figures 11 and 12, consisted of a 7 x 7 square array of Calrod-type electrically heated rods held inside an inner channel by eight evenly distributed horizontal spacer grids^[14]. The channel represents a BWR bundle canister with representative inside dimensions and is constructed of 0.100-inch-thick stainless steel. A simulated 1967 product line upper tie plate held the rods at the top and served as the common electrical ground of all 49 rods. The lower ends of the rods were brazed to the lower head as shown in Figure 11 to retain the rods in

Fig. 10 Heater axial power profiles, BWR- and PWR-FLECHT.

the bottom of the test chamber and to maintain the liquid seal. Longitudinal movement of rods from thermal expansion was permitted by sliding electrical contacts in the upper tie plate.

The stainless steel-clad heaters had an outside diameter of 0.570 inch and a total length of 13 feet 10 inches of which 12 feet was heated. The cladding consisted of an inner layer and an outer layer of stainless steel tubing. The heaters had internal coils of Nichrome wire which required that the surface temperature be limited to 2100°F. The pitch of these coils was varied to provide the desired axial power distribution simulating that of an actual fuel

TABLE II

COMPARISON OF PHYSICAL FEATURES OF BWR- AND PWR-FLECHT BUNDLES

Feature	BWR-FLECHT	PWR-FLECHT
Rod Diameter (in.)	0.570	0.422
Rod Pitch (in.)	0.738	0.563
Rod Cladding, Stainless Steel	304	347
Cladding Thickness (in.)	0.100	0.024
Rod Heating Element	Nichrome Wire	Nichrome or Kanthal Wire
Rod Dielectric	Magnesium Oxide	Boron Nitride
Heating Element Active Length (ft)	12	12
Bundle Size, Number of Rods	49	49 and 100
Number of Active Rods	49	42 and 91
Axial Power Peaking Factor	1.47	1.66
Radial Power Peaking Factor	1.18	1.10
Peak Power (kW/ft)	0.738	1.40 1.24 0.69
Decay Power (% of Actual Reactor Peak Power)	5	6.6
Flow Area Per Rod (in. ²)	0.290	0.176
Wetted Perimeter (in.)	2.46	1.89
Hydraulic Diameter (in.)	0.648	0.532
Heat Transfer Area per Length of Rod (ft)	0.15	0.11
Heat Flux, Reference Peak (kW/ft ²)	0.492	1.125
Heat Capacity of Rod (Btu/ft-°F)	0.0502	0.0393

INC-B-17616

Fig. 11 Bundle and test chamber cross section, BWR-FLECHT.

rod. Insulation was provided by magnesium oxide compacted between the cladding and the heater filament. The heater construction is shown schematically in Figure 13. The radial power peaking, or power distribution, is noted in the top view of the bundle in Figure 12.

Fig. 12 BWR-FLECHT test bundle.

Instrumentation of a BWR-FLECHT heater consisted of up to seven 0.040-inch-diameter, Inconel sheathed, Chromel-Alumel thermocouples imbedded in the inner layer of cladding as shown in Figure 13. A stainless steel tube of 0.036-inch wall thickness was then swaged over the inner layer of cladding to retain the thermocouples and provide the 0.570-inch, smooth outside rod diameter. About 100 thermocouples were distributed radially throughout the SS-4 bundle at elevations of 1.2, 4, 5, 6, 7, 8, 9, and 10.8 feet from the lower end of the heated length. Thermocouples were also located on the inner channel surrounding the bundle.

Fig. 13 Schematic of heater rod, BWR-FLECHT.

1.2 PWR-FLECHT

The PWR-FLECHT test bundle was designed to represent a typical commercial fuel assembly^[9,10]. PWR-FLECHT project experiments utilized both 7 x 7 and 10 x 10 arrays of resistance-type electrical heaters held in radial position by eight sheet-metal grids, one at each end of the bundle and the remaining six spaced equally between the end grids. Located in each bundle were the typical control rod guide thimbles and instrumentation tube elements which do not produce power. The bundles were supported in the test chamber by compression seals at both the top and bottom of the chamber where the rods protrude through the vessel heads as described in Reference 12. End views of the 10 x 10 and 7 x 7 test sections are shown in Figures 14 and 15. Included are the locations of structural elements and the radial power distribution. The stainless steel-clad heaters used were 16 feet long and had a 12-foot heated length and an outside diameter of 0.422 inch. These heaters had internal coils of Nichrome or Kanthal wire of varying pitch to provide the desired axial power distribution. The PWR-FLECHT Group I series using Nichrome filaments was limited to a maximum cladding temperature of 2100°F, and the Group II series using the Kanthal filaments was limited to a cladding temperature of 2400°F. The coils were insulated from the tubes by boron nitride. The

INC-B-17633

Fig. 14 10 x 10 PWR-FLECHT test section.

heater construction is shown in Figure 16 and the axial power profile was given in Figure 10.

Instrumentation of the PWR heaters was distributed radially throughout the bundle. Up to six Chromel-Alumel thermocouples were located on the inside surface of the heater rod cladding. Most rods had three thermocouples positioned at 4, 6, and 10 feet or five thermocouples at 2, 4, 6, 8, and 10 feet from the bottom of the heated length. Measurement of local coolant conditions was made with special steam probes which were radiation shielded and positioned in two

Fig. 15 7 x 7 PWR-FLECHT test section.

INC-A-15116

INC-A-15179

Fig. 16 Schematic of heater rod, PWR-FLECHT.

places in the 100-rod arrays. Other thermocouples were provided to measure local coolant conditions in the bundle and the surface temperature of the thimble. The 49-rod array contained 58 thermocouples and the 100-rod array contained 100. Pressure transducers were attached to taps located in the bottom of the two thimbles to record the pressure at the bottom of the heated length of the bundle.

2. TEST FACILITY

The test facility is made up of the pressure housing or test chamber, coolant supply system, electrical power system, instrumentation, and data acquisition system.

2.1 BWR-FLECHT

The BWR-FLECHT overall test facility is shown schematically in Figure 17. The basic element simulated by the BWR-FLECHT test configuration is represented by the solid lines of Figure 18; the dashed lines represent the BWR reactor geometry surrounding the corresponding element in an actual reactor configuration. As shown in Figures 11 and 18, the dual channel arrangement creates a flow path simulating that occurring between a fuel bundle and the walls of adjacent bundles in a reactor. The channels were both formed from stainless steel square tubing of 0.100-inch wall thickness. The inner channel was 5.278 inches inside across flats, which corresponds to the dimensions of a 1967 product line channel. This dual channel assembly complete with heaters was mounted inside a pressurized cylinder (Figure 11). The heater rods and instrumentation penetrated the pressure vessel lower head. Inlet flooding water was brought through the side of the vessel below the heated length of the bundle and into the outer channel. Holes through the inner channel at this level allowed coolant circulation to the array and between channels simultaneously. The upper vessel head contained the steam outlet. Any entrained liquid escaping the bundle and outer channel fell back into the chamber formed between the vessel and the outer channel. A drain at the elevation of the inlet line allowed drainage of the chamber, if necessary between tests; drainage was not permitted during testing. Differential pressure transducers attached to the bundle and vessel chamber indicated coolant level. Vessel chamber pressure was also monitored by two transducers. The purpose of one of these transducers was to provide the control

INC-C-17615

Fig. 17 BWR-FLECHT test facility - schematic.

Fig. 18 BWR-FLECHT test section and superimposed BWR reactor geometry.

signal for the steam outlet valves to open or close for chamber pressure regulation. The vessel was fully insulated to reduce heat losses.

Demineralized water was supplied to the bundle from the heated steam drums as shown in Figure 17. The demineralized water was pumped through the subcooler where the temperature was reduced as required and into the pump inlet. The flow was regulated by a control valve and the water passed on to the bundle. Test flow was recorded by the flow meter before the water entered the bundle. Drainage was pumped back into the steam drums. Steam was exhausted through the steam outlet valves as required to maintain system pressure. All hot lines were insulated. Thermocouples were located throughout the system (Figure 17) to monitor fluid temperatures.

The heaters had individual power cables attached at the bottom and a common ground attached at the top. Three-phase, 60-cycle power was applied to a set of six transformers. The transformers were connected to step up or step down the

power as required to maintain local power peaking and were adjusted as required prior to testing. To simulate decay power, as depicted in Figure 19, a special function generator was developed to adjust a rheostat automatically. The transformer system maintained the desired power ratios to the rods.

Fig. 19 BWR-FLECHT power decay.

The data acquisition system used during the SS-4 test series included a Hewlett-Packard Dymec System Model 2017-D. This system scanned the thermocouple and wattmeter outputs five times every second. The temperature and power readings were punched as functions of time on a standard eight-channel paper tape and printed on a continuous strip chart. From paper tape, the data were fed to a digital computer for conversion to magnetic tape. The information was used as input for a data print-out code that sorts and arranges the data according to heater rod, elevation, and time. Important thermocouple readings and the pressure transducer and flow meter outputs were displayed on one Sandborne and two Beckman pen-type continuous recorders. To improve on the Dymec sweep time of one data point per channel for each 20 seconds, Idaho Nuclear Corporation installed a 100-channel Accudata Multiplexing System, 16 Accudata Model 120 DC amplifiers, and an Ampex FR-1300 magnetic tape recorder. This system produced 11 data points per channel per second. The thermocouple data and a reference timing signal were recorded on magnetic tape. These data were sorted and stored by run, rod, and

thermocouple number by the IBM-360 computer. From these data the heat transfer coefficients were calculated by use of the DATAR computer code.

2.2 PWR-FLECHT

The PWR-FLECHT test facility schematic is shown in Figure 20. The once-through system which supplied demineralized water to the bundle consisted of a flow housing for the test section, an accumulator to provide coolant feed, a steam generator for back pressure regulation, and the required valves, piping, and auxiliary equipment.

Two flow housings were used, one for the 10 x 10 array and one for the 7 x 7 array. The cross-sectional internal dimensions for these two 0.700-inch-thick carbon steel housings are 5.889 and 4.200 inches, respectively. Clamp-on heaters were installed on the housings, as shown in Figure 21, to obtain the desired initial temperature. The heater rods were held in the housing at both ends by compression seals through which the rods extended. The power leads were attached to these extensions. The upper and lower ends of the flow housing terminated in flanged plenum chambers. The lower plenum served as the coolant inlet chamber. The coolant passed from the lower plenum into the housing through numerous holes in the housing. After rising through the bundle the coolant passed from the housing and into the upper plenum. The upper plenum collected coolant ejected from the bundle, drained the liquid, and vented the vapor. The 10 x 10 housing was equipped with three rectangular quartz view ports. The upper plenum had one port.

The accumulator, a 12-inch-diameter vessel, was designed with external clamp-on heaters to achieve the desired coolant temperature. Pressurized gas cylinders were used to maintain the accumulators at a constant pressure. The steam generator, a 10-inch-diameter vessel of about 20 gallons capacity, also was equipped with external heaters. All equipment, with exception of the inlet and outlet plenums, was fully insulated to reduce heat losses.

Power was supplied by a 1500 kVA transformer connected to three silicon-controlled rectifier (SCR) power control units. To simulate the estimated rate of decay heat generated in a shut down reactor the three SCR units were controlled by an automatic curve-follower programmer which mechanically drove three potentiometers. The potentiometers, each controlling an SCR,

Fig. 20 PWR-FLECHT test facility -- schematic.

Fig. 21 PWR test chamber and bundle - elevation:

INC-B-17617

were offset to obtain the required radial power peaking. The PWR-FLECHT decay power curve thus provided is shown in Figure 22.

Fig. 22 PWR-FLECHT power decay.

Instrumentation external to the bundle included five pressure transducers with taps mounted on the flow housing vertically from 0 to 8 feet to measure axial pressure variations. Bundle inlet and outlet pressures were monitored by pressure cells, and bundle water level was measured by a differential pressure cell. Accumulator water temperature was monitored by thermocouples and the thermocouple output regulated the heater power input. The output of the pressure transducer which monitored the accumulator pressure was recorded. The liquid level was measured by a differential pressure cell and also recorded. The steam generator pressure was monitored by a pressure transducer which fed a recorder-controller which in turn controlled the clamp-on heaters. Steam generator water level was measured by a differential pressure cell and recorded. A Potter turbine flow meter monitored the coolant flow to the housing. The output of this meter was recorded by a flow recorder-controller which in turn governed the pneumatically operated flow control valve shown in Figure 20. The flow housing temperature was monitored by thermocouples on the outer wall of the flow housing. Test section pressure was controlled by regulating flow of steam to the housing and relieving excess pressure to the atmosphere during test runs.

A VIDAR digital data acquisition system was used to scan the heater rod thermocouples at a rate of 25 channels per second. After the data were digitized they were recorded on magnetic tape and later reduced and plotted on an off-line CDC-6600 digital computer. Time correlation with start of flooding was achieved by the VIDAR scan of a synchronous timer that was started with the VIDAR startup and by a flooding event timer that was started at initiation of flooding. Local coolant temperature and pressure were continuously monitored on pen recorders and Visicorders. Housing temperatures were recorded on a small Hewlett-Packard digital data acquisition system. Coolant flow rate was recorded. The power of each radially peaked section of the heater rod array was measured by potential and current transformers. The outputs of these transformers were continuously recorded.

3. TEST PROCEDURE

A brief description of the test operating procedures is provided to aid in understanding the system functions of the BWR-FLECHT facility and the PWR-FLECHT facility during experimental runs.

3.1 BWR-FLECHT

Loop heatup was accomplished with the loop full of water at a temperature below the saturation temperature for the given pressure. The main loop pumps were used to establish flow from the 120-kW heater and steam drums, through the subcooler, through the test vessel, out the steam makeup line, and back to the steam drums. Flow to the heater was at least 10 gpm. Power was applied to the bundle heaters until the system temperature reached the saturation temperature for the test pressure. The system pressure was raised 50 psi above the required test pressure. Pressure was controlled by bleeding coolant from the condenser drain.

To establish test conditions, the bundle power was turned off and the heater power controlled to maintain the test temperature. The main loop pumps were stopped and the bundle drained by the drain pump. The steam drums were drained to about half full. The pressure controller was set to the desired test pressure and the appropriate vent valve was activated to reduce chamber pressure to the required test pressure. The flow control throttle valve was set to the predetermined desired flooding rate.

The transient test was begun by adjusting the bundle power to 250 kW. When the cladding temperature prescribed for test initiation was obtained, the power decay function generator and main loop pumps were started simultaneously. When the water level in the bundle reached the eight-foot elevation, the flow was throttled to maintain the level at this elevation. The system pressure and coolant temperature were adjusted as necessary during the test. When the bundle reached saturation temperature, the power was shut off and the test was terminated.

3.2 PWR-FLECHT

The accumulator was filled with demineralized water and pressurized with gas to about 500 psi above the test pressure of the flow housing. The accumulator was maintained at a constant pressure during the experimental run and the coolant temperature was established by means of external clamp-on heaters. A small amount of water was allowed to enter the test section housing to a level just below the heated length of the test rods.

The housing walls were heated by supplying power to the test bundle for a short period of time and allowing the housing to heat up by radiant heat transfer. Clamp-on heaters along the upper and lower ends of the housing heated these regions, thus allowing the housing to achieve a temperature distribution similar to that of the rods. When the housing walls reached the specified temperature distribution, the housing was then pressurized with steam from the boiler. Coolant drainage and trace heating were utilized to assure that the coolant in the test section housing and in the pipe line between the test section housing and the accumulator was at the correct temperature when water injection was initiated.

Power was supplied to the bundle and the rods were allowed to heat up until the desired initial temperature was reached, at which time flooding was automatically initiated. On initiation of coolant injection, decay of test rod power was automatically started at a programmed rate corresponding to the reactor decay heat. Test housing pressure was regulated by releasing steam to the atmosphere. The ejected water, after partial separation, was drained from the upper plenum, collected in the catch vessel, and measured. The injection of water was terminated manually and the power was turned off after the thermocouples at the upper elevations were quenched.

4. TEST CONDITIONS

The FLECHT experiments were basically investigations of the effect of several independent parameters on the heat transfer process during ECC injection. The parameters of interest were peak power, initial cladding temperature, cold flooding rate, chamber pressure, and inlet coolant subcooling. The BWR-FLECHT SS-4 series primarily examined effects of pressure, whereas the PWR-FLECHT tests examined all five variables. The base line or reference test conditions and ranges of variation which were planned are presented in Table III.

TABLE III
FLECHT PLANNED NOMINAL TEST PARAMETERS

<u>Parameters</u>		<u>BWR-FLECHT</u>	<u>PWR-FLECHT</u>
Peak Power (kW/ft)	Reference	0.738	1.24
	Maximum	---	1.40
	Minimum	---	0.69
Initial Bundle Temperature (°F)	Reference	1300	1600
	Maximum	1600	2200
	Minimum	1300	800
Cold Flooding Rate (in./sec)	Reference	3	6
	Maximum	6	10
	Minimum	3	1
Chamber Pressure (psi)	Reference	60	60
	Maximum	300	90
	Minimum	15	15
Inlet Coolant Subcooling (°F)	Reference	140	140
	Maximum	300	180
	Minimum	50	25

A representative amount of data is reviewed in this report; all of the results of the FLECHT program are not presented. Notably omitted are the PWR-FLECHT work on coolant fallback, different power decay curves, flow blockage, and variable flooding rate. Table IV summarizes the bottom flooding FLECHT data studied in this report. Other tests at lower flooding rates were omitted because of the lack of data or inconsistencies in the data. The data

TABLE IV

FLECHT DATA SUMMARY, BOTTOM FLOODING BWR-FLECHT AND PWR-FLECHT --
SIX-FOOT ELEVATION ON HOTTEST ROD

Project	Test	Pressure (psia)	Peak Power (kW/ft)	Flooding Rate (in./sec)	Inlet Subcooling (°F)	Initial Temperature (°F)	Turnaround Time (sec)	Quench Time (sec)	Bundle Size	Quench Velocity (in./sec)
PWR-FLECHT	0105	56	1.24	6.0	137	801	36	69	7x7	0.71
PWR-FLECHT	0307	58	1.24	9.8	139	1210	5	47	7x7	1.25
PWR-FLECHT	0408	59	1.24	10.0	140	1625	3	47	7x7	1.20
PWR-FLECHT	0509	58	1.24	9.9	137	2017	3	56	7x7	1.03
PWR-FLECHT	0711	15	1.24	5.9	137	1600	15	100	7x7	0.54
PWR-FLECHT	0913	15	1.24	9.8	136	1601	6	56	7x7	0.79
PWR-FLECHT	1002	56	1.24	6.0	137	1605	6	68	7x7	0.81
PWR-FLECHT	1314	57	1.24	5.8	189	1604	8	60	7x7	0.92
PWR-FLECHT	1417	90	1.24	5.8	150	1611	7	72	7x7	0.79
PWR-FLECHT	1615	30	1.24	5.8	130	1594	9	98	7x7	0.58
PWR-FLECHT	1720	61	1.24	5.9	25	1617	7	152	7x7	0.40
PWR-FLECHT	2322	15	1.24	4.0	148	1612	43	185	7x7	0.28
PWR-FLECHT	4027	57	0.69	1.9	141	1592	28	128	10x10	0.49
PWR-FLECHT	4129	60	1.40	1.9	134	1596	55	222	10x10	0.26
PWR-FLECHT	4225	59	1.24	1.9	139	1596	35	190	10x10	0.30
PWR-FLECHT	4526	57	0.69	5.9	139	1608	5	50.5	10x10	1.11
PWR-FLECHT	4628	59	1.40	5.9	150	1606	6	68	10x10	0.65
PWR-FLECHT	5123	55	1.24	1.9	24	1607	45	242	10x10	0.23
PWR-FLECHT	5730	60	1.24	6.0	145	1602	7	63	10x10	0.76
PWR-FLECHT	6155	57	1.24	5.9	143	2212	3	78	7x7	0.78
PWR-FLECHT	6256	60	1.24	3.9	134	2199	4	100	7x7	0.57
PWR-FLECHT	6351	59	1.24	1.0	136	1795	57	246	7x7	0.22
PWR-FLECHT	6553	59	1.24	1.0	148	2012	62	260	7x7	0.22
PWR-FLECHT	6948	58	1.24	1.0	147	1602	95	268	7x7	0.23
PWR-FLECHT	7158	54	1.24	2.0	149	2156	9	153	7x7	0.44
BWR-FLECHT	SS4-10	21	0.738	3.2	115	1400	13	138	7x7	0.62
BWR-FLECHT	SS4-11X	31	0.738	2.8	100	1400	9	134	7x7	0.62
BWR-FLECHT	SS4-12	45	0.738	2.6	130	1430	8	95	7x7	0.98
BWR-FLECHT	SS4-13X	60	0.738	1.0 to 3.2	175	1360	40	104	7x7	0.25 and 0.62
BWR-FLECHT	SS4-14R	60	0.738	3.0	165	1410	7	71	7x7	1.07
BWR-FLECHT	SS4-15Z	58	0.738	3.0	50	1360	8	78	7x7	1.01
BWR-FLECHT	SS4-16X	97	0.738	3.0	180	1365	6	60	7x7	1.20
BWR-FLECHT	SS4-17	195	0.738	3.0	242	1410	2	32	7x7	2.18
BWR-FLECHT	SS4-18	330	0.738	3.0	295	1430	3	34	7x7	2.13
BWR-FLECHT	SS4-19	52	0.738	3.5	145	1745	5	93	7x7	0.78
BWR-FLECHT	SS4-20	58	0.738	5.65	162	1734	4	72	7x7	0.98

as presented represent the actual test conditions and the resulting temperatures for the six-foot elevation on the hot rod. A parametric map for the tests studied is given in Table V.

TABLE V

FLECHT PARAMETERS, BOTTOM FLOODING · BWR-FLECHT AND PWR-FLECHT

Variable		BWR SSI-20 PWR 5730	BWR SSI-152 BWR SSI-14R BWR SSI-13X PWR 4225 PWR 5123 PWR 1720 PWR 1314 PWR 1002	PWR 1417 PWR 1002 PWR 1615 PWR 0711 BWR SSI-18 BWR SSI-17 BWR SSI-16X BWR SSI-14R BWR SSI-12 BWR SSI-11X BWR SSI-10	BWR SSI-20 BWR SSI-19 PWR 6155 PWR 6256 PWR 7158 PWR 0913 PWR 0711 PWR 2322 PWR 0408 PWR 1002 PWR 4225 PWR 6948	PWR 6553 PWR 6351 PWR 6948 PWR 0509 PWR 0408 PWR 0307 PWR 6155 PWR 1002 PWR 0105	PWR 4628 PWR 5730 PWR 4526 PWR 4129 PWR 4225 PWR 4027
Peak Power (kW/ft)	0.738	X		X X X X X X X	X X		
	0.69						
	1.24	X	X X X X X X X	X X X X X X X X X			
	1.40						
Initial Bundle Temperature (°F)	800						
	1200						
	1400						
	1600	X X	X X X X X X	X X X X X X X	X X		
	1800						
	2000						
Cold Flooding Rate (in./sec)	1						
	2		X X X		X		
	3						
	4						
	6	X X	X X X	X X X X X X X	X X X X X X X		
Chamber Pressure (psia)	10						
	15						
	30						
	45						
	60	X X	X X X X X X	X X X X X X X X X	X X X X X X X X X		
	90						
	100						
Inlet Subcooling Temperature (°F)	200						
	300						
	25						
	50						
	100						
	115						
	130						
	140	X	X X X X X X	X X X X X X X X X	X X X X X X X X X		
	160	X					
	180						
	240						
	300						

V. QUALIFICATION OF DATA AND SUPPORTING EVALUATION

Representative surface temperatures and heat transfer coefficients calculated from heater cladding thermocouples in the 7 x 7 BWR-FLECHT array and in both 7 x 7 and 10 x 10 PWR-FLECHT arrays are presented. The radial and axial effects observed in these bundle data are noted and the basis for, and selection of, the most representative data locations for each test are given. The effects on heat transfer coefficient of variation of bundle power, initial cladding temperature, cold flooding rate, chamber pressure, and inlet coolant subcooling are discussed.

1. DATA DEFINITION AND REDUCTION

The primary information derived from the FLECHT tests were temperature-time data recorded by thermocouples located within (BWR-FLECHT) or beneath (PWR-FLECHT) the heater rod cladding. These data were recorded at numerous locations radially and axially throughout all the test bundles. From these temperature-time data the cladding surface temperatures and heat transfer coefficients were calculated. An example relating a typical temperature trace to a corresponding heat transfer coefficient curve is shown in Figure 23. In this plot, as in all subsequent plots in this report, the zero or starting time corresponds to the time at which cooling water entering the test chamber reaches the bottom of the heated length of the bundle rods. This event was monitored by thermocouples located in the coolant path at the bottom of each bundle. Quench time is the time when the observed thermocouple begins to cool rapidly as the cooling regime changes from film boiling to nucleate boiling. Quenching occurs when the incoming liquid has reached and passed the observed thermocouple and the thermocouple has reached liquid saturation temperature. Quenching is characterized by a rapid increase in heat transfer coefficient from below 100 to over 1,000 Btu/hr-ft²-°F. As noted in Figure 23, the classic flow regimes experienced in nuclear bundle simulators with cooling due to bottom flooding as described by Jensen^[7], were evident in the FLECHT experiments.

The bundle power, described as the peak power per foot of rod length on the highest power rod at transient initiation, was carefully controlled and monitored throughout all experiments. The initial bundle temperature

Fig. 23 Typical FLECHT surface temperature and heat transfer coefficient.

(the temperature of the hottest observed spot in the cladding of the hottest rod in the bundle at transient initiation) was used as the indicator during pretest heatup of when to initiate flooding. The planned test initiation temperatures were sometimes exceeded because of instrument error or failure to monitor the location of the peak temperature. The chamber pressure and coolant inlet temperature were satisfactorily recorded in both FLECHT projects. Some problems were experienced with the BWR-FLECHT chamber pressure regulation and coolant temperature control systems as evidenced by Figure 24. To provide workable numbers, these data have been averaged over the test period as shown in the figure. Several of the BWR-FLECHT cold flooding flow rates were somewhat unsteady and also required averaging. In one test,

Fig. 24 BWR-FLECHT Test SS4-20 coolant temperature and pressure.

Test SS4-13X, the transient flooding rate was 1 in./sec for one minute and then the specified 3 in./sec. The data from this test have been included to illustrate variable flooding rate effects. The usual procedure in the BWR-FLECHT experiments was to flood at the specified rate until a pressure corresponding to an eight-foot head of water was measured. Eight feet corresponds to the maximum level to which a BWR core can be flooded during an LOCA (at eight feet all additional coolant spills out the jet pumps of the recirculation system).

No problems were experienced in maintaining or recording bundle pressure, bulk inlet and exit coolant temperatures, or flooding rate in PWR-FLECHT experiments. Local bundle coolant conditions, such as density, temperature, and velocity could not be determined because of inaccuracies or complete lack of data. Unshielded thermocouples used in these FLECHT tests to measure fluid temperatures have proven unreliable as local temperature detectors in a two-phase flow. Such detectors require shielding to prevent radiation heat transfer and to provide protection from impingement by entrained liquid.

Thermocouples were located at specified axial and circumferential positions in the rods and the rods were inserted at specific locations in the bundles; however, no effort was made in the FLECHT program to align the thermocouples in any rod relative to other rods in the bundle. For this reason no reference is made to the circumferential location of any thermocouple. Some thermocouples on rods in the outer row exhibited different behavior than others in the same area, possibly because of the view presented to the thermocouple; that is, another hot rod or the cooler housing or channel may have affected the thermocouple. The rods near the center of the bundles exhibited nearly the same temperature behavior; therefore, any circumferential effect is considered negligible near the bundle center.

Calculations of the local heat transfer coefficients used in this work were accomplished by the DATAR computer code developed by Westinghouse Electric Corporation and are described in Appendix B of Reference 9. The code performs a transient conduction calculation based on the known temperature of the inside diameter of the cladding in PWR-FLECHT or in the middle of the cladding in BWR-FLECHT. The DATAR code calculates the rod surface temperature, surface heat flux, and surface heat transfer coefficient as functions of time by using the known temperature, the known electrical power generation rate at the thermocouple, and the temperature-dependent material properties of the rod at the thermocouple and by assuming the coolant is at the saturation temperature. The code solves the following one-dimensional energy balance and determines the total heat transfer coefficient (the sum of radiation and convection heat transfer coefficients) as though the heat transfer were all due to convection. Axial conduction has been shown to be insignificant^[9]. Therefore:

$$k(T) \frac{dT}{dr} = h_T (T_s - T_{sat}) \quad (1)$$

where

$k(T)$ = temperature-dependent thermal conductivity of rod cladding

$\frac{dT}{dr}$ = rate of temperature change with radial distance at rod surface; derived from power generation rate, rod material properties, and thermocouple readings

h_T = total heat transfer coefficient at rod surface, the quantity to be determined

T_s = temperature at rod surface; derived from power generation rate, rod material properties, and thermocouple readings

T_{sat} = bulk temperature of the coolant and assumed to be saturation temperature at measured chamber pressure.

2. EFFECT OF RADIAL POSITION

Radial effects on temperature and heat transfer coefficient, including size effects of 7 x 7 and 10 x 10 arrays, are discussed through use of data from the six-foot level in each bundle.

2.1 BWR-FLECHT

The surface temperature and heat transfer coefficient for two BWR-FLECHT tests conducted under different conditions are presented in Figures 25 and 26 to demonstrate the radial differences in temperature versus time and heat transfer versus time encountered in the BWR-FLECHT bundle. Data from thermocouples at the six-foot elevation starting at the outer edge and progressing toward the center of the bundle are shown. During the test, several thermocouples on Rod 25, the central rod of this bundle, failed. For this reason, data from Rod 17, which has been observed in numerous BWR-FLECHT experiments to behave similarly to Rod 25, have been substituted. One of the thermocouples on an inner rod (Rod 24) quenches prior to the others in both tests as a result of the low radial power peaking factor at the Rod 24 location (Figure 12). The remaining three thermocouples in both runs quenched at almost the same time. The thermocouple on Rods 17 and 23 generally exhibited the highest temperatures. The heat transfer coefficients, h , reflect these results. These

Fig. 25 Radial differences in temperature and heat transfer coefficient at the six-foot elevation - BWR - FLECHT Test SS4 - 10.

findings indicate that the BWR-FLECHT bundle maintained reasonably constant temperatures and heat transfer coefficients radially except for some slightly (<10%) higher heat transfer coefficients on the outer rods. For subsequent discussions in this report the inner rod, Rod 17, corresponding in behavior to the center rod (Rod 25), is defined to be the hottest rod in this bundle.

Fig. 26 Radial differences in temperature and heat transfer coefficient at the six-foot elevation -- BWR-FLECHT Test SS4-19.

2.2 PWR-FLECHT

Two PWR-FLECHT test runs are examined to determine the radial differences in temperature and heat transfer versus time. One set of plots is presented for the 7 x 7 array and one set for the 10 x 10 array.

Figure 27 presents the temperature and heat transfer coefficient versus time for the 7 x 7 array at the six-foot elevation. The plan view of the bundle shows that the central position is occupied by a control rod thimble which does not produce power, thus only three rods are examined. The outer rod, Rod 1D, has a much lower temperature during the initial stage of the transient, whereas the heat transfer coefficient is twice that of the neighboring location on Rod 2D. The inner rod, Rod 3D, behaves similarly to the adjacent rod, Rod 2D, because the inner rod is adjacent to the thimble. The effect of the thimble to reduce the adjacent rod temperatures is almost totally counteracted by the higher radial power peaking factor of Rod 3D (Figure 15). The temperature and heat transfer curves of all three rods tend to converge as the transient progresses; quenching of all thermocouples occurs at nearly the same time with Rod 3D maintaining a higher temperature somewhat longer. For subsequent discussions Rod 3D is assumed to represent the hottest rod in the 7 x 7 bundle.

The 10 x 10 PWR-FLECHT array radial profiles are examined in Figure 28. The figure indicates that the design of this bundle resulted in a uniform thermal behavior. Rod 1F in the outer row and Rod 2G in the second row exhibited some increase in heat transfer early in the transient. The temperature of the outer rod was below that of the other rods at the start of the transient and remained lower until quenching at which time it converged to the temperatures of the other rods. The temperatures in Figure 28 were compared to determine the hottest rod. Rod 5F was chosen as the hottest rod in this array.

A comparison of the data from the thermocouples at the six-foot elevation on the hottest rods from the two arrays was made to evaluate the effect of bundle size on the PWR-FLECHT results. Results from Test 1002, using the 7 x 7 array, are compared in Figure 29 with results from Test 5730, using the 10 x 10 array under similar conditions. The larger bundle quenched more rapidly, apparently because of a slightly higher coolant inlet subcooling and higher chamber pressure. As will be discussed later, increases in pressure and subcooling tend to enhance cooling. The similarity of the two curves in Figure 29 led to the conclusion that PWR-FLECHT bundles having 100 rods or 49 rods, if operated under the same test conditions, will produce similar results.

Fig. 27 Radial differences in temperature and heat transfer coefficient at the six-foot elevation of the 7 x 7 PWR-FLECHT array - Test 6553.

Fig. 28 Radial differences in temperature and heat transfer coefficient at the six-foot elevation of the 10 x 10 PWR-FLECHT array - Test 4225.

Fig. 29 Effect of bundle size on heat transfer coefficient, PWR - FLECHT, six - foot elevation on the hot rod.

3. EFFECT OF AXIAL POSITION

The effect of axial position on temperature and heat transfer coefficients for both BWR- and PWR-FLECHT bundles is presented for representative thermocouple locations. The time-dependent axial profiles are discussed in Section VI-2.

3.1 BWR-FLECHT

Axial temperature and heat transfer coefficients observed in the BWR-FLECHT experiments are depicted in Figure 30. A complete set of axial plots for the hottest rod is presented in Appendix A; plots are included for each of the 11 runs under examination. The temperature versus time curves of Test SS4-19 followed the usual pattern with the midplane (six-foot elevation) thermocouple starting several hundred degrees above the thermocouples at the four- and eight-foot elevations. The thermocouple at the 1.2-foot elevation exhibited about 500°F lower temperature than thermocouples nearer the midplane. The thermocouple at the 10.8-foot elevation failed during the test.

Fig. 30 Axial differences in temperature and heat transfer coefficient - BWR - FLECHT Test SS4 - 19.

Temperatures at equal distances from the ends of the rod are customarily nearly equal at test initiation.

As the cooldown transient progressed, the lower three thermocouples retained the same relative temperature differences and quenched in order of increasing distance from the bottom of the rod. The thermocouple at the eight-foot elevation quenched slightly later than that at the six-foot elevation.

The thermocouple at the 10.8-foot elevation is assumed to have quenched at a time similar to that at the eight-foot elevation. As shown in Figure 31, under conditions favorable to entrainment of water droplets in the outgoing steam, the upper thermocouples quench prior to the arrival of the advancing wetting front. This phenomenon has been observed in both FLECHT projects and is believed to have occurred when the entrained water passing out the exit of the bundle subsequently wetted the rods from above.

Fig. 31 Effect of water droplet entrainment on quenching at the upper elevations -- BWR - FLECHT Test SS4 - 11X.

The heat transfer coefficients generally had the same value at initiation of the transient. The heat transfer coefficients for the lower elevations reflected the early quenching of the lower temperature positions.

Values of the heat transfer coefficient, h , during the film boiling phase of cooldown remained between 20 and 40 Btu/hr-ft²-°F in the BWR-FLECHT bundle with highest temperature locations normally exhibiting the lowest h values. Thermocouples at the 1.2-foot elevation always reflected the highest heat transfer.

To simplify comparison from test to test, a single thermocouple location was selected for the BWR-FLECHT Bundle SS-4. The midplane (six-foot elevation) on the hottest rod (Rod 17) represents the location of highest temperature in the bundle at transient initiation and is thus the point of greatest interest in studies of ECC system effectiveness and behavior.

3.2 PWR-FLECHT

Test 4225, a low flooding rate (1.9 in./sec) test performed with a 10 x 10 bundle, has been selected to illustrate the axial differences in a PWR-FLECHT array. Figure 32 shows the temperature and heat transfer coefficient at various elevations during this test. A complete set of hottest rod axial heat transfer coefficients is presented in Appendix B. As previously described, the temperatures at the two- and ten-foot heights were approximately the same at transient initiation, whereas those at the four- and eight-foot heights were similar and several hundred degrees higher. The midplane temperature was several hundred degrees higher than the temperatures at the other monitored locations. As cooldown progressed the lower thermocouples quenched in order of increasing elevation as the liquid level moved upward in the bundle. A review of the radial effects indicate that the outer rods (and housing) quenched somewhat earlier than the main body of the array thus forming a concave quench front. Figure 32 illustrates the classic high bundle temperature, low flooding rate shape with upper elevations actually increasing in temperature as lower points were quenching. The thermocouple of the ten-foot elevation in this case wetted from above and quenched prior to that at the eight-foot location.

Fig. 32 Effect of elevation on temperature and heat transfer coefficient -- PWR - FLECHT Test 4225.

The heat transfer coefficients of Test 4225 indicate that all locations exhibited about the same heat transfer at the start of the transient. A value of 4 Btu/hr-ft²-°F has been observed as the initial average coefficient throughout FLECHT.

Figure 32 clearly shows the decrease in heat transfer coefficient with increase in elevation at any given time. This trend is evident throughout the PWR-FLECHT results with the outstanding exception of the ten-foot elevation. Top quenching is seen in Figure 32 to occur just prior to wetting of the eight-foot elevation. In the high flooding rate runs, the upper elevation actually quenched at a rate equal to or greater than the two-foot location.

To simplify subsequent comparisons from test to test, a single thermocouple location (the six-foot elevation on the hottest rod) was selected for each of the rod arrays. In the PWR-FLECHT 7 x 7 array, this location is the six-foot elevation on Rod 3D, and in the 10 x 10 array it is the six-foot elevation on Rod 5F.

4. EFFECTS OF PARAMETRIC VARIATION

The effects on heat transfer coefficient of variation in test parameters are briefly discussed. The major parameters that were investigated by the FLECHT program were listed in Table V. A complete set of traces of heat transfer coefficients for the six-foot elevation of the hot rod versus real time and versus dimensionless time are presented in Appendix C along with smoothed or averaged curves of the heat transfer coefficient versus dimensionless time. Dimensionless time, as used in this work, is defined as the actual time divided by the time to quench at the elevation of interest and is used to relate all heat transfer data to the same reference.

4.1 Power Variation

Power variation experiments in PWR-FLECHT included tests at three power levels measured at the locations of peak power in the bundle. These power levels were investigated at several cold flooding rates with other parameters maintained constant. Figure 33 presents the heat transfer coefficients calculated for six tests. The data for the tests at the lower flooding rates show that the quench time was lengthened with an increase in peak power.

Fig. 33 Effect of peak power on the heat transfer coefficient for the six - foot elevation on the hottest rod - PWR - FLECHT.

The more rapid increase in heat transfer during the lower powered runs at both of the flow rates investigated was caused by a higher velocity quench front as can be seen in Figure 34 in which the quench rates are plotted as a function of power. These rates are taken from the plots of quenching rates of the hottest rod included in Appendix D and represent the rate at which the quench front progresses up the rod. The higher velocity of the quench front in the lower power tests is a result of lower temperature coolant reaching the quench front. As is demonstrated in Section V-4.5, the higher coolant subcooling (lower temperature) is an important parameter affecting the heat transfer in the bundles. The increase in quench time at the higher flooding rates is not as easily detected because of short time spans involved and the resulting overlapping of curves. High entrainment rates with high flooding rates caused this similarity of data. The relative similarity of the plots is evident in the comparison shown in Figure 35 on a dimensionless time base. The smoothed data, which were drawn as estimated averages, are shown to provide a clearer

Fig. 34 Variation of quench rate with power - PWR - FLECHT, six - foot elevation on the hottest rod.

indication of the differences in the data. The magnitudes of the coefficients at temperature turnaround and prior to quenching are almost equal. The conclusion is that an increase in power decreases the heat transfer coefficient and delays the quench time.

4.2 Test Initiation Temperature Variation

PWR-FLECHT tests investigating the effect of the maximum initial cladding temperature were conducted over a range from 800 to 2200°F for several bottom flooding rates with other conditions remaining the same. Figure 36 presents the results of six tests. Tests at lower temperatures were expected to exhibit earlier quenching because the bundle energy was lower; however, the low temperature test at the 1-in./sec flooding rate quenched later than the tests having the higher initial temperatures and the low temperature test at the 6-in./sec flooding rate quenched after the tests at the intermediate temperature. But, when compared on a dimensionless time scale, as in Figure 37, all traces exhibit the same general trends. Figure 38 shows the quench velocities of PWR-

Fig. 35 Variation of heat transfer coefficient with power, dimensionless time base -- PWR-FLECHT, six-foot elevation on the hottest rod.

Fig. 36 Variation of heat transfer coefficient with initial temperature -- PWR - FLECHT, six - foot elevation on the hottest rod.

FLECHT tests as functions of initiation temperature. The conclusion is that the test initiation temperature has little effect on heat transfer coefficient behavior; any differences exhibited by the test results are masked by data scatter.

4.3 Cold Flooding Rate Variation

The cold flooding rate was examined in PWR-FLECHT through a range of 1 to 10 in./sec for varied chamber pressures and at several initial temperatures. Presented in Figure 39 are the 1600-°F, 60-psi data for flooding rates of 1, 2, 6, and 10 in./sec. Increasing the flooding rate reduced the quench time and increased the magnitude of the heat transfer coefficient throughout the transient. Figure 40 demonstrates the overall increase in the heat transfer coefficient with flooding rate increase on a dimensionless time base. This increase in heat transfer was caused by an increased fluid availability for the higher flood rates; thus steam velocity and entrainment increased with a subsequent increase in cooling[7]. The 10-in./sec test quenched at a lower heat

Fig. 37 Variation of heat transfer coefficient with initial temperature, dimensionless time base - PWR-FLECHT, six-foot elevation on the hottest rod.

Fig. 38 Variation of quench rate with initial temperature - PWR-FLECHT, six-foot elevation on the hottest rod.

transfer coefficient than the other tests shown. This earlier quenching is believed to have been the result of additional cooling from liquid which had been entrained in the two-phase flow dropping out the upper end of the bundle.

Fig. 39 Variation of heat transfer coefficient with cold flooding rate - PWR - FLECHT, six - foot elevation on the hottest rod.

The quench rates plotted in Figure 41 demonstrate the effect of flooding rate. At the flooding rates investigated, the quench rate increased with increased flooding rate. The conclusion is that increased flooding rate increases heat transfer and decreases quench time.

4.4 Pressure Variation

The BWR-FLECHT project tests reviewed here investigated the effect of pressure over a range from 21 to 330 psia. The results of these tests are presented in Figure 42. The heat transfer increased significantly with an increase in pressure up to some pressure between 195 and 330 psi. At this undetermined pressure, the improvement in heat transfer was apparently reversed, because the 330-psi test run exhibited a slightly longer time to quench and the heat transfer behavior was similar to that of the 195-psi test. The lower pressure tests at 21 and 31 psi also produced nearly identical plots of the heat transfer coefficient. PWR-FLECHT data shown in Figure 43 for the 56- and 90-psi tests and also from the 15- and 30-psi tests are similarly grouped.

Fig. 40 Variation of heat transfer coefficient with cold flooding rate, dimensionless time base - PWR-FLECHT, six-foot elevation on the hottest rod.

Fig. 41 Quench rate dependence on flooding rate - PWR-FLECHT, six-foot elevation on the hottest rod.

Fig. 42 Effect on pressure on surface temperature and heat transfer coefficient -- BWR - FLECHT, six - foot elevation on the hottest rod.

Figure 44 presents the quench rates plotted as a function of pressure for the BWR-FLECHT and PWR-FLECHT data. The fitted curves for both sets of data approach a slope of zero at upper and lower pressure limits. The greater magnitude of the BWR-FLECHT curve is attributed primarily to the increase in subcooling with increased pressure experienced during the BWR-FLECHT tests as noted in Table IV. As discussed in the next section, Section V-4.5, an increase in subcooling shortens the quench time by increasing the quench

Fig. 43 Effect of pressure on heat transfer coefficient -- PWR - FLECHT, six - foot elevation on the hottest rod.

rate. Total two-phase mixture mass flux, at the upper elevations of the bundle, can be shown to decrease generally with increased pressure. This decrease in mass flux is reflected as an increase in the quench rate (Figure 44) which in turn results in an increase in the amount of water remaining in the bundle. Figure 45 shows, on a dimensionless time base, the overall increase in heat transfer caused by increasing the pressure in BWR-FLECHT. The 21- and 330-psi tests were omitted for clarity; the data from these two tests are coincident with the 31- and 195-psi test data, respectively.

Fig. 44 Variation of quench rate with pressure - BWR-FLECHT and PWR-FLECHT, six-foot elevation on the hottest rod.

The literature was reviewed in an attempt to determine the cause of the apparent reduced influence of pressure on the heat transfer coefficient at the higher pressures. The following paragraphs discuss the application of available correlations to the FLECHT results. No conclusive evidence has been found to explain the apparent reduced pressure influence.

Bromley has suggested a method of determining the effect of pressure on heat transfer coefficient during stable film boiling^[22]. He derived an equation that is similar to Nusselt's equation for horizontal condenser tubes with negligible radiation. Bromley's equation is:

$$h_{co} = C \left[\frac{k^2 \rho (\rho_l - \rho) g \lambda' C_p}{D \Delta t Pr} \right]^{1/4} \quad (1)$$

where the physical properties of the vapor are evaluated at the arithmetic average of the heater surface temperature and the saturation temperature of the liquid and where

Fig. 45 Variation of heat transfer coefficient with pressure, dimensionless time base -- BWR-FLECHT, six-foot elevation on the hottest rod.

h_{co} = convective heat transfer coefficient (Btu/hr-ft²-°F)

C = a constant determined by experiment depending on fluid properties and geometry

D = diameter of heater rod (ft)

Δt = temperature difference between the tube surface temperature and the liquid saturation temperature (°F)

Pr = Prandtl number of vapor = $\frac{\mu C_p}{k}$, where μ is the viscosity of the vapor

k = thermal conductivity of vapor (Btu/hr-ft-°F)

C_p = specific heat of vapor at constant pressure (Btu/lb-°F)

ρ = density of vapor (lb/ft³)

ρ_l = density of liquid at saturation conditions (lb/ft³)

g = acceleration of gravity (4.17×10^8 ft/hr²).

λ' = difference in heat content between the vapor at its arithmetic average temperature and the liquid at saturation temperature (Btu/lb).

Bromley recommends substitutions of the known values for the physical properties in the fourth root term at the pressures of interest to determine the pressure effects.

The BWR-FLECHT tests at 21, 31, 195 and 330 psi are used as examples of the effect of pressure variation on the heat transfer coefficient as determined by Bromley's equation. The following property values are for these pressure conditions:

PROPERTY	Test Pressure (psi)			
	21	31	195	330
$D(\text{ft})$	0.475	0.475	0.475	0.475
$\Delta t(^{\circ}\text{F})$	1170	1150	1030	985
$Pr^{[23]}$	1	1	1	1
$k(\text{Btu/hr-ft-}^{\circ}\text{F})^{[24]}$	0.0235	0.0235	0.0235	0.0235
$C_p(\text{Btu/lb-}^{\circ}\text{F})^{[25]}$	0.48	0.48	0.54	0.63
$\rho(\text{lb/ft}^3)^{[25]}$	0.0345	0.0505	0.351	0.620
$\rho_l(\text{lb/ft}^3)^{[25]}$	58.8	58.8	55.5	52.6

(Contd.)

PROPERTY	Test Pressure (psi)			
	21	31	195	330
$\lambda'(\text{Btu/lb})^{[25]}$	1125	1100	924	845
for solution of Equation (1)				
$\frac{h_{co}}{C} (\text{Btu/hr-ft}^2\text{-}^\circ\text{F})$	26.6	28.6	46.2	54.5

The overall effect of increasing the pressure from 31 to 195 psi, shown by this example, was to increase the heat transfer coefficient by about 60%. A similar comparison between the coefficients calculated for 21 and 330 psi shows an increase of 105%. These average increases can be compared to the 40 to 80% increase seen in the heat transfer coefficients of Figure 45 over the film boiling portion of the transient. As can be seen from the property values, the increases in vapor density, and specific heat with pressure were the primary causes of the increase in the heat transfer coefficient. Use of Bromley's equation does not provide an insight into the cause of heat transfer reduction at high pressures.

The general increase in heat transfer coefficient with increasing pressure has been shown by Addoms to occur also for pool boiling^[26]. However, Addoms noted a reduced effect of pressure on heat transfer at higher pressures.

Wallis and Collier presented a comprehensive coverage of the Bromley and Addoms correlations as well as similar correlations^[27]. They also included a correlation of heat transfer in the liquid deficient region,

$$\frac{h_c D}{k_g} = 0.023 \left\{ \frac{GD}{\mu_g} \left[x + \left(\frac{v_f}{v_g} \right) (1-x) \right] \right\}^{0.8} \left(\frac{C_p \mu}{k} \right)_g^{0.8} \left[1 - 0.1 \left(\frac{\rho_f}{\rho_g} - 1 \right)^{0.4} (1-x)^{0.4} \right] \quad (2)$$

that is similar to the Dittus-Boelter equation. In Equation (2), the subscript g represents vapor and the subscript f represents liquid and,

h_c = convective heat transfer coefficient

$\frac{GD}{\mu_g}$ = the two-phase Reynolds number

x = mixture quality

v = specific volume.

Other nomenclature are the same as those used in Equation (1). Lack of data concerning mixture quality makes solution of Equation (2) impossible. However, an increase in pressure would decrease the vapor specific volume while increasing the vapor specific heat and density. The first two effects would increase the heat transfer coefficient whereas the latter would decrease the coefficient somewhat. If data on quality were available for the FLECHT tests, Equation (2) might be used to explain the reduced effect of pressure at high pressures.

Increasing pressure has been shown in the preceding paragraphs to increase the heat transfer coefficient by increasing the heat transfer capabilities of the vapor in the nucleate boiling and film boiling regions and in the liquid deficient region of two-phase flow. Thus, an increase in heat transfer coefficient with increased pressure is shown to occur in the three regions of steam generation, the areas of the majority of heat transfer in the FLECHT bundles.

Thus, as shown by Figure 45 and the preceding discussion, heat transfer in the FLECHT bundles increased with pressure as predicted by the literature; however, as shown by Figures 42 and 43, this effect has an apparent unexplained upper limit.

4.5 Subcooling Variation

The effect of inlet coolant subcooling on heat transfer coefficient is shown in Figure 46. The coefficient increases with decreased subcooling early in the transient. This trend is reversed late in the transient with the high heat transfer and more rapid quenching attributed to the higher subcooling. These effects are even more evident in Figure 47 on a dimensionless time base.

Figure 48 shows the variation of quench rate with inlet subcooling for two different flooding rates. The quench rate increased (with an attendant decrease in quench time) more rapidly with increasing subcooling at the higher flood rate.

The conclusion was reached that an increase in subcooling caused a early slight drop in heat transfer but later improved cooling and reduced quench time. These results were more pronounced for the high flooding rate data.

Fig. 46 Effect of subcooling on heat transfer coefficient -- PWR - FLECHT, six - foot elevation on the hottest rod.

Fig. 47 Variation of heat transfer coefficient with subcooling, dimensionless time base -- PWR - FLECHT, six - foot elevation on the hottest rod.

Fig. 48 Variation of quench rate with subcooling - PWR-FLECHT, six-foot elevation on the hottest rod.

VI. DISCUSSION OF RESULTS

Heat transfer data are presented in plots both as function of real time and dimensionless time. Two sets of data for the hottest rod, one from each of the FLECHT bundles under similar operating conditions, are examined. The velocities of progression of the quench front up the bundle from these two sets of data are compared. Power profile differences and effects are discussed.

1. COMPARISON OF HEAT TRANSFER DATA FOR BWR-FLECHT AND PWR-FLECHT

The BWR- and PWR-FLECHT tests are compared here on the basis of one test from each project. These tests were performed under similar conditions with the primary exception of peak power which was 0.74 kW/ft for the BWR-FLECHT bundle and 1.24 kW/ft for the PWR-FLECHT bundle. To equate this power difference, the heat transfer coefficients for the hot rod were plotted on a dimensionless time base. As was shown in Figure 35, heat transfer coefficient data from similar tests with differing bundle powers produce similar if not exactly the same curves on a dimensionless time base. Comparison of the power decay curves for these experiments (Figures 19 and 22 of Section IV.2) shows that the PWR-FLECHT curve represents a normalized power of about 20% less than the BWR-FLECHT curve over the first 60 seconds. This difference in decay power is partially compensated for by the difference in radial power peaking between the hottest rods of BWR-FLECHT and PWR-FLECHT (0.98 and 1.10, respectively, for a 10% difference).

The two tests to be compared, Test SS4-20 (BWR-FLECHT) and Test 5730 (PWR-FLECHT) were performed at nominal conditions of 1600°-F initial temperature, 60-psi pressure, 6-in./sec flooding rate, and 140°-F subcooling. Test 5730 was performed with borated coolant (2000 ppm boric acid). Reference 11 notes that the boric acid addition increases heat transfer by a small amount. Actual test conditions were listed in Table IV (Section IV-4) along with results. The comparison curves of heat transfer coefficient on a dimensionless time base presented in Figures 49, 50, and 51, represent data for the hottest rod at the four-, six-, and eight-foot elevations and include smoothed, or averaged,

Fig. 49 Comparison of PWR-FLECHT and BWR-FLECHT heat transfer coefficients at the four-foot elevation on the hottest rods.

Fig. 50 Comparison of PWR - FLECHT and BWR - FLECHT heat transfer coefficients at the six - foot elevation on the hottest rods.

Fig. 51 Comparison of PWR-FLECHT and BWR-FLECHT heat transfer coefficients at the eight-foot elevation on the hottest rods.

curves for clarification. The dimensionless time base is referenced to the quench time at the elevation of interest.

In Figure 49 PWR-FLECHT and BWR-FLECHT heat transfer coefficients at the four-foot elevation both exhibit the same characteristics with the PWR-FLECHT data consistently higher than the BWR-FLECHT values. The smoothed curves show that a difference of 1.33 was maintained from $t^* = 0.4$ to 1.0. The heat transfer data for the six-foot elevation, Figure 50, yielded similar curves. The PWR-FLECHT coefficients were consistently higher than BWR-FLECHT coefficients. Smoothed heat transfer curves constructed from these data with the 1.33 difference reproduce the actual data curves closely. Figure 51 shows the comparison of coefficients calculated for the eight-foot elevation. The BWR-FLECHT tests exhibited an initially higher heat transfer in the first half of the transient whereas the PWR-FLECHT coefficient is slightly larger after $t^* = 0.55$. The flattening of the BWR plot is believed to have been caused by the reduction of coolant flow in the test to simulate the loss of water at eight feet through the jet pumps of a BWR. The conclusion was reached that the heat transfer coefficients at the eight-foot elevation are nearly the same in the two FLECHT projects.

An examination of the BWR-FLECHT test results in Figures 49, 50, and 51 indicates a definite change in heat transfer with elevation. The coefficient at the four-foot elevation was about $30 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$ early in the transient and slowly increased to the time of quenching. The shape of the coefficient for the six-foot elevation was similar to that at the four-foot elevation but with reduced slope. At the eight-foot elevation, the early value of the coefficient was over $40 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$ and this coefficient remained constant nearly to quenching. A similar review of PWR-FLECHT test results indicates greater slopes than BWR-FLECHT results in the plots at all elevations and a reduction of slope with elevation. Early values of the heat transfer coefficients at the four- and six-foot locations were approximately $40 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$ whereas at eight feet the coefficient was about $35 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$. These results indicate that the PWR-FLECHT heat transfer coefficient is 1.33 times higher at four and six feet than the BWR-FLECHT time-averaged coefficient when compared on a dimensionless time base. The significance of this higher coefficient is discussed further in Section VII.

2. AXIAL POWER PROFILE EFFECTS

The power profile of the electrical heater rods employed in both FLECHT projects was of chopped cosine shape simulating approximately 6% of the beginning-of-life operating power of a nuclear reactor bundle. The method of heatup of the bundles for both BWR- and PWR-FLECHT tests resulted in temperature profiles that reflected the shapes of the power profiles. The FLECHT temperature and power profiles are graphically presented in Figures 52 and 53; typical temperature profiles at transient initiation are displayed with the design axial profile. These data were taken from tests performed under similar operating conditions, Test SS4-20 (BWR-FLECHT) and Test 5730 (PWR-FLECHT), and Test SS4-10 (BWR-FLECHT) and Test 6155 (PWR-FLECHT).

The BWR-FLECHT profiles exhibit a steep constant slope at each end of the rod. A small increase in peaking factor to a constant power at the center two feet of the rod caused the temperature profile at midplane to turn over gradually. The PWR-FLECHT profiles, although similar in appearance to the BWR-FLECHT profiles, have several significant differences. The PWR-FLECHT power profile was formed with 23-inch long, constant power increments at each end of the rod and about a 14-inch long, constant power increment at the midplane. Other constant power increments of the rod were seven inches long compared to the 10-inch-long BWR-FLECHT constant power increments. The resulting profiles of power and temperature for PWR-FLECHT (Figure 53) have subsequently steeper slopes than the BWR-FLECHT profiles.

The extrapolated zero-elevation temperature at transient initiation on the hottest rod was 180°F for PWR-FLECHT Test 5730. This temperature was only slightly above the coolant temperature. The corresponding BWR-FLECHT temperature for Test SS4-20 was 800°F. Part of the difference between these temperatures was a result of the difference in midplane temperatures. The temperature of the BWR-FLECHT rod at the six-foot elevation was 1735°F whereas the PWR-FLECHT corresponding temperature was 1602°F. The significance of the difference in temperatures at the lower elevations between BWR-FLECHT Test SS4-20 and PWR-FLECHT Test 5730 at transient initiation is related to the effect on quench velocity. This effect is discussed in the next section.

Fig. 52 BWR - FLECHT axial power and temperature profiles at time of test initiation.

Fig. 53 PWR - FLECHT axial power and temperature profiles at time of test initiation.

The difference in slopes between high temperature tests (BWR-FLECHT Test SS4-20 and PWR-FLECHT Test 6155) and low temperature tests (BWR-FLECHT Test SS4-10 and PWR-FLECHT Test 5730) results from the larger power input to the midplane of the heater rods during the higher temperature tests.

Figures 54 and 55 show axial temperature profiles of BWR-FLECHT Test SS4-20 and PWR-FLECHT Test 5730 for steps of dimensionless time, t^* . The curve at $t^* = 0.24$ (about 17 seconds in real time) of Figure 54 corresponds in shape almost exactly to the $t^* = 0$ curve of Figure 55 and indicates that the PWR-FLECHT test was advanced 17 seconds in time over the comparable BWR-FLECHT test. The lower temperature at the bundle inlet in Test 5730 allowed incoming coolant to rise higher in this bundle than in the BWR-FLECHT bundle before departure from nucleate boiling. Thus, shortly after test initiation, Test 5730 experienced wetting up the rod to a greater height than did Test SS4-20. In general, the wetting front was two feet higher in PWR-FLECHT. The greater coolant depth in PWR-FLECHT resulted in the higher elevations in PWR-FLECHT being in closer proximity to the quench front than in BWR-FLECHT. These higher elevations thus cooled more rapidly due to the lower void fraction nearer the front. This phenomenon is investigated further in the next section.

An interesting phenomenon is evidenced by both temperature profiles shown in Figures 54 and 55. As the coolant enters the bundle and cools the lower end, the point of maximum temperature (hot spot) begins to shift upward (to the right in the figures) indicating that the cooling was less efficient at the higher elevations (midplane and above). The effect of hot spot shift was more pronounced in the PWR-FLECHT results. Although the midplane was not the hottest spot throughout the entire transient in either project, the exclusive use of data for the six-foot elevation covers the initial and highest temperature segment of each test.

Fig. 54 Axial temperature profiles for various dimensionless time, t^* , steps -- BWR - FLECHT Test SS4 - 20, Rod 17.

Fig. 55 Axial temperature profiles for various dimensionless time, t^* , steps -- PWR - FLECHT Test 5730.

3. QUENCH VELOCITY COMPARISONS

Because of the low power peaking at the ends of the rods in PWR-FLECHT bundles, cooling water advanced some finite distance up the rods before the film boiling regime was reached. Similar behavior did not occur in the BWR-FLECHT bundle because of the high initial temperature maintained at the ends of the rods. Plots of quench time with respect to elevation demonstrate how the quench front advanced up the rods. As shown in Figure 56 the liquid front (or quench front) in two PWR-FLECHT tests exhibited the characteristic of an initially high quench rate not evident in the BWR-FLECHT test data.

Fig. 56 Elevation of the quench front progression versus time after flooding.

PWR-FLECHT Tests 5730 and 4526 produced initial quench velocities during the first few seconds of the transients that approximated the cold flooding rates. Near the two-foot elevation, where the rod surface temperature reached the Liedenfrost temperature, film boiling commenced and the rate of quench front progression up the bundle was reduced to a rate dependent on the test parameters. This rate of quench front travel up the rods, the second slope of the curve, is called the average quench velocity, V_q , of the test. In the case of Tests 5730 and 4526 the difference in slope of V_q reflects the difference in peak power.

Because of the high (800°F) zero-elevation temperature, the film boiling regime existed from the very bottom of the heater section of rod for the BWR-FLECHT Test SS4-20. The quench velocity was established at a rate dictated by the bundle power. A constant rate was maintained until an elevation near six feet was reached at which time rapid quenching occurred. The quench fronts of the two tests with similar power and flooding rate, PWR-FLECHT Test 4526 and BWR-FLECHT Test SS4-20, progressed at similar rates, whereas the higher powered PWR-FLECHT Test 5730, with the same flooding rate as Test 4526 exhibited a lower quench rate. The conclusion is that, when all operating parameters were maintained equal in BWR-FLECHT and PWR-FLECHT bundles, the quench rates were equal. A complete set of plots of quench elevation versus time is presented in Appendix D.

VII. CORRELATION OF BWR-FLECHT AND PWR-FLECHT HEAT TRANSFER

The quench rates presented in Appendix D are used to arrive at a correlation between BWR- and PWR-FLECHT heat transfer. This correlation is compared to the test results of BWR-FLECHT Tests SS4-19 and -20 and PWR-FLECHT Tests 4027, 4526, and 5730. The effect of different axial peaking factors is also identified.

An estimate of the Reynolds number must be made to determine whether turbulent flow was established in the two-phase mixture in the bundles. No local velocity, density, and flow rate measurements were made in the FLECHT tests; therefore, overall data are used to determine the bulk Reynolds number.

The law of conservation of mass was applied to the mass flux in the bundle; that is,

$$G_{in} = G_{out} + G_{stored} \quad (3)$$

where

G = mass flow rate per unit area = mass flux

$G = \rho V$

$$G_{in} = \rho_f V_f \quad G_{out} = \rho_m V_m \quad G_{stored} = \rho_f V_q$$

and where the subscripts are defined as

f = saturated fluid

m = mixture of vapor and fluid

q = quench front.

Substituting the respective relationships for G into Equation (3) results in:

$$\rho_f V_f = \rho_m V_m + \rho_f V_q \quad (4)$$

Rearrangement of Equation (4) gives

$$\rho_m V_m = \rho_f (V_f - V_q) = G_{out} \quad (5)$$

One of the low flooding rate tests, PWR-FLECHT Test 6351 at 1.0 in./sec with a 0.22 in./sec quenching rate, was examined. The fluid density was taken as 61 lb/ft³[25]. Thus,

$$G_{out} = 3.97 \text{ lb/ft}^2\text{-sec.}$$

Reynolds number is expressed as

$$Re = \frac{G_{out} D_h}{\mu} = \frac{\rho V D_h}{\mu}$$

where

D_h = hydraulic diameter = 0.0444 ft

μ = assumed viscosity of saturated steam at test pressure (59) psi
 = 11.1×10^{-6} lb/sec-ft [24].

Substitution of D_h , G_{out} , and μ gives $Re = 1.6 \times 10^4$.

Reference 11 describes several tests performed in PWR-FLECHT in which entrained liquid exiting the bundle was captured and measured. These measurements showed that approximately 10% of the exiting mixture was liquid and 90% was steam by weight. The bulk Reynolds number was calculated for Test 6351 with the assumption that the mixture viscosity was equal to the viscosity of 100% steam, which is a good approximation. This low flooding rate test was well into fully developed turbulent flow, and all high flooding rate tests, with higher Reynolds numbers, exhibited turbulent flow.

Although local heat transfer coefficients have been computed, no local coolant mixture conditions such as quality or density can be calculated for the FLECHT tests because of lack of flow data. This lack of information makes selection of the proper two-phase heat transfer correlation difficult and precludes a rigorous solution with any correlation. Many two-phase flow correlations relating heat transfer for turbulent fluid flow in passages are based on the Dittus-Boelter equation for heating a fluid [27,28]:

$$Nu = \frac{h D_h}{k} = 0.023 Re^{0.8} Pr^{0.4}$$

where

Nu = Nusselt number

Re = Reynolds number

Pr = Prandtl number

h = total heat transfer coefficient at the location of interest

k = thermal conductivity.

A modification of this equation is recommended [29,30] to account for parallel flow in bundles of rods with spacing such as that found in power reactors:

$$Nu = C_1 Re^{0.8} Pr^{0.4} \quad (6)$$

with C_1 a constant dependent on fuel rod lattice arrangement; that is,

$$C_1 = 0.042 \left(\frac{\text{pitch}}{\text{diameter}} \right) - 0.024.$$

Substitution of Nusselts and Reynolds numbers into Equation (6) results in

$$\frac{h D_h}{k} = C_1 \text{Pr}^{0.4} \left[\frac{\rho V D_h}{\mu} \right]^{0.8} \quad (7)$$

where

h = total heat transfer coefficient at the location of interest

$D_h = \frac{4}{\pi d} (p^2 - \frac{\pi}{4} d^2) = \text{hydraulic diameter}$

p = center-to-center rod pitch

d = rod diameter

k = thermal conductivity

ρ = density

V = velocity

μ = viscosity.

Rearrangement of Equation (7) gives

$$h = C_1 \frac{k \text{Pr}^{0.4}}{D_h^{0.2}} \left[\frac{\rho V}{\mu} \right]^{0.8} \quad (8)$$

Substituting Equation (5) into Equation (8) and rewriting Equation (8) for the mixture results in

$$h = C_1 \frac{k_m \text{Pr}_m^{0.4}}{D_h^{0.2}} \left[\frac{\rho_f (V_f - V_q)}{\mu_m} \right]^{0.8} \quad (9)$$

Equation (9) presents the final form of the relation to be used to compare the heat transfer coefficients. The subscripts B for BWR and P for PWR are used to represent the respective FLECHT terms in a ratio of coefficients:

$$\frac{h_{\text{PWR}}}{h_{\text{BWR}}} = \frac{C_{1P}}{C_{1B}} \left[\frac{D_{hB}}{D_{hP}} \right]^{0.2} \left[\frac{\text{Pr}_{mP}}{\text{Pr}_{mB}} \right]^{0.4} \frac{k_{mP}}{k_{mB}} \left[\frac{(V_f - V_q)_P}{(V_f - V_q)_B} \right] \frac{\rho_{fP}}{\rho_{fB}} \frac{\mu_{mB}}{\mu_{mP}} \quad (10)$$

The FLECHT tests under comparison, Tests SS4-20 and 5730, were conducted under similar conditions; therefore, the mixture qualities and properties were assumed equal. (Although the mixture quality is known to increase with distance above the quench front^[11] the assumption of equal mixture qualities is expected to result in only a small error in the final correlation.) The saturated fluid velocity is taken as the cold flooding rate in the bundle and the quench velocity is taken as measured in FLECHT.

Cancellation of terms assumed to be equal and substitution of known values into Equation (10) results in a correlation of heat transfer coefficients dependent on bundle hydraulic diameter, lattice size, and fluid velocities which can be expressed as follows:

$$\frac{h_{PWR}}{h_{BWR}} = \frac{C_{1P}}{C_{1B}} \left(\frac{D_{hB}}{D_{hP}} \right)^{0.2} \left[\frac{(V_f - V_q)_P}{(V_f - V_q)_B} \right]^{0.8} \quad (11)$$

Evaluating hydraulic diameters, velocities, and constants for Tests SS4-20 and 5730 results in:

Hydraulic diameter	$D_{hB} = 0.648 \text{ in.}$	$D_{hP} = 0.532 \text{ in.}$
Lattice constant ^[29,30]	$C_{1B} = 0.030$	$C_{1P} = 0.032$
Cold flooding rate from Table IV	$V_{fB} = 5.65 \text{ in./sec}$	$V_{fP} = 6.00 \text{ in./sec}$
Quench front velocity from Table IV	$V_{qB} = 0.98 \text{ in./sec}$	$V_{qP} = 0.76 \text{ in./sec}$

The hydraulic diameters and lattice constants are derived from the bundle geometry as shown in the previous discussion. The cold flooding rate is a test parameter and the quench front velocity was derived from the slopes of the quench elevation versus time curves of Appendix D. For Tests SS4-20 and 5730 the predicted time-averaged heat transfer coefficient ratio is

$$\frac{\bar{h}_P}{\bar{h}_B} = 1.22$$

The calculated PWR-FLECHT time-averaged heat transfer coefficient is approximately 1.22 times higher than the BWR-FLECHT coefficient under these test conditions as compared to 1.33 times higher as shown by the four- and six-foot elevation test data of Figures 49 and 50. The only basic difference in the tests examined was the power level, 0.74 kW/ft for BWR-FLECHT and

1.24 kW/ft for PWR-FLECHT. If the velocities obtained from PWR-FLECHT Test 4526 of Figure 56 performed at 0.69 kW/ft are used in Equation (11), the predicted ratio of time-averaged heat transfer coefficients is

$$\frac{\bar{h}_P}{\bar{h}_B} = 1.13$$

Table VI presents a comparison of time-averaged heat transfer coefficient ratios between the test data and values calculated with Equation (11) for several groups of tests with different peak powers and different flooding rates.

TABLE VI
HEAT TRANSFER COEFFICIENT RATIOS AT FOUR- AND SIX-FOOT ELEVATIONS

Test Number	Flooding Rate (in./sec)	Peak Power (kW/ft)	Test Data Ratio	Calculated Ratio
			$\bar{h}_P/\bar{h}_B$	$\bar{h}_P/\bar{h}_B$
5730	6.0	1.24	1.33	1.22
SS4-20	5.65	0.738		
4526	5.9	0.69	1.18	1.13
SS4-20	5.65	0.738		
4027	1.9	0.69	0.58	0.44
SS4-20	5.65	0.738		
4027	1.9	0.69	0.65	0.66
SS4-19	3.5	0.738		

The use of Equation (11) generally underpredicts the heat transfer coefficient ratio somewhat. This underprediction is believed to be the result of the low axial peaking factor in the lower end of the PWR-FLECHT bundles which allowed the quench front to rise higher on these bundles. This additional rise in quench front caused the front to be closer to the thermocouples in question. The closer proximity of the front provided a lower quality steam with more fluid droplets to act as radiation heat sinks which resulted in higher heat transfer rates.

The author believes that Equation (11) can be used with good accuracy to predict time-averaged heat transfer coefficients for BWR- and PWR-FLECHT

at the four- and six-foot elevations. The theoretical correlation is used to predict the time-averaged heat transfer coefficients for BWR-FLECHT by using PWR-FLECHT heat transfer coefficients and test data (that is, quench rates); or alternately, PWR-FLECHT coefficients are predicted from BWR-FLECHT data. If quench data are lacking, the quench rates can be assumed equal when the flooding rates are equal. Development of a relation of quench velocity to pressure and coolant subcooling is suggested to make the correlation readily adaptable for predicting the heat transfer coefficients of bundles other than those of FLECHT geometry.

Equation (11) accounts for flooding rate, but as discussed in Section V-4, chamber pressure and coolant subcooling exert strong influences and must be maintained constant. As shown in Section V-4, initial bundle temperature and peak power have no appreciable effect on heat transfer over the range investigated.

VIII. CONCLUSIONS AND RECOMMENDATIONS

The temperature and heat transfer data obtained from FLECHT tests have been reviewed and a number of conclusions reached. Recommendations for further work in this area are presented.

1. CONCLUSIONS

Equation (11) predicts a variation between the BWR-FLECHT and PWR-FLECHT heat transfer coefficients that is dependent on the ratio of the differences between coolant flooding rate and quenching rate and inversely dependent on the ratio of hydraulic diameters. The correlation is considered valid on the basis of a comparison of axial temperature and axial quench rate profiles for BWR-FLECHT and PWR-FLECHT bundles.

Nuclear fuel bundle simulators of differing rod diameter and rod pitch operating under similar conditions and with like housing temperatures produce similar results. The geometry effects are reduced to the ratio of hydraulic diameters to the 0.2 power.

The differences in heat transfer coefficients for PWR-FLECHT and BWR-FLECHT described in the results section are believed to be a result of the enhanced cooling developed in the PWR-FLECHT tests by the initially rapid progression of the quench front to about the two-foot elevation. The reduced distance from the quench front to the elevations under investigation acted to increase the local heat transfer at those elevations in the PWR-FLECHT bundle.

The following general effects of parametric variation were the same for both BWR-FLECHT and PWR-FLECHT. Increasing power increased quench time with a decrease in heat transfer coefficient at all times. Initial temperature variation had little effect on either quench time or heat transfer coefficient behavior. An increase in the flooding rate increased the heat transfer coefficient and shortened the quench time. An increase in pressure increased the heat transfer coefficient and decreased the quench time within maximum pressure limits. Additional subcooling increased the heat transfer coefficient and decreased the quench time.

2. RECOMMENDATIONS

Cursory examination of the quench data presented in Section V-4 indicates that a simple correlation can be developed relating the variation of quench velocity with the major test parameters. The quench velocity results from BWR-FLECHT or PWR-FLECHT could then be used in Equation (11) to predict the time-averaged heat transfer coefficient of a different lattice size bundle having similar test conditions.

A method to determine the accuracy of Equation (11) would involve application of the Westinghouse Electric Corporation empirical heat transfer coefficient analysis of Section III-2 to the BWR-FLECHT data. A comparison of the results of this application to BWR-FLECHT with results of the application to comparable PWR-FLECHT tests is expected to verify Equation (11).

A FLECHT-type experiment should be run to verify the effect of rod pitch variation on the heat transfer coefficient. A series of bundles constructed with different heater rod pitches and diameters should be tested with the same test apparatus and under the same operating conditions. Sufficient fluid measurements to determine both fluid inlet and two-phase mixture outlet conditions should be made. Energy balances could then be performed to more completely evaluate the heat transfer results of FLECHT and the conclusions of this report.

The power profile used in PWR-FLECHT resulted in heat transfer coefficients which apparently are too high for direct application to reactor loss-of-coolant response calculations. The profile used in PWR-FLECHT had a low power peaking factor at the lower and upper ends of the rods compared to that in BWR-FLECHT. This low power peaking resulted in low temperature at the ends of the rods. These low temperatures in the PWR-FLECHT bundles allowed water to rise rapidly to about the two-foot elevation. The closer proximity of the quench front to the higher elevations in the bundle resulted in higher heat transfer coefficients at these elevations than in BWR-FLECHT. A less peaked profile should be used to provide a more nearly typical temperature distribution.

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APPENDIX A
PLOTS OF BWR-FLECHT TEMPERATURE AND HEAT TRANSFER
COEFFICIENT VERSUS TIME FOR THE HOTTEST ROD

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APPENDIX A

PLOTS OF BWR-FLECHT TEMPERATURE AND HEAT TRANSFER
COEFFICIENT VERSUS TIME FOR THE HOTTEST ROD

Fig. A-1 Variation of surface temperature and heat transfer coefficient with elevation — BWR-FLECHT Test SS4-10.

Fig. A-2 Variation of surface temperature and heat transfer coefficient with elevation -- BWR - FLECHT Test SS4 - 11X.

Fig. A-3 Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-12.

Fig. A-4 Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-13X.

Fig. A-5 Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-14R.

Fig. A-6 Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-15Z.

Fig. A-7 Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-16X.

Fig. A-8 Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-17.

Fig. A-9 Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-18.

Fig. A-10 Variation of surface temperature and heat transfer coefficient with elevation — BWR-FLECHT Test SS4-19.

Fig. A-11 Variation of surface temperature and heat transfer coefficient with elevation -- BWR-FLECHT Test SS4-20.

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APPENDIX B
PLOTS OF PWR-FLECHT HEAT TRANSFER COEFFICIENT
VERSUS TIME FOR THE HOTTEST ROD

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APPENDIX B
PLOTS OF PWR-FLECHT HEAT TRANSFER COEFFICIENT
VERSUS TIME FOR THE HOTTEST ROD

Fig. B-1 Variation of heat transfer coefficient with elevation -- PWR - FLECHT Test 0105.

Fig. B-2 Variation of heat transfer coefficient with elevation -- PWR - FLECHT Test 0307.

Fig. B-3 Variation of heat transfer coefficient with elevation -- PWR - FLECHT Test 0408.

Fig. B-4 Variation of heat transfer coefficient with elevation -- PWR - FLECHT Test 0509.

Fig. B-5 Variation of heat transfer coefficient with elevation - PWR - FLECHT Test 0711.

Fig. B-6 Variation of heat transfer coefficient with elevation--PWR - FLECHT Test 0913.

Fig. B-7 Variation of heat transfer coefficient with elevation -- PWR - FLECHT Test 1002.

Fig. B-8 Variation of heat transfer coefficient with elevation - PWR - FLECHT Test 1314.

Fig. B-9 Variation of heat transfer coefficient with elevation - PWR - FLECHT Test 1417.

Fig. B-10 Variation of heat transfer coefficient with elevation - PWR - FLECHT Test 1615.

Fig. B-11 Variation of heat transfer coefficient with elevation - PWR - FLECHT Test 1720.

Fig. B-12 Variation of heat transfer coefficient with elevation -- PWR - FLECHT Test 2322.

Fig. B-13 Variation of heat transfer coefficient with elevation -- PWR - FLECHT Test 4027.

Fig. B-14 Variation of heat transfer coefficient with elevation -- PWR - FLECHT Test 4129.

Fig. B-15 Variation of heat transfer coefficient with elevation - PWR - FLECHT Test 4225.

Fig. B-16 Variation of heat transfer coefficient with elevation -- PWR - FLECHT Test 4526.

Fig. B-17 Variation of heat transfer coefficient with elevation - PWR - FLECHT Test 4628.

Fig. B-18 Variation of heat transfer coefficient with elevation -- PWR - FLECHT Test 5123.

Fig. B-19 Variation of heat transfer coefficient with elevation - PWR - FLECHT Test 5730.

Fig. B-20 Variation of heat transfer coefficient with elevation -- PWR - FLECHT Test 6155.

Fig. B-21 Variation of heat transfer coefficient with elevation -- PWR - FLECHT Test 6256.

Fig. B-22 Variation of heat transfer coefficient with elevation -- PWR - FLECHT Test 6351.

Fig. B-23 Variation of heat transfer coefficient with elevation -- PWR - FLECHT Test 6553.

Fig. B-24 Variation of heat transfer coefficient with elevation -- PWR - FLECHT Test 6948.

Fig. B-25 Variation of heat transfer coefficient with elevation -- PWR - FLECHT Test 7158.

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APPENDIX C
PLOTS OF BWR-FLECHT AND PWR-FLECHT HEAT TRANSFER
COEFFICIENT VARIATIONS WITH RESPECT TO
VARIOUS MAJOR PARAMETERS

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APPENDIX C PLOTS OF BWR-FLECHT AND PWR-FLECHT HEAT TRANSFER COEFFICIENT VARIATIONS WITH RESPECT TO VARIOUS MAJOR PARAMETERS

The plots in this appendix show the variation of heat transfer coefficient with changes in the major parameters. The heat transfer coefficients are plotted with respect to real time and dimensionless time with the dimensionless time plot following the corresponding real time plot for a given set of tests. The dimensionless time plots contain heat transfer coefficients obtained from both actual and smoothed data. Parameters varied were peak power, initial temperature, flooding rate, pressure, and coolant subcooling.

Fig. C-1 Variation of heat transfer coefficient with peak power for flooding rates of 2 and 6 in. / sec - PWR - FLECHT, Rod 5 F, six - foot elevation.

Fig. C-2 Variation of heat transfer coefficient with peak power, dimensionless time base -- PWR - FLECHT Tests 4225, 4129, and 4027 at a flooding rate of 2 in./sec.

Fig. C-3 Variation of heat transfer coefficient with peak power, dimensionless time base -- PWR - FLECHT Tests 4526, 5730, and 4628 at a flooding rate of 6 in. / sec.

Fig. C-4 Variation of heat transfer coefficient with initial temperature for flooding rates of 1 and 6 in./sec -- PWR - FLECHT, Rod 3D, six - foot elevation.

Fig. C-5 Variation of heat transfer coefficient with initial temperature, dimensionless time base -- PWR - FLECHT Tests 6948, 6351, and 6553 at a flooding rate of 1 in. / sec.

Fig. C-6 Variation of heat transfer coefficient with initial temperature, dimensionless time base -- PWR - FLECHT Tests 0105, 1002, and 6155 at a flooding rate of 6 in. / sec.

Fig. C-7 Variation of heat transfer coefficient with initial temperature for a flooding rate of 10 in./sec - PWR-FLECHT Tests 0307, 0408, and 0509, Rod 3D, six-foot elevation.

Fig. C-8 Variation of heat transfer coefficient with initial temperature, dimensionless time base -- PWR - FLECHT Tests 0307, 0408, and 0509 at a flooding rate of 10 in. / sec.

Fig. C-9 Variation of heat transfer coefficient with flooding rate -- PWR - FLECHT Tests 6948, 4225, 1002, and 0408, six-foot elevation on the hottest rods.

Fig. C-10 Variation of heat transfer coefficient with flooding rate, dimensionless time base -- PWR - FLECHT Tests 6948, 4225, 1002, and 0408.

Fig. C-11 Variation of heat transfer coefficient with flooding rate - PWR - FLECHT Tests 2322, 0711, and 0913, Rod 3 D, six - foot elevation.

Fig. C-12 Variation of heat transfer coefficient with flooding rate, dimensionless time base - PWR - FLECHT Tests 2322, 0711, and 0913.

Fig. C-13 Variation of heat transfer coefficient with flooding rate - PWR-FLECHT Tests 7158, 6256, and 6155, Rod 3D, six-foot elevation.

Fig C-14 Variation of heat transfer coefficient with flooding rate, dimensionless time base -- PWR - FLECHT Tests 7158, 6256, and 6155.

Fig. C-15 Variation of surface temperature and heat transfer coefficient with flooding rate - BWR - FLECHT Tests SS4 - 19 and SS4 - 20, Rod 17, six - foot elevation.

Fig. C-16 Variation of heat transfer coefficient with flooding rate, dimensionless time base - BWR - FLECHT Tests SS4-19 and SS4-20.

Fig. C-17 Variation of surface temperature and heat transfer coefficient with pressure -- BWR-FLECHT Tests SS4-10, -11X, -12, -14R, -16X, -17, and -18; Rod 17, six-foot elevation.

Fig. C-18 Variation of heat transfer coefficient with pressure, dimensionless time base - BWR - FLECHT Tests SS4-10, -11X, -12 and -14R.

Fig. C-19 Variation of heat transfer coefficient with pressure, dimensionless time base - BWR-FLECHT Tests SS4-16X, -17, and -18.

Fig. C-20 Variation of heat transfer coefficient with pressure - PWR - FLECHT Tests 0711, 1615, 1002, and 1417; Rod 3D, six - foot elevation.

Fig. C-21 Variation of heat transfer coefficient with pressure, dimensionless time base - PWR - FLECHT Tests 0711, 1615, 1002, and 1417.

Fig. C-22 Variation of heat transfer coefficient with subcooling -- PWR - FLECHT Tests 1720, 1002 and 1314; Rod 3D, six - foot elevation.

Fig. C-23 Variation of heat transfer coefficient with pressure, dimensionless time base -- PWR-FLECHT Tests 1720, 1002, and 1314.

Fig. C-24 Variation of heat transfer coefficient with pressure — PWR — FLECHT Tests 5123 and 4225; Rod 5F, six-foot elevation.

Fig. C-25 Variation of heat transfer coefficient with pressure, dimensionless time base — PWR — FLECHT Tests 5123 and 4225.

Fig. C-26 Variation of surface temperature and heat transfer coefficient with subcooling -- BWR-FLECHT Tests SS4-13X, -14R, and -15Z; Rod 17, six-foot elevation.

Fig. C-27 Variation of heat transfer coefficient with subcooling, dimensionless time base -- BWR-FLECHT Tests SS4-13X, -14R, and -15Z.

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APPENDIX D
PLOTS OF BWR-FLECHT AND PWR-FLECHT QUENCH
ELEVATION VERSUS TIME

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APPENDIX D PLOTS OF BWR-FLECHT AND PWR-FLECHT QUENCH ELEVATION VERSUS TIME

The plots in this appendix present the elevation to which the quench front has progressed with respect to time after test initiation. The quench elevation is given for several rods, including the hottest rod in each bundle. The quench rate of the rod (the slope of the plot for that rod) is indicated on each of the plots in in./sec. The initial slopes (between 0 and about six-foot-elevation) of the BWR-FLECHT hottest rod (Rod 17) are used to define quench rates in the BWR-FLECHT tests. The second slopes (generally between two- and eight-foot elevation) for the PWR-FLECHT hottest rod (5E) are used to define quench rate in the PWR-FLECHT tests.

Fig. D-1 Elevation of quench front with respect to time after flooding initiation - BWR - FLECHT Test SS4 - 10.

Fig. D-2 Elevation of quench front with respect to time after flooding initiation -- BWR - FLECHT Test SS4 - 11X.

Fig. D-3 Elevation of quench front with respect of time after flooding initiation - BWR - FLECHT Test SS4-12.

Fig. D-4 Elevation of quench front with respect to time after flooding initiation - BWR - FLECHT Test SS4 - 13X.

Fig. D-5 Elevation of quench front with respect to time after flooding initiation -- BWR - FLECHT Test SS4 - 14R.

Fig. D-6 Elevation of quench front with respect to time after flooding initiation - BWR - FLECHT Test SS4 - 15Z.

Fig. D-7 Elevation of quench front with respect to time after flooding initiation -- BWR - FLECHT Test SS4 - 16X.

Fig. D-8 Elevation of quench front with respect to time after flooding initiation -- BWR - FLECHT Test SS4 - 17.

Fig. D-9 Elevation of quench front with respect to time after flooding initiation -- BWR - FLECHT Test SS4 - 18.

Fig. D-10 Elevation of quench front with respect to time after flooding initiation -- BWR - FLECHT Test SS4 - 19.

Fig. D-11 Elevation of quench front with respect to time after flooding initiation - BWR - FLECHT Test SS4 - 20.

Fig. D-12 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 0105.

Fig. D-13 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 0307.

Fig. D-14 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 0408.

Fig. D-15 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 0509.

Fig. D-16 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 0711.

Fig. D-17 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 0913.

Fig. D-18 Elevation of quench front with respect to time after flooding initiation - PWR - FLECHT Test 1002.

Fig. D-19 Elevation of quench front with respect to time after flooding initiation - PWR - FLECHT Test 1314.

Fig. D-20 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 1417.

Fig. D-21 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 1615.

Fig. D-22 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 1720.

Fig. D-23 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 2322.

Fig. D-24 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 4027.

Fig. D-25 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 4129.

Fig. D-26 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 4228.

Fig. D-27 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 4526.

Fig. D-28 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 4628.

Fig. D-29 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 5123.

Fig. D-30 Elevation of quench front with respect to time after flooding initiation - PWR - FLECHT Test 5730.

Fig. D-31 Elevation of quench front with respect to time after flooding initiation --PWR - FLECHT Test 6155.

Fig. D-32 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 6256.

Fig. D-33 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 6351.

Fig. D-34 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 6553.

Fig. D-35 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 6948.

Fig. D-36 Elevation of quench front with respect to time after flooding initiation -- PWR - FLECHT Test 7158.