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MULTINODE ANALYSIS OF B&W's 205-FUEL-
ASSEMBLY (MARK-C DESIGN) NUCLEAR
PLANTS DURING LOSS-OF-COOLANT
ACCIDENT

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Multinode Analysis of B&W's 205-Fuel-Assembly
(Mark-C Design) Nuclear Plants During
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C. E. Parks, B. M. Dunn, N. H. Shah

Key Words: Multinode Analysis, Nuclear Plant,
Mark-C Fuel Assembly, LOCA

ABSTRACT

Multinode analyses were conducted to evaluate several loss-of-coolant accidents for B&W's nuclear plants with 205 Mark-C fuel assemblies. The multinode blowdown code CRAFT, the reflooding code REFLOOD, and the cladding heatup code THETA1-B were used to perform these analyses at a core power of 3760 MWt. Curves showing the parameters of interest are presented.

Results of the analyses indicate that the emergency core cooling system will terminate the cladding temperature transient and limit the course of the LOCA in accordance with the AEC Interim Acceptance Criteria.

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1. INTRODUCTION

1.1. General

Babcock & Wilcox's nuclear steam system with a core of 205 Mark-C fuel assemblies utilizes a revised loop arrangement in which the steam generator is raised above the reactor vessel. This change permits a significant reduction in the reactor coolant pump inlet piping and eliminates the need for reactor vessel internals vent valves. The relative elevations were chosen to reduce the potential for steam binding following a break in the reactor vessel inlet coolant pipe. The loop arrangement, the reactor vessel internals, and the emergency core cooling systems are described briefly in the appendix of BAW-10045⁹. The effectiveness of the emergency core cooling systems for the B&W reactor design utilizing 205 Mark-C fuel assemblies has been evaluated at a core power of 3760 MWt and a maximum linear heat rate of 15.36 kW/ft.

The Mark-C fuel elements are positioned in a 17×17 array in each fuel assembly. In contrast, the Mark-B fuel elements are positioned in a 15×15 array in each fuel assembly. The primary advantage of the Mark-C design is a reduction in the linear heat rate and consequently lower fuel temperatures. Additional information on the Mark-C, 17×17 fuel element array is presented in section 4.2 and Chapter 15 of the safety analysis report for those B&W 205-fuel-assembly plants referenced in this report.

This multinode loss-of-coolant (LOCA) analysis was performed for a spectrum of break sizes and is based on analytical techniques, assumptions, and procedures similar to those found in part 4, Appendix A, of the AEC Interim Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactors, dated June 29, 1971, as amended on December 18, 1971. The deviations are as follows:

1. Analyses are performed for the spectrum of breaks ranging from 0.5 ft² to the double-ended severance of the largest pipe of the reactor coolant pressure boundary. End of blowdown for the 0.5-ft² break is defined as the time when the reactor vessel bottom head becomes filled with water up to the bottom of the core. This is the only break for which the bottom head fills before zero leak flow is computed.

2. THETA1-B fuel temperatures are based on data from the steady-state code reported in BAW-10044⁴ with added restrictions in accordance with the AEC requirements listed in BAW-1389.¹⁰

3. Since the 205 Mark-C fuel-assembly plants contain no internal vent valves, steam is vented to the reactor building through the broken and intact loops during the reflood period, and the pump resistance K factor is assumed to be that calculated on the basis of a locked rotor.

4. The steam being vented to the reactor building is assumed to be superheated as it passes through the steam generator (see section 2.2). Venting steam through the broken and unbroken loops is not restricted owing to "cold water plugging" because the B&W reactor system provides emergency coolant directly to the reactor vessel through separate nozzles; therefore, blockage of the pipe by the coolant does not occur.

1.2. Summary and Conclusions

For the spectrum of break sizes considered, the results of the analysis indicate that the AEC Interim Acceptance Criteria are met and that the calculated course of the LOCA is limited as follows:

1. The maximum fuel element cladding temperature does not exceed 2300F.
2. The total amount of fuel element cladding that reacts chemically with water or steam is less than 1%.
3. The cladding temperature transient is terminated while the core geometry is still amenable to cooling.
4. The core temperature is reduced and decay heat is removed for an extended period of time.

As shown in section 3, the double-ended rupture in the cold leg piping produces the highest cladding temperature, 2035F. Since this is well below the temperature specified in the Interim Acceptance Criteria, it is concluded that the performance of the emergency core cooling systems is adequate to cool the core for any postulated break in the reactor coolant system.

2. BLOWDOWN, REFLOOD, AND CORE HEATUP MODEL DESCRIPTIONS

2.1. Blowdown Model (CRAFT¹)

The blowdown analysis is the same as presented in BAW-10045⁹. Although the analysis presented herein is for the Mark-C, 17×17 fuel element array, the core's operating hydrodynamic characteristics and the system are essentially identical to those described in BAW-10045 and in the appendix to BAW-10045 entitled, "Typical 205-Fuel-Assembly Plant Features." The 34-node blowdown model is shown in Figure 2-1.

The core power for the analysis was 3760 MWt, and each of the 1800-ft³-capacity core flooding tanks contained 1450 ft³ of water. It was assumed that 40 ft³ of water was contained in the line between the reactor vessel and the first check valve. One high-pressure injection pump was utilized (after a 25-second delay from actuation); however, since this pump was assumed to be the one associated with the broken loop, half of its flow was lost out the break. One low-pressure injection (LPI) pump was used (only for the 0.5-ft² break), but it was not allowed to come on until the flow rate of the core flooding tank had decreased to the point where the pump flow could be established.

2.2. Reflood Model (REFLOOD)

The REFLOOD code² is documented in BAW-10031, Supplement 1, for use in this analysis. The model used is shown in Figure 2-2. To make the code applicable for the 205 Mark-C fuel-assembly plants, the following modifications were made:

1. Entrainment of water by the steam was added so that a variable percentage of the core inlet flow could be removed with the steam that was generated. The amount of water entrained is based on B&W's correlation⁸ for carryover rate fraction. A part of the entrained water was assumed to be converted to saturated steam based on the heat transferred in the upper region of the core.

2. Steam generation was assumed to begin when the water reaches the bottom of the core. Based on a study of the FLECHT data, entrainment is delayed until the water level in the core reaches 18 inches.

3. Heat transfer coefficients for the REFLOOD code are based on FLECHT test results at a flooding rate of 3 inches per second.

4. Since no core flooding tank water or low-pressure injection water enters the RCS piping, steam is vented to the reactor building by way of the reactor vessel in the unbroken loop and the unbroken pipe in the broken loop, as well as directly to the reactor building through the broken pipe. The original code as documented in BAW-10031, Supplement 2, had this capability, but no credit was taken for steam relief around the loops on the vent valve plants.

The normal path resistances (K-factors) were used except in the steam generators and in the piping and pumps at the steam generator outlet. For these paths, the K-factors were increased by the ratio of the specific volume of superheated steam to the specific volume of saturated steam since saturated steam densities were used in the flow equations. The resistances of the steam venting paths as related to the area of the cold leg piping for the double-ended cold leg break are as follows:

<u>Unbroken loop, $A = 8.55 \text{ ft}^2$</u>	<u>Broken loop, $A = 4.28 \text{ ft}^2$</u>
K (core outlet-RV) = 19.59*	K (core outlet-SG outlet) = 1.04
	K (SG outlet-RB; broken pipe) = 14.05*
	K (SG outlet-RV; unbro- ken pipe) = 15.41*

These adjusted K-factors represent the total resistance from the top of the core back to the reactor vessel or to the reactor building. A reactor building pressure of 45 psia was used to determine K-factors for this example. Before the K-factors were adjusted to account for

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* Resistances include pump locked rotor values. .

superheating, the system pressure drops were checked by "inputting" the steady-state operational flow rates to confirm that the system pressure drop was correct.

2.3. Core Heatup Model (THETA1-B)

The THETA1-B code³ was used as specified in part 4, Appendix A, of the AEC Interim Acceptance Criteria. As in BAW-10034, design core peaking factors of 1.55 radial and 1.70 axial were used. The peak linear heat rate for these conditions at 102% power is 15.36 kW/ft. The linear heat rate distribution used for the hot pin in THETA1-B is shown in Figure 2-3. The Groeneveld correlation was used for stable film boiling in the THETA1-B code. The B&W-2 DNB correlation was used to predict departure from nucleate boiling. Section 4 of BAW-10045 discusses the sensitivity of the cladding temperature to heat transfer and power peaking assumptions and presents comparisons using other heat transfer coefficient correlations.

Figure 2-1. Reactor Coolant System —
34-Node Representation

Figure 2-2. Nodal Representation of Core Reflood Model

Figure 2-3. Linear Heat Rate Distribution for 10-Axial Node, THETA1-B Model, Hot Pin Mark-C Fuel Assembly

3. SPECTRUM OF BREAKS ANALYSIS

Seven cold leg breaks and one hot leg break were analyzed. No credit was taken for control rod insertion except for the 0.5-ft² cold leg break, where credit was taken for the minimum tripped rod worth in addition to the voids. Table 3-1 summarizes pertinent parameters and results for the spectrum of breaks considered.

3.1. Split in Hot Leg Pipe With Area Equivalent to Double-Ended Pipe Break (15.75-ft²)

This break represents the largest break size in the reactor coolant system piping and results in a very rapid blowdown and core shutdown, as well as good core flow and core cooling. Also, the water from the core flooding tanks must pass through the core to flow out the break. Therefore, with no assumed adiabatic heatup periods, the peak cladding temperature is lower than those obtained for the large cold leg breaks.

Figures 3-1 through 3-8 show pertinent parameters for this break. As shown in Figure 3-8, the peak cladding temperature is 1392F.

3.2. Reflood Calculations for Cold Leg Breaks

The manner of using the REFLOOD code in these analyses is described since this code affects the cladding temperature calculations during the period after blowdown.

The REFLOOD model given in Figure 2-2 was used for the guillotine ruptures. For splits, node 9 was treated like node 8; i.e., it connected to the reactor vessel instead of to the containment node, and the break was assumed to occur in node 9. For each break, water entrainment was calculated by using the B&W correlation for carryover rate fraction. For each cold leg break at the pump discharge, excepting the 0.5-ft² break which had low-pressure injection, the reactor

vessel was empty at the start of reflood owing to the assumption that all core flooding tank water is lost.

Containment pressure was assumed to be the initial pressure plus 80% of the rise in pressure calculated by the CONTEMPT code. The pressures used in REFLOOD for each break size examined are as follows:

Break size, ft ²	8.55	5.13	3.0	0.5
RB pressure, psia	45	45	41	41

The core reflood rates for the cold leg breaks are shown in Figures 3-9 through 3-14. The corresponding heat transfer coefficients used in the THETA1-B calculations during reflood are shown in Figures 3-15 through 3-20.

The double-ended cold leg pipe break resulted in the highest cladding temperature. To help in understanding the reflood process Figures 3-21 through 3-26 are shown. Figure 3-21 shows the ECCS flow rate to the reactor vessel as a function of time after blowdown. Figure 3-22 shows the water height in the core and downcomer as a function of time. This head difference provides the driving force for steam flow out of the reactor vessel. Figure 3-23 shows the flow from the pumps to the reactor vessel downcomer in the unbroken loop, the flow from the pump to the reactor vessel downcomer in the unbroken pipe in the ruptured loop, the flow from the pump to the reactor building in the broken pipe, and the steam flow from the reactor vessel downcomer to the reactor building.

Figure 3-24 shows the head difference between the downcomer and the core as a function of time. This curve also represents the pressure drop from the core outlet to the top of the downcomer in the reactor vessel and the pressure drop from the core outlet to the reactor building minus the pressure drop from the downcomer to the reactor building. Figure 3-25 relates the amount of water in the core and the amount lost by steaming and entrainment to the amount of water that enters the core region. Figure 3-26 shows the ratio of steam flow rate leaving the core to water flow rate entering the core region as a function of time.

3.3. Double-Ended Cold Leg Break

Double-ended breaks at the pump suction and pump discharge were analyzed. The break at the pump discharge resulted in the highest cladding temperature (2035F). In making the double-ended break at the pump suction, node 27 of Figure 2-1 was divided into two nodes. Leak paths were assumed to go to the containment from both of the nodes, and no communication between these nodes was permitted after the break.

The cases at the pump discharge assumed a pump resistance equivalent to a free-spinning rotor after cavitation. An analysis of a locked rotor case is presented in BAW-10045; the resultant cladding temperature is significantly lower than that for the free-spinning case. Therefore, the locked rotor case is not presented here. Since the 8.55-ft² break at discharge (with the free-spinning rotor) resulted in the highest cladding temperature, Figures 3-27 through 3-42 are presented to explain the system's behavior. Figure 3-27 shows the pressure at the core outlet versus time, and Figure 3-28 shows the core power versus time. Figures 3-29 through 3-37 are computer plots of flows out of the break, flows into and out of the reactor vessel, core flow, and core pressure drop. Figures 3-38 through 3-41 show parameters of interest from the THETA1-B cladding heatup analysis, and Figure 3-42 shows the average core quality versus time. Figures 3-21 through 3-26, referred to in section 3.2, show reflood parameters of interest. Table 3-1 summarizes break results. As stated above, the double-ended break was also run at the pump suction. Figure 3-43 shows the resultant cladding temperature.

3.4. 8.55-ft² Split at Pump Discharge

The 8.55-ft² split at the pump discharge had a blowdown time equal to that of the double-ended break of the same size; however, the peak cladding temperature was 1913F as compared to 2035F for the double-ended break. For the split break, the 8.55-ft² area out of the small piping node causes a higher flow from the reactor vessel and a more negative flow through the core because of the large leakage assumed to appear in node 29 (compare Figures 3-31 and 3-45). During the period of positive core flow (1.5 to 5 seconds), the flow is less than the positive flow that occurs for the double-ended break. Again, this is due to

the larger flow from the reactor vessel for the split break and to the lower flow from the steam generator side of the break. This difference in flow behavior results in the 100F difference in cladding temperature at the end of blowdown and during reflood. To help explain this phenomenon, Figures 3-44 through 3-51 show the flows of interest. Power and pressure versus time are very similar to those shown for the double-ended break. Figure 3-52 shows the smoothed hot spot mass flux, Figure 3-53 shows the hot spot heat transfer coefficient, Figure 3-54 shows hot spot fluid temperature, and Figure 3-55 shows the hot spot cladding temperature transient.

3.5. 5.13-ft² Guillotine Break at Pump Discharge

This break is equivalent to an 8.55-ft² double-ended break with a discharge coefficient of 0.6 instead of 1.0. Available test results show that a discharge coefficient of 0.6 yields better correlations of test results than does a discharge coefficient of 1.0. Therefore, this break points out a conservatism in the AEC Interim Acceptance Criteria, which calls for discharge coefficients of 1.0 since the peak cladding temperature for this case is 1825F. This is 210F lower than the peak temperature for the case with a discharge coefficient of 1.0. Figure 3-55 shows the average core quality versus time, and Figures 3-56 through 3-59 show the hot spot mass flux, heat transfer coefficient, fluid temperature, and cladding temperature.

3.6. 5.13-ft² Split in Cold Leg Pipe at Pump Discharge

This break is the same size as that covered in section 3.5 except that the break is assumed to occur out of one node rather than two. The peak cladding temperature for this case is 1777F. Figures 3-60 through 3-63 show variables of interest for this break.

3.7. 3-ft² Split in Cold Leg Pipe at Pump Discharge

This break resulted in a peak cladding temperature of 1555F. Variables of interest are shown in Figures 3-64 through 3-67.

3.8. 0.5-ft² Split in Cold Leg Pipe at Pump Discharge

For this break, depressurization to reactor building pressure is very slow, mainly due to the large inflow of emergency cooling water. This water comes into the vessel and rapidly fills the downcomer up to

the nozzle elevation of the cold leg pipe. At 144 seconds, the downcomer mixture height on the break side of the vessel reached the bottom of the nozzles. However, at this time, there is a large quantity of steam entrained in this mixture and only 258 ft³ of water in the downcomer volumes. At 151 seconds, these bubbles have either risen out of the mixture or have condensed, and the liquid volume has increased to 713 ft³. Also, at this time, the flow from the vessel to the break node is a mixture of water and steam. At 154 seconds, the liquid level has risen to a height that completely covers the pipe nozzle, and zero quality fluid is leaving the reactor vessel and flowing to the leak node. This large head of water shortly reverses the flow from the core and the bottom head, which had been flowing to the downcomer, and the bottom head becomes filled with water (no steam) at 178 seconds. This is defined as the end of blowdown. Some cooling by the injection water has taken place at this time due to the flow of a two-phase mixture into the core. This small amount of cooling, however, is more than offset by assuming that the core flooding tank water that has come into the vessel is lost. This is very conservative since, as mentioned above, liquid has been flowing out of the vessel for 24 seconds (from 154 to 178 seconds). Even with the subtraction of this core flood tank water, there are approximately 15,000 pounds of water remaining in the vessel. This is not nearly enough to fill the lower head. Therefore, the REFLOOD calculation was begun with this amount of water in the vessel and with a reactor building pressure of 41 psia. An adiabatic heatup period of 10 seconds results while the bottom head is filling up to the bottom of the core. The reflood curve is shown on Figure 3-14, and the corresponding heat transfer coefficient, which was used in THETA1-B, is shown on Figure 3-20. The smoothed hot spot mass flux, the hot spot heat transfer coefficient, fluid temperature, and the cladding temperature are shown on Figures 3-68 through 3-71. (The peak cladding temperature is only 1199F.)

3.9. Analysis of Cold Leg Breaks — Pin Perforation

The foregoing analyses demonstrate that the core can be adequately cooled when using the ground rules specified in the AEC Interim Acceptance Criteria; Table 3-1 summarizes the results. In addition, Table

3-2 shows the percentage of pins expected to perforate for each break size examined. These values were obtained by using the THETA1-B code in conjunction with information showing the percentage of pins at or above a given core power. This method of analysis is conservative for the following reasons:

1. The THETA1-B and CRAFT runs were made using design peaking factors, i.e., a 1.7 inlet axial peak and a 1.55 radial peak. These peaks are not expected to occur.

2. Pins were assumed to perforate at cladding temperatures between 1200 and 1500F. This range of cladding temperature is based on single-pin burst tests performed by the Oak Ridge National Laboratories and by Babcock & Wilcox. In general, the data show pin perforations at temperatures greater than or equal to 1500F for pressure differentials of 1000 psi and less across the cladding. For pins that experience higher pressure differentials (up to 2000 psi), perforations can occur at temperatures as low as 1200F. Therefore, a range of pin perforations is shown in Table 3-2. Curves showing the pressure-temperature relationship are shown in section 6 of reference 6 and reference 7.

Table 3-1. Summary of Break Results

Break size, ft ² /type	Break location	Start of CF tank in- jection, s	End of blowdown, s	Peak cladding temp, F	Metal-water reaction, %	
					Local	Core
15.75/hot leg (split)	Reactor vessel	7.2	15.8	1392	0.012	0.0002
8.55/cold leg (guillotine)	Pump disch	11.9	19.3	2035	2.17	0.073
8.55/cold leg (guillotine)	Pump suction	13.4	24.8	1798	0.70	0.004
8.55/cold leg (split)	Pump disch	11.9	19.3	1913	1.29	0.050
5.13/cold leg (guillotine)	Pump disch	15.4	23.4	1825	0.82	0.009
5.13/cold leg (split)	Pump disch	15.0	23.2	1777	0.52	0.016
3.0/cold leg (split)	Pump disch	22.4	34.2	1555	0.10	0.001
0.5/cold leg (split)	Pump disch	137.0	178.0	1199	0.001	~0

Table 3-2. Percentage of Pin Perforations Vs Break Size

<u>Break size, ft²/type</u>	<u>Break location</u>	<u>Perforation, %</u>
8.55/DE-CLB ^(a)	Pump discharge	50 - 81
8.55/DE-CLB	Pump suction	36 - 71
8.55/CLB (split)	Pump discharge	59 - 82
5.13/DE-CLB	Pump discharge	33 - 72
5.13/CLB (split)	Pump discharge	30 - 71
3/CLB (split)	Pump discharge	4 - 38
0.5/CLB (split)	Pump discharge	0
15.75/HLB ^(b) (split)	--	0 - 51

(a) Double-ended cold leg break.

(b) Hot leg break.

Figure 3-1. Reactor Vessel Pressure at Core Exit for
15.75-ft² Hot Leg Break

Figure 3-2. Power Vs Time for 15.72-ft²
Split in Hot Leg Pipe

Figure 3-3. ECCS Flow Rate to Reactor Vessel as Function of Time for 15.75-ft² Hot Leg Break

Figure 3-4. Break Flow Rate of 15.75-ft² Hot Leg Break

Figure 3-5. Hot Spot Mass Flux for 15.75-ft²
Hot Leg Break

Figure 3-6. Hot Spot Fluid Temperature for 15.75-ft²
Hot Leg Break

Figure 3-7. Hot Spot Heat Transfer Coefficient for 15.75- ft²
Hot Leg Break

Figure 3-8. Hot Spot Cladding Temperature for 15.75-ft²
Hot Leg Break

Figure 3-9. Core Flooding Rate for 8.55-ft² Double-Ended Cold Leg Break

Figure 3-10. Core Flooding Rate for 8.55-ft² Split in Cold Leg Pipe

Figure 3-11. Core Flooding Rate for 5.13-ft² Guillotine
Type Break in Cold Leg Pipe

Figure 3-12. Core Flooding Rate for 5.13-ft² Split in Cold Leg Pipe

Figure 3-13. Core Flooding Rate for 3-ft² Split in Cold Leg Pipe

Figure 3-14. Core Flooding Rate for 0.5-ft² Split in Cold Leg Pipe

Figure 3-15. Post Blowdown Hot Spot Heat Transfer Coefficient for 8.55-ft² Double-Ended Cold Leg Break

Figure 3-16. Post Blowdown Hot Spot Heat Transfer Coefficient for 8.55-ft² Split in Cold Leg Pipe

Figure 3-17. Post-Blowdown Hot Spot Heat Transfer Coefficient for 5.13-ft² Guillotine Break in Cold Leg Pipe

Figure 3-18. Post-Blowdown Hot Spot Heat Transfer Coefficient for 5.13-ft² Split in Cold Leg Pipe

Figure 3-19. Post Blowdown Hot Spot Heat Transfer Coefficient for
3-ft² Split in Cold Leg Pipe

Figure 3-20. Post Blowdown Hot Spot Heat Transfer Coefficient
for 0.5-ft² Split in Cold Leg Pipe

Figure 3-21. ECCS Flow Rate to Reactor Vessel as Function of Time After Blowdown for 8.55-ft² Double-Ended Cold Leg Break

Figure 3-22. Water Height in Core and Downcomer During Reflood for 8.55-ft² Double-Ended Cold Leg Break

Figure 3-23. Steam Flows Through the System During Reflood for 8.55-ft² Double-Ended Cold Leg Break

Figure 3-24. Head Difference Between Downcomer and Core for
8.55-ft² Double-Ended Cold Leg Break

Figure 3-25. History of Emergency Core Coolant Entering Core
for 8.55-ft² Double- Ended Cold Leg Break

Figure 3-28. Power Vs Time for 8.55-ft² Double-Ended Cold
Leg Break at Pump Discharge

Figure 3-29. Leak Flow From Reactor Vessel Side of Break for
8.55-ft² Double-Ended Cold Leg Break at
Pump Discharge

Figure 3-30. Leak Flow From Pump Discharge Side of Break for
8.55-ft² Double-Ended Cold Leg Break at
Pump Discharge

Figure 3-31. Flow at Reactor Vessel Inlet, Broken Loop, Broken Cold Leg, Path 44 - 8.55-ft² Double-Ended Cold Break at Pump Discharge

Figure 3-32. Flow at Reactor Vessel Inlet, Broken Loop, Unbroken Cold Leg, Path 26 - 8.55-ft² Double-Ended Cold Leg Break at Pump Discharge

Figure 3-33. Flow at Reactor Vessel Inlet, Unbroken Loop,
Path 25 — 8.55-ft² Double-Ended Cold Leg
Break at Pump Discharge

Figure 3-34. Flow at Reactor Vessel Outlet, Broken Loop,
Path 24 - 8.55-ft² Double-Ended Cold Leg
Break at Pump Discharge

Figure 3-35. Flow at Reactor Vessel Outlet, Unbroken Loop,
Path 23 - 8.55-ft² Double-Ended Cold Leg
Break at Pump Discharge

Figure 3-36. Core Flow — Path 2 for 8.55-ft² Double-Ended
Cold Leg Break at Pump Discharge

Figure 3-37. Core Pressure Drop for 8.55-ft² Double-Ended
Cold Leg Break at Pump Discharge

Figure 3-38. Smoothed Hot Spot Mass Flux for 8.55-ft² Double-Ended Cold Leg Break at Pump Discharge

Figure 3-39. Hot Spot Heat Transfer Coefficient for 8.55- ft² Double-Ended Cold Leg Break at Pump Discharge

Figure 3-40. Hot Spot Fluid Temperature for 8 55-ft² Double-Ended
Cold Leg Break at Pump Discharge

Figure 3-41. Hot Spot Cladding Temperature for 8.55-ft² Double-Ended Cold Leg Break at Pump Discharge

Figure 3-42. Average Core Quality Vs Time for 8.55-ft² Double-Ended Cold Leg Break at Pump Discharge

Figure 3-43. Hot Spot Cladding Temperature for 8.55-ft² Double-Ended Cold Leg Break at Pump Suction

Figure 3-44. Leak Flow for 8.55-ft² Split in Cold Leg Pipe

Figure 3-45. Flow at Reactor Vessel Inlet, Broken Loop,
Broken Cold Leg, Path 44 - 8.55-ft²
Split in Cold Leg Pipe

Figure 3-46. Flow at Pump Discharge Side of Break, Broken Loop, Broken Cold Leg, Path 43 - 8.55-ft² Split in Cold Leg Pipe

Figure 3-47. Flow at Reactor Vessel Inlet, Broken Loop, Unbroken Cold Leg, Path 26 - 8.55-ft² Split in Cold Leg Pipe

Figure 3-48. Flow at Reactor Vessel Inlet, Unbroken Loop, Path
25 - 8.55-ft² Split in Cold Leg Pipe

Figure 3-49. Flow at Reactor Vessel Outlet, Broken Loop, Path
24 - 8.55-ft² Split in Cold Leg Pipe

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Figure 3-50. Flow at Reactor Vessel Outlet, Unbroken Loop, Path
23 - 8.55-ft² Split in Cold Leg Pipe

Figure 3-51. Core Flow — Path 2 for 8.55-ft² Split in Cold Leg Pipe

Figure 3-52. Smoothed Hot Spot Mass Flux for 8.55-ft²
Split in Cold Leg Pipe

Figure 3.53. Hot Spot Heat Transfer
Coefficient for 8.55-ft²
Split in Cold Leg Pipe

Figure 3-54. Hot Spot Fluid Temperature for 8.55-ft² Split in Cold Leg Pipe

Figure 3-55. Hot Spot Cladding Temperature for 8.55-ft²
Split in Cold Leg Pipe

Figure 3-56. Smoothed Hot Spot Mass Flux for 5.13-ft² Guillotine
Break at Pump Discharge

Figure 3-57. Hot Spot Heat Transfer Coefficient for 5.13-ft² Guillotine Break at Pump Discharge

Figure 3-58. Hot Spot Fluid Temperature for 5.13-ft² Guillotine
Break at Pump Discharge

Figure 3-59. Hot Spot Cladding Temperature for 5.13-ft² Guillotine Break at Pump Discharge

Figure 3-60. Smoothed Hot Spot Mass Flux for 5.13-ft² Split in Cold Leg Pipe

Figure 3-61. Hot Spot Heat Transfer Coefficient for 5.13-ft² Split in Cold Leg Pipe

Figure 3-62. Hot Spot Fluid Temperature for 5.13-ft² Split in Cold Leg Pipe

Figure 3-63. Hot Spot Cladding Temperature for 5.13-ft² Split in Cold Leg Pipe

Figure 3-64. Smoothed Hot Spot Mass Flux for 3-ft² Split in Cold Leg Pipe

Figure 3-65. Hot Spot Heat Transfer Coefficient for 3-ft²
Split in Cold Leg Pipe

Figure 3-66. Hot Spot Fluid Temperature for 3-ft² Split in Cold Leg Pipe

Figure 3-67. Hot Spot Cladding Temperature for 3-ft² Split
in Cold Leg Pipe

Figure 3-68. Smooth Hot Spot Mass Flux for 0.5-ft² Split in Cold Leg Pipe

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Figure 3-69. Hot Spot Heat Transfer Coefficient for 0.5-ft² Split in Cold Leg Pipe

Figure 3-70. Hot Spot Fluid Temperature for 0.5-ft² Split in Cold Leg Pipe

Figure 3-71. Hot Spot Cladding Temperature for 0.5-ft² Split
in Cold Leg Pipe

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