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MULTINODE ANALYSIS OF B&W's
177-FUEL-ASSEMBLY, NON-VENT-VALVE
PLANT DURING LOCA

by

C. E. Parks
L. R. Cartin
K. C. Shieh

BABCOCK & WILCOX
Power Generation Group
Nuclear Power Generation Division
P.O. Box 1260
Lynchburg, Virginia 24505

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Multinode Analysis of B&W's 177-Fuel-Assembly,
Non-Vent-Valve Plant During LOCA

Key Words: LOCA, Multinode, Analysis

ABSTRACT

This analysis documents the multinode evaluation of several loss-of-coolant accidents for B&W's 2772-MWt non-vent-valve, nozzle supported nuclear power plants with prepressurized fuel pins. The basic computer code is discussed, including modifications that have been incorporated to make it more applicable to the system analyzed. Also, basic assumptions and areas of uncertainty and conservatism are discussed. The multinode model was used to perform a core cooling analysis for a spectrum of break sizes in the cold and hot legs of the reactor coolant system. Curves showing hot spot fuel cladding temperature versus time after an assumed break are included for a spectrum of hot and cold leg break sizes.

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1. INTRODUCTION

1.1. General

This report describes an evaluation of the Babcock & Wilcox nuclear steam system with a core of 177 fuel assemblies and a loop arrangement in which the steam generator is raised above the reactor vessel to eliminate the need for reactor vessel internal vent valves. The relative elevations were chosen to eliminate steam binding following a break in the reactor vessel inlet coolant pipe. The effectiveness of the emergency core cooling systems in the event of a reactor coolant system piping failure was evaluated at a core power of 2772 MWt and a maximum linear heat rate of 18.3 kW/ft.

This multinode loss-of-coolant accident (LOCA) analysis was performed for a spectrum of break sizes and was 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, June 29, 1971, as amended on December 18, 1971. The deviations are as follows:

1. Analyses are performed for the spectrum of breaks ranging in size 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-10044. The main difference between this code and that used for calculating initial fuel temperatures as reported in BAW-10034 is an improved model for determining the temperature jump distance.

3. Since the reactor vessel contains no internal vent valves, steam is vented to the reactor building through the broken and intact loops during the reflood period, and it is assumed that the pump resistance K factor is that calculated on the basis of a locked rotor.

4. It is assumed that the steam being vented to the reactor building is 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 show 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 1%.
3. The cladding temperature transient is terminated while the core 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 8.55-ft² split in the cold leg piping at the pump discharge produced the highest cladding temperature (1912F). Since this temperature is well below the maximum 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 (CRAFT)¹

The thermal-hydraulic response of the reactor coolant system during the blowdown phase of the accident was determined analytically by using the CRAFT computer code. A multinode representation of the system was chosen by using the techniques explained in BAW-10034 to ensure that the influence of components with dissimilar thermal and hydraulic characteristics may be accommodated. The resulting 34-node simulation for the system is shown in Figure 2-1, and a legend explaining the noding scheme is given in Table 2-1.

Before a postulated pipe failure the nuclear plant was assumed to be operating in a steady-state condition at 102% of rated power. The system contained 497,000 lbm of coolant, including the pressurizer water mass and possessed an initial total energy of 292×10^6 Btu. The average temperature in the system was 582F with a reactor inlet temperature of 555.3F and a reactor vessel outlet temperature of 608.7F. Flow from four reactor coolant pumps, each delivering 88,000 gpm, was used to circulate the coolant within the system.

Each of the 1410-ft³ core flooding tanks contained 1040 ft³ of water; in addition, it was assumed that 40 ft³ of water was contained in the line between the core flood tank and the first check valve from the reactor vessel.

Except for the 0.5-ft² break, emergency injection during blowdown was supplied only by core flooding tanks. The 0.5-ft² break utilized the high-pressure injection pump associated with the broken loop (50% of the injection water was assumed to enter the unbroken cold leg, and 50% to enter at the break). High-pressure injection was delayed 25 seconds after the receipt of the actuation signal.

2.2. Reflood Model (REFLOOD)²

The REFLOOD code as documented in BAW-10031, Supplement 1, was modified for use in this analysis. The model itself is shown in Figure 2-2. In the analysis it was assumed that 10% of the core region (core plus bypass) flow was entrained by the steam and converted to steam in the upper part of the core. 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 relative to the area of the cold leg piping for the double-ended cold leg break are as follows:

<u>Unbroken loop, A = 8.55 ft²</u>	<u>Broken loop, A = 4.27= ft²</u>
K (Core outlet to RV) = 27.0	K (core outlet - SG outlet) = 1.88
	K (SG outlet - RB; broken pipe) = 21.3
	K (SG outlet - RV; unbroken pipe) = 19.5

These adjusted K factors represent the total resistance from the top of the core back to the reactor vessel or the reactor building. A reactor building pressure of 38 psia was used to determine the K factors for this example. These K factors include the K factors for the reactor coolant pumps if the rotors are assumed to be locked. The building pressures used in the spectrum analysis are as follows:

Break size, ft ²	8.55	5.13	3.0	0.5
Building pressure, psia	38	38	38	34

The building pressures are less than those obtained by taking 80% of the rise in building pressure; this adjustment results in a more conservative reflood analysis. Venting steam is more difficult at lower pressures, and even though containment pressure is held constant in the reflood analysis, the pressure is well below the calculated containment pressure for the time period of interest.

2.3. Core Heatup Model (THETA1-B)³

The THETA1-B code was used to analyze the hot-pin, thermal response to a LOCA as specified in part 4, Appendix A, of the AEC Interim Acceptance Criteria. In the B&W evaluation model, 10 axial regions with six radial fuel nodes and two radial cladding nodes were used to solve the pin heat conduction equation, and the zirconium-water reaction was described by a parabolic rate equation by using the Baker-Just equation with a coefficient of 1.0.

The hot spot temperature transient during the blowdown period was determined from the hot spot mass flux, pressure, power, and core coolant entrance enthalpy as a function of time from CRAFT. A 0.8 flow multiplier was applied to the CRAFT core flow results to obtain the appropriate hot spot mass flux values. A 1.7 inlet axial power shape and a 1.714 radial peak were used. These peaking factors represented maximum design values, and their use yielded a peak linear heat rate of 18.3 kW/ft at 102% power. The linear heat rate distribution used for the hot pin is shown in Figure 2-3. The Groeneveld correlation was also used in the blowdown calculations for stable film boiling, and the B&W-2 DNB correlation was used to predict departure from nucleate boiling.

After blowdown, the hot pin was assumed to undergo adiabatic heatup until the water level in the vessel reached the bottom of the active core. By using the flooding rates calculated by REFLOOD, a set of integrated average flooding rates was obtained. Heat transfer coefficients were then calculated by using correlations derived from PWR-FLECHT data. The hot pin thermal response was then calculated in THETA1-B by using these coefficients for the remainder of the transient.

Table 2-1. Legend for 34-Node Representation
of Reactor Coolant System

<u>Node number</u>	<u>Identification</u>
1	Reactor vessel bottom head
2	Bottom third of core
3	Center third of core
4	Upper third of core
5	Reactor vessel upper plenum
6	Reactor vessel upper head
7	Half volume between plenum assembly and core support shield
8	Half volume between plenum assembly and core support shield
9	Upper part of core bypass volume
10	Half of downcomer volume
11	Half of downcomer volume
12	Lower part of core bypass volume
13	One steam generator upper head
14	Upper one-half of tube volume in one steam generator
15	Lower one-half of tube volume in one steam generator
16	One steam generator lower head
17	Suction piping of two RC pumps and two pump suction volumes
18	Two pump discharge volumes and cold leg piping from pump discharge to re- actor vessel inlet nozzle
19	Hot leg piping from reactor vessel out- let nozzle up to the 90-degree elbow
20	Pressurizer
21	One steam generator upper head
22	Upper one-half of tube volume in one steam generator
23	Lower one-half of tube volume in one steam generator
24	One steam generator lower head
25	Suction piping to one RCS pump and pump suction volume

Table 2-1. (Cont'd)

<u>Node number</u>	<u>Identification</u>
26	Discharge volume of one RCS pump and cold leg piping from pump discharge to reactor vessel inlet nozzle
27	Suction piping to one RCS pump and pump suction volume
28	Discharge volume of one RCS pump and cold leg piping up to and including 30-degree elbow
29	Cold leg piping from 30-degree elbow to reactor vessel inlet nozzle
30	Hot leg piping from reactor vessel outlet nozzle up to 90-degree elbow
31	Hot leg piping from 90-degree elbow to 180-degree elbow
32	Hot leg piping from 90-degree to 180-degree elbow
33	Volume in 180-degree bend in hot leg piping
34	Volume in 180-degree bend in hot leg piping
35	Secondary side of one steam generator
36	Secondary side of one steam generator
37	Reactor building

Figure 2-1. Reactor Coolant System — 34-Node Representation
for CRAFT Blowdown Analysis

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

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. Reflood calculations were based on the methods described in section 3.2 of BAW-10045.

3.1. Splits and Breaks

3.1.1. Split in Hot Leg Pipe With Area Equivalent to Double-Ended Pipe Break (14.1 ft²)

The 14.1-ft² hot leg break represents the largest break possible in the reactor coolant system piping and results in a very rapid blowdown and core shutdown. Since the reactor coolant flow is directed up through the core when flowing toward the break, good core flow and core cooling is maintained during blowdown. Also, with the hot leg break there are no physical phenomena by which the emergency coolant can bypass the core, and a rapid refill of the lower head and quenching of the core occur. Therefore, with no assumed adiabatic heatup period, the peak cladding temperature is lower than the temperatures obtained for the large cold leg breaks.

Figures 3-1 through 3-10 show pertinent parameters for this break. The peak cladding temperature, shown in Figure 3-10, is 1477F.

3.1.2. 8.55-ft² Guillotine Cold Leg Break at Pump Discharge

The 8.55-ft² guillotine cold leg break at pump discharge is characterized by a complete guillotine severance of a cold leg pipe combined with enough pipe movement to ensure that the leak flow from the two ends of the pipe do not interact. For this analysis, a break at

the pump discharge with an assumed pump resistance equivalent to a free-spinning rotor after cavitation was evaluated. Figures 3-11 through 3-21 show the system behavior. Figure 3-11 shows the core pressure history, and Figure 3-12 shows the core power history. Figures 3-13 through 3-21 show the flow out the break, and the flow out of and into the reactor vessel, core flow, and core pressure. In Figure 3-20 note that the core flow remains positive during the time interval between 2 and 7 seconds. In this region, flow that is provided by the pumps in the three unbroken cold legs is greater than the flow from the reactor vessel through the inlet nozzle to the break. This provides a driving force, thus causing water to flow through the core in a positive direction until the pumps cavitate, at which time the flow becomes negative. Therefore, good core cooling occurs initially during the transient, and the maximum cladding temperature occurs during reflooding.

At the end of blowdown, all water that has been injected by the core flooding tanks is discarded (in accordance with the Interim Acceptance Criteria) to ensure that the lower plenum and part of the downcomer are again filled before the core can be cooled. This action results in an adiabatic heatup period of approximately 8.9 seconds during which the cladding temperature increases at a rate of approximately 38F per second. Once core flooding begins, the cladding temperature transient is turned around with a peak temperature of 1900F occurring at 30.5 seconds. Figures 3-22 through 3-28 show reflood parameters. Figures 3-29 through 3-32 show parameters of interest from the THETA1-B pin thermal analysis, and Figure 3-33 shows the average core quality history.

3.1.3. 8.55-ft² Split at Pump Discharge

The 8.55-ft² split in the cold leg piping at pump discharge, equivalent to the double-ended pipe break, results in the highest peak cladding temperature. For the base case in which a pump resistance is assumed to be equivalent to a free spinning rotor after cavitation, the maximum cladding temperature is 1912F, which is 12F higher than that for the equivalent guillotine break. Figures 3-34 through 3-41 show CRAFT flows of interest, and Figures 3-42 through 3-47 describe the results of the thermal and reflood analysis. From Table 3-1 and the

various figures for the 8.55-ft² split and guillotine cold leg breaks, one may conclude that the system responses are similar for both, but cladding temperature differences result from events occurring at slightly different times.

The 8.55-ft² split at the pump discharge was also analyzed by assuming a locked rotor pump resistance. The resultant peak cladding temperature is approximately 1712F. After the pumps cavitate, the driving force for the positive core flow is diminished, and the flow becomes negative again—but at a very low rate. Increasing the pump resistance from a free spinning to a locked rotor causes the core flow to become more negative since less fluid is entering the vessel from the unbroken cold leg pipes. This increase in the magnitude of the core flow decreases the maximum hot spot cladding temperature approximately 200F.

3.1.4. 8.55-ft² Split at Pump Suction

The 8.55-ft² split break at pump suction equivalent to a double-ended pipe break was analyzed. The resultant peak cladding temperature is 1816F, which occurred during reflood; this temperature is 96F lower than that for the worst case at the pump discharge. Figures 3-48 through 3-53 shows some parameters of interest for this particular break.

The smoothed hot spot mass flux shown in Figure 3-48 was considerably higher during the initial part of the transient than that analyzed at the pump discharge; this prompted better core cooling for the blowdown period of the transient. Furthermore, at the end of blowdown, 15,000 pounds of water remained in the vessel after subtracting the core flooding water. This produced a shorter adiabatic heatup period compared to that associated with the same break at the pump discharge, in which case a dry downcomer and a lower head at the end of blowdown are assumed. This 2.8-second decrease in time during the period when zero heat transfer is assumed also acts to make this break less severe at the pump suction location.

3.1.5. 5.13-ft² Guillotine Break at Pump Discharge

The 5.13-ft² guillotine break at the pump discharge is equivalent to an 8.55-ft² guillotine break with a discharge coefficient of

0.6 instead of 1.0. The peak cladding temperature for this break is 1831F. Figures 3-54 through 3-59 present the results of this break. Figure 3-57 shows the cladding temperature transient. Figures 3-54 through 3-56 show the hot spot mass flux, heat transfer coefficient, and fluid temperature, respectively. Figure 3-58 shows the reflood rate, and Figure 3-59 shows the post-blowdown hot spot heat transfer coefficient versus time.

3.1.6. 5.13-ft² Split in Cold Leg Pipe at Pump Discharge

The 5.13-ft² split in the cold leg pipe at pump discharge was analyzed by using a pump resistance equivalent to a free spinning rotor after cavitation and had a peak cladding temperature of 1818F. Figures 3-60 through 3-76 show the system behavior. Figure 3-60 shows the core pressure history, and Figure 3-61 shows the core power history. Figures 3-62 through 3-69 display the flow out of the break, flow out of and into the reactor vessel, core flow, and core pressure drop. Figures 3-70 through 3-75 show parameters of interest from the THETA1-B pin thermal analysis and the reflood analysis, and Figure 3-76 shows the average core quality history. The results for the split and guillotine breaks are similar for the same break area. Thus, the figures presented in this section represent the guillotine break and can serve as a general guide to the systems response during a 5.13-ft² LOCA.

3.1.7. 3-ft² Split in Cold Leg Pipe at Pump Discharge

The peak cladding temperature for the 3-ft² split in the cold leg pipe at the pump discharge is 1640F. Variables of interest for this particular break are shown in Figures 3-77 through 3-82.

The slow depressurization rate of this smaller cold leg break allows more time for core flooding tank water to be injected during blowdown. Thus, for the reflood calculations more dependence is placed on the performance of the high- and low-pressure injection systems to supply emergency coolant water to cool the core. The flooding rate, shown in Figure 3-81, indicates good core cooling for the time interval from 10 to 20 seconds after the end of blowdown; this time interval is long enough to turn the cladding temperature transient around. The flooding rate then drops to a value slightly above 1 in./s, and the

resulting FLECHT heat transfer coefficients are no longer able to continue the cooling of the fuel pin at the same rate. This action results in a second peak in cladding temperature as shown in Figure 3-80. The drop in heat transfer coefficient shown in Figure 3-82 is more drastic for this break than for the larger cold leg breaks. The drop is a function not only of the earlier use of the low flooding rate but also of the FLECHT heat transfer coefficients' dependence on initial cladding temperature. In other words, the FLECHT heat transfer coefficients for the same flooding rate decrease as the initial cladding temperature decreases.

3.1.8. 0.5-ft² Split in Cold Leg Pipe at Pump Discharge

The 0.5-ft² split in the cold leg pipe at the pump discharge results in a peak cladding temperature of 1533F by assuming that the end of blowdown occurs at 147.5 seconds when the reactor vessel lower plenum is filled with water. The smoothed hot spot mass flux, the hot spot heat transfer coefficient, the fluid temperature, and the cladding temperature are shown in Figures 3-83 through 3-86, respectively. The core flooding rate and post-blowdown heat transfer coefficient are shown in Figures 3-87 and 3-88.

3.2. 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 by using design peaking factors, i.e., a 1.7 inlet axial peak and a 1.714 radial peak. These peaks, especially the 1.7 inlet peak, are not expected to occur even during core maneuvering.

2. The higher fuel temperature, which occurs at the end of core life (EOL) assuming no cladding creepdown on the fuel, was assumed

even though design peaking factors are more characteristic of beginning of core life (BOL) conditions.

3. 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 references 6 and 7.

Table 3-1. Summary of Break Results

Break size, ft ² /type	Break location	Start of CF tank injection, s	End of blowdown, s	End of CF tank injection, s	Peak cladding temp, F	Metal-water reaction, %	
						Local	Core
14.1/hot leg (split)	Reactor vessel nozzle	6.85	15.9	36.3	1477	0.0132	0.00102
8.55/cold leg (guillotine)	Pump disch	11.43	18.6	39.6	1900	0.6672	0.02346
8.55/cold leg (split)	Pump disch	11.36	18.6	39.8	1912	0.7395	0.02597
8.55/cold leg (split)	Pump suction	9.88	19.4	38.4	1816	0.3555	0.00791
5.13/cold leg (guillotine)	Pump disch	13.87	22.2	42.4	1831	0.5086	0.01065
5.13/cold leg (split)	Pump disch	13.82	21.8	42.4	1818	0.4516	0.00936
3.0/cold leg (split)	Pump disch	19.89	30.6	48.8	1640	0.2588	0.01088
0.5/cold leg	Pump disch	116.35	147.5	153.5	1533	0.0938	0.00134

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

Break size, ft ² /type	Break location	Perforations, %
8.55/double-ended, cold leg	Pump disch	42 - 61
8.55/cold leg (split)	Pump disch	42 - 59
8.55/cold leg (split)	Pump suction	38 - 57
5.13/double-ended, cold leg	Pump disch	38 - 56
5.13/cold leg (split)	Pump disch	37 - 55
3/cold leg (split)	Pump disch	25 - 51
0.5/cold leg (split)	Pump disch	6 - 29
11.17/hot leg (split)	Reactor vessel nozzle	0 - 50

Figure 3-1. Core Pressure for 14.1-ft² Hot Leg Break

Figure 3-2. Neutron Power for 14.1-ft² Hot Leg Break

Figure 3-3. ECCS Flow Rate to Reactor Vessel for 14.1-ft² Hot Leg Break

Figure 3-4. Leak Flow Rate for 14.1-ft² Hot Leg Break

Figure 3-5. Core Flooding Rate for 14.1-ft² Hot Leg Break

Figure 3-6. Post-Blowdown Heat Transfer Coefficient for 14.1-ft² Hot Leg Break

Figure 3-7. Smoothed Hot Spot Mass Flux for 14.1-ft² Hot Leg Break

Figure 3-8. Hot Spot Fluid Temperature for 14.1-ft² Hot Leg Break

Figure 3-9. Hot Spot Heat Transfer Coefficient for 14.1-ft² Hot Leg Break

Figure 3-10. Hot Spot Cladding Temperature for 14.1-ft² Hot Leg Break

Figure 3-11. Core Pressure for 8.55-ft² Guillotine Cold Leg Break
at Pump Discharge

Figure 3-12. Neutron Power for 8.55-ft² Guillotine Cold Leg Break
at Pump Discharge

Figure 3-13. Leak Flow From Reactor Vessel Side of Break for 8.55-ft²
Guillotine Cold Leg Break at Pump Discharge

Figure 3-14. Leak Flow From Pump Discharge Side of Break for 8.55-ft²
Guillotine Cold Leg Break at Pump Discharge

Figure 3-15. Flow at Reactor Vessel Inlet, Broken Loop, Broken Cold Leg,
Path 44 — 8.55-ft² Guillotine Cold Leg Break at Pump Discharge

Figure 3-16. Flow at Reactor Vessel Inlet. Broken Loop, Unbroken Cold Leg,
Path 26 — 8.55-ft² Guillotine Cold Leg Break at Pump Discharge

Figure 3-17. Flow at Reactor Vessel Inlet, Unbroken Loop, Path 25 — 8.55-ft²
Guillotine Cold Leg Break at Pump Discharge

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Figure 3-18. Flow at Reactor Vessel Outlet, Broken Loop,
Path 24 — 8.55-ft² Guillotine Cold Leg Break
at Pump Discharge

Figure 3-19. Flow at Reactor Vessel Outlet, Unbroken Loop, Path 23 — 8.55-ft²
Guillotine Cold Leg Break at Pump Discharge

Figure 3-20. Core Flow — Path 2 for 8.55-ft² Guillotine Cold Leg Break
at Pump Discharge

Figure 3-21. Core Pressure Drop for 8.55-ft² Guillotine Cold Leg Break at Pump Discharge

Figure 3-22. Core Flooding Rate for 8.55-ft² Guillotine Cold Leg Break at Pump Discharge

Figure 3-23. Post-Blowdown Hot Spot Heat Transfer Coefficient for 8.55-ft² Guillotine Cold Leg Break at Pump Discharge

Figure 3-24. ECCS Flow Rate to Reactor Vessel During Reflood for 8.55-ft² Guillotine Cold Leg Break at Pump Discharge

Figure 3-25. Water Height in Core and Downcomer During Reflood for 8.55-ft² Guillotine Cold Leg Break at Pump Discharge

Figure 3-26. Steam Flows Through System During Reflood for 8.55-ft² Guillotine Cold Leg Break at Pump Discharge

Figure 3-27. Head Difference Between Downcomer and Core During Reflood for 8.55-ft² Guillotine Cold Leg Break at Pump Discharge

Figure 3-28. History of Emergency Core Coolant Entering Core for 8.55-ft² Guillotine Cold Leg Break at Pump Discharge

Figure 3-29. Smoothed Hot Spot Mass Flux for 8.55-ft² Guillotine Cold Leg Break at Pump Discharge

Figure 3-30. Hot Spot Heat Transfer Coefficient for 8.55-ft²
Guillotine Cold Leg Break at Pump Discharge

Figure 3-31. Hot Spot Fluid Temperature for 8.55-ft² Guillotine Cold Leg Break at Pump Discharge

Figure 3-32. Hot Spot Cladding Temperature for 8.55-ft²
Guillotine Cold Leg Break at Pump Discharge

Figure 3-33. Average Core Quality for 8.55-ft² Guillotine Cold Leg Break at Pump Discharge

Figure 3-34. Leak Flow for 8.55-ft² Split in Cold Leg Pipe
at Pump Discharge

Figure 3-35. Flow at Reactor Vessel Inlet, Broken Loop, Broken Cold Leg,
Path 44 — 8.55-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-36. Flow at Pump Discharge Side of Break, Broken Loop, Broken Cold Leg, Path 43 - 8.55-ft² Split in Cold Leg Pipe at Pump Discharge

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Figure 3-37. Flow at Reactor Vessel Inlet, Broken Loop, Unbroken Cold Leg,
Path 26 — 8.55-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-38. Flow at Reactor Vessel Inlet, Unbroken Loop, Path 25 — 8.55-ft²
Split in Cold Leg Pipe at Pump Discharge

Figure 3-39. Flow at Reactor Vessel Outlet, Broken Loop, Path 24 — 8.55-ft²
Split in Cold Leg Pipe at Pump Discharge

Figure 3-40. Flow at Reactor Vessel Outlet, Unbroken Loop, Path 23 — 8.55-ft²
Split in Cold Leg Pipe at Pump Discharge

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Figure 3-41. Core Flow — Path 2 for 8.55-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-42. Core Flooding Rate for 8.55-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-43. Post-Blowdown Hot Spot Heat Transfer Coefficient for 8.55-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-50. Hot Spot Fluid Temperature for 8.55-ft² Split
in Cold Leg Pipe at Pump Suction

Figure 3-51. Hot Spot Cladding Temperature for 8.55-ft²
Split in Cold Leg Pipe at Pump Suction

Figure 3-52. Core Flooding Rate for 8.55-ft² Split in the Cold Leg Pipe at Pump Suction

Figure 3-53. Post-Blowdown Hot Spot Heat Transfer Coefficient for
8.55-ft² Split in Cold Leg Pipe at Pump Suction

Figure 3-54. Smoothed Hot Spot Mass Flux for 5.13-ft² Guillotine Cold Leg Break at Pump Discharge

Figure 3-55. Hot Spot Heat Transfer Coefficient for 5.13-ft² Guillotine Cold Leg Break at Pump Discharge

Figure 3-56. Hot Spot Fluid Temperature for 5.13-ft² Guillotine Cold Leg Break at Pump Discharge

Figure 3-57. Hot Spot Cladding Temperature for 5.13-ft²
Guillotine Cold Leg Break at Pump Discharge

Figure 3-58. Core Flooding Rate for 5.13-ft² Guillotine Cold Leg Break at Pump Discharge

Figure 3-59. Post-Blowdown Hot Spot Heat Transfer Coefficient for 5.13-ft² Guillotine Cold Leg Break at Pump Discharge

Figure 3-60. Core Pressure for 5.13-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-61. Neutron Power for 5.13-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-62. Leak Flow for 5.13-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-63. Flow at Reactor Vessel Inlet, Broken Loop. Broken Cold Leg,
Path 44 — 5.13-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-64. Flow at Reactor Vessel Inlet, Broken Loop, Unbroken Cold Leg,
Path 26 — 5.13-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-65. Flow at Reactor Vessel Inlet, Unbroken Loop, Path 25 — 5.13-ft²
Split in Cold Leg Pipe at Pump Discharge

Figure 3-66. Flow at Reactor Vessel Outlet, Broken Loop, Path 24 — 5.13-ft²
Split in Cold Leg Pipe at Pump Discharge

Figure 3-67. Flow at Reactor Vessel Outlet, Unbroken Loop, Path 23 — 5.13-ft²
Split in Cold Leg Pipe at Pump Discharge

Figure 3-68. Core Flow — Path 2 for 5.13-ft² Split in Cold Leg Pipe
at Pump Discharge

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Figure 3-69. Core Pressure Drop for 5.13-ft² Split in Cold Leg Pipe
at Pump Discharge

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Figure 3-70. Smoothed Hot Spot Mass Flux for 5.13-ft² Split in Cold Leg
Pipe at Pump Discharge

Figure 3-71. Hot Spot Heat Transfer Coefficient for 5.13-ft²
Split in Cold Leg Pipe at Pump Discharge

Figure 3-72. Hot Spot Fluid Temperature for 5.13-ft²
Split in Cold Leg Pipe at Pump Discharge

Figure 3-73. Hot Spot Cladding Temperature for 5.13-ft²
Split in Cold Leg Pipe at Pump Discharge

Figure 3-74. Core Flooding Rate for 5.13-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-75. Post-Blowdown Hot Spot Heat Transfer Coefficient for 5.13-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-76. Average Core Quality for 5.13-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-77. Smoothed Hot Spot Mass Flux for 3-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-78. Hot Spot Heat Transfer Coefficient for 3-ft²
Split in Cold Leg Pipe at Pump Discharge

Figure 3-79. Hot Spot Fluid Temperature for 3-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-82. Post-Blowdown Hot Spot Heat Transfer Coefficient for 3-ft²
Split in Cold Leg Pipe at Pump Discharge

Figure 3-83. Smoothed Hot Spot Mass Flux for 0.5-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-84. Hot Spot Heat Transfer Coefficient for 0.5-ft²
Split in Cold Leg Pipe at Pump Discharge

Figure 3-85. Hot Spot Fluid Temperature for 0.5-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-86. Hot Spot Cladding Temperature for 0.5-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-87. Core Flooding Rate for 0.5-ft² Split in Cold Leg Pipe at Pump Discharge

Figure 3-88. Post-Blowdown Hot Spot Heat Transfer Coefficient for 0.5-ft² Split in Cold Leg Pipe at Pump Discharge

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