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MULTINODE ANALYSIS OF CORE
FLOODING LINE BREAK FOR B&W'S
2568-MWt INTERNALS VENT VALVE PLANTS

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Multinode Analysis of Core Flooding Line
Break for B&W's 2568-MWt Internals Vent
Valve Plants - Revision 1

1

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ABSTRACT

Babcock & Wilcox developed the FOAM Computer Code to predict accurately the consequences of a rupture of a core flooding line. An analysis was carried out using the CRAFT Code to evaluate the hydrodynamics and transient water inventories of the reactor coolant system. The FOAM Code was used to compute a swell level history for the core, and the THETA 1-B Code was used to perform transient fuel pin heatup calculations. The FOAM data are in good agreement with data from tests conducted by General Electric, Hitachi, and Westinghouse and confirm the applicability of the code to reactor analysis.

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1. INTRODUCTION

This report evaluates the effectiveness of the emergency core cooling systems for Babcock & Wilcox's reactor designs with internals vent valves and power levels up to 2568-MWt during the postulated rupture of a core flooding line.

To predict accurately the consequences of this rupture, the swell level within the reactor core must be evaluated realistically. To permit such an evaluation, B&W developed the FOAM computer code. An analytical development of the code is presented along with comparisons of calculational and experimental results which justify the use of the code in B&W's evaluation model.

For this analysis, the CRAFT¹ computer code is used to evaluate the hydrodynamics and transient water inventories of the reactor coolant system. The FOAM code is used to compute a core swell level history, and the THETA 1-B code² is used to perform transient fuel pin heatup calculations. Results for three different power shapes are reported to cover a wide range of operating conditions.

2. SUMMARY AND CONCLUSIONS

Development of the FOAM code utilized an appropriate set of physical relationships. The code analyzes the swell level produced by steady-state pool boiling phenomena during a slow loss-of-coolant transient. Results of comparisons of predicted data and various sets of applicable experimental data indicate that FOAM adequately represents the physical phenomena within a reactor core during periods of relatively quiet pool boiling. Therefore, it is concluded that FOAM is appropriate for use in accident analysis calculations for which such boiling phenomena are expected. | 1

The core flooding line break analysis used three different axial power distributions. The maximum peak cladding temperature was 1199F, which is well below the 2200F limit specified in the Final Acceptance Criteria. Because of the low magnitude of the cladding temperature transients, the effects of metal-water reaction and cladding swelling are negligible. Therefore, the core will not experience structural damage and will always be amenable to cooling. The applicability of this analysis to the Final Acceptance Criteria is demonstrated in Appendix A of BAW-10104, Revision 1.⁹ | 1

The duration of the analysis was chosen so that the vessel is slowly refilling with water and long-term cooling is ensured. Under the ground rules of the analysis, only one HP injection line is used for long-term cooling, and this is shown to be adequate. However, operator action can be taken to ensure more rapid refilling of the reactor vessel. The peak cladding temperatures for the three power shapes considered are summarized below.

Case	Axial location of peak power, ft (a)	Peak linear heat rate, kW/ft	Axial location of max clad- ding temp, ft	Cladding temperature, F
1	5.5	17.8	9.75	731
2	6.8	14.8	9.75	964
3	9.5	12.2	10.75	1199

(a) Measured from bottom of active fuel region.

3. METHOD OF ANALYSIS

3.1. Codes and Usage

The method of analysis used to determine the cladding temperature response is an improved version of the small-leak evaluation model presented in topical report BAW-10052, "Multinode Analysis of Small Breaks for B&W's 2568-MWt Nuclear Plants."³ The use of this model is consistent with the interim policy statement because the flow from the reactor vessel is limited by an insert in the core flooding tank (CFT) nozzle with a cross-sectional area of 0.44 ft².

The core flooding line break is characterized by a rapid blowdown during which a large amount of water in the reactor coolant system is expelled from the primary system. Following the rapid blowdown, a slow blowdown ensues and a quiescent situation exists in the core with core water being boiled off. Eventually, a stable situation exists and makeup equals boiloff.

The CRAFT¹ code is used to analyze the hydrodynamics during the rapid blowdown phase and to calculate the liquid inventory of the core during the quiescent phase. The noding scheme used in the CRAFT analysis is shown in Figure 4-1. There are 12 nodes for the primary system, one for the secondary system, and one for the containment. The break is located in the downcomer, approximately in the center of node 14. The following assumptions are made:

1. The plant is operating at a steady-state power level of 102% of 2568 MWt.
2. The leak is instantaneous (discharge coefficient of 1.0).
3. The reactor trips at a primary system pressure of 2050 psig.
4. Offsite power is lost at the time of the accident.
5. Safety rods begin entering the core one-half second after the reactor trip signal is generated.

6. The reactor coolant pumps trip and coast down at the time of reactor trip.
7. One complete train of the emergency safeguards system - that containing the unbroken CFT line - fails to operate. This failure leaves one core flooding tank and one high-pressure injection system to provide coolant to the reactor vessel.
8. A phase-separation model is used in CRAFT during the entire transient.
9. Other than that calculated by CRAFT, no water is arbitrarily assumed to be lost from the break during CFT injection.

The size of the break, along with the proximity to the vent valves, limits fluid velocities in the loops, lower head, and downcomer to values that preclude entrainment of droplets. Furthermore, since the intact tank is injecting coolant around the vessel 180 degrees from the break, the flow of steam - whether from the vent valves or from the loops - does not cross the injecting nozzle on its way to the break. Steam flow entering the vessel from the loops mixes with water in the downcomer, causing high mixture heights and a flow of mixture out of the break. In this way, CFT water is lost from the primary system.

The FOAM code is used to calculate swell levels during the quiescent period of analysis. The development and the experimental verification of the code are presented in sections 3.2 and 3.3. Input for the FOAM code consists of the power shape, power level, pressure, inlet subcooling, and quiescent water level. Each of the three power shapes considered is divided into 36 axial nodes for FOAM usage. The pressure and the quiescent water level are taken directly from the CRAFT analysis. Because of primary metal heating, the water entering the bottom of the core will be saturated. Therefore, zero inlet subcooling is used. This is substantiated by the CRAFT analysis. Since higher power levels create higher swell levels, the power level of the average channel, rather than the level of the hot channel, is used to determine the swell level. This use is conservative because the swell level will be somewhat higher in the hot bundle.

The THETA 1-B code is run twice to complete the analysis. During the rapid blowdown portion of the analysis, the entire pin length is simulated by using CRAFT input as generated according to the groundrules of the Interim Acceptance Criteria's evaluation model for large breaks. To analyze the slow blowdown period, a second THETA model is run to model only the regions of the core that will be uncovered during the transient. This model is run through the rapid blowdown period using input from the first THETA run, so that the same conditions are obtained at the end of this time period. During the quiescent period, the core will be fully covered for a time: it will then partially uncover and eventually recover. Because of the low qualities found in the core during this time, heat transfer below the swell level will be by pool nucleate boiling. Sufficient flow is input to the THETA model to guarantee credible heat transfer coefficients for the period while the core is covered. While the core is partially uncovered, the THETA input is saturated steam, and the flow rate is assumed to be that of the average channel. This assumption is conservative because the higher power in the hot bundle is expected to generate more steam for the same swell level. The Dittus-Boelter² correlation is used to develop heat transfer coefficients. This use is based on reference 4, which shows that radiation compensates for the decrease in convective heat transfer due to laminar flow, and that the predictions of the Dittus-Boelter equation are in good agreement with the data. After recovering flow is again increased to give credible heat transfer.

3.2. FOAM Code Development and Justification

3.2.1. Development of the FOAM Code

The FOAM code, used to calculate the swell level for a given quiet water level, is based on the following equations:

$$\alpha(x) = \frac{W(x)V_g}{V_B(x)A} \quad (1)$$

$$W(x) = \int_{LS}^x \frac{\phi(x)}{hfg} dx \quad (2)$$

$$W_{out} = W_{in} \quad (3)$$

$$W_{in} = \int_0^{LS} \frac{\phi(x)}{h_{sat} - h_{in}} dx \quad (4)$$

$$W_{out} = \int_{LS}^{SL} \frac{\phi(x)}{hfg} dx \quad (5)$$

$$QL = \int_0^{SL} [1 - \alpha(x)] dx \quad (6)$$

$$\alpha(x) = \text{Wilson } [V_B(x)] \quad (7)$$

$$= 0.136 \left(\frac{\rho_v}{\rho_L - \rho_v} \right)^{0.32} \left[\frac{g_c \sigma}{g(\rho_L - \rho_v)} \right]^{0.19} \left[\frac{V_L(x)}{g \frac{g_c \sigma}{g(\rho_L - \rho_v)}} \right]^{1.78}$$

$$\text{or}$$

$$= 0.75 \left(\frac{\rho_v}{\rho_L - \rho_v} \right)^{0.32} \left[\frac{g_c \sigma}{g(\rho_L - \rho_v)} \right]^{0.19} \left[\frac{V_B(x)}{g \frac{g_c \sigma}{g(\rho_L - \rho_v)}} \right]^{0.78}$$

whichever is less.

Definitions for the symbols used in this report are given in Table 3.1.

Equation (1) can be developed by performing a steady state balance on the steam volume contained within a differential element of core fluid volume. It may be intuitively developed by recognizing that the numerator is the volume of steam passing the point x per unit time. The denominator is the volume of steam which could pass x per unit time if the void fraction at x were 1.0, and is thus the total volume available for fluid. Therefore, equation (1) results from the direct application of the definition of void fraction.

Equation (2) states that the steam flow at any position x above the point at which the coolant switches from subcooled water to saturated water is equal to the power integrated to x divided by the latent heat of vaporization.

Equation (3) states that for a steady state situation the inlet water flow must be equal to the exit steam flow.

Equation (4) is an integral equation relating the inlet subcooled water flow to the point at which subcooled water is completely converted to saturated water. The equation states that the integration of the power from the bottom of the core to the level of complete saturation divided by the degree of subcooling must equal the inlet flow.

Equation (5) gives the outlet steam flow by integrating the power from the saturation point to the top of the swell and dividing by the latent heat of vaporization.

Equation (6) relates the water available to fill the swell to the void fraction as a function of position. It says the volume not occupied by steam between the bottom of the core and the top of the swell must be occupied by water.

Equation (7) gives the Wilson⁵ correlation between void fraction and bubble rise velocity.

These seven equations are simultaneously solved by FOAM to obtain the swell level. The integrals are replaced by summations over discrete core segments. Thus the integral of

$$\int \sigma(x) dx$$

becomes

$$\sum_i \left(\frac{Pf_i + Pf_{i-1}}{2} \right) \bar{\phi} (z_i - z_{i-1})$$

where Pf_i is the peaking factor associated with axial location z_i ; similarly for Pf_{i-1} and z_{i-1} . The term $\bar{\phi}$ denotes the average core linear heating rate. Similarly the integral

$$\int [1 - \alpha(x)] dx$$

becomes

$$\sum_i \left[1.0 - \frac{(\alpha_i + \alpha_{i-1})}{2} \right] (z_i - z_{i-1})$$

The following steps are taken in solving the discrete forms of the equations. The equations are labeled by the original number plus a "D" to indicate the discrete form.

(a) An inlet flow is guessed, and equation (4) is used to find the point at which the fluid becomes saturated. The form of the equation is:

$$\sum_{i=1}^{i(LS)} \left(\frac{Pf_i + Pf_{i-1}}{2} \right) (z_i - z_{i-1}) \bar{\phi} = W_{in} (h_{sat} - h_{in}) \quad 4D)$$

The summation is performed stepwise until, for a given axial position i , the left hand side of the equation is larger than the right hand side. This indicates that LS lies between z_i and z_{i-1} , and a linear interpolation is used to determine LS.

(b) A temporary assumption that enough water exists to cover the core with swell is made and the steam flow at the axial levels is determined by equation (2). The form of the equation is:

$$W_i = \sum_{i(LS)}^i \frac{(Pf_i + Pf_{i-1})}{2} \frac{(z_i - z_{i-1})}{hfg} \bar{\phi} \quad (2D)$$

The evaluation between LS and the next highest node is done by linear interpolation.

(c) Equations 1 and 7 are combined to give:

$$\alpha(x) = \left[0.136 \left(\frac{\rho_v}{\rho_L - \rho_v} \right)^{0.32} \left[\frac{g_c \sigma}{\sqrt{g(\rho_L - \rho_v)}} \frac{1}{d} \right]^{0.19} \left[\frac{W(x)V_g}{A \sqrt{g \frac{g_c \sigma}{g(\rho_L - \rho_v)}}} \right]^{1.78} \right]^{\frac{1}{2.78}} \quad (8)$$

or

$$= \left[0.75 \left(\frac{\rho_v}{\rho_L - \rho_v} \right)^{0.32} \left[\frac{g_c \sigma}{\sqrt{g(\rho_L - \rho_v)}} \frac{1}{d} \right]^{0.19} \left[\frac{W(x)V_g}{A \sqrt{g \frac{g_c \sigma}{g(\rho_L - \rho_v)}}} \right]^{0.78} \right]^{\frac{1}{1.78}}$$

Equation (8) is then used to find the average void fraction between the i th and $(i + 1)$ th axial level by:

$$\bar{\alpha}(i \rightarrow i+1) = \left[0.136 \left(\frac{\rho_v}{\rho_L - \rho_v} \right)^{0.32} \left[\frac{g_c \sigma}{\sqrt{g(\rho_L - \rho_v)}} \frac{1}{d} \right]^{0.19} \left[\frac{(W_i + W_{i+1})V_g}{2A \sqrt{g \frac{g_c \sigma}{g(\rho_L - \rho_v)}}} \right]^{1.78} \right]^{\frac{1}{2.78}} \quad (8D)$$

or

$$= \left[0.75 \left(\frac{\rho_v}{\rho_L - \rho_v} \right)^{0.32} \left[\frac{g_c \sigma}{g(\rho_L - \rho_v)} \right]^{0.19} \left[\frac{(W_i + W_{i+1}) V_g}{2A \sqrt{\frac{g_c \sigma}{g(\rho_L - \rho_v)}}} \right]^{0.78} \right]^{\frac{1}{1.78}}$$

whichever is smaller.

(d) Equation (6) is then used to determine the swell level. The form of the equation is

$$QL = \sum_{i=0}^{i(SL)} [1.0 - \alpha(i+1)] (z_{i+1} - z_i) \quad (6D)$$

The sum is evaluated stepwise until the right side exceeds the left side. At this point, the swell level lies between z_{i+1} and z_i , and linear interpolation is used to fix SL.

(e) Next, the steaming rate is calculated by equation (5). The form of this equation is

$$W_{out} = \sum_{i(LS)}^{i(SL)} \frac{(Pf_i + Pf_{i-1})}{2} (z_i - z_{i-1}) \phi \quad (5D)$$

and the end effects are handled by linear interpolation. Now equation (3) can be used to determine whether the initial estimated value for inlet flow is appropriate. If the estimated W_{in} equals W_{out} to within 0.5%, then the solution is accepted. If not, a new inlet flow is determined by

$$W_{in} = \frac{W_{in \text{ estimated}} + W_{out}}{2}$$

and the process is repeated until an acceptable solution is obtained. Should the iteration fail to converge in 10 tries, the last try is accepted and a failure to converge message is printed.

3.2.2. FOAM Justification

To verify the validity of FOAM as an analytical tool and to justify its use in B&W's small leak evaluation, FOAM predictions were compared with results from past and current experimental tests. The results of tests performed by the General Electric Company, Hitachi Corporation, and the Westinghouse Electric Company confirm the applicability of the code to reactor analysis.

3.2.2.1. General Electric Tests⁶

The General Electric Company conducted a series of tests in 1967 to determine the minimum water level required to keep the core covered with mixture. The test equipment consisted of a core and a downcomer connected by a lower plenum. The core was allowed to boil off water, and a makeup supply was provided to the downcomer to maintain a constant water level. Because the flows involved were very low, the water head in the downcomer was a direct measure of the core liquid volume. Thermocouples were installed in the heater rods to determine the swell level. When a thermocouple was below the swell level, it would register a temperature very close to the system saturation temperature. When the thermocouple was above the swell level, it would be cooled only by steam flow and would thus show an elevated temperature.

In Figure 3-2 the General Electric data and calculations are compared with FOAM calculations for runs at atmospheric pressure and at 100 psia. As can be seen, the FOAM calculations are in good agreement with GE's. The assumptions used for the FOAM calculations are as follows:

1. The inlet temperature is 135F.
2. Complete bundle cooling is defined by a swell level which covers the highest thermocouple (10.8 feet).
3. Atmospheric tests are conducted at 17 psia. (This pressure was obtained from thermocouple readings below the swell level.)

3.2.2.2. Hitachi Tests⁷

A paper presenting the results of these tests, which were performed in Japan, was given at the 1973 ANS meeting at Salt Lake City. The tests were run at atmospheric pressure under conditions similar to those for the GE tests, but more experimental points were obtained and there was less deviation in the data. These data are compared with FOAM data in Figure 3-3; the agreement is excellent.

3.2.2.3. Westinghouse Tests⁸

In March of 1973, the Westinghouse Electric Company conducted a series of tests for the Duke Power Company. These tests were similar to those previously described, but were slow-transient rather than steady-state tests. The system was filled with water, and the heater power was turned on. As the water boiled off, the swell level slowly dropped. The equivalent quiescent water level in the heater bundle was determined by a sight glass connected to the inner vessel. The swell level was determined by thermocouples at various axial locations.

These data are compared with FOAM data in Figures 3-4, 3-5, and 3-6. Figure 3-4 shows the required water level to obtain a 10-foot swell. Figure 3-5 gives the water level required for an 8-foot swell. Figure 3-6 gives the water level required for a 6-foot swell. The tests were run at 100 and 400 psia. At all levels, agreement is very good for the 400-psia tests. The apparent deviation for the 100-psia data is probably due to an inadequate determination of inlet sub-cooling. Even with this deviation, the code produces answers within the error band of the experiment.

3.2.2.4. Conclusion for FOAM Experimental Verification

The FOAM code is based on logically developed equations. The agreement with the results of three different experiments is very good. The results of the Westinghouse tests at 100 psia, which show a minor disagreement with FOAM, could have been caused by inaccuracies in the experimental data. This conclusion is supported by GE's 100-psia test results, which are in good agreement with FOAM results. It is therefore concluded that the FOAM code is an acceptable analytical

tool for use in evaluating level swell during the long-term quasi steady-state transient.

Table 3-1. Definition of Symbols

<u>Symbol</u>	<u>Definition</u>	<u>Unit</u>
$\alpha(x)$	Void fraction	unitless
$W(x)$	Steam flow rate	lbm/s
V_g	Steam specific volume	ft ³ /lbm
A	Flow area	ft ²
x	Axial position	ft
LS	Level at which inlet water becomes saturated	ft
$\phi(x)$	Linear steady-state heat rate	Btu/s-ft
h_{fg}	Latent heat of vaporization	Btu/lbm
W_{out}	Steam flow out of core	lbm/s
W_{in}	Water flow into core	lbm/s
h_{sat}	Enthalpy of saturated water	Btu/lbm
h_{in}	Enthalpy of inlet water	Btu/lbm
SL	Level of swell	ft
QL	Equivalent height of water in swell	ft
$V_B(x)$	Bubble rise velocity	ft/s
ρ_v	Density of saturated steam	lbm/ft ³
ρ_L	Density of saturated water	lbm/ft ³
g_c	Conversion constant	$\frac{32.2 \text{ ft lbm}}{\text{lbf s}^2}$
g	Gravitational constant	32.2 ft/s ²
σ	Surface tension	lbf/ft
d	Hydraulic diameter	ft
Pf	Peaking factor	unitless
z	Axial location	ft
$\bar{\phi}$	Average steady-state linear heat rate	Btu/s-ft
$i(y)$	Index of axial position just above "y"	unitless
$\bar{\alpha}(y)$	Average void fraction over range "y"	unitless

Figure 3-1. CRAFT Model Noding Diagram

Figure 3-2. Comparison of FOAM Calculations to General Electric Calculations and Experimental Data

Figure 3-3. Comparison of FOAM Calculations to the Hitachi Atmospheric Data

Figure 3-4. Comparison of FOAM Calculations to Westinghouse Data for the 10 Foot Elevation

Figure 3-5. Comparison of FOAM Calculations to Westinghouse Data for the 8 Foot Elevation

4. ANALYSIS OF CORE FLOODING LINE BREAK

The break resulting from the complete rupture of a core flooding line is 0.44 ft^2 in area and is located directly outside the downcomer above the cold leg nozzles. The results of this accident are sensitive to the initial axial power shape. To cover the range of plant operation, three different axial shapes were analyzed.

4.1 Results of Analysis

Since the CRAFT model uses only one core path, the code is insensitive to axial power shape, and one CRAFT blowdown applies to all three axial shapes considered. The core flow predicted by CRAFT and used in the THETA model for the first 300 seconds is shown in Figure 4-1. System pressure and core power are shown in Figures 4-2 and 4-3. Figure 4-4 shows the core liquid volume versus time as used in the FOAM code.

During the entire analysis, the CRAFT mixture height is above the top of the core. It is high enough for the mixture to flow out the vent valves and to have an opportunity to be removed from the primary system through the break. Since this effect limits the amount of water in the core region, the code conservatively evaluates the quiescent water level in the core.

The FOAM code is sensitive to axial power shape and was run for each of the three shapes considered. These shapes are shown in Figures 4-5, 4-6, and 4-7. To simplify the analysis, the minimum swell level transient calculated for these three shapes was used for a thermal evaluation of all three shapes. The minimum swell level occurs for the case 2 power shape as shown in Figure 4-8. The dotted line gives the conservative envelope used in the THETA calculations.

As shown in Figure 4-8, it is assumed that the core is uncovered to the 9.5-foot level at 500 seconds and remains uncovered until 650 seconds. Therefore, the second THETA model simulates the top 2.5 feet of core using five axial regions. The peaking factors are taken from the case power shapes at appropriate levels. The cladding temperature histories for the

hot spot for each case are shown in Figures 4-9, 4-10, and 4-11. The case 3 power shape results in the highest calculated cladding temperature, 1199 F. The heat transfer coefficient and fluid temperature for the case 3 analysis are shown in Figures 4-12 and 4-13.

4.2 Alternate Analysis Low-Mixture Level in CRAFT

As mentioned in section 4.1, the conservative phase-separation assumptions in CRAFT limit the water volume remaining in the core. With the mixture level up to the vent valves, injection of additional fluid into the core merely causes more fluid to flow out the vent valve, and there is no appreciable gain in the core liquid volume. To evaluate the effect of this conservatism, a change was made to the CRAFT phase-separation model so that the mixture height as developed by CRAFT was consistent with that evaluated by FOAM. This allowed the CRAFT mixture level to fall below the vent valves. The CRAFT liquid level and the swell level as interpreted by FOAM are shown in Figure 4-14. Note that in this case the mixture does not uncover the core at any time. Since nucleate boiling will exist in the mixture, the cladding temperature will decrease to a value very close to the saturation temperature associated with 70 psia and will stay there. Thus, no temperature calculation was thought to be necessary. This demonstrates the conservatism of the original CRAFT phase-separation model.

Figure 4-1. Core Flow for CFT Line Break

Figure 4-2. System Pressure for CFT Line Break

Figure 4-3. Core Power for CFT Line Break

4-5

Figure 4-4. Inner Vessel Liquid Volume

Figure 4-5. Case 1 Axial Power Shape

Figure 4-6. Case 2 Axial Power Shape

Figure 4-7. Case 3 Axial Power Shape

Figure 4-8. Minimum Core Swell Level for CFT Line Break

Figure 4-9. Case 1 Cladding Temperature Transient

PEAK CLADDING TEMPERATURE FOR CFT LINE BREAK

CASE 1 POWER SHAPE (9 75 FOOT ELEVATION)

Figure 4-10. Case 2 Cladding Temperature Transient

Figure 4-11. Case 3 Cladding Temperature Transient

Figure 4-12. Case 3 Heat Transfer Coefficient

Figure 4-13. Case 3 Fluid Temperature

Figure 4-14. Inner Vessel Fluid Volumes
Using Realistic Phase Separation Model

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