STATUS REPORT

BY THE

DIRECTORATE OF LICENSING

IN THE MATTER OF

BABCOCK AND WILCOX

ECCS EVALUATION MODEL

CONFORMANCE

TO 10 CFR 50, APPENDIX K

8004150758

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1.0 INTRODUCTION

1.1 General

On June 29, 1971, the Atomic Energy Commission published an Interim Statement of Policy establishing acceptance criteria for emergency core cooling systems for light-water-cooled nuclear power reactors. These criteria, which were adopted following a review by the Commission's Regulatory staff and the Advisory Committee on Reactor Safeguards, provided the basis of reasonable assurance that such systems would be effective in the highly unlikely event of a loss-of-coolant accident (LOCA). On November 30, 1971, the Commission announced its decision to hold a legislative-type rulemaking hearing for the purpose of determining whether or not the Interim Policy Statement should be retained as issued, or whether these criteria should be adopted in another form. The hearings lasted a total of 125 days and generated a record of more than 22,000 pages of transcript and thousands of pages of written direct testimony and exhibits.

The Regulatory staff filed its Concluding Statement after considering the entire evidentiary record of the proceeding as well as arguments contained in the Concluding Statements filed by other participants. After oral arguments, the Commission published the new rule on December 28, 1973.

The principal changes from the Interim Policy Statement (IPS) are as follows. The IPS criterion specifying that the Zircaloy clad temperature shall not exceed 2300°F is replaced by two criteria;

where new experimental i...ormation is available or where better calculational methods have been developed.

The wording of the definition of a loss-of-coolant accident has been modified to conform to its long-accepted usage, limiting it to breaks in pipes. The new rule also requires a more complete documentation of the evaluation models.

Prior to the formulation of the current ECCS criteria, the application of the Interim Acceptance Criteria concurrent with technological advances in such areas as fuel densification, flow redistribution within the core during blowdown, and bypass of accumulator water has led to a greater degree of analytical sophistication. This sophistication has modified ECCS models to the extent that it is not surprising that some of the arbitrary conservatisms embodied in the IPS models have been replaced by well-justified quantities capable, in some cases, of effecting an overall decrease in calculated peak clad temperatures when compared to IPS calculations. These calculated lower peak clad temperatures do not imply a laxity of criteria...on the contrary, the current ECCS criteria have become more limiting in such areas as peak clad temperature (2300°F versus 2200°F now) and adding a limit on the maximum allowed local oxidation. The availability of increasing quantities of meaningful data and the growing technical expertise in many complex facets of ECCS evaluation models are allowing these analytical tools to be improved to a precision greater than previously attained. Therefore, 3/18/74 - Meeting - B&W Submits Preliminary Documentation; Volumes 1, 2, 3, 4

3/28/74 - ACRS Subcommittee (Staff)

4/19/74 - Meeting - B&W Submits Sensitivity Studies - AEC Comments on Volumes 1, 2, 3, 4

5/2/74 - Meeting

5/15/74 - AEC Submits Questions on Volumes 1, 2, 3, 4

5/17/74 - ACRS Subcommittee (B&W)

6/1/74 - B&W Submits Topical Report Draft on Final Documentation

6/25/74 - Meeting - RELAP Results

7/2/74 - Meeting - RELAP Results

8/5/74 - B&W Submits Final Documentation of Evaluation Model (References 6, 7, 8, 9, 10)

8/26/74 - Meeting - AEC Questions on Final Documentation

9/28/74 - ACRS Subcommittee (Staff)

10/7/74 - Meeting - AEC Questions on Final Documentation

10/15/74 - Status Report to ACRS (Staff)

1.2 Scope of Review

The ECCS evaluation model developed by Babcock and Wilcox and reviewed by the Regulatory staff is limited to the analytical techniques utilized to predict the course of a postulated loss-of-coolant through a broken pipe. Using the detailed requirements specified in the new rule as a guideline, the staff reviewed each relevant aspect of the Babcock and Wilcox mathematical model and assumptions. Section 4.0 states these requirements, describes how they are met by Babcock of the staff independent calculations and a comparison to analogous Babcock and Wilcox calculations. Section 4.0 addresses each requirement in Appendix K, discusses Babcock and Wilcox conformance and indicates the acceptability of the analytical methods employed in the model. Section 5.0 provides a separate discussion of the staff's evaluation of the Babcock and Wilcox small break model and highlights the important differences from the large break model. Section 6.0 provides an overall status and acceptability of current documentation available on the new Babcock and Wilcox ECCS evaluation model. Section 7.0 provides a list of references utilized by the staff during the review and Appendices A, B, C and D provide greater detail of the staff's review of the containment backpressure calculation, the Babcock and Wilcox code TAFY, the staff's independent calculations, and the momentum equation.

Efforts to assess the impact of specific open items in the Babcock and Wilcox model (unresolved or unacceptable) are currently underway. Advice from the Advisory Committee on Reactor Safeguards will be sought prior to reaching final decisions. This assessment will be published in a separate report.

2.0 THE BABCOCK AND WILCOX ECCS EVALUATION MODEL

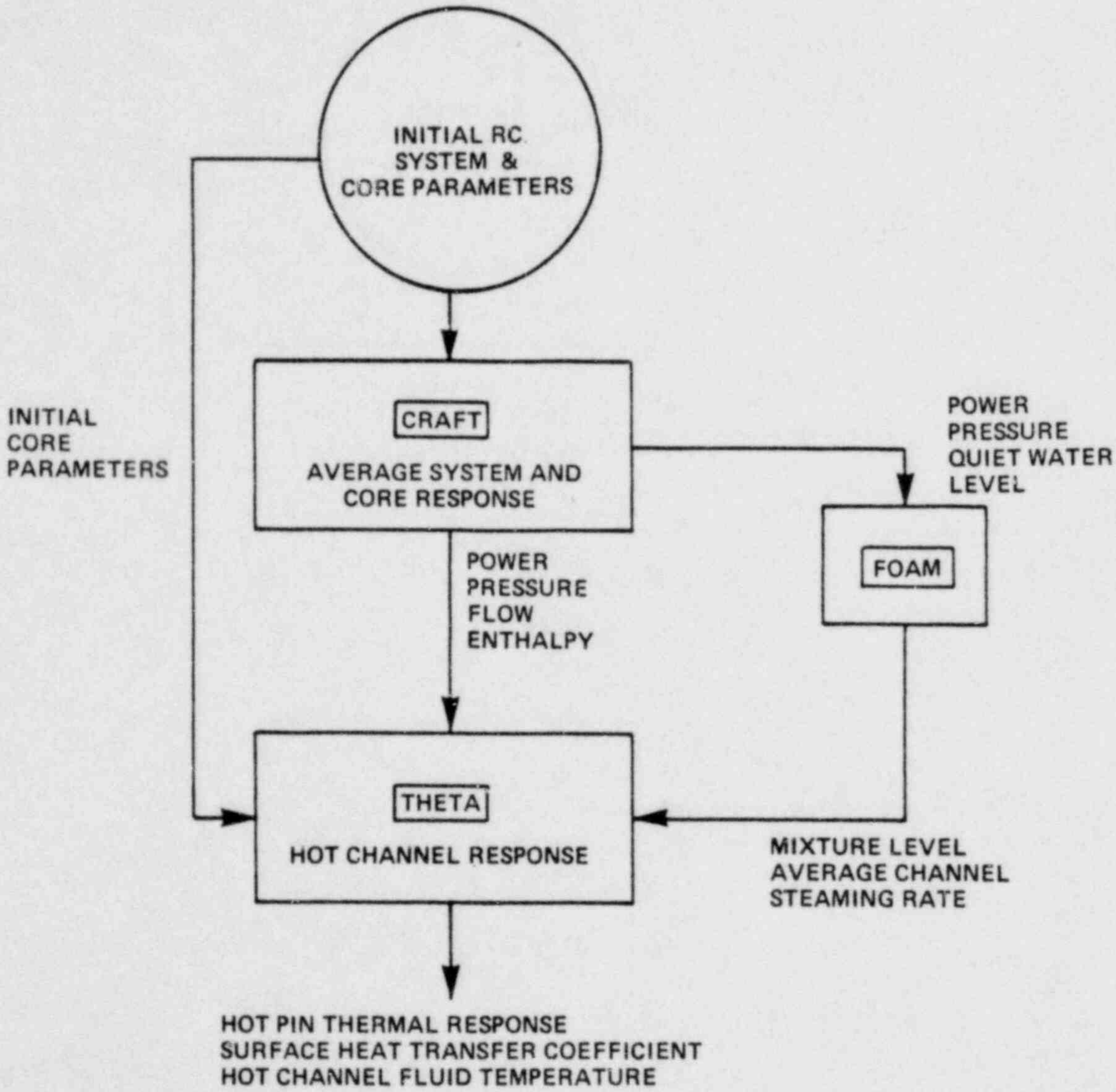
The Babcock and Wilcox evaluation model for large and small breaks is presented in topical reports BAW-10091,(6) BAW-10092,(7) BAW-10093,(8) BAW-10094,(9) and BAW-10095.(10) The large break model is analyzed using the computer code CRAFT 2 to predict the reactor coolant system blowdown transient, the code REFLOOD to determine the length of refill and to calculate flooding rates, and the code THETA 1-B to conduct the heatup calculations over the entire accident. The CONTEMPT code is used to initialize REFLOOD with a conservative backpressure calculation and the code FOAM is utilized in place of CONTEMPT and REFLOCD for the small break analyses. Figures 1 and 2 show the code interfaces in block diagrams. Major changes from codes used previously under the Interim Statement of Policy include the areas of swelling and rupture of the cladding, metal-water reaction rate, bypass model, flow redistribution within the core during blowdown, break noding, momentum flux, post-CHF heat transfer (Dougall-Rohsenow), pump modeling, worst single failure, and reflooding rates. Each of these changes is addressed in Section 4.0 and is discussed in greater detail in the above listed topical reports.

2.1 Overview of the Evaluation Model

2.1.1 ırge Breaks

The transient phenomena associated with a large break LOCA is divided into the three major categories of blowdown, refill, and reflood. As indicated previously, the CRAFT computer code is used to

FIGURE 2. SMALL BREAK ANALYSIS CODE INTERFACES



on a node-center to node-center basis. That is, calculations are conducted from the center of the upstream node to the center of the downstream node. The core barrel vent valves, used on all Babcock and Wilcox plants, are modeled as simple pipes between the reactor vessel outlet plenum and the downcomer. A forward flow k-factor combined with a large reverse flow resistance is used. The reactor core model simulates three distinct radial power regions within the core:

1. The hot bundle
2. The eight adjacent bundles
3. The remaining core

The eight bundles adjacent to the hot bundle are assumed to be at 90 percent of the power of the hot bundle. The steam generators are modeled using four nodes to describe the primary side of the generator and one control volume to represent the secondary side. The performance of the reactor coolant pumps during the LOCA is determined from a homologous pump model. The CRAFT model incorporates bypass of the ECCS water and performs bypass calculations in the flow path that connects the lower head and downcomer.

The REFLOOD code permits analysis of reactor coolant behavior during the core refill and reflood phases of the LOCA. CRAFT results at the end of blowdown define the starting point for the REFLOOD calculations. Since the pin model in CRAFT is different from that in REFLOOD, an energy balance is made so that REFLOOD maintains the correct stored energy in the core. It is intended that the steam generating radial cladding nodes per axial level. The thermal properties for the fuel and cladding are input in tabular form as functions of temperature. Prior to any transient calculations, THETA 1-B is initialized and compared with the stored energy calculations of the steady-state fuel pin code, TAFY[1] (see Appendix B). Calculated results and input parameters from TAFY are input into THETA 1-B. If the average fuel temperatures from THETA 1-B are less than those predicted by TAFY, a multiplier on the as-calculated gas-gap heat transfer coefficient is varied slightly. The intent is to ensure that the initial stored energy calculations in THETA 1-B compare with the steady-state code. Hot spot mass flux, pressure, power, and entering enthalpy as functions of time during blowdown are input from CRAFT into THETA 1-B. THETA 1-B calculates the mode of heat transfer, channel fluid properties, heat transfer from the fuel to the cladding, and from the cladding to the water, changes in pin dimensions due to thermal expansion and pressure differentials, and cladding failure. The input for the cladding failure model is the same as that for the CRAFT model. After the end of blowdown, the hot pin is assumed to undergo adiabatic heat-up until the water level in the vessel reaches the bottom of the core. The reflooding period is then analyzed by interfacing REFLOOD with THETA 1-B. Heat transfer coefficients derived from the PWR-FLECHT data are found for average flooding rates from REFLOOD. These coefficients, along with power and the saturation temperature of the water, are input into THETA 1-B to calculate the pin thermal response and metal-water reaction.

3) Maximum Hydrogen Generation. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
4) Coolable Geometry. Calculated changes in core geometry shall be such that the core remains amenable to cooling.
5) Long-Term Cooling. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Babcock and Wilcox's conformance to these criteria is described as follows:

1) Peak Cladding Temperature. Virtually all of the material discussed in Sections 4.0 and 5.0 of this report pertain to items that affect the peak cladding temperature. Babcock and Wilcox has presented the results of their analysis for a base plant. The results of these analyses show that the peak cladding temperature is below 2200°F for the complete spectrum of break sizes.
2) Maximum Cladding Oxidation. The calculational methods used by Babcock and Wilcox to calculate the extent of cladding oxidation anywhere in the core is discussed in Section 4.1.5 of this report.

period necessary, and the provisions for maintenance of the components and systems required for long term cooling.

We have requested that Babcock and Wilcox address the potential problem of high concentrations of boric acid in the core region during long term cooling.

The Regulatory staff concludes that with the above additional documentation, the Babcock and Wilcox ECCS model and supplemental information regarding the design of the systems needed for long term cooling provides an acceptable means of satisfying the requirements of Criterion 5 of paragraph 50.46.

5) Coolable Geometry. The Babcock and Wilcox ECCS model includes the effect of possible swelling and rupture of the fuel rod cladding and the resulting reduction in flow area. The extent of swelling and rupture that is predicted results in a geometry that is amenable to analysis. Swelling of the fuel clad, if it occurs, will increase the heat transfer area of those fuel rods. Babcock and Wilcox states that the FLECHT test results indicate that blockage increases turbulence which would tend to increase the heat transfer coefficient. However, to be conservative, their model did not include this increased heat transfer rate in their analyses. The Regulatory staff concludes that the feature of the Babcock & Wilcox model that considers possible swelling and rupture of the fuel rod cladding, along with satisfying criteria (1) and (2), are sufficient to satisfy the requirements of Criterion (3) of paragraph 50.46.

3.0 SUMMARY OF STAFF INDEPENDENT CALCULATIONS

3.1 Introduction

The Babcock and Wilcox reactor selected for staff calculations of the LOCA with the computer programs and techniques prescribed in the Water Reactor Evaluation Model is the Oconee Unit One plant. This reactor has two steam generators and four cold legs. The calculations were performed for the 15x15 fuel assembly design. The rated power is 2772 MW(t).

The nodal model used for the RELAP4-EM code blowdown transient consists of 46 volumes and 64 connecting junctions. The plant is modeled with two basic loops. One loop contains a single steam generator and a lumped representation of the two connected intact cold legs. The second loop contains a single steam generator and two cold legs; one is intact and the other represents the postulated double-ended break.

The detailed core thermal response during blowdown is considered in a hot channel calculation with the RELAP4-EM code model consisting of 9 volumes and 16 junctions. In this model, an average core region, a single hot assembly, and a single hot rod are modeled. The hot rod central region is modeled with eight 3-inch axial segments to determine clad rupture conditions, flow blockage, and metal-water reactions in the manner specified in Appendix K of 10 CFR 50.

The reflood transient following refilling of the reactor vessel lower plenum up to the bottom of the active core is computed with the RELAP4-REFLOOD code using a model consisting of 6 volumes and 9

4.0 CONFORMANCE TO APPENDIX K

This section addresses each required and acceptable feature of the Babcock and Wilcox evaluation model as specified in Part III of the new rule. Each paragraph summarizes what the rule requires, discusses what Babcock and Wilcox proposes, whether this proposal meets the rule and, finally, indicates the acceptability of the position.

4.1 Sources of Heat During the LOCA

4.1.1 The Initial Stored Energy in the Fuel

The rule requires that the steady-state temperature distribution and stored energy in the fuel before the loss-of-coolant accident shall be calculated for the burnup that yields the highest calculated cladding temperature (or, optionally, the highest calculated stored energy). To accomplish this, the thermal conductivity of the UO_2 shall be evaluated as a function of burnup and temperature, taking into consideration differences in initial density, and the thermal conductance of the gap between the UO_2 and the cladding shall be evaluated a function of the burnup, taking into consideration fuel densification and expansion, the composition and pressure of the gases within the fuel rod, the initial cold gap dimension with its tolerances, and cladding creep.

A steady-state fuel thermal performance model for use in the LOCA analysis has been approved by the staff for B&W as part of the densification review. The basis of that model is the TAFY program[1] with the appropriate sanctions outlined in the Staff Evaluation (see Appendix application that the volumetric average fuel temperature at the maximum power location in the calculation is equal to or greater than that calculated in the approved version of TAFY.

Babcock and Wilcox states that the initial temperature distribution in the fuel is that which yields the highest calculated cladding temperature and that the initial fuel temperatures used in the CRAFT and THETA codes are consistent with those as determined by TAFY.

To implement the rule requirements, the CRAFT code iterates on the gap gas conductivity until temperatures are achieved which match those from TAFY, the approved B&W steady-state fuel pin model. This is acceptable.

Prior to any transient calculations, the THETA 1-B code is initialized and compared with the stored energy calculations (average fuel temperature) of the AEC-approved steady-state fuel pin code, TAFY. Calculated results and input parameters from TAFY (such as clad and fuel dimensions, surface roughness, Poisson's ratio, radial power shape within the fuel and gas composition) are input into THETA. If the average fuel temperatures from THETA 1-B are less than those predicted by TAFY (differences of about 20 to 30°F have been calculated), the multiplier on the calculated gas-gap heat transfer coefficient is varied slightly. This insures that the initial stored energy calculations in the THETA 1-B code compare with the steady-state code. The use of a .75 multiplier on the TAFY calculated gap conductance was subcritical reactivities of the order of 50 dollars are introduced. By the end of the blowdown period, the fission power rate is a very small (and still decreasing) fraction of the total power being generated (mostly from fission product decay heat). Scram reactivity would add an additional amount to this shutdown reactivity but is neglected for the large break. For the small break, scram is assumed in reducing the power level.

Babcock and Wilcox calculates the fission power history during the transient in the CRAFT code using the moderator inventories and fuel temperatures calculated in that code (as well as pressure signals for scram) and a point kinetics model of normal formulation similar to that used in their IPS calculations. Scram reactivity is not included for large breaks but is for small breaks (less than 0.5 $ft.^2$). Maximum times from signal to insertion are used. The Doppler coefficients to be used are from End-of-Life (EOL) conditions and thus most negative for the cycle. This assumption contributes conservatively toward opposing the initial power decrease. The moderator density coefficient to be used will be at least as positive as that specified as maximum allowable in the Technical Specifications. The moderator coefficient which has been used in the calculations corresponds to the assumption of a zero moderator coefficient (temperature or density) at the initial full power state. This assumption would correspond to an extreme condition for a Beginning-of-Life (BOL) state. Later times in the cycle would be more negative. Babcock and Wilcox plants are expected isotopes are given in ANS-5.1. The production rate of 239 isotopes for current reactor designs averages about 0.7 per fission (generally increasing slightly with burnup over a fuel cycle), and this combined with the decay energy rates of ANS-5.1 and assuming equilibrium concentration (maximum possible) of U-239 and Np-239 gives initial power production (at start of the transient) from actinides of about 0.3% of full power. This is about 4 to 10% of the power from fission product decay over the time period of interest to LOCA. The equilibrium concentration assumption can generally be expected to be conservative since half lives of up to about 2 days are involved. Approximately a week at the Technical Specification maximum peaking and power density conditions to be assumed for LOCA analysis would be required to fully achieve such equilibrium conditions.

Babcock and Wilcox proposes to use the decay properties given in ANS-5.1 at equilibrium concentration conditions, and to use a production term giving an initial power production of 0.36% of full power. This would correspond generally to about a 20% increase over results using an expected end-of-cycle production rate in combination with the ANS-5.1 decay properties. The power and power density to which the full power refers is 102% of licensed power in combination with Technical Specification maximum peaking factors.

curve in the LOCA analysis codes result in a close approximation to the required results. The power and power densities to which this will be referenced will be 102% of licensed power with maximum Technical Specification peaking factors. For the fraction of local generated energy deposited directly in the local hot pin region, Babcock and Wilcox will use 0.973 for the blowdown period and 0.96 for post-blowdown. The former value is a continuation of the fraction used for heat generation within a fuel pin at the initial conditions and the latter value is calculated to occur with decay heat gamma redistribution and includes both structural gamma absorption and gamma smearing effects. These factors, which were calculated and described in BAW-10033,(12) are the same as those used under the Interim Statement of Policy which were reviewed and approved at that time.

Thus, Babcock and Wilcox meets the rule by using the required fission product decay heat rate curve as referenced to the required initial 102% licensed power level and Technical Specification peaking factors power densities, and has justified the use to be employed for gamma redistribution effects by a suitable calculation. The resulting fission product decay heat values and peaking factor reduction for gamma redistribution is acceptable to the Regulatory staff.

4.1.5 Metal-Water Reaction Rate

The rule considers the effects of metal-water reaction on energy release, hydrogen generation and cladding oxidation. The rate of reaction must be calculated using only the Baker-Just equation

The Babcock and Wilcox method for calculating core wide hydrogen generation is adequate provided it can be shown that the temperature response for each axial elevation in the hot channel envelopes the temperature response of other elevations in the core at the same linear heat generation rate, and that the use of a core power distribution which yielded the highest peak cladding temperature envelopes the use of all other core power distributions.

As previously stated, the Babcock and Wilcox method for calculating the total energy release applies the Baker-Just equation to the total reacting surface area of the cladding without steam limitations. However, Babcock and Wilcox does not change the oxide thickness at the time of rupture. Since the metal-water reaction rate is inversely related to the oxide thickness for the current time increment, the B&W method underpredicts the rate of energy release. To be acceptable, the oxide thickness used to calculate metal-water reaction rate must be thinned on both reacting surfaces at the rupture elevation upon the incidence of rupture.

For calculating the minimum fraction of the unreacted cladding wall thickness after oxidation, Babcock and Wilcox reduces the initial cladding thickness. The reduced thickness is based upon an assumption of conservation of cladding cross-sectional area for the maximum swelling expansion predicted. The Babcock and Wilcox heat-up code accounts for both inside and outside metal-water reactions. Their method for calculating the minimum fraction of the unreacted cladding

4.1.7 PWR Primary-to-Secondary Heat Transfer

The rule requires that the heat transferred between the primary and secondary systems through heat exchangers (steam generators) shall be taken into account.

In CRAFT 2, Babcock and Wilcox employs two control volumes and a flow path to describe the tube section of the once-through steam generator. A second control volume is assigned to the secondary side, and heat transfer coefficients in the primary and secondary sides describe heat flow through the tube walls. Each heat transfer coefficient that is calculated is based on the Dittus-Boelter correlation. This procedure accounts for the varying secondary temperature during the transient due to the effects of feedwater flow and secondary-to-primary heat flow. A Babcock and Wilcox sensitivity study compares the base case calculation with an analysis conducted with a high heat transfer coefficient from the secondary side. The study showed very little difference in the final results (variation of 5°F in peak clad temperature).

In the REFLOOD calculation, Babcock and Wilcox used the same basic model with a control volume representing the secondary side of each steam generator. They assume that the secondary side contains water only. The heat flow from the secondary side is determined by a forced convection correlation (Dittus-Boelter) and is corrected by is accommodated in their calculations of gap conductance, zircaloy embrittlement, hydrogen generation and flow blockage.

The Babcock and Wilcox evaluation model calculations for gap conductance at the start of the transient are consistent with their staff-approved fuel rod thermal performance code (see Appendix B). The maximum stored energy is assumed to exist just prior to the postulated LOCA. The associated gap conductance parameters of gap composition and gap dimensions at the initiation of the LOCA are input from those calculated by their thermal performance model. The parameters are then varied throughout the course of the LOCA to account for the transient phenomena. The transient gap conductance calculation considers both the time-variant gap dimensions and the temperature-variant thermal properties.

Babcock and Wilcox has calculated the thermal and elastic contributions to the transient gap dimensions and have used representative coefficients to calculate thermal expansion. They have calculated the elastic contributions to gap size from a thin wall assumption using appropriate elastic material constants. By applying selected data, Babcock and Wilcox has developed an estimate for plastic swelling at rupture (Figure A-2, BAW-10091) and a best estimate for the temperature of rupture (Figure A-1, BAW-10091). The same selected data provides a basis for a Babcock and Wilcox flow blockage table (Appendix F, BAW-10092) which conservatively infers that all swelling which contributes to flow blockage occurs unrestricted by interference with circumjacent rods. The Babcock and Wilcox data

During a postulated LOCA, the temperatures and the heat transfer conditions may vary rapidly and significantly. These complex variations are not easily modelled. The staff considers the Babcock and Wilcox best estimate (Figure A-1, BAW-10091) of their data on temperature of rupture versus differential pressure (plotted as stress) to be a sufficient estimate of the incidence of rupture. Their estimate is enveloped by the somewhat systematically behaved upper bound of available data and the ultimate strength as a function of temperature. Additionally, Babcock and Wilcox overestimates most of their data on plastic circumferential strain at rupture. Their data was selected for heating rate, heating method, method of strain measurement and thermo-mechanical processing. The staff has employed a more exhaustive, uncensored data base and consider that Babcock and Wilcox does not underestimate the degree of swelling below approximately 1700°F (Figure A-2, BAW-10091). The subsequent use of this data (Appendix F, BAW-10092) does not underestimate the magnitude of the coolant flow blocked. However, for ruptures at temperatures above 1700°F, Babcock and Wilcox underestimates the degree of swelling. Approximately 70% circumferential swelling (TCE) will not underestimate the applicable data evaluated by the staff. Babcock and Wilcox should use 70% swelling for rupture temperatures over 1700°F until a more favorable value has been justified.

a. The initial assumed internal pin pressure must exceed the maximum predicted during normal operation for the design being analyzed.

b. If the rod with the highest peak clad temperature ruptures, then the rupture must have occurred prior to the end of blowdown.

c. The curve for strain versus temperature of rupture (Figure A-2, BAW-10091) is valid for ruptur[illegible]eratures not over 1700°F. For rupture temperatures above 1700°F, 70% circumferential swelling will not underestimate the degree of swelling. The use of 70% by Babcock and Wilcox is acceptable to the staff.

4.3 Blowdown Phenomena

4.3.1 Break Characteristics and Flow

4.3.1.1 Break Spectrum

The applicant is required to analyze a spectrum of (a) double-ended pipe breaks ranging in cross-sectional area up to and including that of the largest pipe in the primary coolant system, and (b) longitudinal splits in the largest pipes, with an area equal to the cross-sectional area of the pipe. Babcock and Wilcox has analyzed a range of double-ended pipe breaks up to the largest pipe in the primary coolant system. For the longitudinal split, twice the cross-sectional area of the largest pipe was analyzed. Breaks in the hot-leg piping was limited to the area of the largest pipe only. Babcock and Wilcox believes the latter limitation is justified because there is no discharge coefficient, the range of discharge coefficients shall be extended until the maximum clad temperature calculated by this variation has been achieved.

Babcock and Wilcox has proposed to use the orifice equation for the subcooled portion of primary system blowdown, and the Moody correlation specified above after the leak node quality is 0.0 or greater. In addition, the spectrum of breaks specified has been analyzed for both postulated guillotine and split type breaks. These breaks were analyzed for values of the discharge coefficient of 1.0, 0.8 and 0.6. For both types of break, the highest peak clad temperature was determined to be within the range analyzed. Therefore, it is concluded that the Babcock and Wilcox method and analyses are acceptable.

4.3.1.3 End of Blowdown

For postulated cold leg breaks, the Acceptance Criteria require that all emergency cooling water injected into the inlet lines or the reactor vessel during the bypass period shall in the calculations be subtracted from the reactor vessel calculated inventory. This bypassing shall end in the calculation at a time designated as the "end-of-bypass," after which the expulsion or entrainment mechanisms responsible for the bypassing are calculated not to be effective. The end-of-bypass

When bypass is calculated to occur, the proposed Babcock and Wilcox evaluation model assumes that the ECC water entering the downcomer is lost and it is subtracted out of the downcomer liquid inventory following the pressure search in every time step. The energy associated with this mass is that of the liquid in the downcomer and it is also subtracted. It is the understanding of the staff that the mass and energy removed from the downcomer is lost from the system for the purposes of the blowdown calculations, e.g. it is not deposited in any other control volume. This point, however, should be clarified in the documentation (see p. 2-50 of BAW-10092).

The downcomer in the Babcock and Wilcox evaluation model is represented by a single heterogeneous node. This representation permits separation of steam and water. Typically, the lower portion of the downcomer will be filled with saturated water and steam bubbles while the upper portion will contain steam. Thus, the flow from the downcomer to the break will be predominantly steam.

Experimental evidence on small scale systems indicates that during a fast blowdown the two phases are mixed.[13,14,15] This is expected to apply to the early portion of the blowdown. Once ECC injection starts the physical picture is somewhat different. During the bypass period, a high quality steam/water mixture is flowing upward in the downcomer, while the upper portion of the downcomer is filled with

Cooling Systems for Light-Water Cooled Nuclear Power Reactors. For Noding near the Break, the Rule requires that the noding in the vicinity of and including the broken or split sections of pipe and the points of ECCS injection shall be chosen to permit a reliable analysis of the thermodynamic history in these regions during blowdown.

To assure reliable thermodynamic conditions at the leak node, Babcock and Wilcox performed a sensitivity study by varying the control volumes near the vicinity of the break until a numerically convergent solution was obtained. Convergence was judged on the basis of overall system pressure response and leak flow; if these variables were consistent, there would be no provision for feedback to cause changes in other variables. The staff believes that the convergence criteria based on the above two primary parameters should provide a sound basis for the selection of the number of break nodes. Three models were studied by Babcock and Wilcox: a 2-, 4-, and 6- node cold leg. Figure 5-13 in BAW-10091 shows the noding diagram used.

For the split break, the leak path was always in node 29. A comparison of leak flow versus time for the 2- and 4-node models showed that the 2-node model predicted higher leak rate for the first 11 seconds. Babcock and Wilcox attributed the difference in flow rate to momentum ΔPs. The momentum flux term acted on an average area from the downcomer to the break node in the 2-node model and acted

Like the split break, the plots for leak flow and the core pressure vs time for the 4-node model and the 6-node model are virtually identical.

Originally, the 2-node cold leg model was used in the base case by B&W. On the basis of convergence criteria, the 4-node cold leg model was adopted into the base case model for both split and double-ended breaks. The staff has reviewed the results of this noding study and concludes that B&W's choice of a 4-node cold leg model is in conformance with Appendix K and gives a reliable thermodynamic history in the break region during the blowdown.

4.3.2 Frictional Pressure Drops

The rule requires that the frictional losses in pipes and other components including the reactor core shall be calculated using models that include realistic variation of friction factor with Reynolds number, and realistic two-phase friction multipliers that have been adequately verified by comparison with experimental data, or models that prove at least equally conservative with respect to maximum clad temperature calculated during the hypothetical accident. The modified Baroczy correlation (Baroczy, C. J., "A Systematic Correlation for Two-Phase Pressure Drop," Chem. Engining. Prog. Symp. Series, No. 64, Vol. 62, 1965) or a combination of the Thom correlation (Thom, J. R. S., "Prediction of Pressure Drop During Forced Circulation Boiling of Water," Int. J. of Heat & Mass Transfer, 7,

(4) momentum change due to compressibility, (5) pressure loss resulting from wall friction, (6) pressure loss resulting from area change, and (7) gravitational acceleration. Any omission of one or more of these terms under stated circumstances shall be justified by comparative analyses or by experimental data.

The compliance to the rule for the reviewed momentum equation is discussed in detail in Appendix D and is summarized on a term-by-term basis as follows:

TERM IN EQUATION	STATUS
(1) Temporal Term	ACCEPTABLE
(2) Momentum Convection	ADDITIONAL DOCUMENTATION REQUIRED
(3) Area Change Momentum Flux	ADDITIONAL DOCUMENTATION REQUIRED
(4) Momentum Change Due to Compressibility	ACCEPTABLE
(5) Pressure Loss Resulting from Wall Friction	ACCEPTABLE
(6) Pressure Loss Resulting from Area Change	ADDITIONAL DOCUMENTATION REQUIRED
(7) Gravitational Acceleration	ACCEPTABLE

4.3.4 Critical Heat Flux

The applicant may employ any correlation developed from appropriate steady-state and transient experimental data in predicting critical heat flux during LOCA transients. The computer programs in which the

(a) The Groeneveld correlation shall not be used in the region near its low-pressure singularity.

(b) The nucleate boiling term of the W correlation and the entire McDonough, Milich and King correlation shall not be used during the blowdown after the temperature difference between the clad and the saturated fluid first exceeds 300°F.

(c) Transition boiling heat transfer shall not be reapplied for the remainder of the blowdown and only during reflood, when justified by the local conditions.

Babcock and Wilcox uses the McDonough, Milich and King correlation for transition boiling (between DNB and film boiling), and the Dougall-Rohsenow correlation for film boiling within all the restrictions listed. Since B&W has selected post-CHF heat transfer correlations from those designated as acceptable and meets all restrictions imposed by Appendix K, this is acceptable to the staff.

Pool boiling heat transfer is calculated with the Morgan correlation. Further justification of the switching criterion and the appropriate useful range of the correlation must yet be documented by B&W.

4.3.6 Pump Modeling

The characteristics of rotating primary system pumps (axial flow, turbine, or centrifugal) shall be derived from a dynamic model that includes momentum transfer between the fluid and the rotating member, four regions of pump operation. Approximately 160 tests were performed under two-phase conditions. These data are reported in ANCR-1150, "Experiment Data Report for Single and Two-Phase Steady-State Tests of the 1-1/2 Loop Mod-1 Semiscale System Pump," by D. J. Olson.

Babcock and Wilcox has also performed air-water tests on a one-third scale primary coolant pump. The test results are still under analysis; however, the staff understands from Babcock and Wilcox that the two-phase air-water test results indicate trends similar to those reported above; that is, significant pump head degradation occurs under two-phase conditions.

The pump model proposed by Babcock and Wilcox is similar to the pump model in the RELAP4-EM computer program. To calculate pump degradation under two-phase conditions, a pump head multiplier is applied that is a function of void fraction. The multiplier is applied to the homologous pump data obtained under two-phase conditions and is used to correct the predicted pump head assuming single-phase conditions.

The staff concludes that the Babcock and Wilcox pump model is acceptable.

4.3.7 Core Flow Distribution During Blowdown

The rule requires that the flow rate through the hot region of the core (not greater than the size of one assembly) be calculated a cross-flow path. The equation used for cross-flow resistance is given; however, additional information to justify the approach is required.*

Smoothing of the core flow is accomplished by a filter equation applied to the calculated flow rate; however, further justification is needed with regard to the procedures used.

The Babcock and Wilcox heatup calculations are limited to a 2.4 ft segment of the hot pin. The mass flux and entering enthalpy are supplied from the middle flow path (path 13 in Figure 3-3 of BAW-10091) of the CRAFT hot assembly model. It is our understanding that local enthalpy is calculated in the heatup code from the entering enthalpy using a closed-channel enthalpy balance. The staff has requested confirmation of this aspect of the model and has asked that a comparison of the CRAFT and THETA1-B cladding temperature calculations be provided. In addition, confirmation that the peak clad temperature will always occur in path 13 of the CRAFT calculation is required.

The Babcock and Wilcox representation of the reactor vessel is a combination of homogeneous and heterogeneous nodes. The heterogeneous nodes permit steam and liquid separation. No justification is presented for the assumed use of homogeneous and heterogeneous nodes. No comparison has been shown with CRAFT calculations that utilize only homogeneous nodes.

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* An unpublished Babcock and Wilcox work is referenced (BAW-10092, Reference 9).

be one which minimizes the ECCS available to cool the core, yet allows maximum containment cooling. In lieu of performing a sensitivity study of various single failures to arrive at an overall worst case, Babcock and Wilcox has conservatively elected to assume all containment cooling systems operating for their independent containment calculation, and to assume the diesel failure for the ECCS calculation. These two assumptions, while not mechanistically reasonable, certainly minimize containment pressure and ECCS availability and are acceptable to the staff. Since ECCS performance remains unaffected by containment backpressure for small breaks, the diesel failure continues to be worst-case for breaks less than 0.5 ft^2.

Although generic review of the most damaging single failure of ECCS equipment was conducted, the specific application of a worst single failure criterion will be confirmed by the Regulatory staff on each future project. With each application, the staff will continue to examine the relevant plant system piping and instrumentation diagrams to confirm that appropriate single failure assumptions have been made.

4.4.2 Containment Pressure

Appendix K to 10 CFR 50 of the Commission's regulations requires that the effect of operation of all containment installed pressure reducing systems and processes be included in ECCS evaluations. For

Subsequently, the operation of the containment heat removal systems such as containment sprays and fan coolers will remove steam from the containment atmosphere. When the steam removal rate exceeds the rate of steam addition from the primary system, the containment pressure will decrease from its maximum value.

The B&W CONTEMPT code calculates the containment pressure both during the blowdown and reflood phases of the loss-of-coolant accident. Mass and energy to the containment is provided by the CRAFT-2 code for the blowdown period as described in BAW-10092 and during the reflooding period by the REFLOOD code as described in BAW-10093. The flow of steam through the reactor loops and into the containment during the reflooding period is sensitive to the steam density as determined from the containment pressure. For this reason, iterations may be required between REFLOOD and CONTEMPT to establish consistent reflooding and containment pressure analyses. The Babcock and Wilcox CONTEMPT code considers heat absorption from the containment atmosphere by the action of the passive heat sinks, containment spray systems and ventilation fan coolers.

We have evaluated the Babcock and Wilcox containment pressure model in accordance with Appendix K and have concluded that the model is acceptable provided that conservative input assumptions to the CONTEMPT code are selected.

5. The heat sink structures prescribed in Appendix A will be used for the internal steel and concrete structures. This method should envelope the actual structures. Babcock and Wilcox will, however, provide verification of the actual heat sinks for each plant at a later date. Some of the heat sinks are assumed to be painted in the model based on measurements made in the Category 1 plants.

The staff has reviewed the input assumptions for the Category 1 plants as outlined above and found them acceptable. The staff has also conducted a confirmatory analysis for these plants using the AEC CONTEMPT code and have calculated the same containment backpressure as Babcock and Wilcox. It is, therefore, concluded that the containment backpressure calculated for the Category 1 plants is adequately conservative and, therefore, acceptable.

For other plant sizes, Babcock and Wilcox will provide and justify the specific input assumptions at a later time. This information will include the assumptions made regarding the containment heat sinks containment volume, operation of the active heat removing systems and mass and energy release to the containment. This information is required before we can conclude the containment pressure for any of the plants other than those of Category 1 in Appendix K are acceptable.

The staff believes the applicant's model will calculate a sufficiently conservative containment backpressure to meet the requirements

Figure 3 Multinode Representation of the Reactor Coolant System

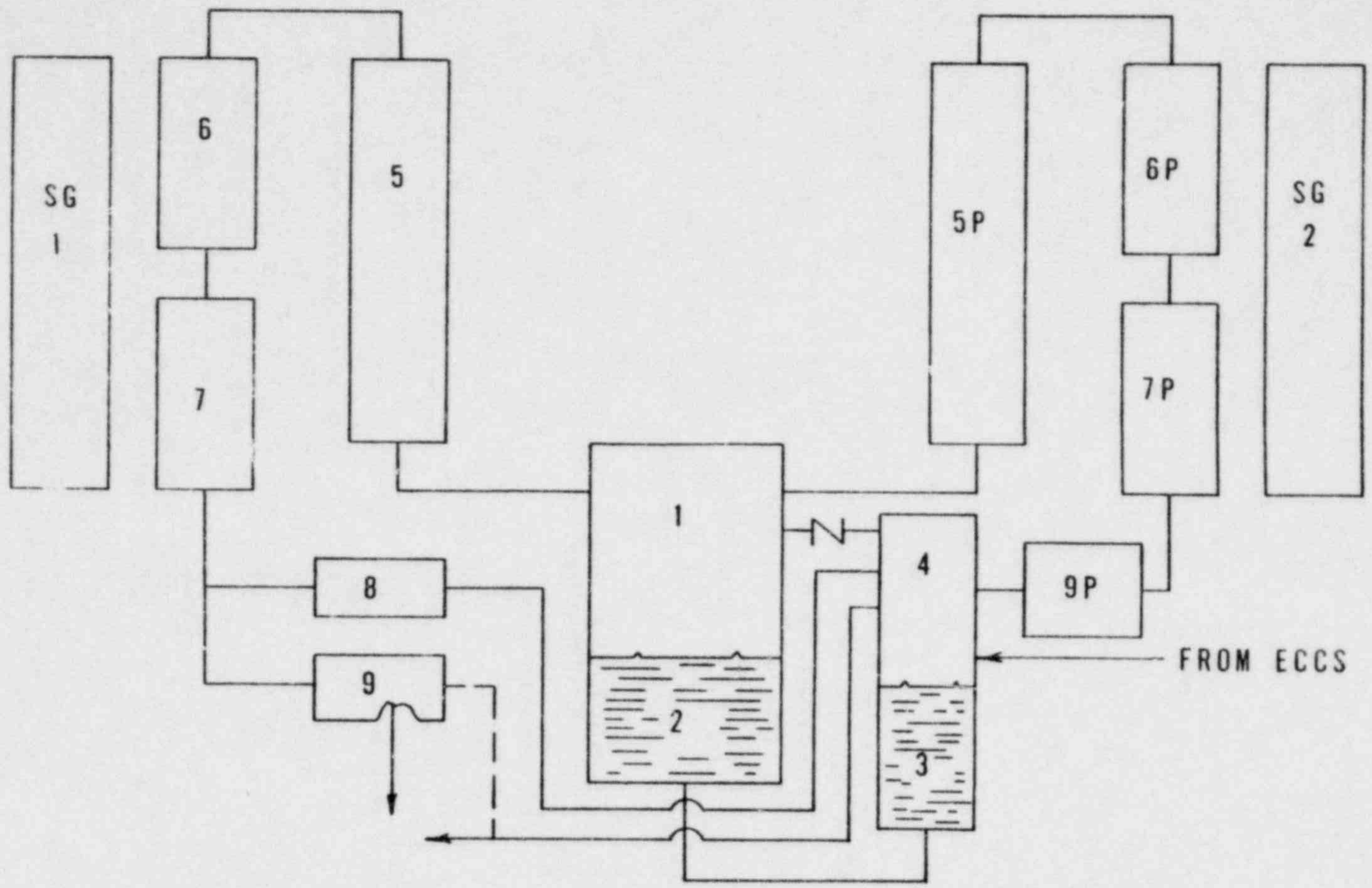
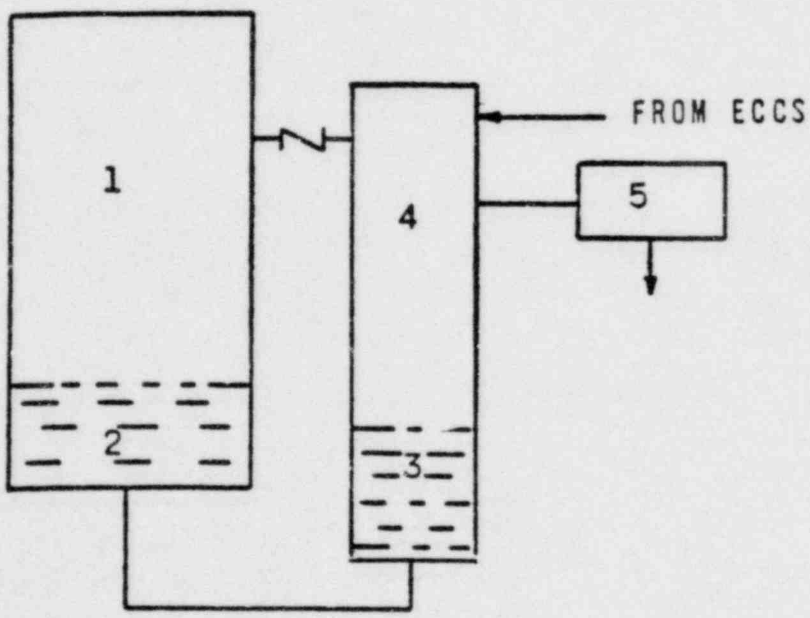


Figure 4 REFLOOD Model of Reactor Coolant System for Cold Leg Break



estimate the heat transfer from the shell metal to the secondary side water and thence through the tube wall to the primary coolant. The principal assumptions are:

- The heat transfer area and heat transfer coefficient are constant.
- The heat transfer coefficient between the shell and secondary water is 1000 BTU/h ft^2°F.
- The secondary water temperature is homogeneous.
- The heat transfer coefficient between the secondary water and the tubes is based on the Nusselt film condensation correlation and is typically 780 BTU/h ft^2°F.
- The coefficient between the tubes and primary coolant is that of Dittus-Boelter, with a two-phase multiplier.

F. Primary Metal Heat Release

This calculation is based on a single metal slab representing the mass of metal in contact with the coolant in each control volume. Energy release is governed by an overall coefficient which is held constant.

G. The ECCS Injection System

The core flood tank flow calculation is based on the assumptions that:

- The momentum flux terms are negligible.
- The gas pressure in the tanks is calculated using the isentropic ideal gas law.

Supplemental information on the interaction of the ECC water with the steam-water mixture issuing from the vent valves has been requested.

J. Sensitivity Analyses

Of the sensitivity runs reported in BAW-10091, those which are pertinent to REFLOOD include:

(a) A 2.6 $ft.^2$ break in which the quantity of primary metal was reduced from its reference value to one-half that value. The resulting reduction by 30°F in peak cladding temperature was stated to "establish that the use of nucleate boiling for the primary metal heat model is appropriately conservative."

(b) The heat transfer coefficient between the steam generator secondary and primary coolant was varied from the reference value of 0.1 times that used for forward (normal operation) heat transfer to 0.35 times the latter value (and hence, typical of nucleate boiling). The result was a decrease by 5°F in peak cladding temperature.

(c) A degradation multiplier which varies with void fraction was used to study the effect of pump resistance. The final sensitivity study demonstrated that as the degradation increases, the core flow becomes negative earlier and the peak cladding temperatures increase. However, with even greater degradation, the negative core flow is increased and the peak clad temperatures decrease. Babcock and Wilcox choose a multiplier based

- The core flooding tank line resistance value of 9.38 is chosen as the maximum expected for Category 1 plants.
- An overall heat transfer coefficient from the primary metal is input by the user and held constant. The primary metal temperatures are input from CRAFT.

4.4.3.3 Staff Evaluation

Additional information has been requested of Babcock and Wilcox to allow the staff to complete the evaluation. Each aspect of the Babcock and Wilcox REFLOOD model described in Section 4.4.3.2 will be addressed when this additional information is reviewed.

4.4.4 Steam Interaction With Emergency Core Cooling Water In Pressurized Water Reactors

The rule requires that the thermal-hydraulic interaction between steam and all emergency core cooling water shall be taken into account in calculating the core reflooding rate. During refill and reflood, the calculated steam flow in unbroken reactor coolant pipes shall be taken to be zero during the time that accumulators are discharging water into those pipes unless experimental evidence is available regarding the realistic thermal-hydraulic interaction between the steam and the liquid.

Babcock and Wilcox states that for plant arrangements where the bottom of the steam generators are below the bottom of the reactor vessel, no credit is taken for steam flow around loops during the should be the effect of ECC subcooling. The benefit for subcooling was determined from Figure 15 of TN-188. The resultant adjusted hot wall delay time was 1.04 seconds which was less than the water free fall time in a vacuum of 1.1 seconds.

The staff finds the extension of the data in Figure 28, in the referenced TN-188, beyond the data range unacceptable. The Babcock and Wilcox adjustment of the delay time based on ECC water subcooling is also unacceptable to the staff since there is no evidence to substantiate any significant subcooling of the water held in the downcomer. The staff accepts hot wall delay times taken from the t_d $(L/s)^{1.5}$ line of Figure 30 of TN-188. This figure contains Semiscale as well as Creare data. The value of "L" should be taken as the distance from the bottom of the injection pipe to the bottom of the core barrel. The value of "s" is the gap size.

In addition to the value taken from Figure 30, the staff requires that the free fall time of water without steam drag be calculated from the bottom of the injection pipe to the bottom of the core barrel.

The staff requires that the total delay time be determined by adding the hot wall delay from Figure 30 of the above reference to the gravity free fall time.

Babcock and Wilcox CRAFT code contains a description of the pressure drop across vent valves during both the blowdown and reflood periods. The vent valves are simulated in CRAFT as a normal flow data, for a range of parameters consistent with the transient to which they are applied.

For peak power locations at or below the 6-ft level, Babcock and Wilcox uses a modified form of the FLECHT heat transfer correlation. The Modified FLECHT correlation has been compared with FLECHT data with constant and variable flooding rates by Babcock and Wilcox and provides a conservative correlation to it. For peak power locations above the 6-ft elevation, Babcock and Wilcox has developed the REFLECHT correlation which is based on an equivalent power condition and a simulated flooding rate. The REFLECHT correlation has been compared by Babcock and Wilcox to both 8-ft and 10-ft data for both constant and variable flooding rate. The staff concludes that the REFLECHT correlation conservatively represents the data.

The staff concludes that the Babcock and Wilcox method of determining the heat transfer coefficient for reflood rates greater than or equal to one inch per second is acceptable for 15 x 15 geometry. Additional information is required to substantiate the Babcock and Wilcox method for calculating reflood heat transfer for the 17 x 17 geometry. Until such a procedure is approved, "h" for 17 x 17 geometry is to be calculated as $0.8\ h_{FLECHT}$.

The rule requires that during refill and during reflood when rates are less than one inch per second, heat transfer calculations shall be based on the assumption that cooling is only by steam, and shall take

5.0 SMALL BREAKS

5.1 Rule Requirements

The rule requires that the entire spectrum of loss-of-coolant breaks be analyzed using an evaluation model containing the required and acceptable features set forth in Appendix K of 10 CFR 50. These features are discussed in Section 4.0 of this report. The rule does not define a separation of the break spectrum into a large or small break region so that this separation has been left to the vendors' justification.

In past B&W analyses of transients over the break spectrum, the 0.5 ft^2 break has been used as the dividing break size between large and small breaks. The justification for this separation is based on the slower system depressurization that would occur for small breaks allowing considerable phase separation between steam generated due to depressurization and heat addition, and the saturated liquid in the reactor system. Such a separation into two separate phases is not considered realistic for large breaks due to the rapid rate of vapor formation and the high flow rates that occur, so that fluid dynamics for the large breaks are based on the existence of a homogeneous two-phase mixture in the reactor system during blowdown. For the small breaks, the slower vapor formation and gradual flow reduction would allow gravity forces on the vapor and liquid phases to produce phase separation, so that the small break model hydrodynamics are based on the existence of a heterogeneous two-phase condition during blowdown.

c. The THETA1-B[9] program to determine hot pin thermodynamics during the LOCA based on fluid conditions determined in the CRAFT and FOAM program calculations.

The blowdown calculation with the CRAFT-2 program for small breaks includes consideration of all the required features imposed by Appendix K of the rule with the exception of bypass of injected ECCS fluid in Paragraph I.C.1.c, and reflood heat transfer in Paragraph I.D.5. Phase separation dynamics in CRAFT-2 are based on the Wilson bubble rise correlation which is adjusted during the quiescent flow period to match two-phase mixture heights determined by the FOAM program. Froth heights and steaming rates determined in the FOAM program are based on average core power and core liquid volume, and these results are used for the hot pin calculation in the THETA1-B program to provide a conservative estimate of hot pin thermodynamics. This process is considered by B&W to be conservative since use of hot channel conditions in FOAM would produce greater froth heights and steaming rates.

5.3 Model Conformance to Rule

The B&W small break model is considered to be in conformance with the rule because the model used to perform the small break analysis is in most respects identical to that used for the large break analysis evaluated in Section 4.0. In using the CRAFT-2 and THETA1-B programs for small break analyses, consideration of the heat sources during the LOCA, swelling and rupture of cladding, and blowdown thermal-hydraulics are performed in almost the identical manner used for large break

6.0 DOCUMENTATION

The rule requires that a complete description of the ECCS evaluation model be furnished. Furthermore, the description is to be sufficiently detailed to permit technical review of the analytical approach. Changes in the evaluation model resulting in a calculated fuel cladding temperature difference of more than 20°F are to be submitted by amendments to the accepted model. A complete listing of each computer program used in the evaluation model is required. Solution convergence must be demonstrated by studies of system modeling or noding and calculational time steps. Also, sensitivity studies must be performed to demonstrate solution convergence through analyses of system modeling or noding and calculational time steps. Such sensitivity studies must also evaluate the effect on the calculated results of variations in noding, phenomena assumed in the calculation to predominate, and values of parameters over their applicable ranges. To the extent practicable, predictions of the evaluation model, or portions thereof, must be compared with applicable experimental information.

Babcock and Wilcox's response to each of the aforementioned items will be acceptable when the additional documentation requested by the staff has been received (see Section 4.0). A detailed description of their evaluation model is included in BAW-10091 through BAW-10095.[6,7,8,9,10] A complete package of the code listings has been made available to the Regulatory staff. Sensitivity studies have been performed and are described in BAW-10091.[6] Comparison with steps. Parametric comparisons of peak fuel and clad temperature, break flow and energy release demonstrate an adequate degree of convergence between the base case and modified base case time steps.

7.0 REFERENCES

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APPENDIX A

CONTAINMENT BACKPRESSURE FOR ECCS EVALUATION

I. Input Information for Model

A. Initial Containment Internal Conditions

The minimum containment gas temperature, minimum containment gas pressure, and maximum humidity encountered under limiting normal operating conditions should be used in the containment model.

B. Initial Outside Containment Ambient Conditions

A reasonably low ambient temperature external to the containment should be used in the containment model.

C. Containment Volume

The maximum net free containment volume should be utilized in the containment backpressure model. This maximum free volume should be determined from the gross containment volume minus the volumes of internal structures such as walls and floors, structural steel, major equipment and piping. The individual volume calculations should reflect the uncertainty in the component volumes.

II. Active Heat Sinks

A. Spray and Fan Cooling Systems

The assumptions for containment cooling systems that are to be utilized in the containment backpressure model should include the assumption of full containment safety systems operating at their

Since the measurement of miscellaneous steel (e.g., equipment supports, gratings and crane, and concrete internal structures) may require extensive effort and be time consuming, the staff has compiled and developed this heat sink data provided by applicants in previous reviews. This would allow analyses in the interim, pending eventual complete identification of the available heat sinks. Therefore, as an internal measure, applicants and licensees may assume the following to size the amount of passive heat sinks:

1. Calculate and use the surface area and thickness of the primary containment steel shell or steel liner and associated anchors and concrete, as appropriate.
2. Estimate the exposed surface area of other steel heat sinks in accordance with Figure A-1 and assume the average thickness to be 3/8 inch.
3. Model the internal concrete structure as a slab with a thickness of 1 foot with an exposed surface area of 160,000 ft^2.

The heat sink thermophysical properties that would be acceptable are shown in Table A.2.

B. Heat Transfer Coefficients

The following conservative condensing heat transfer coefficients for heat transfer to the exposed static heat sink during the blowdown and post-blowdown phases of the accident should be used in the containment model (See also Figure A-2):

The heat transfer coefficients calculated using the above correlations should be applied to all exposed static heat sinks both metal and concrete for both painted and unpainted surfaces.

Heat transfer between adjoining materials in static heat sinks should be based on the assumption of no resistance to heat flow at the material interfaces. An example of this type of interface is the containment liner to concrete interface.

TABLE A.2

HEAT SINK THERMOPHYSICAL PROPERTIES

Material	Density lb/ft^3	Specific Heat Btu/lb-°F	Thermal Conductivity Btu/hr-ft-°F
Concrete	145	0.156	0.92
Steel	490	0.12	27.0

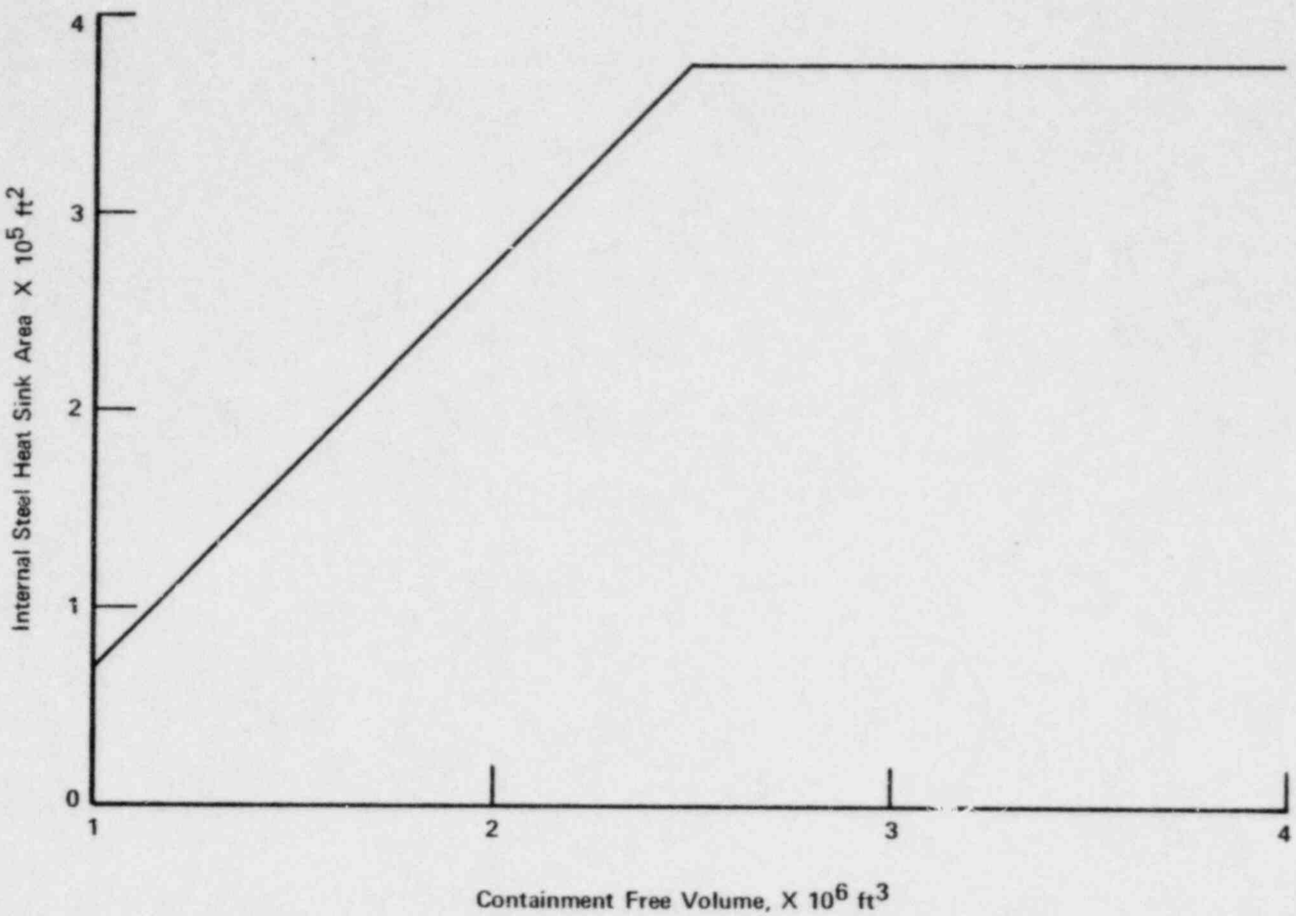

Figure A-1 Area of Steel Heat Sinks Inside Containment

APPENDIX B

EVALUATION OF THE B&W CODE TAFY (July 6, 1973)

I. Evaluation

The B&W computer code TAFY[1] was used to calculate the gap conductance, fuel temperatures, and stored energy giving appropriate considerations to the effects of fuel densification. The code analyzes the transfer of heat generated in the fuel to the coolant outside the cladding by calculating the temperature profile, stored energy, and gap conductance between fuel pellet and clad.

The parameters and phenomena that determine the heat transfer from the fuel to the coolant are either input to the TAFY code or are incorporated as analytical models in the code. The staff reviewed and evaluated the various assumptions for the input and models as used in the TAFY code in its application to B&W fuel.

B&W assumes instantaneous, and anisotropic fuel densification at BOL and no clad creepdown in accordance with the staff guidelines. However, elastic loading of the clad by reactor coolant system pressure is included in TAFY and reduces the diametral gap by approximately 0.5 mil for the fuel rod cladding. The decrease in gap size is approximately offset by the thermal expansion of the cladding.

The thermal expansion of the fuel is calculated in TAFY using the temperature difference between the volumetric average fuel temperature fuel pins, the sorbed gas released from the fuel to the gap, and the fission gases produced and released from the fuel as a function of time. In TAFY the gap conductance is calculated with the conservative assumption that the entire sorbed gas is released at BOL, with the initial amount of sorbed gas being input to the code. The release of the fission gases is treated in TAFY as a function of volumetric average fuel temperature based on the data by Hoffman and Coplin.[5] The conductivity of the gas mixture is based on the evaluation of Holmes and Baernes.[6] The staff concludes that the models in TAFY for the release of sorbed and fission gases and for the gas conductivity are acceptable.

The gap conductance determined by TAFY is based not only on heat transfer by radiation between the fuel pellet and the clad (generally only a small contribution) and by conduction through the gas in the gap, but also is based on heat conduction at a solid to solid, partial contact area which is assumed to exist between the fuel pellet and cladding. The partial contact area, C_A, is calculated by the expression

$$C_A = 0.1 + 0.9 \times 0.1^{(100\ G/D)}$$

where G is the gap size and D is the fuel pellet outside diameter. From this expression it is noted that for any pellet diameter the minimum partial contact area, independent of gap size is $C_A = 0.1$, i.e., 10% of the pellet circumference is in contact with the clad. Babcoc and Wilcox based the concept of a partial contact area on the evaluation by

Restructuring of the fuel is included in TAFY and is based on restructuring temperatures reported by MacEwan.[9] Columnar grain growth is assumed at a temperature of 3200°F based on the BOL temperature distribution in the fuel. The porosity released in the restructuring process is assumed to migrate toward the center of the fuel leading to the formation of a center void and resulting in a density of 100% TD for the restructured fuel. The result of this center void is a reduction in maximum fuel temperature and thus a lower stored energy in the fuel pin. Photomicrographs provided by B&W of cross sections of fuel pellets after high burnup show the existence of center voids and of restructured fuel due to columnar grain growth. Detailed information on the pellet (initial density, enrichment and pellet dimensions) and on the power history of the pellet during exposure were not available.

The staff concludes that although restructuring of the fuel due to columnar grain growth can take place for certain power histories and operating conditions, insufficient information was provided by B&W to establish a temperature of 3200°F as the temperature at which columnar grain growth is initiated, and to attribute a density of 100% TD to the restructured fuel. In addition, fuel densification induced by radiation could be completed before the fuel experiences a temperature of 3200°F and thus foreclose the fuel restructuring due to columnar grain growth leading to the formation of a central void.

are calculated and reported. The staff has compared these parameters deduced from experiments with the corresponding TAFY predictions on the basis of the following ratios:

$$R_T = \frac{\text{code predicted temperature (°F)}}{\text{experimental temperature (°F)}}$$

$$R_h = \frac{\text{experimental gap conductance (Btu/hr-ft}^2\text{-°F)}}{\text{code predicted gap conductance (Btu/hr-ft}^2\text{-°F)}}$$

The code is conservative with respect to an experimental value if these ratios are greater than 1.0.

A comprehensive summary of the comparisons is listed in Table B.1 and discussed below. The initial sorbed gas content, $S(cm^3/g)$, is listed as a parameter in the table. Since no value is given in any of the references for this parameter, its effect was evaluated parametrically with TAFY in some cases.

The data reported by Kjaerheim and Rolstad[7] were obtained for fuel pin geometries with cold diametral gaps ranging from 1.85 mil to 6.61 mil, linear heat rates ranging between approximately 2 to 15 kW/ft and short irradiation times representing BOL conditions. A total of 82 reported experimental fuel temperatures were compared with TAFY predictions. With an assumed initial sorbed gas content of $S = 0.05$, all data are predicted conservatively, i.e., $R_T > 1.0$, with an average value is $\overline{R_T} = 1.21$. These data form the basis for the partial contact term, C_A, used in the TAFY code.

sets of data which, therefore, cannot be combined. The larger set of conservative data[7] overshadows the conclusions that can be drawn from the smaller set of data,[8] which exhibits a wide range of ratios. The use of a sorbed gas content of S = 0.05 is a reasonable estimate in the TAFY calculations for the above experiments in the absence of a measured value, but it is not conservative, particularly since a value of S = 0.02 had been used in the TAFY calculations for some B&W fuel.

Ditmore and Elkins[10] report gap conductances for a cold diametral gap of 15.8 mil and linear heat rates ranging between 16 and 21 kW/ft. The TAFY predicted gap conductances for corresponding conditions for the data reported result in average ratios, $\overline{R_h}$, of 1.72 and 1.20 for the assumed sorbed gas content of 0.05 to 0.0, respectively. However, for the 8 predictions, only 5 were conservative at S = 0.05 and only 3 were conservative at S = 0.0. Figure B-2 shows measured and TAFY predicted values and clearly indicates that the average ratio, $\overline{R_h}$, is strongly influenced in the conservative direction by two of the data points. The minimum ratio for this set of data with the assumption of S = 0.02 is $R_h = 0.76$.

A fourth set of data with a cold diametral gap comparable to the densified gaps of the B&W fuel is reported by Duncan:[11] 12.0 mil gap, 11-24 kW/ft linear heat rate, BOL conditions, 4 measurements. TAFY predicts the reported gap conductance conservatively in all cases and

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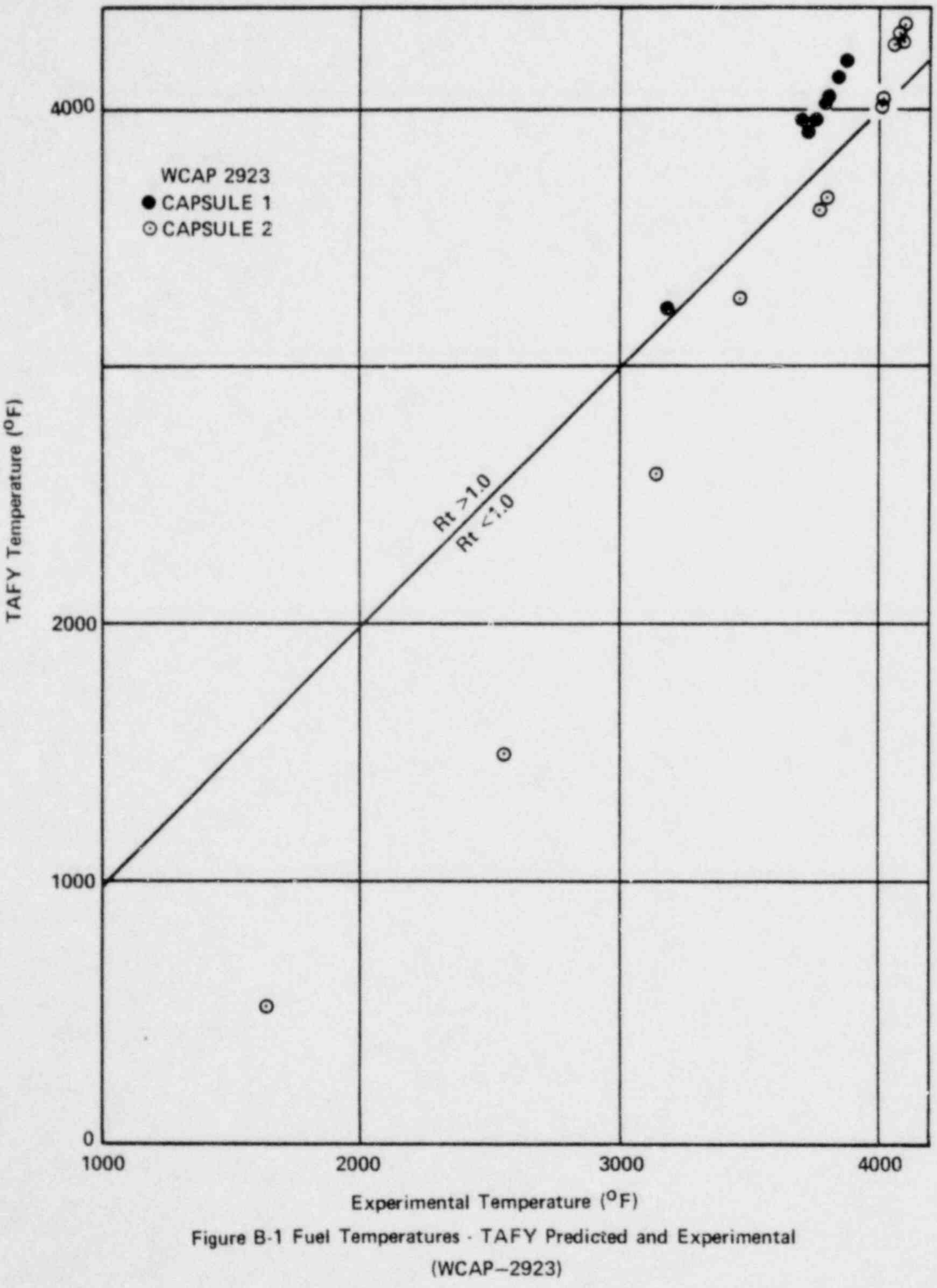

Figure B-1 Fuel Temperatures - TAFY Predicted and Experimental (WCAP–2923)

APPENDIX C

STAFF INDEPENDENT CALCULATION

I. Introduction

The Babcock and Wilcox reactor selected for staff calculations of the postulated LOCA is Oconee Unit No. 1. This pressurized water reactor contains two hot legs and four cold legs, and is designed to operate at a rated power of 2772 MW(t). The reactor contains 177 fuel assemblies and incorporates the lowered primary loop arrangement. The peak linear heat rate assumed in the staff calculation is 16.5 kw/ft. The major plant design parameters are listed in Table 1.

The postulated LOCA was evaluated with the computer programs and calculational procedures described in the Water Reactor Evaluation Model (WREM). The LOCA calculation is performed with the computer codes as follows:

a. RELAP4-EM - Calculates the primary system blowdown and hot rod temperature transient until end of bypass.

b. RELAP4-REFLOOD - Calculates the core reflood rate following the time of bottom core recovery (BOCREC).

c. TOODEE-2 - Calculates the hot rod temperature transient.

II. Blowdown Model

The nodal representation of the primary system is presented in Figure 1, and a description of each node is given in Table 2. The model consists of 46 control volumes (nodes) and 64 junctions. The homogeneous fluid model is assumed in each node except for the pressurizer, core determines that end-of-bypass has occurred when the average velocity in the lower downcomer node is calculated to be zero feet/second following initiation of core flooding tank flow.

The two core flooding tanks are connected directly to the upper downcomer region. Therefore, credit for both tanks is given for the postulated failure of the cold leg pipe.

The Babcock and Wilcox PWRs incorporate once-through steam generators which are modeled with an upper and lower plenum plus two nodes to consider the active tube region. The secondary side of each steam generator is represented as a single node because the pressure and temperatures are nearly uniform during the postulated transient.

III. Hot Channel Model

The core model used to calculate the detailed thermal hydraulic response during blowdown is shown in Figure 2. In this model, time-dependent properties (pressure, temperature, quality and mixture level) obtained from the blowdown calculation are input as boundary conditions on nodes 1 and 8 of the hot channel model (nodes 1 and 14 of the blowdown model). Thus, the response of the core is recalculated including a model for maximum power fuel rod. The fuel nodes and corresponding fluid nodes are presented in Figure 3.

The models for the average core and single hot assembly are the same as in the blowdown calculation. However, a single hot fuel rod is added to the hot assembly fluid node to calculate the detailed thermal response of the peak power fuel rod in the hot assembly. The central two-foot region of the rod is divided into eight 3-inch axial segments (nodes

V. TOODEE 2 Hot Rod Model

The hot rod temperature transient during refill and reflood are calculated with the TOODEE2 computer program. The axial noding of t'e hot rod is similar to the single rod shown in Figure 3 (fuel nodes ;-17).

This calculation starts at the end-of-bypass. At this time, the initial fuel rod temperatures, clad zirconium oxide thickness and cladding plastic strain are obtained from the hot channel calculation (Section III). The calculation assumes adiabatic l:atup until BOCREC. The core reflood rate, fluid inlet temperature and fuel power decay are obtained from the reflood calculation (Section IV). The calculation is terminated after the peak cladding temperature has been determined.

VI. Results

The calculations described in the previous section were applied to Oconee Unit No. 1 to provide an independent evaluation of LOCA transients for comparison with the vendor calculation. Results of the calculations are summarized in Table 4 for a double-ended cold leg break on the discharge side of the pump.

The LOCA transient results for significant parameters are shown in Figure 5 thru 23. The time is noted on the figures for pressurizer liquid depletion, core flooding tank flow initiation and termination, end-of-bypass (EOB), and bottom of the core recovery (BOCREC).

The results of the calculations are presented in the following figures:

FIGURE	DESCRIPTION	COMMENT
18	Temperature transient of ruptured node.	
19	Heat transfer coefficient of rupture node.	
(c)	RELAP4-FLOOD	
20	Core reflooding rate	
21	Total mass in core	
22	Containment Pressure	Calculated by B&W
(d)	TOODEE	
23	Temperature for the hot region	This figure shows temperatures for the fuel and clad rupture and unruptured nodes. The peak clad temperature occured in the ruptured node - 1956°F at 30.6 Sec.

VII. Comparison With Babcock and Wilcox Results

The results presented in the preceding sections are compared with calculations submitted by Babcock and Wilcox (BAW-10091) in Table 5. A graphical comparison of several significant parameters is presented in Figures 24 through 28. Based on these comparisons, the following comments can be made.

(a) Blowdown - The blowdown time and subsequent system depressurization (Figure 24) are noted to be shorter in the staff blowdown calculation. This difference results primarily from the consideration of the momentum flux terms in only simple junctions for the staff calculation whereas B&W includes the momentum flux term in all

TABLE 1

CORE AND PLANT PARAMETERS

Total Power (1.02x2772)	2827 MW(T)
Vessel Coolant Inlet Temperature	556.1 °F
Vessel Coolant Outlet Temperature	607.5 °F
Core Inlet Temperature	555.0 °F
Core Inlet Temperature	608.8 °F
Max. Clad Surface Temperature	658.8 °F
Max. Fuel Centerline Temperature	4587.1 °F
Average Fuel Temperature	2938.7 °F
Total Primary Mass	504,938.0 Lbm
Total Primary Volume	11,600.0 FT^3
Total System Flow Rate	38,306.0 LB/SEC
Total Core Flow Rate	36,901.0 LB/SEC
Pressurizer Pressure	2145.25 PSIA
Secondary Pressure	945.25 PSIA
Accumulator Tanks (2)	600.0 PSIA
Accumulator Mass (2)	133,704.0 Lbm.
Containment Volume	2,000,000.0 FT^3
Hot Leg Diameter	30.0 inches
Cold Leg Diameter	28.0 inches
Cold Leg Break Area	4.2761 FT^2
Core Flow Area	51.91 FT^2
Heat Transfer Area	47,209.2 FT^2

TABLE 2

BLOWDOWN MODEL DESCRIPTION

NODE NO.	DESCRIPTION
1	Lower Plenum Region
2-4	Core - All Assemblies But One
5, 9, 39, 40	Pump Discharge Leg (Broken Loop)
6-8	Core - Hot Assembly Region
10	Containment Building
11	Pump (Unbroken Loop)
12	Pump Discharge Leg (unbroken Loop)
13, 18	Reactor Vessel Annulus (Downcomer)
14	Upper Plenum and Head Region
15	Core Bypass Region
16	Pressurizer Surge Line
17, 29	Core Flooding Tank Surge Line
19-21, 30-32	Hot Legs
22, 33	Steam Generator Inlet Plenum
26	Pump Suction Leg (Double)
27	Pump (Double)
28	Pump Discharge Leg (Double)
37	Pump Suction Leg (Broken Loop)
38	Pump (Broken Loop)
41, 42	Core Flooding Tanks
43	Pressurizer
44, 45	Steam Generator Secondary Region
46	Pump Suction Leg (Unbroken Loop)

TABLE 4

STAFF RESULTS

DOUBLE-ENDED COLD LEG BREAK

Break Area (Ft^2)	8.5
Discharge Coefficient	1.0
Peak Clad Temperature (°F)	1956.0
Peak Clad Temperature, time (Sec.)	30.6
End of Bypass (Sec.)	21.72
Bottom of Core Recovery (Sec.)	29.2
Adiabatic Period (Sec)	7.48
Average Reflood Rate (in/sec.)	2.0

APPENDIX D

EVALUATION OF THE B&W MOMENTUM EQUATION

The Babcock and Wilcox documentation (BAW-10092) on the development and use of the momentum equation has been reviewed. Equation 2-12 of the reference document is the specific equation evaluated. The basic analytical approach is a stream-tube analysis, with fluid motion as one-dimensional homogenous flow. The implementation of the momentum equation is based on the node and branch method, where the time-derivative of branch flow is the variable solved. A term-by-term review of the equation is presented as:

(1) TEMPORAL CHANGE OF MOMENTUM

The temporal term is presented as the time-derivative of branch flow. The branch can consist of several geometries. For geometries such as sudden expansions and for compressible fluids, spatial variations of flow can be expected. The time-derivative of flow for the branch is established on the approximation of uniform flow along the flow path.

The time-derivative of branch flow in the node and branch implementation is evaluated from nodal properties which are spatially displaced from the branch. Thus, the integration of nodal mass and energy equations, and the branch momentum equation are spatially inconsistent. However, it is characteristic of the node and branch technique.

At worst, the inconsistency in the time domain can be bounded as the transport lag from the center of the node to the center of the branch. This in turn depends on the noding geometry and the velocity of the fluid. Typically, the magnitude of the transport lag is small, and thus on this basis, the temporal term is acceptable.

$$W^2 \int_{X_1}^{X_2} \frac{1}{\rho} \cdot \frac{1}{A} \frac{\partial}{\partial x} \left(\frac{1}{A}\right) = \frac{W^2}{2\rho_{fp}} \left(\frac{1}{A_2^2} - \frac{1}{A_1^2}\right)$$

The scope and range of applicability of these assumptions must be established. A possible means of doing this is to compare results of the above technique with those from subcritical two-phase nozzle flow. Pressure ratio and area ratios would have to be chosen to encompass LOCA expected values for a Babcock and Wilcox plant. The result of a sensitivity study are required before a judgement on acceptability can be made.

The partial of density with respect to distance is integrated with area as a constant. The area used is the line-averaged flow path area. This is a reasonable engineering approach since the spatial variation of flow path area is known. The integration results in momentum flux terms which account for compressibility.

The use of homogeneous density is valid at high mass velocity ($G>2x10^6$). For low mass velocities, Weisman's studies[11] have concluded that slip flow modeling gave better correlation with experimental data than homogeneous flow modeling. The use of homogeneous density is currently acceptable to the staff.

(5) PRESSURE LOSS RESULTING FROM WALL FRICTION

The frictional pressure drop term is developed in standard form and utilizes average flow path area and average flow path density in establishing velocity heads. The friction factor is a function of Reynolds number, and when the flow is two-phase, the friction factor is further modified by a two-phase multiplier. The effective length is evaluated from steady state

(7) GRAVITATIONAL ACCELERATION

The gravitational terms of the momentum equation were also reviewed. The static pressure within a node varied with vertical height based on the volume of steam and the volume of froth present. The static pressure as a branch inlet or exit was evaluated as the static pressure within the node at the branch heights. The contribution of gravitational terms associated with a branch was accounted for by using the flow path density. The gravitational terms within the momentum equation are acceptable.