EGG-EAST-8695

TECHNICAL EVALUATION REPORT
RELAP5/MOD2 - B&W
AN ADVANCED COMPUTER PROGRAM
FOR
LIGHT WATER REACTOR LOCA AND NON-LOCA TRANSIENT ANALYSES

C. P. Fineman

February 1990

Idaho National Engineering Laboratory
EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

Prepared for the
U.S. Nuclear Regulatory Commission
Washington D.C. 20555
Under DOE Contract No. DE-AC07-76ID01570
FIN No. D6030

ABSTRACT

A review was completed of the RELAP5/MOD2 - B&W computer program developed by Babcock & Wilcox (B&W) for performing pressurized water reactor (PWR) loss-of-coolant accident (LOCA) licensing analyses. The review consisted of an evaluation of the RELAP5/MOD2 - B&W computer program as well as the modifications made by B&W to the RELAP5/MOD2, cycle 36.04, computer program from which the licensing version of RELAP5/MOD2 - B&W originated. Integral assessment calculations were reviewed to evaluate the validity and proper implementation of the modifications and added models. The review found that RELAP5/MOD2 - B&W met the requirements of 10CFR50, Appendix K, and NUREG-0737, Item II.K.3.30, and it is recommended the code be accepted for use in PWR licensing analyses on the basis that suggested conditions and requirements are followed. The break sizes for which RELAP5/MOD2 - B&W is recommended for use include both large and small breaks. However, prior to using RELAP5/MOD2 - B&W in LOCA applications, B&W must receive approval of their LOCA evaluation model methodology being reviewed separately.

SUMMARY

This report documents the review and evaluation of the Babcock & Wilcox (B&W) computer program RELAP5/MOD2 - B&W for use in performing pressurized water reactor (PWR) loss-of-coolant accident (LOCA) licensing analyses. RELAP5/MOD2 - B&W is capable of analyzing the steady state and transient thermal-hydraulic response of a light-water reactor. The code has features that allow compliance with the requirements of 10CFR50, Appendix K. RELAP5/MOD2 - B&W is based on RELAP5/MOD2, cycle 36.04, developed at the Idaho National Engineering Laboratory (INEL) under Nuclear Regulatory Commission (NRC) sponsorship. The code was submitted to the Office of Nuclear Reactor Regulation for approval as a licensing method. The Office of Nuclear Reactor Regulation requested assistance from the INEL in reviewing RELAP5/MOD2 - B&W.

The RELAP5/MOD2 - B&W code was reviewed and assessed as well as the model additions and modifications made by B&W to RELAP5/MOD2, cycle 36.04. This review was made using the information provided by B&W in the RELAP5/MOD2 - B&W code manual and in B&W's responses to questions submitted by the NRC to B&W. The code was also reviewed to ensure that known updates and corrections to RELAP5/MOD2, cycle 36.04, were included in RELAP5/MOD2 - B&W by B&W or appropriate justification given if excluded. Finally, the code was reviewed for compliance with NRC requirements.

Based on this review, it is recommended that RELAP5/MOD2 - B&W be accepted for performing PWR licensing analyses provided suggested conditions and requirements are followed. The break sizes for which RELAP5/MOD2 - B&W is recommended for use include both large and small breaks. This is based on the adequate RELAP5/MOD2 - B&W analyses of LOFT Test L3-5, Semiscale Test S-LH-1, and Semiscale Test S-04-6. These represent large and small break experiments. However, prior to using RELAP5/MOD2 - B&W in LOCA applications, B&W must receive approval of their LOCA evaluation model methodology being reviewed separately.

PREFACE

This report was prepared for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, by EG&G Idaho, Inc., Energy and Systems Technology Group.

CONTENTS

ABSTRACT ii

SUMMARY iii

PREFACE iv

1. INTRODUCTION 1

2. RELAP5/MOD2 - B&W CODE DESCRIPTION AND ASSESSMENT 3

2.1 General Code Overview 3

2.2 Model Description and Assessment 5

2.2.1 Critical Flow Models 5
2.2.2 Core Heat Transfer Models 6

2.2.2.1 Pre-CHF Correlations 6
2.2.2.2 CHF Correlations 7
2.2.2.3 Post-CHF Correlations 8

2.2.3 Heat Transfer Logic Options 10
2.2.4 Fuel Behavior Models 11

2.3 Assessment Calculations 13

2.3.1 LOFT Test L3-5 13
2.3.2 Semiscale Test S-LH-1 15
2.3.3 Semiscale Test S-04-6 18
2.3.4 Assessment Summary 21

2.4 Phenomena Important to PWR SBLOCA 21

2.4.1 Noncondensible Gas 22
2.4.2 Condensation Heat Transfer 25
2.4.3 Natural Circulation 28

2.5 Calculation of S-UT-8 Phenomena 29

2.6 Loop Seal Clearing Phenomena 29

3. COMPLIANCE WITH NRC REQUIREMENTS 31

4. RELAP5/MOD2 CODE UPDATES 33

5. CONCLUSIONS 34

6. REFERENCES 39

This page intentionally left blank.

TECHNICAL EVALUATION REPORT
RELAP5/MOD2 - B&W
AN ADVANCED COMPUTER PROGRAM
FOR
LIGHT WATER REACTOR LOCA AND NON-LOCA TRANSIENT ANALYSES

1. INTRODUCTION

RELAP5/MOD2 - B&W[1] is a computer program developed by Babcock & Wilcox (B&W) for light water reactor (LWR) system thermal-hydraulic analysis. It provides integral analysis capability of the system and core response to normal and off-normal events during steady state and transients. RELAP5/MOD2 - B&W was adapted from RELAP5/MOD2, cycle 36.04,[2] by B&W for use in recirculating steam generator (RSG) pressurized water reactor (PWR) loss-of-coolant accident (LOCA) analyses, both large break LOCAs (LBLOCAs) and small break LOCAs (SBLOCAs). RELAP5/MOD2 - B&W was submitted to the Nuclear Regulatory Commission (NRC) by B&W for review and acceptance for licensing applications as a method to analyze PWR LOCAs in a manner that conforms to NRC requirements contained in 10CFR50, Appendix K, and other pertinent NRC regulations. RELAP5/MOD2 - B&W will be used by B&W as part of an overall RSG LOCA Evaluation Model (EM) methodology[3] being reviewed separately. Application of RELAP5/MOD2 - B&W to LOCA analyses is contingent on the approval of the RSG LOCA EM methodology. RELAP5/MOD2 - B&W is also being reviewed separately for use in analyzing non-LOCA transients; however, the review documented in this report focused on the code's capability to analyze LBLOCAs and SBLOCAs.

The Office of Nuclear Reactor Regulation (NRR) is responsible for the evaluation and review of computer codes and their proposed applications. The Office of Nuclear Reactor Regulation requested the Idaho National Engineering Laboratory (INEL) provide assistance in the review of the RELAP5/MOD2 - B&W computer code. Specifically, the request for assistance included:

1. Evaluation of RELAP5/MOD2 - B&W as a method to analyze the entire PWR break spectrum.

2. Evaluation of RELAP5/MOD2 - B&W for compliance with requirements contained in 10CFR50, Appendix K, and NUREG-0737, Item II.K.3.30.

3. Assurance that corrections to RELAP5/MOD2, cycle 36.04, the base code for RELAP5/MOD2 - B&W were incorporated into RELAP5/MOD2 - B&W.

Related to the above reviews, NRR also requested INEL review and evaluate B&W's responses to NRC questions regarding the loss-of-coolant modeling applications. The questions were those transmitted to B&W by the NRC in References 4 and 5. Babcock & Wilcox's responses to these questions are contained in References 6 to 10.

This technical evaluation report documents the results of the RELAP5/MOD2 - B&W review and assessment for PWR LOCA analyses. Section 2 provides a brief overview of the history of RELAP5/MOD2 - B&W and its development from RELAP5/MOD2, cycle 36.04. This section also discusses B&W's modifications to the RELAP5 base code and B&W's code assessment work. Section 3 reviews the code for compliance with NRC requirements defined in Appendix K to 10CFR50. Section 4 reviews the implementation status in RELAP5/MOD2 - B&W of all documented updates to RELAP5/MOD2, cycle 36.04, generated by the code developers at the INEL. Section 5 summarizes the conclusions reached from this review and the references are listed in Section 6.

2. RELAP5/MOD2 - B&W CODE DESCRIPTION AND ASSESSMENT

This section presents a brief description of RELAP5/MOD2 - B&W and discusses its relationship to the RELAP5/MOD2, cycle 36.04, code from which it was developed. Then, each of the modifications made by B&W are discussed. The results of RELAP5/MOD2 - B&W integral assessment calculations will then be reviewed followed by a discussion of the code's ability to simulate the phenomena important to PWR SBLOCAs. Finally, the code's ability to simulate the type of phenomena observed in Semiscale Test S-UT-8[11] and loop seal clearing is discussed.

2.1 General Code Overview

RELAP5/MOD2 - B&W is a LWR reactor system transient simulation code based on a nonequilibrium and nonhomogeneous model for two-phase conditions. A full six equation, two fluid model for the vapor-liquid flow field is used. The code formulation of the hydrodynamic components, power sources, heated structures, trips, and control systems provides a flexible method for modeling LWR systems. RELAP5/MOD2 - B&W includes many general component models from which general systems can be simulated. The component models include pumps, valves, pipes, heat structures, reactor point kinetics, accumulators, and control system components. Special process models are included to account for form losses, abrupt area changes, branches, choked flow, boron tracking, and noncondensible gases.

RELAP5/MOD2 - B&W is based on RELAP5/MOD2, cycle 36.04, which was developed under NRC sponsorship. Babcock & Wilcox decided to maintain the same modeling philosophy in RELAP5/MOD2 - B&W as in the base code. The same formulation of the differential equations for the thermal-hydraulic models, the same basic constitutive relations, code architecture, principle solution techniques, and user convenient features were retained in RELAP5/MOD2 - B&W.

Although RELAP5/MOD2 was specifically developed to include the capability to simulate SBLOCAs and operational transients, the code is fully capable of simulating LBLOCAs as well. The application of RELAP5/MOD2 to a wide variety of thermal-hydraulic problems, including

large and small break LOCAs, indicated the formulation of the differential equations and solution techniques provide numerically stable solutions. With RELAP5/MOD2 - B&W maintaining the same basic approach, it can be concluded that it will also provide numerically stable solutions and, therefore, the general code models and structure are adequate for PWR LOCA analyses.

Babcock & Wilcox's assessment of RELAP5/MOD2 - B&W identified several areas that required changes to RELAP5/MOD2 to meet the EM requirements of Appendix K to 10CFR50. As a result, B&W began a code development program in the following areas:

1. Addition of the Moody, extended Henry-Fauske, and Murdock-Bauman critical flow models.

2. Addition of a core heat transfer model.

3. Addition of logic to prevent a return to nucleate boiling and transition boiling.

4. Addition of new fuel rod behavior models to represent fuel rod fission gases, rod cladding swell and rupture and rupture flow blockage, gap conductance, and zircaloy-water reaction.

The modifications made by B&W and their assessment are presented in more detail in Section 2.2.

One model added by B&W was not reviewed for use in a licensing calculation. This was the B&W auxiliary feedwater model for once through steam generators. Because it is B&W's intent to apply RELAP5/MOD2 - B&W to recirculating steam generator plants, this model was not reviewed at this time.

Also, the built-in reactor kinetics data for decay heat calculations in RELAP5/MOD2 - B&W are based on the American Nuclear Society (ANS) 1973 and 1979 standards, but the user has the option of entering different data. Because Appendix K requires the use of 1.2 times the ANS 1971

standard for decay heat calculations in LOCA licensing analyses, B&W should ensure the decay heat used in licensing analyses complies with Appendix K.

2.2 Model Description and Assessment

2.2.1 Critical Flow Models

RELAP5/MOD2 - B&W includes the original Ransom-Trapp critical flow model found in the RELAP5/MOD2 base code, and B&W added the Moody,[12] Extended Henry-Fauske,[13] and Murdock-Bauman[14] critical flow models to RELAP5/MOD2 - B&W. The Extended Henry-Fauske (H-F) model can be used to calculate subcooled critical flow, the Moody model calculates two-phase break flow, and the Murdock-Bauman model calculates superheated steam critical flow. Addition of the Moody model allows the code user to meet the Appendix K requirement that two-phase critical flow during a LOCA be calculated with the Moody model at all break locations.

All the models were incorporated as a tables of mass flux versus stagnation enthalpy and pressure. The critical flow tables may be entered with stagnation properties or the static pressure and enthalpy from the donor cell. If the stagnation properties are chosen, they are calculated from the upstream volume static pressure and enthalpy assuming isentropic flow.

To convert the critical mass flux from the critical flow table to the phasic velocities needed by RELAP5/MOD2 - B&W, several options are available. These include a homogeneous (no slip) model, user input constant slip, Moody's slip model, RELAP5 momentum equation slip, and upstream volume equilibrium quality slip.

Assessment on the implementation of these models was provided by the integral assessment calculation for Semiscale large break Test S-04-6 and small break experiments Semiscale Test S-LH-1 and LOFT Test L3-5. In the EM calculations of these tests, subcooled, two-phase, and superheated steam critical flow were calculated using the extended Henry-Fauske, Moody, and Murdock-Bauman models. The options chosen for calculating the critical flow in the LBLOCA assessment were the use of the static properties and

Moody slip was used with the Moody critical flow model. For SBLOCAs, the options chosen for calculating the critical flow were the use of static properties and a slip ratio of one was used with the Moody critical flow model. The calculated results for Test S-04-6 show the extended Henry-Fauske model slightly overpredicted the subcooled discharge and the Moody model, as expected, overpredicted the two-phase discharge. Both the LBLOCA and SBLOCA results indicate the models using these options were implemented correctly. Thus, the critical flow models in RELAP5/MOD2 - B&W discussed above with the listed options for LBLOCA and SBLOCA calculations, respectively, are recommended for use in LOCA licensing calculations.

2.2.2 Core Heat Transfer Models

Babcock & Wilcox added a new core heat transfer package to RELAP5/MOD2 - B&W for use in licensing calculations. The package is designed to meet the evaluation model requirements of Appendix K to 10CFR50 for large and small break analyses. The correlations used in the core model will be discussed in a sequence following the boiling curve: pre-critical heat flux (pre-CHF) correlations, CHF correlations, and post-CHF correlations. Application of the core model in the assessment calculations showed it was adequate to calculate the core heat transfer response as part of the system calculation in the overall RSG LOCA EM.

2.2.2.1 Pre-CHF Correlations. The pre-CHF correlations are divided into three categories: single-phase liquid, subcooled nucleate boiling, and nucleate boiling correlations. The correlations will be discussed in that order.

The Dittus-Boelter[15] and Rohsenow-Choi[16,17] correlations are used to calculate the single-phase liquid wall to fluid heat transfer in the core. The core heat transfer logic chooses the maximum of the two correlations as the single-phase heat transfer coefficient. This is the same type of logic used in the system heat transfer model (and the RELAP5/MOD2 base code), where the maximum of the Dittus-Boelter, Rohsenow-Choi, and a natural convection correlation is used for the single-phase heat transfer coefficient. The Dittus-Boelter and Rohsenow-Choi correlations are widely used to model single-phase forced

convection heat transfer in reactor transient simulation codes. The use of these correlations in a number of different reactor transient simulation codes has proven them adequate for modeling this type of heat transfer.

Subcooled nucleate boiling heat transfer is modeled using the modified Chen correlation.[18] This is the same correlation used in the system model and the RELAP5 base code. Using the modified Chen correlation to calculate subcooled nucleate boiling in the application of RELAP5/MOD2 and other reactor analysis codes to a variety of problems has proven adequate.

A combination of the Chen,[19] Thom,[20] and Shrock-Grossman[21] correlations is used to calculate the nucleate boiling heat transfer coefficient. The ranges where the correlations are applied are: Chen correlation, pressure less than 750 psia; the Thom correlation, pressure greater than 1000 psia and void fraction less than 0.8; and the Shrock-Grossman correlation, pressure greater than 1000 psia and void fraction greater than 0.9. Linear interpolation is used to connect the correlations. The nucleate boiling correlations and selection logic used in RELAP5/MOD2 - B&W are similar to that used in the THETA1-B[22] code previously approved by the NRC,[23] except the Chen correlation replaced the Jens and Lottes correlation.[24] Also, for qualities above 0.95, the heat flux is a linear interpolation on quality between the nucleate boiling flux and the single-phase vapor flux. The nucleate boiling heat transfer correlations and logic are adequate for use in RELAP5/MOD2 - B&W.

2.2.2.2 <u>CHF Correlations</u>. The CHF correlations used in RELAP5/MOD2 - B&W are divided into high and low flow categories. The high flow correlations and selection logic will be discussed first followed by the low flow correlations and logic.

The high flow correlations and selection logic, with the exception of one high pressure CHF correlation, are similar to the logic in the previously approved THETA1-B code. The high flow correlations are used if the mass flux is greater than 500,000 lbm/h-ft^2. For pressures greater than 1500 psia, the user has the option of choosing one of three correlations developed by B&W for use with their fuel. These are the B&W-2[25] (B&W 15 x 15 fuel bundles), BWC[26] (17 x 17 and zirconium grid

15 x 15 fuel bundles), and BWCMV[27] (17 x 17 mixing vane fuel bundles). The B&W-2 correlation is listed in Appendix K and the BWC and BWCMV correlations were approved by the NRC.[28,29] In the topical report defining B&W's RSG LOCA EM (Reference 3), B&W stated that only the BWCMV correlation will be used in licensing calculations. At pressures below 1300 psia, the Barnett[30] or the modified Barnett (Hughes)[31] CHF correlations are used in the high flow regime. These correlations are listed in Appendix K. Linear interpolation is used to connect the various regions where the individual correlations are applied. The high flow CHF model is considered adequate based on the previous NRC approval of the correlations and logic.

The low flow correlations are used if the mass flux is less than 200,000 lbm/h-ft^2. The high and low flow regions are connected by linear interpolation on mass flux between the high flow CHF correlation evaluated at its low flow limit and the low flow CHF correlation evaluated at its high flow limit. The low flow CHF value is taken to be the maximum of the MacBeth[32] and the Griffith[33] correlations. These correlations and the selection logic are similar to that in the previously approved THETA1-B code. Therefore, the low flow correlations and logic are considered adequate for calculating low flow CHF.

In all regions, there are two additional constraints applied to the CHF calculation. First, a minimum CHF of 90,000 Btu/h-ft^2 is used. The same minimum CHF value is used in THETA1-B. Second, for void fractions greater than 0.8, the value for CHF is taken as the value of the heat flux calculated using the transition boiling heat transfer coefficient. For the McDonough, Milich, and King transition boiling correlation[34] used in RELAP5/MOD2 - B&W, this results in the calculation of a lower CHF value than the use of the CHF correlation alone. Based on the previous NRC approval of the minimum CHF constraint and the fact that the second constraint results in a conservative CHF calculation relative to using only the CHF correlation, these constraints are recommended for acceptance.

2.2.2.3 <u>Post-CHF Correlations</u>. The post-CHF region is divided into transition boiling, film boiling and single-phase vapor plus radiation regimes.

As noted above, the transition boiling heat transfer coefficient is calculated using the McDonough, Milich, and King correlation. This is one of the correlations specifically listed in Appendix K as acceptable for calculating transition boiling heat transfer.

The code user has the option of calculating the film boiling heat transfer coefficient using the maximum of the Condie-Bengston correlation[35] and the Rohsenow-Choi correlation or the maximum of the CSO correlation[36-40] and the Rohsenow-Choi correlation. The maximum of the two correlations is used because single-phase forced convection heat transfer, modeled with the Rohsenow-Choi correlation, provides a lower bound on film boiling heat transfer. A comparison of the Condie-Bengston correlation to rod bundle film boiling data in Reference 35 was reviewed to determine the adequacy of the correlation for calculating film boiling heat transfer. The comparison showed the Condie-Bengston correlation tended to to be slightly conservative in calculating film boiling data. Therefore, the Condie-Bengston correlation is adequate for film boiling heat transfer. In Reference 7, B&W stated that only the combination of Condie-Bengston and Rohsenow-Choi would be used in EM calculations, and, therefore no assessment of the CSO correlation would be provided. Based on this discussion, the CSO correlation cannot be used in a licensing calculation without additional review and approval by the NRC.

For heat transfer to single-phase vapor, the maximum of the McEligot[41] and the Rohsenow-Choi correlations is used to calculate the convection part of the heat flux. Babcock & Wilcox provided additional information on the assessment of the McEligot correlation in their response to question 8, Reference 7. For Reynolds numbers greater than 10,000, data in Reference 42 verified the adequacy of the McEligot correlation. For Reynolds numbers below 10,000, a comparison of the Dittus-Boelter correlation to data in Reference 35 was discussed by B&W. This comparison showed the Dittus-Boelter correlation adequately calculated the single-phase vapor data down to Reynolds numbers of 2000. Because the McEligot correlation will calculate lower heat transfer coefficients than the Dittus-Boelter correlation, the data in Reference 35 support the adequacy of the McEligot correlation down to Reynolds numbers of 2000. Additional data supporting the use of the McEligot correlation for Reynolds

numbers less than 2000 was provided in by B&W in Reference 71. This data showed the McEligot correlation was adequate to calculate the heat transfer coefficient down to Reynolds numbers of 700 to 800 where the Rohsenow-Choi would then be applied. The radiation heat flux portion of single-phase vapor heat transfer is calculated using the correlation developed by Sun,[43] which is the same model used in the RELAP5/MOD2 base code, and is considered adequate.

The heat transfer logic in the post-CHF region selects the heat flux to be the maximum of the transition boiling and film boiling heat fluxes (this is similar to the RELAP5/MOD2 base code) as long as the difference between the wall temperature and the saturation temperature is less than 300°F and the quality is less than 0.95. However, if the transition boiling heat flux is greater than the pre-CHF heat flux, evaluated at the same conditions, then the heat flux is set equal to the film boiling heat flux. Between qualities of 0.95 and 0.999, the heat flux is a linear interpolation on quality between the single-phase vapor and the maximum of the transition or film boiling heat fluxes. When the quality is above 0.999, the heat flux is set equal to the single-phase vapor heat flux. If the wall superheat is greater than 300°F, the same logic is applied except only the film boiling and single-phase vapor correlations are used.

2.2.3 Heat Transfer Logic Options

Appendix K to 10CFR50 requires a code lockout return to nucleate boiling heat transfer once CHF is predicted at an axial fuel rod location during blowdown. Appendix K also requires a lockout regarding the return to transition boiling once the cladding superheat exceeds 300°F during the blowdown phase of a LOCA. These requirements force a degraded heat transfer calculation during the blowdown phase of a LOCA even though local conditions may allow a rewet and a return to nucleate boiling. Babcock & Wilcox added the appropriate heat transfer logic to RELAP5/MOD2 - B&W to allow the user to meet these Appendix K requirements.

The heat transfer logic sets the appropriate flag to true once CHF is exceeded. With this flag set to true, only post-CHF heat transfer correlations are evaluated and used to calculate the heat transfer with one

exception. If the wall temperature is less than the saturation temperature, then single-phase liquid convection or two-phase condensation is allowed by the code. When the wall superheat is greater than 300°F, another flag is set to true, and with this flag true, the code only applies film boiling or convection to steam with the same exception noted above for the nucleate boiling lockout. The exception to the nucleate and transition boiling lockouts discussed above is considered acceptable during the blowdown because, with the wall temperature less than the saturation temperature, the use of the single-phase liquid or two-phase condensation correlations will allow the fluid to heat the cladding.

It should also be noted that B&W does not lockout nucleate boiling in the case where the heat transfer regime changes from nucleate boiling to steam cooling without exceeding CHF. However, should this change in heat transfer result in the wall superheat exceeding 300°F, then return to transition boiling is locked out by the code. The INEL agrees that this logic is in agreement with Appendix K.

2.2.4 Fuel Behavior Models

Fuel behavior models were added to RELAP5/MOD2 - B&W to enable the fuel rod behavior to be calculated during a LOCA. These models include a fuel rod cladding swell and rupture deformation model, a fuel rod gap conductance model, and zircaloy-water reaction. The cladding swell and rupture deformation models are based on NUREG-0630,[44] which is an acceptable data source.

The gap conductance model in the RELAP5/MOD2 - B&W core model is similar to the model used in FRAP-T6-B&W[45] and approved for use in B&W licensing applications in Reference 46. The only difference between the RELAP5/MOD2 - B&W and FRAP-T6-B&W models is that RELAP5/MOD2 - B&W uses a simplified form of the FRAP-T6-B&W gap width model. The RELAP5/MOD2 - B&W model calculates a gap width different from FRAP-T6-B&W only when non-concentric rather than concentric fuel pellets are modeled. In a telephone conversation on November 8, 1989, B&W stated the fuel pellet model chosen for use in RELAP5/MOD2 - B&W depends on which model most accurately matches the fuel stored energy calculated by the steady state

fuel code. In addition, B&W stated that comparing the results of two RELAP5/MOD2 - B&W blowdown calculations that used the different fuel pellet models showed the end-of-blowdown conditions were essentially the same (fuel temperatures within about 2°F). Based on this information, the simplified model is considered adequate.

The zircaloy-water reaction model is based on the Baker-Just model[47] as required by Appendix K.

All these models are described in detail in Section 2.3.2 of the RELAP5/MOD2 - B&W manual.

RELAP5/MOD2 - B&W does not account for prerupture cladding swell in the thermal-hydraulic core flow calculation. Only if the cladding ruptures are the axial friction loss factors adjusted to account for the added flow resistance due to rupture. With respect to rupture, the rods in the hot assembly and the average core are treated the same. Babcock & Wilcox discussed its position on prerupture cladding swell in their response to Question 9, Reference 48, on the RSG LOCA EM. They stated the prerupture flow blockage is small (less than 20%) and results in a smooth area change that will not significantly add to the flow resistance. Babcock and Wilcox referenced data by Hardy[49] to justify this position. They stated any swelling during blowdown would be located near the high power elevations, and any flow diversion that might occur would be quickly recovered. The INEL reviewed additional data on zircaloy cladding behavior during ballooning and rupture in References 50 and 51. This data supports B&W's argument with respect to cladding behavior. This data showed prerupture cladding swell is limited because once cladding swell exceeded 10 to 20% the cladding rapidly expanded to the point of rupture. However, B&W did not provide analyses, test data, or reference material to support their argument that flow diversion effects for cladding swell of 20% or less are minimal. This must be resolved before neglect of prerupture cladding swell can be accepted. The question on flow diversion is also being considered in the INEL review of the RSG LOCA EM. Therefore, resolution of the concern about flow diversion effects for RELAP5/MOD2 - B&W, will be completed as part of the LOCA EM review. However, should B&W's response to the flow diversion question for the RSG LOCA EM review prove inadequate to

resolve this concern, RELAP5/MOD2 - B&W code changes may be required. Also, B&W would need to justify the acceptability of any licensing analyses where cladding swell exceeded 20% but rupture was not calculated.

2.3 Assessment Calculations

RELAP5/MOD2 - B&W was assessed against a LOFT small break experiment, Test L3-5,[52] a Semiscale small break experiment, Test S-LH-1,[53] and a Semiscale large break experiment, Test S-04-6.[54] The LOFT small break assessment calculation will be discussed first followed by Semiscale Tests S-LH-1 and S-04-6.

2.3.1 LOFT Small Break L3-5

LOFT L3-5 was a 2.5% small break LOCA experiment with the pumps tripped at the initiation of the experiment. The break was located in the intact loop cold leg between the pumps and the vessel inlet nozzle. The calculation was run as a EM calculation. Therefore, EM models for break flow, core heat transfer, etc. were used. To try and approximate the measured system depressurization rate, a multiplier of 0.6 was used with the Moody two-phase critical flow model.

During the subcooled blowdown, the calculated system pressure decreased more slowly than the measured pressure. However, the calculated pressure decreased more rapidly than the measured pressure as the system approached saturated conditions. The two pressures are compared in Figure 3-2, Reference 9. Because the measured break flow during the subcooled blowdown is not available, it is not clear why the calculated system depressurization was slower than measured. However, B&W attributed the difference in saturated blowdown to differences in the calculated and measured break flows. The comparison of the calculated and measured break flows in Figure 3-1, Reference 9, confirms this as the calculated break flow after 60 s is higher than the data.

The calculated flow in the intact loop hot leg dropped much more rapidly than the measured flow (see Figure G.2-8, Reference 1). This was due to the pump in the RELAP5/MOD2 - B&W calculation coasting down more

rapidly than the pump in the test (Figure G.2-7, Reference 1). According to B&W, the reason for the difference in pump coastdown is not clear. In their response to question 3b, Reference 9, B&W stated the moment of inertia or the frictional torques of the pump and motor used in the analysis would need to be increased or decreased, respectively, by a factor of 4 to 5 in order to match the measured pump coastdown. The reported moment of inertia for the LOFT pump is 0.0001 that of a full-size RCP. Babcock & Wilcox stated this small a value is surprising in light of the LOFT scaling.

This difference between the calculated and measured pump coastdown is not considered critical to the overall analysis. The rest of the system parameters were adequately calculated in spite of the pump coastdown difference. Furthermore, B&W will not analyze SBLOCAs with the pumps running (see response to question 54d on the B&W LOCA EM methodology, Reference 48). In licensing SBLOCA analyses, the RCPs will coastdown early in the analysis and not play a significant part in the system response. Thus, not matching the pump coastdown in L3-5 is considered acceptable.

The flow comparison in Figure G.2-8 also shows that natural circulation was established once the pump coastdown was completed in both the calculation and the test. This occurred at approximately 35 s in the calculation and at 60 s in the test.

As shown in Figure 3-3, Reference 9, the system mass inventory is underpredicted during the entire transient. Before 60 s, this is probably due to overcalculating the break flow; but, the measured break flow during this time was not available. After 60 s, this was due to overpredicting the critical flow. The system mass distribution was adequately calculated as shown by Figures 3-4 and 3-5, Reference 9. The hot leg was calculated to drain at approximately the same time as in the experiment (Figure 3-4), and B&W stated the cold leg drained more rapidly in the calculation than in the test (Figure 3-5) due to the higher break flow.

These comparisons show that RELAP5/MOD2 - B&W was able to satisfactorily calculate the important phenomena in LOFT small break experiment L3-5.

2.3.2 Semiscale Test S-LH-1

Babcock & Wilcox assessed RELAP5/MOD2 - B&W using data from Semiscale Test S-LH-1. This test was chosen because it represented an integral test designed for PWR SBLOCAs where core uncovery/recovery was encountered; also, it was a test were steam generator liquid holdup and consequent core uncovery were observed. Two assessment calculations were completed for Test S-LH-1 using EM options, a base case calculation using single-phase and two-phase discharge coefficients of 1.0 and a sensitivity calculation where the single- and two-phase discharge coefficients were varied from 1.13 to 0.7. Other model changes were made in the sensitivity study to result in a better comparison between the calculated and measured results. These changes are discussed in more detail below. The base case calculation will be discussed first followed by the sensitivity calculation. All figures for the discussion on the base case and sensitivity calculations for Test S-LH-1 were taken from Reference 9.

For the base case calculation, RELAP5/MOD2 - B&W calculated well the U-tube level response in the intact and broken loop steam generators (Figures 5-5 to 5-8). During the loop seal clearing period, the calculated core level depression matched the test data well (Figure 5-13). As a result of using EM models, the calculated rod heatup, shown in Figures 5-19 and 5-20, exceeded the measured core heatup. The intact loop pump suction seal (Figures 5-9 and 5-10) was calculated to clear at approximately the same time as in the test. In the calculation, the broken loop pump suction seal (Figures 5-11 and 5-12) cleared at the same time as the intact loop seal, and, therefore, about 80 s before the broken loop pump suction seal in the test. Once the loop seals cleared, the fluid in the core began to boil off, and the core began to uncover in the experiment. Less core uncovery was calculated than measured as shown in Figure 5-13. Due to the calculation of a more rapid depressurization after 200 s (see Figure 5-4), the accumulator setpoint was reached at 324 s in the calculation as opposed to 500 s in the test. With the early accumulator injection, the system mass inventory (Figure 5-14) and core collapsed level began to increase in the calculation before they did in the test. However, the depth of the second core uncovery in the calculation was about the same as in the test, in spite of the early accumulator injection; therefore, the calculated peak

cladding temperature (PCT) exceeded the measured PCT by approximately 90°F due to the use of EM methods (Figures 5-19 and 5-20). The PCT was calculated using the SBLOCA EM methodology described in Reference 3.

Babcock & Wilcox addressed the faster depressurization in the base case calculation by changing the discharge coefficients in the sensitivity calculation discussed below. However, overall, the break flow rate comparison showed the break flow in the base case compared well to the measured flow.

The sensitivity calculation of Semiscale Test S-LH-1 was completed with a model that included a number of input changes. These included:

a. Different discharge coefficients were used for subcooled (1.13), two-phase (0.79), and superheated steam discharge conditions (1.13).

b. A special upper downcomer model to force the bypass of the intact loop high pressure safety injection (HPSI) to the broken loop to match the estimated break quality, vessel mass balance, and system mass balance in the test.

c. Revised heat loss modeling. The system heat loss to the environment was reduced based on core and downcomer mass and energy balances during the core boiloff phase.

d. The secondary boundary condition was revised to be based on the measured primary to secondary temperature difference rather than the measured secondary pressure.

e. Changes to several junction connections.

Babcock & Wilcox stated the special upper downcomer model (change b above) was not applicable to full size PWR calculations. The model was required to adequately calculate Test S-LH-1 because the distance between intact and broken loop cold leg nozzles is much smaller in Semiscale than in a full size PWR. This would have the effect of enhancing the

possibility of bypass in Semiscale over a full size PWR and, therefore, required special treatment. Because this geometric difference between the Semiscale system and a full size PWR is real, INEL considers this adequate justification for not applying this model to B&W full size PWR calculations. The INEL notes the other major changes (a and c) are also not directly applicable to a full size PWR. Change a will be covered by the SBLOCA break size spectrum analysis. The heat loss change is only applicable to small scale facilities, where heat loss is a problem, not full size plants.

As a result of the above model changes, the RELAP5/MOD2 - B&W calculated system pressure, core collapsed level, and system mass inventory (Figures 5-4, 5-13, and 5-14) were in much better agreement with the test data during the core boiloff. The steam generator U-tube level response during the loop-seal clearing core level depression was about the same as the base calculation and both compared well to the measured data (Figures 5-5 to 5-8). The core collapsed level during the loop-seal clearing period was calculated to be slightly higher than the base calculation and the data as shown in Figure 5-13. Babcock & Wilcox attributed this to increased draining from the hot leg into the core during the reflux cooling period in the revised calculation (Reference 55). However, the core level was depressed below the top of the core during this period. The intact loop pump suction cleared first in the test and revised calculation and at approximately at the same time (Figures 5-9 and 5-10). The broken loop loop-seal cleared much later in the revised calculation than in the test (605 s versus 262 s) and the base calculation (605 s versus 180 s) as shown in Figures 5-11 and 5-12. Babcock & Wilcox attributed this to differences in broken loop steam generator heat transfer (Reference 55). The RELAP5/MOD2 - B&W output was used by B&W in a rod heatup calculation using the SBLOCA EM methodology mentioned above. This data showed the PCT in the revised calculation was approximately 65°F higher than the measured PCT (see Figure 5-19).

Overall, there are a number of differences between the calculated and measured results presented by B&W. However, because B&W's SBLOCA methodology is based on Appendix K and not best estimate calculations and because the overall method conservatively calculated the PCT, B&W has

demonstrated that RELAP5/MOD2 - B&W was able to adequately calculate the Semiscale system response to Test S-LH-1 based on the code's intended application.

2.3.3 Semiscale Test S-04-6

Semiscale Test S-04-6, was a large break LOCA experiment (200% double-ended offset shear cold leg break) conducted in the Semiscale Mod-1 test facility. Babcock & Wilcox performed two calculations for this assessment, a best estimate, base case calculation using RELAP5/MOD2, cycle 36.04, and a calculation using the options that would be used in an EM licensing calculation. The figures for the Test S-04-6 discussion were taken from Reference 7.

The pressure near the vessel side of the break for the two calculations is compared to the test data in Figure 12.4. This comparison shows the fastest depressurization was calculated by the EM calculation. This was expected because the EM models maximize the system mass depletion, and this maximizes the depressurization rate. The break flow comparison in Figure 12.14 shows that the calculated EM break flow was in fact larger than the RELAP5/MOD2 or the measured break flows from 5 to 15 s.

RELAP5/MOD2 also calculated a faster depressurization than the test data. The biggest difference in the RELAP5/MOD2 and measured depressurization rates occurred during the first 3 to 5 s and was due to the calculated subcooled break flow being greater than the measured break flow. However, once the the calculated and the test break flow transition to two-phase conditions, the calculated break flow agrees very well with the measured break flow. As a result, the depressurization in the RELAP5/MOD2 calculation after 5 s follows the trends of the data fairly well.

The oscillations in all the break flows after 15 s in Figure 12.14 were due to slugs of subcooled emergency core cooling system (ECCS) water moving from the cold leg to the break.

The comparisons of the calculated and measured mass flow rates in the intact loop cold leg and vessel showed good agreement. However, the calculated flow in the intact loop hot leg for both the best estimate and EM calculations underpredicted the measured flow during the first 10 s. Babcock & Wilcox stated (Reference 9, question 2c) this was due to overpredicting the pump side break flow which biased the vessel outlet flow to the broken loop. This caused the intact loop hot leg flow to be underpredicted. However, the calculated and measured intact loop cold leg flows compare well because the intact loop pump drained the pressurizer more rapidly in the calculation than in the experiment.

Comparison of measured and calculated densities at the two break locations and the core inlet showed some differences. At the core inlet, B&W stated (Reference 7, question 12) the two calculated densities were higher than the measured data due to lower heat transfer. At the vessel side of the break (Figure 12.23), the density calculated by the EM analysis overpredicted the density decrease as the cold leg transitions to two-phase conditions, but the EM calculated density increased so that the calculated and measured density compared well after 5 s. The best estimate calculation underpredicted the data from 3 to 10 s. No explanation was given for this difference. At the pump side of the break, according to B&W, the two calculated densities were different than the measured density due to differences between the calculated and measured pressures near this side of the break (Reference 7).

Babcock & Wilcox also compared the calculated and measured fluid temperatures in the system. These comparisons show differences consistent with the pressure differences found in Figure 12.4. For most of the comparisons, the calculated and measured liquid and vapor temperatures follow the system saturation temperature. At the core inlet and in the upper plenum, superheated steam was calculated and measured in the test; however, the amount of superheat was underpredicted by both calculations. This difference is at least partly attributable to the higher calculated rod temperatures in both calculations as discussed below.

Comparison of the EM calculated rod temperature for the high power rod hot spot to the best estimate calculation and the test data, shows the PCT

was significantly overpredicted by the EM model (Figure 12.33). The EM calculated PCT was approximately 1480°F, the best estimate PCT was 1210°F, and the hot rod PCT in the test was 1150°F. This difference was due to the early CHF calculated by the EM model compared to the best estimate calculation and the test data.

For the average power rods, the EM calculated rod temperature for the hot spot was also higher than the best estimate calculation and the test data (Figure 12.32) but the difference was not as great as for the high power rods. This difference was also due to the early CHF calculated by the EM model compared to the best estimate calculation and the test data. This overprediction of the temperatures in the average core is not considered a problem because B&W indicated in a telephone conversation on November 16, 1989 that the limiting PCT for the plants to be analyzed occurs during reflood. In this case, the overprediction of rod temperatures at the end-of-blowdown (EOB) is conservative because it increases the amount of energy to remove from the rods during reflood. However, should the limiting PCT be calculated during blowdown or a blowdown rupture occur, then B&W should verify that overpredicting average core temperatures, through its influence on core hydraulics, does not adversely affect the hot rod PCT calculation.

The highest temperature in Figure 12.32 is the temperature measured on Rod D8-27. This temperature on an average rod exceeded the highest temperature measured on the high power rods and about equaled the peak calculated hot rod temperature in the EM analysis. The similarity between the temperature response of rod D8-27 and the EM hot rod was due to the early CHF of the rod in the experiment and the EM hot rod. For the other rod temperatures in these figures, both calculated and measured, delayed CHF resulted in greater energy removal and lower rod temperatures.

Comparison of the EM and RELAP5/MOD2, Cycle 36.04, calculations showed slightly better cooling in the EM calculation after 12 s. Babcock & Wilcox stated (Reference 9, question 2f) this difference was due to the calculation of higher reverse core flow in the EM case as compared to the Cycle 36.04 case. As a result, the EM analysis analysis calculated a higher heat transfer coefficient. The higher heat transfer coefficient

combined with the higher cladding temperatures of the EM analysis to result in a higher calculated heat flux and better cooling of the cladding relative to the Cycle 36.04 case.

In response to a question on how the PCT calculated by RELAP5/MOD2 - B&W would compare to the measured PCT if the calculated pressure more accurately matched the test data (question 2a, Reference 9), B&W provided the results of an analysis of Test S-04-6 where a discharge coefficient of 0.4 was used for two-phase critical flow. The calculated pressure for this analysis matched the test pressure much more closely than the analyses discussed above. A comparison of hot rod cladding temperatures showed the PCT for this calculation was less than the PCT for the earlier EM calculation, but it was still considerably higher than the measured hot rod PCT.

The data presented by B&W indicates RELAP5/MOD2 - B&W adequately calculates the phenomena associated with large break LOCAs. Conservative blowdown PCTs and end of blowdown temperatures were calculated with the EM model. The use of EM models resulted in the end-of-blowdown being calculated at approximately 18 s in the EM analysis versus 37 s in the test.

2.3.4 Assessment Summary

Based on the assessment results presented by B&W, RELAP5/MOD2 - B&W was able to adequately calculate the system response to both large and small break LOCAs. The comparisons also showed the code was able to calculate vapor superheat. Therefore, it is recommended the code be accepted for integral systems large and small break LOCA licensing analyses.

2.4 Phenomena Important to PWR SBLOCA

In this section, the code's ability to simulate the important phenomena during a PWR SBLOCA will be discussed. This includes its ability to represent a noncondensible gas; condensation heat transfer; and single-phase, two-phase, and reflux natural circulation.

2.4.1 Noncondensible Gas

The RELAP5/MOD2 - B&W hydrodynamic model has the capability to model the presence of a noncondensible gas along with the vapor and liquid phases of water. The noncondensible model in RELAP5/MOD2 - B&W is the same one used in RELAP5/MOD2 and discussed in Reference 2. However, B&W did not assess the noncondensible gas model in the code for the reasons discussed below.

Babcock & Wilcox stated (response to question 4b, Reference 10) that they would not include noncondensibles in their calculations because of the expected negligible effect on SBLOCAs. They stated the volume of noncondensible gases which can be trapped in the primary system during a SBLOCA is too small to impact overall system results. The following potential sources of noncondensible gases were included in B&W's analysis for breaks where the steam generators are needed for core cooling: (1) gases dissolved in the primary coolant system and pressurizer, the charging system, and the refueling water; (2) fission and fill gas in fuel rods; (3) gases generated by the radiolytic decomposition of injected water; and (4) hydrogen released from metal-water reaction. Sources excluded by B&W included the nitrogen dissolved in the accumulator water and nitrogen cover gas in the accumulators. However, these are potential sources of noncondensible gas only for the larger SBLOCAs where the steam generators become a heat source. Babcock & Wilcox considered these sources for the larger SBLOCAs. Considering the first three sources above, B&W calculated the maximum amount of noncondensible released would be 117 ft^3 at 1150 psia and 562°F. This amounts to 29 ft^3 per steam generator or 5.7% of the steam generator tube volume/steam generator (assuming 10% tube plugging). Considering the first three sources plus hydrogen generated by 1% core wide oxidation, B&W calculated the maximum amount of noncondensible released would be 231 ft^3 at 1150 psia and 562°F. This amounts to 58 ft^3 per steam generator or 11% of the steam generator tube volume/steam generator (assuming 10% tube plugging).

Babcock & Wilcox divided SBLOCAs into two categories: those that rely on the steam generators for heat removal and those where the primary system depressurized to below the the secondary pressure. The applicable sources

of noncondensibles and their potential effects on the two SBLOCA categories are discussed below.

For smaller small breaks, noncondensible gases could have a significant effect because the system pressure is above the accumulator setpoint for a long time. For these break sizes, the steam generators must be maintained as a heat sink because energy removal through the break is insufficient to maintain the primary system pressure below the HPSI shutoff head. During the single- and two-phase natural circulation portions of this size SBLOCA, B&W included the amount of noncondensibles added to the primary system from sources 1, 2, and 3. Metal-water reaction was not included as a source because the core is covered by a single- or two-phase mixture in these modes of natural circulation. The worst case assumption for these modes of natural circulation is that all the noncondensible gases collect in the tube region. If the noncondensibles were pushed to the downflow side of the generator, the driving force for natural circulation would be reduced. However, the lower flow would cause greater heating of the fluid on the hot side that would compensate for the loss of cold side driving head. The system would settle into a slightly different condition but able to remove decay heat because the degradation in steam generator heat transfer in this situation is small.

For the reflux mode of cooling, B&W referred to Semiscale tests[56] and single-tube tests[57] that studied the effects of noncondensibles on reflux cooling. According to the Semiscale tests, up to 5% of the tube volume could be occupied by noncondensibles without seriously affecting reflux cooling. Therefore, for SBLOCAs without significant metal-water reaction (i.e., only sources 1 - 3 above are included), the Semiscale tests indicate no effect on steam generator performance. If all four sources are included, B&W calculated 11% of the tube volume could be occupied by noncondensibles at 1150 psia and 562°F. With this tube volume occupied by noncondensible gases, B&W concluded the effect of noncondensible gases is a reduction in condensation heat transfer due to reduced steam generator heat transfer area. However, the net result would be to stabilize at a slightly higher primary pressure (50 psi) than for the case without noncondensible. Babcock & Wilcox noted this 50 psi increase above the steam generator control pressure of 1150 psia would not substantially

reduce the injection capabilities of the charging and safety injection (SI) systems. However, the performance characteristics of SI pumps vary widely in the plants the B&W LOCA EM will be used to analyze. Therefore, B&W should verify on a plant specific basis that neglecting the 50 psi pressure increase in the calculations will not reduce SI flow such that the PCT increases by more than 50°F. If the PCT should change by more than 50°F, additional information would be needed to justify continued neglect of noncondensibles in the analysis.

Therefore, for those SBLOCAs that rely on steam generator heat transfer for energy removal, the effect of noncondensibles are minimal and need not be considered directly in the analysis except as noted above.

For the larger breaks that depressurize to the accumulator setpoint, B&W noted that any effect of noncondensible gases present in the system would occur when the steam generators are a heat source and not relied on for heat removal. Therefore, the effects of noncondensible gases on these heat removal processes need not be considered (Reference 10, question 4b). The INEL notes this would include all possible sources of noncondensible gases (sources 1 to 4 and nitrogen dissolved in the accumulator water and nitrogen cover gas). Babcock & Wilcox (Reference 10, question 4b) considered the potential effects of releasing the accumulator nitrogen cover gas at a time when the system pressure stabilizes at a pressure just below the low pressure injection (LPI) pump dead head pressure. If the accumulators were to release the nitrogen at a critical time, increasing the system pressure and shutting off the LPI flow, the event severity could be increased. To account for this possibility, B&W noted that the charging and SI system have reached runout flow at 200 psia and that they are capable of removing all decay heat after 300 s (for a 3500 MW_t plant). Therefore, SBLOCAs that take longer than 300 s to depressurize to 200 psia do not require LPI injection to mitigate the event. For breaks that do not depressurize below 140 psia, B&W stated that accumulator gas discharge will not occur while the LPI is required for core cooling. For breaks that depressurize below 140 psia, B&W stated that there is sufficient break flow to vent the nitrogen and still remove core decay heat. Therefore, the system depressurization may be slowed but the system would not repressurize and LPI flow is unaffected.

Therefore, for those SBLOCAs that do not rely on steam generator heat transfer for energy removal, the effect of noncondensibles are minimal and need not be considered directly in the analysis.

The INEL reviewed B&W's work and agrees with its conclusion that the effect of noncondensible gases from sources 1 to 4 will be small on the class of breaks where they apply. Therefore, not including noncondensibles from these sources in RELAP5/MOD2 - B&W analyses is considered acceptable. Babcock & Wilcox noted that, for larger SBLOCAs where the steam generators become a heat source, the effects of noncondensible gases from whatever source need not be considered in the analysis. Finally, B&W showed the nitrogen cover gas in the accumulators will not impact larger SBLOCAs by increasing the system pressure enough to shut off LPI flow. Therefore, the effects of noncondensibles can be neglected, except where the noncondensible gases could cause a change in PCT greater than 50°F, in RELAP5/MOD2 - B&W SBLOCA EM analyses. However, it should be noted that the system volume occupied by noncondensible gases would increase as the system pressure and temperature decreased while the plant was cooled to shutdown cooling conditions. At 300 psia and 300°F, the noncondensible gas volumes discussed by B&W, 117 and 231 ft^3, would increase to approximately 330 and 655 ft^3, respectively. With the noncondensible occupying this much of the primary system, the effects of noncondensible gases on steam generator heat removal during the long term cooling analysis would need to be considered. As such, the assumptions regarding noncondensible gases for the short term are not applicable to the post-LOCA long term cooling analysis. Additional justification is required for the application of RELAP5/MOD2 - B&W to long term cooling analyses or the effects of noncondensible gases directly accounted for in the long term cooling analysis.

2.4.2 Condensation Heat Transfer

One of the areas of concern in modeling SBLOCAs that was identified in NUREG-0737, Item II.K.3.30, was the ability to accurately calculate condensation heat transfer rates. The need to confirm this feature of the small break model against applicable experimental data was recognized.

The condensation/vaporization models in RELAP5/MOD2 - B&W are identical to the models used in RELAP5/MOD2. These models are discussed in Section 2.1.3.4 of Reference 1. In assessing these models, B&W (Reference 10, question 4d) referred to a detailed review of the correlations and models in RELAP5/MOD2 performed by the INEL.[58] With respect to the interphase mass transfer models used to calculate condensation and vaporization, the INEL review in Reference 58 noted this was an area of active research, and the code models were an approximate representation of current understanding. While many of the models were considered ad hoc because better information was not available or because the information available is difficult to implement in the code, the INEL review in Reference 58 concluded the models were a reasonable, if incomplete, engineering approximation to interfacial heat transfer.

Babcock & Wilcox discussed (Reference 10, question 4d) RELAP5/MOD2 - B&W's ability to calculate vapor superheat. In the core, B&W noted the code's ability to calculate vapor superheat depends on both the wall heat transfer and interphase heat transfer. The wall heat transfer in the RELAP5/MOD2 - B&W core heat transfer model was reviewed in Section 2.2.2 and found acceptable. A paper by Lin et. al.,[59] was referenced by B&W to discuss the post-CHF interphase heat and mass transfer models. This paper showed RELAP5/MOD2 overpredicted the vapor superheat in high quality tests and underpredicted it in low quality tests. However, the paper also noted that some of the reported data for the low quality test may not have been accurately measured.

The assessment of RELAP5/MOD2 - B&W by B&W included an analysis of Semiscale large break Test S-04-6. This calculation (Reference 7, question 12) showed the code was able to calculate vapor superheat in the core during the last part of the blowdown. The amount of vapor superheat was less than the data, but this is at least partly attributable to the fact the calculated rod temperatures were considerably higher than the data. This means more energy remained in the rods in the calculation and less energy was available to heat the fluid.

In their response to question 4d, Reference 10, B&W noted that even if RELAP5/MOD2 - B&W does not calculate vapor superheat correctly, this will

have little impact on the calculation of the PCT. This is because in the B&W SBLOCA EM methodology only the core collapsed liquid level from RELAP5/MOD2 - B&W is used in the PCT calculation. This level is passed to FOAM2 for the calculation of mixture level and steaming rate. These parameters are then passed to FRAP-T6-B&W for the PCT calculation. In FRAP-T6-B&W, only steam cooling is allowed above the mixture level. This eliminates any cooling due to entrainment of liquid from the two-phase mixture in the core. Babcock & Wilcox provided an assessment of the overall SBLOCA EM methodology in their response to question 5, Reference 9. This assessment showed that the B&W methodology overpredicted the PCT in Semiscale small break Test S-LH-1 by approximately 65 to 90°F. Based on these considerations, the vaporization model in RELAP5/MOD2 - B&W is considered adequate.

The condensation model in RELAP5/MOD2 - B&W is also the same as the RELAP5/MOD2 base code. Developmental assessment of RELAP5/MOD2[60] was one of the references given by B&W (Reference 10, question 4d) to demonstrate the acceptability of the condensation models in RELAP5/MOD2 - B&W. These assessments using data by Bankoff[61] and Aoki[62] were reviewed and showed the condensation model provided reasonable results. Babcock & Wilcox stated their assessment work on LOFT Test L3-5 and Semiscale Tests S-04-6 and S-LH-1 also verified the adequacy of the condensation models in the code. The assessment calculation for Semiscale Test S-04-6 does not show unphysical behavior in the loop flow and cold leg pressure and temperatures during the accumulator injection period (see Figures 12.18, 12.19, 12.6, and 12.29, Reference 7). Similar observations were made by B&W for the SBLOCA assessments using Semiscale Test S-LH-1 and LOFT Test L3-5 (Figures 4.1 to 4.4, Reference 10). Therefore, the INEL considers the condensation model adequate for modeling ECC induced condensation.

With respect to the condensation that could occur on the primary side of the steam generator U-tubes, B&W (Reference 10, question 4d) referred to a study by Nithianandan, et al.,[63] that evaluated the code's ability to model surface condensation. This study used data from B&W single tube experiments and Massachusetts Institute of Technology pressurizer tests. The study concluded the surface condensation model was adequate. The study was reviewed and INEL agrees with the conclusions of the paper. This

indicates the code simulation of the condensation heat transfer in the steam generator U-tubes was adequate.

Based on the information provided by B&W, the condensation model in RELAP5/MOD2 - B&W is adequate for modeling RSG plant SBLOCAs.

2.4.3 Natural Circulation

NUREG-0737, Item II.K.3.30, also identified the need to experimentally and analytically verify the various modes of single-phase and two-phase natural circulation predicted to occur in each vendor's reactor during a SBLOCA.

To demonstrate RELAP5/MOD2 - B&W's ability to calculate natural circulation, B&W compared the calculated flow to the flow from integral experiments where the pumps were tripped. This included LOFT Test L3-5 and MIST data for single-phase natural circulation and Semiscale Test S-LH-1 for two-phase and reflux natural circulation (Reference 10, question 4c). For LOFT Test L3-5, the RELAP5/MOD2 - B&W model underpredicted the measured flow rate for the first 200 s during which single-phase natural circulation occurred. MIST data showed the code adequately calculated single-phase natural circulation.

For Semiscale Test S-LH-1, B&W noted the code was able to calculate two-phase and reflux natural circulation based on a comparison of the calculated liquid and vapor velocities in the U-tubes. In the calculation, B&W stated two-phase natural circulation began at approximately 80 s (at the end of pump coastdown) and lasted until approximately 120 s when countercurrent liquid and vapor velocities were calculated; reflux cooling began at that time. Although separate liquid and vapor velocities were not measured in the test, B&W presented test data to indicate that two-phase natural circulation and reflux cooling occurred in the test during the same time frames as the calculation. Babcock & Wilcox also concluded, based on the good system pressure and break flow comparisons during this time that the code adequately calculated the heat removal associated with two-phase natural circulation and reflux cooling; therefore, the natural circulation calculation was also good.

RELAP5/MOD2 - B&W was able to conservatively or adequately calculate single-phase natural circulation. Babcock & Wilcox also demonstrated the code calculated two-phase and reflux natural circulation under appropriate conditions and at times consistent with those where indirect evidence indicated two-phase and reflux natural circulation were occurring in Semiscale Test S-LH-1. These considerations indicate the code models are adequate to simulate natural circulation during RSG plant SBLOCAs.

2.5 Calculation of S-UT-8 Phenomena

Question 17 of Reference 4 requested B&W to validate the ability of RELAP5/MOD2 - B&W to calculate phenomena similar to that observed during Semiscale Test S-UT-8 and studied in later Semiscale experiments S-LH-1 and S-LH-2 (Reference 53). This included liquid holdup in the upside of the steam generator U-tubes and consequent core level depression. To demonstrate the codes's ability to calculate these phenomena, B&W analyzed Semiscale Test S-LH-1 as discussed in Section 2.3.2 of this report. The results of the Semiscale Test S-LH-1 assessment indicated RELAP5/MOD2 - B&W has the capability to calculate liquid holdup phenomena adequately.

2.6 Loop Seal Clearing Phenomena

The RELAP5/MOD2 interphase drag model tends to overpredict interphase drag. The RELAP5/MOD2 interphase drag model was retained unchanged in RELAP5/MOD2 - B&W. While this resulted in the adequate calculation of steam generator liquid holdup phenomena as discussed above, it is nonconservative for loop seal clearing phenomena. This is because the high drag tends to clear the loop seal of all liquid even in cases where the steam flow is not high enough to entrain the liquid and carry it out of the loop seal. As a result, core uncovery and the PCT may not be accurately calculated in cases where the code predicts the loop seal cleared of liquid when liquid should have been retained.

Babcock & Wilcox noted that RELAP5/MOD2 - B&W tended to overpredicted level swell phenomena in the core in their response to question 59, Reference 48. This discussion defined the quiet water level passed to the FOAM2 code when core uncovery is predicted. Babcock & Wilcox noted

RELAP5/MOD2 - B&W conservatively overpredicted frothing and resulted in too much liquid being carried from the core into the upper plenum, hot legs, and steam generator tube upflow. Therefore, B&W wanted to include this non-core liquid in the quiet water level passed to FOAM2. The INEL notes the same code models involved in the overprediction of frothing in the core will result in the inaccurate calculation of loop seal clearing.

A Finnish assessment of RELAP5/MOD2 against data taken in a full scale loop seal test facility showed the code calculated too much liquid being carried out of the loop seal.[64]

W. Weaver, INEL, indicated that the Finns also assessed RELAP5/MOD3 against the loop seal clearing tests with only slight improvement. The RELAP5/MOD3 interphase drag models are being modified with the intent to better calculate the loop seal clearing tests.[65]

Experimental evidence for partial loop seal clearing was observed in Semisca's small break Tests S-UT-8 and S-07-10D.[66] In addition, partial loop seal clearing, as well as, cyclic loop seal clearing followed by loop seal refill have been observed in some of the small break experiments conducted in the Japan Atomic Energy Research Institute's ROSA-IV Program Large Scale Test Facility (LSTF).[67-69]

The overprediction of drag can be addressed by both model changes and nodalization. The INEL believes that adequate nodalization of the loop seal can overcome this interphase drag model problem. Therefore, this problem will be resolved by requiring B&W to perform a loop seal nodalization sensitivity study as part of the review of the RSG LOCA EM. This study should show that the loop seal nodalization chosen by B&W for use in LOCA licensing calculations results in the highest PCT and that liquid is retained in the loop seal when steam flow conditions would result in liquid retention. Alternately, B&W could provide evidence to demonstrate RELAP5/MOD2 - B&W correctly calculates loop seal phenomena, including cases where liquid was retained in the loop seal for low steam flow conditions. However, should these approaches prove inadequate to resolve this concern, RELAP5/MOD2 - B&W code changes may be required along with assessing the code against several loop seal separate effect tests.

3. COMPLIANCE WITH NRC REQUIREMENTS

Appendix K to 10CFR50 specifies the required and acceptable features of any model used for LOCA licensing analyses. Additional NRC requirements also apply. No attempt will be made to try and address all of B&W's responses in this section, but those responses having a bearing on this licensing assessment or areas of potential concern will be addressed.

All requirements related to simulating the metal-water reaction were met. Babcock & Wilcox incorporated the Baker-Just model into RELAP5/MOD2 - B&W. The code has the capability to prevent the ruptured node from being less than 3 inches in length.

The fuel rod behavior requirements were met by incorporating fuel rod behavior models into the code. This included a dynamic gap conductance model and cladding swell and rupture models based on data from NUREG-0630.

The Moody Critical Flow model was incorporated into RELAP5/MOD2 - B&W. Therefore, it is available for the code user to meet the Appendix K requirement to calculate two-phase critical flow with this model. Appendix K also requires the break size and discharge coefficients be varied to determine the limiting break size, i.e. the break size yielding the highest PCT. The code is capable of varying both the break size and the discharge coefficient.

For LBLOCAs, Appendix K requires the ECC injected into the inlet lines or the reactor vessel during the bypass period be subtracted from the vessel inventory. In the RSG LOCA EM, B&W defined the end of bypass to be the same as the end of blowdown. During the blowdown period, when RELAP5/MOD2 - B&W is used to calculate the system response, ECC bypass is modeled by removing from the inlet annulus node the smaller of 110% of the ECC flow or 99% of all liquid residing in the inlet annulus node. Therefore, the code is capable of meeting this requirement.

To meet the CHF and core heat transfer requirements of Appendix K, B&W added a new core heat transfer algorithm. The algorithm uses Appendix K

recommended CHF and heat transfer correlations, correlations that were previously approved by the NRC, or widely used and accepted correlations.

The post-CHF heat transfer requirements of Appendix K specify return to nucleate boiling be locked out once CHF has occurred during blowdown and a return to transition boiling be locked out if cladding superheat exceeds 300°F during blowdown. Babcock & Wilcox added the appropriate logic to RELAP5/MOD2 - B&W to meet these requirements.

The centrifugal pump model in RELAP5/MOD2 - B&W is the same model used in RELAP5/MOD2. The RELAP5/MOD2 model has proven adequate for a wide variety of transients. The pump homologous curves used in the plant models were discussed in B&W's response to question 14 in Reference 48. The information provided showed the homologous curves to be used in the plant models are based on acceptable data sources.

The Appendix K requirement relating to the thermal-hydraulic interaction between the steam and the ECC water is met. This is because RELAP5/MOD2 - B&W is a two-fluid thermal-hydraulic code with the capability to treat the steam/ECC water interaction included in the basic fluid models. Analysis of LBLOCA and SBLOCA experiments by B&W and analysis of separate effects experiments referenced by B&W indicate these models adequately represent the steam/ECC water interaction.

Appendix K requires the effect of the compressed gas in the accumulator on the reflood rate be considered in LOCA licensing analyses. Because RELAP5/MOD2 - B&W is only used to calculate the blowdown portion of the LOCA, this requirement does not apply. For SBLOCAs, the code does not include models to calculate reflood heat transfer. However, the Appendix K requirement to include acceptable reflood heat transfer models only applies to computer programs used to analyze large break LOCA reflood conditions.

4. RELAP5/MOD2 CODE UPDATES

RELAP5/MOD2 - B&W was developed from the publicly released RELAP5/MOD2, cycle 36.04, code. During the development of RELAP5/MOD2 - B&W, RELAP5/MOD2 was updated to correct errors or to add model improvements. This resulted in the creation of cycles 36.05 and 36.06, where cycle 36.06 was the latest released version of the code. Approval of RELAP5/MOD2 - B&W included a review of the status of these updates.

In response to a question on the status of these updates in RELAP5/MOD2 - B&W, B&W stated (question 45, Reference 48) that all of the cycle 36.05 updates were incorporated into the code. The cycle 36.06 updates were not incorporated into RELAP5/MOD2 - B&W because those updates were specific to the CRAY version of RELAP5/MOD2; therefore, they are not applicable to RELAP5/MOD2 - B&W.

Therefore, RELAP5/MOD2 - B&W contains the error corrections from RELAP5/MOD2, cycles 36.05 and 36.06, that are applicable to the code. This ensures that known errors were corrected.

5. CONCLUSIONS

The RELAP5/MOD2 - B&W submittal by B&W was reviewed to determine the code's acceptability for use in PWR LBLOCA and SBLOCA licensing analyses. Based on this review, it is recommended the code be approved for use in LOCA licensing analyses with the following comments and restrictions:

1. Application of RELAP5/MOD2 - B&W to LOCA analyses is dependent on the approval of the B&W LOCA EM methodology being reviewed separately.

2. It is recommended that RELAP5/MOD2 - B&W be approved for use in integral systems LOCA licensing analyses. For LBLOCAs, this includes the calculation of the system blowdown response. For SBLOCAs, this includes the calculation of the system hydraulic response. The calculation of peak cladding temperature for LOCAs will be performed with BEACH[70] and FRAP-T6-B&W which were reviewed separately. It is recommended RELAP5/MOD2 - B&W be used to analyze the full spectrum of large and small breaks.

3. Babcock & Wilcox will not model noncondensible gases in its SBLOCA system analyses. They demonstrated that the effect on the overall system response from all sources of noncondensible will be negligible for the range of SBLOCAs they intend to apply the methods. However, B&W noted a 50 psi increase above the steam generator control pressure of 1150 psia could result from a worst case noncondensible release. Babcock & Wilcox also stated this would not substantially reduce the injection capabilities of the charging and SI systems. However, the performance characteristics of SI pumps vary widely in the plants the B&W LOCA EM will be used to analyze. Therefore, B&W should verify on a plant specific basis that neglecting the 50 psi pressure increase (resulting from noncondensible gas build up in the steam generator) in the calculations will not reduce SI flow such that the PCT would increase by more than 50°F. If the PCT should change by more than 50°F, additional information would be needed to justify continued neglect of noncondensibles in the analysis. Babcock & Wilcox could analyze a pressure increase less than 50 psi if the smaller increase can be justified from the plant specific analysis.

4. The four modifications reviewed in Section 2.2 of this report are recommended for acceptance for use in LOCA licensing analyses with the following limitations:

a. Babcock & Wilcox stated the CSO film boiling correlation added to the code in the core heat transfer model will not be used for licensing applications. Therefore, the CSO correlation was not reviewed and the correlation cannot be used in a licensing calculation without additional review and approval by the NRC.

b. Based on information provided by B&W and additional information reviewed by INEL, prerupture cladding swell is limited because, once cladding swell exceeds 10 to 20%, the cladding rapidly expands to the point of rupture. However, B&W did not provide analyses, test data, or reference material to support their argument that flow diversion effects for cladding swell of 20% or less are minimal. This must be resolved before neglect of prerupture cladding swell can be accepted. The question on flow diversion is also being considered in the INEL review of the RSG LOCA EM. Therefore, resolution of the concern about flow diversion effects for RELAP5/MOD2 - B&W, will be completed as part of the LOCA EM review. However, should B&W's response to the flow diversion question for the RSG LOCA EM review prove inadequate to resolve this concern, RELAP5/MOD2 - B&W code changes may be required. Also, B&W would need to justify the acceptability of any licensing analyses where cladding swell exceeded 20% but rupture was not calculated.

c. The LBLOCA assessments of the Extended Henry-Fauske and Moody critical flow models added to the code were based on the use of the static properties as input to the critical flow tables and use of Moody slip with the Moody model. The SBLOCA assessments of the same models were based on the use of the static properties as input to the critical flow tables and use of a slip ratio of one with the Moody model. These options must be used in large and small break LOCA licensing calculations, respectively, unless other options are justified in the plant specific submittal.

5. Babcock & Wilcox intends to apply RELAP5/MOD2 - B&W to a variety of Westinghouse three- and four-loop plants and Combustion Engineering (CE) three- and four-loop plants. The review found the code was not plant specific in nature. Therefore, RELAP5/MOD2 - B&W can be applied to any of the proposed Westinghouse and CE plants.

6. The RELAP5/MOD2 interphase drag model tends to overpredict interphase drag. The RELAP5/MOD2 interphase drag model was retained unchanged in RELAP5/MOD2 - B&W. This is nonconservative for loop seal clearing phenomena because the high drag tends to clear the loop seal of all liquid even in cases where the steam flow is not high enough to entrain the liquid and carry it out of the loop seal. As a result, core uncovery and the PCT may not be accurately calculated in cases where the code predicts the loop seal cleared of liquid when liquid should have been retained. This overprediction of drag can be addressed by both model changes and nodalization. The INEL believes that adequate nodalization of the loop seal can overcome the shortcomings of the interphase drag model. Therefore, this problem will be resolved by requiring B&W to perform a loop seal nodalization study as part of the review of the RSG LOCA EM. This study should show that the loop seal nodalization chosen by B&W for use in LOCA licensing calculations results in the highest PCT and that liquid is retained in the loop seal when steam flow conditions would result in liquid retention. Alternately, B&W could provide evidence to demonstrate RELAP5/MOD2 - B&W correctly calculates loop seal phenomena, including cases where liquid was retained in the loop seal for low steam flow conditions. However, should these approaches prove inadequate to resolve this concern, RELAP5/MOD2 - B&W code changes and benchmarking may be required.

7. RELAP5/MOD2 - B&W overpredicted the average core cladding temperatures in the Test S-04-6 assessment calculation. This is not considered a problem because B&W indicated in a telephone conversation on November 16, 1989 that the limiting PCT for the plants to be analyzed occurs during reflood. In this case, the overprediction of rod temperatures at the EOB is conservative because it increases the amount of energy to remove from the rods during reflood. However,

should the limiting PCT be calculated during blowdown or a blowdown rupture occur, then B&W should verify that overpredicting average core cladding temperatures, through its influence on core hydraulics, does not non-conservatively affect the hot rod PCT calculation.

8. The B&W auxiliary feedwater model for once through steam generators, added to the RELAP5/MOD2 base code by B&W, was not reviewed for use in licensing calculations. Because it is B&W's intent to apply RELAP5/MOD2 - B&W to recirculating steam generator plants, this model was not reviewed at this time.

9. The built-in reactor kinetics data for decay heat calculations in RELAP5/MOD2 - B&W are based on the ANS 1973 and 1979 standards. Because Appendix K requires the use of 1.2 times the ANS 1971 standard for decay heat calculations in LOCA licensing analyses, B&W should ensure the decay heat used in licensing analyses complies with Appendix K.

10. While B&W showed the effects of noncondensible gases would be small at primary system pressures of 1150 to 1200 psia, the system volume occupied by noncondensible gases would increase as the plant was cooled to shutdown cooling conditions. At 300 psia and 300°F, the noncondensible gas volumes discussed by B&W, 117 and 231 ft^3, would increase to approximately 330 and 655 ft^3. With this increased volume of the primary system occupied by noncondensible gases, the effects of noncondensible gases on the long term cooling analysis would need to be considered. As such, the assumptions regarding noncondensible gases for the short term are not applicable to the post-LOCA long term cooling analysis. Additional justification is required for the application of RELAP5/MOD2 - B&W to long term cooling analyses or the effects of noncondensible gases directly accounted for in the long term cooling analysis.

11. In Reference 72, B&W supplied information on an error correction for the RELAP5/MOD2 - B&W calculation of the rupture K factor. The change was reviewed and the INEL agrees with B&W on the need for the correction. With the change, the code correctly calculates the rupture K factor.

6. REFERENCES

1. RELAP5/MOD2 - B&W: An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis, BAW-10164P, Rev. 1, Babcock & Wilcox Nuclear Power Group, October 1988.

2. V. H. Ransom, et al., RELAP5/MOD2 Code Manual, NUREG/CR-4312, EGG-2396, August 1985.

3. B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants, BAW-10168P, Rev. 1, Babcock & Wilcox Nuclear Power Group, September 1989.

4. W. Hodges, NRC, ltr to J. H. Taylor, B&W, "Request for Additional Information Regarding BAW-10164P, RELAP5/MOD2 - B&W," March 31, 1988.

5. W. Hodges, NRC, ltr to J. H. Taylor, B&W, "Request for Additional Information Regarding BAW-10164P, RELAP5/MOD2 - B&W,". March 23, 1988.

6. J. H. Taylor, B&W, ltr to J. A. Norberg, NRC, "RELAP5/MOD2 - B&W Topical Report BAW-10164P," JHT/88-153, August 15, 1988.

7. J. H. Taylor, B&W, ltr to J. A. Norberg, NRC, "RELAP5/MOD2 - B&W Topical Report BAW-10164P," JHT/88-215, November 23, 1988.

8. J. H. Taylor, B&W, ltr to J. A. Norberg, NRC, "RELAP5/MOD2 - B&W Topical Report BAW-10164P," JHT/88-231, December 23, 1988.

9. J. H. Taylor, B&W, ltr to J. A. Norberg, NRC, "RELAP5/MOD2 - B&W Topical Report, BAW-10164P," JHT/89-95, May 11, 1989.

10. J. H. Taylor, B&W, ltr to J. A. Norberg, NRC, "RELAP5/MOD2 - B&W Topical Report BAW-10164P," JHT/89-149, July 20, 1989.

11. G. G. Loomis, "Semiscale Liquid Hold-Up Investigations: A Comparison of Results From Small Break LOCA Tests Performed in the Semiscale MOD-2A and MOD-2C Facilities," 13th Water Reactor Safety Information Meeting, Gaithersburg, MD, October 22-25, 1985.

12. F. J. Moody, "Maximum Flow Rate of a Single-Component, Two-Phase Mixture," ASME Paper 64-HT-35, 1964.

13. R. E. Henry and H. K. Fauske, "The Two-Phase Critical Flow of One-Component Mixtures in Nozzles, Orifices, and Short Tubes," ASME Journal of Heat Transfer, 93, Series C, No. 2, 179, May 1971.

14. J. W. Murdock and J. M. Bauman, "The Critical Flow Function for Superheated Steam, Journal of Basic Engineering, Transactions of ASME, September 1964.

15. F. W. Dittus and L. M. K. Boelter, "Heat Transfer in Automobile Radiators of the Tubular Type," Publications in Engineering, University of California, Berkeley, 2, pp. 443-461, 1930.

16. W. M. Rohsenow and H. Y. Choi, Heat, Mass, and Momentum Transfer, Prentice-Hall, Inc., 1961, p. 166.

17. J. M. Delhaye, M. Giot, and M. L. Reithmuller, Thermohydraulics of Two-Phase Systems for Industrial Design and Nuclear Engineering, Hemisphere Publishing Corporation, 1981, pp. 261-262.

18. J. G. Collier, Convection Boiling and Condensation, London: McGraw-Hill Book Company, Inc., 1972.

19. J. C. Chen, "A Correlation for Boiling Heat Transfer to Saturated Fluid in Convertive Flow," Process Design Development, 5, 1966, pp. 322-327.

20. J. R. S. Thom, et al., "Boiling in Subcooled Water During Flow Up Heated Tubes or Annuli," Proceeding of Institute of Mechanical Engineers, 3C180, 1966.

21. V. H. Schrock and L. M. Grossman, Forced Convection Boiling Studies, Final Report on Forced Convection Vaporization Project, TID-14632, 1959.

22. Babcock & Wilcox Revisions to THETA1-B, A Computer Program for Nuclear Reactor Core Thermal Analysis (IN-1445) - Revision 3, BAW-10094, Rev. 3, February 1981.

23. C. O. Thomas, NRC, ltr to J. H. Taylor, B&W, "Acceptance for Referencing of Licensing Topical Report BAW-10094, Revision 3, Babcock & Wilcox Revisions to THETA1-B, A Computer Program for Nuclear Reactor Core Thermal Analysis (IN-1445)," May 1985.

24. W. H. Jens and P. A. Lottes, Analysis of Heat Transfer, Burnout, Pressure Drop, and Density Data for High-Pressure Water, ANL-4627, May 1951.

25. Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000, March 1970.

26. R. H. Wilson, D. A. Farnsworth, and R. H. Stoudt, BWC Correlation of Critical Heat Flux, BAW-10143-A, Babcock & Wilcox, April 1985.

27. D. A. Farnsworth and G. A. Meyer, BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies, BAW-10159P, Babcock & Wilcox, May 1986.

28. C. O. Thomas, NRC, ltr to J. H. Taylor, "Acceptance for Referencing of Licensing Topical Report BAW-10143P, Parts 1 and 2," June 1, 1984.

29. A. Thadani, NRC, ltr to J. H. Taylor, "Acceptance for Referencing of Licensing Topical Report BAW-10159P, BWCMV Correlation of CHF in Mixing Vane Grid Fuel Assemblies, May 1986," February 17, 1989.

30. P. G. Barnett, A Correlation of Burnout Data for Uniformly Heated Annuli and Its Use for Predicting Burnout in Uniformly Heated Rod Bundles, AEEW-R463, September 1966.

31. E. D. Hughes, A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia, IN-1412, July 1970.

32. R. V. Macbeth, Burn-out Analysis Part 4: Application of a Local Conditions Hypothesis to World Data for Uniformly Heated Round Tubes and Rectangular Channels, AEEW-R267, August 1963.

33. C. T. Avedisian, Critical Heat Flux in Countercurrent Flow, M. S. Thesis Massachusetts Institute of Technology, 1974.

34. J. B. McDonough, W. Milich, and E. C. King, Partial Film Boiling with Water at 2000 psia in a Round Tube, Technical Report 62, MSA Research Corporation, 1958.

35. G. L. Yoder, Rod Bundle Film Boiling and Steam Cooling Data Base and Correlation Evaluation, NUREG/CR-4394, ORNL/TM-9628, August 1986.

36. J. C. Chen, R. K. Sundaram, and F. T. Ozkaynak, A Phenomenological Correlation for Post-CHF Heat Transfer, NUREG-0237, June 1977.

37. S. W. Webb and J. C. Chen, Assessment of Post-Critical Heat Flux Models with Lehigh Nonequilibrium Data, NUREG/CR-4353, April 1986.

38. S. W. Webb and J. C. Chen, "Inferring Nonequilibrium Vapor Conditions in Convective Film Boiling," Second International Topical Meeting on Nuclear Reactor Thermal-Hydraulics, Santa Barbara, California, January 11-14, 1983, Thermal-Hydraulics of Nuclear Reactors, Volume 1, pp. 326-334.

39. S. W. Webb and J. C. Chen, "A Two-Region Vapor Generation Rate Model for Convective Film Boiling," First International Workshop on Fundamental Aspects of Post-Dryout Heat Transfer, Salt Lake City, Utah, April 2-4, 1984, pp. 485-508.

40. J. C. Chen, R. K. Sundaram, and K. L. Chen, Analysis of Babcock & Wilcox Data on Film Boiling Heat Transfer, Report TS-811, Lehigh University, March 1, 1981.

41. D. M. McEligot, L. W. Ormond, and H. C. Perkins, "Internal Low Reynolds Number Turbulent and Transitional Gas Flow with Heat Transfer," Journal of Heat Transfer, 88, 1966, p. 239.

42. H. Kumamaru, et al., "Investigation of Uncovered-Bundle Heat Transfer Under High-Pressure Boil-Off Conditions," Nuclear Engineering and Design, 96, 1986, pp. 81-94.

43. K. H. Sun, J. M. Gonzalez, and C. L. Tien, "Calculation of Combined Radiation and Convection Heat Transfer in Rod Bundles Under Emergency Cooling Conditions," Journal of Heat Transfer, Transactions of ASME, 98, 1976, p. 414.

44. D. A. Powers and R. O. Meyer, Cladding Swelling Models for LOCA Analysis, NUREG-0630, April 1980.

45. J. H. Jones, FRAP-T6-B&W -- A Computer Code for the Transient Analysis of Light Water Reactor Fuel Rods, BAW-10165P, Rev. 1, October 1988.

46. A. Thadani, NRC, ltr to J. H. Taylor, B&W, "Acceptance for Referencing of Licensing Topical Report BAW-10165P, FRAP-T6-B&W - A Computer Code for the Transient Analysis of Light Water Reactor Fuel Rods," May 19, 1989.

47. L. Baker and L. C. Just, Studies of Metal-Water Reactions at High Temperatures, III. Experimental and Theoretical Studies of the Zirconium-Water Reaction, ANL-6548, May 1962.

48. J. H. Taylor, B&W, ltr to J. A. Norberg, NRC, "RSG LOCA Topical Report, BAW-10168P," JHT/89-66, March 31, 1989.

49. D. G. Hardy, "High-Temperature Rupture Behavior of Zircaloy Tubing," CONF-730304, USAEC/TIC, Water-Reactor Safety, March 1973.

50. H. M. Chung, et al., "III. Mechanical Properties of Zircaloy Containing Oxygen," Light-Water-Reactor Safety Research Program: Quarterly Progress Report, January - March 1976, ANL-76-49, pp. 34-59.

51. F. J. Erbacher and S. Leistikow, "Zircaloy Fuel Cladding Behavior in a Loss-of-Coolant Accident: A Review," ASTM STP 939, 1987, pp. 451-488.

52. L. T. L. Dao and J. M. Carpenter, Experimental Data Report for LOFT Nuclear Small Break Experiment L3-5/L3-5A, NUREG/CR-1695, EGG-2060, November 1980.

53. G. G. Loomis and J. E. Streit, Results of Semiscale Mod-2C Small Break (5%) Loss-of-Coolant Accident Experiments S-LH-1 and S-LH-2, NUREG/CR-4438, EGG-2424, November 1985.

54. H. S. Crapo, et al., Experimental Data Report for Semiscale MOD-1 Tests S-04-5 and S-04-6 (Baseline ECC Tests), TREE-NUREG-1045, January 1977.

55. Personal communication with J. Klingenfus, B&W, June 21 and 27, 1989.

56. G. G. Loomis, Summary of the Semiscale Program (1965-1986), NUREG/CR-4945, EGG-2509, July 1987, pp. 196-199.

57. D. Hein, R. Rippel, and P. Weiss, "The Distribution of Gas in a U-Tube Heat Exchanger and its Influence on the Condensation Process," International Heat Transfer Conference, Munich, Germany, September 1982.

58. R. A. Dimenna, et al., RELAP5/MOD2 Models and Correlations, NUREG/CR-5194, EGG-2581, August 1988.

59. J. C. Lin, et al., "RELAP5/MOD2 Post-CHF Heat and Mass Transfer Models," International Workshop on Fundamental Aspects of Post-Dryout Heat Transfer, Salt Lake City, Utah, April 1-4, 1984.

60. V. H. Ransom, et al., RELAP5/MOD2 Code Manual, Volume 3: Developmental Assessment Problems, EGG-TFM-7952, December 1987.

61. I. S. Lim, et al., Concurrent Steam/Water Flow in a Horizontal Channel, NUREG/CF-2289, August 1981.

62. S. Aoki, et al., "Direct Contact Condensation of Flowing Steam onto Injected Water," Sixth International Heat Transfer Conference, Toronto, Canada, August 7-11, 1978, Hemisphere Publishing Corporation, Washington, D.C., 1978, paper NRR19, pp. 107-112.

63. C. K. Nithianandan, et al., "RELAP5/MOD2 Models for Surface Condensation in the Presence of Noncondensible Gases," Eighth International Heat Transfer Conference, San Francisco, California, 1986, Hemisphere Publishing Corporation, pp. 1627-1633.

64. P. Vuorio and H. Tuomisto, "Assessment of RELAP5 and SMABRE Phase Separation Models Against Full-Scale Loop Seal Experiments," Third International Topical Meeting on Nuclear Power Plant Thermal Hydraulics and Operation, Seoul, Korea, November 14-17, 1988.

65. Private communication with W. Weaver, INEL.

66. Experimental Data Report for Semiscale MOD-3 Small Break Test S-07-10D (Baseline Test Series), NUREG/CR-1641, EGG-2065, December 1980.

67. Y. Kukita, et al., "Intentional Coolant System Depressurization: Experimental Studies in the ROSA-III and ROSA-IV Programs," CSNI Specialist Meeting on Intentional Coolant System Depressurization, Garching, Federal Republic of Germany, June 12-14, 1989.

68. K. Tasaka, et al., "The Results of 0.5% Small Break LOCA in ROSA-IV LSTF Break Location Parameter Test Series," Proceeding of the 15th Water Reactor Safety Information Meeting, Gaithersburg, MD, 1987.

69. K. Tasaka, et al., "Loop Seal Clearing and Refilling During a PWR Small Break LOCA," Proceeding of the 16th Water Reactor Safety Information Meeting, Gaithersburg, MD, October 24-27, 1988.

70. BEACH: Best Estimate Analysis Core Heat Transfer, A Computer Program for Reflood Heat Transfer During LOCA, BAW-10166P, Rev. 2, Babcock & Wilcox Nuclear Power Group, September 1989.

71. J. H. Taylor, B&W, ltr to V. Wilson, NRC, "RELAP5/MOD2 Topical Report, BAW-10164P," JHT/90-9, January 30, 1990.

72. J. H. Taylor, B&W, ltr to V. Wilson, NRC, "RELAP5/MOD2 Topical Report, BAW-10164P," JHT/90-8, January 30, 1990.

5.4 Responses to Request for Additional Information on Revision 2

This section contains questions transmitted to BWNT by R. C. Jones of the NRC in his letter of October 1993, and responses transmitted by BWNT to the NRC in a letter from J. H. Taylor (JHT/93-279) dated November 16, 1996.

BWNT Response to NRC's Request for Additional Information on BAW-10164, Revision 2, August 1992

Question 1 - Page 2.1-52.3

Please explain any differences between the interphasic friction model for slug flow (equation 2.1.3-30.7, p. 2.1-52.3) used in RELAP5/MOD2 B&W and BEACH. In particular, the multiplier on the Taylor bubble term appears to be different. Also, is smoothing of drag for the junction performed differently? Give the physical basis for any differences in the models.

Response: The interphase friction model for slug flow, given by Equation 2.1.3-30.7 on page 2.1-52.3 of the RELAP5 topical report (BAW-10164, Revision 2, August 1992) is the same as Equation 2.1.3-33 on page 2.1-34 of the currently-approved BEACH topical report (BAW-10166, Revision 3, October 1990). As stated on page 2.1-52.3 of the RELAP5, Revision 2, report, the Taylor bubble multiplier, C_{WSL}, is set equal to one. Accordingly, the multiplier was set to one in the ORNL (Appendix H) and ROSA-IV (Appendix J) benchmarks.

In Revision 4 of the BEACH topical report, a multiplier, $C_{Q_{slg}}$, was added to the Taylor bubble term in Equation 2.1.3-33 on page 2.1-35. Its value is set to 0.25 when the "NEWQUEN" option, which selects the BEACH, Revision 4, reflood options, is selected. BEACH predicted the proper void distribution below the quench front with this constant value, 0.25, for the reflood tests described in Revision 4 of the topical. On the other hand, the ORNL benchmarks show that RELAP5, with C_{WSL} set to one, predicts the proper core void distribution for SBLOCA conditions.

In RELAP5, Revision 2 (and retained in Revision 3), an option was added to the Wilson drag model to modify the base RELAP5 drag smoothing in user-specified junctions (see pages 2.1-52.3 and 2.1-52.4). Exercising the RELAP5, Revision 2, option effectively removes base RELAP5 smoothing from user-selected junctions. At present, the option is only used in SBLOCA applications.

During SBLOCA transients, when the mixture level is in the upper plenum (during the hot leg draining period and the period preceding core uncovery), RELAP5, using the Revision 2 option, predicted the expected void distribution in the upper core and plenum regions, including the discontinuous void behavior at the core-upper plenum boundary. With the base RELAP5 smoothing method, the code tended to flatten the void profile in the upper core region as shown in Figure 2.1.3-3.1 on page 2.1-52.4. When the mixture level is within the core region, RELAP5 calculated the proper void distribution with the base smoothing method. The RELAP5, Revision 2, option is recommended for use in the core-upper plenum interface junctions and in upper plenum junctions for SBLOCA applications. The option is considered part of the SBLOCA evaluation model, BAW-10168, Revision 2, Volume II, October 1992.

Question 2 - Appendix H

In Appendix H, Figure H.1, the outlet junction from the core is labeled as a BRANCH component, but no volume is associated with the component. In RELAP5 MOD2, the BRANCH component has a volume. Has this been changed in RELAP5 MOD2 B&W? Also, please explain the junction shown entering the time dependent volume representing the lower plenum.

Response: There has been no change in RELAP5/MOD2 with respect to designation of BRANCH components as volumes. Figure H.1 is unclear; a revised figure showing this component as a volume is given below. The junction entering the time-dependent volume is not in the model. This has also been corrected in the attached figure.

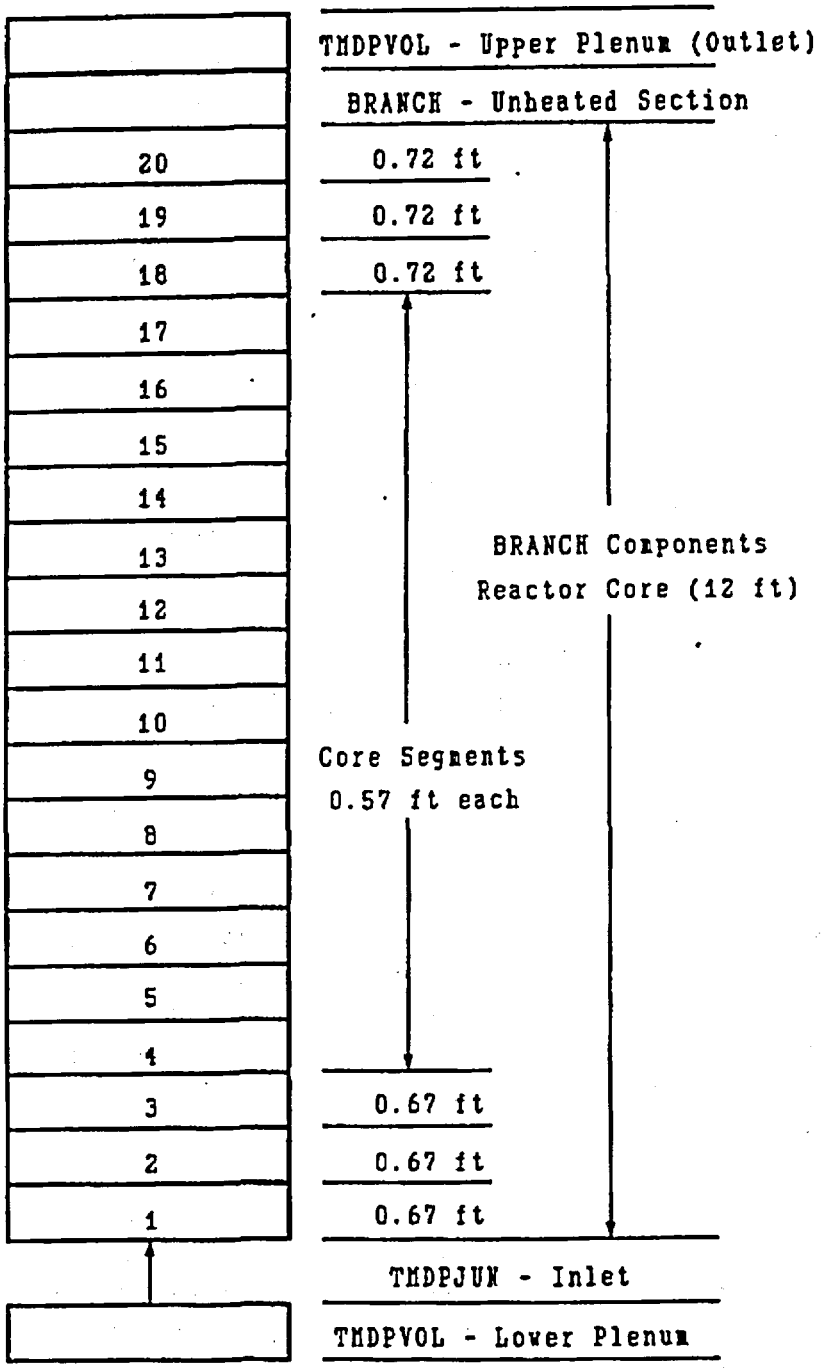


Figure H.1. RELAP5 Model of Hypothetical Reactor Core.

Question 3 - Appendix H, Figures H.28 to H.33

In Figures H.28 through H.33 the vapor temperature is consistently overpredicted by RELAP5 MOD2 B&W. Yet if it were not for the dip in surface temperature caused by the grid effect, the surface temperature would be underpredicted for experiments 3.09.10 i, j, l, and m. This would seem to indicate that the heat transfer coefficient to the vapor predicted by RELAP5 is high and non-conservative. If the vapor temperature were predicted correctly, the fuel surface temperature would be underpredicted. Please discuss the comparisons to the test data and show that these comparisons do not depend on systematically underpredicting the vapor temperature.

Response: The ORNL tests in Appendix H are quasi-steady or steady-state experiments. As such, the energy transfer to the fluid is equal to the decay heat in the test rods, so the heat transfer to the vapor will be the same irrespective of the decrease in the clad temperature at the grid location. Therefore, surface temperatures at locations removed from the grid site are not perturbed by the presence of the grid for tests 3.09.10i, j, l, and m.

The heat transfer coefficient for single-phase vapor is computed using the McEligot-plus-radiation correlation set for both large and small break LOCA. Its use is documented in the approved BWNT evaluation model (BAW-10168, Revision 1, September 1989). The correlation and its implementation are detailed in the approved RELAP5 and FRAP-T6 code topical reports, BAW-10164, Revision 1, October 1988 and BAW-10165, Revision 1, October 1988, respectively. Additional discussion is also presented in the response to Question 8 on the RELAP5 topical report (see BAW-10164, Revision 2, August 1992, pages 5-22 and 5-23), and in the Technical Evaluation Report (TER) for the approved RELAP5 report (see BAW-10164, Revision 2, August 1992, pages 5-219 and 5-220). The response to Question 8 and the TER both document the widely accepted use of the McEligot-plus-radiation correlation set for the calculation of steam cooling in LOCA applications.

The accurate measurement of vapor temperature, using thermocouples in the presence of unheated structures, is difficult at best. ORNL used two methods to obtain vapor temperatures. The first method

consisted of a numerical averaging of the fluid thermocouple measurements within the test assembly. The thermocouples were mounted on four unheated rods used to simulate guide tubes. Because of the proximity of the thermocouples to unheated walls, the measured vapor temperatures were necessarily lower than those next to the hot rods in the test assembly. The second method comprised an energy balance based on assembly power distribution. The method used the bundle thermocouples mounted downstream of the test assembly as an exit condition. Again, thermocouple placement was next to an unheated structure, the top of the test assembly. It is unclear which method was used in the ORNL report[3-1], but either would tend to substantially underpredict the vapor temperature surrounding the hot pin. Experimental vapor temperatures around the hot rod would be expected to be substantially the same as those predicted by RELAP5.

Based on the discussion above, BWNT believes that RELAP5-predicted vapor temperatures are appropriate for use in SBLOCA applications. Furthermore, from the ROSA-IV benchmark, comparisons of the predicted and experimental clad heatup rates during the core boildown period (see BAW-10164, Revision 2, August 1992, Figures J.34 through J.36), confirms that the McEligot-plus-radiation correlation set is properly implemented and suitable for use in SBLOCA licensing calculations.

3-1 Anklam, T. M., et al., "Experimental Investigations of Uncovered-Bundle Heat Transfer and Two-Phase Mixture-Level Swell under High-Pressure Low Heat-Flux Conditions," NUREG/CR-2456, ORNL-5848, March 1882.

Question 4 - Table H.2

The column headings for the pressure and the power density appear to be reversed. Please review this table and make the necessary corrections.

Response: The pressure and power density headings in Table H.2 were indeed inadvertently reversed. The corrected table is given below.

Table H.2. ORNL Thermohydraulics Test Facility (THTF) Benchmark Cases.

Case	Experiment	Power Density (kw/ft)	Pressure (psia)	Mass Flux (lbm/hr/ft^2)
13	3.09.10i	0.68	650	21943.9
14	3.09.10j	0.33	610	9333.4
15	3.09.10k	0.10	580	2306.5
16	3.09.10l	0.66	1090	21461.4
17	3.09.10m	0.31	1010	9313.0
18	3.09.10n	0.14	1030	3395.2
19	3.09.10aa	0.39	590	14938.7
20	3.09.10bb	0.20	560	6961.9
21	3.09.10cc	0.10	520	3706.1
22	3.09.10dd	0.39	1170	14615.7
23	3.09.10ee	0.19	1120	8111.9
24	3.09.10ff	0.08	1090	3561.1

Question 5 - Pages 2.3-83 and I-5

The equation used to compute the critical heat flux stated in Section 2.3.3, Equation 2.3.3-41.1 (q_{crit}) is different than that given by Equation I-1 in Appendix I (CHF). Differences are noted in the a_3x_3 and a_5x_2 terms of Equation 2.3.3-41.1. In addition, the Tong correction factor is missing from Equation 2.3.3-41.1. Please reconcile these differences.

Response: Equation I-1 contains typographical errors; the "ax" terms should be identical to those in Equation 2.3.3-41.1. Equation I-1 should read:

$$CHF = FLS(a_0 + a_1x_1 + a_2x_2 + a_3x_3 + a_4x_1^2 + a_5x_2^2 + a_6x_3|x_3| + a_7x_1x_2 + a_8x_1x_3 + a_9x_2x_3 + a_{10}x_1^3 + a_{11}x_2^3 + a_{12}x_3^3 + a_{13}x_1x_2x_3)/F.$$

The Tong factor is not missing from Equation 2.3.3-41.1; it has been set equal to one as stated on the top of page I-6. Implementation of the Tong factor in this manner was previously approved for the BWCMV CHF correlation as documented in both the RELAP5 and FRAP-T6 approved topical reports, BAW-10164, Revision 1, October 1988 and BAW-10165, Revision 1, October 1988, respectively.

Question 6 - Pages 2.3-83 and I-5

The equation used to compute the spacer grid factor stated as part of Equation 2.3.3-41 is different than that given by Equation I-2 in Appendix I. Differences are noted in the C_4LS and C_5L^2 terms in Equation 2.3.3-41. Please reconcile these differences.

Response: Equation I-2 contains typographical errors; it should be identical to the "FLS" term in Equation 2.3.3-41. Equation I-2 should read:

$$FLS = C_1 + C_2L + C_3S + C_4LS + C_5L^2 + C_6S^2.$$

Question 7 - Pages 2.3-83.1 and I-5

There appears to be a problem with units reconciliation for the X_1 and X_2 terms on pages 2.3-83.1 and I-5. Substituting the English to SI Conversion Factors from Table 2.3.3-1 yields the following:

$$X_1 = e^{\left[\frac{P(psia)}{1000 \times 6894.757 \frac{Pa}{psia}}\right]}$$

$$X_2 = \frac{G(lb/hr\text{-}ft^2)}{10^6 \times 1.356 \times 10^{-3} \frac{kg/m^2\text{-}s}{lbm/hr\text{-}ft^2}}$$

Note that the English pressure and flow units do not cancel. Please recheck these equations. Also, please explain the factors of 1000 and 10^6 in these equations.

Response: On pages 2.3-83.1 and I-5, the units for pressure, P, and mass flux, G, in the BWUMV CHF correlation are incorrectly stated as psia and lbm/hr-ft^2, respectively. The correct unit for pressure is Pa and kg/s-m^2 for mass flux. The 1000 factor in the X_1 equation is a pressure normalization factor having units of psia. The 10^6 factor in the X_2 equation is a mass flux normalization factor having units of lbm/hr-ft^2. Both X_1 and X_2 are non-dimensional quantities.

Question 8 - Appendix J

Please describe the modeling which alters the geometry of the steam generator outlet piping by modifying the vertical angle from 50 degrees to 14 degrees in more detail. Does this affect the volume, length or flow area of any component? Is this modification necessary to model reflux boiling?

Response: This modeling change was made to implement the RELAP5 horizontal stratification model which considers stratified conditions in horizontal components with an inclination angle less than 15 degrees. To facilitate drain-back during the loop draining/reflux cooling phases of an SBLOCA transient, the steam generator inlet piping components (volumes 208-2 and 408-2) were changed to horizontal components with inclination angles of 14 degrees. The reduction in vertical elevation due to the inclination angle change is compensated for in the steam generator inlet plenum (volumes 212 and 412) such that the total elevation change from the hot leg piping to the bottom of the steam generator tube sheet is preserved. This also preserves the overall system hydrostatic head, which is an important first-order effect in SBLOCA transients. Thus, the model change does not alter the transient characteristics of the SBLOCA. The other component geometries such as volume, volume length, volume flow area, and flow resistance are not altered. The modification prevents excessive liquid holdup in the upflow side of the steam generator,

and is necessary to properly simulate reflux boiling and tube draining as demonstrated in the results of the benchmark analysis.

Question 9 - Appendix J

On page J-7, it is stated that the non-equilibrium model is used, except in the core region. We understood that in RELAP5 MOD2, equilibrium nodes cannot be connected to non-equilibrium nodes. Has the code been altered to allow this modeling approach? If so, please give the details of this code change.

Response: RELAP5 has not been modified. Since its release, MOD2 has been fully capable of connecting equilibrium and non-equilibrium nodes. The ROSA-IV facility was benchmarked using equilibrium core nodes so as to replicate BWNT's currently approved SBLOCA EM core modeling (BAW-10168, Revision 1, Volume II, September 1989). Equilibrium core noding is also used in the approved LBLOCA evaluation model (BAW-10168, Revision 1, Volume I, September 1989), and is retained in recent SBLOCA and LBLOCA revisions to the evaluation model, BAW-10168, Revision 2, October 1992 and BAW-10168, Revision 3, October 1993, respectively.

Question 10 - Appendix J

In the Appendix J benchmark, the accumulator flow is stated to be (page J-11) "conservatively less than that of the experiment". Is this a result of the code prediction or was the accumulator modeled to achieve this result? With accumulator injection significantly less than that of the test, how does the benchmark demonstrate that the predictions of key variables, such as fuel surface temperature, are accurate? If the actual accumulator flow were used, this could result in the prediction of additional cooling and lower fuel temperatures.

Response: The accumulator was not modeled to underpredict its flow rate. BWNT experience with SBLOCA analysis has shown the need to increase accumulator line resistances, typically by factors up to 100, to mitigate high initial accumulator flow induced instabilities--flow oscillations and unphysical thermalhydraulic

behavior--in volumes downstream of the injection location. The added resistance, however, does not impact accumulator performance during the core uncovery period--the period of interest--since, in general for SBLOCA, the pressure drop between accumulator and reactor coolant system remains small, 4 psi for the ROSA-IV benchmark during the cladding temperature excursions. A case with unaltered line resistance shows accumulator flows being oscillatory but on average comparable to the benchmark case (increased resistance) during the period of interest (see Figure 10-1, attached herein); Figure 10-2 demonstrates the lack of impact on clad temperature heatup. The resistance change has no consequential effect on peak cladding temperature.

In the ROSA-IV benchmark, accumulator injection is not substantially underpredicted during the important core uncovery period. Major events and system response are calculated in good agreement with the experiment. Slopes on clad temperature curves (Figures J.35 and J.36) are properly predicted, demonstrating appropriate heat transfer during core boil-off. There is no concern regarding the predicted accuracy of key variables.

Use of actual accumulator flow would not cause the clad temperature to be underpredicted. The benchmark shows cladding temperatures begin to rise with the onset of the second core uncovery period, at approximately 320 seconds as shown in Figures J.20, J.35 and J.36. Accumulator injection, however, does not start until about 430 seconds, by which time the predicted cladding temperatures have almost peaked and are already above those of the experimental data. Integrated accumulator flow, a more appropriate measure for SBLOCA, is also reasonably predicted at approximately 550 seconds, by which time all clad temperature excursions have been quenched and core recovery assured (see Figure J.29).

Therefore, the use of actual accumulator flow or one from a line with resistance factors from 1 to 100 will not result in underpredicting clad temperatures. The conclusions stated in Appendix J remain valid and unchanged. The ROSA-IV integral system benchmark demonstrates the appropriateness of RELAP5 for use in SBLOCA applications.

FIGURE 10-1. INTACT LOOP ACCUMULATOR FLOW RATE.

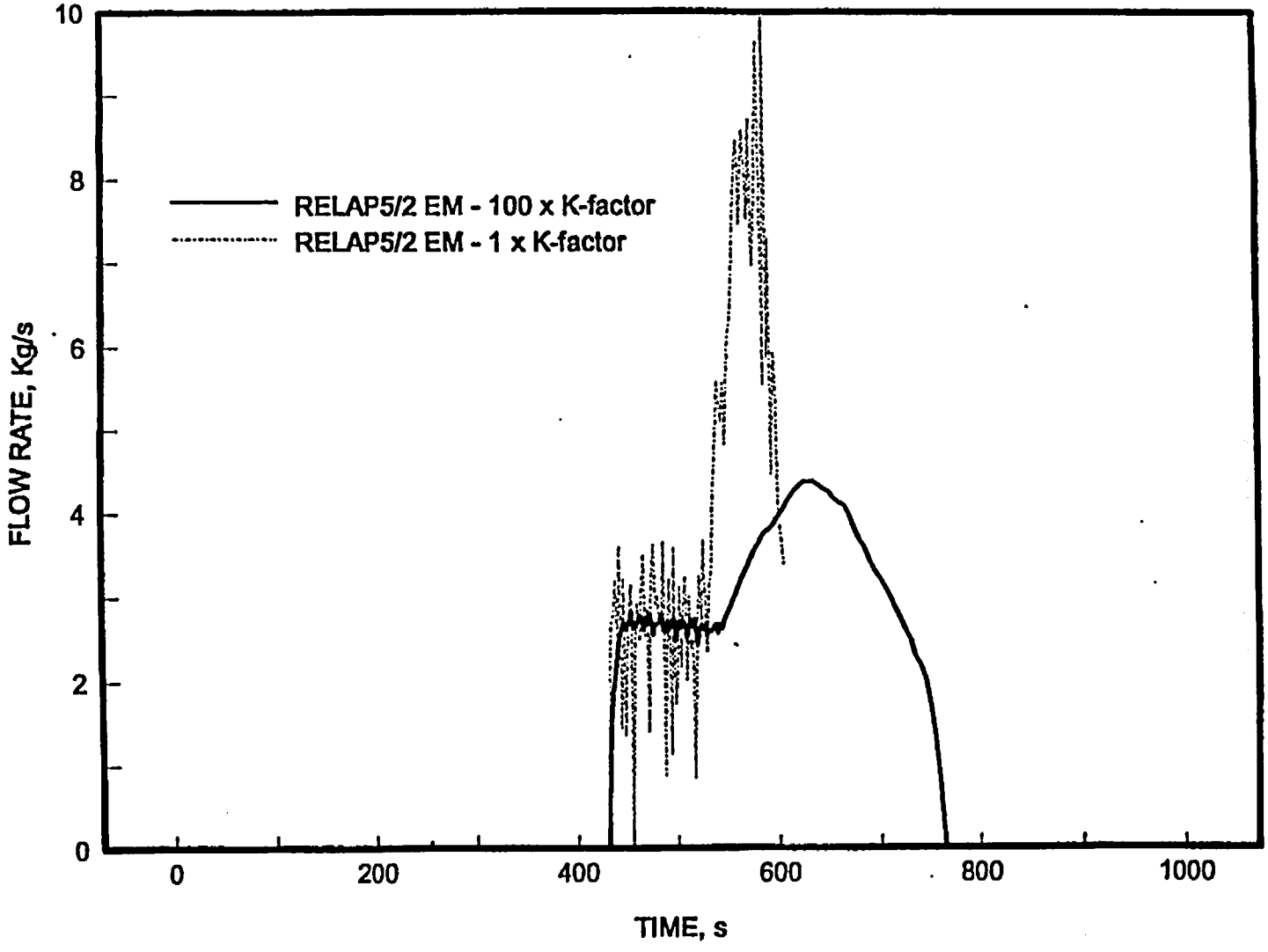
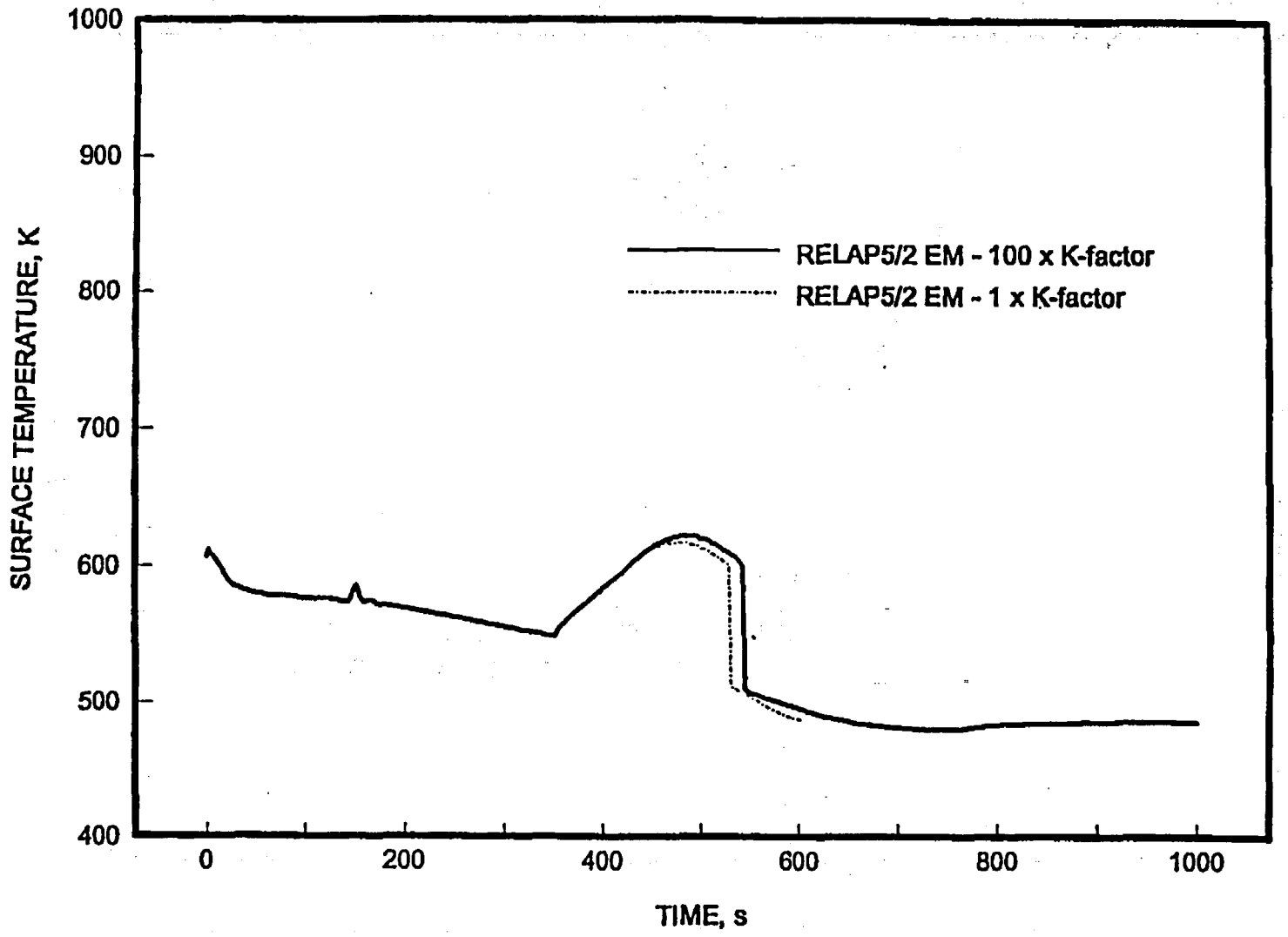


FIGURE 10-2. HOT ROD SURFACE TEMPERATURE - ELEV 3.61 M.



Question 11

No question 11 was transmitted in the request for additional information on BAW-10164, Revision 2, August 1992.

Question 12 - Appendix J

On page J-8, reference is made to Table 2. Should this be Table J.2?

Response: The reference table on page J-8 should be Table J.2.

5.5 Responses to Request for Additional Information on Revision 3

This section contains questions transmitted to BWNT by R. C. Jones of the NRC in his letter of October 1993, and responses transmitted by BWNT to the NRC in a letter from J. H. Taylor (JHT/94-7) dated January 21, 1993 [4].

-3

Requests for Additional Information

BAW-10164P, Revision 3

"RELAP5/MOD2-B&W, An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis"

Question 1 - General

Please clarify the intended usage of the changes made in Revision 3 of the Topical Report. In particular, will any of the new or revised features added to RELAP5/MOD2-B&W be used for Appendix K analysis?

Response

The transmittal letter for Revision 3 summarizes the code modifications specifically added for EM calculations. These can be divided into three general categories: (1) OTSG modeling improvements, (2) EM fuel pin modeling improvements, and (3) fuel rod surface heat transfer modifications. Each of the options discussed in this response is intended for EM use.

The OTSG or IEOTSG model improvements include the BWNT slug-drag and annular mist models, a new secondary side CHF correlation, and a smoothing of the Chen nucleate boiling S_f factor based on void fraction. These changes improve predictions of secondary side steady-state void distributions and liquid inventories, improve transient dry-out predictions, and smooth the heat transfer calculated at near dry-out conditions. The slug-drag model will also be used on the primary side (inside the SG tubes) for SBLOCA applications on B&W-designed plants.

RELAP5/MOD2-B&W will be used to calculate the system thermal-hydraulic and fuel pin thermal response during the blowdown phase

-3

of both large and small break LOCAs. The EM pin model changes improve the internal pin pressure calculations by adding a fuel pin thermal-expansion model in the axial direction. An option to allow fuel-clad mechanical contact for high burnup conditions enables the full range of fuel pin time-in-life conditions to be analyzed. An implicit solution of the Baker-Just metal-water reaction model was added to improve the numerical simulation for applications with blowdown ruptures. The fuel pin axial expansion, closed gap conductance, and implicit metal-water reaction models were formulated based on the NRC-approved FRAP-T6-B&W (BAW-10165) fuel pin performance models.

In addition to these model improvements, several optional conveniences were added to facilitate user input and output requirements. The input options included an automated steady-state fuel pin temperature iteration for matching the fuel pin stored energy specified from the NRC-approved TACO3 (BAW-10162) calculations and an automated BEACH droplet break-up and convective enhancement parameter calculation. For interpretation of calculated results, peak cladding temperature (PCT), time of PCT, location of PCT, maximum local oxidation, location of the maximum oxidation, and channel average oxidation edits were added to the code output.

The fuel pin surface heat transfer model was modified by the addition of a filtered mass flux option to meet Appendix K requirements. An option to increase the surface heat transfer area to account for blowdown pin rupture in the film boiling and steam-plus-radiation heat transfer modes was also added. All of the Revision 3 modifications are intended for use in Appendix K licensing analyses.

Question 2 - Page 2.1-52.4

B&W states that the modifications made to the INEL slug flow regime interphase drag model were developed from numerous benchmarks. However, no further information is given on these benchmarks beyond the calculations in Appendices K and L. Please provide additional information on the benchmarks used to develop this model. Were other benchmarks used?

Response

The BWNT modification of the INEL drag model was formulated by benchmark comparisons completed with the developmental code version. RELAP5/MOD2-B&W results were compared against results obtained with the NRC-approved FOAM2 code (BAW-10064), which uses the Wilson Bubble Rise model. The modifications were further verified by the separate-effects benchmarks to the GE Level Swell Test 1004-3, the Christensen subcooled boiling tests, and the ORNL bundle dryout tests. The results of these benchmarks, which were documented in the developmental assessment, showed good agreement with the experimental data. (The response to Question 14 presents additional ORNL benchmark results performed with Version 19.0.) The BWNT modification was then used in the Version 14.0 benchmarks against steady-state and transient 19-tube OTSG and IEOTSG test data. These steam generator secondary side benchmarks concluded that the default BWNT slug-drag model should be used with the addition of a 0.19 overall multiplier on the annular mist flow regime as shown on page 2.1-53 of the topical report. The OTSG benchmarks were provided in Appendix K. A final integral system benchmark against MIST test 320201 was provided in Appendix L. This benchmark provided drag comparisons in the core, steam generator primary tube regions, and steam generator secondary tube bundle (with the 0.19 annular mist multiplier).

-3

Question 3 - Pages 2.1-52, 2.1-52.4, and 2.1-52.5

With the publication of Revision 3 of BAW-10164P, there are now three separate models for determining the interphase drag in the slug flow regime which are: 1) the base INEL model, 2) the Wilson drag model, and 3) the B&W modified slug-drag model. The user has the option for selecting which of these models will be used for a given calculation. Please discuss how the user will select from among these models for a given analysis.

Response

The selection of the drag model is controlled by the evaluation model. For LBLOCA blowdown applications on RSG and B&W-designed plants, the base INEL drag model is used for all primary components. Both large and small LOCA models on B&W-designed plants use the default BWNT slug-drag model in the secondary tube bundle region with a 0.19 overall multiplier on the annular mist flow regime (see page 2.1-53). For SBLOCA applications in RSG plants, the Wilson drag model option is used for the reactor vessel core, upper plenum, and all vertical components. The B&W designed plants use the Wilson drag model in the core and upper plenum. The default BWNT slug-drag model is used inside the steam generator tubes for small LOCA applications. The default coefficients with the 0.19 multiplier on the annular mist regime are used for OTSG and IEOTSG secondary side applications. Both the B&W-designed plant EM and the RSG plant EM deactivate the Wilson void fraction smoothing option in the RV upper plenum.

-3

Question 4 - Page 2.3.37

On Table 2.3.2-2, please explain why ϵ_{TC} at 1073 K has been changed from 5.22E-3 to 5.14E-3.

Response

The code topical report was in error prior to Revision 3 in reporting the value of the radial strain function, ϵ_{TC}, as 5.22E-3 at 1073 K. The internal code values have always been 5.1395E-3 at 1073.15 K and 5.22E-3 at 1083.15 K. These values are consistent with the MATPRO-Version 11 correlation set (Reference 119 of the topical report).

-3

Question 5 - Page 2.3-46

The quantity β^2 is the fraction of flow area unblocked, not blocked as stated here. Please review and correct this definition.

Response

The quantity β^2 is the fraction of the flow area unblocked. The code formulation is consistent with the correct definition and hence needs no modification. The code topical report text is incorrect and needs to be revised to provide the correct definition.

Additional Typographical Correction: A typographical error was discovered in Equation 2.3.2-20 on page 2.3-36. The equation should have been cast:

$$u_{TC} = (r_{N_{HS}+1} \underset{\uparrow}{+} r_{N_f+2})\ \epsilon_{TC} / 2 . \qquad 2.3.2\text{-}20$$

A plus (+) sign has replaced a minus (-) sign in the above equation at the location indicated by the vertical arrow (↑).

Question 6 - Pages 2.3-37 to 2.3-57

Please describe the qualifications of the clad and fuel axial strain model. Is this model adapted from a previously approved fuel performance code? If so, please describe any differences between this model and the previously approved model.

Response

The clad and fuel axial strain model of RELAP5/MOD2-B&W is used for transient upper pin plenum internal gas volume calculations. The addition of this model improves the predicted internal pin pressure response during the LOCA. The model provides the change in plenum volume term, ΔV_{P_u}, of Equations 2.3.2-51.1 or 2.3.2-51.3 due to changes in fuel rod temperatures from hot initial conditions. This model is comparable to the previously approved fuel performance code FRAP-T6-B&W (BAW-10165, Reference 148 of the RELAP5/MOD2-B&W topical report) with the minor differences outlined below.

The RELAP5/MOD2-B&W model uses the inside clad radius, r_{ic_u}, in calculating the volume change term where as FRAP-T6 uses the fuel pellet radius. Both codes calculate the change in gas plenum length, ΔL_{cf}, as given in Equation 2.3.2-51.4. RELAP5/MOD2-B&W, however, accounts only for the thermal axial strain whereas FRAP-T6 accounts for both thermal and mechanical axial strains. The default thermal material properties ϵ_{ATC} and ϵ_{ATF} of Equations 2.3.2-51.5 and 2.3.2-51.6, respectively, are identical in both codes. RELAP5/MOD2-B&W uses the fuel volume-averaged temperature to calculate the fuel thermal expansion values for each segment, whereas FRAP-T6 uses the fuel temperature value at dish radius locations for the calculation of the thermal expansion function. Also, FRAP-T6 has a term to account for the effect of plutonium content on ϵ_{ATF}.

These differences between the FRAP-T6 models and the RELAP5/MOD2-B&W models are minimal. The FRAP-T6 models are more detailed and complex because it is a fuel pin performance code. The models

-3

extracted from FRAP-T6 for use in RELAP5/MOD2-B&W are LOCA-specific. That is, those parameters which may produce significant effects during the course of a LOCA transient have been integrated into the RELAP5/MOD2-B&W code.

-3

Question 7 - Page 2.3-52

Table 2.3.2-2 referred to on page 2.3-52 could not be located. Please provide a copy.

Response

The requested Table 2.3.2-2 can be found on page 2.3-37 of the topical report and was transmitted to the NRC with the Revision 3 submittal. Question 4 identifies a change unique to the Revision 3 table; therefore that page should have been in the submittal. Nonetheless, a copy of the page is attached with the transmittal of this response.

-3

Question 8 - Appendix K

The feedwater temperature is an important determinant of boiling length. Please clarify where the feedwater temperature is measured. Is the measurement before or after the steam aspiration?

Please describe the modeling of the steam aspiration process. It would seem that node 619 should be at a higher pressure than node 634. If this is so, how does the steam flow against the pressure gradient? How do you assure that the amount of steam flow predicted by the code is correct?

Response

The feedwater temperature was measured with a thermocouple located approximately 3 ft upstream of the aspirator mixer box for the OTSG tests. Therefore, the temperature was measured before mixing occurred with the aspirator flow.

The purpose of the aspirator path is to mix steam with the feedwater to preheat the water to saturation before it enters the tube bundle. Therefore, the feedwater temperature and flow control the aspirator flow. The feedwater and aspirator mixture determine the lower tube bundle flow rate, which directly influences the heat transfer and boiling length.

The pressure gradient that sustains the steam aspiration flow is created by the tube bundle-to-downcomer manometric balance. High boiling contributions in the mixture region lead to steam binding in the upper regions, and result in a bundle collapsed liquid level that remains below the downcomer level. The steam condensation on the subcooled feedwater injected into the top of the downcomer causes a local depressurization, which augments the aspiration flows.

The aspiration process is self-limiting. If too little steam is aspirated, the subcooled downcomer fluid condenses all of the steam creating additional local depressurization that increases the differential pressure across the aspirator. This

differential pressure increase leads to a corresponding increase in aspirator flow. Conversely, if too much steam is aspirated, the downcomer fluid saturates and cannot condense all of the steam. The pressure at the top of the downcomer increases such that the aspirator flow declines. Therefore, the steam aspiration process is self-limiting both in operation and in code applications. The confirmation of the code model is best shown by benchmarks against the 19-tube ARC steady-state and transient tests contained in Appendix K of BAW-10164.

-3

Question 9 - Appendix K

No plots are given comparing the RELAP5/MOD2-B&W results for the steam generator secondary level or mass to the OTSG-LOFW data in Appendix K. Please provide these plots.

Response

For two-phase, high flow applications (where flow losses are high), such as those observed in this OTSG benchmark, the determination of the experimental collapsed level is difficult at best, even during steady-state conditions. Under highly transient conditions, the calculation of collapsed levels is nearly impossible, without recourse to a code calculation that must be presumed accurate. Therefore, no transient collapsed levels or mass inventories were reported with the test results.

To provide information approximating what is requested, the comparison of the steady-state and transient total boiler differential pressures are supplied in Table 9-1 and Figure 9-1, respectively. The calculated steady-state differential was slightly higher than the measured value. This difference was attributed to and consistent with the initial steam flow comparison. The transient boiler pressure differential also reflected the higher steam flow rate calculated during the early portion of the benchmark. The calculated value was filtered to smooth oscillations. The filter used a centered five-point moving mesh averaging technique to smooth the oscillations in the differential pressure curve, given an edit frequency of one point per second.

The initial steady-state comparison of the RELAP5/MOD2-B&W mass inventory versus the value calculated from the test data is provided in Table 9-1. The facility initial secondary mass inventory was generated by adding the dry secondary steam mass to the integrated difference in the experimentally-measured stea flow minus the feedwater flow during the loss-of-feedwater

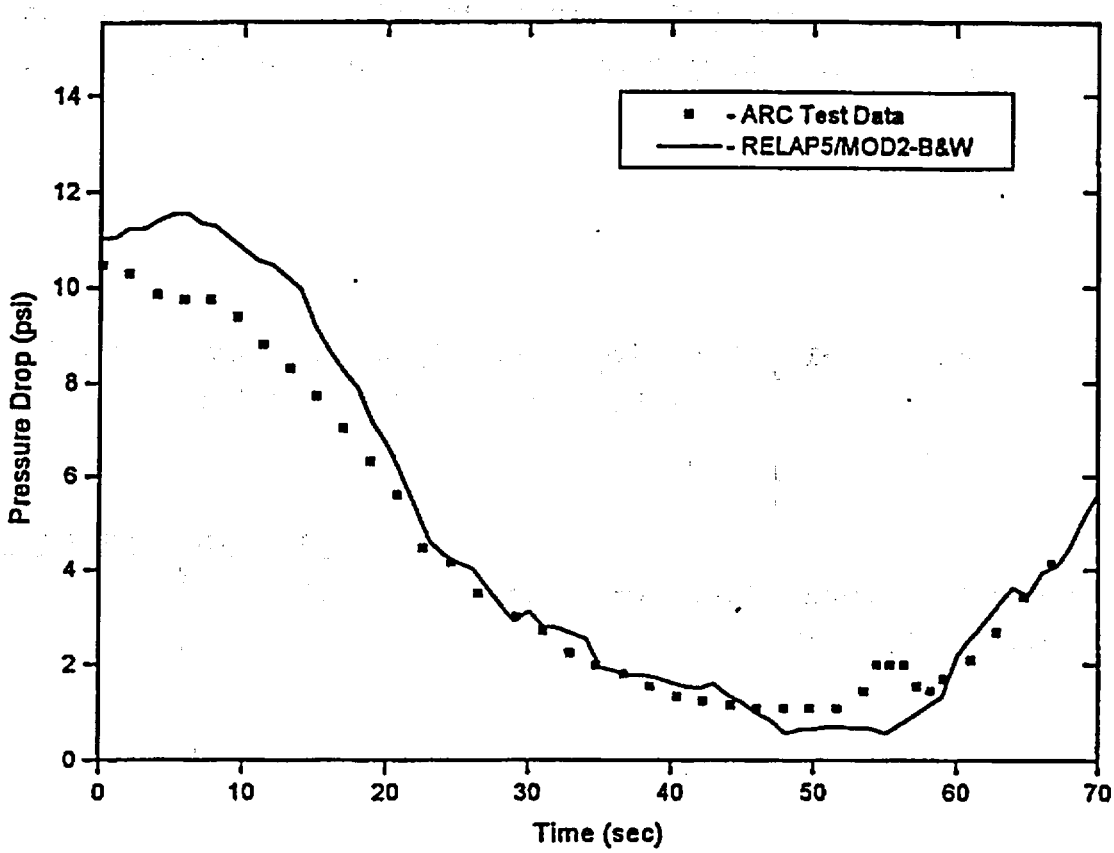
portion of the test. This integrated value was compared against the total inventory from the RELAP5/MOD2-B&W steady-state calculation. Approximately 7 lbm of steam was computed to remain in the dry secondary side prior to the beginning of refill.

Table 9-1. Additional Steady-State Parameters for the 19-Tube Model OTSG LOFW Test.

	Model OTSG	RELAP5/MOD2-B&W
Secondary Total Inventory, lbm	57.0	55.1
Total Boiler ΔP, psid	10.2	11.0

-3

FIGURE 9-1. OTSG LOFW Benchmark Boiler Region Differential Pressure Comparison.



Question 10 - Appendix L, Page L-13

On Table L.1, why is the core power input to the RELAP5 model two percent higher than that used in the test? Please explain the effect of this difference on the benchmark results.

Response

Table L.1 is in error. This table reflects the typical test steady-state conditions that were used as target values from which to initialize the post-test prediction model. Attached is a revised Table L.1 listing the actual initial conditions from Test 320201. These actual test conditions are much closer to the MIST RELAP5/MOD2-B&W initialization values.

The difference in the initial core power is much smaller with the corrected table. Nonetheless, it differs, and the difference relates to MIST operation and post-test benchmarking method. The initial power represents a 3.5 percent scaled core power (where full power is 2700 MW based on a 1/817 scale factor) with an additional 0.4 percent to offset the uncompensated loop heat losses. After break initiation, the power was held at the steady-state value until a MIST operator activated the core decay heat ramp. The total power supplied to the core heaters was controlled by a programmed curve set to match the 1971 ANS decay heat curve with a 1.2 multiplier. This curve was independent of the initial MIST test value other than the intersection point. The activation of the ramp, which occurred upon hot leg saturation (within one minute of the break opening for this test), assured that the calculated and test core power fractions were similar. Thus, the integrated difference in the transient power was quickly minimized. The effect on transient results was therefore negligible or inconsequential.

The initial steady-state values do not exactly match the test data because of the method agreed upon for the MIST post-test prediction analyses. The main goal of this method was to obtain

-3

"unadjusted" post-test prediction results to assess and verify the code formulation via use of the "frozen" input model. Changes in the input model were reserved for cases in which test initial conditions or control functions differed significantly from those specified in the base input model. Any adjustment of the post-test prediction model required justification and documentation. Therefore, many parameters were reviewed prior to performing a code calculation to determine the possible effects on results, given the noted deviations. Differences in the initial parameters listed in the corrected Table L.1 were categorized as having insignificant effects and therefore were not changed for the prediction. In the test initial conditions, the pressurizer surge line and lower pressurizer liquid temperatures were examples of deviations from the RELAP5/MOD2-B&W model that were considered significant. The temperatures controlled the timing of the loss of natural circulation and in turn affected the secondary side pressure response. The secondary response was quite important for this transient, since it controlled the timing of the steam generator reverse heat transfer that activated the secondary side blowdown.

Table L.1. Comparison of MIST Initial Conditions to RELAP5/MOD2-B&W Values.

Parameter, Units	MIST Test 320201 Value	RELAP5/MOD2-B&W Value
Primary Pressure, psia	1733.2	1726.5
Secondary Pressure, psia	1013.9/1014.1	1010.0
Core Exit Temperature, F	592.	593.4
SG Exit Temperature, F	549.	550.3
Core Exit Subcooling, F	23.6	22.0
Core Power to Fluid, Btu/s	119.0	119.5
Pressurizer Level, ft	4.9	5.0
SG Secondary Level, ft	4.7/5.0	5.0
Core Flow Rate, lbm/s	1.86	1.86

Question 11 - Appendix L, Page L-13

On Table L-2, please explain the large difference in the time of hot leg U-bend voiding between versions 5.0 and 14.0 predictions. In particular, describe the change in which loop hot leg U-bend voids first, and the large difference in time of occurrence between the test data and predictions. Also, page -v- states that SBLOCA modifications were made in code version 18.0, yet page L-11 states that the models in version 14.0 are identical to those in version 19.0. Why was an earlier version, 14.0, used for this benchmark? Are there actually no differences in the models, compared to version 19.0?

Response

The differences between the natural circulation flow interruption predictions for Versions 5.0 and 14.0 are attributable to differences in the pressurizer and surge line initial temperature conditions. The fluid temperature in the bottom of the pressurizer was lower than the saturation temperature due to surge line heat losses and insurges that occurred during the steady-state perio In the Version 14.0 benchmark, the pressurizer fluid temperatu was changed to match the actual test data. The colder fluid caused a variation in the hot leg A flashing rates between the two benchmarks. In the Version 5.0 prediction, the pressurizer fluid temperature was saturated, and additional flashing occurred. Thus, the interruption of the loop A flow occurred later than that calculated by the Version 14.0 benchmark.

A difference in the hot leg B interruption time was also reported. This variation was also related to the change in the pressurizer fluid state. The subcooled liquid reduced the flashing rate in the pressurizer, causing a faster RCS depressurization. The faster depressurization allowed the liquid in the B loop hot leg to saturate, flash, and interrupt sooner because of the collection of steam in the hot leg U-bend region.

The test time reported for interruption of natural circulation erroneously given as the onset of voiding in the hot leg U-bend region. The loss of circulation should have been reported as

between 60 and 120 seconds in both hot legs. The interruption time cannot be further resolved, given the scales of the available data plots. The true variation in the timing between the test and prediction is therefore small. The adjustment of the pressurizer liquid temperature improved the predicted RCS initial pressure response as well as the predicted loop flow and primary-to-secondary heat transfer interruption. The integrated effect was observed by the excellent agreement of the secondary side pressures shown in Figure L-4.

During the review of Table L.2, a difference was also noted in reported time of the operator actions to activate HPI, AFW, and the core power ramp. These operator actions were performed between 35 and 55 seconds in the test, not 30 to 42 seconds as previously reported in Table L.2. A revised Table L.2 is provided with the two indicated corrections.

This benchmark was included specifically to validate the BWNT slug-drag model. All of the SBLOCA modifications referred to in Version 18.0 and 19.0 are optional, user-activated models that were not available in Version 14.0 of the code. None of these modifications included changes to the BWNT slug-drag model. This benchmark, performed using Version 19.0 of the code, with the Version 14.0 input model, would reproduce the same results (since none of the new optional code models were activated). Therefore, it was appropriate to present the Version 14.0 benchmark results as those representative of Version 19.0 results.

-3

Table L.2. Sequence of Events.

Event	MIST Observation Seconds	Ver 5.0 Prediction Seconds	Ver 14.0 Prediction Seconds
Leak opened	0	0	0
Primary saturates	12	31	34
Pzr level reaches one foot (HPI, AFW, and DH ramp started)	35-55	60	57
Hot leg U-bend voiding interrupts natural circ. (Loop A/Loop B)	60 to 120 (both)	85/130	130/90
High elev BCM begins (Loop A/Loop B)	170/175	180/185	180/180
Break saturates	190	130	140
Secondary refilled and AFW shutoff (SG A/SG B)	480/480	490/440	480/480
Primary and secondary pressures equalize	1560	1500	1650
Secondary blowdown	1710	1500	1650
CFT injection begins	1920	1680	1800

-3

Question 12 - Appendix L, Page L-6

Page L-6 states that the revised slug drag model was implemented in the core and the shell side of the steam generator. Please confirm that the model used in Appendix L used the B&W modified slug-drag model referred to on page 2-52. Also, please confirm that the default coefficients presented on pages 2.1-52.4 and 2.1-52.5 are used in the benchmarks presented in Appendix L.

Response

The input model for Appendix L used the BWNT slug-drag model with the default coefficients, outlined on page 2.1-52.5, for the core and primary tube regions. The secondary side of the steam generator tube bundle region used the default BWNT slug-drag model with an overall multiplier on the annular mist drag of 0.19.

-3

Question 13 - Figures L-5 and L-6

Figures L-5 and L-6 present the collapsed liquid level in feet. Page L-7 infers that the reference level for the steam generator is the lower tube sheet. The reference level for the reactor vessel is not stated. Please provide the reference location for the reactor vessel and confirm that the reference level for the steam generator secondary is the lower tube sheet.

Response

In the MIST facility all collapsed level comparisons were referenced to the same datum (0.0 ft), which is the elevation of the upper face of the lower tube sheet in the steam generator. In the RELAP5/MOD2-B&W model, the bottom of the reactor vessel is located at 0.848 ft, the core is located between 4.74 ft and 16.74 ft, and the centerlines of the hot and cold leg RV nozzle connections are at 21.25 ft.

Question 14 - Page L-6

In addition to use of the revised interphase drag model in the slug flow regime, the number of volumes in the core region was increased from 3 to 20. The reason given for this change is that greater detail is needed to maintain consistency with the revised models. Presumably, this choice of nodalization results in the good agreement in the collapsed liquid level between the RELAP5/MOD2-B&W code and the MIST data. Use of 20 control volumes in the core obviously provides more detail on the axial flow profile in the core as compared to 3 control volumes. The question arises as to how much of the improved predictions is due to the modeling changes as compared to just increasing the nodal detail using the original RELAP5/MOD2 models.

Please discuss the implications of increasing the number of core volumes from 3 to 20 focusing on the degree of improvement resulting from the use of the new models as compared to just increasing the number of core nodes. Please include a sensitivity evaluation using the original RELAP5/MOD2 interphase drag models with the revised models.

Response

It is known that the INEL slug flow regime drag model overpredicts the interphasic drag in heated regions with small hydraulic diameters. Increased nodalization was the first undocumented attempt by BWNT to improve the core void distribution during the critical phase of plant SBLOCA analyses. The core void distribution was not significantly changed, even with big increases in the number of control volumes. The predictions consistently indicated high interphasic drag that yielded excessive two-phase level swell. The high level swell was not conservative for core cooling calculations. Therefore, the INEL model was considered inappropriate for SBLOCA licensing calculations that could predict clad heatup.

To confirm these findings and address the stated request for a sensitivity study, several benchmarks of test 3.09.10j from the ORNL Thermal-Hydraulics Test Facility were performed. This test is discussed in Appendix H of the RELAP5/MOD2-B&W code topical report. Selection of this test was made because of its similarity to the conditions at the minimum core inventory time of the typical, most-

limiting SBLOCA transient. Since this limiting SBLOCA has a small break area, the RCS depressurization rate is slow. Core recovering is nearly initiated with HPI flow at the approximate time that the RCS reaches the CFT fill pressure. The 0.05-ft^2 to 0.10-ft^2 break range is expected to produce this limiting condition between 20 and 40 minutes into the transient. The approximate core power is 2 percent of a maximum 17 kW/ft, which equals 0.34 kW/ft. This power level, and RCS pressure of approximately 600 psia, are the conditions characteristic of the most-limiting core mixture level encountered during the spectrum of SBLOCA analyses. Therefore, this test provides the most critical conditions for core level swell comparisons.

The first benchmark analysis used the Case 14 model from Appendix H and the INEL slug-drag model with a 24-volume representation. The predicted void distribution is shown in Figure 14-1. The second benchmark also used the INEL drag model with the 24-volume model combined into 4 equal-sized volumes. The curves indicate that the void distribution is poorly predicted by both the 4- and 24-volume models. The increased noding detail did little to improve the prediction because the interphase drag was excessive.

The same two benchmarks were performed using the default BWNT slug-drag model. Figure 14-2 gives the results for both the 4- and 24-volume analyses. As expected, good agreement was obtained with both models. The revised interphase drag was key in the improved behavior. The increased noding detail gives better resolution of the mixture height, but does little to the pool region void distribution.

Figure 14-3 gives the 24-volume model results for the base INEL drag option, default BWNT slug-drag option, and the default Wilson slug-drag option. The BWNT slug-drag and the Wilson drag results both provide good agreement to the experimental data. The quality of the predictions indicate that either model would be acceptable for core drag predictions.

-3

The formulation of the BWNT slug-drag model was based on a 20 axial volume core region. The discretization was identical to that used for the large break BEACH analyses (BAW-10168). This model was selected for consistency between large and small break methods as well as to better resolve the core mixture level for transients that may predict cladding heatup. Three or four axial volumes would be adequate for cases in which the core does not undergo a heatup. In the event, however, that the mixture level does descend into the core region, the mixture height is resolved to within approximately 0.6 ft with the use of the 20-volume model.

Although the BWNT slug-drag modifications were developed with a finely noded core region, the model has been shown to improve the phase separation in a coarsely-noded core or in steam generator secondary side bundles. This marked improvement demonstrates that it is not the number or the size of control volumes, rather it is the interphasic drag model, that is key to predicting proper phase separation in heated regions with small hydraulic diameters.

FIGURE 14-1. RELAP5/MOD2-B&W Predictions of ORNL Test 3.09.10j: 0.33 KW/ft, 610 psia.

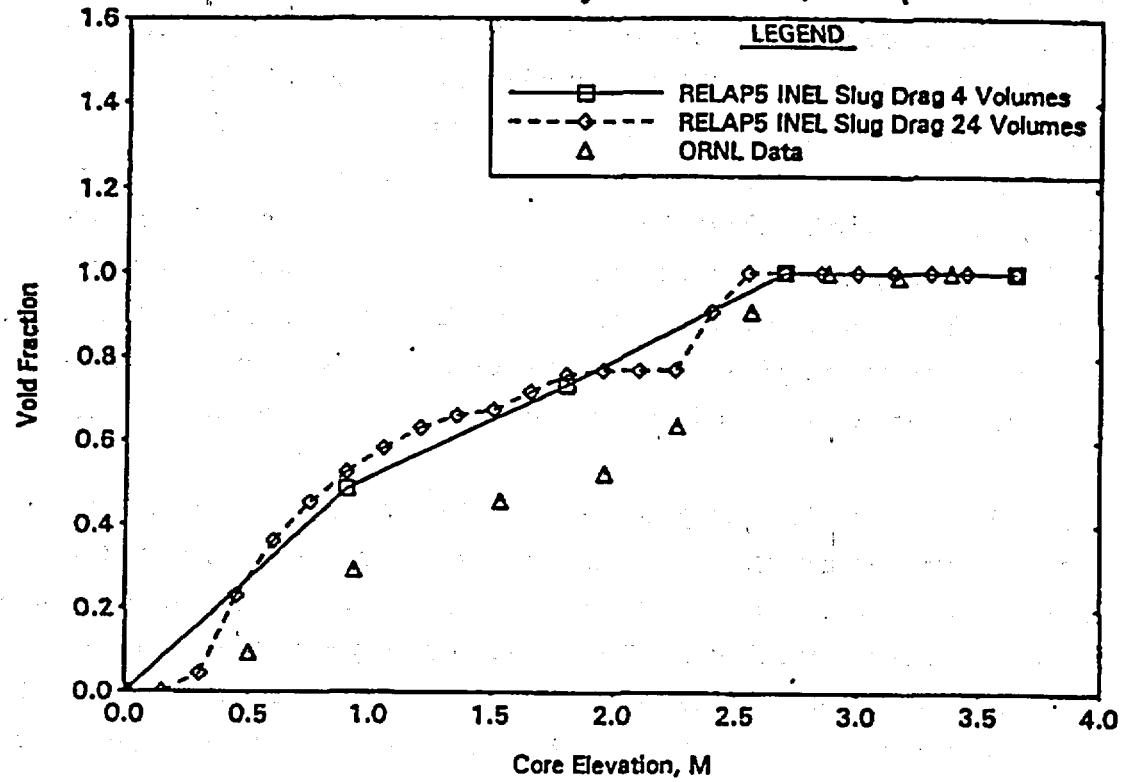
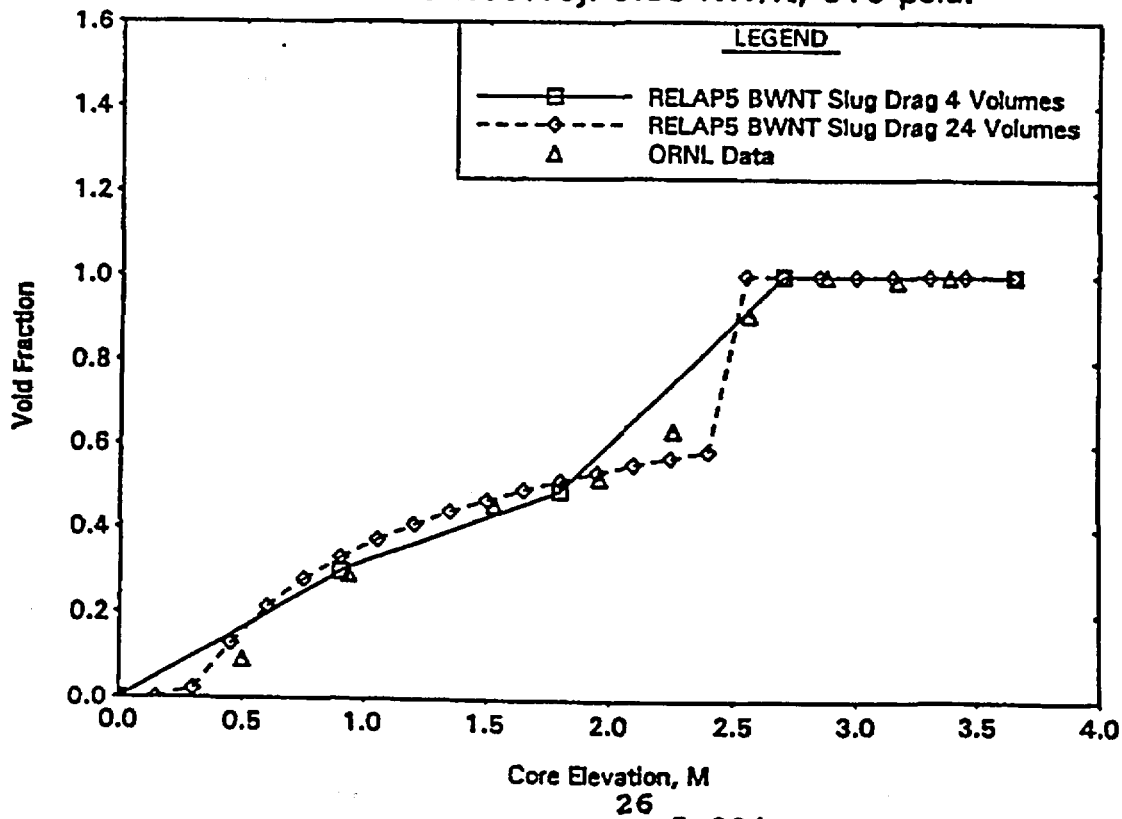
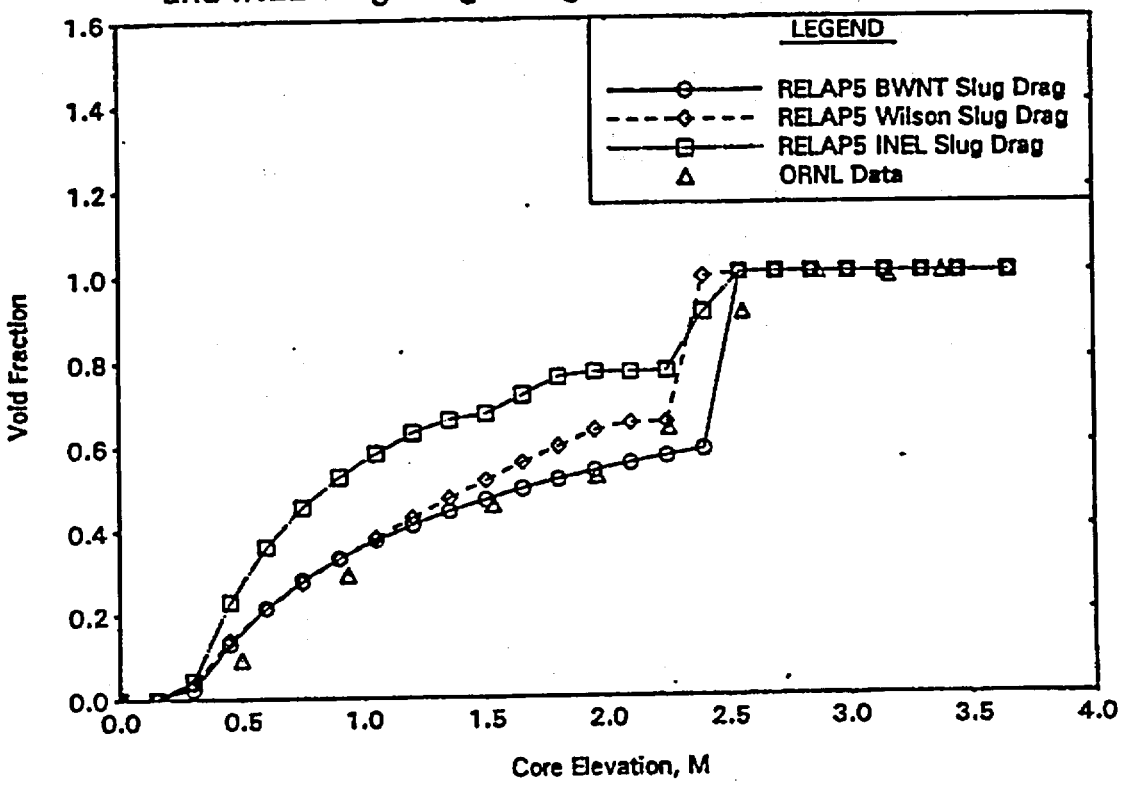


FIGURE 14-2. RELAP5/MOD2-B&W Predictions of ORNL Test 3.09.10j: 0.33 KW/ft, 610 psia.



-3

FIGURE 14-3. RELAP5/MOD2-B&W Predictions with the BWNT, Wilson, and INEL Slug Drag Using the 24 Volume Core Model.



-3

Question 15 - Page L-6

The revised interphase drag model is used to model the secondary side of the steam generator. The number of control volumes used to model the secondary side is not stated in the text. However, Figure L-2 indicates that there are 9 control volumes used on the secondary. If this assertion is true, then the height of each control volume representing the steam generator is much greater than those used in the reactor core.

Please discuss the nodalization of the steam generator system given the apparent need for greater detail in the core region in order to predict the collapsed liquid level as discussed in Question 2.

Response

The control volume heights below the steam generator aspirator elevation are 6.4 ft, and those in the MIST core are 0.6 ft. The effectiveness of the BWNT slug-drag model is not dependent upon the volume heights or number of volumes as indicated in the response to Question 14. Benchmarks with the ARC OTSG data, given in Appendix K, indicate that this level of detail is adequate to predict the governing phenomena in the steam generator. The location of the boiling and transition regions in the OTSG are known in sufficient detail to address level variations. The BWNT slug-drag model improves the void distribution profile and the associated inventory prediction, and as a consequence, the transient primary-to-secondary heat transfer is more accurately calculated. Better predictions of secondary heat transfer improve the accuracy of primary system response predictions.

-3

Question 16 - Appendix L

One of the key variables to be predicted is the fuel surface temperature. No comparison of predictions to test data are shown in the benchmark in Appendix L. Please provide this comparison or explain why such a comparison is inappropriate.

Response

Fuel temperature is an important parameter in SBLOCA analyses but for this particular benchmark the mixture level remained continuously above the top of the core in the analysis as well as in the test. The minimum vessel mixture level was located at the hot leg nozzle elevation. As a result, nucleate boiling removed the fuel stored energy and decay heat contributions after core fluid saturation. The high heat transfer coefficients associated with the nucleate boiling regime maintained the pin surface temperatures within several degrees of the fluid saturation temperature. Since no surface temperature excursions were observed in the test or in the analysis, no comparison of the temperature predictions to the test data were given.

-3

Attachment for Question 7

The following page is a copy of page 2.3-37 containing Table 2.3.2-2.

The radial strain function is defined by either a user input table as a function of cladding temperature for zirconium-based material types other than zircaloy or a built in code correlation set for zircaloy cladding[119] consisting of

$$\epsilon_{TC} = -2.0731 \cdot 10^{-3} + 6.721 \cdot 10^{-6} T_C \qquad 2.3.2\text{-}22$$

for $T_c \leq 1073.15$ K (α phase), and

$$\epsilon_{TC} = -9.4495 \cdot 10^{-3} + 9.7 \cdot 10^{-6} T_C \qquad 2.3.2\text{-}23$$

for $T_C \geq 1273.15$ K (β phase), where T_C is the average cladding temperature (K). In the α phase to β phase transition zone, $1073.15 \text{ K} < T_C < 1273.15$ K, a table lookup is used. Some selected values are listed in Table 2.3.2-2.

Table 2.3.2-2. Thermal Strain of Zircaloy for $1073.15 \text{ K} < T < 1273.15$ K.

T(K)	Radial Strain e_{TC}	Axial Strain e_{ATC}
1073.15	$5.14 \cdot 10^{-3}$	$3.53 \cdot 10^{-3}$
1093.15	$5.25 \cdot 10^{-3}$	$3.50 \cdot 10^{-3}$
1103.15	$5.28 \cdot 10^{-3}$	$3.46 \cdot 10^{-3}$
1123.15	$5.24 \cdot 10^{-3}$	$3.33 \cdot 10^{-3}$
1143.15	$5.15 \cdot 10^{-3}$	$3.07 \cdot 10^{-3}$
1183.15	$4.45 \cdot 10^{-3}$	$1.50 \cdot 10^{-3}$
1223.15	$2.97 \cdot 10^{-3}$	$1.10 \cdot 10^{-3}$
1273.15	$2.90 \cdot 10^{-3}$	$1.40 \cdot 10^{-3}$

Supplemental Information for BAW-10164

RELAP5/MOD2-B&W Revision 2 and 3

1. Given that RELAP5/MOD2-B&W has a two-fluid, six equation formulation, why are equilibrium control volumes used in the core region for RELAP5/MOD2-B&W LOCA analyses?

Response: The EM core heat transfer package used for fuel pin surface heat transfer uses nucleate, transition, and film boiling correlations that were formulated based on equilibrium fluid state conditions. Use of the RELAP5/MOD2-B&W nonequilibrium control volume option can, under certain blowdown conditions, result in higher surface heat transfer than would be calculated with equilibrium fluid conditions. This heat removal is not consistent with the formulation of the correlations and can be nonconservative from a peak cladding temperature perspective. Therefore, equilibrium control volumes are selected for use in the core region to provide appropriate boundary conditions for application of the EM core heat transfer correlations in RELAP5/MOD2-B&W. This approach is consistent with that used in the approved evaluation model, BAW-10168P Revision 1.

2. There are a number of additional user input options includ in the revision 2 and 3 submittals to RELAP5/MOD2-B&W Please identify those options that will be used in evaluation model calculations.

Response: The review of RELAP5/MOD2-B&W revisions is somewhat complicated because the impact on EM calculations of the new options are provided under separate cover. The new code options have been qualified by separate effects benchmarks and calculations. Description of the EM options is not presented in the RELAP5/MOD2-B&W code topical report, since two separate evaluation models reference the latest revisions to the code. The OTSG LOCA EM is contained in BAW-10192P and the RSG LOCA EM is contained in BAW-10168P Revisions 2 and 3. Section 9.0 of the large and small LOCA volumes of each EM is titled "Required Documentation." This section identifies the computer codes used in the EM and lists specific code options controlled by the EM. Also included is a table listing the generic and prescribed code inputs used in each EM. These EM tables are the appropriat locations to provide the necessary user input option control.

3. The BWNT small break LOCA (SBLOCA) evaluation models were revised to use the BWUMV critical heat flux correlation. The correlation is presented in Appendix I of the RELAP5/MOD2-B&W topical report, BAW-10164 Revision 2. Please clarify the value used for the Tong factor and provide justification for use in RELAP5/MOD2-B&W.

Response: The BWUMV correlation is a steady-state, local-condition critical heat flux correlation adapted from the NRC-approved BWCMV (Ref. 1) correlation. The correlation includes the non-uniform Tong factor. The Tong factor, with a value generally greater than one, accounts for non-uniformity in the power shape of the test section. The factor is required when comparing the correlation with steady-state data. As such, the measured-to-predicted comparisons in Appendix I are based on steady-state experiments, and therefore use the Tong factor.

For LOCA analysis, however, BWNT's established practice is to reduce transient CHF conservatism by setting the value of the Tong factor to 1.0. The measured-to-predicted comparisons and statistics in Appendix I demonstrate that the BWUMV correlation is properly formulated and appropriate for steady-state CHF prediction. Setting the Tong factor to 1.0 in transient analysis is a practice applied to all previous CHF correlations including BWCMV. This technique has been approved in all previous LOCA evaluation models, dating back to 1971. Support for the position is primarily based on experiments that do not indicate that fuel pins will experience CHF during the flow coastdown phase of the SBLOCA transient.

Secondarily, it is generally recognized that steady-state CHF correlations under predict transient CHF when CHF is based on local conditions. The following is a summary of supporting references.

Few authors make direct statements on the conservatism of applying steady-state correlations to transient situations. It is common for the author to present the information but leave its

interpretation to the reader. Also, much of the referenc material addresses only large LOCA conditions or extreme transients, since the experimental base for SBLOCA indicates no departures from nucleate boiling.

A survey by Leung (Ref. 2) interprets the observations of various investigators. Section III.B of the survey is mostly appropriate for larger breaks, while Section III.A, flow reduction transients, is reasonably characteristic of SBLOCAs. A review of the works by Schrock (page 25), Maxon and Edwards (page 25), Shiralka et al. (page 27), Letourneau and Green (page 27), Smirnov, Griffith et al., Redfield et al., Cermak et al., Lawson, Morgan et al., and Hicken et al. all confirm the conservatism of transient CHF predictions with steady-state correlations. The evidence presented continues throughout Section III of Leung's work.

Other authors, Tong and Weisman (Section 4.3.2.8 of Ref. 3) Collier (Section 9.6.2 of Ref. 4), Khater and Raithby (Chapter of Ref.5), McIntyre and Merilo (Ref. 6), and Vojtec (Ref. 7) also substantiate the generally held opinion that the prediction of CHF by steady-state correlations during a transient is highly conservative.

The observation that CHF does not occur during the SBLOCA coastdown period derives from integral system tests. Such tests consistently demonstrate that CHF does not occur for SBLOCA. Across the range of SBLOCA integral system tests--Semiscale, LOBI, ROSA, MIST, and LOFT (Ref. 8-13)--no observations of temperature excursions during the flow coastdown phase were made. In these reference tests, none of the cladding temperature excursions was initiated prior to loop seal clearing.

In conclusion three points were made in support of setting the Tong factor to one for SBLOCA CHF calculations.

- The transient application of steady-state CHF correlations with the Tong factor set to one was approved in all previous BWNT evaluation models. Only the CHF correlation, not its method of implementation, is being changed (BWCMV to BWUMV) in this SBLOCA revision to the RELAP5 topical report

- There is broad consensus in the literature that the use of steady-state, local-condition CHF correlations for transient predictions is conservative. Thus, the reduction of conservatism in the SBLOCA evaluation model, through setting the Tong factor to one is appropriate.

- The experimental record for SBLOCA shows that CHF does not occur during the flow coastdown phase of the transient.

Therefore, it is justifiable for BWNT to continue its standard practice of setting the Tong factor equal to one for the prediction of CHF during SBLOCA transients.

References for Supplemental Question 3

1. BAW-10159, "BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies," B&W Fuel Company, July 1990.

2. Leung, J. C. M., NUREG/CR-0056, ANL-78-39, Critical Heat Flux Under Transient Conditions: A Literature Survey, June 1978.

3. Tong, L. S. and J. Weisman, "Thermal Analysis of Pressurized Water Reactors," Second Edition, Published by American Nuclear Society, 1979.

4. Collier, J. G., "Convective Boiling and Condensation," Second Edition, McGraw-Hill, 1981.

5. Khater and Raithby, EPRI NP-1792, Full-Scale Controll Transient Heat Transfer Tests--Analysis Using the FAS Prediction Method, April 1981.

6. McIntyre and Merilo, Controlled Transient CHF Tests in a 5x5 Rod Bundle Under Loss-of-Coolant Accident Conditions, Seventh International Heat Transfer Conference, Munich, Germany, 1982.

7. Vojtec, Investigation of Transient Heat Flux Phenomena and Forced Convection Film Boiling Heat Transfer, Seventh International Heat Transfer Conference, Munich, Germany, 1982.

8. Kmetyk, L. N., "TRAC-PF1/MOD1 Independent Assessment: Semiscale MOD-2A Intermediate Break Test S-IB-3," NUREG/CR-4465, February 1986.

9. Duffield, J. S. and I. Shepherd, "Experience with CATHARE at J. R. C. ISPRA," CATHARE International Seminar, Grenoble, France, May 1988.

10. Barre and Bernard, "The CATHARE Code Strategy and Assessment," Nuclear Engineering and Design 124, 1990.

11. Koizumi, Y., et al., "Temporary Core Liquid Level Depression During A Cold-Leg Small-Break Loss-of-Coolant Accident: T Effect of Break Size and Power Level," Nuclear Technology, Vol. 96, December 1991.

12. Tasaka, K., et al., "The Results of the ROSA-IV LSTF Small-Break LOCA Experiments," Thirteenth Water Reactor Safety Research Meeting, NUREG/CP-0072, Volume 4, February 1986.

13. "Multiloop Integral System Test (MIST): Final Report," NUREG/CR-5395.

14. Jarrell, D. B. and J. M. Divine, "Experiment Data report for LOFT Intermediate Break Experiment L5-1 and Severe Core Transient Experiment L8-2," NUREG/CR-2398, EGG-2136, November 1981.

4. Figure I-6 of BAW-10164 (RELAP5/MOD2-B&W) gives the measured-to-predicted CHF ratios for the BWUMV correlation as a function of mass flux. The number of test points with mass fluxes below 1.25 mlbm/hr-ft^2 appears to be inconsistent with the sum of points given in Table I-3 and Appendix B of BAW-10159 (BWCMV). Please clarify the apparent differences in number of test points versus the number of data points included on that figure. Also, please provide confirmation of the applicable parameter range over which the BWUMV correlation is used in RELAP5/MOD2-B&W SBLOCA applications.

Response: Figure I-6 contains 77 calculated points for mass fluxes less than 1.25 Mlbm/hr-ft^2. The points are comprised of 22 test points from Table I.3 of BAW-10164 Appendix I, 20 test points from BAW-10159 Appendix B, and 35 test points from Table Q1-1 in BAW-10159 Appendix F. The points from Appendix F were considered annular flow points (qualities near 22 percent or above) for BWCMV. The form of the BWCMV CHF equation is a linear function of quality that is invalid for high qualities, which makes annular flow data inappropriate. These points are, on the other hand, appropriate for use in the BWUMV form of the correlation, which extends the quality range to 67 percent. Therefore, these points were included in the data base used to validate the BWUMV correlation.

The thermal-hydraulic parameter ranges used for the BWUMV correlation as implemented in RELAP5/MOD2-B&W are listed as:

$$1300 \text{ psia} \leq \text{Pressure} \geq 2455 \text{ psia}$$

$$0.5 \text{ Mlbm/hr-ft}^2 \leq \text{Mass Flux} \geq 3.871 \text{ Mlbm/hr-ft}^2 .$$

No upper limit on flow or pressure is included in RELAP5/MOD2-B&W. However, SBLOCA EM applications will not result in state conditions that exceed these values. Also, the quality is not explicitly limited in RELAP5/MOD2-B&W, however, in typical SBLOCA analyses the quality will not exceed the BWUMV limit unless it is due to dryout from core uncovering. Under these conditions CHF is unimportant. Additionally, should core uncovering occur, it

is typically at pressures below the minimum BWUMV pressure limi
such that the correlation is not used.

5. Only one MIST benchmark is provided in Revision 3 of BAW-10164. Have any additional benchmarks been performed to qualify the RELAP5/MOD2-B&W system predictions for B&W-designed plants?

Response: The RELAP5/MOD2 code was developed by EG&G, Idaho for predicting the system thermal-hydraulic behavior of a PWR during transient conditions such as a loss-of-coolant-accident (LOCA), particularly SBLOCA. The code has been benchmarked against many tests conducted in the U.S. and internationally. These benchmarks, in general, have confirmed that RELAP5/MOD2 is capable of simulating important system responses including system depressurization and flashing, break mass discharges, two-phase flow phenomena, core heat transfer, and system voiding. Most integral test facilities were scaled to a 4-loop recirculating steam generator (RSG) type reactor design. To provide integral system thermal-hydraulic data on B&W-designed plants, the NRC, BWNT, the B&W Owners Group, and EPRI jointly funded the design, construction, and testing of the Multiloop Integral System Test (MIST) facility. This scaled facility was designed to simulate prototypical SBLOCA phenomena for code benchmarking. The facility contained the unique features of the B&W-designed NSSS with two hot legs, four cold legs, once-through steam generators (OTSGs), and reactor vessel vent valves (RVVVs).

Pretest predictions and posttest predictions (Ref. 1-6) with a frozen input model were performed with RELAP5/MOD2-B&W to verify the simulations of the system responses during SBLOCA and SGTR events. The results consistently showed that the overall code RCS predictions were reasonably accurate. One deficiency that was repeatedly observed in all the predictions was underprediction of the core collapsed level. The source of the problem was high interphasic drag in the slug flow regime, which resulted in retention of too much steam within the two-phase mixture in the vessel. The excessive level swell affected the

core collapsed level predictions but did not have any significant effect on the remainder of the RCS response. Only heated regions with small hydraulic diameters revealed the excessive level swell. For certain tests, prediction of secondary side behavior (a heated region with small hydraulic diameter) also exhibited excessive level swell.

The code versions used for these MIST benchmarks did not have the Wilson or B&W slug-drag model improvements. The drag model changes were the only significant code modifications that would now be used for MIST predictions. A MIST benchmark was chosen to confirm that the overall RCS results would not be appreciably changed between the old version and new version when the new slug flow regimes models were used. That benchmark was also chosen to confirm the improvement in the calculated core and secondary side collapsed levels. Therefore, the benchmark to MIST Test 320201, a scaled 50 cm^2 break in the CLPD, was reanalyzed with the new drag models, and the results were included in Appendix L of BAW-10164. The overall RCS response was quite similar to the earlier calculations. As expected, the core and steam generator secondary side collapsed levels were greatly improved by the drag model change.

Investigation of the other MIST benchmarks referenced on page L-3 of Revision 3 of BAW-10164 led to the conclusion that under comparable boundary conditions the code predictions were reasonably correct except for the core or steam generator level swells due to high interphasic drag in the slug flow regime. For the scaled 10 cm^2 breaks or larger with pumps tripped (Tests 3109AA[1], 3105AA[6], 320302[1], 320503[1], 320604[3], 3404AA[5], 3406AA[1], 350101[4], and 3601AA[2]) the core region collapsed liquid level was underpredicted by about 1.2 to 3.3 ft. It should be noted that the level mismatch was fairly constant, especially when differences in depressurization rates were considered between the test and prediction. Reanalysis of these cases with Version 1[illegible] of RELAP5/MOD2-B&W would greatly improve the core level wit[illegible]

little change in overall RCS response as noted in the benchmark of Test 320201. Revised benchmarks of these MIST tests with the new drag models is not warranted, however, since no new evidence would be provided to substantiate the RELAP5/MOD2-B&W models for SBLOCA applications.

The revised benchmark of Test 320201 along with other benchmarks (GE level swell test 100-4, Christensen subcooled boiling tests, 19-tube OTSG benchmarks, ORNL bundle dryout tests, and the comparison against the NRC-approved FOAM2 code) provide ample justification for the use of RELAP5/MOD2-B&W Version 19 in SBLOCA EM applications. Accurate RCS SBLOCA response for both OTSG plants and RSG plants has been demonstrated. The drag model changes improve the core level swell predictions, such that conservative predictions of cladding temperature excursions will be calculated during periods in which the core mixture level is within the heated core region.

References for supplemental question 5:

1. J. A. Klingenfus and M. V. Parece, "RELAP5/MOD2 MIST Analysis Comparisons," Multiloop Integral System Test (MIST): Final Report, Vol. 10, NUREG/CR-5395, December 1989.

2. C. A. Schamp and J. A. Klingenfus, "RELAP5/MOD2 MIST Post-Test Benchmark of MIST TEST 3601AA - ATOG With Pumps Available," BAW-2033, 77-1171774-00, December 1988.

3. M. K. Smith and M. V. Parece, "RELAP5/MOD2 MIST Post-Test Benchmark of Test No. 320604 - 10 CM^2 Pump Discharge Break," BAW-2029, 77-1171643-00, December 1988.

4. M. B. McGuirk and M. V. Parece, "RELAP5/MOD2 MIST Post-Test Benchmark of Test No. 350101 - 10 CM^2 Primary System Break With High Point Vents," BAW-2032, 77-1168638-00, December 1988.

5. J. C. Seals and P. W. Ploch, "RELAP5/MOD2 MIST Post-Test Benchmark of Test No. 3404AA - Double-Ended Rupture of 10 Steam Generator Tubes," BAW-2031, 77-1171708-00, February 1989.

6. M. K. Smith and M. V. Parece, "RELAP5/MOD2 MIST Post-Test Benchmark of Test No. 3105AA - 10 CM^2 Pump Discharge Break," BAW-2030, 77-1171703-00, December 1988.

6. The B&W high auxiliary feedwater (AFW) model was not reviewed in the original release of BAW-10164 because the model was not used for applications on RSG plants. BWNT requested that this model be reviewed and approved for use in applications on B&W-designed plants. Please provide benchmarks or other supporting information justifying the use of this model in LOCA applications.

Response: The B&W high AFW model was originally developed for use in the AUX code, which calculated the dynamic interaction of the RCS and emergency feedwater system. It was later incorporated into the CRAFT2 code (Ref. 6-1). The NRC reviewed and approved the code model for use on B&W-designed plant EM applications in Reference 6-2. This approved model was later incorporated into RELAP5/MOD2-B&W with some minor calculational improvements.

The CRAFT2 SER summarizes the documentation used to quantify the high AFW tube wetting and heat transfer models. These included AUX benchmarks to plant data, Oconee-1 natural circulation tube wetting data, and ARC flow visualization data. It concludes that the high AFW model has been adequately verified against both separate-effects and integral system test data such that it is acceptable for use in SBLOCA EM applications.

The RELAP5/MOD2-B&W high AFW model consists of an AFW tube wetting model and a wetted tube heat transfer model. The tube wetting model is unchanged from the approved CRAFT2 model. The CRAFT2 heat transfer model has undergone some minor changes and improvements. The most significant change was a restriction of the Drew falling film heat transfer coefficient for subcooled AFW liquid heat transfer. For saturated liquid, the Chen nucleate boiling correlation was used to calculate the heat transfer coefficient for the wetted tubes. The steam heat transfer on the dry tubes was calculated based on the Dittus-Boelter correlation with a steam only Reynolds and Prandtl numbers.

The benchmark that best confirms the performance of the revise heat transfer model was performed as a part of the RELAP5/MOD2-B&W code certification effort. The benchmark is against test data from the MIST facility during steady-state conditions with high AFW injection. Figure 6-1 shows a cross-section of the MIST steam generator, including the tube designations. The AFW enters into the tube bundle approximately 50 feet from the lower tube sheet. Figure 6-2 shows a noding diagram of the RELAP5/MOD2 steam generator model used in the benchmark. The model arrangement above the aspirator is identical to models used in the SBLOCA EM. The primary tube region was separated into two regions. AFW was injected directly onto Tube J. Tubes H, J, and K were included in the AFW wetted region modeled by pipe Component 150. The remaining 16 tubes were included in the dry region modeled by pipe Component 140. Figure 6-3 shows the measured primary side temperature distributions and those predicted by the two channel RELAP5/MOD2 model. The comparisons are excellent. Because this was a separate-effects steady-stat test, the heat transfer coefficients were easily quantified an justified.

The transient-effects modeling of the B&W high AFW model were supported by every MIST test benchmark. The tests began in a steady-state natural circulation condition with high AFW injection. After break initiation, the transient steam generator heat removal from the AFW was very important to the overall behavior of both the primary and secondary sides. The quality of these comparisons confirms the validity of the entire high AFW computer model in numerous integral system benchmarks. This model will be used similarly for typical plant SBLOCA applications. In these analyses, this AFW model will affect the steam generator heat removal and provide the appropriate influence on the RCS system responses.

In summary, RELAP5/MOD2-B&W uses an improved form of the high A model that was approved for EM use in the CRAFT2 code. The

additional separate effects and integral system benchmarks further confirm the validity of the revised model and its appropriateness for use in SBLOCA applications. BWNT believes that, based on the previous NRC-approval and these new benchmarks, ample justification is provided for approving use of this model for B&W-designed plant SBLOCA applications with the RELAP5/MOD2-B&W code.

References for Question 6

6-1. J. J. Cudlin, et al., "CRAFT2 - FORTRAN Program for Digital Simulation of a Multinode Reactor Plant During Loss of Coolant," BAW-10092A, Rev. 3, July, 1985.

6-2. "Safety Evaluation Report for the Babcock and Wilcox Owners Group Small Break Loss-of-Coolant Accident Evaluation Model, CRAFT2 (Rev. 3) (BAW-10192P, Rev. 3 and BAW-10154)," 5/10/85, (Included in Ref. 6-1)

Figure 6-1. MIST STEAM GENERATOR CROSS-SECTION.

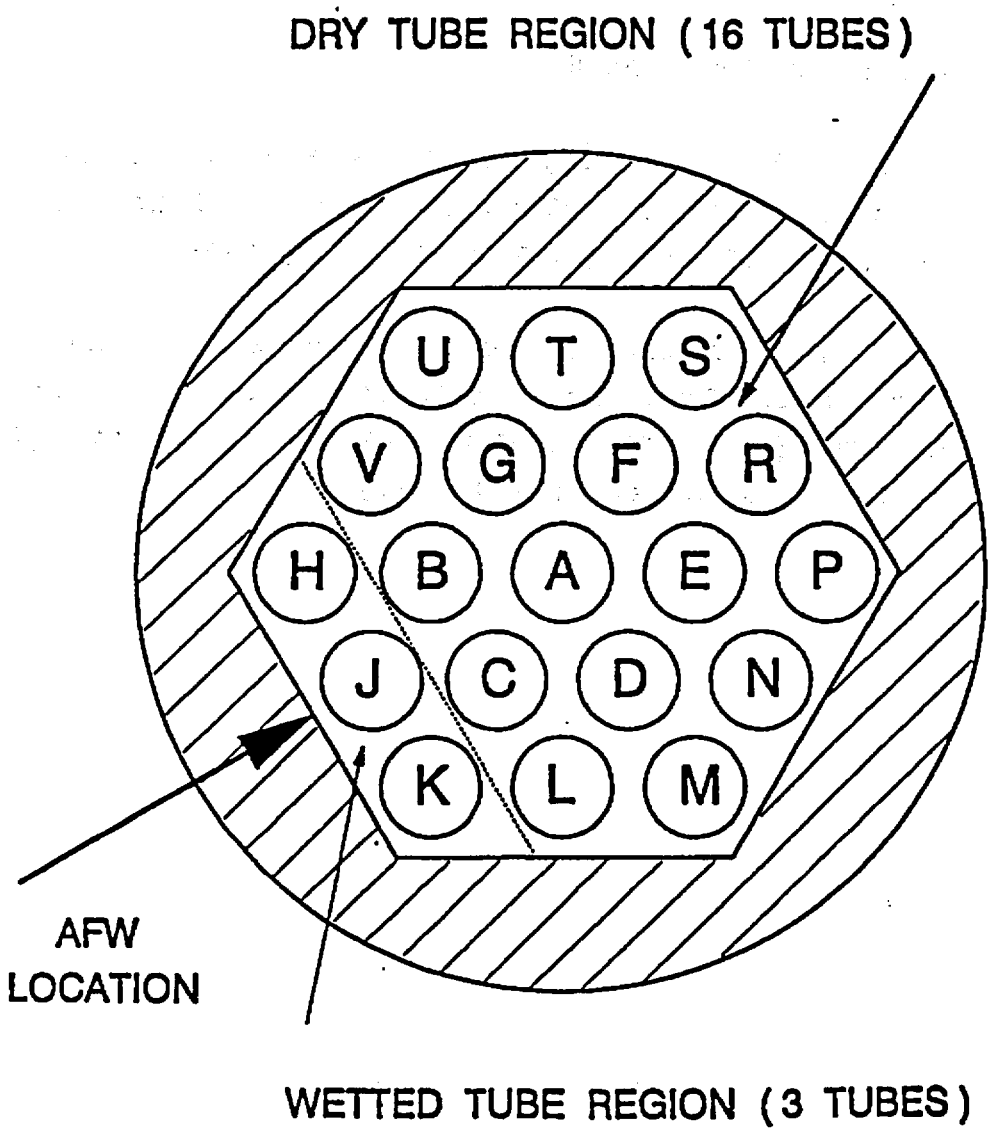


Figure 6-2. MIST STEAM GENERATOR NODING ARRANGEMENT.

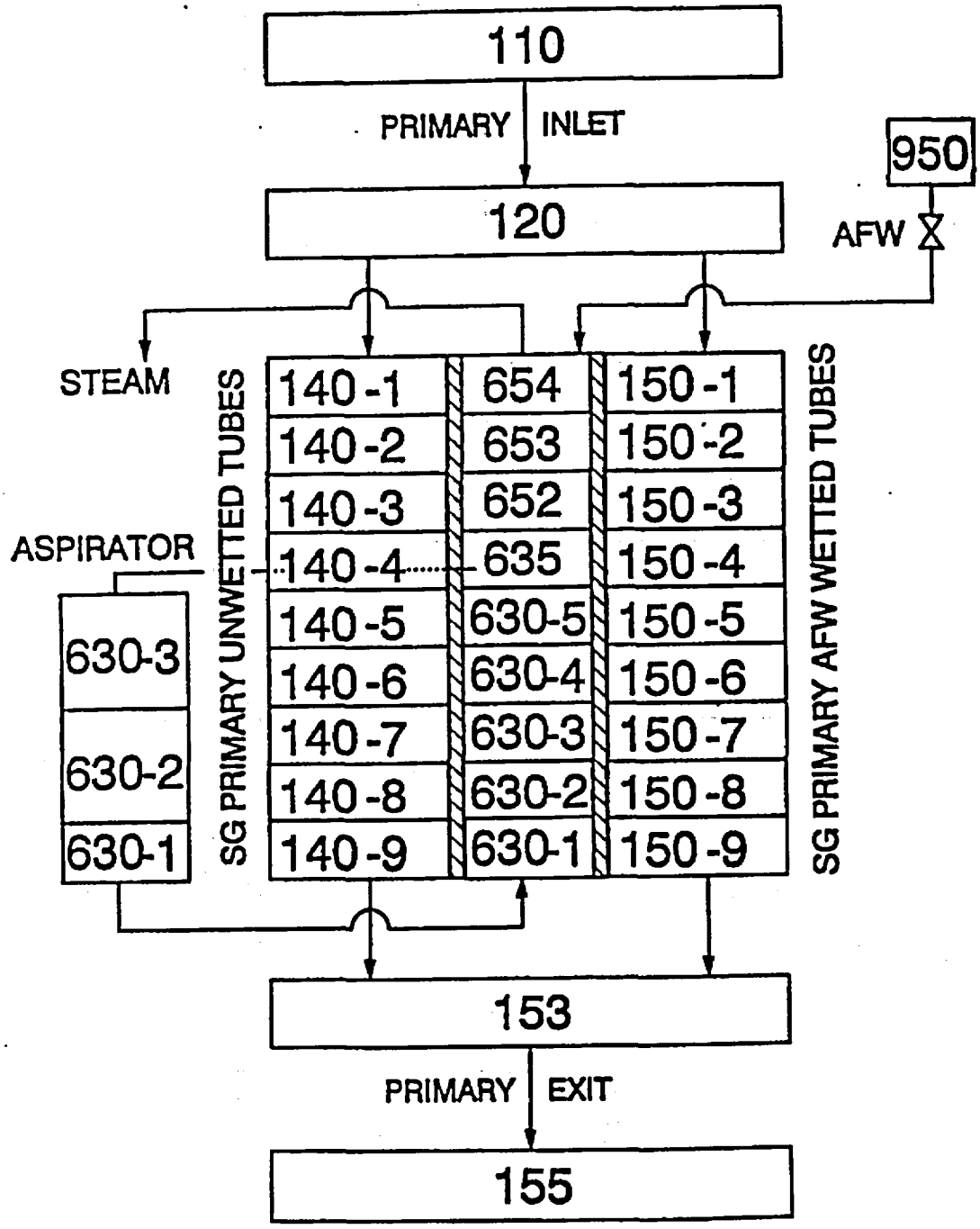
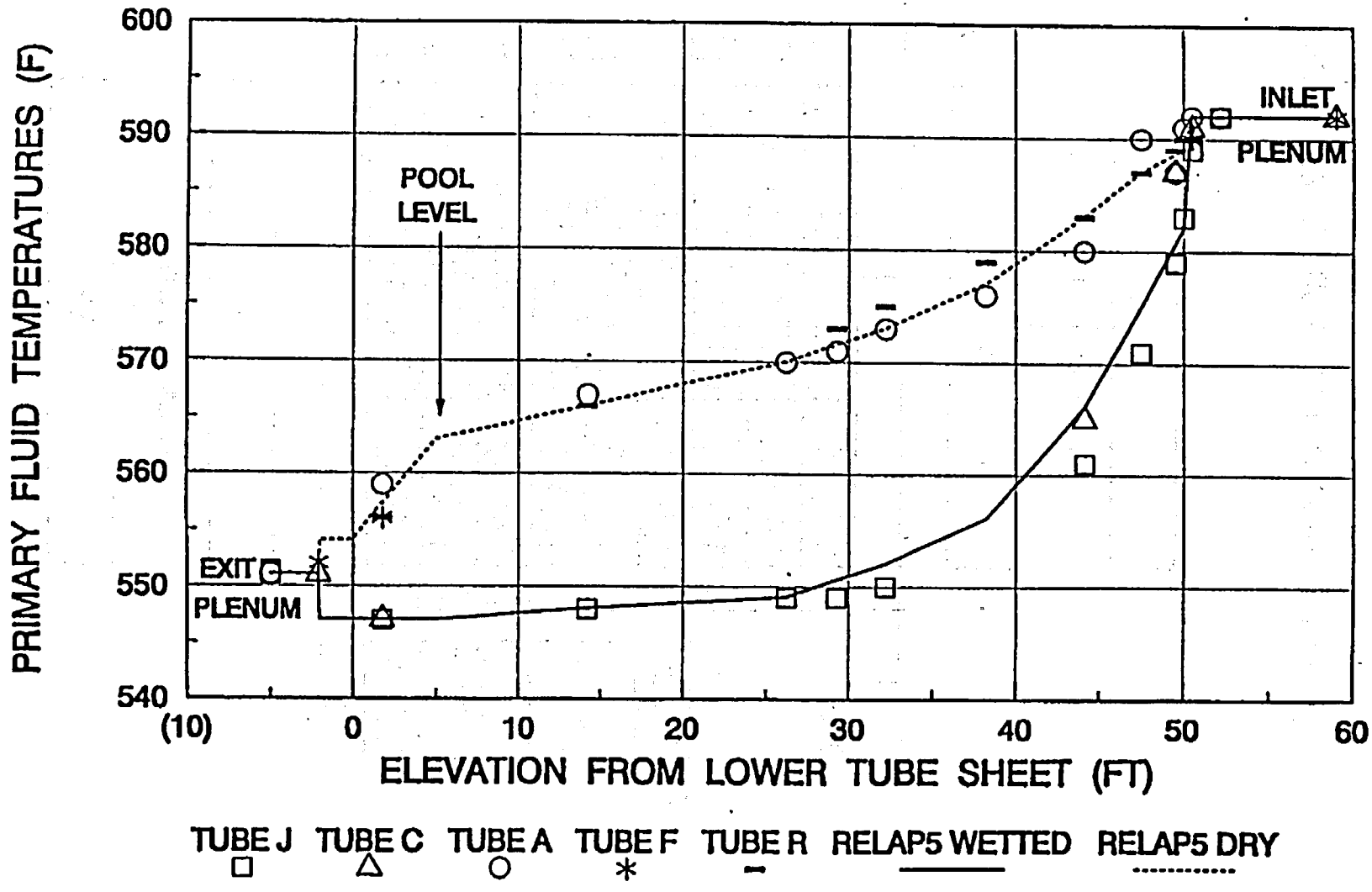


FIGURE 6-3. MIST MEASURED VS PREDICTED PRIMARY TEMPERATURES.



7. The MIST SBLOCA benchmarks provided in previous references did not show fuel temperature excursions. Please give references that justify conservative cladding temperature calculations during periods with fuel temperature excursions.

Response: For all of the MIST tests in which the reactor coolant pumps were not operating, none showed fuel temperature excursions. Therefore, combinations of test benchmarks must be used to support the conservatism of fuel temperature calculations for B&W-designed plants. There are four phenomena that govern the cladding temperature response: (1) the reactor coolant system (RCS) pressure response; (2) the RCS liquid inventory and distribution; (3) the core liquid inventory and mixture level during the core boildown phase, and (4) the core heat transfer above the mixture level during the boildown phase.

The MIST tests provided typical OTSG-plant RCS pressure and inventory responses for code benchmarks, directly addressing the first two phenomena. Good comparisons were obtained in the numerous RELAP5/MOD2-B&W benchmarks given as references in supplemental Question 5. These comparisons demonstrated that the code models, with typical plant nodalization, reproduced the key test thermal-hydraulic behavior for a large number of transients. Good representation of the RCS behavior provides the appropriate forcing functions for liquid inventory distribution, flashing contributions, leak phase determination, HPI flow, and overall transient progression.

Benchmarks of the MIST tests assure that the transient liquid inventory is appropriately calculated. Because the range of MIST experiments does not include cases for which the core mixture level falls into the active region, MIST is not the best benchmark for the core mixture level calculation. The core mixture level calculation during the core boildown phase is best benchmarked by the stand-alone FOAM2 and ORNL benchmarks provided in Appendix H of BAW-10164. The range of core and fuel assembly

geometries to which RELAP5/MOD2-B&W has been benchmarked extended through the FOAM2 benchmarks to include fuel for boiling water reactors (the GE tests), for Japanese reactors (the Hitachi tests), for Westinghouse reactor designs (the Westinghouse tests), and for B&W-designed plants (the B&W tests). The benchmarks confirm both that the RELAP5/MOD2-B&W code correctly determines the mixture level in a prototypical core region and that it does so over a wide range of core and fuel assembly geometries. The ORNL benchmarks also provide comparisons and confirmation of the cladding surface temperature predictions (see Figures H.28 through H.33).

The MIST benchmarks showed accurate prediction of the RCS pressure and liquid inventory during the loop-draining phase. Good predictions of these parameters assure that the SBLOCA will enter the core boildown phase at the appropriate transient time and with the appropriate reactor vessel liquid inventory. The core boildown is then governed by the ECCS injection and core decay heat rates.

Since the mixture level has been shown to be correctly calculated (Appendix H), the determination of the heat transfer for portions of the fuel rods cooled by mixture and by steam must also be correct. The fuel pin heatup is controlled by integrated fuel pin heat addition above the mixture level and by the core steaming rate below the mixture level. Heat transfer below the mixture has repeatedly been shown to be sufficient to maintain the cladding within a few degrees of the coolant saturation temperature. Thus, except for the movement of mixture level, the core steaming rate is a straightforward function of the decay heating rate. Above the mixture, the cladding temperature is controlled by conduction and radiation heat transfer to vapor and by the vapor superheat. For a given cladding temperature excursion, the majority of the heat removal is attributed to the superheating of the vapor and only a small portion (estimated to

be approximately one-quarter) is due directly to other heat transfer mechanisms.

The calculational approach used in RELAP5/MOD2-B&W for the determination of vapor temperature and cladding heat transfer coefficient is straightforward and widely accepted for core boildown conditions. Two examples of temperature excursions during integral system benchmarks are contained in the RELAP5/MOD2-B&W topical BAW-10164, Revision 2. Appendix J contains an SBLOCA benchmark of a ROSA test. Conservative cladding temperatures were shown during the long-term core boildown phase of that transient. Section 5 of this topical (responses to questions) contains a benchmark of Semiscale S-LH-1 beginning on page 5-153. The calculated cladding temperatures given on page 5-187 were conservative compared to the test data. The Technical Evaluation Report (TER) for BAW-10164, Revision 2, (page 5-227 and 5-228) acknowledges this conservatism.

These benchmarks collectively demonstrate that the RELAP5/MOD2-B&W code, with typical plant input models, produces best-estimate RCS thermal-hydraulic responses during SBLOCA transient benchmarks for OTSG and RSG plants. Separate-effects core level swell benchmarks confirm the mixture level calculations and steam cooling tests in conjunction with integral system tests, ROSA and Semiscale, confirm the accuracy of the calculated cladding heatup. These models are ultimately coupled with key conservatisms in the SBLOCA plant analyses. The most important of the conservatisms is the 10CFR50 Appendix K requirement that core decay heat be calculated as 120 percent of ANS 1971 standard. This decay heat (1.2 X ANS 1971 is approximately 35 percent above current best-estimate values) causes significantly more core boiloff, such that core heatup begins much earlier than is realistic. The heatup generally lasts longer, and more ECCS is required to match and exceed the core boiloff rate. The early timing of the heatup and the longer heatup period ensure that conservative cladding temperatures will be calculated.

B&W designed NSSs provide one feature not found in other PWRs. The reactor vessel vent valves (RVVVs) are unique to B&W-designed plants. The RVVVs swing open to provide a direct path for venting of core steam to a cold leg break location. Their design and performance preclude the need for the cold leg pump suctions to be cleared of water for system steam production to be vented to cold leg breaks. Thus, the core liquid inventory reduction and attendant cladding temperature increase required to accomplish loop seal clearing in other PWR designs is eliminated.

The RVVVs are not significant to SBLOCA predictions other than in their function to control the pressure difference between the reactor vessel downcomer and core, and the prevention of the need for loop seal clearing. This function is important, but it does not relate directly to the cladding temperature calculation. The only relationship is through the reactor vessel liquid inventory predictions. The ability of the RELAP5/MOD2-B&W to appropriately model the RVVV influence on system inventory has been adequately benchmarked in the MIST test series. Therefore, further justification of the RVVV performance during an SBLOCA core boildown phase in which it plays a secondary role is not warranted.

The cladding temperature excursion during SBLOCA in current PWR designs is brought about and governed by the same physical processes, regardless of the particular PWR design. BWNT has provided appropriate benchmarks of the ability of RELAP5/MOD2-B&W to simulate these mechanisms. The unique features of the B&W design that effect SBLOCA, the RVVVs and once-through steam generators, have been tested and benchmarked to a test facility that includes a direct modeling of the devices during the SBLOCA phases within which they act to produce differences between the B&W design and other NSS designs. There is, therefore, no need for the inclusion of a core uncovering test specific to the B&W design in the SBLOCA benchmark matrix.

8. Information provided in Supplemental Question 3 stated that a Tong factor of one has been used in LOCA applications since 1971. Please provide a previously-approved reference that explicitly states this licensing position.

Response: BWNT has two examples of NRC-approved LOCA evaluation models that explicitly state how the Tong factor is used. The first EM is for B&W-designed plants (BAW-10104, Rev. 5). THETA1-B (BAW-10094A, Rev. 3) is the core thermal analysis code used in this EM. It states in the first paragraph on page 21 of BAW-10094:

> "A nonuniform flux factor is combined with the Westinghouse W-3, the Babcock & Wilcox B&W-2 and BWC CHF correlations for steady state calculations. However, since the flux factor was not developed for accelerating and decelerating flow situations it is not used during transient computations."

The second EM is for RSG plants (BAW-10168, Rev. 1). FRAP-T6 (BAW-10165, Rev. 1) is the core thermal analysis code used in this EM. It states on page 2.1-37 for the B&W-2 CHF correlation (Equation 2.1.4-30 gives the Tong factor):

> "F_{BW2} = nonuniform axial power shape factor, (optionally used only for steady-state calculations), ..."

The BWC (Equation 2.1.4-32), BWCMV (Equation 2.1.4-34), and W-3 (Equation 2.1.4-35) CHF sections contain similar statements to the one shown above, indicating that the Tong factor is not used for transient applications.

9. Equation 2.1.3-30.6 contains a user-specified coefficient C_{VB}. What value is used for this coefficient?

Response: Equation 2.1.3-30.6 is

$$v_{bub} = C_{VB} \Delta v \quad .$$

A value of 1.0 is used for C_{VB} in all EM SBLOCA applications. This input was originally included to permit sensitivity studies, but it has never been used in any reported application.

During the review of this question, a typographic error was noted in Equation 2.1.3-30.7. The last term in that equation is Δv^2. This term should have been v^2_{bub} from Equation 2.1.3-30.6. However, since C_{VB} is always 1.0, the equation is correct as used in the EM applications.

5.7 Revisions 2 and 3 SER

This section contains the SER/TER transmitted to BWNT by G. M. Holahan of the NRC in his letter of March 14, 1995.

This page intentionally left blank.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 14, 1995

Mr. J.H. Taylor, Manager
Licensing Services
B&W Nuclear Technologies
3315 Old Forrest Road
P.O.Box 10935
Lynchburg, VA 24506-0935

Dear Mr. Taylor:

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT, BAW-10164P, REVISIONS 2 AND 3, "AN ADVANCED COMPUTER PROGRAM FOR LIGHT WATER REACTOR LOCA AND NON-LOCA TRANSIENT ANALYSIS"

The NRC staff has reviewed the topical report BAW-10164P, Revisions 2 and 3, which describe a RELAP5 based computer program for PWR LOCA and non-LOCA transient analysis. Revision 2 was intended for small break LOCA applications and Revision 3 included enhancements to the evaluation model for fuel pin heat transfer, and benchmarks extending application of the code to the once through steam generator plants. In addition to the benchmarks provided in the approved Revision 1 of the code, Revisions 2 and 3 provided comparisons to the ROSA-IV, THTF and the MIST experimental data. The RELAP5 based computer program provides predictions of the physical phenomena which are important during a small break LOCA.

The staff finds BAW-10164P, Revisions 2 and 3 to be acceptable for referencing in LOCA and non-LOCA PWR licensing actions to the extent specified, and under the limitations stated in BAW-10164P, Revisions 2 and 3 and the associated U.S. Nuclear Regulatory Commission safety evaluation report which defines the basis for accepting this topical report.

If the staff's criteria or regulations change so that its conclusions about the acceptability of the report are invalidated, B&W Nuclear Technologies should revise and resubmit their respective documentation or submit justification for the continued effective applicability of the topical report without revising their respective documentation.

The staff was assisted in this review by SCIENTECH, Inc. under contract No. NRC-03-093-031, JCN No. E-2095, Task Order No. 2.

Our safety evaluation (Enclosure 1) is based on the SCIENTECH, Inc. technical evaluation report SCIE-NRC-224-94 which is in Enclosure 2.

Sincerely,

Gary M. Holahan

Gary M. Holahan, Director
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

Enclosures:
As stated

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ATTACHMENT 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

BAW-10164P, REVISIONS 2 AND 3, "AN ADVANCED COMPUTER PROGRAM FOR LIGHT WATER REACTOR LOCA AND NON-LOCA TRANSIENT ANALYSIS"

B&W NUCLEAR TECHNOLOGIES

1.0 INTRODUCTION

By letter dated September 18, 1992, the B&W Nuclear Technologies Company submitted the topical report BAW-10164P, Revisions 2 and 3 for NRC review. The report describes a pressurized water reactor (PWR) thermal-hydraulics transient analysis code for LOCA and non-LOCA transients analysis based on the RELAP5/MOD2 code.

RELAP5/MOD2-B&W is a B&W Nuclear Technologies (BWNT) adaptation of the Idaho National Engineering Laboratory (INEL) RELAP5/MOD2[1] code used for PWR licensing and best estimate thermal hydraulics transient analysis. RELAP5/MOD2 was developed by INEL as a best-estimate computer code for light water reactor transient analysis. B&W Nuclear Technologies has added features to permit use of the RELAP5/MOD2-B&W code for ECCS evaluation model (EM) calculations. The previous revision of the RELAP5-B&W code, Revision 1[2], was approved for use in the analysis of small break and large break LOCAs[3] (SBLOCA and LBLOCA) for recirculating steam generator plants.

Revision 2 of BAW-10164P[5] describes updates for use in performing small break LOCA analysis. These updates include an additional critical heat flux (CHF) correlation referred to as

BWUMV, addition of the Wilson model for determining interphase drag, addition of a counter-current flow limiting (CCFL) model and correction of minor code errors. Benchmarks are included in Revision 2 to specifically address the Wilson interphase drag model and the small break LOCA EM model. The SBLOCA benchmark is against experimental data from the ROSA-IV large scale test facility.

In addition to the correction of minor errors, Revision 3 of BAW-10164P[6] includes enhancements to the EM fuel pin model, EM heat transfer model, and models to support use of the code for analysis of once through steam generator (OTSG) plants. These models include the Becker CHF correlation, further modifications to the slug-drag model, the high auxiliary feedwater model and the Chen nucleate boiling heat transfer coefficient void ramp. Benchmarks against model 19-tube OTSG data and against SBLOCA test data from the MIST facility, which has simulated OTSGs, are included. The modifications to the approved licensing model proposed in References 4 and 5 are the subject of the review and evaluation documented in this report.

2. SUMMARY OF THE TOPICAL REPORT

BAW-10164P Revisions 2 and 3 present best estimate and licensing type calculation for PWRs. Simulation methods are presented for large and small break LOCAs as well as operational transients such as anticipated transients without scram, loss of off-site power, loss of feedwater, and loss of flow. The solution is based on a two energy equation scheme, a two step numerical option, a gap conductance model, constitutive models and control system models. Control system and secondary system components have been added to permit modeling of plant controls, turbines, condensers, and secondary feedwater conditioning systems. Benchmark comparison of code predictions to integral system test

results are also presented.

Revision 2 deals mainly with the small break LOCA. Revision 3 includes enhancements to the evaluation model for fuel pin heat transfer modelling to extent the code applicability to the once through steam generators.

3.0 EVALUATION

Review and evaluation of the RELAP5/MOD2-B&W code includes Revision 2 and Revision 3 of BAW-10164P. Revision 2 provides additional models specifically intended for SBLOCA applications. Revision 3 includes enhancements to the EM fuel pin and heat transfer model, additions and benchmarks which extend application of the code to the once through steam generator plants. An initial review of Revisions 2 and 3 led to generation of Requests for Additional Information[8,9]. Supplemental information was also submitted by BWNT during the review process[10]. Each of the model additions or modifications is discussed and evaluated in the following sections.

3.1 BAW-10164P, Revision 2

3.1.1 Model Changes for the Slug Flow Regime

BWNT added an option for determining the Taylor bubble interphase drag during slug flow based on the Wilson drag model. The Wilson drag model is based on the Wilson bubble rise velocity in a vertical pipe. BWNT applied the Wilson drag model for reflood applications using the BEACH program and is now applying the model for non-reflood applications in RELAP5/MOD2-B&W. These changes are discussed on pages 2.1-51 to 2.1-54 of BAW-10164P. Benchmarks are provided in Appendix H.

In implementing the Wilson drag model for RELAP5/MOD2, BWNT derived an expression for the interphasic friction for Taylor bubbles. Flow was assumed to be in a quasi-steady state. The derivation of this expression was checked. The formulation was determined to be correct. BWNT also incorporated improvements to match the bubble rise data at higher void fractions. An apparent difference between the interphase friction model for slug flow used in RELAP5/MOD2 B&W compared to that used in the BEACH[7] program was questioned. BEACH uses the same Wilson drag model with a different multiplier, on the Taylor bubble term. BWNT responded that the different multipliers were selected based on comparisons to reflood benchmarks in the case of BEACH and small break LOCA benchmarks in the case of RELAP5/MOD2-B&W.

An option to remove smoothing in selected junctions, (not used in BEACH), was added to RELAP5/MOD2-B&W. This option allows smoothing to be bypassed. Since discontinuous void distribution may occur during a small break LOCA, use of this option for small break calculations may be appropriate.

Benchmarks were performed by BWNT using the Wilson drag model against results obtained from the NRC-approved computer code FOAM2[11] and with small break LOCA experiments performed at the Thermal Hydraulic Test Facility (THTF) at the Oak Ridge National Laboratory. These benchmarks are presented in Appendix H to BAW-10164P.

The FOAM2 program, developed by BWNT and previously approved by the NRC, is used to determine whether the water content of a reactor core is sufficient to cover the core with a combination of liquid and two phase mixture based on a given core void distribution. If it is determined that the core is uncovered, FOAM2 calculates the two-phase swell level and steaming rate. The FOAM2 program utilizes the Wilson bubble rise correlation to

directly calculate the core void distribution. The Wilson bubble velocity correlation used in FOAM2 is somewhat different from that used in RELAP5/MOD2-B&W since it does not include the changes made by BWNT to better match the bubble rise data at higher void fractions. BWNT stated that the core void distribution results calculated by RELAP5/MOD2 B&W and FOAM2 should be similar except potentially at higher void fractions because of differences in the formulation of the Wilson model. The benchmarks show that RELAP5/MOD2-B&W predicts void distributions which are comparable to FOAM2 predictions.

Calculations using FOAM2 were performed for reactor powers of 1.5, 2.5, and 5.0 percent of full power. System pressures ranging from 100 to 1600 psia were included in the analysis. BWNT presented plots of core void fraction vs. core elevation comparing the RELAP5/MOD2-B&W results and the FOAM2 results. These plots show acceptable agreement between RELAP5/MOD2-B&W and FOAM2, with RELAP5/MOD2-B&W showing very slightly different results at high void fractions.

The RELAP5/MOD2-B&W code predictions were compared to THTF test results for a range of pressures (520 to 1170 psia), power densities (0.08 to 0.68 kw/ft), and mass flux (3395 to 21943 lbm/hr-ft^2). BWNT presented plots of the results of core void fraction vs. core elevation as well as rod surface and vapor temperature vs. core elevation comparing the RELAP5/MOD2-B&W results and the THTF test results. In general, the void fractions predicted by RELAP5/MOD2-B&W are somewhat higher than the THTF results. Additionally, RELAP5/MOD2-B&W generally overpredicts the vapor and surface temperature relative to the THTF tests. There is a dip in surface temperature in the THTF tests at the core elevation of 11 feet. BWNT attributes the dip to grid effects on the heat transfer rate which are not accounted for in the RELAP5/MOD2-B&W model. The dip in surface temperature

caused by the grid effect, was questioned. The surface temperature would be underpredicted for THTF tests 3.09.10 i, j, l, and m. Underprediction of the surface temperature could indicate that the heat transfer coefficient to the vapor is high and non-conservative. It was requested that BWNT discuss the comparisons between RELAP5/MOD2-B&W and the THTF tests and show that the comparisons do not rely on systematic underprediction of the vapor temperature. In their response, BWNT indicated that the heat transfer coefficient for single phase vapor is computed using the McEligot-plus radiation correlation set. BWNT pointed out that use of the correlation is widely accepted and was reviewed and accepted in previous submittals of RELAP5 and FRAP-T6 topical reports. BWNT also noted that the measurement of vapor temperature is low because the thermocouples were mounted on the unheated rods used to simulate guide tubes. Actual temperatures measured at THTF would be higher and more in agreement with RELAP5/MOD2-B&W. The BWNT response was deemed satisfactory. This benchmark shows that use of the Wilson model for interphase drag produces reasonably accurate predictions of SBLOCA experimental data.

3.1.2 Model Changes for the Annular Mist Flow Regime

BWNT has added an option to RELAP5/MOD2-B&W to include calculation of the overall drag computed for control volumes in an annular mist flow regime. This change is used in the OTSG and MIST benchmarks discussed in Sections 3.2.7 and 3.2.8 of this evaluation.

3.1.3 Counter-Current Flow Limiting Model

BWNT added optional counter-current flow limiting (CCFL) models which are intended for use in predicting flows at the steam generator U-tube inlets and steam generator plenum inlets during the reflux condensation period of a SBLOCA. Addition of the CCFL

model is described on pages 2.1-133 to 2.1-133.3 of BAW-10164P.

The CCFL model modifications, consisting of a correlation for flooding in vertical tubes, are included in the form of a general relationship between the dimensionless vapor flux, j_g^* and the dimensionless liquid flux, j_f^*. The relationship is implemented in the code in a manner similar to the implementation in RELAP5/MOD3. Different values of the correlation parameters are used at the U-tube inlets and at the steam generator plenum inlet. This model was benchmarked against ROSA-IV small break LOCA data in Appendix J of the Topical Report. Flow predictions were in reasonable agreement with the test data. Therefore, the use of the CCFL model at the steam generator plenum and tube inlet, with the parameters used in the benchmark, is acceptable for the analysis of SBLOCAs in recirculating steam generators. As demonstrated by the fact that different correlation parameters are required at the inlet plenum and the tubes, CCFL is very geometry specific. Other uses of the CCFL model will require that the model be validated for that application.

3.1.4 Condensation Heat Transfer Correlation Modifications

Modifications were made to the condensation heat transfer correlation for vertical or horizontal surfaces. These changes are discussed on pages 2.2-31 and 2.2-32 of BAW-10164P.

The Nusselt laminar film correlations for a horizontal surface and for a vertical surface are used in RELAP5/MOD2-B&W. For condensation on a horizontal surface, laminar film condensation in a horizontal tube is assumed. Comparing the formulation given in Collier[12] against that given by BWNT for condensation within a horizontal tube shows that BWNT did not include the equation developed by Rohsenow for the modified latent heat of vaporization. Omission of this equation will have a small effect

for the heat transfer coefficient result. The expression for a vertical surface was found to be in agreement with Collier.

3.1.5 Changes to the Metal-Water Reaction (Swelled Radius) Model

When a fuel rod swells the radius and hence the surface area of the rod will increase in the swelled region. The rate of heat generation and the molar production rate of hydrogen are proportional to the exposed surface area of the clad. These models have been modified by BWNT as described in Section 2.3.2.4 of BAW-10164P to increase the surface area in proportion to the ratio of the swelled clad radius to the cold clad radius. This increase in area is applied to both the clad inside and outside surfaces. Consideration of the increase in clad radius in the swelled region is appropriate and conservative. Both the energy generation rate and the rate of hydrogen production will increase when this model is used compared to the constant surface area model. This model satisfies the requirements of Appendix K and is acceptable for ECCS EM calculations.

3.1.6 Core Heat Transfer Selection Model Modifications

BWNT installed a separate heat transfer option for use in SBLOCA analysis. The changes are discussed on pages 2.3-60 to 2.3-61.2, 2.3-64, 2.3-67, 2.3-83, and 2.3-84 of BAW-10164P. The changes to the switching logic for SBLOCA include the removal of Appendix K restrictions regarding no return to nucleate boiling and the lock into film boiling after the wall superheat exceeds 300° F. The switching logic is unchanged for LBLOCA. BWNT stated that the no return to nucleate boiling and the lock into film boiling restrictions of Appendix K are not applicable to SBLOCA. This assertion is acceptable. While the acceptance criteria of 10 CFR 50.46 apply to LBLOCA, the possible core heatup scenarios following a SBLOCA are varied and more complex than those for the

LBLOCA. The use of a reflood heat transfer model lockout of return to nucleate boiling and prediction of quenching is inappropriate for the small break.

An option to allow use of the BWUMV critical heat flux correlation can be selected by the user depending on the fuel design being evaluated. This correlation is used only at pressures greater than 1300 psia at mass fluxes greater than 500,000 lbm/hr-ft^2 in the core heat transfer selection logic in RELAP5/MOD2-B&W. The BWUMV correlation is reviewed in the next section.

3.1.7 BWUMV Critical Heat Flux Correlation

The BWUMV (B&W Universal Mixing Vane) critical heat flux correlation was developed for the analysis of SBLOCA. The development of the BWUMV correlation is presented in Appendix I to BAW-10164P. BWNT developed this correlation from the database for the previously approved BWCMV correlation[13] with additional mid-flow regime data from three Westinghouse tests.

BWUMV utilizes a third order polynomial fit using three independent variables based on pressure (P), mass flux (G), and quality (X_{eth}). Typographical errors in the CHF equation, units conversion errors, and the FLS equation, were questioned. The responses with the corrected equations and units are provided in Reference 8.

The number of data points and their distribution in the mass flux range below 0.95×10^6 lb/hr-ft^2 was questioned. BWNT indicated in their response that the total number of data points with mass fluxes below 1.25×10^6 lb/hr-ft^2 is 77. In addition to the 22 points from Table I.3 of Appendix I, 20 points are from the data presented in Appendix B of BAW-10159 and 35 test points are from

the data presented in Appendix F of BAW-10159. Of these 77 points, a total of 32 points were measured at a local mass flux of between 0.4×10^6 and 1.0×10^6 lb/hr-ft^2. This total is comparable to the number of points at the lower end of the flow range of the BWCMV correlation.

In evaluating the distribution of points about the measured/predicted ratio for pressure and mass flux, it is noted that the data clustered about 750 psia and 1000 psia on Figure I.5 are not uniformly distributed about the mean value of 1.0. It appears that 10 of the 11 points measured at 750 psia are below the mean and 10 of the 12 points measured at 1000 psia are above the mean. Because of this bias and because of the small number of points, we believe that the BWUMV results are biased at pressures of 750 and 1000 psia. It is recognized that this bias does not affect BWUMV predictions above 1300 psia. However, BWNT did conclude in Appendix I to BAW-10164P that the BWUMV correlation is applicable to CHF calculations for pressures and flow rates above 750 psia. Given the apparent bias in the data points at 750 and 1000 psia, the BWUMV correlation should not be used for CHF calculations at pressures below 1300 psia.

Numerical checks of the BWUMV correlation were performed. These calculations were done to determine how well the BWUMV correlation reproduces predicted results, to determine the behavior of the BWUMV correlation over a range of pressure, flow rate and quality, and to compare the BWUMV to the BWCMV correlation since they were developed from the same database for the most part.

In general, the numerical checks show that the BWUMV and the BWCMV correlations are in agreement within the statistical uncertainty band. Some differences in the results are noted in cases where the flow is varied over the BWCMV range of validity.

These differences are more significant when the pressure is at 1500 and 1800 psia. Evaluations performed where the pressure is varied over the BWCMV range of validity also show some difference between the two correlations.

The statement is made in Appendix I that the Tong factor is set equal to one in the RELAP5/MOD2 B&W implementation of BWUMV. In discussions with BWNT they stated that the Tong factor is included in the development of the BWUMV correlation. It is during the SBLOCA transient analysis that the Tong factor is set equal to 1. BWNT also indicated that their standard practice is to set the Tong factor equal to 1 for LOCA analysis and indicated several references where this practice has been previously approved.

Based on this review, it is concluded that the BWUMV correlation is acceptable for used in RELAP5/MOD2 B&W subject to the restriction that the correlation should not be used below pressures of 1300 psia.

3.1.8 SBLOCA EM Benchmark

BWNT performed a benchmark using RELAP5/MOD2 B&W against a SBLOCA experiment performed at the ROSA-IV facility in 1988. The ROSA-IV facility simulates a recirculating steam generator plant. The results of this benchmark are presented in Appendix J to BAW-10164P. It is important to note that this is not the only benchmark of RELAP5/MOD2-B&W for a SBLOCA. As noted in appendix J, the peak clad temperatures during the experiment are not significant relative to the acceptance criteria. Benchmarks against LOFT and Semiscale SBLOCA data, presented in Section 5 of BAW-10164, provide additional coverage of SBLOCA phenomenology, including clad temperature prediction. This benchmark serves to show that the additional models for SBLOCA, such as CCFL and Wilson drag, are performing correctly and will adequately predict

test data.

The RELAP5/MOD2-B&W model for ROSA-IV, and other benchmarks and applications models, use equilibrium thermodynamics nodes in the core region and non-equilibrium nodes in the remainder of the system. Since the RELAP5/MOD2 Code Manual specifically states that equilibrium nodes cannot be connected to non-equilibrium nodes, BWNT was questioned on node connectivity. In their response BWNT stated that RELAP5/MOD2 has always been fully capable of connecting equilibrium and non-equilibrium nodes, and that this is consistent with the approved EM core modeling. This was confirmed by INEL (the RELAP5 code development organization), that the equilibrium option is obtained using the same basic equations with the interphase heat transfer coefficient set to a very high value. This assures that sufficient heat transfer will occur to keep both phases at saturation conditions. Contrary to the statements in the code manual, it is possible to connect equilibrium and non-equilibrium nodes without adversely affecting the calculational algorithms. BWNT uses equilibrium nodes in the core region to obtain saturation fluid temperature as the boundary condition for the core heat transfer correlations, which were developed on this basis.

When modeling complex systems, it is sometimes necessary to slightly modify the representation to compensate for code model limitations. BWNT used two such modifications in their representation of the ROSA-IV facility. First, the friction factor for the accumulator injection line was increased by a factor of 100 above the nominal value, to eliminate unrealistic injection flow oscillations. Also, the angle of the inlet pipe to the steam generator plenum was decreased to less than 15 degrees to permit use of the horizontal pipe stratification model. BWNT provided justification for these model modifications. In both cases, the BWNT responses adequately explained the need for modifications to overcome code

limitations.

BWNT performed a time step sensitivity study which demonstrated that the 0.05 second time step used for the calculations was adequate.

Results of the calculations showed that the basic thermal-hydraulic phenomena which occurred during the SBLOCA were predicted with reasonable accuracy. Key events were predicted to occur in essentially the correct sequence. Differential pressures in the core and recirculation loops, key determinants of the flows, were adequately predicted. This benchmark further demonstrates that the RELAP5/MOD2-B&W computer program is capable of predicting the important thermal-hydraulic phenomena which occur during a SBLOCA in a recirculating steam generator plant.

3.2 BAW-10164P, Revision 3

3.2.1 Revision to the Slug Flow Drag Model

BWNT incorporated a third option for evaluating the Taylor bubble interphase drag in slug flow. The three models now available are the base INEL model, the Wilson drag model submitted with Revision 2 of BAW-10164P, and the B&W modified slug-drag model, as described on pages 2.1-52.4 and 2.1-52.5 of Revision 3 of BAW-10164P. (The Wilson drag model was reviewed in Section 3.1.1 above).

In the B&W modified slug-drag model, adjustments are made to the interphase friction terms through the use of empirically derived coefficients. BWNT states that these adjustments were based on numerous benchmarks. BWNT listed the benchmarks used which included those documented in Appendices K and L to BAW-10164P. These benchmarks are discussed in Sections 3.2.7 and 3.2.8 of this evaluation.

3.2.2 Modifications to the Chen Heat Transfer Coefficient

The saturated nucleate boiling heat transfer coefficients used in RELAP5/MOD2-B&W are calculated using the Chen heat transfer correlation to model the boiling component of the heat transfer coefficient. The Chen heat transfer correlation employs a nucleate boiling suppression factor S. S was modified by including a multiplicative weighing factor to force S, and the boiling heat transfer coefficient to zero as the steam void fraction approaches one. B&W notes that this ramp is needed for once-through steam generator applications to preclude sharp increases in S that result in disproportionally high overall heat transfer as the void fraction approaches one.

3.2.3 Incorporation of the Becker CHF Model

The Becker CHF correlation for rod bundles is incorporated into RELAP5/MOD2-B&W as described in Section 2.2.2.2 of Revision 3 of BAW-10164P. BWNT states that this correlation was used to obtain better predictions of secondary side heat transfer relative to the Biasi-Zuber correlation used in RELAP5/MOD2 at power levels below 80 percent of full power. The Becker CHF correlation is used up to a pressure of 90 bar (1306 psia). Linear interpolation between the Becker and Biasi-Zuber correlations is performed between 80 and 90 psia to obtain a smooth transition between the two correlations.

Benchmarks comparing results of RELAP5/MOD2-B&W to OTSG test data are presented in Appendix K of BAW-10164P and are discussed in Section 3.2.7 of this SER. These benchmarks show good agreement between RELAP5/MOD2-B&W results using the Becker CHF correlation and the OTSG test data as discussed in Section 3.2.7.

3.2.4 EM Pin Model Modifications

BWNT enhanced the fuel pin model in RELAP5/MOD2-B&W by adding features which had previously been approved for use in the FRAP-T6-B&W[14] and TACO3[15] computer programs. These changes are discussed in Section 2.3.2 of Revision 3 of BAW-10164P. There are three basic areas in which the fuel pin model provides calculations: dynamic fuel/clad gap conductance, fuel rod swelling, and rupture based on the NUREG-0630[16] approach and clad metal-water reaction. Enhancements in Revision 3 include addition of a closed gap contribution to gap conductance to allow modeling of high burnup cases, fuel pin axial expansion, automated clad rupture calculation, implicit metal-water reaction option and automated steady-state gap multiplier option.

The RELAP5/MOD2-B&W enhanced fuel pin model has some differences compared to the previously approved models in FRAP-T6-B&W and TACO3. These are due to the less detailed nature of the model used in RELAP5/MOD2-B&W compared to the fuel performance codes.

The previously approved gap conductance model allowed for modeling of a non concentric fuel stack within the clad, but did not include a contact conductance term in the calculation of total gap conductance. The option to include a contact resistance contribution has been added based upon the model presently used in the TACO3 computer program. Since Appendix K does not prohibit the use of contact resistance, this model is acceptable for cases of high burnup fuel with a closed gap.

The gap gas pressure is used to determine the amount of clad swelling and rupture. One of the variables which determines the gap gas pressure is the amount of volume available for the gas to occupy. A portion of this volume is located in the gas plenum at the top of each fuel rod. When the clad and fuel expand axially

at different rates the volume of the plenum will change. The axial expansion model accounts for this change in plenum volume due to differential thermal expansion of the fuel and clad from the cold condition. Thermal strain correlations from MATPRO[17], or user input fits, are used to determine the amount of fuel and clad axial expansion. Including the effects of axial fuel rod expansion on plenum volume is appropriate for a best-estimate calculation, and is acceptable for Appendix K to 10 CFR 50 LOCA analysis, which does not prohibit including this model. Only minimal differences exist between the enhanced fuel pin model and the FRAP-T6-B&W model. The enhanced model is acceptable for use in ECCS evaluation model calculations.

Addition of an automated clad rupture model introduces heat transfer enhancement downstream of a rupture location. The model used for this option has been previously approved for use in the BEACH computer program. This option automates the calculation of rupture location for use in BEACH. Use of this model in RELAP5/MOD2-B&W is consistent with the Appendix K to 10 CFR 50 approved model and is, therefore, acceptable.

An implicit formulation of the Baker-Just metal water reaction model has been implemented in RELAP5/MOD2-B&W. The model is the same as that which is used in the FRAP-T6-B&W computer code. Appendix K to 10 CFR 50 requires use of the Baker-Just model but does not specify the numerical solution technique to be used. The implicit solution will improve accuracy for calculations which use a larger time step. Use of this option is acceptable for compliance with Appendix K to 10 CFR 50.

Initialization of the RELAP5/MOD2-B&W model must comply with Appendix K to 10 CFR 50 in the sense that initial stored energy in the fuel must be conservatively specified. This is done by adjusting the fuel clad gap conductance to obtain the desired

value of volume average fuel temperature. In earlier versions of the program a user specified multiplier on the gas conductance term of the gap conductance was provided to permit adjustment and matching of volume average fuel temperature. This required that an iterative process be performed by the user. An option has been added to automate this iteration process. The user can specify the desired volume average fuel temperature, rather than the multiplier on the gas conductance term, and the code will iterate to determine the gas conductance multiplier. This is a user convenience feature which is acceptable for ECCS evaluation model calculations.

3.2.5 EM Heat Transfer Model Modifications

BWNT incorporated a filtered flow option to be used with the core heat transfer model. This option was added to facilitate addressing the Appendix K to 10 CFR 50 requirement to eliminate any calculated rapid flow oscillations with a period of less than 0.1 seconds during the LBLOCA blowdown phase. This requirement is mandated because rapid flow oscillations can cause overprediction of the amount of energy removed from the core which would cause peak clad temperature to be underpredicted. The method, described in Section 2.3.3 of BAW-10164P, is consistent with the requirements of Appendix K to 10 CFR 50.

3.2.6 Post-CHF Model Modifications

User defined correction factors were added to the equation used to determine the total wall-to-fluid heat flux due to transition boiling as shown on page 2.3-86 of Revision 3 to BAW-10164P. These constants are defined as $C1_{TR}$ and $C1_{TR1}$ which have a default value of 1.0. BWNT did not identify any benchmarks that used values other than 1.0 for these correction factors.

A user option was added to allow adjustment of the surface heat transfer coefficient following cladding rupture of a fuel pin on page 2.3-89 of Revision 3 to BAW-10164P. If this option is invoked, the heat flux in a ruptured segment single heat structure is multiplied by the ratio of the ruptured to cold outside cladding radius. The heat flux is then computed using the increased surface area resulting from clad swelling and rupture. This change is physically realistic and does not violate Appendix K requirements. It is, therefore, acceptable.

3.2.7 OTSG Benchmarks

BWNT performed two benchmarks of the RELAP5/MOD2-B&W code to demonstrate the adequacy of the OTSG modelling. These benchmarks are discussed in Appendix K to BAW-10164P. The first set of benchmarks are comparisons to steady-state tests performed in 1969 to demonstrate the ability of the code to predict the shell side nucleate boiling length at various power levels. The second benchmark is a comparison to a loss of feedwater flow test performed in 1977 to demonstrate the ability of the code to predict boil-down and refill of a OTSG.

The tests were performed at the Alliance Research Center (ARC) Nuclear Steam Generator Test Facility. This facility provided the capability of testing steam generators at full system pressure and temperature conditions. The primary side of the test loop consisted of a gas-fired furnace to simulate reactor heat input into the primary fluid, a pressurizer, flow control valves, flow measuring elements, and a water conditioning system. The secondary system was a closed circuit test loop consisting of steam flow control valves, steam flow measuring equipment, feedwater heaters, back pressure control valves, a flash tank, circulating pumps, feedwater control valves, feedwater flow measuring equipment, feedwater flow bypass valves, and a water conditioning system.

The model steam generator used for these benchmarks, referred to as the 19-tube OTSG, is a single pass, counterflow, tube and shell heat exchanger. The tube bundle consisted of 19 full length tubes, each 5/8 inch diameter, spaced on a triangular pitch on 7/8 inch centers. Primary inlet flow entered at the top of the steam generator, flowed downward through the tube bundle and exited at the bottom. Secondary feedwater flow entered the tube bundle at the bottom, was boiled as it passed by the outside of the tube bundle, and exited at the top. The feedwater was raised to saturation conditions by mixing the water with steam from the tube region via a steam bleed pipe connected from the secondary side of the generator to the steam/feedwater mixer. A question on where the feedwater temperature is measured was raised. BWNT stated that the feedwater temperature is measured with a thermocouple located approximately three feet upstream of the steam/feedwater mixer.

In the steady state tests, boiling length (dryout location) was determined from primary tube and secondary side thermocouples for a range from 0 to 100 percent of the full scaled power consistent with a 2700 MWth plant. The loss of feedwater flow test initialized to full scaled power consistent with a 2772 MWth plant. The test was initiated by the simultaneous trip of the feedwater pump and closure of the feedwater isolation valve. The steam generator was allowed to boil dry and then the feedwater was restarted. Secondary steam flow and temperature and primary outlet temperature were measured during the test.

The RELAP5/MOD2-B&W model utilized 11 axial control volumes in the primary tube region and in the secondary shell region. Primary to secondary heat transfer was modelled using eleven heat structures between the primary and secondary sides. The external downcomer was modeled with five axial control volumes that represented the piping from the steam/feedwater mixing region to the tube bundle inlet. Feedwater aspiration was provided by a

single junction component that connected the tube bundle region to the external downcomer. A junction connection between the shell side of the heat exchanger and the control volume representing the steam/water mixer is included in the model. Time dependent volume and time dependent junction components were used to set inlet flowrate and temperature of the primary and secondary side coolant.

A question on the modeling of the steam/feedwater mixing or aspiration process, addressed the relative pressures between the control volume representing the mixer and the volume representing the source of aspiration steam from the secondary to the mixer. Assuming a normal flow junction connecting these volumes, the pressure in the secondary must be higher than the mixer in order to provide steam flow to the mixer. BWNT responded that the bundle collapsed liquid level remains below the downcomer level (presumably below the level of the steam/feedwater mixer) during the tests. As a result the pressure gradient that sustains the steam flow from the secondary is maintained by manometric effects. BWNT also noted that this process is self-governing due to changes in the pressure gradient as the downcomer fluid approaches saturation.

BWNT employed some of the features incorporated into RELAP5/MOD2-B&W for the 19-tube OTSG benchmark. These features are summarized below:

* The Becker critical heat flux correlation, discussed in Section 3.2.3, are used on the shell side of the tube heat structure,
* The multipliers defined by the B&W modified slug-drag model and the annular mist model are used, and
* A linear ramp was applied to the Chen boiling suppression

factor. This adjustment to the suppression factor was discussed in Section 3.2.2.

The first set of results for the steady state benchmark compared the boiling lengths predicted by RELAP5/MOD2-B&W to those measured during the 19-tube OTSG tests. BWNT presented tabulated results and a plot comparing the boiling length above the lower tube sheet predicted by RELAP5/MOD2-B&W to the 19-tube OTSG tests at power levels ranging from 20 to 100 percent of the scaled power levels relative to a 2700 MWth plant. The results show that the boiling lengths predicted by RELAP5/MOD2-B&W are in good agreement with the test data. In contrast, results from RELAP5/MOD2 Cycle 36.05 also shown on these plots, differ significantly from the test data below 80 percent scaled power. BWNT attributes this agreement in the boiling length results between RELAP5/MOD2-B&W and the 19-tube OTSG test to the use of the Becker critical heat flux correlation.

For the LOFW benchmark, BWNT presented comparison plots between RELAP5/MOD2-B&W and 19-tube OTSG data for steam flow rate and the primary outlet temperature after the initiation of the LOFW transient. Plots of the primary and secondary system fluid temperatures prior to the initiation of the LOFW test are also presented to show initial conditions. The plots of steam flow rate and primary outlet temperature show that the magnitude and trend of the results are in good agreement between RELAP5/MOD2-B&W and the 19-tube OTSG test.

BWNT was requested to provide comparison plots between RELAP5/MOD2-B&W and 19-tube OTSG data comparing steam generator secondary level or mass. In their response, BWNT noted that determination of collapsed liquid level is difficult to obtain under two phase high flow conditions. BWNT did provide comparisons of the steady state and transient differential pressure in the boiler. The response of BWNT to this question

was reviewed and found adequate.

3.2.8 MIST Benchmarks

As part of their evaluation of the Revision 3 modifications to the RELAP5/MOD2-B&W code, BWNT included an integral system benchmark of a SBLOCA for a simulated reactor system using OTSGs. BWNT included this benchmark as a further check of the modifications made to reduce interphase drag in the slug flow regime (Wilson bubble rise model) and in the annular flow regimes.

The integral system benchmark was performed using data from the Multi-Loop Integral System Test (MIST) facility which is a scale model of a B&W lowered-loop 177 fuel assembly pressurized water reactor. The MIST facility is designed to operate at pressures and temperatures typical of an operating B&W plant. The MIST facility consists of two 19-tube once-through steam generators, a reactor vessel with a heated core and external downcomer, pressurizer with a power operated relief valve, two hot legs and four cold legs. Further information on facility scaling and instrumentation is found in Appendix L to BAW-10164P.

BWNT notes in Appendix L to BAW-10164P, that a number of pre- and post-test predictions have been made for MIST tests using the RELAP5/MOD2 code as a part of the MIST program. These predictions were made with earlier versions of the code, which did not include the recent modifications. The benchmark presented in Appendix L includes a comparison of experimental data with RELAP5/MOD2 results obtained with both the current and earlier versions of the code. This allows an evaluation of the effects of the recent model changes, in particular upon the prediction of collapsed liquid level in the reactor vessel and steam generator secondary. The results show that the code modifications clearly improve the collapsed liquid level

predictions.

The RELAP5/MOD2-B&W model simulates the MIST reactor vessel, downcomer, hot and cold legs, OTSGs, reactor coolant pumps, and other major components. A double flow path connection to the external reactor vessel downcomer is used in this model so that countercurrent two-phase flow can be predicted. The steam generator modelling employs two radial regions to account for tubes directly wetted by auxiliary feedwater (AFW) injection on the shell side of the steam generator. The other region represents the 16 tubes in contact with secondary steam. BWNT notes in their response that the BWNT modified slug-drag model was employed in the core and primary tube region. For the secondary side of the steam generator tube region, the BWNT slug-drag model is used on the annular mist drag.

In the RELAP5/MOD2-B&W model, the core region was modified so that twenty control volumes represent the full height core. BWNT states that this nodding is necessary for consistency with the revised models. BWNT was requested to discuss the implications of increasing the number of core volumes focusing on the degree of improvement resulting from the use of the new interphase drag model as compared to just increasing the number of core nodes.

BWNT performed a nodding sensitivity study based on ORNL THTF Test 3.09.10j discussed in Appendix H to BAW-10164P. In this study, BWNT ran cases using 4 and 24 nodes in the core region using both the INEL drag model and the BWNT slug-drag model. Plots presented by BWNT show good agreement between the RELAP5/MOD2-B&W results using the BWNT slug-drag model for both the 4 and 24 node core models and the THTF data. The results from the INEL model show poor agreement with the THTF data. BWNT notes in their response that the INEL model is known to overpredict the interphase drag in heated regions with small hydraulic diameters. In this context, the BWNT response

regarding the nodalization of the steam generator secondary was reviewed and is acceptable.

MIST test 320201, used for the RELAP5/MOD2-B&W benchmark, simulated a scaled 50 cm^2 pump discharge break. This is reported by BWNT to be the most limiting small break size for B&W designed plants. The MIST facility was initialized in natural circulation mode with the core power scaled to 3.5 percent for this test. BWNT noted that since the MIST facility was capable of only ten percent full-scaled power operation, the facility was initialized to conditions corresponding to 145 seconds after trip. Other initial conditions include primary system pressure corresponding to 22 °F core exit subcooling, pressurizer level of 5 feet above the bottom of the pressurizer, steam generator pressure of 1010 psia, and a steam generator secondary level controlled to five feet above the lower tube sheet by throttling high elevation AFW injection.

The MIST test was initiated by turning off the pressurizer heaters and opening the leak. When the pressurizer level reached one foot, full high pressure injection flow was started and steam generator secondary refill using full capacity AFW was initiated.

BWNT presented comparison plots between RELAP5/MOD2-B&W and MIST for primary pressure, secondary pressure, reactor vessel liquid level and secondary liquid level. A tabulation of the timing of key events is presented in Table L.2 of BAW-10164P. Additional information was provided on the elevation of the steam generators and reactor vessel at the MIST facility. The results for BWNT Versions 5 and 14 of RELAP5/MOD2-B&W are presented. The main difference in these code versions is that Version 14 utilizes the revised interphase drag models.

In general, there is good agreement between the RELAP5/MOD2-B&W

Version 14 results and the MIST data. BWNT noted that the improved prediction of reactor vessel and secondary liquid levels is due to the revised slug drag model. The large difference in timing of the hot leg voiding between Revisions 5 and 14 of RELAP5/MOD2-B&W was questioned. BWNT attributed this to differences in the initial temperature conditions in the pressurizer and surge line used in the Revision 5 and Revision 14 calculations. BWNT also revised Table L.2 which corrects the reported time of operator actions.

In reviewing the MIST benchmark, it is noted that no fuel rod temperature excursion occurred during this test. This issue was discussed with BWNT. BWNT responded that the mixture level in the vessel remained above the core during the test, resulting in removal of the stored energy in the fuel by nucleate boiling. In view of this lack of core uncovery, BWNT noted that MIST is not the best benchmark for the core mixture level calculation. The stand-alone FOAM2 and ORNL benchmarks presented in Appendix H to BAW-10164P were provided to better address the code's predictive capability.

MIST benchmarks address the capability to predict system pressure, liquid inventory and liquid distribution. In this respect, the MIST test 320201 benchmark demonstrates that the modifications made to the code improve its predictive capability in these areas. When considered along with the other benchmarks, including additional MIST benchmarks, BWNT has demonstrated that RELAP5/MOD2 B&W is capable of adequately predicting the system response for an OTSG plant.

In discussing cladding temperature during an SBLOCA, BWNT argues that the cladding temperature excursion during SBLOCA is governed by the same physical processes for all current PWR designs. During core boildown, the vapor temperature and fuel cladding heat transfer are dependent upon local mechanisms and not upon

integral system parameters. Therefore, heatup during core boiloff depends upon the core geometry, but not the integral system geometry. Benchmarks against ROSA-IV, ORNL and Semiscale test data are hence acceptable for demonstrating that RELAP5/MOD2-B&W is capable of adequately predicting fuel clad temperatures during this heatup phase.

3.3 High Auxiliary Feedwater Model

BWNT developed the high auxiliary feedwater model to calculate the B&W OTSG heat transfer correctly during auxiliary feedwater injection from high elevation locations. This model was included in the Revision 1 submittal but was not reviewed because B&W did not intend to use RELAP5/MOD2-B&W for OTSG analysis at that time as noted in Section 5 of the Revision 1 safety evaluation report. BWNT requested a review of this model and has provided additional information to support this request.

The high auxiliary feedwater model was reviewed as part of the CRAFT2 code Topical Report transmittal in 1985. This report included several benchmarks against plant and experimental data. BWNT has made some modifications to the heat transfer models since that time. Therefore, an additional RELAP5/MOD2-B&W benchmark of the revised heat transfer models against test data from the MIST facility was provided by BWNT. Included is a plot comparing RELAP5/MOD2 predicted primary side temperature as a function of elevation above the tube sheet against steady-state MIST measurements. This plot shows good agreement between the RELAP5/MOD2 B&W predictions and the MIST measurements along the length of the steam generator tubes. The steady-state nature of this test allowed the calculated heat transfer coefficients used in the model to be justified. The high AFW model was also used for the additional MIST benchmarks performed by BWNT. These serve to demonstrate the adequacy of the model for transient applications.

Based upon the discussions and benchmarks provided by BWNT, it is concluded that the revisions to the high auxiliary feedwater model have not changed significantly from the model previously accepted. Therefore, we find it acceptable for use in SBLOCA analysis.

3.4 Parameters Used in RELAP5/MOD2-B&W Benchmarks

BWNT has incorporated a number of user specified input parameters in RELAP5/MOD2-B&W. The value of these parameters used in the RELAP5/MOD2 B&W benchmarks are listed in Table 1 of this evaluation. The values of these parameters were selected to improve the agreement between RELAP5/MOD2-B&W and the FOAM2, THTF, OTSG, and MIST benchmark data. The values of the user specified parameters listed in Table 1 are the only acceptable values for LOCA licensing calculations.

One of the motivations driving the initial development of the RELAP5 code was the need to eliminate user choice of modeling options and input dials. When an analysis is performed with RELAP5, one can be certain of what models and fitting parameters are used. The addition of numerous options and dials by BWNT is contrary to this characteristic of RELAP5. The last condition listed above is intended to eliminate the use of user specified dials in keeping with the intent of the RELAP5 approach.

Table 1
Summary of User Specified Parameters Used in the
RELAP5/MOD 2-B&W Benchmarks

Parameter	Description	Page Where Parameter Is Described	Affected Benchmark
C_{WB}	Multiplier on bubble velocity in the Wilson Bubble Rise Model	2.1-52.3 (Revision 2)	FOAM2 code, ORNL THTF tests (Appendix H)
C_{WSL}	Multiplier on the interphase drag predicted for slug flow	2.1-52.3 (Revision 2)	FOAM2 code, ORNL THTF tests (Appendix H)
xms	Multiplier on the interphase drag computed for annular mist flow	2.1-53 (Revision 2)	OTSG benchmarks (Appendix K), MIST benchmarks (Appendix L)
xsg	High void fraction slug coefficient	2.1-52.5 (Revision 3)	OTSG benchmarks (Appendix K), MIST benchmarks (Appendix L)

Table 1
Summary of User Specified Parameters Used in the
RELAP5/MOD 2-B&W Benchmarks

Parameter	Description	Page Where Parameter Is Described	Affected Benchmark
α_s	Low end of void fraction for adjustment	2.1-52.5 (Revision 3)	OTSG benchmarks (Appendix K), MIST benchmarks (Appendix L)
α_{sh}	High end of void fraction for adjustment	2.1-52.5 (Revision 3)	OTSG benchmarks (Appendix K), MIST benchmarks (Appendix L)
xslg	Slope of drag pressure term	2.1-52.5 (Revision 3)	OTSG benchmarks (Appendix K), MIST benchmarks (Appendix L)
cxslg	x-intercept for pressure term	2.1-52.5 (Revision 3)	OTSG benchmarks (Appendix K), MIST benchmarks (Appendix L)

Table 1
Summary of User Specified Parameters Used in the
RELAP5/MOD 2-B&W Benchmarks

Parameter	Description	Page Where Parameter Is Described	Affected Benchmark
α_{gr}	Void fraction at which to begin the S ramp	2.2-22 (Revision 3)	OTSG benchmarks (Appendix K)
$C1_{TR}$, $C1_{TR1}$	User multiplicative constants used in the equation to determine wall-to-fluid heat flux during transition boiling.	2.3-86 (Revision 3)	

4.0 COMPLIANCE WITH NRC REQUIREMENTS

Appendix K to 10-CFR-50 specifies required and acceptable features of ECCS evaluation models. Previous revisions of the RELAP5/MOD2-B&W program, through Rev. 1, have been reviewed and found to satisfy the requirements of Appendix K when used with the approved B&W ECCS methodology, subject to any restrictions cited in the SER.

The modifications documented in Revisions 2 and 3 of BAW-10164P do not affect compliance with any of the required features of Appendix K. Modifications of the fuel pin model are such that the requirements of Appendix K continue to be satisfied. Inclusion of a contact conductance term in the calculation of fuel clad gap conductance does not affect compliance with Appendix K which states that "thermal conductance of the gap between the UO2 and the cladding shall be evaluated as 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." Section B of Appendix K also specifies that "The gap conductance shall be varied in accordance with changes in gap dimensions and any other applicable variables."

Inclusion of axial thermal expansion of the cladding and fuel in the gas plenum volume calculation also does not affect compliance with Appendix K. The plenum volume is used in the calculation of internal rod pressure which is a key determinant of the amount of swelling and rupture. Appendix K requires that "the swelling and rupture calculations shall be based on applicable data in such a way that the degree of swelling and incidence of rupture are not underestimated." Calculation of gas gap pressure using plenum volume which accounts for axial thermal expansion of the fuel and cladding improves accuracy and should not result in underestimating the pressure or the incidence of swelling and

rupture.

Appendix K specifies that the Baker-Just model be used to calculate the metal-water reaction rate, but does not specify the solution method. Therefore, the implicit solution technique is acceptable, given that it yields a mathematically correct solution of the required equation. Appendix K also requires that "The degree of swelling and rupture shall be taken into account in calculations of gap conductance, cladding oxidation and embrittlement, and hydrogen generation." The "swelled radius" modification to the metal-water reaction model accounts for the increase in clad radius, and hence surface area, due to swelling. This model change affects the hydrogen generation and cladding oxidation, and is in compliance with the Appendix K requirements listed above. The iterative technique for determining a multiplier on gap conductance which yields a desired initial stored energy is a user convenience feature which does not affect the previously approved model.

Section C.4.e of Appendix K to 10 CFR 50 states: "After CHF is first predicted at an axial fuel rod location during blowdown, the calculation shall not use nucleate boiling heat transfer correlations at that location subsequently during the blowdown even if the calculated local fluid and surface conditions would apparently justify the reestablishment of nucleate boiling. Heat transfer assumptions characteristic of return to nucleate boiling (rewetting) shall be permitted when justified by the calculated local fluid conditions during the reflood portion of a LOCA." The core heat transfer selection model modifications assure that no return to nucleate boiling will occur before the end of blowdown for the large break LOCA. Since the definitions of blowdown and reflood are inappropriate for SBLOCA, it is acceptable to bypass this "no return to nucleate boiling" requirement in that case. The modification is therefore in compliance for LBLOCA analysis, and acceptable for the SBLOCA

analysis where the requirement is inappropriate.

CHF correlations acceptable for use in LOCA analysis are listed in Sections C.4.b and d. Section C.4.a states that "Correlations developed from appropriate steady-state and transient-state experimental data are acceptable for use in predicting the critical heat flux (CHF) during LOCA transients. The computer programs in which these correlations are used shall contain suitable checks to assure that the physical parameters are within the range of parameters specified for use of the correlation by their respective authors". As discussed in Section 2.1.7 of this report, the BWUMV CHF correlation has been developed using appropriate experimental data for fuel with mixing vanes. The correlation is therefore acceptable for use within the ranges of parameters specified. It is noted that BWNT requested that the correlation be approved for pressures down to 750 psia. Based upon the data comparison provided by BWNT, the staff concludes that the correlation is acceptable down to pressures of 1300 psia and mass fluxes of 500,000 lb/hr-ft^2. Checks which restrict use of the correlation to this range must be included in RELAP5/MOD2-B&W for the code to be acceptable.

Benchmarks against calculations of the approved computer program FOAM2 and THTF experimental test data have shown that addition of the Wilson drag model improves predictions of void distribution in the core region. Appendix K does not list specific requirements in this area. The modification, which improves modeling accuracy, is therefore acceptable. A number of the other code enhancements fall into this same category. They cover areas where Appendix K does not specify required features. These include the annular mist flow regime overall drag multiplier, condensation heat transfer correlation modifications, Wilson slug flow drag model, and the CCFL model. Appropriate justification has been provided to show that these enhancements improve

modeling accuracy.

5.0 CONCLUSIONS AND LIMITATIONS

Modifications made to RELAP5/MOD2-B&W as described in Revisions 2 and 3 of BAW-10164P have been reviewed and evaluated. Based on the benchmarks presented, the staff finds that the models described in version 19 of RELAP5/MOD2-B&W to be acceptable for LOCA and non-LOCA analysis for PWRs with recirculating and OTSGs subject to the following limitations:

* Use of the Wallis and UPTF parameters at the tube bundle and steam generator plenum inlet are acceptable. The parameters used in the CCFL model for any other application must be validated, and the validation reviewed and approved by the staff for that application (see section 3.1.3 of this evaluation).
* The BWUMV correlation is limited to pressures above 1300 psia.
* For large break LOCA ECCS evaluation model calculations, form losses due to ruptured cladding should not be excluded using the user option described in Section 3.2.4 of this evaluation.
* The value of the user specified parameters listed in Table 1 of this evaluation (i.e. those used for the benchmark calculations) are the only acceptable values for LOCA licensing calculations.

Table 2 lists typographical errors that were found during the course of this review. Correction of these errors should be incorporated into the approved version of BAW-10164P. The automated blockage droplet breakup calculation, the implicit formulation of the Baker-Just metal water reaction model and the fuel rod Evaluation Model improvements referred to in Section 5 of the BEACH safety evaluation report[18] were reviewed in this evaluation and found acceptable. Contingency 4 given in Section 5 of the BEACH safety evaluation report is no longer applicable. That contingency states: "Use of the automated blockage droplet breakup calculation, implicit formulation of the Baker-Just metal

water reaction model and the fuel rod Evaluation Model (EM) improvements should be made contingent upon their approval in Revision 3 of BAW-10164P, which describes these updates."

Table 2
Summary of Changes That Should Be Made To The Approved Version of BAW-10164, Revision 2 and 3

Affected Report Revision and Section	Change Summary
Revision 2, Appendix H, Figure H.1	Correction to Figure H.1 in response to Question 2 of the Revision 2 RAI
Revision 2, Appendix H, Table H.2	Correction to Table H.2 in response to Question 4 of the Revision 2 RAI
Revision 2, Section 2.3.3 and Appendix I	Correction to Equation I-1, I-2 and pressure and mass flux units in response to Questions 5, 6, and 7 of the Revision 2 RAI
Revision 2, Appendix I, Table I.3	Duplication in point numbers in Test 160 (point 789) and Test 164 (points 2060 and 2065) should be corrected or clarified.
Revision 2, Appendix J, Page J-8	Reference to Table 2 should be Table J.2 per Question 11 of the Revision 2 TER
Revision 3, Section 2.3.2	Correction of ϵ_{TC} value on Table 2.3.2-2 in response to Question 4 to the Revision 3 RAI.
Revision 3, Page 2.3-46	Correction to β^2 definition in text in response to Question 5 of the Revision 3 RAI. Other correction noted by BWNT on Page 2.3-36 should also be included.
Revision 3, Appendix L, Table L.1	Revision to Table L.1 in response to Question 10 of the Revision 3 RAI.
Revision 3, Appendix L, Table L.2	Revision to Table L.2 in response to Question 11 of the Revision 3 RAI.

5.0 REFERENCES

1. Ransom, V. H., et. al., RELAP5/MOD2 Code Manual -- Volume 1: Code Structures, System Models and Solution Methods and Volume 2: Users Guide and Input Requirements, NUREG/CR-4312 - Volume 1, August, 1985 and NUREG/CR-4312 - Volume 2, December 1985.

2. B&W Nuclear Technologies, RELAP5/MOD2-B&W -- An Advanced Computer Program for Light Water Reactor LOCA and non-LOCA Transient Analysis, BAW-10164P, Revision 1, October 1988.

3. Letter from A. C. Thadani (USNRC) to J. H. Taylor (B&W Nuclear Technologies), Acceptance for Referencing of Topical Report BAW-10164P, Revision 1, RELAP5/MOD2-B&W, An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis, April 18, 1990.

4. Code of Federal Regulations, ECCS Evaluation Models, Chapter 10, Part 50, Appendix K

5. B&W Nuclear Technologies, RELAP5/MOD2-B&W -- An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis, BAW-10164P, Revision 2, August 1992.

6. B&W Nuclear Technologies, RELAP5/MOD2-B&W -- An Advanced Computer Program for Light Water Reactor LOCA and non-LOCA Transient Analysis, BAW-10164P, Revision 3, October 1992.

7. Letter from J. H. Taylor (B&W Nuclear Technologies) to R. C. Jones (USNRC) BEACH Topical Report BAW-10166P, JHT/93-214, August 31, 1993.

8. Letter from J.H. Taylor (B&W Nuclear Technologies), Response to NRC's Request for Additional Information on BAW-10164, Revision 2, August, 1992; RELAP5/MOD2-B&W, An Advanced Computer Program for Light Water Reactor LOCA and NON-LOCA Transient Analysis, JHT/93-279, November 16, 1993

9. Letter from J.H. Taylor (B&W Nuclear Technologies), Response to NRC's Request for Additional Information on BAW-10164, Revision 3, October, 1992; RELAP5/MOD2-B&W, An Advanced Computer Program for Light Water Reactor LOCA and NON-LOCA Transient Analysis, JHT/94-7, January 21, 1994

10. Letter from J.H. Taylor (B&W Nuclear Technologies), Response to NRC's Supplemental Request for Additional Information on BAW-10164, Revision 2, August, 1992; RELAP5/MOD2-B&W, An Advanced Computer Program for Light Water Reactor LOCA and NON-LOCA Transient Analysis, JHT/94-146, September 20, 1994.

This page intentionally left blank.

11. B&W Nuclear Technologies, FOAM2 -- Computer Program to Calculate Core Swell Level and Mass Flow Rate During Small Break LOCA, BAW-10155A, October 1990.

12. Collier, John G., Convective Boiling and Condensation, Mc-Graw-Hill International Book Company, 2nd. edition, 1972

13. B&W Fuel Company, Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies, BAW-10159P-A, July 1990

14. B&W Nuclear Technologies, FRAP-T6-B&W - A Computer Code for the Transient Analysis of Light Water Reactor Fuel Rods, BAW-10165P, October 1988.

15. B&W Nuclear Technologies, TACO3 - Fuel Pin Thermal Analysis Computer Code, BAW-10162P-A, October 1989.

16. D.A. Powers and R.O. Meyers, Cladding Swelling and Rupture Models for LOCA Analysis, NUREG-0630, April, 1980

17. D.L. Hagrman, et. al., MATPRO - Version 11 (Revision 2), NUREG/CR-4079, August 1981

18. D.A. Prelewicz, Technical Evaluation Report - BEACH - Best Estimate Analysis Core Heat Transfer, A Computer Program for Reflood Heat Transfer During LOCA, BAW-10166P, Revision 4, SCIENTECH, Inc., SCIE-NRC-219-93, November, 1993