ANF-91-048(NP)(A)

ANF-91-048(NP)(A)
CORRESPONDENCE

ADVANCED NUCLEAR FUELS CORPORATION

ADVANCED NUCLEAR FUELS CORPORATION METHODOLOGY FOR BOILING WATER REACTORS EXEM BWR EVALUATION MODEL

JANUARY 1993

ANF-91-048(NP)(A)

ANF-91-048(NP)(A)
Correspondence

Issue Date: 1/13/93

ANF-91-048(NP)(A)	ADVANCED NUCLEAR FUELS METHODOLOGY FOR BOILING WATER REACTORS EXEM BWR ECCS EVALUATION MODEL
ANF-91-048(NP)(A) CORRESPONDENCE	LETTERS

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

November 18, 1992

R.A. Copeland, Manager
Reload Licensing
Siemens Nuclear Power Corporation
P.O.Box 130
Richland, Washington 99352-0130

Dear Mr. Copeland:

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT ANF 91-048(P), "ADVANCED NUCLEAR FUELS METHODOLOGY FOR BOILING WATER REACTORS EXEM BWR ECCS EVALUATION MODEL"

The staff reviewed the Topical Report ANF 91-048(P), which describes the Siemens Nuclear Power Corporation's revised EXEM code, the evaluation model for the boiling water reactor (BWR) emergency core cooling system (ECCS). The revisions and improvements made to the code consist of (1) algorithmic changes to improve the numerical stability of the code and (2) a new heat transfer model (modified Dougall-Rohsenow correlation) to replace the existing Dougall-Rohsenow correlation. This code revision is mandated by 10 CFR 50.46 because the aggregate of the calculated peak cladding differences caused by error corrections and modifications exceeds 50°F. Siemens benchmarked the revised code model against the two loop test apparatus (TLTA) test 6406/Run 1, also tested the modified Dougall-Rohsenow correlation against independent experiments, and found them to be conservative.

The staff finds the application of ANF 91-048(P) to be acceptable for referencing in license applications to the extend specified, and under the limitations delineated, in ANF 91-048(P) and the associated U.S. Nuclear Regulatory Commission (NRC) technical evaluation. The evaluation defines the basis for accepting this topical report.

The staff will not repeat its review of the matters found acceptable as described in ANF 91-048(P), when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. Our acceptance applies only to the matters described in the application of ANF 91-048(P).

In accordance with procedures established in NUREG-0390, the staff requests that the Siemens Nuclear Power Corporation publish accepted versions of this topical report, proprietary and non-proprietary, within 3 months of receiving this letter. The accepted version shall include an "A" (designating accepted) following the report identification symbol.

If the staff's criteria or regulations change so that its conclusions about the acceptability of the report are invalidated, Siemens Nuclear Power Corporation and/or the applicants referencing the topical report should revise and resubmit their respective documentation or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

A C Thadani

Ashok C. Thadani, Director
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

Enclosure:
ANF 91-048(P) Evaluation

ENCLOSURE

SAFETY EVALUATION FOR ANF 91-048(P), "ADVANCED NUCLEAR FUELS CORPORATION METHODOLOGY FOR BOILING WATER REACTORS EXEM BWR ECCS EVALUATION MODEL"

1.0 INTRODUCTION

In a letter of July 25, 1991, the Advanced Nuclear Fuels Corporation (ANF; since renamed to Siemens Power Corporation) submitted the topical report ANF-91-048(P), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model," for the staff to review (Refs. 1, 2). Siemens submitted additional information on January 28, 1992 (Ref. 3), May 27, 1992 (Ref. 4), and July 6, 1992 (Ref. 5). The report describes modifications to the existing ANF evaluation model (EM) for the boiling water reactor (BWR) emergency core cooling system (ECCS). These modifications are required by Section 50.46(a)(3)(ii) of Title 10 of the Code of Federal Regulations (10 CFR 50.46 (a)(3)(ii)) because the calculated aggregate of the peak cladding temperature (PCT) differences due to error corrections and modifications exceeds 50°F (Ref. 6). The currently approved EM (EXEM) uses four codes: RODEX2, RELAX, FLEX, and HUXY. This topical report describes proposed changes to the FLEX and RELAX codes.

The RELAX code calculates the system blowdown and the response of the hot channel. The proposed changes in RELAX include modifying the time step control, adding an iterative solution method; eliminating discontinuous transitions, replacing the Dougall-Rohsenow heat transfer correlation with the modified Dougall-Rohsenow heat transfer correlation.

The FLEX code calculates the hydraulic transient for the reflood phase of the loss of coolant accident (LOCA). The applicant proposes making changes in code numerics to reduce computational flow oscillations.

The applicant validated the revised ECCS EM by comparing its results to data from the two loop test apparatus (TLTA) test 6406/Run 1.

2.0 SUMMARY OF THE TOPICAL REPORT

In the topical report the applicant discusses the EM, the proposed revisions, and conformance to Appendix K to 10 CFR part 50, and assesses the proposed EM.

2.1 Evaluation Model

In this section the applicant describes each of the codes that comprise the EM and their function in modeling each phase of the transient. The EM begins with the RODEX2 code calculating the parameter specifications. The RELAX code then calculates the blowdown until the low pressure core spray reaches its rated value. RELAX then performs another calculation that defines the hot

channel conditions including the conditions determining the heat transfer coefficients. The FLEX code calculates the hydraulic characteristics of the reflood phase, and the HUXY code calculates the increase in temperature for the entire LOCA.

2.2 Proposed Revisions to EXEM BWR ECCS EM

The applicant proposed modifying only the RELAX and the FLEX codes of the EM. The applicant modified the RELAX code to improve convergence, eliminate discontinuities, and replace the Dougall-Rohsenow heat transfer correlation with the modified Dougall-Rohsenow correlation. The numerical modifications were based on an iterative hydraulics solution method proposed by Porshing (Ref. 7) and based on finite-difference solutions of the linear differential equations representing conservation of energy, mass, and momentum. The applicant added logic to measure the degree of convergence and determine an optimal time step. The applicant also improved the following: (1) the critical flow model to prevent calculated flow from being reversed at critical flow junctions; (2) the bubble mass integration model, by solving a quadratic equation instead of two linear approximations to determine mass and density; and (3) the drift flux model to provide a smoother transition between differing flow regimes. The RELAX drift flux model was proposed by Ishii (Ref. 8) and was used to replace the existing model. However, Onkawa and Lahey (Ref. 9) noted that the Ishii relations were inconsistent with the Kutateladze flooding correlation and proposed the Onkawa-Lahey modification for the drift velocity formulation in the post-CHF region which is consistent with the Kutateladze flooding correlation. The liquid fraction model in the annular-mist region of the drift flux inhibited conversion, thus, an alternate model (also suggested by Ishii, Ref. 10) was used; the pump model was changed to represent the pump head and pump flow by two linear approximations solved simultaneously; the jet pump model was changed to correct a discontinuity and finally the Dougall-Rohsenow heat transfer correlation was substituted by the modified Dougall-Rohsenow correlation (Ref. 11). The modified form has been found to provide agreement or slightly conservative predictions relative to pertinent experimental data.

The applicant modified the FLEX code to alleviate numerical calculational difficulties that arose from the semi-explicit solution technique used in the code. The specific changes include: correcting the criteria for the core by-pass flow balance convergence modifying the core inlet pressure drop calculation to properly account for two-phase core inlet conditions, improving the low pressure coding logic and editing code to smooth the transition in core inlet density as the lower plenum mixture reaches the core.

The applicant made other changes to FLEX to address calculated negative pressure values during pressure drops caused by a break and by smoothing in the density at the lower plenum outlet. The drift flux model changes at the beginning of the reflood stage, which cause rapid (computed) oscillations in the plenum mixture level and the midplane entrainment. The applicant made other minor changes in FLEX to correct known discrepancies and made minor changes in the format for the input to the code.

The applicant modified the HUXY code to correct an overconservative coefficient in the decay heat model. The corrected decay heat meets the requirements of Appendix K. The applicant made no other changes to HUXY which is an approved code.

2.3 Conformance to Part 10 CFR 50, Appendix K

The only changes that could affect the applicability of Appendix K are the replacement of Dougall-Rohsenow heat transfer correlation and the correction of the decay heat model. The applicant claims that these changes are conservative and result in slightly higher peak clad temperature at the end of the blowdown. Although the changes in the FLEX do not affect the entire EM, they do affect the portions that have been changed. This approach is consistent with the validation procedure for the current versions of RELAX and FLEX. The TLTA nodalization for the RELAX code (blowdown) and the FLEX code (refill) are shown. The applicant ran the TLTA test 6406/Run 1 with the revised model and compared the calculated results to the experimental data.

2.4 Assessment of the Proposed EM

The applicant described the effects of the revised ECCS methodology, discussing separately the effects of the RELAX and FLEX and describing the overall effects. The applicant assessed these effects by comparing the numerical results to the corresponding results from the same problem from the existing ECCS model.

3.0 EVALUATION

3.1 RELAX Simulation of TLTA Test 6406/Run 1

The applicant validated the revised EM by comparing it with the results of the TLTA test 6406/Run 1 to verify the peak cladding temperature (PCT) predicted by the revised EM. The TLTA is a single bundle experiment; thus, the hot channel PCT is calculated without the need to run HUXY. The EM is used in the RELAX and FLEX in a manner identical to the use in licensing applications. This included the effect of the modified Dougall-Rohsenow correlation and a suitable value of a flow multiplier. In the TLTA simulation the applicant added a node in the intact loop centrifugal pump suction line to eliminate flow oscillations at the recirculation line uncovery. These oscillations are one of the reasons for revising the EM. For consistency, the applicant performed the FLEX calculation using the RELAX values calculated at the end of the blowdown phase.

The calculated PCT value is significantly more conservative than that measured in the TLTA.

3.2 FLEX Simulation of TLTA Test 6406/Run 1

After the blowdown phase, simulated by RELAX to 136 seconds, the refill/reflood portion of the TLTA test runs to 200 seconds. The applicant

compared the lower plenum mass measured in TLTA with the value calculated by the revised FLEX and found that the revised FLEX predicted the reflood level well at the mid-level and underpredicted it toward the end of the refill/reflood period. The revised FLEX predicts fairly accurately the steam dome pressure, which is an integral response of many thermodynamic parameters such as critical flow, bypass flow, and phase separation. The key events such as filling of the bypass region, the filling of the lower plenum, and the bundle reflood are timed in a manner consistent with the experimental data.

3.3 Modified Dougall-Rohsenow Correlation

The heat transfer coefficient of the proposed modified Dougall-Rohsenow correlation has been compared to a number of other correlations (Polonik, Groeneveld, Quinn and Dougall-Rohsenow) as a function of pressure (0-2500 psia), temperature (0-2500^{0}F), Quality (0-100 percent) and a mass flow off 0-3.0x10^6 lbm/hr-ft^2. In each of these comparisons, the modified Dougall-Rohsenow correlation has been shown to be reasonably conservative. The applicant compared values calculated using the proposed correlation with data from heat transfer experiments at the Oak Ridge National Laboratory (ORNL) and the Columbia University facility for Combustion Engineering Incorporated. In these applications, an appropriate multiplier was used for the heat transfer coefficient was then applied to the modified Dougall-Rohsenow correlation. The applicant applied this correlation to a set of ORNL data and obtained conservative predictions of the surface peak temperature.

3.4 Applicability of TLTA to a Large BWR

The TLTA facility was designed to 1/624 scale based on a large BWR/6, which contains 624 assemblies. Thus, the TLTA core consists of a single full-scale fuel assembly that is an electrically heated simulation of an 8x8 BWR assembly. The system is scaled by power to volume, and the break is scaled by break area to system volume. Linear dimensions are maintained as close to the prototype as possible. The time scale, the fluid mass and energy distribution, velocities, accelerations and lengths in this facility are essentially the same in the test facility as in the prototype plant (Refs 5 and 12). For the blowdown portion of a large break LOCA, tests on facilities of various scales show the same controlling phenomena. Thus, scale is of no concern for the full scale reactors. However, in the refill/reflood phase, the effects of scale have been observed because component length is preserved but component diameters are significantly reduced. The effects of multiple assemblies in a BWR plant is not preserved in the TLTA experiment. Large scale tests simulating the upper plenum and parallel channels of a multidimensional reactor vessel permit ECC fluid to flow readily to the lower plenum through low power assemblies. Accordingly, the observed TLTA reflood time will likely be longer than the expected reflood time for an actual BWR (Ref. 13). Therefore, when the TLTA phenomena are the controlling phenomena and are applied to a BWR, they are expected to predict delayed reflood and thus increase the calculated PCT, which is a conservative result.

3.5 Assessment of the Revised EXEM BWR ECCS EM

The applicant compared the overall performance of the revised EM to the results of the existing EM. However, the existing model incorporates the Dougall-Rohsenow correlation, which does not satisfy Appendix K requirements. Having established that the changes are conservative, the applicant compared the revised and existing models for phenomenological trends, numerical stability, and solution convergence rather than for absolute values.

The solution for core inlet flow exhibits the same trends as does the solution in the existing version. The same is true for the blowdown break flow and clad temperature. The revision yields the greatest improvements in the relative midplane entrainment for all BWR types for which this code is intended, that is, for BWR/3 to BWR/6. The applicant demonstrated that numerical solution convergence and stability have considerably improved.

4.0 SUMMARY, CONCLUSIONS, AND LIMITATIONS

The staff reviewed a revised version of ANF-91-048(P) for the EXEM BWR ECCS EM. The applicant revised the report to meet the requirements of 10 CFR 50.46 whenever error corrections and modifications result in a value of Delta-T greater than 50^{0}F. The revision included two major parts: numerical modifications to improve code numerical stability and a new heat transfer correlation. The applicant benchmarked the revised EM to the results of the TLTA test 6406/Run 1.

Both revised parts of the EM perform conservatively compared to the data measured in the TLTA. The modified Dougall-Rohsenow heat transfer correlation has been shown to yield conservative results for many experimental measurements. The applicant used a suitable multiplier in the comparison calculations. Licensees will use this multiplier in licensing applications. The staff finds the proposed EXEM BWR ECCS EM, as documented in References 1 to 5, to be acceptable for referencing in BWR LOCA analyses, with the following limitations:

1. The revised model is valid within the range of applicability of the modified Dougall-Rohsenow heat transfer correlation.

2. The staff requires that the revised evaluation model be protected with appropriate quality assurance procedures, subject to auditing by the staff.

3. The phase separation models will be limited to the models used in the topical report.

4. The revised EM will be limited to jet pump plant applications.

The staff recommends that the applicant incorporate the contents of References 3-5 into a single report, for ease of reference.

5.0 REFERENCES

1. Letter from R. A. Copeland, ANF, to NRC, "Transmittal of ANF-91-048(P), EXEM BWR ECCS Evaluation Model," July 25, 1991.

2. ANF-91-048(P), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model," by D. C. Kolesar, et al., July 1991.

3. Letter from R. A. Copeland, Siemens Nuclear Power Corporation to R. C. Jones, "Response to Request for Additional Information, dated December 20, 1991," January 28, 1991.

4. Letter from R. A. Copeland, Siemens Nuclear Power Corporation, to Lambros Lois, "Additional Information Requested in April 1992," May 27, 1992.

5. Letter from R. A. Copeland, Siemens Nuclear Power Corporation, to Lambros Lois, "Additional Information concerning the Large Break LOCA Model Requested in June 1992," July 6, 1992.

6. Code of Federal Regulations, 10 CFR 50.46(a)(3)(ii) and Appendix K.

7. Porshing, T. A., et al, "Stable Numerical Integration of Conservation Equations for Hydraulic Networks," Nuclear Science and Engineering, vol. 43, pp 218-225, February 1971.

8. Ishii, M., "One-Dimensional Drift-Flux Model and Constitutive Equations for Relative Motion Between Phases in Various Two-Phase Flow Regimes, ANL-77-47, October 1977.

9. NES-486, "The Analysis of Proposed BWR Inlet Flow Blockage Experiments in PBF," K. Onkawa and R. T. Lahey, December 1978.

10. Ishii, M., et al, "Phenomenological Modeling of Two-Phase Flow in Water Reactors at ANL," Ninth NRC Water Reactor Safety Research Information Meeting, October 26-30, 1981, Gaithersburg, Maryland.

11. Slaughterbeck, D. C., et al., "Flow Film Boiling Heat Transfer Correlations; Parametric Study with Data Comparisons," 73-HT-50, ASME-AIchE Heat Transfer Conference, Atlanta, Georgia, August 5-8, 1973.

12. Nahavandi, A. N., et al., "Scaling Laws for Modeling Nuclear Reactor Systems," Nuclear Science and Engineering vol. 72, pp. 75-83, 1979.

13. NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," NRC, December 1988.

ADVANCED NUCLEAR FUELS CORPORATION

ANF-91-048(NP)(A)
Issue Date: 1/13/93

ADVANCED NUCLEAR FUELS CORPORATION

METHODOLOGY FOR BOILING WATER REACTORS

EXEM BWR ECCS EVALUATION MODEL

Prepared by

D. C. Kolesar
Engineering Model Development
Research and Product Development

S. E. Jensen
BWR Fuel Engineering
Fuel Engineering and Licensing

K. D. Richert
Engineering Computing Services
Corporate Information Services

/trb

ACKNOWLEDGEMENT

The efforts of Don Rowe and Larry Nielsen of Donald S. Rowe and Associates in the refinement of the FLEX refill/reflood code and Carl Wheeler of Numerical Applications, Inc. in performing the TLTA validation calculations are gratefully acknowledged. In addition, the timely contribution of Dan Braun and Lance Riniker of ANF and Don Rowe of Donald S. Rowe and Associates in performing the assessment calculations are appreciated.

TABLE OF CONTENTS

TABLE OF CONTENTS

LIST OF TABLE

LIST OF FIGURES

LIST OF FIGURES

1.0 INTRODUCTION AND SUMMARY

ANF originally submitted the EXEM BWR Emergency Core Cooling System (ECCS) Evaluation Model (EM)[1] in 1980 for jet pump BWR loss-of-coolant accident (LOCA) analysis. This report documents improvements in the FLEX and the RELAX codes. ANF is also replacing the Dougall-Rohsenow heat transfer correlation in the ECCS EM. The latter is required by the NRC as a consequence of revising the EXEM ECCS EM when a Δ50°F or greater change results from error corrections or modifications to the constituent codes.

The currently approved EXEM BWR ECCS EM methodology employs four major codes. These codes are RODEX2, RELAX, FLEX, and HUXY. This report describes the upgrades made to the FLEX and the RELAX codes. The HUXY and RODEX2 codes were left unchanged by this revision of the ECCS EM. Changes to the RELAX and the FLEX codes are summarized below.

RELAX calculates the system blowdown and hot channel response. This code was modified to replace the current time step control with an automatic time step control based upon numeric code convergence for each time step. This required that code numerics be improved to avoid interference of any discontinuous transitions with the time step controller. Also, the improvement in code numerics assures code convergence and reduces spurious flow oscillations.

The FLEX code computes the system hydraulic transient for the refill/reflood phase in the LOCA. The code robustness was improved to reduce flow oscillations and convergence difficulties in the coupling of core, core bypass, and system regions. These numerical difficulties in plant analyses had forced FLEX users to sometimes use limiting conservative plant

descriptions to bound approved base-line FLEX models.

Validation of the revised ECCS EM was accomplished by benchmarking the EM to experimental data (TLTA test 6406/Run 1). The same benchmark was previously performed in support of the currently approved ECCS EM licensing submittal.

2.0 EVALUATION MODEL DESCRIPTION

The function and interfacing of the component codes of the revised EXEM BWR ECCS EM are unchanged from the currently approved EXEM ECCS EM[1]. Both rely on four major analysis codes. These codes are RELAX, FLEX, HUXY, and RODEX2. The revision of the EXEM BWR EM consists of minor changes to the RELAX and the FLEX codes and a very minor change to the methodology used for computing the blowdown core decay heat. The revised decay heat calculation is consistent with the requirements of Appendix K, and it makes the ANF decay heat calculation during the blowdown consistent with the approved decay heat calculation following blowdown. The methodology for interfacing the codes within the model is illustrated in Figure 2.1.

In the EXEM BWR ECCS EM, both approved and updated, the key role of the FLEX core heat transfer model is to accurately account for the rate of vapor generation, in order to determine the elapsed time from time of rated spray to time of reflood for the heatup code. In contrast, the HUXY heatup model is used for the calculation of the PCT and includes the Appendix K criteria applicable to heatup. Approval for the HUXY code can be traced from the earlier ANF NJP-BWR ECCS EM[2] as well as the currently approved ECCS EM. Subsequent to the release of the currently approved BWR ECCS EM, the GAPEX fuel rod thermal-mechanical response code was replaced by the RODEX2[3] code. This change was approved[3] for both LOCA and non-LOCA licensing applications for both BWRs and PWRs.

A complete analysis for a given break size starts with the specification of fuel parameters using RODEX2. RODEX2 is used to verify the initial stored energy in both the blowdown and the hot channel code, RELAX[1], and the heatup code, HUXY[4,5].

Next, the blowdown calculation is performed using RELAX. The one-dimensional core is represented by an average core channel. This calculation gives the system thermal-hydraulic response during blowdown. RELAX is run from break initiation to the time the ECCS Low

Pressure Core Spray (LPCS) flow reaches its rated value. The RELAX blowdown calculation provides two major sets of output. First, the blowdown calculation provides the upper and lower plenum transient boundary conditions for the hot channel analysis. Second, the blowdown calculation furnishes the transient core power and the system thermal-hydraulic conditions (including average core conditions) at the time the LPCS reaches rated spray.

Following the blowdown calculation, another RELAX run is made to analyze just the maximum power assembly (hot channel) of the core, using information from the blowdown run to supply the core power and the system boundary conditions at the core inlet and exit. The results from this RELAX calculation are heat transfer coefficients and fluid temperature conditions in the hot channel at the plane of interest for input to HUXY.

The FLEX refill/reflood code[1,6,7] is used to perform a liquid inventory calculation during the ECC injection period. Allowing for countercurrent flow through the core and the bypass, FLEX determines the refill rate of the lower plenum due to ECCS water and the subsequent reflood times for the core and the core bypass. Initial conditions for the FLEX calculation are supplied by the RELAX blowdown calculation. The FLEX calculation provides HUXY the time interval from the time of rated LPCS spray injection to the time the two-phase fluid has risen by entrainment in the core to the axial plane of interest (time of hot node reflood).

The HUXY code is used to perform heatup calculations for the entire LOCA transient and yields peak clad temperature and local clad oxidation at the axial plane of interest. These calculations consider thermal-mechanical interactions within the fuel rod. The clad swelling and rupture models from NUREG-0630 have been implemented in the heatup model as described in References 1, 2, 5 and 8. HUXY requires the time of rated core spray from the RELAX blowdown calculation and the time of core bypass reflood and the time of core reflood at the level of interest from the FLEX refill/reflood calculation.

HUXY requires two event times from FLEX. These are the time of reflooding of the core bypass and the time of hot node reflood inside the canister. Following the time of bypass reflood, the heat transfer coefficient on the bypass side of the canister beneath the bypass mixture level is set to a conservative value based upon the modified Bromley pool film boiling correlation[1]. Subsequent to the canister quench, the canister heat transfer coefficient is set to 14,000 BTU/hr-ft^2-°F in conformance to the Appendix K approved modified Yamanouchi correlation. Results from the HUXY calculation are the peak clad temperature (PCT) and maximum fraction of local cladding oxidation (local metal-water reaction).

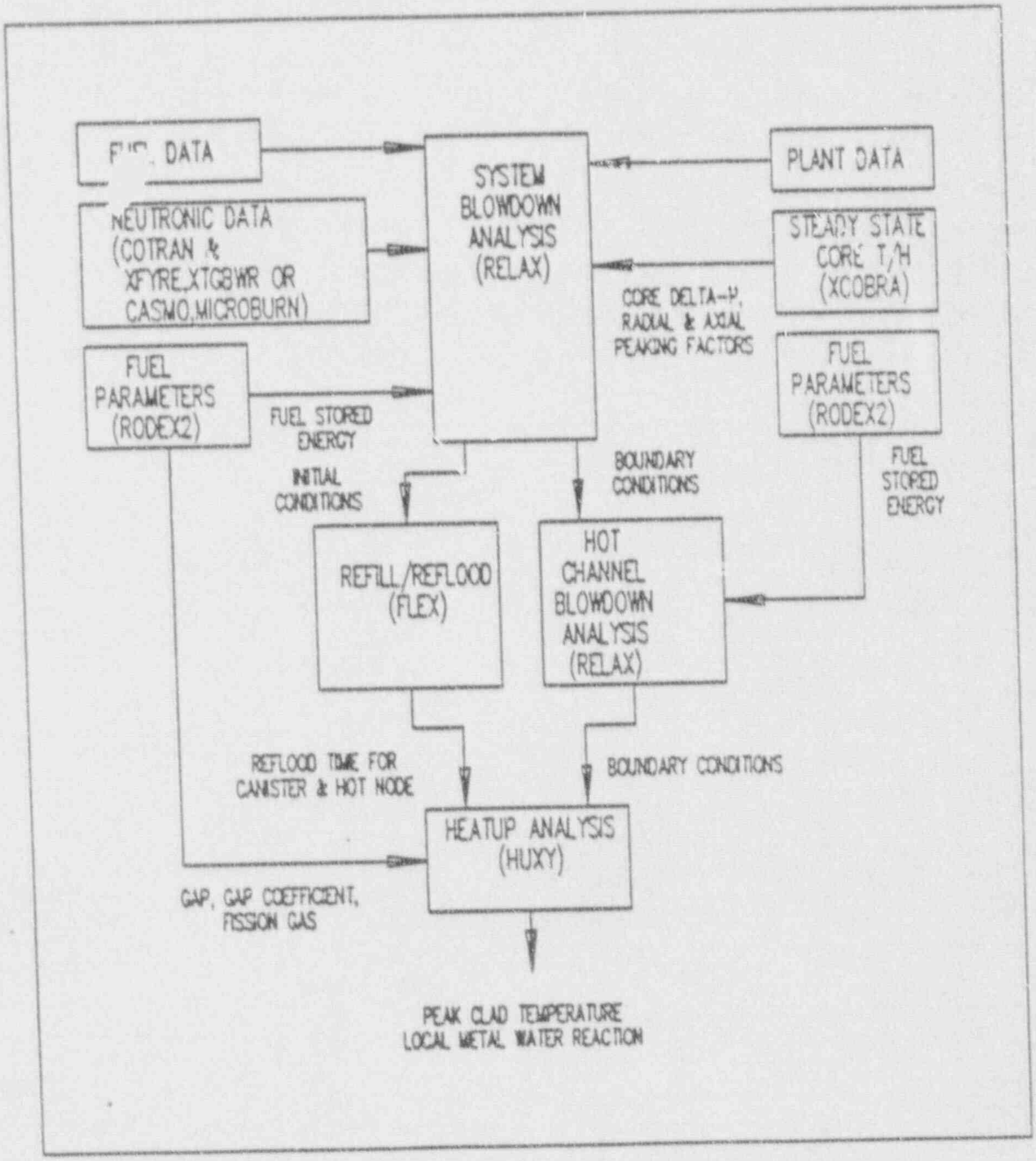
FUEL DATA
NEUTRONIC DATA
(COTRAN &
XFYRE,XTGBWR OR
CASMO,MICROBURN)
FUEL
PARAMETERS
(RODEX2)
SYSTEM
BLOWDOWN
ANALYSIS
(RELAX)
PLANT DATA
STEADY STATE
CORE T/H
(XCOBRA)
CORE DELTA-P,
RADIAL & AXIAL
PEAKING FACTORS
FUEL
PARAMETERS
(RODEX2)
FUEL STORED
ENERGY
INITIAL
CONDITIONS
BOUNDARY
CONDITIONS
FUEL
STORED
ENERGY
REFILL/REFLOOD
(FLEX)
HOT
CHANNEL
BLOWDOWN
ANALYSIS
(RELAX)
REFLOOD TIME FOR
CANISTER & HOT NODE
BOUNDARY CONDITIONS
HEATUP ANALYSIS
(HUXY)
GAP, GAP COEFFICIENT,
FISSION GAS
PEAK CLAD TEMPERATURE
LOCAL METAL WATER REACTION

FIGURE 2.1

3.0 REVISED EXEM BWR ECCS EM

The revised EXEM BWR ECCS EM includes the modification of the RELAX and the FLEX codes. The principal change to these codes was to increase robustness and reduce run time. These changes, plus the previously documented changes to the codes that comprise the currently approved ECCS EM model, will alter the EM results beyond the defined Δ50°F reporting limit[9]. As a consequence of exceeding the reporting limit, the NRC requires that the Dougall-Rohsenow heat transfer correlation be replaced in the ECCS EM. The changes to the EXEM BWR ECCS EM retain the existing Appendix K Evaluation Model approach currently in use at ANF and previously approved by the NRC.

3.1 RELAX Upgrade

RELAX was modified to replace the current automatic time step control with a better automatic time step control based upon numeric code convergence for each time step. This required that code numerics be improved to avoid interference between code models (discontinuous transitions) and the automatic time step controller.

These numerical changes greatly reduce the time step sensitivity of the RELAX code since convergence is assured for each individual time step.

3.1.1 Iterative Solution Method

The numerical solution method used in the currently approved version of RELAX is based on a method proposed by Porsching, et al.[11].

3.1.2 Convergence Measure

Logic has been added to RELAX to measure the degree of convergence at the end of each time step. The term "convergence" is used here in the sense that for the given time step size the nodal pressure, mass and energy terms in the finite difference form of the mass, energy

and momentum equations satisfy these equations to within a specified maximum error.

If convergence is not obtained after a fixed number of iterations, the time step is halved and the calculation is repeated with the new time step.

3.1.3 <u>Automatic Time Step Control</u>

The measured level of convergence is used to determine an optimal time step.

The RELAX upgrade contains logic to be able to recover from previously fatal occurrences of exceeding the bounds of the water property tables, as well as failure to converge in a given number of iterations. This involves trapping these occurrences and resetting variables to values that existed at the beginning of the time step, including the restoration of the special Evaluation Model heat transfer flags to their condition at the beginning of the time step. The calculation then proceeds with a smaller time step. A lower bound of 0.000005 seconds is applied to the time step in this process.

3.1.4 Critical Flow Model

The solution scheme in RELAX requires that flow in junctions with "critical flow" be determined separately by use of the critical flow model instead of the inertial equations. This requires that an estimate of the new flow be made before the flow matrix is solved in order to know when to apply the critical flow model. Under certain circumstances this estimated flow will predict a flow reversal against the pressure gradient. This may in turn result in a failure of the critical flow model since the new upstream fluid state probably will not be covered by the critical flow tables. To correct this problem, a test was included to prevent a reversal in the direction of the estimated flow against the pressure gradient.

3.1.5 Bubble Mass Integration Model

The RELAX Bubble Mass Integration Model was modified to solve a quadratic equation instead of two linear approximations for the determination of the bubble mass and the bubble density at the mixture level.

3.1.6 Drift Flux Model

The drift flux model was modified to provide smoother transitions between the different flow regimes. Logic paths in the coding that would result in default selection of homogeneous flow were eliminated.

3.1.7 Entrainment Fraction

It was found that the model for the liquid fraction in drops (E_d) used in the annular-mist region of the drift flux model inhibited convergence. An alternate model presented by Ishii, et al[14] gives the entrainment fraction as a function of phasic flows as:

$$E_d = \tanh\left[7.25\times10^{-7}\left(j_g^*\sqrt{D^*}\right)^{2.5} Re_l^{0.25}\right]$$

where: j_g^* = dimensionless gas flux given by:

$$j_g^* = \frac{j_g}{\left[\frac{\sigma g \Delta\rho}{\rho_g^2}\left(\frac{\rho_g}{\Delta\rho}\right)\right]^{\frac{1}{4}}}$$

Re_f = total liquid Reynolds number given by:

$$Re_f = \frac{\rho_f j_f D}{\mu_f}$$

D^* = dimensionless diameter given by:

$$D^* = D\sqrt{\frac{g\,\Delta\rho}{\sigma}}$$

In the above equations:

D	=	flow diameter
j_g	=	vapor volumetric flux
j_f	=	liquid volumetric flux
μ_f	=	liquid viscosity

Implementation of the Ishii liquid entrainment model into RELAX required the simultaneous solution of the entrainment fraction with the slip velocity in the slug-annular and annular-mist flow regions, since each is a strong function of the other.

3.1.8 Pump Model

The pump model calculates the new time step pump head based on the previous volume average flow. However, the pump volume average flow is strongly dependent on the pump head, and the explicit coupling of a two may result in diverging values for pump head and pump volume flow. This behavior was eliminated by deriving linear approximations of the two functional relationships between pump head and pump volume flow which are then solved simultaneously.

3.1.9 Jet Pump Model

The jet pump model is coupled to the RELAX solution scheme via pressure drop terms that are added to the total pressure drops for the drive-exit and suction-exit flow paths. However,

these modifications of the pressure drop need to be included in the Jacobian matrix in order to be able to implicitly approximate the pressure drop at the new time.

3.1.10 Modified Dougall-Rohsenow Heat Transfer Correlation

The Dougall-Rohsenow heat transfer flow film boiling correlation was replaced

3.1.11 RELAX Input Changes

The only change to the RELAX code input involves the Time Step Data Cards. To access the revised RELAX coding, a flag (variable NCHK) is set. This flag initiates an iterative solution process with automatic time step selection. User control of time steps in the new methodology

is limited to specification of the maximum increment value (variable DELTM). The minimum time step is a code default selected to preclude useless iteration. This numerical scheme assures mass and energy convergence. The primary control on the code iteration is provided by the normalized percent error tolerance (variable EPSW) in the energy and the mass conservation equations.

3.2 FLEX Upgrade

FLEX code logic was revised to improve code robustness. Application of FLEX to increasingly more complex plant configurations through the years have led to numeric difficulties. These numerical difficulties arise due to the semi-explicit solution technique used, the conservative logic that couples some of the models, and the mapping of FLEX initial conditions from RELAX results at time of rated spray. The difficulties were manifested in either code failure due to iteration troubles or high frequency oscillations in the code predictions. As a consequence, code users had to select more limiting and conservative plant description choices or had to apply conservatism directly to the code results.

The code difficulties have been reduced by revising the core-bypass flow iteration coding and by improving code logic to avoid spurious non-physical depressurization at low pressure and associated disruption of the reflood process. As a result, high frequency oscillations previously seen in results have also been greatly reduced. These changes to the code numerics give the code user the ability to use the approved plant description choices and obtain more realistic (earlier reflood) results.

PAGE 23 - 32 LEFT INTENTIONALLY BLANK

3.2.6 FLEX Input Changes

Three changes have been made to the FLEX code input format. These changes are: (1) Specification of the core bundle inlet and exit pressure loss coefficients through input is possible. (2) The minor edit/plot variable specifications have been simplified. (3) Provision for free format input has been added to both the FLEX code and to the FLXDATA restart option of the RELAX code. If the above features are not desired and if minor edits are not requested, then these input changes can be ignored and prior input decks are backward compatible. It has been found that the core inlet and exit pressure losses are adequately described by the current methodology.

3.3 Revised Procedures for Implementing the Evaluation Model

Since the objective of the revised EXEM BWR ECCS Evaluation Model has been to correct numerical difficulties within the RELAX and the FLEX codes, the overall procedures and methodology does not require significant alteration. The original sensitivity and case studies are unchanged by the new coding and code use procedures. This is demonstrated by the close correspondence to production calculations in the absence of the replacement of the Dougall-Rohsenow correlation (see Section 6.1) and by the good agreement of the TLTA 6406 simulation to experimental data (see Section 5.0).

It has been discovered that the decay heat model in RELAX inadvertently applies a 1.2 multiplier to both the fission product decay and the Actinide decay terms. Appendix K only requires the 1.2 multiplier be applied to the fission product decay component. The currently approved EXEM BWR ECCS EM methodology transfers this unnecessarily conservative decay heat into the HUXY heatup calculation from the start of blowdown until rated LPCS spray is predicted. For the refill/reflood period of the LOCA transient, the decay heat is determined in a manner consistent with Appendix K. That is, the decay heat during refill/reflood period is determined by interpolation of the 1971 draft ANS Standard Decay Heat table presented in the WREM documentation[18]. The methodology for implementing the Appendix K decay heat model in the revised EXEM BWR ECCS methodology is changed to be consistent throughout the LOCA. The ANS Standards Committee draft standard (October 1971) values times 1.2 with Actinide heat added will be used over the entire LOCA transient in the heatup calculation (HUXY code). The

methodology for the system blowdown (RELAX code) and the refill/reflood (FLEX code) phases are left unchanged except as noted below. The resulting model remains in compliance with Appendix K with a conservative bias. The net value of this methodology change has been evaluated for BWR3 and BWR6 plants. The range of reduction of PCT is between 1°F and 2°F.

In addition, the input procedure for RELAX has been changed to set both the core bundle inlet and exit flow areas equal to the core flow area. The result is consistent definitions of pressure loss for standard design (experimental tests), RELAX input, and FLEX input. The current procedure accomplishes the same end, but is not as apparent and more likely to create input errors.

The objective for revising the FLEX code was to reduce numerical problems. However, experience has shown that numerical difficulties can sometimes be further reduced through code input. To accomplish this the system node phase separation model and the mixture level in the upper plenum are important model specification (code input) parameters.

The standard application of the phase separation model to the system nodes (see Figure 5.2) using the currently approved evaluation model is as follows:

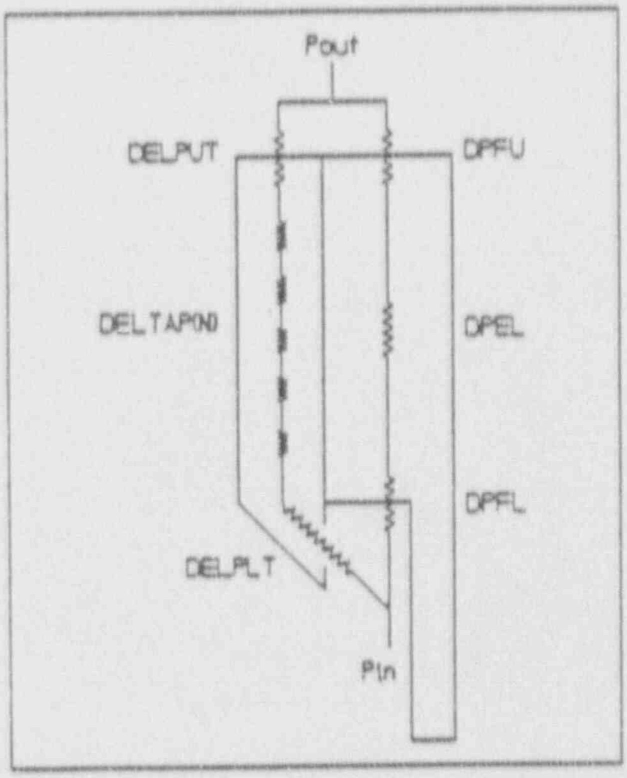

FIGURE 3.1

4.0 CONFORMANCE TO 10 CFR 50, APPENDIX K

The models and codes in the current ANF BWR ECCS Evaluation Model have been approved[1] as being in accordance with Appendix K. This upgrade to the Evaluation Model consists of improvements in RELAX and FLEX code numerics. The numerical improvements do not involve the modification of any Appendix K feature within these codes. Appendix K issues arise only in the conservative replacement of the Dougall-Rohsenow heat transfer coefficient in RELAX and correction of the decay heat specification in the heatup calculation. Otherwise, the Appendix K application of the current ANF Model remains in place, as originally approved by the NRC.

The current RELAX blowdown code, as part of the current ECCS EM, has been approved to be in accordance with Appendix K. In addition to being approved in its own right, its development path includes the EXEM NJP-BWR ECCS EM which was also approved[2] to be in accordance with Appendix K. The only changes to the current version of RELAX are modifications to improve code numerics and replacement of the Dougall-Rohsenow heat transfer model

These numerics modifications do not affect the conformance of the code to Appendix K criteria. Further, the use of the heat transfer correlation is a conservative change in response to new Appendix K requirements. As demonstrated in the comparisons shown, the impact of these changes result in higher peak cladding temperatures at the end of blowdown.

The FLEX refill/reflood code does not contain Appendix K criteria in its core model[1] because it is used to provide a prediction of fluid inventory during the ECC injection period. FLEX is not used as a heatup code for calculation of PCT. The required Appendix K criteria in the currently approved EM and in the revised EM that are applicable to heatup predictions are contained in HUXY and are unchanged. The changes to FLEX involved code logic that reduced purely numeric fluctuations about the mean solution and eliminated code failure when more

realistic core modeling was linked to ultra conservative approximations. These numerical difficulties have required code users to select alternate and limiting conservative problem description choices. The improvement is not obtained by less conservative models added to the code, but by re-enabling problem description choices which have previously been approved, but were not used because of the numerical problems they caused. This lack of model change is demonstrated by comparisons of the currently approved and revised FLEX code versions using the same input.

The fuel stored energy as calculated in RODEX2[3] and fuel deformation and thermal analysis as calculated for the heatup phase by HUXY[4,5] retain their Appendix K status since no alteration of methodology or modification of the codes have been made in the upgrade. However, it was discovered that the decay heat model in RELAX inadvertently applies a 1.2 multiplier to both the fission product and the Actinide terms. This application is more conservative than that required by Appendix K. This is because Appendix K only requires the multiplier be applied to the fission product decay component. The current EXEM BWR ECCS heatup methodology uses this unnecessarily conservative decay heat model from the start of blowdown until rated LPCI spray is predicted. This methodology is changed in the revised EXEM BWR ECCS methodology to be consistent with the currently approved decay heat model following the time of rated LPCS. The ANS Standards Committee (October 1971) fission product decay heat values times 1.2 and the Actinide decay heat without a multiplier will be used over the entire LOCA transient in the heatup calculation (HUXY code). The methodology for the blowdown (RELAX code) and the refill/reflood (FLEX code) phases are left unchanged.

5.0 VALIDATION OF EVALUATION MODEL

This section presents the results of validation tests of the revised EXEM BWR ECCS EM. This validation builds on the results obtained in the validation program presented in the submittal for the currently approved EXEM BWR ECCS EM[1]. Validation of the revised model is accomplished by comparison with experimental data and typical plant calculations made with the approved and revised codes. Comparison with experimental data is accomplished by using the revised EM to predict the behavior of Test 6406[18,20,21] as performed in the Two-Loop Test Apparatus (TLTA).

The TLTA experimental apparatus was used to obtain basic experimental data for hypothetical LOCA events in a BWR. These data were obtained between the middle 1970's and the early 1980's. Use of these data have been primarily to assess the bounding value of PCT as predicted by Evaluation Model methodology. Data from these experiments were used as part of the validation program for the currently approved EXEM BWR ECCS EM submittal. TLTA Test 6406/Run 1 was identified as a reference test[20] in the TLTA experimental program with a full sized full power bundle undergoing blowdown heat transfer with emergency core cooling. For this reason Test 6406 was selected as a key simulation for both the currently approved EXEM BWR ECCS submittal in 1980 and as support for this EM revision.

Test 6406 was run in Configuration 5 of TLTA. TLTA was a small integral facility with an electrically heated full sized bundle used to simulate the core. The facility was volumetrically scaled and included all major BWR component features important to a BWR LOCA. TLTA could sustain prototypic pressures and temperatures during the system transient.

Use of the complete methodology that is summarized in Section 2.0 was not attempted for the simulation of Test 6406. The RELAX and the FLEX codes were used to predict the system blowdown, refill, and reflood periods. Since the core in the TLTA experiment consisted of a single bundle, then analysis of the "hot channel of the core" does not require a separate RELAX (hot channel) run. Further, the EM heatup calculation performed by HUXY is not required because the purpose of the simulation was agreement with experimental results. This focus

implies all features of the test be simulated including actual bundle power and a fully operational ECCS. However, to assure that the remaining conservative features of the revised EM were evaluated, the simulation was performed in the Evaluation Model mode. In addition, the application of RELAX and FLEX, the nodalization of the TLTA experiment, and the code input preparation procedures are consistent with licensing use unless otherwise noted. In total, the results that are reported are expected to be conservative. This approach is consistent with the validation calculations performed in support of the currently approved BWR ECCS EM submittal.

Consistent with the submittal for the currently approved EXEM BWR ECCS EM, the actual electrically heated bundle power and a best-estimate-like break flow multiplier to the Moody saturated critical flow model were used. Use of the actual TLTA power value removes one of the major conservatisms in predicting cladding temperature during blowdown (the 20% addition to the ANS decay power). However, the cladding temperatures will still be affected by the full EM heat transfer model which includes the heat transfer correlation. The earlier submittal for the currently approved EM used a saturated critical flow multiplier of 0.8. This value was originally chosen to provide agreement between analytical and experimental results. However, Slater[22] states that "a discharge coefficient of 0.6 gave best agreement with experimental data for the 100% breaks .. whereas a value of 0.8 gave the best agreement for the 2% breaks". Though TLTA is a small facility, it is believed that the break is better described by the 100% designation.

Figure 5.1 shows the RELAX nodalization diagram for TLTA for blowdown calculations. This nodalization is the same as is being used on plant analyses except for geometric features unique to TLTA; i.e., nodalization associated with the shortened jet pumps, the guide tubes, the break, and the intact loop centrifugal pump suction line. The simulated break is in the pump discharge piping of the recirculation loop which is the right hand loop of the figure.

The reason for adding an additional node in the intact loop centrifugal pump suction line was to eliminate a damped flow oscillation that originated at the point of recirculation line uncovery. The oscillation resulted from the combination of: (1) rapid drop of the downcomer

liquid level defeating the junction smoothing model, (2) liquid level moving back and forth across the entrance to the recirculation line, and (3) liquid surging into the downcomer during the recirculation line uncovery period. Either a phase separated node in the recirculation line or an added node could have cured the difficulty. This phenomenon is not expected in a typical plant application because the downcomer level drops more slowly and the natural damping forces are larger.

In order to be consistent with plant EM analyses, the FLEX calculation was initialized using the conditions calculated by RELAX at the end of the blowdown period (LPCS at rated spray). Since the ECCS spray never reached a steady rated spray condition, the refill/reflood period was initiated when the LPCS reached a near asymptotic value (80% at 136 seconds). This time was characterized by a significant change in the slope of the LPCS delivery curve. The submittal for the currently approved ECCS EM began the FLEX calculation when only the HPCS reached a near asymptotic value (95% with LPCS at 65%). Nodal masses, pressures, passive heat conductor temperatures and heater rod temperatures were initialized by automatic data transfer from the RELAX results. The data transfer process employed the same methodology used in actual licensing analysis. The FLEX system nodalization used in the simulation of Test 6406 is shown schematically in Figure 5.2. This is the standard FLEX nodalization used in ANF EM analyses.

5.1 RELAX Simulation of TLTA Test 6406

The calculated and the measured system pressure responses in the steam dome are shown in Figure 5.3. In addition, the results reported in the submittal for the currently approved EXEM BWR ECCS EM[1] are included in this and subsequent TLTA RELAX figures with the designation "Approved RELAX" code.

PAGES 42 - 45 LEFT INTENTIONALLY BLANK

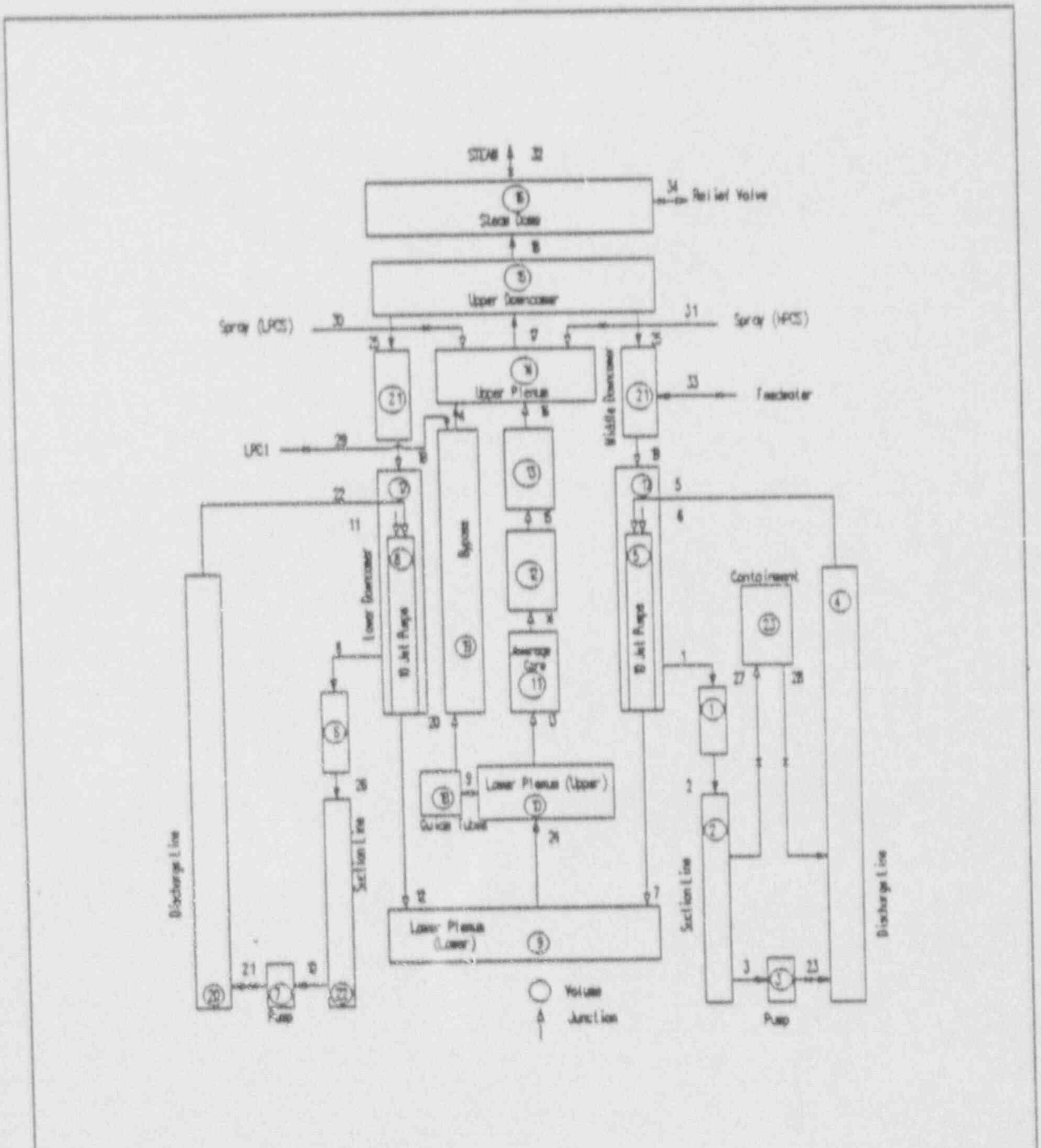
STEAM
Steam Dome
Relief Valve
Upper Downcomer
Spray (LPCS)
Spray (HPCS)
Upper Plenum
Feedwater
Middle Downcomer
LPCI
Lower Downcomer
10 Jet Pumps
Bypass
Average Core
Containment
Discharge Line
Suction Line
Lower Plenum (Upper)
Guide Tubes
Lower Plenum (Lower)
Pump
Volume
Junction

FIGURE 5.1

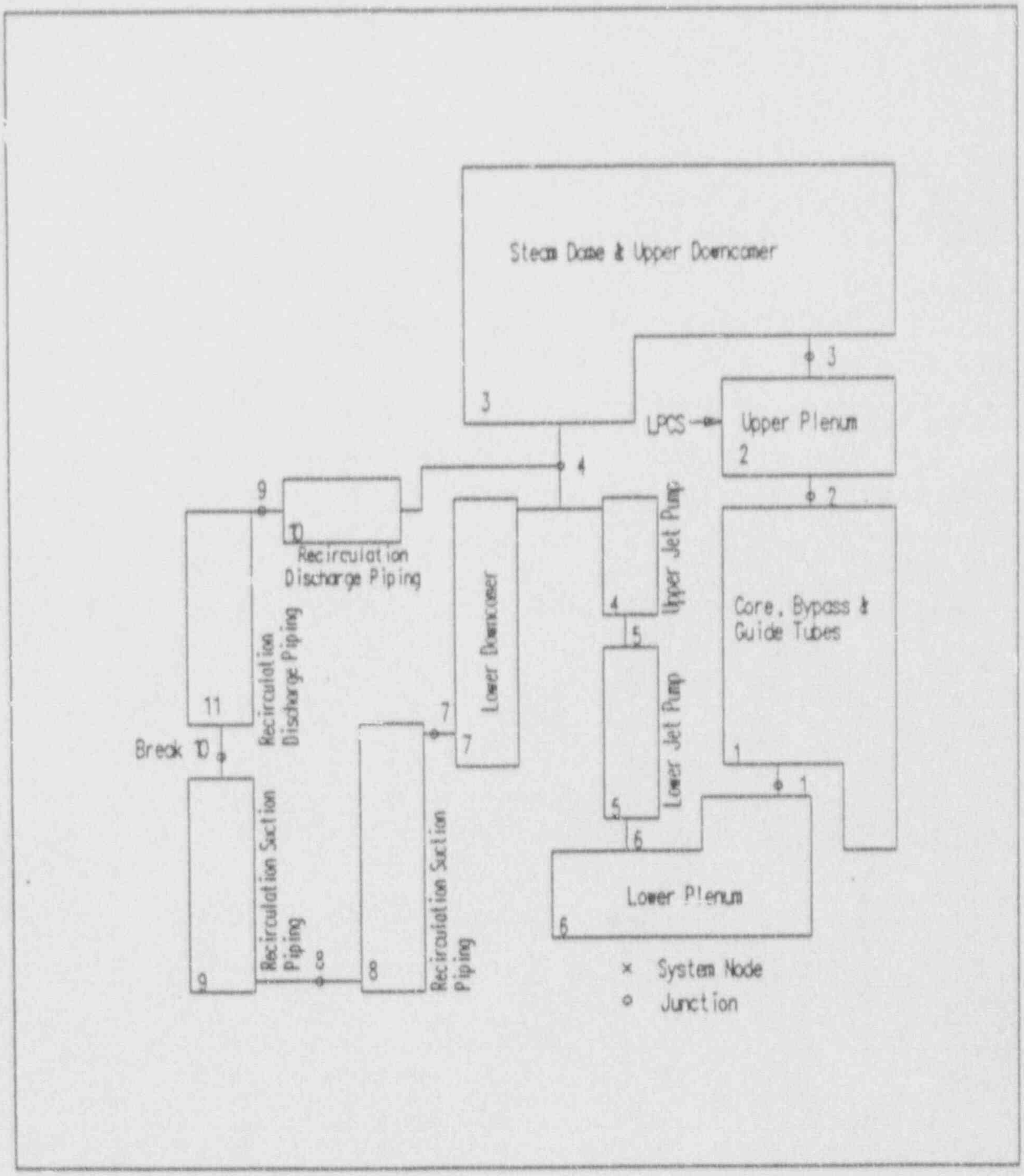
Steam Dome & Upper Downcomer
3
3
LPCS
Upper Plenum
2
4
2
9
10
Recirculation Discharge Piping
Lower Downcomer
Upper Jet Pump
4
5
Core, Bypass & Guide Tubes
Recirculation Discharge Piping
11
Break 10
7
7
Lower Jet Pump
5
1
1
6
Recirculation Suction Piping
Recirculation Suction Piping
Lower Plenum
6
9
8
× System Node
○ Junction

FIGURE 5.2

PAGES 48 - 53 LEFT INTENTIONALLY BLANK

6.0 ASSESSMENT OF EXEM BWR ECCS EM

Section 3.0 described the changes made to the codes and procedures that make up the revised EXEM BWR ECCS Evaluation Model. This section describes the impact of these changes on both individual code results and overall evaluation model results. Section 6.1 illustrates the separate effects of the major components of the RELAX upgrade and the sensitivity improvement inherent in the new code version. Section 6.2 describes the impact of the FLEX upgrade on each plant type that ANF has licensed. Section 6.3 assesses the impact of the revised evaluation model on the peak clad temperature for each of the plant types that ANF has licensed.

6.1 Revised RELAX Code Results

To assess the effect of the code revision three comparisons were made: (1) the currently approved code versus the upgraded code, (2) the upgraded code with and without the

and (3) the upgraded code with a tightening of the convergence criteria.

A comparison of the currently approved RELAX code and the revised RELAX code is provided in the Figures 6.1 through 6.4 for the blowdown phase of a typical BWR4 plant. The upgraded version of RELAX includes the modified heat transfer correlation.

A comparison of the upgraded RELAX code with the currently approved Dougall-Rohsenow correlation and with the modified correlation is provided in the Figures 6.5 and 6.6. Again, the comparison involves a typical BWR4 plant.

The sensitivity of the upgraded PELAX code to the convergence criteria is shown in Figures 6.7 through 6.9.

6.2 Revised FLEX Code Results

To assess the effect of the code upgrade a comparison of the currently approved FLEX code and revised FLEX code results is provided for three different jet pump BWR plant types. The input and input preparation procedures for the FLEX calculations are consistent with the revised evaluation model unless otherwise noted.

6.3 Assessment Applications to BWR Plant Types

This section provides the assessment of the overall effects of applying the revised EXEM BWR EM to typical BWR plant types.

6.3.1 <u>Purpose of Assessment</u>

Changes have been made to the EXEM BWR ECCS evaluation model. The purpose of this assessment is to determine the overall effects of the combined changes on LOCA analysis applications for the various BWR plant types for which ANF provides reload fuel. Questions to be answered include: (1) What are the approximate peak cladding temperatures and extent of metal-water reaction that will be predicted using the new EXEM BWR models? (2) Will the revised EXEM BWR evaluation model predict a different break size or location to be limiting relative to current analyses of record? (3) Will a different assumed worst single failure be more limiting using the revised models?

As noted in Section 3.1, a number of changes have been made to the RELAX code used to calculate the blowdown portion of the LOCA event. Many of these changes are numerical in nature, and while they significantly improve the efficiency and reliability of the code, calculated results are not significantly altered by these changes. Replacement of the Dougall-Rohsenow film boiling heat transfer correlation

will result in reduced heat transfer coefficients and higher temperatures during the blowdown portion of the LOCA.

Following blowdown, the conservative refill or spray cooling period is calculated using the FLEX system code. The principal result of the FLEX code is the time of core reflood. The version of the FLEX code used in current licensing analyses has frequently predicted oscillatory core reflood behavior when using the NRC approved options.

Changes to the FLEX code are expected to eliminate the artificial oscillatory reflood behavior and to allow calculation of a sustained reflood prior to system depressurization.

6.3.2 Assessment Calculations Performed

To assess the effects of revisions to the EXEM BWR evaluation model, large break LOCA calculations were performed for three different jet pump BWR systems: a BWR3, a BWR4, and a BWR6. LOCA behavior and ECC systems in a BWR5 are similar to the BWR6, and the effects of the EXEM BWR models when applied to a BWR5 are expected to be similar to those calculated for the BWR6. These reactor systems encompass the BWR systems using ANF reload fuel.

Current licensing analyses and limits are based on the limiting or worst case LOCA calculation with regard to break size, break configuration, and break location. The limiting break LOCA is determined in part by the duration of the refill time period. Since the revised FLEX code can give different reflood times, the worst or most limiting LOCA break size, location, and configuration could change due to the model revisions. To examine this possibility, a partial break spectrum analysis was performed. The BWR4 plant model which currently has long refill times was chosen for the break spectrum calculations.

It is required that LOCA analysis include a single failure of the ECCS component having the most severe effect on LOCA results. Generally this is the ECCS single failure which yields the lowest ECCS injection rate into the primary system and has been identified by the NSSS vendor. BWR5 and BWR6 reactors have a high pressure core spray (HPCS) ECCS which injects significant ECC water into the system during the blowdown while the system is depressurizing.

Other ECC systems operate later at lower pressure but have higher injection rates. For these systems, the worst single failure may depend on the cooling calculated by the ECCS evaluation model due to the functioning of the HPCS system . To determine whether the worst single failure might change due to the revised EXEM BWR models, an additional single failure calculation was performed for the BWR6 reactor type.

6.3.3 Results for Plant Types Using Revised EXEM BWR ECCS EM

To assess the effects of revisions to the EXEM BWR models on LOCA analysis applications for the BWR3 reactor type, ANF calculated a previously identified limiting large break LOCA for 100% power and 100% flow using the revised EXEM BWR code versions. The LOCA was calculated for the double-ended guillotine break of the recirculation suction line with a discharge coefficient of 1.0 (1.0 DEG/RS). The calculated event times, PCT, and extent of local metal-water reaction using the revised EXEM BWR codes are shown in Table 1. Event times during blowdown prior to the calculated time of rated low pressure core spray change little from the analysis performed with the currently approved model.

Similarly, a typical large break LOCA calculation was performed for the BWR4 reactor system. Results presented for the BWR4 reactor are for the most limiting LOCA break as determined from a partial analysis of the LOCA break spectrum using the revised EXEM BWR codes. The most limiting break from the partial LOCA break spectrum is the large double-ended guillotine break of the recirculation pump discharge line (1.0 DEG/RD). The results of the LOCA calculation for 1.0 DEG/RD break are shown in Table 1 for the BWR4 reactor.

Decreased calculated PCTs result from the shorter refill or spray cooling time period.

Typical results are also shown in Table 1 for the BWR6 system. As with the BWR3 the previously identified limiting break LOCA (1.0 DEG/RD) was recalculated using the revised EXEM BWR evaluation model codes. However, as discussed below using the revised EXEM BWR models, the worst single failure is calculated to change. The results shown in Table 1 for the BWR6 are for the most limiting worst single failure case calculated by ANF which assumes loss of the high pressure core spray (HPCS) system.

The effects of the use of the revised EXEM BWR models on LOCA break spectrum calculations were examined by performing a partial recalculation of the break spectrum for the BWR4 system. Refill time is affected by the EXEM BWR phase separation model selected and the relative refill time for the LOCA breaks can effect the limiting break LOCA from the spectrum, particularly for the BWR4 system. Therefore it is necessary to perform additional break spectrum calculations for each reactor type to redefine the limiting LOCA break appropriate for defining limits using the revised EXEM BWR models. To demonstrate this sensitivity, typical LOCA calculations were performed for six assumed break conditions using the revised EXEM BWR models. The six recirculation discharge line breaks calculated were: 1.0 DEG, 0.6 DEG, 0.4 DEG, 0.4 DES, 0.3 DES, and 0.1 DES.

Applicability of the assumed worst single failure using the revised EXEM BWR evaluation model was assessed by calculations for the BWR6 plant type. These calculations showed that the worst single failure could change from that determined for previous analyses. Thus, the worst single failure assumption may differ with the revised evaluation model. Consequently, the worst single failure assumption should also be addressed for the initial application of the revised EXEM BWR model to a BWR plant type.

6.3.4 Conclusions From Assessments

Conclusions from the assessment calculations can be summarized as follows:

- Application of the revised EXEM BWR evaluation model allows the use of different approved phase separation models

- LOCA break spectrum results can change due to the shortened LOCA refill time. New break spectrum calculations using the revised EXEM BWR evaluation model are necessary to determine the worst or limiting break LOCA when the revised evaluation model is initially applied to each plant type.

- The assumed worst single failure can change using the revised EXEM BWR model. Initial application of the revised EXEM BWR to a plant type must identify the worst single failure to be assumed in LOCA analyses for the plant type.

PAGES 63 - 74 LEFT INTENTIONALLY BLANK

7.0 REFERENCES

1. Exxon Nuclear Methodology For Boiling Water Reactors: EXEM BWR ECCS Evaluation Model, XN-NF-80-19(P)(A), Volumes 2, 2A, 2B, 2C, Exxon Nuclear Company, Inc., September, 1982.

2. The Exxon Nuclear Company WREM-Based NJP-BWR ECCS Evaluation Model and Application to the Oyster Creek Plant, XN-NF-75-55(A), Revision 2, Exxon Nuclear Company, Inc., April, 1977.

3. RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 2, Supplements 1 & 2, Exxon Nuclear Company, Inc., March, 1984.

4. HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option - User's Manual, XN-CC-33(A), Revision 1, Exxon Nuclear Company, Inc., November 1, 1975.

5. BULGEX: A Computer Code to Determine the Deformation and the Onset of Bulging of Zircaloy Fuel Rod Cladding, XN-74-21, Revision 2 and XN-NF-27, Revision 2, Exxon Nuclear Company, Inc., December 31, 1974.

6. A User's Guide for the BWR Refill and Reflood Code FLEX, XN-CC-50, Exxon Nuclear Company, Inc., January, 1982.

7. FLEX User's Guide, USEP87, ANF-88-067(P), Advanced Nuclear Fuels Corporation, February, 1989.

8. Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model, XN-NF-75-41(A), Volume 1, Volume 2 - Appendix A-D, Volume 3 (Rev 2) Supplement 1-4, Supplement 5 (Rev 1), Supplement 6-7, Exxon Nuclear Company, Inc., 1975.

9. Appendix K to 10 CFR 50.

10. D. C. Slaughterbeck, L. J. Ybarrondo, C. F. Obenchain, "Flow Film Boiling Heat Transfer Correlations - Parametric Study with Data Comparisons", 73-HT-50, ASME-AIChE Heat Transfer Conference, Atlanta, Ga., August 5-8, 1973.

11. T. A. Porsching, J. H. Murphy, J. A. Redfield, "Stable Numerical Integration of Conservation Equations for Hydraulic Networks", Nuclear Science and Engineering, Vol.43 (February 1971) pp 218-225.

12. M. Ishii, One-Dimensional Drift-Flux Model And Constitutive Equations For Relative Motion Between Phases In Various Two-Phase Flow Regimes, ANL-77-47 (October 1977).

13. K. Ohkawa, R. T. Lahey, Jr., The Analysis of Proposed BWR Inlet Flow Blockage Experiments in PBF, NES-486, (December 1978).

14. M. Ishii, I. Kataoka, G. Kocamastafaogullari, "Phenomenological Modeling of Two-phase Flow in Water Reactor at ANL", NRC 9th Water Reactor Safety Research Information Meeting, Oct. 26-30, 1981, Gaithersburg.

15. D. M. McEligot, L. W. Ormand, H. C. Perkins, "Internal Low Reynolds Number Turbulent and Transitional Gas Flow with Heat Transfer", Journal of Heat Transfer, Trans. of the ASME, May 1966.

16. Analytical Model for Bottom Reflooding Heat Transfer in Light Water Reactors (The UCFLOOD Code), EPRI-NP-756, August 1978.

17. Loss-of-Coolant Accident and Emergency Core Cooling Models for G.E. Boiling Water Reactors, NEDO-10329, April 1971.

18. WREM: Water Reactor Evaluation Model, NUREG-75/056, Revision 1, Division of Technical Review, NRC, May, 1975.

19. BWR Blowdown/Emergency Core Cooling Preliminary Facility Description Report for the BD/ECC1A Test Phase, GEAP-23592, December 1979.

20. BWR Blowdown/Emergency Core Cooling Fifteenth Quarterly Progress Report, July 1-September 30, 1979, NUREG/CR-1154 Vol. 1, No. 3, GEAP-21304, February, 1980.

21. Electric Power Research Institute, BWR Large Break Simulation Tests - BWR Blowdown/Emergency Core Cooling Program, Volume 2: Appendices I-N, NUREG/CR-2229, EPRI NP-1783, GEAP-24962-2, Vol. 2, March 1981.

22. C. E. Slater, Comparison of Predictions from the Reactor Primary System Decompression Code (RELAP3) with Decompression Data from the Semiscale Blowdown and Emergency Core Cooling (ECC) Project, DR-1680, Idaho Nuclear Corporation, December 1970.

SIEMENS

Issue Date: 1/13/93

July 6, 1992
RAC:091:92

Dr. Lambros Lois
Reactor Systems Branch
Division of Engineering and System Technology
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Lois:

Reference: ANF-91-048(P), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model," Siemens Nuclear Power Corp., July 1991.

Enclosed are the responses to the two additional questions you had concerning the revised EXEM BWR large break LOCA model (Reference) that you are currently reviewing. You requested this information in our telephone conversation in June 1992. Please consider the information contained in these responses to be proprietary to Siemens Nuclear Power Corporation. The affidavit provided with the original submittal of the topical report should satisfy the requirements of 10 CFR 2.790(b) for withholding these responses from public disclosure.

If you need additional information, or if I can be of further help, please contact me at (509) 375-8290.

Very truly yours,

Bob Copeland

R. A. Copeland
Manager, Reload Licensing

Siemens Nuclear Power Corporation

2101 Horn Rapids Road, PO Box 130 Richland, WA 99352-0130 Tel: (509) 375-8100 Fax: (509) 375-8402

bc: SE Jensen
RR Schnepp
LG Riniker
DC Kolesar
HD Curet
LJ Federico
GL Ritter
HE Williamson
file

ADDITIONAL RESPONSES TO NRC CONCERNS ON TOPICAL REPORT ANF-91-048(P)[1]

CONCERN: The SNP responses related to the modified Dougall-Rohsenow heat transfer correlation show that the correlation does not bound the ORNL data. For an Appendix K LOCA-ECCS evaluation model, the correlation must be conservative relative to the data.

RESPONSE: - Overall, the modified Dougall-Rohsenow correlation proposed by SNP yields and the correlation was submitted on this basis. However, at higher pressures and higher quality represented by the ORNL data, the correlation predictions

To provide more conservative predictions

SNP repeated the previous comparison with the ORNL data (Reference 2 Figure 11)

CONCERN: The TLTA test facility is 1/624 scale relative to a large BWR, provide a basis for scaling TLTA results to a large BWR system.

RESPONSE: - The TLTA is an integral test facility designed with the objective of providing typical BWR system behavior on a real time basis. The 1/624 scaling is based on a large BWR/6 which contains 624 fuel assemblies. With this scaling, the TLTA core is a single full-scale fuel assembly which is an electrically heated prototype of an 8x8 BWR assembly. The system is power-to-volume scaled, and the break is scaled on a break area-to-system volume basis. In so far as possible, length dimensions are maintained prototypic of the large BWR system.

With this type of scaling, the time scale, fluid mass and energy distributions, velocities, accelerations and lengths are essentially the same in the test facility as in the prototype plant.(3) This scaling is intended to provide accurate simulation of rate dependent phenomena in the test facility which can be applied directly to the large scale system.

For the blowdown portion of the Large Break LOCA, tests performed with facilities of varying scale for both PWRs and BWRs have confirmed the adequacy of power-to-volume scaling. That is, similar tests performed in systems of different scales indicate the same controlling phenomena, and no significant differences due to scale have been observed which need to be considered in calculating these phenomena for a full-scale reactor relative to small scale test facility. Thus, phenomena observed for the TLTA blowdowns are representative of the expected phenomena in large BWRs. Core boundary conditions in the tests are representative of expected BWR conditions, and the full-scale fuel assembly behavior will be prototypic of the large BWRs. Thus, the TLTA provides a full-scale simulation of the hot channel behavior during blowdown. The EXEM/BWR benchmark demonstrates that EXEM/BWR conservatively predicts TLTA behavior during blowdown.

Effects due to scale have been observed in the refill-reflood period of a BWR LOCA.(4) With power-to-volume scaling, component length is preserved but component diameters are significantly reduced. As a result, multidimensional effects which could occur in the reactor vessel plenum regions of a large BWR may not be observed in small scale tests such as TLTA. Effects resulting from multiple assemblies in a BWR will also not be present in the single bundle TLTA core. For example, the flow of ECC fluid from the upper plenum to the lower plenum in TLTA is via leakage paths and countercurrent flow in the heated bundle. Larger scale tests simulating a multidimensional reactor vessel upper plenum and parallel channels of varying power have shown effects that permit ECC fluid to flow more readily to the lower plenum through low power assemblies than observed in the single channel TLTA. Thus, the time of reflood observed in the TLTA tests will be delayed compared to the expected reflood time for an actual large BWR. The general conclusion from larger scale refill-reflood tests as found on page 6.5-11 of Reference 4 is:

> "These tests showed significant multidimensional and multichannel effects, which are in general, beneficial and result in more effective refill and reflood than that observed in one dimensional tests."

Additional Responses to NRC Concerns
on Topical Report ANF-91-048(P)

The EXEM/BWR refill and reflood model is based on the controlling phenomena of TLTA, and predicts time of reflood consistent with TLTA results. When applied to a large BWR, EXEM/BWR will then predict a delayed reflood time relative to that expected. With the EXEM/BWR model, delaying reflood results in an increase in calculated PCT. Thus, not including the scaling effects benefits during refill and reflood results in a conservative LOCA analysis using EXEM/BWR.

REFERENCES:

(1) "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model," ANF-91-048(P), Siemens Nuclear Power Corporation, July 1991.

(2) Letter, R. A. Copeland (SNP) to Dr. Lambros Lois (NRC), RAC:92:064, dated May 27, 1992.

(3) A. N. Nahavandi, F. S. Castellana, and E. N. Mordkhanian, "Scaling Laws for Modeling Nuclear Reactor Systems," Nuclear Science and Engineering, Vol. 72, Pages 75-83, 1979.

(4) "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230 Final Report, Section 6.5, USNRC, December 1988.

SIEMENS

May 27, 1992
RAC:92:064

Dr. Lambros Lois
Reactor Systems Branch
Division of Engineering and System Technology
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Lois:

Reference: "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model," ANF-91-048(P), Siemens Nuclear Power Corp., July 1981.

Attached is the additional information that you requested to support your review of the referenced topical report. This information was requested in a telephone conversation with Daryl Hershberger in April 1992. Please consider the information contained in these responses to be proprietary to Siemens Nuclear Power. The affidavit required by 10 CFR 2.790(b) to support withholding the responses from public disclosure was provided with the original submittal.

If there are questions, or if additional information is needed, please call Daryl Hershberger or me.

Very truly yours,

Bob Copeland

R. A. Copeland
Manager, Reload Licensing

ecrnm

cc: Mr. R. C. Jones (NRC)
Mr. D. E. Hershberger (SNP)

Siemens Nuclear Power Corporation

Engineering and Manufacturing Facility

2101 Horn Rapids Road, PO Box 130 Richland WA 99352-0130 Tel: (509) 375-8100 Fax: (509) 375-8402

ADDITIONAL INFORMATION
on

Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model (ANF-91-048(P))

Concern 1

> In at least one response, SNP stated that a model had not been changed. SNP should provide information to justify that interaction with models that have changed does not cause the unchanged models to produce non-conservative results.

Response

With regard to possible interactions such that model changes interact with unchanged portions of the model in a non-conservative way, such interactions, if significant, would show up as a difference between calculated results using the old methodology and similar results using the revised models. SNP has not observed any indication of such interactions in the results obtained to date.

Comparisons of results using the changed model with results from the currently approved model are shown in Figures 6.1-6.4 and Figures 6.10-6.15 of ANF-91-048(P). The results using the revised EXEM BWR model are in close agreement with those of the approved model, and there is a general reduction in the oscillations calculated for the flow results. Differences are evident in Figure 6.4 and Figures 6.14-6.15, but these differences relate directly to specific model changes. Thus, the overall comparison of revised model results with approved model results indicates that interactions of the model changes with unchanged portions of the model are insignificant. Likewise, results of the BWR plant calculations using the revised EXEM BWR model are very similar to previous licensing calculations except where model changes have a direct effect on results.

Recirculation pump behavior is one area where the analytical models were not changed but NRC questions indicated concern regarding possible interaction of model changes. Attached Figures 1-6 show calculated pump model results for a BWR-4 using the approved EXEM BWR model (old methodology) and similar results obtained using the revised EXEM BWR model (new methodology). Comparing the calculated results shows that pump parameter results are essentially unchanged by the EXEM BWR model revisions. Thus, interactions of the EXEM BWR model revisions with the recirculation pump models of the approved methodology are not significant.

Figure 1 - BWR-4 Intact Loop Pump Speed (Old Methodology)

Figure 2 - BWR-4 Intact Loop Pump Head (Old Methodology)

Figure 3 - BWR-4 Intact Loop Pump Flow (Old Methodology)

Figure 4 - BWR-4 Intact Loop Pump Speed (New Methodology)

Figure 5 - BWR-4 Intact Loop Pump Head (New Methodology)

Figure 6 - BWR-4 Intact Loop Pump Flow (New Methodology)

ADDITIONAL INFORMATION
on

Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model (ANF-91-048(P))

Concern 2

> SNP describes the application of the revised RELAX and FLEX codes to the TLTA results but does not provide a basis for transformation from application to TLTA to application to a BWR. SNP should provide such a basis.

Response

The Two Loop Test Apparatus (TLTA) is an integral experimental facility designed to simulate BWR behavior under LOCA conditions, particularly the DBA large break LOCA. The volume scaling factor is 1/624 to a BWR-6 reactor.

The experimental facility includes many features representative of a large BWR such as:

1. Two recirculation loops, each containing a jet pump which simulates jet pump phenomena during a LOCA in both intact and broken loops.

2. The pressure vessel and internals simulate the major fluid regions, flow paths, and hardware of a large BWR.

3. The core is represented by a full length electrically heated bundle capable of simulating decay heat power.

4. Various typical ECC systems can be simulated in TLTA.

5. TLTA simulates auxiliary system behavior of a BWR including steam flow isolation, feedwater flow isolation, pressure control, and automatic depressurization systems.

Because of these features, the TLTA experiments simulate the important LOCA phenomena for a large break in a BWR, including the following (from the Compendium of ECCS Research for Realistic LOCA Analysis, NUREG 1230 Table 5-4):

1. Break Flow.

2. Channel and bypass axial flow and void distribution.

3. CCFL at UCSP and channel inlet orifices.

4. Core heat transfer including DNB, dryout, RNB, surface to surface radiation.

Additional Information on ANF-91-048(P)
Concern 2 (continued)

5. Entrainment and de-entrainment in core and upper plenum.

6. Separator behavior.

7. Recirculation pump behavior.

8. Phase separation and mixture level behavior.

9. Guide tube and lower plenum flashing.

10. Mixture level in downcomer.

The capability of analytical models to predict basic phenomena applies for both experimental facilities and for large BWR systems where these phenomena occur. Since the TLTA was designed to simulate BWR LOCA phenomena, the capability to predict the important phenomena in the TLTA indicates similar predictive capability for the large BWRs. Thus, SNP demonstrated the capability of the revised EXEM BWR evaluation model to predict phenomena associated with a BWR LOCA by comparison with TLTA measured results for these phenomena.

ADDITIONAL INFORMATION
on

Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model (ANF-91-048(P))

Concern 3

> SNP has provided a comparison of the modified Dougall-Rohsenow correlation with the Dougall-Rohsenow correlation to demonstrate that the former is conservative. Since the NRC considers the Dougall-Rohsenow correlation to be non-conservative, SNP's comparison does not demonstrate that the modified Dougall-Rohsenow correlation is conservative or applicable when applied to reactor conditions. SNP should provide a comparison of calculated results using SNP's code with measured data.

Response

Justification of the correlation was done by comparing calculated results against test data over the appropriate range of BWR LOCA conditions. SNP has attached additional justification for this correlation. The additional justification includes:

1. Comparisons of the base correlation against additional test data over the appropriate BWR LOCA range of conditions to strengthen the justification for the base correlation,

2. Code calculations against steady-state bundle test data to provide justification for application of the correlation with the RELAX code using calculated bundle conditions, and

3. Additional information relative to the effects of using the correlation on the results of the integral TLTA test predictions to demonstrate conservatism of the overall EXEM BWR model in computing LOCA cladding temperatures.

Base Correlation Justification - In selecting the modified Dougall-Rohsenow (MDR) post CHF correlation to replace the non-conservative Dougall-Rohsenow (DR) correlation, SNP evaluated the MDR correlation against three major sources of data:

1. The data comparisons from the ASME heat transfer paper 73-HT-50[1],

2. Data from the Oak Ridge National Laboratory (ORNL) Thermal-Hydraulic Test Facility (THTF)[2], and

3. Combustion Engineering test data from the Columbia University Heat Transfer Test Facility (CE/Columbia tests)[3].

Additional Information on ANF-91-048(P)
Concern 3 (continued)

The ASME paper compared several correlations against test data, and showed comparisons for the MDR correlation for ranges of pressure, wall temperature, fluid quality, and mass flux. The conclusion from these comparisons is that

The ORNL data included 22 steady-state tests, however, only the tests with system pressures applicable for BWR LOCA conditions were considered. Attached Figure 11 shows a summary of the MDR predicted heat transfer coefficient versus the observed test values. Again, the correlation predicted the data fairly well for all the parameter ranges. Pressure data were predicted well between 800-1000 psi.

The CE/Columbia tests cover a more extensive range of conditions than the ORNL THTF tests, and the summary comparison of MDR predicted versus observed data is shown in Figure 12, attached.

Justification of Correlation Against Separate Effects Data - To assess the capability of the modified Dougall-Rohsenow correlation in conjunction with the EXEM BWR RELAX code to predict post CHF heat transfer, SNP performed additional calculations for several of the ORNL steady state post CHF heat transfer tests. For these calculations, the test bundle was modeled and test conditions were input at the bundle inlet. Heat transfer and fluid conditions within the bundle were then calculated using the RELAX code. Calculations were performed for the ORNL tests shown in Table 1, and the calculated temperature results compared to the measured temperatures are given in Figures 13-18. The temperatures shown are limited to the bundle region where post CHF heat transfer using the modified Dougall-Rohsenow was calculated with RELAX.

Additional Information on ANF-91-048(P)
Concern 3 (continued)

Justification of Correlation Against Integral Test Data - The final test of the LOCA evaluation model and associated correlations is the conservative prediction of peak cladding temperature. Large integral system test data representative of BWR conditions are the best available data against which to evaluate the overall evaluation model results. Figure 5.8 of ANF-91-048(P) shows the RELAX predicted peak rod temperature versus the measured TLTA integral test data.

References:

1. D. C. Slaughterbeck, L. J. Ybarrondo, and C. F. Obenchain, "Flow Film Boiling Heat Transfer Correlations - Parametric Study and Data Comparisons," ASME Paper 73-HT-50, August 1973.

2. G. L. Yoder, et. al., "Dispersed Flow Film Boiling in Rod Bundle Geometry - Steady State Heat Transfer Data and Correlation Comparisons," NUREG/CR-2435 (ORNL-5822), Oak Ridge Tenn., March 1982.

3. D. M. Turner, "Blowdown Heat Transfer Program - Film Boiling Heat Transfer Coefficients (CE/Columbia Tests)," CENPD-152 Rev. 1, Windsor, Conn., October 1975.

Figure 7 - Heat Transfer Coefficients as a Function of Pressure

Figure 8 - Heat Transfer Coefficients as a Function of Wall Temperature

Figure 9 - Heat Transfer Coefficients as a Function of Quality

Figure 10 - Heat Transfer Coefficients as a Function of Mass Flux

Figure 11 - Calculated vs Measured Heat Transfer Coefficient (ONRL Data Summary)

Figure 12 - Calculated vs Measured Heat Transfer Coefficient
(CE/Columbia Data Summary)

Siemens Nuclear Power Corporation Proprietary

ADDITIONAL INFORMATION
on

Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors
EXEM BWR ECCS Evaluation Model (ANF-91-048(P))

Table 1 - ORNL Test Data for [illegible] AX Calculations

Test Case	Pressure (psia)	Mass Flow (lb/s)	Bundle Power (Mw)	ORNL Reference
K	634.8	3.05	2.89	NUREG/CR-2435, ORNL-5822
O	866.7	4.15	3.51	NUREG/CR-2435, ORNL-5822
P	874.3	7.03	5.37	NUREG/CR-2435, ORNL-5822
R	952.4	4.91	4.13	NUREG/CR-2435, ORNL-5822
IF	584.1	2.32	2.21	NUREG/CR-3502, ORNL/TM-8794
IG	564.6	3.49	2.91	NUREG/CR-3502, ORNL/TM-8794

Figure 13 - Calculated vs Measured Temperatures (ORNL Test K)

Figure 14 - Calculated vs Measured Temperatures (ORNL Test O)

Figure 15 - Calculated vs Measured Temperatures (ORNL Test P)

Figure 16 - Calculated vs Measured Temperatures (ORNL Test R)

Figure 17 - Calculated vs Measured Temperatures (ORNL Test IF)

Figure 18 - Calculated vs Measured Temperatures (ORNL Test IG)

SIEMENS

January 28, 1992
LB/RAC:005:92

Mr. R. C. Jones, Chief
Reactor Systems Branch
Division of Engineering and System Technology
Office of Nuclear Reactor Regulation
Mail Station P1-137
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Jones:

Reference: Letter, R. C. Jones (USNRC) to R. A. Copeland (SNP), "ANF-91-048(P), Request for Additional Information (TAC No. M81274)," December 20, 1991.

Attached are the responses to the questions transmitted in the referenced letter. This additional information is in support of the updated EXEM BWR large break LOCA evaluation model review. Please consider the information in the responses to be proprietary to Siemens Nuclear Power Company. The affidavit accompanying the original submittal should comply with the 10 CFR 2.790(b) requirements to withhold this information from public disclosure.

If there are additional questions or if I can be of further help, please contact me at (509) 375-8290.

Very truly yours,

Bob Copeland

R. A. Copeland
Manager, Reload Licensing

cc: Dr. L. Lois (USNRC)

Siemens Nuclear Power Corporation

2101 Horn Rapids Road, PO Box 130 Richland, WA 99352-0130 Tel: (509) 375-8100 Fax: (509) 375-8402

RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION ON TOPICAL REPORT ANF-91-048(P)

Question 1

In the evaluation model (EM) description, it is stated that RODEX2 is used to ensure that the initial stored energy estimated in RELAX and in HUXY is the same or higher than [that] estimated in RODEX2.

What does the model provide if the estimated RODEX2 energy is smaller than the RELAX and HUXY estimates?

Response

Appendix K to 10 CFR 50 requires that a conservative initial stored energy be used in LOCA analyses. NRC has reviewed and approved SNP's RODEX2 code for this purpose. To assure that SNP's transient LOCA codes, RELAX and HUXY, use an initial stored energy consistent or conservative with respect to the approved RODEX2 results, SNP procedures require that the initial average fuel temperature (stored energy) calculated by RELAX and HUXY be equal to or greater than that calculated by RODEX2 for identical conditions.

This method for assuring a conservative initial stored energy is part of the currently approved EXEM/BWR model and is unchanged by the revisions described in ANF-91-048(P).

Responses to NRC Questions
Topical Report ANF-91-048

January 27, 1992

Question 2

> Are there break sizes for which LPCS (due to high pressure) never reaches rated flow? If yes, what is the event sequence and how is the water level determined?

Response

A break size for which LPCS (due to high pressure) never reaches rated flow has not been calculated in SNP analyses. For large break LOCAs analyzed by SNP the breaks are sufficiently large that depressurization and rated LPCS flow are calculated to occur. For smaller break size LOCAs calculated by SNP, actuation of the automatic depressurization system (ADS) has been predicted. This system in effect converts a small break into a large break sufficient to depressurize the reactor system and permit the low pressure ECC systems (such as LPCS) to function.

Responses to NRC Questions
Topical Report ANF-91-048

January 27, 1992

Question 3

Following the time of bypass reflood, the heat transfer coefficient on the bypass side of the canister beneath the bypass mixture level is set at a "conservative" value based upon the modified Bromley pool film boiling correlation.

It is not clear what constitutes a conservative value and whether such value can be found for all break sizes. Please comment.

Response

This treatment of canister heat transfer is part of the currently approved EXEM/BWR model, and is not changed by the revisions submitted in ANF-91-048(P).

Responses to NRC Questions
Topical Report ANF-91-048 January 27, 1992

Question 4 In the revised EXEM BWR ECCS EM, it is not stated whether the estimated DT greater than 50°F is positive or negative. Please provide the sign.

Response

The final calculated peak cladding temperature using the revised EXEM/BWR Evaluation Model is expected to decrease by more than 50°F when compared with calculations performed with the current code versions.

The statement on Page 1 of ANF-91-048(P) regarding the estimated ΔPCT of 50°F relates to the accumulation of the effects of minor changes to the approved Evaluation Model which have been periodically reported to NRC. These accumulated changes for the current SNP EXEM/BWR Evaluation Model are approaching a PCT decrease of 50°F. If the accumulated PCT decrease were to exceed 50°F, the revised ECCS rule requires that the Dougall-Rohsenow heat transfer correlation in the current Evaluation Model no longer be used under conditions where it is non-conservative. A major reason for submittal of the revised EXEM/BWR model at this time is to address the issue of replacing the non-conservative Dougall-Rohsenow correlation.

Responses to NRC Questions
Topical Report ANF-91-048

January 27, 1992

Question 5

> In the measure of the convergence, the terms Uerr and Perr are calculated, however, the definition of these terms requires the value from the solution which at the time is not known. If the values are functions of time step, then the definitions should be modified. Please comment.

Response

The convergence test is performed at the end of the calculation for each time step, at which time the latest iterate values for all terms at the end of that time step are known. Uerr and Perr are calculated for each iteration of each time step and must be less than the given convergence value before starting the next time step.

Responses to NRC Questions
Topical Report ANF-91-048

January 27, 1992

Question 6

> In the critical flow model correction, it seems that the objective is to prevent flow reversal against the pressure gradient. However, neither how it is done or what is the physics of the model and its impact on the EM are discussed. Please elaborate.

Response

The objective of the critical flow model correction is to prevent premature termination of the calculation due to the use of the wrong donor volume fluid properties in the critical flow table search.

Responses to NRC Questions
Topical Report ANF-91-048

January 27, 1992

Question 7

As in 6 above, there is no explanation of the logic added to mitigate the bubble model degeneracies. Please discuss.

Response

When a volume with a mixture level becomes completely full or completely empty of liquid, there is a jump or step change in the corresponding junction properties for junctions at the top or bottom of the volume. Such a discontinuity can introduce non-physical oscillations in the simulation and prevent convergence since the junction properties can alternate across the discontinuity on successive iterations. To prevent such numerical oscillations,

Responses to NRC Questions
Topical Report ANF-91-048

January 27, 1992

Question 8

On the drift flux model: Why was the Ishii correlation used since it was known to have a built-in inconsistency with the Kutateladze flooding correlation? What is the physical basis for the smoothing α function?

Response

The Ishii correlation was chosen and implemented as part of the original EXEM/BWR model. At that time, SNP was not aware of an inconsistency with the Kutateladze flooding correlation, and no problems have been observed in SNP analyses due to this inconsistency. The EXEM[illegible] revisions described in ANF-91-048(P) do not change the basic drift flux model correlatio[illegible] address numerical problems which can occur under specialized conditions.

Question 9:

Isn't the upper limit of 1800°F for the modified Dougall-Rohsenow correlation low? What is the corresponding limit for the present Dougall-Rohsenow correlation?

Response

Question 10

In Section 3.0, the modifications include (1) changing the solution method to an iterative scheme and (2) eliminating significant discontinuous transitions. However, the changes in paragraphs 3.1.4 on the critical flow, 3.2.1 on the numerical convergence, 3.2.3 on the small pressure model, and 3.2.4 on the lower plenum outlet density smoothing, there is no discussion [of the effect of] the proposed changes on the EM. For example, zeroing the large off-diagonal term in the Jacobian will affect the absolute value of the solution in 3.2.1. Please discuss potential effects of the numerical fixes.

Response

(RELAX) - To the degree that this question refers to the RELAX code (Paragraph 3.1.4), please refer to the response to Question 6.

(FLEX) - The bulk of the changes to the FLEX code either do not significantly change previous results or allow the code to continue running where it would not previously run.

Responses to NRC Questions
Topical Report ANF-91-048

January 27, 1992

Question 11

> On page 40, either choice of the critical flow multiplier is based on agreement of experimental and analytical results. Please, justify the proposed change of the critical flow coefficient in terms of the requirements of the EM.

Response

There is no change in the EM choice of critical flow multiplier associated with the EXEM/BWR revisions described in ANF-91-048(P). The discussion on page 40 relates only to validation calculations made for comparison with the TLTA test. Conservatisms required in EM licensing calculations are removed from the test comparison analysis so that the desired phenomena can be assessed directly against data without distortion due to these conservatisms. Therefore, the SNP TLTA simulation was run with measured power input and a critical flow multiplier intended to give the most realistic calculation of the TLTA blowdown. Calculated results can then be compared directly against measured temperatures and event times.

For licensing analyses with the revised EXEM/BWR model, SNP will continue to perform the required spectrum of LOCA breaks, and the worst of these breaks will be analyzed to show conformance to 10 CFR 50.46 criteria.

Responses to NRC Questions
Topical Report ANF-91-048

January 27, 1992

Question 12

From the application of the revised RELAX model on the TLTA results, it is concluded that it conservatively predicts the bundle temperature[.] [A]s shown in Figure 5.8[,] consider the following comments:

- Temperature data are given only 140 sec, i.e. the end of blowdown.

Response

In the revised EXEM/BWR methodology, changes were made to the RELAX and FLEX codes. The validation calculation was focused on the effects of the RELAX and FLEX changes compared to the TLTA data.

- P(t) in Figures 5.3 and 5.9; how can substantial over-prediction of rod temperature (Figure 5.8) result in under-prediction of dome pressure?

Response

- What is the rod temperature measurement between 140 and 200 seconds?

Response

- Aren't figs. 5.8 and 5.10 inconsistent?

Response

This question is related to an apparent thermodynamic inconsistency between saturated pressure and temperature posed above. This issue has been addressed in part 2 of this question.

- Considering similar behavior in an actual reactor [it] is not clear that fig. 5.8 represents a conservative result. Please discuss why 5.8 is conservative. Include in the discussion any potential affect of pressure on the refill rate.

Response

Responses to NRC Questions
Topical Report ANF-91-048

January 27, 1992

Question 13

In Section 5.2 which discusses the FLEX simulation of TLTA, a conservative prediction is claimed, however, there is no comparison to experimental data, there is no discussion of potential effects of the revised RELAX and it does not consider TLTA in the perspective of an actual BWR. Please respond to above comments.

Response

The revised model as submitted in ANF-91-048(P) focuses on changes in results for the benchmark provided with the original EXEM/BWR model (XN-NF-80-19 (P)(A), Volume 2) and on the changes in results from sample calculations for BWRs due to the revisions. The ANF-91-048(P) submittal provides essentially the same RELAX and FLEX comparisons with experimental data as provided with the original benchmark. The object was to view the composite of the 1980 and the 1991 submittals as the support documentation for the revised EXEM/BWR model.

The TLTA test was chosen for the original benchmark as being the test most representative of large BWR LOCA phenomena available at the time. The test simulated LOCA blowdown phenomena, core heatup, and vessel reflood conditions. Section 5.1 of ANF-91-048(P) discusses the TLTA simulation using the revised RELAX code.

Responses to NRC Questions
Topical Report ANF-91-048

January 27, 1992

Question 14

> In Section 6.0, the results of the presently licensed version of EXEM are compared to the revised version. However, the staff has essentially disqualified the Dougall-Rohsenow correlation, thus comparison to these results does not prove the claimed conservatism of the proposed modified Dougall-Rohsenow correlation. This combined with the limited comparison to the TLTA 6406 makes the entire report to appear as very weakly justified. Please comment.

Response

Siemens has evaluated the adequacy of the modified Dougall-Rohsenow correlation by comparison with data available in the open literature. These data include test results from the Oak Ridge National Laboratory Thermal-Hydraulic Test Facility and results shown in ASME papers by Slaughterbeck Et. Al.[1] In general, for BWR application the modified Dougall-Rohsenow correlation conservatively under predicts the experimentally determined heat transfer coefficient results.

The results shown in Figures 6.5 and 6.6 demonstrate the effects of use of the modified Dougall-Rohsenow correlation in place of the disqualified Dougall-Rohsenow correlation in the current EM model when applied to a BWR-4 system. The effect is to increase the calculated cladding temperature both in the hot channel and average core blowdown calculations relative to the current EM model results. These results confirm the additional conservatism of the modified Dougall-Rohsenow correlation when applied to a licensing analysis.

(1) D. C. Slaughterbeck, L. J. Ybarrondo, and C. F. Obenchain, "Flow Film Boiling Heat Transfer Correlations –Parametric Study with Data Comparisons," ASME paper 73-HT-50, ASME-AIChE Heat Transfer Conference, Atlanta, Ga., August 5-8, 1973.

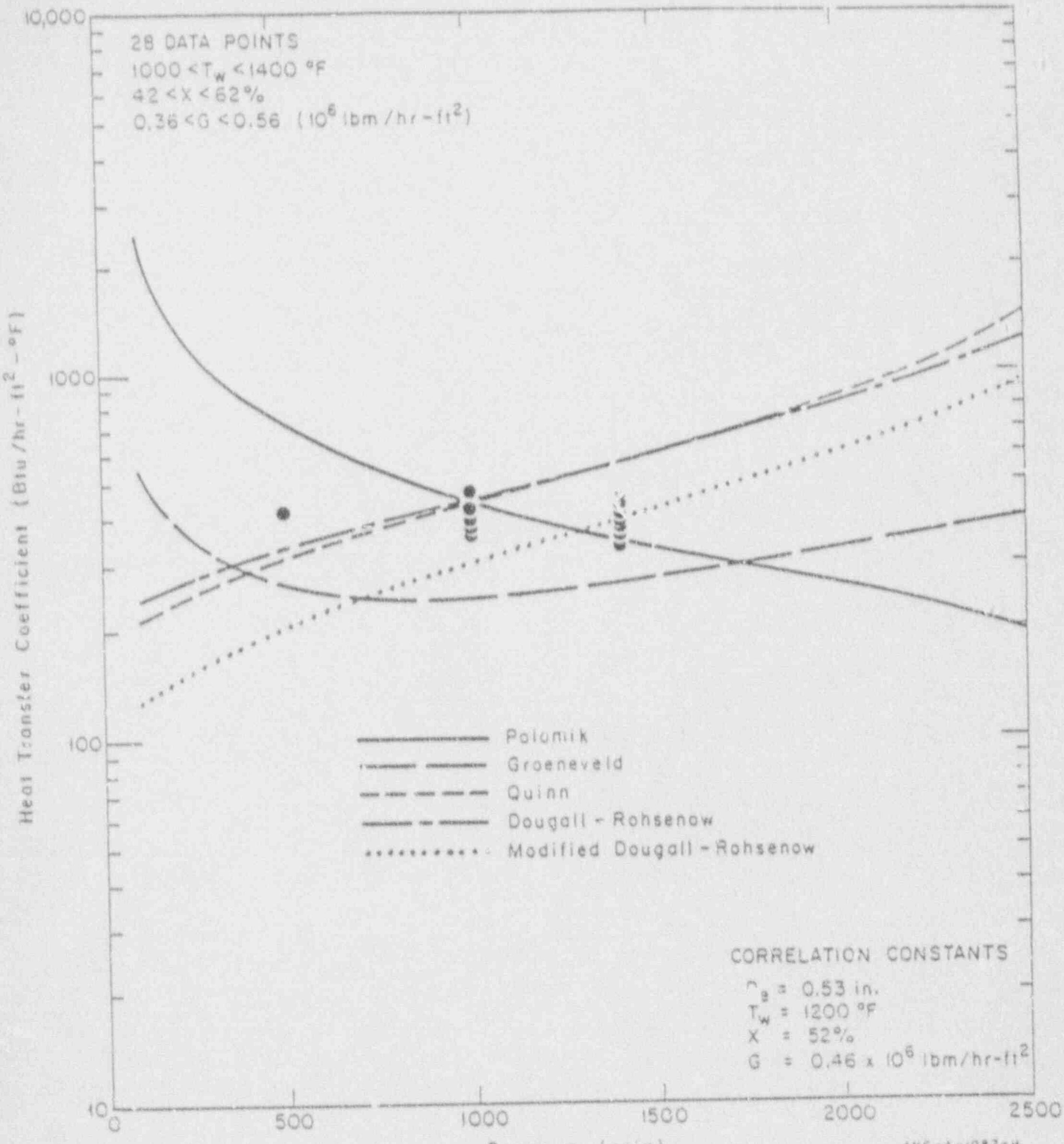

FIG. 2 HEAT TRANSFER COEFFICIENTS AS A FUNCTION OF PRESSURE

January 28, 1992

bc: H. D. Curet
L. J. Federico
D. E. Hershberger
J. G. Ingham
S. E. Jensen
D. C. Kolesar
J. N. Morgan
K. D. Richert
G. L. Ritter
H. E. Williamson
File

ANF-91-048(NP)(A)

ANF-91-048(NP)(A)
CORRESPONDENCE

Issue Date: 1/13/93

ADVANCED NUCLEAR FUELS CORPORATION

METHODOLOGY FOR BOILING WATER REACTORS

EXEM BWR EVALUATION MODEL

Distribution

USNRC/T.E Hershberger (15)