FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT ANP-10311P, REVISION 0

"COBRA-FLX: A CORE THERMAL-HYDRAULIC-ANALYSIS CODE TOPICAL REPORT"

AREVA NP, INC.

PROJECT NO. 728

1.0 INTRODUCTION

By letter dated March 31, 2010, AREVA NP, Inc. (AREVA) submitted Topical Report (TR) ANP-10311P, Revision 0, "COBRA-FLX: A Core Thermal-Hydraulic Analysis Code Topical Report" (Reference 1), to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. The purpose of this TR is to support stand-alone application[1] of the COBRA-FLX code for nuclear core thermal-hydraulic simulations. The COBRA-FLX code is intended to replace the current subchannel codes XCOBRA-IIIC (TR XN-75-21(P)(A), Revision 2) and LYNXT (TR BAW-10156-A, Revision 1), within the methodology applications where these codes are currently used. The general application of the COBRA-FLX code for safety-related analyses includes determining core flow distribution (including lateral crossflow velocities), core and fuel assembly pressure drop, local core coolant conditions, minimum departure from nucleate boiling ratio (MDNBR), and steady-state or transient core thermal-hydraulic conditions.

The NRC staff sent an initial Request for Additional Information (RAI) dated November 17, 2010 (Reference 2). AREVA responded to the RAI by the submittal of "Response to Request for Additional Information on ANP-10311P, 'COBRA-FLX: A Core Thermal-Hydraulic Analysis Code Topical Report'" (Reference 3).

2.0 REGULATORY EVALUATION

This safety evaluation (SE) reviews the COBRA-FLX thermal-hydraulic code for stand-alone application to nuclear core thermal-hydraulic analysis for steady-state and transient conditions. The transient analysis capabilities of this code are as defined in Table 1-1 of Reference 1.

The extent of review and approval requested by AREVA (as per Table 1-1 of Reference 1) is specified as the following:

a) application to pressurized water reactors (PWRs) only

b) MDNBR calculations using eight AREVA-specific critical heat flux (CHF) correlations previously approved for use with LYNXT or LYNX2 and revalidated with COBRA-FLX

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[1] The COBRA-FLX code is the thermal-hydraulic module for the core simulator ARTEMIS within the ARCADIA® code package developed for world-wide application of a converged code system within AREVA for neutronic and thermal-hydraulic core design and safety evaluation. Application of the COBRA-FLX code within the ARCADIA® package is supported by a separate submittal (TR ANP-10297P), and is not considered in this evaluation.

c) two numerical solution methods, specifically:

    i. SCHEME-Pressure (P) solution method (with two optional Successive Over-Relaxation (SOR) solvers)

    ii. Pressure-Velocity (PV) solution method

d) steady-state applications for safety-related hydraulic and thermal-hydraulic calculations

e) transient applications for safety-related analyses where the core surface heat flux is specified as a boundary condition obtained using an NRC approved code

The fuel rod model in COBRA-FLX and the rewetting model for post-CHF heat transfer will not be used for safety-related analysis, and are specifically excluded from this review. Additional limitations are specified for the empirical correlations that will be used in licensing calculations. These empirical correlations are listed in Table 1-2 and Appendix A of the TR (Reference 1) and are summarized as the following:

a) water properties (from the International Association for the Properties of Water and Steam-Industrial Formulation 97 (IAPWS-IF97))

b) friction factor correlation constants

    i. Lehman friction factor (with or without Szablewski correction)

    ii. wall viscosity correction option

c) two-phase friction multiplier – homogeneous model only

d) bulk void correlation – Chexal-Lellouche (using the full curve fit routine or tables with interpolation)

e) subcooled void correlation – Saha-Zuber

f) subcooled boiling profile fit correlation – Zuber-Staub

g) nucleate boiling forced convection heat transfer correlation – Chen

h) post-departure from nucleate boiling (DNB) forced convection heat transfer correlation – Groeneveld 5.7

i) single-phase convection heat transfer correlations

    i. Sieder-Tate for normal flow conditions

    ii. McAdams natural convection correlation for very low flow conditions

The numerous other bulk void correlations, subcooled boiling correlations, two-phase friction multiplier correlations, and heat transfer correlations in the code are specifically excluded from consideration for use in licensing calculations.

This SE is based on regulations applicable to steady-state and transient analysis methods found in Sections 50.34 and 50.46 of Title 10 of the *Code of Federal Regulations* (10 CFR) as well as Chapter 15 of the Standard Review Plan (SRP) (Reference 4). SRP 15.0.2, "Review of Transient and Accident Analysis Methods," describes areas of NRC staff review for transient and accident analysis methods such as Required Documentation, Evaluation Model, Code Assessment, and Uncertainty Analysis. The NRC staff's review of this TR submittal on the COBRA-FLX code is based on the SRP guidance in the areas of Required Documentation and Code Assessment, independent evaluation of the technical merit of the submittal, and other applicable regulations associated with the review of TRs. The COBRA-FLX code is intended for application to steady-state and transient conditions.

3.0 TECHNICAL EVALUATION

The COBRA-FLX code models fluid flow and heat transfer in the reactor core. Local flow and heat transfer behavior determines the rate of heat removal from the nuclear fuel, which directly determines the local surface temperature on the fuel rods and temperature distribution within the fuel. COBRA-FLX predicts the axial and lateral flow, pressure, and temperature (enthalpy) distributions in rod bundle arrays for flow conditions where wall shear forces predominate fluid-fluid shear forces. Typical applications of the COBRA-FLX code include predictions of core-wide flow and enthalpy distributions as well as the pressure drop for steady-state and transient conditions. The code is also used for CHF calculations and DNB ratio (DNBR) predictions.

COBRA-FLX was developed from the COBRA 3-CP code, which is a thermal-hydraulic code used in support of reactor core reload applications. COBRA 3-CP was developed by Siemens/Kraftwerk Union (KWU) from the COBRA-IIIC/MIT-2 code. These codes utilize the subchannel analysis concept developed for reactor core analysis, solving conservation equations for mass, momentum, and energy to obtain fluid enthalpy and flow distributions as well as momentum pressure drop. The flow field is assumed to be incompressible and homogeneous, with constitutive models incorporated to account for subcooled boiling and liquid/vapor slip. Surface heat flux and wall temperatures are calculated using empirical correlations for forced convection heat transfer. DNB is determined in the thermal-hydraulic solution using empirical CHF correlations. In addition, CHF correlations can be used to calculate DNBR values, as the ratio of predicted CHF versus calculated heat flux at the fuel rod surface, to assess thermal margin during normal operations.

3.1 Basic Conservation Equations

Sections 2.1 and 2.2 of the TR (Reference 1) present the transient conservation equations for mass, energy, and momentum for a single-component two-phase mixture. The governing equations for subchannel geometry are derived from an integral balance on an arbitrary fixed (Eulerian) control volume. The formulation of the basic equations and simplifying models for two-phase flow and lateral momentum exchange are consistent with well established and well verified approaches for subchannel analysis of reactor cores. The derivation of the conservation equations is essentially identical to that of earlier subchannel codes (e.g., COBRA-IIIC, XCOBRA-IIIC, COBRA-IIC/MIT, VIPRE-1, LYNX2, and LYNXT), reflecting the historical development of the subchannel analysis methodology.

3.2 Numerical Solution Methodology

Section 2.3 of the TR (Reference 1) presents detailed descriptions of the two numerical solution methods used in the COBRA-FLX code to solve the thermal-hydraulic equations derived from the mixture balance laws. This includes a detailed derivation of the finite difference formulation of the subchannel equations from the basic conservation equations presented in Section 2.2 of the TR (Reference 1), and development of the matrix formulation for the solution of each of the conservation equations (i.e., mass, momentum, and energy). The two solution methods are defined as the SCHEME solution method (also referred to as the P-solution method) and the PV-solution method. Both methods solve the same set of finite differential equations, using the same models and correlations for heat transfer, friction losses, fluid state, and two-phase flow. The two methods differ mainly in their treatment of the flow and pressure fields, and in the solution of the energy equation.

In the SCHEME solution method, developed originally for normal upflow in the reactor core, the solution begins by solving the momentum equations (lateral and axial), then the continuity and energy equations at each axial level in succession. In the PV-solution, the momentum equations are used to obtain a tentative flow solution based on the results at the previous level, then flow and pressure are adjusted to satisfy continuity using a Newton-Raphson iteration. The converged flow and pressure solution is then used to solve the energy equation.

When properly implemented, both methods yield the same results for a given set of boundary conditions. In general, the SCHEME solution is somewhat faster than the PV-solution method, and consequently is the recommended approach for most applications. However, the PV-solution method is required for conditions where axial flows can be locally very small, in a reverse direction during the simulation, or if the magnitude of the crossflow velocities is large relative to the axial flows.

These solution methods have been implemented successfully in a number of subchannel codes, and have been shown to be robust and reliable when applied within the appropriate range of their underlying assumptions. The addition of optional SOR solvers to the P-solution method does not substantially change the nature of the numerical solution. However, it does introduce additional steps in the verification of this solution method, as it must be shown to yield the same results when used with the alternative solvers.

3.3 Verification and Validation

Section 5 of the TR (Reference 1) presents results of verification and validation of COBRA-FLX for the intended range of application of the code. The document appropriately recognizes that there are two main aspects to verification and validation of the code. First, it must be verified that the solution methods are correctly implemented and yield the same results for the same boundary conditions. Second, it must be shown that the solution methods in the code yield results that are in good agreement with relevant experimental data modeling fluid behavior in rod bundle geometries.

As presented in Section 5 of the TR (Reference 1), the documentation is incomplete for the verification that the solution methods yield the same results for the same boundary conditions. However, additional information supplied in responses to RAI A.1 and RAI A.2 (Reference 3), provided detailed documentation of a range of test cases showing the equivalence of results obtained with the different solution options and solvers available in COBRA-FLX. For a summary of all of the RAI questions developed in this review, see Attachment 1 of this SE.

Similarly, the documentation of the validation of the solution methods in COBRA-FLX presented in Section 5 of the TR (Reference 1) is limited and incomplete. The TR ANP-10311P, Revision 0 (Reference 1) appropriately takes credit for prior work on validation of the general subchannel code capabilities, based on the referenced documentation for the LYNX2, LYNXT, and COBRA 3-CP codes. However, additional work is required to support the validation specific to the COBRA-FLX code. This was supplied in the responses to RAIs B.3, B.4, B.5, and B.6 (Reference 3). The detailed results presented in the RAI responses show that the COBRA-FLX code can appropriately represent the flow and enthalpy distribution in a subchannel array for the range of boundary conditions for typical applications of the code. The RAI responses also develop important user guidance for input parameters defining lateral flow resistances in the rod array as well as appropriate placement of axial resistances representing grid spacer losses within the finite node divisions of the model; in order to assure consistency between the results obtained with the two main solution options in the code.

3.4 Constitutive Models

Empirical correlations comprise the constitutive models required to achieve closure of the set of conservation equations solved by the COBRA-FLX code. Appendix A of the TR (Reference 1) lists and describes the empirical correlations available in the code for fluid state, friction losses in single- and two-phase flow, and the void/quality relationship. As noted above in Section 2.0 of this SE, this review evaluated only the specific correlations identified for use in licensing calculations. These are summarized as following:

a. water properties (from IAPWS-IF97)

b. friction factor correlation constants

    i. Lehman friction factor (with or without Szablewski correction)

    ii. wall viscosity correction option

c. two-phase friction multiplier – homogeneous model only

d. bulk void correlation – Chexal-Lellouche (using the full curve fit routine or tables with interpolation)

e. subcooled void option – Saha-Zuber

f. subcooled boiling profile fit – Zuber-Staub

g. nucleate boiling forced convection heat transfer correlation – Chen

h. post-DNB forced convection heat transfer correlation – Groeneveld 5.7

i. single-phase convection heat transfer correlations

    i. Sieder-Tate for normal flow conditions

    ii. McAdams natural convection correlation for very low flow conditions

3.4.1 Water Properties

The properties for water, which are used in the COBRA-FLX code to determine the fluid state (e.g., density, enthalpy, viscosity, and thermal conductivity) as a function of pressure and temperature, are based on the industrial standard for water and steam, IAPWS-IF97. This is a database developed and qualified by an international organization, similar to the American Society of Mechanical Engineers (ASME) Steam Tables typically used in the United States for specifying fluid properties in a subchannel code, and contain essentially the same values for water properties. The IAPWS database is as acceptable as the ASME Steam Tables for this purpose.

3.4.2 Friction Factor Models

The friction factor correlations for single- and two-phase flow are for the most part venerable and well validated models in widespread general use in subchannel codes. The exception is the Lehmann correlation for the friction factor, which corrects the standard Reynolds number correlation to account for surface roughness, and the additional correction to the Lehmann correlation to account for the “entrance effect” of the grid spacers on turbulence in the flow field. The usefulness of these corrections when applied to the discretized numerical mesh of a finite difference model is arguable, but it is inarguable that they do attempt to capture physical effects that are generally ignored in subchannel modeling.

On a purely mathematical basis, these factors result in very small corrections to the “standard” friction factor values, when applied in the range typical of reactor operations. It does no harm to include them in the code, and could, in the long run, prove helpful as increasing computational resources allow more detailed and finer scaled models of the flow field in the reactor core.

3.4.3 Void Models

The void correlation in a subchannel code of this type is extremely important, in that it must compensate for the effects of the physical modeling simplification resulting from assuming that two-phase flow can be treated as a homogeneous fluid. The void correlation is relied upon to capture the effects of subcooled boiling and phase slip, to appropriately represent the relationship between thermodynamic quality and void fraction. Licensing calculations with the COBRA-FLX code use the Chexal-Lellouche model, which is a well-known and well validated correlation based on the drift flux representation of two-phase flow. This correlation has been qualified against a large range of steady-state two-phase or two-component (air/water) test data for thermodynamic conditions in geometries typical of PWR and boiling water reactor fuel assemblies.

This model is an excellent choice for representing the two-phase flow behavior in COBRA-FLX. However, as implemented in the COBRA-FLX code, the Chexal-Lellouche model is not consistent with the model as described in the primary reference. Two components of the model, consisting of the correlation for the initiation of subcooled void formation, and the profile-fit correlation relating local quality and void fraction, are not the models presented in the primary reference for these components of the Chexal-Lellouche model. The specific correlations used in COBRA-FLX are Saha-Zuber (for the initiation of subcooled void), and Zuber-Staub (for the profile-fit correlation).

This discrepancy between the COBRA-FLX code and the published form of the Chexal-Lellouche model generated RAI C.7, requesting justification for the substitution of these two correlations for the corresponding components of the original model. The response provided this justification by comparison of COBRA-FLX code calculations to measured data for two-phase flow conditions. In these calculations, the Chexal-Lellouche void correlation was used with the Saha-Zuber and Zuber-Staub correlations. The response included comparisons to steady-state and transient data. The steady-state data was for the ATRIUM 10XM fuel design obtained in AREVA's thermal-hydraulic test loop KATHY at Karlstein, Germany in 2009. The transient data consisted of Nuclear Power Engineering Group (NUPEC) transient boiling experiments for a single channel.

Evaluation of these comparisons showed that for a wide range of conditions, the results obtained with the Chexal-Lellouche model, as implemented in COBRA-FLX, gives very good agreement with the two-phase experimental measurements. Of particular significance is the agreement shown between the COBRA-FLX predictions and measured void fraction from the steady-state tests. Over the range of conditions with void fractions near zero (subcooled boiling) to void fractions near 1.0 (saturated steam), the calculated results match the measured data within the measurement uncertainty. These results indicate that for the intended application of the COBRA-FLX code, the modified version of the Chexal-Lellouche model is acceptable.

3.4.4 Heat Transfer Models

In subchannel codes such as COBRA-FLX, forced convection heat transfer correlations are used to calculate surface temperatures, including cladding surface temperatures, for steady-state and transient conditions. COBRA-FLX uses the typical approach of specifying different correlations for the different regions of the boiling curve, consisting of single-phase liquid, nucleate boiling, post-CHF film boiling, and single-phase vapor. In applications of COBRA-FLX for licensing calculations, heat transfer coefficients in the nucleate boiling region use the Chen correlation, which is formulated to capture both subcooled boiling and fully developed bulk boiling conditions. For post-CHF conditions, licensing calculations use the Groeneveld 5.7 correlation. Both of these correlations are appropriate for the range of conditions of licensing calculations, and are widely used in subchannel codes throughout the nuclear industry.

For single-phase flow conditions, the COBRA-FLX code uses the Sieder-Tate correlation, rather than the Dittus-Boelter correlation typically used in this region. In addition, for very low flow conditions, the code uses a natural convection heat transfer correlation developed by McAdams. The TR, ANP-10311P, Revision 0 (Reference 1), does not document the applicability of these correlations to typical code applications, which resulted in the generation of RAI C.10 in Reference 2. Additional information in the response to RAI C.10 (Reference 3) on the development and range of applicability of these correlations provided appropriate justification for their use in the COBRA-FLX code for licensing calculations.

3.4.5 CHF Correlations

The COBRA-FLX code includes eight CHF correlations previously developed using the LYNXT or LYNX2 subchannel codes. The specific correlations are listed in Table 1 below.

CHF Correlation	Developed For:	Developed Using:
ACH-2	US EPR	LYNXT
BHTP	HTP fuel design	LYNXT
BWU-Z	Mark-BW17 fuel with MSMGs	LYNXT
BWU-Z	Mark-BW17 fuel without MSMGs	LYNXT
BWCMV-A	Mark BW17 fuel with or without MSMGs	LYNXT
BWCMV	AREVA 15x15, 17x17; and WE 17x17 OFA	LYNX2
BWU-N	Non-mixing vane fuel designs	LYNX2
BWC	15x15 fuel designs	LYNX2

Table 1: CHF Correlations Included in COBRA-FLX

These correlations have been approved by the NRC for reload licensing calculations, but because the CHF correlations use local subchannel conditions predicted by either the LYNXT or LYNX2 code, NRC approval is limited to applications only with the specific subchannel code used in the correlation's derivation and to other codes for which an application extension has been approved by the NRC. In order to use these correlations with the COBRA-FLX code in licensing calculations, it is necessary to demonstrate that each correlation can be expected to yield essentially the same results for the local subchannel conditions predicted with COBRA-FLX as it would with its base code.

Appendix C of TR ANP-10311P, Revision 0 (Reference 1), presents a detailed validation for each of these correlations, with the original databases used to develop the correlations with the LYNXT or LYNX2 code. The measured to predicted ratio (M/P) for each test point was calculated using each correlation in COBRA-FLX and evaluated using statistical methods used to assess the original M/P results obtained with the LYNXT or LYNX2 code. The statistical properties of the fit to the database were compared to those of the original fit with LYNXT or LYNX2, as was the resulting DNBR limit value obtained with the COBRA-FLX results.

This is the appropriate approach for qualifying a CHF correlation with a subchannel code different from the code used to derive it. The results presented in TR ANP-10311P, Revision 0 (Reference 1), show that these eight CHF correlations give essentially the same results as were obtained in the original development of the correlation, and the DNBR limits for applications with the COBRA-FLX code are equivalent to the limits developed for these correlations when used with their original subchannel code.

4.0 LIMITATIONS AND CONDITIONS

1. The fuel rod model in COBRA-FLX and the rewetting model for post-CHF heat transfer will not be used for safety-related analysis, and are specifically excluded from this review. Additional limitations are specified for the empirical correlations that will be used in licensing calculations. These empirical correlations are listed in Table 1-2 and Appendix A of the TR (Reference 1) and are summarized as the following:

a) water properties (IAPWS-IF97)

b) friction factor correlation constants

   i. Lehman friction factor (with or without Szablewski correction)

   ii. wall viscosity correction option

c) two-phase friction multiplier – homogeneous model only

d) bulk void correlation – Chexal-Lellouche (using the full curve fit routine or tables with interpolation)

e) subcooled void correlation – Saha-Zuber

f) subcooled boiling profile fit correlation – Zuber-Staub

g) nucleate boiling forced convection heat transfer correlation – Chen

h) post-departure from nucleate boiling (DNB) forced convection heat transfer correlation –Groeneveld 5.7

i) single-phase convection heat transfer correlations

   i. Sieder-Tate for normal flow conditions

   ii. McAdams natural convection correlation for very low flow conditions

2. This review examined only the specific models and correlations requested by the applicant, as summarized in Section 2.0 of this SE. These are the only models and correlations that may be used in licensing calculations with the COBRA-FLX subchannel code. The fuel rod model in COBRA-FLX and the rewetting model for post-CHF heat transfer shall not be used for safety related analysis, and are specifically excluded from this review.

5.0 CONCLUSION

The documentation provided in TR ANP-10311P, Revision 0 (Reference 1), and in responses to the RAI questions (see Attachment 1 of this SE) demonstrate that the COBRA-FLX thermal-hydraulic code is suitable for stand-alone application to nuclear core thermal-hydraulic analysis for steady-state and transient conditions. The basic equation set and numerical solution methods are correctly implemented and appropriate to the widely used subchannel methodology. This review examined only the specific models and correlations requested by the applicant, as summarized in Section 2.0 of this SE. These are the only models and correlations that may be used in licensing calculations with the COBRA-FLX subchannel code. The fuel rod model in COBRA-FLX and the rewetting model for post-CHF heat transfer shall not be used for safety related analysis, and are specifically excluded from this review.

In summary, the NRC staff finds it acceptable to use the COBRA-FLX code in place of the current subchannel codes XCOBRA-IIIC and LYNXT, within the methodology applications where these codes are currently used. The validation of the eight CHF correlations listed in

Appendix C of TR ANP-10311P, Revision 0 (Reference 1), shows that these correlations can be used with the COBRA-FLX code, with the DNBR limit values calculated for their respective databases using the COBRA-FLX code.

6.0 REFERENCES

1. AREVA NP, Inc., "ANP-10311P, Revision 0, 'COBRA-FLX: A Core Thermal-Hydraulic Analysis Code Topical Report,'" March 31, 2010, ADAMS Accession No. ML101550172.

2. Cruz, Holly D., letter to Salas, Pedro, AREVA NP Inc., "Request for Additional Information on ANP-10311P, Revision 0, 'COBRA-FLX: A Core Thermal-Hydraulic Analysis Code,'" November 17, 2010, ADAMS Accession No. ML103350388.

3. Salas, Pedro, AREVA NP Inc., letter to U.S. NRC, Document Control Desk, "Response to a Request for Additional Information (RAI) on ANP-10311P, 'COBRA-FLX: A Core Thermal-Hydraulic Analysis Code Topical Report,'" January 28, 2011, ADAMS Accession No. ML110310596.

4. U.S. NRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, December 2005.

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