September 17, 2004

Mr. James F. Mallay
Director, Regulatory Affairs
Framatome ANP
3315 Old Forest Road
P.O. Box 10935
Lynchburg, Virginia 24506-0935

SUBJECT: DRAFT SAFETY EVALUATION FOR TOPICAL REPORT BAW-10244P, "MARK-BW CHF CORRELATION APPLIED WITH XCOBRA-IIIC" (TAC NO. MC0671)

Dear Mr. Mallay:

On September 3, 2003, Framatome ANP (FANP) submitted Topical Report (TR) BAW-10244P, "MARK-BW CHF Correlation Applied with XCOBRA-IIIC" to the staff for review. Enclosed for FANP's review and comment is a copy of the staff's draft safety evaluation (SE) for the TR.

Pursuant to 10 CFR 2.390, we have determined that the enclosed draft SE does not contain proprietary information. However, we will delay placing the draft SE in the public document room for a period of ten working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects. If you believe that any information in the enclosure is proprietary, please identify such information line-by-line and define the basis pursuant to the criteria of 10 CFR 2.390. After ten working days, the draft SE will be made publicly available, and an additional ten working days are provided to you to comment on any factual errors or clarity concerns contained in the SE. The final SE will be issued after making any necessary changes and will be made publicly available. The staff's disposition of your comments on the draft SE will be discussed in the final SE.

To facilitate the staff's review of your comments, please provide a marked-up copy of the draft SE showing proposed changes and provide a summary table of the proposed changes.

If you have any questions, please contact Michelle C. Honcharik at 301-415-1774.

Sincerely,

/RA/

Robert Gramm, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 728

Enclosure: Draft Safety Evaluation

September 17, 2004

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Director, Regulatory Affairs
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DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

BAW-10244P, "MARK-BW CHF CORRELATION APPLIED WITH XCOBRA-IIIC"

FRAMTOME ANP

PROJECT NO. 728

1.0 INTRODUCTION

By letter dated September 3, 2003 (Reference 1), as supplemented by letter dated May 21, 2004 (Reference 2), Framatome ANP (FANP) submitted topical report (TR) BAW-10244P, "Mark-BW CHF [critical heat flux] Correlations Applied with XCOBRA-IIIC." The purpose of the submittal is to justify applying the BWU CHF correlations to the Mark-BW fuel design using the XCOBRA-IIIC code (Reference 3). The original approved BWU CHF correlations had been applied with the LYNXT thermal-hydraulic code (References 4 and 5), and the only new aspect is the use of the BWU CHF correlations with a different thermal-hydraulic computer code.

FANP is a joint venture of the companies Framatome and Siemens. This new company has created opportunities for the fuel designs previously developed within one former company to be analyzed with the thermal-hydraulic code previously developed by the other former company. The need for NRC approval of BAW-10244, "Mark-BW Critical Heat Flux Correlations" is to support the use of CHF correlations for a fuel product design (the Mark-BW) developed by the former Framatome company to be combined with the reload analysis methodology developed by the former Siemens company. The XCOBRA-IIIC code is integral to the reload analysis methodology developed by Siemens and is currently used for Combustion Engineering (CE) and Westinghouse-type plants.

2.0 REGULATORY EVALUATION

Section 50.36 of Title 10 of the *Code of Federal Regulations* (10 CFR) requires that safety limits be included in the plant-specific technical specifications (TS). Pursuant to 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 10, "Reactor design," the reactor core and associated coolant, control, and protective systems, are required to be designed with an appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operational, including anticipated operational occurrences. To ensure compliance with GDC 10, the NRC staff will confirm that the vendor performed the departure from nucleate boiling (DNB) analyses using NRC-approved methodologies as described in NUREG-0800, "Standard Review Plan," Section 4.4.

3.0 TECHNICAL EVALUATION

3.1 CHF Test Programs

The tests on the mixing vane spacer addressed in BAW-10244P were performed at the Columbia University Heat Transfer Research Facility (HTRF). The HTRF is a ten megawatt electric facility capable of testing full length (up to 14 foot heated length) rod arrays in matrices up to a 6 X 6 matrix. HTRF testing conditions cover the full range of operating conditions with pressure up to 2500 pounds per square inch-atmosphere (psia), mass velocities up to 3.5 million pounds per hour per square foot (Mlb/hr-ft^2), and inlet temperatures approaching saturation. The test on the non-mixing spacer addressed in BAW-10244P was performed at the Babcock & Wilcox (B&W) Alliance Research Center (ARC).

Individual CHF tests for the Mark-BW non-midspan mixing (MSM) spacer, the Mark-BW MSM grid, and the non-mixing spacer are summarized in Table 2.1 of Reference 1. The test BW 17.0 was not included in the database, because it was not included in the approved BWU CHF correlation (Reference 4).

3.2 Calculation for Local Thermal-Hydraulic Conditions and Data Analysis

The accurate prediction of CHF in operating reactors requires analysis with a subchannel thermal-hydraulic analysis code to predict the local coolant conditions at any point in the core. The data from each CHF observation within a test includes the variables for test section power, flow, inlet temperature, pressure, and CHF location (rod and axial location). Each test section was modeled for analysis with the XCOBRA-IIIC thermal-hydraulic computer code. The XCOBRA-IIIC code produces the local thermal-hydraulic conditions (mass velocity, thermodynamic quality, heat flux, etc.) axially along the test section heated length. The local condition results at the actual observed location of CHF, along with the test section global variables, are then compared to the calculated CHF.

The individual local condition results from analyses of the data with XCOBRA-IIIC are tabulated in Appendices A, B, and C of Reference 1 for the respective databases. It is important to check the individual results for bias with respect to either the dependent or any of the independent variables in the development and verification of any correlation. A justification is provided in BAW-10244P that: (1) there is no independent variable bias because an examination of the plots of the measured to predicted (M/P) CHF ratio against the independent variables of mass velocity, pressure, and quality for each grid type show that there is no major deviation from a 3.5δ (standard deviation) horizontal line; (2) there is no dependent variable bias because there are no significant deviations of the data grouping about the 45 degree line of any of the cases; and (3) histograms of the individual M/P results confirm the normal (or quasi normal for the non-mixing spacer) distribution of the database for each design type.

The NRC staff has reviewed the justifications and found them acceptable because an approved methodology was used and none of the groupings are outside the traditionally accepted 5 percent CHF uncertainty band.

3.3 BWU-Z and BWU-N CHF Correlations

The BWU-Z and BWU-N CHF correlations are parts of the approved BWU correlations (Reference 4). The form of the correlation has been implemented into XCOBRA-IIIC without any changes. The form consists of the uniform part and non-uniform flux shape of the BWU CHF correlation. The individual BWU coefficients for the MARK-BW non-MSM spacer, the Mark-BW MSM grid, and the non-mixing spacer including a performance factor are shown in Table 3.1 of Reference 1 and are also given in References 4 and 5.

The codes LYNXT and XCOBRA-IIIC have almost identical subchannel modeling capability and produce virtually the same results when identical modeling is employed. Both are derived from the original COBRA code written in the 1960's. The primary difference between LYNXT and XCOBRA-IIIC modeling is the treatment of the subchannel hydraulic resistance. LYNXT has the capability for discrete form loss coefficients for each different subchannel type in the bundle being analyzed. XCOBRA-IIIC utilizes a single average form loss coefficient for all of the subchannels in the bundle. The impact of using non-discrete form losses in the correlation verification using XCOBRA-IIIC results in a slightly higher CHF design limit as compared to LYNXT based results.

The NRC staff has reviewed the quality control of the code evaluation for satisfying the limitations imposed on the application of the approved BWU CHF. In response to the staff's request for additional information (RAI) (Reference 2), FANP states that: (1) both LYNXT and XCOBRA-IIIC identify any of the local variables that violate the ranges of the specific correlation being used; (2) the range violation reports are contained both in the body of the code output and in a separate error file for both codes; and (3) all safety-related calculations, and quality assurance of such calculations, are governed by established FANP quality assurance procedures. The NRC staff has found the justification acceptable.

3.4 Departure from Nucleate Boiling Ratio (DNBR) Design Limits and Correlation Applicability

The use of CHF equations in pressurized water reactor analyses is facilitated by the definition of the DNBR, which is defined as a ratio of calculated CHF at a given location divided by actual heat flux at that location. The DNBR is a measure of the local thermal margin to film boiling. A DNBR value of 1.0 implies transition to film boiling at that location. The higher the DNBR (above 1.0), the greater the margin to film boiling. In design analyses, DNBR values are calculated throughout the core for a given core condition. Calculation of the minimum core DNBR and comparison of this minimum with a design limit ($DNBR_L$) provides protection against departure from nucleate boiling in the core. The $DNBR_L$ is the lowest DNBR that can be calculated for any given core condition on the limiting fuel rods in the reactor while still maintaining a 95 percent confidence that 95 percent of these limiting fuel rods are not in film boiling. This approach of DNBR 95/95 limits is used to develop the BWU design limit for each of the boiling water reactor correlations.

Reference 2 indicates that the difference in modeling of the form loss coefficients results in an increase of about two percent on the design limit when using the XCOBRA-IIIC code as compared to using the LYNXT code as stated above in Section 3.3 of this evaluation.

Therefore, during plant-specific applications of the BWU correlations using the XCOBRA-IIIC code, the predicted minimum DNBRs will be approximately two percent higher than the minimum DNBR predictions using the BWU CHF correlations with the LYNXT.

The NRC staff has reviewed the ranges of applicability for the BWU correlations against the BWU database in BAW-10244P. The acceptable ranges are provided in Tables 1 through 3 attached to this safety evaluation (SE) for the Mark-BW non-MSM spacer, the Mark-BW MSM grid, and the non-mixing spacer, respectively. The ranges of applicability differ slightly from those listed in Reference 4, because a different analysis code is used (LYNXT used in Reference 4 and XCOBRA-IIIC used here).

4.0 CONCLUSION

The NRC staff has reviewed BAW-10244P and the responses to the staff's RAI to determine the acceptability of the Mark-BW CHF correlations applied with XCOBRA-IIIC and has concluded the following:

1. The BWU CHF correlation applies only to the following fuel data bases:

   (a) BWU-Z, Mark-BW non-MSM database;

   (b) BWU-Z, Mark-BW MSM database; and

   (c) BWU-N, non-mixing vane database; for the ranges of applicability summarized in Tables 1 through 3 attached to this SE, respectively.

2. The correlation should only be used within the limits specified by Tables 1 through 3 attached to this SE, for the above three fuel types.

3. If conditions fall outside of the range of applicability of BWU correlation, then the CHF shall be assumed to occur as stated in Reference 4.

4. The review does not apply to the application of the correlations in a thermal-hydraulic safety analysis code other than XCOBRA-IIIC. Should the correlation be incorporated into a different safety analysis code or model, then an additional review would be needed addressing the specifics of the code application to the prediction of CHF.

5.0 ADMINISTRATIVE ERRORS

Administrative errors in two areas were noted: (1) on page 4-2, Reference 11 should be Reference 13, and (2) on page 4-8, Design Limit DNBR above 1500 psia 1.22 should be 1.23. During a June 30, 2004, conference call, FANP stated that they will correct both errors when the "A" version of BAW-10244P is published.

6.0 REFERENCES

1. Letter from James F. Mallay, Framatome ANP to USNRC, Request for Review of BAW-10244P, "Mark-BW CHF Correlations Applied with XCOBRA-IIIC," dated September 3, 2003. (ADAMS Accession Nos. ML032530106 letter, ML032530061 for non-proprietary version, and ML032530067 for proprietary version)

2. Letter from James F. Mallay, Framatome ANP to USNRC, Response to Request for Additional Information - BAW-10244(P) Revision 0, "Mark-BW CHF Correlations Applied with XCOBRA-IIIC," dated May 21, 2004. (ADAMS Accession No. ML041470203)

3. XN-NF-75-21(P)(A), Revision 2, "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Operation," January 1986. (ADAMS Accession No. 8605140222)

4. BAW-10199P-A, BWU Critical Heat Flux Correlations, Framatome Cogema Fuels, dated August 19, 1996. (ADAMS Accession Nos. 9609040275 letter, 9609040388 non-proprietary version, 9609040390 proprietary version)

5. Letter from T. Coleman, Framatome Cogema Fuels to USNRC BAW-10199P, Addendum 2, "Application of the BWU-Z CHF Correlation to MarkBW17 Fuel Design with Mid-Span Mixing Grids," dated November 22, 2000. (ADAMS Accession Nos. ML003774307 for letter and ML003774341 for proprietary version)

Attachments:
1. Table 1: BWU-Z Ranges of Applicability Mark-BW Non-MSM Database
2. Table 2: BWU-Z Ranges of Applicability Mark-BW MSM Database
3. Table 3: BWU-N Ranges of Applicability Non-Mixing Vane Database

Principal Contributor: T. Huang

Date: September 17, 2004

Table 1: BWU-Z Ranges of Applicability Mark-BW Non-MSM Database

Pressure (psia)	400 - 2465
Mass Velocity (Mlb/hr-ft^2)	0.351 - 3.577
Thermodynamic Quality at Critical Heat Flux	less than 0.731
Thermal-Hydraulic Computer Code	XCOBRA-IIIC
Spacer	Mark-BW Non-MSM
Design Limit DNBR	
Above 1000 psia	1.22
700 - 1000 psia	1.23
Below 700 psia	1.62
BWU Coefficients	BWU-Z as Reported in Table 3.1 of Reference 1

Table 2: BWU-Z Ranges of Applicability Mark-BW MSM Database

Pressure (psia)	400 - 2465
Mass Velocity (Mlb/hr-ft^2)	0.351 - 3.577
Thermodynamic Quality at CHF	less than 0.731
Thermal-Hydraulic Computer Code	XCOBRA-IIIC
Spacer	Mark-BW MSM
Design Limit DNBR	
Above 1000 psia	1.22
700 - 1000 psia	1.23
Below 700 psia	1.62
BWU Coefficients	BWU-Z as Reported in Table 3.1 of Reference 1

Table 3: BWU-N Ranges of Applicability Non-Mixing Vane Database

Pressure (psia)	788 - 2616
Mass Velocity (Mlb/hr-ft^2)	0.272 - 3.775
Thermodynamic Quality at CHF	less than 0.690
Thermal-Hydraulic Computer Code	XCOBRA-IIIC
Spacer	Non-Mixing Vane
Design Limit DNBR	
Above 1500 psia	1.23
1200 1500 psia	1.31
Below 1200 psia	1.41
BWU Coefficients	BWU-N as Reported in Table 3.1 of Reference 1