April 10,2003

Mr. James Mallay
Director, Regulatory Affairs
Framatome ANP
3315 Old Forest Road
Lynchburg, VA 24501

SUBJECT: EVALUATION OF FRAMATOME ANP PRELIMINARY SAFETY CONCERN (PSC) 2-00 RELATING TO CORE FLOOD LINE BREAK AND OPERATOR ACTION TIME (TAC NO. MA9973)

Dear Mr. Mallay:

On September 26, 2000, Framatome ANP informed the NRC staff of a preliminary safety concern wherein use of the reactor coolant pump two-phase degradation curve contained in the approved evaluation model could underpredict peak fuel cladding temperature following postulated core flood line breaks and some cold leg breaks. In some core flood line break cases, preliminary recalculations with a different degradation curve predicted that the 10 CFR 50.46 2200°F acceptance criterion was exceeded when the reactor coolant pumps were tripped two minutes after the loss of reactor coolant system subcooling margin. Calculations with a pump trip at one minute predicted the acceptance criterion would be met. Additional information was provided in letters dated April 2, 2001, and February 5, 2002.

The NRC staff has evaluated anticipated operator action time in response to the identified breaks. We find that a combination of suitable instrumentation, emergency procedures, and operator training can provide acceptable justification for a one minute operator response time to trip the reactor coolant pumps following loss of subcooling margin in the Babcock and Wilcox-designed nuclear steam supply systems. The example submittals provided by the licensees for Three Mile Island Unit 1 and Crystal River Unit 3, in combination with our evaluation as documented in the enclosed safety evaluation, are suitable justification for a one-minute response time if referenced and confirmed to be accurate in licensee-docketed requests.

The April 2, 2001, letter summarized the results of numerous evaluation model calculations to predict Babcock and Wilcox-designed reactor coolant system response to breaks in the core flood lines or cold legs. The evaluations considered three reactor coolant pump two-phase degradation curves identified as the M1 curve, the general use 'RELAP5-Default' head degradation curve, and a minimum degradation curve known as the M3-Modified curve. The M3 curve generally predicted significantly higher peak clad temperatures when reactor coolant pumps continued to run following a core flood line break. The Babcock and Wilcox Owners Group (B&WOG) concluded the M3 curve was an adequate bound and should be used for these cases. The staff agrees that of the three curves considered, the M3 curve provides the most conservative peak clad temperature behavior for these cases. However, the B&WOG provided no information to support the contention that the M3 curve was actually conservative.

Consequently, the staff concluded that the B&WOG did not establish that the M3 curve is conservative and the staff's acceptance of the analysis results is contingent upon each licensee establishing that the M3 curve is conservative for its plant. This should be accomplished through plant specific submittals.

In the event that any comments or questions arise, please contact Drew Holland at (301) 415-1436.

Sincerely,

/RA/

Herbert N. Berkow, Director
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 728

Enclosure: Safety Evaluation

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Enclosure: Safety Evaluation

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FRAMATOME ANP PRELIMINARY SAFETY CONCERN (PSC) 2-00

RELATING TO CORE FLOOD LINE BREAK AND OPERATOR ACTION TIME

PROJECT NO. 728

1.0 INTRODUCTION

Section 50.46(a)(1)(i) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires that emergency core cooling system (ECCS) performance be calculated for a sufficient number of loss-of-coolant accident (LOCA) sizes, locations, and other properties to assure that the most severe LOCAs are evaluated. Appendix K to 10 CFR Part 50 adds further requirements for licensees who do not use realistic evaluation models (EMs) to assess LOCAs.

Licensees typically use staff-approved EMs that contain conservative assumptions such as degraded reactor coolant pump (RCP) performance curves and the worst credible single failure in addition to an assumed loss of offsite power (LOOP). These conservative assumptions have been reported to be non-conservative for some conditions that may occur in Babcock and Wilcox (B&W)-designed reactor coolant systems (RCSs). The principle impact addressed here involves continued provision of offsite power following initiation of a LOCA in combination with an assumption of less degradation in the RCP performance curve. The Babcock and Wilcox Owners Group (B&WOG) predicted this combination could lead to higher peak clad temperatures (PCTs) than reported with the staff-approved EM and, in some cases, the 10 CFR 50.46 2200°F acceptance criterion was exceeded when the RCPs were tripped two minutes after loss of RCS subcooling margin (SCM), consistent with previously approved operator action times. Calculations with a pump trip at one minute predicted the acceptance criterion would be met.

The B&WOG summarized the results of numerous EM calculations to predict B&W-designed RCS response to breaks in the core flood lines (CFLs) or cold legs. The evaluations considered three RCP two phase degradation curves identified as the M1 curve, the general use 'RELAP5-Default' head degradation curve, and a minimum degradation curve known as the M3-Modified curve. The M3 curve generally predicted significantly higher PCTs when RCPs continued to run following a CFL break. The B&WOG concluded the M3 curve was an adequate bound and should be used for these cases.

2.0 REGULATORY BASIS

2.1 History

On September 12, 2000, via telephone and on September 26, 2000 via Reference 1, the B&WOG informed the NRC of a preliminary safety concern (PSC) related to a CFL break in B&W-designed RCSs. The B&WOG designated this concern as PSC 2-00. The concern was that temperature predictions for certain LOCAs, if analyzed with different assumptions than used in the EM, could exceed design basis predictions and, in some cases, could exceed 10 CFR 50.46 acceptance criteria limits. Section 50.46 (b)(1) specifies a maximum fuel element cladding temperature of 2200°F.

The assumptions and the nature of their impact were identified as follows:

1. Historic design basis calculations for larger breaks include an assumption of LOOP at the time of reactor trip. This causes loss of the RCPs. If the RCPs continue to run, more RCS water is lost for some postulated pipe breaks. This can cause core uncovery and a significant temperature rise when RCPs are eventually tripped.[1]

2. The approved EM RCP pump characteristic curve overpredicts head degradation during the early portion of a CFL break and some cold leg pump discharge (CLPD) breaks. Since less pump degradation results in additional core uncovery and higher PCTs for these cases, the B&WOG concluded that the approved EM model was non-conservative for these cases. Use of a pump characteristic curve that excluded the traditional conservative pump degradation allowance caused an increase in predicted PCT.

The NRC staff immediately assessed PSC 2-00 and determined that it was of low safety concern because:

1. The concern did not apply in the most likely case where full ECCS capability exists.

2. The concern that 10 CFR 50.46(b) requirements are not met appeared to be limited to a break in the CFL. This potential break represents a small contributor to the approximately 10^{-3}/reactor-year likelihood of a LOCA in the size range of concern.

3. There is no concern unless SCM has been lost. Licensees operating B&W-designed RCSs use emergency operating procedures (EOPs) that instruct operators to immediately trip RCPs upon loss of SCM. This instruction is committed to memory during training. Based upon experience, the staff judged that realistic operator response would be sufficiently rapid to preclude excessive PCTs even if predicted by conservative EMs. (The B&WOG reported that RCP trip at or before one minute precluded excessive PCTs.)

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[1]Operators are instructed to trip RCPs immediately upon loss of SCM in the RCS. For design basis calculations, the staff has historically required a two-minute delay before credit can be taken for this operator action.

Consequently, the staff judged that PSC 2-00 was not a significant safety concern but that the 10 CFR 50.46 and Appendix K issues remained to be resolved.[2]

2.2 Proposed PSC 2-00 Resolution

A final report addressing PSC 2-00 was provided on April 2, 2001 (Reference 3). This stated that "a two-phase pump degradation sensitivity study was performed to define the appropriate, yet conservative, two-phase RCP head degradation curve model that should be used in the PSC 2-00 analyses...." The RCP two-phase degradation studies followed the classical EM approach of analyzing two-phase RCP performance curves that were derived from the Semiscale pump tests. The study involved running a general use curve, minimum bound curve, and maximum bound curve to determine which pump performance curve produces the most limiting core cooling consequences. Use of the upper bound, or maximum two-phase head degradation model known as the M1 curve, and the general use 'RELAP5-Default' head degradation curve produced similar results with non-limiting overall PCTs. Use of the lower bound, or minimum two-phase head degradation curve known as the M3-Modified curve, minimized the residual core liquid inventory and produced significantly worse results. Therefore, all revised PSC 2-00 plant cases with delayed RCP trip used this limiting head degradation curve.

Reported CFL break analysis results for plant-specific RCPs assuming the M3 two-phase degradation curve are summarized in Table 1.

<table>
<tr><th colspan="6">Table 1. Reported PCTs With M3 Two-Phase Degradation Curve</th></tr>
<tr><th rowspan="2">Item</th><th colspan="5">PCT, °F*</th></tr>
<tr><th>Crystal River (CR) @ 2568MWt</th><th>Three Mile Island (TMI) @ 2568MWt</th><th>Oconee @ 2568MWt</th><th>Arkansas Nuclear One @ 2772MWt</th><th>Davis-Besse @ 2966MWt</th></tr>
<tr><td>2 minute RCP trip</td><td>>2200</td><td>>2200</td><td>≤1346: 2nd HPI @ 10 min
>2200: 1 HPI, ¾ Power
≤946: 1 HPI, ½ Power</td><td>1051</td><td>≤962</td></tr>
<tr><td>1 minute RCP trip</td><td>718</td><td>717</td><td>715: 1 HPI, ¾ Power</td><td>-</td><td>-</td></tr>
<tr><td colspan="6">*Power levels used in the analyses were 1.02 times the listed values.</td></tr>
</table>

[2]The NRC reported in Three Mile Island Inspection Report 05000289/2000-006 (Reference 2) that it reviewed the licensee's operability evaluation applicable to this concern and that "the licensee concluded, based on past observations of operator performance during unannounced simulator training scenarios, there was reasonable assurance operators would carry-out the proceduralized actions and immediately trip the RCPs in less than one minute upon loss of subcooling margin and, therefore, determined the ECCSs continued to remain operable." The inspector did not identify any concerns with the licensee's operability evaluation.

These values are consistent with Arkansas and Davis-Besse meeting the 10 CFR 50.46 acceptance criterion of 2200°F with a two-minute RCP trip. A two-minute RCP trip time results in a PCT prediction in excess of the acceptance criterion for CR and TMI, but a one minute reactor trip time results in an acceptable prediction. Oconee is predicted to meet the criterion with a two-minute RCP trip if a second high pressure injection (HPI) pump is credited to be initiated at 10 minutes or if the plant is operating at half power if the second HPI pump is not available. Based on these results, Reference 3 concluded that there was no safety concern if, upon loss of SCM, RCP trip was accomplished as follows:

Table 2. RCP Trip Times That Result In PCT < 2200 °F

Plant	RCP trip time that results in predicted PCT < 2200 °F
Crystal River	One minute
Three Mile Island	One minute
Oconee	Two minutes if at full power with two HPI pumps available immediately with the third pump available at 10 minutes, or if at half power with two HPI pumps available
Arkansas	Two minutes
Davis-Besse	Two minutes

3.0 <u>EVALUATION</u>

3.1 Evaluation of Acceptability of One Minute Reactor Trip

Part of the PCS 2-00 reporting addressed acceptability of assuming operators will trip RCPs within one minute of occurrence of a loss of SCM. In response to a staff request to provide evidence that the operators would be capable of tripping the RCPs within one minute, the B&WOG provided input from TMI and CR (Reference 4). The staff's questions, a summary of the responses, and the staff's evaluation are as follows:

(1) **Question**. Provide applicable EOPs or describe any significant differences between the EOPs and the generic emergency procedures guidelines contained in Volumes 1 and 2 of Revision 09 of the Technical Basis Document (Reference 5). Describe operator training applicable to ensuring RCP trip within one minute of loss of SCM. Assuming this is a simulator "critical task," include RCP trip time history from operator requalification testing.

Summary of response. Both licensees stated that tripping the RCPs is the initial immediate step of the EOP on loss of SCM with the reactor shut down. They further indicated that this immediate step is committed to memory in training and does not require the operator to read the procedure in order to take the action. This action is a "critical task" practiced and tested many times during training and evaluation. Although delay time is not routinely recorded, the task is typically accomplished within 10 to 30 seconds, even with a simulated failure of the SCM alarm function. Once licensed, there have been no recorded failures of operators performing this task during requalification examinations.

Staff evaluation. The staff concludes that the described procedures and training are satisfactory to ensure that operators are capable of successfully tripping the RCPs within one minute of loss of SCM.

(2) **Question**. Identify the information needed by the operators and describe how the information is provided visually and audibly. Address the safety "pedigree" of the applicable displays and describe alternates, including operator response, if the primary information sources are lost. Include photographs of the key visual information sources.

Summary of response. Both licensees stated that the indication to trip the RCPs is loss of SCM. The TMI indication of subcooling is contained in dual digital indicators visible from the main console, the Shift Technical Advisor station, and the Shift Manager's desk. There are two independent safety-grade instrument channels for SCM. In addition to the digital displays, these instrument channels feed an overhead annunciator indicating SCM $<25°F$.

The primary CR SCM indications are located on dual Safety Parameter Display System (SPDS) monitors on the main control board directly above the reactor control panel. Normally, SCM is displayed at the top of each SPDS monitor. If incore temperature exceeds the SCM limit, the monitor sounds a unique audible alarm, the SCM display is enlarged to cover the entire display page, it changes background color to yellow or red depending on the limit exceeded, and starts a visible timer to count in seconds. The two redundant channels of SPDS are powered by independent emergency power sources and meet the requirement for Regulatory Guide 1.97 Type A, Category 1 instrumentation. In addition, two SCM slave displays mounted on the main control board provide duplicate indication of the SPDS SCM and brighten and flash when the SCM limit is exceeded.

Staff evaluation. The staff concludes that the described SCM information is sufficient, is displayed satisfactorily, and the displays are sufficiently redundant and safety-grade to ensure that the information required by the operators to trip the RCPs on loss of SCM is available.

(3) **Question**. Identify RCP trip controls and describe how they operate. Include a description of the relative location of the controls and the applicable displays.

Summary of response. Both licensees indicated that the controls are standard General Electric Type SBM "pistol grip" control switches located on the main console within easy view of the SCM displays. The operator action is to turn the handle to the "stop" position for each RCP.

Staff evaluation. The staff concludes that the described operator action is simple and straightforward enough to ensure that the operators are capable of tripping the RCPs within one minute of loss of SCM.

3.2 Evaluation Model Changes Applicable to PSC 2-00

Reference 3 stated that the analyses used to characterize CFL break behavior were performed in compliance with the approved RELAP5/MOD2 EM with two exceptions:

(1) Use of the RELAP5/MOD2 void-dependent cross-flow model that was under review, and

(2) Use of a more conservative M3-modified two-phase head degradation curve.

In Reference 6, the staff reported that the void-dependent cross-flow model was acceptable, but qualified its approval with respect to future changes in the small break LOCA methodology that could significantly affect either the form of the cross-flow model or the cross-flow coefficient values. Application of the RELAP5/MOD2 EM with the incorporated crossflow model to the CFL break is unaffected by this qualification. Therefore, Exception (1) above, is no longer an exception and its use as part of the EM for the CFL break characterizations is acceptable.

With respect to the two-phase degradation curve, the staff agrees that the M3 curve is a better representation of PCT behavior for CFL break licensing basis analyses than the M1 or the RELAP5-default curves. However, the B&WOG has not established that the M3 curve is conservative for these analyses. Therefore, the staff's acceptance of the analysis results is contingent upon each licensee establishing that the M3 curve is conservative for each licensee's plant.[3] The staff will accept the analysis results if the M3 curve is acceptably shown to be conservative.

4.0 <u>CONCLUSIONS</u>

The staff agrees that of the three curves considered, the M3 curve provides the most conservative PCT behavior for these cases. However, the B&WOG provided no information to support the contention that the M3 curve was actually conservative. Consequently, the staff concluded that the B&WOG did not establish that the M3 curve is conservative and the staff's acceptance of the analysis results is contingent upon each licensee establishing that the M3 curve is conservative for its plant.

The staff finds that a combination of suitable instrumentation, EOPs, and operator training can provide acceptable justification for a one minute operator response time to trip the RCPs following loss of SCM in the B&W-designed RCSs. The example submittals provided by the licensees for TMI Unit 1 and for CR Unit 3, in combination with the staff's evaluation as documented above, are suitable justification for a one-minute response time if referenced and confirmed to be accurate in licensee-docketed requests.

[3]Licensee reference to a staff-approved topical report that establishes suitable conservatism for its RCPs for the conditions of concern is acceptable.

The staff further finds that the analysis results summarized herein are acceptable provided the M3 two-phase RCP degradation curve is acceptably shown to be conservative for each licensee's plant.

5.0 REFERENCES

1. Kelly, Joseph, "Report of Preliminary Safety Concern Related to Core Flood Line Break with 2-Minute Operator Action Time," Letter to NRC from Framatome Technologies, FTI-00-2433, September 26, 2000.

2. Rogge, John F., "NRC's Integrated Three Mile Island Report 05000289/2000-006," NRC Region I Letter to Mark E. Warner, Vice President, AmerGen Energy Company, October 25, 2000.

3. Firth, David J., "Transmittal of Final Report on the Evaluation of PSC 2-00 Relating to Core Flood Line Break with 2-Minute Operator Action Time," Letter to NRC from Framatome ANP, FANP-01-988, ADAMS ML010950222, April 2, 2001.

4. Mallay, James F., "Responses to an Informal Request for Supporting Information," Letter to NRC from Framatome ANP, NRC:02:011, February 5, 2002.

5. "Emergency Operating Procedures Technical Bases Document," Framatome Technologies, 74-1152414-09, March 31, 2000.

6. Barnett, Leslie W., "Safety Evaluation of Framatome Technologies Topical Report BAW-10164P, Revision 4, 'RELAP5/MOD2-B&W, An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analyses,' (TAC Nos. MA8465 and MA8468)," Letter to James F. Mallay, Framatome ANP, from Acting Director, Project Directorate IV, Division of Licensing Project Management, NRR, NRC, April 9, 2002.

Principal Contributor: W. Lyon, NRR

Date: April 10, 2003