September 20, 2005

Mr. Ronnie L. Gardner
Manager, Site Operations
and Regulatory Affairs
Framatome ANP
3315 Old Forest Road
Lynchburg, VA 24501

SUBJECT: FINAL SAFETY EVALUATION FOR FRAMATOME ANP TOPICAL REPORT ANF-1358(P), REVISION 3, "THE LOSS OF FEEDWATER HEATING TRANSIENT IN BOILING WATER REACTORS" (TAC NO. MC4260)

Dear Mr. Gardner:

By letter dated August 19, 2004, Framatome ANP (FANP) submitted Topical Report (TR) ANF-1358(P), Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," to the U.S. Nuclear Regulatory Commission (NRC) staff. By letter dated August 22, 2005, an NRC draft safety evaluation (SE) regarding our approval of ANF-1358, Revision 3, was provided for your review and comments. FANP had no comments on the draft SE.

The NRC staff has found that ANF-1358, Revision 3, is acceptable for referencing in licensing applications for General Electric-designed boiling-water reactors to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that FANP publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include a "-A" (designating accepted) following the TR identification symbol.

Enclosure 1 transmitted herewith contains sensitive unclassified information.
When separated from Enclosure 1, this document is decontrolled.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, FANP and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

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

Project No. 728

Enclosures: 1. Final SE (Proprietary)
2. Final SE (Non-proprietary)

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Enclosures: 1. Draft SE (Proprietary)
2. Draft SE (Non-proprietary)

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PROPRIETARY SE: ML052650348 PKG.: ML
ACCESSION NO.: LTR/NON-PROP.SE: ML052650334 NRR-043 ***No substantive changes**

OFFICE	PDIV-1/PM	PDIV-1/LA	Tech Branch*	PDIV-1/SC	PDIV-2/SC(A)	PDIV/D
NAME	MHoncharik	LFeizollahi for DJohnson	FAkstulewicz	DTerao	DCollins	HBerkow
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT ANF-1358(P), REVISION 3

"THE LOSS OF FEEDWATER HEATING TRANSIENT IN BOILING WATER REACTORS"

FRAMATOME ANP

PROJECT NO. 728

1.0 INTRODUCTION AND BACKGROUND

In a letter dated August 19, 2004, Framatome ANP (FANP), submitted topical report (TR) ANF-1358(P), Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," for the NRC staff to review (References 1 and 2). At the NRC staff's request, FANP submitted additional information on April 15 and 27, and June 3, 2005 (References 3, 4, and 5). The TR describes revisions made to a previously approved TR ANF-1358(P)(A), Revision 1, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," September 1992 (Reference 6). The methodology described in Reference 6 defined the Minimum Critical Power Ratio (MCPR) following a Loss of Feedwater Heating (LFWH) transient event as a function of the [ ]

The proposed revision is based on an expanded database and resulted in changes to the coefficients of the previously approved correlation. The purpose of expanding the database is to extend the range of applicability of the methodology so that it can be applied to current core designs. The proposed revision also extends the methodology to the determination of the Linear Heat Generation Rate (LHGR).

The LFWH transient event is an infrequent anticipated operational occurrence (AOO), which results in an increase in the core inlet subcooling due to the loss of one or more feedwater heaters, producing a higher core power level. The increase in core thermal power causes an increase in the LHGR and a reduction in the core MCPR, potentially resulting in this event being the limiting event when establishing the reload MCPR Operating Limit (MCPROL).

The generic methodology is a parametric description of the fuel/system response. The parametric description was developed using the results of over one thousand applications of the currently approved core simulation methodology in TR XN-NF-80-19(P)(A), "Advanced Nuclear Fuels Methodology for Boiling Water Reactors: Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology," and TR EMF-2158(P)(A), Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," (References 7 and 8). Postulated LFWH events initiated from actual boiling-water reactor (BWR) operating state points and fuel loadings were evaluated to derive the correlation. Applying the correlation yields a bounding MCPROL for the LFWH event.

The LFWH transient can occur in two ways:

1) a steam extraction line to a feedwater heater is closed, or

2) the feedwater is bypassed around a feedwater heater.

The first case produces a gradual drop in the temperature of the feedwater. In the second case, the feedwater bypasses the heater and no heating occurs. Both cases cause a decrease in the temperature of the feedwater entering the reactor vessel. The decrease in feedwater temperature results in an increase in the core inlet subcooling, which collapses voids and thus increases the core average power and shifts the axial power distribution towards the bottom of the core. Voids begin to build up at the bottom again because of this axial shift, acting as negative feedback to the void collapse process. This feedback moderates the core power increase. This feedback also tends to flatten the core radial power distribution.

The LFWH event is analyzed with either the FANP 3-D core simulator model MICROBURN-B (Reference 7) or MICROBURN-B2 (Reference 8). These computer codes were reviewed and approved by the NRC staff. The LFWH event is a slow (>100 seconds) transient and can be modeled by analyzing equilibrium conditions at the initial and final state points.

The methodology employed involves evaluating the LFWH event at a large number of reactor operating state points (power, flow, exposure, and control rod pattern) obtained from several operating BWRs, over many fuel load cycles. The plant types (BWR/3, 4, 5, and 6) are diverse in the respect that they have different power densities, core designs, core average void fraction, fuel types, cycle lengths, and feedwater temperatures.

The assumptions used in the analysis include the following:

1) The reactor is in steady-state equilibrium before and after the event.

2) The xenon concentration does not change during the event.

3) The plant is conservatively assumed to be operating in the manual flow control mode, and the total core flow remains constant.

4) A reactor scram is not assumed.

FANP confirmed the applicability of this approach by transient analyses and plant startup tests and, as mentioned earlier, it was previously approved by the NRC staff (Reference 6). Actual and projected state points from eight operating BWR plants were used as initial conditions. The proposed revision is based on an expanded database which includes 1686 simulated LFWH events, representing 1069 operating state points from 26 operating/projected cycles, including the BWR/6 Maximum Extended Operating Domain. The database includes various mixtures of Global Nuclear Fuels (GNF) and FANP fuel assemblies, and the most recent fuel types (geometry and enrichments) and combinations of fuel types, various fuel loading schemes, and control rod sequences associated with modern fuel and core designs.

FANP incorporated the LHGR calculations in ANF-1358(P), Revision 3. FANP reported that the LHGR analyses demonstrated that during the event, the maximum LHGR for FANP and GNF fuel was

within the acceptable range of the AOO LHGR limit. These LHGR bounding values can be compared to cycle-specific AOO limits.

2.0 REGULATORY EVALUATION

This methodology is applicable to BWR/3, 4, 5, and 6 plant types for present and future operating cycles, provided that the limitations and conditions listed in Section 4.0 of this safety evaluation (SE) are met. The NRC staff has performed its review consistent with the procedures outlined in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 15.0.2 (III), "Review of Transient and Accident Analysis Methods." The proposed revision meets the requirements delineated in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criterion (GDC) 10, "Reactor design."

3.0 TECHNICAL EVALUATION

FANP used NRC-approved codes and methodologies to develop the bounding Critical Power Ratio (CPR) correlation. The operating state points evaluated by FANP represented a large number of reactor operating state points (power, flow, exposure, and control rod pattern) obtained from several operating BWRs, over many fuel load cycles. The plant types (BWR/3, 4, 5, and 6) are diverse in the respect that they have different power densities, core designs, core average void fraction, fuel types, cycle lengths and feedwater temperatures. In Reference 3, in response to the NRC staff's request for additional information (RAI), FANP confirmed that the data used in developing the correlation also included the state points that represent the minimum core flow at rated power conditions. These data included the actual state points of the high-power/high-flow and the high-power/low-flow which includes actual control rod patterns for a variety of BWR plants and different BWR classes. In particular, a LFWH event was simulated whenever a significant change in the control rod pattern was made. The incremental exposure between events is approximately 500 MWd/MTU or less. The LFWH analysis is an evaluation of how the [ ]. The results, as presented, show that the bounding correlation has no dependency on the core thermal power and core flow.

From the results presented in ANF-1358(P), Revision 3, the NRC staff assessed the effects of different types of fuel designs, and their effects in a mixed core. The fuel types included in the database were [ ]. The results suggest that, with the fuel types used in the database, there was no obvious trend and the effects of various fuel types were negligible. In Reference 5, in response to the NRC staff RAI, FANP submitted supplemental information about the applicability of the report to other fuel types. FANP stated that in order to confirm the applicability of the TR to fuel types not included in the database, FANP will document additional calculations using the methodology described in ANF-1358(P), Revision 3, for any fuel design that is not currently included. The additional calculations will be at minimum LFWH calculations for one additional representative cycle, which includes the new fuel type. This analysis will demonstrate that the correlation is still applicable to the new fuel type by showing that all of the residuals from the correlation are less than 0.0, as presented in ANF-1358(P), Revision 3. FANP will perform the calculations under the guidelines provided in Generic Letter 83-11, Supplement 1, "Licensee Qualifications for Performing Safety Analyses," (Reference 9) and they will be consistent with the methodology described in the approved TR EMF-2245(P)(A), Revision 0, "Application of Siemens Power

Corporation's Critical Power Correlations to Co-Resident Fuel" (Reference 10). In addition, FANP will demonstrate that the ratio of LHGRs for the limiting assemblies of that fuel type are less than the ratio used for the mechanical overpower analysis associated with that fuel type. This analysis will cover the anticipated operation of these fuel assemblies. It was further stated that the additional calculations will be maintained at the FANP offices and will be available for NRC audit.

FANP evaluated the bias of the data calculated using the bounding CPR relation and found that the revised correlation provided conservative results for each plant and cycle, with no obvious trends or biases that would offset the conservatism of the correlation.

The analysis has demonstrated that the MCPR after a LFWH event can be directly correlated to the MCPR prior to the LFWH event by plant operating parameters. The analytical model developed from this analysis was adjusted accordingly from the currently approved model described in Reference 6 to bound all of the calculated results. In Reference 4, in response to a question from the NRC staff, FANP further stated that the bounding correlation for determining delta-CPR reported in ANF-1358(P), Revision 3, gives slightly more conservative values than in the originally approved Reference 6, because it includes the original data, as well as additional data, that covers more diverse core conditions. Additional conservatism was incorporated in defining the bounding fit coefficients.

Under normal operations and AOOs, GDC 10 requires that fuel and cladding be protected from excessive strain and overheating. To protect against such failures, FANP imposes requirements that the fuel centerline temperature cannot exceed the melting point and the cladding strain during a transient cannot exceed 1 percent. [

] This limit is a result of performing the fuel centerline melt and the cladding transient strain analyses using the NRC-approved methodology and criteria. The mechanical analyses were performed using the NRC-approved methodology as described in the TR EMF-85-74, Revision 0, Supplement 1 and Supplement 2(P)(A), “RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model” (Reference 11). The mechanical design criteria are contained in the NRC-approved TR ANF-89-98(P)(A) Revision 1, “Generic Mechanical Design Criteria for BWR Fuel Designs” (Reference 12). In Reference 4, in response to the NRC staff's RAI, FANP further stated that typically, [

] FANP has shown in ANF-1358(P), Revision 3, that the increase in LHGR for the LFWH event was found to be less than the AOO LHGR limit.

The NRC staff concludes that the results of the analysis presented in ANF-1358(P), Revision 3, are applicable to BWR plants, given that all the limitations and conditions outlined in Section 4.0 of this SE are observed.

4.0 <u>LIMITATIONS AND CONDITIONS</u>

The following restrictions are imposed on the use of ANF-1358(P), Revision 3:

1) The methodology applies to BWR/3, BWR/4, BWR/5, and BWR/6 plants, and the fuel types which were part of the database ([

]), provided that the exposure, [

] are within the range covered by the data points presented in Reference 2.

2) To confirm applicability of the correlation to fuel types outside the database, FANP will perform additional calculations using the methodology, as described in Section 3.0 of this SE. In addition, FANP calculations will be consistent with the methodology described in Reference 10 and comply with the guidelines and conditions identified in the associated NRC staff SE.

3) The methodology applies only to the MCPROL and the LHGR for the LFWH event.

5.0 CONCLUSION

The NRC staff reviewed ANF-1358(P), Revision 3, which is a revised version to the previously approved Reference 6. Both TRs describe a generic methodology for evaluating the LFWH transient. ANF-1358(P), Revision 3, is based on an expanded database, including current core designs, and resulted in changes to the coefficients of the CPR correlation approved earlier by the NRC staff. ANF-1358(P), Revision 3, also extends the methodology to the determination of the LHGR. The NRC staff concluded that:

1) FANP used results from NRC staff-approved computer codes to develop this methodology and the revised CPR correlation.

2) The database of LFWH events represents a wide range of operating state points for BWRs, as well as various types of fuel design. For the fuel types outside the database, FANP will perform additional calculations, as outlined in Section 3.0, in order to justify applicability of the correlation.

3) The revised correlation, by design, yields conservative results relative to those calculated using NRC-approved methodologies.

4) The results have no obvious trends or biases that affect the conservatism of the revised correlation.

Therefore, within the restrictions noted in Section 4.0, the NRC staff finds ANF-1358(P), Revision 3 acceptable for referencing in licensing submittals.

6.0 REFERENCES

1. Letter from J. F. Mallay, Framatome ANP to NRC, "Request for Review and Approval of ANF-1358(P) Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," dated August 19, 2004. Agencywide Documents Access and Management System (ADAMS) Accession No. ML042390335.

2. ANF-1358(P), Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," by P. Wang, June 2004. ADAMS Accession No. for non-proprietary version is ML042390407, and ML042390425 for the proprietary version.

3. Letter from Framatome ANP to NRC, "Request for Additional Information - ANF-1358(P) Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," dated April 15, 2005. ADAMS Accession No. for non-proprietary version is ML052170292, and ML051090418 for the proprietary version.

4. Letter from Framatome ANP to NRC, "Request for Additional Information - ANF-1358(P) Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," dated April 27, 2005. ADAMS Accession No. for non-proprietary version is ML051190250, and ML051190255 for the proprietary version.

5. Letter from Framatome ANP to NRC, "Request for Additional Information - ANF-1358(P) Revision 3, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," dated June 3, 2005. ADAMS Accession No. ML051580374.

6. ANF-1358(P)(A), Revision 1, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Advanced Nuclear Fuels Corporation, September 1992.

7. XN-NF-80-19(P)(A), Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, "Advanced Nuclear Fuels Methodology for Boiling Water Reactors: Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology," Advanced Nuclear Fuels Corporation, November 1990.

8. EMF-2158(P)(A), Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," Siemens Power Corporation, October 1999.

9. Generic Letter 83-11, Supplement 1, "Licensee Qualifications for Performing Safety Analyses," dated June 24, 1999. ADAMS Accession No. ML031080345.

10. EMF-2245(P)(A), Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," August 2000. ADAMS Accession No. for package is ML003753187, ML003753223 for proprietary version, and ML003753200 for non-proprietary version.

11. EMF-85-74, Revision 0, Supplement 1 and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," February 1998.

12. ANF-89-98(P)(A), Revision 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," May 1995.

Principal Contributor: M. Razzaque

Date: September 20, 2005