May 11, 2001

Mr. James F. Mallay
Director, Regulatory Affairs
Framatome ANP, Richland, Inc.
2101 Horn Rapids Road
Richland, WA 99352

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT EMF-2310(P), REVISION 0, "SRP CHAPTER 15 NON-LOCA METHODOLOGY FOR PRESSURIZED WATER REACTORS" (TAC NO. MA7192)

Dear Mr. Mallay:

The NRC staff has completed its review of Topical Report EMF-2310(P), Revision 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors" submitted by Framatome ANP Richland, Inc. (FRA-ANP, previously known as Siemens Power Corporation (SPC)) on November 22, 1999, and supplemented by letter dated January 26, 2001.

On the basis of our review, the staff finds the subject report to be acceptable for referencing in license applications to the extent specified, and under the limitations delineated in the report, and in the enclosed safety evaluation (SE). The SE defines the basis for NRC acceptance of the report.

Pursuant to 10 CFR 2.790, we have determined that the enclosed SE does not contain proprietary information. However, we will delay placing the SE in the public document room for a period of ten (10) working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. 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 2.790.

The staff will not repeat its review and acceptance of the matters described in the report, when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with the procedures established in NUREG-0390, the NRC requests that FRA-ANP publish accepted versions of the report, including the safety evaluation, in the proprietary and non-proprietary forms within 3 months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an "-A" (designating accepted) following the report identification symbol. The accepted versions shall also incorporate all communications between FRA-ANP and the staff during this review.

Should our criteria or regulations change so that our conclusions as to the acceptability of the report are no longer valid, FRA-ANP and the licensees referencing the topical report will be expected to revise and resubmit their respective documentation, or to submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

/RA/

Stuart A. Richards, Director
Project Directorate IV and Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 702

Enclosure: Safety Evaluation

Should our criteria or regulations change so that our conclusions as to the acceptability of the report are no longer valid, FRA-ANP and the licensees referencing the topical report will be expected to revise and resubmit their respective documentation, or to submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,
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Stuart A. Richards, Director
Project Directorate IV and Decommissioning
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Project No. 702

Enclosure: Safety Evaluation

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT EMF-2310(P), REVISION 0

"SRP CHAPTER 15 NON-LOCA METHODOLOGY FOR

PRESSURIZED WATER REACTORS"

PROJECT NO. 702

1.0 INTRODUCTION

Framatome ANP Richland Inc. (FRA-ANP), formerly known as Siemens Power Corporation (SPC) submitted Topical Report EMF-2310(P), Revision 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors" (Reference 1), on November 22, 1999 (Reference 2), for NRC review and approval for application of the S-RELAP5 thermal-hydraulic analysis computer code (Reference 3), to Chapter 15 non loss-of-coolant accident (LOCA) transients. The application to use S-RELAP5 is as a replacement for the NRC approved code ANF-RELAP (References 4 and 5). S-RELAP5 is an updated version of ANF-RELAP. The application of S-RELAP5 to the analysis of the small-break LOCA (SBLOCA) under the guidance of 10 CFR Part 50, Appendix K was previously approved by the staff (Reference 6).

The stated goal of FRA-ANP is to apply a single computer code to the analysis of both LOCA and non-LOCA transient events. The code of choice is to be one that has had wide industry acceptance and application. To achieve this goal the decision was made to modify the approved ANF-RELAP code in such a way as to bring it up to a standard that incorporates the thermal-hydraulic code RELAP5/MOD2 (Reference 7), the fuel design code RODEX2 (Reference 8), along with codes specifically needed for LOCA analysis into a single system calculation. In so doing the RELAP5/MOD2 code was modified to include selected models from the RELAP5/MOD3 code (Reference 9), improved numerics, and models necessary to satisfy the requirements of 10 CFR Part 50, Appendix K, though not necessary for the non-LOCA transient events.

The XCOBRA-IIIC code (Reference 10), will continue to be used to obtain the final predicted Minimum Departure from Nucleate Boiling Ratio (MDNBR) for each non-LOCA transient event. The core conditions calculated for the reactor coolant system (RCS) by S-RELAP5 will be used as input to the XCOBRA-IIIC core and subchannel methodology to predict the event-specific MDNBR.

2.0 CODE APPLICABILITY

During the course of review of S-RELAP5 for application to SBLOCA transients, extensive examination of the code numerics, two-fluid equations, heat transfer, point kinetics, and general assessment took place. At that time requests for additional information (RAIs) were developed relating to the code itself (Reference 11), and responded to by FRA-ANP (Reference 12). Meetings were held with the Advisory Committee on Reactor Safeguards (ACRS) and Thermal-Hydraulic Phenomena Subcommittee regarding the modeling within S-RELAP5. The meetings with the ACRS and the reviews conducted by its members and their consultants were considered in the preparation of the staff's RAIs.

The RELAP5 computer code is a light water reactor transient analysis code developed for the NRC for use in rulemaking, licensing audit calculations, evaluation of operator guidelines, and as a basis for nuclear power plant analyses. RELAP5 is a general purpose code that, in addition to calculating the behavior of a RCS during a transient, can be used for simulation of a wide variety of hydraulic and thermal transients in both nuclear and non-nuclear systems involving mixtures of steam, water, non-condensable gas, and solute. The RELAP5 code is based on a nonhomogeneous and nonequilibrium model for the two-phase system. Solution is by a partially implicit numerical scheme to permit economical calculation of system transients. The objective of the RELAP5 development effort was to produce a code that included important first-order effects necessary for accurate prediction of system transients but that was sufficiently simple and cost effective so that parametric or sensitivity studies were possible.

The code includes many generic component models from which general systems can be simulated. These component models include pumps, valves, pipes, heat releasing or absorbing structures, reactor point kinetics, electric heaters, jet pumps, turbines, separators, accumulators, and control and trip system components. In addition, special process models are included for effects such as form loss, flow at an abrupt area change, branching, choked flow, counter-current flow limit (CCFL), boron tracking, and noncondensable gas transport. The code also incorporates many user conveniences such as extensive input checking, free-form input, internal plot capability, restart, renodalization, and variable output edits.

The S-RELAP5 code evolved from FRA-ANP's ANF-RELAP code, a modified RELAP5/MOD2 version used by FRA-ANP for pressurized water reactor (PWR) plant licensing analyses that included the SBLOCA analysis, steam line break analysis, and PWR non-LOCA Updated Final Safety Analysis Report (UFSAR) Chapter 15 event analyses. During the modifications to permit realistic analyses, enhancements were made to incorporate the requirements of 10 CFR Part 50, Appendix K SBLOCA analysis. The code structure was also modified to be similar to that of RELAP5/MOD3. This included incorporation of the RELAP5/MOD3 reactor kinetics, control systems and trip systems models.

Some of the major modifications made to RELAP5/MOD2 and ANF-RELAP to produce the S-RELAP5 code include the following:

1. Multi-Dimensional Capability - Two-dimensional treatment has been added to the hydrodynamic field equations. This capability can handle the Cartesian and cylindrical coordinate systems.

2. Energy Equations - The energy equations were modified to conserve the energy transported into and out of a control volume, thus correcting the tendency of the RELAP5 codes to produce an energy error when a large pressure gradient exists between two adjacent control volumes.

3. Numerical Solution of Hydrodynamic Field Equations - Where the RELAP5 codes use a Gaussian elimination solver to reduce the hydrodynamic finite-difference equations to a pressure equation, S-RELAP5 uses algebraic manipulation.

4. State of Steam-Noncondensable Mixture - At very low steam quality, the ideal gas equation is used for both steam and noncondensable gas. This permits calculation of state relations for both steam and noncondensable gas at low steam quality and also the presence of pure noncondensable gas below the ice point.

5. Hydrodynamic Constitutive Models - Significant modifications were made to the RELAP5 interphase friction and interphase mass transfer models. Some of the flow regime (two-phase flow) transient criteria were modified to be consistent with published data. Transient flow regimes are introduced for smoothing the constituent models. Most of the RELAP5/MOD2 partition functions were only slightly modified if at all. A more accurate wall friction factor approximation replaces the Colebrook equation.

6. Heat Transfer Model - The RELAP5/MOD2 use of different heat transfer correlations in reflood was eliminated. The Dittus-Boelter single phase steam heat transfer correlation was replaced with the Sleicher-Rouse correlation which gives higher steam temperatures and has a smaller uncertainty range.

7. Choked Flow - The Moody critical flow model was implemented for 10 CFR Part 50, Appendix K calculations. The modification of ANF-RELAP to use an iterative scheme to compute the equation of state at the choked plane rather than using the previous time step information was also implemented.

8. Counter-Current Flow Limit - The Kutateladze type CCFL correlation of ANF-RELAP was replaced with the Bankoff form. This conforms with RELAP5/MOD3.

9. Component Models - The pump model includes the EPRI pump performance degradation data, and the pump head term in the fluid field equations was made more implicit. The ICECON containment code was incorporated to run concurrently with S-RELAP5. User guidelines were implemented to specify a replacement procedure for modeling the accumulator model.

10. Fuel Models - The RODEX2 fuel deformation and conductivity models were incorporated for SBLOCA applications. The flow diversion model of TOODEE2 was implemented to account for the effect of cladding rupture on heat transfer. The Baker-Just metal-water reaction model was implemented as required by 10 CFR Part 50, Appendix K.

11. Code Architecture - Modifications were made to bring the S-RELAP5 code into conformance with the description of the RELAP5/MOD3 code architecture. This

includes writing the code in FORTRAN 77 and maintaining a common source for all computer versions.

Many of the above noted modifications made in the development of S-RELAP5 are not applicable to non-LOCA transient analysis but were reviewed during the overall code review performed for the application to the SBLOCA. The non-LOCA transient events are described in Reference 13. Specific event application of S-RELAP5 is given in Table 1.

Table 1

Applicable SRP Chapter 15 Events

Event	SRP No.
15.1 - Increase in Heat Removal by the Secondary System	
Decrease in Feedwater Flow	15.1.1
Increase in Feedwater Flow	15.1.2
Increase in Steam Flow	15.1.3
Inadvertent Opening of Steam Generator Relief/Safety Valve	15.1.4
Steam System Piping Failures Inside and Outside Containment	15.1.5
15.2 - Decrease in Heat Removal by Secondary System	
Loss of Outside External Load (LOEL)	15.2.1
Turbine Trip	15.2.2
Loss of Condenser Vacuum	15.2.3
Closure of Main Steam Isolation Valve	15.2.4
Steam Pressure Regulator Failure	15.2.5
Loss of Non-Emergency AC Power to the Station Auxiliaries	15.2.6
Loss of Normal Feedwater (LONF) Flow	15.2.7
Feedwater System Piping Breaks Inside and Outside Containment	15.2.8
15.3 - Decrease in Reactor Coolant Flow Rate	
Loss of Forced Reactor Coolant Flow (LOCF)	15.3.1
Flow Controller Malfunctions	15.3.2
Reactor Coolant Pump (RCP) Rotor Seizure	15.3.3
RCP Shaft Break	15.3.4

Table 1
(continued)

15.4 - Reactivity and Power Distribution Anomalies	
Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal From a Subcritical or Low Power Startup Condition	15.4.1
Uncontrolled RCCA Bank Withdrawal at Power	15.4.2
RCCA Misoperation	15.4.3
Dropped Rod/Bank	15.4.3.1
Single Rod Withdrawal	15.4.3.2
Statically Misaligned RCCA	15.4.3.3
Startup of an Inactive Loop at an Incorrect Temperature	15.4.4
Chemical and Volume Control System (CVCS) Malfunction that Results in a Decrease of Boron Concentration (Boron Dilution)	15.4.6
Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (Misloaded Assembly)	15.4.7
Spectrum of Rod Ejection Accidents	15.4.8
15.5 - Increase In Reactor Coolant Inventory	
Inadvertent Operation of the Emergency Core Cooling System that Increases Reactor Coolant Inventory	15.5.1
CVCS Malfunction that Increases Reactor Coolant Inventory	15.5.2
15.6 - Decreases in Reactor Coolant Inventory	
Inadvertent Opening of a Pressurizer Pressure Relief Valve	15.6.1
Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	15.6.2
Radiological Consequences of Steam Generator Tube Rupture (SGTR)	15.6.3

Application of S-RELAP5 is to be made to Combustion Engineering (CE) 2x4 plants and Westinghouse 3 and 4-loop plants.

3.0 STAFF APPROACH TO REVIEW

The staff performed an extensive review of the S-RELAP5 code during the review of the code for application to the SBLOCA events. Review for application to non-LOCA transient events focused on the results of assessment cases and comparison to calculations performed using the approved ANF-RELAP code.

4.0 CODE ASSESSMENT

The assessment of S-RELAP5 for application to Chapter 15 non-LOCA transient events consists of calculation of four Loss-of-Fluid Test (LOFT) program transient tests and comparison with ANF-RELAP calculations of various transients. The four LOFT tests that were calculated are:

- LOFT L6-1 Loss of Load
- LOFT L6-2 Loss of Primary Flow
- LOFT L6-3 Excessive Steam Load
- LOFT L6-5 Loss of Feedwater

The four LOFT tests represent SRP Category 15.1 - Increase in Heat Removal by the Secondary System; SRP Category 15.2 - Decrease in Heat Removal by Secondary System; and SRP Category 15.3 - Decrease in Reactor Coolant Flow Rate events.

The major parameters for each of the tests were predicted well in both trend, timing and magnitude. For the loss of load case, the steam generator and pressurizer levels were underpredicted (conservative) by both S-RELAP5 and ANF-RELAP, while both the hot leg and cold leg temperatures were conservatively overpredicted. For the loss of primary flow case, again the hot and cold leg temperatures were overpredicted. For the case of excessive steam load, the secondary and pressurizer levels were conservatively predicted and the reactor power prediction was within 2.5 percent of that measured at the peak. Again, the hot and cold leg temperatures were conservatively predicted. For the loss of feedwater case, the steam generator secondary side liquid level showed a small oscillation about the measured value, a low pressurizer pressure prediction and conservative predictions of hot and cold leg temperatures and steam generator steam flow.

Overall, the predictions of the LOFT transient tests show good agreement with the measured results and good comparison with the calculated results of ANF-RELAP. The simulations performed by the code included modeling of automatic control components and systems such as pressurizer sprays and heaters, feedwater control, pressure control, steam generator level control, and reactor power.

Chapter 15 transient events are to be analyzed considering the following:

- Timing of Loss-of-Offsite Power
- Mitigating Systems
- Operator Actions

- Single Failures
- Number of Loops Operating
- Axial and Radial Power Distributions

In performing the analyses, the analyst is expected to select values and equipment for the above in accordance with the guidelines provided in the appropriate Regulatory Guides, Standard Review Plan, and computer code user guides to ensure conservative calculations are performed. In addition, the analyst must ascertain where it is appropriate to use nominal or technical specification values for the initial core power level, initial reactor coolant flow rate, initial reactor coolant temperature, initial reactor pressure and pressurizer level, moderator temperature reactivity coefficient, Doppler reactivity coefficient, reactor protection system trip and equipment setpoints and delay times, and scram characteristics. The sample problems provided being of a generic nature, assume nominal values for these parameters for the given plant designs being analyzed.

The comparisons with event-specific ANF-RELAP calculations included both CE and Westinghouse design transients. The events analyzed fall into two major categories: Anticipated Operational Occurrences and Postulated Accidents. Specific sample problems provided in Reference 1, include the main steam line break (MSLB), both pre- and post-scram, LOEL, LONF, LOCF, the uncontrolled rod bank withdrawal at power, and SGTR.

The staff finds that the assessment performed in support of the S-RELAP5 application to Chapter 15 non-LOCA events is adequate in that it compares code results with ANF-RELAP results for the selected LOFT transients and for plant calculations. Specific plant applications may still require additional supporting assessment calculations should plant specific features or conditions be outside the range of the generic assessments.

5.0 EVALUATION OF S-RELAP5

The staff's SE on the ANF-RELAP code application to Chapter 15 non-LOCA events (Reference 5), identified six restrictions on use of the code. The staff notes the following regarding those restrictions:

- The stated application of the S-RELAP5 code is for the events listed above in Table 1. There are other computer codes and methodologies employed for evaluation of the events not listed in the table. For each licensing basis event analyzed, the applicant must, as always, justify the methodology used whether by reference to S-RELAP5 or whatever methodology has been used.

- Analysis of Chapter 15, Events 15.6.2 and 15.6.3, on radiological consequences is beyond the scope of the S-RELAP5 computer code. However, where primary and secondary mass and energy release are used as the principal source of the radiological components of the event, S-RELAP5 is capable of providing that information.

- The S-RELAP5 documentation provides support for use of the code in cases for which upper head voiding occurs and cases where the boron tracking model is used.

- The S-RELAP5 documentation provides sufficient support for use of the code in cases for which natural circulation cooldown is to be calculated.

- As is the case in reviewing all generic topical report applications, submittals for specific plants and events must include justification of the nodalization used, input parameters, options selected, and all of the parameters that influence the progression of the event and its mitigation.

- The S-RELAP5 code incorporates the RODEX2 fuel analysis code. Although the application of the S-RELAP5 code to Chapter 15 non-LOCA events uses a conservative input value of the fuel rod gap conductance, should it be necessary, or desirable, an analyst does have available the RODEX2 capability as an integral part of the S-RELAP5 code.

The staff finds that the modifications performed in developing the S-RELAP5 code satisfy the restrictions that had been placed on the ANF-RELAP code when applied to Chapter 15 non-LOCA events. The staff encourages and supports efforts to develop methodologies capable of analyzing a broad spectrum of events rather than separate methodologies for each event which must be analyzed. The staff also notes, however, that a generic topical report describing a code such as S-RELAP5 cannot provide full justification for each specific individual plant application. The individual applicant must still provide justification for the specific application of the code which is expected to include as a minimum, the nodalization, defense of the chosen parameters, any needed sensitivity studies, justification of the conservative nature of the input parameters, and calculated results.

The MSLB is one of the most challenging non-LOCA events for a PWR. The analysis performed with the S-RELAP5 code provides the thermal-hydraulic response of the RCS combined with the RELAP5 point kinetics model. A detailed fuel failure calculation is performed by using the thermal-hydraulic conditions predicted with detailed fuel and cladding calculations from XCOBRA-IIIC (Reference 10) and a neutronics code for the highest powered fuel assemblies. Since the report under review is intended to be generic in nature, an extensive evaluation is performed to determine the most important phenomena affecting the CE configuration as well as the Westinghouse configuration. The calculation is then broken down into two phases: pre-scram and post-scram. The calculation is further complicated in having to separate those plants which still have high boron concentration storage tanks from those which have removed the tanks. The neutronic response is significantly different for the cases with or without high concentration boron injection.

Although the remainder of the transients for which S-RELAP5 is to be applied are less severe than the MSLB, FRA-ANP evaluates the important phenomena for the non-LOCA events to determine that the code is capable of predicting the phenomenological response for the plant being analyzed. Although this is not a phenomena identification and ranking table in the strictest sense, the goal is achieved in showing that the important phenomena are adequately represented.

The staff concludes that the S-RELAP5 code, with conservative input modeling assumptions, is capable of addressing the thermal-hydraulic response of the target non-LOCA events in a conservative manner in keeping with the staff's SRP guidance and is, therefore, an acceptable replacement for the ANF-RELAP code.

6.0 CONCLUSIONS

The staff supports the efforts of applicants to integrate codes for analysis of accidents and transients rather than manual transfer of information between the codes. Integrating the thermal-hydraulic, fuel rod performance, and other codes, permits a smoother and more accurate prediction of the performance of the system under accident conditions.

The staff finds that the modifications performed in developing the S-RELAP5 code satisfy the restrictions that had been placed on the ANF-RELAP code when applied to Chapter 15 non-LOCA events. The staff encourages and supports efforts to develop methodologies capable of analyzing a broad spectrum of events rather than separate methodologies for each event which must be analyzed. The staff also notes, however, that a generic topical report describing a code such as S-RELAP5 cannot provide full justification for each specific individual plant application. The individual applicant must still provide justification for the specific application of the code which is expected to include as a minimum, the nodalization, defense of the chosen parameters, any needed sensitivity studies, justification of the conservative nature of the input parameters, and calculated results.

The staff finds that the assessment performed in support of the S-RELAP5 application to Chapter 15 non-LOCA events is adequate in that it compares code results with ANF-RELAP results for the selected LOFT transients and for plant calculations. Specific plant applications may still require additional supporting assessment calculations should plant specific features or conditions be outside the range of the generic assessments.

The staff concludes that the S-RELAP5 code is capable of addressing the thermal-hydraulic response of the target non-LOCA events in a conservative manner and is, therefore, an acceptable replacement for the ANF-RELAP code.

7.0 REFERENCES

1. EMF-2310(P), Rev. 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Siemens Power Corporation, November 1999.

2. Letter, SPC to NRC, dated November 22, 1999.

3. EMF-2100(P), Rev. 2, "S-RELAP5 Models and Correlations Code Manual," Siemens Power Corporation, January 2000.

4. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, May 1992.

5. Letter, NRC to Siemens Nuclear Power Corporation, "Acceptance for Referencing of Topical Report ANF-89-151(P), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," dated March 16, 1992.

6. Letter, NRC to Framatome ANP, Richland, Inc., "Acceptance for Referencing of Licensing Topical Report EMF-2328(P), Revision 0, 'PWR Small Break LOCA Evaluation Model, S-RELAP5 Based'," dated March 15, 2001.

7. NUREG/CR-4312, EGG-2396, Rev. 1, "RELAP5/MOD2 Code Manual," March 1987.

8. XN-NF-81-58(P)(A), Rev. 2, Supplements 1 and 2, "RODEX2: Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, March 1984.

9. NUREG/CR-5535, INEL-95/0174, "RELAP5/MOD3 Code Manual," August 1995.

10. XN-75-21(P)(A), Rev. 2, "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," Exxon Nuclear Company, January 1986.

11. Letter NRC to SPC dated December 11, 2000.

12. Letter SPC to NRC dated January 26, 2001.

13. NUREG-0800, "USNRC Standard Review Plan," U. S. Nuclear Regulatory Commission, Washington, DC 20555, July 1981.

Principal Contributor: R. R. Landry

Date: May 11, 2001