May 31, 2000

Mr. James F. Mallay
Director, Nuclear Regulatory Affairs
Siemens Power Corporation
2101 Horn Rapids Road
Richland, WA 99352

SUBJECT: SIEMENS POWER CORPORATION RE: REQUEST FOR CONCURRENCE ON SAFETY EVALUATION REPORT CLARIFICATIONS (TAC NO. MA6160)

Dear Mr. Mallay:

References:

1. Letter from J. Mallay (SPC) to USNRC, "Request for Concurrence on SER Clarifications," July 28, 1999.

2. Letter from J. Mallay (SPC) to USNRC, "Revision to Request for Concurrence on SER Clarifications," August 9, 1999.

3. Letter from J. Mallay (SPC) to USNRC, "Revisions to Attachment 1 of Letter NRC:99:030, Request for Concurrence on SER Clarifications," October 12, 1999.

4. XN-NF-84-105(P)(A), Volume 1 and Volume 1, Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Company, February 1987.

5. ANF-913(P)(A), Volume 1 Revision 1 and Volume 1, Supplements 2, 3, and 4, "COTRANSA2: A Computer Program for BWR Transient Analysis," Advanced Nuclear Fuels Corporation, August 1980.

6. ANF-89-98(P)(A), Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Company, May 1995.

7. XN-NF-82-06(P)(A), Supplement 1, Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Supplement 1, "Extended Burnup Qualification of ENC 9x9 BWR Fuel," Advanced Nuclear Fuels Corporation, May 1988.

8. ANF-89-014(P)(A), "Revision 1 and Supplements 1 and 2, Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X Reload Fuel," Advanced Nuclear Fuels Corporation, October 1991.

9. ANF-1 125(P)(A), Supplement 1 and Supplement 2, "ANFB Critical Power Correlation," Advanced Nuclear Fuels Corporation, April 1990.

10. EMF-1 997(P)(A), Revision 0, "ANFB-1 0 Critical Power Correlation," Siemens Power Corporation, July 1998.

The staff has reviewed your July 28 and October 12, 1999, letters that requested NRC staff concurrence on Siemens Power Corporation interpretations of safety evaluation (SE) restrictions and the supplementary information which was supplied. The letters referred to two broad categories of interpretations: (1) transient analysis with XCOBRA-T and COTRANSA2; and (2) fuel design issues. The fuel design issues are further broken into sections on boiling water reactor (BWR) issues, linear heat generation rate (LHGR) issues and critical heat flux (CHF) correlation issues.

XCOBRA-T and COTRANSA2 are part of a code system that is used primarily for Anticipated Operation Occurrence (AOO) analysis to predict the transient critical power ratio. Most of the restrictions in the SE on XCOBRA-T and COTRANSA2 relate to the fact that it cannot model negative bypass flow (i.e., reversed flow). Siemens Power Corporation (SPC) has traditionally interpreted these restrictions to mean that these codes, when used together, cannot be used to model transients where negative bypass flow significantly contributes to the predicted CHF. In order to ensure that the staff concurs with the SPC interpretation, SPC has, as its current practice, sent a letter to the staff describing the transients to which they have applied this interpretation. In addition to the specific transients approved in the XCOBRA-T and COTRANSA2 SEs, SPC applies these codes to three broad categories: (1) pressurization transients, (2) change in coolant inventory, and (3) change in core flow. The staff is unaware of any transients which fit into these categories which are significantly affected by negative bypass flow. Therefore, the staff concurs with the SPC interpretation that the XCOBRA-T and COTRANSA2 methodology may be used to analyze all of the transients in the enclosed table. SPC should continue to assess the applicability of these codes to all of the transients listed in the table to ensure that the assumptions regarding the significance of negative bypass flow remain valid. These clarifications are applicable to References 2 and 3.

The staff has also reviewed the information supplied regarding BWR and LHGR clarifications. The information relates to the application of RODEX2A, BWR generic fuel design criterion, thermal-hydraulic stability, and nuclear characteristics of fuel, and LHGR limit curves. The information supplied discusses SPC plans to raise the burnup, major versus minor fuel changes, and the generation and application of thermal limits. The staff concurs with the SPC interpretations regarding BWR and LHGR clarifications. These clarifications are applicable to References 4, 5, and 6.

Finally, SPC presented information describing measures taken during the application of its critical power correlation to ensure that when conditions are predicted which go beyond the database used to derive the correlation, either the calculation is stopped or the situation is treated conservatively. These measures are intended to ensure that the correlation is used as described in the approved licensing topical report which justifies the acceptability of the correlation. The staff considers the proposed remedies to be acceptable provided that the code provides the analyst with a report of the fact that it corrected an attempt to use the correlation

out-of-bounds. This report will notify the analyst to scrutinize the input and output to look for any errors which could have caused the prediction to go out of bounds. These clarifications are applicable to References 7 and 8.

Sincerely,

/***RA/***

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

Project No. 702

Enclosure: Table

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Sincerely,

/***RA/***

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

Project No. 702

Enclosure: Table

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TABLE

15.1.1-15.1.3	Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Demand
15.2.1-15.2.5	Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)
15.2.7	Loss of Normal Feedwater Flow
15.3.1-15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions
15.3.3-15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
15.4.4-15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
15.5.1	Inadvertent Operation of ECCS that Increases Reactor Coolant Inventory
15.6.1	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve
15.8	Anticipated Transients Without Scram (the Initial Pressurization Only)