September 13, 2000

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

SUBJECT: ACCEPTANCE FOR REFERENCING OF SIEMENS POWER CORPORATION TOPICAL REPORT EMF-2292(P), REVISION 0, "ATRIUM™-10: APPENDIX K SPRAY HEAT TRANSFER COEFFICIENTS" (TAC NO. MA6785)

Dear Mr. Mallay:

Topical Report EMF-2292(P), Revision 0, "ATRIUM™-10: Appendix K Heat Transfer Coefficients," was submitted for NRC review by the Siemens Power Corporation (SPC) by letter dated September 27, 1999.

This topical report presents SPC experimental results and HUXY computer code calculations that substantiate that the application of 10 CFR Part 50 Appendix K convective heat transfer coefficients during loss of coolant accident spray cooling of the ATRIUM™-10 fuel design is conservative. A similar topical report demonstrating the conservatism of the Appendix K spray heat transfer coefficients for the ANF 9x9 fuel design with an internal water canister has previously been approved.

The staff has reviewed the topical report and finds it acceptable for referencing in licensing actions as stated in our enclosed safety evaluation (SE).

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 SPC 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 a "-A" (designating accepted) following the report identification symbol. The accepted versions shall also incorporate all communications between SPC 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, SPC 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 by Stephen Dembek for/

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

In accordance with the procedures established in NUREG-0390, the NRC requests that SPC 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 SPC 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, SPC 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
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 702

Enclosure: Safety Evaluation

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT EMF-2292 (P), REVISION 0

"ATRIUM™-10: APPENDIX K SPRAY HEAT TRANSFER COEFFICIENTS"

SIEMENS POWER CORPORATION

PROJECT NO. 702

1.0 INTRODUCTION AND BACKGROUND

By letter of September 27, 1999 (Reference 1), Siemens Power Corporation (SPC) requested NRC review and acceptance for referencing in licensing actions, Topical Report EMF-2292(P), Revision 0, "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients."

Appendix K, paragraphs I.D.6 and I.D.7, of Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50) specifies for reactors with jet pumps, convective heat transfer coefficients be used under spray cooling conditions for the fuel rods and channel box for a 7x7 fuel rod array design. SPC has conducted a series of experimental tests in their fuel cooling test facility (FCTF) (Reference 2) for their ATRIUM™-10 fuel design (Reference 3), a 10x10 fuel rod array with an internal water canister (IWC). These tests were performed to confirm that the spray cooling convective heat transfer coefficients prescribed in 10 CFR Part 50 Appendix K for 7x7 rod arrays are conservative for the 10x10 array when used for loss of coolant accidents (LOCA) analyses in jet pump boiling water reactors (BWR). SPC previously performed a similar test series for its ANF 9x9 rod array with IWC fuel design (Reference 4), which was reviewed and approved by the NRC.

SPC uses a multi-rod heat-up model, embodied in the HUXY computer code (References 5 and 6), to calculate fuel rod response during the core spray cooling period of a LOCA, in accordance with 10 CFR Part 50, Appendix K. The HUXY code results, using the convective heat transfer coefficients prescribed in Appendix K, were compared with experimental results obtained from a series of fuel heatup tests to demonstrate conservatism for the ATRIUM™-10 fuel design.

2.0 TECHNICAL EVALUATION

The topical report describes the 10x10 test bundle mechanical design and the series of 23 experimental tests performed in the FCTF and further describes the multiple HUXY calculations performed for each test. The selection of the experimental conditions were chosen to simulate the heat transfer response for BWR 3, 4, and 5/6 plant designs. The HUXY calculations simulate each test condition, but use the Appendix K prescribed heat transfer coefficients. The radiation emissivity input data are conservatively chosen to maximize the calculated cladding temperature. The topical report presents the experimental data and provides a comparison of

these data with the corresponding HUXY calculations, in terms of the maximum temperature as a function of spray cooling time.

For each test the maximum cladding temperature predicted by HUXY exceeded the maximum measured cladding temperature.

2.1 Experimental Tests

The test bundle simulated a full-scale ATRIUM-10 BWR fuel assembly, including the upper tie plate, electrically heated rods, eight prototypical spacers, and an IWC. The inlet orifice, lower tie plate and outer assembly channel box were also simulated. The heated rods used a typical axial power profile. Thermocouples were placed at five elevations above the bottom of the rod heated length to span the location of the peak cladding temperature. The general initial and transient test conditions were maintained to be similar to the previously conducted 9x9 spray heat transfer tests approved in Reference 4.

2.2 HUXY Input

The HUXY calculations used the initial power, decay profile, spray initiation and test termination times from the experimental test data. The measured initial average temperature of the instrumented rods and the channels were input to HUXY, along with the local and axial power peaking factors for each elevation to be modeled. The approved assembly channel quenching methodology from Reference 5 was used for the HUXY calculations. The prescribed Appendix K spray heat transfer coefficients for rods and channels were applied at spray initiation. Further conservatism was introduced by assigning no delay time to the IWC quench calculations.

2.3 Comparison of Analytical Results and Experimental Data

The topical report first presented results of individual rod temperature comparisons between code calculations and test results during the transient test heat-up period prior to spray initiation and power decay, to confirm the adequacy of the test configuration and the computer model. The comparisons of calculated versus measured rod temperature after spray initiation were then displayed, along with a comparison of the 9x9 IWC and ATRIUM-10 data, for each class of BWRs being modeled. The measured temperature histories for the three rods that achieved the highest temperature were compared with the HUXY predicted temperatures for the highest three rods at each of four axial thermocouple elevations for each transient test sequence. In all cases, the final measured three highest test temperatures were less than or equal to the three highest HUXY predicted temperatures for each test sequence. HUXY, with the Appendix K coefficients, did not predict temperature turnaround and decrease at any axial location for any of the tests, although some test measurements did show temperature decreases.

3.0 EVALUATION

SPC conducted a series of 23 tests for the ATRIUM-10 bundle and performed multiple HUXY predictive calculations for each test. The calculations simulated each test condition, but used the heat transfer coefficients prescribed in Appendix K. The results show that, for each test, the maximum predicted cladding temperature exceeded the maximum measured cladding temperature. This demonstrates that the use of the Appendix K heat transfer coefficients in the

emergency core cooling system (ECCS) analyses of ATRIUM-10 fuel bundle design is conservative. The results were also compared to similar tests previously conducted that were reviewed and approved by Reference 4 for the ANF 9x9 internal canister fuel design, showing similar conservatism. The staff finds that the SPC calculations consistently and conservatively overpredict the measured peak cladding temperatures. This finding shows the topical report to be an acceptable demonstration of the conservatism of the application of the Appendix K coefficients.

4.0 CONCLUSIONS

The staff has reviewed the evaluations described in Topical Report EMF-2292(P), Revision 0, "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients." The staff also reviewed the limitations imposed on SPC ECCS model applications from prior topical report SERs (References 7 and 8) and the proposed application is consistent with the limitations. The staff also reviewed the previous similar SPC 9x9 spray cooling tests and comparisons with the ATRIUM™-10 results to substantiate the data trends and the continued conservatism of application of the Appendix K coefficients. The reported test results, along with the references, demonstrate the acceptability of the application of Appendix K coefficients for use during the spray cooling period of LOCA analyses for jet pump BWR plants with ATRIUM™-10 reload fuel assemblies. The staff concludes that, as discussed above, Topical Report EMF-2292(P), Revision 0 is acceptable for referencing in licensing applications.

5.0 REFERENCES

1. Letter from J. F. Mallay (SPC) to U. S. NRC, "Request for Review of EMF-2292(P), Revision 0, "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," NRC:99:041, September 27, 1999.

2. XN-NF-81-46, "Exxon Nuclear Company Jet Pump BWR Refill and Reflood Fuel Cooling Test Program," August 10, 1981.

3. EMF-98-006(P), Revision 0, "Mechanical Design Evaluation for Siemens Power Corporation ATRIUM™-10 BWR Reload Fuel to 54 MWd/kgU Assembly Exposure," January 1998.

4. ANF-CC-33(P)(A), Supplement 2, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option," Siemens Power Corporation, January 1991.

5. XN-CC-33(P)(A), Revision 1, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option," Exxon Nuclear Company, November 1975.

6. EMF-CC-102(P), Revision 1, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option," Siemens Power Corporation, February 10, 1999.

7. ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," Advanced Nuclear Fuels Corporation, January 1993.

8. ANF-91-048(P)(A), Supplements 1 and 2, "BWR Jet Pump Model Revision for RELAX," Siemens Power Corporation, October 1997.

Principal Contributor: E. Kendrick

Date: September 13, 2000