February 10, 2000

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

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT EMF-92-081, REVISION 1, "STATISTICAL SETPOINT/TRANSIENT METHODOLOGY FOR WESTINGHOUSE TYPE REACTORS," (TAC NO. MA4593)

Dear Mr. Mallay:

The staff has completed its review of the subject topical report submitted by Siemens Power Corporation (SPC) by letter dated December 21, 1998. 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 of the matters described in the report, and found acceptable when the report appears as a reference in license applications, except to ensure 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 procedures established in NUREG-0390, the NRC requests that SPC publish accepted versions of this report, including the safety evaluation, in 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Should our criteria or regulations change so that our conclusions as to the acceptability of the report are no longer valid, SPC and the licensees referencing the topical report will be expected to revise and resubmit their respective documentation, or to submit justification for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,

/***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, 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/***
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

DISTRIBUTION:
File Center
PUBLIC
PDIV-2 Reading
SRichards (clo)
JWermiel
OGC
ACRS

*See Previous Concurrence

To receive a copy of this document, indicate **"C"** in the box								
OFFICE	PDIV-2/PM	C	PDIV-2/LA	C	SRXB*		PDIV-2/SC	PDIV/D
NAME	NKalyanam*		EPeyton*		JWermiel*		SDembek*	SRichards
DATE	02/07/00		02/07/00		12/10/99		02/07/00	02/08/00

DOCUMENT NAME: C:\MA4593-closure letter with ~.wpd
OFFICIAL RECORD COPY

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO SIEMENS POWER CORPORATION LICENSING TOPICAL REPORT

EMF-92-081(P), REVISION 1,

"STATISTICAL SETPOINT/TRANSIENT METHODOLOGY

FOR WESTINGHOUSE TYPE REACTORS"

1.0 INTRODUCTION

By letter dated December 21, 1998, Siemens Power Corporation (SPC) proposed to revise the methodology used for statistical setpoint and transient analysis of Westinghouse type reactors (Ref. 1). The revised methodology incorporates new ways to statistically combine the uncertainties in the trip setpoints and limiting conditions of operation (LCOs). Additionally, a new methodology for calculating trip setpoints and verifying trip systems during transients is described. These changes to the setpoint methodology will facilitate automating the methodology, decreasing the user effect and the potential for introducing user errors.

This safety evaluation (SE) evaluates the changes incorporated into the methodology and will not reiterate the findings of the previous SE for the methodology that is unchanged.

2.0 METHODOLOGY CHANGES

The licensee requested approval for revising the statistical setpoint and transient analysis methodology previously approved for Westinghouse type reactors. The revision describes in greater detail how SPC confirms departure from nucleate boiling (DNB), fuel centerline melt (FCM), and hot-leg saturation protection, and incorporates new ways to combine the uncertainties of those calculations.

Overpower Delta Temperature (OPΔT) Reactor Trip Setpoint

The OPΔT setpoint provides 95 percent probability at a 95 percent confidence level that fuel melt will not occur during transients and anticipated operational occurrences (AOOs.) The form of the OPΔT setpoint trip is the same as used by Westinghouse (Ref. 2). The methodology used in determining coefficients K4 and K6 was not revised by this submittal.

To calculate the setpoint coefficients, a FCM limit based on the operating cycle and core design is needed. This FCM limit is expressed in terms of KW/ft; thus, the FCM limit is expressed as a function of a limit on linear heat generation rate (LHGR). The FCM limit is a cycle specific parameter which is calculated for each reload using the RODEX2 code, a quasi-static fuel rod performance code used by SPC (Refs. 3 and 4.) The calculation of the FCM limit accounts for the gadolinia concentration, burnup history, axial power shape, and periodic power spikes (to account for the scram delay time). To correlate the FCM limit to a LHGR limit, melt curves for the fuel rods are generated. These melt curves provide a relationship between melt power and the rod burnup and gadolinia concentration. Thus, the power at which FCM begins for each rod type is identified and through a relationship is converted into a LHGR. The FCM limit is the minimum LHGR for all fuel types divided by the fraction of power generated in the rod. Revising the methodology facilitates automation of the calculation process and reducing the user effect on the calculation results.

The trip reset function is designed to accommodate events when the axial power shape distribution undergoes large changes resulting in core power distributions which have a total peaking in excess of Fq. This function is part of the OPΔT trip function. It compensates the setpoint value when the actual difference between the normalized fluxes from the top and bottom detectors of the power range nuclear ion chambers (ΔI) differs from the assumed value. Multiple axial shapes are used in determining the reset function so all potential axial shapes which could result in a peaking in excess of Fq are included in the calculation. Since the OPΔT trip equation is expressed in terms of core ΔT, the FCM power level probabilities for the axial shapes of interest are also converted into an expression for core ΔT. This calculation is performed at minimum pressure to maximize the core ΔT over the power level probability distribution. The minimum reset function can be expressed in terms of the core ΔT and the FCM core ΔT. A bounding reset function is determined which will prevent FCM for all axial shapes. This bounding reset function accounts for the uncertainties included in the calculation of the OPΔT reset function including the uncertainty in ΔI and protects against FCM with a 95/95 confidence protection.

The methods used to confirm the OPΔT trip are now included in the topical report. Previous versions stated that the confirmation had been performed but the methodology used for confirming the trip was not described. This confirmation methodology is used when the coefficients and reset function are known, i.e., when they are provided to SPC or are known from a previous calculation. It takes credit for the protection provided by the overtemperature delta temperature (OTΔT) trip function and main steam safety valves (MSSVs) by excluding operational areas where these actions protect from FCM. The confirmation is performed in two parts. First, the nominal margin is calculated, the difference between the trip and the limit using all variables at their nominal values. Then, the nominal margin is adjusted for uncertainties to obtain the statistically adjusted margin. The statistically adjusted margin between the trip power and the FCM power is verified to be positive with at least a 95 percent probability at a 95 percent confidence level.

Overtemperature Delta Temperature (OTΔT) Reactor Trip Setpoint

The OTΔT reactor trip setpoint provides 95/95 confidence that neither DNB nor hot-leg saturation will occur during normal operation, operational transients, and AOOs. The AOOs that the OTΔT trip protects against are uncontrolled power ascension and a core power redistribution. The form of the OTΔT setpoint trip is the same as used by Westinghouse. This revision does not include a change in the methodology used to determine the trip coefficients.

To confirm core safety limit lines (CSLLs), a set of OTΔT trip coefficients are calculated which bound the CSLLs. This determination is made by plotting the CSLLs as functions of core average temperature and ΔT for each pressure and finding the trip coefficient values that will actuate the OTΔT trip before the CSLLs are reached.

The trip reset function is designed to protect against axial power shapes that are more limiting then design axial power shape. This function is part of the OTΔT trip function. It reduces the value of the trip point to reflect an increase in the hot channel factors which could result in localized DNB. Multiple axial shapes are considered for the determination of the reset function and the axial shapes that yield the minimum departure from nucleate boiling ratio (DNBR) are chosen for inclusion in the calculation. The response surface for the core ΔT most sensitive to uncertainties is determined from deterministic and nominal calculation results. The range of conditions considered in the calculation include those that are within the allowed pressure and are not excluded by the MSSV limit and saturation line curves. The margin at the most sensitive points is determined to define the statistical core ΔT penalty for DNB conditions. This penalty is shown as a ΔT uncertainty adjustment probability table in which each uncertainty parameter is explicitly modeled.

The methods used to confirm OTΔT DNB protection are described in the report. Previous topical reports stated that the confirmation could be performed but they did not provide the methodology for confirming OTΔT. This confirmation methodology is used when the coefficients and reset function are known, i.e., when they are provided to SPC or are known from a previous calculation. This methodology demonstrates that the margin between the power corresponding to DNB or hot-leg saturation and the trip is positive. To begin the calculation, the DNB-limiting axial power shapes are found and reduced to a representative group. Nominal and deterministic power cases corresponding to DNB are used to calculate the most sensitive point. This point is used to develop a probability distribution in power at DNB which includes uncertainties in the radial peaking factor, the engineering factor, and DNBR correlation, and the flow. In calculating the trip function, the uncertainties are all converted to power and combined to create a probability distribution in the trip margin. These uncertainties include the loop temperatures, the reactor coolant system (RCS) pressure, and the ΔI. The trip margin and power at DNB probability distributions are combined to create a probability distribution in the margin. This margin between the power corresponding to DNB or hot-leg saturation and the trip is verified to be positive with at least a 95 percent probability at a 95 percent confidence level and confirms that there is at least a 95/95 confidence of protection from DNB.

Statistical Transient Analysis Methodology

The statistical transient analysis provides 95 percent probability at a 95 percent confidence limit for protecting the specified acceptable fuel design limits (SAFDLs) and pressure limits.

The calculation of the trip setpoint follows a similar calculation path as the OPΔT and OTΔT trip setpoints. Transient analysis is performed using nominal and deterministic values to develop the most sensitive point and the corresponding response surface for the point. This portion is performed using SPC's approved GSUAM methodology (Ref 5). GSUAM is a methodology to statistically combine uncertainties and create response surfaces which are used to determine the probability of conservatively remaining below the limiting parameter. In determining the trip setpoint, the plant specific uncertainties from the trip uncertainty are included in the probability distribution. The combination of the two probability distributions can be performed by either of two methods. The resultant probability distribution is compared to the results of the Monte Carlo run and the most limiting of the two calculations that will prevent transient limit violation with 95 percent probability at a 95 percent confidence level is selected at the technical specification (TS) limit. The statistical trip value is used as the TS limit in cases where the trip is either the OPΔT or OTΔT trip, and the setpoints were calculated based on the setpoint analysis. The trip setpoint calculation is performed using the same methodology for DNB, FCM, and system pressure to determine the 95/95 probability confidence trip setpoint.

When transient analysis involves multiple trips, the probability distributions for each trip can be evaluated independently and the overall probability for the respective parameter of interest (DNB, FCM, or system pressure limit) can be determined. This is shown through probabilistic techniques to provide 95/95 confidence.

The method used to demonstrate that the overall probability distribution difference between the calculated setpoint parameter and the limit will protect the limit follows the same methodology scheme as confirming the OPΔT and OTΔT trips. For DNB, the parameters affecting the transient system behavior and minimum departure from nucleate boiling ration (MDNBR) are varied. The margin is obtained by subtracting the DNBR value that corresponds to DNB from the calculated MDNBR. This margin is verified to be positive with at least a 95/95 confidence and accounts for the uncertainties in the calculations. This methodology to confirm at least a 95/95 confidence between the calculated trip and the limit is performed for DNB, FCM, peak kW/ft and system pressure. In the simplified DNB method, the parameters affecting the transient system behavior are set to their deterministic limit while the parameters for MDNBR calculation are still varied. The simplified FCM margin confirmation is similar although the uncertainty in the peak LHGR is directly calculated and a deterministic approach is used to determine the FCM limit.

Neutronics Analysis

This section has not been revised.

3.0 EVALUATION

The SPC revision to the methodology for the OPΔT and OTΔT trips, and the statistical analysis uses statistical and probabilistic methods that are standard textbook techniques that are applied in a consistent manner. These techniques use standard statistical techniques of combining the uncertainties to create a response surface for determining the probability of remaining below or above the limit value which was previously approved for use by SPC (Ref. 5.) The new techniques that are used, compared to the previously approved methodology, for combining the uncertainties incorporated into the setpoint methodology, are statistically valid applications which allow SPC to automate the methodology. This determination was made by comparing SPC's methods to methods in statistics books and verifying the statistical applications with the NRC statistical expert. The subsets of variables treated statistically were reviewed and determined to be properly treated, combined based on dependence or independence, and incorporated in the methodology. In the confirmation of margin calculation, treating the one variable subset at their conservative deterministic values results in a conservative confirmation of the margin. The new methodology confirms the core safety limit lines, and extends the transient methodology to postulated accidents and events which have no trip, and therefore, adds additional safety verification to the overall methodology. Incorporating protection of the secondary system pressure limit into the transient methodology also adds conservatism.

4.0 CONCLUSIONS

Based on our review, the staff concludes that the proposed topical report is acceptable. This acceptance is subject to the following conditions which SPC agreed to by letter dated December 7, 1999 (Ref. 6):

1. The methodology includes a statistical treatment of specific variables in the analysis; therefore, if additional variables are treated statistically SPC should re-evaluate the methodology and document the changes in the treatment of the variables. The documentation will be maintained by SPC and will be available for NRC audit.

2. The steam generator safety valve (SGSV) limit line provides an upper limit on the temperature range for setpoint verification. The upper limit on the temperature range should be adjusted to reflect the steam generator plugging level.

5.0 REFERENCES

1. Letter from James F. Mallay (SPC) to the U.S. Nuclear Regulatory Commission, submitting Topical Report EMF-92-081(P), Revision 1, "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," December 21, 1998.

2. Siemans Power Corporation Topical Report, EMF-92-081(P), Revision 0 and Supplement 1, "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," February 1994.

3. Exxon Nuclear Methodology for "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," XN-NF-81-58(P)(A), Revision 2 and Supplements 1 and 2, March 1984.

4. Exxon Nuclear Methodology for "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," ANF-81-58(P)(A), Revision 2 and Supplements 3 and 4, June 1990.

5. Exxon Nuclear Methodology for "Generic Statistical Uncertainty Analysis Methodology," XN-NF-22(P)(A), November 1983.

6. Letter from James F. Mallay (SPC) to the U.S. Nuclear Regulatory Commission, Conditions for Topical Report EMF-92-081(P), Revision 1, "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," December 7, 1999.

Principal Contributor: U. Shoop

Date: February 10, 2000