July 12, 2000

Mr. James F. Mallay
Director, Nuclear Regulatory Affairs
Siemens Power Corporation
2101 Horn Rapids Road
Richland, WA 99352-0130

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT EMF-1961(P), REVISION 0, "STATISTICAL SETPOINT/TRANSIENT METHODOLOGY FOR COMBUSTION ENGINEERING TYPE REACTORS" (TAC NO. MA4659)

Dear Mr. Mallay:

The Nuclear Regulatory Commission (NRC) has completed the review of the subject topical report submitted by the Siemens Power Corporation (SPC) by letter dated December 21, 1998. The report is acceptable for referencing in licensing applications to the extent specified, and under the limitations delineated in the report, and in the associated NRC safety evaluation (SE) which is enclosed. The safety evaluation 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 and delay adding it to the Agencywide Documents Access and Management Systems Publicly Available Records System (ADAMS PARS) Library 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.

We do not intend to repeat our 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, we request that SPC publish accepted versions of this report, proprietary and non-proprietary, 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.

Should our criteria or regulations change so that our conclusions about acceptability of the report are invalidated, 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 reports 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 about acceptability of the report are invalidated, 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 reports 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

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* SE from staff dated June 20, 2000, used with minor editorial changes

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT EMF-1961(P), REVISION 0,

"STATISTICAL SETPOINT/TRANSIENT METHODOLOGY

FOR COMBUSTION ENGINEERING TYPE REACTORS"

SIEMENS POWER CORPORATION

1.0 INTRODUCTION

By letter dated December 21, 1998, Siemens Power Corporation (SPC) submitted Topical Report EMF-1961(P), Revision 0, "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors" that described a new methodology for statistical setpoint and transient analysis of Combustion Engineering (CE) type reactors (Reference 1). The methodology includes ways to statistically combine the uncertainties for analyzing limiting conditions of operation (LCOs), limiting safety system settings (LSSSs), and transients. The new methodology uses SPC's previously approved generic statistical uncertainty analysis methodology (GSUAM) (Reference 2); a methodology to statistically combine uncertainties and create response surfaces which are used to determine the probability of conservatively remaining below the limiting parameter. The methodology is based on CE plants with thermal margin/low pressure (TM/LP) LSSS, local power density (LPD) LSSS, LPD LCO, and departure from nucleate boiling (DNB) LCO. The new statistical methodology will facilitate automating the methodology, decreasing the user effect and the potential for introducing user errors.

2.0 METHODOLOGY

The vendor requested approval to use a new statistical setpoint and transient analysis methodology for CE type reactors. The topical report describes how SPC extends their previously approved statistical method (Reference 3) to include additional transients and incorporate new techniques for combining the uncertainties.

2.1 LCOs and LSSSs

The LCOs and LSSSs protect against fuel failure in loss-of-coolant accidents (LOCAs), prevent DNB, and meet specified acceptable fuel design limits (SAFDLs) for fuel centerline melt (FCM). LOCA limits are based on the linear heat generation rate (LHGR) used in the LOCA analysis, DNB limits are based on correlations approved by the NRC, and FCM limits are calculated for each reload cycle and fuel design. In determining the LCO and LSSS limits, a deterministic method with values set at the most limiting conservative values or the statistical method can be utilized.

2.2 LPD LCO and LPD LSSS

The LPD LCO and LSSS protect against FCM. The LPD LSSS monitors the power level of the reactor and trips the reactor when the power level exceeds the setpoint corresponding to the axial shape index (ASI). The LPD LCO prevents the LPD from exceeding the LHGR limit of the LOCA analysis. In developing the limits, both functions use the worst axial power distribution and the technical specification radial power distribution for a given ASI (an augmented radial power distribution is used for cases when the control elements are inserted). To provide additional conservatism to the LPD LCO and LSSS calculations, the most limiting axial power distribution for a given ASI is used in setting the limit.

To calculate the LPD LSSS and LCO, an 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 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 (References 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. Including the method to calculate the FCM limit in the statistical setpoint/transient methodology for CE type reactors is a new addition to the topical report. The FCM limit methodology facilitates automation of the calculation process, reducing the user effect on the calculation results.

The LPD LSSS calculation is performed over a series of axial power shapes. For each shape, the FCM power is calculated. The difference between the trip power and the FCM power is calculated to determine the margin between the two. This margin between the two powers must be positive for all values of ASI considered to confirm that the LPD LSSS protects against FCM with a 95 percent confidence level at a 95 percent probability. During the calculation of the trip power, the uncertainties from measurements and calculational uncertainties are included. These uncertainties are included using standard statistical methods to combine them into the calculated margin. This calculation is an iterative process. If the initial estimate of the LPD LSSS does not provide 95/95 confidence protection, then it is made more restrictive until the power margin provides 95/95 confidence protection.

The LPD LCO is performed in the same manner as the LPD LSSS; however, the LPD LCO prevents the plant from exceeding a reduced LHGR during operation. The LPD LCO is used when the in-core detectors are not in service. Therefore, the LHGR value used for developing the LPD LCO is equal to or less than the LHGR value used in the LOCA analysis.

2.3 TM/LP LSSS and DNB LCO

The TM/LP LSSS and DNB LCO protects against DNB. The TM/LP LSSS provides a reactor trip when the limiting fuel pin approaches DNB and the DNB LCO provides protection from DNB during anticipated operational occurrences. These protective functions are determined using

an iterative process which provides protection with at least a 95 percent probability at a 95 percent confidence level.

The TM/LP LSSS provides protection by tripping the reactor at a pressure which will preclude DNB and hot-leg saturation. Calculating the LSSS begins with the development of axial power shapes for the operating cycle and then the reduction of the number of shapes by a deterministic method, resulting in a set of bounding axial power shapes for each ASI used for setting the limit. This set of axial power shapes is used to perform sensitivity studies to obtain the most sensitive point, the point that shows the greatest change in the calculated pressure. A response surface is developed around the most sensitive point providing the pressure at DNB. The response surface is comprised of all parametric variations statistically combined and includes the appropriate uncertainties. Using the response surface calculation, a table of parametric variation with the corresponding DNB pressure is obtained.

The trip margin is defined as the 95 percent lower limit at a 95 percent confidence level of the difference between the DNB pressure (or hot-leg saturation pressure) to the trip pressure. Demonstrating that the trip margin is positive confirms that there is 95/95 confidence of protection from DNB. Calculation of the margin takes credit for the protection provided by the LPD LSSS and main steam safety valves (MSSVs) by excluding operational areas where these actions provide protection. The confirmation of positive trip margin is performed for the hot-leg saturation and DNB pressures. The confirmation of the hot-leg saturation pressure margin is performed in two steps. First, the nominal margin is calculated. This is defined as the difference between the trip pressure and the saturation pressure corresponding to the temperature points of the vessel exit and inlet. Then, the nominal margin is adjusted for uncertainties to obtain the statistically adjusted margin. Similarly, for the confirmation of DNB pressure, the nominal margin is calculated for the difference between the DNB pressure and the trip pressure and then the margin is statistically adjusted to account for the uncertainties. These statistically adjusted margins are verified to be positive with at least a 95 percent probability at a 95 percent confidence level to demonstrate that they adequately protect against DNB.

The DNB LCO is performed in the same manner as the TM/LP LSSS; however, the DNB LCO iterates on power instead of pressure. Additionally, no credit is taken for the protection provided by the LPD LSSS and MSSVs.

2.4 Transient Analysis Methodology

The statistical transient analysis provides 95 percent probability at a 95 percent confidence limit that the reactor protection system, in conjunction with the LCOs and LSSSs, will ensure that the SAFDLs and pressure limits will not be exceeded.

The calculation of the trip setpoint follows a similar calculation path as that for establishing the LPD, DNB and TM/LP, LSSSs and LCOs. Transient analyses are 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 (Reference 2). 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 response surfaces, the parameter

uncertainties are included in the probability distribution. The trip setpoint is performed using the same methodology for DNB, FCM, and system pressure to demonstrate the 95/95 probability confidence of protection.

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 methods used to confirm margin demonstrate that the overall probability distribution difference between the calculated setpoint parameter and the limit will protect the limit. For DNB, the parameters affecting the transient system behavior and minimum departure from nucleate boiling ratio (MDNBR) are varied. The margin is obtained by subtracting the DNBR value that corresponds to DNB from the calculated MDNBR. This margin is confirmed for 95/95 confidence and accounts for the uncertainties. This confirmation methodology is performed for DNB, FCM, 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.

2.5 Neutronics Analysis

Core average axial power distributions and the corresponding internal and external ASIs are used in the setpoint and transient analysis. These ordered pairs of axial power distributions and ASIs are cycle-specific parameters generated from core simulation techniques. The original core simulation calculations provide the design total peaking and radial peaking factors. Therefore, they are modified to bound reactor operation by using the technical specification limit values.

3.0 EVALUATION

The SPC methodology for the LPD, DNB, and TM/LP, LSSSs and LCOs, uses statistical and probabilistic analytical methods that are standard textbook techniques and are applied in a consistent manner. These methods use standard statistical techniques to combine 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 (Reference 2). The new techniques that are used, compared to the previously approved methodology, for combining the uncertainties incorporated into the setpoint methodology, are statically valid applications which allow SPC to automate the methodology. The staff made this determination 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 extends the transient methodology to postulated accidents and events which have no trip, and therefore, adds additional safety verification to the overall methodology.

4.0 CONCLUSIONS

Based on its 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 March 3, 2000 (Reference 4):

1. This methodology is approved only for CE type reactors which use protection systems as described in the topical report.

2. 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.

5.0 REFERENCES

1. Letter from James F. Mallay (SPC) to the U.S. Nuclear Regulatory Commission, submitting Topical Report EMF-1961(P), Revision 0, "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," December 21, 1998.

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

3. Exxon Nuclear Company Topical Report XN-NF-507(P)(A), Supplements 1 and 2, "ENC Setpoint Methodology for C.E. Reactors: Statistical Setpoint Methodology," September 1986.

4. Letter from James F. Mallay (SPC) to the U.S. Nuclear Regulatory Commission, accepting to the Conditions in Topical Report EMF-1961(P), "SER Conditions for EMF-1961(P), “Statistical Setpoint/ Transient Methodology for Combustion Engineering Type Reactors,"” March 3, 2000.

Principal Contributor: Undine Schoop

Date: July 12, 2000