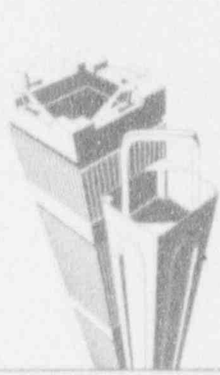
SIEMENS

EMF-92-081(NP)(A)

EMF-92-081(NP)(A)
Supplement 1

Statistical Setpoint/Transient Methodology For Westinghouse Type Reactors

February 1994



Siemens Power Corporation

Nuclear Division

EMF-92-081(NP)(A)

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Supplement 1

Issue Date: 3/1/94

EMF-92-081(NP)(A)	Statistical Setpoint/Transient Methodology For Westinghouse Type Reactors
EMF-92-081(NP)(A) Supplement 1	Correspondence

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 10, 1993

Mr. R. A. Copeland
Manager, Reload Licensing
Siemens Nuclear Power Corporation
2101 Horn Rapids Road
P.O. Box 130
Richland, WA 99352-0130

Dear Mr. Copeland:

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT EMF-92-081, "STATISTICAL SETPOINT/TRANSIENT METHODOLOGY FOR WESTINGHOUSE TYPE REACTORS" (TAC NO. M85061)

The staff has reviewed the topical report submitted by Siemens Nuclear Power Corporation by letter dated May 29, 1992. The report is acceptable for referencing in license applications to the extent specified and under the limitations stated in the enclosed report and U.S. Nuclear Regulatory Commission (NRC) evaluation. The evaluation defines the basis for acceptance of the report.

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 assure that the material presented applies to the specific plant involved. NRC acceptance applies only to the matters described in the report. In accordance with procedures established in NUREG-0390, the NRC requests that Siemens Nuclear Power Corporation publish accepted versions of the 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 and an -A (designating accepted) following the report identification symbol.

If the NRC's criteria or regulations change so that its conclusion that the report is acceptable is invalidated, Siemens Nuclear Power Corporation and/or the applicant referencing the topical report will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the topical report without revision of the respective documentation.

Sincerely,

Ashok C. Thadani, Director
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

Enclosure:
EMF-92-081 Evaluation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATING TO TOPICAL REPORT EMF-92-081(P)
"STATISTICAL SETPOINT/TRANSIENT METHODOLOGY FOR WESTINGHOUSE TYPE REACTORS"
SIEMENS NUCLEAR POWER CORPORATION

1. INTRODUCTION

In a letter of May 29, 1992, from R. A. Copeland to T. E. Murley (NRC), Siemens Nuclear Power Corporation (SNPC) submitted topical report EMF-92-081(P), "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," for NRC review. The report describes the SNPC methodology for performing statistical transient analyses, including the statistical development of the trip protective systems, utilized in Westinghouse-type pressurized water reactors.

The NRC staff was supported in this review by its consultant, Brookhaven National Laboratory. The staff has adopted the findings recommended in the consultant's technical evaluation report (TER) which is attached.

2. EVALUATION

The attached TER provides the evaluation.

3. CONCLUSIONS

The staff has reviewed the SNPC topical report EMF-92-081(P) and the supporting documentation submitted in response to staff requests for additional information. On the basis of this review, the staff concludes that EMF-92-081(P) is acceptable for referencing in licensing actions by SNPC with respect to the statistical setpoint methodology for Westinghouse reactors, subject to the limitations stated in Section 4.0 of the attached TER. In a separate action, the HTP DNB correlation has been approved by the NRC and, therefore, may be used with the SNPC statistical setpoint methodology described in EMF-92-081(P).

TECHNICAL EVALUATION REPORT

TECHNICAL EVALUATION OF THE SNPC STATISTICAL SETPOINT/TRANSIENT METHODOLOGY TOPICAL REPORT EMF-92-081(P)

J. F. Carew

November 20, 1993

Prepared for the
Office of Nuclear Regulatory Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

NRC FIN L-2589-3, Task-1

Reactor Analysis Group
Applied Technologies Division
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TECHNICAL EVALUATION REPORT

Topical Report Title:	SNPC Statistical Setpoint/Transient Methodology
Topical Report Number:	EMF-92-081(P)
Report Issue Date:	May 1992
Originating Organization:	Siemens Nuclear Power Corporation

1.0 INTRODUCTION

By letter dated May 29, 1992 (Reference-1), the Siemens Nuclear Power Corporation (SNPC) has submitted the Statistical Setpoint/Transient Methodology Topical Report - EMF-92-081(P) for NRC review and approval. The topical report provides the methodology that SNPC intends to use in determining the trip protective system settings for Westinghouse (W) type reactors. A detailed description of the methodology and the treatment of setpoint parameter uncertainties together with a sample calculation are included in the Topical Report. The primary difference relative to the presently accepted SNPC methods for Westinghouse reactors is that in the proposed methodology the calculation and measurement uncertainties are treated statistically rather than deterministically. The statistical approach used is similar in many respects to the SNPC statistical setpoint methodology that has been approved for application to Combustion Engineering (CE) plants (Reference-2). The SNPC methodology makes use of response surfaces, together with Monte Carlo sampling techniques and Second-Order Error Propagation (SOERP) methods, to determine appropriate setpoint uncertainty tolerances. The construction and testing of the response surfaces is based, in part, on the approved GSUAM - Generic

Statistical Uncertainty Analysis Methodology (References 3 and 4). The setpoint methods are intended for the determination of the Overpower-ΔT (OPΔT) and Overtemperature-ΔT (OTΔT) setpoints which provide 95/95 protection for the fuel melt and DNBR limits, respectively. The methodology will also be used for analyzing transient events and determining or verifying the reactor protective system trips.

The review of the Topical Report focused on the conservatism included in the determination of the OPΔT and OTΔT setpoint parameters and the statistical method used for determining the 95/95 uncertainty limits. The SNPC methodology is summarized in the following Section-2, and the technical evaluation of the important issues raised during this review is presented in Section-3. The technical position is given in Section-4.

2.0 SUMMARY OF THE TOPICAL REPORT

2.1 Overpower-ΔT Reactor Trip

The Overpower-ΔT reactor trip provides 95/95 protection from fuel melt during operational transients and Anticipated Operational Occurrences (AOOs). The form of the OPΔT trip is the same as used by Westinghouse (Reference-5). The trip includes a constant term with coefficient K_4, and a term proportional to the core-average temperature with coefficient K_6 to account for the effects of coolant density and heat capacity on the relationship between ΔT and the core-average temperature. An additional core-average temperature term with coefficient K_5 is included to account for transient effects such as piping and thermal delays. Compensation for the time dependence of the measured ΔT and core-average temperature is accounted for in the

OPΔT trip equation using the precalculated sets of time-constants (τ_1, τ_2, τ_3) and (τ_6, τ_7), respectively.

The steady-state coefficients K_4 and K_6 are determined so that for the design peaking factor F_Q, the core power level at which fuel centerline melt occurs is avoided. The power peaking measurement uncertainties and the engineering factor are included in the K_4-coefficient using a Monte Carlo procedure and the remaining uncertainties are subtracted directly from the value of K_4.

The calculation of the K_4 and K_6 OPΔT temperature coefficients is based on the design basis peaking-F_Q. In order to account for power distributions more severe than the design peaking, the OPΔT limit is calculated for a range of axial power distributions. A joint probability distribution is then determined in order to allow for uncertainty in both the power level and the axial flux difference-ΔI. The power distribution adjustment to the OPΔT limit - the F(ΔI) reset function -is determined using this joint probability distribution and by requiring that the fuel-melt power level be avoided with 95% probability and 95% confidence.

2.2 Overtemperature-ΔT Trip

The Overtemperature-ΔT trip provides the required protection from DNB and hot-leg saturation during normal operation and Anticipated Operational Occurrences(AOOs). The form of the Overtemperature-ΔT trip is the same as that used by Westinghouse (Reference-5) and includes a constant term K_1, a term proportional to the core-average coolant temperature with coefficient K_2, and a term proportional to the pressurizer pressure with coefficient K_3. The time dependence of the measured ΔT and the core-average temperature are accounted for in the

OTΔT limit using the predetermined sets of time constants (τ_1, τ_2, τ_3) and (τ_4, τ_5, τ_6), respectively. The OTΔT steady-state coefficients K_1, K_2 and K_3 are determined to provide DNB and hot-leg saturation protection at the 95/95 probability/confidence-level. The DNB uncertainty margin is calculated using a response surface for ΔT. A similar analysis is performed for the hot-leg saturation ΔT setpoint. The response surfaces are determined using the SNP approved GSUAM procedures. The statistical uncertainty penalty is determined using a Monte Carlo approach, rather than the SOERP method, and includes the response surface parameter uncertainty as well as the trip input temperature and pressure measurement uncertainties.

In order to provide DNB protection for axial power shapes more severe than the design axial distribution, an F(ΔI) reset trip adjustment is included in the OTΔT trip function. The first step in determining the reset function is to identify the DNB limiting axial power distribution for each ΔI interval. A ΔT response surface is then generated for this axial distribution at the conditions determined to be most sensitive to the uncertainties. The resulting ΔT distribution is then used to determine the joint probability distribution. The F(ΔI) adjustment is calculated from this joint probability function.

2.3 Neutronics Analysis

Both the OPΔT limit for fuel melt and the OTΔT limit for DNB protection depend on the core axial power distribution. This dependence is included in the trip functions via the F(ΔI) reset function. The relationship between the local peaking (F_Q and $F_{\Delta H}$) and ΔI are determined using a three-dimensional neutronics model. The axial flux difference and power distribution are calculated as a function of core operating conditions.

3.0 SUMMARY OF THE TECHNICAL EVALUATION

The Topical Report EMF-92-081(P) provides a detailed description of the statistical methodology that SNPC intends to use for the setpoint determination for Westinghouse type reactors. The review focused on the applicability and conservatism of the methods used for calculating the trip parameters (K_1-K_6), and the validity of the statistical approach employed in determining the allowance for design and measurement uncertainties. Several important technical issues were identified during the initial review which required additional information and clarification from SNPC. This information was requested in Reference-6 and was provided in the SNPC response included in References 7-9. This evaluation is based on the description and examples presented in the topical report and the supporting information provided in References 7-9. The evaluation of the major issues raised during this review are summarized in the following.

3.1 Setpoint Methodology

3.1.1 Transient Effects

The time-dependent effects associated with instrument delay and piping lag are accounted for by the K_5-transient coefficient in the OPΔT trip setpoint equation. These effects are considered to be hardware related and, in Response-17 of Reference-7, SNPC has indicated that this parameter will not be modified with the application of the EMF-92-081(P) Methodology. The OPΔT trip is designed to protect against slowly evolving transients in which there is no significant fuel temperature overshoot. In Response-3 of Reference-7, SNPC has indicated that it will verify the adequacy of this term and the protection of the fuel centerline temperature from fuel melt by performing event specific transient analyses.

The time-dependent processing and compensation of the OPΔT and OTΔT trip input signals is performed using the seven time constants τ_1 - τ_7. These plant-specific equipment-related constants will be provided from information supplied by the reactor vendor. These time constants will not be recalculated as part of the SNPC setpoint methodology, however, the adequacy of the values will be confirmed by SNPC in plant-specific transient analyses (Response-19, Reference-7).

3.1.2 Response Surface Methods

The statistical uncertainty analyses for the DNB and hot-leg saturation limit lines and the statistical transient analyses are performed using response surfaces generated with the SNPC GSUAM methodology. The response surfaces are typically low order polynomials in the independent variables and the specific functional forms are selected to provide the required accuracy. Specifically, the selection is based on the fitting statistics, residual plots and the overall goodness of the fit. In Response-9, SNPC has indicated that the response surface fitting error is determined and included in the Monte Carlo uncertainty analysis.

The methods used to construct the response surface are generally conservative. However, the SNPC methodology response surface selection criteria does not in general result in the base-point having the maximum statistical uncertainty. In Response-8 of Reference-7 and Response-1 of Reference-9, however, SNPC demonstrates that for the case of the DNB and hot-leg saturation lines and the Chapter-4 DNB transients this selection criteria does result in the maximum statistical uncertainty and is therefore acceptable for these applications. However, the validity of this criteria is based on the specific selection of uncertainty variables and the assumed uncertainty estimates. If additional variables are added to the DNB and/or hot-leg saturation response surfaces or the uncertainty estimates change, this criteria should be reevaluated.

The proposed response surface selection criteria is not applied to the Chapter-4 low flow and pressurization transients. In Response-2 of Reference-8, SNPC has indicated that for these transients the analyses will be performed at the operating conditions which result in the most conservative transient setpoints.

3.1.3 Axial Shape Reset Function - F(ΔI)

The axial shape reset function F(ΔI) provides the reduction in the ΔT trips to account for axial power distributions more severe than the design peaking. In Response-12 of Reference-7, SNPC has indicated that the limiting conditions of operation have been included in the determination of F(ΔI). Limiting core power level and axial power shape combinations are selected. Severe axial xenon distributions are used to determine the axial power distribution as a function of fuel rod burnup, power level and control rod insertion.

The axial power distributions used to determine the DNB OTΔT F(ΔI) penalty for axial power shape depend on the core power level. The most DNB limiting axial power shape is selected and provides a bounding F(ΔI) penalty function.

3.1.4 DNB Quality Limit

The DNB correlations are typically only applicable below a specific upper quality limit. The SNPC HTP DNB correlation is limited to qualities below the HTP quality limit while the application of the W-3 correlation is limited to qualities less than 15%. While the hot-leg saturation limit line will typically limit the exit quality to less than the HTP limit, this may not be the case for other DNB correlations. However, SNPC has indicated in Response-2 (Reference-7) that the XCOBRA-IIIC code used to calculate the DNB OTΔT limit line provides a specific edit to insure that the DNB correlation is within the applicable quality limit.

3.1.5 Steam Generator Safety Valve Limit Line

The Steam Generator Safety Valve (SGSV) limit line provides the lower limit on the DNB ΔT setpoint at high core average temperatures. If the steam generator tube plugging levels are unchanged SNPC will use the vendor SGSV limit line. However, if tube plugging levels increase, SNPC should adjust the SGSV line as described in Response-16 of Reference-7.

3.2 Application of the Methodology

3.2.1 Fuel and Plant Designs

The setpoint methodology makes specific assumptions concerning the plant configuration and core design. The proposed methods are applicable to Westinghouse plants which have OPΔT and OTΔT protective system trips of the form given in Sections 2.1 and 3.1, respectively, of the topical report. The methodology is applicable to fuel designs for which SPNC has NRC-approved methods for evaluating DNB and fuel melt.

In Response-1 of Reference-7, SNPC notes three potential simplifications of the application of the statistical methodology. First, in plant-specific applications certain variables may be treated conservatively by taking them at their deterministic limits rather than treating them statistically. Second, certain licensing applications will only require the verification of a prior vendors setpoints. In this case, the setpoint coefficients K_i will not be recalculated but rather an independent verification of the setpoints will be performed. Finally, for certain plants the total uncertainty allowance associated with the trip channel may be provided by the reactor vendor. In this case, the individual trip channel component uncertainties will be replaced by a single statistical total uncertainty allowance.

3.2.2 Setpoint and Event Applications

The setpoint/transient analysis methodology is applicable to the setpoint determination for specific reactor trips. The steady-state methods of Chapters 2 and 3 are applicable to the determination of the OPΔT and OTΔT trip setpoints. The transient analysis methods of Chapter-4 are intended for the analysis of limiting DNB and pressurization events, and the determination of the OPΔT, OTΔT and other specified trip setpoints. The transient analysis methods of Chapter-4 may be used to either determine or verify these setpoints (Response-18, Reference-7).

3.2.3 Code and Methods Approvals

The proposed setpoint methodology employs several SNPC core performance codes/correlations including RODEX2, XCOBRA-IIIC, XTG, and the XNB and HTP DNB correlations. In Response-11 of Reference-7, SNPC has indicated that, except for the HTP DNB correlation, all of these methods have been approved. The HTP DNB correlation is presently under review and it must receive NRC approval before it may be used with the statistical setpoint methodology.

3.2.4 Allowance for Uncertainties

The EMF-92-081(P) methodology performs a statistical determination of the 95/95 probability/confidence-level uncertainty allowance for the Reactor Protection System trip setpoints. This analysis requires plant-specific uncertainty estimates for the important measurement uncertainty components. In Response-6 of Reference-7, SNPC has indicated that the uncertainty values for the average coolant temperature, core power level, pressurizer pressure, bistable, channel linearity and reproductibility, and axial flux difference measurement will be provided by the reactor vendor. The plant-specific power peaking uncertainties used will be based on the Technical Specification values.

4.0 TECHNICAL POSITION

The Siemens Nuclear Power Corporation Statistical Setpoint/Transient Methodology Tropical Report EMF-92-081(P) and supporting documentation provided in References 7 and 8 have been reviewed in detail. Based on this review, it is concluded that the SNPC methodology is acceptable for determining the Reactor Protective System setpoints for Westinghouse type reactors subject to the conditions stated in Section-3 of this evaluation and summarized in the following.

1) DNB and Hot-leg Saturation Response Surfaces

The validity of the maximum arithmetic difference criteria for determining the most conservative DNB and hot-leg saturation response surfaces is based on the specific selection of uncertainty variables and their estimated uncertainties. If additional uncertainty variables are added to the DNB and/or hot-leg saturation response surfaces or the uncertainty estimates change, this criteria should be reevaluated (Section-3.1.2).

2) Low Flow and Pressurization Transient Response Surfaces

The response surface base-point selection criteria does not determine the maximum statistical uncertainty for the low flow and pressurization transients. For these transients the base-point should be selected at the operating conditions which result in the most conservative transient setpoints (Section-3.1.2).

3) <u>Steam Generator Tube Plugging</u>

The steam generator safety valve line provides the lower limit on the DNB ΔT setpoint at high core average temperatures. If steam generator tube plugging levels increase the SGSV line should be adjusted (Section-3.1.5).

4) <u>HTP DNB Correlation</u>

The HTP DNB correlation is presently under review and it must receive NRC approval before it may be used with the statistical setpoint methodology (Section-3.2.3).

REFERENCES

1. "EMF-92-081(P) Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors, Siemens Nuclear Power Corporation, May 1992," Letter, R.A. Copeland (SNPC) to Thomas E. Murley (NRC), May 29, 1992.

2. "ENC Setpoint Methodology for CE Reactors: Statistical Setpoint Methodology," XN-NF-507(P) (A), Supplements 1 and 2, Exxon Nuclear Company, Richland, WA 99352, September, 1986.

3. "Generic Statistical Uncertainty Analysis Methodology," XN-NF-81-22(P) (A), Exxon Nuclear Company, Richland, WA 99352, November 1983.

4. "Expanded Generic Statistical Uncertainty Analysis Methodology," XN-NF-507(P) (A), Supplement 1, Appendix A, Exxon Nuclear Company, Richland, WA 99352, September, 1986.

5. "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," WCAP-8745(P), Westinghouse Electric Corporation, Pittsburgh, PA 15230, March 1977.

6. "Request for Additional Information EMF-92-081(P), Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," Letter, R.C. Jones (USNRC) to R.A. Copelend (SPC), July 16, 1993.

7. "Responses to NRC Questions on SPC Statistical Transient Methodology," Letter, R.A. Copeland (SPC) to R.C. Jones (NRC), August 13, 1993.

8. "Responses on the SPC Statistical Transient Methodology," Letter, R.A. Copeland (SPC) to L. Kopp (NRC), September 24, 1993.

9. "Response to Question on EMF-92-081(P)," Letter, R.A. Copeland (SPC) to L. Kopp (NRC), October 5, 1993.

Siemens Nuclear Power Corporation

EMF-92-081(NP)(A)

STATISTICAL SETPOINT/TRANSIENT METHODOLOGY

FOR WESTINGHOUSE TYPE REACTORS

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May 1992

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1.0 INTRODUCTION

This report describes Siemens Nuclear Power Corporation's (SNP) methodology for statistical transient analyses, including the statistical development of the trip protective systems, utilized in Westinghouse type pressurized water reactors (PWRs). The methods described in this report are conceptually the same as those in SNP's approved statistical setpoint methodology for Combustion Engineering plants[1]. The primary differences are in the forms of the trip equations and the NSSS vendor specific terminology. The method for statistically combining uncertainties is that presented in SNP's approved Generic Statistical Uncertainty Analysis Methodology[9,11] (GSUAM).

Section 2.0 of this report provides the bases and methods used to develop the overpower ΔT (OPΔT) trip setpoint function. Section 3.0 gives the bases and methods used to develop the overtemperature ΔT (OTΔT) trip setpoint function. Section 4.0 provides the methodology utilized in the statistical transient analyses. Section 5.0 describes the methods used to generate the core power distributions employed in the verification of the $F(\Delta I)$ trip reset functions. Section 6 contains an example of the statistical setpoint/transient methodology as applied to a typical three-loop Westinghouse plant.

2.0 OVERPOWER ΔT REACTOR TRIP SETPOINT

The OPΔT is designed to protect the Specified Acceptable Fuel Design Limit (SAFDL) on fuel centerline melt during normal operation, operational transients and AOOs, such that with 95% probability and 95% confidence no location in the core experiences fuel centerline melting. AOOs which may challenge the SAFDL are those which result in an uncontrolled power ascension or a drastic core power redistribution. Typical events to be considered are:

- Uncontrolled RCCA Withdrawal
- Uncontrolled Boron Dilution
- Increased Feedwater Flow
- Decreased Feedwater Heating
- Excessive Load Increase

2.1 Description of the Overpower ΔT Trip Function

The general format of the OPΔT setpoint is:

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_0 \left(K_4 - K_5 \left(\frac{\tau_7 S}{1+\tau_7 S}\right)\left(\frac{1}{1+\tau_6 S}\right) T - K_6 \left(T\left(\frac{1}{1+\tau_6 S}\right) - T'\right) - F(\Delta I)\right) \tag{2.1}$$

where:

ΔT	=	measured ΔT, F-degrees;
ΔT_0	=	indicated ΔT at rated thermal power, F-degrees;
T	=	indicated average reactor coolant temperature, °F
T'	=	indicated average reactor coolant temperature at rated thermal power, °F
K_4	=	a preset, manually adjustable bias
K_5	=	a constant that compensates for piping and thermal delays, 1/°F;
K_6	=	a constant that accounts for the effects of coolant density and heat capacity on the relationship between ΔT and thermal power, 1/°F;
S	=	Laplace transform operator, 1/sec;
$(1+\tau_1{*}S)/(1+\tau_2{*}S)$	=	Lead-lag compensator on measured ΔT
$1/(1+\tau_3{*}S)$	=	Lag compensator on measured ΔT
$1/(1+\tau_6{*}S)$	=	Lag compensator on measured average coolant temperature
$\tau_7 S/(1+\tau_7{*}S)$	=	the function generated by the rate-lag controller for average coolant temperature dynamic compensation
$F(\Delta I)$	=	a function of the indicated flux difference between the top and bottom detectors of the power range nuclear ion chambers (ΔI)

The excore neutron flux detectors measure the relative powers in the top (P_t) and bottom (P_b) of the core such that ΔI is defined as (P_t - P_b). The relative powers also define the axial offset (AO):

$$AO = \frac{P_t - P_b}{P_t + P_b} \tag{2.2}$$

The axial offset is a relative measure of the skewness of the axial power distribution.

2.2 Overpower ΔT Setpoint Calculation Overview

The calculational procedure for generating the OPΔT setpoint equation consists of three steps:

1) A fuel centerline melt limit is determined which accounts for the dynamic effects not compensated for by the K_5 transient term or the lead-lag compensator on measured ΔT. A discussion of the criterion and methods used in determining this limit is presented in Section 2.3.
2) The OPΔT K_4 and K_6 trip coefficients are determined which prevent the plant from violating the determined fuel centerline melt limit with 95% probability and 95% confidence. A specified design F_Q is used in the determination of these coefficients. Section 2.4 presents the method with which these coefficients are determined.
3) The F(ΔI) reset trip function is determined. The F(ΔI) function addresses the impact on the core protection limits from core power distributions more adverse than the design F_Q used in the generation of the K_4 and K_6 trip coefficients. The procedure for evaluating the F(ΔI) trip reset function is described in Section 2.5.

The process for determining the OPΔT trip equation is very similar to SNP's approved Combustion Engineering plant statistical LPD analysis methodology[1].

2.3 Overpower ΔT Transient Compensation and Fuel Melt Limit Determination

The first step in the statistical development of the OPΔT trip equation is the determination of a fuel centerline melt limit which accounts for the dynamic effects not compensated for by the K_5 transient term and the lead-lag compensator on measured ΔT. The OPΔT K_4 and K_6 trip coefficients, and the F(ΔI) function are all generated in a manner which assumes

In a transient event, due to trip processing and scram time delays, the peak power reached will be above the power level at which trip occurs. To cover this transient over-shoot, There are two transient components in the OPΔT setpoint. There is the lead-lag compensator which adjusts the measured ΔT and there is the K_5 rate-lag controller term, which adjusts for the dynamic variation of the average coolant temperature. The K_5 coefficient and the time constant values are plant specific and must be chosen to reduce sensor noise and to provide protection to the centerline melt SAFDL during transients. In general, the K_5 trip coefficient and the time constants will not change due to changes in fuel design or changes in fuel operating limits. This is because these terms depend primarily upon sensor, signal processing and loop transport delays which are system and not core related.

The K_5 transient compensator is designed to adjust the OPΔT trip for transient events in which the average coolant temperature is increasing. That is, events in which power is increasing. The adequacy of the compensator is dependent on the range of power ascension rates that can occur in a given plant. To determine the transient over-shoot, a series of power excursion events are modeled using an approved transient simulation code. A temperature adjustment is determined which bounds that of any AOO which could result in the occurrence of fuel melt. A transient compensated fuel melt temperature is determined by adjusting an approved fuel melt temperature curve by a bounding transient compensation adjustment.

To determine the temperature of the various fuel rods contained in the core, each of the limiting fuel rod types are analyzed with an approved fuel rod model. These analyses determine the temperature of a given rod as a function of burnup and power level. The resultant temperatures are then compared with a transient compensated fuel melt temperature to determine the respective fuel melt power level as a function of burnup for each of the various fuel types.

Over the cycle of consideration, the determined fuel melt power levels and respective assembly peaking factors are compared to determine an LHGR limit for the peak F_Q location in the core. This limit is set so that when the peak F_Q location in the core is at or below this power level, no location in the core will undergo fuel melt. The assembly peaking factors for the cycle of concern are generated with an approved three-dimensional neutronics code.

2.4 Overpower ΔT Trip Coefficient Generation

The which need to be accounted for have been incorporated into the fuel centerline melt limit. By removing the transient terms and the F(ΔI) reset trip function, which is addressed in Section 2.5, the OPΔT setpoint equation can be simplified to:

$$\Delta T \leq \Delta T_0 \left(K4 - K6 \left(T - T' \right) \right) \qquad (2.3)$$

Using this equation, a set of K_4 and K_6 trip coefficients is determined which will prevent the core power level at which centerline melt may occur from being obtained. Then, K_4 is adjusted to result in the Technical Specification setpoint equation.

Part A in equation 2.5 represents the nominal results while part B represents the uncertainties.

The resultant factor is multiplied onto (A) to get the core protection limit which the OPΔT trip will be shown to protect.

The steady-state form of the OPΔT setpoint with the F(ΔI) reset trip function removed was presented in equation 2.3. This can be rearranged to be:

$$\frac{\Delta T}{\Delta T_0} \leq K4 - K6\,(T - T') \tag{2.6}$$

The allowable K_4 constant is closely related to the core protection limit power for centerline melt, equalling the ratio of the vessel ΔT at the protection limit to the vessel ΔT at the nominal rated power operating point. This ratio has a small pressure dependence. Because a smaller value of K_4 provides more margin between the fuel centerline temperature at trip conditions and the temperature corresponding to the SAFDL on centerline melt, K_4 is conservatively defined by:

The value of to occur within the pressure range delimited by the uncertainty adjusted high and low pressurizer pressure trip setpoints. Due to the physical variation of the specific heat of water with pressure, this will occur at the minimum allowable pressure.

The K_6 term of the OPΔT equation accounts for the effects of the increasing vessel average temperature on the relationship between vessel ΔT and core power. At constant pressure and power, an increased vessel average temperature results in a smaller vessel ΔT due to the increase in specific heat of water with temperature. Therefore, the OPΔT setpoint must be reduced at average temperatures above the nominal average temperature. The value of K_6 is developed using the following equation:

To assure that the OPΔT setpoint is equation is conservative, the K_6 coefficient must be maximized.

These generated K_4 and K_6 coefficients result in the OPΔT trip equation. The uncertainty adjusted trip, which goes into the plant Technical Specifications, is determined by A list of typical uncertainty parameters included in the OPΔT setpoint analysis is presented in Table 2.1. All of these uncertainties, with the exception of

To determine the statistical adjustment, a Monte Carlo analysis is performed in which the assumed distribution of each parameter is modeled explicitly. The 95/95 bounding value is the resultant Technical Specifications value for the plant. The distributions assumed for each parameter are From the central limit theorem the resultant distribution should be approximately normal, whether the individual distributions are assumed to be normal or not.

2.5 Overpower ΔT F(ΔI) Trip Reset Function

The OPΔT were developed so as to prevent fuel centerline melt from occurring with 95% probability and 95% confidence with the power peaking at the design F_Q.

The OPΔT F(ΔI) trip reset function is designed to account for the effects of core power distributions which result in F_Q's more adverse than the design F_Q.

The value of ΔI for a given core configuration is calculated from:

$$\Delta I = AO * P \tag{2.9}$$

where:

where:	ΔI	=	axial flux difference
	AO	=	axial offset
	P	=	a conservatively low power fraction at which melt could occur for a respective axial distribution and associated F_Q

The value of the AO employed is that calculated from the core average axial power distribution F(Z) and thus represents the core average axial offset. The resultant ΔI value represents the minimum indicated ΔI at which fuel melt can take place. A description of the method used to generate the axial distributions utilized in the OPΔT F(ΔI) analysis is presented in Section 5.

The steps involved in the statistical development of the OPΔT F(ΔI) reset trip function are:

- Determination of the limiting nominal fuel melt power levels as a function of ΔI (P_{NOM})

- Calculation of required F(ΔI) corrections given core ΔT's of melt

A discussion of these steps employed in the OPΔT F(ΔI) reset function statistical analysis follows.

For each of the generated axial distributions, the fuel centerline melt power level is determined A great number of the generated axial distributions have approximately the same ΔI. For a given small ΔI range, the axial distribution producing the lowest fuel melt power level is determined and retained.

The nominal power level at which fuel melt is precluded for a given ΔI range is specified by:

Accounting for uncertainties, the core power level at which fuel melt is precluded occur is specified by the equation:

This equation and the fuel melt equation for the design F_Q, equation 2.4, are effectively the same.

The core power level of fuel melt is determined accounting for uncertainties. These power levels are identified as P_{DET} values. The same uncertainties considered in the generation of assumed here. Those axial distributions having P_{DET} values greater than the design F_Q core power limit

Around the P_{NOM} values power probability distribution tables are generated. In the statistical calculation of the uncertainty adjusted allowed power level, equation 2.11, the parameters To determine the probability density of the multiplicative factor on P_{NOM} and bias factors, a Monte Carlo simulation is performed. In this calculation the distribution of each of the respective uncertainty parameters is explicitly modeled.

A joint probability curve is then generated for the power dependent probability and the uncertainty distribution of ΔI. Mathematically this can be expressed as:

$$J(F_{\Delta I}, F_{POWER}) = 0.05 \qquad (2.12)$$

where J is the joint probability distribution. The result is the power level as a function of ΔI at which fuel melt is prevented with a 95% probability and 95% confidence.

Each of the P_{NOM} points results in a series of 95/95 power levels and ΔI's. The most limiting of all power levels as a function of ΔI are identified. These respective power levels, identified as P_{UNC} values, are the power levels which the OPΔT trip and respective F(ΔI) reset must prevent from occurring.

The OPΔT trip equation is in units of core ΔT. Therefore, to make the units consistent, the fuel melt core ΔT's are calculated for each of the respective P_{UNC} power levels. In this calculation

The static form of the OPΔT setpoint equation (Eq. 2.1) is:

$$\Delta T \le \Delta T_0 \left(K4 - K6\,(T - T') - F(\Delta I)\right) \tag{2.13}$$

For the purposes of determining the required F(ΔI) reset trip function this equation can be reduced to:

$$\Delta T \le \Delta T_0 \left(K4 - F(\Delta I)\right) \tag{2.14}$$

In operation, the core ΔT must remain below the right hand side of the side of the equation. Translating this into the prevention of fuel melt, the ΔT of fuel melt, ΔT_{FM}, must remain above this right hand side, or:

$$\Delta T_{FM} \ge \Delta T_0 \left(K4 - F(\Delta I)\right) \tag{2.15}$$

rearranging this equation results:

$$F(\Delta I) \ge K4 - \frac{\Delta T_{FM}}{\Delta T_0} \tag{2.16}$$

The required F(ΔI) correction is calculated for each of the ΔT_{FM} values. A bounding F(ΔI) correction is determined to prevent centerline melt for F_Q's more severe than the design F_Q.

A sample plot of a bounding OPΔT F(ΔI) function is presented in Figure 2.1.

Table 2.1: Typical Uncertainties Applied in OPΔT Setpoint Analysis

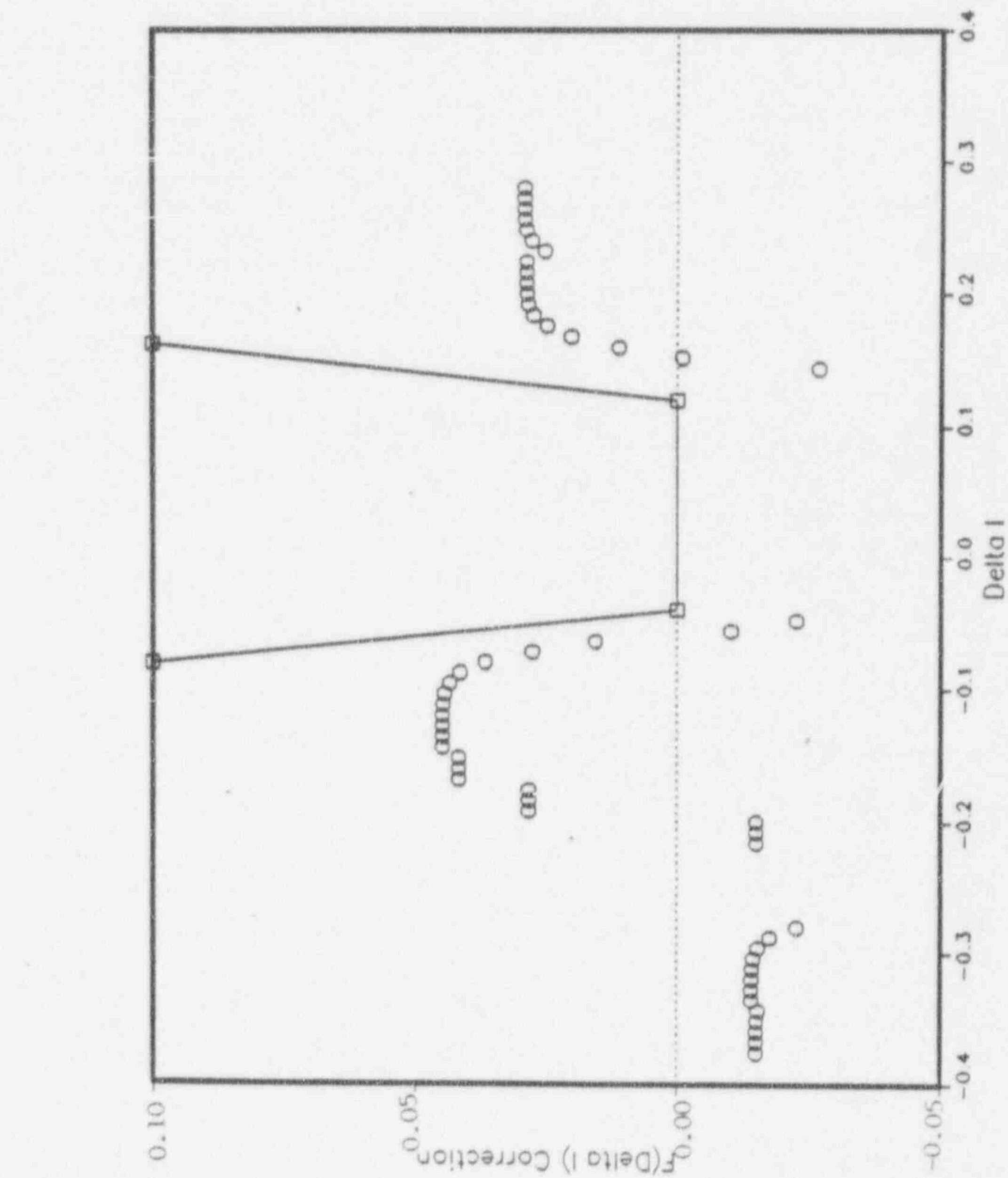

Figure 2.1: Sample OPΔT F(ΔI) Function

3.0 OVERTEMPERATURE ΔT REACTOR TRIP SETPOINT

The OTΔT trip is designed to protect the SAFDL on DNB and to prevent boiling in the vessel hot legs during normal operation, operational transients and AOOs. The trip is set such that with 95% probability and 95% confidence, neither DNB nor hot leg saturation will occur. The prevention of boiling in the hot legs permits the measured temperature differences across the vessel to be used to calculate power for use in the protection system. AOOs which may challenge the OTΔT trip are those which result in an uncontrolled power ascension or a drastic core power redistribution. Typical events to be considered are:

- Uncontrolled RCCA Withdrawal
- Uncontrolled Boron Dilution
- Increased Feedwater Flow
- Decreased Feedwater Heating
- Excessive Load Increase

The onset of DNB in the core or boiling in the hot legs is dependent upon the average coolant temperature, the primary system pressure and the core thermal power.

3.1 Description of the Overtemperature ΔT Trip Function

The general format of the OTΔT trip setpoint is:

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_0 \left(K_1 - K_2 \left(\frac{1+\tau_4 S}{1+\tau_5 S}\right) \left(T\left(\frac{1}{1+\tau_6 S}\right) - T'\right) + K_3 (P - P') - F(\Delta I)\right) \quad (3.1)$$

where:

ΔT	=	measured ΔT, F-degrees
ΔT_0	=	indicated ΔT at rated thermal power, F-degrees
T	=	indicated average reactor coolant temperature, °F
T'	=	indicated average reactor coolant temperature at rated thermal power, °F
P	=	pressurizer pressure, psig
P'	=	nominal pressurizer pressure at rated conditions, psig
K_1	=	a preset, manually adjustable bias
K_2	=	a constant that compensates for piping and thermal delays and for changes in design limits with temperature, 1/°F
K_3	=	a constant that compensates for changes in design limits with pressure, 1/psi
S	=	Laplace transform operator, 1/sec
$(1+\tau_1{*}S)/(1+\tau_2{*}S)$	=	Lead-lag compensator on measured ΔT
$1/(1+\tau_3{*}S)$	=	Lag compensator on measured ΔT
$(1+\tau_4{*}S)/(1+\tau_5{*}S)$	=	Lead-lag compensator on average coolant temperature
$1/(1+\tau_6{*}S)$	=	Lag compensator on measured average coolant temperature
$F(\Delta I)$	=	a function of the indicated flux difference between the top and bottom detectors of the power range nuclear ion chambers (ΔI)

3.2 Overtemperature ΔT Setpoint Calculation Overview

There are two main parts to the statistical development of the OTΔT setpoint equation. There is the generation of the OTΔT trip coefficients, K1, K2 and K3, and there is the determination of the F(ΔI) reset trip function. The OTΔT setpoint fulfills many of the same protective functions as does the thermal margin/low pressure (TM/LP) trip setpoint in Combustion Engineering type plants. In consequence, by the nature of the conditions being prevented, the employed OTΔT calculational processes are very similar to those presented in SNP's approved statistical TM/LP setpoint procedure[1].

These results are the basis for determining the adequacy of the desired trip function. The methodology for the statistical determination of the OTΔT trip coefficients K1, K2, and K3 is presented in Section 3.3. The statistical OTΔT F(ΔI) methodology is presented in Section 3.4.

All of the statistical OTΔT setpoint calculations are performed The transient compensator terms of the equation are verified in the performance of the Standard Review Plan Chapter 15 transient analyses. A discussion of the Westinghouse type plant statistical transient analysis methodology is presented in Section 4.

3.3 Overtemperature ΔT Trip Coefficient Generation

This section will present the methodology for the statistical calculation of the OTΔT K_1, K_2 and K_3 trip coefficients. The steady-state form of the OTΔT setpoint equation (eq. 3.1) is:

$$\Delta T \leq \Delta T_0 \left(K_1 - K_2 (T - T') + K_3 (P - P') - F(\Delta I)\right) \qquad (3.2)$$

By removing the F(ΔI) reset trip function, which is addressed in Section 3.4, this equation can be further reduced to:

$$\Delta T \leq \Delta T_0 (K_1 - K_2 (T - T') + K_3 (P - P')) \qquad (3.3)$$

The calculational process for the OTΔT setpoint is:

- Determine average coolant temperature and core ΔT's resulting in the target MDNBR and/or hot leg saturation assuming nominal parameter values (NOMSCAN)

- Calculate acceptable OTΔT trip coefficients for statistically adjusted conditions

A more detailed description of these calculational processes follows.

In the NOMSCAN portion of the OTΔT statistical setpoint development procedure the average coolant temperature and core ΔT are varied

The range of average coolant temperatures and core ΔT's to be considered is set by the plant's OPΔT setpoint equation line and by a limit line representing the Steam Generator Safety Valve (SGSV).

Those parameters which

are not going to be treated statistically are set to their respective deterministic values. Table 3.1 provides a list of typical parameters considered statistically by the OTΔT analysis. Figure 3.1 presents a graphical representation of the results of a typical NOMSCAN analysis.

In the NOMSCAN calculation all input parameters

The sufficiency of this axial distribution relative to the setting of the ΔI "dead-band" is verified in the OTΔT F(ΔI) analysis (Section 3.4).

The uncertainty variation methods for the generation of the response surface are given in SNP's approved GSUAM methodology[2,3].

If it is desirable and justifiable a number of different DNB ΔT response surfaces can be generated and applied over respectively bounding spaces (e.g., high pressure and low pressure). The purpose of this modification would be to apply more representative yet conservative uncertainty adjustments.

To determine the statistical core ΔT penalty for DNB conditions, a Monte Carlo calculation is performed. The Monte Carlo calculation is performed using equation 3.4, which contains the response surface

In this series of Monte Carlo calculations, the parameters of the response surface,

The 95/95

minimum value of S is the objective statistical DNB ΔT adjustment.

The hot leg saturation DETSCAN calculation

To determine the statistical hot leg saturation core ΔT penalty, a Monte Carlo calculation is performed about the hot leg saturation conditions

This hot leg statistical penalty determination differs from that performed for DNB only in that the response surface are different. The hot leg saturation Monte Carlo calculation is performed on the equation:

The 95/95 minimum value of S is the objective statistical hot leg saturation ΔT adjustment.

The statistical DNB and hot leg saturation ΔT adjustments are applied

With the uncertainty adjusted points a set of limit lines are determined. These adjusted limit lines specify the conditions at which DNB and hot leg saturation are prevented with at least 95% probability and 95% confidence.

A set of OTΔT trip coefficients is determined which will bound the uncertainty adjusted safety limit lines. These coefficients are subsequently compared with the trip coefficients supported by the statistical transient analyses (Section 4.0).

3.4 Overtemperature ΔT F(ΔI) Verification

The OTΔT setpoint equation (Eq. 3.1) is developed so as to prevent the occurrence of DNB in the core with 95% probability and 95% confidence. The purpose of the F(ΔI) reset trip function is to compensate for axial distributions which are more limiting than the design axial distribution used in the development of the OTΔT K_1, K_2, and K_3 trip coefficients. As in the statistical development of the OTΔT trip coefficients, the statistical OTΔT F(ΔI) process is very similar to SNP's approved Combustion Engineering statistical TM/LP setpoint analysis[1].

The process for evaluating the OTΔT F(ΔI) reset trip function has many of the same components as the DNB portions of the OTΔT base calculation. The steps are:

- Identify the axial distributions most limiting with respect to DNB as a function of ΔI (AXSCAN)

- Convert the required trip ΔT adjustments into trip units to establish the F(ΔI) reset trip func ion

A description of these calculational processes follows.

The axial distributions which are used in the evaluation of the OTΔT F(ΔI) reset trip function are those axials which result in

The procedure for the generation of the axial distributions is presented in Section 5.0.

In the NOMSCAN portion of the statistical OTΔT F(ΔI) reset trip function procedure,

In the DETSCAN portion of the statistical OTΔT F(ΔI) methodology, a response surface of core ΔT is determined. The process used to generate this response surface is very similar to that used in the generation of the base OTΔT DNB response surface.

The uncertainty variation methods for the generation of the response surface are given in SNP's approved GSUAM methodology[2,3]. A list of typical uncertainty parameters considered in the OTΔT F(ΔI) verification is presented in Table 3.1.

If so desired, and justifiable,

The purpose of this would be to apply more representative yet conservative uncertainty adjustments.

To determine the core ΔT uncertainty probability distribution, a Monte Carlo calculation is performed

The result of the Monte Carlo calculation is a ΔT uncertainty adjustment probability table. In this calculation the distributions of each of the respective uncertainty parameters is explicitly modeled.

A joint probability curve is then generated for the ΔT adjustment probability table and the uncertainty distribution of ΔI. Mathematically this can be expressed as:

$$J(F_{\Delta T}, F_{\Delta I}) = 0.05 \qquad (3.7)$$

where:

J	=	joint probability distribution
$F_{\Delta T}$	=	probability of given ΔT adjustment
$F_{\Delta I}$	=	probability of given ΔI adjustment

The joint probability curves are applied to the respective NOMSCAN files to determine the core ΔT's and ΔI's at which DNB is prevented from occurring with 95% probability and 95% confidence.

A plot of a representative comparison is presented in Figure 3.2. A sample OTΔT analysis is presented in Section 6.0.

Table 3.1: Typical Uncertainties Applied in OTΔT Setpoint Analysis

Uncertainty Source	Uncertainty Value*

Table 3.2: Typical Uncertainties Applied in OTΔT Saturation Temperature Calculation

Uncertainty Source	Uncertainty Value*

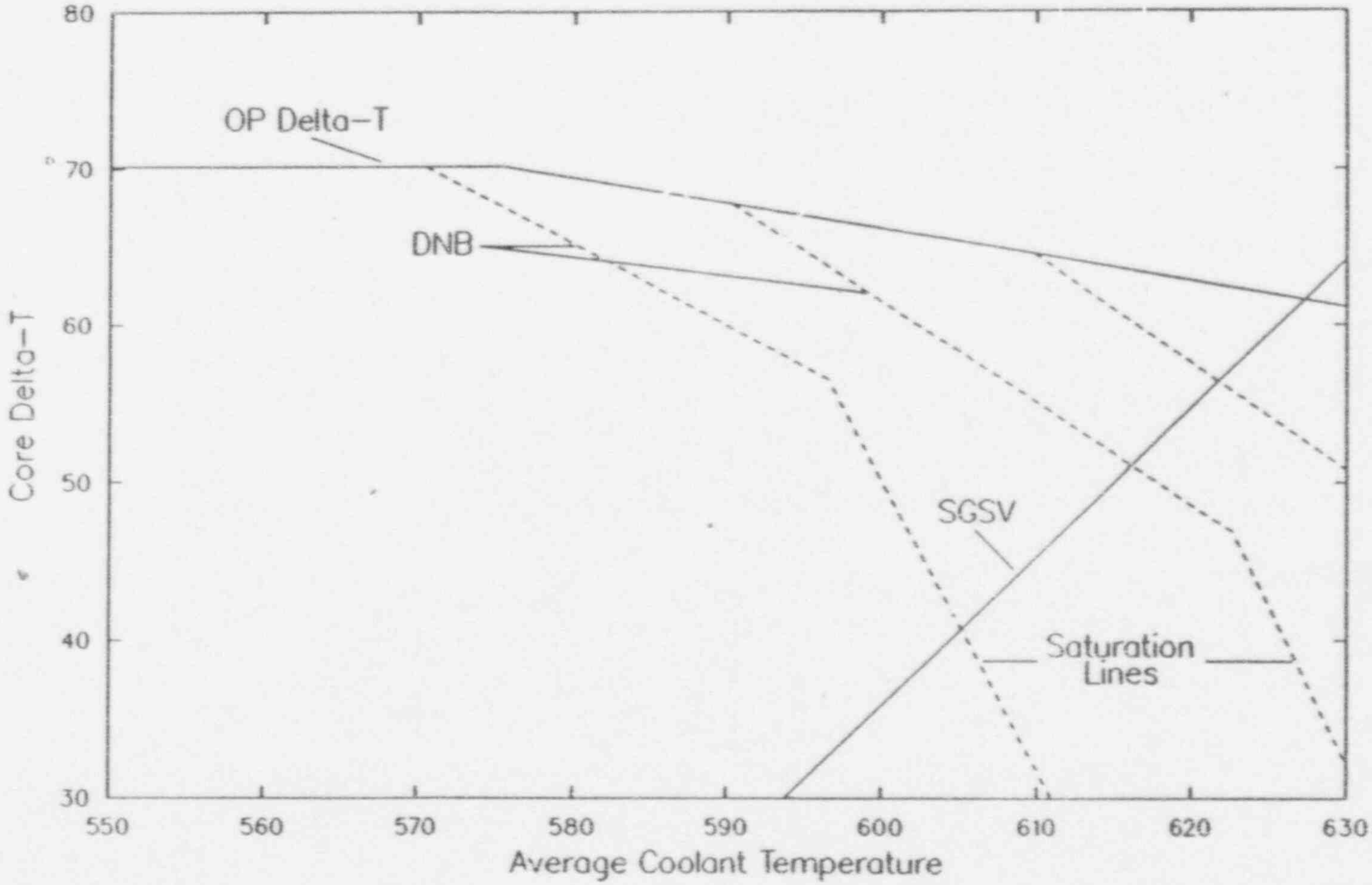

Figure 3.1: Sample OTΔT NOMSCAN Plot (1850, 2150, 2400 psia)

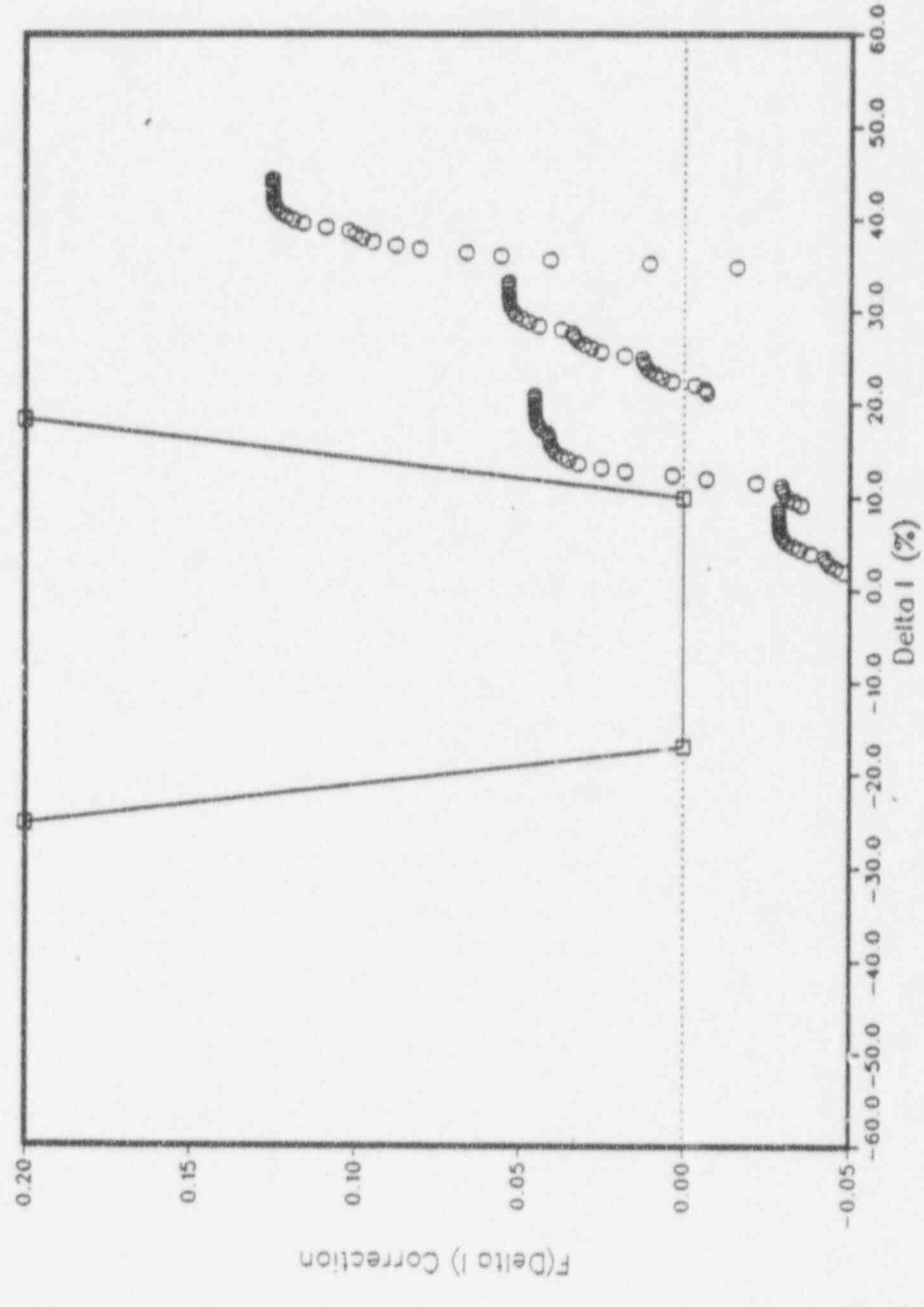

Figure 3.2: Sample OTΔT F(ΔI) Plot

4.0 STATISTICAL TRANSIENT ANALYSIS METHODOLOGY

The Westinghouse type plant statistical transient analyses will be performed in a manner which results in the protective system setpoint required to prevent exceeding the respective limit with 95% probability and 95% confidence. For example, the uncontrolled rod withdrawal event is protected by the OTΔT and high flux trip setpoints. The value of one or both of these trips would be determined which prevents the occurrence of the respective limit of concern (e.g., DNBR).

The first step in the transient analysis process is

These transient simulations will be done according to an approved methodology.

The statistical transient calculational process for Westinghouse type reactors is very similar to the process outlined for the OTΔT and OPΔT trip setpoints. The calculational steps are listed below.

A more detailed description of the calculational process follows.

in the statistical transient DETSCAN analysis, all of the uncertainty parameters are set to their respective values. The trip setpoint is then determined which will produce the limit (e.g. DNBR or RCS overpressure).

In the statistical transient NOMSCAN analysis, the uncertainty parameters which are treated statistically are set to their values. All of the parameters which are not treated statistically are set in accordance with an approved deterministic transient analysis methodology.

The method of parameter variation is according to SNP's approved GSUAM methodology[2,3].

To determine the 95% probability/95% confidence uncertainty adjusted trip coefficient, a Monte Carlo calculation is performed on the generated response surface. In addition to the uncertainty parameters in the response surface, the trip specific uncertainties are also included in the Monte

Carlo calculation. The various parameter uncertainties and their respective distributions are explicitly modeled in the Monte Carlo calculation. The values and distributions of the various uncertainty plant parameters will be justified on a plant specific basis.

The 95/95 trip coefficient from the Monte Carlo calculation is

to determine the trip statistical uncertainty adjustment. This statistical uncertainty adjustment is applied

. The most limiting of the resultant trips is the setpoint which will prevent the transient from violating the respective limit with 95% probability and 95% confidence and will be used in the setting of the Technical specification value. If the trip being considered is either the OPΔT or OTΔT, the resultant trip is compared with that which was generated in the statistical setpoint analysis. The most limiting of the statistical trip values is used in the setting of the Technical Specifications.

A list of parameters typically treated statistically in the transient analyses is presented in Table 4.1. A sample statistical transient analysis utilizing the method described above is presented in Section 6.

As an alternative to the above described method, the Rod Control Cluster Drop (RCCD) and Loss of Coolant Flow transients may be performed with a method very similar to that described in SNP's approved Combustion Engineering type plant statistical transient analysis methodology[1]. The protective systems for the two plant designs are functionally equivalent for these particular events.

Postulated accidents are allowed to have a limited amount of fuel failures caused by the penetration of DNB.

Table 4.1: Parameters Typically Treated Statistically in Transient Analyses

5.0 NEUTRONICS ANALYSIS

The purpose of this section is to describe the generation of the core power distributions utilized in the determination of the F(ΔI) portion of the reactor trip functions. An F(ΔI) adjustment is used in the overpower ΔT trip function to account for the effect of variations in the core axial power distribution on the margin to fuel centerline melt. Another F(ΔI) adjustment is utilized in the overtemperature ΔT trip function to account for the effects of variations in the core axial power distribution on the margin to DNB. The F(ΔI) terms are functions of ΔI, the difference in power (neutron flux) between the top half and bottom half of the core.

The following is a description of the procedure used in establishing the reactor core model which generates the axial power distributions (APD) used in the setpoint analyses.

1. For the reactor being analyzed, a three-dimensional model is assembled and depleted for previous cycles. Results from these calculations provide the necessary fuel assembly exposure distribution for the cycle being analyzed.

2. For the core design being analyzed the three-dimensional model is depleted to simulate the expected behavior of the core during the cycle.

5. From the cases in Step 4 the core power and core maximum values of $F_{\Delta H}$ and F_Q are obtained

6.0 SAMPLE CALCULATIONS

6.1 Sample Calculation Introduction

This section presents a sample implementation of the statistical setpoint and transient methodology for Westinghouse type plants. In this demonstration, the OPΔT and OTΔT setpoints will be developed. In addition, a sample statistical uncontrolled rod withdrawal analysis will be performed. The OPΔT reactor trip setpoint provides protection of the SAFDL on fuel centerline melt. The OTΔT reactor trip setpoint provides protection of the SAFDL on DNB and precludes the occurrence of coolant saturation at the vessel exit. Final confirmation of the OPΔT and OTΔT setpoint equations is provided by the plant transient analysis.

A summary of the OPΔT and OTΔT setpoints developed via this methodology is provided in Section 6.2 of this document. In addition to the resultant trips, the respective plant conditions assumed during the calculations are also provided in Section 6.2. The calculations and results of the sample OPΔT setpoint development are given in Section 6.3. In Section 6.4, the results of a sample OTΔT calculation are provided. Section 6.5 presents a sample statistical analysis of the uncontrolled rod withdrawal event.

6.2 Sample Calculation Summary

OPΔT and OTΔT reactor trip setpoints have been developed for a typical 3-loop Westinghouse plant operating at 2300 MWt. These setpoints are to respectively provide designed protection of the SAFDLs on fuel centerline melt, coolant saturation at the vessel exit and DNB for Anticipated Operational Occurrences (AOO). The plant conditions assumed in the development of these trips are presented in Table 6.2.1. The applied uncertainties are presented in the respective analysis sections. The resultant OPΔT and OTΔT setpoint equations are given in Tables 6.2.2 and 6.2.3 respectively.

These analyses were performed in accordance with the SNP methodology described in the previous sections of this report. Important design inputs to the setpoint calculations are provided in Sections 6.3 and 6.4. A statistical uncontrolled rod withdrawal transient analysis was also performed that confirmed the OTΔT K_1 coefficient generated in the setpoint analysis. The important inputs to the statistical transient analysis are present in Section 6.5.

Table 6.2.1: Sample Calculation Plant Conditions

Design Parameters	Value
Core Thermal Power	2300 MWt
Rated Power Vessel Ave. Temp.	575.4 °F
Primary System Pressure	2250 psia
Minimum Allowed Pressure	1850 psia
Maximum Allowed Pressure (High Pressurizer Trip)	2400 psia
Vessel Coolant Flow (Minimum)	97.3 Mlbm/hr
Bypass Flow Fraction	4.5%
Steam Generator Pressure (Dome)	800 psia
Steam Flow per Steam Generator	3.37 Mlbm/hr
Number of Fuel Assemblies	157
Number of Fuel Rods/Assembly	204
Fuel Rod OD	0.424
Active Fuel Length	144 in.
Design $F_{\Delta H}$	1.70
Design F_Q^T	3.01

Table 6.2.2: Overpower ΔT Reactor Trip Setpoint for a 3-Loop Westinghouse Plant

$$\Delta T\frac{(1+\tau_1 S)}{(1+\tau_2 S)}\left(\frac{1}{1+\tau_3 S}\right) \le \Delta T_0\left(K_4-K_5\left(\frac{\tau_7 S}{1+\tau_7 S}\right)\left(\frac{1}{1+\tau_6 S}\right)T-K_6\left(T\left(\frac{1}{1+\tau_6 S}\right)-T'\right)-F(\Delta I)\right)$$

where:

ΔT	=	measured ΔT, F-degrees
ΔT_0	=	indicated ΔT at rated thermal power, F-degrees
T	=	indicated average reactor coolant temperature, °F
T'	=	575.4 °F, indicated average reactor coolant temperature at rated thermal power
K_4	$\le$	1.110
K_5	=	0.20/ °F for increasing average temperature and 0 for decreasing average temperature, 1/°F;
K_6	=	0.0027 for $T > T'$ and 0 for $T_{avg} \le T'$
S	=	Laplace transform operator, 1/sec;
$(1+\tau_1 {*} S)/(1+\tau_2 {*} S)$	=	Lead-lag compensator on measured ΔT
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 = 0$ sec, $\tau_2 = 0$ sec
$1/(1+\tau_3 {*} S)$	=	Lag compensator on measured ΔT
τ_3	=	Time constant utilized in lag compensator for ΔT, $\tau_3 = 0$ sec
$1/(1+\tau_6 {*} S)$	=	Lag compensator on measured average coolant temperature
τ_6	=	Time constant utilized in lag compensator for measured average coolant temperature, $\tau_6 = 0$ sec
$\tau_7 S/(1+\tau_7 S)$	=	the function generated by the rate-lag controller for average coolant temperature dynamic compensation
τ_7	=	Time constant utilized in rate-lag controller for average coolant temperature dynamic compensation, $\tau_7 = 10$ sec
$F(\Delta I)$	=	See description on next page and Figure 6.2.1

Table 6.2.2 (Cont.)

$F(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers, with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for q_t-q_b between +12 percent and -4 percent, $F(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core, respectively, and q_t+q_b is total THERMAL POWER in percent of RATED THERMAL POWER (2300 MWt), $F(\Delta I) = 0$. For every 2.4% below the rated power (2300 MWt) level, the permissible positive flux difference range is extended by +1 percent. For every 2.4% below the rated power (2300 MWt) level, the permissible negative flux difference range is extended by -1 percent.

(ii) for each percent that the magnitude of q_t-q_b exceeds +12 percent, the ΔT trip setpoint shall be automatically reduced by 2.4 percent of its value at RATED THERMAL POWER (2300 MWt).

(iii) for each percent that the magnitude of q_t-q_b exceeds -4 percent, the ΔT trip setpoint shall be automatically reduced by 2.4 percent of its value at RATED THERMAL POWER (2300 MWt).

Table 6.2.3: Overtemperature ΔT Reactor Trip Setpoint for a 3-Loop Westinghouse Plant

$$\Delta T\frac{(1+\tau_1 S)}{(1+\tau_2 S)}\left(\frac{1}{1+\tau_3 S}\right) \le \Delta T_0\left(K_1 - K_2\left(\frac{1+\tau_4 S}{1+\tau_5 S}\right)\left(T\left(\frac{1}{1+\tau_6 S}\right) - T'\right) + K_3(P-P') - F(\Delta I)\right)$$

where:

ΔT	=	measured ΔT, F-degrees
ΔT_0	=	indicated ΔT at rated thermal power, F-degrees
T	=	indicated average reactor coolant temperature, °F
T'	=	575.4 °F, indicated average reactor coolant temperature at rated thermal power
P	=	pressurizer pressure, psig
P'	=	2235 psig, nominal pressurizer pressure at rated conditions
K_1	$\le$	1.189
K_2	=	0.01228 (1/F-degrees)
K_3	=	0.00089 (1/psi)
S	=	Laplace transform operator, 1/sec;
$(1+\tau_1{*}S)/(1+\tau_2{*}S)$	=	Lead-lag compensator on measured ΔT
τ_1, τ_2	=	Time constant utilized in lead-lag compensator on measured ΔT, $\tau_1 = 0.0$ sec
$1/(1+\tau_3{*}S)$	=	Lag compensator on measured ΔT
τ_3	=	Time constant utilized in lag compensator on measured ΔT, $\tau_3 = 0.0$ sec
$(1+\tau_4{*}S)/(1+\tau_5{*}S)$	=	Lead-lag compensator on average coolant temperature
τ_4	=	Time constant utilized in lead-lag compensator on average coolant temperature, $\tau_4 = 20.0$ sec
τ_5	=	Time constant utilized in lead-lag compensator on average coolant temperature, $\tau_5 = 3.0$ sec
$1/(1+\tau_6{*}S)$	=	Lag compensator on measured average coolant temperature
τ_6	=	Time constant utilized in lag compensator on measured average coolant temperature, $\tau_6 = 0.0$ sec

Table 6.2.3 (Cont.)

$F(\Delta I)$	=	a function of the indicated flux difference between the top and bottom detectors of the power range nuclear ion chambers (ΔI). See discussion below and Figure 6.2.2.

$F(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers, with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for q_t-q_b between +10 percent and -17 percent, $F(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core, respectively, and q_t+q_b is total THERMAL POWER in percent of RATED THERMAL POWER (2300 MWt), $F(\Delta I) = 0$. For every 2.4% below the rated power (2300 Mwt) level, the permissible positive flux difference range is extended by +1 percent. For every 2.4% below the rated power (2300 MWt) level, the permissible negative flux difference range is extended by -1 percent.

(ii) for each percent that the magnitude of q_t-q_b exceeds +10 percent, the ΔT trip setpoint shall be automatically reduced by 2.4 percent of its value at RATED THERMAL POWER (2300 MWt).

(iii) for each percent that the magnitude of q_t-q_b exceeds -17 percent, the ΔT trip setpoint shall be automatically reduced by 2.4 percent of its value at RATED THERMAL POWER (2300 MWt).

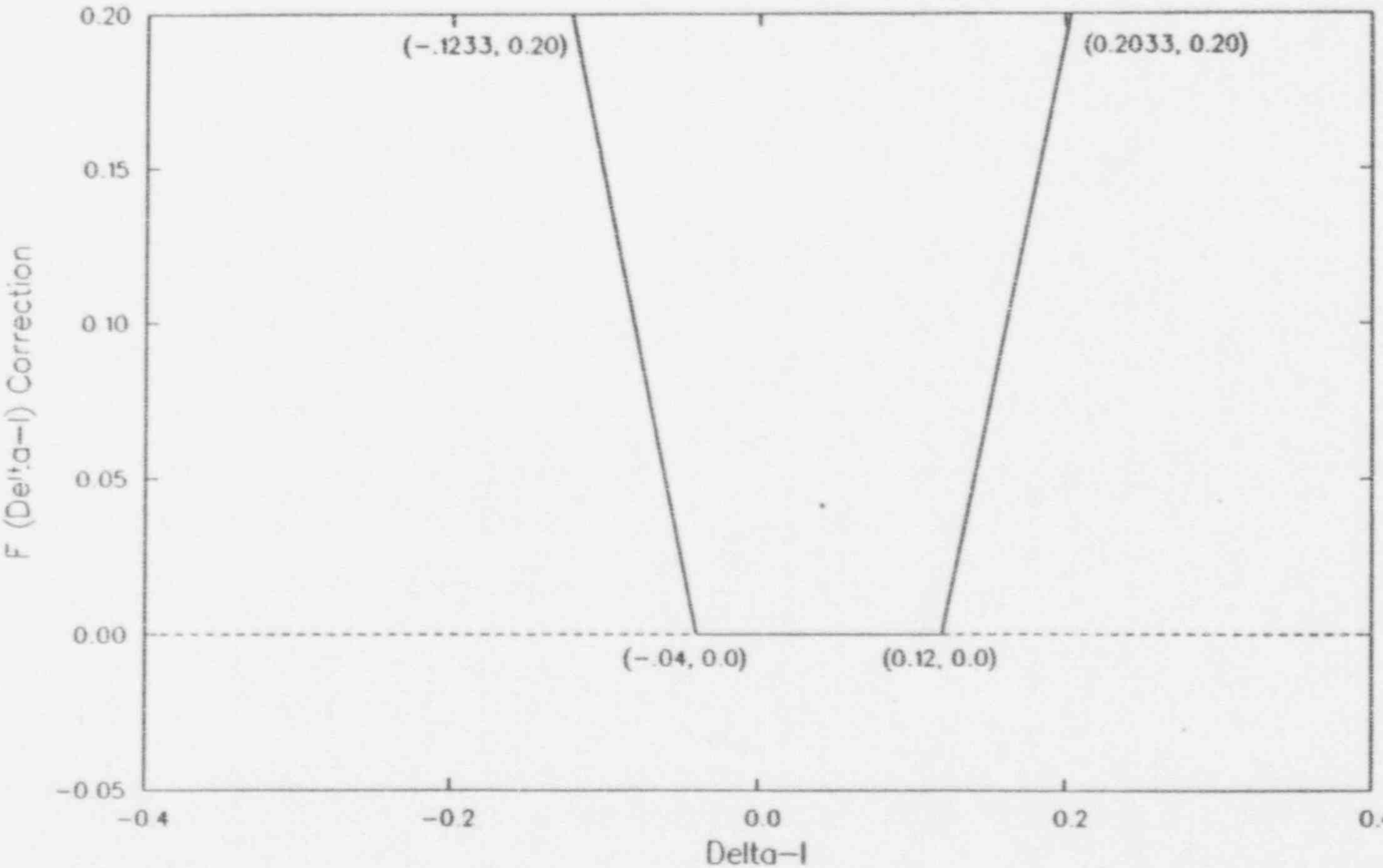

Figure 6.2.1: F(ΔI) Function for OPΔT Setpoint

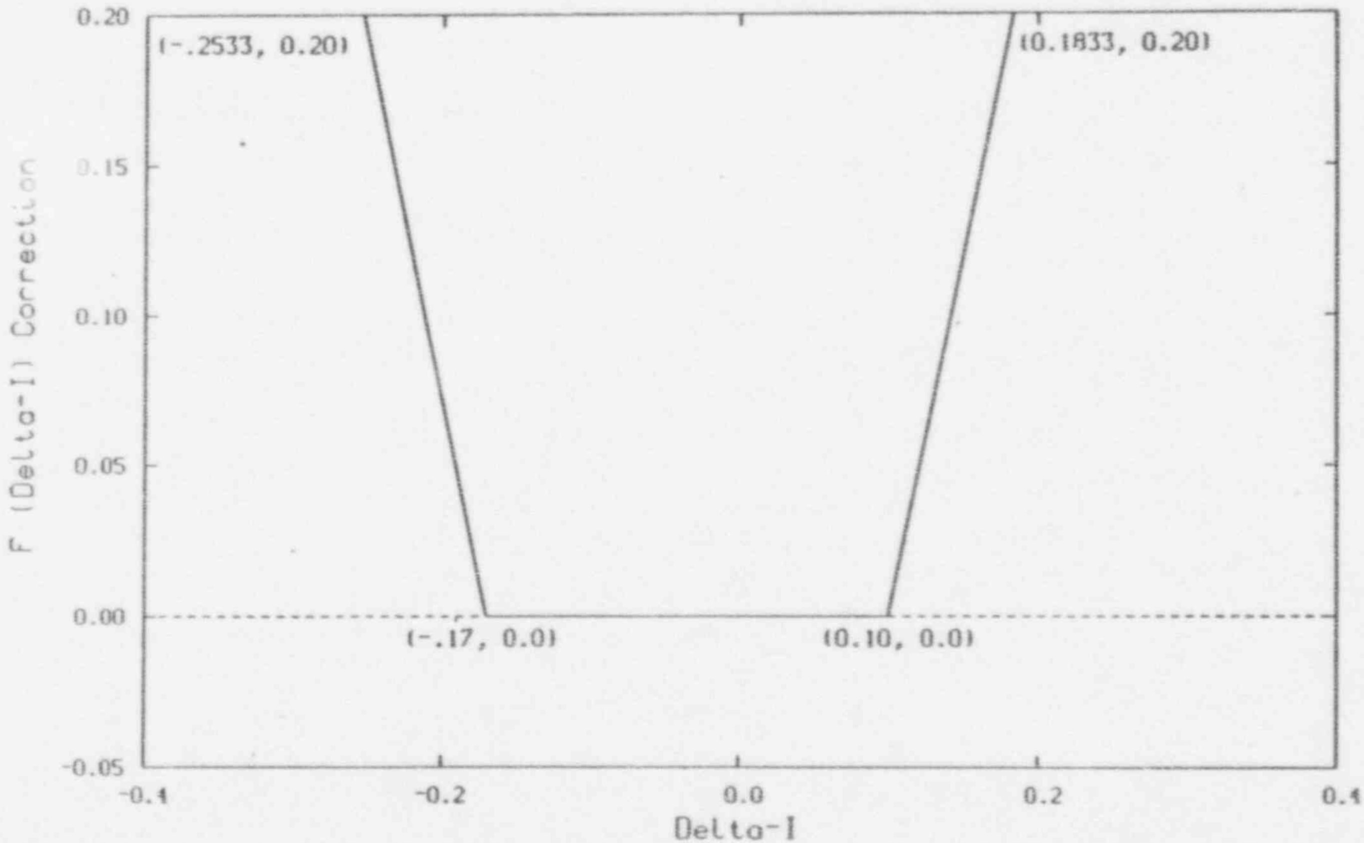

Figure 6.2.2: F(ΔI) Function for OTΔT Setpoint

6.3 Overpower ΔT Setpoint Equation

6.3.1 Fuel Melt Limit Determination

The OPΔT trip coefficients are set so as to prevent the occurrence of fuel centerline melt with 95% probability and 95% confidence. Before the trip coefficients can be determined, it is first necessary to determine the LHGR at which fuel melt will take place. This is done by comparing the fuel centerline temperatures, calculated by RODEX2[4], with the respective temperatures at which fuel melt will occur. Figure 6.3.1 depicts the LHGR fuel melt limit as a function of assembly burnup. This LHGR limit curve is overlayed with a series of XTGPWR[5] assembly peaking and exposure maps to determine the allowed LHGR as a function of cycle depletion. The LHGR limit as a function of the sample calculation cycle depletion is depicted in Figure 6.3.2.

6.3.2 OPΔT Trip Coefficient Determination

The overpower ΔT setpoint function is listed in Table 6.2.2. A list of all of the uncertainties assumed in the performance of the sample statistical OPΔT analysis is provided in Table 6.3.1. The setpoint function was developed to protect the core protection limit of 118% power over the range of pressures delineated by the high and low pressurizer pressure trips. The design F_Q was set

The convolution of the uncertainties was performed by a Monte Carlo analysis.

This K_4 trip coefficient was adjusted

The K4 trip coefficient adjustment is determined by a Monte Carlo analysis. A plot of the resultant cumulative probability distribution is presented in Figure 6.3.3. The resultant 95/95 minimum K_4 trip coefficient This was truncated to the recommended Technical Specification trip coefficients of:

6.3.3 F(ΔI) Reset Trip Function for OPΔT Setpoint

The OPΔT F(ΔI) trip reset function is designed to account for the effects on fuel temperature of core power distributions more adverse than the design peaking. The OPΔT F(ΔI) reset trip function is listed in Table 6.2.2. The OPΔT F(ΔI) function is depicted in Figure 6.2.1.

The uncertainties used in the statistical OPΔT F(ΔI) verification

The axial distributions analyzed were generated with the methodology presented in Section 5. A bounding set of power adjustment/ΔI joint probability curves were generated for the range of fuel melt power levels A representative joint probability curve is presented in Figure 6.3.4. The joint probability adjustment curves are applied to determine the required correction as a function of ΔI. A comparison of thes ›ints and the proposed OPΔT F(ΔI) function is presented in Figure 6.3.5. As shown, the proposed F(ΔI) function bounds the calculated points. If the F(ΔI) function did not bound the calculated points, either the function or the design F_Q would require modification.

Table 6.3.1: OPΔT Statistical Uncertainty Parameters

Figure 6.3.1: Assembly LHGR Limit vs. Assembly Exposure

Figure 6.3.2: LHGR Limit as a Function of Cycle Depletion

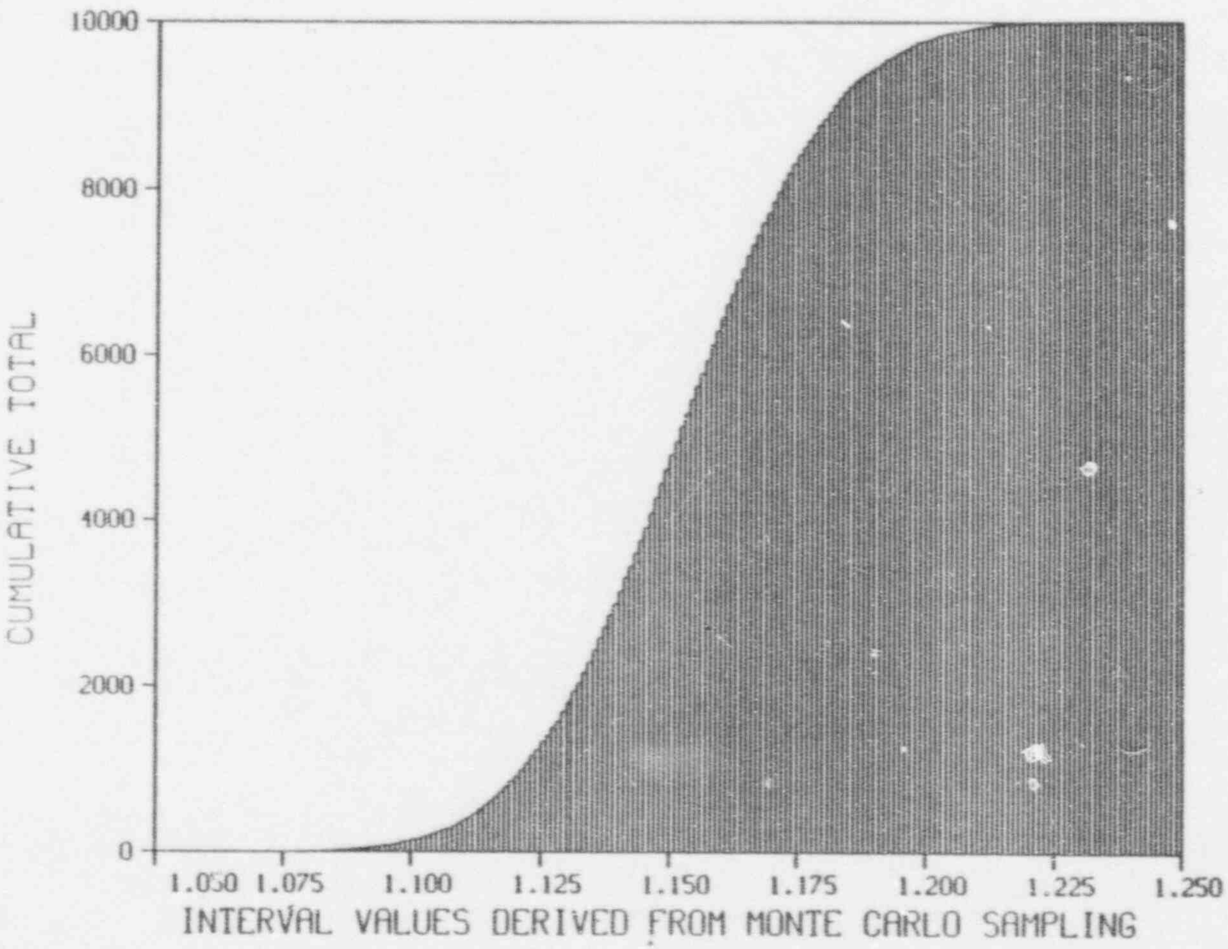

Figure 6.3.3: OPΔT Cumulative Probability Distribution

Figure 6.3.4: OPΔT Power Adjustment/ΔI Joint Probability Curve for Nominal Power of 116.1 %

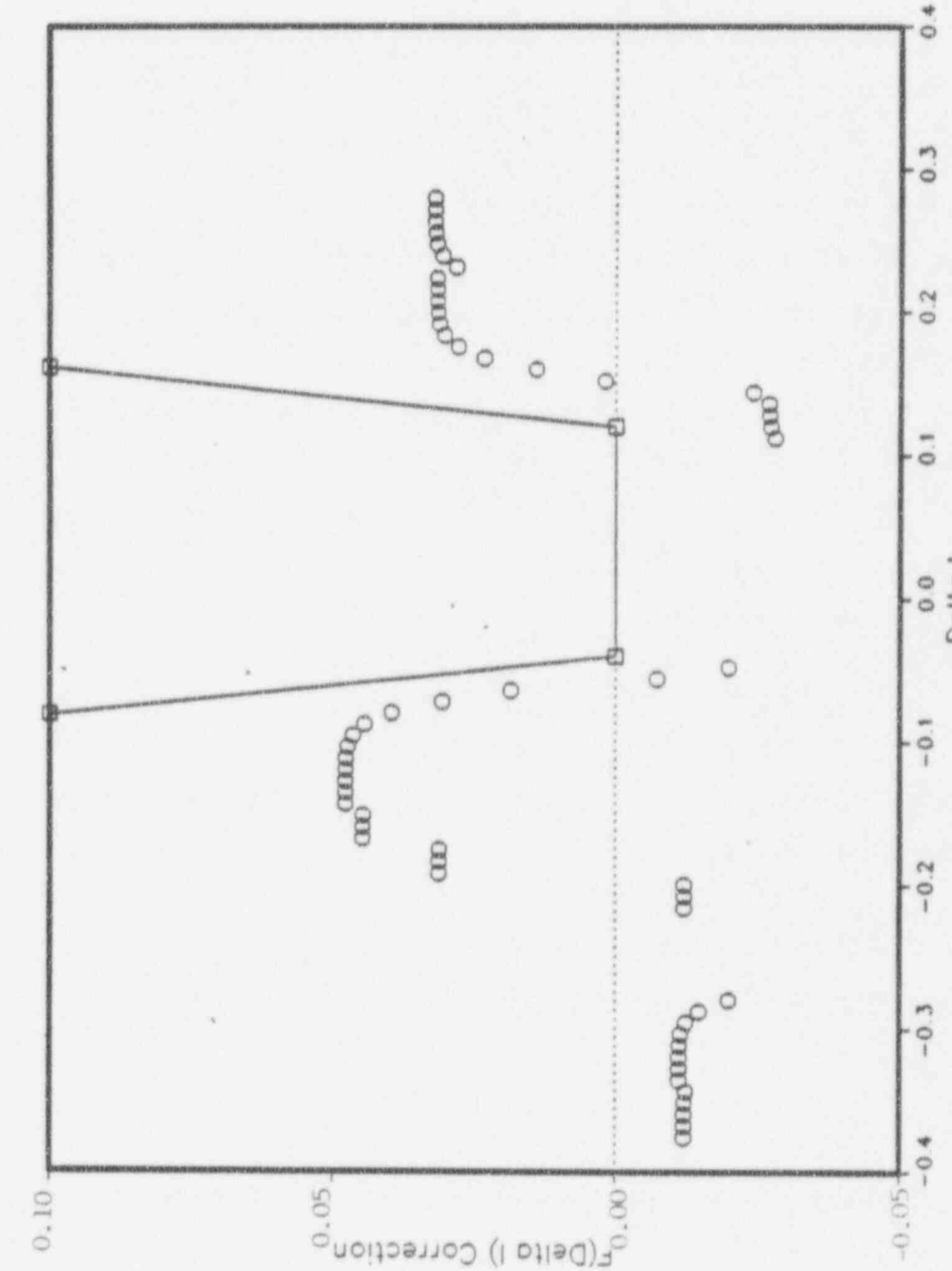

Figure 6.3.5: OPΔT F(ΔI) Reset Trip Function

6.4 Overtemperature ΔT Setpoint Equation

6.4.1 Determination of Overtemperature ΔT Setpoint Coefficients

The OTΔT setpoint coefficients are set so as to prevent the occurrence of DNB or hot leg saturation with 95% probability and 95% confidence. The process used to determine the coefficients is provided in Section 3.3. The DNB calculations were all performed with SNP's XCOBRA-IIIC computer code(6).

The adequacy of the axial distribution is subsequently verified in Section 6.4.2, the OTΔT F(ΔI) reset trip function analysis.

A response surface was generated according to SNP's approved GSUAM methodology(2,3). The parameters varied in the generation of the DNB ΔT response surface are identified in Table 6.4.1. A comparison of the fitted versus actual DNB ΔT data points is presented in Figure 6.4.1.

A Monte Carlo calculation was performed to determine the 95%/95% statistical DNB ΔT adjustment. The Monte Carlo calculation used equation 6.1, listed below.

A hot leg saturation NOMSCAN/DETSCAN calculational sequence was performed

A response surface was generated according to SNP's GSUAM methodology[2,3]. The parameters varied in the generation of the hot leg saturation response surface are identified in Table 6.4.2. A comparison of the fitted versus actual hot leg saturation data points is presented in Figure 6.4.2.

A Monte Carlo calculation was performed to determine the hot leg saturation 95%/95% ΔT adjustment. The Monte Carlo calculation used equation 6.2 listed below.

The DNB and hot leg saturation statistical ΔT adjustments were applied to

to get the uncertainty adjusted conditions. These conditions represent the locations where the occurrence of DNB and hot leg saturation are prevented with 95% probability and 95% confidence. A set of OTΔT trip coefficients was developed which bounded these respective points. A comparison of the trip equation lines and the respective conditions of DNB and hot leg saturation is presented in Figure 6.4.3. The trip coefficients were:

6.4.2 F(ΔI) Reset Trip Function for Overtemperature ΔT Setpoint

The OTΔT F(ΔI) trip reset function is designed to account for the effects on MDNBR of core power distributions more adverse than the design values. The OTΔT F(ΔI) trip reset function is listed in Table 6.2.1. A plot of this function is presented in Figure 6.2.2.

The uncertainties used in the statistical OTΔT F(ΔI) analysis were

The OTΔT F(ΔI) function methodology is described in Section 3.4. The OTΔT F(ΔI) analysis was performed using the procedure described in Section 5. A NOMSCAN/DETSCAN calculational sequence was performed

A response surface was generated about this point using the parameters identified in Table 6.4.1. The parameters were varied in the generation of this response surface according to SNP's approved GSUAM methodology[2,3]. A comparison of the fitted versus actual core ΔT data points is presented in Figure 6.4.4.

A Monte Carlo calculation was performed

A plot of the cumulative probability is presented in Figure 6.4.5. A joint probability distribution of core ΔT and ΔI adjustment was generated based on this resultant probability table. The uncertainty distribution on ΔI The resultant joint probability uncertainty adjustment was applied

at which the occurrence of DNB is prevented with a 95% probability and 95% confidence.

A plot of the required core ΔT adjustments, converted into F(ΔI) units, and the proposed F(ΔI) function is presented in Figure 6.4.6.

Table 6.4.1: OTΔT Statistical DNB Parameters

Table 6.4.2: OTΔT Statistical Hot Leg Saturation Parameters

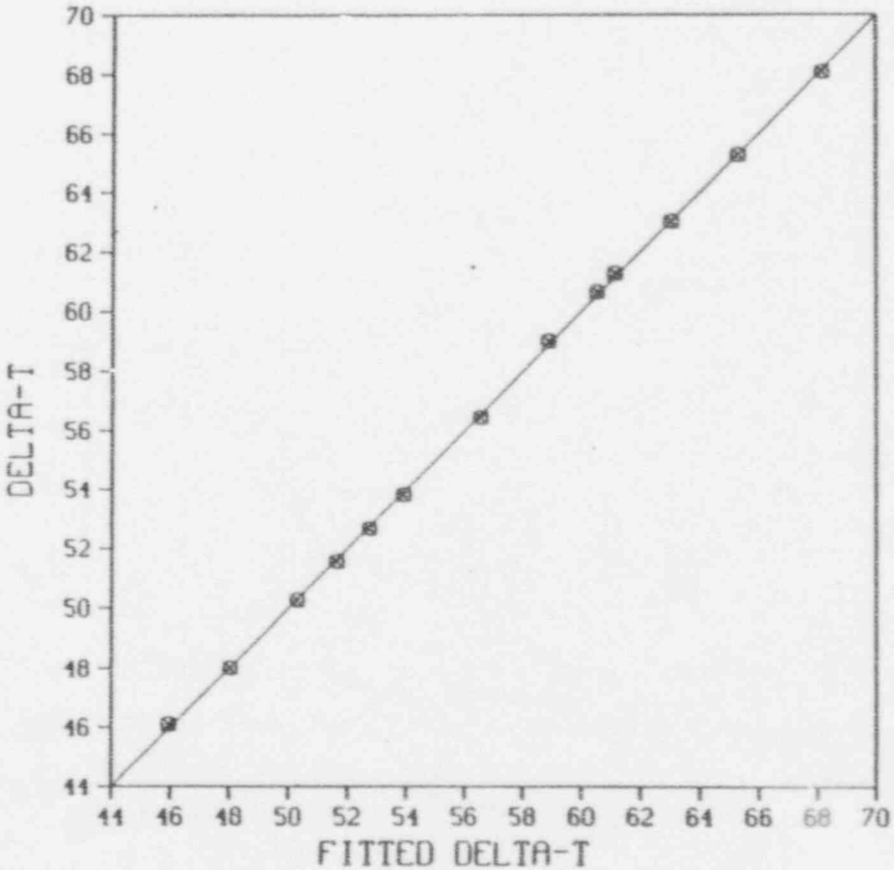

Figure 6.4.1: DNB Response Surface Fitted vs. Actual Comparison

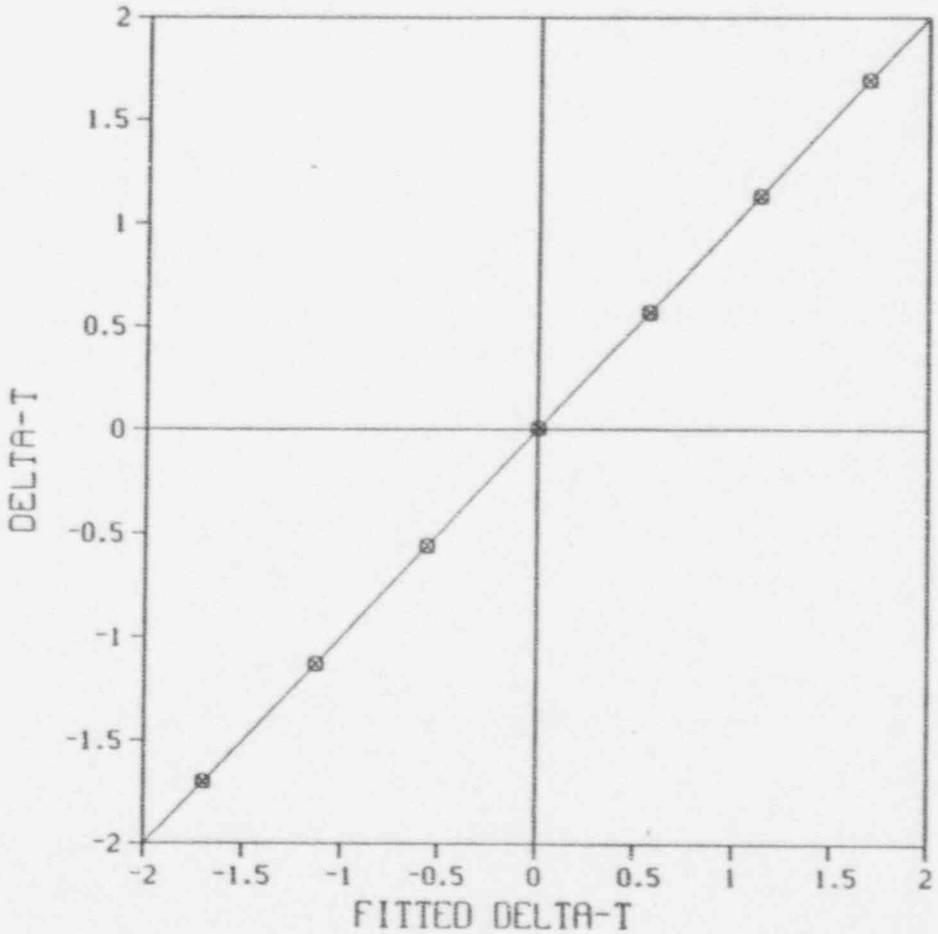

Figure 6.4.2: Hot Leg Saturation Response Surface Fitted vs. Actual Comparison

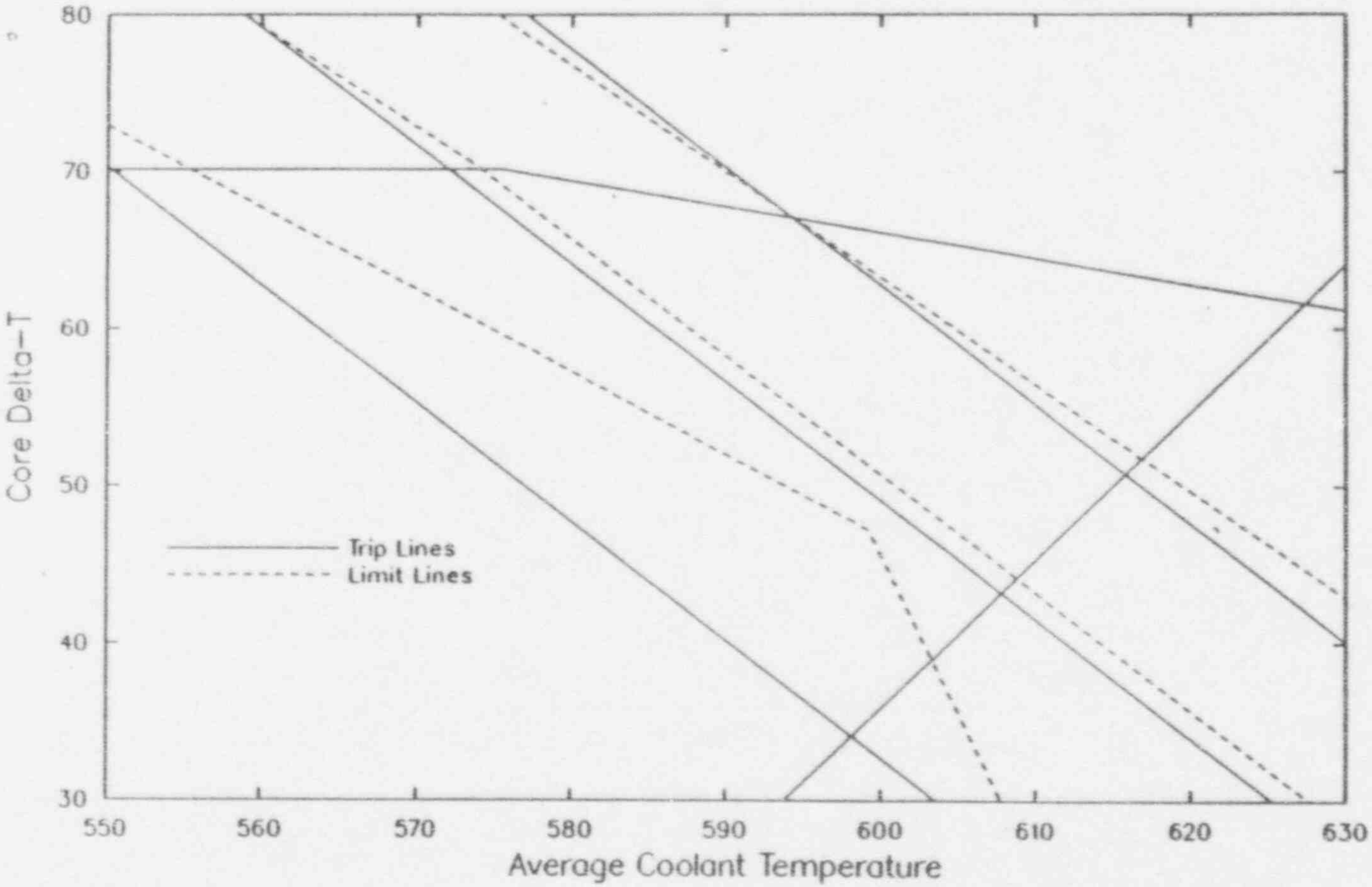

Figure 6.4.3: Verification of OTΔT Trip Coefficients (1850, 2150 and 2400 psia)

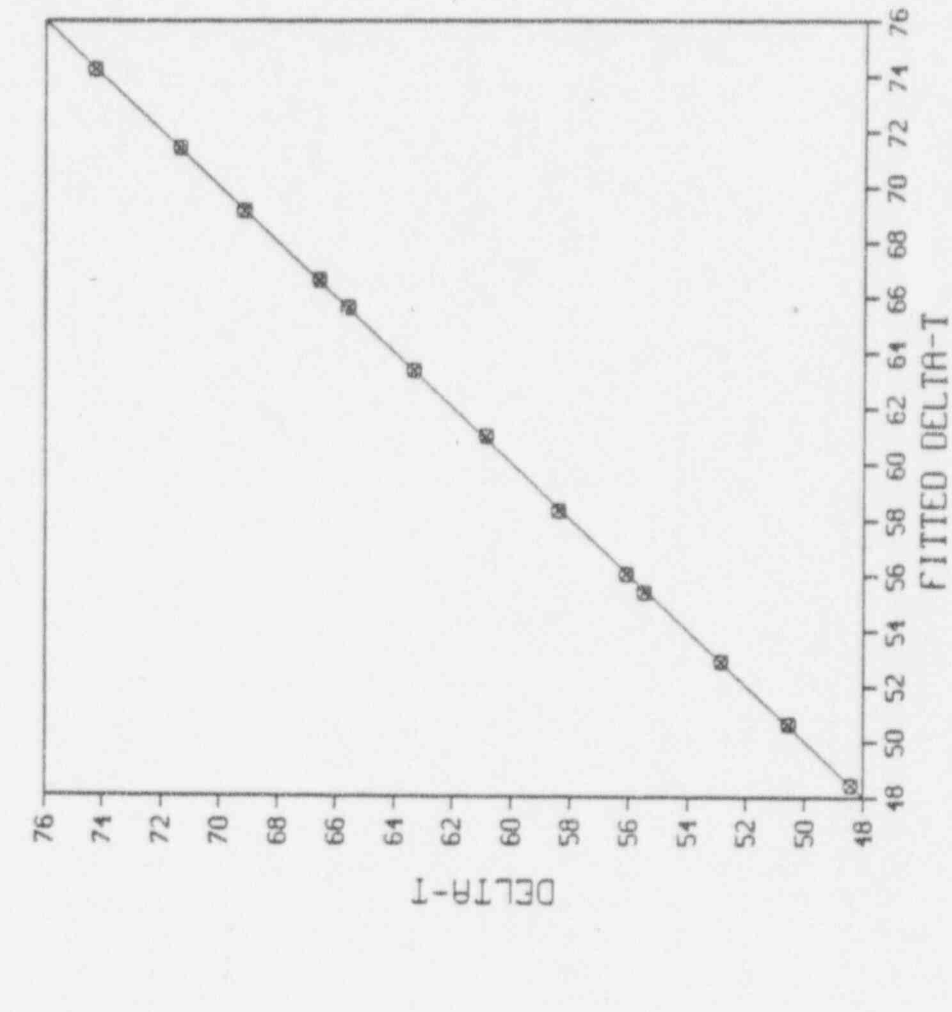

Figure 6.4.4: OTΔT F(ΔI) Response Surface Fitted vs. Actual Comparison

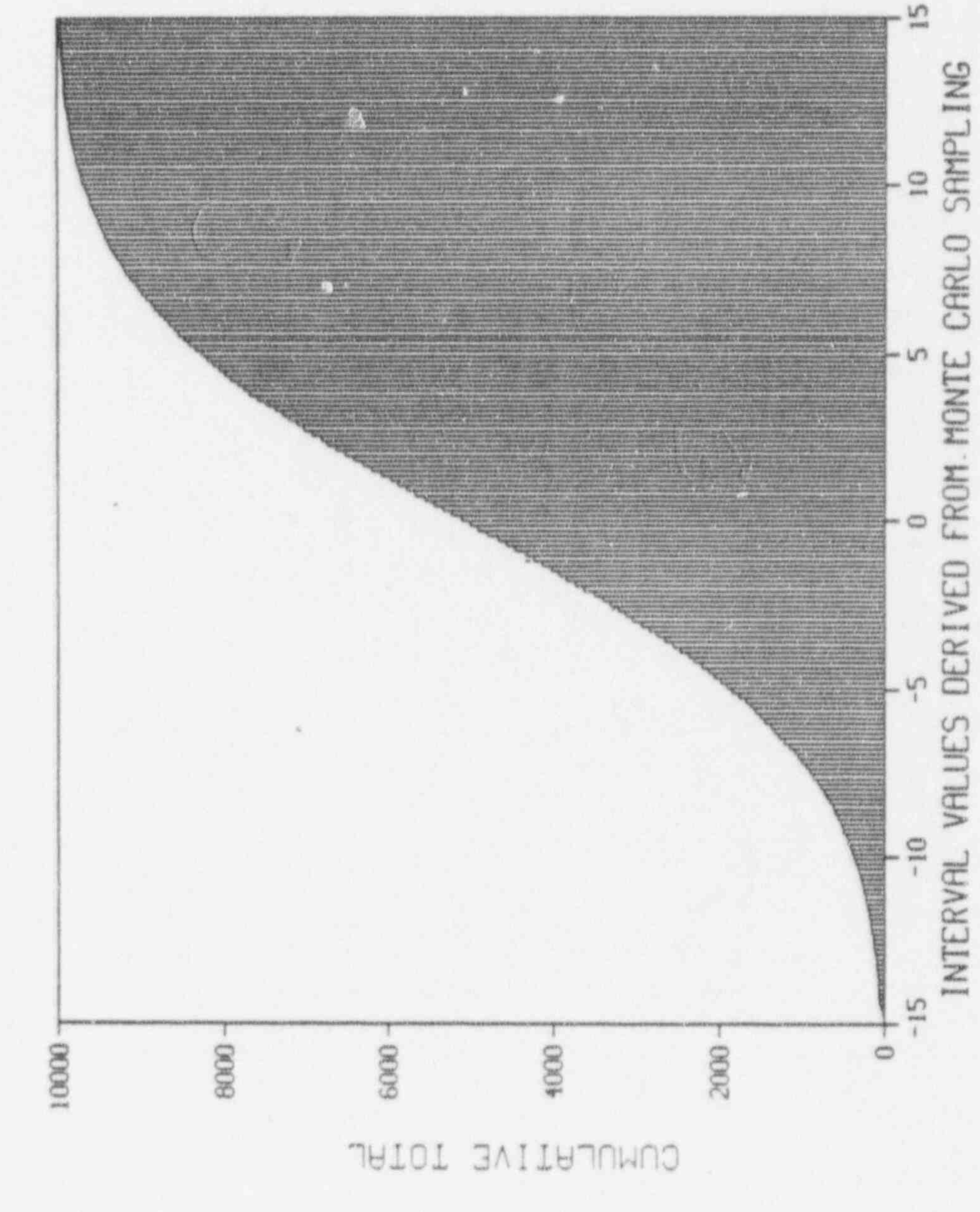

Figure 6.4.5: OTΔT F(ΔI) Monte Carlo Calculation Cumulative Distribution

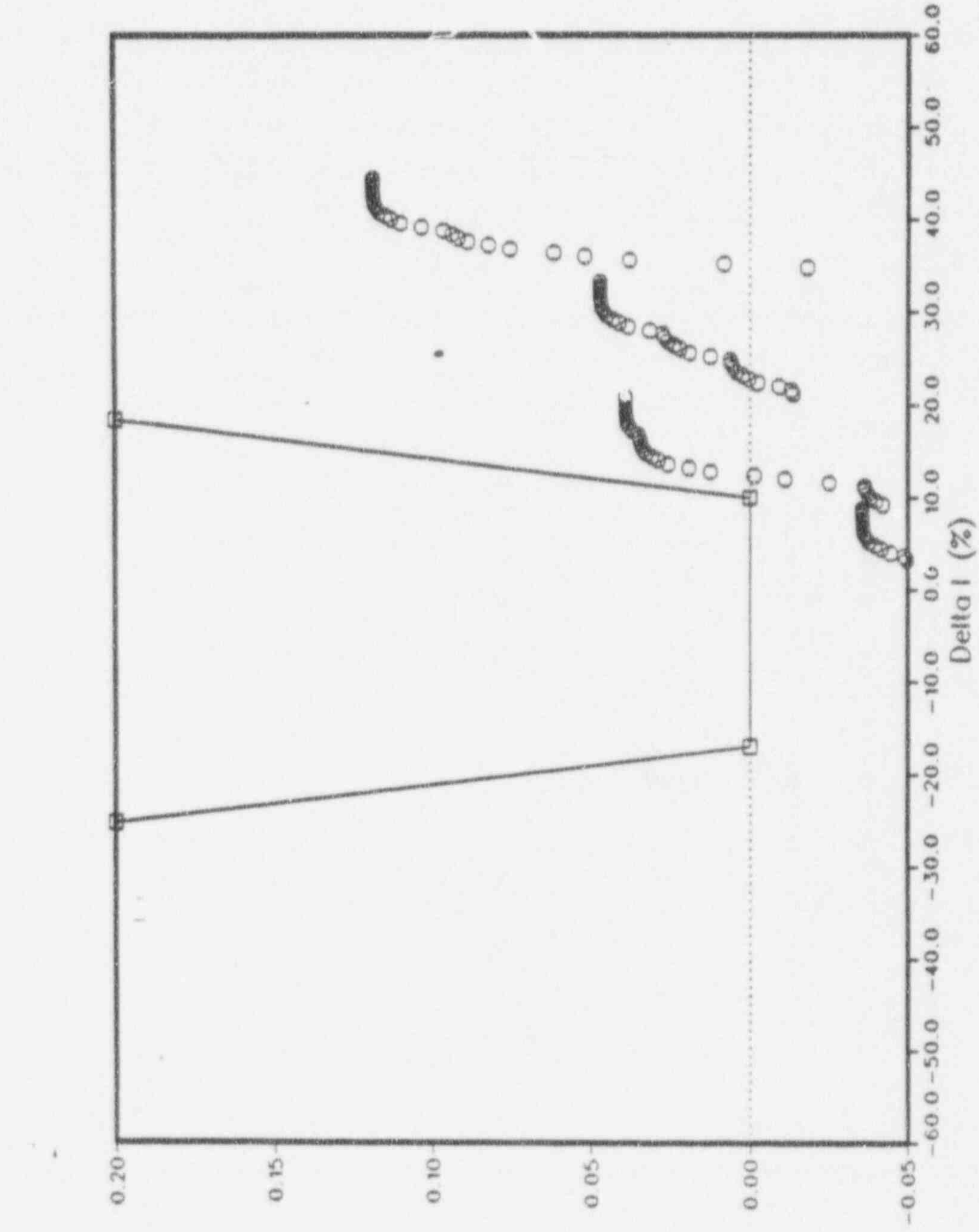

Figure 6.4.6: OTΔT F(ΔI) Plot

6.5 Sample Statistical Uncontrolled Rod Withdrawal Analysis

To demonstrate the statistical transient analysis methodology for Westinghouse type reactors, a sample uncontrolled rod withdrawal analysis was performed. The result of this analysis was an OTΔT trip setpoint which prevents the occurrence of DNB with 95% probability and 95% confidence. The methodology employed by this calculation was that described in Section 4.0 of this report.

The statistical uncontrolled rod withdrawal event considered statistically those parameters listed in Table 6.5.1.

The first step in the calculational process is to determine the reactivity insertion rate

A response surface was generated

In this response surface, the parameters listed in Table 6.5.1 were varied according to SNP's GSUAM methodology[2,3]. A plot of the comparison of the response surface fitted versus observed data points is presented in Figure 6.5.2.

To determine the statistical uncertainty adjustment to K_1 a Monte Carlo analysis was performed on equation 6.4.

The statistical K_1 adjustment was applied

which prevent the occurrence of DNB with 95% probability and 95% confidence.

Table 6.5.1: Statistical Uncertainty Parameters in Sample Uncontrolled Rod Withdrawal Analysis Response Surface

Figure 6.5.1: Statistical Uncontrolled Rod Withdrawal Analysis NOMSCAN/DETSCAN Results

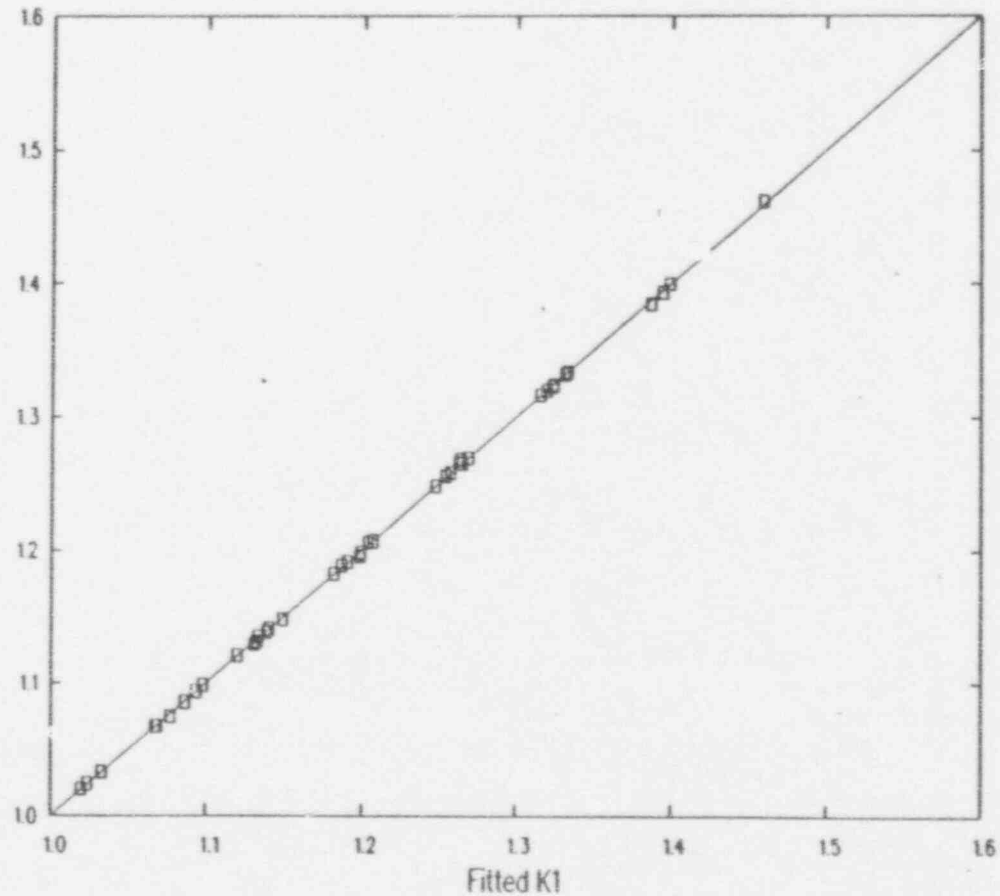

Figure 6.5.2: Statistical Uncontrolled Rod Withdrawal Analysis Response Surface Fit Comparison

7.0 REFERENCES

1) ENC Setpoint Methodology for C.E. Reactors: Statistical Setpoint Methodology, XN-NF-507(P)(A), Supplements 1 and 2, Exxon Nuclear Company, Richland, WA 99352, September, 1986.

2) Generic Statistical Uncertainty Analysis Methodology, XN-NF-81-22(P)(A), Exxon Nuclear Company, Richland, WA 99352, November, 1983.

3) Expanded Generic Statistical Uncertainty Analysis Methodology, XN-NF-507(P)(A), Supplement 1, Appendix A, Exxon Nuclear Company, Richland, WA 99352, September, 1986.

4) RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, XN-NF-81-58(A), Exxon Nuclear Company, Richland, WA 99352, February, 1983.

5) XTG - A Two-Group Three-Dimensional Reactor Simulator Utilizing Coarse Mesh Spacing, XN-CC-28(A), Revision 3, Exxon Nuclear Company, Richland, WA 99352, April, 1975.

6) XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation", XN-NF-75-21(A), Revision 2, Exxon Nuclear Company, Richland, WA 99352, January, 1986.

7) Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events, ANF-84-73(P), Revision 3, Advanced Nuclear Fuels Corporation, Richland, WA 99352, May, 1988.

SUPPLEMENT 1

CORRESPONDENCE

SIEMENS

August 13, 1993
RAC:93:127

Mr. R. C. Jones, Chief
Reactor Systems Branch
Division of Engineering and System Technology
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Jones:

Responses to NRC Questions on SPC Statistical Transient Methodology

References: (1) Letter, R. C. Jones (USNRC) to R. A. Copeland (SPC), "Request for Additional Information EMF-92-081(P), 'Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors'," July 16, 1993.

(2) Letter, R. A. Copeland (SPC) to T. E. Murley (USNRC), RAC:065:92, May 29, 1992.

Attached are the responses to the additional information requested in the Reference 1 letter. These responses are needed for the NRC review of Siemens Power Corporation's statistical setpoint/transient methodology for Westinghouse type of reactors.

Siemens Power Corporation considers the information contained in the attached responses to be proprietary. The affidavit supplied with the original submittal (Reference 2) should satisfy the requirements of 10 CFR 2.790(b) to support the withholding of the attachment from public disclosure.

If there are questions, or if additional information is needed, please contact me at (509) 375-8290.

Very truly yours,

Bob Copeland

R. A. Copeland, Manager
Product Licensing

/smg

Attachment

cc: Dr. J. Carew, BNL
Mr. L. I. Kopp, USNRC

bcc: F. T. Adams
D. M. Brown
R. C. Gottula
L. E. Hansen
J. S. Holm
H. G. Shaw
F. B. Skogen
L. D. O'Dell
File/LB

Question 1 - To what type of Westinghouse plants and fuel designs is the EMF-92-081(P) methodology applicable ? Besides the trip parameters (K_i, τ_i) and uncertainties, what plant specific-methodology adjustments are required ?

Response - The SPC methodology is considered applicable to W plants equipped with OPΔT and OTΔT reactor trips of the form given in Sections 2.1 and 3.1 of Reference 1, respectively. The methodology is applicable to fuel designs for which SPC has or obtains approved models and methodologies for centerline temperature and DNB evaluations. These fuel designs will include all SPC fuel designs for W reactors.

The SPC methodology is written to cover the complete set of calculations required to develop new setpoints from scratch. In some instances, plant applications will require the verification of existing setpoints rather than the development of new ones. In these instances, the trip coefficients are already specified, and it is necessary only to verify that the trip setpoint adequately protects the SAFDLs. The computation of the trip coefficients is deleted from the analysis. The remainder of the analyses are retained, if a deterministic uncertainty analysis proves overly conservative. A conservative deterministic uncertainty may be applied in the F(ΔI) verification analysis, for computational convenience.

Some plants are already provided with a statistical trip channel uncertainty allowance by the reactor vendor. The total trip channel statistical allowance, termed Total Allowance (TA), is embedded in the plant Technical Specifications monitoring and surveillance procedures. It is therefore desirable that the precise value of TA be retained in SPC setpoint analyses.

Because the TA encompasses a two-sided 95% probability level, it is taken as a $\pm 2\sigma$ uncertainty band. It is assumed normal, per the response to Question 6 below. Because the TA typically includes all applicable ΔI uncertainties, joint power-ΔI probability curves need not be generated.

Plant or application-specific methodology adjustments are formulated to retain the general statistical approach detailed in the SPC methodology report. The use of conservative deterministic uncertainty treatments in the place of or in conjunction with statistical treatments is an acceptable alternative to the full statistical treatment described in the report. The design bases of the OTΔT and OPΔT reactor trips and the functional bases of the SPC methodology are preserved under any adjustments that prove necessary.

Question 2 - Typical OTΔT trips limit the outlet quality to less than the limit of the DNB correlation (e.g., 15%). How is this limit incorporated in the SPC setpoint?

Response - The SPC setpoint methodology will utilize primarily the HTP DNB correlation, which has a quality limit well above the W-3 correlation limit of 15%. Typically, the hot-leg saturation limit will be encountered before the HTP quality limit. The XCOBRA-IIIC code prints a warning message when the quality limit is exceeded, to alert the analyst to this problem if it occurs.

Question 3 - Describe the power excursion transient used to determine the transient-compensated fuel temperature adjustment.

Response - The OPΔT trip is designed to protect against slowly evolving events in which transient fuel temperature overshoot is not a significant concern. It is not designed to protect against the quickly evolving events which result in significant power overshoots. Inclusion of a transient compensation bias is thus inconsistent with the trip design basis.

Transient fuel centerline temperature will be verified in the transient analyses on an event by event basis. Since the effect of power overshoot is evaluated on an event-by-event basis in the transient analyses, a transient overshoot bias compensation is not required in the OPΔT trip analysis.

Question 4 - Discuss how the selection of the high pressurizer and the hot-leg saturation temperature results in a maximum value for K_6.

Response - In general, vessel ΔT at a given power level varies reciprocally with the specific heat of water. As temperature increases at given power, the specific heat of water increases, and vessel ΔT decreases. The maximum temperature achievable at a given pressure is the saturation temperature. Thus, for given power level and pressure, the saturation temperature results in the maximum value of specific heat and the minimum value of vessel ΔT.

As pressure increases, the saturation temperature increases. The maximum temperature permitted by the OTΔT setpoint is the saturation temperature at the maximum operating pressure. Selection of the maximum operating pressure and corresponding saturation temperature maximizes the value of the specific heat, and minimizes the value of the vessel ΔT at given power level. By inspection of Equation 2.8, Page 8, of Reference 1, a minimum value of ΔT yields a maximum value of K_6.

Because the inlet coolant density decreases at higher pressures (assuming saturated conditions at the hot leg) due to the higher saturation temperatures, the resulting decrease in mass flow rate through the core will tend to increase the ΔT at a given power level as pressure increases. The effect of higher pressure on specific heat is the stronger dependence, and dominates the effect of pressure on the computed ΔT. Hence, the effect of the thermodynamic properties of water are such that selection of the highest pressure achievable in the core (at saturated conditions) leads to the desired goal of maximizing the value of K_6.

Question 5 - Maximizing K_6 appears to provide a conservative trip for $T_{avg} > T'$ and a non-conservative trip for $T_{avg} < T'$. How will a conservative OPΔT trip be insured for $T_{avg} < T'$?

Response -

as noted in typical plant Technical Specifications.

ensures that the trip function is conservative, because the trip circuitry ensures that $\Delta T/\Delta T_0$ is always equal to or less than K_4.

Question 6 - How will the uncertainties and their distributions of Tables 2.1, 3.1, and 3.2, and the trip processing and time delays of Table 4.1 be verified in plant-specific applications?

Response -

are typically supplied by the utility based on information provided by the reactor vendor. In some plant applications, the Technical Specification total statistical allowance (TA) associated with a reactor trip channel subsumes these uncertainties, and is used in their stead. These uncertainties are characteristic of the reactor measurement and instrumentation systems, and are not affected by changes in the fuel design or plant operating point. They continue to be applicable without the need for further verification.

The power peaking measurement uncertainties are typically provided in the plant Technical Specifications. SPC's adoption of these uncertainties is justified on a generic basis as part of SPC's approved neutronics methodology.

The DNB correlation uncertainty listed in Table 3.1 is justified by comparison to DNB test data. The DNB correlation statistics resulting from this comparison are described in a topical report that is reviewed by the NRC.

The trip processing and time delays listed in Table 4.1 may be treated deterministically in the statistical transient analyses. In this case, the bounding values are typically taken from information supplied by the reactor vendor or from the plant Technical Specifications. These bounding values are characteristic of reactor systems, and are not expected to change with fuel design or plant operating point. They continue to be applicable without the need for further justification.

In other cases, the trip processing and delay times are treated statistically. Statistical analyses of plant systems data are performed to justify the uncertainties that are to be used in the statistical transient analyses. The uncertainty distribution and its descriptive statistics result from these analyses.

In most cases, the uncertainties for the reactor measurement and instrumentation systems have been reported only as a simple uncertainty band. The uncertainty distributions and their descriptive statistics are typically not provided by the reactor vendor. SPC therefore makes the following conservative assumptions about the uncertainty distributions and their descriptive statistics:

Question 7 - Describe in detail the method used to calculate the joint probability distribution and the ΔI-dependent power limit for determining the OPΔT and OTΔT F(ΔI) reset functions?

Response -

Uncertainties in the Trip at Fixed Flux Difference

The OTΔT trip calculates a hot leg-to-cold leg temperature difference at which the reactor should be tripped to protect against DNB at a 95/95 level. The trip uses the average of the hot leg and cold leg temperatures, the pressurizer pressure and the axial flux difference to calculate the trip setpoint.

$$\delta T = \delta T_0 * (k_1 - k_2 * (T_{AVE} - T_{AVE0}) + k_3 * (P - P_0) - F(\delta I)),$$

where the subscript "0" denotes nominal conditions and F(δI) is the adjustment for the axial flux difference.

The OPΔT trip is quite similar, although somewhat simpler, and protects against fuel centerline melt. In general the description below, which is directed towards the OTΔT trip, applies to this trip also.

A probability distribution is determined for δT which utilizes all uncertainties except those associated with the measurement of the axial flux difference.

A response surface-Monte Carlo method is used for this purpose.

Statistical Combination of δT and δI Uncertainties

In addition to the probability distribution for δT, a second probability distribution, is created to The appropriate measure of the proximity of the OTΔT trip to the MDNBR limit is determined by evaluating a combination of these two probability distributions which results in protection at the 95/95 limit.

The joint probability distribution is written, in general, as an integral over the joint probability density function.

$$F_J(\Delta T, \Delta I) = \int f_J(x,y)\ dxdy\ ,$$

where the integration over x and y is over the domain bounded by ΔT and ΔI. The appropriate interpretation of the argument (variable greater or less than argument) depends on which direction from the nominal point the distribution is being evaluated. In this example, it will be assumed that $\delta T < \Delta T$ and that $\delta I < \Delta I$ for simplicity. For the other three cases, the domain of integration, as reflected in the limits used below, would need to be altered

The one-sided 95/95 limit is established by setting this joint probability equal to 0.05.

Question 8 - In determining the reference point for the (setpoint and transient analysis) response surfaces,

Response - The individual NOMSCAN/DETSCAN differences are combined

Question 9 - Provide the form of the response surfaces for ΔT_{DNB} of Equation-3.4, ΔT_{sat} of Equation-3.5, ΔT_{unc} of Equation-3.6, and the ΔTRIP for the Chapter 4 transient analysis.

Response - The form of the various response surfaces is selected at the discretion of the analyst. The bases for the selection include the fitting statistics, the residual plots, and a plot of the response variable vs. the data points to which the response surface is fit. The subsequent Monte Carlo analyses include an allowance for response surface fitting error. Care is taken to ensure that the experimental design is sound, and that the response surface is of sufficiently low order in the independent variables to preclude overfit.

Some of the terms in some instances have been rejected because they do not substantially enhance the fit. The analyst is free to use any response surface form that provides a fit of the requisite accuracy.

Question 10 - Provide a comparison of the W and SNPC OPΔT and OTΔT setpoints and discuss any significant differences.

Response - The methodology will be applied to verify an existing setpoint or to develop a new setpoint. In either case, the fundamental form of the setpoint equations remains intact, and the transient elements of the setpoint are unchanged. The values of the K_1, K_2, and K_3 constants in the OTΔT setpoint and the values of the K_4 and K_6 constants in the OPΔT setpoint may be revised in developing new setpoints. In that case, the supporting analyses will utilize the Reference 1 methods to ensure a conservative setpoint. In verification applications, the value of the K_1, K_2, K_3, K_4 and K_6 constants will remain unchanged as well. Thus, SPC and W utilize the same form for the setpoints and develop or evaluate them from equivalent design bases.

Question 11 - Are the DNB and fuel melt codes and methods used to determine the ΔT setpoints approved for application to W plants?

Response - Yes, the codes, correlations, and methodologies utilized in the ΔT setpoint analyses are individually submitted to the NRC for approval. The RODEX2 code is used to determine fuel melt limits. It is approved for application to SPC fuel designs. DNB calculations are performed using SPC approved thermal hydraulic methodologies, and will primarily be performed with the HTP DNB correlation, which is at present under NRC Staff review. SPC's approved XNB correlation may be utilized in some cases. DNB correlations approved in the future may be used as well.

Question 12 - How is it assured that the limiting operating conditions have been included in the determination of the axial power shapes?

Response - To assure that the limiting operating conditions have been covered, the axial power shapes are generated

The axial xenon shapes are obtained from xenon oscillation simulations

The xenon oscillation simulations are induced by inserting rods

then rapidly removing them. The change in the xenon distribution during the following hours produces axial xenon shapes

achievable during operation. These adverse xenon shapes are then applied to rodded and unrodded core configurations

This produces axial power shapes which are more limiting than those which are reasonably achievable.

The axial power shapes are highly dependent on the power level at which they are generated. In the development or verification of the OPΔT F(ΔI) function,

These are the axial distributions for which the OPΔT F(ΔI) trip reset action is required.

Question 13 - The shape of the F(ΔI) function typically depends on the power level. How is this dependence accounted for in the NOMSCAN calculation of the OTΔT F(ΔI) reset function?

Response - Of the seven W reactors that SPC has fueled, six have no power dependence in their F(ΔI) trip reset functions.

In one reactor, the F(ΔI) trip reset does permit a broadening of the deadband at power levels below rated for steady-state operation. If a power excursion event occurs, the deadband will automatically narrow to the rated power range as power approaches rated, causing an appropriate reset of the trip setpoint. Limiting transient events tend to terminate at power levels at or above rated. Consequently, the broader deadband permitted at reduced power levels is not expected to significantly affect the performance of the trip during limiting transient events that trip on OTΔT.

In the event that a limiting event terminates on an OTΔT trip at a power level below rated, the DNBR may be evaluated

In this way, a conservative evaluation of the event and the OTΔT performance is assured. Should DNBR limits not be met, a revision to the F(ΔI) function would be required.

The F(ΔI) gain, G, establishes the required trip reset for a given excursion of ΔI beyond the deadband. For plants in which deadband relaxation at reduced power is permitted, the relaxation is computed using the same gain. The same DNB safety margin is preserved with the increased deadband at reduced power as is preserved for an equivalent ΔI increase at or above rated power.

For the W reactors with which SPC has experience, no additional treatment of the F(ΔI) reset is required for reduced power operation. This is because either the deadband is not relaxed at reduced power, or because the deadband is relaxed at a rate which preserves the margin to the DNB safety limit.

Question 14 - How will this methodology be applied to mixed core loadings with fuel from multiple vendors?

Response - SPC setpoint analyses are structured to support SPC fuel operating in a mixed core environment. These analyses employ SPC's approved mixed core thermal hydraulic methodology. Usually, the previous vendor's analyses are relied upon to support the non-SPC fuel in the core. Existing Technical Specification power peaking limits are typically retained for the non-SPC fuel in the mixed core, to ensure the continued applicability of the previous vendor's analyses. The DNB and fuel centerline temperature effects of diversion cross flow between assemblies in the mixed core are explicitly considered in each application, to ensure that the previous vendor's fuel remains within the bounds of its supporting safety analysis.

Question 15 - What is the typical power level implicit in the $\Delta I = 0$ OPΔT setpoint?

Response - The power level at the OPΔT setpoint with zero F(ΔI) compensation is computed for each application of the OPΔT setpoint methodology. A recent application of the methodology resulted in a power level of rated, corresponding to an uncertainty-adjusted K_4 value

Question 16 - Is the equation presently used to determine the SGSV opening employed in the SNPC methodology? If not, justify any differences.

Response - If plant operating conditions and steam generator tube plugging levels are essentially unchanged, the SGSV limit line may be taken from previous vendor's analyses. If tube plugging levels increase, the line is adjusted

Alternatively, the line may be developed as described below.

The SGSV limit line provides an upper limit on the coolant average temperature based on the power level and the secondary side saturation temperature at the steam generator safety valve relief setpoint. The SGSV limit on T_{avg} is fixed by the following equation:

$$q=(UA)*\Delta T_{P-S}$$

where ΔT_{P-S} is the classical log mean temperature difference, (UA) is the overall primary to secondary heat transfer coefficient, and q is the heat transferred per unit time. The temperatures employed in computing the ΔT_{P-S} are the primary hot and cold leg temperatures, the secondary inlet temperature, and the secondary steam temperature. The secondary steam temperature is equal to the saturation temperature at the maximum uncertainty-adjusted steam generator safety valve setpoint pressure.

The primary average coolant temperature is computed as the mean of the hot and cold leg temperatures. The hot and cold leg temperatures are consistent with the power and flow levels. The primary-secondary heat transfer coefficient (UA) is deduced from known plant operating conditions. The average primary coolant temperature is determined at various power levels, and mapped onto the T_{avg} - ΔT coordinate system.

Question 17 - Describe the transient calculations used to determine the OPΔT trip dynamic compensation factor K_5. Why is the indicated value of K_5 a factor of about 10 larger than typical values?

Response - In general, SPC does not intend to either develop or to modify the K_5 constant. The first two paragraphs of page 5 of the reference 1 document discuss the transient signal compensation for the OPΔT trip. The last sentence of the first paragraph is intended to indicate that SPC will not develop or modify these time constants or K_5 under the present methodology.

The "1/°F" that appears in the setpoint descriptions is intended to indicate the proper units for the particular constant, rather than a specific value.

Question 18 - What transients will be analyzed with the statistical methods of Chapter 4? What trips, other than the OPΔT and OTΔT trips, will be determined or verified using these analyses?

Response - The methods of Chapter 4 are intended to apply to limiting DNB transients and limiting pressurization (depressurization) events.

Question 19 - The OPΔT plant-equipment related time constants τ_1, τ_2, τ_3, and τ_6 of Table 6.2.2 (which are taken to be zero) and the OTΔT dynamic compensation time constants τ_4, and τ_5 of Table 6.2.3 are substantially smaller than typical values selected for these constants. What is the basis for these specific values and how will they be determined in plant-specific applications?

Response - The plant-specific equipment related time constants are provided by the utility from information supplied by the reactor vendor. The SPC methodology is not applied to develop or change the values for the various time constants in the OPΔT and OTΔT trips. The adequacy of the existing time constants is verified in the plant transient analyses.

Question 20 - In the rod control cluster drop and loss of flow events, how are axial power distributions more severe than the design axial distribution accounted for? How does this differ from the W approach?

Response - For the loss of flow event, which does not ordinarily result in an OTΔT reactor trip,

For rod control cluster assembly (RCCA) drop events that do not result in an OTΔT trip or do not involve significant rod bank motion,

The axial might also be conservatively selected as is described below for RCCA drop events which result in an OTΔT trip.

For those RCCA drop events which result in an OTΔT trip, the most limiting axial at the terminal power level of the event is considered;

The generation of these axial shapes is discussed in the response to Question 12.

Question 21 - Is the RODEX2 calculated fuel melt LHGR adjusted to account for transient effects? How are the uncertainties in the RODEX2 calculation of the fuel melt LHGR accounted for?

Response - The RODEX2 code is essentially a steady-state code. The RODEX2 fuel temperature predictions are

This conservatism is deemed sufficient to account for the uncertainties inherent in fuel temperature prediction. The approved HUXY code may be used for transient fuel temperature calculations.

SIEMENS

September 24, 1993
RAC:93:147

Mr. Larry Kopp
Reactor Systems Branch
Division of Engineering and System Technology
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Kopp:

Responses on the SPC Statistical Transient Methodology

Reference: Letter, R. A. Copeland (SPC) to T. E. Murley (USNRC), RAC:065:92, May 29, 1992.

Attached are the responses to the additional questions on the Siemens Power Corporation Statistical Transient Methodology currently under your review.

Please consider the information in these responses to be proprietary to Siemens Power Corporation. The affidavit supplied with the original submittal (Reference) should satisfy the requirements of 10 CFR 2.790(b) to support the withholding of the attachment from public disclosure.

If there are questions, or if I can be of further help, please call me at (509) 375-8290.

Very truly yours,

Bob Copeland

R. A. Copeland, Manager
Product Licensing

/smg

Attachment

cc: Dr. J. Carew (BNL)

bcc: F. T. Adams
D. M. Brown
R. C. Gottula
J. S. Holm
L. D. O'Dell
H. G. Shaw
File

Siemens Power Corporation

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RESPONSES ON THE SPC STATISTICAL TRANSIENT METHODOLOGY

Question 1: Please provide assurance that all combinations of axial power distribution and power level are protected by the OTΔT F(ΔI) trip reset function generated under the statistical setpoint methodology.

Response:

Question 2: Please support the conservatism of using
criterion in the statistical transient analyses for the pressurizer pressure trip and the low flow trip.

Response:

SIEMENS

October 5, 1993
RAC:93:158

Mr. L. I. Kopp
Reactor Systems Branch
Division of Engineering and System Technology
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Kopp:

Response to Question on EMF-92-081(P)

Reference: Letter, R. A. Copeland (SPC) to T. E. Murley (USNRC), RAC:065:92, May 29, 1992.

Attached is the response to the additional question Dr. Carew had about the statistical transient methodology currently under your review.

Please consider the information contained in this response to be proprietary to Siemens Power Corporation. The affidavit supplied with the original submittal (reference) should satisfy the requirements of 10 CFR 2.790(b) to support the withholding of the attachment from public disclosure.

If there are questions, or if I can be of further help, please call me at (509) 375-8290.

Very truly yours,

Bob Copeland

R. A. Copeland, Manager
Product Licensing

/smg

Attachment

cc: Dr. J. Carew (BNL)

Response to Question on EMF-92-081(P)

Question 1: In determining the reference point for the OTΔT transient analysis response surfaces, are the individual NOMSCAN/DETSCAN differences combined using an algebraic or a statistical root-mean-square (rms) method? If the statistical root-mean-square method is not used, provide the basis for concluding that the reference point yields a bounding rms-statistical uncertainty for all points of interest.

Response: The individual NOMSCAN/DETSCAN differences are combined as

In typical applications, the arithmetic criterion and the RMS criterion yield essentially the same result, as discussed below. The arithmetic criterion is considered to be an acceptable alternate to the RMS criterion in applications.

FAX NO. 1608
PAGE 2 OF 3

Table 1 Uncertainties and Sensitivities Considered in DNB Response Surfaces
(Maximum Flow Sensitivity)

Table 2 Uncertainties and Sensitivities Considered in DNB Response Surfaces
(Minimum Flow Sensitivity)

EMF-92-081(NP)(A)

EMF-92-081(NP)(A)
Supplement 1

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STATISTICAL SETPOINT/TRANSIENT METHODOLOGY

FOR WESTINGHOUSE TYPE REACTORS

Distribution

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