REACTOR ENGINEERING

TU ELECTRIC
Generating Division

RXE-90-006-NP

Power Distribution Control Analysis
and
Overtemperature N-16 and Overpower N-16
Trip Setpoint Methodology

February 1991

John T. Bosma
Mark A. Grace

Reviewed: Mickey R Killgore Date: 2/24/91
Mickey R. Killgore
Supervisor, Reactor Physics

Reviewed: Stephen M. Maier Date: 2/25/91
Stephen M. Maier
Supervisor, Transient Analysis

Approved: Date: 2-25-91
Randall L. Janne
Manager, Nuclear Fuel

Approved: Ausaf Husain Date: 2/25/91
Ausaf Husain
Director, Reactor Engineering

PROPRIETARY INFORMATION

This document contains information proprietary to Advanced Nuclear Fuels Corporation; it is submitted in confidence and is to be used solely for the purpose for which it is furnished and returned upon request. This document and such information is not to be reproduced, transmitted, disclosed or used otherwise, in whole or in part, without authorization of Advanced Nuclear Fuels Corporation.

Advanced Nuclear Fuels Corporation
600 108th Avenue NE
P.O. Box 90777
Bellevue, WA 98009-0777

DISCLAIMER

The information contained in this report was prepared for the specific requirements of Texas Utilities Electric Company (TUEC), and may not be appropriate for use in situations other than those for which it was specifically prepared. TUEC PROVIDES NO WARRANTY HEREUNDER, EXPRESSED OR IMPLIED, OR STATUTORY, OF ANY KIND OR NATURE WHATSOEVER, REGARDING THIS REPORT OR ITS USE, INCLUDING BUT NOT LIMITED TO ANY WARRANTIES ON MERCHANTABILITY OR FITNESS FOR A PARTICULAR PURPOSE.

By making this report available, TUEC does not authorize its use by others, and any such use is forbidden except with the prior written approval of TUEC. Any such written approval shall itself be deemed to incorporate the disclaimers of liability and disclaimers of warranties provided herein. In no event shall TUEC have any liability for any incidental or consequential damages of any type in connection with the use, authorized or unauthorized, of this report or the information in it.

TABLE OF CONTENTS

PAGE

PAGE

PAGE

LIST OF TABLES

LIST OF FIGURES

FIGURE		PAGE

FIGURE PAGE

FIGURE PAGE

CHAPTER 1

INTRODUCTION

In order to support reload design, licensing, and operation of Comanche Peak Steam Electric Station (CPSES) Units 1 and 2, an Overpower N-16 and Overtemperature N-16 trip setpoint methodology, along with a power distribution control analysis methodology, is required.

The Overpower N-16 and Overtemperature N-16 trip setpoints are designed to protect against fuel centerline melting, departure from nucleate boiling (DNB), and hot leg saturation during postulated transients (ANSI N18.2 Condition II events). The design bases and the methodology for the calculation of the Overpower N-16 and Overtemperature N-16 trip setpoints, including the treatment of the core power distribution effects and trip channel uncertainties, are presented in this report. The calculations required to define the power distribution effects are based on the methodology developed and employed by [

].

Power distribution control is utilized to assure that the axial power distributions and associated peaking factors during normal operation are maintained within the limits assumed within the safety analyses. The basic concept of the power distribution control procedures is that the variation in the core axial power distribution during reactor operation can be controlled by maintaining the axial flux difference within predefined target bands. TU Electric has adopted the principal attributes of the [ ] for power distribution control analysis.

CHAPTER 2

SUMMARY

The power distribution control analysis and Overpower and Overtemperature reactor trip setpoint methodology which will be utilized in support of design, licensing, start-up, and operation of CPSES Units 1 and 2 has been defined.

The methodology used for calculating the safety settings for the Overpower N-16 (OP N-16) and Overtemperature N-16 (OT N-16) reactor trip functions is described, including the determination of compensation terms. The compensation terms are a function of the axial flux difference and account for variations in the axial power distributions.

The Overpower N-16 reactor trip setpoint is developed to limit the maximum linear heat generation rate in the fuel, thereby precluding fuel centerline melt during postulated transients. The overpower protection limit is first established, then the compensation term is determined. Based on the analytical results obtained to date, a single overpower limit will preclude fuel centerline melt for all anticipated axial power distributions; therefore, it is anticipated that the compensation term will not be required for CPSES.

The Overtemperature N-16 reactor trip setpoint has been developed to provide primary protection against departure from nucleate boiling (DNB) and hot leg saturation during postulated transients. The safety settings for the Overtemperature N-16 reactor trip are selected assuming a fixed reference axial power distribution; then, compensation terms are determined which account for variations in the axial power distribution.

The variation in the axial power distribution with axial flux difference must be determined to establish normal operating limits as well as to quantify the Overpower N-16 and the Overtemperature N-16 compensation terms. TU Electric has adopted the principal attributes of [ ] for generation of the axial power distributions required for both applications. Using this prescriptive methodology, enveloping sets of axial power distributions are developed using [ ].

[ ]

[

].

I addition, a method is presented to determine an expanded axial flux difference range in which reactor operation is permitted for periods of short duration. It is demonstrated that operation outside the specified axial flux difference target band, but within the expanded range, is permitted for periods of short duration since control of the variation in the power distribution is maintained.

The methodology described in this report allows TU Electric to establish the Overpower N-16 and Overtemperature N-16 trip setpoints including the compensation terms. In addition, the power distribution control analysis is presented which allows TU Electric to establish axial flux difference target bands and define normal operating power distribution shapes for input to Loss of Coolant Accident (LOCA) and non-LOCA transient analyses.

CHAPTER 3

CORE THERMAL OVERPOWER AND OVERTEMPERATURE PROTECTION

The bases of the Overpower Nitrogen-16 (OP N-16) and the Overtemperature N-16 (OT N-16) protection trips and the methods used by TU Electric to determine the trip setpoints are presented in this chapter. These trip functions provide primary protection against departure from nucleate boiling (DNB), hot-leg boiling, and fuel centerline melting (excessive linear heat generation rate) during postulated transients (ANSI N18.2 Condition II events).

3.1 Functional Description

3.1.1 Nitrogen-16 Monitoring System

The Nitrogen-16 (N-16) activity in the primary coolant water is a process parameter used for continuous measurement of the reactor power level. The N-16 activity is formed by high energy (>10 Mev) neutron activation of Oxygen-16 contained in the water. The N-16 activity in the primary coolant is directly proportional to the integrated fast neutron flux throughout the core and therefore provides a means to directly measure the total fission rate and total core power.

The N-16 activity in the primary coolant is monitored by measuring, outside each of the four hot-leg pipes, the gamma radiation resulting from the decay of the N-16. This N-16 activity in each hot-leg is converted to a power signal that is used by the OP N-16 and OT N-16 protection trip functions.

3.1.2 Overpower N-16 Trip Description

The thermal overpower protection function will trip the reactor when the measured N-16 power in any two out of four channels exceeds the setpoint (one channel per reactor coolant loop). The OP N-16 setpoint is given by the following equation:

$$Q_{sp} = K_4 - f(\Delta I) \qquad (3\text{-}1)$$

where:

Q_{sp} = The OP N-16 setpoint which is compared to the measured N-16 power signal, fraction of Rated Thermal Power (RTP)

K_4 = A preset, manually adjustable bias, fraction of RTP

$f(\Delta I)$ = A function of the neutron flux difference between the average of the upper two and the average of the lower two calibrated ex-core ion chamber readings, fraction of RTP.

The neutron flux information for the $f(\Delta I)$ function is provided by the four-section power range neutron flux detectors. The flux difference, ΔI, is defined as the average of the upper two calibrated signals (P_t) minus the average of the lower two calibrated signals (P_b).

3.1.3 Overtemperature N-16 Trip Description

The thermal overtemperature protection function will trip the reactor when the measured N-16 power in any two out of four channels exceeds the setpoint (one channel per reactor coolant loop). The OT N-16 setpoint is given by the following equation:

$$Q_{sp} = K_1 - K_2 \left(\frac{1+\tau_1 s}{1+\tau_2 s} T_c - T_c^{o}\right) + K_3 (P - P^{o}) - f(\Delta I) \quad (3\text{-}2)$$

where:

Q_{sp} = The OT N-16 setpoint which is compared to the measured N-16 power signal, fraction of RTP

K_1 = Preset manually adjustable bias, fraction of RTP

K_2 = Preset gain which compensates for the effect of temperature on the design limits, fraction of RTP/°F

K_3 = Preset gain which compensates for the effect of pressure on the design limits, fraction of RTP/psi

T_c = Measured value of cold leg temperature, °F

T_c^0 = Reference cold leg temperature at RTP, °F

P = Measured value of pressurizer pressure, psig

P^0 = Reference pressurizer pressure at RTP, psig

$\frac{1+\tau_1 s}{1+\tau_2 s}$ = Lead/Lag compensation on measured cold leg temperature

$f(\Delta I)$ = A function of the neutron flux difference between the average of the upper two and the average of the lower two calibrated ion chamber readings, fraction of RTP

The three process parameter inputs to the OT N-16 equation are generated as follows. The fast response in-line T_c measurement is provided by a resistance temperature detector (RTD) in a thin wall thermowell installed in the reactor coolant cold leg piping in each loop. The lead/lag function compensates for inherent instrument delays and piping lags between the reactor core and the loop temperature detectors. The adequacy of the lead/lag compensation in the reactor trip setpoints is supported by the transient analyses. The required one pressurizer pressure parameter per loop (or channel) is obtained from separate sensors connected to pressure taps at the top of the pressurizer. The neutron flux difference information for the OT N-16 trip is the same as that for the OP N-16 trip.

3.2 Overpower N-16 Trip

The Overpower N-16 trip setpoint is determined by first calculating maximum linear heat generation rate (LHGR) that does not result in fuel centerline melt, $LHGR_{clm}$. For each of the axial power distributions calculated, the peak pellet linear heat generation rate, $LHGR_{pp}$, is determined and compared to the $LHGR_{clm}$. If the $LHGR_{pp}$ exceeds the $LHGR_{clm}$ limit, a flux difference trip reset function, $f(\Delta I)$, is calculated which will effectively reduce the OP N-16 trip setpoint to protect the reactor from variations in power distributions that may potentially violate the $LHGR_{clm}$ limit.

3.2.1 Calculational Bases

The purpose of the thermal overpower protection system trip is to prevent fuel centerline melting during Condition II events. The fuel temperature design basis is stated in the FSAR as follows: during modes of operation associated with Condition I and Condition II events, there is at least a 95 percent probability that the peak kW/ft fuel rods will not exceed the UO_2 melting temperature at the 95 percent confidence level. By precluding UO_2 melting, the fuel geometry is preserved and possible adverse effects of molten UO_2 on the cladding are eliminated.

To preclude fuel centerline melting, an overpower limit corresponding to a fuel centerline temperature of 4700 °F is selected. This limit is significantly less than the actual UO_2 melting temperature for design fuel burnups $\leq$ 60,000 MWD/MTU and is consistent with the current licensing basis as presented in the FSAR.

3.2.2 Calculation of the Overpower N-16 Setpoint

Fuel centerline melt is precluded by limiting the $LHGR_{pm}$ to a design value less than the $LHGR_{clm}$ limit. This limit on the LHGR is used in justifying the core protection limit, P_{cpl}, for the Overpower N-16 reactor trip setpoint. The core protection limit is set at a fractional power level given by the following equation:

$$P_{cpl} \leq \frac{LHGR_{clm}}{LHGR_{RTP} * F_Q} \qquad (3\text{-}3)$$

where:

P_{cpl} = core protection limit, fraction of RTP

$LHGR_{clm}$ = peak pellet LHGR limit which precludes centerline melt, kW/ft

$LHGR_{RTP}$ = core average LHGR at RTP, kW/ft

F_Q = design total power peaking factor, including uncertainties

Typically a core protection limit of 1.18 is selected as a bounding value to preclude fuel centerline melting. Thus, for the Overpower N-16 setpoint equation (equation 3-1), a K_4 value of 1.18 represents the maximum allowable power fraction during operation. Allowances for core axial power distribution effects and trip channel uncertainties are described in Sections 3.2.3 and 3.2.4, respectively.

3.2.3 Core Power Distribution Effects

The OP N-16 setpoint equation contains a compensation term, f(ΔI), that accounts for the effects of highly skewed axial power distributions. The f(ΔI) trip reset function effectively reduces the OP N-16 setpoint to protect the reactor from variations in the axial power distribution.

A set of axial power distributions is developed [

]

[

]. A detailed discussion on the generation of the axial power distributions is contained in Section 3.4. For each of the axial power distributions, [

]:

[ ] (3-4)

[

]:

[ ] (3-5)

solving for [ ] yields:

[ ] (3-6)

If the analysis shows that [ ], then the core protection limit, or K_4 value in the OP N-16 setpoint equation, is adequate to protect the reactor, and development of the f(ΔI) trip reset function is not required. It is anticipated that the f(ΔI) trip reset function will not be required to preclude fuel centerline melting for CPSES when a core protection limit of 1.18 is selected. However, for completeness, the approach used to determine the f(ΔI) trip reset function is described in the following steps:

1. For each [ ] calculated, determine the axial offset (AO) and flux difference (ΔI):

   $$AO = (P_T - P_B) / (P_T + P_B) \qquad (3\text{-}7)$$

   $$\Delta I = (P_T - P_B) \qquad (3\text{-}8a)$$

   or alternatively:

   $$\Delta I = [\quad] * AO \qquad (3\text{-}8b)$$

where:

AO = axial offset, fraction

$\Delta I =$ flux difference, fraction

$P_T =$ power in the top of the core, fraction of RTP

$P_B =$ power in the bottom of the core, fraction of RTP

The values of AO and ΔI are typically multiplied by 100 to permit discussion in terms of percent.

[

].

A sample calculation using the Overpower N-16 setpoint methodology was performed using design data for CPSES Unit 1, Cycle 1. For each of the axial power distributions generated (Section 3.4), the $LHGR_{pp}$ was calculated. Figure 3.3 displays a plot of the $LHGR_{pp}$ versus axial flux difference. For this sample calculation, the calculated $LHGR_{pp}$ never exceeded the $LHGR_{clm}$ limit; therefore, the [ ] calculation was not performed. The data presented in Figure 3.3 demonstrates that the Overpower N-16 trip setpoint with K_4 = 1.18 adequately protects the core from exceeding the $LHGR_{clm}$ limit for all axial power distributions generated; therefore, the f(ΔI) trip reset function is zero. Over 4,300 axial power distributions were analyzed in this Overpower N-16 setpoint calculation.

3.2.4 <u>Overpower N-16 Trip Channel Uncertainties</u>

The OP N-16 setpoint equation calculated in Sections 3.2.2 and 3.2.3 represents the maximum allowable power during operation. However, the final OP N-16 setpoint, for inclusion into the Technical Specifications, is determined by adjusting K_4 and f(ΔI) based on appropriate allowances for uncertainties and equipment and measurement errors. These allowances are calculated as part of the statistical setpoint study.

3.3 Overtemperature N-16 Trip

The calculation of the OT N-16 trip setpoint consists of three steps. In the first step, core safety limit curves are calculated based on DNB and hot-leg boiling considerations assuming a reference core axial power distribution. In the second step, the OT N-16 setpoint equation terms are calculated based on the Core Safety Limit curves, the OP N-16 setpoint, and physical limitations on temperature due to the opening of the steam generator safety valves. The third step determines the compensating term, f(ΔI), which accounts for core axial power distributions more severe with respect to DNB than the reference power distribution.

3.3.1 Calculational Bases for Core Safety Limits

The OT N-16 trip function is designed to ensure operation within the DNB design basis and the hot-leg boiling limit during Condition II events. The DNB design basis is stated in the FSAR as follows: there will be at least a 95 percent probability that DNB will not occur on the limiting fuel rods during Condition II events at the 95 percent confidence level (95/95). DNB is evaluated in terms of the departure from nucleate boiling ratio (DNBR), which is the ratio of the predicted DNB heat flux calculated by an empirical correlation to the actual local heat flux. Standard

TU Electric DNB analysis methodology (1) is used to calculate the DNBR for this application. The criterion on DNBR is that the minimum DNBR in the hot channel must not be less than the design limit value. The DNBR design limit value is higher than the correlation 95/95 DNBR limit to provide sufficient margin to offset DNBR penalties resulting from plant and cycle-specific considerations and generic issues. The specification of this design DNBR limit value and the evaluation of the margin penalties is beyond the scope of this submittal.

The hot-leg boiling limit is not a core protection limit, but was originally established to ensure that the temperature difference across the reactor vessel (ΔT) remains proportional to the reactor power, since the Westinghouse Overtemperature and Overpower protection systems were based on ΔT. Although Comanche Peak Units 1 and 2 do not use ΔT in the Overtemperature and Overpower protection system, the hot-leg boiling limit is conservatively maintained to ensure that there are no voids in the hot-leg which could potentially affect the N-16 signal.

The DNB and hot-leg boiling limits are dependent upon, for a given volumetric flow rate, the reactor power, coolant temperature, and pressure. Reactor Core Safety Limit curves define a region of permissible operation and are determined for a range of reactor operating conditions assuming a reference core axial power distribution. The effects on the DNB limits due to core axial power distributions which differ from the reference core axial power distribution are accounted for with the $f(\Delta I)$ trip reset function (see Section 3.3.3).

The reactor Core Safety Limits are curves of reactor vessel inlet temperature versus reactor power fraction which satisfy the DNB and hot-leg boiling limits (Figure 3.4). The reactor Core Safety Limit curves are calculated for a range of pressures between the low and high pressurizer pressure reactor trip setpoints.

The DNB limit line is determined using the standard TU Electric DNB analysis metnodology as described in Reference 1. The core DNB limits are generated with the following assumptions:

1. A reference core axial power distribution is used (typically, a chopped cosine with a peak to average power ratio of 1.55). The effect of other axial power distributions is presented in Section 3.3.3.
2. The nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, is increased at power levels below RTP by the following equation:

$$F^N_{\Delta H} = F_{\Delta H}^{RTP} (1 + PF_{\Delta H} * (1 - P)) \qquad (3\text{-}9)$$

where:

$F_{\Delta H}^{RTP}$ = the $F^N_{\Delta H}$ design limit at RTP (typically 1.55)

$PF_{\Delta H}$ = the part power multiplier (typically .2 or .3)

P = the fraction of RTP

For power levels at or above RTP, the $F^N_{\Delta H}$ is equal to $F_{\Delta H}^{RTP}$.

3. The primary coolant flow is assumed to be the Thermal Design Flow.
4. The maximum core bypass flow is assumed.

The hot-leg boiling limit lines are loci of points for which the reactor vessel hot-leg temperature is equal to the saturation temperature. The allowable reactor vessel inlet temperature which may lead to hot-leg saturation is calculated using an energy balance on the reactor vessel:

$$h_{in} = h_f + Q_{core}/\dot{m} \qquad (3\text{-}10)$$

where:

h_{in} = inlet enthalpy which results in hot-leg boiling, BTU/lbm

h_f = enthalpy of saturated liquid, BTU/lbm

Q_{core} = reactor core power, BTU/hr

$\dot{m}$ = reactor vessel inlet mass flow rate, lbm/hr

The hot-leg boiling limits are calculated using the Thermal Design Flow rate and are generated for the same range of pressure conditions as the DNB limit lines. Typically, at lower power and pressure, the hot-leg boiling limits are more restrictive than the DNB limits, as illustrated in Figure 3.4. At higher power (usually greater than 80 percent) and higher pressure, the DNB limit is typically the more restrictive limitation.

3.3.2 Calculation of the Overtemperature N-16 Setpoint Equation

Consistent with the current licensing basis as described in the FSAR, certain constraints limit the range over which the OT N-16 trip function must protect. The OP N-16 trip provides an upper limit on the power range, the high and low pressurizer pressure trips limit the pressure range, and the opening of the steam generator safety valves imposes an upper limit on the temperature range.

The steam generator safety valves place an upper limit on the secondary side steam temperature which is approximately constant at the saturation temperature corresponding to the safety valve pressure setpoint. The primary side temperature cannot exceed this saturation temperature plus the temperature drop across the steam generator tubes.

Therefore, the steam generator safety valves impose an upper limit on the primary side temperature.

The locus of points for the conditions when the steam generator safety valves are open is termed the "steam generator safety valve (SGSV) opening line." The fundamental log mean temperature difference equation provides the relationship between the reactor vessel inlet temperature, core power, and steam generator secondary temperature.

$$Q_{core} = UA(T_{out} - T_{in})/\ln((T_{out} - T_{in})/(T_{in} - T_s))$$
$$= \dot{m}\ (h_{out} - h_{in}) \qquad (3\text{-}11)$$

where:

- Q_{core} = core power, BTU/hr
- UA = overall heat transfer coefficient from primary to secondary side at nominal conditions, BTU/hr-°F
- T_{out} = reactor vessel outlet temperature, °F
- T_{in} = reactor vessel inlet temperature, °F
- T_s = saturation temperature of the secondary side, °F
- $\dot{m}$ = reactor vessel inlet mass flow rate, lbm/hr
- h_{out} = reactor vessel outlet enthalpy BTU/lbm
- h_{in} = reactor vessel inlet enthalpy BTU/lbm

The SGSV opening line is calculated in terms of reactor vessel inlet temperature versus core power and intersects with the Core Safety Limit lines as illustrated in Figure 3.5. The intersections of the SGSV opening line with the Core Safety Limits are determined by evaluating points along the Core Safety Limit until equation (3-11) is satisfied.

For the reference core axial power distribution and steady state conditions, the equation for the OT N-16 setpoint is:

$$Q_{sp} = K_1 - K_2 \ (T_c - T_c^{\circ}) + K_3 \ (P - P^{\circ}) \qquad (3\text{-}12)$$

where the terms are defined in equation (3-2). The calculation of the constants K_1, K_2, and K_3, in the OT N-16 equation is as follows:

1. The Core Safety Limits are plotted on a vessel inlet temperature versus core power coordinate system (Figure 3.4).
2. The OP N-16 trip set int line, plotted on Figure 3.5, shows the upper limit on power which must be considered in calculating the OT N-16 setpoint.
3. The SGSV opening line, plotted on Figure 3.5, provides the upper limit on temperature which must be considered in calculating the OT N-16 setpoint.

4. The values of K_1, K_2, and K_3 are then calculated such that the OT N-16 trip lines lie below the Core Safety Limits. The value of the constants are determined from the intersection points of the OP N-16 trip line and the SGSV opening line with the Core Safety Limits (points A_1, A_2, B_1, ..., D_2, on Figure 3.5).

For example, the core power, inlet temperature, and pressure at points A_1, A_2, and D_1, are substituted into the steady state OT N-16 setpoint equation (3-12) to provide three equations and three unknowns. The constants K_1, K_2, and K_3 are determined from solving the three simultaneous equations. In this example, the OT N-16 trip lines are parallel to points A_1 and A_2, and intersect with point D_1.

The OT N-16 setpoint equation solutions for all possible combinations of intersection points are calculated. Each equation is tested to ensure the Core Safety Limits are protected over the entire range. Generally, two or three equations are found which provide protection over the entire range. One equation is selected based on operating margin considerations. Figure 3.6 shows an acceptable set of OT N-16 trip setpoint lines.

3.3.3 Core Axial Power Distribution Effects

The OT N-16 setpoint equation is based on the reactor Core Safety Limits which are generated assuming a reference core axial power distribution (typically a 1.55 chopped cosine). Core axial power distributions that are more adverse than the reference shape, with respect to DNB, affect the reactor Core Safety Limit curves and are accounted for with the OT N-16 $f(\Delta I)$ trip reset function.

The criterion used to evaluate the core axial power distribution effects is that the DNB design basis must be satisfied. The hot-leg boiling limit is unaffected by variations in core axial power distributions.

The steady state form of the OT N-16 setpoint is given by the following equation:

$$Q_{sp} = K_1 + K_2\ (T_c - T_c^{o}) + K_3\ (P - P^{o}) - f(\Delta I) \qquad (3\text{-}13)$$

or solving for $f(\Delta I)$:

$$f(\Delta I) = K_1 + K_2\ (T_c - T_c^{o}) + K_3\ (P - P^{o}) - Q_{sp} \qquad (3\text{-}14)$$

where the terms are defined in Equation 3-2.

Standard TU Electric DNB analysis methodology is used to determine the effects of core axial power distributions using the same assumptions as those described in Section 3.3.1, except that a variety of core axial power distributions are considered rather than the assumption of a reference core axial power distribution. The standard TU Electric DNB analysis methods include the use of the VIPRE-01 computer code. However for this analysis, the DNB evaluations are first performed with a "VIPRE-equivalent" computer code to screen out the non-limiting axial power shapes. This code uses a single-channel model and includes correction factors to account for the multi-dimensional effects such as flow redistribution and cross channel mixing. The correction factors are developed based on a large number of VIPRE-01 calculations and thus allow the VIPRE-equivalent code to predict a minimum DNBR with good accuracy when compared to VIPRE-01. The advantage of this code is that the computer run time is very fast, and thus analyses for a large number of axial power distributions can quickly be performed with high accuracy. The VIPRE-01 code is then used to analyze the limiting axial power distributions.

The "new" core DNB limit line due to the effects of an adverse core axial power distribution is illustrated in Figure 3.7. As can be seen from Figure 3.7, a portion of the "new" core DNB limit line is violated (i.e., unprotected by the OT N-16 setpoint). The largest violation of the core DNB limit lines occurs at the intersection of the OT N-16 trip and the OP N-16 trip as illustrated in Figure 3.7. By keeping the slope of the OT N-16 setpoint constant and uniformly reducing the setpoint such that the OT N-16 setpoint intersects at the "new" T-in, protection is assured for the entire Core Safety Limit lines. Figure 3.8 illustrates the new OT N-16 setpoint with the axial flux difference trip reset compensation.

In order to calculate the $f(\Delta I)$ trip reset function, a set of core axial power distributions is developed which consider various combinations of [

].

Development of the power distributions is discussed in Section 3.4.

DNBR calculations are performed for each of the core axial power distributions at a set of core conditions (power, pressure, and temperature) which result in the largest potential violation of the core DNB limit lines. Figure 3.9 shows the DNBR analysis results for a sample of core axial power distributions using the limiting set of core conditions. From this figure it is observed that, within a flux difference range (i.e., deadband), the reference core axial power distribution used to generate the Core Safety

Limits bounds any calculated core axial power distribution, and no OT N-16 setpoint reduction is needed (i.e., the DNBR is greater than the limit value). Beyond the deadband, an OT N-16 setpoint reduction is required to prevent violating the DNBR limit value.

With the power level conservatively held constant at the intersection of the Overtemperature and Overpower N-16 setpoint value (typically, 118% of RTP), an iterative calculation is performed for each core axial power distribution that is more adverse than the reference core axial power distribution, with the reactor vessel inlet temperature being decreased to give a minimum DNBR which is equal to the design basis limit value. The decrease in the reactor vessel inlet temperature thus required (Required Temperature Reduction, RTR) is recorded with the flux difference (ΔI) associated with the core axial power distribution being analyzed. A bounding envelope of RTR versus axial flux difference is generated (see Figure 3.10) and is verified to be conservative over the entire pressure range. RTR is defined to be a positive quantity for a decrease in temperature.

The RTR versus axial flux difference curve is described by denoting the deadband range and slopes of the lines beyond the deadband through the following equations:

$$RTR = S_{pos}\ (\Delta I - D_{pos}) \qquad \text{for } \Delta I > D_{pos} \qquad (3\text{-}15a)$$

$$RTR = 0. \qquad \text{for } D_{neg} < \Delta I < D_{pos} \qquad (3\text{-}15b)$$

$$RTR = S_{neg}\ (\Delta I - D_{neg}) \qquad \text{for } \Delta I < D_{neg} \qquad (3\text{-}15c)$$

where:

S_{pos}, S_{neg} = Slope of the RTR lines on the positive and negative sides of the deadband respectively, °F/%ΔI

D_{pos}, D_{neg} = Deadband limits or breakpoints at the positive and negative sides of the RTR envelopes respectively, %ΔI

with RTR being defined by the following equation:

$$RTR = T_i - T_{new} \qquad (3\text{-}16)$$

where:

T_i = Reference reactor vessel inlet temperature corresponding to the intersection of the Overtemperature and Overpower N-16 setpoints, °F

T_{new} = Allowed reactor vessel inlet temperature for a particular axial flux difference to satisfy DNBR design basis, °F

Combining Equations 3-15 (a, b, and c) with equation 3-16 and solving for T_{new} gives:

$$T_{new} = T_i - S_{pos}\ (\Delta I - D_{pos}) \quad \text{for} \quad \Delta I > D_{pos} \qquad (3\text{-}17a)$$

$$T_{new} = T_i \quad \text{for} \quad D_{neg} < \Delta I < D_{pos} \qquad (3\text{-}17b)$$

$$T_{new} = T_i - S_{neg}\ (\Delta I - D_{neg}) \quad \text{for} \quad \Delta I < D_{neg} \qquad (3\text{-}17c)$$

The deadband of the $f(\Delta I)$ function corresponds to the maximum range of ΔI for which the RTR is equal to zero; therefore, no $f(\Delta I)$ trip reset is required. The slope of the $f(\Delta I)$ function is obtained by substituting T_{new} from Equations 3-17a and 3-17c for T_c in Equation 3-14 and taking the derivative of $f(\Delta I)$ with respect to ΔI.

For the positive ΔI line:

$$\frac{d}{d\ \Delta I} f(\Delta I) = \frac{d}{d\ \Delta I}(K_1 + K_2(T_i - S_{pos}(\Delta I - D_{pos}) - T_c^{\circ}) + K_3(P - P^{\circ}) - Q_{sp})$$

$$= -\ K_2\ S_{pos} \qquad (3\text{-}18)$$

For the negative ΔI line:

$$\frac{d}{d\ \Delta I} f(\Delta I) = \frac{d}{d\ \Delta I}(K_1 + K_2(T_i - S_{neg}(\Delta I - D_{neg}) - T_c^{\circ}) + K_3(P - P^{\circ}) - Q_{sp})$$

$$= -\ K_2\ S_{neg} \qquad (3\text{-}19)$$

Figure 3.11 shows an example of the f(ΔI) trip reset function. The f(ΔI) trip reset function thus ensures that the OT N-16 setpoint is reduced sufficiently to prevent DNB for core axial power distributions more severe than the reference core axial power distribution.

3.3.4 Overtemperature N-16 Trip Channel Uncertainties

The OT N-16 setpoint equation calculated in Sections 3.3.2 and 3.3.3 represents the maximum allowable power during operation. However, the final OT N-16 setpoint, for inclusion into the Technical Specifications, is determined by adjusting K_1 and f(ΔI) based on appropriate allowances for uncertainties and equipment and measurement errors. These allowances are calculated as part of the statistical setpoint study.

3.4 Generation of Axial Power Distributions

TU Electric has adopted the principal attributes of the [ ] for generating the axial power distributions used in the determination of the f(ΔI) trip reset function for the OP N-16 and OT N-16 trips. The [ ] model discussed in Chapter 5 is used to generate an [ ].

[

].

By considering [

] of axial power distributions can be obtained that will assure that the f(ΔI) trip reset function is determined based upon the most limiting core configurations. [

]

[

]

[

].

Table 3.1

Equilibrium Axial Offsets as a Function of Cycle Exposure for Comanche Peak Steam Electric Station Unit 1, Cycle 1, Demonstration Calculation

[ ]

Table 3.2

Control Bank D Insertion Positions [ ]

Reactor Power Level (%)	Control Rod Position, Steps Withdrawn 228 steps withdrawn is ARO						
	228	181	162	134	81	28	0
25	X	X	X	X	X	X	X
50	X	X	X	X	X	X	
75	X	X	X	X	X		
100	X	X	X	X			
118	X	X	X	X			

Figure 3.1 [ ] as a Function of ΔI (Typical)

[ ]

Figure 3.2 Determination of the f(ΔI) Trip Reset Function for the OP N-16 Trip Setpoint (Typical)

Figure 3.3 Calculated Peak Pellet LHGR Versus Axial Offset, Comanche Peak Steam Electric Station Unit 1, Cycle 1

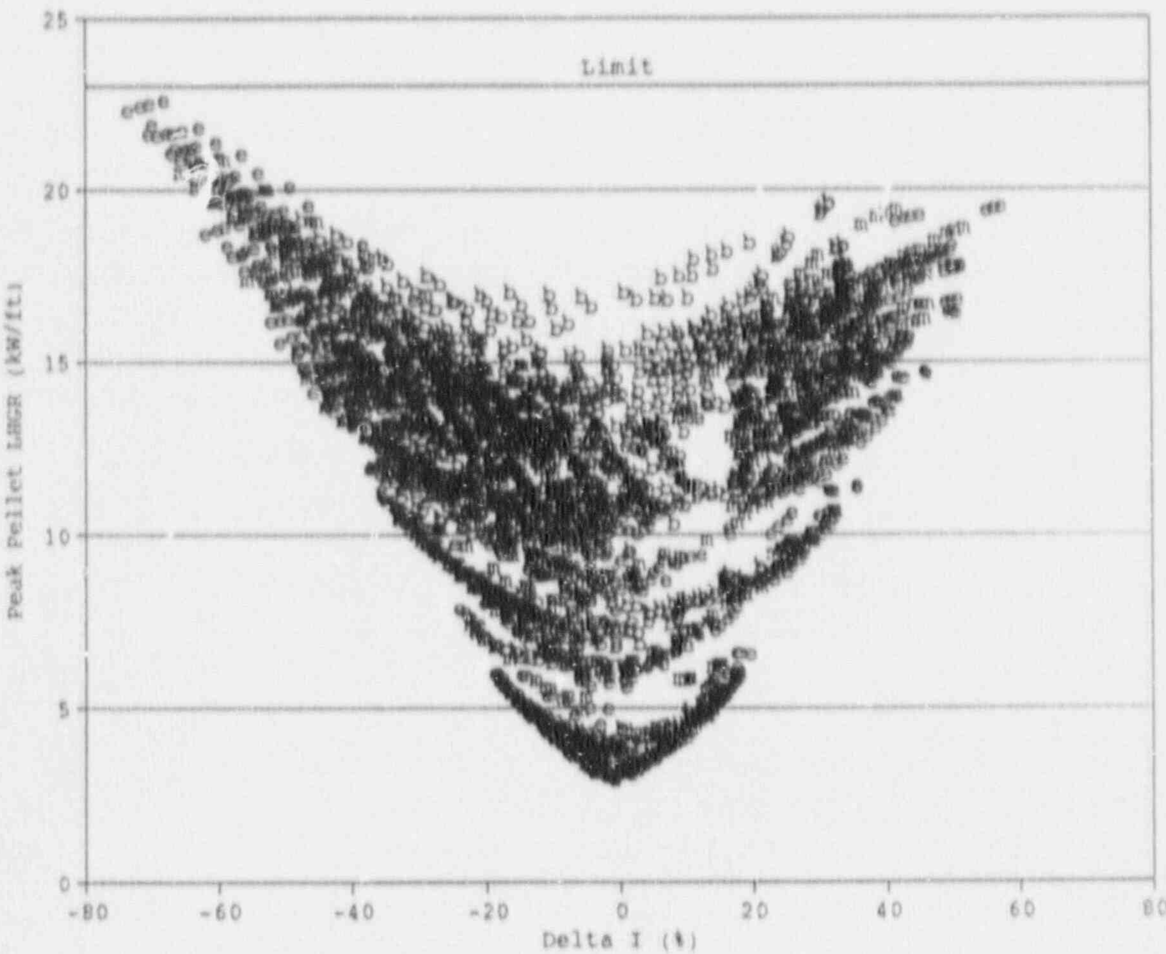


Figure 3.4   Reactor Core Safety Limits (Typical)

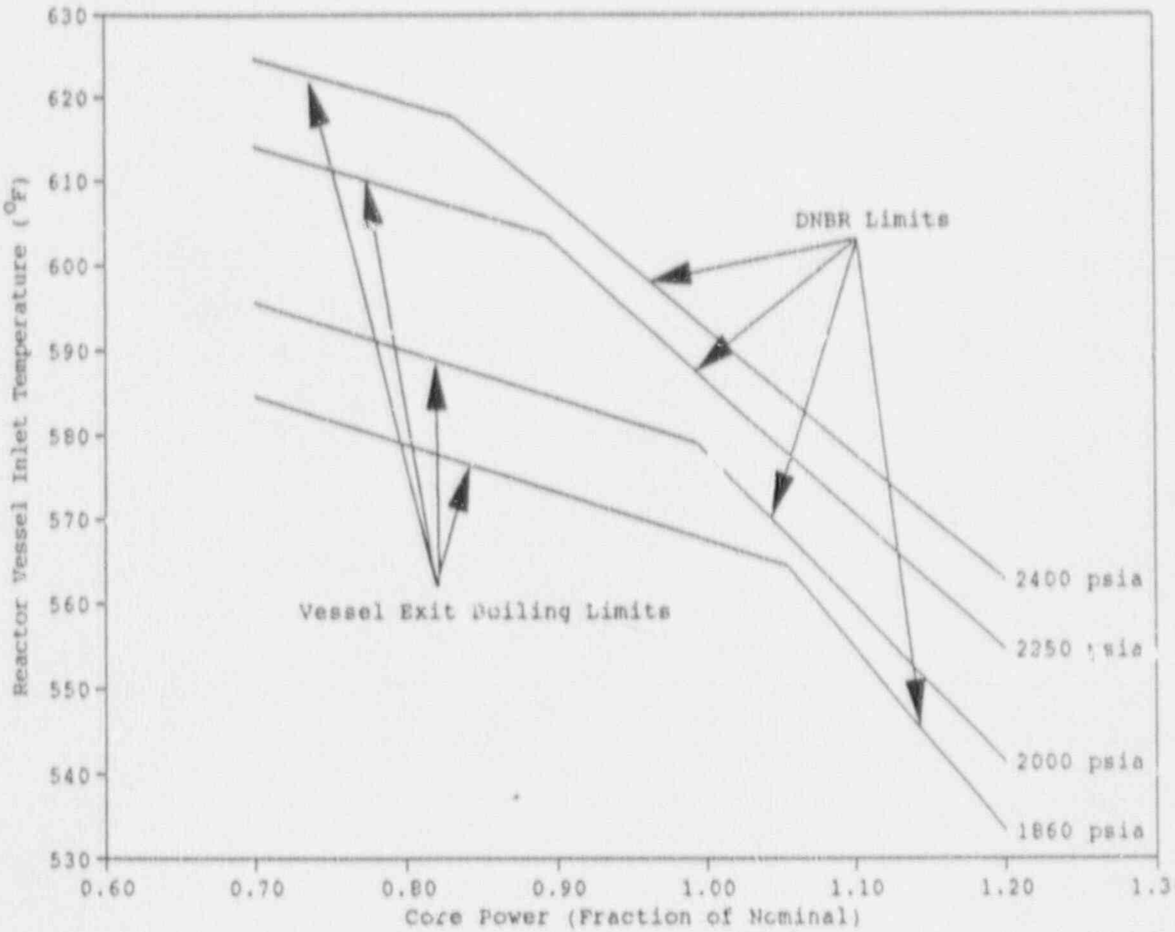


Figure 3.5 Intersection Points Used to Determine OT N-16 Protection

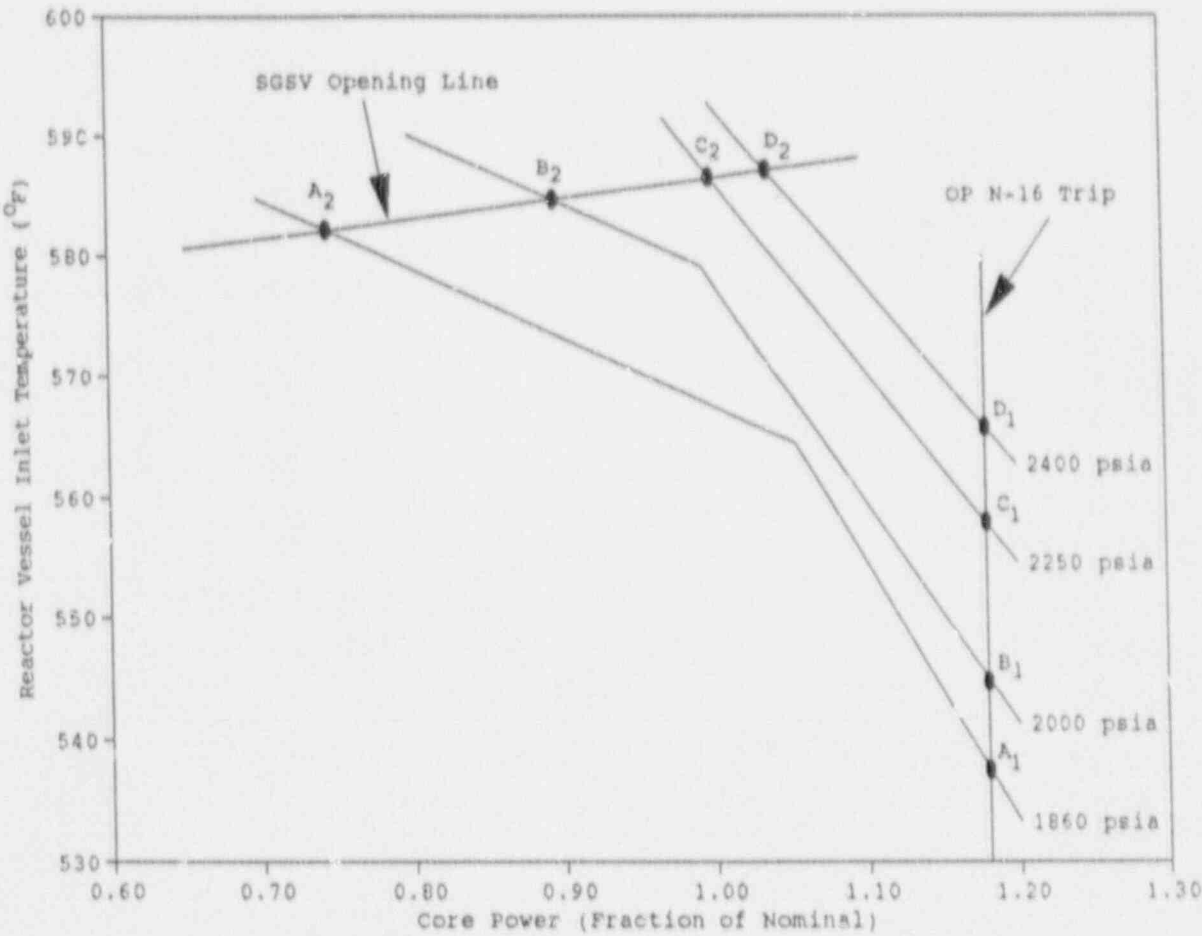


Figure 3.6 Illustration of OT N-16 Protection

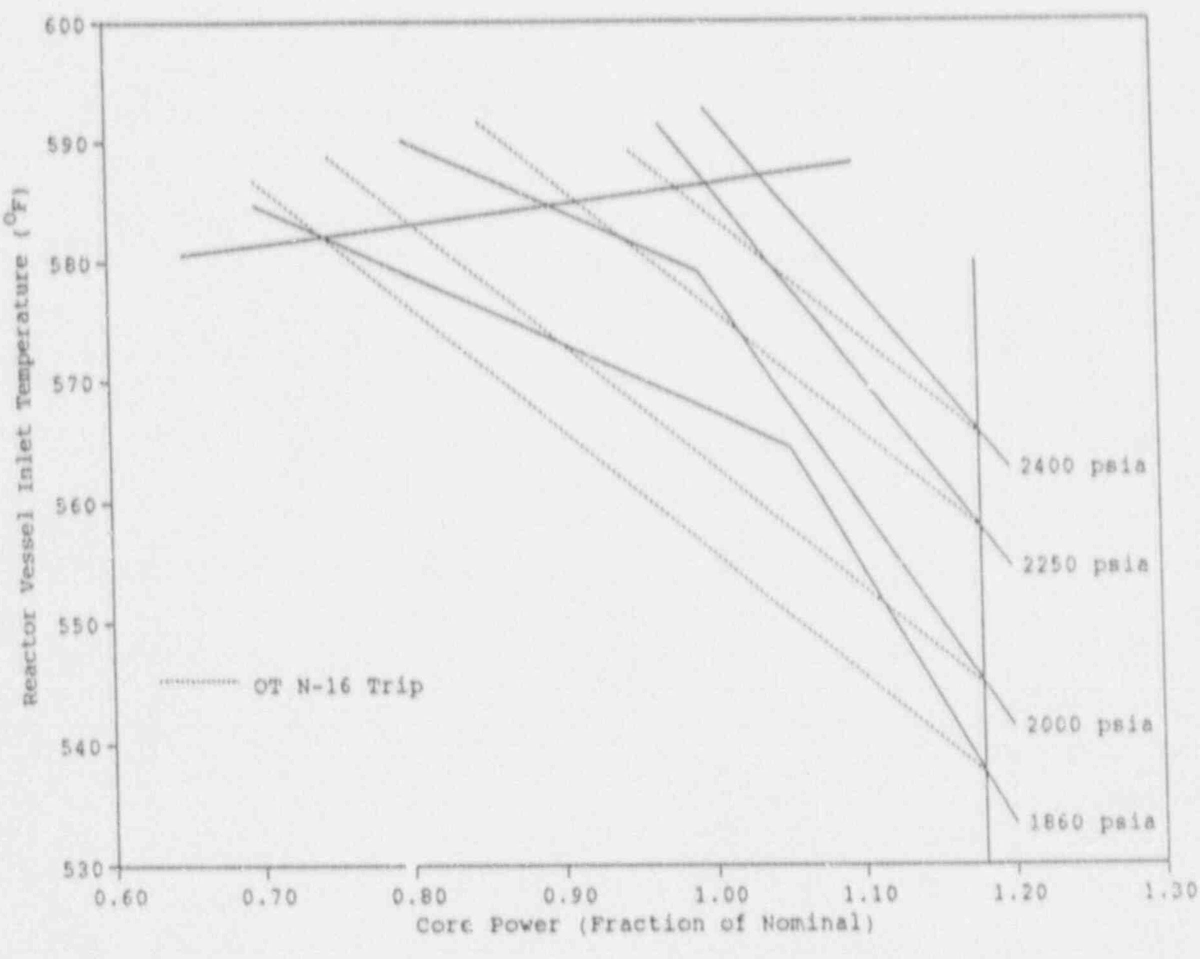


Figure 3.7 Illustration of the Effects of an Adverse Axial Power Distribution

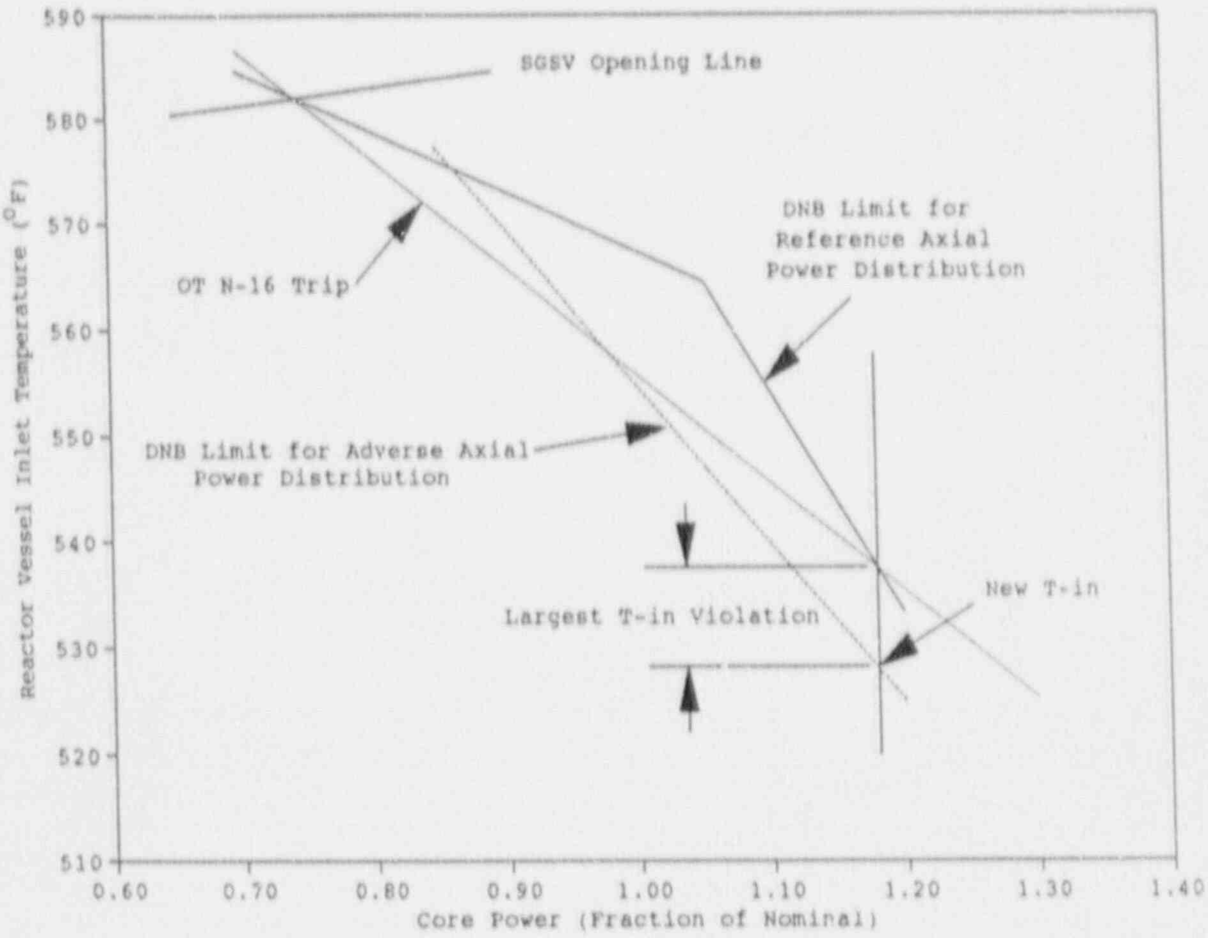


Figure 3.8 Illustration of OT N-16 Setpoint with f(ΔI) Compensation

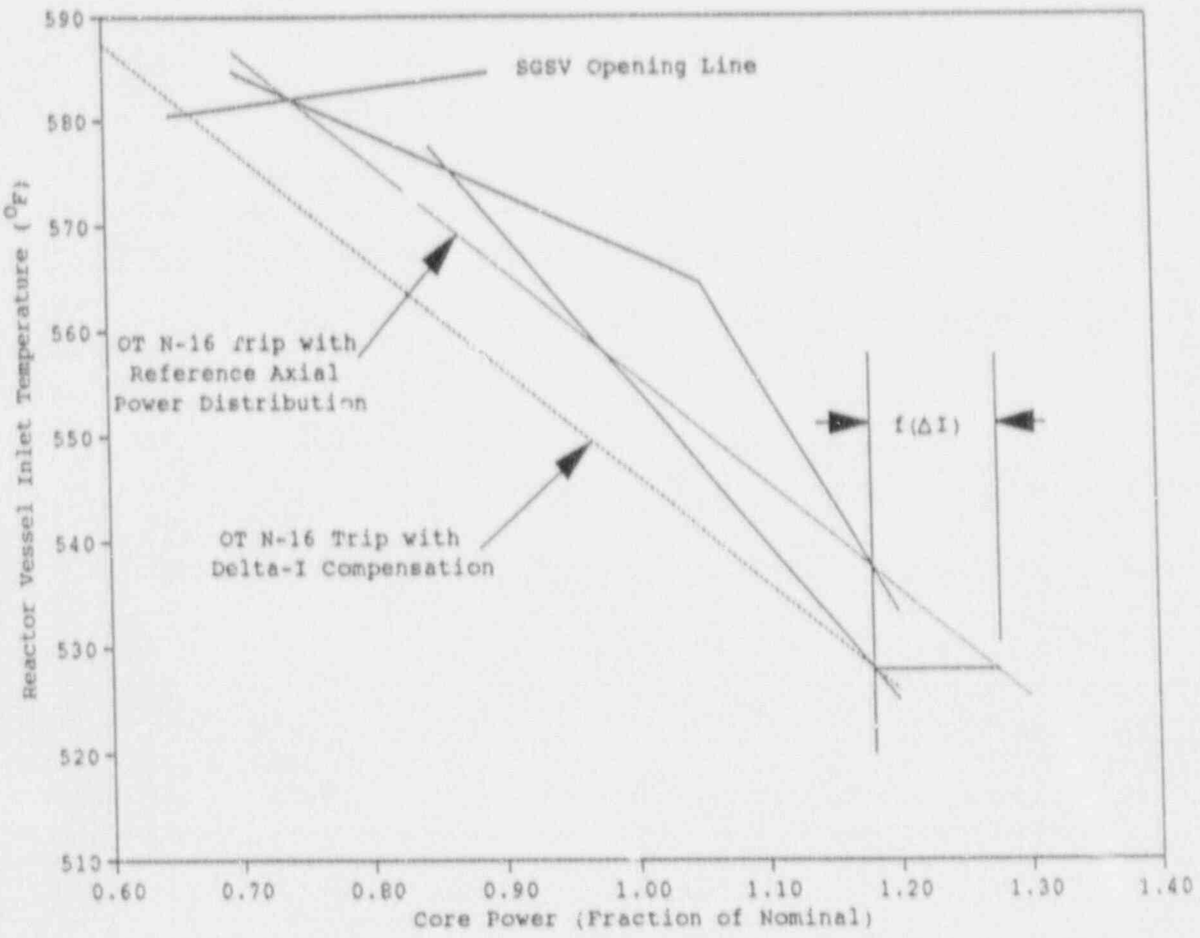


Figure 3.9 Minimum DNBR Results for Various Axial Power Distributions (Typical)

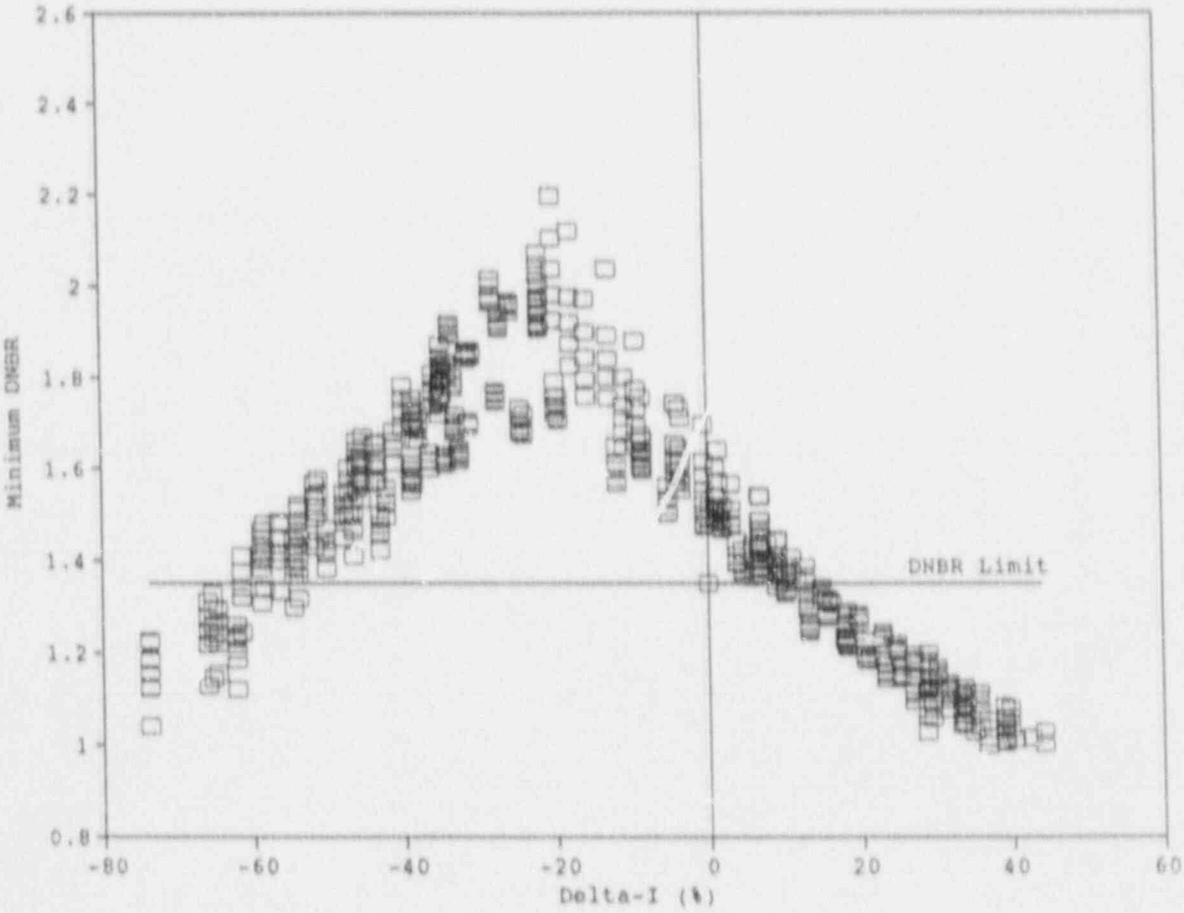


Figure 3.10 Envelope of Required Temperature Reduction Versus Axial Flux Difference (Typical)

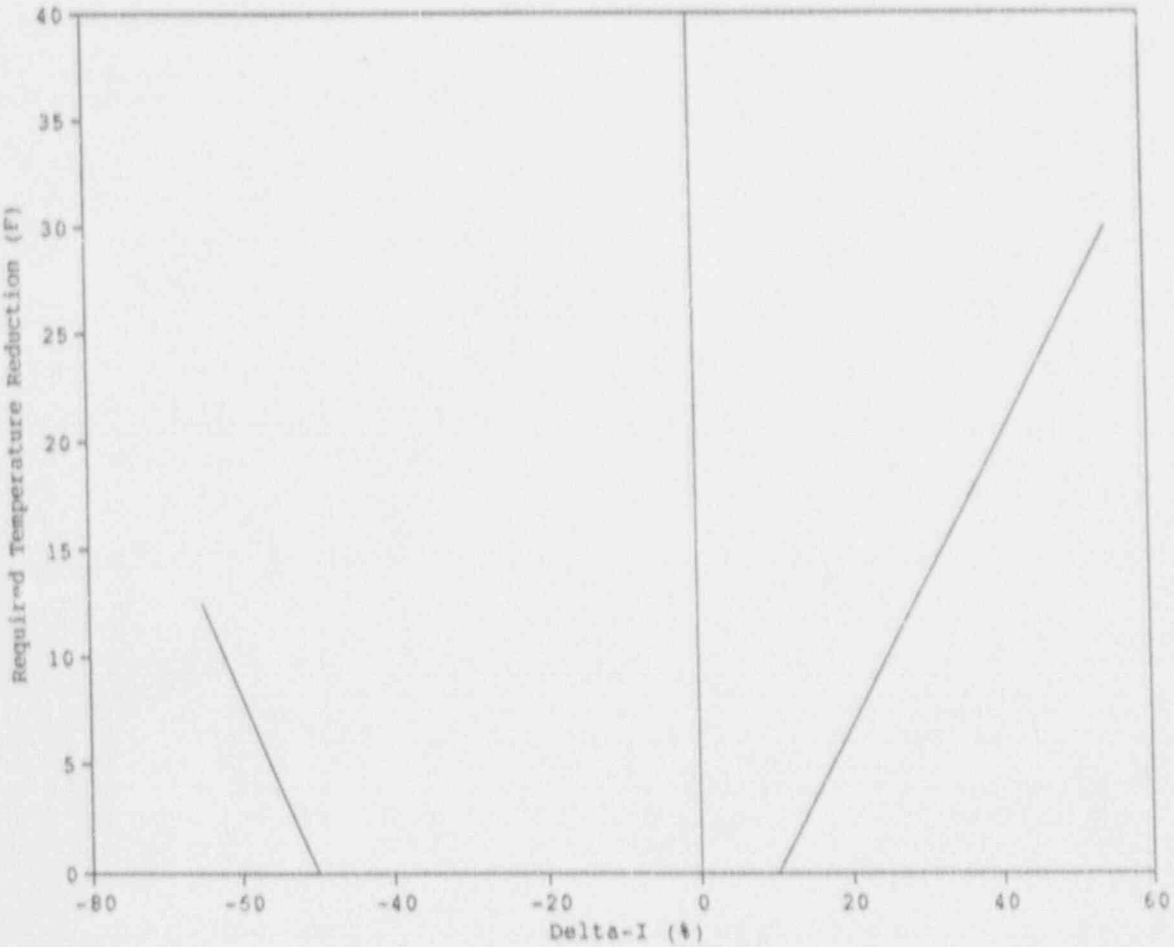


Figure 3.11 f(ΔI) Trip Reset Function (Typical)

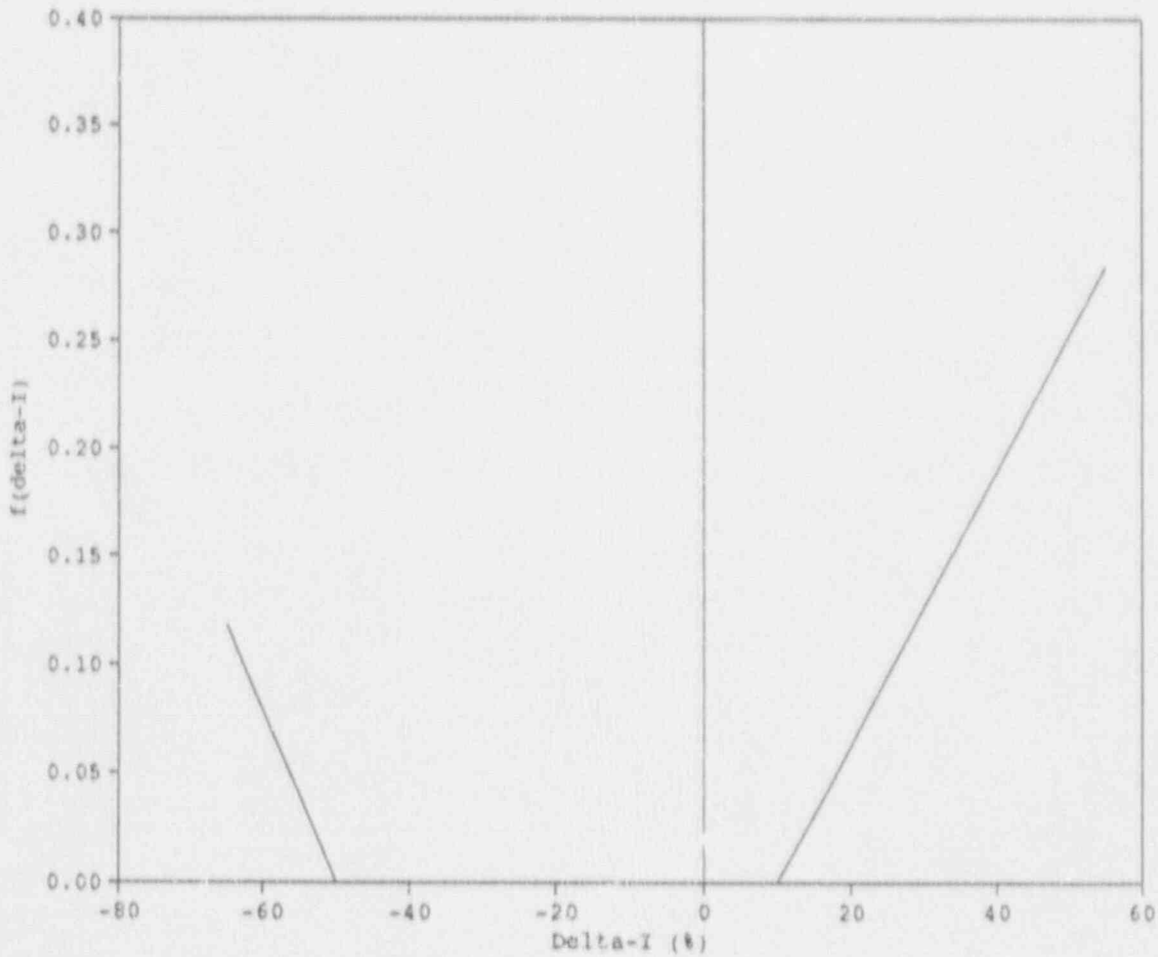


Figure 3.12 Illustration of Axial Offset During Xenon Oscillations Generated at BOL for [ ] Equilibrium Axial Offset Conditions, Comanche Peak Steam Electric Station Unit 1, Cycle 1

[ ]

CHAPTER 4

POWER DISTRIBUTION CONTROL ANALYSIS

4.1 Introduction

The axial power distribution control procedure employed at Comanche Peak Steam Electric Station (CPSES) requires the control of the axial flux difference within a specified target band. This axial flux difference target band is established in order to protect safety margins for: the loss-of-coolant accident (LOCA), departure from nucleate boiling (DNB) during anticipated transients, and local power (kW/ft) limits during normal operation.

Currently, CPSES utilizes the Westinghouse Constant Axial Offset Control (CAOC) (2) method of power distribution control. [

]. TU Electric has adopted the principal attributes of the [ ] and intends to apply them to support future operation of CPSES.

The methodology used by TU Electric to support the constant axial offset power distribution control relies on the determination of the maximum possible variation in the total power peaking factor, $F_Q^T(z)$, at a given height. This variation is, in turn, combined with the measured $F_Q^T(z)$ to project the axially dependent maximum total power peaking factor which could be achieved as a result of varying the axial flux difference within the specified target band.

4.2 Implementation of Power Distribution Control Procedure

The factor T(z) is defined as the maximum anticipated increase in the total peaking factor, $F_Q^T(z)$, from the equilibrium value, when the axial offset is maintained within the target band (non-equilibrium operation). The application of the T(z) factor to verify cc iance with the F_Q^T operating limit is summarized below:

1. Determine the axial offset target band as follows:

$$AO_{lim} = AO_{eq} \pm AO_{band} * P_{RTP} / P$$

where:

AO_{lim}	=	axial offset target band, percent
AO_{eq}	=	equilibrium axial offset, percent
AO_{band}	=	axial offset band width, percent
P_{RTP}	=	Rated Thermal Power, MWth
P	=	reactor operating power, MWth,

NOTE:

Above a fractional power of 0.9, the measured axial offset must be maintained within the target band. Below a fractional power of 0.9 the measured axial offset is allowed to deviate from the target band for one hour out of each twenty-four consecutive hours, provided that the measured axial offset remains within a broader, but specified axial offset band. If this requirement is violated, the core relative power must be reduced below 0.5.

2. Measure the $F_Q^T(z)$ distribution for equilibrium xenon, full power conditions including all uncertainties.
3. Augment the measured $F_Q^T(z)$ with the T(z) factor to determine the maximum anticipated $F_Q^T(z)$ when operating within the axial offset target band.
4. If the product $T(z)*F_Q^T(z)$ exceeds the Technical Specification limit on $F_Q^T(z)$, a power reduction is required.

4.3 Calculation of T(z), the Maximum Variation in $F_Q^T(z)$

The factor T(z) is developed to determine the maximum expected increase in the equilibrium $F_Q^T(z)$ distribution which could occur when the reactor is operated in accordance with the corresponding power distribution control procedures. The

TU Electric T(z) factor is analogous to the [

]. T(z) is determined by performing calculations within the full range of axial offsets permitted by the power distribution control procedures. A series of daily load follow cycle calculations are performed while maintaining the axial offset at the extremes of the axial offset target band. The variations in the $F_Q^T(z)$ distributions resulting from these load follow simulations are bounding with respect to operation within the axial offset target band.

The core average axial offset and flux difference are defined as:

$$AO = (P_T - P_B) / (P_T + P_B) \quad (4\text{-}1)$$

$$\Delta I = (P_T - P_B) \quad (4\text{-}2)$$

or alternatively

$$\Delta I = AO * (P_T + P_B) \quad (4\text{-}3)$$

where:

AO	=	core average axial offset, fraction
ΔI	=	axial flux difference, fraction
P_T	=	power in the top of the core, fraction of RTP
P_B	=	power in the bottom of the core, fraction of RTP

The values of ΔI and AO are typically multiplied by 100 to permit discussion in terms of percent.

The $F_Q^T(z)$ distribution is determined by a [

] and is discussed in section 5.3.

The T(z) factor is calculated as the ratio of the maximum $F_Q^T(z)$ calculated for each of the daily load follow cycles to the equilibrium base depletion $F_Q^T(z)$ distribution de{ `rmined from operating at the equilibrium axial offset. The operating limit on $F_Q^T(z)$ increases as reactor power is reduced; therefore, this effect is included in the T(z) calculation by multiplying $F_Q^T(z)$ by the fractional power level before calculating T(z):

$$T(z) = \frac{F_Q^T(z,j) * P_j}{F_Q^T(z,equil)} \tag{4-4}$$

where:

$F_Q^T(z,j)$ = Total peaking factor for daily load follow cycle case j

$F_Q^T(z,equil)$ = Total peaking factor for equilibrium base depletion

P_j = Power level for daily load follow cycle case j, fraction of RTP

A bounding approach is used to develop the T(z) factor such that the core average axial offset, a measurable parameter, can be used to verify the acceptability of reactor operation at non-equilibrium conditions. [

].

4.4 <u>Reactor Load Follow Simulations</u>

The load follow simulations analyzed in the calculation of the T(z) factor were developed to envelop all axial power distributions that are anticipated during normal operating modes (3,7,8). The load follow simulations consist of a regular pattern of power reductions and increases. The reactor power is reduced from 100% to 50% power over a three hour period, and maintained at 50% for six hours. The power is then increased to 100% over a three hour period and maintained at 100% for 12 hours, which concludes the daily load cycle. This daily load cycle is continued for five twenty-four hour periods as depicted in Figure 4.1.

[

].

[

]

[

].

The T(z) factor is calculated as a function of exposure by utilizing the results from these six daily load follow cycles. Typically the calculations are performed at BOL, MOL, and EOL. For BOL, a cycle exposure of 500 MWD/MTU is chosen to include the effects of equilibrium xenon and samarium. The MOL case is selected near the middle of cycle, while EOL is selected at approximately 80% of the total cycle exposure such that sufficient soluble boron remains to permit load follow.

[

]. A limiting T(z) is then calculated by the addition of an arbitrary 1% conservatism to the maximum composite line. The resultant curves are shown in Figure 4.3.

4.5 <u>Operation Outside the Target Axial Offset Limits</u>

The power distribution control procedures are designed to limit the variation in the power distribution during reactor operation by limiting the variation in the core power axial offset. However, reactor operation outside the axial offset target band for periods of short duration is permitted while continuing to maintain control of the variation in the power distribution.

In the determination of the maximum composite T(z) distribution, it was observed that limiting values for T(z) were always defined [

]. It is therefore concluded that the increase in the operating limit on $F_Q^T(z)$ as the power level decreases is always greater than the actual increase in $F_Q^T(z)$ as the power level decreases and the axial offset is maintained within the target band. As a result, reactor operation with the axial offset outside the target band is permitted provided the relative power level is below 90 percent. The time outside the target band is limited to one hour in any consecutive twenty-four hour period to preserve the xenon distribution. Preserving the xenon distribution will prevent xenon oscillations and subsequent power distribution oscillations with axial offsets outside the analyzed target band.

[

]

[

]:

[ ]

Calculations to determine the T-factor are typically performed at 50%, 70% and 90% power at the cycle exposures used to determine T(z). At each of these points, the T-factor and the ΔAO are calculated. An example is shown in Figure 4.4. The positive and negative ΔAO values for the one-hour limit are determined by locating the ΔAO where the expected variation will equal the maximum variation (T-factor equals 1.0). The axial offset limits at each power level are converted to ΔI to define the one-hour limits. Figure 4.5 illustrates a typical allowable deviation from the target axial flux difference.

4.6 Methodology Applied to CPSES Unit 1 Cycle 1

4.6.1 T(z) Distribution Calculation

CPSES Unit 1, Cycle 1, has been selected to illustrate application of the TU Electric power distribution control methodology. The exposure dependent T(z) distribution was calculated [

]. In these examples, two axial offset target band widths have been selected. For the BOL case, an asymmetric target band width of +3% and -5% has been chosen. Such a target band width is representative of the operating constraints which would be utilized should the cycle design result in very little margin to the F_Q^T limit. At EOL an asymmetric axial offset band width of +3% and -12% has been selected. This band width is typical of those utilized for cores designed for improved load follow capability.

[

].

[

].

In Reference 7, [

]. Although the [ ] and the T(z) distribution depend upon the physical characteristics of the reactor, it is instructive to make a [ ] the [ ] and the TU Electric calculation for CPSES. The variation in the $F_Q^T(z)$ is principally determined by the width of the axial offset target bands; therefore, the [ ] and the T(z) calculated for CPSES for similar axial offset target band widths should be similar in magnitude.

[ ]. This compares to the maximum T(z) for a +3% band width in the top of the core of 1.083 at BOL and 1.089 at EOL. [ ]

[ ]. This compares to a T(z) calculation for a -5% band width in the bottom of the core of 1.116 at BOL.

4.6.2 Operation Outside the Target Band

Reactor operation outside the axial offset target band can be permitted for periods of short duration while maintaining control of the variation in the power distribution. To illustrate the methodology used to determine the expanded axial offset band, a calculation has been performed utilizing CPSES Unit 1, Cycle 1.

The demonstration calculation was performed at BOL conditions for reactor operation at 90% power. Restart calculations were initiated from the positive and negative axial offset limits, and the axial offset was forced outside the target bands for up to one hour. The T-factor was determined, and is displayed in Figure 4.10.

The one-hour limits are determined by locating the Δ. ' values at the positive and negative ΔAO limits where the expected variation, T(z), equals the maximum variation. From Figure 4.10, the negative limit is determined to be -9%ΔAO, and the positive limit is determined to be +5%ΔAO. These axial offset limits are converted from units of ΔAO to ΔI at 90% power, and become -8%ΔI and +4%ΔI for the negative and positive one-hour target band widths, respectively.

Figure 4.1 Daily Load Follow Cycle Used in T(z) Analysis

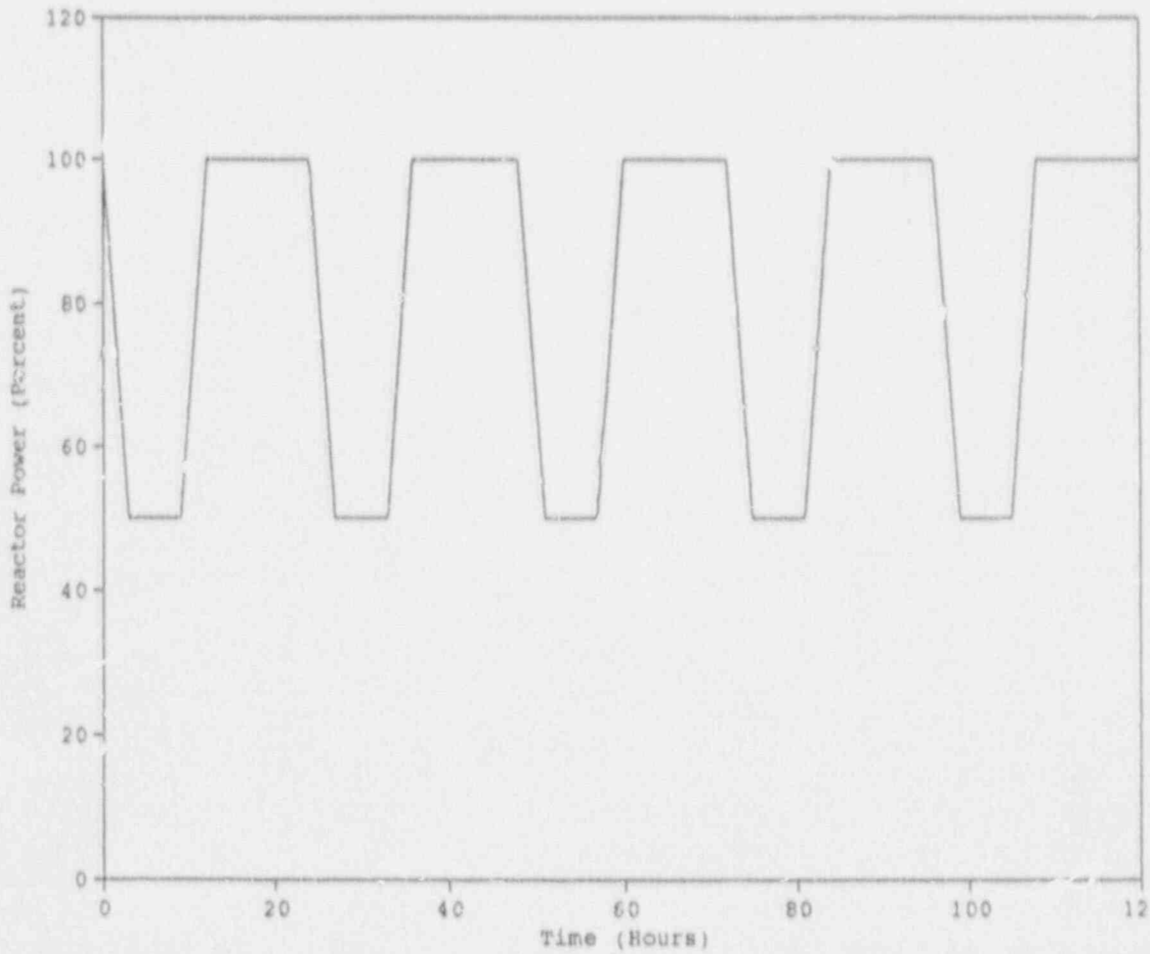


Figure 4.2 T(z) Distribution for Daily Load Follow Cases (Typical)

Figure 4.3 Maximum Composite and Limiting T(z) Distributions (Typical)

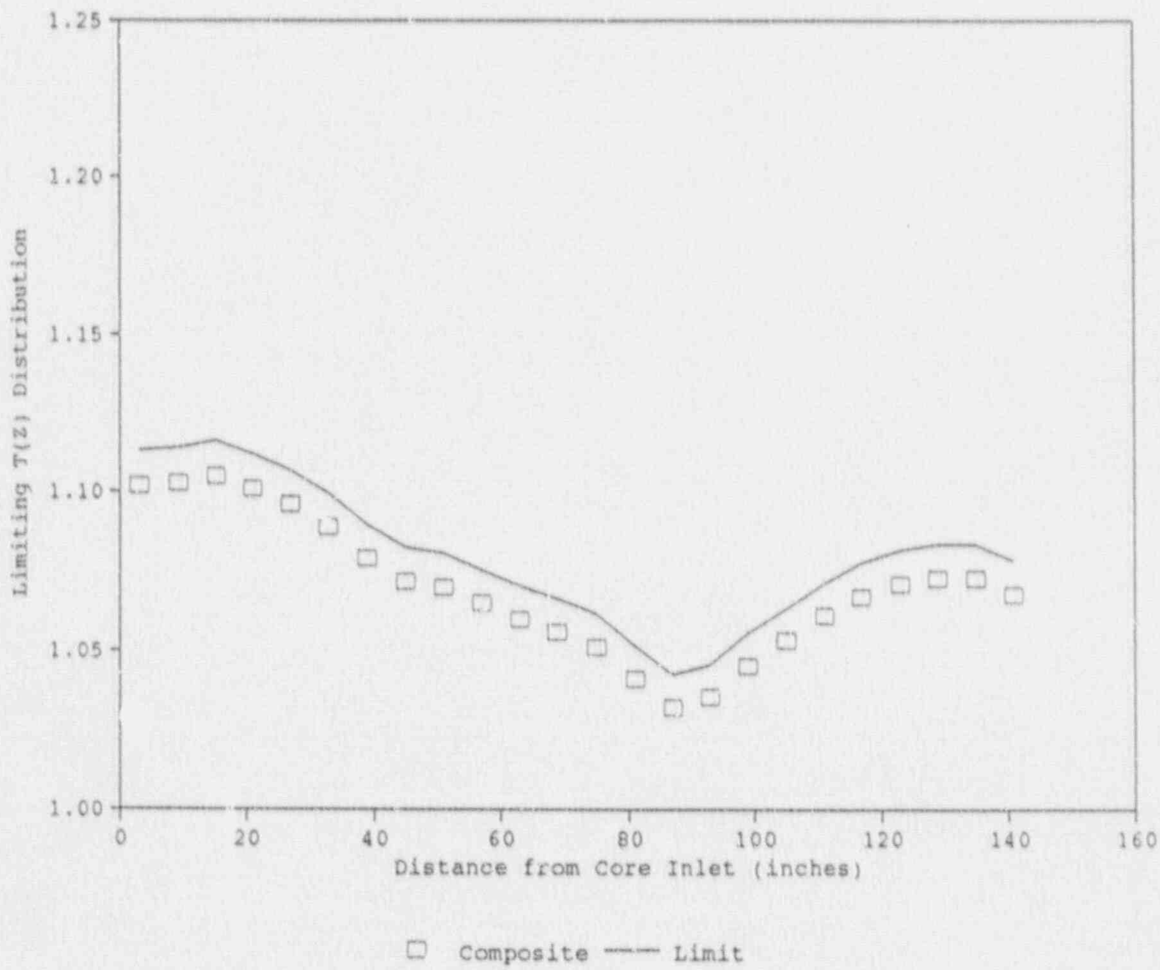


Figure 4.4 T-factor as a Function of Delta Axial Offset (Typical)

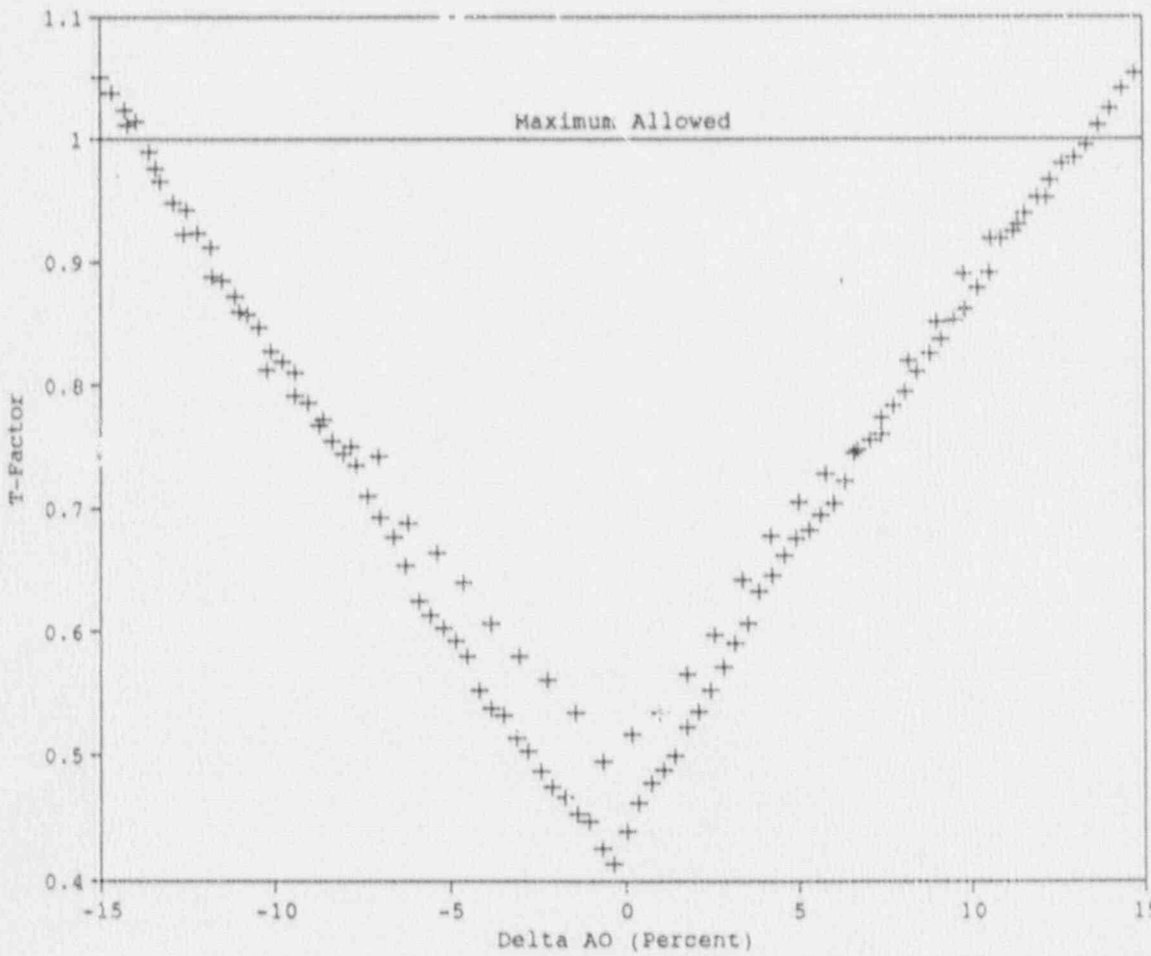


Figure 4.5 Allowable Deviation from Target Flux Difference for Operation Outside of Target Flux Bands (Typical)

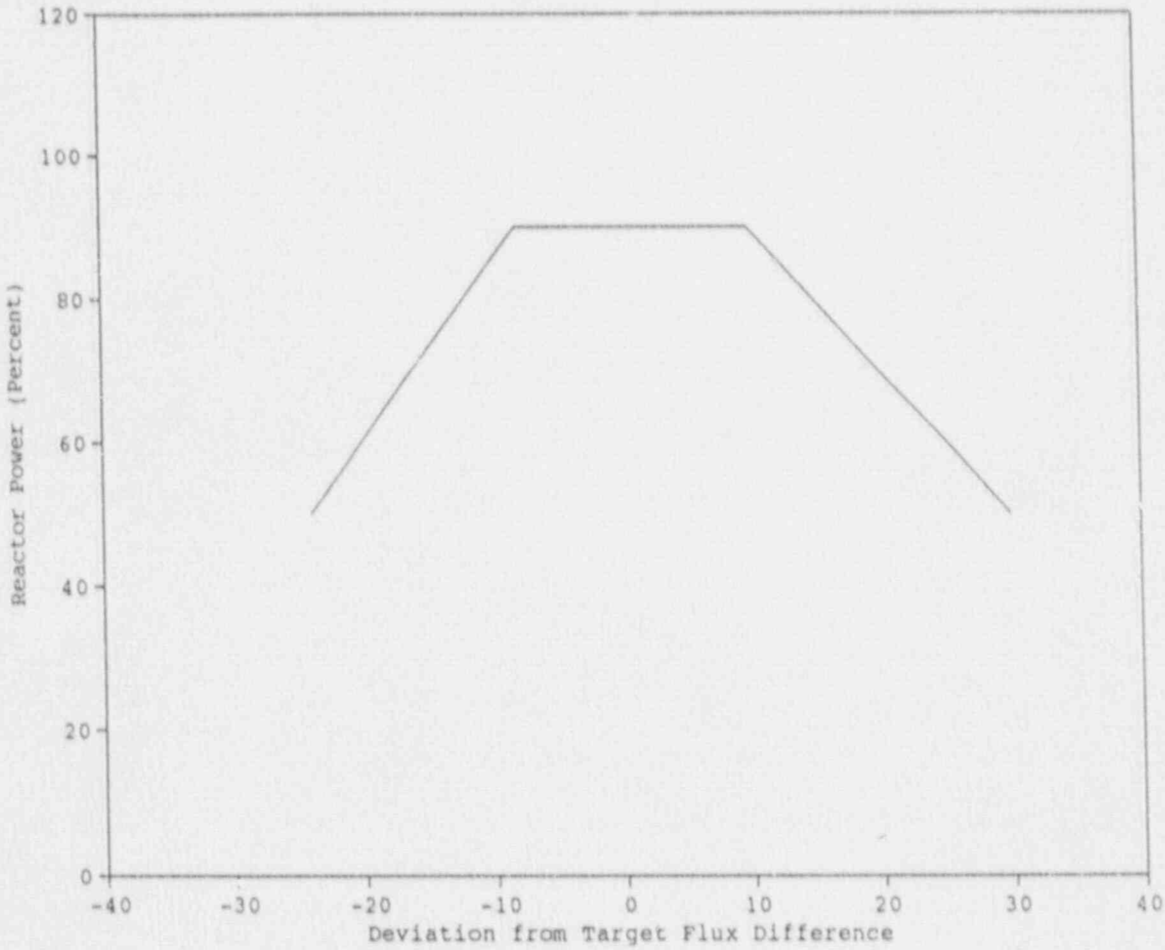


Figure 4.6 T(z) Distribution, +3%, -5% Target Axial Offset Band, BOL, [ ], Comanche Peak Steam Electric Station Unit 1, Cycle 1, Demonstration

Figure 4.7 Maximum Composite and Limiting T(z) Distributions, +3%, -5% Target Axial Offset Band, BOL, Comanche Peak Steam Electric Station Unit 1, Cycle 1, Demonstration

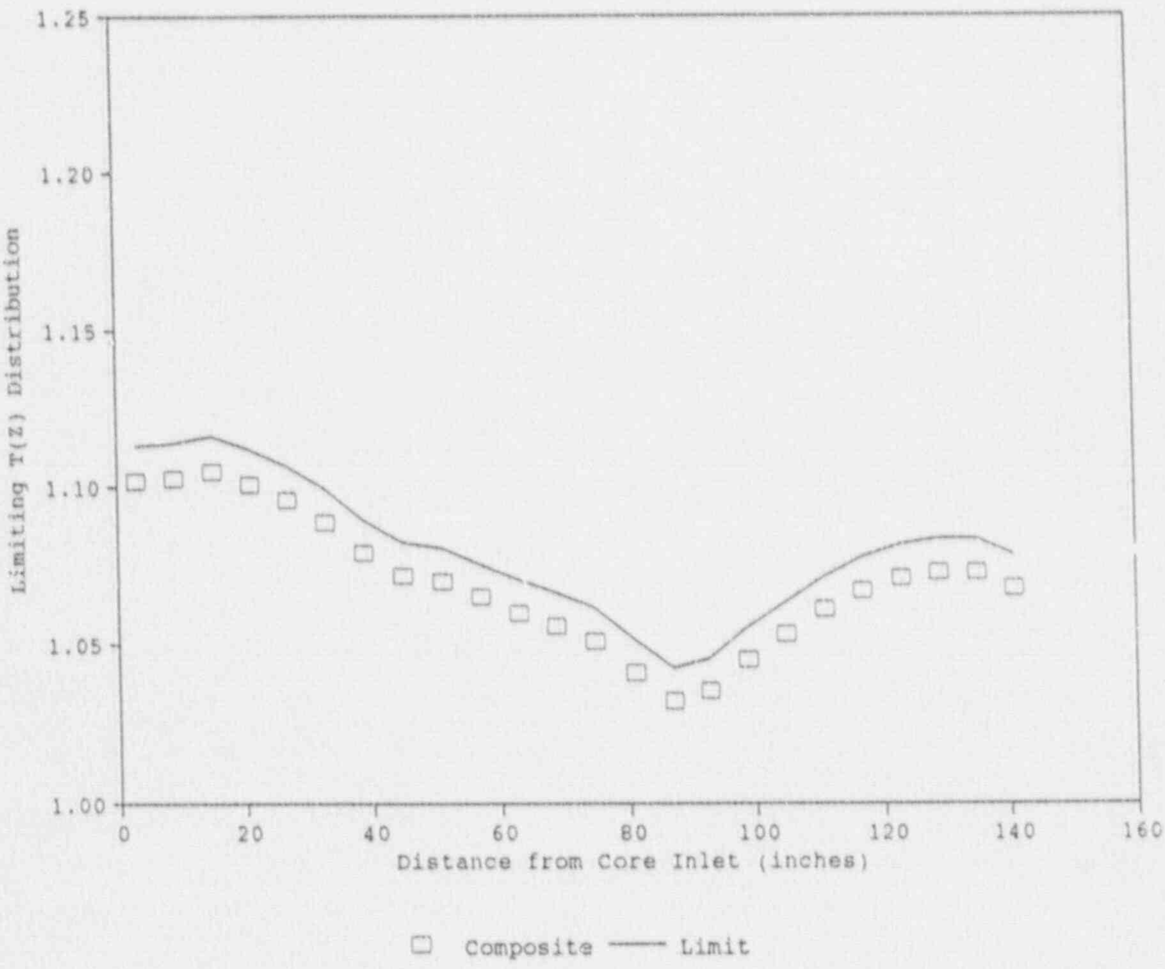


Figure 4.8 T(z) Distribution for Daily Load Follow, +3%, -12% Target Axial Offset Band, EOL, [ ], Comanche Peak Steam Electric Station Unit 1, Cycle 1, Demonstration

Figure 4.9 Maximum Composite and Limiting T(z) Distributions, +3%, -12% Target Axial Offset Band, EOL, Comanche Peak Steam Electric Station Unit 1, Cycle 1, Demonstration

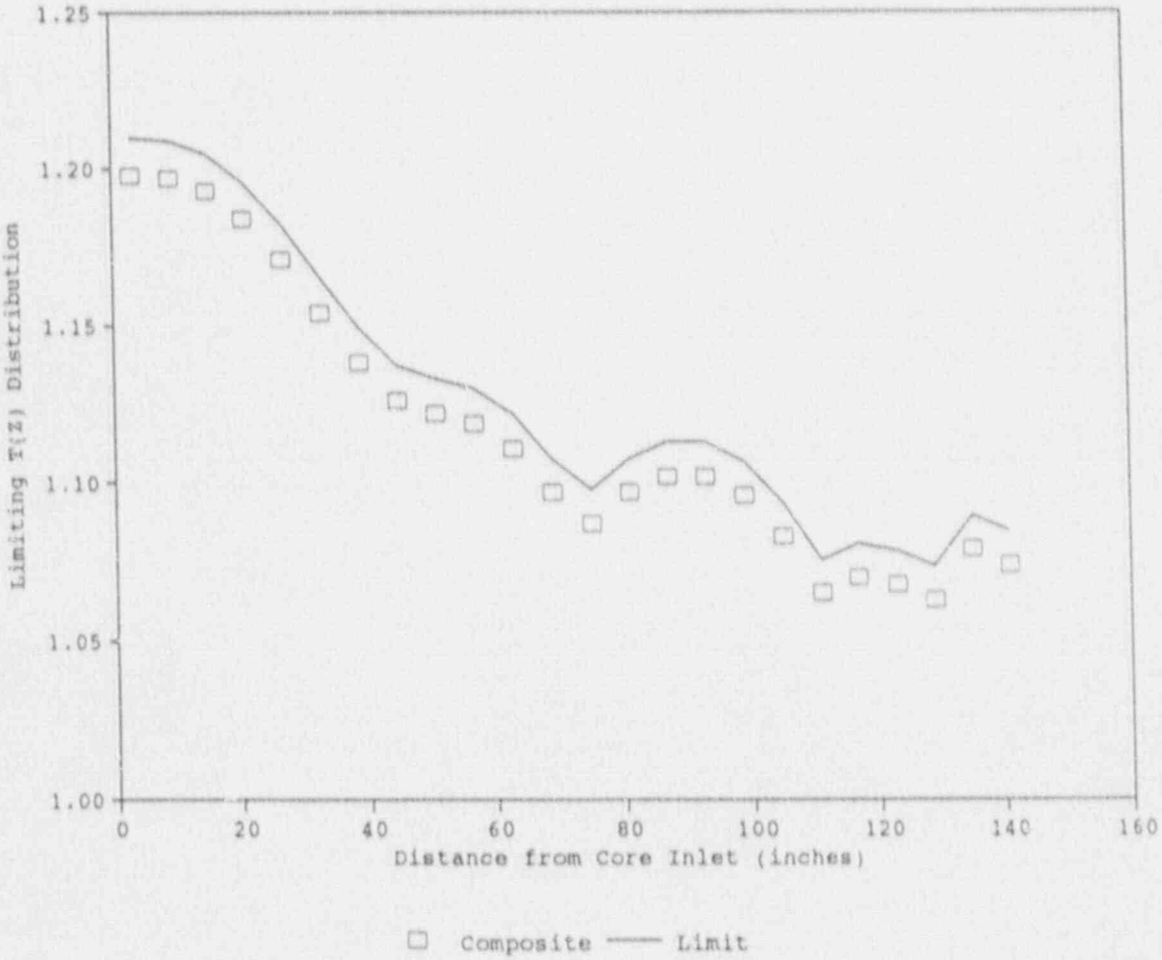
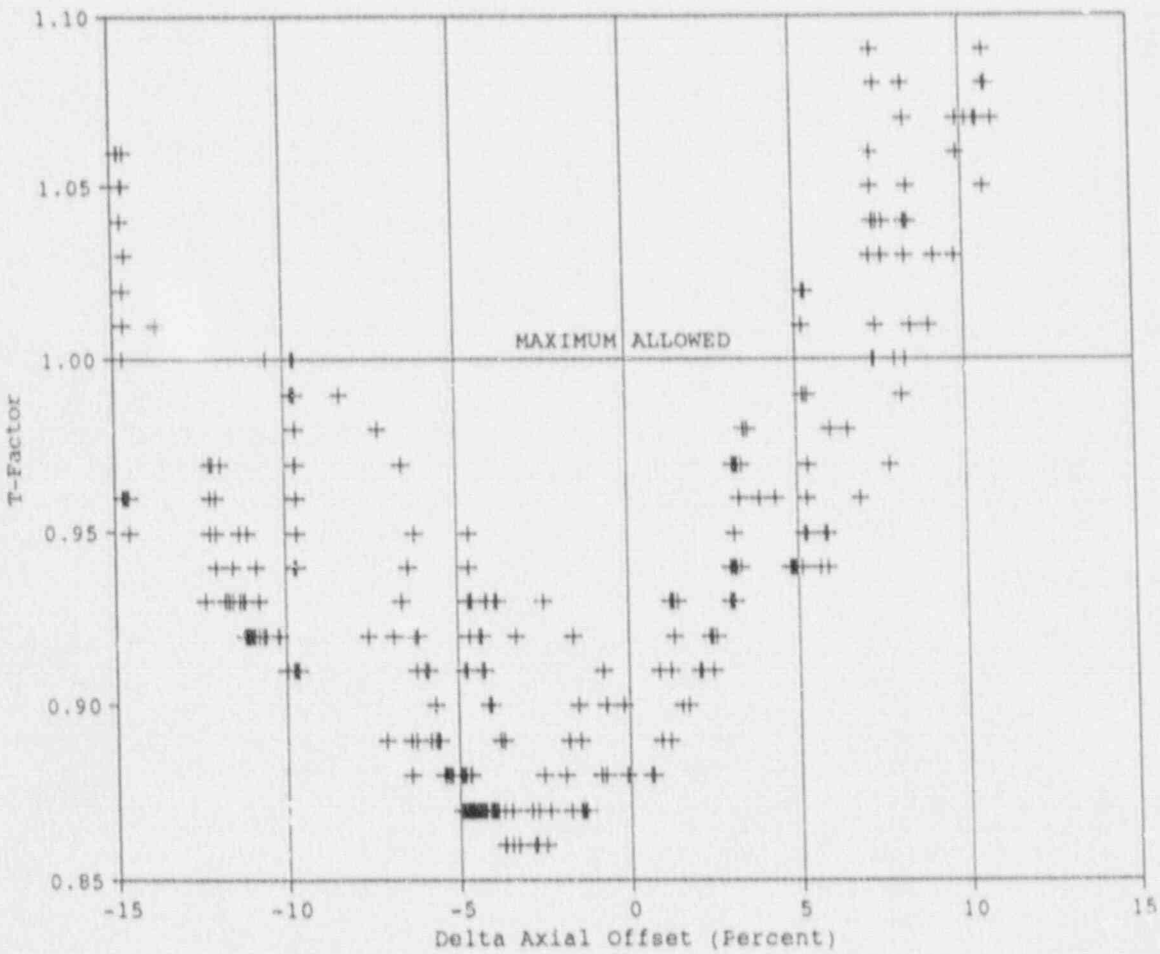


Figure 4.10 T-factor as a Function of Delta Axial Offset, BOL, 90% power, +3%, -5% Target Band, Comanche Peak Steam Electric Station Unit 1, Cycle 1, Demonstration



CHAPTER 5

NEUTRONIC MODELS AND METHODS

5.1 Calculational Approach

The neutronics analysis supporting the setpoint methodology and power distribution control analysis is computationally intensive; therefore, the analysis is performed in

[

].

The advanced two-group nodal code, SIMULATE-3, is used to

[

].

The total peaking factor, F_Q^T, is derived by a synthesis of

[

].

5.1.1 Computer Code Descriptions

5.1.1.1 The [ ] Computer Code

[

].

[

].

5.1.1.2 The CASMO-3 Computer Code

The fuel assembly cross sections used in the [ ] and SIMULATE-3 models are generated by the CASMO-3 (11) computer code. CASMO-3 is a multi-group, two-dimensional transport theory code for burnup calculations on BWR and PWR assemblies or simple pin cells. The code handles a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array with allowance for fuel rods loaded with gadolinium, burnable absorber rods, cluster control rods and incore instrument channels.

A detailed discussion on the use of CASMO-3 by TU Electric in support of steady state reactor physics analyses required for design, licensing, start-up and operation of CPSES is included in Reference 12.

5.1.1.3 The SIMULATE-3 Computer Code

The SIMULATE-3 (13) computer code is an advanced two-group nodal code used to perform steady state core physics calculations. The neutronics model in SIMULATE-3 solves a two energy group neutron diffusion equation using second-order polynomials to represent the transverse leakage. These polynomials represent the intra-nodal flux distribution for both fast and thermal energy groups. The solution can be obtained in one, two or three dimensions. SIMULATE-3 employs discontinuity factors to permit discontinuous flux on the node boundary. The discontinuity factors are obtained from the same CASMO-3 cases which generate two-group assembly average cross sections.

A detailed discussion on the use of SIMULATE-3 by TU Electric to perform steady state reactor physics analyses is included in Reference 12.

5.1.2 <u>Fuel Assembly Model (Cross Section Generation)</u>

The fuel assembly cross sections were generated using the CASMO-3 fuel assembly model detailed in Reference 12. Attributes of the cross sections used in the [ ] model can be summarized:

1. Conventional macroscopic cross sections extracted from the PDQ block of CASMO-3;
2. Thermal group cutoff at 0.625 eV;
3. Xenon built-in through I/Xe chain;
4. Average temperature and inlet temperature depletions;
5. Control rod branch cases; and
6. Moderator temperature branch cases.

CASMO-3 average temperature depletions, inlet temperature depletions, and moderator temperature branch cases are used to correct the macroscopic cross sections for the spectral effects of variation in moderator density.

Control rod branch cases are used to determine the effect of control rod insertion on the macroscopic cross sections. The control rod cross sections are applied as "delta" cross sections to the fueled regions of the core. These "delta" cross sections are determined explicitly from the CASMO-3 base depletion and CASMO-3 control rod branch cases. A linear exposure dependence of the delta cross sections is assumed in the [ ] models.

5.1.3 [ ] Core Model

[

];

[

].

5.1.4 [ ] Core Model

[

];

[

].

[

].

5.2 [ ]

[

].

5.2.1 Comanche Peak Unit 1 Cycle 1 Core Description

CPSES Unit 1 is a Westinghouse four-loop 3411 MWth, pressurized water reactor. The core consists of 193 fuel assemblies with a 17x17 fuel pin array in each assembly. Cycle 1 of this unit utilizes Westinghouse fuel of standard design and includes 944 burnable absorber rods containing borosilicate glass. The fuel enrichments for Cycle 1 are 1.6 w/o U-235, 2.4 w/o U-235, and 3.1 w/o U-235. At the time this analysis was begun, hafnium control rods were scheduled to be used in this unit. Subsequently, silver-indium-cadmium control rods were actually utilized. Hafnium control rods were modeled in all calculations and comparisons presented herein.

Figure 5.1 shows the core loading plan for Cycle 1, including control rod locations. A list of the fuel types for this cycle and the number of assemblies of each type is given in Table 5.1.

5.2.2 Boron Letdown [ ]

[

].

5.2.3 Differential and Integral Rod Worth [ ]

[

]

[

].

5.2.4 Radial Power Distribution [ ]

[

].

5.2.5 Axial Power and Exposure Distribution [ ]

[

].

5.2.6 Xenon Worth [ ]

[

]

[

].

5.3 Calculation of Total Peaking Factor, F_Q^T

The total peaking factor, F_Q^T, defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, is determined from [

]. The method for determining the total peaking factor is summarized in the following sections.

5.3.1 Determination of F_Q^T

The total peaking factor can be expressed as a function of axial position, z:

[ ] (5-1)

[

]

[ ]:

$$\left[\qquad \right] \tag{5-2}$$

Given the hot rod axial power distribution, and the following definition for the nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$:

$$F^N_{\Delta H} = F_{\Delta H}^{RTP} (1 + PF_{\Delta H} * (1 - P)) \tag{5-3}$$

where:

$F_{\Delta H}^{RTP}$ = the $F_{\Delta H}^N$ design limit at power $\geq$ RTP (typically 1.55)

$PF_{\Delta H}$ = the part power multiplier (typically .2 or .3)

P = the fraction of RTP

the total peaking factor F_Q^T can be determined.

$$F_Q^T \quad = \quad \max \{ F^N_{\Delta H} * F_h(z) \} * F_E * F_A \tag{5-4}$$

5.3.2 Determination of Radial Peaking Factor and Power Spike Factor

The radial peaking factor used in the [ ]

[

].

The calculations are performed at nominal rod worth as well as with conservative increases in rod worth (typically 10% and 30%). Additional conservatism is included by preventing the $R(z,z_o)$ factor from decreasing as control rods are inserted more deeply into the core. An example of the $R(z,z_o)$ factor, a composite of the calculations performed for CPSES Unit 1, Cycle 1, is shown in Figure 5.15. The $R(z,z_o)$ factor for node $z=z'$ increases as the control bank D enters the axial node $z_0=z'$, then remains relatively constant as the control bank is inserted deeper. When control bank C enters the axial node $z=z'$, the $R(z,z_o)$ factor begins to increase again. The $R(z,z_o)$ factor peaks at a maximum value of 1.21.

The densification power spike factor, $S(z)$ is dependent upon the fuel type (vendor dependent), and may vary as a function of axial position. An example of the $S(z)$ factor is shown in Figure 5.16.

Table 5.1

Core Loading Description, Comanche Peak Steam Electric Station Unit 1, Cycle 1

Enrichment U-235	Number of Burnable Absorbers	Number of Assemblies
1.6	0	65
2.4	0	4
2.4	4	8
2.4	8	32
2.4	12	12
2.4	24	8
3.1	0	32
3.1	3	4
3.1	4	12
3.1	9	8
3.1	24	8

Table 5.2

[ ] in Assembly Average Relative Powers, SIMULATE-3 [ ], Comanche Peak Steam Electric Station Unit 1, Cycle 1

[ ]

Table 5.3

Axial Offset [ ], SIMULATE-3 [ ],
Comanche Peak Steam Electric Station Unit 1, Cycle 1

Exposure MWD/MTU	Percent Axial Offset		
	SIMULATE-3	[ ]	[ ]
0	-6.5		
150	-7.9		
500	-7.5		
1000	-6.9		
2000	-5.6		
4000	-3.7		
6000	-3.[illegible]		
8000	-3.2		
10000	-2.8		
12000	-2.7		
13000	-1.5		

Figure 5.1 Quarter-Core Loading Arrangement, Comanche Peak Steam Electric Station Unit 1, Cycle 1

	H	G	F	E	D	C	B	A
8	1.6 CB D	2.4 12	1.6 CB A	2.4 8	1.6 SDB E	2.4 8	1.6 CB C	3.1 4
9	2.4 12	1.6	2.4 8	1.6	2.4 8	1.6 SBD B	3.1 24	3.1
10	1.6 CB A	2.4 8	1.6 CB C	2.4 8	1.6	2.4 4	1.6 CB B	3.1 4
11	2.4 8	1.6	2.4 8	1.6	2.4 12	1.6 SDB C	3.1 9	3.1
12	1.6 SDB E	2.4 8	1.6	2.4 12	2.4 CB D	2.4 24	3.1 SDB A	
13	2.4 8	1.6 SDB B	2.4 4	1.6 SDB D	2.4 24	3.1 3	3.1	
14	1.6 CB C	3.1 24	1.6 CB B	3.1 9	3.1 SDB A	3.1		
15	3.1 4	3.1	3.1 4	3.1				

enrichment (w/o U-235)
number of BAs
control bank or shutdown bank

Figure 5.2 SIMULATE-3 [ ] Boron Letdown [ ], ARO, HFP, Comanche Peak Steam Electric Station Unit 1, Cycle 1

[ ]

Figure 5.3 SIMULATE-3 [ ] Differential Control Rod Worth of Control Banks D and C with 113 Steps Overlap, BOL, HFP, Comanche Peak Steam Electric Station Unit 1, Cycle 1

Figure 5.4 SIMULATE-3 [ ] Integral Control Rod Worth of Control Banks D and C with 113 Step Overlap, BOL, HFP, Comanche Peak Steam Electric Station Unit 1, Cycle 1

Figure 5.5 SIMULATE-3 [ ] Differential Control Rod Worth of Control Banks D and C with 113 Step Overlap, EOL, HFP, Comanche Peak Steam Electric Station Unit 1, Cycle 1

Figure 5.6 SIMULATE-3 [ ] Integral Control Rod Worth of Control Banks D and C with 113 Step Overlap, EOL, HFP, Comanche Peak Steam Electric Station Unit 1, Cycle 1

Figure 5.7 SIMULATE-3 [ ], HFP, 150 MWD/MTU, ARO, Relative Power Distribution [ ], Comanche Peak Steam Electric Station Unit 1, Cycle 1

Figure 5.8 SIMULATE-3 [ ], HFP, 6000 MWD/MTU, ARO, Relative Power Distribution [ ], Comanche Peak Steam Electric Station Unit 1, Cycle 1

Figure 5.9 SIMULATE-3 [ ], HFP, 12000 MWD/MTU, ARO, Relative Power Distribution [ ], Comanche Peak Steam Electric Station Unit 1, Cycle 1

Figure 5.10 SIMULATE-3 [ ] ARO Core Average Axial Power Distribution [ ]: A) BOL, HFP; B) MOL, HFP; C) EOL, HFP; Comanche Peak Steam Electric Station Unit 1, Cycle 1

[

]

Figure 5.11 SIMULATE-3 [ ] ARO Core Average Axial Exposure Distribution [ ]: A) BOL, HFP; B) MOL, HFP; C) EOL, HFP; Comanche Peak Steam Electric Station Unit 1, Cycle 1

Figure 5.12 SIMULATE-3 [ ] Integral Xenon Worth [ ] Following Plant Start-up, BOL, HFP, Comanche Peak Steam Electric Station Unit 1, Cycle 1

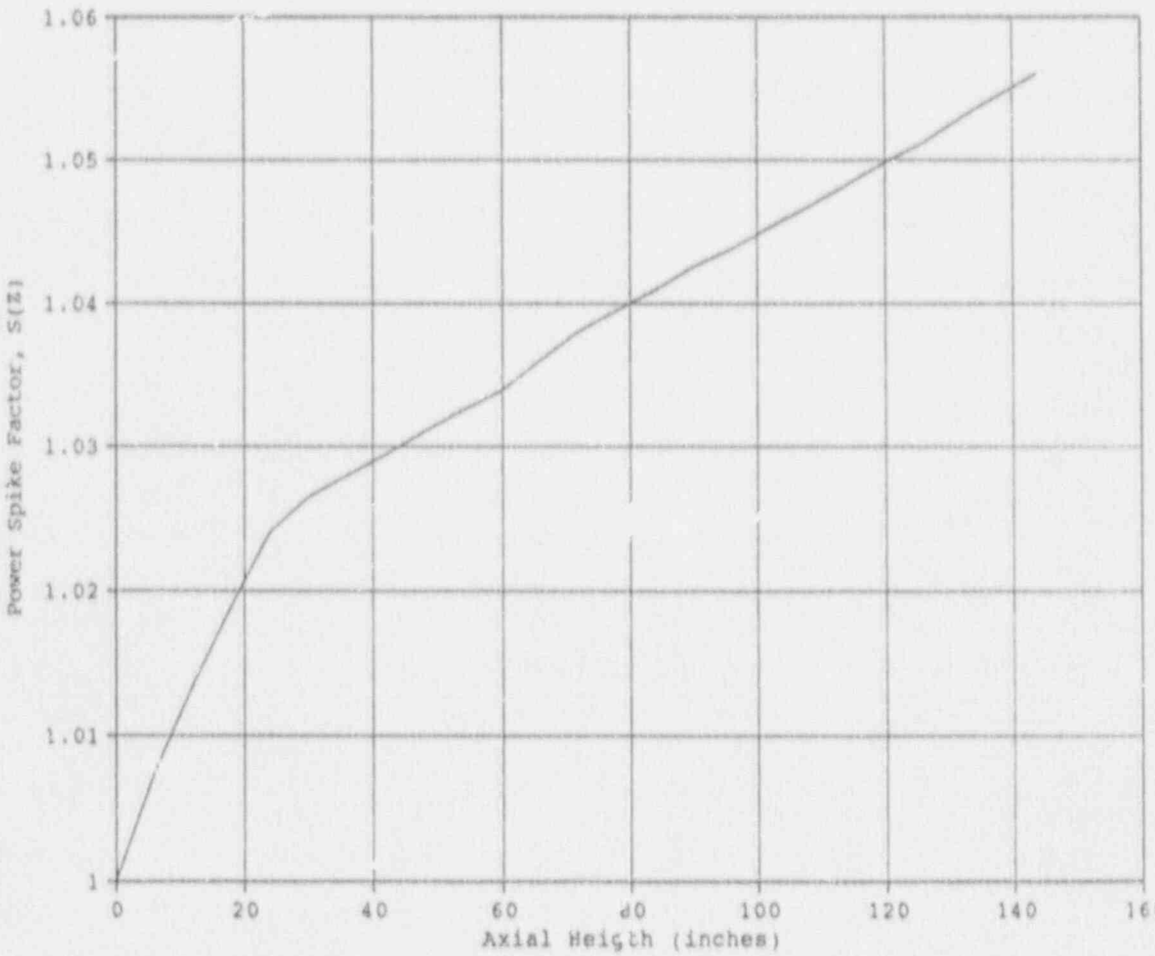
Figure 5.13 SIMULATE-3 [ ] Integral Xenon Worth [ ] Following Plant Trip After Steady State Operation at HFP, BOL, Comanche Peak Steam Electric Station Unit 1, Cycle 1

Figure 5.14 [ ] Integral Xenon Worth [ ] Following Plant Trip After Steady State Operation at HFP, EOL, Comanche Peak Steam Electric Station Unit 1, Cycle 1

Figure 5.15 [

]

Figure 5.16 Densification Power Spike Factor as a Function of Core Height (Typical)



CHAPTER 6

CONCLUSIONS

Development of the Overpower N-16 and Overtemperature N-16 reactor trip setpoint methodology to be utilized by TU Electric has been completed. The methodology draws heavily on standard industry practice for specification of such setpoints. Calculation of the axial power distributions for establishment of the f(ΔI) trip reset functions relies on the approach developed and [ ].

TU Electric has adopted the principal attributes of [

]. The analytical bases of the methodology have been presented and the application of that methodology to CPSES Unit 1, Cycle 1 has been demonstrated.

It is concluded that the TU Electric Overpower N-16 and Overtemperature N-16 trip setpoint methodology and power distribution control analysis and is acceptable for use in support of design, licensing, start-up and operation of CPSES.

CHAPTER 7

REFERENCES

1. Y. X. Sung, H. B. Giap, "VIPRE-01 Core Thermal-Hydraulic Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications," RXE-89-002, TU Electric, June, 1989.

2. T. Morita, L. R. Scherpereel, N. J. Dzikowski, R. E. Radcliffe, and D. M. Lucoff, "Topical Report Power Distribution Control and Load Following Procedures," WCAP-8403, Westinghouse Electric Corporation, September 1974.

3. [ ].

4. [ ].

5. [ ].

6. [ ].

7. [ ].

8. [

].

9. [

].

10. [

].

11. M. Edenius, A. Ahlin, and B. Forssen, "CASMO-3: A Fuel Assembly Burnup Program User's Manual," Studsvik/NFA - 88/48, Studsvik of America, September, 1988.

12. D. J. Edwards, L. E. Kostyniak, F. A. Monger, R. M. Rubin, C. E. Willingham, "Steady State Reactor Physics Methodology," RXE-89-003-P, TU Electric, July, 1989.

13. K. S. Smith, J. A. Umbarger, and D. M. Ver Planck, "SIMULATE-3: Advanced Three-Dimensional Two-Group Reactor Analysis Code User's Manual," Studsvik/SOA - 88/1, Studsvik of America, March, 1988.