PRAIRIE ISLAND UNITS 1 AND 2

SAFETY EVALUATION WITH INCREASED ENTHALPY RISE FACTOR

NSPNAD-8406

May 1984

Prepared by Dary Rautmann Date 5/17/84

Reviewed by Tom Kapitz Date 5/21/84

Approved by Roger O. Anderson Date 5/21/84

LEGAL NOTICE

This report was prepared by or on behalf of Northern States Power Company (NSP). Neither NSP, nor any person acting on behalf of NSP:

a. Makes any warranty or representation, express or implied, with respect to the accuracy, completeness, usefulness, or use of any information, apparatus, method or process disclosed or contained in this report, or that the use of any such information, apparatus, method, or process may not infringe privately owned rights; or

b. Assumes any liabilities with respect to the use of, or for damages resulting from the use of, any information, apparatus, method, or process disclosed in the report.

DISTRIBUTION

C. E. Larson, Director Nuclear Generation	MS3
G. H. Neils, Gen Mgr Hdqtrs Nuc Group	MS4
L. R. Eliason, Gen Mgr Nuclear Plants	MS3
E. L. Watzl, PI Plant Manager	PI
M. A. Klee, Supt Nuclear Engineering (2)	PI
D. M. Musolf, Mgr Nuclear Support Services (25)	MS4
C. A. Bonneau, Supt Core Analysis	MS4
T. L. Breene, Engineer II	MS4
D. H. Peterson, Mgr Fuel Supply	MS4
J. L. Karalus, Adm Fuel Supply Contracts	MS4
D. A. Rautmann, Supt Safety Analysis	MS4
L. McCarten, Nuclear Engineer	MS4
W. J. Lax, Engineer II	MS4
D. W. Dean, Engineer I	MS4
P. J. Riedel, Engineer I	MS4
J. K. Kapitz, Engineer II	MS4
T. J. Tasto, Engineer Associate	MS4
J. M. Garrick, Engineer Associate	MS4
C. S. Gantner, Engineer Associate	MS4
P. D. Pankratz, Engineer Associate	MS4
Exxon Nuclear Inc. (5)	Richland, WA
R. L. Grow	UAI
R. C. Kern	UAI
Nuclear Analysis Department File System (4)	MS4

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION

This report summarizes the calculations completed by NSPNAD in support of the Margin Improvement Program - Phase II for Prairie Island Units 1 and 2.

In Phase II of the Margin Improvement Program, the Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}$, is increased from 1.55 to 1.65.

The LOCA analysis was performed by the current fuel vendor (Exxon) and the results are reported in reference 1.

The non-LOCA analyses were performed by the NSP Nuclear Analysis Department for Prairie Island Unit 1 Cycle 9 and Unit 2 Cycle 8, the currently operating cycles. These analyses have been divided into four distinct parts consisting of: calculations for the limiting transients, evaluation of the rod bowing penalty, generation of the safety limit curves, and evaluation of the adequacy of the overtemperture ΔT trip function. The basic methodology and the analytical results which were obtained are presented in this report.

2.0 CALCULATIONAL MODELS AND METHODOLOGY

2.1 Calculational Models

The increased enthalpy rise factor calculations were performed using the NSPNAD Reload Safety Evaluation Methods for PWRs.(3) There have been no changes made to the codes since the analysis of Prairie Island Unit 2 Cycle 8.(5)

2.2 Methodology

2.2.1 Transient Analysis

The non-LOCA transient analyses were performed by the NSP Nuclear Analysis Department (NSPNAD) using methods(2)(3) which have been approved by the U.S. Nuclear Regulatory Commission (NRC). These methods have been demonstrated to be conservative for a complete spectrum of Reload Safety Evaluation transients, except LOCA, so that they can be used to perform analysis relating to licensing actions.

2.2.2 Rod Bow Penalty

The methodology for calculating the reduction in MDNBR due to rod bow for PI 1 Cycle 9 and PI 2 Cycle 8 with $F\Delta H$ = 1.55 is described in reference 4. After the RSE reports for these cycles(5)(6) were published, Exxon received NRC approval of a new computational method for evaluating the effect of fuel rod bowing on MDNBR.(7) The new method still uses the equation:

$$MDNBR_B = MDNBR_{NB}\ (1 - \delta_B)$$

where:

$MDNBR_{NB}$ = MDNBR for nonbowed fuel
$MDNBR_B$ = MDNBR for bowed fuel
δ_B = fractional reduction in MDNBR due to fuel rod bowing

The new method has changed the calculation of δ_B. δ_B is now defined as:

$$\delta_B = 0.0 \text{ for } 0 \le \Delta C/C_o \le 0.5$$

and

$$\delta_B = \frac{\Delta C/C_o - 0.5}{0.5} * \delta_{BOW} \text{ for } \frac{\Delta C}{C_o} > 0.5$$

This change in methodology was made because rod bow DNB test results have shown that the penalty on MDNBR is negligible until the fractional gap closure is well in excess of 0.5. This new methodology substantially reduces the rod bow penalty.

2.2.3 Safety Limit Curves

The safety limit curves define the regions of acceptable operation with respect to average temperature, power, and pressurizer pressure. The boundaries of acceptable operation are defined in terms of thermal overpower limit (fuel melting), thermal overtemperature limit (cladding damage based on DNB considerations), and locus of points where the steam generator safety valves open. These limits are used to set the overpower and overtemperature ΔT trip setpoints.

The thermal overpower limit to prevent fuel melting is protected by the thermal overpower trip. Fuel melting calculations are predominantly dependent on the total peaking factor, F_Q, and the fuel type analyzed, and relatively independent of the enthalpy rise factor, $F_{\Delta H}$. Previous calculations were based on Westinghouse fuel with an F_Q of 2.58.[8] The Exxon fuel type is sufficiently similar to the Westinghouse design so that for an F_Q of 2.32, the current Technical Specifications value, the previous analysis, including the overpower trip setpoints, will bound the present core. Therefore, increasing $F_{\Delta H}$ from 1.55 to 1.65 will not effect the thermal overpower limit calculations.

The opening of the steam generator safety valves impose a physical limit on the reactor power and temperature, dependent only on the reactor system characteristics and independent of peaking factors. Therefore, an increase in $F_{\Delta H}$ from 1.55 to 1.65 will not effect this limit.

Increasing $F_{\Delta H}$ from 1.55 to 1.65 will significantly impact the results of the thermal overtemperature limit calculations.

For the overtemperature limit, the following four limiting criteria are used:

1. Vessel exit temperature < 650 °F (design temperature limits).

2. Vessel exit temperature < saturation temperature (ensures power ~ ΔT).

3. Minimum DNBR > 1.3 (fuel damage limit).

4. Hot channel exit quality < 15% (limit for W-3 CHF correlation).

The first two criteria result in a single limit on vessel exit temperature. This limit is easily determined from an energy balance on the vessel at different pressures.

$$h_{in} + Q/m = h_{out}$$

where:

Q = core power, Btu/hr
m = vessel design flow rate = 68.2 E + 6 lbm/hr
h_{in} = inlet enthalpy, Btu/lbm
$h_{out} = \left.\begin{matrix} h_f \\ h\ (650\ °F) \end{matrix}\right\}$ lowest value, Btu/lbm

The third criteria on DNBR is evaluated using the thermal margin methodology described in Reference 3 Appendix C. The following key assumptions are made in generating the DNB limits.

- An 1/8 core COBRA IIIC/MIT hot channel model is used.

- The radial peaking factor, $F^N_{\Delta H}$, will obey the following equation at powers below 100%, provided the control rod insertion limits are observed.

  $$F^N_{\Delta H}(P) = 1.65\ [1 + .2\ (1-P)]$$

  where P is the fraction of rated core power.

  At power levels above 100%, the value of $F^N_{\Delta H}$ is 1.65.

- The reference axial power distribution, an upskewed cosine with a peak to average value, F^N_Z of 1.365 is used. The effect of other axial power distributions will be discussed in section 2.2.4.

- The coolant flow rate will be the Thermal Design Flow, which is 68.2 Mlbm/hr.

This model is then iterated to find the inlet temperature (and hence average temperature) that will give a MDNBR of 1.3 for a given pressure and power level. The resulting functions of T_{avg} versus core power, at different pressures, define the limits for this criteria.

The fourth criteria limits the hot channel outlet quality to less than + 15%. This criteria assures that the quality in the hot node (point of MDNBR) will also be less than + 15% which is the upper range of applicability of the W-3 DNB correlation for the range of pressures over which the core DNB limits are required. This criteria is evaluated using the same model used for the MDNBR criteria only now the hot channel model is iterated to give an exit quality of + 15%.

2.2.4 Overtemperature ΔT Setpoint Verification

The thermal overtemperature limit is protected against fuel damage by the overtemperature ΔT reactor trip which compares the difference of the measured hot and cold leg temperatures to a dynamically calculated setpoint. The setpoint in Laplace notation is given by.

$$\Delta T^{setpoint}_{OT} = \Delta T_o^{OT} \left(K_1 - K_2 \frac{(1+\tau_1 s)}{(1+\tau_2 s)} (T^{ave}(s) - T_o^{OT}) + K_3 (P(s) - P_o) - f_{OT}(\Delta I)\right)$$

where:

ΔT_o^{OT} = Indicated ΔT at rated power

T_o^{OT} = Reference average temperature

P_o = Reference pressurizer pressure

$T^{ave}(s)$ = Measured average temperature

$P(s)$ = Measured pressurizer pressure

K_1, K_2, K_3 = Constants (> 0)

τ_1, τ_2 = Time constants

$f_{OT}(\Delta I)$ = A function of the measured difference between top and bottom excore detectors (≥0)

For steady state operation with $\Delta I=0$, $\Delta T_{OT}^{setpoint}$ reduces to:

$\Delta T_{OT}^{setpoint}$ (steady state) =

$$\Delta T_o^{OT} (K_1 - K_2 (T^{ave} - T_o^{OT}) + K_3 (P - P_o))$$

K_1 includes an error allowance for calibration and instrument errors. $\Delta T_{OT}^{setpoint}$ (steady state) is then compared to the safety limit curves for the region where the thermal overtemperature limits are bounding. If $\Delta T_{OT}^{setpoint}$ (steady state) $\leq \Delta T$ safety limit, then the current setpoint is acceptable. If not, then the setpoint constants must be recalculated. The procedure for calculation of the overtemperature ΔT trip setpoint constants is described in Reference 9.

For transient operation, the dynamic terms, τ_1 and τ_2, in the overtemperature ΔT trip function compensate for inherent instrument delays and piping lags between the reactor core and the loop temperature sensors. These affects are system dependent and independent of peaking factors. Thus, an increase in $F_{\Delta H}$ will not effect these parameters.

The final consideration is to determine the adequacy of the f_{OT} (ΔI) function. This parameter is included in the setpoint, because the Westinghouse methodology relating to minimun DNB utilizes a conservative axial power profile which is applicable only when the axial offset (which is related to ΔI) is within a certain operating band near zero. Power distribution control at the PI units during normal operations keeps the axial offset essentially within this band. Thus, if the power distribution becomes adversely skewed outside this band, the setpoint is appropriately reduced by $f(\Delta I)$ to ensure that the same thermal margins are maintained. The Westinghouse methodology consists of running a selected number of operational transients which span a wide range of anticipated load follows, boron dilutions with the rod controller in the manual and automatic mode, cooldown accidents, and rod withdrawal transients.

The axial power distributions which result during these transients are then used to compute the setpoint reductions to maintain a fixed thermal margin at full power conditions. The $f(\Delta I)$ is then set to bound all of these points to provide the necessary protection.

The NSP methodology approached this analysis in a similar manner. The core safety limit curves were developed using a reference hot channel axial power distribution (i.e. upskewed cosine) for $\Delta I_{core} = 0$. The effect of normal and offnormal operation on the hot channel power distribution and on ΔI_{core} is determined using the methods described in Reference 2. These distributions are obtained during the normal design analysis phase and bound both normal and transient operation including Condition II events. These distributions are input to the COBRA hot channel model and the power level and inlet temperature varied until the thermal margin (section 2.2.3) is the same as the reference case ($\Delta I = 0$). The steady state $\Delta T_{OT}^{setpoint}$ equation is then solved for for $f_{OT}(\Delta I)$.

$$f_{OT}(\Delta I) = K_1 - K_2 (T^{ave} - T_o^{OT}) + K_3 (P - P_o) - \frac{\Delta T_{OT}^{setpoint}}{\Delta T_o^{OT}}$$

$\Delta T_{OT}^{setpoint}$ is taken to be the calculated ΔT at the safety limit line. Since the overtemperature ΔT trip setpoint bounds the safety limit curves for all pressures (using the reference distribution), $f_{OT}(\Delta I{=}0)$ Reference ≤ 0. Therefore, the calculated $f_{OT}(\Delta I)$ distribution must be biased so that $f_{OT}(\Delta I{=}0)$ Reference = 0.

This biased distribution is then used to determine whether the current $f_{OT}(\Delta I)_{setpoint}$ distribution is bounding (see section 3.5). This calculation is only performed at one pressure. Varying the pressure will only change the bias and will not effect the results.

3.0 RESULTS

3.1 Input Parameters

The analyses in this report are performed for Prairie Island Unit 1 Cycle 9 and Unit 2 Cycle 8, the currently operating cycles. Thermal-hydraulic parameters for full power operation are summarized in Table 3.1-1. All other parameters including neutronics, setpoints and delays are outlined in References 5 and 6. In all cases, the parameters used in the analyses bound the actual values for the Prairie Island plants.

3.2 Transient Analysis

For steady state operation at HFP, raising the $F_{\Delta H}$ limit to 1.65 reduces the MDNBR by about 11% and the MDNBR margin by about 29% for both cycles. Table 3.2 1 summarizes the results for PI 1 Cycle 9 and PI 2 Cycle 8.

For non LOCA transients, changes in $F_{\Delta H}$ only impact events for which fuel damage is a consideration. For the PI units, the limiting transients with respect to the thermal margin fuel damage criterion on DNBR are: The RCC assembly misalignment, slow rod withdrawal at power, locked rotor, the large steamline break, and the rod ejection transient.

The RCC assembly misalignment, large steamline break, and rod ejection transient will not be effected by a change in the Technical Specification $F_{\Delta H}$ limit for the currently operating cycles, i.e. PI 1 Cycle 9 and PI 2 Cycle 8. These transients are initiated from offnormal initial conditions, i.e. dropped or misaligned RCCA at HFP for the RCCA misalignment transient and HZP on rod stuck out for the steamline break transient and ejected rod at HZP and HFP for the ejected rod transient. The peak ng factors for these transients typically will excced the Technical Specifications criteria. These transients are therefore evaluated using the calculated peaking factors, including reliability factors rather than the Technical Specification limits. The calculated transient response is therefore dependent on the core loading and is independent of any change in the T.S.

limit on $F_{\Delta H}$ once cycle operation has begun. Therefore, the previous calculations (reference 5 and 6) are still valid for PI 1 Cycle 9 and PI 2 Cycle 8, respectively.

The locked rotor transient, a class IV transient, is evaluated based on the number of failed fuel pins. The failure criteria used is a MDNBR of less than 1.3, evaluated using the W-3 correlation. A pins census ($F_{\Delta H}$ per pin) is done based on the calculated power distribution, including reliability factors, at HFP. From this the MDNBR for each pin is determined. The calculated pin power distribution, at HFP, is dependent only on the core loading, once cycle operation has begun, and will not be effected by a change in the T.S. limit on $F_{\Delta H}$. Therefore, the previous calculations (ref 5 and 6) are still valid, for PI 1 Cycle 9 and PI 2 Cycle 8, respectively.

The slow rod withdrawal transient assumes initial operation at HFP with $F_{\Delta H}$ at the T.S. limit. This transient must therefore be reanalyzed using an $F_{\Delta H}$ of 1.65. The slow rod withdrawal event was reanalyzed and the resulting transient DNBR is shown in Figures 3.2-1 and 3.2-2, for PI 2 Cycle 8 and PI 1 Cycle 9, respectively, assuming values of $F_{\Delta H}$ = 1.55 and 1.65. The former case shows the transient terminating earlier relative to the latter case because the scram was calculated to occur earlier. The difference in scram times is due to the fact that the 1.65 case included a more conservative evaluation of the overtemperature delta T trip setpoint instrument and measurement uncertainties than the 1.55 case. This was done in order to make the analysis more consistant with current Westinghouse methodology. The results show sufficient margin with respect to meeting the 1.3 limiting criterion.

3.3 Rod Bow Analysis

The most limiting operational transient for PI 1 Cycle 9 and PI 2 Cycle 8 is the slow rod withdrawal transient. For this event the MDNBR was calculated to be 1.441 and 1.440, respectively. A further reduction in DNBR will be caused by rod bowing at high exposures. The methodology for calculation of the DNBR reduction due to rod bow is explained in section 2.2.2.

The maximum anticipated fractional gap closure was calculated at an assembly average exposure of 49,500 GWD/MTU. This exposure will bound the peak pellet exposure limit of 55,000 MWD/MTU.

The results of the rod bow penalty calculations are shown on Table 3.3.-1. Both plants will lose an additional 3.4% on MDNBR due to rod bowing. This brings the MDNBR during a slow rod withdrawal transient down to 1.392 and 1.391 for PI 1 Cycle 9 and PI 2 Cycle 8, respectively.

The results of this analysis show that by using an $F_{\Delta H}$ of 1.65, together with the new rod bow penalty methodology, both Prairie Island 1 Cycle 9 and Prairie Island 2 Cycle 8 are adequately protected for all the transients and accidents for which fuel damage is prohibited.

3.4 Safety Limit Curves

The safety limit curves were generated using the methodology described in section 2.2.3. These curves were generated using a Prairie Island Unit 1 Cycle 9 hot channel model. The MDNBR limit curve was established on the basis of 1.365 which includes an additional 5% over the required limit to compensate for cycle to cycle variations in the hot channel model, thereby making this analysis generic to Prairie Island units with TOPROD fuel. The 5% margin was established based on historical data which shows 1% to 2% variations in DNBR between cycles with similar fuel types. The applicability of this number (5%) will be checked as part of the Reload Safety Evaluation process.

The solid curves of Figure 3.4-1 represent the loci of points of thermal power, coolant pressure, and coolant average temperature for which either the coolant enthalpy at the core exit is limiting or the DNB ratio is limiting. For the 1685 psig and 1985 psig curves, the coolant average enthalpy at the core exit is equal to saturated water enthalpy below power levels of 81% and 65%, respectively. For the 2235 psig and 2385 psig curves, the coolant average temperature at the core exit is equal to 650 °F below power levels of 58% and 66%, respectively. For all four curves, the DNBR is equal to 1.3 at higher power levels. The area of safe operation is below these curves.

3.5 Overtemperature ΔT Setpoint Verficiation

The applicability of the current overtemperature ΔT trip setpoint was verified as described in section 2.2.4. The overtemperature ΔT setpoints used in the analysis are given in Table 3.4-1.

The overtemperature ΔT trip limit (with $\Delta I=0$) was compared to the safety limit curves generated in section 3.4. The results of this comparison for the two limiting pressures of 1700 and 2400 psia, the low and high pressurizer pressure trip setpoints respectively, are shown in Figures 3.5-1 and 3.5-2, respectively. In all cases, the overtemperature ΔT trip setpoint is seen to provide adequate thermal margins with respect to the safety limits, since the setpoint always lies below all the limits.

Figures 3.5-3 and 3.5-4 show the results of the $f(\Delta I)$ function adequacy study for Prairie Island Unit 2 Cycle 8 and Unit 1 Cycle 9, respectively. These results show that the current setpoint adequately covers this core design. In fact, with the exception of the one point at $\Delta I = 31$, for PI 1 Cycle 9, the reference power distribution is more conservative so that the need for the the $f(\Delta I)$ function is only minimal. These results confirm the fact that the reference axial power distribution has been chosen very conservatively and bounds all shapes. The adequacy of the $f(\Delta I)$ function will be checked each cycle as part of the Reload Safety Evaluation process.

TABLE 3.1-1

Parameter Values Used in Transient Analysis

	Analysis Input Value
Core	
Total Core Heat Output, Mw (102%)	1,683.0*
Heat Generated in Fuel, %	97.4
System Pressure, psia	2,220.0**
Hot Channel Factors	
Total Peaking Factor, F_Q^T	2.32
Enthalpy Rise Factor, $F_{\Delta H}^N$	1.65
Total Coolant Flow, lb/hr	68.20×10^6
Effective Core Flow, lb/hr	65.13×10^6
Reactor Inlet Temperature, °F	534.5***
Steam Generators	
Calculated Total Steam Flow, lb/hr	7.26×10^6
Steam Temperatue, °F	510.8
Feedwater Temperature, °F	427.3

* Includes +2% uncertainty
** Includes -30 psi uncertainty
*** Includes +4 °F uncertainty

TABLE 3.2-1

Summary of BOC, HFP Base Case MDNBR

	$F_{\Delta H}$=1.55	$F_{\Delta H}$=1.65	MDNBR % Change	Margin % Change
PI 1 Cycle 9	2.083	1.859	-10.8	-28.6
PI 2 Cycle 8	2.114	1.881	-11.0	-28.6

TABLE 3.3-1

Slow Rod Withdrawal
Transient and Thermal Margin Results

Prairie Island 1 Cycle 9

	$F\Delta H=1.55$	$F\Delta H=1.65$
Rod Ave. Heat Flux @ time of MDNBR (Btu/hr ft^2)	341,335	366,953
$MDNBR_{NB}$	1.760	1.441
$(\Delta C/C_o)$ 95/95	0.485	0.5493
δ_{BOW}	0.328	0.348
δ_B	0.159	0.0338
$MDNBR_B$	1.484	1.392

Prairie Island 2 Cycle 8

	$F\Delta H=1.55$	$F\Delta H=1.65$
Rod Ave. Heat Flux @ time of MDNBR (Btu/hr ft^2)	340,930	366,588
$MDNBR_{NB}$	1.790	1.440
$(\Delta C/C_o)$ 95/95	0.485	0.5493
δ_{BOW}	0.327	0.348
δ_B	0.1586	0.0348
$MDNBR_B$	1.5	1.391

TABLE 3.4-1

Prairie Island Units 1 and 2

Overtemperature ΔT Trip Setpoints

ΔT_o^{OT} = 64.3 °F*

T_o^{OT} = 567.3

P_o = 2235 psig

K_1 = 1.202**

K_2 = 0.009

K_3 = 0.000566

$f(\Delta I)$ = a) for qt - qb within -12 to +9 percent $F(\Delta I)$ = 0

b) for each percent that the magnitude of qt - qb exceeds +9 percent the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.5% rated power

c) for each percent that the magnitude of qt - qb exceeds -12 percent the ΔT trip setpoint shall be automatically reduced by an equivalent of 1.5% rated power.

* ΔT_o at design flow

** Including uncertainties

Slow Rod Withdrawal
Minimum DNB Ratio vs. Time

Figure 3.2-1
Prairie Island 2, Cycle 8

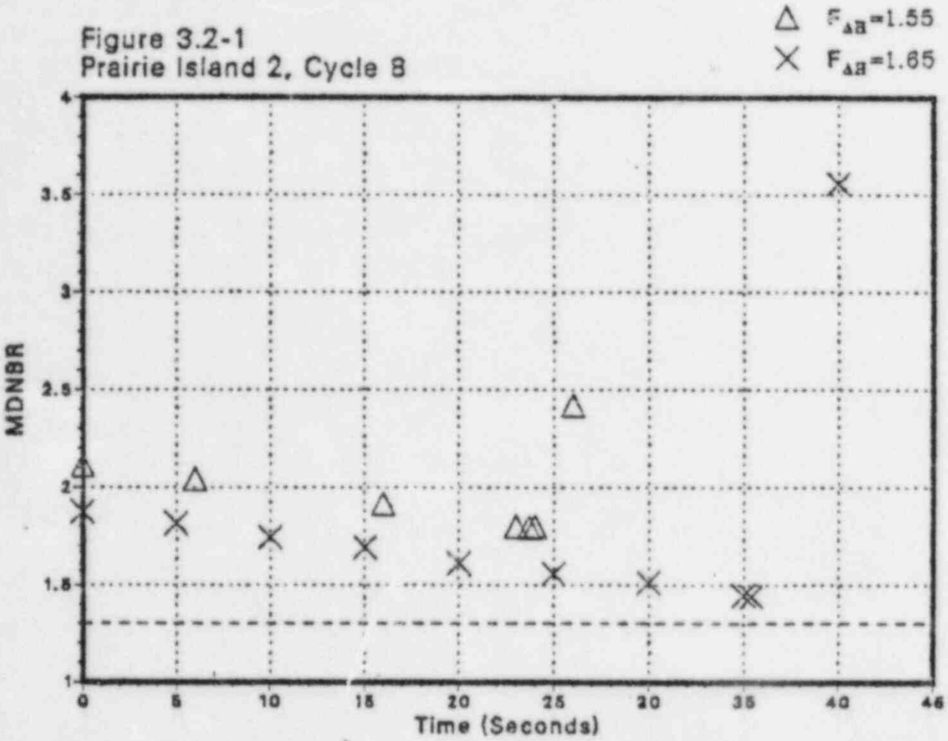
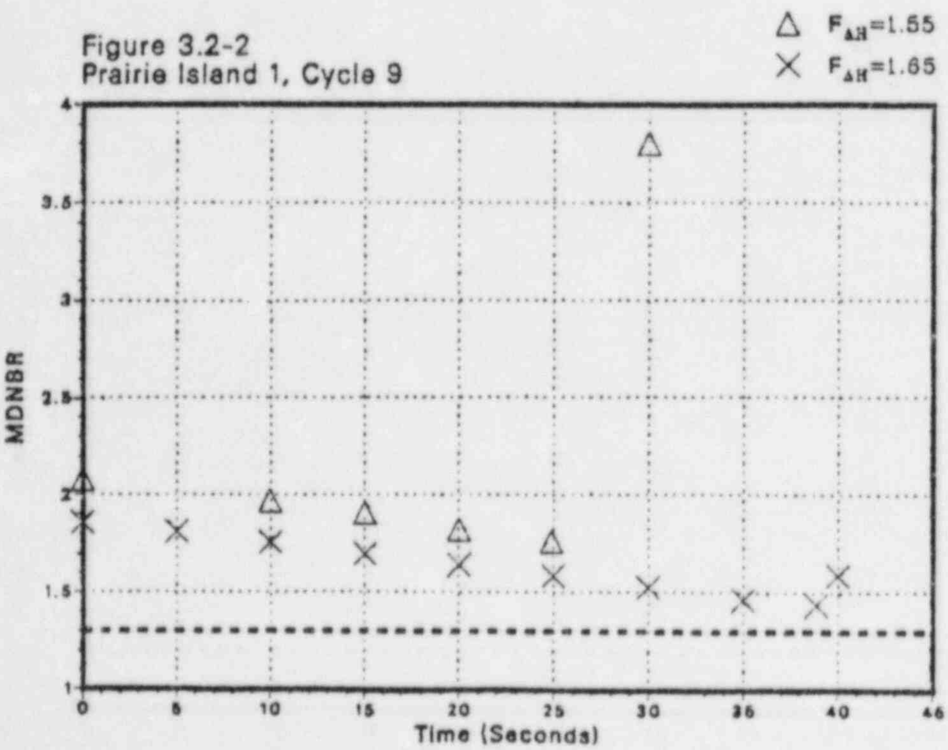


Figure 3.2-2
Prairie Island 1, Cycle 9

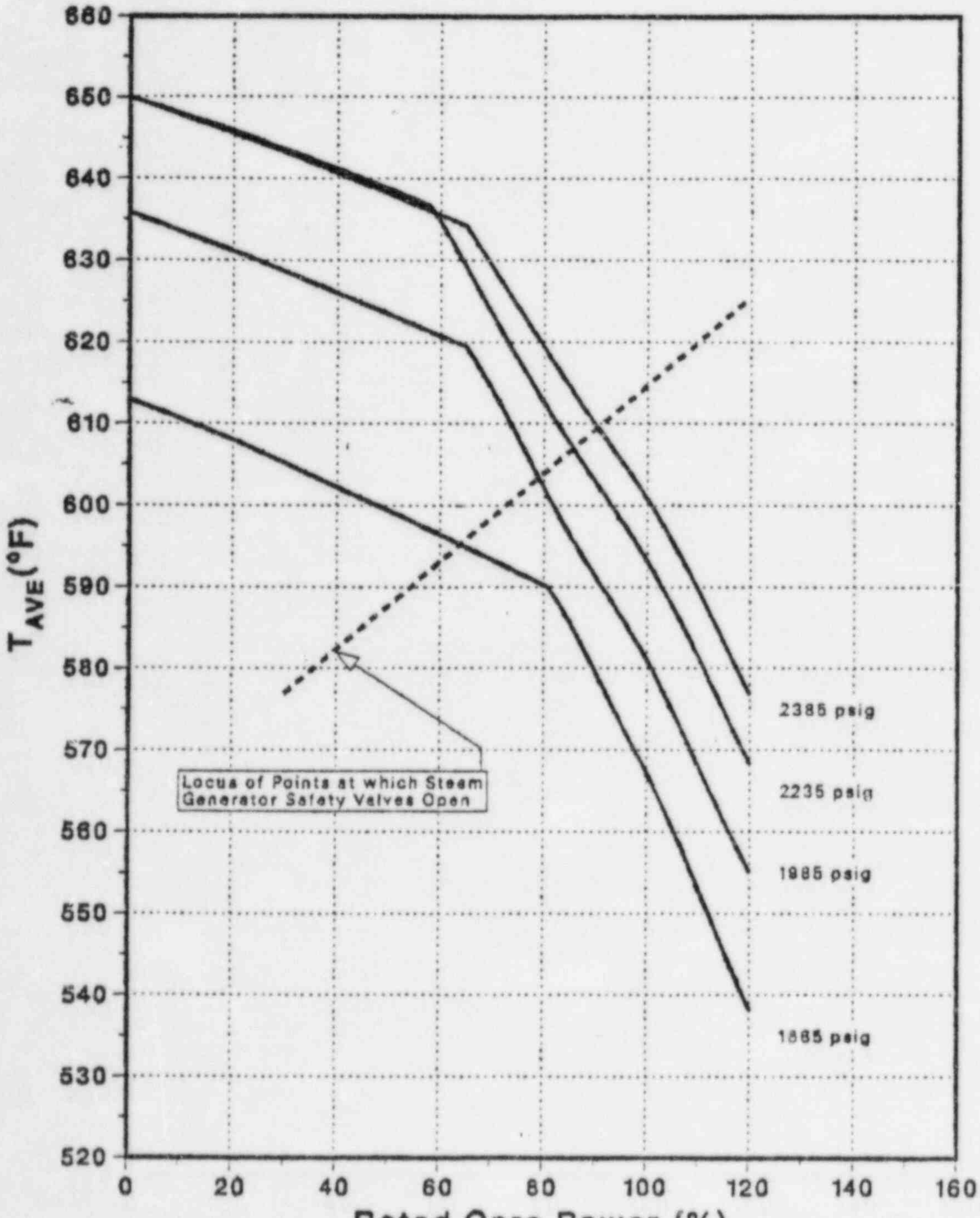


NSP

Prepared by NSP Safety Analysis, 4/23/84

Core Average Temperature Limits vs. Core Average Power

Figure 3.4 -1

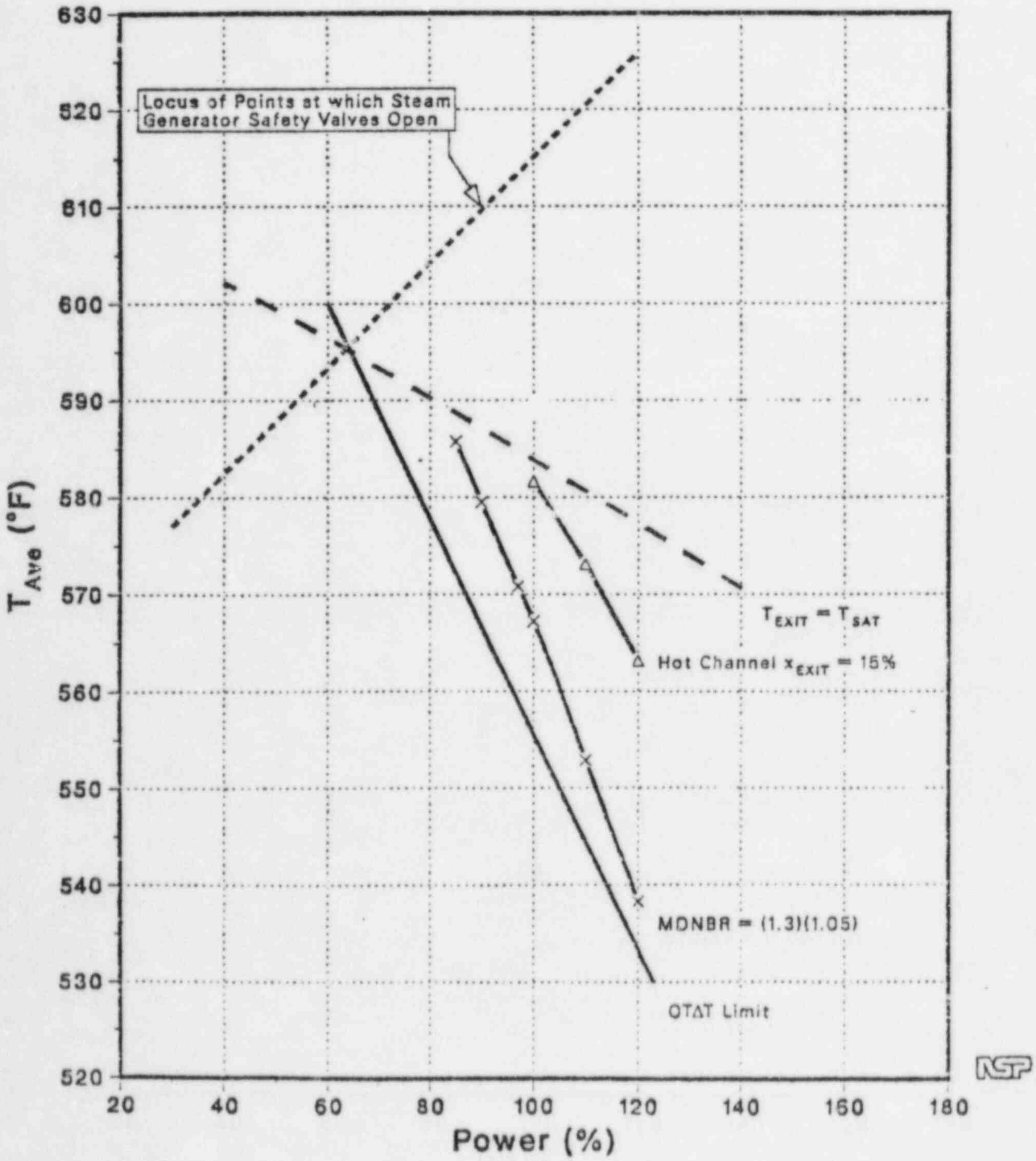


Prepared by NSP Safety Analysis, 3/16/84

FIGURE 3.5-1

Core Average Temperature Limits vs. Core Average Power

FΔh = 1.65, Fq = 2.32, Upskewed
1700 psia



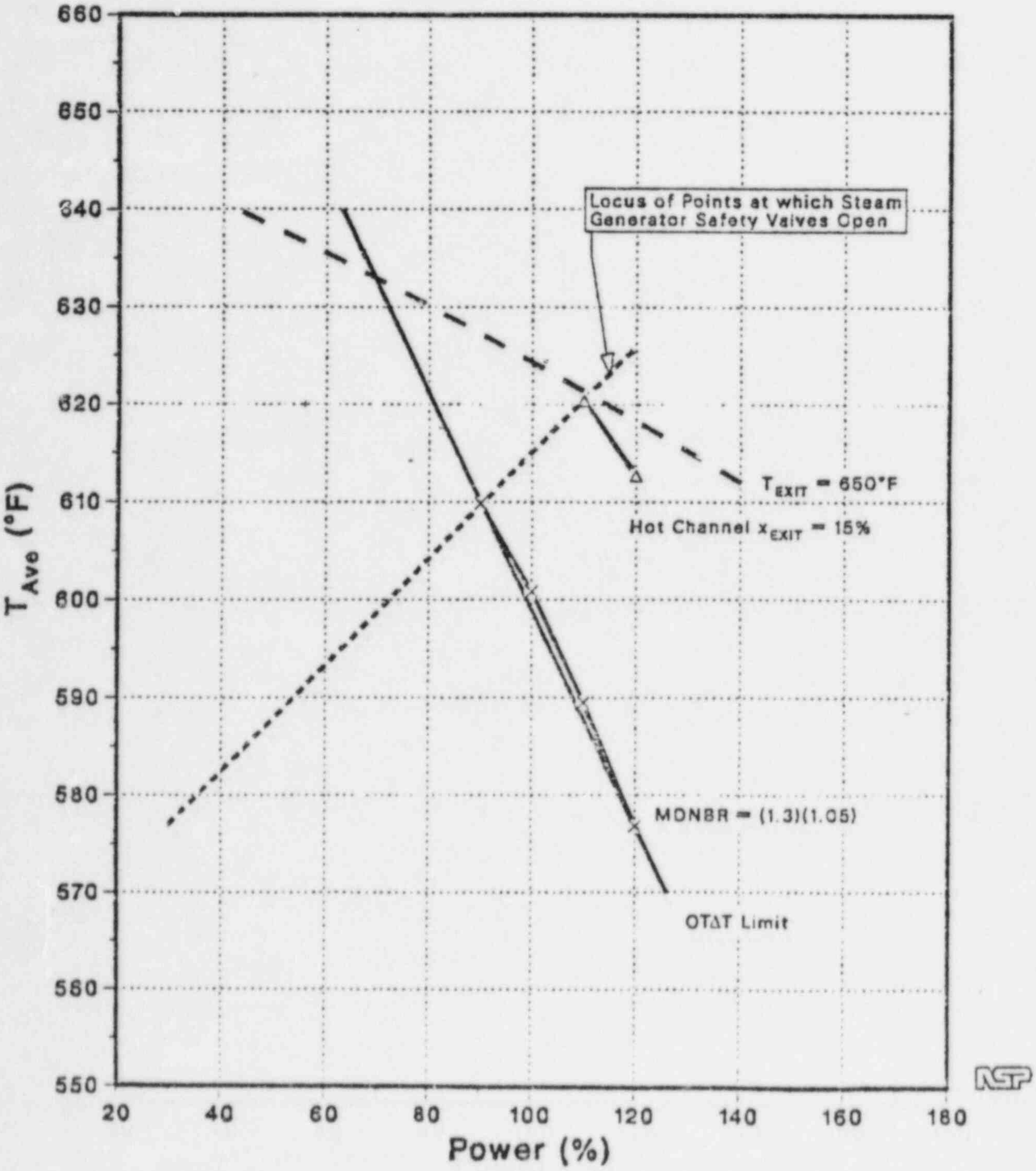
Prepared by NSP Safety Analysis, 2/28/84

FIGURE 3.5-2

Core Average Temperature Limits vs. Core Average Power

FΔh = 1.65, Fq = 2.32, Upskewed

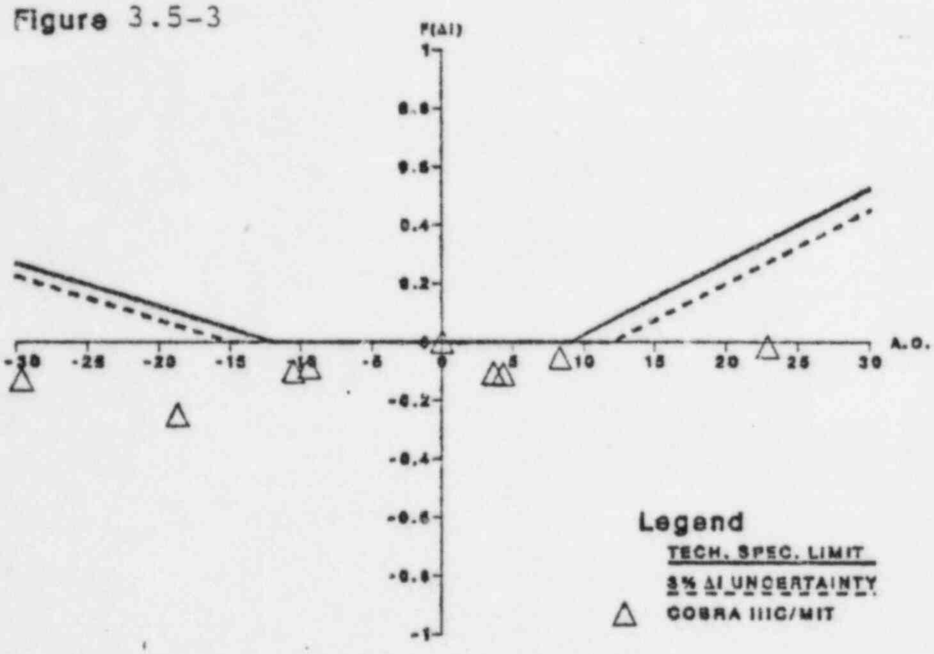
2400 psia



Prepared by NSP Safety Analysis, 2/28/84

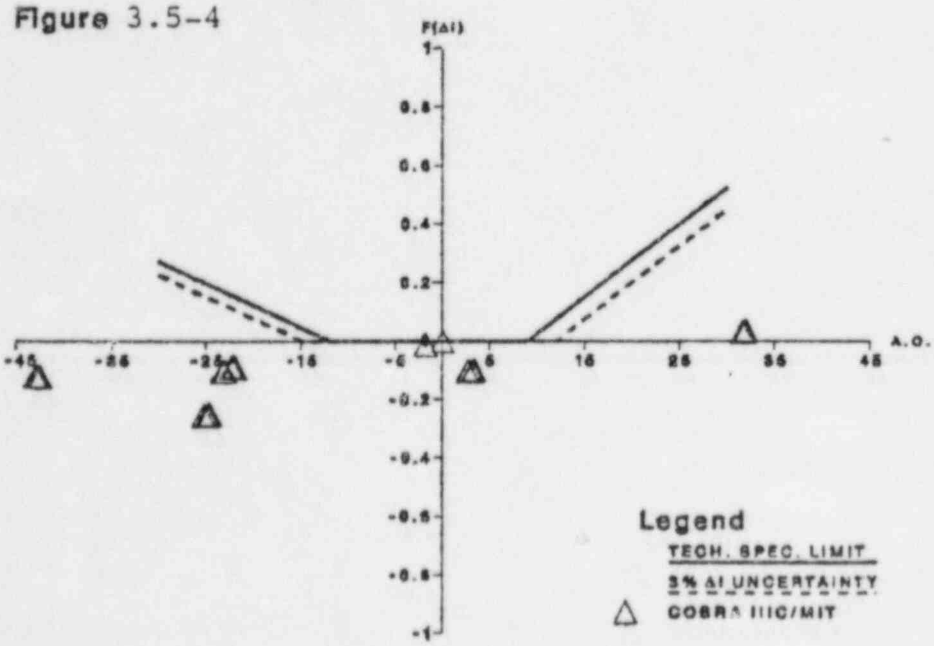
PRIARE ISLAND UNIT 2 CYCLE 3
$F(\Delta I)_{calc.}$ vs. Axial Offset

Figure 3.5-3



PRIARE ISLAND UNIT 1 CYCLE 9
$F(\Delta I)_{calc.}$ vs. Axial Offset

Figure 3.5-4



Prepared by NSP Safety Analysis, 4/24/84

4.0 Summary and Conclusions

Analyses have been performed to demonstrate an increase in the Technical Specifications limit for $F_{\Delta H}$ from 1.55 to 1.65 for the two Prairie Island units will not result in any reduction in safety margins. This analysis covered the entire spectrum from anticipated transients through design basis LOCA's. The LOCA analysis was performed by Exxon[1] which resulted in a slight reduction in FQ to meet all licensing acceptance criteria. The non-LOCA analysis was performed by NSPNAD utilizing our in-house analytical methods. This analysis demonstrated that the limiting transients with respect to minimum DNBR fuel damage are not adversely effected with respect to meeting the limiting criterion. In addition, the overtemperature ΔT trip setpoint has been shown to provide adequate protection to the safety limits and the existing $f_{OT}(\Delta I)$ function protects against all possible skewed axial power distributions.

5.0 References

1. "Prairie Island Units 1 and 2 Limiting Break LOCA ECCS Analysis with Increased Enthalpy Rise Factor" XN-NF-84-03, February 1983.

2. Qualification of Reactor Physics Methods for Application to PI Units" NSPNAD-8101P, December 1982.

3. "Reload Safety Evaluation Methods for Application to PI Units" NSPNAD-8102P, December 1982.

4. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing" Supplement 1, June 1979.

5. "Prairie Island Units 1 Cycle 9 Final Reload Design Report (RSE)" NSPNAD-8313P, October 1983.

6. "Prairie Island Unit 2 Cycle 8 Final Reload Design Report (RSE)" NSPNAD-8305P Rev.2, November 1983.

7. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing" Supplement 4, October 1983.

8. WCAP 8091, "Fuel Densification Prairie Island Nuclear Generating Plant Unit No. 1", March 1973.

9. WCAP 8746, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions", March 1977.