EXHIBIT H

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

License Amendment Request Dated January 13, 1986

PRAIRIE ISLAND UNITS 1 AND 2

SAFETY EVALUATION

OF

INCREASED FQ, FΔH, AND ISOTHERMAL TEMPERATURE COEFFICIENT

NSPNAD-8600

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1.0 INTRODUCTION

This report summarizes the calculations performed by the Northern States Power Nuclear Analysis Department (NSPNAD) in support of Technical Specification changes to FQ, FΔH, and isothermal temperature coefficient. The new FQ and FΔH limits are 2.30 and 1.60 The FΔH equation includes a new multiplier on the limit at reduced power, increasing from 0.2 to 0.3. The Technical Specification on isothermal temperature coefficient allows a +5 pcm/°F ITC from 0% to 70% power, and a 0.0 pcm/°F ITC above 70% power.

NSPNAD has analyzed the transient response of Prairie Island with the new Technical Specifications. The results show that Prairie Island still meets all transient acceptence criteria with the new Technical Specifications and therefore they involve no unreviewed safety questions.

Section 2 of this report describes the calculational models and methodology used for this analysis.

Section 3 contains the thermal-hydraulic design analysis.

Section 4 contains the accident analysis results.

2.0 CALCULATIONAL MODELS AND METHODOLOGY

2.1 Calculational Models

These calculations have been performed using the NSPNAD Reload Safety Evaluation Methods for PWRs (Reference 1). These methods have been submitted to the NRC for approval and are currently being reviewed.

2.2 Methodology

For this analysis NSPNAD has evaluated the limiting transients for Prairie Island. These limiting transients have been identified previously by Westinghouse (Reference 2), Exxon (Reference 3 and 4) and NSPNAD (Reference 1). The limiting transients are:

1. Fast Control Rod Withdrawal
2. Slow Control Rod Withdrawal
3. Loss of Power to Both Reactor Coolant Pumps
4. Locked Rotor in One Reactor Coolant Pump
5. Loss of Electric Load
6. Large Steam Line Break
7. Small Steam Line Break
8. Rupture of Control Rod Drive Mechanism Housing (RCCA Ejection)

The transients that are not reanalyzed are:

1. Uncontrolled RCC Assembly Withdrawal From a Subcritical Position
2. Startup of Inactive Loop
3. Feedwater System Malfunction
4. Excessive Load Increase
5. Loss of AC Power.

These transients have not been limiting in the past. The new Technical Specifications will not change the relative worth of various transients, so these transients will continue to be non-limiting. This conclusion is supported by the following:

A. While the new FQ and FΔH will affect the initial steady state MDNBR, it will not change the relative change in MDNBR. NSPNAD has performed Prairie Island transient analysis at several different values of FQ and FΔH without seeing any change in the relative severity of the accients.

B. Through the process of creating reload safety evaluations for several cycles at Prairie Island, NSPNAD has performed transient analysis over a wide range of ITC values. Some analyses have included a positive ITC and no change in the relative severity of the accidents was observed.

These results form the basis for concluding that the limiting transients identified in the past analysis will continue to be limiting under the new Technical Specifications on ITC and FQ, FΔH.

3.0 THERMAL HYDRAULIC DESIGN ANALYSIS

This section provides results of the thermal hydraulic design analyses for Prairie Island 1.

3.1 Design Criteria

The thermal and hydraulic design performance requirements for Prairie Island fuel are as follows.

1. The minimum departure from nucleate boiling ratio (MDNBR) will be:

   1.3 at overpower for ENC fuel using the W-3 correlation with corrections for non-uniform axial heating, cold wall effects, and a reduction in MDNBR due to fuel rod bowing.

   1.17 at overpower for Westinghouse fuel using the WRB-1 correlation with corrections for non-uniform axial heating and a reduction in MDNBR due to fuel rod bowing.

2. The fuel must be thermally and hydraulically compatible with the existing fuel and the reactor core throughout the life cycle of the fuel.

3. The maximum fuel temperature at design overpower shall not exceed the fuel melting temperature for ENC fuel and shall not exceed 4700 °F for Westinghouse fuel.

4. For ENC fuel the cladding upper temperature limits shall not exceed.

Inner surface temperature	850 °F
Outer surface temperature	675 °F
Average volumetric temperature	750 °F

3.2 Core Hydraulic Compatibility

The hydraulic compatibility of the Prairie Island fuel is discussed in Reference 5.

3.3 Thermal Margin

The most limiting transient for Prairie Island is the dropped rod - auto control event. The minimum DNBR was calculated to be 1.662 in an ENC assembly, using the W-3 correlation. Table 3.1 provides reference conditions for the analysis. Details of the plant transient analysis for Prairie Island are given in Section 4.1.2.4

3.4 Effect of Fuel Rod Bow on Thermal Hydraulic Performance

3.4.1 Rod Bow as Applied to DNBR Analysis - ENC Fuel

The calculation of the DNBR reduction as a result of rod bow considers both DNB tests with rod bow and the degree of bowing.

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

and,

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

$$\delta_B = 2.0 \times (\Delta C/C_o - 0.5) \times \delta_{BOW} \qquad \text{for } 0.5 \le \Delta C/C_o \le 1.0$$

where

$MDNBR_{NB}$ = MDNBR for nonbowed fuel

$MDNBR_B$ = MDNBR for bowed fuel

δ_B = fractional reduction in MDNBR due to bowing

$\frac{\Delta C}{C_o}\ 95/95$ = anticipated fractional gap closure as a function of exposure

δ_{bow} = MDNBR reduction associated with bow to contact.

The calculation of DNB rod bow to contact penalty is based on DNB tests with rod bow as referred to in the NRC's Interim Safety Evaluation Report on Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors. Exxon Nuclear Company's detailed methodology for calculating fuel rod bowing and its MDNBR effect is given in Reference 4.

The maximum anticipated fractional gap closure is based on rod bow measurements of ENC fuel similar to the Prairie Island design and currently being used in operating reactors. Reference 6 presents the results of rod bow measurements taken on ENC PWR reload fuel.

The data base obtained from the above measurements include approximately 11,000 independent measurements of rod-to-rod spacings for interior as well as peripheral rod bows. After two cycles of operation, the results indicate rod bow which is a small fraction of that required for bow-to-bow contact. Application of this data to the ENC TOPROD design is in accordance with the aforementioned SER and includes a 1.2 multiplier to account for cold-to-hot variations in measured rod spacings and a 1.5 multiplier to account for batch-to-batch variation. The maximum anticipated fractional gap closure through the fuel lifetime is:

$$\frac{\Delta C}{C_o} \; 95/95 = 0.5493$$

This value is valid for assembly exposures up to 49.5 GWD/MTU, which bound operation up to 55 GWD/MTU peak pellet exposure. Table 3.2 provides a comparison of key results and rod bow penalties from the analysis of the "dropped rod" event in section 4.0. The heat flux and pressure parameters in Table 3.2 correspond to the values calculated at the time of MDNBR. For conservatism, 60 psia has been added to the pressure prior to calculating the rod bow penalties shwon in Table 3.2. The bowed and unbowed MDNBR results are well above the allowable 1.3 value. Thus, no reduction in allowable reactor peaking is required as a result of a change in MDNBR due to rod bow for ENC fuel.

3.4.2 Rod Bow as Applied to DNBR Analysis - Westinghouse Fuel

The calculation of the DNBR reduction due to rod bowing for Westinghouse fuel is similar to that for ENC fuel (Reference 7).

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

where δ_B is given as a function of assembly average burnup. For 0.400 in O.D. OFA fuel the value of δ_B is:

0.050	(full flow)
0.055	(low flow)

This represents the rod bow penalty at an average assembly burnup of 33,000 MWD/MTU. While the amount of rod bowing increases beyond this exposure, the fuel is not capable of achieving limiting peaking factors due to the decrease in fissionable isotopes and the buildup of fission product inventory. The physical burndown effect is greater than the rod bowing effects which would be calculated based on the amount of bow predicted at those burnups.

Therefore, for the purpose of evaluating effects of rod bow on Westinghouse fuel, 33,000 MWD/MTU represents the maximum burnup of concern.

The bowed and unbowed MDNBR results are well above the allowable 1.17 value. Thus, no reduction in allowable reactor peaking is required as a result of a change in MDNBR due to rod bow for Westinghouse fuel.

3.5 Fuel Temperature Analysis

The fuel temperature analyses for Prairie Island are given in References 8 (ENC) and 9 (Westinghouse).

3.6 Safety Limit Curves

Safety limit curves for Prairie Island are given in Technical Specifications section 2.1. These curves define the region of acceptable operation in terms of core average temperature, power, and pressure. One of these limits is the thermal overtemperature limit, which prevents cladding damage based on DNB considerations. Due to the fact that DNB ratios are a function of core loading, the applicability of these limits will be verified by NSP on a cycle specific basis.

The thermal overtemperature limit is imposed on the reactor by the overtemperature ΔT reactor trip. This trip function is designed to trip the reactor before it exceeds the limits defined in the Technical Specifications in order to prevent DNB induced cladding damage. This trip function consists of two parts, $\Delta T_{OT}^{setpoint}$ and $f(\Delta I)$. These two terms are combined to give a final trip setting $\Delta T_{OT}^{trip} = \Delta T_{OT}^{setpoint} - f(\Delta I)$.

NSP has evaluated both components of the overtemperature ΔT trip function and found that they remain valid for operation of Prairie Island with an $F\Delta H$ that follows the equation.

$$F\Delta H(P) = 1.60\ [1 + 0.3\ (1-P)] \qquad P \le 1.0$$
$$= 1.60 \qquad p > 1.0$$

where P = fraction of rated power (1650 MWth)

TABLE 3.1

Prairie Island Thermal Hydraulic Reference Conditions

Reactor Conditions	Nominal
Rated Core Power (MWt)	1650 (100%)
Total Reactor Flow Rate (Mlb/hr)	68.62
Active Core Flow Rate (Mlb/hr)	64.50
Core Coolant Inlet Temperature (°F)	530.5
Core Pressure (psia)	2250.0
Power Distribution	
Overall Peaking (F_Q)	2.32
Radial x Local	1.60
Engineering Factor	1.03

TABLE 3.2

Dropped Rod - Auto Control

Transient and Thermal Margin Results

Rod Heat Flux @ time of MDNBR (Btu/hr ft^2)	326,406
$MDNBR_{NB}$	1.662
$(\Delta c/c_o)$ 95/95	0.5493
δ_{BOW}	0.2193
δ_B	0.0216
$MDNBR_B$	1.626
Margin to DNBR Limit (%)	20

4.0 ACCIDENT AND TRANSIENT ANALYSIS

This safety evaluation was performed to help answer the following questions that must be addressed as part of all safety evaluations (Reference 10).

a. Does it create a possibility for an accident or malfunction of a different type than evaluated previously in the USAR or subsequent commitments?

b. Does it increase the probability of occurence of an accident or malfunction of equipment important to safety previously analyzed in the USAR or subsequent commitments?

c. Does it increase the consequences of any accident or malfunction of equipment important to safety previously analyzed in the USAR or subsequent commitments?

d. Is the margin of safety defined in the bases for any Technical Specification reduced?

As part of all safety evaluations, NAD evaluates all accidents in the USAR to determine if the current anlayses bound the design. All accidents that are not bounded by previous analysis are reanalyzed and the results are presented in the RSE.

4.1 Plant Transient Analysis

This documents the NSP Nuclear Analysis Department's analysis of plant operational transients for Prairie Island.

This analysis includes the following:

ITC

The current Tech Specs require that the Isothermal Temperature Coefficient (ITC) be negative (except during low power physics tests) at all times. The new Tech Specs are designed to allow a +5.0 pcm/°F ITC up to 70% power, and a zero or negative ITC above 70%. All transients have been analyzed using a +5.0 pcm/°F ITC at full power in order to bound operation with the new limits.

Thimble Plug Removal

NSP has recently received approval to remove the thimble plug assemblies from the assemblies that have neither control rods or source assemblies (Reference 11). Our analysis included the effect of thimble plug removal and the associated increase in core bypass flow.

Departure from Nucleate Boiling Limits

Previous analysis of thermal margin to MDNBR limits was performed using the W-3 CHF correlation in the COBRA-IIIC/MIT computer program. NSP has since replaced the COBRA-IIIC/MIT program with the VIPRE-01 program. The ENC fuel will continue to be evaluated using the W-3 correlation. The OFA fuel uses the new WRB-1 CHF correlation. This correlation has a lower design limit of 1.17 (versus 1.3 for the W-3 correlation). Both types of fuel are analyzed separately for DNB margin and the fuel type showing the least margin is included in the graphs for each transient.

FQ, FΔH

Small and large break LOCA analysis are being performed by Westinghouse for PI 1 Cycle 11. This analysis will include both Westinghouse OFA and ENC TOPROD fuel. The NSP transient analysis used a FQ of 2.32 and an FΔH of 1.60. This will bound the LOCA analysis values of 2.30 and 1.60. In addition, the FΔH limit equation at reduced power was changed as shown in section 3.6.

The analysis shows that the calculated transient, the dropped rod-auto control, is well above the acceptable MDNBR of 1.3 for ENC fuel and 1.17 for Westinghouse fuel.

In addition to the dropped rod described above, the pump seizure, a Class IV event, showed a calculted MDNBR of < 1.3. The number of fuel rods which would potentially experience DNB in the transient is calculated to be less than 8%. For all transients, the maximum pressurizer pressure was less than 2750 psia. The latter pressure corresponds to 110% of the design pressure of 2500 psia.

4.1.1 Input Parameters

The steam line breaks are initiated from hot shutdown conditions. All other transients are initiated from 102% of full power conditions. For full power operation, an axial peaking factor, F_Z, of 1.379 located at X/L = 0.6, and a total peaking, F_Q, of 2.32 is assumed. Other thermal hydraulic parameters for full power operation are summarized in Table 4.2.

Reactor trip setpoints for Prairie Island Units 1 and 2, along with setpoints and delay times used in the analysis, are given in Table 4.3. The setpoints used in the analysis are essentially the same as those in the FSAR analysis. In all cases, the setpoints used in the analysis bound the actual setpoints for the Prairie Island plants to account for instrumentation errors and uncertainties.

4.1.2 Transient Analysis Results

4.1.2.1 Fast Control Rod Withdrawal

This transient assesses plant response to a control rod withdrawal, with a reactivity insertion rate of 8.2 E-4 ΔK/sec, from full power. All automatic reactor control systems are assumed inoperable.

The transient response of the NSSS for this case is shown in Figures 4.1 through 4.6. The reactor trip is generated on high neutron power (setpoint at 118%) at .94 seconds. The pressurizer pressure rises to 2308 psia at 4.9 seconds. The vessel average temperature rises by less than 1.5 °F at 4 seconds and then drops off. The DNB ratio drops from its initial value of 2.157 to a minimum of 2.046 at 2.0 seconds after the start of the transient.

The acceptance criteria for this transient are that the minimum DNBR be not less than 1.3 and that the maximum reactor coolant and main steam system pressure not exceed 110% of their design values. This transient meets all acceptance criteria.

4.1.2.2 Slow Control Rod Withdrawal

This transient assesses plant response to a control rod withdrawal with a reactivity insertion rate of 2.52 E-5 ΔK/sec during full power operation. The reactivity insertion rate was selected to minimize DNBR during the transient. All automatic reactor control systems are assumed inoperable.

The transient response of the NSSS for this case is shown in Figures 4.7 through 4.12. The reactor trip is generated on overpower ΔT at 57.1 seconds. The pressurizer pressure rises to 2481 psia at 60.2 seconds. The vessel average temperature rises by less than 6 °F and then drops off. The DNB ratio drops from its initial value of 2.157 to a minimum of 1.794 at 57.2 seconds after the start of the transient.

The acceptance criteria are that the minimum DNBR be not less than 1.3 and that the maximum reactor coolant and main steam system pressure not exceed 110% of their design values. This transient meets all acceptance criteria.

4.1.2.3 Loss of External Electric Load

This transient considers plant response from full power when a loss of load results in a turbine trip. Simultaneous reactor trip initiated by the turbine stop valves is conservatively neglected. Rather the reactor is scrammed later in the transient by the pressurizer overpressure trip signal. All automatic reactor control systems, as well as the steam generator relief valves, are assumed inoperable. Steam dump and bypass are also neglected.

The transient response of the NSSS for this case is shown in Figures 4.13 through 4.18. At the start of the transient, the turbine stop valves close and the secondary side pressure rises rapidly to the safety valve setpoint at 14 seconds and is limited to that pressure by relief through the safety valves. The primary system pressurizes rapidly due to the loss of heat sink and the reactor is scrammed at 5.4 seconds on a high pressurizer pressure trip signal. Pressurizer safety valve opening occurs and a peak pressure of 2501 psia is calculated at 6.6 seconds. Core power remains relatively constant up until the time of reactor trip. Because of the primary system pressurization, the DNB ratio increases and remains above its initial 2.157 value.

The acceptance criteria for this transient are that the minimum DNBR be not less than 1.3 and that the maximum reactor coolant and main steam system pressure not exceed 110% of their design values. This transient meets all acceptance criteria.

4.1.2.4 Dropped Rod - Auto Control

In this transient a full length RCCA is assumed to be released by the stationary gripper coils and to fall into a fully inserted position in the core. A dropped RCCA typically results in a reactor trip signal due to the power range negative neutron flux rate circuitry. The core power distribution, from an absolute value point of view, is not adversely effected during the short interval prior to reactor trip. The drop of a single RCCA may or may not result in a negative flux rate reactor trip. If a trip does not occur, a single failure of the controller circuitry can cause a transient power overshoot. The power overshoot combined with the higher peaking factors associated with a dropped rod could conceivably challenge the 1.3 MDNBR limit. The transient response of the NSSS for this case is shown in Figures 4.19 - 4.24.

The MDNBR drops from its initial value of 1.689 to 1.675 at 35.0 seconds. After a slight rise to 1.686 at 70.0 seconds, the MDNBR begins to drop and reaches a value of 1.663 at 10 minutes. The analysis is terminated at 10 minutes because of:

a. The slow decrease in MDNBR from 70 seconds on is due to the analysis using a HFP ITC of +5.0 pcm/°F. This causes the power to continue to rise after the initial rod pull. This would not occur if the ITC was at 0.0 pcm/°F (maximum value allowed at HFP).

b. It is assumed in the analysis that the operators would notice the dropped rod and correct the situation within 10 minutes.

A maximum of pressurizer pressure of 2232 psia occurs at 101 seconds. The acceptance criteria for this transient are that the minimum DNBR be not less than 1.3 and that the maximum reactor coolant and main steam pressure not exceed 110% of their design values. This transient meets all acceptance criteria.

4.1.2.5 Loss of Reactor Coolant Flow

This transient considers the loss of reactor coolant flow associated with the simultaneous coastdown of both primary system coolant pumps. Following the loss of two pumps at power, a reactor trip is actuated by either low voltage or open pump circuit breakers since the incident is due to the simultaneous loss of power for all pump buses. Both the low voltage and pump breaker reactor trip circuitry meet the single failure criteria and therefore cannot be negated by a single failure. The time from the loss of power to all pumps to the initiation of control rod assembly motion to shutdown reactor is taken as 2.1 seconds. This is a conservative assessment of the delay. All automatic reactor control systems are assumed inoperable.

The transient response of the NSSS for this case is shown in Figures 4.25 through 4.30. The MDNBR drops from its initial value of 2.157 to a minimum of 1.865 at 3.5 seconds into the transient due to an increase in the power to flow ratio. A maximum pressurizer pressure of 2372 psia is calculated to occur at 6.5 seconds into the transient.

The acceptance criteria for this transient are that the minimum DNBR be not less than 1.3 and that the maximum reactor coolant and main steam system pressure not exceed 110% of their design values. This transient meets all acceptance criteria.

4.1.2.6 Locked Pump Rotor

The locked pump rotor transient is a Class IV event that considers plant response from full power operation when one of the two primary coolant pumps is postulated to abruptly seize. Reactor scram and trip of the feedwater pumps due to low primary coolant flow is conservatively assumed to occur at 0.9 second after pump seizure. All automatic reactor control systems are assumed inoperable.

The plant response for this transient is shown in Figures 4.31 to 4.36. The calculated MDNBR drops below 1.3 at approximately 1.1 seconds after pump seizure. The duration of time for which the DNBR is less than 1.3 is less than 4.5 seconds. The number of fuel rods statistically calculated to experience DNB for this Class IV transient is less than 8%. A maximum pressurizer pressure of 2502 psia is calculated to occur at 3.25 seconds into the transient.

The acceptance criteria for the locked rotor analysis are as follows:

1. The maximum reactor coolant and main steam system pressures must not exceed 110% of the design values.

2. The maximum clad temperature calculated to occur at the core hot spot must not exceed 2700 °F.

This transient meets all acceptance criteria.

4.1.2.7 Large Steam Line Break

This transient considers plant response, with a withdrawn (stuck) control rod cluster from hot shutdown (full flow), due to depressurization of the secondary system such as might occur for a large steam line break. Hot shutdown is considered since the steam generator secondary side water inventory is at a maximum which maximizes the duration and magnitude of the primary loop cooldown. This cooldown in conjunction with a negative moderator temperature coefficient at end-of-cycle conditions maximizes the severity of the transient.

The transient response of the NSSS for this case is shown in Figures 4.37 through 4.41. The secondary depressurization for the large steamline break transient is calculated using the Moody curve for fL/D = 0 for a break at the S.G. exit. Cooldown and depressurization of the primary system is fairly rapid.

The signal for high pressure safety injection is conservatively assumed to occur at 6.0 seconds as a result of high containment pressure. Borated HPSI flow does not actually commence however until after a 10 second delay or until 16 seconds into the transient. In the interim, the cooldown, in conjunction with the large negative moderator temperature coefficient, is sufficient to overcome the assumed minimum shutdown margin and begin a return to power after about 23 seconds into the transient.

Minimum capability for injection of high concentration boric acid solution corresponding to the most restrictive single failure in the Safety Injection System is assumed. This corresponds to the flow delivered by one high head safety injection pump. Low concentration boric acid (1950 ppm) must be swept from the safety injection lines downstream of the boric acid tank isolation valves prior to the delivery of high concentration boric acid (20,000 ppm) to the main coolant loops. This effect has been allowed for in the analysis.

Core heat flux reaches a maximum value of 15.5% at about 39 seconds after the start of the transient after which point boron reaching the core begins to reduce power. The MDNBR at approximately the time of peak heat flux was calculated to be 3.7 using the W-3 correlation. In calculating this MDNBR a conservative peaking factor of 9.4 was used in the calculations.

The containment pressure response has not been reevaluated for Prairie Island 1 Cycle 11. The containment pressure response is a system dependent parameter and is relatively independent of fuel type. The containment pressure response is therefore bounded by the analysis in the FSAR.

The acceptance criteria for the large steam line break are as follows:

1. The maximum reactor coolant and main steam system pressures must not exceed 110% of the design values.

2. The maximum containment pressure must not exceed the Technical Specification limit of 46 psig.

This transient meets all acceptance criteria.

4.1.2.8 Small Steam Line Break

The small steam line break transient is an accident similar to the large steam line break except that the initial break flow is only 25% of normal rated steam flow versus the 620% in the large steam line break transient. The small steam line break transient is intended to envelope a steam generator relief valve failure.

Figures 4.42 through 4.45 show plant response during this transient. Primary system cooldown is less rapid in this transient then in the large steam line break transient. Depressurization of the pressurizer is correspondingly less rapid so that in this transient the signal for HPSI due to low pressurizer pressure is not reached until 112 seconds after the start of the transient. In this small steam line break transient primary system cooldown is not sufficient to overcome the assumed minimum shutdown margin prior to boration from the HPSI and hence the reactor does not return to power.

The acceptance criteria for this transient are identical to those of the large steam line break. This transient meets all acceptance criteria.

4.2 LOCA-ECCS Analysis

Large and small LOCA-ECCS analysis for Prairie Island Unit 1 Cycle 11 have been performed by Westinghouse and transmitted under separate cover.

4.3 Rod Ejection Analysis

A Control Rod Ejection Accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of a Rod Cluster Control Assembly (RCCA) and drive shaft. The consequence of this mechanical failure is a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

The rod ejection accident has been evaluated with the procedures developed in Reference 1. The ejected rod worths, hot pellet peaking factors, delayed neutron fractions and Doppler coefficients were taken as conservative values which bound the Prairie Island analysis.

The pellet energy deposition resulting from an ejected rod was evaluated explicitly at HFP and HZP initial conditions. The HFP pellet energy deposition was calculated to be 136 cal/gm. The HZP pellet energy deposition was calculated to be 147 cal/gm. These values bound both ENC and Westinghouse fuel. The rod ejection accident was found to result in energy deposition of less than the 280 cal/gm limit as stated in Regulatory Guide 1.77. The significant parameters for the analysis, along with the results are summarized in Table 4.4.

TABLE 4.1

Summary of Prairie Island Transient Margins

(Calculated Value/Acceptance Criteria)

Transient	MDNBR	Pressure (psia) RCS	MSL	# Failed Pins (%)	Clad Temp. (F)	Fuel Enthalpy (cal/gm)
Rod Withdrawal at Power						
Fast	2.046/1.3	2308/2750	885/1210	-	-	-
Slow	1.794/1.3	2481/2750	1048/1210	-	-	-
Turbine Trip	2.157/1.3	2501/2750	1109/1210			
2/2 Pump Trip	1.865/1.3	2372/2750	1069/1210	-	-	-
Locked Rotor	-	2502/2750	1099/1210	8	NC/2700	-
Dropped Rod	1.662/1.3	2232/2750	757/1210	-	-	-
MSL Break *	-	2250/2750	1046/1210	0	NC/2700	-
Ejected Rod-HZP	-	2500/3000	-	NC	761/2700	147/280
Ejected Rod-HFP	-	2380/3000	-	NC	739/2700	136/280

NC - Not calculated for each cycle

* - Also a Tech. Spec. limit of peak containment pressure 46 psig, which is not calculated.

TABLE 4.2

Parameter Values Used in Full Power 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*
Hot Channel Factors	
Total Peaking Factor, F_Q^T	2.32
Enthalpy Rise Factor, $F_{\Delta H}^N$	1.60
Total Coolant Flow, lb/hr	68.62×10^6
Effective Core Flow, lb/hr	64.50×10^6
Reactor Inlet Temperature, °F	534.5
Steam Generators	
Calculated Total Steam Flow, lb/hr	7.23×10^6
Steam Temperature, °F	510.8
Feedwater Temperature, °F	427.3
Tubes Plugged, %	5.0**

* Locked Rotor is initiated from 2280 psia

** MSL break conservatively assumes no plugging

TABLE 4.3

Prairie Island Units 1 and 2 Trip Setpoints

	Setpoint	Used in Analysis	Delay Time
High Neutron Flux	108%	118%	0.5 sec
Low Reactor Coolant Flow	93%	87%	0.6 sec
High Pressurizer Pressure	2388 psia	2425 psia	1.0 sec
Low Pressurizer Pressure	1915 psia	1700 psia	1.0 sec
High Pressurizer Water Level	85% of Span	100% of Span	1.5 sec
Low-Low Steam Generator Water Level	13% of Span	0% of Span	1.0 sec
Overtemperature ΔT*	TAVEo = 567.3F	TAVEo = 567.3F	6.0 sec
	Po = 2250 psia	Po = 2250 psia	
Overpower ΔT**	TAVEo = 567.3	TAVEo = 567.3	6.0 sec
High Pressure Safety Injection	1842 psia	1800 psia	10 sec

* The overtemperature ΔT trip is a function of pressurizer pressure, coolant average temperature, and axial offset. The TAVEo and Po setpoints are contained within the functional relationship.

** The overpower ΔT trip is a function of coolant average temperature and axial offset. The TAVEo setpoint is contained within the functional relationship.

TABLE 4.4

Prairie Island Ejected Rod Analysis

	HZP	HFP
Maximum Control Rod Worth (pcm)	607	167
Doppler Defect (pcm)	1243	1243
Shutdown Margin (pcm)	1631	1631
Delayed Neutron Fraction	0.004975	0.004884
Power Peaking Factor, F_Q	9.252	3.576
Energy Deposition (cal/gm)	147	136

Prairie Island
DYNODE-P Fast Rod Withdrawal

Figure 4.1
K Effective

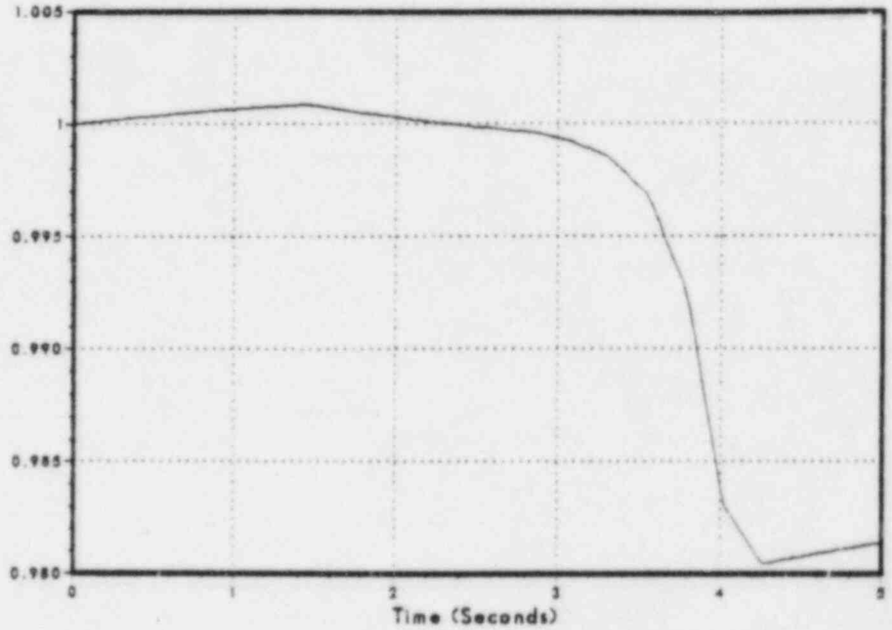


Figure 4.2
Absolute Power

NSP

Prepared by NSP Safety Analysis, 9/28/85

Prairie Island
DYNODE-P Fast Rod Withdrawal

Figure 4.3
Core Average Heat Flux

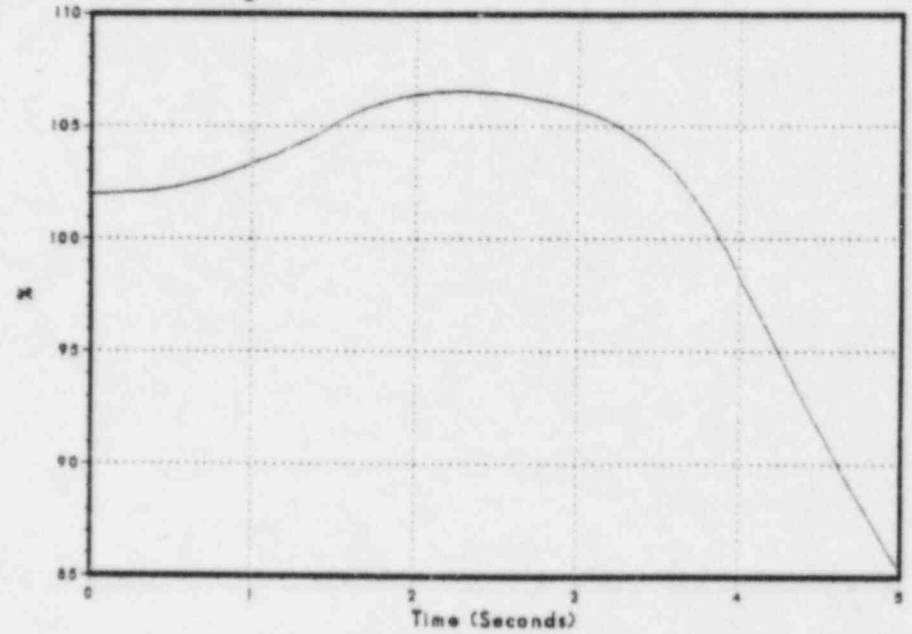
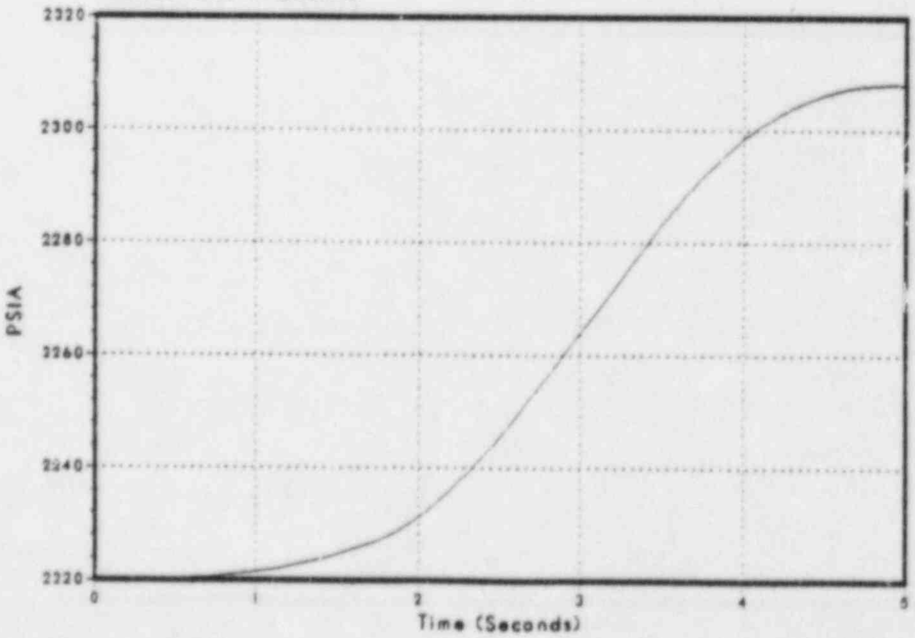


Figure 4.4
Pressurizer Pressure



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Prairie Island
DYNODE-P Fast Rod Withdrawal

Figure 4.5
Vessel Average Temperature

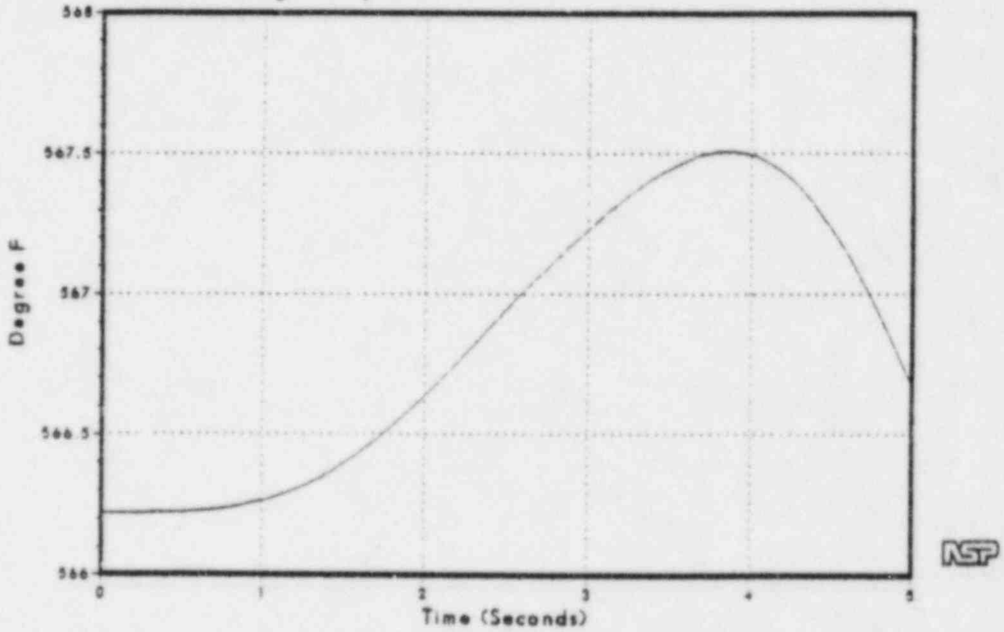
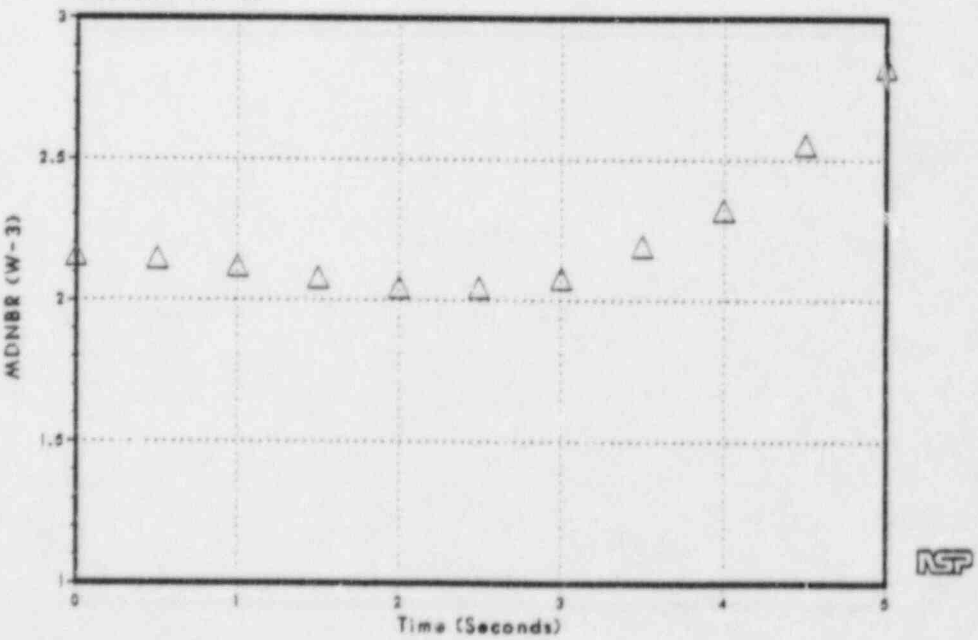


Figure 4.6
MDNBR (W-3)



Prepared by NSP Safety Analysis, 9/28/85

Prairie Island
DYNODE-P Slow Rod Withdrawal

Figure 4.7
K Effective

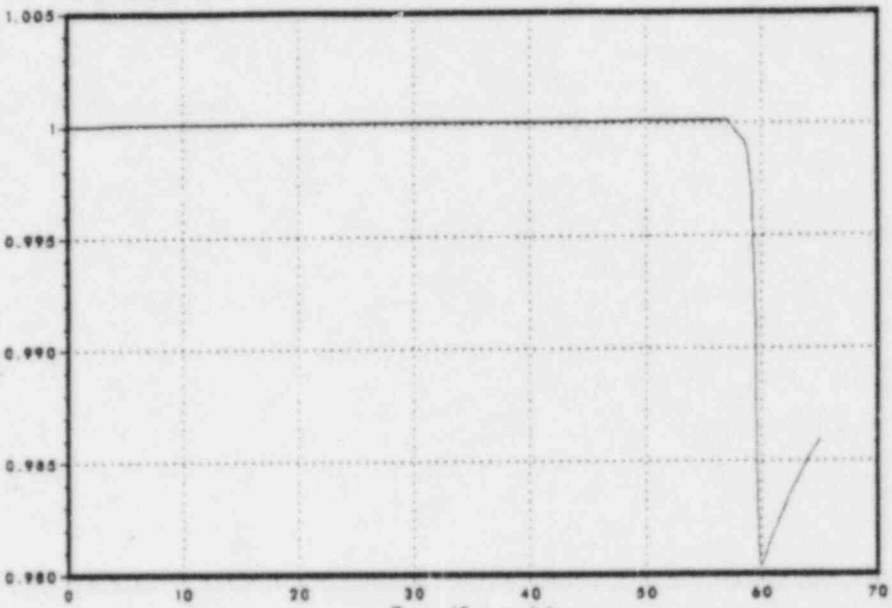
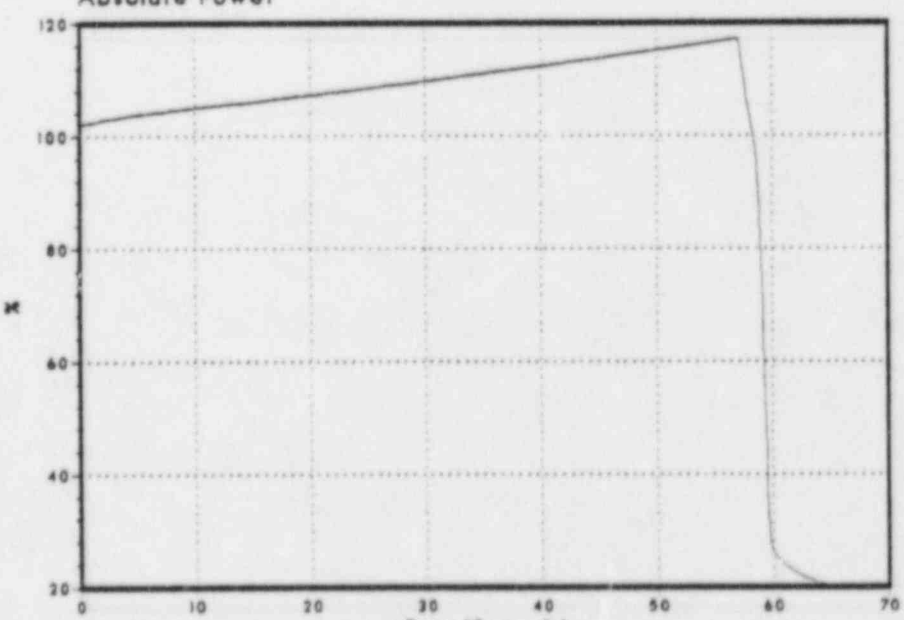


Figure 4.8
Absolute Power



NSP

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Prairie Island
DYNODE-P Slow Rod Withdrawal

Figure 4.9
Core Average Heat Flux

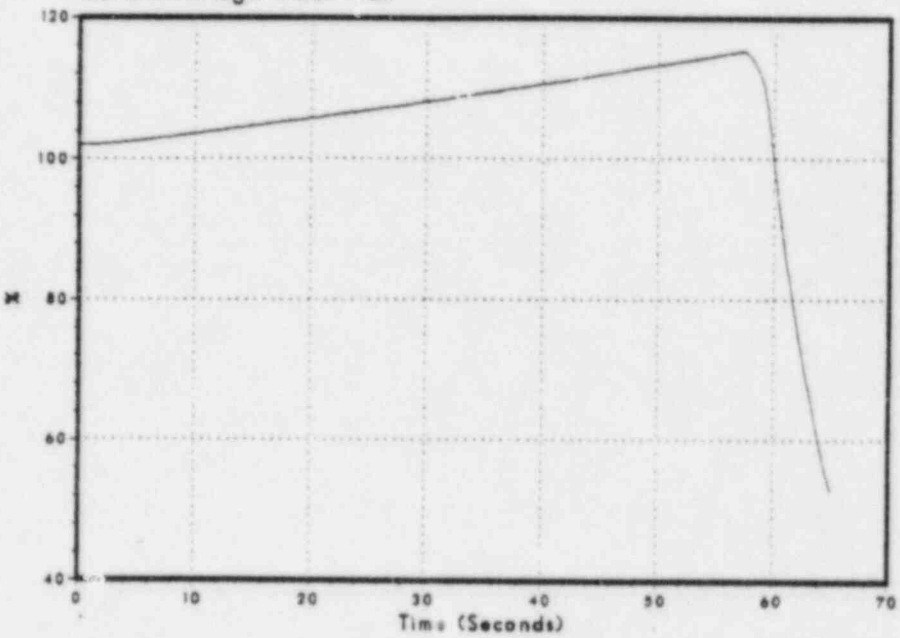
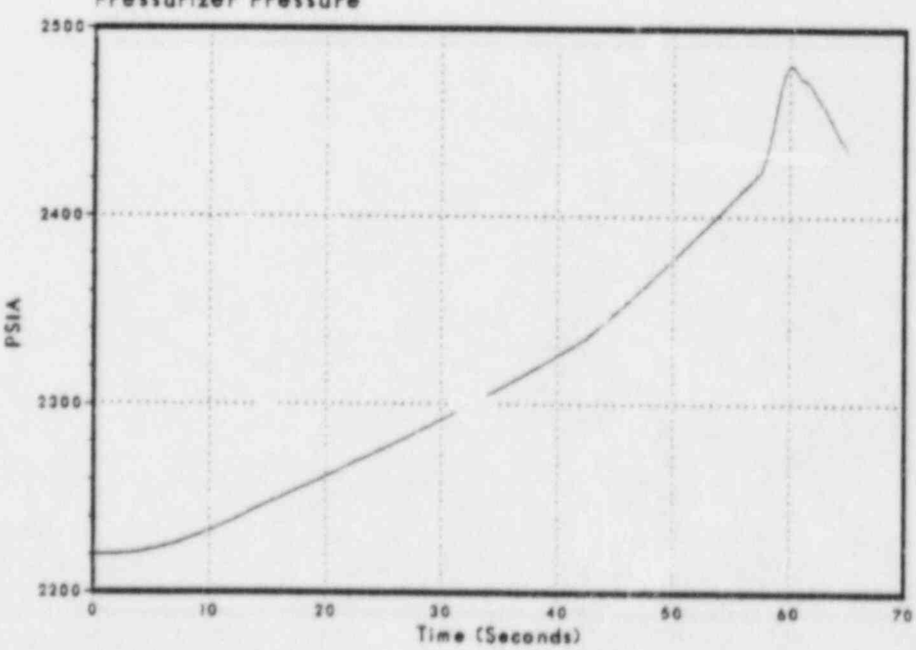


Figure 4.10
Pressurizer Pressure



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Prairie Island
DYNODE-P Slow Rod Withdrawal

Figure 4.11
Core Inlet Temperature

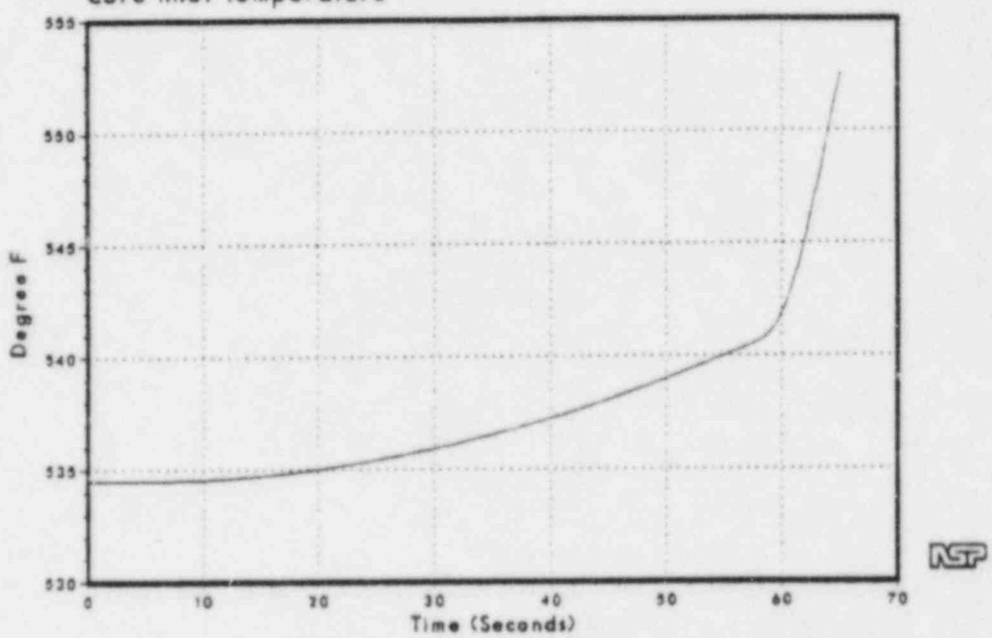
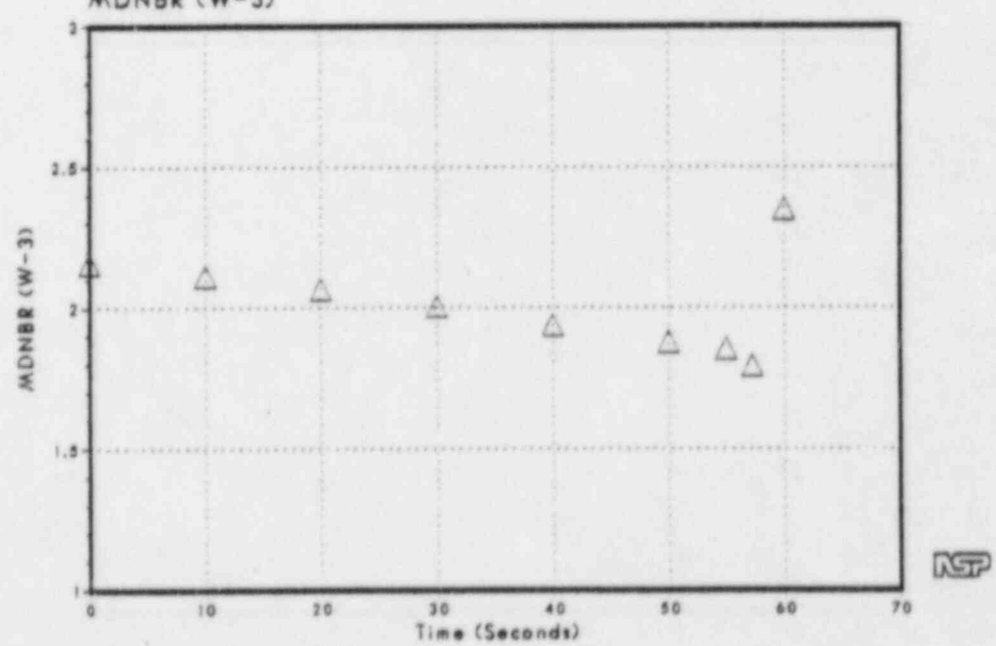


Figure 4.12
MDNBR (W-3)



Prepared by NSP Safety Analysis, 9/28/85

Prairie Island
DYNODE-P Turbine Trip

Figure 4.13
K Effective

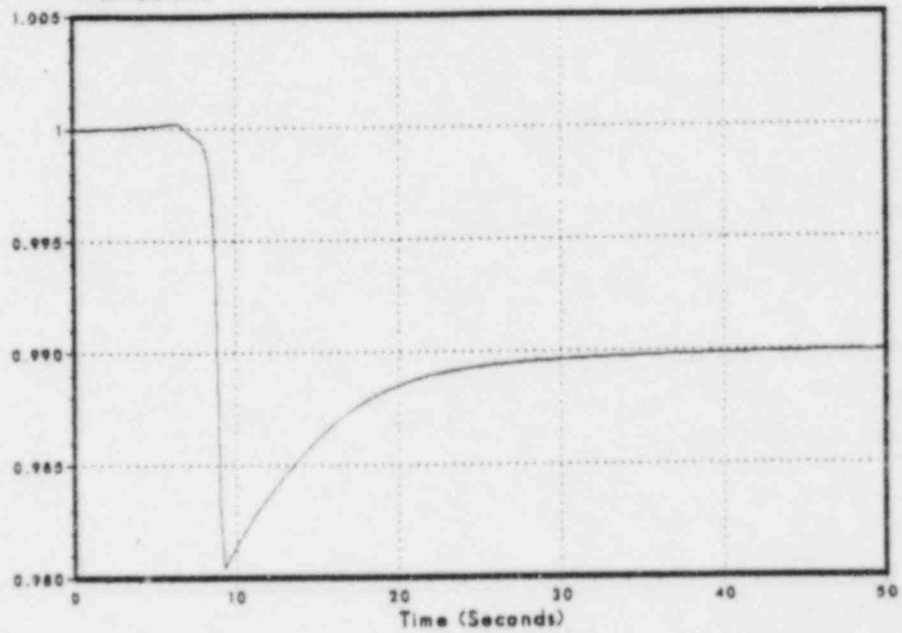
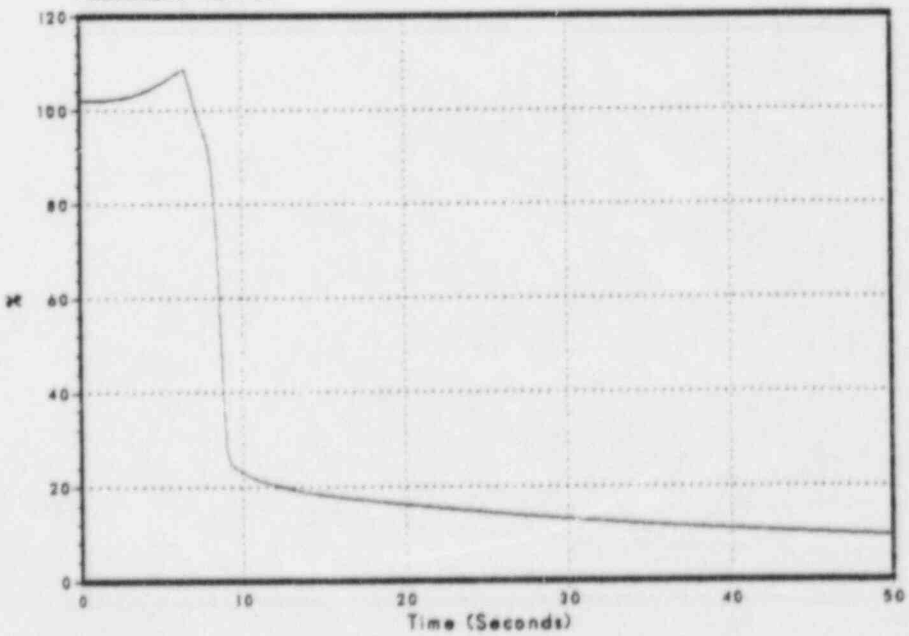


Figure 4.14
Absolute Power



NSP

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Prairie Island
DYNODE-P Turbine Trip

Figure 4.15
Core Average Heat Flux

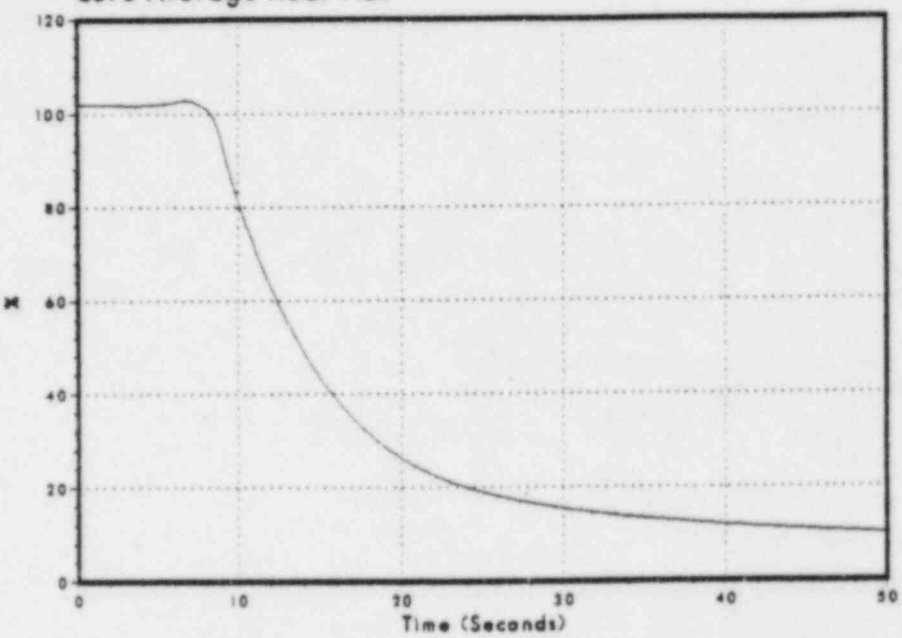
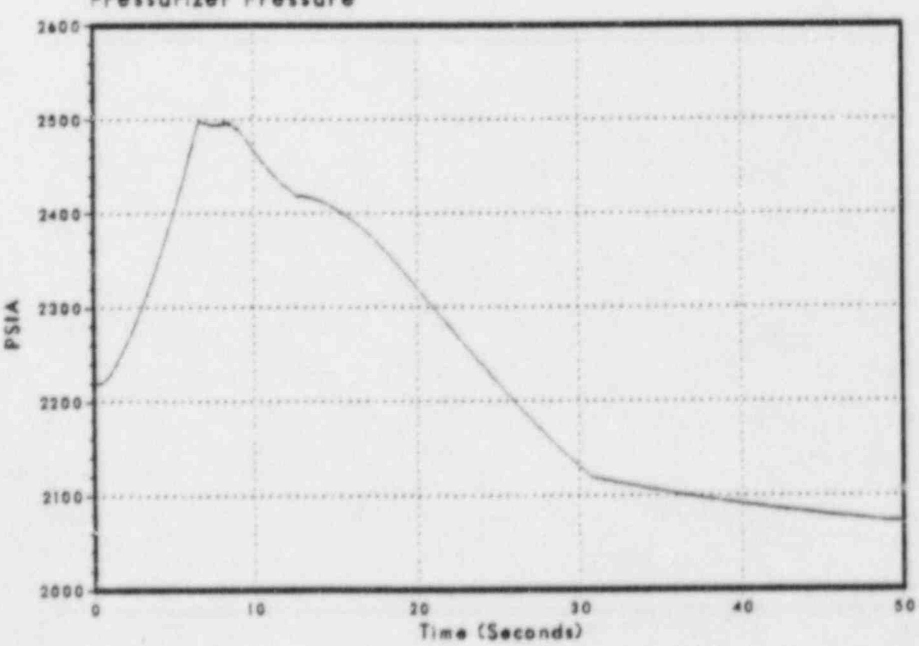


Figure 4.16
Pressurizer Pressure



NSP

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Prairie Island
DYNODE-P Turbine Trip

Figure 4.17
Core Inlet Temperature

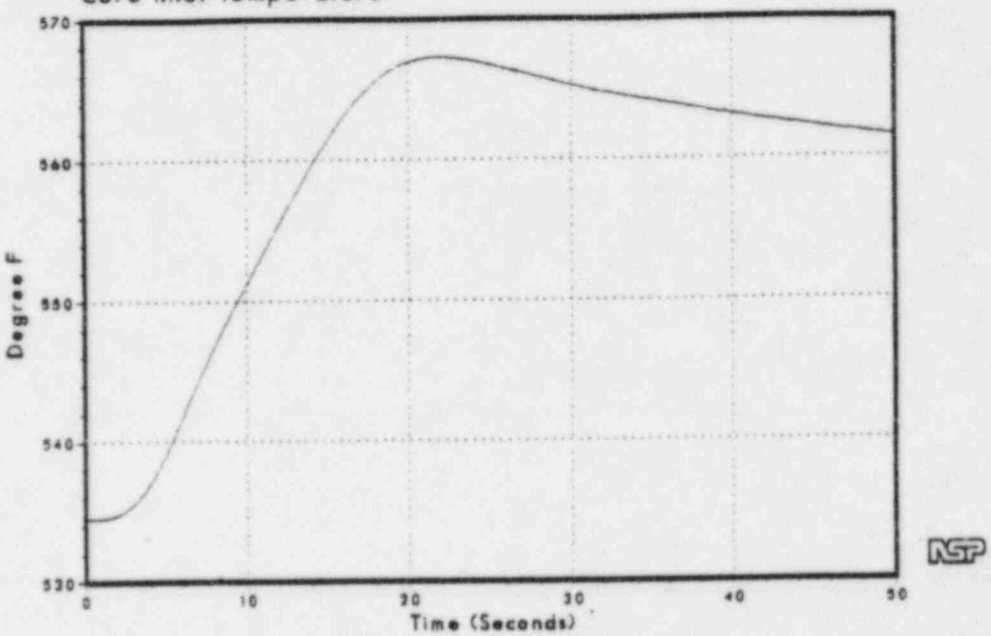
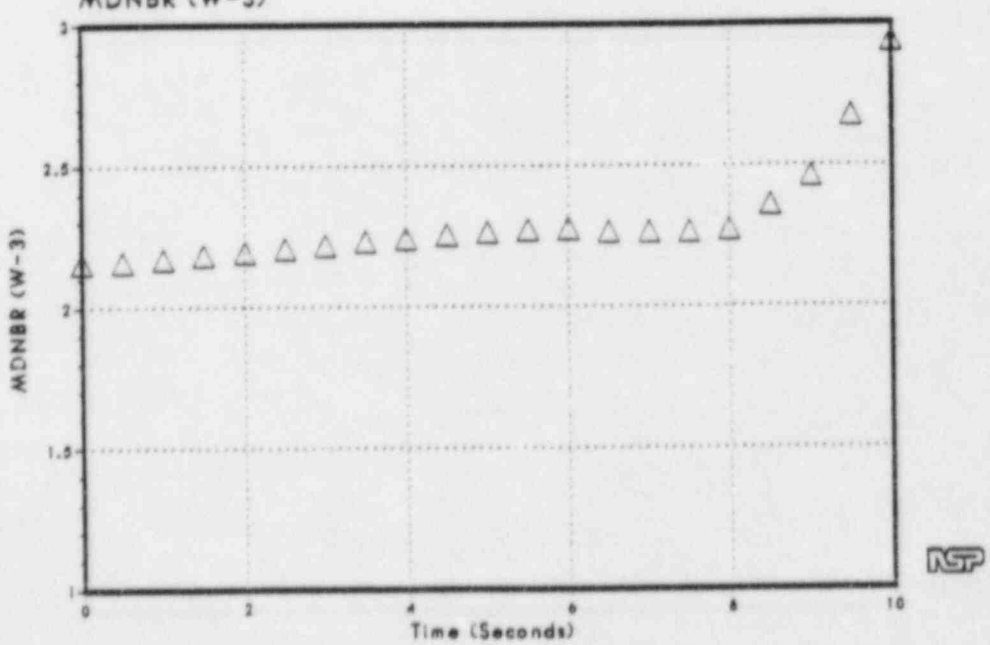


Figure 4.18
MDNBR (W-3)



Prepared by NSP Safety Analysis. 9/25/85

Prairie Island
DYNODE-P Dropped Rod, EOC

Figure 4.19
K Effective

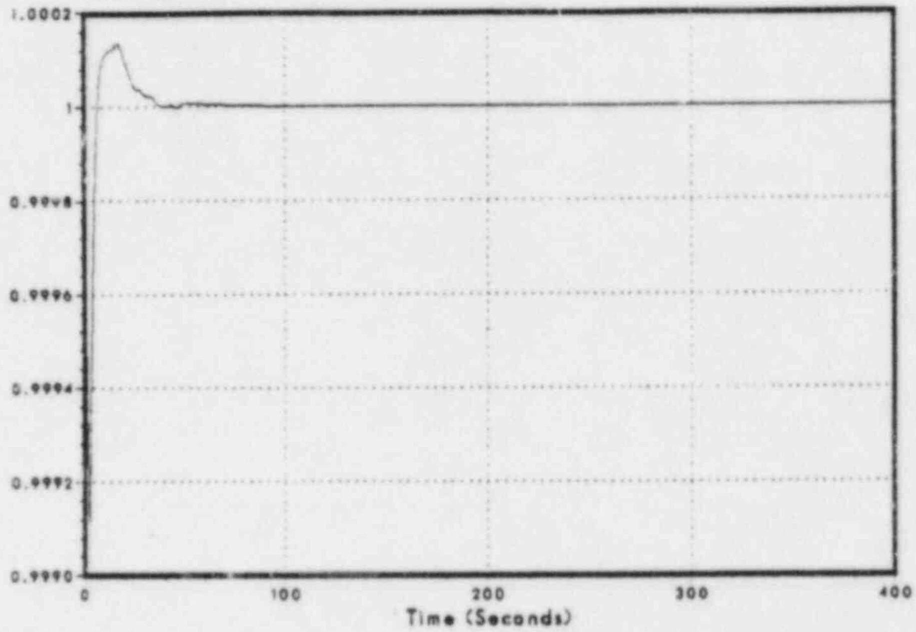
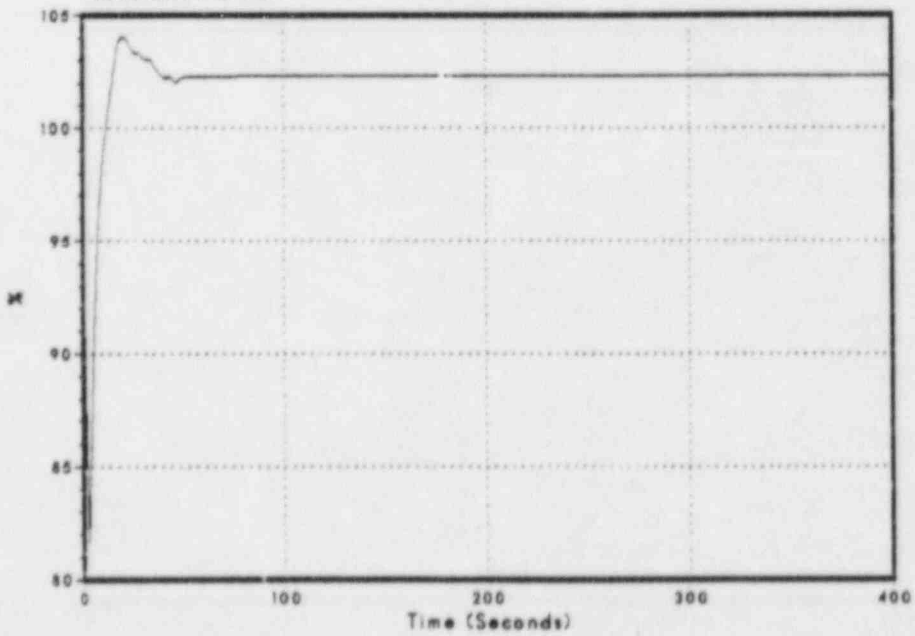


Figure 4.20
Absolute Power



NSP

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Prairie Island
DYNODE-P Dropped Rod, EOC

Figure 4.21
Core Average Heat Flux

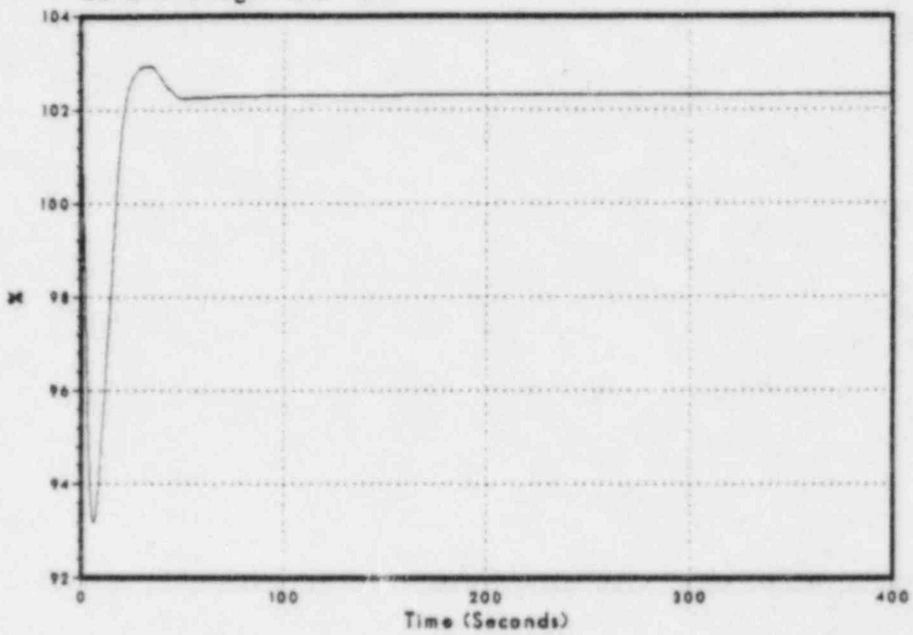
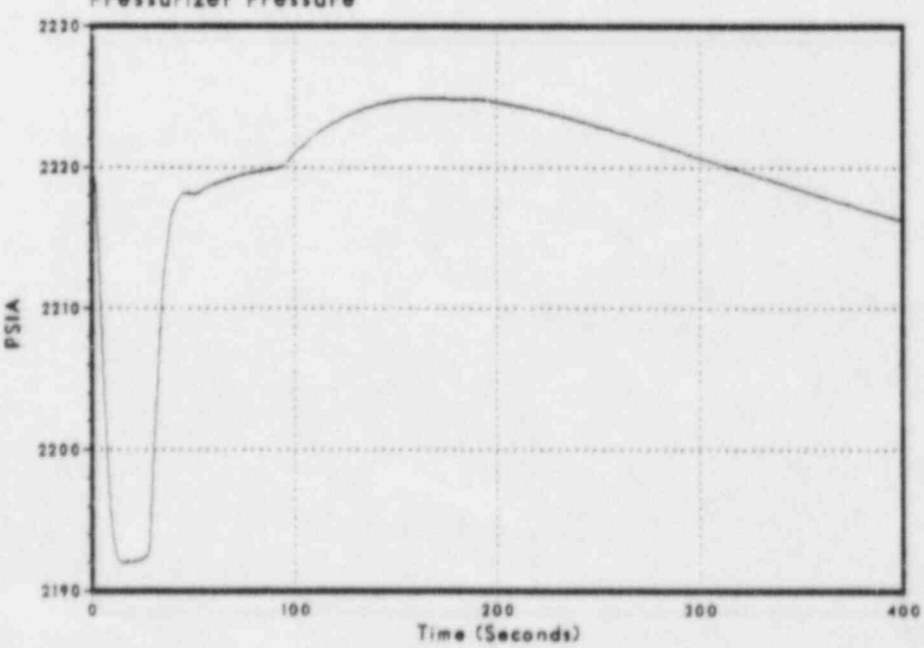


Figure 4.22
Pressurizer Pressure



NSP

Prepared by NSP Safety Analysis, 11/18/85

Prairie Island
DYNODE-P Dropped Rod. EOC

Figure 4.23
Vessel Average Temperature

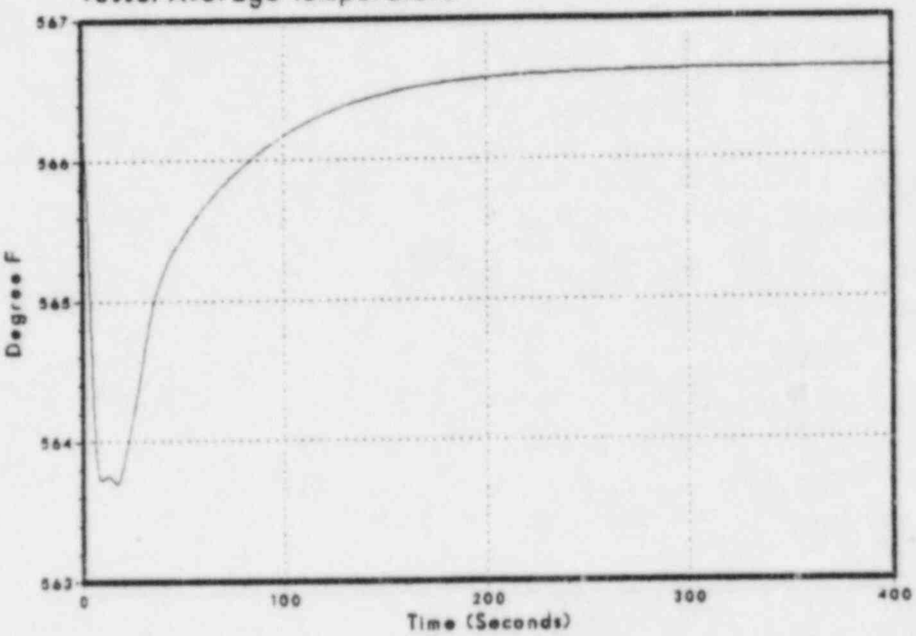
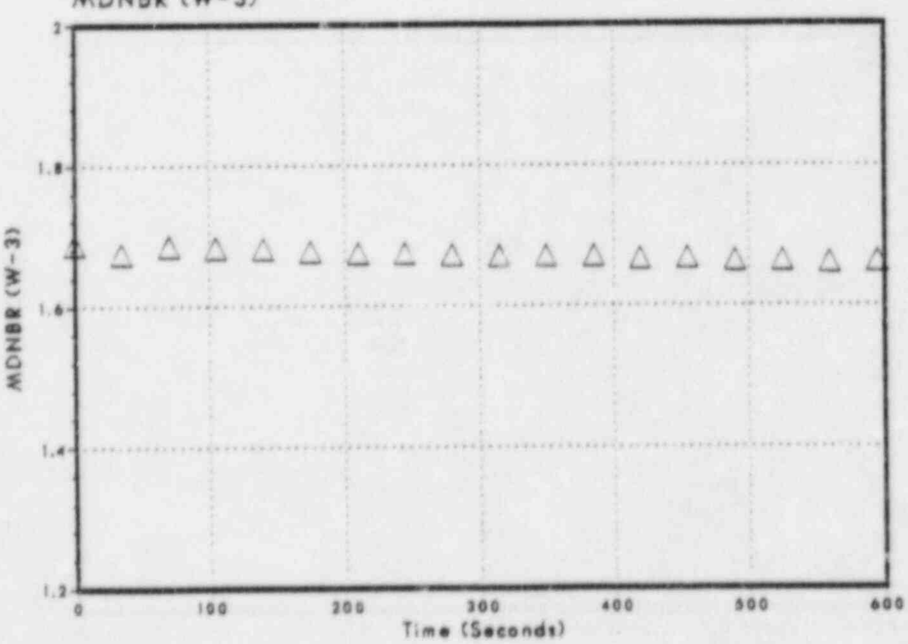


Figure 4.24
MDNBR (W-3)



Prepared by NSP Safety Analysis, [illegible]

Prairie Island
DYNODE-P 2/2 Pump Trip

Figure 4.25
K Effective

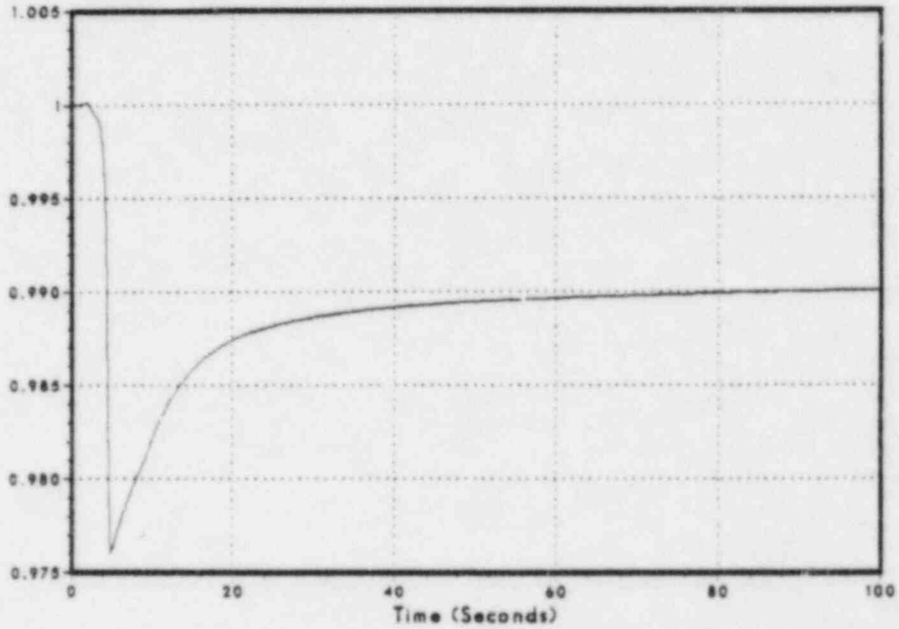
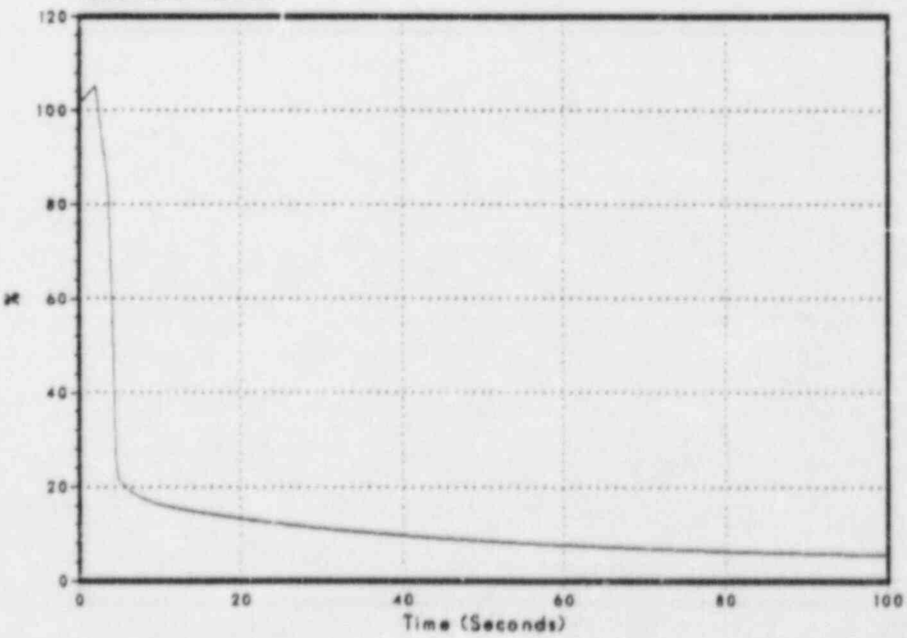


Figure 4.26
Absolute Power



NSP

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Prairie Island
DYNODE-P 2/2 Pump Trip

Figure 4.27
Core Average Heat Flux

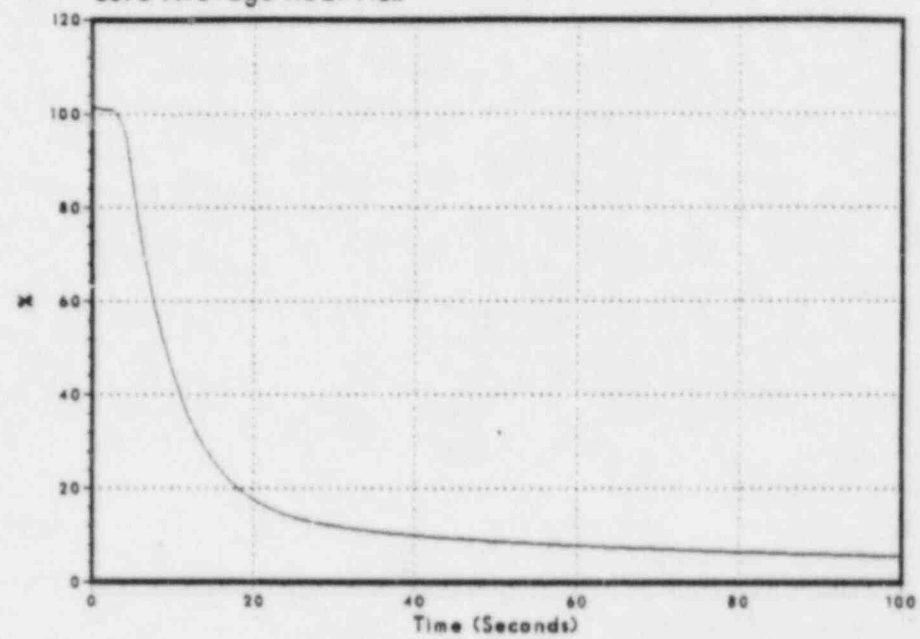
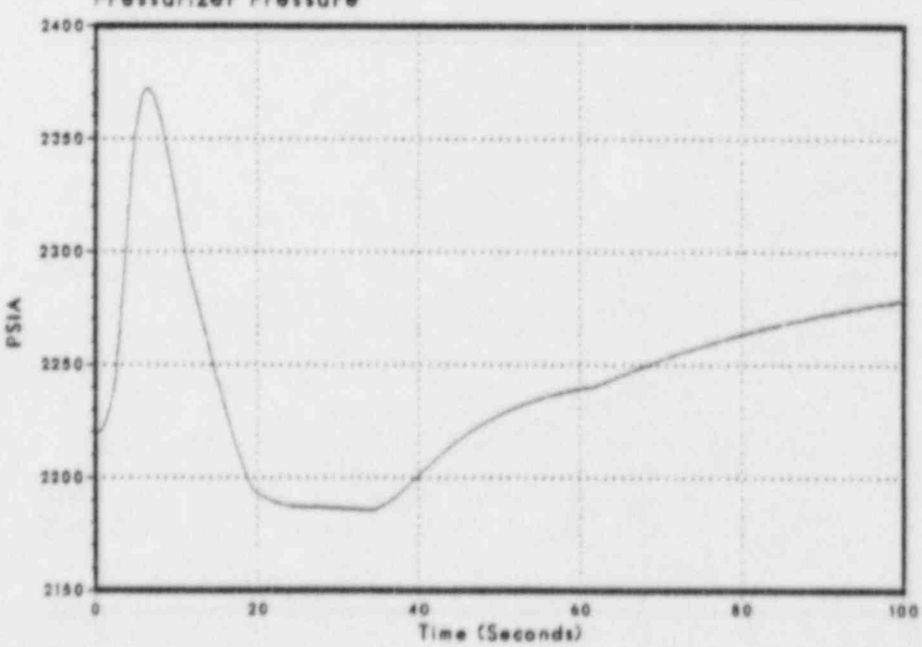


Figure 4.28
Pressurizer Pressure



NSP

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Prairie Island
DYNODE-P 2/2 Pump Trip

Figure 4.29
Core Flow

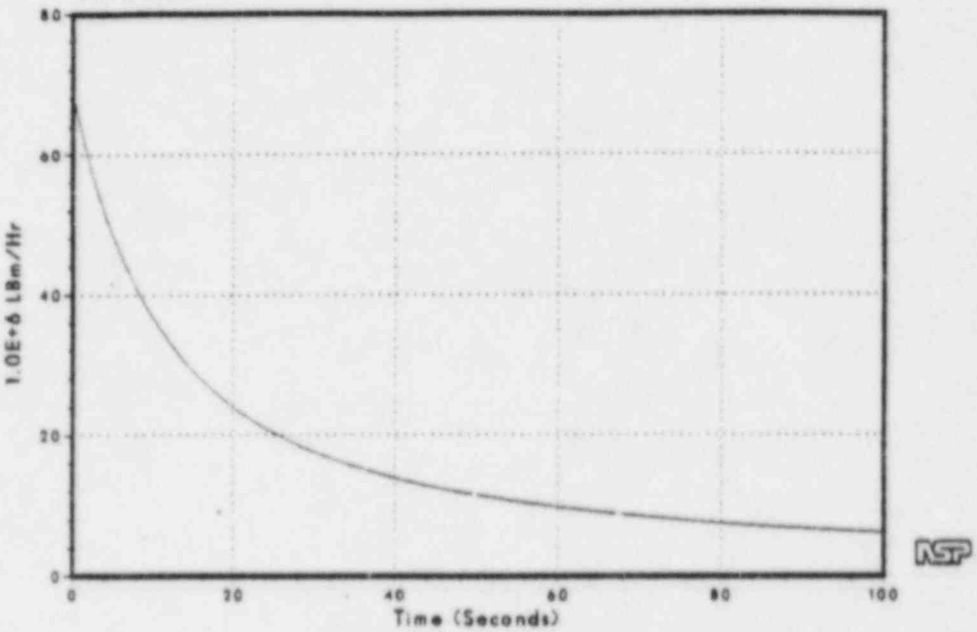
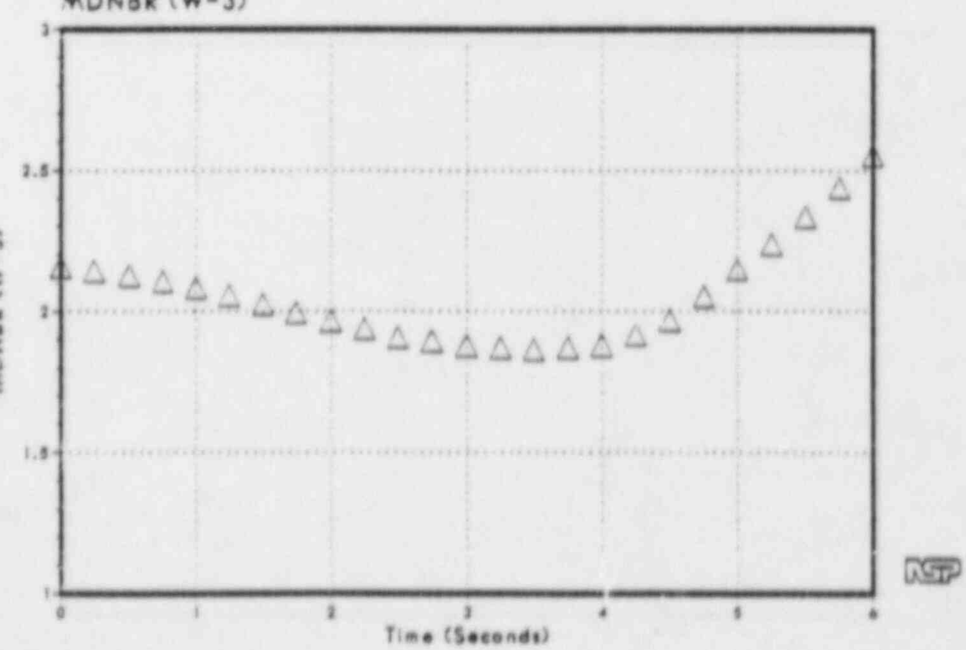


Figure 4.30
MDNBR (W-3)



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Prairie Island
DYNODE-P Locked Rotor

Figure 4.31
K Effective

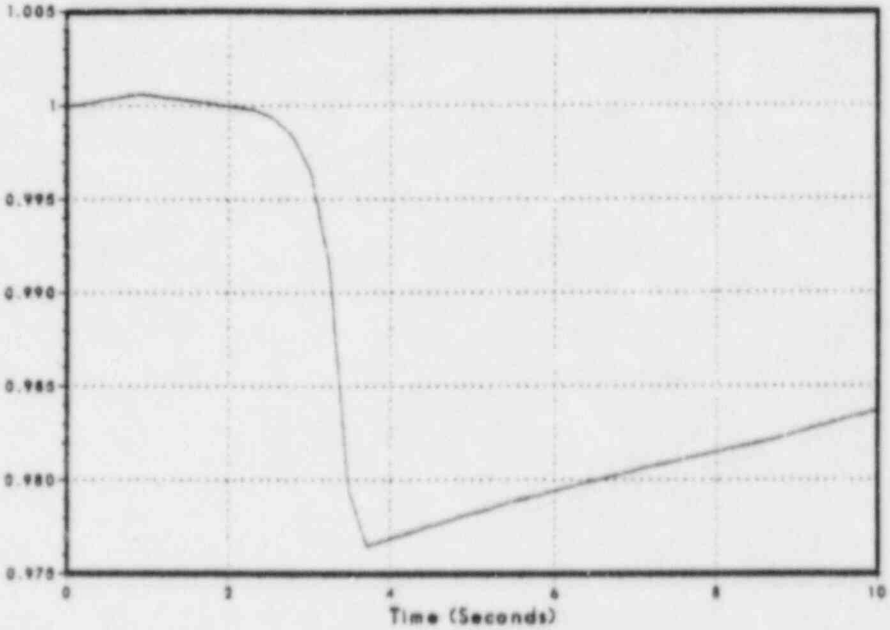
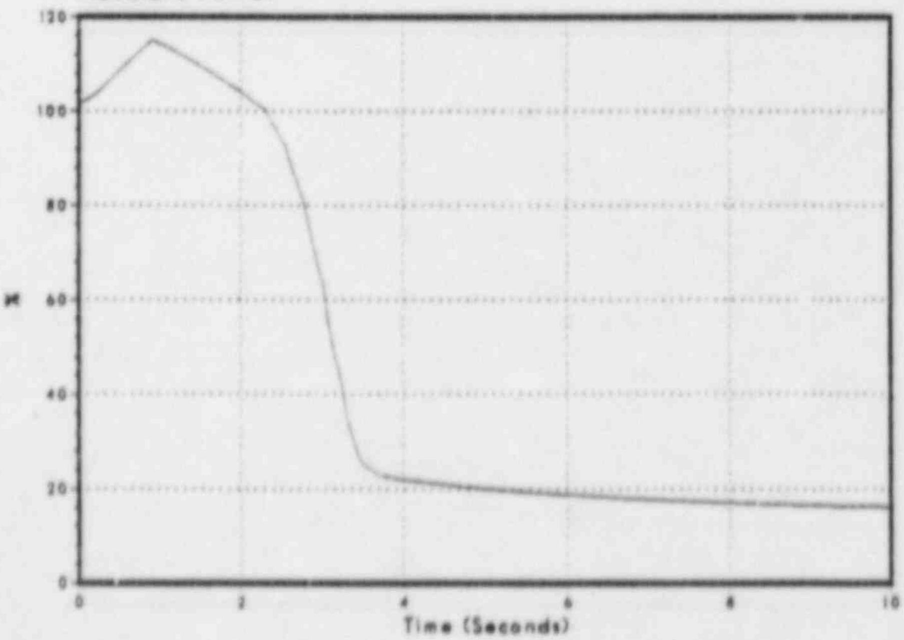


Figure 4.32
Absolute Power



NSP

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Prairie Island
DYNODE-P Locked Rotor

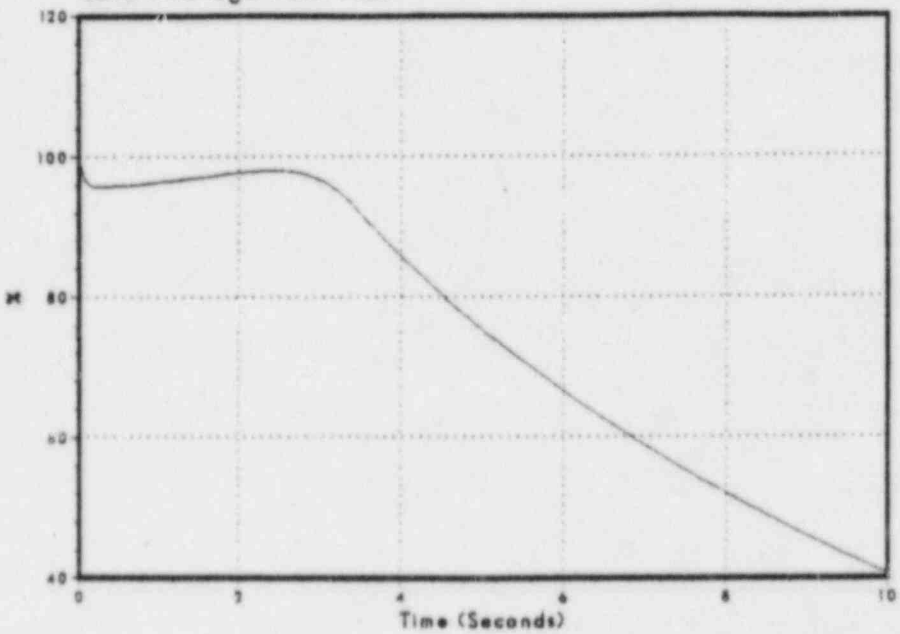

Figure 4.33
Core Average Heat Flux

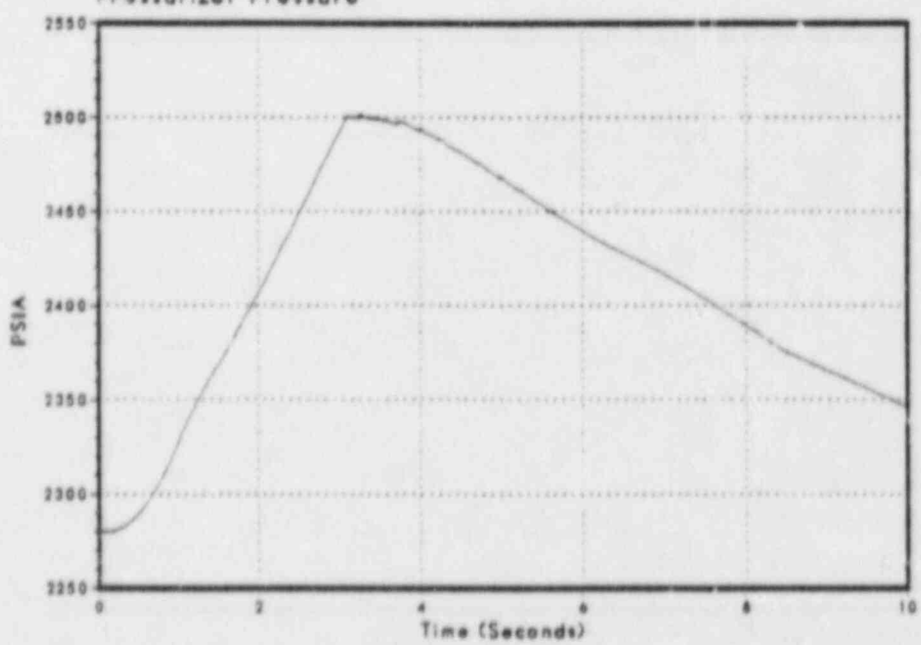

Figure 4.34
Pressurizer Pressure

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Prairie Island
DYNODE-P Locked Rotor

Figure 4.35
Core Flow

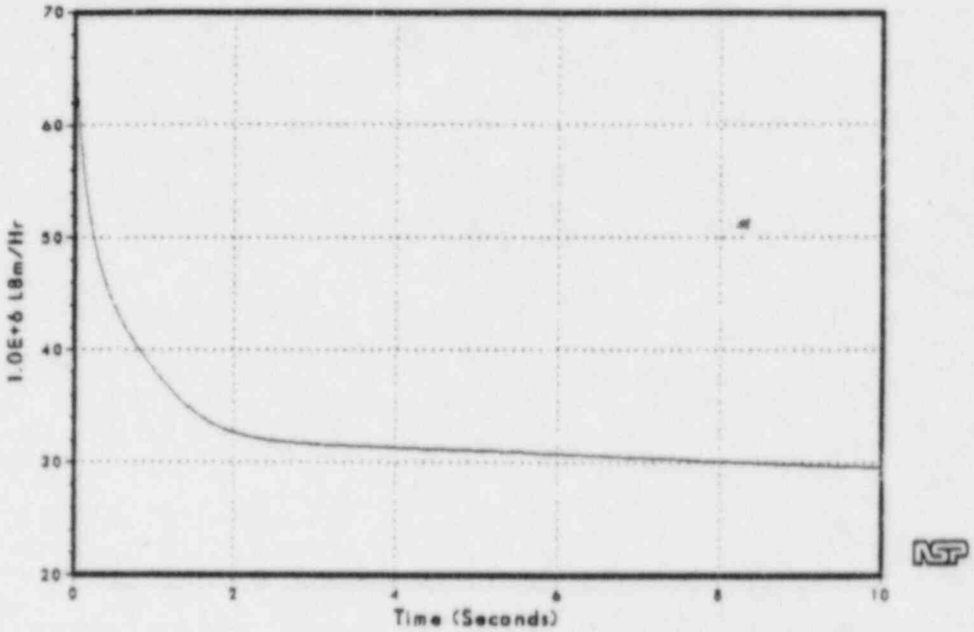
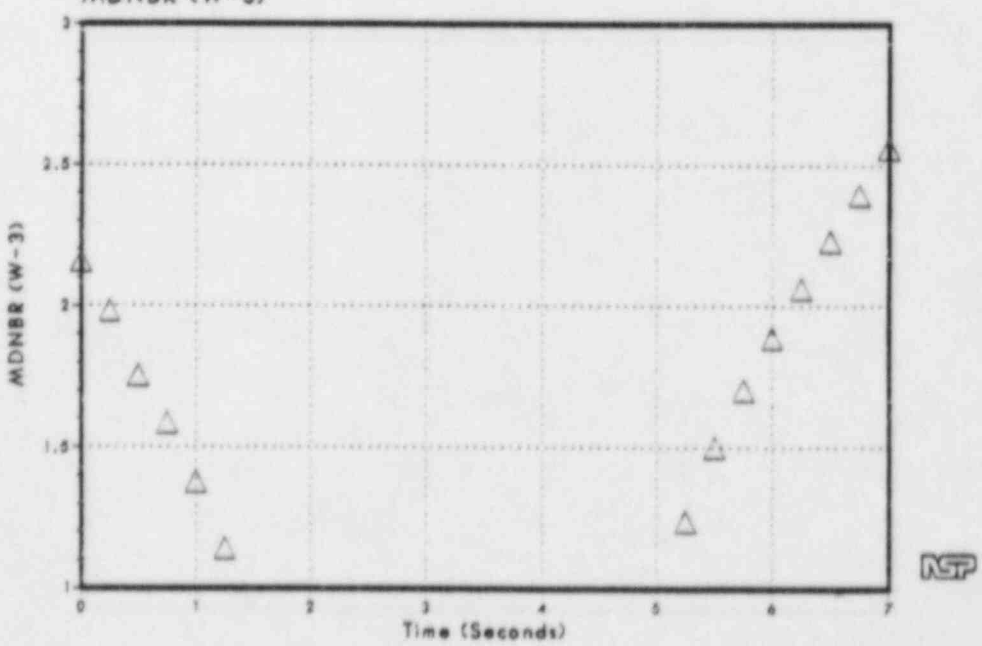


Figure 4.36
MDNBR (W-3)



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Prairie Island
DYNODE-P Large MSL Break

Figure 4.37
K Effective

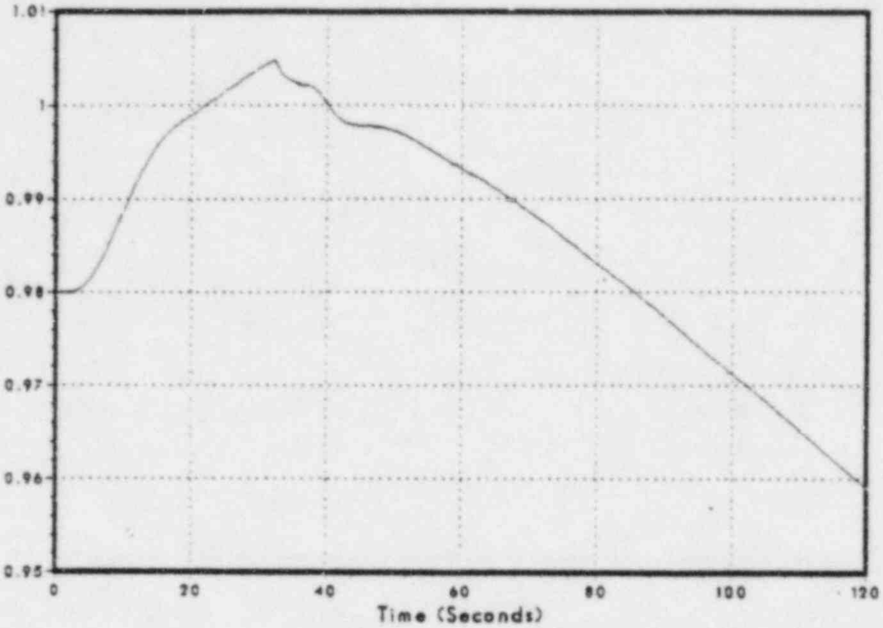
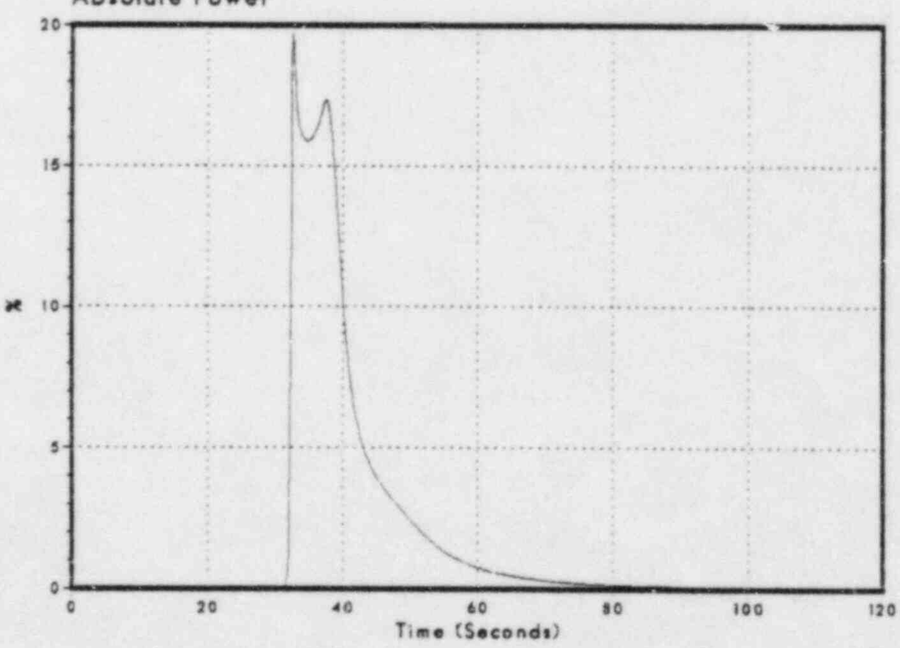


Figure 4.38
Absolute Power



NSP

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Prairie Island
DYNODE-P Large MSL Break

Figure 4.39
Core Average Heat Flux

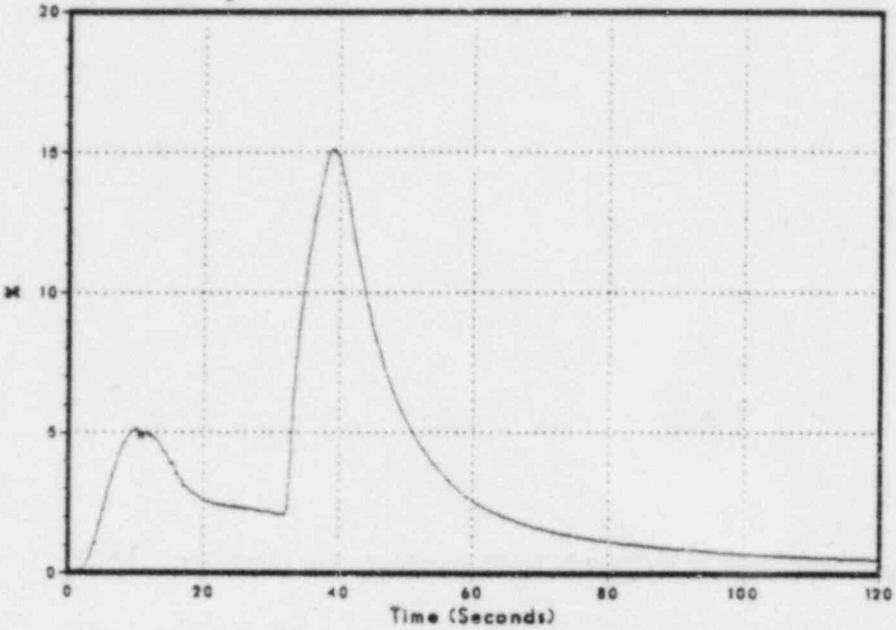
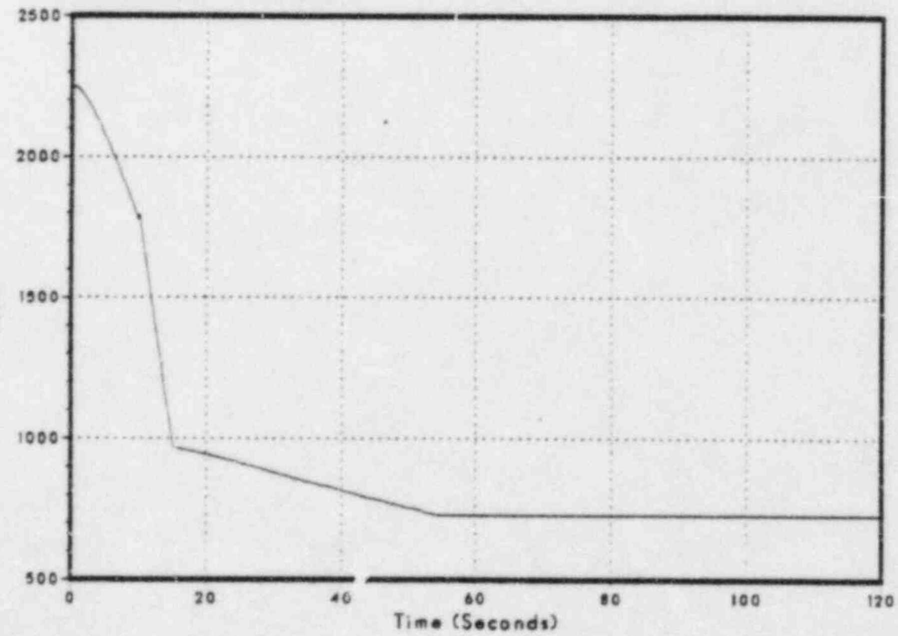


Figure 4.40
Pressurizer Pressure



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NSP

Prairie Island
DYNODE-P Large MSL Break

Figure 4.41
Pressurizer Liquid Level

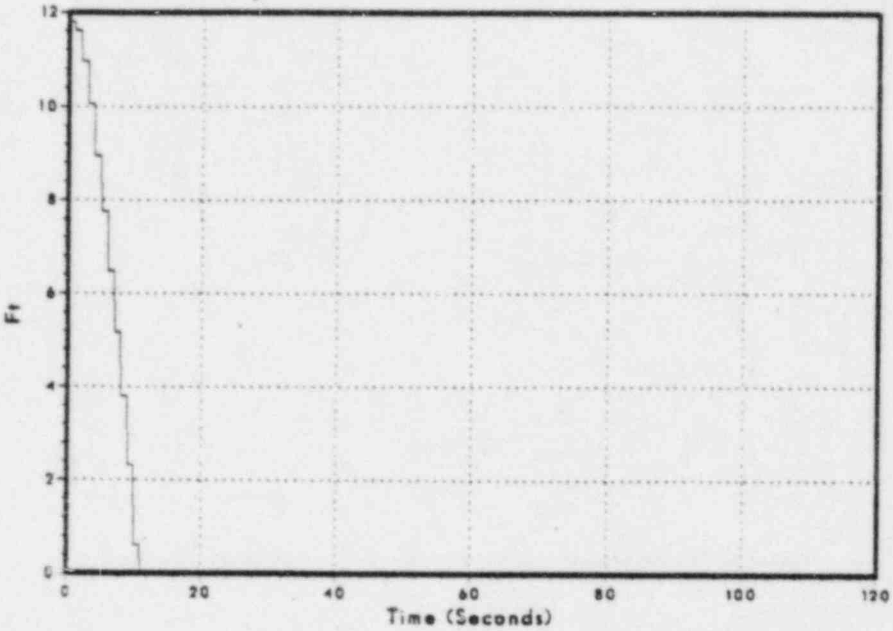
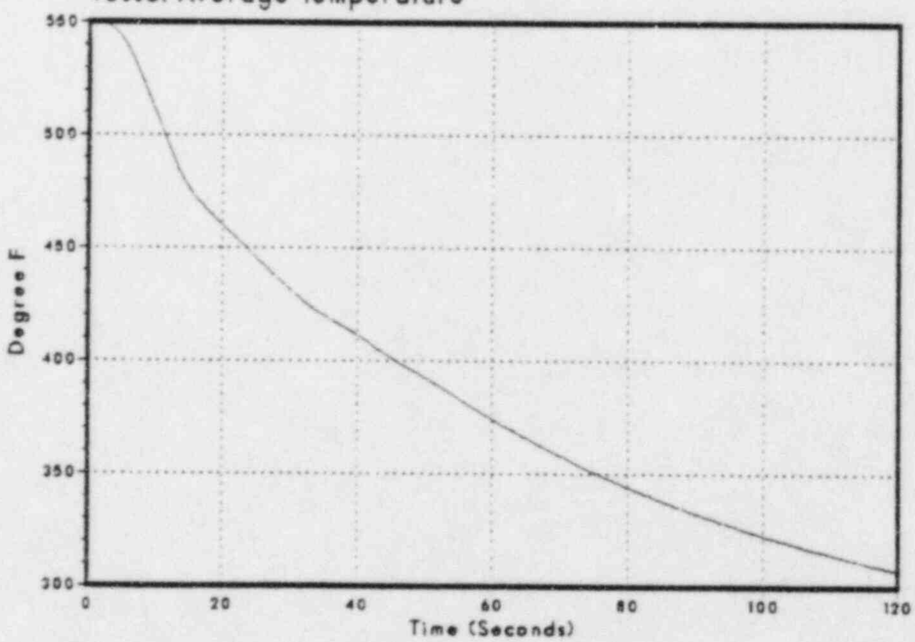


Figure 4.42
Vessel Average Temperature



NSP

Prepared by NSP Safety Analysis, 10/11/85

Prairie Island
DYNODE-P Small MSL Break

Figure 4.43
K Effective

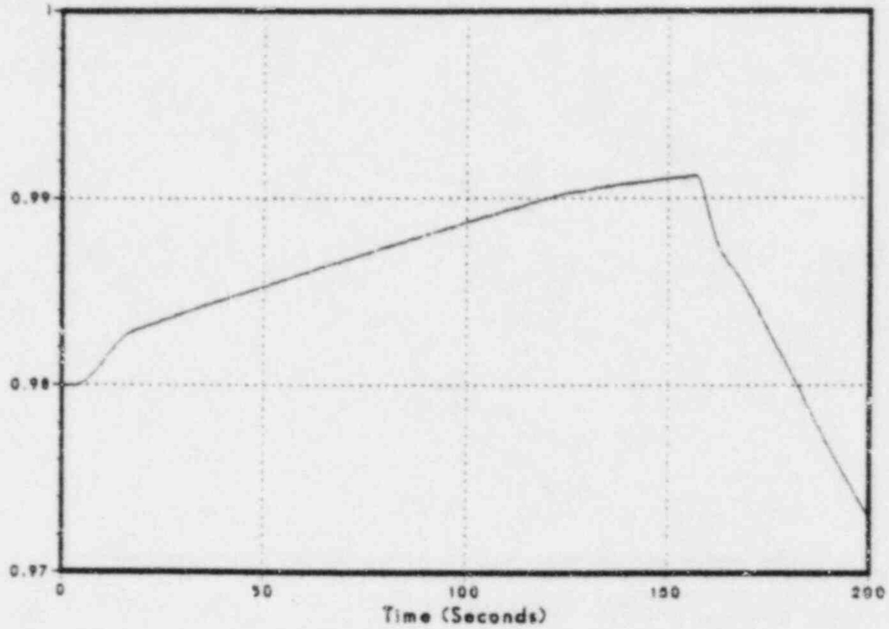
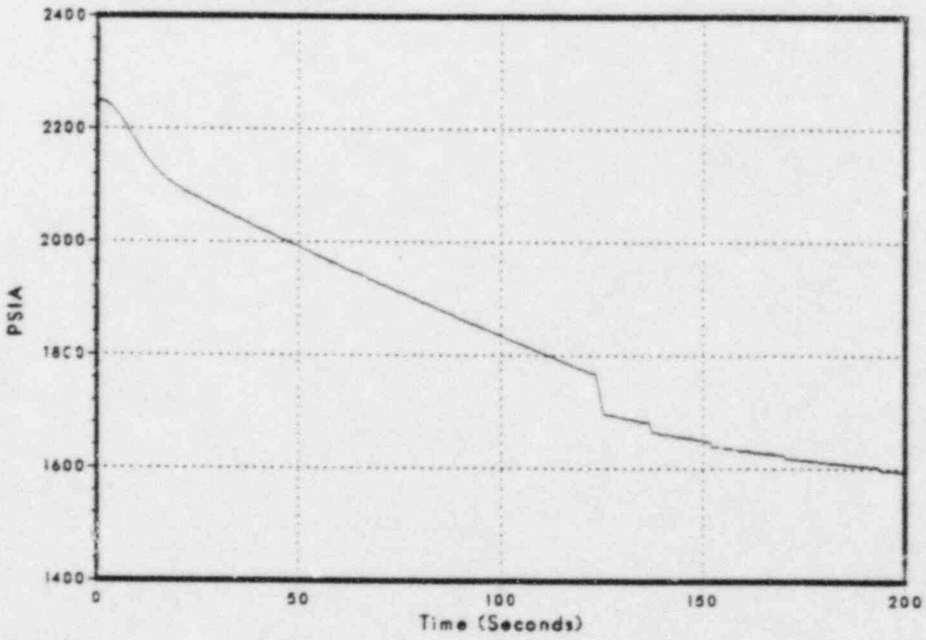


Figure 4.44
Pressurizer Pressure



NSP

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Prairie Island
DYNODE-P Small MSL Break

Figure 4.45
Pressurizer Liquid Level

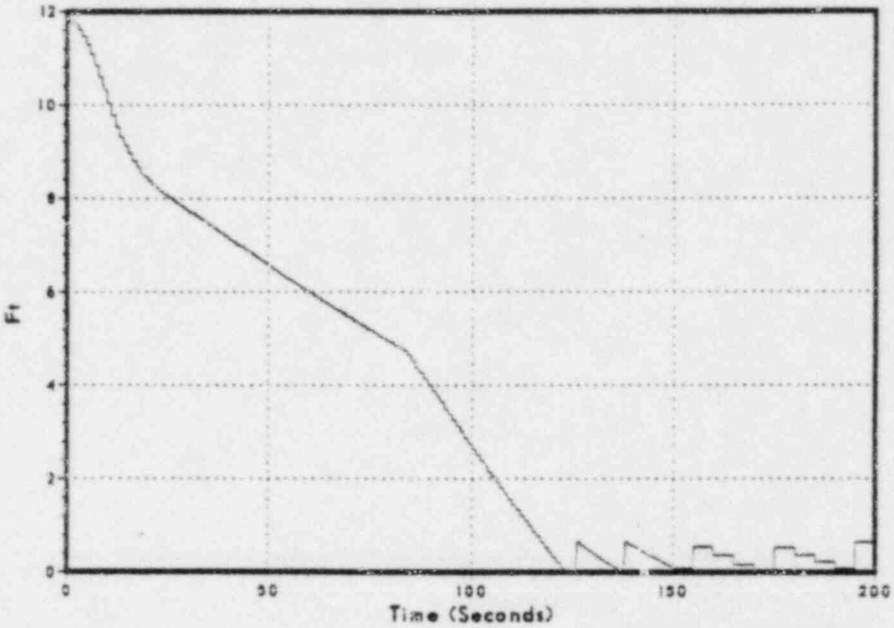
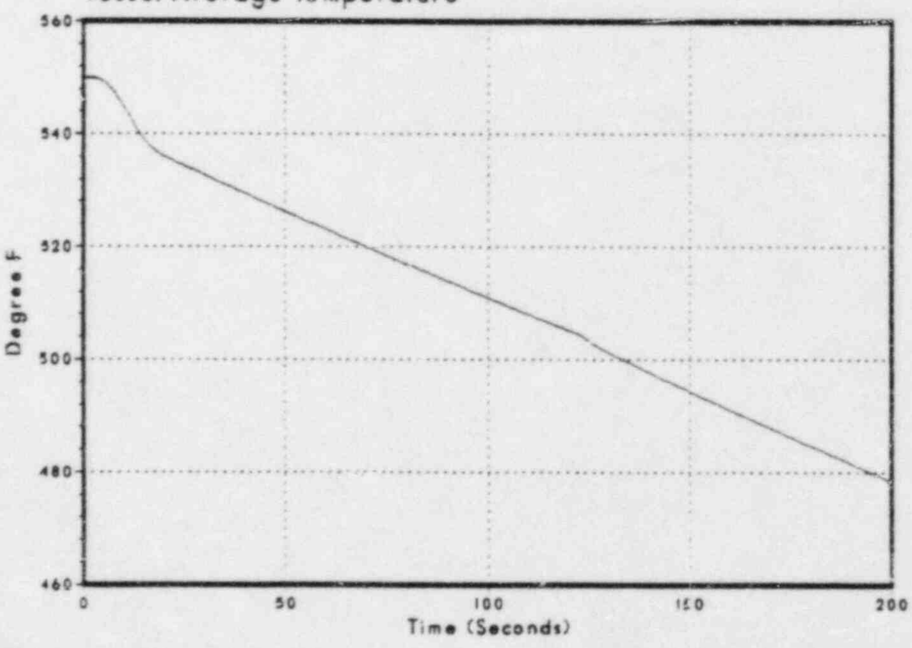


Figure 4.46
Vessel Average Temperature



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1) NSPNAI
March n Methods for Application to PI Units",

2) Prairie

3) XN-NF-78-
Units 1 a the Prairie Island Nuclear Plant,

4) XN-NF-80-6(
TOPROD Fuel Prairie Island Units 1 and 2 with ENC

5) Westinghouse
OFA and ENC T(Compatability of Westinghouse 14x14
, 1985.

6) XN-NF-75-32(P)(A
Supplement 1, 2, for Evaluating Fuel Ro Bowing",

7) WCAP-8691 Rev 1, "on, 1979.

8) XN-NF-80-61, "Prair lant Safety Analysis eport", Rev 1, March 1981.

9) Westinghouse Letter 85 ibit tion," March 12 1985.

10) Northern States Power C te Ad ve Work Instru ions, N1AWI 5.1.9, "Safety Eva

11) NSPNAD-8412, "Prairie Isl 1 and 2 aluation for RCC Guide Thimble Plug Rem ril, 19