Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods For Application to PI Units NSPNAD-8102-NPA, Revision 7

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Northern States Power Company
Nuclear Analysis & Design

RELOAD SAFETY EVALUATION METHODS

FOR

APPLICATION TO PRAIRIE ISLAND UNITS

NSPNAD-8102-NPA

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ABSTRACT

This document is a Topical Report describing the Northern States Power Company (NSP) reload safety evaluation methods for application to the Prairie Island Units.

The report addresses the methods for the calculation of cycle specific physics parameters and their comparison to the bounding values used in the accident analyses. In addition, a brief summary is presented of the NSP safety analysis experience and calculational results for Prairie Island.

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1.0 INTRODUCTION

This report addresses the methods for the calculation of Prairie Island cycle specific physics parameters and their comparison to the bounding values used in the accident analysis. It also addresses the method for determining the minimum required shutdown margin.

A brief description of the general physics calculational procedures is reviewed in section 2. General methods are described for each of the key physics parameters of interest in reload safety evaluations.

Cycle specific physics calculations and their comparisons to the safety analyses are described for each accident in section 3. The specific applications of the reliability factors described in reference 1 are also presented in this section.

A general description is given in section 3 of the accidents that are sensitive to physics parameters and are therefore of concern for a reload evaluation. For each accident, a discussion of the general input assumptions, consequences and sensitivities to various physics characteristics is provided.

Calculations of core physics parameters for the purpose of performing reload safety evaluations requires an intimate knowledge of the safety analyses to which cycle specific comparisons are to be made. Specifically, one must understand the manner in which the bounding physics parameters have been used in each of the analyses and the conservatism inherent in the values chosen. In order to acquire such an understanding, Northern States Power (NSP) has developed models for performing various safety analyses for Prairie Island and has performed representative calculations for the incidents of importance for a reload evaluation. A summary of the results of these calculations is included in section 3 to demonstrate NSP safety analysis experience and to exemplify the expertise required to make the determinations as to whether or not an accident must be re-analyzed and to perform the necessary analyses for a given fuel cycle.

A determination of those analyses which are affected by a reload design has been performed. Section 3.0 identifies the analyses which require review and itemizes the physics parameters that change necessitating an analysis review. The specific bounding values for each analysis are provided in the cycle specific Reload Safety Evaluation Report utilizing the most up-to-date analysis.

Appendix A gives an overview of the computer code package that is used to simulate the transients and accidents listed in this report. A discussion of computer code uncertainties is also included in this section.

Appendix B gives a description of the DYNODE-P computer code which is used to simulate the transient response of the Nuclear Steam supply System (NSSS).

Appendix C gives a description of the COBRA IIIC/MIT computer code which is used to simulate the thermal-hydraulic response of the hot coolant channel. A discussion of the NSP thermal margin methodology is also included in this appendix.

Appendix D gives a description of the TOODEE2 computer code which is used to simulate the thermal response of the hot fuel rod and associated coolant channel under-transient conditions. A discussion of the NSP fuel thermal response methodology is also included in this appendix.

Appendix E gives a description of the CONTEMPT-LT/026 computer code which is used to simulate the transient response of the containment. A discussion of the NSP containment analysis methodology is also included in this appendix.

Appendix F gives a description of the VIPRE-01 computer code which is to replace COBRA-IIIC/MIT for simulating the thermal hydraulic response of the hot channel. A discussion of the NSP thermal margin methodology using VIPRE-01 is also included.

Appendix G describes the methodology for determining the minimum required shutdown margin for a dilution accident for various system configurations and injection flow rates with the reactor subcritical.

2.0 GENERAL PHYSICS METHODS

In this section the general physics calculational methods are described for application to reload safety evaluations for Prairie Island.

Cycle specific calculations, the application of reliability factors, biases and comparisons to the safety analyses are discussed in Section 3 for each accident considered. Reference 6 contains detailed procedures for calculating the cycle specific parameters for each accident.

2.1 *Moderator Temperature Reactivity Coefficient, α_M*

Definition: α_M is the change in core reactivity associated with a 1°F change in average moderator temperature at constant average fuel temperature.

Calculations of α_M are performed in three dimensions with the nodal model (1). The average moderator temperature is varied while the independent core parameters such as core power level, control rod position and RCS boron concentration are held constant. Dependent core parameters such as power distribution and moderator temperature distribution are permitted to vary as dictated by the changes in core neutronics and thermal-hydraulics. The average fuel temperature is held constant and no changes in nodal xenon inventory are permitted.

2.2 *Power Reactivity Defect, $\Delta\rho_p$*

Definition: $\Delta\rho_p$ is the change in core reactivity associated with a change in core average power level.

Calculations of $\Delta\rho_p$ are performed in three dimensions with the nodal model (1). Core power is varied while all other independent parameters such as rod position and RCS boron concentration are held constant. Dependent core parameters such as power distribution, average fuel and moderator temperatures, and moderator temperature distribution are permitted to vary as dictated by the changes in core neutronics and thermal-hydraulics feedback. No changes in nodal xenon inventory are permitted.

2.3 *Doppler Reactivity Coefficient, α_D*

Definition: α_D is the change in core reactivity associated with a 1°F change in average fuel temperature at constant average RCS moderator temperature.

α_D is computed as the difference between power defect, $\Delta\rho_P$, and the moderator coefficient, α_M as shown below.

$$\alpha_D = \frac{\Delta\rho_p - \Delta T_M \ \alpha_M}{\Delta T_F}$$

2.4 Boron Reactivity Coefficient, α_B

Definition: α_B is the change in reactivity associated with a 1 PPM change in core average soluble boron concentration.

Calculations of α_B are performed in three dimensions with the nodal model (1). The core average boron concentration is varied while the <u>independent</u> core parameters such as core power level and control rod position are held constant. <u>Dependent</u> core parameters such as power distribution and moderator temperature distribution are permitted to vary as dictated by the changes in core neutronics and thermal-hydraulics. No changes in nodal xenon inventory are permitted.

2.5 Shutdown Margin, SDM

Definition: SDM is the amount of reactivity by which the core would be subcritical following a reactor trip, assuming the most reactive control rod is stuck out of the core and no changes in xenon or RCS boron concentration.

Case #1 At power condition with rods at the power dependent insertion limits.

Case #2 Hot zero power condition with all rods in except the stuck rod. No changes in xenon or boron are assumed.

Case #3 HZP with rod position from Case #1. The dependent core parameters such as power distribution and temperature distribution are permitted to vary as dictated by the changes in core neutronics and thermal-hydraulics feedback. All spatial effects and rod insertion allowances are explicitly accounted for in each calculation. The SDM is computed as the change in core reactivity between case 1 and 2. This value is conservatively reduced using Case #3, model reliability factors RF_i (Reference 1), and biases.

These factors are applied to the inserted rod worth, the temperature defect and the Doppler defect.

2.6 Scram Reactivity Curve, ($\Delta\rho_{SCRAM}(t)$)

Definition: $\Delta\rho_{SCRAM}(t)$ is the rod worth inserted into the core as a function of time after rod release. The most reactive rod is assumed to remain fully withdrawn.

The independent core parameters such as power level, RCS boron concentration and xenon inventory are held constant during the insertion. The dependent parameters such as flux distribution are permitted to vary as dictated by the changes in core neutronics.

A conservatively slow scram curve is generated by making the following assumptions:

1. The integral of the scram curve is based on an initial rod position at or below power dependent insertion limits.
2. The shape of the scram curve is based on an initial rod position of full out. The rods are assumed to move uniformly together. This provides the longest possible delay to significant reactivity insertion.
3. The positional insertion dependence is converted into a time dependent function using empirical data relating rod position to time after rod release. The empirical data is normalized such that the total time to rod insertion is equal to or greater than the limits defined by the Technical Specifications.
4. The xenon distribution is that which causes the minimum shutdown margin.
5. Instantaneous redistribution of flux is assumed to occur during the rod insertion.

2.7 Nuclear Heat Flux Hot Channel Factor, F_Q

Definition: F_Q is the maximum local fuel rod linear power density divided by the core average fuel rod linear power density.

Calculations of F_Q are based on three dimensional power distributions obtained with the nodal model (1) coupled with local peak pin to assembly power ratios obtained from the quarter core PDQ model (1). Model reliability factors and biases are used to increase F_Q to a conservative value.

2.8 Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}$

Definition: $F_{\Delta H}$ is the ratio of the integral of linear power along the rod on which minimum DNBR occurs to the core average integral rod power.

Calculations of $F_{\Delta H}$ are based on three-dimensional power distributions obtained with the nodal model (1) coupled with the local peak pin to assembly power ratio obtained from the quarter core PDQ model (1). Model reliability factors and biases are used to increase $F_{\Delta H}$ to a conservative value.

2.9 *Effective Delayed Neutron Fraction, β_{eff}*

Definition: β_{eff} is the core effective total delayed neutron fraction.

Nodal values of β, for core delay neutron group i, are determined by weighing the delayed neutron fraction from each fissile isotope by the local fission sharing as determined from CASMO II. This local result is then power weighted using the nodal 3D power distribution. The importance factor I, applied as ≡ .97, accounts for the effects of reduced fast fissioning, increased resonance escape, and decreased fast leakage by the delayed neutrons. β_{eff} is the product of β and I, where $\beta=\Sigma\beta_i$.

2.10 *Prompt Neutron Lifetime, ℓ **

Definition: ℓ* is the average time from the emission of a prompt neutron in fission to the absorption of the neutron somewhere in the reactor.

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3.0 SAFETY EVALUATION METHODS

This section addresses the evaluation of the cycle specific physics parameters with respect to the bounding values used in the safety analyses. Specific methods are described for each accident or transient by which the determination is made as to whether or not any re-analysis is required. For each accident or transient the following material is described:

a. Definition of Accident - a brief description of the causes and consequences.
b. Accident Analysis - a brief description of the methods employed and discussion of the sensitive physics parameters. Included is a list of the acceptance criteria.
c. NSP Safety Analysis Experience - a brief summary of the NSP calculational experience and results of the comparisons of their models to the Prairie Island Final Safety Analysis Report (Reference 2).
d. Cycle Specific Physics Calculations - a description of the specific physics calculations performed each cycle for the purposes of a reload safety evaluation.
e. Reload Safety Evaluation - A description of the comparisons of the cycle specific physics characteristics and the bounding values used in the safety analysis. Specific applications of the model reliability factors and biases are also addressed. Biases and reliability factors are to be applied in the following form:

- **Moderator Temperature Coefficient (α_M)**

  Apply in a conservative direction as follows:

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- **Doppler Coefficient (α_D)**

  Apply in a conservative direction as follows:

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- **Boron Reactivity Coefficient (α_B)**

  Apply in a conservative direction using the following form:

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- **Scram Reactivity ($\Delta\rho_{SCRAM}(t)$)**

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- **Nuclear Heat Flux Hot Channel Factor (F_Q)**

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- **Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)**

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- **Effective Delayed Neutron Fraction:** β_{eff}

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- **Prompt Neutron Lifetime (ℓ*)**

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- **Rod Worth ($\Delta\rho_R$)**

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The specific numerical values assigned as the bounding values for each accident for purposes of performing the Prairie Island reload safety evaluations will be presented in the cycle specific Reload Safety Evaluation Report.

If an accident or transient requires re-analysis because any one of the cycle specific physics parameters exceeds the current bounding value, the re-analysis will be performed utilizing the transient analysis methodology as described herein for that specific event and which has been qualified by the presented results.

3.1 *Uncontrolled RCCA Withdrawal from a Subcritical Condition*

3.1.1 Definition of Accident

An uncontrolled addition of reactivity due to uncontrolled withdrawal of a Rod Cluster Control Assembly (RCCA) results in a power excursion. The nuclear power response is characterized by a very fast rise terminated by the reactivity effect of the negative fuel temperature coefficient. After the initial energy release, the reactor power is reduced by this inherent feedback and the accident is terminated by a reactor trip. Due to the small amount of energy released to the core coolant, pressure and temperature excursions are minimal during this accident.

3.1.2 Accident Analysis

The uncontrolled RCCA withdrawal from a subcritical condition is analyzed using a dynamic simulation incorporating point neutron kinetics, including delayed neutrons and decay heat; fuel, clad, and gap heat conduction; and channel coolant thermal-hydraulics. The reactivity effects due to moderator and fuel temperature effects, as well as that due to control rod insertion after trip, are included.

The core is assumed initially to be at hot zero power, HZP. Power is supplied to the RCCA drive mechanisms such that no more than two banks may be withdrawn simultaneously. The maximum reactivity insertion due to the rods is therefore conservatively assumed to be due to two banks of maximum worth moving simultaneously at maximum speed through the region of highest differential worth.

The magnitude of the power peak reached during the transient is strongly dependent upon the Doppler reactivity coefficient for a given rate of reactivity insertion. A value conservatively small in absolute magnitude, which generally occurs at Beginning of Cycle (BOC), is assumed for the accident analysis. The magnitude of the power spike is relatively insensitive to the value of moderator temperature reactivity coefficient chosen. The least negative value, occurring at BOC, maximizes the calculated consequences of the accident.

In calculating reactivity due to control rod insertion by reactor trip, the most adverse combination of instrument and setpoint errors and time delays is assumed. The power range - low range trip setpoint is assumed to be 10% (of full power) above its nominal value. The most reactive rod is assumed to stick in the fully withdrawn position when the trip signal is actuated.

As long as the reactivity insertion remains small compared to β_{eff}, the total delayed neutron yield, the shortest reactor period during the transient will remain large compared to ℓ^*, the mean neutron lifetime. In this case, the transient core power response is relatively insensitive to the value of ℓ^* and is determined predominately by the yields and decay constants of the delayed neutron precursors. The postulated initial core pressure and temperature are conservatively taken as the minimum and maximum, respectively, that are consistent with the assumed rod and power configurations.

The results of the analysis are compared to the following acceptance criteria:

a. The maximum power density in the fuel must be less than that at which center line melting or other modes of fuel failure occur.
b. The minimum departure from nucleate boiling ratio (DNBR) must be greater than 1.3 using the W-3 correlation or 1.17 using the WRB-1 correlation, whichever is applicable to the specific fuel type.

3.1.3 NSP Safety Analysis Experience

NSP has analyzed the uncontrolled RCCA withdrawal from a subcritical condition transient using input assumptions consistent with the Prairie Island FSAR (Reference 2).

The models described in Appendix A were used to analyze the case of a rapid (8.2 x 10^{-4} Δk/sec) RCC assembly withdrawal from subcritical. The results of these calculations are compared to Figures 14.1-3, 14.1-4, and 14.1-5 of Reference 2 in Figures 3.1-1 to 3.1-5.

The NSP model predicts higher fuel, clad, and coolant temperatures than those of Reference 2, however, the NSP fuel model is consistent with the Doppler and moderator reactivity coefficients used so that the nuclear power and core average heat flux compare well with the Reference 2 results.

A sensitivity study showing the effect of initial power level on peak heat flux was performed and the results are compared to Figure 14.1-2 of Reference 2 in Figure 3.1-6.

This study was only run at one reactivity insertion rate, i.e. 8.2E-4 Δk/sec, the same insertion rate that was used to generate Figures 3.1-1 to 3.1-5, however, the results compare well to the FSAR results.

3.1.4 Cycle Specific Physics Calculations

These calculations are performed at the most limiting core conditions found during the cycle, e.g. the point in time, and the power level, that gives the most conservative result with respect to the acceptance criteria. Sensitivity studies are conducted to determine the limiting conditions accounting for the effects of control rods, xenon, power level, temperature, etc. for each parameter.

a. **Moderator Temperature Coefficient (α_M)**

Calculations of α_M are performed in accordance with the general procedures described in Section 2.0.

b. **Doppler Temperature Coefficient (α_D)**

Calculations of α_D are performed in accordance with the general procedures described in Section 2.0.

c. **Scram Reactivity Curve, ($\Delta\rho_{SCRAM}(t)$)**

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2.0.

d. **Effective Delayed Neutron Fraction (β_{eff})**

Calculations of β_{eff} are performed in accordance with the general procedures described in Section 2.0.

e. **Maximum Reactivity Insertion Rate ($\Delta\rho/\Delta t$)**

The assumption is made that two banks of highest worth will be withdrawn simultaneously at maximum speed. 0This value requires two components: the maximum withdrawal speed in inches per second, and the maximum differential reactivity insertion per inch for the two maximum worth rod banks moving in 100% overlap. This result is obtained by first calculating the two banks which have the maximum worths. These two banks are then moved simultaneously at HZP. A rod worth reliability factor and bias are applied to the integral worth by position.

3.1.5 Reload Safety Evaluations

Each of the physics parameters calculated above are adjusted to include the model reliability factors, RF_i and biases, B_i. These adjusted values are the cycle specific parameters which are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable if the following inequalities are met:

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The integral of the bounding value of the scram curve, $\Delta\rho_{SCRAM}(t)$, is taken as that rod worth required to produce the shutdown margin assumed in the safety analysis for the most limiting cycle specific core conditions.

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Figure 3.1- 1: Uncontrolled Control Rod Withdrawal From Subcritical

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Figure 3.1- 2: Uncontrolled Control Rod Withdrawal From Subcritical

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Figure 3.1- 3: Uncontrolled Control Rod Withdrawal From Subcritical

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Figure 3.1- 4: Uncontrolled Control Rod Withdrawal From Subcritical

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Figure 3.1- 5: Uncontrolled Control Rod Withdrawal From Subcritical

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Figure 3.1- 6: Uncontrolled Control Rod Withdrawal From Subcritical

3.2 *Uncontrolled RCCA Withdrawal at Power*

3.2.1 Definition of Accident

An uncontrolled RCCA withdrawal at power results in a gradual increase in core power followed by an increase in core heat flux. The resulting mismatch between core power and steam generator heat load results in an increase in reactor coolant temperature and pressure. The fuel in the reactor core could eventually encounter departure from nucleate boiling if the power excursion were not checked by the reactor protection system. Depending on the initial power level and rate of reactivity insertion, the following trips serve to prevent fuel damage or overpressurization of the coolant system: high nuclear power, over temperature and over power ΔT, high pressurizer level, and high pressurizer pressure. For the more rapid rates of reactivity insertion, the maximum power reached during the transient will exceed the power at the time the trip setpoint is reached by an amount proportional to the insertion rate and the time delay associated with trip circuitry and rod motion.

3.2.2 Accident Analysis

The uncontrolled RCCA withdrawal at a power condition is analyzed using a dynamic simulation incorporating point neutron kinetics, reactivity effects of moderator, fuel and rods, and decay heat. A simulation of the reactor vessel, steam generator tube and shell sides, pressurizer, and connecting piping is required to evaluate the coolant pressure and core inlet temperature response and their effect on core thermal margins. The reactor trip system, main steam and feedwater systems, and pressurizer control systems are also included in the model. This model calculates the response of the average core channel thermal-hydraulic conditions and heat generation and is coupled to a detailed model of the hot channel. This latter model calculates the departure from nucleate boiling ratio (DNBR) as a function of time during the accident.

In order to maximize the peak power during the transient, the fuel and moderator temperature coefficients used in the analysis are the least negative likely to be encountered. The least negative Doppler and moderator coefficients are normally encountered at BOC.

The reactivity reduction due to reactor trip is calculated by considering the most adverse combination of instrument and setpoint errors and time delays. The rate of reactivity insertion corresponding to the trip of the RCC assemblies is calculated assuming that the most reactive assembly is stuck in the fully withdrawn position.

Since the reactivity insertion rate determines which protective system function will initiate termination of the accident, a range of insertion rates must be considered. Relatively rapid insertion rates result in reactor trip due to high nuclear power. The maximum rate is bounded by that calculated assuming that the two highest worth banks, both in their region of highest incremental worth, are withdrawn at their maximum speed. Relatively slow rates of reactivity insertion result in a slower transient which is terminated by an Overtemperature ΔT trip signal, or in some cases, a high pressurizer pressure signal. The minimum rate which need be considered in the analysis is determined by reducing the reactivity insertion rates until the analysis shows no further change in DNBR.

The accepted criteria for this accident are that the maximum pressures in the reactor coolant and main steam systems do no exceed 110% of design values and that cladding integrity be maintained limiting the minimum DNB ratio greater than 1.3 (W-3) or 1.17 (WRB-1), whichever is applicable.

3.2.3 NSP Safety Analysis Experience

NSP has analyzed the uncontrolled RCCA withdrawal from a full power condition transient using input assumptions consistent with the Prairie Island FSAR (Reference 2).

The models described in Appendix A were used to analyze the following two control rod withdrawal transients from full power:

- Fast rate (8.2×10^{-4} ΔK/sec)
- Slow rate (3.0×10^{-5} Δk/sec)

The transient response of the NSSS for the fast rate case is compared to Figures 14.1-6 and 14.1-7 of Reference 2 in Figures 3.2-1 - 3.2-3. The reactor trip is generated on high neutron power for this case. The T_{AVE} and pressure responses are slightly more severe using the NSP models, however, the NSP models show the same trends as the FSAR results.

The corresponding NSSS results for the slow rate case are compared to Figures 14.1-8 and 14.1-9 of Reference 2 in Figures 3.2-5 - 3.2-7. The NSP results predict a slower power ramp and corresponding by slower pressure and T_{AVE} increases. A reactor trip is generated on Overtemperature ΔT for this case. The NSP model uses a dynamic simulation of the setpoint generator and predicts a trip at a slightly lower power level than Reference 2.

Figures 3.2-4 and 3.2-8 show the transient hot channel DNBR comparisons to Figures 14.1-7 and 14.1-9 of Reference 2 for the fast and slow rate cases respectively.

The NSP hot channel DNBR analyses were computed using both a single closed channel model, a multichannel 1/8 assembly model, and an 1/8 core lumped subchannel model. It is believed that the single channel model and the 1/8 assembly model will provide a better comparison to the FSAR, however, the 1/8 core model is more accurate and will be used for licensing analyses. A detailed description of the NSP Thermal Margin methodology is given in Appendix C.

The NSP single channel model predicts a MDNBR of 1.726 for the fast withdrawal case and 1.641 for the slow withdrawal case as compared to 1.63 and 1.36 from Reference 2 respectively. Note that the initial MDNBR of the NSP model is slightly higher, (≈.038), tending to bias the results throughout the transient.

3.2.4 Cycle Specific Physics Calculations

These calculations are performed at the most limiting core conditions found during the cycle, e.g., the point in time and the power level that gives the most conservative result with respect to the acceptance criteria. Sensitivity studies are conducted to determine these limiting conditions accounting for the effects of control rods, xenon, power level, temperature, etc. for each parameter.

a. Moderator Temperature Coefficient (α_M)

Calculations of α_M are performed in accordance with the general procedures described in Section 2.0.

b. Doppler Temperature Coefficient (α_D)

Calculations of α_D are performed in accordance with the general procedure described in Section 2.0.

c. Scram Reactivity Curve ($\Delta\rho_{SCRAM}(t)$)

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2.0.

d. Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)

The maximum core $F_{\Delta H}$ is assumed to remain within the current limits as defined in the Technical Specifications for allowable combinations of axial offset and power level. For Prairie Island, the continuous surveillance of the power distribution is accomplished with the excore detectors using the Exxon PDC-IIa[3] scheme.

e. **Effective Delayed Neutron Fraction (β_{eff})**

Calculations of β_{eff} are performed in accordance with the general procedures described in Section 2.0.

f. **Maximum Reactivity Insertion Rate ($\Delta\rho/\Delta t$)**

Calculations similar to those described in Section 3.1.4 © are performed at the full power, and constant equilibrium xenon conditions.

3.2.5 Reload Safety Evaluations

Each of the physics parameters calculated above are adjusted to include the model reliability factors, RF_i, and biases, B_i. These adjusted values are the cycle specific parameters which are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable if the following inequalities are met:

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The integral of the bounding value of the scram curve, $\Delta\rho_{SCRAM}(t)$, is taken as that rod worth required to produce the shutdown margin assumed in the safety analysis for the most limiting cycle specific core conditions.

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Figure 3.2- 1: Uncontrolled Control Rod Withdrawal From Full Power - Fast Rate

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Figure 3.2- 2: Uncontrolled Control Rod Withdrawal From Full Power - Fast Rate

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Figure 3.2- 3: Uncontrolled Control Rod Withdrawal From Full Power - Fast Rate

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Figure 3.2- 4: Uncontrolled Control Rod Withdrawal From Full Power - Fast Rate

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Figure 3.2- 5: Uncontrolled Control Rod Withdrawal From Full Power - Slow Rate

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Figure 3.2- 6: Uncontrolled Control Rod Withdrawal From Full Power - Slow Rate

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Figure 3.2- 7: Uncontrolled Control Rod Withdrawal From Full Power - Slow Rate

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Figure 3.2- 8: Uncontrolled Control Rod Withdrawal From Full Power - Slow Rate

3.3 *Control Rod Misalignment*

3.3.1 Definition of Accident

In the analysis of this accident, one or more rod cluster control assemblies (RCCA) is assumed to be statically misplaced from the normal or allowed position. This situation might occur if a rod was left behind when inserting or withdrawing banks, or if a single rod was to be withdrawn. Full power operation under these conditions could lead to a reduction in DNBR and is subject to limitations specified in the plant Technical Specifications.

3.3.2 Accident Analysis

In the analysis of misaligned control rods, $F_{\Delta H}$ will be determined for the most limiting configuration. In general, the worst case is that with Bank D fully inserted, except for a single withdrawn assembly, since Bank D is the only bank which may be inserted at full power. In practice, multiple independent alarms would alert the operator well before the postulated conditions are approached.

The limiting value of $F_{\Delta H}$ is input to a steady state thermal-hydraulic subchannel calculation to determine the departure from nucleate boiling ratio (DNBR). This calculation assumes the most adverse combination of steady state errors applied to core neutron flux level, coolant pressure, and coolant temperature at the core inlet.

The acceptance criteria for this accident are that the calculated DNBR is not less than 1.3 (W-3) or 1.17 (WRB-1), whichever is applicable, and that fuel temperature and cladding strain limits consistent with the acceptance criteria of Standard Review Plan 4.2 are not exceeded.

3.3.3 NSP Safety Analysis Experience

NSP has analyzed the control rod misalignment accident using input consistent with the Prairie Island Final Safety Analysis Report (Reference 2). Using the methods described in Appendix A, the control rod misalignment incident was analyzed using a hot channel factor ($F_{\Delta H}$) of 1.92. The DNBR obtained was 1.470 using a multichannel, 1/8 assembly hot channel model which is in agreement with the Reference 2 result of greater than 1.38.

3.3.4 Cycle Specific Physics Calculations

These calculations are performed at the most limiting core conditions found during the cycle, e.g., the point in time and the power level that gives the most conservative result with respect to the acceptance criteria. Sensitivity studies are conducted to determine these limiting conditions accounting for the effects of control rods, xenon, power level, temperature, etc. for each parameter.

The nuclear enthalpy rise hot factor ($F_{\Delta H}$) is calculated for this accident consistent with the procedure described in Section 2.0. The maximum $F_{\Delta H}$ for a control rod misalignment at full power is calculated with Bank D at the full power insertion limit (FPIL) and one rod cluster of Bank D fully withdrawn.

3.3.5 Reload Safety Evaluations

The $F_{\Delta H}$ calculated above is conservatively adjusted to account for the model reliability factor, $RF_{F\Delta H}$, and bias, $B_{F\Delta H}$. Additionally, a further adjustment is made to account for the maximum initial quadrant tilt condition (T) allowed by the Technical Specifications. The resultant $F_{\Delta H}$ is then compared to the value used in the safety analysis as follows:

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3.4 *Control Rod Drop*

3.4.1 Definition of Accident

In the analysis of this accident, a full-length RCCA is assumed to be released by the gripper coils and to fall into a fully inserted position in the core.

A dropped rod cluster control assembly (RCCA) typically results in a reactor trip signal from the power range negative neutron flux rate circuitry. The core power distribution is not adversely affected during the short interval prior to reactor trip.

The drop of a single RCCA may not result in a reactor trip. The calculated consequences for this event are dependent upon whether the reactor is being operated in an automatic or manual mode. For operation in the manual mode, the plant is brought back to full power with an assembly fully inserted and a reduction in core thermal margins may result because of a possible increased hot channel peaking factor. If a rod drop event occurs when the reactor is in the automatic mode, the reactor control system responds to both the reactor power drop, as seen by the excore detectors, (mismatch between turbine power and reactor power) and the decrease in the core average temperature and attempts to restore both quantities to their original values. This restoration of reactor power by the reactor control system may result in some power overshoot. This power overshoot combined with the possible increased hot channel peaking factor (due to the inserted RCCA) will cause a reduction in the core thermal margin.

3.4.2 Accident Analysis

The analysis for the control rod drop accident is divided into two parts; (1) determination of which dropped rods will trip the negative flux rate scram system and thus require no further analysis and (2) the determination of the consequences of the transient for rods which do not cause a trip.

For the part (1) analysis of which rods will trip the negative flux rate scram system, a dynamic simulation is performed using the DYNODE-P code (Appendix B) using the following input parameters and assumptions.

- Dropped rod specific physics parameters determined using static methods described in section 2.0 (listed in section 3.4.5.1).

- The relative tilt as seen by the excore detectors is evaluated by correlating the core edge power densities to the relative excore detector readings.
- A nominal negative flux rate trip setpoint of 5 percent reactor total power (RTP) with a time constant of 2 seconds; with uncertainty, a setpoint of 6.9 percent RTP with a time constant of 2 seconds is used.
- Two out of four (with worst failure) excore detector rate trip system logic.

Only single RCCA drops were examined. Multiple drops could result from a single failure, however, it was found that only the lowest worth single RCCA drops did not cause a flux rate trip and therefore multiple drops were not examined. If this situation were to change in future cycles, multiple drops would be examined.

For the part (2) analysis of the transient consequences, a dynamic simulation is performed, for those rods which do not cause a trip, using the DYNODE-P code (Appendix B) including the rod control system. The following input parameters and assumptions are used.

- Dropped rod specific physics parameters determined using static methods described in section 2.0 (listed in section 3.4.5.2).
- The relative tilt as seen by the excore detectors is evaluated by correlating the core edge power densities to the excore detector readings. The detector tilt is conservatively assumed constant throughout the transient.
- Excore detectors averaged response (with worst failure) as input to the rod controller.
- Technical Specification control bank insertion.
- Rod controller withdrawal deadband ignored.
- The calculation is performed assuming full power with the most adverse combination of steady state errors applied to core neutron flux level, coolant pressure, and coolant temperature at the core inlet.
- Least negative moderator and Doppler coefficients are used to maximize the transient power overshoot, since the reactor temperature increases over its initial value.

Sensitivity studies were performed on rod worth, rod drop time, and initial rod speed error bias. It was found that the largest rod worth, the slowest rod drop time, and the largest initial rod speed error bias caused the worst transient response.

For each rod, three xenon conditions are examined, equilibrium, highly positive and highly negative axial offset. If these three cases do not produce similar results, i.e. reactor trip or no trip, a sensitivity study is performed to find the highest worth for that particular rod which will not cause a trip. Bounding values for excore tilt (Part 1 analysis) and for excore tilt and peaking factors (Part 2 analysis) are used.

Beginning and end of cycle conditions were examined. End of cycle conditions result in the maximum rod worth not to cause a reactor negative flux rate trip (part 1 analysis) and thus result in the highest power overshoot in the part 2 analysis. Beginning of cycle conditions result in lower transient MDNBR due to the higher peaking factors.

For the case with the rod controller in manual, the core is assumed to return to hot full power conditions under manual control.

The DNB response during a dropped rod transient is evaluated using the methodology described in Appendix C. This methodology is modified slightly since the dropped rod transient causes core power redistribution and hence loss of core symmetry. A full core VIPRE model is used, rather than a 1/8 core model as in Appendix C, to account for the non-symmetric core radial power distribution. Dropped rod specific peaking factors and power distributions are obtained using the physics methods described in section 2.0, and are conservatively assumed constant throughout the transient. The acceptance criteria for the accident are that the DNBR calculated is not less than 1.3 (W-3) or 1.17 (WRB-1), whichever is applicable, and that fuel temperature and cladding strain limits consistent with acceptance criteria of Standard Review Plan 4.2 are not exceeded.

3.4.3 NSP Safety Analysis Experience

NSP has analyzed the control rod drop accident using input consistent with the Prairie Island Final Safety Analysis Report (Reference 2), i.e. manual rod control. A hot channel factor ($F_{\Delta H}$) of 1.62 is used with a multichannel 1/8 assembly hot channel COBRA model (Appendix C). The DNBR obtained was 1.934 which is in good agreement with the Reference 2 result of "greater than 1.9."

Prairie Island Unit 2 Cycle 8 and Unit 1 Cycle 9 have been analyzed using the methodology described in section 3.4.2. A summary of the part (1) results is shown on tables 3.4-1 and 3.4-2 for PI 2 Cycle 8 and PI 1 Cycle 9 respectively. The worst case results, with respect to MDNBR, for part (2) of the analyses, are shown in Figures 3.4-1 through 3.4-5 and 3.4-6 through 3.4-10 for PI 2 Cycle 8 and PI 1 Cycle 9 respectively.

The transient response is similar for both cycles. The initial rod drop causes a drop in reactor power and is followed by a corresponding drop in vessel average temperature and pressure. The automatic rod controller (Figure 3.4-11) responds to the sensed mismatch between vessel average and reference temperature and between turbine and reactor power and starts to withdraw the control bank rods. The excore detectors magnify the power mismatch signal. The rod controller continues to pull the control bank rods until the rod speed error signal drops below (in absolute magnitude) the lockup temperature (-1.0 °F). When the rods stop, the rod speed error begins to increase (in absolute magnitude) and then slowly decreases as the power and temperature approach a steady state. If the increase in rod speed error, after a rod pull, is sufficient to reach the deadband temperature, (-1.5 °F) then a second rod pull is initiated and the cycle continues until a new steady state condition is reached. For the PI 1 Cycle 9 analysis, four separate rod pulls are initiated with a total of 134 pcm being added by the automatic rod controller.

Figure 3.4-12 shows the actual response of the automatic rod controller including all of the functions shown in Figure 3.4-11 for PI 1 Cycle 9. Figure 3.4-13 shows the calculated response of the controller variables without simulating the controller circuitry. The automatic circuitry causes an additional three rod pulls (29 pcm), after the initial pull, and thus causes a power overshoot.

The results for PI 1 Cycle 9 and PI 2 Cycle 8 show transient MDNBRs of 1.586 and 1.913, respectively. The large variation in MDNBR between the two cycles is due to the variation in F_Q (dropped rod). The core average transient responses are very similar.

3.4.4 Cycle Specific Physics Calculations

These calculations are performed at the most limiting core conditions found during the cycle, e.g. the point in time that gives the most conservative value of the parameter in question. Sensitivity studies are conducted to determine these limiting conditions accounting for the effects of control rods, xenon, power level, temperature, etc.

a. Moderator Temperature Coefficient (α_M)

Calculations of α_M are performed in accordance with the general procedure described in section 2.0.

b. **Doppler Temperature Coefficient (α_D)**

Calculations of α_D are performed in accordance with the general procedure described in section 2.0.

c. **Nuclear Heat Flux Hot Channel Factor (F_Q)**

The nuclear heat flux hot channel factor, F_Q, is calculated for all possible dropped rods, consistent with the procedure described in section 2.0. Each rod is dropped at full power.

d. **Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)**

The nuclear enthalpy rise hot channel factor, $F_{\Delta H}$, is calculated for all possible dropped rods, consistent with the procedure described in section 2.0. Each rod is dropped at full power, equilibrium xenon and with highly skewed power shapes (both positive and negative axial offsets). The $F_{\Delta H}$ is then conservatively adjusted for allowed quadrant tilt.

e. **Effective Delayed Neutron Fraction (β_{eff})**

The value of β_{eff} is calculated in accordance with the general procedure described in section 2.0. Cycle specific calculations are performed for each dropped rod condition.

f. **Prompt Neutron Lifetime (ℓ^*)**

The value of ℓ^* is calculated in accordance with the general procedure described in section 2.0. Cycle specific calculations are performed for each dropped rod condition.

g. **Dropped Rod Worth ($\Delta\rho_{DROP}$)**

Calculations of the dropped rod worth are performed with the nodal model in three dimensions. Cycle specific calculations are performed for each dropped rod. All calculations are performed with the control rods at the FPIL. The worth of the dropped rod includes considerations of equilibrium as well as highly skewed power shapes (both positive and negative axial offsets).

h. **Control Bank Worth ($\Delta\rho_{CONTROL}$)**

Calculations of the integral worth of Bank D (control bank) are performed with the nodal model in three dimensions. Cycle specific calculations are performed for those dropped rods and sets of conditions which have been calculated not to cause a reactor negative flux rate trip.

i. **Excore Tilt**

Calculations of the relative tilt as seen by the excore detectors are performed by correlating the core edge power densities based on the distance squared between the edge assemblies and the detector. A 2% allowed tilt (Technical Specifications) is included in the calculations.

3.4.5 Reload Safety Evaluations

Each of the physics parameters calculated are adjusted to include the model reliability factors, RF_i, and biases, B_i, (Reference 1). These adjusted values are used in both parts of the rod drop analysis.

3.4.5.1 Part 1 Flux Rate Trip Analysis

The adjusted physics parameters are calculated for all possible dropped rods, consistent with the procedure described in Section 2.0. These parameters are used to determine the spectrum of rods which will not trip the reactor when dropped. The physics parameters should be adjusted as follows.

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3.4.5.2 Part 2 Transient Consequences

The adjusted physics parameters are calculated, for those rods which do not cause a negative flux rate trip, consistent with the procedure described in section 2.0. These parameters are used to determine the transient response following a rod drop. The physics parameters should be adjusted as follows.

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Table 3.4- 1: PI 2 Cycle 8 - Dropped Rod Results

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Table 3.4- 2: PI 1 Cycle 9 - Dropped Rod Results

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Figure 3.4- 1: PI 2 Cycle 8 Dropped Rod - EOC

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Figure 3.4- 2: PI 2 Cycle 8 Dropped Rod - EOC

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Figure 3.4- 3: PI 2 Cycle 8 Dropped Rod - EOC

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Figure 3.4- 4: PI 2 Cycle 8 Dropped Rod - EOC

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Figure 3.4- 5: PI 2 Cycle 8 Dropped Rod - EOC

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Figure 3.4- 6: PI 1 Cycle 9 Dropped Rod - EOC

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Figure 3.4- 7: PI 1 Cycle 9 Dropped Rod - EOC

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Figure 3.4- 8: PI 1 Cycle 9 Dropped Rod - EOC

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Figure 3.4- 9: PI 1 Cycle 9 Dropped Rod - EOC

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Figure 3.4- 10: PI 1 Cycle 9 Dropped Rod - EOC

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Figure 3.4- 11: Functional Block Diagram - Automatic Rod Control System

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Figure 3.4- 12: PI 1 Cycle 9 Dropped Rod - EOC - Rod Controller Response

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Figure 3.4- 13: PI 1 Cycle 9 Dropped Rod - EOC - Rod Controller Response

3.5 *Uncontrolled Boron Dilution*

3.5.1 Definition of Accident

The accident considered here is the malfunction of the chemical and volume control system in such a manner as to deliver unborated water at the maximum possible flowrate to the reactor coolant system under full power conditions. It is assumed that with the reactor subcritical, a dilution accident is recognized and terminated by operator intervention before loss of shutdown margin (see Appendix G). With the reactor in automatic control, the power and temperature increase from boron dilution at power results in the insertion of the RCC assemblies and a decrease in shutdown margin. Rod insertion limit alarms would alert the operator to isolate the source of unborated water and initiate boration prior to the time that shutdown margin was lost. With the reactor in manual control, the power and temperature rise due to boron dilution would eventually result in an overtemperature ΔT reactor trip if the operator did not intervene. After such a trip, the operator is expected to isolate the unborated water source and initiate boration procedures.

3.5.2 Accident Analysis

The system transient response to an uncontrolled boron dilution is simulated using a detailed model of the plant which includes the core, reactor vessel, steam generators, pressurizer, and connecting piping. The model also includes a simulation of the charging and the letdown systems, pressurizer control systems, and the reactor protection systems. Reactivity effects due to the fuel and moderator feedbacks, coolant boron concentration, and control rod motion after trip are included in the analysis. This model provides the transient response of average core power, reactor coolant pressure, and coolant temperature at the core inlet which are applied as forcing functions to a thermal-hydraulic simulation of the hot channel. The hot channel model uses the W-3 or WRB-1 correlation to calculate the departure from nucleate boiling ratio in the hot channel.

The reactivity due to boron dilution is calculated by assuming the maximum possible charging flow and minimum reactor coolant volume and taking into account the effect of increasing boron worth as dilution continues. The core burnup and corresponding boron concentration are selected to yield the most limiting combination of moderator temperature coefficient, Doppler temperature coefficient and spatial power distribution. This is normally the BOC condition. The minimum shutdown margin allowed by the Technical Specifications is conservatively assumed to exist prior to the initiation of the transient. The maximum time delay is assumed to exist between the time the trip setpoint is reached and the rods begin to move into the core. The most reactive rod is assumed to remain in its fully withdrawn position after receipt of the trip signal.

The acceptance criteria for this accident are that pressures in the reactor coolant system and main steam system do not exceed 110% of the respective design pressures, and that fuel clad integrity is maintained by limiting the minimum DNBR greater than 1.3 (W-3) or 1.17 (WRB-1), whichever is applicable.

3.5.3 NSP Safety Analysis Experience

NSP has analyzed a chemical and volume control system malfunction resulting in a decrease in the boron concentration of the reactor coolant. The analysis was performed using the model described in Appendix A with input consistent with the FSAR (Reference 2). The results are compared to those presented in Section 14.1.4 of Reference 2. Sensitivity studies indicate that critical input parameters in an analysis of the boron dilution accident are the moderator temperature coefficient, the boron worth coefficient, and the parameters used in the overtemperature ΔT trip set point algorithm.

The NSSS and hot channel transient response calculated by the NSP model are shown in Figures 3.5-1 to 3.5-4. No corresponding transient results are given in Reference 2, however, reactor trip on overtemperature ΔT was stated to occur at 78 seconds. The trip time calculated using the NSP model was 77 seconds, also on overtemperature ΔT, with a 6.0 second delay.

From Figure 14.1-10 of Reference 2, the minimum DNBR corresponding to this rate of reactivity insertion is 1.365. The NSP multichannel 1/8 assembly hot channel model result is 1.633.

3.5.4 Cycle Specific Physics Calculations

These calculations are performed at the most limiting core conditions found during the cycle, e.g., the point in time and the power level that gives the most conservative result with respect to the acceptance criteria. Sensitivity studies are conducted to determine these limiting conditions accounting for the effects of control rods, xenon, power level, temperature, etc. for each parameter.

a. Moderator Temperature Coefficient (α_M)

Calculations of α_M are performed using the methods described in Section 2. Cycle specific calculations for this accident are made at unrodded full power.

b. Doppler Temperature Coefficient (α_D)

Calculations of α_D are performed in accordance with the procedure described in Section 2.

c. Boron Concentration Reactivity Coefficient (α_B)

Calculations of α_B are performed using methods described in Section 2. Cycle specific calculations for this accident are made at unrodded full power.

d. Shutdown Margin (SDM)

The shutdown margin will be calculated and compared with the limiting assumptions used in the safety analysis using the methods described in Section 2.0.

e. Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)

The maximum core $F_{\Delta H}$ is assumed to remain within the current limits as defined in the Technical Specifications for allowable combinations of axial offset and power level. For Prairie Island, the continuous surveillance of the power distribution is accomplished with the excore detectors using the Exxon PDC-IIa[3] scheme.

3.5.5 Reload Safety Evaluations

All the cycle specific parameters discussed above are adjusted to include model reliability factors, RF_i, and biases, B_i. These results are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable if the following inequalities are met:

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Figure 3.5- 1: Chemical and Volume Control System Malfunction

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Figure 3.5- 2: Chemical and Volume Control System Malfunction

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Figure 3.5- 3: Chemical and Volume Control System Malfunction

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Figure 3.5- 4: Chemical and Volume Control System Malfunction

3.6 *Startup of an Inactive Coolant Loop*

3.6.1 Definition of Accident

Since there are no isolation valves or check valves in the Prairie Island reactor coolant system, operation of the plant with an inactive loop causes reversed flow through that loop. If there is a thermal load on the steam generator in the inactive loop, the hot leg coolant in that loop will be at a lower temperature than the core inlet temperature. The startup of the pump in the idle loop results in a core flow increase and the injection of cold water into the core, followed by a rapid reactivity and power increase. The resulting increase in fuel temperature limits the power rise due to Doppler feedback. Above 10% rated power, however, the reactor protection system prevents operation with an inactive loop, and consequently the temperature differential in an inactive loop would be small enough to minimize the accident consequences. Furthermore, the Prairie Island Technical Specifications do not permit operation with a reactor coolant pump out of service except during low power physics testing.

3.6.2 Accident Analysis

The system transient responses to an inactive loop startup is simulated using a detailed model which includes the core, reactor vessel, steam generators, main steam and reactor coolant piping, and the plant control and protection systems. This model calculates the time-dependent behavior of the average core power, coolant pressure, and core inlet flow and temperature which are supplied as forcing functions to a model of the hot channel for calculation of DNBR.

The accident is analyzed using the most negative moderator temperature coefficient calculated to occur during the cycle. This is normally the EOC condition. No credit is taken for reactivity reduction caused by reactor trip.

The reactor is initially assumed to be operating at 12% of rated power with reverse flow through the inactive loop. This includes a 2% uncertainty for calibration error above the 10% power setpoint for single loop operation in the reactor protection system. The assumption of this high initial power level is conservative, since it maximizes the temperature difference between the hot leg and cold leg in the inactive loop. The most adverse combination of initial coolant pressure and core inlet temperature is chosen to minimize the margin to core DNB limits.

The acceptance criteria for this accident are that the maximum pressures in the reactor coolant and main steam systems do not exceed 110% of design values and that cladding integrity be maintained limiting the minimum DNB ratio greater than 1.3 (W-3) or 1.17 (WRB-1), whichever is applicable.

3.6.3 NSP Safety Analysis Experience

NSP has analyzed the inactive loop start-up accident using the models and methods described in Appendix A. The results obtained are compared to the results presented in Section 14.1-5 of Reference 2.

The flow in both loops was linearly ramped to the nominal value in 10 seconds as stated in Reference 2.

Figures 3.6-1 to 3.6-5 provide a comparison of NSSS transient response to Figures 14.1-14, 14.1-15(a) and 14.1-15(b) of Reference 2. The results of the NSP model compare well with those of Reference 2.

The same transient was run using a dynamic flow simulation of the RCS pumps. The NSP model shows that the inactive pump will reach full speed in approximately 22 seconds as compared to 20 seconds as stated in Section 14.1.5 of Reference 2.

3.6.4 Cycle Specific Physics Calculations

These calculations are performed at the most limiting core conditions found during the cycle, e.g., the point in time and the power level that gives the most conservative result with respect to the acceptance criteria. Sensitivity studies are conducted to determine these limiting conditions accounting for the effects of control rods, xenon, power level, temperature, etc. for each parameter.

a. Moderator Temperature Coefficient (α_M)

Calculations of α_M are performed in accordance with the general procedures described in Section 2. Specific calculations for this accident are performed for hot zero power, rodded, no xenon conditions. The model bias, B_m is included in the calculations.

b. Doppler Temperature Coefficient (α_D)

Calculations of α_D are performed in accordance with the general procedures described in Section 2.

c. **Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)**

The maximum core $F_{\Delta H}$ is assumed to remain within the current limits as defined in the Technical Specifications for allowable combinations of axial offset and power level. For Prairie Island the continuous surveillance of the power distribution is accomplished with the excore detectors using the Exxon PDC-IIa[3] scheme.

3.6.5 Reload Safety Evaluations

Each of the physics parameters calculated above are conservatively adjusted to include the model reliability factors, RF_i, and biases, B_i. These adjusted values are the cycle specific parameters which are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable if the following inequalities are met:

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Figure 3.6- 1: Start-up of an Inactive Coolant Loop

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Figure 3.6- 2: Start-up of an Inactive Coolant Loop

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Figure 3.6- 3: Start-up of an Inactive Coolant Loop

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Figure 3.6- 4: Start-up of an Inactive Coolant Loop

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Figure 3.6- 5: Start-up of an Inactive Coolant Loop

3.7 *Feedwater System Malfunction*

3.7.1 Definition of Accident

Two classes of accidents are to be considered under this classification: Those that result in a decrease in feedwater temperature and those that result in an increase in feedwater flow. Either condition will result in an increased heat transfer rate in the steam generators causing a decrease in the reactor coolant temperature and an increased core power level due to negative reactivity coefficients and/or control system action. For the case of a decrease in feedwater temperature, the worst accident which may be postulated involves opening the bypass valve which diverts flow around the feedwater heaters. For the case of an increase in feedwater flow rate, the worst accident which may be postulated involves the full opening of a feedwater control valve. For this case, sustained high feedwater flow rate would ultimately result in a reactor trip due to high steam generator water level.

3.7.2 Accident Analysis

The feedwater system malfunction transient is analyzed using a dynamic simulation which includes core kinetics and heat transfer, reactor vessel and coolant piping, steam generators, pressurizer, and control systems. Pertinent variables obtained from the NSSS simulation are then applied as forcing functions to a separate thermal-hydraulic model of the core hot channel which calculates DNBR.

Two cases are analyzed. The first case is for a reactor without automatic control and with the least negative moderator coefficient. This is normally the BOC condition. This represents the situation where the reactor has the least inherent transient response capability. In this case, the core power slowly increases due to Doppler reactivity effects until the core power level again matches the load demand and a new steady state is achieved. The reactor does not trip. The core power increase is caused by a coolant temperature decrease which has the effect of increasing the margin to DNB.

The second case analyzed assumes that the reactor automatic control system responds to the decreasing coolant temperature and matches reactor power to load demand. A conservatively large (in absolute value) negative moderator temperature coefficient is assumed to exist. This is normally the EOC condition. This case results in a somewhat higher final core power level than the uncontrolled case without moderator feedback; this in turn results in a net decrease in DNBR but the decreased coolant temperature again maintains a significant margin above the 1.3 limit.

The core neutronic characteristics which exert a significant influence on the calculated results of this transient are the Doppler and moderator reactivity coefficients. The most negative Doppler temperature coefficient calculated to occur during the cycle is used in the analysis to maximize the power increase. For such slow rates of reactivity addition as are encountered, the transient response is insensitive to the value of ℓ^*, the thermal neutron lifetime. Trip reactivity insertion characteristics are not relevant, since the reactor does not trip.

The acceptance criteria for the feedwater system malfunction transient are that cladding integrity be maintained by limiting the minimum DNBR to be greater than 1.3 (W-3) or 1.17 (WRB-1), whichever is applicable, and that maximum pressure in the reactor coolant and main steam system not exceed 110% of the design pressure.

3.7.3 NSP Safety Analysis Experience

Not all classes of feedwater system malfunction transient have been analyzed. Specifically, the transient that was analyzed was the opening of the feedwater heater bypass valve. The NSP safety analysis experience in this area is represented by the analysis of a decrease in feedwater temperature transient using the model described in Appendix A. This calculation has been performed using input consistent with the Prairie Island FSAR (Reference 2).

The models used correspond to BOC conditions without control and EOC conditions with control. Figures 14.1-16 and 14.1-17, 14.1-18 and 14.1-19 of Reference 2 were used to obtain forcing functions of the transient feedwater-steam flow, and feedwater enthalpy for the two cases respectively.

The response of the NSSS for the BOC case is compared to Figures 14.1-16 and 14.1-17 of Reference 2 in Figures 3.7-1 to 3.7-4. The hot channel transient DNBR was computed using the NSP 1/8 assembly multichannel model and is compared to Figure 14.1-16 of Reference 2 in Figure 3.7-5 for the BOC case.

The EOC comparisons to Figures 14.1-18 and 14.1-19 of Reference 2 are shown in Figures 3.7-6 through 3.7-10.

The NSP models predict the same trends throughout the transient as the FSAR results.

3.7.4 Cycle Specific Physics Calculations

These calculations are performed at the most limiting core conditions found during the cycle, e.g., the point in time and the power level that gives the most conservative result with respect to the acceptance criteria. Sensitivity studies are conducted to determine these limiting conditions accounting for the effects of control rods, xenon, power level, temperature, etc., for each parameter.

a. **Moderator Temperature Coefficient (α_M)**

Calculations of α_M are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations are performed to determine the least negative α_M at full power conditions and the most negative α_M under all operating conditions. The model bias is included in these calculations.

b. **Doppler Temperature Coefficient (α_D)**

Calculations of α_D are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations for this accident are performed as a function of power level over the full operating range from 0-100% power.

c. **Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)**

The maximum core $F_{\Delta H}$ is assumed to remain within the current limits as defined in the Technical Specifications for allowable combinations of axial offset and power level. For Prairie Island, the continuous surveillance of the power distribution is accomplished with the excore detectors using the Exxon PDC-IIa[3] scheme.

3.7.5 Reload Safety Evaluations

Each of the physics parameters calculated above are adjusted to include the model reliability factors RF_i and biases, B_i. These adjusted values are the cycle specific parameters which are then compared to the bounding values assumed in the same analysis. The cycle specific parameters are acceptable with regard to feedwater malfunction transients if the following inequalities are met:

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Figure 3.7- 1: Decrease in Feed Water Temperature - BOC Without Reactor Control

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Figure 3.7- 2: Decrease in Feed Water Temperature - BOC Without Reactor Control

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Figure 3.7- 3: Decrease in Feed Water Temperature - BOC Without Reactor Control

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Figure 3.7- 4: Decrease in Feed Water Temperature - BOC Without Reactor Control

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Figure 3.7- 5: Decrease in Feed Water Temperature - BOC Without Reactor Control

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Figure 3.7- 6: Decrease in Feed Water Temperature - EOC With Reactor Control

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Figure 3.7- 7: Decrease in Feed Water Temperature - EOC With Reactor Control

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Figure 3.7- 8: Decrease in Feed Water Temperature - EOC With Reactor Control

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Figure 3.7- 9: Decrease in Feed Water Temperature - EOC With Reactor Control

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Figure 3.7- 10: Decrease in Feed Water Temperature - EOC With Reactor Control

3.8 Excessive Load Increase

3.8.1 Definition of Accident

An excessive load increase accident is defined as a rapid increase in steam generator steam flow that causes a power mismatch between core heat generation and secondary side load demand. The ensuing decrease in reactor coolant temperature results in a core power increase due to Doppler and moderator feedback and/or control system action. Only steam flow increases within the capability of the turbine control valves are considered here; larger flow increases are considered in connection with main steam line rupture accidents (Section 3.14).

3.8.2 Accident Analysis

The analysis of the excessive load increase transient is analyzed using a dynamic simulation which includes the reactor core, reactor vessel, steam generators, pressurizer and connecting piping. The main steam and feedwater systems and control and protection systems are also modeled. The departure from nucleate boiling ratio is computed using a separate model of the hot channel thermal-hydraulic behavior and the appropriate CHF correlation. This model is coupled to the NSSS simulation which supplies core power and coolant temperature and pressure as a function of time.

The transient is initiated by imposing a rapid increase in steam flow to 120% of rated full power flow. Initial pressurizer pressure, reactor coolant temperature, and core power are assumed at their extreme steady state values to minimize the calculated margin to DNB. Typically, four cases are analyzed: moderator reactivity coefficient at minimum and maximum values; with and without automatic reactor control.

For the cases without control, the case with the least negative moderator coefficient shows a large coolant temperature decrease relative to the power increase, the net effect is to increase the DNBR. The case with a large negative moderator coefficient shows a larger increase in power and a decrease in DNBR. The cases with reactor control show similar behavior but here the control system acts to maintain average coolant temperature by increasing power, so the DNBR decreases in both cases. However, all cases presented in the Prairie Island FSAR (Reference 2) exhibit a large margin to the 1.3 limit.

Reactor trip does not occur during any of the transients considered, consequently scram reactivity insertion characteristics are not factors in the evaluation of this accident. Moderator and Doppler reactivity coefficients are the most significant kinetics parameters. The most negative Doppler temperature coefficient is assumed to provide the most conservative evaluation, since it maximizes the core power increase, as does the most negative moderator coefficient. The acceptance criteria for this accident are that the fuel cladding integrity be maintained by limiting the minimum DNBR to not less than 1.3 (W-3) or 1.17 (WRB-1), whichever is applicable, and reactor coolant and main steam system maximum pressures not greater than 110% of the design pressures.

3.8.3 NSP Safety Analysis Experience

No specific excessive load increase transient has been analyzed. The NSP safety analysis experience in this area is represented by the analysis of an increase in heat removal (cooldown) by the secondary system transient, specifically, a decrease in feedwater temperature. The results of this analysis have been presented in the preceding section (3.7.3). In addition, calculations have been performed for the steam line break accident which represents the most limiting case for accidents in this category. Results of these calculations are presented in Section 3.14.

3.8.4 Cycle Specific Physics Calculations

These calculations are performed at the most limiting core conditions found during the cycle, e.g., the point in time and the power level that gives the most conservative result with respect to the acceptance criteria. Sensitivity studies are conducted to determine these limiting conditions accounting for the effects of control rods, xenon, power level, temperature etc. for each parameter.

a. Moderator Temperature Coefficient (α_M)

Calculations of α_M are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations are performed to determine the minimum values of moderator coefficient at full power equilibrium xenon conditions.

b. Doppler Temperature Coefficient (α_D)

Calculations of α_D are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations for this accident are performed at the full power equilibrium xenon conditions.

c. **Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)**

The maximum core $F_{\Delta H}$ is assumed to remain within the current limits as defined in the Technical Specifications for allowable combinations of axial offset and power level. For Prairie Island, the continuous surveillance of the power distribution is accomplished with the excore detectors using the Exxon PDC-IIa[3] scheme.

3.8.5 Reload Safety Evaluations

Each of the physics parameters calculated above are adjusted to include the model reliability factors, RF_i, and biases, B_i. These adjusted values are the cycle specific parameters which are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable with regard to excessive load increase transients if the following inequalities are met:

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3.9 *Loss of External Load*

3.9.1 Definition of Accident

The most likely source of a complete loss of load is a turbine-generator trip. Above approximately 10% power, a turbine trip generates a direct reactor trip which is signaled from either of two diverse inputs: release of autostop oil or stop valve closure. If credit is taken for the steam bypass system and pressurizer control system, there is no significant increase in reactor coolant temperature or pressure. To provide a conservative assessment of the accident, however, no credit is taken for direct reactor trip, steam bypass actuation, or pressurizer pressure control. Under these assumptions, both secondary and primary pressures increase rapidly and a reactor trip is generated by the high pressurizer pressure signal. This accident is primarily of concern from the standpoint of demonstrating the adequacy of overpressurization protection, since the hot channel DNBR increases (or decreases only slightly) during the accident.

3.9.2 Accident Analysis

The loss of external load accident is analyzed using a detailed digital model of the nuclear steam supply system and associated control and protected systems. Core kinetics, heat transfer, reactor coolant and steam generator secondary side temperature and pressures, steam feedwater flow rates, and pressurizer liquid level are some of the variables computed by the model.

Four transients are analyzed pertaining to a loss of load accident: moderator reactivity coefficient at minimum and maximum values, with and without automatic reactor control.

Typically the minimum and maximum moderator reactivity coefficients occur at BOC and EOC conditions respectively. The following reactor conditions are assumed: No credit is taken for the direct reactor trip caused by the turbine trip or for the steam bypass system. The secondary side pressure rises to the safety valve setpoint and is limited to that pressure by steam relief through the safety valves. Scram on high pressurizer pressure mitigates the consequences of this accident and prevents water relief through the pressurizer relief and safety valves.

The worst case with respect to overpressurization assumes no control rod motion prior to reactor trip and no credit for pressurizer relief or spray valves. In this case, the moderator reactivity coefficient is assumed to be the least negative value occurring at BOC.

The acceptance criteria for this accident are that the maximum pressures in the reactor coolant and main steam systems do not exceed 110% of design values and that cladding integrity be maintained limiting the minimum DNB ratio greater than 1.3 (W-3) or 1.17 (WRB-1), whichever is applicable.

3.9.3 NSP Safety Analysis Experience

NSP has analyzed the loss of load accident using input consistent with the FSAR. The models described in Appendix A were used to analyze the loss of external load transient for four cases:

A. BOC conditions with reactor control.
B. EOC conditions with reactor control.
C. BOC conditions without reactor control.
D. EOC conditions without reactor control.

These transients are simulated by closing the turbine stop valves rapidly. The anticipated reactor trip on stop valve closure is disabled and trip occurs on high pressurizer pressure.

The transient hot channel DNBR calculations are done using the NSP multichannel 1/8 assembly model. The results of the case A calculations are compared to Figures 14.1-38 and 14.1-39 of Reference 2 in Figures 3.9-1 to 3.9-6.

The results of the case B calculations are compared to Figures 14.1-40 and 14.1-41 of Reference 2 in Figures 3.9-7 to 3.9-12.

These two cases show significant deviations from the FSAR results due to the use of a more realistic representation of the pressurizer relief valves in the NSP model. This model includes simulation of the valve opening and closing time delay and stroke times.

The results of case C are compared to Figures 14.1-42 and 14.1-43 of Reference 2 in Figures 3.9-13 to 3.9-18.

The results of case D are compared to Figures 14.1-44 and 14.1-45 of Reference 2 in Figures 3.9-19 to 3.9-24.

3.9.4 Cycle Specific Physics Calculations

These calculations are performed at the most limiting core conditions found during the cycle, e.g., the point in time and the power level that gives the most conservative result with respect to the acceptance criteria. Sensitivity studies are conducted to determine these limiting conditions accounting for the effect of control rods, xenon, power level, temperature, etc. for each parameter.

a. **Moderator Temperature Coefficient (α_M)**

Calculations of α_M are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations are performed to determine the least negative value of the moderator coefficient at the full power condition. The model bias, B_M, is included in the calculations.

b. **Doppler Temperature Coefficient (α_D)**

Calculations of α_D are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations for this accident are performed at the full power equilibrium xenon condition.

c. **Scram Reactivity Curve ($\Delta\rho_{SCRAM}(t)$)**

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations for this accident are performed for the full power condition.

d. **Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)**

The maximum core $F_{\Delta H}$ is assumed to remain within the current limits as defined in the Technical Specifications for allowable combinations of axial offset and power level. For Prairie Island, the continuous surveillance of the power distribution is accomplished with the excore detectors using the Exxon PDC-IIa[3] scheme.

3.9.5 Reload Safety Evaluations

Each of the physics parameters calculated above are adjusted to include the model reliability factors, RF_i, and biases, B_i. These adjusted values are the cycle specific parameters which are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable with regard to transients if the following inequalities are met:

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The integral of the bounding value of the scram curve, $\Delta\rho_{SCRAM}(t)$, is taken as that rod worth required to produce the shutdown margin assumed in the safety analysis for the most limiting cycle specific core conditions.

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Figure 3.9- 1: Loss of External Load - BOC With Reactor Control

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Figure 3.9- 2: Loss of External Load - BOC With Reactor Control

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Figure 3.9- 3: Loss of External Load - BOC With Reactor Control

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Figure 3.9- 4 : Loss of External Load - BOC With Reactor Control

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Figure 3.9- 5: Loss of External Load - BOC With Reactor Control

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Figure 3.9- 6: Loss of External Load - BOC With Reactor Control

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Figure 3.9- 7: Loss of External Load - EOC With Reactor Control

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Figure 3.9- 8: Loss of External Load - EOC With Reactor Control

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Figure 3.9- 9: Loss of External Load - EOC With Reactor Control

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Figure 3.9- 10: Loss of External Load - EOC With Reactor Control

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Figure 3.9- 11: Loss of External Load - EOC With Reactor Control

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Figure 3.9- 12: Loss of External Load - EOC With Reactor Control

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Figure 3.9- 13: Loss of External Load - BOC Without Reactor Control

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Figure 3.9- 14: Loss of External Load - BOC Without Reactor Control

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Figure 3.9- 15: Loss of External Load - BOC Without Reactor Control

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Figure 3.9- 16: Loss of External Load - BOC Without Reactor Control

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Figure 3.9- 17: Loss of External Load - BOC Without Reactor Control

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Figure 3.9- 18: Loss of External Load - BOC Without Reactor Control

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Figure 3.9- 19: Loss of External Load - EOC Without Reactor Control

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Figure 3.9- 20: Loss of External Load - EOC Without Reactor Control

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Figure 3.9- 21: Loss of External Load - EOC Without Reactor Control

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Figure 3.9- 22: Loss of External Load - EOC Without Reactor Control

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Figure 3.9- 23: Loss of External Load - EOC Without Reactor Control

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Figure 3.9- 24: Loss of External Load - EOC Without Reactor Control

3.10 *Loss Of Normal Feedwater Flow*

3.10.1 Definition of Accident

This accident is defined as a complete loss of normal feedwater. Realistically, the plant's auxiliary feedwater pumps would be actuated and supply sufficient feedwater to both steam generators to dissipate residual and decay heat after reactor trip. To provide a margin of conservatism, however, only one of the two auxiliary feedwater pumps is assumed to deliver feedwater to one of the two steam generators. Under this assumption, the steam generator not receiving auxiliary feedwater suffers a degradation of heat transfer capability and the reactor coolant system temperature and pressure increase as a result of decay heat following reactor trip. Traditionally, an additional conservatism has been applied to the analysis of the loss of feedwater accident by assuming that the reactor coolant pumps are tripped and coastdown to natural circulation conditions, further degrading the heat transfer capability of both steam generators. When analyzed in this manner, the accident corresponds to a loss of non-emergency A.C. power.

3.10.2 Accident Analysis

The loss of normal feedwater accident is analyzed using a dynamic simulation model which includes the reactor and reactor coolant system and the secondary plant systems. The model includes a simulation of the natural circulation flow existing in the reactor coolant system subsequent to the assumed coastdown of the reactor coolant pumps. The model also includes the heat source due to the decay of fission products since the reactor trips on a low steam generator level signal early in the transient, and this decay heat constitutes the main energy source thereafter.

The results of the analysis of the loss of normal feedwater accident are not sensitive to the values of the core neutronics parameters. The reactor is tripped very early in the transient by decreasing steam generator levels. Since this occurs well before steam generator heat transfer capability has been reduced, the margin to DNB is not reduced significantly prior to reactor trip. The maximum reactor coolant temperature occurs approximately 2000 seconds after accident initiation and is not significantly affected by the core neutron power transient, since decay of fission products is the major energy source over most of this time interval. The decay heat is conservatively calculated by assuming that the fission products are initially in equilibrium at the existing core power level.

The acceptance criteria for this accident are that pressures in the reactor coolant and main steam systems not exceed 110% of design pressure and that the minimum DNBR occurring during the accident be not less than 1.3 (W-3) or 1.17 (WRB-1), whichever is applicable.

3.10.3 NSP Safety Analysis Experience

NSP has analyzed the loss of normal feedwater accident using input data consistent with the Prairie Island FSAR.

The NSSS transient response to a loss of normal feedwater accident was analyzed using the models described in Appendix A. The results are compared to the corresponding results reported in Section 14.1-10 of Reference 2. The accident is assumed to occur as a result of isolating both steam generators from their normal supply of feedwater. One steam generator receives flow from one auxiliary feedpump; the other SG dries out due to steam release through the safety valves. A trip of both reactor coolant pumps, postulated to occur simultaneously, results in a further degradation of heat transfer capability.

The results obtained from the NSP model are compared to Figures 14.1-46(a), (b), and (c) of Reference 2 in Figures 3.10-1 to 3.10-3.

The fact that the steam generator water level response is approximately the same in both analyses indicates that compatible initial shell side mass inventories and transient safety valve flow rates were used in both the NSP and the Reference 2 analyses.

A volume balance based on the thermal expansion of the RCS fluid indicates that the pressurizer volume surge is consistent with the RCS temperature calculated by the NSP model. In general, the results of the NSP model and those reported in Reference 2 show the same trends.

3.10.4 Cycle Specific Physics Calculations

The loss of normal feedwater transient is not sensitive to core physics parameters since the reactor is assumed to trip in the initial stages (i.e., 2 seconds) of the transient. This trip occurs due to a low-low steam generator level signal well before the heat transfer capability of the steam generator is reduced. The transient is then driven by the decay heat from the tripped reactor. Also, the loss of flow transient analyzed in Section 3.11 is considered a more severe transient of this type. Therefore no comparisons will be made for Reload Safety Evaluations.

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Figure 3.10- 1: Loss of Normal Feed Water

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Figure 3.10- 2: Loss of Normal Feed Water

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Figure 3.10- 3: Loss of Normal Feed Water

3.11 *Loss Of Reactor Coolant Flow - Pump Trip*

3.11.1 Definition of Accident

The accident considered here is the simultaneous loss of electrical power to both of the reactor coolant pumps. As a result of loss of driving head supplied by the pumps, the coolant flow through the core begins to decrease. This decrease in flow rate is retarded by the hydraulic inertia of the fluid itself and the flywheels on the pump motors. The reactor is tripped by any one of several diverse and redundant signals which monitor coolant flow conditions. This trip results in a power reduction before the thermal-hydraulic conditions in the core approach those which could result in damage to the fuel. Loss of power to one of the pumps with both pumps initially operating may also be considered, but the consequences are less severe than for the two pump trips. Seizure of the reactor coolant pump shaft is considered in Section 3.12.

3.11.2 Accident Analysis

The loss of forced reactor coolant flow accident is analyzed using a detailed model of the reactor coolant system thermal-hydraulics. The conservation of momentum and continuity equations for the coolant, coupled to a representation of the pump hydraulics and speed coastdown, are solved to compute the system flowrates as a function of time. Reactor core neutron kinetics and heat transfer equations are coupled to the flow coastdown equations in order to compute heat flux and coolant temperatures in the reactor. A simulation of the steam generators and pressurizer is also included in the model. A separate model analyzes the transient response of the core hot channel, using conditions supplied by the NSSS Model as input, and computes the departure from nucleate boiling ratio (DNBR).

The initial conditions for the accident analysis assume the most adverse combination of power, core inlet temperature, and pressurizer pressure including allowances for steady state error so that the initial margin to DNB is the minimum expected during steady state operation.

The power transient is analyzed using the least negative value of moderator reactivity coefficient calculated to occur during the cycle. For the sake of conservatism, a value of zero is used in the FSAR analysis even though the moderator coefficient is expected to remain negative for all normal operating conditions. The Loss of Flow-Pump Trip transient is relatively insensitive to the Doppler coefficient. For the sake of conservatism, the most negative expected Doppler coefficient is assumed.

The reactivity reduction due to control rod insertion after trip is calculated by assuming the most adverse delay time expected to occur between loss of power to the pump and the initiation of rod motion. Upon reactor trip, it is assumed that the most reactive RCC assembly is stuck in its fully withdrawn position, resulting in a minimum insertion of negative reactivity. The trip reactivity insertion dominates the power response and is the most important neutronics input parameter. The acceptance criteria for the loss of reactor coolant flow accident are that the minimum DNBR be not less than 1.3 (W-3) or 1.17 (WRB-1), whichever is applicable, and that the maximum reactor coolant and main steam system pressure not exceed 110% of their design values.

3.11.3 NSP Safety Analysis Experience

NSP has analyzed the loss of reactor coolant flow accident using input consistent with the FSAR (Reference 2).

The models described in Appendix A were used to analyze the following transient cases:

- Loss of power to one reactor coolant pump with two pumps initially running (1/2 pump trip)
- Loss of power to two reactor coolant pumps with two pumps initially running (2/2 pump trip)

The results of these two cases are compared to the corresponding results of Section 14.1-8 of Reference 2.

Figure 3.11-1 provides a comparison of the transient core flow fraction for the 2/2 pump trip case to Figure 14.1-29 of Reference 2. In Figure 3.11-5, the transient core flow response predicated from the NSP model is compared to the corresponding results from Figure 14.1-26 of Reference 2 for the 1/2 pump trip case.

Figures 3.11-2 and 3.11-3 compare the nuclear power and core average heat flux for the loss of reactor coolant flow accident to Figure 14.1-30 of Reference 2.

Figures 3.11-6 and 3.11-7 compare the transient neutron flux and core average heat flux for the loss of reactor coolant flow accident to Figure 14.1-27 of Reference 2.

Figures 3.11-4 and 3.11-8 compare the transient DNBR computed using the NSP models to Figures 14.1-31 and 14.1-28 of Reference 2 respectively.

DNBR analyses were computed using a single closed channel model, the multichannel 1/8 assembly model, and the 1/8 core model. It is believed that the single channel model will provide a better comparison to the FSAR, however, the 1/8 core model is more accurate and will be used for licensing analyses.

3.11.4 Cycle Specific Physics Calculations

These calculations are performed at the most limiting core conditions found during the cycle, e.g., the point in time and the power level that gives the most conservative result with respect to the acceptance criteria. Sensitivity studies are conducted to determine these limiting conditions accounting for the effects of control rods, xenon, power level, temperature, etc. for each parameter.

a. **Moderate Temperature Coefficient (α_M)**

Calculations of α_M are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations are performed to determine the least negative value of the moderator coefficient at the full power condition. The model bias, B_M, is included in the calculations.

b. **Doppler Temperature Coefficient (α_D)**

Calculations of α_D are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations for this accident are performed to determine the most negative value at full power conditions.

c. **Scram Reactivity Curve ($\Delta\rho_{SCRAM}(t)$)**

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations for this accident are performed for the full power condition.

d. **Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)**

The maximum core $F_{\Delta H}$ is assumed to remain within the current limits as defined in the Technical Specifications for allowable combinations of axial offset and power level. For Prairie Island the continuous surveillance of the power distribution is accomplished with the excore detectors using the Exxon PDC-IIa[3] scheme.

e. **Shutdown Margin (SDM)**

The shutdown margin will be calculated and compared with the limiting assumptions used in the safety analysis using the methods described in Section 2.0.

3.11.5 Reload Safety Evaluations

Each of the physics parameters calculated above are adjusted to include the model reliability factors, RF_i, and biases, B_i. These adjusted values are the cycle specific parameters which are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable with regard to loss of reactor coolant flow pump trip transients if the following inequalities are met:

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The integral of the bounding value of the scram curve $\Delta\rho_{SCRAM}(t)$, is taken as that rod worth required to produce the shutdown margin assumed in the safety analysis for the most limiting cycle specific core conditions.

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Figure 3.11- 1: Loss of RCS Flow - 2/2 Pump Trip

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Figure 3.11- 2: Loss of RCS Flow - 2/2 Pump Trip

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Figure 3.11- 3: Loss of RCS Flow - 2/2 Pump Trip

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Figure 3.11- 4: Loss of RCS Flow - 2/2 Pump Trip

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Figure 3.11- 5: Loss of RCS Flow - ½ Pump Trip

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Figure 3.11- 6: Loss of RCS Flow - ½ Pump Trip

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Figure 3.11- 7: Loss of RCS Flow - ½ Pump Trip

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Figure 3.11- 8: Loss of RCS Flow - ½ Pump Trip

3.12 Loss Of Reactor Coolant Flow - Locked Rotor

3.12.1 Definition of Accident

The accident postulated is the instantaneous seizure of the rotor of a single reactor coolant pump. Flow through the affected loop is rapidly reduced, leading to a reactor trip initiation due to low flow. The sudden decrease in core flow while the reactor is at power results in a degradation of core heat transfer and departure from nucleate boiling in some of the fuel rods.

3.12.2 Accident Analysis

The analysis of the locked reactor coolant pump rotor is performed using a detailed model of the reactor coolant system thermal-hydraulics. The conservation of momentum and continuity equations for the coolant, coupled to a representation of the pump hydraulic characteristics, are solved to compute the system flowrates as a function of time. Reactor core neutron kinetics and transient heat transfer equations are coupled to the flow equations in order to compute the core heat flux and coolant temperatures in the reactor. A simulation of the pressurizer and steam generators is also included in the model. Separate models compute the thermal-hydraulic response of the coolant hot channel and fuel hot spot using conditions supplied by the NSSS model as input. These models compute heat flux, fuel and clad temperatures, and DNBR for a conservative evaluation of the extent of fuel damage which could occur during a locked rotor accident.

The initial conditions for the accident analysis assume the most adverse combination of power, core inlet temperature, and pressurizer pressure including allowances for steady state error so that the initial margin to DNB is the minimum expected during steady state operation. For purposes of evaluating the reactor coolant system pressure transient, the initial pressure is assumed as the maximum expected during normal operation including allowances for instrumentation error and controller tolerances.

The power transient is analyzed using the least negative value of moderator reactivity coefficient calculated to occur during the cycle. For the sake of conservatism, a value of zero is assumed in the FSAR analysis even though only negative values are expected at normal operating conditions. The most negative Doppler reactivity coefficient is used in the analysis since this results in maximum hot spot temperatures. Trip reactivity insertion characteristics are calculated by assuming the maximum time delay between a low flow signal and control rod motion. It is further assumed that the most reactive RCC assembly is stuck in a fully withdrawn position.

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 2750°F.
3. The number of fuel rods calculated to experience a DNBR of less than 1.3 (W-3) or 1.17 (WRB-1), whichever is applicable, should not exceed the number which are required to fail in order that the doses due to released activity will exceed the limits of 10CFR 100. This limit is currently the maximum number of failed fuel rods calculated in the FSAR (Reference 2).

3.12.3 NSP Safety Analysis Experience

NSP has analyzed the locked rotor accident for the Prairie Island Plant, using input consistent with the FSAR.

The models described in Appendix A were used to analyze the locked rotor accident assuming the condition of a locked rotor in one coolant loop with two pumps initially running. The results of this analysis are compared to the corresponding results of Section 14.1-8 of Reference 2.

Figures 3.12-1 and 3.12-2 compare the transient response of the core flow rate and RCS pressure to Figures 14.1-32 and 14.1-34 of Reference 2 respectively for the locked rotor case. The NSP model predicts that the pressurizer safety valve setpoint of 2500 psia is never reached with a maximum pressurizer pressure of 2474 at 4.3 seconds. The Reference 2 results show a maximum reactor coolant system pressure of 2737 psia at 2.5 seconds.

In Figures 3.12-3, 4 and 5, the transient DNBR during the locked rotor accident are compared to Figure 14.1-35 of Reference 2 for peaking factors, $F_{\Delta H}$, of 1.58, 1.456 and 1.411 respectively, using the NSP hot channel models. The analyses show the sensitivity of the MDNBR to the initial $F_{\Delta H}$. The NSP models predict greater decreases in MDNBR than do the FSAR calculations due to the significant difference in predicted pressurizer pressure response. The NSP model predicts that 17.45% of the rods will reach a DNBR lower than 1.3 during the locked rotor accident compared to 20% predicted in Section 14.1-8 of Reference 2.

Figure 3.12-6 compares the transient clad temperature response at the hot spot for the locked rotor accident to the corresponding results of Figure 14.1-37 of Reference 2. The NSP model predicts a maximum of 1671°F and the Reference 2 analysis a maximum of 1680°F.

A sensitivity study was performed on the hot spot cladding temperature response with respect to the surface heat transfer coefficient. Typically, a 25 Btu/hr-ft^2°F change in the heat transfer coefficient produced a 50°F change in peak cladding temperature.

3.12.4 Cycle Specific Physics Calculations

These calculations are performed at the most limiting core conditions found during the cycle e.g., the point in time and the power level that gives the most conservative result with respect to the acceptance criteria. Sensitivity studies are conducted to determine these limiting conditions accounting for the effects of control rods, xenon, power level, temperature, etc. for each parameter.

a. Moderator Temperature Coefficient (α_M)

Calculations of α_M are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations are performed to determine the least negative value at the full power condition. The model bias, B_M, is included in the calculations.

b. Doppler Temperature Coefficient (α_D)

Calculations of α_D are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations for this accident are performed to determine the most negative value at full power conditions.

c. Scram Reactivity Curve ($\Delta\rho_{scram}(t)$)

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations are performed to determine the integral rod worth as a function of core height during the trip.

d. Effective Delayed Neutron Fraction (β_{eff})

Calculations of β_{eff} are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations are performed at full power conditions.

e. Fuel Pin Census

Calculation of the number of fuel pins (pin census) versus $F_{\Delta H}$ is performed in accordance with the general procedures described in Section 2.0. The calculations determine the number of fuel pins above the limiting value of $F_{\Delta H}$ above which the DNBR equals 1.3 (W-3) or 1.17 (WRB-1), whichever is applicable.

f. Shutdown Margin (SDM)

The shutdown margin will be calculated and compared with the limiting assumptions used in the safety analysis using the methods described in Section 2.0.

3.12.5 Reload Safety Evaluations

Each of the physics parameters calculated are adjusted to include the model reliability factors, RF_i, and biases, B_i. These adjusted values are then compared to the bounding values assumed in the safety analysis. The cycle specific parameters are acceptable with regard to the locked rotor accident if the following inequalities are met:

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Figure 3.12- 1: Loss of RCS Flow - Locked Rotor

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Figure 3.12- 2: Loss of RCS Flow - Locked Rotor

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Figure 3.12- 3: Loss of RCS Flow - Locked Rotor

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Figure 3.12- 4: Loss of RCS Flow - Locked Rotor

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Figure 3.12- 5: Loss of RCS Flow - Locked Rotor

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Figure 3.12- 6: Loss of RCS Flow - Locked Rotor

3.13 Fuel Handling Accident

3.13.1 Definition of Accident

The accident considered is the sudden release of the gaseous fission products held in the voids between the pellets and cladding of one fuel assembly. The activities associated with this accident would be released either inside the Containment Building or the Auxiliary Building. A high radiation level on the Auxiliary building vent monitor would automatically activate the special ventilation system with subsequent absolute and charcoal filtration. In calculating the offsite exposure from the accident, however, it is assumed that the activity is discharged to the atmosphere at ground level from the Auxiliary Building since this maximizes the offsite doses.

3.13.2 Accident Analysis

The gap activity is calculated based on fission gas buildup in the fuel and subsequent diffusion to the fuel rod gap at rates dependent upon the operating temperature. The calculation assumes that the assembly with the maximum gap activity is the one which is damaged. Only that fraction of fission gases which has diffused into the gap and plenum regions of the fuel pin would be available for immediate release. This fraction is calculated based on a conservative evaluation of the temperature and power distribution in the highest powered assembly in the last six weeks prior to shutdown. This activity is further reduced by decay during the 100 hours elapsing after shutdown before removal of the vessel head.

The activity present in the fuel rod gaps consists predominately of halogens and noble gases. Decontamination factors are applied to account for halogen depletion by the pool water; all the noble gas inventory is assumed to escape from the pool water surface. Dispersal of the activity escaping the Auxiliary Building is calculated using the Gaussian plume dispersion formula, taking credit for building wake dilution. Using conservative radiological formulae, the activity concentrations at the site exclusion boundary are converted to integrated whole body and thyroid doses. These doses are then compared to the acceptance criteria set forth in 10CFR 100.

3.13.3 NSP Safety Analysis Experience

NSP has not analyzed this accident. The Westinghouse and Exxon analysis is reviewed for each reload to determine its applicability to the current core design.

3.13.4 Cycle Specific Physics Calculations

The maximum hot channel factor, F_Q, for all levels is calculated for equilibrium hot full power conditions with the rods at the full power insertion limits. This value is determined for core exposure 1.5 GWD/MU before EOC. Calculations of F_Q are performed in accordance with the procedure described in Section 2.7.

3.13.5 Reload Safety Evaluations

The F_Q calculated above is conservatively adjusted to include the reliability factor, RF_{FQ}, and bias, B_{FQ}. This value is then compared to the value assumed in the accident analysis. The comparison is acceptable if the following inequality is satisfied:

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3.14 Main Steam Line Break

3.14.1 Definition of Accident

The accident considered here is the complete severance of a pipe inside containment at the exit of the steam generator with the plant initially at no load conditions and both reactor coolant pumps running. The resulting uncontrolled steam release causes a rapid reduction in reactor coolant temperature and pressure as the secondary side is depressurized. If the most reactive RCC assembly is assumed stuck in its fully withdrawn position, there is a possibility that the core will become critical and return to power due to the negative moderator coefficient. A return to power is potentially a problem mainly because of the high hot channel factors which exist with a stuck RCC assembly. The core is ultimately restored to a subcritical condition by boric acid injection via the Emergency Core Cooling System. The zero power case is considered because the stored energy of the system is at a minimum and steam generator secondary inventory is at a maximum under these conditions, thus increasing the severity of the transient.

3.14.2 Accident Analysis

The analysis of the steam line rupture accident is performed using a detailed, multi-loop model of the core, reactor coolant system and pressurizer, steam generators, and main steam supply system. The steam flow through the severed steam line is calculated using a critical flow model. Conservation equations for the steam generator shell side mass and energy inventory are solved to predict the temperatures and pressures existing throughout the transient. Heat transfer from the reactor coolant system to the steam generators is calculated based on instantaneous fluid conditions and empirical correlations. The analytical model includes a representation of the reactor vessel upper head volume in order to predict the transient response of the reactor coolant pressure subsequent to draining the pressurizer. A simulation of the safety injection system and boron spatial kinetics allows calculation of the core coolant boron concentration and its influence on core neutron kinetics. The representation of core moderator density reactivity effects must include allowances for the large change in density which the coolant undergoes as the system temperature falls. A detailed thermal-hydraulic model of the hot channel is coupled to the system simulation and provides a calculation of the departure from nucleate boiling ratio during the transient.

The core neutronics parameters input to the model are evaluated at the core conditions which yields the most limiting values of moderator and Doppler reactivity coefficients, spatial power distribution, and shutdown margin. This is normally the EOC condition, since the moderator temperature coefficient is most negative and the shutdown margin is minimum. Trip reactivity transient insertion characteristics need not be input to the analysis, since the reactor is assumed to be initially shutdown with minimum shutdown margin.

The moderator reactivity coefficient is also calculated assuming the most reactive rod is stuck in its fully withdrawn position, and includes the local reactivity feedback from the high neutron flux in the vicinity of the stuck rod.

The acceptance criteria for the main 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 clad temperature calculated to occur at the core hot spot must not exceed 2750°F.
3. The number of fuel rods calculated to experience a DNBR of less than 1.3 (W-3, when RCS pressure is >1000 psia) or 1.45 (W-3, when RCS pressure is ≥500 psia but ≤ 1000 psia) (Reference 7), should not exceed the limits of 10CFR 100. This limit is currently the maximum number of failed fuel rods calculated in the FSAR (Reference 2).

3.14.3 NSP Safety Analysis Experience

NSP has analyzed the main steam line break using input consistent with the Prairie Island FSAR (Reference 2). The models described in Appendix A were used to analyze the following transient cases:

a. A break at the exit of the steam generator with safety injection and offsite power assumed available.
b. A break at the exit of the steam generator, with safety injection but without offsite power assumed available.
c. A break downstream of the flow measuring nozzle with safety injection and offsite power assumed available.
d. A break downstream of the flow measuring nozzle with safety injection but without offsite power assumed available.
e. A break equivalent to 247 lbm/sec at 1100 psia with safety injection and offsite power assumed available.

The results of case A are compared to those reported in Figure 14.2-6, 14.2-13, and 14.2-12 of Reference 2 in Figures 3.14-1 to 3.14-6. The NSP results show a slightly slower blowdown in the initial stages of the transient and a slightly higher blowdown in the final stages, which is reflected in the containment pressure curve. The containment pressure response is shown in Figure 3.14-6. Three curves are shown; the FSAR results, the NSPNAD CONTEMPT-LT results (CNP000/82) using mass and energy release data calculated from the FSAR analysis and the NSPNAD CONTEMPT-LT results (CNP012/81) using mass and energy release data calculated from the DYNODE-P analysis.

The containment pressure response is fairly insensitive to all parameters except the mass and energy release rates. The CONTEMPT-LT results show good comparison to the FSAR results when consistent sets of release data are used (CNP000/82). The peak containment pressures compare within 1.5%.

The hot channel model described in Appendix C was used to analyze the hot channel DNBR at the five steady state conditions listed in Table 3.14-1. The minimum DNBR calculated using the NSP multichannel 1/8 assembly model was 1.596, compared to Section 14.2-5 of Reference 2 which states that the DNBR was greater than 1.37 for all cases.

The results of case B are compared to those reported in Figure 14.2-8 of Reference 2 in Figures 3.14-12 to 3.14-16.

The results of case C are compared to those reported in Figure 14.2-5 of Reference 2 in Figures 3.14-7 to 3.14-11.

The results of case D are compared to those reported in Figure 14.2-7 of Reference 2 in Figures 3.14-17 to 3.14-21.

The results of case E are compared to those reported in Figure 14.2-9 of Reference 2 in Figures 3.14-22 to 3.14-24.

An important parameter which is input to the model is the boron concentration in the Boron Injection Tank. The value used in the NSP model corresponds to the minimum value permitted by the Technical Specifications. A conservative evaluation of the least negative boron coefficient is also input to the model.

3.14.4 Cycle Specific Physics Calculations

These calculations are performed at the most limiting core conditions found during the cycle, e.g., the point in time and the power level that gives the most conservative value of the parameter in question. Sensitivity studies are conducted to determine these limiting conditions accounting for the effects of control rods, xenon, power level, temperature, etc. for each parameter.

a. Moderator Temperature Coefficient (α_M)

Calculations of α_M are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations are performed at EOC with hot full power boron concentrations in order to obtain the most negative coefficient. Additionally, calculations are made with all rods in except for the most reactive RCCA stuck out at 1000 psia as a function of core average temperature. Using this functional value of $\alpha_M(T)$, k_{eff} is calculated versus temperature assuming an initial 2% shutdown condition at 547°F. Model biases are included in all calculations.

b. Doppler Temperature Coefficient (α_D)

Calculations of α_D are performed in accordance with the general procedure described in Section 2.0.

c. Boron Reactivity Coefficient (α_B)

The boron coefficient is calculated consistent with the description given in Section 2.0, and is calculated for HZP conditions.

d. Shutdown Margin (SDM)

The shutdown margin is calculated consistent with the description given in Section 2.0, and is calculated for HZP and HFP conditions. Model biases are included in the calculations.

e. Fuel Pin Census

Calculation of the number of fuel pins (pin census) versus $F_{\Delta H}$ is performed in accordance with the general procedures described in Section 2.0. The calculations determine the number of fuel pins above the limiting value of $F_{\Delta H}$ above which the DNBR equals 1.3 (W-3, when RCS pressure is >1000 psia) or 1.45 (W-3 when RCS pressure is ≥500 psia but ≤1000 psia) (Reference 7).

3.14.5 Reload Safety Evaluations

Each parameter calculated above is conservatively adjusted to include the model reliability factors, RF_i, and biases, B_i. These results are then compared to the bounding values assumed in the safety analysis. For K_{eff} versus temperature during cooldown, the reliability factors are applied to the calculation of the moderator temperature coefficient prior to the determination of K_{eff}. Uncertainties applied to the shutdown margin (SDM) include reliability factors for the rod worth, moderator temperature defect, and Doppler temperature defect as discussed in Section 2.4. The cycle specific parameters are acceptable if the following inequalities are met:

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Table 3.14- 1: Steady State Conditions For Hot Channel Analysis of Steam Line Break

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*   Values taken from case DNP119/81

**   Values taken from FSAR Table 14.2-1

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Figure 3.14- 1: Main Steam Line Break - At S.G. Exit With A.C.

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Figure 3.14- 2: Main Steam Line Break - At S.G. Exit With A.C.

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Figure 3.14- 3: Main Steam Line Break - At S.G. Exit With A.C.

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Figure 3.14- 4: Main Steam Line Break - At S.G. Exit With A.C.

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Figure 3.14- 5: Main Steam Line Break - At S.G. Exit With A.C.

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Figure 3.14- 6: Main Steam Line Break - At S.G. Exit With A.C.

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Figure 3.14- 7: Main Steam Line Break - Downstream of Flow Restrictor With A.C.

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Figure 3.14- 8: Main Steam Line Break - Downstream of Flow Restrictor With A.C.

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Figure 3.14- 9: Main Steam Line Break - Downstream of Flow Restrictor With A.C.

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Figure 3.14- 10: Main Steam Line Break - Downstream of Flow Restrictor With A.C.

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Figure 3.14- 11: Main Steam Line Break - Downstream of Flow Restrictor With A.C.

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Figure 3.14- 12: Main Steam Line Break - At S.G. Exit Without A.C.

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Figure 3.14- 13: Main Steam Line Break - At S.G. Exit Without A.C.

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Figure 3.14- 14: Main Steam Line Break - At S.G. Exit Without A.C.

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Figure 3.14- 15: Main Steam Line Break - At S.G. Exit Without A.C.

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Figure 3.14- 16: Main Steam Line Break - At S.G. Exit Without A.C.

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Figure 3.14- 17: Main Steam Line Break - Downstream of Flow Restrictor Without A.C.

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Figure 3.14- 18: Main Steam Line Break - Downstream of Flow Restrictor Without A.C.

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Figure 3.14- 19: Main Steam Line Break - Downstream of Flow Restrictor Without A.C.

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Figure 3.14- 20: Main Steam Line Break - Downstream of Flow Restrictor Without A.C.

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Figure 3.14- 21: Main Steam Line Break - Downstream of Flow Restrictor Without A.C.

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Figure 3.14- 22: Main Steam Line Break - 247 lbm at 1100 PSIA With A.C.

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Figure 3.14- 23: Main Steam Line Break - 247 lbm at 1100 PSIA With A.C.

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Figure 3.14- 24: Main Steam Line Break - 247 lbm at 1100 PSIA With A.C.

3.15 Control Rod Ejection

3.15.1 Definition of Accident

This accident is postulated to result from the unlikely failure of a control rod mechanism pressure housing followed by ejection of an RCC assembly by the reactor coolant system pressure. If a rod inserted in a high worth region of the core were to be ejected, the rapid reactivity insertion and unfavorable power distribution which would result might cause localized fuel rod damage.

3.15.2 Accident Analysis

The analysis of the control rod ejection analysis requires a model of the neutron kinetics coupled to models of the fuel and clad transient conduction and the thermal-hydraulics of the coolant channel. In practice, model sophistication has varied from point kinetics to three-dimensional spatial kinetics. When three-dimensional calculations are not employed, the reactivity feedbacks must be corrected using weighting factors to account for the spatial dimensions not included in the model. The thermal-hydraulic model used includes a multi-nodal radial model of fuel, gap, and clad conduction; and a multi-nodal axial model of the coolant channel. Since the calculations result in maximum fuel enthalpies less than those corresponding to catastrophic fuel failures, the system pressure surge is calculated on the basis of conventional heat transfer from the fuel. The pressure surge model includes prompt heat generation in the coolant (so called "direct moderator heating"), fluid transport in the system, heat transfer in the steam generators, and the action of relief and safety valves. No credit is taken for control rod pressure housing.

The maximum ejected rod worth is calculated with all control banks at their maximum permissible insertion for the power level of interest.

The moderator reactivity effect is included in the model by correlating reactivity with moderator density, thereby including effects of coolant temperature, pressure, and voiding. The Doppler reactivity effect is typically correlated as a function of either fuel temperature or power. The highest boron concentration corresponding to the initial reactor state is assumed in the calculation of the moderator feedback. The largest temperature rises during the transient, and hence the largest reactivity effects, occur in channels where the power is higher than average. This means that the reactivity feedback is larger than that predicted by a single average channel analysis. As a result, when a three-dimensional space-time kinetics calculation is not performed, weighting factors are applied as multipliers to the average channel Doppler feedback reactivity to account for spatial reactivity feedback effects.

The results of the accident analysis are relatively insensitive to β_{eff}, the effective delayed neutron fraction, and ℓ^* the prompt neutron lifetime, except in those cases in which the ejected rod worth approaches or exceeds β_{eff}. In these cases, the minimum value calculated for the assumed initial reactor state is used in the accident analysis. The results are relatively insensitive to ℓ^* in the range of values normally encountered in commercial pressurized water reactors. Minimum values of ℓ^* are used in the accident analysis.

Control rod reactivity insertion during the trip is obtained by combining a differential rod worth curve with a rod velocity curve, based on maximum design limit values for scram insertion times. The reactor trip delay time is calculated by combining the maximum time delays involved in the instrumentation and actuation circuitry.

The acceptance criteria for the control rod ejection accident are as follows:

1. The average hot spot fuel enthalpy must be less than 280 calories/gram.
2. The maximum reactor coolant system pressure must be less than the pressure that will cause stresses to exceed the emergency condition stress limit; assumed to be 120% of design pressure.
3. The maximum clad temperature calculated to occur at the core hot spot must not exceed 2750 °F.
4. The number of fuel rods calculated to experience a DNBR of less than 1.3 (W-3) or 1.17 (WRB-1), whichever is applicable, should not exceed the number which are required to fail in order that the doses due to released activity will exceed the limits of 10CFR 100. This limit is currently the maximum number of failed fuel rods calculated in the FSAR (Reference 2).

3.15.3 NSP Safety Analysis Experience

NSP has analyzed the ejection rod accidents and compared the results to a Westinghouse Topical Report on rod ejection accidents analysis (Reference 4).

The models described in Appendix A were used to analyze the control rod ejection accident at the following four initial conditions:

1. Zero Power Beginning of Life (ZPBOL)
2. Full Power Beginning of Life (FPBOL)
3. Zero Power End of Life (ZPEOL)
4. Full Power End of Life (FPEOL)

The results of these calculations are compared to those reported in Chapter 4 of Reference 4 for equivalent cases (same initial power, core burnup, ejected rod worth, and transient peaking factor). Reference 4 provides documentation of generic results for Westinghouse Pressurized Water Reactors; consequently, those results are applicable to the Prairie Island Plant. In all cases, comparisons are made to the Reference 4 results based on a Doppler reactivity weighting factor of 1.6.

The core nuclear power response and energy release and hot spot fuel temperatures are compared to the results of Figure 4.1 and 4.3 of Reference 4 in Figures 3.15-1 and 3.15-2 for the ZPBOL case.

Similar comparisons with Figures 4.2 and 4.4 of Reference 4 for the FPBOL cases are presented in Figure 3.15-3 and 3.15-4.

Figures 3.15-5, and 3.15-6 show the comparisons with the results of Figures 4.1 and 4.2 of Reference 4 for the core nuclear power and energy release for the ZPEOL and FPEOL cases, respectively.

The results of the comparison for the maximum fuel rod temperatures and enthalpies at the hot spot are given in Table 3.15-1.

An energy balance was performed at the hot spot for the ZPEOL case out to the time at which maximum fuel temperature occurred. This energy balance verified that the NSP hot spot model results are consistent with the energy release.

The NSP results show very good comparison with those of Reference 4 for the ZP cases. The NSP results for the FP cases are more conservative than those of Reference 4. This conservatism is due to the fact that the NSP models do not predict as large a negative doppler reactivity insertion as is seen in Figure 4.2 of Reference 4. A sensitivity study was performed to determine the effect of increasing the fuel rod gap coefficient on energy released. The NSP model used to generate the results of Figures 3.15-1, 3.15-3, 3.15-5, and 3.15-6 uses a conservatively low gap coefficient, i.e. 550 Btu/hr ft^{2}°F. The effect of increasing the gap coefficient is to reduce the specific heat of the UO_2 which increases the relative transient temperature rise and hence increases the Doppler feedback, provided the same Doppler Defect vs. Power Level curve is used for all cases. For the

FPBOL case, a gap coefficient of 10000 Btu/hr ft^2°F resulted in a reduction of 19.4% in the energy release at 5 seconds from the NSP results presented in Figure 3.15-3. For the ZPBOL case, this resulted in a 38.2% decrease in the energy release at 5 seconds from the results of Figure 3.15-1. Note that this brings the NSP results into approximate agreement with those of Reference 4. Therefore, the use of a low average gap coefficient (550 Btu/hr ft^2°F) in the NSP model is a conservative approach.

A sensitivity study was also performed for the peak cladding temperature as a function of the fuel rod surface heat transfer coefficient. For zero power conditions, a change of 50 Btu/hr ft^2°F in the heat transfer coefficient produced about a 425°F change in peak cladding temperature. For full power conditions, a change of 50 Btu/hr ft^2°F in the heat transfer coefficient produced about a 330°F change in peak cladding temperature.

3.15.4 Cycle Specific Physics Calculations

These calculations are performed at the most limiting core conditions found during the cycle, e.g., the point in time and the power level that gives the most conservative result with respect to the acceptance criteria. Sensitivity studies are conducted to determine these limiting conditions accounting for the effects of control rods, xenon, power level, temperature, etc. for each parameter.

a. Moderator Temperature (α_M) and Density Coefficient ($\alpha\rho$)

Calculations of α are performed in accordance with general procedures described in Section 2. Cycle specific values are computed at HZP, HFP to determine the least negative moderator temperature coefficient for each of the four conditions.

b. Doppler Temperature Coefficient (α_D)

The values of α_D are calculated in accordance with the general procedures described in Section 2.0. Full and zero power core conditions are considered in order to determine the least negative value of α_D.

c. Scram Reactivity Curve ($\Delta\rho_{SCRAM}(t)$)

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations for this accident are performed for the full and zero power conditions.

d. **Heat Flux Hot Channel Factor (F_Q)**

The maximum F_Q for all levels is calculated for each of the cases investigated as described above for the determination of the maximum $\Delta\rho_{EJECT}$. The maximum value of F_Q does not necessarily correspond to the maximum value of $\Delta\rho_{EJECT}$. As discussed above, all calculations of F_Q for the instantaneous ejected rod cases do not take credit for the moderator or Doppler feedback mechanisms.

e. **Effective Delayed Neutron Fraction (β_{eff})**

The value of β_{eff} is calculated in accordance with the general procedures given in Section 2.0. Cycle specific calculations are performed for both the full power and zero power conditions to determine the appropriate full power and zero power values.

f. **Prompt Neutron Lifetime (ℓ^*)**

The value of ℓ^* is calculated in accordance with the general procedures given in Section 2.0. Cycle specific calculations are performed for both the full and zero power conditions.

g. **Maximum Ejected Rod Worth ($\Delta\rho_{EJECT}$)**

Calculations of the ejected rod conditions are performed with the nodal model in three-dimensions. No credit is taken for either moderator or Doppler reactivity feedback mechanisms. All calculations are performed with the control rods at the PDIL. Cycle specific calculations are performed for both the full power and zero power conditions. The search for the most reactive rod includes all rods initially inserted at the PDIL. The maximum worth of the ejected rod includes considerations of transient xenon conditions such as maximum positive axial offset.

h. **Fuel Pin Census**

Calculation of the number of fuel pins (pin census) versus $F_{\Delta H}$ is performed in accordance with the general procedures described in Section 2.0. The calculations determine the number of fuel pins above the limiting value of $F_{\Delta H}$ above which the DNBR equals 1.3 (W-3) or 1.17 (WRB-1), whichever is applicable.

3.15.5 Reload Safety Evaluations

Each of the physics parameters calculated for this accident are adjusted to include the model reliability factors, RF_i, and biases, B_i. In calculating $\alpha\rho$, the model reliability factor and bias are first applied to α_M then $\alpha\rho$ is determined. These adjusted values are the cycle specific parameters to be compared to the bounding values used in the safety analysis.

The cycle specific parameters are acceptable with regard to the ejected rod accident if the following inequalities are met:

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Table 3.15- 1: Comparison of Rod Ejection Maximum Fuel Rod Enthalpies and Temperature

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Figure 3.15- 1: Rod Ejection - HZP, BOL

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Figure 3.15- 2: Rod Ejection - HZP, BOL

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Figure 3.15- 3: Rod Ejection - HFP, BOL

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Figure 3.15- 4: Rod Ejection - HFP, BOL

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Figure 3.15- 5: Rod Ejection - HZP, EOL

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Figure 3.15- 6: Rod Ejection - HZP, EOL

3.16 Loss Of Coolant Accident

3.16.1 Definition of Accident

The loss of coolant accident is defined as the rupture of the reactor coolant system piping or any line connected to the system, up to and including a double-ended guillotine rupture of the largest pipe. Ruptures of small flow area would cause coolant expulsion at a rate which would allow replacement at the same rate via the charging pumps and an orderly shutdown would be possible. A larger rupture would result in a net loss of reactor coolant inventory and a decreasing pressurizer water level and pressure. A Safety Injection Signal would be actuated resulting in a reactor trip, the injection of borated water into the reactor coolant system, isolation of the normal feedwater, and initiation of auxiliary feedwater supply. When the reactor coolant system depressurizes to 700 psia, the nitrogen bubble in the accumulator tanks expands, forcing additional water into the reactor coolant system. For large breaks, void formation in the core coolant during the initial blowdown phase results in almost immediate power reduction down to decay heat levels.

3.16.2 Accident Analysis

An analysis of the loss of coolant accident is performed to demonstrate the effectiveness of the emergency core cooling system (ECCS) to meet the criteria of 10CFR50.46 and in preventing radioactive releases which would violate the criteria of 10CFR100. This analysis is usually performed in four steps:

1. A system blowdown analysis is performed to obtain the time-dependent behavior of core power, system pressure, flowrates, and other relevant variables. The digital model employed in this calculation is a detailed representation of the primary and secondary systems, including the hot fuel assemblies and the remainder of the core; the reactor vessel downcomer, upper plenum, upper head, and lower plenum regions; the steam generators, pressurizer, and associated piping; and the safety injection systems. The model uses a lumped "node and flow-path" approach to compute the space and time variations of the thermal-hydraulic conditions of the primary and secondary systems. Some of the phenomena which must be considered in the blowdown analysis are coolant flows between regions; heat transfer between primary and secondary fluids, and between system metal surfaces and fluids contacting

them; the hydraulic interactions of system components such as reactor coolant pumps; fuel rod swelling and rupture; and the behavior of emergency core coolant as it is injected into a system undergoing rapid decompression.

2. An analysis of the core hot channel is conducted using a detailed thermal-hydraulic model supplied with time-varying boundary conditions from the blowdown analysis. These calculations must consider cross flow between regions and any flow blockage calculated to occur as a result of clad swelling or rupture. The calculated flow must be smoothed to eliminate calculated oscillations with a period of less than 0.1 seconds. This model is used during the period extending from the beginning of blowdown to the end of ECCS bypass.
3. A re-flood model continues the system blowdown analysis through the period of ECCS re-flood of the reactor core. Due to the complexity of the phenomena occurring, empirical correlations of experimental data are used to define such variables as carryover fraction, heat transfer coefficients, natural convection in the secondary side of the steam generators, and slip flow in the ruptured loop cold leg nozzle.
4. A thermal calculation of the temperature transient in the hot fuel rod during refill and re-flood is accomplished using a fourth model. As in the re-flood model, empirical correlations of measured data are employed to represent complex phenomena such as flow blockage due to clad swelling and rupture. Metal - water chemical reaction and radiation from the fuel rod surface are included in the hot rod model.

Detailed requirements for ECCS evaluation models are described in 10CFR Part 50, Appendix K.

Certain data related to core neutronics are required as input to the ECCS evaluation model described above. These items consist of the data required to calculate the continuing fission energy generation prior to shutdown by voiding and boron injection; the data necessary to calculate fission product decay heat subsequent to reactor shutdown; and data relating to the initial spatial power distribution.

The fission power history prior to reactor shutdown is calculated from the reactor kinetics equations, with terms included to account for fuel temperature and moderator density feedback, control rod insertion, and injection of borated water. For the larger breaks, reactor shutdown usually occurs due to coolant void formation, while for smaller breaks, scram reactivity insertion is required. A conservative calculation is assured by assuming the minimum plausible values for the various components in the reactivity balance. The moderator feedback is calculated using the boron concentration which corresponds to the core status when the Technical Specification requirement relating to moderator temperature coefficient is just met. Moderator reactivity is input to the transient calculation as a function of core coolant density. The Doppler reactivity feedback is usually much smaller than that resulting from coolant voiding.

Reactor trip may be actuated by one of several signals; the particular trip setpoint first reached and the time of trip are dependent on break size, particularly for small breaks. For large breaks, trip occurs due to high containment pressure or safety injection actuation; while for smaller breaks, pressurizer low pressure or low level actuates the trip.

The trip reactivity insertion is calculated assuming the most reactive rod to remain in its fully withdrawn position and using a rod drop time corresponding to the Technical Specification limit. Large break accidents do not exhibit significant sensitivity to trip reactivity.

Fission product and actinide decay energy sources are calculated in accord with the requirements of Appendix K of 10 CFR Part 50. Infinite operating time is assumed prior to accident initiation.

The spatial power distribution used in the ECCS evaluation analysis is chosen as the most limiting from the several calculated to occur over the lifetime of the core. Axial power shapes with maximum near the core mid-plane generally result in the most severe accident consequences. This is because the upper portions of the core are cooled to a greater extent during the flow reversal which occurs early in blowdown, and the lower portion of the core is cooled quickly by the initial stages of re-flood. The initial hot spot peaking factor, F_Q, plays an important role in determining the severity of the worst cladding temperature response in the core. Because of the rapid degradation in heat transfer following the break, the temperature profile within the fuel rod tends to flatten out thereby increasing the cladding temperature. In addition, larger values of F_Q will result in less effective heat transfer during the re-flood period at the hot spot. Thus, a larger value of F_Q will produce a more severe cladding temperature response.

3.16.3 NSP Safety Analysis Experience

NSP has not analyzed this accident. The Westinghouse and Exxon analysis is reviewed for each reload to determine its applicability to the current core design.

3.16.4 Cycle Specific Physics Calculations

These calculations are performed at the most limiting core conditions found during the cycle, e.g., the point in time and the power level that gives the most conservative result with respect to the acceptance criteria. Sensitivity studies are conducted to determine these limiting conditions accounting for the effects of control rods, xenon, power level, temperature, etc. for each parameter.

a. **Moderator Temperature Coefficient (α_M)**

Calculations of α_M are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations for this accident are performed to determine the moderator coefficients over the operating range of 0 - 100% power under various conditions of xenon inventory.

b. **Doppler Temperature Coefficient (α_D)**

Calculations of α_D are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations for this accident are performed as a function of power level over the range 0 to 100% power.

c. **Boron Reactivity Coefficient (α_B)**

The boron coefficient is calculated consistent with the description given in Section 2, and is calculated for HZP and HFP conditions.

d. **Shutdown Margin (SDM)**

The shutdown margin will be calculated and compared with the limiting assumptions used in the safety analysis using the methods described in Section 2.0.

e. **Scram Reactivity Curve ($\Delta\rho_{SCRAM}(t)$)**

Calculations of the scram reactivity curve are performed in accordance with the general procedures described in Section 2.0. Cycle specific calculations for this accident are performed at BOC and EOC for the full power condition.

f. Heat Flux Hot Channel Factor (F_Q)

The maximum core F_Q for all levels is assumed to remain within the current Technical Specifications for all allowable combinations of axial offset and power level. For Prairie Island, the continuous surveillance of the power distribution is accomplished with the excore detectors using the Exxon PDC-IIa[3] scheme.

g. Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)

The maximum core $F_{\Delta H}$ are assumed to remain within the current limits as defined in the Technical Specifications for allowable combinations of axial offset and power level. For Prairie Island, the continuous surveillance of the power distribution is accomplished with the excore detectors using the Exxon PDC-IIa[3] scheme.

3.16.5 Reload Safety Evaluation

Each of the physics parameters calculated for this accident are adjusted to include the model reliability factors, RF_i, and biases, B_i. In calculating $\alpha\rho$, the model reliability factor and bias are first applied to α_M then $\alpha\rho$ is determined. These adjusted values are the cycle specific parameters to be compared to the bounding values used in the safety analysis.

The cycle specific parameters are acceptable with regard to the ejected rod accident if the following inequalities are met.

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The integral of the bounding value of the scram curve, $\Delta\rho_{SCRAM}(T)$, is taken as that rod worth required to produce the shutdown margin assumed in the safety analysis for the most limiting cycle specific core conditions.

3.17 *Fuel Misloading Accident*

3.17.1 Definition of Accident

The fuel misloading event consists of the inadvertent loading and operation of a fuel assembly in an improper position. Two types of loading errors should be considered: the interchange of two assemblies so that one or both are in improper core locations and the improper orientation of an assembly. Only the interchange of two assemblies will be considered because it is not possible to load an assembly with an improper orientation. The improper orientation is not possible because the fuel manipulation crane is capable of engaging the fuel assembly in only one orientation through the use of asymmetric orientation holes and the manipulation crane itself can not be rotated. Full power operation with a misloaded fuel assembly could lead to a reduction in DNBR and is subject to limitations specified in the plant Technical Specifications.

3.17.2 Accident Analysis

To ensure a misloading event will not occur, several administrative procedures and technical specifications are implemented during and after core loading. The applicable Prairie Island technical specifications are 3.8 Refueling and Fuel Handling and 3.11 Core Surveillance Instrumentation. These procedures require that no fuel movement is permitted without use of a fuel movement log approved by the Superintendent Nuclear Engineering. The fuel transfer logs, located in the control room, containment and the spent fuel pit area are required to be signed off after each move and all moves are under the direct supervision of an SRO with no other concurrent responsibilities and independently witnessed and verified by the fuel accountability person. The technical specifications require a power map using the incore moveable detector system following each fuel loading. This map will be used to confirm the proper fuel arrangement within the core. After fuel shuffling has been completed, the entire core is video taped to permanently record the positions of all fuel assemblies in the core.

The analysis of the fuel misloading accident will consist of a sufficient number of misloadings to ensure the worst undetected misloading has been analyzed. The analysis will be done using the DP5 program (Reference 1) to determine the effects of the misloading on $F_{\Delta H}$ and F_Q.

The results of the analysis must meet one of the following acceptance criteria:

a. All misloadings are detectable.
b. All undetectable misloadings are within the nuclear uncertainty and therefore inconsequential.
c. Any undetectable misloadings larger than the nuclear uncertainty will have offsite consequences that are a small fraction of 10CFR Part 100 guidelines.

3.17.3 NSP Safety Analysis Experience

NSP has analyzed the fuel misloading accident using a typical core loading which is representative of the Prairie Island reload cores. The analysis considered misloadings in the unmonitored locations only. Any misloadings in monitored locations would be detected.

The analysis was done by calculating the actual F_Q and $F_{\Delta H}$ from the power distribution and the measured F_Q and $F_{\Delta H}$ using the detector reaction rates from only the monitored bundles. It was then determined that if the actual F_Q and $F_{\Delta H}$ violated the technical specifications, the detector system would also show F_Q and $F_{\Delta H}$ (including nuclear uncertainty) violated the technical specifications.

The actual F_Q and $F_{\Delta H}$ were determined by combining the 3-dimensional nodal power distribution from DP5 (Reference 1) and the pin-to-box factors calculated from a discrete pin quarter core PDQ (Reference 1). The measured F_Q and $F_{\Delta H}$ were determined using the DETECTOR program (Reference 5) inputting the detector reaction rates from the monitored bundles. The detector reaction rates were calculated from the DP5 power distribution using the SIGMA program (Reference 1).

Table 3.17-1 shows a summary of the cases run. The "*" on Table 3.17-1 shows that for each violation of the actual F_Q or $F_{\Delta H}$, the measured F_Q or $F_{\Delta H}$ will also show a violation. Figures 3.17-1 through 3.17-17 show the change in the detector reaction rates for the monitored bundles as a result of the misloadings.

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, analysis confirm that the resulting power distribution effects which are undetectable by the incore moveable detector system are within the nuclear uncertainty and therefore inconsequential.

3.17.4 Cycle Specific Physics Calculations

The misloaded bundle analysis showed that the incore moveable detector system will detect all misloadings which are of consequence. Therefore the analysis need not be repeated for each cycle because the same detector system and procedures to ensure a misloading will not occur will be used.

Table 3.17- 1: Summary of Misloadings Analyzed

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Actual - from power distribution

Measured - from DETECTOR program including nuclear uncertainty

*   denotes technical specification violation

Figure 3.17- 1: Misloaded Bundle

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Figure 3.17- 17: Misloaded Bundle

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4.0 REFERENCES

1. Northern States Power Company, Prairie Island Unit 1, Topical Report titled "Qualification of Reactor Physics Methods for Application to Prairie Island".
2. Northern States Power Company, Prairie Island Nuclear Power Plant, Final Safety Analysis Report.
3. Exxon Nuclear Company, Inc., "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II", XN-NF-77-57(A), XN-NF-77-57 Supp.1(A), May, 1981.
4. D. H. Fisher, Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods", WCAP-7588, Revision 1-A, January 1975.
5. S. C. Cohen, P. H. Siang and R. J. Shanstrom, "CONFORM: The Exxon Nuclear Revised Core; Codes for Operating Reactor Evaluation" XN-NF-CC-48, March 1979.
6. Northern States Power Company, NAD Policies and Procedures, "Prairie Island Reload Safety Evaluation", NAP2.102T.
7. Letter from Kathleen C. Midock (Westinghouse) to Keith Higar (NSP), 94NS*-G-004, February 14, 1994, "Utilization of W-3 DNB correlation below 1000 psia".

APPENDIX A

This section describes the NSP computer programs that were used to simulate the response of the nuclear steam supply system (NSSS) and the thermal-hydraulic response of the hot coolant channel and hot spot in the core for the transients and accidents listed in this report.

A.1 Overview

The DYNODE-P program (DYNODE-P) is used to analyze the transient response of the Nuclear Steam Supply System (NSSS). This program is described in detail in Reference A1. DYNODE-P provides a simulation of the core average power, the core average fuel temperature, and the core average coolant channel thermal-hydraulic responses.

The COBRA-IIIC/MIT program (Reference A2) is used to analyze the transient response of the hot channel in the core. COBRA-IIIC/MIT provides a simulation of the thermal-hydraulic response of the coolant channels and associated fuel rods within the core.

The TOODEE-2 program (Reference A3) is used to compute the transient temperature response of the hot fuel spot for certain accidents. TOODEE-2 provides a simulation of the hot fuel rod and associated coolant channel. This program is used only if the COBRA-IIIC/MIT hot channel analysis yields a departure from nucleate boiling ratio (DNBR) which is less than the value corresponding to the 95% probability limit at 95% confidence level.

The CONTEMPT-LT program (Reference A5) is used to compute the transient temperature-pressure response of the containment during a steamline break accident.

The sequence of calculations and interfaces of these programs are as follows: DYNODE-P is run to obtain the core average power and the RCS thermal-hydraulic response. The transient core average power, core inlet coolant temperature and flowrate, and RCS pressure responses along with the appropriate core spatial power distribution and hot channel relative flow rate are then input into COBRA-IIIC/MIT to obtain the hot channel transient DNBR. Similar information is also input into TOODEE-2 to analyze the thermal response of the hot fuel spot in those cases requiring this analysis. The transient steam generator mass and energy release obtained from DYNODE-P are input to CONTEMPT-LT to analyze the containment pressure response for a main steamline break.

A.2 Computer Code Uncertainties

During the NSP/NRC meeting of July 21, 1982, the NRC expressed a requirement that the uncertainties of a computer program must be established before it can be approved for use in safety analyses. The case relating to the NRC acceptance of the GE ODYN code was cited as an example to support this point of view. Dick Kern, of NAI, responded that this requirement was not necessary for the NSP methods for the reasons he expressed at that time. NSP believes that these reasons, which are presented in this letter, are still valid.

The validation of the NSP Reload Safety Evaluation methods is based on comparisons with the Prairie Island FSAR results. The FSAR analyses are a set of conservative calculations in that they were based on pessimistic assumptions and employed computer programs which were acceptable standards for performing these types of analyses. Hence, the NSP method of qualification relates to comparisons of one licensing type analysis to another, and these comparisons have little or no relationship to those which would be made on the basis of best estimate (realistic) analyses.

In the case of ODYN, this code was qualified for over pressurization transients on the basis of comparisons with the Peach Bottom 2 and KKM turbine trip test data. These comparisons were made using best-estimate calculations. In this case, GE could not qualify ODYN by comparisons to a conservative analysis, since none other existed at that point in time. Their earlier code (REDY) had been demonstrated to be non-conservative for overpressurization transients on the basis of the turbine trip tests. Thus, since the qualification was made on a best-estimate basis, the question of code uncertainties was appropriate in order that adequately conservative models could be developed using that program by factoring in these uncertainties.

APPENDIX A REFERENCES:

A1. R.C. Kern, "DYNODE-P version 3: A Nuclear Steam Supply System Transient Simulation for Pressurized Water Reactors-User Manual," NAI 80-70, revision 0.

A2. Robert W. Bowring and Pablo Moreno, "COBRA IIIC/MIT Computer Code Manual," MIT, March 1976.

A3. G.N. Lauben, "TOODEE-2: A Two-Dimensional Time Dependent Fuel Element Thermal Analysis Program," NUREG-75/057, May 1975.

A4. G.R. Horn, F.E. Panisko, "Users Guide for GAPCON: A Computer Program to Predict Fuel-to-Cladding Heat Transfer Coefficients in Oxide Fuel Pins," HEDL-TME72-128, September, 1972.

A5. L.L. Wheat, et al. "CONTEMPT-LT: A Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident," ANCR-1219, June, 1975.

APPENDIX B

B.1 DYNODE-P Code (Reference 1)

The response of the NSSS of a PWR under transient and accident conditions is analyzed with the DYNODE-P program. This program includes a simulation of the components of a PWR NSSS which significantly influence the response of the system to transient conditions. Geometry options are provided to permit representation of any of the current PWR designs.

The major features of DYNODE-P are:

- Point kinetics model for core power transients with major feedback mechanisms and decay heat represented.
- Power forced mode option for hot channel analyses.
- Multinode radial fuel rod and multinode axial coolant channel representations in the core.
- Conservation of mass, energy, volume, and boron concentration for the reactor coolant system (RCS). Conservation of momentum is optional.
- Detailed non-equilibrium pressurizer model including spray and heater systems and safety and relief valves.
- Explicit representation of the shell side of the steam generators including conservation of mass, energy, and volume.
- Explicit representation of the main steam system with isolation, check, dump, bypass, and turbine valves including conservation of mass, energy, momentum, and volume.
- Representation of the reactor protective and high pressure safety injection systems.
- Provisions for simulating a variety of transients and accidents including a break in the main steam system, asymmetric loop transients, and steam generator tube ruptures.

The basic input parameters relating to the initial conditions are:

- Core geometry and initial thermal-hydraulic characteristics.
- Initial RCS pressure and pressurizer level, core inlet enthalpy, RCS flow distribution, and RCS boron concentration.
- Initial core power level and distribution.

- RCS, steam generator, and main steam system volume distributions and hydraulic characteristics.
- Initial steam generator pressures and levels and heat transfer data.

The input parameters required to obtain the transient response are:

- Core kinetics characteristics including control rod motion.
- Reactor coolant system inertias, pressure loss coefficients, and pump hydraulic and torque characteristics.
- Control system characteristics.
- Main and auxiliary feedwater characteristics.
- Valve characteristics.
- Safety systems characteristics.
- Transient power demand.

The major output consists of the following list of parameters which are edited at select time points during the transient:

- Core variables
  - Average power
  - Fuel rod temperature and heat flux
  - Coolant enthalpies, temperature, and mass
  - Kinetics variables including k_{eff}
- RCS variables
  - Mass, energy, and boron distribution of the coolant
  - Loop flow rates
  - Pressurizer pressure and level
  - Safety system variables
  - Pressure control system variables
  - Reactor coolant pump speeds, torques, and developed heads
- Steam generator variables
  - Pressure and levels
  - Masses
  - Heat loads
  - Feedwater and steam flows
- Main steam system variables
  - Pressure and mass distributions
  - Steam flows

APPENDIX B REFERENCES

B.1 R. C. Kern, "DYNODE-P version 3: A Nuclear Steam Supply System Transient Simulation for Pressurized Water Reactors - User Manual," NAI 80-70, revision 0.

APPENDIX C

C.1 COBRA IIIC/MIT Code (Reference 1)

A thermal-hydraulic hot channel analysis is performed for each transient to determine the minimum DNBR using the COBRA IIIC/MIT code.

COBRA-IIIC/MIT was developed by the Massachusetts Institute of Technology and computes the flow and enthalpy distribution in the fuel rod bundles or core for both steady state and transient conditions on a subchannel basis. This development was carried out under a two phase sponsorship, the first phase being sponsored by the Electric Power Research Institute (EPRI) and the second by New England Electric System and Northeast Utilities Services Co.

The coolant regions under analysis are divided into computational cells. The conservation equations of mass, energy, and momentum for the fluid are solved for each cell where the independent variables (enthalpy, pressure and velocity) are averaged. Heat and momentum sources and sinks due to fixed solids (such as fuel rods) are considered.

The heat transfer regimes from subcooled to superheated forced convection including departure from nucleate boiling (DNB), and turbulent and diversion cross flows are considered in the subchannel analysis.

The basic input parameters are:

- Fuel rod and channel geometries
- Fluid thermal-hydraulic parameters
- Heat flux distribution
- Turbulent mixing parameters
    - Core pressure drop parameters
    - Transient forcing functions.
        - Core inlet temperature
        - Core inlet flow
        - System average pressure
        - Core power or heat flux

The major time-dependent output parameters are:

- Subchannel DNB
- Subchannel flow heat fluxes
- Subchannel fluid properties
- Fuel rod heat fluxes and DNB ratios

C.2 Thermal Margin Methodology

The NSP Thermal Margin methodology is based on the use of the COBRA IIIC/MIT[1] code. No modifications have been made to the 1976 version of the code. Thermal margins are established on the basis of plant specific transient analysis performed with the DYNODE-P[2] code.

The COBRA IIIC/MIT program is used to evaluate coolant density, mass velocity, enthalpy, vapor voids and static pressure distributions along parallel flow channels in a three-dimensional PWR core under all expected operating conditions.

In order to evaluate the predictive capabilities of the NSP methodology, the COBRA IIIC/MIT code was benchmarked to the Prairie Island FSAR[3] results. These results are shown in Reference 4. An effort was made to construct a model, for benchmarking, which was comparable to that which was used to do the original analysis (Westinghouse 1969).

Two models were developed, a single hot channel model, and an 1/8 hot assembly model. The single channel model uses a constant assembly gap flow fraction (obtained from Reference 5) whereas the 1/8 assembly model includes this flow in the mixing calculations. Both models use an $F_{\Delta H}$ engineering peaking factor of 1.01. This factor includes the effects of inlet flow maldistribution, flow mixing, flow redistribution, and tolerances in pellet diameter, density and enrichment, full rod diameter, pitch and bowing. Key input parameters for both the single channel and the 1/8 assembly models are shown in Tables C-1 and C-2 respectively.

The FSAR shows an initial MDNBR of 1.885. This compares favorably with the NSP single channel (within 0.2%) and 1/8 assembly (within 0.4%) steady state results. The transient calculations are run as pseudo steady state time points using heat flux rather than core power as the transient forcing term. The results show good comparisons. The benchmarks are not expected to show exact comparison since the transient forcing functions for heat flux, pressure, core inlet temperature and core inlet mass flow are slightly different (the NSP calculations are based on the DYNODE-P benchmarks of the FSAR analyses). In general, however, the maximum transient change in MDNBR and the transient trends are followed well.

Once the initial models were benchmarked with good results, it became necessary to examine the validity of these simple models to correctly predict core-wide flow distribution effects. It was questionable whether an engineering peaking factor on $F_{\Delta H}$ could account for assembly to full core mixing affects for a mixed core loading, i.e. Westinghouse, Exxon Standard and Exxon TOPROD, since these different fuel types have different thermal-hydraulic properties, i.e. different pellet diameters, rod diameters, and pressure loss coefficients. The NSPNAD thermal margin methodology was therefore expanded to evaluate a full core (symmetric 1/8 core) rather than a single channel or 1/8 assembly. The 1/8 core model will be used exclusively in all future RSE calculations.

The core-wide model provides an overall analysis of the thermal hydraulic behavior of the core. The hot quarter assembly is modeled by individual subchannels each consisting of an individual or a limited number of fuel rods. The remainder of the core is modeled on an assembly-by-assembly basis. Each channel is divided axially into increments of equal lengths. Resistance to crossflow and coolant mixing between adjacent channels is considered. Flow redistribution due to localized hydraulic resistances (i.e. spacer grids) is also predicted. The effects of local variations in power, fuel rod and pellet fabrication and fuel rod spacing are also considered. An iterative procedure is used to adjust the inlet flow distribution to give a uniform exit pressure distribution while maintaining the same total coolant mass flow rate. Table C-3 gives a summary of the key input parameters for the 1/8 core model. All of the effects included in the $F_{\Delta H}$ engineering factor are therefore accounted for explicitly in the model, so that the 1.01 peaking factor can be dropped from the model.

The following section gives a detailed description of the modeling techniques used and key input parameter sensitivity studies performed.

Radial Power Distribution

Individual rod power generation and lumped rod power generation is represented in the code-wide model. Radial power distributions are obtained from the nuclear design analysis (Reference 6). The radial hot assembly average power is artificially increased to the predetermined $F_{\Delta H}$. The radial power distribution within the hot assembly is conservatively assumed to be the worst assembly distribution as determined from the nuclear design analysis of all assemblies. This worst case distribution is then conservatively overlaid on the actual hot assembly location. The hot rod power is increased by an engineering factor F_Q^E, of 3%. This factor accounts for tolerances in pellet diameter, density and enrichment fuel rod diameter, pitch and bowing at the hot spot. Due to code modeling limitations, the hot spot factor is conservatively applied to the entire hot rod.

Two radial hot assembly power distributions were analyzed to determine the worst case, a conservatively flat and a typical locally peaked distribution. The conservatively flat distribution would be expected to be the most limiting since the flatter power distribution would reduce the coolant mixing effects in the hot assembly.

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* Hot node axial flow/Core average inlet flow

Axial Power Distribution

A constant axial flux distribution is superimposed on the radial distribution in order to yield the heat generated in each individual rod (or lumped rod) at any elevation. A cosine distribution with a conservatively high peak, 1.72, is used. A sensitivity study was performed for three axial power distributions, a middle peaked cosine, a down-skewed and an up-skewed cosine shape (see figure 3). It should be noted that the skewed cases are well out of the operating band of the power distribution control scheme used.

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The up-skewed case was found to produce the most conservative results and will be used for reload calculations. The cosine shape was used for this study to maintain the compatibility of the model to previous calculations.

Inlet Flow Distribution

The initial inlet mass flow distribution is assumed constant. An iterative procedure is then used to adjust the inlet flow distribution to give a uniform exit pressure distribution while maintaining the same total coolant flow rate. The core inlet pressure distribution is assumed constant. Reference 5 found the analyses are insensitive to the inlet pressure distribution. The axial distributions for three channels are shown in figure 4. The three channels shown are: 45 - the hot channel, 1 - the core center assembly lumped channel and 81 - a core peripheral lumped channel.

Assembly Crossflow

The transverse momentum equation provides for the evaluation of cross flow between adjacent channels. The equation contains two parameters, SL and KIJ which must be established by the user. Where:

SL = Transverse Momentum Factor = s/l
KIJ = Crossflow Resistance Coefficient
s = gap spacing
l = centroid distance

Reasonable values for these parameters are, SL = 0.25 and KIJ = 0.15. Sensitivity studies were performed for $0.0 \leq KIJ \leq 25.0$ and $0.01 \leq SL \leq 0.5$. The most conservative assumptions are those that increase the crossflow effect. As KIJ, the crossflow resistance decreases, crossflow increases. As SL increases, crossflow increases. The results are relatively insensitive to both these parameters.

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$R_f = D_k/P$

Where D_k = centroid to centroid distance and P = rod pitch. This model did not impact the results and will not be used in future analysis.

Coolant Mixing

The reference turbulent mixing parameter β is:

$$\beta = 0.0062 \ Re^{-0.1}$$

This value of $\beta$was developed for fully developed smooth tube conditions, and as such, represents the minimum value expected for turbulent mixing. Since this reference value of β is the anticipated minimum, the sensitivity of increasing β was only considered. A small value of β is included in the study for comparison purposes only and represents an unrealistically low value.

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The reference value of β produces expectedly conservative results with respect to MDNBR.

Axial Increments

The size of the length step in the axial direction was studied. Three cases were run in which the core was divided into 25, 40 and 50 axial increments. The three cases produced consistent results.

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Convergence Criteria

The convergence criteria used are those which are recommended in the code manual (Reference 1), i.e. a flow convergence error of 0.001 with a maximum of 60 iterations allowed. Most cases typically converge within 6-7 iterations. To ensure a stable solution has been reached, the steady state base case was allowed to iterate for 200 times. The results are shown in Figures C-5, C-6 and C-7.

The solution does oscillate divergently with iteration, however the amplitude of the oscillations at the point of convergence is sufficiently small to be insignificant, e.g. 0.04% hot node mass flux and 0.003% hot node enthalpy.

Hot Channel Identification

Once the hot rod has been identified, the hot quarter assembly which contains the hot rod is represented by single subchannels and the hot subchannel is identified as the one having the lowest MDNBR. Once identified, the hot channel dimensions are reduced to account for tolerances specified by the vendor on fuel rod diameter and pitch.

Flux Peaking at Edge of Hot Assembly

In valid subchannel analysis, sufficient detail of the regions surrounding the hot channel must be considered. If a case is specified where the hot channel occurred on the edge of the hot quarter assembly, the hot channel would be adjacent to a quarter assembly lumped channel. The basic philosophy upon which subchannel analyses are based is thus not being satisfied. Therefore a new geometry must be described such that the hot channel is interior to a region of equivalently sized subchannels. This new geometry representation is accomplished by representing the adjacent quarter assembly on a detailed subchannel basis (similar to the hot quarter assembly) rather than a lumped channel basis.

Steady State Results

The 1/8 core model should give identical results to the simple models for the full power steady state case since the assembly to full core peaking factor, used in the simple models, was developed at these conditions.

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Transient Analysis

The Cobra IIIC/MIT transient analysis forcing functions for heat flux pressure, inlet mass flux, and inlet temperature are obtained from the DYNODE-P[2] system analysis. The

DYNODE-P fuel model accounts for the heat capacity of the rods so that heat flux rather than power is input to COBRA.

For transients in which the core flow changes rapidly, i.e. Pump Trip and Locked Rotor, the COBRA IIIC/MIT model is run in the transient mode. For slow transients in which the core flow remains approximately constant, i.e. Rod Withdrawal and Turbine Trip, the COBRA IIIC/MIT model is run in the steady state mode. A sensitivity study was run to determine the time step size required in the transient mode. The case run was the 2/2 Pump Trip.

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It was determined that a time step size of 0.25 seconds was sufficient to analyze the loss of flow transients.

Several transient benchmarks (Reference 3) were rerun using the new 1/8 core model. The transients run were, Fast Rod Withdrawal, Slow Rod Withdrawal, 2/2 Pump Trip, ½ Pump Trip and the Locked Rotor. The results are shown in figures 3.2-4, 3.2-8, 3.11-4, 3.11-8, 3.12-3 and 3.12-5 of Reference 4 respectively. The results would be expected to give comparable results to the 1/8 assembly case since the $F_{\Delta H}$ peaking factor used in the 1/8 assembly case was developed for this loading (i.e. Cycle 1). The 1/8 core model would, however, be expected to give significantly different results than a 1/8 assembly model for current loadings which include 3 different fuel types. It can be seen that the results compare very well to previous analyses as expected.

Table C- 1: Single Channel Model

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Table C- 2: 1/8 Assembly Model

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Table C- 3: 1/8 Core Model

Proprietary Information Deleted

1. Robert W. Bowring and Pablo Moreno, "COBRA IIIC/MIT Computer Code Manual," MIT, March 1976.
2. R.C.Kern, "DYNODE-P version 3: A Nuclear Supply System Transient Simulation for Pressurized Water Reactors - User Manual," NAI 80-70, revision 0.
3. Northern States Power Company, Prairie Island Nuclear Power Plant, Final Safety Analysis Report.
4. Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods for Application to PI Units, NSPNAD-8102P, December 1981.
5. Topical Report, Application of the THINC Program to PWR Design, WCAP 7359-L, August 1969.
6. Prairie Island Nuclear Power Plant, Qualification of Reactor Physics Methods, NSPNAD-8101P, December 1981.

Figure C- 1: COBRA IIIC/MIT 1/8 Core Model Channel Layout

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Figure C- 2: COBRA IIIC/MIT 1/8 Core Model Channel Layout

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Figure C- 3: COBRA IIIC/MIT Axial Power Shape

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Figure C- 4: COBRA IIIC/MIT Axial Flow vs. Channel Length

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Figure C- 5: COBRA IIIC/MIT Mass Flux Convergence

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Figure C- 6: COBRA IIIC/MIT Enthalpy Convergence

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Figure C- 7: COBRA IIIC/MIT Crossflow Convergence

APPENDIX D

D.1 TOODEE2 Code (Reference 1)

In those cases where the DNBR limit may be exceeded (as, for example, in the case of the locked rotor incident), transient clad and fuel temperatures of the hot rod must also be obtained to assure that design criteria are met. The hot rod analysis is performed using the TOODEE2 code.

TOODEE-2 computes the thermal response of a fuel rod and associated coolant channel under transient conditions.

TOODEE-2 solves the conservation of energy equation in the fuel rod and the conservation of mass and energy in the coolant channel over the entire length of the core. The fuel-cladding gap model is the same as in the GAPCON programs (Reference 5). Material properties are computed based on local conditions. Cladding deformation is taken into account.

Zr-H_2O reaction is also considered as part of the total heat source. Heat transfer regimes from subcooled to superheated forced convection are considered.

The major input parameters are:

- Fuel rod and coolant channel geometries and properties
- Initial power level and distribution
- Initial temperature distribution
- Time dependent forcing functions
- Average power
- Inlet flow and temperature
- Saturation temperature

The major time-dependent output parameters are:

- Temperature distribution in fuel rod
- Fuel rod surface and gap conditions
- Energy in the fuel

D.2 Fuel Thermal Response Methodology

NSP currently uses TOODEE 2[1] version TOD78265. No modifications have been made to this version. The NSP methodology described in Reference 2 limits the use of the TOODEE 2 code to non-LOCA type transients.

KEY INPUT PARAMETERS

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ASSUMPTIONS

The inlet temperature is set equal to the saturation temperature during the transient in order to maximize fuel pin temperatures.

The vendor-supplied fuel temperature profile is used as the initial condition.

The cladding surface heat transfer coefficient, H_S, is input as a constant which is the lowest value calculated during the transient using the Sandberg, et al[4] correlation for low flow stable film boiling. This is an iterative process since H_S (Sandberg) is a function of film temperature, where

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The initial iteration is calculated using the results obtained from H_S = H (Groenveld 5.9) which is programmed internally in the code. The difference between the FSAR results and the final and initial iteration, calculated from the NSP methodology, for the Locked Rotor benchmark (Figure 3.12-6 of Reference 2) is shown in Figure D.1.

RESULTS

The NSP methodology provides a very conservative calculation of the transient hot spot fuel pin temperatures. The benchmarks shown in figures 3.12-6, 3.15-2 and 3.15-4 of Reference 2 show good comparison to the FSAR analyses.

APPENDIX D REFERENCES

1.) G. N. Lauben, "TOODEE-2: A Two-Dimensional Time Dependent Fuel Element Thermal Analysis Program," NUREG-75/057, May 1975.

2.) Prairie Island Nuclear Power Plant, "Reload Safety Evaluation Methods for Application to P.I. Units," NSPNAD-8102P.

3.) R. C. Kern, DYNODE-P version 3: A Nuclear Steam Supply System Transient Simulation for Pressurized Water Reactors - User's Manual," NAI 8-70 revision 0.

4.) Northern States Power Company, Prairie Island Nuclear Power Plant, "Final Safety Analysis Report."

5.) G. R. Horn, F. E. Panisko, "Users Guide for GAPCON: A Computer Program to Predict Fuel-to-Cladding Heat Transfer Coefficients in Oxide Fuel Pins," HEDL-TME72-128, September, 1972.

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Figure D- 1: Locked Rotor - Clad Temperature vs. Time

APPENDIX E

E.1 CONTEMPT-LT/026 Code (Reference 1)

The transient response of the containment during a main steamline break accident is analyzed with the CONTEMPT-LT program. CONTEMPT-LT can model a PWR dual dry containment with an annulus region. This program calculates the time variation of compartment pressures, temperatures, mass and energy inventories, heat structure temperature distributions, and energy exchange with adjacent compartments. Models are provided to describe fan cooler and cooling spray engineered safety systems.

The basic input parameters are:

- Containment compartment geometries and initial conditions.
- Heat structure geometries and material properties.
- Mass and energy addition rates.
- Engineered safety system description.
- Leakage rates.
- Heat transfer coefficients.

The major time-dependent output parameters are:

- Compartment pressure, temperature, mass and energy distributions.
- Heat structure temperature distributions.

E.2 Containment Analysis Methodology

The mass and energy release models in DYNODE-P will be used in conjunction with the CONTEMPT-LT/026 code to do containment pressure calculations for steam line break and feedwater line break transients.

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APPENDIX E REFERENCES

1. L. L. Wheat, et al. "CONTEMPT-LT: A Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident," ANCR-1219, June, 1975.
2. Northern States Power Company, Prairie Island Nuclear Power Plant Final Safety Analysis Report, Amendment 33, 4/9/73.

Table E- 1: Structural Heat Sinks*

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NOTE: Concrete structures inside containment **NOT** used in the calculation include:

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* The structural heat sink data has been revised since the FSAR calculations were performed. The original data is used in the benchmark calculations in this report, however the structural heat sink data will be updated to the current values for all future RSE calculations.

Table E- 2: Steam Generator Mass and Energy Blowdown

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Figure E- 1: Containment Fan Cooler Heat Removal Rate

APPENDIX F

F.1 VIPRE-01 Code

NSP has developed an alternate MDNBR analysis methodology based on the VIPRE-01 computer code. This new methodology will be used in place of the methodology based on COBRA-IIIC/MIT described in Appendix C.

The VIPRE-01 computer program (Reference F.1) was developed by Battelle, Pacific Northwest Laboratories under sponsorship of the Electric Power Research Institute (EPRI). VIPRE-01 has been written based on the strength of the COBRA code series. The COBRA-IIIC, COBRA-IV-I, and COBRA-WC codes were also developed by Battelle under sponsorship of the Atomic Energy Commission, the Nuclear Regulatory Commission and the Department of Energy. The COBRA-IIIC/MIT code was written by MIT under sponsorship of EPRI in order to be the thermal hydraulic part of the MEKIN coupled neutronics - thermal hydraulic code.

NSPNAD is replacing COBRA IIIC/MIT with VIPRE-01 for the following reasons: The large geometries used by NSPNAD can cause convergence problems in COBRA IIIC/MIT. VIPRE-01 has improved numerics which eliminates convergence problems for all practical purposes. VIPRE-01 also has many features which increase it's flexibility and ease of use compared to the COBRA codes. Of the new features, the multiple axial power profiles, gap dependent turbulent mixing, and iteration on chosen parameters to a specified MDNBR are the ones of most use for the NSPNAD work. VIPRE-01 is written in standard Fortran and has extensive documentation associated with it, both of which are definite improvements over the COBRA codes.

The basic computational processes of VIPRE-01 and COBRA-IIIC/MIT are similar. VIPRE-01 solves the conservation of mass, axial and lateral momentum, and energy equations for fluid enthalpy, axial flow rate, lateral flow per unit length, and momentum pressure drop. The flow field is assumed to be incompressible and homogeneous, although nonmechanistic models are included for subcooled boiling and liquid/vapor slip in two-phase flow. VIPRE-01 has an expanded choice of correlations for CHF, CPR, two-phase flow, and heat transfer, plus provisions for user coded correlations in each case. VIPRE-01 also includes many user oriented features such as multiple case options, the ability to simulate a wide variety of different geometries, free format input, and several dump/restart options.

The basic input parameters and time-dependent output parameters are similar to those of COBRA-IIIC/MIT (see Appendix C).

F.2 *Comparisons to COBRA-IIIC/MIT*

In order to assess the capabilities of the VIPRE-01 code, NSPNAD performed extensive benchmarks of VIPRE-01 to COBRA-IIIC/MIT. These benchmarks used the NRC approved thermal methodology described in Appendix C and cover both steady-state and transient simulations.

F.2.1 Steady-State Calculations

In order to assess the steady-state capabilities of VIPRE-01, the Prairie Island thermal Overtemperature safety limit curves have been reproduced using VIPRE-01. These curves were previously calculated using the COBRA-IIIC/MIT code (Reference F.2).

The thermal Overtemperature limit curves define the regions of acceptable operation with respect to average temperature, power, and pressurizer pressure. One of the boundaries of these regions is MDNBR > 1.3 (Fuel Damage Limit).

The operating conditions at which this condition is met are found at various pressures and power levels by iterating on inlet temperature until the MDNBR = 1.3. These points are then graphed at various pressures in terms of average temperature and power.

The curves at 1830, 2000, 2250, and 2500 psia have been reproduced. These curves are based on the Prairie Island Unit 1, Cycle 9 model in which an added 5% margin on DNB has been included; i.e. DNBR = 1.365. Figures F.1 and F.2 show the results graphically. As can be seen, VIPRE-01 matches COBRA-IIIC/MIT very well. This is true over the entire range of different pressures and power levels represented in these curves. In cases where there are minor differences, the VIPRE-01 results are slightly more conservative.

F.2.2 Transient Calculations

A series of transient calculations have been run to assess the transient capabilities of VIPRE-01. These cases are based on the limiting transients and accidents Prairie Island Unit 1 Cycle 9 analysis and include:

- Fast RCCA withdrawal from power,
- Turbine trip,
- 2/2 pump trip,
- Locked pump rotor.

This set of transients and accidents cover all the important DNBR related events with respect to power, flow, pressure, and subcooling boundary conditions.

The results of these cases were compared to similar COBRA-IIIC/MIT results. Figures F.3 through F.6 show these comparisons. The VIPRE-01 results show very good agreement with the COBRA-IIIC/MIT results. The VIPRE-01 results tend to be slightly more conservative for rapidly changing transients. This additional conservatism is because VIPRE-01 shows a slightly higher crossflow out of the hot channel, which leads to higher quality and lower critical heat fluxes. However, this effect is small and the conclusion to be drawn is that VIPRE-01 gives similar (or slightly conservative) results when compared to COBRA-IIIC/MIT in the transient mode.

F.2.3 Failed Pin Analysis

One of the FSAR acceptance criteria for the locked rotor transient is that the number of fuel rods calculated to experience DNB (and thus fail) should not exceed the number required to fail in order to exceed the release activity limits of 10 CFR 100. This analysis is performed by iterating on $F_{\Delta H}$ until the MDNBR limit is just met (1.3 for W-3). The number of failed pins is then determined by calculating how many pins are above this $F_{\Delta H}$ using a pin census.

The number of failed pins has been calculated for Prairie Island 1 Cycle 9 using both COBRA-IIIC/MIT and VIPRE-01. Table F-1 shows these results. VIPRE-01 calculates that an $F_{\Delta H}$ of 1.275 will give a MDNBR of 1.3, which converts to 7.1% of the pins experiencing DNB and failing. COBRA-IIIC/MIT gives an $F_{\Delta H}$ of 1.278 for a MDNBR of 1.3, which then gives 6.6% pins failing.

These two results are almost identical with VIPRE-01 being slightly more conservative.

F.2.4 Axial Power Shape Sensitivity

As part of a previous analysis to justify an increase in the Prairie Island $F_{\Delta H}$ limit (Reference F.2), NSPNAD examined the validity of the $f(\Delta I)$ function used on the Overtemperature ΔT trip setpoint. This function is needed, because the Westinghouse methodology for calculating the Overtemperature ΔT setpoint is based on an axial power profile with a zero axial offset (center peaked cosine).

The f(ΔI) function is used to lower the setpoint when highly skewed axial power shapes are encountered (large positive or negative axial offsets).

The NSPNAD methodology for determining the validity of the f(ΔI) function for current cycles involves using a variety of different axial power shapes in COBRA-IIIC/MIT. These axial power shapes are obtained during the normal design process and bound both normal and transient operation including Condition II events. These shapes include a wide range of both positive (up-skewed) and negative (down-skewed) axial offset conditions. The model with each of these power shapes is then iterated on power level until the MDNBR limit of 1.3 is met. The reactor conditions (average temperature, pressure, axial offset) where this limit is met are then input into the Overtemperature ΔT trip equation, including f(ΔI) penalty and uncertainties, to ensure that the reactor would trip before the DNBR limit is violated.

The Prairie Island Unit 1 Cycle 9 calculations for validating the f(ΔI) function have been repeated using the VIPRE-01 code.

Figure F.7 shows the calculated powers required at the various axial power distributions to yield a MDNBR of 1.365 (1.3 + 5% margin). Figure F.8 shows these various powers as a function of axial offsets. These graphs demonstrate that the two codes both predict similar MDNBR for similar axial shapes. This is true over a wide range of different axial shapes with no significant deviations at either highly up-skewed or down-skewed shapes. The VIPRE results tend to be slightly more conservative.

F.2.5 Conclusions

The NSPNAD benchmarks demonstrate that the VIPRE-01 code produces similar and slightly more conservative results to COBRA-IIIC/MIT and therefore VIPRE-01 is an acceptable tool for analyzing thermal margin at Prairie Island Units 1 and 2.

F.3 Use of the WRB-1 Critical Heat Flux Correlation

Westinghouse has developed a new critical heat flux correlation based exclusively on the large data base of mixing vane bundle CHF test results they have collected. The new correlation, called the WRB-1 ("Westinghouse Rod Bundle") correlation, has been reviewed and approved for use by the NRC (Reference F.3). This new correlation has a reactor design criterion on MDNBR of 1.17 which allows considerable margin improvement over the W-3 limit of 1.3.

NSP purchased the WRB-1 correlation as part of the current fuel contract. NSPNAD has coded the WRB-1 correlation into the VIPRE-01 code using the "User-Defined CHF Correlation" subroutine. This subroutine was included with the purpose of allowing users to easily add new CHF correlations to the code. This is another improvement over the COBRA series of codes.

In order to verify that the WRB-1 limit of 1.17 is still valid for use with the NSP coding in VIPRE-01, a series of Westinghouse CHF tests have been simulated by NSPNAD. These tests were designed specifically to model the 14 x 14 Westinghouse Improved Optimized fuel assembly used at Prairie Island.

The statistical results of NSP's benchmarking of these tests is shown in Table F-2.

A MDNBR 95/95 limit using these results is (using the method described in Reference F.3)

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This value is less than the 1.17 limit derived for the WRB-1 correlations by Westinghouse. This is expected since the 1.17 limit is based on a less conservative (M/P) average value of 1.0079 and a larger standard deviation of 0.0859. Therefore, the 1.17 limit with the NSP VIPRE-01/WRB-1 combination will be used as a conservative criterion for thermal margin.

The Westinghouse analysis of the same set of test conditions yielded a measured to predicted average of 1.0571 with a standard deviation of 0.0771. These results are in very good agreement to the Westinghouse results on similar fuel types and therefore demonstrates the ability of the WRB-1 to predict OFA fuel CHF.

The MDNBR will now be evaluated using the WRB-1 correlation instead of the W-3 correlation for Westinghouse Improved Optimized Fuel Assemblies (Imp.OFA). This means that wherever in Chapter 3 of this document the acceptance criteria for a particular transient is listed as a MDNBR of greater than 1.3 using the W-3 correlation, this is now replaced by a MDNBR of greater than 1.17 using the WRB-1 correlation for Westinghouse Imp. OFA.

F.4 Thermal Margin Methodology Using The VIPRE-01 Code

The use of the VIPRE-01 code has allowed improvements to be made to the NSPNAD thermal margin methodology. These improvements are made possible by certain features of the

VIPRE-01 code that are not available in COBRA-IIIC/MIT. These are mainly in the area of turbulent mixing and axial power shapes. Along with these methodology changes, the MDNBR will now be evaluated using the WRB-1 correlation instead of the W-3.

Radial Power Distribution

The radial power of the hottest fuel rod has been altered in the new methodology. In COBRA-IIIC/MIT, the FQ engineering factor (F_Q^E) was conservatively applied to the entire hot rod power. This was due to a code limitation which allows only one axial power shape to be applied to all fuel rods. In VIPRE-01 this engineering factor will be applied to the peak power node of the hottest fuel rod, as per its definition.

Axial Power Distribution

A sensitivity study has been performed on various axial power profiles with the WRB-1 correlation. This included a wide range of shapes possible under the power distribution control procedures used at Prairie Island. The FZ for each shape was calculated by the formula

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where K(Z) is calculated at the location of peak FZ from the relationship given in the Prairie Island Technical Specifications.

The results of the study are shown in Table F-3. The shape with the most positive axial offset is the most limiting. This is expected since the location of MDNBR is in the upper portion of the core. This axial shape will then be verified for each cycle to ensure that it bounds the axial offset conditions permitted by the Prairie Island power distribution control.

Axial shapes that are outside this envelope are accounted for in the f(ΔI) penalty applied to the Overtemperature ΔT and Overpower ΔT trip setpoints. Previously NSPNAD had ignored this function and used a conservative axial shape to bound all conditions. The new methodology takes credit for this trip function and limits the axial shapes to those permitted by the axial offset control at Prairie Island.

Turbulent Mixing

The effect of turbulent mixing is ignored in the current NSPNAD methodology using COBRA. This is because the assumptions used for the turbulent mixing models are designed for mixing between subchannels and are not adequate for mixing between lumped assemblies. VIPRE has the ability to specify turbulent mixing parameters for each channel connection individually. This will be used to specify no turbulent mixing between either lumped assemblies or lumped assemblies and subchannels, while turbulent mixing between subchannels will be modeled as

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The turbulent momentum factor, FTM, is set equal to 0.8. This value is recommended by the VIPRE manual.

Time Step Sensitivity Studies

A series of time step sensitivity studies have been run to determine the time step size appropriate for each transient. The sensitivity study has been performed on the following transients:

- Turbine Trip,
- Fast Rod Withdrawal,
- Slow Rod Withdrawal,
- Loss of Flow (2/2 Pump Trip).

The results of these studies are shown in Figures F.9 through F.12. The smallest time step results are graphed as solid lines. As can be seen, the MDNBR calculated at any particular time is essentially independent of the time step size chosen. The primary difference between time step sizes is degree of detail revealed for each transient and the amount of CPU time required. The optimal time step size is chosen for each transient based on consideration of these two items.

The time step size chosen for each transient is shown in Table F-4 along with the results of the sensitivity studies. As can be seen in the slow rod withdrawal results, the MDNBR calculated by modeling the transient as a series of steady state cases chosen at various time steps is conservative with respect to the transient calculations. This steady state calculational method is therefore acceptable for slowly changing transients, such as the slow rod withdrawal, dropped rod, main steam line break, etc.

Major Features

The major features of the revised model are presented in Table F-5.

APPENDIX F REFERENCES

1. EPRI NP-2511-LLM, V1-V3; VIPRE-01 Computer Code Manual," April 1983.

2. NSPNAD-8406, Revision 1, "Prairie Island Units 1 and 2, Safety Evaluation with Increased Enthalpy Rise Factor," September 1984.

3. WCAP-8762-P-A, "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," July 1984.

Table F- 1: Locked Rotor Failed Pin Analysis

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Table F- 2: VIPRE-01/WRB-1 - 4x4 OFA Test Results

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Table F- 3: Axial Power Shape Study

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Table F- 4: Time Step Sensitivity Results - MDNBR vs. Time Step Size

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Table F- 5: VIPRE-01 1/8 Core Model - Westinghouse Improved Optimized Fuel

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Figure F- 1: Thermal Overtemperature Limits

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Figure F- 2: Thermal Overtemperature Limits

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Figure F- 3: Prairie Island 1 Cycle 9 - Turbine Trip

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Figure F- 4: Prairie Island 1 Cycle 9 - Fast Rod Withdrawal

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Figure F- 5: Prairie Island 1 Cycle 9 - 2/2 Pump Trip

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Figure F- 6: Prairie Island 1 Cycle 9 - Locked Rotor

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Figure F- 7: Axial Power Profile Study

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Figure F- 8: Axial Power Profile Study

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Figure F- 9: Prairie Island 1 Cycle 9 - Turbine Trip

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Figure F- 10: Prairie Island 1 Cycle 9 - Slow Rod Withdrawal

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Figure F- 11: Prairie Island 1 Cycle 9 - Fast Rod Withdrawal

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Figure F- 12: Prairie Island 1 Cycle 9 - 2/2 Pump Trip

Appendix G

G.1 Introduction

This appendix describes the methodology for determining the minimum required Shutdown Margin (SDM) for a dilution accident for various system configurations and dilution flow rates with the reactor subcritical. Dilution accidents with the reactor critical are analyzed using methodology described in section 3 of this topical.

Section G.2 provides some background information on the dilution transient.

Section G.3 describes the methodology and assumptions used in determining the required SDM.

Section G.4 lists the acceptance criteria for the dilution accident.

Section G.5 contains an example of how the methodology may be utilized to generate the required SDM for various system configurations and dilution flow rate.

G.2 Background Information

Unborated water may be added to the reactor coolant system to increase core reactivity. If this happens inadvertently because of operator error or equipment malfunction, there is an unwanted increase in core reactivity and a decrease in shutdown margin. Termination of the unplanned dilution event relies on operator action to stop the unplanned dilution before the shutdown margin is eliminated. The length of time the operators have to recognize the event and terminate the dilution is dependant on the mass of the RCS being diluted, the dilution flow rate (typically referred to as charging flow), and the initial shutdown margin.

G.3 Methodology and Assumptions

The methodology used in determining the required SDM is based on the following equation for determining the boron concentration as a function of time for a fixed mass and a given dilution rate.

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The above equation is based on the assumptions that:

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Other key assumptions utilized in the methodology are:

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One of two approaches may be utilized to determine the required SDM for a given system configuration and injection flow rate.

Approach number 1:

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Approach number 2:

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G.4 *Acceptance Criteria*

The acceptance criteria for a dilution accident when the reactor is subcritical is the time between the initiation of the dilution and a complete loss of SDM must be greater than or equal to 24 minutes.

G.5 *Example of Application of Methodology*

This section provides an example of how the above methodology could be used to generate a table of required shutdown margins for various modes of operations and number of pumps in service. For this example the following modes of operations were analyzed with 1, 2, & 3 pumps running:

Mode 3 with the RCS temperature ≥ 520°F
Mode 3 with the RCS temperature ≥ 350°F but < 520°F
Mode 4 i.e. RCS temperature ≥ 200°F but < 350°F
Mode 5 i.e. RCS temperature < 200°F

The following assumptions and system conditions were used in this example:

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The following table lists the boron concentrations (in ppm) that correspond to various SDMs for the given RCS temperatures. All boron concentrations are for the limiting time in life (i.e. maximum boron concentration), zero Xenon concentration, and assume the highest worth rod remains fully withdrawn.

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For this example approach number 2 described in the previous section was utilized. The following table lists the assumed SDM for each condition being analyzed.

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The following table lists the calculated time between the initiation of the dilution and a complete loss of SDM listed in the previous table.

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Since all calculated times are larger than the acceptance criteria of 24 minutes, assumed SDMs are acceptable for use as required SDMs.

This example analyzed the limiting time in core life (i.e. maximum boron concentration) and thus, the required SDMs would be conservative for the entire cycle. If desirable, the methodology could be applied at additional exposure points generating tables or curves of required SDM versus exposures for any desirable plant configuration.

APPENDIX H

H.1 Changes Made In Revision 7 To This Report

The format of this document has changed slightly between Revisions 6 and 7. Most of this was due to the use of a different word processor which reformatted the entire document. For reference, alterations made in Revision 7 are summarized below.

Rev. 6 Page Number(s)	Rev. 7 Page Number(s)	Description of Change
1	1*	Revision number and date were updated.
2,3	2*,3*	Margins were adjusted.
4-12	4-12*	Provided additional sections in Table of Contents, adjusted titles of various figures (but not the actual figures).
13-14	13-14	Text was added to describe the addition of Appendix G.
47	49	Changed text to show VIPRE is used rather than COBRA.
50	52	The equation for Moderator Temperature Coefficient was corrected. An error introduced in Rev. 6 in the inequality for excore tilt was also corrected.
66, 67	69, 70*	Corrected the character for delta in the text.* Text was added to describe the addition of Appendix G.
68	71, 72	Clarification of shutdown margin.
84, 97, 133, 145	88, 101, 138, 150	Provided additional information for the Doppler Coefficient.
131	136	Corrected number of pumps assumed to trip in each figure.
133, 145	138, 150	Added Shutdown Margin to the Cycle Specific Parameters list and text describing it.
155	160	Correct typo on pressure at which steamline break occurs.
157	163	Pressure under (e) is corrected.
188	194	The equations for β_{eff} and $\Delta\rho_{EJECT}$ were corrected.
200	206	The equation for Boron Reactivity Coefficient was corrected.
257	264*	Corrected the units in two columns.
263	270*	Changed $f(\Delta)$ and Δ to $f(\Delta I)$ and ΔT.
n/a	293-298	Added Appendices G and H.
various	various*	Corrected spelling and grammar, added spaces to separate words, corrected subject / verb agreement, etc.

* No revision bars are provided for spelling and grammar corrections or text format changes.