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) or 1.17 (WRB-1), whichever is applicable~~, 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.

(W-3, when RCS pressure is > 1000 psia) or 1.45 (W-3, when RCS pressure is ≥ 500 psia but ≤ 1000 psia) (Reference 7).

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.~~

(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 Evaluation

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:

CYCLE SPECIFIC PARAMETER		SAFETY ANALYSIS PARAMETER
a. Keff(T_M)	$\leq$	Keff(T_M) bounding
b. α_D*(1-RF_D)	$\leq$	α_D (least negative bounding value)
c. α_B+B_B+RF_B	$\leq$	α_B (least negative bounding value)
d. SDM	$\geq$	SDM (bounding)
e. # of fuel pins above $F_{\Delta H}$ DNBR = 1.3 ~~(W-3)~~ (W-3, for RCS pressure > 1000 psia) or DNBR = ~~1.17 (WRB-1)~~ 1.45 (W-3, for RCS pressure $\geq$ 500 psia but $\leq$ 1000 psia (Reference 7)	$\leq$	20%

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, Novemoer 1, 1981.

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 Rev.4, March 26, 1984.

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".

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. 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 Evaluation

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:

CYCLE SPECIFIC PARAMETER		SAFETY ANALYSIS PARAMETER
a.	$K_{eff}(T_M)$	$\leq K_{eff}(T_M)$ bounding
b.	$\alpha_D*(1-RF_D)$	$\leq \alpha_D$ (least negative bounding value)
c.	$\alpha_B + B_B + RF_B$	$\leq \alpha_B$ (least negative bounding value)
d.	SDM	$\geq$ SDM (bounding)
e.	# of fuel pins above $F_{\Delta H}$ DNBR = 1.3 (W-3, for RCS Pressure > 1000 psia) or DNBR = 1.45 (W-3, for RCS pressure ≥ 500 psia but ≤1000 psia) (Reference 7)	≤ 20%

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, November 1, 1981.

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", NAP2.102? Rev.4, March 26, 1984.

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".