KEWAUNEE NUCLEAR POWER PLANT

RELOAD SAFETY EVALUATION
CYCLE 23
SEPTEMBER 1998

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RELOAD SAFETY EVALUATION

FOR

KEWAUNEE CYCLE 23

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1.0 SUMMARY

The Kewaunee Nuclear Power Plant is scheduled to shut down for the Cycle 22-23 refueling in October 1998. Startup of Cycle 23 is forecast for December 1998.

This report presents an evaluation of the Cycle 23 reload and demonstrates that the reload will not adversely affect the safety of the plant. Those accidents which could potentially be affected by the reload core design are reviewed.

Details of the calculational model used to generate physics parameters for this Reload Safety Evaluation are described in Reference 1 and Appendix A of this report. Accident Evaluation methodologies applied in this report are detailed in Reference 2. These reports have been previously reviewed and approved by the NRC as shown in References 3 and 4. The current physics model reliability factors are discussed in Section 5 of this report.

An evaluation, by accident, of the pertinent reactor parameters is performed by comparing the reload analysis results with the current bounding safety analysis values. The evaluations performed in this document employ the current Technical Specification (Reference 5) limiting safety system settings and operating limits as amended by Proposed Amendment 152 (Reference 10). Proposed Amendment 152 is required for full power operation of Reload Cycle 23.

It is concluded that the Cycle 23 design is more conservative than results of previously docketed accident analyses and implementation of this design will not introduce an unreviewed safety question since:

1. the probability of occurrence or the consequences of an accident will not be increased,

2. the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report will not be created and,

3. the margin of safety as defined in the basis for any technical specification will not be reduced.

This conclusion is based on the assumption that there is adherence to plant operating limitations and Technical Specifications (Reference 5) as amended by Proposed Amendment 152 (Reference 10) and Cycle 22 is shut down within a $\pm$500 MWD/MTU window of the nominal design End of Cycle (EOC) burnup of 16,500 MWD/MTU.

2.0 CORE DESIGN

2.1 Core Description

The reactor core consists of 121 fuel assemblies of 14 x 14 design. The core loading pattern, fuel assembly identification, and gadolinia loading for Cycle 23 are presented in Figure 2.1.1.

Table 2.1.1 displays the Cycle 23 fuel characteristics including region identification, initial enrichment, number of previous duty cycles, fuel rod design, grid design, and gadolinia loading. The SPC "heavy" fuel assemblies contain approximately 406 KgU (per assembly) versus approximately 378 KgU in the SPC standard fuel assembly design. Descriptions of the fuel designs are provided in References 6 and 11.

Fuel assemblies with two or three previous duty cycles are loaded on the core periphery flat region to lower power in that region and reduce reactor vessel fluence (Reference 7) in the critical reactor vessel locations. Fuel duty during this fuel cycle will assure peak fuel rod burnups less than the maximum burnup recommended by the fuel vendor. The Cycle 23 fuel loading pattern is capable of achieving a burnup of 17,256 MWD/MTU operating at full power, based on an end of Cycle 22 burnup of 16,500 MWD/MTU.

Table 2.1.1

Cycle 23 Fuel Characteristics

REGION	NUMBER OF ASSEMBLIES	NUMBER OF DUTY CYCLES	INITIAL U235 ENRICHMENT (GAD LOAD)	FUEL ROD DESIGN	GRID DESIGN
20	1	2	3.4	Standard	Bi-M
22	8	3	3.7	Standard	Bi-M
23	4	2	3.8	Standard	Bi-M
23	12	2	4.1	Standard	Bi-M
23	8	2	4.1	Heavy	HTP
24	24	1	4.1	Standard	HTP
24	12	1	4.5	Standard	HTP
24	8	1	4.5 (4 rods - 4%)	Heavy	HTP
25	8	0	4.1 (8 rods - 8%)	Heavy	HTP
25	12	0	4.1 (12 rods - 8%)	Heavy	HTP
25	8	0	4.5 (4 rods - 4%)	Heavy	HTP
25	8	0	4.5 (8 rods - 4%)	Heavy	HTP
25	8	0	4.5 (8 rods - 8%)	Heavy	HTP

Bi-M denotes the SPC bi-metallic grid design.

HTP denotes the SPC High Thermal Performance grid design.

Figure 2.1.1

Cycle 23 Loading Pattern

	1	2	3	4	5	6	7	8	9	10	11	12	13
A						Z23 3.7	A77 4.1	Z20 3.7					
B				B68 4.1	NEW-C3 4.5 4GAD4	NEW-C5 4.5 8GAD8	B84 4.5	NEW-C5 4.5 8GAD8	NEW-C3 4.5 4GAD4	B58 4.1			
C			B73 4.1	NEW-C4 4.5 8GAD4	B79 4.5	B94 4.5 4GAD4	NEW-C2 4.1 12GAD8	B89 4.5 4GAD4	B78 4.5	NEW-C4 4.5 8GAD4	B56 4.1		
D		B51 4.1	NEW-C4 4.5 8GAD4	A62 3.8	NEW-C1 4.1 8GAD8	A91 4.1	A81 4.1	A96 4.1	NEW-C1 4.1 8GAD8	A56 3.8	NEW-C4 4.5 8GAD4	B66 4.1	
E		NEW-C3 4.5 4GAD4	B75 4.5	NEW-C1 4.1 8GAD8	A72 4.1	NEW-C2 4.1 12GAD8	B63 4.1	NEW-C2 4.1 12GAD8	A88 4.1	NEW-C1 4.1 8GAD8	B81 4.5	NEW-C3 4.5 4GAD4	
F	Z21 3.7	NEW-C5 4.5 8GAD8	B87 4.5 4GAD4	A94 4.1	NEW-C2 4.1 12GAD8	B67 4.1	B69 4.1	B64 4.1	NEW-C2 4.1 12GAD8	A95 4.1	B93 4.5 4GAD4	NEW-C5 4.5 8GAD8	Z19 3.7
G	A85 4.1	B86 4.5	NEW-C2 4.1 12GAD8	A80 4.1	B52 4.1	B62 4.1	W29 3.4	B53 4.1	B54 4.1	A90 4.1	NEW-C2 4.1 12GAD8	B85 4.5	A88 4.1
H	Z18 3.7	NEW-C5 4.5 8GAD8	B90 4.5 4GAD4	A97 4.1	NEW-C2 4.1 12GAD8	B70 4.1	B65 4.1	B60 4.1	NEW-C2 4.1 12GAD8	A92 4.1	B88 4.5 4GAD4	NEW-C5 4.5 8GAD8	Z28 3.7
I		NEW-C3 4.5 4GAD4	B83 4.5	NEW-C1 4.1 8GAD8	A89 4.1	NEW-C2 4.1 12GAD8	B55 4.1	NEW-C2 4.1 12GAD8	A74 4.1	NEW-C1 4.1 8GAD8	B76 4.5	NEW-C3 4.5 4GAD4	
J		B72 4.1	NEW-C4 4.5 8GAD4	A65 3.8	NEW-C1 4.1 8GAD8	A98 4.1	A79 4.1	A93 4.1	NEW-C1 4.1 8GAD8	A69 3.8	NEW-C4 4.5 8GAD4	B71 4.1	
K			B61 4.1	NEW-C4 4.5 8GAD4	B82 4.5	B92 4.5 4GAD4	NEW-C2 4.1 12GAD8	B91 4.5 4GAD4	B77 4.5	NEW-C4 4.5 8GAD4	B74 4.1		
L				B57 4.1	NEW-C3 4.5 4GAD4	NEW-C5 4.5 8GAD8	B80 4.5	NEW-C5 4.5 8GAD8	NEW-C3 4.5 4GAD4	B59 4.1			
M						Z17 3.7	A84 4.1	Z15 3.7					

PRELIMINARY CYCLE TWENTY-THREE

ASSEMBLY ID
INITIAL ENRICHMENT
GADOLINIA LOADING

2.2 Operating Parameters and Design Limits

Cycle 23 core design is based on the following operating conditions and limits.

2.2.1 Operating Conditions

- Power Rating (MWTH)........................ 1650
- System Pressure (PSIA) 2250
- Core Average Moderator Temperature, HZP (°F) 547
- Core Average Moderator Temperature, HFP (°F) 562

2.2.2 Operating Limits

A. Nuclear peaking factor limits are as follows:

(i) FQ(Z) limits

a) For SPC heavy fuel:

$FQ(Z) \leq (2.35/P) * K(Z)$ for $P > 0.5$
$FQ(Z) \leq 4.70 * K(Z)$ for $P \leq 0.5$

b) For SPC standard fuel:

$FQ(Z) \leq (2.28/P) * K(Z)$ for $P > 0.5$
$FQ(Z) \leq 4.56 * K(Z)$ for $P \leq 0.5$

K(Z) is the function given in Figure TS 3.10-2 of Reference 5 and Z is the core height.

(ii) FΔH limits

a) For SPC heavy fuel: $F\Delta H \leq 1.70 * (1 + 0.2 * (1-P))$

b) For SPC standard fuel: $F\Delta H \leq 1.55 * (1 + 0.2 * (1-P))$

P is the fraction of full power at which the core is operating. A mixed core thermal hydraulic penalty has been evaluated (References 8 and 9) for the SPC standard Bi-M fuel assemblies.

B. The moderator temperature coefficient at operating conditions shall be less than +5.0 pcm/°F for $0\% \leq P \leq 60\%$, shall be negative for $P > 60\%$, and shall be less than -8.0 pcm/°F for 95% of the time at hot full power (Reference 5).

C. With the most reactive rod stuck out of the core, the remaining control rods shall be able to shut down the reactor by a sufficient reactivity margin:

1.0% at Beginning of Cycle (BOC)

2.0% at End of Cycle (EOC)

D. The power dependent rod insertion limits (PDIL) are presented in Figure TS 3.10-3 of Reference 5.

E. The indicated axial flux difference shall be maintained within a $\pm$ 5% band about the target axial flux difference above 90 percent power. Figure TS 3.10-5 of Reference 5 shows the axial flux difference limits as a function of core power. Reference 5 also provides limits on temporary operation allowed within the 3.10.b.11.a. line envelope (see Figure TS 3.10-5 of Reference 5) at power levels between 50 percent and 90 percent.

F. At refueling conditions a boron concentration of 2100 ppm will be sufficient to maintain the reactor subcritical by 5 percent $\Delta k/k$ with all rods inserted and will maintain the core subcritical with all rods out.

2.3 Scram Worth Insertion Rate

The most limiting scram curve is that curve which represents the slowest trip reactivity insertion rate normalized to the minimum shutdown margin. The Cycle 23 minimum shutdown margin is 2.23 percent at end of cycle hot full power conditions. Figure 2.3.1 compares the Cycle 23 minimum scram insertion curve to the current bounding safety analysis curve.

It is concluded that the minimum trip reactivity insertion rate for Cycle 23 is conservative with respect to the bounding value. Thus, for accidents in which credit is taken for a reactor trip, the proposed reload core will not adversely affect the results of the safety analysis due to trip reactivity assumptions.

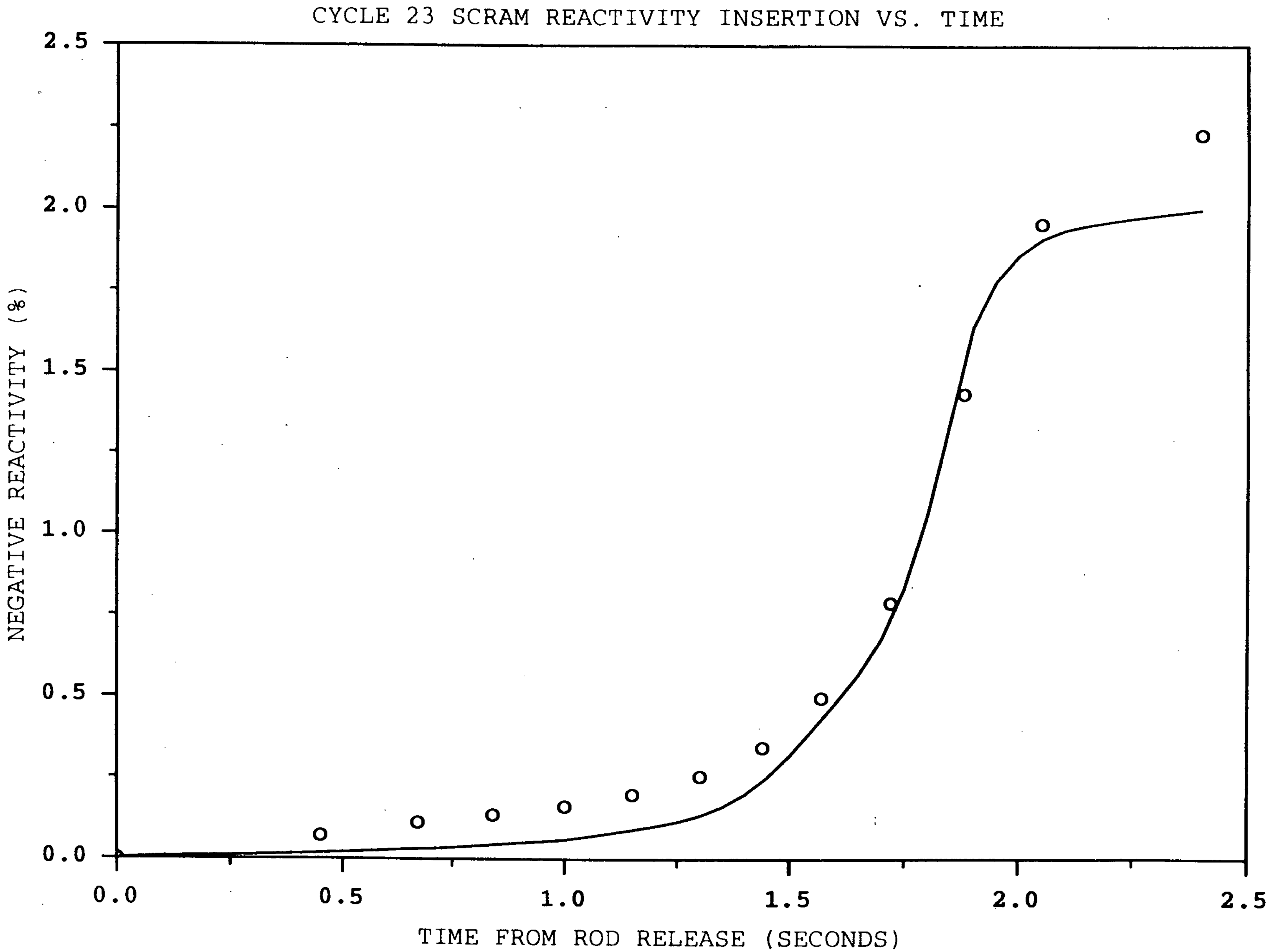
CYCLE 23 SCRAM REACTIVITY INSERTION VS. TIME
NEGATIVE REACTIVITY (%)
2.5
2.0
1.5
1.0
0.5
0.0
0.0
0.5
1.0
1.5
2.0
2.5
TIME FROM ROD RELEASE (SECONDS)

Figure 2.3.1

2.4 Shutdown Window

An evaluation of the maximum full power equilibrium peaking factors versus EOC 22 burnup is presented in Table 2.4.1. The values shown have conservatisms applied in accordance with Reference 1.

It is concluded that if the shutdown of Cycle 22 occurs within the burnup window, the Cycle 23 peaking factors will not be significantly affected and will not exceed their limiting values.

Table 2.4.1

Peaking Factor Versus Shutdown Burnup

	FΔHN#		FQ@	
	Cycle 23	**Limit**	**Cycle 23**	**Limit**
EOC 22 - 500 MWD/MTU	1.59	1.70	2.25	2.35
EOC 22 Nominal	1.59	1.70	2.26	2.35
EOC 22 + 500 MWD/MTU	1.60	1.70	2.27	2.35

\# The peak FΔH occurred in an SPC Heavy assembly. All SPC Standard assemblies met the SPC Standard FΔH limit of 1.55.

@ The peak FQ occurred in an SPC Heavy assembly. All SPC Standard assemblies met the SPC Standard FQ limit of 2.28.

2.5 Moderator Temperature Coefficient

An evaluation of the Cycle 23 hot full power moderator temperature coefficient is presented in Table 2.5.1. The calculated Cycle 23 value at Beginning of Cycle (BOC) is compared to the MTC upper bound limit of -8.0 pcm/°F. Cycle 23 MTC must be less than the upper bound limit for 95% of the scheduled time at HFP due to anticipated transient without scram (ATWS) concerns. Since MTC is less than the limit at BOC, and becomes increasingly negative with cycle exposure, it will be less than the upper bound limit for 95% of scheduled time at HFP. It is concluded that the Cycle 23 MTC is conservative with respect to the bounding value. Therefore, the Cycle 23 core will not adversely affect the results of the ATWS safety analysis.

Table 2.5.1

Moderator Temperature Coefficient

Reload Safety Evaluation Value		**Current Safety Analysis**	**Units**
-12.2	≤	-8.0	pcm/°Fm

3.0 ACCIDENT EVALUATIONS

Table 3.0.1 presents the latest safety analyses performed for the accidents which are evaluated in Sections 3.1 through 3.16 of this report. The bounding values derived from these analyses are shown in Table 3.0.2 and will be applied in the Cycle 23 accident evaluations.

Table 3.0.1

Kewaunee Nuclear Power Plant

List of Current Safety Analyses

Accident	Current Safety Analysis
Uncontrolled RCCA Withdrawal From a Subcritical Condition	Ref. 6, 10
Uncontrolled RCCA Withdrawal at Power	Ref. 6, 10
Control Rod Drop	Ref. 6, 10
RCC Assembly Misalignment	Ref. 6, 10
CVCS Malfunction	Ref. 6, 10
Startup of an Inactive RC Loop	Ref. 6, 10
Excessive Heat Removal Due to FW System Malfunctions	Ref. 6, 10
Excessive Load Increase Incident	Ref. 6, 10
Loss of Reactor Coolant Flow Due to Pump Trip Due to Underfrequency Trip	Ref. 6, 10
Locked Rotor Accident	Ref. 6, 10
Loss of External Electrical Load	Ref. 6, 10
Loss of Normal Feedwater	Ref. 6, 10
Fuel Handling Accident	Ref. 6, 10
Rupture of a Steam Pipe	Ref. 6, 10
Rupture of CR Drive Mechanism Housing	Ref. 6, 10
Large Break LOCA	Ref. 6, 10
Small Break LOCA	Ref. 6, 10

Table 3.0.2

Safety Analyses Bounding Values

Parameter	Lower Bound	Upper Bound	Units
Moderator Temp. Coefficient			
Most Negative	-40.0	---	pcm/°Fm
$0 \leq P \leq 60\%$	---	+5.0	pcm/°Fm
$P > 60\%$	---	0.0	pcm/°Fm
95% of time at HFP	---	-8.0	pcm/°Fm
URW from subcritical	---	+10.0	pcm/°Fm
Doppler Coefficient	-2.32	-1.0	pcm/°Ff
Differential Boron Worth	-11.2	-7.25	pcm/ppm
Delayed Neutron Fraction	.00485	.00706	---
Prompt Neutron Lifetime	15	N/A	μsec
Shutdown Margin	1.0 (BOC) 2.0 (EOC)	N/A N/A	% $\Delta\rho$
Differential Rod Worth of 2 Banks Moving	N/A	82	pcm/sec
Ejected Rod Cases			
HFP, BOL			
ßeff	.0055	N/A	---
Rod Worth	N/A	.30	% $\Delta\rho$
FQ	N/A	5.03	---
HFP, EOL			
ßeff	.0050	N/A	---
Rod Worth	N/A	.42	% $\Delta\rho$
FQ	N/A	4.6	---
HZP, BOL			
ßeff	.0055	N/A	---
Rod Worth	N/A	.91	% $\Delta\rho$
FQ	N/A	8.2	---
HZP, EOL			
ßeff	.0050	N/A	---
Rod Worth	N/A	.92	% $\Delta\rho$
FQ	N/A	12.8	---

3.1 Evaluation of Uncontrolled Rod Withdrawal from Subcritical

An uncontrolled addition of reactivity due to uncontrolled withdrawal of a Rod Cluster Control Assembly (RCCA) results in a power excursion.

The most important parameters are the reactivity insertion rate and the doppler coefficient. A maximum reactivity insertion rate produces a more severe transient while a minimum (absolute value) doppler coefficient maximizes the nuclear power peak. Of lesser concern are the moderator coefficient and delayed neutron fraction which are chosen to maximize the peak heat flux.

Table 3.1.1 presents a comparison of Cycle 23 physics parameters to the current safety analysis values for the Uncontrolled Rod Withdrawal from a Subcritical Condition.

Since the pertinent parameters from the proposed Cycle 23 reload core are conservatively bounded by those used in the current safety analysis, an uncontrolled rod withdrawal from subcritical accident will be less severe than the transient in the current safety analysis. Therefore, the implementation of the Cycle 23 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.1.1

Uncontrolled Rod Withdrawal From Subcritical

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-1.96	≤	10.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.32	≤	-1.0	pcm/°Ff
C) Differential Rod Worth of Two Moving Banks	.113	≤	.116	$/sec
D) Scram Worth vs. Time	See Section 2.3			
E) Delayed Neutron Fraction	.00651	≤	.00706	---
F) Prompt Neutron Lifetime	24	≥	15	μsec

3.2 Evaluation of Uncontrolled Rod Withdrawal at Power

An uncontrolled control rod bank 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 minimum absolute value of the doppler and moderator coefficients serves to maximize peak neutron power, while the delayed neutron fraction is chosen to maximize peak heat flux.

Table 3.2.1 presents a comparison of the Cycle 23 physics parameters to the current safety analysis values for the Uncontrolled Rod Withdrawal at Power Accident.

Since the pertinent parameters from the proposed Cycle 23 reload core are conservatively bounded by those used in the current safety analysis, an uncontrolled rod withdrawal at power accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 23 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.2.1

Uncontrolled Rod Withdrawal at Power

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-7.22	≤	0.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.32	≤	-1.0	pcm/°Ff
C) Differential Rod Worth of Two Moving Banks	.113	≤	.116	$/sec
D) FΔHN	1.59*	≤	1.70	---
E) Scram Worth vs. Time	See Section 2.3			
F) Delayed Neutron Fraction	.00651	≤	.00706	---

* The peak FΔHN occurred in an SPC Heavy assembly. All SPC Standard assemblies met the SPC Standard FΔH limit of 1.55.

3.3 Evaluation of Control Rod Misalignment

The static misalignment of an RCCA from its bank position does not cause a system transient; however, it does cause an adverse power distribution which is analyzed to show that core Departure from Nuclear Boiling Ratio (DNBR) limits are not exceeded.

The limiting core parameter is the peak FΔH in the worst case misalignment of Bank D fully inserted with one of its RCCAs fully withdrawn at full power.

Table 3.3.1 presents a comparison of the Cycle 23 FΔHN versus the current safety analysis FΔH limit for the Control Rod Misalignment Accident.

Since the pertinent parameter from the proposed Cycle 23 reload core is conservatively bounded by that used in the current safety analysis, a control rod misaligument accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 23 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.3.1

Control Rod Misalignment

Parameter	Reload Safety Evaluation Value		Current Safety Analysis
A) FΔHN	1.98	≤	2.02

3.4 Evaluation of Dropped Rod

The release of a full length control rod or control rod bank by the gripper coils while the reactor is at power causes the reactor to become subcritical and produces a mismatch between core power and turbine demand. The dropping of any control rod bank will produce a negative neutron flux rate trip with no resulting decrease in thermal margins. Dropping of a single RCCA or several RCCA's from the same bank may or may not result in a negative rate trip, and therefore the radial power distribution must be considered.

Table 3.4.1 presents a comparison of the Cycle 23 physics parameters to the current safety analysis values for the Dropped Rod Accident.

Since the pertinent parameters from the proposed Cycle 23 reload core are conservatively bounded by that used in the current safety analysis, a dropped rod accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 23 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.4.1

Dropped Rod

Parameter	Reload Safety Evaluation Value		Current Safety Analysis	Units
A) FΔHN	1.99	≤	2.02*	---

* Limit is 1.85 for SPC Standard fuel with Bi-M spacers. All Cycle 23 SPC Standard fuel with Bi-M spacers met the 1.85 limit.

3.5 Evaluation of Uncontrolled Boron Dilution

The malfunction of the Chemical and Volume Control System (CVCS) is assumed to deliver unborated water to the Reactor Coolant System (RCS).

Although the boron dilution rate and shutdown margin are the key parameters in this event, additional parameters are evaluated for the manual reactor control case. In this case core thermal limits are approached and the transient is terminated by a reactor trip on over-temperature ΔT.

Table 3.5.1 presents a comparison of Cycle 23 physics analysis results to the current safety analysis values for the Uncontrolled Boron Dilution Accident for refueling, full power, and startup conditions.

Since the pertinent parameters from the proposed Cycle 23 reload core are conservatively bounded by those used in the current safety analysis, an uncontrolled boron dilution accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 23 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.5.1

Uncontrolled Boron Dilution

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
i) Refueling Conditions				
A) Shutdown Margin	6.1	≥	5.0	%
ii) At-Power Conditions				
A) Moderator Temp. Coefficient	-7.22	≤	0.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.32	≤	-1.0	pcm/°Ff
C) Reactivity Insertion Rate by Boron	.0017	≤	.0023	$/sec
D) Shutdown Margin	2.23	≥	1.0	%
E) FΔHN	1.59*	≤	1.70	---
F) Delayed Neutron Fraction	.00651	≤	.00706	---
iii) Startup Conditions				
A) Critical Boron Concentration (ARI)	1113	≤	1300	ppm

* See footnote on Page 20

3.6 Evaluation of Startup of an Inactive Loop

The startup of an idle reactor coolant pump in an operating plant would result in the injection of cold water from the idle loop hot leg into the core which causes a rapid reactivity insertion and subsequent core power increase.

The moderator temperature coefficient is chosen to maximize the reactivity effect of the cold water injection. Doppler temperature coefficient is chosen conservatively low (absolute value) to maximize the nuclear power rise. The power distribution ($F\Delta H$) is used to evaluate the core thermal limit acceptability.

Table 3.6.1 presents a comparison of the Cycle 23 physics calculation results to the current safety analysis values for the Startup of an Inactive Loop Accident.

Since the pertinent parameters from the proposed Cycle 23 reload core are conservatively bounded by those used in the current safety analysis, the startup of an inactive loop accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 23 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.6.1

Startup of an Inactive Loop

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-34.3	≥	-40.0	pcm/°Fm
B) Doppler Coefficient	-1.89	≤	-1.0	pcm/°Ff
C) FΔHN	1.59*	≤	1.70	---

* See footnote on Page 20

3.7 Evaluation of Feedwater System Malfunction

The malfunction of the feedwater system such that the feedwater temperature is decreased or the flow is increased causes a decrease in the RCS temperature and an attendant increase in core power level due to negative reactivity coefficients and/or control system action.

Minimum and maximum moderator coefficients are evaluated to simulate both BOC and EOC conditions. The doppler reactivity coefficient is chosen to maximize the nuclear power peak.

A comparison of Cycle 23 physics calculation results to the current safety analysis values for the Feedwater System Malfunction Accident is presented in Table 3.7.1.

Since the pertinent parameters from the proposed Cycle 23 reload core are conservatively bounded by those used in the current safety analysis, a feedwater system malfunction will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 23 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.7.1

Feedwater System Malfunction

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
i) Beginning of Cycle				
A) Moderator Temp. Coefficient	-7.22	≤	0.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.32	≤	-1.0	pcm/°Ff
ii) End of Cycle				
A) Moderator Temp. Coefficient	-30.15	≥	-40.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.34	≤	-1.0	pcm/°Ff
iii) Beginning and End of Cycle				
C) FΔHN	1.59*	≤	1.70	---

* See footnote on Page 20

3.8 Evaluation of Excessive Load Increase

An excessive load increase causes a rapid increase in steam generator steam flow. The resulting mismatch between core heat generation and secondary side load demand results in a decrease in reactor coolant temperature which causes a core power increase due to negative moderator feedback and/or control system action.

This event results in a similar transient as that described for the feedwater system malfunction and is therefore sensitive to the same parameters.

Table 3.8.1 presents a comparison of Cycle 23 physics results to the current safety analysis values for the Excessive Load Increase Accident.

Since the pertinent parameters from the proposed Cycle 23 reload core are conservatively bounded by those used in the current safety analysis, an excessive load increase accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 23 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.8.1

Excessive Load Increase

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
i) Beginning of Cycle				
A) Moderator Temp. Coefficient	-7.22	≤	0.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.32	≤	-1.0	pcm/°Ff
ii) End of Cycle				
A) Moderator Temp. Coefficient	-30.15	≥	-40.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.34	≤	-1.0	pcm/°Ff
iii) Beginning and End of Cycle				
C) FΔHN	1.59*	≤	1.70	---

* See footnote on Page 20

3.9 Evaluation of Loss of Load

A loss of load is encountered through a turbine trip or complete loss of external electric load. To provide a conservative assessment of this event, no credit is taken for direct turbine/reactor trip, steam bypass, or pressurizer pressure control, and the result is a rapid rise in steam generator shell side pressure and reactor coolant system temperature.

Minimum and maximum moderator coefficients are evaluated to simulate both BOC and EOC conditions. The doppler reactivity coefficient is chosen to maximize the nuclear power and heat flux transient. The power distribution (FΔH) and scram reactivity are evaluated to ensure thermal margins are maintained by the reactor protection system.

A comparison of Cycle 23 physics parameters to the current safety analysis values for the Loss of Load Accident is presented in Table 3.9.1.

Since the pertinent parameters from the proposed Cycle 23 reload core are conservatively bounded by those used in the current safety analysis, a loss of load accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 23 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.9.1

Loss of Load

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
i) Beginning of Cycle				
A) Moderator Temp. Coefficient	-7.22	≤	0.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.62	≥	-2.32	pcm/°Ff
ii) End of Cycle				
A) Moderator Temp. Coefficient	-30.15	≥	-40.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.64	≥	-2.32	pcm/°Ff
iii) Beginning and End of Cycle				
C) FΔHN	1.59*	≤	1.70	---
D) Scram Worth Versus Time	See Section 2.3			

* See footnote on Page 20

3.10 Evaluation of Loss of Normal Feedwater

A complete loss of normal feedwater is assumed to occur due to pump failures or valve malfunctions. An additional conservatism is applied by assuming the reactor coolant pumps are tripped, further degrading the heat transfer capability of the steam generators. When analyzed in this manner, the accident corresponds to a loss of offsite power.

The short term effects of the transient are covered by the Loss of Flow Evaluation (Section 3.11), while the long term effects, driven by decay heat, and assuming auxiliary feedwater additions and natural circulation RCS flow, have been shown not to produce any adverse core conditions.

The Loss of Feedwater Transient is not sensitive to core physics parameters and therefore no comparisons will be made for the Reload Safety Evaluation.

3.11 Evaluation of Loss of Reactor Coolant Flow Due to Pump Trip

The simultaneous loss of power or frequency decay in the electrical buses feeding the reactor coolant pumps results in a loss of driving head and a flow coast down. The effect of reduced coolant flow is a rapid increase in core coolant temperature. The reactor is tripped by one of several diverse and redundant signals before thermal hydraulic conditions approach those which could result in fuel damage.

The doppler temperature coefficient is compared to the most negative value since this results in the slowest neutron power decay after trip. The moderator temperature coefficient is least negative to cause a larger power rise prior to the trip. Trip reactivity and $F\Delta H$ are evaluated to ensure core thermal margin.

Table 3.11.1 presents a comparison of Cycle 23 calculated physics parameters to the current safety analysis values for the Loss of Reactor Coolant Flow Due to Pump Trip Accident.

Since the pertinent parameters from the proposed Cycle 23 reload core are conservatively bounded by those used in the current safety analysis, a loss of reactor coolant flow due to pump trip accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 23 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.11.1

Loss of Reactor Coolant Flow Due to Pump Trip

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-7.22	≤	0.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.64	≥	-2.32	pcm/°Ff
C) FΔHN	1.59*	≤	1.70	---
D) Scram Worth Versus Time	See Section 2.3			
E) Fuel Temperature	2092	≤	2135	°F

* See footnote on Page 20

3.12 Evaluation of Loss of Reactor Coolant Flow Due to Locked Rotor

This accident is an instantaneous seizure of the rotor of a single reactor coolant pump resulting in a rapid flow reduction in the affected loop. The sudden decrease in flow results in DNB in some fuel rods.

The minimum (absolute value) moderator temperature coefficient results in the least reduction of core power during the initial transient. The large negative doppler temperature coefficient causes a slower neutron flux decay following the trip as does the large delayed neutron fraction.

Table 3.12.1 presents a comparison of Cycle 23 physics parameters to the current safety analysis values for the Locked Rotor Accident.

Since the pertinent parameters from the proposed Cycle 23 reload core are conservatively bounded by those used in the current safety analysis, a locked rotor accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 23 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.12.1

Loss of Reactor Coolant Flow Due to Locked Rotor

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-7.22	≤	0.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.64	≥	-2.32	pcm/°Ff
C) Delayed Neutron Fraction	.00651	≤	.00706	---
D) Percent Pins > Limiting FΔHN (DNBR=1.14)	9.2*	≤	40.0	%
E) Scram Worth Versus Time	See Section 2.3			
F) FQ	2.27	≤	2.35	---
G) Fuel Temperature	2092	≤	2135	°F

* All of the pins that exceeded their Limiting FΔHN were in SPC Heavy fuel. No SPC standard fuel pins exceeded their Limiting FΔHN. The SPC standard fuel Limiting FΔHN corresponds to a DNBR of 1.30.

3.13 Evaluation of Main Steam Line Break

The break of a main steam line inside containment at the exit of the steam generator causes an uncontrolled steam release and a reduction in primary system temperature and pressure. The negative moderator coefficient produces a positive reactivity insertion and a potential return to criticality after the trip. The doppler coefficient is chosen to maximize the power increase.

Shutdown margin at the initiation of the cooldown and reactivity insertion and peak rod power ($F\Delta H$) during the cooldown are evaluated for this event. The ability of the safety injection system to insert negative reactivity and reduce power is minimized by using the least negative boron worth coefficient.

Table 3.13.1 presents a comparison of Cycle 23 calculated physics parameters to the current safety analysis values for the main steam line break accident. Figure 3.13.1 compares core Keff during the cooldown to the current bounding safety analysis curve.

Since the pertinent parameters from the proposed Cycle 23 reload core are conservatively bounded by those used in the current safety analysis, a main steam line break accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 23 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.13.1

Main Steam Line Break

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Shutdown Margin	2.23	≥	2.00	%Δρ
B) FΔH	5.24	≤	5.25	---
C) Doppler Temp. Coefficient	-1.32	≤	-1.0	pcm/°Ff
D) Boron Worth Coefficient	-7.32	≤	-7.25	pcm/ppm

CYCLE 23 MAIN STEAM LINE BREAK
KEFFECTIVE VS. TEMPERATURE

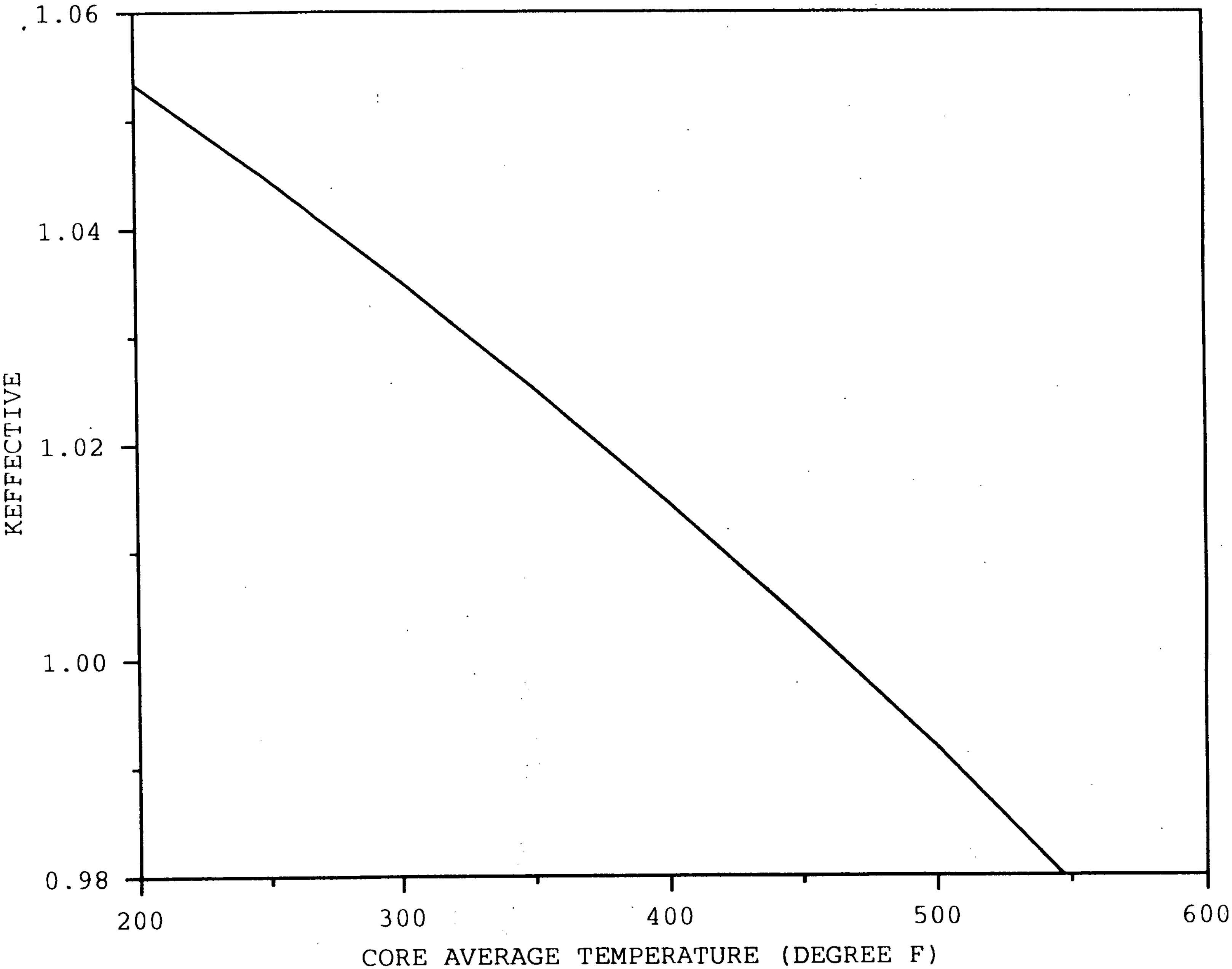

Figure 3.13.1

3.14 Evaluation of Rod Ejection Accidents

The ejected rod accident is defined as a failure of a control rod drive pressure housing followed by the ejection of a RCCA by the reactor coolant system pressure.

Tables 3.14.1 through 3.14.4 present the comparison of Cycle 23 calculated physics parameters to the current safety analysis values for the Rod Ejection Accident at zero and full power, BOC and EOC core conditions.

Since the pertinent parameters from the proposed Cycle 23 reload core are conservatively bounded by those used in the current safety analysis, a rod ejection accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 23 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.14.1

Rod Ejection Accident at

HFP, BOC

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-7.22	≤	0.0	pcm/°Fm
B) Delayed Neutron Fraction	.00613	≥	.00550	---
C) Ejected Rod Worth	.07	≤	0.30	%Δρ
D) Doppler Temp. Coefficient	-1.32	≤	-1.0	pcm/°Ff
E) Prompt Neutron Lifetime	24	≥	15	μsec
F) FQN	2.38	≤	5.03	---
G) Scram Worth Versus Time	See Section 2.3			

Table 3.14.2

Rod Ejection Accident at

HZP, BOC

Parameter	Reload Safety Evaluatiou Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-1.96	≤	5.0	pcm/°Fm
B) Delayed Neutron Fraction	.00613	≥	.00550	---
C) Ejected Rod Worth	0.53	≤	0.91	%Δρ
D) Doppler Temp. Coefficient	-2.10	≤	-1.0	pcm/°Ff
E) Prompt Neutron Lifetime	24	≥	15	μsec
F) FQN	5.5	≤	8.2	---
G) Scram Worth Versus Time	See Section 2.3			

Table 3.14.3

Rod Ejection Accident at

HFP, EOC

Parameter	Reload Safety Evaluatiou Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-22.55	≤	0.0	pcm/°Fm
B) Delayed Neutron Fraction	.00527	≥	.00500	---
C) Ejected Rod Worth	0.13	≤	0.42	%Δρ
D) Doppler Temp. Coefficient	-1.34	≤	-1.0	pcm/°Ff
E) Prompt Neutron Lifetime	27	≥	15	μsec
F) FQN	3.1	≤	4.6	---
G) Scram Worth Versus Time	See Section 2.3			

Table 3.14.4

Rod Ejection Accident at

HZP, EOC

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-17.0	≤	5.0	pcm/°Fm
B) Delayed Neutron Fraction	.00527	≥	.00500	---
C) Ejected Rod Worth	0.80	≤	0.92	%Δρ
D) Doppler Temp. Coefficient	-2.72	≤	-1.0	pcm/°Ff
E) Prompt Neutron Lifetime	27	≥	15	μsec
F) FQN	9.8	≤	12.8	---
G) Scram Worth Versus Time	See Section 2.3			

3.15 Evaluation of Fuel Handling Accident

This accident is the sudden release of the gaseous fission products held within the fuel cladding of one fuel assembly. The fraction of fission gas released is based on a conservative assumption of high power in the fuel rods.

Table 3.15.1 presents a comparison of the maximum Cycle 23 FΔHN to the current safety analysis FΔHN limit for the Fuel Handling Accident.

Since the pertinent parameter from the proposed Cycle 23 reload core is conservatively bounded by that used in the current safety analysis, a fuel handling accident will be less severe than the accident in the current analysis. Therefore, the implementation of the Cycle 23 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.15.1

Fuel Handling Accident

Parameter	Reload Safety Evaluation Values		Current Safety Analysis
A) FΔHN	1.59	≤	1.70

3.16 Evaluation of Loss of Coolant Accident

The Loss of Coolant Accident (LOCA) 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.

The principal reload design parameters which affect the results of LOCA analysis are shown in Table 3.16.1. Table 3.16.1 presents the comparison of Cycle 23 physics calculation results to the current safety analysis values for the Loss of Coolant Accident.

Since the pertinent parameters from the proposed Cycle 23 reload core are conservatively bounded by those used in the current safety analysis, a loss of coolant accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 23 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.16.1

Loss of Coolant Accident

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A. FQ	2.27	≤	2.35@	
B. FΔH	1.59*	≤	1.70	
C. Fuel Features	SPC Heavy, HTP	=	SPC Heavy, HTP#	
D. Max No. of Non-Uranium Rods	Zero	≤	Zero	
E. Fuel Design For Max. Fuel Ave. Temp.	SPC Heavy, HTP	=	SPC Heavy, HTP#	
F. Max. Assy. Ave. Peaking Factor ($P\tilde{H}A$)	1.422	≤	1.514	
G. Fuel Design For Max. Core Power Deposited in Fuel	SPC Heavy, HTP	=	SPC Heavy, HTP#	
H. Most Negative Axial Offset at 100% Power	-9.5	≥	-30.0	%
I. Most Positive Axial Offset at 100% Power	4.9	≤	+13.0	%
J. Max. Core Ave. Power in Lower Power Assy. Before 1500 MWD/MTU	0.45	≤	0.45	
K. Max. Core Ave. Power in Lower Power Assy. Beyond 1500 MWD/MTU	0.50	≤	0.60	
L. Max 95/95 Power for the Hot Rod	14.16	≤	14.66	kw/ft

@ 2.28 for SPC Standard fuel. See FQ footnote on Page 11.

* See footnote on Page 20.

Transition core effects for non-feed SPC Standard fuel have been evaluated.

3.17 Power Distribution Control Verification

The total peaking factor FQT relates the maximum local power density to the core average power density. The FQT is determined by both the radial and axial power distributions. The radial power distribution is relatively fixed by the core loading pattern design. The axial power distribution is controlled by the Technical Specifications (Reference 5).

FQT(Z) are determined by calculations performed at full power, equilibrium core conditions, at exposures ranging from BOC to EOC. Conservative factors which account for potential power distribution variations allowed by the power distribution control specifications, manufacturing tolerances, and measurement uncertainties are applied to the calculated FQT(Z).

Figure 3.17.1 compares the calculated FQT(Z), including uncertainty factors, to the FQT(Z) limits. These results demonstrate that the power distributions expected during Cycle 23 operation will not preclude full power operation under the power distribution control specifications currently applied (Reference 5).

CYCLE 23 MAX(FQ * P) VS. AXIAL CORE HEIGHT

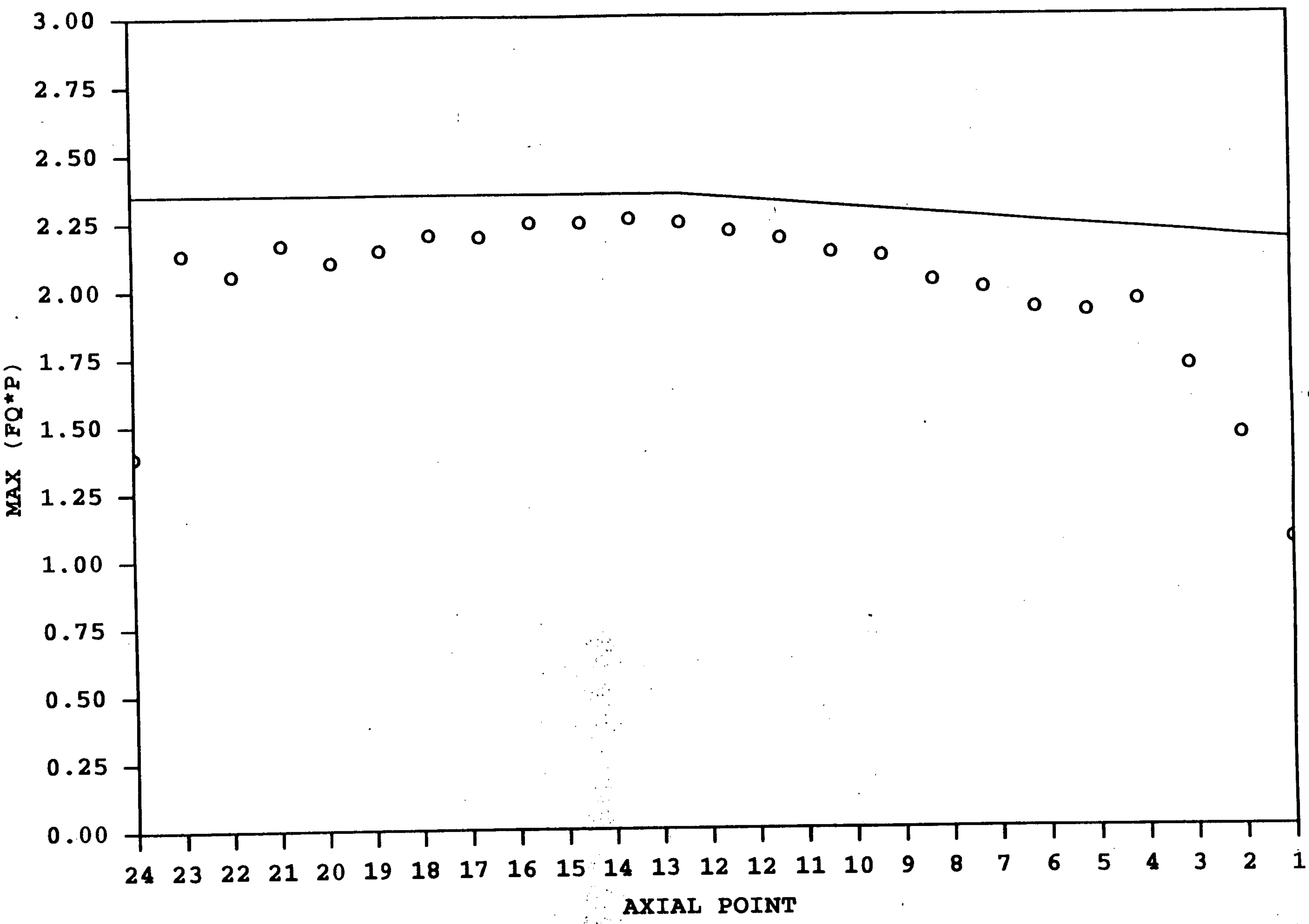

Figure 3.17.1

4.0 TECHNICAL SPECIFICATIONS

Proposed Amendment 152 to the Kewaunee Nuclear Power Plant is required for Reload Cycle 23 and Technical Specifications have been submitted for review and approval (Reference 10).

5.0 STATISTICS UPDATE

Measurements and calculations of Cycles 19, 20, and 21 are incorporated into the FQN and FΔH statistics data base. The moderator temperature coefficient statistics data base includes results from Cycles 13 through 22. The reliability and bias factors used for the Cycle 23 Reload Safety Evaluations are presented in Tables 5.0.1 and 5.0.2.

Table 5.0.1

Reliability Factors

Parameter	Reliability Factor	Bias
FQN	See Table 5.0.2	---
FΔH	4.82%	0
Rod Worth	10.0%	0
Moderator Temperature Coefficient	2.3 pcm/°F	3.1 pcm/°F
Doppler Coefficient	10.0%	0
Boron Worth	5.0%	0
Delayed Neutron Parameters	3.0%	0

Table 5.0.2

FQN Reliability Factors

Core Level	1.05* σNode	RF (%)
1 (Bottom)	.0741	12.23
2	.0633	10.52
3	.0289	5.38
4	.0338	6.06
5	.0297	5.49
6	.0264	5.05
7	.0269	5.11
8	.0230	4.61
9	.0265	5.06
10	.0234	4.66
11	.0253	4.91
12	.0251	4.88
13	.0253	4.91
14	.0231	4.63
15	.0234	4.66
16	.0249	4.85
17	.0299	5.52
18	.0248	4.84
19	.0332	5.98
20	.0290	5.39
21	.0483	8.20
22	.0397	6.91
23	.0823	13.53
24 (Top)	.0825	13.56

6.0 REFERENCES

1. Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, topical report entitled, "Qualification of Reactor Physics Methods for Application to Kewaunee," dated September 29, 1978.

2. Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, topical report WPSRSEM-NP-A entitled, "Reload Safety Evaluation Methods for Application to Kewaunee," Revision 2, dated October 1988.

3. Safety Evaluation Report by the Office of Nuclear Reactor Regulation: "Qualification of Reactor Physics Methods for Application to Kewaunee," dated October 22, 1979.

4. Safety Evaluation Report by the Office of Nuclear Reactor Regulation: "Reload Safety Evaluation Methods for Application to Kewaunee," dated April 1988.

5. Wisconsin Public Service Corporation Technical Specifications for the Kewaunee Nuclear Power Plant. Docket Number 50-305, Amendment No. 137, dated June 9, 1998.

6. Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, Updated Safety Analysis Report.

7. NRC Letter 89-061, from C. R. Steinhardt to U.S. NRC Document Control Desk, dated May 12, 1989.

8. EMF 96-038, May 1996, "Kewaunee Mixed Core Thermal Hydraulic Compatibility Report."

9. Letter from J. T. Holly to Cycle 22 Reload Safety Evaluation File, "Cycle 22 Mixed Core Thermal Hydraulic Evaluation," dated July 16, 1996.

10. Letter from C. R. Steinhardt to U.S. Nuclear Regulatory Commission Document Control Desk, submitting "PA 152 to Kewaunee Technical Specifications," dated April 15, 1998.

11. Letter from M. L. Marchi to U.S. Nuclear Regulatory Commission Document Control Desk, "KEW-19 Fuel and Cycle 23 Reload Design Changes," dated September 5, 1997.

Appendix A

Reactor Physics Methods for Application to Kewaunee Reload Cycle 23

References:

A1. Safety Evaluation Report by the Office of Nuclear Reactor Regulation: "Qualification of Reactor Physics Methods for Application to Prairie Island," dated February 17, 1983

A2. Cycle 22 Lead Fuel Assembly Monitoring Report

A3. Gadolinia Benchmark Report, dated April 22, 1998

Cycle 23 is the first Kewaunee reload which relies primarily on gadolinia as a burnable absorber. The implementation of gadolinia requires a change to the Kewaunee reactor physics methods. This change involves the addition of the MICBURN computer code to the WPS core analysis system. MICBURN is being added to more accurately model the gadolinia burnable absorber. MICBURN has been approved and implemented for gadolinia reload core designs at Prairie Island (Reference A1). MICBURN's gadolinia cross-section and number density results are provided as input to the WPS cross-section generator, CPM-2, for each of the gadolinia fuel designs. All other physics methods in the WPS core analysis system remain the same as the methods used in previous non-gadolinia reload core designs.

The changes in the Kewaunee reactor physics methods have been verified through comparisons to Cycle 22 lead fuel assembly in-core measurements (Reference A2) and through benchmark analyses against Siemens Power Corporation (SPC) calculations for a representative Kewaunee full reload core design with gadolinia (Reference A3). The comparisons of calculation (WPS) to calculation (SPC) and calculation (WPS) to measurement are consistent with current and past non-gadolinia core neutronic comparisons. Therefore, the reactor physics model uncertainty factors will be consistent with current and past model uncertainties. These model uncertainty factors will be conservatively applied in the Cycle 23 reload core design and safety evaluation to assure that all Technical Specification design and safety analysis criteria are satisfied during Cycle 23 operation.