NRC-01-116

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Kewaunee / Point Beach Nuclear
Operated by Nuclear Management Company, LLC

November 29, 2001

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Ladies/Gentlemen:

DOCKET 50-305
OPERATING LICENSE DPR-43
KEWAUNEE NUCLEAR POWER PLANT
CYCLE 25 RELOAD SAFETY EVALUATION

This letter transmits 3 copies of the Reload Safety Evaluation for the Kewaunee Cycle 25 reload core. The evaluation has shown that the Cycle 25 core design is more conservative than the design used in our accident analyses. Therefore, the Cycle 25 reload design does not create an unreviewed safety question, and the reload is proceeding in accordance with the provisions of 10 CFR 50.59(a)(1).

Sincerely,

Kyle A. Hoops
Manager-Kewaunee Plant

JTH

Attachment

cc: US NRC Senior Resident Inspector
US NRC Region III

IE26 1/3
two copies sent to John Lamb
Rec'd 12/19/01

KEWAUNEE NUCLEAR POWER PLANT

RELOAD SAFETY EVALUATION
CYCLE 25
NOVEMBER 2001

NUCLEAR MANAGEMENT COMPANY, LLC
WISCONSIN PUBLIC SERVICE CORPORATION
WISCONSIN POWER & LIGHT COMPANY

RELOAD SAFETY EVALUATION

FOR

KEWAUNEE CYCLE 25

Prepared By: J. J. Voskuil Date: 11-9-2001
Senior Engineer

Reviewed By: D. J. Wanner Date: 11-9-2001
Project Manager

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1.0 SUMMARY

The Kewaunee Nuclear Power Plant shut down for the Cycle 24-25 refueling in September 2001. Startup of Cycle 25 is scheduled for December 2001.

This report presents an evaluation of the Cycle 25 reload core design and demonstrates that the reload core design 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 References 1 and 2. Accident Evaluation methodologies that are applied in this report are detailed in Reference 3. References 1 and 3 have been previously reviewed and approved by the NRC as shown in References 4 and 5. The current physics model reliability factors are discussed in Section 5 of this Reload Safety Evaluation report.

An evaluation, by accident, of the pertinent reactor parameters is performed by comparing the reload core design analysis results with the current bounding safety analysis values. The evaluations performed in this document employ the current Technical Specification (Reference 6) limiting safety system settings and operating limits. The evaluations are based on Cycle 24 shutting down within a ±500 MWD/MTU window of the nominal Cycle 24 end of cycle (EOC) burnup of 15,500 MWD/MTU.

It is concluded that the Cycle 25 reload core design is more conservative than results of the current bounding accident analyses and implementation of this reload core design will not introduce an unreviewed safety question since:

1. the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety will not be increased,

2. the possibility for an accident or malfunction of equipment important to safety 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 contents of this report, which show that the Cycle 25 reload core design is more conservative than the results of the current bounding analyses and that implementation of the Cycle 25 reload core design will not introduce an unreviewed safety question. With the exception of the four Westinghouse LUAs, which are being utilized in accordance with Reference 7, the Cycle 25 reload fuel is of the same design as the existing Cycle 24 reload fuel. The Cycle 25 reload analyses have been performed with previously approved methods (References 4 and 5).

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, assembly identification, control rod bank identification, instrument thimble I.D., thermocouple I.D., and burnable absorber configurations for Cycle 25 are presented in Figure 2.1.1. Table 2.1.1 displays Cycle 25 fuel characteristics including region identification, initial enrichment, number of previous duty cycles, fuel rod design, grid design, and gadolinia loading. The four Westinghouse 422V+ lead use assemblies (LUAs) contain approximately 403 KgU (per assembly) versus approximately 406 KgU in the FRA-ANP Heavy fuel assemblies. Descriptions of the fuel designs are provided in References 8 through 11. (NOTE: On January 1, 2001 a merger of Siemens Power Generation and the Framatome Group was completed. As a result, the name of Siemens Power Corporation (SPC) was changed to Framatome ANP (FRA-ANP). Therefore, the FRA-ANP Heavy fuel design discussed in this Reload Safety Evaluation report is manufactured by the same personnel at the same facility to the same design specifications under the same quality assurance program as the SPC Heavy fuel design used in past Kewaunee reloads. The only thing that has changed is the name on the letterhead of the design documents.)

Fuel assemblies with two or three previous duty cycles are loaded on the core periphery flat region to reduce power in that region, thereby reducing reactor vessel fluence (Reference 12) in the critical reactor vessel locations. The Cycle 25 fuel loading pattern is capable of achieving a burnup of 16,391 MWD/MTU operating at full power, based on a nominal end of Cycle 24 burnup of 15,500 MWD/MTU.

Table 2.1.1

Cycle 25 Fuel Characteristics

REGION	VENDOR	NUMBER OF ASSEMBLIES	NUMBER OF DUTY CYCLES	INITIAL U235 ENRICHMENT (GAD LOAD)	FUEL ROD DESIGN	GRID DESIGN*
25	FRA-ANP	8	2	4.1 (8 rods – 8%)	Heavy	HTP
25	FRA-ANP	9	2	4.1 (12 rods – 8%)	Heavy	HTP
25	FRA-ANP	8	2	4.5 (4 rods – 4%)	Heavy	HTP
25	FRA-ANP	8	2	4.5 (8 rods – 4%)	Heavy	HTP
25	FRA-ANP	8	2	4.5 (8 rods – 8%)	Heavy	HTP
26	FRA-ANP	20	1	4.1 (8 rods – 8%)	Heavy	HTP
26	FRA-ANP	8	1	4.5 (4 rods – 4%)	Heavy	HTP
26	FRA-ANP	4	1	4.5 (8 rods – 4%)	Heavy	HTP
26	FRA-ANP	8	1	4.5 (8 rods – 8%)	Heavy	HTP
27	FRA-ANP	16	0	4.5 (8 rods – 4%)	Heavy	HTP
27	FRA-ANP	20	0	4.5 (8 rods – 8%)	Heavy	HTP
27	Westinghouse	4	0	3.3 None	422V+	ZIRLO

* - HTP denotes the FRA-ANP High Thermal Performance mid-grid design. ZIRLO denotes the Westinghouse mid-grid design. The FRA-ANP top and bottom grids are bi-metallic (Zircaloy and Inconel). The Westinghouse top and bottom grids are Inconel.

FIGURE 2.1.1

Cycle 25 Loading Patttern

	1	2	3	4	5	6	7	8	9	10	11	12	13
A						C55 4.1 8GAD8	D66 4.1 8GAD8	C58 4.1 8GAD8					
B				D58 4.1 8GAD8	D76 4.5 4GAD4	E51 4.5 8GAD4	E88 3.3	E64 4.5 8GAD4	D75 4.5 4GAD4	D59 4.1 8GAD8			
C			C64 4.1 12GAD8	E55 4.5 8GAD4	E83 4.5 8GAD8	C92 4.5 8GAD8	D80 4.5 8GAD4	C93 4.5 8GAD8	E69 4.5 8GAD8	E60 4.5 8GAD4	C65 4.1 12GAD8		
D		D63 4.1 8GAD8	E62 4.5 8GAD4	D87 4.5 8GAD8	D69 4.1 8GAD8	C71 4.5 4GAD4	E81 4.5 8GAD8	C78 4.5 4GAD4	D55 4.1 8GAD8	D83 4.5 8GAD8	E52 4.5 8GAD4	D65 4.1 8GAD8	
E		D78 4.5 4GAD4	E68 4.5 8GAD8	D62 4.1 8GAD8	C81 4.5 8GAD4	E84 4.5 8GAD8	C60 4.1 12GAD8	E78 4.5 8GAD8	C86 4.5 8GAD4	D61 4.1 8GAD8	E79 4.5 8GAD8	D72 4.5 4GAD4	
F	C51 4.1 8GAD8	E59 4.5 8GAD4	C88 4.5 8GAD8	C76 4.5 4GAD4	E74 4.5 8GAD8	C96 4.5 8GAD4	D84 4.5 8GAD8	C80 4.5 8GAD4	E70 4.5 8GAD8	C74 4.5 4GAD4	C87 4.5 8GAD8	E63 4.5 8GAD4	C54 4.1 8GAD8
G	D54 4.1 8GAD8	E87 3.3	D79 4.5 8GAD4	E71 4.5 8GAD8	C61 4.1 12GAD8	D89 4.5 8GAD8	C69 4.1 12GAD8	D85 4.5 8GAD8	C62 4.1 12GAD8	E86 4.5 8GAD8	D82 4.5 8GAD4	E90 3.3	D70 4.1 8GAD8
H	C57 4.1 8GAD8	E54 4.5 8GAD4	C89 4.5 8GAD8	C77 4.5 4GAD4	E80 4.5 8GAD8	C95 4.5 8GAD4	D90 4.5 8GAD8	C85 4.5 8GAD4	E82 4.5 8GAD8	C75 4.5 4GAD4	C90 4.5 8GAD8	E66 4.5 8GAD4	C56 4.1 8GAD8
I		D71 4.5 4GAD4	E73 4.5 8GAD8	D68 4.1 8GAD8	C84 4.5 8GAD4	E72 4.5 8GAD8	C59 4.1 12GAD8	E77 4.5 8GAD8	C79 4.5 8GAD4	D52 4.1 8GAD8	E75 4.5 8GAD8	D73 4.5 4GAD4	
J		D60 4.1 8GAD8	E57 4.5 8GAD4	D88 4.5 8GAD8	D56 4.1 8GAD8	C73 4.5 4GAD4	E67 4.5 8GAD8	C72 4.5 4GAD4	D53 4.1 8GAD8	D86 4.5 8GAD8	E53 4.5 8GAD4	D64 4.1 8GAD8	
K			C70 4.1 12GAD8	E65 4.5 8GAD4	E85 4.5 8GAD8	C91 4.5 8GAD8	D81 4.5 8GAD4	C94 4.5 8GAD8	E76 4.5 8GAD8	E56 4.5 8GAD4	C68 4.1 12GAD8		
L				D57 4.1 8GAD8	D74 4.5 4GAD4	E58 4.5 8GAD4	E89 3.3	E61 4.5 8GAD4	D77 4.5 4GAD4	D67 4.1 8GAD8			
M						C53 4.1 8GAD8	D51 4.1 8GAD8	C52 4.1 8GAD8					

PRELIMINARY CYCLE TWENTY-FIVE

ASSEMBLY ID
INITIAL ENRICHMENT
GADOLINIA LOADING

2.2 Operating Parameters and Design Limits

The Cycle 25 reload core design is based on the following operating parameters and design limits.

2.2.1 Operating Parameters

- 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 Design Limits

A. Nuclear peaking factor limits are as follows:

(i) FQ(Z) limits

a) For FRA-ANP 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 Westinghouse 422V+ LUA fuel (administrative limits):

$FQ(Z) \leq (2.17/P) * K(Z)$ for $P > 0.5$
$FQ(Z) \leq 4.34 * K(Z)$ for $P \leq 0.5$

K(Z) is the function given in Figure TS 3.10-2 of Reference 6 and Z is the core height.

(ii) $F\Delta H$ limits

a) For FRA-ANP Heavy fuel:

$F\Delta H \leq 1.70 * (1 + 0.2 * (1-P))$

b) For Westinghouse 422V+ LUA fuel (administrative limit):

$F\Delta H \leq 1.55 * (1 + 0.2 * (1-P))$

P is the fraction of full power at which the core is operating. The administrative limits placed on the Westinghouse 422V+ LUAs are sufficient to accommodate the thermal hydraulic differences between the LUAs and the co-resident FRA-ANP Heavy fuel assemblies (References 13 and 14).

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 6).

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:

i) 1.0% at Beginning of Cycle (BOC)

ii) 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 6.

E. The indicated axial flux difference shall be maintained within a ± 5% band about the target axial flux difference above 90 percent power. Figure TS 3.10-5 of Reference 6 shows the axial flux difference limits as a function of core power. Reference 6 also provides limits on temporary operation allowed within the 3.10.b.11.a. line envelope (see Figure TS 10.3-5 of Reference 6) at power levels between 50 percent and 90 percent.

F. At refueling conditions a boron concentration of 2200 ppm will be sufficient to maintain the reactor subcritical by 5% $\Delta k/k$ with all rods inserted and will maintain the core subcritical with all rods out (Reference 6).

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 N-1 rod worth reduced by the rod worth reliability factor. Figure 2.3.1 compares the Cycle 25 minimum scram insertion curve to the current bounding safety analysis curve.

It is concluded that the minimum trip reactivity insertion rate for Cycle 25 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.

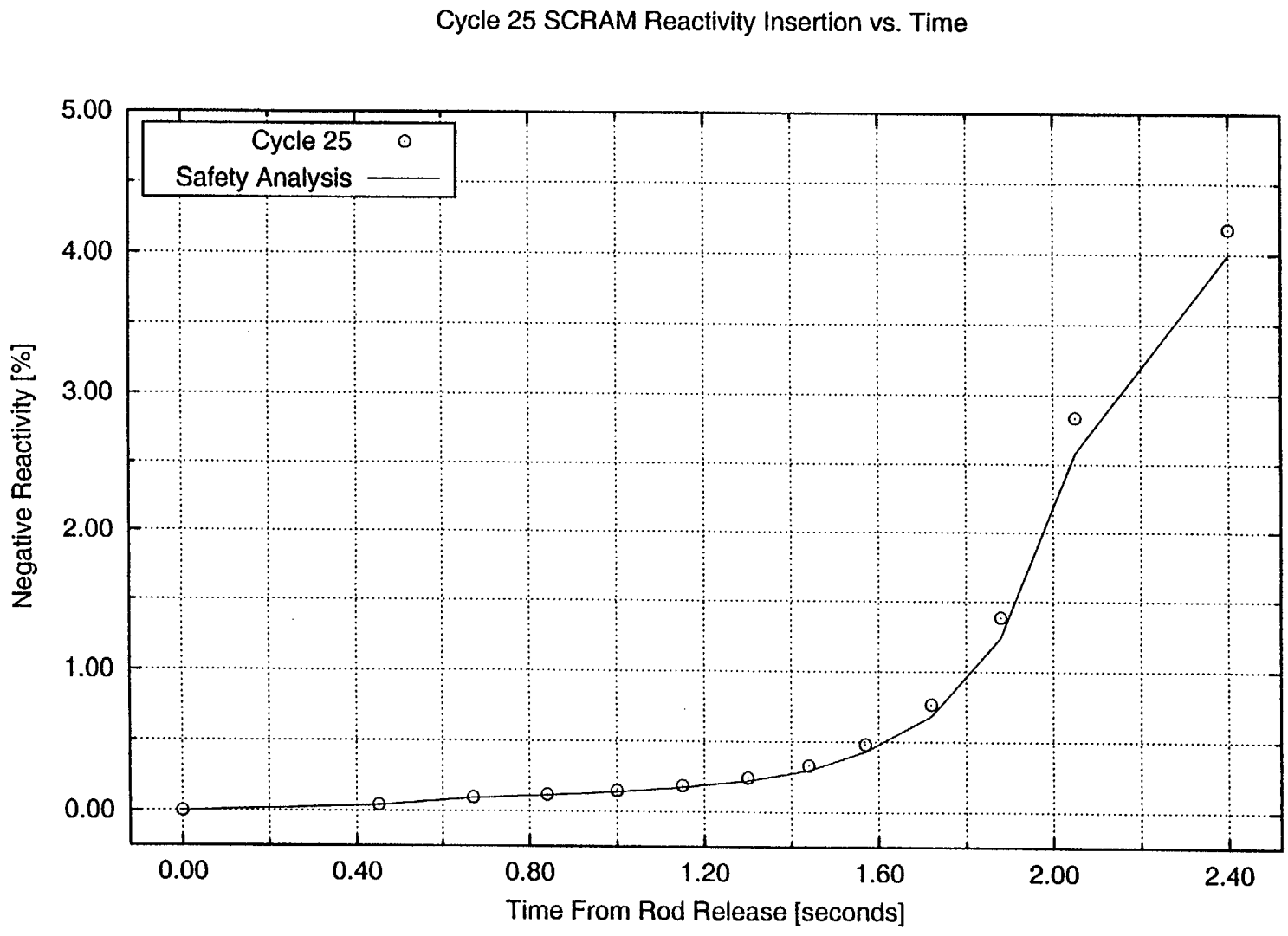

Cycle 25 SCRAM Reactivity Insertion vs. Time
Cycle 25
Safety Analysis
Negative Reactivity [%]
5.00
4.00
3.00
2.00
1.00
0.00
0.00
0.40
0.80
1.20
1.60
2.00
2.40
Time From Rod Release [seconds]

Figure 2.3.1

2.4 Shutdown Window

An evaluation of the maximum full power equilibrium peaking factors versus EOC 24 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 refueling shutdown of Cycle 24 occurs within the burnup window, the Cycle 25 peaking factors will not be significantly affected and will not exceed their limiting values.

Table 2.4.1

Cycle 25 Peaking Factor Versus Cycle 24 Shutdown Burnup

	FΔH		FQ	
	Cycle 25	**Limit**	**Cycle 25**	**Limit**
EOC 24 = 15000 MWD/MTU	HVY 1.64	1.70	2.25	2.35
(Nominal - 500 MWD/MTU)	LUA 1.51	1.55*	2.14	2.17*
EOC 24 = 15500 MWD/MTU	HVY 1.64	1.70	2.25	2.35
(Nominal)	LUA 1.52	1.55*	2.15	2.17*
EOC 24 = 16000 MWD/MTU	HVY 1.64	1.70	2.25	2.35
(Nominal + 500 MWD/MTU)	LUA 1.53	1.55*	2.16	2.17*

* Administrative limit for the four Westinghouse 422V+ LUAs.

2.5 Moderator Temperature Coefficient

An evaluation of the Cycle 25 hot full power moderator temperature coefficient is presented in Table 2.5.1. The calculated Cycle 25 hot full power value at beginning of cycle (BOC) is compared to the MTC upper bound limit of -8.0 pcm/°F. The Cycle 25 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 the 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 25 MTC is conservative with respect to the bounding value. Therefore, the Cycle 25 reload core design will not adversely affect the results of the ATWS safety analysis.

Table 2.5.1

Moderator Temperature Coefficient

Parameter	Reload Safety Evaluation Value		Current Safety Analysis	Units
A) Full Power Moderator Temp. Coefficient	-13.2	≤	-8.0	pcm/°F_m

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 25 accident evaluations.

Based on the results of References 13 and 14, restricting the Westinghouse 422V+ LUAs to $FQ \leq 2.17$ and $F\Delta H \leq 1.55$, is sufficient for the results of the existing safety analyses performed with FRA-ANP Heavy fuel at $FQ = 2.35$ and $F\Delta H = 1.70$ to bound the Westinghouse 422V+ LUAs in the Cycle 25 reload core.

Table 3.0.1

Kewaunee Nuclear Power Plant

List of Current Safety Analyses

Accident	Current Safety Analysis
Uncontrolled RCCA Withdrawal From a Subcritical Condition	Reference 15
Uncontrolled RCCA Withdrawal at Power	Reference 15
Control Rod Misalignment	Reference 15
Control Rod Drop	Reference 15
CVCS Malfunction	Reference 15
Startup of an Inactive Loop	Reference 15
Excessive Heat Removal Due to FW System Malfunction	Reference 15
Excessive Load Increase	Reference 15
Loss of External Load	Reference 15
Loss of Normal Feedwater Flow	Reference 15
Loss of Reactor Coolant Flow – Pump Trip	Reference 15
Loss of Reactor Coolant Flow - Locked Rotor	Reference 15
Main Steam Line Break	Reference 15
Control Rod Ejection	Reference 15
Fuel Handling Accident	Reference 15
Loss of Coolant Accident	Reference 16

Table 3.0.2

Safety Analyses Bounding Values

Parameter	Lower Bound	Upper Bound	Units
Moderator Temp. Coefficient			
$0 \leq P \leq 60\%$	-40.0	+5.0	pcm/°F_m
$P > 60\%$	-40.0	0.0	pcm/°F_m
95% of time at HFP	---	-8.0	pcm/°F_m
URW from subcritical only	---	+5.0	pcm/°F_m
Doppler Coefficient	-2.32	-1.0	pcm/°F_f
Differential Boron Worth	-11.2	-7.0	pcm/ppm
Delayed Neutron Fraction	0.00485	0.00706	---
Prompt Neutron Lifetime	15	--	μsec
Shutdown Margin	1.0 (BOC) 2.0 (EOC)	-- --	% $\Delta\rho$
Differential Rod Worth of 2 Banks Moving	--	82	pcm/sec
Ejected Rod Cases			
HFP, BOL			
ßeff	0.0055	N/A	---
Rod Worth	N/A	0.30	% $\Delta\rho$
FQ	N/A	5.03	---
HFP, EOL			
ßeff	0.0050	N/A	---
Rod Worth	N/A	0.42	% $\Delta\rho$
FQ	N/A	4.6	---
HZP, BOL			
ßeff	0.0055	N/A	---
Rod Worth	N/A	0.91	% $\Delta\rho$
FQ	N/A	8.2	---
HZP, EOL			
ßeff	0.0050	N/A	---
Rod Worth	N/A	0.92	% $\Delta\rho$
FQ	N/A	12.8	---

Table 3.0.2 (cont.)

Safety Analyses Bounding Values

Parameter	Lower Bound	Upper Bound	Units
Normalized Scram Worth Insertion Rate			
HFP Magnitude	---	4.0	% Δρ
HZP Manitude	---	2.5	% Δρ
Locked Rotor			
FΔH	--	1.462	--
Pins Above FΔH	--	35	%
Control Rod Misalignment FΔH	--	2.174	--
Boron Dilution			
Differential Boron Worth	-0.015864	--	$/ppm
ARO HFP EQXE PPM	---	1600	ppm
ARI PPM (Startup Cond.)	---	1300	ppm
Refueling Condition SDM	5.0	--	% Δk/k

3.1 Evaluation of Uncontrolled RCCA Withdrawal from a Subcritical Condition

An uncontrolled addition of reactivity due to an 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 the delayed neutron fraction, which are chosen to maximize the peak heat flux.

Table 3.1.1 presents a comparison of Cycle 25 physics parameters to the current safety analysis values for the Uncontrolled RCCA Withdrawal from a Subcritical Condition.

Since the pertinent parameters from the proposed Cycle 25 reload core design are conservatively bounded by those used in the current safety analysis, an Uncontrolled RCCA Withdrawal from a Subcritical Condition accident will be less severe than the transient in the current safety analysis. Therefore, the implementation of the Cycle 25 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.1.1

Uncontrolled RCCA Withdrawal From a Subcritical Condition

Parameter	Reload Safety Evaluation Value		Current Safety Analysis	Units
A) Doppler Temp. Coefficient	-1.34	≤	-1.0	pcm/°F_f
B) Moderator Temp. Coefficient	-2.27	≤	5.0	pcm/°F_m
C) Differential Rod Worth of Two Moving Banks	0.065	≤	0.116	$/sec
D) Scram Worth vs. Time	See Section 2.3			
E) Delayed Neutron Fraction	0.00647	≤	0.00706	---
F) Prompt Neutron Lifetime	23	≥	15	μsec

3.2 Evaluation of Uncontrolled RCCA 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 25 physics parameters to the current safety analysis values for the Uncontrolled Rod Withdrawal at Power accident.

Since the pertinent parameters from the proposed Cycle 25 reload core design 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 safety analysis. Therefore, the implementation of the Cycle 25 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.2.1

Uncontrolled RCCA Withdrawal at Power

Parameter	Reload Safety Evaluation Value		Current Safety Analysis	Units
A) Doppler Temp. Coefficient	-1.34	≤	-1.0	pcm/°F_f
B) Moderator Temp. Coefficient	-8.37	≤	0.0	pcm/°F_m
C) Differential Rod Worth of Two Moving Banks	0.065	≤	0.116	$/sec
D) Scram Worth vs. Time	See Section 2.3			
E) FΔH	Heavy 1.64 LUA 1.52	≤	1.70 1.55	--- ---
F) Delayed Neutron Fraction	0.00647	≤	.00706	---

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 that is analyzed to show that core Departure from Nuclear Boiling Ratio (DNBR) limits are not exceeded.

The limiting core parameter is the peak $F\Delta H$ for 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 25 $F\Delta H$ versus the current safety analysis $F\Delta H$ limit for the Control Rod Misalignment accident.

Since the pertinent parameter from the proposed Cycle 25 reload core design is conservatively bounded by that used in the current safety analysis, a Control Rod Misalignment accident will be less severe than the transient in the current safety analysis. Therefore, the implementation of the Cycle 25 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) $F_{\Delta H}$, rod misalignment	2.083	≤	2.174

3.4 Evaluation of Control Rod Drop

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 25 physics parameter to the current safety analysis value for the Control Rod Drop accident.

Since the pertinent parameter from the proposed Cycle 25 reload core design is conservatively bounded by that used in the current safety analysis, a Control Rod Drop accident will be less severe than the transient in the current safety analysis. Therefore, the implementation of the Cycle 25 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.4.1

Control Rod Drop

Parameter	Reload Safety Evaluation Value		Current Safety Analysis	Units
A) $F\Delta H$, rod drop	2.036	≤	2.174	---

3.5 Evaluation of Chemical and Volume Control System Malfunction

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 would likely be terminated by a reactor trip on over-temperature ΔT if the operator did not intervene.

Table 3.5.1 presents a comparison of Cycle 25 physics analysis results to the current safety analysis values for the Chemical and Volume Control System Malfunction accident for refueling, startup, and full power core conditions.

Since the pertinent parameters from the proposed Cycle 25 reload core design are conservatively bounded by those used in the current safety analysis, a Chemical and Volume Control System Malfunction accident will be less severe than the transient in the current safety analysis. Therefore, the implementation of the Cycle 25 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.5.1

Chemical and Volume Control System Malfunction

Parameter	Reload Safety Evaluation Value		Current Safety Analysis	Units
i) Refueling Conditions				
A) Shutdown Margin	5.9	≥	5.0	% Δρ
ii) At-Power Conditions				
A) Doppler Temp. Coefficient	-1.34	≤	-1.0	Pcm/°F_f
B) Moderator Temp. Coefficient	-8.37	≤	0.0	pcm/°F_m
C) Boron Dilution Differential Boron Worth	-0.0154	≥	-0.0015864	$/ppm
D) Shutdown Margin	1.82	≥	1.0	% Δρ
E) FΔH	Heavy 1.64	≤	1.70	---
	LUA 1.52	≤	1.55	---
F) Delayed Neutron Fraction	0.00647	≤	0.00706	---
G) ARO Boron Concentration	1376	≤	1600	ppm
H) Scram Worth Versus Time	See Section 2.3			
iii) Startup Conditions				
A) Critical Boron Concentration (ARI)	1272	≤	1300	ppm

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 25 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 25 reload core design 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 safety analysis. Therefore, the implementation of the Cycle 25 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 Value		Current Safety Analysis	Units
A) Doppler Coefficient	-1.94	≤	-1.0	pcm/°F_f
B) Moderator Temp. Coefficient	-35.1	≥	-40.0	pcm/°F_m
C) $F_{\Delta H}$	Heavy 1.64	≤	1.70	---
	LUA 1.52	≤	1.55	---

3.7 Evaluation of Excessive Heat Removal Due to 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 25 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 25 reload core design are conservatively bounded by those used in the current safety analysis, a Feedwater System Malfunction accident will be less severe than the transient in the current safety analysis. Therefore, the implementation of the Cycle 25 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.7.1

Excessive Heat Removal Due to Feedwater System Malfunction

Parameter	Reload Safety Evaluation Value		Current Safety Analysis	Units
i) Beginning of Cycle				
A) Doppler Temp. Coefficient	-1.34	≤	-1.0	pcm/°F_f
B) Moderator Temp. Coefficient	-8.37	≤	0.0	pcm/°F_m
ii) End of Cycle				
A) Doppler Temp. Coefficient	-1.36	≤	-1.0	pcm/°F_f
B) Moderator Temp. Coefficient	-31.37	≥	-40.0	pcm/°F_m
iii) Beginning and End of Cycle				
C) $F_{\Delta H}$	Heavy 1.64	≤	1.70	---
	LUA 1.52	≤	1.55	---
D) Scram Worth Versus Time	See Section 2.3			

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 accident and is therefore sensitive to the same parameters.

Table 3.8.1 presents a comparison of Cycle 25 physics results to the current safety analysis values for the Excessive Load Increase accident.

Since the pertinent parameters from the proposed Cycle 25 reload core design 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 safety analysis. Therefore, the implementation of the Cycle 25 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 Value		Current Safety Analysis	Units
i) Beginning of Cycle				
A) Doppler Temp. Coefficient	-1.34	≤	-1.0	pcm/°F_f
B) Moderator Temp. Coefficient	-8.37	≤	0.0	pcm/°F_m
ii) End of Cycle				
A) Doppler Temp. Coefficient	-1.36	≤	-1.0	pcm/°F_f
B) Moderator Temp. Coefficient	-31.37	≥	-40.0	pcm/°F_m
iii) Beginning and End of Cycle				
C) $F_{\Delta H}$	Heavy 1.64 LUA 1.52	≤ ≤	1.70 1.55	--- ---
D) Scram Worth Versus Time	See Section 2.3			

3.9 Evaluation of Loss of External 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 ($F\Delta H$) and scram reactivity are evaluated to ensure thermal margins are maintained by the reactor protection system.

A comparison of Cycle 25 physics parameters to the current safety analysis values for the Loss of External Load accident is presented in Table 3.9.1.

Since the pertinent parameters from the proposed Cycle 25 reload core design are conservatively bounded by those used in the current safety analysis, a Loss of External Load accident will be less severe than the transient in the current safety analysis. Therefore, the implementation of the Cycle 25 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.9.1

Loss of External Load

Parameter	Reload Safety Evaluation Value		Current Safety Analysis	Units
i) Beginning of Cycle				
A) Moderator Temp. Coefficient	-1.64	≥	-2.32	pcm/°F_f
B) Doppler Temp. Coefficient	-8.37	≤	0.0	pcm/°F_m
ii) End of Cycle				
A) Doppler Temp. Coefficient	-1.66	≥	-2.32	pcm/°F_f
B) Moderator Temp. Coefficient	-31.37	≥	-40.0	pcm/°F_m
iii) Beginning and End of Cycle				
C) Scram worth Versus Time	See Section 2.3			
D) $F_{\Delta H}$	Heavy 1.64 LUA 1.52	≤ ≤	1.70 1.55	--- ---

3.10 Evaluation of Loss of Normal Feedwater Flow

A complete loss of normal feedwater flow 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 Reactor Coolant Flow – Pump Trip evaluation (see 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 Normal Feedwater Flow accident is not sensitive to core physics parameters since the reactor is assumed to trip in the initial stages of the transient. This trip occurs well before the heat transfer capability of the steam generator is reduced. The decay heat then drives the transient from the tripped reactor. Also, the loss of flow due to pump trip transient discussed in Section 3.11 is considered a more severe transient of this type. Therefore no core physics parameter comparisons will be made for the Reload Safety Evaluation.

3.11 Evaluation of Loss of Reactor Coolant Flow - Pump Trip

The loss of power and/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 25 calculated physics parameters to the current safety analysis values for the Loss of Reactor Coolant Flow - Pump Trip accident.

Since the pertinent parameters from the proposed Cycle 25 reload core design are conservatively bounded by those used in the current safety analysis, a Loss of Reactor Coolant Flow - Pump Trip accident will be less severe than the transient in the current safety analysis. Therefore, the implementation of the Cycle 25 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.11.1

Loss of Reactor Coolant Flow - Pump Trip

Parameter	Reload Safety Evaluation Value		Current Safety Analysis	Units
A) Doppler Temp. Coefficient	-1.66	≥	-2.32	pcm/°F_f
B) Moderator Temp. Coefficient	-8.37	≤	0.0	pcm/°F_m
C) Scram Worth Versus Time	See Section 2.3			
D) FΔH	Heavy 1.64	≤	1.70	---
	LUA 1.52	≤	1.55	---
E) Fuel Temperature	2080	≤	2130	°F

3.12 Evaluation of Loss of Reactor Coolant Flow - 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 departure from nucleate boiling (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 25 physics parameters to the current safety analysis values for the Loss of Reactor Coolant Flow - Locked Rotor accident.

Since the pertinent parameters from the proposed Cycle 25 reload core design are conservatively bounded by those used in the current safety analysis, a Loss of Reactor Coolant Flow - Locked Rotor accident will be less severe than the transient in the current safety analysis. Therefore, the implementation of the Cycle 25 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.12.1

Loss of Reactor Coolant Flow - Locked Rotor

Parameter	Reload Safety Evaluation Value		Current Safety Analysis	Units
A) Doppler Temp. Coefficient	-1.66	≥	-2.32	pcm/°F_f
B) Moderator Temp. Coefficient	-8.37	≤	0.0	pcm/°F_m
C) Scram Worth Versus Time	See Section 2.3			
D) Delayed Neutron Fraction	0.00647	≤	0.00706	---
E) Percent Pins > Limiting FΔH, locked rotor (DNBR=1.14)	19.3	≤	35.0	%
F) FQ	Heavy 2.25	≤	2.35	---
	LUA 2.16	≤	2.17	---
G) Fuel Temperature	2080	≤	2130	°F

3.13 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 histories in the fuel rods.

Table 3.13.1 presents a comparison of the maximum Cycle 25 $F\Delta H$ to the current safety analysis $F\Delta H$ limit for the Fuel Handling accident.

Since the pertinent parameter from the proposed Cycle 25 reload core design 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 safety analysis. Therefore, the implementation of the Cycle 25 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.13.1

Fuel Handling Accident

Parameter	Reload Safety Evaluation Value		Current Safety Analysis
A) $F^{\Delta H}$, fuel handling	1.64	≤	1.70

3.14 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 scram 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.

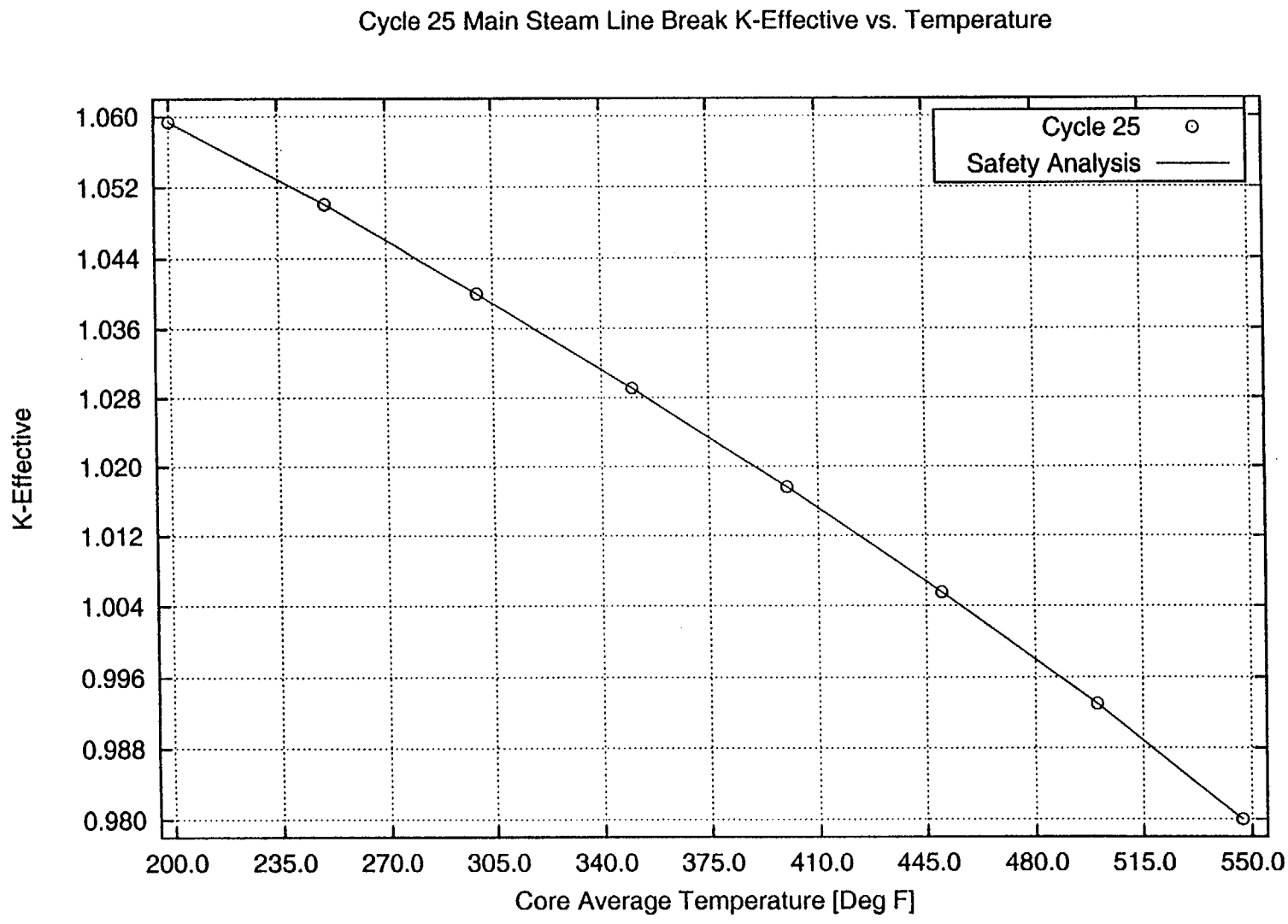
Table 3.14.1 presents a comparison of Cycle 25 calculated physics parameters to the current safety analysis values for the Main Steam Line Break accident. Figure 3.14.1 compares the core K-Effective during the cooldown to the current bounding safety analysis curve.

Since the pertinent parameters from the proposed Cycle 25 reload core design 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 safety analysis. Therefore, the implementation of the Cycle 25 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.14.1

Main Steam Line Break

Parameter	Reload Safety Evaluation Value		Current Safety Analysis	Units
A) K-Effective Versus Temperature	See Figure 3.14.1			
B) Shutdown Margin	2.001	≥	2.00	%Δρ
C) FΔH, steam line break	4.84	≤	8.00	---
D) Doppler Temp. Coefficient	-1.34	≤	-1.0	pcm/°F_f
E) Boron Worth Coefficient	-7.00	≤	-7.0	pcm/ppm


Cycle 25 Main Steam Line Break K-Effective vs. Temperature
Cycle 25
Safety Analysis
K-Effective
1.060
1.052
1.044
1.036
1.028
1.020
1.012
1.004
0.996
0.988
0.980
200.0
235.0
270.0
305.0
340.0
375.0
410.0
445.0
480.0
515.0
550.0
Core Average Temperature [Deg F]

3.15 Evaluation of Control Rod Ejection

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 3.15.1 through 3.15.4 present the comparison of Cycle 25 calculated physics parameters to the current safety analysis values for the Control Rod Ejection accident at zero and full power, BOC and EOC core conditions.

Since the pertinent parameters from the proposed Cycle 25 reload core design are conservatively bounded by those used in the current safety analysis, a Control Rod Ejection accident will be less severe than the transient in the current safety analysis. Therefore, the implementation of the Cycle 25 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.15.1

Control Rod Ejection at

HFP, BOC

Parameter	Reload Safety Evaluation Value		Current Safety Analysis	Units
A) Doppler Temp. Coefficient	-1.34	≤	-1.0	pcm/°F_f
B) Moderator Temp. Coefficient	-8.37	≤	0.0	pcm/°F_m
C) Delayed Neutron Fraction	0.00609	≥	0.00550	---
D) Ejected Rod Worth	0.09	≤	0.30	%Δρ
E) Scram Worth Versus Time	See Section 2.3			
F) FQ, rod ejection	2.59	≤	5.03	---
G) Prompt Neutron Lifetime	23.0	≥	15.0	μsec

Table 3.15.2

Control Rod Ejection at

HZP, BOC

Parameter	Reload Safety Evaluation Value		Current Safety Analysis	Units
A) Doppler Temp. Coefficient	-2.29	≤	-1.0	pcm/°F_f
B) Moderator Temp. Coefficient	-2.27	≤	5.0	pcm/°F_m
C) Delayed Neutron Fraction	0.00609	≥	0.00550	---
D) Ejected Rod Worth	0.67	≤	0.91	%Δρ
E) Scram Worth Versus Time	See Section 2.3			
F) FQ, rod ejection	5.93	≤	8.20	---
G) Prompt Neutron Lifetime	23.0	≥	15.0	μsec

Table 3.15.3

Control Rod Ejection at

HFP, EOC

Parameter	Reload Safety Evaluation Value		Current Safety Analysis	Units
A) Doppler Temp. Coefficient	-1.36	≤	-1.0	pcm/°F_f
B) Moderator Temp. Coefficient	-24.21	≤	0.0	pcm/°F_m
C) Delayed Neutron Fraction	0.00527	≥	0.00500	---
D) Ejected Rod Worth	0.11	≤	0.42	%Δρ
E) Scram Worth Versus Time	See Section 2.3			
F) FQ, rod ejection	2.86	≤	4.60	---
G) Prompt Neutron Lifetime	26.0	≥	15.0	μsec

Table 3.15.4

Control Rod Ejection at

HZP, EOC

Parameter	Reload Safety Evaluation Value		Current Safety Analysis	Units
D) Doppler Temp. Coefficient	-2.85	≤	-1.0	pcm/°F_f
A) Moderator Temp. Coefficient	-5.92	≤	5.0	pcm/°F_m
B) Delayed Neutron Fraction	0.00527	≥	0.00500	---
C) Ejected Rod Worth	0.66	≤	0.92	%Δρ
G) Scram Worth Versus Time	See Section 2.3			
F) FQ, rod ejection	8.32	≤	12.8	---
E) Prompt Neutron Lifetime	26.0	≥	15.0	μsec

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 reactor coolant system, up to and including a double-ended guillotine rupture of the largest pipe.

The principal reload design parameters that affect the results of LOCA analyses are shown in Table 3.16.1. Table 3.16.1 presents the comparison of Cycle 25 physics calculation results to the current safety analysis values for the Loss of Coolant accident.

Since the pertinent parameters from the proposed Cycle 25 reload core design 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 safety analysis. Therefore, the implementation of the Cycle 25 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 Value	Required Inequality	Current Safety Analysis	Units
A. FQ	Heavy 2.25 LUA 2.16	≤ ≤	2.35 2.17	-- --
B. F$^{\Delta}$H	Heavy 1.64 LUA 1.52	≤ ≤	1.70 1.55	-- --
C. Max. Assy. Ave. Peaking Factor	1.462	≤	1.514	--
D. Axial Offset at 100% Power a) Most Negative b) Most Positive	 -9.0 +6.1	 ≥ ≤	 -30.0 +13.0	 % %
E. Max. Core Ave. Power in Lower Power Assy a) Before 1500 MWD/MTU b) After 1500 MWD/MTU	 0.48 0.53	 ≤ ≤	 0.50 0.60	 -- --
F. Max 95/95 Power for the Hot Rod	14.593	≤	14.661	kw/ft

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 Technical Specifications (Reference 6).

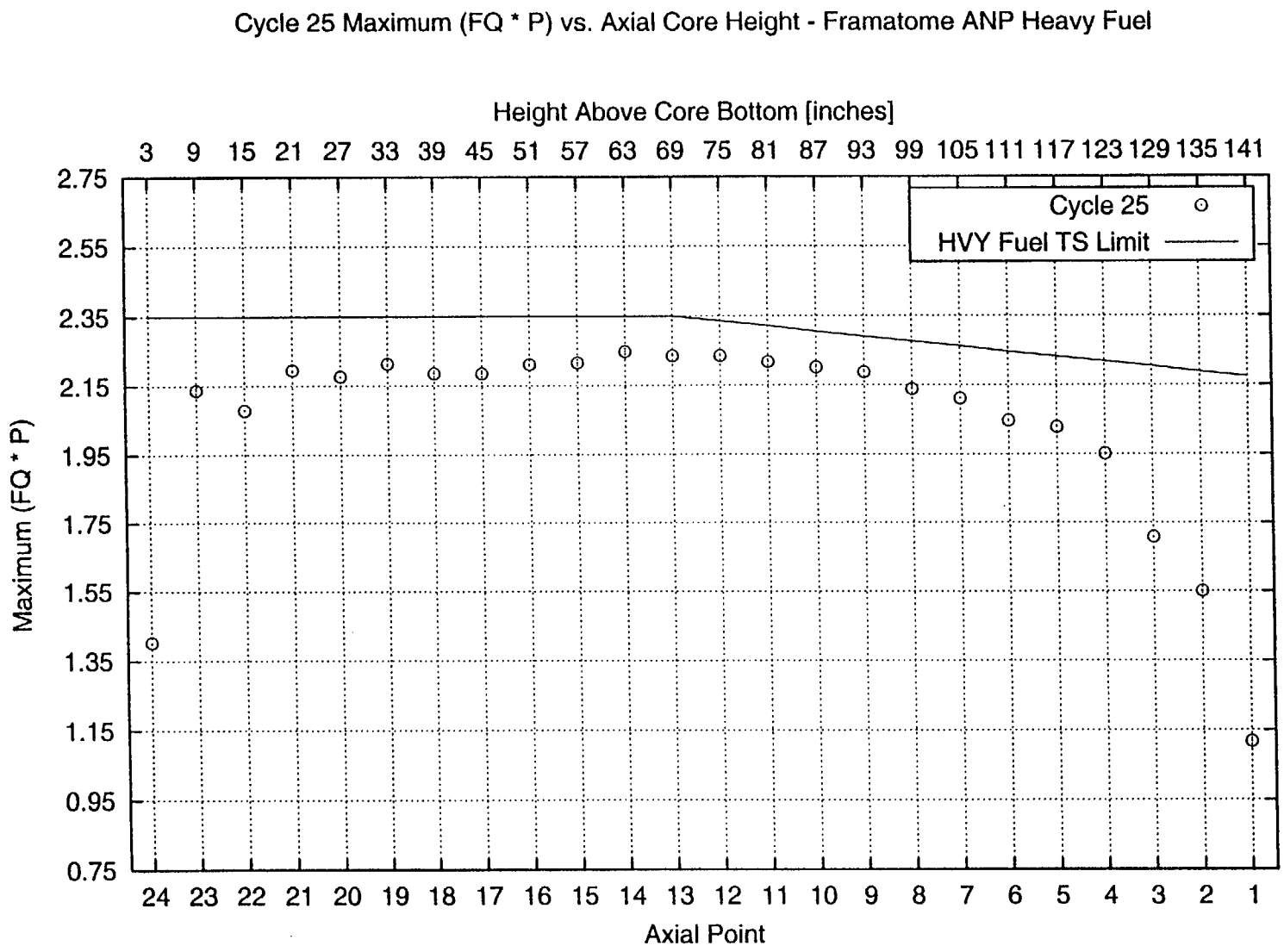
FQT(Z) are determined by calculations performed at full power, equilibrium core conditions, at exposures ranging from BOC to EOC. Conservative factors which account for potential power distribution variations allowed by the power distribution control procedures, manufacturing tolerances, and measurement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

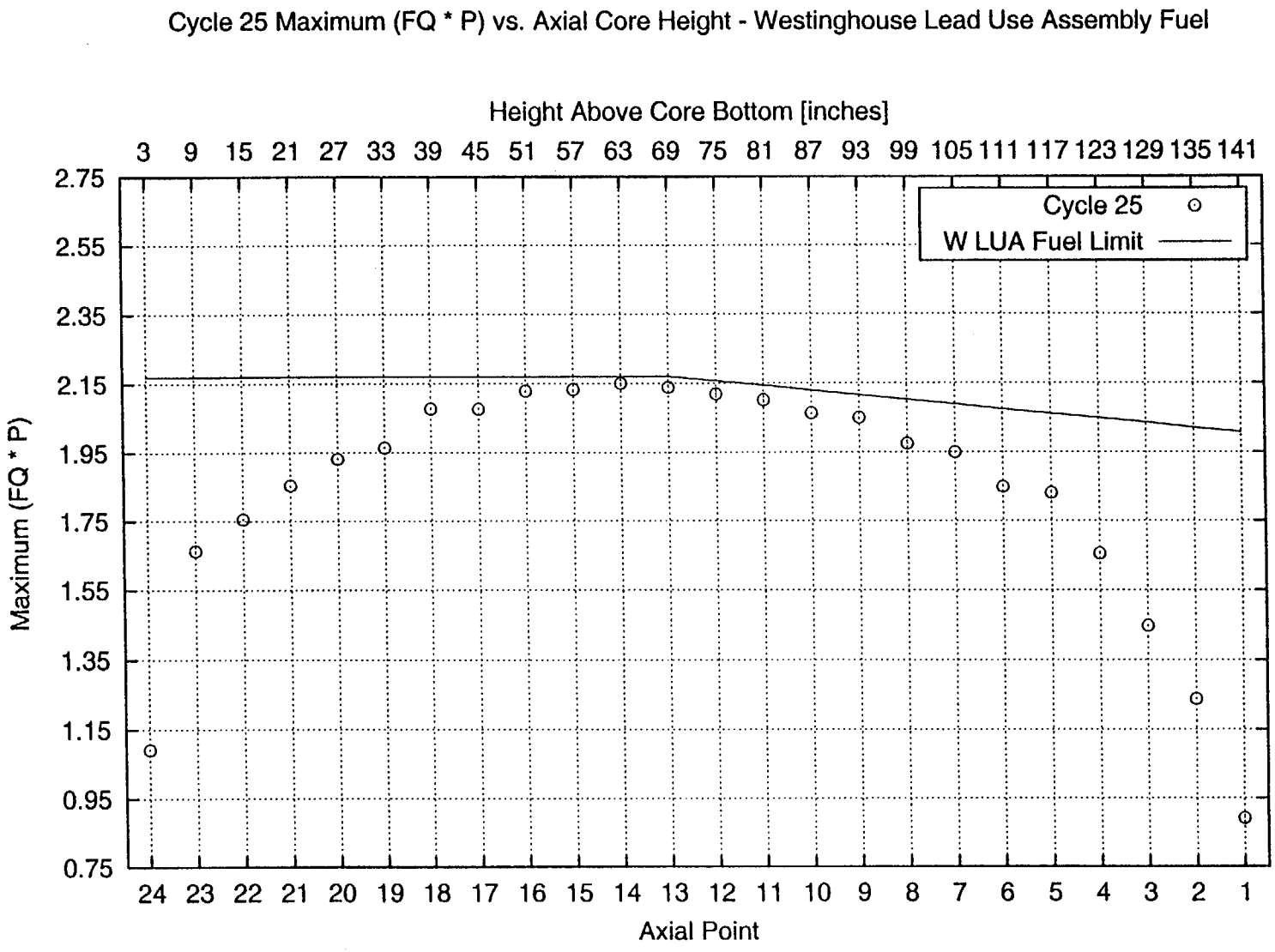
Table 3.17.1 compares the power distribution parameters to their respective limits. Figure 3.17.1 displays the calculated FQT(Z), including uncertainty factors, to the FQT(Z) limits. These results demonstrate that the power distributions expected during Cycle 25 operation will not preclude full power operation under the power distribution control specifications currently applied (Reference 6).

Table 3.17.1

Power Distribution Control Verification

Parameter	Reload Safety Evaluation Value	Required Inequality	Current Safety Analysis	Units
A. FQ	See Figures 3.17.1 and 3.17.2			
B. FΔH	Heavy 1.64 LUA 1.52	≤ ≤	1.70 1.55	-- --


Cycle 25 Maximum (FQ * P) vs. Axial Core Height - Framatome ANP Heavy Fuel
Height Above Core Bottom [inches]
3 9 15 21 27 33 39 45 51 57 63 69 75 81 87 93 99 105 111 117 123 129 135 141
Cycle 25
HVY Fuel TS Limit
Maximum (FQ * P)
2.75
2.55
2.35
2.15
1.95
1.75
1.55
1.35
1.15
0.95
0.75
24 23 22 21 20 19 18 17 16 15 14 13 12 11 10 9 8 7 6 5 4 3 2 1
Axial Point


Cycle 25 Maximum (FQ * P) vs. Axial Core Height - Westinghouse Lead Use Assembly Fuel
Height Above Core Bottom [inches]
3 9 15 21 27 33 39 45 51 57 63 69 75 81 87 93 99 105 111 117 123 129 135 141
Cycle 25
W LUA Fuel Limit
Maximum (FQ * P)
2.75
2.55
2.35
2.15
1.95
1.75
1.55
1.35
1.15
0.95
0.75
24 23 22 21 20 19 18 17 16 15 14 13 12 11 10 9 8 7 6 5 4 3 2 1
Axial Point

4.0 TECHNICAL SPECIFICATIONS

No amendments to the Kewaunee Nuclear Power Plant Technical Specifications (Reference 6) are required for Reload Cycle 25. However, in the event that the total control rod worth measured during startup physics testing is more than 6.0% less than the predicted total control rod worth, a revision to Technical Specification Figure TS 3.10-1, "Required Shutdown Reactivity vs. Reactor Boron Concentration," may be needed before the Cycle 25 hot full power all rods out boron concentration reaches 700 ppm.

5.0 STATISTICS UPDATE

Measurements and calculations of Cycles 21, 22 and 23 are incorporated into the FQN and FΔH statistics database. The moderator temperature coefficient statistics database includes results from Cycles 13 through 24. The reliability and bias factors used for the Cycle 25 Reload Safety Analyses 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	4.75%	0
Rod Worth		
Steam Line Break	6.0%*	0
All Other Analyses	10.0%	0
Moderator Temperature Coefficient	2.1 pcm/°F	3.2 pcm/°F
Doppler Coefficient	10.0%	0
Boron Worth	5.0%	0
Delayed Neutron Fractions	3.0%	0
Delayed Neutron Lifetimes	5.0%	0

* - In the event that the total control rod worth measured during startup physics testing is over 6.0% less than the predicted total worth, a reanalysis of the steam line break accident and a revision to Technical Specification Figure TS 3.10-1, "Required Shutdown Reactivity vs. Reactor Boron Concentration," may be needed before the Cycle 25 hot full power all rods out boron concentration reaches 700 ppm.

Table 5.0.2

FQN Reliability Factors

Core Level	σNode	RF (%)
1 (Bottom)	0.0789	13.62
2	0.0628	10.94
3	0.0302	5.77
4	0.0335	6.26
5	0.0311	5.90
6	0.0261	5.18
7	0.0265	5.24
8	0.0234	4.81
9	0.0262	5.19
10	0.0235	4.82
11	0.0247	4.99
12	0.0246	4.97
13	0.0245	4.96
14	0.0234	4.81
15	0.0231	4.77
16	0.0252	5.06
17	0.0292	5.62
18	0.0258	5.14
19	0.0330	6.18
20	0.0284	5.51
21	0.0488	8.65
22	0.0378	6.91
23	0.0741	12.82
24 (Top)	0.0837	14.43

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, Updated Safety Analysis Report, Revision 16, dated December 1, 2000.

3. Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, topical report WPSRSEM-NP entitled, "Reload Safety Evaluation Methods for Application to Kewaunee," Revision 3, dated September 2000.

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

5. Safety Evaluation Report by the Office of Nuclear Reactor Regulation: "Topical Report WPSRSEM-NP Revision 3, Reload Safety Evaluation Methods, Nuclear Management Company LLC, Kewaunee Nuclear Power Plant, Docket No. 50-305," dated September 10, 2001.

6. Wisconsin Public Service Corporation Technical Specifications for the Kewaunee Nuclear Power Plant. Docket Number 50-305, Amendment No. 157, dated September 5, 2001.

7. "Core Reloads of Advanced Design Fuel Assemblies," Wisconsin Public Service letter NRC-91-084, from K.H. Evers to NRC Document Control Desk, dated June 19, 1991.

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9. Characteristics Specification, EMF - CS-797, Rev. 4, Siemens Power Corporation, dated June 22, 1998.

10. Characteristics Specification, EMF - CS-939, Rev. 1, Siemens Power Corporation, dated January 20, 2000.

11. Characteristics Specification, EMF - CS-1098, Rev. 2, Framatome ANP Richland, Inc., dated June 28, 2001.

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

13. "Lead Test Assembly Evaluation for LOCA Analyses," Westinghouse letter KEW-RTSR-01-009, from R.H. Owoc to D.J. Wanner, dated August 1, 2001.

14. "Westinghouse 422V+ Lead Use Assembly T/H Assessment," calc note by D.J. Wanner, dated October 19, 2001.

15. PSA 2001 Notebook (maintained in Fuel Services filing system).

16. "Kewaunee Nuclear Power Plant Steam Generator Replacement and T_{avg} Operating Window Program," Westinghouse licensing report dated November 2000.