KEWAUNEE NUCLEAR POWER PLANT

RELOAD SAFETY EVALUATION
KEWAUNEE CYCLE VIII
FEBRUARY 1982

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SAFETY

EVALUATION

FOR

KEWAUNEE

CYCLE VIII

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1.0 INTRODUCTION

The Kewaunee Nuclear Power Plant is in its seventh cycle of operation. Refueling shutdown for Cycle 7 is scheduled for April, 1982 with startup of Cycle 8 forecast for May, 1982.

This report presents an evaluation of the Cycle 8 reload and demonstrates that the core 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. Accident Evaluation methodologies applied in this report are detailed in Reference 2. These reports have been previously reviewed (3). The current 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 (4) limiting safety system setpoints and operating limits including the burnup dependent power

peaking limits described in section 2.2 where applicable.

It has been concluded that the Cycle 8 design is more conservative than results of previously docketed accident analyses. This conclusion is based on the assumptions that:

1. Cycle 7 operation is terminated after 10,500 ± 500 MWD/MTU.
2. There is adherence to plant operating limitations, and Technical Specifications (4).

2.0 CORE DESIGN

2.1 Core Description

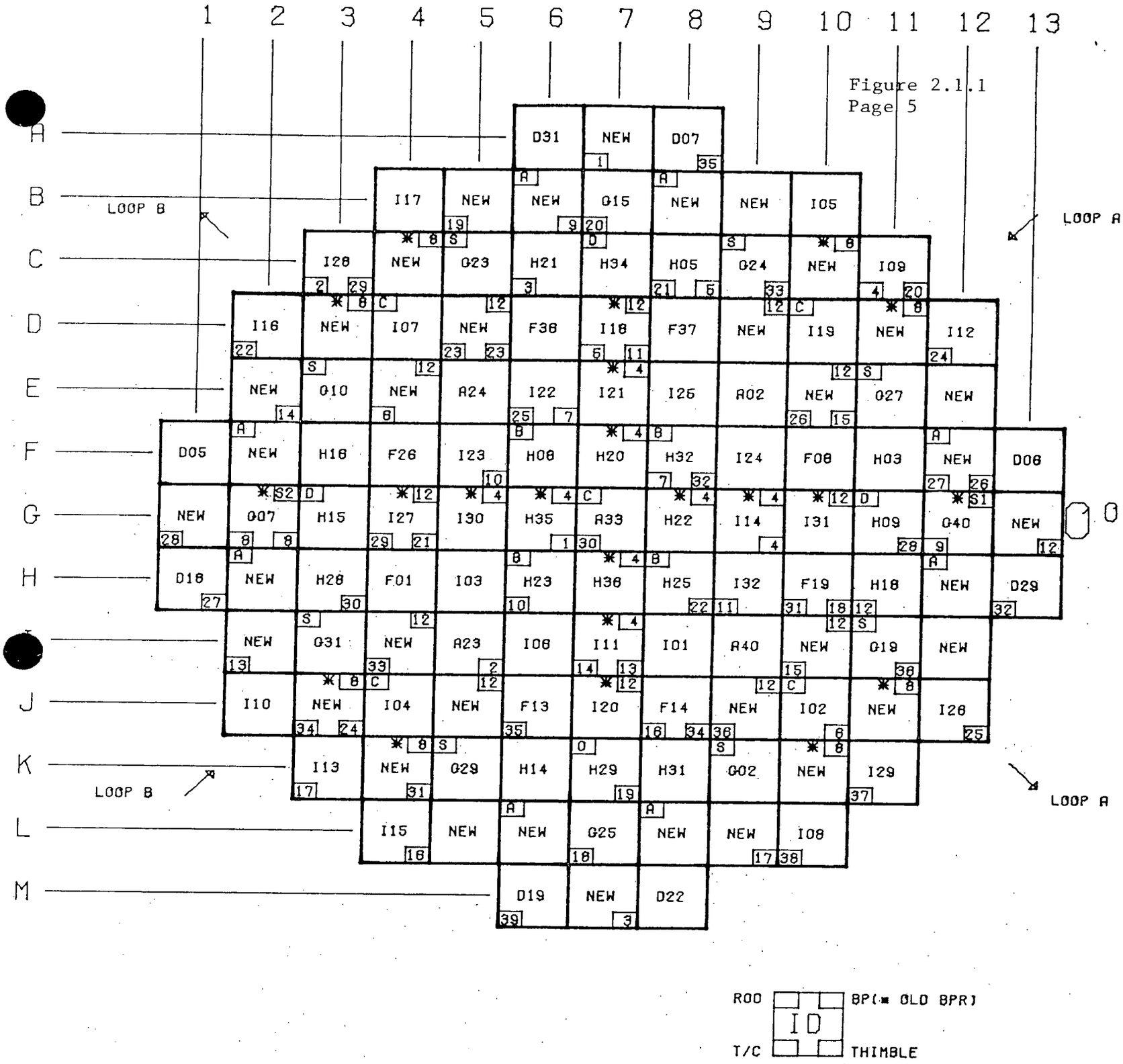
The reactor core consists of 121 fuel assemblies of 14 X 14 design. The core loading pattern, assembly identification, RCCA bank identification, instrument thimble I.D., thermocouple I.D., and burnable poison rod configurations for Cycle 8 are presented in Figure 2.1.1. The Cycle 8 reload core will employ 28 Burnable Poison Rod Assemblies (BPRA'S) containing 96 fresh and 144 partially depleted burnable poison rods.

Thirty-six new Exxon assemblies enriched to 3.2 w/o U235 will reside with sixty-four partially depleted Exxon and twenty-one partially depleted Westinghouse assemblies. Table 2.1.1 displays the core breakdown by region, enrichment and previous cycle duty.

Table 2.1.1

CYCLE 8 FUEL CHARACTERISTICS

Region	Vendor	Initial W/O U235	Number of Previous Duty Cycles	No. of Assemblies
1	W	2.2	1	5
4	W	3.3	4	8
6	W	3.1	3	8
7	ENC	3.2	2	12
8	ENC	3.2	1	4
8	ENC	3.2	2	16
9	ENC	3.2	1	32
10	ENC	3.2	0	36 (FEED)



Kewaunee Cycle 8
Loading Pattern

2.2 Design Objectives and Operating Limits

Power Rating	1650 MWTH
System Pressure	2250 PSIA
Core Average Moderator Temperature (HZP)	547 Degrees F
Core Average Moderator Temperature (HFP)	562 Degrees F

Cycle 8 core design is based on the following design objectives and operating limits.

A. Nuclear Peaking Factor Limits are as follows:

(i) FQ(Z) Limits for all Westinghouse Electric Corp. Fuel

$FQ(Z) \leq (2.22/P) * K(Z)$ for $P > 0.5$
$FQ(Z) \leq (4.44) * K(Z)$ for $P \leq 0.5$

(ii) FQ(Z) Limits for Exxon Nuclear Company Fuel
$FQ(Z) \leq (FQT(Ej)/P) * K(Z)$ for $P > 0.5$
$FQ(Z) \leq 4.42 * K(Z)$ for $P \leq 0.5$

(iii) FΔH Limits for all Fuel
$F\Delta HN \leq 1.55(1 + 0.2(1-P))$ for Exposure ≤ 24000 MWD/MTU
$F\Delta HN \leq 1.52(1 + 0.2(1-P))$ for Exposure > 24000 MWD/MTU

Where P is the fraction of full power at which the core is operating
K(Z) is the function given in Figure 2.2.1
FQT(Ej) is the function given in Figure 2.2.2
Ej is the fuel rod exposure for which FQ is measured
Z is the core height location FQ

B. The moderator temperature coefficient at operating conditions shall be negative.

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 BOC

2.0 % at EOC

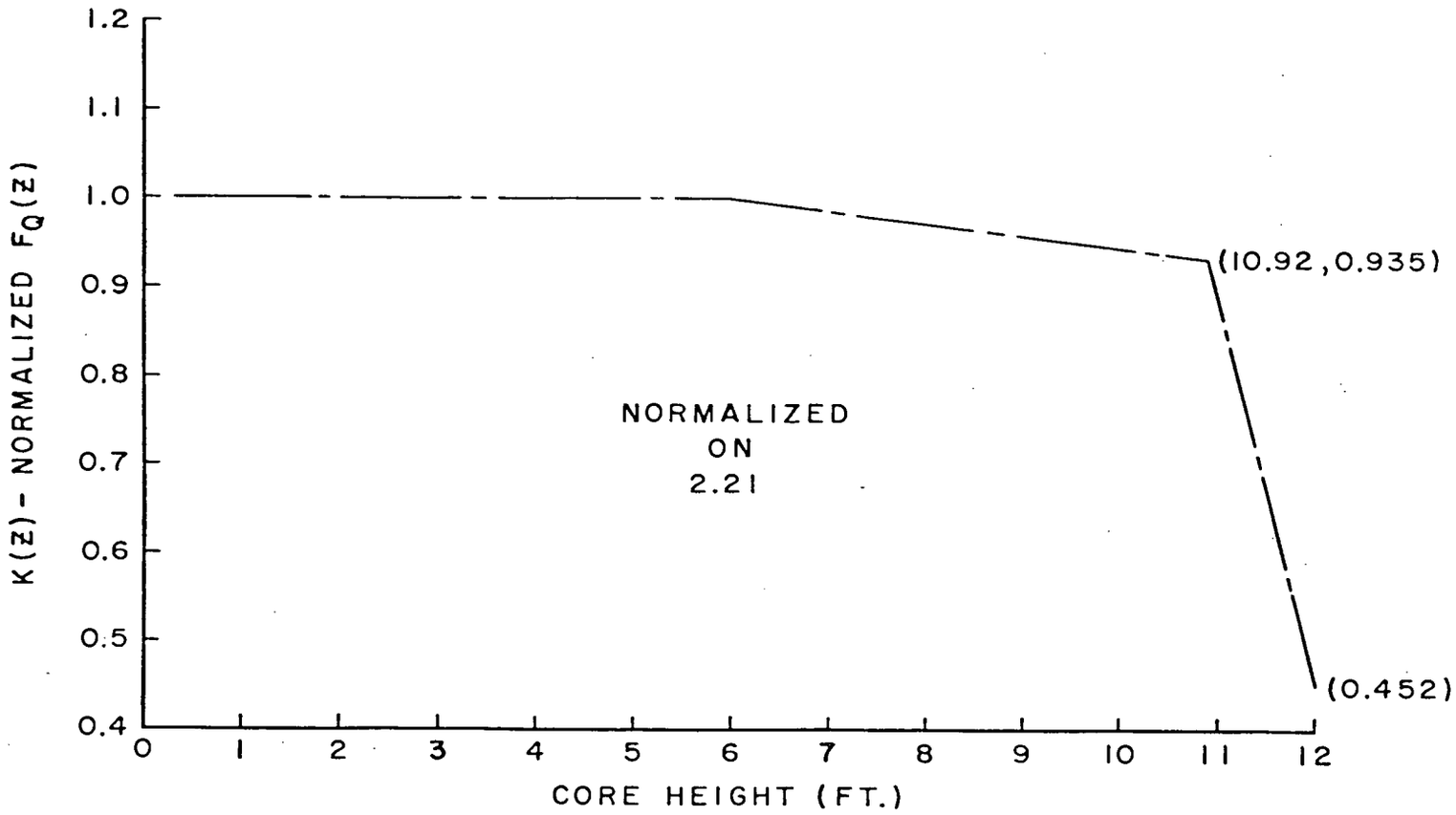
D. The Fuel Loading Pattern shall be capable of generating approximately 10,300 MWD/MTU.

E. The Power Dependent Rod Insertion Limits (PDIL) are presented in Figure 2.2.3. These limits are obtained from Reference 4.

F. The indicated axial flux difference shall be maintained within a ±5% band about the target axial flux difference above 90% power. Figure 2.2.4 shows the axial flux difference limits as a function of core power. These limits are obtained from Reference 4.

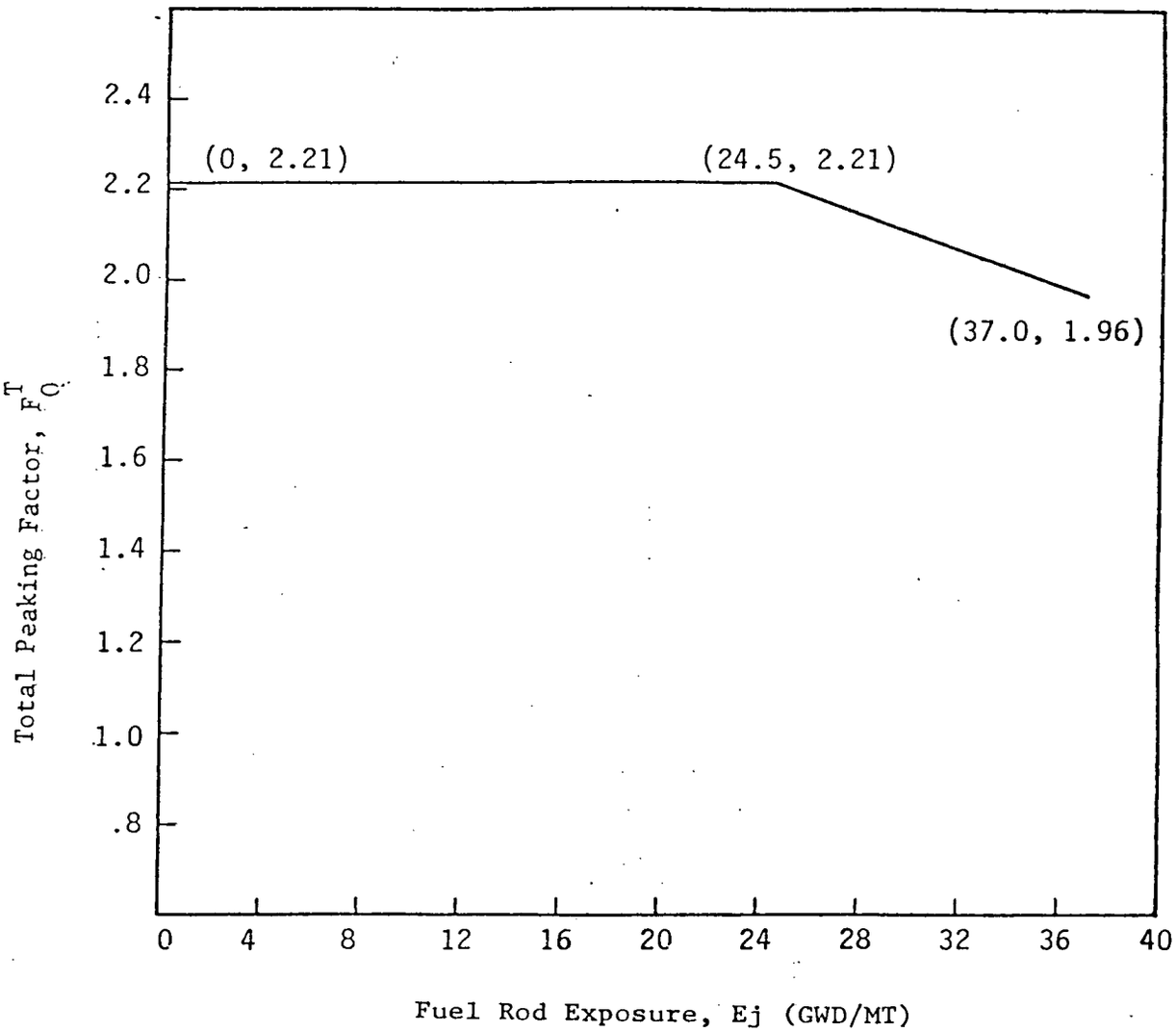
G. A refueling boron concentration of 2100 ppm will be sufficient to maintain the reactor subcritical by 10% $\Delta k/k$ in the cold condition with all rods inserted and will maintain the core subcritical with all rods out of the core.

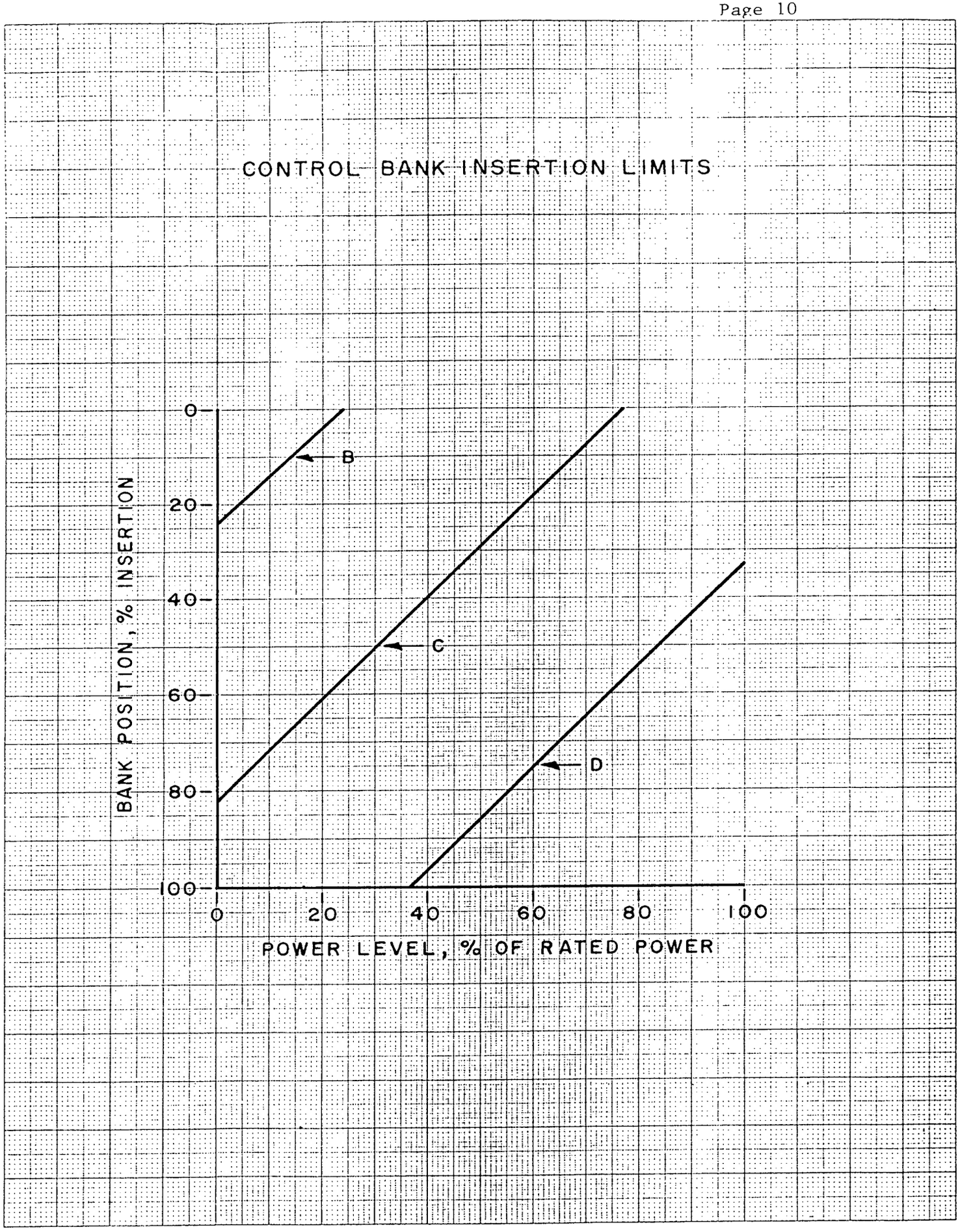
H. Fuel duty expected during this reload will not result in peak fuel rod burnups greater than those analysed by the respective fuel vendors.

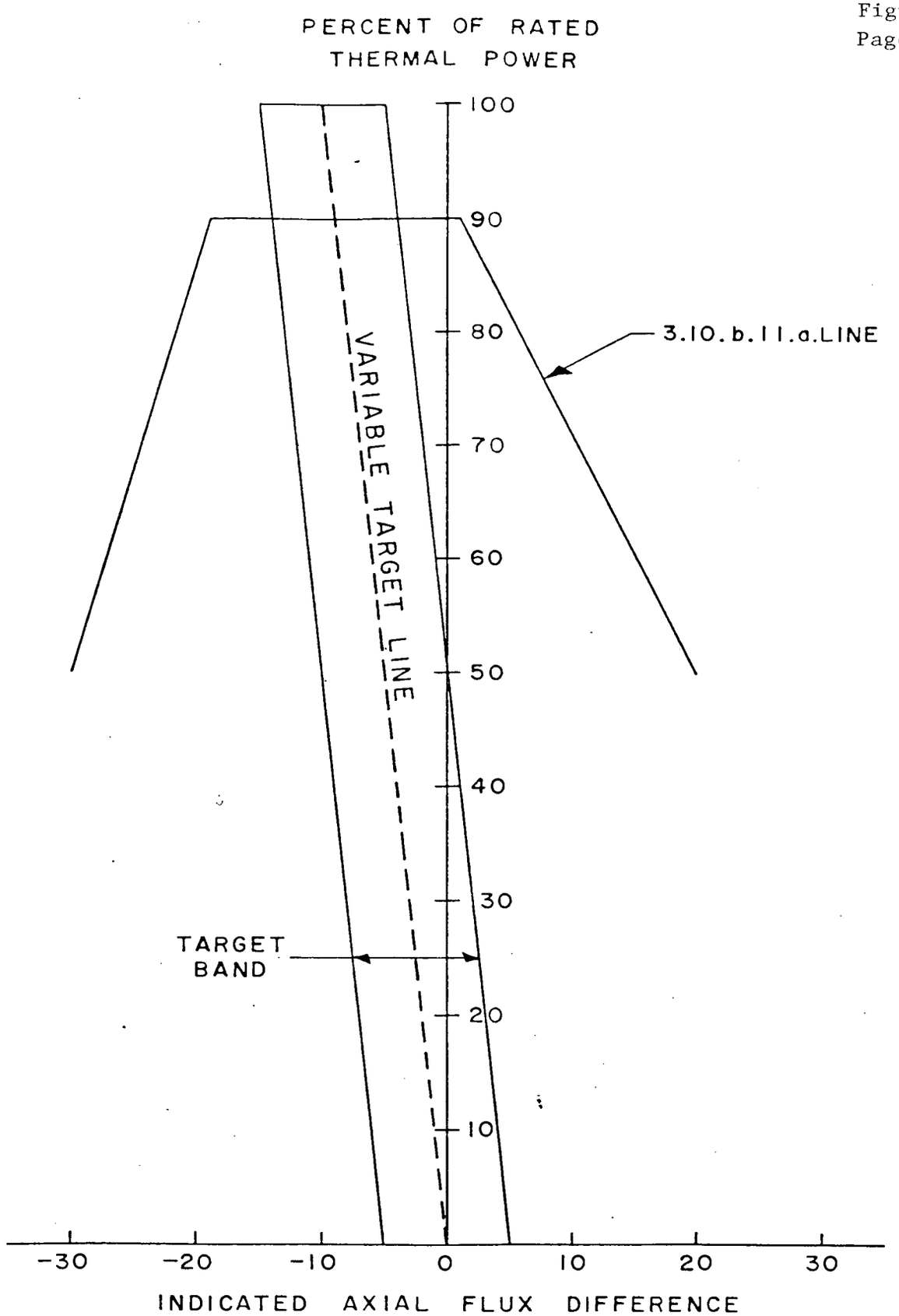


Hot Channel Factor
Normalized Operating Envelope

Kewaunee F_Q^T versus Rod Exposure







Target Band on Indicated Flux Difference
As a Function of Operating Power Level (Typical)

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 8 minimum shutdown margin is 2.55% at end of cycle hot full power conditions. The minimum reload design scram curve is conservatively bounded by the scram curve used in the current accident analyses.

It is concluded that the minimum trip reactivity insertion rate for Cycle 8 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 analyses due to trip reactivity assumptions.

2.4 Shutdown Window

An evaluation of the full power equilibrium peaking factor variation at BOC 8 versus EOC 7 burnup is presented in Table 2.4.1. The values presented have conservatisms applied in accordance with References 1 and 9.

The EOC 7 design shutdown window of ±500 MWD/MTU will not significantly affect the Cycle 8 peaking factors if refueling shutdown of Cycle 7 occurs within this window.

TABLE 2.4.1

Peaking Factor Sensitivity to Shutdown Window

	FΔH		FQ	
	Cycle 8	Limit	Cycle 8	Limit
BOC 8 - 500 MWD/T	1.508	1.55	2.109	2.21
BOC 8 NOMINAL	1.502	1.55	2.129	2.21
BOC 8 + 500 MWD/T	1.517	1.55	2.150	2.21

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 8 accident evaluations.

Table 3.0.1

Kewaunee Nuclear Power Plant

List of Safety Analyses

Accident	Current Analysis	Ref. No.
Uncontrolled RCCA Withdrawal From a Subcritical Condition	2/78 (Cycle 4-RSE)	7
Uncontrolled RCCA Withdrawal at Power	2/78 (Cycle 4-RSE)	7
Control Rod Drop	1/27/71 (AM7-FSAR)	6
RCC Assembly Misalignment	1/27/71 (AM7-FSAR)	6
CVCS Malfunction	1/27/71 (AM7-FSAR)	6
Startup of an Inactive RC Loop	1/27/71 (AM7-FSAR)	6
Excessive Heat Removal Due to FW System Malfunctions	1/27/71 (AM7-FSAR)	6
Excessive Load Increase Incident	1/27/71 (AM7-FSAR)	6
Loss of Reactor Coolant Flow	3/73 (WCAP-8903)	8
Locked Rotor Accident	2/78 (Cycle 4-RSE)	7
Loss of External Electrical Load	1/27/71 (AM7-FSAR)	6
Loss of Normal Feedwater	8/31/73 (AM33-FSAR)	6
Fuel Handling Accidents	1/27/71 (AM7-FSAR)	6
Rupture of a Steam Pipe	4/13/73 (AM28-FSAR)	6
Rupture of CR Drive Mechanism Housing	2/78 (Cycle 4-RSE)	7
RC System Pipe Rupture (LOCA) Westinghouse	12/10/76 (AM40-FSAR)	6
Zirc - Water Addendum	12/14/79	12
Clad Hoop Stress Addendum	1/8/80	13
RC System Pipe Rupture (LOCA) Exxon	1/79 (XN-NF-79-1)	11

Table 3.0.2
Safety Analyses Bounding Values

Parameter	Lower Bound	Upper Bound	Units
Moderator Temperature Coefficient	-35.0	0.0	pcm/°F
Doppler Coefficient	-2.32	-1.0	pcm/°F
Differential Boron Worth	-11.2	N/A	pcm/ppm
Delayed Neutron Fraction	.0050	.0071	
Prompt Neutron Lifetime	20	N/A	μ sec
Shutdown Margin	1.0	2.0	%Δρ
Differential Rod Worth of 2 Banks Moving	N/A	82	pcm/sec
Ejected Rod Cases			
HFP, BCL			
βeff	.0055	N/A	
Rod Worth	N/A	.30	%Δρ
FQ	N/A	5.03	
HFP, ECL			
βeff	.0050	N/A	
Rod Worth	N/A	.42	%Δρ
FQ	N/A	5.1	
HZP, BOL			
βeff	.0055	N/A	
Rod Worth	N/A	.92	%Δρ
FQ	N/A	13.0	
HZP, ECL			
βeff	.0050	N/A	
Rod Worth	N/A	.92	%Δρ
FQ	N/A	13.0	

3.1 Evaluation of Uncontrolled Rod Withdrawal from Subcritical

Table 3.1.1 presents a comparison of Cycle 8 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 8 reload core are conservatively bounded by those used in the current safety analysis, an uncontrolled rod withdrawal from a subcritical condition will be less severe than the transient in the current analysis. The implementation of the Cycle 8 reload core design, therefore, 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	2.1	≤	10.0	pcm/°Fm
B) Doppler Temp. Coefficient	-2.3	≤	-1.0	pcm/°Ff
C) Differential Worth of Two Moving Banks	48	≤	82	pcm/sec
D) Scram Worth vs. Time	See Section 2.3			
E) Delayed Neutron Fraction	.00545	≥	.0050	

3.2 Evaluation of Uncontrolled Rod Withdrawal at Power

Table 3.2.1 presents a comparison of the Cycle 8 physics parameters to the current safety analysis values for the Uncontrolled Rod Withdrawal at Power Accident.

The application of the reliability factor to the moderator coefficient calculated at HZP, no xenon core conditions results in a slightly positive value. It is anticipated that BOC Startup Physics Test measurements will demonstrate that the moderator coefficient will be negative at operating conditions.

Since the pertinent parameters from the proposed Cycle 8 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. The implementation of the Cycle 8 reload core design, therefore, 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	2.1*	≤	0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.6	≤	-1.0	pcm/°Ff
C) Differential Rod Worth Of Two Moving Banks	48	≤	82	pcm/sec
D) FΔHN	1.53	≤	1.55	
E) Scram Worth vs. Time	See Section 2.3			

* Moderator Temperature Coefficient will be verified negative at Startup Testing.

3.3 Evaluation of Control Rod Misalignment

Table 3.3.1 presents a comparison of the Cycle 8 $F^{\Delta H}_{N}$ versus the current safety analysis $F^{\Delta H}_{N}$ limit for the Misaligned Rod Accident.

Since the pertinent parameter from the proposed Cycle 8 reload core 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 analysis. The implementation of the Cycle 8 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.3.1

CONTROL ROD MISALIGNMENT ACCIDENT

Parameter	Reload Safety Evaluation Value		Current Safety Analysis
A) $F_{\Delta H}^{N}$	1.87	$\leq$	1.92

3.4 Evaluation of Dropped Rod

A comparison of the Cycle 8 $F\Delta HN$ to the current safety analysis $F\Delta HN$ limit for the Dropped Rod Accident is presented in Table 3.4.1.

Since the pertinent parameter from the proposed Cycle 8 reload core is 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. The implementation of the Cycle 8 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.4.1

DROPPED ROD ACCIDENT

Parameter	Reload Safety Evaluation Values		Current Safety Analysis
A) $F^{\Delta H}_{N}$	1.67	$\leq$	1.92

3.5 Evaluation of Uncontrolled Boron Dilution

Table 3.5.1 presents a comparison of Cycle 8 physics analysis results to the current safety analysis values for the Uncontrolled Boron Dilution Accident for refueling and full power core conditions.

The application of the reliability factor to the moderator coefficient calculated at HZP, no xenon core conditions results in a slightly positive value. It is anticipated that BOC Startup Physics Test measurements will demonstrate that the moderator coefficient will be negative at operating conditions.

Since the pertinent parameters from the proposed Cycle 8 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. The implementation of the Cycle 8 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.5.1

UNCONTROLLED BORON DILUTION ACCIDENT

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
i) Refueling Conditions				
A) Shutdown Margin (ARI)	11.4	≥	10.0	%Δρ
ii) At Power Conditions				
A) Moderator Temp. Coefficient	2.1*	≤	0.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.6	≤	-1.0	pcm/°Ff
C) Reactivity Insertion Rate by Boron	1.48	≤	1.60	pcm/sec
D) Shutdown Margin	2.55	≥	1.0	%Δρ
E) FΔHN	1.53	≤	1.55	

* Moderator Temperature Coefficient will be verified negative at Startup Testing.

3.6 Evaluation of Startup of an Inactive Loop

Table 3.6.1 presents a comparison of Cycle 8 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 8 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. The implementation of the Cycle 8 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.6.1

STARTUP OF AN INACTIVE LOOP ACCIDENT

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temperature Coefficient	-32.2	≥	-35.0	pcm/°Fm
B) Doppler Temperature Coefficient	-2.24	≤	-1.0	pcm/°Ff
C) FΔHN	1.53	≤	1.55	

3.7 Evaluation of Feedwater System Malfunction

A comparison of Cycle 8 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 8 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. The implementation of the Cycle 8 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.7.1

FEEDWATER SYSTEM MALFUNCTION ACCIDENT

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temperature Coefficient	-2.1	$\leq$	0	pcm/°Fm
B) Doppler Temperature Coefficient	-1.6	$\leq$	-1.0	pcm/°Ff
C) FΔHN	1.53	$\leq$	1.55	
D) Moderator Temperature Coefficient (maximum)	-29.2	$\geq$	-35.0	pcm/°Fm

3.8 Evaluation of Excessive Load Increase

Table 3.8.1 presents a comparison of Cycle 8 physics results to the current safety analysis values for the Excessive Load Increase Accident.

Since the pertinent parameters from the proposed Cycle 8 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. The implementation of the Cycle 8 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.8.1

EXCESSIVE LOAD INCREASE ACCIDENT

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temperature Coefficient (minimum)	-2.1	≤	0	pcm/°Fm
B) Moderator Temperature Coefficient (maximum)	-29.2	≥	-35.0	pcm/°Fm
C) Doppler Temperature Coefficient (BOL)	-1.6	≤	-1.0	pcm/°Ff
D) FΔHN	1.53	≤	1.55	

3.9 Evaluation of Loss of Load

A comparison of Cycle 8 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 8 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. The implementation of the Cycle 8 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.9.1

LOSS OF LOAD ACCIDENT

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-2.1	≤	0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.60	≤	-1.0	pcm/°Fm
C) FΔHN	1.53	≤	1.55	
D) Scram Worth Versus Time	See Section 2.3			

3.10 Evaluation of Loss of Normal Feedwater

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

Table 3.11.1 presents a comparison of Cycle 8 calculational 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 8 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. The implementation of the Cycle 8 reload core design, therefore, 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	-2.1	$\leq$	0	pcm/^{o}Fm
B) Doppler Temp. Coefficient	-2.26	$\geq$	-2.32	pcm/^{o}Ff
C) FΔHN	1.53	$\leq$	1.55	
D) Scram Worth Versus Time	See Section 2.3			

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

Table 3.12.1 presents a comparison of Cycle 8 physics parameters to the current safety analysis values for the Locked Rotor Accident.

Since the pertinent parameters from the proposed Cycle 8 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. The implementation of the Cycle 8 reload core design, therefore, 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	-2.1	≤	0	pcm/°Fm
B) Doppler Temp. Coefficient	-2.26	≥	-2.32	pcm/°Ff
C) Delayed Neutron Fraction	.00545	≥	.0051	
D) Percent Pins > Limiting FΔHN (DNBR=1.3)	24.0	≤	40.0	%
E) Scram Worth Versus Time	See Section 2.3			

3.13 Evaluation of Main Steam Line Rupture

The minimum Cycle 8 shutdown margin is compared to that assumed in the safety analysis in Table 3.13.1. Figure 3.13.1 compares the Cycle 8 keff versus moderator temperature at 1000 psia to the current safety analysis limiting cooldown reactivity curve.

Since the pertinent parameters from the proposed Cycle 8 reload core are conservatively bounded by those used in the current safety analysis, a main steam line rupture accident will be less severe than the transient in the current analysis. The implementation of the Cycle 8 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

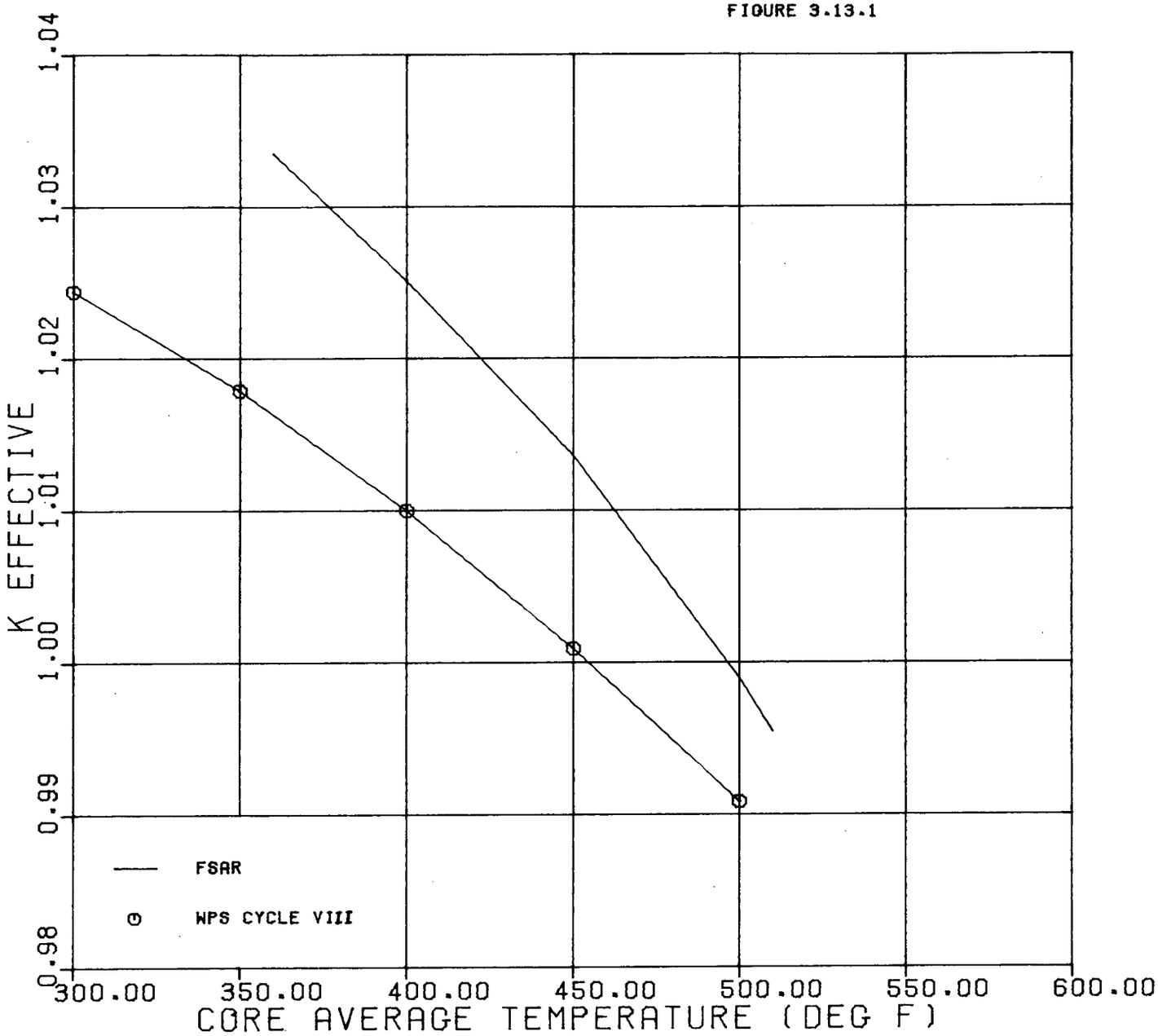
Table 3.13.1

MAIN STEAM LINE RUPTURE ACCIDENT

Parameter	Reload Safety Evaluation Value		Current Safety Analysis	Unit
A) Shutdown Margin	2.55	≥	2.0	% Δρ

VARIATION OF REACTIVITY, WITH CORE TEMPERATURE AT 1000 PSIA FOR THE END OF LIFE RODDED CORE WITH ONE ROD STUCK (ZERO POWER)

FIGURE 3.13.1



3.14 Evaluation of Rod Ejection Accidents

Tables 3.14.1 thru 3.14.4 present the comparison of Cycle 8 calculated physics parameters to the current safety analysis values for the Rod Ejection Accident at zero and full power, BOL and EOL core conditions.

The application of the reliability factor to the moderator coefficient calculated at HZP, BOL, no xenon core conditions results in a slightly positive value. It is anticipated that BOC Startup Physics Test measurements will demonstrate that the moderator coefficient will be negative at operating conditions.

Since the pertinent parameters from the proposed Cycle 8 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. The implementation of the Cycle 8 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.14.1

ROD EJECTION ACCIDENTS

HFP, BOL

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-2.1	≤	0	pcm/°Fm
B) Delayed Neutron Fraction	.00615	≥	.0055	
C) Ejected Rod Worth	.18	≤	.30	%Δρ
D) Doppler Temp. Coefficient	-1.6	≤	-1.0	pcm/°Ff
E) Prompt Neutron Lifetime	29.8	≥	20	μsec
F) FQN	3.1	≤	5.03	
E) Scram Worth Versus Time	See Section 2.3			

Table 3.14.2
ROD EJECTION ACCIDENTS

HZP, BOL

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	2.1*	≤	0	pcm/°Fm
B) Delayed Neutron Fraction	.00615	≥	.0055	
C) Ejected Rod Worth	.49	≤	.91	% Δρ
D) Doppler Temp. Coefficient	-1.6	≤	-1.0	pcm/°Ff
E) Prompt Neutron Lifetime	29.8	≥	20	μsec
F) FQN	4.9	≤	11.2	
G) Scram Worth Versus Time	See Section 2.3			

* Moderator Temperature Coefficient will be verified negative at Startup Testing.

Table 3.14.3
ROD EJECTION ACCIDENTS

HFP, EOL

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-14.9	≤	0	pcm/°Fm
B) Delayed Neutron Fraction	.00545	≥	.0050	
C) Ejected Rod Worth	.30	≤	.42	% Δρ
D) Doppler Temp. Coefficient	-1.85	≤	-1.0	pcm/°Ff
E) Prompt Neutron Lifetime	32.7	≥	20	μsec
F) FQN	3.2	≤	5.1	
G) Scram Worth Versus Time	See Section 2.3			

Table 3.14.4
ROD EJECTION ACCIDENTS

HZP, EOL

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-10.1	≤	0	pcm/°Fm
B) Delayed Neutron Fraction	.00545	≥	.0050	
C) Ejected Rod Worth	.78	≤	.92	% Δρ
D) Doppler Temp. Coefficient	-1.85	≤	-1.0	pcm/°Ff
E) Prompt Neutron Lifetime	32.7	≥	20	μsec
F) FQN	7.0	≤	13.0	
E) Scram Worth Versus Time	See Section 2.3			

3.15 Evaluation of Fuel Handling Accident

Table 3.15.1 presents a comparison of the Cycle 8 FQN, calculated at end of Cycle 8 less 2.0 GWD/MTU, to the current safety analysis FQN limit for the Fuel Handling Accident.

Since the pertinent parameter from the proposed Cycle 8 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. The implementation of the Cycle 8 reload core design, therefore, 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) FQN	1.99	$\leq$	2.53

3.16 Evaluation of Loss of Coolant Accident

Table 3.16.1 presents the comparison of Cycle 8 physics calculation results to the current safety analysis values for the Loss of Coolant Accident.

Since the pertinent parameters from the proposed Cycle 8 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. The implementation of the Cycle 8 reload core design, therefore, 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
A) Scram Worth Versus Time	See Section 2.3	
B) FQ	See Section 3.17	

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 procedures defined in Section 2.2 of this report (9).

Following these procedures, 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).

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 8 operation will not preclude full power operation under the power distribution control specifications currently applied (10).

MAX (FQ * P REL) VS AXIAL
CORE HEIGHT CYCLE 8
S3D 81355.0830

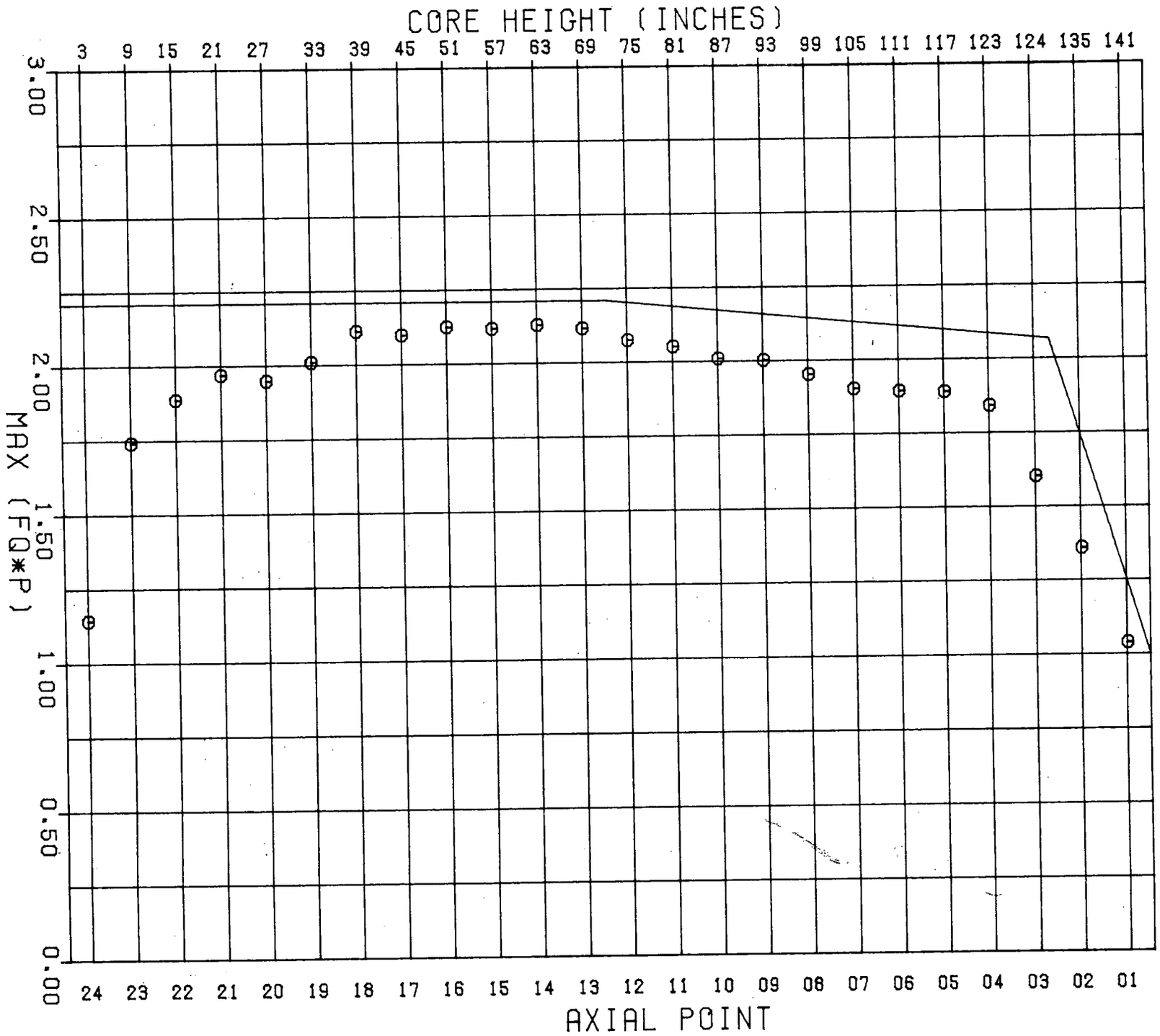


Figure 3.17.1
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4.0 TECHNICAL SPECIFICATIONS

No Technical Specification Amendments are required as a result of this reload core implementation.

5.0 STATISTICS UPDATE

In an effort to provide continuing assurance of the model applicability, Cycle 6 measurements and calculations were added to the statistics data base prior to model applications to the Cycle 8 Reload Analysis. The reliability and bias factors applicable to Cycle 8 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	
$F^{\Delta}H$	4.0%	0
Rod Worth	10.0%	0
Moderator Temperature Coefficient	5.36 PCM/^{o}F	-3.11 PCM/^{o}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	Node	RF (%)
1 (Bottom)	0.082	14.17
2	0.036	6.60
3	0.035	6.41
4	0.030	5.67
5	0.033	6.20
6	0.038	7.00
7	0.039	7.05
8	0.038	6.91
9	0.032	6.05
10	0.031	5.90
11	0.030	5.69
12	0.029	5.58
13	0.024	4.92
14	0.026	5.11
15	0.024	4.85
16	0.024	4.84
17	0.021	4.54
18	0.019	4.25
19	0.024	4.91
20	0.020	4.33
21	0.039	7.13
22	0.026	5.19
23	0.078	13.40
24 (Top)	0.060	10.46

6.0 REFERENCES

1. Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, Topical Report Titled, "Qualification of Reactor Physics Methods for Application to Kewaunee."

2. Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, Topical Report Titled, "Reload Safety Evaluation Methods for Application to Kewaunee."

3. Safety Evaluation by the Office of Nuclear Reactor Regulation on 'Qualifications of Reactor Physics Methods for Application to Kewaunee' Report. October 22, 1979

4. Wisconsin Public Service Corporation, Technical Specifications for Kewaunee Nuclear Power Plant.

5. Exxon Nuclear Company, "Generic Mechanical and Thermal Hydraulic Design for Exxon Nuclear 14 X 14 Reload Fuel Assemblies with Zircaloy Guide Tubes for Westinghouse 2-Loop Pressurized Water Reactors." November 1978

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

7. "Reload Safety Evaluation," Kewaunee Nuclear Power Plant Cycles 2, 3, and 4.

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9. R.J. Burnside and J.S. Holm, "Exxon Nuclear Power Distribution Control For Pressurized Water Reactors, Phase II" XN-NF-77-57 Exxon Nuclear Company, Inc. January 1978

10. Proposed Amendment 48 to the KNPP Technical Specifications. Letter from E.R. Mathews to D.G. Eisenhut, November 23, 1981.

11. "ECCS Analysis for Kewaunee Using ENC WREM-IIA PWR Evaluation Model (2)," XN-NF-79-1 Exxon Nuclear Company, Inc. January 1979

12. ECCS Reanalysis - ZIRC/Water Reaction Calculation. Letter from E.R. Mathews to A. Schwencer, December 14, 1979.

13. Clad Swelling and Fuel Blockage Models. Letter from E.R. Mathews to D.G. Eisenhut, January 8, 1980.