KEWAUNEE NUCLEAR POWER PLANT

RELOAD SAFETY EVALUATION
KEWAUNEE CYCLE V
APRIL 1979

CORPORATION

HT COMPANY

C COMPANY

R E L O A D

S A F E T Y

E V A L U A T I O N

F O R

K E W A U N E E

C Y C L E   V

April, 1979

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Table of Contents

RSE Cycle 5

List of Tables

List of Figures

1.0 INTRODUCTION

The Kewaunee Nuclear Power Plant is in its fourth cycle of operation. Refueling shutdown for Cycle 5 is scheduled for May, 1979 with startup forecast for early June, 1979.

This report presents an evaluation of the Cycle 5 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 design core are reviewed. Details of the Reload Safety Evaluation accident evaluation methodologies applicable to reload cores are presented in the topical report, "Reload Safety Evaluation Methods for Application to Kewaunee", submitted February, 1979.

Details of the calculational model used to generate physics parameters for this Reload Safety Evaluation are presented in the topical report "Qualification of Reactor Physics Methods for Application to Kewaunee" submitted September, 1978.

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 used the current Technical Specification limiting safety system setpoints.

It has been concluded that the Cycle 5 design is more conservative than results of previously docketed accident analyses. This conclusion is based on the assumptions that:

1. Cycle 4 operation is terminated after 11,500 $\pm$ 500 MWD/MTU.
2. There is adherence to plant operating limitations.

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 and control rod, incore thimble, thermocouple, and burnable poison rod locations for Cycle 5 are presented in Figure 2.1.1. The location of depleted burnable poisons and source assemblies are indicated in Figure 2.1.1 by * and ■ respectively.

Table 2.1.1 shows the fuel enrichment and densities by region. Forty fresh assemblies in this reload are of the Exxon design which is neutronically and thermal hydraulically compatible to the current resident fuel assemblies. Reference 6, which has been submitted on the Northern States Power Prairie Island docket, describes the Exxon 14 x 14 fuel design.

The nominal design end of Cycle 5 burnup is 10,060 MWD/MTU.

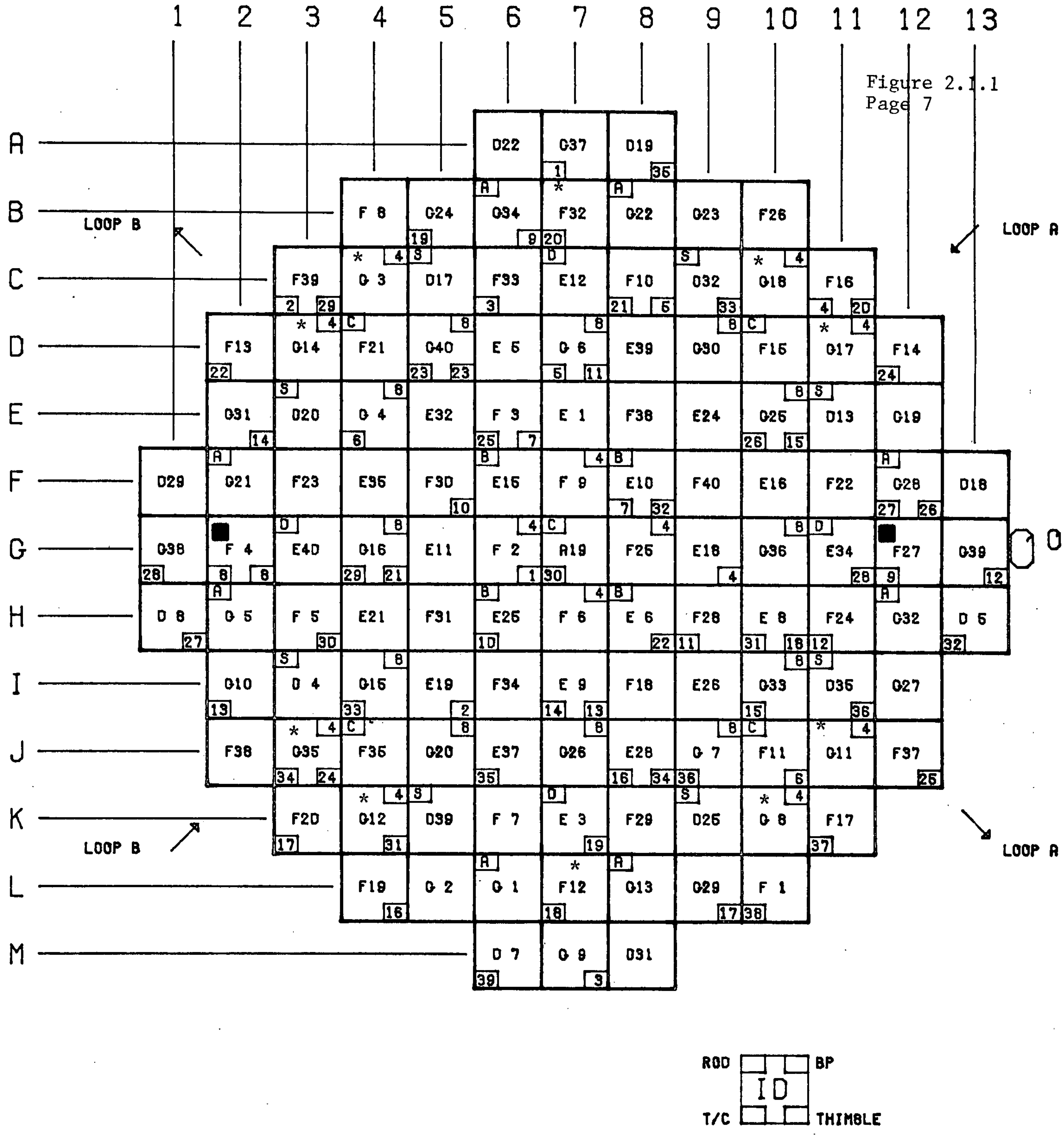
Table 2.1.1

FUEL ENRICHMENT AND DENSITY CORE 5

Region	W/O U_{235}	Percent Theoretical Density	No. of Assemblies
A	2.20	93.6	1
D	3.28	94.5	16
E	3.30	94.5	24
F	3.10	94.5	40
G *	3.20	94.0	40

*Exxon supplied fuel

Figure 2.1.1
Page 7



*Once Depleted BPR's
■ New Source Locations

CORE FIVE

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)	563 Degrees F

Cycle 5 core design is based on the following design objectives and operating limits.

A. Nuclear Peaking Factor Limits are as follows:

$FQ(Z) \leq (2.16/P) * K(Z)†$ for $P > .5$

$FQ(Z) \leq 4.50 * K(Z)†$ for $P \leq .5$

$F\Delta_H^N \leq 1.55(1 + .2(1-P))$ P = Relative Reactor Power

†K(Z) is shown in Figure 2.2.1

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

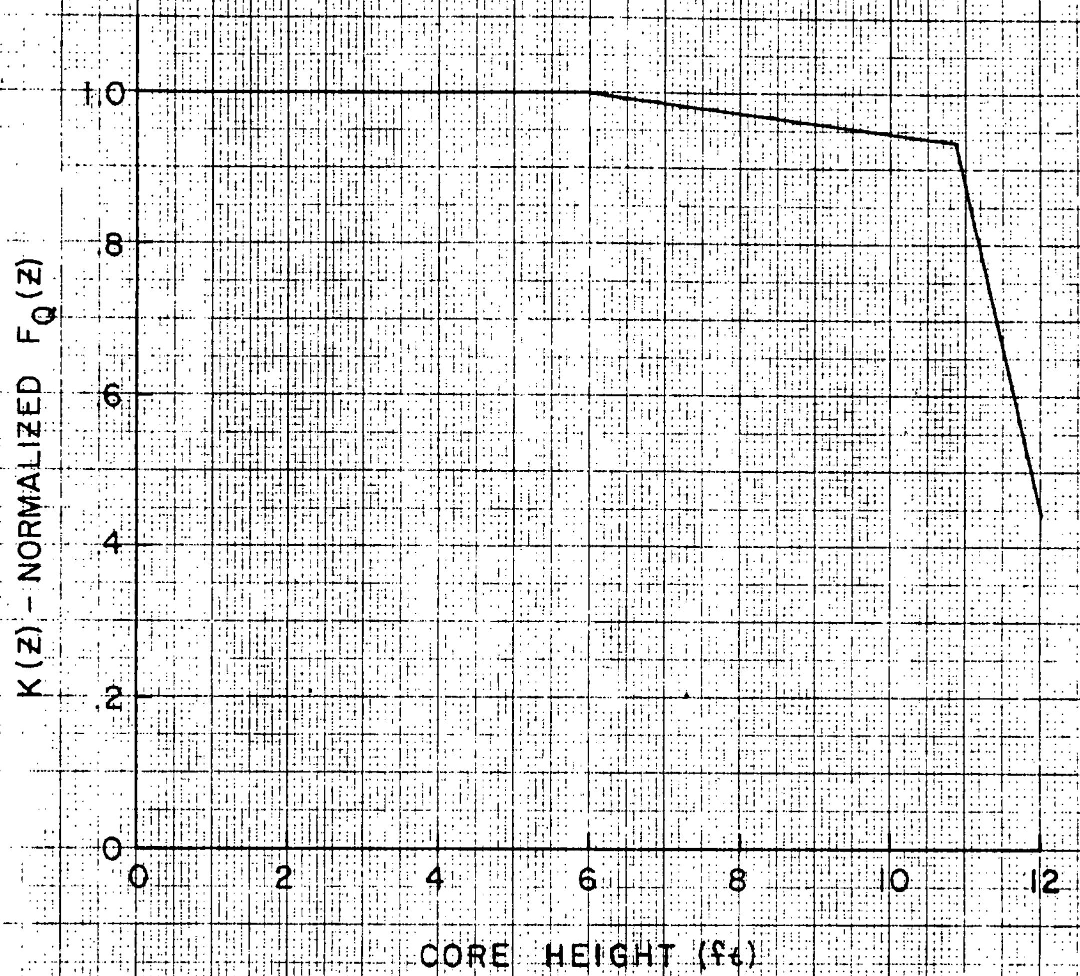
D. The Fuel Loading Pattern shall be capable of generating approximately 10,060 MWD/MTU.

E. The Power Dependent Rod Insertion Limits (PDIL) are presented in Figure 2.2.2. These limits are derived from Reference 3.

F. The indicated axial flux difference shall be maintained within a $\pm$ 5% band about the target axial flux difference above 90% power. Figure 2.2.3 shows the axial flux difference limits as a function of core power. These limits are derived from Reference 3.

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.

Figure 2.2.1



HOT CHANNEL FACTOR
NORMALIZED OPERATING ENVELOPE

CONTROL BANK INSERTION LIMITS

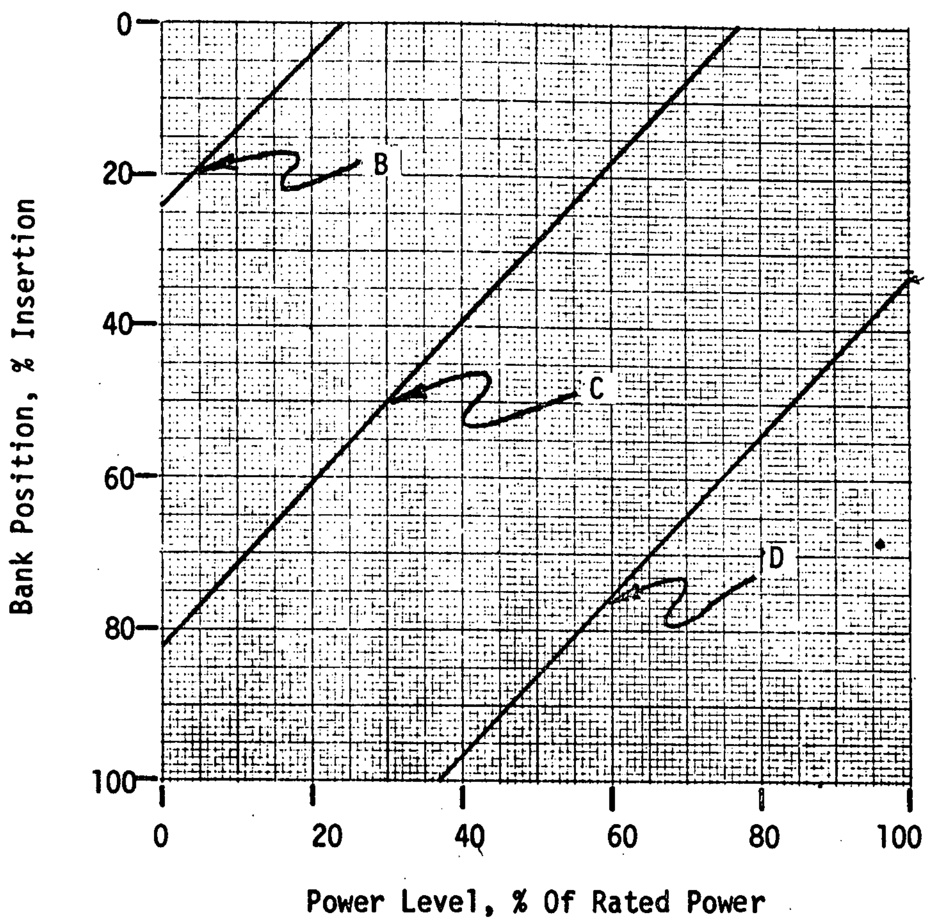
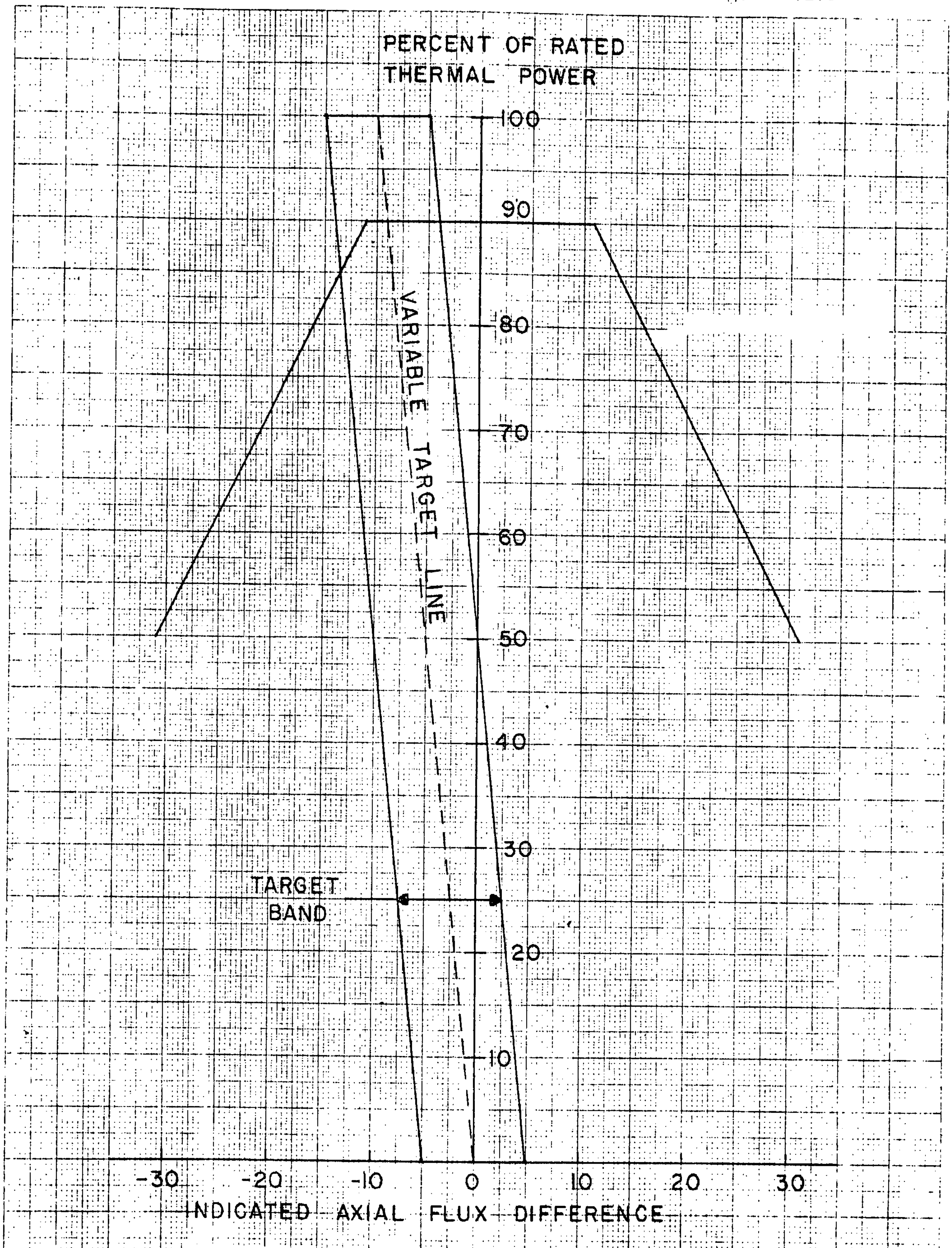


Figure 2.2.3



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 Core 5 minimum shutdown margin is 2.53% which occurs at hot full power and end of cycle. This minimum reload design curve has been compared to the bounding scram curve used in the accident analyses.

It is concluded that the minimum trip reactivity insertion rate for Cycle 5 is conservative with respect to the bounding value.

Thus, for all accidents in which credit is taken for a reactor trip, the proposed reload core will not adversely effect the results of the safety analyses due to trip reactivity assumptions.

2.4 Shutdown Window

An evaluation of the sensitivity of the Cycle 5 peaking factors to the Cycle 4 EOC burnup is presented in Table 2.4.1

It is evident that the EOC4 design shutdown window of $\pm$ 500 MWD/MTU will not significantly effect the Cycle 5 peaking factors, and therefore no further safety evaluation is required for the proposed core reload if refueling shutdown of the previous core occurs within this window.

Table 2.4.1

Peaking Factor Sensitivity to Shutdown Window

	FQ		FΔH	
	Cycle 5	Limit	Cycle 5	Limit
EOC 4 - 500	2.0469	2.16	1.521	1.55
EOC 4 Nom.	2.0428	2.16	1.517	1.55
EOC 4 + 500	2.0381	2.16	1.532	1.55

3.0 ACCIDENT EVALUATIONS

Table 3.0.1 presents the latest safety analyses performed for the accidents which are evaluated in Sections 3.1 to 3.16 of the RSE. The bounding values derived from these analyses are shown in Table 3.0.2 and will be applied in the Cycle 5 accident evaluations.

Table 3.0.1

Kewaunee Nuclear Power Plant

List of Safety Analyses

Accident	Latest Analysis
Uncontrolled RCCA Withdrawal From a Subcritical Condition	2/78 (Core 4 - RSE)
Uncontrolled RCCA Withdrawal at Power	2/78 (Core 4 - RSE)
Control Rod Drop	1/27/71 (AM7 - FSAR)
RCC Assembly Misalignment	1/27/71 (AM7 - FSAR)
CVCS Malfunction	1/27/71 (AM7 - FSAR)
Startup of an Inactive RC Loop	1/27/71 (AM7 - FSAR)
Excessive Heat Removal Due to FW System Malfunctions	1/27/71 (AM7 - FSAR)
Excessive Load Increase Incident	1/27/71 (AM7 - FSAR)
Loss of Reactor Coolant Flow	3/73 (WCAP-8903)
Locked Rotor Accident	2/78 (Core 4 - RSE)
Loss of External Electrical Load	1/27/71 (AM7 - FSAR)
Loss of Normal Feedwater	8/31/73 (AM33 - FSAR)
Fuel Handling Accidents	1/27/71 (AM7 - FSAR)
Rupture of a Steam Pipe	4/13/73 (AM28 - FSAR)
Rupture of a CR Drive Mechanism Housing	2/78 (Core 4 - RSE)
RC System Pipe Rupture (LOCA)	12/10/76 (AM40 - FSAR)

Table 3.0.2

Safety Analyses Bounding Values

Parameter	Lower Bound	Upper Bound	Units
Moderator Temperature Coefficient	-35.0	0.0	pcm/^{o}F
Doppler Coefficient	-2.32	-1.0	pcm/^{o}F
Differential Boron Worth	-11.2	N/A	pcm/ppm
Delayed Neutron Fraction	.0051	.0071	
Prompt Neutron Lifetime	20	N/A	μsec
Shutdown Margin	1.0	2.0	% Δρ
Differential Rod Worth of 2 Banks Moving	N/A	82	pcm/sec
Ejected Rod Cases			
HFP, BOL			
βeff	.0055	N/A	
Rod Worth	N/A	.30	% Δρ
FQ	N/A	5.03	
HFP, EOL			
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	.91	% Δρ
FQ	N/A	11.2	
HZP, EOL			
βeff	.0050	N/A	
Rod Worth	N/A	.92	% Δρ
FQ	N/A	13.0	

Table 3.1.1

Uncontrolled Rod Withdrawal From Subcritical

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temperature Coefficient	$+2.5 * 10^{-5}$	$\leq$	$1.0 * 10^{-4}$	$\Delta\rho/^{o}Fm$
B) Doppler Temperature Coefficient	$-1.481 * 10^{-5}$	$\leq$	$-1.0 * 10^{-5}$	$\Delta\rho/^{o}Ff$
C) Differential Worth of Two Moving Banks	$2.23 * 10^{-4}$	$\leq$	$8.2 * 10^{-4}$	Δk/sec
D) Scram Worth Insertion Rate	See Section 2.3			

3.1 Evaluation of Uncontrolled Rod Withdrawal from Subcritical Accident

Table 3.1.1 presents a comparison of Cycle 5 physics parameters to the current safety analysis values for the Uncontrolled Rod Withdrawal from Subcritical Accident.

Since the pertinent parameters from the proposed Cycle 5 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 analysis. The implementation of the Cycle 5 reload core design, therefore, will not adversely effect the safe operation of the Kewaunee Plant.

3.2 Evaluation of Uncontrolled Rod Withdrawal at Power Accident

Table 3.2.1 presents a comparison of the Cycle 5 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 measurements at operating conditions will demonstrate a moderator coefficient which is conservative with respect to the current safety analysis.

Since the pertinent parameters from the proposed Cycle 5 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 5 reload core design, therefore, will not adversely effect 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 Temperature Coefficient (minimum feedback)	$2.5 * 10^{-5}$	$\nleq$	0	$\Delta\rho/^{o}Fm$
B) Doppler Temperature Coefficient (minimum feedback)	$-1.152 * 10^{-5}$	$\leq$	$-1.0 * 10^{-5}$	$\Delta\rho/^{o}Ff$
C) Differential Rod Worth In Motion (maximum)	$2.23 * 10^{-4}$	$\leq$	$8.2 * 10^{-4}$	Δk/sec
D) $F\Delta_H^N$	1.52	$\leq$	1.55	
E) Scram Worth vs. Time	See Section 2.3			

3.3 Evaluation of Control Rod Misalignment Accident

Table 3.3.1 presents a comparison of the Cycle 5 $F\Delta_H^N$ versus the current safety analysis $F\Delta_H^N$ limit for the misaligned rod accident.

Since the pertinent parameter from the proposed Cycle 5 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 5 reload core design, therefore, will not adversely effect the safe operation of the Kewaunee Plant.

Table 3.3.1

CONTROL ROD MISALIGNMENT ACCIDENT

Parameter	Reload Safety Evaluation Value		Current Safety Analysis
A) $F\Delta_H^N$	1.79	$\leq$	1.92

3.4 Evaluation of Dropped Rod Accident

A comparison of the Cycle 5 $F\Delta_H^N$ to the current safety analysis $F\Delta_H^N$ limit for the dropped rod accident is presented in Table 3.4.1.

Since the pertinent parameter from the proposed Cycle 5 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 5 reload core design, therefore, will not adversely effect the safe operation of the Kewaunee Plant.

Table 3.4.1

DROPPED ROD ACCIDENT

Parameter	Reload Safety Evaluation Values		Current Safety Analysis
A) $F\Delta^N_H$	1.65	$\leq$	1.92

3.5 Evaluation of Uncontrolled Boron Dilution Accident

Table 3.5.1 presents a comparison of Cycle 5 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 measurements at operating conditions will demonstrate a moderator coefficient which is conservative with respect to the current safety analysis.

Since the pertinent parameters from the proposed Cycle 5 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 5 reload core design, therefore, will not adversely effect the safe operation of the Kewaunee Plant.

Table 3.5.1

UNCONTROLLED BORON DILUTION ACCIDENT

i) Refueling Conditions

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Shutdown Margin (ARI)	11.1	$\geq$	10.0	$\%\Delta K$

ii) At Power Conditions

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temperature Coefficient (minimum feedback)	$2.5 * 10^{-5}$	$\nleq$	0	$\Delta\rho/^{o}Fm$
B) Doppler Temperature Coefficient (minimum feedback)	$-1.152 * 10^{-5}$	$\leq$	$-1.0 * 10^{-5}$	$\Delta\rho/^{o}Ff$
C) Reactivity Insertion Rate by Boron (maximum)	$1.34 * 10^{-5}$	$\leq$	$1.6 * 10^{-5}$	$\Delta\rho/sec$
D) Shutdown Margin (BOL)	2.74	$\geq$	1.0	$\%\Delta\rho$
E) $F\Delta^{N}_{H}$	1.52	$\leq$	1.55	

3.6 Evaluation of Startup of an Inactive Loop Accident

Table 3.6.1 presents a comparison of Cycle 5 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 5 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 5 reload core design, therefore, will not adversely effect 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	$-33.6 * 10^{-5}$	$\geq$	$-35.0 * 10^{-5}$	$\Delta\rho/^{o}Fm$
B) Doppler Temperature Coefficient	$-1.442 * 10^{-5}$	$\leq$	$-1.0 * 10^{-5}$	$\Delta\rho/^{o}Ff$
C) $F\Delta_H^N$	1.52	$\leq$	1.55	

3.7 Evaluation of Feedwater System Malfunction

A comparison of Cycle 5 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 5 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 5 reload core design, therefore, will not adversely effect 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 (HFP)	$-1.4 * 10^{-5}$	$\leq$	0	$\Delta\rho/^{o}Fm$
B) Doppler Temperature Coefficient	$-1.152 * 10^{-5}$	$\leq$	$-1.0 * 10^{-5}$	$\Delta\rho/^{o}Ff$
C) $F\Delta_{H}^{N}$	1.52	$\leq$	1.55	
D) Moderator Temperature Coefficient (maximum)	$-33.6 * 10^{-5}$	$\geq$	$-35.0 * 10^{-5}$	$\Delta\rho/^{o}Fm$

3.8 Evaluation of Excessive Load Increase Accident

Table 3.8.1 presents a comparison of Cycle 5 physics results to the current safety analysis values for the Excessive Load Increase Accident.

Since the pertinent parameters from the proposed Cycle 5 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 5 reload core design, therefore, will not adversely effect 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)	$-1.4 * 10^{-5}$	$\leq$	0	$\Delta\rho/^{o}Fm$
B) Moderator Temperature Coefficient (maximum)	$-28.3 * 10^{-5}$	$\geq$	$-35.0 * 10^{-5}$	$\Delta\rho/^{o}Fm$
C) Doppler Temperature Coefficient (BOL)	$-1.15 * 10^{-5}$	$\leq$	$-1.0 * 10^{-5}$	$\Delta\rho/^{o}Ff$
D) $F\Delta_{H}^{N}$ (BOL)	1.52	$\leq$	1.55	$\Delta\rho/^{o}Ff$

3.9 Evaluation of Loss of Load Accident

A comparison of Cycle 5 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 5 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 5 reload core design, therefore, will not adversely effect 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 Temperature Coefficient	$-1.4 * 10^{-5}$	$\leq$	0	$\Delta\rho/^{\circ}Fm$
B) Doppler Temperature Coefficient	$-1.15 * 10^{-5}$	$\leq$	$-1.0 * 10^{-5}$	$\Delta\rho/^{\circ}Fm$
C) $F\Delta_H^N$	1.52	$\leq$	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 reload safety evaluations.

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

Table 3.11.1 presents a comparison of Cycle 5 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 5 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 5 reload core design, therefore, will not adversely effect the safe operation of the Kewaunee Plant.

Table 3.11.1

LOSS OF REACTOR COOLANT FLOW DUE TO PUMP TRIP

Parameters	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temperature Coefficient	$-1.4 * 10^{-5}$	$\leq$	0	$\Delta\rho/^{\circ}Fm$
B) Doppler Temperature Coefficient	$-1.53 * 10^{-5}$	$\geq$	$-2.32 * 10^{-5}$	$\Delta\rho/^{\circ}Ff$
C) $F\Delta_{H}^{N}$	1.52	$\leq$	1.55	
D) Scram Worth vs. 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 5 physics parameters to the current safety analysis values for the Locked Rotor Accident.

Since the pertinent parameters from the proposed Cycle 5 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 5 reload core design, therefore, will not adversely effect 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 Temperature Coefficient	$-1.4 * 10^{-5}$	$\leq$	0	$\Delta\rho/^{o}F_{m}$
B) Doppler Temperature Coefficient	$-1.53 * 10^{-5}$	$\geq$	$-2.32 * 10^{-5}$	$\Delta\rho/^{o}F_{f}$
C) Delayed Neutron Fraction	.00529	$\geq$	.0051	
D) Percent Pins > Limiting $F\Delta^{N}_{H}$ (DNBR=1.3)	31.3	$\leq$	40.0	
E) Scram Worth vs. Time	See Section 2.3			

3.13 Evaluation of Main Steam Line Rupture Accident

The minimum Cycle 5 shutdown margin is compared to that assumed in the safety analysis in Table 3.13.1. Figure 3.13.1 shows the comparison of the Cycle 5 keff versus Temperature cooldown curve at 1000 psia to the current safety analysis curve.

Since the pertinent parameters from the proposed Cycle 5 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 5 reload core design, therefore, will not adversely effect the safe operation of the Kewaunee Plant.

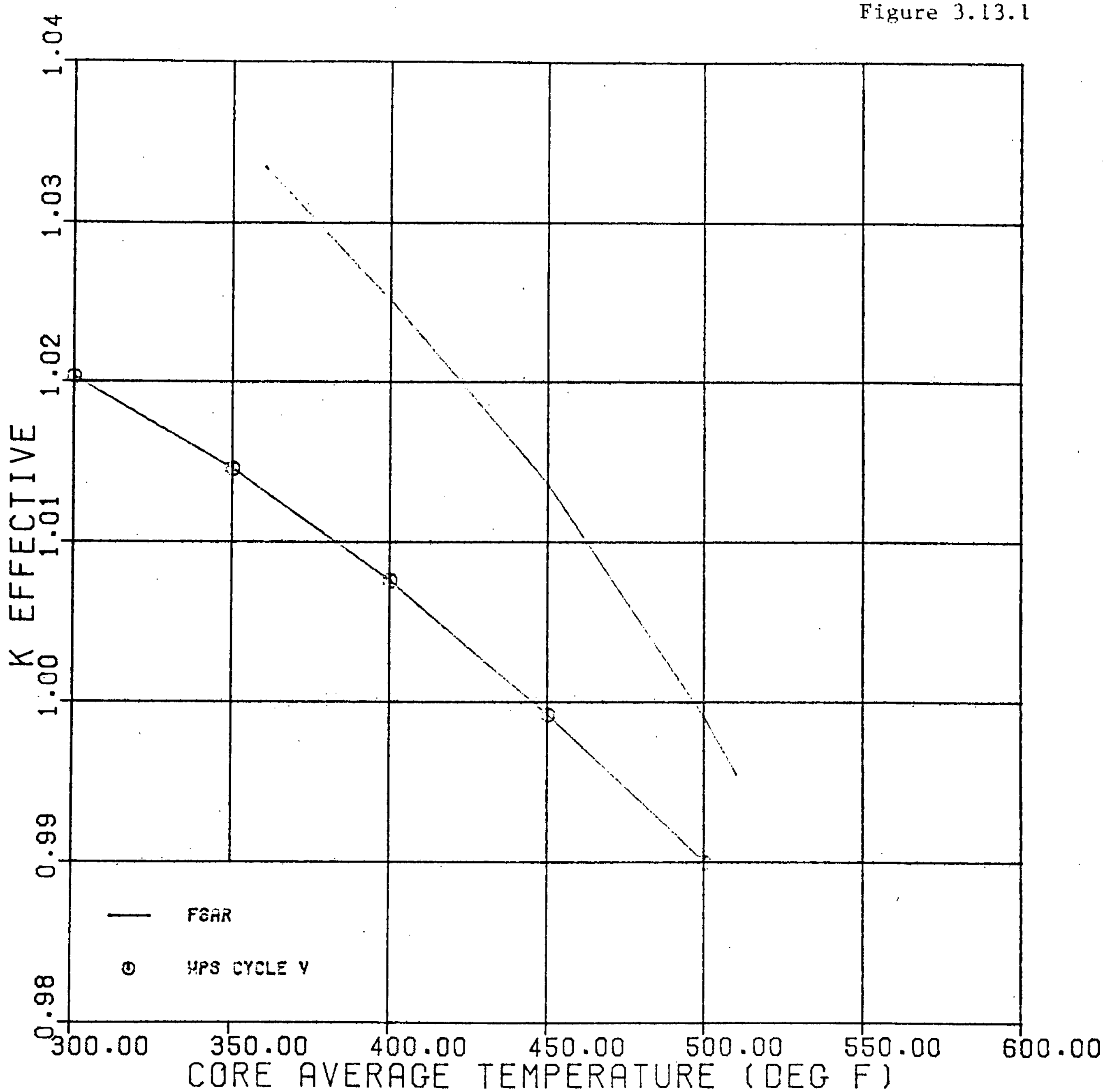
Table 3.13.1

MAIN STEAM LINE RUPTURE ACCIDENT

Parameter	Reload Safety Evaluation Value		Current Safety Analysis	Unit
A) Shutdown Margin	2.53	$\geq$	2	$\% \Delta\rho$

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 5 calculated physics parameters to the current safety analysis values for the Rod Ejection Accident at zero and full power, BOL and EOL core conditions.

Since the pertinent parameters from the proposed Cycle 5 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 5 reload core design, therefore, will not adversely effect 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 Temperature Coefficient	$-1.4 * 10^{-5}$	$\leq$	0	$\Delta\rho/^{o}Fm$
B) Delayed Neutron Fraction	.00599	$\geq$	.0055	
C) Ejected Rod Worth	.285	$\leq$	.30	$\%\Delta\rho$
D) Doppler Temperature Coefficient	$-1.15 + 10^{-5}$	$\leq$	$-1.0 * 10^{-5}$	$\Delta\rho/^{o}Ff$
E) Prompt Neutron Lifetime	24.9	$\geq$	20	μsec
F) F_Q^N	2.59	$\leq$	5.03	
G) Scram Worth vs. 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 Temperature Coefficient	$2.5 * 10^{-5}$	$\not\leq$	0	$\Delta\rho/^{o}Fm$
B)	Delayed Neutron Fraction	.00599	$\geq$	.0055	
C)	Ejected Rod Worth	.62	$\leq$	.91	$\%\Delta\rho$
D)	Doppler Temperature Coefficient	$-1.48 * 10^{-5}$	$\leq$	$-1.0 * 10^{-5}$	$\Delta\rho/^{o}Ff$
E)	Prompt Neutron Lifetime	24.9	$\geq$	20	μsec
F)	F_Q^N	7.41	$\leq$	11.2	
G)	Scram Worth vs. Time	See Section 2.3			

Table 3.14.3

ROD EJECTION ACCIDENTS HFP, EOL

Parameter		Reload Safety Evaluation Values		Current Safety Analysis	Units
A)	Moderator Temperature Coefficient	$-13.7 * 10^{-5}$	$\leq$	0	$\Delta\rho/^{o}Fm$
B)	Delayed Neutron Fraction	.00529	$\geq$	.0050	
C)	Ejected Rod Worth	.23	$\leq$	.42	%$\Delta\rho$
D)	Doppler Temperature Coefficient	$-1.24 * 10^{-5}$	$\leq$	$-1.0 * 10^{-5}$	$\Delta\rho/^{o}Ff$
E)	Prompt Neutron Lifetime	26.5	$\geq$	20	μsec
F)	F_Q^N	3.15	$\leq$	5.1	
G)	Scram Worth vs Time	See Section 2.3			

Table 3.14.4 HZP, EOL

ROD EJECTION ACCIDENTS

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temperature Coefficient	$-9.7 * 10^{-5}$	$\leq$	0	$\Delta\rho/^{o}Fm$
B) Delayed Neutron Fraction	.00529	$\geq$	.0050	
C) Ejected Rod Worth	.74	$\leq$	.92	$\%\Delta\rho$
D) Doppler Temperature Coefficient	$-2.07 * 10^{-5}$	$\leq$	$-1.0 * 10^{-5}$	$\Delta\rho/^{o}Ff$
E) Prompt Neutron Lifetime	26.5	$\geq$	20	μsec
F) F_Q^N	6.22	$\leq$	13.0	
G) Scram Worth vs. Time	See Section 2.3			

3.15 Evaluation of Fuel Handling Accident

Table 3.15.1 presents a comparison of the Cycle 5 F_Q^N calculated at EOC-1.5 GWD/MTU to the current safety analysis F_Q^N limit for the Fuel Handling Accident.

Since the pertinent parameter from the proposed Cycle 5 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 5 reload core design, therefore, will not adversely effect the safe operation of the Kewaunee Plant.

Table 3.15.1

FUEL HANDLING ACCIDENT

Parameter	Reload Safety Evaluation Values		Current Safety Analysis
A) F_Q^N	1.69	$\leq$	2.53

3.16 Evaluation of Loss of Coolant Accident

Table 3.16.1 presents the comparison of Cycle 5 physics calculation results to the current safety analysis values for the Loss of Coolant Accident.

Since the pertinent parameters from the proposed Cycle 5 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 5 reload core design, therefore, will not adversely effect 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 vs. Time	See Section 2.3	
B) F_Q	See Section 3.17	

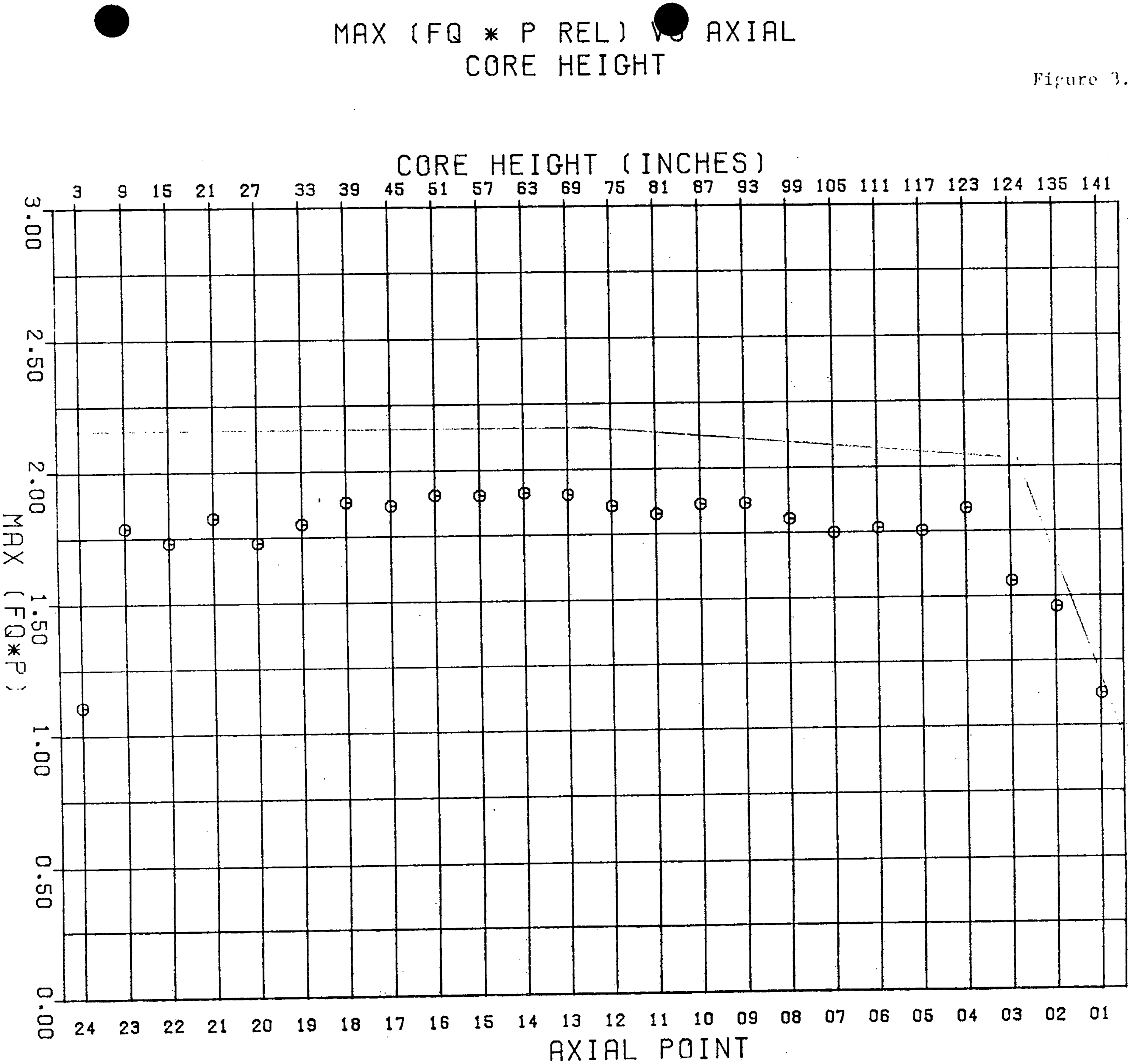
3.17 Constant Axial Offset Control (CAOC) Verification

The total peaking factor F_Q^T relates the maximum local power density to the core average power density. The F_Q^T 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.

Following these procedures, F_Q^T is determined by calculations performed for core conditions from BOL to EOL. The calculated core conditions include load follow maneuvers, xenon transients, and various control rod configurations. The limiting core conditions investigated include severely perturbed axial power distributions to assure that the resultant peaking factors are conservative.

Figure 3.17.1 presents the results of the F_Q^T comparison. These results demonstrate that $F_Q^T * P_{REL}$ is maintained below the F_Q^T limit (Section 2.2) during Core 5 operation, where P_{REL} is the core average power normalized to full power.

Figure 3.17.1



MAX (FQ * P REL) VS AXIAL
CORE HEIGHT
CORE HEIGHT (INCHES)
3 9 15 21 27 33 39 45 51 57 63 69 75 81 87 93 99 105 111 117 123 124 135 141
3.00
2.50
2.00
1.50
1.00
0.50
0.00
MAX (FQ*P)
24 23 22 21 20 19 18 17 16 15 14 13 12 11 10 09 08 07 06 05 04 03 02 01
AXIAL POINT

4.0 TECHNICAL SPECIFICATIONS

There will be no revisions or additions to the Kewaunee Technical Specifications due to the implementation of the Core 5 reload design.

5.0 REFERENCES

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

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

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

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

5. D. H. Risher, Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods", WCAP-7588, Revision 1-A, January, 1975

6. 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

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