# KEWAUNEE NUCLEAR POWER PLANT

# RELOAD SAFETY EVALUATION

# **KEWAUNEE CYCLE 14**

## **NOVEMBER 1987**

# WISCONSIN PUBLIC SERVICE CORPORATION

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#### **RELOAD SAFETY EVALUATION**

#### FOR

**KEWAUNEE CYCLE 14** 

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#### 1.0 Introduction

The Kewaunee Nuclear Power Plant is scheduled to shut down for the Cycle 13-14 refueling in March of 1988. Startup of Cycle 14 is forecast for April 1988.

This report presents an evaluation of the Cycle 14 reload and demonstrates that the reload will not adversely affect the safety of the plant. Those accidents which could potentially be affected by the reload core design are reviewed.

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

An evaluation, by accident, of the pertinent reactor parameters is performed by comparing the reload analysis results with the current bounding safety analysis values. The evaluations performed in this document employ the current Technical Specification (4) limiting safety system setpoints and operating limits.

It is concluded that the Cycle 14 design is more conservative than results of previously docketed accident analyses and implementation of this design will not introduce an unreviewed safety question since:

- the probability of occurrence or the consequences of an accident will not be increased,
- the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report will not be created and,

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

This conclusion is based on these assumptions; Cycle 13 is shut down within a +500 MWD/MTU, -200 MWD/MTU window of the nominal design EOC burnup, and there is adherence to plant operating limitations and Technical Specification (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 14 are presented in Figure 2.1.1.

Forty new Advanced Nuclear Fuel (ANF) assemblies enriched to 3.4 w/o U235 will reside with eighty-one partially depleted ANF assemblies. Table 2.1.1 displays the core breakdown by region, enrichment and number of previous duty cycles.

The Cycle 14 reload core will employ 24 burnable poison rod assemblies (BPRA'S) containing 160 fresh and 80 partially depleted burnable poison rods.

### Table 2.1.1

### Cycle 14 Fuel Characteristics

Region	Initia1 <u>W/O_U235</u>	Number of Previous Duty Cycles	Assemblies
10	3.2	3	1
10	3.2	3	4
13	3.4	2	4
13	3.4	3	8
14	3.4	2	32
15	3.4	1	32
16	3.4	0	40 (FEED)

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FIGURE 2.1.1





#### CYCLE 14 LØADING PATTERN

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2.2 Design Objectives and Operating Limits

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

Core Average Moderator Temperature (HFP) 561° F

Cycle 14 core design is based on the followng design objectives and operating limits.

A. Nuclear peaking factor limits are as follows:

(i) FQ(Z) limits

FQ(Z)  $\leq$  (2.23/P) \* K(Z) for P > 0.5 FQ(Z)  $\leq$  4.46 \* K(Z) for P  $\leq$  0.5

(ii) F<sub>∆</sub>H limits

 $F \land HN < 1.55(1 + 0.2(1-P))$ 

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 Z is the core height

- 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 11,200 MWD/MTU.
- E. The power dependent rod insertion limits (PDIL) are presented in Figure 2.2.2. These limits are those currently specified in Reference 4.
- 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 currently specified in Reference 4.
- G. At refueling conditions a boron concentration of 2100 ppm will be sufficient to maintain the reactor subcritical by approximately 10%  $\Delta k/k$ with all rods inserted and will maintain the core subcritical with all rods out.
- H. Fuel duty during this fuel cycle will assure peak fuel rod burnups less than the maximum burnup recommended by the fuel vendor.

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#### FIGURE 2.2.2



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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 14 minimum shutdown margin is 2.40% at end of cycle hot full power conditions. Figure 2.3.1 compares the Cycle 14 minimum scram insertion curve to the current bounding safety analysis curve.

It is concluded that the minimum trip reactivity insertion rate for Cycle 14 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.

FIGURE 2.3.1	F	١G	JRE	2.	3.	1
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Scram Worth vs. Time



2.4 Shutdown Window

An evaluation of the full power equilibrium peaking factor variation at BOC 14 versus EOC 13 burnup is presented in Table 2.4.1. The values shown have conservatisms applied in accordance with references 1 and 9.

It is concluded that if refueling shutdown of Cycle 13 occurs within the burnup window the Cycle 14 peaking factors will not be significantly affected and will not exceed their limiting values.

Ti	ab	1	е	2	•	4	•	1	

	For H		FQ		
	Cycle 14	<u>Limit</u>	Cycle 14	<u>Limit</u>	
EOC 13 - 200 MWD/MTU	1.51	1.55	2.12	2.23	
EOC 13 Nominal	1.50	1.55	2.13	2.23	
EOC 13 + 500 MWD/MTU	1.50	1.55	2.16	2.23	

Peaking Factor Versus Cycle 13 Shutdown Burnup

#### 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 14 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 Clad Hoop Stress Addendum	12/14/79 1/8/80	10 11
Exxon	10/01/84 (XN-NF-84-31. Rev.1	) 12

### Table 3.0.2

Safety	Analyses	Bounding	Values
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Parameter	Lower Bound	Upper Bound	<u>Units</u>
Moderator Temp. Coefficient	-40.0	0.0	pcm/°F
Doppler Coefficient	-2.32	-1.0	pcm/°F
Differential Boron Worth	-11.2	-7.7	pcm/ppm
Delayed Neutron Fraction	.00485	.0071	
Prompt Neutron Lifetime	15	N/A	μ sec
Shutdown Margin	1.0 (BOC) 2.0 (EOC)	N/A N/A	<b>%</b> Δρ
Differential Rod Worth of 2 Banks Moving	N/A	82	pcm/sec
Ejected Rod Cases			
HFP, BOL β eff Rod Worth FQ	.0055 N/A N/A	N/A .30 5.03	<b>%</b> Δρ
HFP, EOL ßeff Rod Worth FQ	.0050 N/A N/A	N/A .42 5.1	<b>%</b> Δρ
HZP, BOL βeff Rod Worth FQ	.0055 N/A N/A	N/A .91 11.2	<b>%</b> Δρ
HZP,EOL βeff Rod Worth FQ	.D050 N/A N/A	N/A .92 13.0	<b>%</b> Δρ

#### 3.1 Evaluation of Uncontrolled Rod Withdrawal from Subcritical

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

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

Table 3.1.1 presents a comparison of Cycle 14 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 14 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 14 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

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### Uncontrolled Rod Withdrawal From Subcritical

Pa	rameter	Reload Safety Evaluation Values		Current <u>Safety Analysis</u>	<u>Units</u>
A)	Moderator Temp. Coefficient	0.37	<u>۲</u>	10.0	pcm/°Fm
B)	Doppler Temp. Coefficient	-2.2	<u>&lt;</u>	-1.0	pcm/°Ff
C)	Differential Wort of Two Moving Ban	h ks.048	<u>&lt;</u>	.116	\$/sec
D)	Scram Worth vs. Time	See Section 2	.3		
E)	Delayed Neutron Fraction	.00544	>	00485	

3.2 Evaluation of Uncontrolled Rod Withdrawal at Power

An uncontrolled control rod bank withdrawal at power results in a gradual increase in core power followed by an increase in core heat flux. The resulting mismatch between core power and steam generator heat load results in an increase in reactor coolant temperature and pressure.

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

Table 3.2.1 presents a comparison of the Cycle 14 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, BOC, 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 14 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 14 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

#### Table 3.2.1

### Uncontrolled Rod Withdrawal at Power

Par	rameter	Reload Safety Evaluation Value	<u>s</u>	Current Safety Analys	is <u>Units</u>
A)	Moderator Temp. Coefficient	0.37*	٢	0.0	pcm/°Fm
B)	Doppler Temp. Coefficient	-1.1	<u>&lt;</u>	-1.0	pcm/°Ff
C)	Differential Ro Worth Of Two Moving Banks	d .048	<u>&lt;</u>	.116	\$/sec
D)	FAHN	1.54	٢	1.55	
E)	Scram Worth vs. Time	See Sectio	n 2.	3	· · ·
F)	Delayed Neutron Fraction	0.00544	· <u>&gt;</u>	0.00485	
*Mc St	oderator Tempera artup Testing.	ture Coefficient	wi]	1 be verified	negative at

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#### 3.3 Evaluation of Control Rod Misalignment

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

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

Table 3.3.1 presents a comparison of the Cycle 14  $F_{\Delta}$  HN versus the current safety analysis  $F_{\Delta}$  H limit for the Misaligned Rod Accident.

Since the pertinent parameter from the proposed Cycle 14 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 14 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∆HN	1.85	<u>&lt;</u>	1.92



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#### 3.4 Evaluation of Dropped Rod

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

A comparison of the Cycle 14 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 14 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 14 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

### Dropped Rod Accident

Parameter	Reload Safety Evaluation Value		Current Safety Analysis
A) FAHN	1.65	<u>&lt;</u>	1.92



#### 3.5 Evaluation of Uncontrolled Boron Dilution

The malfunction of the Chemical and Volume Control System (CVCS) is assumed to deliver unborated water to the reactor coolant system.

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

Table 3.5.1 presents a comparison of Cycle 14 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, BOC, 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 14 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 14 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

#### Table 3.5.1

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### Uncontrolled Boron Dilution Accident

	Pa	rameter	Re' Evalu	load Sa uation	afety Values		Curr Safety	ent Analysis	Units
i)	Re	fueling Condition	<u>15</u>						
	A)	Shutdown Margin	1	10.0		2	10.	0	<b>%</b> ∆ρ
ii)	<u>At</u>	-Power Conditions	<u>5</u>						
	A)	Moderator Temp. Coefficient		0.37*		<u>&lt;</u>	0.	0	pcm/°Fm
	B)	Doppler Temp. Coefficient	-	-1.1		<u>&lt;</u>	-1.	0	pcm/°Ff
	C)	Reactivity Inser Rate by Boron	tion	.0021		<u>&lt;</u>	•	0023	\$/sec
	D)	Shutdown Margin		2.40		2	1.	00	<b>%</b> ∆ρ
	E)	F∆HN	•	1.54		<u>&lt;</u>	1.	55	
	F)	Delayed Neutron Fraction		0.0054	4	2	0.	00485	

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

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 14 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 14 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 14 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

Pa	rameter	Reload Safety Evaluation Values		Current Safety Analysis	Unit
A)	Moderator Temp. Coeffient	-34.5	<u>&gt;</u>	-40.0	pcm/°Fm
B)	Doppler Coefficient	-1.5	<u>&lt;</u>	-1.0	pcm/°Ff
C)	F∆HN	1.54	<u>&lt;</u>	1.55	

#### 3.7 Evaluation of Feedwater System Malfunction

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

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

A comparison of Cycle 14 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 14 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 14 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 <u>Safety Analysis</u>	<u>Units</u>
i)	Beginning of Cycle				
	A) Moderator Temp. Coefficient	-5.2	<u>&lt;</u>	0.0	pcm/°Fm
	B) Doppler Temp. Coefficient	-1.29	<u>&gt;</u>	-2.32	pcm/°Ff
ii)	End of Cycle				•
	A) Moderator Temp. Coefficient	-31.0	<u>&gt;</u>	-40.0	pcm/°Fm
	B) Doppler Temp. Coefficient	-1.2	<u>۲</u>	-1.0	pcm/°Ff
	C) FAHN	1.54	<u>&lt;</u>	1.55	

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#### 3.8 Evaluation of Excessive Load Increase

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

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

Table 3.8.1 presents a comparison of Cycle 14 physics results to the current safety analysis values for the Excessive Load Increase Accident.

Since the pertinent parameters from the proposed Cycle 14 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 14 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

### Table 3.8.1

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#### Excessive Load Increase Accident

	Parameter	Reload Safety Evaluation Values		Current Safety Analysis	<u>Units</u>
i)	Beginning of Cycle				·
	A) Moderator Temp. Coefficient	-5.2	<u>&lt;</u>	0.0	pcm/°Fm
	B) Doppler Temp. Coefficient	-1.29	2	-2.32	pcm/°Ff
ii)	End of Cycle				
	A) Moderator Temp. Coefficient	-31.0	2	-40.0	pcm/°Fm
	B) Doppler Temp. Coefficient	-1.2	<u>&lt;</u>	-1.0	pcm/°Ff
	C) F∆HN	1.54	<	1.55	

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#### 3.9 Evaluation of Loss of Load

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

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

A comparison of Cycle 14 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 14 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 14 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

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#### Table 3.9.1

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### Loss of Load Accident

	Parameter	Reload Safety Evaluation Values		Current Safety Analysis	<u>Units</u>
i)	Beginning of Cycle				
	A) Moderator Temp. Coefficient	-5.2	<u>&lt;</u>	0.0	pcm/°Fm
	B) Doppler Temp. Coefficient	-1.1	<u>&lt;</u>	-1.0	pcm/°Ff
ii)	End of Cycle			· · · .	
	A) Moderator Temp. Coefficient	-31.0	<u>&gt;</u>	-40.0	pcm/°Fm
	B) Doppler Temp. Coefficient	-1.41	2	-2.32	pcm/°Ff
	C) F∆HN	1.54	<u>&lt;</u>	1.55	
	D) Scram Worth Versus Time	See Section 2	2.3	3	

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#### 3.10 Evaluation of Loss of Normal Feedwater

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

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

The Loss of Feedwater Transient is not sensitive to core physics parameters and therefore no comparisons will be made for the Reload Safety Evaluation. 3.11 Evaluation of Loss of Reactor Coolant Flow Due to Pump Trip

The simultaneous loss of power to 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 14 calculated physics parameters to the current safety analysis values for the Loss of Reactor Coolant Flow Due to Pump Trip Accident.

Since the pertinent parameters from the proposed Cycle 14 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 14 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

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### Loss of Reactor Coolant Flow Due to Pump Trip

Pa	rameter	Reload Safety Evaluation Values	5	Curi Safety	rent Analysis	<u>Units</u>
A)	Moderator Temp. Coefficient	-5.2	<u>&lt;</u>	0.	.0	pcm/°Fm
B)	Doppler Temp. Coefficient	-1.41	<u>&gt;</u>	-2.	.32	pcm/°Ff
C)	F∆HN	1.54	<u>&lt;</u>	1.	. 55	
D)	Scram Worth Versus Time	See Section	2.3	3		

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

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

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

Table 3.12.1 presents a comparison of Cycle 14 physics parameters to the current safety analysis values for the Locked Rotor Accident.

Since the pertinent parameters from the proposed Cycle 14 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 14 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

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# Loss of Reactor Coolant Flow Due to Locked Rotor

Pa	rameter	Reload Safety Evaluation Values		Current <u>Safety Analysis</u>	<u>Units</u>
A)	Moderator Temp. Coefficient	-5.2	<u>۲</u>	0.0	pcm/°Fm
B)	Doppler Temp. Coefficient	-1.41	<u>&gt;</u>	-2.32	pcm/°Ff
C)	Delayed Neutron Fraction	0.00544	<u>&gt;</u>	0.00485	
D)	Percent Pins > Limiting FAHN (DNBR=1.3)	18.4	<u>&lt;</u>	40.0	%
E)	Scram Worth Versus Time	See Section 2	.3		
F)	FQ	2.13	<u>&lt;</u>	2.23	

3.13 Evaluation of Main Steam Line Break

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

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

Table 3.13.1 presents a comparison of Cycle 14 calculated physics parameters to the current safety analysis values for the main steam line break accident. Figure 3.13.1 compares core Keff during the cooldown to the current bounding safety analysis curve.

Since the pertinent parameters from the proposed Cycle 14 reload core are conservatively bounded by those used in the current safety analysis, a main steam line break accident will be less severe than the transient in the current analysis. The implementation of the Cycle 14 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Pa	rameter	Reload Safety Evaluation Value		Current Safety Analysis	<u>Unit</u>
A)	Shutdown Margin	2.40	2	2.00	<b>%</b> Δρ
B)	F∆H	3.9	<u>&lt;</u>	8.8	
C)	Doppler Temp. Coefficient	-1.1	<u>&lt;</u>	-1.0	pcm/°Ff
D)	Boron Worth Coefficient	-7.8	<u>&gt;</u>	-7.7	pcm/ppm

#### Main Steam Line Break Accident





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#### 3.14 Evaluation of Rod Ejection Accidents

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

Tables 3.14.1 thru 3.14.4 present the comparison of Cycle 14 calculated physics parameters to the current safety analysis values for the Rod Ejection Accident at zero and full power, BOC and EOC core conditions.

The application of the reliability factor to the moderator coefficient calculated at HZP, BOC, 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 14 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 14 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

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### Rod Ejection Accidents

### HFP, BOC

<u>Pa</u>	rameter	Reload Safety Evaluation Values		Current Safety Analysis	<u>Units</u>
A)	Moderator Temp. Coefficient	-5.2	<u>&lt;</u>	0.0	pcm/°Fm
B)	Delayed Neutron Fraction	0.00619	2	0.00550	
C)	Ejected Rod Worth	0.08	<u>&lt;</u>	0.30	<b>%</b> Δρ
D)	Doppler Temp. Coefficient	-1.1	<u>&lt;</u>	-1.0	pcm/°Ff
E)	Prompt Neutron Lifetime	29.0	2	15.0	μSeC
F)	FQN	2.61	<u>&lt;</u>	5.03	
G)	Scram Worth	See Section 2	2		



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### Rod Ejection Accidents

### HZP, BOC

Pai	rameter	Reload Safety Evaluation Values		Current Safety Analysis	<u>Units</u>
A)	Moderator Temp. Coefficient	0.37*	<u>&lt;</u>	0.0	pcm/°Fm
B)	Delayed Neutron Fraction	0.00619	2	0.00550	
C)	Ejected Rod Worth	0.47	<u>&lt;</u>	0.91	<b>%</b> Δρ.
D)	Doppler Temp. Coefficient	-1.7	٢	-1.0	pcm/°Ff
E)	Prompt Neutron Lifetime	29.0	2	15.0	μsec
F)	FQN	5.61	<u>&lt;</u>	11.2	· ·
G)	Scram Worth Versus Time	See Section 2.	3		

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

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### Rod Ejection Accidents

### HFP, EOC

Par	rameter	Reload Safety Evaluation Values		Current Safety Analysis	<u>Units</u>
A)	Moderator Temp. Coefficient	-16.9	<u>&lt;</u>	0.0	pcm/°Fm
B)	Delayed Neutron Fraction	0.00544	2	0.00500	
C)	Ejected Rod Worth	0.12	<u>&lt;</u>	0.42	% Δρ
D)	Doppler Temp. Coefficient	-1.16	٢	-1.0	pcm/°Ff
E)	Prompt Neutron Lifetime	32.0	2	15.0	μSec
F)	FQN	3.05	<u>&lt;</u>	5.1	
G)	Scram Worth Versus Time	See Section 2.	3		

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Rod Ejection Accidents

### HZP, EOC

Pa	rameter	Reload Safety Evaluation Values		Current Safety Analysis	<u>Units</u>
A)	Moderator Temp. Coefficient	-4.9	<u>&lt;</u>	0.0	pcm/°Fm
B)	Delayed Neutron Fraction	0.00544	<u>&gt;</u>	0.00500	
C)	Ejected Rod Worth	0.64	<u>&lt;</u>	0.92	%Δp
D)	Doppler Temp. Coefficient	-2.23	<u>&lt;</u>	-1.0	pcm/°Ff
E)	Prompt Neutron Lifetime	32.0	<u>&gt;</u>	15.0	μsec
F)	FQN	8.45	<u>&lt;</u>	13.0	
E)	Scram Worth Versus Time	See Section 2.3	3		

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#### 3.15 Evaluation of Fuel Handling Accident

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

The maximum FQ expected during this period is evaluated within the restrictions of the power distribution control procedures.

Table 3.15.1 presents a comparison of the Cycle 14 FQN, calculated at end of Cycle 14 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 14 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 14 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

### Fuel Handling Accident

Parameter	Reload Safety Evaluation Values	Current <u>Safety Analysis</u>		
A) FQN	1.96	<u>&lt;</u>	2.53	

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#### 3.16 Evaluation of Loss of Coolant Accident

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The Loss of Coolant Accident is defined as the rupture of the reactor coolant system piping or any line connected to the system, up to and including a double-ended guillotine rupture of the largest pipe.

The principal parameters which affect the results of LOCA analysis are the fuel stored energy, fuel rod internal pressures, and decay heat. These parameters are affected by the reload design dependent parameters shown in Table 3.16.1.

The initial conditions for the LOCA analyses are assured through limits on fuel design, fuel rod burnup, and power distribution control strategies.

Table 3.16.1 presents the comparison of Cycle 14 physics calculation results to the current safety analysis values for the Loss of Coolant Accident.

Since the pertinent parameters from the proposed Cycle 14 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 14 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

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### Loss of Coolant Accident

Parameter	Reload Safety Evaluation Values	Current <u>Safety Analysis</u>		
A) Scram Worth Versus Time	See Section 2.3			
B) FQ	See Section 3.17			
C) F∆H	1.54	<	1.55	

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#### 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 (9) described in Section 2.2 of this report.

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 14 operation will not preclude full power operation under the power distribution control specifications currently applied (4). FIGURE 3.17.1

#### MAX (FO \* P REL ) VS AXIAL CORE HEIGHT CYCLE 14 S3D 87280.1029



### 4.0 TECHNICAL SPECIFICATIONS

No Technical Specification changes are required as a result of this reload.

#### 5.0 STATISTICS UPDATE

In an effort to provide continuing assurance of the model applicability, Cycle 12 measurements and calculations were added to the statistics data base prior to model applications to the Cycle 14 Reload Analysis. The reliability and bias factors applicable to Cycle 14 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∆H	3.5%	0
Rod Worth	10.0%	0
Moderator Temperature Coefficient	4.33 PCM/°F	1.17 PCM/°F
Doppler Coefficient	10.0%	0
Boron Worth	5.0%	0
Delayed Neutron Parameters	3.0%	0

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### Table 5.0.2

### FQN Reliability Factors

Core Level	oNode	<u>RF (%)</u>
1 (Bottom)	.115	19.77
2	.049	8.63
3	.030	5.75
4	.029	5.60
5	.030	5.75
6	.029	5.57
7	.026	5.22
8	.028	5.38
9	.025	5.05
10	.028	5.42
11	.024	4.93
12	.025	4.98
13	.021	4.54
14	.021	4.54
15	.021	4.48
16	.022	4.68
17	.022	4.66
18	.021	4.52
19	.028	5.42
20	.029	5.65
21	.045	8.08
22	.043	7.70
23	.078	13.45
24 (Top)	.089	15.23

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#### 6.0 REFERENCES

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