KEWAUNEE NUCLEAR POWER PLANT

RELOAD SAFETY EVALUATION
CYCLE 21
MAY 1995

WISCONSIN PUBLIC SERVICE CORPORATION
WISCONSIN POWER & LIGHT COMPANY
MADISON GAS & ELECTRIC COMPANY

RELOAD SAFETY EVALUATION

FOR

KEWAUNEE CYCLE 21

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1.0 SUMMARY

The Kewaunee Nuclear Power Plant is scheduled to shut down for the Cycle 20-21 refueling in April 1995. Startup of Cycle 21 is forecast for May 1995.

This report presents an evaluation of the Cycle 21 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 References 1 and 15. Accident Evaluation methodologies applied in this report are detailed in Reference 2. These reports have been previously reviewed and approved by the NRC as shown in References 3 and 4. The current physics model reliability factors are discussed in Section 5 of this report.

An evaluation, by accident, of the pertinent reactor parameters is performed by comparing the reload analysis results with the current bounding safety analysis values. The evaluations performed in this document employ the current Technical Specification (Reference 5) limiting safety system settings and operating limits.

It is concluded that the Cycle 21 design is more conservative than results of previously docketed accident analyses and implementation of this design will not introduce an unreviewed safety question since:

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

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

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

This conclusion is based on the assumption that there is adherence to plant operating limitations and Technical Specifications (Reference 5).

2.0 CORE DESIGN

2.1 Core Description

The reactor core consists of 121 fuel assemblies of 14 x 14 design. The core loading pattern, assembly identification, control rod bank identification, instrument thimble I.D., thermocouple I.D., and burnable poison rod configurations for Cycle 21 are presented in Figure 2.1.1.

Twenty (20) new Siemens Power Corporation (SPC) standard assemblies enriched to 3.8 w/o U235, twenty (20) SPC standard assemblies enriched to 4.1 w/o U235, and eight (8) SPC "heavy" lead test assemblies enriched to 4.1 w/o U235 will reside with 69 partially depleted SPC standard assemblies and four (4) Westinghouse OFA assemblies. The 8 SPC "heavy" lead test assemblies contain 405 KgU (per assembly) versus 378 KgU in the SPC standard fuel design. Descriptions of the fuel designs are provided in Ref. 6 for SPC standard, Ref. 16,17 for Westinghouse OFA, and Ref. 20 for SPC heavy.

Table 2.1.1 displays the core breakdown by region, enrichment, and number of previous duty cycles.

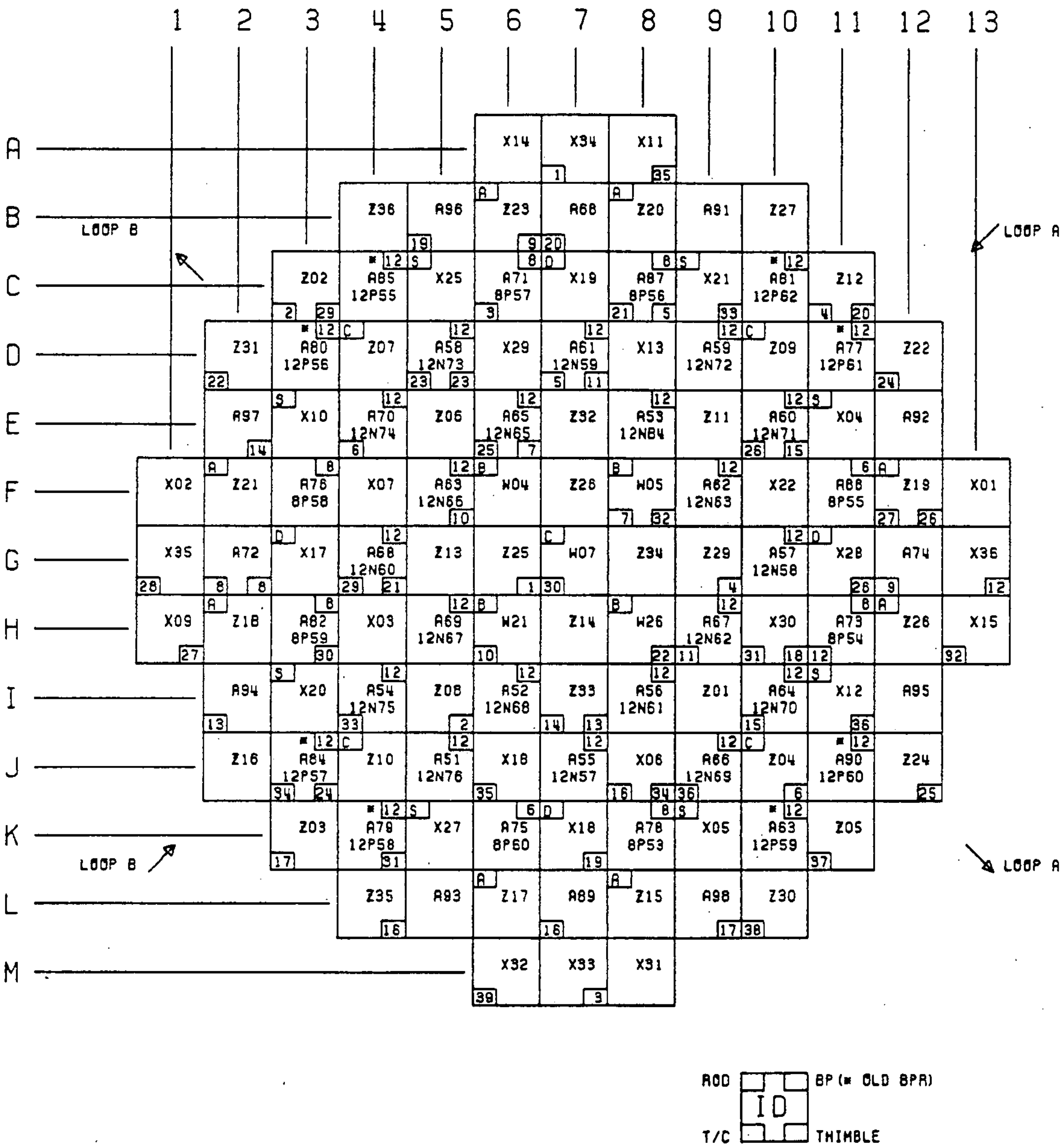
The Cycle 21 reload core will employ 36 burnable poison rod assemblies (BPRAs) containing 304 fresh and 96 partially depleted burnable poison rods. Fuel assemblies with two or three previous duty cycles are loaded on the core periphery flat region to lower power in that region and reduce reactor vessel fluence (Reference 14) in the critical reactor vessel locations. The Cycle 21 fuel loading pattern is capable of achieving a burnup of 15,930 MWD/MTU operating at full power, based on an end of Cycle 20 burnup of 11,453 MWD/MTU.

Table 2.1.1

Cycle 21 Fuel Characteristics

Region	Region Identifier	Initial W/O U235	Number of Previous Duty Cycles	Assemblies
20	W	3.4	2	5
21	X	3.4	2	28
21	X	3.1 (OFA)	2	4
22	Z	3.5	1	12
22	Z	3.7	1	24
23	A	3.8	0	20
23	A	4.1	0	20
23	A	4.1 (Heavy)	0	8

Figure 2.1.1



CYCLE TWENTY-ONE

2.2 Operating Conditions and Limits

Cycle 21 core design is based on the following operating conditions and limits.

2.2.1 Operating Conditions

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

2.2.2 Operating Limits

A. Nuclear peaking factor limits are as follows:

(i) FQ(Z) limits

a) For SPC standard fuel:

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

K(Z) is the function given in Figure 2.2.1

Z is the core height

b) For Westinghouse OFA fuel, the FQ(Z) limit is the SPC standard fuel limit less 10% (Reference 13).

c) For SPC heavy fuel, the FQ(Z) limit is the SPC standard fuel limit less 5.3% (Ref. 14).

(ii) $F\Delta H$ limits

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

Where P is the fraction of full power at which the core is operating.

(iii) The Westinghouse OFA and the SPC heavy fuel will not be limiting with respect to power distribution and LOCA analysis assumptions (Ref. 12).

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

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

1.0% at Beginning of Cycle (BOC)

2.0% at End of Cycle (EOC)

D. The power dependent rod insertion limits (PDIL) are presented in Figure 2.2.2. These limits are those currently specified in Reference 5.

E. The indicated axial flux difference shall be maintained within a $\pm$ 5% band about the target axial flux difference above 90 percent power. Figure 2.2.3 shows the axial flux difference limits as a function of core power. These limits are currently specified in Reference 5, which also provides limits on temporary operation allowed within the 3.10.b.11.a. line envelope at power levels between 50 percent and 90 percent.

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

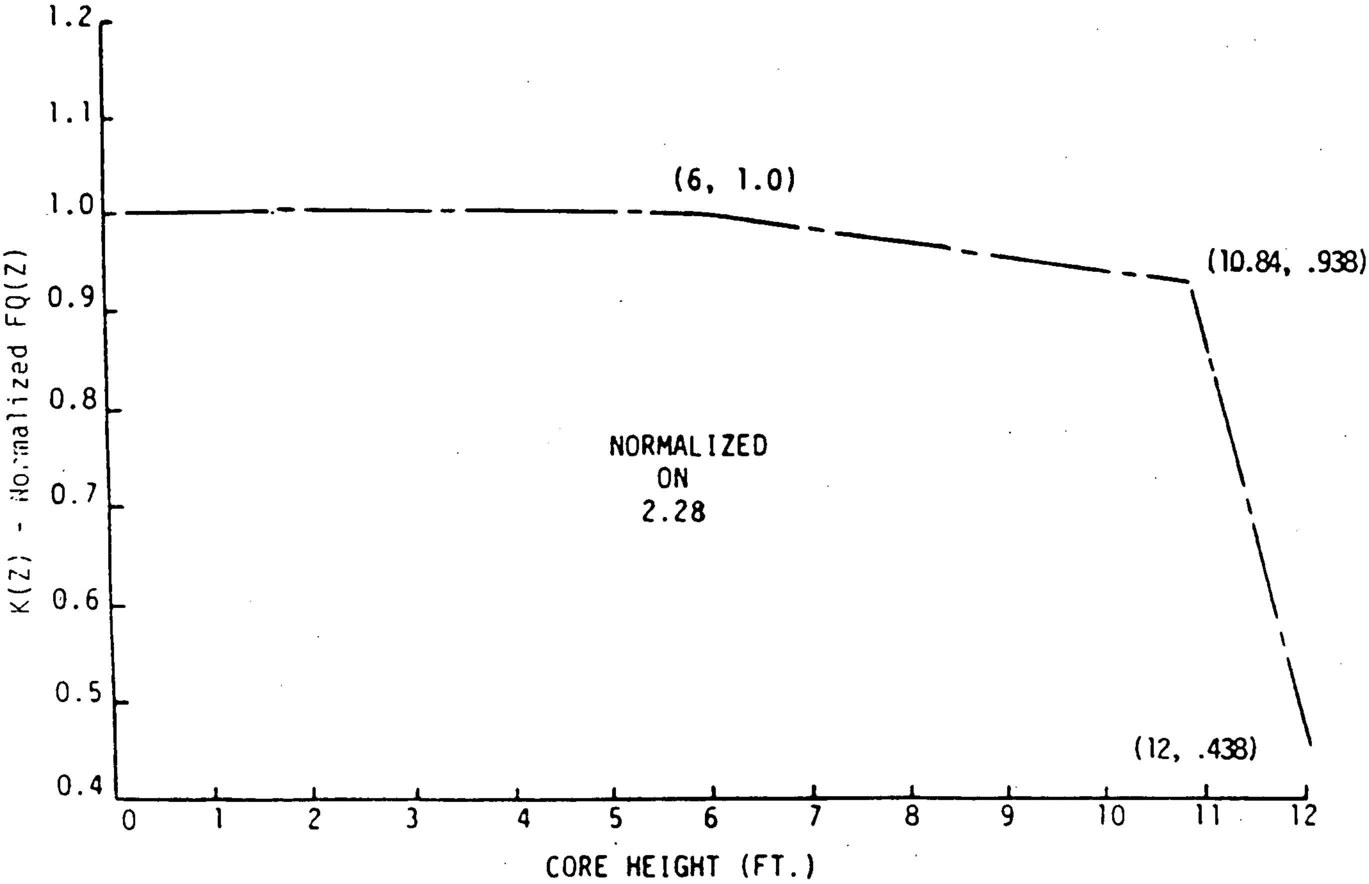
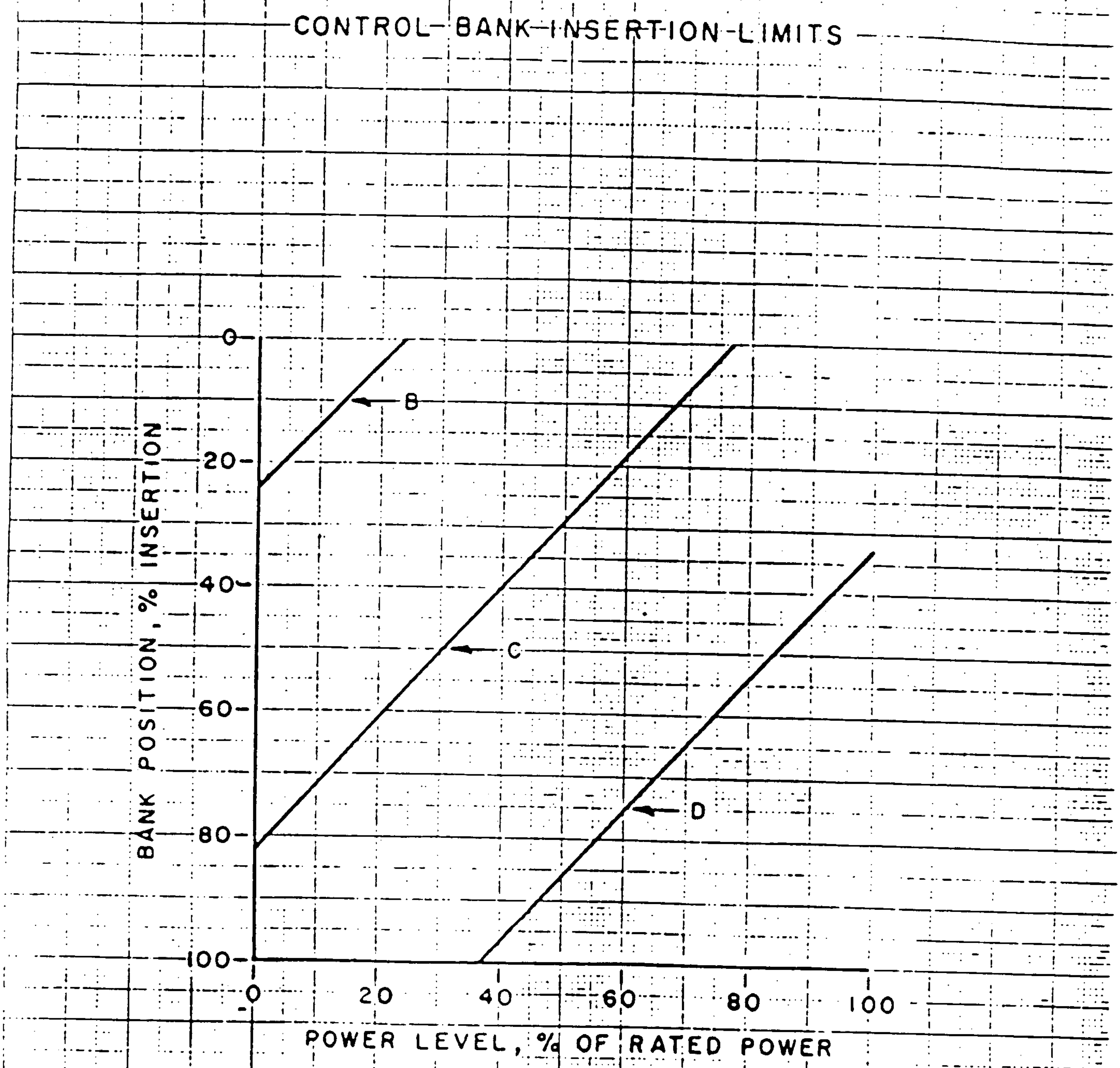


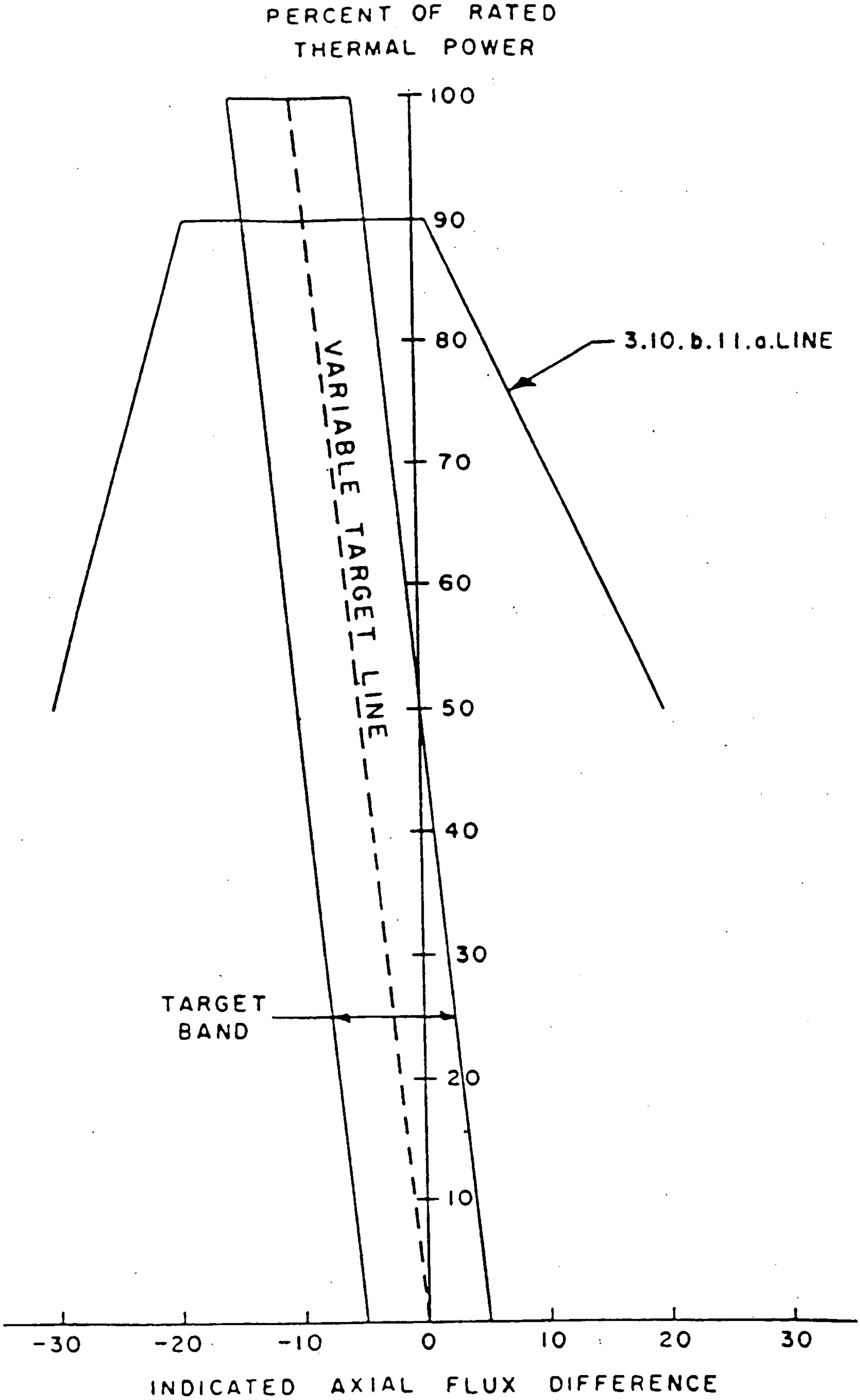
FIGURE 2.2.1 - HOT CHANNEL FACTOR
NORMALIZED OPERATING ENVELOPE

FIGURE 2.2.2



CONTROL BANK INSERTION LIMITS
BANK POSITION, % INSERTION
0
20
40
60
80
100
B
C
D
0
20
40
60
80
100
POWER LEVEL, % OF RATED POWER

FIGURE 2.2.3



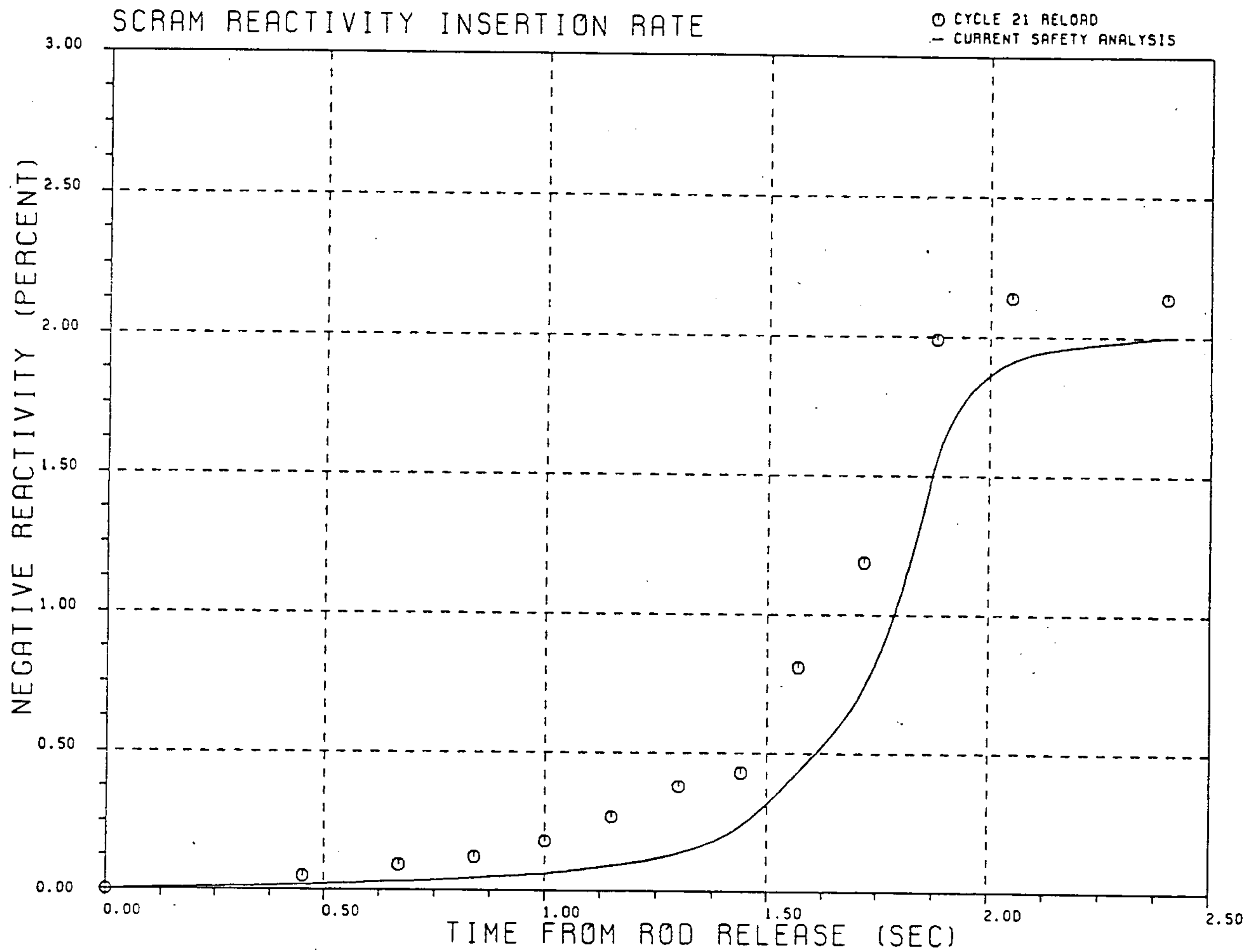
Target Band on Indicated Flux Difference
As a Function of Operating Power Level (Typical)

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 Cycle 21 minimum shutdown margin is 2.01 percent at end of cycle hot full power conditions. Figure 2.3.1 compares the Cycle 21 minimum scram insertion curve to the current bounding safety analysis curve.

It is concluded that the minimum trip reactivity insertion rate for Cycle 21 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



SCRAM REACTIVITY INSERTION RATE
CYCLE 21 RELOAD
CURRENT SAFETY ANALYSIS
NEGATIVE REACTIVITY (PERCENT)
TIME FROM ROD RELEASE (SEC)
3.00
2.50
2.00
1.50
1.00
0.50
0.00
0.00
0.50
1.00
1.50
2.00
2.50

2.4 Shutdown Window

An evaluation of the maximum full power equilibrium peaking factors is presented in Table 2.4.1. The values shown have conservatisms applied in accordance with References 1 and 7.

It is concluded that the Cycle 21 peaking factors are within their limiting values.

Table 2.4.1

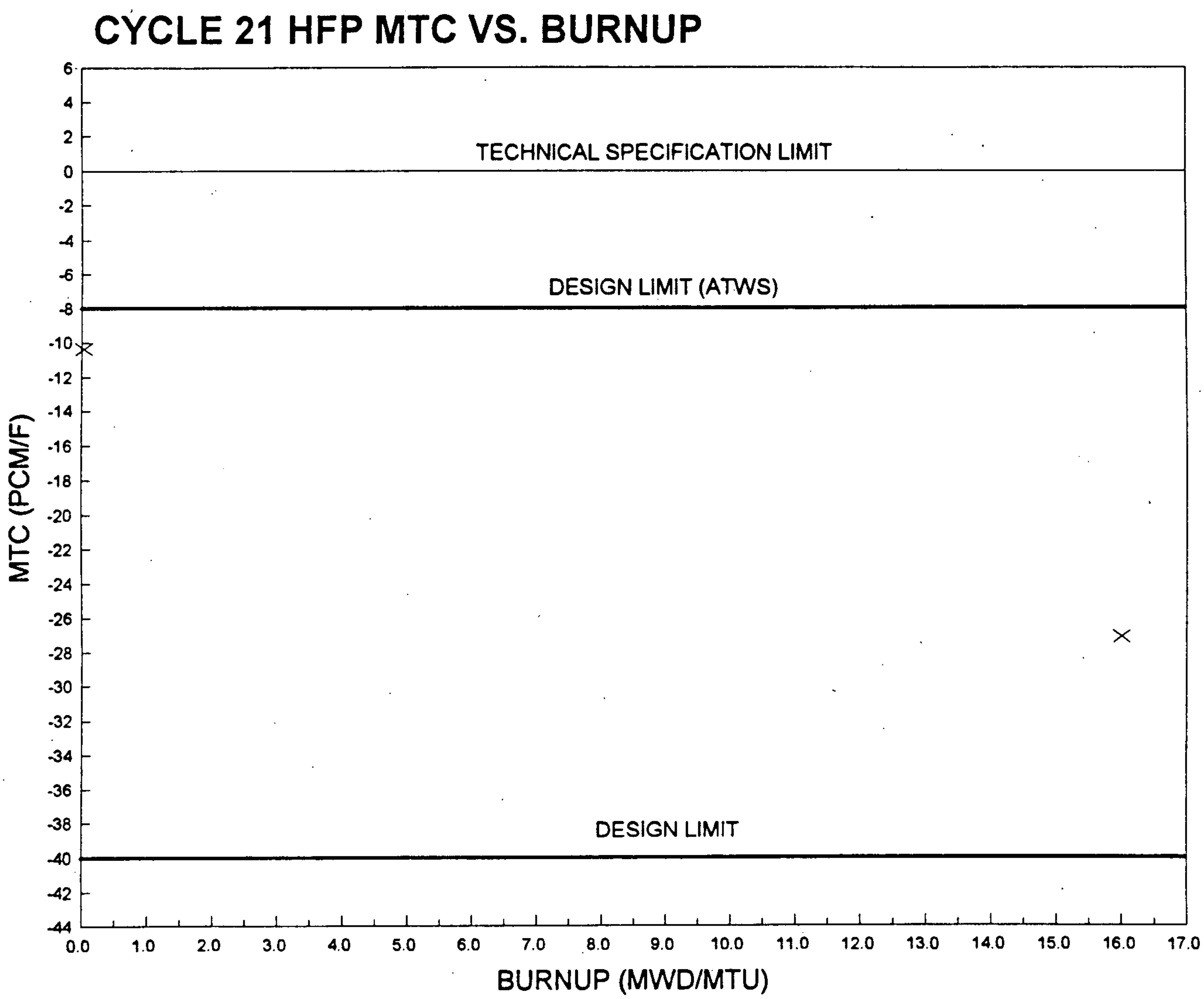
Peaking Factor Evaluation

	FΔH		**FQ**	
	Cycle 21	**Limit**	**Cycle 21**	**Limit**
EOC 20 Nominal	1.55	1.55	2.26	2.28

2.5 Moderator Temperature Coefficient

An evaluation of the Cycle 21 hot full power moderator temperature coefficient is presented in Figure 2.5.1. The calculated Cycle 21 values are compared to the MTC limit of -8.0 pcm/°F. Cycle 21 MTC must be less than the upper bound limit for 95% of the scheduled time at HFP due to anticipated transient without scram (ATWS) concerns. It is concluded that the Cycle 21 MTC is conservative with respect to the bounding value. Therefore, the Cycle 21 core will not adversely affect the results of the ATWS safety analysis.

Figure 2.5.1


CYCLE 21 HFP MTC VS. BURNUP
TECHNICAL SPECIFICATION LIMIT
DESIGN LIMIT (ATWS)
DESIGN LIMIT
MTC (PCM/F)
BURNUP (MWD/MTU)
6
4
2
0
-2
-4
-6
-8
-10
-12
-14
-16
-18
-20
-22
-24
-26
-28
-30
-32
-34
-36
-38
-40
-42
-44
0.0
1.0
2.0
3.0
4.0
5.0
6.0
7.0
8.0
9.0
10.0
11.0
12.0
13.0
14.0
15.0
16.0
17.0

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 21 accident evaluations.

Table 3.0.1

Kewaunee Nuclear Power Plant

List of Current Safety Analyses

Accident	Current Safety Analysis	Ref. No.
Uncontrolled RCCA Withdrawal From a Subcritical Condition		8, 11
Uncontrolled RCCA Withdrawal at Power		8, 11
Control Rod Drop		8, 11
RCC Assembly Misalignment		8, 11
CVCS Malfunction		8, 11
Startup of an Inactive RC Loop		8, 11
Excessive Heat Removal Due to FW System Malfunctions		8, 11
Excessive Load Increase Incident		8, 11
Loss of Reactor Coolant Flow		
Due to Pump Trip		8, 11
Due to Underfrequency Trip		8, 11
Locked Rotor Accident		8, 11
Loss of External Electrical Load		8, 11
Loss of Normal Feedwater		8, 11
Fuel Handling Accidents		8
Rupture of a Steam Pipe		8, 11
Rupture of CR Drive Mechanism Housing		8, 11
RC System Pipe Rupture (LOCA)		9, 10

Table 3.0.2

Safety Analyses Bounding Values

Parameter	Lower Bound	Upper Bound	Units
Moderator Temp. Coefficient			
Most Negative	-40.0	---	pcm/°Fm
$0 \leq P \leq 60\%$	---	+5.0	pcm/°Fm
$P > 60\%$	---	0.0	pcm/°Fm
95% of time at HFP	---	-8.0	pcm/°Fm
URW from subcritical only	---	+10.0	pcm/°Fm
Doppler Coefficient	-2.32	-1.0	pcm/°Ff
Differential Boron Worth	-11.2	-7.7	pcm/ppm
Delayed Neutron Fraction	.00485	.00706	---
Prompt Neutron Lifetime	15	N/A	μsec
Shutdown Margin	1.0 (BOC) 2.0 (EOC)	N/A N/A	% Δρ
Differential Rod Worth of 2 Banks Moving	N/A	82	pcm/sec
Ejected Rod Cases			
HFP, BOL			
ßeff	.0055	N/A	---
Rod Worth	N/A	.30	% Δρ
FQ	N/A	5.03	---
HFP, EOL			
ßeff	.0050	N/A	---
Rod Worth	N/A	.42	% Δρ
FQ	N/A	5.1	---
HZP, BOL			
ßeff	.0055	N/A	---
Rod Worth	N/A	.91	% Δρ
FQ	N/A	11.2	---
HZP, EOL			
ßeff	.0050	N/A	---
Rod Worth	N/A	.92	% Δρ
FQ	N/A	13.0	---

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 21 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 21 reload core are conservatively bounded by those used in the current safety analysis, an uncontrolled rod withdrawal from subcritical accident will be less severe than the transient in the current safety analysis. Therefore, the implementation of the Cycle 21 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.1.1

Uncontrolled Rod Withdrawal From Subcritical

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	+0.87	≤	10.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.29	≤	-1.0	pcm/°Ff
C) Differential Rod Worth of Two Moving Banks	.056	≤	.116	$/sec
D) Scram Worth vs. Time	See Section 2.3			
E) Delayed Neutron Fraction	.00656	≤	.00706	---
F) Prompt Neutron Lifetime	26	≥	15	μsec

3.2 Evaluation of Uncontrolled Rod Withdrawal at Power

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

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

Table 3.2.1 presents a comparison of the Cycle 21 physics parameters to the current safety analysis values for the Uncontrolled Rod Withdrawal at Power Accident.

Since the pertinent parameters from the proposed Cycle 21 reload core are conservatively bounded by those used in the current safety analysis, an uncontrolled rod withdrawal at power accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 21 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.2.1

Uncontrolled Rod Withdrawal at Power

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-4.65	≤	0.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.29	≤	-1.0	pcm/°Ff
C) Differential Rod Worth of Two Moving Banks	.056	≤	.116	$/sec
D) FΔHN	1.55	≤	1.55	---
E) Scram Worth vs. Time	See Section 2.3			
F) Delayed Neutron Fraction	.00656	≤	.00706	---

3.3 Evaluation of Control Rod Misalignment

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

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

Table 3.3.1 presents a comparison of the Cycle 21 FΔHN versus the current safety analysis FΔH limit for the Misaligned Rod Accident.

Since the pertinent parameter from the proposed Cycle 21 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. Therefore, the implementation of the Cycle 21 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.3.1

Control Rod Misalignment

Parameter	Reload Safety Evaluation Value		Current Safety Analysis
A) FΔHN	1.97	≤	2.03

3.4 Evaluation of Dropped Rod

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

Table 3.4.1 presents a comparison of the Cycle 21 physics parameters to the current safety analysis values for the Dropped Rod Accident.

Since the pertinent parameters from the proposed Cycle 21 reload core are conservatively bounded by that used in the current safety analysis, a dropped rod accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 21 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.4.1

Dropped Rod

Parameter	Reload Safety Evaluation Value		Current Safety Analysis	Units
A) FΔHN	1.55	≤	1.55	---
B) Doppler Temp. Coefficient	-1.29	≤	-1.0	pcm/°Ff
C) Delayed Neutron Fraction	.00656	≤	.00706	---
D) Excore Tilt (Control)	.86	≥	.80	---
E) Full Power Insertion Limit Worth (BOL)	329	≤	400	pcm
F) Full Power Insertion Limit Worth (EOL)	432	≤	450	pcm
G) Moderator Temperature Coefficient (BOL)	-4.65	≤	0.0	pcm/°Fm
H) Moderator Temperature Coefficient (EOL)	-21.0	≤	-17.0	pcm/°Fm

3.5 Evaluation of Uncontrolled Boron Dilution

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

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

Table 3.5.1 presents a comparison of Cycle 21 physics analysis results to the current safety analysis values for the Uncontrolled Boron Dilution Accident for refueling and full power core conditions.

Since the pertinent parameters from the proposed Cycle 21 reload core are conservatively bounded by those used in the current safety analysis, an uncontrolled boron dilution accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 21 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.5.1

Uncontrolled Boron Dilution

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
i) Refueling Conditions				
A) Shutdown Margin	6.5	≥	5.0	%
ii) At-Power Conditions				
A) Moderator Temp. Coefficient	-4.65	≤	0.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.29	≤	-1.0	pcm/°Ff
C) Reactivity Insertion Rate by Boron	.0019	≤	.0023	$/sec
D) Shutdown Margin	2.01	≥	1.0	%
E) FΔHN	1.55	≤	1.55	---
F) Delayed Neutron Fraction	.00656	≤	.00706	---

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 rcactivity effect of the cold water injection. Doppler temperature coefficient is chosen conservatively low (absolute value) to maximize the nuclear power rise. The power distribution (FΔH) is used to evaluate the core thermal limit acceptability.

Table 3.6.1 presents a comparison of the Cycle 21 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 21 reload core are conservatively bounded by those used in the current safety analysis, the startup of an inactive loop accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 21 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.6.1

Startup of an Inactive Loop

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-36.6	≥	-40.0	pcm/°Fm
B) Doppler Coefficient	-1.83	≤	-1.0	pcm/°Ff
C) FΔHN	1.55	≤	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 21 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 21 reload core are conservatively bounded by those used in the current safety analysis, a feedwater system malfunction will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 21 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.7.1

Feedwater System Malfunction

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
i) Beginning of Cycle				
A) Moderator Temp. Coefficient	-4.65	≤	0.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.29	≤	-1.0	pcm/°Ff
ii) End of Cycle				
A) Moderator Temp. Coefficient	-29.82	≥	-40.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.29	≤	-1.0	pcm/°Ff
iii) Beginning and End of Cycle				
C) $F_{\Delta H}^{N}$	1.55	≤	1.55	---

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 21 physics results to the current safety analysis values for the Excessive Load Increase Accident.

Since the pertinent parameters from the proposed Cycle 21 reload core are conservatively bounded by those used in the current safety analysis, an excessive load increase accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 21 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.8.1

Excessive Load Increase

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
i) Beginning of Cycle				
A) Moderator Temp. Coefficient	-4.65	≤	0.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.29	≤	-1.0	pcm/°Ff
ii) End of Cycle				
A) Moderator Temp. Coefficient	-29.82	≥	-40.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.29	≤	-1.0	pcm/°Ff
iii) Beginning and End of Cycle				
C) FΔHN	1.55	≤	1.55	---

3.9 Evaluation of Loss of Load

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

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

A comparison of Cycle 21 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 21 reload core are conservatively bounded by those used in the current safety analysis, a loss of load accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 21 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.9.1

Loss of Load

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
i) Beginning of Cycle				
A) Moderator Temp. Coefficient	-4.65	≤	0.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.58	≥	-2.32	pcm/°Ff
ii) End of Cycle				
A) Moderator Temp. Coefficient	-29.82	≥	-40.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.59	≥	-2.32	pcm/°Ff
iii) Beginning and End of Cycle				
C) FΔHN	1.55	≤	1.55	---
D) Scram Worth Versus Time	See Section 2.3			

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

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

Table 3.11.1 presents a comparison of Cycle 21 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 21 reload core are conservatively bounded by those used in the current safety analysis, a loss of reactor coolant flow due to pump trip accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 21 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.11.1

Loss of Reactor Coolant Flow Due to Pump Trip

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-4.65	≤	0.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.59	≥	-2.32	pcm/°Ff
C) FΔHN	1.55	≤	1.55	---
D) Scram Worth Versus Time	See Section 2.3			
E) Fuel Temperature	2090	≤	2100	°F

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 21 physics parameters to the current safety analysis values for the Locked Rotor Accident.

Since the pertinent parameters from the proposed Cycle 21 reload core are conservatively bounded by those used in the current safety analysis, a locked rotor accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 21 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.12.1

Loss of Reactor Coolant Flow Due to Locked Rotor

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-4.65	≤	0.0	pcm/°Fm
B) Doppler Temp. Coefficient	-1.59	≥	-2.32	pcm/°Ff
C) Delayed Neutron Fraction	.00656	≤	.00706	---
D) Percent Pins > Limiting FΔHN (DNBR=1.3)	28.9	≤	40.0	%
E) Scram Worth Versus Time	See Section 2.3			
F) FQ	2.26	≤	2.28	---
G) Fuel Temperature	2090	≤	2100	°F

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 (FΔH) 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 21 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 21 reload core are conservatively bounded by those used in the current safety analysis, a main steam line break accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 21 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

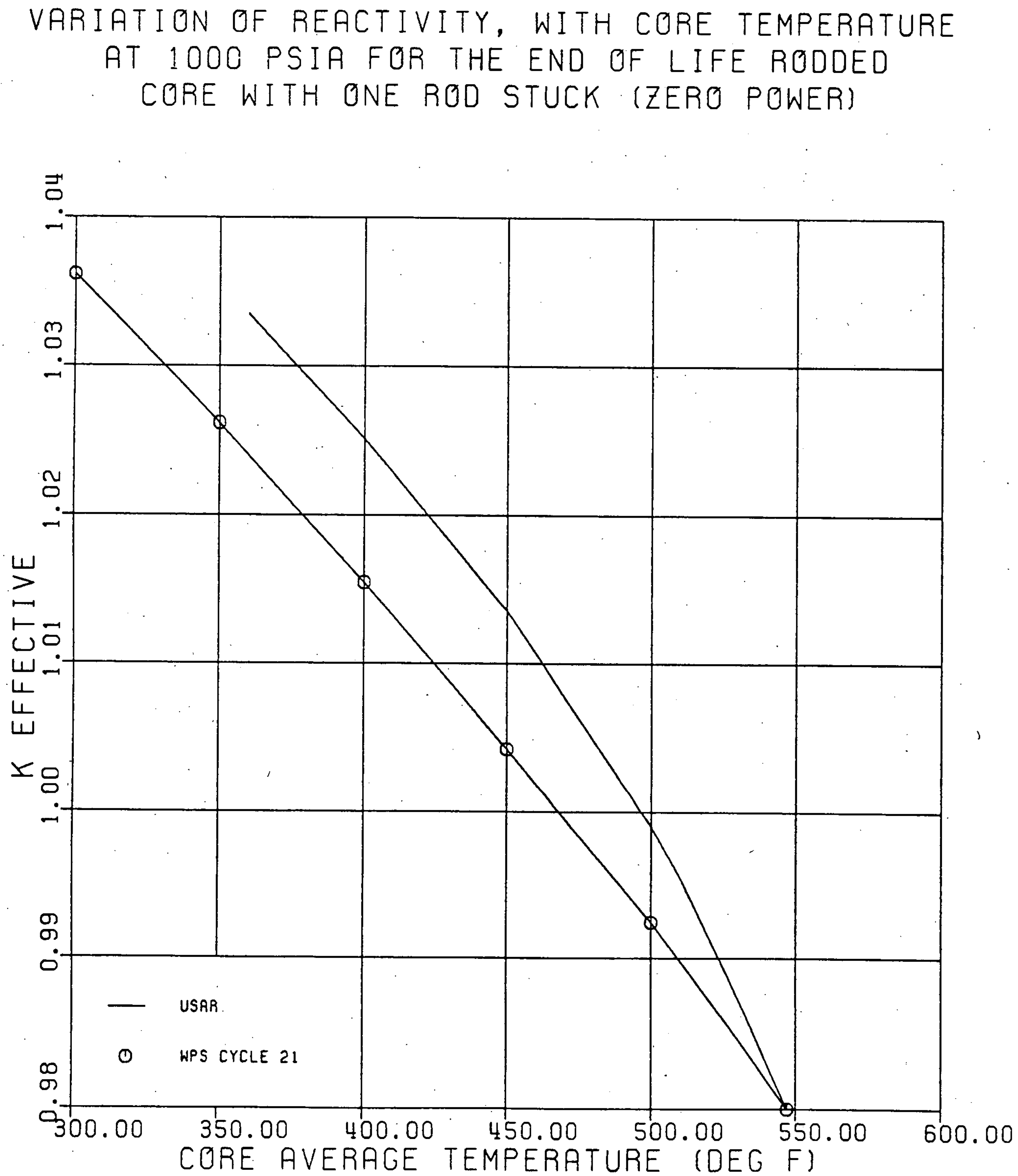
Table 3.13.1

Main Steam Line Break

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Shutdown Margin	2.01	≥	2.00	%Δρ
B) FΔH	≤ 4.25*	≤	4.25	---
C) Doppler Temp. Coefficient	-1.29	≤	-1.0	pcm/°Ff
D) Boron Worth Coefficient	-8.0	≤	-7.7	pcm/ppm

* Based on engineering judgement

Figure 3.13.1


VARIATION OF REACTIVITY, WITH CORE TEMPERATURE
AT 1000 PSIA FOR THE END OF LIFE RODDED
CORE WITH ONE ROD STUCK (ZERO POWER)
K EFFECTIVE
1.04
1.03
1.02
1.01
1.00
0.99
0.98
USAR
WPS CYCLE 21
300.00
350.00
400.00
450.00
500.00
550.00
600.00
CORE AVERAGE TEMPERATURE (DEG F)

3.14 Evaluation of Rod Ejection Accidents

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

Tables 3.14.1 through 3.14.4 present the comparison of Cycle 21 calculated physics parameters to the current safety analysis values for the Rod Ejection Accident at zero and full power, BOC and EOC core conditions.

Since the pertinent parameters from the proposed Cycle 21 reload core are conservatively bounded by those used in the current safety analysis, a rod ejection accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 21 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.14.1

Rod Ejection Accident at

HFP, BOC

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-4.65	≤	0.0	pcm/°Fm
B) Delayed Neutron Fraction	.00618	≥	.00550	---
C) Ejected Rod Worth	.07	≤	0.30	%$\Delta\rho$
D) Doppler Temp. Coefficient	-1.30	≤	-1.0	pcm/°Ff
E) Prompt Neutron Lifetime	26.0	≥	15.0	μsec
F) FQN	2.31	≤	5.03	---
G) Scram Worth Versus Time	See Section 2.3			

Table 3.14.2

Rod Ejection Accident at

HZP, BOC

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	+0.87	≤	5.0	pcm/°Fm
B) Delayed Neutron Fraction	.00618	≥	.00550	---
C) Ejected Rod Worth	0.59	≤	0.91	%$\Delta\rho$
D) Doppler Temp. Coefficient	-2.10	≤	-1.0	pcm/°Ff
E) Prompt Neutron Lifetime	26.0	≥	15.0	μsec
F) FQN	5.02	≤	11.2	---
G) Scram Worth Versus Time	See Section 2.3			

Table 3.14.3

Rod Ejection Accident at

HFP, EOC

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-21.02	≤	0.0	pcm/°Fm
B) Delayed Neutron Fraction	.00518	≥	.00500	---
C) Ejected Rod Worth	0.12	≤	0.42	%$\Delta\rho$
D) Doppler Temp. Coefficient	-1.31	≤	-1.0	pcm/°Ff
E) Prompt Neutron Lifetime	29.4	≥	15.0	μsec
F) FQN	2.86	≤	5.1	---
G) Scram Worth Versus Time	See Section 2.3			

Table 3.14.4

Rod Ejection Accident at

HZP, EOC

Parameter	Reload Safety Evaluation Values		Current Safety Analysis	Units
A) Moderator Temp. Coefficient	-7.28	≤	5.0	pcm/°Fm
B) Delayed Neutron Fraction	.00518	≥	.00500	---
C) Ejected Rod Worth	0.69	≤	0.92	%$\Delta\rho$
D) Doppler Temp. Coefficient	-2.40	≤	-1.0	pcm/°Ff
E) Prompt Neutron Lifetime	29.4	≥	15.0	μsec
F) FQN	7.61	≤	13.0	---
G) Scram Worth Versus Time	See Scction 2.3			

3.15 Evaluation of Fuel Handling Accident

This accident is the sudden release of the gaseous fission products held within the fuel cladding of one fuel assembly. The fraction of fission gas released is based on a conservative assumption of high power in the fuel rods 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 maximum Cycle 21 FQN calculated during the last 2.0 GWD/MTU of the cycle to the current safety analysis FQN limit for the Fuel Handling Accident.

Since the pertinent parameter from the proposed Cycle 21 reload core is conservatively bounded by that used in the current safety analysis, a fuel handling accident will be less severe than the accident in the current analysis. Therefore, the implementation of the Cycle 21 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.15.1

Fuel Handling Accident

Parameter	Reload Safety Evaluation Values		Current Safety Analysis
A) FQN	1.98	≤	2.53

3.16 Evaluation of Loss of Coolant Accident

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

The principal parameters which affect the results of LOCA analysis are the fuel stored energy, fuel rod internal pressures, and dccay 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 21 physics calculation results to the current safety analysis values for the Loss of Coolant Accident.

Since the pertinent parameters from the proposed Cycle 21 reload core are conservatively bounded by those used in the current safety analysis, a loss of coolant accident will be less severe than the transient in the current analysis. Therefore, the implementation of the Cycle 21 reload core design will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.16.1

Loss of Coolant Accident

Parameter	Reload Safety Evaluation Values		Current Safety Analysis
A) Scram Worth Versus Time	See Section 2.3		
B) FQ	See Section 3.17		
C) FΔH	1.55	≤	1.55

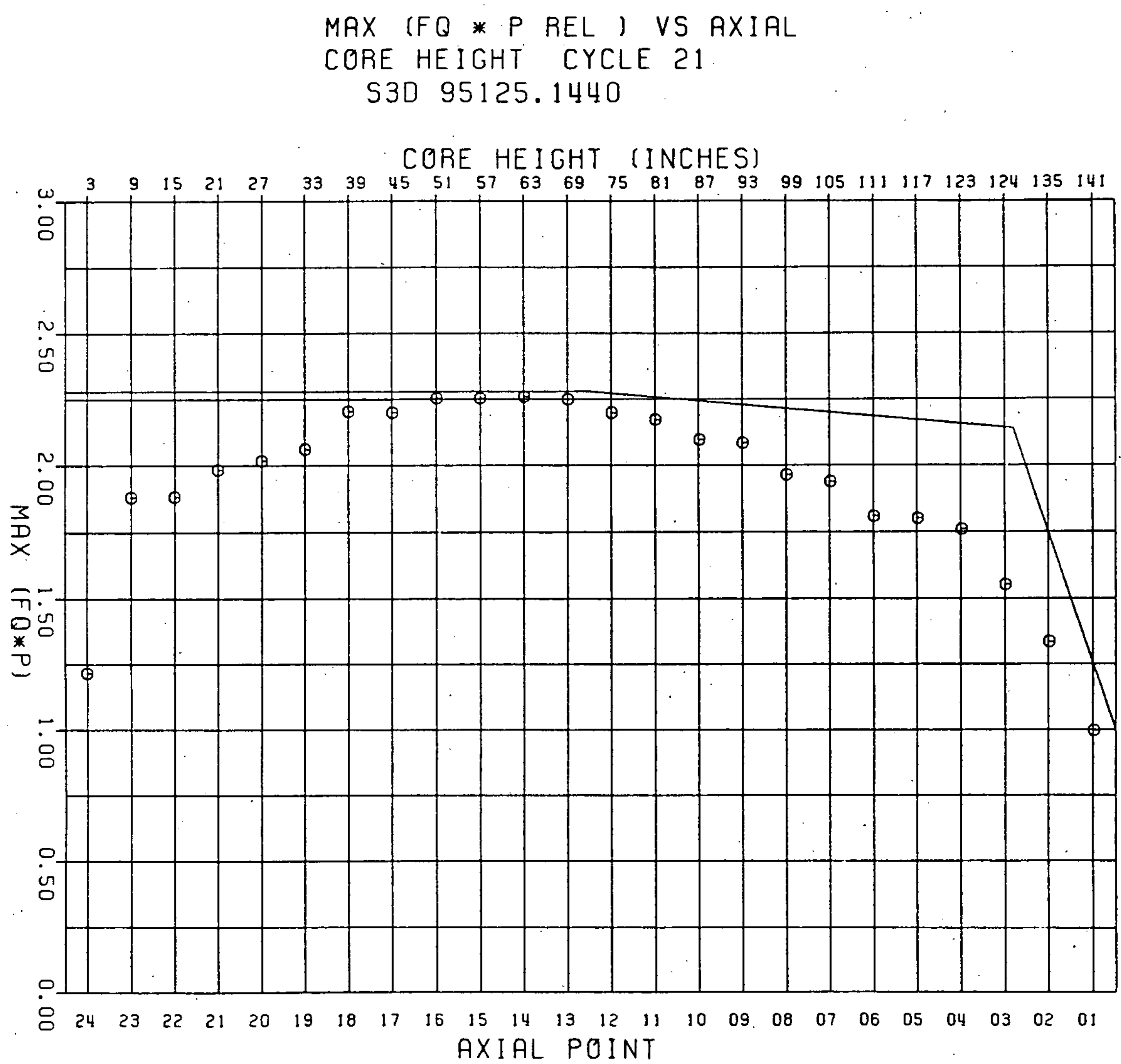
3.17 Power Distribution Control Verification

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 (Reference 7) 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 21 operation will not preclude full power operation under the power distribution control specifications currently applied (Reference 5).

Figure 3.17.1



MAX (FQ * P REL) VS AXIAL
CORE HEIGHT CYCLE 21
S3D 95125.1440
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

The following Technical Specification revisions are in effect for Cycle 21:

The moderator temperature coefficient upper bound limit was revised. The MTC limit is +5.0 pcm/°F for $0 \leq P \leq 60\%$ and 0.0 for $P > 60\%$ (Ref. 5).

The RWST minimum boron concentration is 2400 ppm (Ref. 5).

5.0 STATISTICS UPDATE

Measurements and calculations of Cycles 18, 19, and 20 are incorporated into the FQN and FΔH statistics data base. The moderator temperature coefficient statistics data base includes results from Cycles 13 through 20. The reliability and bias factors used for the Cycle 21 Reload Safety Analyses are presented in Tables 5.0.1 and 5.0.2.

Table 5.0.1

Reliability Factors

Parameter	Reliability Factor	Bias
FQN	See Table 5.0.2	---
FΔH	4.08%	0
Rod Worth	10.0%	0
Moderator Temperature Coefficient	2.7 pcm/°F	3.1 pcm/°F
Doppler Coefficient	10.0%	0
Boron Worth	5.0%	0
Delayed Neutron Parameters	3.0%	0

Table 5.0.2

FQN Reliability Factors

Core Level	σNode	RF (%)
1 (Bottom)	.0585	10.23
2	.0476	8.46
3	.0186	4.20
4	.0211	4.51
5	.0237	4.85
6	.0189	4.23
7	.0206	4.45
8	.0187	4.21
9	.0196	4.32
10	.0167	3.98
11	.0163	3.93
12	.0169	4.00
13	.0161	3.91
14	.0163	3.93
15	.0172	4.03
16	.0167	3.98
17	.0210	4.50
18	.0184	4.17
19	.0263	5.21
20	.0248	5.00
21	.0452	8.07
22	.0357	6.59
23	.0806	13.91
24 (Top)	.0752	13.00

6.0 REFERENCES

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

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

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

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

5. Wisconsin Public Service Corporation Technical Specifications for the Kewaunee Nuclear Power Plant. Docket Number 50-305, Amendment No. 117, dated April 3, 1995.

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. 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.

8. Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, Updated Safety Analysis Report, Revision 9.

9. "Kewaunee Limiting Break LOCA/ECCS Analysis" XN-NF-85-98, Revision 1, Advanced Nuclear Fuels Corp., March 1987.

10. "Kewaunee Large Break LOCA/ECCS Analysis with 25% Steam Generator Tube Plugging", EMF-95-075(P), April 1995.

11. Kewaunee Engineering Support Request (ESR) 95-023.

12. WPS letter from C. R. Steinhardt to U.S. Nuclear Regulatory Commission, Docket 50-305, dated June 19, 1991 "Core Reloads of Advanced Design Fuel Assemblies".

13. SPC letter from H. G. Shaw (SPC) to S. F. Wozniak dated December 11, 1992, "Disposition of LBLOCA Analysis for Kewaunee with Four Westinghouse Lead Assemblies".

14. SPC Report EMF-94-207 dated November 1994, "Kewaunee HTP Lead Assembly Compatibility Report".

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