# KEWAUNEE NUCLEAR POWER PLANT

# RELOAD SAFETY EVALUATION
# CYCLE 20
# FEBRUARY 1994

WISCONSIN PUBLIC SERVICE CORPORATION
WISCONSIN POWER & LIGHT COMPANY
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RELOAD SAFETY EVALUATION

FOR

KEWAUNEE CYCLE 20

| | | |
|---|---|---|
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## 1.0 SUMMARY

The Kewaunee Nuclear Power Plant is scheduled to shut down for the Cycle 19-20 refueling in April 1994. Startup of Cycle 20 is forecast for May 1994.

This report presents an evaluation of the Cycle 20 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 20 design is more conservative than results of previously docketed accident analyses and implementation of this design will not introduce an unreviewed safety question sinee:

1. the probability of occurrenee 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 redueed.

This conclusion is based on these assumptions: There is adherence to plant operating limitations and Technical Specifications (Reference 5), and Cycle I9 is shut down within a +300 MWD/MTU, -300 MWD/MTU window of the nominal design End of Cycle (EOC) burnup of I1,500 MWD/MTU.

## 2.0 CORE DESIGN

### 2.1 Core Description

The reactor core consists of 121 fuel assemblies of 14 x I4 design. The core loading pattern, assembly identification, control rod bank identification, instrument thimble I.D., thermocouple I.D., and burnable poison red configurations for Cycle 20 are presented in Figure 2.I.1.

Twelve (12) new Siemens Power Corporation (SPC) standard assemblies enriched to 3.5 w/o U235 and twenty-four (24) SPC standard assemblies enriched to 3.7 w/o U235 will reside with 81 partially depleted SPC assemblies and four (4) Westinghouse Electric OFA assemblies. Table 2.1.1 displays the core breakdown by region, enrichment, and number of previous duty cycles. Reference 6 describes the SPC 14 x 14 design. References 16 and 17 describe the Westinghouse OFA design.

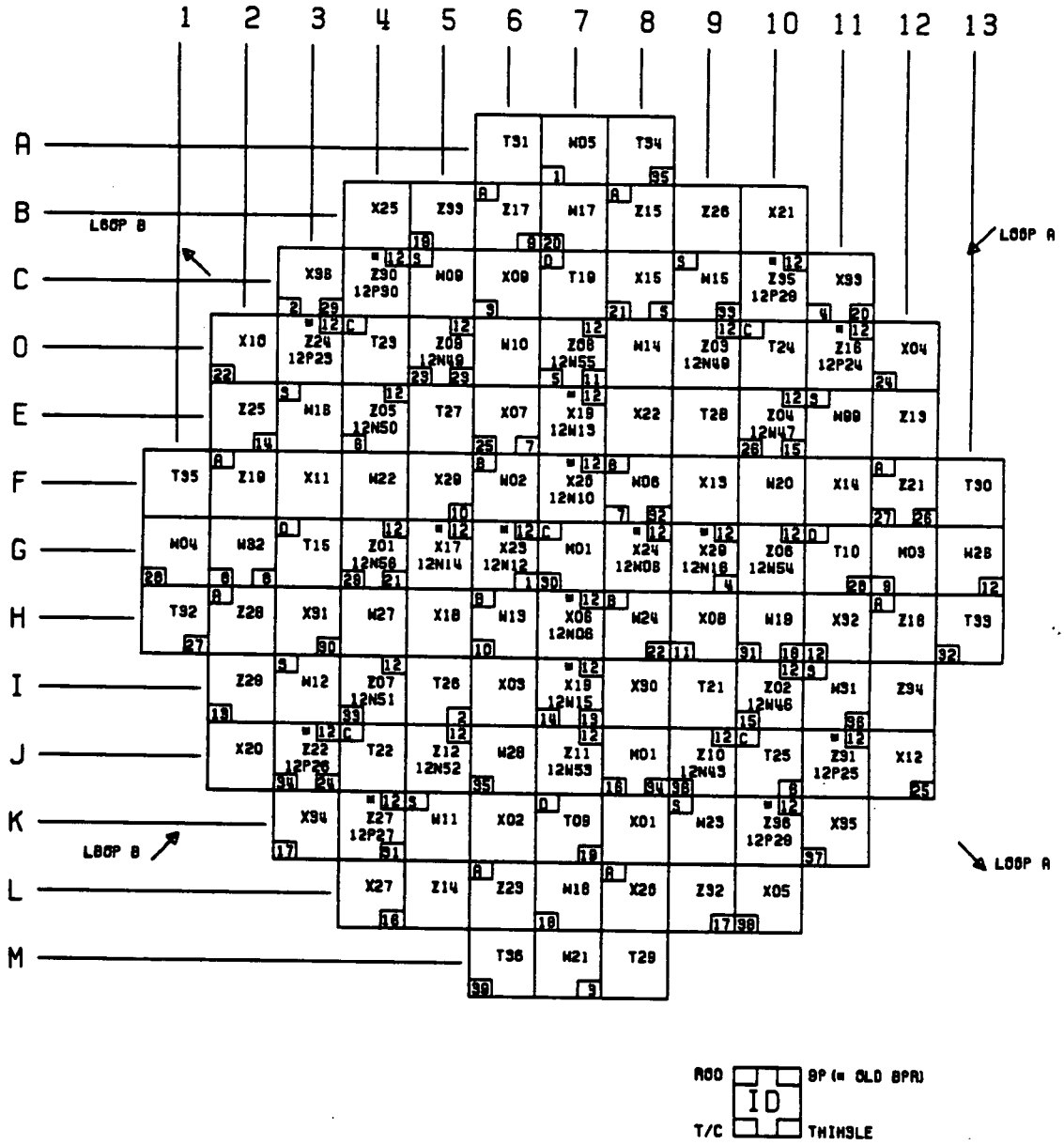
The Cycle 20 reload core will employ 28 burnable poison rod assemblies (BPRAs) containing I44 fresh and 192 partially depleted burnable poison rods. Fuel assemblies with two or three previous duty cycles are loaded on the core periphery flat region to lower power in that region and reduce reactor vessel fluence (Reference I4) in the critical reactor vessel locations.

Table 2.1.1

Cycle 20 Fuel Characteristics

| Region | Region Identifier | Initial W/O U235 | Number of Previous Duty Cycles | Assemblies |
|---|---|---|---|---|
| 13 | M | 3.400 | 3 | 1 |
| 18 | T | 3.400 | 2 | 4 |
| 18 | T | 3.400 | 3 | 8 |
| 18 | T | 3.500 | 3 | 8 |
| 20 | W | 3.400 | 2 | 28 |
| 21 | X | 3.400 | 1 | 32 |
| 21 | X | 3.100 | 1 | 4 |
| 22 | Z | 3.500 | 0 | 12 |
| 22 | Z | 3.700 | 0 | 24 |

Figure 2.1.1




CYCLE 20
LOADING PATTERN

## 2.2 Operating Conditions, Limits, and Design Objectives

Cycle 20 core design is based on the following operating conditions, limits, and design objectives.

### 2.2.1 Operating Conditions

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

### 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 19).

(ii) FΔH 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:

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 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 line 3.10.b.11.a. 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.

2.2.3 Design Objectives

A. The fuel loading pattern shall be capable of generating approximately 10,700 MWD/MTU based on a nominal end of Cycle 19 burnup of 11,500 MWD/MTU.

B. Fuel duty during this fuel cycle will assure peak fuel rod burnups less than the maximum burnup recommended by the fuel vendors.

C. The fuel loading pattern shall be a "lower" neutron leakage design in order to reduce vessel fluence in critical reactor vessel locations.

D. The Westinghouse Electric OFA assemblies will not be limiting with respect to power distribution and LOCA analysis assumptions (Reference 18).

FIGURE 2.2.1

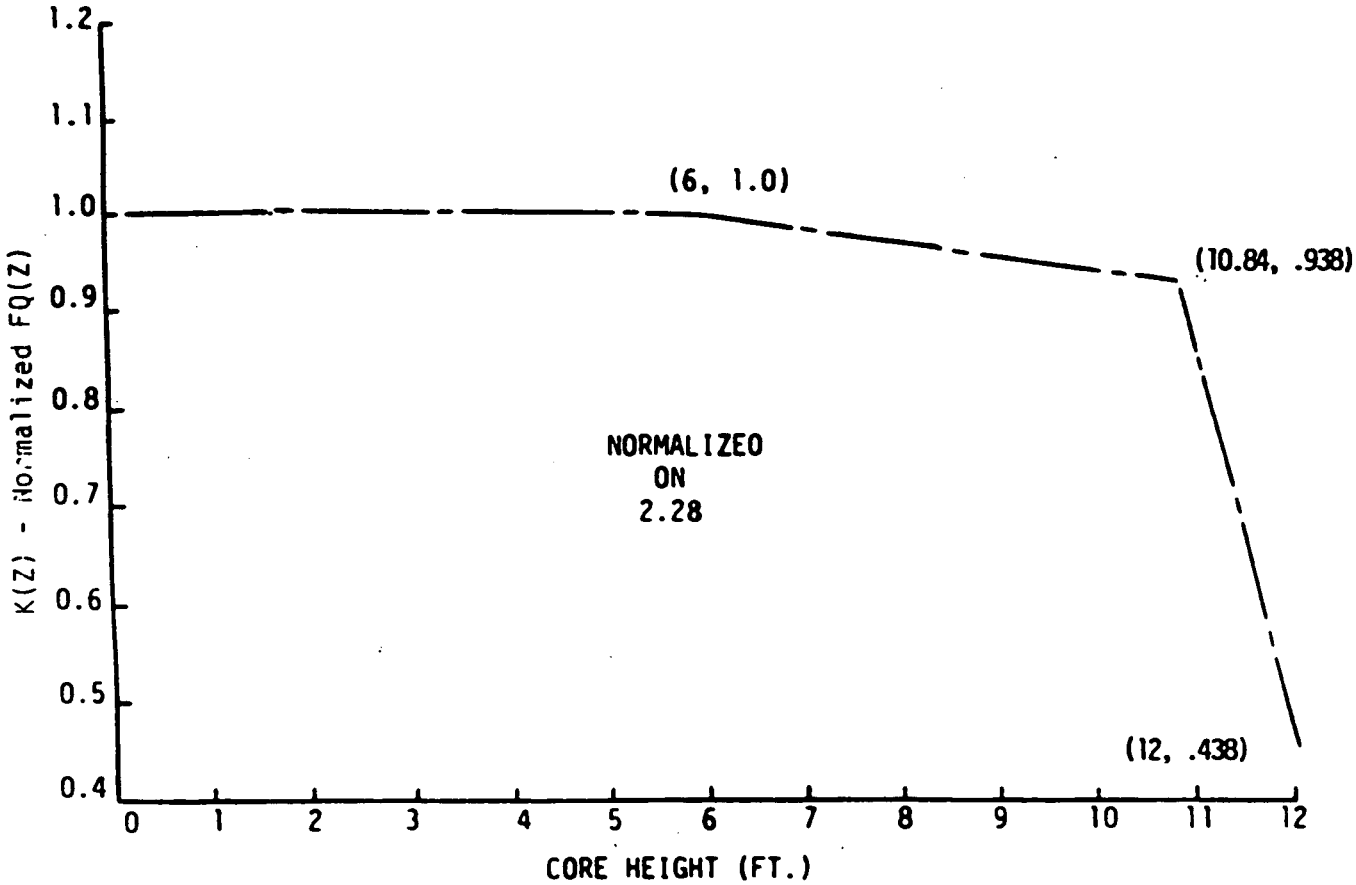



FIGURE 2.2.1 - HOT CHANNEL FACTOR
NORMALIZED OPERATING ENVELOPE

FIGURE 2.2.2

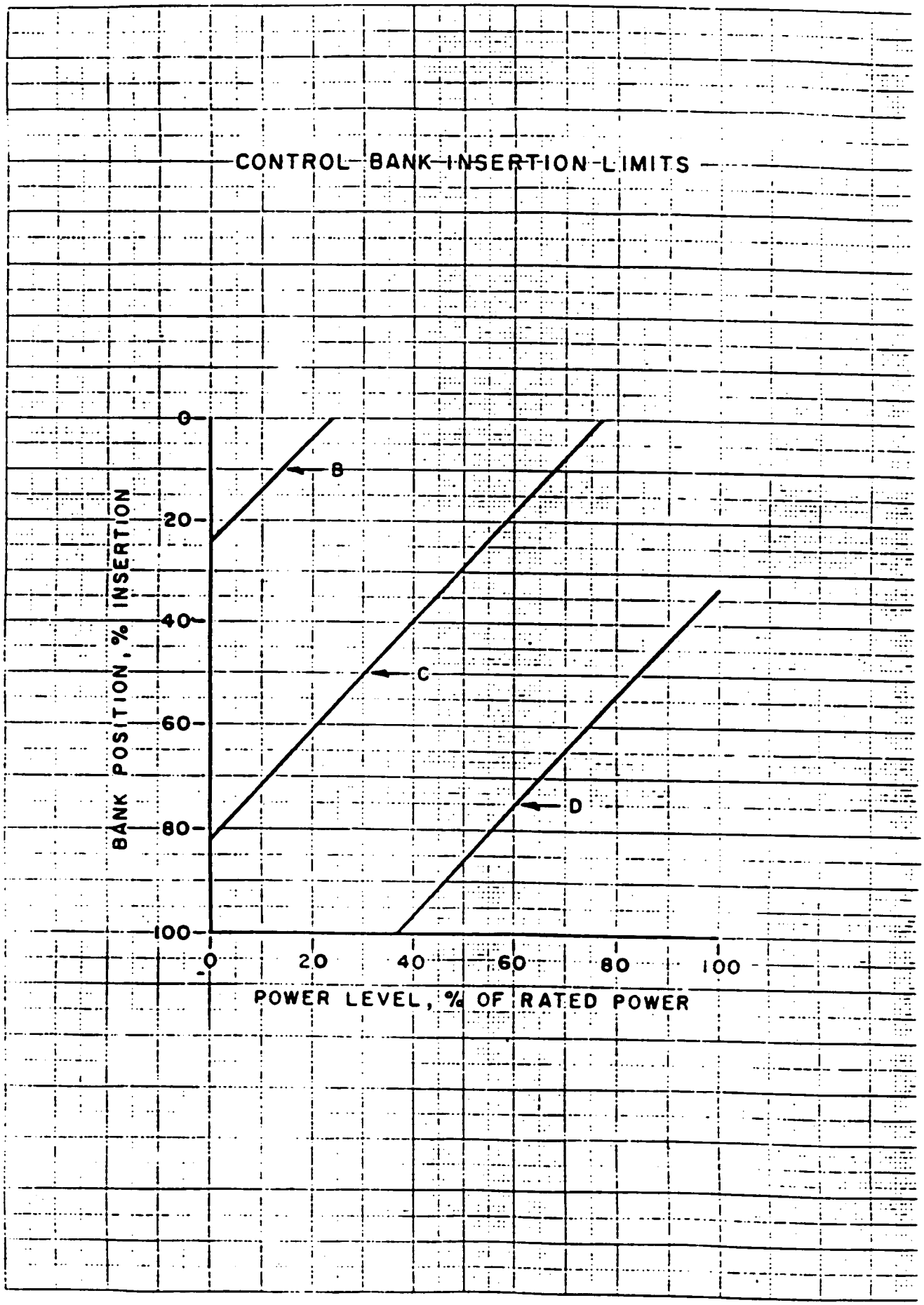
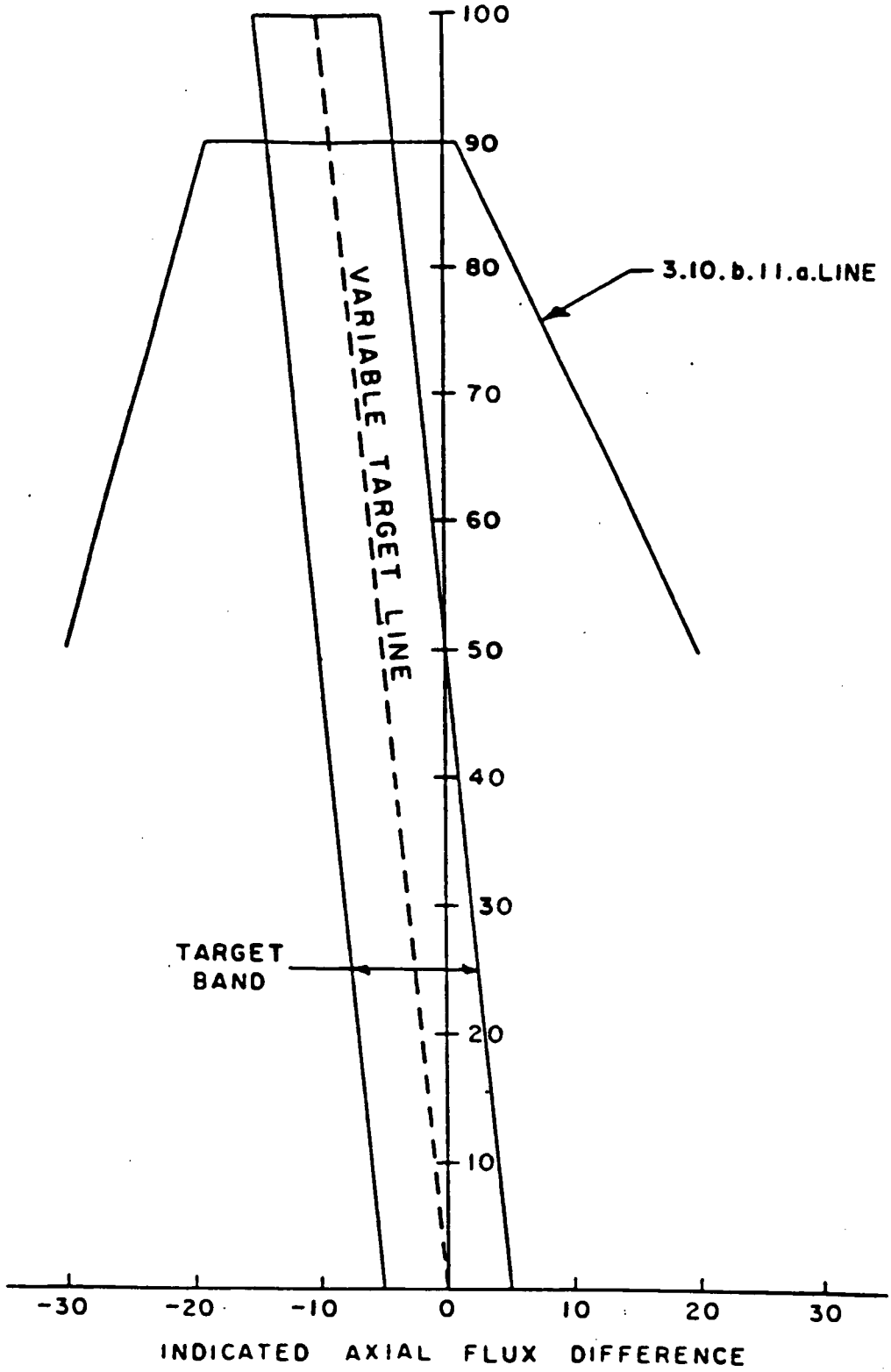

CONTROL BANK INSERTION LIMITS
BANK POSITION, % INSERTION
0
20
40
60
80
100
B
C
D
0
20
40
60
80
100
POWER LEVEL, % OF RATED POWER

FIGURE 2.2.3



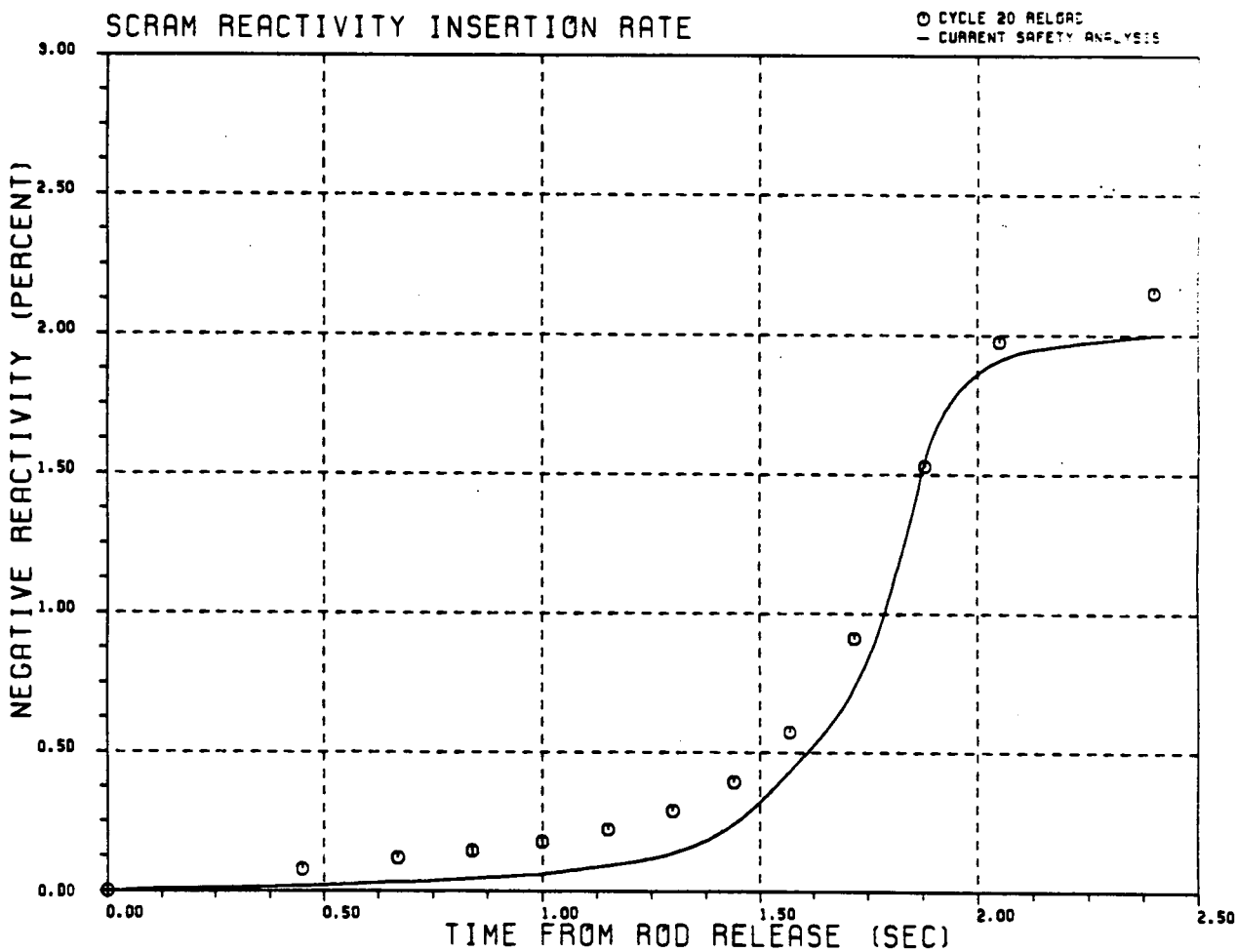

Target Band on Indicated Flux Difference
As a Function of Operating Power Level (Typical)

## 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 20 minimum shutdown margin is 2.15 percent at end of cycle hot full power conditions. Figure 2.3.1 compares the Cycle 20 minimum scram insertion curve to the current bounding safety analysis curve.

It is concluded that the minimum trip reactivity insertion rate for Cycle 20 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 20 RELOAD
CURRENT SAFETY ANALYSIS
NEGATIVE REACTIVITY (PERCENT)
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
TIME FROM ROD RELEASE (SEC)

## 2.4 Shutdown Window

An evaluation of the maximum full power equilibrium peaking factors versus EOC 19 burnup is presented in Table 2.4.1. The values shown have conservatisms applied in accordance with References 1 and 7.

It is concluded that if the refueling shutdown of Cycle 19 occurs within the burnup window, the Cycle 20 peaking factors will not be significantly affected and will not exceed their limiting values.

Table 2.4.1

Peaking Factor Versus Cycle 19 Shutdown Burnup

| | FΔH | | FQ | |
|---|---|---|---|---|
| | Cycle 20 | Limit | Cycle 20 | Limit |
| EOC 19 - 300 MWD/MTU | 1.53 | 1.55 | 2.13 | 2.28 |
| EOC 19 Nominal | 1.52 | 1.55 | 2.14 | 2.28 |
| EOC 19 + 300 MWD/MTU | 1.53 | 1.55 | 2.15 | 2.28 |

## 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 20 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) | 9 |
| Uncontrolled RCCA Withdrawal at Power | 2/78 (Cycle 4-RSE) | 9 |
| Control Rod Drop | 7/1/91 (Rev. 9-USAR) | 8 |
| RCC Assembly Misalignment | 7/1/91 (Rev. 9-USAR) | 8 |
| CVCS Malfunction | 1/27/71 (AM7-USAR) | 8 |
| Startup of an Inactive RC Loop | 1/27/71 (AM7-USAR) | 8 |
| Excessive Heat Removal Due to FW System Malfunctions | 1/27/71 (AM7-USAR) | 8 |
| Excessive Load Increase Incident | I/27/71 (AM7-USAR) | 8 |
| Loss of Reactor Coolant Flow | | |
| Due to Pump Trip | 3/73 (WCAP-8092) | 10 |
| Due to Underfrequency | 7/88 (Rev. 6-USAR) | 8 |
| Locked Rotor Accident | 2/78 (Cycle 4-RSE) | 9 |
| Loss of External Electrical Load | 1/27/71 (AM7-USAR) | 8 |
| Loss of Normal Feedwater | 8/31/73 (AM33-USAR) | 8 |
| Fuel Handling Accidents | 1/27/71 (AM7-USAR) | 8 |
| Rupture of a Steam Pipe | 4/13/73 (AM28-USAR) | 8 |
| Rupture of CR Drive Mechanism Housing | 2/78 (Cycle 4-RSE) | 9 |
| RC System Pipe Rupture (LOCA) | 12/10/76 (AM40-USAR) | 8 |
| Westinghouse | | |
| Zirc - Water Addendum | 12/14/79 | 11 |
| Clad Hoop Stress Addendum | 1/8/80 | 12 |
| Advanced Nuclear Fuels Corporation | 10/01/84 (XN-NF-84-31, Rev. 1) | 13 |

Table 3.0.2

Safety Analyses Bounding Values

| Parameter | Lower Bound | Upper Bound | Units |
|---|---|---|---|
| Moderator Temp. Coefficient | -40.0 | 0.0 | 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)<br>2.0 (EOC) | N/A<br>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 20 physics parameters to the current safety analysis values for the Uncontrolled Rod Withdrawal from a Subcritical Condition.

Sinee the pertinent parameters from the proposed Cycle 20 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. The implementation of the Cycle 20 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.1.1

Uncontrolled Rod Withdrawal From Subcritical

| Parameter | Reload Safety Evaluation Values | | Current Safety Analysis | Units |
|---|---|---|---|---|
| A) Moderator Temp. Coefficient | -1.75 | ≤ | 10.0 | pcm/°Fm |
| B) Doppler Temp. Coefficient | -1.28 | ≤ | -1.0 | pcm/°Ff |
| C) Differential Rod Worth of Two Moving Banks | .054 | ≤ | .116 | $/sec |
| D) Scram Worth vs. Time | See Section 2.3 | | | |
| E) Delayed Neutron Fraction | .00630 | ≤ | .00706 | — |
| F) Prompt Neutron Lifetime | 28 | ≥ | 15 | μsec |

## 3.2 Evaluation of Uncontrolled Rod Withdrawal at Power

An uncontrolled control rod bank withdrawal at power results in a gradnal increase in core power followed by an increase in core heat flux. The resulting mismatch between core pewer 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 20 physics parameters to the current safety analysis values for the Uncontrolled Rod Withdrawal at Power Accident.

Since the pertinent parameters from the proposed Cycle 20 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 20 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.2.1

Uncontrolled Rod Withdrawal at Power

| Parameter | Reload Safety Evaluation Values | | Current Safety Analysis | Units |
|---|---|---|---|---|
| A) Moderator Temp. Coefficient | -1.75 | ≤ | 0.0 | pcm/°Fm |
| B) Doppler Temp. Coefficient | -1.28 | ≤ | -1.0 | pem/°Ff |
| C) Differential Rod Worth of Two Moving Banks | .054 | ≤ | .116 | $/sec |
| D) FΔHN | 1.52 | ≤ | 1.55 | — |
| E) Scram Worth vs. Time | See Section 2.3 | | | |
| F) Delayed Neutron Fraction | .00630 | ≤ | .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 20 FΔHN versus the current safety analysis FΔH limit for the Misaligned Rod Accident.

Since the pertinent parameter from the propesed Cycle 20 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 20 reload core design, therefore, 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 | 1.87 | ≤ | 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 20 physics parameters to the current safety analysis values for the Dropped Rod Accident.

Since the pertinent parameters from the proposed Cycle 20 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. The implementation of the Cycle 20 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.4.1

Dropped Rod

| Parameter | Reload Safety Evaluation Valne | | Cnrrent Safety Analysis | Units |
|---|---|---|---|---|
| A) FΔHN | 1.52 | ≤ | 1.55 | --- |
| B) Doppler Temp. Coefficient | -1.28 | ≤ | -1.0 | pcm/°Ff |
| C) Delayed Neutron Fraction | .00630 | ≤ | .00706 | --- |
| D) Excore Tilt (Control) | .86 | ≥ | .80 | --- |
| E) Full Power Insertion Limit Worth (BOL) | 313 | ≤ | 400 | pcm |
| F) Full Power Insertion Limit Worth (EOL) | 417 | ≤ | 450 | pcm |
| G) Moderator Temperature Coefficient (BOL) | -7.93 | ≤ | 0.0 | pcm/°Fm |
| H) Moderator Temperature Coefficient (EOL) | -20.91 | ≤ | -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 ease. 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 20 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 20 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 20 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.5.1

Uncontrolled Boron Dilution

| Parameter | Reload Safety Evaluation Valnes | | Current Safety Analysis | Units |
|---|---|---|---|---|
| i) Refueling Conditions | | | | |
| A) Shutdown Margin | 10.9 | ≥ | 5.0 | % |
| ii) At-Power Conditions | | | | |
| A) Moderator Temp. Coefficient | -1.75 | ≤ | 0.0 | pcm/°Fm |
| B) Doppler Temp. Coefficient | -1.28 | ≤ | -1.0 | pcm/°Ff |
| C) Reactivity Insertion Rate by Boron | .0021 | ≤ | .0023 | $/sec |
| D) Shutdown Margin | 2.15 | ≥ | 1.00 | % |
| E) FΔHN | 1.52 | ≤ | 1.55 | --- |
| F) Delayed Neutron Fraction | .00630 | ≤ | .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 reactivity effect of the cold water injection. Doppler temperature coefficient is chosen conservatively low (absolutc value) to maximize the nuclear power rise. The power distribution ($F\Delta H$) is used to evaluatc the core thermal limit acceptability.

Table 3.6.1 presents a comparison of the Cycle 20 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 20 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 20 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.6.1

Startup of an Inactive Loop

| Parameter | Reload Safety Evalnation Valnes | | Cnrrent Safety Analysis | Units |
|---|---|---|---|---|
| A) Moderator Temp. Coefficient | -34.08 | ≥ | -40.0 | pcm/°Fm |
| B) Doppler Coefficient | -1.85 | ≤ | -1.0 | pcm/°Ff |
| C) FΔHN | 1.52 | ≤ | 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 20 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 20 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 20 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.7.1

Feedwater System Malfunction

| Parameter | Reload Safety Evaluation Valnes | | Cnrrent Safety Aualysis | Units |
|---|---|---|---|---|
| i) Beginning of Cycle | | | | |
| A) Moderator Temp. Coefficient | -7.93 | ≤ | 0.0 | pcm/°Fm |
| B) Doppler Temp. Coefficient | -1.28 | ≤ | -1.0 | pcm/°Ff |
| ii) End of Cycle | | | | |
| A) Moderator Temp. Coefficient | -28.63 | ≥ | -40.0 | pcm/°Fm |
| B) Doppler Temp. Coefficient | -1.28 | ≤ | -1.0 | pcm/°Ff |
| iii) Beginning and End of Cycle | | | | |
| C) FΔHN | 1.52 | ≤ | 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 eauses 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 20 physics results to the current safety analysis values for the Excessive Load Increase Accident.

Since the pertinent parameters from the proposed Cycle 20 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 20 reload core design, therefore, 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 | -7.93 | ≤ | 0.0 | pcm/°Fm |
| B) Doppler Temp. Coefficient | -1.28 | ≤ | -1.0 | pcm/°Ff |
| ii) End of Cycle | | | | |
| A) Moderator Temp. Coefficient | -28.63 | ≥ | -40.0 | pcm/°Fm |
| B) Doppler Temp. Coefficient | -1.28 | ≤ | -1.0 | pcm/°Ff |
| iii) Beginning and End of Cycle | | | | |
| C) FΔHN | 1.52 | ≤ | 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 fiux 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 20 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 20 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 20 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.9.1

Loss of Load

| Parameter | Reload Safety Evaluation Values | | Curreut Safety Analysis | Units |
|---|---|---|---|---|
| i) Beginning of Cycle | | | | |
| A) Moderator Temp. Coefficient | -7.93 | ≤ | 0.0 | pcm/°Fm |
| B) Doppler Temp. Coefficient | -1.57 | ≥ | -2.32 | pcm/°Ff |
| ii) End of Cycle | | | | |
| A) Moderator Temp. Coefficient | -28.63 | ≥ | -40.0 | pcm/°Fm |
| B) Doppler Temp. Coefficient | -1.56 | ≥ | -2.32 | pcm/°Ff |
| iii) Beginning and End of Cycle | | | | |
| C) FΔHN | 1.52 | ≤ | 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 $F\Delta H$ are evaluated to ensure core thermal margin.

Table 3.11.1 presents a comparison of Cycle 20 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 20 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 20 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.11.1

Loss of Reactor Ccolant Flow Due to Pump Trip

| Parameter | Reload Safety Evaluatiou Values | | Current Safety Analysis | Units |
|---|---|---|---|---|
| A) Moderator Temp. Coefficient | -7.93 | ≤ | 0.0 | pcm/°Fm |
| B) Doppler Temp. Coefficient | -1.57 | ≥ | -2.32 | pcm/°Ff |
| C) FΔHN | 1.52 | ≤ | 1.55 | --- |
| D) Scram Worth Versus Time | See Section 2.3 | | | |
| E) Fuel Temperature | 2035 | ≤ | 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 20 physics parameters to the current safety analysis values for the Locked Rotor Accident.

Since the pertinent parameters from the proposed Cycle 20 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 20 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.12.1

Loss of Reactor Coolant Flow Due to Locked Rotor

| Parameter | Reload Safety Evaluatiou Values | | Current Safety Analysis | Units |
|---|---|---|---|---|
| A) Moderator Temp. Coefficient | -7.93 | ≤ | 0.0 | pcm/°Fm |
| B) Doppler Temp. Coefficient | -1.57 | ≥ | -2.32 | pcm/°Ff |
| C) Delayed Neutron Fraction | .00630 | ≤ | .00706 | --- |
| D) Percent Pins > Limiting FΔHN (DNBR=1.3) | 25.85 | ≤ | 40.0 | % |
| E) Scram Worth Versus Time | See Section 2.3 | | | |
| F) FQ | 2.15 | ≤ | 2.28 | --- |
| G) Fuel Temperature | 2035 | ≤ | 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 20 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 20 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 20 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

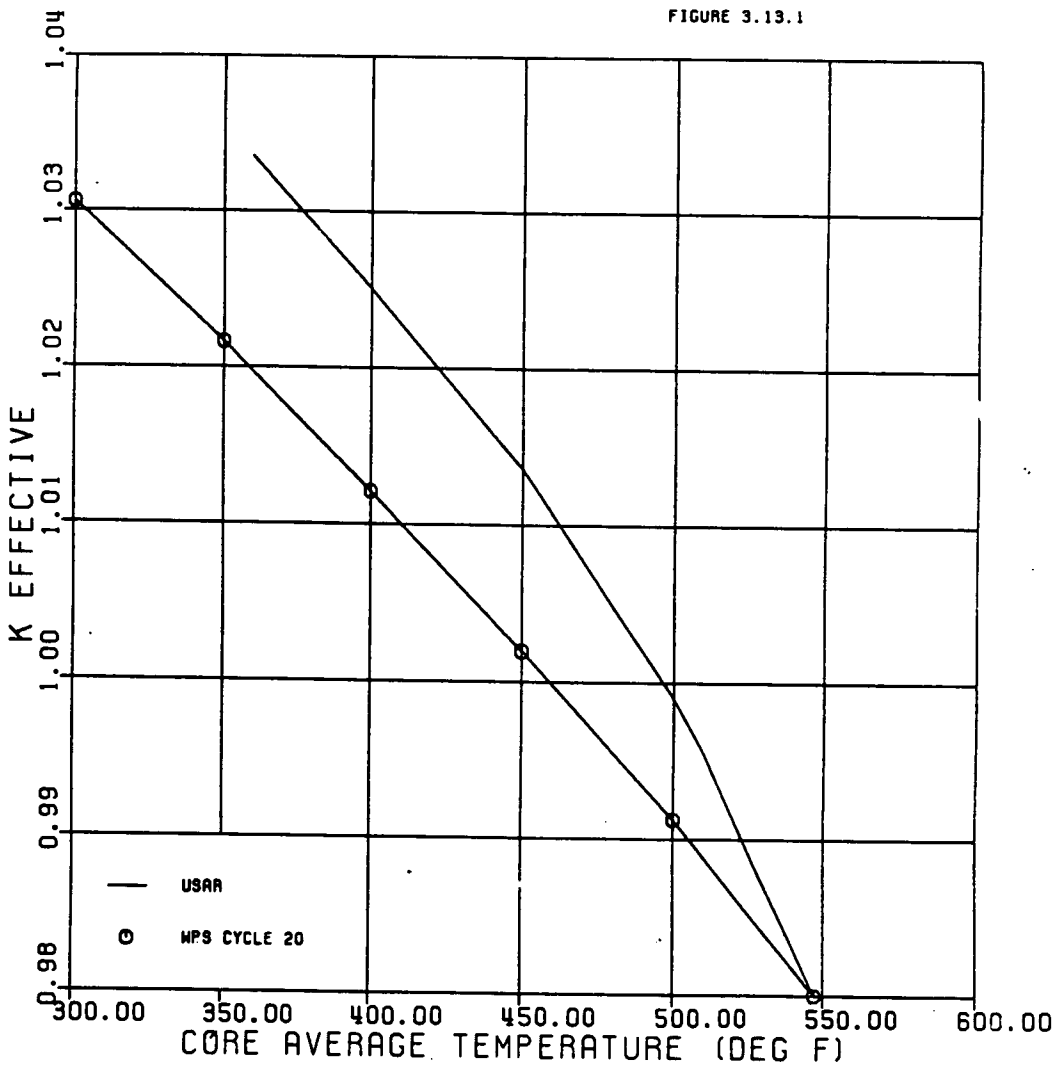
Table 3.13.1

Main Steam Line Break

| Parameter | Reload Safety Evaluatiou Values | | Curreut Safety Analysis | Units |
|---|---|---|---|---|
| A) Shutdown Margin | 2.15 | ≥ | 2.00 | %Δρ |
| B) FΔH | 4.18 | ≤ | 6.6 | --- |
| C) Doppler Temp. Coefficient | -1.28 | ≤ | -1.0 | pcm/°Ff |
| D) Boron Worth Coefficient | -8.4 | ≤ | -7.7 | pcm/ppm |

**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)



## 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 20 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 20 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 20 reload core design, therefore, will not adversely affect the safe operation of the Kewaunce Plant.

Table 3.14.1

Rod Ejection Accident at

HFP, BOC

| Parameter | Reload Safety Evaluation Values | | Current Safety Analysis | Units |
|---|---|---|---|---|
| A) Moderator Temp. Coefficient | -7.93 | ≤ | 0.0 | pcm/°Fm |
| B) Delayed Neutron Fraction | .00593 | ≥ | .00550 | --- |
| C) Ejected Rod Worth | .07 | ≤ | 0.30 | $\%\Delta\rho$ |
| D) Doppler Temp. Coefficient | -1.29 | ≤ | -1.0 | pcm/°Ff |
| E) Prompt Neutron Lifetime | 28.0 | ≥ | 15.0 | $\mu$sec |
| F) FQN | 2.25 | ≤ | 5.03 | --- |
| G) Scram Worth Versus Time | See Section 2.3 | | | |

Table 3.14.2

Rod Ejection Aecident at

HZP, BOC

| Parameter | Reload Safety Evalnation Values | | Current Safety Analysis | Units |
|---|---|---|---|---|
| A) Moderator Temp. Coefficient | -1.75 | ≤ | 0.0 | pcm/°Fm |
| B) Delayed Neutron Fraction | .00593 | ≥ | .00550 | --- |
| C) Ejected Rod Worth | 0.45 | ≤ | 0.91 | $\%\Delta\rho$ |
| D) Doppler Temp. Coefficient | -2.21 | ≤ | -1.0 | pcm/°Ff |
| E) Prompt Neutron Lifetime | 28.0 | ≥ | 15.0 | $\mu$sec |
| F) FQN | 4.68 | ≤ | 11.2 | --- |
| G) Scram Worth Versus Time | Sce Section 2.3 | | | |

Table 3.14.3

Rod Ejection Accident at

HFP, EOC

| Parameter | Reload Safety Evalnation Values | | Current Safety Analysis | Units |
|---|---|---|---|---|
| A) Moderator Temp. Coefficient | -20.91 | ≤ | 0.0 | pcm/°Fm |
| B) Delayed Neutron Fraction | .00522 | ≥ | .00500 | --- |
| C) Ejected Rod Worth | 0.09 | ≤ | 0.42 | $\%\Delta\rho$ |
| D) Doppler Temp. Coefficient | -1.28 | ≤ | -1.0 | pcm/°Ff |
| E) Prompt Neutron Lifetime | 30.7 | ≥ | 15.0 | $\mu$sec |
| F) FQN | 2.42 | ≤ | 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 Aualysis | Units |
|---|---|---|---|---|
| A) Moderator Temp. Coefficient | -7.02 | ≤ | 0.0 | pcm/°Fm |
| B) Delayed Neutron Fraction | .00522 | ≥ | .00500 | --- |
| C) Ejected Rod Worth | 0.66 | ≤ | 0.92 | $\%\Delta\rho$ |
| D) Doppler Temp. Coefficient | -2.49 | ≤ | -1.0 | pcm/°Ff |
| E) Prompt Neutron Lifetime | 30.7 | ≥ | 15.0 | $\mu$sec |
| F) FQN | 7.33 | ≤ | 13.0 | --- |
| G) Scram Worth Versus Time | Sce Section 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 20 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 20 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 20 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.15.1

Fuel Handling Accident

| Parameter | Reload Safety Evaluation Values | | Current Safety Aualysis |
|---|---|---|---|
| A) FQN | 1.99 | ≤ | 2.53 |

## 3.16 Evaluation of Loss of Coolant Aecident

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 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 20 physics calculation results to the current safety analysis values for the Loss of Coolant Accident.

Since the pertinent parameters from the proposed Cycle 20 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 20 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.16.1

Loss of Coolant Aecident

| Parameter | Reload Safety Evaluatiou Values | | Current Safety Analysis |
|---|---|---|---|
| A) Scram Worth Versus Time | Sec Section 2.3 | | |
| B) FQ | See Section 3.17 | | |
| C) FΔH | 1.52 | ≤ | 1.55 |

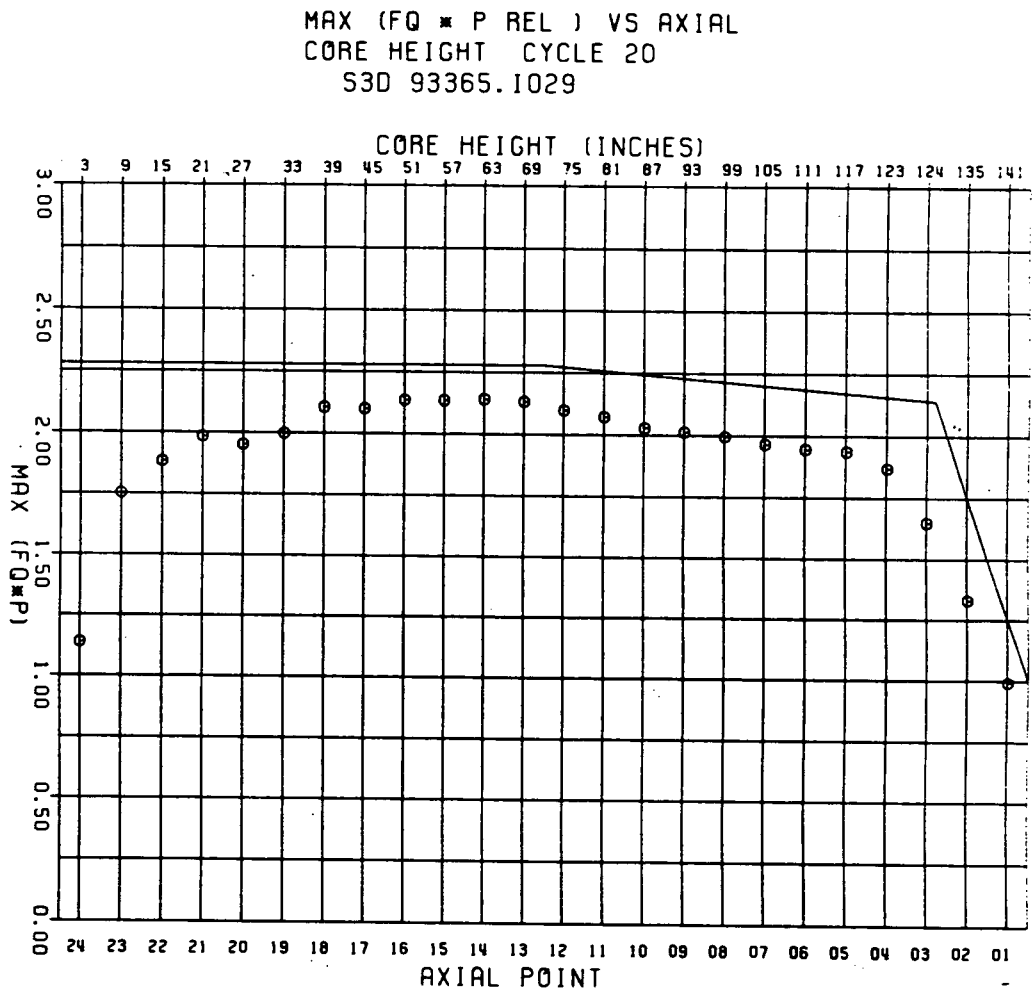
## 3.17 Power Distribution Control Verification

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 20 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 20
S3D 93365.1029
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
MAX (FQ*P)
3.00
2.50
2.00
1.50
1.00
0.50
0.00
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

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

## 5.0 STATISTICS UPDATE

Measurements and calculations of Cycles 16, 17, and 18 are incorporated into the FQN and FΔH statistics data base. The moderator temperature coefficient statistics data base includes results from Cycles 13 through 19. The reliability and bias factors used for the Cycle 20 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 | 4.67% | 0 |
| Rod Worth | 10.0% | 0 |
| Moderator Temperature Coefficient | 2.1 pcm/°F | 2.9 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) | .0579 | 10.13 |
| 2 | .0526 | 9.26 |
| 3 | .0201 | 4.38 |
| 4 | .0220 | 4.63 |
| 5 | .0217 | 4.59 |
| 6 | .0188 | 4.22 |
| 7 | .0194 | 4.30 |
| 8 | .0177 | 4.09 |
| 9 | .0188 | 4.22 |
| 10 | .0167 | 3.97 |
| 11 | .0163 | 3.93 |
| 12 | .0171 | 4.03 |
| 13 | .0172 | 4.04 |
| 14 | .0168 | 3.98 |
| 15 | .0185 | 4.18 |
| 16 | .0183 | 4.16 |
| 17 | .0225 | 4.69 |
| 18 | .0194 | 4.30 |
| 19 | .0269 | 5.29 |
| 20 | .0252 | 5.06 |
| 21 | .0463 | 8.25 |
| 22 | .0368 | 6.75 |
| 23 | .0825 | 14.22 |
| 24 (Top) | .0710 | 12.31 |

## 6.0 REFERENCES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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 Kewaunce," April 1988.

5. Wisconsin Public Service Corporation Technical Specifications for the Kewaunee Nuclear Power Plant. Docket Number 50-305, Amendment No. 104, dated December 9, 1993.

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17. Westinghouse letter PSA-91-233 from D. J. Cesare to S. F. Wozniak dated November 11, 1991.

18. 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".

19. 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".