# KEWAUNEE NUCLEAR POWER PLANT

## RELOAD SAFETY EVALUATION
## KEWAUNEE CYCLE X
## FEBRUARY 1984

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER & LIGHT COMPANY

MADISON GAS & ELECTRIC COMPANY

RELOAD SAFETY EVALUATION

FOR

KEWAUNEE CYCLE X

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## 1.0 INTRODUCTION

The Kewaunee Nuclear Power Plant will be shutdown in March 1984 for the the Cycle 9-10 refueling. Startup of Cycle 10 is forecast for May, 1984.

This report presents an evaluation of the Cycle 10 reload and demonstrates that the core reload will not adversely affect the safety of the plant. Those accidents which could potentially be affected by the reload core design are reviewed.

Details of the calculational model used to generate physics parameters for this Reload Safety Evaluation are described in Reference (1). Accident Evaluation methodologies applied in this report are detailed in Reference (2). These reports have been previously reviewed (3). The current 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. The burnup dependent power peaking limits described in section 2.2 are modified (14) as described in section 4.0 of this report.

It has been concluded that the Cycle 10 design is more conservative than results of previously docketed accident analyses. This conclusion is based on the assumptions that:

1. Cycle 9 operation is terminated after 10,500 (+300, -500) MWD/MTU.
2. There is adherence to plant operating limitations and Technical Specifications (4, 14).

## 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 10 are presented in Figure 2.1.1.

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

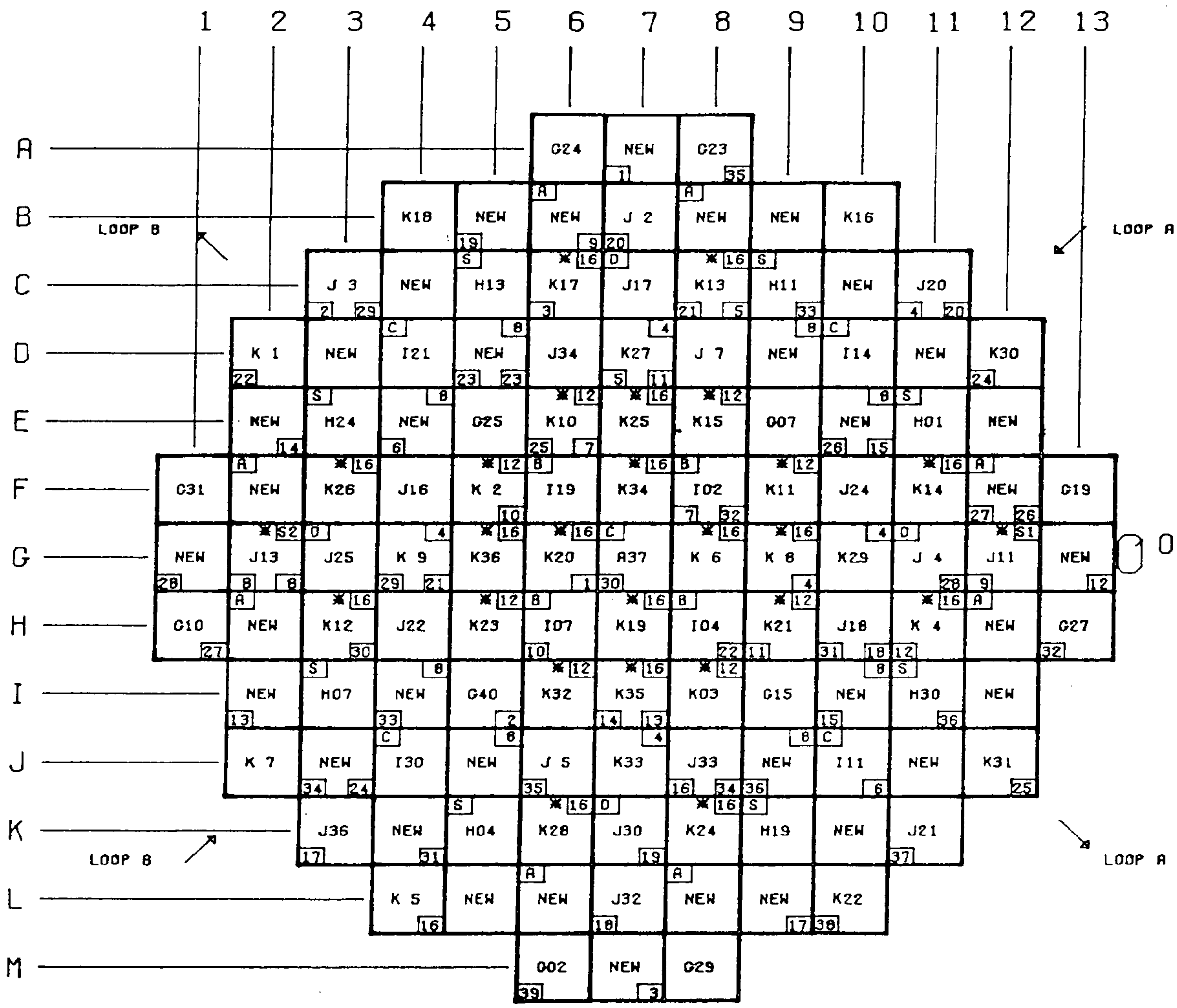
The Cycle 10 reload core will employ 36 burnable poison rod assemblies (BPRA'S) containing 64 fresh and 352 partially depleted pyrex poison rods, and 16 wet annular burnable poison rods.

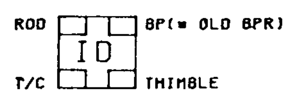
Table 2.1.1

Cycle 10 Fuel Characteristics

| Region | Initial Vendor | Number of Previous W/O U235 | Number of Duty Cycles | Assemblies |
|---|---|---|---|---|
| 1 | W | 2.2 | 1 | 1 |
| 7 | ENC | 3.2 | 3 | 12 |
| 8 | ENC | 3.2 | 2 | 8 |
| 9 | ENC | 3.2 | 2 | 8 |
| 10 | ENC | 3.2 | 2 | 20 |
| 11 | ENC | 3.4 | 1 | 36 |
| 12 | ENC | 3.2 | 0 | 36 (FEED) |

Figure 2.1.1




Kewaunee Cycle 10
Loading Pattern

## 2.2 Design Objectives and Operating Limits

Power Rating 1650 MWTH

System Pressure 2250 PSIA

Core Average Moderator Temperature (HZP) 547 degrees F

Core Average Moderator Temperature (HFP) 561 degrees F

Cycle 10 core design is based on the following design objectives and operating limits.

A. Nuclear peaking factor limits are as follows:

(i) FQ(Z) limits for all Westinghouse Electric Corp. fuel

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

(ii) FQ(Z) limits for Exxon Nuclear Company fuel - Ref. (14)

$FQ(Z) \leq (FQT(Ej)/P) * K(Z)$ for $P > 0.5$
$FQ(Z) \leq 4.42 * K(Z)$ for $P \leq 0.5$

(iii) $F\Delta H$ limits for all fuel

$F\Delta HN \leq 1.55(1 + 0.2(1-P))$ for exposure $\leq$ 24000 MWD/MTU
$F\Delta HN \leq 1.52(1 + 0.2(1-P))$ for exposure $>$ 24000 MWD/MTU

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

K(Z) is the function given in Figure 2.2.1
FQT(Ej) is the function given in Figure 2.2.2 - Ref. (14)
Ej is the fuel rod exposure for which FQ is measured
Z is the core height location FQ

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

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

1.0 % at BOC

2.0 % at EOC

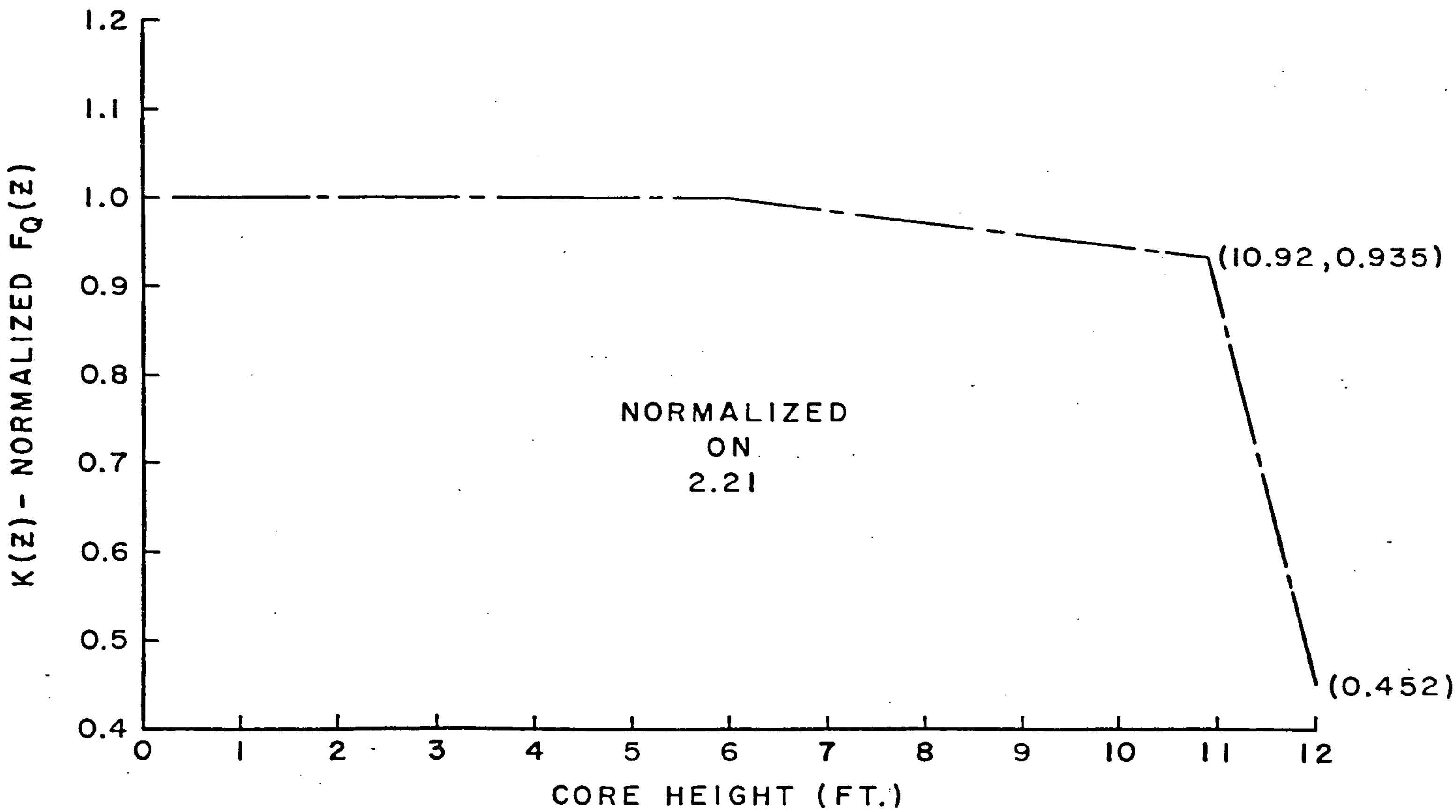
D. The fuel loading pattern shall be capable of generating approximately 10,000 MWD/MTU.

E. The power dependent rod insertion limits (PDIL) are presented in Figure 2.2.3. These limits are those currently specified in Reference 4.

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

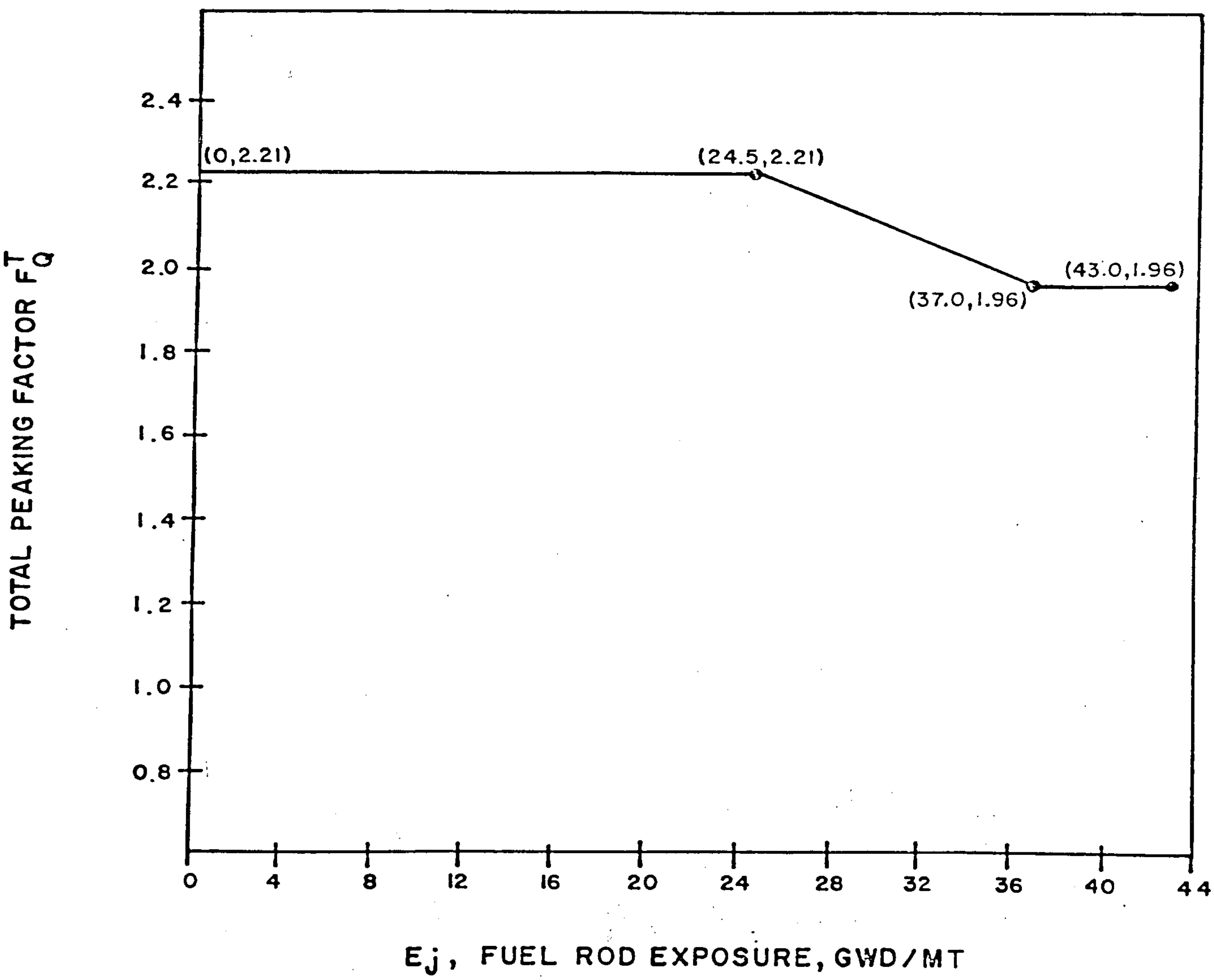
G. A refueling boron concentration of 2100 ppm will be sufficient to maintain the reactor subcritical by 10% $\Delta k/k$ in the cold condition with all rods inserted and will maintain the core subcritical with all rods out of the core.

H. Fuel duty during this fuel cycle will assure peak fuel rod burnups less than those maximum burnups recommended by the respective fuel vendors.




Hot Channel Factor
Normalized Operating Envelope

KEWAUNEE $F_Q^T$ VERSUS ROD EXPOSURE




(Proposed T.S. Amendment 56, Ref. 14)

Figure 2.2.3

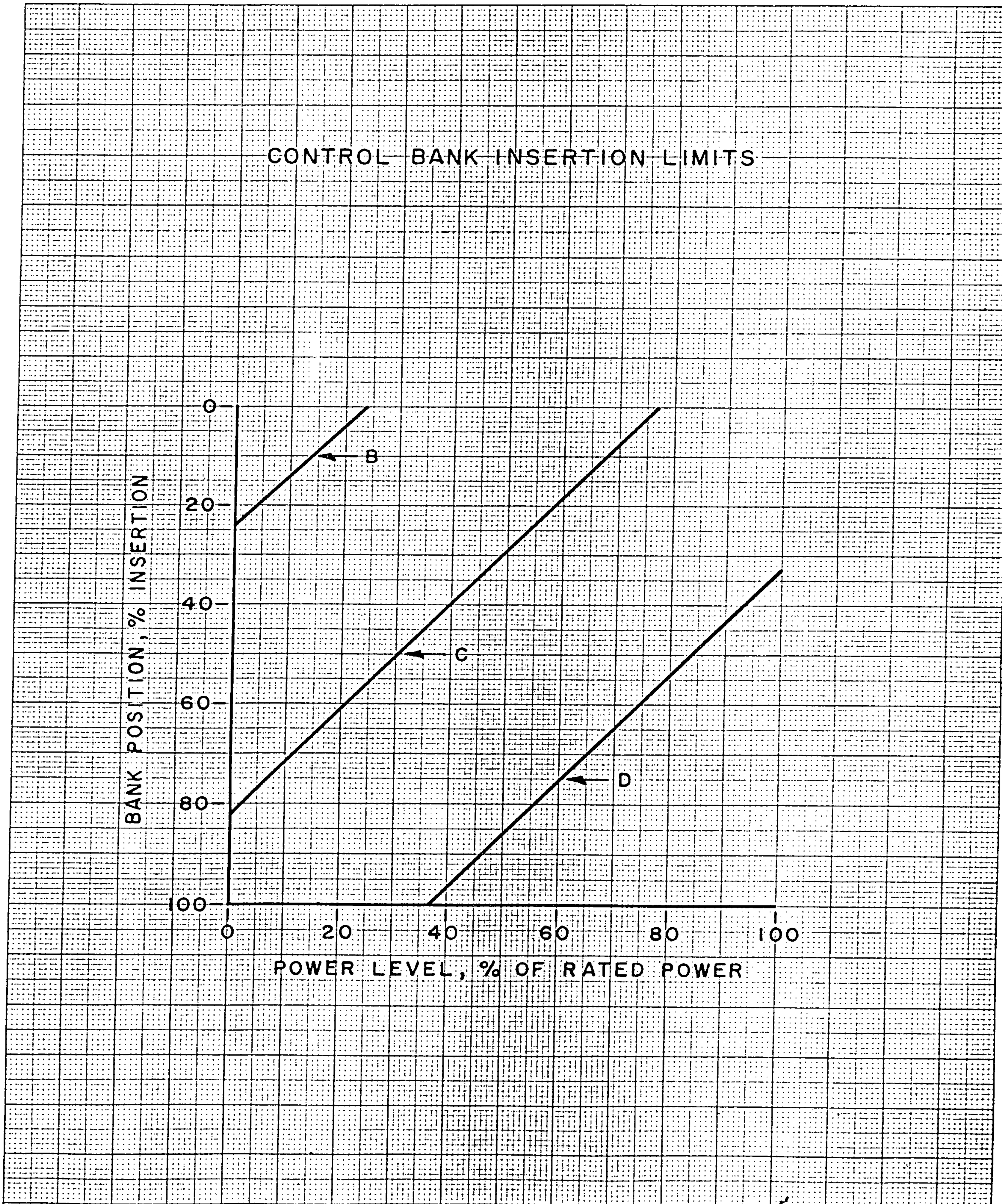
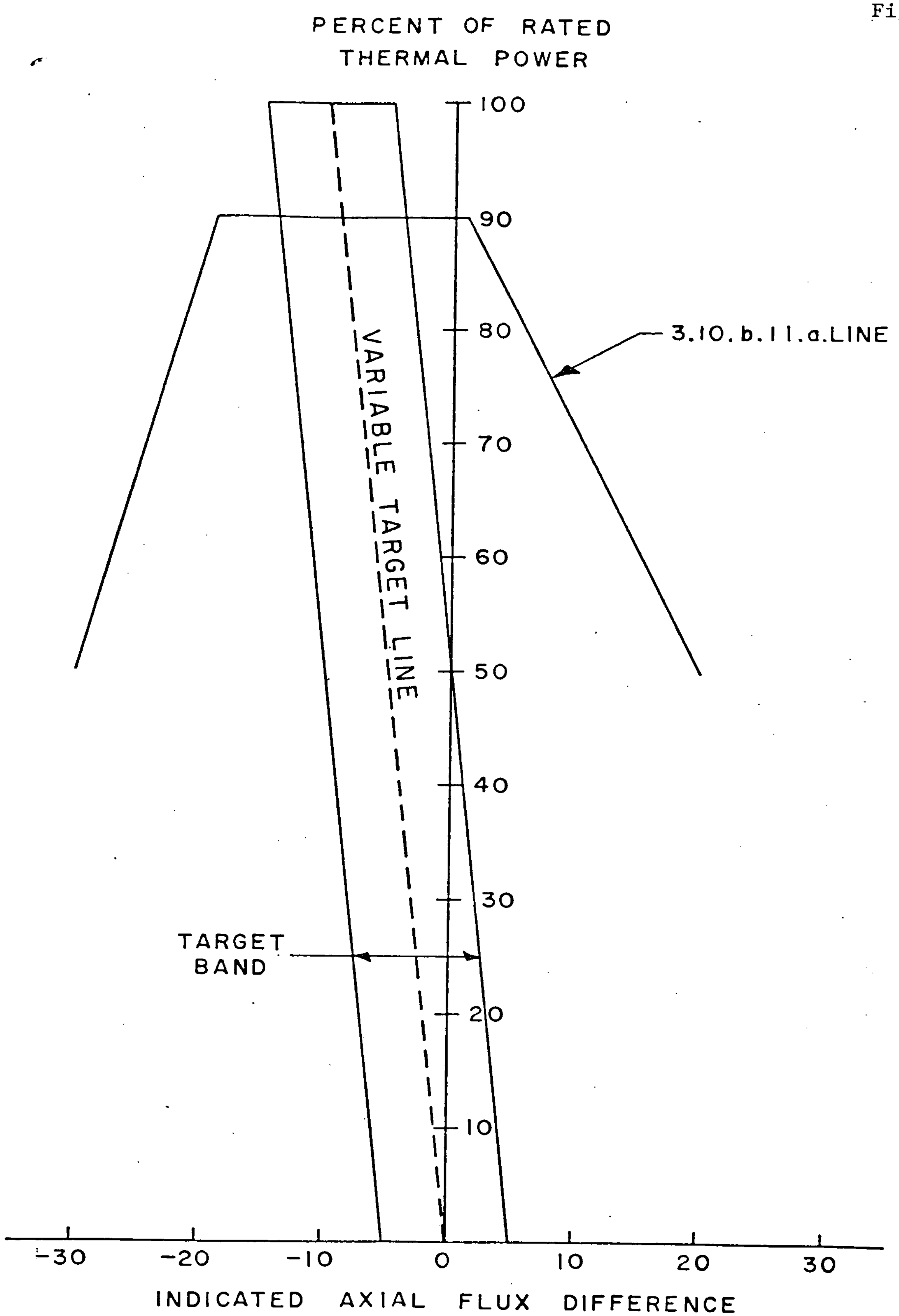

CONTROL BANK INSERTION LIMITS
BANK POSITION, % INSERTION
POWER LEVEL, % OF RATED POWER
0
20
40
60
80
100
0
20
40
60
80
100
B
C
D

Figure 2.2.4




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 10 minimum shutdown margin is 2.06% at end of cycle hot full power conditions.

It is concluded that the minimum trip reactivity insertion rate for Cycle 10 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.

## 2.4 Shutdown Window

An evaluation of the full power equilibrium peaking factor variation at BOC 10 versus EOC 9 burnup is presented in Table 2.4.1. The values have the conservatisms applied in accordance with references 1 and 9.

The EOC 9 shutdown burnup will not significantly affect the Cycle 10 peaking factors if refueling shutdown of Cycle 9 occurs within the burnup window.

Table 2.4.1

Peaking Factor at Actual Beginning of Cycle Burnup

| | $F_{\Delta H}$ | | $F_Q$ | |
|---|---|---|---|---|
| | Cycle 10 | Limit | Cycle 10 | Limit |
| BOC 10 (+300 EOC9) | 1.534 | 1.55 | 2.145 | 2.21 |
| BOC 10 (Nominal EOC9) | 1.529 | 1.55 | 2.138 | 2.21 |
| BOC 10 (-500 EOC 9) | 1.519 | 1.55 | 2.125 | 2.21 |

## 3.0 ACCIDENT EVALUATIONS

Table 3.0.1 presents the latest safety analyses performed for the accidents which are evaluated in Sections 3.1 through 3.16 of this report. The bounding values derived from these analyses are shown in Table 3.0.2 and will be applied in the Cycle 10 accident evaluations.

Table 3.0.1

Kewaunee Nuclear Power Plant

List of Safety Analyses

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

Table 3.0.2

Safety Analyses Bounding Values

| Parameter | Lower Bound | Upper Bound | Units |
|---|---|---|---|
| Moderator Temperature Coefficient | -40.0 | 0.0 | pcm/°F |
| Doppler Coefficient | -2.32 | -1.0 | pcm/°F |
| Differential Boron Worth | -11.2 | N/A | pcm/ppm |
| Delayed Neutron Fraction | .0050 | .0071 | |
| Prompt Neutron Lifetime | 20 | N/A | μ sec |
| Shutdown Margin | 1.0 | 2.0 | %Δρ |
| Differential Rod Worth of 2 Banks Moving | N/A | 82 | pcm/sec |
| Ejected Rod Cases | | | |
| HFP, BOL | | | |
| βeff | .0055 | N/A | |
| Rod Worth | N/A | .30 | %Δρ |
| FQ | N/A | 5.03 | |
| HFP, EOL | | | |
| βeff | .0050 | N/A | |
| Rod Worth | N/A | .42 | %Δρ |
| FQ | N/A | 5.1 | |
| HZP, BOL | | | |
| βeff | .0055 | N/A | |
| Rod Worth | N/A | .92 | %Δρ |
| FQ | N/A | 13.0 | |
| HZP, EOL | | | |
| βeff | .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 10 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 10 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 10 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.1.1

Uncontrolled Rod Withdrawal From Subcritical

| Parameter | Reload Safety Evaluation Values | | Current Safety Analysis | Units |
|---|---|---|---|---|
| A) Moderator Temp. Coefficient | 2.1 | ≤ | 10.0 | pcm/°Fm |
| B) Doppler Temp. Coefficient | -1.8 | ≤ | -1.0 | pcm/°Ff |
| C) Differential Worth of Two Moving Banks | 27.7 | ≤ | 82.0 | pcm/sec |
| D) Scram Worth vs. Time | See Section 2.3 | | | |
| E) Delayed Neutron Fraction | .00633 | ≤ | .0071 | |

## 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 10 physics parameters to the current safety analysis values for the Uncontrolled Rod Withdrawal at Power Accident.

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

Since the pertinent parameters from the proposed Cycle 10 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 10 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.2.1

Uncontrolled Rod Withdrawal at Power

| | Parameter | Reload Safety Evaluation Values | | Current Safety Analysis | Units |
|---|---|---|---|---|---|
| A) | Moderator Temp. Coefficient | 2.1* | ≤ | 0.0 | pcm/°Fm |
| B) | Doppler Temp. Coefficient | -1.30 | ≤ | -1.0 | pcm/°Ff |
| C) | Differential Rod Worth Of Two Moving Banks | 27.7 | ≤ | 82.0 | pcm/sec |
| D) | FΔHN | 1.53 | ≤ | 1.55 | |
| E) | Scram Worth vs. Time | See Section 2.3 | | | |
| F) | Delayed Neutron Fraction | .00633 | ≤ | 0.0071 | |

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

### 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 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 10 FΔHN versus the current safety analysis FΔHN limit for the Misaligned Rod Accident.

Since the pertinent parameter from the proposed Cycle 10 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 10 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.3.1

Control Rod Misalignment Accident

| Parameter | Reload Safety Evaluation Value | | Current Safety Analysis |
|---|---|---|---|
| A) $F_{\Delta H}^{N}$ | 1.90 | ≤ | 1.92 |

## 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 10 FΔHN to the current safety analysis FΔHN limit for the Dropped Rod Accident is presented in Table 3.4.1.

Since the pertinent parameter from the proposed Cycle 10 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 10 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.4.1

Dropped Rod Accident

| Parameter | Reload Safety Evaluation Value | | Current Safety Analysis |
|---|---|---|---|
| A) $F \Delta h N$ | 1.71 | ≤ | 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 10 physics analysis results to the current safety analysis values for the Uncontrolled Boron Dilution Accident for refueling and full power core conditions.

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

Since the pertinent parameters from the proposed Cycle 10 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 10 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.5.1

Uncontrolled Boron Dilution Accident

| Parameter | Reload Safety Evaluation Values | | Current Safety Analysis | Units |
|---|---|---|---|---|
| i) Refueling Conditions | | | | |
| A) Shutdown Margin (ARI) | 12.0 | ≥ | 10.0 | %Δρ |
| ii) At-Power Conditions | | | | |
| A) Moderator Temp. Coefficient | 2.1* | ≤ | 0.0 | pcm/°Fm |
| B) Doppler Temp. Coefficient | -1.3 | ≤ | -1.0 | pcm/°Ff |
| C) Reactivity Insertion Rate by Boron | 1.39 | ≤ | 1.60 | pcm/sec |
| D) Shutdown Margin | 2.06 | ≥ | 1.0 | %Δρ |
| E) FΔHN | 1.53 | ≤ | 1.55 | |

* Moderator Temperature Coefficient will be verified negative at startup testing.

## 3.6 Evaluation of Startup of an Inactive Loop

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 Cycle 10 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 10 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 10 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.6.1

Startup of an Inactive Loop Accident

| Parameter | Reload Safety Evaluation Values | | Current Safety Analysis | Units |
|---|---|---|---|---|
| A) Moderator Temp. Coefficient | -36.8 | ≥ | -40.0 | pcm/°Fm |
| B) Doppler Coefficient | -1.7 | ≤ | -1.0 | pcm/°Ff |
| C) FΔHN | 1.53 | ≤ | 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 BOL and EOL conditions. The doppler reactivity coefficient is chosen at a minimum (absolute) value to maximize the nuclear power peak.

A comparison of Cycle 10 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 10 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 10 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.7.1

Feedwater System Malfunction Accident

| | Parameter | Reload Safety Evaluation Values | | Current Safety Analysis | Units |
|---|---|---|---|---|---|
| A) | Moderator Temp. Coefficient | -2.2 | ≤ | 0.0 | pcm/°Fm |
| B) | Doppler Temp. Coefficient | -1.3 | ≤ | -1.0 | pcm/°Ff |
| C) | FΔHN | 1.53 | ≤ | 1.55 | |
| D) | Moderator Temp. Coefficient (maximum) | -32.4 | ≥ | -40.0 | pcm/°Fm |

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

Since the pertinent parameters from the proposed Cycle 10 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 10 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.8.1

Excessive Load Increase Accident

| Parameter | Reload Safety Evaluation Values | | Current Safety Analysis | Units |
|---|---|---|---|---|
| A) Moderator Temp. Coefficient (minimum) | -2.2 | ≤ | 0.0 | pcm/°Fm |
| B) Moderator Temp. Coefficient (maximum) | -32.4 | ≥ | -40.0 | pcm/°Fm |
| C) Doppler Temp. Coefficient | -1.3 | ≤ | -1.0 | pcm/°Ff |
| D) FΔHN | 1.53 | ≤ | 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.

A minimum moderator temperature coefficient maximizes the power transient and heatup prior to reactor trip while the large (negative) doppler coefficient retards the power coast down following reactor trip. The power distribution (FΔH) and scram reactivity are evaluated to ensure thermal margins are maintained by the reactor protection system.

A comparison of Cycle 10 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 10 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 10 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.9.1

Loss of Load Accident

| Parameter | Reload Safety Evaluation Values | | Current Safety Analysis | Units |
|---|---|---|---|---|
| A) Moderator Temp. Coefficient | -2.2 | ≤ | 0.0 | pcm/°Fm |
| B) Doppler Temp. Coefficient | -1.8 | ≥ | -2.32 | pcm/°Ff |
| C) $F_{\Delta H}^N$ | 1.53 | ≤ | 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 to both 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 flux 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 10 calculational physics parameters to the current safety analysis values for the Loss of Reactor Coolant Flow Due to Pump Trip Accident.

Since the pertinent parameters from the proposed Cycle 10 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 10 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.11.1

Loss of Reactor Coolant Flow Due to Pump Trip

| | Parameter | Reload Safety Evaluation Values | | Current Safety Analysis | Units |
|---|---|---|---|---|---|
| A) | Moderator Temp. Coefficient | -2.2 | ≤ | 0.0 | pcm/°Fm |
| B) | Doppler Temp. Coefficient | -1.8 | ≥ | -2.32 | pcm/°Ff |
| C) | FΔHN | 1.53 | ≤ | 1.55 | |
| D) | Scram Worth Versus Time | See Section 2.3 | | | |

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

Since the pertinent parameters from the proposed Cycle 10 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 10 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.12.1

Loss of Reactor Coolant Flow Due to Locked Rotor

| Parameter | Reload Safety Evaluation Values | | Current Safety Analysis | Units |
|---|---|---|---|---|
| A) Moderator Temp. Coefficient | -2.2 | ≤ | 0.0 | pcm/°Fm |
| B) Doppler Temp. Coefficient | -1.8 | ≥ | -2.32 | pcm/°Ff |
| C) Delayed Neutron Fraction | 0.00633 | ≤ | 0.0071 | |
| D) Percent Pins > Limiting FΔHN (DNBR=1.3) | 24.4 | ≤ | 40.0 | % |
| E) Scram Worth Versus Time | See Section 2.3 | | | |

### 3.13 Evaluation of Main Steam Line Rupture

The rupture 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.

Shutdown margin and reactivity insertion from the cooldown are evaluated against those used in the accident analysis.

The minimum Cycle 10 shutdown margin is compared to that assumed in the safety analysis in Table 3.13.1. Figure 3.13.1 compares the Cycle 10 keff versus moderator temperature at 1000 psia to the current safety analysis limiting cooldown reactivity curve.

Since the pertinent parameters from the proposed Cycle 10 reload core are conservatively bounded by those used in the current safety analysis, a main steam line rupture accident will be less severe than the transient in the current analysis. The implementation of the Cycle 10 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

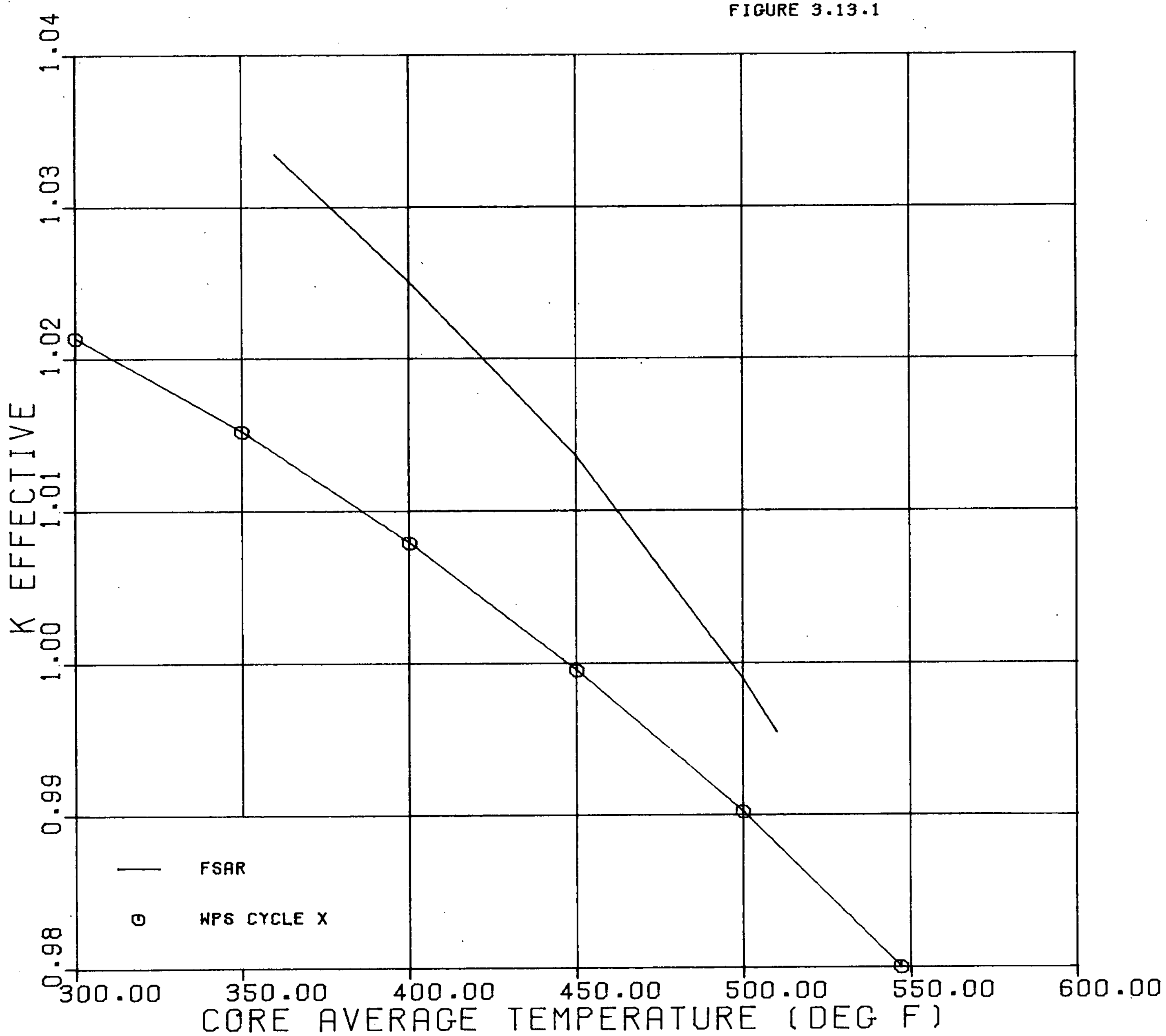
Table 3.13.1

Main Steam Line Rupture Accident

| Parameter | Reload Safety Evaluation Value | | Current Safety Analysis | Unit |
|---|---|---|---|---|
| A) Shutdown Margin | 2.06 | ≥ | 2.0 | %$\Delta\rho$ |

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

FIGURE 3.13.1



## 3.14 Evaluation of Rod Ejection Accidents

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

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

Since the pertinent parameters from the proposed Cycle 10 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 10 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.14.1

Rod Ejection Accidents

HFP, EOL

| | Parameter | Reload Safety Evaluation Values | | Current Safety Analysis | Units |
|---|---|---|---|---|---|
| A) | Moderator Temp. Coefficient | -2.2 | ≤ | 0.0 | pcm/°Fm |
| B) | Delayed Neutron Fraction | 0.0063 | ≥ | 0.0055 | |
| C) | Ejected Rod Worth | 0.09 | ≤ | 0.30 | %Δρ |
| D) | Doppler Temp. Coefficient | -1.3 | ≤ | -1.0 | pcm/°Ff |
| E) | Prompt Neutron Lifetime | 29.3 | ≥ | 20.0 | μsec |
| F) | FQN | 2.39 | ≤ | 5.03 | |
| G) | Scram Worth Versus Time | See Section 2.3 | | | |

Table 3.14.2

Rod Ejection Accidents

HZP, BOL

| | Parameter | Reload Safety Evaluation Values | | Current Safety Analysis | Units |
|---|---|---|---|---|---|
| A) | Moderator Temp. Coefficient | 2.1* | ≤ | 0.0 | pcm/°Fm |
| B) | Delayed Neutron Fraction | 0.0063 | ≥ | 0.0055 | |
| C) | Ejected Rod Worth | 0.53 | ≤ | 0.91 | %Δρ |
| D) | Doppler Temp. Coefficient | -1.3 | ≤ | -1.0 | pcm/°Ff |
| E) | Prompt Neutron Lifetime | 29.3 | ≥ | 20.0 | μsec |
| F) | FQN | 5.55 | ≤ | 11.2 | |
| G) | Scram Worth Versus Time | See Section 2.3 | | | |

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

Table 3.14.3

Rod Ejection Accidents

HFP, EOL

| | Parameter | Reload Safety Evaluation Values | | Current Safety Analysis | Units |
|---|---|---|---|---|---|
| A) | Moderator Temp. Coefficient | -13.7 | ≤ | 0.0 | pcm/°Fm |
| B) | Delayed Neutron Fraction | 0.0053 | ≥ | 0.0050 | |
| C) | Ejected Rod Worth | 0.12 | ≤ | 0.42 | %Δρ |
| D) | Doppler Temp. Coefficient | -1.48 | ≤ | -1.0 | pcm/°Ff |
| E) | Prompt Neutron Lifetime | 32.1 | ≥ | 20.0 | μsec |
| F) | FQN | 2.81 | ≤ | 5.1 | |
| G) | Scram Worth Versus Time | See Section 2.3 | | | |

Table 3.14.4

Rod Ejection Accidents

HZP, EOL

| | Parameter | Reload Safety Evaluation Values | | Current Safety Analysis | Units |
|---|---|---|---|---|---|
| A) | Moderator Temp. Coefficient | -8.6 | ≤ | 0.0 | pcm/°Fm |
| B) | Delayed Neutron Fraction | 0.0053 | ≥ | 0.0050 | |
| C) | Ejected Rod Worth | 0.63 | ≤ | 0.92 | %Δρ |
| D) | Doppler Temp. Coefficient | -1.48 | ≤ | -1.0 | pcm/°Ff |
| E) | Prompt Neutron Lifetime | 32.1 | ≥ | 20.0 | μsec |
| F) | FQN | 7.24 | ≤ | 13.0 | |
| E) | Scram Worth Versus Time | See 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 Cycle 10 FQN, calculated at end of Cycle 10 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 10 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 10 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.15.1

Fuel Handling Accident

| Parameter | Reload Safety Evaluation Values | | Current Safety Analysis |
|---|---|---|---|
| A) FQN | 1.96 | ≤ | 2.53 |

### 3.16 Evaluation of Loss of Coolant Accident

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

Since the pertinent parameters from the proposed Cycle 10 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 10 reload core design, therefore, will not adversely affect the safe operation of the Kewaunee Plant.

Table 3.16.1

Loss of Coolant Accident

| Parameter | Reload Safety Evaluation Values | Current Safety Analysis |
|---|---|---|
| A) Scram Worth Versus Time | See Section 2.3 | |
| B) FQ | See Section 3.17 | |

## 3.17 Power Distribution Control Verification

The total peaking factor FQT relates the maximum local power density to the core average power density. The FQT is determined by both the radial and axial power distributions. The radial power distribution is relatively fixed by the core loading pattern design. The axial power distribution is controlled by the procedures defined in Section 2.2 of this report (9).

Following these procedures, FQT(Z) are determined by calculations performed at full power, equilibrium core conditions, at exposures ranging from EOC 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 10 operation will not preclude full power operation under the power distribution control specifications currently applied (10).

MAX (FQ * P REL ) VS AXIAL
CORE HEIGHT CYCLE 10
S3D 83346.1029

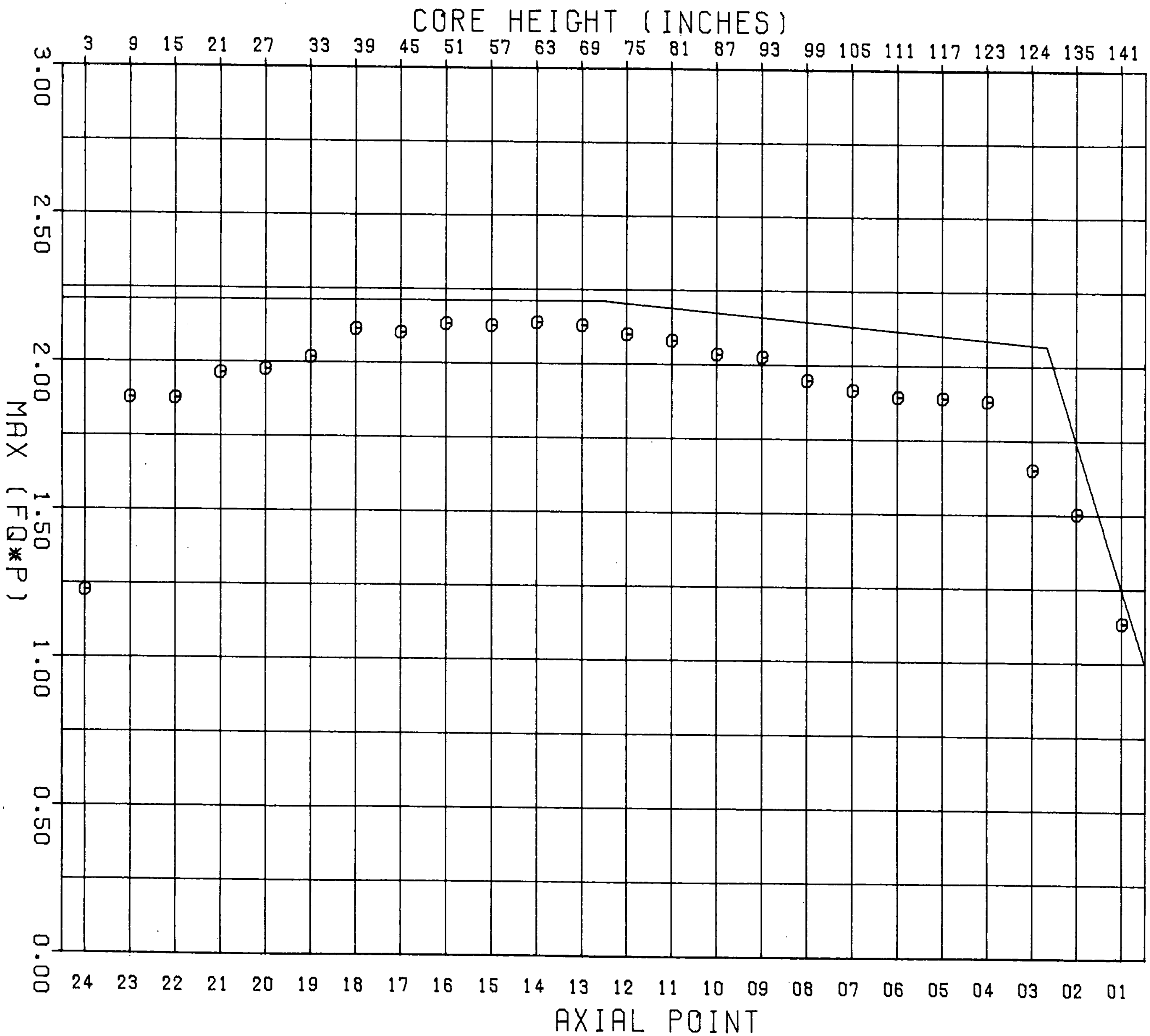



Figure 3.17.1

## 4.0 TECHNICAL SPECIFICATIONS

Figure 3.10-6 of the Kewaunee Nuclear Power Plant Technical Specifications specifies the total peaking factor for fuel manufactured by Exxon Nuclear Corporation as a function of fuel rod exposure to 37.0 GWD/MT. Some fuel rods manufactured by Exxon Nuclear Company will exceed 37.0 GWD/MT during Cycle 10.

Proposed Amendment 56 to the Kewaunee Nuclear Power Plant Technical Specifications, submitted to the NRC on December 14, 1983 provided the justification for extending the exposure function to 43.0 GWD/MT. This change has been previously reviewed and accepted for other facilities by the NRC. Approval is expected prior to the time that exposures on Exxon fuel exceed 37.0 GWD/MT.

No other revisions or additions to the Kewaunee Nuclear Power Plant Technical Specifications are required due to the implementation of the Cycle 10 reload design.

## 5.0 STATISTICS UPDATE

In an effort to provide continuing assurance of the model applicability, Cycle 8 measurements and calculations were added to the statistics data base prior to model applications to the Cycle 10 Reload Analysis. The reliability and bias factors applicable to Cycle 10 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 | 4.3% | 0 |
| Rod Worth | 10.0% | 0 |
| Moderator Temperature Coefficient | 7.43 PCM/°F | -2.13 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 | Node | RF (%) |
|---|---|---|
| 1 (Bottom) | 0.091 | 15.57 |
| 2 | 0.044 | 7.82 |
| 3 | 0.034 | 6.33 |
| 4 | 0.028 | 5.48 |
| 5 | 0.034 | 6.37 |
| 6 | 0.038 | 6.96 |
| 7 | 0.038 | 7.00 |
| 8 | 0.037 | 6.85 |
| 9 | 0.032 | 6.05 |
| 10 | 0.031 | 5.95 |
| 11 | 0.030 | 5.74 |
| 12 | 0.029 | 5.63 |
| 13 | 0.024 | 4.87 |
| 14 | 0.025 | 5.09 |
| 15 | 0.023 | 4.80 |
| 16 | 0.023 | 4.80 |
| 17 | 0.021 | 4.50 |
| 18 | 0.021 | 4.53 |
| 19 | 0.025 | 4.98 |
| 20 | 0.023 | 4.79 |
| 21 | 0.042 | 7.52 |
| 22 | 0.035 | 6.46 |
| 23 | 0.087 | 15.02 |
| 24 (Top) | 0.077 | 13.36 |

## 6.0 REFERENCES


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

2. Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, topical report entitled, "Reload Safety Evaluation Methods for Application to Kewaunee."

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

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

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

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

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

8. WCAP 8093, "Fuel Densification Kewaunee Nuclear Power Plant," March 1973.

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

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

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

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

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

14. Proposed Amendment 56 to the KNPP Technical Specifications. Letter from C.W. Giesler to H.R. Denton, December 14, 1983.