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U. S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

Donald C. Cook Nuclear Plant Unit 1
CORE OPERATING LIMITS REPORT

Indiana Michigan Power Company, the licensee for Donald C. Cook Nuclear Plant Unit 1, is submitting the Core Operating Limits Report (COLR) for Unit 1 Cycle 25 in accordance with Technical Specification 5.6.5. Revision 0 of the Unit 1 Cycle 25 COLR is provided as an enclosure to this letter.

There are no new or revised commitments in this letter. Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Manager, at (269) 466-2649.

Sincerely,

Joel P. Gebbie
Site Vice President

DMB/kmh

Enclosure:

Donald C. Cook Nuclear Plant Unit 1 Cycle 25 Core Operating Limits Report, Revision 0

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ENCLOSURE TO AEP-NRC-2013-44

Donald C. Cook Nuclear Plant Unit 1 Cycle 25

Core Operating Limits Report
Revision 0

Donald C. Cook Nuclear Plant
Unit 1 Cycle 25
Core Operating Limits Report

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Donald C. Cook Nuclear Plant Unit 1 Cycle 25 design has been prepared in accordance with the requirements of Technical Specification 5.6.5.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

- a. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, July 1985
- b. WCAP-8385, Power Distribution Control and Load Following Procedures – Topical Report, September 1974
- c. WCAP-10216-P-A, Rev. 1A, Relaxation of Constant Axial Offset Control/ F_Q Surveillance Technical Specification, February 1994
- d. Plant-specific adaptation of WCAP-16009-P-A, Revision 1, Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM), as approved by NRC Safety Evaluation dated October 17, 2008.
- e. WCAP-12610-P-A, VANTAGE+ Fuel Assembly Reference Core Report, April 1995
- f. WCAP-8745-P-A, Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions, September 1986
- g. WCAP-13749-P-A, Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement, March 1997
- h. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, Optimized ZIRLO™, July 2006

The Technical Specifications affected by this report are listed below:

- | | |
|-------|--|
| 2.1.1 | Reactor Core Safety Limits |
| 3.1.1 | SHUTDOWN MARGIN (SDM) |
| 3.1.3 | Moderator Temperature Coefficient (MTC) |
| 3.1.5 | Shutdown Bank Insertion Limits |
| 3.1.6 | Control Bank Insertion Limits |
| 3.2.1 | Heat Flux Hot Channel Factor ($F_Q(Z)$) |
| 3.2.2 | Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$) |
| 3.2.3 | AXIAL FLUX DIFFERENCE (AFD) |
| 3.3.1 | Reactor Trip System (RTS) Instrumentation |
| 3.4.1 | RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits |
| 3.9.1 | Boron Concentration |

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using the NRC-approved methodologies specified in Technical Specification 5.6.5.

2.1 SAFETY LIMITS

2.1.1 Reactor Core Safety Limits (Specification 2.1.1)

In Modes 1 and 2, the combination of thermal power, pressurizer pressure, and the highest loop average temperature (T_{avg}) shall not exceed the limits as shown in Figure 6 for 4 loop operation.

2.2 REACTIVITY CONTROL

2.2.1 SHUTDOWN MARGIN (SDM) (Specification 3.1.1)

Shutdown margin shall be greater than or equal to 1.3% $\Delta k/k$ for $T_{avg} > 200^\circ\text{F}$

Shutdown margin shall be greater than or equal to 1.0% $\Delta k/k$ for $T_{avg} \leq 200^\circ\text{F}$

2.2.2 Moderator Temperature Coefficient (MTC) (Specification 3.1.3)

- a. The Moderator Temperature Coefficient (MTC) limits are:

The BOL/ARO-MTC shall be less positive or equal to the value given in Figure 1.

The EOL/ARO/RTP-MTC shall be less negative or equal to $-4.54\text{E-}4 \Delta k/k/^\circ\text{F}$.

This limit is based on a T_{avg} program with HFP vessel T_{avg} of 554.0 to 558.0 $^\circ\text{F}$.

Where: ARO stands for All Rods Out
BOL stands for Beginning of Cycle Life
EOL stands for End of Cycle Life
RTP stands for Rated Thermal Power
HFP stands for Hot Full Thermal Power

- b. The MTC Surveillance limit is:
The 300 ppm/ARO/RTP-MTC should be less negative or equal to $-3.84\text{E-}4 \Delta\text{k/k/}^\circ\text{F}$ at a HFP vessel T_{avg} of 554.0 to 558.0 °F.
- c. The Revised Predicted near-EOL 300 ppm MTC shall be calculated using Figure 7 and the following algorithm:

Revised Predicted MTC = Predicted MTC + AFD Correction + Predicted Correction*

* Predicted Correction is $-0.30\text{E-}4 \Delta\text{k/k/}^\circ\text{F}$.

If the Revised Predicted MTC is less negative than the SR 3.1.3.2 limit (COLR 2.2.2.b) and all of the benchmark data contained in the surveillance procedure are met, then a MTC measurement in accordance with SR 3.1.3.2 is not required.

- d. The MTC Surveillance limit is:
The 60 ppm/ARO/RTP-MTC should be less negative or equal to $-4.41\text{E-}4 \Delta\text{k/k/}^\circ\text{F}$ at a HFP vessel T_{avg} of 554.0 to 558.0 °F

2.2.3 Shutdown Bank Insertion Limits (Specification 3.1.5)

The shutdown rods shall be withdrawn to at least 228 steps.

2.2.4 Control Bank Insertion Limits (Specifications 3.1.6)

- a. The control rod banks shall be limited in physical insertion as shown in Figure 2.
- b. Successive Control Banks shall overlap by 100 steps. The sequence for Control Bank withdrawal shall be Control Bank A, Control Bank B, Control Bank C and Control Bank D.

2.3 POWER DISTRIBUTION LIMITS

2.3.1 AXIAL FLUX DIFFERENCE (AFD) (Specification 3.2.3)

- a. The Allowable Operation Limits are provided in Figure 3.
- b. The AFD target band is $\pm 5\%$ for a cycle average accumulated burnup ≥ 0.0 MWD/MTU.

2.3.2 Heat Flux Hot Channel Factor ($F_Q(Z)$) (Specification 3.2.1)

$$F_Q^C(Z) \leq \frac{CF_Q}{P} * K(Z) \quad \text{for } P > 0.5$$

$$F_Q^C(Z) \leq 2 * CF_Q * K(Z) \quad \text{for } P \leq 0.5$$

$$F_Q^W(Z) \leq \frac{CF_Q}{P} * K(Z) \quad \text{for } P > 0.5$$

$$F_Q^W(Z) \leq 2 * CF_Q * K(Z) \quad \text{for } P \leq 0.5$$

Where: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

- a. $CF_Q = 2.15$
- b. $K(Z)$ is provided in Figure 4.
- c. $F_Q^C(Z)$ is the measured hot channel factor including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.
- d. $W(Z)$ is provided in Table 1 for $\pm 5\%$ AFD target band.
- e. $F_Q^W(Z) = F_Q^C(Z) \times W(Z) \times F_P$

The $W(z)$ values are generated assuming that they will be used for a full power surveillance. When a part power surveillance is performed, the $W(z)$ values should be multiplied by the factor $1/P$, when P is > 0.5 . When P is ≤ 0.5 , the $W(z)$ values should be multiplied by the factor $1/(0.5)$, or 2.0. This is consistent with the adjustment in the $F_Q(z)$ limit at part power conditions.
- f. For Cycle 25, $F_P = 1.02$ for all burnups associated with Note 2a of SR 3.2.1.2. When no penalty is required, $F_P = 1.00$.

2.3.3 Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$) (Specification 3.2.2)

$$F_{\Delta H}^N \leq CF_{\Delta H} * (1 + PF_{\Delta H} * (1-P))$$

Where: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

- a. $CF_{\Delta H} = 1.545$
- b. $PF_{\Delta H} = 0.3$
- c. $F_{\Delta H}^N$ is the measured Enthalpy Rise Hot Channel Factor including a 4% measurement uncertainty.

2.4 INSTRUMENTATION

2.4.1 Reactor Trip System (RTS) Instrumentation (Specification 3.3.1)

The Overtemperature ΔT and Overpower ΔT setpoints are as shown in Figure 5.

2.5 REACTOR COOLANT SYSTEM

2.5.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits (Specification 3.4.1)

- a. Pressurizer Pressure shall be ≥ 2018 psig⁺
- b. Reactor Coolant System T_{AVG} shall be $\leq 580.5^\circ\text{F}$ ⁺
- c. Reactor Coolant System Total Flow Rate shall be $\geq 362,900$ gpm

2.6 REFUELING OPERATIONS

2.6.1 Boron Concentration (Specification 3.9.1)

The boron concentration of all filled portions of the Reactor Coolant System, the refueling canal and the refueling cavity shall be greater than or equal to 2400 ppm⁺⁺.

⁺ These are Safety Analysis values. With readability allowance, the corresponding values are 578.2°F for T_{avg} , and 2050 psig for Pressurizer Pressure.

⁺⁺ This concentration bounds the condition of $K_{eff} \leq 0.95$ which includes a 1% $\Delta k/k$ conservative allowance for uncertainties. The boron concentration of 2400 ppm includes a 50 ppm conservative allowance for uncertainties.

FIGURE 1

MODERATOR TEMPERATURE COEFFICIENT (MTC) LIMITS

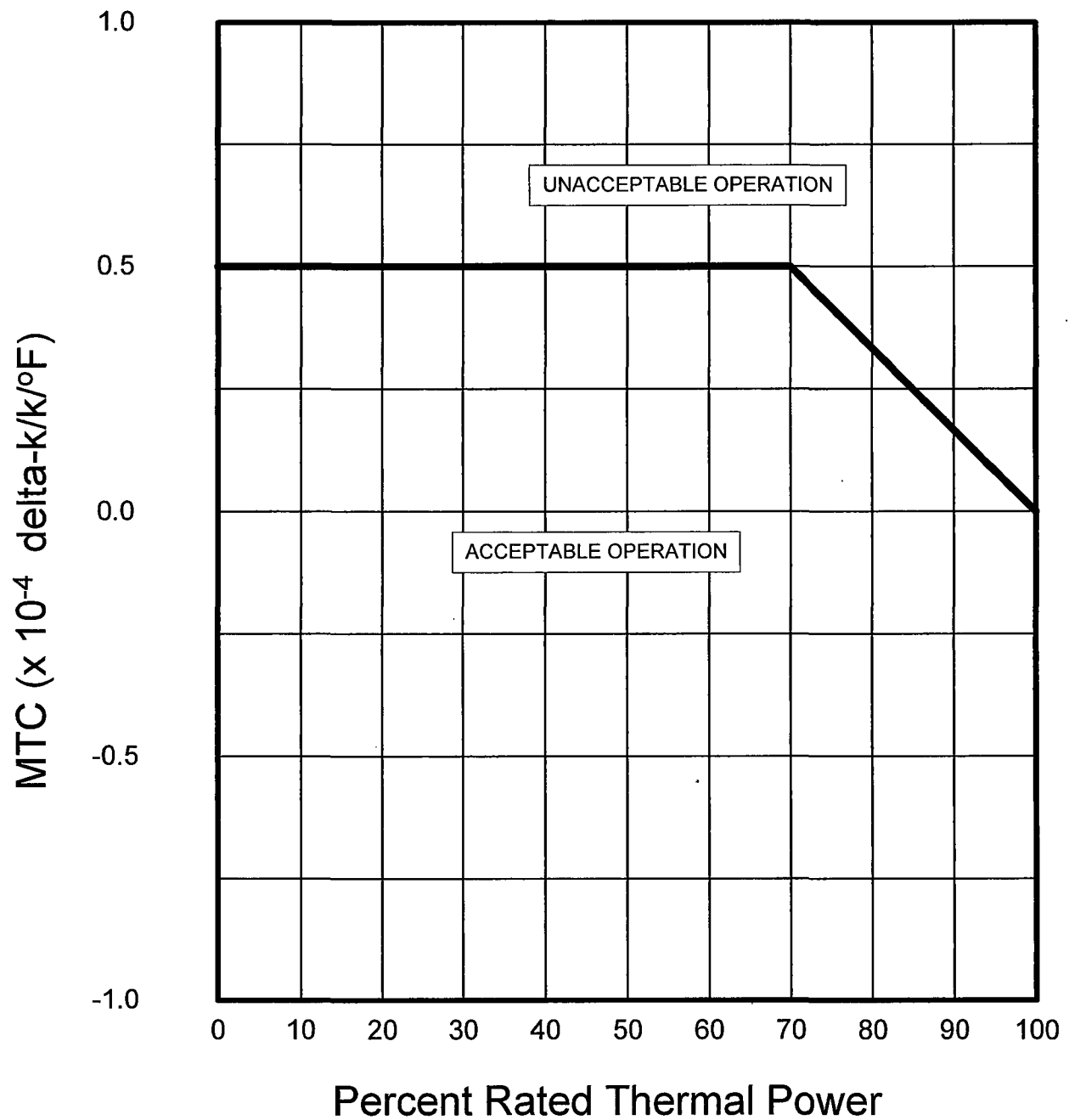


FIGURE 2

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER

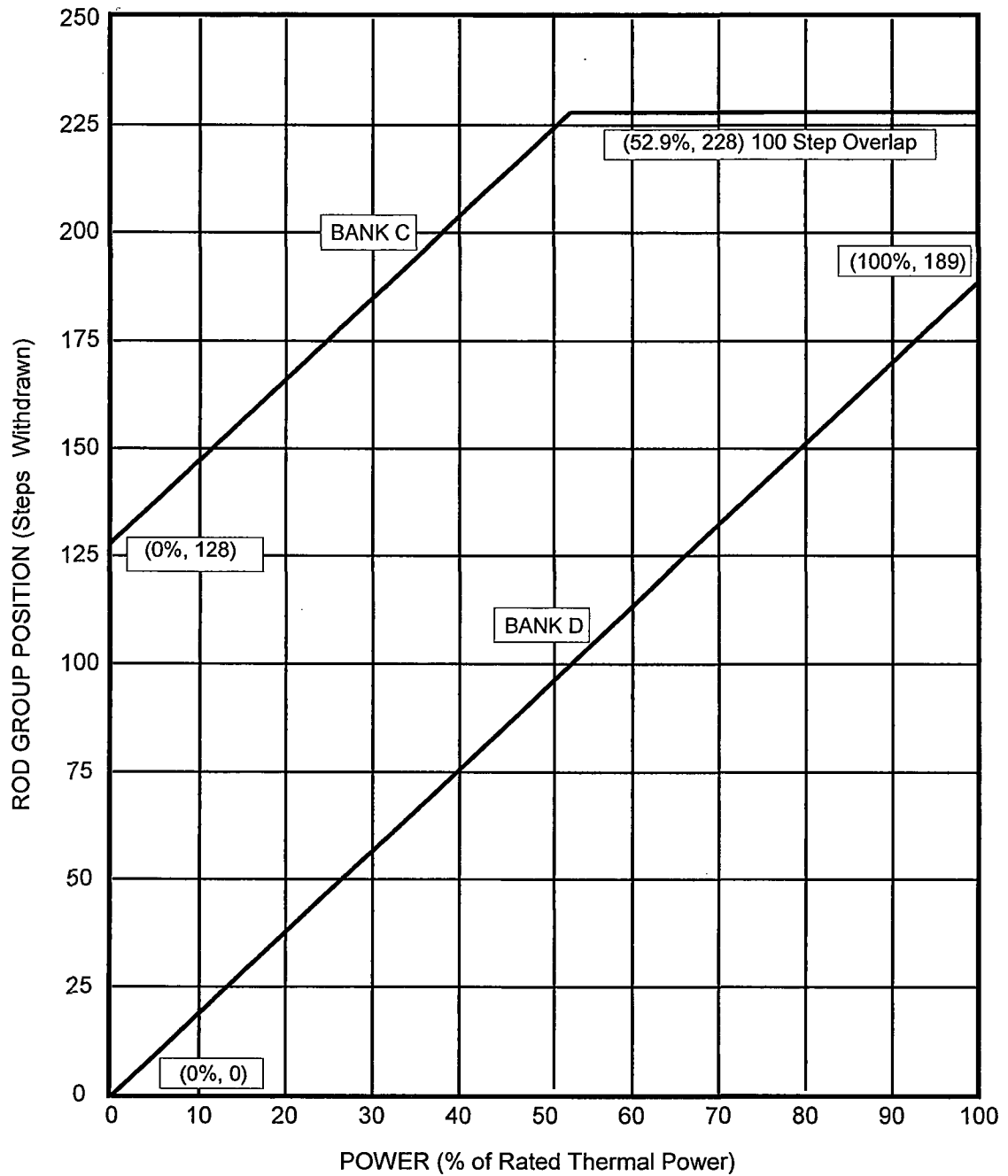


FIGURE 3

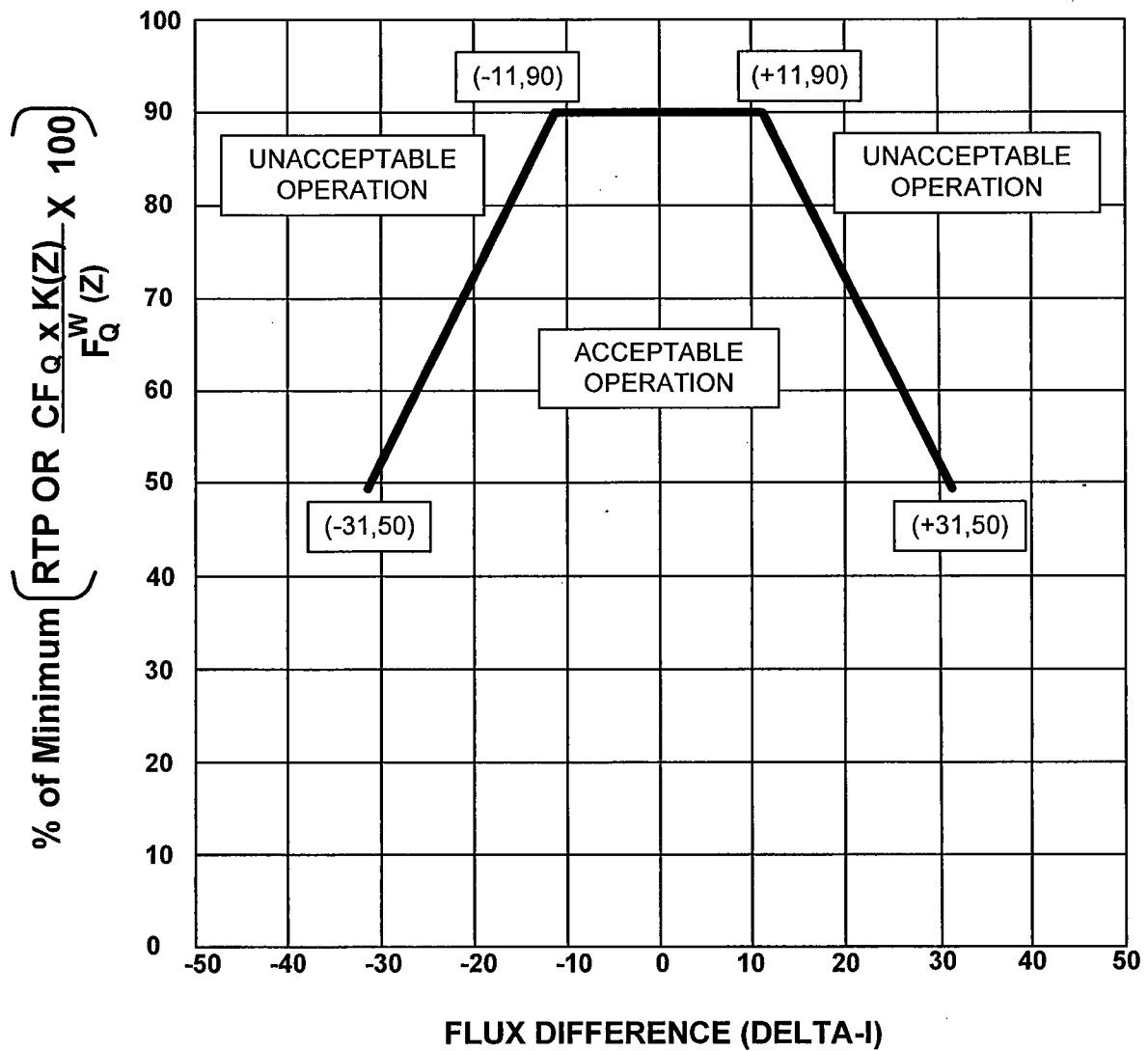
AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED
THERMAL POWER (RTP)

FIGURE 4

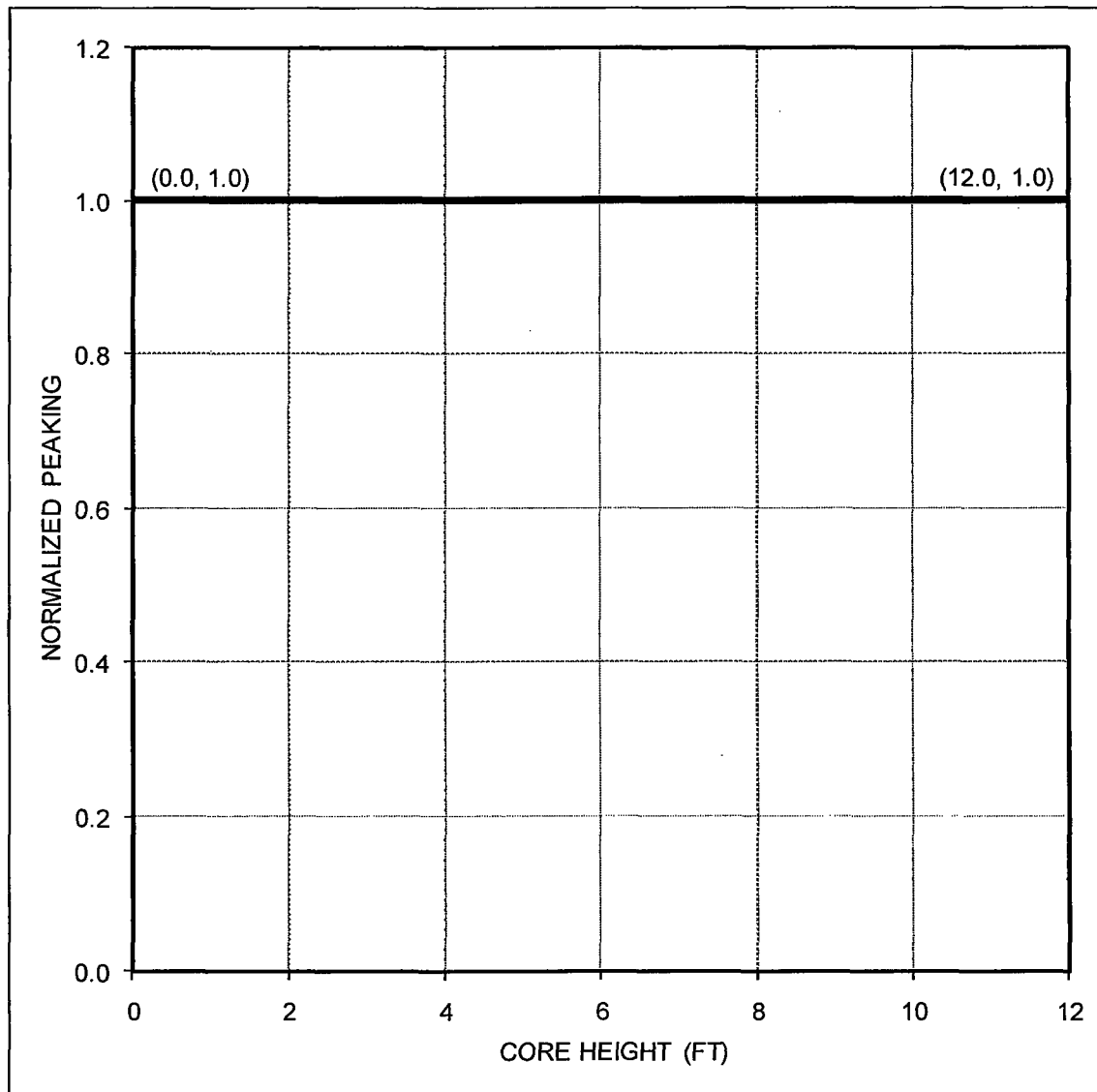
K(Z) – NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

FIGURE 5**(Page 1 of 2)****Reactor Trip System Instrumentation Trip Setpoints****Overtemperature ΔT Trip Setpoint**

$$\text{Overtemperature } \Delta T \leq \Delta T_o [K_1 - K_2 \left[\frac{1 + \tau_1 S}{1 + \tau_2 S} \right] (T - T') + K_3 (P - P') - f_1 (\Delta I)]$$

Where:	ΔT	=	Measured RCS ΔT , °F
	ΔT_o	=	Indicated ΔT at RATED THERMAL POWER, °F
	T	=	Average temperature, °F
	T'	=	Nominal T_{avg} at RATED THERMAL POWER (≤ 574.0 °F)
	P	=	Pressurizer pressure, psig
	P'	=	Nominal RCS operating pressure (2085 psig)
	$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
	τ_1, τ_2	=	Time constants utilized in the lead-lag controller for T_{avg} $\tau_1 \geq 22$ secs. $\tau_2 \leq 4$ secs.
	S	=	Laplace transform operator, sec^{-1}
	K_1	\leq	1.35 *
	K_2	\geq	0.0230/°F
	K_3	\geq	0.00110/psi
	$f_1 (\Delta I)$	=	-0.33 {37% + ($q_t - q_b$)} when $q_t - q_b \leq -37\%$ RTP 0% of RTP when $-37\% \text{ RTP} < q_t - q_b \leq 3\% \text{ RTP}$ +2.34 {($q_t - q_b$) - 3%} when $q_t - q_b > 3\% \text{ RTP}$

where q_t and q_b are percent RATED THERMAL POWER in the upper and lower halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent RATED THERMAL POWER.

* This is a Safety Analysis value. Refer to Technical Requirements Manual for nominal value of this coefficient used in programming the trip setpoint.

FIGURE 5
(Page 2 of 2)
Overpower ΔT Trip Setpoint

$$\text{Overpower } \Delta T \leq \Delta T_o \left[K_4 - K_5 \left[\frac{\tau_3 S}{1 + \tau_3 S} \right] T - K_6 (T - T'') - f_2(\Delta T) \right]$$

Where:	ΔT	=	Measured RCS ΔT , °F
	ΔT_o	=	Indicated ΔT at RATED THERMAL POWER, °F
	T	=	Average temperature, °F
	T''	=	Nominal T_{avg} at RATED THERMAL POWER (≤ 562.1 °F)
	K_4	\leq	1.172 *
	K_5	\geq	0.0177/°F for increasing average temperature ; $K_5 = 0$ for decreasing average temperature
	K_6	\geq	0.0015/°F for T greater than T'' ; $K_6 = 0$ for T less than or equal to T''

$$\frac{\tau_3 S}{1 + \tau_3 S} = \text{The function generated by the rate lag controller for } T_{avg} \text{ dynamic compensation}$$

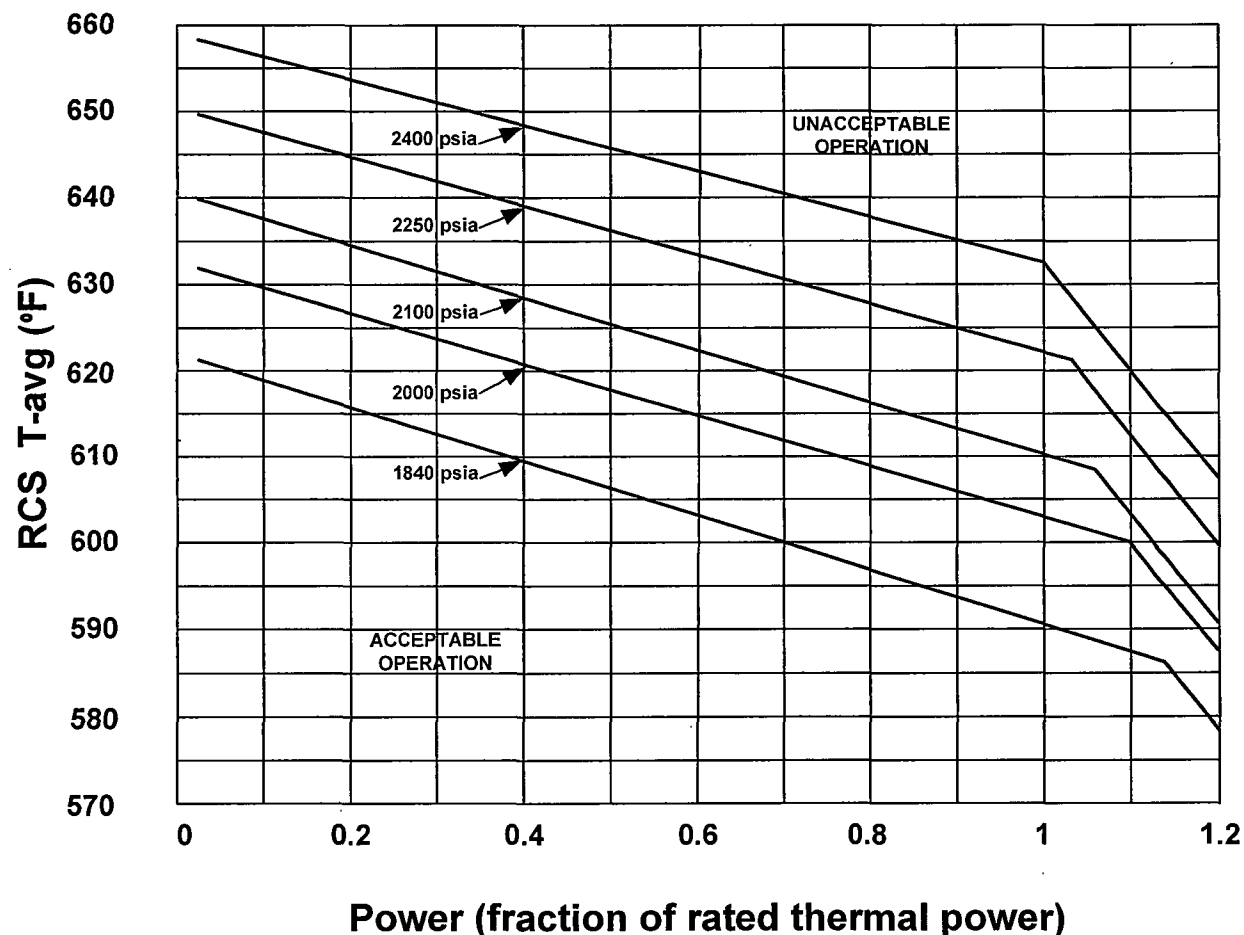
$$\tau_3 = \text{Time constant utilized in the rate lag controller for } T_{avg} \quad \tau_3 \geq 10 \text{ secs.}$$

$$S = \text{Laplace transform operator, sec}^{-1}$$

$$f_2(\Delta T) = 0.0$$

* This is a Safety Analysis value. Refer to Technical Requirements Manual for nominal value of this coefficient used in programming the trip setpoint.

FIGURE 6
Reactor Core Safety Limits

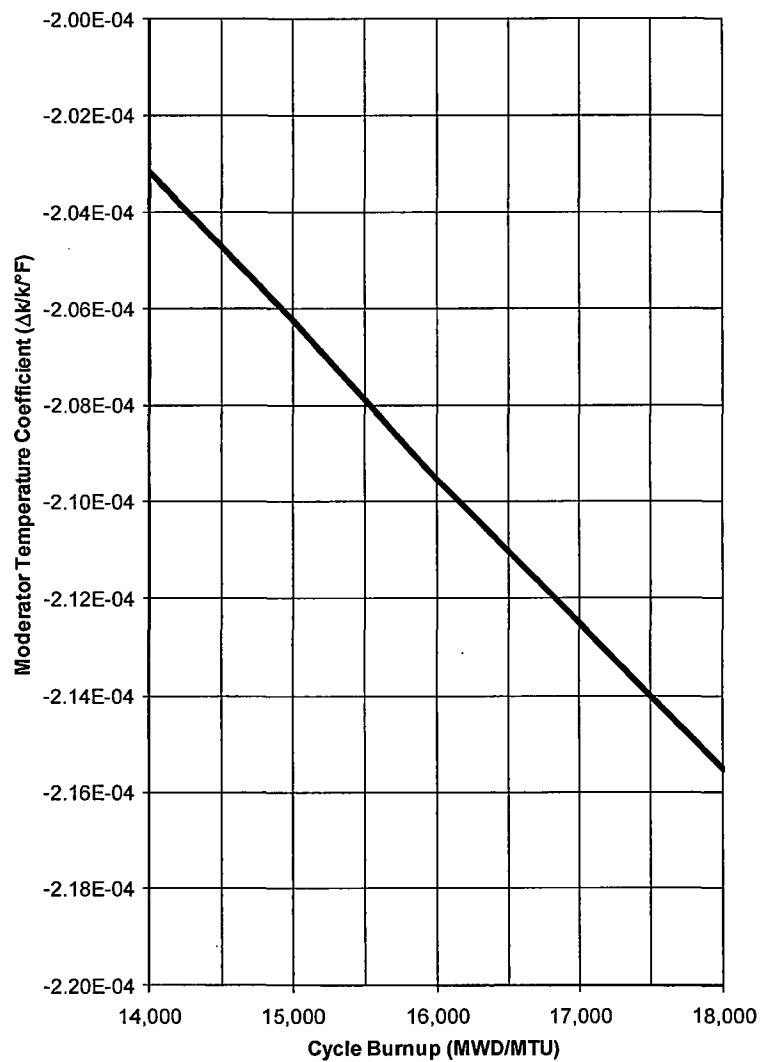


DESCRIPTION OF SAFETY LIMITS

<u>PRESSURE</u> <u>(PSIA)</u>	<u>Power</u> <u>(frac)</u>	<u>Tavg</u> <u>(°F)</u>	<u>Power</u> <u>(frac)</u>	<u>Tavg</u> <u>(°F)</u>	<u>Power</u> <u>(frac)</u>	<u>Tavg</u> <u>(°F)</u>
1840	0.02	620.86	1.136	586.17	1.2	577.94
2000	0.02	632.79	1.094	600.31	1.2	586.52
2100	0.02	639.85	1.068	608.72	1.2	591.77
2250	0.02	649.96	1.031	620.83	1.2	599.40
2400	0.02	659.52	0.996	632.42	1.2	606.63

UNIT 1 Reactor Core Safety Limits

FIGURE 7
Unit 1 Cycle 25 Predicted HFP ARO 300 PPM MTC
Versus Burnup



Burnup (MWD/MTU)	MTC (pcm/°F)	MTC ($\Delta k/k^{\circ}F$)
14000	-20.316	-2.0316E-04
15000	-20.621	-2.0621E-04
16000	-20.955	-2.0955E-04
17000	-21.252	-2.1252E-04
18000	-21.554	-2.1554E-04

