



FirstEnergy Nuclear Operating Company

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L-01-128

U. S. Nuclear Regulatory Commission  
Attention: Document Control Desk  
Washington, DC 20555-0001

**Subject: Beaver Valley Power Station, Unit No. 1  
Docket No. 50-334, License No. DPR-66  
Cycle 15 Reload and Core Operating Limits Report**

Beaver Valley Power Station, Unit No. 1 completed the fourteenth cycle of operation on August 31, 2001, with a burnup of 17,627.56 MWD/MTU. This letter describes the Cycle 15 reload design, provides a copy of the Core Operating Limits Report (COLR) in accordance with Technical Specification 6.9.5.d, and documents our review in accordance with 10 CFR 50.59 including our determination that no Technical Specification or license amendment was required for the Cycle 15 reload design.

The Beaver Valley Power Station Unit No. 1 reactor core features a low leakage pattern. During the 1R14 refueling, 9 Region 1, 24 Region 14A and 28 Region 15A fuel assemblies were discharged and replaced with 1 reinserted Region 13A, 28 fresh Region 17A fuel assemblies enriched to 4.20 nominal weight percent, and 32 fresh Region 17B fuel assemblies enriched to 4.80 nominal weight percent. Cycle 15 is the first cycle with Robust Fuel Assemblies (RFA) in the core (Regions 17A & 17B). The NRC-approved Fuel Criteria Evaluation Process (WCAP-12488-A, October 1994) was used to address the design modifications for the 17 x 17 RFA (NSD-NRC-98-5796, October 13, 1998).

FirstEnergy Nuclear Operating Company has performed a review of this reload core design to determine those parameters affecting the design basis limits and the safety analyses for postulated accidents described in the Updated Final Safety Analysis Report (UFSAR). The analytical methods used to determine the core operating limits meet the criteria specified in Technical Specification 6.9.5.a, b and c. This core reload has been designed using the Revised Thermal Design Procedure approved by Technical Specification Amendment No. 239 and using the updated conditions approved by Technical Specification Amendment No. 243. The Cycle 15 reactor core reload evaluation concluded that the implementation of the 17 x 17 RFA core will not adversely affect the safety of the plant. The reload evaluation also concluded that the core design will not require any Technical Specification changes, and will not require a license amendment due to new safety analyses changes, fission product barrier design basis

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limits, or methods of evaluation as described in the UFSAR, pursuant to 10 CFR 50.59. The Beaver Valley Onsite Safety Committee has concurred with the conclusions of the reload evaluation.

The Core Operating Limits Report (COLR) is enclosed in accordance with Technical Specification 6.9.5.d. The COLR has been updated for this cycle by revising the  $F_{xy}$  and maximum  $F_Q^{T*} P_{REL}$  criteria.

If there are any questions concerning this matter, please contact Mr. Thomas S. Cosgrove, Manager, Regulatory Affairs at 724-682-5203.

Sincerely,



Lew W. Myers

- c: Mr. L. J. Burkhart, Project Manager
- Mr. D. M. Kern, Sr. Resident Inspector
- Mr. H. J. Miller, NRC Region I Administrator
- Mr. L. E. Ryan (BRP/DEP)

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4.1 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report provides the cycle specific parameter limits developed in accordance with the NRC approved methodologies specified in Technical Specification Administrative Control 6.9.5.

Specification 3.1.3.5 Shutdown Rod Insertion Limits

The shutdown rods shall be withdrawn to at least 225 steps.

Specification 3.1.3.6 Control Rod Insertion Limits

Control Banks A and B shall be withdrawn to at least 225 steps.

Control Banks C and D shall be limited in physical insertion as shown in Figure 4.1-1.

Specification 3.2.1 Axial Flux Difference

NOTE: The target band is  $\pm 7\%$  about the target flux from 0% to 100% RATED THERMAL POWER.

The indicated Axial Flux Difference:

- a. Above 90% RATED THERMAL POWER shall be maintained within the  $\pm 7\%$  target band about the target flux difference.
- b. Between 50% and 90% RATED THERMAL POWER is within the limits shown on Figure 4.1-2.
- c. Below 50% RATED THERMAL POWER may deviate outside the target band.

Specification 3.2.2  $F_Q(Z)$  and  $F_{xy}$  Limits

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

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

Where:  $CF_Q = 2.3$                        $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

$K(Z)$  = the function obtained from Figure 4.1-3.

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The  $F_{xy}$  limits [ $F_{xy}(L)$ ] for RATED THERMAL POWER within specific core planes shall be:

$$F_{xy}(L) = F_{xy}(RTP)(1 + PF_{xy} * (1-P))$$

Where: For all core planes containing D-Bank:

$$F_{xy}(RTP) \leq 1.71$$

For unrodded core planes:

$$F_{xy}(RTP) \leq 1.77 \text{ from 1.8 ft. elevation to 2.3 ft. elevation}$$

$$F_{xy}(RTP) \leq 1.82 \text{ from 2.3 ft. elevation to 5.8 ft. elevation}$$

$$F_{xy}(RTP) \leq 1.85 \text{ from 5.8 ft. elevation to 8.5 ft. elevation}$$

$$F_{xy}(RTP) \leq 1.82 \text{ from 8.5 ft. elevation to 9.0 ft. elevation}$$

$$F_{xy}(RTP) \leq 1.75 \text{ from 9.0 ft. elevation to 10.2 ft. elevation}$$

$$PF_{xy} = 0.2$$

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

Figure 4.1-4 provides the maximum total peaking factor times relative power ( $F_Q^T * P_{rel}$ ) as a function of axial core height during normal core operation.

Specification 3.2.3  $F_{\Delta H}^N$

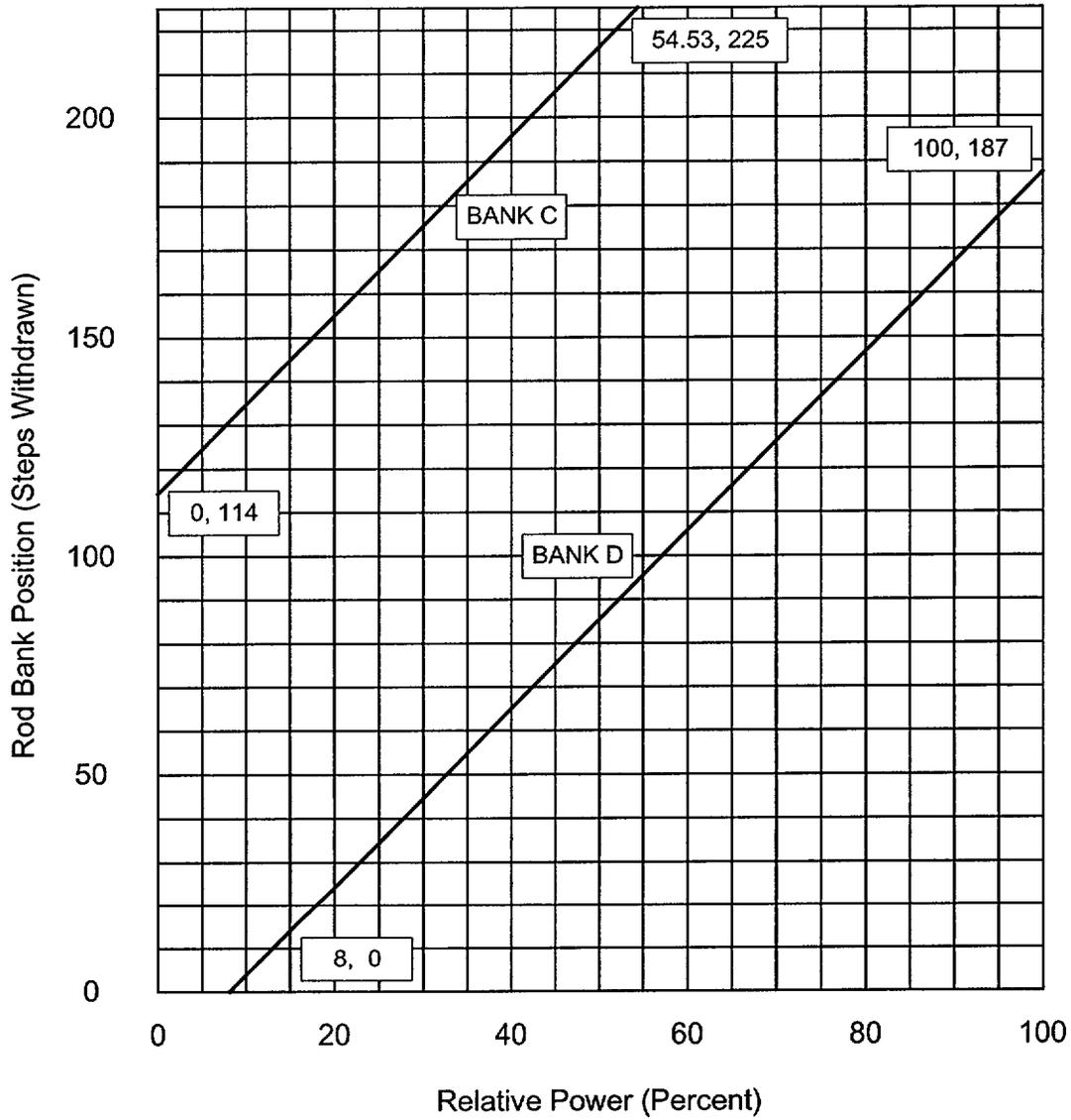
$$F_{\Delta H}^N \leq CF_{\Delta H} * (1 + PF_{\Delta H} (1-P))$$

Where:  $CF_{\Delta H} = 1.62$

$$PF_{\Delta H} = 0.3$$

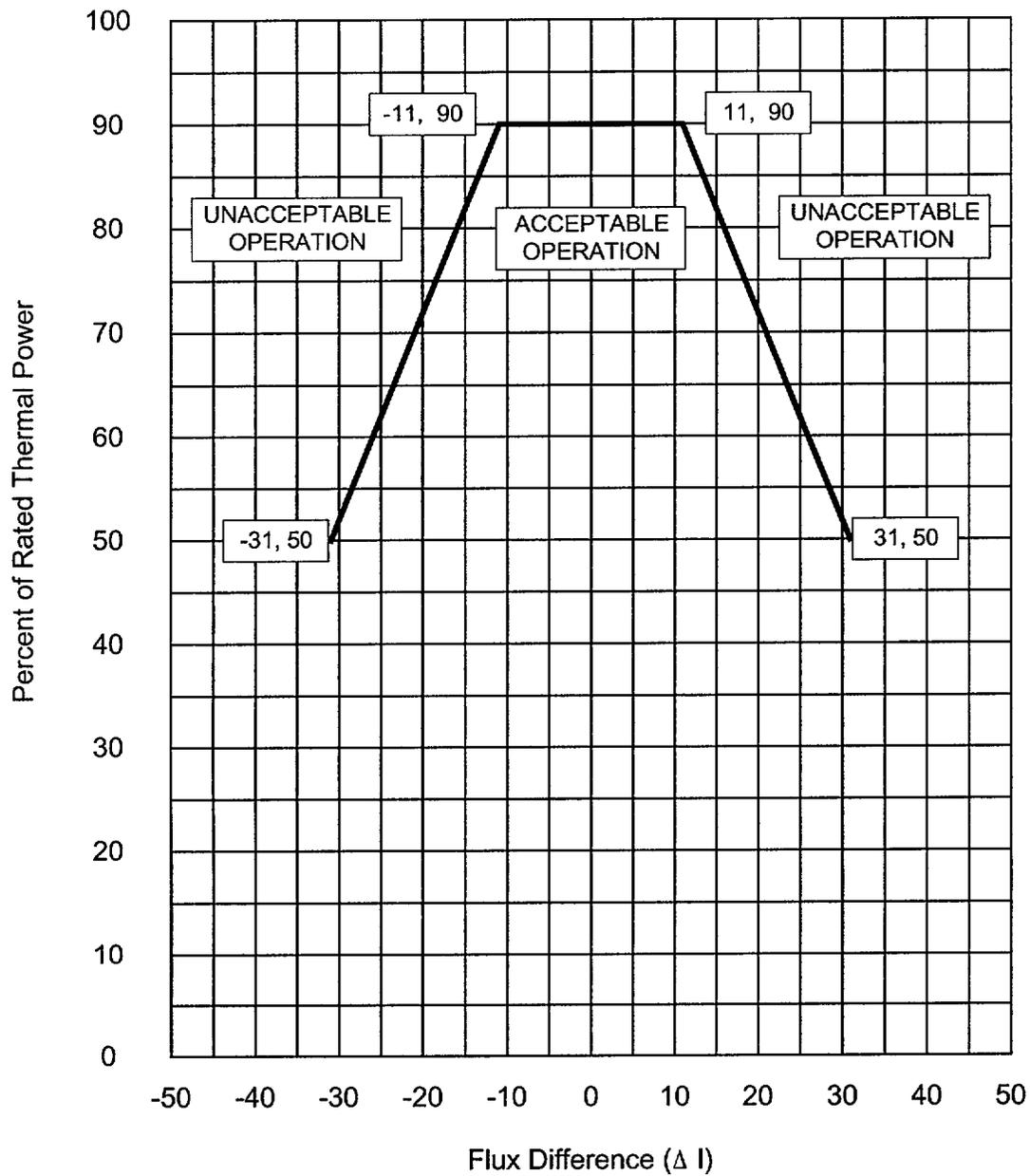
$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

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**FIGURE 4.1-1  
CONTROL ROD INSERTION LIMITS**

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**FIGURE 4.1-2  
AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF  
RATED THERMAL POWER**

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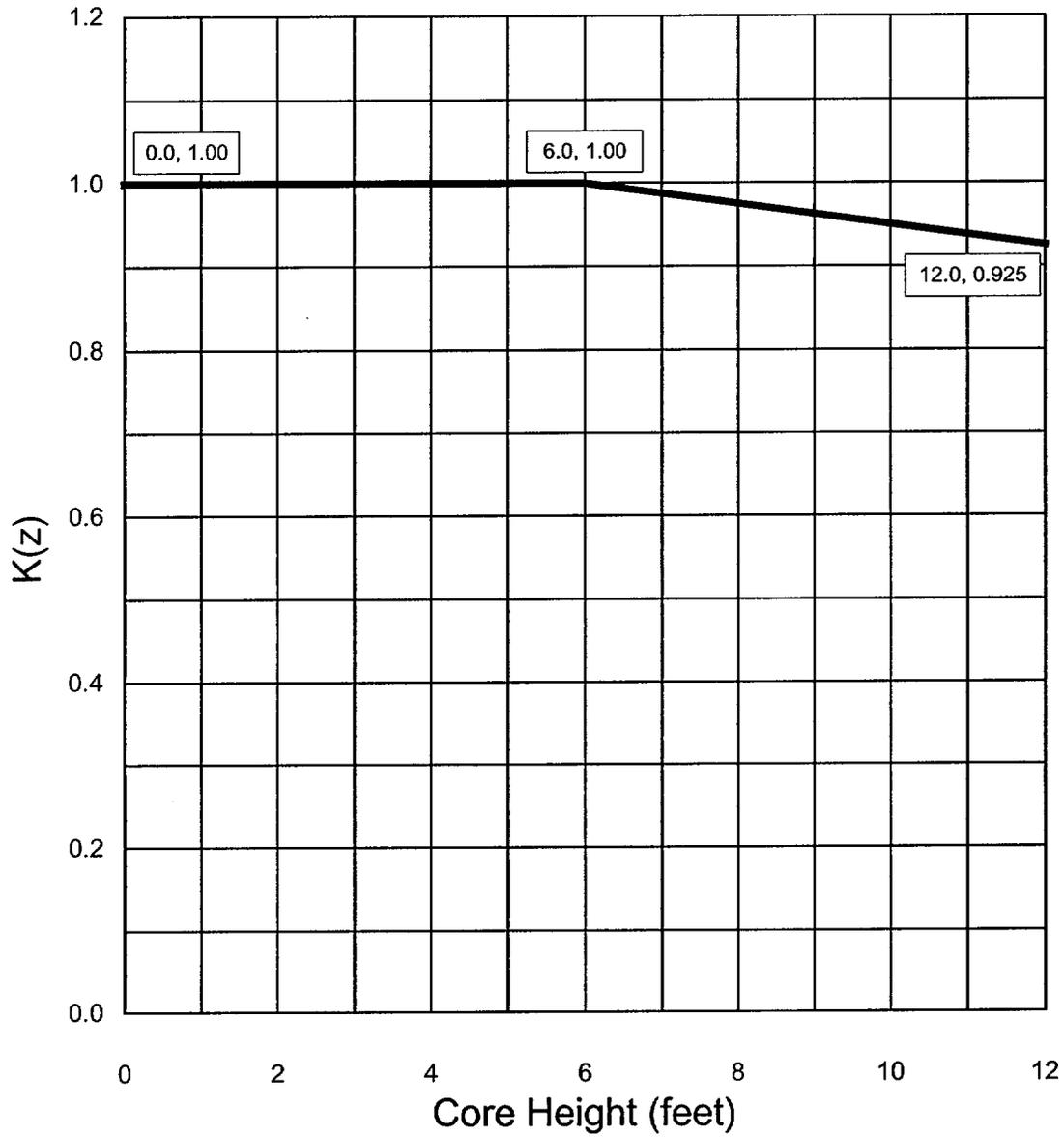
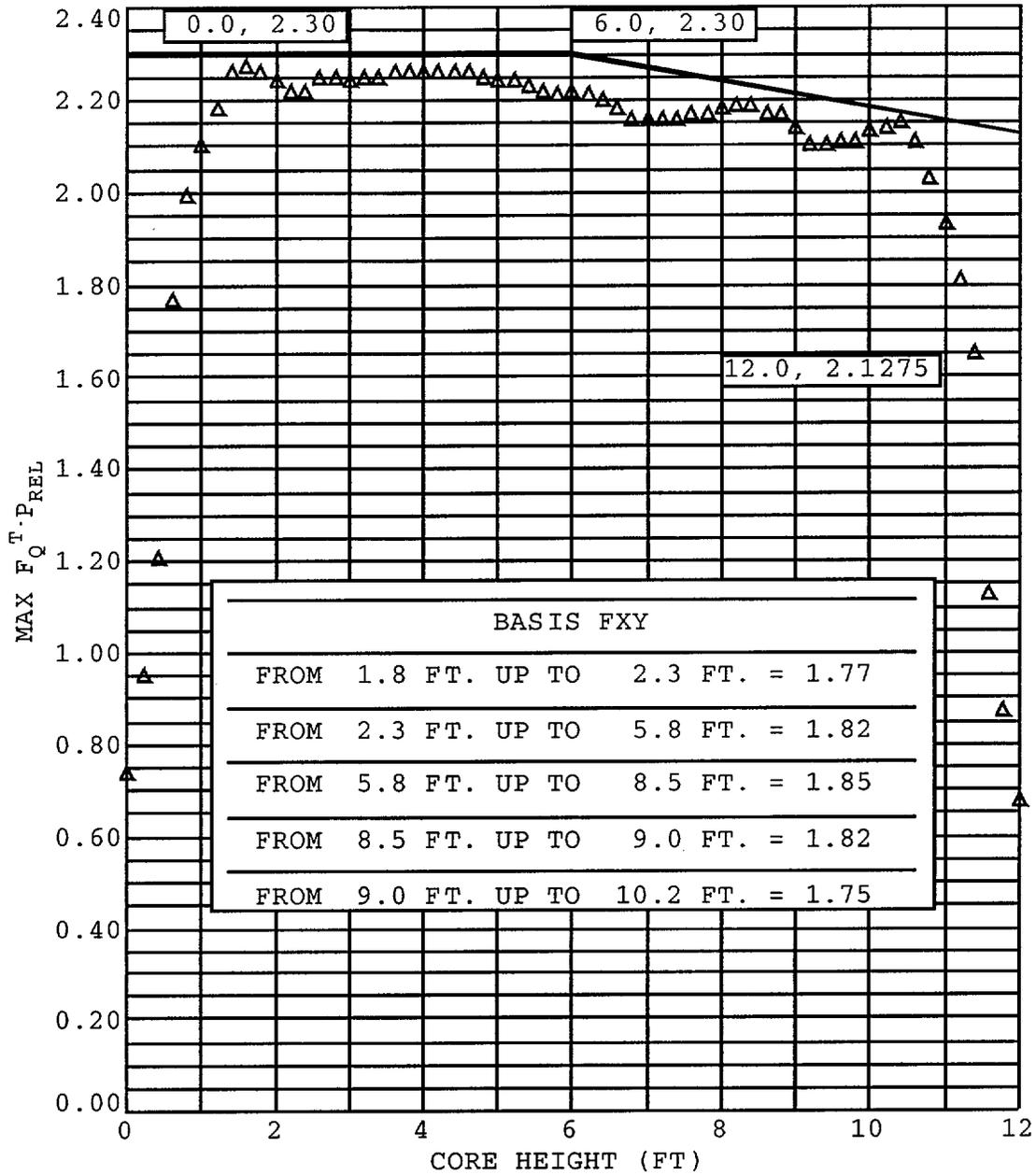


FIGURE 4.1-3  
F<sub>QT</sub> NORMALIZED OPERATING ENVELOPE, K(Z)

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**FIGURE 4.1-4**  
**MAXIMUM ( $F_Q^T \cdot P_{REL}$ ) VS AXIAL CORE HEIGHT**  
**DURING NORMAL OPERATION**

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Specification 3.3.1.1 Reactor Trip System Instrumentation Setpoints, Table 3.3-1 Table Notations A and B

Overtemperature  $\Delta T$  Setpoint Parameter Values:

<u>Parameter</u>	<u>Value</u>
Overtemperature $\Delta T$ reactor trip setpoint	$K1 \leq 1.259$
Overtemperature $\Delta T$ reactor trip setpoint Tav <sub>g</sub> coefficient	$K2 \geq 0.01655/^{\circ}\text{F}$
Overtemperature $\Delta T$ reactor trip setpoint pressure coefficient	$K3 \geq 0.000801/\text{psia}$
Tav <sub>g</sub> at RATED THERMAL POWER	$T' \leq 576.2^{\circ}\text{F}$
Nominal Pressurizer Pressure	$P' \geq 2250 \text{ psia}$
Measured reactor vessel average temperature lead/lag time constants	$\tau_1 \geq 30 \text{ secs}$ $\tau_2 \leq 4 \text{ secs}$

$f(\Delta I)$  is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

- (i) for  $q_t - q_b$  between -36 percent and +15 percent,  $f(\Delta I) = 0$  (where  $q_t$  and  $q_b$  are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and  $q_t + q_b$  is total THERMAL POWER in percent of RATED THERMAL POWER).
- (ii) for each percent that the magnitude of  $(q_t - q_b)$  exceeds -36 percent, the  $\Delta T$  trip setpoint shall be automatically reduced by 2.08 percent of its value at RATED THERMAL POWER.
- (iii) for each percent that the magnitude of  $(q_t - q_b)$  exceeds +15 percent, the  $\Delta T$  trip setpoint shall be automatically reduced by 1.59 percent of its value at RATED THERMAL POWER.

Overpower  $\Delta T$  Setpoint Parameter Values:

<u>Parameter</u>	<u>Value</u>
Overpower $\Delta T$ reactor trip setpoint	$K4 \leq 1.0916$
Overpower $\Delta T$ reactor trip setpoint Tav <sub>g</sub> rate/lag coefficient	$K5 \geq 0.02/^{\circ}\text{F}$ for increasing average temperature

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Overpower  $\Delta T$  Setpoint Parameter Values (continued):

<u>Parameter</u>	<u>Value</u>
Overpower $\Delta T$ reactor trip setpoint $T_{avg}$ heatup coefficient	$K6 \geq 0.00128/^{\circ}\text{F}$ for $T > T''$ $K6 = 0/^{\circ}\text{F}$ for $T \leq T''$
$T_{avg}$ at RATED THERMAL POWER	$T'' \leq 576.2^{\circ}\text{F}$
Measured reactor vessel average temperature rate/lag time constant	$\tau_3 \geq 10$ secs

Specification 3.2.5 DNB Parameters

<u>Parameter</u>	<u>Indicated Value</u>
Reactor Coolant System $T_{avg}$	$T_{avg} \leq 580.0^{\circ}\text{F}^{(1)}$
Pressurizer Pressure	Pressure $\geq 2215$ psia <sup>(2)</sup>
Reactor Coolant System Total Flow Rate	Flow $\geq 267,400$ gpm <sup>(3)</sup>

- (1) The Reactor Coolant System (RCS)  $T_{avg}$  value includes allowances for rod control operation and verification via control board indication.
- (2) The pressurizer pressure value includes allowances for pressurizer pressure control operation and verification via control board indication.
- (3) The RCS total flow rate includes allowances for normalization of the cold leg elbow taps with a beginning of cycle precision RCS flow calorimetric measurement and verification on a periodic basis via control board indication.

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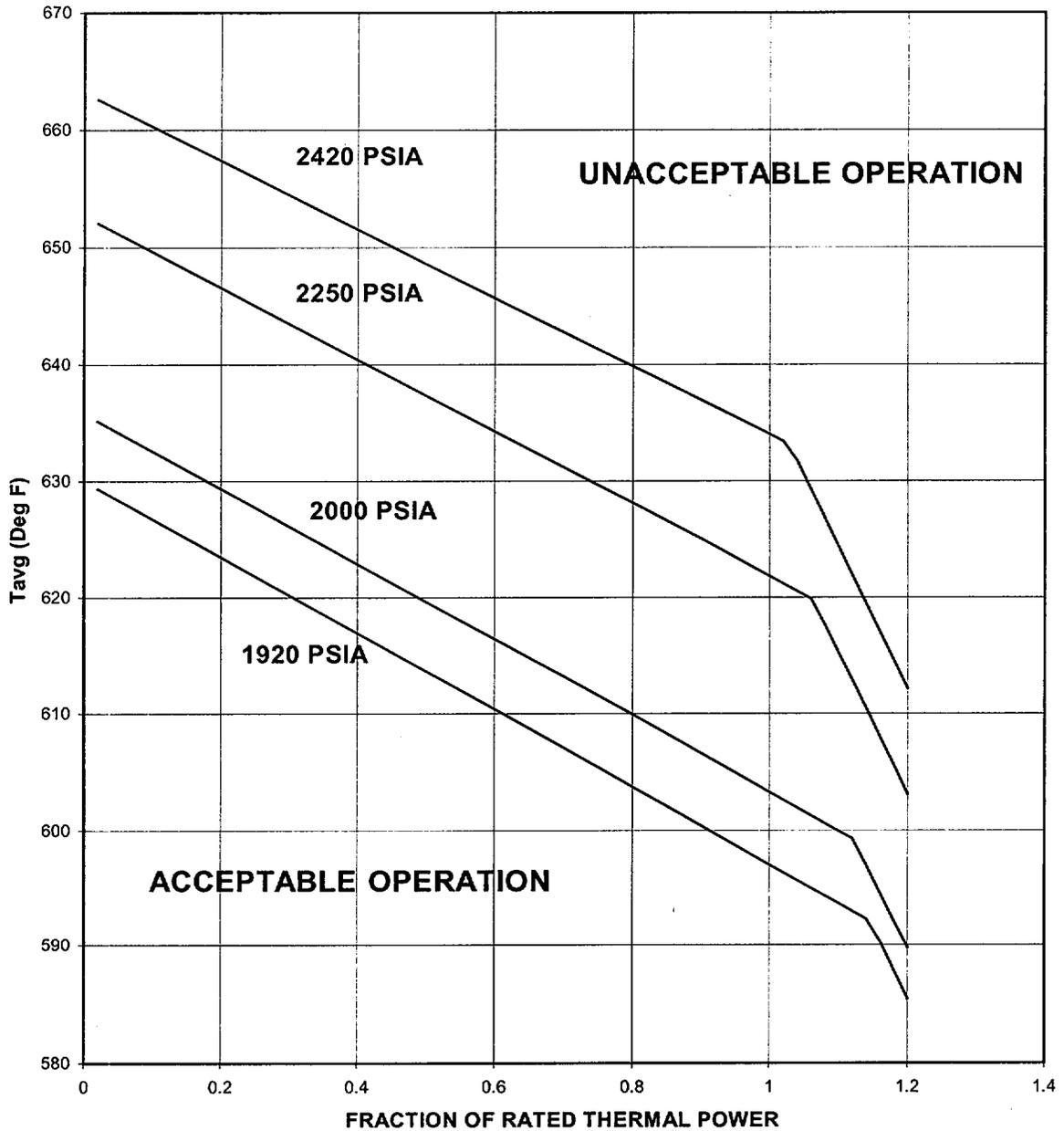


Figure 4.1-5  
REACTOR CORE SAFETY LIMIT  
THREE LOOP OPERATION  
(Technical Specification Safety Limit 2.1.1)