FENOC

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October 9, 2003
L-03-158

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

Subject: Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, License No. NPF-73
Cycle 11 Reload and Core Operating Limits Report

Beaver Valley Power Station, Unit No. 2 completed the tenth cycle of operation on September 12, 2003, with a burnup of 20,307.76 MWD/MTU. This letter describes the Cycle 11 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 11 reload design.

The Beaver Valley Power Station Unit No. 2 reactor core features a low leakage pattern. During the 2R10 refueling, 1 Region 8B, 12 Region 10A, 20 Region 10B and 28 Region 11A assemblies were discharged and replaced with 1 reinserted Region 8A, 36 fresh Region 13A fuel assemblies enriched to 4.40 nominal weight percent and 24 fresh Region 13B fuel assemblies enriched to 4.95 nominal weight percent.

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. The Cycle 11 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 limits, or methods of evaluation as described in the UFSAR, pursuant to 10 CFR 50.59. The Beaver Valley Plant Onsite Review Committee has concurred with the conclusions of the reload evaluation.

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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.

No new commitments are contained in this submittal. If there are any questions concerning this matter, please contact Mr. Larry R. Freeland, Manager, Regulatory Affairs/Performance Improvement at 724-682-5284.

Sincerely,

L. W. Pearce

L. William Pearce

Attachment

c: Mr. T. G. Colburn, NRR Senior Project Manager
Mr. P. C. Cataldo, NRC Sr. Resident Inspector
Mr. H. J. Miller, NRC Region I Administrator
Mr. L. E. Ryan (BRP/DEP)

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 ±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 ±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

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

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

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

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

*As indicated by the group demand counter

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:

$F_{xy}(RTP) \leq 1.76$ from 1.8 ft. elevation to 2.3 ft. elevation

$F_{xy}(RTP) \leq 1.80$ from 2.3 ft. elevation to 3.7 ft. elevation

$F_{xy}(RTP) \leq 1.83$ from 3.7 ft. elevation to 5.8 ft. elevation

$F_{xy}(RTP) \leq 1.84$ from 5.8 ft. elevation to 7.4 ft. elevation

$F_{xy}(RTP) \leq 1.81$ from 7.4 ft. elevation to 9.0 ft. elevation

$F_{xy}(RTP) \leq 1.72$ 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 $F^N_{\Delta H}$

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

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

$PF_{\Delta H} = 0.3$

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

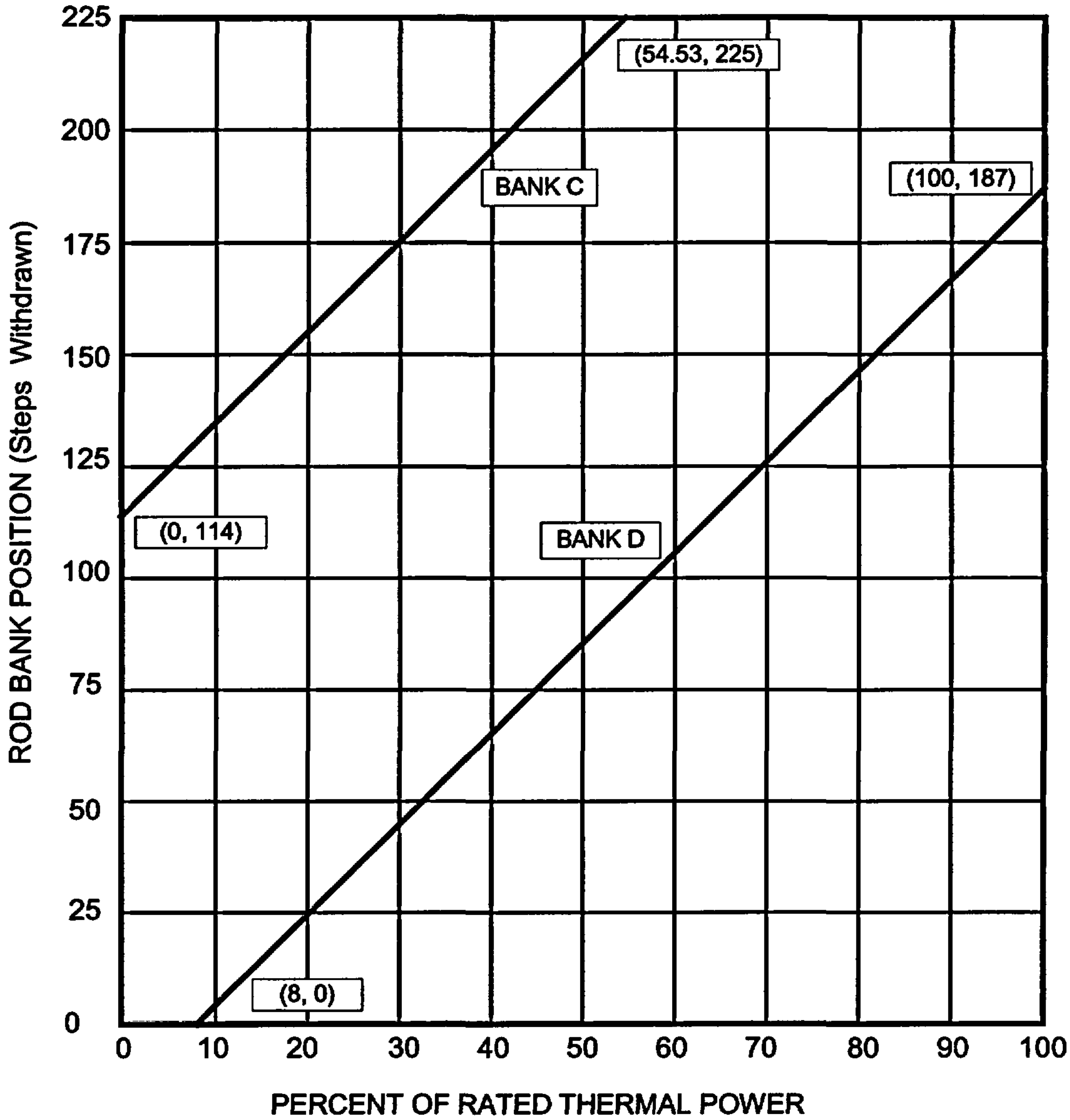

FIGURE 4.1-1

CONTROL ROD INSERTION LIMITS AS A

FUNCTION OF RATED POWER LEVEL

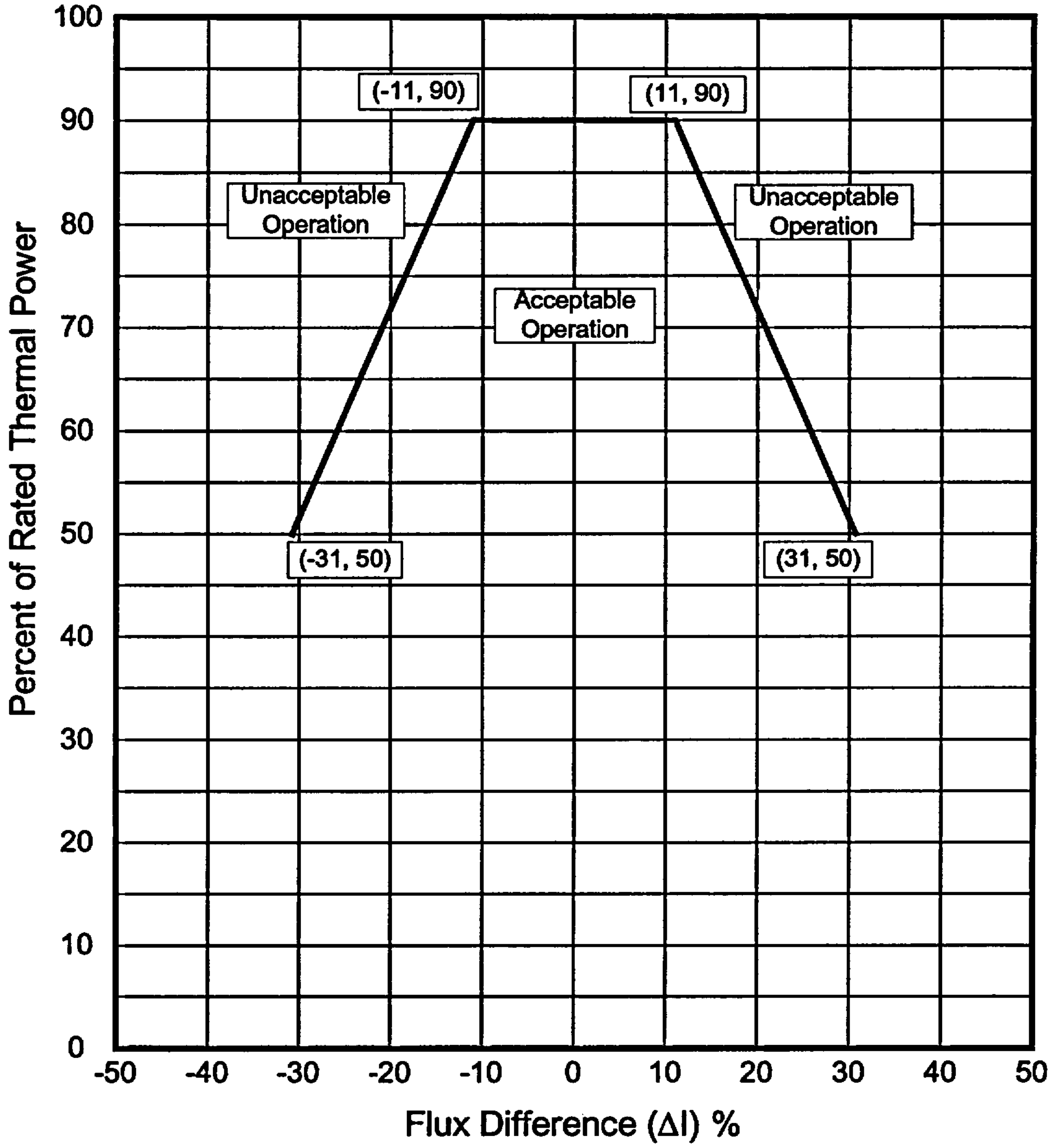

FIGURE 4.1-2

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF

RATED THERMAL POWER

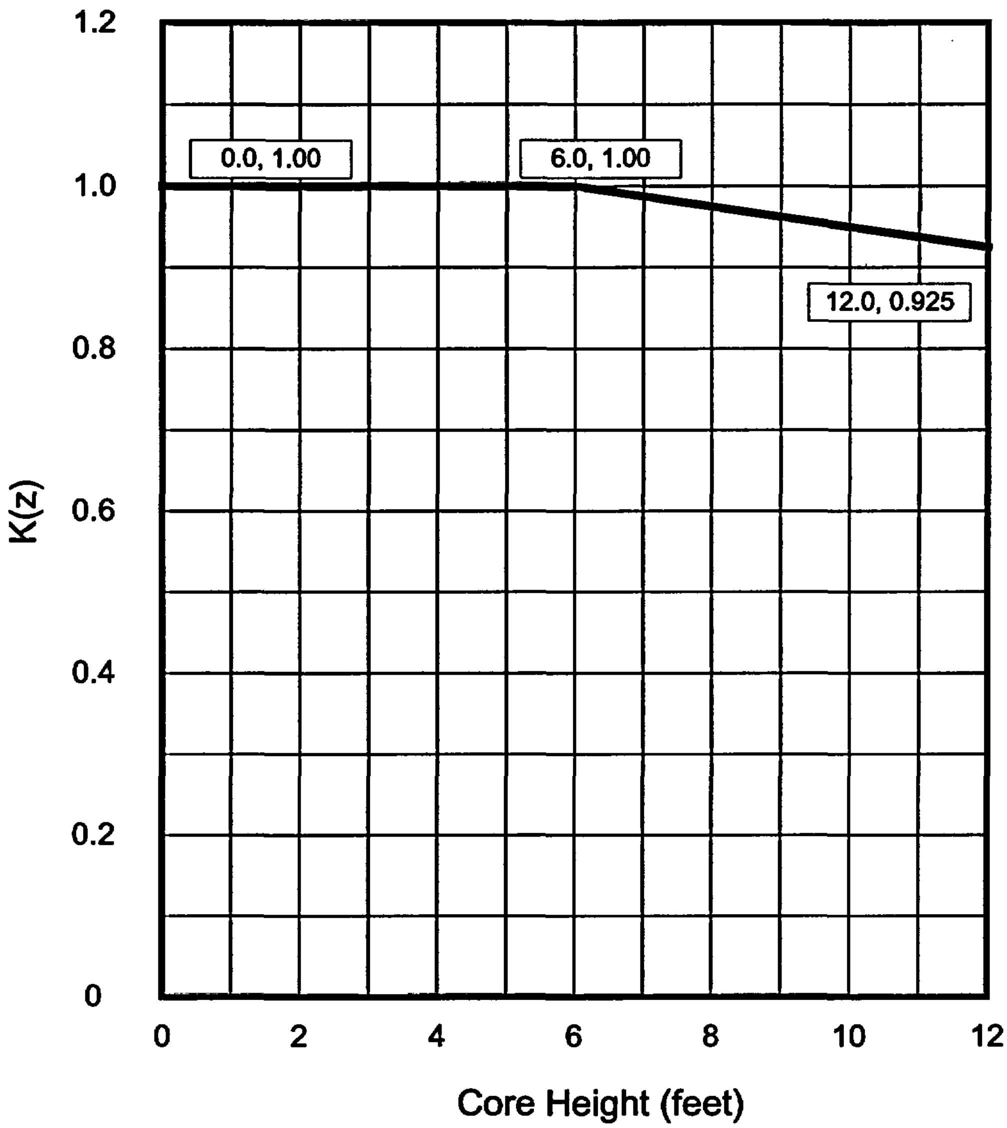

FIGURE 4.1-3

$F_Q T$ NORMALIZED OPERATING ENVELOPE, K(Z)

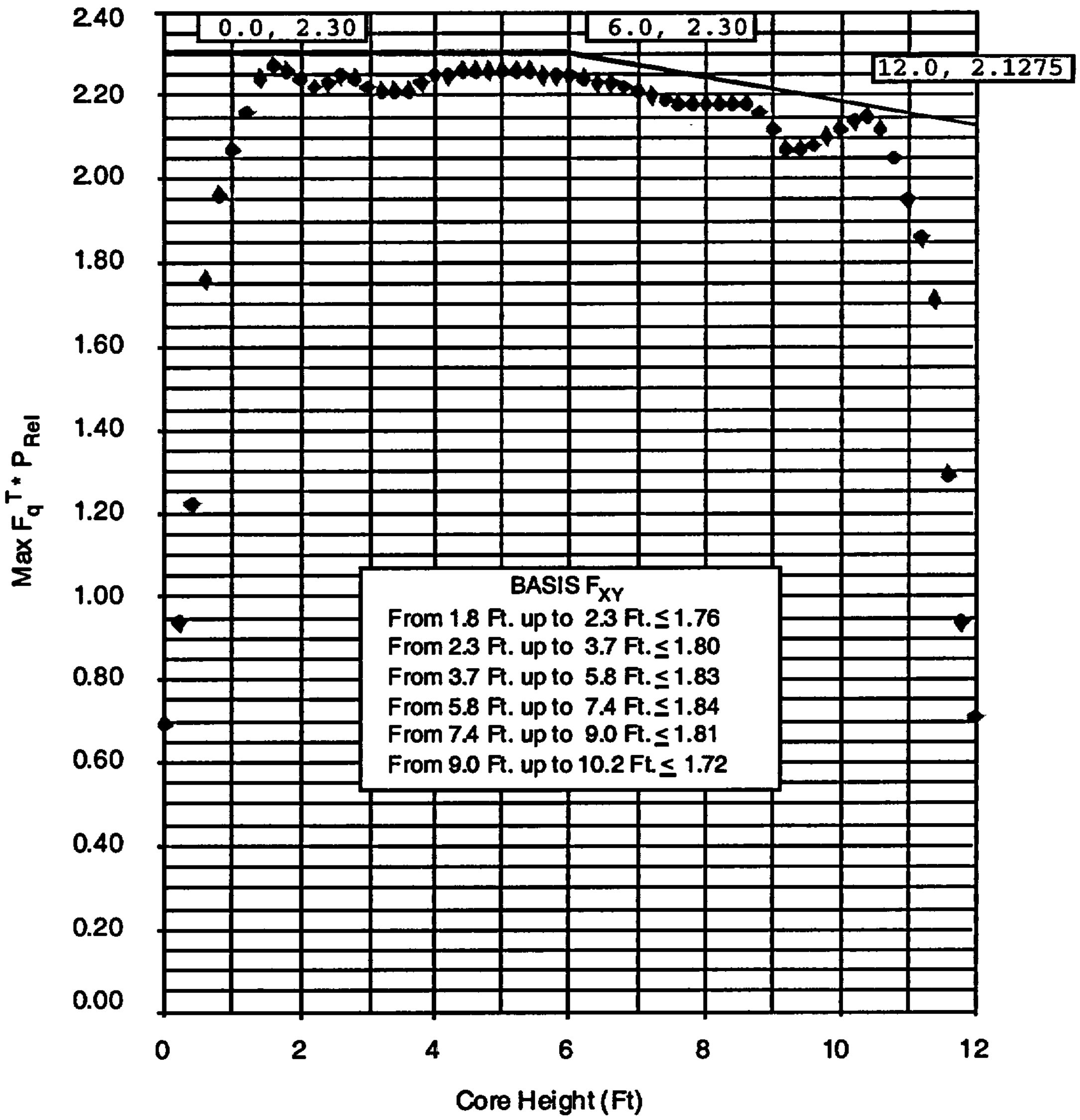

FIGURE 4.1-4

MAXIMUM (F_QT*Prel) VS. AXIAL CORE HEIGHT

DURING NORMAL OPERATION

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

Overtemperature ΔT Setpoint Parameter Values:

Parameter	Value
Overtemperature ΔT reactor trip setpoint	$K1 \leq 1.311$
Overtemperature ΔT reactor trip setpoint Tavg coefficient	$K2 \geq 0.0183/°F$
Overtemperature ΔT reactor trip setpoint pressure coefficient	$K3 \geq 0.00082/psia$
Tavg at RATED THERMAL POWER	$T' \leq 576.2°F$
Nominal pressurizer pressure	$P' \geq 2250$ psia
Measured reactor vessel ΔT lead/lag time constants	$\tau_1 \geq 8$ sec $\tau_2 \leq 3$ sec
Measured reactor vessel ΔT lag time constant	$\tau_3 \leq 0$ sec
Measured reactor vessel average temperature lead/lag time constants	$\tau_4 \geq 30$ sec $\tau_5 \leq 4$ sec
Measured reactor vessel average temperature lag time constant	$\tau_6 \leq 0$ sec

f (ΔI) 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 -32% and +11%, $f_1(\Delta I) = 0$, 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 -32%, the ΔT Trip Setpoint shall be automatically reduced by 1.46% of its value at RATED THERMAL POWER; and

(iii) For each percent that the magnitude of $q_t - q_b$ exceeds +11%, the ΔT Trip Setpoint shall be automatically reduced by 1.56% of its value at RATED THERMAL POWER.

Overpower ΔT Setpoint Parameter Values:

Parameter	Value
Overpower ΔT reactor trip setpoint	$K4 \leq 1.094$
Overpower ΔT reactor trip setpoint Tavg rate/lag coefficient	$K5 \geq 0.00/°F$ for increasing average temperature $K5 = 0/°F$ for decreasing average temperature
Overpower ΔT reactor trip setpoint Tavg heatup coefficient	$K6 \geq 0.0012/°F$ for $T > T''$ $K6 = 0/°F$ for $T \leq T''$
Tavg at RATED THERMAL POWER	$T'' \leq 576.2°F$
Measured reactor vessel ΔT lead/lag time constants	$\tau_1 \geq 8$ sec $\tau_2 \leq 3$ sec
Measured reactor vessel ΔT lag time constant	$\tau_3 \leq 0$ sec
Measured reactor vessel average temperature lag time constant	$\tau_6 \leq 0$ sec
Measured reactor vessel average temperature rate/lag time constant	$\tau_7 \geq 0$ sec

Specification 3.2.5 DNB Parameters

Parameter	Indicated Value
Reactor Coolant System Tavg	Tavg ≤ 579.9°F[(1)]
Pressurizer Pressure	Pressure ≥ 2214 psia[(2)]
Reactor Coolant System Total Flow Rate	Flow ≥ 267,200 gpm[(3)]

(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 plant process computer. If periodic verification of flow rate is performed via the process computer, the required flow value is ≥ 267,200 gpm.

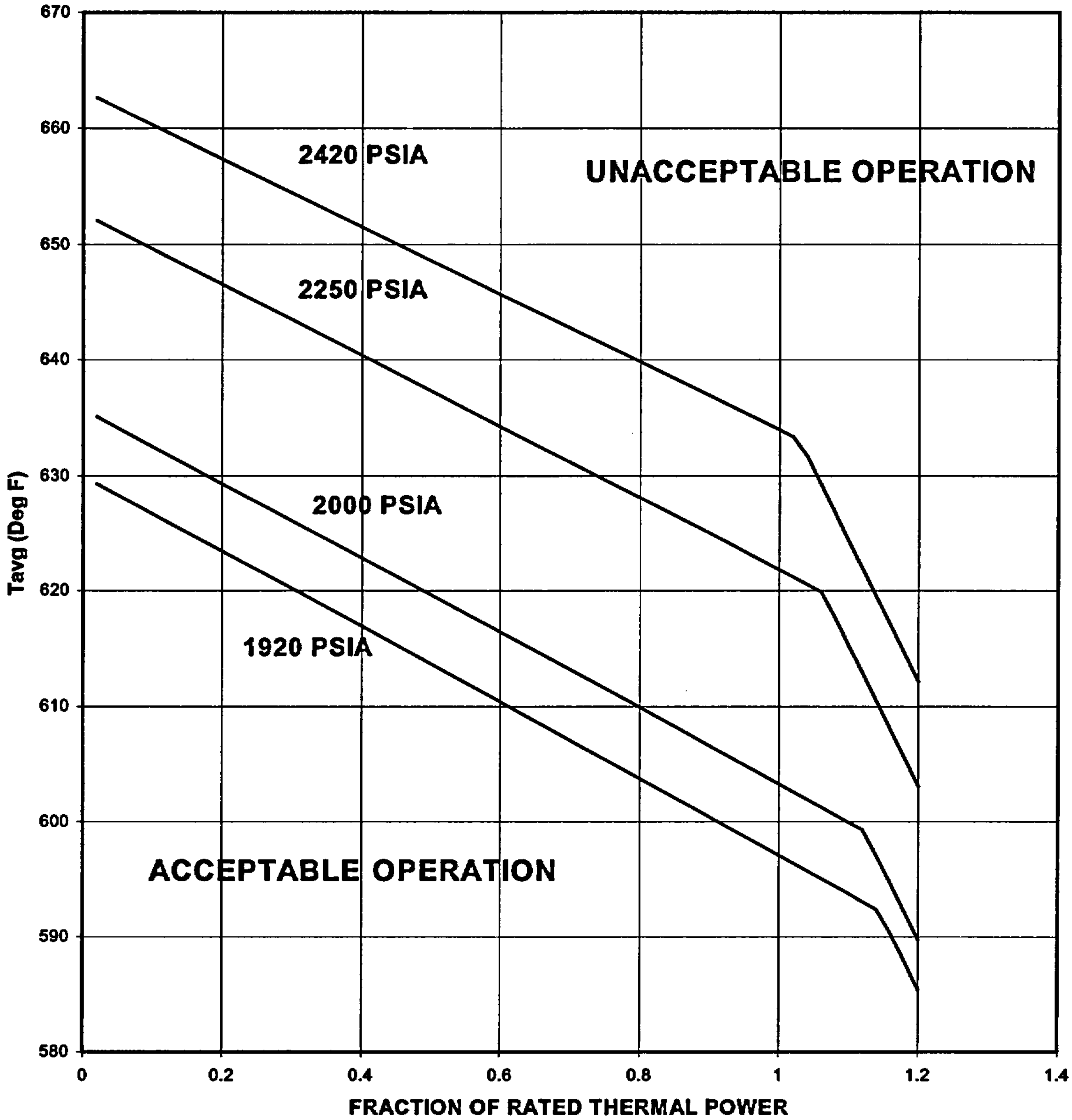

Figure 4.1-5
REACTOR CORE SAFETY LIMIT
THREE LOOP OPERATION
(Technical Specification Safety Limit 2.1.1)