WOLF CREEK
NUCLEAR OPERATING CORPORATION

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Docket No. 50-482: Wolf Creek Generating Station Cycle 13 Core Operating Limits Report

Gentlemen:

Enclosed is Revision 0 of the Wolf Creek Generating Station Cycle 13 Core Operating Limits Report (COLR). This document is being submitted pursuant to Section 5.6.5 of the Wolf Creek Generating Station (WCGS) Technical Specifications.

There are no commitments associated with this submittal. If you should have any questions regarding this submittal, please contact me at (620) 364-4034, or Mr. Tony Harris at (620) 364-4038.

Very truly yours,

Richard A. Muench

RAM/pb

Enclosure

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WOLF CREEK
NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION
CYCLE 13

CORE OPERATING LIMITS REPORT
Revision 0

March, 2002

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1.0 CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT (COLR) for Wolf Creek Generating Station Cycle 13 has been prepared in accordance with the requirements of Technical Specification 5.6.5.

The core operating limits that are included in the COLR affect the following Technical Specifications:

- 2.1.1 Reactor Core Safety Limits
- 3.1.1 Shutdown Margin (SDM)
- 3.1.3 Moderator Temperature Coefficient (MTC)
- 3.1.4 Rod Group Alignments
- 3.1.5 Shutdown Bank Insertion Limits
- 3.1.6 Control Bank Insertion Limits
- 3.1.8 PHYSICS TEST Exceptions - MODE 2
- 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) (FQ Methodology)
- 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)
- 3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology)
- 3.3.1 Reactor Trip System (RTS) Instrumentation
- 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation
- 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits
- 3.9.1 Boron Concentration

The portions of the Technical Specification Bases affected by the report are listed below:

ASA B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the subsections below:

2.1 Reactor Core Safety Limits (SL 2.1.1)

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits in Figure 2.1.

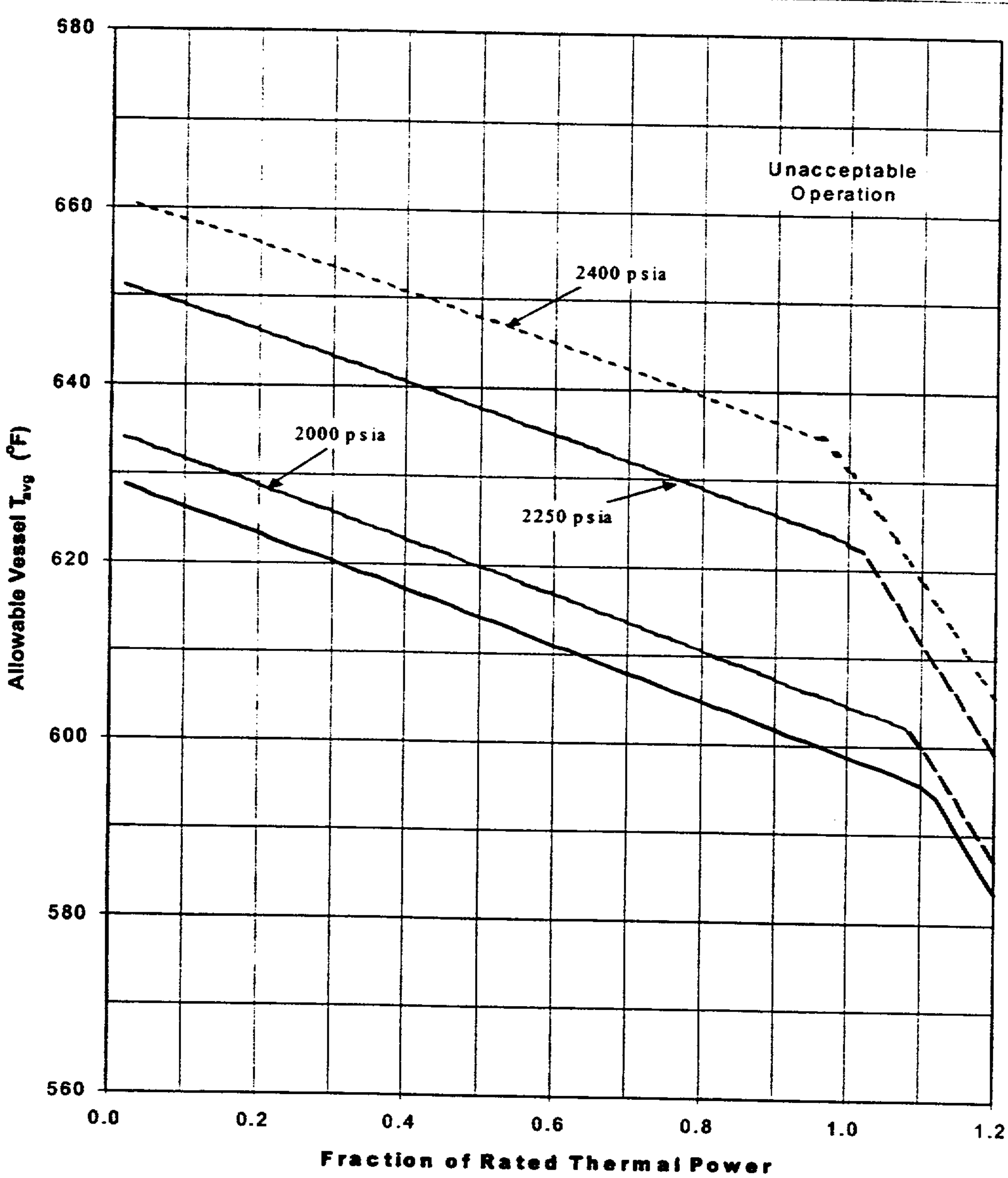

Figure 2.1
Reactor Core Safety Limits

2.2 Moderator Temperature Coefficient (MTC) (LCO 3.1.3, SR 3.1.3.2)

The MTC shall be less positive than the limit provided in Figure 2.2.

The MTC shall be less negative than -50 pcm/°F.

The 300 PPM MTC Surveillance limit is -41 pcm/°F (equilibrium, all rods withdrawn, RATED THERMAL POWER condition).

The 60 PPM MTC Surveillance limit is -46 pcm/°F (equilibrium, all rods withdrawn, RATED THERMAL POWER condition).

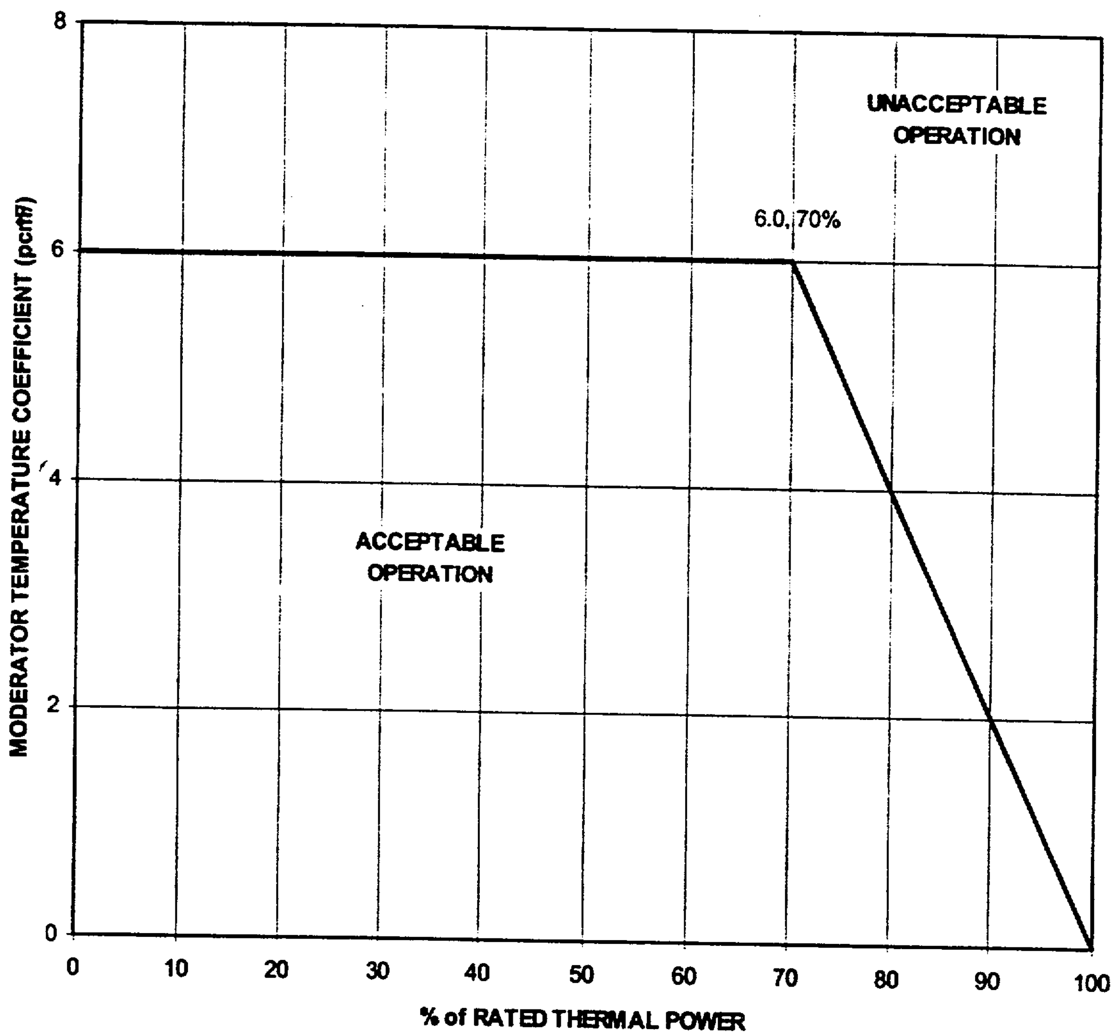

Figure 2.2
Moderator Temperature Coefficient Vs.
RATED THERMAL POWER

2.3 Shutdown Bank Insertion Limits (LCO 3.1.5)

The shutdown banks shall be fully withdrawn (i.e., positioned within the interval of ≥ 222 and ≤ 231 steps withdrawn).

2.4 Control Bank Insertion Limits (LCO 3.1.6)

The Control Bank insertion, sequence, and overlap limits are specified in Figure 2.4.

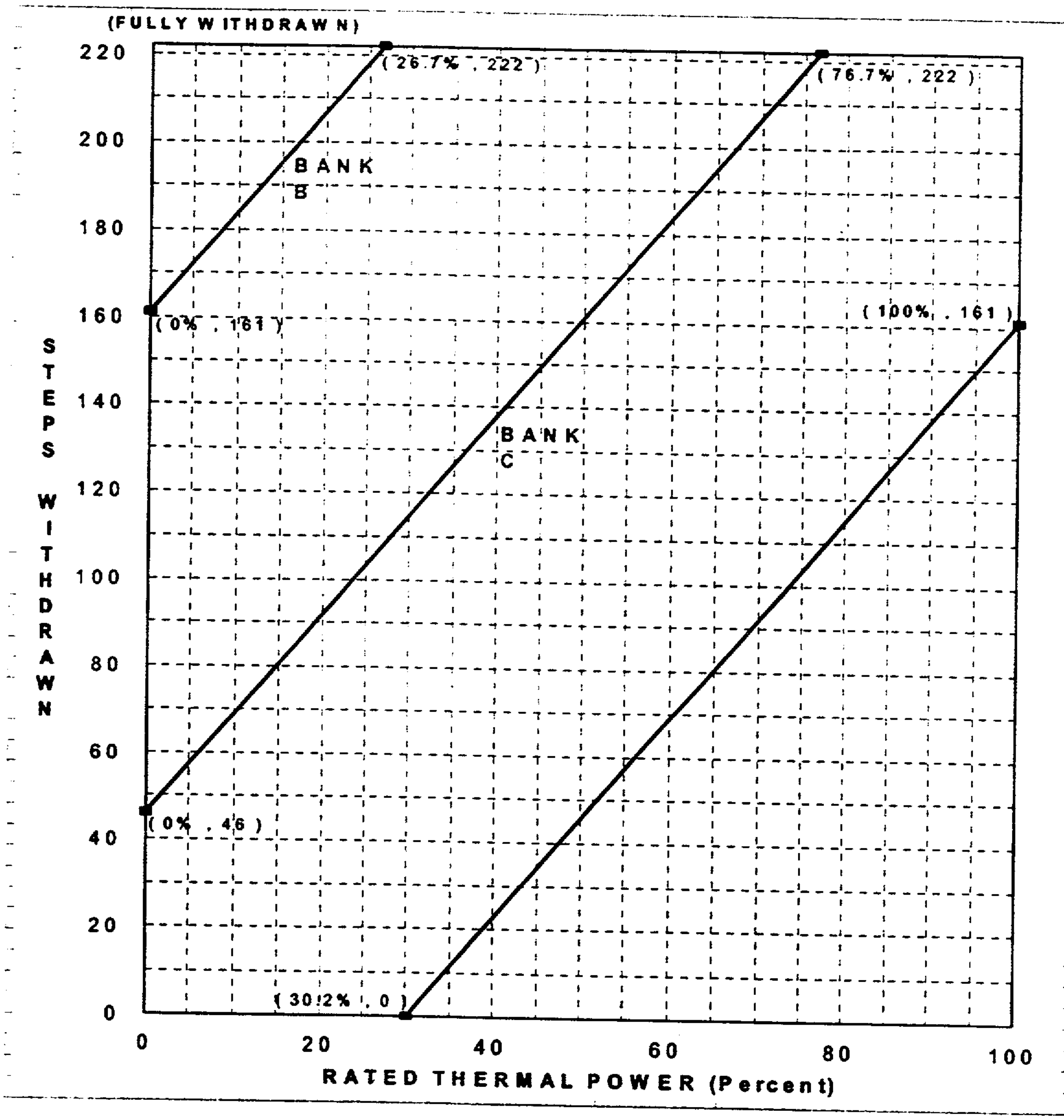

Figure 2.4
Control Bank Insertion, Sequence, and Overlap Limits Vs. THERMAL POWER - Four Loop Operation

Fully withdrawn shall be the condition where control banks are at a position within the interval of ≥ 222 and ≤ 231 steps withdrawn.

2.5 AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology) (LCO 3.2.3)

The indicated AXIAL FLUX DIFFERENCE (AFD) allowed operational space is defined by Figure 2.5.

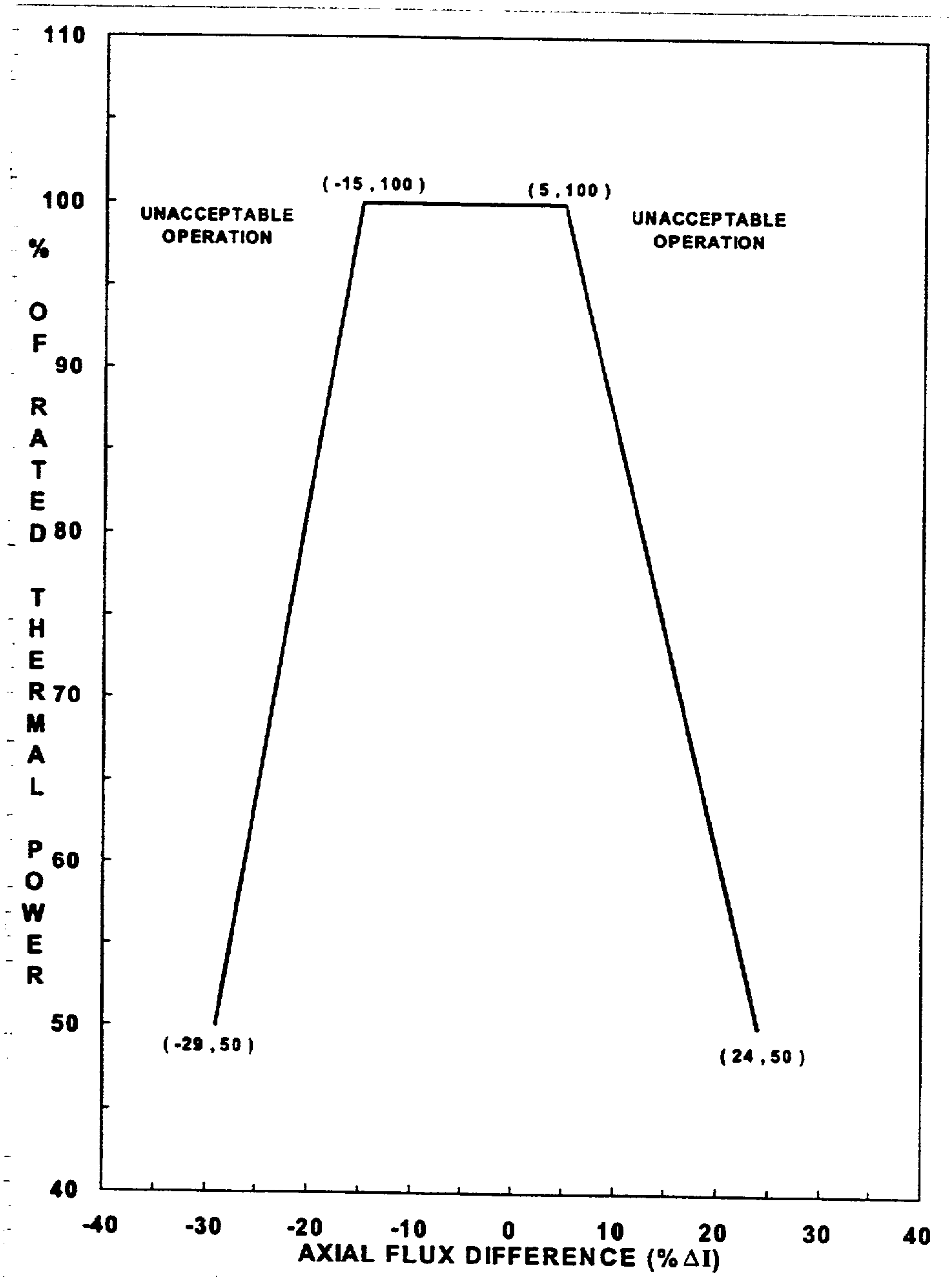

Figure 2.5
AXIAL FLUX DIFFERENCE Limits as a Function of RATED THERMAL POWER

2.6 Heat Flux Hot Channel Factor ($F_Q(Z)$)(F_Q Methodology) (LCO 3.2.1, SR 3.2.1.2)

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

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

where, $P = \dfrac{\text{THERMALPOWER}}{\text{RATEDTHERMALPOWER}}$

$CFQ = F_Q^{RTP}$

F_Q^{RTP} = $F_Q(Z)$ limit at RATED THERMAL POWER (RTP)

= 2.50, and

$K(Z)$ = as defined in Figure 2.6.

$$F_Q^C(Z) = F_Q^M(Z)(1.03)(1.05) = F_Q^M(Z)(1.0815)$$

where, $F_Q^M(Z)$ = Measured value of $F_Q(Z)$ from incore flux map

$$F_Q^W(Z) = F_Q^C(Z)W(Z)$$

where, $W(Z)$ = a cycle dependent function that accounts for power distribution transients encountered during normal operation (see Appendix A).

See Appendix A for:

F_Q Penalty Factor

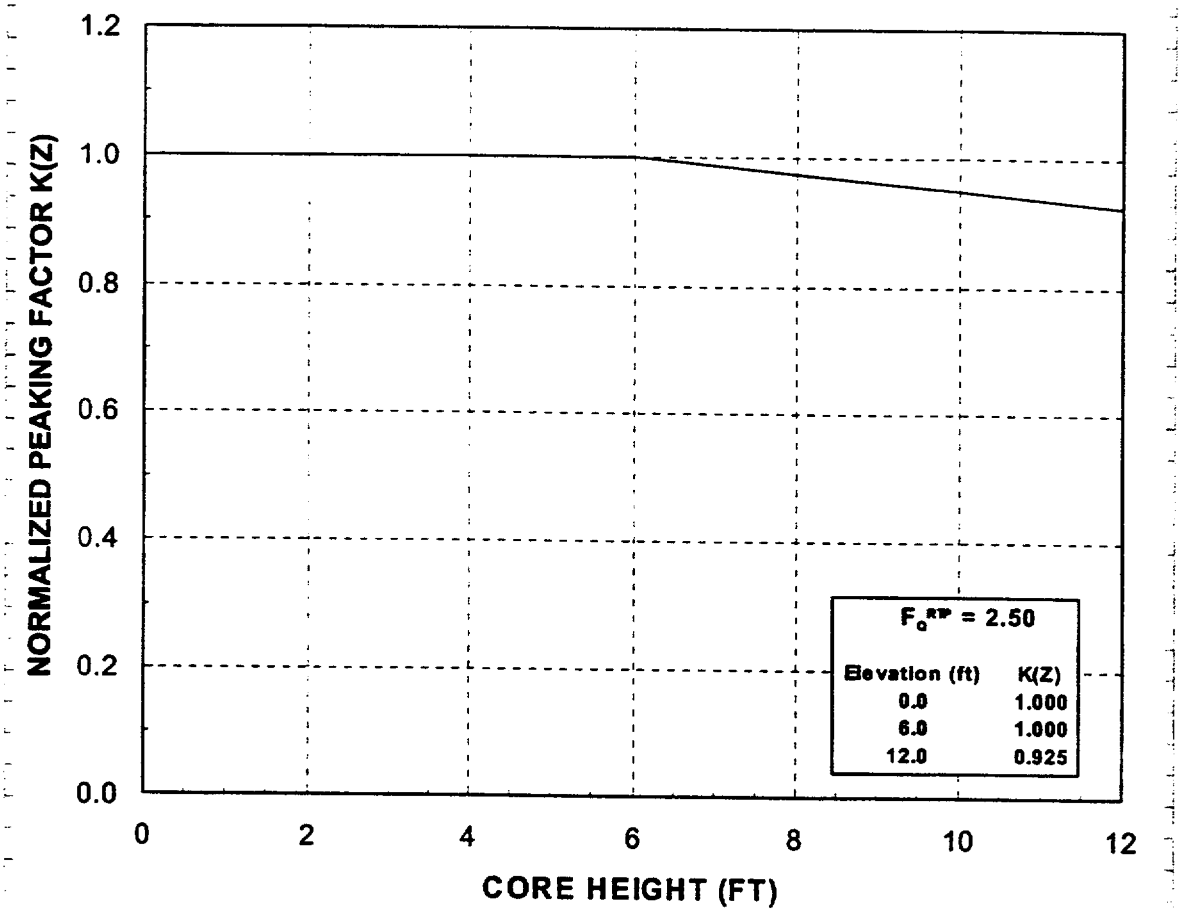

Figure 2.6
K(Z) - Normalized Peaking Factor Vs. Core Height

2.7 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) (LCO 3.2.2)

$F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq F^{RTP}_{\Delta H}\left[1.0 + PF_{\Delta H}(1.0 - P)\right]$$

Where, $F^{RTP}_{\Delta H}$ = $F^N_{\Delta H}$ limit at RATED THERMAL POWER (RTP)

= 1.586

$PF_{\Delta H}$ = power factor multiplier for $F^N_{\Delta H}$

= 0.3

P = $\dfrac{THERMAL\ POWER}{RATED\ THERMAL\ POWER}$

$F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used since an uncertainty of 4% for incore measurement of $F^N_{\Delta H}$ has been included in the above limit.

2.8 Reactor Trip System Overtemperature ΔT Setpoint Parameter Values (LCO 3.3.1)

Parameter	Value
Overtemperature ΔT reactor trip setpoint	$K_1 = 1.10$
Overtemperature ΔT reactor trip setpoint T_{avg} coefficient	$K_2 = 0.0137/°F$
Overtemperature ΔT reactor trip setpoint pressure coefficient	$K_3 = 0.000671/psig$
Nominal T_{avg} at RTP	$T' \leq 586.5°F$
Nominal RCS operating pressure	$P' \geq 2235$ psig
Measured RCS ΔT lead/lag constant	$\tau_1 = 6$ sec $\tau_2 = 3$ sec
Measured RCS ΔT lag constant	$\tau_3 = 2$ sec
Measured RCS average temperature lead/lag constant	$\tau_4 = 16$ sec $\tau_5 = 4$ sec
Measured RCS average temperature lead/lag constant	$\tau_6 = 0$ sec

$f_1(\Delta I) = -0.0227\{23\% + (q_t - q_b)\}$ when $(q_t - q_b) < -23\%$ RTP

0% of RTP when -23% RTP $\leq (q_t - q_b) \leq 5\%$ RTP

$0.0184\{(q_t - q_b) - 5\%\}$ when $(q_t - q_b) > 5\%$ RTP

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

2.9 Reactor Trip System Overpower ΔT Setpoint Parameter Values (LCO 3.3.2)

Parameter	Value
Overpower ΔT reactor trip setpoint	$K_4 = 1.10$
Overpower ΔT reactor trip setpoint T_{avg} rate/lag coefficient	$K_5 =$ 0.02/°F for increasing T_{avg} = 0/°F for decreasing T_{avg}
Overpower ΔT reactor trip setpoint T_{avg} heatup coefficient	$K_6 =$ 0.00128/°F for $T > T''$ = 0/°F for $T \leq T''$
Indicated T_{avg} at RTP (calibration temperature for ΔT instrumentation)	$T'' \leq 586.5$°F
Measured RCS ΔT lead/lag constant	$\tau_1 = 6$ sec $\tau_2 = 3$ sec
Measured RCS ΔT lag constant	$\tau_3 = 2$ sec
Measured RCS average temperature lead/lag constant	$\tau_6 = 0$ sec
Measured RCS average temperature rate/lag constant	$\tau_7 = 10$ sec

$f_2(\Delta I) =$ 0% RTP for all ΔI

2.10 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits (LCO 3.4.1)

Parameter	Indicated Value
Pressurizer pressure	Pressure ≥ 2220 psig
RCS average temperature	$T_{avg} \leq 590.5$ °F
RCS total flow rate	Flow ≥ 371,000 gpm

2.11 Boron Concentration (LCO 3.9.1)

The refueling boron concentration shall be greater than or equal to 2300 PPM.

2.12 SHUTDOWN MARGIN (LCO 3.1.1, 3.1.4, 3.1.5, 3.1.6, & 3.1.8)

The SHUTDOWN MARGIN shall be greater than or equal to 1300 pcm (1.3% $\Delta k/k$).

2.13 Departure from Nucleate Boiling Ratio (DNBR) Limits (B 3.4.1, ASA)

Safety Analysis DNBR Limit	1.76
WRB-2 Design Limit DNBR	1.23

APPENDIX A

A. Input relating to LCO 3.2.1:

$$W(Z) = \frac{F_Q(Z)^{\text{max transient}}}{F_Q(Z)^{\text{steady state}}}$$

These values are issued in a controlled report which will be provided on request.

Input relating to SR 3.2.1.2

Cycle Burnup	$F_Q^W(Z)$ Penalty Factor
0	2.00
21200	2.00

Note: All cycle burnups outside of the above table shall use a 2% penalty factor for compliance with SR 3.2.1.2. Linear interpolation should be used for intermediate cycle burnups.

B. <u>Approved Analytical Methods for Determining Core Operating Limits</u>

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

1. WCNOC Topical Report TR 90-0025 W01, "Core Thermal Hydraulic Analysis Methodology for the Wolf Creek Generating Station." (ET 90-0140, ET 92-0103)

   NRC Safety Evaluation Report dated October 29, 1992, for the "Core Thermal Hydraulic Analysis Methodology for the Wolf Creek Generating Station."

2. WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

   NRC Safety Evaluation Report dated January 17, 1989, for the "Acceptance for Referencing of Licensing Topical Report WCAP-11397, Revised Thermal Design Procedure."

3. WCNOC Topical Report NSAG-006, "Transient Analysis Methodology for the Wolf Creek Generating Station" (ET-91-0026, ET 92-0142, WM 93-0010, WM 93-0028).

   NRC Safety Evaluation Report dated September 30, 1993, for the "Transient Analysis Methodology for the Wolf Creek Generating Station."

4. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification," February 1994.

   NRC Safety Evaluation Report dated November 26, 1993, "Acceptance for Referencing of Revised Version of Licensing Topical Report WCAP-10216-P, Rev. 1, Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification" (TAC No. M88206).

5. WCNOC Topical Report NSAG-007, "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station" (ET 92-0032, ET 93-0017).

   NRC Safety Evaluation Report dated March 10, 1993, for the "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station."

6. NRC Safety Evaluation Report dated March 30, 1993, for the "Revision to Technical Specification for Cycle 7" (NA 92-0073, NA 93-0013, NA 93-0054).

7. WCAP-10266-P-A, Revision 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," March 1987.

   NRC letter dated November 13, 1986, "Acceptance for Referencing of Licensing Topical Report WCAP-10266 "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code.""

   WCAP-10266-P-A, Addendum 1, Revision 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code Addendum 1: Power Shape Sensitivity Studies," December 1987.

   NRC letter dated September 15, 1987, "Acceptance for Referencing of Addendum 1 to WCAP-10266, BASH Power Shape Sensitivity Studies."

   WCAP-10266-P-A, Addendum 2, Revision 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code Addendum 2: BASH Methodology Improvements and Reliability Enhancements," May 1988

   NRC letter dated January 20, 1988, "Acceptance for Referencing Topical Report Addendum 2 to WCAP-10266, Revision 2, "BASH Methodology Improvements and Reliability Enhancements."

8. WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988.

   NRC Safety Evaluation Report dated May 17, 1988, "Acceptance for Referencing of Westinghouse Topical Report WCAP-11596 - Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores."

9. WCAP 10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," September 1988.

   NRC letter dated June 23, 1986, "Acceptance for Referencing of Topical Report WCAP 10965-P and WCAP 10966-NP."

10. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995.

    NRC Safety Evaluation Reports dated July 1, 1991, "Acceptance for Referencing of Topical Report WCAP-12610, 'VANTAGE+ Fuel Assembly Reference Core Report' (TAC NO. 77258)."

    NRC Safety Evaluation Report dated September 15, 1994, "Acceptance for Referencing of Topical Report WCAP-12610, Appendix B, Addendum 1, 'Extended Burnup Fuel Design Methodology and ZIRLO Fuel Performance Models' (TAC NO. M86416)."

11. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Function." September 1986.

    NRC Safety Evaluation Report dated April 17, 1986, "Acceptance for Referencing of Licensing Topical Report WCAP-8745(P)/8746(NP), 'Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions.'"