WOLF CREEK
NUCLEAR OPERATING CORPORATION

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Manager Regulatory Affairs

November 4, 2015

RA 15-0081

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

Subject: Docket No. 50-482: Wolf Creek Generating Station Cycle 21 Core Operating Limits Report, Revision 1

Gentlemen:

Enclosed is Revision 1 of the Wolf Creek Generating Station Cycle 21 Core Operating Limits Report (COLR). Revision 1 incorporates changes associated with the implementation of Amendment No. 213. This document is being submitted pursuant to Section 5.6.5 of the Wolf Creek Generating Station Technical Specifications.

This letter contains no commitments. If you have any questions concerning this matter, please contact me at (620) 364-4204.

Sincerely,

Cynthia R. Hafenstine

Cynthia R. Hafenstine

CRH/rlt

Enclosure

cc: M. L. Dapas (NRC), w/e
C. F. Lyon (NRC), w/e
N. H. Taylor (NRC), w/e
Senior Resident Inspector (NRC), w/e

A001
NRR

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION
CYCLE 21

CORE OPERATING LIMITS REPORT
Revision 1

September 2015

Prepared by:	Jeff Blair	9/8/15 Date
Reviewed by:	Keith Colussy	9/9/15 Date
Approved by:	Gregory S. Kinn	9/17/15 Date

1.0 CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT (COLR) for Wolf Creek Generating Station Cycle 21 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 Alignment Limits

3.1.5 Shutdown Bank Insertion Limits

3.1.6 Control Bank Insertion Limits

3.1.8 PHYSICS TESTS Exceptions - MODE 2

3.2.1 Heat Flux Hot Channel Factor $(F_Q(Z))$ (F_Q Methodology)

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

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

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

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.

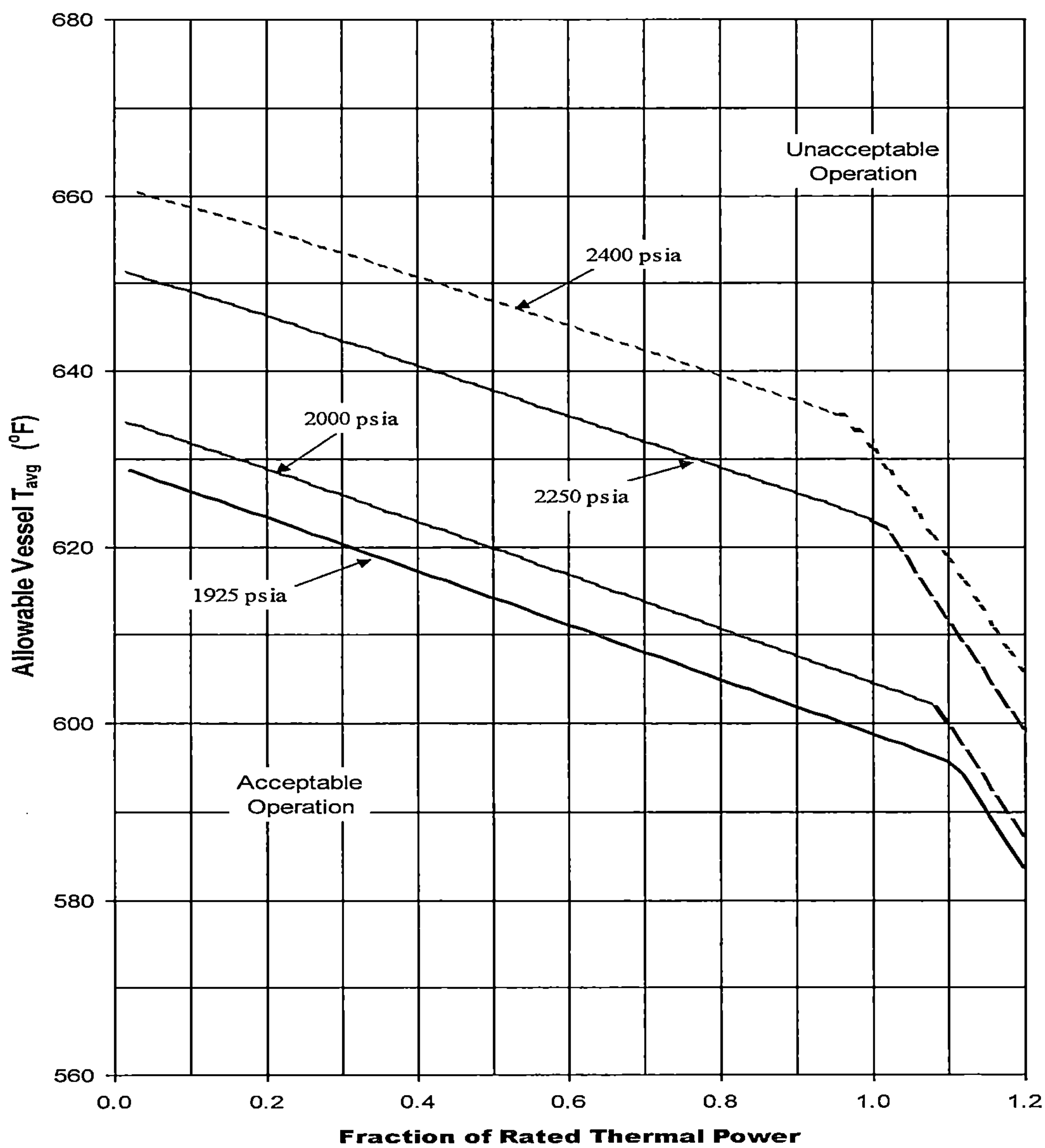

Figure 2.1
Reactor Core Safety Limits

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).

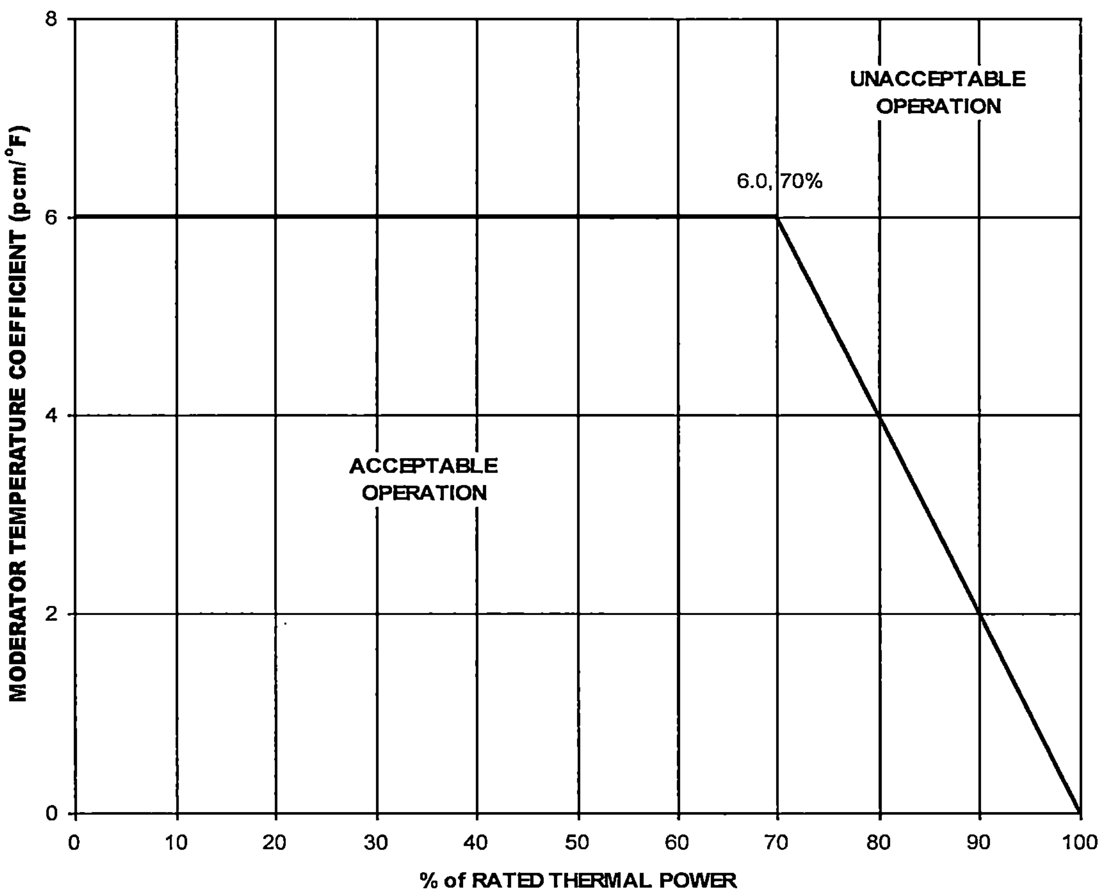

Figure 2.2
Moderator Temperature Coefficient Vs.
THERMAL POWER (%)

2.3 <u>Shutdown Bank Insertion Limits</u> (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 <u>Control Bank Insertion Limits</u> (LCO 3.1.6)

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

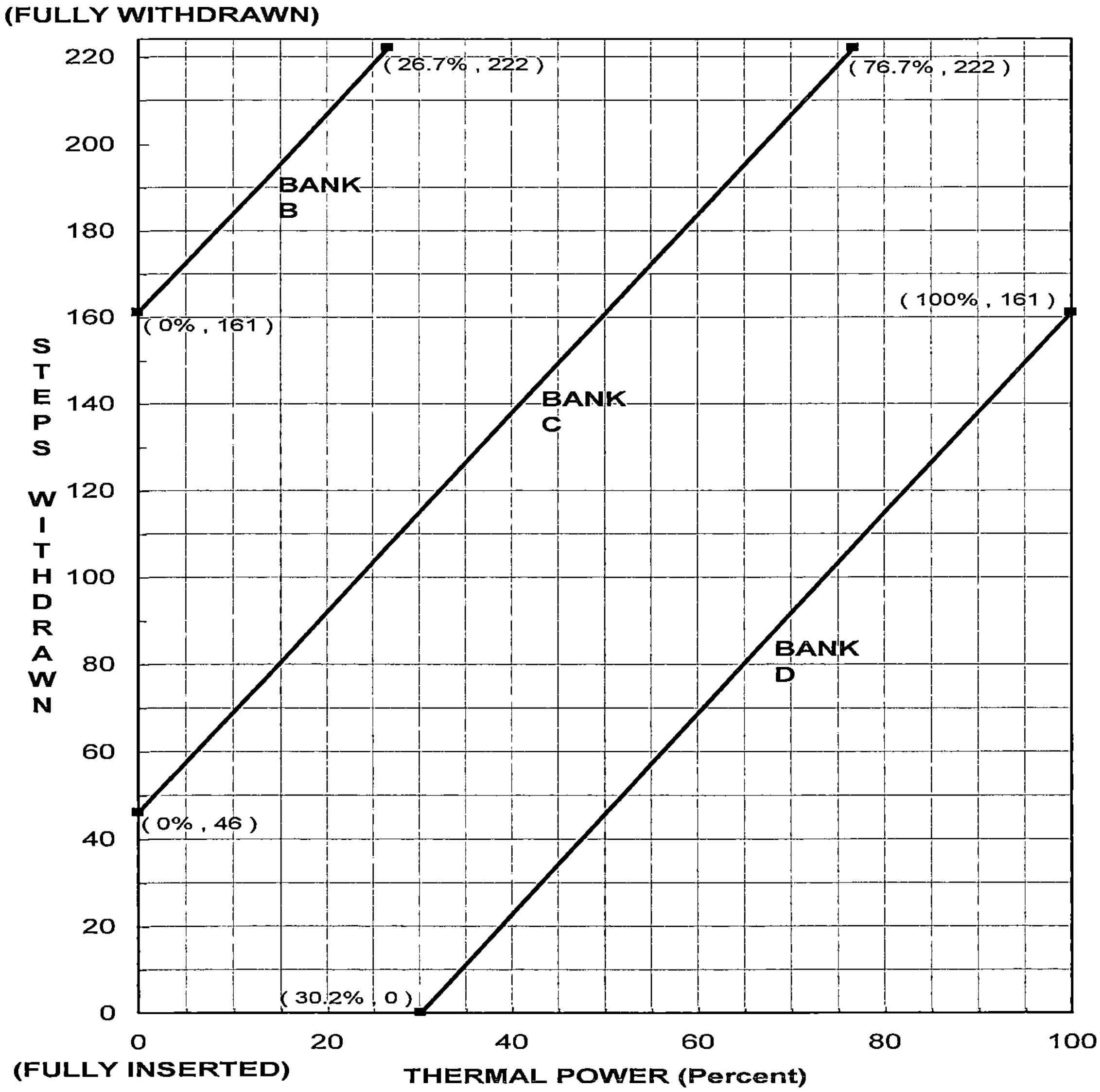

Figure 2.4
Control Bank Insertion, Sequence, and Overlap Limits Vs. THERMAL POWER (%) - Four Loop Operation

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.

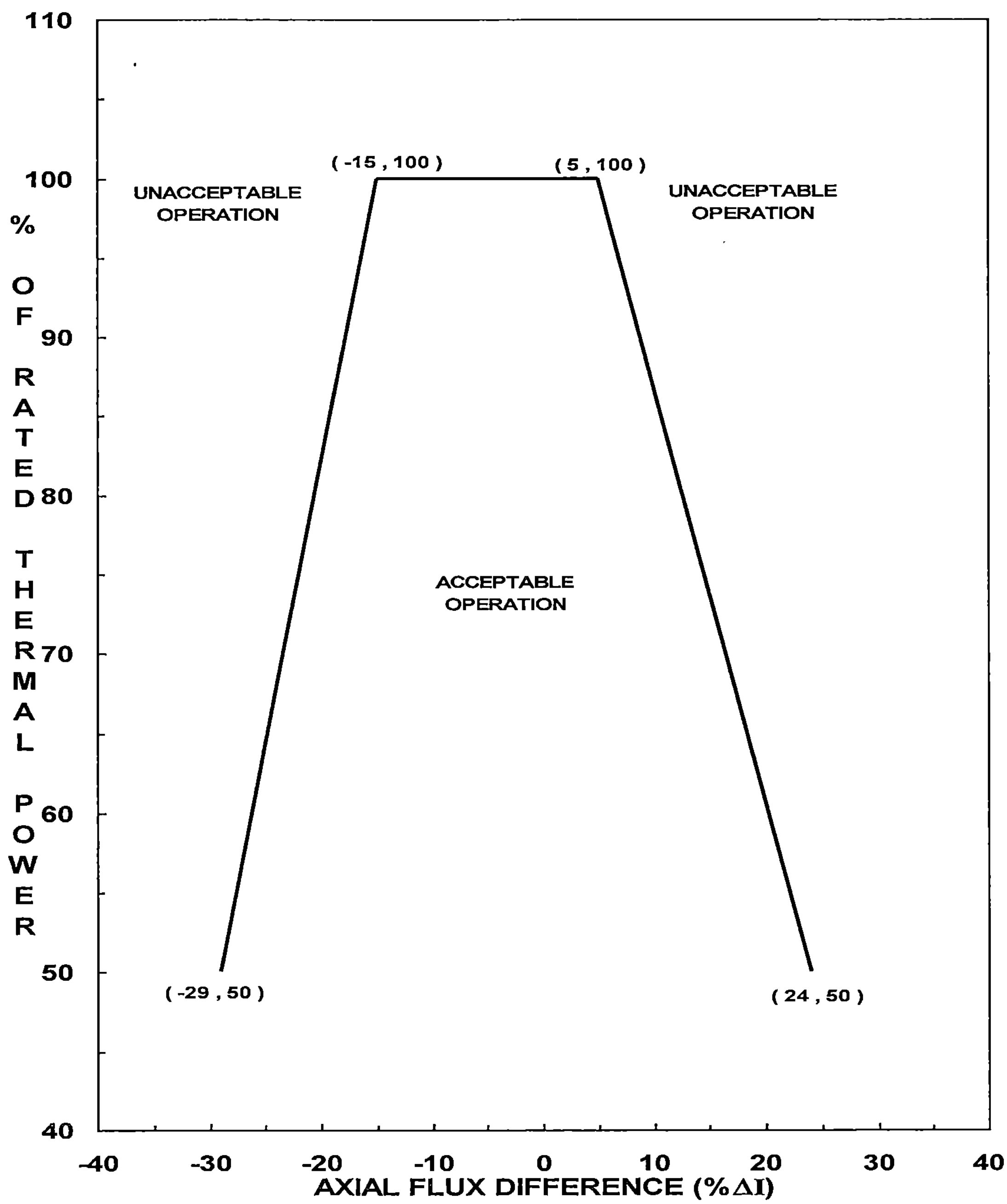

Figure 2.5
AXIAL FLUX DIFFERENCE Limits as a Function of THERMAL POWER (%)

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

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

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

where, P = $\dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

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.

$F_Q^M(Z)$ is the measured value of $F_Q(Z)$, inferred from a power distribution measurement obtained with the Movable Incore Detector System (MIDS) or the Power Distribution Monitoring System.

Measurement uncertainty is applied as follows.

$F_Q^C(Z) = F_Q^M(Z)(1.03)(1.05) = F_Q^M(Z)(1.0815)$ when $F_Q^M(Z)$ is obtained from MIDS.

$F_Q^C(Z) = F_Q^M(Z)(1.03)(U_{QU})$ when $F_Q^M(Z)$ is obtained from PDMS.

Manufacturing tolerances are accounted for in the 1.03 Engineering uncertainty factor. Measurement uncertainty for MIDS is accounted for in the 1.05 factor. PDMS measurement uncertainty is accounted for in the U_{QU} factor, and it is determined by PDMS.

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

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

When using the PDMS, $F_Q^W(Z)$ uses $F_Q^C(Z)$ that is determined from an $F_Q^M(Z)$ that reflects full-power steady-state conditions rather than current conditions.

See Appendix A for: F_Q Penalty Factor.

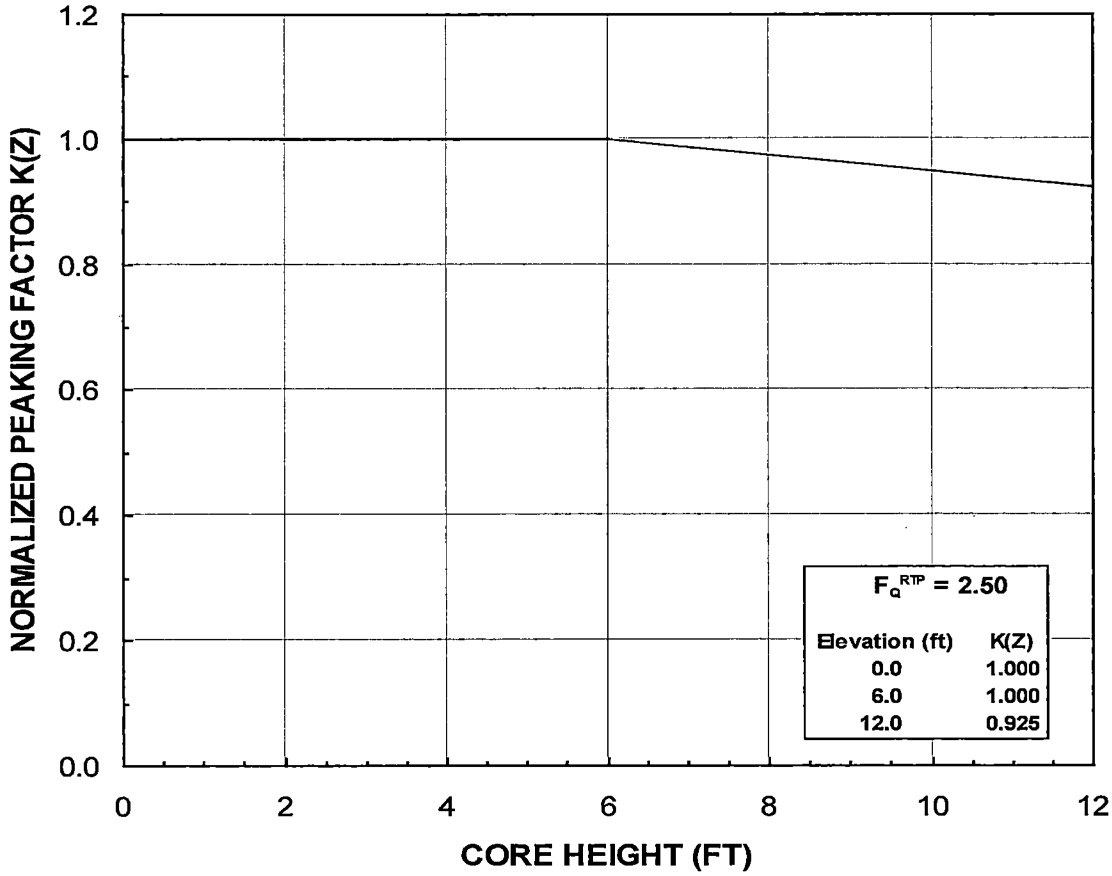

Figure 2.6
K(Z) - Normalized Peaking Factor Vs. Core Height

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

$F_{\Delta H}^{N}$ shall be limited by the following relationship:

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

Where, $F_{\Delta H}^{RTP}$ = $F_{\Delta H}^{N}$ limit at RATED THERMAL POWER (RTP)

= 1.650

$PF_{\Delta H}$ = power factor multiplier for $F_{\Delta H}^{N}$

= 0.3

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

$F_{\Delta H}^{N}$ = $F_{\Delta H}^{N}$ is the measured value of $F_{\Delta H}^{N}$, inferred from a power distribution measurement obtained with the Movable Incore Detector System (MIDS) or the Power Distribution Monitoring System (PDMS). Measurement uncertainty is applied as follows.

When $F_{\Delta H}^{N}$ is obtained from MIDS, the measured value is multiplied by 1.04.

When $F_{\Delta H}^{N}$ is obtained from PDMS, the measured value is increased by an uncertainty factor ($U_{\Delta H}$), and the factor is determined by PDMS, with a lower limit of 4%.

2.8 Reactor Trip System Overtemperature ΔT Setpoint Parameter Values (LCO 3.3.1, Table 3.3.1-1, Note 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 3.3.1, Table 3.3.1-1, Note 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} ≤ 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% Δk/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:

$$W(Z) = \frac{F_Q(Z)^{\text{max transient}}}{F_Q(Z)^{\text{steady state}}} \times \frac{1}{P}, \quad \textit{for } P > 0.5$$

$$W(Z) = \frac{F_Q(Z)^{\text{max transient}}}{F_Q(Z)^{\text{steady state}}} \times \frac{1}{0.5}, \quad \textit{for } P \leq 0.5$$

where, $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

$F_Q(Z)^{max\ transient}$ = Maximum ($F_Q(Z)\ x\ p$) calculated over the entire range of power shapes analyzed for Condition I operations (p = power at which maximum occurs).

$F_Q(Z)^{steady\ state}$ = ($F_Q(Z)\ x\ p$) calculated at full power (p = 1.0) equilibrium conditions.

The W(z) values are generated at full power equilibrium conditions (P = 1.0). W(z) values specific to part-power conditions may also be generated; these can be used for part-power surveillance measurements, rather than the full-power W(z) values. For these part-power W(z) values, the $F_Q(Z)^{steady\ state}$ (denominator in above equations) is generated at the specific anticipated surveillance conditions. W(Z) values are issued in controlled reports which will be provided on request.

Input relating to SR 3.2.1.2

Cycle Burnup (MWD/MTU)	$F_Q(Z)$ Penalty Factor (%)
≥ 0 to ≤ 150	3.08
348	2.76
546	2.38
743 to 8456	2.00
8654	2.06
8852	2.26
9049	2.44
9247	2.40
9445	2.38
9643	2.36
9840	2.31
10038	2.23
10236	2.15
10434	2.06
≥ 10631	2.00

Cycle Burnup (MWD/MTU)	$F_Q(Z)$ Exclusion Zone (% [INCORE mesh points])	
	Top	Bottom
≤ 8,000	15 [11]	15 [11]
> 8,000	10 [7]	10 [7]

B. Approved Analytical Methods for Determining Core Operating Limits

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."

   EPRI Topical Report NP-7450(A), "RETRAN-3D – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," including NRC Safety Evaluation Report dated January 25, 2001, "Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," (TAC No. MA4311)." RETRAN-3D code is only utilized in the RETRAN-02 mode.

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-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)," Revision 0, January 2005.

   NRC letter dated November 5, 2004,"Final Safety Evaluation for WCAP-16009-P, Revision 0, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)" (TAC NO. MB9483)."

8. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004.

   NRC Safety Evaluation dated March 18, 2004, "Final Safety Evaluation for Westinghouse Topical Report WCAP-16045-P, Revision 0, "Qualification of the Two-Dimensional Transport Code PARAGON."

9. WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology," August 2007.

   NRC Safety Evaluation dated February 23, 2007, "Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) WCAP-16045-P-A, Addendum 1, "Qualification of the NEXUS Nuclear Data Methodology" (TAC NO. MC9606)."

10. WCAP 10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," September 1986.

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

11. 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)."

12. 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.'"