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Kewaunee Nuclear Power Plant
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Kewaunee Nuclear Power Plant – Core Operating Limits Report (COLR),
Cycle 27, Revision 0

In accordance with the requirements of Technical Specification 6.9.a.4, enclosed is Revision 0 of the Cycle 27, Core Operating Limits Report (COLR).

If you have any questions, please contact Mr. Gerald Riste at (920) 388-8424.

This letter contains no new commitments and no revisions to existing commitments.

Thomas Coutu
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Enclosure

cc: Administrator, Region III, USNRC
Senior Resident Inspector, Kewaunee, USNRC
Project Manager, Kewaunee, USNRC
Public Service Commission of Wisconsin

ADD 1

TRM 2.1

Kewaunee Nuclear Power Plant

CORE OPERATING LIMITS REPORT
(COLR)

CYCLE 27

REVISION 0

Approved


PORC Chairman

10/12/01
Date

04-095
Mtg.#

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CORE OPERATING LIMITS REPORT CYCLE 27

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Kewaunee Nuclear Power Plant (KNPP) has been prepared in accordance with the requirements of Technical Specification (TS) 6.9.a.4.

A cross-reference between the COLR sections and the KNPP Technical Specifications affected by this report is given below:

COLR Section	KNPP TS	Description
2.1	2.1	Reactor Core Safety Limits
2.2	3.10.a	Shutdown Margin
2.3	3.1.f.3	Moderator Temperature Coefficient
2.4	3.10.d.1	Shutdown Bank Insertion Limit
2.5	3.10.d.2	Control Bank Insertion Limits
2.6	3.10.b.1.A	Heat Flux Hot Channel Factor ($F_Q(Z)$)
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	3.10.b.6.C.i	
2.7	3.10.b.1.B	Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)
2.8	3.10.b.8	Axial Flux Difference (AFD)
2.9	2.3.a.3.A	Overtemperature ΔT Setpoint
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2.11	3.10.k	RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling
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	3.10.m.1	
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Figure 6		Penalty Factor, F_p , for $F_Q^{EO}(Z)$
Figure 7		Axial Flux Difference

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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 6.9.a.4.

2.1 Reactor Core Safety Limits

The combination of rated power level, coolant pressure, and coolant temperature shall not exceed the limits shown in COLR Figure 1 (1772 MWt). The safety limit is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

2.2 Shutdown Margin

2.2.1 When the reactor is subcritical prior to reactor startup, the SHUTDOWN margin shall be at least that shown in COLR Figure 2.

2.3 Moderator Temperature Coefficient

2.3.1 When the reactor is critical and $\leq 60\%$ RATED POWER, the moderator temperature coefficient shall be ≤ 5.0 pcm/ $^{\circ}$ F, except during LOW POWER PHYSICS TESTING. When the reactor is $> 60\%$ RATED POWER, the moderator temperature coefficient shall be zero or negative.

2.3.2 The reactor will have a moderator temperature coefficient no less negative than -8 pcm/ $^{\circ}$ F for 95% of the cycle time at full power.

2.4 Shutdown Bank Insertion Limit

2.4.1 The shutdown rods shall be fully withdrawn (≥ 226 steps and ≤ 231 steps) when the reactor is critical or approaching criticality.

2.5 Control Bank Insertion Limit

2.5.1 The control banks shall be limited in physical insertion; insertion limits are shown in COLR Figure 4.

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2.6 Nuclear Heat Flux Hot Channel Factor ($F_Q^N(Z)$)

2.6.1 $F_Q^N(Z)$ Limits for Fuel

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.35)/P \times K(Z) \text{ for } P > 0.5 \quad \text{[FRA-ANP Hvy]}$$

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (4.70) \times K(Z) \text{ for } P \leq 0.5 \quad \text{[FRA-ANP Hvy]}$$

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.28)/P \times K(Z) \text{ for } P > 0.5 \quad \text{[FRA-ANP Std]}$$

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (4.56) \times K(Z) \text{ for } P \leq 0.5 \quad \text{[FRA-ANP Std]}$$

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.50)/P \times K(Z) \text{ for } P > 0.5 \quad \text{[422 V+]}$$

$$F_Q^N(Z) \times 1.03 \times 1.05 \leq (5.00) \times K(Z) \text{ for } P \leq 0.5 \quad \text{[422 V+]}$$

where:

P is the fraction of full power at which the core is OPERATING

K(Z) is the function given in Figure 3

Z is the core height location for the F_Q of interest

2.6.2 The measured $F_Q^{EQ}(Z)$ hot channel factors under equilibrium conditions shall satisfy the following relationship for the central axial 80% of the core for fuel:

$$F_Q^{EQ}(Z) \times 1.03 \times 1.05 \times W(Z) \times F_p \leq (2.35)/P \times K(Z) \quad \text{[FRA-ANP Hvy]}$$

$$F_Q^{EQ}(Z) \times 1.03 \times 1.05 \times W(Z) \times F_p \leq (2.28)/P \times K(Z) \quad \text{[FRA-ANP Std]}$$

$$F_Q^{EQ}(Z) \times 1.03 \times 1.05 \times W(Z) \times F_p \leq (2.5)/P \times K(Z) \quad \text{[422 V+]}$$

where:

P is the fraction of full power at which the core is OPERATING

K(Z) is the function given in Figure 3

Z is the core height location for the F_Q of interest

F_p is the $F_Q^{EQ}(Z)$ penalty factor described in 2.6.3.

W(Z) is the function given in Figure 5

$F_Q^{EQ}(Z)$ is a measured F_Q distribution obtained during the target flux determination

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2.6.3 The penalty factor of 1.0 shall be used for TS 3.10.b.6.A and TS 3.10.b.6.B. The penalty factor provided in Figure 6 shall be used for TS 3.10.b.6.C.i. The penalty factor for all burnups outside the range of Figure 6 shall be 2%.

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

2.7.1 $F_{\Delta H}^N$ Limits for Fuel

$$F_{\Delta H}^N \times 1.04 \leq 1.38 [1 + 0.3(1-P)] \quad \text{[FRA-ANP Hvy]}$$

$$F_{\Delta H}^N \times 1.04 \leq 1.70 [1 + 0.3(1-P)] \quad \text{[422 V+]}$$

$$F_{\Delta H}^N \times 1.04 \leq 1.38 [1 + 0.3(1-P)] \quad \text{[FRA-ANP Std]}$$

where:

P is the fraction of full power at which the core is OPERATING

2.8 Axial Flux Difference (AFD)

2.8.1 The Axial Flux Difference (AFD) acceptable operation limits are provided in Figure 7.

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2.9 Overtemperature ΔT Setpoint

Overtemperature ΔT setpoint parameter values:

- ΔT_0 = Indicated ΔT at RATED POWER, %
T = Average temperature, °F
T' \leq 573.0 °F
P = Pressurizer Pressure, psig
P' = 2235 psig
K₁ = 1.195
K₂ = 0.015/°F
K₃ = 0.00072/psig
 τ_1 = 30 seconds
 τ_2 = 4 seconds
f(ΔI) = An even function of the indicated difference between top and bottom detectors of the power range nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RATED POWER, such that
- (a) For $q_t - q_b$ within -15, +10 %, $f(\Delta I) = 0$
 - (b) For each percent that the magnitude of $q_t - q_b$ exceeds +10 % the ΔT trip setpoint shall be automatically reduced by an equivalent of 1.51 % of RATED POWER.
 - (c) For each percent that the magnitude of $q_t - q_b$ exceed -15 % the ΔT trip setpoint shall be automatically reduced by an equivalent of 3.78% of RATED POWER.

2.10 Overpower ΔT Setpoint

Overpower ΔT setpoint parameter values:

- ΔT_0 = Indicated ΔT at RATED POWER, %
T = Average temperature, °F
T' \leq 573.0 °F
K₄ \leq 1.095
K₅ \geq 0.0275/°F for increasing T; 0 for decreasing T
K₆ \geq 0.00103/°F for T > T' ; 0 for T < T'
 τ_3 = 10 seconds
f(ΔI) = 0 for all ΔI

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2.11 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

2.11.1 During steady state power operation, T_{avg} shall be $< 576.7^{\circ}\text{F}$ for control board indication or $< 576.5^{\circ}\text{F}$ for computer indication.

2.11.2 During steady state power operation, Pressurizer Pressure shall be > 2217 psig for control board indication or > 2219 psig for computer indication

2.11.3 During steady state power operation, reactor coolant total flow rate shall be $\geq 186,000$ gpm.

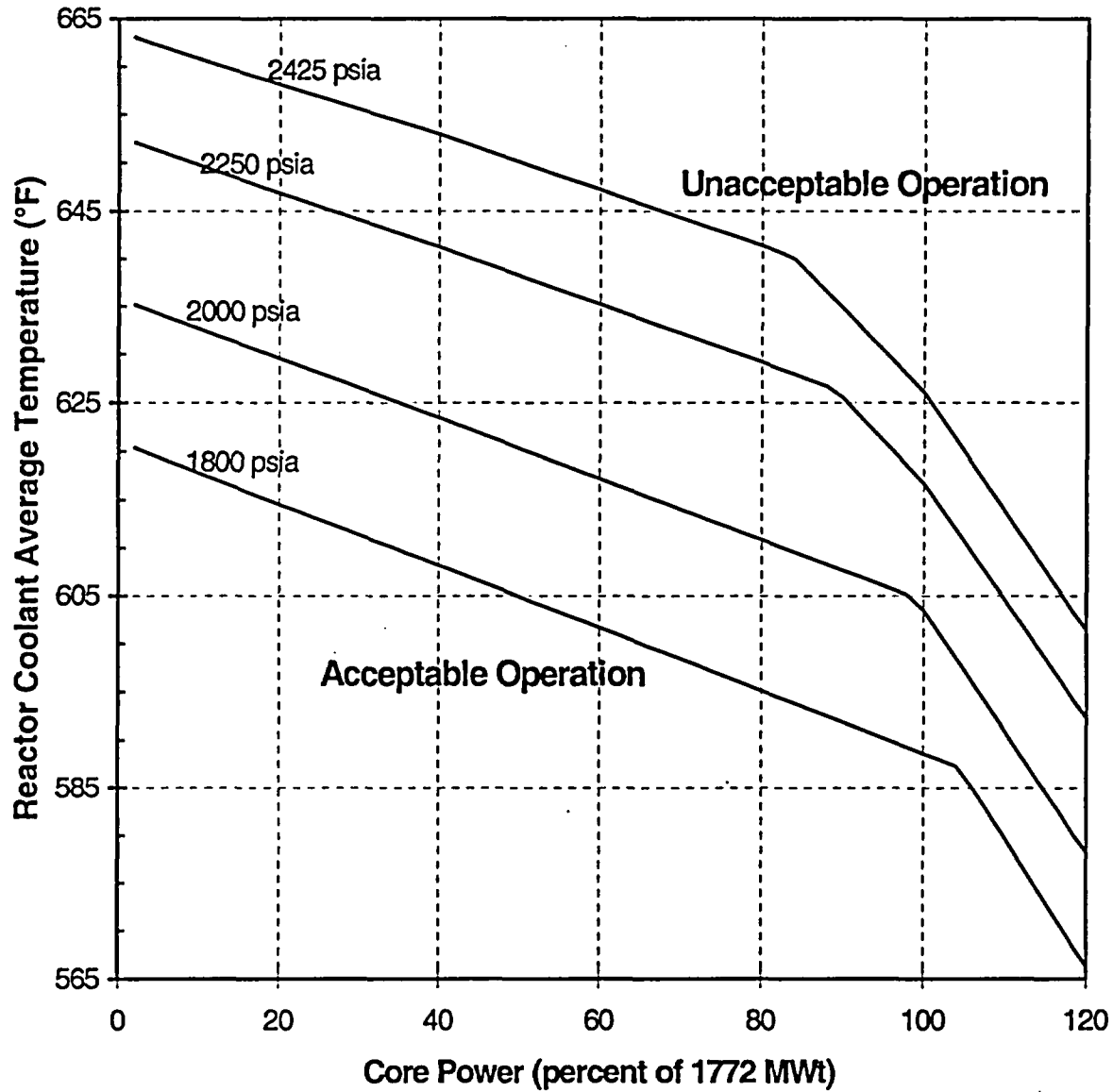
2.12 Refueling Boron Concentration

2.12.1 When there is fuel in the reactor, a minimum boron concentration of 2500 ppm and a shutdown margin of $\geq 5\% \Delta k/k$ shall be maintained in the Reactor Coolant System during reactor vessel head removal or while loading and unloading fuel from the reactor.

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Figure 1

Reactor Core Safety Limits Curve (1772 Mwt)
(Cores Containing 422V+ fuel)



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Figure 2
Required Shutdown Reactivity vs. Boron
Concentration

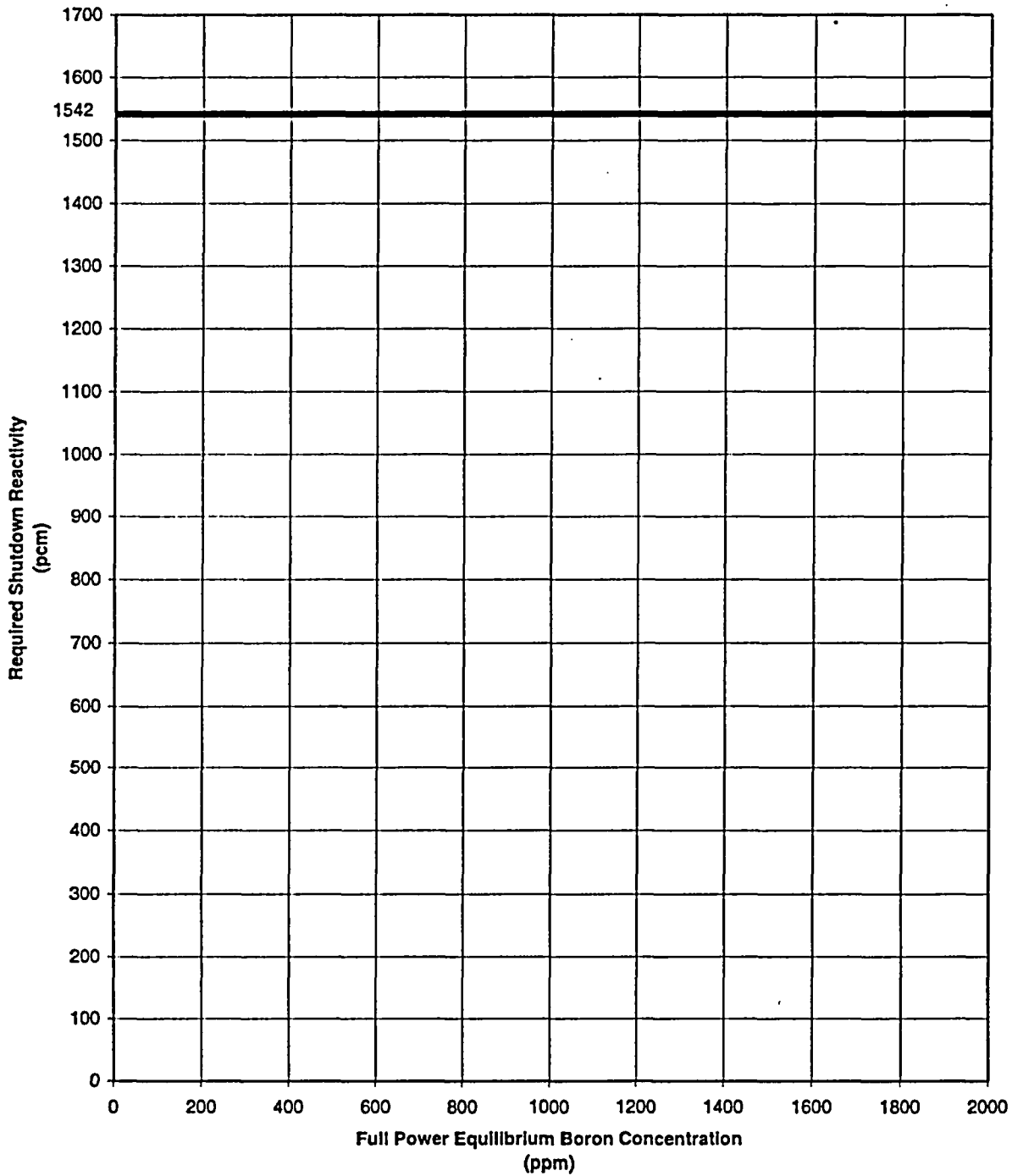
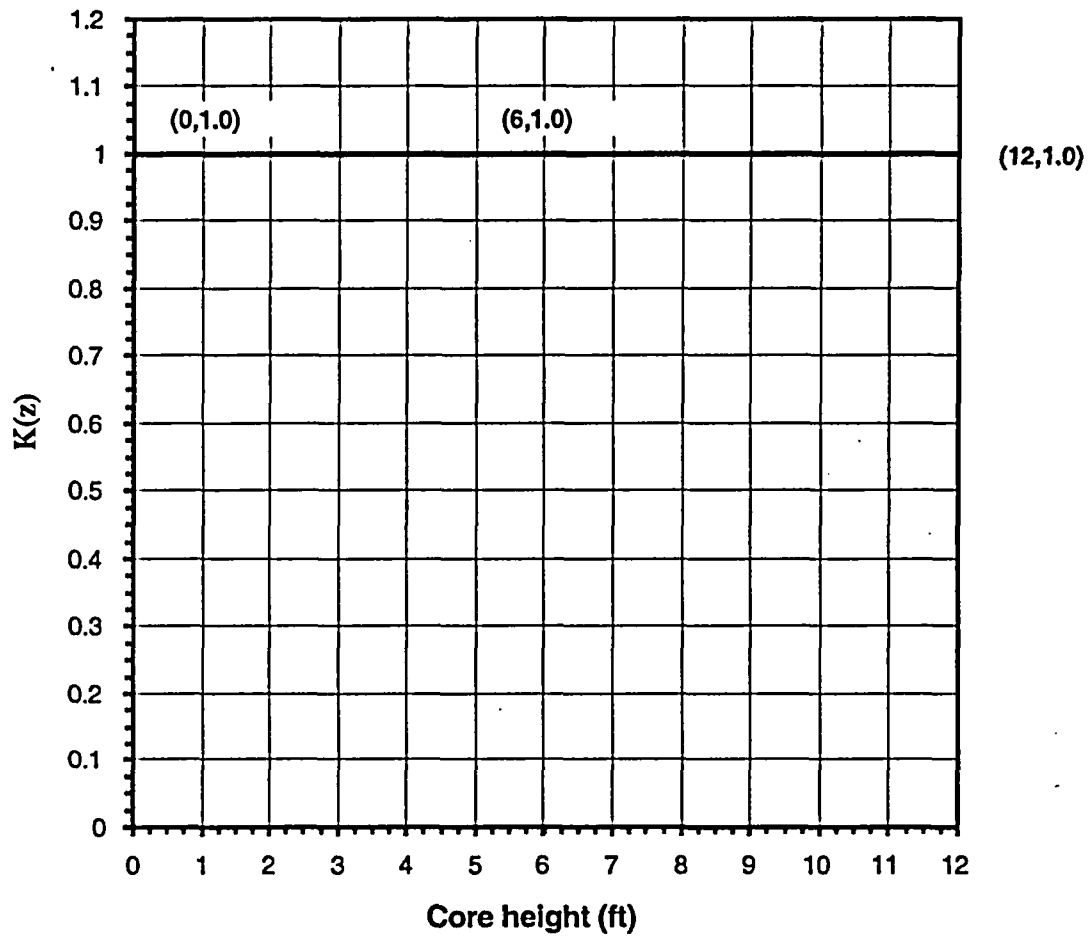
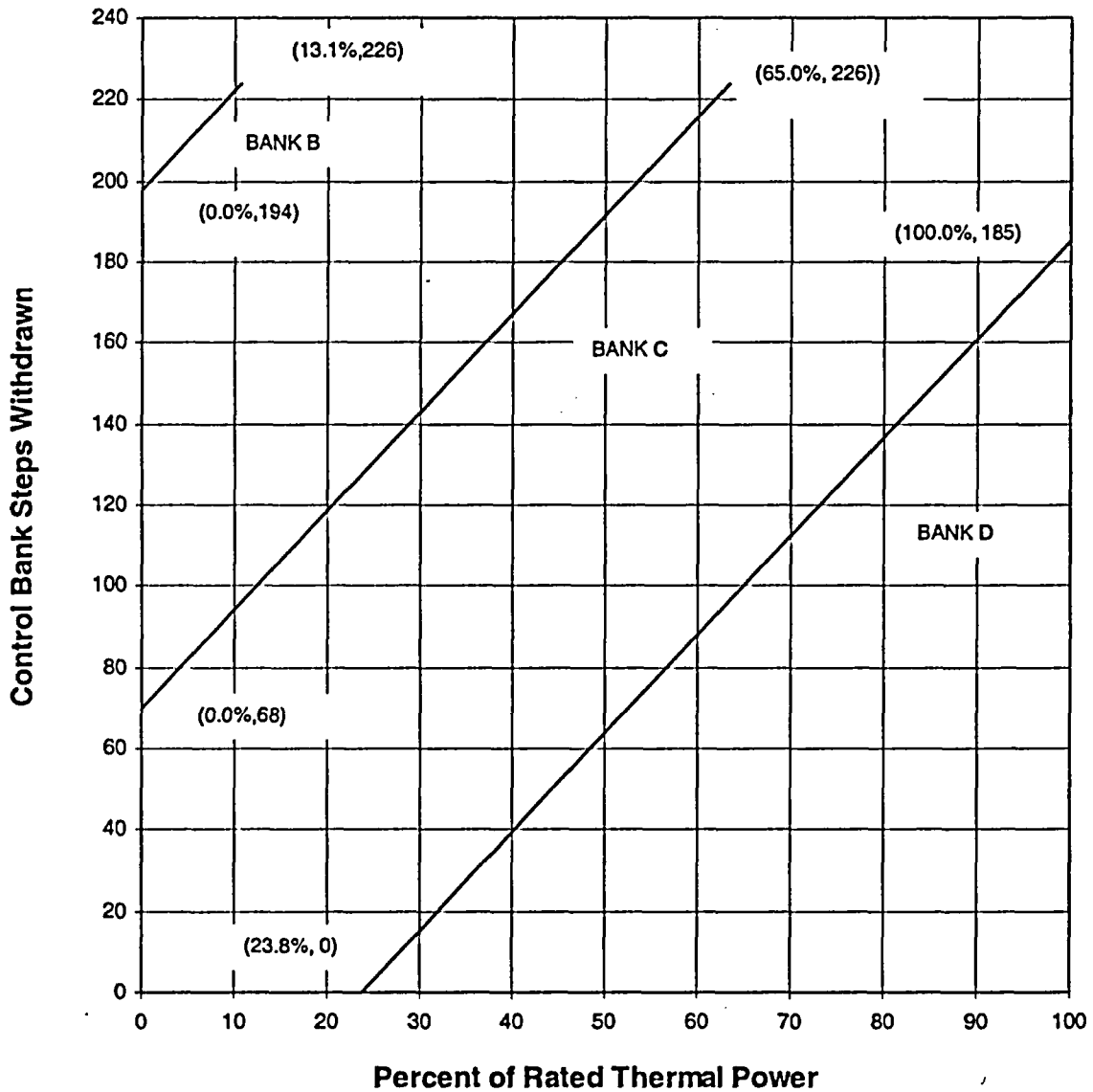


Figure 3
Hot Channel Factor Normalized Operating Envelope



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Figure 4
 Control Bank Insertion Limits



Fully withdrawn shall be the condition where control rods are at a position within the interval ≥ 226 and ≤ 231 steps withdrawn.

Note: The Rod Bank Insertion Limits are based on a control bank tip-to-tip distance of 126 steps.

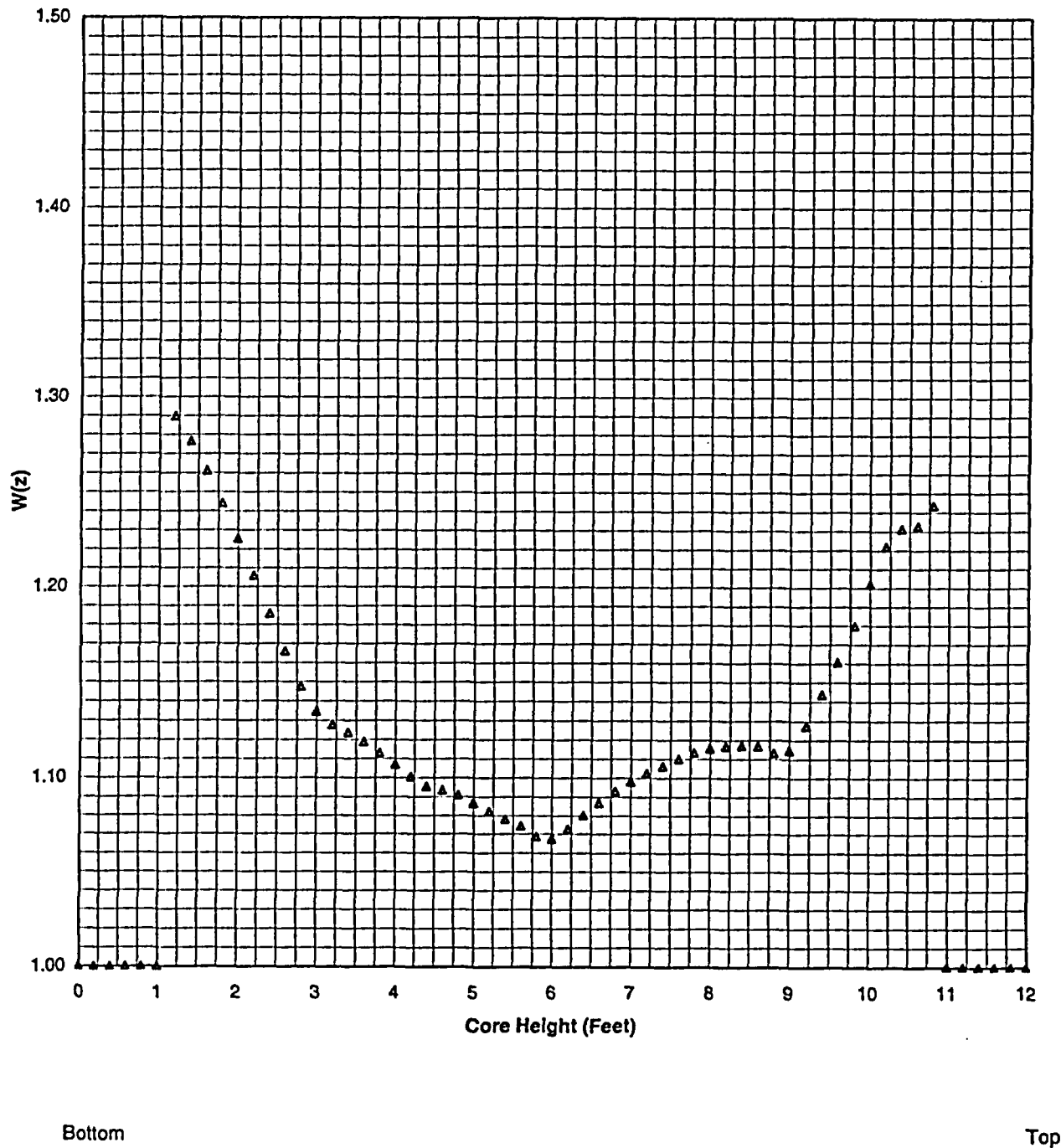
CORE OPERATING LIMITS REPORT CYCLE 27

Figure 5
 Composite RAOC Summary

Core Height (ft)	150 MWD/MTU	4000 MWD/MTU	8000 MWD/MTU	10000 MWD/MTU	12000 MWD/MTU	16000 MWD/MTU
0.0	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
0.2	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
0.4	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
0.6	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
0.8	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
1.0	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
1.2	1.2898	1.2165	1.1548	1.1514	1.1411	1.1855
1.4	1.2767	1.2053	1.1451	1.1418	1.1317	1.1765
1.6	1.2613	1.1923	1.1339	1.1310	1.1212	1.1665
1.8	1.2441	1.1778	1.1217	1.1190	1.1097	1.1555
2.0	1.2255	1.1622	1.1085	1.1062	1.0975	1.1436
2.2	1.2060	1.1457	1.0972	1.0931	1.0849	1.1312
2.4	1.1859	1.1291	1.0895	1.0797	1.0724	1.1198
2.6	1.1660	1.1127	1.0815	1.0662	1.0603	1.1119
2.8	1.1477	1.0951	1.0730	1.0588	1.0473	1.1041
3.0	1.1347	1.0859	1.0674	1.0562	1.0421	1.0967
3.2	1.1277	1.0849	1.0652	1.0548	1.0419	1.0925
3.4	1.1236	1.0830	1.0639	1.0542	1.0457	1.0922
3.6	1.1188	1.0808	1.0623	1.0570	1.0493	1.0927
3.8	1.1131	1.0783	1.0603	1.0612	1.0526	1.0928
4.0	1.1069	1.0750	1.0580	1.0652	1.0557	1.0924
4.2	1.1005	1.0732	1.0583	1.0691	1.0587	1.0917
4.4	1.0955	1.0740	1.0634	1.0735	1.0623	1.0915
4.6	1.0935	1.0752	1.0685	1.0779	1.0672	1.0929
4.8	1.0910	1.0757	1.0726	1.0816	1.0719	1.0965
5.0	1.0869	1.0749	1.0757	1.0840	1.0756	1.0996
5.2	1.0823	1.0741	1.0778	1.0853	1.0782	1.1011
5.4	1.0781	1.0748	1.0795	1.8610	1.0804	1.1020
5.6	1.0742	1.0762	1.0807	1.0871	1.0825	1.1022
5.8	1.0688	1.0793	1.0838	1.0912	1.0830	1.1047
6.0	1.0674	1.0834	1.0860	1.0963	1.0868	1.1111
6.2	1.0727	1.0873	1.0900	1.1004	1.0963	1.1184
6.4	1.0802	1.0818	1.1025	1.1069	1.1069	1.1244
6.6	1.0870	1.0988	1.1141	1.1193	1.1167	1.2950
6.8	1.0930	1.1056	1.1245	1.1323	1.1254	1.1331
7.0	1.0981	1.1111	1.1341	1.1439	1.1336	1.1366
7.2	1.1024	1.1160	1.1431	1.1546	1.1417	1.1405
7.4	1.1062	1.1215	1.1527	1.1641	1.1496	1.1429
7.6	1.1101	1.1262	1.1612	1.1723	1.1563	1.1440
7.8	1.1134	1.1298	1.1684	1.1790	1.1616	1.1437
8.0	1.1156	1.1323	1.1743	1.1843	1.1657	1.1420
8.2	1.1168	1.1337	1.1790	1.1880	1.1684	1.1391
8.4	1.1171	1.1341	1.1825	1.1902	1.1700	1.1349
8.6	1.1168	1.1339	1.1847	1.1911	1.1697	1.1301
8.8	1.1133	1.1291	1.1837	1.1871	1.1700	1.1321
9.0	1.1146	1.1302	1.1831	1.1873	1.1726	1.1380
9.2	1.1275	1.1449	1.1875	1.1980	1.1821	1.1412
9.4	1.1438	1.1592	1.1952	1.2083	1.1950	1.1443
9.6	1.1607	1.1741	1.2027	1.2222	1.2085	1.1468
9.8	1.1799	1.1893	1.2122	1.2371	1.2216	1.1511
10.0	1.2020	1.2037	1.2235	1.2477	1.2330	1.1653
10.2	1.2219	1.2171	1.2337	1.2525	1.2437	1.1889
10.4	1.2312	1.2292	1.2425	1.2561	1.2585	1.2093
10.6	1.2328	1.2396	1.2568	1.2594	1.2742	1.2255
10.8	1.2435	1.2493	1.2683	1.2654	1.2894	1.2358
11.0	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
11.2	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
11.4	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
11.6	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
11.8	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
12.0	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000

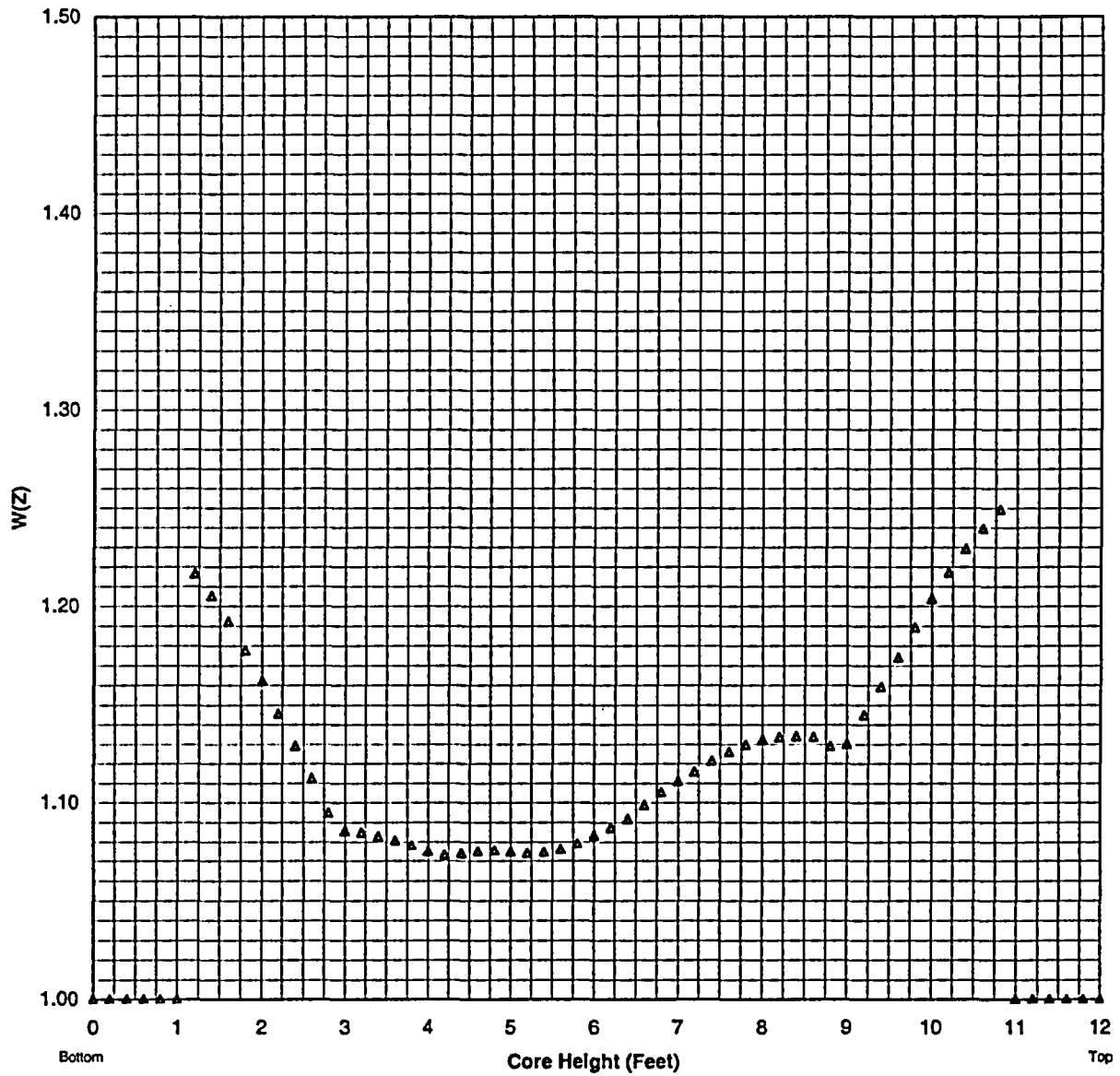
CORE OPERATING LIMITS REPORT CYCLE 27

Figure 5a
RAOC Summary of W(Z) at 150 MWD/MTU



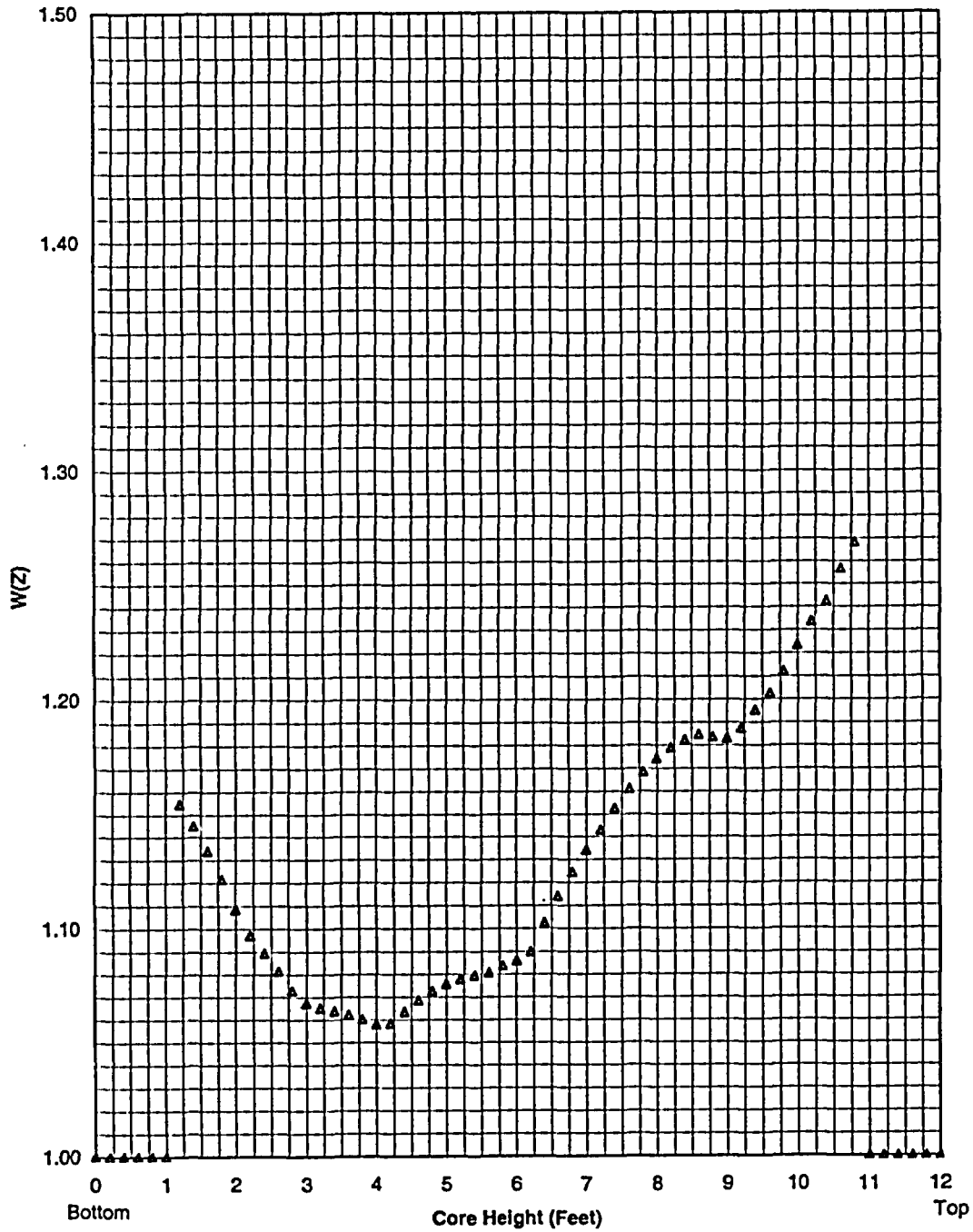
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Figure 5b
RAOC Summary of W(Z) at 4000 MWD/MTU



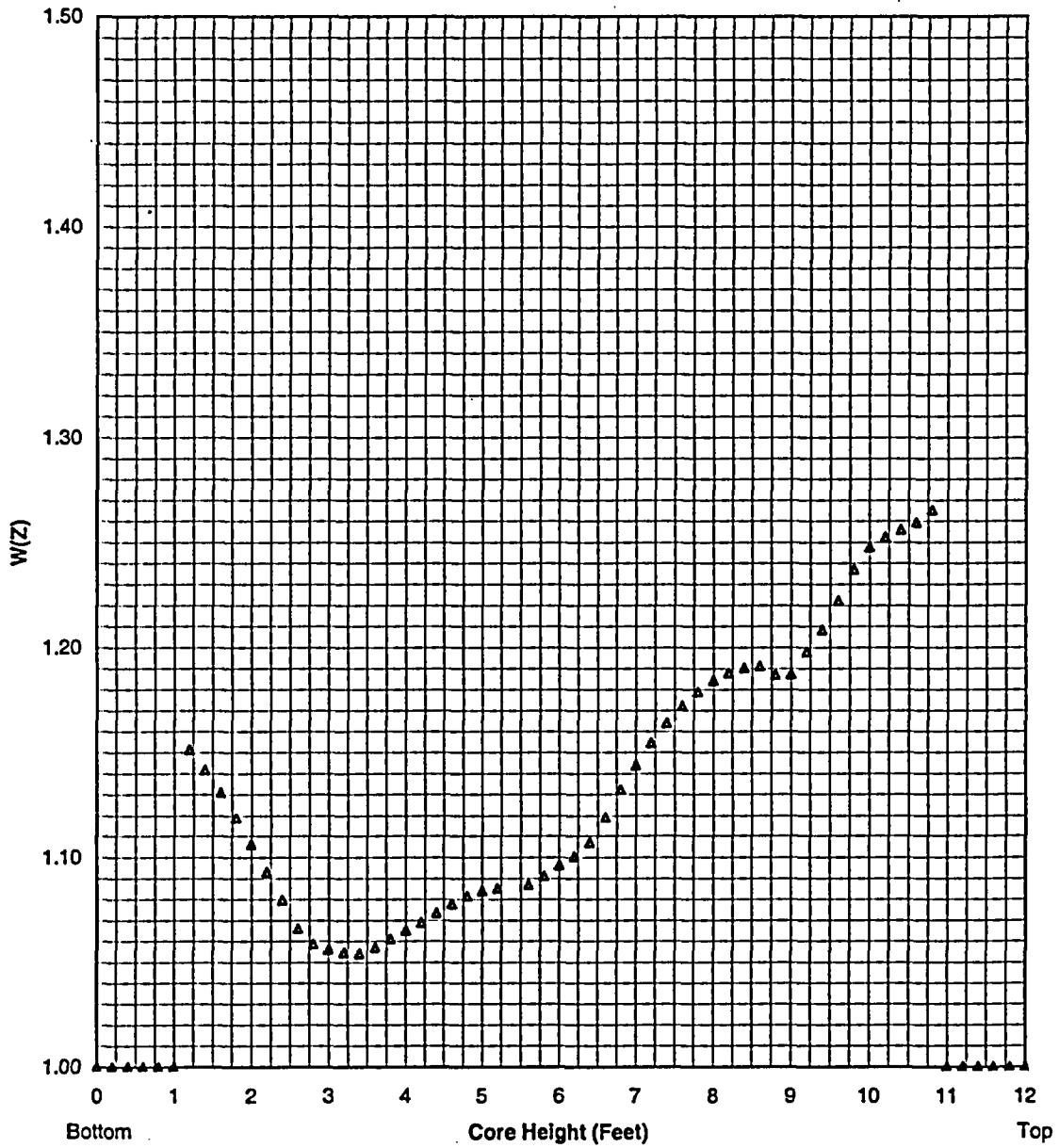
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Figure 5c
RAOC Summary of W(Z) at 8000 MWD/MTU



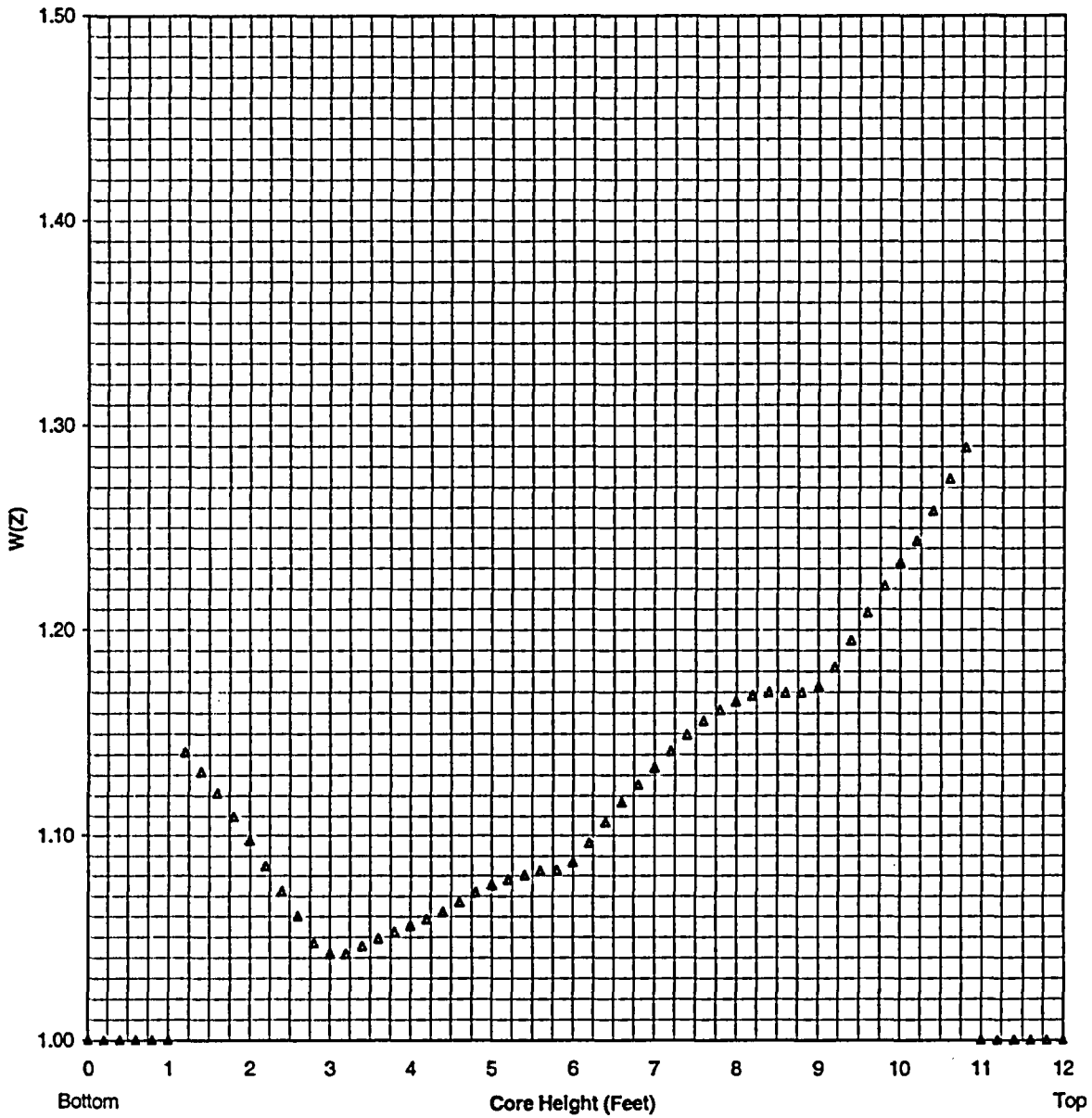
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Figure 5d
RAOC Summary of $W(Z)$ at 10000 MWD/MTU



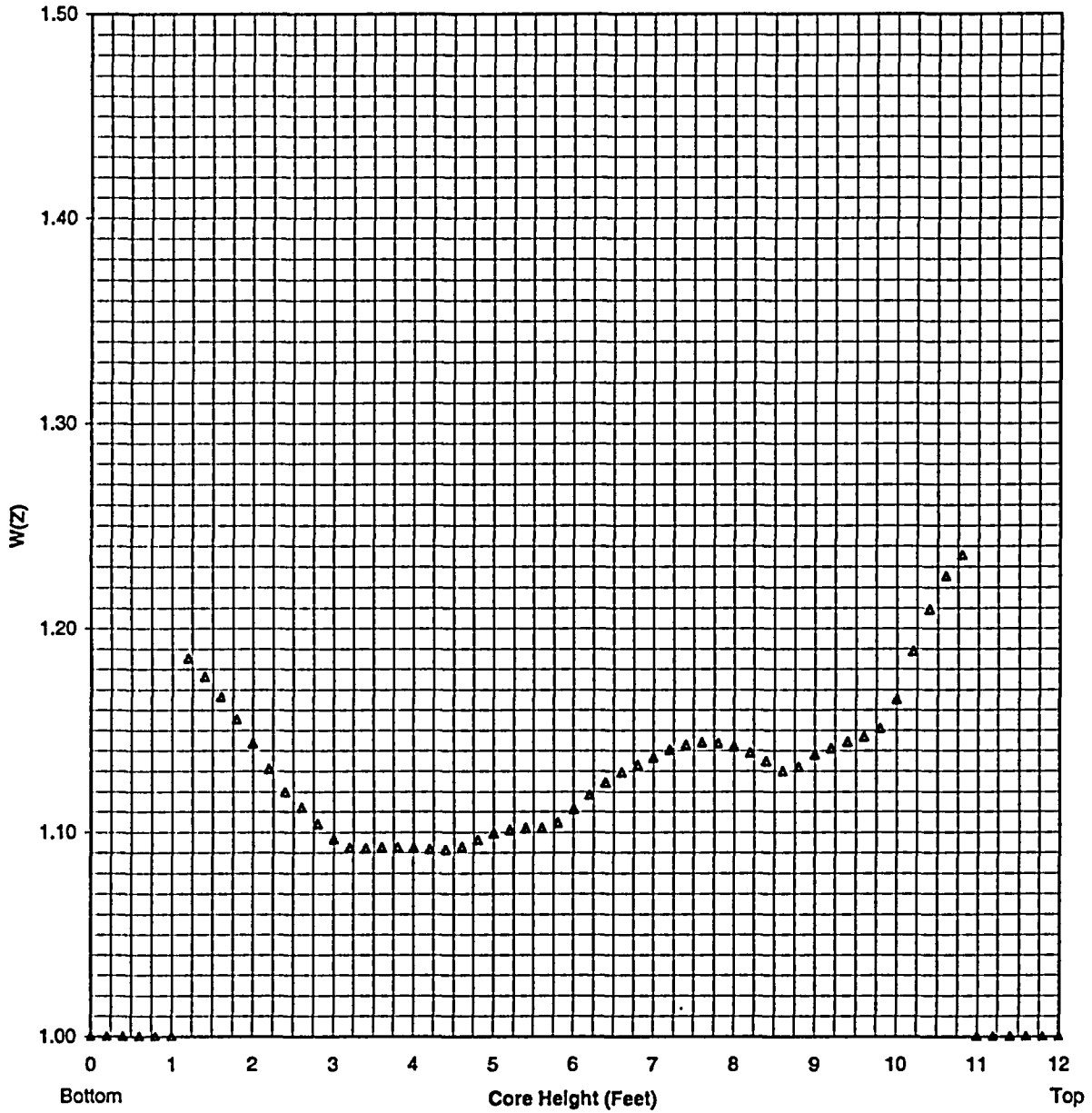
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Figure 5e
RAOC Summary of W(Z) at 16000 MWD/MTU



CORE OPERATING LIMITS REPORT CYCLE 27

Figure 5f
RAOC Summary of W(Z) at 16000 MWD/MTU



CORE OPERATING LIMITS REPORT CYCLE 27

Figure 6
Penalty Factor, F_p (%), for $F_a^{EQ}(Z)$

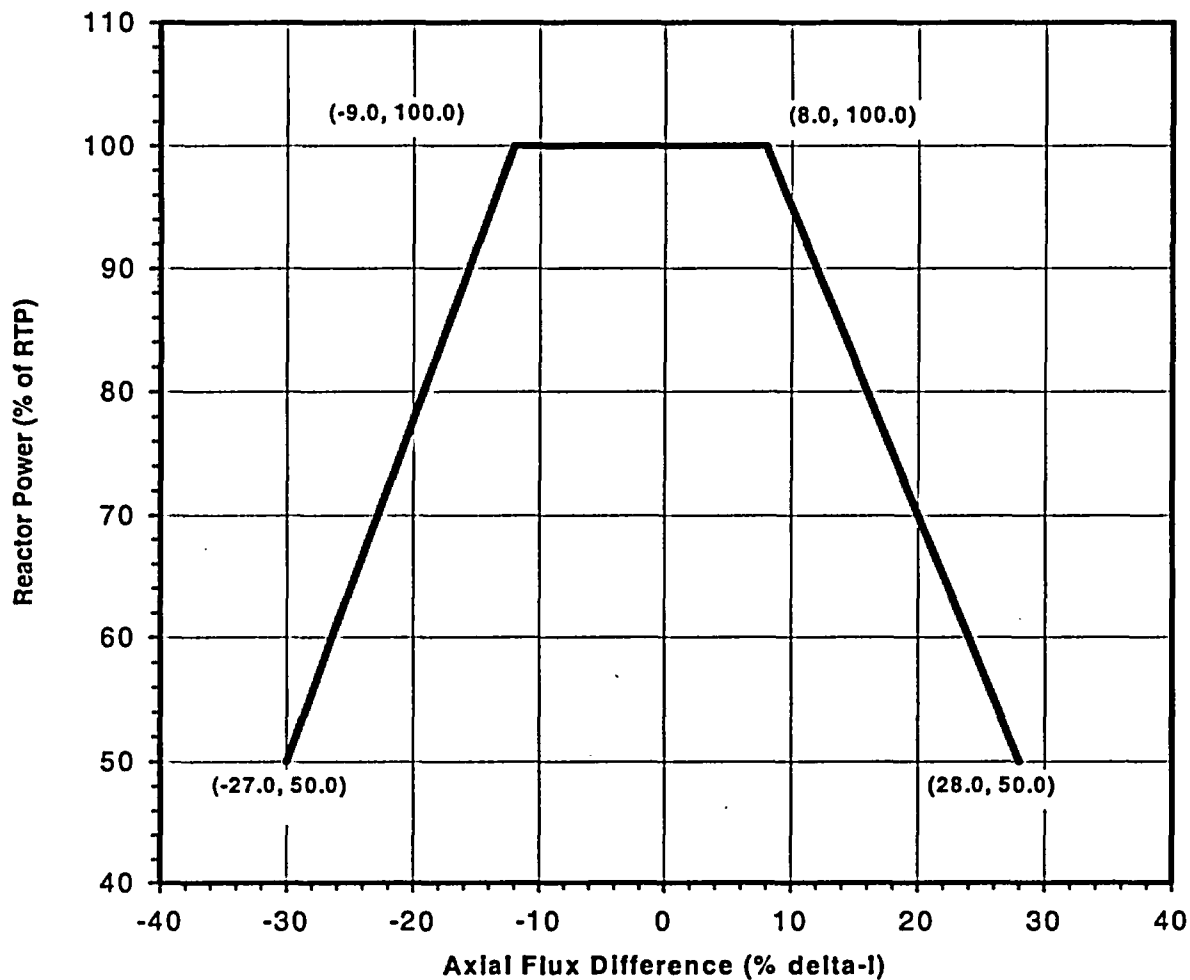
Cycle Burnup (MWD/MTU)	Penalty Factor F_p (%)
150	2.00
19700	2.00

Note: Linear interpolation is adequate for intermediate cycle burnup.

All cycle burnups outside the range of the table shall use a penalty factor, F_p , of 2.0%.
Refer to TS 3.10.b.6.C.

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Figure 7
Axial Flux Difference



Note: This figure represents the Relaxed Axial Offset Control (RAOC) band used in safety analyses, it may be administratively tightened depending on in-core flux map results. Refer to TS 3.10.b.7.

CORE OPERATING LIMITS REPORT CYCLE 27

Table 1

NRC Approved Methodologies for COLR Parameters

<u>COLR Section</u>	<u>Parameter</u>	<u>NRC Approved Methodology</u>
2.1	Reactor Core Safety Limits	<p>WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.</p> <p>Safety Evaluation by the Office of Nuclear Reactor Regulation on "Qualifications of Reactor Physics Methods for Application to Kewaunee" Report, dated August 21, 1979, report date September 29, 1978.</p> <p>Kewaunee Nuclear Power Plant-Review for Kewaunee Reload Safety Evaluation Methods Topical Report WPSRSEM-NP, Revision 3 (TAC NO MB0306) dated September 10, 2001.</p>
2.2	Shutdown Margin	<p>WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985</p> <p>Safety Evaluation by the Office of Nuclear Reactor Regulation on "Qualifications of Reactor Physics Methods for Application to Kewaunee" Report, dated August 21, 1979, report date September 29, 1978.</p>
2.3	Moderator Temperature Coefficient	<p>WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.</p> <p>Safety Evaluation by the Office of Nuclear Reactor Regulation on "Qualifications of Reactor Physics Methods for Application to Kewaunee" Report, dated August 21, 1979, report date September 29, 1978.</p>
2.4	Shutdown Bank Insertion Limit	<p>WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.</p> <p>Safety Evaluation by the Office of Nuclear Reactor Regulation on "Qualifications of Reactor Physics Methods for Application to Kewaunee" Report, dated August 21, 1979, report date September 29, 1978.</p>

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Table 1 (cont)

NRC Approved Methodologies for COLR Parameters

<u>COLR Section</u>	<u>Parameter</u>	<u>NRC Approved Methodology</u>
		Kewaunee Nuclear Power Plant-Review for Kewaunee Reload Safety Evaluation Methods Topical Report WPSRSEM-NP, Revision 3 (TAC NO MB0306) dated September 10, 2001.
2.5	Control Bank Insertion Limits	<p>WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.</p> <p>Safety Evaluation by the Office of Nuclear Reactor Regulation on "Qualifications of Reactor Physics Methods for Application to Kewaunee" Report, dated August 21, 1979, report date September 29, 1978.</p> <p>Kewaunee Nuclear Power Plant-Review for Kewaunee Reload Safety Evaluation Methods Topical Report WPSRSEM-NP, Revision 3 (TAC NO MB0306) dated September 10, 2001.</p>
2.6	Heat Flux Hot Channel Factor	<p>WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification," February 1994.</p> <p>Safety Evaluation by the Office of Nuclear Reactor Regulation on "Qualifications of Reactor Physics Methods for Application to Kewaunee" Report, dated August 21, 1979, report date September 29, 1978.</p> <p>WCAP-12945-P-A (Proprietary), "Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Accident Analysis," Volume I, Rev.2, and Volumes II-V, Rev.1, and WCAP-14747 (Non-Proprietary), March 1998.</p>

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Table 1 (cont)

NRC Approved Methodologies for COLR Parameters

<u>COLR Section</u>	<u>Parameter</u>	<u>NRC Approved Methodology</u>
		ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, dated December 1991.
	(F _d (Z))	WCAP-14449-P-A, "Application Of Best Estimate Large Break LOCA Methodology To Westinghouse PWRs With Upper Plenum Injection," Revision1, and WCAP-14450-NP-A, Rev.1 (Non-Proprietary), October 1999. WCAP-12610-P-A, "Vantage+ Fuel Assembly Reference Core Report," April 1995.
	Model	WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Using the NOTRUMP Code," August 1985. Note: WCAP-10081-A as referenced in TS which is the non-prop version of WCAP-10054-P-A.
	NOTRUMP	WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.
2.7	Nuclear Enthalpy Rise Hot Channel Factor (F ^N _{ΔH})	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985. Safety Evaluation by the Office of Nuclear Reactor Regulation on "Qualifications of Reactor Physics Methods for Application to Kewaunee" Report, dated August 21, 1979, report date September 29, 1978. Kewaunee Nuclear Power Plant-Review forKewaunee Reload Safety Evaluation Methods Topical Report WPSRSEM-NP, Revision 3 (TAC NO MB0306) dated September 10, 2001.

CORE OPERATING LIMITS REPORT CYCLE 27

Table 1 (cont)

NRC Approved Methodologies for COLR Parameters

<u>COLR Section</u>	<u>Parameter</u>	<u>NRC Approved Methodology</u>
		XN-82-06 (P)(A) Revision 1 and Supplements 2, 4, and 5, "Qualifications of Exxon Nuclear Fuel for Extended Burnup, Exxon Nuclear Company, dated October 1986.
		ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups for 62 GWd/MTU," Advanced Nuclear Fuels Corporation, dated December 1991.
		EMF-92-116 (P)(A) Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation, dated February 1999.
2.8	Axial Flux Difference	WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset (AFD) Control- F_0 Surveillance Technical Specification," February 1994.
		Safety Evaluation by the Office of Nuclear Reactor Regulation on "Qualifications of Reactor Physics Methods for Application to Kewaunee" Report, dated August 21, 1979, report date September 29, 1978.
		Kewaunee Nuclear Power Plant-Review for Kewaunee Reload Safety Evaluation Methods Topical Report WPSRSEM-NP, Revision 3 (TAC NO MB0306) dated September 10, 2001.
2.9	Reactor Protection System (RPS) Instrumentation-Overtemperature ΔT	WCAP-8745-P-A, "Design Bases For The Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.

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Table 1 (cont)

NRC Approved Methodologies for COLR Parameters

<u>COLR Section</u>	<u>Parameter</u>	<u>NRC Approved Methodology</u>
		<p>Kewaunee Nuclear Power Plant-Review for Kewaunee Reload Safety Evaluation Methods Topical Report WPSRSEM-NP, Revision 3 (TAC NO MB0306) dated September 10, 2001.</p> <p>CENP-397-P-A, "Improved Flow Measurement Accuracy Using Cross Flow Ultrasonic Flow Measurement Technology," Rev. 1, May 2000.</p>
2.10	Reactor Protection System (RPS) Instrumentation-Overpower ΔT	<p>WCAP-8745-P-A, "Design Bases For The Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.</p> <p>Kewaunee Nuclear Power Plant-Review for Kewaunee Reload Safety Evaluation Methods Topical Report WPSRSEM-NP, Revision 3 (TAC NO MB0306) dated September 10, 2001.</p> <p>CENP-397-P-A, "Improved Flow Measurement Accuracy Using Cross Flow Ultrasonic Flow Measurement Technology," Rev. 1, May 2000.</p>
2.11	RCS Pressure, Temperature, and Flow Departure From Nucleate (DNB) Limits	<p>WCAP-11397-P-A, "Revised Thermal Design Procedure, "April 1989, for those events analyzed using RTDP</p> <p>WCAP-15591, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology-Kewaunee Nuclear Power Plant (Power Uprate to 1757 MWt-NSSS Power with Feedwater Venturis, or 1780 MWt-NSSS Power with Ultrasonic Flow Measurements, and 54F Replacement Steam Generators), Revision 1, December</p>

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2002, Proprietary. Safety Evaluation dated
July 8, 2003.

Table 1 (cont)

NRC Approved Methodologies for COLR Parameters

<u>COLR Section</u>	<u>Parameter</u>	<u>NRC Approved Methodology</u>
		Kewaunee Nuclear Power Plant-Review for Kewaunee Reload Safety Evaluation Methods Topical Report WPSRSEM-NP, Revision 3 (TAC NO MB0306) dated September 10, 2001.
		WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 for those events not utilizing RTDP.
2.12	Boron Concentration	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
		Safety Evaluation by the Office of Nuclear Reactor Regulation on "Qualifications of Reactor Physics Methods for Application to Kewaunee" Report, dated August 21, 1979, report date September 29, 1978.