Dominion Resources Services, Inc.
Innsbrook Technical Center
5000 Dominion Boulevard, 2SE, Glen Allen, VA 23060

Dominion®

January 7, 2014

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

Serial No. 13-643
NLOS /ETS
Docket No. 50-338
License No. NPF-4

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA POWER STATION UNIT 1
CYCLE 24 CORE OPERATING LIMITS REPORT, REVISION 2

Pursuant to North Anna Technical Specification 5.6.5.d, attached is a copy of the Dominion Core Operating Limits Report for North Anna Unit 1 Cycle 24 Pattern BUS, Revision 2.

If you have any questions regarding this submittal, please contact Mr. Thomas Shaub at (804) 273-2763.

Sincerely,

T. R. Huber, Director
Nuclear Licensing and Operations Support
Dominion Resources Services, Inc.
for Virginia Electric and Power Company

Attachment:
1. Core Operating Limits Report for North Anna Unit 1 Cycle 24 - Pattern BUS, Revision 2.

Commitments made in this letter: None

A001
NRR

cc: U.S. Nuclear Regulatory Commission - Region II
Marquis One Tower
245 Peachtree Center Ave., NE, Suite 1200
Atlanta, Georgia 30303-1257

Mr. J. E. Reasor, Jr.
Old Dominion Electric Cooperative
Innsbrook Corporate Center
4201 Dominion Blvd.
Suite 300
Glen Allen, Virginia 23060

NRC Senior Resident Inspector
North Anna Power Station

Dr. V. Sreenivas
NRC Project Manager
U. S. Nuclear Regulatory Commission
One White Flint North
Mail Stop O8 G-9A
11555 Rockville Pike
Rockville, Maryland 20852-2738

ATTACHMENT 1

CORE OPERATING LIMITS REPORT FOR NORTH ANNA UNIT 1 CYCLE 24 PATTERN BUS, REVISION 2

NORTH ANNA POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)

N1C24 CORE OPERATING LIMITS REPORT

INTRODUCTION

The Core Operating Limits Report (COLR) for North Anna Unit 1 Cycle 24 has been prepared in accordance with North Anna Technical Specification 5.6.5. The technical specifications affected by this report are listed below:

TS 2.1.1	Reactor Core Safety Limits
TS 3.1.1	Shutdown Margin (SDM)
TS 3.1.3	Moderator Temperature Coefficient (MTC)
TS 3.1.4	Rod Group Alignment Limits
TS 3.1.5	Shutdown Bank Insertion Limit
TS 3.1.6	Control Bank Insertion Limits
TS 3.1.9	PHYSICS TESTS Exceptions – Mode 2
TS 3.2.1	Heat Flux Hot Channel Factor
TS 3.2.2	Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)
TS 3.2.3	Axial Flux Difference (AFD)
TS 3.3.1	Reactor Trip System (RTS) Instrumentation
TS 3.4.1	RCS Pressure, Temperature, and Flow DNB Limits
TS 3.5.6	Boron Injection Tank (BIT)
TS 3.9.1	Boron Concentration

In addition, a technical requirement (TR) in the NAPS Technical Requirements Manual (TRM) refers to the COLR:

TR 3.1.1 Boration Flow Paths – Operating

The analytical methods used to determine the core operating limits are those previously approved by the NRC and discussed in the documents listed in the References Section.

Cycle-specific values are presented **in bold.** Text in *italics* is provided for information only.

REFERENCES

1. VEP-FRD-42, Rev. 2.1-A, "Reload Nuclear Design Methodology," August 2003.

   Methodology for:
   TS 3.1.1 – Shutdown Margin,
   TS 3.1.3 – Moderator Temperature Coefficient,
   TS 3.1.4 – Rod Group Alignment Limits
   TS 3.1.5 – Shutdown Bank Insertion Limit,
   TS 3.1.6 – Control Bank Insertion Limits,
   TS 3.1.9 – Physics Tests Exceptions – Mode 2,
   TS 3.2.1 – Heat Flux Hot Channel Factor,
   TS 3.2.2 – Nuclear Enthalpy Rise Hot Channel Factor
   TS 3.5.6 – Boron Injection Tank (BIT) and
   TS 3.9.1 – Boron Concentration

2. Plant-specific adaptation of WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," as approved by NRC Safety Evaluation Report dated February 29, 2012.

   Methodology for: TS 3.2.1 – Heat Flux Hot Channel Factor

3. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

   Methodology for: TS 3.2.1 – Heat Flux Hot Channel Factor

4. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

   Methodology for: TS 3.2.1 – Heat Flux Hot Channel Factor

5. WCAP-12610-P-A, "VANTAGE+ FUEL ASSEMBLY – REFERENCE CORE REPORT," April 1995.

   Methodology for:
   TS 2.1.1 – Reactor Core Safety Limits
   TS 3.2.1 – Heat Flux Hot Channel Factor

6. VEP-NE-2, Rev. 0-A, Statistical DNBR Evaluation Methodology, June 1987.

   Methodology for:
   TS 3.2.2 – Nuclear Enthalpy Rise Hot Channel Factor and
   TS 3.4.1 – RCS Pressure, Temperature and Flow DNB Limits

7. VEP-NE-1, Rev. 0.1-A, Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications, August 2003.

    Methodology for:
    TS 3.2.1 – Heat Flux Hot Channel Factor and
    TS 3.2.3 – Axial Flux Difference

8. WCAP-8745-P-A, Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions, September 1986.

    Methodology for:
    TS 2.1.1 – Reactor Core Safety Limits and
    TS 3.3.1 – Reactor Trip System Instrumentation

9. WCAP-14483-A, Generic Methodology for Expanded Core Operating Limits Report, January 1999.

    Methodology for:
    TS 2.1.1 – Reactor Core Safety Limits,
    TS 3.1.1 – Shutdown Margin,
    TS 3.1.4 – Rod Group Alignment Limits
    TS 3.1.9 – Physics Tests Exceptions – Mode 2
    TS 3.3.1 – Reactor Trip System Instrumentation,
    TS 3.4.1 – RCS Pressure, Temperature, and Flow DNB Limits
    TS 3.5.6 – Boron Injection Tank (BIT) and
    TS 3.9.1 – Boron Concentration

10. BAW-10227P-A, Rev. 0, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," February 2000.

    Methodology for:
    TS 2.1.1 – Reactor Core Safety Limits and
    TS 3.2.1 – Heat Flux Hot Channel Factor

11. EMF-2103 (P) (A), Rev. 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," April 2003.

    Methodology for: TS 3.2.1 - Heat Flux Hot Channel Factor

12. EMF-96-029 (P) (A), Rev. 0, "Reactor Analysis System for PWRs," January 1997.

    Methodology for: TS 3.2.1 - Heat Flux Hot Channel Factor

13. BAW-10168P-A, Rev. 3, "RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," December 1996. Volume II only (SBLOCA models).

    Methodology for: TS 3.2.1 - Heat Flux Hot Channel Factor

14. DOM-NAF-2, Rev. 0.2- P-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," including Appendix A, "Qualification of the F-ANP BWU CHF Correlations in the Dominion VIPRE-D Computer Code," and Appendix C, "Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code," August 2010.

    Methodology for:
    TS 3.2.2 – Nuclear Enthalpy Rise Hot Channel Factor and
    TS 3.4.1 – RCS Pressure, Temperature and Flow DNB Limits

15. WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," July 2006.

    Methodology for:
    TS 2.1.1 – Reactor Core Safety Limits and
    TS 3.2.1 – Heat Flux Hot Channel Factor

Note: In some instances, the North Anna COLR lists multiple methodologies that are used to verify a single Technical Specification parameter. This is due to the transition from AREVA fuel to Westinghouse fuel which requires the use of different vendor proprietary methodologies to verify the two fuel products meet the applicable regulatory limits.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

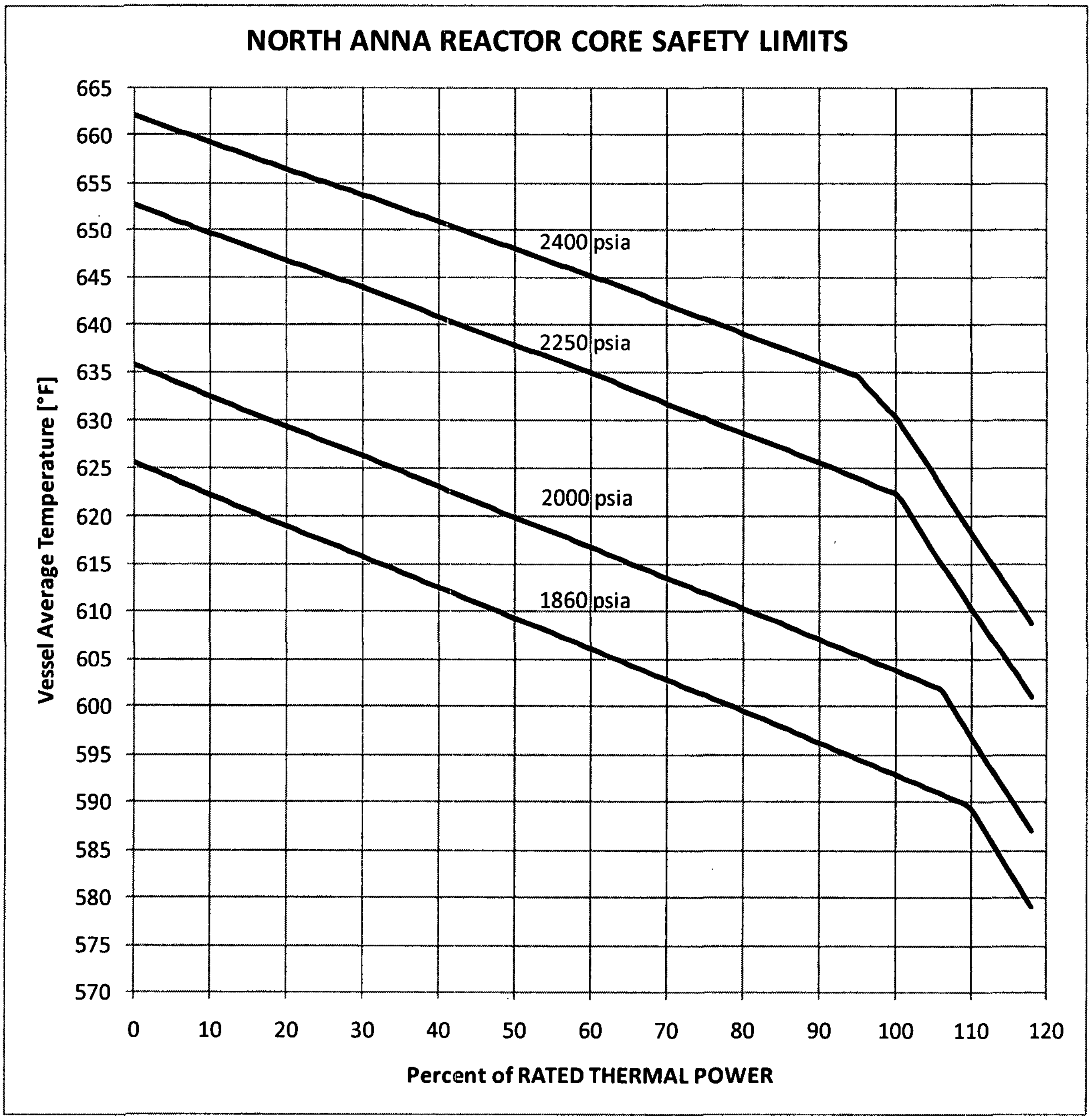
2.1.1 Reactor Core SLs

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 specified in **COLR Figure 2.1-1**; and the following SLs shall not be exceeded.

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to the 95/95 DNBR criterion for the DNB correlations and methodologies specified in the References Section.

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup, for Westinghouse fuel and < 5173°F, decreasing by 65°F per 10,000 MWD/MTU of burnup, for AREVA fuel.

COLR Figure 2.1-1



3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be ≥ **1.77 % Δk/k.**

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3 The MTC shall be maintained within the limits specified below. The upper limit of MTC is **+0.6 x 10^{-4} Δk/k/°F**, when < 70% RTP, and **0.0 Δk/k/°F** when ≥ 70% RTP.

The BOC/ARO-MTC shall be ≤ **+0.6 x 10^{-4} Δk/k/°F** (upper limit), when < 70% RTP, and ≤ **0.0 Δk/k/°F** when ≥ 70% RTP.

The EOC/ARO/RTP-MTC shall be less negative than **–5.0 x 10^{-4} Δk/k/°F** (lower limit).

The MTC surveillance limits are:

The 300 ppm/ARO/RTP-MTC should be less negative than or equal to **–4.0 x 10^{-4} Δk/k/°F** [Note 2].

The 60 ppm/ARO/RTP-MTC should be less negative than or equal to **–4.7 x 10^{-4} Δk/k/°F** [Note 3].

SR 3.1.3.2 Verify MTC is within **–5.0 x 10^{-4} Δk/k/°F** (lower limit).

Note 2: If the MTC is more negative than **–4.0 x 10^{-4} Δk/k/°F**, SR 3.1.3.2 shall be repeated once per 14 EFPD during the remainder of the fuel cycle.

Note 3: SR 3.1.3.2 need not be repeated if the MTC measured at the equivalent of equilibrium RTP-ARO boron concentration of ≤ 60 ppm is less negative than **–4.7 x 10^{-4} Δk/k/°F.**

3.1.4 Rod Group Alignment Limits

Required Action A.1.1 Verify SDM to be ≥ **1.77 % Δk/k.**

Required Action B.1.1 Verify SDM to be ≥ **1.77 % Δk/k.**

Required Action D.1.1 Verify SDM to be ≥ **1.77 % Δk/k.**

3.1.5 Shutdown Bank Insertion Limits

LCO 3.1.5 Each shutdown bank shall be withdrawn to at least **225** steps.

Required Action A.1.1 Verify SDM to be ≥ **1.77 % Δk/k.**

Required Action B.1 Verify SDM to be ≥ **1.77 % Δk/k.**

SR 3.1.5.1 Verify each shutdown bank is withdrawn to at least **225** steps.

3.1.6 Control Bank Insertion Limits

LCO 3.1.6 Control banks shall be limited in physical insertion as shown in **COLR Figure 3.1-1.** Sequence of withdrawal shall be A, B, C and D, in that order; and the overlap limit during withdrawal shall be **97** steps.

Required Action A.1.1 Verify SDM to be ≥ **1.77 % Δk/k.**

Required Action B.1.1 Verify SDM to be ≥ **1.77 % Δk/k.**

Required Action C.1 Verify SDM to be ≥ **1.77 % Δk/k.**

SR 3.1.6.1 Verify estimated critical control bank position is within the insertion limits specified in **COLR Figure 3.1-1**.

SR 3.1.6.2 Verify each control bank is within the insertion limits specified in **COLR Figure 3.1-1**.

SR 3.1.6.3 Verify each control bank not fully withdrawn from the core is within the sequence and overlap limits specified in **LCO 3.1.6 above**.

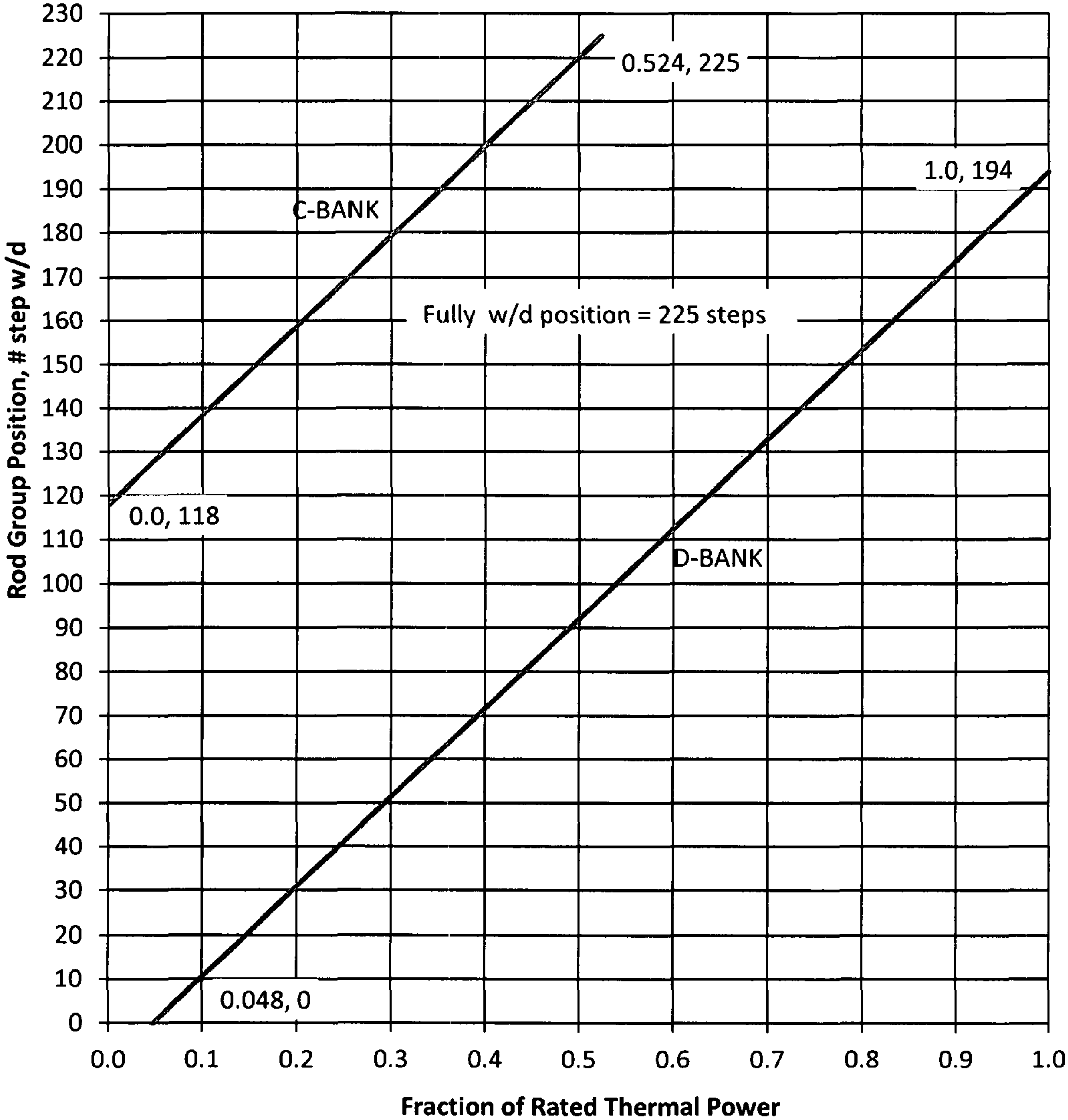
3.1.9 PHYSICS TESTS Exceptions – MODE 2

LCO 3.1.9.b SDM is ≥ **1.77 % Δk/k.**

SR 3.1.9.4 Verify SDM to be ≥ **1.77 % Δk/k.**

COLR Figure 3.1-1

North Anna 1 Cycle 24
Control Rod Bank Insertion Limits



3.2 POWER DISTRIBUTION LIMITS

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

LCO 3.2.1 $F_Q(Z)$, as approximated by $F_Q^M(Z)$, shall be within the limits specified below.

CFQ = 2.32

The Measured Heat Flux Hot Channel Factor, $FQ^M(Z)$, shall be limited by the following relationships:

$$F_Q^M(Z) \le \frac{CFQ}{P}\frac{K(Z)}{N(Z)} \qquad for\ P>0.5$$

$$F_Q^M(Z) \le \frac{CFQ}{0.5}\frac{K(Z)}{N(Z)} \qquad for\ P\le 0.5$$

where: $P = \dfrac{THERMAL\ POWER}{RATED\ THERMAL\ POWER}$; and

K(Z) is provided in **COLR Figure 3.2-1**

N(Z) is a cycle-specific non-equilibrium multiplier on $F_Q^M(Z)$ to account for power distribution transients during normal operation, provided in **COLR Table 3.2-1.**

The discussion in the Bases Section B 3.2.1 for this LCO requires the application of a cycle dependent non-equilibrium multiplier, N(Z), to the CFQ limit. N(Z) accounts for power distribution transients encountered during normal operation. As function N(Z) is dependent on the predicted equilibrium $F_Q(Z)$ and is sensitive to the axial power distribution, it is typically generated from the actual EOC burnup distribution that can only be obtained after the shutdown of the previous cycle. The cycle-specific N(Z) function is presented in **COLR Table 3.2-1**.

COLR Table 3.2-1
N1C24 Normal Operation N(Z)

NODE	HEIGHT (FEET)	0 to 1000 MWD/MTU	1000 to 3000 MWD/MTU	3000 to 5000 MWD/MTU	5000 to 7000 MWD/MTU	7000 to 9000 MWD/MTU
10	10.2	1.128	1.139	1.143	1.144	1.121
11	10.0	1.128	1.147	1.148	1.148	1.128
12	9.8	1.133	1.155	1.154	1.152	1.137
13	9.6	1.140	1.161	1.161	1.156	1.148
14	9.4	1.143	1.166	1.165	1.156	1.153
15	9.2	1.144	1.168	1.168	1.158	1.157
16	9.0	1.150	1.173	1.173	1.170	1.170
17	8.8	1.155	1.176	1.176	1.181	1.181
18	8.6	1.158	1.177	1.176	1.186	1.186
19	8.4	1.159	1.175	1.176	1.185	1.185
20	8.2	1.162	1.173	1.176	1.186	1.185
21	8.0	1.162	1.169	1.175	1.185	1.184
22	7.8	1.162	1.164	1.175	1.185	1.183
23	7.6	1.160	1.160	1.174	1.180	1.184
24	7.4	1.157	1.157	1.171	1.173	1.187
25	7.2	1.152	1.152	1.167	1.167	1.189
26	7.0	1.147	1.147	1.163	1.163	1.189
27	6.8	1.145	1.145	1.161	1.162	1.190
28	6.6	1.143	1.143	1.158	1.162	1.188
29	6.4	1.134	1.135	1.150	1.158	1.185
30	6.2	1.123	1.123	1.137	1.152	1.178
31	6.0	1.118	1.116	1.131	1.149	1.175
32	5.8	1.113	1.108	1.121	1.142	1.167
33	5.6	1.100	1.091	1.098	1.126	1.148
34	5.4	1.092	1.079	1.081	1.112	1.131
35	5.2	1.092	1.077	1.077	1.107	1.122
36	5.0	1.096	1.080	1.079	1.109	1.120
37	4.8	1.099	1.082	1.079	1.111	1.121
38	4.6	1.101	1.086	1.082	1.114	1.123
39	4.4	1.102	1.091	1.087	1.115	1.121
40	4.2	1.102	1.097	1.091	1.115	1.120
41	4.0	1.104	1.104	1.096	1.115	1.120
42	3.8	1.113	1.112	1.104	1.113	1.116
43	3.6	1.127	1.121	1.114	1.115	1.115
44	3.4	1.137	1.128	1.122	1.124	1.123
45	3.2	1.146	1.135	1.129	1.137	1.137
46	3.0	1.157	1.144	1.135	1.147	1.147
47	2.8	1.170	1.154	1.142	1.155	1.155
48	2.6	1.182	1.163	1.153	1.158	1.159
49	2.4	1.193	1.172	1.164	1.166	1.167
50	2.2	1.203	1.180	1.174	1.183	1.183
51	2.0	1.213	1.187	1.183	1.195	1.195
52	1.8	1.222	1.195	1.192	1.197	1.197

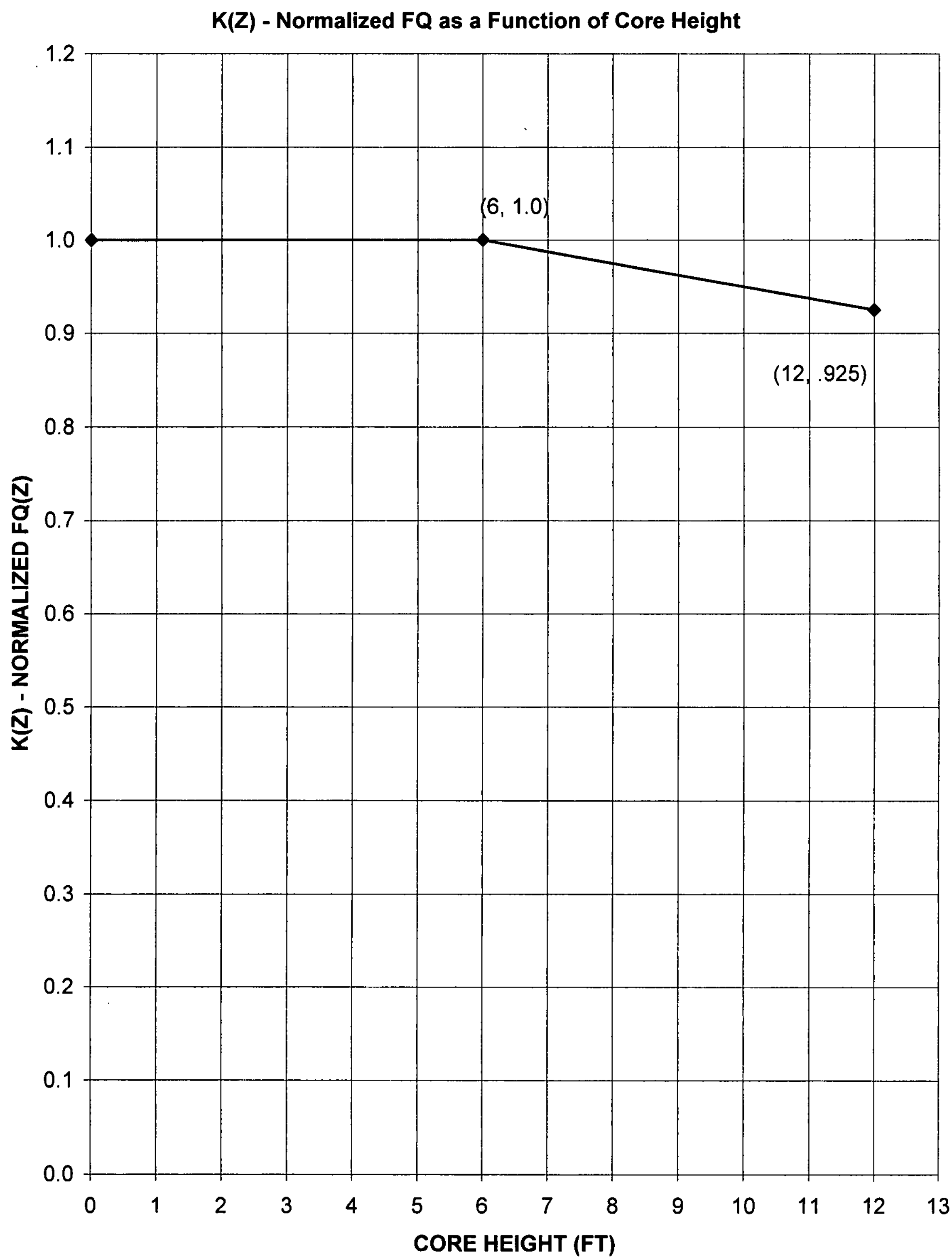
These decks are generated for normal operation flux maps that are typically taken at full power ARO. Additional N(z) decks may be generated, if necessary, consistent with the methodology described in the RPDC topical (Reference 7). EOR is defined as Hot Full Power End of Reactivity.

COLR Table 3.2-1 (continued)
N1C24 Normal Operation N(Z)

NODE	HEIGHT (FEET)	9000 to 11000 MWD/MTU	11000 to 13000 MWD/MTU	13000 to 15000 MWD/MTU	15000 to 17000 MWD/MTU	17000 to EOR MWD/MTU
10	10.2	1.119	1.120	1.112	1.119	1.119
11	10.0	1.118	1.118	1.111	1.119	1.119
12	9.8	1.123	1.121	1.108	1.118	1.118
13	9.6	1.133	1.127	1.109	1.117	1.117
14	9.4	1.138	1.128	1.111	1.112	1.112
15	9.2	1.143	1.132	1.120	1.113	1.113
16	9.0	1.156	1.144	1.139	1.123	1.123
17	8.8	1.168	1.158	1.159	1.138	1.139
18	8.6	1.173	1.164	1.165	1.146	1.148
19	8.4	1.175	1.170	1.170	1.157	1.160
20	8.2	1.180	1.182	1.182	1.175	1.179
21	8.0	1.182	1.189	1.189	1.188	1.194
22	7.8	1.182	1.192	1.192	1.193	1.200
23	7.6	1.183	1.193	1.193	1.200	1.207
24	7.4	1.187	1.197	1.197	1.211	1.218
25	7.2	1.189	1.198	1.198	1.217	1.226
26	7.0	1.189	1.198	1.198	1.219	1.228
27	6.8	1.190	1.197	1.197	1.221	1.231
28	6.6	1.188	1.195	1.195	1.222	1.231
29	6.4	1.185	1.196	1.196	1.222	1.232
30	6.2	1.178	1.197	1.197	1.219	1.229
31	6.0	1.175	1.198	1.198	1.218	1.229
32	5.8	1.167	1.194	1.194	1.211	1.223
33	5.6	1.147	1.184	1.184	1.196	1.209
34	5.4	1.132	1.173	1.173	1.181	1.194
35	5.2	1.130	1.168	1.168	1.174	1.188
36	5.0	1.131	1.159	1.163	1.169	1.182
37	4.8	1.130	1.145	1.155	1.161	1.172
38	4.6	1.127	1.131	1.145	1.154	1.164
39	4.4	1.123	1.122	1.134	1.148	1.159
40	4.2	1.119	1.123	1.130	1.144	1.154
41	4.0	1.119	1.133	1.134	1.140	1.148
42	3.8	1.118	1.142	1.141	1.134	1.135
43	3.6	1.122	1.150	1.150	1.128	1.124
44	3.4	1.130	1.156	1.156	1.122	1.122
45	3.2	1.144	1.161	1.161	1.122	1.133
46	3.0	1.156	1.163	1.165	1.130	1.145
47	2.8	1.168	1.168	1.167	1.145	1.160
48	2.6	1.176	1.174	1.169	1.153	1.170
49	2.4	1.185	1.185	1.170	1.163	1.181
50	2.2	1.199	1.199	1.170	1.175	1.196
51	2.0	1.208	1.208	1.171	1.186	1.209
52	1.8	1.209	1.209	1.173	1.194	1.220

These decks are generated for normal operation flux maps that are typically taken at full power ARO. Additional N(z) decks may be generated, if necessary, consistent with the methodology described in the RPDC topical (Reference 7). EOR is defined as Hot Full Power End of Reactivity.

COLR Figure 3.2-1



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

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified below.

$$\mathbf{F^N_{\Delta H} \leq 1.587\{1 + 0.3(1 - P)\}}$$

where: $$P = \frac{THERMAL\ POWER}{RATED\ THERMAL\ POWER}$$

SR 3.2.2.1 Verify $F^N_{\Delta H}$ is within limits specified above.

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3 The AFD in % flux difference units shall be maintained within the limits specified in the applicable **COLR Figure (3.2-2-A or 3.2-2-B).**

COLR Figure 3.2-2-A
Applicable Burnup: BOC* to 5000 MWD/MTU

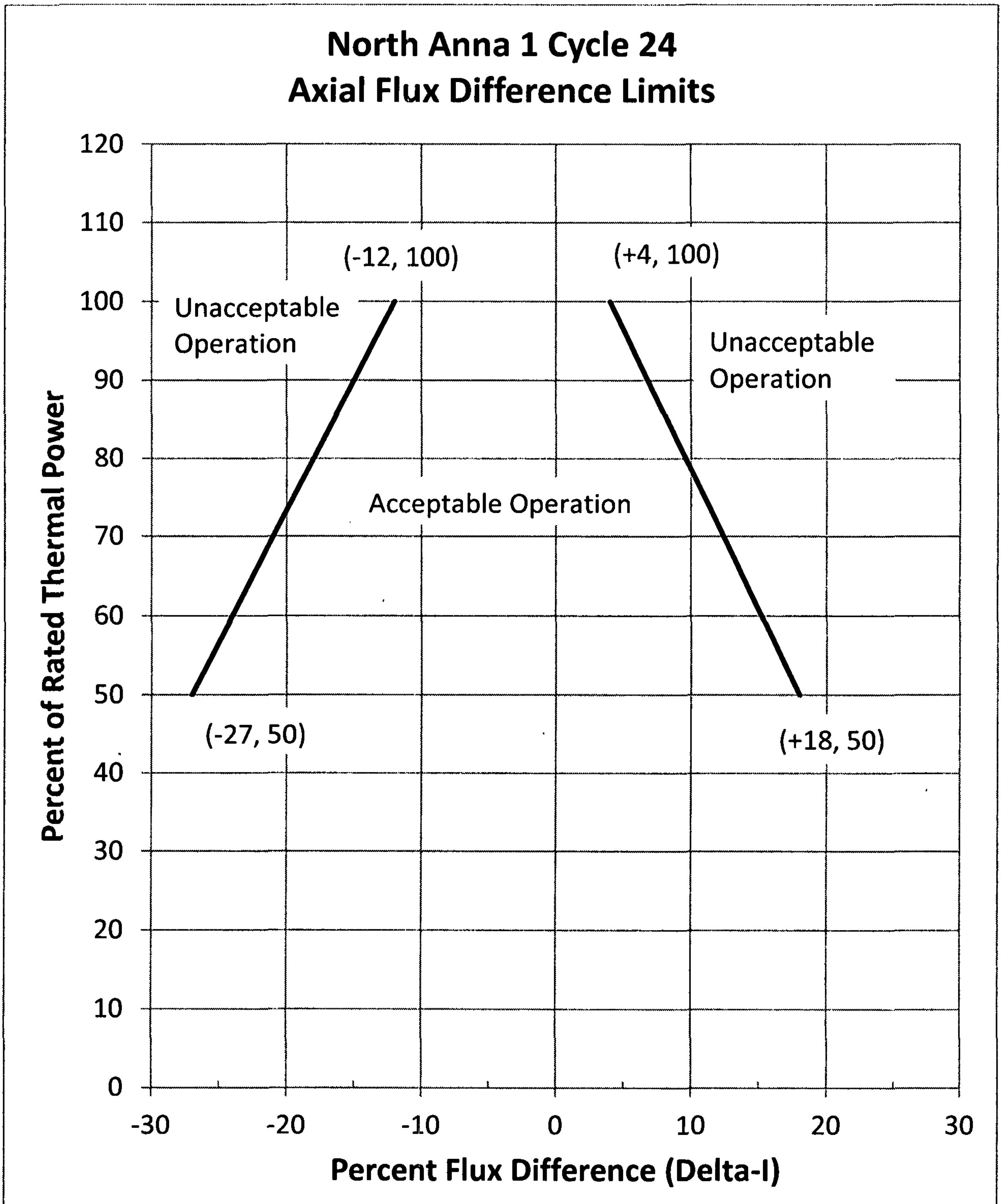
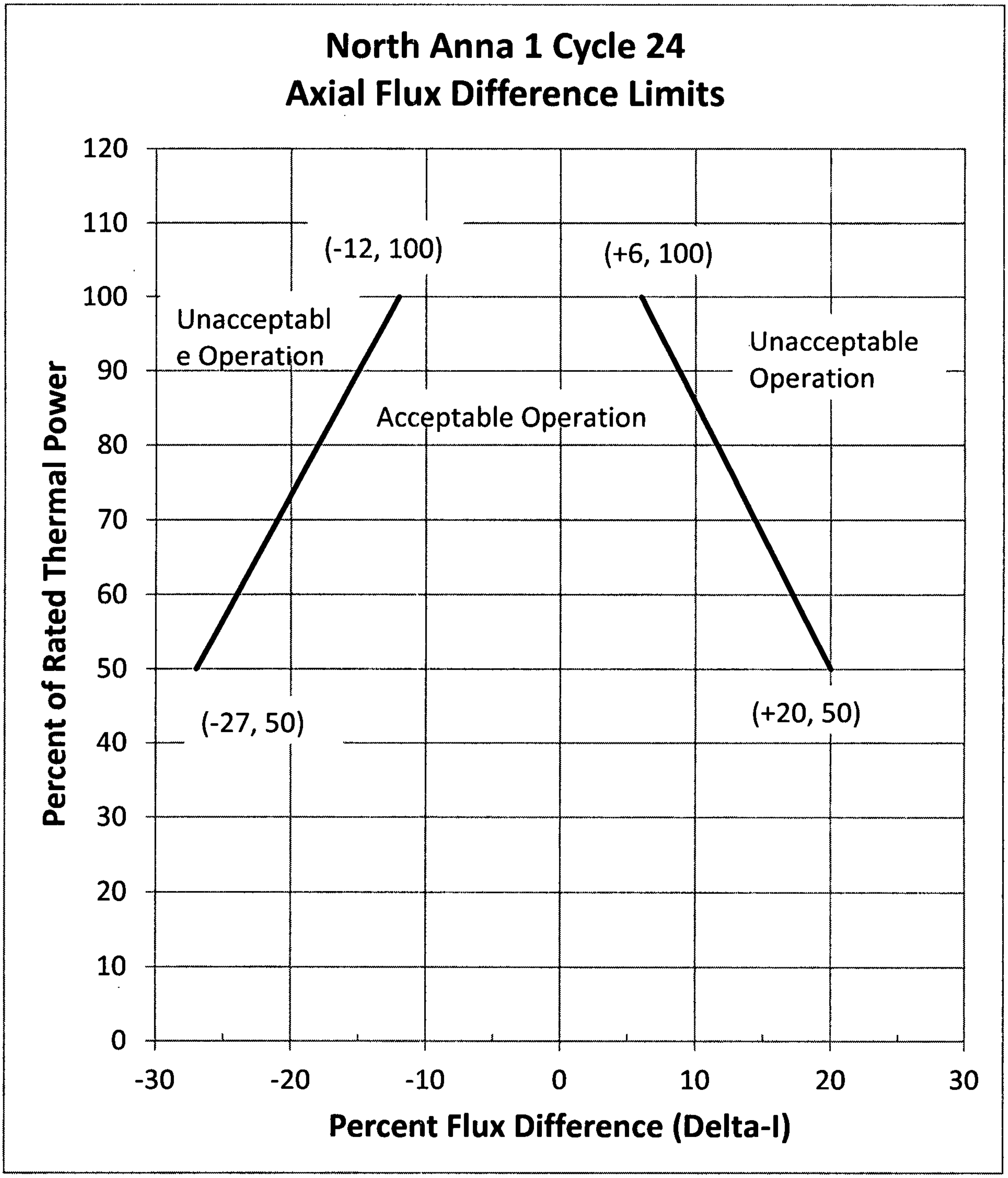


*Figure 3.2.-2-A was implemented at a core burnup of approximately 2000 MWD/MTU.

COLR Figure 3.2-2-B
Applicable Burnup: 5000 MWD/MTU to EOC



3.3.1 Reactor Trip System (RTS) Instrumentation

TS Table 3.3.1-1 Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following nominal trip setpoint by more than 2% of ΔT span, with the numerical values of the parameters as specified below.

$$\Delta T \leq \Delta T_0 \left\{ K_1 - K_2 \frac{(1+\tau_1 s)}{(1+\tau_2 s)} [T - T'] + K_3 (P - P') - f_1(\Delta I) \right\}$$

where: ΔT is measured RCS ΔT, °F
ΔT_0 is the indicated ΔT at RTP, °F
s is the Laplace transform operator, sec^{-1}
T is the measured RCS average temperature, °F
T′ is the nominal T_{avg} at RTP, **≤ 586.8 °F**
P is the measured pressurizer pressure, psig
P′ is the nominal RCS operating pressure, **≥ 2235 psig**

$K_1 \leq$ **1.2715** $K_2 \geq$ **0.02174 /°F** $K_3 \geq$ **0.001145 /psig**

τ_1, τ_2 = time constants utilized in the lead-lag controller for T_{avg}
$\tau_1 \geq$ **23.75 sec** $\tau_2 \leq$ **4.4 sec**

$(1+\tau_1 s)/(1+\tau_2 s)$ = function generated by the lead-lag controller for T_{avg} dynamic compensation

$f_1(\Delta I) \geq$ **0.0291**{-13.0 – ($q_t - q_b$)}	when ($q_t - q_b$) < **–13.0%** RTP
0	when **–13.0%** RTP ≤ ($q_t - q_b$) ≤ **+7.0%** RTP
0.0251{($q_t - q_b$) – **7.0**}	when ($q_t - q_b$) > **+7.0%** 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.

TS Table 3.3.1-1 Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following nominal trip setpoint by more than 2% of ΔT span, with the numerical values of the parameters as specified below.

$$\Delta T \le \Delta T_0 \left\{ K_4 - K_5 \left[\frac{\tau_3 s}{1+\tau_3 s} \right] T - K_6 [T - T'] - f_2(\Delta I) \right\}$$

where: ΔT is measured RCS ΔT, °F.

ΔT_0 is the indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature, °F.

T′ is the nominal T_{avg} at RTP, **≤ 586.8 °F.**

K_4 **≤ 1.0865**

K_5 **≥0.0198 /°F** for increasing T_{avg}
0 /°F for decreasing T_{avg}

K_6 **≥0.00162 /°F** when T > T′
0 /°F when T ≤ T′

τ_3 = time constant utilized in the rate lag controller for T_{avg}

τ_3 **≥ 9.5 sec**

$\tau_3 s/(1+\tau_3 s)$ = function generated by the rate lag controller for T_{avg} dynamic compensation

$f_2(\Delta I)$ = **0, for all ΔI.**

3.4 REACTOR COOLANT SYSTEM (RCS)

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

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure is greater than or equal to **2205 psig**;

b. RCS average temperature is less than or equal to **591 °F**; and

c. RCS total flow rate is greater than or equal to **295,000 gpm**.

SR 3.4.1.1 Verify pressurizer pressure is greater than or equal to **2205 psig**.

SR 3.4.1.2 Verify RCS average temperature is less than or equal to **591 °F**.

SR 3.4.1.3 Verify RCS total flow rate is greater than or equal to **295,000 gpm**.

SR 3.4.1.4 ------------------------------NOTE------------------------------
Not required to be performed until 30 days after ≥ 90% RTP.

Verify by precision heat balance that RCS total flow rate is ≥ **295,000 gpm**.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.6 Boron Injection Tank (BIT)

Required Action B.2 Borate to a SDM ≥ **1.77 % Δk/k at 200 °F**.

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity shall be maintained ≥ **2600 ppm**.

SR 3.9.1.1 Verify boron concentration is within the limit specified above.

TRM 3.1 REACTIVITY CONTROL SYSTEMS

TR 3.1.1 Boration Flow Paths – Operating

Required Action D.2 Borate to a SHUTDOWN MARGIN ≥ **1.77 % Δk/k at 200 °F,** after xenon decay.