# Progress Energy

U. S. Nuclear Regulatory Commission
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SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
CYCLE 17 CORE OPERATING LIMITS REPORT- REVISION 1

Ladies and Gentlemen:

In accordance with Technical Specifications (TS) 6.9.1.6.4, Carolina Power & Light Company, doing business as Progress Energy Carolinas, Inc., submits Revision 1 to the Harris Nuclear Plant (HNP) Cycle 17 Core Operating Limits Report (COLR).

Enclosure 1 provides a summary of Revision 1 to the Cycle 17 COLR.  Enclosure 2 contains a copy of Revision 1 of the Cycle 17 COLR.

This document contains no new regulatory commitments.

Please refer any questions regarding this submittal to me at (919) 362-3137.

Sincerely,

David H. Corlett
Supervisor - Licensing/Regulatory Programs
Harris Nuclear Plant

DHC/jmd

Enclosures:  1.  Summary of Changes – HNP Cycle 17 COLR Revision 1
2.  Attachment 9 to HNP Procedure PLP-106, "Technical Specification Equipment List Programs and Core Operating Limits Report – Rev. 1"

cc:  Mr. J. D. Austin, NRC Sr. Resident Inspector, HNP
Ms. A. T. Billoch Colon, NRC Project Manager, HNP
Mr. V. M. McCree, NRC Regional Administrator, Region II

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NRR

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SUMMARY OF CHANGES - HNP CYCLE 17 COLR REVISION 1

| Page | Section | Change Description |
|---|---|---|
| 4 | 2.0.18 | Added EC 83899 to references. |
| 3, 4, 6 | 2.10, 4.3.b, and 5.2.4 | Changed AP-603, FSAR Revisions to REG-NGGC-0101 as AP-603 was superceded (Editorial Correction). PRR-481239 |
| 58 | Attachment 9, Sheet 2 of 13 | Changed the 300 ppm/ARO/RTP-MTC value in step 2.2.2 from -42.1 pcm/°F to -44.1 pcm/°F per EC 83899. |
| 57-69 | Attachment 9 | Changed title of the entire attachment from Rev. 0 to Rev. 1. |
| All | All | Updated footer to reflect Rev. 51 of PLP-106 |

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Attachment 9 to HNP Procedure PLP-106,
"Technical Specification Equipment List Program and
Core Operating Limits Report," Rev. 1
(13 Pages)

Harris Unit 1 Cycle 17
Core Operating Limits Report - Rev. 1

1.0   CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Shearon Harris Unit 1 Cycle 17 has been
prepared in accordance with the requirements of Technical Specification 6.9.1.6.

The Technical Specifications affected by this report are listed below:

   3/4.1.1.2    SHUTDOWN MARGIN - Modes 3, 4, and 5

   3/4.1.1.3    Moderator Temperature Coefficient

   3/4.1.3.5    Shutdown Rod Insertion Limit

   3/4.1.3.6    Control Rod Insertion Limits

   3/4.2.1      Axial Flux Difference

   3/4.2.2      Heat Flux Hot Channel Factor - $F_Q(Z)$

   3/4.2.3      Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}$

   3/4.9.1.a    Boron Concentration During Refueling Operations

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
NRC-approved methodologies specified in Technical Specification 6.9.1.6 and given in
Section 3.0.

2.1   SHUTDOWN MARGIN - Modes 3, 4, and 5 (Specification 3/4.1.1.2)

   The SHUTDOWN MARGIN versus RCS boron concentration - Modes 3, 4, and 5 is
   specified in Figure 1.

2.2   Moderator Temperature Coefficient (Specification 3/4.1.1.3)

   1.   The Moderator Temperature Coefficient (MTC) limits are:

        The Positive MTC Limit (ARO/HZP) shall be less positive than
        +5.0 pcm/°F for power levels up to 70% RTP with a linear ramp to
        0 pcm/°F at 100% RTP.

        The Negative MTC Limit (ARO/RTP) shall be less negative than
        -50 pcm/°F.

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2.2     Moderator Temperature Coefficient (Specification 3/4.1.1.3) (continued)

        2.      The MTC Surveillance limit is:

                The 300 ppm/ARO/RTP-MTC should be less negative than or equal to
                -44.1 pcm/°F.

                where:          ARO stands for All Rods Out
                                HZP stands for Hot Zero THERMAL POWER
                                RTP stands for RATED THERMAL POWER

2.3     Shutdown Rod Insertion Limit (Specification 3/4.1.3.5)

        Fully withdrawn for all shutdown rods shall be greater than or equal to 225
        steps.

2.4     Control Rod Insertion Limit (Specification 3/4.1.3.6)

        The control rod banks shall be limited in physical insertion as specified in
        Figure 2.  Fully withdrawn for all control rods shall be greater than or equal
        to 225 steps.

2.5     Axial Flux Difference (Specification 3/4.2.1)

        The AXIAL FLUX DIFFERENCE (AFD) target band is specified in Figure 3.

2.6     Heat Flux Hot Channel Factor - $F_Q(Z)$ (Specification 3/4.2.2)

        1.      The $F_Q(Z)$ Limit as referenced in TS 3.2.2 is:

                $F_Q(Z) \leq F_Q^{RTP} * K(Z)/P$ for P > 0.5

                $F_Q(Z) \leq F_Q^{RTP} * K(Z)/0.5$ for P $\leq$ 0.5

                where:

                a.      P = THERMAL POWER/RATED THERMAL POWER

                b.      $F_Q^{RTP}$ = 2.41

                c.      K(Z) = the normalized $F_Q(Z)$ as a function of core height, as
                        specified in Figure 4.  For P<0.42, K(Z) may be set equal to 1.0
                        for all axial elevations.

        2.      V(Z) Curves versus core height for PDC-3 Operation, as used in T.S.
                4.2.2, are specified in Figures 5 through 6.  The first V(Z) curve
                (Figure 5) is valid for Cycle 17 burnups from 0 up to but not including
                15000 MWD/MTU. The second V(Z) curve (Figure 6) is valid for Cycle 17
                burnups greater than or equal to 15000 MWD/MTU to a maximum cycle energy
                of 21176 MWD/MTU.

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2.7    Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}$ (Specification 3/4.2.3)

$$F_{\Delta H} \leq F_{\Delta H}^{RTP} * (1 + PF_{\Delta H} * (1 - P))$$

where:

a.  P = THERMAL POWER/RATED THERMAL POWER

b.  $F_{\Delta H}^{RTP}$ = $F_{\Delta H}$ Limit at RATED THERMAL POWER = 1.66

c.  $PF_{\Delta H}$ = Power Factor Multiplier for $F_{\Delta H}$ = 0.35

$F_{\Delta H}$ = Enthalpy rise hot channel factor obtained by using the movable incore detectors to obtain a power distribution map, with the measured value of the nuclear enthalpy rise hot channel factor ($F_{\Delta H}^{N}$) increased by an allowance of 4% to account for measurement uncertainty.

2.8    Boron Concentration for Refueling Operations (Specification 3/4.9.1.a)

Through the end of Cycle 17, the boron concentration required to maintain $K_{eff}$ less than or equal to .95 is equal to 2217 ppm.  Boron concentration must be maintained greater than or equal to 2217 ppm during refueling operations.

3.0    METHODOLOGY REFERENCES

1.    XN-75-27(A) (June 1975) and Supplements 1 (September 1976), 2 (December 1977), 3 (November 1980), 4 (December 1985), and 5 (February 1987), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," Exxon Nuclear Company, Richland, WA 99352. (Not used for Cycle 17.)

      (Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - Modes 3, 4, and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).

2.    ANF-89-151(A), and Correspondence, "ANF-RELAP Methodology for Pressurized Water Reactors:  Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, Richland, WA 99352, May 1992.

      (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

3.    XN-NF-82-21(A), Revision 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, Richland, WA 99352, September 1983.

      (Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

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3.0    METHODOLOGY REFERENCES (continued)

4.    XN-75-32(A), (April 1975) Supplements 1 (July 1979), 2 (July 1979), 3
(January 1980), and 4 (October 1983), "Computational Procedure for
Evaluating Fuel Rod Bowing," Exxon Nuclear Company, Richland, WA 99352.

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor, and
3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

5.    EMF-84-093(A), Revision 1, "Steamline Break Methodology for PWRs,"
Siemens Power Corporation, February 1999.

(Methodology for Specification 3.1.1.3 - Moderator Temperature
Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control
Bank Insertion Limits, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel
Factor).

6.    EMF-2087(A), Revision 0, "SEM/PWR-98: ECCS Evaluation Model for PWR
LBLOCA Applications," Siemens Power Corporation, June 1999.

(Methodology for Specification 3.2.1 - Axial Flux Difference,
3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise
Hot Channel Factor).

7.    XN-NF-78-44(A), "A Generic Analysis of the Control Rod Ejection Transient
for Pressurized Water Reactors," Exxon Nuclear Company,
Richland, WA 99352, October 1983.

(Methodology for Specification 3.1.3.5 - Shutdown Bank Insertion Limits,
3.1.3.6 - Control Bank Insertion Limits, and 3.2.2 - Heat Flux Hot
Channel Factor).

8.    ANF-88-054(A), "PDC-3:  Advanced Nuclear Fuels Corporation Power
Distribution Control for Pressurized Water Reactors and Application of
PDC-3 to H. B. Robinson Unit 2," Advanced Nuclear Fuels Corporation,
Richland, WA 99352, October 1990.

(Methodology for Specification 3.2.1 - Axial Flux Difference, and
3.2.2 - Heat Flux Hot Channel Factor).

9.    AREVA NP Setpoint methodology as described by:

EMF-92-081(A), and Supplement 1, "Statistical Setpoint/Transient
Methodology for Westinghouse Type Reactors," Siemens Power Corporation,
Richland, WA 99352, February 1994.

EMF-92-081(A), Revision 1, "Statistical Setpoint/Transient Methodology
for Westinghouse Type Reactors," Siemens Power Corporation, February
2000.

(Methodology for Specification 3.1.1.3 - Moderator Temperature
Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control
Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux
Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel
Factor).

10.    EMF-92-153(A), Revision 1, "HTP: Departure from Nucleate Boiling
Correlation for High Thermal Performance Fuel," Siemens Nuclear Power
Corporation, Richland, WA 99352, January 2005.

(Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel
Factor).

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3.0   METHODOLOGY REFERENCES (continued)

11.   XN-NF-82-49(A), Revision 1, April 1989 and XN-NF-82-49(P), Revision 1,
      Supplement 1, December 1994, "Exxon Nuclear Company Evaluation Model EXEM
      PWR Small Break Model," Exxon Nuclear Company, Richland, WA 99352.

      (Methodology for Specification 3.2.1 - Axial Flux Difference,
      3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise
      Hot Channel Factor).

12.   EMF-96-029(A), Volumes 1 and 2, "Reactor Analysis Systems for PWRs,
      Volume 1 - Methodology Description, Volume 2 - Benchmarking Results,"
      Siemens Power Corporation, January 1997.

      (Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - Modes 3, 4,
      and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown
      Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits,
      3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor,
      3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron
      Concentration).

13.   EMF-2328 (A), Revision 0, "PWR Small Break LOCA Evaluation Model, S-
      RELAP5 Based," Siemens Power Corporation, March 2001
      (Not used in Cycle 17)

      (Methodology for Specification 3.2.1 - Axial Flux Difference, and
      3.2.2 - Heat Flux Hot Channel Factor), and 3.2.3 - Nuclear Enthalpy Rise
      Hot Channel Factor).

14.   Mechanical Design Methodologies

      XN-NF-81-58(A), Revision 2 and Supplements 1 and 2, "RODEX2 Fuel Rod
      Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company,
      March 1984.

      ANF-81-58(A), Revision 2 and Supplements 3 and 4, "RODEX2 Fuel Rod
      Thermal Mechanical Response Evaluation Model," Advanced Nuclear Fuels
      Corporation, June 1990.

      XN-NF-82-06(A), Revision 1 and Supplements 2, 4, and 5, "Qualification of
      Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company, October
      1986.

      ANF-88-133(A), and Supplement 1, "Qualification of Advanced Nuclear
      Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced
      Nuclear Fuels Corporation, December 1991.

      XN-NF-85-92(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation
      Examination and Thermal Conductivity Results," Exxon Nuclear Company,
      November 1986.

      EMF-92-116(A), Revision 0, "Generic Mechanical Design Criteria for PWR
      Fuel Designs," Siemens Power Corporation, February 1999.

      (Methodologies for Specification 3.2.1 - Axial Flux Difference,
      3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise
      Hot Channel Factor).

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4.0   OTHER REQUIREMENTS

4.1   Movable Incore Detection System

1.   Operability: The Movable Incore Detection System shall be OPERABLE with:

R       a.   At least 38 detector thimbles at the beginning of cycle (where the beginning of cycle is defined in this instance as a flux map determination that the core is loaded consistent with design),

        b.   A minimum of 38 detector thimbles for the remainder of the operating cycle,

        c.   A minimum of two detector thimbles per core quadrant, and

        d.   Sufficient movable detectors, drive, and readout equipment to map these thimbles.

2.   Applicability: When the Movable Incore Detection System is used for:

        a.   Recalibration of the Excore Neutron Flux Detection System, or

        b.   Monitoring the QUADRANT POWER TILT RATIO, or

        c.   Measurement of $F_{\Delta H}$ and $F_Q(Z)$

3.   Surveillance Requirements: The Movable Incore Detection System shall be demonstrated OPERABLE, within 24 hours prior to use, by irradiating each detector used and determining the acceptability of its voltage curve when required for:

        a.   Recalibration of the Excore Neutron Flux Detection System, or

        b.   Monitoring the QUADRANT POWER TILT RATIO, or

        c.   Measurement of $F_{\Delta H}$ and $F_Q(Z)$

4.   Bases

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F_{\Delta H}$, a full incore flux map is used.

Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring QUADRANT POWER TILT RATIO when one Power Range channel is inoperable.

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4.0   OTHER REQUIREMENTS (continued)

R    5.    Evaluation Requirements

In order to change the requirements concerning the number and location of operable detectors, the NRC staff deems that a rigorous evaluation and justification is required. The following is a list of elements that must be part of a 50.59 determination and available for audit if the licensee wishes to change the requirements:

a.    How an inadvertent loading of a fuel assembly into an improper location will be detected,

b.    How the validity of the tilt estimates will be ensured,

c.    How adequate core coverage will be maintained,

d.    How the measurement uncertainties will be assured and why the added uncertainties are adequate to guarantee that measured nuclear heat flux hot channel factor, nuclear enthalpy rise hot channel factor, radial peaking factor and quadrant power tilt factor meet Technical Specification limits, and

e.    How the Movable Incore Detection System will be restored to full (or nearly full) service before the beginning of each cycle.

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# Figure 1
# Shutdown Margin Versus RCS Boron Concentration
# Modes 3, 4, and 5/Drained
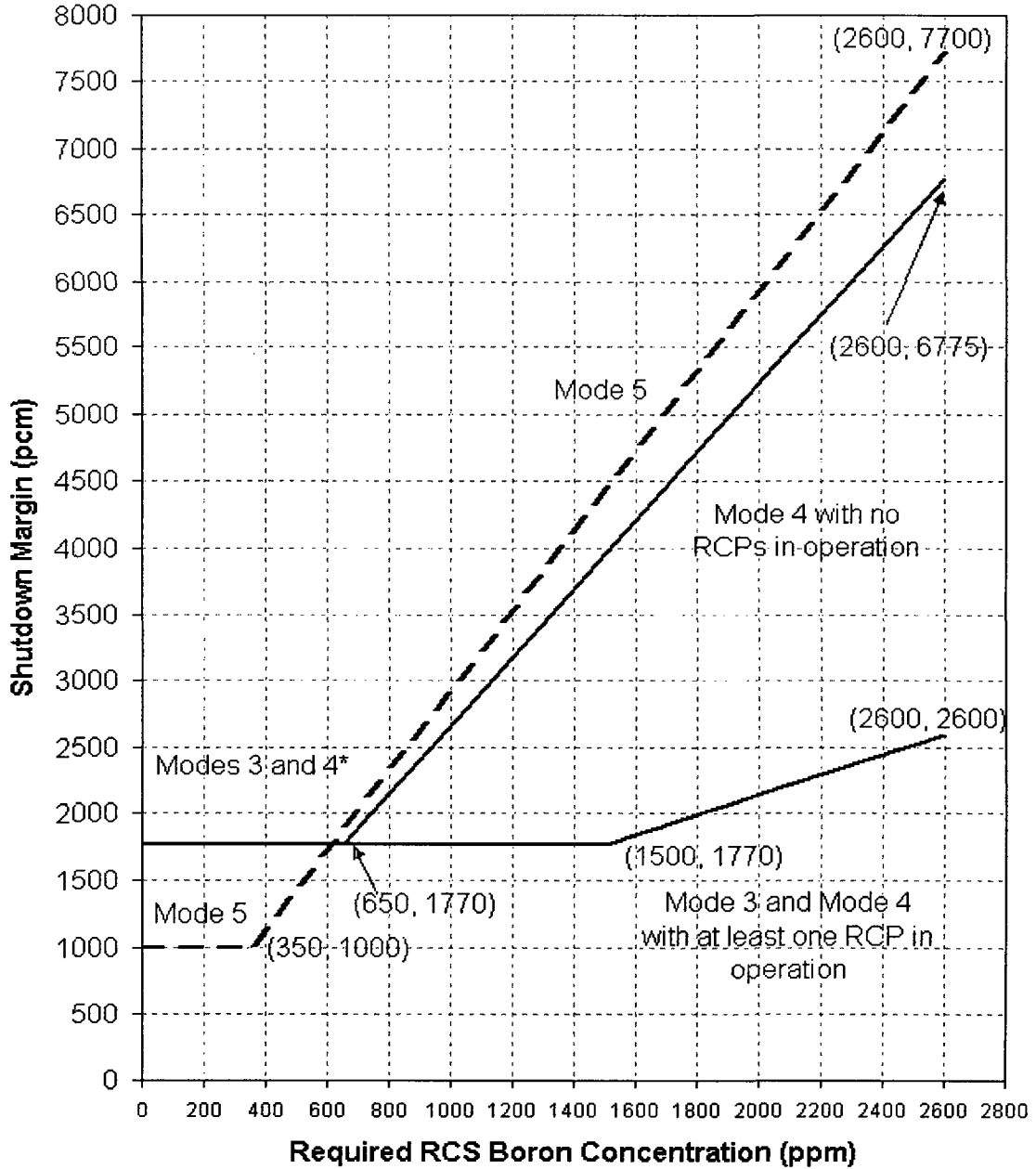


* Applicable to Mode 4, with or without RCPs in operation

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# Figure 2
# Rod Group Insertion Limits Versus Thermal Power
# (Three Loop Operation)
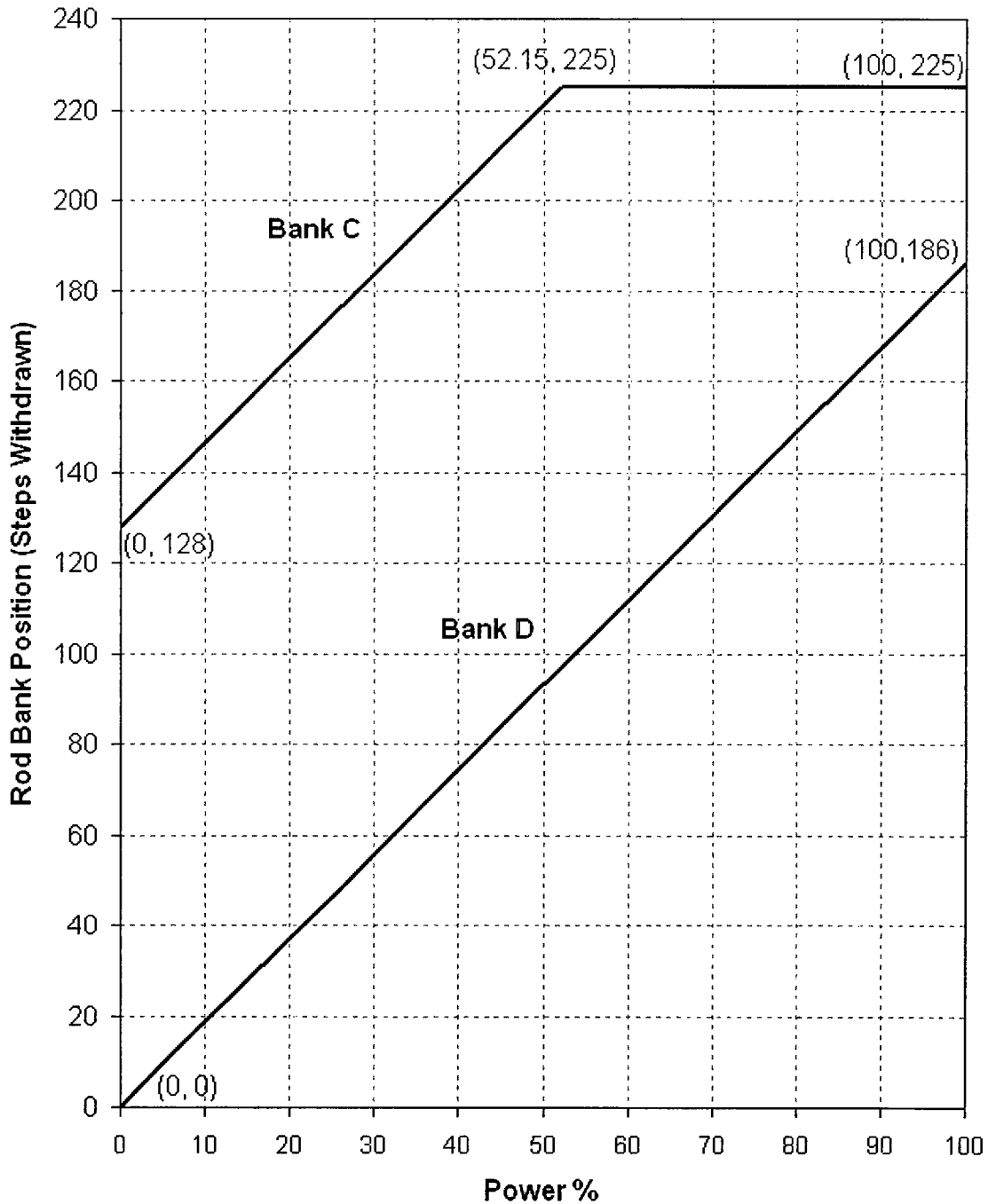


Notes: 1. Fully withdrawn position shall be greater than or equal to 225 steps.
2. Control Banks A and B must be withdrawn from the core prior to power operation.

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# Figure 3
# Axial Flux Difference Limits as a Function of
# Rated Thermal Power



Axial Flux Difference (% Delta I)
(Deviation from Target AFD)
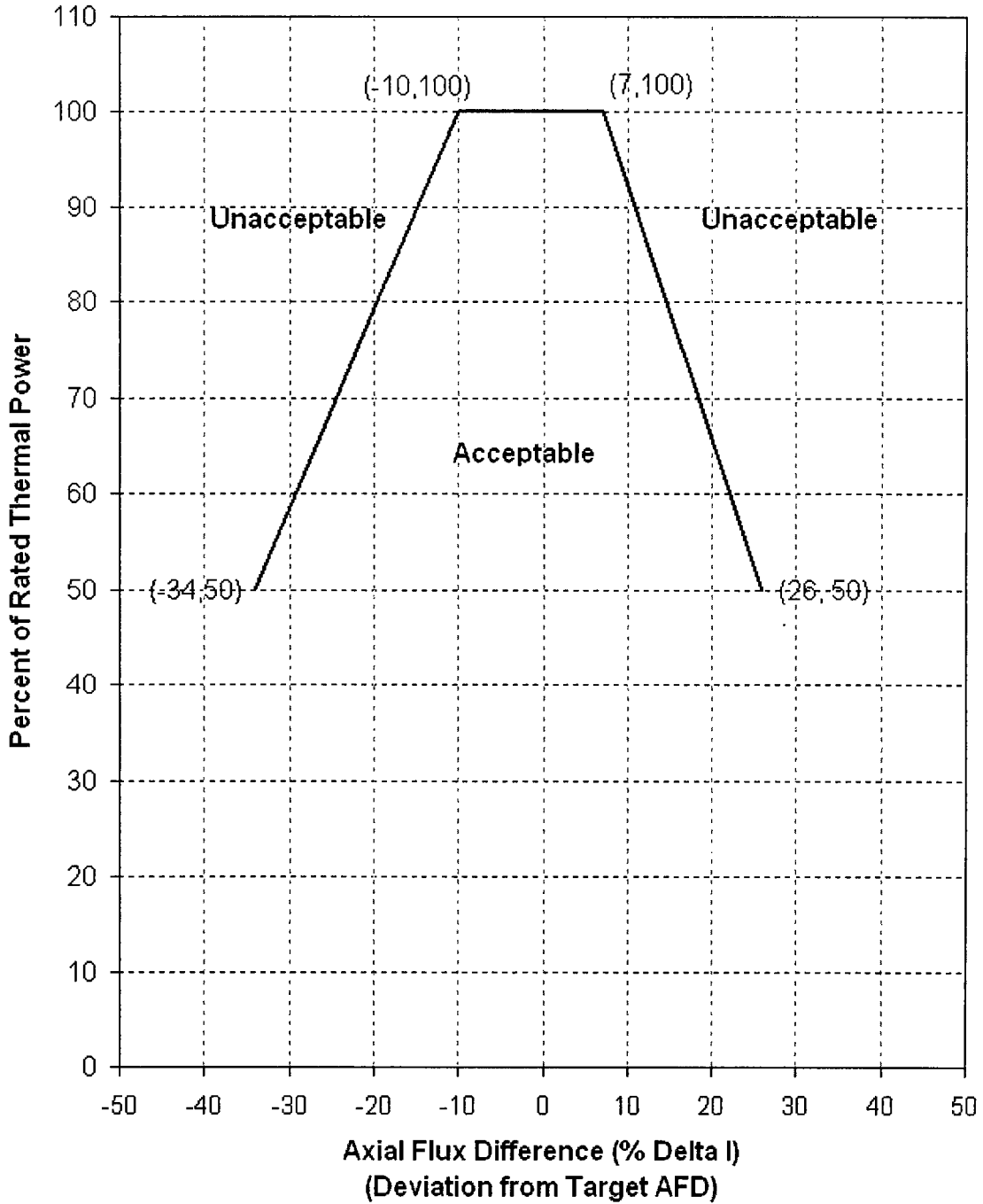
Note: At power levels less than HFP, the deviation is applied to the target AFD
    appropriate to that power level. The target AFD varies linearly between
    the HFP target and zero at zero power.

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# Figure 4

# K(Z) - Local Axial Penalty Function for $F_Q(Z)$
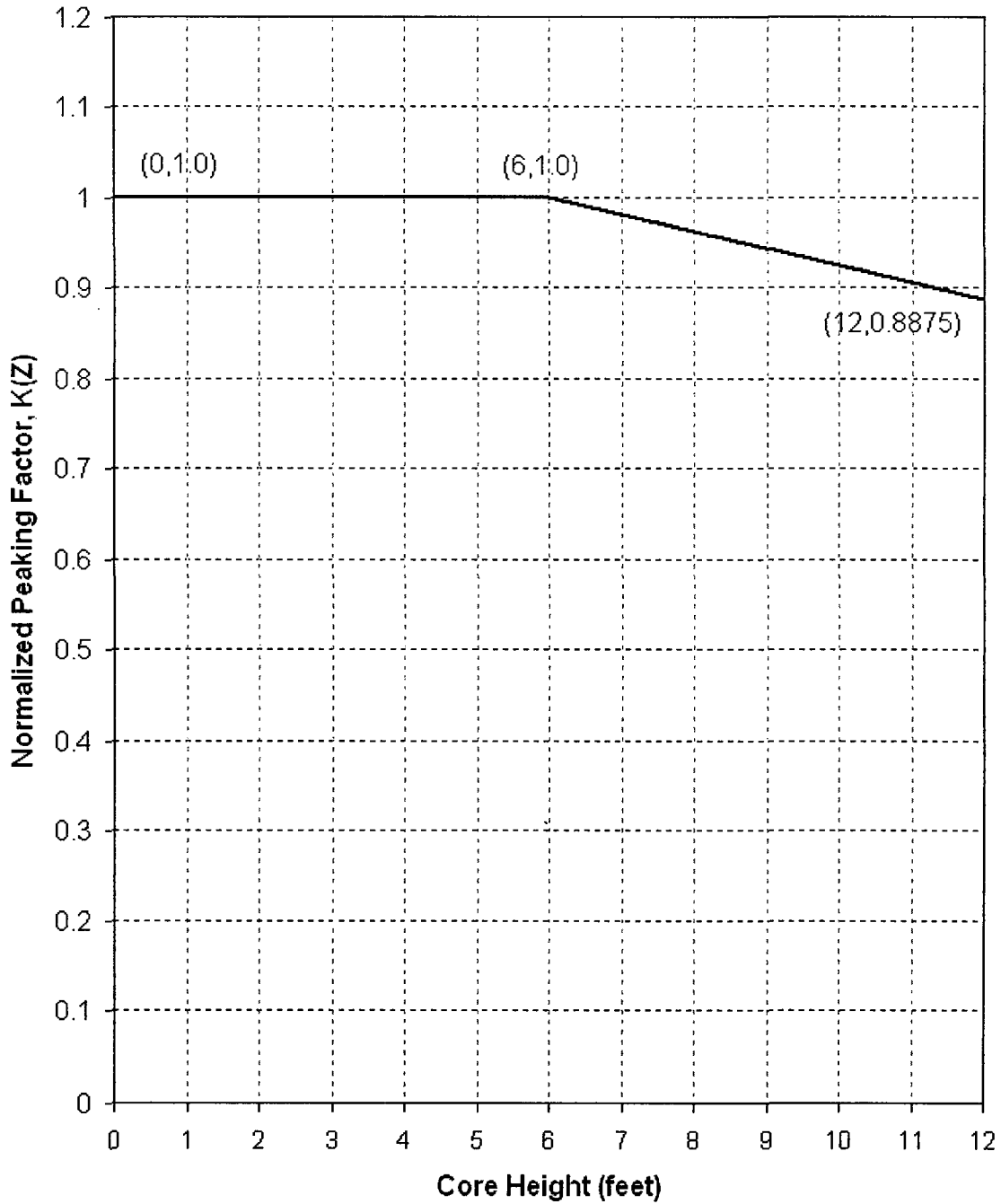


Note: For power levels below 42% RTP, the K(Z) at all axial elevations is 1.0. It is conservative to apply
the above figure to all power levels below 42% RTP.

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# Figure 5
## Cycle 17 V(Z) Versus Core Height
## 0 MWD/MTU ≤ burnup < 15000 MWD/MTU



| Height (feet) | V(Z) |
|---|---|
| 0.0 | 1.000 |
| 0.2 | 1.000 |
| 0.4 | 1.000 |
| 0.6 | 1.000 |
| 0.8 | 1.000 |
| 1.0 | 1.000 |
| 1.2 | 1.000 |
| 1.4 | 1.000 |
| 1.6 | 1.000 |
| 1.8 | 1.190 |
| 2.0 | 1.182 |
| 2.2 | 1.173 |
| 2.4 | 1.164 |
| 2.6 | 1.155 |
| 2.8 | 1.144 |
| 3.0 | 1.135 |
| 3.2 | 1.126 |
| 3.4 | 1.118 |
| 3.6 | 1.116 |
| 3.8 | 1.122 |
| 4.0 | 1.128 |
| 4.2 | 1.134 |
| 4.4 | 1.138 |
| 4.6 | 1.141 |
| 4.8 | 1.144 |
| 5.0 | 1.152 |
| 5.2 | 1.157 |
| 5.4 | 1.163 |
| 5.6 | 1.168 |
| 5.8 | 1.170 |
| 6.0 | 1.171 |
| 6.2 | 1.170 |
| 6.4 | 1.167 |
| 6.6 | 1.163 |
| 6.8 | 1.161 |
| 7.0 | 1.164 |
| 7.2 | 1.166 |
| 7.4 | 1.166 |
| 7.6 | 1.164 |
| 7.8 | 1.162 |
| 8.0 | 1.158 |
| 8.2 | 1.151 |
| 8.4 | 1.144 |
| 8.6 | 1.135 |
| 8.8 | 1.125 |
| 9.0 | 1.114 |
| 9.2 | 1.111 |
| 9.4 | 1.111 |
| 9.6 | 1.114 |
| 9.8 | 1.118 |
| 10.0 | 1.123 |
| 10.2 | 1.130 |
| 10.4 | 1.000 |
| 10.6 | 1.000 |
| 10.8 | 1.000 |
| 11.0 | 1.000 |
| 11.2 | 1.000 |
| 11.4 | 1.000 |
| 11.6 | 1.000 |
| 11.8 | 1.000 |
| 12.0 | 1.000 |

• Top and bottom 15% excluded per Technical Specification 4.2.2.2.g

• For all power levels below 50% RTP, the v(z) data at all axial levels is 1.0. It is conservative to apply the above figure to power levels below 50% RTP.

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# Figure 6
## Cycle 17 V(Z) Versus Core Height
## 15000 MWD/MTU ≤ burnup ≤ 21176 MWD/MTU



| Height (feet) | | V(Z) |
|---|---|---|
| 0.0 | • | 1.000 |
| 0.2 | • | 1.000 |
| 0.4 | • | 1.000 |
| 0.6 | • | 1.000 |
| 0.8 | • | 1.000 |
| 1.0 | • | 1.000 |
| 1.2 | • | 1.000 |
| 1.4 | • | 1.000 |
| 1.6 | • | 1.000 |
| 1.8 | | 1.201 |
| 2.0 | | 1.190 |
| 2.2 | | 1.178 |
| 2.4 | | 1.167 |
| 2.6 | | 1.155 |
| 2.8 | | 1.144 |
| 3.0 | | 1.133 |
| 3.2 | | 1.125 |
| 3.4 | | 1.125 |
| 3.6 | | 1.132 |
| 3.8 | | 1.143 |
| 4.0 | | 1.153 |
| 4.2 | | 1.162 |
| 4.4 | | 1.172 |
| 4.6 | | 1.181 |
| 4.8 | | 1.189 |
| 5.0 | | 1.196 |
| 5.2 | | 1.202 |
| 5.4 | | 1.206 |
| 5.6 | | 1.209 |
| 5.8 | | 1.208 |
| 6.0 | | 1.205 |
| 6.2 | | 1.210 |
| 6.4 | | 1.217 |
| 6.6 | | 1.223 |
| 6.8 | | 1.226 |
| 7.0 | | 1.228 |
| 7.2 | | 1.227 |
| 7.4 | | 1.223 |
| 7.6 | | 1.219 |
| 7.8 | | 1.211 |
| 8.0 | | 1.202 |
| 8.2 | | 1.190 |
| 8.4 | | 1.176 |
| 8.6 | | 1.160 |
| 8.8 | | 1.141 |
| 9.0 | | 1.123 |
| 9.2 | | 1.109 |
| 9.4 | | 1.105 |
| 9.6 | | 1.106 |
| 9.8 | | 1.107 |
| 10.0 | | 1.117 |
| 10.2 | | 1.142 |
| 10.4 | • | 1.000 |
| 10.6 | • | 1.000 |
| 10.8 | • | 1.000 |
| 11.0 | • | 1.000 |
| 11.2 | • | 1.000 |
| 11.4 | • | 1.000 |
| 11.6 | • | 1.000 |
| 11.8 | • | 1.000 |
| 12.0 | • | 1.000 |

- Top and bottom 15% excluded per Technical Specification 4.2.2.2.g

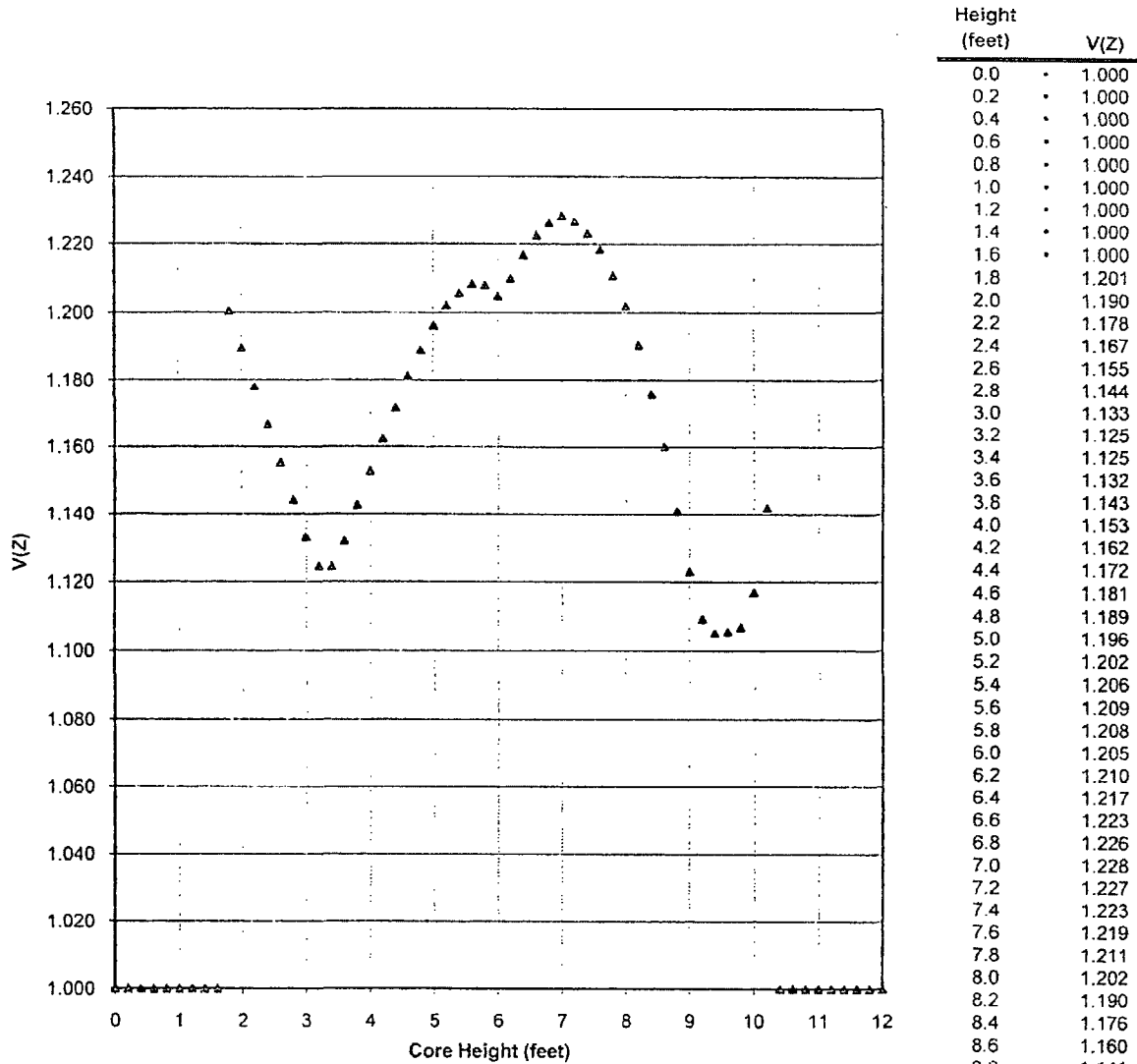- For all power levels below 50% RTP, the v(z) data at all axial levels is 1.0. It is conservative to apply the above figure to power levels below 50% RTP.