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TS 5.6.5

April 12, 2021

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Calvert Cliffs Nuclear Power Plant, Unit No. 1
Renewed Facility Operating License No. DPR-69
NRC Docket No. 50-318

Subject: **Core Operating Limits Report for Unit 2, Cycle 24, Revision 0**

Pursuant to Calvert Cliffs Nuclear Power Plant Technical Specification 5.6.5, the attached Core Operating Limits Report for Unit 2, Cycle 24, Revision 0 (Attachment 1), is provided for your records.

Please replace the Unit 2 Core Operating Limits Report in its entirety, with the attached Revision 0.

There are no regulatory commitments contained in this correspondence.

Should you have questions regarding this matter, please contact Mr. Larry D. Smith at (410) 495-5219.

Respectfully,

Joseph A. Dullinger
Plant Manager

JAD/LDS/lmd

Attachment: (1) Core Operating Limits Report for Unit 2, Cycle 24, Revision 0

<u>ATTACHMENT (1)</u>

# CORE OPERATING LIMITS REPORT

# FOR

# UNIT 2, CYCLE 24, REVISION 0

Calvert Cliffs Nuclear Power Plant

April 12, 2021

# Calvert Cliffs Nuclear Power Plant Core Operating Limits Report

## COLR

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## Unit 2 Cycle 24

**Revision 0**

Effective Date: 03/03/2021

Schearer, Timothy A Digitally signed by Schearer, Timothy A DN: cn=Schearer, Timothy A Date: 2021.02.16 12:43:05 -05'00'

**Responsible Engineer / Date**

Digitally signed by Kelliher, Andrew P DN: cn=Kelliher, Andrew P Date: 2021.02.16 12:53:06 -05'00'

**Independent Reviewer / Date**

Broderick, Alexander J. Digitally signed by Broderick, Alexander J. DN: cn=Broderick, Alexander J. Date: 2021.02.17 16:07:50 -05'00'

**Station Qualified Reviewer / Date**

2021.02.18 10:51:25 -06'00'

**Sr. Manager – PWR Core Design / Date**

Exelon Generation.

**CORE OPERATING LIMITS REPORT**
**CALVERT CLIFFS UNIT 2, CYCLE 24**

The following limits are included in this Core Operating Limits Report:



The following figures are included in this Core Operating Limits Report:

**UNIT 2**
**CORE OPERATING LIMITS REPORT**
**LIST OF EFFECTIVE PAGES**

| Page No. | Rev. No. |
|---|---|
| 1 | 0 |
| 2 | 0 |
| 3 | 0 |
| 4 | 0 |
| 5 | 0 |
| 6 | 0 |
| 7 | 0 |
| 8 | 0 |
| 9 | 0 |
| 10 | 0 |
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## INTRODUCTION

This report provides the cycle-specific limits for operation of Calvert Cliffs Unit 2, Cycle 24. It contains the limits for:

Shutdown Margin (SDM)
Moderator Temperature Coefficient (MTC)
Control Element Assembly (CEA) Alignment
Regulating Control Element Assembly (CEA) Insertion Limits
Linear Heat Rate (LHR)
Total Integrated Radial Peaking Factor ($F_r^T$)
Axial Shape Index (ASI)
Reactor Protective System (RPS) Instrumentation – Operating
RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits
Boron Concentration

In addition, this report contains a number of figures which give limits on the parameters listed above. If any of the limits contained in this report are exceeded, corrective action will be taken as defined in the Technical Specifications.

This report has been prepared in accordance with the requirements of Technical Specifications. The cycle specific limits have been developed using the NRC-approved methodologies given in the "List of Approved Methodologies" section of this report and in the Technical Specifications.

## COLR Revision 0

Initial release of the Unit 2 Cycle 24 (U2C24) COLR. U2C24 may operate in all plant modes.

## DEFINITIONS

### Axial Shape Index (ASI)

ASI shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core.

$$ASI = \frac{\textbf{lower} - \textbf{upper}}{\textbf{lower} + \textbf{upper}} = Y_E$$

The Axial Shape Index ($Y_I$) used for the trip and pretrip signals in the Reactor Protection System (RPS) is the above value ($Y_E$) modified by an appropriate multiplier (A) and a constant (B) to determine the true core axial power distribution for that channel.

$$Y_I = AY_E + B$$

### Total Integrated Radial Peaking Factor - $F_r^T$

The Total Integrated Radial Peaking Factor is the ratio of the peak pin power to the average pin power in an unrodded core.

## LICENSING RESTRICTIONS

1) For the Asymmetric Steam Generator Transient analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.8, the methodology shall be revised to capture the asymmetric core inlet temperature distribution and application of local peaking augmentation factors. The revised methodology shall be applied to Calvert Cliffs Unit 2 core reload designs starting with Cycle 19.

2) For the Seized Rotor Event analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.8, the methodology shall be revised to capture the asymmetric core inlet flow distribution. The revised methodology shall be applied to Calvert Cliffs Unit 2 core reload designs starting with Cycle 19.

3) For the Control Element Assembly Ejection analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.11, the cycle-specific hot zero power peak average radial fuel enthalpy is calculated based on a modified power dependent insertion limit with Control Element Assembly Bank 3 assumed to be fully inserted (only in the analysis, not in actual plant operations). This revised methodology shall be applied to Calvert Cliffs Unit 2 core reload designs starting with Cycle 19.

4) The Small Break Loss of Coolant accident performed in accordance with the methodology of Technical Specification 5.6.5.b.9 shall be analyzed using a break spectrum with augmented detail related to break size. This revised methodology shall be applied to Calvert Cliffs Unit 2 core reload designs starting with Cycle 19.

5) Core Operating Limits Report Figures 3.1.6, 3.2.3, and 3.2.5 shall not be changed without prior NRC review and approval until an NRC-accepted generic, or Calvert Cliffs-specific, basis is developed for analyzing the Control Element Assembly Rod Bank Withdrawal Event, the Control Element Assembly Drop, and the Control Element Assembly Ejection (power level-sensitive transients) at full power conditions only.

6) Approval of the use of S-RELAP5 (Technical Specification 5.6.5.b.8) is restricted only to those safety analyses that confirm acceptable transient performance relative to the specified acceptable fuel design limits. Prior transient specific NRC approval is required to analyze transient performance relative to reactor coolant pressure boundary integrity until NRC-approval is obtained for a generic or Calvert Cliffs-specific basis for the use of the methodology in Technical Specification 5.6.5.b.8 to demonstrate reactor coolant pressure boundary integrity.

| NOTE: The NRC has issued a letter that allows S-RELAP5 to be used for the transient-specific application of the methodology to CCNPP only as described in the letter pertaining to PSV setpoints. It is not a generic approval of the methodology. |
|---|

Ref: Letter from Alexander N. Chereskin (NRC) to Bryan C. Hanson (Exelon) dated December 30, 2015, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 – Issuance of Amendment Re: Revision to Pressurizer Safety Valve Technical Specifications (CAC Nos. MF3541 and MF3542)

7) For the RODEX2-based fuel thermal-mechanical design analysis performed in accordance with the methodology of Technical Specification 5.6.5.b.3, Calvert Cliffs Unit 2 core reload designs (starting with Cycle 19) shall satisfy the following criteria:

   a. Predicted rod internal pressure shall remain below the steady state system pressure.

   b. The linear heat generation rate fuel centerline melting safety limit shall remain below 21.0 KW/ft.

8) For the Control Element Assembly Ejection analysis, Calvert Cliffs Unit 2 core reloads (starting with Cycle 19) shall satisfy the following criteria:

   a. Predicted peak radial average fuel enthalpy when calculated in accordance with the methodology of Technical Specification 5.6.5.b.11 shall remain below 200 cal/g.

   b. For the purpose of evaluating radiological consequences, should the S-RELAP5 hot spot model predict fuel temperature above incipient centerline melt conditions when calculated in accordance with the methodology of Technical Specification 5.6.5.b.8, a conservative radiological source term (in accordance with Regulatory Guide 1.183, Revision 0) shall be applied to the portion of fuel beyond incipient melt conditions (and combined with existing gap source term), and cladding failure shall be presumed.

9) The approval of the emergency core cooling system evaluation performed in accordance with the methodology of Technical Specification 5.6.5.b.7 shall be valid only for Calvert Cliffs Unit 2, Cycle 19. To remove this condition, Calvert Cliffs shall obtain NRC approval of the analysis of once- and twice-burned fuel for core designs following Unit 2 Cycle 19.

| |
|---|
| <u>NOTE</u>: The revised methodology was submitted and received NRC approval in December 2012. This license condition is satisfied; however since NRC approval was obtained via letter and not LAR, this license condition is still listed in Appendix C of the Tech. Specs. and has been retained here for consistency. |

Ref: Letter from Douglas V. Pickett (NRC) to George H. Gellrich (CCNPP) dated February 18, 2011, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 – Amendment Re: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel (TAC Nos. ME2831 and ME2832)

10) Up to two Framatome PROtect™ Lead Test Assemblies utilizing Chromium-coated M5® cladding and chromia doped pellets may be placed in limiting regions of the core for up to 3 cycles commencing with the implementation of Amendment 317.

11) The safety limits specified in TS 2.1.1.2 regarding fuel centerline melt temperature for Framatome fuel, <5081°F, decreasing by 58°F per 10,000 MWD/MTU and adjusted for burnable poison per XN-NF-79-56 (P)(A), Revision 1, Supplement 1 is not applicable for the Framatome PROtect™ Lead Test Assemblies utilizing Chromium-coated M5® cladding and chromia doped pellets for up to 3 cycles commencing with the implementation of Amendment 317.

12) The requirement that the RODEX2 predicted rod internal pressure shall remain below the steady state system pressure is not applicable for the Framatome PROtect™ Lead Test Assemblies utilizing Chromium coated M5® cladding and chromia doped pellets for up to 3 cycles commencing with the implementation of Amendment 317.
   - License Condition #12 supersedes License Condition #7a only for the Lead Test Assembly.

## CYCLE SPECIFIC LIMITS FOR UNIT 2, CYCLE 24

### 3.1.1 Shutdown Margin (SDM) (SR 3.1.1.1)

*Tavg > 200 °F - Modes 3 and 4:*

The shutdown margin shall be ≥ 3.5% $\Delta\rho$.

*Tavg ≤ 200 °F - Mode 5:*

The shutdown margin shall be ≥ 3.0% $\Delta\rho$.

### 3.1.3 Moderator Temperature Coefficient (MTC) (SR 3.1.3.2)

The Moderator Temperature Coefficient (MTC) shall be less negative than -3.1 x $10^{-4}$ $\Delta\rho$/°F at rated thermal power.

### 3.1.4 Control Element Assembly (CEA) Alignment (Action 3.1.4.B.1)

The allowable time to realign a CEA is 120 minutes when the pre-misaligned $F_r^T$ is ≤ 1.65 and zero (0) minutes when the pre-misaligned $F_r^T$ is > 1.65.

The pre-misaligned $F_r^T$ value used to determine the allowable time to realign the CEA shall be the latest measurement taken within 5 days prior to the CEA misalignment. If no measurements have been taken within 5 days prior to the misalignment and the full core power distribution monitoring system is unavailable then the time to realign is zero (0) minutes.

### 3.1.6 Regulating Control Element Assembly (CEA) Insertion Limits (SR 3.1.6.1 and SR 3.1.6.2)

The regulating CEA groups insertion limits are shown on COLR Figure 3.1.6.

Figure 3.1.6 will not be changed unless the requirements in Licensing Restriction 5 are met.

### 3.2.1 Linear Heat Rate (LHR) (SR 3.2.1.2 and SR 3.2.1.4)

The linear heat rate shall not exceed the limits shown on COLR Figure 3.2.1-1.

The axial shape index power dependent control limits are given in COLR Figure 3.2.1-2.

When using the excore detector monitoring system (SR 3.2.1.2):

The alarm setpoints are equal to or less than the ASI limits; therefore when the alarms are adjusted, they provide indication to the operator that ASI is not within the limits.

The axial shape index alarm setpoints are shown as a function of fraction of thermal power on COLR Figure 3.2.1-2.

When using the incore detector monitoring system (SR 3.2.1.4):

The alarm setpoints are adjusted to protect the Linear Heat Rate limits shown on COLR Figure 3.2.1-1 and uncertainty factors are appropriately included in the setting of these alarms.

The uncertainty factors for the incore detector monitoring system are:

1. A measurement-calculational uncertainty factor of 1.07
2. An engineering uncertainty factor of 1.03,

3.a For measured thermal power less than or equal to 50 percent but greater than 20 percent of rated full core power a thermal power measurement uncertainty factor of 1.035.

3.b For measured thermal power greater than 50 percent of rated full core power a thermal power measurement uncertainty factor of 1.020.

### 3.2.3 Total Integrated Radial Peaking Factor ($F_r^T$) (SR 3.2.3.1)

The calculated value of $F_r^T$ **shall be limited to** $\leq 1.65$.

If the calculated $F_r^T$ exceeds the above limit, the allowable combinations of thermal power, CEA position, and $F_r^T$ are shown on COLR Figure 3.2.3.

Figure 3.2.3 will not be changed unless the requirements in Licensing Restriction 5 are met.

### 3.2.5 Axial Shape Index (ASI) (SR 3.2.5.1)

The axial shape index and thermal power shall be maintained equal to or less than the limits of COLR Figure 3.2.5 for CEA insertions specified by COLR Figure 3.1.6.

Figure 3.2.5 will not be changed unless the requirements in Licensing Restriction 5 are met.

### 3.3.1 Reactor Protective System (RPS) Instrumentation - Operating (Reactor Trip Setpoints) (TS Table 3.3.1-1)

The Axial Power Distribution - High trip setpoint and allowable values are given in COLR Figure 3.3.1-1.

The Thermal Margin/Low Pressure (TM/LP) trip setpoint is given in COLR Figures 3.3.1-2 and 3.3.1-3. The allowable values are to be not less than the larger of (1) 1875 psia or (2) the value calculated from COLR Figures 3.3.1-2 and 3.3.1-3.

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

The RCS DNB parameters for pressurizer pressure, cold leg temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure $\geq$ 2200 psia;

b. RCS cold leg temperature ($T_c$) $\leq$ 548°F; and

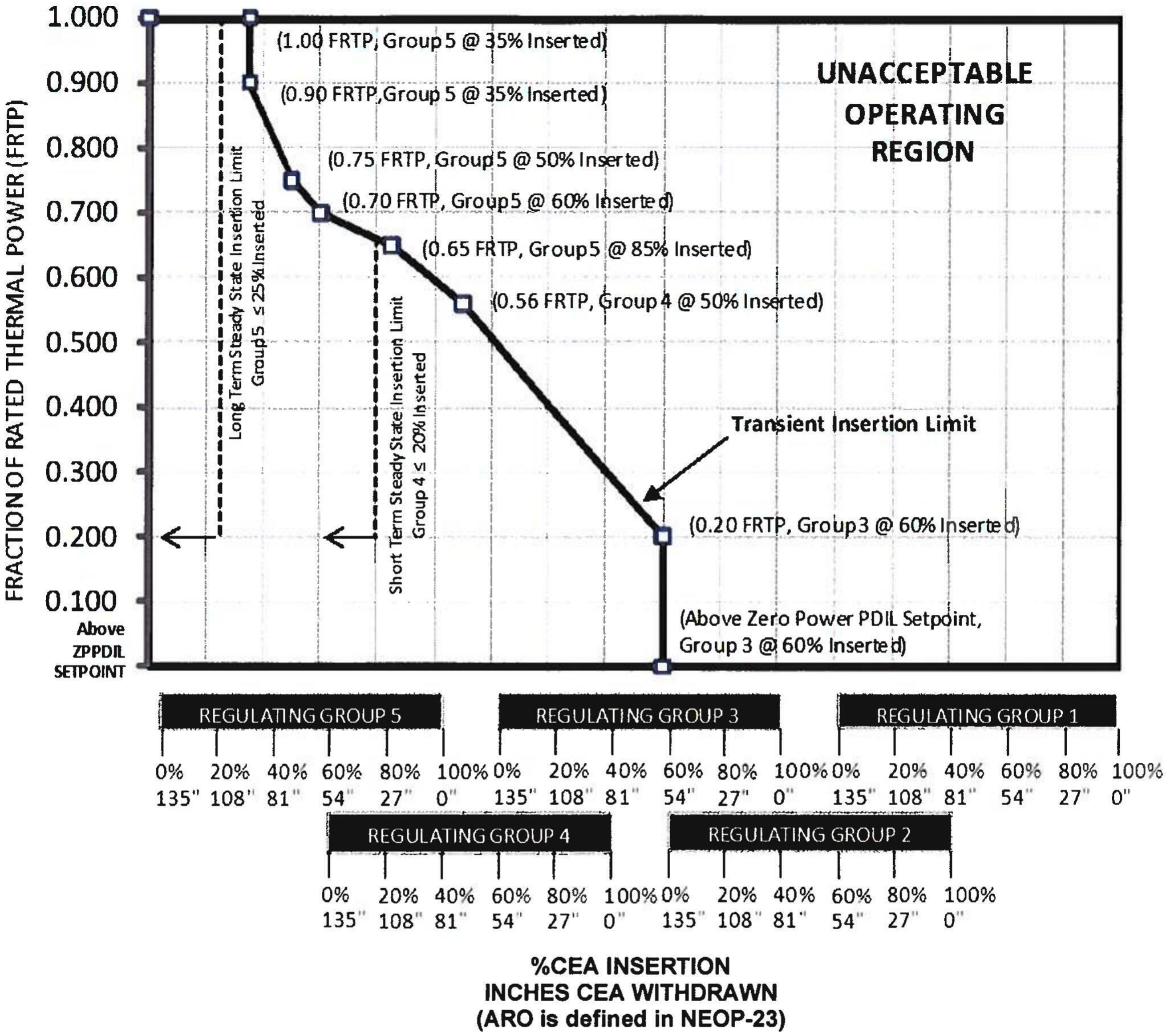
c. RCS total flow rate $\geq$ 370,000 gpm.

### 3.9.1 Boron Concentration (SR 3.9.1.1)

The refueling boron concentration will maintain the $k_{eff}$ at 0.95 or less (including a 1% $\Delta k/k$ conservative allowance for uncertainties). The refueling boron concentration shall be maintained uniform. For Mode 6 operation the RCS temperature must be maintained $\leq$ 140°F.

## U2C24 Refueling Boron Concentration Limits

| | UNIT 2 CYCLE 24 | |
|---|---|---|
| U2C24 Cycle Average Exposure | 0 GWD/MTU | ≥ 16 GWD/MTU |
| Number of Credited CEAs | 0 | 0 |
| Post-Refueling UGS or RV Head Lift Height Restrictions. | No Restriction | No Restriction |
| Minimum Required Refueling Boron Concentration:<br><br>This number includes:<br>▪ Chemistry Sampling Uncertainty<br>▪ Boron-10 Depletion Allowance<br>▪ Margin for dilution of refueling pool between low and high level alarms<br>▪ Allowance for temporary rotations of fuel assemblies<br>▪ Extra Conservatism for unlimited number of empty locations during refueling operations. | ≥ 2560 ppm | ≥ 2560 ppm |




<u>Note</u>:

Per Tech Spec Bases 3.1.5 and 3.1.6, CEAs are considered to be fully withdrawn at 129 inches.

**Figure 3.1.6**

**CEA Group Insertion Limits vs. Fraction of Rated Thermal Power**

This figure cannot be changed without prior NRC approval.

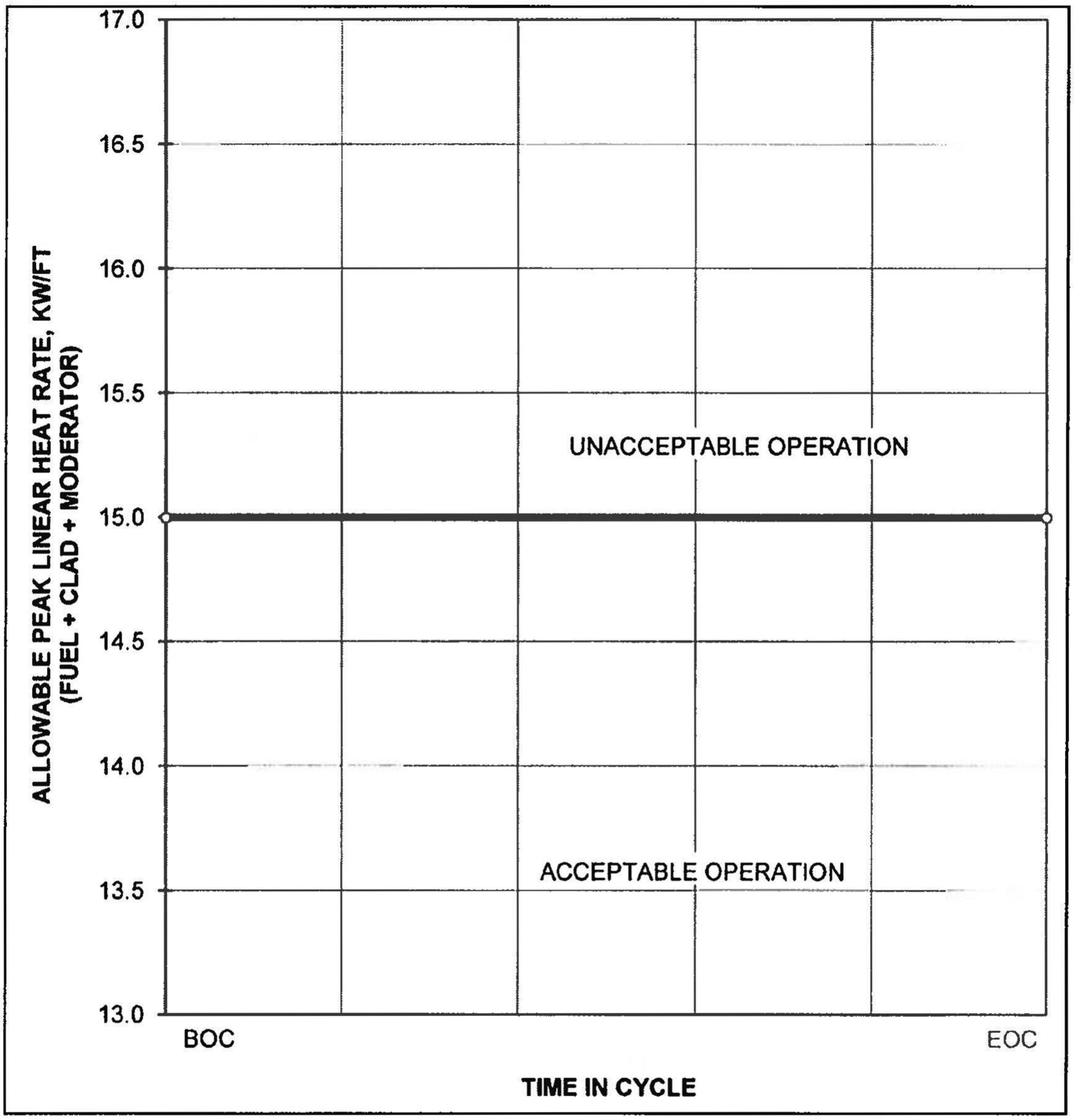



**Figure 3.2.1-1**

**Allowable Peak Linear Heat Rate vs. Time in Cycle**

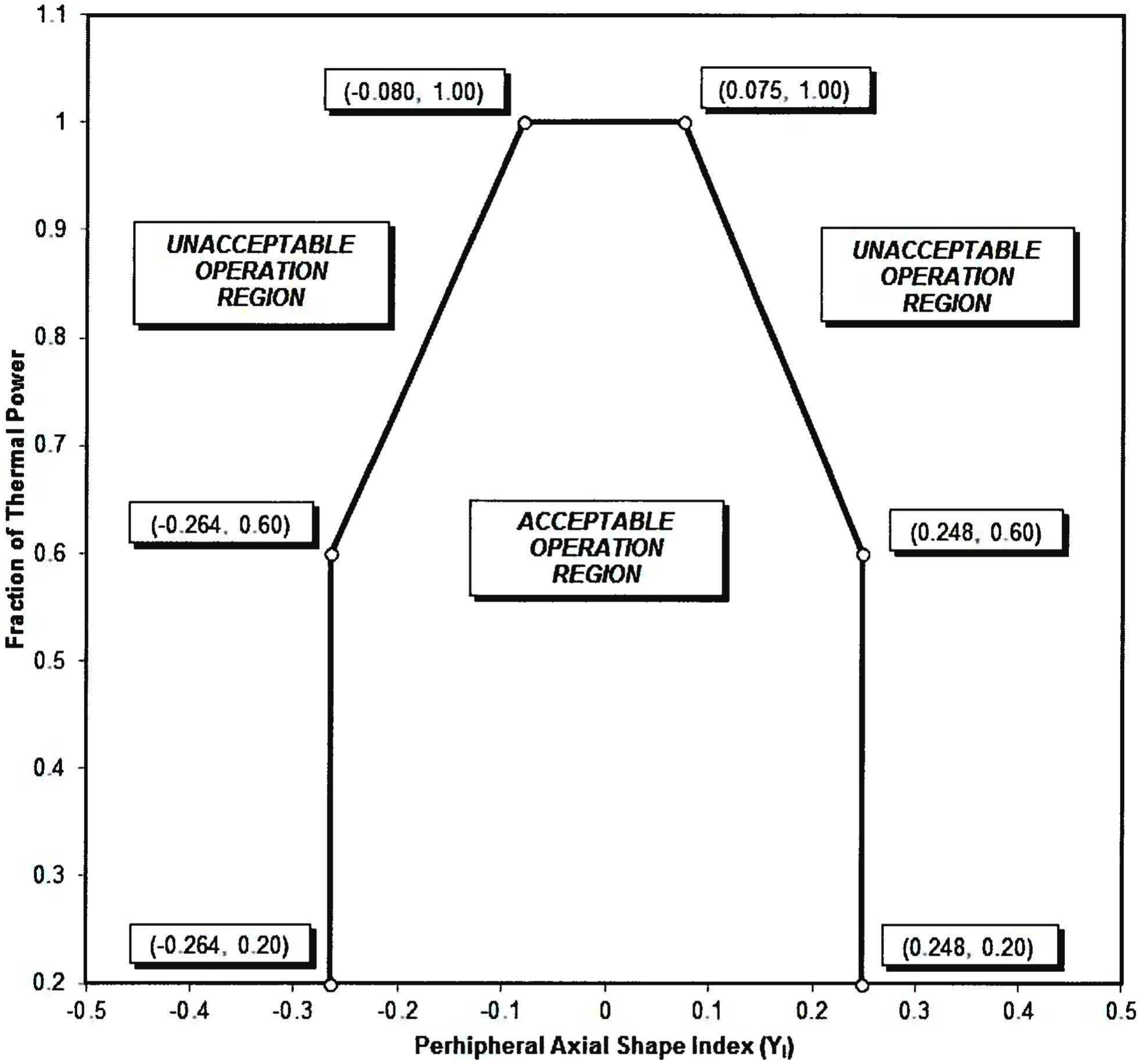



**Figure 3.2.1-2**

**Linear Heat Rate Axial Flux Offset Control Limits**

(AXIAL SHAPE INDEX limits for Linear Heat Rate when using Excore Detector Monitoring System)

(LCO Limits are not needed below 20% thermal power per SE00433)

(See NEOP-23 for Operational Limits)

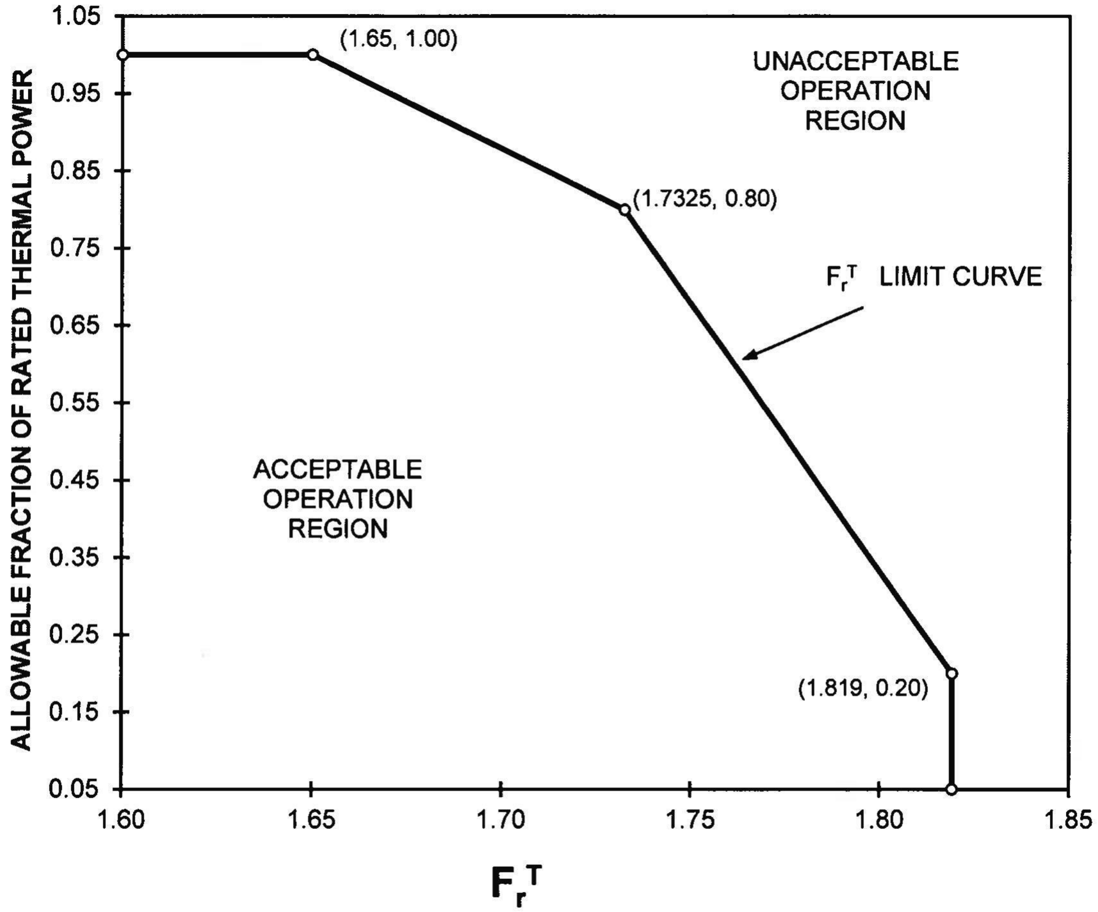



**Figure 3.2.3**

**Total Integrated Radial Peaking Factor ($F_r^T$) vs. Allowable Fraction of Rated Thermal Power**

**While operating with $F_r^T$ greater than 1.65, withdraw CEAs to or above the Long Term Steady State Insertion Limits (Figure 3.1.6)**

This figure cannot be changed without prior NRC approval.

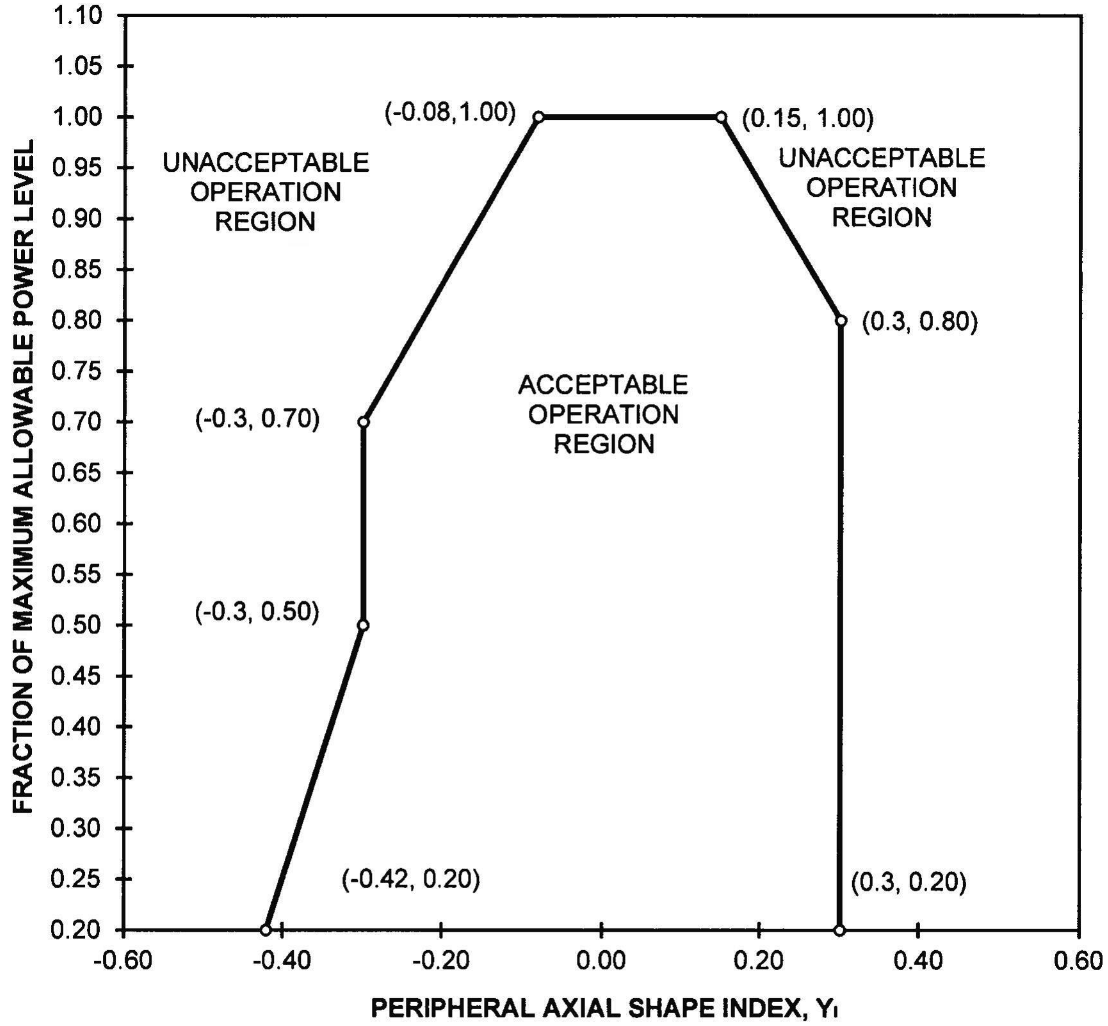



**Figure 3.2.5**

**DNB Axial Flux Offset Control Limits**

(LCO Limits are not needed below 20% thermal power per SE00433)

(See NEOP-23 for Operational Limits)

This figure cannot be changed without prior NRC approval.

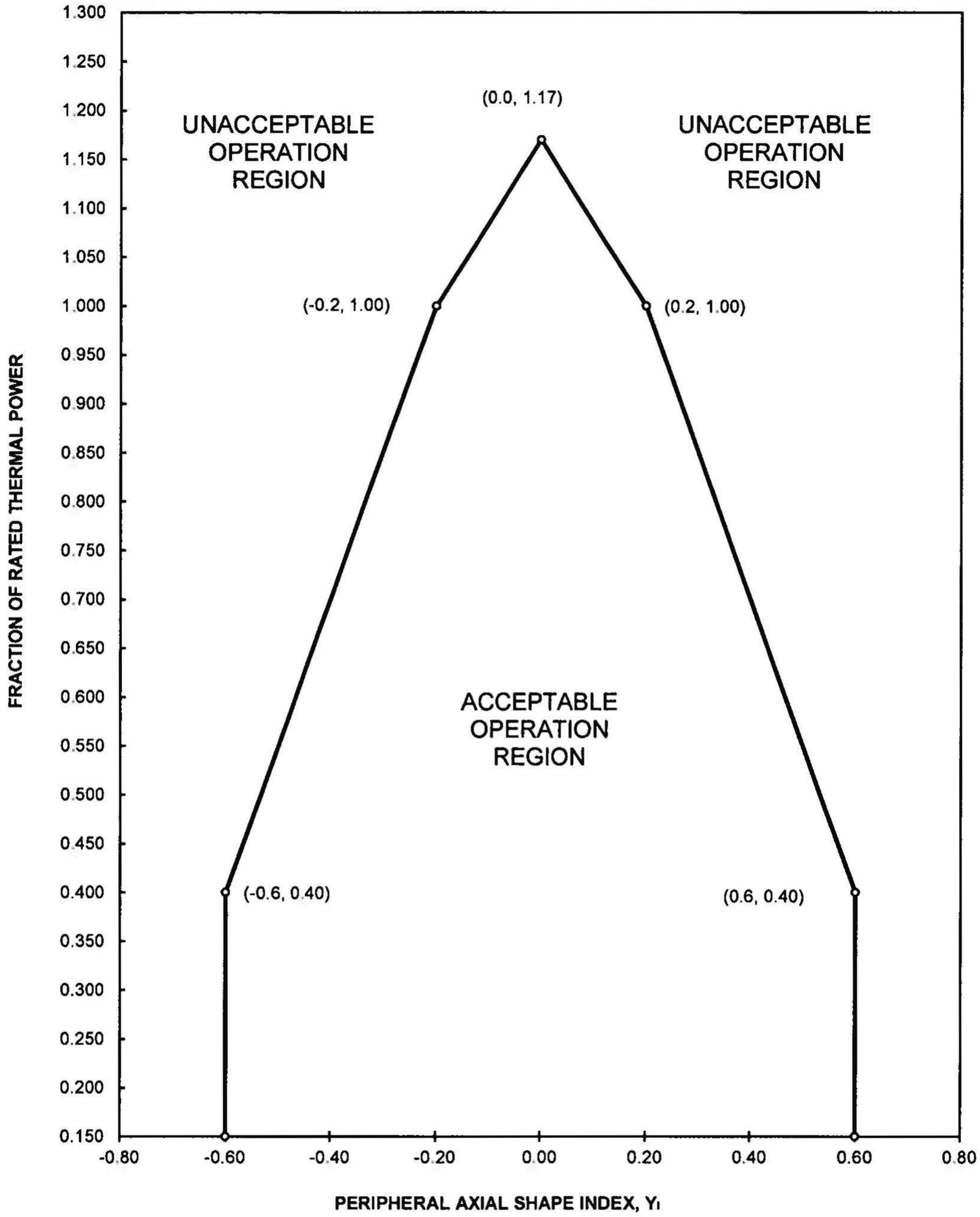



**Figure 3.3.1-1**

**Axial Power Distribution - High Trip Setpoint**
**Peripheral Axial Shape Index vs. Fraction of Rated Thermal Power**

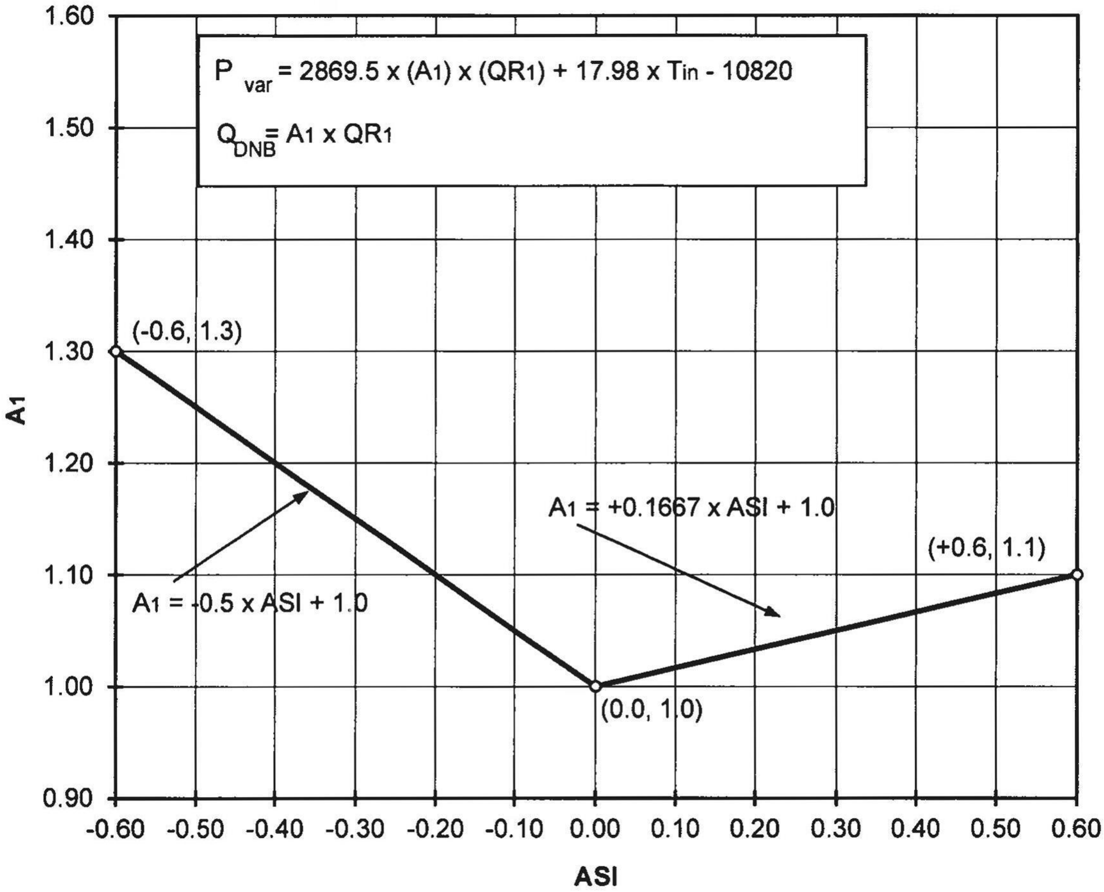



**Figure 3.3.1-2**

**Thermal Margin/Low Pressure Trip Setpoint - Part 1**
**(ASI vs. $A_1$)**

$$P_{var}^{Trip} = 2869.5 \times (A1) \times (QR1) + 17.98 \times T_{in} - 10820$$

$$Q_{DNB} = A1 \times QR1$$

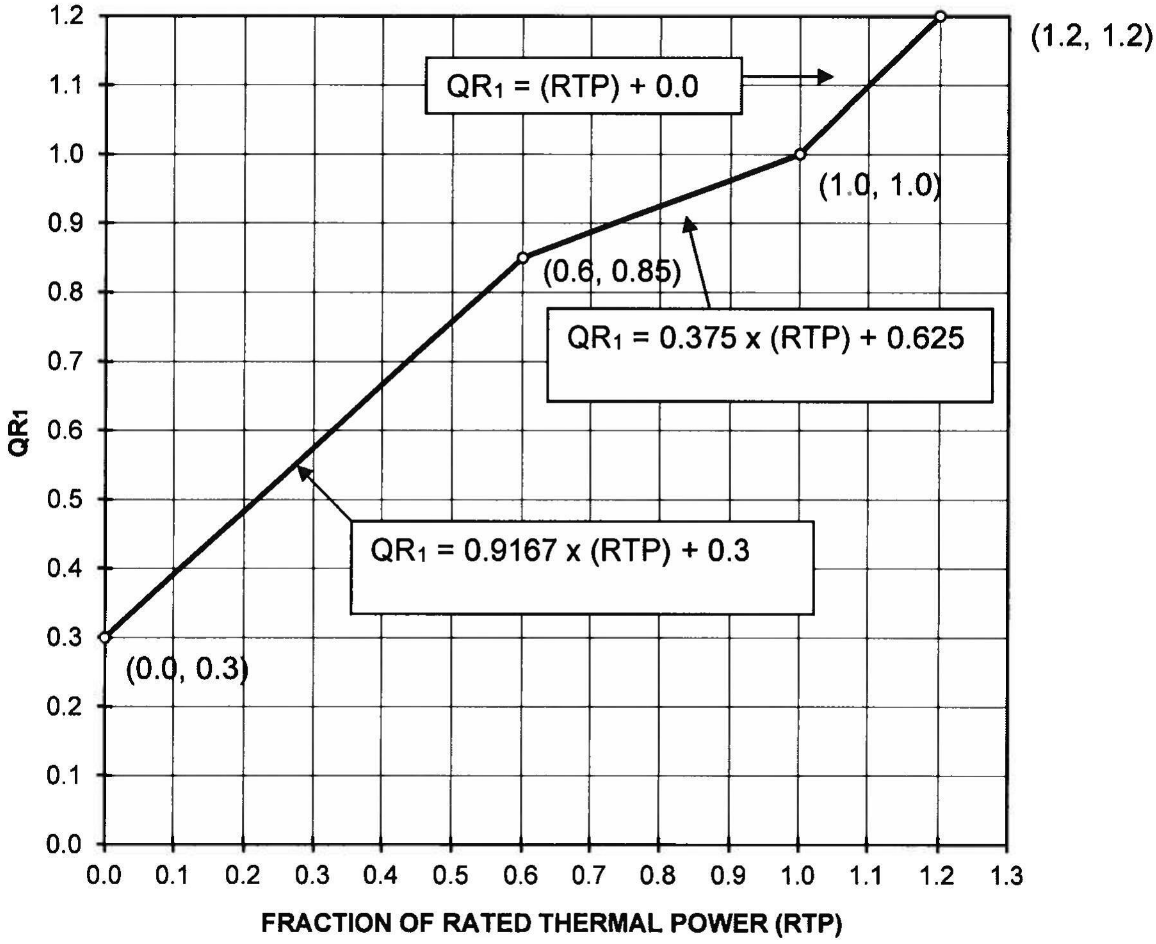



**Figure 3.3.1-3**

**Thermal Margin/Low Pressure Trip Setpoint - Part 2**
**(Fraction of Rated Thermal Power vs. $QR_1$)**

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## LIST OF APPROVED METHODOLOGIES

1. 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, December 1991

2. BAW- 10240(P)(A), Revision 0, "Incorporation of M5 Properties in Framatome ANP Approved Methods" Framatome ANP, May 2004

3. EMF-92-116(P)(A), Revision 0, Supplement 1(P)(A), Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs" AREVA Inc. , February 2015 [*Licensing Restriction 7*]

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

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

6. EMF-1961 (P)(A), Revision 0, "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation, July 2000

7. EMF-2103 (P)(A), Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors" Framatome ANP, April 2003 [*Licensing Restriction 9*]

8. EMF-2310(P)(A), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors" Framatome ANP, May 2004 [*Licensing Restrictions 1, 2, 6, and 8b*]

9. EMF-2328(P)(A), Revision 0, Supplement 1 (P)(A), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based" AREVA, December 2016 [*Licensing Restriction 4*]

10. XN-NF-75-32(P)(A), Supplements 1, 2, 3 & 4, "Computational Procedure for Evaluating Fuel Rod Bowing" Exxon Nuclear Company Inc., February 1983

11. XN-NF-78-44(NP)(A), “A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors” Exxon Nuclear Company Inc., October 1983 [*Licensing Restrictions 3 and 8a*]

12. XN-NF-79-56(P)(A), Revision 1 and Supplement 1, "Gadolinia Fuel Properties for LWR Fuel Safety Evaluation" Siemens Power Corporation, October 1981

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

14. XN-NF-82-21(P)(A), Revision 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations" Exxon Nuclear Company Inc., August 1983

15. XN-NF-85-92(P)(A), Revision 0, "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results" Exxon Nuclear Company Inc., September 1986

16. CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 2: Combination of System Parameter Uncertainties in Thermal Margin Analyses for Calvert Cliffs Units 1 and 2," January 1980 [*Not used for this fuel cycle*]

17. CEN-191(B)-P, "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2," December 1981 [*Not used for this fuel cycle*]

18. Letter from Mr. D. H. Jaffe (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated June 24, 1982, Unit 1 Cycle 6 License Approval (Amendment No. 71 to DPR-53 and SER) [Approval to CEN-124(B)-P (three parts) and CEN-191(B)-P)] [*Not used for this fuel cycle*]

19. CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986 [*Not used for this fuel cycle*]

20. CENPD-206-P-A, "TORC Code, Verification and Simplified Modeling Methods," June 1981 [*Not used for this fuel cycle*]

21. CENPD-225-P-A, "Fuel and Poison Rod Bowing," June 1983 [*Not used for this fuel cycle*]

22. CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993 [*Not used for this fuel cycle*]

23. CENPD-139-P-A, "C-E Fuel Evaluation Model Topical Report," July 1974 [*Not used for this fuel cycle*]

24. CEN-161-(B)-P-A, "Improvements to Fuel Evaluation Model," August 1989 [*Not used for this fuel cycle*]

25. CEN-161-(B)-P, Supplement 1-P, "Improvements to Fuel Evaluation Model," April 1986 [*Not used for this fuel cycle*]

26. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated February 4, 1987, Docket Nos. 50-317 and 50-318, "Safety Evaluation of Topical Report CEN-161-(B)-P, Supplement 1-P, Improvements to Fuel Evaluation Model" (Approval of CEN-161(B), Supplement 1-P) [*Not used for this fuel cycle*]

27. CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," May 1990 [*Not used for this fuel cycle*]

28. CENPD-135, Supplement 5-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977 [*Not used for this fuel cycle*]

29. CENPD-387-P-A, Latest Approved Revision, "ABB Critical Heat Flux Correlations for PWR Fuel" [*Not used for this fuel cycle*]

30. CENPD-404-P-A, Latest Approved Revision, "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs". [*Not used for this fuel cycle*]

31. WCAP-11596-P-A, "Qualification of the PHOENIX-P, ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988. [*Not used for this fuel cycle*]

32. WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," September 1986. [*Not used for this fuel cycle*]

33. WCAP-10965-P-A Addendum 1, "ANC: A Westinghouse Advanced Nodal Computer Code; Enhancements to ANC Rod Power Recovery," April 1989. [*Not used for this fuel cycle*]

34. WCAP-16072-P-A, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs," August 2004. [*Not used for this fuel cycle*]

35. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004. [*Not used for this fuel cycle*]