FPL

Florida Power & Light Company, 6501 South Ocean Drive, Jensen Beach, FL 34957

March 28, 2001

L-2001-074
10 CFR 50.36

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

Re: St. Lucie Unit 1
Docket No. 50-335
Cycle 16 Core Operating Limits Report – Revision 2

Pursuant to St. Lucie Unit 1 Technical Specification (TS) 6.9.1.11.d, Florida Power & Light Company (FPL) is submitting the Core Operating Limits Report (COLR) Revision 2 for operating cycle 16.

Technical Specification 6.9.1.11.d requires that the COLR, including any mid-cycle revisions or supplements, be provided to the NRC upon issuance for each reload cycle. This COLR revision implements License Amendment 171 for the remainder of cycle 16. License Amendment 171 relocated the shutdown margin limits from TS to the COLR. Accordingly, enclosed is a copy of the *St. Lucie Unit 1, Cycle 16 Core Operating Limits Report*, Revision 2.

Please contact us if there are any questions about this submittal.

Very truly yours,

Rajiv S. Kundalkar

Rajiv S. Kundalkar
Vice President
St. Lucie Plant

RSK/GRM

Enclosure

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant

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ST. LUCIE UNIT 1, CYCLE 16

CORE OPERATING LIMITS REPORT

Revision 2

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1.0 INTRODUCTION

This CORE OPERATING LIMITS REPORT (COLR) describes the cycle-specific parameter limits for operation of St. Lucie Unit 1 Cycle 16. It contains the limits for the following as provided in Section 2.

Moderator Temperature Coefficient

Full Length CEA Position - Misalignment > 15 Inches

Regulating CEA Insertion Limits

Linear Heat Rate

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

DNB Parameter - AXIAL SHAPE INDEX

Refueling Operations - Boron Concentration

SHUTDOWN MARGIN – T_{avg} Greater Than 200°F

SHUTDOWN MARGIN – T_{avg} Less Than or Equal To 200°F

This report also contains the necessary figures which give the limits for the above listed parameters.

Terms appearing in capitalized type are DEFINED TERMS as defined in Section 1.0 of the Technical Specifications.

This report is prepared in accordance with the requirements of Technical Specification 6.9.1.11.

2.0 CORE OPERATING LIMITS

2.1 Moderator Temperature Coefficient (TS 3.1.1.4)

The moderator temperature coefficient (MTC) shall be less negative than -32 pcm/^{0}F at RATED THERMAL POWER.

2.2 Full Length CEA Position - Misalignment > 15 Inches (TS 3.1.3.1)

The time constraints for full power operation with the misalignment of one full length CEA by 15 or more inches from any other CEA in its group are shown in Figure 3.1-1a.

2.3 Regulating CEA Insertion Limits (TS 3.1.3.6)

The regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits shown on Figure 3.1-2, with CEA insertion between the Long Term Steady State Insertion Limits and the Power Dependent Insertion Limits restricted to:

a. ≤ 4 hours per 24 hour interval,

b. ≤ 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. ≤ 14 Effective Full Power Days per calendar year.

2.4 Linear Heat Rate (TS 3.2.1)

The linear heat rate shall not exceed the limits shown on Figure 3.2-1.

The AXIAL SHAPE INDEX power dependent control limits are shown on Figure 3.2-2.

During operation, with the linear heat rate being monitored by the Excore Detector Monitoring System, the AXIAL SHAPE INDEX shall be maintained within the limits of Figure 3.2-2.

During operation, with the linear heat rate being monitored by the Incore Detector Monitoring System, the Local Power Density alarm setpoints shall be adjusted to less than or equal to the limits shown on Figure 3.2-1.

2.5 TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T (TS 3.2.3)

The calculated value of F_r^T shall be limited to $\leq$ 1.70.

The power dependent F_r^T limits are shown on Figure 3.2-3.

2.6 DNB Parameters - AXIAL SHAPE INDEX (TS 3.2.5)

The AXIAL SHAPE INDEX shall be maintained within the limits specified in Figure 3.2-4.

2.7 Refueling Operations - Boron Concentration (TS 3.9.1)

With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling cavity shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1000 pcm conservative allowance for uncertainties, or

b. A boron concentration of $\geq$ 1720 ppm, which includes a 50 ppm conservative allowance for uncertainties.

2.8 SHUTDOWN MARGIN – T_{avg} Greater Than 200^{0}F (TS 3.1.1.1)

The SHUTDOWN MARGIN shall be greater than or equal to 3600 pcm.

2.9 SHUTDOWN MARGIN – T_{avg} Less Than or Equal To 200^{0}F (TS 3.1.1.2)

The SHUTDOWN MARGIN shall be greater than or equal to 2000 pcm.

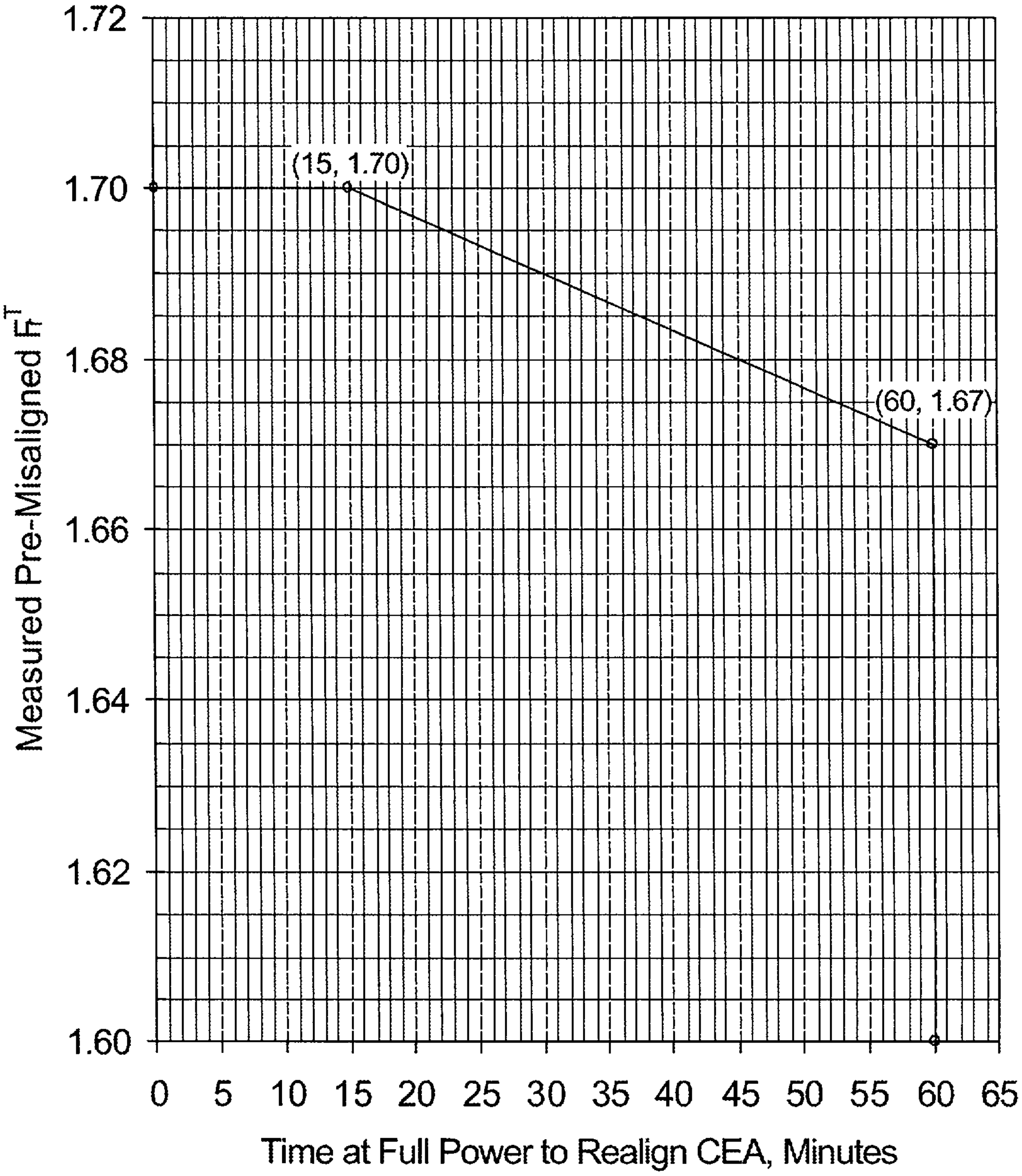


FIGURE 3.1-1a
Allowable Time to Realign CEA vs. Initial F_r^T

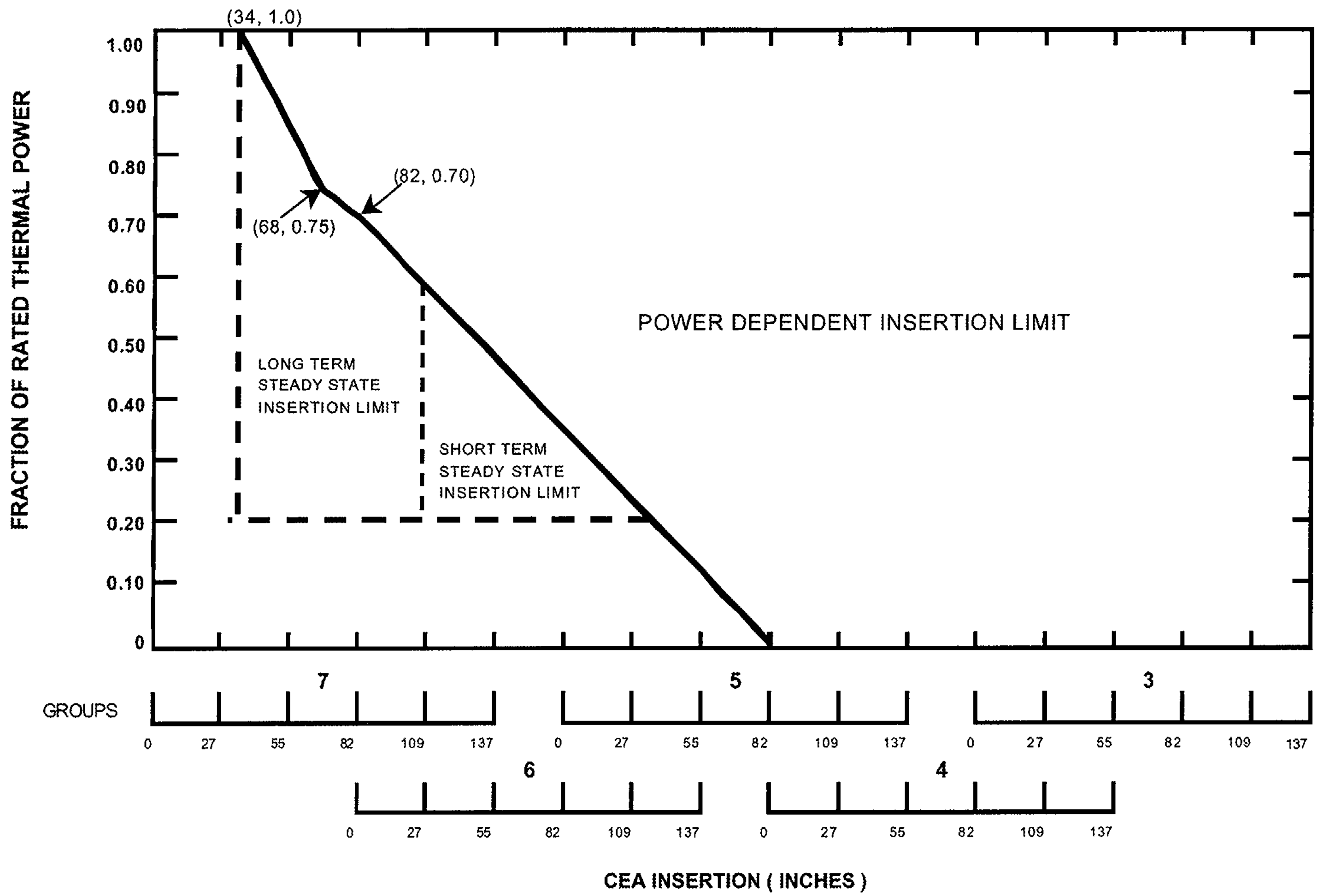
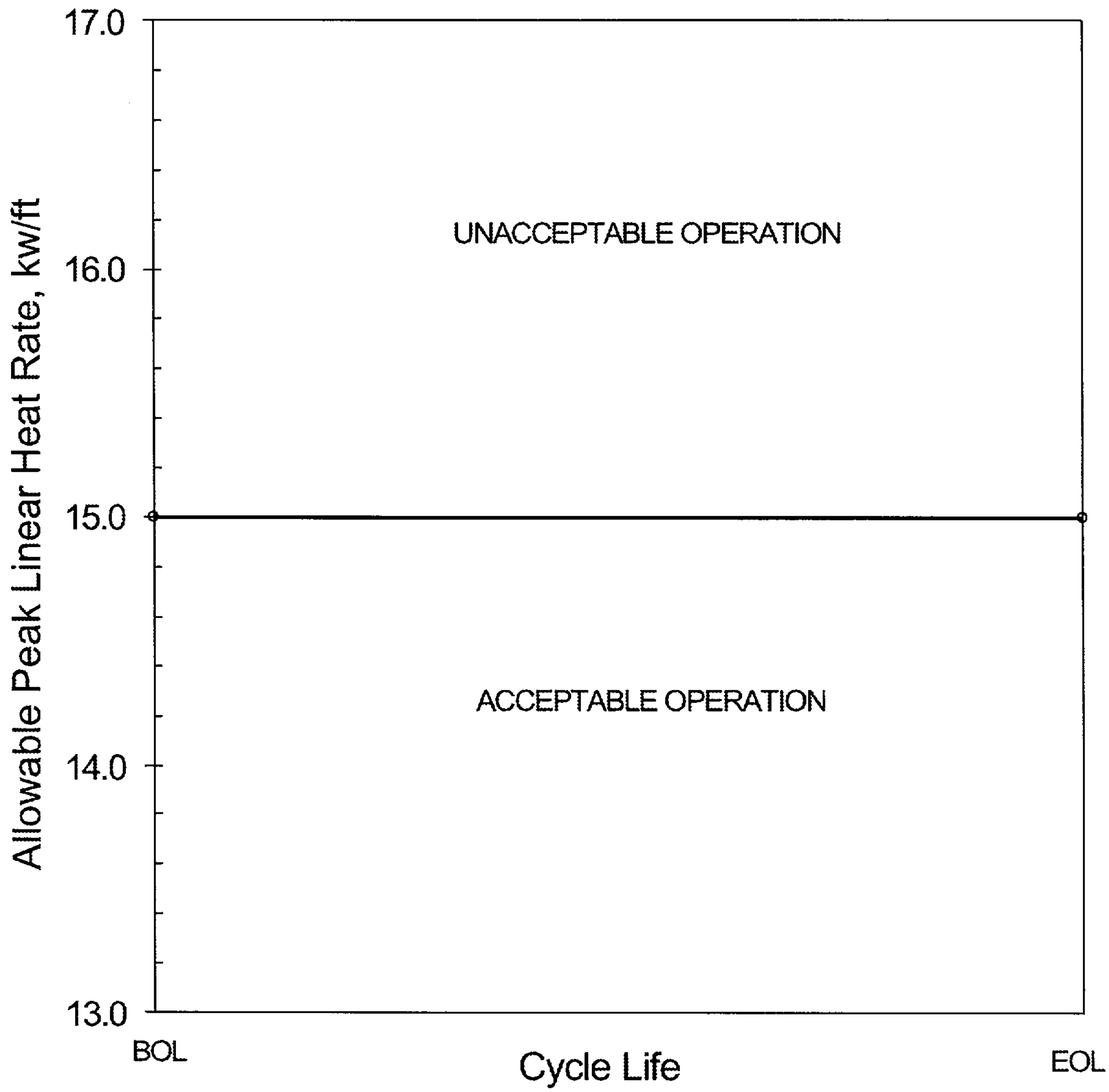
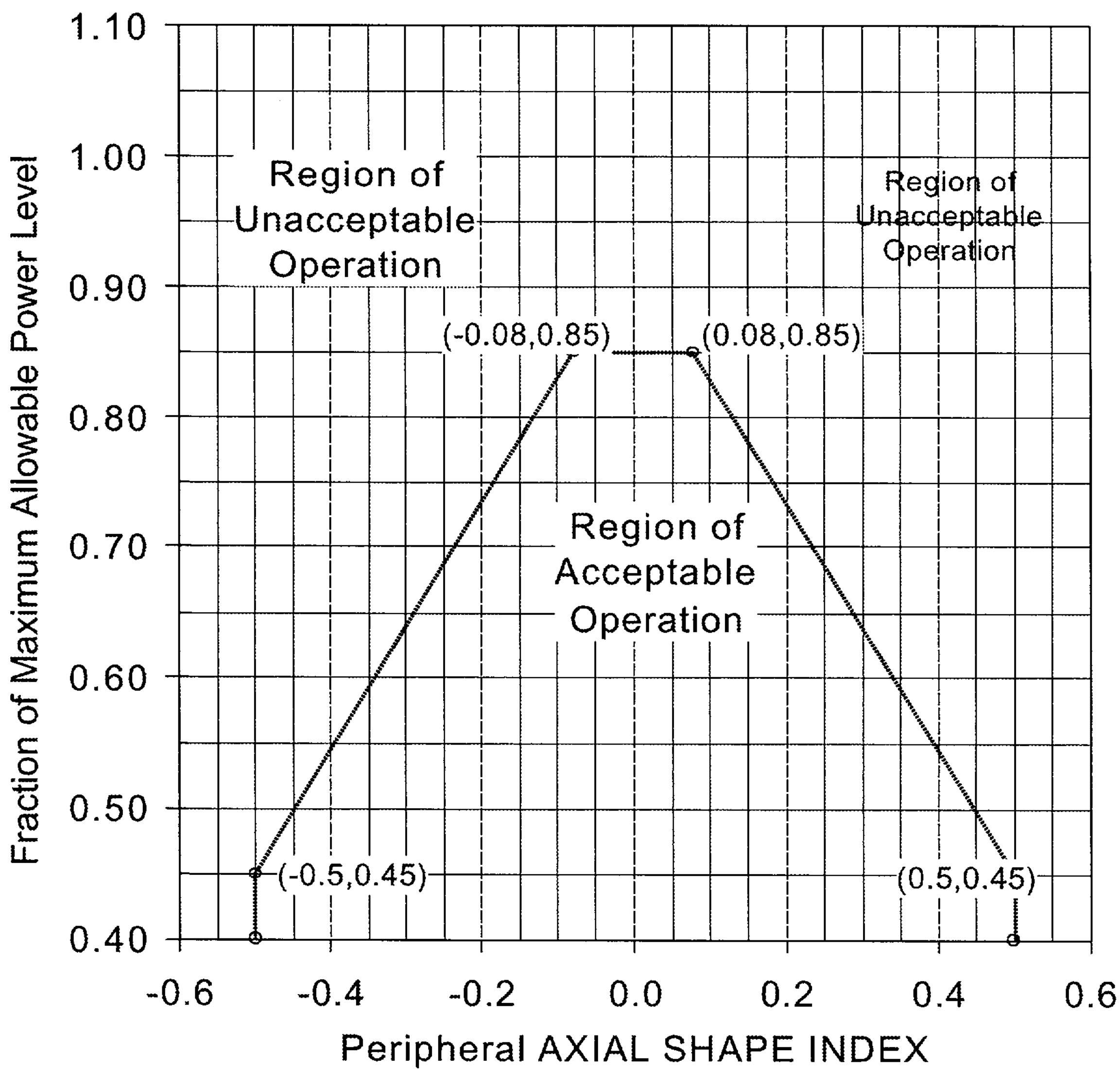


FIGURE 3.1-2
CEA Insertion Limits vs. THERMAL POWER
(4 Reactor Coolant Pumps Operating)



(Fuel + Clad + Moderator)

FIGURE 3.2-1
Allowable Peak Linear Heat Rate vs. Burnup



(Not Applicable Below 40% Power)

FIGURE 3.2-2
AXIAL SHAPE INDEX vs. Maximum Allowable Power Level

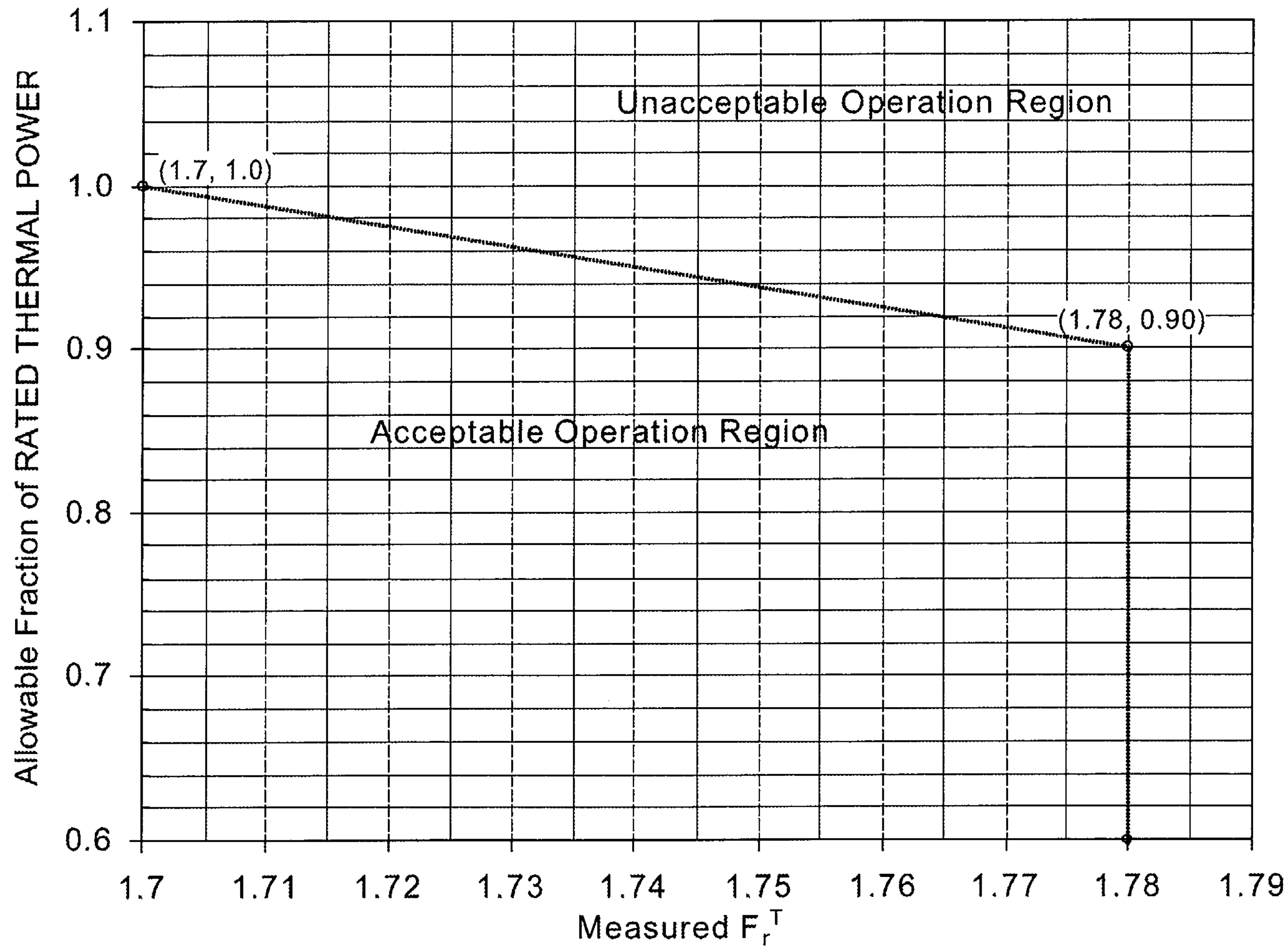
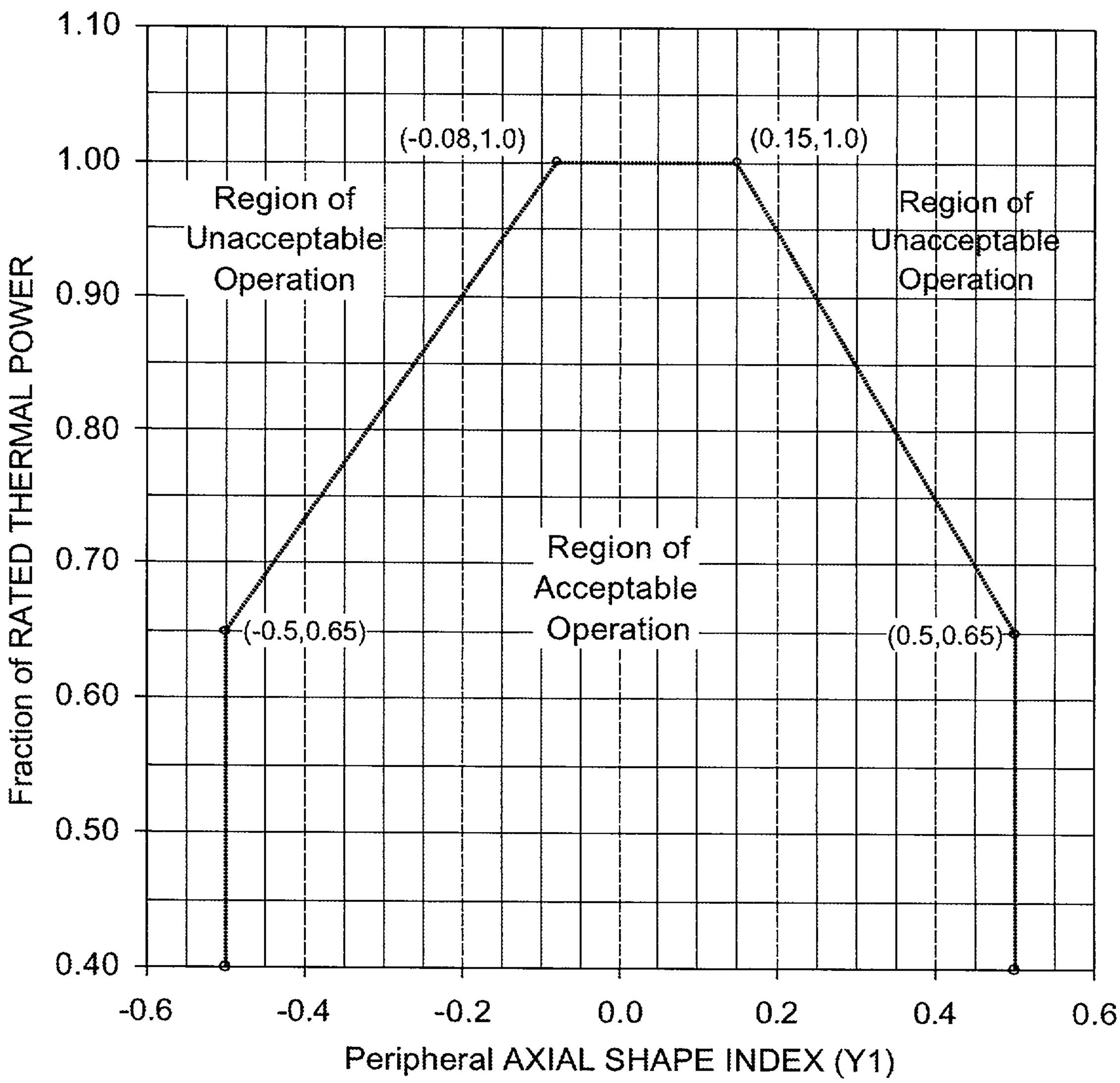


FIGURE 3.2-3
Allowable Combinations of THERMAL POWER and F_r^T



(Not Applicable Below 40% Power)

FIGURE 3.2-4
AXIAL SHAPE INDEX Operating Limits vs. THERMAL POWER
(Four Reactor Coolant Pumps Operating)

3.0 LIST OF APPROVED METHODS

The analytical methods used to determine the core operating limits are those previously approved by the NRC, and are listed below.

1. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988 (Westinghouse Proprietary)

2. NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants," Florida Power & Light Company, January 1995

3. XN-75-27(A) and Supplements 1 through 5, [also issued as XN-NF-75-27(A)], "Exxon Nuclear Neutronic(s) Design Methods for Pressurized Water Reactors," Exxon Nuclear Company, Inc. / Advanced Nuclear Fuels Corporation, Report and Supplement 1 dated April 1977, Supplement 2 dated December 1980, Supplement 3 dated September 1981 (P), Supplement 4 dated December 1986 (P), and Supplement 5 dated February 1987 (P)

4. ANF-84-73(P)(A) Revision 5, Appendix B, & Supplements 1 and 2, "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuels Corporation, October 1990

5. 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., September 1983

6. a) ANF-84-93(P)(A) and Supplement 1, [also issued as XN-NF-84-93(P)(A)], "Steamline Break Methodology for PWRs," Advanced Nuclear Fuels Corporation, March 1989

   b) EMF-84-093(P)(A) Revision 1, "Steam Line Break Methodology for PWRs," Siemens Power Corporation, February 1999 (This document is a Revision to ANF-84-93)

7. XN-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company, Inc., October 1983

8. Siemens Small Break LOCA methodology as defined by:

   a) XN-NF-82-49(P)(A) Revision 1, "Exxon Nuclear Company Evaluation Model EXEM PWR Small Break Model," Advanced Nuclear Fuels Corporation, April 1989

   b) XN-NF-82-49(P)(A) Revision 1 Supplement 1, "Exxon Nuclear Company Evaluation Model Revised EXEM PWR Small Break Model," Siemens Power Corporation, December 1994

9. 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

10. XN-NF-621(P)(A) Revision 1, "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company, Inc., September 1983

11. EXEM PWR Large Break LOCA Evaluation Model as defined by:

    a) 1. XN-NF-82-20(P)(A) Revision 1 Supplement 2, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Exxon Nuclear Company, Inc., February 1985

       2. XN-NF-82-20(P)(A) Revision 1 and Supplements 1, 3 and 4, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Advanced Nuclear Fuels Corporation, January 1990

       3. XN-NF-82-20(P)(A) Revision 1 Supplement 6, "EXEM/PWR Large Break LOCA ECCS Model Updates," Siemens Power Corporation, June 1998

    b) XN-NF-82-07(P)(A) Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, Inc., November 1982

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

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

    d) XN-NF-85-16(P)(A) Volume 1, and Supplements 1, 2 and 3; Volume 2, Revision 1 and Supplement 1, "PWR 17x17 Fuel Cooling Test Program," Advanced Nuclear Fuels Corporation, February 1990

    e) XN-NF-85-105(P)(A) and Supplement 1, "Scaling of FCTF Based Reflood Heat Transfer Correlation for Other Bundle Designs," Advanced Nuclear Fuels Corporation, January 1990

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

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

13. 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

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

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

16. XN-NF-507(P)(A), Supplements 1 and 2, "ENC Setpoint Methodology for C. E. Reactors: Statistical Setpoint Methodology," Exxon Nuclear Company, Inc., September 1986

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

18. EMF-92-153(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation, March 1994

19. 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

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