FPL

April 29, 2024
L-2024-071
10 CFR 50.36

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-00001

Re: St. Lucie Nuclear Plant, Unit 2
Docket 50-389

Cycle 27 Core Operating Limits Report

Pursuant to St. Lucie Unit 2 Technical Specification (TS) 6.9.1.11.d, Florida Power & Light Company (FPL) is submitting Revision 1 of the Core Operating Limits Report (COLR) for operating cycle 27. Revision 1 reflects changes of the conversion from Current Technical Specifications (CTS) to Improved Technical Specifications (ITS).

Should you have any questions regarding this submission, please contact Mr. Kenneth Mack, Fleet Licensing Manager, at 561-904-3635.

Sincerely,

Paul Rasmus
General Manager, Regulatory Affairs
Florida Power & Light Company

Enclosure St. Lucie Unit 2, Cycle 27 Core Operating Limits Report, Revision 1

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, St. Lucie Nuclear Plant
USNRC Resident Inspector, St. Lucie Nuclear Plant

ST. LUCIE UNIT 2, CYCLE 27

CORE OPERATING LIMITS REPORT

Revision 1

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Approved by:	R. Norring	4/19/24 Date

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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 2. It contains the limits for the following as provided in Section 2.

Moderator Temperature Coefficient (MTC),

Control Element Assembly (CEA) Alignment,

Regulating CEA Insertion Limits,

Linear Heat Rate (LHR) and Axial Shape Index (ASI),

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

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

Boron Concentration,

SHUTDOWN MARGIN (SDM) – T_{avg} Greater Than 200 ^{0}F,

SHUTDOWN MARGIN (SDM) – T_{avg} Less Than or Equal To 200 ^{0}F,

Reactor Core Safety Limits (SLs).

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 5.6.3.

2.0 CORE OPERATING LIMITS

2.1 Moderator Temperature Coefficient (TS 3.1.3)

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

The maximum positive limit shall be:

a. +7 pcm/°F with THERMAL POWER ≤ 70% RTP, and
b. +2 pcm/°F with THERMAL POWER > 70% RTP.

2.2 Control Element Assembly (CEA) Alignment (TS 3.1.4)

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

2.3 Regulating CEA Insertion Limits (TS 3.1.6 and 3.1.7)

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 Days, and

c. ≤ 14 Effective Full Power Days per 365 EFPD.

2.4 Linear Heat Rate (TS 3.2.1) and Axial Shape Index (TS 3.2.4)

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.

Excore Detector Monitoring System
During operation, with the linear heat rate (LHR) being monitored by the Excore Detector Monitoring System, the AXIAL SHAPE INDEX required by TS 3.2.1 shall be maintained within the limits of Figure 3.2-2.

Incore Detector Monitoring System
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.

The AXIAL SHAPE INDEX required by TS 3.2.4 shall be maintained within the limits shown on Figure 3.2-4.

The instrumentation AXIAL SHAPE INDEX (Y_I) used for the trip and pretrip signals in the reactor protection system is the AXIAL SHAPE INDEX value (Y_E) modified by an appropriate multiplier (A) and a constant (B) to determine the true core axial power distribution for that channel ($Y_I = AY_E + B$). Where Y_E is the power level detected by the lower excore nuclear instrument detectors (L) less the power level detected by the upper excore nuclear instrument detectors (U), divided by the sum of these power levels, [$Y_E = (L-U) / (L+U)$].

2.5 <u>TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T</u> (TS 3.2.2)

The calculated value of F_r^T shall be limited to ≤ 1.65.

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

2.6 <u>RCS Pressure, Temperature and Flow Departure from Nucleate Boiling (DNB) Limits</u> (TS 3.4.1)

The following DNB-related parameters shall be maintained within the limits shown on Table 3.2-2:

a. Cold Leg Temperature
b. Pressurizer Pressure

2.7 <u>Boron Concentration</u> (TS 3.9.1)

With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the boron concentration of all filled portions of the Reactor Coolant System, the refueling canal, and the refueling canal 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, or

b. A boron concentration of greater than or equal to 1900 ppm.

2.8 <u>SHUTDOWN MARGIN – T_{avg} Greater Than 200 ^{0}F</u> (TS 3.1.1)

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

2.9 <u>SHUTDOWN MARGIN – T_{avg} Less Than or Equal To 200 ^{0}F</u> (TS 3.1.1)

The SHUTDOWN MARGIN shall be greater than or equal to 3000 pcm.

2.10 Reactor Core SLs (TS 2.1)

The fuel melt limit is defined as [(2790 - 17.9 x P - 3.2 x B) x 1.8 + 32] °F, where P is the maximum weight percent of Gadolinia (%) and B is the maximum pin burnup (GWD/MTU).

Table 3.2-2

DNB MARGIN LIMITS

PARAMETER	FOUR REACTOR COOLANT PUMPS OPERATING
Cold Leg Temperature (narrow Range)	535^0F** $\leq$ T $\leq$ 551^0F
Pressurizer Pressure*	2225 psia $\leq$ P_{PZR} $\leq$ 2350 psia**

* Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% per minute of RATED THERMAL POWER or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

** Applicable only if power level ≥ 70% of RATED THERMAL POWER.

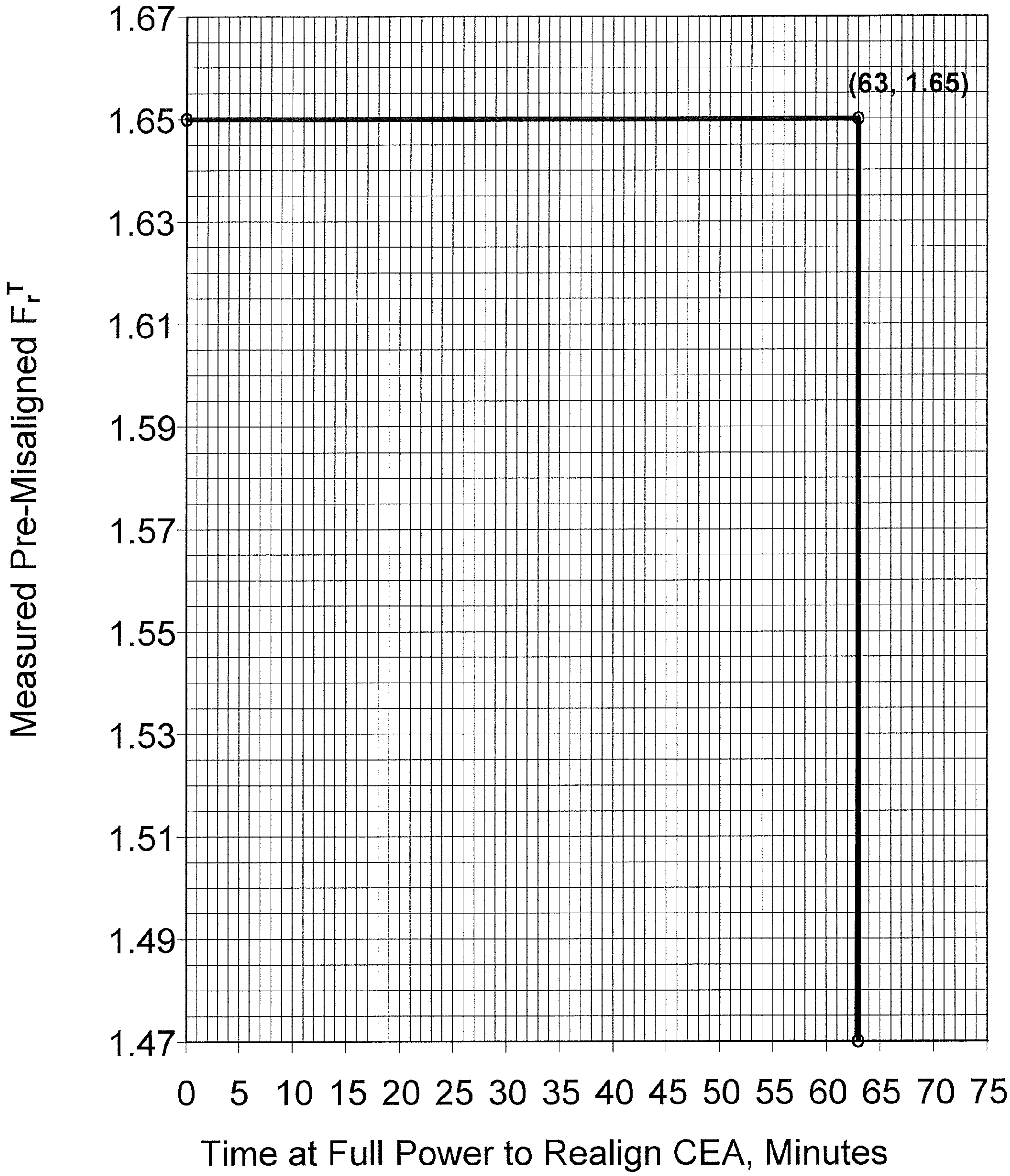


FIGURE 3.1-1a
Allowable Time to Realign CEA vs. Initial F_r^T

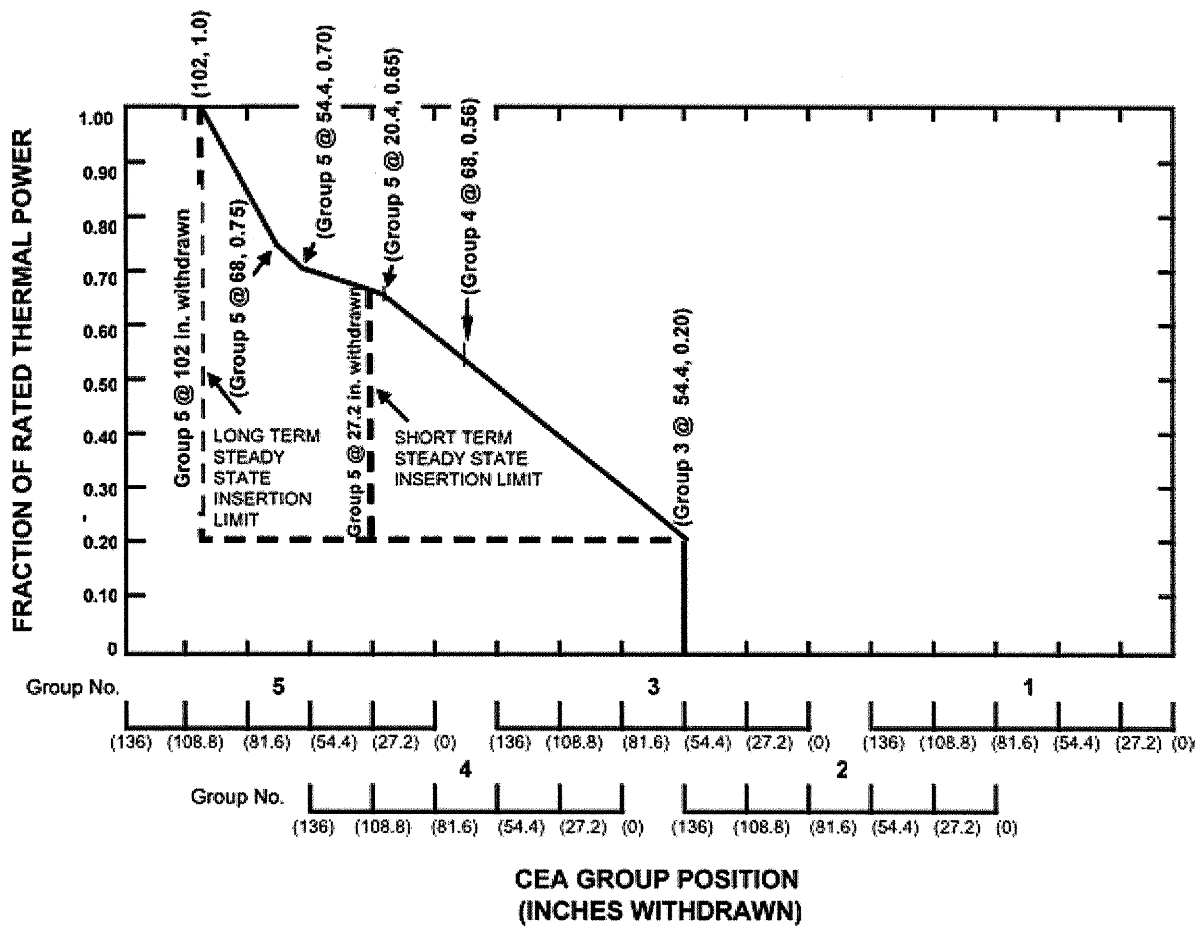
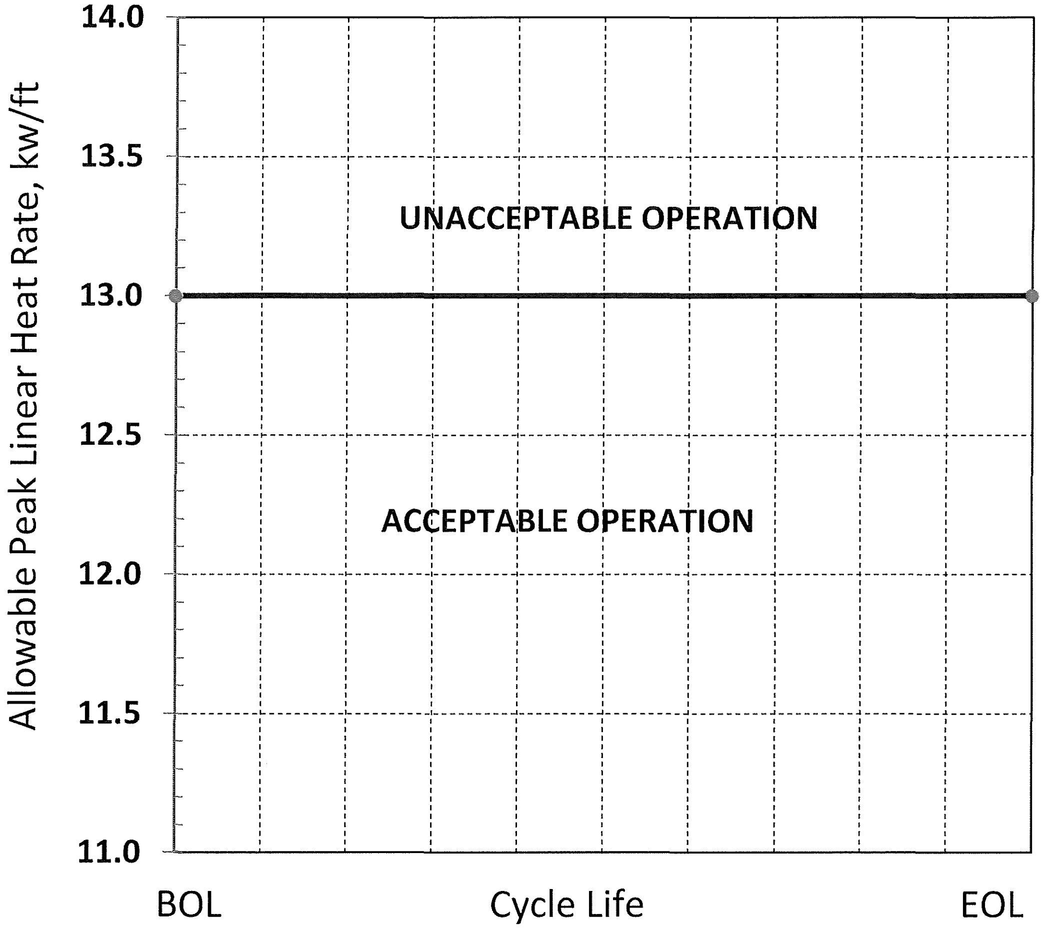
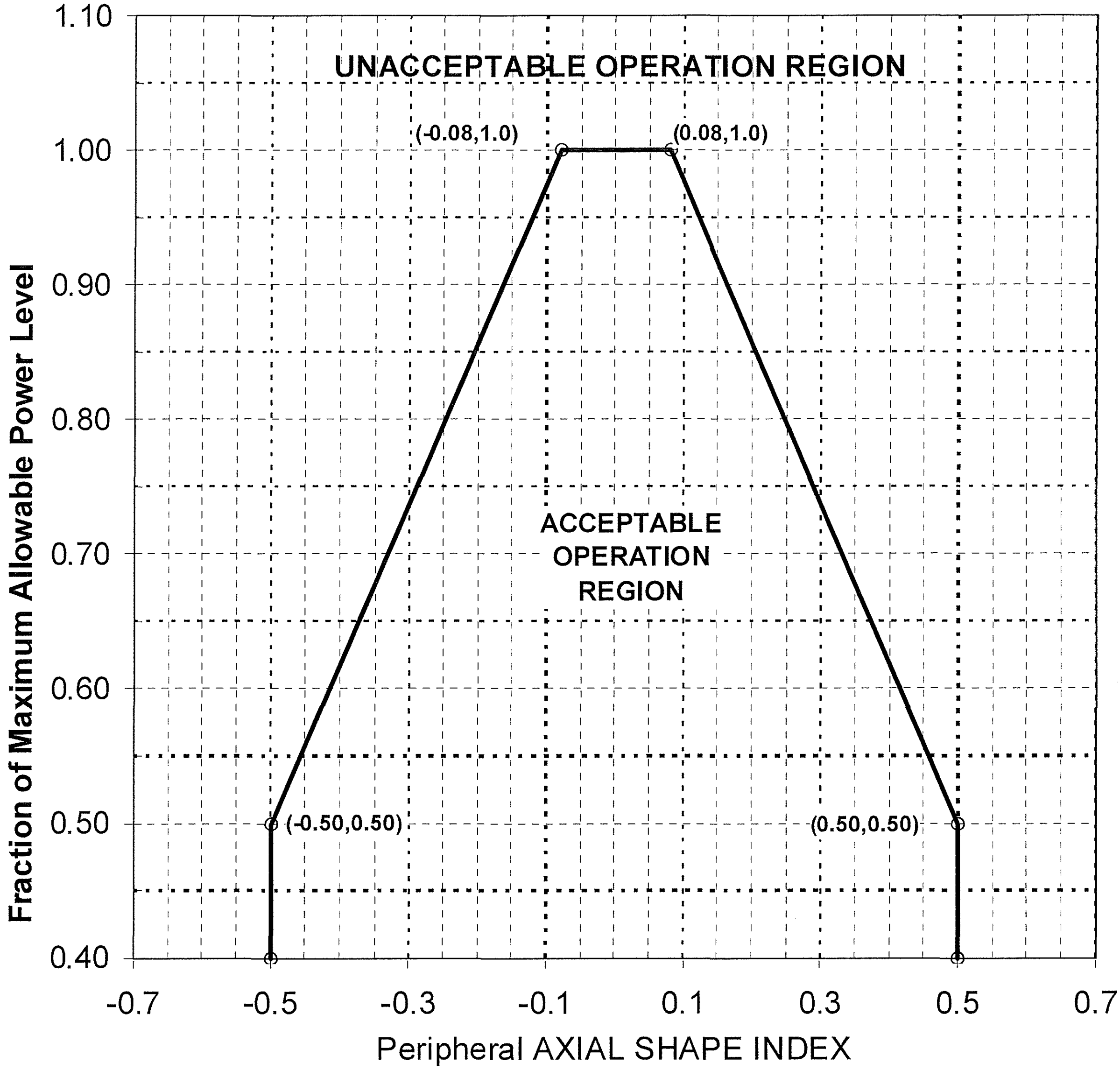


FIGURE 3.1-2
CEA Group Insertion Limits vs. THERMAL POWER



(Fuel + Clad + Moderator)

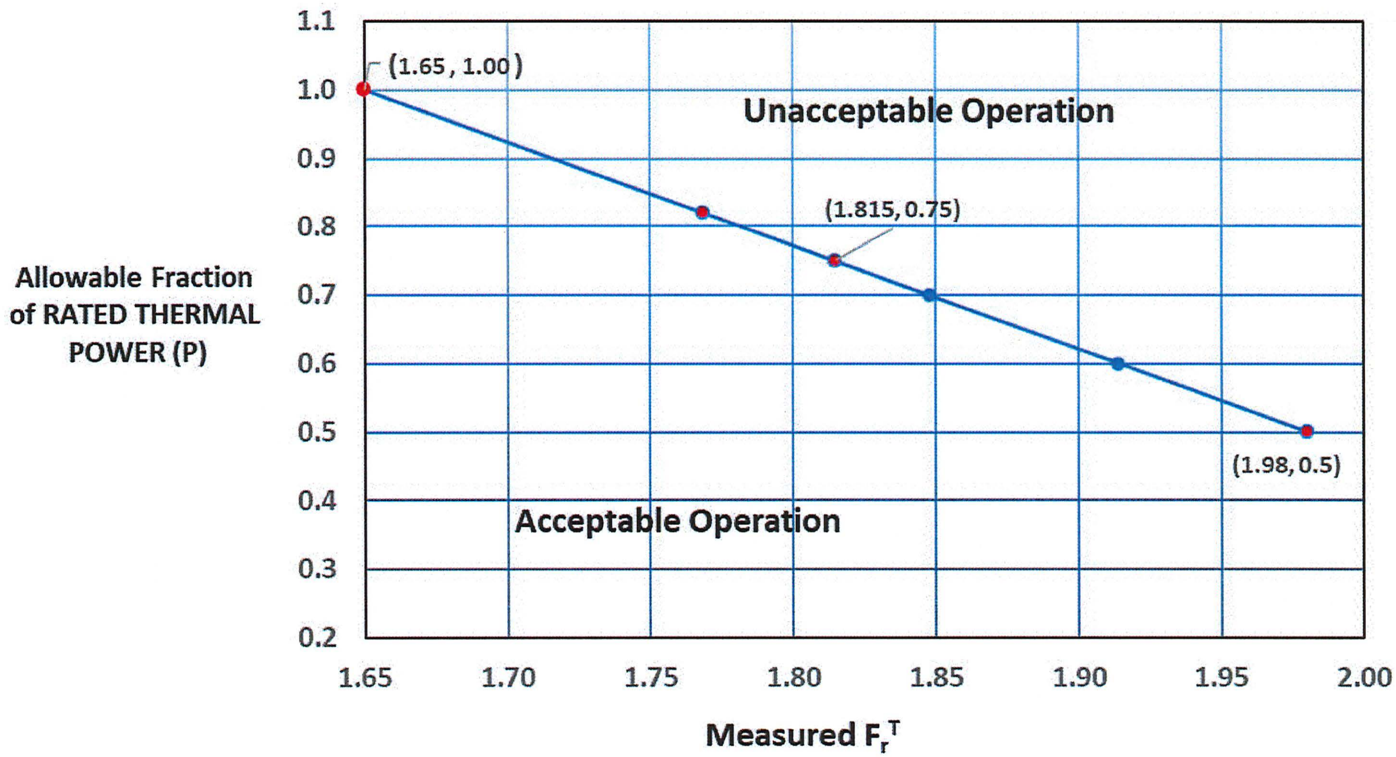
FIGURE 3.2-1
Allowable Peak Linear Heat Rate vs. Burnup



(Not Applicable Below 40% Power)

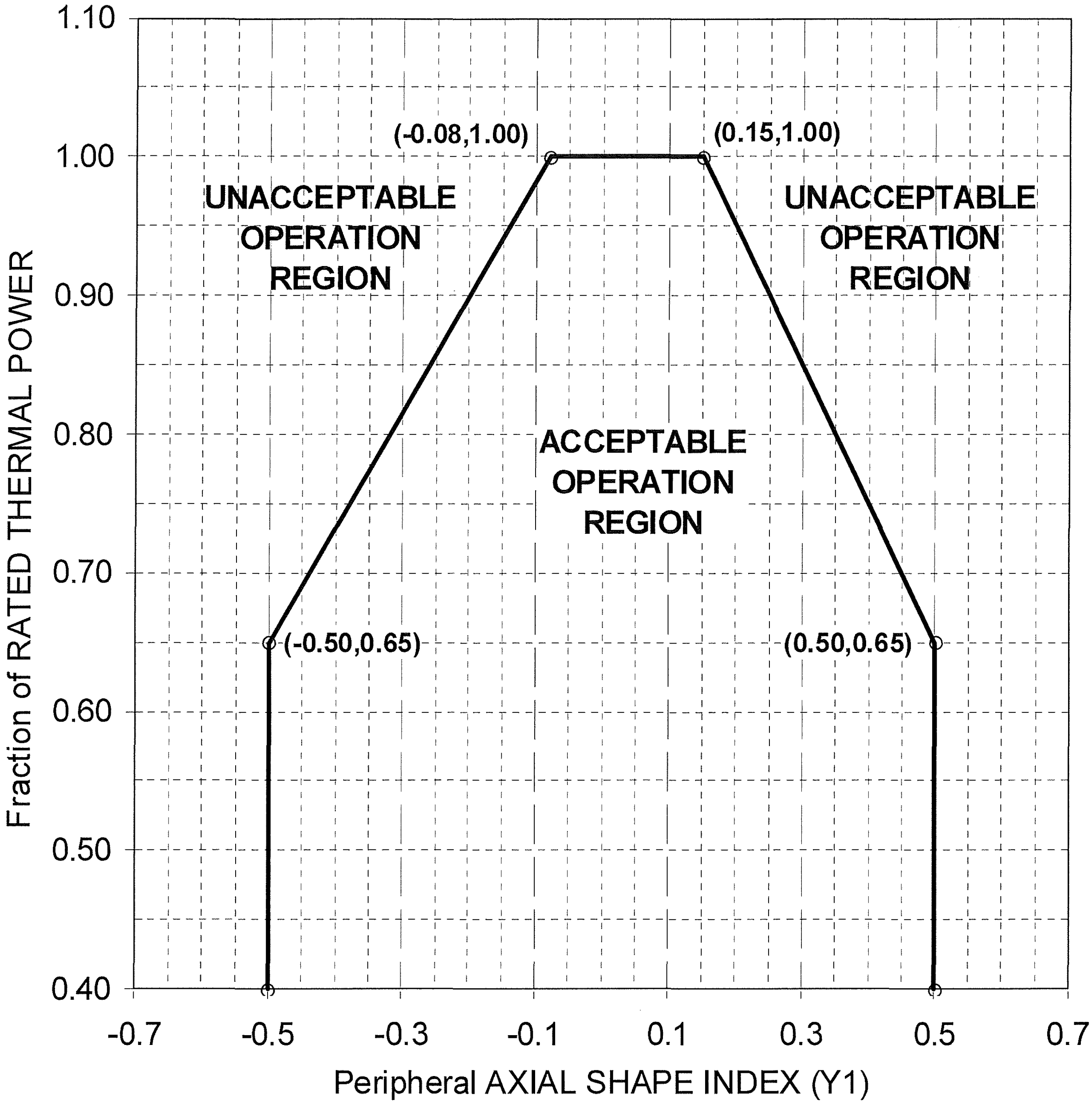
FIGURE 3.2-2
AXIAL SHAPE INDEX vs. Maximum Allowable Power Level

Note: AXIAL SHAPE INDEX limits for Linear Heat Rate when using Excore Detector Monitoring System



$F_r^T = 1.65 \times [1+0.4 \times (1 - P)]$

FIGURE 3.2-3
Allowable Combinations of THERMAL POWER and F_r^T
(The expression specified in the Figure may be used for F_r^T at all power levels)



(Not Applicable Below 40% Power)

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

(AXIAL SHAPE INDEX limits for DNB)

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 (NRC SER dated June 9, 1995), & Supplement 1, August 1997

3. Deleted.

4. XN-NF-79-56(P)(A) Revision 1 and Revision 1 Supplement 1, “Gadolinia Fuel Properties for LWR Fuel Safety Evaluation”, October 1981.

5. CENPD-275-P, Revision 1-P-A, "C-E Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorbers," May 1988, & Revision 1-P Supplement 1-P-A, April 1999

6. Deleted.

7. Deleted.

8. CEN-123(F)-P, "Statistical Combination of Uncertainties Methodology Part 1: CE Calculated Local Power Density and Thermal Margin/Low Pressure LSSS for St. Lucie Unit 1," December 1979

9. Deleted.

10. CEN-123(F)-P, "Statistical Combination of Uncertainties Methodology Part 3: CE Calculated Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for St. Lucie Unit 1," February 1980

11. CEN-191(B)-P, "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2," December 1981

12. Letter, J. W. Miller (NRC) to J. R. Williams, Jr. (FPL), Docket No. 50-389, Regarding Unit 2 Cycle 2 License Approval (Amendment No. 8 to NPF-16 and SER), November 9, 1984 (Approval of CEN-123(F)-P (three parts) and CEN-191(B)-P)

13. Deleted.

14. Letter, J. A. Norris (NRC) to J. H. Goldberg (FPL), Docket No. 50-389, "St. Lucie Unit 2 - Change to Technical Specification Bases Sections '2.1.1 Reactor Core' and '3/4.2.5 DNB Parameters' (TAC No. M87722)," March 14, 1994 (Approval of CEN-371(F)-P)

15. Deleted.

16. Deleted.

17. Deleted.

18. Deleted.

19. CENPD-225-P-A, "Fuel and Poison Rod Bowing," June 1983

20. CENPD-139-P-A, "C-E Fuel Evaluation Model Topical Report," July 1974

21. CEN-161(B)-P-A, "Improvements to Fuel Evaluation Model," August 1989

22. CEN-161(B)-P, Supplement 1-P-A, "Improvements to Fuel Evaluation Model," January 1992

23. CENPD-132, Supplement 3-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS," June 1985

24. CENPD-133, Supplement 5-A, "CEFLASH-4A, A FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis," June 1985

25. CENPD-134, Supplement 2-A, "COMPERC-II, a Program for Emergency Refill-Reflood of the Core," June 1985

26. CENPD-135-P, Supplement 5, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977

27. Letter, R. L. Baer (NRC) to A. E. Scherer (CE), "Evaluation of Topical Report CENPD-135, Supplement #5," September 6, 1978

28. CENPD-137, Supplement 1-P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," January 1977

29. CENPD-133, Supplement 3-P, "CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident," January 1977

30. Letter, K. Kniel (NRC) to A. E. Scherer (CE), "Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P," September 27, 1977

31. CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," January 1977

32. Letter, C. Aniel (NRC) to A. E. Scherer (CE), "Evaluation of Topical Report CENPD-138, Supplement 2-P," April 10, 1978

33. Letter, W. H. Bohlke (FPL) to Document Control Desk (NRC), "St. Lucie Unit 2, Docket No. 50-389, Proposed License Amendment, MTC Change from -27 pcm to -30 pcm," L-91-325, December 17, 1991

34. Letter, J. A. Norris (NRC) to J. H. Goldberg (FPL), "St. Lucie Unit 2 - Issuance of Amendment Re: Moderator Temperature Coefficient (TAC No. M82517)," July 15, 1992

35. Letter, J. W. Williams, Jr. (FPL) to D. G. Eisenhut (NRC), "St. Lucie Unit No. 2, Docket No. 50-389, Proposed License Amendment, Cycle 2 Reload," L-84-148, June 4, 1984

36. Letter, J. R. Miller (NRC) to J. W. Williams, Jr. (FPL), Docket No. 50-389, Regarding Unit 2 Cycle 2 License Approval (Amendment No. 8 to NPF-16 and SER), November 9, 1984 (Approval of Methodology contained in L-84-148)

37. Deleted.

38. Deleted.

39. Deleted.

40. Deleted.

41. Deleted.

42. CEN-348(B)-P-A, Supplement 1-P-A, "Extended Statistical Combination of Uncertainties," January 1997

43. CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," May 1990

44. Deleted.

45. Deleted.

46. Deleted.

47. Deleted.

48. CEN-396(L)-P, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/KG for St. Lucie Unit 2," November 1989 (NRC SER dated October 18, 1991, Letter J. A. Norris (NRC) to J. H. Goldberg (FPL), TAC No. 75947)

49. CENPD-269-P, Rev. 1-P, "Extended Burnup Operation of Combustion Engineering PWR Fuel," July 1984

50. CEN-289(A)-P, "Revised Rod Bow Penalties for Arkansas Nuclear One Unit 2," December 1984 (NRC SER dated December 21, 1999, Letter K. N. Jabbour (NRC) to T. F. Plunkett (FPL), TAC No. MA4523)

51. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998

52. CENPD-140-A, "Description of the CONTRANS Digital Computer Code for

Containment Pressure and Temperature Transient Analysis," June 1976

53. Deleted.

54. Deleted.

55. CENPD-387-P-A, Revision 000, "ABB Critical Heat Flux Correlations for PWR Fuel," May 2000

56. CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001

57. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998

58. WCAP-12610-P-A & CENPD-404-P-A, Addendum 2-A, "Westinghouse Clad Corrosion Model for ZIRLO and Optimized ZIRLO," October 2013.

59. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology'" July 1985

60. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control; FQ Surveillance Technical Specification," February 1994

61. WCAP-11397-P-A, (Proprietary), "Revised Thermal Design Procedure," April 1989

62. WCAP-14565-P-A, (Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999

63. WCAP-14565-P-A, Addendum 1-A, Revision 0, "Addendum 1 to WCAP-14565-P-A Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code," August 2004

64. Letter, W. Jefferson, Jr. (FPL) to Document Control Desk (USNRC), "St. Lucie Unit 2 Docket No. 50-389: Proposed License Amendment WCAP-9272 Reload Methodology and Implementing 30% Steam Generator Tube Plugging Limit," L-2003-276, December, 2003 (NRC SER dated January 31, 2005, Letter B. T. Moroney (NRC) to J. A. Stall (FPL), TAC No. MC1566)

65. WCAP-14882-P-A, Rev. 0, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.

66. WCAP-7908-A, Rev. 0, "FACTRAN-A FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod," December 1989.

67. WCAP-7979-P-A, Rev. 0, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," January 1975.

68. WCAP-7588, Rev. 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," January 1975.

69. WCAP-16045-P-A, Revision 0, “Qualification of the Two-Dimensional Transport Code PARAGON,” August 2004.

70. 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.

71. 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.

72. XN-75-27(A) and Supplements 1 through 5, "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," Exxon Nuclear Company, 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).

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

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

75. 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.

76. EMF-92-116(P)(A), Rev. 0, and Supplement 1(P)(A), Rev. 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," February 1999 and February 2015.

77. BAW-10240(P)(A), Rev. 0, "Incorporation of M5™ Properties in Framatome ANP Approved Methods," Framatome ANP, Inc., May 2004.

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

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

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

81. EMF-2310(P)(A), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Framatome ANP, Inc., May 2004.

82. XN-75-32(P)(A), Supplements 1, 2, 3, and 4, "Computational Procedure for Evaluating Fuel Rod Bowing," October 1983.

83. BAW-10231P-A Revision 1, "COPERNIC Fuel Rod Design Computer Code," January 2004.

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

85. EMF-2328 (P)(A) Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2001.