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Technical Specification 5.6.5

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

Core Operating Limits Report – Revision 16

Dear Sir or Madam:

Entergy Nuclear Operations, Inc. is providing revision 16 of the Palisades Nuclear Plant (PNP) Core Operating Limits Report (COLR). This report is submitted in accordance with the requirements of PNP Technical Specification 5.6.5.d. Enclosure 1 contains a summary of changes from the previous revision. Enclosure 2 contains revision 16 of the COLR.

Summary of Commitments

This letter contains no new commitments and no revision to existing commitments.

Laurie Lahti

Laurie A. Lahti
Licensing Manager
Palisades Nuclear Plant

Enclosures (2)

CC Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades, USNRC

**ENCLOSURE 1**
**PALISADES CORE OPERATING LIMITS REPORT**
**REVISION 16**

**SUMMARY OF CHANGES**

Documentation for all analytical methods used to determine the core operating limits currently listed in Technical Specification Section 5.6.5 is added to COLR Section 3.0, with documentation revision numbers and dates.

**ENCLOSURE 2**

**PALISADES CORE OPERATING LIMITS REPORT**
**REVISION 16**

10 Pages Follow

Procedure No COLR
Revision 16
Issued Date 5/1/2008

## PALISADES NUCLEAR PLANT

# TITLE: CORE OPERATING LIMITS REPORT

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**Entergy Nuclear Palisades, LLC**
**Entergy Nuclear Operations, Inc.**
**Docket No 50-255**
**License No DPR-20**

# Core Operating Limits Report

## 1.0 INTRODUCTION

This Core Operating Limits Report for Palisades has been prepared in accordance with the requirements of Technical Specification 5.6.5. The Technical Specifications Limiting Conditions for Operation (LCOs) affected by this report are listed below:

| Section | Title | LCO |
|---|---|---|
| 2.1 | SHUTDOWN MARGIN (SDM) | 3.1.1 |
| | | 3.1.6 |
| | | 3.9.1 |
| 2.2 | Regulating Rod Group Position Limits | 3.1.6 |
| 2.3 | Linear Heat Rate (LHR) | 3.2.1 |
| 2.4 | Total Radial Peaking Factor | 3.2.2 |
| 2.5 | AXIAL SHAPE INDEX (ASI) | 3.2.4 |
| 2.6 | PCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits | 3.4.1 |

## 2.0 OPERATING LIMITS

The cycle specific parameter limits for the specifications listed in Section 1 are presented in the following subsections. These limits have been developed using the NRC-approved methodologies specified in Section 3.0.

2.1 SHUTDOWN MARGIN (SDM)

2.1.1 MODES 1 and 2 (LCO 3.1.6 Regulating Rod Group Position Limits) - The minimum SDM requirement is 2% with the most reactive rod fully withdrawn. The rod insertion limit (PDIL) is discussed in Section 2.2 and shown in Figure 2.2-1.

2.1.2 MODES 3, 4 and 5, Loops Filled (LCO 3.1.1 SHUTDOWN MARGIN) - The SDM requirement is ≥ 2% for normal cooldowns and heatups.

2.1.3 MODE 5, Loops Not Filled (LCO 3.1.1 SHUTDOWN MARGIN) - The SDM requirement is ≥ 3.5% assuming $T_{ave}$ of 60°F.

2.1.4 MODE 6 (LCO 3.9.1 Boron Concentration) - The SDM requirement is specified in the definition of REFUELING BORON CONCENTRATION.

2.2 Regulating Rod Group Position Limits

a. If the reactor is critical, to implement the limits on SHUTDOWN MARGIN, individual rod worth and hot channel factors, the limits on control rod regulating group insertion shall be established as shown on Figure 2.2-1.

b. If the reactor is subcritical, the rod position at which criticality could be achieved if the control rods were withdrawn in normal sequence shall not be lower than Group 2 at 72 inches (ie, ~ 45% control rod insertion).

c. The sequence of withdrawal of the regulating groups shall be 1, 2, 3, 4.

d. An overlap of control banks in excess of 40% shall not be permitted.

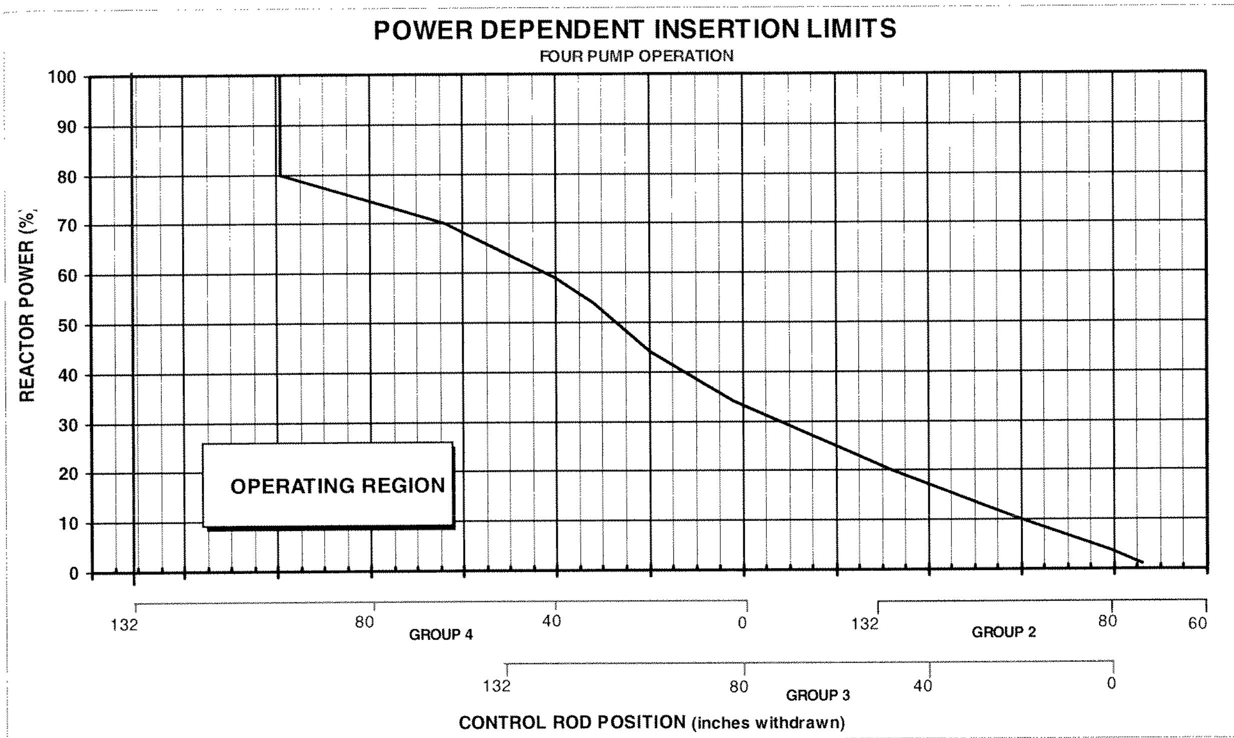


**Figure 2.2-1 Regulating Rod Group Position Limits**

**<u>NOTE</u>:** A regulating rod is considered fully withdrawn at ≥128 inches.

2.3 Linear Heat Rate (LHR)

The LHR in the peak powered fuel rod shall not exceed the following:

$$LHR \leq LHR_{TS} \times F_A(z)$$

Where:

$LHR_{TS}$ = Maximum allowable LHR shown in Table 2.3-1.

$F_A(z)$ = Allowable LHR as a function of peak power location shown in Figure 2.3-1.

**Table 2.3-1 - Linear Heat Rate Limit**

| | |
|---|---|
| Peak Rod | 15.28 (kW/ft) |

To ensure that the design margin of safety is maintained, the determination of both the incore alarm setpoints and the Allowable Power Level takes into account the local LHR measurement uncertainty factors given in Table 2.4-2, an engineering uncertainty factor and a thermal power measurement uncertainty factor (values given in Technical Specification Basis B 3.2.1).

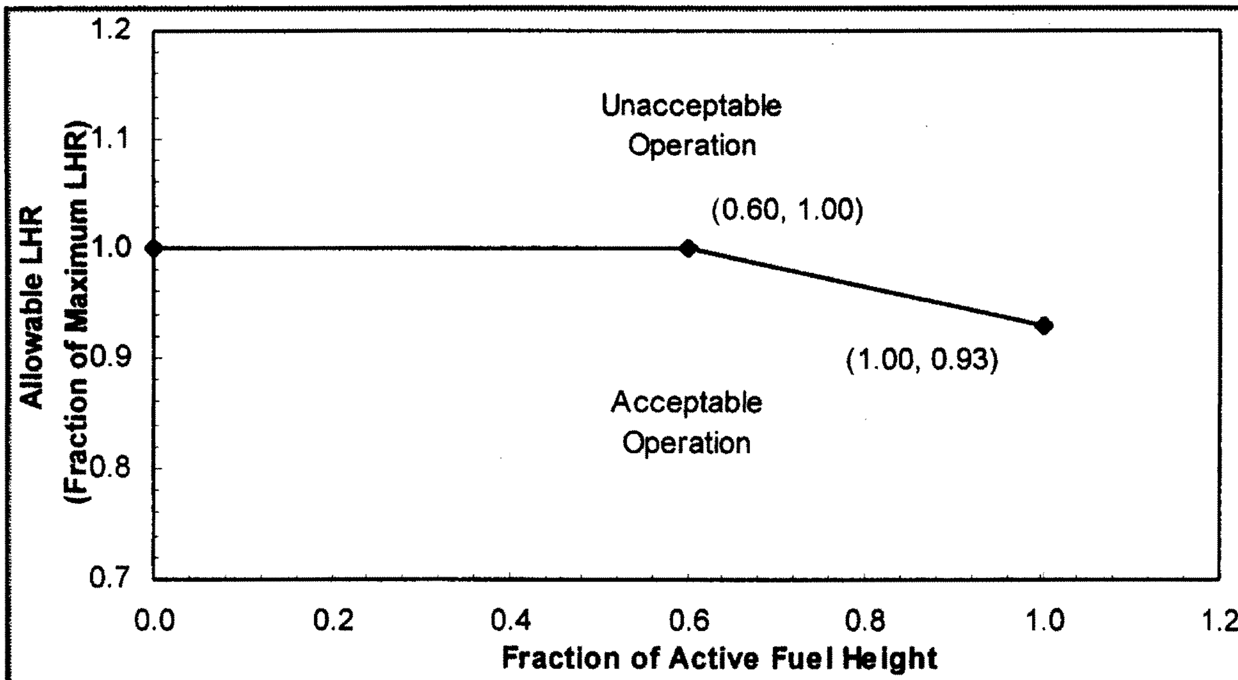


**Figure 2.3-1 - Allowable LHR as a Function of Peak Power Location**

2.4 Total Radial Peaking Factor

The radial peaking factor shall not exceed the following:

for $P \geq 0.5$

$$F_r \leq F_r^{TS} \times [1.0 + 0.3 \times (1 - P)]$$

and for $P < 0.5$,

$$F_r \leq F_r^{TS} \times 1.15$$

Where:

$F_r$ = Measured $F_r^T$,

$F_r^{TS}$ = Peaking Factor Limits (Table 2.4-1),

$P$ = Fraction of rated power.

**Table 2.4-1 - Peaking Factor Limits, $F_r^{TS}$**

| All Fuel Types |
|---|
| 2.04 |

To ensure that the design margin of safety is maintained, the determination of radial peaking factors takes into account the appropriate measurement uncertainty factors given in Table 2.4-2.

To ensure that the design margin of safety is maintained with respect to the alternative source term radiological consequence analysis assumptions as restricted by footnote 11 of Table 3 of Regulatory Guide 1.183, an evaluation shall be performed on the pin power/burnup of the core design against the following criteria:

- Fewer than 21 rods in any one assembly violate the "54/6.3" criterion.
- Fewer than 20 assemblies in any core design contain at least one rod that violates the "54/6.3" criterion.
- All rods that violate the "54/6.3" criterion have a rod average linear heat generation rate of less than 6.7 kW/ft.
- All rods that violate the "54/6.3" criterion have a rod burnup of less than 58.5 GWD/MTU.
- In any assembly containing any rods that violate the "54/6.3" criterion there are at least four times as many rods that have total radial peaking factor of less than ¾ of the total radial peaking factor limit of 2.04.

## TABLE 2.4-2 POWER DISTRIBUTION MEASUREMENT UNCERTAINTY FACTORS

| | LHR | $F_r^T$ |
|---|---|---|
| Measurement Uncertainty | 0.0500 | 0.0425 |

## 2.5 AXIAL SHAPE INDEX (ASI)

The ASI limit for the $T_{inlet}$ function is shown in Figure 2.5-1.

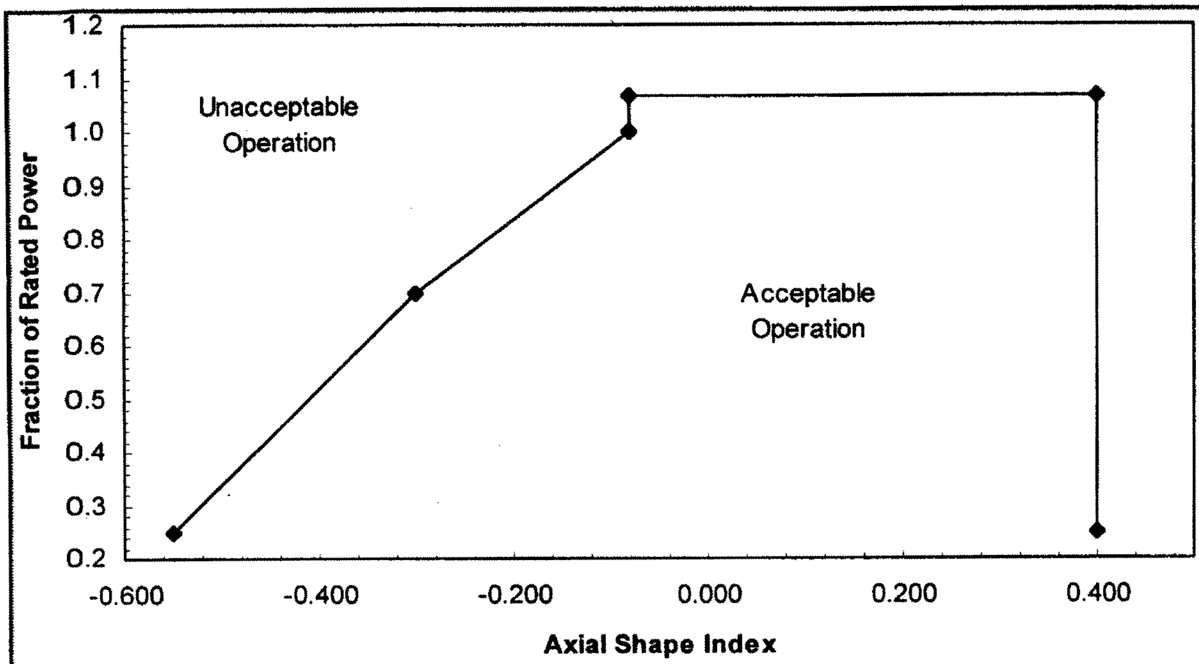


**Figure 2.5-1 - ASI Limit for Tinlet Function**

Break Points:

| | |
|---|---|
| -0.550, | 0.250 |
| -0.300, | 0.700 |
| -0.080, | 1.000 |
| -0.080, | 1.065 |
| +0.400, | 1.065 |
| +0.400, | 0.250 |

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

a. Pressurizer pressure $\geq$ 2010 psia and $\leq$ 2100 psia

b. PCS cold leg temperature $\leq$ 544 °F

c. PCS total flow rate $\geq$ 352,000 gpm

## 3.0 ANALYTICAL METHODS

The analytical methods used to determine the core operating limits are those previously reviewed and approved by the NRC, specifically those described in the Technical Specification Section 5.6.5 list of methodology documents. The Technical Specification 5.6.5 list is repeated below with revision numbers and dates added. Specific application of these methodologies to Palisades is described in the cycle's most current safety analysis reports.

The analytical methods used to determine the radial peaking factor measurement uncertainty factors are described in FSAR, Section 3.3.2.5.

1. EMF-96-029(P)(A) Volumes 1 and 2, "Reactor Analysis System for PWRs," Siemens Power Corporation, January 1997.
(LCOs 3.1.1, 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

2. ANF-84-73 Appendix B (P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events, Revision 5, July 1990," Advanced Nuclear Fuels Corporation. (Bases report not approved) (LCOs 3.1.1, 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

3. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, Revision 1, September 1983.
(LCOs 3.2.1, 3.2.2, & 3.2.4)

4. EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs," Revision 1, February 1999.
(LCOs 3.1.1, 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

5. XN-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company, October 1983. (Bases document not approved)
(LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

6. EMF-2310 (P)(A), Revision 1, Framatome ANP, Inc., May 2004, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors."
(LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

7. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company, October 1983. (LCOs 3.1.6, 3.2.1, & 3.2.2)

8. 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.
(LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

9. EMF-92-153(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation, Revision 1, January 2005. (LCOs 3.2.1, 3.2.2, & 3.2.4)

10. XN-NF-621(P)(A), "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company, Revision 1, September 1983. (LCOs 3.2.1, 3.2.2, & 3.2.4)

11. 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, October 1986. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

12. 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. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

13. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, November 1986. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

14. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation, Revision 0, February 1999. (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

15. EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," Siemens Power Corporation, June 1999. (LCOs 3.1.6, 3.2.1, & 3.2.2)

16. ANF-87-150 Volume 2, "Palisades Modified Reactor Protection System Report: Analysis of Chapter 15 Events," Advanced Nuclear Fuels Corporation, June, 1988. [Approved for use in the Palisades design during the NRC review of license Amendment 118, November 15, 1988] (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.4.1)

17. EMF-1961(P)(A), Revision 0, Siemens Power Corporation, July 2000, "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors." (LCOs 3.1.6, 3.2.1, 3.2.2, 3.2.4, & 3.4.1)

18. EMF-2328 (P)(A), Revision 0, Framatome ANP, Inc., March, 2001, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based." (LCOs 3.1.6, 3.2.1, & 3.2.2)

19. BAW-2489P, "Revised Fuel Assembly Growth Correlation for Palisades, Revision 0, March 2005." (LCOs 3.1.6, 3.2.1, 3.2.2, & 3.2.4)

20. EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors, Revision 0, April 2003." (LCOs 3.1.6, 3.2.1, & 3.2.2)