



Entergy Nuclear Operations, Inc.
1340 Echelon Parkway
Jackson, MS 39213-8298
Tel 601-368-5573

Jeffery A. Hardy
Regulatory Assurance Manager

10 CFR 50.4

PNP 2019-009

February 14, 2019

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

Subject: Core Operating Limits Report – Fuel Cycle 27

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

Dear Sir or Madam:

Enclosed is Revision 20 of the Palisades Nuclear Plant (PNP) Core Operating Limits Report (COLR). This COLR revision is submitted in accordance with the requirements of PNP Technical Specification 5.6.5d, and is applicable to the current PNP fuel cycle 27.

This letter contains no new commitments.

Should you have any questions concerning this letter, or require additional information, please contact Jeff Erickson at 269-764-2375.

Respectfully,

A handwritten signature in blue ink, appearing to be "JAH" followed by a stylized flourish.

JAH/jse

Enclosure: Palisades Nuclear Plant Core Operating Limits Report, Revision 20

cc: NRC Region III Regional Administrator
NRC Senior Resident Inspector – Palisades Nuclear Plant
NRC Project Manager – Palisades Nuclear Plant

PNP 2019-009

ENCLOSURE

Palisades Nuclear Plant Core Operating Limits Report

Revision 20

11 pages follow

PALISADES NUCLEAR PLANT

TITLE: CORE OPERATING LIMITS REPORT

Approved: JPMiksa / 2/14/2019
Procedure Sponsor Date

Process Applicability Exclusion

☐

New Procedure/Revision Summary:

Revision 20

Specific Changes:

Revision 20:

Missing information is restored to Figure 2.2-1, "Regulating Rod Group Position Limits," per CR-PLP-2017-05111 and DRN-17-00721.

Revision 19:

Updated 3.0, Analytical Methods, Item 14 (EMF-92-116(P)(A), adding Supplement 1 (DRN-17-00185)

Revision 18:

See changes in Section 2.4, Total Radial Peaking Factor

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:

<u>Section</u>	<u>Title</u>	<u>LCO</u>
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 $\geq 2\%$ for normal cooldowns and heatups.
- 2.1.3 MODE 5, Loops Not Filled (LCO 3.1.1 SHUTDOWN MARGIN) - The SDM requirement is $\geq 3.5\%$ 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.

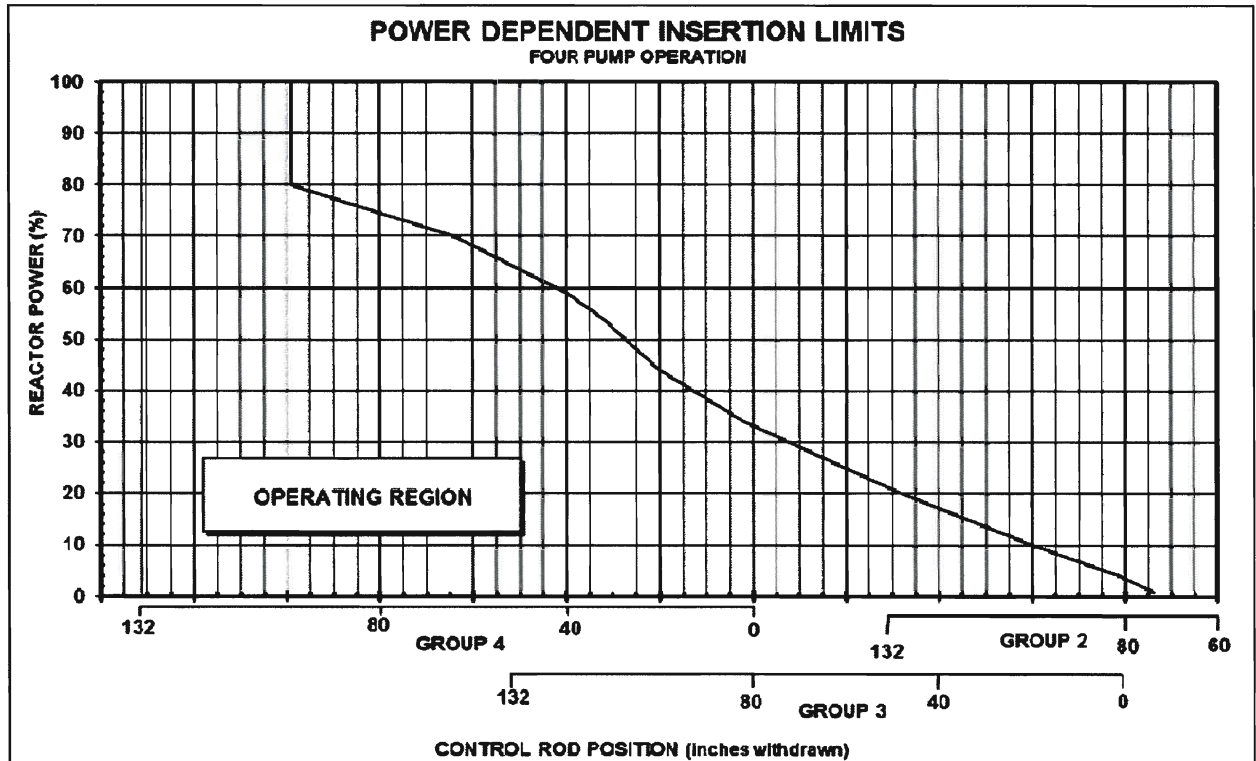


Figure 2.2-1 Regulating Rod Group Position Limits

NOTE: 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:

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

Where:

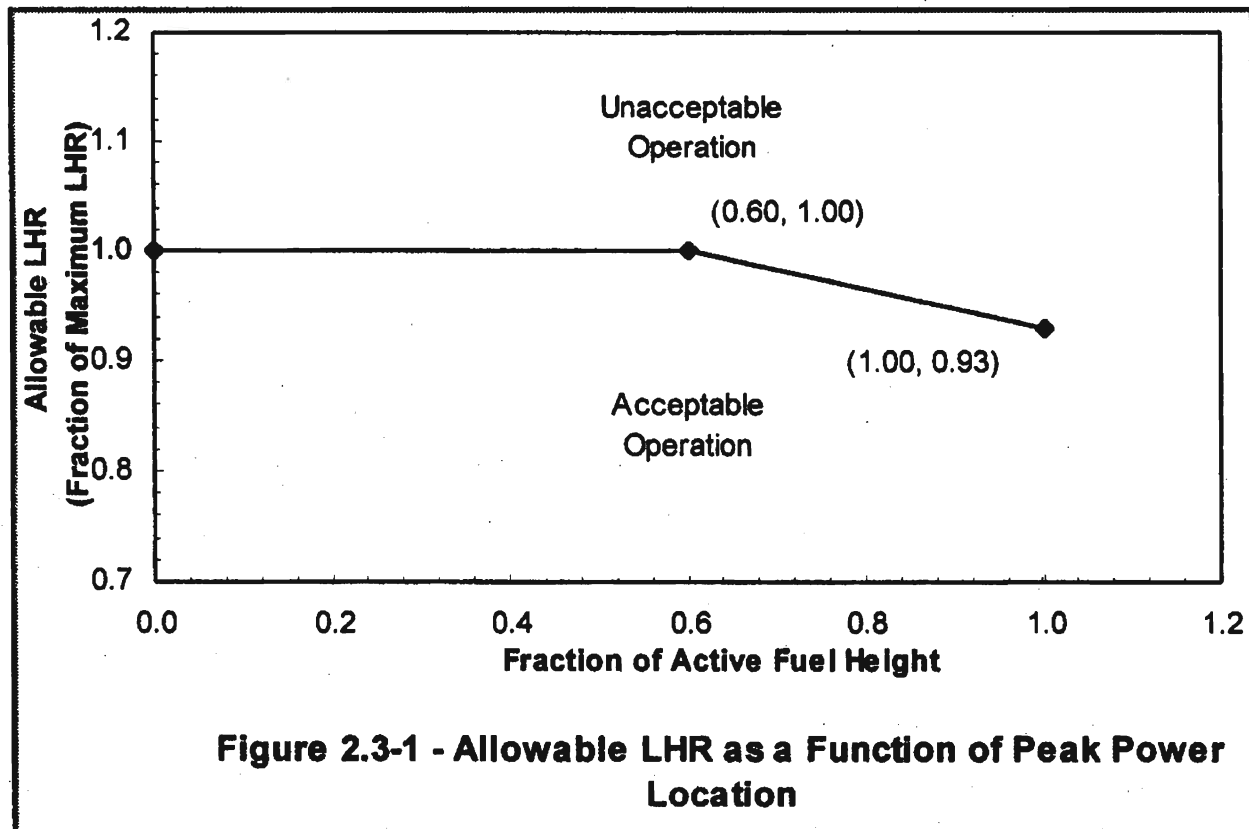
LHR_{TS} = Maximum allowable LHR shown in Table 2.3-1.

$F_A(z)$ = 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).



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:

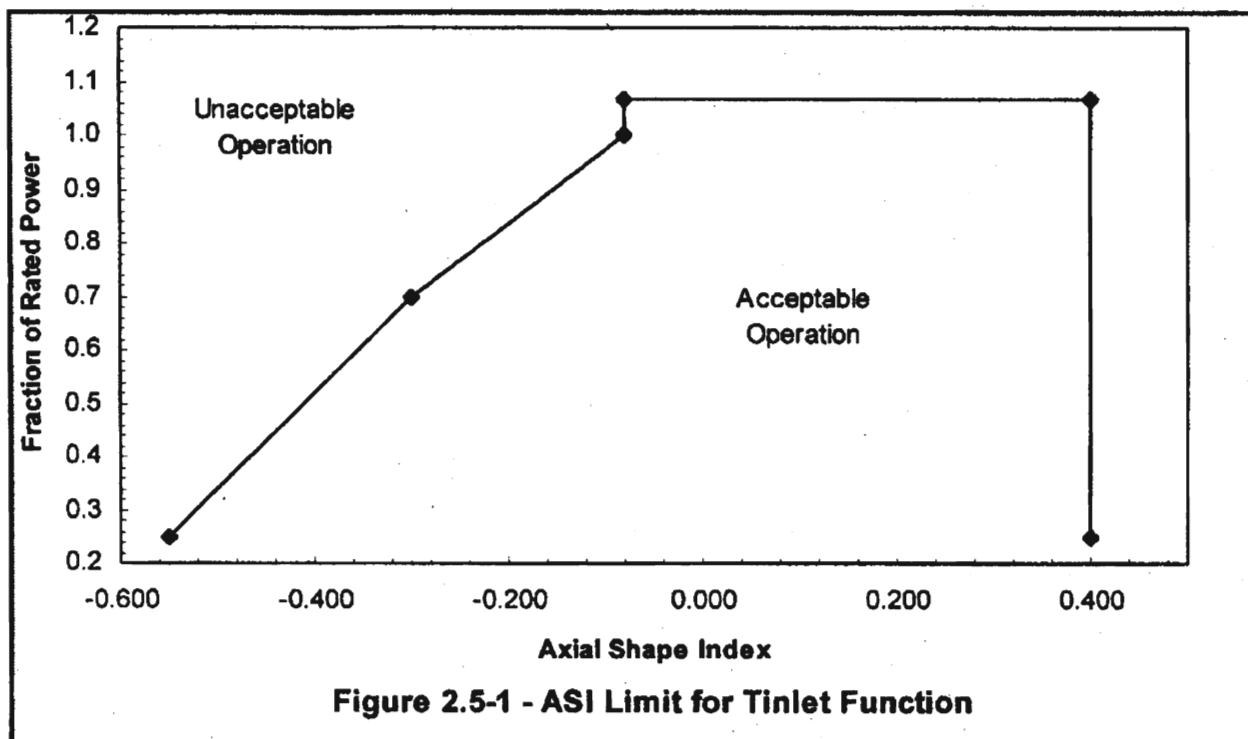
- Fewer than 42 rods in any one assembly violate the "54/6.3" criterion.
- Fewer than 42 assemblies in any core design contain at least one rod that violates the "54/6.3" criterion.
- All rods that violate the "54/6.3" criterion have a rod average linear heat generation rate of less than 6.7 kW/ft.
- 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 $\frac{3}{4}$ 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.



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 ≥ 2010 psia and ≤ 2100 psia
- b. PCS cold leg temperature ≤ 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)
- 14A. 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)
- 14B. EMF-92-116(P)(A) Revision 0, Supplement 1, "Generic Mechanical Design Criteria for PWR Fuel Designs," AREVA Inc, Revision 0, February 2015. (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)
21. BAW-10240(P)-A, Revision 0, May 2004, "Incorporation of M5 Properties in Framatome ANP Approved Methods." (LCOs 3.1.6, 3.2.1, 3.2.2, 3.2.4, & 3.4.1)