PALISADES PLANT

CORE OPERATING LIMITS REPORT

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Core Operating Limits Report

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report for Palisades has been prepared in accordance with the requirements of Technical Specification 6.9.1 (f). The Technical Specifications affected by this report are listed below:

Section	Title	Specification
2.1	ASI Limits for T_{inlet} Function	3.1.1
2.2	Regulating Group Insertion Limits	3.10.5
2.3	Linear Heat Rate (LHR) Limits	3.23.1
2.4	Radial Peaking Factor Limits	3.23.2

NOTE:

Any procedure or document previously referencing the Technical Specifications for any operating limit that has been moved to the COLR should be viewed as referencing the COLR until the applicable procedures or documents are revised.

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 ASI Limits for T_{inlet} Function (Technical Specification 3.1.1)

The ASI limit for the T_{inlet} function is shown in Figure 2.1.

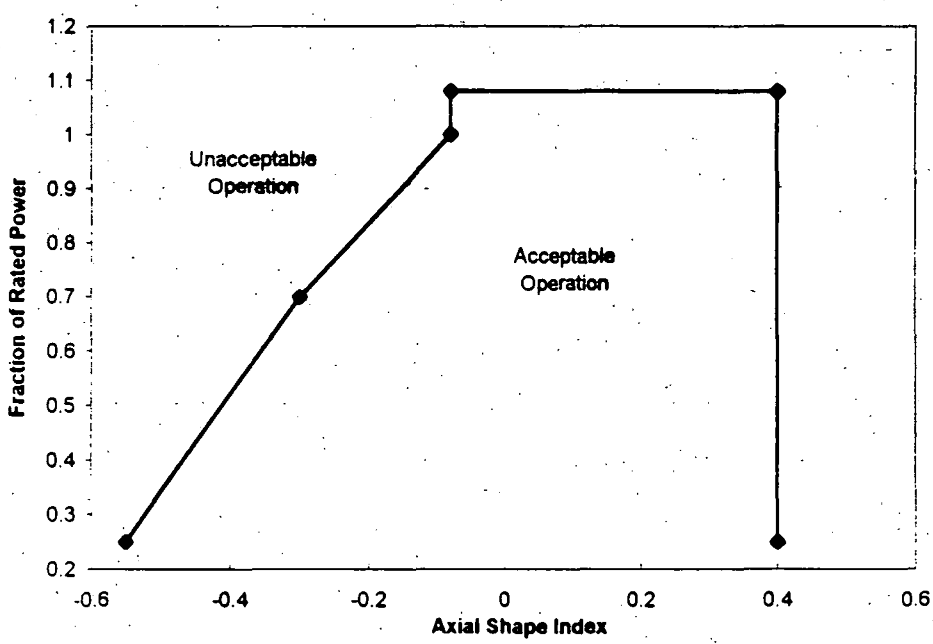

Figure 2.1 - ASI Limit for Tinlet Function

Break Points:

-0.550,	0.250
-0.300,	0.700
-0.080,	1.000
-0.080,	1.080
+0.400,	1.080
+0.400,	0.250

2.2 <u>Regulating Group Insertion Limits</u> (Technical Specification 3.10.5)

a. 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.

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

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

d. If the reactor is subcritical, the rod position at which criticality could be achieved if the control rods were withdrawn in normal sequence shall not be lower than the insertion limit for zero power shown on Figure 2.2.

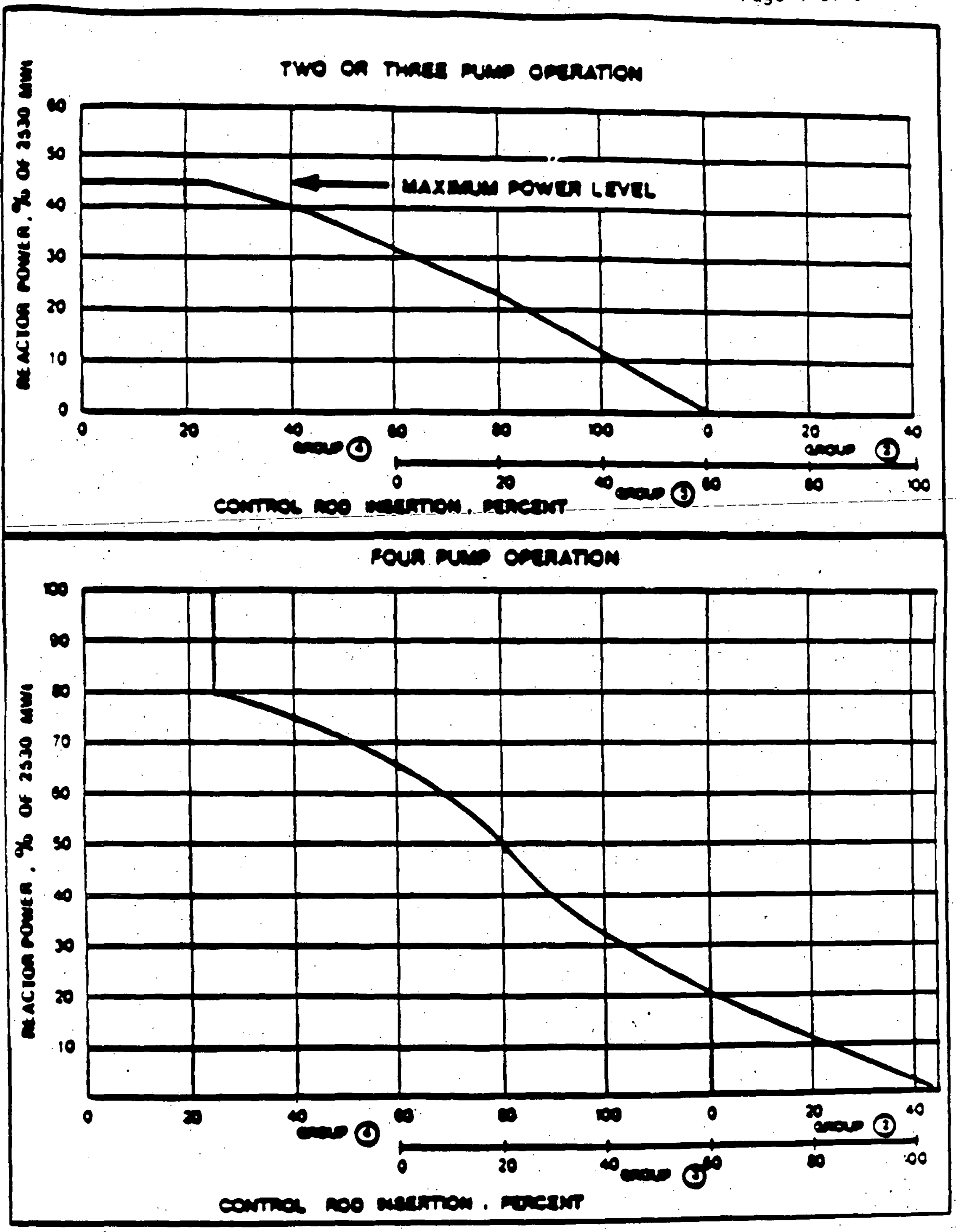

Figure 2.2 - Power Dependent Control Rod Insertion Limit

2.3 <u>Linear Heat Rate (LHR) Limits</u> (Technical Specification 3.23.1)

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 2.1.

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

Table 2.1 - Linear Heat Rate Limit

Peak Rod	15.28 (kW/ft)

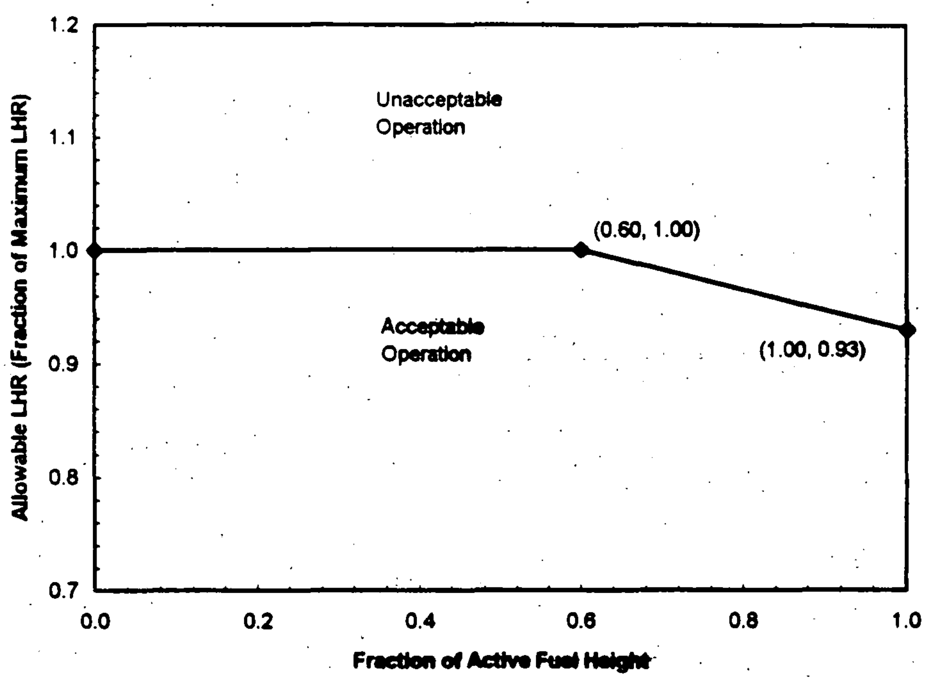

Figure 2.3 - Allowable LHR as a Function of Peak Power Location

2.4 Radial Peaking Factor Limits (Technical Specification 3.23.2)

The radial peaking factor shall not exceed the following:

for $P \geq 0.5$

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

and for $P < 0.5$,

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

Where:

F_r = Measured F_r^A or F_r^T ,

F_r^{TS} = Maximum allowable F_r^A or F_r^T (Table 2.2),

P = Fraction of rated power.

Table 2.2 - Peaking Factor Limits, F_r^{TS}

Peaking Factor	Reload L & M	Reload N	Reload O & P
Assembly F_r^A	1.57	1.66	1.76
Peak Rod F_r^T	1.92	1.92	2.04

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 following documents:

3.1 XN-75-27(A), and Supplements 1 through 5, "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," Exxon Nuclear Company, dated April 1977, Supplement 1 dated September 1976, Supplement 2 dated December 1980, Supplement 3 dated September 1981, Supplement 4 dated December 1986, Supplement 5 dated February 1987.

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

3.3 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.

3.4 ANF-84-093(P)(A), and Supplement 1, "Steamline Break Methodology for PWRs," Advanced Nuclear Fuels Corporation, March 1989.

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

3.6 EXEM PWR Large Break LOCA Model as defined by:

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

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

XN-NF-81-58(A), Revision 2 and Supplements 1 through 4, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, Revision 2 and Supplements 1 and 2 dated March 1984, Supplements 3 and 4 dated June 1990.

XN-NF-85-16(A), Volume 1, Supplements 1 through 3, and Volume 2, Revision 1 and Supplement 1, "PWR 17x17 Fuel Cooling Tests Program," Exxon Nuclear Company, February 1990.

XN-NF-85-105(A), and Supplement 1, "Scaling of FCTF Based Reflood Heat Transfer Correlation for other Bundle Designs," Exxon Nuclear Company, January 1990.

3.7 XN-NF-78-44(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company, October 1983.

3.8 ANF-1224(P)(A), and Supplement 1, "Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Advanced Nuclear Fuels Corporation, April 1990.

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

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

Specific application of these methodologies to Palisades is described in EMF-95-016 Rev. 1, "Palisades Cycle 12 Safety Analysis Report," dated July 1995.