Attachment 1

Millstone Nuclear Power Station Unit No. 2

Core Operating Limits Report - Cycle 13, Revision 0

February 1995

Millstone Unit No. 2

Cycle 13

Core Operating Limits Report

2.0 — CORE OPERATING LIMITS REPORT, CYCLE 13

1. CORE OPERATING LIMITS REPORT

This Core Operating Limits Report for Millstone 2 has been prepared in accordance with the requirements of Technical Specification 6.9.1.7. The Technical Specifications affected by this report are listed below:

Section	Specification	
2.1	3/4.1.1.1	SHUTDOWN MARGIN — T_{avg} > 200°F
2.2	3/4.1.1.2	SHUTDOWN MARGIN — $T_{avg} \leq$ 200°F
2.3	3/4.1.1.4	Moderator Temperature Coefficient
2.4	3/4.1.3.6	Regulating CEA Insertion Limits
2.5	3/4.2.1	Linear Heat Rate
2.6	3/4.2.3	TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR — F_r^T
2.7	3/4.2.6	DNB Margin

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

2. OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using the NRC-approved methodologies specified in Section 3.

2.1 SHUTDOWN MARGIN — T_{avg} > 200°F (Specification 3/4.1.1.1)

The SHUTDOWN MARGIN shall be $\geq$ 3.6% ΔK/K

2.2 SHUTDOWN MARGIN — $T_{avg} \leq$ 200°F (Specification 3/4.1.1.2)

The SHUTDOWN MARGIN shall be $\geq$ 3.6% ΔK/K

2.3 Moderator Temperature Coefficient (Specification 3/4.1.1.4)

The moderator temperature coefficient shall be:

a. Less positive than 0.7×10^{-4} ΔK/K/°F whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER,

b. Less positive than 0.4×10^{-4} $\Delta K/K/{}^\circ F$ whenever THERMAL POWER is > 70% of RATED THERMAL POWER,

c. Less negative than -2.8×10^{-4} $\Delta K/K/{}^\circ F$ at RATED THERMAL POWER.

2.4 Regulating CEA Insertion Limits (Specification 3/4.1.3.6)

The regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits shown in Figure 2.4-1. CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits is restricted to:

a. $\leq$ 4 hours per 24 hour interval,

b. $\leq$ 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. $\leq$ 14 Effective Full Power Days per calendar year.

2.5 Linear Heat Rate (Specification 3/4.2.1)

The Linear heat rate, including heat generated in the fuel, clad and moderator, shall not exceed 15.1 kw/ft.

During operation with the linear heat rate being monitored by the Excore Detector Monitoring System, the AXIAL SHAPE INDEX shall remain within the limits of Figure 2.5-1.

During operation with the linear heat rate being monitored by the Incore Detector Monitor System, the alarm setpoints shall be adjusted to less than or equal to the limit when the following factors are appropriately included in the setting of the alarms:

1. A measurement-calculational uncertainty factor of 1.07,

2. An engineering uncertainty factor of 1.03, and

3. A THERMAL POWER measurement uncertainty factor of 1.02.

2.6 TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR — F_r^T (Specification 3/4.2.3)

The calculated value of F_r^T shall be $\leq$ 1.69.

2.6.1 The Power Dependent F_r^T limits are shown in Figure 2.6-1.

2.6 DNB Margin (Specification 3/4.2.6)

The DNB margin shall be preserved by maintaining the cold leg temperature, pressurizer pressure, reactor coolant flow rate, and AXIAL SHAPE INDEX within the following limits:

Parameter		Limits Four Reactor Coolant Pumps Operations
a.	Cold Leg Temperature	$\leq$ 549°F
b.	Pressurizer Pressure	$\geq$ 2225 psia*
c.	Reactor Coolant Flow Rate	$\geq$ 360,000 gpm
d.	AXIAL SHAPE INDEX	FIGURE 2.7-1

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

3. ANALYTICAL METHODS

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

3.1 XN-75-27(A), Rev. 0 and Supplements 1 through 5, "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," Exxon Nuclear Company. Rev. 0 dated June 1975, 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), Rev. 3, "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuel Corporation, dated May 1988.

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

3.4 ANF-84-93(A), Rev. 0 and Supplement 1, "Steamline Break Methodology for PWR's," Advanced Nuclear Fuels Corporation. Rev. 0 dated March 1989, Supplement 1 dated March 1989.

3.5 XN-75-32(A), Supplements 1, 2, 3, and 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company, dated October 1983.

3.6 XN-NF-82-49(A), Rev. 1 and Supplement 1, "Exxon Nuclear Company Evaluation Model EXEM PWR Small Break Model," Advanced Nuclear Fuels Corporation, both reports dated April 1989.

3.7 EXEM PWR Large Break LOCA Evaluation Model as defined by:

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

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

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

d. XN-NF-85-16(A), Volume 1 through Supplement 3; Volume 2, Rev. 1 and Supplement 1, "PWR 17x17 Fuel Cooling Tests Program," Exxon Nuclear Company. All reports dated February 1990.

e. XN-NF-85-105(A), Rev. 0 and Supplement 1, "Scaling of FCTF Based Reflood Heat Transfer Correlation for Other Bundle Designs," Exxon Nuclear Company. Both reports dated January 1990.

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

3.9 XN-NF-621(A), Rev. 1, "Exxon Nuclear DNB Correlation of PWR Fuel Design," Exxon Nuclear Company, dated September 1983.

An acceptable Millstone 2 specific application of these analytical methodologies is described in ANF-88-126, "Millstone Unit 2 Cycle 10 Safety Analysis Report," dated October 1988.

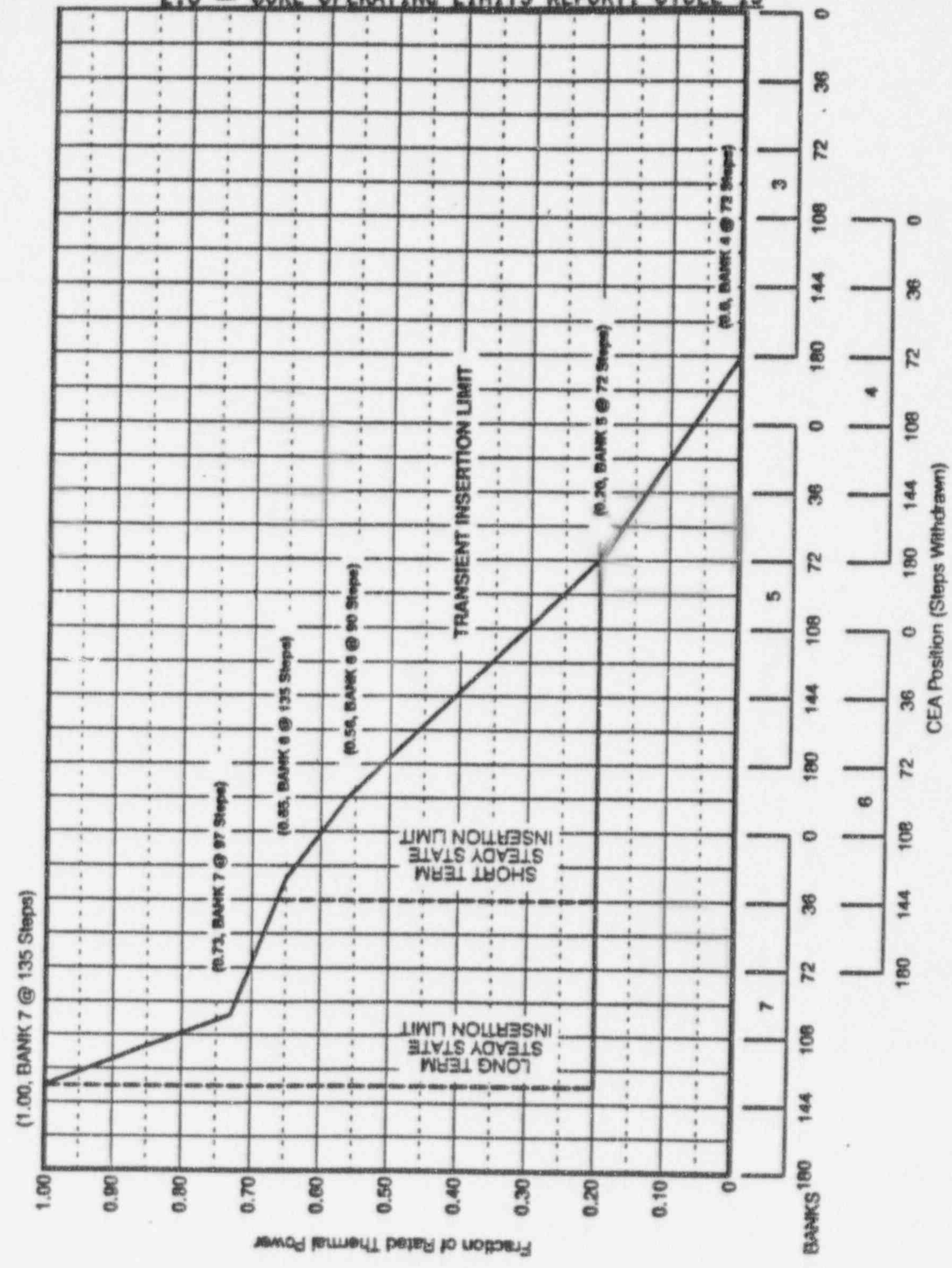

FIGURE 2.4—1
CEA Insertion Limit vs. THERMAL POWER with
Four Reactor Coolant Pumps Operating

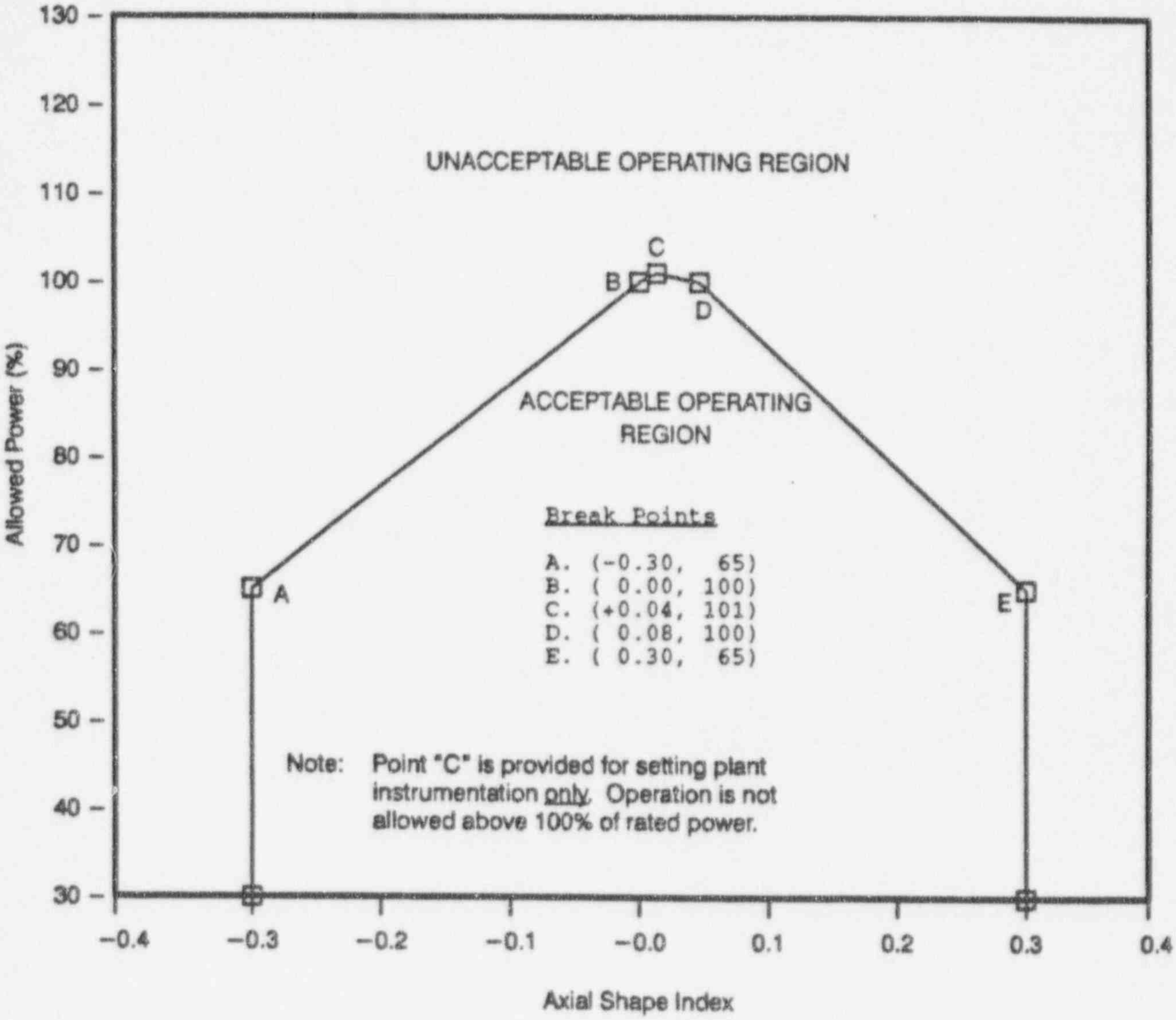

FIGURE 2.5—1
AXIAL SHAPE INDEX vs. PERCENT OF ALLOWABLE POWER LEVEL

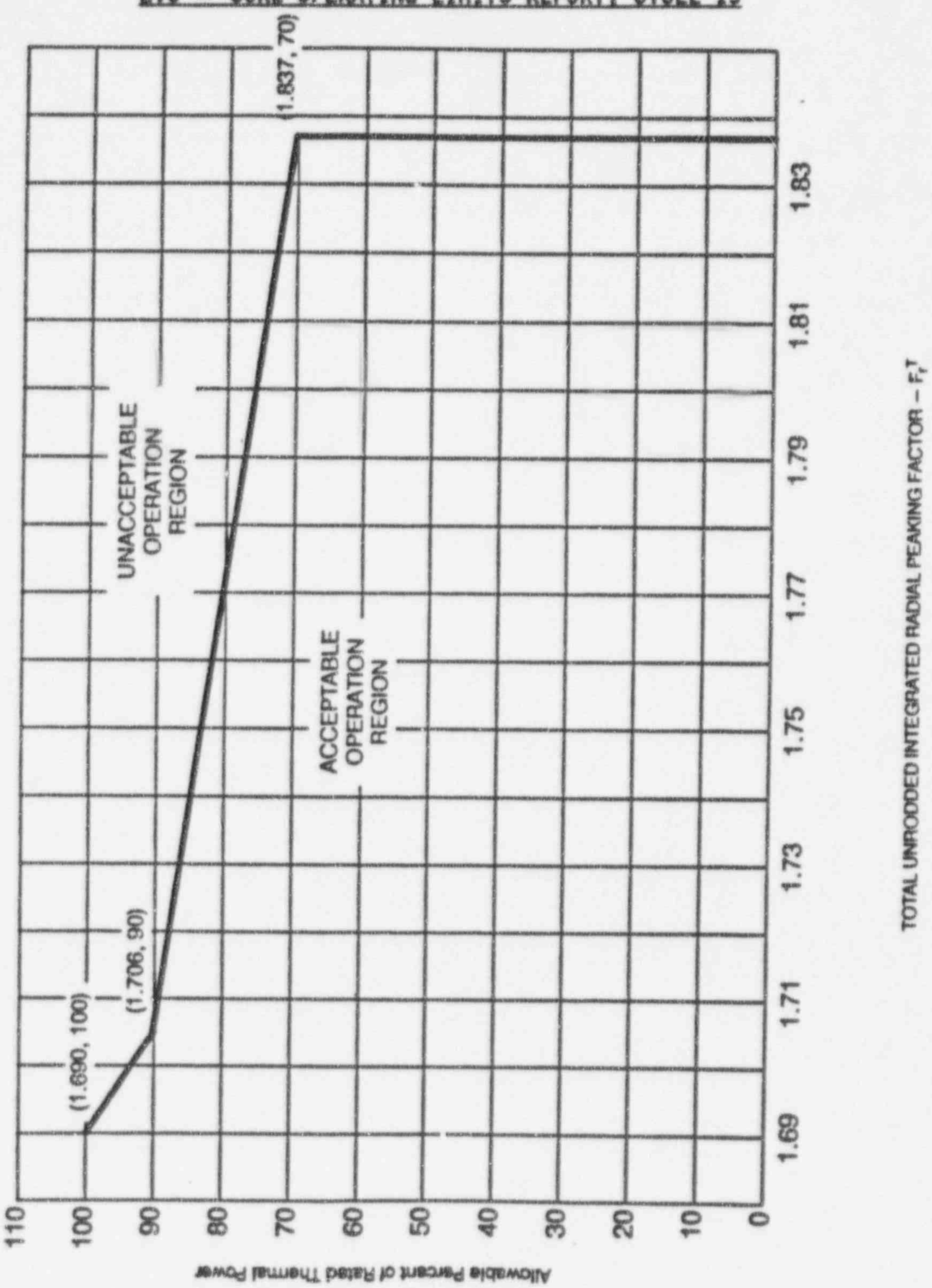

FIGURE 2.6—1
TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR
vs. Allowable Rated Thermal Power

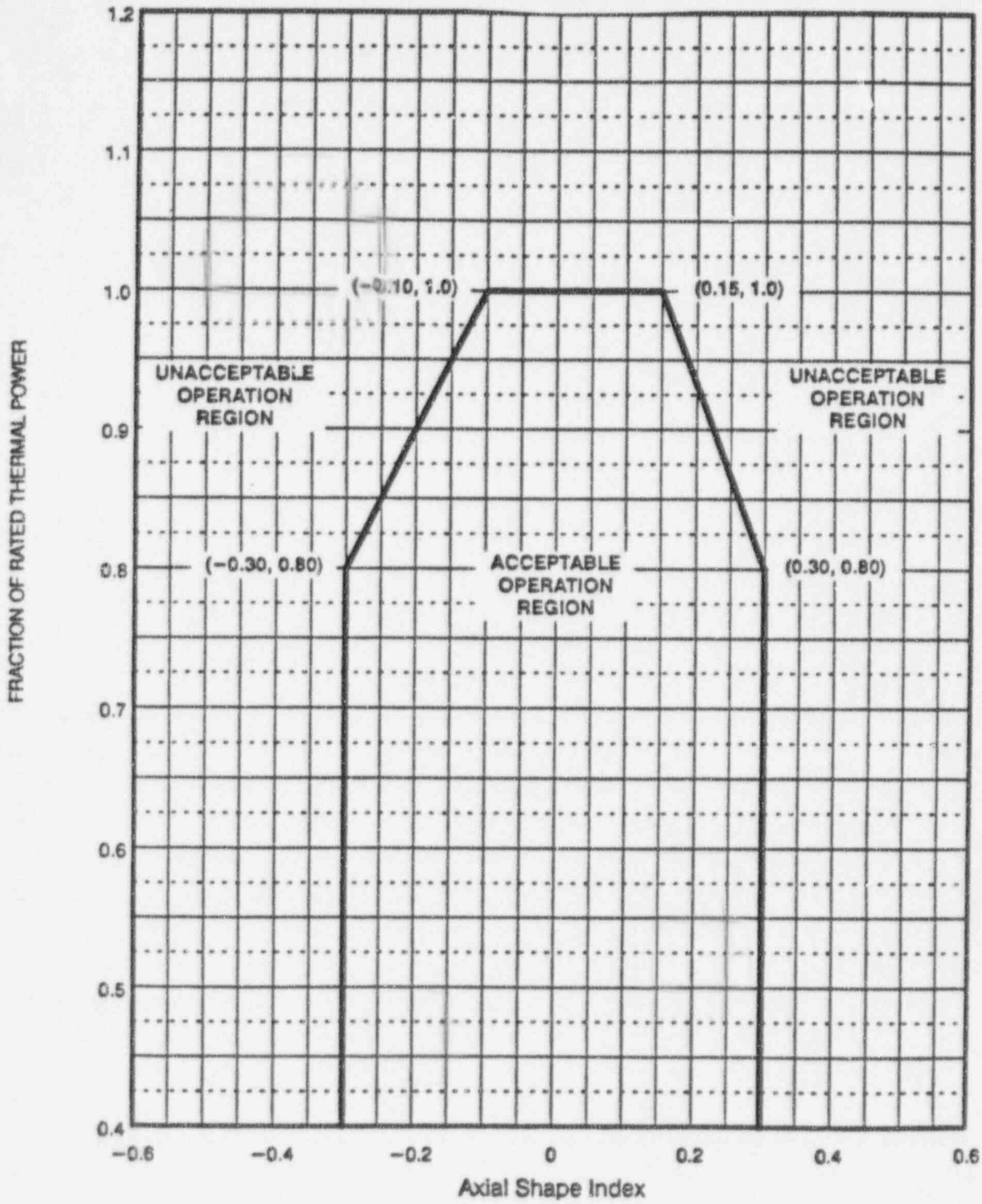

FIGURE 2.7—1
AXIAL SHAPE INDEX Operating Limits with
Four Reactor Coolant Pumps Operating