

June 19, 1985

Docket No. 50-336

Mr. John F. Opeka, Senior Vice President  
Nuclear Engineering and Operations  
Northeast Nuclear Energy Company  
P. O. Box 270  
Hartford, Connecticut 06141-0270

**DISTRIBUTION:**

Docket File	LTremper
NRC PDR	RDiggs
L PDR	ACRS-10
SECY	WJones
ORB#3 Rdg	TBarnhart-4
HThompson	MVirgilio
PMKreutzer-3	EJordan
DBOsborne	LJHarmon
OELD	CMiles
Gray File +4	BGrimes

Dear Mr. Opeka:

The Commission has issued the enclosed Amendment No. 99 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit No. 2, in response to your application dated February 6, 1985 as supplemented June 5 and June 11, 1985.

These revisions to the Technical Specifications modify the allowable region of operation when the core power distribution is monitored by the Excore Detector Monitoring System. These revisions reflect changes in Cycle 7 operating characteristics and allow operation in fuel Cycle 7.

A copy of our Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

/S/

D. B. Osborne, Project Manager  
Operating Reactors Branch #3  
Division of Licensing

**Enclosures:**

1. Amendment No. 99 to DPR-65
2. Safety Evaluation

cc w/enclosures:  
See next page

ORB#3:DL  
PMKreutzer  
6/17/85

ORB#3:DL  
DBOsborne  
6/17/85

ORB#3:DL  
EButcher  
6/17/85

OELD  
6/18/85

AD:OR:DL  
GCLainas  
6/19/85

8507050069 850619  
PDR ADOCK 05000336  
P PDR

Mr. John F. Opeka  
Northeast Nuclear Energy Company

Millstone Nuclear Power Station  
Unit No. 2

cc:  
Gerald Garfield, Esq.  
Day, Berry & Howard  
Counselors at Law  
City Place  
Hartford, Connecticut 06103-3499

Mr. Wayne D. Romberg  
Superintendent  
Millstone Nuclear Power Station  
P. O. Box 128  
Waterford, Connecticut 06385

Regional Administrator, Region I  
U.S. Nuclear Regulatory Commission  
Office of Executive Director for  
Operations  
631 Park Avenue  
King of Prussia, Pennsylvania 19406

Mr. Edward J. Mroczka  
Vice President, Nuclear Operations  
Northeast Nuclear Energy Company  
P. O. Box 270  
Hartford, Connecticut 06101

Mr. Charles Brinkman, Manager  
Washington Nuclear Operations  
C-E Power Systems  
Combustion Engineering, Inc.  
7910 Woodmont Avenue  
Bethesda, Maryland 20814

Mr. Lawrence Bettencourt, First Selectman  
Town of Waterford  
Hall of Records - 200 Boston Post Road  
Waterford, Connecticut 06385

Northeast Utilities Service Company  
ATTN: Mr. Richard R. Laudonat, Manager  
Generation Facilities Licensing  
Post Office Box 270  
Hartford, Connecticut 06101

Arthur Heubner, Director  
Radiation Control Unit  
Department of Environmental  
Protection  
State Office Building  
Hartford, Connecticut 06116

Mr. John Shedlosky  
Resident Inspector/Millstone  
Box 811  
Niantic, Connecticut 06357

Office of Policy & Management  
ATTN: Under Secretary Energy  
Division  
80 Washington Street  
Hartford, Connecticut 06115



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

NORTHEAST NUCLEAR ENERGY COMPANY  
THE CONNECTICUT LIGHT AND POWER COMPANY  
THE WESTERN MASSACHUSETTS ELECTRIC COMPANY  
DOCKET NO. 50-336  
MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2  
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 99  
License No. DPR-65

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Northeast Nuclear Energy Company, et al. (the licensee), dated February 6, 1985 as supplemented June 5 and June 11, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
  - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
  - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
  - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

8507050072 850619  
PDR ADOCK 05000336  
P PDR

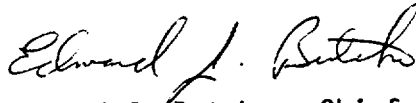
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-65 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 99, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Edward J. Butcher, Chief  
Operating Reactors Branch #3  
Division of Licensing

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: June 19, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 99

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are provided to maintain document completeness.

Remove Pages

IV

3/4 2-1  
3/4 2-2  
3/4 2-4  
3/4 2-5  
3/4 2-6  
3/4 2-7

3/4 2-8

3/4 2-9

Insert Pages

IV

3/4 2-1  
3/4 2-2  
3/4 2-4  
3/4 2-5  
3/4 2-6  
3/4 2-7  
3/4 2-7(a)  
3/4 2-8  
2-8(a)  
3/4 2-9

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>	<u>PAGE</u>
3/4.0 APPLICABILITY.....	3/4 0-1
<u>3/4.1 REACTIVITY CONTROL SYSTEMS</u>	
3/4.1.1 BORATION CONTROL.....	3/4 1-1
Shutdown Margin - $T_{avg} > 200^{\circ}\text{F}$ .....	3/4 1-1
Shutdown Margin - $T_{avg} \leq 200^{\circ}\text{F}$ .....	3/4 1-3
Boron Dilution .....	3/4 1-4
Moderator Temperature Coefficient (MTC).....	3/4 1-5
Minimum Temperature for Criticality.....	3/4 1-7
3/4.1.2 BORATION SYSTEMS.....	3/4 1-8
Flow Paths - Shutdown.....	3/4 1-8
Flow Paths - Operating.....	3/4 1-10
Charging Pump - Shutdown.....	3/4 1-12
Charging Pumps - Operating.....	3/4 1-13
Boric Acid Pumps - Shutdown.....	3/4 1-14
Boric Acid Pumps - Operating.....	3/4 1-15
Borated Water Sources - Shutdown.....	3/4 1-16
Borated Water Sources - Operating.....	3/4 1-18
3/4.1.3 MOVABLE CONTROL ASSEMBLIES.....	3/4 1-20
Full Length CEA Group Position.....	3/4 1-20
Position Indicator Channels.....	3/4 1-24
CEA Drop Time.....	3/4 1-26
Shutdown CEA Insertion Limit.....	3/4 1-27
Regulating CEA Insertion Limits.....	3/4 1-28

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>	<u>PAGE</u>
<u>3/4.2 POWER DISTRIBUTION LIMITS</u>	
3/4.2.1 LINEAR HEAT RATE .....	3/4 2-1
3/4.2.2 TOTAL PLANAR RADIAL PEAKING FACTOR - $F_{xy}^T$ .....	3/4 2-5
3/4.2.3 TOTAL INTEGRATED RADIAL PEAKING FACTOR - $F_r^T$ .....	3/4 2-9
3/4.2.4 AZIMUTHAL POWER TILT .....	3/4 2-10
3/4.2.5 FUEL RESIDENCE TIME .....	3/4 2-12
3/4.2.6 DNB MARGIN .....	3/4 2-13
<u>3/4.3 INSTRUMENTATION</u>	
3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION .....	3/4 3-1
3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION .....	3/4 3-10
3/4.3.3 MONITORING INSTRUMENTATION .....	3/4 3-26
Radiation Monitoring .....	3/4 3-26
Incore Detectors .....	3/4 3-30
Seismic Instrumentation .....	3/4 3-32
Meteorological Instrumentation .....	3/4 3-36
Remote Shutdown Instrumentation .....	3/4 3-39
Chlorine Detection Systems .....	3/4 3-42
Fire Detection Instrumentation .....	3/4 3-43
Accident Monitoring .....	3/4 3-46
<u>3/4.4 REACTOR COOLANT SYSTEM</u>	
3/4.4.1 COOLANT LOOPS AND COOLANT CIRCULATION .....	3/4 4-1
Startup and Power Operation .....	3/4 4-1
Hot Standby .....	3/4 4-1a
Shutdown .....	3/4 4-1b
3/4.4.2 SAFETY VALVES .....	3/4 4-2
3/4.4.3 RELIEF VALVES .....	3/4 4-3

### 3/4.2 POWER DISTRIBUTION LIMITS

#### LINEAR HEAT RATE

#### LIMITING CONDITION FOR OPERATION

---

3.2.1 The linear heat rate shall not exceed the limits shown on Figure 3.2-1.

APPLICABILITY: MODE 1.

#### ACTION:

During operation with the linear heat rate being monitored by the Incore Detector Monitoring System, comply with the following ACTION:

With the linear heat rate exceeding its limit, as indicated by four or more coincident incore channels, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

- a. Restore the linear heat rate to within its limits within 1 hour, or
- b. Be in at least HOT STANDBY within the next 6 hours.

During operation with the linear heat rate being monitored by the Excore Detector Monitoring System, comply with the following ACTIONS:

With the linear heat rate exceeding its limit, as indicated by the AXIAL SHAPE INDEX being outside of the power dependent limits on the Power Ratio Recorder and with the THERMAL POWER:

- a. Above 100% of the allowable power level determined by Specification 4.2.1.2.c, within 15 minutes either restore the AXIAL SHAPE INDEX to within the allowable limits as required per Technical Specification 3.2.2 or reduce THERMAL POWER to  $\leq$  100% of the allowable power level determined by Specification 4.2.1.2.c.
- b.  $<$  100% of the allowable power level determined by Specification 4.2.1.2.c, either restore the AXIAL SHAPE INDEX to within the allowable limits as required per Technical Specification 3.2.2 within 1 hour from initially exceeding the linear heat rate limit or be in HOT STANDBY within the next 4 hours.

#### SURVEILLANCE REQUIREMENTS

---

4.2.1.1 The linear heat rate shall be determined to be within its limits by continuously monitoring the core power distribution with either the excore detector monitoring system or with the incore detector monitoring system.



## POWER DISTRIBUTION LIMITS

### SURVEILLANCE REQUIREMENTS (Continued)

---

4.2.1.2 Excore Detector Monitoring System - The excore detector monitoring system may be used for monitoring the core power distribution by:

- a. Verifying at least once per 12 hours that the full length CEAs are withdrawn to and maintained at or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6.
- b. Verifying at least once per 31 days that the AXIAL SHAPE INDEX alarm setpoints are adjusted to within the allowable limits as required per Technical Specification 3.2.2.
- c. Verifying at least once per 31 days that the AXIAL SHAPE INDEX is maintained within the allowable limits as required per Technical Specification 3.2.2, where 100 percent of the allowable power represents the maximum THERMAL POWER allowed by the existing Reactor Coolant Pump Combination.

4.2.1.3 Incore Detector Monitoring System - The incore detector monitoring system may be used for monitoring the core power distribution by verifying that the incore detector Local Power Density alarms:

- a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days.
- b. Have their alarm setpoint adjusted to less than or equal to the limits shown on Figure 3.2-1 when the following factors are appropriately included in the setting of these alarms:
  1. Flux peaking augmentation factors as shown in Figure 4.2-1,
  2. A measurement-calculational uncertainty factor of 1.07,
  3. An engineering uncertainty factor of 1.03,
  4. A linear heat rate uncertainty factor of 1.01 due to axial fuel densification and thermal expansion, and
  5. A THERMAL POWER measurement uncertainty factor of 1.02.

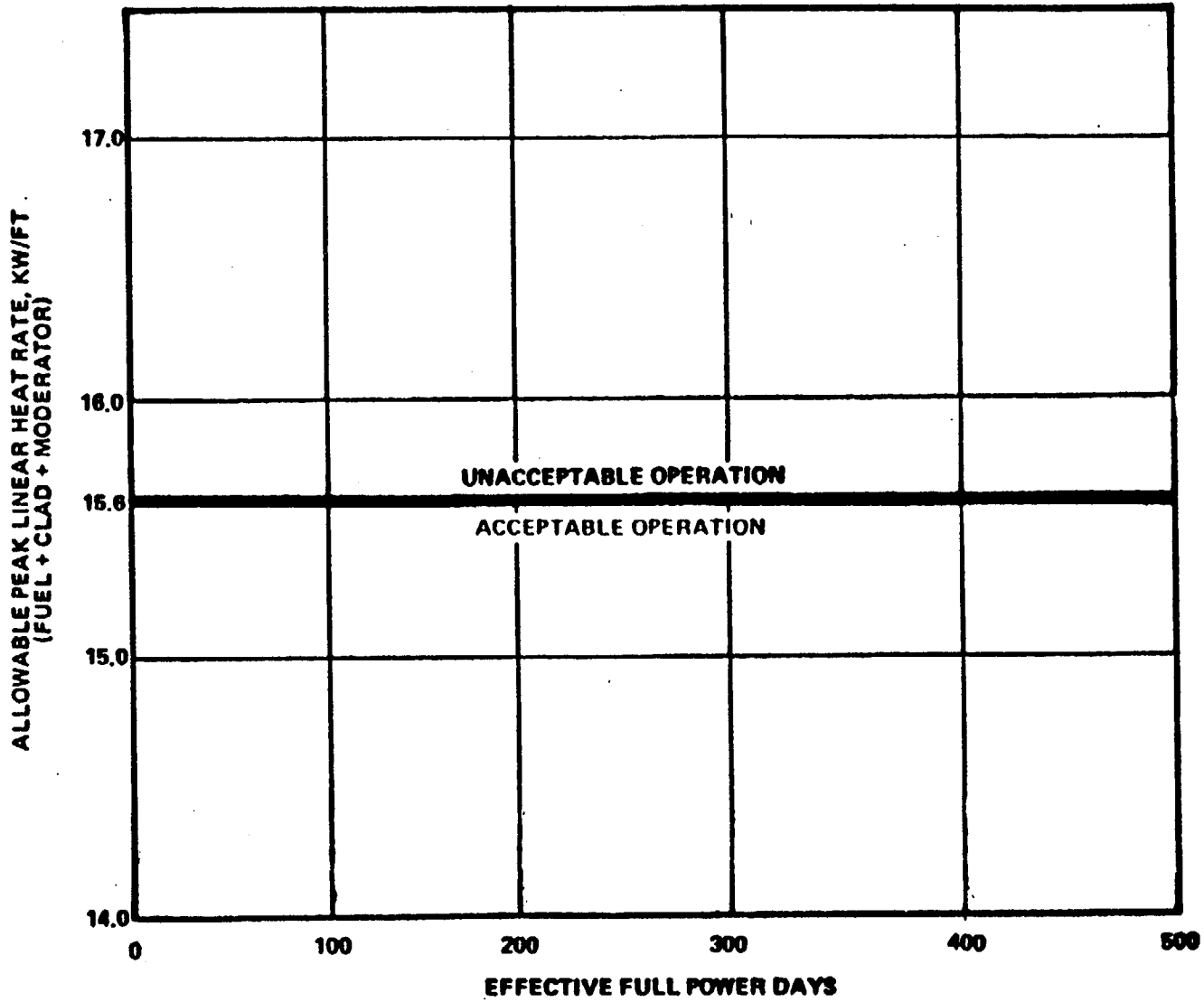


Figure 3.2-1 Allowable Peak Linear Heat Rate vs Burnup

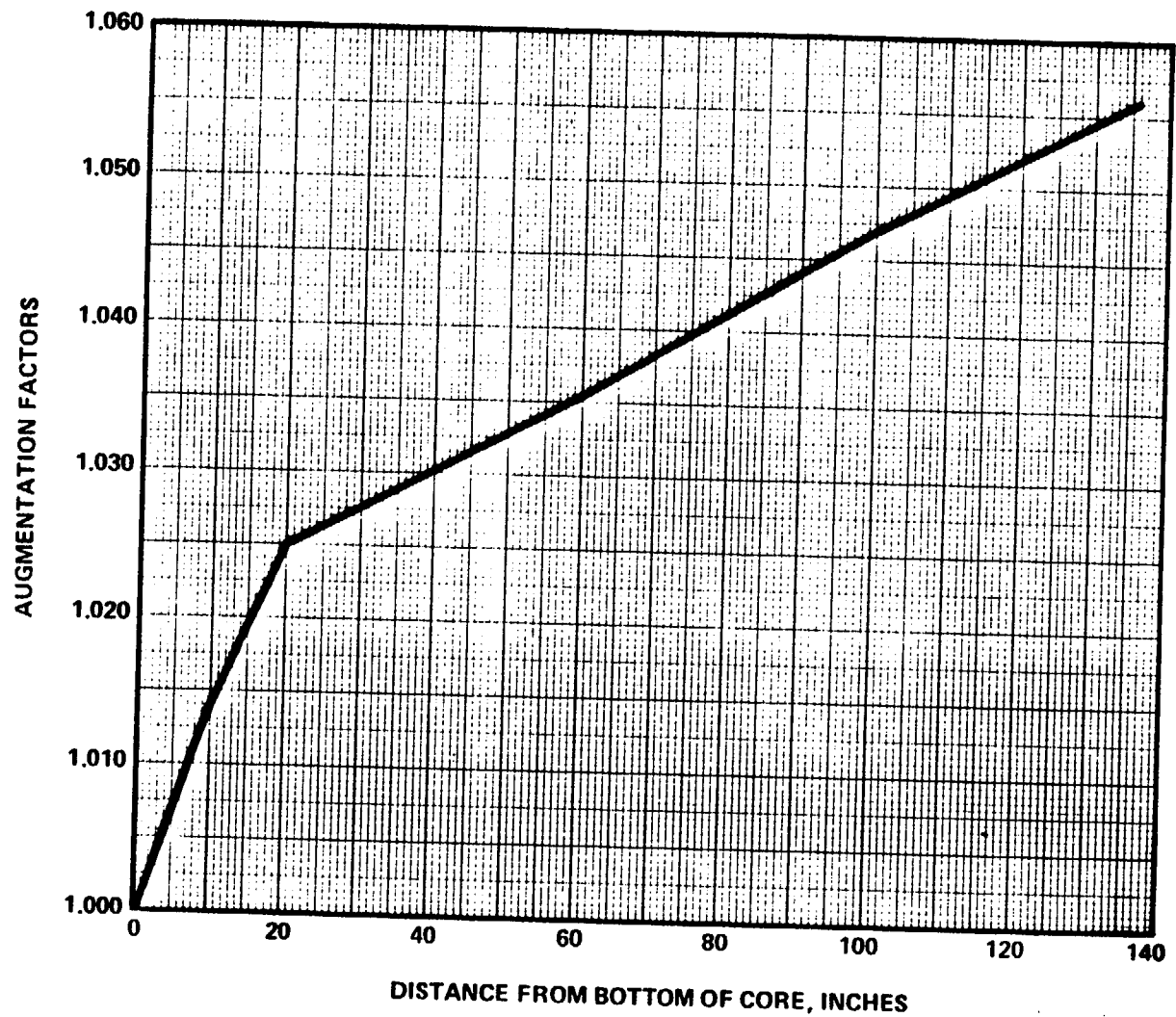


FIGURE 4.2-1 Augmentation Factors vs Distance From Bottom of Core

## POWER DISTRIBUTION LIMITS

TOTAL PLANAR RADIAL PEAKING FACTOR -  $F_{xy}^T$

### LIMITING CONDITION FOR OPERATION

3.2.2 Meet either of 3.2.2.1 or 3.2.2.2.

3.2.2.1 The calculated value of  $F_{xy}^T$ , defined as  $F_{xy}^T = F_{xy} (1+T_q)$ , shall be limited to  $\leq 1.62$  with the AXIAL SHAPE INDEX alarm setpoints adjusted consistent with the limits shown on Figure 3.2-2a, or

3.2.2.2 The calculated value of  $F_{xy}^T$ , defined as  $F_{xy}^T = F_{xy} (1+T_q)$ , shall be limited to  $\leq 1.719$  with the AXIAL SHAPE INDEX alarm setpoints adjusted consistent with the limits shown on Figure 3.2-2b.

APPLICABILITY: MODE 1\*.

#### ACTION:

- a. With  $F_{xy}^T > 1.62$  and the AXIAL SHAPE INDEX alarm setpoints adjusted consistent with the limits shown on Figure 3.2-2a, within 6 hours either:
  - 1) Reduce THERMAL POWER to bring the combination of THERMAL POWER and  $F_{xy}^T$  to within the limits of Figure 3.2-3a and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limit of Specification 3.1.3.6, or
  - 2) Apply the limits of Specification 3.2.2.2 and Figure 3.2-3b and within 72 hours adjust the AXIAL SHAPE INDEX alarm setpoints consistent with the limits shown on Figure 3.2-2b, or
  - 3) Be in at least HOT STANDBY.
- b. With  $F_{xy}^T > 1.719$  and the AXIAL SHAPE INDEX alarm setpoints adjusted consistent with the limits shown on Figure 3.2-2b, within 6 hours either:
  - 1) Reduce THERMAL POWER to bring the combination of THERMAL POWER and  $F_{xy}^T$  to within the limits of Figure 3.2-3b and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limit of Specification 3.1.3.6, or
  - 2) Be in at least HOT STANDBY.

\*See Special Test Exception 3.10.2.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

---

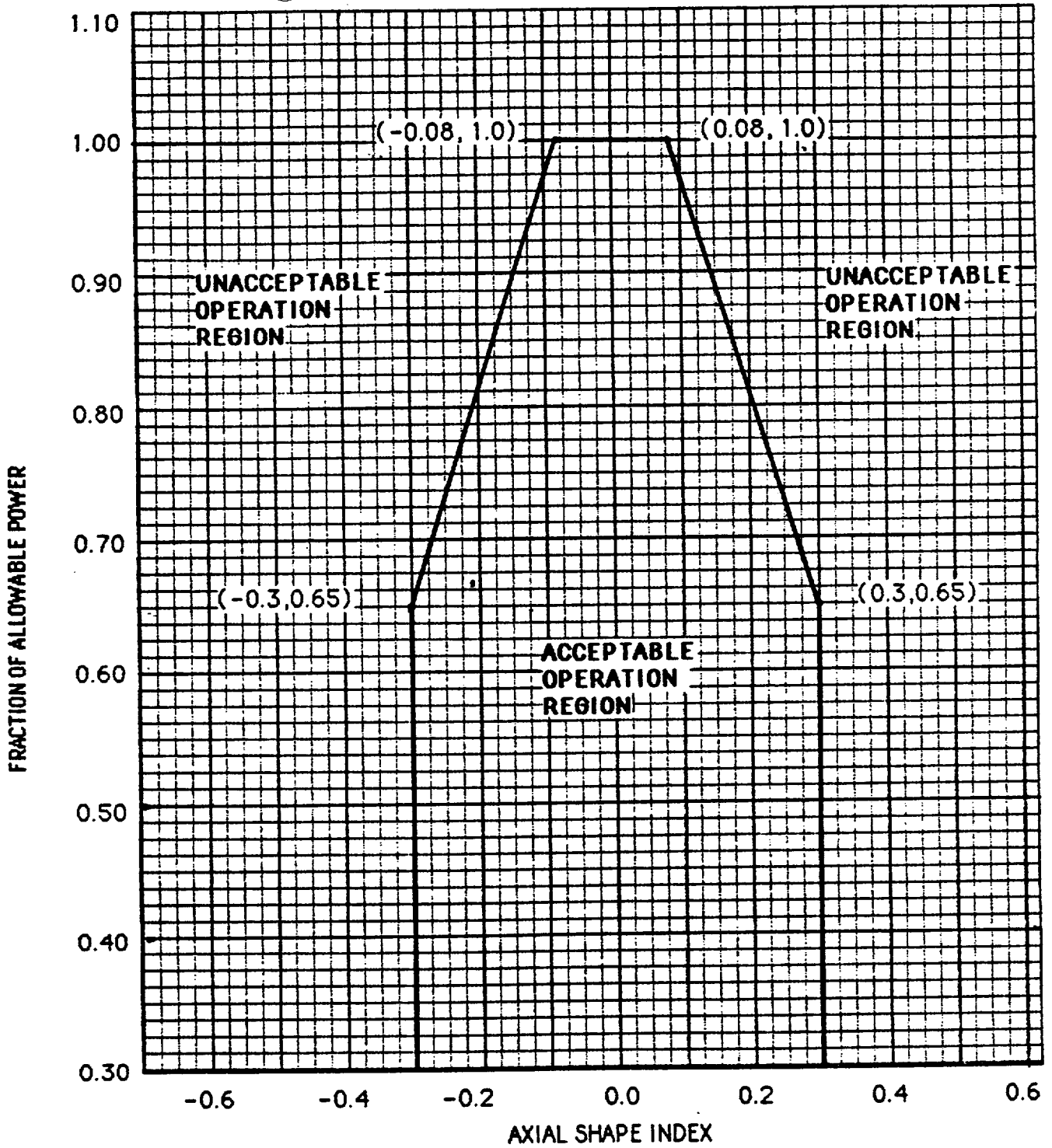
4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2  $F_{xy}^T$  shall be calculated by the expression  $F_{xy}^T = F_{xy} (1+T_q)$  and  $F_{xy}^T$  shall be determined to be within its limit at the following intervals:

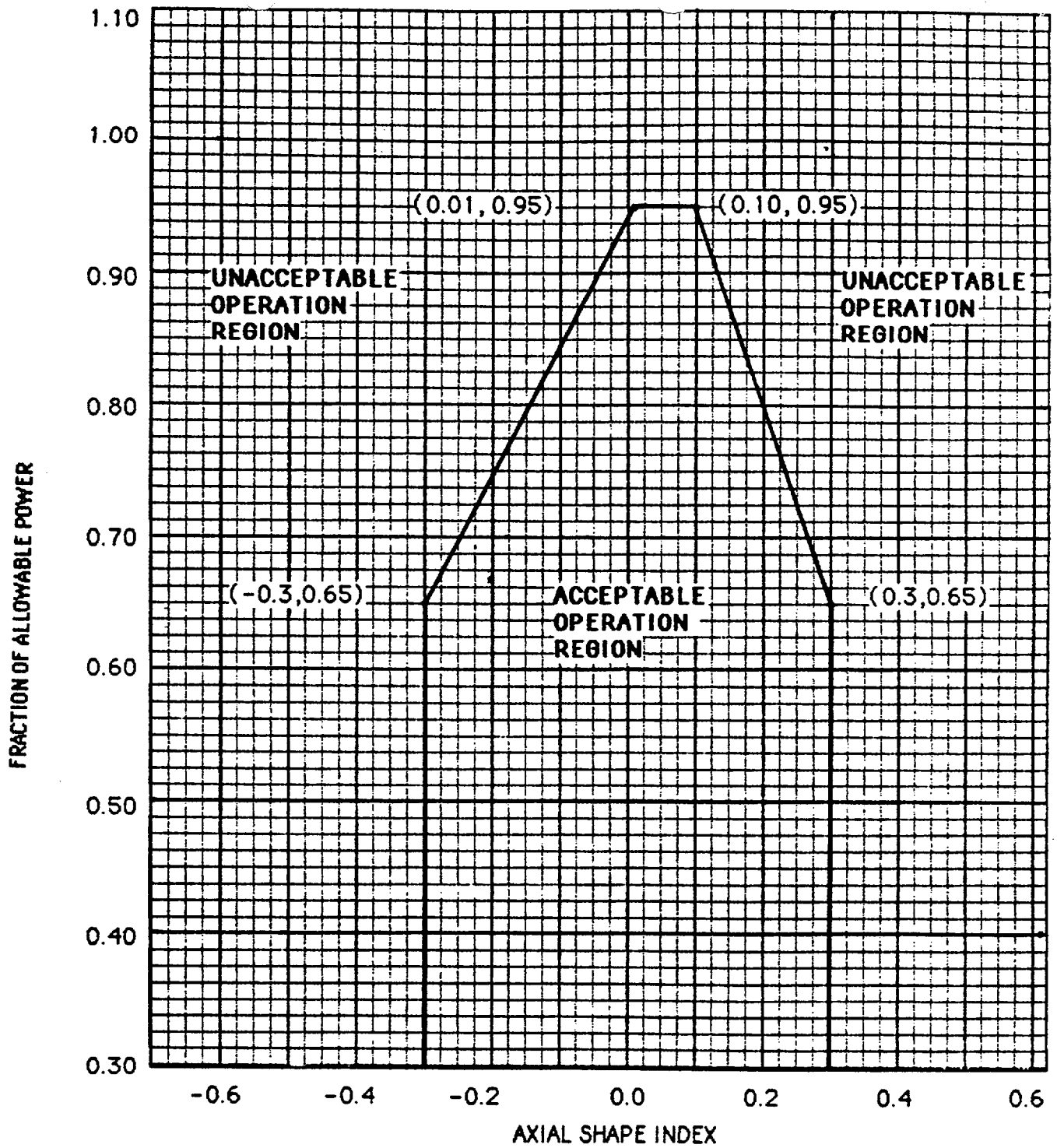
- a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,
- b. At least once per 31 days of accumulated operation in MODE 1, and
- c. Within four hours if the AZIMUTHAL POWER TILT ( $T_q$ ) is  $> 0.02$ .

4.2.2.3  $F_{xy}$  shall be determined each time a calculation of  $F_{xy}^T$  is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. This determination shall be limited to core planes between 15% and 85% of full core height inclusive and shall exclude regions influenced by grid effects.

4.2.2.4  $T_q$  shall be determined each time a calculation of  $F_{xy}^T$  is required and the value of  $T_q$  used to determine  $F_{xy}^T$  shall be measured value of  $T_q$ .

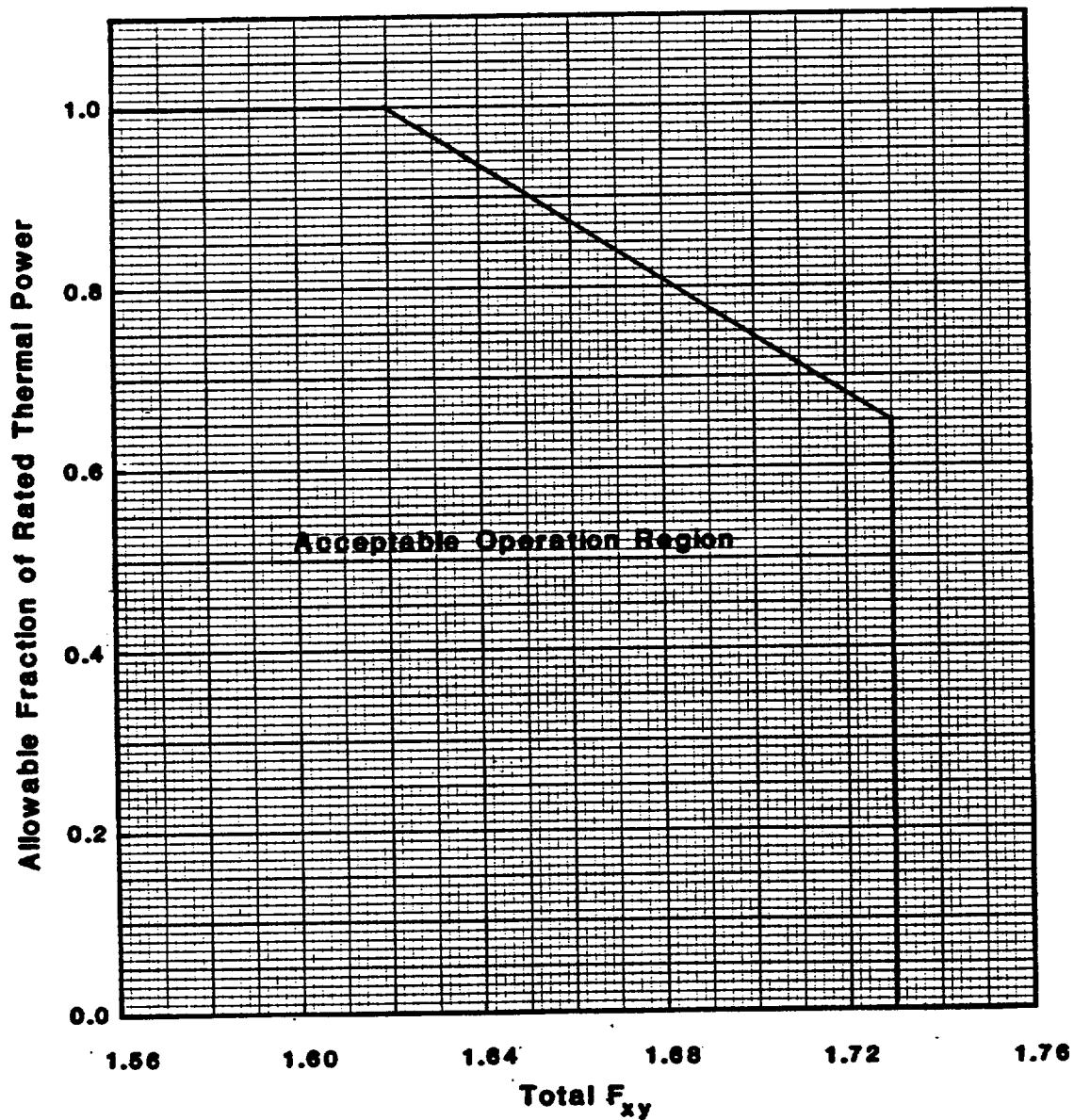


**FIGURE 3.2-2a AXIAL SHAPE INDEX vs FRACTION OF ALLOWABLE POWER LEVEL per Specification 4.2.1.2c**

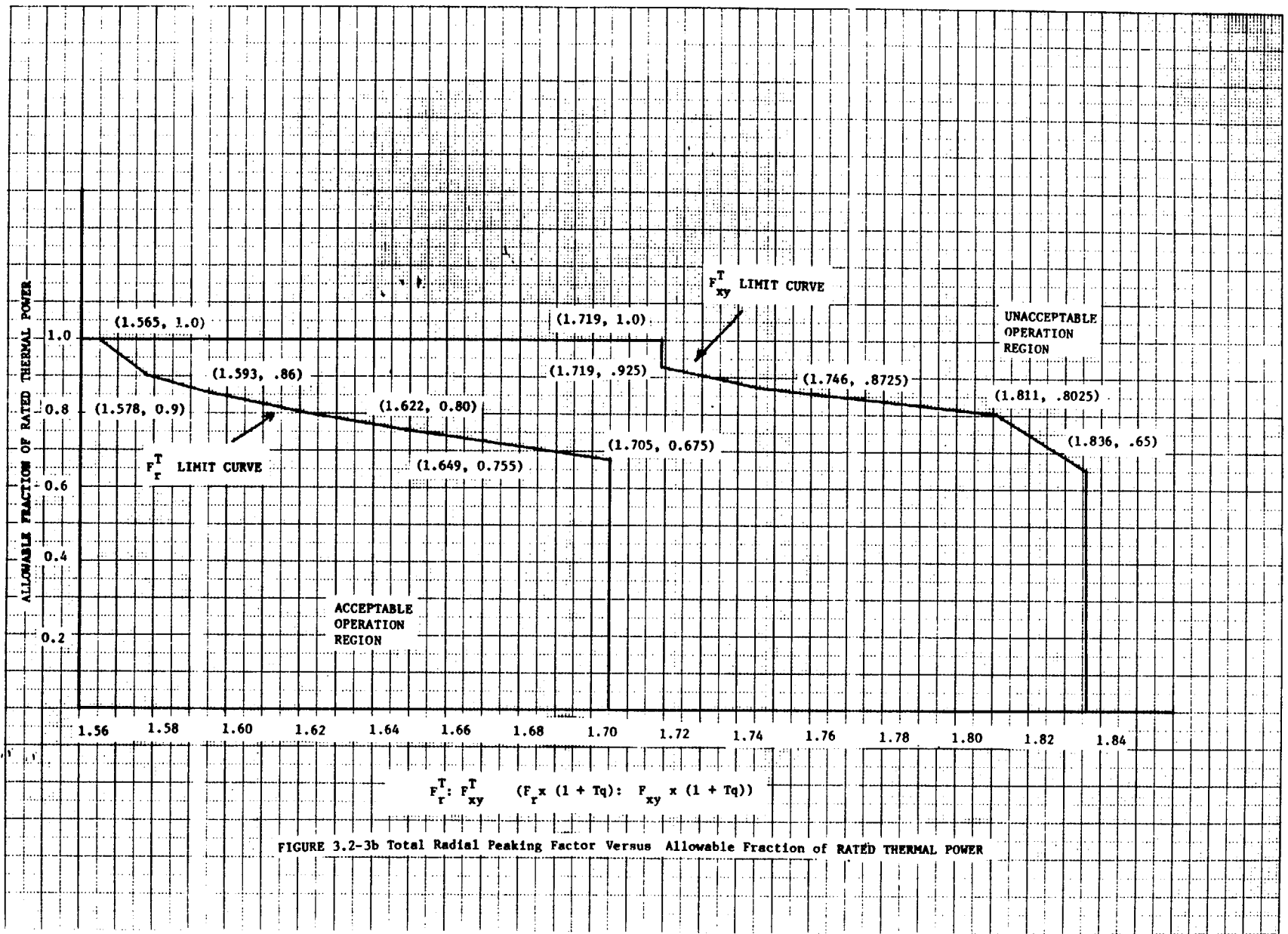


**FIGURE 3.2-2b AXIAL SHAPE INDEX vs FRACTION OF ALLOWABLE POWER LEVEL  
per Specification 4.2.1.2c**

**FIGURE 3.2-3a Total Radial Peaking Factor vs Allowable Fraction of Rated Thermal Power**







## POWER DISTRIBUTION LIMITS

TOTAL INTEGRATED RADIAL PEAKING FACTOR -  $F_r^T$

### LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of  $F_r^T$ , defined as  $F_r^T = F_r(1+T_q)$  shall be limited to  $\leq 1.565$ .

APPLICABILITY: MODE 1\*.

#### ACTION:

With  $F_r^T > 1.565$ , within 6 hours either:

- a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and  $F_r^T$  to within the limits of Figure 3.2-3b and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or
- b. Be in at least HOT STANDBY.

### SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2  $F_r^T$  shall be calculated by the expression  $F_r^T = F_r(1+T_q)$  and  $F_r^T$  shall be determined to be within its limit at the following intervals:

- a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,
- b. At least once per 31 days of accumulated operation in MODE 1, and
- c. Within four hours if the AZIMUTHAL POWER TILT ( $T_q$ ) is  $> 0.020$ .

4.2.3.3  $F_r$  shall be determined each time a calculation of  $F_r^T$  is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump Combination.

4.2.3.4  $T_q$  shall be determined each time a calculation of  $F_r^T$  is required and the value of  $T_q$  used to determine  $F_r^T$  shall be the measured value of  $T_q$ .

\*See Special Text Exception 3.10.2.

## POWER DISTRIBUTION LIMITS

### AZIMUTHAL POWER TILT - $T_q$

#### LIMITING CONDITION FOR OPERATION

3.2.4 The AZIMUTHAL POWER TILT ( $T_q$ ) shall not exceed 0.02.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER\*.

#### ACTION:

- a. With the indicated AZIMUTHAL POWER TILT determined to be  $> 0.02$  but  $\leq 0.10$ , either correct the power tilt within two hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the TOTAL PLANAR RADIAL PEAKING FACTOR ( $F_{xy}^T$ ) and the TOTAL INTEGRATED RADIAL PEAKING FACTOR ( $F_r^T$ ) are within the limits of Specifications 3.2.2 and 3.2.3.
- b. With the indicated AZIMUTHAL POWER TILT determined to be  $> 0.10$ , operation may proceed for up to 2 hours provided that the TOTAL INTEGRATED RADIAL PEAKING FACTOR ( $F_r^T$ ) and TOTAL PLANAR RADIAL PEAKING FACTOR ( $F_{xy}^T$ ) are within the limits of Specifications 3.2.2 and 3.2.3. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided the THERMAL POWER level is restricted to  $< 20\%$  of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

#### SURVEILLANCE REQUIREMENT

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit by:

- a. Calculating the tilt at least once per 7 days when the Channel High Deviation Alarm is OPERABLE,

\* See Special Test Exception 3.10.2.



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 99 TO DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1. INTRODUCTION

In Reference 1, Northeast Nuclear Energy Company (NNECO) submitted a license amendment request and the preliminary Reload Safety Analyses (RSA) in support of the Millstone Unit No. 2, Cycle 7 reload. The final Reload Safety Analysis was provided in Reference 2. As indicated in these submittals, the bases on which the Cycle 7 reload was analyzed were documented in a "Basic Safety Report" (BSR) (Ref. 3), and in the Cycle 6 Reload Safety Analysis (Reference 4). The BSR, as supplemented by Reference 5 serves as the reference fuel assembly and safety analysis report for the use of Westinghouse fuel at Millstone 2 (a Combustion Engineering plant). Reference 6 documents the NRC staff's review and acceptance of the BSR. The analysis and evaluation of the reload was accomplished using the methodology of Reference 7. This methodology was approved in Reference 8.

In Reference 1, NNECO informed the Staff that due to the elevated levels of radioactive iodine and other fission products identified during Cycle 6 operation, NNECO anticipated the discovery of a number of fuel assemblies with leaking fuel rods during the refueling outage for Cycle 7.

Since that time, NNECO performed fuel sipping identifying 16 fuel assemblies with failed fuel rods. In addition, visual examination revealed several fuel assemblies to have broken holddown springs. NNECO is replacing all leaking fuel assemblies with a combination of reconstituted and previously discharged fuel assemblies. These changes have necessitated a revised loading pattern (Reference 2) for Cycle 7 operation.

8507050075 850619  
PDR ADDCK 05000336  
P PDR

## 1.1 General Description

The Millstone 2 reactor core is comprised of 217 fuel assemblies. Each fuel assembly has a skeletal structure consisting of five (5) Zircaloy guide thimble tubes, nine (9) Inconel grids, a stainless steel bottom nozzle, and a stainless steel top nozzle. One hundred seventy-six fuel rods are arranged in the grids to form a 14x14 array. The fuel rods consist of slightly enriched uranium dioxide ceramic pellets contained in Zircaloy-4 tubing which is plugged and seal welded at the ends to encapsulate the fuel.

Nominal core design parameters utilized for Cycle 7 are as follows:

Core Power (MWt)	2,700
System Pressure (psia)	2,250
Reactor Coolant Flow (GPM)	350,000
Core Inlet Temperature (°F)	549
Average Linear Power Density (kw/ft)	6.065
(based on best estimate hot, densified core average stack height of 136.4 inches)	

The feed fuel for the Millstone 2, Cycle 7 core consists of twenty-four (24) zoned-enrichment interior feed assemblies, each containing sixty (60) fuel rods at 2.62 w/o U235 and one-hundred sixteen (116) fuel rods at 2.91 w/o U235, and forty-eight (48) zoned-enrichment peripheral assemblies, each containing sixty (60) fuel rods at 2.91 w/o U235 and one-hundred sixteen (116) fuel rods at 3.29 w/o U235. The zoned-enrichment assembly configuration contains 12 lower enrichment fuel pins around each of the five control rod water holes. The feed fuel will replace twenty (20) Combustion Engineering (CE) Batch A assemblies, one (1) CE Batch B assembly, and fifty-one (51) Westinghouse Batch F assemblies. An additional five (5) Westinghouse Batch F assemblies will be discharged from the end of Cycle 6, and will be replaced by five (5) Westinghouse Batch F assemblies which were removed from the core

at the end of Cycle 5. Due to fuel defects in Cycle 6, and subsequent symmetry considerations, fourteen (14) Westinghouse Batch G assemblies, seven (7) Westinghouse Batch F assemblies (these Batch F and G assemblies were removed from the core at the end of Cycle 5), and four (4) CE Batch A assemblies (discharged at the end of Cycle 1) are needed as well. As a result of fuel reconstitution, the fuel rods from seven (7) Westinghouse reload assemblies to be used in Cycle 7 have been placed in Combustion Engineering (CE) skeletons. Also, twenty-one (21) fuel rods have been replaced with stainless steel rods in Cycle 7. The twenty-one stainless steel rods are distributed among eleven (11) fuel assemblies, with the number of stainless steel rods in each of these assemblies ranging from one to five. A summary of the Cycle 7 fuel inventory is given in Table 1.

TABLE 1  
Millstone Unit 2 Cycle 7  
Core Loading

<u>Region</u>	<u>Type</u>	<u>Number of Assemblies</u>	<u>Initial Enrichment w/oU235</u>	<u>%Theoretical Density</u>	<u>BOC** Burnup Average (MWD/MTU)</u>
A	CE	4	1.93	95.0	15960
F1	<u>W</u>	4	2.70	94.5	25200
F2	<u>W</u>	5	3.30	94.9	22200
F2	<u>W*</u>	3	3.30	94.9	21560
G1	<u>W</u>	19	2.72	95.0	23470
G2	<u>W</u>	32	3.19	94.7	19290
G2	<u>W*</u>	4	3.19	94.7	9970
H1	<u>W</u>	30	2.73	95.2	13790
H2	<u>W</u>	44	3.22	94.8	9560
J1	<u>W</u>	24	2.62/2.91	95.2/95.1	0
J2	<u>W</u>	48	2.91/3.29	95.1/95.2	0

\*Westinghouse fuel reassembled using CE skeletons.

\*\*EOL Cycle 6 burnup assumed: 11,500 MWD/MTU.

## 2. FUEL SYSTEM DESIGN

The fuel system design for Millstone Unit 2, Cycle 6 is the same as that approved (Ref. 6) for Cycles 4, 5, and 6. That is, approval of the BSR constituted approval of the use of a mixed core of Combustion Engineering and Westinghouse fabricated fuel assemblies. The replacement of CE fuel with Westinghouse fuel at each reloading would eventually lead to a core with all Westinghouse fuel.

As described in Reference 2, the reload redesign utilizes a combination of reconstituted and previously discharged fuel assemblies to replace leaking fuel assemblies. Since this redesign uses previously approved fuel assembly types, and since the redesign and the reinserted CE assemblies will not receive greater than design exposure, the redesign is acceptable from the fuel system point of view.

At the end of Cycle 5, NNECO identified broken holddown springs on 15 fuel assemblies. Initial plans were to effect replacement of the broken holddown springs. The procedure developed was utilized successfully on one fuel assembly. However, NNECO decided that the irradiated fuel repair procedure involved a high risk with the potential for damaging fuel assemblies, particularly fuel pins, during the repair.

NNECO therefore reached the conclusion and provided supporting analysis (Ref. 10) that operation of Cycle 6 with 9 fuel assemblies, each with a single broken holddown spring, was acceptable and prudent. The analysis provided by NNECO characterizes the breaks to the holddown springs, provides justification that the breaks were caused by excessive vibratory motion during reactor operation, discusses fretting wear, loose parts, control rod jamming and the probability of multiple fractures, and concludes that operation of Cycle 6 with the 9 assemblies having broken holddown springs would be acceptable. This is primarily because the number of active turns of the springs is only slightly decreased by the types of breaks observed. Future new fuel would have newly designed springs. We found this acceptable.

Nine assemblies identified to have broken holddown springs at the end of Cycle 6 will be reloaded for Cycle 7 operation. In addition 4 assemblies from Cycle 5 needed to provide symmetry in the loading pattern and which have broken holddown springs will be utilized. We find this acceptable based upon the finding for Cycle 6 and the lack of any problems observed in operation of Cycle 6 with 9 fuel assemblies having broken holddown springs. No broken holddown springs were identified in Batch H fuel at the end of Cycle 6 operation. Batch H fuel had a new top nozzle design intended to eliminate the problem, which has proven to be the case for the one cycle of exposure of the 74 assemblies in Batch H.

### 3. NUCLEAR DESIGN

The nuclear design procedures and models used for the analysis of the Millstone Unit 2 Cycle 7 reload core (References 1 and 2) are the same as those used for Cycle 6. These are documented in the Millstone Unit 2 Basic Safety Report (BSR), (Reference 3) and have been approved (Reference 6) for the analysis of the Millstone Unit 2 core using Westinghouse reload fuel beginning with Cycle 4. In addition, the methods described in Reference 7 document the methodology used by Westinghouse for performing this as well as other reloads. This methodology was approved in Reference 8.

The physics analysis of the reload specifically included the zoned-enrichment fuel assemblies, the 21 stainless steel rods in reconstituted fuel assemblies, and the loading pattern of the various fuel types described in Section 1.1 above, in order to determine maximum linear heat rates achievable in normal operation, control rod worths for the shutdown margin evaluation, and the Cycle 7 kinetics characteristics for use in the accident evaluation. Also included in the analysis is substitution of full strength control rods for part strength control rods in the lead control element assembly (CEA) bank. This hardware change was implemented during the refueling outage. Because these calculations were performed with approved methods, they are acceptable.

In Reference 2, Table 2, the kinetics parameters for the Cycle 7 reload redesign are given. These are all within the current limits with a small exception in the



least negative above 30% power Doppler temperature coefficient and in the maximum delayed neutron fraction. Both of these parameters had the same values in Cycle 6. The conclusion there was that no reanalysis was necessary because the potential effects were small. This was found acceptable for Cycle 6 and continues to be acceptable for Cycle 7. Two accidents were reanalyzed for other reasons, and are discussed under Accident Analysis, Section 5.

The control rod worths and shutdown requirements for the Cycle 7 design are presented in Table 3 of Reference 2 and compared with previous Cycle 6 values. At EOC 7, the reactivity worth with all control rods inserted assuming the highest worth rod is stuck out of the core is 6.26% and assuming a 10% reduction to allow for uncertainty. The reactivity worth required for shutdown, including the contribution required to control the steamline break event at EOC 7, is 5.89%. Therefore, sufficient control rod worth is available to accommodate the reactivity effects of the steamline break at the worst time in core life allowing for the most reactive control rod stuck in the fully withdrawn position and also allowing for calculational uncertainties. We have reviewed the calculated control rod worths and the uncertainties in these worths based upon comparison of calculations with experiments presented in the BSR and in previous Westinghouse reports. On the basis of our review, we conclude that the NNECo's assessment of reactivity control is suitably conservative and that adequate negative reactivity worth has been provided by the control system to assure shutdown capability assuming the most reactive control rod is stuck in the fully withdrawn position.

#### 4. THERMAL-HYDRAULIC DESIGN

Millstone 2 Cycle 7 utilized the Basic Safety Report (Ref. 3) which was approved by the staff in Reference 6. The Basic Safety Report was also used as the basis for Cycle 4, 5, and 6 operation.

As discussed in the BSR, the Westinghouse fuel assemblies have been designed and shown through testing to be hydraulically compatible with all resident Millstone 2 fuel assemblies. The stainless steel rods in the reconstituted fuel assemblies were treated as heated rods in the THINC DNB analysis. This is conservative since it results in higher subchannel enthalpy predictions.

No significant variations in thermal margins result from the Cycle 7 reload. The Cycle 7 analysis takes a partial credit of 3.0% of the net conservatism which exists between convoluting and summing the uncertainties of various measured plant power parameters in terms of power. This partial credit was applied in previous cycles and its approval is discussed in more detail in the Cycle 4 Reload Safety Evaluation Report (Ref. 9); therefore, we find operation of Cycle 7 acceptable.

## 5. ACCIDENT ANALYSIS

As a result of the change to full strength CEAs in the lead CEA bank, the value of the ejected rod worth for the HFP ejected rod accident for Cycle 7 increased to 0.28%  $\Delta k/k$ . The licensee therefore provided a reanalysis of this event. The results show that the energy deposition increased from 171 cal/gm for the reference analysis to 185 cal/gm for the Cycle 7 analysis. This is below the criterion of 200 cal/gm established as a limit for this accident in the BSR, and is therefore acceptable.

The split enrichment fuel assembly design flattens the power peaking by placing slightly lower enrichment fuel pins around the large water holes in the fuel assembly. In order to assess the effect of this flattening on a limiting DNB event, the loss of flow accident for Cycle 7 was reanalyzed. The results show the MDNBR to be 1.30, which is acceptable.

In Reference 11, the licensee provided a reanalysis of the small break LOCA. This was done because there was an inconsistency between the Technical Specification requirement on axial shape index (ASI) and the ASI assumptions used in the approved small break LOCA analysis. The inconsistency was documented in Millstone Unit 2 Licensee Event Report 85-001-0.

The most negative ASI input to the approved analysis was an ASI of 0.14. The Technical Specifications allow the 100% power ASI to be no more negative than -0.10. If the 0.06 ASI uncertainty is introduced into the analysis, the most negative upper bound to ASI becomes -0.16, which is inconsistent with the -0.14 ASI input to the small break LOCA model. Reference 11 provides the results of a small break LOCA analysis which allowed the ASI value to be -0.16. The calculated peak clad temperature increased from 1971°F to 2035°F. This is below the acceptance criterion of 2200°F for the small break LOCA, and is therefore acceptable.

## 6. TECHNICAL SPECIFICATIONS

Technical Specification changes proposed by the licensee in Reference 1 and as clarified in reference 12 are acceptable as follows:

The main Technical Specification change proposed by the licensee trades range in radial peaking for more range in axial shape index (ASI). For monitoring of the power distribution with excore detectors, the maximum radial peak is specified in the Technical Specifications by a limit on the total planar radial peaking factor,  $F_{xy}$ . The maximum axial peak is specified by limits on the ASI. The product of the radial and axial peaks is the core peaking factor, which is proportional to the maximum peak linear heat rate. The maximum allowable peak linear heat rate in turn is limited as a result of the LOCA analysis to 15.6 kw/ft in normal operation of the powerplant. A decrease in the allowable value of  $F_{xy}$  can be offset with an increase in the allowable value of ASI without changing the limiting achievable linear heat rate.

The current Millstone Unit 2 Technical Specifications define an allowable ASI envelope for  $F_{xy} \leq 1.791$ . Also defined is a power derate curve if the  $F_{xy}$  limit cannot be met. The proposed Technical Specification change defines an expanded allowable ASI envelope and an appropriate power derate envelope if  $F_{xy} \leq 1.62$ . The licensee has indicated in Reference 1 that an analysis was performed to verify that the current envelope is unaffected by the hardware change in the

lead CEA bank. The licensee further indicated that an additional analysis was performed to verify this for the new derate envelope for  $F_{xy} \leq 1.62$ , and that the appropriate ASI envelope for this  $F_{xy}$  was calculated with the approved methods of Reference 3.

The licensee also proposed to delete the indexing parameters M and N. These parameters specify allowable power levels when less than all reactor coolant pumps are used and when excore detectors are used for monitoring and the  $F_{xy}$  limit is exceeded. Reference 12 contained a page which was inadvertently left out of the Reference 1 submittal. This page deleted reference to the indexing parameters M and N. These parameters had previously been deleted on another page submitted with Reference 1. An additional technical specification page was included with Reference 12 which corrected a typo contained in the Reference 1 submittal. The way the revised Specifications have been written makes the change administrative, and it is therefore acceptable.

Since the proposed Technical Specification changes were calculated and evaluated with approved methods, and since they do not alter the maximum peak linear heat rate achievable in normal operation of the powerplant, the changes are acceptable.

## 7. ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(9). Pursuant to 10 CFR §51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8. CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: June 19, 1985

Principal Contributor:

M. Dunnenfeld

8. REFERENCES

1. W. G. Council (NNECO), letter to J. R. Miller (NRC), with "Cycle Refueling-Preliminary Reload Safety Analysis," February 6, 1985.
2. W. G. Council (NNECO), letter to J. R. Miller (NRC), with "Cycle 7 Refueling-Reload Safety Analysis," June 5, 1985.
3. "Basic Safety Report," Westinghouse proprietary report for Millstone Unit 2, Docket Number 50-336, submitted via letter, W. G. Council (NU) to R. Reid (NRC), March 6, 1980.
4. W. G. Council (NNECO), letter to J. R. Miller (NRC), November 17, 1983.
5. W. G. Council (NNECO), letter to R. A. Clark, November 17, 1981.
6. L. S. Rubenstein (NRC), memorandum for T. M. Novak, "SER Input on Millstone Unit 2 BSR," February 16, 1982.
7. Bordelon, F. M. et.al., "Westinghouse Reload Safety Methodology", WCAP-9272, March 1978.
8. C. O. Thomas (NRC), letter to E. P. Rahe, Jr. (W), "Acceptance for Referencing of Licensing Topical Report WCAP-9272(P)/9273(NP)", May 28, 1985.
9. W. G. Council (NNECO), letter to R. A. Clark, June 3, 1980.
10. W. G. Council (NNECO), letter to J. R. Miller (NRC), December 1, 1983.
11. W. G. Council (NNECO), letter to J. R. Miller (NRC), April 11, 1985.
12. W. G. Council (NNECO), letter #B11569 to J. R. Miller (NRC), "Proposed Revisions to Technical Specifications", June 11, 1985.