NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

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February 6, 1985

Docket No. 50-336
B11437

Director of Nuclear Reactor Regulation
Attn: Mr. James R. Miller
Operating Reactors Branch #3
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References:
(1) W. G. Counsil letter to R. Reid, dated March 6, 1980.

(2) R. A. Clark letter to W. G. Counsil, dated June 22, 1981.

(3) R. A. Clark letter to W. G. Counsil, dated January 12, 1982.

(4) R. A. Clark letter to W. G. Counsil, dated February 18, 1982.

(5) W. G. Counsil letter to R. A. Clark, dated November 17, 1983.

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
Cycle 7 Refueling - Preliminary Reload Safety Analysis
Proposed Revisions to Technical Specifications

The preliminary Reload Safety Analysis, submitted in support of the Millstone Unit No. 2 Cycle 7 reload, is attached. This report presents preliminary information concerning the Cycle 7 reload. Coolant activity measurements in Cycle 6 indicate a potential for changes to the fuel inventory expected for Cycle 7. Due to the uncertainty in the fuel inventory, the final Cycle 7 Reload Safety Analysis cannot be performed until after the shutdown of Cycle 6 when the determination of the exact fuel inventory available for use in Cycle 7 is made. The purpose of the attached report is to provide a preliminary description of the expected characteristics of the Cycle 7 reload, and to provide the bases for all changes to the Technical Specifications anticipated at this time. The Reload Safety Analysis will be submitted after the Cycle 7 reload design has been completed.

In Reference (1), Northeast Nuclear Energy Company (NNECO) provided the NRC Staff with the Basic Safety Report (BSR). The BSR serves as the reference fuel assembly and safety analysis report for the use of Westinghouse fuel at Millstone Unit No. 2. References (2), (3), and (4) document the Staff's acceptability of this report. In Reference (5), NNECO presented the Staff with

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the Millstone Unit No. 2, Cycle 6, Reload Safety Analyses. The BSR, as supplemented by Reference (5), provides the basis against which the Cycle 7 reload has been preliminarily evaluated.

Cycle 7 operation will necessitate certain changes to the Plant Technical Specifications. Therefore, pursuant to 10 CFR 50.90, NNECO hereby proposes to amend its operating license, DPR-65, by incorporating the revisions identified in the Attachment into the Millstone Unit No. 2 Technical Specifications. These revisions reflect changes in Cycle 7 operating characteristics. The proposed changes to the Millstone Unit No. 2 Technical Specifications modify the allowable region of operation when the core power distribution is monitored by the Excore Detector Monitoring System. A new curve for the allowable thermal power vs axial shape index has been developed for the case when the total radial peaking factor (Fxy) is less than 1.62. This curve allows a wider range of operation than the current curve developed for the case when Fxy is less than 1.719. The lower value of Fxy allows for operation at a higher thermal power and a larger axial shape index. The change establishes two curves to be used when the core power distribution is monitored by the Excore Detector Monitoring System. The new curve applies for Fxy values less than or equal to 1.62 and the current curve applies for Fxy values less than or equal to 1.719. The parameter "N" has been removed from surveillance 9.2.1.2.c because the relationship between the allowable value of Fxy and power level is already included in the axial shape index monitoring tents.

The curve of allowable thermal power vs axial shape index is used to assure that peak linear heat rate assumed in the LOCA analysis is not exceeded. For Millstone Unit No. 2, the maximum allowable peak linear heat rate is 15.6 kw/ft.

The proposed changes trade range in radial peaking for more range in axial shape index. The maximum radial peak is specified in the Technical Specification as a limit on Fxy. The maximum axial peak is specified by limits on the axial shape index. The allowable axial shapes will still assure that the limit on maximum peak linear heat rate of 15.6 kw/ft. is met. The increase in the allowable value of the axial shape index will be offset by a decrease in the allowable value for Fxy without changing the design basis value for linear heat rate. Since the design basis value for linear heat rate is unchanged, safety analyses involving linear heat rate are not impacted by the change.

The changes also have no impact upon non-LOCA transients. The curve in Technical Specification 3.2.6 (Figure 3.2-4) provides the shapes to be input to all DNBR design basis analyses. Since the allowable thermal power vs axial shape index in the modified Technical Specification is still bounded by the curve in Technical Specification 3.2.6, the transients for which DNBR is a concern are unaffected by the change. Additional justification for the changes is included in the Attachment.

The attached proposed changes have been reviewed pursuant to 10 CFR 50.59 and have not been found to constitute an unreviewed safety question. The analyses discussed herein support this conclusion in that none of the criteria of 10 CFR 50.59(a)(2) are compromised. Specifically, the proposed changes do not impact the previously derived maximum allowable linear heat rate or other parameters which could adversely impact plant transient or accident analyses.

NNECO has reviewed the attached proposed license amendment pursuant to the requirements of 10 CFR 50.91(a) and has determined that the changes do not involve a significant hazards determination. The basis for this conclusion is that none of the criteria delineated in 10 CFR 50.92 have been compromised. That is, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated; or create the possibility of a new or different kind of accident from any accident previously evaluated; or involve a significant reduction in a margin of safety. A comparison of the contents of this amendment request with the list of examples of amendments in 48 FR 14870 not likely to involve significant hazards considerations reveals that example (iii) is applicable, in that the changes proposed are the result of a core reloading and no fuel assemblies significantly different from those found previously acceptable to the NRC for previous cores at Millstone Unit No. 2 are involved. No significant changes were made to the acceptance criteria for the Technical Specifications, the analytical methods used to demonstrate conformance with the Technical Specifications and regulations are not significantly changed, and the NRC has previously found such methods acceptable as documented in References (2) through (4). As described above, previously approved methods have been utilized to trade margin between axial shape index and total radial peaking factor to improve operational flexibility. It is also noted that none of the examples provided as amendments likely to involve significant hazards considerations are applicable to this proposal.

The Millstone Unit No. 2 Nuclear Review Board has reviewed and approved the attached proposed changes and concurs with the above determinations.

In accordance with the requirements of 10 CFR 50.91(b), a copy of this document is being provided to the State of Connecticut.

In accordance with 10 CFR 50.170.12(c) an amendment application fee of $150 is enclosed with this amendment request.

We remain available to assist the Staff by any means to facilitate the review of the attached proposed changes. It is requested that these changes be approved prior to startup from the upcoming refueling outage, currently estimated to occur during June, 1985. We anticipate submittal of the Final Reload Safety Analysis Report on or about May 15, 1985.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

W. G. Counsil

W. G. Counsil
Senior Vice President

C. F. Sears

By: C. F. Sears
Vice President

cc: Mr. Kevin McCarthy
Director, Radiation Control Unit
Department of Environmental Protection
State Office Building
Hartford, CT 06116

STATE OF CONNECTICUT)
) ss. Berlin
COUNTY OF HARTFORD)

Then personally appeared before me C. F. Sears, who being duly sworn, did state that he is Vice President of Northeast Nuclear Energy Company, a Licensee herein, that he is authorized to execute and file the foregoing information in the name and on behalf of the Licensee herein and that the statements contained in said information are true and correct to the best of his knowledge and belief.

Jennifer F. Powers
Notary Public
My Commission Expires March 31, 1990

Docket No. 50-336

Attachment

Millstone Nuclear Power Station, Unit No. 2

Cycle 7

- Cycle 7 Preliminary Reload Safety Analysis
- Justification of Technical Specification Revisions
- Proposed Revisions to Technical Specifications

February, 1985

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1.0 Introduction

1.1 Objectives

This report presents preliminary information concerning the core reload of the Millstone Nuclear Power Station Unit No. 2, Cycle 7. Coolant activity measurements in Cycle 6 indicate a potential for changes to the fuel inventory expected for Cycle 7. Due to the uncertainty in the fuel inventory, the Cycle 7 reload safety analysis will be performed after the shutdown of Cycle 6 and the determination of the exact fuel inventory available for use in Cycle 7 is made. The purpose of this report is to provide a preliminary description of the expected characteristics of the Cycle 7 reload, and to provide the bases for all changes to the Technical Specifications anticipated at this time. The Reload Safety Analysis Report will be submitted after the Cycle 7 reload design has been completed.

1.2 Fuel Inventory for Cycle 7

The feed fuel for the Millstone Unit No. 2, Cycle 7 core will consist of 24 split-enrichment interior feed assemblies, each containing 60 fuel rods at 2.6 w/o and 116 fuel rods at 2.9 w/o, and 48 split-enrichment peripheral feed assemblies, each containing 60 fuel rods at 2.9 w/o and 116 fuel rods at 3.3 w/o. The feed fuel will replace 20 Combustion Engineering (CE) Batch A assemblies, 1 CE Batch B assembly, and 51 Westinghouse Batch F assemblies which will be discharged from the core at the end of Cycle 6. An additional 5 Westinghouse Batch F assemblies will be discharged at the end of Cycle 6 and will be replaced by 5 Batch F assemblies which were removed from the core at the end of Cycle 5. Twenty-four CE Batch A assemblies from Cycle 1 will also be available for use in Cycle 7. Additional fuel inventory will be available from the reconstitution of fuel assemblies which were removed from the core at the end of Cycle 5 with indication of failed fuel rods (Ref. 1).

2.0 Cycle 7 Preliminary Physics Characteristics

2.1 Normal Inventory Reload Design

A core loading pattern was developed for the Cycle 7 reload based on the assumption of no fuel failure in Cycle 6. The parameters which have been historically the most limiting for Millstone Unit No. 2 were analyzed using this loading pattern, namely, the Moderator Temperature Coefficient (MTC), radial peaking factor, and the available Shutdown Margin (SDM) at the most limiting condition during the cycle. The current limits on MTC, radial peaking factor, and SDM were met with this reload design. Table 1 provides the control rod worths and requirements at the most limiting condition during the cycle.

2.2 Potential Impact of Inventory Changes

In order to provide early identification of potential problem areas in the Cycle 7 reload design, loading pattern scoping studies were performed based on three different assumed fuel inventory scenarios. The basic assumption of the scoping studies was that the extent of fuel damage in Cycle 6 is similar to that of Cycle 5, as indicated by the coolant activity measurements. The three scenarios assumed different distributions of the fuel failure in Cycle 6 in order to simulate the effects of a variety of possible reactivity and burnup distributions. Loading patterns were established for each of the three scenarios, and the MTC, radial peaking factor (Fr), total planar radial peaking factor (Fxy), and SDM were analyzed. In comparison with past behavior of Millstone Unit No. 2, the parameter which was most affected by the atypical fuel inventories of the scoping studies was the planar radial peaking factor. The best estimate Fxy calculated in the scoping studies showed an increase of approximately 3 percent over the values determined in previous cycles. Based on the results of these three studies, a Technical Specification change on Fxy is not anticipated; however, the Cycle 7 loading pattern based on the actual fuel inventory may necessitate changes.

3.0 Justification for Technical Specification Changes

Analyses were performed in order to justify Technical Specification changes proposed at the present time for Millstone Unit No. 2, Cycle 7. The results of these analyses are presented here.

3.1 Hardware Change to Lead CEA Bank

In anticipation of hardware changes to be made to the lead Control Element Assembly (CEA) bank during the plant outage following the Cycle 6 shutdown, an evaluation was performed to determine the impact of the hardware changes on the current Technical Specifications and safety analysis inputs. The part-strength control rods in the lead CEA bank are to be replaced by full-strength control rods, making the lead bank CEAs identical in composition to the remainder of the CEA banks.

Parameters which are input to the safety analysis and which would be affected by the proposed hardware change were analyzed. These parameters are radial peaking factor, dropped rod peaking factor, ejected rod worth and peaking factor, and shutdown margin. The only input parameter to the safety analysis which exceeded the current limit established by the Basic Safety Report (BSR) (Ref. 2) and the Cycle 6 reload safety analysis (Ref. 3) was the HFP ejected rod worth.

The ejected rod accident at HFP was reanalyzed using the methodology described in Reference 4. The parameters used in the analysis are given in Reference 5, with the exception of the ejected rod worth and the reactor coolant flow. The reactor coolant flow used is given in the Cycle 6 reload safety analysis (Ref. 3). The value of the ejected rod worth used in this analysis was 0.28% . The results of this analysis are given in Table 2. The sequence of events for this accident is given in Table 3. The nuclear power transient and the hot spot fuel and clad temperature transients are shown in Figures 1 through 4. The results demonstrate that the limiting criteria given in Reference 2 for this accident are not exceeded. The average enthalpy of the hottest fuel pellet does not exceed the damage threshold of 200 cal/gm.

An analysis was performed in order to verify the applicability of the current Technical Specifications on the Axial Shape Index alarm setpoints for operation using the excore detector monitoring system, the fuel centerline melt trip, the Limiting Condition for Operation (LCO) on ASI for Departure from Nucleate Boiling (DNB), and the Thermal Margin/Low Pressure (TM/LP) trip. The Condition I and II power shapes were analyzed using the methodology described in Reference 4 with the proposed rod configuration. The results of these analyses showed that the proposed hardware change does not require changes to these Technical Specifications.

3.2 Total Planar Radial Peaking Factor and Linear Heat Rate Monitoring

The Axial Shape Index (ASI) envelope currently used to monitor the linear heat rate while operating on excore detectors was generated using the power-dependent radial peaking factor (Fxy) given in Technical

Specification Figure 3.2-3 in accordance with the methodology described in Reference 2. The maximum Fxy allowed by the current Technical Specification is 1.719 at full power. The current envelope was verified to be unaffected by the proposed hardware change to the lead CEA bank as described in Section 3.1. An additional analysis was performed using the power-dependent Fxy shown in Figure 5, with a maximum of 1.62 at full power. The lead CEA bank hardware change was accounted for in the additional analysis. Given the Fxy relationship of Figure 5, the power shape analysis showed that it is acceptable to expand the ASI operating envelope to the limits shown in Figure 6. It is proposed that the surveillance requirements on the linear heat rate and the Technical Specification on the Total Planar Radial Peaking Factor (Fxy) be modified as given in the Appendix to allow the use of the wider operating range when Fxy is ≤ 1.62.

4.0 References

1. Osborne, D., "Meeting Summary", Notes of NUSCO/NRC Meeting of October 3, 1984 at Bethesda, MD, October 26, 1984.

2. Millstone Unit No. 2, "Millstone Unit No. 2 Basic Safety Report", Docket No. 50-336, March, 1980.

3. W. G. Counsil letter to J. R. Miller, dated November 4, 1983.

4. Bordelon, F. M., et. al., "Westinghouse Reload Safety Evaluation Methodology", WCAP-9273, March, 1978.

5. W. G. Counsil letter to R. A. Clark, dated November 17, 1981.

TABLE 1

SHUTDOWN REQUIREMENTS AND MARGINS

MILLSTONE UNIT 2 - CYCLE 7

NORMAL FUEL INVENTORY ASSUMED

Control Rod Worth (%Δρ)	EOC 7
All Rods Inserted	8.36
All Rods Inserted Less Worst Stuck Rod	6.86
(1) Less 10%	6.17
Control Rod Requirements	
Reactivity Defects (combined Doppler, Tavg, Void, and Redistribution Effects)	2.58
Rod Insertion Allowance	0.42
(2) Total Requirements	3.00
Shutdown Margin ((1)-(2)) (%Δρ)	3.17
Required Shutdown Margin (%Δρ)	2.90

TABLE 2

RESULTS OF THE CEA EJECTION ACCIDENT ANALYSIS

	HFP
Max. fuel pellet average temperature, °F	4228
Max. fuel center temperature, °F	5011
Max. clad average temperature, °F	2465
Max. fuel pellet center melting, percent	8.44
Max. fuel stored energy, cal/gm	186

TABLE 3

SEQUENCE OF EVENTS, CEA EJECTION INCIDENT

HFP Case

Time	Event	Setpoint or Value
0.0	Initiation of Transient	-
0.04	High Power Trip Signal Generated	112 percent
0.1	CEA Fully Ejected	-
0.13	Peak Nuclear Flux Reached	See Fig. 1
0.94	CEA Insertion Begins	-
0.36	Peak Fuel Temperature Reached	See Fig. 2
3.54	CEA's Reach 90 percent Insertion	-

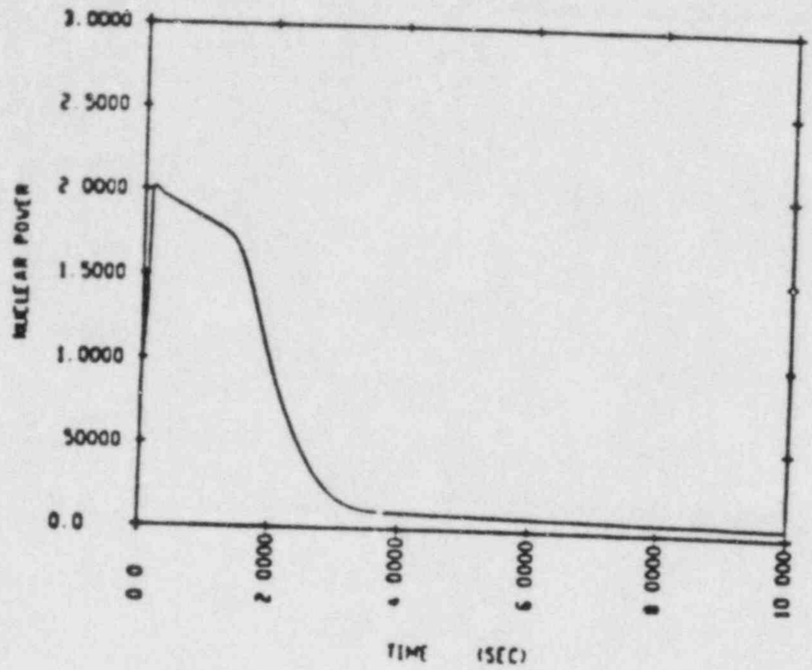


Figure 1 Rod Ejection - HFP
Nuclear Power versus Time

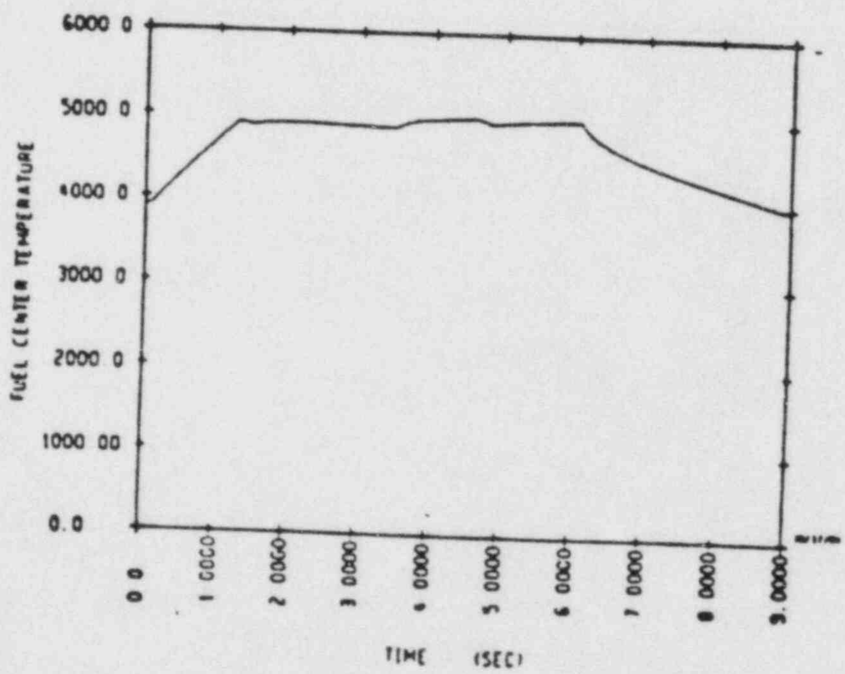


Figure 2 Rod Ejection - HFP
Fuel Center Temperature versus Time

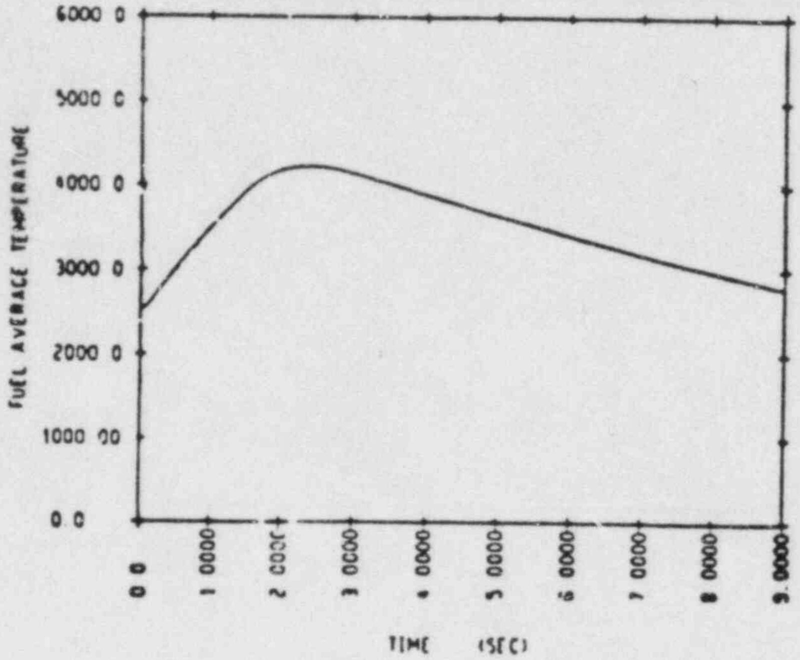


Figure 3 Rod Ejection - HFP
Fuel Average Temperature versus Time

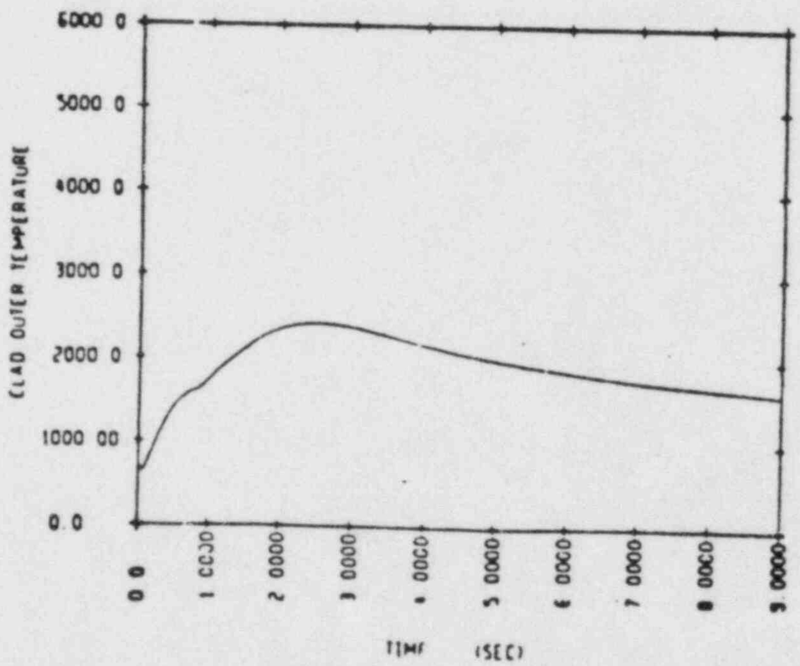
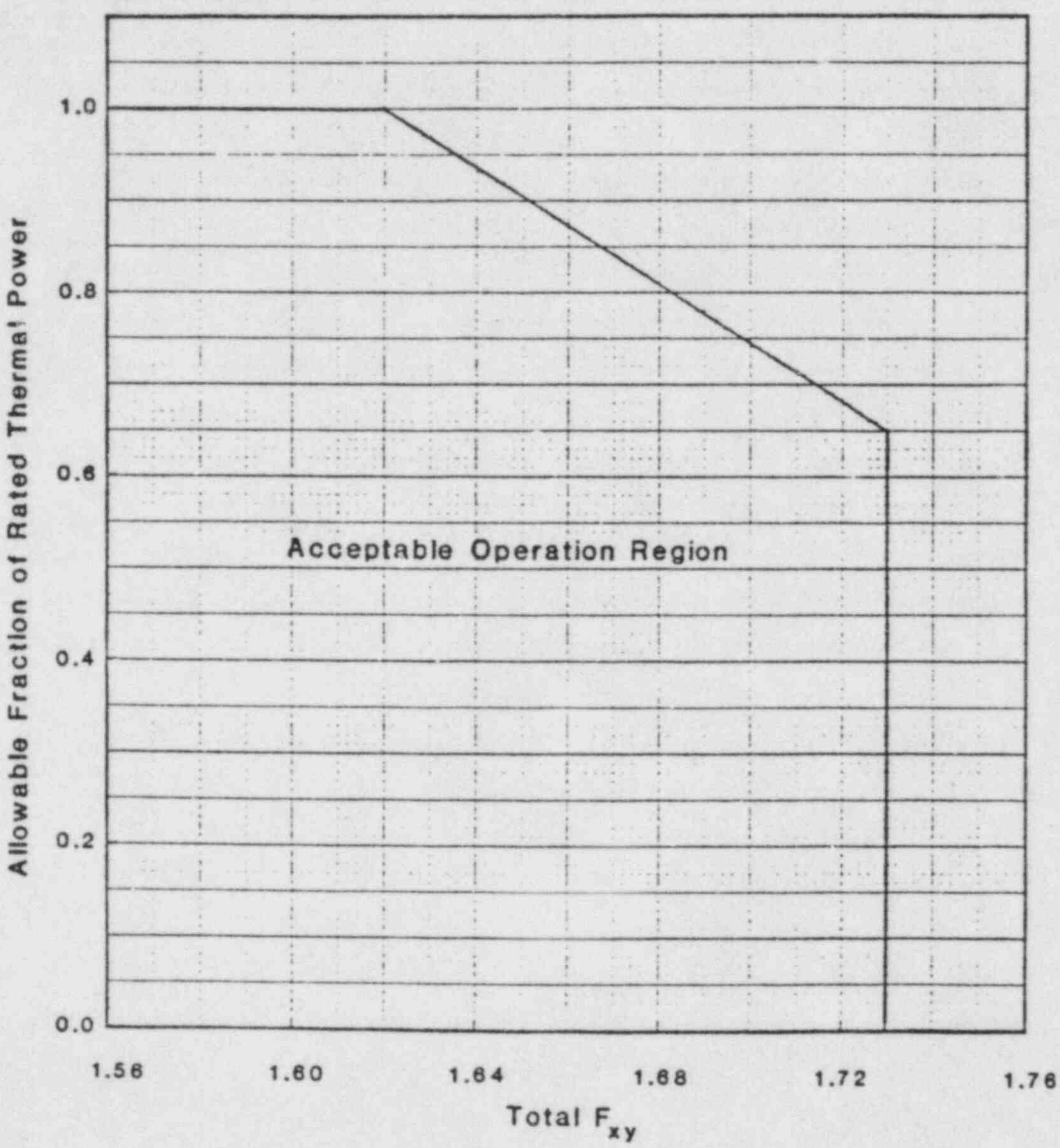


Figure 4 Rod Ejection - HFP
Clad Temperature versus Time

FIGURE 5 Total Radial Peaking Factor vs Allowable Fraction of Rated Thermal Power



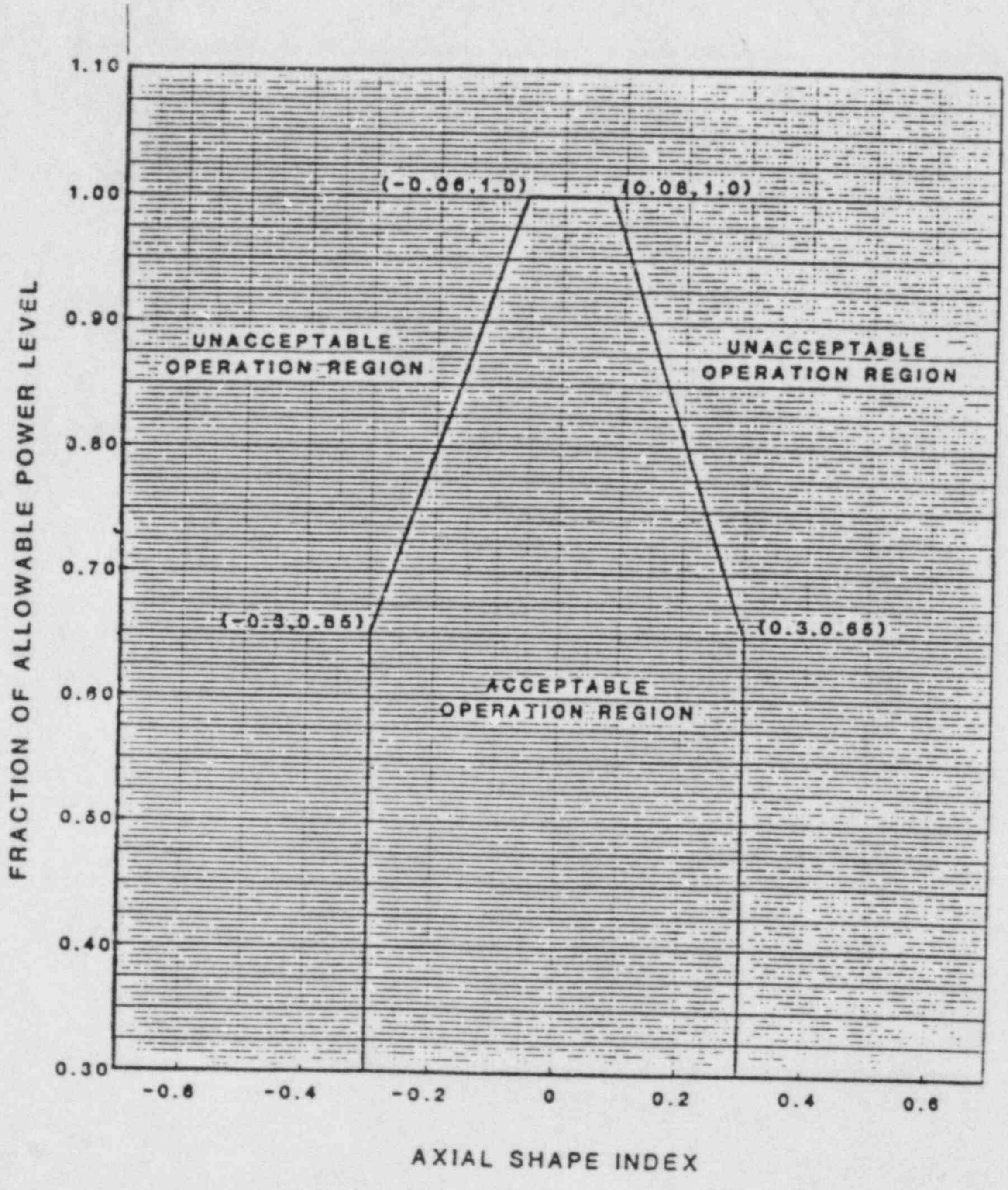


FIGURE 6 Axial Shape Index vs Fraction of Allowable Power Level per Specification 4.2.1.2c