YAEC-1324

MAINE YANKEE

CYCLE 7

CORE PERFORMANCE ANALYSIS

September, 1982

Major Contributors:

Reactor Physics Group

G. M. Solan
A. S. DiGiovine
S. VanVolkinburg

Transient Analysis Group

P. J. Guimond
M. P. LeFrancois
K. R. Rousseau

Nuclear Evaluations and Support Group

K. E. St. John
D. K. Beller

LOCA Analysis Group

W. J. Szymczak
R. C. Harvey

Fuel Management Group

E. C. Biemiller

Yankee Atomic Electric Company
Nuclear Services Division
1671 Worcester Road
Framingham, Massachusetts 01701

APPROVALS

Approved By: *P.A. Bergeron* — *9/14/82* (Date)

P. A. Bergeron, Manager
Transient Analysis Group

Approved By: *R.J. Cacciapouti* — *9/14/82* (Date)

R. J. Cacciapouti, Manager
Reactor Physics Group

Approved By: *Ausaf Husain* — *9/14/82* (Date)

A. Husain, Manager
LOCA Analysis Group

Approved By: *Stephen P. Schultz* — *9/14/82* (Date)

S. P. Schultz, Manager
Nuclear Evaluations and Support Group

Approved By: *E. E. Pilat* — *9/14/82* (Date)

E. E. Pilat, Manager
Fuel Management Group

Approved By: *Bruce C. Slifer* — *9/15/82* (Date)

B. C. Slifer, Manager
Nuclear Engineering Department

DISCLAIMER

This document was prepared by Yankee Atomic Electric Company ("Yankee") pursuant to a contract between Yankee and Maine Yankee Atomic Power Company ("Maine Yankee"). The use of information contained in this document by anyone other than Maine Yankee, or for any purpose other than for which it is intended, is not authorized, and with respect to any unauthorized use, neither Yankee nor its officers, directors, agents, or employees assume any obligation, responsibility, or liability, or makes any warranty or representation concerning the contents of this document or its accuracy or completeness.

ABSTRACT

This report presents design and calculational analysis results pertinent to the operation of Cycle 7 of the Maine Yankee Atomic Power Station. These include core fuel loading, fuel description, reactor power distributions, control rod worths, reactivity coefficients, the results of the safety analyses performed to justify plant operation, the startup test program and the Reactor Protective System (RPS) setpoints assumed in the safety analysis. The analysis results, in conjunction with the startup test results, RPS setpoints and Technical Specifications, serve as the basis for ensuring safe operation of Maine Yankee during Cycle 7.

ACKNOWLEDGEMENTS

The authors would like to express their gratitude to E. J. DeCouteau, J. Pappas, J. P. Gorski, and W. J. Waterman for their assistance, and to the Word Processing Center staff for typing the report.

TABLE OF CONTENTS

		Page
	APPROVALS	ii
	DISCLAIMER	iii
	ABSTRACT	iv
	ACKNOWLEDGEMENTS	v
	LIST OF TABLES	x
	LIST OF FIGURES	xii
1.0	INTRODUCTION	1
2.0	OPERATING HISTORY	3
2.1	Cycles 1 and 1A	3
2.2	Cycle 2	3
2.3	Cycles 3 and 4	3
2.4	Cycles 5 and 6	4
3.0	RELOAD CORE DESIGN	7
3.1	General Description	7
3.1.1	Core Fuel Loading	7
3.1.2	Core Burnable Poison Loading	7
3.1.3	Core Loading Pattern	8
3.1.4	Assembly Exposure History	8
3.1.5	CEA Group Configuration	9
3.2	Fuel System Design	10
3.2.1	Fuel Mechanical Design	10
3.2.2	Thermal Design	12
3.2.3	Thermal-Hydraulic Design	14
3.3	Control Element Assembly Design	17
4.0	PHYSICS ANALYSIS	37
4.1	Fuel Management	37
4.2	Core Physics Characteristics	37
4.3	Power Distributions	37
4.4	CEA Group Reactivity Worths	39
4.5	Doppler Reactivity Coefficients and Defects	39
4.6	Moderator Reactivity Coefficients and Defects	39
4.7	Soluble Boron and Burnable Poison Reactivity Effects	40
4.8	Kinetics Parameters	40
4.9	Safety-Related Characteristics	41
4.9.1	CEA Group Insertion Limits	41
4.9.2	CEA Ejection Results	41

TABLE OF CONTENTS
(continued)

Page

4.9.3 CEA Drop Results.................................... 41

4.9.3.1 Design Analysis Results.................... 41
4.9.3.2 Post-CEA Drop Restrictions................. 42

4.9.4 Available Scram Reactivity............................ 43
4.9.5 Shutdown Margin Requirements.......................... 45
4.9.6 Augmentation Factors.................................. 47

4.10 Methodology and Methodology Revisions...................... 47

4.10.1 Summary of Physics Methodology Documentation....... 47
4.10.2 CEA Ejection Analysis Physics Input................ 47

4.10.2.1 Background and Introduction................ 47
4.10.2.2 Present Licensing Method................... 48
4.10.2.3 Changes to Licensing Method................ 49

4.10.2.3.1 Core Average Doppler Defect Calculation........... 50
4.10.2.3.2 Core Average Doppler Defect Uncertainty........... 50

5.0 SAFETY ANALYSIS REVIEW.. 92

5.1 General.. 92

5.1.1 Initial Operating Conditions.......................... 92
5.1.2 Core Power Distributions.............................. 93
5.1.3 Reactivity Coefficients............................... 94
5.1.4 Shutdown CEA Characteristics.......................... 95
5.1.5 Reactor Protective System Setpoints and Time Delays.................................. 96

5.2 Summary.. 98

5.3 Anticipated Operational Occurrences for Which the RPS Assures No Violation of SAFDLs....................... 99

5.3.1 Control Element Assembly Withdrawal.................. 99
5.3.2 Boron Dilution.. 100

5.3.2.1 Dilution During Refueling.................. 101
5.3.2.2 Dilution During Startup.................... 102
5.3.2.3 Dilution During Hot and Cold Shutdown...... 102
5.3.2.4 Dilution During Drained RCS Conditions..... 103

TABLE OF CONTENTS
(continued)

		Page
5.3.2.5	Dilution During Hot Standby, Critical, and Power Operation	104
5.3.2.6	Failure to Borate Prior to Cooldown	104
5.3.3	Excess Load Incident	105
5.3.4	Loss of Load Incident	106
5.3.5	Loss of Feedwater Incident	106
5.4	Anticipated Operational Occurrences Which are Dependent on Initial Overpower Margin for Protection Against Violation of SAFDLs	107
5.4.1	Loss-of-Coolant Flow	107
5.4.2	Full Length CEA Drop	108
5.5	Postulated Accidents	109
5.5.1	Steam Line Rupture	109
5.5.2	Steam Generator Tube Rupture	111
5.5.3	Seized Rotor Accident	111
5.5.4	CEA Ejection	112
5.5.5	Loss of Coolant	113
5.5.5.1	Introduction and Summary	113
5.5.5.2	Large Break LOCA Analysis	113
5.5.5.2.1	Break Spectrum Analysis	114
5.5.5.2.2	Burnup Sensitivity Studies	115
5.5.5.2.3	Cosine Axial Power Distribution Study	115
5.5.5.3	Small Break LOCA Analysis	116
5.6	Methodology Revisions	117
5.6.1	YAEC-1 DNB Correlation	117
5.7	Plant Hardware Modifications	118
5.7.1	Placement of CEAs in Former Part-Length CEA Locations	118
6.0	RPS SETPOINTS	137
6.1	General	137
6.2	Thermal Margin/Low Pressure Trip	138
6.3	Symmetric Offset Trip	139
6.4	Limiting Conditions for Operation	141

TABLE OF CONTENTS
(continued)

		Page
6.4.1	CEA Drop Symmetric Offset Limits	141
6.4.2	Loss-of-Coolant Flow Symmetric Offset Limits	142
7.0	STARTUP TEST PROGRAM	146
7.1	Low Power Physics Tests	146
7.2	Power Escalation Tests	147
7.3	Acceptance Criteria	147
8.0	CONCLUSIONS	150
9.0	REFERENCES	151

LIST OF TABLES

Number	Title	Page
2.1	Operating History Summary	5
2.2	Fuel Assembly Types by Cycle	6
3.1	Cycle 7 Assembly Description	18
3.2	Cycle 7 Core Loading	19
3.3	Mechanical Design Features of Cycle 7 Fuel	20
3.4	Centerline and UO_2 Melt Temperature Comparison	21
3.5	Cycle 7 Ratio of Maximum Radial Relative Pin Powers - Maximum in Type J Fuel to Maximum in Core	22
3.6	Cycle 7 Determination of ENC Rod Bow MDNBR Penalty	23
3.7	Cycle 7 Thermal-Hydraulic Parameters at Full Power	24
4.1	Cycles 3, 6, and 7 Nuclear Characteristics	53
4.2	Cycle 7 Maximum Radial Relative Pin Power-Design and As-Built Shim Loadings	54
4.3	Cycles 3, 6, and 7 CEA Group Worths at HFP	55
4.4	Cycles 3, 6 and 7 Core Average Doppler Defect	56
4.5	Cycles 3, 6 and 7 Core Average Doppler Coefficient	57
4.6	Cycle 7 Core Average Doppler Defect with Pre-Ejected Weighting for Full Power CEA Ejection	58
4.7	Cycle 7 Core Average Doppler Defect with Pre-Ejected Weighting for Zero Power CEA Ejection	59
4.8	Cycles 3, 6 and 7 Moderator Temperature Coefficients	60
4.9	Cycles 3, 6 and 7 ARI Moderator Defect with Worst Stuck CEA	61
4.10	Cycles 3 and 7 Kinetics Parameters	62
4.11	Cycle 7 CEA Ejection Results	63
4.12	Cycles 6 and 7 CEA Drop Results at BOC	64
4.13	Cycles 6 and 7 CEA Drop Results at EOC	65
4.14	Cycle 7 Available Scram Reactivity	66

LIST OF TABLES
(continued)

Number	Title	Page
4.15	Cycle 7 Required Scram Reactivity	67
4.16	Cycles 6 and 7 Augmentation Factors	68
4.17	Cycles 6 and 7 Core Radial Pin Power Census for Augmentation Factor Calculation	69
4.18	Cycles 6 and 7 Quarter Assembly Power Census for Augmentation Factor Calculation	70
4.19	Physics Methodology Documentation	71
4.20	Comparison of Full Power CEA Ejection Nominal Physics Parameters	72
4.21	Comparison of Zero Power CEA Ejection Nominal Physics Parameters	73
4.22	Comparison of Doppler Weighting Factors for Various Control Rod Ejection Analyses	74
5.1	Cycle 7 Safety Parameters	119
5.2	Cycle Safety Analysis - Incidents Considered	123
5.3	Cycle 7 Safety Analysis - Summary of Results	124
5.4	Cycle 7 Required Initial Boron Concentrations to Allow Fifteen Minutes Margin to Criticality for Dilutions from Shutdown Conditions with the RCS Filled	127
5.5	Cycle 7 Required Initial Boron Concentrations to Allow Thirty Minutes Margin to Criticality for Dilutions from Shutdown Conditions with the RCS Drained	128
5.6	Summary of Results for Cycle 7 Boron Dilution Events	129
5.7	Cycle 7 Full Power CEA Ejection Accident Results	130
5.8	Cycle 7 Zero Power CEA Ejection Accident Results	131
5.9	Cycle 7 Burnup Sensitivity Results	132
5.10	Cycle 7 Cosine Axial Power Shape Results	133
5.11	Comparison of Thermal Margin for Limiting Cycle 7 Power Distributions to FSAR Design Power Distribution	134
7.1	Cycle 7 Startup Test Acceptance Criteria	149

LIST OF FIGURES

Number	Title	Page
3.1	Cycle 7 Burnable Poison Shim Assembly Locations	25
3.2	Cycle 7 Assembly Loading Pattern	26
3.3	Cycle 7 Calculated Assembly Exposures at BOC	27
3.4	Cycle 6 Burnup Distribution by Assembly at 10,000 MWD/MT	28
3.5	Cycle 7 CEA Group Locations	29
3.6	Cycles 1-6 and Cycle 7 CEA Group 5 Configurations	30
3.7	Centerline Temperature Vs. LHGR at BOC for Type J Fuel	31
3.8	Centerline Temperature Vs. LHGR at BOC for Type K Fuel	32
3.9	Centerline Temperature Vs. LHGR at BOC for Type L Fuel	33
3.10	Centerline Temperature Vs. LHGR at EOC for Type J Fuel	34
3.11	Centerline Temperature Vs. LHGR at EOC for Type K Fuel	35
3.12	Centerline Temperature Vs. LHGR at EOC for Type L Fuel	36
4.1	Cycle 7 Assembly Relative Power Densities BOC (50 MWD/MT), HFP, ARO	75
4.2	Cycle 7 Assembly Relative Power Densities MOC (6,000 MWD/MT), HFP, ARO	76
4.3	Cycle 7 Assembly Relative Power Densities EOC (12,000 MWD/MT), HFP, ARO	77
4.4	Cycle 7 Assembly Relative Power Densities BOC (50 MWD/MT), HFP, CEA Bank 5 Inserted	78
4.5	Cycle 7 Assembly Relative Power Densities MOC (6,000 MWD/MT), HFP, CEA Bank 5 Inserted	79
4.6	Cycle 7 Assembly Relative Power Densities EOC (12,000 MWD/MT), HFP, CEA Bank 5 Inserted	80
4.7	Cycle 7 Allowable Unrodded Radial Peak Vs. Cycle Average Burnup	81
4.8	Cycle 7 Arrangement of Shim Loadings - As-Built Calculation	82
4.9	Cycle 7 Power Dependent Insertion Limit (PDIL) for CEAs	83

LIST OF FIGURES
(continued)

Number	Title	Page
4.10	Cycles 6 and 7 Maximum Radial Peaking Vs. Worth for Dropped CEAs	84
4.11	Cycle 7 Dropped CEA Radial Peaking with Power Restriction - ARO plus Dropped A at BOC	85
4.12	Cycle 7 Dropped CEA Radial Peaking with Power Restriction - ARO plus Dropped B at BOC	86
4.13	Cycle 7 Dropped CEA Radial Peaking with Power Restriction - ARO plus Dropped A at EOC	87
4.14	Cycle 7 Dropped CEA Radial Peaking with Power Restriction - ARO plus Dropped B at EOC	88
4.15	Cycle 7 Shutdown Margin Equation and Required Scram Reactivity	89
4.16	Cycle 7 Required Shutdown Margin Vs. RCS Boron Concentration	90
4.17	Cycle 7 Limiting Single Gap Power Peaking	91
5.1	Cycle 7 Allowable 3 Loop Steady-State Coolant Conditions	135
5.2	Design Power Distributions	136
6.1	Cycle 7 Thermal Margin/Low Pressure Trip Setpoint Part 1, (YI Versus A1)	143
6.2	Cycle 7 Thermal Margin/Low Pressure Trip Setpoint Part 2, (Fraction of Rated Thermal Power Versus QR1)	144
6.3	Cycle 7 Symmetric Offset Function, Three Pump Operation	145

1.0 INTRODUCTION

This report provides justification for the operation of Maine Yankee during the next fuel cycle, Cycle 7. The Cycle 7 refueling will involve the discharge of 73 assemblies and the insertion of 72 new fuel assemblies and one burned Type E assembly from Core 2. The new fuel assemblies (designated Batch L) are being provided by Exxon Nuclear (ENC) and are similar in design to the ENC fuel remaining from Cycles 5 and 6. Several minor design changes associated with the Type L fuel are discussed in Section 3.2.1.

The ENC fuel designs are similar but not identical to the single CE fuel assembly inserted in the core center from Cycle 2. Small differences exist in both the mechanical and hydraulic characteristics. Important differences in the mechanical design are discussed in Section 3.2.1 of (33). Differences in the hydraulic characteristics are covered in Section 3.2.3.

The proposed operating conditions for Cycle 7 are: rated core thermal power 2630 MWt, a maximum steady-state operating pressure of 2275 psia, and a maximum indicated core inlet temperature of 550°F. In addition, operation is allowed over a pressure range from 2075 psia to 2225 psia by imposing a limit on the maximum core inlet temperature at the lower pressures to preserve the DNB margin. This assures that DNB performance is the same for all possible limiting temperature and pressure combinations. These conditions are consistent with the "Stretch Power" conditions proposed in (1) and currently allowed by the Maine Yankee operating license.

Since the core power level and core inlet temperature were outside the bounds considered in the FSAR (2), the full spectrum of transients and accidents were re-analyzed in (3) as part of the 2630 MWt stretch power proposal (1). The staff approved operation at 2630 MWt in (4) based on the analysis in (3). The ACRS also reviewed the stretch power application (1) and concurred with the Staff's finding that Maine Yankee can be safely operated at 2630 MWt (5).

The analysis in (3), therefore, now serves as the "Reference Safety Analysis" for Maine Yankee. This analysis and the analysis reported in (6), justifying operation with a slight positive Moderator Temperature Coefficient

(MTC), the analyses submitted in (7), justifying Cycle 4 operation, (33), justifying Cycle 5 operation, and (37) and (38), justifying Cycle 6 operation, will be used herein to demonstrate continued safe operation at power levels up to 2630 MWt through Cycle 7.

This report contains sections dealing with the fuel mechanical, thermal-hydraulic, physics and safety analysis aspects of the operation of Cycle 7. An evaluation of ECCS performance, setpoints for the Thermal Margin/Low Pressure (TM/LP) and Symmetric Offset Trips, and a description of the Startup Test Program are included. Except as noted, the methods used in these analyses are in accordance with those described in (8-16) and (35). These methods have been approved by the NRC for use on Maine Yankee in (17) and (18). Methods used in safety-related analyses for the fuel mechanical design evaluations are based on the Exxon Nuclear generic models which have received prior approval by the NRC.

Unique features of Cycle 7 are:

1. Low-leakage core design (Sections 3.1 and 4.1).

2. Installation of additional Control Element Assemblies and reconfiguration of CEA Bank 5 (Section 3.1.5).

3. Modifications to the reactor physics methodology for the CEA Ejection Analysis (Section 4.10.2).

4. First application of the YAEC-1 CHF correlation in determining LSSS (Section 5.6).

Details of each change are provided in the sections indicated.

2.0 OPERATING HISTORY

The operating history of Maine Yankee has consisted of seven cycles designated as 1, 1A, and 2 through 6. The significant operating conditions and durations of the cycles are defined in Table 2.1. The fuel assembly types loaded by cycle are given in Table 2.2.

2.1 Cycles 1 and 1A

The initial Maine Yankee core consisted of unpressurized, low density fuel designated as Core 1 design fuel assemblies. Cycle 1 operation was restricted and terminated due to leaking fuel assemblies.

Cycle 1A consisted of operation after the leaking fuel assemblies from the initial core were replaced with fresh fuel designated as Replacement Fuel (RF) assemblies. The mechanical design of the RF assemblies was essentially the same as Core 1 design fuel. The significant difference in the design was the pressurization of the fuel rod with helium sufficient to prevent creep collapse of the fuel rod cladding and improve gap heat transfer. The replacement fuel assemblies performed successfully during Cycle 1A.

2.2 Cycle 2

Cycle 2 consisted entirely of fresh assemblies designated as Core 2 design fuel. Mechanical design changes were made in comparison to the Core 1 design fuel. These comprised pre-pressurization, higher fuel density, and smaller diameter pellets. A detailed discussion of the design changes was provided in (30). The Core 2 design fuel performed successfully. Subsequent to Cycle 2 operation, burnable poison shim failures were discovered in the Type E assemblies. Corrective action consisted of replacement of all Type E shims with water-filled zircaloy rods prior to reinsertion in subsequent cycles.

2.3 Cycles 3 and 4

Cycle 3 consisted of fresh fuel assemblies of the Core 2 design and Replacement Fuel assemblies reinserted from Cycle 1A. The performance integrity of the Cycle 3 fuel had been demonstrated through irradiation in Cycles 2 and 1A, respectively. All fuel performed successfully during Cycle 3.

Cycle 4 consisted of all fuel assemblies of the Core 2 design. Slight design changes to the fresh Type I fuel were made and discussed in Section 3.2.1 of (7). New fuel and once-burned fuel assemblies from Cycle 2 were inserted and the replacement fuel discharged. A small number of leaking fuel assemblies were discovered near end-of-cycle.

2.4 Cycles 5 and 6

Cycles 5 and 6 consisted of fuel assemblies of the Core 2 CE design and fresh assemblies designed by ENC. A detailed discussion of the ENC design assemblies was provided in (33). Five Core 2 design leaking assemblies returned to the core in Cycle 5 were repaired by replacement of fuel rods with fresh, low enrichment Core 2 design fuel (34 rods) or water-filled zircaloy rods (10 rods). The fuel has performed successfully during Cycles 5 and 6.

TABLE 2.1

MAINE YANKEE OPERATING HISTORY SUMMARY

Cycle	Date of Power Escalation	Core Power Level Licensed (MWt)	Core Power Level Operated (%)	Cycle Burnup (MWD/MT)
1	11/3/72	2440	50-80[(1)]	10367
1A	10/12/74	2440	80[(1)]	4500
2	6/29/75	2440	100	17395
3	6/17/77	2630[(2)]	93	11075
4	8/23/78	2630	97[(3)]	10496
5	3/17/80	2630	97	10796
6	7/20/81	2630	97	11500[(4)]

(1) Power decrease and primary system pressure decrease to 1800-2000 psia due to leaking fuel.

(2) Licensed power increase from 2440 MWt/2100 psia operation to 2630 MWt/2250 psia operation.

(3) Power restriction due to secondary plant limitations (turbine).

(4) Estimated.

TABLE 2.2

MAINE YANKEE
FUEL ASSEMBLY TYPES BY CYCLE

Fuel Type	Enrichment (w/o U-235)	Assembly Mechanical Design Type	Number of Fuel Assemblies by Cycle 1	1A	2	3	4	5	6
A	2.01	CE-Core 1	69	57	-	-	-	-	-
B	2.40	CE-Core 1	80	24	-	-	-	-	-
C	2.95	CE-Core 1	68	64	-	-	-	-	-
RF	2.33	CE-RF	-	2	-	-	-	-	-
RF	1.93	CE-RF	-	70	-	65	-	-	-
D	1.95	CE-Core 2	-	-	69	-	-	-	-
E	2.52	CE-Core 2	-	-	80	12	61	1	1
F	2.90	CE-Core 2	-	-	68	68	12	-	-
G	2.73	CE-Core 2	-	-	-	32	32	32	-
H	3.03	CE-Core 2	-	-	-	40	40	40	-
I	3.03	CE-Core 2	-	-	-	-	72	72	72
J	3.00	ENC	-	-	-	-	-	72	72
K	3.00	ENC	-	-	-	-	-	-	72

integrity of the Cycle 3 fuel had been demonstrated through irradiation in Cycles 2 and 1A, respectively. All fuel performed successfully during Cycle 3.

Cycle 4 consisted of all fuel assemblies of the Core 2 design. Slight design changes to the fresh Type I fuel were made and discussed in Section 3.2.1 of (7). New fuel and once-burned fuel assemblies from Cycle 2 were inserted and the replacement fuel discharged. A small number of leaking fuel assemblies were discovered near end-of-cycle.

2.4 Cycles 5 and 6

Cycles 5 and 6 consisted of fuel assemblies of the Core 2 CE design and fresh assemblies designed by ENC. A detailed discussion of the ENC design assemblies was provided in (33). Five Core 2 design leaking assemblies returned to the core in Cycle 5 were repaired by replacement of fuel rods with fresh, low enrichment Core 2 design fuel (34 rods) or water-filled zircaloy rods (10 rods). The fuel has performed successfully during Cycles 5 and 6.

TABLE 2.1

MAINE YANKEE OPERATING HISTORY SUMMARY

Cycle	Date of Power Escalation	Core Power Level Licensed (MWt)	Core Power Level Operated (%)	Cycle Burnup (MWD/MT)
1	11/3/72	2440	50-80(1)	10367
1A	10/12/74	2440	80(1)	4500
2	6/29/75	2440	100	17395
3	6/17/77	2630(2)	93	11075
4	8/23/78	2630	97(3)	10496
5	3/17/80	2630	97	10796
6	7/20/81	2630	97	11500(4)

(1) Power decrease and primary system pressure decrease to 1800-2000 psia due to leaking fuel.

(2) Licensed power increase from 2440 MWt/2100 psia operation to 2630 MWt/2250 psia operation.

(3) Power restriction due to secondary plant limitations (turbine).

(4) Estimated.

TABLE 2.2

MAINE YANKEE
FUEL ASSEMBLY TYPES BY CYCLE

Fuel Type	Enrichment (w/o U-235)	Assembly Mechanical Design Type	Number of Fuel Assemblies by Cycle						
			1	1A	2	3	4	5	6
A	2.01	CE-Core 1	69	57	-	-	-	-	-
B	2.40	CE-Core 1	80	24	-	-	-	-	-
C	2.95	CE-Core 1	68	64	-	-	-	-	-
RF	2.33	CE-RF	-	2	-	-	-	-	-
RF	1.93	CE-RF	-	70	-	65	-	-	-
D	1.95	CE-Core 2	-	-	69	-	-	-	-
E	2.52	CE-Core 2	-	-	80	12	61	1	1
F	2.90	CE-Core 2	-	-	68	68	12	-	-
G	2.73	CE-Core 2	-	-	-	32	32	32	-
H	3.03	CE-Core 2	-	-	-	40	40	40	-
I	3.03	CE-Core 2	-	-	-	-	72	72	72
J	3.00	ENC	-	-	-	-	-	72	72
K	3.00	ENC	-	-	-	-	-	-	72

3.0 RELOAD CORE DESIGN

3.1 General Description

3.1.1 Core Fuel Loading

The core of Maine Yankee Cycle 7 consists of 217 fuel assemblies of the type and quantity detailed in Table 3.1. The single Type E assembly is of the Core 2 mechanical design and has irradiation exposure from Cycle 2. Assembly Type J was introduced in Cycle 5 and has irradiation exposure from Cycles 5 and 6. Assembly Type K was introduced in Cycle 6. Assembly Type L is fresh fuel to be introduced in Cycle 7. Assembly Types J, K, and L are ENC design fuel. The significant neutronic difference in the Type L fuel is the increase in enrichment to 3.30 w/o U-235. The total number of fuel rods by assembly type for Cycle 7 is also given in Table 3.1. The core loading by fuel type is given in Table 3.2.

3.1.2 Core Burnable Poison Loading

Burnable poison shim rods are located in selected assemblies in Cycle 7. The total number of shim rods and locations by assembly type is detailed in Table 3.1. The shim locations in the assemblies are illustrated in Figure 3.1. As described in Section 2.2, the single Type E assembly shim rod locations contain water-filled zircaloy rods with end plugs to restrict the flow in these rods.

All burnable poison shims are composed of B_4C in Al_2O_3. The mechanical design of the ENC burnable poison shims was addressed in (33). Design changes for the Type L shims are discussed in Section 3.2. The ENC shim irradiation integrity has been demonstrated in the Types J and K assemblies during Cycles 5 and 6.

The Type L burnable poison shims were fabricated in three separate batch lots. The as-built poison loading of one of the three batch lots fell slightly outside the design specification. The differences in burnable poison loading among the three batches were addressed and accommodated in the core loading pattern, as discussed in detail in Section 4.3.

3.1.3 Core Loading Pattern

The fuel assembly locations designated for Maine Yankee Cycle 7 are given for the first quadrant in Figure 3.2. They are given relative to the previous locations of the Type E assembly in Cycle 2 and the Types J and K assemblies in Cycle 6. The appropriate rotation index relative to the previous assembly position in the core is also given for each assembly. The loading and rotations of the other quadrants are such that mirror symmetry exists with respect to the quadrant boundary lines.

The Cycle 7 loading pattern incorporates a low-leakage design, achieved by placement of fresh fuel assemblies in selected core interior locations and twice-burned fuel assemblies on the core edge. The benefits of such a core design are:

1) A less severe moderator defect with cooldown at end-of-cycle, providing greater shutdown margin for cooldown transients;

2) Reduced irradiation exposure to the reactor pressure vessel, thus reducing the rate of irradiation embrittlement;

3) Extended cycle full-power lifetime due to reduced neutron leakage;

4) Preferred fuel rod power and exposure histories from fuel performance and mechanical integrity considerations; and

5) Improved stability to axial xenon oscillations near end-of-cycle.

The increased Type L enrichment and low-leakage fuel management have resulted in increased burnable poison shim requirements for the Type L fuel to 416 rods, compared to 176 rods each for the Types J and K fuel.

3.1.4 Assembly Exposure History

The calculated exposure history of the Cycle 6 fuel assemblies at BOC is given in Figure 3.3. The exposures are based on an expected cycle length of 11,500 MWD/MT for Cycle 6 and the obtained cycle length of 10,800 MWD/MT for Cycle 5. Table 3.2 gives BOC average exposures by fuel type. The Cycle 7 BOC average exposure for the core is approximately 11,440 MWD/MT.

The exposure history of the assemblies utilized in the analysis is demonstrated to be accurate by comparison with incore detector plant data. Figure 3.4 is a comparison of predicted and actual burnup assembly data late in Cycle 6. The excellent agreement demonstrates a high confidence in the prediction of the core depletion behavior.

3.1.5 CEA Group Configuration

The Control Element Assembly (CEA) group configuration for the lead regulating bank, CEA Bank 5, has been modified for Cycle 7. The requirements for increased available scram reactivity to maintain shutdown margin have necessitated such changes. Figure 3.5 shows the CEA group locations in the quarter core and Figure 3.6 details the CEA Bank 5 finger locations in Cycles 1-6 and in Cycle 7. The configuration changes are:

1) The eight part-strength CEAs (i.e., one or two active B_4C fingers, with the remaining fingers stainless steel) in the original Bank 5 locations have been replaced with full-strength CEAs. This change provides increased available scram reactivity.

2) The four part-strength CEAs with two active B_4C fingers have been added to Bank 5 in former part-length CEA locations for better local power distribution control. These four CEA locations are nonscrammable and do not contribute to the available scram reactivity.

In addition to providing additional scram reactivity, the reactivity worth of CEA Bank 5 is significantly increased. The effects of the changes are most evident in the CEA ejection results in Section 4.9.2. Minor changes to the methodology for ejected CEAs are detailed in Section 4.10 to accommodate the CEA ejection results and maintain appropriate margins.

The increased CEA Bank 5 reactivity worth and spatial distribution improves the core symmetric offset control effected with the regulating bank. As such, less CEA insertion and, consequently, less local linear heat rate change is required for the same change in symmetric offset, placing a less severe duty cycle on the fuel.

The original CEA Bank 5 (9 CEAs) and the former part-length CEA bank (4 CEAs) are not electrically connected as a single CEA bank. As such, their movements will be administratively controlled for positioning as a single CEA bank. To accommodate this movement, the physics input to the RPS setpoint analysis (Section 6) has included power distribution cases sufficient to justify differences in insertion in these two regulating groups subject to the CEA group insertion limits, as discussed in Section 4.9.1. This difference in insertion is addressed in the Technical Specification changes.

3.2 Fuel System Design

3.2.1 Fuel Mechanical Design

The mechanical design of the Type E fuel assemblies fabricated by CE has been reported in previous submittals [(7), (15), (20), and (30)]. The Types J, K, and L fuel assemblies fabricated by ENC have been designed to maintain mechanical, material, and chemical compatibility with all other fuel and structures in the reactor core. Table 3.3 lists the mechanical design features of all fuel types.

The ENC fuel design is similar to the Core 2 fuel design, with the following differences:

1) The fuel rod cladding has a nominal wall thickness of 31 mils (versus 28 mil nominal wall thickness for Core 2 design fuel).

2) Fuel pellets are short and dished with a nominal fuel density of 94% T.D.

3) The bi-metallic fuel rod spacers are a Zircaloy-4 structure with Inconel springs.

4) To allow removal of fuel rods for replacement or inspection either prior to or after irradiation, the fuel assembly upper tie plate is mechanically locked to the Zircaloy-4 guide tubes and may be easily removed and replaced. This design feature also facilitates replacement of guide tube wear sleeves.

The detailed fuel assembly description and mechanical design criteria and the design considerations for the Types J, K, and L reload fuel have been described in (33). The fuel pellet design changes for the Type L fuel include pellet length and dish depth, as described in (48). The fuel rod plenum length in the Type L fuel has been increased slightly due to the removal of the lower fuel rod insulator disc. The effect of the lower insulator disc was evaluated by ENC and found to be inconsequential relative to fuel design performance.

The burnable poison shim pellets used in the Type L fuel assemblies are described in Table 3.3. The mechanical design criteria and design considerations for the Type L shims are the same as those used for the Types J and K shims. The Type L shim pellets were fabricated in three batch lots which comprise both chamfered and unchamfered pellets.

3.2.2 Thermal Design

Fuel assembly types and quantities for the core of Maine Yankee Cycle 7 are given in Table 3.1. Fuel assembly Type E is of the Combustion Engineering (CE) design. Fuel assembly Types J, K, and L are of Exxon Nuclear Corporation (ENC) design. Fuel designated Type L represents the new fuel assemblies being provided by ENC.

Fuel thermal calculations were performed for all fuel types. These calculations were performed using the GAPEX (21) computer program. The methodology of calculation for Cycle 7 is essentially that used in previous reload analyses with the following two enhancements introduced during Cycle 6 analysis:

1) The appropriate steady-state history effects for the fuel rod thermal performance were modeled by utilizing the time-dependent power factors calculated in the physics analysis for the fuel rods of interest. Applicable power and burnup uncertainty factors were incorporated.

2) The method uses the NRC NUREG-0418 modeling approach for burnup enhancement on the fission gas release prediction.

The GAPEX code calculates pellet-to-clad gap conductance from a combination of theoretical and empirical models which predict fuel and cladding thermal expansion, fission gas release, pellet swelling, pellet densification, pellet cracking, and fuel and cladding thermal conductivity.

The thermal effects analysis encompassed a study of fuel rod response as a function of the detailed cycle burnup and power. The fuel rod types and power histories examined in detail include:

1) Maximum power rod - Type L fuel

2) Maximum power rod - Type K fuel

3) Maximum power rod - Type J fuel

Figures 3.7 through 3.9 demonstrate the effect of Linear Heat Generation Rate - LHGR - on fuel temperature at beginning-of-cycle conditions. Figures 3.10 through 3.12 demonstrate the effect of LHGR on fuel temperature at end-of-cycle conditions. Table 3.4 lists UO_2 melting temperature and centerline temperature for the rods of interest at various points in life and various power levels. The calculated internal fuel rod pressures are less than Operating Coolant System pressure throughout Cycle 7 operation (48).

The result of the fuel performance calculations indicates that the thermal performance is not significantly different than the previously reported performance of fuel Types E, J, and K and that the conclusions set forth in (33), (15) and (7) continue to be applicable.

The conclusions are summarized below:

1) A 21 kW/ft limit on LHGR is a conservative limit for avoiding fuel centerline melt for Type E, K, and L fuel.

2) A 20 kW/ft limit on LHGR is conservative for Type J fuel.

3) A conservative value for use in transient analysis assumptions for the average channel gap coefficient is 600 Btu/hr-ft^2-^{o}F.

As noted in 2) above, a 20 kW/ft LHGR limit is applicable to the Type J fuel for Cycle 7. This is reflected in the proposed changes to the LHGR SAFDL, Technical Specification 2.2, for Cycle 7. Table 3.5 provides a comparison of the maximum radial relative pin power for the Type fuel to the core maximum radial pin power throughout Cycle 7. It is clear from Table 3.5 that the Type J fuel is never limiting for Cycle 7 in spite of the lower LHGR limit. Therefore, subsequent portions of this report will refer to the LHGR SAFDL for Cycle 7 in terms of the 21 kW/ft limit corresponding to the limiting fuel types in the core.

3.2.3 Thermal-Hydraulic Design

Steady-state and transient DNBR analysis of the Cycle 7 core have been performed using the COBRA-IIIC computer program (22), in the manner described in (8) and (9), and the COBRA-IV computer program (39) as described below. The models reflect the intended Cycle 7 coolant conditions with uncertainties, the Cycle 7 power distributions with uncertainties, the assembly flow distribution due to differences in hydraulic characteristics and inlet flow maldistribution, and the specific geometry of the Maine Yankee fuel assemblies.

An eighth core assembly-by-assembly COBRA-IV model was used to determine hot assembly enthalpy rise flow factors. This model explicitly represents each fuel assembly in the eighth core in the specific location it will reside for Cycle 7 operation, and accounts for the differences in hydraulic characteristics between the single CE fuel assembly and the ENC fuel which comprises the rest of the core.

The COBRA-IV program is used due to problem size limitations on Yankee's version of COBRA-IIIC. At least 34 radial power regions are required to model each fuel assembly in an eighth core region. COBRA-IIIC allows only 20. Rather than modify the COBRA-IIIC coding to increase the allowed problem size, it was more convenient to use COBRA-IV. COBRA-IV and COBRA-IIIC produce similar results for problems of a size that can be executed with both.

Enthalpy rise flow factors were calculated with the eighth core COBRA-IV model for the potential hot assembly locations assuming power distributions typical of the full range of CEA insertion for Cycle 7. The inlet flow maldistribution imposed on this model is based on the result of flow measurements taken in scale model flow tests of the Maine Yankee reactor vessel reported in (36) and the FSAR (2). The resulting hot assembly flow factors are 1.0 for power distributions where the CEAs are restricted to insertion less than allowed by the 60% power PDIL (Technical Specification 3.10), and 0.95 for power distributions with deeper CEA insertions. The enthalpy rise flow factor for the single E-16, CE assembly, which is centrally located in the core is 1.0 for all CEA configurations. These flow factors are conservatively applied in the Cycle 7 COBRA-IIIC subchannel analysis models by reducing the inlet flow for the entire model by the appropriate flow factor.

The magnitude of the additional system hydraulic resistance added to the core by the replacement of the 72 CE fuel assemblies with 72 ENC assemblies of higher resistance is small. The total flow delivered by the reactor coolant pumps remains higher than the design flow rate assumed in the safety analysis, 360,000 gpm, which remains unchanged for Cycle 7.

The potential effects of fuel rod bow on thermal-hydraulic performance has also been evaluated for Cycle 7 operation. For CE fuel (23) presented a model for treating the effects of fuel rod bowing on thermal-hydraulic performance. This model requires a reduction of DNBR as shown below:

Bundle Average Burnup	% DNBR Reduction
0 - 15000 MWD/MTU	0
15000 - 24000 MWD/MTU	0
24000 - 33000 MWD/MTU	3.0

The Maine Yankee fuel management scheme is such that the CE fuel assembly is never the DNB limiting assembly in Cycle 7. To evaluate whether this non-limiting assembly could become limiting with a fuel rod bow penalty, the peak pin in the CE assembly was compared to the corresponding parameters at the limiting core location. It was found that the minimum difference between the limiting location in the core and the peak pin in the CE assembly is greater than 23%. This large difference more than accommodates the additional penalty for fuel rod bowing. The maximum expected assembly average burnup for the CE assembly at the end of Cycle 7 is less than 33,000 MWD/MTU.

Allowances for rod pitch, bow and clad diameter variations for the ENC fuel remains at 1.00. Allowances for manufacturing tolerances on rod pitch and clad diameter, if considered in the most adverse situation, would result in a maximum channel closure in the vicinity of 10%. Using the methodology of (34), the maximum channel gap closure due to fuel rod bowing for the ENC fuel assembly with the highest burnup during Cycle 7 is less than 28% (Table 3.6). Tests performed at Columbia (24) indicate that a degradation in DNB performance is not experienced until channel closures exceed 50%. Therefore, a flow factor of 1.0 is justified for the ENC fuel in Cycle 7 since the channel closure resulting from rod pitch, bow and clad diameter considerations for any ENC fuel during Cycle 7 will be less than 50%.

Table 3.7 contains a list of the pertinent thermal-hydraulic design parameters used for both safety analysis and for generating reactor protection system setpoint information. The list also includes the corresponding thermal-hydraulic parameters from (3), the Reference Analysis and (37), Cycle 6, for comparison.

3.3 Control Element Assembly (CEA) Design

The eight full-strength CEAs fabricated for use in Cycle 7 by CE contain minor design improvements compared to the original design CEAs. The CEAs are dimensionally and neutronically equivalent to the original design CEAs. The improvements are:

1) A Ag-In-Cd slug has been added to the tip of the center CEA finger, which formerly contained only B_4C pellets. This change will reduce swelling due to B_4C irradiation near the CEA tip.

2) The nose cap design for the center CEA finger has been replaced by the same nose cap design used in the outer four fingers, making all the CEA fingers interchangeable. A two-inch extra region of poison material in the center finger only has thus been eliminated.

3) The Ag-In-Cd slugs used in the CEA tips contain a hole drilled completely through them. In the original CEA design, these holes were plugged to prevent B_4C particles from entering. Hot cell studies have shown the B_4C pellets do not disintegrate and the plugs are unnecessary.

Table 3.1

Maine Yankee Cycle 7
Assembly Description

Assembly Designation	Exposure History	Number of Fuel Rods per Assembly	Initial w/o U-235 Fuel	Number of Shim Locations per Assembly	Initial mg B-10 per inch in Shims	Number of Assemblies in Core	Total Shim Locations	Total Fuel Rods
E-16	Cycle 2	160	2.52	16	29.0*	1	16*	160
J-0	Cycles 5, 6	176	3.00	0	–	48	0	8448
J-4	Cycles 5, 6	172	3.00	4	23.8	4	16	688
J-8	Cycles 5, 6	168	3.00	8	23.8	20	160	3360
K-0	Cycle 6	176	3.00	0	–	48	0	8448
K-4	Cycle 6	172	3.00	4	23.8	4	16	688
K-8	Cycle 6	168	3.00	8	23.8	20	160	3360
L-0	Fresh	176	3.30	0	–	16	0	2816
L-4	Fresh	172	3.30	4	23.8	12	48	2064
L-8	Fresh	168	3.30	8	23.8	40	320	6720
L-12	Fresh	164	3.30	12	23.8	4	48	656
Core Totals						217	784	37408

* E-16 shims replaced with water-filled rods for Cycle 7

Table 3.2

Maine Yankee Cycle 7
Core Loading

Assembly Type	Number of Assemblies	KGU per Assembly	Total KGU	Average Exposure at BOC*
E-16	1	353.36	353.36	17583
J-0	48	381.12	18293.76	21982
J-4	4	372.46	1489.84	26268
J-8	20	363.80	7276.00	26176
K-0	48	381.12	18293.76	9391
K-4	4	372.46	1489.84	13352
K-8	20	363.80	7276.00	13364
L-0	16	381.12	6097.92	0
L-4	12	372.46	4469.52	0
L-8	40	363.80	14552.00	0
L-12	4	355.14	1420.56	0
			81012.56	11441

* Based on End-of-Cycle 6 at 11,500 MWD/MT.

Table 3.3

Mechanical Design Features of Cycle 7 Fuel

	Type E	Types J and K	Type L
Fuel Assembly			
Overall length	156.718**	156.718	156.718
Spacer grid size (max. square)	8.115	8.115	8.115
Retention grid	0	0	0
No. zircaloy grids	8	0	0
No. inconel grids	1	0	0
No. bimetallic grids	0	9	9
Fuel rod growth clearance	1.021	1.300	1.300 min.
Fuel Rod			
Active fuel length	136.7	136.7	136.7
Plenum length	8.575	8.6	8.80
Clad OD	0.440	0.440	0.440
Clad ID	0.384	0.378	0.378
Clad wall thickness	0.028	0.031	0.031
Pellet OD	0.3765	0.370	0.370
Pellet length	0.450	(***)	(***)
Dish depth	0.023	0.005	0.008
Clad material	Zr-4	Zr-4	Zr-4
Pellet density initial	95%	94.0%	94.0%
Initial pressure	(*)	(***)	(***)
Poison Rods			
Overall rod length	146.513	146.500	146.500
Clad OD	0.440	0.440	0.440
Clad ID	0.388	0.378	0.378
Clad wall thickness	0.026	0.031	0.031
Pellet OD	0.376	0.353	0.353
Clad material	Zr-4	Zr-4	Zr-4

* YAEC 1099P
** All length dimensions are in inches
*** XN-NF-79-52

Table 3.4

Centerline and UO_2 Melt Temperature Comparison

Fuel Type	Melt Temperature (°F)	Power Level (kW/ft)	Centerline Temperature (°F)
BOC J	4862	19	4466
		20	4680
		21	4883
K	4942	19	4346
		20	4561
		21	4767
L	5054	19	4590
		20	4746
		21	4894
EOC J	4800	19	4572
		20	4784
		21	4986
K	4867	19	4346
		20	4561
		21	4767
L	4955	19	4334
		20	4549
		21	4754

Table 3.5

Maine Yankee Cycle 7
Ratio of Maximum Radial Relative Pin Powers - Maximum
in Type J Fuel to Maximum in Core

Rodded Condition
HFP, Equilibrium Conditions of ARO

Regulating Banks Inserted	Ratio of Maximum Radial Relative Pin Powers BOC 50 MWD/MT	MOC 6K MWD/MT	EOC 12K MWD/MT
ARO	0.751	0.767	0.739
Bank 5	0.817	0.773	0.722
Banks 5 + 4	0.709	0.739	0.746
Banks 5 + 4 + 3	0.656	0.690	0.710
Banks 5 + 4 + 3 + 2	0.663	0.697	0.704
Banks 5 + 4 + 3 + 2 + 1	0.672	0.682	0.676

Table 3.6

Cycle 7 Determination of ENC Rod Bow MDNBR Penalty

Maximum EOC 7 ENC Assembly Average Burnup (1)	= 42.973 GWD/MTU
Fractional Gap Closure Due to Fuel Rod Bow (2), delta C/C_o	= 0.2768
DNBR Rod Bow Penalty (3)	= 0.0

(1) Includes 2% calorimetric power uncertainty plus 10% physics uncertainty

(2) From Reference 34, delta C/C_c = 0.0533 + .0052 * BU

where C_o = nominal initial gap width
BU = assembly average burnup (GWD/MTU)

(3) Penalty = sigma_B, where $\text{MDNBR}_{\text{Bowed}} = \text{MDNBR}_{\text{Unbowed}} (1 - \text{sigma}_B)$
and

sigma_B = 0.0 for delta C/C_o less than 0.50.

Table 3.7

Maine Yankee Cycle 7
Thermal-Hydraulic Parameters at Full Power

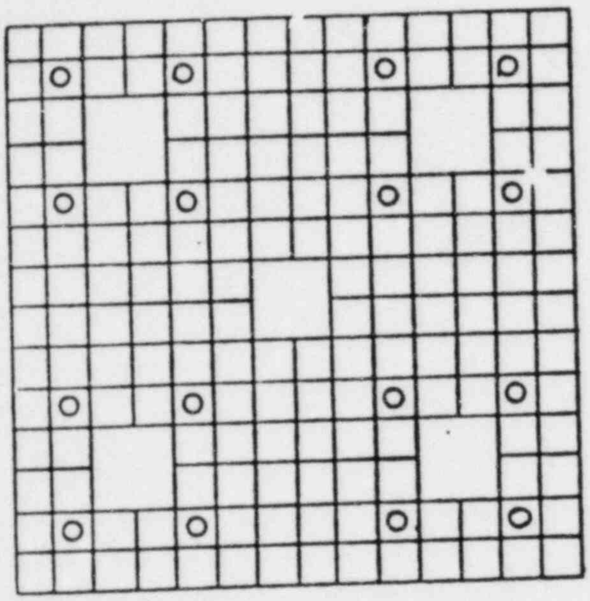
General Characteristics	Units	Cycle 5	Cycle 6	Cycle 7
Total Heat Output	MWT	2630	2630	2630
	10^6 Btu/hr	8976	8976	8976
Fraction of Heat Generated in Fuel Rod		0.975	0.975	0.975
Pressure				
Nominal	psig	2235	2235	2235
Minimum in Steady-State	psig	2185	2060	2060
Maximum in Steady-State	psig	2285	2260	2260
Design Inlet Temperature (steady-state)	°F	554	546-554	546-554
Total Reactor Coolant Flow (design)	10^6 lb/hr	134.6	136.0-134.6	136.0-134.6
Coolant Flow Through Core (design)	10^6 lb/hr	130.7	132.1-130.7	132.1-130.7
Hydraulic Diameter (nominal channel)	ft	0.044	0.044	0.044
Average Mass Velocity	10^6 lb/hr-ft^2	2.444	2.47-2.444	2.47-2.444
Pressure Drop Across Core (design flow)	psi	9.7	10.18	10.35
Total Pressure Drop Across Vessel (Based on nominal dimensions and design flow)	psi	32.4	32.9	33.1
Core Average Heat Flux	Btu/hr-ft^2	178,742*	177,981*	179,769*
Total Heat Transfer Area	ft^2	48,978*	49,188*	48,684*
Film Coefficient at Average Conditions	Btu/hr-ft^2°F	5,640	5,636	5,636
Maximum Clad Surface Temperature	°F	656	656	656
Average Film Temperature Difference	°F	31.7	31.6	31.9
Average Linear Heat Rate of Rod	kW/ft	6.03*	6.01*	6.07*
Average Core Enthalpy Rise	Btu/lb	68.7	68.7	68.7

Calculational Factors	Cycle 5 CE	Cycle 6 ENC	Cycle 6 CE	Cycle 7 ENC	Cycle 7 CE
Engineering Heat Flux Factor	1.03	1.03	1.03	1.03	1.03
Engineering Factor on Hot Channel Heat Input	1.03	1.03	1.03	1.03	1.03
Flow Factors					
Inlet Plenum Non-Uniform Distribution	1.05	1.05	1.05	1.00***	1.00***
Rod Pitch, Bowing and Clad Diameter	1.065	1.00	1.065**	1.00	1.065**

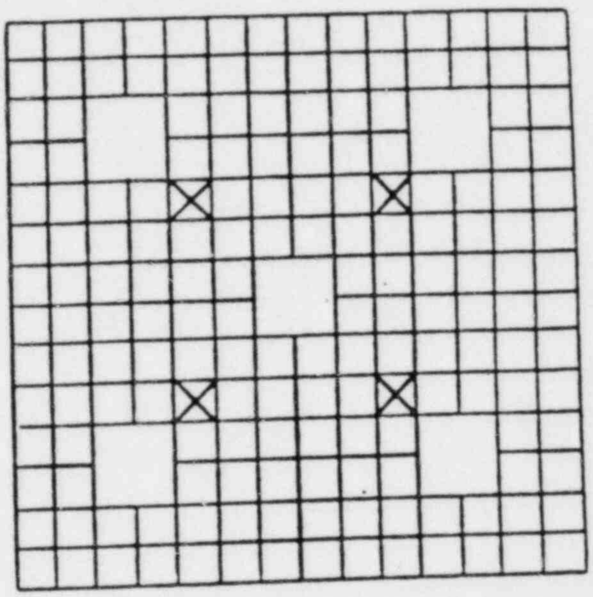
* Allows for axial shrinkage due to fuel densification.
** Additional 3% penalty required due to high burnup of CE fuel. See Section 3.2.3.
*** See discussion in Section 3.2.3.

FIGURE 3.1

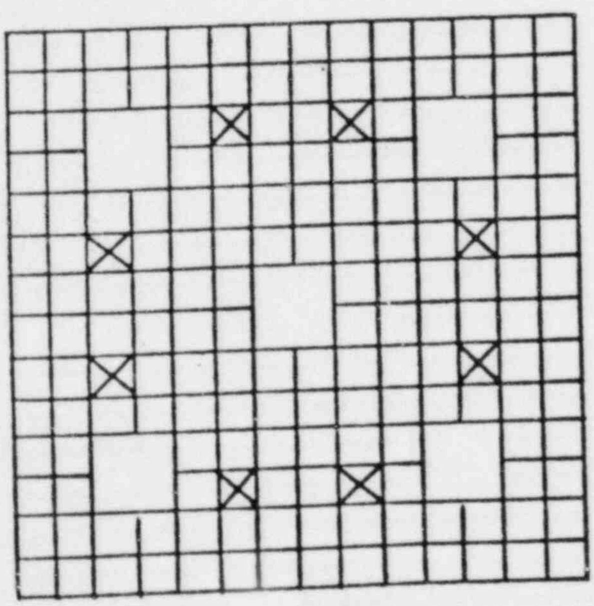
Maine Yankee Cycle 7
Burnable Poison Shim Assembly Locations



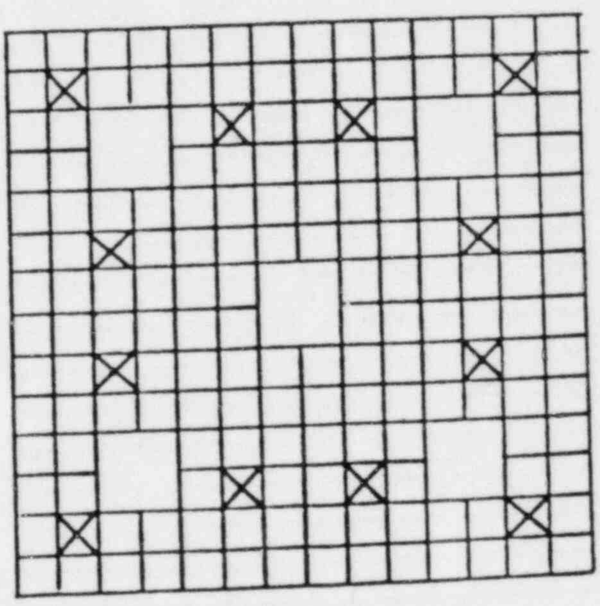
16 Water Rod Assembly (E-16)



4 Shim Assembly (J-4,K-4,L-4)



8 Shim Assembly (J-8,K-8,L-8)



12 Shim Assembly (L-12)

Legend: ○ Water-filled rods

⊠ B_4C in Al_2O_3 shim rods

Figure 3.2
MAINE YANKEE CYCLE 7
ASSEMBLY LOADING PATTERN

FUEL TYPE		
E	E-16 62 10 0	...Cycle 7 Location ...Cycle 2 Location ...Rotation Index*
J,K	J-8 1 16 2	...Cycle 7 Location ...Cycle 6 Location ...Rotation Index**
L	L-0 2	...Cycle 7 Location (Fresh Fuel)

							J-8 1 16 2	L-0 2
				J-0 3 25 2	L-0 4	L-4 5	L-8 6	K-8 7 9 2
			L-8 8	L-8 9	K-0 10 3 0	K-0 11 2 1	J-0 12 27 0	K-8 13 22 1
		L-8 14	L-4 15	K-0 16 1 0	J-0 17 34 3	L-8 18	J-0 19 43 1	L-8 20
	J-0 21 32 2	L-8 22	K-0 23 45 0	J-8 24 20 0	[illegible]-8 25 11 1	K-0 26 4 2	J-0 27 18 2	J-0 28 36 0
	L-0 29	K-0 30 21 0	J-0 31 41 1	J-8 32 38 3	L-12 33	K-0 34 5 1	K-0 35 8 1	J-0 36 53 0
	L-4 37	K-0 38 54 3	L-8 39	K-0 40 29 2	K-0 41 37 3	K-4 42 15 0	K-8 43 12 3	K-8 44 7 0
J-8 45 23 2	L-8 46	J-0 47 49 0	J-0 48 51 3	J-0 49 39 2	K-0 50 14 3	K-8 51 47 1	J-4 52 33 0	L-8 53
L-0 54	K-8 55 9 1	K-8 56 22 0	L-8 57	J-0 58 59 0	J-0 59 61	K-8 60 55 0	L-8 61	E-16 62 10 0

* Clockwise multiple of 90° relative to Cycle 2 location in quadrant.
** Clockwise multiple of 90° relative to Cycle 6 location in quadrant.

Figure 3.3
Maine Yankee Cycle 7
Calculated Assembly Exposures at BOC

82/03/03. — 14.32.28. MYC7 EOC6 @ 11.5K MWD/MT ARO CALCULATED BURNUPS AT BOC

CORE AVERAGE EXPOSURE = 11440.9
MAX ASSEMBLY EXPOSURE = 26934.0 @ LOC 1

CORE POSITION/ASSEMBLY NUMBER .
FUEL TYPE ....
CEA BANK TYPE ....
ASSEMBLY BURNUP (MWD/MT) ......

Core Position	Assembly Number	Fuel Type	CEA Bank Type	Assembly Burnup (MWD/MT)
A14	1	J-8		26934.0
A12	2	L-0		0.0
B17	3	J-0		23306.0
B16	4	L-0	* C *	0.0
B15	5	L-4		0.0
B13	6	L-8	* 1 *	0.0
B11	7	K-8		12289.0
C18	8	L-8		0.0
C17	9	L-8	* A *	0.0
C16	10	K-0		7516.0
C15	11	K-0	* C *	9640.0
C13	12	J-0		21332.0
C11	13	K-8	* 5 *	12289.0
D19	14	L-8		0.0
D18	15	L-4	* 5 *	0.0
D17	16	K-0		7500.0
D16	17	J-0	* A *	24338.0
D15	18	L-8		0.0
D13	19	J-0	* 3 *	21794.0
D11	20	L-8		0.0
E20	21	J-0		23306.0
E19	22	L-8	* A *	0.0
E18	23	K-0		7500.0
E17	24	J-8		25937.0
E16	25	J-8		25537.0
E15	26	K-0	* 2 *	10430.0
E13	27	J-0		21254.0
E11	28	J-0		19748.0
F20	29	L-0	* C *	0.0
F19	30	K-0		7516.0
F18	31	J-0	* A *	24338.0
F17	32	J-8		25537.0
F16	33	L-12	* 6 *	0.0
F15	34	K-0		12475.0
F13	35	K-0		8785.0
F11	36	J-0		19984.0
G20	37	L-4		0.0
G19	38	K-0	* C *	9640.0
G18	39	L-8		0.0
G17	40	K-0	* 2 *	10430.0
G16	41	K-0		12475.0
G15	42	K-4	* B *	13352.0
G13	43	K-8		14445.0
G11	44	K-8	* 4 *	13354.0
H21	45	J-8		26934.0
J20	46	L-8	* 1 *	0.0
J19	47	J-0		21332.0
J18	48	J-0	* 3 *	21794.0
J17	49	J-0		21254.0
J16	50	K-0		8785.0
J15	51	K-8		14445.0
J13	52	J-4	* B *	26267.0
J11	53	L-8		0.0
K21	54	L-0		0.0
L20	55	K-8		12289.0
L19	56	K-8	* 5 *	12289.0
L18	57	L-8		0.0
L17	58	J-0		19748.0
L16	59	J-0		19984.0
L15	60	K-8	* 4 *	13354.0
L13	61	L-8		0.0
L11	62	E-16	* 5 *	17583.0

BATCH NUMBER	BATCH ID	BATCH AVERAGE EXPOSURE
1	E-16	17583.0
2	J-0	21981.7
3	J-4	26267.0
4	J-8	26175.8
5	K-0	9391.0
6	K-4	13352.0
7	K-8	13364.4
8	L-0	0.0
9	L-4	0.0
10	L-8	0.0
11	L-12	0.0

Figure 3.4

Maine Yankee Cycle 6
Burnup Distribution by Assembly
INCA Versus Predicted
at 10,000 MWD/MT Cycle Exposure

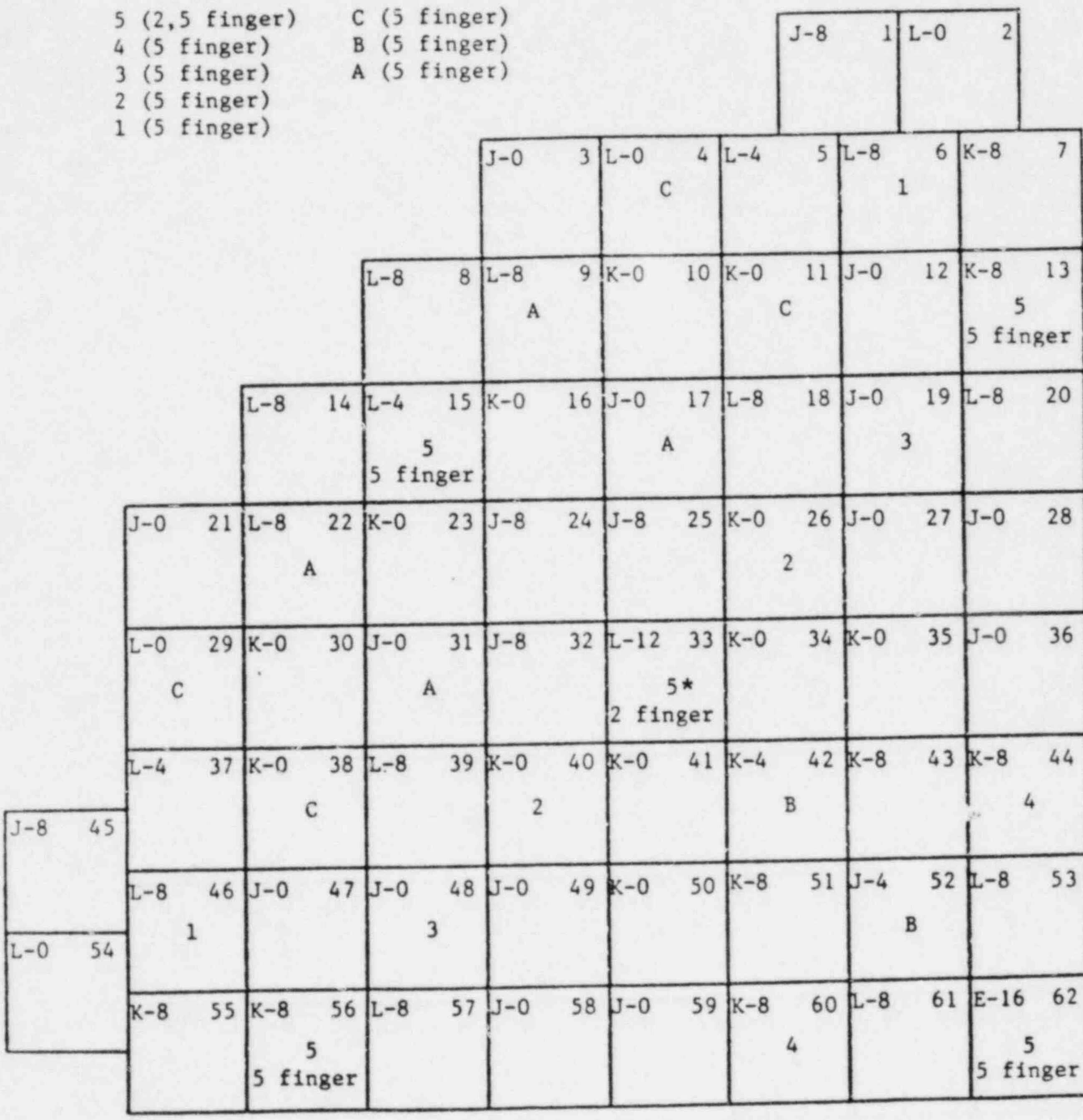
Assembly Type and INCA Location . . . .
Assembly Exposure (MWD/MT) INCA . . . .
Calculated . . . .
Percent Difference . . . .

			K-0 8 6616 6451 -2.5	K-0 21 8503 8283 -2.6	
	K-0 15 6579 6473 -1.6	K-0 31 8960 9052 +1.0	K-0 11 10800 10841 +0.4	I-0 25 32659 32623 -0.1	K-8 4 11796 11406 -3.3
K-0 16 7655 7572 -1.1	K-8 33 10610 10572 -0.4	I-0 13 29556 29706 +0.5	J-8 28 23759 23766 +0.1	K-8 7 12662 12420 -1.9	I-0 20 31534 31681 +0.5
K-4 34 11553 11546 -0.1	J-8 14 25306 25281 -0.1	I-0 30 30366 30646 +0.9	J-0 10 19000 19507 +2.7	I-4 24 33783 33998 +0.6	J-8 3 24251 24309 +0.2
	I-0 32 29194 29238 +0.2	J-0 12 21364 21661 +1.4	I-0 27 29278 29566 +1.0	J-0 6 19911 19739 -0.9	I-0 19 31013 31318 +1.0
		J-4 29 24615 24682 +0.3	J-0 9 22581 22842 +1.2	I-4 23 34743 33945 -2.3	J-0 2 18390 18210 -1.0
			I-4 26 32705 32372 -1.0	J-0 5 20146 20318 +0.9	I-4 18 34037 33922 -0.3
				I-0 22 29000 29115 +0.4	J-0 1 18644 18429 -1.2
					E-16 17 27377 27055 -1.2

Fuel Type	Exposure INCA	Exposure Calculated	Difference (%)
E-16	27377	27055	-322 (-1.2%)
I-0	30371	30536	165 (0.5%)
I-4	33965	33697	-268 (-0.8%)
J-0	20253	20398	145 (0.7%)
J-4	24615	24682	67 (0.3%)
J-8	24476	24481	5 (0.0%)
K-0	8185	8112	- 73 (-0.9%)
K-4	11553	11546	- 7 (-0.1%)
K-8	11668	11478	-190 (-1.6%)
Core	20948	20956	8
Cycle	9995	10000	5

Figure 3.5
Maine Yankee Cycle 7
CEA Group Locations

Regulating CEA Group	Shutdown CEA Group
5 (2,5 finger)	C (5 finger)
4 (5 finger)	B (5 finger)
3 (5 finger)	A (5 finger)
2 (5 finger)	
1 (5 finger)	



* non-scrammable CEA locations

Figure 3.6
Maine Yankee CEA Bank 5 Configurations

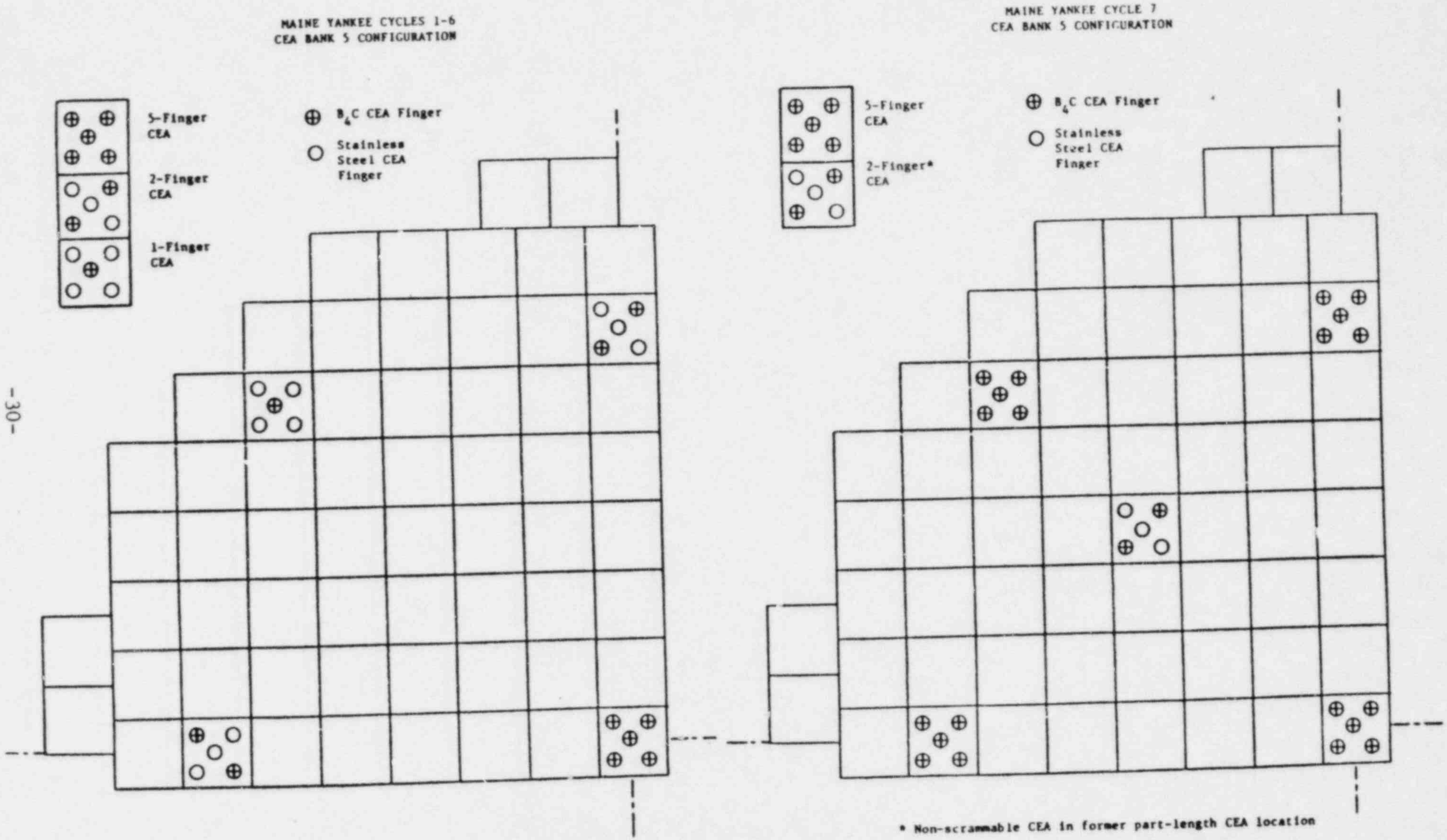


FIGURE 3-7

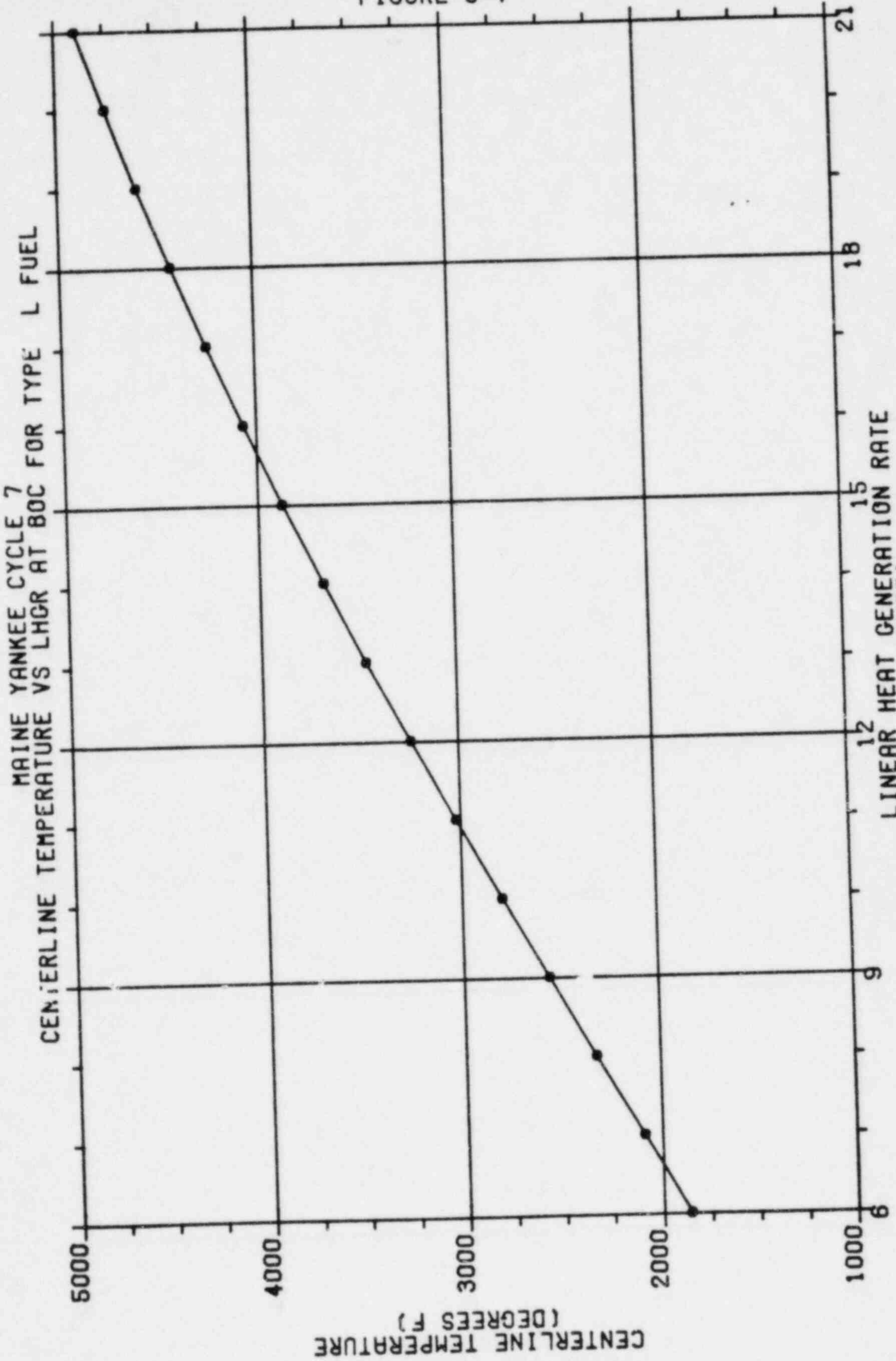

MAINE YANKEE CYCLE 7
CENTERLINE TEMPERATURE VS LHGR AT BOC FOR TYPE L FUEL
LINEAR HEAT GENERATION RATE
(KW/FT)
6
9
12
15
18
21
CENTERLINE TEMPERATURE
(DEGREES F)
1000
2000
3000
4000
5000

FIGURE 3-8

MAINE YANKEE CYCLE 7
CENTERLINE TEMPERATURE VS LHGR AT BOC FOR TYPE K FUEL

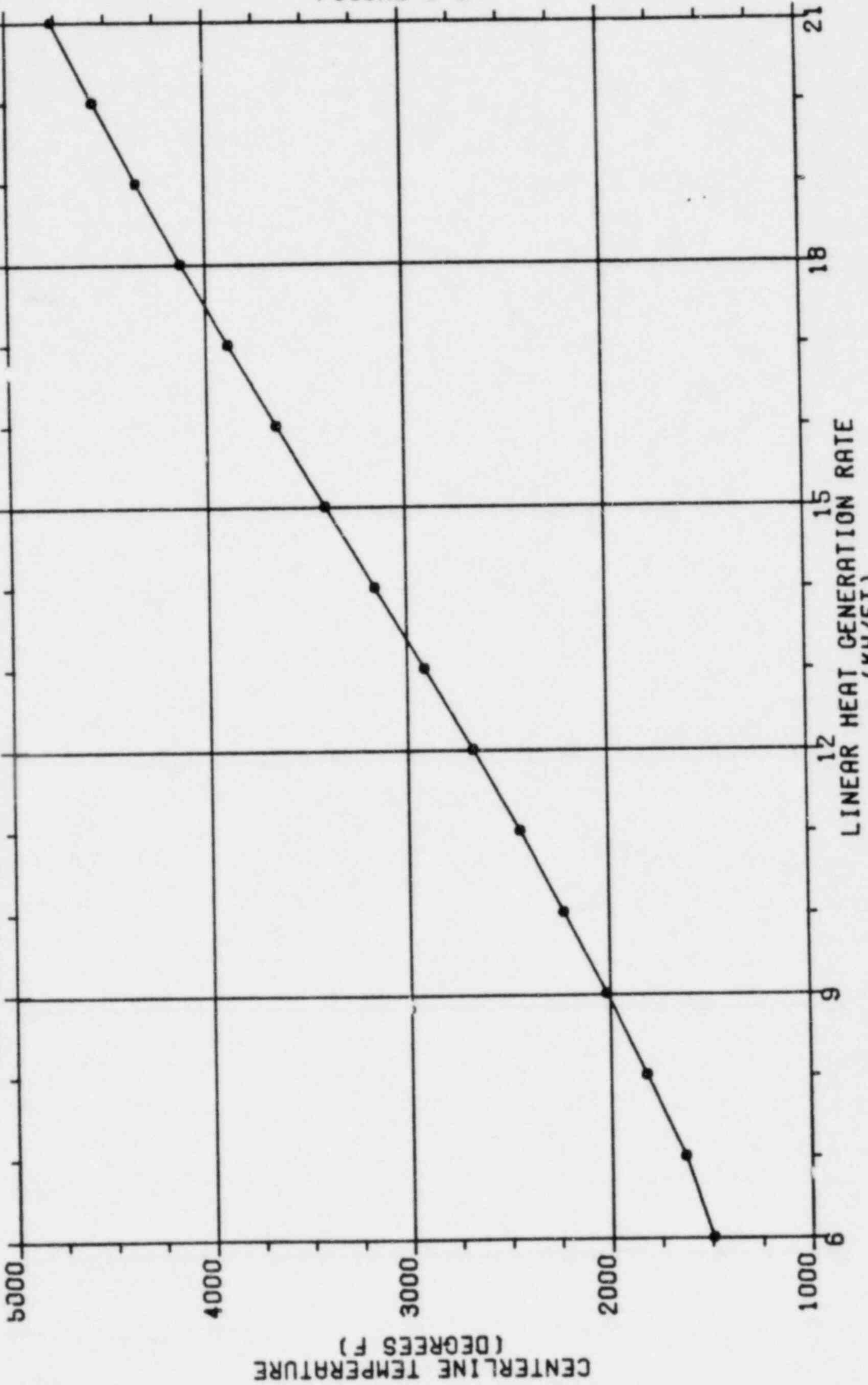


FIGURE 3-9

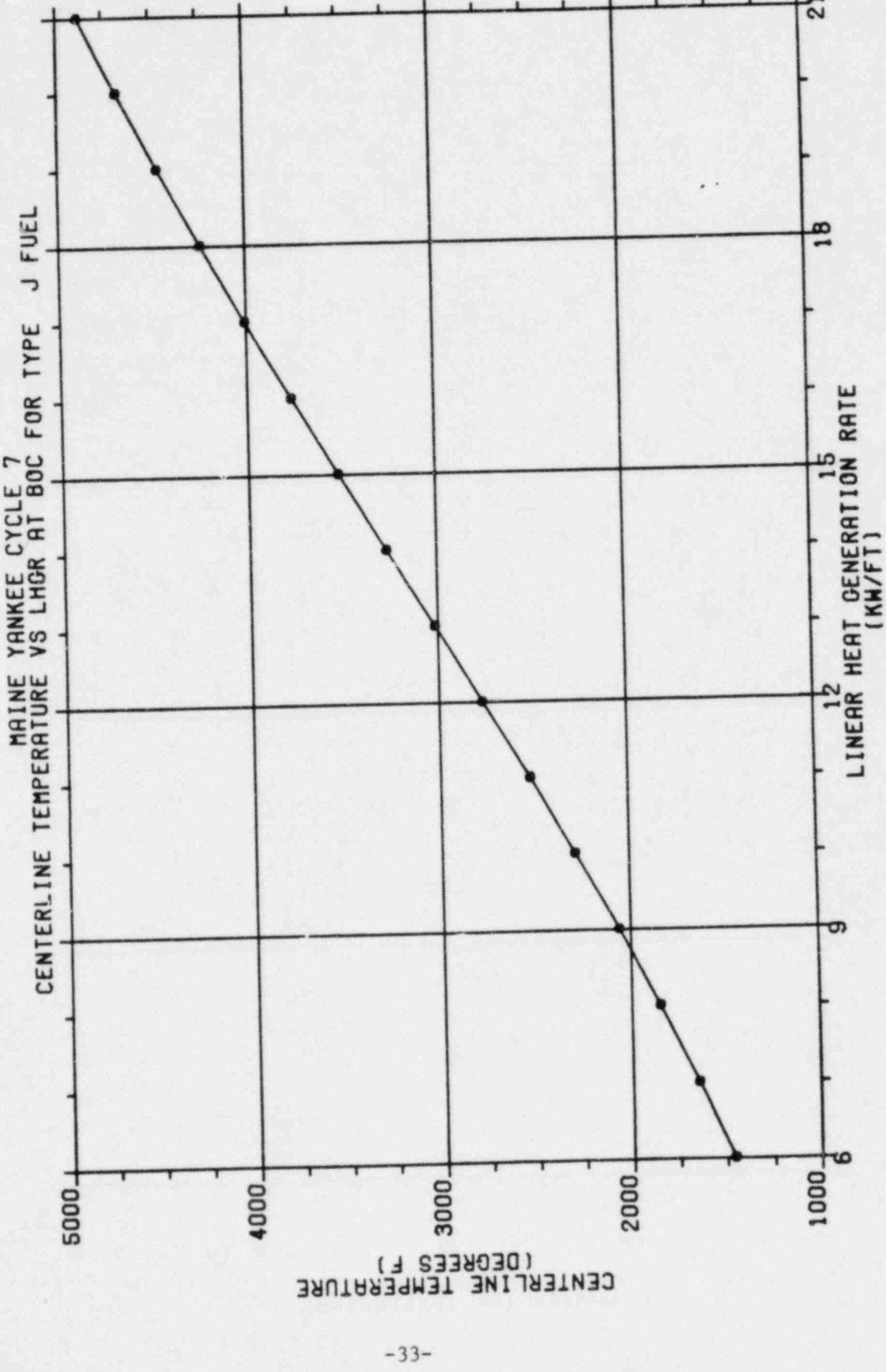
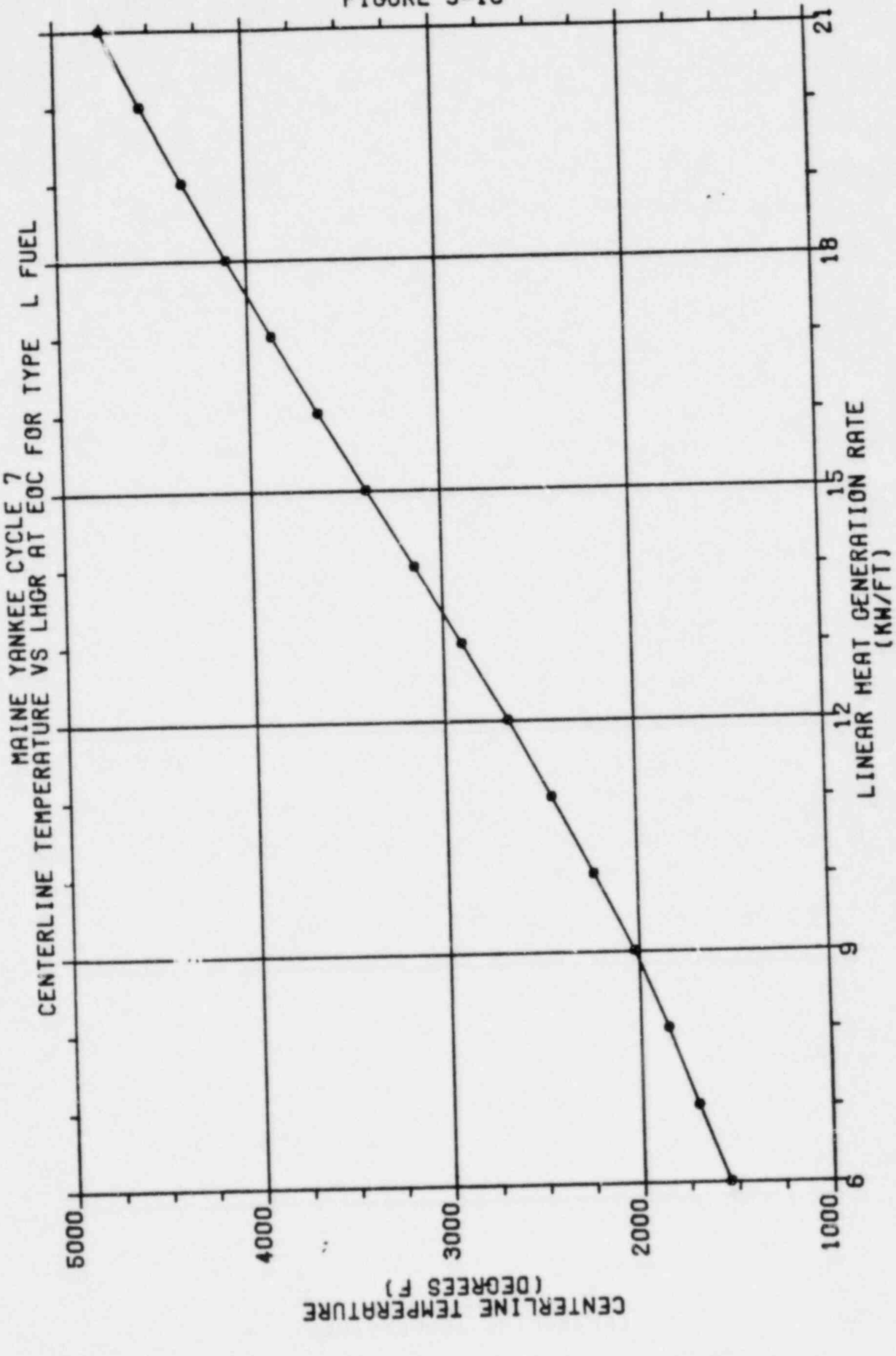

MAINE YANKEE CYCLE 7
CENTERLINE TEMPERATURE VS LHGR AT BOC FOR TYPE J FUEL
LINEAR HEAT GENERATION RATE
(KW/FT)
CENTERLINE TEMPERATURE
(DEGREES F)
5000
4000
3000
2000
1000
6
9
12
15
18
21

FIGURE 3-10



MAINE YANKEE CYCLE 7
CENTERLINE TEMPERATURE VS LHGR AT EOC FOR TYPE L FUEL
LINEAR HEAT GENERATION RATE
(KW/FT)
6
9
12
15
18
21
CENTERLINE TEMPERATURE
(DEGREES F)
1000
2000
3000
4000
5000

FIGURE 3-11

MAINE YANKEE CYCLE 7
CENTERLINE TEMPERATURE VS LHGR AT EOC FOR TYPE K FUEL

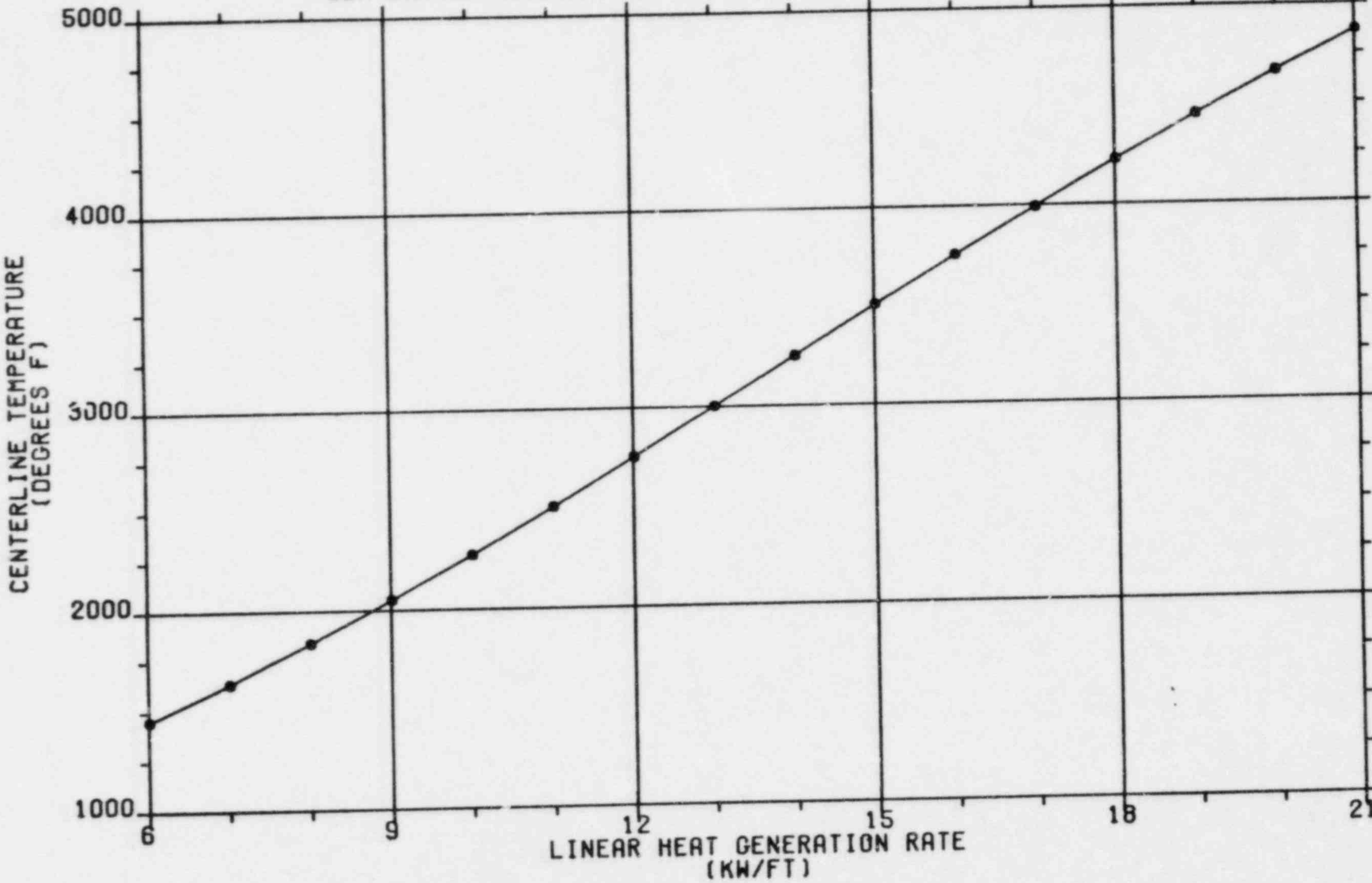
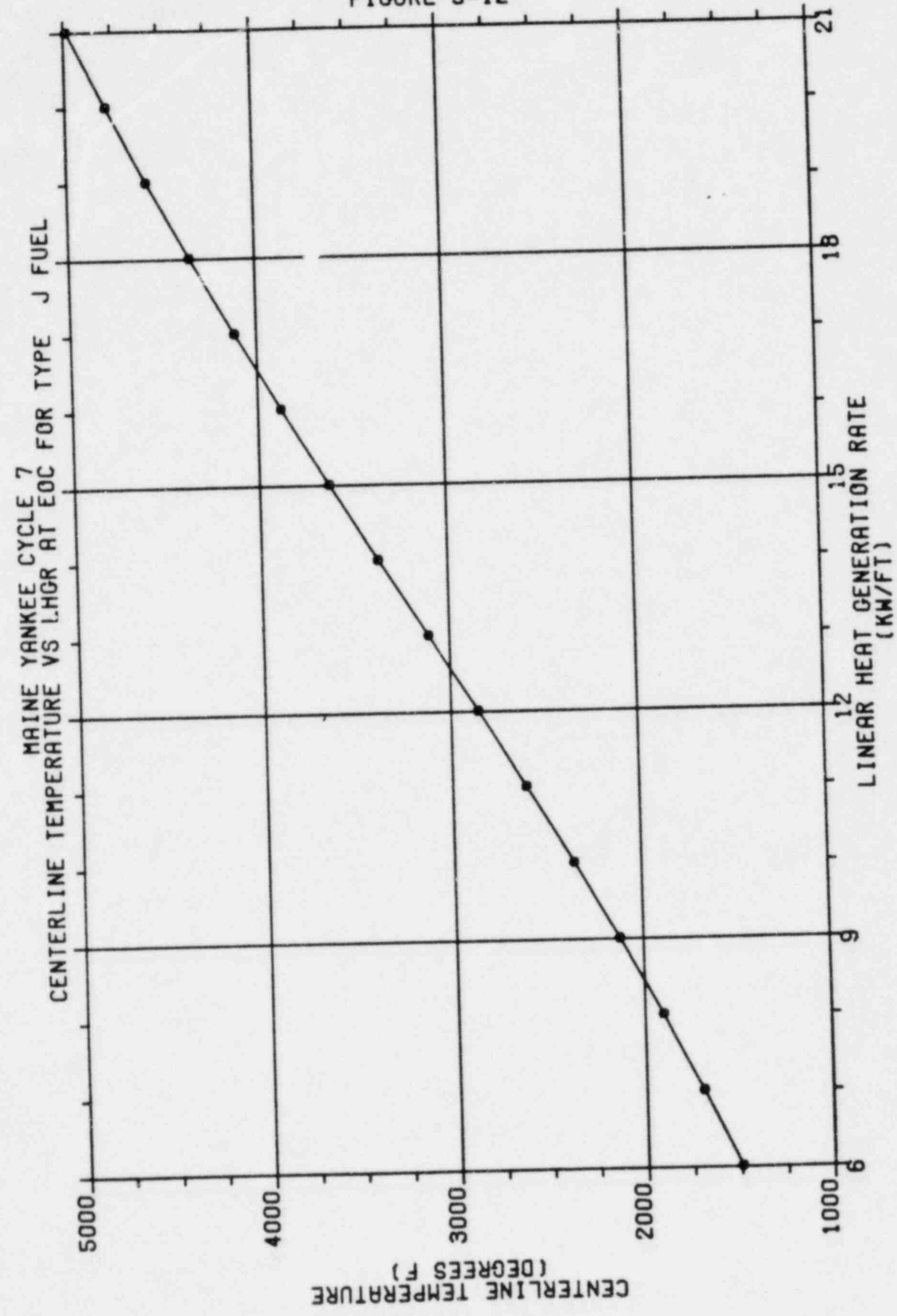


FIGURE 3-12


MAINE YANKEE CYCLE 7
CENTERLINE TEMPERATURE VS LHGR AT EOC FOR TYPE J FUEL
LINEAR HEAT GENERATION RATE
(KW/FT)
6
9
12
15
18
21
CENTERLINE TEMPERATURE
(DEGREES F)
1000
2000
3000
4000
5000

4.0 PHYSICS ANALYSIS

4.1 Fuel Management

Maine Yankee Cycle 7 consists of irradiated and fresh fuel assemblies as described in Section 3.1.1. The core layout is given in Figure 3.2. Cycle 7 is expected to attain a cycle average full power lifetime of 12,000 MWD/MT. A low-leakage loading pattern is employed, as described in Section 3.1.3.

4.2 Core Physics Characteristics

The primary physics characteristics of the Reference Cycle (Cycle 3), Cycle 6 and Cycle 7 are given in Table 4.1. The Cycle 7 characteristics differ from those of Cycles 3 and 6 based on the following significant changes, as discussed previously:

1) Increased Type L fuel enrichment;

2) Low-leakage fuel management;

3) Increased use of burnable poison shims in the Type L fuel; and

4) Increased reactivity worth resulting from the reconfiguration of CEA Bank 5.

These different physics characteristics are discussed in the following sections.

4.3 Power Distributions

Assembly relative power densities for Cycle 7 at Hot Full Power (HFP), equilibrium xenon conditions are presented for unrodded and rodded (CEA Bank 5 In) configurations. Figure 3.5 shows the locations of the CEA groups.

The unrodded power distributions at BOC (50 MWD/MT), MOC (6000 MWD/MT) and EOC (12,000 MWD/MT) are presented in Figures 4.1 through 4.3. The unrodded maximum 1-pin radial peak power is 1.492 at BOC, 1.438 at MOC, and 1.446 at EOC. The rodded (CEA Bank 5 In) power distributions at BOC, MOC and EOC are presented in Figures 4.4 through 4.6. The rodded maximum 1-pin radial peak power is 1.581 at BOC, 1.552 at MOC, and 1.560 at EOC for spot Bank 5 insertion at these times in core life.

The unrodded radial peaking is comparable to previous cycles. The CEA Bank 5 radial peaking is increased due to the CEA Bank 5 modifications by 5 to 10% in comparison to Cycle 6. This is reflected in a more restrictive insertion allowance for CEA Bank 5, as discussed in Section 4.9.1.

The allowable unrodded radial peaking (with uncertainties) versus exposure for Cycle 7 is included in the plant Technical Specifications for the purpose of comparison to measured values. Such a comparison demonstrates the adequacy of the peaking values utilized in the safety analyses. The values are shown in Figure 4.7 and show that the maximum radial peaking occurs early in the cycle.

The unrodded core power distributions are slightly asymmetric due to non-octant symmetric burnup gradients across the octant and quadrant boundary assemblies. The quadrant analysis presented overpredicts the slight asymmetry the full core will exhibit, providing a conservative analysis of the peaking effects.

Subsequent to the design analysis, it was determined that the as-built burnable poison shims were fabricated in three separate batch lots, as discussed in Section 3.1.2. The average poison loading of the lots were 24.62, 23.96, and 23.08 mg B-10 per inch, relative to the design specification of 23.8 mg B-10 per inch.

To minimize the impact on the design analysis, the shims were preferentially loaded in the specific assembly locations detailed in Figure 4.8. The as-built depletion analysis was performed and the effect on the maximum 1-pin radial peaking is given in Table 4.2. The differences are minimal and have been conservatively accounted for in safety analyses.

4.4 CEA Group Reactivity Worths

The CEA group configurations were shown in Figure 3.5 and the modifications to the lead regulating CEA Bank 5 were discussed in Section 3.1.5. The CEA group worths at HFP are presented in Table 4.3 for Cycles 3, 6, and 7. The CEA Bank 5 modifications provide the largest increase in CEA group worth compared to previous cycles. The low-leakage fuel management also contributes to a general increase in CEA worths due to power distribution weighting effects. CEA group reactivity worths are verified by the startup test program and the associated acceptance criteria.

4.5 Doppler Reactivity Coefficients and Defects

The fuel temperature, or Doppler, components of reactivity are presented in Tables 4.4 and 4.5 for nominal conditions in Cycles 3, 6, and 7. The total core average Doppler defect from 4000^{o}F is given in Table 4.4 and the core average Doppler coefficient in Table 4.5. The values in Cycles 3, 6, and 7 are similar. Uncertainties of 25% are conservatively applied to the coefficient and defect values prior to transient analysis applications.

Separate core average Doppler defect calculations appropriate for application to the ejected CEA cases have been performed for Cycle 7 and are discussed in detail in Section 4.10.2. The calculations utilize the 3D pre-ejected configuration weighting to obtain a more representative Doppler defect. The full and zero power case Doppler defects for the ejected CEA cases are given in Tables 4.6 and 4.7 for specified times in core life. As discussed in Section 4.10.2.3, an uncertainty of 15% is applied prior to transient analysis application of these defects.

4.6 Moderator Reactivity Coefficients and Defects

The moderator temperature coefficients at nominal operating HFP and HZP, critical boron conditions are presented in Table 4.8 for Cycles 3, 6, and 7. Relative to Cycle 6, the Cycle 7 MTC at BOC, HZP is less positive, due primarily to the increased fresh fuel enrichment and number of burnable poison shims. These factors overcome the increased critical boron level at HZP for Cycle 7 (1244 vs. 1180 ppm, from Table 4.1). The end of Cycle 7 MTC values are less negative than those of Cycle 6, due primarily to the power distribution differences of the low-leakage core design. An uncertainty of $\pm$ 0.5 x 10^{-4} $\Delta\rho/^{o}F$ is conservatively applied to MTC values prior to transient analysis use and the startup test program demonstrates the validity of such values.

The moderator defect appropriate to the scrammed (ARI) less worst stuck CEA configuration is given in Table 4.9 for Cycles 3, 6, and 7. This defect curve yields a conservative moderator reactivity increase versus temperature or density, while accounting for the effects of loss in total CEA worth and the worst stuck CEA. Starting in Cycle 6, this calculation has been performed at BOC, high soluble boron and EOC, no soluble boron conditions. Since the soluble boron concentration is the most significant factor in determining the moderator defect, a boron-concentration dependent minimum required shutdown margin has been incorporated in the plant Technical Specifications, as discussed in Section 4.9.5. An uncertainty of 15% is applied to the moderator defect values in cooldown transients from HZP. An uncertainty of 25% is applied in cooldown transients from HFP, where moderator redistribution effects are an additional reactivity component.

4.7 Soluble Boron and Burnable Poison Reactivity Effects

The soluble boron and burnable poison shim reactivity effects are shown in Table 4.1 for Cycles 3, 6, and 7. The critical boron concentrations for Cycle 7 at BOC are higher than those of Cycle 6, due primarily to the increased fresh fuel enrichment. The increased number and worth of burnable poison shims in Cycle 7 have limited this critical soluble boron difference. The inverse boron worths for Cycles 6 and 7 are comparable and reflect the different critical boron concentrations.

4.8 Kinetics Parameters

The total delayed neutron fractions and prompt neutron generation time for Cycles 3, 6, and 7 are presented in Table 4.1. The values are comparable and the differences reflect the effects of core average exposure and power weighting. Table 4.10 details the delayed neutron fractions and lifetimes by delayed neutron group for Cycles 3 and 7 at HFP, ARO conditions. Kinetics parameters for HFP and HZP conditions, both unrodded and rodded, are calculated for appropriate application in transient analysis cases and a 10% uncertainty is applied in a conservative manner.

4.9 Safety-Related Characteristics

4.9.1 CEA Group Insertion Limits

The CEA group insertion limits are given in the Technical Specifications and Figure 4.9. The Power-Dependent Insertion Limit (PDIL) for CEAs provides for sufficient available scram reactivity at all power levels and times-in-cycle-life. It also specifies the allowable CEA configurations for scoping analysis of dropped, ejected, and withdrawn CEAs. The CEA group insertion limits are more restrictive than previous cycles, due primarily to the modification of CEA Bank 5.

4.9.2 CEA Ejection Results

The calculated worths and planar radial maximum 1-pin powers resulting from the worst ejected CEAs for beginning, middle, and end-of-cycle are shown in Table 4.11 for Cycle 7. Hot full and hot zero power conditions are considered. No credit is taken for feedback effects in these calculations.

The Cycle 7 values are increased relative to previous cycles, due primarily to the increased reactivity worth of the modified CEA Bank 5. Section 4.10 addresses the CEA ejection physics methodology and methods modifications.

4.9.3 CEA Drop Results

4.9.3.1 Design Analysis Results

The calculated worths of the most limiting dropped CEAs for Cycles 6 and 7, with the resulting maximum 1-pin radial powers, are given in Tables 4.12 and 4.13 for beginning and end-of-cycle. Since Cycle 4, this analysis has utilized a local pinwise Doppler feedback methodology which was verified by a special at-power CEA drop test performed during the Cycle 4 startup physics tests (27).

The calculations are performed for all CEA drops at potentially limiting power levels, determined to be greater than 60% power for Cycle 7. CEA drops from ARO and CEA Bank 5 inserted are those considered for Cycle 7 based on the CEA insertion limits. Due to the more restrictive CEA insertion limits for Cycle 7, CEA drops from the CEA Banks 5 + 4 inserted configuration are not considered as they were in previous cycles.

The CEA drop results in Tables 4.12 and 4.13 are compared for Doppler feedback conditions of 80% of rated thermal power. In Cycles 5 and 6, CEA drops from this power level were determined to be the most limiting in terms of the RPS setpoint analysis. For Cycle 7, detailed separate envelopes of maximum percent increase in radial peaking versus reactivity worth of the dropped CEA are calculated for various power levels and presented in Figure 4.10. The results are similar to those witnessed in Cycle 6, also presented in the figure.

In the design analysis for dropped CEAs, the 2D radial peaking increases in Figure 4.10 are combined with the most limiting radial and axial peaking allowed by the symmetric offset limits to obtain total peaking for the given power level. This peaking, increased by 10% for uncertainties, is accommodated in the RPS setpoint generation in Section 6.0. The design analysis is based on the instantaneous peaking increase associated with a dropped CEA.

4.9.3.2 Post-CEA Drop Restrictions

During Cycle 6 operation, it was realized that the Technical Specifications allowed for operation of up to 4 hours with a dropped CEA without a required power reduction. Such a situation leads to further increased radial peaking due to preferential xenon burnout in the high-power core region far from the dropped CEA. This situation was reported to the NRC in (46) and corrective action taken for Cycle 6.

Analyses for Cycle 7 were performed to determine the required rate of power level reduction to be bounded by the present design analysis method in Section 4.9.3.1. Three-dimensional nodal calculations were performed to address this problem. The results, presented in this section, indicate that the following actions are required to maintain the core within the limits of the design analysis following a dropped CEA:

1) Decrease thermal power by at least 10% of rated power within one-half hour;

2) Decrease thermal power by at least 20% of rated power within one hour;

3) Maintain thermal power at or below this reduced power level; and

4) Limit CEA insertion to the insertion level corresponding to the pre-drop thermal power.

The present design analysis, utilizing 2D calculations to determine radial peaking for dropped CEAs, is conservative relative to a 3D analysis with appropriate feedbacks in that it yields both a higher dropped CEA worth and a higher maximum increase in radial peaking. Credit is taken for the conservatism in peaking to justify continued operation for one-half hour. The power reductions described above assure that proper limits are maintained for longer term operation. The plant Technical Specification changes for Cycle 7 reflect these restrictions.

The 2D and 3D calculational results for the worst core peripheral (CEA Type A dual) and core central (CEA Type B dual) CEA drops are presented in Figures 4.11 through 4.14 for beginning and end-of-cycle. The figures demonstrate that the radial peaking remains stable and within the limits of the design analysis with the above specified power level restrictions.

4.9.4 Available Scram Reactivity

The available scram reactivity from both Hot Full Power (HFP) and Hot Zero Power (HZP) conditions at BOC and EOC is tabulated in Table 4.14. In addition to uncertainties, allowances for the worst stuck CEA and the power dependent insertion limit for CEAs are included. The CEA programming allowance corresponds to the loss in available scram reactivity due to movement of all CEAs a maximum of 3 inches (4 steps) into the active core.

The required scram reactivity at the Hot Zero Power (HZP) condition is determined from the requirements of the steam line rupture analysis in Section 5.5.1 and the other safety analyses in Section 5. The required scram reactivity at HZP must be sufficient to prevent a return-to-criticality following the most limiting steam line rupture event from HZP. It also must be greater than assumed in other safety analyses from HZP. The available scram reactivity at HZP with uncertainties, from Table 4.14, must be greater than the required scram reactivity at HZP.

In addition, the required scram reactivity at HZP, when added to the additional scram reactivity provided by the CEA insertion limits versus power from Figure 4.9, must be sufficient to prevent a return-to-criticality following a steam line rupture event from any power level. It must also be greater than the value assumed in other safety analyses from at-power conditions.

The steam line rupture analyses are performed from both Hot Full Power (HFP) and HZP conditions (38). They explicitly account for the moderator defect as a function of moderator temperature, and Doppler defect as a function of fuel temperature, with the uncertainties stated in Sections 4.5 and 4.6. Other safety analyses are also performed from both HZP and HFP conditions. The CEA insertion limits versus power are designed to provide increased available scram reactivity proportional to the increased power level. This assures that intermediate power level conditions are covered by analysis of the HZP and HFP cases.

The steam line rupture analysis provides the minimum required worth in CEAs for cooldown events from HFP and HZP conditions to maintain subcriticality. In addition, other safety analyses have implicitly assumed minimum required worth in CEAs, as stated in Section 5.1.4. These are compared, in Table 4.15, to the available scram reactivity with uncertainties from Table 4.14. The table demonstrates that, in each condition and time in cycle life, the available scram reactivity with uncertainties is greater than the required scram reactivity.

Compliance with the startup test criteria on CEA worths in Section 7 demonstrates the available scram reactivity with uncertainties in Table 4.14. As such, it also demonstrates CEA worth in excess of the required scram reactivities.

The minimum required worth in CEAs for the steam line rupture analysis is calculated at typical beginning and end-of-cycle conditions, corresponding to Cycle 7 RCS soluble boron conditions of 910 and 0 ppm, respectively. The boron concentration determines the severity of the moderator temperature defect and has the most direct impact on the minimum required worth in CEAs. The result is that the minimum required shutdown margin, as discussed in the next section, can be expressed as a function of RCS soluble boron concentration in the Technical Specifications.

4.9.5 Shutdown Margin Requirements

Shutdown margin is defined as the sum of:

1) the reactivity by which the reactor is subcritical in its present condition, and

2) the reactivity associated with the withdrawn trippable CEAs less the reactivity associated with the highest worth withdrawn trippable CEA.

For a critical reactor, the shutdown margin must be maintained by sufficient available scram reactivity. The required and available scram reactivity comparison in Table 4.15 is the result of calculations which demonstrate adequate shutdown margin by bounding all the critical operating conditions for Cycle 7. This is so, provided the CEA insertion limits and assumptions inherent in them are fulfilled. These assumptions are:

1) the available scram reactivity calculations,

2) the operability of all trippable CEAs, and

3) the CEA drop time to 90% of full insertion in less than 2.7 seconds.

The shutdown margin requirement in the Technical Specifications for Cycle 7 is expressed in equation form as:

$SDM = 4.21 - 0.00310\ C + 2.53\ P$

when C is less than 326 ppm, and

$SDM = 3.20 + 2.53\ P$

when C is greater than or equal to 326 ppm

where

SDM is the required shutdown margin in percent reactivity

C is RCS boron concentration in ppm

P is reactor power level in fraction of rated thermal power.

This representation allows for calculation of the minimum required shutdown margin for any RCS boron concentration and power level. This shutdown margin representation is demonstrated, in Figure 4.15, to bound the required scram reactivities of Table 4.15 from both HFP and HZP conditions. Based on the discussion in Section 4.9.4, meeting the startup test criteria on CEA worths in Section 7 demonstrates the calculated available scram reactivity with uncertainties and thus demonstrates compliance with the required shutdown margin.

The minimum required shutdown margin is given for selected power levels in Figure 4.16 and the Technical Specifications to provide a well-defined requirement as a function of key plant parameters. This specification permits the development of procedures which preserve the minimum required shutdown margin. Under normal operating conditions, the CEA insertion limits provide such assurance. In the event of an inoperable or slow CEA, such procedures would apply.

4.9.6 Augmentation Factors

The set of augmentation factors applied to Cycle 7 has been determined for noncollapsed fuel clad using the calculational model described in (28). Augmentation factors have been conservatively calculated by statistically combining the single gap peaking factors of Figure 4.17 and the worst (i.e., flattest) radial quarter assembly power census during the cycle in Table 4.18. The complete Cycle 7 pin census at this time in cycle life is compared to the most limiting Cycle 6 pin census in Table 4.17.

The augmentation factors are calculated for changes in fuel density of 0.5, 1.5 and 2.5%. The 2.5% fuel density change is the maximum anticipated change and was the density change assumed for the Cycle 5 and 6 augmentation factors. For Cycle 7, resintering test data has been obtained for the Types J, K and L fuel assembly pellet lots which conservatively justifies use of augmentation factors appropriate to a 1.5% fuel density change. Table 4.16 provides the Cycle 7 augmentation factors for a 1.5% fuel density change and compares those to the Cycle 6 and 7 augmentation factors for a 2.5% fuel density change.

The augmentation factors are incorporated as a power spike penalty in all calculations of core power to incipient fuel centerline melt, as part of the RPS setpoint analyses in Section 6.0.

4.10 Methodology and Methodology Revisions

4.10.1 Summary of Physics Methodology Documentation

A summary of the reference report and supplemental documentation for the application of physics methodology to Maine Yankee since Cycle 3 is given in Table 4.19. The reference physics methodology report is YAEC-1115 (14).

4.10.2 CEA Ejection Analysis Physics Input

4.10.2.1 Background and Introduction

The CEA ejection analysis method employed since Cycle 3 contains conservatisms based on both analytical model assumptions and selection of input parameters. This method has adequately addressed the worst CEA ejections witnessed in Cycles 3-6.

The available scram reactivity requirements for Cycle 7 have increased so as to achieve an increased shutdown margin for the stream line break transient. To achieve this, the part-strength Bank 5 CEAs have been replaced for Cycle 7 with full-strength CEAs as described in Section 3.1.5. As a result, the CEA ejection physics parameters are more limiting for Cycle 7.

The nominal key physics parameters for the CEA ejection from Cycle 6, Cycle 7, and the FSAR analysis of Cycle 1 are given in Tables 4.20 and 4.21 for the worst ejections from full power and zero power, respectively. The key parameters of the worst cases, the FSAR EOC cases, bound those of Cycles 6 and 7. The FSAR analysis, however, used space-time analysis to remove conservatisms inherent in the point kinetics approach.

Changes are proposed in the physics input parameters to the CEA ejection analysis which are justified by the nature of the transient and supported by higher-order calculations.

4.10.2.2 Present Licensing Methodology

The CHIC-KIN program has been used in the analysis of the CEA ejection and seized rotor transients. Its application is described in (10). Improvements for Cycle 6 were described in (37). The code uses point kinetics reactivity assumptions in determining the core power response with time.

The referenced document for the physics parameter analysis is YAEC-1115 (14). This document describes the computer codes and their application in the calculation of these parameters. The following key physics parameter inputs are required for the CEA ejection analysis. Other physics input parameters are of lesser significance.

1) Determination of the worst static ejected CEA worth from HFP and HZP conditions at BOC and EOC. Sufficient scoping and judgment is applied to assure that these limiting ejections bound intermediate

power level and cycle burnup conditions. Static calculated worths are conservative relative to higher order space-time calculated worths. A 15% uncertainty is applied to the static ejected CEA worths for conservatism.

2) Determination of the worst pin power peaking and pin power census from the post-ejected static condition from HFP and HZP conditions at BOC and EOC. These cases have historically been the worst static ejected CEA worth cases. The static pin power peaking and census are more limiting than the higher order space-time calculated values. A 10% uncertainty is applied to the pin power peaking census for conservatism when comparing to the fuel failure limit.

3) A core average Doppler defect curve is supplied for each time in core life analyzed. The defect curve is typically from an unrodded case, in which local power weighting effects are not significant. A 25% uncertainty is applied to the Doppler defect for conservatism.

4) Delayed neutron parameters are supplied for each case analyzed. These parameters correspond to the pre-ejected condition. A 10% uncertainty is applied for conservatism to the delayed neutron parameters. The pre-ejection weighted delayed neutron parameters are typically conservative in comparison to post-ejection weighted parameters.

4.10.2.3 Changes to Licensing Methodology

There are two proposed changes to the licensing methodology, discussed in the following sections, dealing with the physics parameter inputs. The following remain unchanged:

1) The CHIC-KIN program and its application;

2) The static ejected worth, pin power peaking, pin power census and the stated uncertainties for each parameter;

3) The delayed neutron parameters and the stated uncertainties;

The proposed changes are described in the following sections.

4.10.2.3.1 Core Average Doppler Defect Calculation

The core average Doppler defect curve is typically calculated for the unrodded condition. As such, the local power weighting effects are minimal. The uncertainties are conservatively applied to the defect curve in the direction of worsening the effects of the given transient.

It is proposed that a core average Doppler defect curve with local power weighting based on the explicit pre-ejected power shape be applied for each of the ejected CEA cases. The core average Doppler defect curve thus derived is more representative of the given conditions than the unrodded curve. Most significant is that it is a conservative curve in application to the ejected CEA case. This is because the pre-ejected power weighting is the least peaked, or flattest shape witnessed during the transient.

This pre-ejected weighting results in typical increases in the core average Doppler defect of approximately 10-15% relative to the nominal unrodded weighting over the core average fuel temperature ranges of interest (1,000-2,000°F for HFP cases, 500-1,500°F for HZP cases). These defect curves for Cycle 7 are contained and described in Section 4.5.

4.10.2.3.2 Core Average Doppler Defect Uncertainty

The core average Doppler defect uncertainty typically included for transient analysis application is 25%. This uncertainty addresses items such as:

1) Uncertainties in cross sections and their temperature dependencies;

2) Uncertainties in burnups which affect isotopic distributions;

3) Uncertainties in nominal power distributions which influence the local Doppler reactivity weighting;

4) Uncertainties in the local doppler reactivity weighting during the course of the transient due to changing conditions.

For all transient cases, the uncertainties in Items (1)-(3) are applicable. For CEA ejections, however, the local Doppler reactivity weighting is a significant factor in limiting the transient. Thus, the component in Item (4) results in an important benefit in limiting the consequences of CEA ejections.

A number of studies of CEA ejections [(2), (44) and (45)] for typical PWR conditions have documented the magnitude of the local Doppler reactivity weighting by comparison of space-time to point kinetics analysis results. These studies typically define a spatial Doppler weighting factor. This factor may be defined as the multiplier to the nominal Doppler defect in the point kinetics calculation which yields the same total core energy response as the space-time (i.e., spatially weighted Doppler) solution.

Such analysis was performed for the explicit Cycle 1 worst ejected CEA cases from HFP and HZP by Combustion Engineering and documented in the FSAR (2). Subsequent licensing methods used by CE have demonstrated that this FSAR analysis technique is conservative [(44), Appendix B]. The FSAR analysis developed a linear relationship between the spatial Doppler weighting factor and ejected CEA worth for the same Bank 5 ejection locations which have been the most limiting in all subsequent cycles. The result was a doppler weighting factor (with 20% reduction for conservatism included) of 72% per dollar of ejected CEA worth. Stated another way, a spatial Doppler weighting factor of 1.72 would be applicable to an ejected CEA worth of one dollar.

A summary of the results of higher-order calculated spatial Doppler weighting factors for control rod ejections are given in Table 4.22. The spatial Doppler weighting factor for each case is expressed as percent increase per dollar of ejected reactivity for purposes of comparison. It is recognized that the details of the ejected CEA case determine the specific amount of applicable spatial Doppler weighting. Nevertheless, there is a sizeable spatial Doppler reactivity component which is correlated to the magnitude of the CEA ejection.

Based on the supporting higher-order calculation results and the particular nature of the transient, a reduction in the uncertainty applied to the core average Doppler defect from 25% to 15% is thus proposed, in application to CEA ejections only. Such a reduction recognizes the inherent conservatisms in the non-spatial point kinetics representation of the Doppler defect for this particular transient.

TABLE 4.1

MAINE YANKEE CYCLES 3, 6 and 7

NUCLEAR CHARACTERISTICS

	Cycle 3	Cycle 6	Cycle 7
Core Characteristics			
Exposure (MWD/MT)			
Core Average at BOC	7,000	11,100	11,400
Cycle Length at Full Power	10,200	10,800	12,000
Reactivity Coefficients - ARO			
Moderator Temperature Coefficient ($10^{-4}\Delta\rho/^{o}F$)			
HFP,BOC	-0.3[illegible]	-0.83	-0.61
HFP,EOC	-1.98*	-2.55	-2.30
Fuel Temperature Coefficient ($10^{-5}\Delta\rho\ /^{o}F$)			
HZP,BOC	-1.70	-1.69	-1.64
HFP,BOC	-1.30	-1.28	-1.29
HZP,EOC	-1.80	-1.78	-1.77
HFP,EOC	-1.37	-1.37	-1.39
Kinetics Parameters - ARO			
Total Delayed Neutron Fraction (β_{eff})			
HFP,BOC	0.00611	0.00582	0.00607
HFP,EOC	0.00517	0.00516	0.00518
Prompt Neutron Generation Time (10^{-6} Sec.)			
HFP,BOC	29.3	27.0	26.3
HFP,EOC	32.3	30.6	30.1
Control Characteristics			
Control Elements Assemblies			
Number Full/Part Length	77/8	77/0	81/0**
Total CEA Scrammable Worth (%$\Delta\rho$)			
HFP,BOC	9.18	8.44	9.03
HFP,EOC	9.56	9.52	10.04
Burnable Poison Rods			
Number B_4C-Al_2O_3/Borosilicate Glass	756/0	448/16	768/0
Total Worth at HFP,BOC (%$\Delta\rho$)	1.4	1.2	1.9
Critical Soluble Boron at BOC,ARO (ppm)			
HZP,NoXe,PkSm	1,075	1,180	1,244
HFP,NoXe,PkSm	995	1,084	1,145
HFP,Equilibrium Xe	782	855	909
Inverse Boron Worths (ppm/%$\Delta\rho$)			
HZP,BOC	84	94	96
HFP,BOC	89	100	102
HZP,EOC	74	80	82
HFP,EOC	79	87	87

* Conditions of 2440 MWt/2100 psia operation.

** Four full-length CEA's are non-Scrammable in Cycle 7.

Table 4.2

Maine Yankee Cycle 7

Maximum Radial Relative Pin Power
Design and As-Built Shim Loadings
HFP, ARO Equilibrium Conditions

Cycle Burnup MWD/MT	Maximum Radial Relative Pin Power (INCA location) Design	As-Built	Percent Difference*
50	1.492(10)	1.490(10)	-0.1
500	1.468(10)	1.466(10)	-0.1
1K	1.471(10)	1.469(10)	-0.1
2K	1.467(10)	1.465(10)	-0.1
4K	1.455(10)	1.452(10)	-0.2
6K	1.438(10)	1.435(10)	-0.2
8K	1.427(10)	1.424(29)	-0.2
10K	1.437(10)	1.443(29)	+0.4
12K	1.446(29)	1.449(29)	+0.2

* Percent Difference = $\frac{\text{(As-Built)-(Design)}}{\text{(Design)}} \times 100$

TABLE 4.3

MAINE YANKEE CYCLES 3, 6 and 7

CEA GROUP WORTHS AT HFP

	Worths (%Δρ)					
	Cycle 3		Cycle 6		Cycle 7	
	BOC	EOC	BOC	EOC	BOC	EOC
Shutdown CEA Groups						
Banks C + B + A	5.86	6.02	5.12	6.02	5.39	6.17
Regulating CEA Groups						
Bank 5*	0.55	0.64	0.49	0.68	1.17	1.45
Banks 5 + 4	0.90	0.97	0.79	0.95	1.59	1.83
Banks 5 + 4 + 3	1.74	1.90	1.62	1.83	2.46	2.76
Banks 5 + 4 + 3 + 2	2.49	2.73	2.58	2.54	3.19	3.40
Banks 5 + 4 + 3 + 2 + 1	3.32	3.54	3.32	3.50	4.31	4.78
All CEA Groups						
Banks 5 + 4 + 3 + 2 + 1 + C + B + A	9.18	9.56	8.44	9.52	9.70	10.95

* Bank 5 has been redesigned for Cycle 7 to provide additional reactivity worth.

TABLE 4.4

MAINE YANKEE CYCLES 3, 6, 7
CORE AVERAGE DOPPLER DEFECT

Doppler Defect (x $10^{-4}\,\Delta\rho$)

Fuel Resonance Temperature ^{o}F	Cycle 3* BOC	Cycle 3* EOC	Cycle 6 BOC	Cycle 6 EOC	Cycle 7 BOC	Cycle 7 EOC
4000	0	0	0	0	0	0
3750	19.4	20.9	-	-	-	-
3500	39.4	42.5	41.1	44.4	41.6	45.0
3250	59.9	64.8	-	-	-	-
3000	81.2	87.8	84.7	91.6	85.6	92.7
2750	103.1	111.6	-	-	-	-
2500	125.9	136.2	131.4	142.1	132.7	143.6
2250	149.7	161.9	-	-	-	-
2000	174.5	188.6	182.0	197.0	183.8	198.8
1750	200.5	216.7	-	-	-	-
1500	228.1	246.5	237.8	257.3	239.9	259.6
1232			270.5	292.6	273.0	295.1
1000	288.7	311.9	301.3	325.6	303.5	328.0
800			329.5	356.1	331.8	358.5
532	358.5*	387.4*	372.1	402.2	373.2	403.2
300	394.3	426.0	411.4	444.3	413.8	446.9
200	-	-	430.7	465.1	433.1	467.6
100	-	-	-	-	453.9	490.0
0	-	-	-	-	476.6	514.4

* at 525^{o}F

TABLE 4.5

MAINE YANKEE CYCLES 3, 6, 7
CORE AVERAGE DOPPLER COEFFICIENT

Doppler Coefficient (x $10^{-4} \Delta\rho/^{o}F$)

Fuel Resonance Temperature ^{o}F	Cycle 3* BOC	Cycle 3* EOC	Cycle 6 BOC	Cycle 6 EOC	Cycle 7 BOC	Cycle 7 EOC
100	-	-	-	-	.2170	.2336
200	-	-	-	-	.2005	.2153
300	-	-	.187	.2015	.1871	.2011
400	-	-	.176	.1890	.1765	.1901
532	.159*	.172*	.169	.1778	.1642	.1770
576.4	-	-	.165	.1742	.1602	.1730
800	.144**	.156**	.147	.1585	.1470	.1584
1000	.131	.141	.137	.1470	.1378	.1485
1232	.121***	.121***	.128	.1370	.1287	.1391
1500	.114	.123	.117	.1265	.1192	.1288
2000	.102	.110	.1065	.1155	.1073	.1159
2500	.093	.101	.0970	.1055	.0982	.1061
3000	.086	.094	.0900	.0975	.0911	.0986
3500	.081	.088	.0845	.0915	.0856	.0926

* at 525^{o}F
** at 750^{o}F
*** at 1250^{o}F

TABLE 4.6

MAINE YANKEE CYCLE 7
CORE AVERAGE DOPPLER DEFECT
WITH PRE-EJECTED WEIGHTING
FOR FULL POWER CEA EJECTION

Fuel Resonance Temperature ^oF	Doppler Defect ($\times 10^{-4} \Delta\rho$) With Bank 5 Inserted BOC 50 MWD/MT	MOC 6K MWD/MT	EOC 12K MWD/MT
4000	0.0	0.0	0.0
3500	46.3	47.9	49.5
3000	95.8	98.9	102.1
2500	149.3	153.8	158.6
2000	207.9	213.5	220.0
1500	273.5	279.8	288.1
1232	312.6	319.1	328.2
1000	349.3	355.8	365.6
532	433.8	439.7	451.3

TABLE 4.7

MAINE YANKEE CYCLE 7
CORE AVERAGE DOPPLER DEFECT
WITH PRE-EJECTED WEIGHTING
FOR ZERO POWER CEA EJECTION

Fuel Resonance Temperature °F	Doppler Defect ($\times 10^{-4} \Delta\rho$) With Banks 5+4+3 Inserted BOC 0 MWD/MT	MOC 6K MWD/MT	EOC 12K MWD/MT
4000	0.0	0.0	0.0
3500	35.9	23.9	14.3
3000	76.3	50.2	30.1
2500	123.1	79.5	47.7
2000	179.0	112.8	67.9
1500	247.6	152.3	92.3
1232	291.1	178.5	108.7
1000	333.4	208.5	129.9
800	374.2	246.5	169.6
600	420.0	297.4	235.7
532	432.6	310.9	251.8

TABLE 4.8

MAINE YANKEE CYCLES 3, 6 AND 7
MODERATOR TEMPERATURE COEFFICIENTS

Conditions: HFP and HZP, ARO, Critical Boron

MTC ($10^{-4}\Delta\rho/^{o}F$)

Case Conditions	Cycle 3* BOC	Cycle 3* EOC	Cycle 6 BOC	Cycle 6 EOC	Cycle 7 BOC	Cycle 7 EOC
HFP, EqXe, EqSm	-0.47	-2.24	-0.83	-2.55	-0.61	-2.30
HZP, NoXe, PkSm	+0.24	-	+0.34	-	+0.21	-
HZP, NoXe, EqSm	-	-1.40	-	-1.45	-	-1.12

* Cycle 3 HZP values referenced to 525^{o}F, Cycles 6 and 7 HZP values referenced to 532^{o}F.

TABLE 4.9

MAINE YANKEE CYCLES 3, 6, 7
ARI MODERATOR DEFECT WITH WORST STUCK CEA

Moderator Temperature °F	Moderator Defect ($\times 10^{-4}\Delta\rho$)** Cycle 3* BOC	Cycle 3* EOC	Cycle 6 BOC	Cycle 6 EOC	Cycle 7 BOC	Cycle 7 EOC
	-	0 ppm	848 ppm	0 ppm	910 ppm	0 ppm
576.4	-	-138.0	-49.5	-134.7	-57.2	-129.2
532 - (525)*	-	0.0	0.0	0.0	0.0	0.0
500	-	43.5	16.5	62.5	19.6	60.8
450	-	111.0	34.8	138.8	31.5	132.0
400	-	167.3	49.4	197.7	44.5	190.0
350	-	217.0	64.5	244.8	54.0	243.0
300	-	261.6	83.2	327.0	64.0	292.2
250	-	298.5	104.0	369.5	75.0	335.0
200	-	-	-	-	82.0	370.0
150	-	-	-	-	88.0	400.0
100	-	-	-	-	91.5	423.0
68	-	-	-	-	93.9	430.9

* Cycle 3 values referenced to 0 at 525°F.

** Moderator defect at a constant 2250 psia for the specified temperatures.

TABLE 4.10

MAINE YANKEE CYCLES 3 and 7

KINETICS PARAMETERS

Conditions: HFP, ARO, Critical Boron

Time in Cycle Life	Delayed Neutron Group	Cycle 3 Effective Fraction	Cycle 3 Lifetime (Sec^{-1})	Cycle 7 Effective Fraction	Cycle 7 Lifetime (Sec^{-1})
BOC	1	0.00018	0.0126	0.00018	0.0126
	2	0.00128	0.0305	0.00128	0.0305
	3	0.00116	0.1163	0.00116	0.1167
	4	0.00237	0.3116	0.00236	0.3129
	5	0.00083	1.1652	0.00082	1.1716
	6	0.00029	3.0253	0.00029	3.0220
	TOTAL	0.00611		0.00607	
EOC	1	0.00014	0.0126	0.00014	0.0127
	2	0.00111	0.0304	0.00112	0.0304
	3	0.00097	0.1193	0.00098	0.1196
	4	0.00197	0.3185	0.00198	0.3196
	5	0.00072	1.1833	0.00072	1.1910
	6	0.00025	2.9831	0.00024	2.9857
	TOTAL	0.00517		0.00518	

TABLE 4.11

MAINE YANKEE CYCLE 7

CEA EJECTION RESULTS

	Time in Core Life		
	BOC	MOC	EOC
Maximum 1-Pin Radial Peak			
HFP Bank 5 In Ejected 5 (INCA Location 20)	3.11	3.72	4.04
HZP Banks 5 + 4 + 3 In Ejected 5 (INCA Location 34)	6.67	6.81	7.83
Maximum Ejected Worth (%$\Delta\rho$)			
HFP Bank 5 In Ejected 5 (INCA Location 20)	0.234	0.321	0.364
HZP Banks 5 + 4 + 3 In Ejected 5 (INCA Location 34)	0.514	0.559	0.692

<u>TABLE 4.12</u>

<u>MAINE YANKEE CYCLES 6 AND 7</u>
<u>CEA DROP RESULTS AT BOC</u>

CEA Group Positions Before Drop	Dropped CEA Type	Dropped CEA Worth (%Δρ) Cycle 6	Dropped CEA Worth (%Δρ) Cycle 7	Maximum 1-Pin Radial Power* Cycle 6	Maximum 1-Pin Radial Power* Cycle 7
ARO	A	0.106	0.107	1.60	1.69
ARO	B	0.158	0.163	1.61	1.70
ARO	C	0.090	0.106	1.56	1.69
ARO	1	0.051	0.070	1.52	1.61
Bank 5 In**	A	0.106	0.098	1.70	1.78
Bank 5 In	B	0.150	0.186	1.71	1.88
Bank 5 In	C	0.092	0.108	1.66	1.80
Bank 5 In	1	0.047	0.066	1.62	1.70
Banks 5+4 In	A	0.124	-	1.84	-
Banks 5+4 In	B	0.087	-	1.79	-
Banks 5+4 In	C	0.111	-	1.81	-
Banks 5+4 In	1	0.061	-	1.75	-

* Pre-Drop Maximum 1-Pin radial powers:

	ARO	Bank 5 In	Banks 5+4 In
Cycle 6	1.420	1.506	1.608
Cycle 7	1.492	1.581	-

Post-Drop Maximum 1-Pin radial power represent 80% of 2630 MWt conditions.

** Bank 5 configuration changed for Cycle 7.

TABLE 4.13

MAINE YANKEE CYCLES 6 AND 7
CEA DROP RESULTS AT EOC

CEA Group Positions Before Drop	Dropped CEA Type	Dropped CEA Worth (%$\Delta\rho$) Cycle 6	Dropped CEA Worth (%$\Delta\rho$) Cycle 7	Maximum 1-Pin Radial Power* Cycle 6	Maximum 1-Pin Radial Power* Cycle 7
ARO	A	0.113	0.114	1.65	1.64
ARO	B	0.147	0.166	1.66	1.65
ARO	C	0.102	0.106	1.63	1.62
ARO	1	0.068	0.078	1.59	1.57
Bank 5 In**	A	0.114	0.105	1.63	1.75
Bank 5 In	B	0.148	0.181	1.65	1.79
Bank 5 In	C	0.104	0.112	1.61	1.76
Bank 5 In	1	0.063	0.077	1.57	1.69
Banks 5+4 In	A	0.132	-	1.85	-
Banks 5+4 In	B	0.080	-	1.79	-
Banks 5+4 In	C	0.123	-	1.84	-
Banks 5+4 In	1	0.078	-	1.78	-

* Pre-Drop Maximum 1-Pin radial powers:

	ARO	Bank 5 In	Banks 5+4 In
Cycle 6	1.469	1.452	1.617
Cycle 7	1.446	1.560	-

Post-Drop Maximum 1-Pin radial power represent 80% of 2630 MWt conditions.

** Bank 5 configuration changed for Cycle 7.

TABLE 4.14

MAINE YANKEE CYCLE 7

AVAILABLE SCRAM REACTIVITY

	Worths (%$\Delta\rho$)			
	BOC		EOC	
	HFP	HZP	HFP	HZP
Scrammable CEA Worth*	9.13	8.66	10.04	9.53
Stuck CEA Worth	1.65	1.57	1.78	1.70
PDIL CEA Worth	0.07	2.00	0.15	2.56
CEA Programming Allowance	0.04	0.04	0.11	0.25
Available Scram CEA Worth - Nominal	7.37	5.05	8.00	5.02
- Including Uncertainties	6.63	4.55	7.20	4.52

* ARI CEA worth less non-scrammable CEA worth (Bank 5 2-finger CEAs).

TABLE 4.15

MAINE YANKEE CYCLE 7
REQUIRED SCRAM REACTIVITY

	Worth (%Δρ) for Time in Cycle Life and RCS Soluble Boron Concentration			
	BOC 910 ppm		EOC 0 ppm	
	HFP	HZP	HFP	HZP
Available Scram Reactivity With Uncertainties (Table 4.14)	6.63	4.55	7.20	4.52
Minimum Required Worth in CEAs Assumed				
- Steam Line Rupture Event (Section 5.5.1)	3.05	1.39	6.74	4.21
- Safety Analyses (Section 5)	5.70	3.20	5.70	3.20
Required Scram Reactivity[1]	5.70	3.20	6.74	4.21
Excess from Required to Available Scram Reactivity	0.93	1.35	0.46	0.31

[1]Maximum of either the minimum required worth in CEAs assumed for the steam line rupture event or other safety analyses in Section 5.

TABLE 4.16

MAINE YANKEE CYCLES 6 AND 7
AUGMENTATION FACTORS

Core Height (%)	Core Height (inches)	Change in Fuel Density Due to Densification 2.5% Cycle 6	2.5% Cycle 7	1.5% Cycle 7
98.5	134.7	1.057	1.059	1.048
86.8	118.6	1.051	1.052	1.042
77.9	106.5	1.047	1.046	1.038
66.2	90.5	1.041	1.040	1.033
54.4	74.4	1.035	1.033	1.028
45.6	62.3	1.030	1.028	1.025
33.8	46.2	1.024	1.021	1.020
22.1	30.2	1.017	1.015	1.015
13.2	18.1	1.011	1.010	1.010
1.5	2.0	1.001	1.001	1.001

TABLE 4.17

MAINE YANKEE CYCLES 6 AND 7
CORE RADIAL PIN POWER CENSUS
FOR AUGMENTATION FACTOR CALCULATION*

Radial Pin Power Interval	Number of Pins in Core: Cycle 6 at 10,000 MWD/MT	Number of Pins in Core: Cycle 7 at 500 MWD/MT
1.401 - 1.469	340	264
1.301 - 1.400	1172	1716
1.201 - 1.300	2200	5424
1.101 - 1.200	5236	8424
1.001 - 1.100	10992	6020
0 - 1.000	17788	15560
Total	37728	37408

*Flattest pin cens is during the cycle.

TABLE 4.18

MAINE YANKEE CYCLES 6 AND 7 QUARTER ASSEMBLY POWER CENSUS FOR AUGMENTATION FACTOR CALCULATION*

Cycle 6 at 10,000 MWD/MT		Cycle 7 at 500 MWD/MT	
Power Density	Number of Quarter Assemblies	Power Density	Number of Quarter Assemblies
1.3488	4	1.2849	8
1.3486	12	1.2830	8
1.3427	4	1.2748	16
1.3422	4	1.2744	4
1.3253	4	1.2665	8
1.3249	4	1.2645	8
1.3215	4	1.2628	8
1.3208	4		

*Flattest quarter assembly census during the cycle.

Table 4.19

Maine Yankee Physics Methodology Documentation

Description of Methodology	Supporting Documentation	Reference	Application in Cycle
Reactor Physics Methods - Reference Report	YAEC-1115	14	3
Reactor Protective System Setpoint Analysis - Reference Report	YAEC-1110	8	3
Extension of Fine Mesh Diffusion Theory and Nodal Physics Methods to Reactivity Parameter Calculations and a Change in the Nodal Neutronic Coupling Model	PC No. 64, Section 4.8 WMY 78-102, Attachment B	7 27	4
Introduction of Local Pointwise Doppler Feedback Effects in Two-Dimensional Pinwise Diffusion Theory Calculations for Dropped CEAs and Special CEA Drop Test at 50% Power for Method Verification	PC No. 64, Section 4.8 WMY 78-102, Attachment C	7 27	4
Uncertainty Applied to Moderator Reactivity Defect from Hot Zero Power Reduced from 25 to 15%	PC No. 84, YAEC-125[illegible], Section 4.7	37	6

Table 4.20

Maine Yankee
Comparison of Full Power CEA Ejection
Nominal Physics Parameters

	Time in Cycle Life					
	BOC			EOC		
Parameter	Cycle 6	Cycle 7	FSAR	Cycle 6	Cycle 7	FSAR
Delayed Neutron Fraction	0.0058	0.0061	0.0069	0.0052	0.0052	0.0052
Ejected CEA Worth (% Delta Rho)	0.09	0.23	0.36	0.19	0.36	0.39
Ejected CEA Reactivity in Dollars	0.16	0.38	0.52	0.37	0.69	0.75*
Maximum 1-Pin Radial Peak	2.19	3.11	3.85	3.28	4.04	4.58*

* Worst reactivity and peaking case from HFP for these cycles.

Table 4.21

Maine Yankee
Comparison of Zero Power CEA Ejection
Nominal Physics Parameters

Parameter	Time in Cycle Life					
	BOC			EOC		
	Cycle 6	Cycle 7	FSAR	Cycle 6	Cycle 7	FSAR
Del. ved Neutron Fraction	0.0058	0.0064	0.0069	0.0052	0.0053	0.0052
Ejected CEA Worth (% Delta Rho)	0.21	0.51	0.55	0.34	0.69	1.07
Ejected CEA Reactivity in Dollars	0.36	0.80	0.80	0.65	1.30	2.06*
Maximum 1-Pin Radial Peak	4.55	6.67	5.85	6.25	7.83	7.93*

* Worst reactivity and peaking case from HZP for these cycles.

Table 4.22

Comparison of Doppler Weighting Factors For Various Control Rod Ejection Analyses

Analysis (Reference)	Conditions	Ejected Worth (Dollars)	Doppler Weighting Factor (DWF)	% Increased Doppler per Dollar Ejected
MY FSAR (2)	HFP	0.58	1.43	72*
		0.83	1.60	72*
	HZP	0.87	1.62	72*
		2.27	2.64	72*
CE (44)	HZP	1.61	1.98	61**
B&W (45)	HFP	0.79	1.81	103
	HZP	1.31	2.85	141

* 20% conservatism included.

** Conservatism included - unspecified.

Figure 4.1
Maine Yankee Cycle 7
Assembly Relative Power Densities
BOC (50 MWD/MT) , HFP, ARO

82/02/24. — 20.28.31. MYC7 50 MWD/MT (FDC6 @ 11.5K)

CORE SUMMARY OF MAXIMUM FUEL ASSEMBLY POWER

DESCRIPTION	MAX.VALUE	ASSEMBLY
ASSEMBLY AVG.	1.27448	39
MAX. FUEL ROD	1.49167	39
MAX. CHANNEL	1.42406	39

CORE POSITION	ASSEMBLY NUMBER	FUEL TYPE	CEA BANK TYPE	ASSEMBLY AVERAGE POWER	MAXIMUM FUEL ROD POWER	MAXIMUM CHANNEL POWER
A14	1	J-8		.38687	.69523	.67856
A12	2	L-0		.79286	1.19445	1.14592
B17	3	J-0		.40552	.84875	.81365
B16	4	L-0	C	.94184	1.45375	1.39162
B15	5	L-4		1.08128	1.47554	1.41428
B13	6	L-8	1	1.13284	1.42094	1.34710
B11	7	K-8		1.03731	1.14812	1.11776
C18	8	L-8		.64471	1.18473	1.14438
C17	9	L-8	A	1.04796	1.40461	1.33558
C16	10	K-0		1.19315	1.36925	1.31772
C15	11	K-0	C	1.22656	1.35820	1.30966
C13	12	J-0		.98070	1.12052	1.11237
C11	13	K-8	5	1.08124	1.15700	1.11956
D19	14	L-8		.64480	1.18486	1.14451
D18	15	L-4	5	1.14655	1.43728	1.36659
D17	16	K-0		1.13277	1.30807	1.25429
D16	17	J-0	A	.91314	.99737	.99388
D15	18	L-8		1.27433	1.49130	1.42372
D13	19	J-0	3	.95686	1.03372	1.02717
D11	20	L-8		1.21072	1.38014	1.31095
E20	21	J-0		.40590	.84936	.81432
E19	22	L-8	A	1.04830	1.40499	1.33592
E18	23	K-0		1.13277	1.30817	1.25438
E17	24	J-8		.83154	.93395	.92781
E16	25	J-8		.84816	.91682	.91348
E15	26	K-0	2	1.14152	1.34838	1.31281
E13	27	J-0		.97809	1.09729	1.09188
E11	28	J-0		.96874	1.04368	1.03409
F20	29	L-0	C	.94227	1.45434	1.39219
F19	30	K-0		1.19372	1.36990	1.31832
F18	31	J-0	A	.91349	.99711	.99382
F17	32	J-8		.84762	.91739	.91346
F16	33	L-12	6	1.17292	1.39331	1.32701
F15	34	K-0		1.16873	1.33939	1.31210
F13	35	K-0		1.21660	1.33945	1.29582
F11	36	J-0		1.01366	1.09178	1.08178
G20	37	L-4		1.08160	1.47585	1.41458
G19	38	K-0	C	1.22684	1.35843	1.30987
G18	39	L-8		1.27448	1.49167	1.42406
G17	40	K-0	2	1.14175	1.34871	1.31310
G16	41	K-0		1.16852	1.33973	1.31239
G15	42	K-4	B	1.15250	1.28132	1.23321
G13	43	K-8		1.10509	1.22562	1.18308
G11	44	K-8	4	1.10994	1.18988	1.14566
H21	45	J-8		.38718	.69557	.67892
J20	46	L-8	1	1.13300	1.42113	1.34727
J19	47	J-0		.98056	1.12074	1.11259
J18	48	J-0	3	.95658	1.03332	..02681
J17	49	J-0		.97807	1.09760	1.09216
J16	50	E-0		1.21665	1.33914	1.29896
J15	51	K-8		1.10525	1.22582	1.18331
J13	52	J-4	B	.90810	.95129	.94384
J11	53	L-8		1.21904	1.39412	1.34507
K21	54	L-0		.79304	1.19464	1.14611
L20	55	K-8		1.03737	1.14812	1.11773
L19	56	K-8	5	1.08107	1.15688	1.11948
L18	57	L-8		1.21044	1.37982	1.31064
L17	58	J-0		.96844	1.04349	1.03389
L16	59	J-0		1.01304	1.09072	1.08146
L15	60	K-8	4	1.10993	1.16985	1.14556
L13	61	L-8		1.21898	1.39403	1.34501
L11	62	E-16	5	.99343	1.07946	1.03734

Figure 4.2
Maine Yankee Cycle 7
Assembly Relative Power Densities
MOC (6K MWD/MT), HFP; ARO

02/02/24. — 20.32.24. MYC7 6K MWD/MT (EOC6 @ 11.5K)

CORE SUMMARY OF MAXIMUM FUEL ASSEMBLY POWER

DESCRIPTION	MAX.VALUE	ASSEMBLY
ASSEMBLY AVG.	1.28889	18
MAX. FUEL ROD	1.43753	18
MAX. CHANNEL	1.39871	53

CORE POSITION/ASSEMBLY NUMBER	A14 1	A12 2
FUEL TYPE	J-8	L-0
CEA BANK TYPE		
ASSEMBLY AVERAGE POWER	.43238	.85852
MAXIMUM FUEL ROD POWER	.75161	1.25978
MAXIMUM CHANNEL POWER	.73832	1.22628

CORE POSITION/ASSEMBLY NUMBER	B17 3	B16 4	B15 5	B13 6	B11 7
FUEL TYPE	J-0	L-0	L-4	L-8	K-8
CEA BANK TYPE		• C •		• 1 •	
ASSEMBLY AVERAGE POWER	.42548	.90946	1.08947	1.22373	1.12413
MAXIMUM FUEL ROD POWER	.81976	1.33751	1.42686	1.42622	1.24518
MAXIMUM CHANNEL POWER	.79605	1.29471	1.38014	1.37580	1.19040

CORE POSITION/ASSEMBLY NUMBER	C18 8	C17 9	C16 10	C15 11	C13 12	C11 13
FUEL TYPE	L-8	L-8	K-0	K-0	J-0	K-8
CEA BANK TYPE		• A •		• C •		• 5 •
ASSEMBLY AVERAGE POWER	.70020	1.07950	1.12726	1.17008	1.00385	1.15100
MAXIMUM FUEL ROD POWER	1.19810	1.35828	1.24733	1.29286	1.10269	1.24411
MAXIMUM CHANNEL POWER	1.15825	1.30861	1.21703	1.26732	1.09564	1.18858

CORE POSITION/ASSEMBLY NUMBER	D19 14	D18 15	D17 16	D16 17	D15 18	D13 19	D11 20
FUEL TYPE	L-8	L-4	K-0	J-0	L-8	J-0	L-8
CEA BANK TYPE		• 5 •		• A •		• 3 •	
ASSEMBLY AVERAGE POWER	.70025	1.16441	1.10418	.90850	1.28889	.96528	1.26924
MAXIMUM FUEL ROD POWER	1.19815	1.38691	1.25342	.97016	1.43753	1.04521	1.41083
MAXIMUM CHANNEL POWER	1.15829	1.33497	1.21411	.96823	1.39031	1.04051	1.36435

CORE POSITION/ASSEMBLY NUMBER	E20 21	E19 22	E18 23	E17 24	E16 25	E15 26	E13 27	E11 28
FUEL TYPE	J-0	L-8	K-0	J-8	J-8	K-0	J-0	J-0
CEA BANK TYPE		• A •				• 2 •		
ASSEMBLY AVERAGE POWER	.42577	1.07966	1.10410	.84778	.86420	1.09184	.94411	.94513
MAXIMUM FUEL ROD POWER	.82020	1.35840	1.25345	.92526	.91511	1.22620	1.01801	1.02093
MAXIMUM CHANNEL POWER	.79654	1.30872	1.21413	.92270	.91251	1.20696	1.01457	1.01603

CORE POSITION/ASSEMBLY NUMBER	F20 29	F19 30	F18 31	F17 32	F16 33	F15 34	F13 35	F11 36
FUEL TYPE	L-0	K-0	J-0	J-8	L-12	K-0	K-0	J-0
CEA BANK TYPE	• C •		• A •		• 6 •			
ASSEMBLY AVERAGE POWER	.90965	1.12753	.90871	.86366	1.21903	1.09483	1.11835	.95753
MAXIMUM FUEL ROD POWER	1.33770	1.24761	.97030	.91541	1.39353	1.20572	1.21525	1.00924
MAXIMUM CHANNEL POWER	1.29491	1.21729	.96844	.91236	1.34629	1.19068	1.16287	1.00167

CORE POSITION/ASSEMBLY NUMBER	H21 45	G20 37	G19 38	G18 39	G17 40	G16 41	G15 42	G13 43	G11 44
FUEL TYPE	J-8	L-4	K-0	L-8	K-0	K-0	K-4	K-8	K-8
CEA BANK TYPE			• C •		• 2 •		• B •		• 4 •
ASSEMBLY AVERAGE POWER	.43265	1.08959	1.17014	1.28888	1.09196	1.09503	1.09203	1.07139	1.09534
MAXIMUM FUEL ROD POWER	.75186	1.42692	1.29288	1.43745	1.22639	1.20542	1.20217	1.14795	1.16761
MAXIMUM CHANNEL POWER	.73858	1.38021	1.26733	1.39024	1.20712	1.19086	1.15810	1.10362	1.12259

CORE POSITION/ASSEMBLY NUMBER	J20 46	J19 47	J18 48	J17 49	J16 50	J15 51	J13 52	J11 53
FUEL TYPE	L-8	J-0	J-0	J-0	K-0	K-8	J-4	L-8
CEA BANK TYPE	• 1 •		• 3 •				• B •	
ASSEMBLY AVERAGE POWER	1.22377	1.00366	.96479	.94407	1.11839	1.07161	.92488	1.28501
MAXIMUM FUEL ROD POWER	1.42614	1.10275	1.04465	1.01821	1.21512	1.14820	.96168	1.42862
MAXIMUM CHANNEL POWER	1.37579	1.09571	1.04018	1.01476	1.16275	1.10383	.95769	1.39871

CORE POSITION/ASSEMBLY NUMBER	K21 54	L20 55	L19 56	L18 57	L17 58	L16 59	L15 60	L13 61	L11 62
FUEL TYPE	L-0	K-8	K-8	L-8	J-0	J-0	K-8	L-8	E-16
CEA BANK TYPE			• 5 •				• 4 •		• 5 •
ASSEMBLY AVERAGE POWER	.85860	1.12410	1.15082	1.26897	.94490	.95701	1.09542	1.28500	1.03625
MAXIMUM FUEL ROD POWER	1.25983	1.24509	1.24396	1.41054	1.02077	1.00933	1.16775	1.42858	1.09885
MAXIMUM CHANNEL POWER	1.22633	1.19030	1.18845	1.36406	1.01587	1.00121	1.12271	1.39869	1.07233

Figure 4.3
Maine Yankee Cycle 7
Assembly Relative Power Densities
EOC (12K MWD/MT), HFP, ARO

82/02/24. — 20.33.36.   MYC7 12K MWD/MT (EOC6 @ 11.5K)

CORE SUMMARY OF MAXIMUM FUEL ASSEMBLY POWER

DESCRIPTION	MAX.VALUE	ASSEMBLY
ASSEMBLY AVG.	1.31748	53
MAX. FUEL ROD	1.44601	33
MAX. CHANNEL	1.40063	53

CORE POSITION/ASSEMBLY NUMBER	FUEL TYPE	CEA BANK TYPE	ASSEMBLY AVERAGE POWER	MAXIMUM FUEL ROD POWER	MAXIMUM CHANNEL POWER
A14 1	J-8		.45738	.77809	.76397
A12 2	L-0		.85259	1.19801	1.18206
B17 3	J-0		.45898	.83336	.81371
B16 4	L-0	* C *	.89592	1.24876	1.22371
B15 5	L-4		1.09052	1.35891	1.32241
B13 6	L-8	* 1 *	1.25076	1.41196	1.34726
B11 7	K-8		1.09681	1.18455	1.15113
C18 8	L-8		.78851	1.24582	1.21280
C17 9	L-8	* A *	1.15151	1.37491	1.31715
C16 10	K-0		1.09797	1.16776	1.15952
C15 11	K-0	* C *	1.12660	1.21761	1.20820
C13 12	J-0		.99026	1.06772	1.06316
C11 13	K-8	* 5 *	1.12146	1.18149	1.14842
D19 14	L-8		.78855	1.24585	1.21233
D18 15	L-4	* 5 *	1.22482	1.42389	1.34763
D17 16	K-0		1.11262	1.24671	1.22236
D16 17	J-0	* A *	.92247	.97255	.96726
D15 18	L-8		1.31297	1.43574	1.35915
D13 19	J-0	* 3 *	.96275	1.03570	1.03000
D11 20	L-8		1.29322	1.41600	1.34261
E20 21	J-0		.45926	.83377	.81416
E19 22	L-8	* A *	1.15163	1.37499	1.31721
E18 23	K-0		1.11254	1.24672	1.22237
E17 24	J-8		.88025	.95170	.94828
E16 25	J-8		.89415	.95206	.94357
E15 26	K-0	* 2 *	1.07020	1.14891	1.14573
E13 27	J-0		.92182	.97556	.96961
E11 28	J-0		.92584	1.00653	1.00191
F20 29	L-0	* C *	.89604	1.24886	1.22381
F19 30	K-0		1.09816	1.16795	1.15969
F18 31	J-0	* A *	.92266	.97276	.96727
F17 32	J-8		.89566	.95163	.94340
F16 33	L-12	* 6 *	1.29987	1.44601	1.38505
F15 34	K-0		1.05730	1.12104	1.11590
F13 35	K-0		1.04950	1.12292	1.10705
F11 36	J-0		.91196	.95371	.94958
G20 37	L-4		1.09062	1.35899	1.32247
G19 38	K-0	* C *	1.12666	1.21764	1.20822
G18 39	L-8		1.31297	1.43572	1.35914
G17 40	K-0	* 2 *	1.07031	1.14905	1.14579
G16 41	K-0		1.05746	1.12120	1.11604
G15 42	K-4	* B *	1.03542	1.12147	1.11369
G13 43	K-8		1.01582	1.05697	1.03537
G11 44	K-8	* 4 *	1.04564	1.10640	1.07620
H21 45	J-8		.45765	.7[illegible]33	.76423
J20 46	L-8	* 1 *	1.25082	1.41196	1.34726
J19 47	J-0		.99014	1.06782	1.06325
J18 48	J-0	* 3 *	.96254	1.03550	1.02980
J17 49	J-0		.92181	.97573	.96979
J16 50	K-0		1.04855	1.12286	1.10699
J15 51	K-8		1.01601	1.05717	1.03561
J13 52	J-4	* B *	.92139	.98403	.97843
J11 53	L-8		1.31748	1.44211	1.40063
K21 54	L-0		.85267	1.19908	1.18212
L20 55	K-8		1.09683	1.18454	1.15112
L19 56	K-8	* 5 *	1.12139	1.18144	1.14938
L18 57	L-8		1.29309	1.41585	1.34248
L17 58	J-0		.92570	1.00646	1.00184
L16 59	J-0		.91153	.95286	.94977
L15 60	K-8	* 4 *	1.04571	1.10651	1.07629
L13 61	L-8		1.31747	1.44205	1.40062
L11 62	E-16	* 5 *	1.05866	1.10075	1.08309

Figure 4.4
Maine Yankee Cycle 7
Assembly Relative Power Densities
BOC (50 MWD/MT), HFP, CEA Bank 5 Inserted
82/04/09. — 08.53.16. MYC7 50 MWD/MT (EOC6 @ 11.5K) HFP BANK 5

CORE SUMMARY OF MAXIMUM FUEL ASSEMBLY POWER

DESCRIPTION	MAX.VALUE	ASSEMBLY
ASSEMBLY AVG.	1.42754	50
MAX. FUEL ROD	1.58109	35
MAX. CHANNEL	1.51845	35

CORE POSITION/ASSEMBLY NUMBER	A14 1	A12 2
FUEL TYPE	J-8	L-0
CEA BANK TYPE		
ASSEMBLY AVERAGE POWER	.39930	.79345
MAXIMUM FUEL ROD POWER	.69972	1.16898
MAXIMUM CHANNEL POWER	.68518	1.11852

B17 3	B16 4	B15 5	B13 6	B11 7
J-0	L-0	L-4	L-8	K-8
	* C *		* 1 *	
.39815	.99368	1.13829	1.10266	.92487
.85481	1.54397	1.56633	1.43517	1.01502
.81793	1.47597	1.50181	1.36598	.97287

C18 8	C17 9	C16 10	C15 11	C13 12	C11 13
L-8	L-8	K-0	K-0	J-0	K-8
	* A *		* C *		* 5 *
.43867	.93108	1.22436	1.28747	.91477	.67264
.81414	1.31360	1.44804	1.40118	1.12772	.83713
.78915	1.25936	1.39329	1.35019	1.12265	.78291

D19 14	D18 15	D17 16	D16 17	D15 18	D13 19	D11 20
L-8	L-4	K-0	J-0	L-8	J-0	L-8
	* 5 *		* A *		* 3 *	
.43883	.56967	.91461	.90883	1.35380	.98040	1.15825
.81441	.81855	1.16152	1.04542	1.57565	1.05968	1.42890
.78942	.78112	1.12002	1.03922	1.50527	1.05113	1.36162

E20 21	E19 22	E18 23	E17 24	E16 25	E15 26	E13 27	E11 28
J-0	L-8	K-0	J-8	J-8	K-0	J-0	J-0
	* A *				* 2 *		
.39864	.93162	.91474	.72270	.81374	1.22420	1.10079	1.09343
.85564	1.31432	1.16201	.75418	.94446	1.49738	1.25399	1.15904
.81882	1.26008	1.12049	.74779	.93520	1.45496	1.25105	1.15057

F20 29	F19 30	F18 31	F17 32	F16 33	F15 34	F13 35	F11 36
L-0	K-0	J-0	J-8	L-12	K-0	K-0	J-0
* C *		* A *		* 6 *			
.99441	1.22525	.90937	.81332	.94602	1.26270	1.42739	1.20858
1.54501	1.44911	1.04538	.94481	1.24799	1.50620	1.58109	1.29217
1.47698	1.39430	1.03940	.93541	1.15936	1.48148	1.51845	1.28126

G20 37	G19 38	G18 39	G17 40	G16 41	G15 42	G13 43	G11 44
L-4	K-0	L-8	K-0	K-0	K-4	K-8	K-8
	* C *		* 2 *		* B *		* 4 *
1.13898	1.28812	1.35425	1.22462	1.26309	1.31961	1.30058	1.30608
1.56729	1.40187	1.57640	1.49791	1.50671	1.44380	1.44623	1.41866
1.50272	1.35087	1.50599	1.45544	1.48193	1.37934	1.39012	1.37140

H21 45
J-8
.39976
.70030
.68579

J20 46	J19 47	J18 48	J17 49	J16 50	J15 51	J13 52	J11 53
L-8	J-0	J-0	J-0	K-0	K-8	J-4	L-8
* 1 *		* 3 *				* B *	
1.10319	.91492	.98032	1.10092	1.42754	1.30082	1.00893	1.25817
1.43586	1.12831	1.05949	1.25425	1.58085	1.44652	1.11344	1.55489
1.36665	1.12322	1.05094	1.25130	1.51808	1.39044	1.10296	1.48603

K21 54
L-0
.79392
1.16959
1.11911

L20 55	L19 56	L18 57	L17 58	L16 59	L15 60	L13 61	L11 62
K-8	K-8	L-8	J-0	J-0	K-8	L-8	E-16
	* 5 *				* 4 *		* 5 *
.92526	.67273	1.15821	1.09323	1.20794	1.30611	1.25814	.68318
1.01550	.83737	1.42882	1.15870	1.29103	1.41862	1.55491	.83331
.97335	.78294	1.36154	1.15022	1.28099	1.37136	1.48605	.74440

Figure 4.5
Maine Yankee Cycle 7
Assembly Relative Power Densities
MOC (6K MWD/MT), HFP, CEA Bank 5 Inserted

82/04/09. — 08.56.57.   MYC7 6K MWD/MT (EOC6 @ 11.5K) HFP BANK 5

CORE SUMMARY OF MAXIMUM FUEL ASSEMBLY POWER

DESCRIPTION	MAX. VALUE	ASSEMBLY
ASSEMBLY AVG.	1.38018	39
MAX. FUEL ROD	1.55178	61
MAX. CHANNEL	1.50590	61

CORE POSITION/ASSEMBLY NUMBER	A14 1	A12 2
FUEL TYPE	J-8	L-0
CEA BANK TYPE		
ASSEMBLY AVERAGE POWER	.45451	.87591
MAXIMUM FUEL ROD POWER	.76954	1.25459
MAXIMUM CHANNEL POWER	.75855	1.21748

B17 3	B16 4	B15 5	B13 6	B11 7
J-0	L-0	L-4	L-8	K-8
	* C *		* 1 *	
.42193	.97394	1.16381	1.20553	1.01300
.83474	1.44218	1.49799	1.44658	1.11365
.80901	1.39390	1.45177	1.39013	1.06278

C18 8	C17 9	C16 10	C15 11	C13 12	C11 13
L-8	L-8	K-0	K-0	J-0	K-8
	* A *		* C *		* 5 *
.48064	.96677	1.17069	1.24204	.93625	.70145
.83059	1.28028	1.33348	1.34565	1.12100	.88364
.81335	1.24302	1.30613	1.31373	1.11462	.82070

D19 14	D18 15	D17 16	D16 17	D15 18	D13 19	D11 20
L-8	L-4	K-0	J-0	L-8	J-0	L-8
	* 5 *		* A *		* 3 *	
.48074	.57370	.89745	.91219	1.37996	.98897	1.20526
.83074	.82426	1.11208	1.02592	1.53165	1.04988	1.41641
.81350	.77531	1.08238	1.02314	1.48332	1.04355	1.36886

E20 21	E19 22	E18 23	E17 24	E16 25	E15 26	E13 27	E11 [illegible]
J-0	L-8	K-0	J-8	J-8	K-0	J-0	J-0
	* A *				* 2 *		
.42230	.96709	.8974[illegible]	.73963	.82539	1.16878	1.06340	1.06685
.83535	1.28065	1.11231	.76738	.94704	1.35874	1.17052	1.10677
.80966	1.24340	1.08262	.76461	.94047	1.33408	1.16797	1.10034

F20 29	F19 30	F18 31	F17 32	F16 33	F15 34	F13 35	F11 36
L-0	K-0	J-0	J-8	L-12	K-0	K-0	J-0
* C *		* A *		* 6 *			
.97433	1.17119	.91255	.82495	.95618	1.17169	1.31503	1.14657
1.44268	1.33905	1.02592	.94720	1.25927	1.35155	1.43813	1.19905
1.39440	1.30668	1.02328	.94052	1.14117	1.34121	1.40187	1.19096

G20 37	G19 38	G18 39	G17 40	G16 41	G15 42	G13 43	G11 44
L-4	K-0	L-8	K-0	K-0	K-4	K-8	K-8
	* C *		* 2 *		* B *		* 4 *
1.16421	1.24237	1.38018	1.16905	1.17197	1.24699	1.26297	1.29118
1.49841	1.34598	1.53202	1.35907	1.35189	1.34127	1.36189	1.37631
1.45218	1.31404	1.48369	1.33438	1.34151	1.28807	1.30451	1.32203

H21 45
J-8
.45491
.77000
.75903

J20 46	J19 47	J18 48	J17 49	J16 50	J15 51	J13 52	J11 53
L-8	J-0	J-0	J-0	K-0	K-8	J-4	L-8
* 1 *		* 3 *				* B *	
1.20588	.93629	.98883	1.06348	1.31516	1.26327	1.02485	1.31813
1.44696	1.12136	1.04970	1.17074	1.43902	1.36223	1.09842	1.55169
1.39051	1.11496	1.04338	1.16818	1.40170	1.30483	1.09125	1.50582

K21 54
L-0
.87624
1.25500
1.21787

L20 55	L19 56	L18 57	L17 58	L16 59	L15 60	L13 61	L11 62
K-8	K-8	L-8	J-0	J-0	K-8	L-8	E-16
	* 5 *				* 4 *		* 5 *
1.01326	.70149	1.20519	1.06670	1.14602	1.29131	1.31814	.69506
1.11397	.88379	1.41635	1.10675	1.19806	1.37646	1.55178	.85291
1.06307	.82084	1.36879	1.10008	1.19049	1.32216	1.50590	.76830

Figure 4.6
Maine Yankee Cycle 7
Assembly Relative Power Densities
EOC (12K MWD/MT), HFP, CEA Bank 5 Inserted

82/04/09. — 08.58.47. MYC7 12K MWD/MT (EOC6 @ 11.5K) HFP BANK 5

CORE SUMMARY OF MAXIMUM FUEL ASSEMBLY POWER

DESCRIPTION	MAX.VALUE	ASSEMBLY
ASSEMBLY AVG.	1.42218	39
MAX. FUEL ROD	1.55979	39
MAX. CHANNEL	1.47876	39

CORE POSITION/ASSEMBLY NUMBER	A14 1	A12 2
FUEL TYPE	J-8	L-0
CEA BANK TYPE		
ASSEMBLY AVERAGE POWER	.50210	.91226
MAXIMUM FUEL ROD POWER	.83249	1.24983
MAXIMUM CHANNEL POWER	.82025	1.22896

B17 3	B16 4	B15 5	B13 6	B11 7
J-0	L-0	L-4	L-8	K-8
	* C *		* 1 *	
.45834	.97933	1.20284	1.28234	1.02946
.85668	1.37718	1.48836	1.46591	1.11648
.83464	1.34657	1.43543	1.40816	1.08003

C18 8	C17 9	C16 10	C15 11	C13 12	C11 13
L-8	L-8	K-0	K-0	J-0	K-8
	* A *		* C *		* 5 *
.53775	1.03115	1.15500	1.22429	.94798	.69063
.85946	1.28423	1.27956	1.30502	1.12536	.89244
.85286	1.25026	1.26330	1.28945	1.11640	.82215

D19 14	D18 15	D17 16	D16 17	D15 18	D13 19	D11 20
L-8	L-4	K-0	J-0	L-8	J-0	L-8
	* 5 *		* A *		* 3 *	
.53783	.58138	.89674	.92929	1.42200	1.00060	1.24453
.85958	.82149	1.09149	1.04144	1.55961	1.06031	1.42559
.85298	.76502	1.07925	1.03675	1.47859	1.05324	1.36407

E20 21	E19 22	E18 23	E17 24	E16 25	E15 26	E13 27	E11 28
J-0	L-8	K-0	J-8	J-8	K-0	J-0	J-0
	* A *				* 2 *		
.45868	1.03139	.89673	.75754	.83839	1.13442	1.03842	1.04812
.85722	1.28451	1.09165	.79126	.96445	1.25735	1.11486	1.08941
.83522	1.25051	1.07942	.78685	.95579	1.24584	1.10917	1.08765

F20 29	F19 30	F18 31	F17 32	F16 33	F15 34	F13 35	F11 36
L-0	K-0	J-0	J-8	L-12	K-0	K-0	J-0
* C *		* A *		* 6 *			
.97960	1.15535	.92960	.83800	.95971	1.09826	1.22077	1.08603
1.37749	1.27995	1.04143	.96455	1.23727	1.24127	1.32153	1.12492
1.34689	1.26368	1.03690	.95580	1.10438	1.23766	1.30287	1.12211

H21 45	G20 37	G19 38	G18 39	G17 40	G16 41	G15 42	G13 43	G11 44
J-8	L-4	K-0	L-8	K-0	K-0	K-4	K-8	K-8
		* C *		* 2 *		* B *		* 4 *
.50248	1.20314	1.22452	1.42218	1.13464	1.09847	1.15537	1.17873	1.21434
.83291	1.48869	1.30527	1.55979	1.25760	1.24148	1.21257	1.23989	1.26232
.82069	1.43572	1.28968	1.47876	1.24606	1.23789	1.18335	1.20781	1.23317

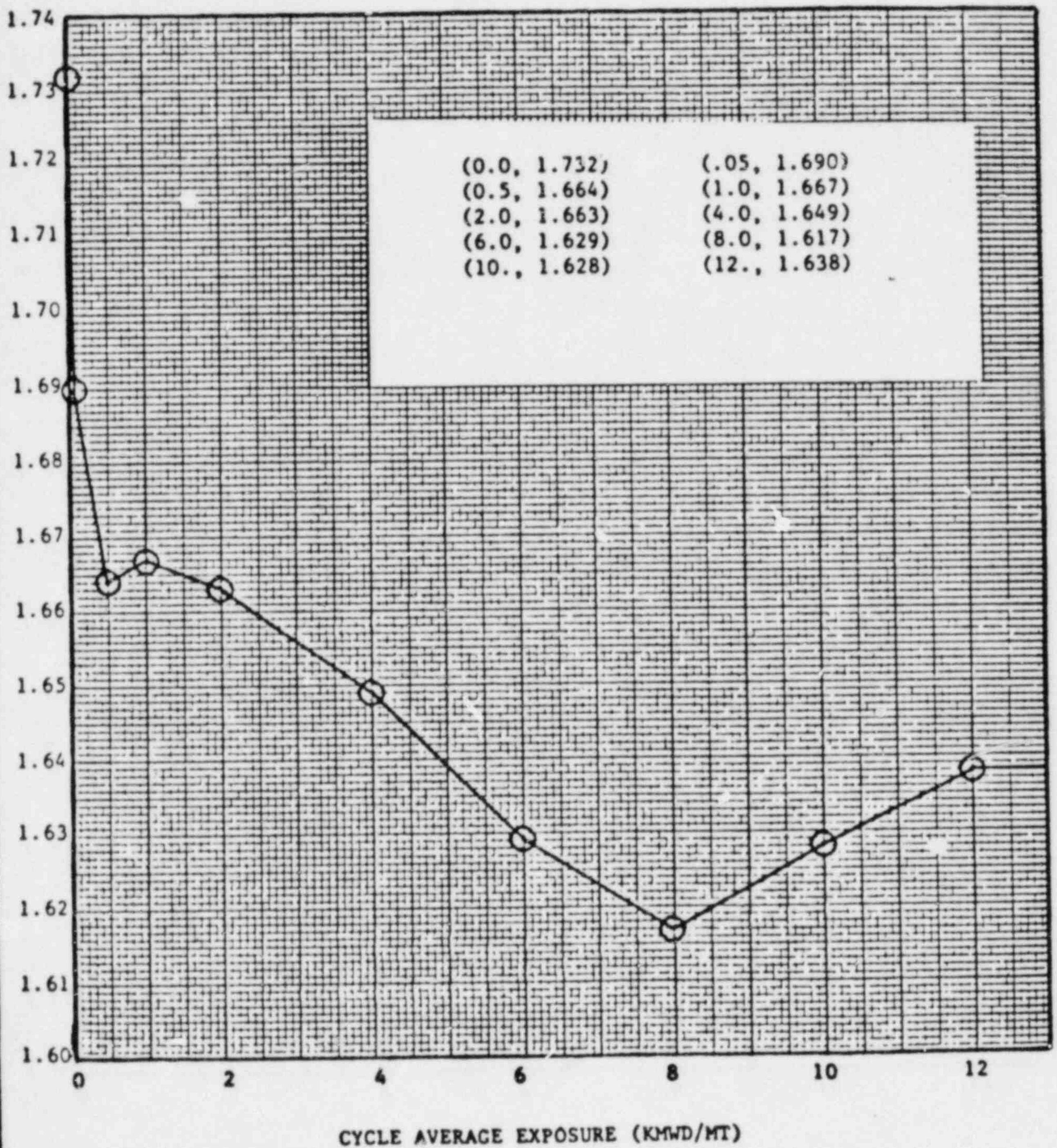
K21 54	J20 46	J19 47	J18 48	J17 49	J16 50	J15 51	J13 52	J11 53
L-0	L-8	J-0	J-0	J-0	K-0	K-8	J-4	L-8
	* 1 *		* 3 *				* B *	
.91254	1.28264	.94801	1.00049	1.03850	1.22088	1.17898	.99829	1.31565
1.25015	1.46624	1.12565	1.06027	1.11504	1.32148	1.24016	1.04737	1.50486
1.22928	1.40846	1.11669	1.05314	1.10936	1.30279	1.20803	1.04089	1.44282

L20 55	L19 56	L18 57	L17 58	L16 59	L15 60	L13 61	L11 62
K-8	K-8	L-8	J-0	J-0	K-8	L-8	E-16
	* 5 *				* 4 *		* 5 *
1.02969	.69069	1.24452	1.04803	1.08556	1.21445	1.31566	.67632
1.11674	.89258	1.42559	1.08943	1.12409	1.26244	1.50491	.84566
1.08028	.82228	1.36407	1.08767	1.12163	1.23328	1.44288	.76380

Note: 1. This curve includes 10% calculational uncertainty

2. $F_R^T = F_R^P * 1.03$

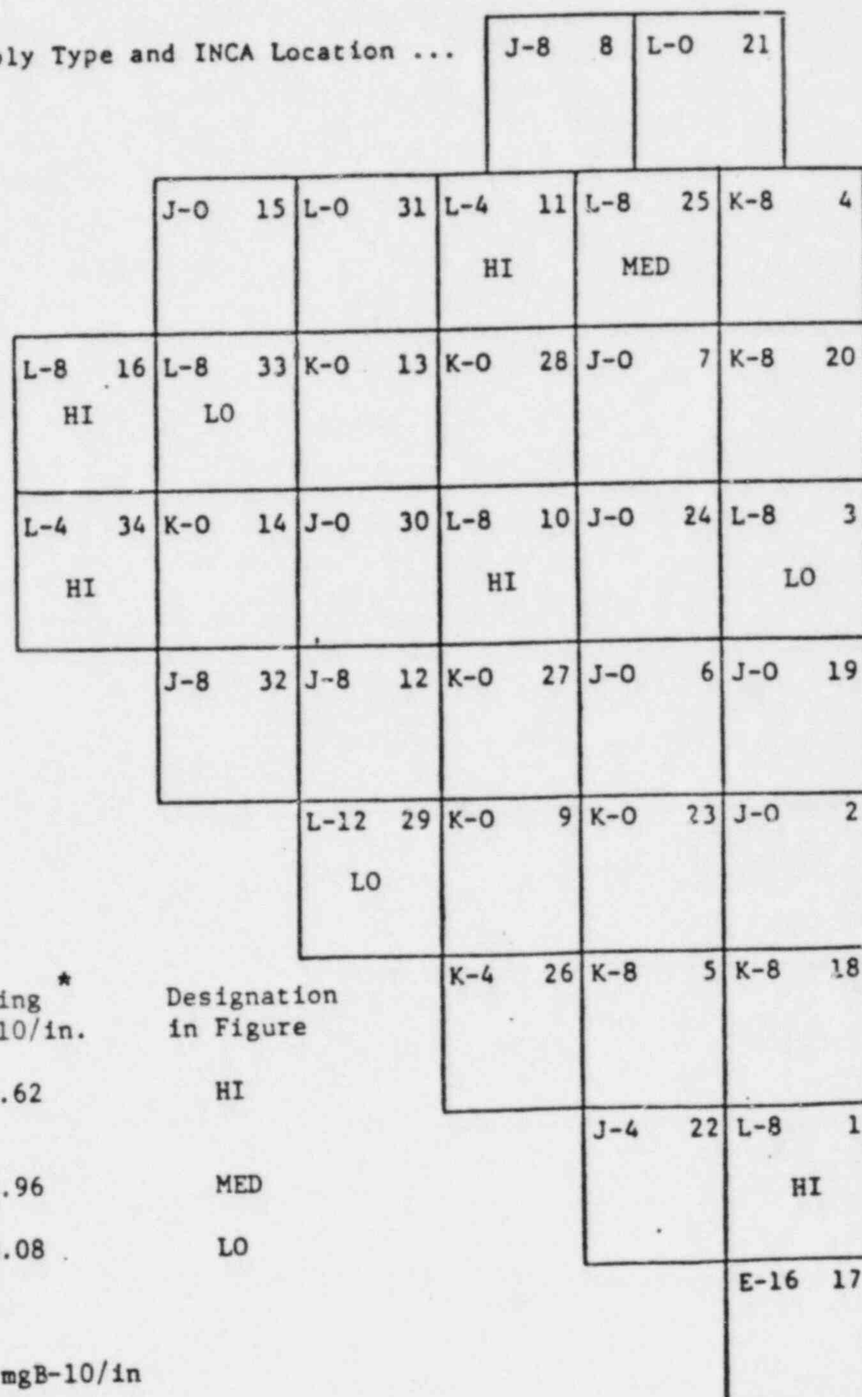
3. Measured F_R^T should be augmented by measurement uncertainty (8%) before comparison to this curve.



MAINE YANKEE	Allowable Unrodded Radial Peak Versus Cycle Average Burnup	Figure 4.7

Figure 4.8
Maine Yankee Cycle 7
Arrangement of Shim Loadings
As-Built Calculation

Assembly Type and INCA Location ...



Assembly Type	No. of Assemblies	Loading* mgB-10/in.	Designation in Figure
L-4	12	24.62	HI
L-8	20		
L-8	8	23.96	MED
L-8	12	23.08	LO
L-12	4		

* design shim loading is 23.8 mgB-10/in

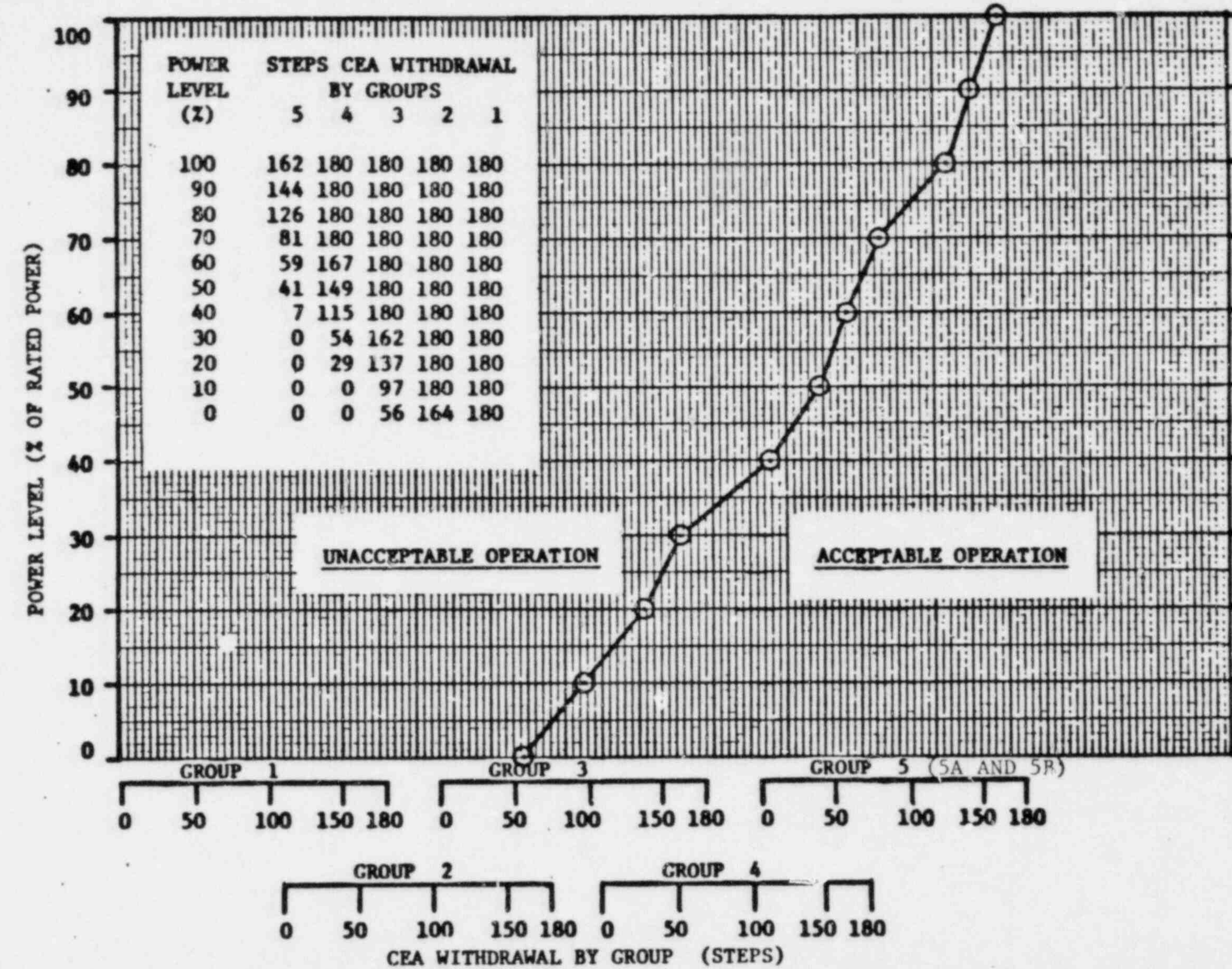
MAINE YANKEE

Power-Dependent Insertion Limit
(P D I L)
for CEA's

Figure 4.9

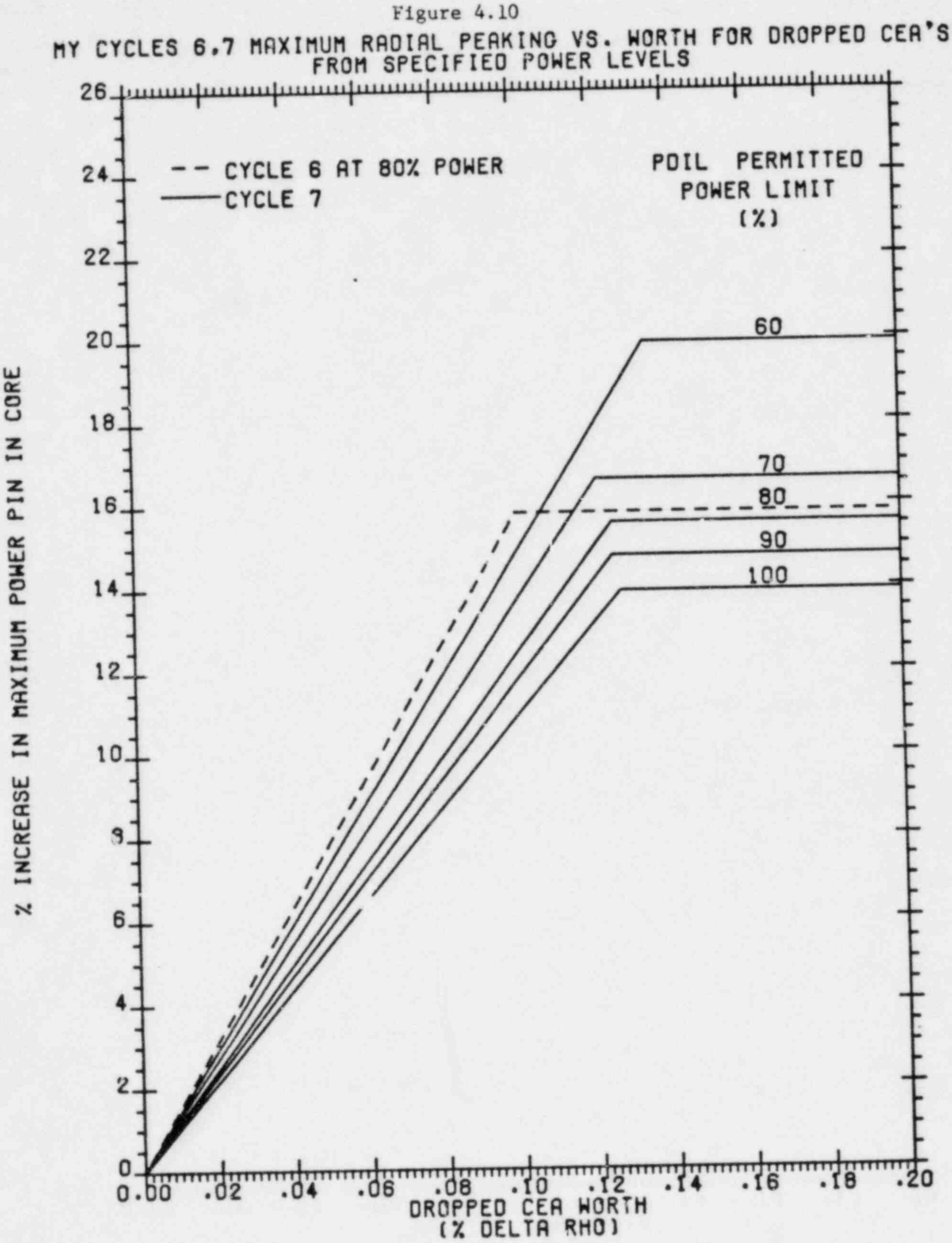
POWER LEVEL (% OF RATED POWER) VS. CEA WITHDRAWAL (STEPS) FOR THREE LOOP OPERATION
OR (% OF 0.63 X RATED POWER) FOR TWO LOOP OPERATION

POWER LEVEL (%)	STEPS CEA WITHDRAWAL BY GROUPS 5	4	3	2	1
100	162	180	180	180	180
90	144	180	180	180	180
80	126	180	180	180	180
70	81	180	180	180	180
60	59	167	180	180	180
50	41	149	180	180	180
40	7	115	180	180	180
30	0	54	162	180	180
20	0	29	137	180	180
10	0	0	97	180	180
0	0	0	56	164	180



NOTE: IN APPLYING THIS RESTRICTION TO TWO LOOP OPERATION, THE RATED POWER FOR TWO LOOP OPERATION MUST BE USED WITHIN FOUR (4) HOURS OF ATTAINING TWO LOOP CONDITIONS

Figure 4.10



MY CYCLES 6,7 MAXIMUM RADIAL PEAKING VS. WORTH FOR DROPPED CEA'S FROM SPECIFIED POWER LEVELS
- - CYCLE 6 AT 80% POWER
—— CYCLE 7
POIL PERMITTED POWER LIMIT (%)
60
70
80
90
100
% INCREASE IN MAXIMUM POWER PIN IN CORE
0
2
4
6
8
10
12
14
16
18
20
22
24
26
0.00
.02
.04
.06
.08
.10
.12
.14
.16
.18
.20
DROPPED CEA WORTH (% DELTA RHO)

Figure 4.11 MYC7 DROPPED CEA RADIAL PEAKING WITH POWER RESTRICTION
ARO + DROPPED A AT BOC MAXIMUM 1-PIN PEAKING VS. TIME

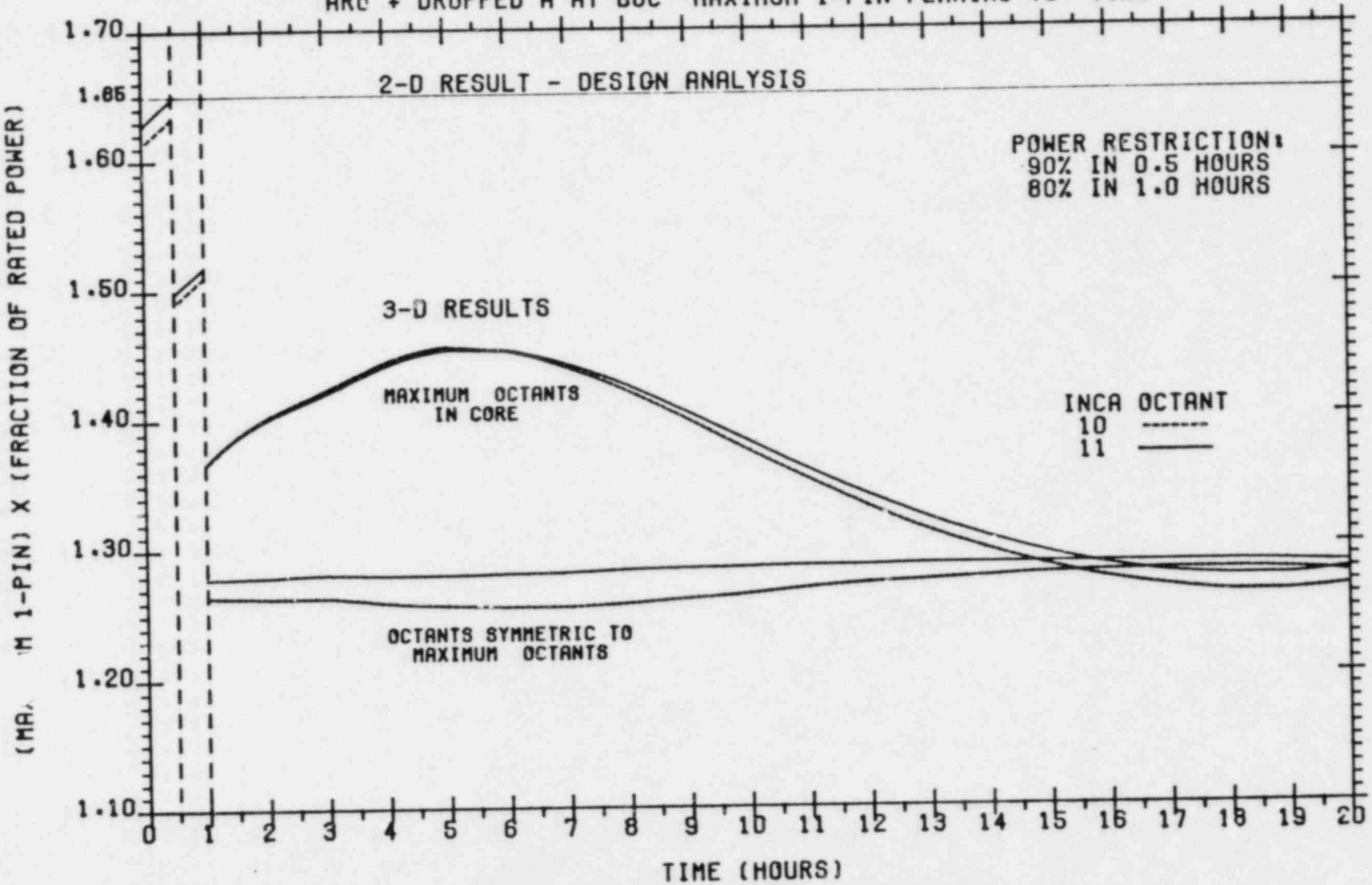


Figure 4.12 MYC7 DROPPED CEA RADIAL PEAKING WITH POWER RESTRICTION
ARO + DROPPED B AT BOC MAXIMUM 1-PIN PEAKING VS. TIME

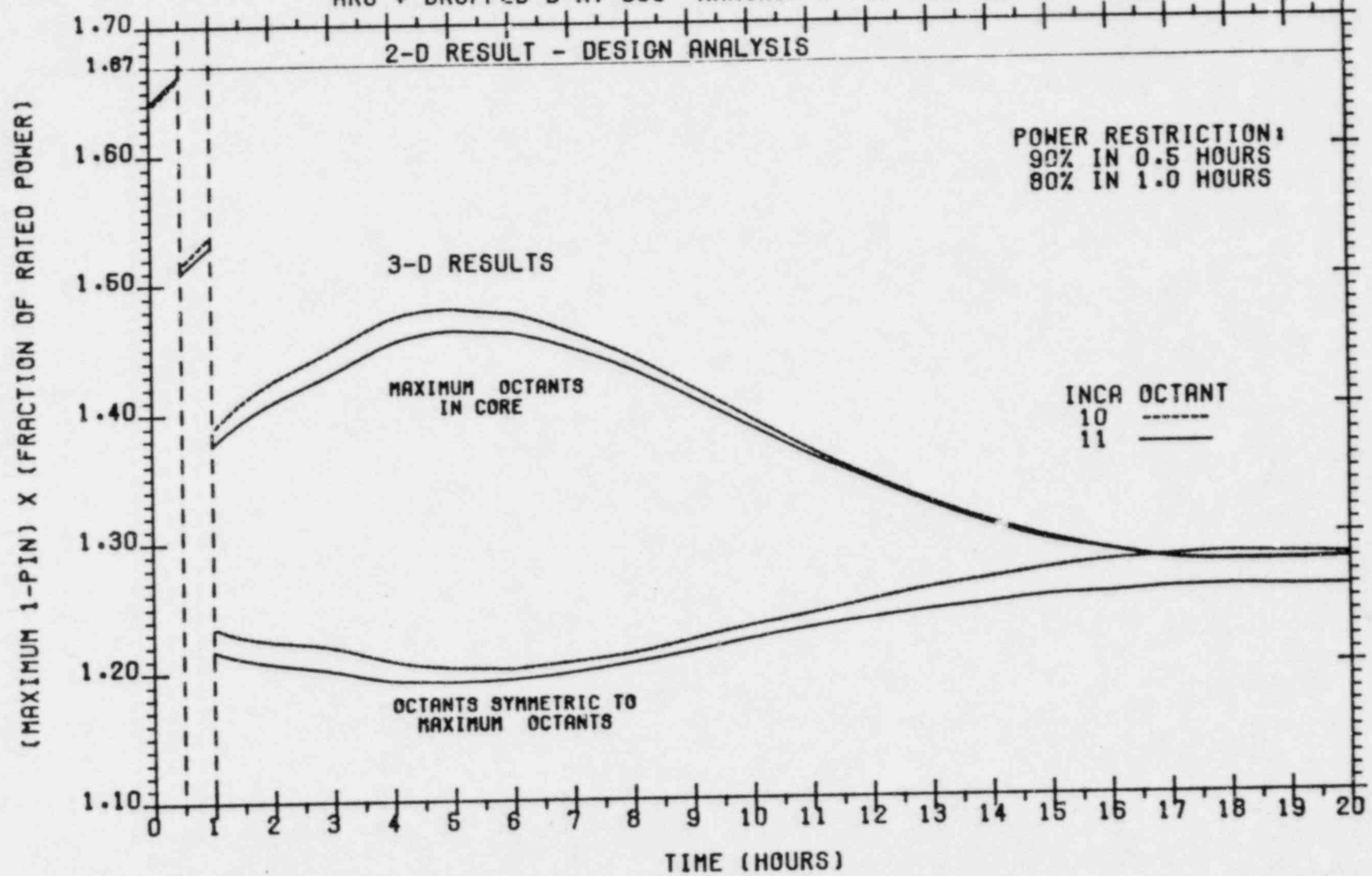
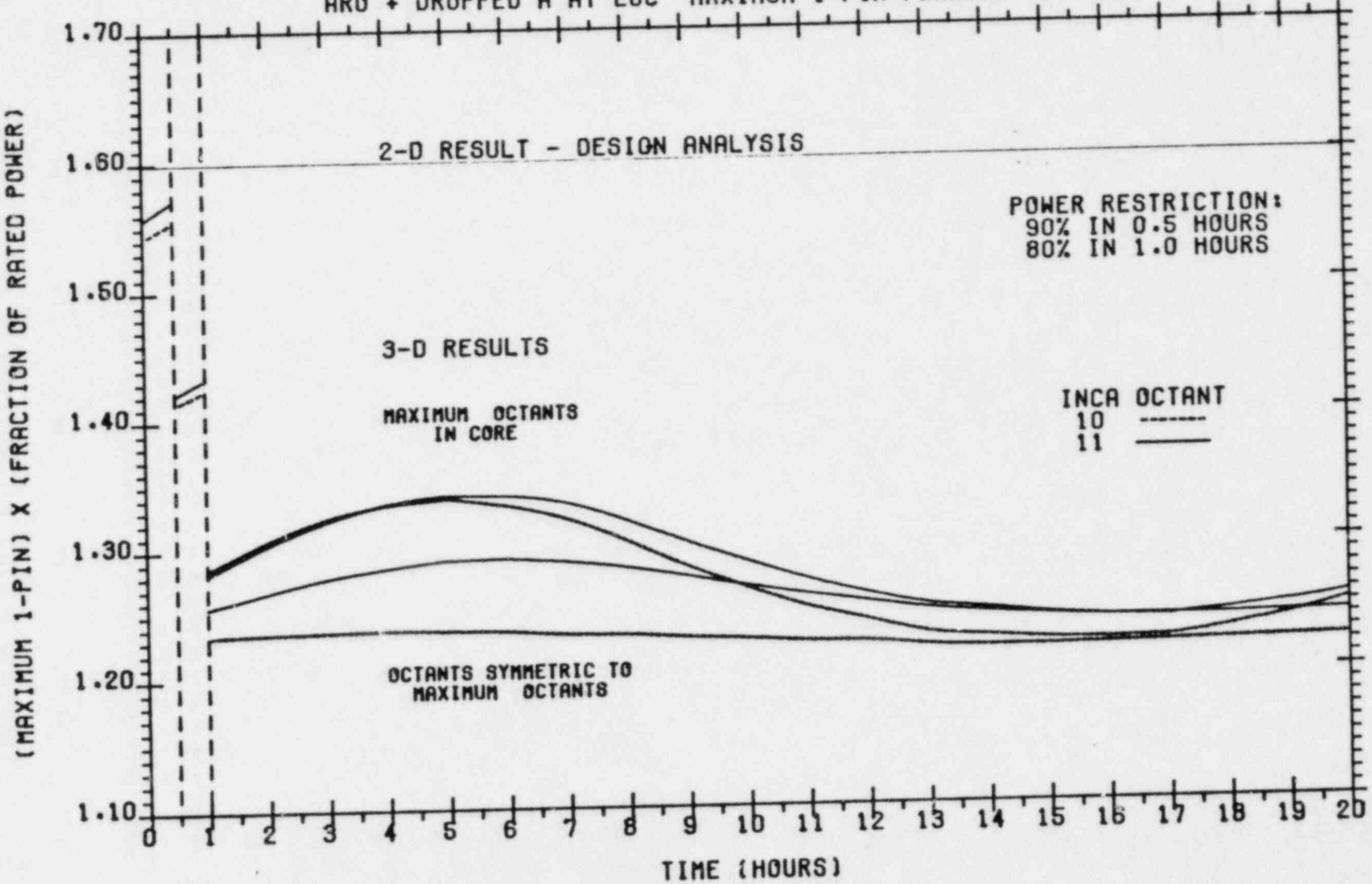


Figure 4.13 MYC7 DROPPED CEA RADIAL PEAKING WITH POWER RESTRICTION
ARO + DROPPED A AT EOC MAXIMUM 1-PIN PEAKING VS. TIME



-87-

Figure 4.14 MYC7 DROPPED CEA RADIAL PEAKING WITH POWER RESTRICTION
ARO + DROPPED B AT EOC MAXIMUM 1-PIN PEAKING VS. TIME

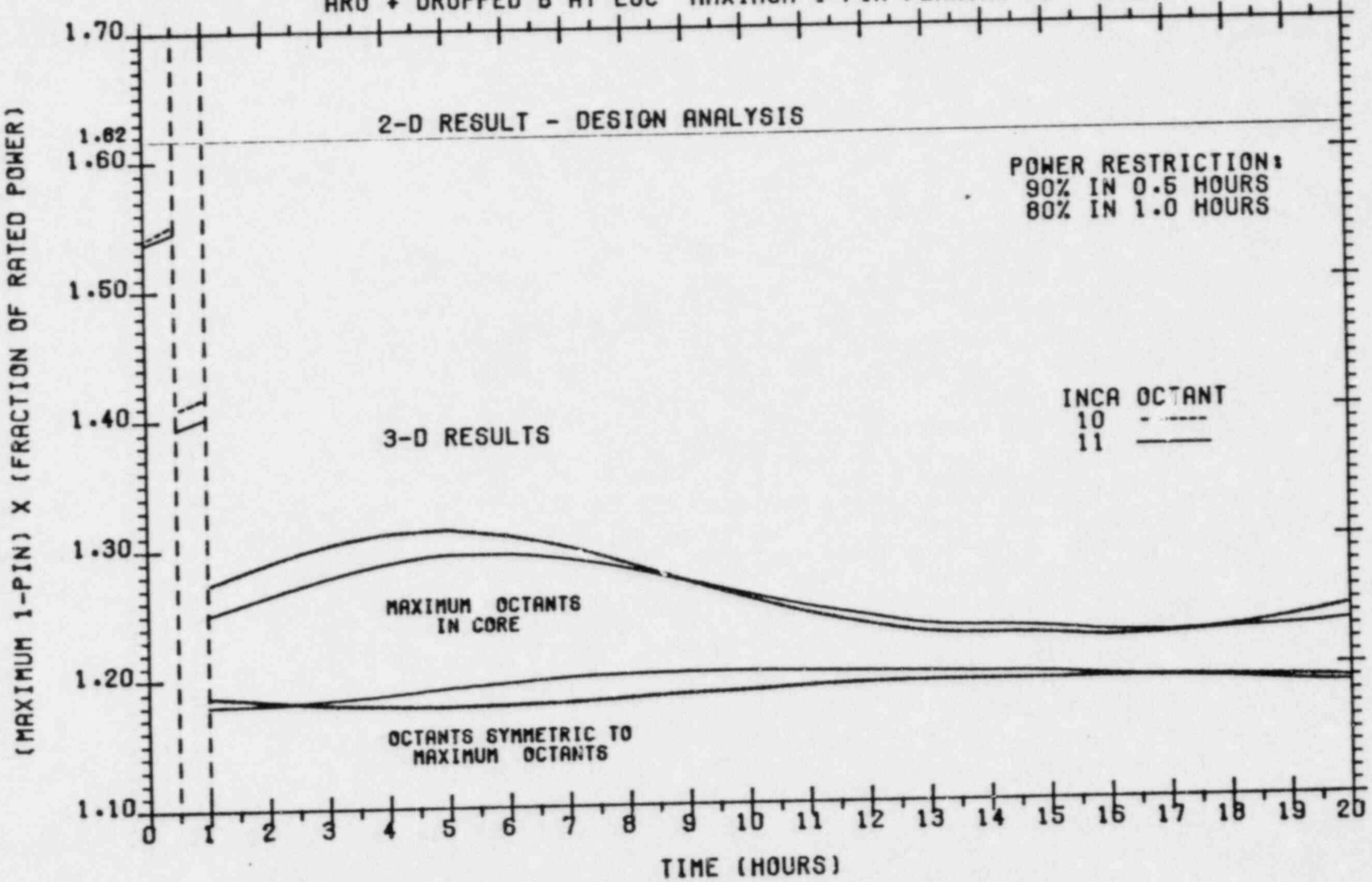
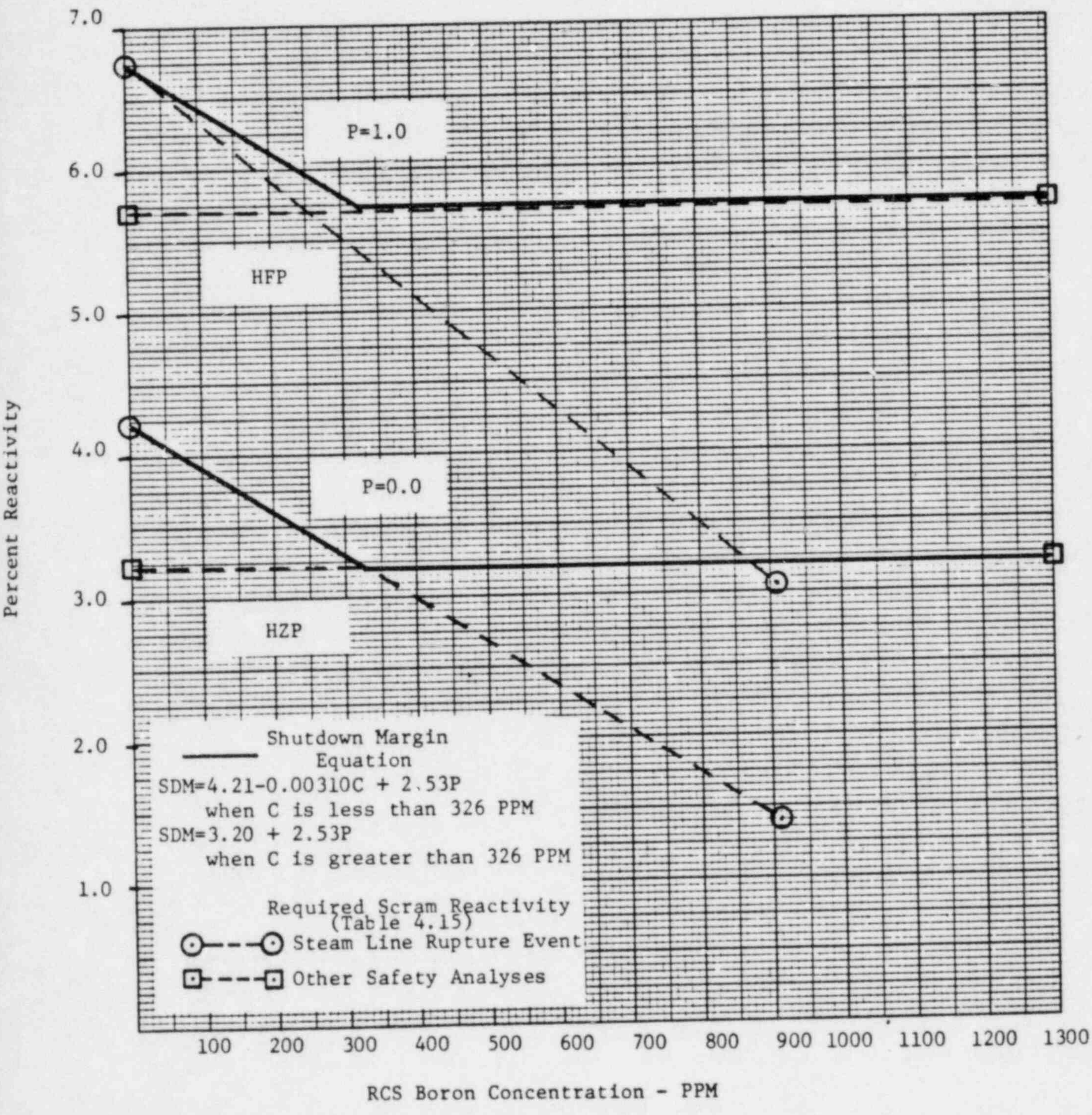
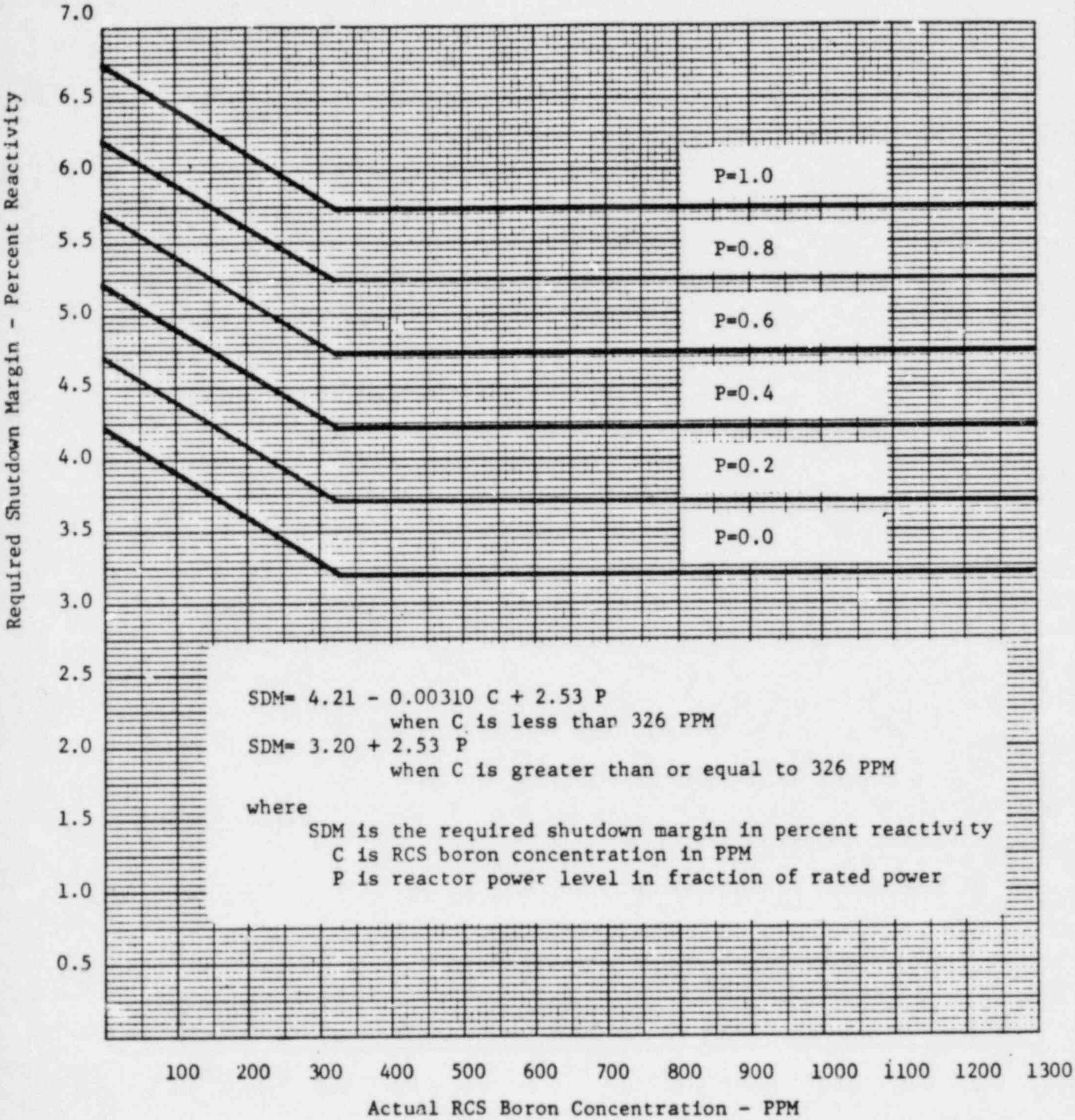


Figure 4.15
Maine Yankee Cycle 7
Shutdown Margin Equation
and Required Scram Reactivity



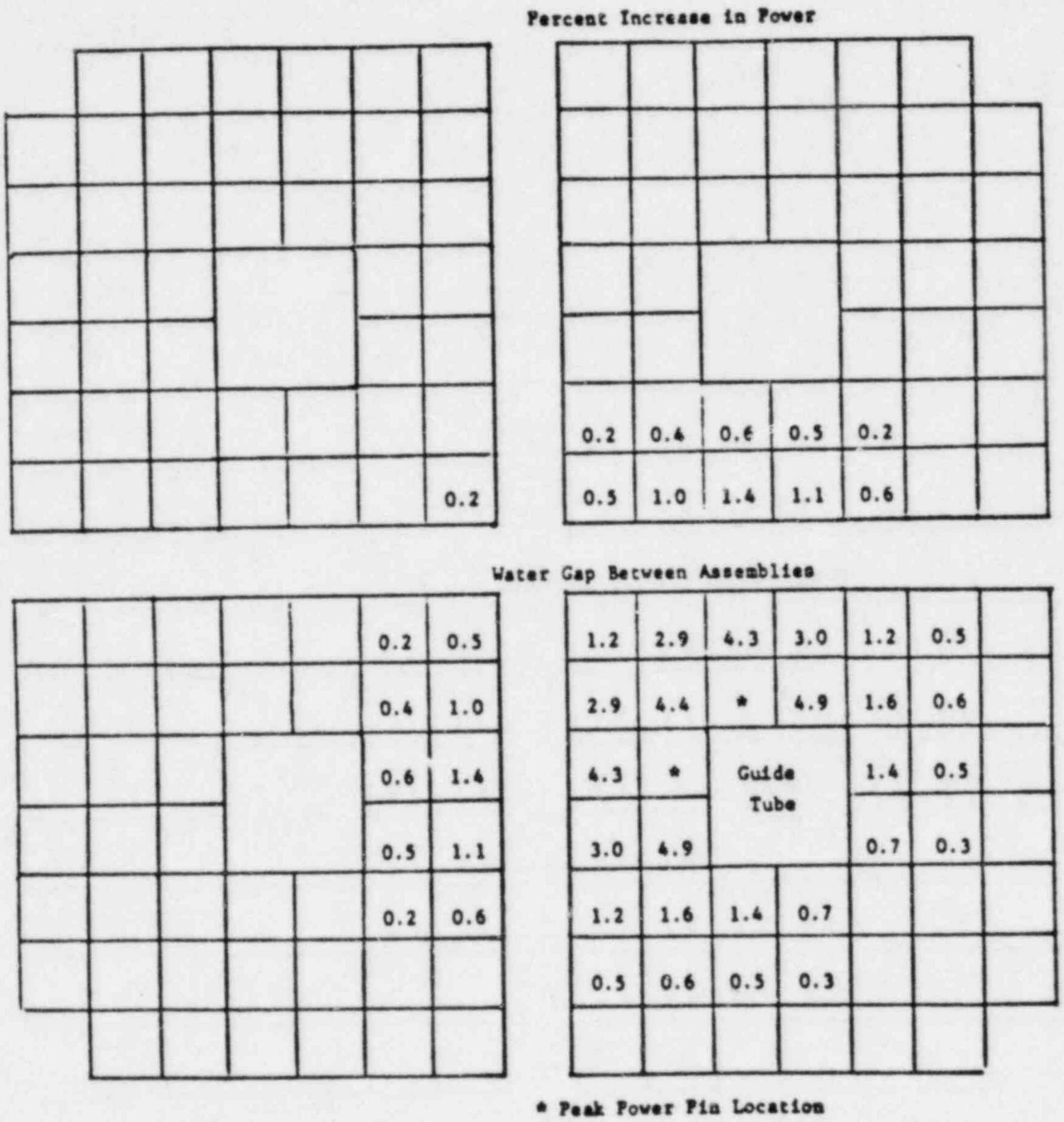


Maine Yankee	Required Shutdown Margin Versus RCS Boron Concentration	Figure 4.16

Figure 4.17

Maine Yankee Cycle 7

Limiting Single Gap Power Peaking



Legend: $\boxed{4.9}$ ... Peaking in peak power pin location (ungapped rod) due to a single gap at the indicated location.

5.0 SAFETY ANALYSIS

5.1 General

A review of the Reference Safety Analysis for operation of Maine Yankee during Cycle 7 is presented in this section. The parameters which influence the results of the safety analysis are listed in Table 5.1. Values are provided for the Reference Analysis, for Cycle 6 and Cycle 7. Currently, the Reference Analysis for Maine Yankee is the Cycle 3 stretch power analysis.

These parameters may be divided as follows: 1) initial operating conditions, 2) core power distributions, 3) reactivity coefficients, 4) shutdown CEA characteristics, and 5) Reactor Protection System Setpoints and Time Delays. A discussion of the differences between Cycle 7, Cycle 6, and the Reference Analysis values for the classes of parameters listed above is contained in Subsections 5.1.1 through 5.1.5.

5.1.1 Initial Operating Conditions

The initial conditions assumed in the safety evaluations considered in this section are listed in Table 5.1. These conditions are conservative with respect to intended Cycle 7 operation in that uncertainties are included to account for measurement errors associated with plant instrumentation. The uncertainties include:

a) A two percent allowance for calorimetric error in core thermal power.

b) A four degree allowance for measurement error on reactor coolant temperature.

c) A twenty-five psi allowance for measurement error on main coolant pressure.

Variable conditions on core inlet temperature and pressure are allowed in Technical Specification 3.10, shown in Figure 5.1. These are based on preserving DNB overpower margin for all possible combinations of temperature and pressure. The preservation of DNB overpower margin assures that the minimum DNBR reported for each of the incidents considered remains conservative for operation at the lower system pressure.

5.1.2 Core Power Distributions

The power distribution in the core, and in particular, the peak heat flux and enthalpy rise, are of major importance in determining core thermal margin. The procedure used in the Reference Safety Analysis was to set the initial conditions (inlet temperature, power, pressure, CEA insertion, and axial power distribution) and through analysis assure that enough initial overpower margin is available to prevent the violation of acceptable criteria for each incident analyzed.

This procedure is continued for Cycle 7. If the available overpower margin is not sufficient for the set of initial conditions, new power distributions are selected, which are controlled by the symmetric offset pre-trip alarm (Section 6.0), until it is demonstrated that sufficient margin exists.

As a starting point the Reference Safety Analysis assumed the FSAR design power distribution ($F_z = 1.68$ and $F^N_{\text{delta H}} = 1.49$) shown in Figure 5.2. The power distributions in Figure 5.2 cover the spectrum expected in the operation of the reactor through its lifetime and are considered a design set. In most cases considered in (3) and (6), acceptable performance was demonstrated with the use of the design power distributions. As indicated in Table 5.11, the design power distribution, at the full power heat flux, results in lower DNBR than any of the Cycle 7 predicted power distributions within the symmetric offset pre-trip alarm band (S/O LCO band), evaluated at their respective maximum power level limit as defined in the PDIL (Technical Specifications 3.10.A.1).

In addition, values are presented in Table 5.11 for Pd-Po, the percent rated thermal power margin between Pd, the power level at which the MDNBR for a given power distribution would equal the SAFDL on DNB, and Po, the initial maximum power level allowed by the CEA insertion limit for that rod configuration. This is a more precise indicator of relative DNB margin between power distributions than initial steady-state MDNBR, due to variation in the subchannel location in which MDNBR is predicted at nominal conditions versus limiting conditions. For Cycle 7, the thermal power margin (Pd-Po) for the FSAR design power distribution at full power conditions is the same or lower than the thermal margin for the limiting Cycle 7 power distributions within the LCO band at both full and lower power levels. This is true for both the W-3 and YAEC-1 DNB SAFDLS. Hence, thermal margins calculated using the FSAR design power distribution are conservative for Cycle 7.

Furthermore, it is obvious from Table 5.11 that the W-3 DNB correlation predicts conservative margins to DNB with respect to the proposed YAEC-1 correlation. These .acts justify the margins to DNB calculated with the W-3 correlation and FSAR design power distribution in (3) and (6) as being bounding for Cycle 7, provided other influential factors affecting a given transient are bounded as well.

The values for initial steady-state MDNBR and thermal power margin for the FSAR design power distribution using the reference analysis and Cycle 5 full power thermal design conditions are as or more limiting than any of the Cycle 7 power distributions, including the FSAR design distribution at HFP conditions for Cycle 7. This is due to the flow penalty required for mixed core thermal hydraulics for Cycle 5 and the rod pitch, bow and clad diameter enthalpy rise factor applied to the CE fuel which was the limiting fuel type in the reference analysis. This fact justifies the use of values for MDNBR reported in (6) and (33) as being limiting for Cycle 7 as well, provided that other influential factors are also bounded in Cycle 7 for a specific transient.

Power peaking associated with the CEA drop and the CEA ejection events for Cycle 7 are compared with reference values in Table 5.1. The effect of differences between Cycle 7, Cycle 6, and the Reference Analysis for the CEA drop and CEA ejection are discussed in Sections 5.4.2 and 5.5.4.

5.1.3 Reactivity Coefficients

The transient response of the reactor system is dependent on reactivity feedback effects, in particular, the moderator and the fuel temperature reactivity coefficients. Nominal values for each of the above feedback coefficients are given in Sections 4.5 and 4.6. The moderator temperature coefficient and All-Rods-In (ARI) moderator defect for Cycle 7 are bounded by Cycle 6 values. The Doppler coefficients for Cycle 7 are essentially identical with those of Cycle 6. Variations in the above parameters will influence each transient in a different manner. Therefore, the effect of the difference in reactivity coefficients noted above on the Reference Safety Analysis is discussed on an event by event basis.

A proposed modification to the Technical Specification for Cycle 7 operation is to change the allowable value for MTC in the power range greater than 70% RTP (rated thermal power) from "less positive than 0.0" to "less positive than $+0.5 \times 10^{-4}$ delta rho/^{o}F". The analyses performed in (6), which were conservatively performed at HFP conditions with MTC equal to $+0.5 \times 10^{-4}$ delta rho/^{o}F, support this change and any new analysis performed continues to consider the MTC at HFP, BOC to be as positive as $+.5 \times 10^{-4}$ delta rho/^{o}F.

The effective neutron lifetime, delayed neutron fractions, and decay constants are functions of fuel burnup and the fuel loading pattern. The Cycle 7 kinetics parameters are compared to the corresponding Reference Cycle values in Section 4.8. Small differences that are experienced from cycle to cycle have an insignificant impact on the response of the plant for all transients except the CEA Ejection. In the CEA Ejection Accident, the ratio of the ejected rod worth to effective delayed neutron fraction is extremely sensitive in determining the course of the power response. An evaluation of this event for Cycle 7 is provided in Section 5.5.4.

5.1.4 Shutdown CEA Characteristics

The negative reactivity insertion following a reactor trip is a function of the acceleration of the CEA and the variation of CEA worth as a function of position. The Reference Safety Analysis considered this function in three separate parts: 1) the CEA position versus time, 2) the normalized reactivity worth versus rod position, and 3) the total negative reactivity inserted following a scram.

The CEA position versus time assumed in the Reference Safety Analysis was provided as Figure 4.2 in (3). This curve reflects a conservative rod insertion time of 3.0 seconds. This curve is based on results from plant measurements and is not expected to change from cycle to cycle. Furthermore, CEA drop times are measured at each refueling as part of the startup test program to verify this assumption.

The normalized reactivity worth versus rod position assumed in the Reference Safety Analysis was provided as Figure 4.3 in (3). This curve is sensitive to axial power distribution and is based on the minimum reactivity insertion for a variety of axial power distributions. The normalized reactivity worth versus rod position was calculated for limiting Cycle 7 axial power distributions and was compared to the curve assumed in the Reference Safety Analysis. The normalized reactivity worth versus rod position assumed in the Reference Safety Analysis is conservative for Cycle 7 for events limiting at HFP. The normalized control rod negative reactivity insertion versus time curve presented in Figure 4-4 of (3), which was obtained from a synthesis of the aforementioned functions, is likewise conservative in application to Cycle 7 for HFP events.

The normalized reactivity worth versus position curve from (3) was modified for the HZP condition for Cycle 6 (37). A more conservative function was derived from Cycle 6 power distributions at HZP. This was compared with the normalized reactivity worth versus rod position curves determined for limiting Cycle 7 axial power distributions at zero power conditions. The Cycle 6 curve remains bounding for Cycle 7.

Values assumed in the Reference Safety Analysis and for Cycle 7 for the total negative reactivity inserted following a scram are given in Table 5.1. Comparison of the scram worths assumed in the Reference Safety Analyses and the values assumed in the Cycle 7 safety analysis indicate that the Cycle 7 values bound the Reference Safety Analysis values. The values of scram reactivity specified in Table 4.15 bound those assumed in the safety analysis supporting operation of Cycle 7.

5.1.5 Reactor Protective System Setpoints and Time Delays

The reactor is protected by the Reactor Protective System (RPS) and Engineered Safeguards Features (ESF). In the event of an abnormal transient, the Reactor Protective System is set to trip the reactor and prevent unacceptable core damage. The elapsed time between the time when the setpoint condition exists at the sensor and the time when the trip breakers are open, is defined as the trip delay time. The values of the trip setpoints and instrumentation delay times used in the Reference Safety Analysis are provided in Table 4.7 of (3).

These trip setpoints are independent of the core loading pattern and thus remain unchanged for Cycle 7. New analysis presented here as a result of a change in safety parameter assumes the same RPS setpoints and time delays as the Reference Safety Analysis.

As indicated in (3) the Reference Safety Analysis assumes no credit for the High Rate of Change of Power, Thermal Margin/Low Pressure, (TM/LP), or Symmetric Offset Trip Functions. This remains unchanged for analysis performed for Cycle 7. Credit is taken for the functioning of the Variable Overpower Trip System in several areas. First, in limiting the initial power distributions considered in setting the Symmetric Offset Trip System setpoints as a function of power level, as discussed in Section 6.0. Second, in limiting the power increase possible during CEA bank withdrawals, as discussed in Section 5.3.1, and other power increase transients for power levels lower than HFP. Third, credit is taken for the floor of the variable overpower trip, setpoint at 20%, for the low power CEA ejection transient.

The TM/LP and Symmetric Offset Trips are cycle dependent. They are derived from the predicted core behavior as described in (8) and Section 6.0. The Cycle 7 setpoints for the TM/LP and Symmetric Offset Trips for 3-loop operation are presented in Section 6.0.

5.2 Summary

Each transient and accident considered in (3) and (6) is reviewed and/or re-evaluated for Cycle 7. The incidents considered are categorized as follows:

1) Anticipated Operational Occurrences (AOO) for which the Reactor Protection System (RPS) assures that no violation of Specified Acceptable Fuel Design Limits (SAFDL) will occur.

2) Anticipated Operational Occurrences (AOO) for which an initial steady-state overpower margin must be maintained in order to assure acceptable results.

3) Postulated Accidents.

The incidents considered are listed in Table 5.2.

In most cases the parameters considered in (3) and (6) and for Cycle 6 in (37) and (38) bound the Cycle 7 values. For those transients where the parameters for Cycle 7 are outside the bounds considered in previous Safety Analyses a new analysis is provided. These are:

1) Boron Dilution
2) CEA Ejection
3) CEA Withdrawal
4) CEA Drop
5) Seized RCP Rotor

A summary of results for Cycle 7 is presented in Table 5.3.

Unless otherwise noted, values of MDNBR reported are based on the use of the W-3 DNB correlation. As discussed in Section 5.1.2, the margins to DNB calculated with the W-3 DNB correlation are more conservative than would be calculated with the proposed YAEC-1 DNB correlation (40). Thus, MDNBR values from previous cycles need not be recalculated with the YAEC-1 DNB correlation for Cycle 7.

5.3 Anticipated Operational Occurrences for which the RPS Assures No Violation of SAFDLs

The incidents in this category were analyzed in the Reference Safety Analyses for the 2630 MWt Uprate and Positive MTC submittals for Maine Yankee, (3) and (6). Selected cases were reanalyzed in (7) to account for changes in the Cycle 4 core physics characteristics. These analyses showed that the incidents in this category do not violate the SAFDLs; the primary coolant system pressure limit; or the 10CFR20 site boundary dose limits. The changes considered in the present analysis do not significantly affect the NSSS response during these transients. This assures that the conclusions relative to primary system pressure and site boundary dose remain valid.

Protection against violation of the SAFDLs continues to be assured by the RPS (Section 6.0). Setpoints are generated for the TM/LP and Symmetric Offset Trips which include the changes in power distributions associated with Cycle 7. Sections 5.3.1 through 5.3.5 review the Anticipated Operational Occurrences for which the RPS assures no violation of the SAFDLs.

5.3.1 Control Element Assembly Bank Withdrawal

The Reference Safety Analysis for this event, (3) and (6) demonstrates that the most severe CEA withdrawal transient occurs for a combination of reactivity addition rate and time in core life that results in the slowest reactor power rise to a level just below the Variable Overpower Trip. This combination of parameters maximizes the core thermal heat flux and core inlet temperature and results in the minimum DNBR. Since the analysis in (3) and (6) assumed the FSAR design power distribution at HFP conditions, which is bounding with respect to limiting combinations of initial power level and power distribution within the symmetric offset LCO for Cycle 7 at HFP and lower power levels, the cases analyzed in (3) and (6) are bounding for Cycle 7.

The Reference Safety Analysis considered parametric analyses at full power (2630 MWt) for Moderator Temperature Coefficient (MTC) and Reactivity Addition Rate. The ranges analyzed were $+0.5 \times 10^{-4}$ delta rho/^{o}F to -3.0×10^{-4} delta rho/^{o}F and 0 to 0.7×10^{-4} delta rho/sec. As indicated in Table 5.1 the Cycle 7 predicted value of MTC, with uncertainty, is -2.8×10^{-4} delta rho/^{o}F. Reference 3 showed the MDNBR to occur at an MTC of -2.9×10^{-4} delta rho/^{o}F for this event, with less negative MTC resulting in higher MDNBR. Table 5.1 also shows a slightly higher maximum rate of reactivity addition for Cycle 7. Reference (3) showed that high rates of reactivity addition result in a faster rise of core power to the Variable Overpower Trip Setpoint and values of MDNBR less limiting than for slower transients.

The CEA bank 5 division into two separately moveable subgroups created a new class of CEA bank withdrawal event, withdrawal of a CEA bank subgroup. Complete withdrawals of each of the CEA bank 5 subgroups were analyzed from initial conditions corresponding to the limiting power distributions within the S/O LCO alarm band for each power level.

The MDNBR for a CEA bank withdrawal event for Cycle 7 occurs for the withdrawal of CEA bank 5 subgroup B from an initial power level of 76% rated power, assuming the CEAs to be initially positioned at the corresponding insertion limit. The MDNBR for this event is 1.37. The peak RCS pressure for a CEA bank withdrawal is provided in Table 5.3 and is less than the ASME design overpressure limit of 2750 psia. Peak RCS pressure for this event remains bounded by the Loss of Load.

5.3.2 Boron Dilution

The Boron Dilution Incident was addressed in (3), (29), (43) and the FSAR. Inadvertent dilution of the Reactor Coolant System was considered under a variety of plant conditions which could result in either an inadvertent power generation or loss of shutdown margin if sufficient time were not available for the operator to take corrective action.

Small changes in boron concentrations resulting from the Cycle 7 reload have an insignificant impact on the conclusions reached in the Reference Analysis. An evaluation of this incident was performed for Cycle 7 for events postulated during refueling, shutdown, startup, hot standby and power operation conditions. Table 5.6 presents a summary of the results of this review for Cycle 7.

5.3.2.1 Dilution During Refueling

Assumptions made in the Cycle 7 evaluation for dilutions during refueling are consistent with those made in (3) and (29).

The limiting dilution in (3) was based on the maximum capacity of the CVCS via the normal makeup and letdown flow paths (200 gpm each). The limiting dilution event in (29) was based on the maximum flow of the Primary Water Makeup System (250 gpm). Both analyses assumed letdown flows equal to the dilution flowrates and minimum reactor vessel water volumes of 2599 ft^3 (volume below lower lip of reactor vessel nozzles). Hence, the Primary Makeup Water System dilution is the limiting dilution under refueling conditions.

Based on the Cycle 7 core loading, the critical boron concentration under cold conditions (68^{o}F) during refueling is 1044 ppm. The minimum initial reactor vessel boron concentration which will prevent an inadvertent criticality within 30 minutes is:

1536 ppm (Case No. 3, Reference 29 dilution)

This value is lower than that specified for Cycle 6, even though the critical boron concentration is higher for Cycle 7 than for Cycle 6. For convenience, Cycle 6 (37) and previous cycles used an overly conservative dilution event, Case No. 4 of (29), together with a minimum reactor vessel water level at the lower lip of the RV nozzles to determine required initial boron concentrations for refueling conditions. Case No. 4 of (29) postulated a valve misalignment during startup of the slipstream purification system for operation with the RHR System. As outlined in (29), in order for Case No. 4 to occur RHR must be operating. Thus, level cannot be as low as postulated. Assumption of the minimum required volume for RHR operation as the dilution volume for Case No. 4 results in Case No. 3 ot (29) becoming the limiting dilution event under refueling conditions. Case No. 3 of (29) involves a valve misalignment during final adjustment of RCS boron concentration via the Primary Water System, after the head has been removed and prior to flooding the refueling cavity. Thus, RV level must be assumed to be at the minimum level for this event. Case No. 3 is conservative, but also is less limiting than the previous use of Case No. 4 with the artificially low volume. Thus the required boron concentration for Cycle 7 refueling is lower than for Cycle 6.

Therefore, it is concluded that if the reactor vessel boron concentration is maintained at or greater than 1536 ppm during Cycle 7 refueling, it would require a continuous dilution at the maximum possible rate for 30 minutes to achieve an inadvertent criticality. This is ample time for the operator to acknowledge the audible count rate signal and take corrective action to cut off the source of the dilution.

5.3.2.2 Dilution During Cold and Hot Shutdown with RCS Filled

Dilutions during cold and hot shutdown were addressed in (43). The assumptions in (43) remain unchanged for Cycle 7. The limiting dilution is via the CVCS (200 gpm), and the RCS is assumed to be filled (no credit taken for pressurizer volume). The highest worth CEA is assumed to be stuck out of the core, the loop stop valves open and either RHR or RCP on. Required minimum Reactor Coolant System initial boron concentrations to allow 15 minutes margin to criticality are listed in Table 5.4, along with the boron concentration required to meet the Technical Specification 5% delta K/K subcriticality requirement for shutdown conditions. The boron concentrations required by the Technical Specification 5% delta K/K subcriticality requirement conservatively bound those required to meet the 15-minute requirement for margin to criticality during boron dilution events.

5.3.2.3 Dilution During Hot and Cold Shutdown with Drained RCS Conditions

Dilutions during shutdown conditions with the RCS partially drained were addressed in (29) and (43). In order to conservatively bound any partially drained configuration with one or more reactor coolant loop isolated, the assumption is made that only the portion of the reactor vessel below the lower lip of the nozzle is filled. With the exception of the CEA of highest worth, which is assumed to be stuck out of the core, and 1% delta K/K of Bank A, which is procedurally withdrawn during cooldowns from hot standby to approximately 350^{o}F, all CEAs are assumed to be inserted in the core. The limiting dilution in this situation is Case No. 3 of (29). As discussed in 5.3.2.1, since Case No. 4 of (29) requires RHR to be operational, it is less limiting, although it was previously used conservatively in (43) by assuming the minimum reactor vessel volume instead of the volume required for RHR operations. This was overly conservative and unduly restrictive on plant operation. Hence, Case No. 3 is now assumed as the appropriate limiting case at the low reactor vessel level condition.

The required initial Reactor Coolant System boron concentrations to allow 30 minutes margin to criticality during drained RCS conditions are given in Table 5.5. Thirty minutes margin is used to bound mid-cycle "refueling" situations where the reactor vessel head may be removed to perform maintenance operations. Table 5.5 also shows the boron concentrations required to meet the 5% delta K/K Technical Specification subcriticality requirement for shutdown conditions. Administrative procedures ensure that the higher of the two values in Table 5.5 are used during drained RCS conditions, thus a minimum of 30 minutes margin to criticality will be provided for the limiting boron dilution event from drained conditions.

5.3.2.4 Dilution During Startup

To evaluate the Boron Dilution Incident during startup for Cycle 7, the same assumptions were used as in the analysis in (3) except for the following:

1) Initial boron concentration is 1536 ppm as determined from the results of the dilution during refueling presented above.

2) The critical boron concentration under cold conditions ($68^{o}F$, 2% uncertainty and all CEAs withdrawn) is 1318 ppm. At hot standby conditions, the critical boron concentration ARO is 1456 ppm. (The assumption of ARO bounds configurations expected during CEA drop time testing when each CEA bank is fully withdrawn prior to individually dropping each CEA. This includes performing the CEA drop time tests during isolated loop conditions.)

Results of a Boron Dilution Incident during startup were reported in Figure 4.3-3 or (3) for a range of initial and critical boron concentrations. Based on the above assumptions, the minimum time required to reduce the boron concentration to the maximum critical value of 1456 ppm is 18.9 minutes. The large change in time to criticality is a result of the lower required boron concentration for refueling conditions discussed in 5.3.2.1 above. Boron dilution for startup is performed under strict procedures and administrative controls. With the CEAs in their normal prescribed position (i.e., nearly all single CEAs fully inserted) the time required to achieve criticality is significantly longer.

5.3.2.5 Dilution at Hot Standby and at Power

The assumptions made for boron dilution events during hot standby and at power in (3) remain the same for Cycle 7. However, the hot standby critical boron concentration with uncertainty is higher, 1456 ppm versus 1275 ppm. The results for Cycle 7 using Figures 4.3-4 and 4.3-5 of (3) are summarized below:

	Maximum Reactivity Insertion Rate
Dilution at Hot Standby	1.02×10^{-5} delta rho/sec
Dilution at Power	9.1×10^{-6} delta rho/sec

The consequences of events with such small reactivity addition rates are bounded by the results reported in Section 5.3.1 for the CEA Withdrawal Incident. Based on the maximum reactivity addition rate it would take approximately 17 minutes of continuous dilution at the maximum charging rate to completely absorb a 1% delta K/K shutdown margin. Because of the available alarms and indications, there is ample time and information to allow the operator to take corrective action.

5.3.2.6 Failure to Borate Prior to Cooldown

Because of the large negative moderator temperature coefficient at EOL, any decrease in primary coolant temperature adds reactivity to the reactor core. Consequently, during the process of cooling down the Primary System for refueling or repairs, it is necessary to borate in order to compensate for this reactivity addition.

The failure to add boron during cooldown was evaluated on the basis of the following assumptions:

(a) The moderator temperature coefficient is the most negative value expected with all rods in the core, including uncertainties.

(b) The reactor is initially 1% subcritical at an average temperature of 550°F (a more conservative condition than the nominal 532°F).

(c) The primary system temperature is reduced at the rate of 100°F/hr, the maximum cooling rate permitted.

In order to make the reactor critical from these initial conditions, the average coolant temperature must be reduced from 550°F to about 495°F. This temperature reduction requires approximately 33 minutes to accomplish. This is ample time for the operator to diagnose the condition and take the necessary corrective action.

5.3.3 Excess Load Incident

An Excess Load Incident is an event where a power-energy removal mismatch is established leading to a decrease in the reactor coolant average temperature and pressure. Hence, when the moderator temperature coefficient of reactivity is negative, unintentional increases in reactor power may occur. Thus, the Excess Load Incident as reported in (3) is most limiting at EOC where the moderator temperature coefficient is most negative.

Tables 5.1 and 4.8 indicate that the Cycle 7 MTC with uncertainty is slightly more negative than the value assumed in the Reference Safety Analysis (3). Since the Cycle 4 MTC was more negative than the value assumed in the Reference Safety Analysis, the Excess Load Incident had been re-analyzed for Cycle 4 in (7). The analysis in (7) assumed a MTC of 3.17×10^{-4} delta rho/°F which is more negative than the value predicted for Cycle 7 including uncertainty. Therefore, the analysis presented in (7) bounds Cycle 7 operation. Values for MDNBR were revised in (33) to account for ENC fuel. Whereas MDNBRs predicted with the FSAR power distribution and Cycle 5 core average heat flux are more limiting than those predicted for Cycle 7 HFP power distributions, the MDNBR results for the HFP incident reported in (33) remain bounding for Cycle 7. The value for MDNBR for the HZP excess load incident is revised for Cycle 7 to account for potential operation at higher than nominal cold leg temperatures. The analysis in (7) assumed the initial cold leg temperature to be 532°F. Assuming an initial cold leg temperature of 554°F results in a value of MDNBR for the zero power excess load incident greater than 3.0 if no credit is taken for the variable overpower trip signal at 20% power. If credit is taken for the variable overpower trip at 20%, MDNBR for this event is greater than 5.0.

5.3.4 Loss of Load Incident

The prime concern of the loss of load transient is protection of the primary and secondary reactor coolant systems from overpressurization.

System parameters which have a major influence upon the severity of the pressure excursion are the initial power level, initial RCS pressure, steam generator pressure, primary and secondary safety relief valve capacities and setpoints, high pressurizer pressure reactor trip setpoint, and moderator temperature coefficient. The Cycle 7 limiting values for these parameters are the same as those assumed in the Reference Safety Analyses. The pressure transient results from the Reference Safety Analyses remain unchanged for Cycle 7. Peak RCS pressure is noted in Table 5.3 and is less than the ASME design overpressure limit of 2750 psia. As discussed in Section 5.1.2, the minimum DNBR for this event is bounded by the Cycle 5 value reported in (33). Therefore, MDNBR is greater than 1.77.

5.3.5 Loss of Feedwater Incident

Results of the loss of feedwater incident are sensitive to initial power level, reactor coolant system pressure and temperature, steam generator pressure, steam generator inventory, primary and secondary relief and safety valve capacity, moderator temperature coefficient, and low steam generator level trip setpoint. The limiting Cycle 7 values of all of these parameters are unchanged from those assumed in the Reference Safety Analysis. However, as discussed in (37), the Reference Loss of Feedwater Analysis did not account for the loss of SG inventory due to the continuous blowdown of water during normal operation to the feedwater demineralizing system. The additional loss of SG inventory was accounted for in (37) and reduced the minimum time available to initiate auxiliary FW from the reference analysis value of 15 minutes to 13 minutes. Maine Yankee has automated the initiation of the Auxiliary Feedwater System (AFW). Following a loss of main feedwater, the Auxiliary Feedwater System will start immediately upon receipt of a low Steam Generator level signal. Hence, Steam Generator dryout is not expected to occur for this event. As discussed in Section 5.1.2 the MDNBR for this event is bounded by the Cycle 5 value of 1.68. Peak RCS pressure for this event is provided in Table 5.3 and is less than the ASME design overpressure limit of 2750 psia.

5.4 Anticipated Operational Occurrences Which are Dependent on Initial Overpower Margin for Protection Against Violation of SAFDLs

The incidents in this category rely on the provision of adequate initial overpower margin to assure that they do not result in violation of the SAFDLs. In order to demonstrate that the incidents of this category do not violate the SAFDLs, primary system pressure limits, or site boundary dose limits (10CFR20) under Cycle 7 conditions, these incidents are reviewed here with the parameters listed in Table 5.1.

5.4.1 Loss-of-Coolant Flow

Results of the Loss-of-Coolant Flow Analysis are sensitive to initial overpower DNB margin, rate of flow degradation, low reactor coolant flow reactor trip setpoint, available scram reactivity, and moderator temperature coefficient. The limiting overpower DNB margin within the LCO envelope for Cycle 7 is greater than that assumed in the reference analysis in (6) of this event inherent in the use of the FSAR design power distribution, as discussed in Section 5.1.2. The assumptions pertaining to MTC, low reactor coolant flow trip setpoint, and rate of coolant flow degradation remain the same as in the Reference Safety Analysis for this event in (6). The available shutdown margin assumed for Cycle 7 (Table 5.1) bounds that assumed for the analysis in (6). Thus, the minimum DNBR for the three pump loss of flow from 100% power using the FSAR design power distribution is greater than 1.50.

5.4.2 Full Length CEA Drop

The drop of a full length CEA results in a distortion of the core power distribution and could lead to the violation of Specified Acceptable Fuel Design Limits (SAFDL). As discussed in Section 6.4.1, the LCO symmetric offset band is designed to restrict permissible initial operating conditions such that the SAFDL for DNB and fuel centerline melt are not exceeded for this incident.

Previous analysis of this incident identified the limiting transient as one initiated from near full power. To cover all potentially limiting conditions the CEA drop for Cycle 7 was evaluated from power levels ranging from 66% to 100% of 2630 MWt. The CEA drop was found not to be limiting for power levels less than 66%.

Power distributions used in the evaluation of DNBR and proximity to fuel centerline melt were selected at each power level from the limiting cases within the symmetric offset LCO alarm band, as well as the FSAR design power distribution, in the manner described in Section 6.4.1.

The initial percent increase in peaking as a function of dropped CEA worth for Cycle 7 is given in Figure 4.10. The value for the maximum increase in peaking for any dropped CEA from Figure 4.10 was conservatively (Section 4.9.3.1) applied at each power level considered.

The CEA drop analysis also considers the increased peaking which results from xenon redistribution during the period of time operation with a dropped CEA is allowed by the Technical Specifications (Section 4.9.3). The percent increase in peaking from Figure 4.10 was conservatively augmented by the increase in peaking due to xenon redistribution at subsequent points in time, assuming operation consistent with the power level reductions required by the proposed changes to the Technical Specifications. The margins to the SAFDLs were then determined for the limiting power distributions within the symmetric offset LCO band allowed for the existing power level at any point in time assuming the CEAs to be inserted no deeper than allowed by the insertion limit associated with the pre-drop power level. This is consistent with the proposed changes to the Technical Specifications outlined in Section 4.9.3.

The worst case full length CEA drop with respect to DNB reported in (3) was the minimum worth CEA that results in the maximum increase in peaking. Thus, for conservatism the plant response assumed in the Cycle 7 evaluation was based on a worth of 0.10% delta rho.

The results of the DNB evaluation for Cycle 7 indicate that the limiting full length CEA drop is one initiated from 100% power with a MDNBR 1.38 using the FSAR design power distribution. All other cases were found to be less limiting.

The worst case full length CEA drop with respect to fuel centerline melt is one initiated from power distributions at the edge of the symmetric offset LCO band at each power level. The maximum allowable steady-state linear heat rate required to assure that the maximum linear heat generation rate after the drop does not violate the SAFDL of 21 kW/ft is given in Table 5.1. These limits are reflected in deriving the LCO band on symmetric offset presented in Section 6.0.

5.5 Postulated Accidents

The incidents in this category were previously analyzed in (3), (6), (7), (37), and (38). For the conditions in those reports it was demonstrated that each of these incidents met the appropriate accident criteria. Each of these incidents have been reviewed below and results of new analyses reported when Cycle 7 conditions warranted re-analysis of the accident.

5.5.1 Steam Line Rupture

The key parameters which have changed are the EOC Moderator Temperature Defect and available shutdown CEA worth. As indicated in Table 4.9, the Moderator Temperature Defect is slightly smaller for Cycle 7 than assumed in (38), the Cycle 6 Analysis. In addition, new CEAs have been added to the MY core which result in higher available shutdown CEA worths than considered for Cycle 6. Boron reactivity worths for Cycle 7 are higher than those used in calculating the injected boron reactivity in (38). The minimum shutdown reactivities reported for the limiting steam line break transients for Cycle 6 (38) are, therefore, conservative and bound Cycle 7.

Reference (38) identified the limiting steam line breaks for Maine Yankee to be:

HFP - SLB with failure of MFWRV to close with RCP tripped by operator at SIAS plus 30 seconds.

HZP - SLB with failure of one HPSI pump to start with RCP tripped by operator at SIAS plus 30 seconds.

In both cases the reactor was predicted to remain subcritical. These analyses contain a number of conservative assumptions; for example, pure steam blowdown, constant heat transfer from primary to secondary, use of worst case uncertainties on key parameters, and the assumption of the highest worth CEA stuck out of the core. We have shown (47) that the conservatism used, in terms of reactivity, can account for at least 4.7% delta rho when evaluating margins of subcriticality. Thus, the subcriticality margins reported in (38) along with the substantial reactivity tied up in conservatism assure that the Maine Yankee reactor will remain subcritical following a major SLB.

The analysis in (38) evaluated cases with and without off-site power. Although the cases without off-site power were severe cases, the analysis in (38) concluded that the loss of off-site power cases were not the most limiting events. The analysis in (38) of the HZP loss of off-site power case assumed that two HPSI pumps were available. If we assume that one HPSI pump fails to start for this case, then the loss of off-site power case becomes slightly more limiting at zero power. The results of re-analysis demonstrate that for the zero power SLB, the limiting transient is the case with failure of a HPSI pump to start with coincident loss of off site power. However, the margin to criticality for this case is no less than the minimum reported in (38). Thus, the minimum required shutdown margins to preclude a return to critical are based on the analysis in (38).

Administrative restrictions were imposed on allowable CEA insertions in (42). This was done in order to provide sufficient shutdown margin for MSLB events when the plant is operated with indicated cold leg temperatures greater than the normal RCS temperature program. This practice will be continued during Cycle 7 operation.

5.5.2 Steam Generator Tube Rupture

The parameter in Table 5.1 to which this accident is most sensitive is primary system pressure. The analysis of this accident in (3) and the FSAR assumed the nominal primary system pressure of 2250 psia and demonstrated that the resulting site boundary dose would not violate 10CFR100 limits. Since, the nominal operating pressure remains unchanged from the value assumed in the Reference Analysis the conclusions of (3) remain valid for this cycle.

5.5.3 Seized Rotor Accident

The consequences of the seized rotor accident are sensitive to the initial overpower DNB margin, core power distribution, assumed rate of flow degradation, low reactor coolant flow trip setpoint, MTC and to the primary to secondary leakage flow rate. Most of these factors remain unchanged. As discussed in Section 5.1.2, the thermal margin for Cycle 7 is bounded by that of Cycle 5, thus the limiting radial power factor determined in the Cycle 5 analysis remains limiting for Cycle 7.

The fraction of fuel failure for the seized rotor accident was re-analyzed using the limiting HFP conditions for Cycle 7.

Analysis results indicate that less than 8% of the fuel experiences DNBR less than 1.3 (W-3) as compared to 8.7% for Cycle 6. These results were calculated assuming a very conservative time for reactor trip based on the rate of flow decrease from a RCP coastdown instead of seizure together with an instantaneous reduction to two-thirds the three RCP flow rate. Radiological Release Analyses show these results to be well within the bounds of 10CFR100. Furthermore, as pointed out in (41), DNB is an overly conservative and inappropriate criteria for clad failure. It is expected that a detailed fuel temperature-time history analysis for this event would show no fuel failure. Use of the YAEC-1 DNB correlation, with a DNBR limit of 1.17, results in less than 3% of the fuel experiencing DNB. Therefore, the predicted consequences of this accident are acceptable.

5.5.4 CEA Ejection

The consequences of a CEA Ejection Accident are most sensitive to ejected CEA worth, effective delayed neutron fraction (Beff), and post-ejected 3-D peak. Specifically, the most severe transient occurs for the maximum ejected CEA worth, minimum effective delayed neutron fraction and maximum post-ejected 3-D peak. A comparison between the limiting values assumed in past Safety Analysis, (3), (6) and (37), the Cycle 6 analysis, and those predicted for Cycle 7 including uncertainties is presented in Table 5.1.

In each case, the values for Cycle 7 exceed one or more of the parameters assumed in the Reference Safety Analysis. A reanalysis of each event has been performed using the methodology described in (10) and (35).

A summary of the results for the HFP CEA Ejection Accident is presented in Table 5.7. Reference Safety Analysis results for this accident indicated no clad damage occurred and only a small percentage of fuel experienced incipient centerline melting. Reanalysis of the HFP cases for Cycle 7 shows less than .03% clad damage (radially averaged fuel enthalpy greater than 200 calories per gram), less than 0.6% of the fuel experiences incipient centerline melting (centerline fuel enthalpy greater than 250 calories per gram), and less than 0.2% of the fuel is fully molten (centerline fuel enthalpy greater than 310 calories per gram) at the centerline.

A summary of the results for the HZP CEA Ejection Accident for Cycle 7 are presented in Table 5.8. Less than 0.5% clad damage occurs, less than 0.8% of the fuel experiences incipient centerline melting, and less than 0.3% is fully molten at the centerline.

With the exception of the reduction in uncertainty applied to the Doppler defect and use of the Doppler defect calculated from the case specific pre-ejection power distribution (discussed in Section 4.9.1.3.1), the analysis assumptions are unchanged from those presented in (6). No Doppler weighting factors were used. The use of point kinetics was shown in (49) to be a conservative method of analysis for this event.

5.5.5 Loss of Coolant

5.5.5.1 Introduction and Summary

The LOCA analyses performed for Cycle 5 and Cycle 6 presented in (33, 37) serve as the reference LOCA analyses for Cycle 7. The results of the review of the reference analyses contained herein conclude that Appendix K criteria are met with the Cycle 7 fuel types operating at the following limits (same as for Cycle 5 and Cycle 6):

Fresh Fuel	13.5 kW/ft	CAB less than 792 MWD/MTU	X/L greater than 0.50
(Type L)	14.0 kW/ft	CAB greater than 792 MWD/MTU	X/L greater than 0.50
	16.0 kW/ft		X/L less than 0.50
Exposed Fuel	14.0 kW/ft		X/L greater than 0.5
(Types E, J, K)	16.0 kW/ft		X/L less than 0.5

where CAB is cycle average burnup and X/L is fraction of core height.

Fresh fuel is comprised only of ENC Type L fuel. Exposed fuel is comprised of ENC Types J and K fuel and CE Type E.

5.5.5.2 Large Break LOCA Analysis

Large break LOCA calculations were performed for Cycle 5 and Cycle 6 using YAEC's WREM based Generic PWR ECCS Evaluation Model (16). These calculations consisted of break spectrum, burnup and axial power distribution analyses for Cycle 5 (33) and selected sensitivity studies for Cycle 6 (37). The following subsections discuss the review of these analyses for application to Cycle 7.

5.5.5.2.1 Break Spectrum Analysis

The Cycle 7 core is slightly different from the reference core (Cycle 5) in the area of core hydraulics and core physics. This is due to the Cycle 7 core loading of 72 fresh ENC fuel assemblies, 144 exposed ENC fuel assemblies and 1 exposed CE assembly versus the reference core loading of 72 fresh ENC assemblies and 145 exposed CE assemblies.

In the reference core break spectrum analysis, the hot assembly region was modeled as an Exxon assembly and the core region modeled as an average of the Exxon and CE fuel assemblies. In Cycle 7, the hot assembly loss coefficients remain unchanged while the core region loss coefficients differ slightly from reference core values due to the fuel loading. A 4.2% increase in the overall average core region forward loss coefficient was noted, while the overall average core region reverse loss coefficient increased by 2.6%. These are very minor perturbations to the blowdown hydraulics when one considers the overall system hydraulics driving the blowdown response. As such, no reanalysis of the break spectrum blowdown response is deemed necessary due to the slight differences in core hydraulics as noted above. Similarly, hot channel and reflood analysis will be likewise unaffected to the extent that results will not change significantly from the reference analysis.

With regards to core physics assumptions, those utilized in the reference analysis bound the expected Cycle 7 parameters. The most important kinetics parameter in the blowdown calculation is the moderator density defect. Moderator density defect data employed in the reference analysis was assumed to be the least negative that could exist at Maine Yankee and as such, this data amply bounds Cycle 7 data. Other kinetics data used in the reference analysis such as B/l*, and Doppler defect are representative of Cycle 7 values. Thus, the reference break spectrum analysis is unaffected by specific Cycle 7 physics parameters since bounding assumptions are employed.

The break spectrum analysis is performed at the maximum stored energy point for fresh fuel which occurs at BOL. Thus, enhanced fission gas release, which is invoked at burnups beyond 20,000 MWD/MTU, does not affect the break spectrum analysis results.

Similarly, it was shown in the reference analysis that the effect of NUREG-0630 clad swelling and rupture data was to reduce peak clad temperatures. Therefore, the reference cycle break spectrum analysis is conservative with respect to this data.

Summarizing, the reference analysis break spectrum is directly applicable to Cycle 7 operation because the minor cycle dependent design differences will not change the break spectrum results. Additionally, clad swelling and rupture and fission gas enhancement do not alter the break spectrum results. Therefore, the worst large break identified in reference analysis (1.0 DECLS) is valid for Cycle 7.

5.5.5.2.2 Burnup Sensitivity Studies

The burnup sensitivity analyses performed for Cycles 5 and 6 (33, 37) were examined for applicability to Cycle 7. These analyses examined Exxon fuel from BOL up through two cycles worth of exposure. CE fuel was examined for burnups representing up to three cycles of exposure. Cycle 7 will now have Exxon fuel exposed for three fuel cycles. Thus, the only burnup point not explicitly addressed in the reference analyses is three-cycle exposed Exxon fuel (i.e., Type J fuel at EOC7).

Since the break spectrum results do not change and bounding burnups were assumed for each fuel type at the points analyzed, the Cycle 5 and Cycle 6 burnup study results would be directly applicable to Cycle 7 except for one point, i.e., EOC7 for Batch J (ENC) fuel since this point had not been analyzed. The Batch J (ENC) fuel has been evaluated at EOC7 using the same top skewed axial power profile assumed in the reference analyses. The results given in Table 5.9 demonstrate that operating Cycle 7 at the linear heat generation limits specified for positive axial offsets in Section 5.5.5.1 satisfies Appendix K criteria (31).

5.5.5.2.3 Cosine Axial Power Distribution Study

The cosine axial power distribution study performed in the reference analyses was similarly re-examined for its applicability to Cycle 7. As stated above, the three-cycle exposed Exxon fuel had not been evaluated (J, EOC7). This point, EOC7 for Batch J (ENC) fuel, was rerun using the same cosine power shape presented in (33) and (37). The results of this cosine axial power distribution case are summarized in Table 5.10. Based on these results, the operation of Cycle 7 at linear heat generation limits specified for zero or negative axial offset in Section 5.5.5.1 satisfies Appendix K criteria (31).

5.5.5.3 Small Break LOCA Analyses

The small break LOCA analysis performed by Combustion Engineering (32) for Cycle 4 considered a spectrum of cold leg breaks varying in size from 0.1 to 0.5 ft^2. Results showed that the limiting break size is the 0.5 ft^2 break with a peak clad temperature of 1348°F, well below the acceptance criteria of 10CRF50.46. A demonstration analysis of the limiting break performed for Cycle 5 (33) utilizing Yankee methodology yielded a peak clad temperature of 1230°F, well below the 10CFR50.46 acceptance criteria and Maine Yankee large break results. Thus, small break LOCAs for Maine Yankee were shown to be not limiting. Since small break LOCA results are insensitive to fuel type, being primarily decay heat driven and system dependent, these small break results for Maine Yankee to date are directly applicable to Cycle 7.

5.6 Methodology Revisions

5.6.1 YAEC-1 DNB Correlation

The W-3 CHF correlation with a DNBR limit of 1.30 has been used, in conjunction with the COBRA-IIIC program (9), as the Specified Acceptable Fuel Design Limit (SAFDL) on DNBR in the Reference Safety Analyses of Maine Yankee. Reference 40 proposed a new DNBR SAFDL for Maine Yankee based on use of the YAEC-1 CHF correlation with a DNBR limit of 1.17 and use of the COBRA-IIIC code. The proposed DNBR SAFDL based on YAEC-1 has been used in the generation of the LSSS and TM/LP trip setpoints, for Cycle 7. The margin available between normal operating pressure and required trip pressure is significantly increased by use of the new SAFDL. This will result in greater operational flexibility and plant reliability due to a reduction in the potential for spurious (noise) trips, and the ability to operate at high power over the entire width of the Symmetric Offset LCO band, while maintaining adequate margin to trip pressures. This application of YAEC-1 requires approval of the reference topical report (40).

The YAEC-1 CHF correlation is based on the results of CHF experiments performed at Columbia University with geometries identical to Maine Yankee 14 x 14 fuel. The CHF limit of 1.17 was statistically derived in (40) as the 95/95 confidence level limit for DNBR using the YAEC-1 correlation and the COBRA-IIIC code. As such, it provides the required confidence that the probability of a fuel rod experiencing DNB during normal operation and anticipated operational occurrences is small.

The proposed DNB SAFDL has also been used to quantify the fraction of fuel which experiences DNB and is presumed to release its gap activity for purposes of radiological consequence analysis of the Seized RCP rotor event for Cycle 7 in Section 5.5.3.

Comparisons of initial steady-state operating margins to DNB were reported in Table 5.1 for both the W-3 and YAEC-1 SAFDLs.

It is anticipated that future safety analyses will be performed using the YAEC-1 DNB SAFDL exclusively.

5.7 Plant Hardware Modifications

5.7.1 Placement of Non-Scrammable Full Length CEAs Into the Former Part Length CEA Locations

This change is discussed in Sections 3.1.5, 4.4 and 4.9.1. The effects of the additional CEAs have been accounted for in the review of the CEA Drop, Steam Line Break, CEA Ejection and CEA bank withdrawal transients, as well as in the generation of the Symmetric Offset and Thermal Margin/Low Pressure Trip Setpoints and Symmetric Offset LCO band.

Table 5.1

Maine Yankee Safety Parameters

Parameter	Units	Reference Cycle 3 Including Uncertainties	Cycle 6 Including Uncertainties	Cycle 7 Including Uncertainties
- Planar Radial Peaking Factor Bank 5 Inserted to PDIL		1.58 (100%)[2]	1.609 (100%)[2]	1.58 (100%)[2]
- Axial Peak for Shape Resulting in MDNBR		1.42 (100%)[2]	1.43 (100%)[2]	1.49 (100%)[2]
- Augmentation Factors		1.0 to 1.067[1]	1.0 to 1.057[1]	1.0 to 1.047[1]
- Moderator Temperature Coefficient	10^{-4} delta rho/^{o}F	0 to -2.74	+.5 to -3.05	+.5 to -2.80
- Ejected CEA Worth				
BOC Zero Power	% delta rho	.396	.237	.514
BOC Full Power	% delta rho	.210	.150	.234
EOC Zero Power	% delta rho	.544	.394	.692
EOC Full Power	% delta rho	.230	.222	.364
- Ejected CEA 3D Peak				
BOC Zero Power		13.32	8.42	11.21
BOC Full Power		5.53	4.05	5.22
EOC Zero Power		14.08	11.56	13.15
EOC Full Power		5.59	6.06	6.79
- Dropped CEA Integral Worth	% delta rho	0 to .30	0 to .30	0 to .30

Table 5.1
(Continued)

Parameter	Units	Reference Cycle 3 Including Uncertainties	Cycle 6 Including Uncertainties	Cycle 7 Including Uncertainties
- Dropped CEA Integral Radial Peak		Figure 4.4-1 of Reference 3	Figure 4.8 of Reference 37	Figure 4.10
- Power Level (including 2% uncertainty)	MWt	2683	2683	2683
- Maximum Reactor Coolant Inlet Temperature	^oF	554	546 - 554	546 - 554
- Reactor Coolant System Pressure	psia	2200 - 2300	2050 - 2300	2050 - 2300
- Reactor Coolant System Flow Rate	10^6 lbs/hr	134.6(5)	134.6(5) - 136.0(6)	134.6(5) - 136.0(6)
- Axial Power Distributi n	Symmetric Offset	+.14 (100%)	+.14 (100%)	+.14 (100%)
- Power Dependent Insertion Limit		Figure 4.9 of Reference 20	Figure 4.9 of Reference 20	Figure 4.9
- Initial Steady-State(4)(7) Minimum DNB Ratio	W-3(3) YAEC-1	1.881 1.977	1.836 1.923	1.872 1.961
- Initial Allowable Linear Heat Generation Rate (excluding LOCA)(8)	kW/ft	16.2	18.1	100(10) 18.4(11) 90 18.3 80 18.0 66 17.5
- Maximum Possible Rate of Reactivity Addition(9)	delta rho/sec	0.7×10^{-4}	0.86×10^{-4}	1.24×10^{-4}

Table 5.1
(Continued)

Parameter	Units	Reference Cycle 3 Including Uncertainties	Cycle 6 Including Uncertainties	Cycle 7 Including Uncertainties
- Available Scram Reactivity Assumed in Safety Analysis	% delta rho			
HFP, BOC		4.0	4.0 6.39[13][14]	4.0
HZP, BOC		2.0	2.0 2.9[15]	2.0 2.9[15]
HFP,EOC		5.7 6.5[12]	5.7 6.74[16]	5.7 6.74[16]
HZP,EOC		2.9	2.9 4.21[17]	2.9 3.2[18] 4.21[17]

Table 5.1
(Continued)

Notes

1) Applies only in fuel centerline melt calculations.

2) With limiting cycle power distribution limited by symmetric offset pre-trip alarm. Power level refers to conditions allowed by PDIL for that cycle.

3) Using the COBRA-IIIC cold wall correction factor.

4) FSAR design power distribution ($F^N_{delta\ H}$ = 1.49, F_z = 1.68).

5) Based on Reactor Coolant System pressure of 2200 psia and reactor coolant inlet temperature of 554°F.

6) Based on Reactor Coolant System pressure of 2050 psia and reactor coolant inlet temperature of 546°F.

7) Includes 2% calorimetric power uncertainty and 3% allowance for maximum tilt allowed by Technical Specification 3.10.

8) Based on CEA Drop Incident described in Section 5.3.2 and energy deposited in fuel rod.

9) For CEA bank withdrawal transient.

10) Power level % of Rated Thermal Power (RTP).

11) KW/FT.

12) EOC, HFP steam line break assumed 6.5% delta rho.

13) Loss of coolant flow assumed 6.39% delta rho.

14) HFP, EOC CEA ejection assumed 6.39% delta rho.

15) HZP, BOC CEA ejection assumed 2.9% delta rho.

16) HFP, EOC steam line break assumed 6.74% delta rho.

17) HZP, EOC steam line break assumed 4.21% delta rho.

18) HZP, EOC CEA ejection assumed 3.2% delta rho.

Table 5.2

Maine Yankee
Cycle 5 - Incidents Considered

A. Anticipated Operational Occurrences for which the RPS assures no violation of SAFDLs:

1. Control Element Assembly Withdrawal
2. Boron Dilution
3. Excess Load
4. Loss of Load
5. Loss of Feedwater

B. Anticipated Operational Occurrences which are dependent on Initial Overpower Margin for protection against violation of SAFDLs:

1. Loss of Coolant Flow
2. Full Length CEA Drop

C. Postulated Accidents:

1. CEA Ejection
2. Steam Line Rupture
3. Steam Generator Tube Rupture
4. Seized Rotor
5. Loss of Coolant

Table 5.3

Maine Yankee Cycle 7
Safety Analysis
Summary of Results

Incident	Section	Criteria	Reference Analysis	Cycle 6	Cycle 7
CEA Withdrawal	5.2.1	MDNBR = 1.30** RCS pressure 2750 psia	MDNBR = 1.51 RCS pressure 2570 psia	MDNBR = 1.44 RCS pressure 2570 psia	MDNBR greater than 1.44 RCS pressure 2570 psia
Boron Dilution	5.2.2	Subcritical: Sufficient time for operator action Critical: MDNBR 1.30**	Subcritical: Refueling-65 min. Startup-3.2 hours Critical: Bounded by CEA withdrawal	Subcritical: Refueling-30 min. Startup-1.5 hours Critical: Bounded by CEA withdrawal	Subcritical: Refueling-30 min. Startup-18 minutes Critical: Bounded by CEA withdrawal
CEA Drop	5.3.2	MDNBR 1.30** Local LHGR less than 21 kW/ft	MDNBR 1.36 Local LHGR less than 21 kW/ft	MDNBR 1.43 Local LHGR less than 21 kW/ft	MDNBR 1.38 Local LHGR less than 21 kW/ft
Loss of Coolant Flow	5.3.1	MDNBR 1.3**	MDNBR = 1.50	MDNBR = 1.53	MDNBR greater than 1.50
Seized Pump Rotor	5.4.3	A sufficiently low fraction of rods with MDNBR less than 1.3**	4.2% of rods with MDNBR less than 1.3	8.7% of rods with MDNBR less than 1.3	8% of rods with MDNBR less than 1.3 3% of rods with MDNBR less than 1.17**
Excess Load	5.2.4	MDNBR = 1.3**	MDNBR = 1.8	MDNBR = 1.68	MDNBR greater than 1.68
Loss of Load	5.2.5	MDNBR = 1.3** RCS pressure 2750 psia	MDNBR = 1.85 RCS pressure 2689 psia*	MDNBR = 1.77 RCS pressure 2689 psia*	MDNBR greater than 1.77 RCS pressure 2689 psia*

Table 5.3 (Continued)

Incident	Section	Criteria	Reference Analysis	Cycle 6	Cycle 7
Loss of Feedwater	5.2.6	1. Sufficient time for initiation of auxiliary feedwater system 2. RCS pressure 2750 psia	1. Aux. feedwater system required 15 minutes following event 2. Peak RCS pressure 2600 psia*	1. Aux. feedwater water system required 13 min. following event 2. Peak RCS pressure 2600 psia*	1. Aux. feedwater water system required 13 min. following event 2. Peak RCS pressure 2600 psia*
Steam Line Rupture	5.4.1	Maintain fuel rod integrity	Fuel rod integrity is maintained since reactor does not return critical	Fuel rod integrity is maintained since reactor does not return critical	Fuel rod integrity is maintained since reactor does not return critical
Steam Generator Tube	5.4.2	10CFR100	Radiological does well within 10CFR100	Reference analysis unchanged by Cycle 6 reload	Reference analysis unchanged by Cycle 7 reload
CEA Ejection	5.4.4	10CFR100	No clad damage thus no radiological release	No clad damage thus no radiological release	Less than 1% clad damage thus radiological release bounded by LOCA
LOCA	5.4.5	10CFR100	10CFR100	Doses unaffected by Cycle 6 reload	Doses unaffected by Cycle 7 reload
		10CFR50.46	Cycle 5 is new reference analysis	PCT=2181°F clad oxidation = 10.25% Less than 1% hydrogen generation	PCT=2149°F clad oxidation = 5.52% Less than 1% hydrogen generation
Steam Line Rupture Outside Containment	-	10CFR100	10CFR100	Reference analysis unchanged by Cycle 6 reload	Reference analysis unchanged by Cycle 7 reload
Feedwater Line Rupture Outside Containment	-	10CFR100	Bounded by steam line rupture	Reference analysis unchanged by Cycle 6 reload	Reference analysis unchanged by Cycle 7 reload

Table 5.3 (Continued)

Incident	Section	Criteria	Reference Analysis	Cycle 6	Cycle 7
Containment Pressure	-	Peak pressure 55 psig containment design pressure	Peak pressure 55 psig	Reference analysis unchanged by Cycle 6 reload	Reference analysis unchanged by Cycle 7 reload
Fuel Handling Incident	-	10CFR100	10CFR100	Reference analysis unchanged by Cycle 6 reload	Reference analysis unchanged by Cycle 7 reload
Waste Gas System Failure	-	10CFR100	10CFR100	Reference analysis unchanged by Cycle 6 reload	Reference analysis unchanged by Cycle 7 reload
Spent Fuel Cask Drop	-	10CFR100	NA	NA	NA
Radioactive Liquid Waste System Leak	-	10CFR100	10CFR100	Reference analysis unchanged by Cycle 6 reload	Reference analysis unchanged by Cycle 7 reload

* (3), (6), and (37) reported RCS pressure corresponding to peak pressurizer pressure, peak system pressure occurs at the RCP discharge and is given above for each event.

** (40) proposed changes in this criteria to allow use of a DNBR limit of 1.17 with the YAEC-1 CHF correlation (see Section 5.6).

Table 5.4

Required Initial RCS Boron Concentrations to Allow 15 Minutes Margin to Criticality for Dilutions from Shutdown Conditions with the RCS Filled

		Required Ci (ppm)	
		Boron Dilution	5% delta K/K*
BOC			
	532°F	878	1046
	300°F	1015	1112
	68°F	1003	1114
EOC			
	532°F	0	146
	300°F	244	331
	68°F	276	378

* Margin of subcriticality required by Technical Specifications for shutdown conditions.

Table 5.5

Required Initial RCS Boron Concentrations
to Allow 30 Minutes Margin to Criticality
for Dilutions from Shutdown Conditions
with the RCS Drained*

		Boron Dilution	Required Ci (ppm) 5% delta K/K**
BOC			
	532°F	1225	1046
	300°F	1433	1112
	68°F	1459	1114
EOC			
	532°F	0	146
	300°F	319	331
	68°F	400	378

* Level = lower lip of RV nozzles.

** Margin of subcriticality required by Technical Specifications for shutdown conditions.

Table 5.6

Summary of Boron Dilution Incident Results
for Cycle 7

(A) Operating Mode	(B) Minimum Technical Specification Shutdown Margin Requirement	(C) Minimum Time to Absorb (B) Minutes	(D) Acceptance Criteria(5)
Refueling	5% delta K/K	30	30
Cold Shutdown			
Filled RCS	5% delta K/K	15	15
Drained RCS	5% delta K/K(1)	30(1)	30(1)
Hot Shutdown			
Filled RCS	5% delta K/K	15	15
Drained RCS	5% delta K/K(1)	30(1)	30(1)
Startup	3.2% delta K/K(6)	18.9(2)	15
Hot Standby	3.2% delta K/K(6)	17(3)	15
Power Operation	3.2% delta K/K(6)	17(3)	15
Failure to Borate Prior to Cooldown	3.2% delta K/K(6)	33(4)	15

(1) 30 minutes margin is used to provide sufficient margin for drained conditions where the head is removed. These are classed as "refueling" conditions in the Technical Specification definition of refueling. Margin quoted is for initial boron concentrations administratively required for these conditions.

(2) Margin quoted assumes initial boron concentration at refueling value for Cycle 7, 1536 ppm.

(3) Time to absorb a 1% delta K/K shutdown margin. Times to absorb specified 3.2% delta K/K minimum shutdown margin would be significantly longer.

(4) Cooldown rate assumed to be 100^{o}F.

(5) Time span between event initiation and criticality.

(6) Minimum value for shutdown margin specified in Technical Specification 3.10.

Table 5.7

Cycle 7
Full Power CEA Ejection Accident Results

	BOC	EOC
Total Average Enthalpy of Hottest Fuel Pellet (cal/gm)	198	203
Total Centerline Enthalpy of Hottest Fuel Pellet (cal/gm)	339	351
Fraction of Rods that Suffer Clad Damage (Average Enthalpy 200 cal/gm)	0	.0002
Fraction of Fuel Having at Least Incipient Centerline Melting (Centerline Enthalpy 250 cal/gm)	.0010	.0060
Fraction of Fuel Having at Least a Full Molten Centerline Condition (Centerline Enthalpy 310 cal/gm)	.0003	.002

Table 5.8

Cycle 7
Zero Power CEA Ejection Accident Results

	BOC	EOC
Total Average Enthalpy of Hottest Fuel Pellet (cal/gm)	162	242
Total Centerline Enthalpy of Hottest Fuel Pellet (cal/gm)	212	362
Fraction of Rods that Suffer Clad Damage (Average Enthalpy 200 cal/gm)	0	.0044
Fraction of Fuel Having at Least Incipient Centerline Melting (Centerline Enthalpy 250 cal/gm)	0	.0078
Fraction of Fuel Having at Least a Fully Molten Centerline Condition (Centerline Enthalpy 310 cal/gm)	0	.0026

Table 5.9

Maine Yankee Cycle 7

Burnup Sensitivity Study

Results

Case* Description	PLHGR (kW/ft)	Hot Pin Burnup (MWD/MTU)	Peak Cladding Temperature, ^{o}F	Cycle 7 Equivalent
LBOC7	13.5	0.	1978	JBOC5
L25D7	14.0	1294	2140	J25D5
LEOC7	14.0	17452	2149	JEOC5
KBOC7	14.0	17452	2149	JEOC5
KEOC7	14.0	33573	2109	JEOC6
JBOC7	14.0	33573	2109	JEOC6
JEOC7	14.0	44350	1943	---

* Key example - LBOC7 is Batch L fuel at beginning of Cycle 7 conditions.

Table 5.10

Maine Yankee Cycle 7

Cosine Axial Power Shape Study

Results

Case* Description	PLHGR (kW/ft)	Hot Pin Burnup (MWD/MTU)	Peak Cladding Temperature, ^{o}F	Cycle 7 Equivalent
LBOC7	16.0	0.0	1849	JBOC5
KBOC7	16.0	17452	1854	IBOC5
JBOC7	16.0	30473	1768	IBOC6
JEOC7	16.0	44350	1926	---

* Key example - LBOC7 is Batch L fuel at beginning of Cycle 7 conditions.

Table 5.11

Cycle 7
Comparison of Thermal Margin for Limiting Cycle 7 Power Distributions to FSAR Design Power Distribution

Power Level	Power Distribution	W-3(1) MDNBR	W-3(1) (Pd-Po)	YAEC-1 MDNBR	YAEC-1 (Pd-Po)
100	FSAR(3)	1.872	28	1.961	42
100	(2)(3)	1.904	28	1.994	42
90	(2)(3)	1.892	30	1.989	43
77	(2)(3)	2.039	31	2.130	43
66	(2)(3)	2.159	40	2.312	50
47.5	(2)(3)	3.523	50	3.73	59
100	FSAR(3)(4)	1.843	25	1.919	39
100	FSAR(3)(5)	1.881	25	1.977	39

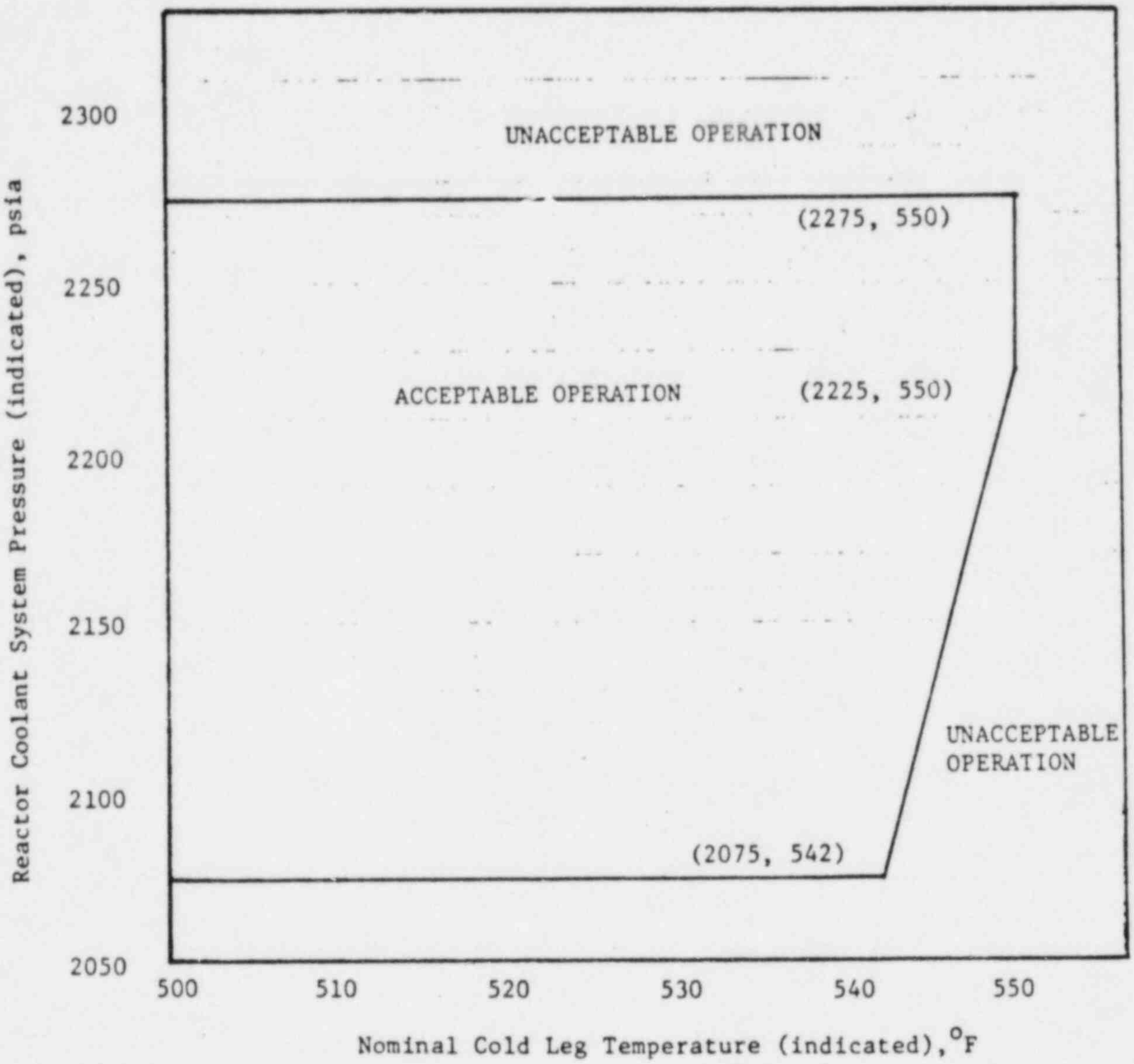
(1) Using COBRA-IIIC cold wall correction factor.

(2) Limiting Cycle 7 power distribution within symmetric offset pre-trip alarm band for indicated power level for Cycle 7.

(3) At 2200 psia, 554 degrees F.

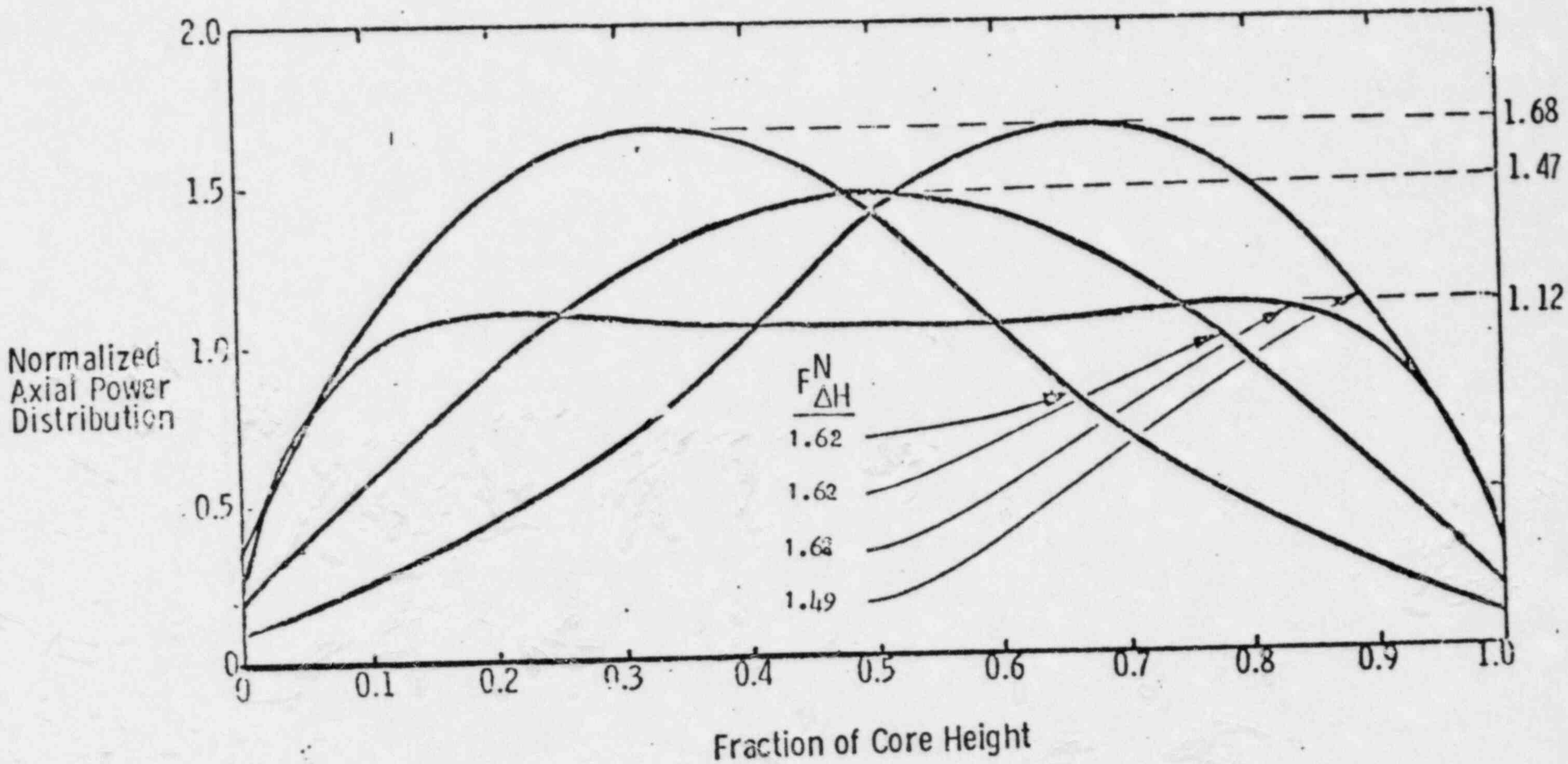
(4) Cycle 5.

(5) Reference Analysis (Cycle 3).



MAINE YANKEE	Allowable 3 Loop Steady State Coolant Conditions	Figure 5.1

Figure 5.2
Maine Yankee Design
Power Distributions



6.0 RPS SETPOINTS

6.1 General

The Maine Yankee Reactor Protection System (RPS) initiates a reactor trip signal on the following parameters:

- Variable nuclear or delta-T overpower
- Thermal margin/low pressure
- Symmetric offset
- Rate of change of nuclear power
- Reactor coolant flow
- Pressurizer pressure
- Steam generator pressure
- Steam generator water level

While one or more of the above trips may provide protection during various anticipated transients and accidents, the final level of protection is provided by the Variable Overpower, Thermal Margin/Low Pressure (TM/LP) and the Symmetric Offset Trips, in conjunction with the plant Limiting Conditions for Operation (LCO). This final level of protection assures that Specified Acceptable Fuel Design Limits (SAFDL) are not exceeded during any Anticipated Operational Occurrence (AOO). Details of the Maine Yankee RPS Setpoint Methodology for calculating the required Limiting Safety System Settings (LSSS) for the TM/LP, Symmetric Offset Trip System and LCO were given in (8). The following paragraphs of this section summarize the method and provides the required LSSS for Cycle 7 operation at 2630 MWt. The LSSS for Cycle 7 operation at 2630 MWt reflect a variable overpower trip setpoint of:

$Q + 10$, or 106.5 (whichever is smaller) for $10<Q<100$
and 20 for $Q<10$

where,

Q = percent thermal or nuclear power

6.2 Thermal Margin/Low Pressure Trip

The TM/LP trip provides protection against exceeding the SAFDL Departure from Nucleate Boiling (DNB). The SAFDL previously used for Maine Yankee, as described in Section 3 of the FSAR and in (8), is maintaining a W-3 DNB ratio of greater than 1.3. Thus, the TM/LP Trip System provides a reactor trip for a given set of conditions before a W-3 DNBR of 1.3 is calculated. A new thermal margin criteria has been proposed in (40), use of the YAEC-1 CHF correlation with a DNBR limit of greater than 1.17. Setpoints for the TM/LP trip have been determined using both criteria, the W-3 and the YAEC-1. Utilization of the setpoints calculated with the SAFDL based on YAEC-1 require approval of the reference topical report (40). The setpoints calculated for the YAEC-1 SAFDL provide greater operating margins and are provided here as the preferred set.

The Maine Yankee TM/LP Trip System monitors reactor inlet temperature, power, peripheral Symmetric Offset (SO), and Reactor Coolant System (RCS) pressure. It utilizes these parameters as inputs to an analog calculator and initiates a reactor trip when the RCS pressure, calculated by equation (1) below, is equal to the measured value of RCS pressure.

$$P^{trip} = A\,Q_{DNB} + BT_c + C \qquad (1)$$

where

A, B and C are constants

T_c = cold leg temperature

Q_{DNB} = Al x QR_1

A_1 = SO function

QR_1 = power level function

The LSSS for the TM/LP Trip System consists of the constants A, B, C and the function Al and QR1.

The constants A, B and C are determined from a reference set of DNB thermal limit lines which define the locus of points as a function of power and pressure, for all possible ranges in reactor coolant inlet temperature, at which a SAFDL on DNB would exist. The range of inlet temperatures considered is limited by the secondary safety valve setpoints and the steam generator low pressure trip.

The power distribution for which the reference set of DNB thermal limit lines is based is consistent with the power distribution that both the QR1 and A1 functions are normalized.

The QR1 and A1 functions are derived by examining the effect that CEA position and symmetric offset have on the reference set of thermal limit lines. The specific constants and functions derived for the LSSS include various measurement and calculational uncertainties. The required constants and functions for the TM/LP for Cycle 7 operation at 2630 MWt are given in Figures 6.1 and 6.2 for the YAEC-1 correlation. The setpoints derived using the YAEC-1 DNB correlation provide an additional 100 psi margin to the trip signal at the normal HFP operating point.

6.3 Symmetric Offset Trip

The Symmetric Offset Trip provides protection against the violation of SAFDL on fuel centerline melt. Based on previous studies (15, 20, 33) for the results contained herein for the enhanced fission gas release model (Section 3.2.2), a conservative value of 21 kW/ft is selected as the Linear Heat Generation Rate (LHGR) corresponding to incipient fuel centerline melt for the limiting fuel types in Cycle 7 (Types K and L). The lower LHGR limit of 20.0 kW/ft for the J fuel was shown to be non-limiting for Cycle 7 in Section 3.2.2.

The Maine Yankee Symmetric Offset Trip function monitors both core power and peripheral shape index and initiates a trip signal when the combination reaches a pre-selected setpoint. The required relationship between core power and peripheral symmetric offset corresponding to incipient centerline melting at the hot spot is defined directly from specific power distributions as follows:

$$P_L = \frac{100 \times 21.0}{F_r \, F_e \, F_A(z) \, F(Z) \, W_{max\ avg}} \qquad (2)$$

where:

F_L = core power to incipient fuel centerline melt, % rated power

F_r = pin radial peaking factor

$F_{(z)}$ = assembly axial peaking factor

F_e = engineering heat flux factor

$F_A(z)$ = power spike penalty due to the postulated formation of gaps between pellets

W_{avg} = full power core average LHGR deposited in rod augmented for axial stack shortening due to fuel densification.

Since $F_A(z)$ is a function of axial position, the maximum product of $F_r\ F_A(z)\ F(z)$ is selected for determining P_L in equation (2). The results from the above determination of P_L considering a spectrum of possible exposures, xenon conditions, and rod positions are plotted versus peripheral SO. The peripheral SO is calculated for each power distribution and is related to the excore detector response as follows:

$$S = A\ S_E + b \qquad (3)$$

where:

S = Peripheral SO

A = Shape annealing factor

S_E = External SO generated from excore detector signals

b = Bias term

Envelopes of the data - P_L versus S - are arranged for each specific CEA position analyzed based on the Power Dependent Insertion Limit (PDIL). The LSSS for the SO trip is derived from this curve by defining the allowable SO band as a function of core power. Maximum core power for each CEA position is limited by the variable overpower trip setting. The envelope defining the SO trip includes various uncertainties associated with the monitored and calculated parameters. The required SO trip function for Cycle 7 2630 MWt operation is unchanged from Cycle 6 and is given in Figure 6.3.

6.4 Limiting Conditions for Operation

Sections 6.2 and 6.3 have outlined the manner in which the Reactor Protection System functions through the Thermal Margin/Low Pressure Symmetric Offset and Variable Overpower Trips to automatically shut the reactor down to prevent the fuel from exceeding SAFDL. However, there are some AOO for which these trips would not function to provide the necessary protection, but for which operating allowances on symmetric offset in conjunction with other Reactor Protection System trips do prevent exceeding SAFDL. The limiting incidents of this class are the CEA drop and the loss-of-coolant flow.

6.4.1 CEA Drop Symmetric Offset Limits

Following a CEA Drop Incident, the RPS does not normally initiate a reactor trip, unless the moderator temperature coefficient is not sufficiently negative such that the reduction in primary average temperature required to overcome the negative reactivity of the dropped CEA produces a low pressurizer pressure trip. However, for boron conditions that exist later in core life, the absence of a turbine runback following a CEA drop will tend to restore the reactor to near full power with an adversely distorted power distribution. Therefore, it is necessary to maintain the linear heat rate and thermal-hydraulic operating limits within a band to assure that SAFDLs are not exceeded during the transient.

The required operating band is determined by finding the increase in the maximum three-dimensional power peak resulting from a CEA drop, then reducing the 21 kW/ft envelope by this amount. This results in the operating band shown in Figure 6.3. The minimum DNB ratio and void fraction is calculated with the most adverse power distribution existing at this limit and verified to be within design limits. If the thermal-hydraulic parameters are not within design limits, the operating band on symmetric offset is reduced until the specific criteria is met. The results for the CEA drop for Cycle 7 at 2630 MWt are reported in Section 5.4.2.

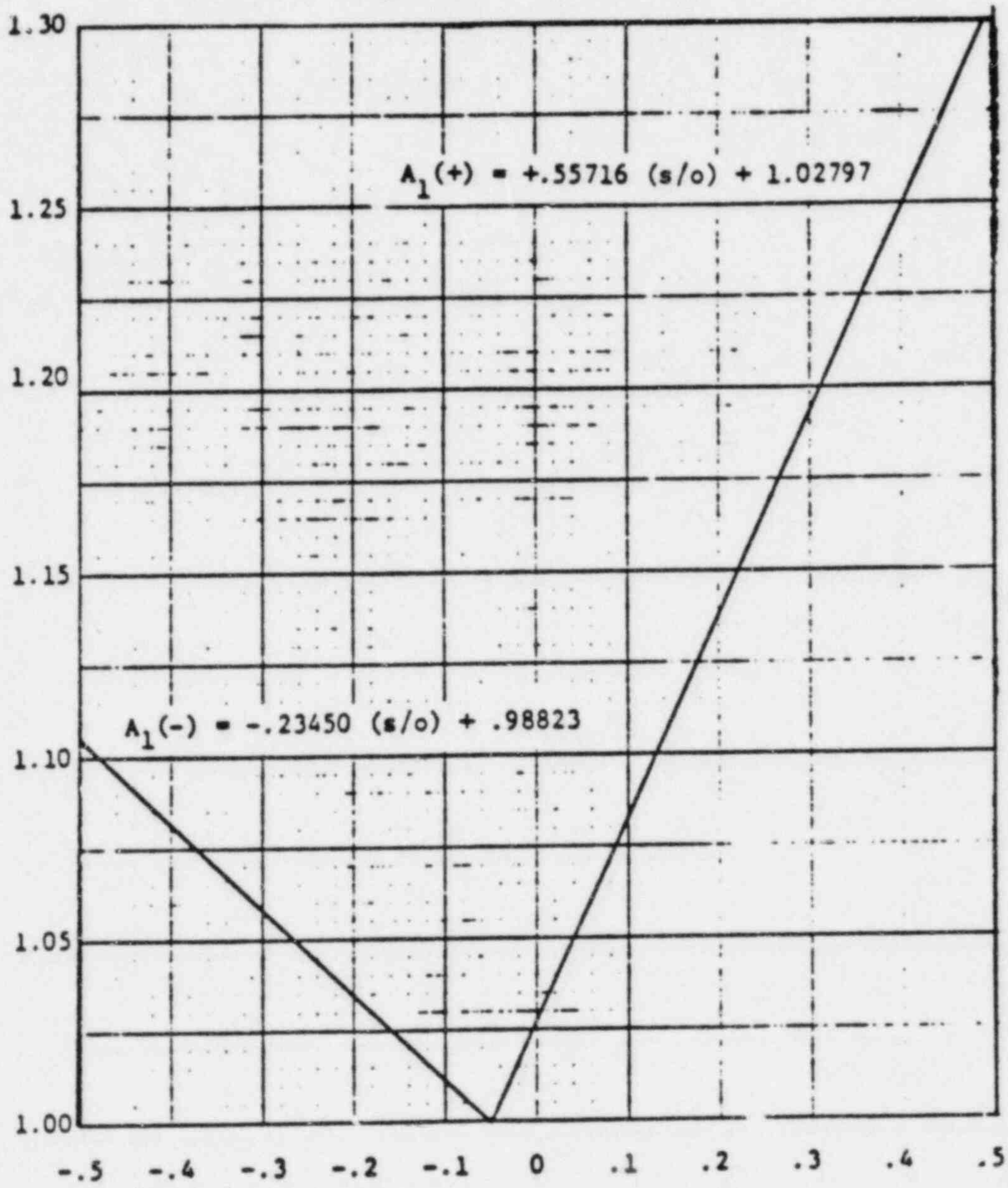
6.4.2 Loss-of-Coolant Flow Symmetric Offset Limits

During a Loss-of-Coolant Flow Incident, the reactor is tripped on low flow to prevent the fuel from exceeding design limits. However, since distributions defined by the operating band on symmetric offset will persist during the course of the incident. Therefore, to verify that fuel design limits are not exceeded during a Loss of Flow Incident, the thermal-hydraulic parameters are computed at the most adverse power distributions existing within the operating band determined for the CEA Drop Incident (linear heat rate limits need not be considered since power distributions do not change adversely during the incident). If the thermal limits are exceeded with this operating band, the operating band is reduced until results are acceptable. The results of the Loss-of-Coolant Flow Incident for Cycle 7 at 2630 MWt are reported in Section 5.4.1.

Where: $Q_{DNB} = A_1 \times QR_1$

and $P_{var}^{trip} = 1998.0\ Q_{DNB} + 17.9\ T_{in} - 10053$

T_{in} = Cold leg temperature, ^{o}F

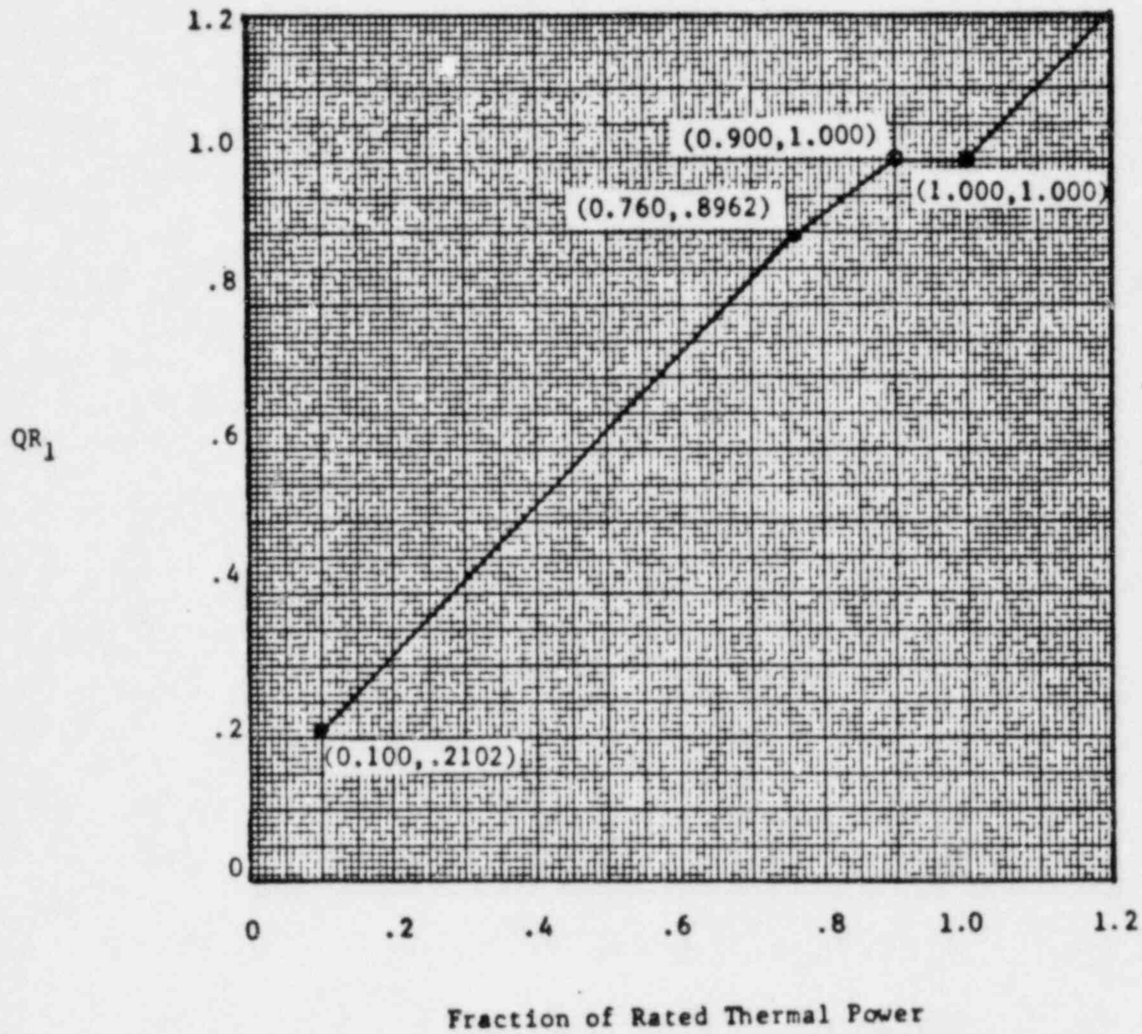


MAINE YANKEE	Thermal Margin/Low Pressure Trip Setpoint Versus (Y_I versus A_1)	Figure 6.1

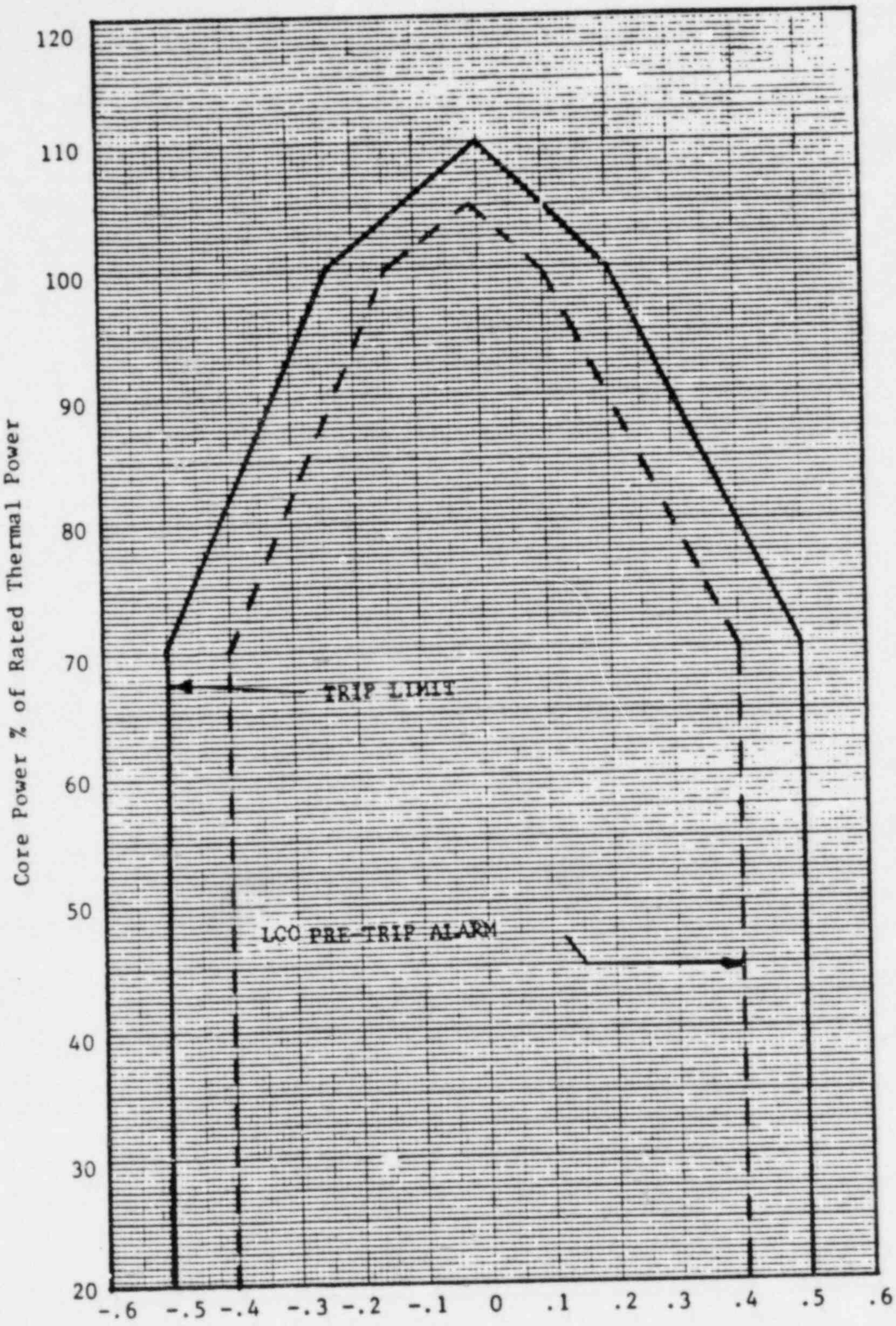
Where: $Q_{DNB} = A_1 \times QR_1$

and $P_{var}^{trip} = 1998.0\ Q_{DNB} + 17.9\ T_{in} - 10053$

T_{in} = Cold leg temperature



MAINE YANKEE	Thermal Margin/Low Pressure Trip Setpoint Part 2 (Fraction of Rated Thermal Power versus QR_1)	Figure 6.2



Symmetric Offset = $A\frac{U-L}{U+L} + B$

MAINE YANKEE	Symmetric Offset Function, Three Pump Operation	Figure 6.3

7.0 STARTUP TEST PROGRAM

The startup test program includes low power physics and power escalation tests for the purposes of:

1) Verifying that the core is correctly loaded and there are no anomalies present which could cause problems later in the cycle;

2) Verifying that the calculational model used will correctly predict core behavior during the cycle.

The low power physics tests are conducted at a power level less than 2.0% of rated full power with a primary system temperature and pressure of approximately 532°F and 2250 psia, respectively.

7.1 Low Power Physics Tests

The following reactor parameters are measured at the low power conditions:

1) Critical boron concentration is determined at unrodded and selected rodded positions.

2) The integral worth at the hot zero power condition of control rod groups 1, 2, 3, 4 and 5 in the non-overlap condition. The total of the worths of these groups must be within ±10% of the predicted value. If this condition is not met, then Banks A, B and C will be measured and the sum of the worths of all the groups must be within ±10% of predicted.

3) The most limiting near full power ejected CEA worth is measured at the pre-ejection conditions by the boron dilution technique.

4) The isothermal temperature coefficient is measured by trending moderator temperature and reactivity changes. The measurement is performed at unrodded and a rodded condition.

5) Control rod drop times are measured by monitoring reed switch voltage for position indication versus time. All full length CEA drop times are measured.

7.2 Power Escalation Tests

Power escalation tests assure the performance of various primary and secondary plant systems. Plant parameters are stabilized and test data taken at approximately 48% and approximately 100% of rated power.

The following plant parameters are evaluated at the above power levels, or as indicated:

1) Core radial power distributions at essentially unrodded conditions at the above power levels are determined using the fixed incore detector system.

2) Isothermal temperature coefficients are derived at 48% power by partially closing the steam turbine governor valves which increase reactor coolant system temperature. The result is a change in moderator temperature and power level from which the coefficient is inferred.

7.3 Acceptance Criteria

Acceptance criteria for the prediction of key core parameters are defined in Table 7.1. The permissible deviations from predicted values are selected to insure the adequacy of the transient analysis. In these tests, the nominal measured value is compared to the nominal calculated value, the latter corrected for any difference between the measurement and calculational conditions.

If the criteria in Table 7.1 are met, verification is obtained that the core characteristics conform to those assumed in the safety analysis. In particular, compliance with Specification 3.10.B.1 is demonstrated by the CEA worth and drop time measurements, provided all trippable CEAs remain operable.

If the initial criteria in Table 7.1 are not met, additional measurements, as prescribed by the table, are performed. In addition, any deviations are evaluated relative to the assumptions in the safety analysis for the given core parameters. Such deviations and their evaluations are reported to the Staff. A startup test summary report will be available on-site within 90 days of the completion of the startup tests.

Table 7.1

Maine Yankee Cycle 7
Startup Test Acceptance Criteria

	Measurement	Conditions	Criteria
1.	Critical Boron Concentration	Hot zero power, near all-rods-out	Measurement within $\pm 1\% \Delta\rho$ of predicted value
2.	CEA Bank Worths	Hot zero power, CEA Banks 1+2+3+4+5 in the non-overlap condition	Total worth within $\pm 10\%$ of the predicted value
3.	CEA Bank Worths	Hot zero power, CEA Banks A+B+C+1+2+3+4+5 in the non-overlap condition	If the criteria in Measurement (2) is not met, the total worth of all CEA banks must be within $\pm 10\%$ of the predicted value.
4.	Ejected CEA Worth	Hot zero power, pre-ejection CEA banks inserted for measurement of the most limiting near full power ejected CEA	Ejected CEA worth no more than 15% greater than the predicted value
5.	Isothermal Temperature Coefficient	Hot zero power, near all-rods-out	Measurement within $\pm 0.5 \times 10^{-4} \Delta\rho$ /°F of predicted value
6.	Control Rod Drop	Operating temperature, insertion to 90%	Drop times no greater than 2.70 seconds
7.	Radial Power Distribution	At or slightly below 50% power, near all-rods-out	Each assembly average power within $\pm 10\%$ of predicted value
8.	INCA Tilt Monitoring for Symmetry Verification	5-48% rated power, near all-rods-out, tilt is monitored at 5% power intervals	Tilt trends to less than 3.0% for greater than 50% power operation

8.0 CONCLUSIONS

The results of analyses presented herein have demonstrated that design criteria as specified in the FSAR and the NRC ECCS Acceptance Criteria (31) will be met for operation of Maine Yankee during Cycle 7. Table 5.3 summarizes the results of each incident analyzed; including the Reference Cycle result and the appropriate design limit. This table illustrates that Specified Acceptable Fuel Design Limits (SAFDL) on DNB and fuel centerline melt, the primary coolant system ASME code pressure limit, and the 10CFR100 site boundary dose limits are not violated for any of the incidents considered.

The maximum computed peak clad temperature following a LOCA for operation within the limits specified in Section 5.5.5.1 is 2149^{o}F and is below the 2200^{o}F limit given in 10CFR50.46. Maximum calculated cladding oxidation and hydrogen generation are 5.52% and less than 1%, respectively.

Based on these results, it is concluded that operation of the Maine Yankee Atomic Power Station can continue through Cycle 7 within the limits of the proposed Technical Specifications without endangering the health and safety of the public.

9.0 REFERENCES

1. Maine Yankee Letter to USNRC, WMY 77-75, dated August 1, 1977.

2. Maine Yankee Atomic Power Station Final Safety Analysis Report (FSAR).

3. P. A. Bergeron, P. J. Guimond, J. DiStefano, "Justification for 2630 MWt Operation of the Maine Yankee Atomic Power Station", YAEC-1132, dated July 1977.

4. USNRC Letter to Yankee Atomic dated January 17, 1978. USNRC Letter to Yankee Atomic dated April 11, 1978.

5. ACRS Letter to J. M. Hendrie, Chairman USNRC from S. Lawroski, Chairman ACRS, dated June 7, 1978.

6. P. J. Guimond, P. A. Bergeron, "Justification for Operation of the Maine Yankee Atomic Power Station with a Positive Moderator Temperature Coefficient", YAEC-1148, dated April 1978.

7. Maine Yankee Letter to USNRC, WMY 78-62, dated June 26, 1978, "Maine Yankee Proposed Change No. 64".

8. P. A. Bergeron, D. J. Denver, "Maine Yankee Reactor Protection System Setpoint Methodology", YAEC-1110, dated September 1976.

9. R. N. Gupta, "Maine Yankee Core Thermal-Hydraulic Model Using COBRA IIIC", YAEC-1102, dated June 1976.

10. R. N. Gupta, "Maine Yankee Core Analysis Model Using CHIC-KIN", YAEC 1103, dated September 1976.

11. T. R. Hencey, "GEMINI-II - A Modified Version of the GEMINI Computer Program", YAEC-1068, dated April 1974.

12. P. A. Bergeron, "Maine Yankee Plant Analysis Model Using GEMINI-II", YAEC-1101, dated June 1976.

13. W. J. Szymczak, "Maine Yankee Plant Accident Analysis Model Using FLASH-4", YAEC-1104, dated November 1976.

14. D. J. Denver, E. E. Pilat, R. J. Cacciapouti, "Application of Yankee's Reactor Physics Methods to Maine Yankee", YAEC-1115, dated October 1976.

15. P. A. Bergeron, "Maine Yankee Fuel Thermal Performance Evaluation Model", YAEC-1099P, dated February 1976 (Proprietary).

16. YAEC-1160, "Application of YANKEE WREM-BASED Generic PWR ECCS Evaluation Model to Maine Yankee", July 1978.

17. USNRC Letter, R. W. Reid to R. H. Groce, dated May 27, 1977.

18. USNRC Letter to Yankee Atomic dated January 17, 1979.

19. Letter, R. H. Groce (YAEC) to B. H. Grier (NRC), "Transmittal of MY Startup Test Report, Cycle 5", WMY 80-93, (June 4, 1980).

20. Maine Yankee Letter to USNRC, WMY 76-129, dated November 18, 1976, "Maine Yankee Proposed Change No. 52".

21. XN-73-25 GAPEX: A Computer Program for Predicting Pellet-to-Cladding Heat Transfer Coefficients, August 13, 1973.

22. D. S. Rowe, "COBRA IIIC: A Digital Computer Program for Steady-State and Transient Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements", BNWL-1695 (March 1973).

23. USNRC Letter to Yankee Atomic, dated January 7, 1977.

24. E. S. Markowski, L. Lee, R. Biderman, J. E. Casterlin, "Effect of Rod Bowing on CHF in PWR Fuel Assemblies", ASME paper 77-HT-91.

25. CEN-93(M)-P, "Maine Yankee Reactor Operation with Modified CEA Guide Tubes", June 21, 1978.

26. Maine Yankee Letter to USNRC, dated October 12, 1979.

27. Maine Yankee Letter to USNRC, WMY 78-102, dated November 15, 1978, "Maine Yankee Startup Test Report".

28. XN-207, "Power Spike Model for Pressurized Water Reactor Fuel", March 1974.

29. Maine Yankee Letter to USNRC, WMY 78-2, dated January 5, 1978.

30. Maine Yankee Letter to USNRC, WMY 75-28, dated March 27, 1975, "Maine Yankee Core 2 Performance Analysis".

31. "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors", Federal Register, Vol. 39, No. 3-Friday, January 4, 1974.

32. Maine Yankee Letter to USNRC, WMY 77-87, dated September 22, 1977.

33. Maine Yankee Letter to USNRC, WMY 79-143, dated December 5, 1979; Attachment A, YAEC-1202, "Maine Yankee Cycle 5 Core Performance Analysis", P. Bergeron, et al.

34. XN-75-32 (NP) Supplement 2 - "Computation Procedures for Evaluating Fuel Rod Bowing", July 1979.

35. R. E. Helfrich, "Thermal-Hydraulic Analysis of PWR Fuel-Element Transients Using the CHIC-KIN Code", YAEC-1241, March 1981.

36. Combustion Engineering Report, TR-DT-34, "The Hydraulic Performance of the Maine Yankee Reactor Model", June 1971.

37. YAEC-1259, "Maine Yankee Cycle 6 Core Performance Analysis", Attachment to MYAPC Letter to USNRC, FMY-81-65, Proposed Change No. 84, dated April 28, 1981.

38. Cycle 6 MSLB Analysis, Attachment to MYAPC Letter to USNRC, FMY 81-162, dated October 29, 1981.

39. C. L. Wheeler, et al., "COBRA-IV-I: An Interim Version of COBRA for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores", BNWL-1962 (March 1976).

40. J. Handschuh, "DNBR Limit Methodology and Application to the Maine Yankee Plant," YAEC-1296P, January 1982, Attachment to YAEC Letter to USNRC, FYR-82-41, MN-82-78, dated April 8, 1982.

41. R. VanHouten, "Fuel Rod Failure as a Consequence of Departure from Nucleate Boiling or Dryout", NUREG-0562, Office of Nuclear Regulatory Research, USNRC, Washington, D. C. 20555 (June 1979).

42. MYAPC Letter to USNRC, MN-82-165, "Maine Yankee Reportable Occurrence 82-23/01T-0", dated August 19, 1982.

43. MYAPC Letter to USNRC, MN-82-53, "Boron Dilution During Hot and Cold Shutdown (Mode 5 Operation)," dated March 18, 1982.

44. CENPD-190, "CE Method for Control Element Assembly Ejection Analysis", Combustion Engineering, January 1976.

45. S. Bian, "Application of Reactivity Weighting to Rod Ejection Accident Analysis in a Pressurized Water Reactor", Nucl. Tech., Vol. 41, 401-407 (1978).

46. MYAPC Letter to USNRC, MN-82-157, "Maine Yankee Reportable Occurrence 82-21/01T-0", dated August 11, 1982.

47. USNRC letter to MYAPC, dated December 11, 1982.
"Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 60 to License No. DPR-36, Maine Yankee Atomic Power Company, Maine Yankee Atomic Power Station, Docket No. 50-309", attached to USNRC Letter to MYAPC, dated December 11, 1981.

48. XN-NF-81-85, "Mechanical Design Report Supplement for Exxon Nuclear Maine Yankee XN-1 through XN-4 Extended Burnup Program", November 1981.

49. J. B. Yasinsky, "On the Use of Point Kinetics for the Analysis of Rod-Ejection Accidents", Nuclear Science and Engineering: 39, 241-256 (1970).