Commonwealth Edison Company
Dresden Generating Station
6500 North Dresden Road
Morris, IL 60450
Tel 815-942-2920

ComEd

July 24, 1998

JMHLTR: #98-0213

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Dresden Nuclear Power Station Unit 3
Mid-Cycle Core Operating Limits Report
NRC Docket No. 50-249

References:
a) Siegel (USNRC) to T. Kovach (ComEd) dated February 8, 1990. Approving the Technical Specification Amendment and Core Operating Limits Report per Generic Letter 88-16.

b) J.M. Heffley to USNRC dated May 2, 1998 (Transmittal to the NRC of Dresden Unit 2, Cycle16 COLR)

The purpose of this letter is to transmit the revised Core Operating Limits Report (COLR) and revised Reload Licensing Analysis for Unit 3 pursuant to the requirements of Technical Specification 6.9.A.6.c. and Reference (a). Specifically, ComEd has revised the plant transient analysis to allow use of a bounding Main Steam Flow and Feedwater Flow. The changes to the transient analysis, which are reflected in the attached Reload Licensing Analysis and the COLR, have been reviewed by ComEd under the provisions of 10CFR50.59. These reviews have resulted in no changes to the SL-MCPR and no Unreviewed Safety Questions.

A001

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PDR ADOCK 05000249
P PDR

The use of the bounding steam and feedwater flow, on Unit 2 transient and reload licensing analysis were reflected in the Reference (b) submittal. Please address any questions concerning this letter to Frank Spangenberg, Regulatory Assurance Manager, extension 3800.

Sincerely,

J.M. Heffley
Site Vice President
Dresden Station

Attachment: A. Core Operating Limits Report for Dresden Station Unit 3, Cycle 15 (Revised)
B. Reload Licensing Analysis

cc. Regional Administrator, Region lll
L.W. Rossbach, Dresden Project Manager, NRR
K. Reimer, Senior Resident Inspector, Dresden
Office of Nuclear Facility Safety-IDNS

50-249 CE DRESDEN 3

MID-CYCLE CORE OPERATING LIMITS REPORT

REC'D W?LTR DTD 07/24/98....9807290058

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SECTION A

Core Operating Limits Report

DRESDEN STATION UNIT 3

CYCLE 15

June 1998

ISSUANCE OF CHANGES SUMMARY

Affected Section	Affected Pages	Summary of Changes	Date
All	1-1 through 5-6	Incorporated Reference to TSUP Section Number/Deleted References to Custom TS.	06/97
References	iii	Identified Analyses of Record for D3C15.	06/97
2.2 and Figure 2.2-1 and Table 2.2-1	2-1, 2-2 and 2-3	Included MAPLHGR limits for D3C15 9x9-2 and ATRIUM-9B reload fuel.	06/97
3.2 and Figure 3.2-1	3-1 and 3-2	Included SLHGR limits for D3C15 ATRIUM-9B reload fuel.	06/97
4.2 and Figure 4.2-1	4-1 and 4-2	Included TLHGR limits for D3C15 ATRIUM-9B reload fuel.	06/97
5.2 and Table 5.2-1	5-1 and 5-2	Simplified from Figure 5.2-1A, since OLMCPRs are not scram time dependent.	06/97
5.2 and Table 5.2-1, Figure 5.2-1 and 5.2-2	5-2, 5-4 and 5-5	Revised to reflect new Operating Limit MCPRs for 9x9-2 and ATRIUM-9B reload fuel, Deleted previous Figure 5.2-1, because Operating Limits MCPR's were performed using only the Technical Specification scram times and, thus, are not scram time dependent	06/97
Table 5.2-2	5-3	Added a table of the OLMCPR adders for turbine bypass valve opening time degradation	06/97
Figure 2.2-1, Figure 3.2-1, and Figure 4.2-1	2-2, 3-2, and 4-2	The table of information in Figure 2.2-1 was split into two tables, Figures 3.2-1 and 4.2-1 were changed to say N/A if an limit did not exist at that exposure.	06/97
References	iv	The SPC letter documenting the 0.01 adder for the reduced dome pressure operation was added	9/97
Table 2.3-1	2-3	Corrected action step to be consistent with TSUP by changing 3.6.A Action d to 3.6.A.1d. Done at Dresden.	9/97

Table 5.2-1	5-2	Increased the Operating Limit Minimum Critical Power Ratio by 0.02 due to operation at up to 15 psi below the analyzed pressure (0.01) and a conservative (0.01) for future potential additional MCPR penalties (i.e. for operation at a higher steam flow rate).	9/97
Table 5.2-1	5-2	Added statement that no NSS limits are presented for this cycle. Only TSSS limits are presented in the COLR	9/97
Issuance of Changes Summary	i, ii	The table of changes increased by one page, therefore affecting the page numbering of the following pages, and the Table of Contents	9/97
References	iv	The D3C15 Reload Licensing documents (Reference numbers 4 and 5) were updated for their most recent revisions due to the incorporation of an increased steam flow rate in the analyses.	6/98
Table 5.2-1	5-2	Added a footnote explaining that the 9/97 additional 0.01 MCPR penalty is not necessary to support the operation in increased steam flow.	6/98
Section B	Attachments 2 & 3	Attached the latest revision of the D3C15 SPC Reload Analysis and excerpts from the SPC Transient Analysis	6/98

SECTION A
TABLE OF CONTENTS

REFERENCES

1. Commonwealth Edison Company Docket No. 50-249, Dresden Nuclear Power Station, Unit 3, Facility Operating License DPR-25.

2. Letter, D.M. Crutchfield to All Power Reactor Licensees and Applicants, Generic Letter 88-16, Concerning the Removal of Cycle-Specific Parameter Limits from Technical Specifications.

3. EMF-97-031(P), Dresden LOCA-ECCS Analysis MAPLHGR Limits For ATRIUM-9B And 9x9-2 Fuel, Siemens Power Corporation, May 1997, NFS NDIT # 970081.

4. EMF-96-141, Revision 1, Dresden Unit 3 Cycle 15 Reload Analysis, Siemens Power Corporation, June 1998, NFS NDIT# NFS-97-0085, Sequence 1.

5. EMF-97-047, Dresden Unit 3 Cycle 15 Plant Transient Analysis with Increased Steam Flow, Siemens Power Corporation, June 1998, NFS NDIT # NFS-97-0084, Sequence 1.

6. Dresden Unit 3 Cycle 15 Neutronic Licensing Report, NFS NDIT # 970028.

7. EMF-92-149(P) And EMF-92-149(P) Supplement 1, Revision 1, Dresden Units 2 And 3 Generic Coastdown Analysis With ATRIUM-9B, Siemens Power Corporation, September 1996, NFS NDIT # 960137.

8. SPC letter, Dresden Unit 3 Cycle 15 MAPLHGR Limits Versus Assembly Average Exposure, DEG:97:048, D.E. Garber to R.J. Chin, June 5, 1997, NFS NDIT # 970111.

9. SPC letter, Dresden Reduced Dome Pressure Analyses, DEG:97:102, D.E. Garber to R.J. Chin, August 6, 1997, NFS NDIT #970117 Rev. 1.

LIST OF FIGURES

LIST OF TABLES

1.0 ROD BLOCK MONITOR (RBM)

1.1 Technical Specification Reference

Technical Specification 3.3.M. - Rod Block Monitor (RBM)

1.2 Description

The Rod Block Monitor Upscale Instrumentation Setpoints are determined from the relationships shown in Table 1.2-1.

TABLE 1.2-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION:	TRIP LEVEL SETTING:
Rod Block Monitor Upscale (Flow Bias)	
Dual Loop Operation	Less than or equal to (0.65 W_d plus 55)*
Single Loop Operation	Less than or equal to (0.65 W_d plus 51)*

*W_d - percent of drive flow required to produce a rated core flow of 98 Mlb/hr.

2.0 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

2.1 Technical Specification Reference

Technical Specificiation 3.11.A - AVERAGE PLANAR LINEAR HEAT GENERATION RATE

2.2 Description

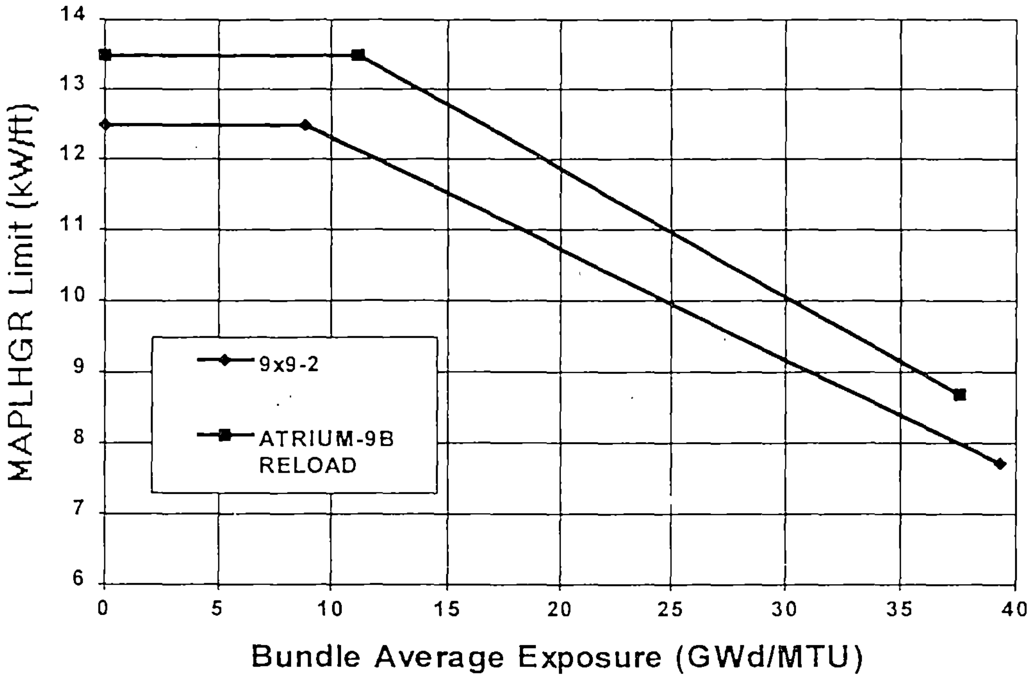
The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Limit versus Bundle Average Exposure for each fuel type is determined from Figure 2.2-1.

2.3 MAPLHGR Limit Equipment Out of Service Multipliers

The appropriate multiplicative factors, during power operation with equipment out of service, to apply to the base MAPLHGR limits specified in Section 2.2 are shown in Table 2.3-1.

FIGURE 2.2-1

MAPLHGR LIMIT VS. BUNDLE AVERAGE EXPOSURE



Bundle Average Exposure (GWD/MTU)	MAPLHGR Limit 9x9-2 (kW/ft)
0	12.5
8.8	12.5
39.3	7.7

Bundle Average Exposure (GWD/MTU)	MAPLHGR Limit ATRIUM-9B Reload Fuel (kW/ft)
0	13.5
11.1	13.5
37.5	8.7

TABLE 2.3-1

EQUIPMENT OUT OF SERVICE MAPLHGR LIMIT MULTIPLIERS

Technical Specification	Title of Technical Specification	Scenario	Multiplicative Factor, 9x9-2 and ATRIUM-9B
3.11.A & 3.6.A.1.d	Average Planar Linear Heat Generation Rate and Recirculation Loops	Single Loop Operation (SLO)	0.90

3.0 STEADY STATE LINEAR HEAT GENERATION RATE

3.1 Technical Specification Reference

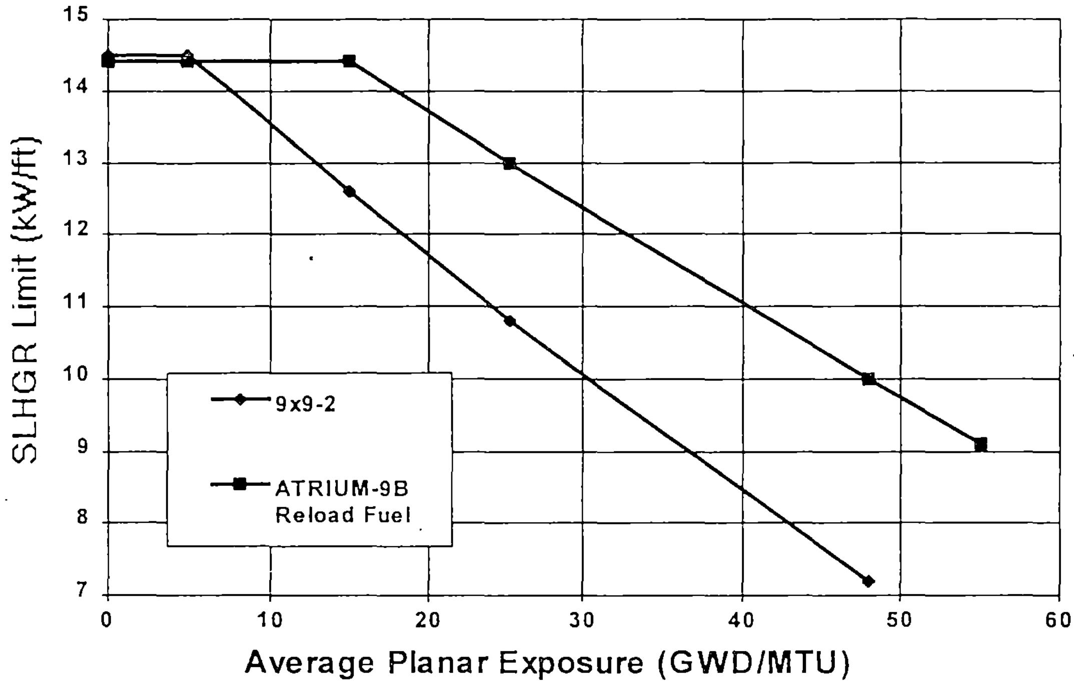
Technical Specification 3.11.D - STEADY STATE LINEAR HEAT GENERATION RATE

3.2 Description

The Steady State LHGR (SLHGR) limit versus Average Planar Exposure for each fuel type is determined from Figure 3.2-1.

FIGURE 3.2-1

STEADY STATE LHGR (SLHGR) LIMIT VS. AVERAGE PLANAR EXPOSURE



Average Planar Exposure (GWD/MTU)	SLHGR Limit 9x9-2 (kW/ft)	SLHGR Limit ATRIUM-9B Reload Fuel (kW/ft)
0	14.5	14.4
5.0	14.5	14.4
15.0	12.6	14.4
25.2	10.8	13.0
48.0	7.2	10.0
55.0	N/A	9.1

4.0 TRANSIENT LINEAR HEAT GENERATION RATE

4.1 Technical Specification Reference

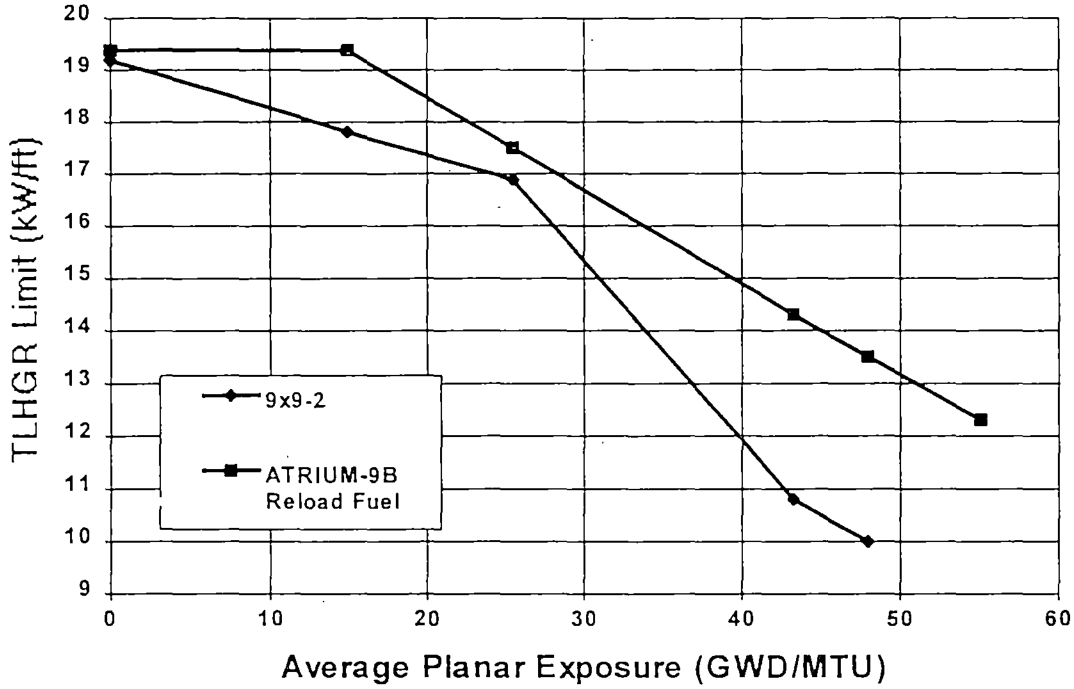
Technical Specification 3.11.B - TRANSIENT LINEAR HEAT GENERATION RATE

4.2 Description

The Transient LHGR (TLHGR) limit versus Average Planar Exposure for each fuel type is determined from Figure 4.2-1.

FIGURE 4.2-1

TRANSIENT LHGR (TLHGR) LIMIT VS. AVERAGE PLANAR EXPOSURE



Average Planar Exposure (GWD/MTU)	TLHGR Limit 9x9-2 (kW/ft)	TLHGR Limit ATRIUM-9B Reload Fuel (kW/ft)
0.0	19.2	19.4
15.0	17.8	19.4
25.4	16.9	17.5
43.2	10.8	14.3
48.0	10.0	13.5
55.0	N/A	12.3

5.0 MINIMUM CRITICAL POWER RATIO

5.1 Technical Specification Reference

Technical Specification 3.11.C - MINIMUM CRITICAL POWER RATIO

5.2 Description

a. The Operating Limit MCPRs for D3C15 are listed in Table 5.2-1. The OLMCPRs calculated for D3C15 are based on Technical Specification Scram Insertion Speeds (3.3.E).

b. For operation with a degraded turbine bypass valve opening time, the OLMCPR adder in Table 5.2-2 must be added to the Operating Limit MCPR determined from Table 5.2-1. Linear interpolation between the data points is permissible.

c. During Manual Flow Control, the Operating Limit MCPR for each fuel type at reduced core flow conditions can be determined from whichever is greater:

   i. Figure 5.2-1 using the curve and the appropriate flow rate.

   ii. The Operating Limit MCPR determined from Table 5.2-1, and supplemented by Table 5.2-2 when appropriate.

d. During Automatic Flow Control, the Operating Limit MCPR for each fuel type at reduced flow rates can be determined from Figure 5.2-2 using the appropriate flow rate and the Operating Limit MCPR, which is obtained from Table 5.2-1, and supplemented by Table 5.2-2 when appropriate. Linear interpolation between the curves on Figure 5.2-2 is permissible.

TABLE 5.2-1

OPERATING LIMIT MCPR
FOR 9x9-2 AND ATRIUM-9B RELOAD FUEL

Operating Scenario	Operating Limit MCPR[1,4]
Normal Operation	1.48
Normal Operation with Feedwater Heaters Out of Service	1.48
Single Loop Operation	1.49
Coastdown[2]	1.52
Coastdown and SLO Operation[3]	1.53

[1]Note that the Operating Limit MCPR is not a function of the average CRD scram insertion time for the current operating cycle other than assuming the Technical Specification average CRD scram insertion time limits (3.3.E) are met. For simplification of implementing the limits for D3C15, only limits corresponding to the Technical Specification Scram Speeds have been specified. The MCPR Operating Limits presented are based on Technical Specification Scram Speeds and bound the Nominal Scram Speed Operating Limits.

[2]The 0.04 MCPR penalty during Coastdown includes the effects of Feedwater Heater(s) Out of Service and Single Loop Operation.

[3]For coastdown and SLO, the 0.01 adder to the MCPR Safety Limit is still necessary.

[4]The MCPR Operating Limits contain a 0.01 adder to the reload licensing results due to operation at slightly less than design pressure. Originally (the September 1997 version of the D3C15 COLR), the MCPR Operating Limit had an additional 0.01 conservatism for future potential operation with increased steam flow (for a total 0.02 adder to the reload licensing results). However, when the operation at increased steam flow was analyzed, the extra 0.01 in the MCPR Operating Limit was not necessary and the MCPR Operating Limits were not changed. Therefore, the above MCPR Operating Limits contain a 0.01 adder for operation at slightly less than design pressure and a 0.01 generic conservatism, that can be used for future operational flexibility.

TABLE 5.2-2

TURBINE BYPASS VALVE DEGRADATION OLMCPR ADDERS

Equivalent Bypass Valve Delay Time (msec)*	OLMCPR Adder for ATRIUM-9B and 9x9-2 Fuel
50	0.00
150	0.02
250	0.03
350	0.04
450	0.04
550	0.05
700	0.05
900	0.05
No Bypass	0.05

*Delay is relative to the time of TSV full closure.

Linear interpolation can be used for purposes of selecting a conservative OLMCPR adders for equivalent delay times not specifically listed in the table.

Turbine Bypass Valve Degradation OLMCPR Adders

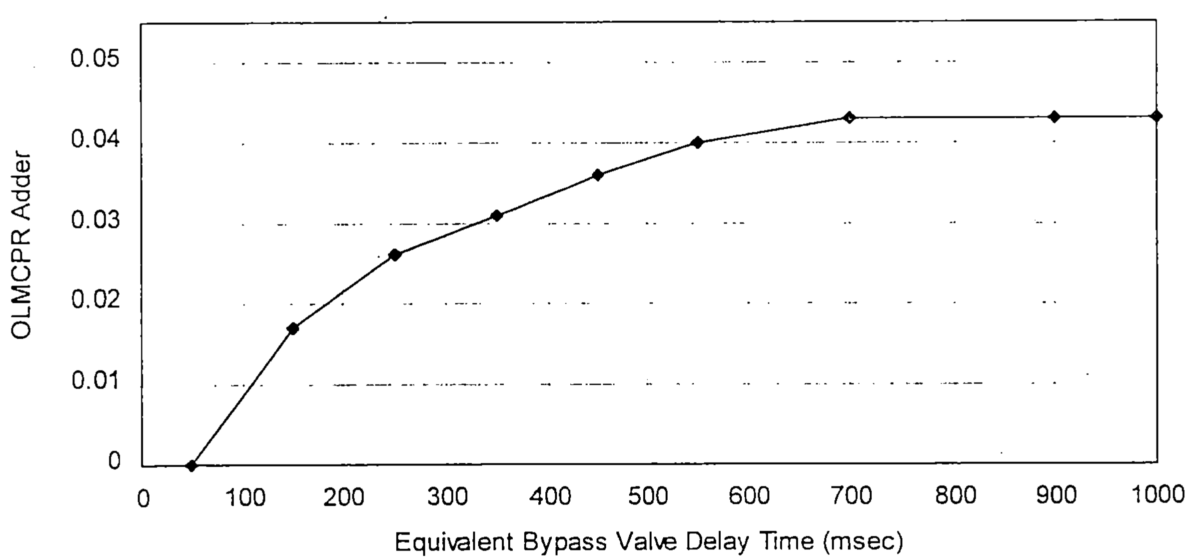
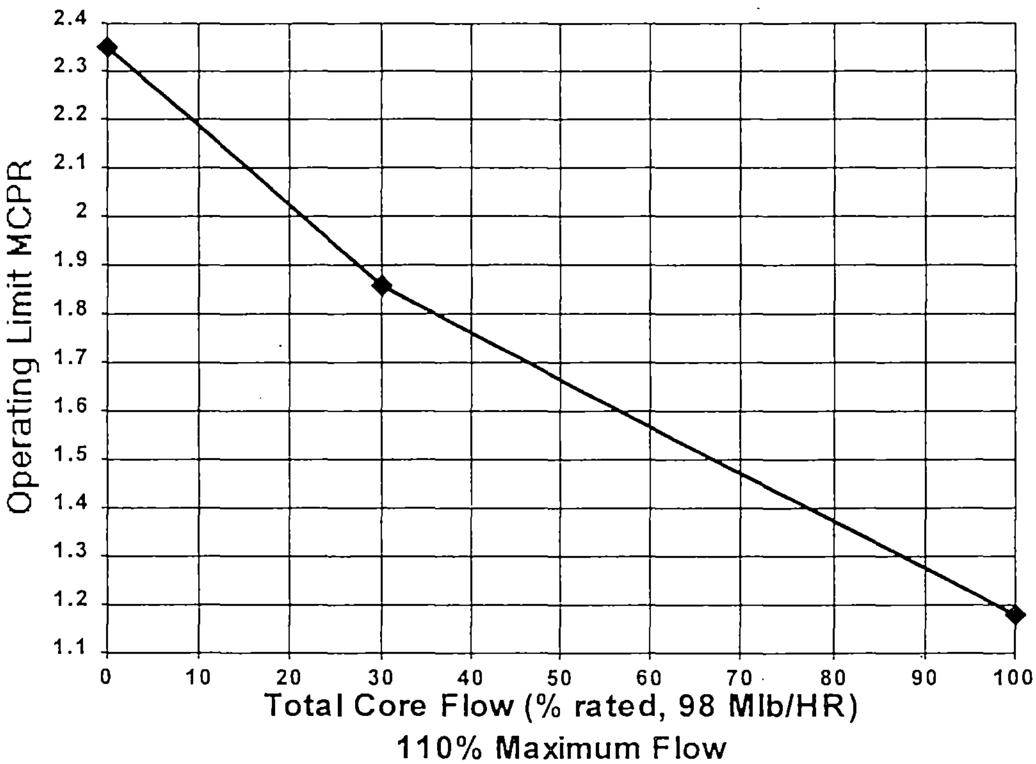


FIGURE 5.2-1

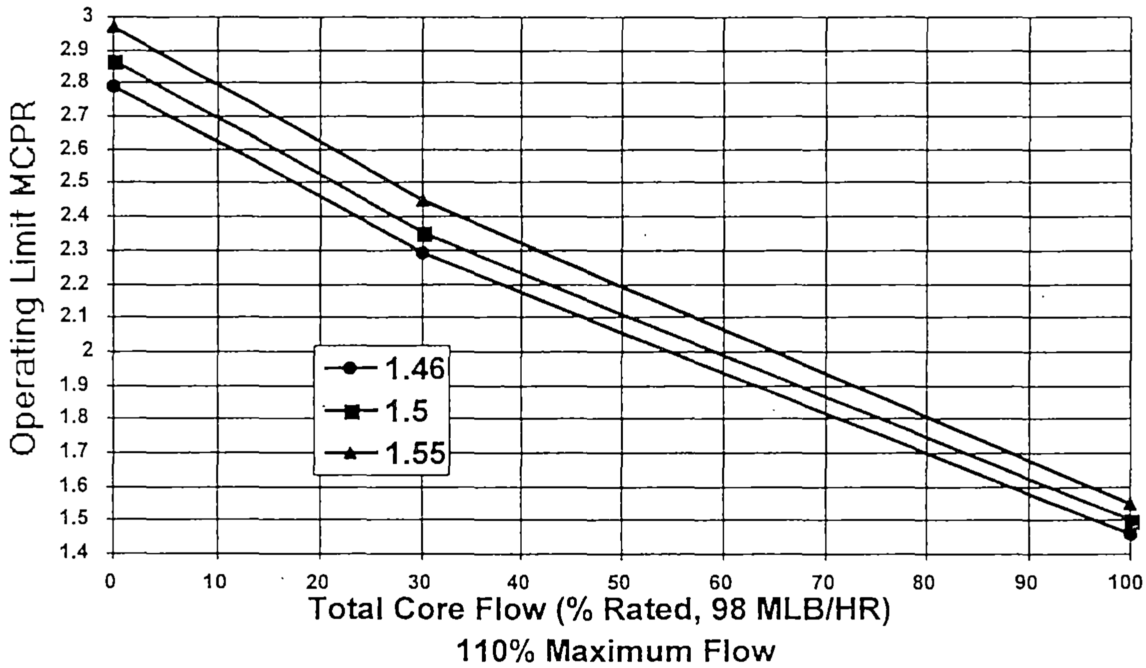
OPERATING LIMIT MCPR FOR MANUAL FLOW CONTROL



Total Core Flow (% Rated)	Operating Limit MCPR for 9x9-2 and ATRIUM-9B Reload Fuel
100	1.18
30	1.86
0	2.35

FIGURE 5.2-2

OPERATING LIMIT MCPR FOR AUTOMATIC FLOW CONTROL
FOR ATRIUM-9B and 9x9-2 FUEL[1]



Total Core Flow (% Rated)	Operating Limit MCPR 9x9-2 and ATRIUM-9B Reload Fuel 1.46	1.50	1.55
100	1.46	1.50	1.55
30	2.29	2.35	2.45
00	2.79	2.87	2.97

[1] Although analyzed for core flows from 0% to 100%, Technical Specification 3.3.N prohibits AFC operation below 65% core flow.

6.0 METHODOLOGIES

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in the latest approved revision or supplement of the topical reports describing the methodology. For Dresden Unit 3, the NRC approved topical reports are:

1) ANF-1125(P)(A) and Supplements 1 and 2, "Critical Power Correlation - ANFB", April 1990.

2) ANF-524(P)(A), "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence", XN-NF-524(P)(A) Revision 2, Supplement 1 Revision 2, Supplement 2, November 1990.

3) XN-NF-79-71(P)(A), "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," Revision 2 Supplements 1, 2, and 3, March 1986.

4) XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis", Volume 1 and Supplements 1 and 2, March 1983.

5) XN-NF-80-19(P)(A), "Advanced Nuclear Fuels Methodology for Boiling Water Reactors," Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, November 1990.

6) XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," Volumes 2, 2A, 2B, 2C, September 1982.

7) XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description", Volume 3 Revision 2, January 1987.

8) XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads", Volume 4, Revision 1, June 1986.

9) XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump Boiling Water Reactors Reload Fuel," Revision 1, September 1986.

10) ANF-913(P)(A), "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, August 1990.

11) XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1 Extended Burnup Qualification of ENC 9x9 BWR Fuel," May 1988.

12) ANF-89-014(P)(A), "Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel", October 1991.

13) ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs," Revision 1 and Revision 1 Supplement 1, May 1995.

14) ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," January 1993.

15) Commonwealth Edison Company Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," and associated Supplements on Neutronic Licensing Analyses (Supplement 1) and LaSalle County Unit 2 Benchmarking (Supplement 2).

ATTACHMENT B

RELOAD LICENSING ANALYSIS

Section B

Dresden Unit 3 Cycle 15

Reload Transient Analysis Results

June 1998

Section B
Table of Contents

Attachment 1

Dresden Unit 3 Cycle 15

Neutronics Licensing Report

NUCLEAR FUEL SERVICES DEPARTMENT
NUCLEAR DESIGN INFORMATION TRANSMITTAL

☒ SAFETY RELATED ☐ NON-SAFETY RELATED ☐ REGULATORY RELATED	Originating Organization ☒ Nuclear Fuel Services ☐ Other (specify) ______	NDIT No. 970028 Rev. No. 0 Page 1 of 19

Station Dresden Unit 3 Cycle 15 Generic ______

To: Russell D. Freeman

Subject Dresden Unit 3 Cycle 15 Neutronic Licensing Report (NLR)

David A. Pheglev Preparer	[signature] Preparer's Signature	5/9/97 Date
Theodore P. Shannon Reviewer	[signature] Reviewer's Signature	5/13/97 Date
Adelmo S. Pallotta NFS Supervisor	[signature] NFS Supervisor's Signature	5/13/57 Date

Status of Information: ☒ Verified
☐ Unverified
☐ Engineering Judgement

Method and Schedule of Verification for Unverified NDITs: ______

Description of Information: Results and basis of neutronic licensing caclulations for Dresden Unit 3 Cycle 15

Purpose of Information: Provide Station and NFS BSS group with neutronic licensing results

Source of Information Calculation Notes NFS:BNDD:97-015 and NFS:BNDD:97-038

Supplemental Distribution: J.W. Keffer (Dresden), D.A. Worthington, Dresden Central File, D3C15 Letterbook
DG Central File

~~9806250113~~ 19pp

Licensing Basis

This document, in conjunction with References 1 and 2, provides the licensing basis for Dresden Unit 3 Reload 14, Cycle 15. The calculations that support this report are given in References 3 through 10.

Table of Contents

DHP 3/20/97
JRF 3/20/97

I. Nuclear Design Analysis

I.1 Fuel Bundle Nuclear Design Analysis

Assembly Average Enrichment (w/o U-235)

SPC ATRIUM-9B 3.26 9Gd3.5/11Gd5.5/9Gd4.5	(DRC-8L)	3.26
SPC ATRIUM-9B 3.26 9Gd3.5/11Gd5.5	(DRC-8H)	3.26
SPC ATRIUM-9B 3.39 6Gd3.5/6Gd4.5/6Gd5.5	(DRC-8A)	3.39

Radial Enrichment and Burnable Poison Distribution

SPC ATRIUM-9B 3.43 9Gd3.5 (DRC-8L and 8H)	Figure 1
SPC ATRIUM-9B 3.44 11Gd5.5 (DRC-8L and 8H)	Figure 2
SPC ATRIUM-9B 3.78 9Gd4.5 (DRC-8L)	Figure 3
SPC ATRIUM-9B 3.78 11Gd5.5 (DRC-8H)	Figure 4
SPC ATRIUM-9B 3.62 6Gd3.5 (DRC-8A)	Figure 5
SPC ATRIUM-9B 3.62 6Gd4.5 (DRC-8A)	Figure 6
SPC ATRIUM-9B 3.88 6Gd5.5 (DRC-8A)	Figure 7

Axial Enrichment and Burnable Poison Distribution

SPC ATRIUM-9B 3.26 9Gd3.5/11Gd5.5/9Gd4.5	Figure 8
SPC ATRIUM-9B 3.26 9Gd3.5/11Gd5.5	Figure 8
SPC ATRIUM-9B 3.39 6Gd3.5/6Gd4.5/6Gd5.5	Figure 8

DAP 3/20/97

I.2 Core Nuclear Design Analysis

I.2.1 Core Configuration and Licensing Exposure Limits

Bundle Type	Cycle Loaded	Number in Core
SPC 9x9-2 3.13 8Gd2.0/8Gd3.0	12	84
SPC 9x9-2 3.13 8Gd2.0/8Gd4.0	12	12
SPC 9x9-2 2.95 8Gd3.0	13	48
SPC 9x9-2 2.95 9Gd3.5	13	116
SPC 9x9-2 2.95 7Gd3.0	13	52
SPC 9x9-2B 3.13 7Gd3.5	14	72
SPC 9x9-2B 3.13 8Gd3.5/9Gd4.5	14	108
SPC ATRIUM-9B 3.26 9Gd3.5/11Gd5.5/9Gd4.5	15	104
SPC ATRIUM-9B 3.26 9Gd3.5/11Gd5.5	15	72
SPC ATRIUM-9B 3.39 6Gd3.5/6Gd4.5/6Gd5.5	15	56

	Core Average Exposure	Cycle Incremental Exposure
Exposure at EOC N-1		
Nominal EOC N-1 (MWD/MTU)	26199.6	8900.0
Short EOC N-1 (MWD/MTU)	25699.6	8400.0
Exposure at EOC N-1		
Shutdown Reactivity Calculations, (MWD/MTU)	25394.9	8095.2

Cycle 15 neutronics analyses are valid for EOC N-1 exposures greater than 8400.0 MWD/MT. The exposure window that validates the pressurization transients can be found in Reference 2.

DAP 5/9/97
TAf 5/12/97

I.2.2 Core Reactivity Characteristics

All values reported below are with zero xenon and are for 68°F moderator temperature. The MICROBURN-B cold BOC K-effective bias is 1.0070 (see Reference 17).

BOC Cold K-Effective, All Rods Out		1.10471
BOC Cold K-Effective All Rods In		0.96027
BOC Cold K-Effective, Strongest Rod Out		0.99691
BOC Shutdown Margin, % ΔK		1.00
Minimum Shutdown Margin, % ΔK		1.00
Reactivity Defect (R-value) Total, % ΔK		0.04
Boron Slumping, % ΔK	0.04	
SDM Decrease from BOC, % ΔK	0.00	
Standby Liquid Control System Shutdown Margin, Cold Condition, 600 ppm (% ΔK)		4.767

II. Control Rod Withdrawal Error

Analysis was performed at 100% power, 100% flow, unblocked conditions only.

Distance Withdrawn (ft)	ΔCPR
12	0.32

The design complies with the SPC 1% plastic strain criteria via conformance to the transient LHGR limits.

DHP 5/9/97
TPf 5/12/97

III. Fuel Loading Error

The fuel loading error, including fuel mislocation and misorientation, is classified as an accident. By demonstrating that the fuel loading error meets the more stringent Anticipated Operational Occurrence (AOO) requirements, the offsite dose requirement is assured to be met. Because the events listed below result in a ΔCPR value that is less than that of the limiting transient, the AOO requirements and hence the off-site dose requirements are met for the fuel loading error.

The values reported below bound all fuel types found in the core.

Event	ΔCPR
Mislocated Bundle	0.28
Misoriented Bundle	0.10

For the fuel loading error, the design complies with the SPC 1% plastic strain criteria via conformance to the transient LHGR limits.

DAP 3/20/97
TPS 3/2

IV. Control Rod Drop Accident

This Analysis was performed using a rod sequence that bounds the Dresden-supplied rod sequence (to be used in D3C15) as described by References 22 and 23 (rod arrays 1-4) and Reference 18 (rod arrays 5-8). Note that the 0.32%Δk adder mentioned below is included in this analysis to account for possible rod mispositioning errors.

Dropped Control Rod Worth without 0.32 %Δk adder, %Δk	0.793
Dropped Control Rod Worth with 0.32 %Δk adder, %Δk	1.113
Doppler Coefficient, 1/k Δk/dT	-10.44E-06 $(^{\circ}F)^{-1}$
Effective Delayed Neutron Fraction used	0.0055
Four-Bundle Local Peaking Factor	1.28
Maximum Deposited Fuel Rod Enthalpy with 0.32 %Δk adder, (Cal/gm)	168
Number of Rods Greater than 170 Cal/gm with 0.32%Δk adder	0

V. Loss of Feedwater Heating

The loss of feedwater heating event is analyzed at 100% of rated power, 87% and 100% of rated flow and an assumed inlet temperature decrease of 200°F.

Event	ΔCPR
Loss of Feedwater Heating	0.22

The design complies with the SPC 1% plastic strain criteria via conformance to the transient LHGR limits.

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TR 5/12

VI. Maximum Exposure Limit Compliance

Note that these exposures are based on the nominal Cycle 14 exposure, 8900 MWD/MT, and an End Of Cycle 15 core exposure of 26,386 MWD/MT (this is the licensing basis core exposure at EOC15 per Reference 2). See References 19-21 for fuel assembly exposure limits.

Projected Peak Assembly Exposure (MWD/MTU) (Assembly A3U013 @ 35-36 - 9x9-2 Fuel)	39,330
SPC 9x9-2 Assembly Exposure Limit (MWD/MTU)	40,000
Projected Peak Pellet Exposure (MWD/MTU) (Assembly A3U006 @ 25-36-8 - 9x9-2 Fuel)	53,811
SPC 9x9-2 Pellet Exposure Limit (MWD/MTU)	55,000

The Data found above is for 9x9-2 fuel. The ATRIUM-9B fuel will not be near the Exposure limits set for it at the end of Cycle 15. The projected peak exposures for ATRIUM-9B fuel are listed below.

Projected ATRIUM-9B Peak Assembly Exposure (MWD/MT) (Assembly A3X198 @ 23-36)	18,004
SPC ATRIUM-9B Assembly Exposure Limit (MWD/MT)	48,000
Projected ATRIUM-9B Peak Pellet Exposure (MWD/MT) (Assembly A3X062 @ 51-28-5)	27,810
SPC ATRIUM-9B Pellet Exposure Limit (MWD/MT)	60,000

VII. Spent Fuel Pool and New Fuel Vault Criticality Compliance

For the D3C15 reload, there are three new SPC ATRIUM-9B assembly types consisting of 7 unique lattices, as identified in Section I.1. As described in the Reference 13 and 14 transmittals, all three fresh bundle types comply with the spent fuel pool and new fuel vault criticality limits.

VII.1 New Fuel Vault Criticality Compliance

All the new assemblies comply with the fresh fuel vault criticality limits of enrichment less than 5.00 wt% U-235 (lattice average) and gadolinia content greater than 6 rods at 2.0 wt% Gd_2O_3. Reference 11 details the analysis showing that the above enrichment/Gd limits insure compliance with the Reference 15 UFSAR section.

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VII.2 Spent Fuel Pool Criticality Compliance

All the new assemblies comply with the spent fuel pool criticality limits of enrichment less than 4.30 wt% U-235 (lattice average) and gadolinia content greater than 6 rods at 2.0 wt% Gd_2O_3. Reference 12 details the analysis showing that the above enrichment/Gd limits insure compliance with the Reference 16 Technical Specifications section.

VIII. References

1. EMF-96-139, "Dresden Unit 3 Cycle 15 Plant Transient Analysis," May 1997.

2. EMF-96-141, "Dresden Unit 3 Cycle 15 Reload Analysis," May 1997.

3. Calcnote NFS:BNDD:97-002, "Dresden Unit 3 Cycle 15 Neutronic Licensing Basepoint and SDM Calculations," Revision 0, 1/29/97.

4. Calcnote NFS:BNDD:97-003, "Dresden Unit 3 Cycle 15 CRDA Analysis," Revision 0, 3/6/97.

5. Calcnote NFS:BNDD:96-042, "Dresden 3 Cycle 15 Fuel Assembly Mislocation Calculations," Revision 0, 3/4/97.

6. Calcnote NFS:BNDD:96-044, "Dresden 3 Cycle 15 Loss of Feedwater Heating Analysis," Revision 0, 2/3/97.

7. Calcnote NFS:BNDD:97-006, "Dresden Unit 3 Cycle 15 Rod Withdrawal Error Analysis," Revision 0, 2/28/97.

8. Calcnote NFS:BNDD:97-011, "Dresden Unit 3 Cycle 15 Standby Liquid Control Calculations," Revision 0, 2/24/97.

9. Calcnote NFS:BNDD:97-010, "Dresden Unit 3 Cycle 15 Misoriented Bundle Analysis," Revision 0, 3/7/97.

10. Calcnote NFS:BNDD:97-016, "Dresden Unit 3 Cycle 15 Analysis of 030697 Loading Pattern," Revision 0, 3/13/97.

11. EMF-96-148(P), "Criticality Safety Analysis for ATRIUM-9B Fuel, Dresden and Quad Cities New Fuel Storage Vault," Revision 1, September 1996.

12. EMF-94-098(P), "Criticality Safety Analysis for ATRIUM-9B Fuel, Dresden Units 2 and 3 Spent Fuel Storage Pool," Revision 1, January 1996.

13. Letter NFS:BSS:96-005, "D3C15 New Fuel Storage Vault and Spent Fuel Storage Pool Criticality Analysis (Current Technical Specification)," C.H. Nguyen to J.W. Keffer, January 16, 1996.

DHP 5/7/97
TPH 5/1

14. Letter NFS:BSS:96-098, "D3C15 New Fuel Storage Vault and Spent Fuel Storage Pool Criticality Analysis (Current Technical Specification) - Extra 56 Fuel Bundles," C.H. Nguyen to R. Kundalkar, May 28, 1996.

15. "Dresden Updated Final Safety Analysis Report," Section 9.1.1.3.

16. "Dresden Units 2 and 3 Technical Specifications," Section 5.6.A.

17. Calcnote NFS:BND:95-038, "Selection of D3C15 Target Eigenvalues," Revision 0, 4/12/96.

18. Letter "Dresden Control Rod Withdrawal Sequences," from Carlos de la Hoz to Ronald J. Chin, 9/16/94.

19. XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," September 1986.

20. XN-NF-82-06(P)(A), Supplement 1, Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup, Supplement 1 Extended Burnup Qualification of ENC 9x9 BWR Fuel," May 1988.

21. EMF-96-040(P), Revision 1, "Fuel Design Report for Dresden Unit 3, Cycle 15 ATRIUM-9B Fuel Assemblies," August 1996.

22. Letter "Control Rod Arrays and Bank Positions for D3C15," from R.D. Freeman to Dr. R. Chin, July 16, 1996, Doc. I.D.# 5028404.

23. Letter "New BPWS 'X' Rod Pull Order," from R.D. Freeman to Dr. R. Chin, August 10, 1995.

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TBR 5/12/97

1 1.80	2 2.20	3 2.70	3 2.70	4 3.30	4 3.30	4 3.30	3 2.70	3 2.70
2 2.20	G 2.70	4 3.30	4 3.30	G 2.70	5 4.06	5 4.06	G 2.70	4 3.30
3 2.70	4 3.30	4 3.30	4 3.30	4 3.30	5 4.06	5 4.06	5 4.06	5 4.06
3 2.70	4 3.30	4 3.30				3 2.70	5 4.06	5 4.06
4 3.30	G 2.70	4 3.30		Internal Water Channel		5 4.06	G 2.70	5 4.06
4 3.30	5 4.06	5 4.06				5 4.06	5 4.06	5 4.06
4 3.30	5 4.06	5 4.06	3 2.70	5 4.06	5 4.06	G 2.70	5 4.06	5 4.06
3 2.70	G 2.70	5 4.06	5 4.06	G 2.70	5 4.06	5 4.06	G 2.70	5 4.06
3 2.70	4 3.30	5 4.06	5 4.06	5 4.06	5 4.06	5 4.06	5 4.06	4 3.30

1	Rods (1)	at 1.80 w/o U-235
2	Rods (2)	at 2.20 w/o U-235
3	Rods (10)	at 2.70 w/o U-235
4	Rods (18)	at 3.30 w/o U-235
G	Rods (9)	at 2.70 w/o U-235 + 3.5 w/o Gd
5	Rods (32)	at 4.06 w/o U-235

Figure 1
SPC ATRIUM-9B 3.43 9Gd3.5 (DRC-8L and 8H)
Enrichment Distribution

1 1.80	2 2.20	3 2.70	3 2.70	4 3.30	4 3.30	4 3.30	3 2.70	3 2.70
2 2.20	G 2.70	4 3.30	4 3.30	G 2.70	5 4.06	5 4.06	G 2.70	4 3.30
3 2.70	4 3.30	4 3.30	4 3.30	4 3.30	5 4.06	5 4.06	5 4.06	5 4.06
3 2.70	4 3.30	4 3.30				G 2.70	5 4.06	5 4.06
4 3.30	G 2.70	4 3.30		Internal Water Channel		5 4.06	G 2.70	5 4.06
4 3.30	5 4.06	5 4.06				5 4.06	5 4.06	5 4.06
4 3.30	5 4.06	5 4.06	G 2.70	5 4.06	5 4.06	G 2.70	5 4.06	5 4.06
3 2.70	G 2.70	5 4.06	5 4.06	G 2.70	5 4.06	5 4.06	G 2.70	5 4.06
3 2.70	4 3.30	5 4.06	5 4.06	5 4.06	5 4.06	5 4.06	5 4.06	4 3.30

1	Rods (1)	at 1.80 w/o U-235
2	Rods (2)	at 2.20 w/o U-235
3	Rods (8)	at 2.70 w/o U-235
4	Rods (18)	at 3.30 w/o U-235
G	Rods (11)	at 2.70 w/o U-235 + 5.5 w/o Gd
5	Rods (32)	at 4.06 w/o U-235

Figure 2
SPC ATRIUM-9B 3.44 11Gd5.5 (DRC-8L and 8H)
Enrichment Distribution

1 1.80	2 2.50	3 3.10	3 3.10	6 3.30	6 3.30	6 3.30	3 3.10	7 2.70
2 2.50	G 3.10	4 3.86	4 3.86	G 3.10	5 4.39	5 4.39	G 3.10	4 3.86
3 3.10	4 3.86	4 3.86	4 3.86	4 3.86	5 4.39	5 4.39	5 4.39	5 4.39
3 3.10	4 3.86	4 3.86				3 3.10	5 4.39	5 4.39
6 3.30	G 3.10	4 3.86		Internal Water Channel		5 4.39	G 3.10	5 4.39
6 3.30	5 4.39	5 4.39				5 4.39	5 4.39	5 4.39
6 3.30	5 4.39	5 4.39	3 3.10	5 4.39	5 4.39	G 3.10	5 4.39	5 4.39
3 3.10	G 3.10	5 4.39	5 4.39	G 3.10	5 4.39	5 4.39	G 3.10	5 4.39
7 2.70	4 3.86	5 4.39	5 4.39	5 4.39	5 4.39	5 4.39	5 4.39	4 3.86

1	Rods (1)	at 1.80 w/o U-235
2	Rods (2)	at 2.50 w/o U-235
3	Rods (8)	at 3.10 w/o U-235
G	Rods (9)	at 3.10 w/o U-235 + 4.5 w/o Gd
4	Rods (12)	at 3.86 w/o U-235
5	Rods (32)	at 4.39 w/o U-235
6	Rods (6)	at 3.30 w/o U-235
7	Rods (2)	at 2.70 w/o U-235

Figure 3
SPC ATRIUM-9B 3.78 9Gd4.5 (DRC-8L)
Enrichment Distribution

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TPS 3/2

1 1.80	2 2.50	3 3.10	3 3.10	6 3.30	6 3.30	6 3.30	3 3.10	7 2.70
2 2.50	G 3.10	4 3.86	4 3.86	G 3.10	5 4.39	5 4.39	G 3.10	4 3.86
3 3.10	4 3.86	4 3.86	4 3.86	4 3.86	5 4.39	5 4.39	5 4.39	5 4.39
3 3.10	4 3.86	4 3.86				G 3.10	5 4.39	5 4.39
6 3.30	G 3.10	4 3.86		Internal Water Channel		5 4.39	G 3.10	5 4.39
6 3.30	5 4.39	5 4.39				5 4.39	5 4.39	5 4.39
6 3.30	5 4.39	5 4.39	G 3.10	5 4.39	5 4.39	G 3.10	5 4.39	5 4.39
3 3.10	G 3.10	5 4.39	5 4.39	G 3.10	5 4.39	5 4.39	G 3.10	5 4.39
7 2.70	4 3.86	5 4.39	5 4.39	5 4.39	5 4.39	5 4.39	5 4.39	4 3.86

1	Rods (1)	at 1.80 w/o U-235
2	Rods (2)	at 2.50 w/o U-235
3	Rods (6)	at 3.10 w/o U-235
G	Rods (11)	at 3.10 w/o U-235 + 5.5 w/o Gd
4	Rods (12)	at 3.86 w/o U-235
5	Rods (32)	at 4.39 w/o U-235
6	Rods (6)	at 3.30 w/o U-235
7	Rods (2)	at 2.70 w/o U-235

Figure 4
SPC ATRIUM-9B 3.78 11Gd5.5 (DRC-8H)
Enrichment Distribution

3/20

1 2.00	2 2.30	7 3.00	7 3.00	4 3.53	4 3.53	4 3.53	7 3.00	3 2.75
2 2.30	7 3.00	4 3.53	G 3.00	4 3.53	5 4.30	5 4.30	G 3.00	4 3.53
7 3.00	4 3.53	4 3.53	4 3.53	4 3.53	4 3.53	5 4.30	5 4.30	5 4.30
7 3.00	G 3.00	4 3.53	Internal Water Channel			7 3.00	5 4.30	5 4.30
4 3.53	4 3.53	4 3.53				5 4.30	5 4.30	5 4.30
4 3.53	5 4.30	4 3.53				5 4.30	G 3.00	5 4.30
4 3.53	5 4.30	5 4.30	7 3.00	5 4.30	5 4.30	7 3.00	5 4.30	5 4.30
7 3.00	G 3.00	5 4.30	5 4.30	5 4.30	G 3.00	5 4.30	7 3.00	4 3.53
3 2.75	4 3.53	5 4.30	5 4.30	5 4.30	5 4.30	5 4.30	4 3.53	7 3.00

1	Rods (1)	2.00 w/o U-235
2	Rods (2)	2.30 w/o U-235
3	Rods (2)	2.75 w/o U-235
4	Rods (21)	3.53 w/o U-235
G	Rods (6)	3.00 w/o U-235 + 3.5 w/o Gd
5	Rods (28)	4.30 w/o U-235
7	Rods (12)	3.00 w/o U-235

Figure 5

SPC ATRIUM-9B 3.62 6Gd3.5 (DRC-8A)
Enrichment Distribution

1 2.00	2 2.30	7 3.00	7 3.00	4 3.53	4 3.53	4 3.53	7 3.00	3 2.75
2 2.30	7 3.00	4 3.53	G 3.00	4 3.53	5 4.30	5 4.30	G 3.00	4 3.53
7 3.00	4 3.53	4 3.53	4 3.53	4 3.53	4 3.53	5 4.30	5 4.30	5 4.30
7 3.00	G 3.00	4 3.53				7 3.00	5 4.30	5 4.30
4 3.53	4 3.53	4 3.53		Internal Water Channel		5 4.30	5 4.30	5 4.30
4 3.53	5 4.30	4 3.53				5 4.30	G 3.00	5 4.30
4 3.53	5 4.30	5 4.30	7 3.00	5 4.30	5 4.30	7 3.00	5 4.30	5 4.30
7 3.00	G 3.00	5 4.30	5 4.30	5 4.30	G 3.00	5 4.30	7 3.00	4 3.53
3 2.75	4 3.53	5 4.30	5 4.30	5 4.30	5 4.30	5 4.30	4 3.53	7 3.00

1	Rods (1)	2.00 w/o U-235
2	Rods (2)	2.30 w/o U-235
3	Rods (2)	2.75 w/o U-235
4	Rods (21)	3.53 w/o U-235
G	Rods (6)	3.00 w/o U-235 + 4.5 w/o Gd
5	Rods (28)	4.30 w/o U-235
7	Rods (12)	3.00 w/o U-235

Figure 6

SPC ATRIUM-9B 3.62 6Gd4.5 (DRC-8A) Enrichment Distribution

1 2.00	2 2.30	8 3.00	3 3.30	4 3.78	4 3.78	4 3.78	3 3.30	8 3.00
2 2.30	3 3.30	4 3.78	G 3.53	4 3.78	5 4.55	5 4.55	G 3.53	4 3.78
8 3.00	4 3.78	4 3.78	4 3.78	4 3.78	4 3.78	5 4.55	5 4.55	5 4.55
3 3.30	G 3.53	4 3.78	Internal Water Channel			3 3.30	5 4.55	5 4.55
4 3.78	4 3.78	4 3.78				7 4.30	5 4.55	5 4.55
4 3.78	5 4.55	4 3.78				5 4.55	G 3.53	5 4.55
4 3.78	5 4.55	5 4.55	3 3.30	7 4.30	5 4.55	9 3.53	5 4.55	5 4.55
3 3.30	G 3.53	5 4.55	5 4.55	5 4.55	G 3.53	5 4.55	9 3.53	4 3.78
8 3.00	4 3.78	5 4.55	5 4.55	5 4.55	5 4.55	5 4.55	4 3.78	3 3.30

1	Rods (1)	2.00 w/o U-235
2	Rods (2)	2.30 w/o U-235
3	Rods (8)	3.30 w/o U-235
4	Rods (21)	3.78 w/o U-235
G	Rods (6)	3.53 w/o U-235 + 5.5 w/o Gd
5	Rods (26)	4.55 w/o U-235
7	Rods (2)	4.30 w/o U-235
8	Rods (4)	3.00 w/o U-235
9	Rods(2)	3.53 w/o U-235

Figure 7
SPC ATRIUM-9B 3.88 6Gd5.5 (DRC-8A)
Enrichment Distribution

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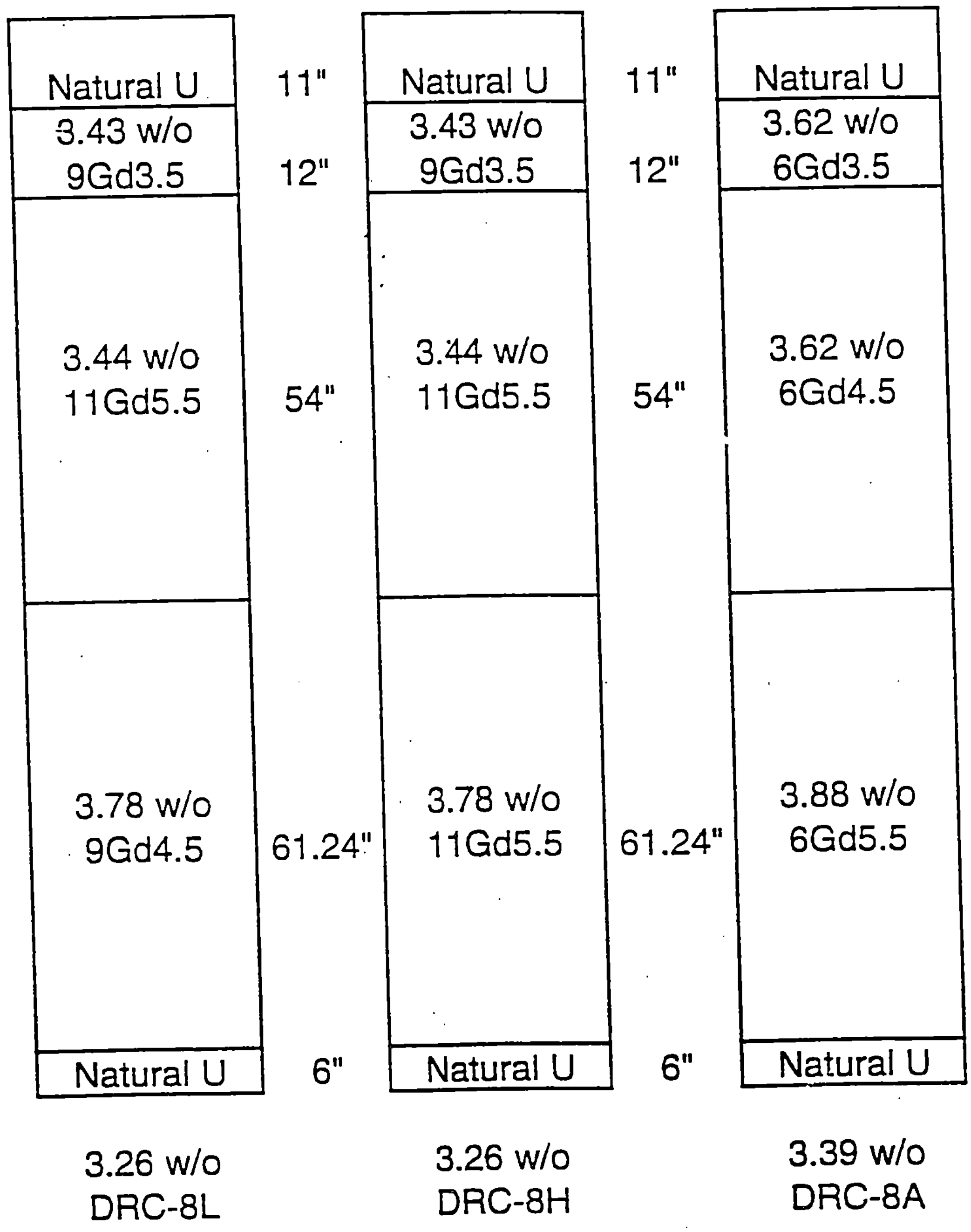
Figure 8. DR315 ATRIUM-9B Bundle Axial Designs



ATTACHMENT A - NEUTRONIC LICENSING PROCEDURE REFERENCES

1. NFS-ND-900, "Nuclear Design Procedures," Revision 4, 8/20/96, Appendix A, "Performing Nuclear Design Routine Controlled Analysis using SPC Methods," Section 5.9, "Design Shutdown Margin Calculations," Revision 1, 9/2/92.

2. NFS-ND-900, "Nuclear Design Procedures," Revision 4, 8/20/96, Appendix A, "Performing Nuclear Design Routine Controlled Analysis using SPC Methods," Section 8.2, "Bundle Misorientation Calculations," Revision 0, 8/9/93.

3. NFS-ND-900, "Nuclear Design Procedures," Revision 4, 8/20/96, Appendix A, "Performing Nuclear Design Routine Controlled Analysis using SPC Methods," Section 8.3, "Fuel Assembly Mislocation Calculations," Revision 0, 8/10/93.

4. NFS-ND-900, "Nuclear Design Procedures," Revision 4, 8/20/96, Appendix A, "Performing Nuclear Design Routine Controlled Analysis using SPC Methods," Section 8.4, "Rod Withdrawal Error Calculations," Revision 0, 9/21/93.

5. NFS-ND-900, "Nuclear Design Procedures," Revision 4, 8/20/96, Appendix A, "Performing Nuclear Design Routine Controlled Analysis using SPC Methods," Section 8.5 "Control Rod Drop Accident Analysis," Revision 0, 7/22/94.

6. NFS-ND-900, "Nuclear Design Procedures," Revision 4, 8/20/96, Appendix A, "Performing Nuclear Design Routine Controlled Analysis using SPC Methods," Section 8.7, "Standby Liquid Control System (SBLC) Worth Calculations," Revision 0, 4/6/93.

7. NFS-ND-900, "Nuclear Design Procedures," Revision 4, 8/20/96, Appendix A, "Performing Nuclear Design Routine Controlled Analysis using SPC Methods," Section 8.8, "Loss of Feedwater Heating Transient Analysis," Revision 2, 5/20/94.

8. NFS-ND-900, "Nuclear Design Procedures," Revision 4, 8/20/96, Appendix A, "Performing Nuclear Design Routine Controlled Analysis using SPC Methods," Section 8.6, "Reload Licensing Report," Revision 0, 11/2/93.

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Attachment 2

Dresden Unit 3 Cycle 15

Reload Analysis Report

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PDR ADOCK 05000249
P PDR

Siemens Power Corporation - Nuclear Division

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DATE: 6/22/98

EMF-96-141
Revision 1
Issue Date:

Dresden Unit 3 Cycle 15
Reload Analysis

Prepared by:

A W Will

A. W. Will, Engineer
BWR Safety Analysis
Nuclear Engineering

and

D C Serell

D. C. Serell, Engineer
BWR Safety Analysis
Nuclear Engineering

June 1998

paj

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Nature of Change

Title: **Dresden Unit 3 Cycle 15 Reload Analysis**

Superseded Issue: 0

Item	Page No.	Nature of Change

GENERAL COMMENT: The report is revised to include analysis results and licensing limits for new plant operating conditions. Specifically, the revised parameters are a steam flow of 9.90 Mlbm/hr, feedwater flow of 9.87 Mlbm/hr, control rod drive flow of 0.03 Mlbm/hr, and a feedwater temperature range of 340°F–350°F. Revised LOCA analysis results and extended 9x9-2 LHGR limits are included.

Item	Page No.	Nature of Change
1	3	Added text describing extended LHGR limits for 9x9-2 fuel.
2	4	Changed SLMCPR Reference to 9.16.
3	5	Updated Thermal Power and Feedwater Flow Rate at SLMCPR. Added a footnote to the feedwater temperature in Section 3.3.1.
4	8	Added statements of stability analysis applicability with increased steam flow.
5	9	Replaced results in Section 5.1 with results from analyses performed with revised operating conditions. Revised footnote (a).
6	10	Updated ASME overpressurization results and added footnote (a).
7	11	Updated ΔCPR results and MCPR limits. Added text to footnote (a), and added footnote (c).
8	14	Extended steady-state LHGR limits for 9x9-2 fuel to 50.9 GWd/MTU and updated the reference to Reference 9.14.
9	16	Updated References 9.3, 9.4, and 9.5 to current reports. Added References 9.14, 9.15, and 9.16.
10	19, 20	Eliminated maximum F-eff value from Figures 3.2 and 3.3.
11	35	Updated Figure 7.1 consistent with Item 8 above.
12	A-2	Added footnote to single-loop MCPR limit determination.
13	A-3	Added footnote to SLO MAPLHGR multiplier.
14	A-3	Updated Reference A.2 to latest revision. Added Reference A.3.

NOTE: Changed items are further identified with (|) in left margins.

Contents

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(Continued)

Tables

Figures

Dresden Unit 3 Cycle 15 Reload Analysis

1.0 INTRODUCTION

This report provides the results of the analysis performed by Siemens Power Corporation - Nuclear Division (SPC-ND) in support of the Cycle 15 reload for Dresden Unit 3. This report is intended to be used in conjunction with the SPC-ND topical Report XN-NF-80-19(P)(A), Volume 4, Revision 1, *Application of the ENC Methodology to BWR Reloads*, which describes the analyses performed in support of this reload, identifies the methodology used for those analyses, and provides a generic reference list. Section numbers in this report are the same as corresponding section numbers in XN-NF-80-19(P)(A), Volume 4, Revision 1. Methodology used in this report which supersedes XN-NF-80-19(P)(A), Volume 4, Revision 1, is referenced in Section 8.0. The NRC Technical Limitations presented in the methodology documents, including the documents referenced in Section 8.0, have been satisfied by these analyses.

For Dresden Unit 3 Cycle 15, Commonwealth Edison Company (ComEd) has responsibility for portions of the reload safety analysis. This document describes only the Cycle 15 analyses performed by SPC-ND; ComEd analyses are described elsewhere. This document alone does not necessarily identify the limiting events or the appropriate operating limits for Cycle 15. The limiting events and operating limits must be determined in conjunction with results from ComEd analyses.

The Dresden Unit 3 Cycle 15 core consists of a total of 724 fuel assemblies, including 232 unirradiated DRC-8 and DRC-8A ATRIUM™-9B[a] assemblies and 492 irradiated SPC-ND 9x9-2 and 9x9-2B assemblies. The reference core configuration is described in Section 4.2.

The design and safety analyses reported in this document were based on the design and operational assumptions in effect for Dresden Unit 3 during the previous operating cycle. The effects of channel bow are explicitly accounted for in the safety limit analysis. Increased core

[a] ATRIUM is a trademark of Siemens.

flow was not evaluated for Cycle 15. SPC-ND has performed time step size sensitivity studies to assure that the numerics solution in the COTRANSA2 code converged.

Analyses and limits presented in this report support operation with various equipment out of service (EOOS). The EOOS conditions addressed in this report include:

- Feedwater heaters out of service
- Relief valve out of service
- Safety/relief valve safety function out of service (ASME events)
- Up to 40% TIP channels (equivalent of up to 2 TIP machines) out of service
- Single-loop operation

2.0 FUEL MECHANICAL DESIGN ANALYSIS

Applicable SPC-ND Fuel Design Reports — References 9.1, 9.7, 9.8, 9.14

To assure that the power history for the fuel to be irradiated during Cycle 15 of Dresden Unit 3 is bounded by the assumed power history in the fuel mechanical design analysis, LHGR operating limits have been specified. In addition, LHGR limits for Anticipated Operational Occurrences have been specified in the references. The LHGR limits for 9x9-2 fuel were extended from a planar exposure of 48.0 GWd/MTU (Reference 9.1) to 50.9 GWd/MTU (Reference 9.14). Steady-state and transient LHGR limits are provided in Section 7.2.3 and in Figures 7.1 and 7.2 for both ATRIUM-9B and 9x9-2 fuel. The bundle exposure limit of 40 GWd/MTU for the 9x9-2 fuel (Reference 9.1) is not changed.

3.0 THERMAL-HYDRAULIC DESIGN ANALYSIS

3.2 Hydraulic Characterization

3.2.1 Hydraulic Compatibility

Component hydraulic resistances for the constituent fuel types in the Dresden Unit 3 Cycle 15 core have been determined in single-phase flow tests of full-scale assemblies. The hydraulic demand curves for SPC-ND 9x9-2 and ATRIUM-9B fuel in the Dresden Unit 3 core are provided in Reference 9.7 (Figure 5.2 in the reference).

3.2.3 Fuel Centerline Temperature

9x9-2	Reference 9.1, Figure 3.21
ATRIUM-9B	Reference 9.7, Figure 3.2

3.2.5 Bypass Flow

Calculated Bypass Flow Fraction at 100% power/100% flow at EOC[a]	10.9%

3.3 MCPR Fuel Cladding Integrity Safety Limit (SLMCPR)

Safety Limit MCPR - 1.08[b],[c]	Reference 9.16

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[a] Includes water rod/channel flow.

[b] Analyses performed support a two-loop MCPR safety limit of 1.08 or greater. Operating limits are based on the Technical Specification two-loop MCPR safety limit of 1.08.

[c] Includes the effects of channel bow, up to 40% of the TIP channels (equivalent of up to 2 TIP machines) out of service, a 2000 EFPH calibration interval, and up to 50% of the LPRMs out of service.

3.3.1 Coolant Thermodynamic Condition

Thermal Power (at SLMCPR)	4260 MWt
Feedwater Flow Rate (at SLMCPR)	16.5 Mlb/hr
Core Pressure	1030 psia
Feedwater Temperature	340.1 °F[a]

3.3.2 Design Basis Radial Power Distribution

Figure 3.1 shows the limiting radial power distribution used in the MCPR Fuel Cladding Integrity Safety Limit analysis.

3.3.3 Design Basis Local Power Distribution

Figures 3.2 and 3.3 show the conservative local power distributions used in the MCPR Fuel Cladding Integrity Safety Limit analysis.

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[a] Disposition of increased feedwater temperature associated with increased steam flow is presented in Reference 9.15.

4.0 NUCLEAR DESIGN ANALYSIS

4.1 Fuel Bundle Nuclear Design Analysis

Assembly Average Enrichment	
ATRIUM-9B (DRC-8)	3.26 wt%
(DRC-8A)	3.39 wt%
Radial Enrichment Distribution	
SPCA9-343L-9G3.5	Figure 4.1
SPCA9-344L-11G5.5	Figure 4.2
SPCA9-378L-11G5.5	Figure 4.3
SPCA9-378L-9G4.5	Figure 4.4
SPCA9-362L-6G3.5	Figure 4.5
SPCA9-362L-6G4.5	Figure 4.6
SPCA9-388L-6G5.5	Figure 4.7
Axial Enrichment Distribution	Figures 4.8 and 4.9
Burnable Absorber Distribution	Figures 4.8 and 4.9
Non-Fueled Rods	Figures 4.1—4.7
Neutronics Design Parameters	Table 4.1
Maximum Lattice k_∞ [a]	
ATRIUM-9B	References 9.9 and 9.10

(a) The ATRIUM-9B is bounded by the referenced analysis.

4.2 Core Nuclear Design Analysis

4.2.1 Core Configuration

	Figure 4.10
Core Exposure at EOC14, MWd/MTU (nominal value)	26,292
Core Exposure at BOC15, MWd/MTU (from nominal EOC14)	14,527
Core Exposure at EOC15, MWd/MTU (licensing basis)	26,386

NOTE: Analyses in this report are applicable to a core exposure of 26,386 MWd/MTU. Generic coastdown analyses (References 9.6 and 9.11) are applicable for Cycle 15 provided full power capability is lost prior to reaching a core exposure of 26,386 MWd/MTU.

< Cycle 15 short window exposure to be furnished by ComEd. >

4.2.2 Core Reactivity Characteristics

< This data is to be furnished by ComEd. >

4.2.4 Core Hydrodynamic Stability

The results of the evaluation of decay ratio for several points along the current exclusion region boundary of the power/flow operation map are shown below. These results show that the maximum STAIF decay ratio throughout the cycle occurs at the intercept of the APRM rod block line and the natural circulation flow line. This analysis was performed using the design basis step-through control rod pattern projection, hence, it explicitly models the effects of Cycle 15 exposure. The calculated decay ratios are for demonstrating and tracking relative stability from cycle to cycle. These results are applicable for increased feedwater temperature conditions associated with a 1% increase in rated steam flow. The resulting lower core inlet subcooling will have an insignificant effect on stability.

	%Power / %Flow State Points	Decay Ratio (ΔDR[a]) Global		Regional	
1.	58 / 33[b]	0.97		0.87	
2.	67 / 41[c]	0.84		0.77	
3.	73 / 45[d]	0.76	(−.10)	0.72	(.04)
4.	63 / 45[e]	0.52		0.48	
5.	34 / 28[f]	0.41	(−.11)	0.35	(.00)
6.	39 / 38[g]	0.23		0.21	

For reactor operation under conditions of coastdown, feedwater heaters out of service, and single-loop, it is possible that higher decay ratios could be achieved than are shown for normal operation. Operation under these conditions will be acceptable in Cycle 15 as long as operating procedures and precautions defined by the NRC (Reference 9.12) and BWROG (Reference 9.13) for Interim Corrective Actions are followed.

---

[a] $DR_{CY15} - DR_{CY14}$ values are in parentheses.

[b] APRM rod block line — natural circulation flow.

[c] APRM rod block line — two-pump minimum flow.

[d] APRM rod block line — 45% flow.

[e] 100% rod line — 45% flow.

[f] 70% rod line — natural circulation flow.

[g] 70% rod line — two-pump minimum flow.

5.0 ANTICIPATED OPERATIONAL OCCURRENCES

Applicable Generic Transient Analysis Report — Reference 9.2

5.1 Analysis of Plant Transients at Rated Conditions — References 9.3, 9.6 and 9.11

Limiting Transients: Load Rejection No Bypass (LRNB)
Feedwater Controller Failure (FWCF)
Loss of Feedwater Heating (LFWH)

Event	Power (%)	Flow (%)	Maximum Heat Flux (%)	Peak Neutron Flux (%)	Maximum Pressure (psig)	ΔCPR[a]	Model
LRNB[b]	100	100	128	670	1298	0.35/0.36	COTRANSA2
FWCF[b],[c]	100	100	133	528	1166	0.37/0.38	COTRANSA2
LFWH	[d]	[d]	[d]	[d]	[d]	[d]	[d]

5.2 Analysis for Reduced Flow Operation — Reference 9.3

Limiting Transient: Recirculation Flow Increase Transient (RFIT)
(Pump Run-Up Event)

5.3 Analysis for Reduced Power Operation — Reference 9.3

Limiting Transient: Feedwater Controller Failure (FWCF)

---

[a] ΔCPR results for second-cycle 9x9-2/first-cycle ATRIUM-9B fuel types with revised operating conditions.

[b] Based on Technical Specification limiting scram performance parameters.

[c] Feedwater heaters out of service (100°F reduction in feedwater temperature).

[d] This data to be furnished by ComEd.

5.4 ASME Overpressurization Analysis[a],[b]

Limiting Event	MSIV Closure
Worst Single Failure	Valve Position Scram
Maximum Pressure (Lower Plenum)	1339 psig
Maximum Steam Dome Pressure[c]	1313 psig

5.5 Control Rod Withdrawal Error

Starting Control Pattern for Analysis — Figure 5.1

< This data is to be furnished by ComEd. >

5.6 Fuel Loading Error

< This data is to be furnished by ComEd. >

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[a] Overpressurization results are for operation with revised operating conditions.

[b] Coastdown operation requires a generic pressure penalty of 5.0 psid.

[c] Maximum steam dome pressure corresponds to the analysis that resulted in the maximum vessel pressure.

5.7 Determination of Thermal Margins

Summary of Thermal Margin Requirements

Event	Power (%)	Flow (%)	ΔCPR[a]	MCPR Limit[a]
LRNB	100	100	0.35 / 0.36	1.43 / 1.44[b]
FWCF[c]	100	100	0.37 / 0.38	1.45 / 1.46[b]
CRWE	[d]	[d]	[d]	[d]

MCPR Operating Limit at Rated Conditions[e]

Prior to End of Licensing Basis Exposure (Section 4.2.1)	1.46
During Coastdown	1.50[f]

MCPR Operating Limits at Off-Rated Conditions[b]

Reduced Flow MCPR Limits:

Manual Flow Control	Figure 5.2
Automatic Flow Control	Figure 5.3

---

[a] Values for second-cycle 9x9-2/first-cycle ATRIUM-9B fuel types with revised operating conditions.

[b] Based on plant Technical Specification two-loop MCPR safety limit of 1.08 and Technical Specification limiting scram performance parameters.

[c] Feedwater heaters out of service (100°F reduction in feedwater temperature).

[d] This data is to be furnished by ComEd.

[e] The presented limits are applicable to both ATRIUM-9B and 9x9-2 fuel in the Cycle 15 core. These limits may need to be increased if ComEd CRWE analysis results are more limiting.

[f] Generic MCPR penalty of 0.04 is added to the MCPR operating limit to support coastdown operation beginning at EOFP (References 9.6 and 9.11). This penalty is not necessary if the station elects to monitor to the core thermal power limit in Figure 2.1 in Reference 9.11. If the 0.04 adder is applied, the core thermal power limit provided in Figure 2.2 in Reference 9.11 must be maintained.

6.0 POSTULATED ACCIDENTS

6.1 Loss-of-Coolant Accident

6.1.1 Break Location Spectrum — Reference 9.4

6.1.2 Break Size Spectrum — Reference 9.4

6.1.3 MAPLHGR Analyses — Reference 9.5

The MAPLHGR limits of Reference 9.5 are valid for the Dresden Unit 3 9x9-2 (ANF-5, ANF-6, DRC-7) and ATRIUM-9B (DRC-8 and DRC-8A) fuels for Cycle 15 operation. MAPLHGR limits are presented in Section 7.2.1.

Limiting Break: Double-Ended Guillotine Pipe Break
Recirculation Pump Suction Line
1.0 Discharge Coefficient
LPCI Valve Failure - DBA Single Failure

Peak clad temperature, peak metal water reaction (MWR), and total core MWR are 1920°F, <1.09% locally, and <0.12% core wide, respectively for 9x9-2 fuel with flow measurement uncertainties. For ATRIUM-9B fuel, PCT, peak MWR, and total core MWR are 1838°F, <0.80% locally, and <0.12% core wide, respectively with flow measurement uncertainties. The 9x9-2 fuel is the limiting fuel type for Cycle 15.

6.2 Control Rod Drop Accident

< This data is to be furnished by ComEd. >

7.0 TECHNICAL SPECIFICATIONS

7.1 Limiting Safety System Settings

7.1.1 MCPR Fuel Cladding Integrity Safety Limit

MCPR Safety Limit (all fuel) 1.08[a],[b]

7.1.2 Steam Dome Pressure Safety Limit

Pressure Safety Limit 1345 psig

7.2 Limiting Conditions for Operation

7.2.1 Average Planar Linear Heat Generation Rate

Planar Average Exposure (GWd/MTU)	9x9-2 MAPLHGR (kW/ft)	ATRIUM-9B MAPLHGR (kW/ft)
0	12.5	13.5
15	12.5	13.5
20	11.9	13.5
55	7.7	9.3
60	---	8.7

As long as bundle exposures are within the range of exposures considered in the LOCA analyses, the specified MAPLHGR limits remain valid during coastdown operation.

(a) Analyses performed support a two-loop MCPR safety limit of 1.08 or greater. Operating limits are based on the Technical Specification two-loop MCPR safety limit of 1.08.

(b) Includes the effects of channel bow, up to 40% of the TIP channels (equivalent of up to 2 TIP machines) out of service, a 2000 EFPH calibration interval, and up to 50% of the LPRMs out of service.

7.2.2 Minimum Critical Power Ratio

Rated Conditions MCPR Limit Based on Technical Specification Scram Times	[a]
Off-Rated Conditions MCPR Limits:	
Manual Flow Control	Figure 5.2
Automatic Flow Control	Figure 5.3

7.2.3 Linear Heat Generation Rate — Figure 1 of Reference 9.14 and Figure 2.1 of Reference 9.7

Steady-State LHGR Limits

9x9-2 Fuel		ATRIUM-9B Fuel[b]	
Planar Average Exposure (GWd/MTU)	LHGR (kW/ft)	Planar Average Exposure (GWd/MTU)	LHGR (kW/ft)
0.0	14.5	0.0	14.4
5.0	14.5	15.0	14.4
25.2	10.8	55.0	9.1
48.0	7.2		
50.9	6.7		

The transient linear heat generation rate curve is Figure 2 of Reference 9.14 for 9x9-2 and Figure 2.2 of Reference 9.7 for ATRIUM-9B. These figures are presented in the report as Figures 7.1 and 7.2[b] for convenience.

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[a] Based on limiting results from Section 5.7 or analyses within ComEd's scope of responsibility. The MCPR operating limit is based on a Technical Specification two-loop MCPR safety limit of 1.08 and the limiting ΔCPR for Cycle 15.

[b] ATRIUM-9B planar exposure is limited to 55 GWd/MTU based on a peak pellet exposure of 60 GWd/MTU.

8.0 METHODOLOGY REFERENCES

See XN-NF-80-19(P)(A), Volume 4, Revision 1 for a complete bibliography.

8.1 *COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analysis*, ANF-913(P)(A), Volume 1, Revision 1, Supplements 1, 2, 3 and 4, Advanced Nuclear Fuels Corporation, August 1990.

8.2 *Advanced Nuclear Fuels Critical Power Methodology for Boiling Water Reactors*, XN-NF-524(P)(A), Revision 2, and Supplements, Advanced Nuclear Fuels Corporation, November 1990.

8.3 *ANFB Critical Power Correlation*, ANF-1125(P)(A), Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.

8.4 *Advanced Nuclear Fuels Methodology for Boiling Water Reactors: Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology*, XN-NF-80-19(P)(A), Volume 1 and Supplement 3, Appendix F and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

8.5 *STAIF: A Computer Program for BWR Stability Analysis in the Frequency Domain*, EMF-CC-074(P)(A), Volume 1 and Volume 2, Siemens Power Corporation - Nuclear Division, July 1994.

9.0 ADDITIONAL REFERENCES

9.1 *Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel*, XN-NF-85-67(P)(A), Revision 1, Exxon Nuclear Company, Inc., September 1986.

9.2 *Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors*, XN-NF-79-71(P), Revision 2, including Supplements 1 through 3(P)(A), Exxon Nuclear Company, Inc., March 1986.

9.3 *Dresden Unit 3 Cycle 15 Plant Transient Analysis With Increased Steam Flow*, EMF-97-047, Siemens Power Corporation - Nuclear Division, June 1998.

9.4 *LOCA Break Spectrum Analysis for Dresden Units 2 and 3*, EMF-97-025(P), Revision 1, Siemens Power Corporation - Nuclear Division, May 1997.

9.5 *Dresden LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM™-9B and 9x9-2 Fuel*, EMF-98-007(P), Siemens Power Corporation - Nuclear Division, January 1998.

9.6 *Dresden Units 2 and 3 Generic Coastdown Analysis,* EMF-92-149(P), Siemens Power Corporation - Nuclear Division, October 1992.

9.7 *Fuel Design Report for Dresden 3, Cycle 15 ATRIUM™-9B Fuel Assemblies*, EMF-96-040(P), Revision 1, Siemens Power Corporation - Nuclear Division, August 1996.

9.8 *Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X BWR Reload Fuel*, ANF-89-014(P)(A) Revision 1 and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, October 1991.

9.9 *Criticality Safety Analysis for Dresden Units 2 and 3 Spent Fuel Storage Pool*, EMF-94-098(P), Revision 1, Siemens Power Corporation - Nuclear Division, January 1996.

9.10 *Criticality Safety Analysis for ATRIUM™-9B Fuel Dresden and Quad Cities New Fuel Storage Vaults*, EMF-96-148(P), Revision 1, Siemens Power Corporation - Nuclear Division, September 1996.

9.11 *Dresden Units 2 and 3 Generic Coastdown Analysis for ATRIUM™-9B*, EMF-92-149(P), Supplement 1, Revision 1, Siemens Power Corporation - Nuclear Division, September 1996.

9.12 "Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors," NRC Generic Letter 94-02, U.S. Nuclear Regulatory Commission, July 11, 1994.

9.13 "BWR Owners' Group Guidelines for Stability Interim Corrective Action," BWR Owners' Group Letter BWROG-94078, June 6, 1994.

9.14 Letter, J. H. Riddle (SPC) to R. J. Chin (ComEd), "Extension of LHGR Curve for Higher Planar Exposure for 9x9-2 Fuel at Dresden," JHR:97:200, May 20, 1997.

9.0 ADDITIONAL REFERENCES (Continued)

9.15 Letter, J. H. Riddle (SPC) to R. J. Chin (ComEd), "Clarification of Design Operating Conditions in Transient, LOCA and Safety Limit Analyses," JHR:97:088, March 3, 1997.

9.16 Letter, D. E. Garber (SPC) to R. J. Chin (ComEd), "Dresden Unit 3 Cycle 15 SLMCPR Results with Updated ATRIUM™-9B Additive Constants and Various Additive Constant Uncertainty Approaches," DEG:98:116, April 6, 1998.

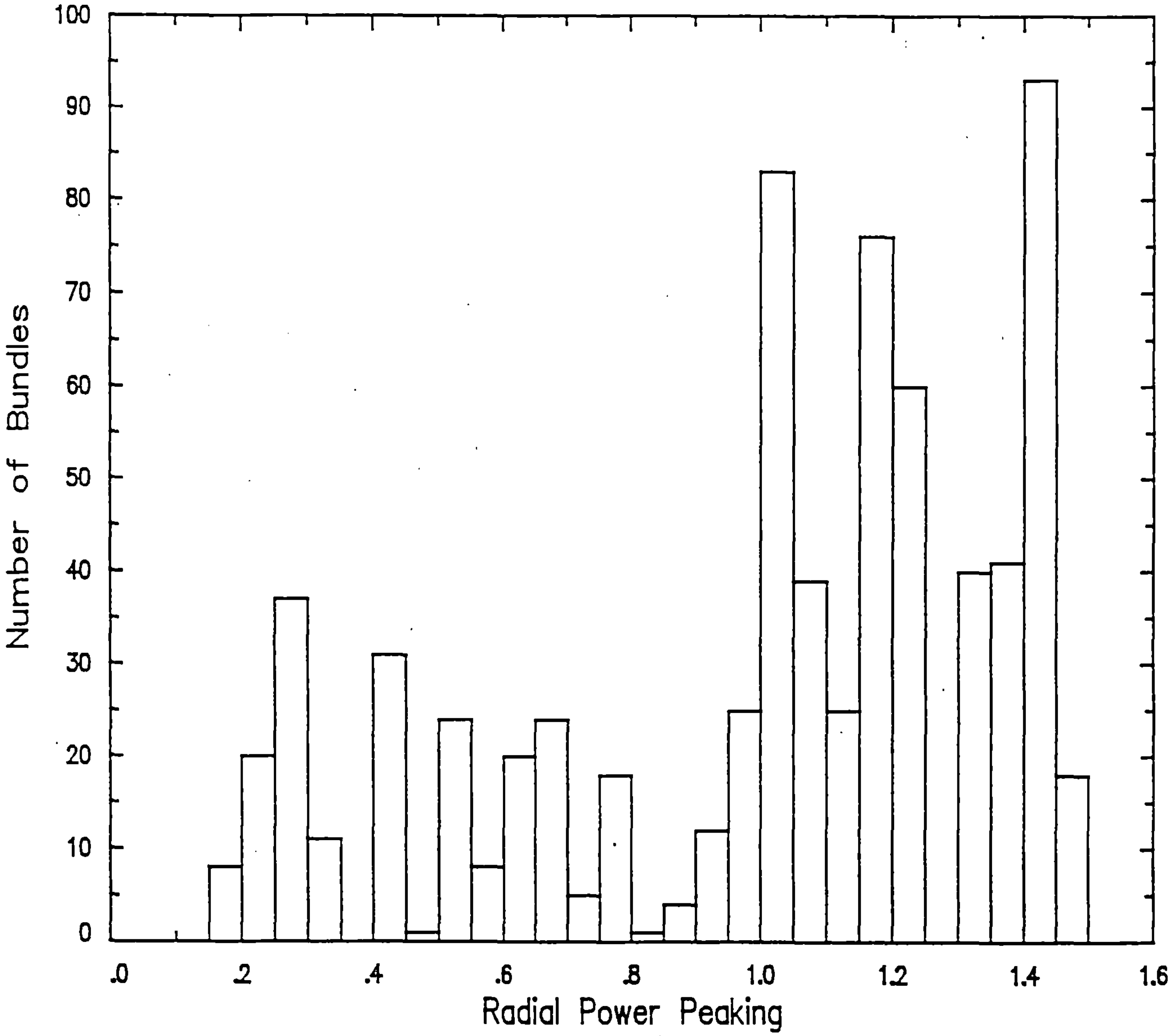


Figure 3.1

Design Basis Radial Power Distribution for SLMCPR Determination

Control Rod Corner

Control Rod Corner

0.997	0.995	1.036	1.005	1.087	1.073	1.078	1.005	1.063
0.995	0.974	1.026	1.000	0.878	1.068	1.057	0.866	1.051
1.036	1.026	1.001	1.031	1.041	1.102	1.023	1.004	1.082
1.005	1.000	1.031				0.869	0.973	1.033
1.087	0.878	1.041		Internal Water Channel		1.075	0.764	1.012
1.073	1.068	1.102				1.028	0.937	0.998
1.078	1.057	1.023	0.869	1.075	1.028	0.761	0.932	1.007
1.005	0.866	1.004	0.973	0.764	0.937	0.932	0.759	1.046
1.063	1.051	1.082	1.033	1.012	0.998	1.007	1.046	1.022

Maximum Local Power: 1.102

Figure 3.2

Design Basis Local Power Distribution for SPC-ND ATRIUM-9B Fuel (SPCA9-326B-11GZ-80M) Uncontrolled at 15,000 MWd/MTU and 70% Void for SLMCPR Determination

Control Rod Corner

Control Rod Corner

1.005	0.983	1.053	1.028	1.086	1.075	1.081	1.026	1.044
0.983	0.990	1.029	0.916	0.986	1.064	1.057	0.887	1.056
1.053	1.029	1.008	1.040	1.044	1.000	1.025	1.005	1.083
1.028	0.916	1.040	Internal Water Channel			0.890	0.969	1.034
1.086	0.986	1.044				1.070	0.955	1.010
1.075	1.064	1.000				1.030	0.773	1.008
1.081	1.057	1.025	0.890	1.070	1.030	0.803	0.936	1.016
1.026	0.887	1.005	0.969	0.955	0.773	0.936	0.803	0.956
1.044	1.056	1.083	1.034	1.010	1.008	1.016	0.956	0.961

Maximum Local Power: 1.086

Figure 3.3

Design Basis Local Power Distribution for SPC-ND ATRIUM-9B Fuel (SPCA9-339B-6GZ-80M) Uncontrolled at 17,500 MWd/MTU and 70% Void for SLMCPR Determination

Table 4.1

Neutronic Design Values

Number of Fuel Assemblies	724
Rated Thermal Power, MWt	2527
Rated Core Flow, Mlbm/hr	98.0
Core Inlet Subcooling, Btu/lbm	22.7[a]
Moderator Temperature, °F	546
Channel Thickness, inch	0.080
Channel Internal Face-to-Face Dimension, inch	5.278
Fuel Assembly Pitch, inch	6.0
Wide Water Gap Thickness, inch	0.750
Narrow Water Gap Thickness, inch	0.374

Control Rod Data[b]

Absorber Material	B_4C
Total Blade Span, inch	9.810
Total Blade Support Span, inch	1.580
Blade Thickness, inch	0.312
Absorber Rods Per Blade	84
Absorber Rod OD, inch	0.188
Absorber Rod ID, inch	0.138
Absorber Density, % of theoretical	70

(a) Based on actual operating experience.

(b) The control rod data represents original equipment control blades at Dresden which were modeled in the licensing analyses. Dresden FSAR Section 4.6.2.1 indicates that reactivity characteristics of replacement control blades closely match original equipment blades.

Control Rod Corner

Control Rod Corner

1 1.800	2 2.200	3 2.700	3 2.700	4 3.300	4 3.300	4 3.300	3 2.700	3 2.700
2 2.200	5 2.700	4 3.300	4 3.300	5 2.700	6 4.060	6 4.060	5 2.700	4 3.300
3 2.700	4 3.300	4 3.300	4 3.300	4 3.300	6 4.060	6 4.060	6 4.060	6 4.060
3 2.700	4 3.300	4 3.300		Internal		3 2.700	6 4.060	6 4.060
4 3.300	5 2.700	4 3.300		Water		6 4.060	5 2.700	6 4.060
4 3.300	6 4.060	6 4.060		Channel		6 4.060	6 4.060	6 4.060
4 3.300	6 4.060	6 4.060	3 2.700	6 4.060	6 4.060	5 2.700	6 4.060	6 4.060
3 2.700	5 2.700	6 4.060	6 4.060	5 2.700	6 4.060	6 4.060	5 2.700	6 4.060
3 2.700	4 3.300	6 4.060	6 4.060	6 4.060	6 4.060	6 4.060	6 4.060	4 3.300

1 Rods (1) -- 1.800 wt% U-235
2 Rods (2) -- 2.200 wt% U-235
3 Rods (10) -- 2.700 wt% U-235
4 Rods (18) -- 3.300 wt% U-235
5 Rods (9) -- 2.700 wt% U-235 + 3.50 wt% Gd_2O_3
6 Rods (32) -- 4.060 wt% U-235

Figure 4.1

Dresden Unit 3 Cycle 15
SPCA9-343L-9G3.5 Enrichment Distribution

Control Rod Corner

Control Rod Corner

1 1.800	2 2.200	3 2.700	3 2.700	4 3.300	4 3.300	4 3.300	3 2.700	3 2.700
2 2.200	5 2.700	4 3.300	4 3.300	5 2.700	6 4.060	6 4.060	5 2.700	4 3.300
3 2.700	4 3.300	4 3.300	4 3.300	4 3.300	6 4.060	6 4.060	6 4.060	6 4.060
3 2.700	4 3.300	4 3.300				5 2.700	6 4.060	6 4.060
4 3.300	5 2.700	4 3.300		Internal Water Channel		6 4.060	5 2.700	6 4.060
4 3.300	6 4.060	6 4.060				6 4.060	6 4.060	6 4.060
4 3.300	6 4.060	6 4.060	5 2.700	6 4.060	6 4.060	5 2.700	6 4.060	6 4.060
3 2.700	5 2.700	6 4.060	6 4.060	5 2.700	6 4.060	6 4.060	5 2.700	6 4.060
3 2.700	4 3.300	6 4.060	6 4.060	6 4.060	6 4.060	6 4.060	6 4.060	4 3.300

1 Rods (1) -- 1.800 wt% U-235
2 Rods (2) -- 2.200 wt% U-235
3 Rods (8) -- 2.700 wt% U-235
4 Rods (18) -- 3.300 wt% U-235
5 Rods (11) -- 2.700 wt% U-235 + 5.50 wt% Gd_2O_3
6 Rods (32) -- 4.060 wt% U-235

Figure 4.2

Dresden Unit 3 Cycle 15
SPCA9-344L-11G5.5 Enrichment Distribution

Control Rod Corner

Control Rod Corner

1 1.800	2 2.500	3 3.100	3 3.100	4 3.300	4 3.300	4 3.300	3 3.100	5 2.700
2 2.500	6 3.100	7 3.860	7 3.860	6 3.100	8 4.390	8 4.390	6 3.100	7 3.860
3 3.100	7 3.860	7 3.860	7 3.860	7 3.860	8 4.390	8 4.390	8 4.390	8 4.390
3 3.100	7 3.860	7 3.860				6 3.100	8 4.390	8 4.390
4 3.300	6 3.100	7 3.860		Internal Water Channel		8 4.390	6 3.100	8 4.390
4 3.300	8 4.390	8 4.390				8 4.390	8 4.390	8 4.390
4 3.300	8 4.390	8 4.390	6 3.100	8 4.390	8 4.390	6 3.100	8 4.390	8 4.390
3 3.100	6 3.100	8 4.390	8 4.390	6 3.100	8 4.390	8 4.390	6 3.100	8 4.390
5 2.700	7 3.860	8 4.390	8 4.390	8 4.390	8 4.390	8 4.390	8 4.390	7 3.860

1 Rods (1) -- 1.800 wt% U-235
2 Rods (2) -- 2.500 wt% U-235
3 Rods (6) -- 3.100 wt% U-235
4 Rods (6) -- 3.300 wt% U-235
5 Rods (2) -- 2.700 wt% U-235
6 Rods (11) -- 3.100 wt% U-235 + 5.50 wt% Gd_2O_3
7 Rods (12) -- 3.860 wt% U-235
8 Rods (32) -- 4.390 wt% U-235

Figure 4.3

Dresden Unit 3 Cycle 15
SPCA9-378L-11G5.5 Enrichment Distribution

Control Rod Corner

Control Rod Corner

1 1.800	2 2.500	3 3.100	3 3.100	4 3.300	4 3.300	4 3.300	3 3.100	5 2.700
2 2.500	6 3.100	7 3.860	7 3.860	6 3.100	8 4.390	8 4.390	6 3.100	7 3.860
3 3.100	7 3.860	7 3.860	7 3.860	7 3.860	8 4.390	8 4.390	8 4.390	8 4.390
3 3.100	7 3.860	7 3.860	Internal Water Channel			3 3.100	8 4.390	8 4.390
4 3.300	6 3.100	7 3.860				8 4.390	6 3.100	8 4.390
4 3.300	8 4.390	8 4.390				8 4.390	8 4.390	8 4.390
4 3.300	8 4.390	8 4.390	3 3.100	8 4.390	8 4.390	6 3.100	8 4.390	8 4.390
3 3.100	6 3.100	8 4.390	8 4.390	6 3.100	8 4.390	8 4.390	6 3.100	8 4.390
5 2.700	7 3.860	8 4.390	8 4.390	8 4.390	8 4.390	8 4.390	8 4.390	7 3.860

1	Rods (1)	--	1.800 wt% U-235
2	Rods (2)	--	2.500 wt% U-235
3	Rods (8)	--	3.100 wt% U-235
4	Rods (6)	--	3.300 wt% U-235
5	Rods (2)	--	2.700 wt% U-235
6	Rods (9)	--	3.100 wt% U-235 + 4.50 wt% Gd_2O_3
7	Rods (12)	--	3.860 wt% U-235
8	Rods (32)	--	4.390 wt% U-235

Figure 4.4

Dresden Unit 3 Cycle 15
SPCA9-378L-9G4.5 Enrichment Distribution

Control Rod Corner

Control Rod Corner

1 2.000	2 2.300	3 3.000	3 3.000	4 3.530	4 3.530	4 3.530	3 3.000	5 2.750
2 2.300	3 3.000	4 3.530	6 3.000	4 3.530	7 4.300	7 4.300	6 3.000	4 3.530
3 3.000	4 3.530	4 3.530	4 3.530	4 3.530	4 3.530	7 4.300	7 4.300	7 4.300
3 3.000	6 3.000	4 3.530	Internal Water Channel			3 3.000	7 4.300	7 4.300
4 3.530	4 3.530	4 3.530				7 4.300	7 4.300	7 4.300
4 3.530	7 4.300	4 3.530				7 4.300	6 3.000	7 4.300
4 3.530	7 4.300	7 4.300	3 3.000	7 4.300	7 4.300	3 3.000	7 4.300	7 4.300
3 3.000	6 3.000	7 4.300	7 4.300	7 4.300	6 3.000	7 4.300	3 3.000	4 3.530
5 2.750	4 3.530	7 4.300	7 4.300	7 4.300	7 4.300	7 4.300	4 3.530	3 3.000

1	Rods (1)	--	2.000 wt% U-235
2	Rods (2)	--	2.300 wt% U-235
3	Rods (12)	--	3.000 wt% U-235
4	Rods (21)	--	3.530 wt% U-235
5	Rods (2)	--	2.750 wt% U-235
6	Rods (6)	--	3.000 wt% U-235 + 3.50 wt% Gd_2O_3
7	Rods (28)	--	4.300 wt% U-235

Figure 4.5

Dresden Unit 3 Cycle 15
SPCA9-362L-6G3.5 Enrichment Distribution

Control Rod Corner (top)

Control Rod Corner (left)

1 2.000	2 2.300	3 3.000	3 3.000	4 3.530	4 3.530	4 3.530	3 3.000	5 2.750
2 2.300	3 3.000	4 3.530	6 3.000	4 3.530	7 4.300	7 4.300	6 3.000	4 3.530
3 3.000	4 3.530	4 3.530	4 3.530	4 3.530	4 3.530	7 4.300	7 4.300	7 4.300
3 3.000	6 3.000	4 3.530		Internal		3 3.000	7 4.300	7 4.300
4 3.530	4 3.530	4 3.530		Water		7 4.300	7 4.300	7 4.300
4 3.530	7 4.300	4 3.530		Channel		7 4.300	6 3.000	7 4.300
4 3.530	7 4.300	7 4.300	3 3.000	7 4.300	7 4.300	3 3.000	7 4.300	7 4.300
3 3.000	6 3.000	7 4.300	7 4.300	7 4.300	6 3.000	7 4.300	3 3.000	4 3.530
5 2.750	4 3.530	7 4.300	7 4.300	7 4.300	7 4.300	7 4.300	4 3.530	3 3.000

1 Rods (1) -- 2.000 wt% U-235
2 Rods (2) -- 2.300 wt% U-235
3 Rods (12) -- 3.000 wt% U-235
4 Rods (21) -- 3.530 wt% U-235
5 Rods (2) -- 2.750 wt% U-235
6 Rods (6) -- 3.000 wt% U-235 + 4.50 wt% Gd_2O_3
7 Rods (28) -- 4.300 wt% U-235

Figure 4.6

Dresden Unit 3 Cycle 15
SPCA9-362L-6G4.5 Enrichment Distribution

Control Rod Corner

Control Rod Corner

1 2.000	2 2.300	3 3.000	4 3.300	5 3.780	5 3.780	5 3.780	4 3.300	3 3.000
2 2.300	4 3.300	5 3.780	6 3.530	5 3.780	7 4.550	7 4.550	6 3.530	5 3.780
3 3.000	5 3.780	5 3.780	5 3.780	5 3.780	5 3.780	7 4.550	7 4.550	7 4.550
4 3.300	6 3.530	5 3.780				4 3.300	7 4.550	7 4.550
5 3.780	5 3.780	5 3.780		Internal Water Channel		8 4.300	7 4.550	7 4.550
5 3.780	7 4.550	5 3.780				7 4.550	6 3.530	7 4.550
5 3.780	7 4.550	7 4.550	4 3.300	8 4.300	7 4.550	9 3.530	7 4.550	7 4.550
4 3.300	6 3.530	7 4.550	7 4.550	7 4.550	6 3.530	7 4.550	9 3.530	5 3.780
3 3.000	5 3.780	7 4.550	7 4.550	7 4.550	7 4.550	7 4.550	5 3.780	4 3.300

1	Rods (1)	--	2.000 wt% U-235
2	Rods (2)	--	2.300 wt% U-235
3	Rods (4)	--	3.000 wt% U-235
4	Rods (8)	--	3.300 wt% U-235
5	Rods (21)	--	3.780 wt% U-235
6	Rods (6)	--	3.530 wt% U-235 + 5.50 wt% Gd_2O_3
7	Rods (26)	--	4.550 wt% U-235
8	Rods (2)	--	4.300 wt% U-235
9	Rods (2)	--	3.530 wt% U-235

Figure 4.7

Dresden Unit 3 Cycle 15
SPCA9-388L-6G5.5 Enrichment Distribution

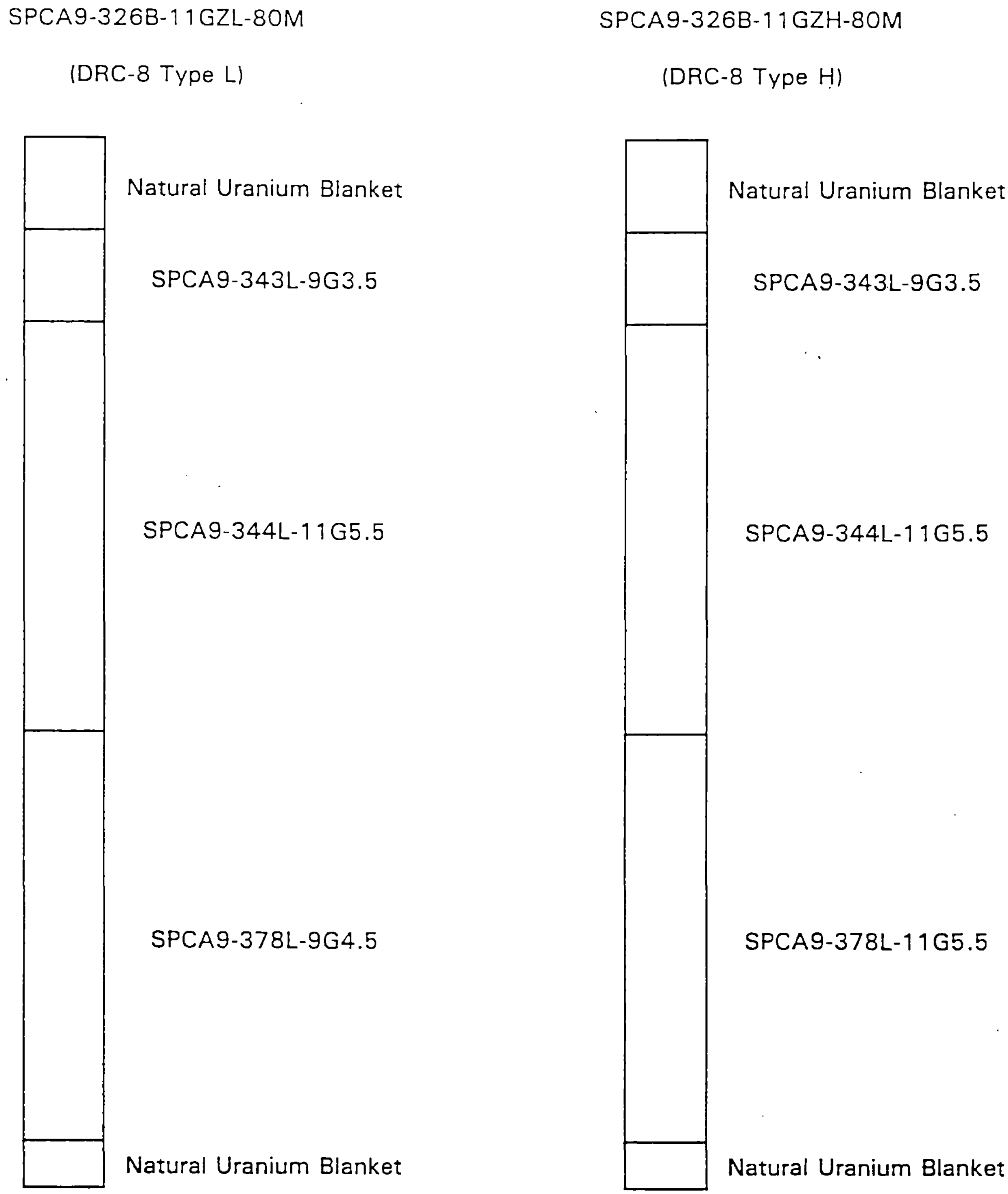


Figure 4.8

Dresden Unit 3 Reload Batch DRC-8
Axial Fuel Assembly Design

SPCA9-339B-6GZ-80M

(DRC-8A)

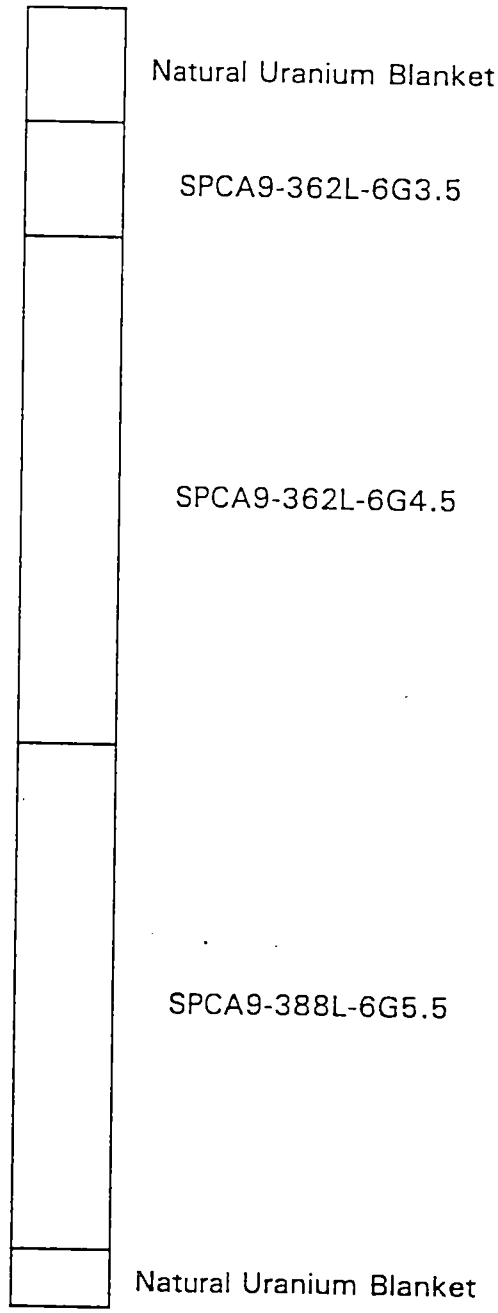


Figure 4.9

Dresden Unit 3 Reload Batch DRC-8A
Axial Fuel Assembly Design

E2	H0	F1	D2	D2	H0	G1	D2	D2	H0	F1	E2	C2	D2	A3
H0	G1	I0	F1	H0	F1	I0	G1	J0	G1	H0	J0	I0	D2	B3
F1	I0	D2	J0	G1	I0	F1	I0	F1	H0	G1	I0	D2	D2	A3
D2	F1	J0	D2	E2	G1	J0	D2	D2	D2	J0	F1	I0	D2	A3
D2	H0	G1	E2	D2	H0	G1	C2	E2	H0	C2	H0	C2	E2	A3
H0	F1	I0	G1	H0	E2	H0	G1	J0	G1	H0	I0	C2	A3	
G1	I0	F1	J0	G1	H0	E2	H0	G1	I0	F1	G1	A3		
D2	G1	I0	D2	C2	G1	H0	G1	E2	J0	G1	F1	B3		
D2	J0	F1	D2	E2	J0	G1	E2	G1	H0	D2	C2	A3		
H0	G1	H0	D2	H0	G1	I0	J0	H0	A3	A3	A3			
F1	H0	G1	J0	C2	H0	F1	G1	D2	A3	A3				
E2	J0	I0	F1	H0	I0	G1	F1	C2	A3					
C2	I0	D2	I0	C2	C2	A3	A3	A3						
D2	D2	D2	D2	E2	A3									
A3	B3	A3	A3	A3										

X Y

X = Fuel Type

Y = Cycles Irradiated

A	84	SPC 9x9-2	3.13 wt% U-235	ANF-5L
B	12	SPC 9x9-2	3.13 wt% U-235	ANF-5H
C	48	SPC 9x9-2	2.95 wt% U-235	ANF-6L
D	116	SPC 9x9-2	2.95 wt% U-235	ANF-6H
E	52	SPC 9x9-2	2.95 wt% U-235	ANF-6A
F	72	SPC 9x9-2B	3.13 wt% U-235	DRC-7L
G	108	SPC 9x9-2B	3.13 wt% U-235	DRC-7H
H	104	SPC ATRIUM-9B	3.26 wt% U-235	DRC-8L
I	72	SPC ATRIUM-9B	3.26 wt% U-235	DRC-8H
J	56	SPC ATRIUM-9B	3.39 wt% U-235	DRC-8A

Figure 4.10

Dresden Unit 3 Cycle 15
Reference Loading Map
(Quarter-Core Symmetric Loading)

< This data is to be furnished by ComEd. >

Figure 5.1

**Starting Control Rod Pattern for
Control Rod Withdrawal Analysis**

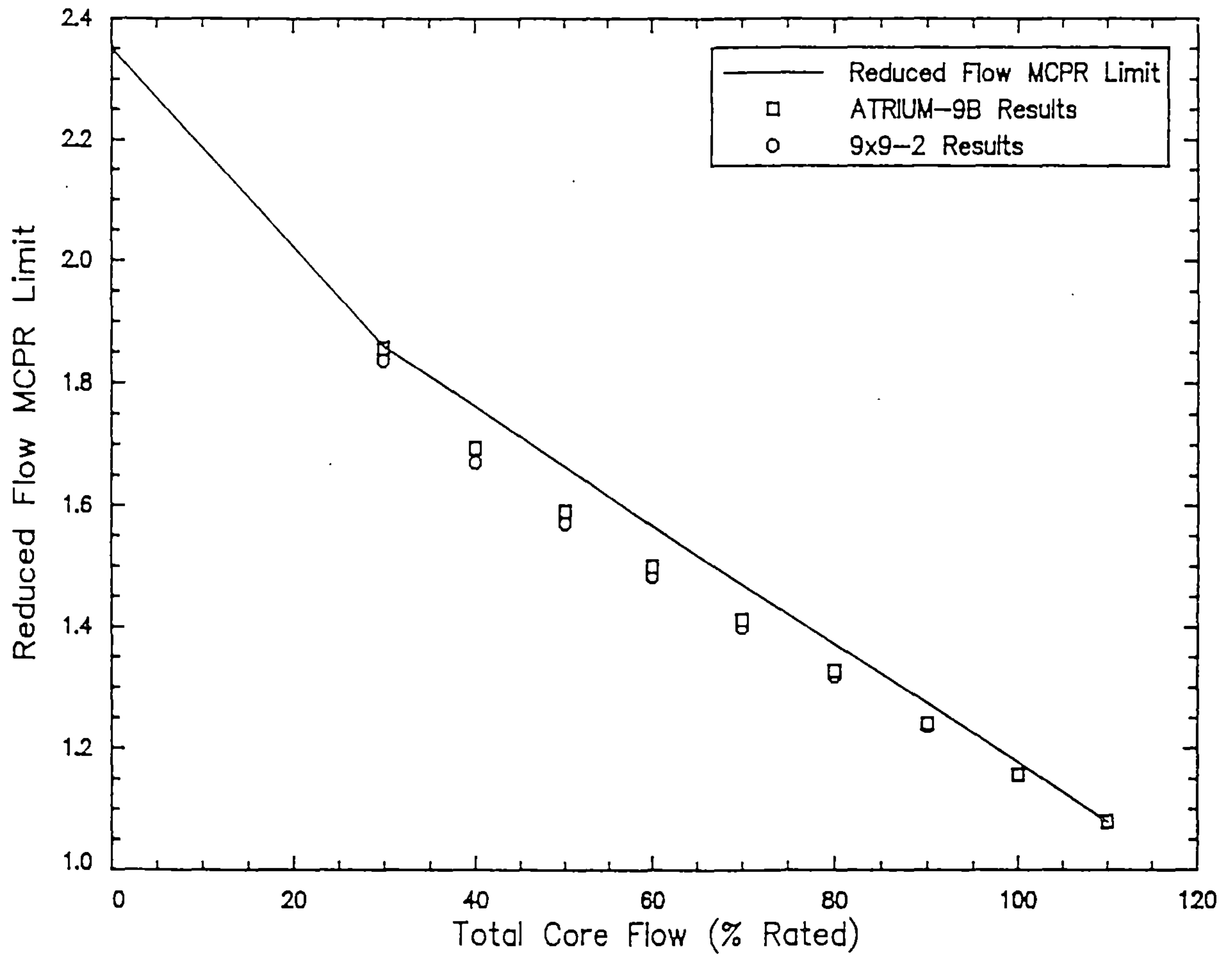


Figure 5.2

Reduced Flow MCPR Limit for
Manual Flow Control (SLMCPR = 1.08)
Applicable to ATRIUM-9B and 9x9-2 Fuel

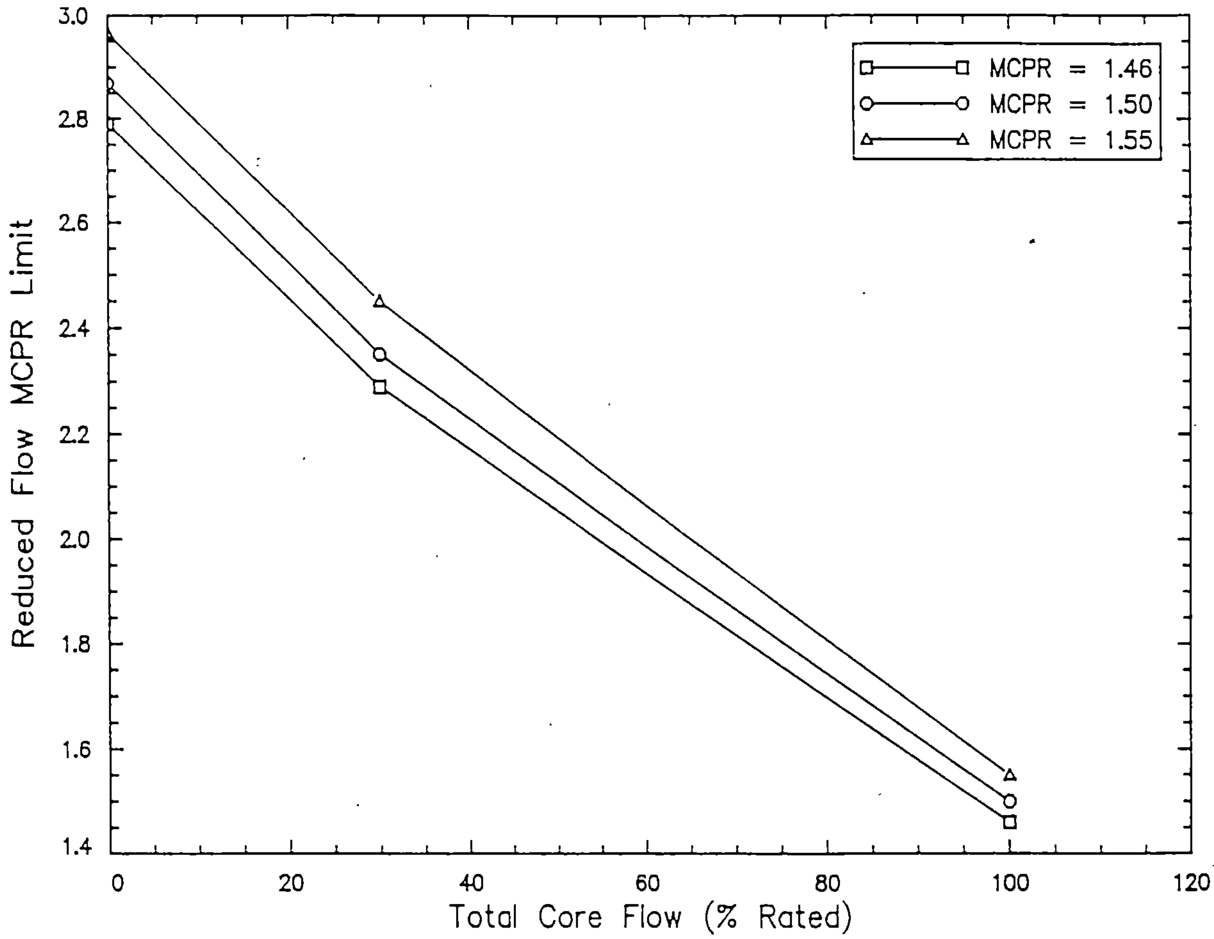


Figure 5.3

Reduced Flow MCPR Limit for
Automatic Flow Control
Applicable to ATRIUM-9B and 9x9-2 Fuel

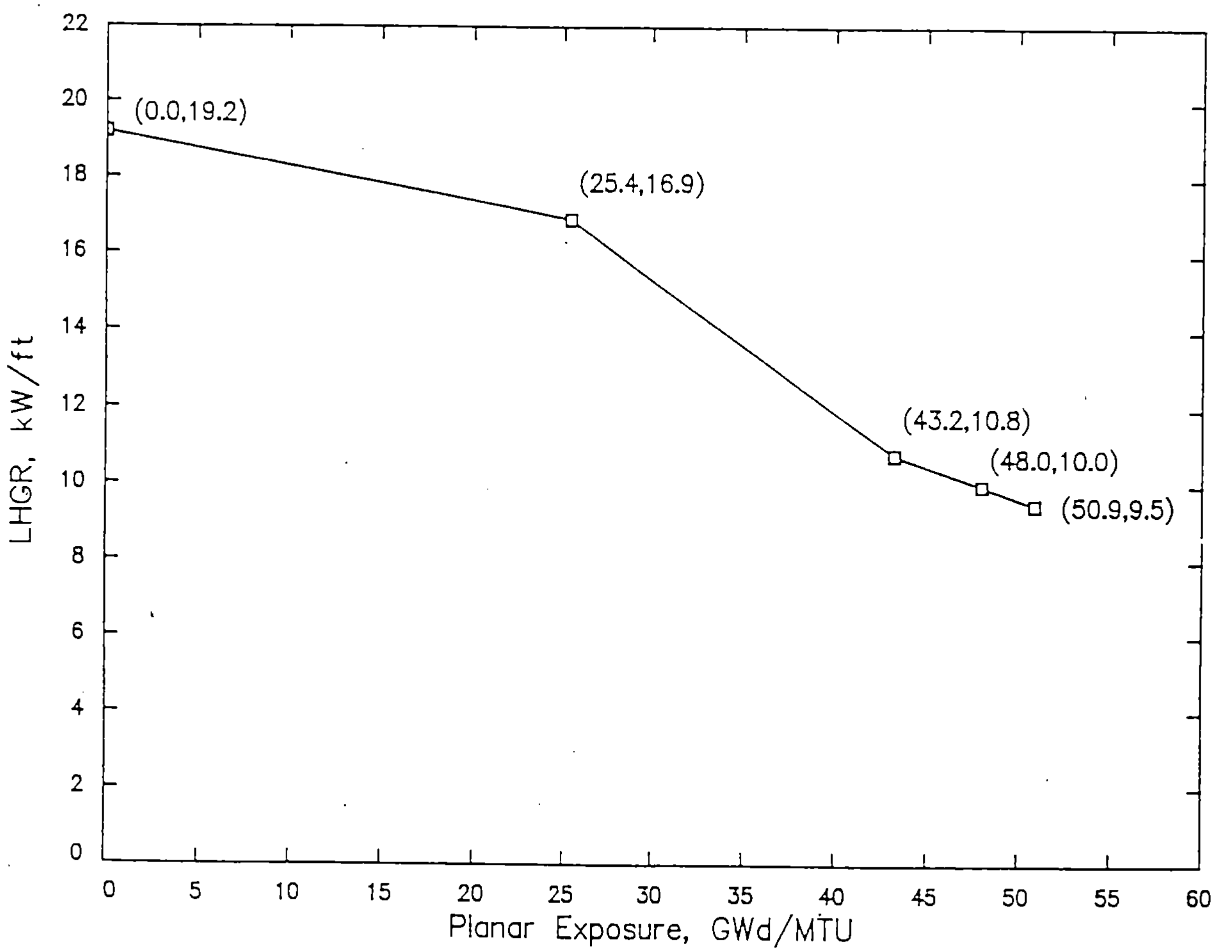


Figure 7.1

Protection Against Power Transient LHGR Limit for 9x9-2 Fuel

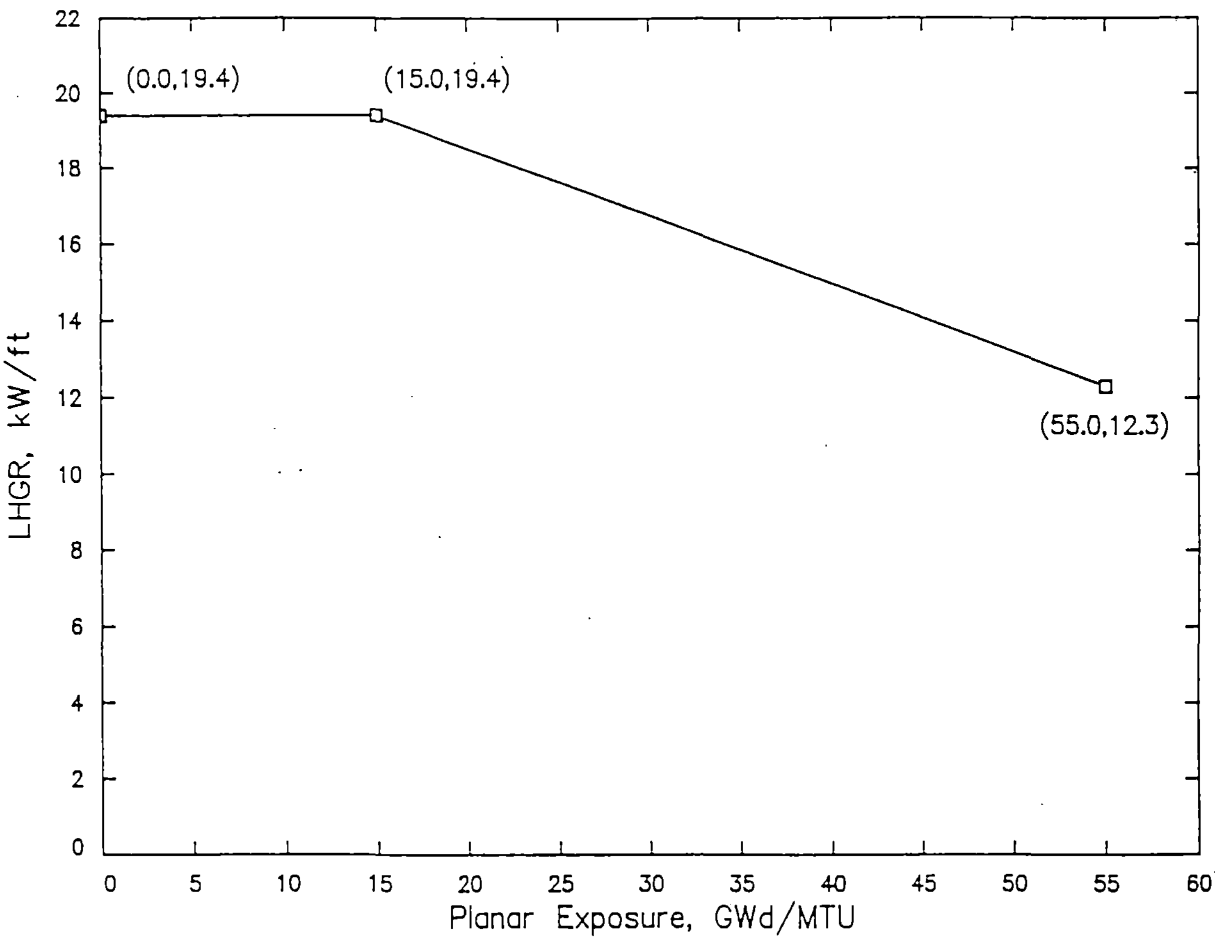


Figure 7.2

Protection Against Power Transient LHGR Limit for ATRIUM-9B Fuel

APPENDIX A

Single-Loop Operation

A.1 ANTICIPATED OPERATIONAL OCCURRENCES

Analyses have been provided which demonstrate the safety of plant operation with a single recirculation loop out of service for an extended period of time (Reference A.1). These analyses confirm that during single-loop operation, the plant cannot reach the normal bundle power levels and nodal power levels that are possible when both recirculation systems are in operation. The physical interdependence between core power and recirculation flow rate inherently limits the core to less than rated power. Because the SPC-ND fuel was designed to be compatible with the coresident fuel in thermal-hydraulic, nuclear, and mechanical design performance, and because the SPC-ND methodology has given results which are consistent with those of the previous analyses for normal two-loop operation, the analyses performed by the NSSS supplier for single-loop operation are also applicable to single-loop operation with fuel and analyses provided by SPC-ND.

A.2 MCPR SAFETY LIMIT

It is conservative to use the reduced flow two-loop operating MCPR limit or full flow MCPR operating limit plus 0.01 (whichever is greatest) for single-loop operations. These limits conservatively bound all transients from SLO conditions. The reduced flow MCPR limit is to protect against boiling transition during flow excursions to maximum two-pump flow; excursions to such high flows are not possible during single-loop one-pump operation. Thus, conservatively maintaining this two-loop limit assures that there is even more thermal margin under single-loop conditions than under two-loop full power/full flow conditions.

A.3 REDUCED FLOW OPERATION

It is conservative to use the reduced flow two-loop operating MCPR limit or full flow MCPR operating limit plus 0.01 (whichever is greatest) for single-loop operations.[a] This method is applied for operation up to end of full power and for coastdown. These limits conservatively bound all transients from SLO conditions. The reduced flow MCPR limit is to protect against boiling transition during flow excursions to maximum two-pump flow; excursions to such high flows are not possible during single-loop one-pump operation. Thus, conservatively maintaining this two-loop limit assures that there is even more thermal margin under single-loop conditions than under two-loop full power/full flow conditions.

A.4 SINGLE-LOOP PUMP SEIZURE ACCIDENT

Pump seizure is a postulated accident where the operating recirculation pump suddenly stops rotating. This causes a rapid decrease in core flow, a decrease in the rate at which heat can be transferred from the fuel rods, and a decrease in the critical power ratio. COTRANSA2, XCOBRA, and XCOBRA-T are used to calculate the MCPR for SPC-ND fuel during a pump seizure from single-loop operation.

COTRANSA2 was used to simulate system response to a pump seizure in single-loop operation. The core was assumed to be operating at the $MCPR_f$ limit at 58% core flow. The operating recirculation pump speed was reduced to zero causing a sudden decrease in active jet pump drive flow. During the event the inactive jet pump diffuser flow went from negative flow to positive flow.

Thermal-hydraulic analysis using SPC-ND safety limit methodology has shown that less than 10% of the rods in the core would experience boiling transition during the event. The Dresden FSAR indicates that a LOCA with a failure of 45% of the fuel rods in the core results in an off-site dose of less than 10% of 10CFR100 limits. Therefore, the two-loop manual flow control $MCPR_f$ limit below 58% flow provides the required protection such that any postulated fuel failures would not result in exceeding a small fraction of the 10CFR100 requirements.

---

[a] This conservative determination of MCPR limits remains applicable for the revised operating conditions analyzed in Reference A.3.

A.5 MAPLHGR LIMITS

SPC-ND performed LOCA analyses from single-loop conditions and determined an appropriate SLO MAPLHGR multiplier of 0.90 for 9x9-2 and ATRIUM-9B fuel.[a] The ECCS analysis results are presented in Reference A.2. All calculations were performed with the NRC-approved EXEM/BWR ECCS Evaluation Model according to Appendix K of 10CFR50.

A.6 REFERENCES

A.1 *Dresden Unit 3 Cycle 11 Plant Transient Analysis*, ANF-87-096, Advanced Nuclear Fuels Corporation, September 1987.

A.2 *Dresden LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM™-9B and 9x9-2 Fuel - Single-Loop Analysis*, EMF-98-007(P) Supplement 1, Siemens Power Corporation - Nuclear Division, January 1998.

A.3 *Dresden Unit 3 Cycle 15 Plant Transient Analysis With Increased Steam Flow*, EMF-97-047, Siemens Power Corporation - Nuclear Division, June 1998.

[a] The SLO MAPLHGR multiplier of 0.90 remains applicable for the revised operating conditions analyzed in Reference A.3.

Dresden Unit 3 Cycle 15
Reload Analysis

Distribution

D. E. Garber, 38 (15)

Notification List
(E-Mail Notification)

D. J. Braun
O. C. Brown
D. G. Carr
R. J. DeMartino
M. E. Garrett
J. L. Maryott
R. R. Schnepp
D. C. Serell
A. W. Will
J. A. White

Attachment 3

Dresden Unit 3 Cycle 15

Excerpts from Plant Transient Analysis Report

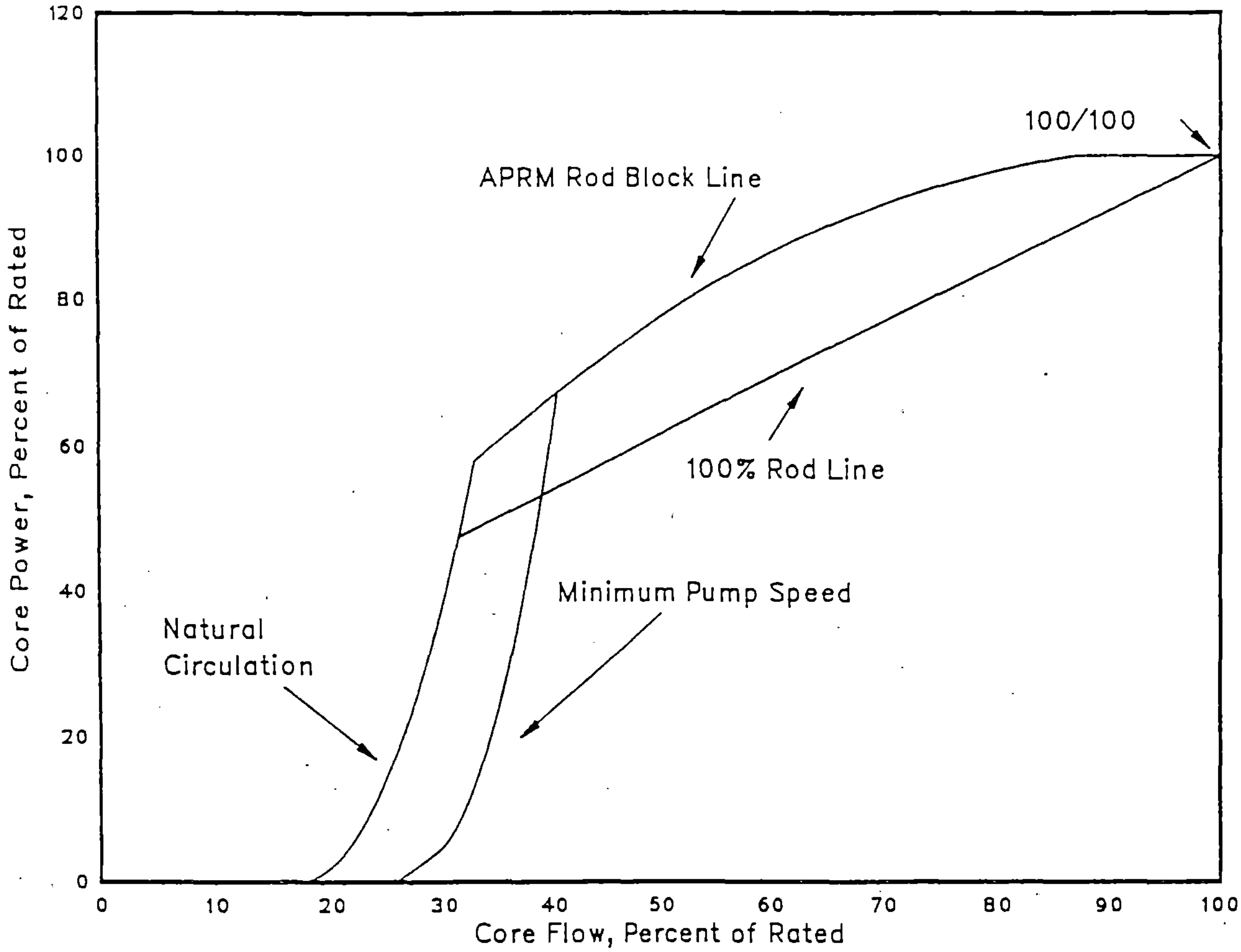


Figure 1.1

**Dresden Unit 3
Operating Power/Flow Map**

Table 2.1

Dresden Unit 3 Cycle 15
ΔCPRs at Rated Power With Increased Steam Flow

Transient	ΔCPR[a] 9x9-2	ΔCPR[a] ATRIUM-9B
Load Rejection No Bypass[b]		
100% Power / 100% Flow	0.35	0.36
100% Power / 87% Flow	0.33	0.36
Feedwater Flow Controller Failure[b]		
100% Power / 100% Flow	0.37	0.37
100% Power / 87% Flow	0.34	0.37
100% Power / 100% Flow - FHOOS[c]	0.37	0.38
100% Power / 87% Flow - FHOOS[c]	0.35	0.38
Loss of Feedwater Heating	[d]	[d]

[a] ΔCPRs presented are for second-cycle 9x9-2 fuel and first-cycle ATRIUM-9B fuel.

[b] ΔCPR based on Technical Specification scram performance.

[c] FHOOS - feedwater heaters out of service (100 °F reduction in feedwater temperature).

[d] Analysis of the LFWH is the responsibility of ComEd for Dresden Unit 3 Cycle 15.

Table 2.2

Dresden Unit 3 Cycle 15
Thermal Margin Summary With Increased Steam Flow

MCPR Operating Limit[a]

Transient	OLMCPR for 9x9-2/ATRIUM-9B	
	Up to EOFP	Coastdown
Feedwater Controller Failure (100%P / 100%F - FHOOS) (100%P / 87%F - FHOOS)	1.46	1.50[b]

Maximum Pressurization
(psig)

Transient	Steam Dome	Lower Plenum	Steam Lines
MSIV Closure Without Position Scram (ASME) (100%P / 100%F)	1318[c]	1344[c]	1318[c]

(a) Based on a plant technical specification two-loop MCPR safety limit of 1.08 and analysis of the limiting system transient analyzed in this report. The actual cycle operating limit may be higher if analyses within ComEd's scope of responsibility result in a ΔCPR higher than those in Table 2.1.

(b) Generic MCPR penalty of 0.04 is added to the MCPR operating limit to support coastdown operation beginning at EOFP (References 17 and 18). This penalty is not necessary if the station elects to monitor to the core thermal power limit in Figure 2.1 in Reference 17. If the 0.04 adder is applied, the core thermal power limit provided in Figure 2.2 in Reference 17 must be maintained.

(c) Generic pressure penalty of 5.0 psid was added to the results from the limiting end of full power case to support coastdown operation (Reference 17).

Table 2.3

Dresden Unit 3 Cycle 15[a]
Results of Plant Transient Analysis With Increased Steam Flow

Event	Maximum Neutron Flux (% of Rated)	Maximum Core Average Heat Flux (% of Rated)	Maximum Vessel[b]/ Dome Pressure (psig)
Load Rejection No Bypass (100%P / 100%F)	670	128	1298 / 1271
Load Rejection No Bypass (100%P / 87%F)	570	127	1299 / 1277
Feedwater Flow Controller Failure FHOOS (100%P / 100%F)	528	133	1166 / 1137
Feedwater Flow Controller Failure FHOOS (100%P / 87%F)	478	132	1163 / 1138
Feedwater Flow Controller Failure (100%P / 100%F)	620	132	1204 / 1173
Feedwater Flow Controller Failure (100%P / 87%F)	530	130	1200 / 1174
MSIV Closure ASME Analysis (100%P / 100%F)	326	128	1339 / 1313
MSIV Closure ASME Analysis (100%P / 87%F)	320	125	1339 / 1316

[a] Bounding state points.

[b] Lower plenum pressure.

Table 3.1

Dresden Unit 3
Design Reactor and Plant Conditions

Reactor Thermal Power	2527 MWt
Total Core Flow	98.0 Mlbm/hr
Core Active Flow	87.3 Mlbm/hr
Core Bypass Flow[a]	10.7 Mlbm/hr
Core Inlet Enthalpy	523.0 Btu/lbm
Vessel Pressures	
Steam Dome	1020 psia
Core Exit (upper plenum)	1030 psia
Lower Plenum	1053 psia
Turbine Pressure	964 psia
Feedwater/Steam Flow[b]	9.9 Mlbm/hr
Feedwater Enthalpy	321.1 Btu/lbm
Recirculating Pump Flow (per pump)	17.8 Mlbm/hr

[a] Includes water rod/channel flow.

[b] Feedwater flow is set equal to steam flow because control rod drive flow is not modeled separately in the analysis.

Table 3.2

Dresden Unit 3
Significant Parameter Values Used in Analysis

High Neutron Flux Trip	3032.4 MWt
Control Rod Insertion Time	3.5 sec / 90% inserted[a]
Time to Deenergize Pilot Scram Solenoid Valves	200 msec
Time to Sense Fast Turbine Control Valve Closure	80 msec[b]
Time From High Neutron Flux Trip To Control Rod Motion	290 msec[c]
Turbine Stop Valve Stroke Time	100 msec
Turbine Stop Valve Position Trip	90% open
Turbine Control Valve Stroke Time (total)	150 msec
Core Average Fuel/Cladding Gap[d] Conductance (cycle-specific value)	789.2 Btu/hr-ft^2-°F

(a) Includes a 0.2-second time delay to deenergize scram pilot valve solenoids.

(b) Includes a 50-msec delay for RPS logic transfer and a 30-msec delay until signal is received by RPS logic.

(c) Includes a 90-msec delay for signal to reach solenoid valves and a 200-msec delay for pilot scram solenoid valves to deenergize.

(d) Calculated by SPC for the Cycle 15 core using RODEX2.

Table 3.2

Dresden Unit 3
Significant Parameter Values Used in Analysis
(Continued)

Safety/Relief Valve Performance Settings[a]	
Target Rock Safety/Relief Valve (1 valve)	
Capacity Per Valve (relief)	155.0 lbm/sec at 1120 psig[b]
Capacity Per Valve (safety)	159.5 lbm/sec at 1112.4 psig[c]
Power Relief Valves Capacity (4 valves)[d]	
Capacity Per Valve	155.0 lbm/sec at 1120 psig
Safety Valves Capacity (8 valves)	
Capacity Per Valve	171.8 lbm/sec at 1277.2 psig
Target Rock Valve Delay/Stroke	967/200 msec[b]
Power Relief Valves Delay/Stroke	967/200 msec
MSIV Stroke Time	3.0 sec
MSIV Position Trip Set Point	90% open
Condenser Bypass Valve Performance	
Total Capacity	1085 lbm/sec
Delay to Opening (from demand)	150 msec
Opening Time (entire bank, maximum demand)	1.0 sec
Fraction of Energy Generated in Fuel	0.965[e]
Vessel Water Level (above separator skirt)	
Normal	30 in
Range of Operation (lower bound)	20 in
High Level Trip	60 in
Maximum Feedwater Runout Flow (2 pumps)	3311 lbm/sec
Recirculating Pump Trip Set Point	1250 psig Steam Dome Pressure

[a] Valve set points are given in Reference 16.

[b] The relief valve mode of the Target Rock SRV is conservatively modeled with Dresser RV flow capacity and set points.

[c] For ASME overpressurization event, Target Rock SRV safety function is not credited.

[d] One relief valve at the lowest set point is not credited.

[e] Reference 12.

Table 3.3

Control Characteristics[a]

Sensor Time Constants	
Pressure	100 msec
Steam Flow/Feedwater Flow	250 msec
Level	1 sec
Feedwater Control Mode	Single Element[b]
Feedwater 100% Mismatch	
Water Level Error	45 in
Pressure Regulator Settings	
Lead	1.0 sec
Lag	6.0 sec
Gain	3.33%/psid
Bypass Flow Signal Bias	3.0%
Combined Steam Flow Limiter Setting	1.05
Turbine Maximum Steam Flow	2816.67 lbm/sec
Recirculation Flow Control Mode	Manual

(a) The transients considered in cycle-specific analyses are mitigated by reactor scram which has a response that is faster than the feedwater control system response. The inclusion of the control system in the analysis model results in a more realistic calculated plant response. The representative parameters used in the analysis may not be bounding, but their effects on pressure and thermal margins are insignificant.

(b) Dresden Unit 3 plans to have a modification in the feedwater control system. Dresden licensing analyses are insensitive to the feedwater control system algorithms or settings. Single-element mode provides slightly more conservative results compared to manual or three-element control mode for all events (Reference 11).

Table 3.4

Dresden Unit 3 Cycle 15
Comparison of LRNB and TTNB Results With Increased Steam Flow

State Point	Maximum Neutron Flux (% of Rated)	Maximum Core Average Heat Flux (% of Rated)	Maximum[a] Vessel Pressure (psig)	ΔCPR[b]
100% Power / 100% Flow				
LRNB	669.7	128.4	1297.9	0.35 / 0.36
TTNB	664.8	128.4	1298.3	0.35 / 0.35

[a] Lower plenum pressure.

[b] Values for second-cycle 9x9-2/first-cycle ATRIUM-9B fuel. LRNB ΔCPRs are 0.0002 higher than corresponding TTNB ΔCPRs. The difference in the ATRIUM-9B results is due to rounding.

Table 3.5

Turbine Bypass Valve Degradation Study
ΔCPR Results With Increased Steam Flow

Bypass Valve Delay Time[a] msec	Second-Cycle 9x9-2 Δ(ΔCPR)[b]	First-Cycle ATRIUM-9B Δ(ΔCPR)[b]
50	0.000	0.000
100	0.010	0.010
150	0.017	0.019
250	0.026	0.029
350	0.031	0.033
450	0.035	0.037
550	0.039	0.041
700	0.042	0.044
No Bypass	0.042	0.044

(a) Delay is relative to time of TSV full closure (TSV closure takes 100 msec).

(b) Relative to ΔCPR for case with 50-msec delay (FWCF at 100% power/87% flow with FHOOS in Table 2.1).

Table 3.6

Input for MCPR Safety Limit Analysis

Fuel-Related Uncertainties[a]

Parameter	Source Document	Statistical Treatment
ANFB Correlation[b]	References 4, 15, 20, 21, and 25	Convoluted
Radial Peaking Factor[c]	References 13 and 21	Convoluted
Local Peaking Factor	Reference 5	Convoluted
Assembly Flow Rate	Reference 14	Convoluted
Channel Bow Local Peaking Factor	Reference 3	Convoluted

Plant Measurement Uncertainties

Parameter	Units	Value[d]	Uncertainty Percent	Statistical Treatment
Feedwater Flow Rate	Mlbm/hr	16.5[e]	2.306	Convoluted
Feedwater Temperature	°F	340.1[f]	2.36	Convoluted
Core Pressure	psia	1030	1.42	Convoluted
Total Core Flow	Mlbm/hr	98.0	2.50	Convoluted
Core Power	MWt	4260[e]		Allowed to vary with heat balance

(a) Fuel related uncertainties are proprietary and can be found in the indicated references.

(b) Additive constant uncertainty values are used.

(c) Radial peaking factor uncertainty includes allowances for up to 40% (equivalent of 2 TIP machines) of the TIP machines out of service (with POWERPLEX®-II CMSS SUBTIP methodology), LPRM recalibration interval up to 2000 EFPH, and LPRM failures up to 50% with POWERPLEX®-II CMSS bypass methodology on or off.

(d) Values are from analysis to support a two-loop MCPR safety limit of 1.08 provided in Reference 25. The Cycle 15 SLMCPR is 1.08 from Technical Specifications.

(e) Feedwater flow rate and core power were increased above design values to attain desired core MCPR for safety limit evaluation, consistent with Reference 3 methodology.

(f) Disposition of increased feedwater temperature associated with increased steam flow is presented in Reference 26.

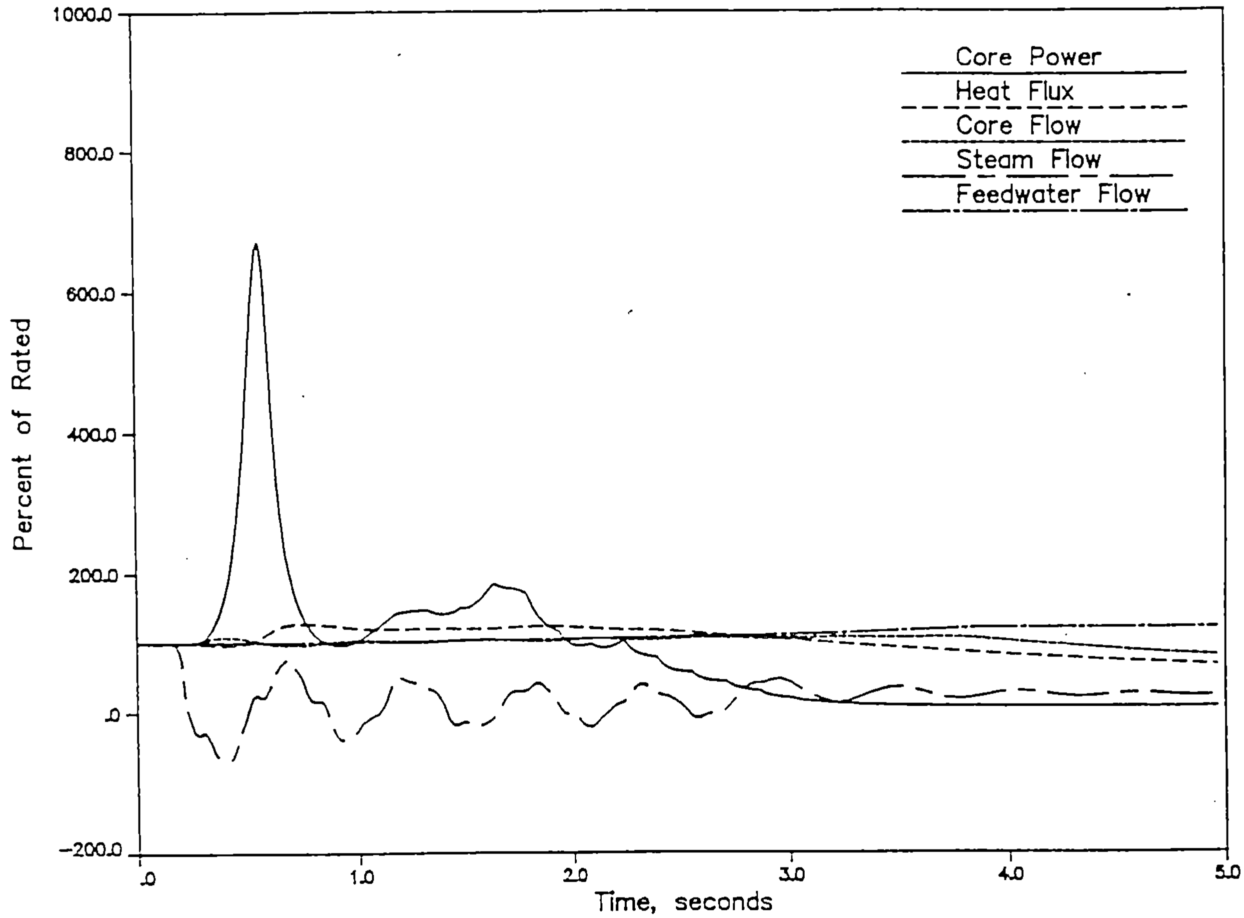


Figure 3.1

Load Rejection No Bypass at 100/100 -
Key Parameters

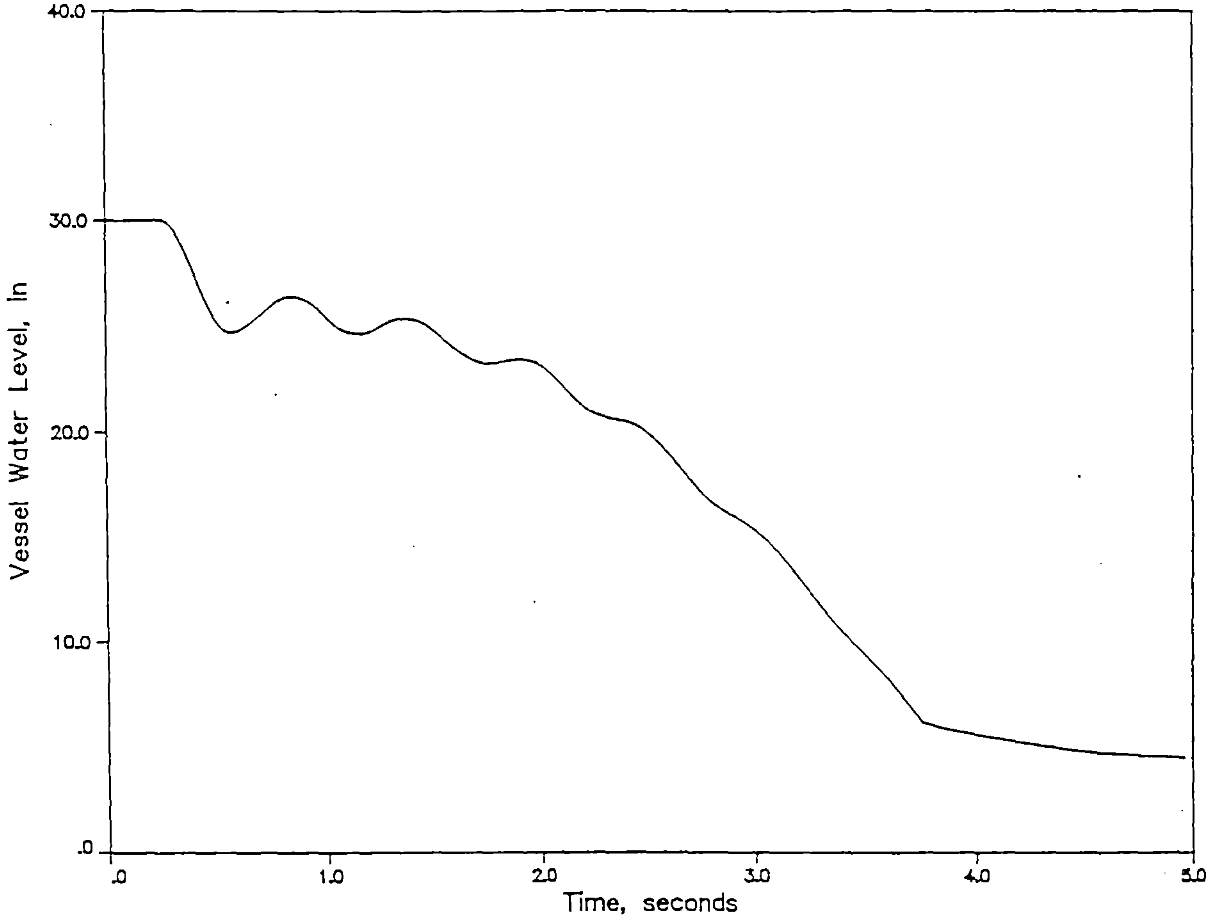


Figure 3.2

Load Rejection No Bypass at 100/100 -
Vessel Water Level
(Referenced to Instrument Zero)

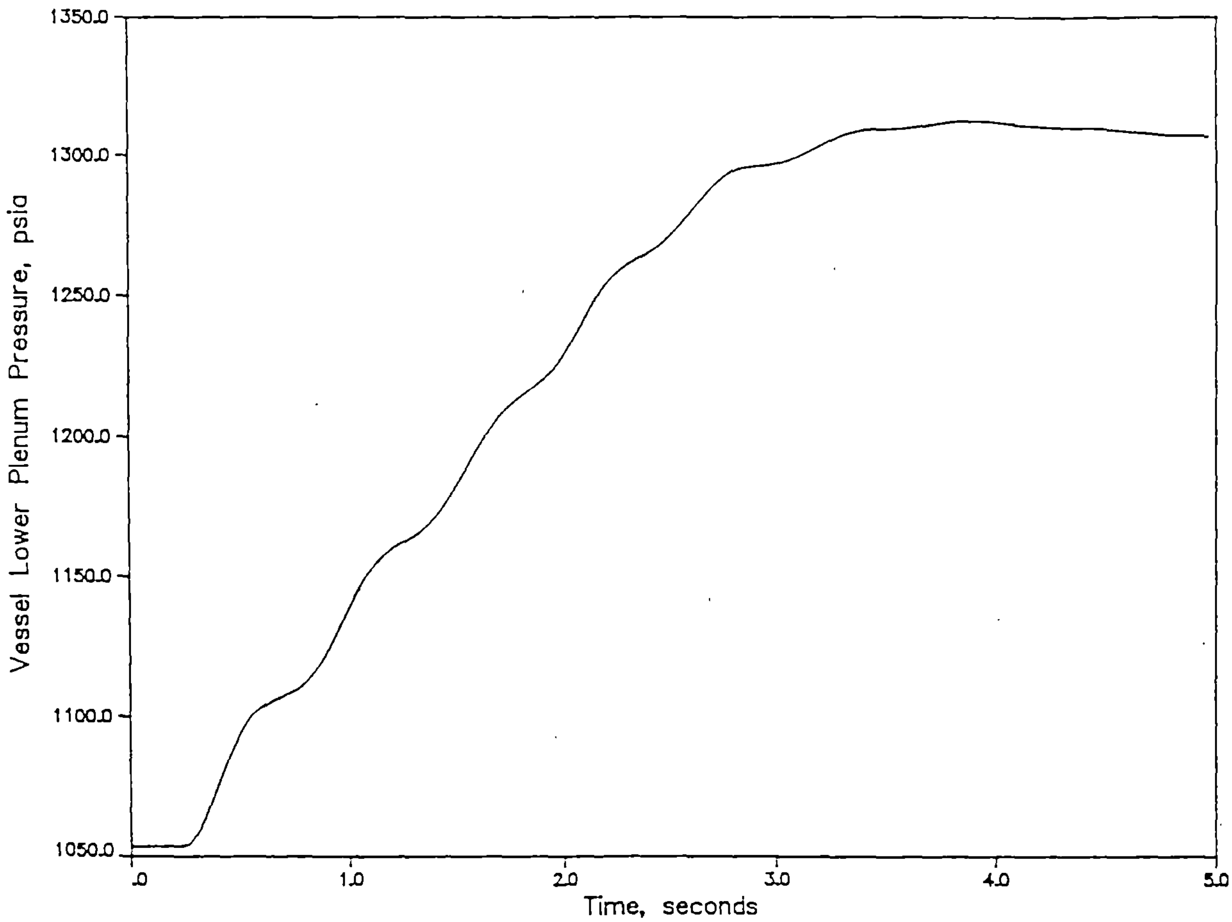


Figure 3.3

Load Rejection No Bypass at 100/100 -
Vessel Pressure Response

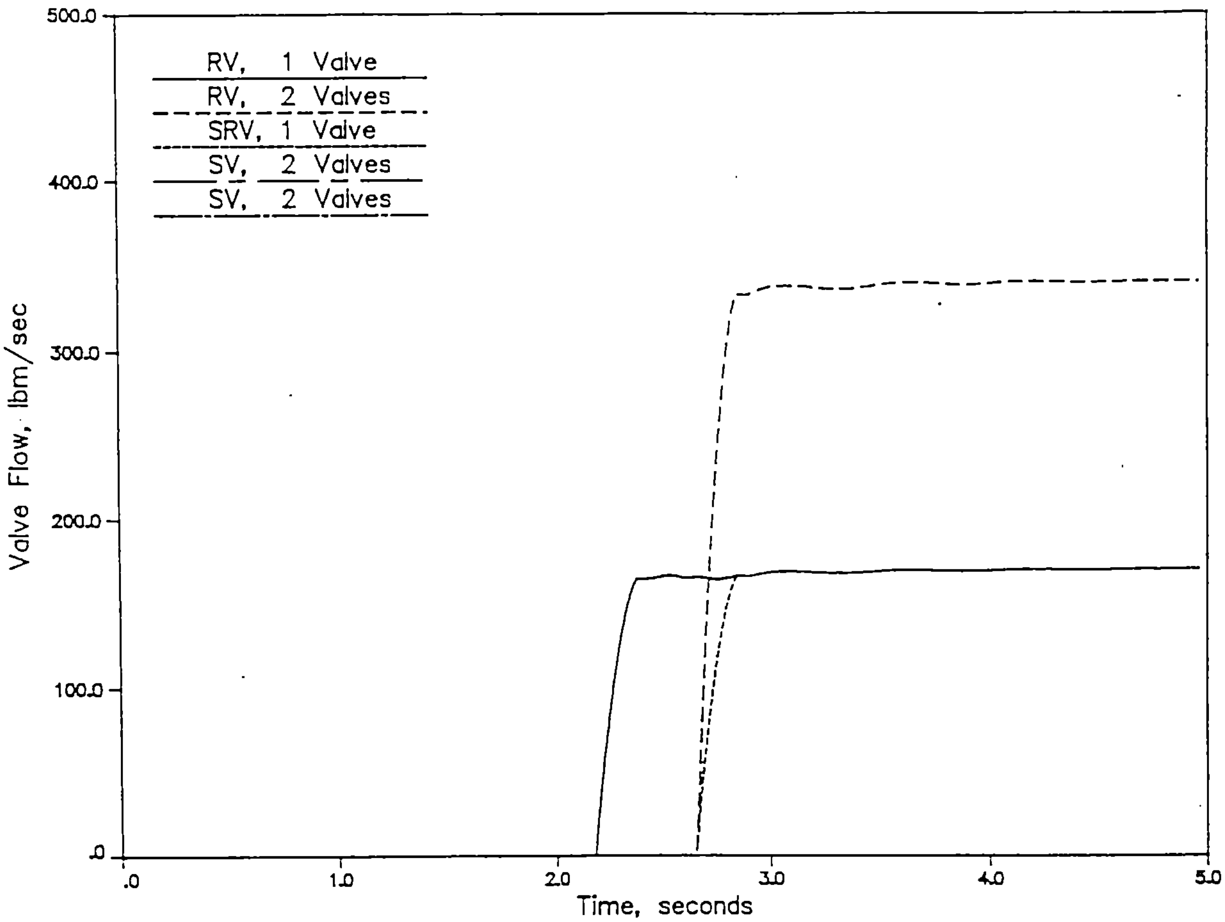
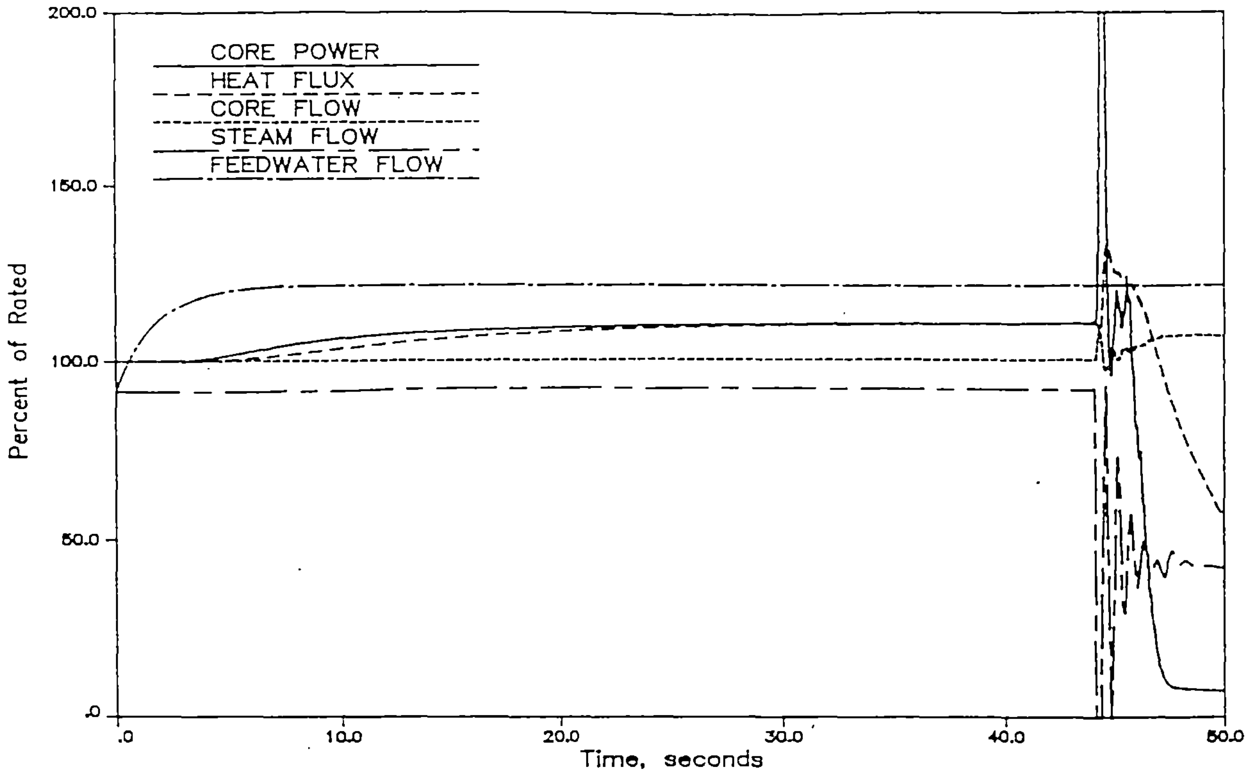


Figure 3.4

Load Rejection No Bypass at 100/100 -
Safety/Relief Valve Flows



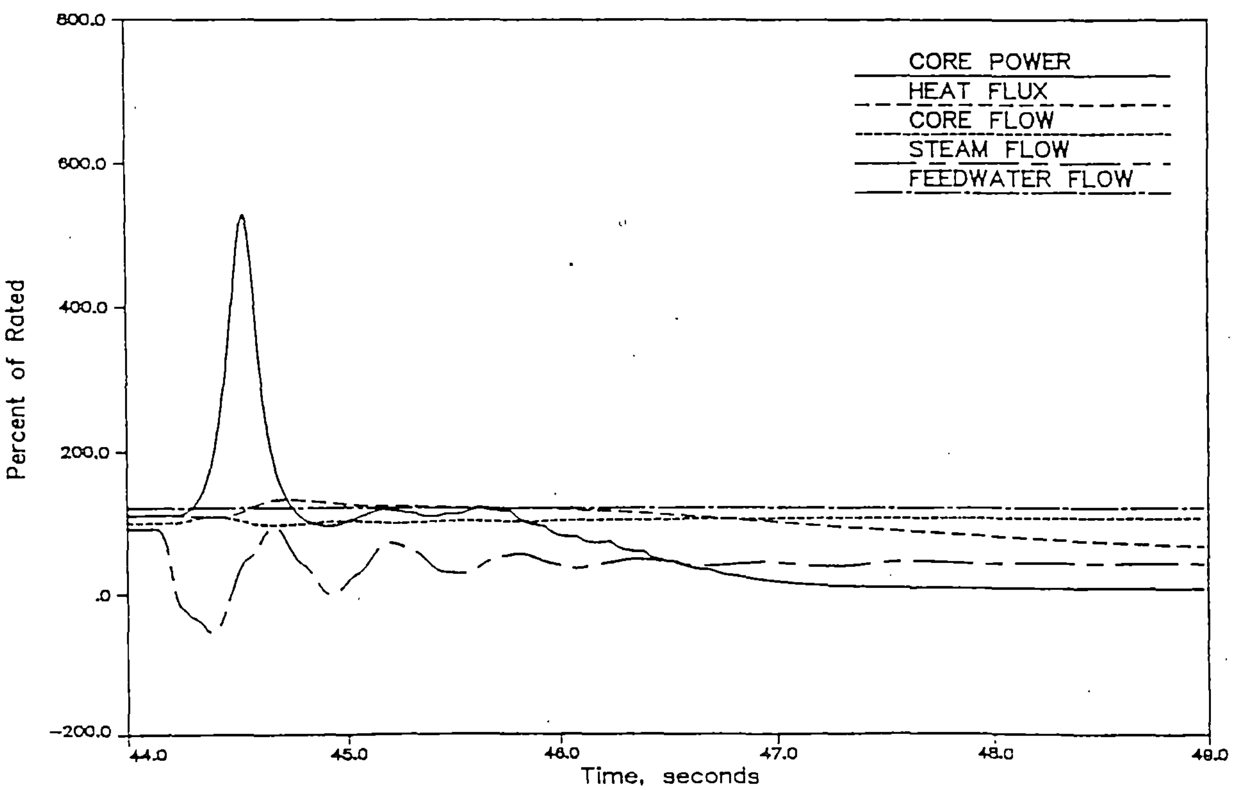
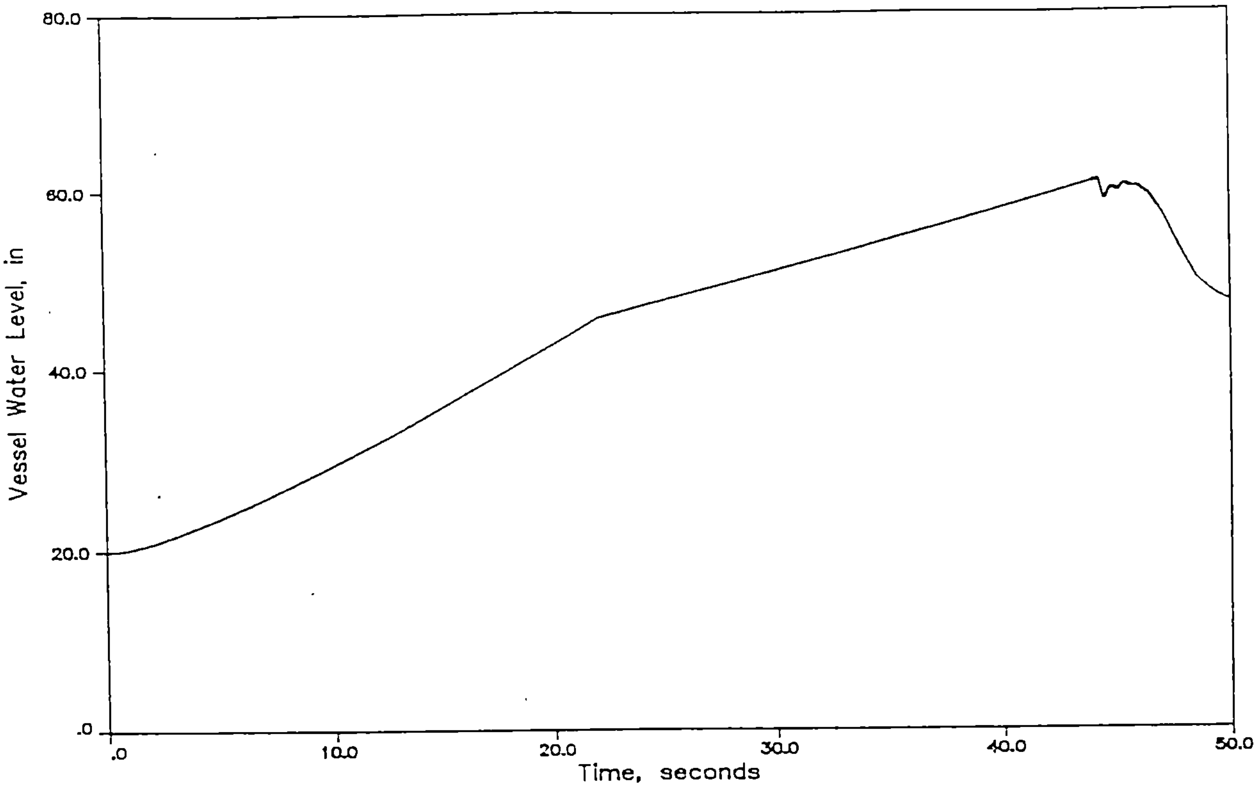


Figure 3.5

Feedwater Controller Failure at 100/100 FHOOS -
Key Parameters



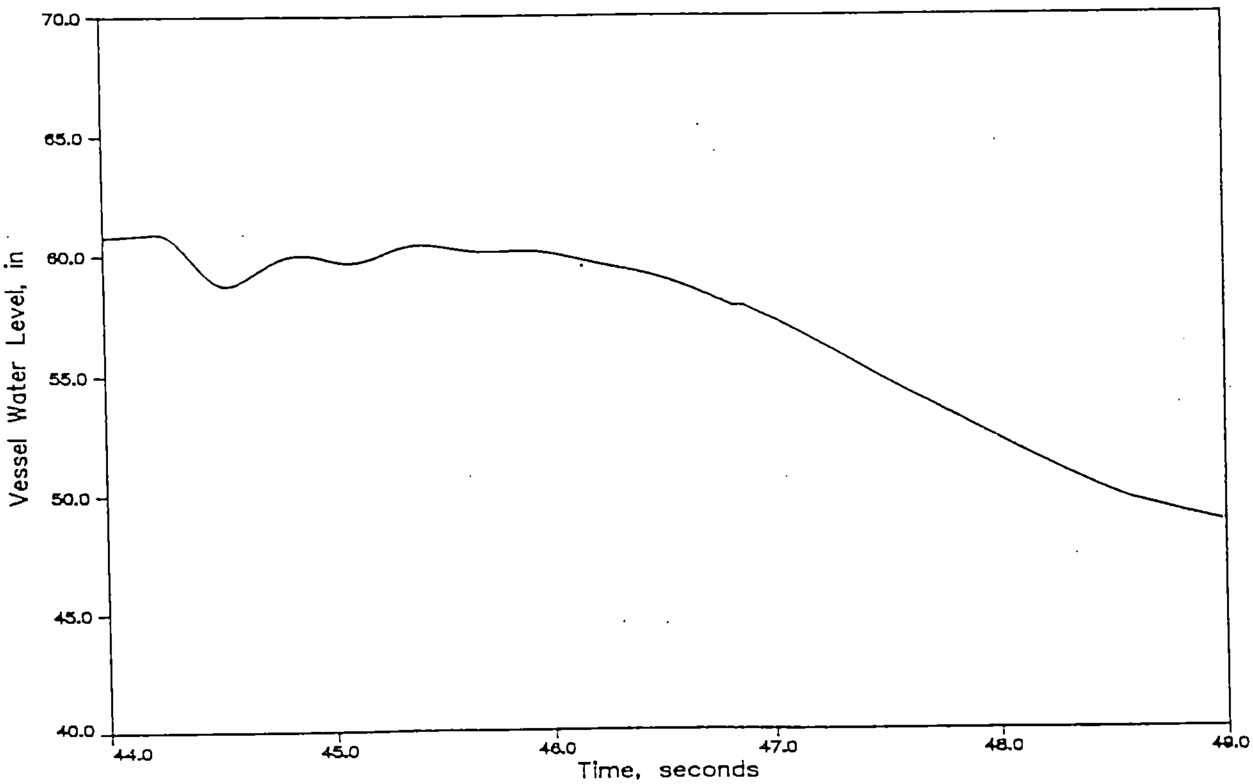
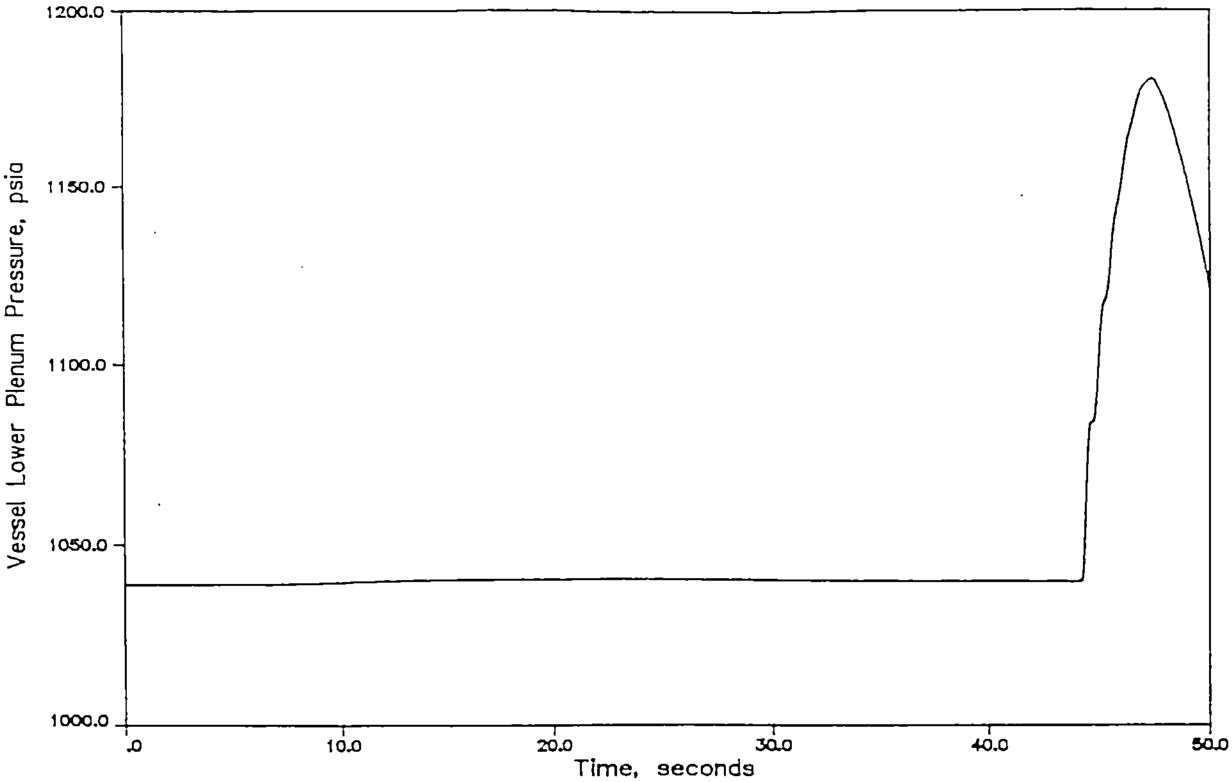


Figure 3.6

Feedwater Controller Failure at 100/100 FHOOS -
Vessel Water Level
Referenced to Instrument Zero



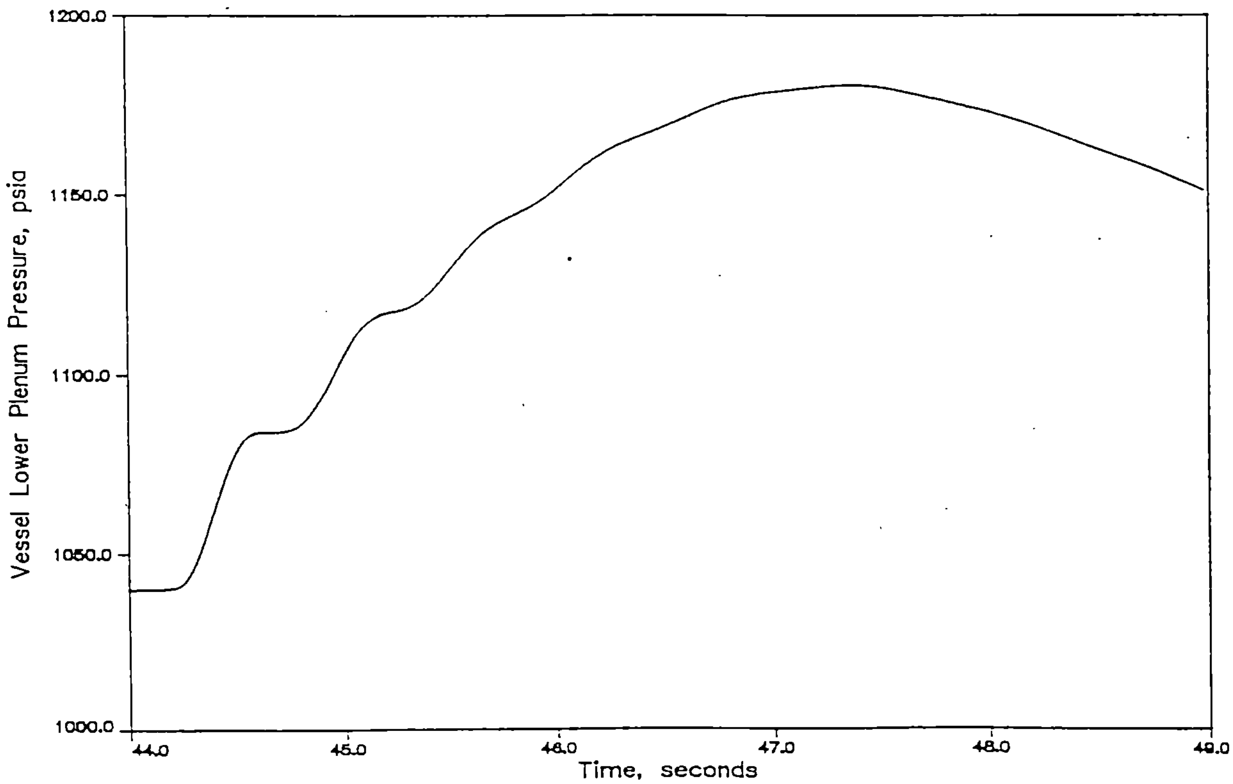
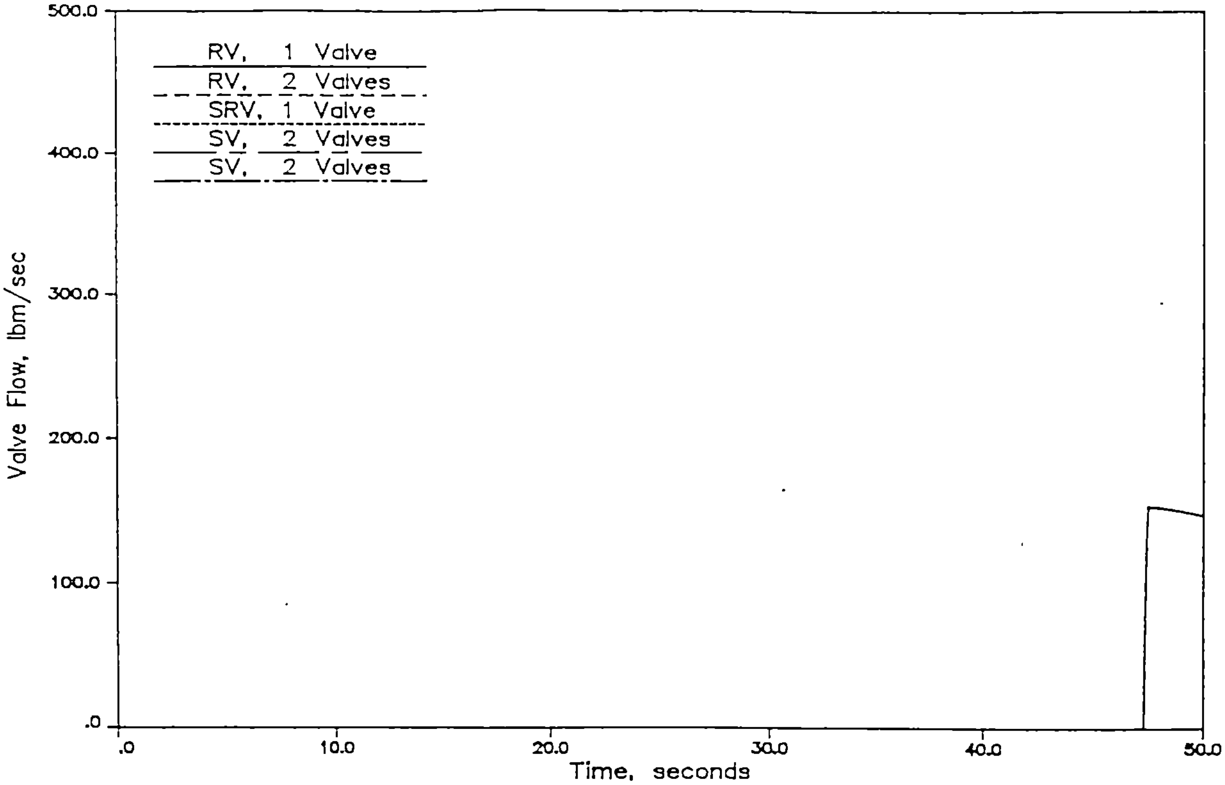


Figure 3.7

Feedwater Controller Failure at 100/100 FHOOS -
Vessel Pressure Response



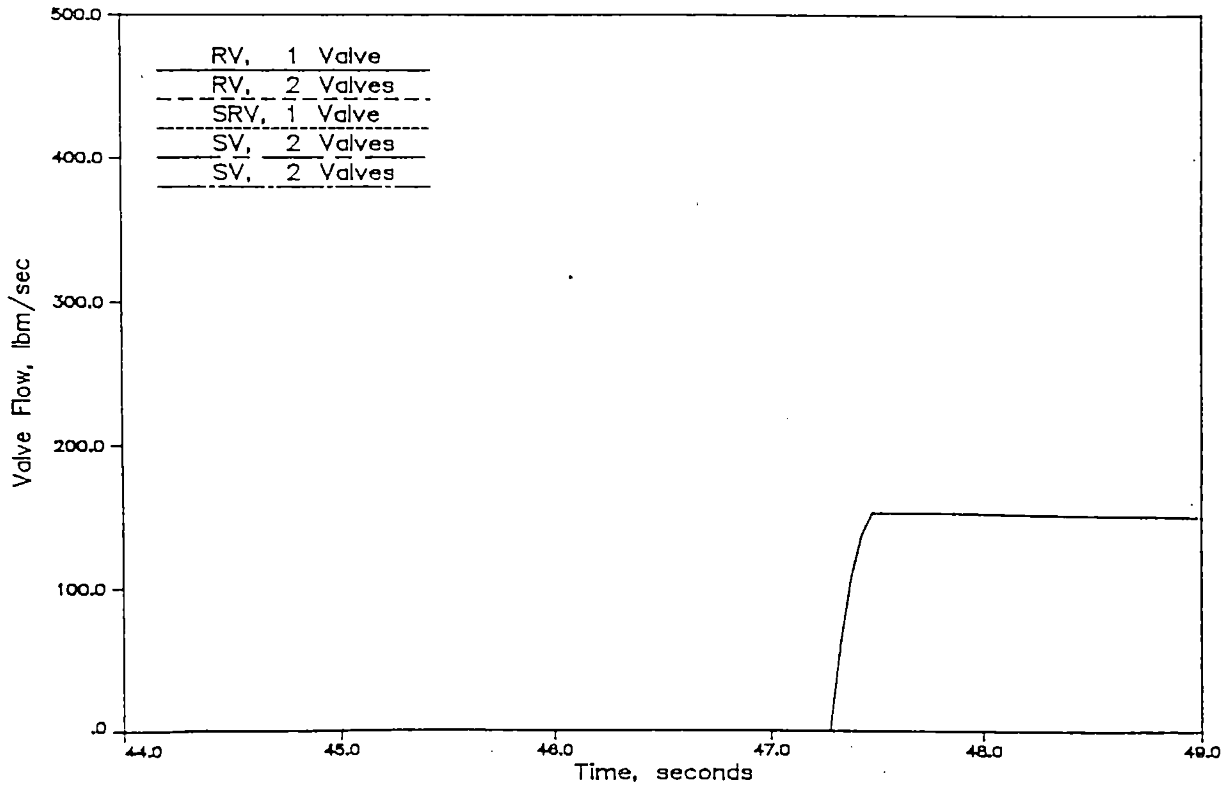


Figure 3.8

Feedwater Controller Failure at 100/100 FHOOS - Safety/Relief Valve Flows

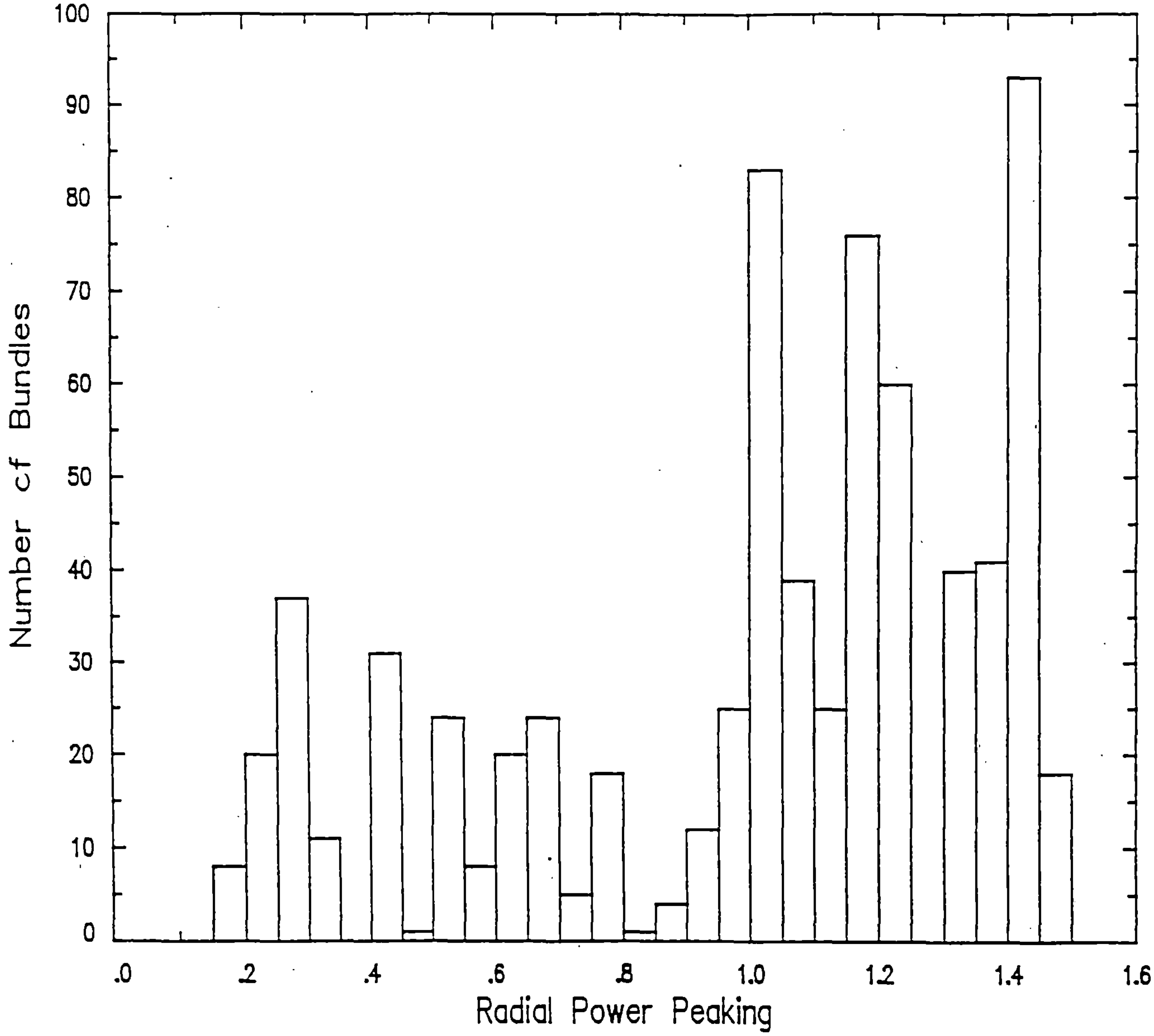


Figure 3.9

Design Basis Radial Power Distribution
for SLMCPR Determination

Control Rod Corner

Control Rod Corner

0.997	0.995	1.036	1.005	1.087	1.073	1.078	1.005	1.063
0.995	0.974	1.026	1.000	0.878	1.068	1.057	0.866	1.051
1.036	1.026	1.001	1.031	1.041	1.102	1.023	1.004	1.082
1.005	1.000	1.031		Internal		0.869	0.973	1.033
1.087	0.878	1.041		Water		1.075	0.764	1.012
1.073	1.068	1.102		Channel		1.028	0.937	0.998
1.078	1.057	1.023	0.869	1.075	1.028	0.761	0.932	1.007
1.005	0.866	1.004	0.973	0.764	0.937	0.932	0.759	1.046
1.063	1.051	1.082	1.033	1.012	0.998	1.007	1.046	1.022

Maximum Local Power: 1.102

Figure 3.10

Design Basis Local Power Distribution for
SPC-ND ATRIUM-9B Fuel (SPCA9-326B-11GZ-80M)
Uncontrolled at 15,000 MWd/MTU and 70% Void
for SLMCPR Determination

Control Rod Corner

Control Rod Corner

1.005	0.983	1.053	1.028	1.086	1.075	1.081	1.026	1.044
0.983	0.990	1.029	0.916	0.986	1.064	1.057	0.887	1.056
1.053	1.029	1.008	1.040	1.044	1.000	1.025	1.005	1.083
1.028	0.916	1.040		Internal		0.890	0.969	1.034
1.086	0.986	1.044		Water		1.070	0.955	1.010
1.075	1.064	1.000		Channel		1.030	0.773	1.008
1.081	1.057	1.025	0.890	1.070	1.030	0.803	0.936	1.016
1.026	0.887	1.005	0.969	0.955	0.773	0.936	0.803	0.956
1.044	1.056	1.083	1.034	1.010	1.008	1.016	0.956	0.961

Maximum Local Power: 1.086

Figure 3.11

Design Basis Local Power Distribution for
SPC-ND ATRIUM-9B Fuel (SPCA9-339B-6GZ-80M)
Uncontrolled at 17,500 MWd/MTU and 70% Void
for SLMCPR Determination

Table 4.1

Dresden Unit 3 Cycle 15
Results Summary of ASME Overpressurization Analyses
With Increased Steam Flow

Transient	Maximum Pressurization (psig)	
	Steam Dome	Lower Plenum
MSIV Closure		
(100%P / 100%F)	1312.9	1338.7
(100%P / 87%F)	1315.6	1338.7
TSV Closure		
(100%P / 87%F)	1315.9	1338.5
TCV Closure		
(100%P / 87%F)	1315.9	1338.5

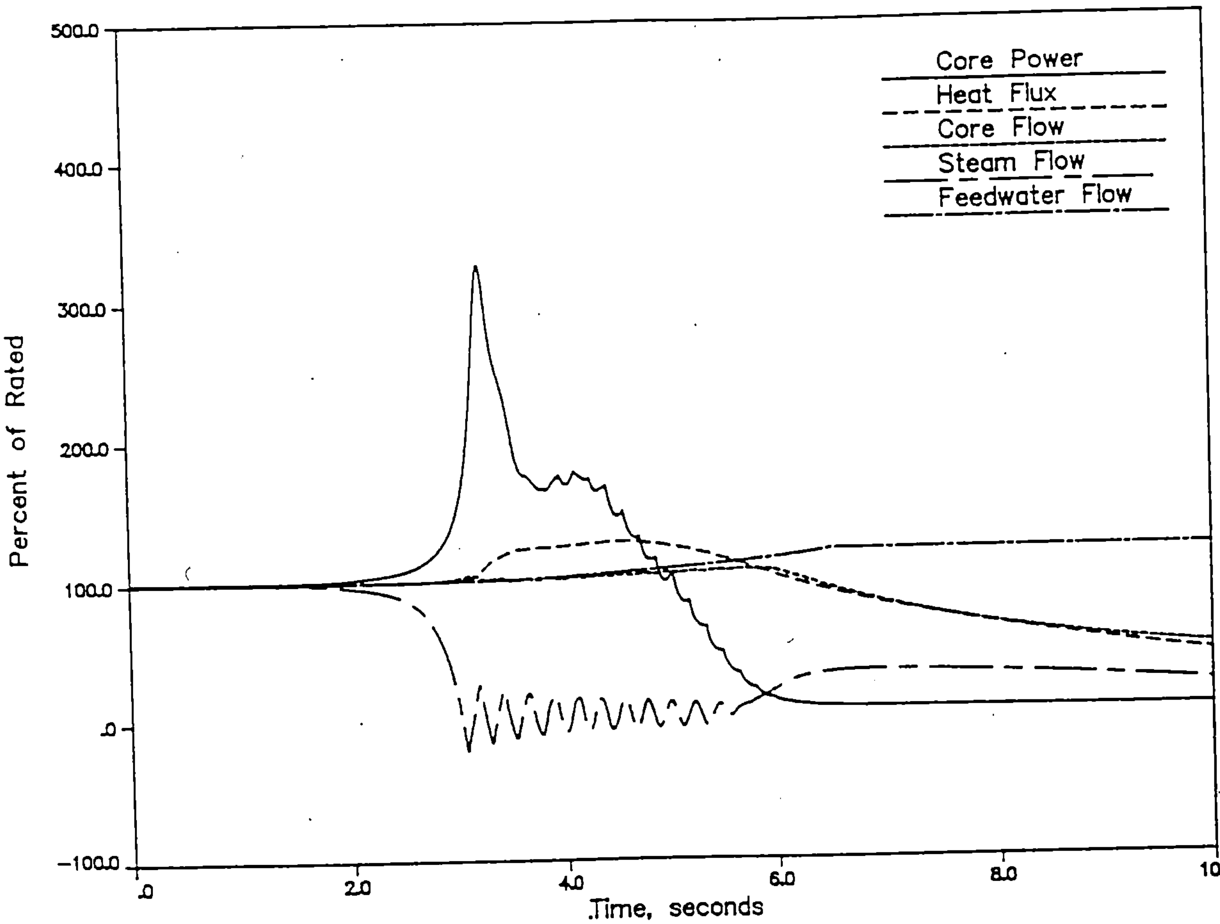


Figure 4.1

MSIV Closure at 100/100 -
Key Parameters

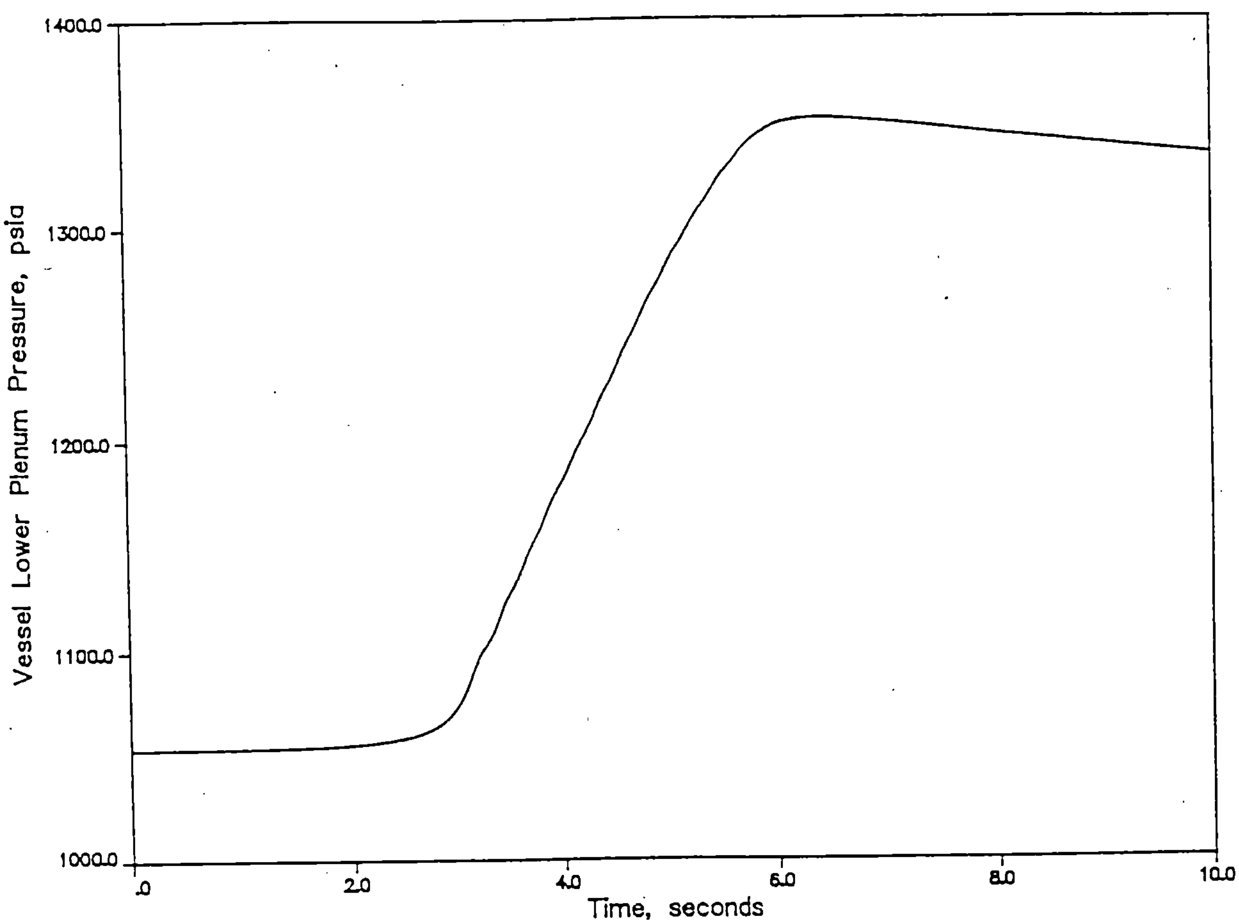


Figure 4.2

**MSIV Closure at 100/100 -
Vessel Pressure Response**

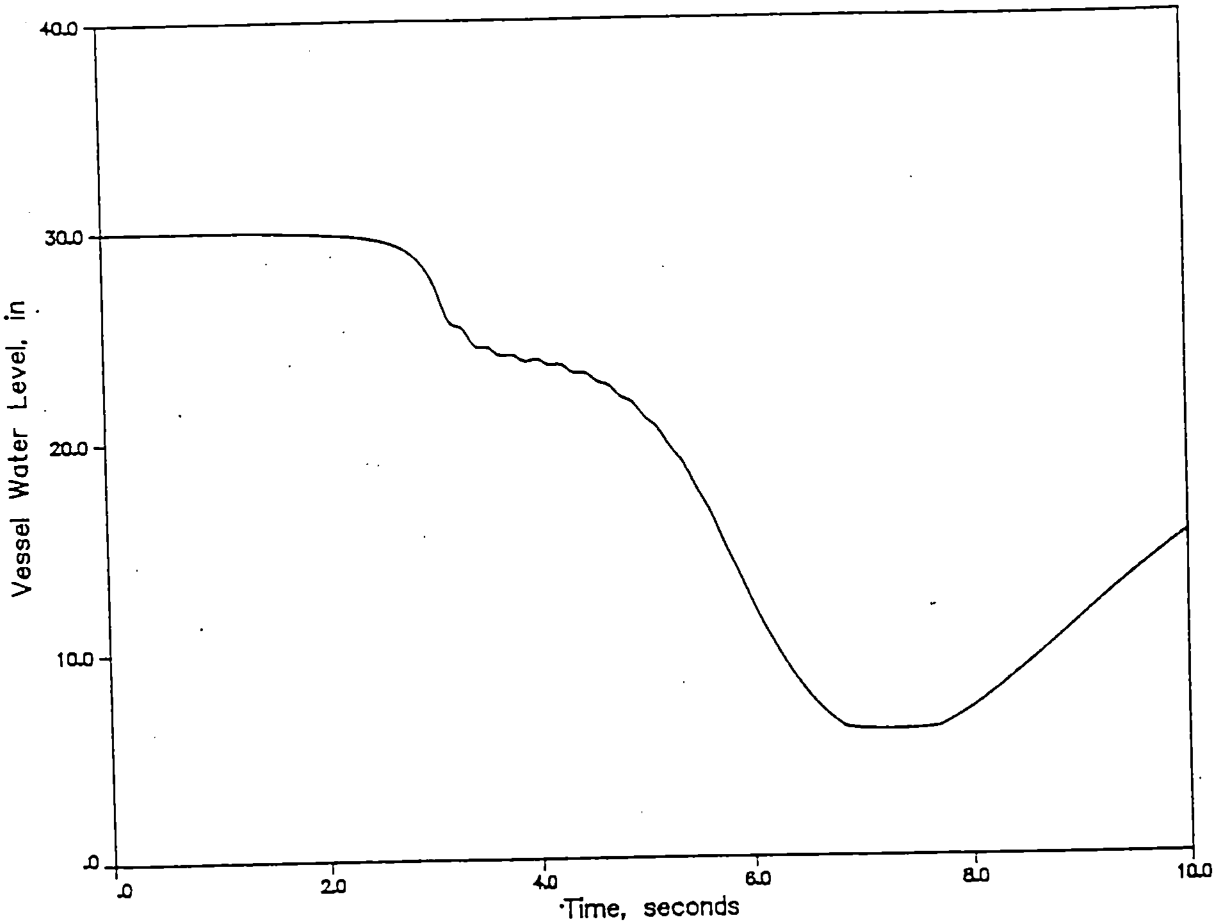


Figure 4.3

MSIV Closure at 100/100 -
Vessel Water Level
(Referenced to Instrument Zero)

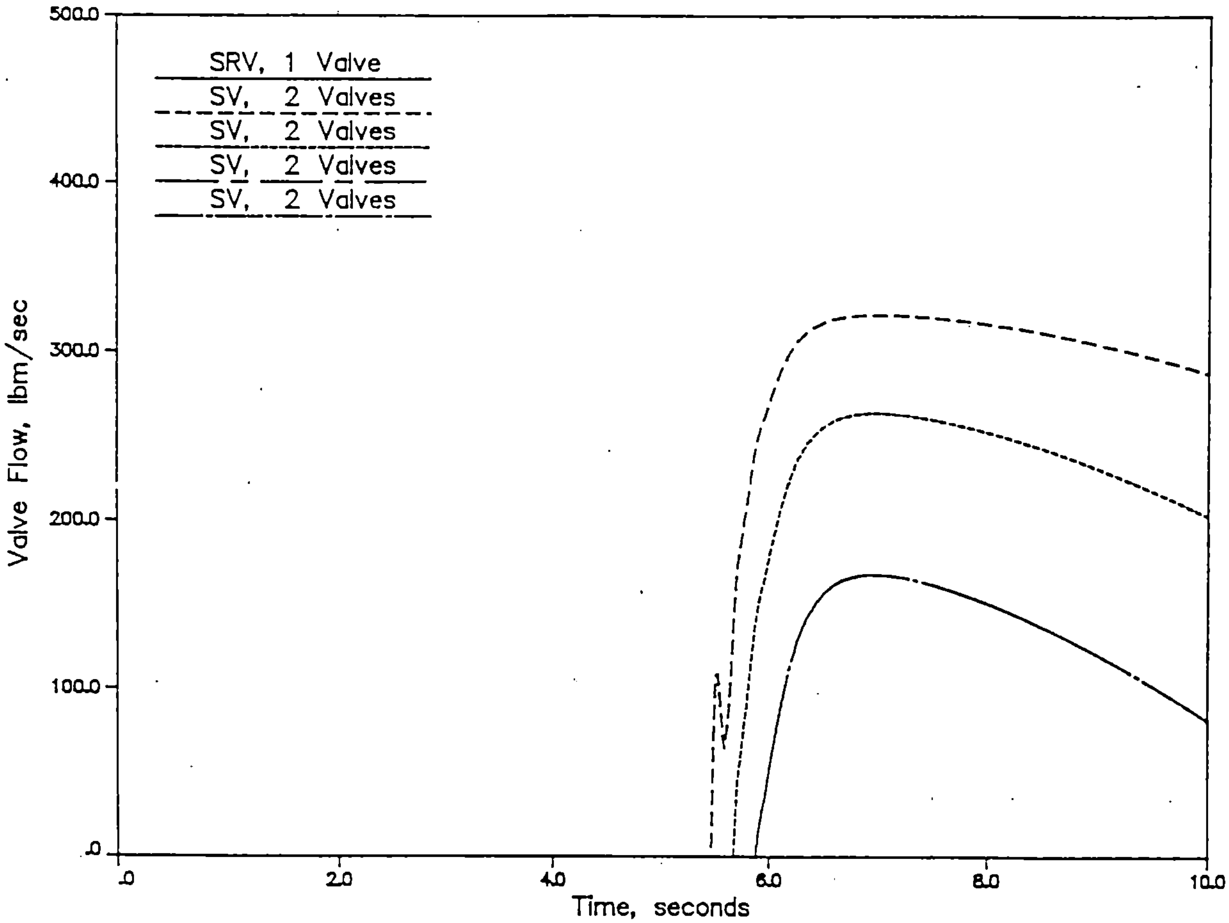


Figure 4.4

MSIV Closure at 100/100 - Safety Valve Flow Rates
(SRV Assumed Inoperable)

Table 5.1

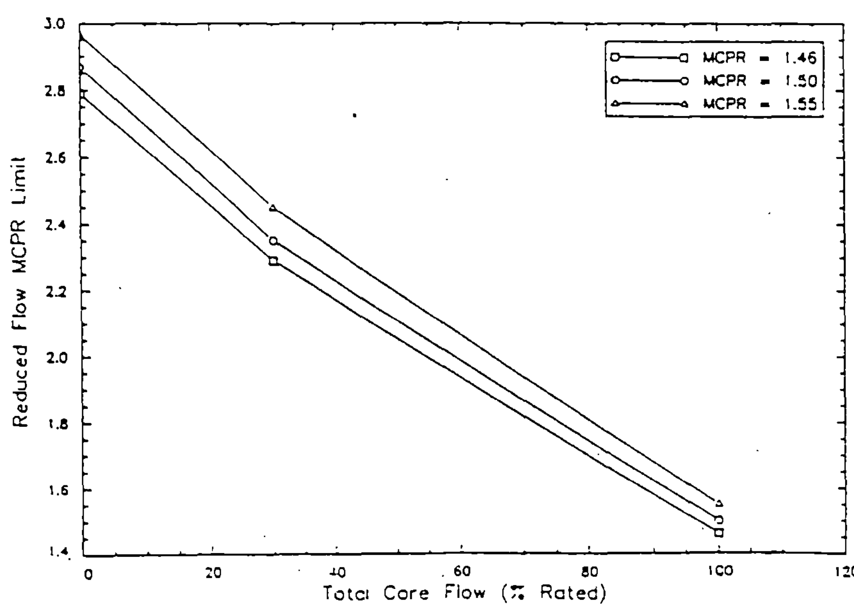
Automatic Flow Control
Excursion Path

Recirculating Flow (% of Rated)	Power (% of Rated)
100	100
90	92
80	84
70	76
60	68
50	60
40	52
30	44

Table 5.2

Reduced Flow MCPR Limits for
Automatic Flow Control
(ATRIUM-9B and 9x9-2 Fuel)

Recirculation Flow (% of Rated)	$MCPR_f$ Limit for OLMCPR = 1.46	$MCPR_f$ Limit for OLMCPR = 1.50	$MCPR_f$ Limit for OLMCPR = 1.55
100	1.46	1.50	1.55
30	2.29	2.35	2.45
0	2.79	2.87	2.97



NOTE: Larger view of the above graph is found on page 5-10 (Figure 5.1).

Table 5.3

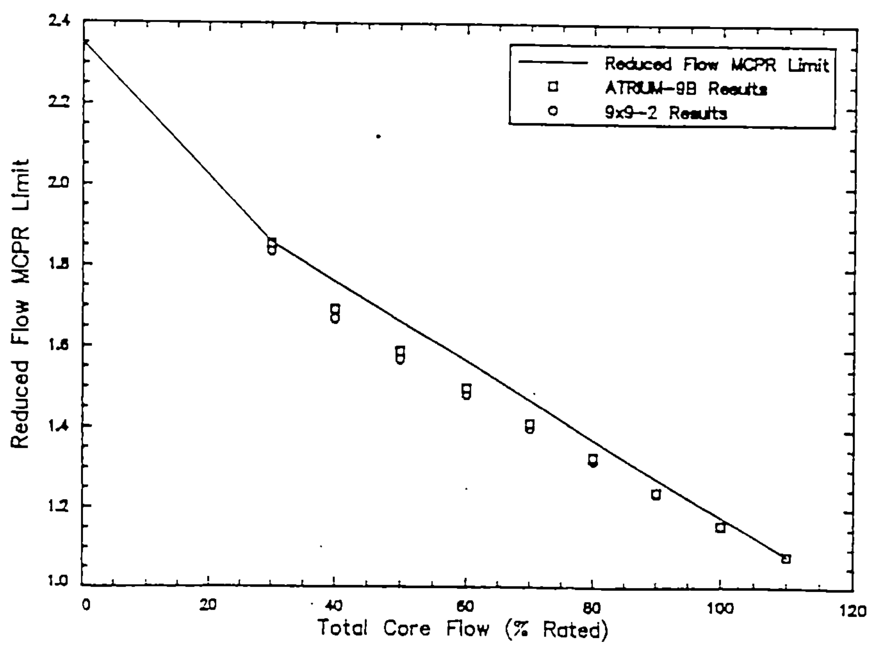
Manual Flow Control
Excursion Path

Recirculating Flow (% of Rated)	Power (% of Rated)
110	120
100	111
90	102
80	93
70	83
60	74
50	65
40	56
30	47

Table 5.4

Reduced Flow MCPR Limits for
Manual Flow Control
(ATRIUM-9B and 9x9-2 Fuel)

Recirculation Flow (% of Rated)	MCPR Limit
100	1.18
30	1.86
0	2.35



NOTE: Larger view of the above graph is found on page 5-11 (Figure 5.2).

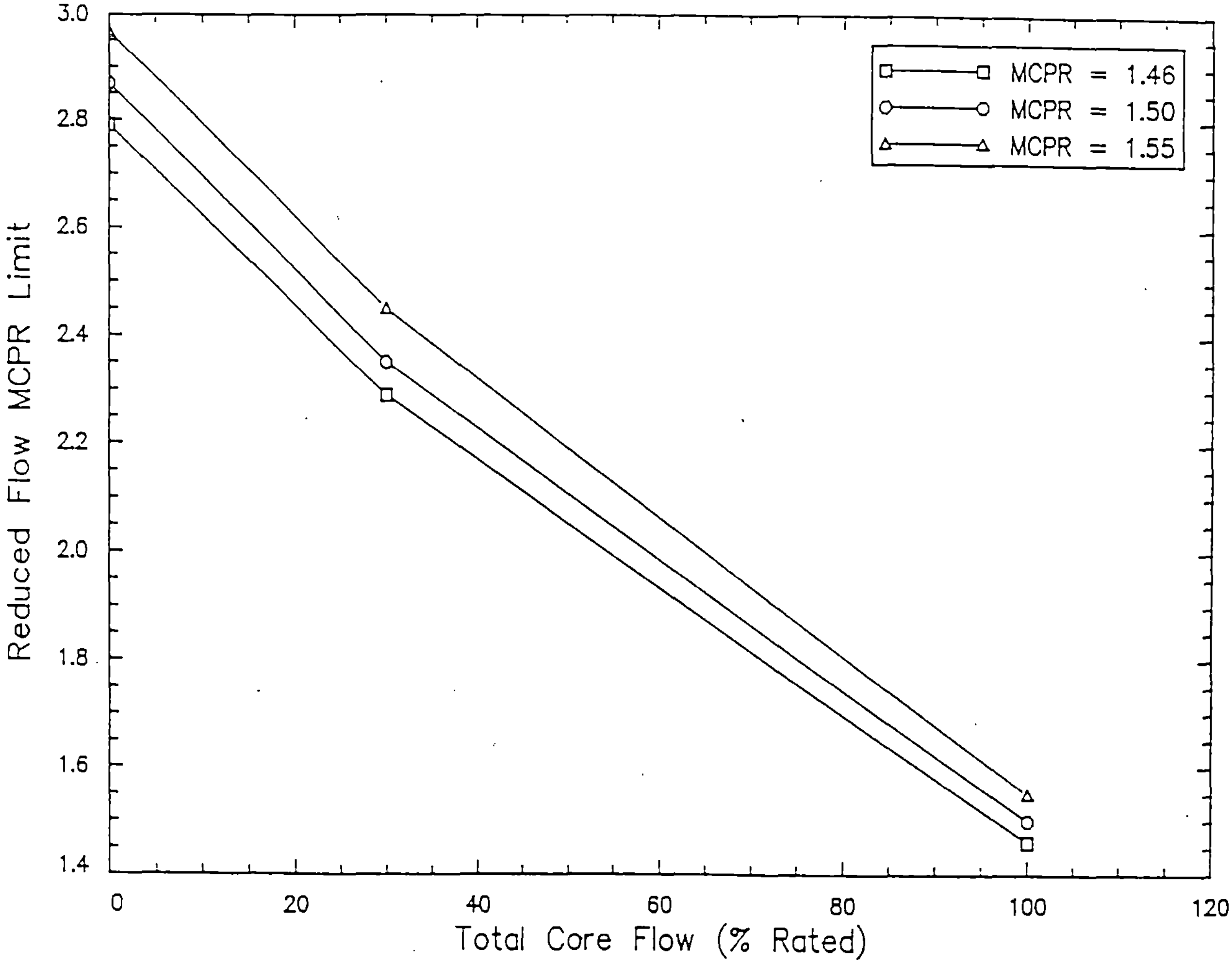


Figure 5.1

Reduced Flow MCPR Limit for Automatic Flow Control (ATRIUM-9B and 9x9-2 Fuel)

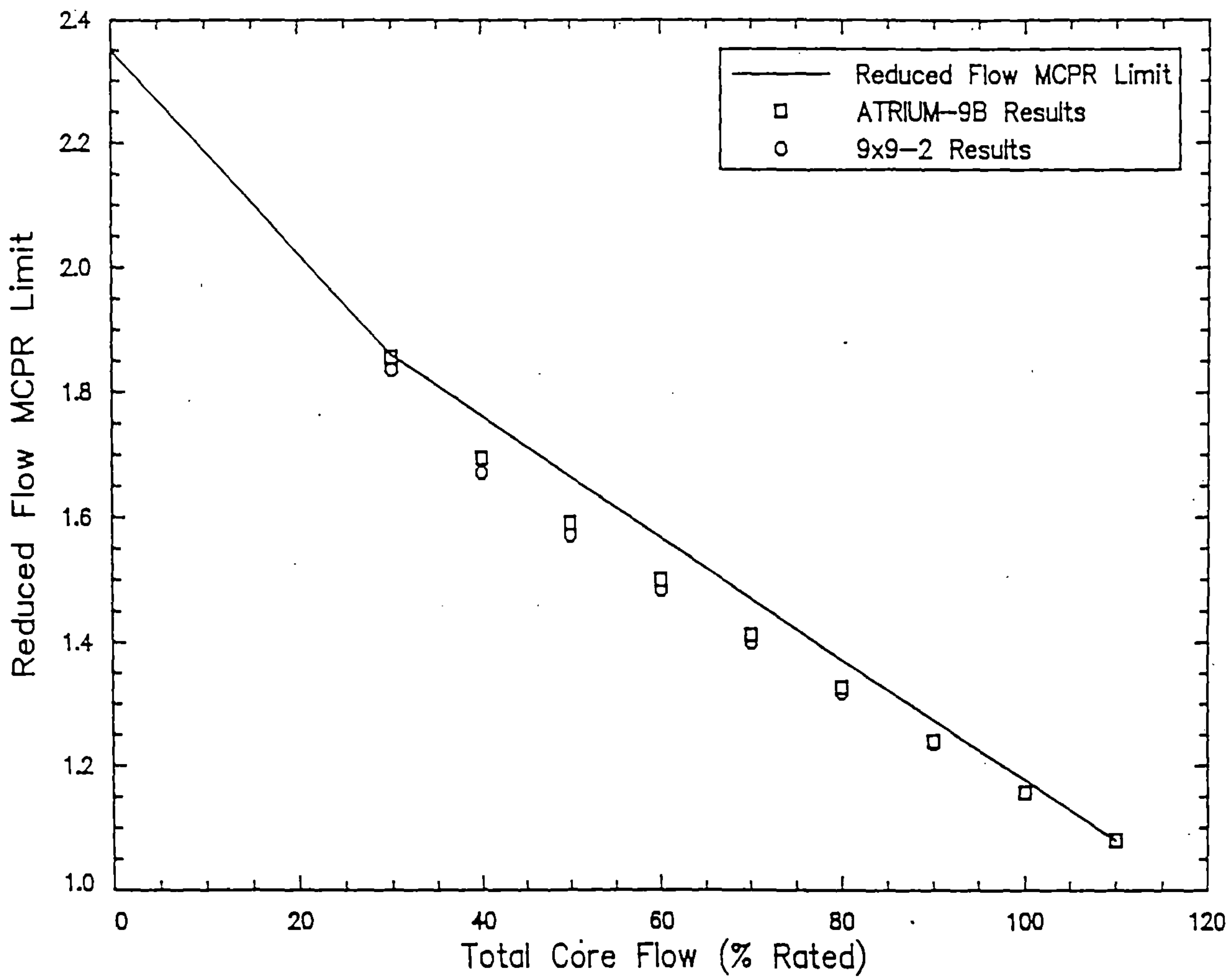


Figure 5.2

Reduced Flow MCPR Limit for Manual Flow Control (SLMCPR = 1.08) (ATRIUM-9B and 9x9-2 Fuel)

Table A.1

SLO Reactor and Plant Conditions

Reactor Thermal Power (81.3%)	2054.5 MWt
Total Recirculation Flow (58%)	56.84 Mlbm/hr
Core Bypass Flow	6.07 Mlbm/hr[a]
Core Inlet Enthalpy	507.8 Btu/lbm
Vessel Pressures	
Steam Dome	993.8 psia
Core Exit	999.0 psia
Lower Plenum	1012.6 psia
Turbine Pressure	958.1 psia
Feedwater/Steam Flow	7.80 Mlbm/hr
Feedwater Enthalpy	294.2 Btu/lbm

[a] Includes water rod/channel flow.

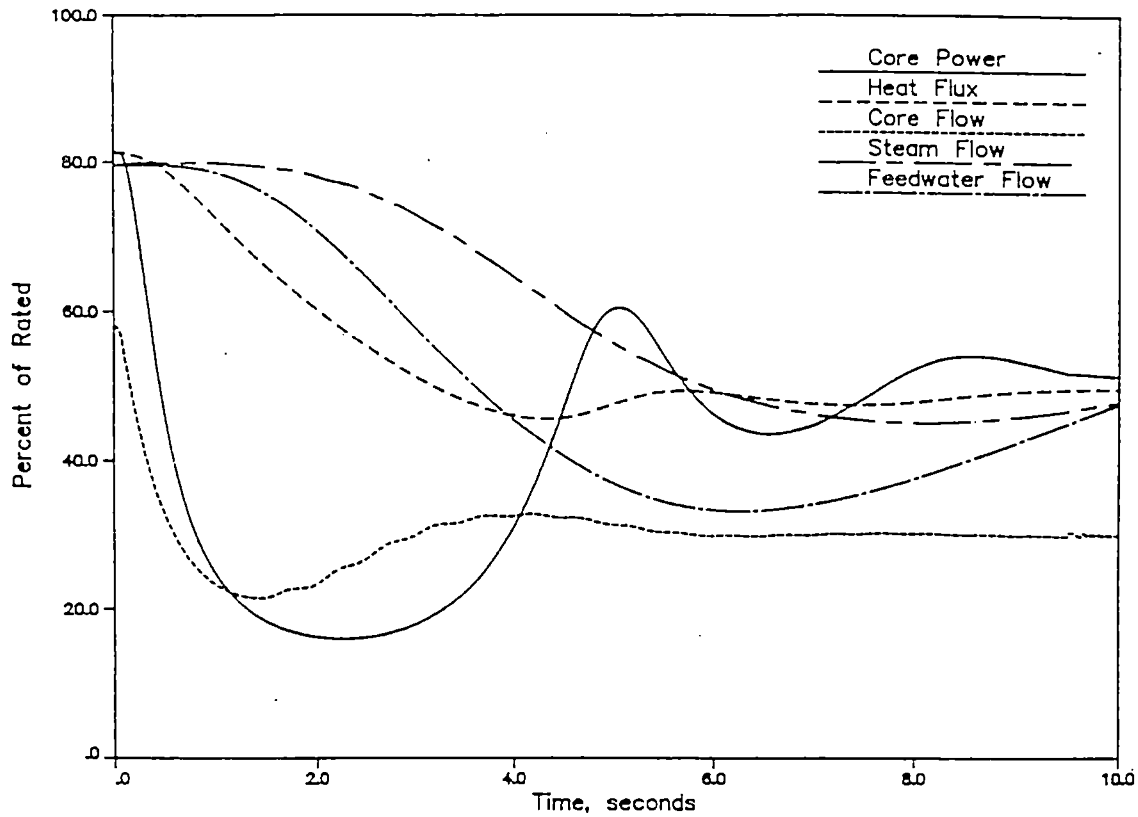


Figure A.1

Single-Loop Operation Pump Seizure - Key Parameters

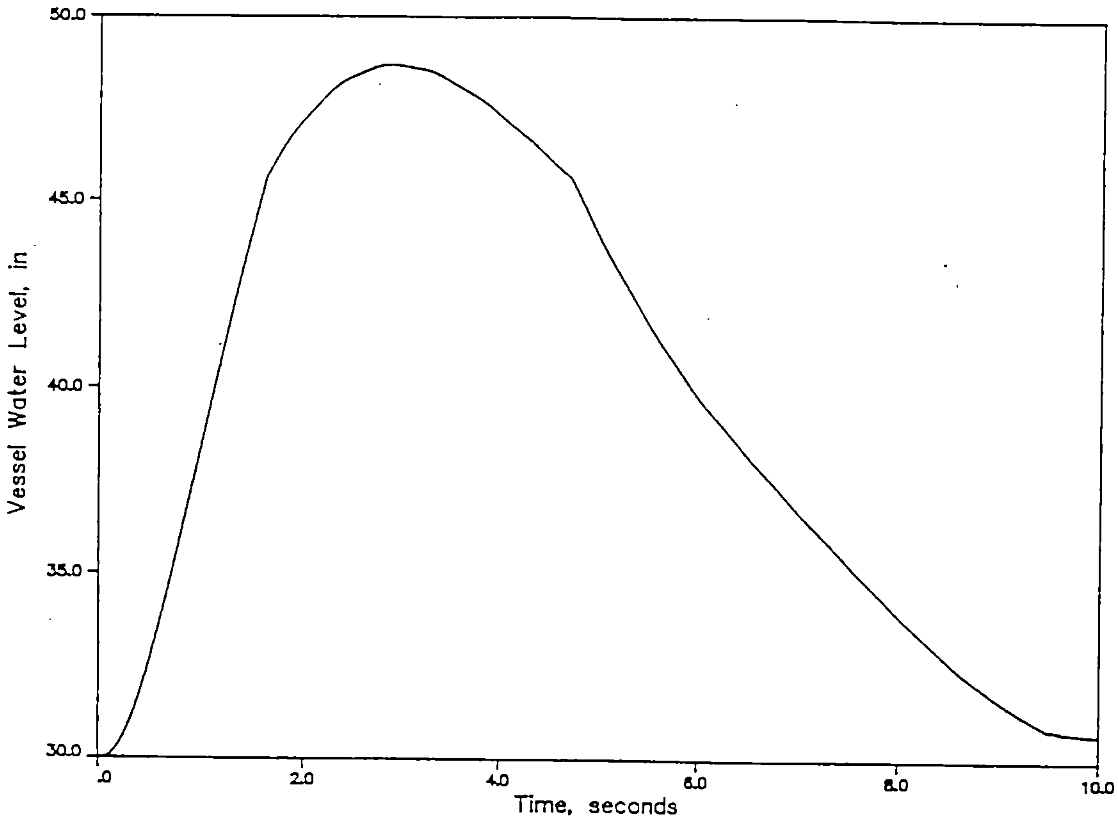


Figure A.2

Single-Loop Operation Pump Seizure - Vessel Water Level
(Referenced to Instrument Zero)

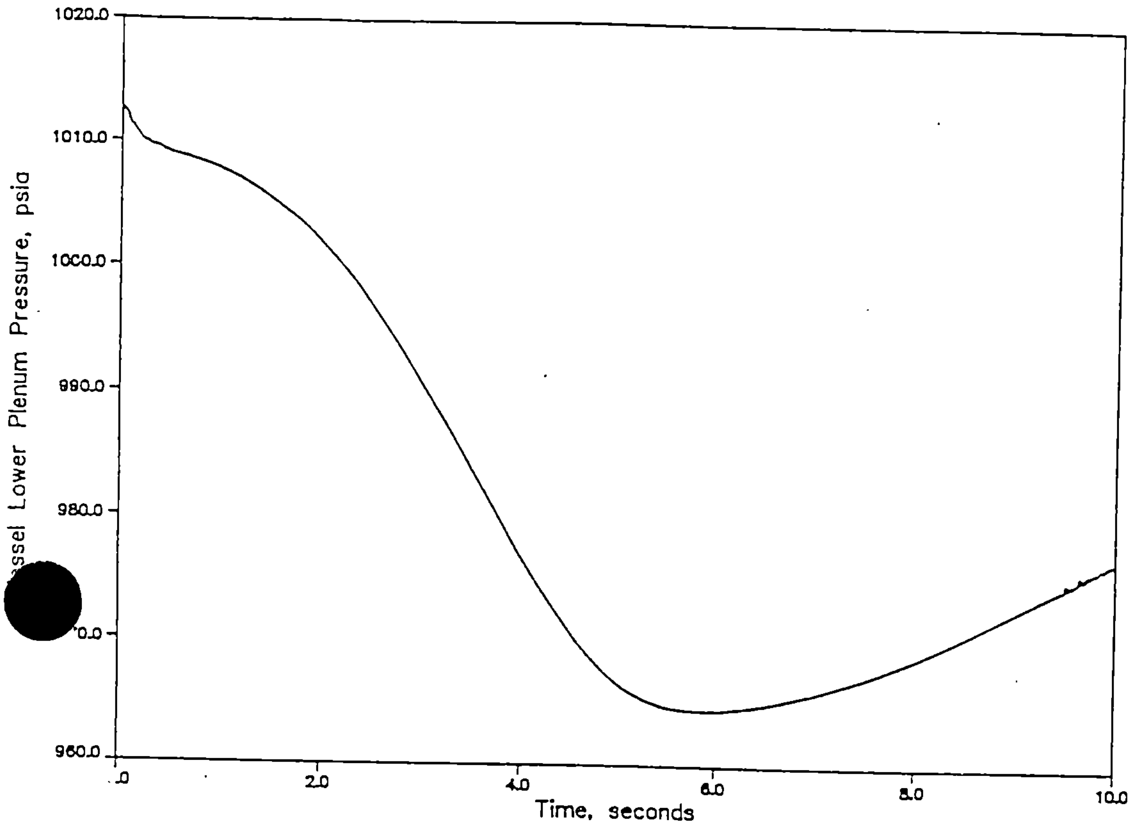


Figure A.3

Single-Loop Operation Pump Seizure - Vessel Pressure Response