XN-NF-85-57

DRESDEN UNIT 3 CYCLE 10 RELOAD ANALYSIS

Design and Safety Analyses
For ENC XN-3 9x9 Reload Fuel

SEPTEMBER 1985

RICHLAND, WA 99352

EXXON NUCLEAR COMPANY, INC.

ERRATA NO. 1

XN-NF-85-57

Dresden Unit 3, Cycle 10 Reload Analysis

1. Make the following pen-and-ink changes in the issued document, XN-NF-85-57:

   a. Page 3, Section 3.2.3, "Fuel Centerline Temperature;" change "153%" to "124%" in the line "Centerline Temperature at 153% Power." (Correction of typographical error)

2. Post this sheet inside the front cover of the corrected report.

XN-NF-85-57
Issue Date: 9/12/85

DRESDEN UNIT 3 CYCLE 10 RELOAD ANALYSIS

Design and Safety Analyses
For ENC XN-3 9x9 Reload Fuel

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DRESDEN UNIT 3 CYCLE 10 RELOAD ANALYSIS

Design and Safety Analyses
For ENC XN-3 9x9 Reload Fuel

1.0 INTRODUCTION

This report provides the results of the analyses performed by Exxon Nuclear Company (ENC) in support of the Cycle 10 reload for Dresden Unit 3, which is scheduled to commence operation in Spring 1986. This report is intended to be used in conjunction with ENC topical report XN-NF-80-19(P), 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), Volume 4, Revision 1.

The Dresden Unit 3 Cycle 10 core will comprise a total of 724 fuel assemblies, including 176 unirradiated ENC XN-3 9x9 assemblies, 408 previously irradiated ENC-fabricated 8x8 assemblies, and 140 previously irradiated Type P8x8R assemblies fabricated by General Electric. No G.E.-fabricated 7x7, 8x8, or 8x8R fuel assemblies are to be irradiated in the Dresden Unit 3 Cycle 10 core. 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. Additional information and the results of design studies covering the development of 9x9 fuel assemblies for BWR reloads are contained in Reference 9.8.

2.0 FUEL MECHANICAL DESIGN ANALYSIS

Applicable Fuel Design Report: Reference 9.1

The expected power history for the 9x9 fuel to be irradiated during Cycle 10 of Dresden Unit 3 is bounded by the assumed power history in the fuel mechanical design analysis. The mechanical qualification of ENC 8x8 fuel as presented in Reference 9.4 remains valid for Cycle 10 operation of Dresden Unit 3.

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 10 core have been determined in single phase flow tests of full scale assemblies. The compatibility of the ENC XN-1 and XN-2 8x8 fuel types with representative G.E. 8x8 fuel has been demonstrated generically in XN-NF-80-19(P), Volume 4. Figure 3.1 illustrates the hydraulic demand curves for ENC and G.E. 8x8 fuel and ENC 9x9 fuel in the Dresden Unit 3 core. The XN-3 9x9 fuel performance falls between that of the ENC 8x8 fuel and that of the G.E. 8x8 fuel, indicating adequate compatibility for coresidence in the Dresden core.

3.2.3 Fuel Centerline Temperature	
Exposure at Minimum Margin Point	5000 MWD/MT
Centerline Temperature at ~~153%~~ 124% Power	4115 F
Melting Point of Fuel	5050 F
Margin to Centerline Melting	935 F
3.2.5 Bypass Flow	
Calculated Bypass Flow Fraction	10.0%

3.3 MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT

3.3.1 Coolant Thermodynamic Condition	
Rated Thermal Power	2527 MWt
Feedwater Flowrate (at SLMCPR)	12.6 Mlbm/hr
Steam Dome Pressure (at SLMCPR)	1015 psia
Feedwater Temperature	345 F

3.3.2 Design Basis Radial Power Distribution

See Figure 3.2

3.3.3 Design Basis Local Power Distribution

See Figures 3.3 through 3.6

4.0 NUCLEAR DESIGN ANALYSIS

4.1 FUEL BUNDLE NUCLEAR DESIGN ANALYSIS

Assembly Average Enrichment	3.13%
Radial Enrichment Distribution	
ENC XN-3 9x9	Figure 4.1
ENC XN-3A 9x9	Figure 4.2
Axial Enrichment Distribution	Uniform 3.35% with 6" natural urania ends
Burnable Poisons	Figs. 4.1, 4.2
Non-Fueled Rods	Figs. 4.1, 4.2
Neutronic Design Parameters	Table 4.1
Maximum Lattice k-infinite	1.235

4.2 CORE NUCLEAR DESIGN ANALYSIS

4.2.1 Core Configuration	Figure 4.3
Core Exposure at EOC9, MWD/MT	
Nominal Value	19,754
Shutdown Reactivity Calculations	19,524
Core Exposure at BOC10, MWD/MT	13,026
Core Exposure at EOC10, MWD/MT	21,665
4.2.2 Core Reactivity Characteristics	
BOC Cold K-effective, All Rods Out	1.1095
BOC Cold K-effective, All Rods In	0.9594
BOC Cold K-effective, Strongest Rod Out	0.9874

Technical Specification R-Value	0.0022
Includes 0.0004 to account for B4C settling in control rod tubes	
Standby Liquid Control System Reactivity, Cold Conditions, 600 ppm	0.9420
4.2.4 Core Hydrodynamic Stability	Figure 4.4
Maximum Decay Ratio Values	
100% Flow Control Line	0.33

5.0 ANTICIPATED OPERATIONAL OCCURRENCES

Applicable Generic Transient Analysis Report — Reference 9.2

5.1 ANALYSIS OF PLANT TRANSIENTS AT RATED CONDITIONS — Reference 9.3

Limiting Transient(s): Load Rejection Without Bypass (LRWB)
Feedwater Controller Failure (FWCF)
Loss of Feedwater Heating (LFWH)

Event	Power	Flow	Maximum Heat Flux	Maximum Power	Maximum Pressure	Delta-CPR*	Model
LRWB	100%	100%	109.5%	275%	1273 psia	0.28	COTRANSA
FWCF	100%	100%	112.1%	185%	1191 psia	0.20	COTRANSA
LFWH	100%	100%	120.0%	120%	1064 psia	0.20	PTSBWR3

*Delta-CPR results for most limiting fuel type

5.2 ANALYSES FOR REDUCED FLOW OPERATION Reference 9.3

Limiting Transient(s): Recirculation Flow Increase (RFIT)

5.4 ASME OVERPRESSURIZATION ANALYSIS Reference 9.3

Limiting Event	Containment Isolation
Worst Single Failure	Direct Scram
Maximum Pressure	1341 psia
Maximum Steam Dome Pressure	1316 psia

5.5 CONTROL ROD WITHDRAWAL ERROR

Starting Control Rod Pattern for Analysis Figure 5.1

Rod Block Reading	Distance Withdrawn	Delta-CPR
105%	3.5 ft	0.07/0.07*
106%	4.0	0.10/0.10
107%	4.5	0.11/0.11
108%	5.0	0.13/0.13
109%	5.5	0.14/0.14
110%**	6.0	0.15/0.15

*Delta-CPR values for 8x8/9x9 fuel
**Rod Block Monitor setting selected for Cycle 10 operation

5.6 FUEL MISLOADING ERROR

Maximum LHGR	12.8 kW/ft
Minimum MCPR	1.577
Maximum Delta-CPR	0.19

5.7 DETERMINATION OF THERMAL MARGINS

Summary of Thermal Margin Requirements

Event	Power	Flow	Delta-CPR*	MCPR Limit*
LRWB	100%	100%	0.24/0.28	1.29/1.33
LFWH	100%	100%	0.20/0.20	1.25/1.25
FWCF	100%	100%	0.17/0.20	1.22/1.25
CRWE	100%	100%	0.15/0.15	1.20/1.20

* Limits for 8x8 fuel/9x9 fuel

MCPR Operating Limits at Rated Conditions

Fuel Type	MCPR Limit
9x9 Fuel	1.33
8x8 Fuel	1.29

MCPR Operating Limits at Off-Rated Conditions

Reduced Flow MCPR Limits

Manual and Automatic Flow Control	Figure 5.2
Automatic Flow Control	Figure 5.3

6.0 POSTULATED ACCIDENTS

6.1 LOSS-OF-COOLANT ACCIDENT

6.1.1 Break Location Spectrum	Reference 9.6
6.1.2 Break Size Spectrum	Reference 9.6
6.1.3 MAPLHGR Analyses for XN-3, XN-3A 9x9 Fuel	Reference 9.7

See Reference 9.5 for MAPLHGR analyses for XN-1 and XN-2 fuel. Existing MAPLHGR limits for these fuel types remain valid for Cycle 10 operation.

Limiting Break: Double-ended guillotine pipe break
Recirculation pump suction line
1.0 Discharge Coefficient

Bundle Average Exposure	MAPLHGR	Peak Clad Temperature	Peak Local MWR
0	11.40	2006 F	2.20%
5,000	11.75	2045 F	2.44%
10,000	11.40	1893 F	0.91%
15,000	10.55	1805 F	0.63%
20,000	9.70	1710 F	0.44%
25,000	8.85	1623 F	0.29%
30,000	8.00	1529 F	0.18%
35,000	7.15	1421 F	0.12%
40,000	6.30	1309 F	0.08%

6.2 CONTROL ROD DROP ACCIDENT — Reference 8.1

Dropped Control Rod Worth	0.0074
Doppler Coefficient, 1/k dk/dT	$-10.5x(10)^{-6}$
Effective Delayed Neutron Fraction	0.0052
Four-Bundle Local Peaking Factor	1.278
Maximum Deposited Fuel Rod Enthalpy, cal/gm	109

7.0 TECHNICAL SPECIFICATIONS

7.1 LIMITING SAFETY SYSTEM SETTINGS

7.1.1 MCPR Fuel Cladding Integrity Safety Limit

MCPR Safety Limit 1.05

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 Limits for XN-3 & XN-3A 9x9 Fuel

Bundle Average Exposure	MAPLHGR
0	11.40 kW/ft
5,000	11.75
10,000	11.40
15,000	10.55
20,000	9.70
25,000	8.85
30,000	8.00
35,000	7.15
40,000	6.30

7.2.2 Minimum Critical Power Ratio

Rated Conditions MCPR Limits

Fuel Type	Limit
9x9 Fuel	1.33
8x8 Fuel	1.29

Off-Rated Conditions MCPR Limits

Reduced Flow MCPR Limits

Manual and Automatic Flow Control	Figure 7.1
Automatic Flow Control	Figure 7.2

7.3 SURVEILLANCE REQUIREMENTS

7.3.1 Scram Insertion Time Surveillance

Individual control rod drive insertion times shall be monitored in accordance with existing Technical Specification requirements. If the average insertion time to the 90% insertion point for all the control rod drives in the core, based on the most recent observation for each drive, exceeds 2.73 seconds, the MCPR operating limit for each fuel type in the core shall be increased by an amount determined by:

$$OLMCPR = MCPRs + (Tav - 2.73) * MCPRt,$$

where OLMCPR is the MCPR operating limit, MCPRs is the Technical Specification MCPR operating limit based on compliance with the statistical assumptions, Tav is the average control rod insertion time to 90%, and MCPRt is 0.130 for 8x8 fuel and 0.143 for 9x9 fuel.

This surveillance requirement does not supersede the Control Rod Drive operability requirement or the scram insertion time requirements specified elsewhere.

7.3.3 Procedural Controls

Procedural controls on fuel rod local linear heat generation rate shall be established such that the power distribution assumptions of the mechanical design analysis remain applicable.

8.0 METHODOLOGY REFERENCES

See XN-NF-80-19, Volume 4 for complete bibliography.

9.0 ADDITIONAL REFERENCES

9.1 "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," XN-NF-85-67(P), Exxon Nuclear Company, Richland, Washington (July 1985).

9.2 "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," XN-NF-79-71(P), Revision 2, Exxon Nuclear Company, Richland, Washington (November 1981).

9.3 "Dresden Unit 3 Cycle 10 Plant Transient Analysis," XN-NF-85-62, Exxon Nuclear Company, Richland, Washington (September 1985).

9.4 "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," XN-NF-81-21(A), Revision 1, Exxon Nuclear Company, Richland, Washington (January 1982).

9.5 "Dresden Unit 3 LOCA Analysis Using the EXEM/BWR Evaluation Model," XN-NF-81-75, Exxon Nuclear Company, Richland, Washington

9.6 "Generic Jet Pump BWR3 LOCA Analysis Using the EXEM Evaluation Model," XN-NF-81-71(A), Exxon Nuclear Company, Richland, Washington (August 1981).

9.7 Dresden Unit 3 LOCA-ECCS Analysis MAPLHGR Results for ENC 9x9 Fuel," XN-NF-85-63, Exxon Nuclear Company, Richland, Washington (September 1985).

9.8 "Demonstration of 9x9 Assemblies for BWRs," EPRI NP-1580-5, Electric Power Research Institute, Palo Alto, California (May 1984).

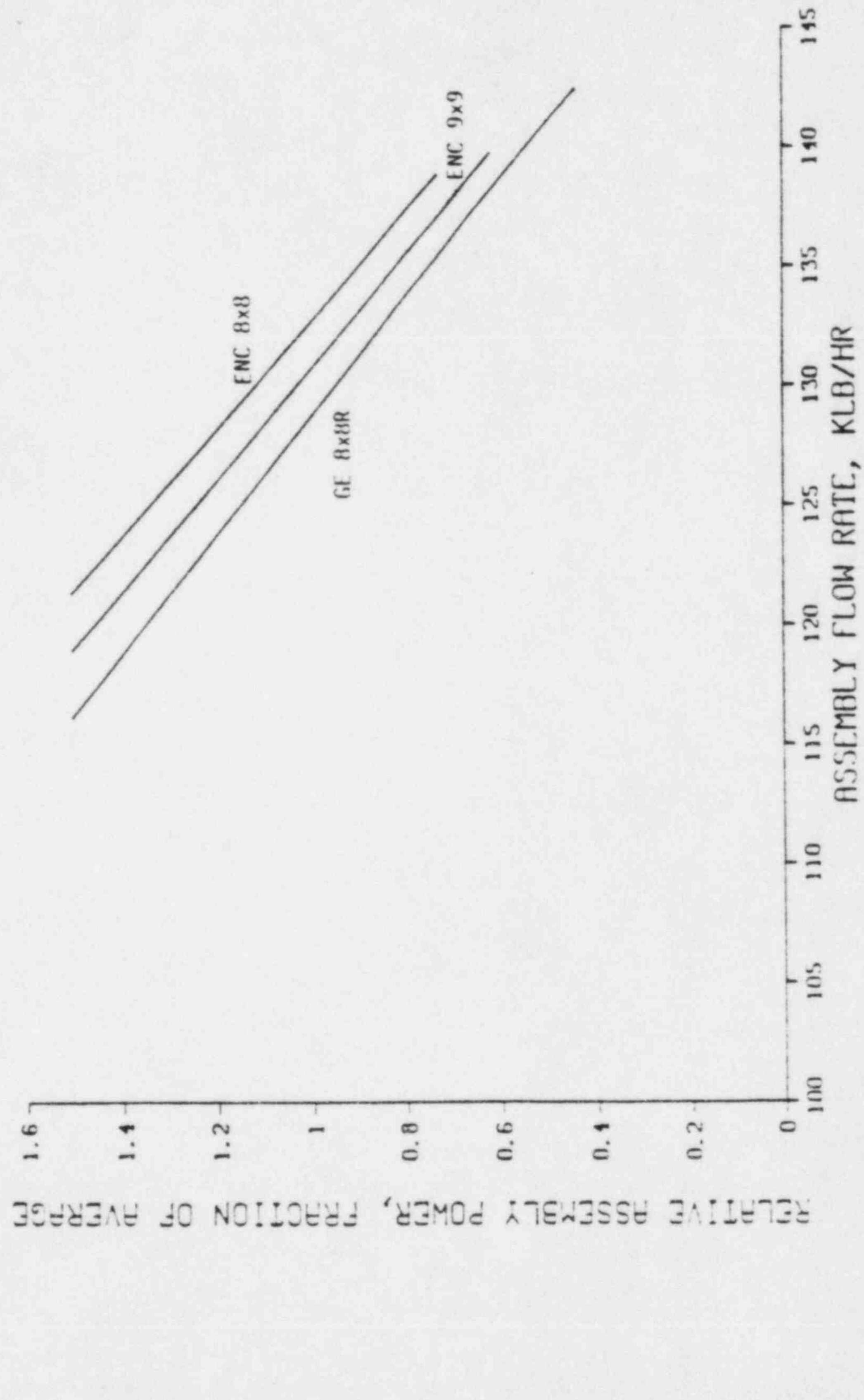


Figure 3.1 Hydraulic Demand Curves for Dresden Units 3 Cycle 10 Core

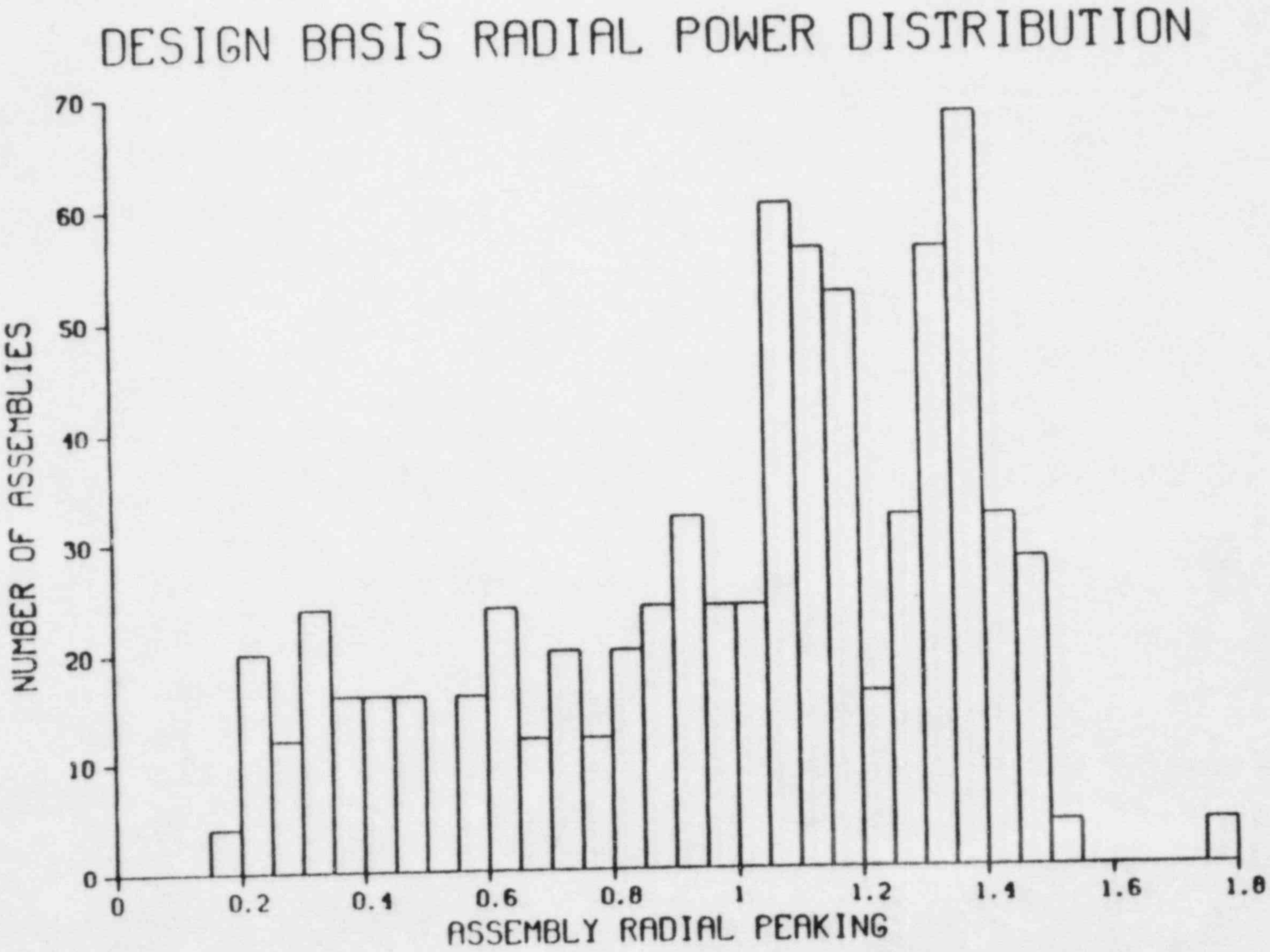


Figure 3.2 Design Basis Radial Power Distribution

LL 0.88	L 0.93	ML 0.98	M 1.07	H 1.12	H 1.12	M 1.06	ML 0.97	L 0.92
L 0.97	ML 0.99	M 1.04	ML* 0.85	H 1.03	H 1.03	ML* 0.85	H 1.12	ML 0.97
L 0.94	M 1.09	M 0.99	H 1.01	H 0.99	H 1.00	H 1.02	ML* 0.85	M 1.06
ML 1.04	ML* 0.92	H 1.05	H 0.99	H 1.00	W 0.00	H 1.00	H 1.03	H 1.12
ML 1.04	M 1.07	H 1.05	H 1.01	W 0.00	H 1.00	H 0.99	H 1.03	H 1.12
L 0.95	M 1.10	M 1.00	H 1.03	H 1.01	H 0.99	H 1.01	ML* 0.85	M 1.07
L 0.98	ML* 1.01	ML 0.94	M 1.00	H 1.05	H 1.05	M 0.99	M 1.04	ML 0.98
LL 0.89	L 0.95	ML* 1.01	M [illegible]	M [illegible]	ML* 0.92	M 1.09	ML 0.99	L 0.93
LL 0.94	LL 0.89	L 0.98	L 0.95	ML 1.04	ML 1.04	L 0.94	L 0.97	LL 0.88

Figure 3.3 Design Basis Local Power Distribution
ENC XN-3 and XN-3A 9x9 Fuel

L 0.99	ML 0.99	ML 0.94	M 1.06	M 1.05	M 1.07	ML 0.94	ML 0.98
ML 1.01	ML* 0.93	M 1.04	H 1.02	H 1.01	ML* 0.83	M 1.04	ML 0.94
ML 0.99	M 1.07	H 1.02	H 0.98	H 0.99	H 1.00	ML* 0.83	M 1.07
ML 0.98	M 1.06	H 1.01	H 0.98	W 0.00	H 0.99	H 1.01	M 1.05
ML 0.99	M 1.07	H 1.03	H 0.93	H 0.98	H 0.98	H 1.02	M 1.06
ML 1.01	ML 0.93	M 1.05	H 1.03	H 1.01	H 1.02	M 1.04	ML 0.94
L 0.99	ML* 0.99	ML 0.93	M 1.07	M 1.06	M 1.07	ML* 0.93	ML 0.99
LL 0.95	L 0.99	ML 1.01	ML 0.99	ML 0.98	ML 0.99	ML 1.01	L 0.99

W I D E (vertical, left side)

W I D E

Figure 3.4 Design Basis Local Power Distribution for ENC XN-2 8x8 Fuel

L 0.99	ML 0.97	ML 0.95	M 1.05	M 1.04	M 1.05	ML 0.95	ML 0.98
ML 0.99	ML* 0.94	M 1.03	H 1.03	H 1.03	ML* 0.88	M 1.04	ML 0.95
ML 0.97	M 1.05	H 1.04	H 1.01	H 1.01	H 1.02	ML* 0.88	M 1.05
ML 0.97	M 1.04	H 1.03	H 1.01	W 0.00	H 1.01	H 1.03	M 1.04
ML 0.97	M 1.04	H 1.03	H 1.01	H 1.01	H 1.01	H 1.03	M 1.05
ML 0.98	ML 0.94	M 1.03	H 1.03	H 1.03	H 1.04	M 1.03	ML 0.95
L 0.98	ML* 0.95	ML 0.94	M 1.04	M 1.04	M 1.05	ML* 0.94	ML 0.97
LL 0.99	L 0.96	ML 0.96	ML 0.97	ML 0.97	ML 0.97	ML 0.99	L 0.99

Figure 3.5 Design Basis Local Power Distribution for ENC XN-1 8x8 Fuel

L 0.97	ML 0.95	MH 1.02	MH 1.01	MH 1.00	MH 1.01	M 0.96	ML 0.95
ML 0.96	MH 1.02	MH* 0.99	H 1.06	MH* 0.96	H 1.05	MH 0.99	M 0.96
ML 0.95	MH 1.01	H 1.07	H 1.05	H 1.03	H 1.03	H 1.05	MH 1.01
ML 0.95	MH 1.01	H 1.07	W 0.00	H 1.05	H 1.03	MH* 0.96	MH 1.00
ML 0.95	MH* 1.01	MH 0.99	MH 0.99	W 0.00	H 1.05	H 1.06	MH 1.01
ML 0.96	MH 1.02	MH 1.00	MH 0.99	H 1.07	H 1.07	MH* 0.99	MH 1.02
L 0.96	ML 0.94	MH 1.02	MH* 1.01	MH 1.01	MH 1.01	MH 1.02	ML 0.95
LL 0.99	L 0.96	ML 0.96	ML 0.95	ML 0.95	ML 0.95	ML 0.96	L 0.97

WIDE (vertical, left side)

WIDE

Figure 3.6 Design Basis Local Power Distribution for G.E. 8x8 Fuel Types

LL	L	ML	M	H	H	M	ML	L
L	ML	M	ML*	H	H	ML*	H	ML
L	M	M	H	H	H	H	ML*	M
ML	ML*	H	H	H	W	H	H	H
ML	M	H	H	W	H	H	H	H
L	M	M	H	H	H	H	ML*	M
L	ML*	ML	M	H	H	M	M	ML
LL	L	ML*	M	M	ML*	M	ML	L
LL	LL	L	L	ML	ML	L	L	LL

W I D E

W I D E

LL RODS (5) --- 1.50 W/O U235
L RODS (12) --- 2.20 W/O U235
ML RODS (11) --- 2.84 W/O U235
M RODS (16) --- 3.72 W/O U235
H RODS (27) --- 4.34 W/O U235
ML* RODS (8) --- 2.84 W/O U235 + 4.00 W/O GD203
W RODS (2) --- INERT WATER ROD

FIGURE 4.1 ENRICHMENT DISTRIBUTION FOR 9D3.35- 8G4.0 (XN-3)

LL	L	ML	M	H	H	M	ML	L
L	ML*	M	ML*	H	H	ML*	H	ML
L	M	M	H	H	H	H	ML*	M
ML	ML*	H	H	H	W	H	H	H
ML	M	H	H	W	H	H	H	H
L	M	M	H	H	H	H	ML*	M
L	ML	ML*	M	H	H	M	M	ML
LL	L	ML	M	M	ML*	M	ML*	L
LL	LL	L	L	ML	ML	L	L	LL

W
I
D
E

W I D E

```
 LL  RODS ( 5)  ---  1.50 W/O U235
  L  RODS (12)  ---  2.20 W/O U235
 ML  RODS (10)  ---  2.84 W/O U235
  M  RODS (16)  ---  3.72 W/O U235
  H  RODS (27)  ---  4.34 W/O U235
ML*  RODS ( 9)  ---  2.84 W/O U235 + 4.00 W/O GD203
  W  RODS ( 2)  ---  INERT WATER ROD
```

FIGURE 4.2 ENRICHMENT DISTRIBUTION FOR 9D3.35- 9G4.0 (XN-3A)

B2	B2	D0	B2	B2	B2	D0	B2	B2	B2	D0	B2	B2	C1	A3
B2	D0	C1	D0	B2	D0	C1	D0	B2	D0	C1	D0	C1	C1	A3
D0	C1	B2	C1	D0	C1	B2	C1	D0	B2	D0	B2	C1	A3	A3
B2	D0	C1	B2	B2	D0	C1	B2	B2	D0	C1	D0	C1	A3	A3
B2	B2	D0	B2	B2	B2	D0	B2	B2	B2	D0	C1	C1	A3	A3
B2	D0	C1	D0	B2	D0	C1	D0	B2	D0	B2	C1	A3	A3	
D0	C1	B2	C1	D0	C1	B2	C1	D0	C1	D0	C1	A3		
B2	D0	C1	B2	B2	D0	C1	B2	B2	D0	C1	A3	A3		
B2	B2	D0	B2	B2	B2	D0	B2	B2	C1	C1	A3	A3		
B2	D0	B2	D0	B2	D0	C1	D0	C1	B2	A3	A3			
D0	C1	D0	C1	D0	B2	D0	C1	C1	A3	A3				
B2	D0	B2	D0	C1	C1	C1	A3	A3	A3					
B2	C1	C1	C1	C1	A3	A3	A3	A3						
C1	C1	A3	A3	A3	A3									
A3	A3	A3	A3	A3										

xy — x = Fuel Type, y = Cycles Irradiated

Fuel Type	Number of Assemblies	Description
A	140	GE 8x8R 2.65 w/o U-235
B	224	XN1 8x8 2.69 w/o U-235
C	184	XN2 8x8 2.83 w/o U-235
D	176	XN3 9x9 3.13 w/o U-235

Figure 4.3 Dresden Unit 3 Cycle 10 Reference Loading Pattern by Fuel Type (One Quarter of Symmetrical Core Loading)

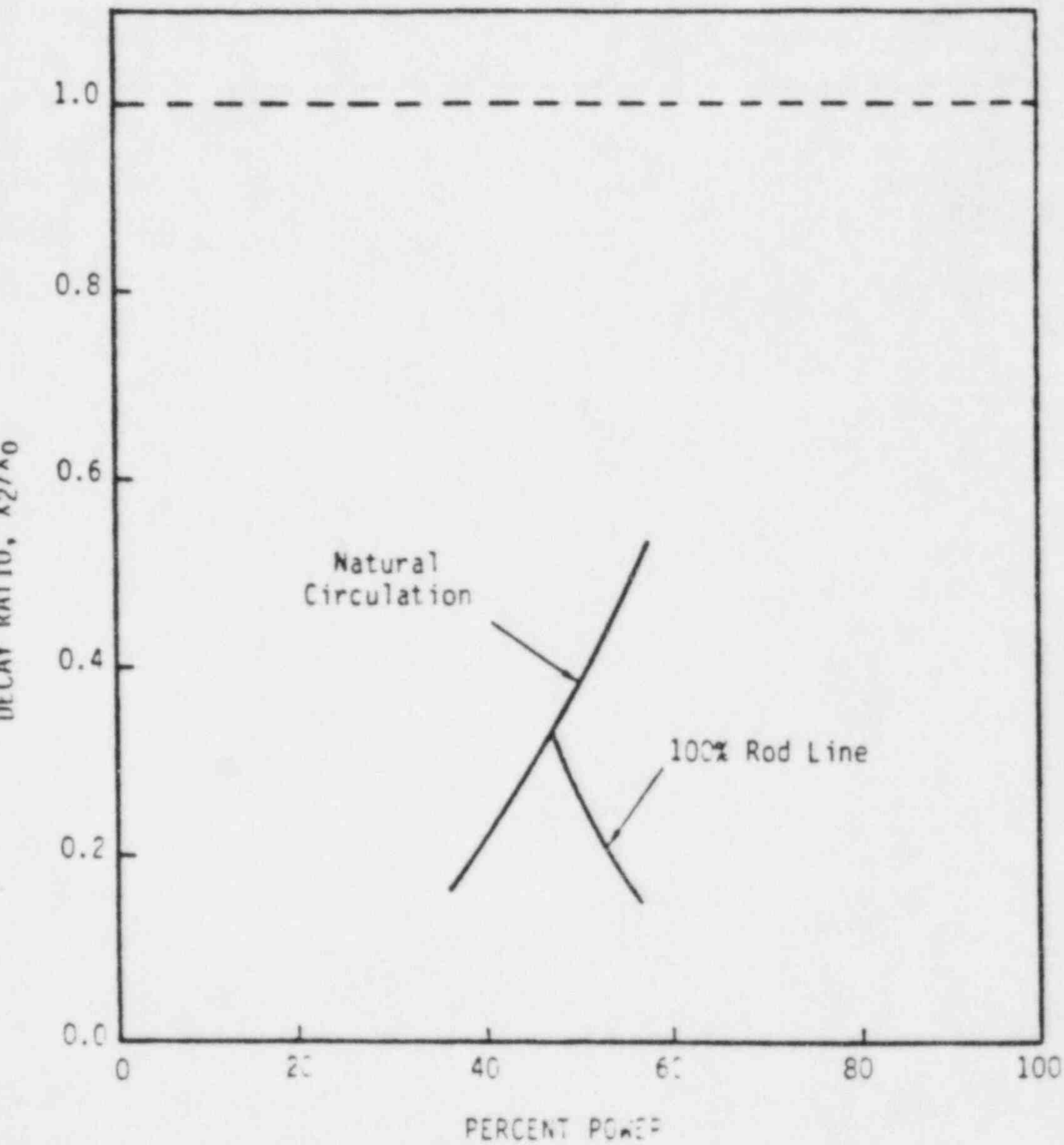
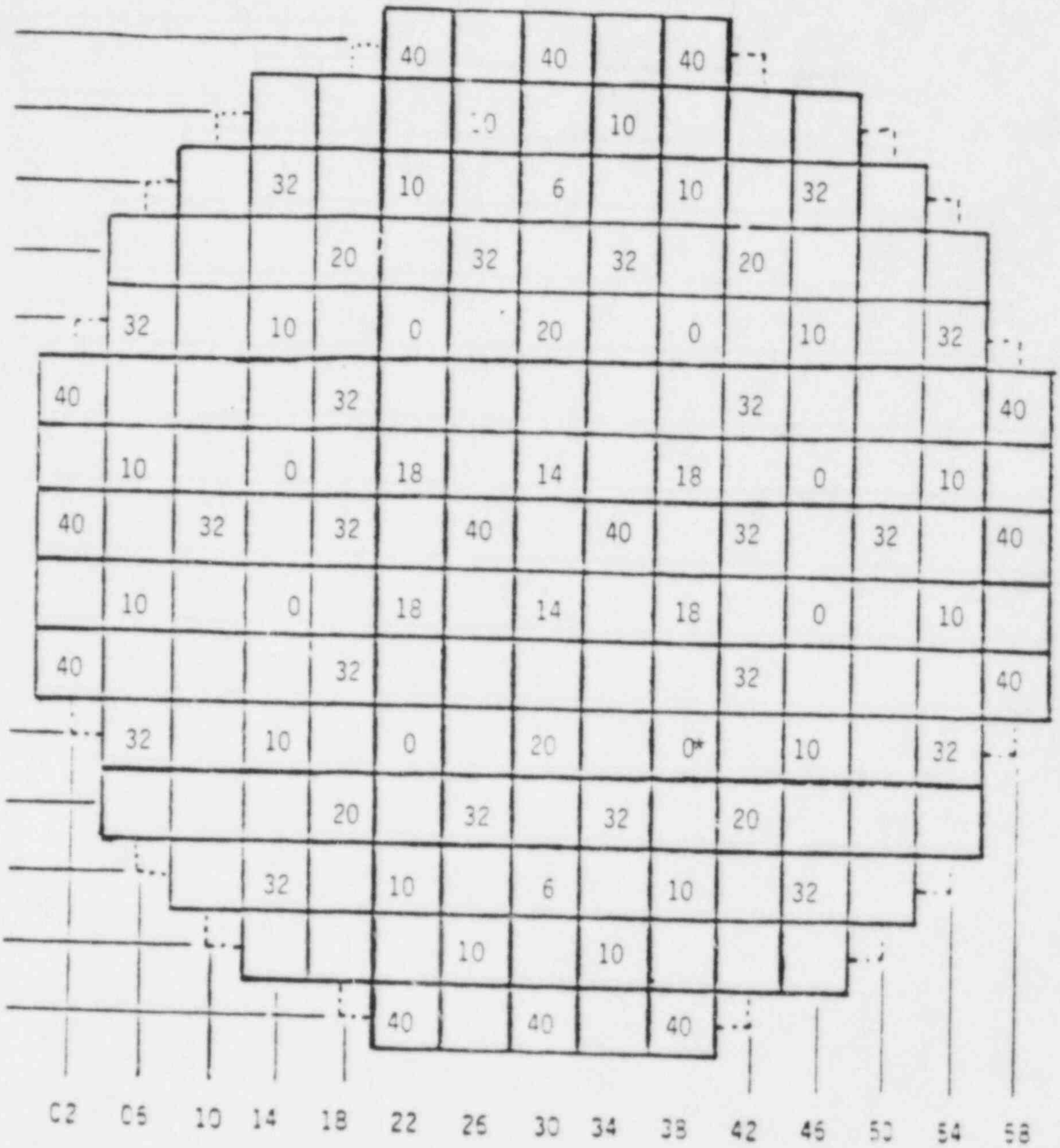


Figure 4.4 Decay Ratio vs. Reactor Power for Dresden 3 Cycle 10

Table 4.1 Neutronic Design Values

Fuel Pellet	Reference 9.1
Fuel Rod	Reference 9.1
Fuel Assembly	Reference 9.1
Core Data	
Number of fuel assemblies	724
Rated thermal power, MW	2527
Rated core flow, Mlbm/hr	98.0
Core inlet subcooling, BTU/lbm	24.6
Moderator temperature, F	546
Channel thickness, inch	0.080
Fuel assembly pitch, inch	6.0
Wide water gap thickness, inch	0.750
Narrow water gap thickness, inch	0.374
Control Rod Data	
Absorber material	B4C, Hf
Total blade span, inch	9.750
Total blade support span, inch	1.562
Blade thickness, inch	0.3120
Blade face-to-face internal dimension, inch	0.200
Absorber rods per blade	84
Absorber rod outside diameter, inch	0.188
Absorber rod inside diameter, inch	0.138
Absorber density, % of theoretical	70



Note: *Control rod being withdrawn, rod positions in notches, full in = 0, full out = blank or 48

Figure 5.1 Starting Control Rod Pattern for Control Rod Withdrawal Analysis

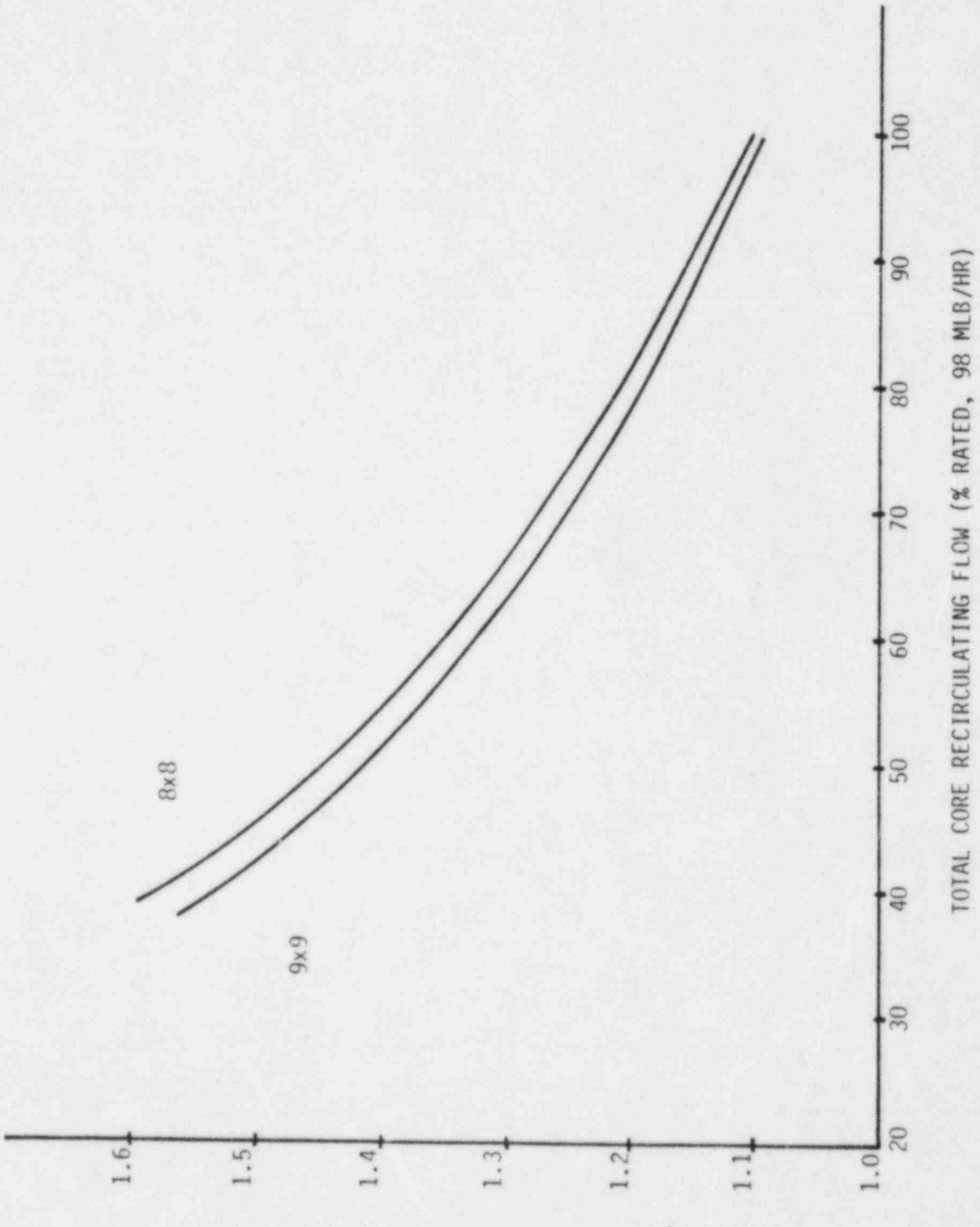


Figure 5.2 Reduced Flow MCPR Limit - All Conditions

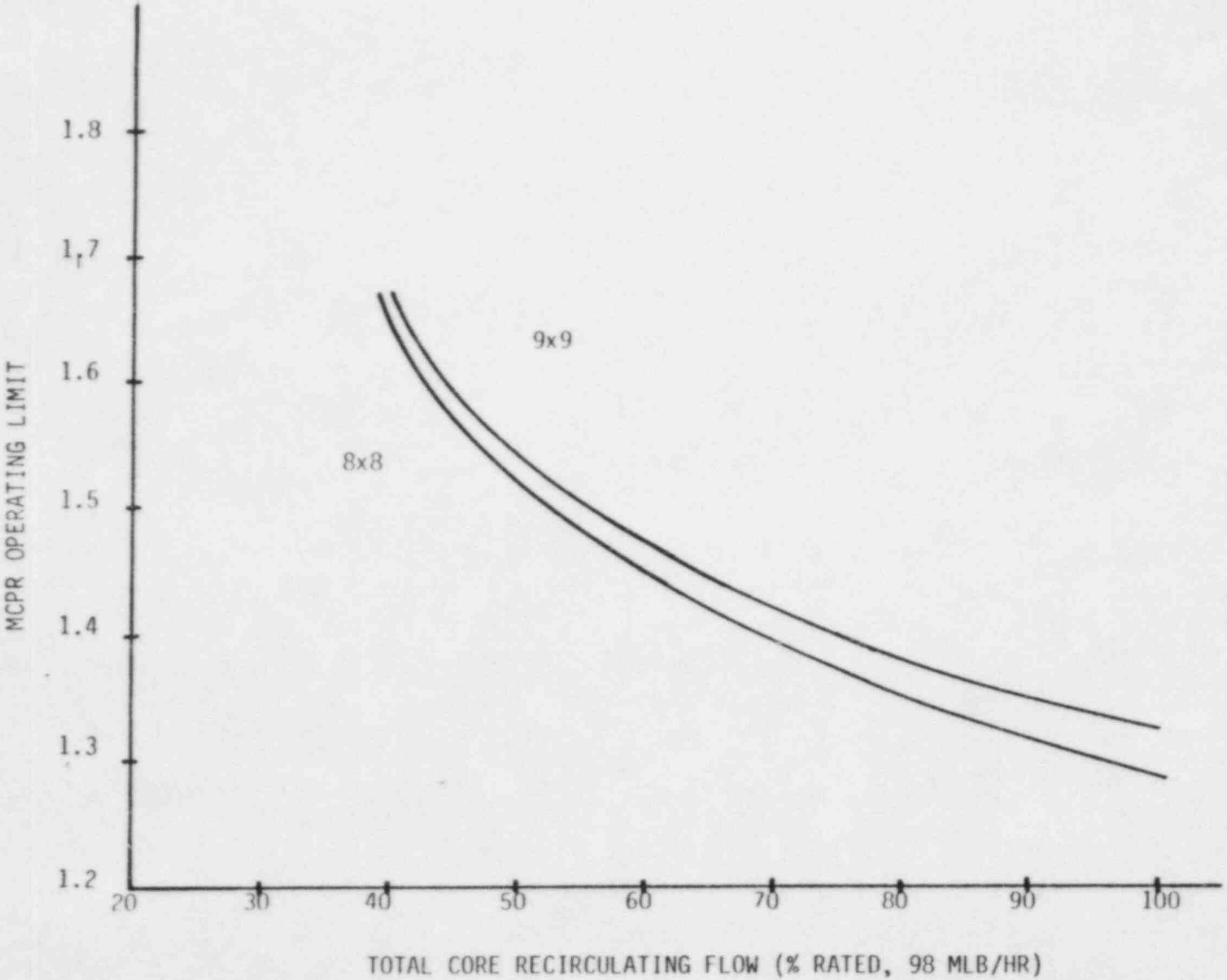


Figure 5.3 Reduced Flow MCPR Limit - Automatice Flow Control

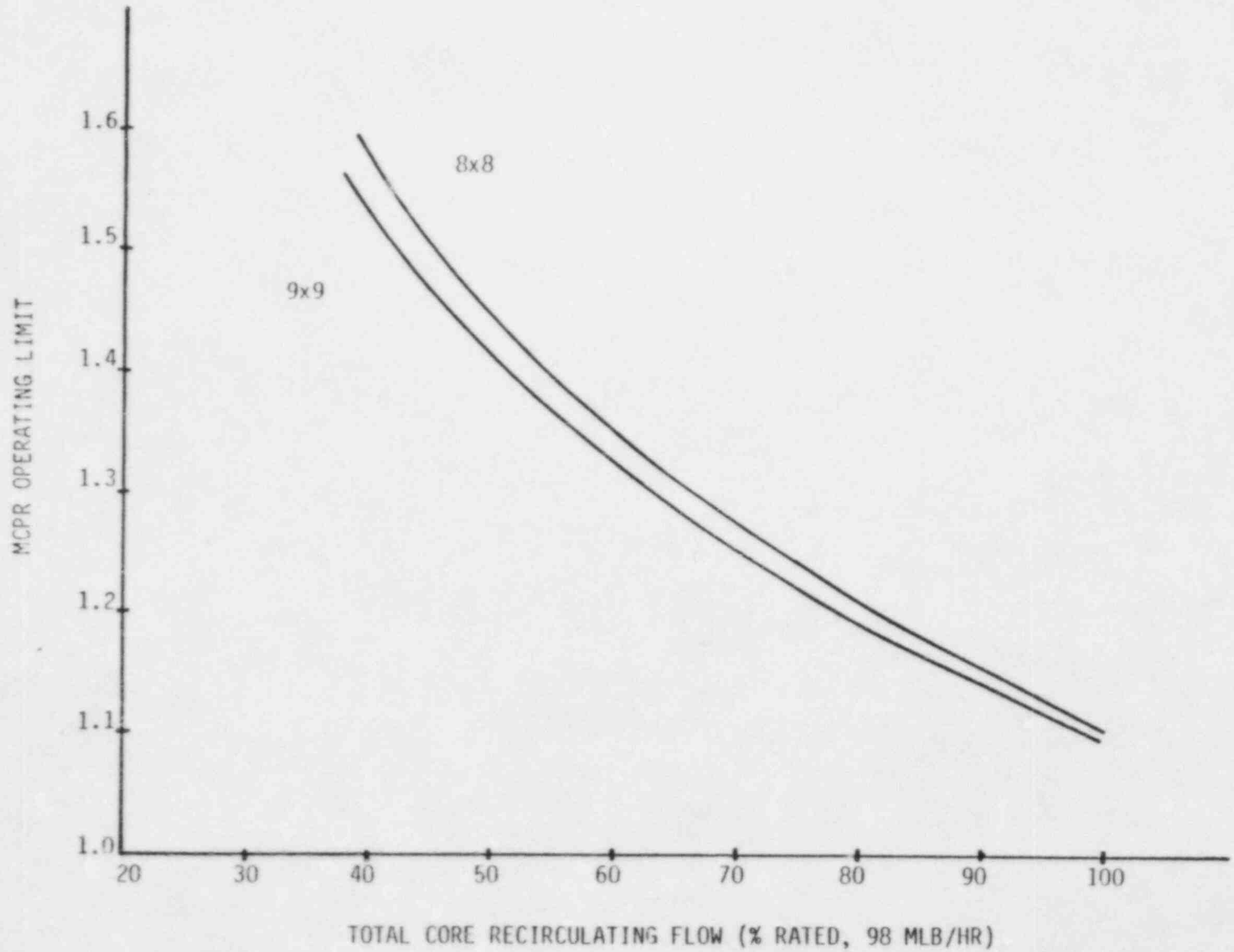


Figure 7.1 Reduced Flow MCPR Tech. Spec. Limit - All Conditions

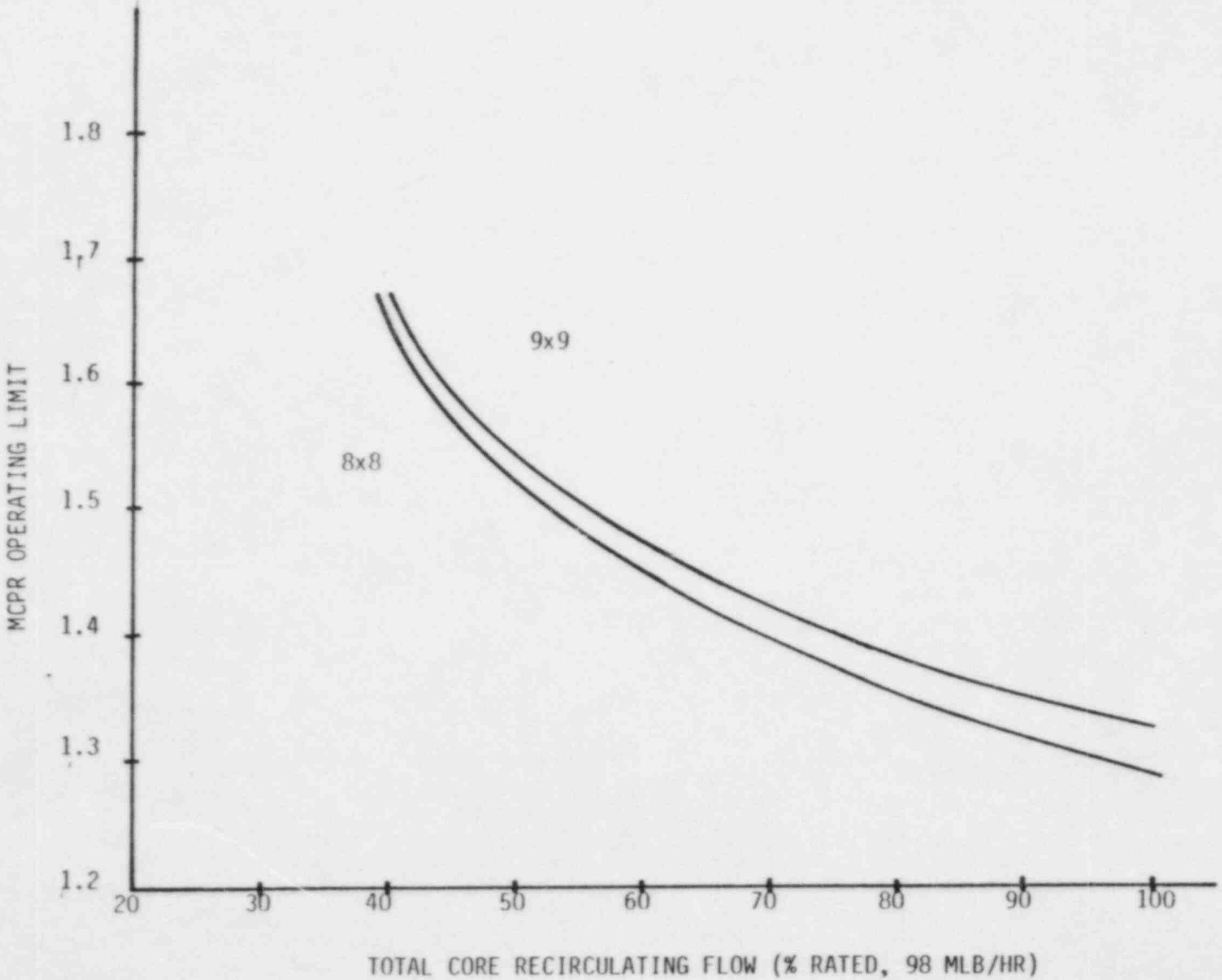


Figure 7.2 Reduced Flow MCPR Tech. Spec. Limit - Automatic Flow Control

APPENDIX A

SINGLE LOOP OPERATION

The NSSS supplier has provided analyses which demonstrate the safety of plant operation with a single recirculation loop out of service for an extended period of time. These analyses restrict the overall operation of the plant to lower bundle power levels and lower nodal power levels than are allowed 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 ENC fuel was designed to be compatible with the coresident fuel in thermal hydraulic, nuclear, and mechanical design performance, and because the ENC 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 ENC.

With a single recirculation loop in operation, the NSSS supplier's analyses support continued operation with an increase of 0.01 in the MCPR safety limit. Because of the similarity between the ENC XN-3 9x9 fuel type and the other fuel types making up the remainder of the core, and because of the similarity in the magnitude of the uncertainties which determine the MCPR safety limit, this small increase in the safety limit value is also appropriate for operation with ENC fuel and analyses. For Cycle 10 operation with both recirculation loops in operation, the MCPR safety limit is 1.05, which is the same value as was used for the previous cycle. For Cycle 10 operation with a single recirculation loop in service, the MCPR safety limit is 1.06, which is also the same value as was used for the previous cycle.

The consequences of core-wide transients at the reduced power and flow conditions necessitated by single loop operation are bounded by the consequences of these events at rated conditions. The additional conservatism imposed by the reduced flow MCPR operating limits specified in the main body of this report assures that the MCPR safety limit will not be violated during anticipated operational occurrences with a single recirculation loop in service. No modification to the delta-CPR defining the rated conditions MCPR operating limit is required, and the reduced flow MCPR limit curve remains conservatively applicable during single loop operation. Because the reduced flow MCPR limit curves are based on equipment performance which physically cannot happen during single loop operation, the added conservatism present in the curves compensates for the penalties associated with increased uncertainties in the MCPR safety limit and control rod drive performance. The reduced flow MCPR limit curves are applicable without modification during single loop operation.

The stability characteristics of the Cycle 10 core are equivalent to or better than those of the previous cycle core. Reactor operation within the limitations which assured adequate stability for the previous cycle will continue to assure adequate stability for Cycle 10. In addition, the stability analyses reported in Section 4.2.4 of the main body of this report cover the operating region of Single Loop Operation. The calculated decay ratio is 0.53, which is within the appropriate acceptance criterion for Single Loop Operation.

The NSSS supplier justified MAPLHGR reduction factors for resident fuel types 8x8, 8x8R, and P8x8R during single loop operation. Because the ENC 8x8 fuel is very similar to the P8x8R fuel in design and operational characteristics, the MAPLHGR reduction factors defined by P8x8R fuel are applicable to the ENC 8x8 fuel for single loop operation. The LOCA analyses performed in support of the Cycles 8 and 10 reloads demonstrate a similarity in ECCS performance for cores fueled with 8x8 and 9x9 fuel, with the 9x9 fuel exhibiting a consistently lower value of Peak Cladding Temperature (PCT). Because the LOCA performance of the various fuel types is similar and because the 9x9 fuel exhibits a consistently lower PCT, the MAPLHGR reduction factors for 8x8 fuel are conservatively applicable to 9x9 fuel.

APPENDIX B

FORMULATION OF MCPR LIMITS

The original Exxon Nuclear analyses performed in support of operation of Dresden Unit 3 as reported in XN-NF-81-76 defined thermal margin requirements in terms of a minimum critical power ratio (MCPR) as formulated in XN-NF-524, Revision 0. This MCPR formulation contained a flow iteration during the calculation of the critical heat flux associated with any given fuel operating state. During the NRC Staff review of XN-NF-524 and the Dresden license amendment application, it was determined that differences between the proposed ENC formulation of MCPR and the existing formulation in the Dresden design basis were unnecessarily confusing. Cycle 8 and 9 operation of Dresden Unit 3 and Cycle 9 and 10 operation of Dresden Unit 2 were authorized on the basis of this flow-iterative formulation of MCPR, but the topical report, XN-NF-524, was revised (Revision 1) to incorporate a constant flow MCPR formulation consistent with the previous design basis. Revision 1 of XN-NF-524 was approved generically for BWR applications, and Dresden continued to operate with the flow-iterative MCPR under the provisions of 10 CFR 50.59(a)(1).

The introduction of the ENC XN-3 9x9 fuel necessitated a revision to the plant Technical Specifications in order to specify operating limits for the new fuel type. At this time, new, constant flow MCPR operating limits are provided for all incore fuel types consistent with the approved ENC methodology in XN-NF-524(A), Revision 1. All MCPR limits reported in this document are quoted on a constant flow basis.

APPENDIX C

RECIRCULATION PIPE REPLACEMENT PROGRAM

Following the completion of Cycle 9 operation, the Dresden Unit 3 plant was altered to replace cracked stainless steel piping in the recirculation systems. Although the original plant configuration was followed closely, there were some differences in dimensions between the original piping and the replacement piping.

Exxon Nuclear has evaluated the geometric differences between the original and replacement piping. The contained volumes and piping spool configurations are sufficiently similar that the results of earlier analyses performed by Exxon Nuclear based on the original plant configuration are valid within the reporting accuracy of the design basis documents.

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