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ADVANCED NUCLEAR FUELS CORPORATION

SUSQUEHANNA UNIT 2 CYCLE 3 RELOAD ANALYSIS DESIGN AND SAFETY ANALYSES.

NOVEMBER 1987

AN AFFILIATE OF KRAFTWERK UNION
KWU

ADVANCED NUCLEAR FUELS CORPORATION

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SUSQUEHANNA UNIT 2 CYCLE 3 RELOAD ANALYSIS

Design and Safety Analyses

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AN AFFILIATE OF KRAFTWERK UNION
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1.0 INTRODUCTION

This report provides the results of the analyses performed by Advanced Nuclear Fuels Corporation (ANF)* in support of the Cycle 3 reload for Susquehanna Unit 2, which is scheduled to commence operation in the spring of 1988. This report is intended to be used in conjunction with ANF topical report XN-NF-80-19(P)(A), Volume 4, Revision 1, "Application of the Exxon Nuclear Company 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. However, LHGR mechanical design limits (Reference 9.1) and plant transient simulation model developments (Reference 9.2) have been revised by ANF subsequent to NRC approval of XN-NF-80-19(P)(A), Volume 4, Revision 1. Both References 9.1 and 9.2 have been approved by the NRC for use in referencing in license applications. Section numbers in this report are the same as corresponding section numbers in XN-NF-80-19(P)(A), Volume 4, Revision 1.

The Susquehanna Unit 2 Cycle 3 core will comprise a total of 764 fuel assemblies, including 236 unirradiated ANF XN-2 9x9 assemblies, 324 irradiated ANF XN-1 9x9 assemblies, 112 irradiated General Electric 8x8R fuel assemblies (central region), and 92 irradiated GE 8x8R assemblies in the peripheral region. 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 Susquehanna Unit 2 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 9.3.

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*Formerly Exxon Nuclear Company (ENC).

2.0 FUEL MECHANICAL DESIGN ANALYSIS

Applicable ANF Fuel Design Report: Reference 9.1

To assure that the expected power history for the fuels to be irradiated during Cycle 3 of Susquehanna Unit 2 is bounded by the assumed power history in the fuel mechanical design analysis, LHGR operating limits (Figure 3.3 of Reference 9.1) have been specified. In addition, an LHGR transient operating limit for Anticipated Operating Occurrences (Figure 3.4 of Reference 9.1) has been specified for ANF 9x9 fuel. Additional information on rod bow, as requested in the NRC's safety evaluation report for Reference 9.1, has been transmitted in Reference 9.4.

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 Susquehanna Unit 2 Cycle 3 core have been determined in single phase flow tests of full scale assemblies. Figure 3.1 shows the hydraulic demand curves for ANF 9x9 fuel and GE 8x8R fuel in the Susquehanna Unit 2 core. The similar hydraulic performance indicates compatibility for co-residence in the Susquehanna Unit 2 core.

3.2.3 Fuel Centerline Temperature

Applicable Generic Report	Reference 9.1

3.2.5 Bypass Flow

Calculated Bypass Flow Fraction at 104% Power/100% Flow	10.1%

3.3 MCPR Fuel Cladding Integrity Safety Limit

Safety Limit MCPR = 1.06

3.3.1 Coolant Thermodynamic Condition

Rated Thermal Power	3293 MWt
Feedwater Flowrate (at SLMCPR)	16.1 Mlbm/hr
Core Pressure (at SLMCPR)	1042.9 psia
Feedwater Temperature	383°F

3.3.2 <u>Design Basis Radial Power Distribution</u>

See Figure 3.2

3.3.3 <u>Design Basis Local Power Distribution</u>

See Figures 3.3 through 3.6

4.0 NUCLEAR DESIGN ANALYSIS

4.1 Fuel Bundle Nuclear Design Analysis

Assembly Average Enrichment	3.33%
Radial Enrichment Distribution	Figure 4.1 and 4.2
Axial Enrichment Distribution	Uniform 3.44% with 6" natural uranium top blanket
Burnable Poisons	Figure 4.1 and 4.2

Note: Burnable poisons are distributed uniformly over the enriched length of the designated rods. The natural urania axial blanket sections do not contain burnable absorber material.

Non-Fueled Rods	Figure 4.1 and 4.2
Neutronic Design Parameters	Table 4.1

4.2 Core Nuclear Design Analysis

4.2.1 Core Configuration

Figure 4.3

Core Exposure at EOC2, MWd/MTU	18350.7
Core Exposure at BOC3, MWd/MTU	10911.2
Core Exposure at EOC3, MWd/MTU	21740.8
Maximum Cycle 3 Licensing Exposure Limit, MWd/MTU	22076

4.2.2 Core Reactivity Characteristics

BOC Cold K-effective, All Rods Out	1.11353
BOC Cold K-effective, Strongest Rod Out	0.98524
Reactivity Defect (R-Value)	0.00% rho
Standby Liquid Control System Reactivity, Cold Conditions, 660 ppm	0.98348

4.2.4 Core Hydrodynamic Stability

Power/Flow Map — Figure 4.4

Power/Flow State Points	Decay Ratio (COTRAN)
64/42*	0.82
69/47**	0.75
66/45**	0.75

*Two pump minimum flow - APRM Rod Block intercept point. Extended operation at lower flow is not allowed by Technical Specifications.

**Operation at less than 45% flow requires APRM/LPRM surveillance. In addition, operation at power/flow combinations above and to the left of the line connecting these two points requires APRM/LPRM surveillance. See Figure 4.4.

5.0 ANTICIPATED OPERATIONAL OCCURRENCES

Applicable Generic Transient
Analysis Methodology Report — References 9.5 & 9.7

5.1 Analysis Of Plant Transients At Rated Conditions — Reference 9.6

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

Event	Power*	Flow	% Rated Maximum Heat Flux	% Rated Maximum Power	Maximum Pressure (psia)	Delta CPR**	Model
LRWB	100%	100%	116.2%	267%	1194	0.24	COTRANSA/ XCOBRA-T
FWCF	100%	100%	116.8%	233%	1179	0.23	COTRANSA/ XCOBRA-T
LFWH	100%	100%	121.3%	123%	1078	0.16	PTSBWR3/ XCOBRA

Single Loop Operation: Appendix A

5.2 Analyses For Reduced Flow Operation — Reference 9.6

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

*104% power used in analysis as design bases.

**Delta-CPR results for most limiting fuel type.

5.3 Analyses For Reduced Power Operation — Reference 9.6

Limiting Transient(s): Feedwater Controller Failure (FWCF)

% Power	Transient	Delta CPR ANF 9x9	Delta CPR GE 8x8R
104	FWCF	0.23	0.20
80	FWCF	0.25	0.23
65	FWCF	0.28	0.26
40	FWCF	0.31	0.28

5.4 ASME Overpressurization Analysis — Reference 9.6

Limiting Event	Full MSIV Isolation
Worst Single Failure	Direct Scram
Maximum Pressure	1297 psig
Maximum Steam Dome Pressure	1281 psig

5.5 Control Rod Withdrawal Error (CRWE)

Starting Control Rod Pattern for Analysis — Figure 5.1

Rod Block Setting	100% Flow: Distance Withdrawn (ft)	100% Flow: Delta CPR
105	4.0	0.22
106*	4.5	0.24
107	5.0	0.26
108*	5.0	0.26

---

*Rod Block Monitor settings recommended for Cycle 3 operation.

5.6 Fuel Loading Error

Maximum Delta CPR 0.16

5.7 Determination Of Thermal Margins

Summary of Thermal Margin Requirements

Event	Power	Flow	Delta CPR*	MCPR Limit
LRWB	100%**	100%	0.24	1.30
FWCF	100%**	100%	0.23	1.29
LFWH	100%**	100%	0.16	1.22
CRWE	100%	100%	0.24 at 106% RBM 0.26 at 108% RBM	1.30 1.32

MCPR Operating Limits at Rated Conditions

MCPR Operating Limit
1.30 at 106% RBM
1.32 at 108% RBM

Reduced Flow MCPR Limits Figure 5.2

Power Dependent MCPR Operating Limit Results for Cycle 3:

% Power/% Flow	Limiting Transient	ANF 9x9	GE 8x8R
100**/100	LRWB	1.30	1.27
80/100	FWCF	1.31	1.29
65/100	FWCF	1.34	1.32
40/100	FWCF	1.37	1.34

*Delta CPR results for most limiting fuel type.

**104% power used in analysis as design bases.

6.0 POSTULATED ACCIDENTS

6.1 Loss-Of-Coolant Accident

Seismic-LOCA: Appendix B

6.1.1 Break Location Spectrum
Reference 9.8

6.1.2 Break Size Spectrum
Reference 9.8

6.1.3 MAPLHGR Analyses

ANF 9x9 Fuel Reference 9.9

Limiting Break: Double-ended guillotine pipe break
Recirculation pump discharge line
0.4 Discharge Coefficient

Bundle Average Exposure (GWD/MTU)	MAPLHGR (kw/ft)	Peak Clad Temperature* (Degree F)	Peak Local MWR** (Percent)
0	10.2	2060	3.9
5	10.2	2069	3.7
10	10.2	2121	3.7
15	10.2	2140	4.8
20	10.2	2147	5.2
25	9.6	2016	2.7
30	8.9	1839	1.0
35	8.2	1752	0.7
40	7.5	1676	0.5

*Peak clad temperatures for XN-1 and XN-2 fuel are bounded by these results.

**Metal Water Reaction.

6.2 <u>Control Rod Drop Accident</u> Section 8.0

Dropped Control Rod Worth, mk	13.5
Doppler Coefficient, 1/k dk/dT	$-10.6 \times (10)^{-6}$
Effective Delayed Neutron Fraction	0.0058
Four-Bundle Local Peaking Factor	1.34
Maximum Deposited Fuel Rod Enthalpy, cal/gm	205
Number of Rods Exceeding 170 cal/gm	<250

7.0 TECHNICAL SPECIFICATIONS

7.1 Limiting Safety System Settings

7.1.1 MCPR Fuel Cladding Integrity Safety Limit

MCPR Safety Limit 1.06

7.1.2 Steam Dome Pressure Safety Limit

Pressure Safety Limit (as measured in steam dome) 1325 psig

Analysis shows that a steam dome pressure safety limit of 1358 psig is allowed but the 1325 psig value used in Cycle 2 is to be conservatively retained.

7.2 Limiting Conditions For Operation

7.2.1 Average Planar Linear Heat Generation Rate Limits

Bundle Average Exposure (GWD/MT)	MAPLHGR Limits (kw/ft) ANF 9x9 Fuel
0	10.2
5	10.2
10	10.2
15	10.2
20	10.2
25	9.6
30	8.9
35	8.2
40	7.5

7.2.2 Minimum Critical Power Ratio

MCPR Operating Limits at Rated Conditions:

MCPR Operating Limit

1.30 at 106% RBM
1.32 at 108% RBM

MCPR Operating Limits at Off-Rated Conditions:

At Reduced Flow — Figure 5.2

Total Core Recirculation Flow (% Rated)	Reduced Flow MCPR Operating Limit
100	1.12
96	1.14
92	1.16
83	1.20
76	1.23
60	1.31
50	1.44
40	1.61

At Reduced Power

Power Level (% Rated)	Reduced Power MCPR Operating Limit
100*	1.30
80	1.31
65	1.34
40	1.37

*104% power used in analysis as design bases.

7.2.3 LHGR Limits

LHGR Limits	Figures 3.3 and 3.4 of Reference 9.1

7.3 Surveillance Requirements

7.3.1 Scram Insertion Time Surveillance

Thermal limits established in Section 5.0 are based on minimum acceptable scram insertion performance as defined in the Technical Specifications. No additional surveillance for scram insertion is required for validation of thermal limits.

7.3.2 Stability Surveillance

Power/Flow Map	Figure 4.4

The Unit 2 Cycle 2 Technical Specifications require APRM/LPRM surveillance to the left of the 45% Constant Flow line and above the 80% Rod Block line. Based on core hydrodynamic stability analyses, operation at power/flow combinations above and to the left of the line connecting the 66% Power/45% Flow and 69% Power/47% Flow points but below the APRM Rod Block line needs to be added to the APRM/LPRM surveillance requirement (see Section 4.2.4).

8.0 METHODOLOGY REFERENCES

See XN-NF-80-19(P)(A), Volume 4, Revision 1 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)(A), Rev. 1, Advanced Nuclear Fuels Corporation*, Richland, Washington, September 4, 1986.

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

9.3 "Demonstration of 9x9 Assemblies for BWRs," EPRI NP-3468, Electric Power Research Institute, Palo Alto, California, May 1, 1984.

9.4 Letter, G. N. Ward (ANF) to G. C. Lainas (NRC), "Additional Information on Rod Bow," serial no. GNW:021:87, dated March 11, 1987.

9.5 "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," XN-NF-79-71(P), Revision 2, Advanced Nuclear Fuels Corporation, Richland, Washington, November 16, 1981.

9.6 "Susquehanna Unit 2 Cycle 3 Plant Transient Analysis," ANF-87-125, Rev. 2, Advanced Nuclear Fuels Corporation, Richland, Washington, November 1987.

9.7 "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," XN-NF-84-105(P)(A), Volume 1 and Volume 1, Supplements 1 and 2, Advanced Nuclear Fuels Corporation, Richland, Washington, February 1987.

9.8 "Generic LOCA Break Spectrum Analysis BWR 3 & 4 with Modified Low Pressure Coolant Injection Logic Using the EXEM Evaluation Model," XN-NF-84-117(P), Advanced Nuclear Fuels Corporation, Richland, Washington, December 1984.

9.9 "Susquehanna LOCA-ECCS Analysis MAPLHGR Results for ENC 9x9 Fuel," XN-NF-86-65, Advanced Nuclear Fuels Corporation, Richland, Washington, May 1986.

9.10 "Principal Reload Fuel Design Parameters, Fuel Design, Susquehanna Unit 2 Reload XN-2," XN-NF-1058, Advanced Nuclear Fuels Corporation, Richland, Washington, March 1987.

*Formerly Exxon Nuclear Company.

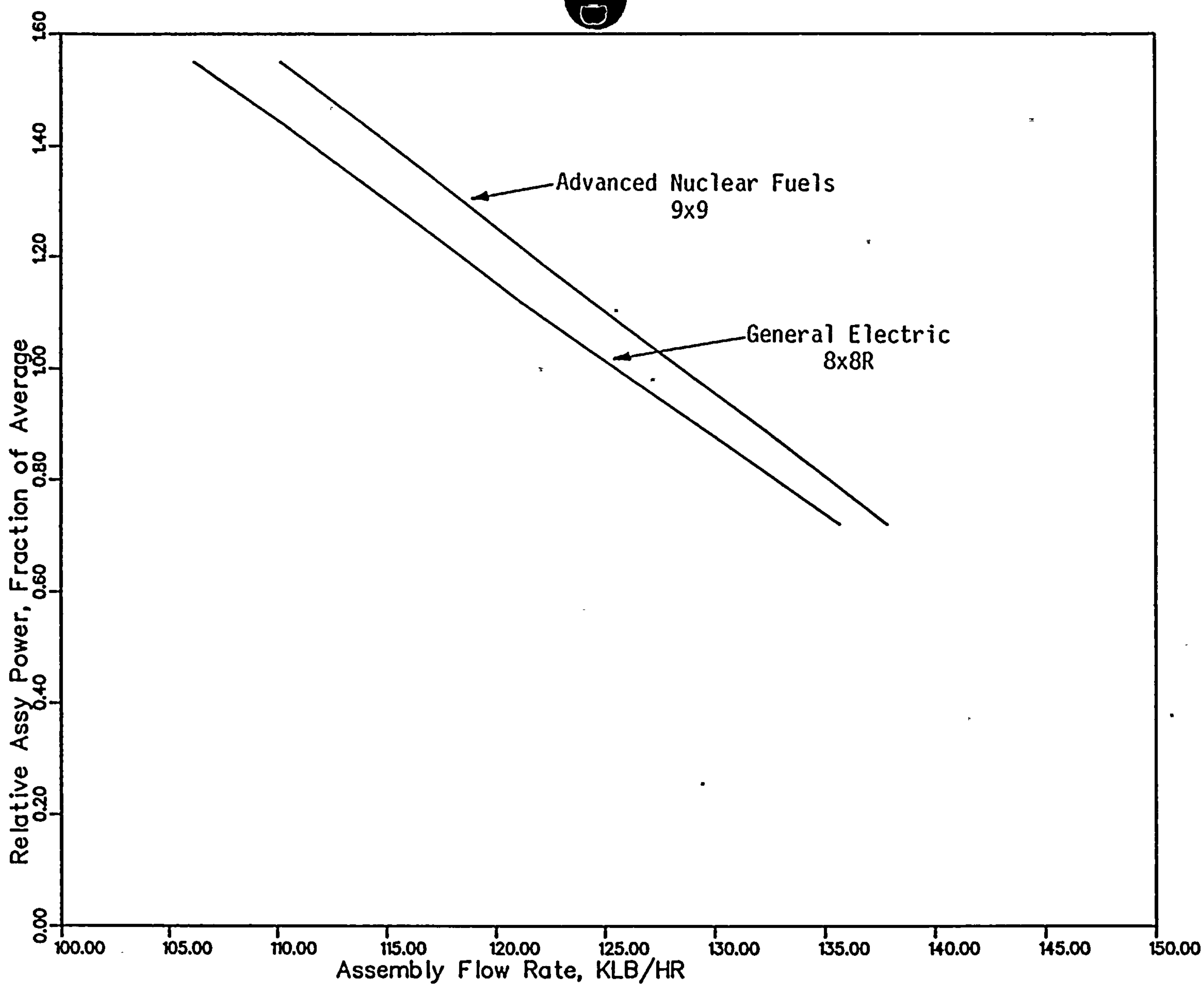

Figure 3.1 Susquehanna Unit 2 Cycle 3 Hydraulic Demand Curve
Power vs. Flow

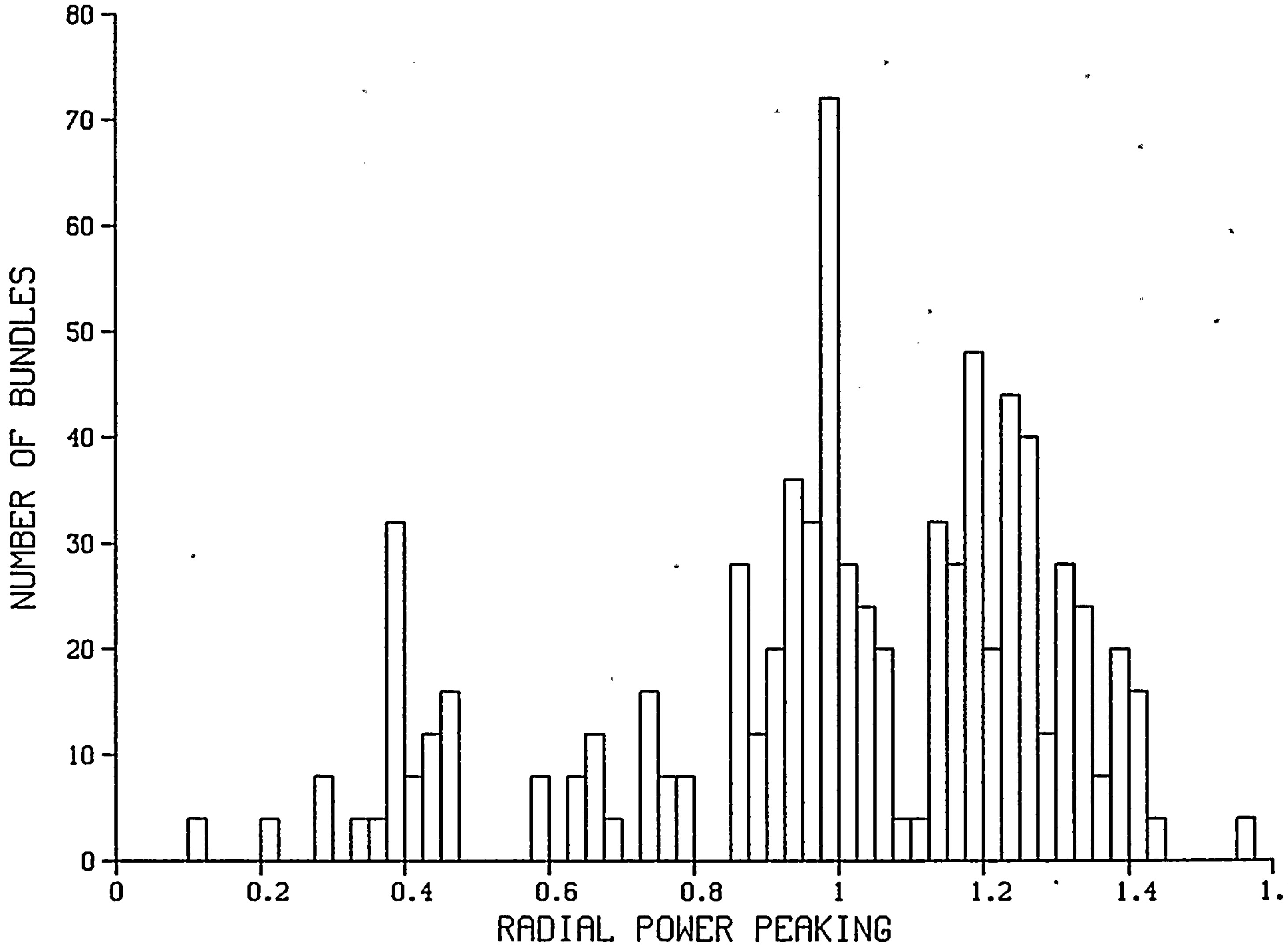

Figure 3.2 Susquehanna [illegible] 2 Cycle 3 Design Basis Radial Power

0.88	0.91	0.96	1.04	1.02	1.04	0.96	1.00	0.96
0.91	0.93	0.98	1.07	0.91	1.07	0.97	1.04	1.01
0.96	0.98	0.90	1.04	1.03	1.04	1.04	0.99	0.96
1.04	1.07	1.04	1.00	0.99	1.00	1.05	0.94	1.04
1.02	0.91	1.03	0.99	0.00	0.98	1.05	1.07	1.04
1.04	1.07	1.04	1.00	0.98	0.00	1.03	0.94	1.05
0.96	0.97	1.04	1.05	1.05	1.03	1.06	1.00	0.97
1.00	1.04	0.99	0.94	1.07	0.94	1.00	0.94	1.01
0.96	1.01	0.96	1.04	1.04	1.05	0.97	1.01	0.97

Figure 3.3
Design Basis Local Power Distribution
Advanced Nuclear Fuels XN-2 9X9 Fuel

0.91	0.92	0.95	1.01	1.01	1.01	0.96	0.98	0.95
0.92	0.94	0.98	0.97	1.05	0.95	0.99	0.95	0.98
0.95	0.98	0.93	1.06	1.05	1.06	1.05	0.97	0.96
1.01	0.97	1.06	1.03	1.03	1.04	1.07	1.06	1.02
1.01	1.05	1.05	1.03	0.00	1.01	1.07	1.06	1.01
1.01	0.95	1.06	1.04	1.01	0.00	1.04	0.96	1.02
0.96	0.99	1.05	1.07	1.07	1.04	1.06	1.00	0.96
0.98	0.95	0.97	1.06	1.06	0.96	1.00	0.95	0.98
0.95	0.98	0.96	1.02	1.01	1.02	0.96	0.98	0.96

Figure 3.4
Design Basis Local Power Distribution
Advanced Nuclear Fuels XN-1 9X9 Fuel

1.03	1.00	1.00	1.00	1.00	1.00	1.01	1.03
1.00	0.98	1.00	1.02	1.02	1.03	1.00	1.01
1.00	1.00	1.01	1.01	1.01	0.90	1.03	1.00
1.00	1.02	1.01	0.89	0.00	1.01	1.02	1.00
1.00	1.02	1.01	0.00	0.89	1.01	0.99	1.00
1.00	1.03	0.90	1.01	1.01	0.98	1.00	1.00
1.01	1.00	1.03	1.02	0.99	1.00	0.98	1.00
1.03	1.01	1.00	1.00	1.00	1.00	1.00	1.03

Figure 3.5
Design Basis Local Power Distribution
General Electric (Central) 8X8R Fuel

1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
1.00	1.00	1.00	1.00	0.00	1.00	1.00	1.00
1.00	1.00	1.00	0.00	1.00	1.00	1.00	1.00
1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

Figure 3.6
Design Basis Local Power Distribution
General Electric (Peripheral) 8X8R Fuel

TABLE 4.1 NEUTRONIC DESIGN VALUES

Fuel Pellet	Reference 9.10
Fuel Rod	Reference 9.10
Fuel Assembly	Reference 9.10
Core Data	
Number of fuel assemblies	764
Rated thermal power, MW	3293
Rated core flow, Mlbm/hr	100
Core inlet subcooling, Btu/lbm	24.0
Moderator temperature, F	548.8
Channel thickness, inch	.080
Fuel assembly pitch, inch	6.00
Wide water gap thickness, inch	0.562
Narrow water gap thickness, inch	0.562
Control Rod Data	
Absorber material	B_4C
Total blade span, inch	9.75
Total blade support span, inch	1.58
Blade thickness, inch	0.260
Blade face-to-face internal dimension, inch	0.200
Absorber rods per blade	76
Absorber rod outside diameter, inch	0.188
Absorber rod inside diameter, inch	0.138
Absorber density, % of theoretical	70.0

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

```
LL RODS ( 1)  ---  1.45 W/O U235
 L RODS ( 5)  ---  1.95 W/O U235
ML RODS (16)  ---  2.55 W/O U235
 M RODS (20)  ---  3.27 W/O U235
MH RODS (13)  ---  4.23 W/O U235
 H RODS (15)  ---  4.66 W/O U235
M* RODS ( 9)  ---  3.27 W/O U235 + 4.00 W/O GD203
W  RODS ( 2)  ---  INERT WATER ROD
```

Figure 4.1 Susquehanna Unit 2 Cycle 3 Enrichment Distribution for the ANF92-344L-9G4 XN-2 Fuel Lattice

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

LL RODS (1) --- 1.45 W/O U235
L RODS (5) --- 1.95 W/O U235
ML RODS (16) --- 2.55 W/O U235
M RODS (19) --- 3.27 W/O U235
MH RODS (13) --- 4.23 W/O U235
H RODS (15) --- 4.66 W/O U235
M* RODS (10) --- 3.27 W/O U235 + 5.00 W/O GD203
W RODS (2) --- INERT WATER ROD

Figure 4.2 Susquehanna Unit 2 Cycle 3 Enrichment Distribution for the ANF92-344L-10G5 XN-2 Fuel Lattice

A2	C1	A2	C1	A2	C1	A2	C1	D0	C1	A2	C1	E0	C1	A2
C1	D0	C1	D0	C1	A2	C1	D0	C1	A2	C1	E0	C1	C1	A2
A2	C1	D0	A2	D0	C1	D0	A2	D0	C1	D0	C1	E0	C1	A2
C1	D0	A2	D0	C1	D0	C1	D0	A2	D0	C1	E0	C1	C1	A2
A2	C1	D0	C1	A2	C1	D0	C1	D0	A2	D0	C1	E0	C1	A2
C1	A2	C1	D0	C1	C1	A2	D0	A2	E0	C1	E0	C1	C1	A2
A2	C1	D0	C1	D0	A2	C1	C1	D0	C1	E0	C1	E0	C1	A2
C1	D0	A2	D0	C1	D0	C1	E0	C1	E0	C1	D0	C1	A2	
D0	C1	D0	A2	D0	A2	D0	C1	E0	C1	E0	A2	B2		
C1	A2	C1	D0	A2	E0	C1	E0	C1	C1	C1	A2	A2		
A2	C1	D0	C1	D0	C1	E0	C1	E0	C1	A2				
C1	E0	C1	E0	C1	E0	C1	D0	A2	A2					
E0	C1	E0	C1	E0	C1	E0	C1	B2	A2					
C1	C1	C1	C1	C1	C1	C1	A2							
A2	A2	A2	A2	A2	A2	A2								

XY = Fuel Type X
Burned Y Cycles

Fuel Type	No. of Bundles	Description
A	196	GE 8X8 Type III 2.19 w/o U-235
B	8	GE 8X8 Type II 1.76 w/o U-235
C	324	XN-1 ENC92-331B-7G4
D	140	XN-2 ANF92-333B-9G4
E	96	XN-2 ANF92-333B-10G5

Figure 4.3 Susquehanna Unit 2 Cycle 3 Reference Core Loading

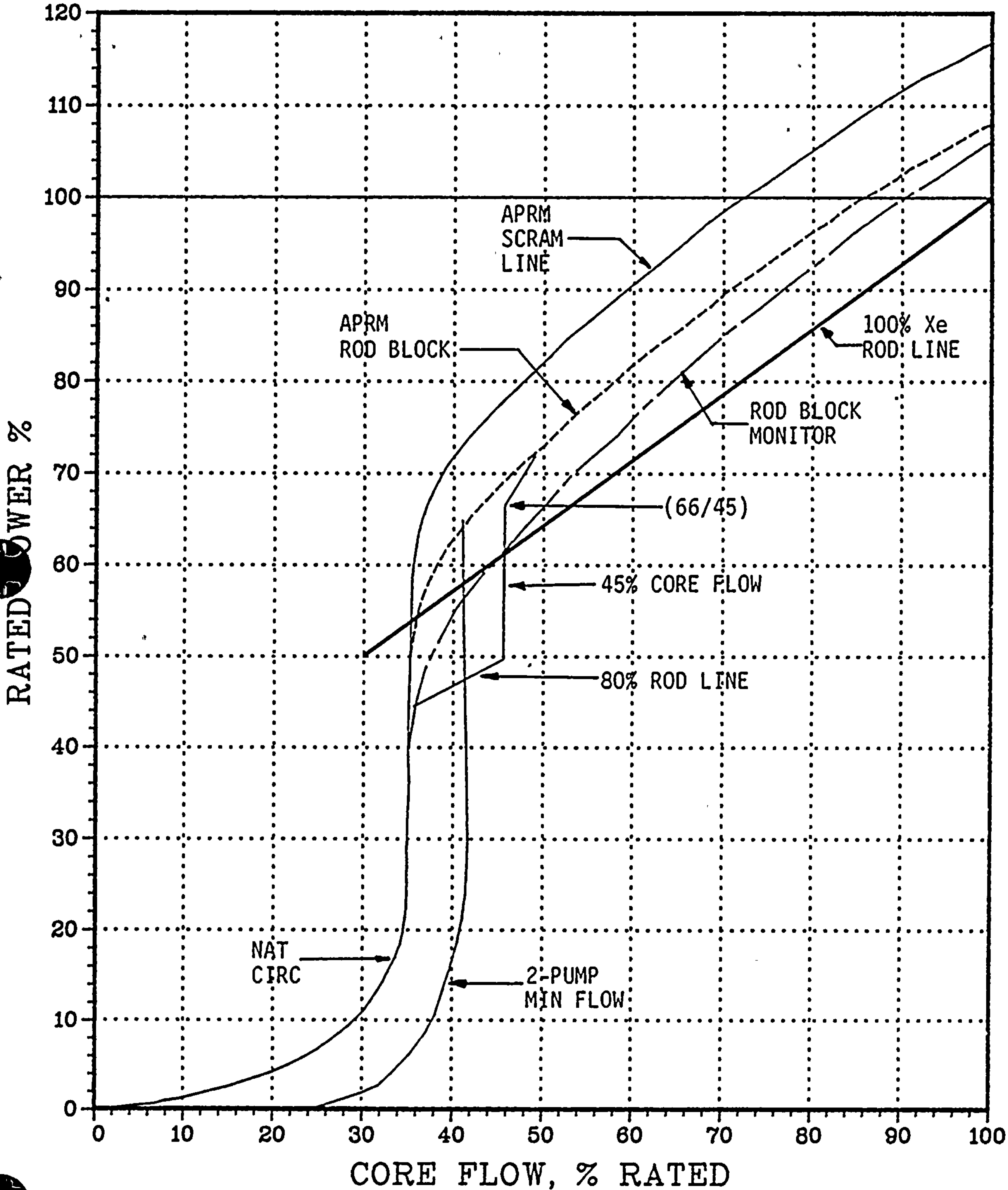

Figure 4.4 Susquehanna Unit 2 Cycle 3 - Core Power vs. Core Flow

	2	6	10	14	18	22	26	30	34	38	42	46	50	54	58	
59					--	--	--	--	--	--	--					59
55				--	--	12	--	00	--	12	--	--				55
51			--	--	20	--	26	--	26	--	20	--	--			51
47		--	--	00	--	12	--	08	--	12	--	00	--	--		47
43	--	--	20	--	20	--	--	--	--	--	20	--	20	--	--	43
39	--	12	--	08	--	08	--	00	--	08	--	08	--	12	--	39
35	--	--	26	--	--	--	44	--	44	--	--	--	26	--	--	35
31	--	00	--	04	--	00	--	00	--	00	--	04	--	00	--	31
27	--	--	26	--	--	--	44	--	44	--	--	--	26	--	--	27
23	--	12	--	08	--	08	--	00*	--	08	--	08	--	12	--	23
19	--	--	20	--	20	--	--	--	--	--	20	--	20	--	--	19
15		--	--	00	--	12	--	08	--	12	--	00	--	--		15
11			--	--	20	--	26	--	26	--	20	--	--			11
7				--	--	12	--	00	--	12	--	--				7
3					--	--	--	--	--	--	--					3
	2	6	10	14	18	22	26	30	34	38	42	46	50	54	58	

Cycle Exposure 0.0 MWD/MTU
Control Rod Density 23.3 %

Control Rod Being Withdrawn = 00*
Rod Fully Inserted = 00
Rod Fully Withdrawn = --

Figure 5.1 Susquehanna Unit 2 Cycle 3 Control Rod Withdrawal Error Analysis Limiting Initial Control Rod Pattern

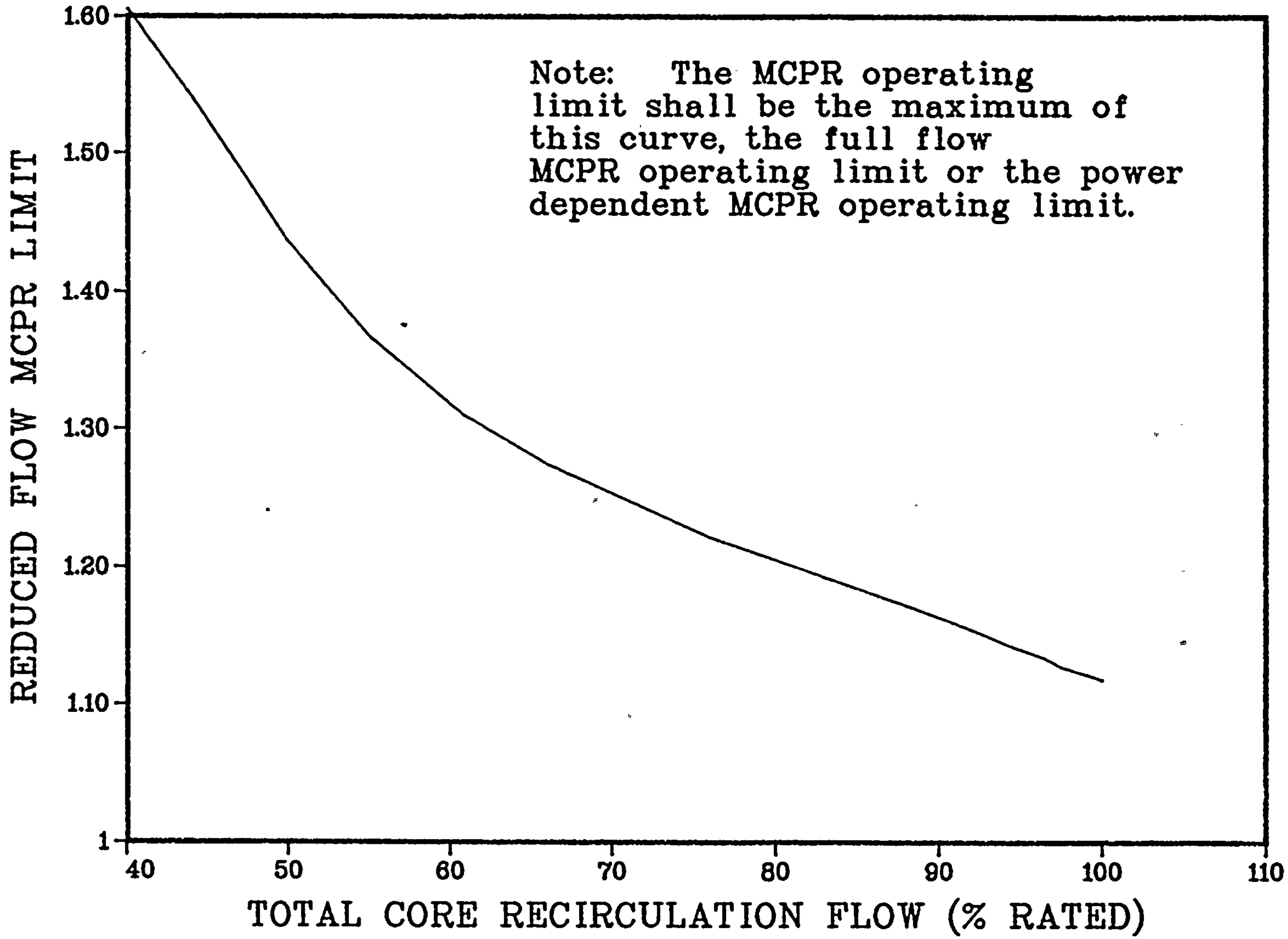

Figure 5.2 Susquehanna Unit 2 Cycle 3 Flow MCPR Operating Limit

APPENDIX A

SINGLE LOOP OPERATION

This Appendix provides limits and justification of those limits for Single Loop Operation (SLO).

A.1 ANTICIPATED OPERATIONAL OCCURRENCES

Reference A.1

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. ANF fuel was designed to be compatible with the co-resident fuel in thermal hydraulic, nuclear, and mechanical design performance. The ANF methodology has given results which are consistent with those of previous analyses for normal two-loop operation. Many analyses performed by the NSSS supplier for single loop operation are also applicable to single loop operation with fuel and analyses provided by ANF.

For single loop operation, the NSSS vendor found that an increase of 0.01 in the MCPR safety limit was needed to account for the increased flow measurement uncertainties and increased tip uncertainties associated with single pump operation. ANF has evaluated the effects of the increased flow measurement uncertainties on the safety limit MCPR and found that the NSSS vendor determined increase in the allowed safety limit MCPR is also applicable to ANF fuel during single loop operation. Thus, increasing the safety limit MCPR by 0.01 for single loop operation (1.07) with ANF fuel is sufficiently conservative to also bound the increased flow measurement uncertainties for single loop operation.

The limiting MCPR operating limit for single loop operation is conservatively set using the limiting pump seizure accident delta CPR plus the single loop operation MCPR safety limit. This limit together with the $MCPR_f$ curve for two loop operation plus .01 and the $MCPR_p$ curve for two loop operation plus .01 conservatively bound all transients.

The Technical Specifications require APRM/LPRM surveillance to the left of the 45% Constant Flow line and above the 80% Rod Block line. Based on core hydrodynamic stability analyses for Cycle 3, operation at power/flow combinations above and to the left of the line connecting the 66% Power/45% Flow and 69% Power/47% Flow points needs to be added to the APRM/LPRM surveillance requirements. Figure 4.4 shows the core power versus core flow established for Cycle 3.

A.2 POSTULATED ACCIDENTS

Reference A.2

ANF performed LOCA analyses for single loop conditions and has determined that the MAPLHGR limit curve (Section 7.2) for two-loop operation is also applicable to single loop operation for ANF 9x9 fuels.

REFERENCES

A.1 "Susquehanna Unit 2 Cycle 2 Single Loop Operation Analysis," XN-NF-86-146, Advanced Nuclear Fuels Corporation, Richland, WA 99352, November 1986.

A.2 "Susquehanna LOCA Analysis for Single Loop Operation," XN-NF-86-125, Advanced Nuclear Fuels Corporation, Richland, WA 99352, November 1986.

APPENDIX B

SEISMIC-LOCA EVALUATION

The structural response of Advanced Nuclear Fuels Corporation's (ANF's) 9x9 fuel is similar to the structural response of the GE 8x8R fuel it replaces in the Susquehanna Unit 2 core. Therefore, the seismic-LOCA structural response evaluation performed in support of the initial core remains applicable and continues to provide assurance that control blade insertion will not be inhibited following the occurrence of the design basis seismic-LOCA event.

The physical and structural properties of the 9x9 and the 8x8 fuel types which are important to the dynamic response of the fuel are summarized in Table B.1. The close agreement between the important parameters for the ANF 9x9 and GE 8x8R fuel types indicates that the structural response would be very similar for both fuel types.

Similarity in the natural frequencies of the two fuel types mentioned above is further assured by the stiffness of the fuel assembly channel box. Both fuel types use the same fuel assembly channel box, and the channel box dominates the overall dynamic response of the incore fuel. ANF calculations show that approximately 97% of the stiffness of a fuel assembly is attributable to the stiffness of the channel box. For this reason, the dynamic structural response of the reload core is essentially that of the initial core, and the original seismic-LOCA analysis remains applicable. Deformation of the channel to the point that control blade insertion is inhibited is not predicted to occur.

TABLE B.1 COMPARISON OF PHYSICAL AND STRUCTURAL CHARACTERISTICS FOR 8X8 AND 9X9 FUEL ASSEMBLIES

Property	Fuel Types	
	ANF 9x9	GE 8x8R
Assembly Weight, lbs	580	600
Number of Spacers	7	7
Overall Assembly Length, in	171.29	171.40
Assembly Frequencies, cps		
Mode 1	1.9	*
2	3.7	
3	6.5	
4	10.4	
5	15.5	
6	21.9	
7	29.1	

*GE proprietary

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SUSQUEHANNA UNIT 2 CYCLE 3 RELOAD ANALYSIS

Design and Safety Analyses

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