XN-NF-87-23

SUSQUEHANNA UNIT 1 CYCLE 4 RELOAD ANALYSIS

Design and Safety Analyses

APRIL 1987

RICHLAND, WA 99352

ADVANCED NUCLEAR FUELS CORPORATION

AN AFFILIATE OF KRAFTWERK UNION

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ADVANCED NUCLEAR FUELS CORPORATION

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SUSQUEHANNA UNIT 1 CYCLE 4 RELOAD ANALYSIS

Design and Safety Analyses

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TABLE OF CONTENTS

TABLE OF CONTENTS
(Continued)

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION

This report provides the results of the analyses performed by Advanced Nuclear Fuels Corporation (ANF)* in support of the Cycle 4 reload for Susquehanna Unit 1, which is scheduled to commence operation in the fall of 1987. 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 1 Cycle 4 core will comprise a total of 764 fuel assemblies, including 240 unirradiated ANF XN-3 9x9 assemblies, 296 irradiated ANF XN-2 8x8 assemblies, 192 irradiated ANF XN-1 8x8 assemblies, and 36 previously irradiated 8x8R assemblies fabricated by General Electric. 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 1 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.

*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 4 of Susquehanna Unit 1 is bounded by the assumed power history in the fuel mechanical design analysis, LHGR operating limits (Figures 3.1 and 3.3 of Reference 9.1) have been specified. In addition, a LHGR transient operating limit for Anticipated Operating Occurrences (Figure 3.4 of Reference 9.1) has been specified for ANF 8x8 and 9x9 fuels. 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 1 Cycle 4 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 ANF 8x8 fuel in the Susquehanna Unit 1 core. The similar hydraulic performance indicates compatibility for co-residence in the Susquehanna Unit 1 core. The remaining General Electric fuel assemblies are in the peripheral (low power) region of the core, and do no represent limiting locations in operation of the plant.

3.2.2 Thermal Margin Performance, Comparison

Core Configuration	ANF 9x9 MCPR	ANF 8x8 MCPR
Cycle 4, Mixed Core	1.325	1.298
All ANF 8x8 Core	--	1.294
All ANF 9x9 Core	1.333	--

3.2.3 Fuel Centerline Temperature

Applicable Generic Report Reference 9.1

3.2.5 Bypass Flow

Calculated Bypass Flow Fraction at 100% Power/100% Flow* 9.4%

---

*Does not include flow through water rods (.42%).

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)	15.4 Mlbm/hr
Steam Pressure (at SLMCPR)	1030 psia
Feedwater Temperature	383°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 3.7

4.0 NUCLEAR DESIGN ANALYSIS

4.1 Fuel Bundle Nuclear Design Analysis

Assembly Average Enrichment	3.31%
Radial Enrichment Distribution	Figure 4.1
Axial Enrichment Distribution	Uniform 3.42% with 6" natural uranium top blanket
Burnable Poisons	Figure 4.1

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
Neutronic Design Parameters	Table 4.1

4.2 Core Nuclear Design Analysis

4.2.1 Core Configuration — Figure 4.2

Core Exposure at EOC3, MWD/MTU	18606
Core Exposure at BOC4, MWD/MTU	10590
Core Exposure at EOC4, MWD/MTU	21590
Maximum Cycle 4 Licensing Exposure Limit, MWD/MTU	21700

4.2.2 Core Reactivity Characteristics

BOC Cold K-effective, All Rods Out	1.11441
BOC Cold K-effective, Strongest Rod Out	0.98399
Reactivity Defect (R-Value)	0.00% rho
Standby Liquid Control System Reactivity, Cold Conditions, 660 ppm	0.98070

4.2.4 Core Hydrodynamic Stability

Power/Flow State Points	Decay Ratio (COTRAN)
64/42*	0.74
68/45**	0.66

*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. This point exhibits the highest power to flow ratio in the region where surveillance not required.

5.0 ANTICIPATED OPERATIONAL OCCURRENCES

Applicable Generic Transient
Analysis Methodology Report — Reference 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%	113.6%	259%	1195	0.22	COTRANSA/ XCOBRA-T
FWCF	100%	100%	115.3%	236%	1179	0.21	COTRANSA/ XCOBRA-T
LFWH	100%	100%	124.1%	125%	1079	0.17	PTSBWR3/ XCOBRA

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 8x8	Delta CPR ANF 9x9
100	FWCF	0.19	0.21
80	FWCF	0.21	0.24
65	FWCF	0.22	0.25
40	FWCF	0.24	0.28

5.4 ASME Overpressurization Analysis — Reference 9.6

Limiting Event	Full MSIV Isolation
Worst Single Failure	Direct Scram
Maximum Pressure	1304 psig
Maximum Steam Dome Pressure	1288 psig

5.5 Control Rod Withdrawal Error (CRWE)

Starting Control Rod Pattern for Analysis — Figure 5.1

	100% Flow	
Rod Block Setting	Distance Withdrawn (ft)	Delta CPR
105	4.0	0.15
106	4.5	0.16
107	5.0	0.17
108*	5.5	0.18

*Rod Block Monitor setting selected for Cycle 4 operation.

5.6 Fuel Loading Error

Maximum Delta CPR 0.08

5.7 Determination Of Thermal Margins

Summary of Thermal Margin Requirements

Event	Power*	Flow	Delta CPR**	MCPR Limit
LRWB	100%	100%	0.22	1.28
LFWH	100%	100%	0.17	1.23
FWCF	100%	100%	0.21	1.27
CRWE	100%	100%	0.18 at 108% RBM	1.24

MCPR Operating Limits at Rated Conditions

MCPR Operating Limit
1.28

Reduced Flow MCPR Limits Figure 5.2

Power Dependent MCPR Operating Limit Results for Cycle 4:

% Power/% Flow	Limiting Transient	ANF 8x8	ANF 9x9
100/100	LRWB	1.25	1.28
80/100	FWCF	1.27	1.30
65/100	FWCF	1.28	1.31
40/100	FWCF	1.30	1.34

*104% power used in analysis as design bases.

**Delta CPR results for most limiting fuel type.

6.0 POSTULATED ACCIDENTS

6.1 Loss-Of-Coolant Accident

6.1.1 Break Location Spectrum Reference 9.8

6.1.2 Break Size Spectrum Reference 9.8

6.1.3 MAPLHGR Analyses

ANF XN-3 (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	1675	0.5

*Peak clad temperatures shown are for ANF 9x9 fuel in Susquehanna Unit 2. Changes in PCT, due to minor fuel design changes, were investigated for ANF 9x9 fuel in Susquehanna Unit 1, and found to be insignificant. Calculations were performed at exposure intervals of 5 GWD/MTU from BOL to 20 GWD/MT, and at 30 and 40 GWD/MT. All PCT's were lower, the largest change was 21°F decrease in PCT at 10 GWD/MTU and the smallest was 1°F decrease in PCT at 20 GWD/MTU.

**Metal Water Reaction.

ANF XN-1 (8x8) Fuel Reference 9.10

Limiting Break: Double-ended guillotine pipe break
Recirculating 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	13.0	2074	1.9
5	13.0	2093	2.0
10	13.0	2116	2.1
15	13.0	2136	2.2
19	13.0	2147	2.3
25	11.5	1977	1.6
30	10.4	1846	1.0
35	10.4	1852	1.2

6.2 Control Rod Drop Accident Section 8.0

Dropped Control Rod Worth, mk	6.2
Doppler Coefficient, 1/k dk/dT	$-11.0 \times (10)^{-6}$
Effective Delayed Neutron Fraction	0.0050
Four-Bundle Local Peaking Factor	1.34
Maximum Deposited Fuel Rod Enthalpy, cal/gm	91

*Peak clad temperatures shown are for ANF XN-1 8x8 fuel. The possible change in PCTs due to the fuel design changes in ANF XN-2 8x8 fuel was investigated at three exposures. A 22°F increase resulted at BOL, no change at 10 GWD/MTU, and a 15°F increase resulted at 19 GWD/MTU.

**Metal Water Reaction.

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 3 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 XN-3 9x9 Fuel	ANF XN-1 & XN-2 8x8 Fuel
0	10.2	13.0
5	10.2	13.0
10	10.2	13.0
15	10.2	13.0
19		13.0
20	10.2	
25	9.6	11.5
30	8.9	10.4
35	8.2	10.4
40	7.5	

7.2.2 Minimum Critical Power Ratio

MCPR Operating Limits at Rated Conditions:

MCPR Operating Limit

1.28

MCPR Operating Limits at Off-Rated Conditions:

At Reduced Flow — Figure 5.2

Total Core Recirculation Flow (% Rated)	Reduced Flow MCPR Operating Limit
105	1.06
90	1.11
70	1.19
50	1.34
42	1.42

At Reduced Power

Power Level (% Rated)	Reduced Power MCPR Operating Limit
100	1.28
80	1.30
65	1.31
40	1.34

7.2.3 LHGR Limits

LHGR Limits	Figures 3.1, 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

Stability surveillance established to provide assurance of stable operation during Cycle 3 shall be continued during Cycle 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 1 Cycle 4 Plant Transient Analysis," XN-NF-87-22, Advanced Nuclear Fuels Corporation, Richland, Washington, April 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 "Susquehanna Unit 1 LOCA-ECCS Analysis, MAPLHGR Results," XN-NF-84-119, Advanced Nuclear Fuels Corporation, Richland, Washington, December 1984.

*Formerly Exxon Nuclear Company.

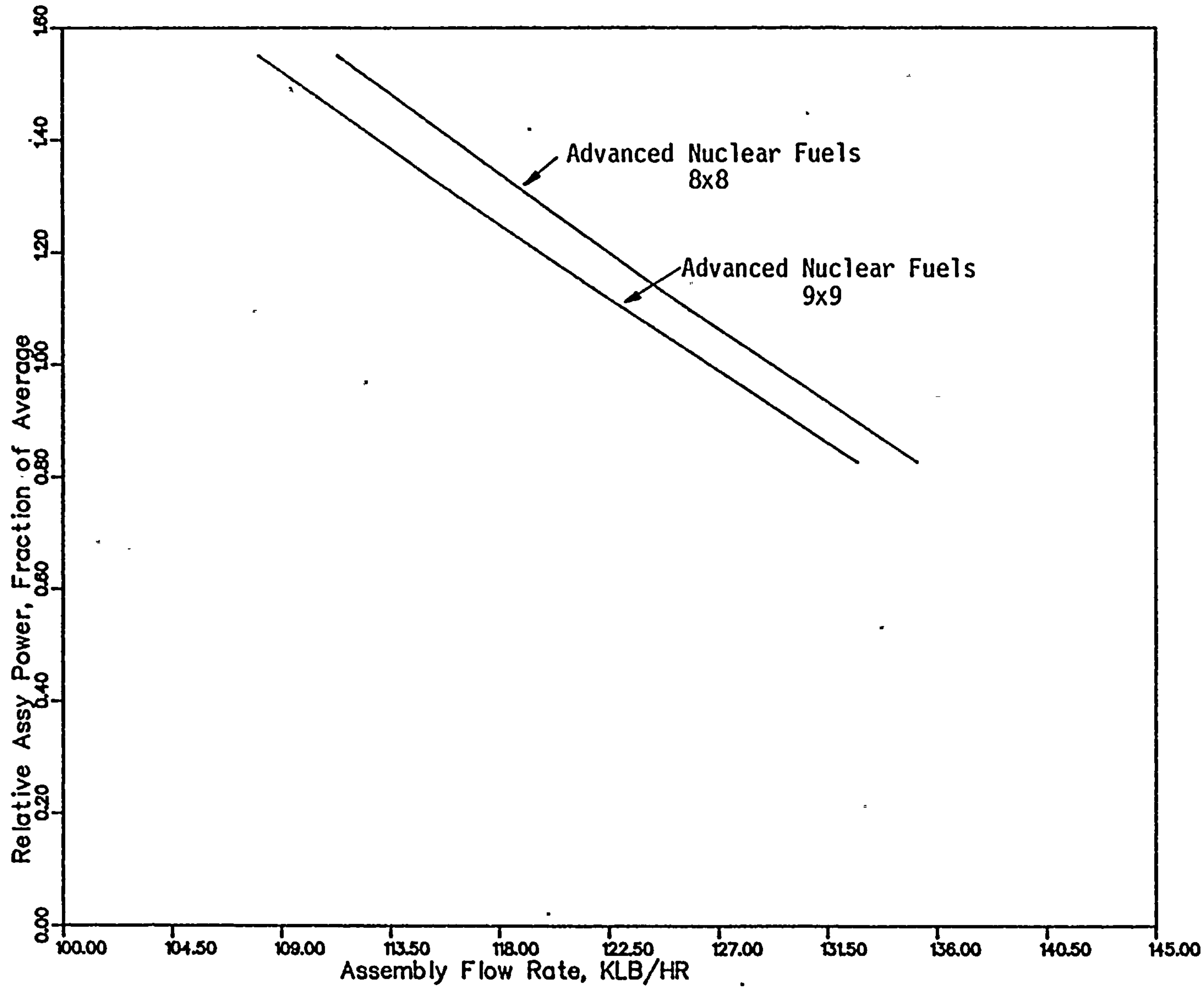

Figure 3.1 Susquehanna Unit 1 Cycle 4 Hydraulic Demand Curve
Power vs. Flow

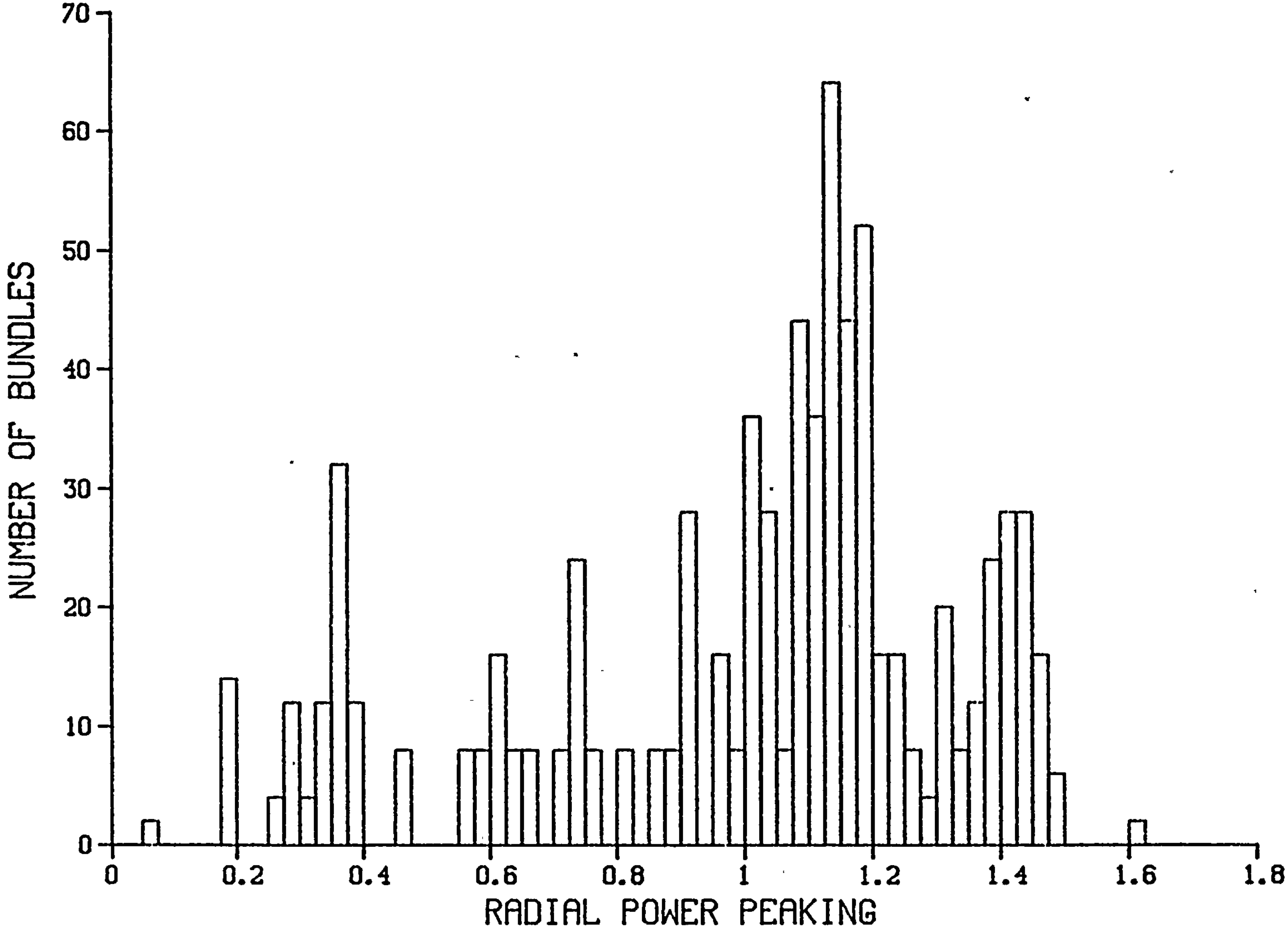

Figure 3.2 Susquehanna Unit 1 Cycle 4 Design Basis Radial Power

* 0.88	0.91	0.96	1.04	1.03	1.04	0.96	1.00	0.97
0.91	0.93	0.98	1.06	0.92	1.07	0.97	1.04	1.00
0.96	0.98	0.90	1.04	1.02	1.04	1.04	0.99	0.96
1.04	1.06	1.04	0.99	0.99	1.00	1.05	0.94	1.04
1.03	0.92	1.02	0.99	0.00	0.97	1.04	1.06	1.04
1.04	1.07	1.04	1.00	0.97	0.00	1.03	0.97	1.05
0.96	0.97	1.04	1.05	1.04	1.03	1.07	0.99	0.97
1.00	1.04	0.99	0.94	1.06	0.97	0.99	0.95	1.01
0.97	1.00	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-3 9X9 Fuel

* W-W corner, rod adjacent to control blade location.

* 0.95	0.95	0.98	1.01	1.01	1.02	1.00	0.99
0.95	1.01	0.92	1.05	1.04	0.98	0.95	1.00
0.98	0.92	1.04	1.02	1.01	1.03	0.98	1.02
1.01	1.05	1.02	0.00	0.94	1.01	1.04	1.01
1.01	1.04	1.01	0.94	0.00	1.02	1.05	1.01
1.02	0.98	1.03	1.01	1.02	1.04	0.92	1.02
1.00	0.95	0.98	1.04	1.05	0.92	0.97	1.00
0.99	1.00	1.02	1.01	1.01	1.02	1.00	0.99

Figure 3.4
Design Basis Local Power Distribution
Advanced Nuclear Fuels XN-2 8X8 Fuel

* W-W corner, rod adjacent to control blade location.

* 0.99	0.98	0.99	1.01	1.01	0.99	0.98	0.99
0.98	0.98	1.03	0.93	1.02	1.03	1.01	0.98
0.99	1.03	1.01	1.01	1.00	1.01	0.94	0.99
1.01	0.93	1.01	0.00	0.96	1.00	1.03	1.01
1.01	1.02	1.00	0.96	0.00	1.01	0.98	1.01
0.99	1.03	1.01	1.00	1.01	1.02	1.04	1.02
0.98	1.01	0.94	1.03	0.98	1.04	0.97	1.01
0.99	0.98	0.99	1.01	1.01	1.02	1.01	1.02

Figure 3.5
Design Basis Local Power Distribution
Advanced Nuclear Fuels XN-1(Central) 8X8 Fuel

* W-W corner, rod adjacent to control blade location.

* 0.96	0.97	1.00	1.03	1.03	1.00	0.97	0.96
0.97	0.98	1.05	0.91	1.03	1.05	1.02	0.97
1.00	1.05	1.02	1.01	1.00	1.01	0.93	1.00
1.03	0.91	1.01	0.00	0.93	1.00	1.03	1.03
1.03	1.03	1.00	0.93	0.00	1.00	0.97	1.03
1.00	1.05	1.01	1.00	1.00	1.02	1.05	1.04
0.97	1.02	0.93	1.03	0.97	1.05	0.97	1.03
0.96	0.97	1.00	1.03	1.03	1.04	1.03	1.01

Figure 3.6
Design Basis Local Power Distribution
Advanced Nuclear Fuels XN-1(Peripheral) 8X8 Fuel

* W-W corner, rod adjacent to control blade location.

* 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 3.7
Design Basis Local Power Distribution
General Electric 8X8R Fuel - Conservatively Flat

* W-W corner, rod adjacent to control blade location.

TABLE 4.1 NEUTRONIC DESIGN VALUES

<u>Fuel Pellet</u>	Reference 9.2
<u>Fuel Rod</u>	Reference 9.2
<u>Fuel Assembly</u>	Reference 9.2
<u>Core Data</u>	
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
<u>Control Rod Data</u>	
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) --- 2.53 W/O U235
M RODS (20) --- 3.27 W/O U235
MH RODS (13) --- 4.19 W/O U235
H RODS (15) --- 4.62 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 1 Cycle 4 (XN-3) Central Enrichment Distribution

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	B2	C1	D0	B2	D0	B2	D0	B2	D0	B2	D0	C1	D0	C1	A3
2	C1	D0	C1	D0	C1	D0	B2	D0	C1	D0	C1	D0	C1	C1	B2
3	D0	C1	B2	B2	C1	B2	D0	B2	C1	B2	D0	C1	D0	C1	B2
4	B2	D0	B2	D0	C1	D0	B2	D0	C1	D0	C1	D0	C1	C1	B2
5	D0	C1	C1	C1	B2	C1	D0	B2	D0	B2	C1	C1	D0	C1	B2
6	B2	D0	B2	D0	C1	D0	C1	D0	C1	D0	C1	D0	C1	C1	B2
7	D0	B2	D0	B2	D0	C1	B2	B2	C1	C1	D0	C1	B2	C1	A3
8	B2	D0	B2	D0	B2	D0	B2	D0	C1	D0	C1	D0	C1	A3	
9	D0	C1	C1	C1	D0	C1	C1	C1	D0	C1	B2	C1	B2		
10	B2	D0	B2	D0	B2	D0	C1	D0	C1	D0	C1	B2	A3		
11	D0	C1	D0	C1	C1	C1	D0	C1	B2	C1	B2				
12	C1	D0	C1	D0	C1	D0	C1	D0	C1	B2					
13	D0	C1	D0	C1	D0	C1	B2	C1	B2	A3					
14	C1	C1	C1	C1	C1	C1	C1	A3							
15	A3	B2	B2	B2	B2	A3	A3								

XY = Fuel Type X
Burned Y Cycles

Fuel Type	No. of Bundles	Description
A	36	GE 8X8 Type III 2.19 w/o U-235
B	192	XN-1 ANF 8X8 2.72 w/o U-235
C	296	XN-2 ANF 8X8 2.89 w/o U-235
D	240	XN-3 ANF 9X9 3.42 w/o U-235

Figure 4.2 Susquehanna Unit 1 Cycle 4 Reference Core Loading Plan

	2	6	10	14	18	22	26	30	34	38	42	46	50	54	58	
59					--	--	--	--	--	--	--					59
55				--	--	00	--	00	--	00	--	--				55
51			--	--	--	--	--	--	--	--	--	--	--			51
47		--	--	04	--	08	--	16	--	08	--	04	--	--		47
43	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	43
39	--	00	--	08	--	00	--	18	--	00	--	08	--	00	--	39
35	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	35
31	--	00	--	16	--	18	--	08	--	18	--	16	--	00	--	31
27	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	27
23	--	00	--	08	--	00	--	18	--	**	--	08	--	00	--	23
19	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	19
15		--	--	04	--	08	--	16	--	08	--	04	--	--		15
11			--	--	--	--	--	--	--	--	--	--	--			11
7				--	--	00	--	00	--	00	--	--				7
3					--	--	--	--	--	--	--					3
	2	6	10	14	18	22	26	30	34	38	42	46	50	54	58	

** Control Rod Being Withdrawn
Rod Position in Notches Withdrawn
    Full in = 00
    Full Out = --

Figure 5.1 Susquehanna Unit 1 Cycle 4 Control Rod Withdrawal Error Analysis Control Rod Pattern

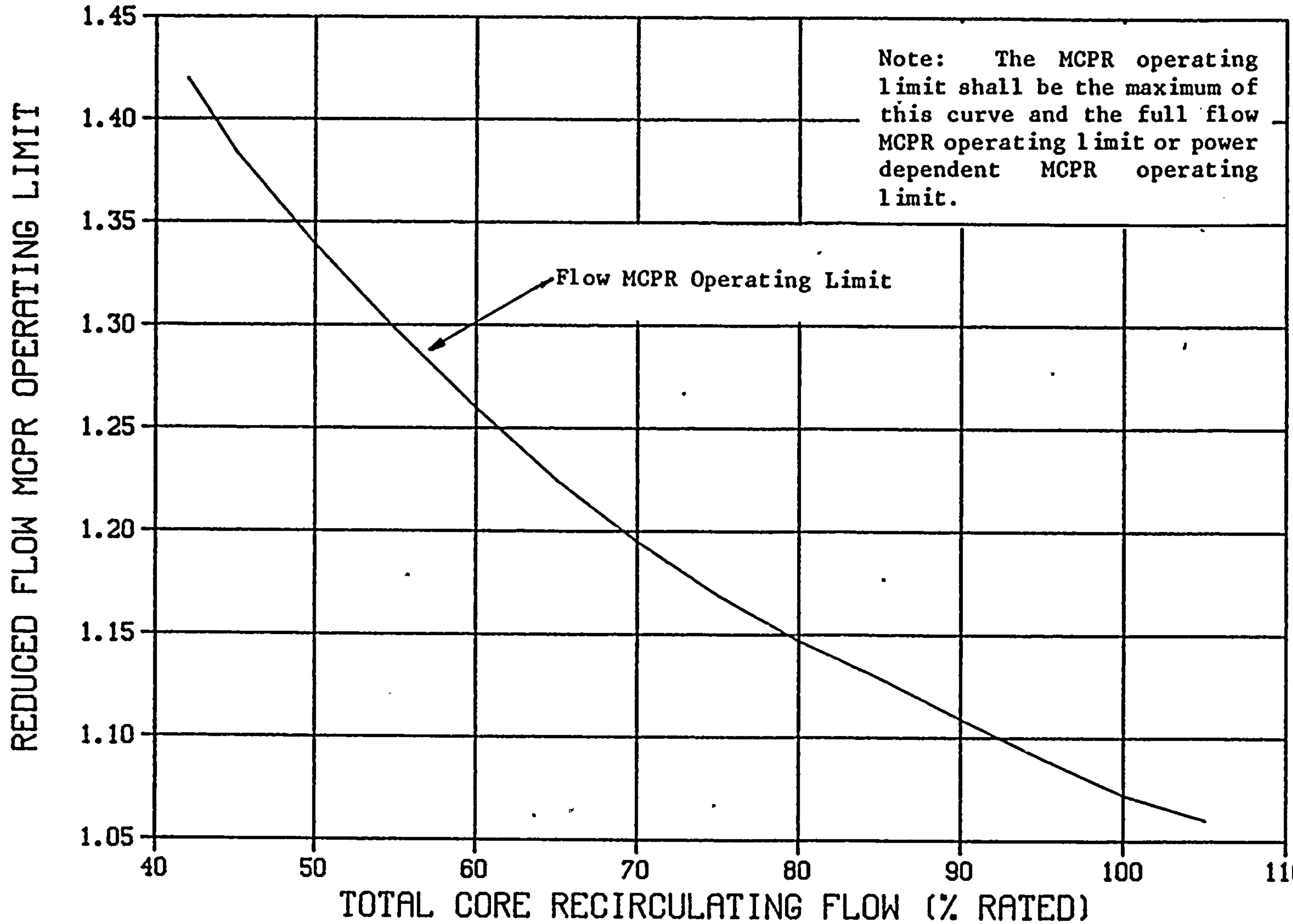

Figure 5.2 Susquehanna Unit 1 Cycle 4 Reduced Flow MCPR Operating Limit

APPENDIX A

SEISMIC-LOCA EVALUATION

The structural response of Advanced Nuclear Fuels Corporation's (ANF's) 9x9 fuel is the same as the structural response of the 8x8 fuel it replaces in the Susquehanna Unit 1 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 A1. The close agreement between the important parameters for the ANF 9x9 and Exxon 8x8 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 A1 COMPARISON OF PHYSICAL AND STRUCTURAL CHARACTERISTICS FOR 8X8 AND 9X9 FUEL ASSEMBLIES

Property		Fuel Types		
		ANF 9x9	ANF 8x8	GE 8x8R
Assembly Weight, lbs		580	596	600
Number of Spacers		7	7	7
Overall Assembly Length, in		171.29	171.29	171.40
Assembly Frequencies, cps				
Mode	1	1.9	1.7	*
	2	3.7	3.5	
	3	6.5	6.5	
	4	10.4	10.8	
	5	15.5	16.6	
	6	21.9	24.2	
	7	29.1	33.9	

*GE proprietary

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SUSQUEHANNA UNIT 1 CYCLE 4 RELOAD ANALYSIS

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