XN-NF-86-60

SUSQUEHANNA UNIT 2 CYCLE 2
RELOAD ANALYSIS
DESIGN AND SAFETY ANALYSES

MAY 1986

RICHLAND, WA 99352

EXXON NUCLEAR COMPANY, INC.

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SUSQUEHANNA UNIT 2 CYCLE 2 RELOAD ANALYSIS

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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 Exxon Nuclear Company (ENC) in support of the Cycle 2 reload for Susquehanna Unit 2, which is scheduled to commence operation in September 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 Susquehanna Unit 2 Cycle 2 core will comprise a total of 764 fuel assemblies, including 324 unirradiated ENC XN-1 9x9 assemblies, and 440 previously irradiated Type P8x8R 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 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 9.1.

2.0 FUEL MECHANICAL DESIGN ANALYSIS

Applicable ENC 9x9 Fuel Design Report: Reference 9.2

To assure that the expected power history for the 9x9 fuel to be irradiated during Cycle 2 of Susquehanna Unit 2 is bounded by the assumed power history in the fuel mechanical design analysis, an LHGR operating limit (Figure 3.3 of Reference 9.2) has been specified for ENC 9x9 fuel. In addition, an LHGR operating limit for Anticipated Operating Occurence (Figure 3.4 of Reference 9.2) has been specified for ENC 9x9 fuel.

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 2 core have been determined in single phase flow tests of full scale assemblies. Figure 3.1 illustrates the hydraulic demand curves for ENC 9x9 fuel and G.E. 8x8 fuel in the Susquehanna Unit 2 core. The similar hydraulic performance indicates adequate compatibility for co-residence in the Susquehanna cores.

3.2.2 Thermal Margin Performance, Comparison

Core Configuration	ENC Fuel MCPR	GE Fuel MCPR
All GE Fuel	----	1.28
All ENC Fuel	1.33	----
Mixed Core	1.34	1.27

3.2.3 Fuel Centerline Temperature

Exposure at Minimum Margin Point	5000 MWD/MT
Centerline Temperature at 120% Power	4157 F
Melting Point of Fuel	5080 F
Margin to Centerline Melting	923 F

3.2.5 Bypass Flow

Calculated Bypass Flow Fraction at 100% Power/100% Flow	10.3%
Calculated Bypass Flow Fraction at 100% Power/108% Flow	9.8%

3.3 <u>MCPR Fuel Cladding Integrity Safety Limit</u>

Safety Limit MCPR = 1.06

3.3.1 Coolant Thermodynamic Condition

Rated Thermal Power	3293 MWt
Feedwater Flowrate (at SLMCPR)	16.2 Mlbm/hr
Steam Dome Pressure (at SLMCPR)	1010 psia
Feedwater Temperature*	420 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 and 3.4

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*Conservative assumption relative to MCPR Safety Limit Monte Carlo procedure.

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 urania top blanket
Burnable Poisons	Fig. 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	Fig. 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 EOC1, MWD/MT	12220
Core Exposure at BOC2, MWD/MT	7805
Core Exposure at EOC2, MWD/MT	18305

Note: Cycle 2 safety analyses are valid for EOC1 exposure from -1000 MWD/MT to +780 MWD/MT from the nominal value reported above.

4.2.2 Core Reactivity Characteristics

BOC Cold K-effective, All Rods Out	1.10813
BOC Cold K-effective, Strongest Rod Out	0.97320
Reactivity Defect (R-Value)	0.04% rho
Standby Liquid Control System Reactivity, Cold Conditions, 660 ppm	0.9758

4.2.4 Core Hydrodynamic Stability

Power/Flow State Points	Decay Ratio (COTRAN)
38.5/30	< 0.30
58/60	< 0.15
68/45	0.59

These state points bound the surveillance region. A COTRANSA2 and additional COTRAN calculations will be provided at a later date and startup tests are scheduled for demonstration of the stability performance.

5.0 ANTICIPATED OPERATIONAL OCCURRENCES

Applicable Generic Transient Analysis Report — Reference 9.3

5.1 Analysis of Plant Transients at Rated Conditions

Reference 9.4 & 9.5

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 (psia)	Delta CPR**	Model
LRWB	100%	100%	114.3%	274%	1213	0.17	COTRANSA
FWCF	100%	100%	114.7%	245%	1180	0.15	COTRANSA
LFWH	100%	100%	N/A	N/A	N/A	0.08	XTGBWR

Note: See Appendix A for single-loop operation.
See Appendix C for Final Feedwater Temperature Reduction (FFTR) and Increased Core Flow (ICF).

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*104% power used in analysis as design bases.

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

5.2 Analyses for Reduced Flow Operation — Reference 9.4

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

5.4 ASME Overpressurization Analysis — Reference 9.4

Limiting Event	Full MSIV Isolation
Worst Single Failure	Direct Scram
Maximum Pressure	1315 psig
Maximum Steam Dome Pressure	1301 psig

5.5 Control Rod Withdrawal Error (CRWE)

Starting Control Rod Pattern for Analysis

For 100% Flow @ 106 RBM setting	Figure 5.1
For 100% Flow @ 108 RBM setting	Figure 5.2
For 108% Flow @ 106/108 RBM setting	Figure 5.3

	100% Flow		108% Flow	
Rod Block Setting	Distance Withdrawn (ft)	Delta CPR	Distance Withdrawn (ft)	Delta CPR
105	4.0	0.15	4.5	0.22
106*	4.5	0.17	5.0	0.23
107	5.0	0.18	6.0	0.24
108*	6.0	0.21	8.0	0.26

*Rod Block Monitor settings selected for Cycle 2 operation

5.6 Fuel Loading Error

Maximum Delta-CPR 0.15

5.7 Determination of Thermal Margins

Summary of Thermal Margin Requirements

Event	Power	Flow	Delta-CPR	MCPR Limit
LRWB	100%	100%	0.17	1.23
LFWH	100%	100%	0.08	1.14
FWCF	100%	100%	0.15	1.21
CRWE	100%	100%	0.17 at 106% RBM	1.23
CRWE	100%	100%	0.21 at 108% RBM	1.27
FWCF*	100%	100%	0.17	1.23
LRWB*	100%	100%	0.15	1.21
LRWB	100%	108%	0.17	1.23
LFWH	100%	108%	0.08	1.14
FWCF	100%	108%	0.16	1.22
CRWE	100%	108%	0.23 at 106% RBM	1.29
CRWE	100%	108%	0.26 at 108% RBM	1.32
FWCF*	100%	108%	0.17	1.23
LRWB*	100%	108%	0.15	1.21

Note : Events are results of bounding analyses (1.06 safety limit used).

MCPR Operating Limits at Rated Conditions

Rod Block Setting	MCPR Operating Limit
106%	1.23
108%	1.27

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*At Final Feedwater Temperature Reduction of 65°F.

MCPR Operating Limits at Off-Rated Conditions.

At 100% Power/108% Flow

Rod Block Setting	MCPR Operating Limit
106%	1.29
108%	1.32

At Reduced Flow Figure 5.4

Power Dependent MCPR Operating Limit

% Power/% Flow	Nominal Feedwater Temp		FFTR*	
	GE P8x8R	ENC 9x9	GE P8x8R	ENC 9x9
100/100	1.20	1.21	1.22	1.23
80/100	1.28	1.30	1.26	1.28
65/100	1.29	1.31	1.30	1.32
40/100	1.32	1.35	1.32	1.35
100/108	1.21	1.22	1.22	1.23
80/108	1.26	1.28	1.26	1.28
65/108	1.29	1.31	1.30	1.32
40/108	1.33	1.36	1.32	1.36

*65°F reduction in Feedwater Temperature.

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-1 9x9 Fuel

Reference 9.7

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

Bundle Average Exposure (GWD/MT)	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

6.2 Control Rod Drop Accident

Reference 8.1

Dropped Control Rod Worth, mk	8.3
Doppler Coefficient, 1/k dk/dT	$-9.5 \times (10)^{-6}$
Effective Delayed Neutron Fraction	0.0045
Four-Bundle Local Peaking Factor	1.35
Maximum Deposited Fuel Rod Enthalpy, cal/gm	147

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 of 1359 can be allowed but the 1325 psig value used in Cycle 1 is to be conservatively retained.

7.2 Limiting Conditions for Operation

7.2.1 Average Planar Linear Heat Generation Rate Limits for XN-1 9x9 Fuel

Bundle Average Exposure (GWD/MT)	MAPLHGR (kw/ft)
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

Rod Block Setting	MCPR Operating Limit
106%	1.23
108%	1.27

MCPR Operating Limits at Off-Rated Conditions

At 100% Power/108% Flow

Rod Block Setting	MCPR Operating Limit
106%	1.29
108%	1.32

At Reduced Flow Figure 5.4

Power Dependent MCPR Operating Limit

% Power/% Flow	Nominal Feedwater Temp		FFTR*	
	GE P8x8R	ENC 9x9	GE P8x8R	ENC 9x9
100/100	1.20	1.21	1.22	1.23
80/100	1.28	1.30	1.26	1.28
65/100	1.29	1.31	1.30	1.32
40/100	1.32	1.35	1.32	1.35
100/108	1.21	1.22	1.22	1.23
80/108	1.26	1.28	1.26	1.28
65/108	1.29	1.31	1.30	1.32
40/108	1.33	1.36	1.32	1.36

7.2.3 LHGR Limits

LHGR Limits	Figure 3.3 and 3.4 of Reference 9.2

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 1 shall be continued during Cycle 2.

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*65°F reduction in Feedwater Temperature.

8.0 METHODOLOGY REFERENCES

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

9.0 ADDITIONAL REFERENCES

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

9.2 "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel", XN-NF-85-67(P), Rev. 1, Exxon Nuclear Company, Richland, Washington (April 1986).

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

9.4 "Susquehanna Unit 2 Cycle 2 Plant Transient Analysis", XN-NF-86-55, Exxon Nuclear Company, Richland, Washington (MAY 1986).

9.5 "A Generic Analysis of the Loss of Feedwater Heating Transient for Boiling Water Reators", XN-NF-900(P), Exxon Nuclear Company, Richland, WA (February 1986).

9.6 "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), Exxon Nuclear Company, Richland, Washington (December 1984).

9.7 "Susquehanna LOCA-ECCS Analysis MAPLHGR Results for ENC 9x9 Fuel", XN-NF-86-65, Exxon Nuclear Company, Richland, Washington (May 1986).

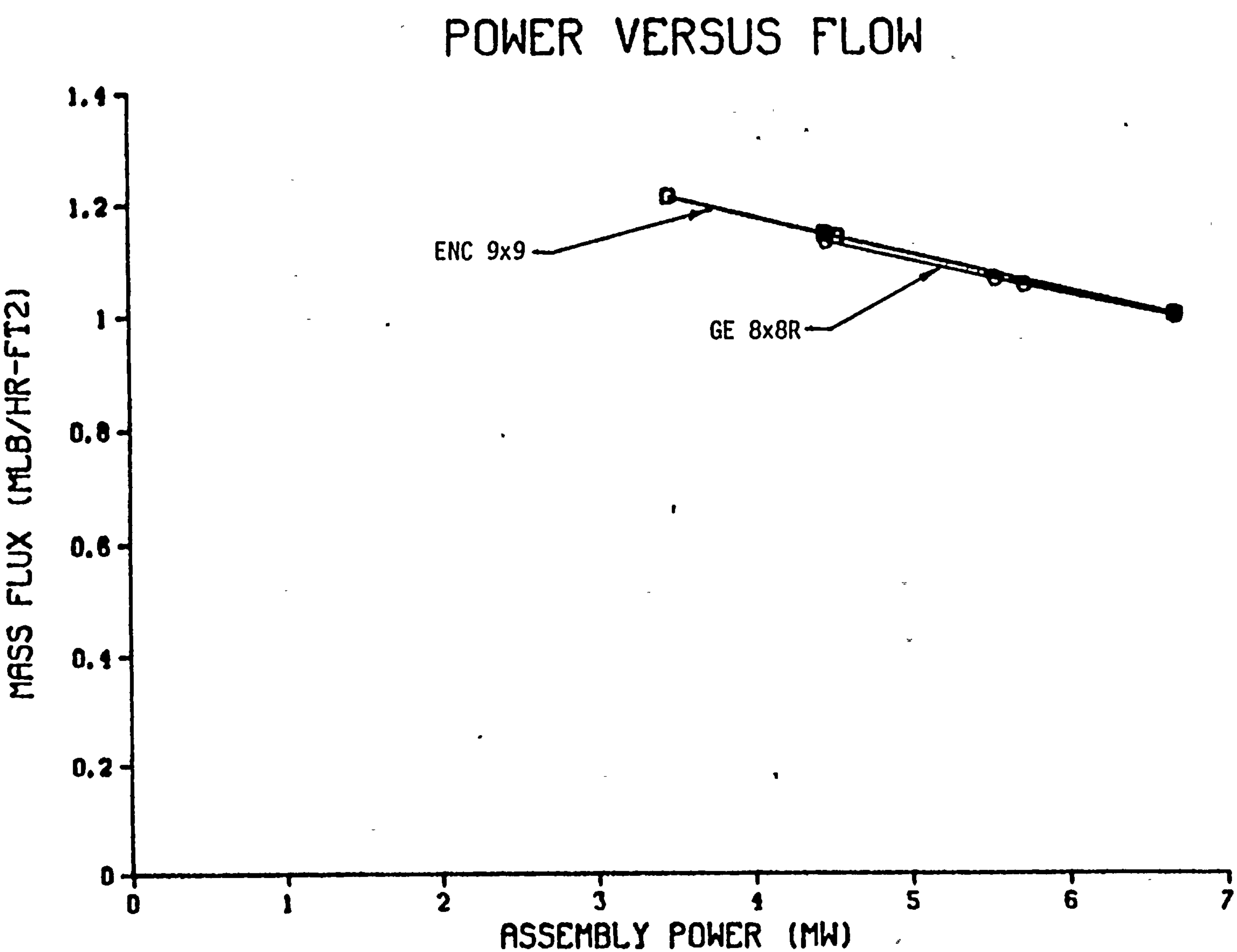

Figure 3.1 Hydraulic Demand Curves For Susquehanna Unit 2 Cycle 2 Core

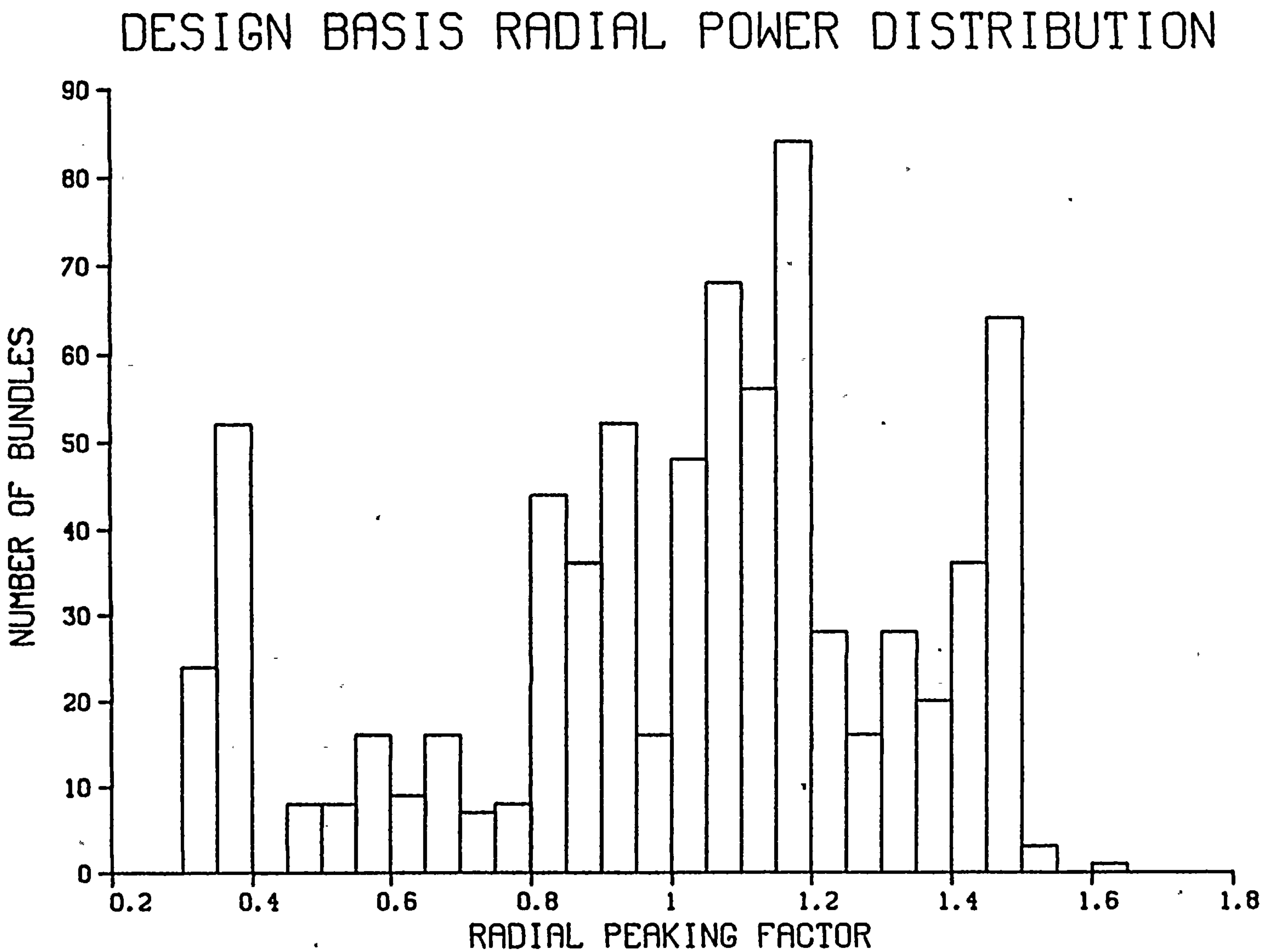

Figure 3.2 Susquehanna Unit 2 Cycle 2 Safety Limit Radial Power Histogram

* 0.88	0.91	0.97	1.04	1.03	1.04	0.97	1.01	0.97
0.91	0.94	0.98	0.94	1.05	0.93	0.99	0.94	1.01
0.97	0.98	0.90	1.04	1.03	1.05	1.04	0.97	0.97
1.04	0.94	1.04	1.00	1.00	1.01	1.05	1.07	1.04
1.03	1.05	1.03	1.00	0.00	0.97	1.05	1.06	1.04
1.04	0.93	1.05	1.01	0.97	0.00	1.02	0.95	1.05
0.97	0.99	1.04	1.05	1.05	1.02	1.06	1.00	0.97
1.01	0.94	0.97	1.07	1.06	0.95	1.00	0.95	1.02
0.97	1.01	0.97	1.04	1.04	1.05	0.97	1.02	0.97

FIGURE 3.3
DESIGN BASIS LOCAL POWER DISTRIBUTION
ENC XN-1 9X9 FUEL

* Rod adjacent to control blade location.

* 1.03	1.00	0.99	0.99	0.99	0.99	1.00	1.03
1.00	0.97	0.99	1.02	1.03	1.03	0.99	1.00
0.99	0.99	1.02	1.01	1.02	0.91	1.03	0.99
0.99	1.02	1.01	0.91	0.00	1.02	1.02	0.99
0.99	1.03	1.02	0.00	1.02	1.01	0.99	0.99
0.99	1.03	0.91	1.02	1.01	0.98	0.99	0.99
1.00	0.99	1.03	1.02	0.99	0.99	0.97	1.00
1.03	1.00	0.99	0.99	0.99	0.99	1.00	1.03

FIGURE 3.4
DESIGN BASIS LOCAL POWER DISTRIBUTION
G.E. 8X8R FUEL

* Rod adjacent to control blade location.

Table 4.1

Neutronic Design Values

Core Data

Number of fuel assemblies	764
Rated thermal power, MW	3293
Rated core flow, Mlbm/hr	100.0
Core inlet subcooling, BTU/lbm	24.0
Moderator temperature, F	549
Channel thickness, inch	0.080
Fuel assembly pitch, inch	6.0
Water gap thickness, inch	0.562
Narrow water gap thickness, inch	0.562

Control Rod Data

Absorber material	B4C
Total blade span, inch	9.75
Total blade support span, inch	1.58
Blade thickness, inch	0.26
Blade face-to-face internal dimension, inch	0.20
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	M	MH	M*	M	ML	ML
ML	M	M*	H	H	H	MH	M*	ML
M	M	H	H	H	H	H	MH	M
M	MH	H	H	W	MH	H	MH	M
M	M*	H	H	MH	W	MH	M*	M
ML	M	MH	H	H	MH	MH	M	ML
ML	ML	M*	MH	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 (18) --- 2.58 w/o U235
M Rods (20) --- 3.27 w/o U235
MH Rods (13) --- 4.18 w/o U235
H Rods (15) --- 4.68 w/o U235
M* Rods (7) --- 3.27 w/o U235 + 4.00 w/o Gd2O3
W Rods (2) --- Inert Water Rods

Figure 4.1 Susquehanna Unit 2 XN-1 3.42 w/o Central Enrichment Distribution

A1	A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	A1
A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1
C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	A1
A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1
C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	A1
A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1
C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	A1
A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	A1	
C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	A1		
A1	C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	A1	B1		
C0	A1	C0	A1	C0	A1	C0	A1	C0	A1	A1				
A1	C0	A1	C0	A1	C0	A1	C0	A1	A1					
C0	A1	C0	A1	C0	A1	C0	A1	A1	B1					
A1	C0	A1	C0	A1	C0	A1	A1							
A1	A1	A1	A1	A1	A1	A1								

XY = Fuel Type X
Burned Y Cycles

Fuel Type	No. of Bundles	Description
A	432	GE 8x8 Type III 2.19 w/o U-235
B	8	GE 8x8 Type II 1.76 w/o U-235
C	324	ENC 9x9 3.31 w/o 7Gd4.0

Figure 4.2 Susquehanna Unit 2 Cycle 2 Reference Core Loading

	02	06	10	14	18	22	26	30	34	38	42	46	50	54	58	
59					--	--	--	--	--	--	--					59
55				--	--	14	--	28	--	14	--	--				55
51			--	--	28	--	--	--	--	--	28	--	--			51
47		--	--	14	--	06	--	12	--	06	--	14	--	--		47
43	--	--	28	--	--	--	--	--	--	--	--	--	28	--	--	43
39	--	14	--	06	--	16	--	28	--	16	--	06	--	14	--	39
35	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	35
31	--	28	--	12	--	28	--	00*	--	28	--	12	--	28	--	31
27	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	27
23	--	14	--	06	--	16	--	28	--	16	--	06	--	14	--	23
19	--	--	28	--	--	--	--	--	--	--	--	--	28	--	--	19
15		--	--	14	--	06	--	12	--	06	--	14	--	--		15
11			--	--	28	--	--	--	--	--	28	--	--			11
07				--	--	14	--	28	--	14	--	--				07
03					--	--	--	--	--	--	--					03
	02	06	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 2 Cycle 2 Control Rod Withdrawal Error Analysis
Initial Control Rod Pattern For 106 RBM Setting
100% Flow Case

	02	06	10	14	18	22	26	30	34	38	42	46	50	54	58	
59					--	--	--	--	--	--	--					59
55				--	--	08	--	04	--	08	--	--				55
51			--	--	--	--	--	--	--	--	--	--	--			51
47		--	--	08	--	10	--	06	--	10	--	08	--	--		47
43	--	--	--	--	40	--	36	--	36	--	40	--	--	--	--	43
39	--	08	--	10	--	16	--	08	--	16	--	10	--	08	--	39
35	--	--	--	--	36	--	34	--	34	--	36	--	--	--	--	35
31	--	04	--	06	--	08	--	20	--	08	--	06	--	04	--	31
27	--	--	--	--	36	--	--	--	--	--	36	--	--	--	--	27
23	--	08	--	10	--	16	--	00*	--	16	--	10	--	08	--	23
19	--	--	--	--	40	--	--	--	--	--	40	--	--	--	--	19
15		--	--	08	--	10	--	06	--	10	--	08	--	--		15
11			--	--	--	--	--	--	--	--	--	--	--			11
07				--	--	08	--	04	--	08	--	--				07
03					--	--	--	--	--	--	--					03
	02	06	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 5.2 Susquehanna Unit 2 Cycle 2 Control Rod Withdrawal Error Analysis Initial Control Rod Pattern For 108 RBM Setting 100% Flow Case.

	02	06	10	14	18	22	26	30	34	38	42	46	50	54	58	
59					--	--	--	--	--	--	--					59
55				--	--	00	--	16	--	00	--	--				55
51			--	--	16	--	--	--	--	--	16	--	--			51
47		--	--	00	--	06	--	08	--	06	--	00	--	--		47
43	--	--	16	--	--	--	--	--	--	--	--	--	16	--	--	43
39	--	00	--	06	--	08	--	10	--	08	--	06	--	00	--	39
35	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	35
31	--	16	--	08	--	10	--	00*	--	10	--	08	--	16	--	31
27	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	27
23	--	00	--	06	--	08	--	10	--	08	--	06	--	00	--	23
19	--	--	16	--	--	--	--	--	--	--	--	--	16	--	--	19
15		--	--	00	--	06	--	08	--	06	--	00	--	--		15
11			--	--	16	--	--	--	--	--	16	--	--			11
07				--	--	00	--	16	--	00	--	--				07
03					--	--	--	--	--	--	--					03
	02	06	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 5.3 Susquehanna Unit 2 Cycle 2 Control Rod Withdrawal Error Analysis
Initial Control Rod Pattern
108% Flow Case

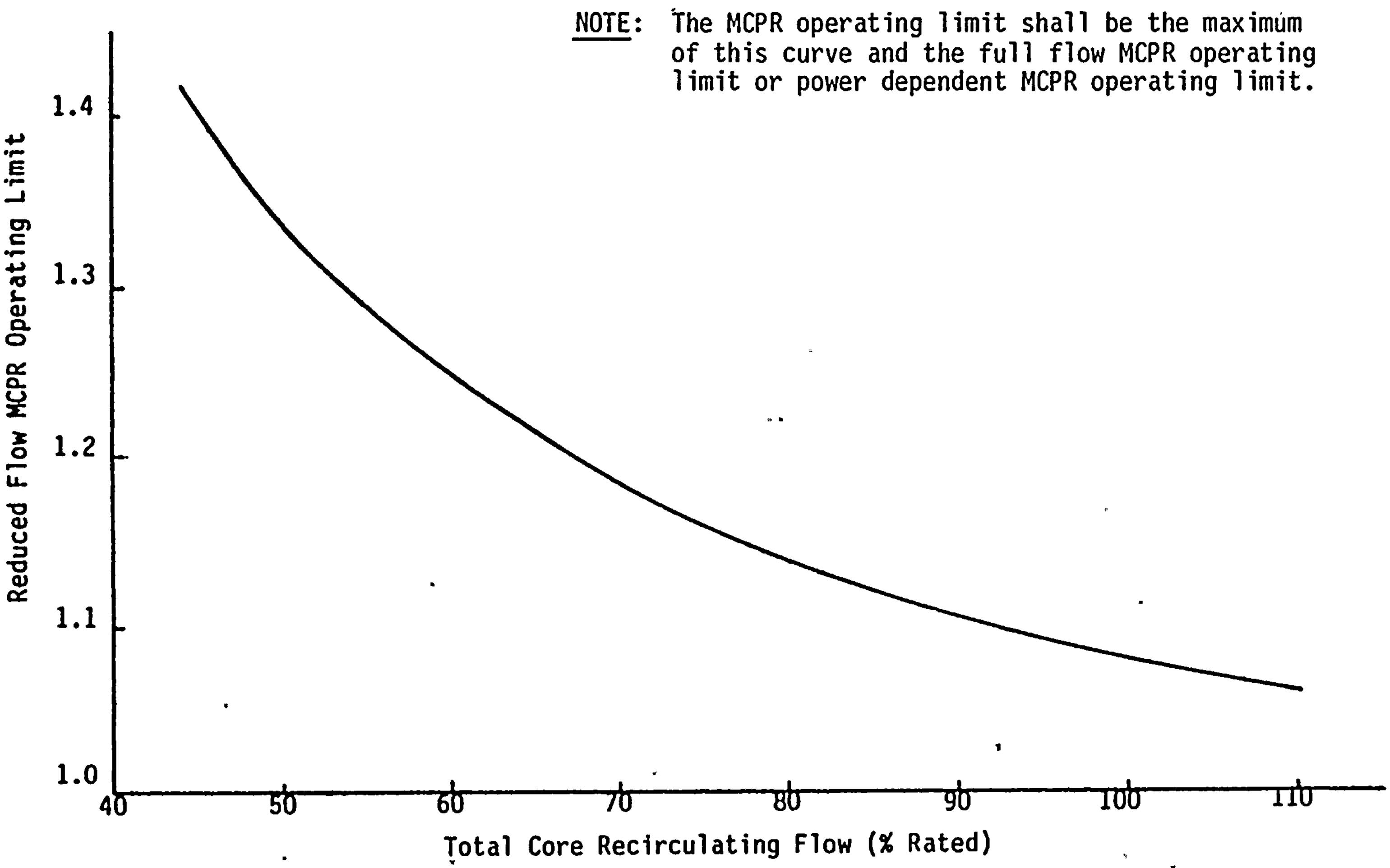

Figure 5.4 Reduced Flow MCPR Operating Limit

APPENDIX A

SUSQUEHANNA UNIT 2 SINGLE-LOOP OPERATION WITH ENC 9X9 FUEL

Analyses have been performed for Susquehanna Units 1 and 2 for normal two pump operation both by the NSSS vendor and Exxon Nuclear Company (ENC). Generally, both analyses showed similar results and yielded comparable allowed operating limits. Since the ENC and vendor 8x8 fuel designs are very similar, this result is to be expected. ENC analysis for 8x8 and 9x9 fuel in the Susquehanna reactors justifies modification of 8x8 calculated MAPLHGR limits to be appropriate for 9x9 fuel on an equal planar power basis.

The ability to operate the Susquehanna reactors with only one recirculation pump running is highly desirable in the event that a recirculation pump or other component maintenance renders one loop inoperative. In order to justify single-loop operation, the NSSS vendor has performed additional accident and transient analyses for single-loop operating conditions (Reference A.1). The single-loop operation analysis generally showed that operation within the full-power two pump operating limits will assure that the safety limit is not violated and that substantial margin to the safety limit exists for single-loop operation due to the reduced power. For these cases, ENC fuel will likewise experience the benefit of the power reduction and application of two pump full-power limits for the ENC fuel designs is conservative and appropriate. This Appendix discusses appropriate limits for Susquehanna Unit 2 Cycle 2 operation with ENC 9x9 fuel and their bases.

A.1 ROD WITHDRAWAL ERROR

The rod block system is designed to stop rod withdrawal at a minimum critical power ratio (MCPR) higher than the fuel cladding safety limit. For

single-loop operation, a procedure has been established for correcting the APRM rod block equation to account for the discrepancy between actual flow and indicated flow in the active loop. This procedure preserves the original relationship between APRM rod block and actual effective drive flow when operating with a single-loop. The APRM scram trip settings are flow biased in the same manner as the APRM rod block setting. Modification to the rod block equation and lower power assures the MCPR safety limit is not violated. This applies for both 8x8 and 9x9 ENC fuel designs.

A.2 TRANSIENT MCPR LIMITS

Operating with one recirculation loop results in a maximum power output which is about 25% below that which is attainable for two pump operation. Therefore, the NSSS vendor single-loop analysis showed that the consequences of abnormal operation transients will be considerably less severe than those analyzed from a two-loop operational mode. These results are shown in Table 15.c.3-3 of Reference A.1. The limiting transients from an allowed MCPR operating limit of 1.38 gave transient MCPRs of 1.20-1.21 which are well above the GE safety limit of 1.07 with a 0.13-0.14 margin in CPR. For pressurization, flow increase, flow decrease, and cold water injection transients, results for two-loop operation bound both the thermal and overpressure consequences of one-loop operation. It was concluded that the MCPR operating limits established for two-pump operation are also applicable to single-loop operation conditions. This is true even for the increased safety limit associated with single-loop operation (see A.4).

The increased MCPR margin for single-loop operation at reduced power is also applicable to ENC fuel designs. Therefore, the operating MCPR limits established for two-pump operation with ENC fuel will be conservative when applied to single-loop operation for the same reasons as for the vendor fuel. This applies for both 8x8 and 9x9 fuel designs. Applicability of two-pump limits for single pump operation is discussed phenomenologically in the following section.

A.3 ABNORMAL OPERATING TRANSIENTS

MCPR limits established for full flow two loop operation are conservative for single loop operation because of the physical phenomena related to part-power part-flow operation, not because of features in reactor analysis models or compatible fuel designs. A review of the most limiting delta CPR transients for single-loop operation was conducted. Under single-loop conditions, steady state operation cannot exceed approximately 75% power and 60% core flow because of the capability of the recirculation loop pump. Thus, the MCPR limit at maximum power is higher than the two-pump operating MCPR limit due to the flow dependent MCPR function. This flow dependence is based on a flow increase transient from runup of two pumps. Flow runups from a single recirculation pump would be much less severe but the conservative two pump limit is retained.

A.3.1 Load Rejection Without Bypass

The limiting system transient for the Susquehanna Units is the Load Rejection Without Bypass (LRWB) pressurization transient. In this transient, the primary phenomena is the pressurization caused by abruptly stopping the steam flow through rapid closure of the turbine control valve. When the rapid pressurization reaches the core it causes a power excursion due to void collapse.

At reduced power and flow there is a corresponding reduction in steam flow. With lower steam flow the maximum pressurization of the core is reduced in comparison to rated conditions when the control valve is closed. The resulting power excursion and associated delta CPR are reduced below those of the full power full flow case.

Thus the MCPR limits based on LRWB analyses at full power are conservatively applicable to the lower powers associated with single loop conditions based on the physics of the transient. Furthermore, LRWB analyses by GE[A.2] and

preliminary ENC analyses at reduced power and flow conditions with two-loop operation confirm this trend, and GE analyses[A.2] under single-loop conditions also confirm this trend.

A.3.2 Feedwater Controller Failure

The second most limiting transient for Susquehanna is the Feedwater Controller Failure (FWCF). This transient is also less severe at the reduced power and flow conditions associated with single-loop operation.

This transient assumes the feedwater controller fails to maximum demand and allows the maximum amount of subcooled feedwater into the downcomer. When this cooler water reaches the core, the power rises. The core power rise is terminated through a turbine trip scram initiated by a high water level trip in the downcomer due to the additional amount of feedwater being injected.

At the reduced recirculation flows the subcooling in the downcomer due to high feedwater injection takes longer to transverse to the core such that a high level trip occurs before the core power rises as much as in the full power case. In the subsequent pressurization transient, the result of turbine trip is less severe for the reduced powers in transients from single-loop conditions because of the reasons discussed in the LRWB transient.

Thus, because of the slower transport phenomena caused by the lower flow in the downcomer and because of the lower steam line flow in the pressurization portion of the transient, and the higher full-power MCPR limit, the FWCF has larger margin to the operating limit in single-loop operation than in full-power two-loop operation.

A.3.4 Summary

It is very conservative to use the reduced flow two-loop operating MCPR limit for single-loop operations. 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 SAFETY LIMIT MCPR

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. ENC 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 ENC fuel during single-loop operation. Thus, increasing the safety limit MCPR by 0.01 for single-loop operation (1.07) with ENC fuel is sufficiently conservative to also bound the increased flow measurement uncertainties for single-loop operation.

A.5 MAPLHGR LIMITS

The NSSS vendor has also evaluated the changes in the two-loop MAPLHGR limits required to permit single-loop operation. A multiplier of 0.81 is to be applied to the appropriate two-loop MAPLHGR limit to obtain the MAPLHGR limit for single-loop operation. The need to reduce the allowed MAPLHGR arises because of the conservative assumption of early boiling transition (at 0.1 sec) in the LOCA-ECCS analysis applied for single-loop operation at reduced core flow.

To support operation of Susquehanna Unit 2 with Exxon Nuclear Company (ENC) 9x9 fuel with a single recirculating pump operating, the GE MAPLHGR limits for the highest enriched GE 8x8R fuel design with a multiplier of 0.81 are to be applied on an equal planar power basis to ENC 9x9 fuel for single-loop operation. The basis for this is two-fold:

1) The phenomena which require the reduction in MAPLHGR limits are a result of operation of the Susquehanna Unit 2 system with single active recirculation loop, and are therefore, equally applicable to both GE and ENC fuel designs, and

2) For the expected exposures during Cycle 2 operation the analysis methods used by GE have yielded conservative MAPLHGR limits relative to the MAPLHGR limits obtained using the ENC approved analysis models. Therefore, applying the more conservative GE MAPLHGR limit to ENC fuel provides a limit which assures conformance to NRC 10 CFR 50.46 criteria.

The major difference between operation with both recirculation pumps running and operating with only one active recirculation pump are reduced operating core flow, reduced core power, and reverse flow through the inactive loop jet pumps. Flow dependent MCPR limits assure reduced maximum assembly power during single-loop operation. The primary system coolant inventory and LOCA break conditions are essentially unchanged from the two-loop operation. Thus, the uncovery of the jet pump suction, recirculation suction line uncovery, and system depressurization rate would be expected to change little between one and two-loop operation. The phenomena associated with these key parameters largely determine LOCA analysis results for both ENC and GE analyses. The analyses performed by GE confirm this system behavior in that the limiting pipe break LOCA is essentially unchanged from the two-loop analysis, as are the break size and core uncovery and reflood times. Although ENC LOCA analysis methods differ from those of GE, similar results would be expected

from an ENC analysis because the phenomena are governed by the system parameters.

The principal LOCA concern associated with single-loop operation is the possibility of the LOCA break occurring in the operating loop, in which case there is no coastdown of an intact loop recirculation pump to sustain jet pump and core flow during the early portion of the system blowdown. An early boiling transition (CHF) may result from this early loss of flow capability.

To account for this possibility, GE derived a single-loop operation MAPLHGR multiplier of 0.81 to be used with calculated two-loop MAPLHGR limits during single-loop operation. The analyses which determined this multiplier assumed a near instantaneous boiling transition (0.1 sec) even though a longer boiling transition time may have been calculated using approved models. This assumption is very conservative when applied to the GE fuel and would be even more conservative when applied to ENC 9x9 fuel because of lower stored energy in 9x9 fuel.

The major difference between the ENC and GE methodologies that would effect analysis differences between single and two-loop operation is in the blowdown heat transfer. ENC's more mechanistic model calculates boiling transition times that are equivalent to or later than those reported from the GE model, and the ENC model explicitly calculates the blowdown heat transfer throughout the blowdown period while the GE model assumes an adiabatic heatup period. Thus, the conservative approach taken in the GE analysis of assuming an early boiling transition (0.1 sec) for single-loop operation would yield a greater penalty using ENC methodology than for the more conservative GE methods. For this reason, limits based on the more conservative GE analysis are recommended. ENC's more mechanistic heat transfer during the GE adiabatic heatup period would partially offset this effect, thus, risking the recommended limits conservative.

Application of GE calculated 8x8 MAPLHGR limits modified on equal planar power basis for ENC 9x9 fuel for single-loop operation will conservatively assure that the NRC criteria of 10 CFR 50.46 will be met for the following reasons:

1) Since ENC has performed LOCA analyses for a number of BWRs under two-loop operation and MAPLHGR limits for ENC 8x8 fuel are higher than the equivalent GE 8x8 fuel limits in all cases for bundle exposures less than 19,000 MWD/Mt, an ENC analysis for the similar single-loop operating conditions would be expected to also yield MAPLHGR limits equal to or higher than those obtained by GE.

2) The MAPLHGR reduction factor to protect against early boiling transition determined by GE is based on a conservative early boiling transition assumption which is even more conservative when applied to 9x9 fuel.

3) ENC analysis for two-loop operation at expected exposures in Cycle 2 of Susquehanna Unit 2 with 9x9 fuel justifies MAPLHGR limits equal to or greater than the GE 8x8 design on an equivalent planar power basis. That is, 9x9 MAPLHGR limits are equal or greater than the GE 8x8 limits times the ratio of heated rods in the 8x8 assembly to heated rods in the 9x9 assembly. On this basis 8x8 MAPLHGR limits can be conservatively modified for application to 9x9 fuel.

For Cycle 2 of Susquehanna Unit 2 single-loop operation with ENC 9x9 fuel, a MAPLHGR limit corresponding to 0.81 times the MAPLHGR limits for the highest enriched Cycle 1 GE fuel type can be conservatively used. These 8x8 MAPLHGR limits are to be adjusted by the ratio (62/79) to be on an equal planar power basis for 9x9 fuel.

A.6 STABILITY

Susquehanna Units 1 and 2 have adopted a detect and suppress approach to avoid unstable reactor operation. This is consistent with single-loop operation requirements stated in NRC Generic Letter #86-09 (Reference A.3). The detect and suppress criteria will be conservatively applicable to ENC 9x9 fuel in the Susquehanna Unit 2 reactor.

A.7 REFERENCES

A.1 General Electric Co., "Susquehanna Single-Loop Operation Analysis", GP84-142, General Electric Co., June 1984.

A.2 "Extended Load Line Limit Analyses for Susquehanna Steam Electric Station Unit 1", NEDO22128, General Electric Co., May 1982.

A.3 "Technical Resolution of Generic Issue No. B-59-(N-1) Loop Operation in BWRs and PWRs", (Generic Letter No. 86-09), March 31, 1986.

APPENDIX B

SEISMIC-LOCA EVALUATION

The structural response of Exxon Nuclear's 9x9 fuel is the same as the structural response of the 8x8 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 B1. The close agreement between the important parameters for the two 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 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. ENC 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 B1

COMPARISON OF PHYSICAL AND STRUCTURAL CHARACTERISTICS FOR 8x8 AND 9x9 FUEL ASSEMBLIES

Property	Fuel Types		
	ENC 9x9	ENC 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.

APPENDIX C

INCREASED CORE FLOW (ICF) AND FINAL FEEDWATER TEMPERATURE REDUCTION (FFTR) PLANT TRANSIENTS RESULTS

Load rejection without bypass, feedwater controller failure, and MSIV closure were evaluated at increased core flow and final feedwater temperature reduction combinations. The delta-CPR's are given in Tables C.1 and C.2.

Table C.1

RESULTS OF SYSTEM PLANT TRANSIENT ANALYSIS AT
INCREASED CORE FLOW AND AT REDUCED FEEDWATER TEMPERATURE

Load Rejection Without Bypass

% Power/% Flow	Maximum Neutronic Flux (% Rated)	Maximum Core Average Heat Flux (% Rated)	Maximum System Pressure (psia)	Delta CPR
100/100 (FFTR)*	253	112.9	1191	0.15
100/108 (NFT)**	241	112.1	1210	0.17
100/108 (FFTR)	222	110.8	1187	0.15

ASME Overpressure (MSIV Closure) (psig)

	Vessel Dome	Vessel Lower Plenum	Steam Line
100/100 (FFTR)	1264	1279	1265
100/108 (NFT)	1290	1307	1296
100/108 (FFTR)	1257	1274	1259

*Final Feedwater Temperature Reduction (65°F).

**Nominal Feedwater Temperature.

Table C.2

FEEDWATER CONTROLLER FAILURE DELTA CPR
OF ICF AND FFTR ANALYSIS

	Nominal Feedwater Temp.		FFTR*	
% Power/% Flow	GE P8x8R	ENC 9x9	GE P8x8R	ENC 9x9
100 / 100	0.14	0.15	0.16	0.17
80 / 100	0.22	0.24	0.20	0.22
65 / 100	0.23	0.25	0.24	0.26
40 / 100	0.26	0.29	0.26	0.29
100 / 108	0.15	0.16	0.16	0.17
80 / 108	0.20	0.22	0.20	0.22
65 / 108	0.23	0.25	0.24	0.26
40 / 108	0.27	0.30	0.26	0.30

*65°F reduction in Feedwater Temperature.

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SUSQUEHANNA UNIT 2 CYCLE 2 RELOAD ANALYSIS

Design and Safety Analyses

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