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SUSQUEHANNA UNIT 1
CYCLE 3 RELOAD ANALYSIS

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SUSQUEHANNA UNIT 1 CYCLE 3 RELOAD ANALYSIS

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SUSQUEHANNA UNIT 1 CYCLE 3 RELOAD ANALYSIS

1.0 INTRODUCTION

This report provides the results of the analyses performed by Exxon Nuclear Company (ENC) in support of the Cycle 3 reload for Susquehanna Unit 1, 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 Susquehanna Unit 1 Cycle 3 core will comprise a total of 764 fuel assemblies, including 296 unirradiated ENC XN-2 8x8 assemblies, 192 once-irradiated ENC XN-1 8x8 assemblies, and 276 twice-irradiated Type III 8x8 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.

2.0 FUEL MECHANICAL DESIGN ANALYSIS

Applicable Fuel Design Report: References 9.1 and 9.5

Qualification analyses provided in the two references are both applicable to the Susquehanna Unit 1 XN-2 fuel.

The expected power history for the fuel to be irradiated during Cycle 3 of Susquehanna Unit 1 is bounded by the assumed power history in the fuel mechanical design analysis.

3.0 THERMAL HYDRAULIC DESIGN ANALYSIS

3.2 HYDRAULIC CHARACTERIZATION

3.2.3 Fuel Centerline Temperature

Exposure at Minimum Margin Point	5000 MWD/MT
Centerline Temperature at 120% Power	4057 F
Melting Point of Fuel	5000 F
Margin to Centerline Melting	943 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	3293 MW
Feedwater Flowrate (at SLMCPR)	15.6 Mlb/hr
Steam Dome Pressure (at SLMCPR)	1045 psia
Feedwater Temperature	382.7 F

3.3.2 Design Basis Radial Power Distribution

See Figure 3.1

3.3.3 Design Basis Local Power Distribution

See Figure 3.2

4.0 NUCLEAR DESIGN ANALYSIS

4.1 FUEL BUNDLE NUCLEAR DESIGN ANALYSIS

Assembly Average Enrichment	2.89 w/o
Radial Enrichment Distribution	Figure 4.1
Axial Enrichment Distribution	Uniform 2.98 w/o with 6 in. natural urania at top
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 EOC2	14,550 MWD/MT
Core Exposure at BOC3	7,770 MWD/MT
Core Exposure at EOC3	18,270 MWD/MT

Note: Cycle 3 safety analyses are valid for EOC2 exposure from -700 MWD/MT to +700 MWD/MT from the nominal value reported above.

4.2.2 Core Reactivity Characteristics

BOC Cold K-effective, All Rods Out	1.11157
BOC Cold K-effective, Strongest Rod Out	0.97790
Reactivity Defect(R-Value)	0.27% rho
Standby Liquid Control System Reactivity, Cold Conditions, 660 ppm, Keffective	0.95539

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)
Loss of Feedwater Heating (LFWH)
Feedwater Controller Failure (FWCF)

Event	Power	Flow	Maximum Heat Flux	Maximum Power	Maximum Pressure	Delta-CPR	Model
LRWB	100%	100%	108.4%	259%	1213 psia	0.17	COTRANSA
LFWH	100%	100%	112.0%	118%	1067	0.15	PTSBWR3
FWCF	100%	100%	108.6%	227%	1177	0.15	COTRANSA

5.2 ANALYSES FOR REDUCED FLOW OPERATION — Reference 9.4

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

5.4 ASME OVERPRESSURIZATION ANALYSIS — Reference 9.3

Limiting Event	MSIV Closure
Worst Single Failure	MSIV Position Scram
Maximum Pressure	1275 psig
Maximum Steam Dome Pressure	1254 psig

5.5 CONTROL ROD WITHDRAWAL ERROR

Starting Control Rod Pattern for Analysis — Figure 5.1

Rod Block Reading	Distance Withdrawn	Delta-CPR
105%	5.0 ft	0.18
106%*	5.5 ft	0.19
107%	7.0 ft	0.21
108%*	9.0 ft	0.23

*Rod block settings of 106% and 108% selected for Cycle 3

5.6 FUEL MISLOADING ERROR

Maximum Delta-CPR — 0.18

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%	0.15	1.21
FWCF	100%	100%	0.15	1.21
CRWE(106%)	100%	100%	0.19	1.25
CRWE(108%)	100%	100%	0.23	1.29

MCPR Operating Limits at Rated Conditions

Fuel Type	MCPR Limit (106%)	MCPR Limit (108%)
ENC 8x8	1.25	1.29
G.E. 8x8	1.25	1.29

MCPR Operating Limits at Off-Rated Conditions

Reduced Flow MCPR Limits	Figure 5.2

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	Reference 9.7

Limiting Break: Double-Ended Guillotine Break
Recirculation Pump Discharge Line
0.4 Break Coefficient
(0.4 DEG/PD)

Bundle Average Exposure	MAPLHGR	Peak Clad Temperature +	Peak Local MWR
0 MWD/MT	13.0 kW/ft	2074 F	1.9%
5.000	13.0	2093	2.0
10.000	13.0	2116	2.1
15.000	13.0	2136	2.2
19.000	13.0	2147	2.3
25.000	11.5	1977	1.6
30.000	10.4	1846	1.0
35.000	10.4	1852	1.2

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

6.2 CONTROL ROD DROP ACCIDENT

Reference 8.1

Dropped Control Rod Worth	5.8 mk
Doppler Coefficient	-9.5 x (10)-6 1/k dk/dT
Effective Delayed Neutron Fraction	0.0050
Four-Bundle Local Peaking Factor	1.30
Maximum Deposited Fuel Rod Enthalpy	83 cal/gm

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	1352 psig

7.2 LIMITING CONDITIONS FOR OPERATION

7.2.1 Average Planar Linear Heat Generation Rate Limits for ENC XN-1 and XN-2 8x8 Fuel

Bundle Average Exposure	MAPLHGR
0 MWD/MT	13.0 kW/ft
5.000	13.0
10,000	13.0
15.000	13.0
19,000	13.0
25.000	11.3*
30,000	9.8*
35.000	8.3*

*Values lower than those supported in the ECCS analysis are used for consistency with the mechanical design analysis

7.2.2 Minimum Critical Power Ratio

Rated Conditions MCPR Limits

Fuel Type	Limit (106%)	Limit (108%)
ENC 8x8 XN-1,2	1.25	1.29
GE 8x8 Type III	1.25	1.29

Off-Rated Conditions MCPR Limits

Reduced Flow MCPR Limit Figure 5.2

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 time is required for validation of thermal limits.

7.3.2 Stability Surveillance

Stability surveillance established to provide assurance of stable operation during Cycle 2 shall be continued during Cycle 3.

7.3.3 Procedural Controls

Procedural controls shall be established to assure that the operation of the fuel remains within the power distribution assumptions of the fuel mechanical design analysis.

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 (July 1985).

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

9.3 "Susquehanna Unit 1 Cycle 3 Plant Transient Analysis," XN-NF-85-130, Exxon Nuclear Company (December 1985).

9.4 "Susquehanna Unit 1 Cycle 3 Plant Transient Analysis," XN-NF-85-130, Exxon Nuclear Company (December 1985).

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

9.6 "Generic LOCA Break Spectrum Analysis Jet-Pump BWR 3/4 With Modified Low Pressure Coolant Injection Logic," XN-NF-84-117, Exxon Nuclear Company (December 1984).

9.7 "Susquehanna Unit 1 LOCA-ECCS Analysis, MAPLHGR Results," XN-NF-84-119, Exxon Nuclear Company (December 1984).

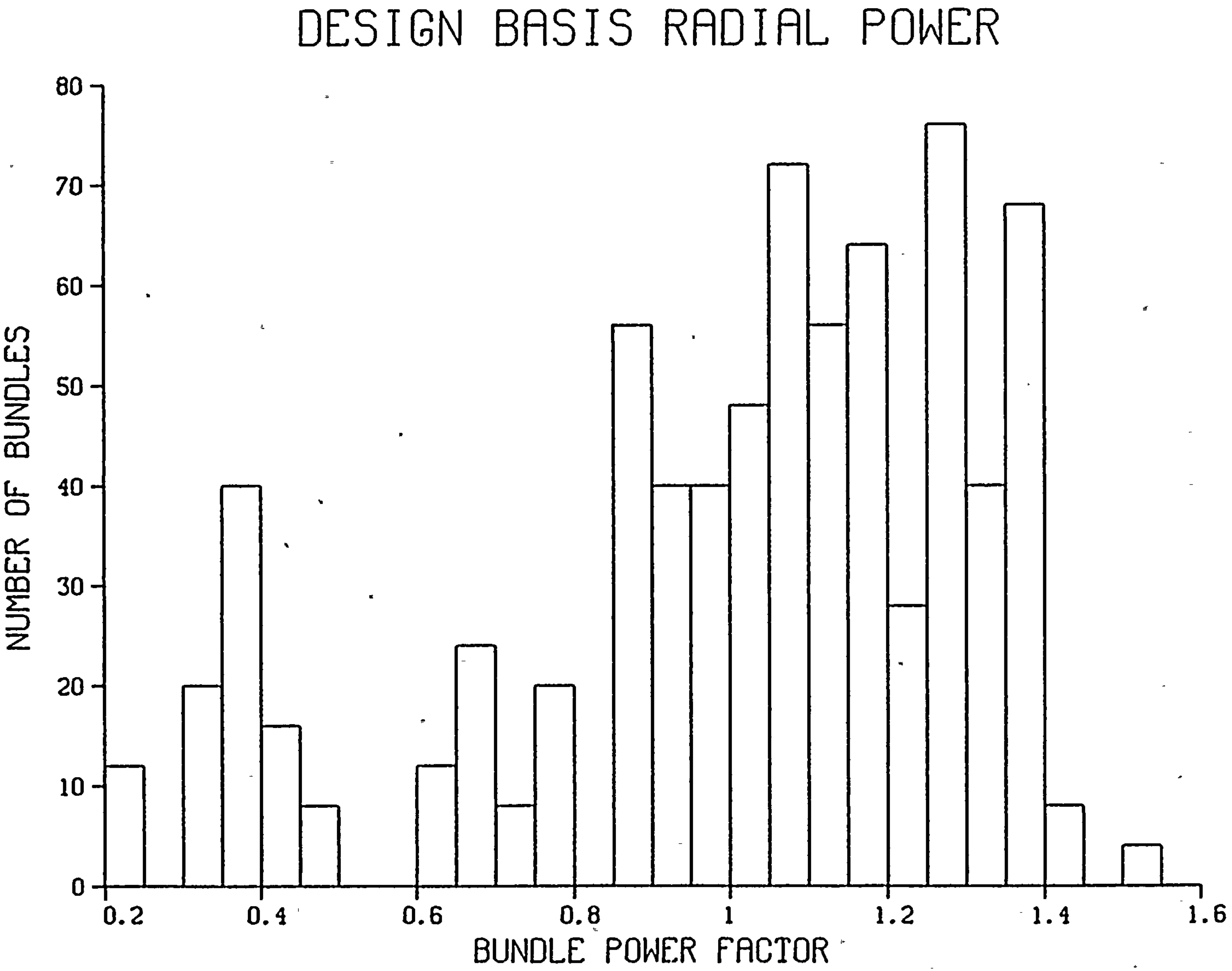


Figure 3.1 Susquehanna Unit 1 Cycle 3 Safety Limit Radial Power Histogram

LL 0.92	L 0.95	LM 0.98	HM 1.02	HM 1.02	HM 1.04	LM 1.01	L 0.99
L 0.95	HM 1.02	LM* 0.91	H 1.05	H 1.05	HM 0.97	LM* 0.94	LM 1.02
LM 0.98	LM* 0.91	H 1.04	H 1.01	H 1.00	H 1.02	HM 0.97	HM 1.04
HM 1.02	H 1.05	H 1.01	W 0.00	HM 0.90	H 1.00	H 1.05	HM 1.02
HM 1.02	H 1.05	H 1.00	HM 0.90	W 0.00	H 1.01	H 1.05	HM 1.02
HM 1.04	HM 0.97	H 1.02	H 1.00	H 1.01	H 1.04	LM* 0.90	HM 1.04
LM 1.01	LM* 0.94	HM 0.97	H 1.05	H 1.05	LM* 0.90	LM 0.96	LM 1.02
L 0.99	LM 1.02	HM 1.04	HM 1.02	HM 1.02	HM 1.04	LM 1.02	L 0.99

Figure 3.2 Susquehanna Unit 1 Cycle 3 Safety Limit Local Peaking Factors (ENC Fuel)

LL	L	LM	HM	HM	HM	LM	L
L	HM	LM*	H	H	HM	LM*	LM
LM	LM*	H	H	H	H	HM	HM
HM	H	H	W	HM	H	H	HM
HM	H	H	HM	W	H	H	HM
HM	HM	H	H	H	H	LM*	HM
LM	LM*	HM	H	H	LM*	LM	LM
L	LM	HM	HM	HM	HM	LM	L

LL Rods (1) --- 1.45 w/o U235
L Rods (5) --- 1.95 w/o U235
LM Rods (9) --- 2.48 w/o U235
HM Rods (21) --- 2.94 w/o U235
H Rods (20) --- 3.72 w/o U235
LM* Rods (6) --- 2.48 w/o U235 + 4.00 w/o Gd2O3
W Rods (2) --- Inert Water Rod

Figure 4.1 Susquehanna Unit 1 XN-2 8x8 Fuel Enrichment Distribution

A2	A2	C0	A2	C0	A2	C0	A2	C0	A2	C0	A2	C0	B1	A2
A2	C0	B1	C0	B1	C0	B1	C0	B1	C0	B1	C0	B1	C0	A2
C0	B1	A2	A2	C0	A2	C0	A2	C0	A2	C0	A2	C0	B1	A2
A2	C0	A2	C0	B1	C0	B1	C0	B1	C0	B1	C0	B1	C0	A2
C0	B1	C0	B1	A2	A2	C0	A2	C0	A2	C0	A2	C0	B1	A2
A2	C0	A2	C0	A2	C0	B1	C0	B1	C0	B1	C0	B1	B1	A2
C0	B1	C0	B1	C0	B1	A2	A2	C0	A2	C0	A2	C0	A2	A2
A2	C0	A2	C0	A2	C0	A2	C0	B1	C0	B1	C0	B1	A2	
C0	B1	C0	B1	C0	B1	C0	B1	A2	A2	C0	B1	A2		
A2	C0	A2	C0	A2	C0	A2	C0	A2	A2	B1	A2	A2		
C0	B1	C0	B1	C0	B1	C0	B1	C0	B1	A2				
A2	C0	A2	C0	A2	C0	A2	C0	B1	A2					
C0	B1	C0	B1	C0	B1	C0	B1	A2	A2					
B1	C0	B1	C0	B1	B1	A2	A2							
A2	A2	A2	A2	A2	A2	A2								

XY = Fuel Type X
Burned Y Cycles

Fuel Type	No. of Bundles	Description
A	276	GE 8x8 Type III 2.19 w/o U-235
B	192	XN-1 8x8 2.72 w/o U-235
C	296	XN-2 8x8 2.89 w/o U-235

Figure 4.2 Susquehanna Unit 1 Cycle 3 Reference Core Loading

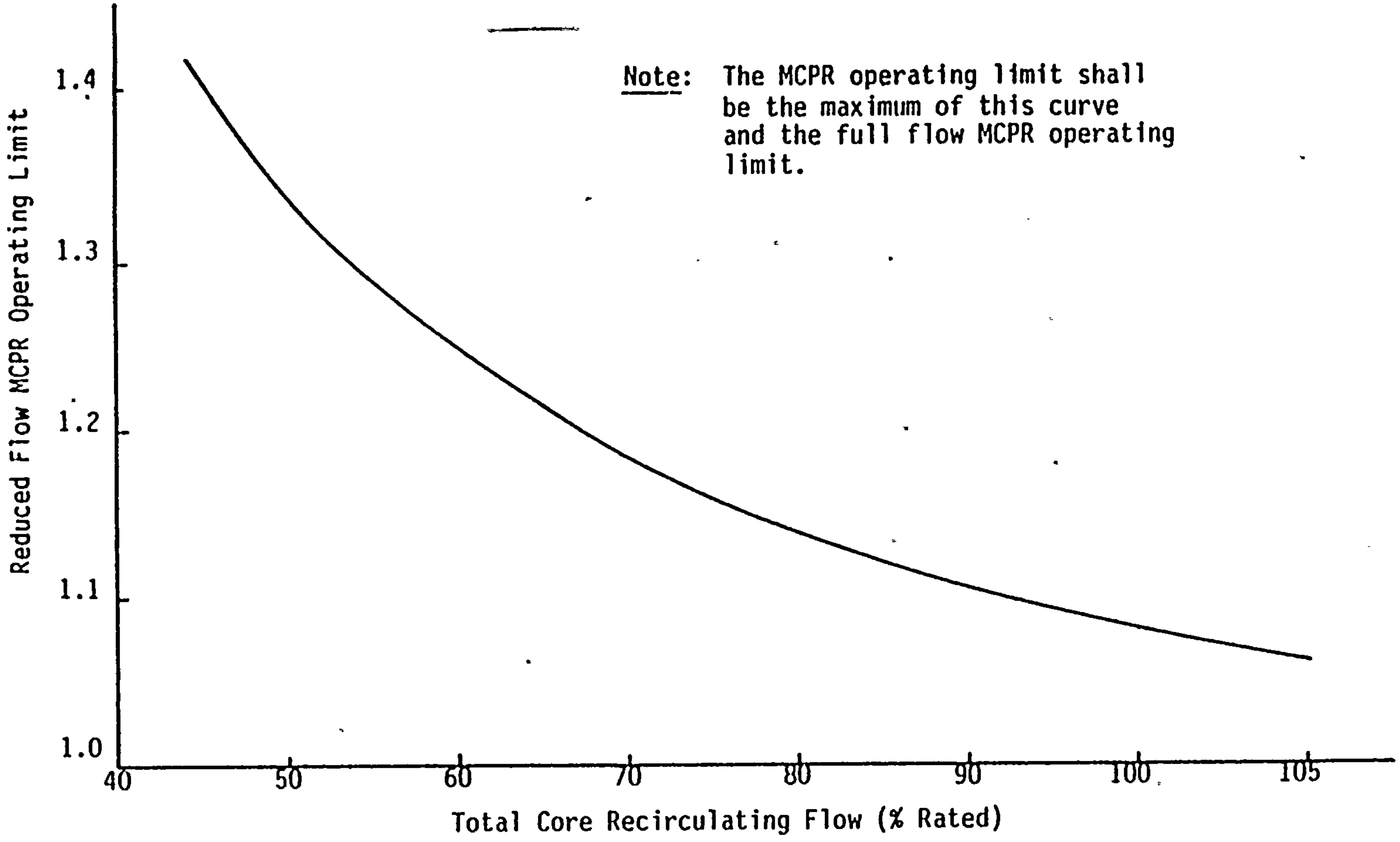
Table 4.1 Neutronic Design Values

Fuel Pellet	Reference 9.1
Fuel Rod	Reference 9.1
Fuel Assembly	Reference 9.1
Core Data	
Number of fuel assemblies	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	0.080
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	B4C
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

	02	06	10	14	18	22	26	30	34	38	42	46	50	54	58	
59					--	--	--	--	--	--	--					59
55				--	--	30	--	--	--	30	--	--				55
51			--	--	00	--	00	--	00	--	00	--	--			51
47		--	--	--	--	36	--	--	--	36	--	--	--	--		47
43	--	--	00	--	12	--	12	--	12	--	12	--	00	--	--	43
39	--	30	--	36	--	--	--	--	--	--	--	36	--	30	--	39
35	--	--	00	--	12	--	00	--	00	--	12	--	00	--	--	35
31	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	31
27	--	--	00	--	12	--	00	--	--*	--	12	--	00	--	--	27
23	--	30	--	36	--	--	--	--	--	--	--	36	--	30	--	23
19	--	--	00	--	12	--	12	--	12	--	12	--	00	--	--	19
15		--	--	--	--	36	--	--	--	36	--	--	--	--		15
11			--	--	00	--	00	--	00	--	00	--	--			11
07				--	--	30	--	--	--	30	--	--				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 1 Cycle 3 Control Rod Withdrawal Error Analysis Initial Control Rod Pattern



Note: The MCPR operating limit shall be the maximum of this curve and the full flow MCPR operating limit.

Figure 5.2 Reduced Flow MCPR Operating Limit

APPENDIX A

SINGLE LOOP OPERATION

1.0 INTRODUCTION

The NSSS supplier, General Electric (GE), 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.

2.0 LOSS-OF-COOLANT ACCIDENT ANALYSIS

To support operation of Susquehanna Unit 1 with ENC 8x8 fuel with a single recirculation pump operating. ENC recommends the conservative use of GE fuel MAPLHGR limits for the similar Type III 8x8 fuel design with a multiplier of

0.81 applied for single loop operation. The basis for this recommendation is as follows:

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

(2) The analytical methods used by GE have yielded conservative MAPLHGR limits relative to the MAPLHGR limits obtained using the approved ENC analytical methods.

Therefore, applying the more conservative GE MAPLHGR limits to ENC fuel provides a limit which assures conformance with the criteria of 10 CFR 50.46.

The major differences 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 conditions. 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 G.E., similar results would be expected from an ENC analysis because the phenomena are governed by the system parameters.

The major difference between the ENC and GE methodologies that would impact 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 using an adiabatic heatup period and assuming an early boiling transition (0.1 sec) for single loop operation would add more conservatism to an ENC single loop analysis than to a GE analysis.

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 similarly conservative when applied to the equivalent ENC 8x8 fuel.

Application of the GE calculated MAPLHGR limits to ENC fuel for single loop operation will conservatively assure that the 10 CFR 50.46 criteria will be met for the following reasons:

(1) Since ENC has performed LOCA analyses for Susquehanna Unit 1 for two loop operation and obtained MAPLHGR limits for ENC fuel which are higher than the equivalent GE limits, an ENC analysis for the similar single loop operating conditions would also be expected to give 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 also conservative when applied to the equivalent ENC 8x8 fuel because the physical characteristics of the fuel are essentially equivalent. The similarities between ENC and GE fuel are shown in Table A-1. The only differences worth noting are the smaller pellet to cladding gap and more open tie plate of the ENC fuel design. Both of these design features are beneficial in an ECCS analysis. The smaller gap results in less initial stored energy. The more open upper tie plate allows more countercurrent liquid downflow to cool the fuel and refill the core faster.

(3) ENC two loop MAPLHGR limits, in addition to being higher than the GE limits, have significant margin to peak cladding temperature limits for most exposure conditions. This further assures that GE results will be conservative relative to ENC calculations.

3.0 ANTICIPATED OPERATIONAL OCCURRENCES

MCPR limits established for full flow two loop operation are conservative for single loop operation because of the physical phenomena related to partial power, partial flow operation rather than modeling features. A review of the most limiting transients for single loop operation was conducted for thermal margin effects. Under single loop conditions, steady state operation cannot exceed approximately 75% power and 60% core flow because of the capability of the operating recirculation pump. Thus, the technical specification change to allow single loop operation with the MCPR limits established for two loop operation up to 100% power are conservative for single loop operation.

3.1 Load Rejection Without Bypass (LRWB)

The limiting transient for the Susquehanna units is the Load Rejection Without Bypass pressurization transient. In this transient, the primary phenomenon 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 thermal margin reduction are reduced below those of the full power 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 (Ref. 1) and preliminary ENC analyses are reduced power and flow conditions with two loop operation confirm this trend, and GE analyses (Ref. 2) under single loop conditions also confirm this trend.

3.2 Feedwater Controller Failure (FWCF)

The second most limiting transient for Susquehanna is the Feedwater Controller Failure to Maximum Demand. 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 traverse the core such that a high level trip occurs before the core power level rises as much as in the full power case. In the subsequent pressurization transient, the result of the turbine trip is less severe for the reduced powers in transients from single loop conditions because of the reasons discussed in the LRWB transient section.

Thus, because of the slower transport phenomena caused by the lower flow in the downcomer and because of the lower steamline flow in the pressurization portion of the transient, the FWCF has larger margin to the operating limit.

3.3 Summary

It is very conservative to use the reduced flow MCPR limit for single loop operation. The reduced flow MCPR limit is to protect against boiling transition during flow excursions to greater than full flow; excursions to such high flows are not possible during single loop conditions. 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.

References

(1) "Extended Load Line Limit Analysis for Susquehanna Steam Electric Station Unit 1," NEDO-22128, General Electric Company (May 1982).

(2) "Susquehanna Single Loop Operation Analysis." GP-84-142, General Electric Company (June 1984).

Table A-1 Similarities in ENC and GE Fuels for SSES Unit 1 Affecting ECCS Analysis

Fuel Parameter	ENC 8x8	GE 8x8R
Number of fuel rods	62	62
Fuel rod OD (in)	0.484	0.483
Pellet-cladding cold gap (in)	0.0085	0.009
Pellet OD (in)	0.4055	0.410
Rod pitch (in)	0.641	0.640
Bare bundle hydraulic diameter (ft)	0.0452	0.0446
Bare bundle rod flow area (in2)	15.96	15.82
Upper tie plate flow area (in2)	15.4	11.4
Prepressurization (atm)	3	3

SUSQUEHANNA UNIT 1 CYCLE 3 RELOAD ANALYSIS

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