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WNP-2 CYCLE 2 RELOAD ANALYSIS

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WNP-2 CYCLE 2 RELOAD ANALYSIS

Design and Safety Analyses

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 the Supply System Nuclear Project Number 2 (WNP-2). WNP-2 is scheduled to commence Cycle 2 operation in June 1986. This report is intended to be used in conjunction with ENC topical report XN-NF-80-19(A), Volume 4, "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(A), Volume 4. Appendix A of this report presents a seismic-LOCA evaluation of ENC 8x8 fuel, and Appendix B addresses single loop operation.

The WNP-2 Cycle 2 core will comprise a total of 764 fuel assemblies, including 132 unirradiated ENC XN-1 8x8 assemblies, and 632 previously irradiated 8x8 assemblies fabricated by General Electric (GE). The reference core configuration is described in Section 4.2.

As a result of a change in energy requirements since the Cycle 2 analysis commenced, the number of unirradiated ENC assemblies to be actually loaded in Cycle 2 is to be lower than the 196 8x8 assemblies quoted in the System Transient Analysis Report (Reference 9.3). The control rod withdrawal error (CRWE) event was evaluated for the appropriate core loading of 132 ENC 8x8 assemblies and found to be the

most limiting transient. Since the core loading used in the system transient analysis approximates the actual core loading, it is expected that the results presented herein are representative to the Cycle 2 reactor core. Thus, the review of this current document by the U.S. NRC can begin and proceed as there are expected to be minor changes, if any, needed in this documentation after the reanalysis of the system transients with the final loading pattern.

The design and safety analyses reported in this document were based on the design and operational assumptions in effect for WNP-2 during the previous operating cycle except the analyses were expanded to encompass increased core flow up to 106%.

2.0 FUEL MECHANICAL DESIGN ANALYSIS

Applicable Fuel Design Report: Reference 9.1 and 9.6

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

The expected power history for the fuel to be irradiated during Cycle 2 of WNP-2 is bounded by the assumed power history in the fuel mechanical design analyses. ENC analyses have confirmed that centerline melting and 1% uniform total clad strain will not be exceeded during all anticipated operational occurrences. Furthermore, the centerline melt criteria are more restrictive than the 1% uniform total clad strain criteria. In addition the internal rod pressure in the ENC fuel was conservatively calculated to be below system pressure for Cycle 2 exposures using this assumed power history.

3.0 THERMAL HYDRAULIC DESIGN ANALYSIS

3.1 DESIGN CRITERIA

3.1.3 Fuel Centerline Temperature

Exposure at Minimum Margin Point	5000 MWD/MTM
Centerline Temperature at 120% Power	4065°F
Melting Point of Fuel	5000°F
Margin to Centerline Melting	935°F

3.2 HYDRAULIC CHARACTERIZATION

3.2.5 Bypass Flow

Calculated Bypass Flow Fraction	11.6%

3.3 MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT

3.3.1 Coolant Thermodynamic Condition

Core Power	3754 MWt
Core Inlet Enthalpy	527.7 BTU/lbm
Steam Dome Pressure	1031 psia
Feedwater Temperature	420°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	2.72 w/o U-235
Radial Enrichment Distribution	Figure 4.1
Axial Enrichment Distribution	Uniform 2.89 w/o U-235 with 6 inch top and bottom natural uranium blankets
Burnable Poisons	Figure 4.1
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 EOC1 (MWD/MTM)	8,161
Core Exposure at BOC2 (MWD/MTM)	7,424
Core Exposure at EOC2 (MWD/MTM)	13,528

4.2.2 Core Reactivity Characteristics

BOC Cold K-effective, All Rods Out	1.1062
BOC Cold K-effective, Strongest Rod Out	0.9689
Reactivity Defect/R-Value, % delta k/k	1.43
Standby Liquid Control System (SBLC)	
Reactivity, 660 PPM Boron, K-effective	0.9595

4.2.3 Stability Analysis

Reactor Core Stability	Figure 4.3
Maximum Decay Ratio Value 100% Flow Control Line	0.45

4.2.4 Core Hydrodynamic Stability

WNP-2 has adopted a detect and suppress approach to assuring hydrodynamic stability during plant operation. Because of differences in the relative pressure drop characteristics of the upper and lower tie plate designs between the ENC XN-1 fuel and G.E. 8x8R fuel, the ENC fuel is slightly more stable than the G.E. fuel. The detect and suppress operating requirements in the Technical Specifications implemented during Cycle 1 are to continue in Cycle 2 to provide assurance of stable core operation for Cycle 2.

The stability analysis results shown in Figure 4.3 are included only to be informative, and stability analyses are not necessary to support the detect and suppress plant operational approach to hydrodynamic stability.

5.0 ANTICIPATED OPERATIONAL OCCURRENCES

Applicable Transient Analysis Report Reference 9.2

5.1 ANALYSIS OF PLANT TRANSIENTS AT RATED CONDITIONS Reference 9.3

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

Transient*	Maximum Heat Flux	Maximum Power	Maximum Pressure	Delta-CPR
LRWB	116%	320%	1170 psig	0.18
FWCF	109%	147%	1158 psig	0.08
LFWH	123%	126%	1069 psig	0.16

5.2 ANALYSES FOR REDUCED FLOW OPERATION Reference 9.3

Limiting Transient: Recirculation Flow Increase

5.3 ASME OVERPRESSURIZATION ANALYSIS Reference 9.3

Limiting Event	MSIV Closure
Worst Single Failure	MSIV Position Scram Trip
Maximum Pressure	1317 psig
Maximum Steam Dome Pressure	1288 psig

5.4 CONTROL ROD WITHDRAWAL ERROR

Initial Control Rod Pattern for CRWE Analysis Figure 5.1

Rod Block Setting	Distance Withdrawn (ft)	ENC Delta-CPR	GE Delta-CPR
106%**	4.0	0.21	0.22
107%	4.5	0.23	0.24
108%	5.0	0.24	0.26

5.5 FUEL LOADING ERROR

Delta CPR 0.11

---

*All transients are based on measured plant scram insertion data, see Section 7.2.3.1.

**Rod Block Setting (RBS) of 106% for Cycle 2.

5.6 DETERMINATION OF THERMAL MARGINS

Summary of Thermal Margin Requirements

Event	Delta-CPR	MCPR Limit	Model
LRWB	0.18*	1.24	COTRANSA
FWCF	0.08	1.14	COTRANSA
LFWH	0.16	1.22	PTSBWR3
CRWE	0.22	1.28**	XTGBWR

MCPR Operating Limits at Rated Conditions

Fuel Type	MCPR Limit (106% RBS)
ENC XN-1 8x8	1.27
GE P8x8R	1.28

MCPR Operating Limits at Off-Rated Conditions Figure 5.2

Reduced Flow MCPR Limit Reference 9.3

6.0 POSTULATED ACCIDENTS

6.1 Loss-of-Coolant Accident

---

*0.18 at 104% power/100% flow and 0.19 at 104% power/106% flow.

**1.27 for ENC fuel, 1.28 for GE fuel.

6.1.1 Break Location Spectrum Reference 9.4

6.1.2 Break Size Spectrum Reference 9.4

6.1.3 MAPLHGR Analyses Reference 9.5

Limiting Break: Split Break in the Recirculation Suction Piping With an Area Equal to Sixty Percent of the Double-Ended Cross-Sectional Pipe Area

Bundle Average Exposure (MWD/MTM)	MAPLHGR (kw/ft)	Peak Clad Temperature, °F	Peak Local MWR, %
0	13.0	1765	0.49
5,000	13.0	1766	0.48
10,000	13.0	1765	0.47
15,000	13.0	1772	0.47
20,000	13.0	1788	0.54
25,000	11.3	1699	0.34
30,000	9.4	1521	0.17
35,000	7.9	1397	0.10

6.2 Control Rod Drop Accident Reference 9.7

Dropped Control Rod Worth, mK	6.6
Doppler Coefficient dK/KdT, 1/°F	-9.5×10^{-6}
Effective Delayed Neutron Fraction	0.0050
Four-Bundle Local Peaking Factor	1.26
Maximum Deposited Fuel Rod Enthalpy (cal/gm)	98.

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 1345 psig

7.2 Limiting Conditions for Operation

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

Bundle Average Exposure (MWD/MTM)	MAPLHGR (Kw/ft)
0	13.0
5,000	13.0
10,000	13.0
15,000	13.0
20,000	13.0
25,000	11.3
30,000	9.4
35,000	7.9

7.2.2 Minimum Critical Power Ratio

Rated Conditions MCPR Limits (100% to 106% flow)

Fuel Type	Limit
ENC XN-1 8x8	1.27
GE P8x8R	1.28

Off-Rated Conditions MCPR Limit

Reduced Flow MCPR Limit Figure 5.2

7.2.3 Surveillance Requirements

7.2.3.1 Scram Insertion Time Surveillance

The ENC reload safety analyses were performed using the control rod insertion times shown below which are based on plant data. In the event that plant surveillance shows these scram insertion times may be exceeded, the plant thermal margin limits are to default to the values which correspond to the technical specification control rod scram times.

Position Inserted From Fully Withdrawn	Average Rod Time in Seconds as Defined in Footnote*
Notch 45	0.404
Notch 39	0.660
Notch 25	1.504
Notch 5	2.624

The limiting transient using technical specification control rod scram times is the generator load rejection without bypass. The MCPR for both ENC and GE fuel during Cycle 2 is 1.32 using the technical specification control rod speeds. The use of technical specification control rod insertion times on all of the core transients investigated has been shown to have no effect on the definition of the limiting transient, but only to change the magnitude of the delta CPR. This is discussed in reference 9.2.

7.2.3.2 Stability Surveillance

Detect and Suppress Procedures in accordance with the surveillance requirements implemented in the Technical Specifications during Cycle 1 are applicable during Cycle 2.

---

*Slowest measured average control rod insertion time to specified notches for each group of four control rods arranged in a 2x2 array.

7.2.3.3 Mechanical Design LHGR Surveillance

The mechanical design linear heat generation rate (LHGR) limit versus average planar exposure for ENC 8x8 reload fuel is shown in Figure 7.1. This figure was developed from information contained in reference 9.1, and the region of permissible operation is shown.

Table 4.1 Neutronic Design Values

Fuel Pellet	
Fuel Material	UO_2 Sintered Pellets
Density, g/cc	10.36
% of T.D.	94.5
Diameter	
Enriched Fuel	0.4055
Natural Fuel	0.4045
Fuel Rod	
Fuel Length, inches	150
Cladding Material	Zircaloy-2
Clad I.D., inches	0.414
Clad O.D., inches	0.484
Fuel Assembly	
Number of Fuel Rods	62
Number of Inert Water Rods	2
Fuel Rod Enrichments	Figure 4.1
Fuel Rod Pitch, inches	0.641
Fuel Assembly Loading, KgU	176.0

Table 4.1 Neutronic Design Values (Cont.)

<u>Core Data</u>	
Number of fuel assemblies	764
Rated thermal power, MW	3323
Rated core flow, Mlbm/hr	108.5
Core inlet subcooling, BTU/lbm	19.0
Reactor Pressure, psia	1019.0
Channel thickness, inch	0.100
Fuel assembly pitch, inch	6.00
Water gap thickness (symmetric), inch	0.522
<u>Control Rod Data</u>	
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

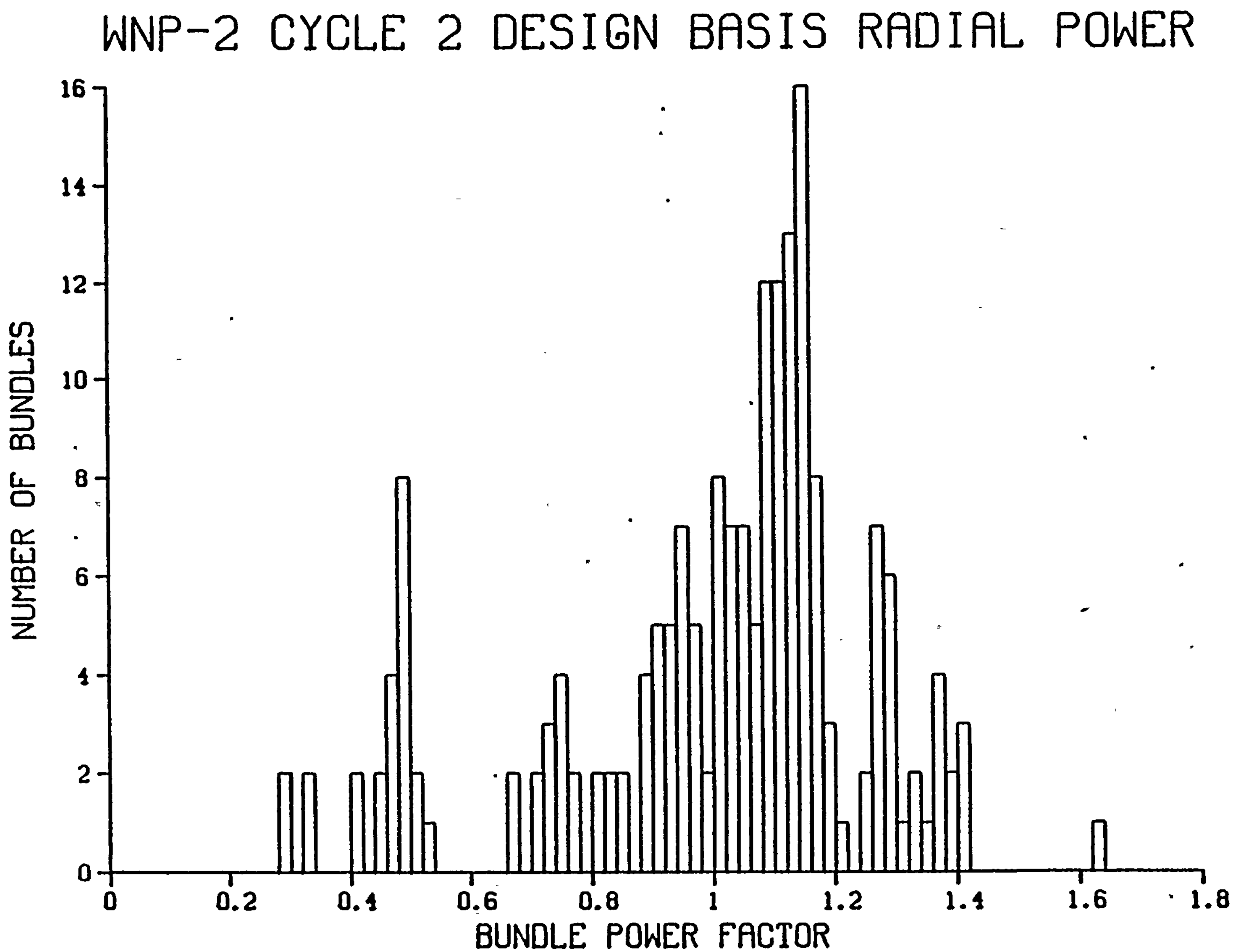


Figure 3.1 Radial Power Histogram For 1/4 Core Safety Limit Model

LL 0.91	L 0.95	ML 1.02	M 1.06	M 1.06	ML 1.02	L 0.95	LL 0.91
L 0.95	ML 0.97	H 1.08	ML* 0.87	H 1.04	H 1.07	M 1.04	L 0.95
ML 1.02	H 1.08	H 1.01	H 1.00	H 0.98	H 1.00	ML* 0.90	ML 1.02
M 1.06	ML* 0.87	H 1.00	W 0.00	M 0.90	H 0.98	H 1.04	M 1.06
M 1.06	H 1.04	H 0.98	M 0.90	W 0.00	H 0.99	M 0.93	M 1.05
ML 1.02	H 1.07	H 1.00	H 0.98	H 0.99	H 1.00	H 1.07	M 1.08
L 0.95	M 1.04	ML* 0.90	H 1.04	M 0.93	H 1.07	ML* 0.96	ML 1.07
LL 0.91	L 0.95	ML 1.02	M 1.06	M 1.05	M 1.08	ML 1.07	L 1.03

Figure 3.2 WNP-2 Cycle 2 Safety Limit
Local Power Factors (ENC Fuel)

XN-CH-0524

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

LL RODS (3) --- 1.50 W/O U235
L RODS (7) --- 2.00 W/O U235
ML RODS (9) --- 2.57 W/O U235
M RODS (16) --- 2.94 W/O U235
H RODS (22) --- 3.54 W/O U235
ML* RODS (5) --- 2.57 W/O U235 + 2.00 W/O GD203
W RODS (2) --- INERT WATER ROD

Figure 4.1 WNP-2 Cycle 2 (XN-1) Enriched Zone Enrichment Distribution

A	B	B	B	B	B	B	B	B	B	B	B	B	B	A
B	C	A	C	B	C	A	C	B	C	B	C	B	B	A
B	A	B	B	A	A	B	B	A	B	A	B	B	B	A
B	C	B	C	A	C	B	C	A	C	B	C	B	B	A
B	B	A	A	B	B	A	A	B	B	A	B	B	B	A
B	C	A	C	B	C	A	C	B	C	B	C	B	B	A
B	A	B	B	A	A	B	B	A	B	B	B	B	B	A
B	C	B	C	A	C	B	C	B	C	B	C	B	A	
B	B	A	A	B	B	A	B	B	B	B	B	A		
B	C	B	C	B	C	B	C	B	C	B	A	A		
B	B	A	B	A	B	B	B	B	B	A				
B	C	B	C	B	C	B	C	B	A					
B	B	B	B	B	B	B	B	A	A					
B	B	B	B	B	B	B	A							
A	A	A	A	A	A	A								

Fuel Type	Number of Assemblies	Description
A	200	GE 8x8 Type II 1.76 w/o U-235
B	432	GE 8x8 Type III 2.19 w/o U-235
C	132	XN-1 8x8 2.72 w/o U-235

Figure 4.2 WNP-2 Cycle 2 Reference Loading Pattern by Fuel Type (One Quarter of Symmetrical Core Loading)

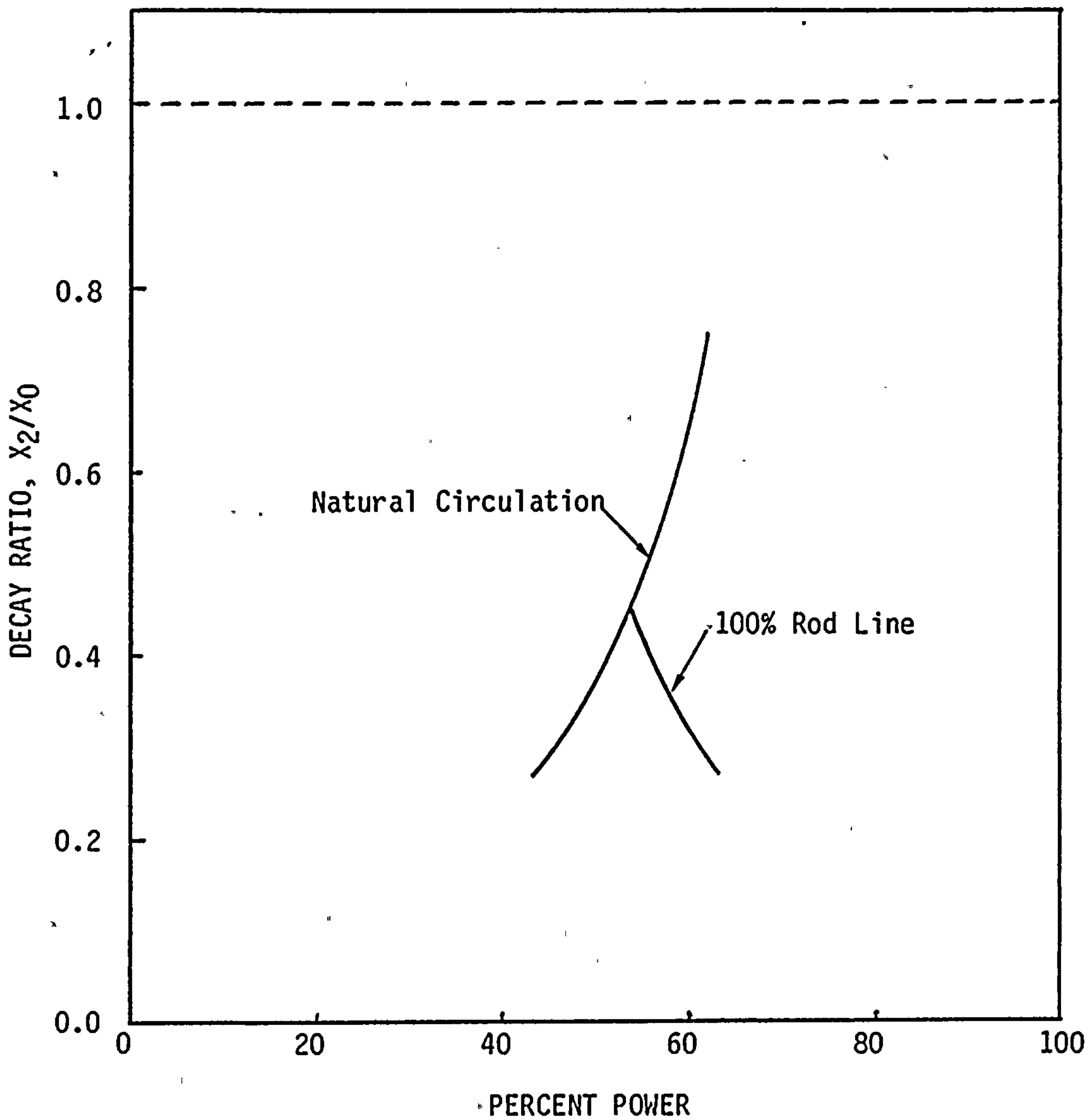


Figure 4.3 Decay Ratio vs. Reactor Power for WNP-2 Cycle 2

	02	06	10	14	18	22	26	30	34	38	42	46	50	54	58	
59					--	--	--	--	--	--	--					59
55				--	--	00	--	36	--	00	--	--				55
51			--	--	--	--	--	--	--	--	--	--	--			51
47		--	--	00	--	00	--	00	--	00	--	00	--	--		47
43	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	43
39	--	00	--	00	--	10	--	12	--	10	--	00	--	00	--	39
35	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	35
31	--	36	--	00	--	12	--	00*	--	12	--	00	--	36	--	31
27	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	27
23	--	00	--	00	--	10	--	12	--	10	--	00	--	00	--	23
19	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	19
15		--	--	00	--	00	--	00	--	00	--	00	--	--		15
11			--	--	--	--	--	--	--	--	--	--	--			11
07				--	--	00	--	36	--	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 5.1 WNP-2 Cycle 2 Control Rod Withdrawal Analysis
Initial Control Rod Pattern

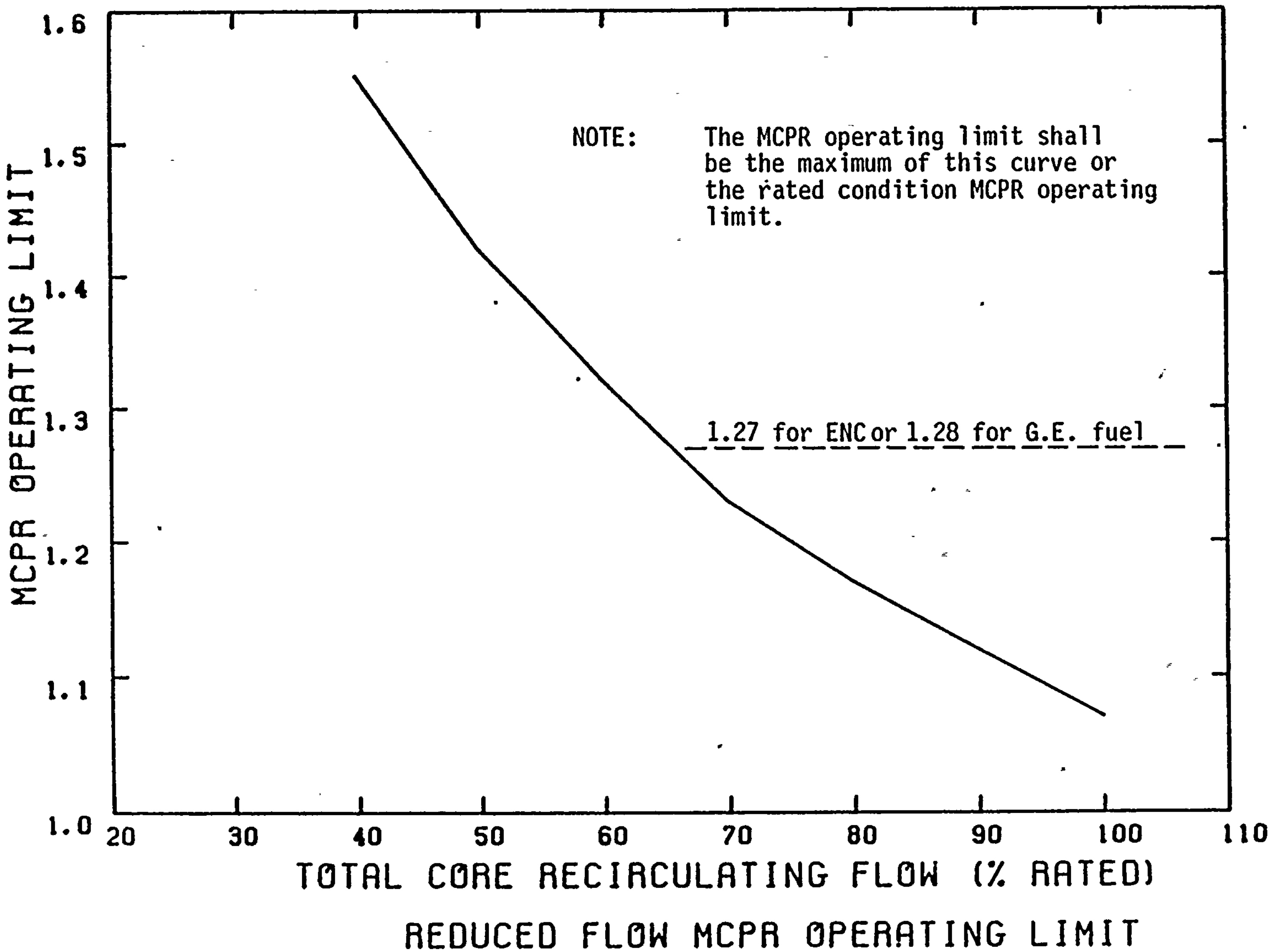
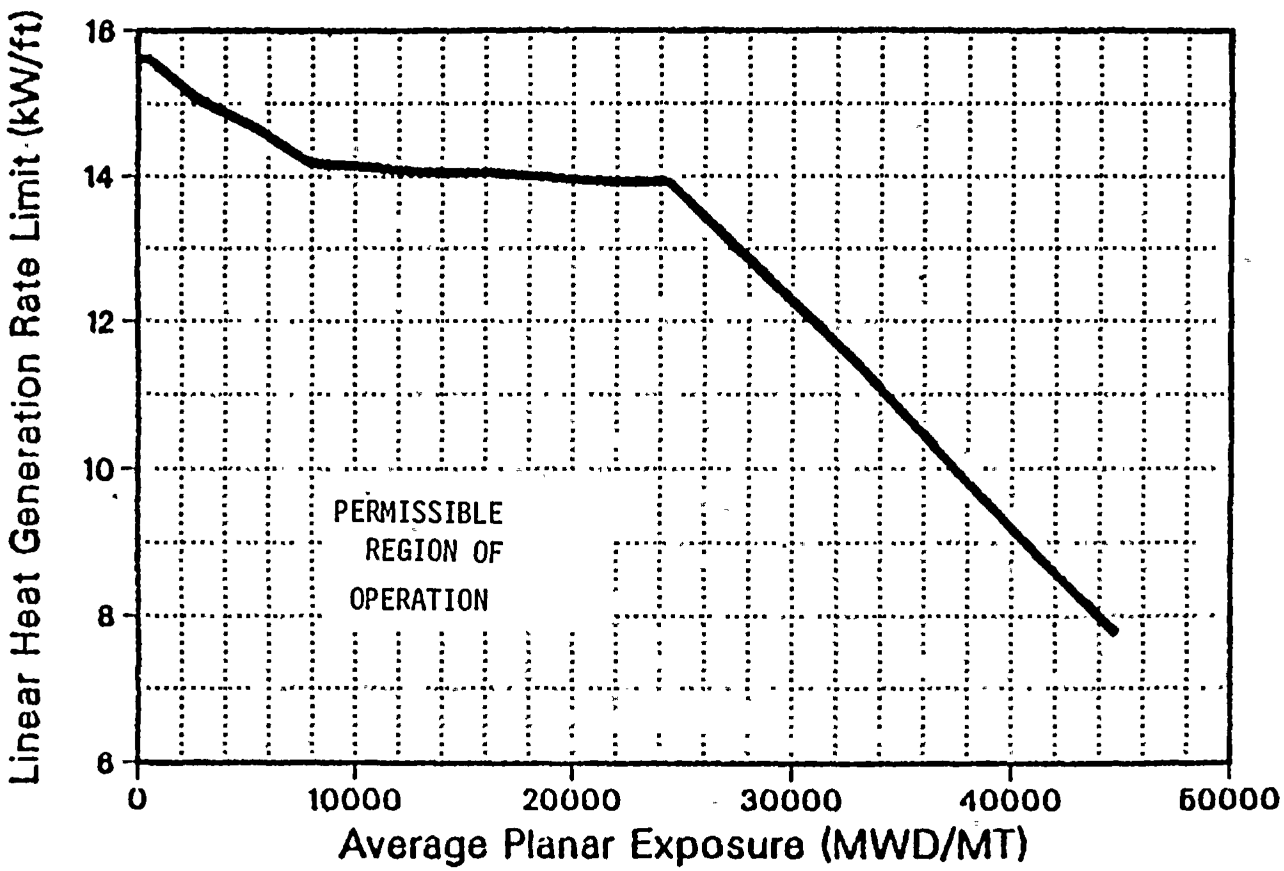


Figure 5.2 Reduced Flow MCPR Operating Limit



EXP	LHGR
0	15.62
610	15.62
2,580	15.10
5,230	14.71
7,940	14.19
10,470	14.13
13,220	14.06
15,990	14.06
18,780	14.00
21,590	13.93
24,420	13.93
27,280	13.08
30,150	12.24
33,050	11.40
35,960	10.47
38,900	9.56
41,830	8.65
44,760	7.77

Figure 7.1 LINEAR HEAT GENERATION RATE (LHGR) LIMIT VERSUS AVERAGE PLANAR EXPOSURE EXXON 8X8 FUEL

9.0 ADDITIONAL REFERENCES

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9.2 R.H. Kelley, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," XN-NF-79-71(P), Revision 2, Exxon Nuclear Co., Inc., Richland, WA 99352 (November 1981).

9.3 J.E. Krajicek, "WNP-2 Cycle 2 Plant Transient Analysis," XN-NF-85-143, Revision 1, Exxon Nuclear Co., Inc., Richland, WA 99352 (February, 1985).

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APPENDIX A

SEISMIC-LOCA EVALUATION

The structural response of the ENC XN-1 8x8 fuel is the same as the structural response of the GE P8x8R fuel it replaces in the WNP-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 geometric properties of the ENC XN-1 8x8 and the GE P8x8R 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 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 an equivalent fuel assembly channel box, and the channel box dominates the overall dynamic response of the incore fuel. ENC calculations show that more than 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 A.1 Comparison of Physical and Geometric Characteristics for ENC and GE 8x8 Assemblies

Property	Fuel Type	
	ENC	GE
Fuel Rod Pitch, inch	.641	.640
Number of Spacers	7	7
Assembly Weight, lbs	589	600
Assembly Length, inch	176.05	176.16
Fuel Rod Diameter, inch	.484	.483
Cladding Thickness, inch	.035	.032

APPENDIX B

SINGLE LOOP OPERATION
(SLO)

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 GE for single loop operation are also applicable to single loop operation with fuel and analyses provided by ENC.

With a single recirculation loop in operation, the GE analyses supported continued operation with an increase of 0.01 in the MCPR safety limit. Because of the similarity between the ENC XN-1 8x8 fuel type and the other fuel types making up the remainder of the core, and because of the similarity in the magnitude of the uncertainties which determine the MCPR safety limit, this small increase in the safety limit value is also appropriate for operation with ENC fuel and analyses. For Cycle 2 operation with both recirculation loops in operation, the MCPR safety limit is 1.06, which is the same value as was used for the previous cycle. For Cycle 2 operation with a

single recirculation loop in service, the MCPR safety limit is 1.07, which is also the same value as was used for the previous cycle.

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

The stability characteristics of the Cycle 2 core are equivalent to or better than those of the previous cycle core. Reactor operation within the monitoring which assured adequate stability using the detect and suppress approach for the previous cycle will continue to assure adequate stability for Cycle 2.

To support operation of WNP-2 with a core composed of GE P8x8R and ENC 8x8 fuel with a single recirculation pump operating, ENC recommends the conservative use of GE fuel MAPLHGR limits for the similar GE P8x8R fuel design with a multiplier of 0.84 applied for single loop operation. The basis for this recommendation is as follows:

The phenomena which require the reduction in MAPLHGR limits are a result of operation of the WNP-2 system with a single active recirculation loop, and are equally applicable to both GE and ENC fuel designs; and

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.

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WNP-2 CYCLE 2 RELOAD ANALYSIS

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