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ADVANCED NUCLEAR FUELS CORPORATION

WNP-2 CYCLE 5 RELOAD ANALYSIS

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1.0 INTRODUCTION

This report summarizes the results of the analyses performed by Advanced Nuclear Fuels Corporation (ANF) in support of the Cycle 5 reload for the Supply System Nuclear Project Number 2 (WNP-2). WNP-2 is scheduled to commence Cycle 5 operation in June 1989. This report is intended to be used in conjunction with ANF topical report XN-NF-80-19(A), 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(A), Volume 4, Revision 1.

Final feedwater temperature reduction (FFTR) analysis to support cycle extension was performed for WNP-2. This FFTR analysis is applicable for a condition with all the control rods out with normal feedwater temperature. That is, additional MCPR limit changes are applicable when reactor operation is being extended by reduction of the feedwater temperature.

The WNP-2 Cycle 5 core will comprise a total of 764 fuel assemblies: including 140 ANF 8x8 unirradiated assemblies; 2 ANF 9x9-IX unirradiated Lead Fuel Assemblies (LFA); 2 ANF 9x9-9X unirradiated LFA's; 152 once irradiated ANF 8x8 assemblies; 148 twice irradiated ANF 8x8 assemblies; 128 thrice irradiated ANF 8x8 assemblies; and 192 irradiated P8x8R assembiles from the Cycle 1 core fabricated by General Electric (GE). The ANF 9x9 Lead Fuel Assembly (LFA) licensing information is given in Appendix A. 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 WNP-2 during the previous operating cycle which encompass core flow up to 106% of the design basis value.

2.0 FUEL MECHANICAL DESIGN ANALYSIS

Applicable Fuel Design Report: Reference 9.8

The expected power history for the fuel to be irradiated during Cycle 5 of WNP-2 is bounded by the assumed power history in the fuel mechanical design analyses.

3.0 THERMAL HYDRAULIC DESIGN ANALYSIS

3.1 Design Criteria

3.1.3 Fuel Centerline Temperature

The LHGR curve in Figure 3.4 of Reference 9.8 shows that the ANF 8x8 fuel centerline temperature will be below the melting point at 120% over power. The LHGR curve in Reference 9.8 is greater than 120% above the LHGR limit curve in Reference 9.1. Therefore, fuel centerline melt is protected for all ANF 8x8 exposures within the bounds of the referenced LHGR curves.

3.2 Hydraulic Characterization

3.2.5 Bypass Flow

Calculated Bypass Flow Fraction (100% power/106% flow) 10.7%

3.3 MCPR Fuel Cladding Integrity Safety Limit

3.3.1 Coolant Thermodynamic Condition

Core Power	3950 MWt
Core Inlet Enthalpy	525.6 Btu/lbm
Steam Dome Pressure	1021 psia
Feedwater Temperature	414°F

3.3.2 Design Basis Radial Power Distribution

See Figure 3.1.

3.3.3 Design Basis Local Power Distribution

See Figures 3.2, 3.3, 3.4 and 3.5.

4.0 NUCLEAR DESIGN ANALYSIS

4.1 Fuel Bundle Nuclear Design Analysis

Assembly Average Enrichment	2.62 w/o U-235
Radial Enrichment Distribution	Figure 4.1
Axial Enrichment Distribution	Uniform 2.79 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

Note: The reload includes 4 ANF 8x8 assemblies of the 2.64 w/o U-235 design loaded in Cycle 4 and described in the Cycle 4 Reload Analysis Report ANF-88-02 and 4 9x9 LFA's of the 2.53/2.59 w/o U-235 design described in Appendix A.

4.2 Core Nuclear Design Analysis

4.2.1 Core Configuration

Core Configuration	Figure 4.2
Core Exposure at EOC4 (MWd/MTU)	16,700
Core Exposure at BOC5 (MWd/MTU)	12,300
Core Exposure at EOC5 (MWd/MTU)	18,100

4.2.2 Core Reactivity Characteristics

BOC Cold k-eff, All Rods Out	1.1133
BOC Cold k-eff, Strongest Rod Out	0.9868
Reactivity Defect (R-Value)	0.0
Standby Liquid Control System (SBLC) 660 ppm Boron, Cold k-eff	0.9633

4.2.4 Core Hydrodynamic Stability

%Power/%Flow State Points	Decay Ratio (COTRAN)
65/45*	0.49
47/27.6**	0.89
42/23.8***	0.82

---

*45 percent flow - APRM Rod Block intercept point.

**Two pump minimum flow - 46 percent power.

***Natural circulation flow - APRM Rod Block intercept point.

5.0 ANTICIPATED OPERATIONAL OCCURRENCES

Applicable Transient Analysis Report — Reference 9.3

5.1 Analysis Of Plant Transients At Increased Core Flow Conditions

References 9.3 and 9.11

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

Transient analyses for WNP-2 Cycle 2 anticipated operational events showed that delta CPR values at design basis conditions are bounded by delta CPR values at design basis power (104%) and increased core flow conditions (106%). Thus Cycle 5 analyses results at increased core flow conditions are conservatively applicable to rated flow conditions.

Cycle 5 specific analyses of transient events were performed for two recirculation pump operation conditions, with the recirculation pump trip (RPT) in service and out of service, and for two scram conditions which are normal scram speed (NSS) and technical specification scram speed (TSSS). Analyses were performed at end-of-cycle exposures which produced the results shown in following table. Generic analyses were performed for FFTR to extend cycle operation (Reference 9.11).

The loss of feedwater heating event was analyzed on a plant specific bounding value basis and the delta CPR results are bounding values for WNP-2.

Transient*	% Power/ % Flow	Maximum Heat Flux %	Maximum Power %	Maximum Pressure psig	Delta CPR GE Fuel	Delta CPR ANF Fuel
LRNB, NSS RPT Operable	104/106	121	403	1169	0.28	0.25
LRNB, NSS RPT Inoperable	104/106	127	501	1181	0.35	0.31
LRNB, TSSS RPT Operable	104/106	127	454	1174	0.35	0.31
LRNB, TSSS RPT Inoperable	104/106	132	594	1189	0.41	0.35
FWCF, NSS RPT Operable	47/106	54	163	1026	0.23	0.20
LOFH	N/A	N/A	N/A	N/A	0.09	0.09

5.2 <u>Analyses For Reduced Flow Operation</u> References 9.3 and 9.11

Limiting Transient: Recirculation Flow Increase

5.3 <u>Analysis For Reduced Power Operation (SLO)</u> References 9.12, 9.13, and 9.14

ANF has performed analyses for WNP-2 which demonstrate the safety of plant operation with a single recirculation loop out of service for an extended period of time. These analyses were performed for the most limiting transient events, the pump seizure accident and the loss-of-coolant-accident (LOCA) for the maximum extended power state during WNP-2 single-loop operation (SLO). The transient analysis and pump seizure accident analysis are documented in Reference 9.12, and the LOCA analysis is documented in Reference 9.13. The conclusions presented in these documents are applicable to future cycles with ANF fuel and have been reviewed by the U. S. Nuclear Regulatory Commission, Reference 9.14; the SLO limits from the USNRC review are summarized below.

*Normal scram speed (NSS) is based on measured plant scram insertion data, see Section 7.2.3.1.

SLO MCPR Operating Limit for ANF and GE fuel	1.35

Two-loop MAPLHGR limits which are shown in Section 6.1.3 for ANF fuel apply during SLO. For GE fuel the reduction of the MAPLHGR limit to a value of 0.84 times the two recirculation loop operation MAPLHGR limit for SLO remains unchanged.

5.4 <u>ASME Overpressurization Analysis</u> — References 9.3 and 9.11

Limiting Event	MSIV Closure
Worst Single Failure	MSIV Position Scram Trip
Maximum Pressure	1315 psig
Maximum Steam Dome Pressure	1286 psig

5.5 <u>Control Rod Withdrawal Error</u>

Initial Control Rod Pattern for CRWE Analysis	Figure 5.1

Rod Block Monitor Setting	Distance Withdrawn (ft)	ANF Fuel Delta-CPR	GE Fuel Delta-CPR
106%	5.5	0.17	0.17
107%*	6.0	0.19	0.19
108%	7.0	0.21	0.21

5.6 <u>Loading Error for Reload Fuels</u>

	With Loading Error	Correctly Loaded Core
Maximum LHGR, kW/ft	16.6	13.3
Minimum MCPR	1.26	1.44

5.7 <u>Determination Of Thermal Margins</u>

Summary of Thermal Margin Requirements

*Rod Block Monitor Setting (RBM) of 107%.

All system transient results were analyzed at the more limiting increased flow conditions (106%) rather than rated flow conditions (100%). LRNB results for the more limiting power (design basis condition - 104%) were used for this transient.

These calculated results are based on end of cycle conditions and increased core flow (106%).

Event	Equipment Operational Status	Delta CPR GE Fuel	Delta CPR ANF Fuel	MCPR Limit GE Fuel	MCPR Limit ANF Fuel	Model
LRNB	RPT Operable, NSS	0.28	0.25	1.34	1.31	COTRANSA/XCOBRA-T
LRNB	RPT Inoperable, NSS	0.35	0.31	1.41	1.37	" "
LRNB	RPT Operable, TSSS	0:35	0.31	1.41	1.37	" "
LRNB	RPT Inoperable, TSSS	0.41	0.35	1.47	1.41	" "
FWCF	RPT Operable, NSS	0.23	0.20	1.29	1.26	" "
LOFH	N/A	0.09	0.09	1.15	1.15	XTGBWR

Note: For cycle extension with reduced feedwater temperature, add 0.02 to delta CPR/MCPR LRNB and FWCF transient results in the above table.

MCPR Operating Limits At Rated Condition For Cycle Exposures Less Than EOC -2000 MWd/MTU are based on the CRWE (100% To 106% Flow)

Fuel Type	MCPR Limit (107% RBM)
ANF	1.25
GE	1.25

MCPR Operating Limits At Rated Condition From EOC -2000 MWd/MTU To EOC (100% To 106% Flow) With Normal Feedwater Temperature

Fuel Type	MCPR Limit
ANF	1.31
GE	1.34

MCPR Operating Limits At Rated Condition Beyond All Rods Out With Reduced Feedwater Temperature (100% To 106% Flow And Thermal Coastdown) Point (EOC5)

Fuel Type	MCPR Limit
ANF	1.33
GE	1.36

MCPR Limits at Off-Rated Conditions	Figures 5.2 and 5.3
Reduced Flow MCPR Limit	References 9.3 and 9.11

6.0 POSTULATED ACCIDENTS

6.1 Loss-Of-Coolant Accident

6.1.1 Break Location Spectrum

Reference 9.4

6.1.2 Break Size Spectrum

Reference 9.4

6.1.3 MAPLHGR Analyses (ANF Fuel - Two-Loop Operation and SLO)

References 9.5, 9.13 and 9.14

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	1779	0.50
5,000	13.0	1755	0.45
10,000	13.0	1761	0.46
15,000	13.0	1765	0.46
20,000	13.0	1771	0.51
25,000	11.3	1659	0.32
30,000	9.4	1513	0.16
35,000	7.9	1385	0.09

Heatup analysis shows insignificant changes in PCT's and local MWR, but no change in MAPLHGR limits, from the MAPLHGR analysis for the earlier ANF 8x8 fuel design which is shown in Reference 9.5.

6.2 Control Rod Drop Accident

Reference 9.7

Dropped Control Rod Worth, mK	8.1
Doppler Coefficient dk/kdT, 1/°F	-10.0×10^{-6}
Effective Delayed Neutron Fraction	0.0050
Four-Bundle Local Peaking Factor	1.17
Maximum Deposited Fuel Rod Enthalpy (cal/gm)	121

*For the ANF-4(6Gd2) fuel design PCT's and MWR's.

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	1346 psig

7.2 Limiting Conditions For Operation

7.2.1 Average Planar Linear Heat Generation Rate Limits For ANF 8x8 Fuel

Bundle Average Exposure (MWd/MTU)	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

For single-loop operation these limits also apply to ANF Fuel when using a SLO MCPR limit of at least 1.35.

7.2.2 Minimum Critical Power Ratio

Rated Condition MCPR Operating Limit Up To EOC -2000 MWd/MTU Exposure (100% To 106% Flow)

Fuel Type	Limit (107% RBM)
ANF	1.25
GE	1.25

Rated Conditions MCPR Operating Limits From EOC -2000 MWd/MTU To EOC (100% To 106% Flow)

Fuel Type	Limit
ANF	1.31
GE	1.34

Thermal Coastdown and FFTR Rated Condition MCPR Operating Limit Beyond All Rods Out Point With Reduced Feedwater Temperature (100% to 106% Flow)

Fuel Type	Limit
ANF	1.33
GE	1.36

Reduced Flow MCPR Limit (all cycle exposures) Figures 5.2 and 5.3

Single-Loop Operation (SLO) MCPR Limit (all cycle exposures)

Fuel Type	Limit
ANF	1.35
GE	1.35

7.2.3 Surveillance Requirements

7.2.3.1 Scram Insertion Time Surveillance

The ANF reload safety analyses were labeled NSS (Normal Scram Speed) 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 (TSSS) control rod scram times (see Section 5.7).

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

7.2.3.2 Stability Surveillance

Core hydrodynamic stability analyses require slight modification to the Technical Specifications which preclude operation in specified power/flow regions. The results of these analyses support operation below a line defined by the following power/flow[1] points: 42% Power/23.8% Flow, 47% Power/27.6% Flow, and 65% Power/45% Flow (see Section 4.2.4).

Surveillance requirements remain unchanged for Cycle 5, e.g., surveillance is required when operating in a power flow region above the 80% rod line and less than 45% core flow.

7.2.3.3 Technical Specification LHGR Surveillance

The Technical Specification linear heat generation rate (LHGR) limit versus average planar exposure for ANF 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.

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

TABLE 4.1 NEUTRONIC DESIGN VALUES

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

TABLE 4.1 NEUTRONIC DESIGN VALUES (Continued)

<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	008.0
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	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

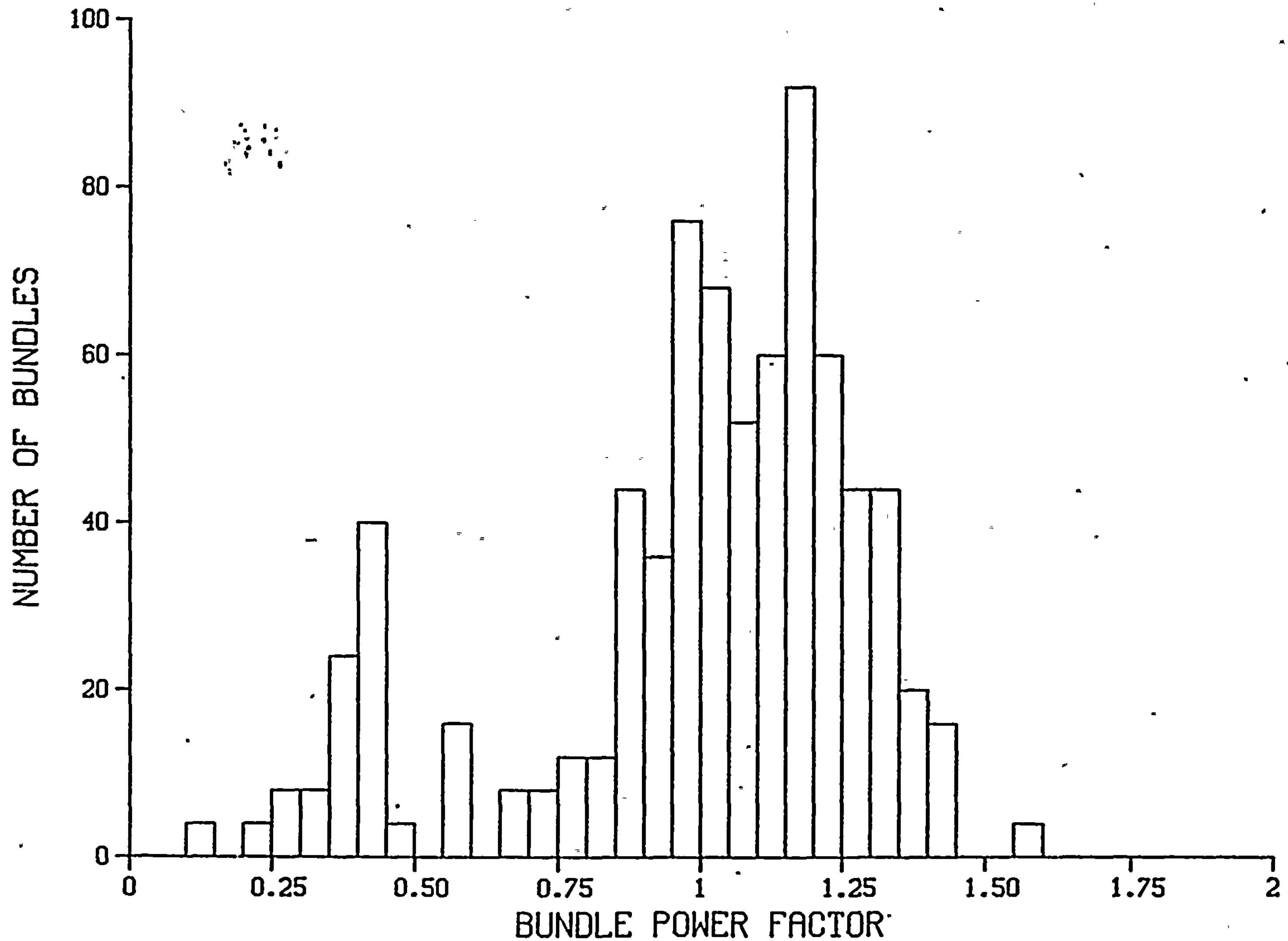


FIGURE 3.1 RADIAL POWER HISTOGRAM FOR 1/4 CORE SAFETY LIMIT MODEL

.936	.977	1.023	1.015	1.011	1.041	1.076	1.052
.977	1.011	.907	1.042	1.035	.932	.962	1.075
1.023	.907	1.017	.988	.974	.996	.931	1.040
1.015	1.042	.988	.000	.850	.972	1.033	1.009
1.011	1.035	.974	.850	.000	.985	1.038	1.011
1.041	.932	.996	.972	.985	1.012	.901	1.043
1.076	.962	.931	1.033	1.038	.901	.976	1.078
1.052	1.075	1.040	1.009	1.011	1.043	1.078	1.054

FIGURE 3.2 WNP-2 CYCLE 5 SAFETY LIMIT LOCAL PEAKING FACTORS (ANF-4 FUEL)

.944	.962	1.011	1.044	1.043	1.010	.960	.943
.962	.980	1.064	.894	1.033	1.059	1.034	.961
1.011	1.064	1.010	.994	.982	1.002	.915	1.010
1.044	.894	.994	.000	.907	.980	1.032	1.042
1.043	1.033	.982	.907	.000	.988	.952	1.041
1.010	1.059	1.002	.980	.988	1.004	1.060	1.065
.960	1.034	.915	1.032	.952	1.060	.966	1.053
.943	.961	1.010	1.042	1.041	1.065	1.053	1.019

FIGURE 3.3 WNP-2 CYCLE 5 SAFETY LIMIT LOCAL PEAKING FACTORS
(ANF XN-3 FUEL)

.950	.963	1.000	1.027	1.026	.999	.963	.950
.963	.981	1.052	.920	1.033	1.049	1.020	.963
1.000	1.052	1.017	1.005	.997	1.011	.936	1.000
1.027	.920	1.005	.000	.935	.996	1.033	1.027
1.026	1.033	.997	.935	.000	1.002	.971	1.027
.999	1.049	1.011	.996	1.002	1.016	1.054	1.042
.963	1.020	.936	1.033	.971	1.054	.973	1.029
.950	.963	1.000	1.027	1.027	1.042	1.029	1.003

FIGURE 3.4 WNP-2 CYCLE 5 SAFETY LIMIT LOCAL PEAKING FACTORS
(ANF XN-2 FUEL)

.967	.969	.997	1.019	1.019	.996	.968	.966
.969	.981	1.044	.932	1.030	1.042	1.013	.968
.997	1.044	1.017	1.008	1.001	1.012	.944	.997
1.019	.932	1.008	.000	.947	1.000	1.030	1.019
1.019	1.030	1.001	.947	.000	1.006	.976	1.020
.996	1.042	1.012	1.000	1.006	1.017	1.047	1.032
.968	1.013	.944	1.030	.976	1.047	.975	1.020
.966	.968	.997	1.019	1.020	1.032	1.020	1.003

FIGURE 3.5 WNP-2 CYCLE 5 SAFETY LIMIT LOCAL PEAKING FACTORS
(ANF XN-1 FUEL)

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

LL RODS (1) --- 1.50 W/O U235
L RODS (5) --- 2.00 W/O U235
ML RODS (9) --- 2.50 W/O U235
M RODS (21) --- 2.64 W/O U235
H RODS (20) --- 3.43 W/O U235
ML* RODS (6) --- 2.50 W/O U235 + 2.00 W/O GD203
W RODS (2) --- INERT WATER ROD

FIGURE 4.1 WNP-2 CYCLE 5 ENRICHED ZONE ENRICHMENT DISTRIBUTION

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	D	C	F	C	D	C	F	C	D	C	D	C	D	F	B
2	C	H	B	H	D	H	C	H	D	H	B	H	D	H	B
3	F	B	E	B	F	B	F	B	F	D	F	D	D	F	A
4	C	H	B	F	C	H	B	H	C	H	B	H	B	H	A
5	D	D	F	C	G	C	F	B	F	D	F	D	D	F	B
6	C	H	B	H	C	E	C	H	C	H	B	H	C	D	B
7	F	C	F	B	F	C	H	B	E	D	F	D	F	D	A
8	C	H	B	H	B	H	B	F	C	H	B	H	B	A	
9	D	D	F	C	F	C	E	C	I	D	E	F	A		
10	C	H	D	H	D	H	D	H	D	C	C	A	A		
11	D	B	F	B	F	B	F	B	E	C	C				
12	C	H	D	H	D	H	D	H	F	A					
13	D	D	D	B	D	C	F	B	A	A					
14	F	H	F	H	F	D	D	A							
15	B	B	A	A	B	B	A								

Fuel Type Type	Number of Assemblies	Description
A	56	GE 8x8 Type II 1.76 w/o U-235 (Cycle 1)
B	136	GE 8x8 Type III 2.19 w/o U-235 (Cycle 1)
C	128	ANF 8x8 2.72 w/o U-235 (Cycle 2)
D	148	ANF 8x8 2.72 w/o U-235 (Cycle 3)
E	24	ANF 8x8 2.72 w/o U-235 (Cycle 4)
F	128	ANF 8x8 2.64 w/o U-235 (Cycle 4)
G	4	ANF 8x8 2.64 w/o U-235 (Cycle 5)
H	136	ANF 8x8 2.62 w/o U-235 (Cycle 5)
I	4	ANF 9x9 Lead 2.53/2.59 w/o U-235 (Cycle 5)

FIGURE 4.2 WNP-2 CYCLE 5 REFERENCE LOADING PATTERN BY FUEL TYPE (ONE QUARTER OF SYMMETRICAL CORE LOADING)

	2	6	10	14	18	22	26	30	34	38	42	46	50	54	58	
59					--	--	--	--	--	--	--					59
55				--	--	00	--	40	--	00	--	--				55
51			--	--	--	--	--	--	--	--	--	--	--			51
47		--	--	36	--	16	--	00	--	16	--	36	--	--		47
43	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	43
39	--	00	--	16	--	00	--	36	--	00	--	16	--	00	--	39
35	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	35
31	--	40	--	00	--	36	--	00*	--	36	--	00	--	40	--	31
27	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	27
23	--	00	--	16	--	00	--	36	--	00	--	16	--	00	--	23
19	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--	19
15		--	--	36	--	16	--	00	--	16	--	36	--	--		15
11			--	--	--	--	--	--	--	--	--	--	--			11
7				--	--	00	--	40	--	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 WNP-2 CYCLE 5 CONTROL ROD WITHDRAWAL ANALYSIS
INITIAL CONTROL ROD PATTERN

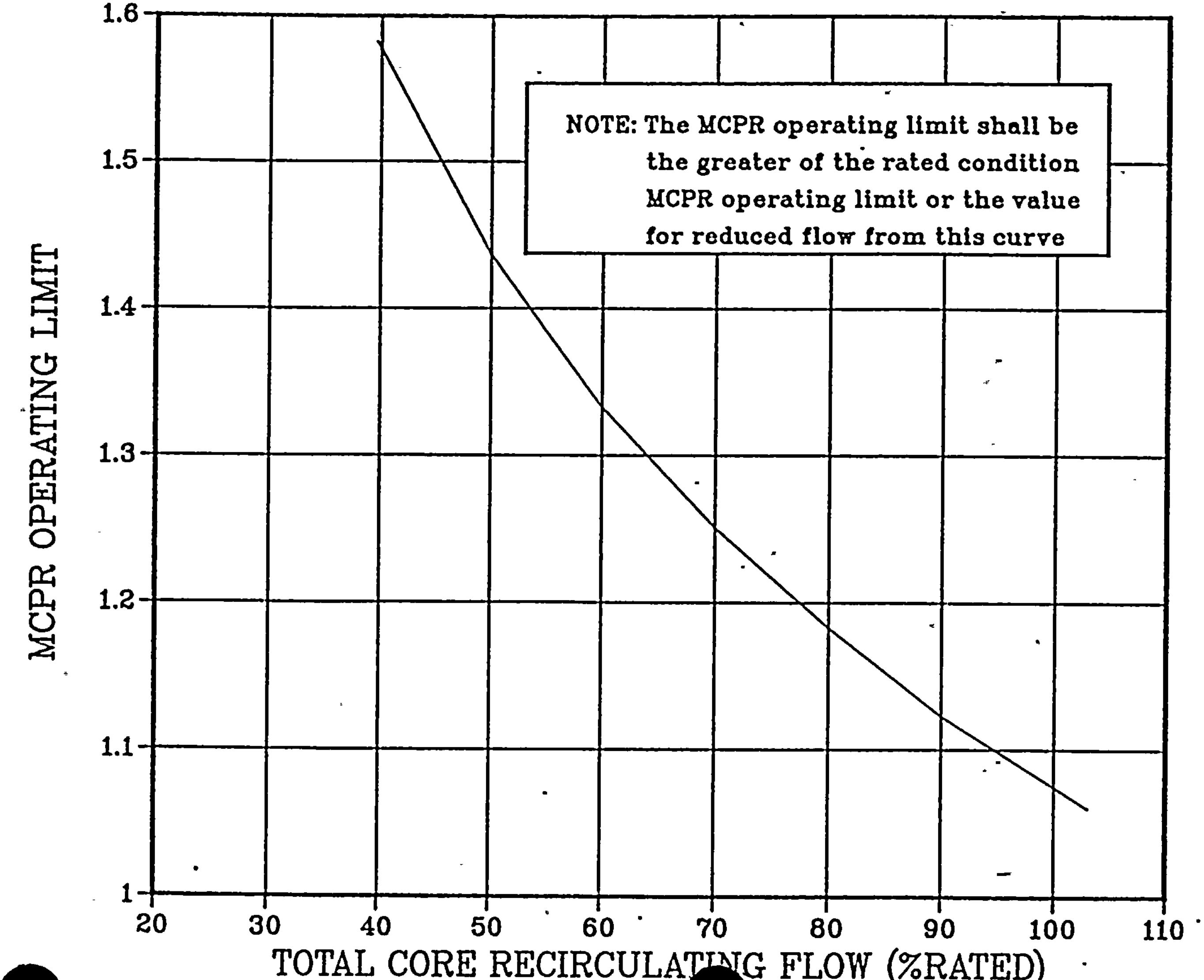


FIGURE 5.2 REDUCED FLOW MCPR OPERATING LIMIT FOR NORMAL FEEDWATER TEMPERATURE

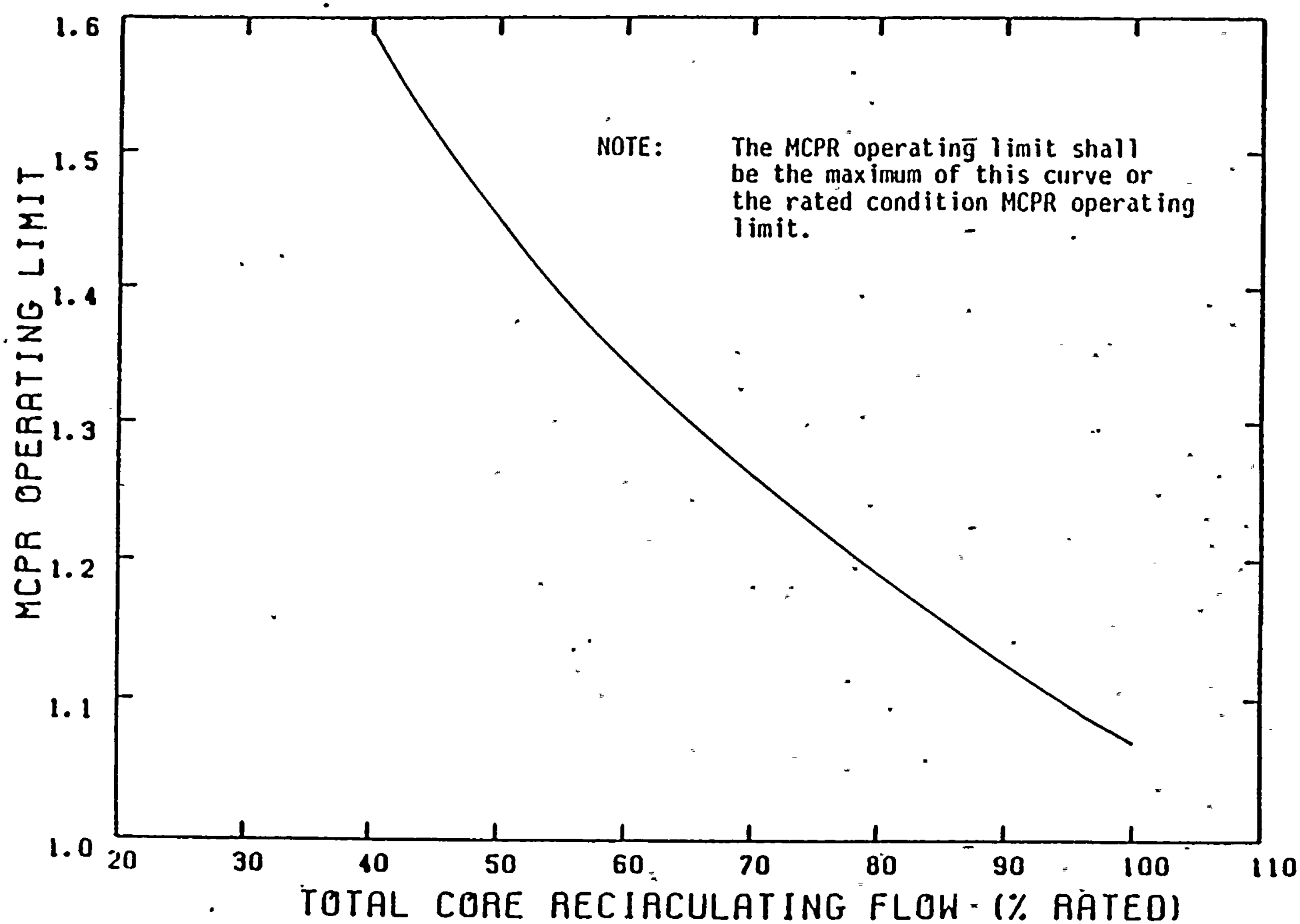
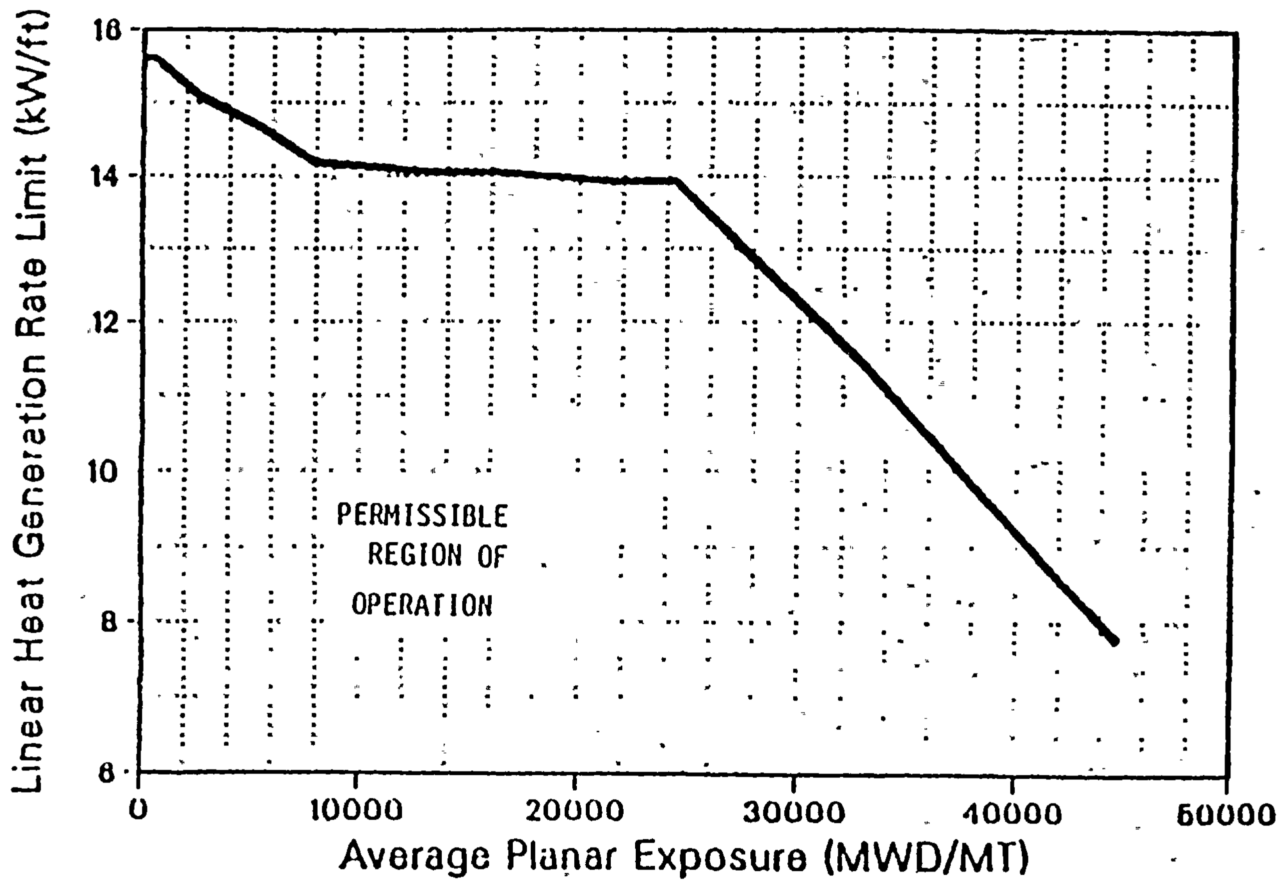


FIGURE 5.3 REDUCED FLOW MCPR OPERATING LIMIT FOR FFTR OPERATION



EXP	LHGR
0	15.62
510	15.62
2,680	15.10
5,230	14.71
7,840	14.18
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
33,060	11.40
35,960	10.47
38,900	9.56
41,830	8.66
44,760	7.77

FIGURE 7.1 LINEAR HEAT GENERATION RATE (LHGR) LIMIT VERSUS AVERAGE PLANAR EXPOSURE, ANF 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 Company, Inc., Richland, WA 99352, November 1981.

9.3 J. E. Krajicek, "WNP-2 Cycle 5 Plant Transient Analysis," ANF-89-01, Advanced Nuclear Fuels Corporation, Richland, WA 99352, January 1989.

9.4 J. E. Krajicek, "LOCA Break Spectrum for a BWR 5," XN-NF-85-138(P), Exxon Nuclear Company, Inc., Richland, WA 99352, December 1985.

9.5 D. J. Braun, "WNP-2 LOCA-ECCS Analysis, MAPLHGR Results," XN-NF-85-139, Exxon Nuclear Company, Inc., Richland, WA 99352, December 1984.

9.6 M. H. Smith, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," XN-NF-81-21(P), Revision 1, Supplement 1, Exxon Nuclear Company, Inc., Richland, WA 99352, March 1985.

9.7 "Exxon Nuclear Methodology for Boiling Water Reactors-Neutronics Methods for Design and Analysis," XN-NF-80-19(A), Volume 1 and Supplements, Exxon Nuclear Company, Inc., Richland, WA 99352, May 1980.

9.8 "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," XN-NF-85-67(A), Revision 1, Exxon Nuclear Company, Inc., Richland, WA 99352, September 1986.

9.9 "Exxon Nuclear Methodology for Boiling Water Reactors Neutronics Methods for Design Analysis," XN-NF-80-19(A), Volume 1, Supplements 1 and 2, Exxon Nuclear Company, Inc., Richland, WA 99352, March 1983.

9.10 J. B. Edgar, Letter to WPPSS, Supplemental Licensing Analysis Results, ENWP-86-0067, Exxon Nuclear Company, Inc., Richland, WA 99352, April 15, 1986.

9.11 J. E. Krajicek, "WNP-2 Plant Transient Analysis With Final Feedwater Temperature Reduction," XN-NF-87-92 and XN-NF-87-92, Supplement 1, Advanced Nuclear Fuels Corporation, Richland, WA 99352, June 1987 and May 1988.

9.12 J. E. Krajicek, "WNP-2 Single Loop Operation Analysis," ANF-87-119, Advanced Nuclear Fuels Corporation, Richland, WA 99352, September 1987.

9.13 J. E. Krajicek and T. Tahvili, "WNP-2 LOCA Analysis For Single Loop Operation," ANF-87-118, Advanced Nuclear Fuels Corporation, Richland, WA 99352, September 1987.

9.14 Letter, R. B. Samworth, USNRC, to G. C. Sorensen, WPPSS, Subject: Issuance Of Amendment No. 62 To Facility Operating License No. NPF-21-WPPSS Nuclear Project 2 (TAC No. 67538), August 5, 1988.

APPENDIX A

9X9-IX AND 9X9-9X LEAD FUEL ASSEMBLIES (LFA'S)

A.1 INTRODUCTION

Evaluations have been performed consistent with ANF methodology ("Exxon Nuclear Methodology for Boiling Water Reactors", XN-NF-80-19) to establish a licensing basis for two ANF 9x9-IX and two ANF 9x9-9X Lead Fuel Assemblies (LFA) in the WNP-2 Cycle 5 core. Justification is provided which demonstrates the applicability of the WNP-2 Cycle 5 operating limits to these four LFA's unless stated otherwise.

The insertion of only four ANF 9x9 LFA's in the Cycle 5 core will have negligible effects upon core wide transient performance. However, some 9x9 LFA specific analyses have been performed to assure that the Cycle 5 operating limits are also applicable to the LFA's. Fuel specific LHGR and MAPLHGR limits have been developed for these LFA's and are presented in this appendix.

A.2 FUEL MECHANICAL DESIGN

A mechanical design analysis showing that the 9x9-IX and 9x9-9X fuel meet approved criteria will be documented in ANF-89-014(P).

The dynamic response of the LFA's is expected to be almost identical to that of the 8x8 already in the core. This is due to the fact that the fuel assembly stiffness is provided by the assembly channel, which is the same in both designs. The mass of the LFA's is very close to that of the 8x8's. It thus follows that the dynamic response should be the same.

A.3 THERMAL HYDRAULIC DESIGN

The 9x9 LFA's are hydraulically compatible with the co-resident ANF 8x8 fuel assemblies based on a comparison of fuel component hydraulic resistances.

Steady state thermal hydraulic analysis has shown that even though the ANF 9x9 LFA design has a somewhat smaller flow area than the ANF 8x8 design, no reduction in thermal margin is experienced in the Cycle 5 core. This is due to the increased critical power performance of the ANF 9x9 LFA design relative to the ANF 8x8 design at WNP-2 Cycle 5 conditions.

A.4 NUCLEAR DESIGN

The average enrichment and enrichment distribution for the 9x9-IX and 9x9-9X fuel assemblies have been selected to match, as closely as possible, the neutronic performance of the four 8x8 XN-3 2.64 w/o U-235 reload assemblies included in the Cycle 5 reload. The fuel assembly average enrichment, including six-inch top and bottom natural uranium blankets, is 2.53 w/o U-235 for the 9x9-IX design and 2.59 w/o U-235 for the 9x9-9X design. The average enrichment of the 138 inch central portion of the fuel assembly is 2.69 w/o U-235 for the 9x9-IX and 2.75 w/o U-235 for the 9x9-9X. Each 9x9 assembly contains six fuel rods containing Gd_2O_3 blended with 2.51 w/o U-235. The 9x9 fuel assembly contains 72 fueled rods and one central water channel displacing nine rod positions. The key neutronic design parameters for the ANF 9x9 LFA designs are presented in Table A.1 along with the corresponding values for the ANF XN-3 8x8 reload fuel design.

The nuclear characteristics of the 9x9 LFA's are similar to the characteristics of the ANF 8x8 fuel. The effect of replacing four ANF 8x8 assemblies with the four ANF 9x9 LFA's on the Cycle 5 core neutronics is negligible. The maximum cold uncontrolled non-voided k_∞ of the 9x9 fuel is 1.215 compared to the maximum k_∞ of 1.229 for the XN-3 8x8 fuel; thus the 9x9 fuel is compatible with the 8x8 fuel for fuel storage.

The LFA's were included in the core-wide stability analysis reported in Section 4.2.4. Local instability tests were performed on 9x9 leads in a BWR-3; no detectable difference was noted in stability performance relative to the co-resident 8x8 fuel.

A.5 ANTICIPATED OPERATIONAL OCCURRENCES

Analyses of the WNP-2 Cycle 5 limiting transients have been performed for ANF 8x8, ANF 9x9 LFA's, and GE P8x8R fuel. It has been shown that using the XN-3 ANF CHF correlation, the bundle power required to produce transition boiling in an ANF 9x9 LFA is higher than that for an ANF 8x8 bundle. That is, when an ANF 9x9 LFA bundle is modeled as an 8x8 bundle with equivalent conditions, there is margin to the MCPR safety limit during all AOO's. The Cycle 5 Safety Limit Analysis considered the LFA's such that the MCPR safety limit of 1.06 is also applicable to the LFA's. Therefore, the ANF 9x9 LFA's can be monitored to the ANF 8x8 fuel limits.

A.6 POSTULATED ACCIDENTS

Since heatup is primarily a planar and not an axial phenomena, the appropriate bundle power limit is derived from a LOCA analysis is the peak bundle planar power. The ANF 9x9 LFA's have better cooling during LOCA conditions relative to an ANF 8x8 fuel assembly due to the lower stored energy in the fuel rods, a greater surface area provided by the larger number of fuel rods, and more inert surface from the central water channel. Thus, a LOCA analysis for the ANF 9x9 LFA's would yield lower Peak Cladding Temperatures (PCT's) and metal-water reactions than an ANF 8x8 assembly at the same bundle peak planar power. The MAPLHGR limits for the ANF 9x9 LFA's restrict the peak bundle planar power to that analyzed for the ANF 8x8 fuel and assure that the USNRC criteria are met for the ANF 9x9 LFA's in Cycle 5.

The fuel loading error was analyzed for the ANF 9x9 LFA's. Results show that if the loading error went undetected, the offsite consequences would remain well within the guidelines specified in 10 CFR Part 100.

A.7 TECHNICAL SPECIFICATIONS

All operational limits used for ANF 8x8 fuel are applicable to the ANF 9x9 LFA's except for fuel type specific MAPLHGR limits and the 9x9-IX and 9x9-9X LHGR limits. The LHGR limits for the 9x9-IX and 9x9-9X LFA's are shown in Figures A.4 and A.5 respectively, and the MAPLHGR limits for the LFA's are shown in Figure A.6. The numerical values of Figure A.6 are 0.861 (62/72)

times the MAPLHGR values of Section 7.2.1. The LFA single-loop operation (SLO) limits are bounded by the two-loop operation limits.

TABLE A.1 ANF 9X9-IX AND 9X9-9X LEAD FUEL ASSEMBLY NEUTRONIC DESIGN VALUES

Fuel Pellets	Reload XN-3 8x8	9x9-IX Except Gd Rods	9x9-9X and IX Gd Rods
Fuel Material	UO_2 Sintered Pellets	UO_2 Sintered Pellets	UO_2 Sintered Pellets
Density			
g/cc	10.36	10.55	10.36
% of TD	94.5	96.26	94.5
Diameter, inch			
Enriched Fuel	0.4055	0.3740	0.3665
Natural Fuel	0.4045	0.3740	0.3665
Fuel Rods			
Fuel Length, inch	150	150	150
Cladding Material	Zircaloy-2	Zircaloy-2	Zircaloy-2
Cladding Liner Material	N/A	Zirconium	N/A
Clad I.D., inch	0.414	0.3807	0.373
Clad O.D., inch	0.484	0.431	0.431

TABLE A.1 ANF 9X9-IX AND 9X9-9X LEAD FUEL ASSEMBLY NEUTRONIC DESIGN VALUES (CONTINUED)

Water Rods	Reload XN-3 8x8	9x9-IX Except Gd Rods	9x9-9X and IX Gd Rods
Number	2	N/A	N/A
Cladding I.D., inch	0.414	N/A	N/A
Cladding O.D., inch	0.484	N/A	N/A
Central Water Channel			
Outside Width, inch	N/A	1.65	1.65
Thickness, inch	N/A	0.0285	0.0285
Fuel Assembly Data			
Number of Fuel Rods	62	72	72
Fuel Rod Enrichment	Figure A.1	Figure A.2	Figure A.3
Fuel Rod Pitch, inch	0.641	0.569	0.569
Fuel Assembly Loading, KgU	176.0	176.8	167.7

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) --- 1.94 W/O U235
ML RODS (9) --- 2.50 W/O U235
M RODS (16) --- 2.86 W/O U235
H RODS (22) --- 3.43 W/O U235
ML* RODS (5) --- 2.50 W/O U235 + 2.00 W/O GD203
W RODS (2) --- INERT WATER ROD

FIGURE A.1 XN-3 8X8 ENRICHED ZONE ENRICHMENT DISTRIBUTION

L	M	H	H	H	H	H	M	L
M	H	H	M*1	H	H	H	H	M
H	H	M*1	H	M	H	M*1	H	H
H	M*1	H	W	W	W	H	H	H
H	H	M	W	W	W	M	H	H
H	H	H	W	W	W	H	H	H
H	H	M*1	H	M	H	M*2	H	H
M	H	H	H	H	H	H	H	M
L	M	H	H	H	H	H	M	L

L RODS (4) --- 1.92 W/O U235
M RODS (12) --- 2.51 W/O U235
H RODS (50) --- 2.82 W/O U235
M*1 RODS (5) --- 2.51 W/O U235 + 1.80 W/O GD203
M*2 RODS (1) --- 2.51 W/O U235 + 4.50 W/O GD203
W RODS (9) --- INERT WATER ROD

FIGURE A.2 9X9-IX ENRICHED ZONE ENRICHMENT DISTRIBUTION

L	M	H	H	H	H	H	M	L
M	H	H	M*1	H	H	H	H	M
H	H	M*1	H	M	H	M*1	H	H
H	M*1	H	W	W	W	H	H	H
H	H	M	W	W	W	M	H	H
H	H	H	W	W	W	H	H	H
H	H	M*1	H	M	H	M*2	H	H
M	H	H	H	H	H	H	H	M
L	M	H	H	H	H	H	M	L

L RODS (4) --- 1.92 W/O U235
M RODS (12) --- 2.51 W/O U235
H RODS (50) --- 2.90 W/O U235
M*1 RODS (5) --- 2.51 W/O U235 + 1.80 W/O GD203
M*2 RODS (1) --- 2.51 W/O U235 + 4.50 W/O GD203
W RODS (9) --- INERT WATER ROD

FIGURE A.3 9X9-9X ENRICHED ZONE ENRICHMENT DISTRIBUTION

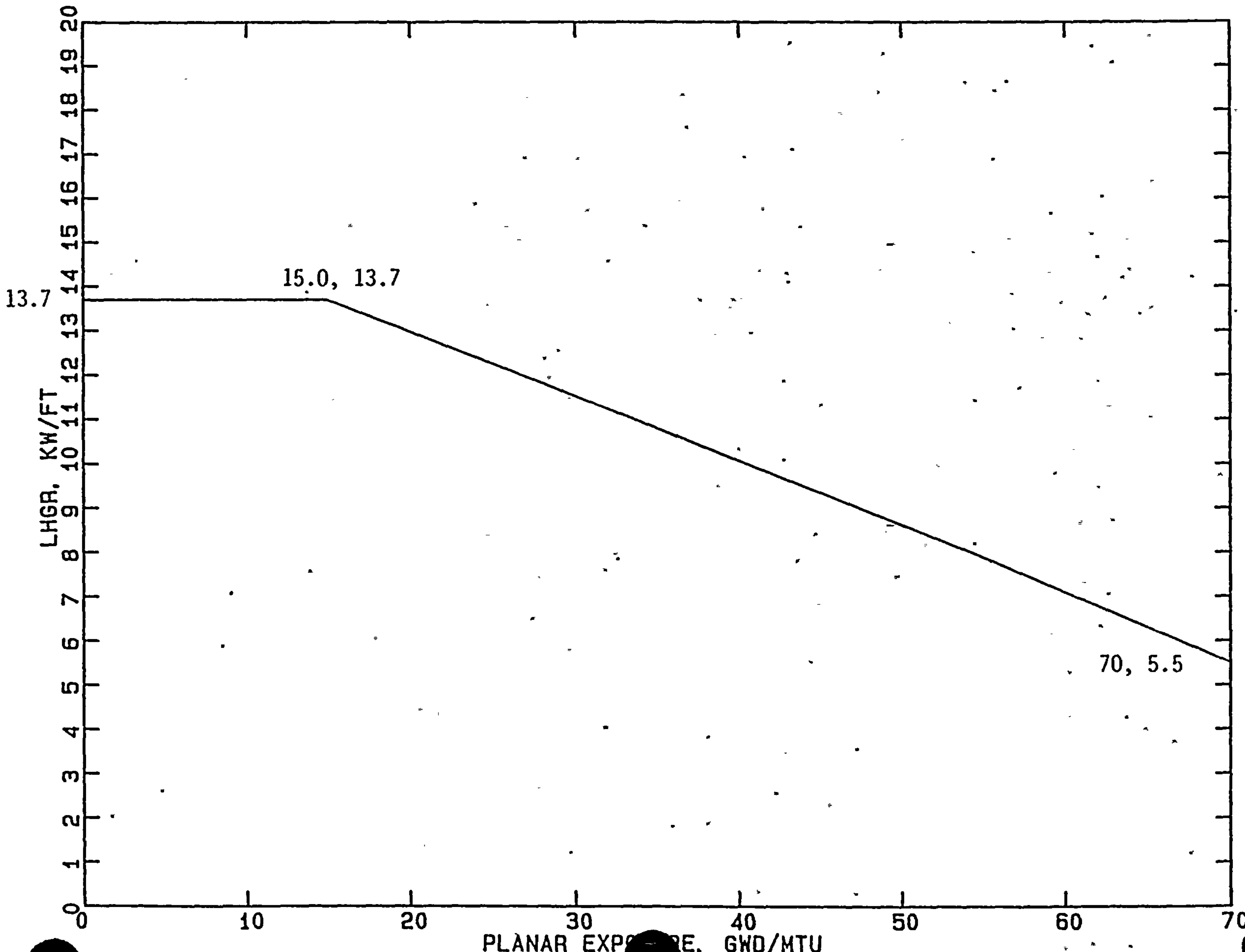

FIGURE A.4 LHGR LIMIT FOR 9X9-IX FUEL

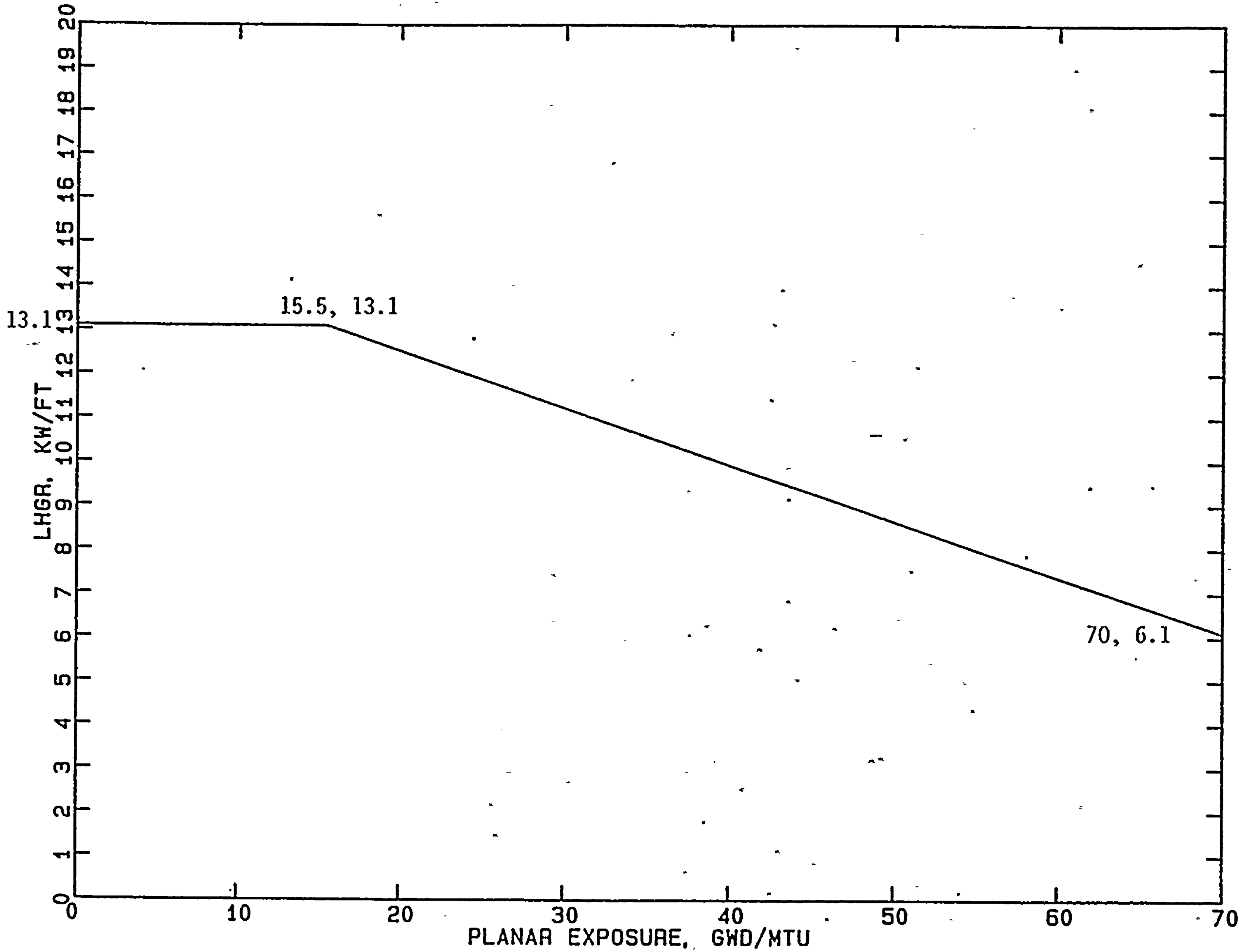

FIGURE A.5 LHGR LIMIT FOR 9X9-9X FUEL

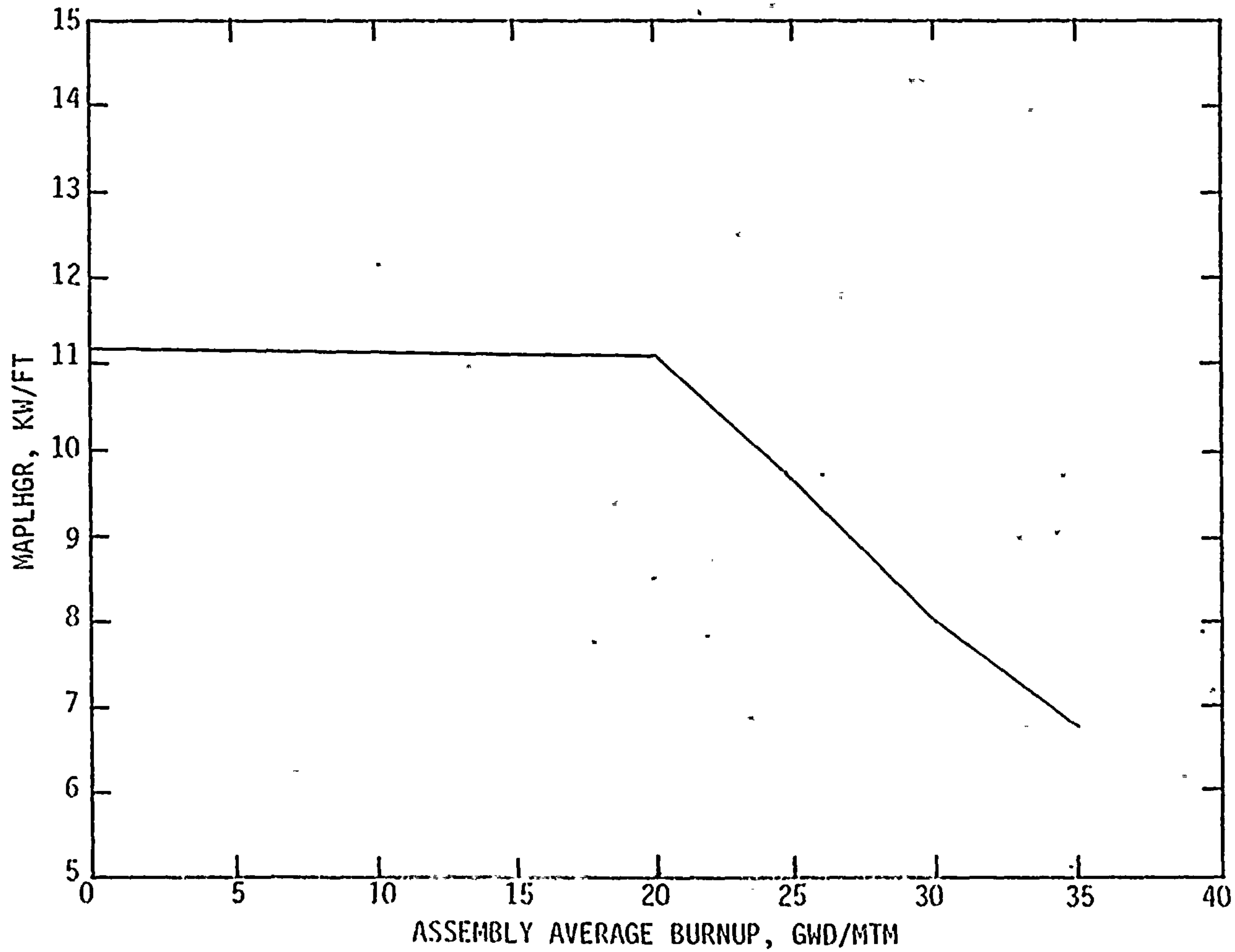

FIGURE A.6 ANF 9X9-IX AND 9X9-9X MAPLHGR LIMITS

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WNP-2 CYCLE 5 RELOAD ANALYSIS

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