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Grand Gulf Unit 1 Cycle 6 Reload Analysis

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GRAND GULF UNIT 1 CYCLE 6 RELOAD ANALYSIS

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1.0 INTRODUCTION

This report provides the results of the analyses performed by Siemens Nuclear Power Corporation (SNP) in support of the Cycle 6 reload for Grand Gulf Unit 1. This report is intended to be used in conjunction with SNP te[illegible]cal report XN-NF-80-19(A), Volume 4, Revision 1, "Ap[illegible]ation of the ENC Methodology to [illegible] Reloads," which describes the analyses performed in support of this reload, identifies [illegible] used for those analyses, and provides a generic reference list. Section num[illegible]t are [illegible]e same as corresponding section numbers in XN-NF-80-19(A), Volume 4 [illegible]ology used in this report which supersedes XN-NF-80-19(A), Volume 4, [illegible]ed as appropriate.

The NSSS vendor performed extensive safety a[illegible]yses for Grand Gulf Unit 1 in conjunction with the extension of the power/flow operating map to the Maximum Extended Operating Domain (MEOD) in Cycle 1 (Reference 1). These analyses established appropriate operating limits for MEOD operation. The initial reload of SNP fuel in Grand Gulf Unit 1 occurred in Cycle 2. In support of the initial reload of SNP fuel, extensive additional safety analyses were performed by SNP to either justify the NSSS vendor operating limits or, where necessary, to provide appropriate limits for SNP fuel using SNP methodologies (Reference 2). Subsequent SNP analyses supported an additional reload of SNP fuel in Cycle 3 (Reference 9), Cycle 4 (Reference 12), and Cycle 5 (Reference 15).

Changes from Cycle 5 to Cycle 6 for Grand Gulf Unit 1 include an additional reload of SNP fuel resulting in a core comprised of twice burned SNP 8x8 designs and four SNP 9x9-5 LTAs, once burned SNP 9x9-5 fuel, and fresh SNP 9x9-5 fuel. The 9x9-5 reload fuel is mechanically, neutronically, and thermal hydraulically compatible with the co-resident 8x8 and 9x9-5 fuel inserted in previous cycles. The cycle length remains 18 months and the nominal cycle energy is 1748 GWd. A reload batch design composed of 272 assemblies with axial enriched zoning and up to 3.38 w/o U235 assembly average enrichment containing axially varying Gd_2O_3 is used to meet the cycle energy requirements. A portion of each assembly contains from eight to ten Gd_2O_3 rods. The balance of the core is composed of 240 twice burned SNP 8x8 reload fuel assemblies, 4 twice burned 9x9-5 lead fuel assemblies, and 284 once burned SNP 9x9-5 reload fuel assemblies.

The design and safety analyses reported in this document were based on design and operational assumptions in effect for Grand Gulf Unit 1 during Cycle 5 operation and conditions bounding Cycle 6 operation. The $MCPR_p$ and $MCPR_f$ limits have been revised to reflect SNP calculated limits. Provision has been made in the flow dependent MCPRs for "loop manual" operation (Reference 11). Analyses were performed at EOC-30 EFPD, at EOC, and at EOC+30 EFPD providing limits for Cycle 6 that are cycle exposure dependent. The analyses also included support of the power/flow operation map for MEOD as shown in Figure 1.1. MCPR values were determined using the ANFB Critical Power Correlation (Reference 8.9). Monitoring to the plant thermal limits presented in this report will be performed using SNP's core monitoring methodology, POWERPLEX® CMSS, in accordance with SNP's thermal limits methodology, THERMEX (Reference 8.6).

SNP evaluated the LOCA-seismic response and operation with feedwater heaters out of service for Cycle 2 and subsequent cycles. These evaluations remain applicable for Cycle 6. The Cycle 6 SLO analyses are performed using SNP methodology (References 5 and 8.1 through 8.18). The Cycle 6 results supersede the previous cycle's results.

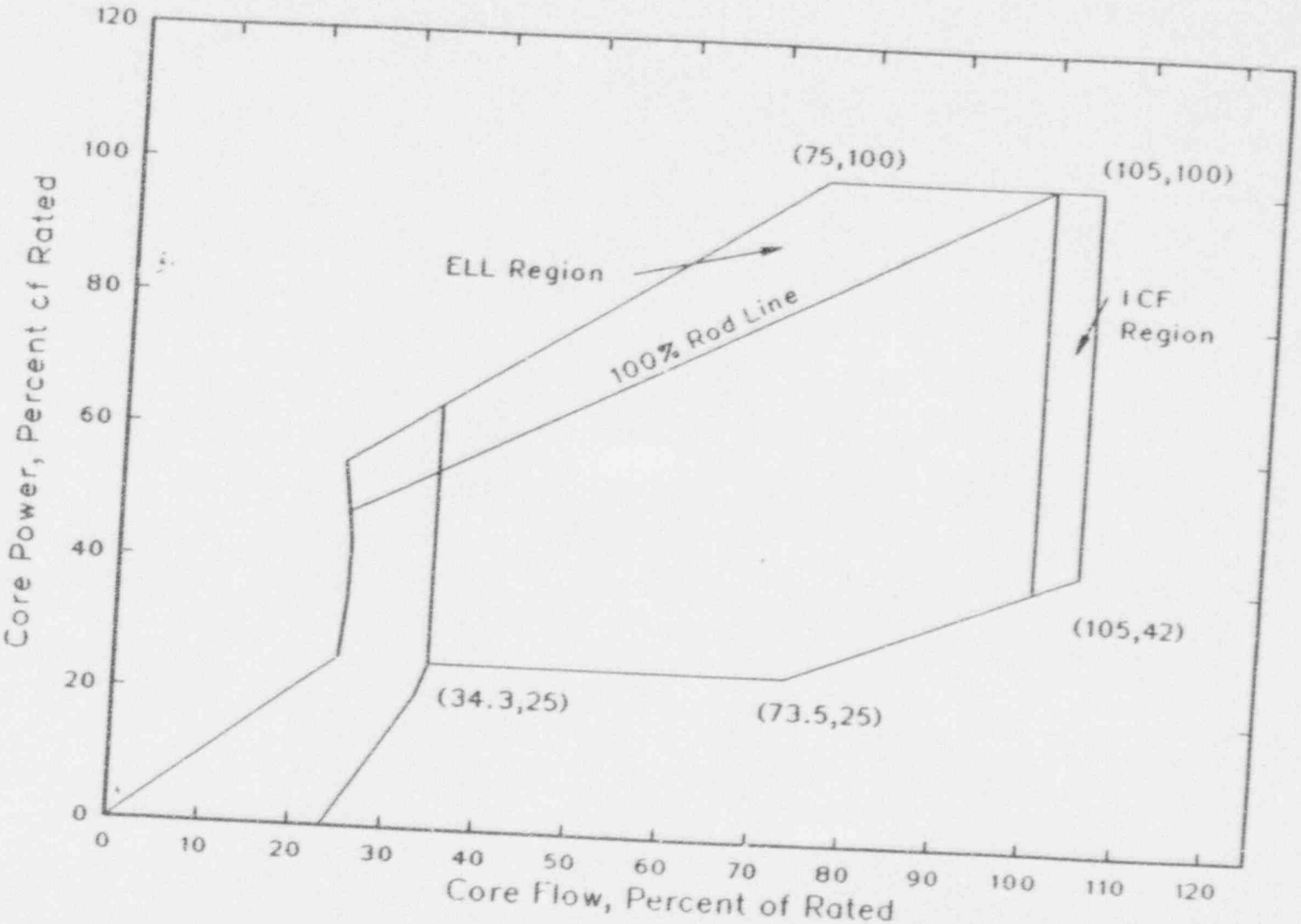

FIGURE 1.1 POWER/FLOW MAP USED FOR GRAND GULF UNIT 1 MEOD ANALYSIS

2.0 FUEL MECHANICAL DESIGN ANALYSIS

Applicable Fuel Design Report: References 3, 10, and 13

Qualification analyses provided in the references are applicable to the Grand Gulf Unit 1 SNP fuel assemblies. Minor mechanical design changes are discussed in Reference 14.

The expected power history for the fuel to be irradiated during Cycle 6 is bounded by the design LHGR of Figure 4.1 of Reference 16 and Figure 3.1 of Reference 13.

Seismic/LOCA analysis results for Cycle 5 reported in Appendix A of Reference 15 remain valid for Cycle 6.

3.0 THERMAL HYDRAULIC DESIGN ANALYSIS

3.2 Hydraulic Characterization

3.2.3 Fuel Centerline Temperature

Fuel Centerline Melting is protected by the transient LHGR limit given in References 13 and 16.

3.2.5 Bypass Flow

Calculated Bypass Flow (Exclusive of Water Rod Flow at 104.2%P/108%F)	10.6%

3.3 MCPR Fuel Cladding Integrity Safety Limit

See Reference 4	1.06* 1.07**

3.3.1 Nominal Coolant Condition in Safety Limit Monte Carlo Analysis

Core Power	5074 MWt
Core Inlet Enthalpy	520.5 Btu/lbm
Reference Pressure	1050 psia
Feedwater Temperature	420°F
Feedwater Flow Rate	21.8 Mlbm/hr

3.3.2 Design Basis Radial Power Distribution

See Figure 3.1

3.3.3 Design Basis Local Power Distribution

See Figure 3.2

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* The 1.06 includes effects for channel bow.

** For single loop operation the safety limit MCPR increases to 1.07 due to increased uncertainties associated with SLO.

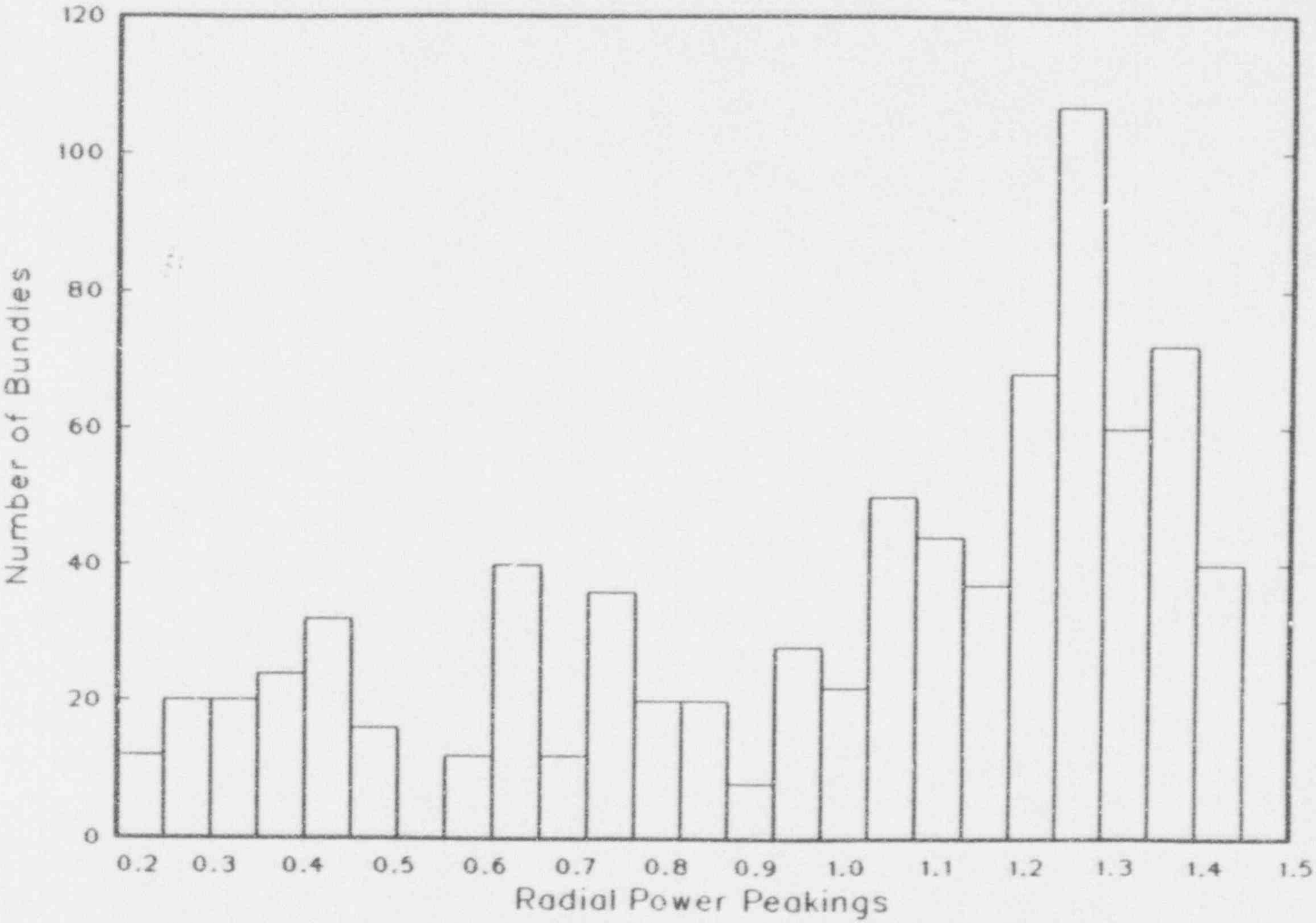

FIGURE 3.1 GRAND GULF UNIT 1 CYCLE 6 SAFETY LIMIT DESIGN RADIAL HISTOGRAM

CONTROL ROD

0.986	1.025	1.018	1.030	1.063	1.030	1.018	1.025	0.986
1.025	0.967	1.047	0.989	0.814	0.989	1.047	0.966	1.025
1.018	1.047	1.028	0.970	0.994	0.968	1.027	1.047	1.019
1.030	0.989	0.970	0.897	0.000	1.050	0.970	0.990	1.031
1.063	0.814	0.994	0.000	0.000	0.000	0.999	0.814	1.064
1.030	0.989	0.968	1.050	0.000	0.889	0.982	0.993	1.032
1.018	1.047	1.027	0.970	0.999	0.982	1.035	1.051	1.020
1.025	0.966	1.047	0.990	0.814	0.993	1.051	0.967	1.027
0.986	1.025	1.019	1.031	1.064	1.032	1.020	1.027	0.987

FIGURE 3.2 GRAND GULF UNIT 1 CYCLE 6 SAFETY LIMIT DESIGN BASIS LOCAL POWER DISTRIBUTION

4.0 NUCLEAR DESIGN ANALYSIS

4.1 Fuel Bundle Nuclear Design Analysis

Assembly Average Enrichment, w/o U235	3.38 ANF-1.5 H 2.94 ANF-1.5 L
Radial Enrichment Distribution	See Reference 10
Axial Enrichment Distribution	Figure 4.1
Burnable Poisons	Figure 4.1
Location of Non-Fueled Rods	See Reference 10
Neutronic Design Parameters	Table 4.1

4.2 Core Nuclear Design Analysis

4.2.1 Core Configuration

Core Configuration	Figure 4.2
Core Exposure at EOC5	24805 MWd/MTU
Core Exposure at BOC6	13385 MWd/MTU
Core Exposure at EOC6	25831 MWd/MTU
Maximum Cycle 6 Licensing Exposure Limit	26649 MWd/MTU

4.2.2 Core Reactivity Characteristics[1],[2]

BOC6 Cold K-effective, All Rods Out	1.11869
BOC6 Cold K-effective, All Rods In	0.95220
BOC6 Cold K-effective, Strongest Rod Out	0.98914
Reactivity Defect/R-Value[3]	.07% Delta-K/K
Standby Liquid Control System Reactivity, 660 PPM Cold Conditions, K-effective	0.96850

[1]Includes calculational bias.

[2]Evaluated at nominal EOC5-818 MWd/MTU.

[3]The R-Value will be revised based on actual EOC5 conditions.

4.2.4 Core Hydrodynamic Stability

Core hydrodynamic stability is addressed by the licensee.

TABLE 4.1 NEUTRONIC DESIGN VALUES

Fuel Assembly (9x9-5)

Number of fuel rods	76
Number of inert water rods	5
Fuel rod enrichments	See Reference 10
Fuel rod pitch, inches	0.563
Fuel assembly loading, kgU	
ANF-1.5 H	175.70
ANF-1.5 L	175.57

Core Data

Number of fuel assemblies	800
Rated thermal power, MWt	3833
Rated core flow, Mlbm/hr	112.5
Core inlet subcooling, Btu/lbm	22.2
Moderator temperature, °F	551
Channel thickness, inch	0.120
Fuel assembly pitch, inch	6.0
Sym. water gap thickness, inch	0.545

Control Rod Data

Absorber material	B4C
Total blade span, inch	9.804
Total blade support span, inch	1.55
Blade thickness, inch	0.328
Blade face-to-face internal dimension, inch	0.238
Absorber rods per blade (wing)	72 (18)
Absorber rod outside diameter, inch	0.22
Absorber rod inside diameter, inch	0.166
Absorber density, percent of theoretical	70

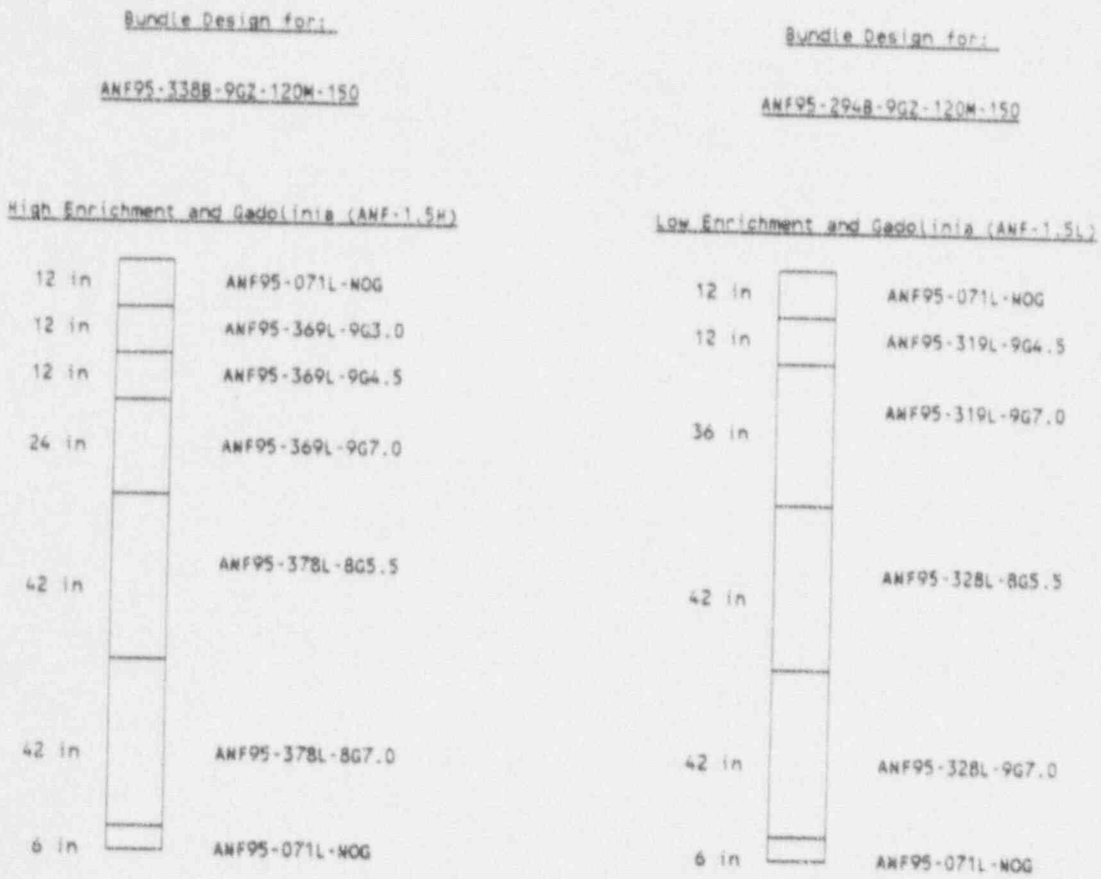

FIGURE 4.1 GRAND GULF UNIT 1 CYCLE 6 BUNDLE DESIGNS

A2	C1	F0	C1	F0	D1	F0	D1	F0	C1	F0	D1	A2	D1	A2	A2
C1	F0	D1	A2	D1	F0	D1	F0	D1	F0	A2	E0	A2	A2	A2	A2
F0	D1	A2	D1	F0	D1	F0	D1	F0	C1	E0	D1	E0	D1	A2	
C1	A2	D1	F0	C1	F0	D1	F0	C1	F0	A2	E0	A2	D1	A2	
F0	D1	F0	C1	F0	D1	F0	C1	F0	C1	E0	D1	E0	A2	A2	
D1	F0	D1	F0	D1	F0	C1	F0	C1	F0	A2	E0	A2	D1	A2	
F0	D1	F0	D1	F0	C1	F0	D1	F0	C1	E0	C1	E0	A2	A2	
D1	F0	D1	F0	C1	F0	D1	E0	D1	E0	A2	E0	A2	A2		
F0	D1	F0	C1	F0	C1	F0	D1	A2	A2	E0	D1	A2			
C1	F0	C1	F0	C1	F0	C1	E0	A2	A2	A2	C1	A2			
F0	A2	E0	A2	E0	A2	E0	A2	E0	A2	D1	C1	A2			
D1	E0	D1	E0	D1	E0	D1	E0	D1	C1	C1	B2				
A2	A2	E0	A2	E0	A2	E0	A2	A2	A2	A2					
D1	A2	D1	D1	A2	D1	A2	A2								
A2	A2	A2	A2	A2	A2	A2									
A2	A2														

XY — X= Fuel Type, Y= Cycles Irradiated

A	240	SNP 8x8 3.37 w/o U-235 (ANF-1.3)
B	4	SNP 9x9 3.25 w/o U-235 (ANF-1.3)
C	180	SNP 9x9 3.42 w/o U-235 (ANF-1.4)
D	104	SNP 9x9 3.42 w/o U-235 (ANF-1.4)
E	100	SNP 9x9 3.38 w/o U-235 (ANF-1.5)
F	172	SNP 9x9 2.94 w/o U-235 (ANF-1.5)

FIGURE 4.2 GRAND GULF UNIT 1, CYCLE 6 REFERENCE CORE LOADING PATTERN
(QUARTER CORE, REFLECTIVE SYMMETRY)

5.0 ANTICIPATED OPERATIONAL OCCURRENCES

Applicable Generic Transient Methodology Report — References 5, 8.8

5.1 Analysis of Plant Transients

Reference 4

(Applicable at rated conditions)

Transient	Delta-CPR*		
	EOC-30 EFPD	EOC	EOC+30 EFPD
LRNB	0.14	0.16	0.18
LFWH**	0.09	0.09	0.09
CRWE***	0.10	0.10	0.10
FWCFNB	0.13	0.15	0.16

* Limiting values.
** Applicable at all conditions.
*** Statistically determined, Reference 6.

Exposure Dependent Limit - $MCPR_e$ — Figure 5.5

5.2 Analyses For Reduced Flow Operation

Reference 4

$MCPR_f$ — Figure 5.1

$LHGRFAC_f$ — Figure 5.3

5.3 Analyses For Reduced Power Operation

Reference 4

$MCPR_p$ — Figure 5.2

$LHGRFAC_p$ — Figure 5.4

5.4 ASME Overpressurization Analysis

Reference 4

Limiting Event — MSIV Closure

Worst Single Failure — MSIV Position Scram Trip

5.5 Control Rod Withdrawal Error — Reference 6

Values of delta-CPR as a function of core power level resulting from a CRWE transient were developed in Reference 6 on a generic basis for BWR/6 class of plants (including Maximum Extended Operating Domain operation). Analysis has been performed demonstrating continued applicability of the generic CRWE analysis results.

5.6 Fuel Loading Error — Reference 8.1

	With Loading Error	Correctly Loaded Core
Maximum LHGR, kW/ft	12.97	11.80
Minimum MCPR*	1.21	1.31

*Determined using ANFB Critical Power Correlation.

5.7 Determination of Thermal Limits

The results of the analyses presented in Sections 5.1, 5.2, and 5.3 are used for the determination of the operating limit. Section 5.1 provides the results of analyses at rated conditions, including the operating limit as a function of exposure in the cycle ($MCPR_e$, Figure 5.5). Sections 5.2 and 5.3 provide for the determination of operating limit at off-rated conditions of reduced flow and reduced power operation ($MCPR_f$, Figure 5.1 and $MCPR_p$, Figure 5.2). The highest value of MCPR from among the ones presented in these figures for the operating condition of the reactor is to be selected as the operating limit of interest.

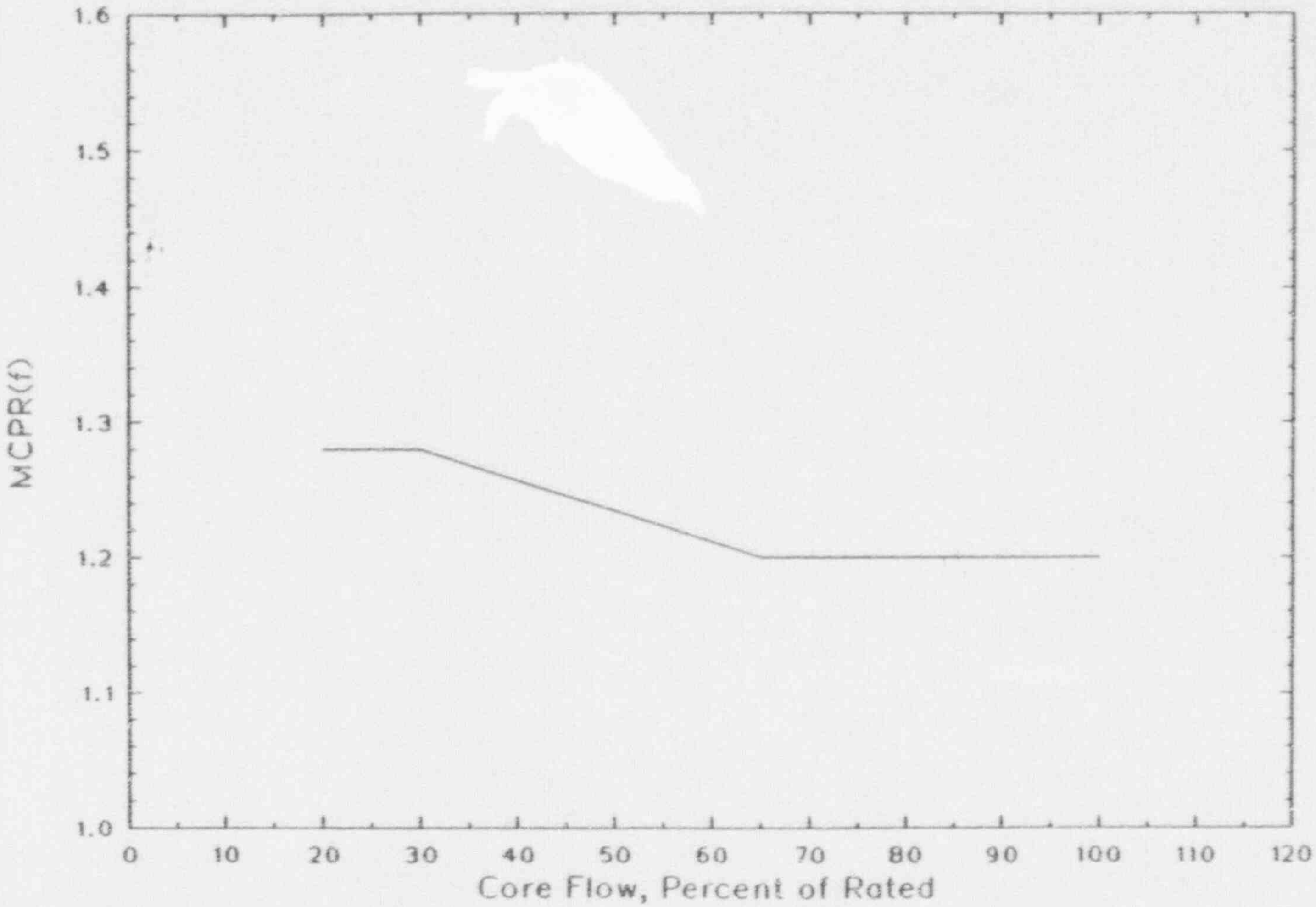

FIGURE 5.1 FLOW DEPENDENT MCPR LIMITS FOR GRAND GULF UNIT 1 CYCLE 6

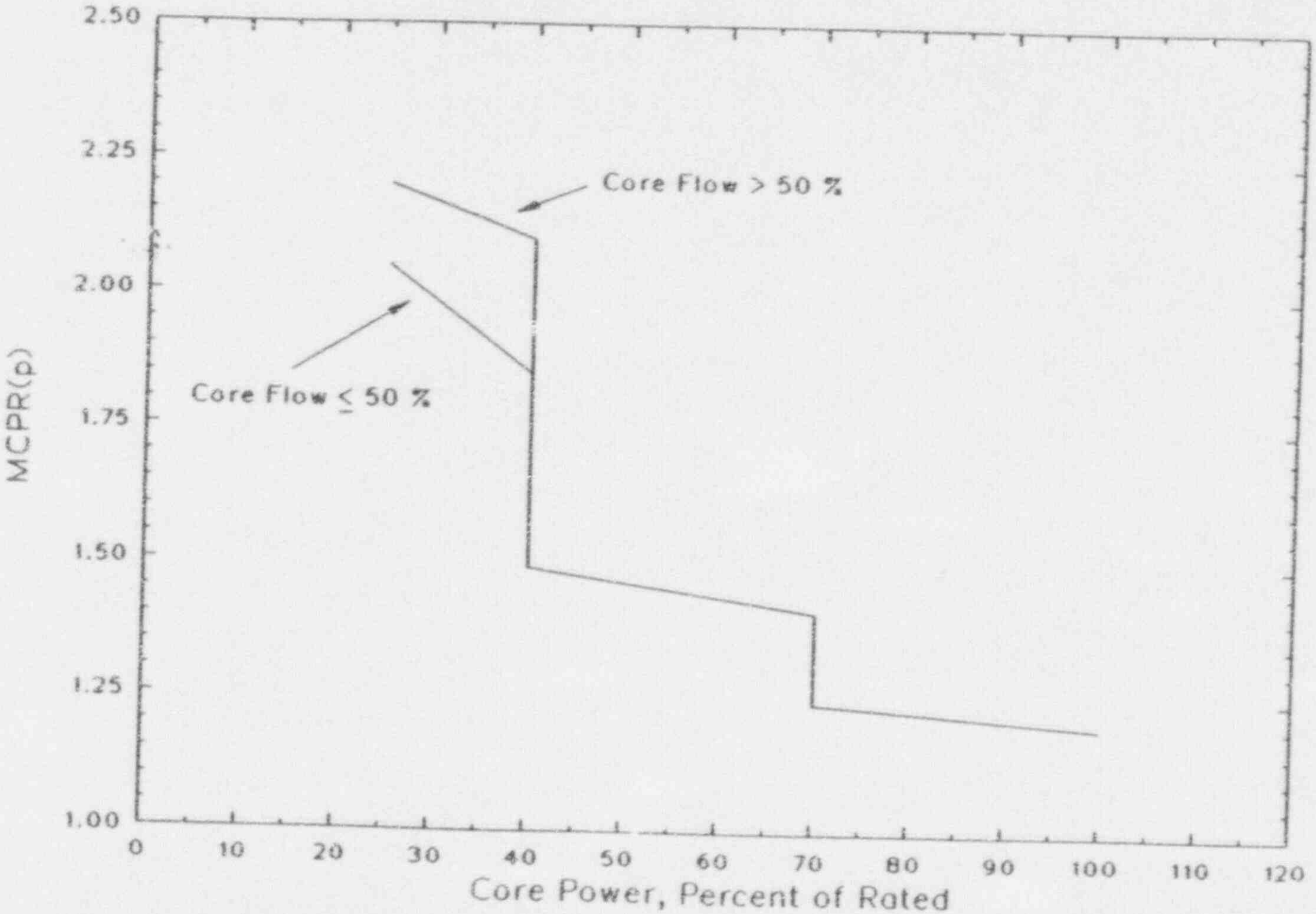

FIGURE 5.2 POWER DEPENDENT MCPR LIMITS FOR GRAND GULF UNIT 1 CYCLE 6

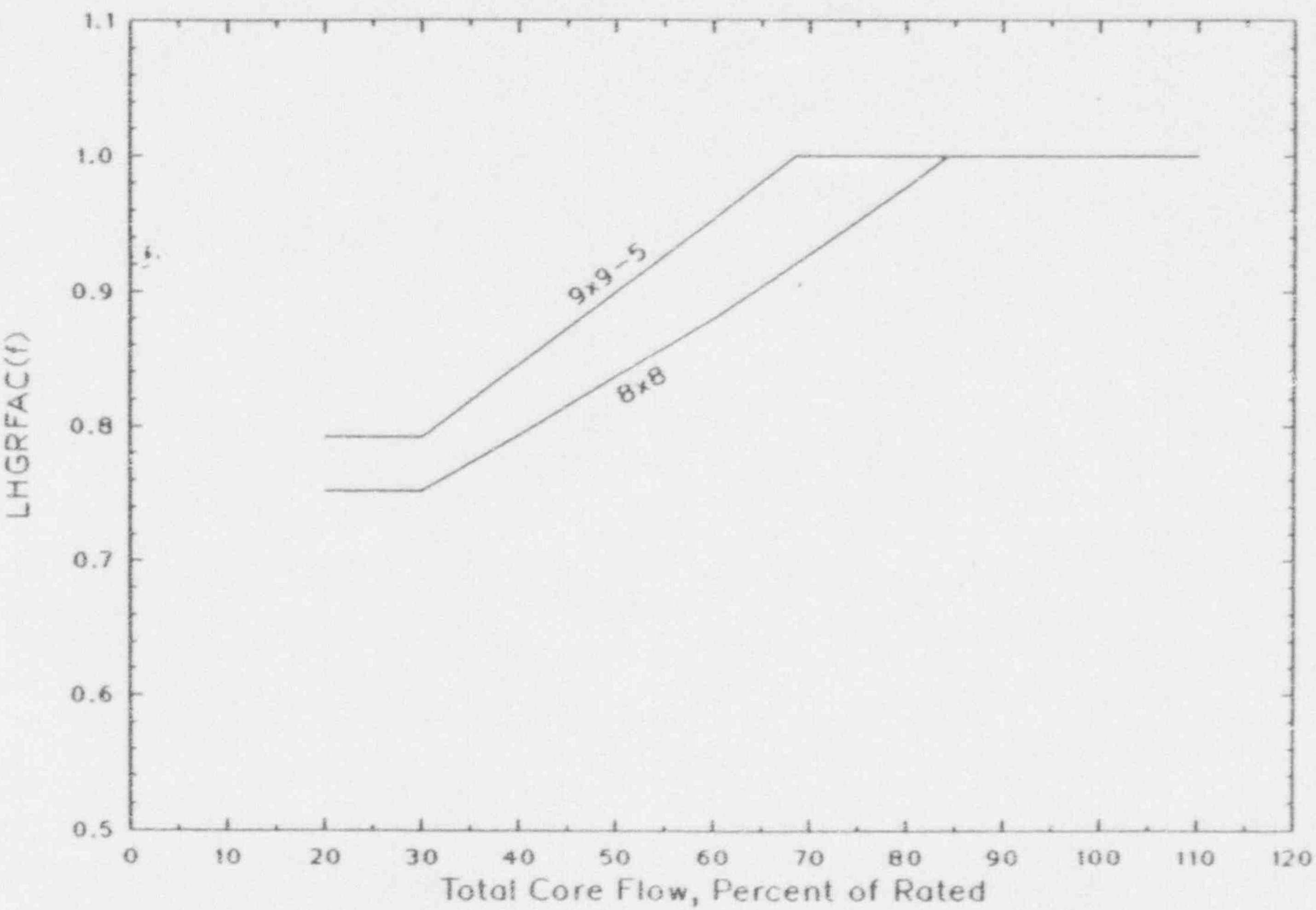

FIGURE 5.3 FLOW DEPENDENT LHGRFAC VALUE FOR GRAND GULF UNIT 1 CYCLE 6

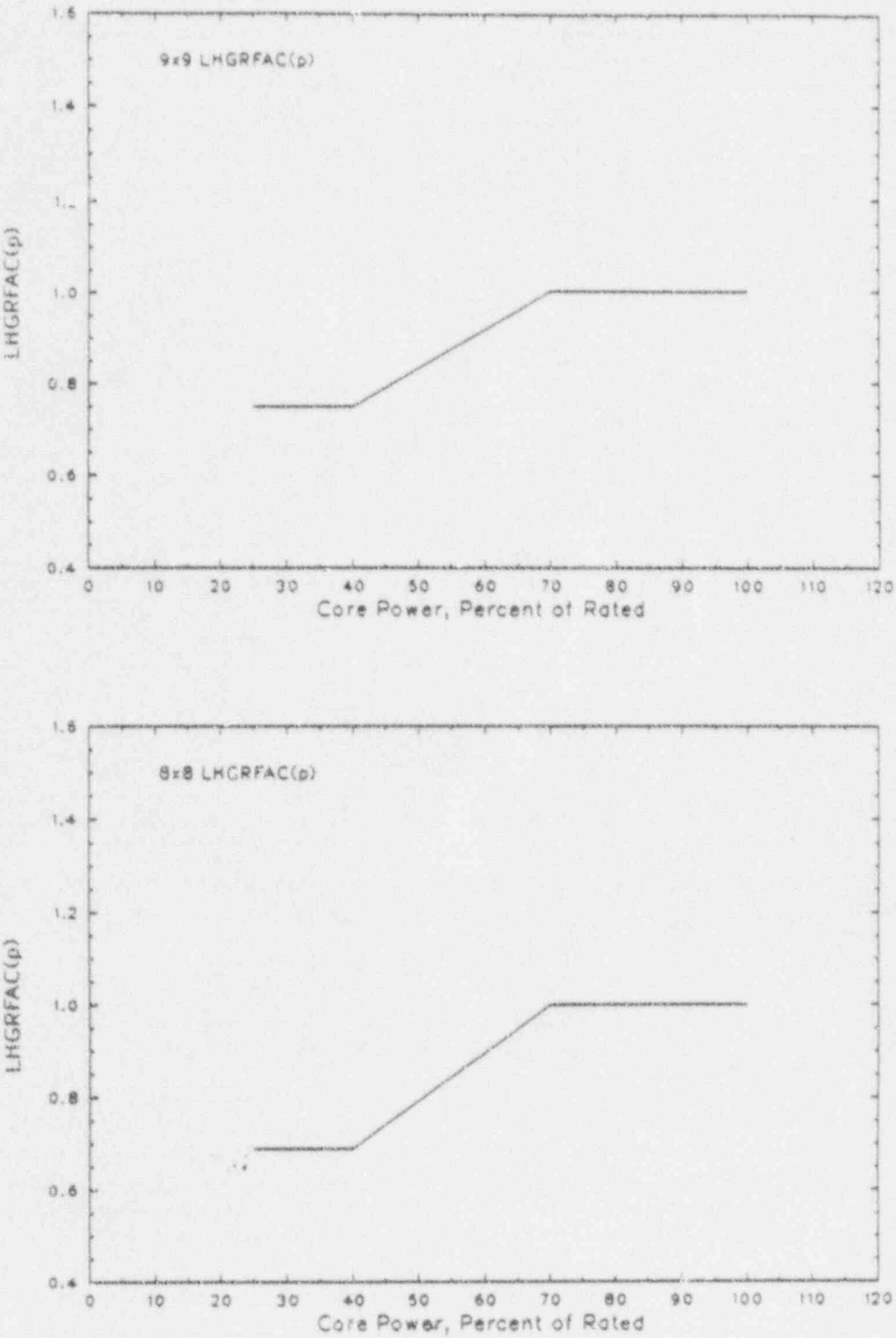

FIGURE 5.4 POWER DEPENDENT LHGRFAC VALUE FOR GRAND GULF UNIT 1 CYCLE 6

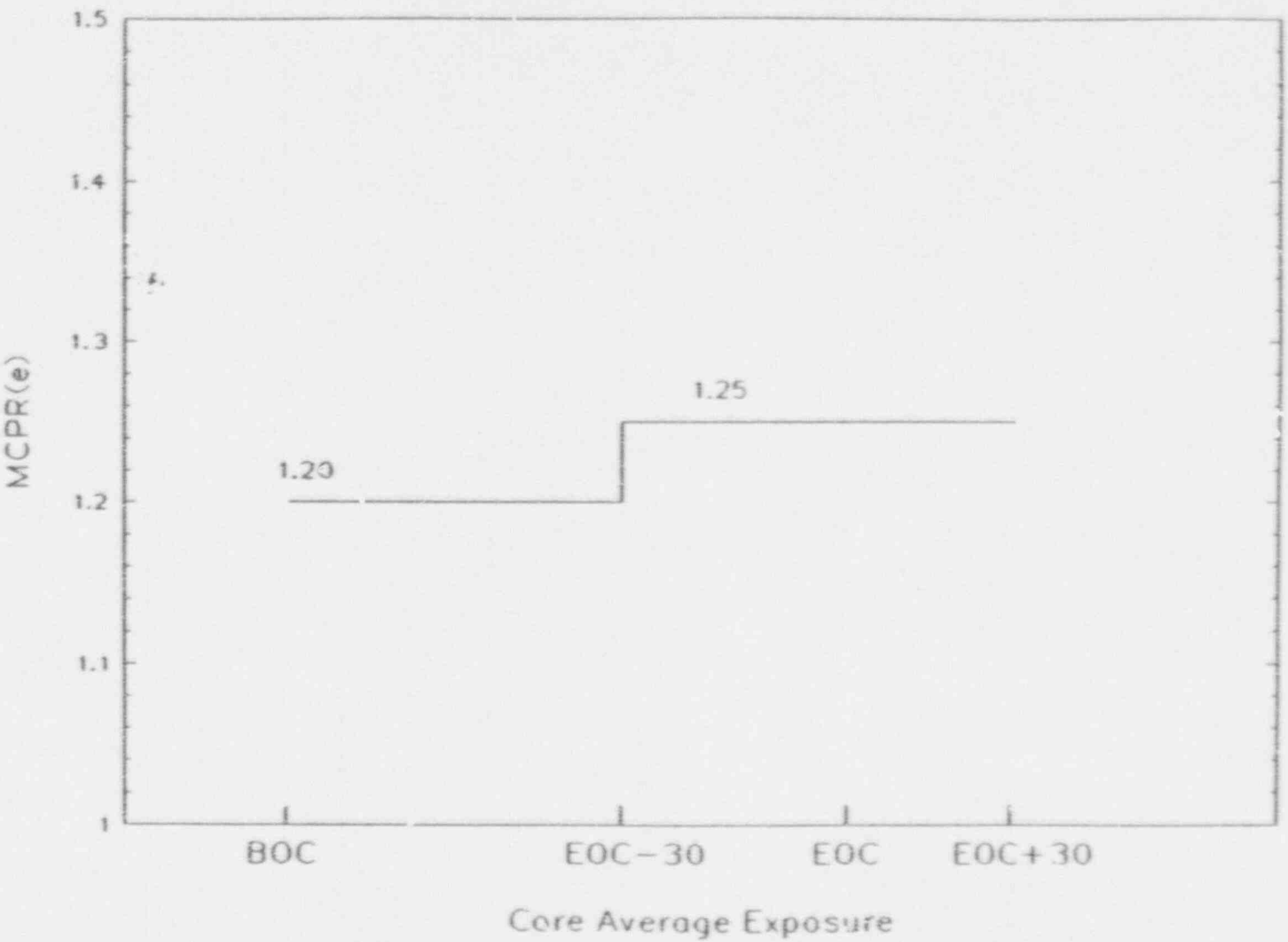

FIGURE 5.5 EXPOSURE DEPENDENT MCPR LIMITS FOR GRAND GULF UNIT 1 CYCLE 6

6.0 POSTULATED ACCIDENTS

6.1 Loss-Of-Coolant Accident

6.1.1 Break Location Spectrum — Reference 7

6.1.2 Break Size Spectrum — Reference 7

6.1.3 MAPLHGR Analysis For SNP 8x8 and 9x9-5 Fuel — References 8 and 12

Limiting Break: Double-Ended Guillotine Pipe Break in
Recirculation Pump Discharge Line with
1.00 Discharge Coefficient (1.0 DEG/RD)

The spray heat transfer coefficients identified in 10CFR50 Appendix K are used for the 9x9-5 fuel in an identical manner as in the previous approved analysis for Grand Gulf 1 (Reference 15). This includes the use of 5 BTU/hr-ft^2-°F for all of the unheated surfaces including the five water rods.

MAPLHGR results for the two reload fuel types are reported below:

	Maximum PCT (°F)	Peak Local Metal Water Reaction (%)
8x8 Fuels	1691	0.3
9x9 Fuels	1713	0.5

The core wide metal water reaction is less than 0.1%.

The MAPLHGR limits for 8x8 and 9x9-5 are shown in Figure 6.1. These are bounding limits. The 9x9-5 limits are bounding for the LTA. The 8x8 limits are provided in Reference 8. For single-loop operation, a reduction factor of 0.86 is applied to the two-loop MAPLHGR limits shown in Figure 6.1. Application of this reduction factor ensures that the PCT for a single-loop operation LOCA is bounded by the two-loop LOCA analysis.

6.2	Control Rod Drop Accident	Reference 8.1
	Dropped Control Rod Worth, mk	11.4
	Doppler Coefficient, $\Delta K/K/°F$	-10.4×10^{-6}
	Effective Delayed Neutron Fraction	5.40×10^{-3}
	Four-Bundle Local Peaking Factor	1.225
	Maximum Deposited Fuel Rod Enthalpy, cal/g	166

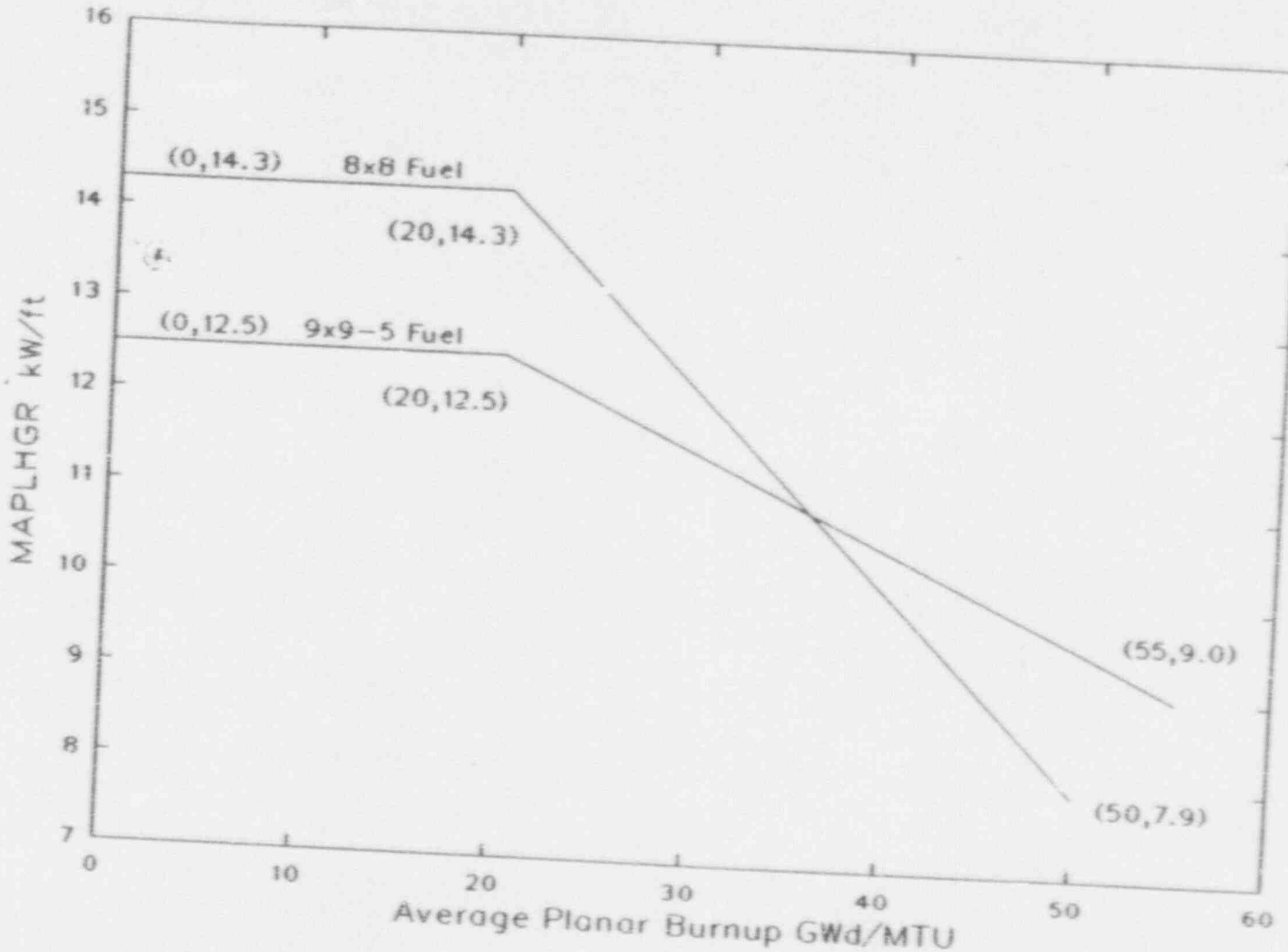

FIGURE 6.1 MAPLHGR VS AVERAGE PLANAR EXPOSURE FOR SNP 8X8 AND 9X9-5 RELOAD FUEL

7.0 TECHNICAL SPECIFICATIONS

7.1 Limiting Safety System Settings

7.1.1 MCPR Fuel Cladding Integrity Safety Limit

Safety Limit MCPR	1.06* 1.07**

7.1.2 Steam Dome Pressure Safety Limit

Pressure Safety Limit	1325 psig

7.2 Limiting Conditions For Operation

7.2.1 Average Planar Linear Heat Generation Rate for SNP Fuel

The following MAPLHGR limits are consistent with 10CFR50.46 requirements. The MAPLHGR limit is not used to protect the design basis LHGR limits for the fuel types co-resident in Cycle 6.

Average Planar Exposure	MAPLHGR 8x8	MAPLHGR 9x9-5
0.0 GWd/MTU	14.3 kW/ft	12.5 kW/ft
20.0	14.3	[illegible]5
50.0	7.9	9.5
55.0	-	9.0

For single-loop operation, a reduction factor of 0.86 is applied to the above two-loop MAPLHGR limits.

---

* The 1.06 safety limit accounts for channel bow.

** A safety limit of 1.07 is to be applied during single loop operation.

7.2.2 Minimum Critical Power Ratio

MCPR(f)	Figure 5.1
MCPR(p)	Figure 5.2
MCPR(e)	Figure 5.5

7.2.3 Linear Heat Generation Rate For SNP Fuel

The LHGR limits for SNP 8x8 fuel for Grand Gulf 1 have been extended to support Cycle 6 operation. These limits, which are based on Figure 4.1 of Reference 16, are as follows:

Average Planar Exposure	LHGR
0.00 GWd/MTU	16.0 kW/ft
25.40	14.1
40.00	10.0
55.00	8.0

The LHGR limits for 9x9-5 fuel, based on Figure 3.1 of Reference 13, for SNP reload fuel during Cycle 6 operation are as follows:

Average Planar Exposure	LHGR
0.00 GWd/MTU	13.1 kW/ft
15.50	13.1
55.00	8.0

$LHGRFAC_f$ and $LHGRFAC_p$ multipliers are applied directly to the Technical Specification LHGR limits for each fuel type at reduced power and/or flow conditions to ensure protection of the limits.

LHGRFAC Multipliers for Off-Nominal Conditions:

LHGRFAC(f)	Figure 5.3
LHGRFAC(p)	Figure 5.4

7.3 Surveillance Requirements

7.3.1 Scram Insertion Time Surveillance

Thermal margins are based on analyses in which scram performance was assumed consistent with the Technical Specification limits. No additional surveillance for scram performance is required above that already being done for conformance to Technical Specifications.

7.3.2 Stability Surveillance

Core stability surveillances have been addressed by the Licensee in TS 4.4.1.1.1.

8.0 METHODOLOGY REFERENCES

Section 8 References 8.1 through 8.18 are contained in the following report:

"Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," XN-NF-80-19(A), Volume 4, Revision 1, Exxon Nuclear Company, Richland, Washington (March 1985).

Reference 8.6 is superseded by:

8.6 "Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description," XN-NF-80-19(P)(A), Volume 3, Revision 2 (January 1987).

References 8.9 and 8.18 are superseded by:

8.9 "ANFB Critical Power Correlation," ANF-1125(P)(A), and Supplements 1 and 2 (April 1990).

Reference 8.10 is superseded by:

8.10 "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors," ANF-524(P)(A), Revision 2, and Supplements 1 and 2 (November 1990).

9.0 REFERENCES

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2. "Grand Gulf Unit 1 Cycle 2 Reload Analysis," XN-NF-86-35, Revision 3, Exxon Nuclear Company, Richland, WA, August 1986.

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

4. "Grand Gulf Unit 1 Cycle 6 Plant Transient Analysis," EMF-91-168, Siemens Nuclear Power Corporation, Richland, WA, October 1991.

5. "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analysis," ANF-913(P)(A), Volume 1, Revision 1 and Supplements 2, 3, and 4, August 1990.

6. "BWR/6 Generic Rod Withdrawal Error Analysis, MCPRp," XN-NF-825(A), Exxon Nuclear Company, Richland, WA, May 1986, and XN-NF-825(P)(A), Supplement 2, October 1986.

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10. "Grand Gulf Unit 1 Reload ANF-1.5 Design Report, Mechanical, Thermal Hydraulic, and Neutronic Design for Advanced Nuclear Fuels 9x9-5 Fuel Assemblies," ANF-91-080(P), Advanced Nuclear Fuels Corporation, Richland, WA, July 1991.

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12. "Grand Gulf Unit 1 Cycle 4 Reload Analysis," ANF-88-149, Advanced Nuclear Fuels Corporation, Richland, WA, November 1988.

13. "Generic Mechanical Design for Advanced Nuclear Fuels 9x9-5 BWR Reload Fuel," ANF-88-152(P)(A) with Amendment 1 and Supplement 1, Advanced Nuclear Fuels Corporation, Richland, WA, November 1990.

14. Letter, R. A. Copeland (ANF) to R. C. Jones (NRC), "Minor Mechanical Design Change," March 5, 1991 (RAC:026:91).

15. "Grand Gulf Unit 1 Cycle 5 Reload Analysis," ANF-90-022, Revision 2, Advanced Nuclear Fuels Corporation, Richland, WA, August 1990.

16. "Grand Gulf Unit 1 XN-1.3, Cycle 4 Mechanical Design Report," ANF-88-183(P), Supplement 1, Siemens Nuclear Power Corporation, Richland, WA, August 1991.

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