ADVANCED NUCLEAR FUELS CORPORATION

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GRAND GULF UNIT 1 CYCLE 5 RELOAD ANALYSIS

Prepared by

R B Macduff

R. B. Macduff
BWR Fuel Engineering
Fuel Engineering and Licensing

David Hershberger

D. E. Hershberger
BWR Fuel Engineering
Fuel Engineering and Licensing

Mike Hibbard

M. J. Hibbard
BWR Fuel Engineering
Fuel Engineering and Licensing

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During the NRC review of the nuclear peaking uncertainties of the MICROBURN-B methodology, ANF was informed that the proposed TIP asymmetry uncertainty as presented in Reference 14 would require further extensive review. ANF was also informed that concurrence to use the currently accepted value would allow the NRC to complete remaining actions associated with the issuance of the MICROBURN-B SER without further technical review by the NRC staff. ANF agreed to the use of the currently accepted value as stated in Reference 15.

The change in uncertainty value required ANF to evaluate the impact upon analyses performed for the Cycle 5 licensing campaign for Grand Gulf Unit 1 as provided in ANF-90-021 and ANF-90-022.

Revision 2 of this report is issued to effect the changes in results associated with the increase in TIP asymmetry uncertainty. Text changes from Revision 1 are indicated by revision bars in the left margin of the report. Figures 5.1, 5.2, and 5.5 are also revised.

1.0 INTRODUCTION

This report provides the results of the analyses performed by Advanced Nuclear Fuels Corporation (ANF) in support of the Cycle 5 reload for Grand Gulf Unit 1. 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. Methodology used in this report which supersedes XN-NF-80-19(A), Volume 4, Revision 1, is referenced as appropriate.

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

Changes from Cycle 4 to Cycle 5 for Grand Gulf Unit 1 include an additional reload of ANF fuel resulting in a core comprised of once and twice burned ANF 8x8 designs, four ANF 9x9-5 LTAs, and fresh ANF 9x9-5 design. The 9x9-5 reload fuel is mechanically, neutronically, and thermal hydraulically compatible with the co-resident 8x8 fuel inserted in previous cycles. The cycle length remains 18 months and the nominal cycle energy remains 1698 GWd. A reload batch design composed of 284 assemblies enriched to 3.42 w/o U235 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 272 once exposed ANF 8x8 reload fuel assemblies, four once exposed 9x9-5 lead fuel assemblies and 240 twice exposed ANF 8x8 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 4 operation and conditions bounding Cycle 5 operation. The $MCPR_p$ and $MCPR_f$ limits have been verified or revised to reflect ANF calculated limits. As in Cycle 4, provision has been made in the flow dependent MCPRs for "loop manual" operation as well as "non-loop manual" operation (Reference 11). Analyses were performed at EOC-2000 MWd/MTU, at EOC, and at EOC + 30 EFPD providing limits for Cycle 5 that are cycle exposure dependent. The analyses also included support of the power/flow operation map for Maximum Extended Operating Domain 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 ANF's core monitoring methodology, POWERPLEX® CMSS, in accordance with ANF's thermal limits methodology, THERMEX (Reference 8.6).

The ANF evaluation for Grand Gulf Unit 1 Single Loop Operation (SLO) without condenser bypass and LOCA-seismic considerations were confirmed for Cycle 2 and subsequent cycles. Since the Cycle 5 SLO analyses are performed using new methodology (References 5 and 8.1 through 8.18), the Cycle 5 results supersede the Cycle 2 results.

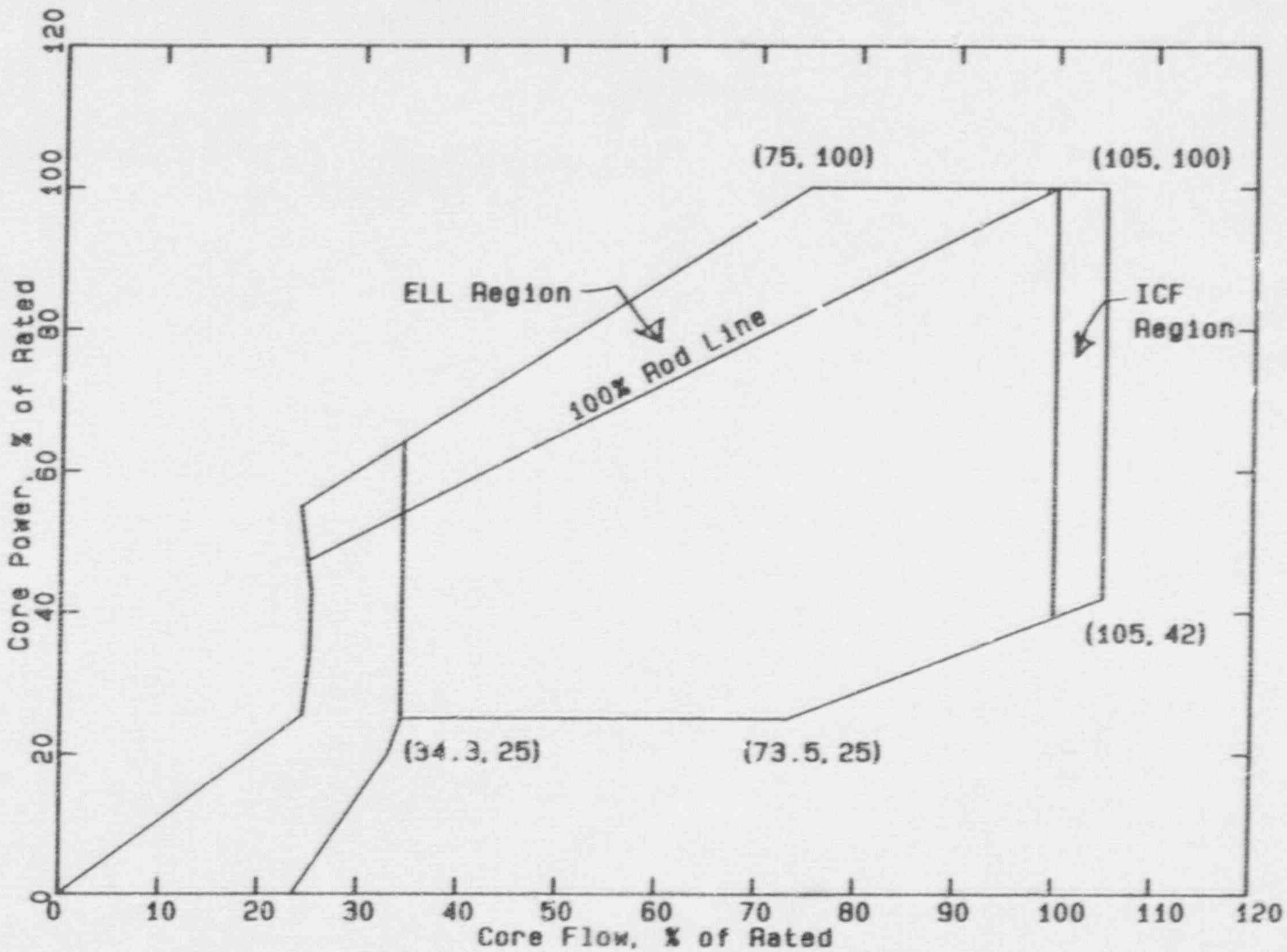

Figure 1.1 Power/Flow Map Used for Grand Gulf Unit 1 MEOD Analysis

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 ANF fuel assemblies.

The expected power history for the fuel to be irradiated during Cycle 5 is bounded by the design LHGR of Figure 3.1 of References 3 and 13.

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 3 and 13.

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	1.09*

3.3.1 Nominal Coolant Condition in Monto Carlo Analysis

Core Power	4754 MWt
Core Inlet Enthalpy	522.3 Btu/lbm
Reference Pressure	1050 psia
Feedwater Temperature	420°F
Feedwater Flow Rate	20.43 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 1.09 includes effects for channel bow and single loop operation.

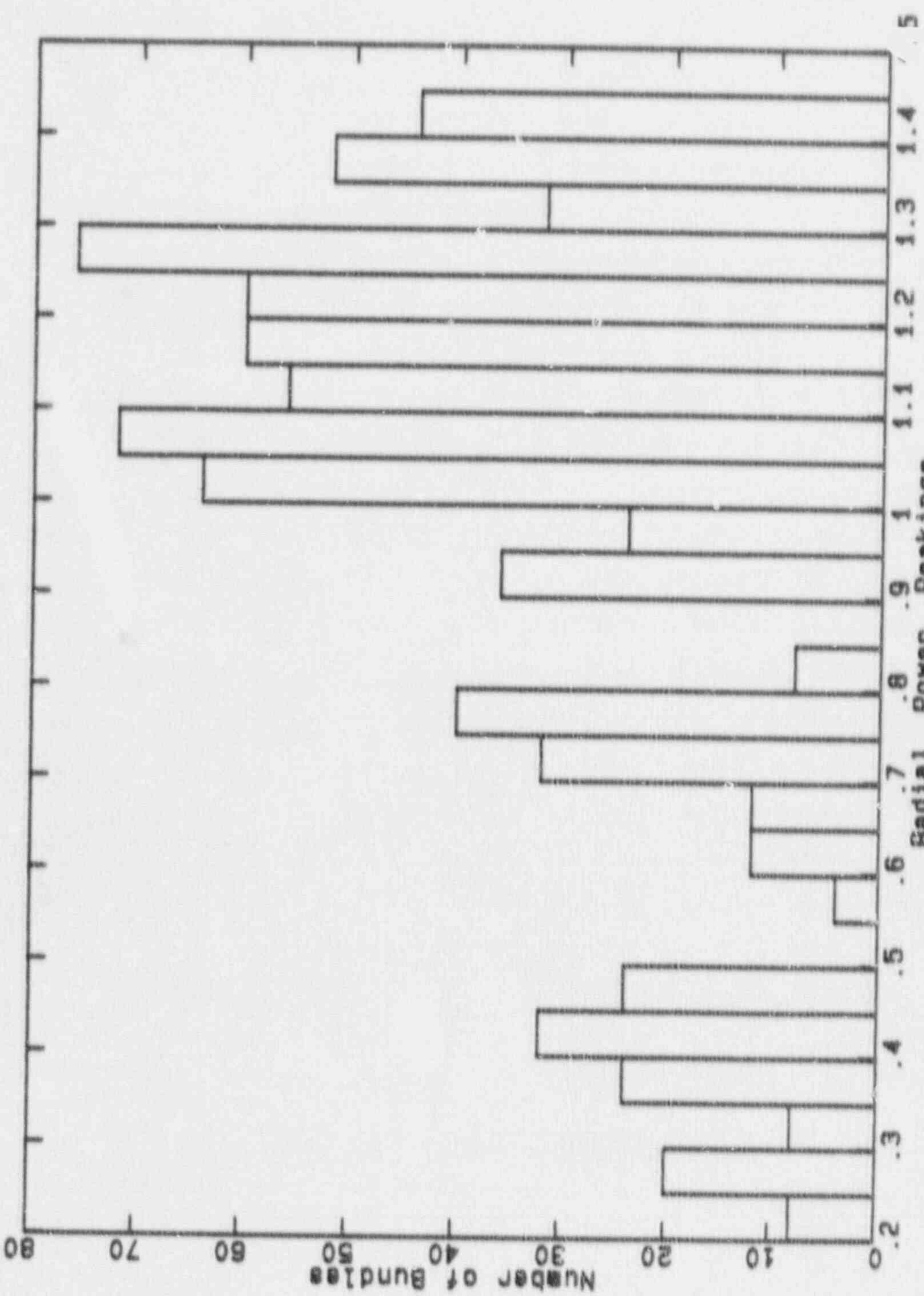

Figure 3.1 Grand Gulf Unit 1 Cycle 5 Safety
Limit Design Radial Histogram

1.116	1.127	1.108	1.117	1.107	1.116	1.107	1.126	1.116
1.127	0.786	1.007	0.973	0.636	0.971	1.004	0.785	1.126
1.108	1.007	0.949	0.954	0.976	0.946	0.944	1.004	1.106
1.117	0.973	0.954	0.735	0.000	1.045	0.946	0.970	1.116
1.107	0.636	0.976	0.000	0.000	0.000	0.976	0.633	1.106
1.116	0.971	0.946	1.045	0.000	0.706	0.956	0.972	1.116
1.107	1.004	0.944	0.946	0.976	0.956	0.949	1.006	1.107
1.126	0.785	1.004	0.970	0.633	0.972	1.006	0.784	1.126
1.116	1.126	1.106	1.116	1.106	1.116	1.107	1.126	1.116

Figure 3.2 Grand Gulf Unit 1 Cycle 5 Safety Limit Design Basis
Local Power Distribution

4.0 NUCLEAR DESIGN ANALYSIS

4.1 Fuel Bundle Nuclear Design Analysis

Assembly Average Enrichment	3.42 w/o
Radial Enrichment Distribution	Figures 4.1 - 4.3
Axial Enrichment Distribution	Uniform 3.80 w/o with 12" natural uranium at top and 6" at bottom
Burnable Poisons	Figures 4.1 - 4.3

Note: Burnable poisons are not distributed uniformly over the enriched length of the designated rods. The natural uranium axial blanket sections do not contain burnable absorber material.

Location of Non-Fueled Rods	Figures 4.1 - 4.3
Neutronic Design Parameters	Table 4.1

4.2 Core Nuclear Design Analysis

4.2.1 Core Configuration	Figure 4.4
Core Exposure at EOC4	23016 MWd/MTU
Core Exposure at BOC5	12872 MWd/MTU
Core Exposure at EOC5	24948 MWd/MTU
Maximum Cycle 5 Licensing Exposure Limit	25766 MWd/MTU

4.2.2 Core Reactivity Characteristics[(1),(2)]

BOC5 Cold K-effective, All Rods Out	1.12245
BOC5 Cold K-effective, All Rods In	0.95342
BOC5 Cold K-effective, Strongest Rod Out	0.98956
Reactivity Defect/R-Value (Minimum occurs at 0 MWd/MTU)	0.0% Delta-K/K
Standby Liquid Control System Reactivity, 660 PPM Cold Conditions, K-effective	0.97085

(1)Includes calculational bias.

(2)Evaluated at nominal EOC4-825 MWd/MTU

4.2.4 Core Hydrodynamic Stability

The results of Cycle 5 core hydrodynamic stability analyses continue to confirm the applicability of the previous cycles analyses results. The presence of 9x9-5 fuel in the Cycle 5 core does not change the conclusions of the stability analysis of the previous cycles.

Table 4.1 Neutronic Design Values

<u>Fuel Assembly (9x9-5)</u>

Number of fuel rods	76
Number of inert water rods	5
Fuel rods enrichments	Figures 4.1 - 4.3
Fuel rod pitch, inches	0.563
Fuel assembly loading, KgU	
ANF-1.4 H	175.16
ANF-1.4 L	175.59

<u>Core Data</u>

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

<u>Control Rod Data</u>

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

L1	ML1	M1	MH1	MH1	MH1	M1	ML1	L1
ML1	LL1*	MH1	H2	LL2*	H2	MH1	LL1*	ML1
M1	MH1	MH1	H2	H2	H2	MH1	MH1	M1
MH1	H2	H2	LL2	W	H2	H2	H2	MH1
MH1	LL2*	H2	W	W	W	H2	LL2*	MH1
MH1	H2	H2	H2	W	LL2	H2	H2	MH1
M1	MH1	MH1	H2	H2	H2	MH1	MH1	M1
ML1	LL1*	MH1	H2	LL2*	H2	MH1	LL1*	ML1
L1	ML1	M1	MH1	MH1	MH1	M1	ML1	L1

L1 Rods (4) --- 2.67 w/o U235
ML1 Rods (8) --- 3.33 w/o U235
M1 Rods (8) --- 3.66 w/o U235
MH1 Rods (24) --- 3.98 w/o U235
H2 Rods (22) --- 4.73 w/o U235
LL2 Rods (2) --- 2.27 w/o U235
LL1* Rods (4) --- 2.27 w/o U235 + 5.5 or 7.0 w/0 Gd_2O_3
LL2* Rods (4) --- 2.27 w/o U235 + 5.5 or 7.0 w/0 Gd_2O_3
W Rods (5) --- Inert Water Rod

FIGURE 4.1 GRAND GULF UNIT 1 CYCLE 5,
ANF-1.4 ANF3E^E8GXS95 ENRICHMENT DISTRIBUTION

L1	ML1	M1	MH1	MH1	MH1	M1	ML1	L1
ML1	LL1*	MH1	H2	LL2*	H2	MH1	LL1*	ML1
M1	MH1	MH1	H2	H2	H2	MH1	MH1	M1
MH1	H2	H2	LL2	W	H2	H2	H2	MH1
MH1	LL2*	H2	W	W	W	H2	LL2*	MH1
MH1	H2	H2	H2	W	LL2*	H2	H2	MH1
M1	MH1	MH1	H2	H2	H2	MH1	MH1	M1
ML1	LL1*	MH1	H2	LL2*	H2	MH1	LL1*	ML1
L1	ML1	M1	MH1	MH1	MH1	M1	ML1	L1

L1 Rods (4) --- 2.67 w/o U235
ML1 Rods (8) --- 3.33 w/o U235
M1 Rods (8) --- 3.66 w/o U235
MH1 Rods (24) --- 3.98 w/o U235
H2 Rods (22) --- 4.73 w/o U235
LL2 Rods (1) --- 2.27 w/o U235
LL1* Rods (4) --- 2.27 w/o U235 + 3.0, 4.5, 5.5 or 7.0 w/O Gd_2O_3
LL2* Rods (5) --- 2.27 w/o U235 + 3.0, 4.5, 5.5 or 7.0 w/O Gd_2O_3
W Rods (5) --- Inert Water Rod

FIGURE 4.2 GRAND GULF UNIT 1 CYCLE 5,
ANF-1.4 ANF380E9GXS95 ENRICHMENT DISTRIBUTION

L1	ML1	M1	MH1	MH1	MH1	M1	ML1	L1
ML1	LL1*	MH1	H2	LL2*	H2	MH1	LL1*	ML1
M1	MH1	MH1	H2	H2	H2	MH1	MH1	M1
MH1	H2	H2	LL2*	W	H2	H2	H2	MH1
MH1	LL2*	H2	W	W	W	H2	LL2*	MH1
MH1	H2	H2	H2	W	LL2*	H2	H2	MH1
M1	MH1	MH1	H2	H2	H2	MH1	MH1	M1
ML1	LL1*	MH1	H2	LL2*	H2	MH1	LL1*	ML1
L1	ML1	M1	MH1	MH1	MH1	M1	ML1	L1

L1 Rods (4) --- 2.67 w/o U235
ML1 Rods (8) --- 3.33 w/o U235
M1 Rods (8) --- 3.66 w/o U235
MH1 Rods (24) --- 3.98 w/o U235
H2 Rods (22) --- 4.73 w/o U235
LL1* Rods (4) --- 2.27 w/o U235 + 4.5, 5.5 or 7.0 w/O Gd_2O_3
LL2* Rods (6) --- 2.27 w/o U235 + 4.5, 5.5 or 7.0 w/O Gd_2O_3
W Rods (5) --- Inert Water Rod

FIGURE 4.3 GRAND GULF UNIT 1 CYCLE 5,
ANF-1.4 ANF380E10GXS95 ENRICHMENT DISTRIBUTION

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1	A2	C1	E0	C1	E0	A2	F0	A2	F0	A2	F0	A2	F0	C1	A2	A2
2	C1	F0	A2	F0	C1	E0	C1	F0	A2	E0	C1	F0	C1	C1	B2	A2
3	E0	A2	F0	C1	B2	C1	E0	C1	F0	C1	E0	B2	F0	C1	A2	
4	C1	F0	D1	E0	C1	F0	A2	F0	C1	E0	C1	F0	B2	C1	B2	
5	E0	C1	A2	C1	E0	C1	C1	A2	F0	C1	E0	C1	F0	C1	A2	
6	B2	E0	C1	F0	C1	E0	C1	F0	C1	E0	C1	F0	B2	C1	A2	
7	F0	C1	E0	A2	C1	C1	E0	A2	F0	B2	F0	C1	F0	C1	A2	
8	A2	F0	C1	F0	A2	F0	A2	E0	C1	E0	C1	F0	C1	A2		
9	F0	A2	F0	C1	F0	C1	F0	C1	E0	C1	F0	C1	B2			
10	A2	E0	C1	E0	C1	E0	A2	E0	C1	F0	A2	C1	A2			
11	F0	C1	E0	C1	E0	C1	F0	C1	F0	B2	C1	B2	A2			
12	B2	F0	A2	F0	C1	F0	C1	F0	C1	C1	A2	B2				
13	F0	C1	F0	B2	F0	B2	F0	C1	B2	A2	A2					
14	C1	C1	C1	C1	C1	C1	C1	A2								
15	A2	B2	B2	A2	A2	A2	A2									
16	B2	A2														

XY — X = FUEL TYPE, Y = CYCLES IRRADIATED

FUEL TYPE	NUMBER OF ASSEMBLIES (FULL CORE)	DESCRIPTION
A	164	ANF 8X8 XN-1.2 3.01 W/O U-235 6GD AT 4.0%
B	76	ANF 8X8 XN-1.2 3.01 W/O U-235 8GD AT 4.0%
C	272	ANF 8X8 ANF-1.3 3.37 W/O U-235 8GD AT 4.0% \ 5.0%
D	4	ANF 9X9 ANF-1.3 3.25 W/O U-235 8GD AT 5.0% \ 6.0%
E	104	ANF 9X9 ANF-1.4 3.42 W/O U-235 10GD AXIALLY ZONED
F	180	ANF 9X9 ANF-1.4 3.42 W/O U-235 9GD AXIALLY ZONED

Figure 4.4 Grand Gulf Unit 1 Cycle 5 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

(Applicable at rated conditions) — Reference 4

Transient	Delta-CPR*		
	EOC-2000 MWd/MTU	EOC	EOC+30 EFPD
LRNB	0.06	0.20	0.21
LFWH**	0.09	0.09	0.09
CRWE***	0.10	0.10	0.10
FWCFNB	0.08	0.13	–

* 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

Maximum Vessel Pressure	1291 psig
Maximum Dome Pressure	1269 psig

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). Power dependent limits of MCPR are based on these results as well as the results from the Cycle 5 specific transient analysis (Reference 4).

5.6 Fuel Loading Error — Reference 8.1

	With Loading Error	Correctly Loaded Core
Maximum LHGR	14.33	12.83
Minimum MCPR*	1.21	1.28

*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.

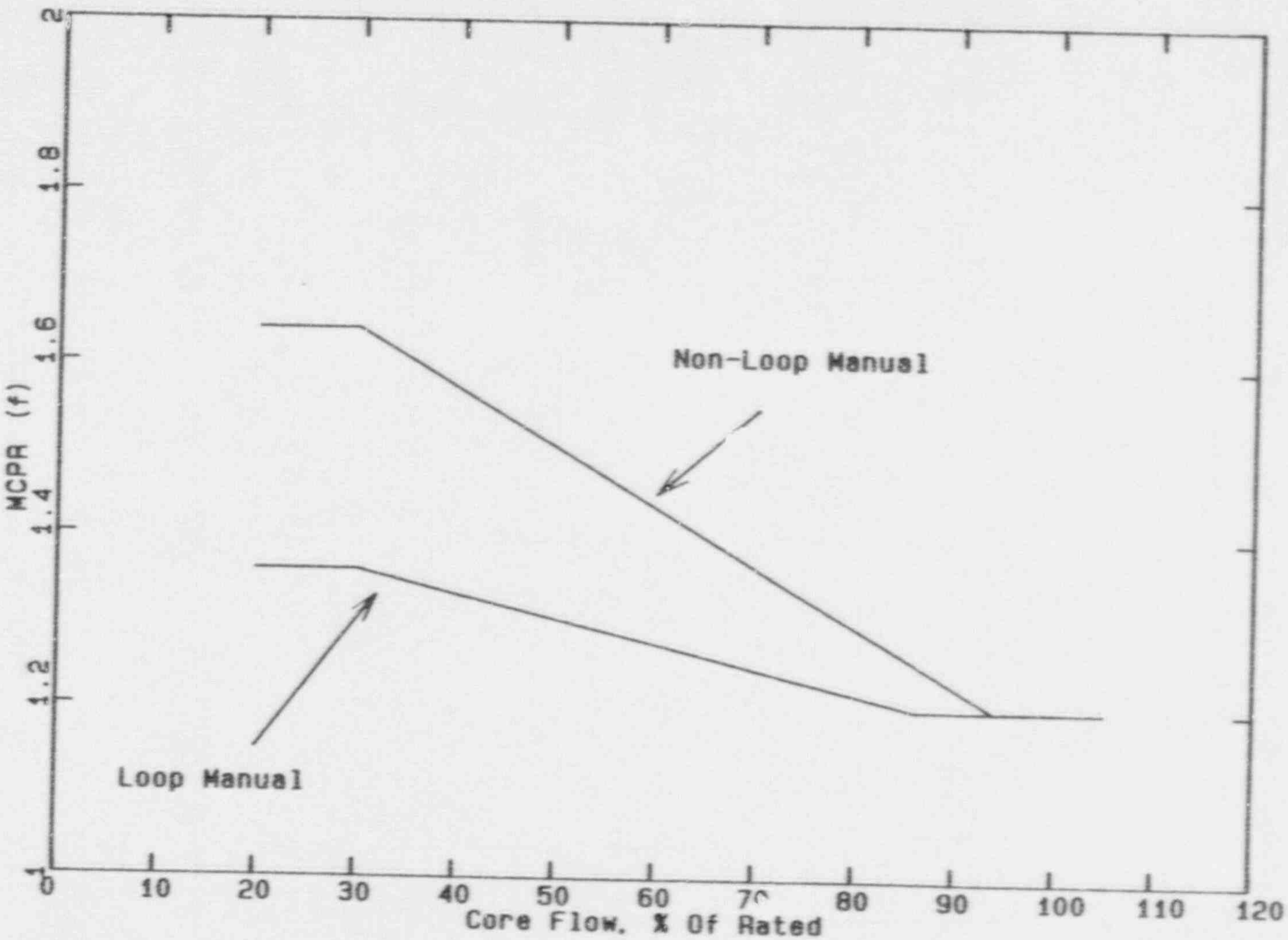

Figure 5.1 Flow Dependent MCPR Limits For Grand Gulf Unit 1 Cycle 5

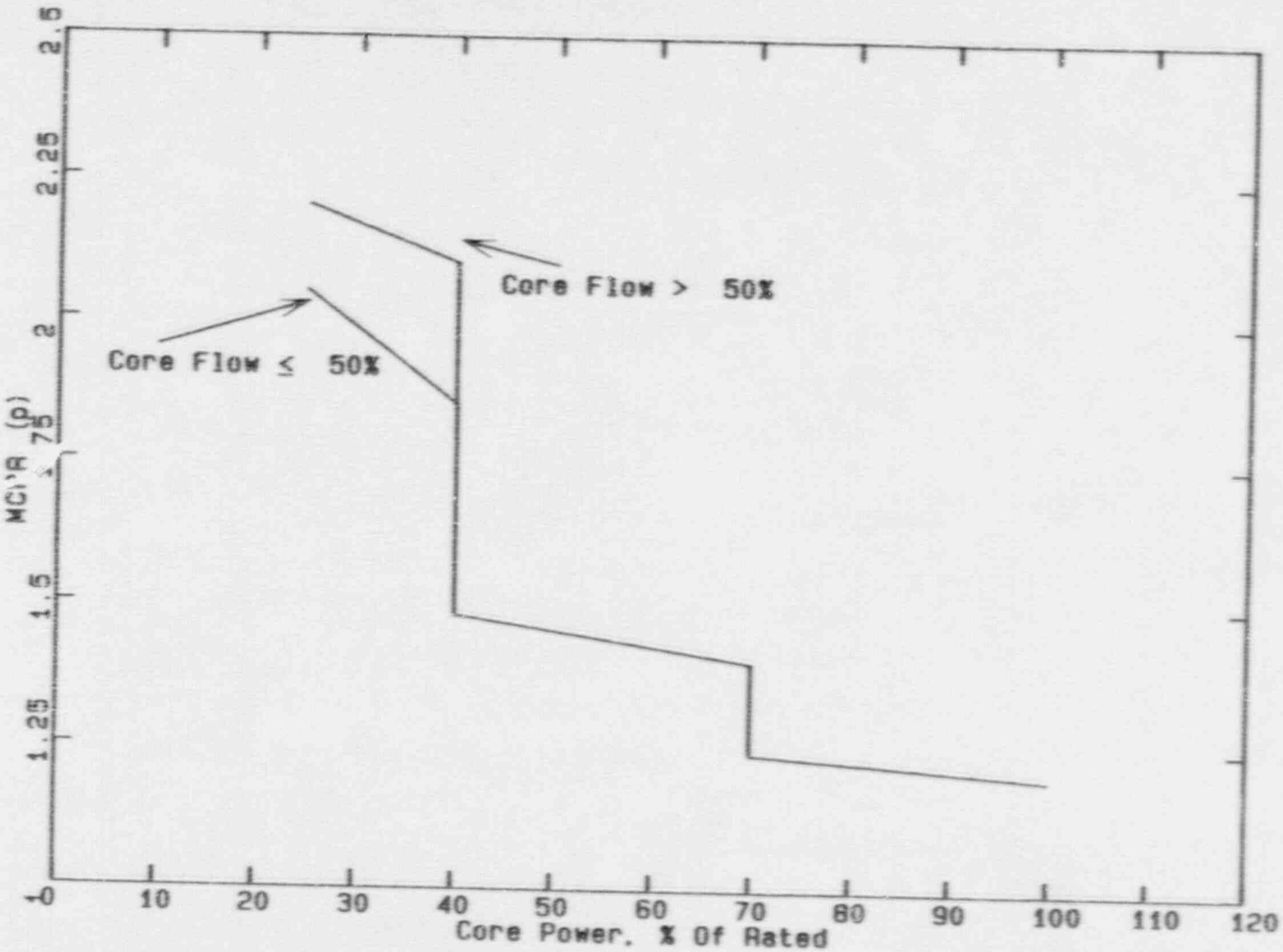

Figure 5.2 Power Dependent MCPR Limits For Grand Gulf Unit 1 Cycle 5

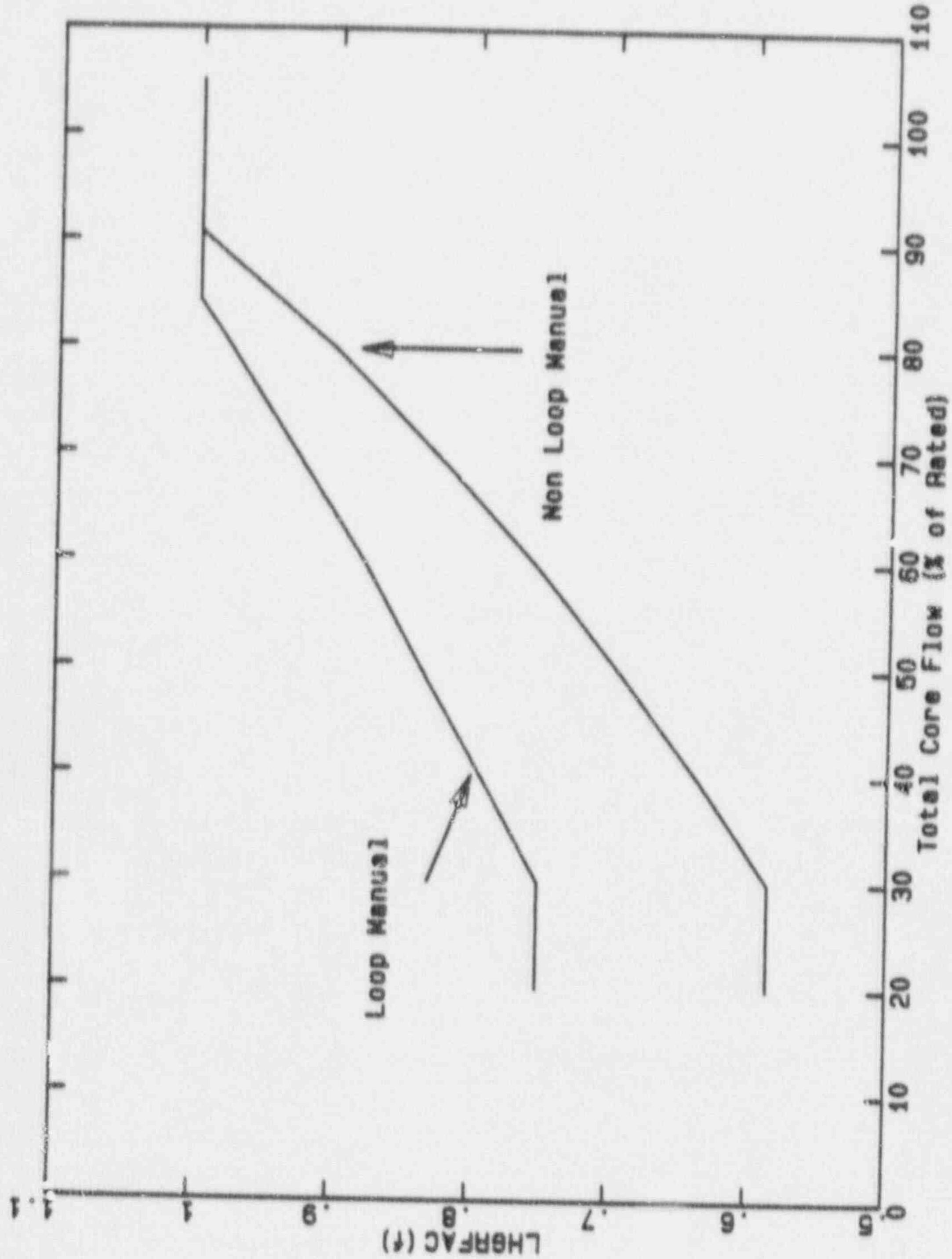

Figure 5.3 Flow Dependent LHGRFAC Value for Grand Gulf Unit 1 Cycle 5

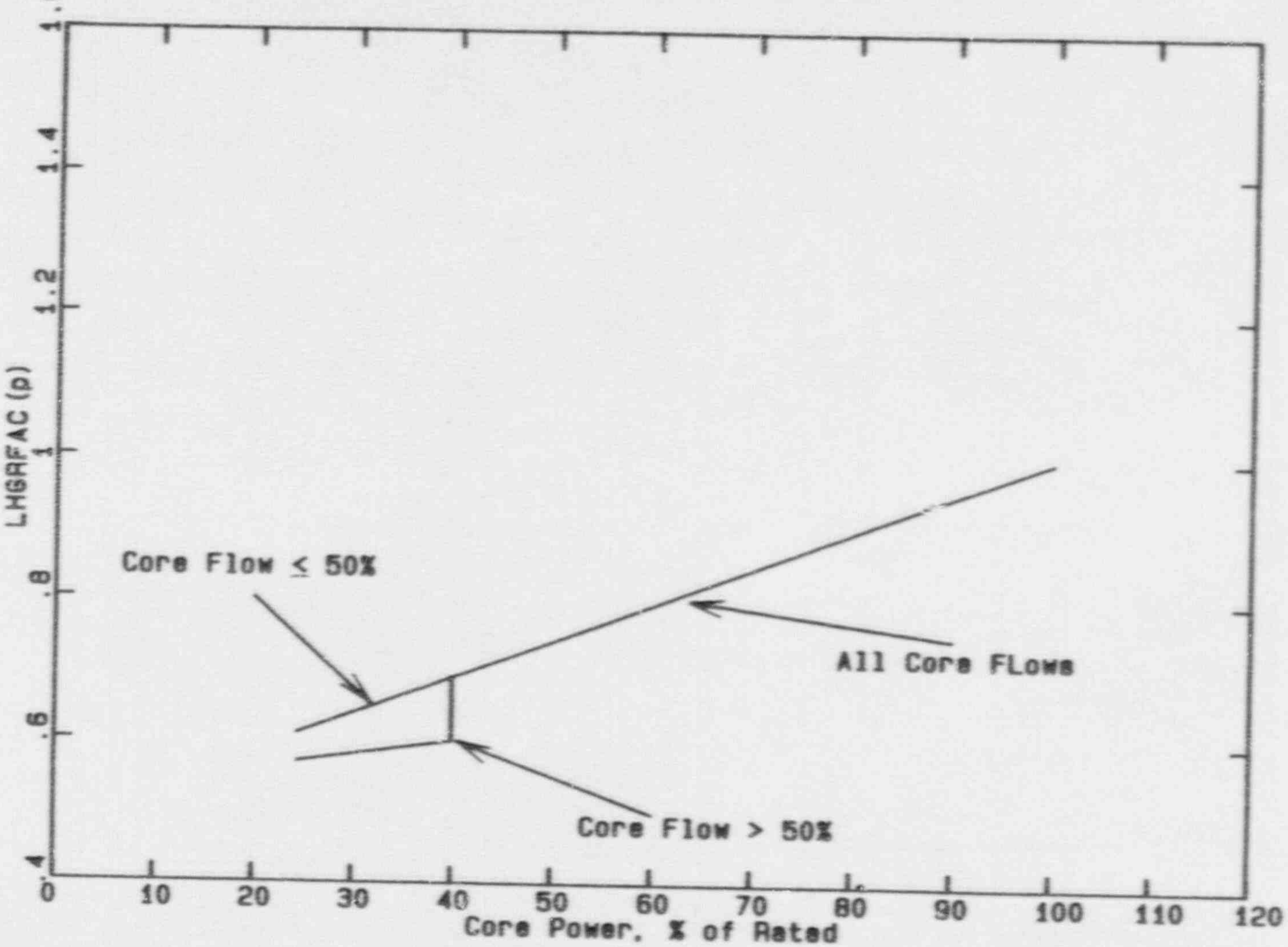

Figure 5.4 Power Dependent LHGRFAC Value for Grand Gulf Unit 1 Cycle 5

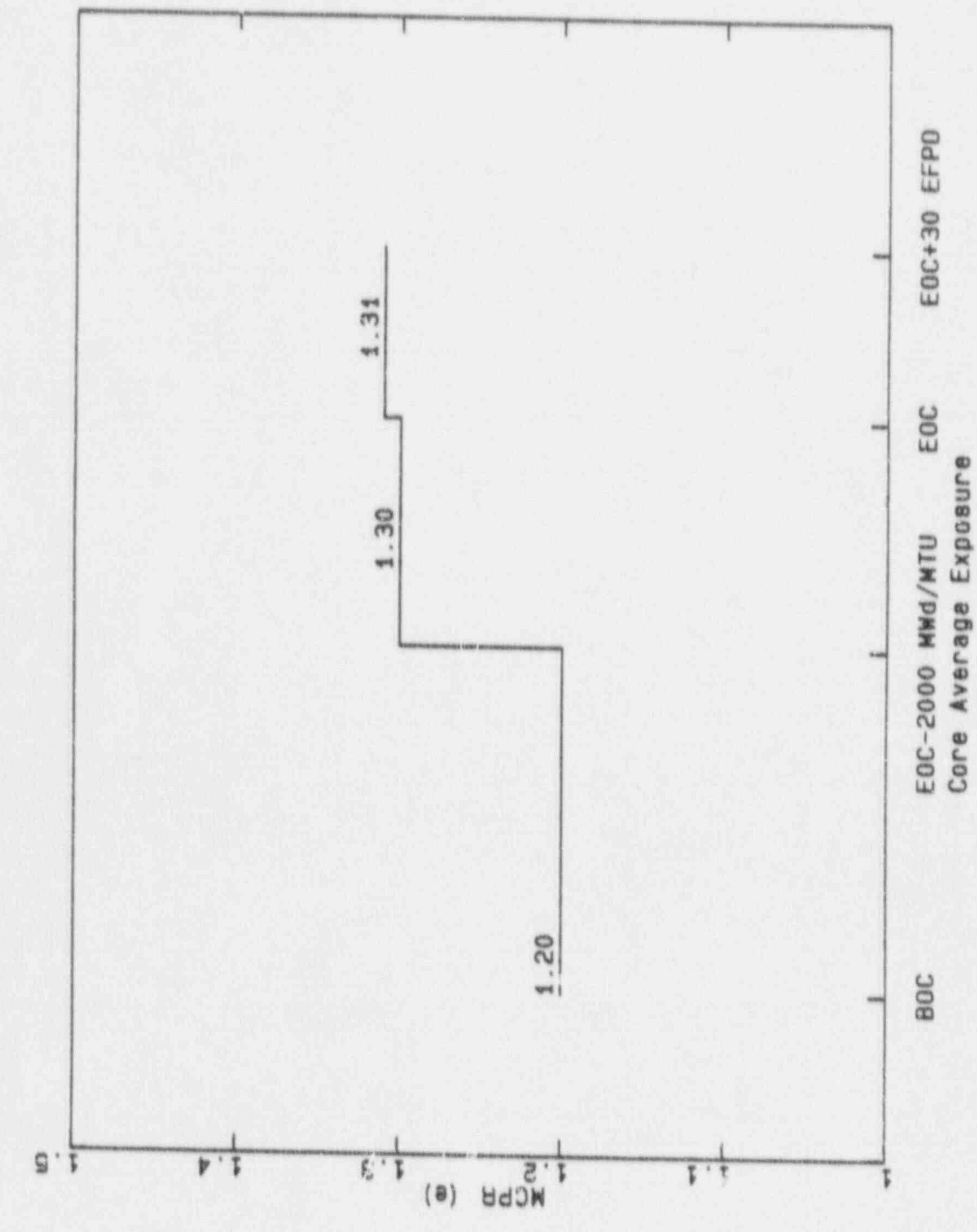

Figure 5.5 Exposure Dependent MCPR Limits For Grand Gulf Unit 1 Cycle 5

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 ANF 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 are used for the ANF 9x9-2 fuel design. 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	1696	0.4

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 0.80 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	8.8 mk
	Doppler Coefficient	-10.4×10^{-6} $\Delta K/K/^{\circ}F$
	Effective Delayed Neutron Fraction	5.47×10^{-3}
	Four-Bundle Local Peaking Factor	1.439
	Maximum Deposited Fuel Rod Enthalpy	192 cal/g

The Control Rod Drop Accident analysis is unaffected by the lowering of the BPWS operability requirement from 20% power to 10% power.

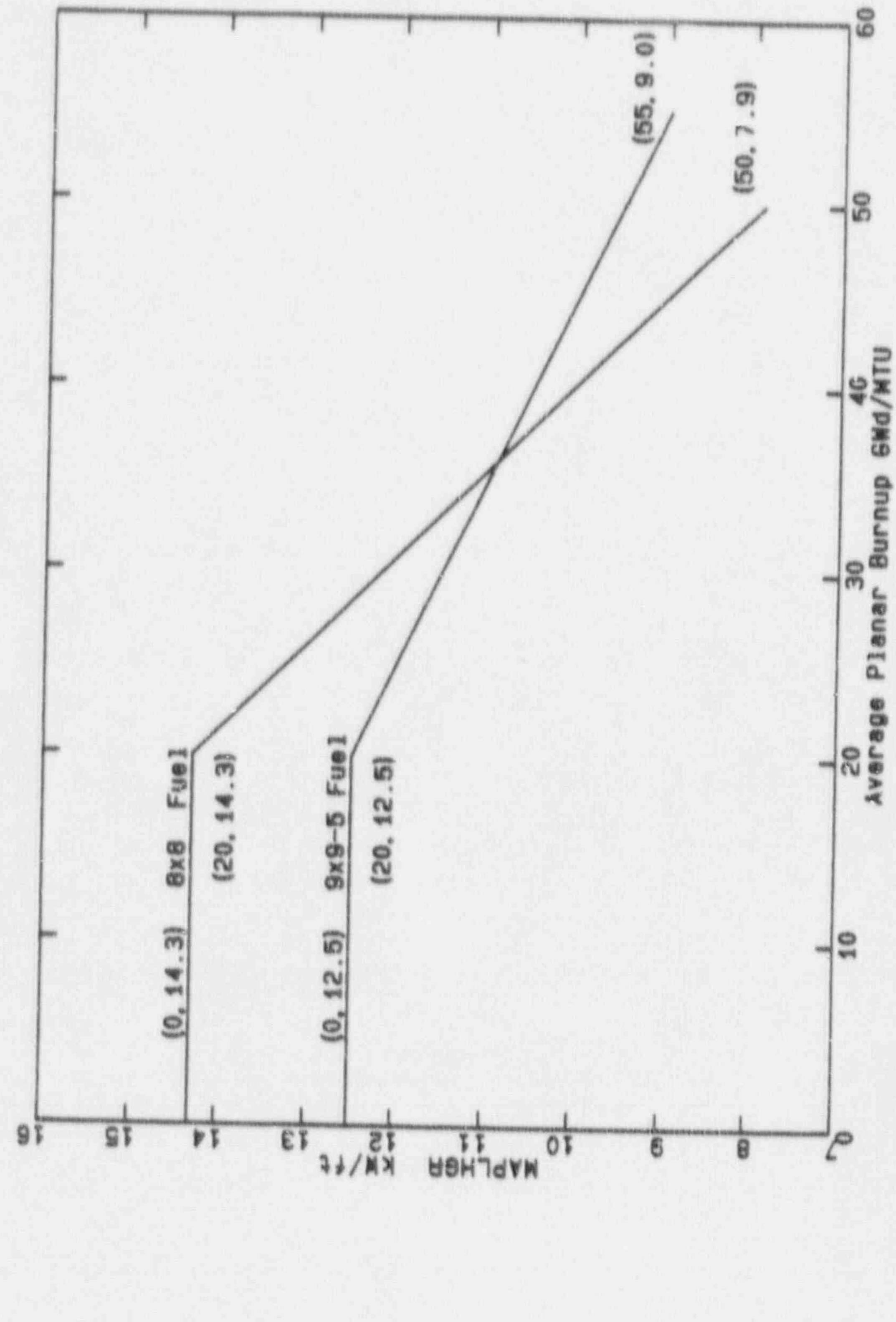

Figure 5.1 MAPLHGR vs Average Planar Exposure for ANF 8x8 and ANF 9x9-5 Reload Fuel

7.0 TECHNICAL SPECIFICATIONS

7.1 Limiting Safety System Settings

7.1.1 MCPR Fuel Cladding Integrity Safety Limit

Safety Limit MCPR	1.09*

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 ANF Fuel

The following MAPLHGR limits are consistent with 10CFR50.46 requirements. Unlike previous cycles, the MAPLHGR limit is not used to protect the design basis LHGR limits for the fuel types co-resident in Cycle 5.

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

For single-loop operation, a reduction factor of 0.8 is applied to the above two-loop MAPLHGR limits.

*The 1.09 safety limit accounts for channel bow and 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 ANF Fuel

The LHGR limits for Grand Gulf 1 as previously analyzed remain applicable for ANF 8x8 fuel during Cycle 5 operation. These limits are extended to cover the exposure range for Cycle 5. These limits, which are based on Figure 3.1 of Reference 3, are as follows:

Average Planar Exposure	LHGR
0.00 GWd/MTU	16.0 kW/ft
25.40	14.1
50.00	6.98

The LHGR limits for 9x9-5 fuel, based on Figure 3.1 of Reference 13 for ANF reload fuel during Cycle 5, 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 3/4, 4.1.1 (Technical Specification Amendment No. 62).

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, Supplement 1(P)(A) (April 1990).

Reference 8.10 is superseded by:

8.10 "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors," ANF-524(P), Revision 2, and Supplements, April 1989.

9.0 REFERENCES

1. Letter, Lester L. Kintner (USNRC) to O. D. Kingsley, Jr. (MP&L), "Technical Specification Changes to Allow Operation with One Recirculation Loop and Extended Operating Domain," August 15, 1986.

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 5 Plant Transient Analysis," ANF-90-021, Revision 2, Advanced Nuclear Fuels Corporation, Richland, WA, August 1990.

5. "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analysis," ANF-913, Volume 1, Supplements 1, 2, and 3.

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.

7. "Generic LOCA Break Spectrum Analysis for BWR/6 Plants," XN-NF-86-37(P), Exxon Nuclear Company, Richland, WA, April 1986.

8. "Grand Gulf Unit 1 LOCA Analysis," XN-NF-86-38, Exxon Nuclear Company, Richland, WA, June 1986.

9. "Grand Gulf Unit 1 Cycle 3 Reload Analysis," ANF-87-67, Revision 1, Advanced Nuclear Fuels Corp., Richland, WA, August 1987.

10. "Grand Gulf Unit 1 Reload ANF-1.4, Cycle 5 Mechanical, Thermal Hydraulic, and Neutronic Design for Advanced Nuclear Fuels 9x9-5 Fuel Assemblies," ANF-89-171(P) Volumes 1 and 2, Advanced Nuclear Fuels Corporation, Richland, WA, January 1990.

11. "Grand Gulf Nuclear Station Unit 1 Revised Flow Dependent Thermal Limits," NESDQ-88-003, MSU System Services Inc., November 1988.

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), Amendment 1, September 1989, Advanced Nuclear Fuels Corporation, Richland, WA.

14. Letter, R. A. Copeland (ANF) to Director, NRR (NRC), "Submittal of MICROBURN-B," dated March 8, 1989 (RAC:0[illegible]:90).

15. Letter, R. A. Copeland (ANF) to Lambros Lois (NRC), "TIP Asymmetry Uncertainty," dated July 20, 1990 (RAC:083:90).

APPENDIX A SEISMIC/LOCA-ANF 9x9-5

The acceptability for Grand Gulf Unit 1 of the ANF 9x9-5 fuel seismic-LOCA performance is qualified by its similarity to the GE 8x8 fuel originally licensed to operate in Grand Gulf Unit 1. The 9x9-5 fuel will exhibit essentially the same static and dynamic response as the GE 8x8 since it has essentially the same dynamic and hydraulic characteristics as identified below and is subjected to the same dynamic excitation.

The dynamic input to the reload fuel will be the same as that for the existing fuel since it will be installed at the same location and there are no significant changes which would affect the overall response of the reactor pressure vessel (RPV) and its pedestal. The dynamic response of the assemblies is dependent on the mass and stiffness properties of the fuel elements which determine their natural frequencies.

Table A.1 presents, for comparison, fuel assembly properties for the GE 8x8, ANF 8x8, and ANF 9x9 fuel. Based on the data presented, the important dynamic characteristics for the various fuel bundles are similar.

The channeled fuel assembly dynamic response is primarily a function of the channel. Because channels of a similar design are used for both the 8x8 and 9x9 fuel, then, the in-reactor dynamic characteristics of the channeled fuel assembly for these fuel types would essentially be identical. This is confirmed in the analysis documented in the Susquehanna Unit 2 Cycle 2 reload analysis, XN-NF-86-60 Appendix B, where the seismic-LOCA performance of the 8x8 and 9x9 assemblies are compared. The ANF analysis reported in XN-NF-81-51(P)(A), "LOCA-Seismic Structural Response of an Exxon Nuclear Company BWR Jet Pump Fuel Assembly," dated May 1986, used a channel allowable stress of 24,000 psi at 545°F. The NRC has concluded that the ANF value of 24,000 psi is conservative relative to the GE channel faulted allowable stress. The Cartech channel uses the same material as GE and has a limiting faulted allowable stress

(1.2 x σyp) greater than 28,360 psi at 545 °F. Thus, design margin exists when either GE or Cartech channels are used.

The pressure drop of different fuel designs can be compared from calculations performed for typical full core loadings of the respective designs at the rated conditions of flow and power. The results can be considered in terms of overall pressure drop and in terms of fuel assembly drop. The overall pressure drop considers the pressure drop from the orifice inlet to the top of the upper tie plate while the fuel assembly pressure drop subtracts out the orifice pressure drop. The results of the typical BWR-6 analysis show that the ANF 8x8 and the ANF 9x9-5 fuel designs have lower pressure drops than the comparable GE 8x8 fuel design.

In comparing overall pressure drops, the ANF 8x8 fuel shows an 8% lower pressure drop than the GE 8x8 fuel. The ANF 9x9-5 fuel shows a 2% lower pressure drop than the GE 8x8 fuel.

Focusing on the fuel assembly pressure drop, the ANF 8x8 fuel shows a 12% lower pressure drop than the GE 8x8 fuel while the ANF 9x9-5 fuel shows a 3% lower pressure drop than the GE 8x8 fuel.

In summary, the 9x9-5 dynamic and hydraulic characteristics are essentially the same as those of the fuel it replaces. Therefore, the results of previous analyses are applicable to the 9x9-5.

Table A.1 Fuel Assembly Properties

Property	GE8x8*	9x9	8x8	9x9-5
Active Fuel Length (in)	150.0	150.0	150.0	150.0
Fuel Rod OD (in)	0.483	0.424	0.484	0.443/0.417
Pellet OD (in)	0.410	0.3565	0.405	0.375/0.353
Fuel Rod Pitch (in)	0.636	0.563	0.636	0.563
Spacer Pitch (in)	20.15	20.15	20.15	20.15
Number of Water Rods	2	2	2	5
Fuel Assembly Weight (lb)	600	574	585	583
Channel Length (in)	167.4	167.4	167.4	137.4
Channel Wall Thickness (in)	0.120	0.120	0.120	0.120
Channel Weight (lb)	98.7	98.7	98.7	98.7
Channel Minimum Inside Envelope (in)	5.205	5.205	5.205	5.205

*Estimated value.

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