Standard No.: GGNS-MS-48.0
Revision: 1
Date: 11-12-93

Grand Gulf Nuclear Station
Core Operating Limits Report
Safety-Related

GRAND GULF NUCLEAR STATION **NUCLEAR PLANT ENGINEERING**

REVIEW AND APPROVAL SHEET

STANDARD NO.: GGNS-MS-48.0 REVISION: 1

STANDARD TITLE: Core Operating Limits Report

This document specifies items related to nuclear safety YES [X] NO [ ]

Signatures certify that the above standard was originated, verified, reviewed or waived and approved as noted below:

ORIGINATED BY: G. E. Broadbent DATE: 11/10/93

VERIFIED BY: Ken L. Walker DATE: 11/11/93

REVIEWED BY: M. D. Withrow DATE: 11/12/93
Cognizant Group Supervisor

NPE SECTION	REVIEWED BY	REVIEW WAIVED BY	DATE
ELECTRICAL		[illegible signature]	11/12/93
CIVIL		[illegible signature]	11/12/93
MECHANICAL	M. D. Withrow		11/12/93

ANII: N/A DATE:
(Insert N/A if not applicable)

APPROVED BY: M. D. Withrow for DGB DATE: 11/12/93
Responsible Manager

SAFETY EVALUATION APPLICABILITY REVIEW FORM

A) Document Evaluated: Standard GGNS-MS-48.0, Revision 1

B) Description of the Proposed Change: Per GNRI-93-008, Amendment 106 to the Grand Gulf Operating License, Entergy committed to removing certain reactor physics parameters from the Technical Specifications and placing them in a separate report prepared for each fuel cycle. Standard GGNS-MS-48.0 is the Core Operating Limits Report (COLR) and establishes these parameters. Revision 1 to GGNS-MS-48.0 reports the Cycle 7 core operating limits.

PRE-SCREENING

Check the applicable boxes below. If any of the boxes are checked, neither a safety evaluation applicability review nor a safety evaluation is necessary and steps C, D, E, and F may be skipped. The preparer and reviewer must sign at the bottom of the form.

__ The change is editorial only.

__ 10CFR50.54 applies to the change instead of 10CFR50.59.

X An approved safety evaluation covering all aspects of this subject already exists.
Reference SE# The Cycle 7 core operating limits have been evaluated in the Cycle 7 reload safety evaluation, 93-0100-R00.

__ The change, in its entirety, has been approved by the NRC.
Reference: ________________

__ The change is an FSAR change that meets the exclusion criteria outlined in Site Directive G4.803

Safety Evaluation Applicability Review

If any of the following questions are answered "yes", then a full 50.59 Safety Evaluation must be completed.

C) Does the proposed change or activity represent a change to the Technical Specifications?

YES __ Explain:
NO __

D) Does the proposed change or activity represent:

(1) A change to the facility which alters, or has the potential to alter, the information, operation, function or ability to perform the function of a system, structure or component described in the SAR?

YES __ Explain:
NO __

(2) A change to a procedure which alters, or has the potential to alter, a procedure described, outlined or summarized in the SAR?

YES __ Explain:
NO __

(3) A test or experiment not described in the SAR or which requires that a system be operated in an abnormal manner that is not described or previously analyzed in the SAR?

YES __ Explain:
NO __

PREPARER G. E. Broadbent — Eng. II — 11/10/93
Name — Job Title — Date

REVIEWER Ken L. Walker — Sr. Engr. — 11/11/93
Name — Job Title — Date

If the preparer performed an applicability review, the reviewer should check below to indicate by which means the independent review reached the same conclusions.

✓ Reviewed the applicability review documentation.

__ Completed an independent applicability review.

__ Performed a verbal review with the preparer.

REVISION STATUS SHEET

STANDARD REVISION SUMMARY

REVISION	ISSUE DATE	DESCRIPTION
0	April 1, 1993	Issued for use
1	11-12-93	Issued for use

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PAGE REVISION STATUS

PAGE NO.	REVISION	PAGE NO.	REVISION
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APPENDIX/ATTACHMENT STATUS

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1.0 PURPOSE

On October 4, 1988, the NRC issued Generic Letter 88-16 (Reference 28) encouraging licensees to remove cycle-specific parameter limits from Technical Specifications and to place these limits in a formal report to be prepared by the licensee. As long as the parameter limits were developed with NRC-approved methodologies, the letter indicated that this would remove unnecessary burdens on licensee and NRC resources.

On October 29, 1992, Entergy Operations complied with this letter by submission of a Proposed Amendment to the Grand Gulf Operating License (Reference 29). This document requested changes to the GGNS Technical Specifications to remove certain reactor physics parameter limits that change each fuel cycle. This amendment committed to placing these operating limits in a separate Core Operating Limits Report (COLR) which will be defined in Technical Specifications. This PCOL was approved by the NRC by SER dated January 21, 1993 (Reference 30).

The development of this COLR is the responsibility of Design Engineering. The purpose of Standard GGNS-MS-48.0 is to develop the Core Operating Limits Report from the available supporting documents for each fuel cycle. This standard will be revised accordingly for each fuel cycle or remaining portion of a fuel cycle.

2.0 SCOPE

As defined in Technical Specification 1.7a, the COLR is the GGNS document that provides the core operating limits for the current fuel cycle. This document is prepared in accordance with Technical Specification 6.9.1.11 for each reload cycle using NRC-approved analytical methods.

The Cycle 7 core operating limits included in this report are:

1. the Average Planar Linear Heat Generation Rate (APLHGR) limits for each fuel type for both two-loop and single-loop operation. (Technical Specification 3.2.1),

2. the Minimum Critical Power Ratio (MCPR) operating limit including the power (as a function of exposure) and flow dependent curves. (Technical Specification 3.2.3), and

3. the Linear Heat Generation Rate (LHGR) limit for each fuel type including the power and flow dependent parametric adjustment factor curves, $LHGRFAC_p$ and $LHGRFAC_f$, respectively. (Technical Specification 3.2.4)

The cycle-specific MCPR safety limits are documented in Technical Specification 2.1.2.

3.0 REFERENCES

This section contains the methodology and cycle-specific references used in the safety analysis of Grand Gulf Cycle 7. The supplements and revisions of the current analytical methodology references are included below in accordance with Technical Specification 6.9.1.11, Core Operating Limits Report.

METHODOLOGY REFERENCES:

1.) XN-NF-79-71(P), Revision 2 including Supplements 1, 2, and 3, Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, Exxon Nuclear Company, Inc., Richland, WA, November 1981. Approved by NRC letter dated October 24, 1986.

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

3.) XN-NF-80-19(P)(A), Volume 1 Supplements 3 and 4, Advanced Nuclear Fuels Methodology for Boiling Water Reactors: Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology, Advanced Nuclear Fuels Corporation, Inc., Richland, WA, November 1990.

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

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

6.) ANF-1125 (P)(A) and Supplements 1 and 2, ANFB Critical Power Correlation, Advanced Nuclear Fuels Corporation, Richland, WA, April 1990.

7.) XN-NF-84-105(P)(A), Volume 1 and Supplements 1 and 2, XCOBRA-T: A Computer Code for BWR Transient Thermal Hydraulic Core Analysis, Exxon Nuclear Company, Inc., Richland, WA, February 1987.

8.) XN-NF-573(P), RAMPEX Pellet-Clad Interaction Evaluation Code for Power Ramps, Exxon Nuclear Company, Inc., Richland, WA, May 1982. Approved by NRC letter dated August 28, 1990.

9.) XN-NF-81-58(P)(A) and Supplements 1 and 2, Revision 2, RODEX2: Fuel Rod Thermal-Mechanical Response Evaluation Model, Exxon Nuclear Company, Inc., Richland, WA, March 1984.

10.) XN-NF-85-74(P)(A), RODEX2A (BWR): Fuel Rod Thermal-Mechanical Response Evaluation Model, Exxon Nuclear Company, Inc., Richland, WA, August 1986.

11.) XN-CC-33(P)(A), Revision 1, HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option, Exxon Nuclear Company, Inc., Richland, WA, November 1975.

12.) XN-NF-825(P)(A) Supplement 2, BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$ for Plant Operation Within the Extended Operating Domain, Exxon Nuclear Company, Inc., Richland, WA, October 1986.

13.) XN-NF-81-51(P)(A), LOCA-Seismic Structural Response of an Exxon Nuclear Company BWR Jet Pump Fuel Assembly, Advanced Nuclear Fuels Corporation, Richland, WA, May 1986.

14.) XN-NF-84-97(P)(A), LOCA-Seismic Structural Response of an ENC 9x9 BWR Jet Pump Fuel Assembly, Exxon Nuclear Company, Inc., Richland, WA, August 1986.

15.) XN-NF-86-37(P), Generic LOCA Break Spectrum Analysis for BWR/6 Plants, Exxon Nuclear Company, Inc., Richland, WA, April 1986. Approved by NRC letter dated October 24, 1986.

16.) XN-NF-82-07(P)(A), Revision 1, Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model, Exxon Nuclear Company, Inc., Richland, WA, November 1982.

17.) XN-NF-80-19(A), Volumes 2, 2A, 2B, and 2C, Exxon Nuclear Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model, Exxon Nuclear Company, Inc., Richland, WA, September 1982.

18.) XN-NF-79-59(P)(A), Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies, Exxon Nuclear Company, Inc., Richland, WA, November 1983.

19.) ANF-1358(P)(A), Revision 1 and Correspondence, The Loss of Feedwater Heating in Boiling Water Reactors, Siemens Power Corporation, Richland, WA, September 1992.

CURRENT CYCLE REFERENCES:

20.) EMF-93-050, Grand Gulf Unit 1 Cycle 7 Plant Transient Analysis, Siemens Power Corporation, Richland, WA, June 1993.

21.) EMF-93-051, Grand Gulf Unit 1 Cycle 7 Reload Analysis, Siemens Power Corporation, Richland, WA, June 1993.

22.) ANF-92-190(P), Grand Gulf 1 ANF-1.6 Design Report, Mechanical, Thermal-Hydraulic and Neutronic Design for Advanced Nuclear Fuels 9X9-5 Fuel Assemblies, Advanced Nuclear Fuels, Richland, WA, December 1992.

23.) ANF-86-133, Revision 4, Principal ECCS and Plant Transient Analysis Parameters Grand Gulf Unit 1, Advanced Nuclear Fuels Corporation, Richland, WA, June 1991.

24.) EMF-91-172, Grand Gulf Unit 1 LOCA Analysis for Single Loop Operation, Siemens Power Corporation, Richland, WA, October 1991.

25.) ANF-88-152(P)(A) with Amendment 1 and Supplement 1, Generic Mechanical Design for Advanced Nuclear Fuels 9X9-5 BWR Reload Fuel, Advanced Nuclear Fuels Corporation, Richland, WA, November 1990.

CYCLE 6 REFERENCES:

26.) ANF-91-080(P), Grand Gulf 1 ANF-1.5 Design Report, Mechanical, Thermal-Hydraulic and Neutronic Design for Advanced Nuclear Fuels 9X9-5 Fuel Assemblies, Advanced Nuclear Fuels, Richland, WA, July 1991.

CYCLE 5 REFERENCES:

27.) ANF-89-171(P), Volumes 1 and 2, Grand Gulf 1 ANF-1.4 Design Report, Mechanical, Thermal-Hydraulic and Neutronic Design for Advanced Nuclear Fuels 9X9-5 Fuel Assemblies, Advanced Nuclear Fuels, Richland, WA, January 1990.

GENERAL REFERENCES:

28.) MAEC-88/0313, Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications", October 4, 1988.

29.) GNRO-92-00093, Proposed Amendment to Grand Gulf Operating License, PCOL-92/07, dated October 29, 1992.

30.) GNRI-93-0008, Amendment 106 to Grand Gulf Operating License, January 21, 1993.

4.0 DEFINITIONS

1. Average Planar Linear Heat Generation Rate (APLHGR) - the APLHGR shall be applicable to a specific planar height and is equal to the sum of the linear heat generation rates for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle. (Technical Specification 1.3)

2. Average Planar Exposure - the Average Planar Exposure shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle. (Technical Specification 1.2)

3. Critical Power Ratio (CPR) - the ratio of that power in the assembly which is calculated by application of the ANFB boiling correlation (Reference 6) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power. (Technical Specification 1.8)

4. Core Operating Limits Report (COLR) - The Grand Gulf Nuclear Station specific document that provides core operating limits for the current reload cycle in accordance with Technical Specification 6.9.1.11. (Technical Specification 1.7a)

5. Linear Heat Generation Rate (LHGR) - the LHGR shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length. (Technical Specification 1.21)

6. Minimum Critical Power Ratio (MCPR) - the MCPR shall be the smallest CPR which exists in the core. (Technical Specification 1.25)

7. MCPR Safety Limit - the minimum value of the CPR at which the fuel could be operated with the expected number of rods in boiling transition not exceeding 0.1% of the fuel rods in the core. (Reference 20, Section 3.4)

5.0 GENERAL REQUIREMENTS

This section reports the Grand Gulf Cycle 7 core operating limits. These limits are taken from Reference 21 Sections 5.7, 6.1.3, and 7.2.3. As discussed in Technical Specifications 2.1.1 and 2.1.2, these operating limits are applicable when thermal power is greater than 25% of rated power.

Average Planar Linear Heat Generation Rates (Technical Specification 3.2.1)
During two-loop operation, all AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits in Figure 5.1.

During single-loop operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limits shown in Figure 5.1 multiplied by 0.86.

Minimum Critical Power Ratio (MCPR) (Technical Specification 3.2.3)
The MCPR shall be equal to or greater than the $MCPR_f$ and $MCPR_p$ limits at the indicated core flow and thermal power, for the exposure range, as shown in Figures 5.2, 5.3, 5.4, and 5.5.

Linear Heat Generation Rate (LHGR) (Technical Specification 3.2.4)
The LHGR shall not exceed the limits shown in Figure 5.6 as multiplied by the smaller of either the flow-dependent LHGR factor ($LHGRFAC_f$) of Figure 5.7 or the power-dependent LHGR factor ($LHGRFAC_p$) of Figure 5.8.

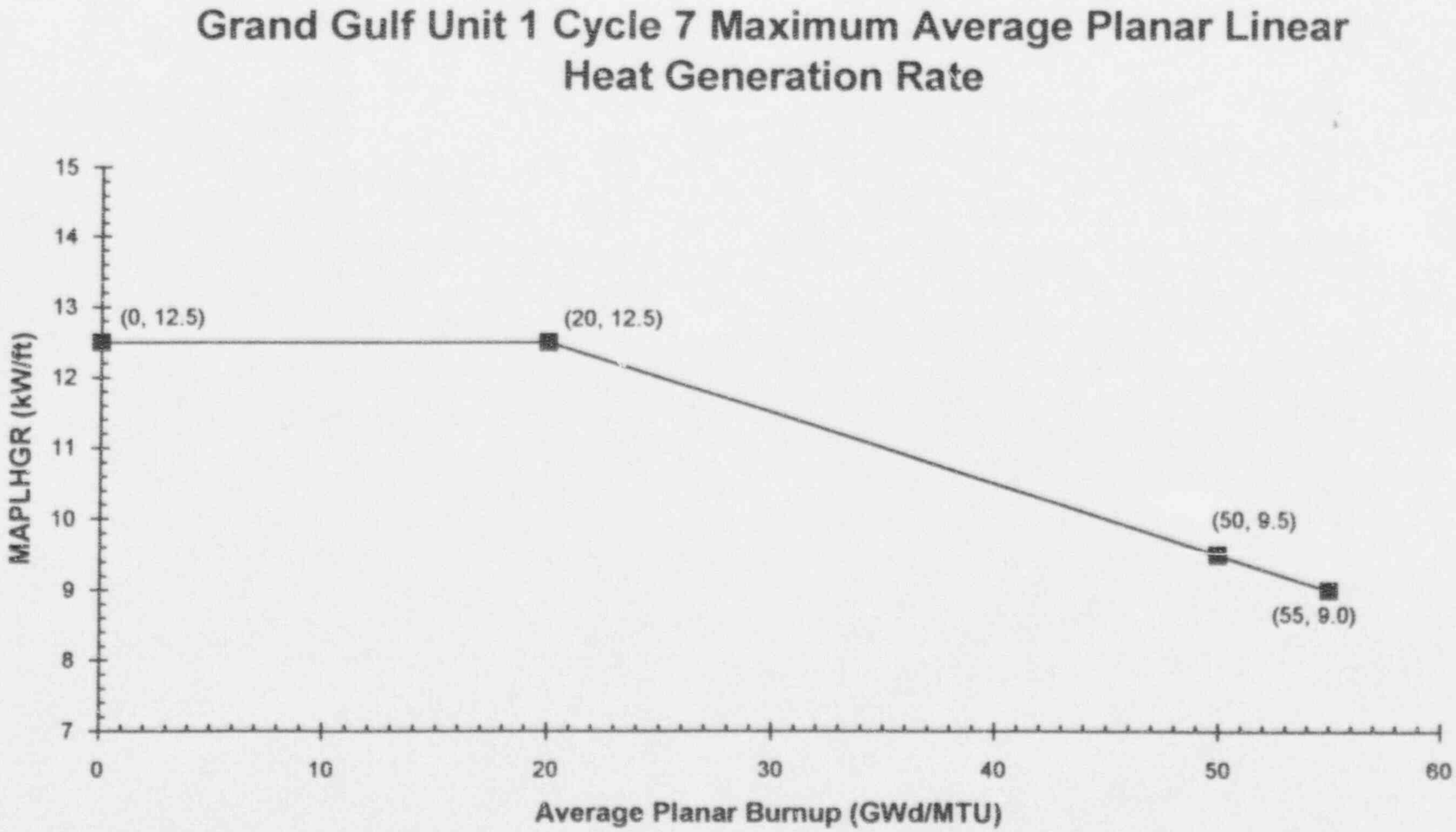

Figure 5.1 MAPLHGR Limits for Grand Gulf Unit 1 Cycle 7

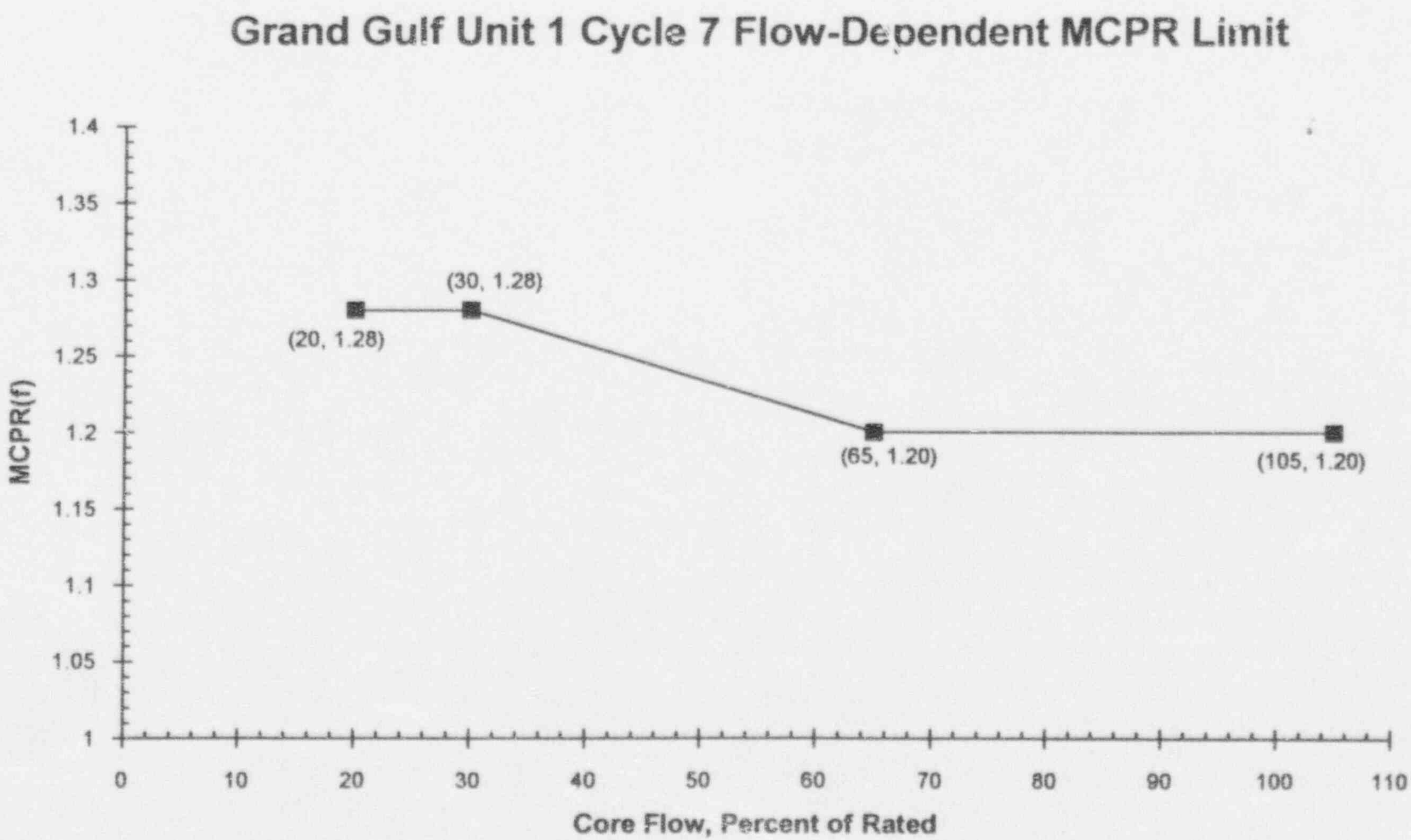

Figure 5.2 Flow Dependent MCPR Limits for Grand Gulf Unit 1 Cycle 7

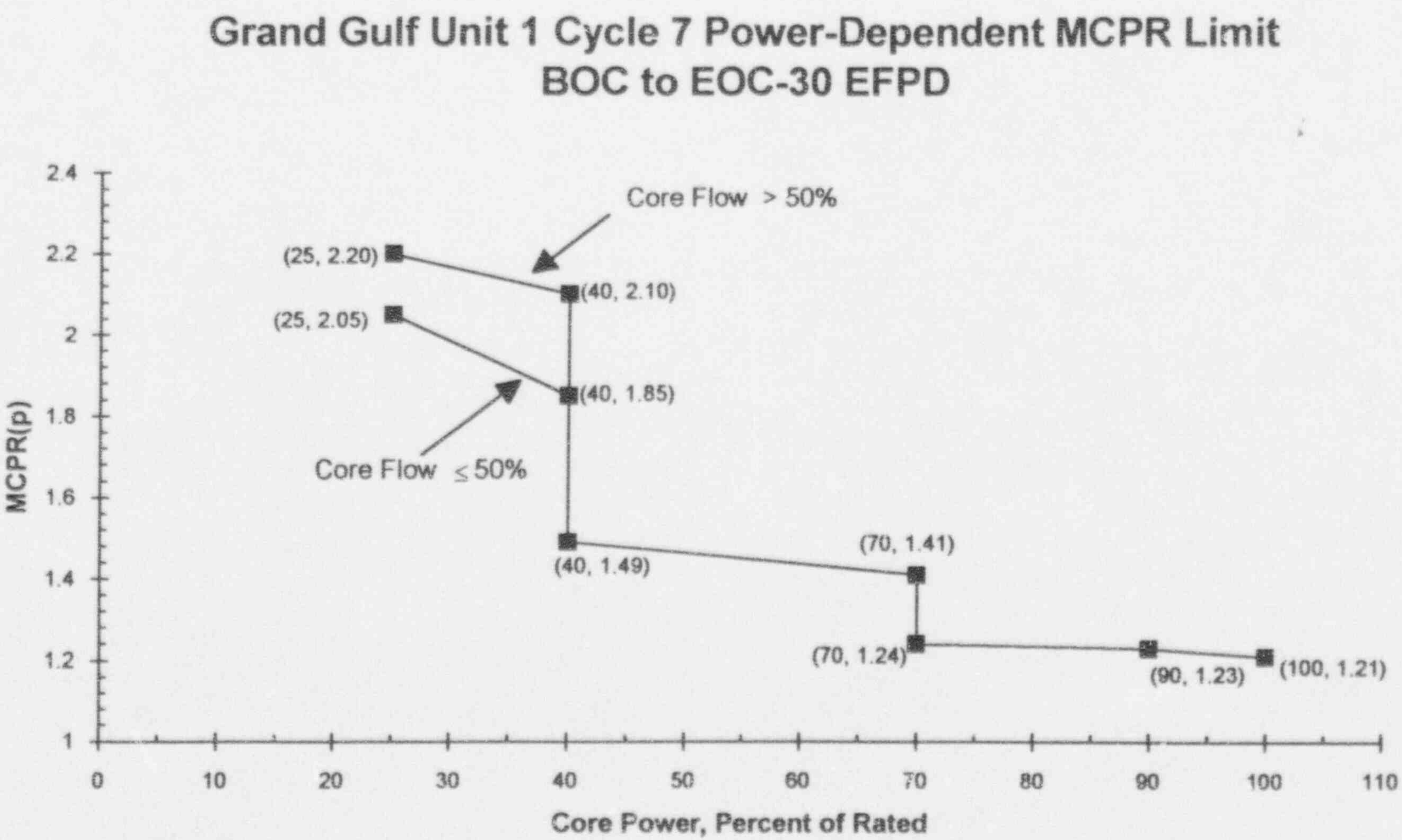

Figure 5.3 Power Dependent MCPR Limits for Grand Gulf Unit 1 Cycle 7 for Exposures From BOC to EOC-30 EFPD

Grand Gulf Unit 1 Cycle 7 Power-Dependent MCPR Limit
EOC-30 EFPD to EOC

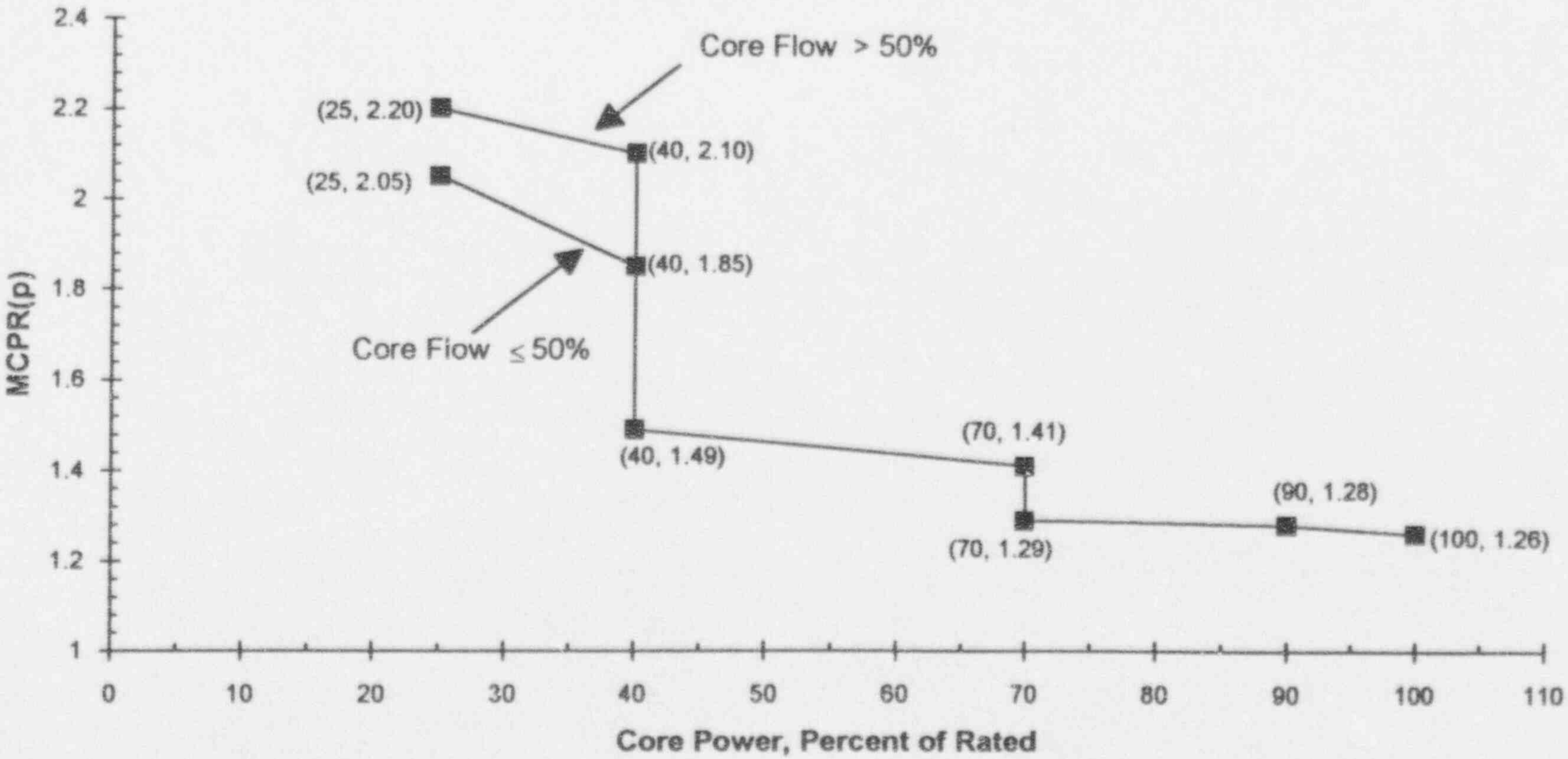

Figure 5.4 Power Dependent MCPR Limits for Grand Gulf Unit 1 Cycle 7 for Exposures From EOC-30 EFPD To EOC

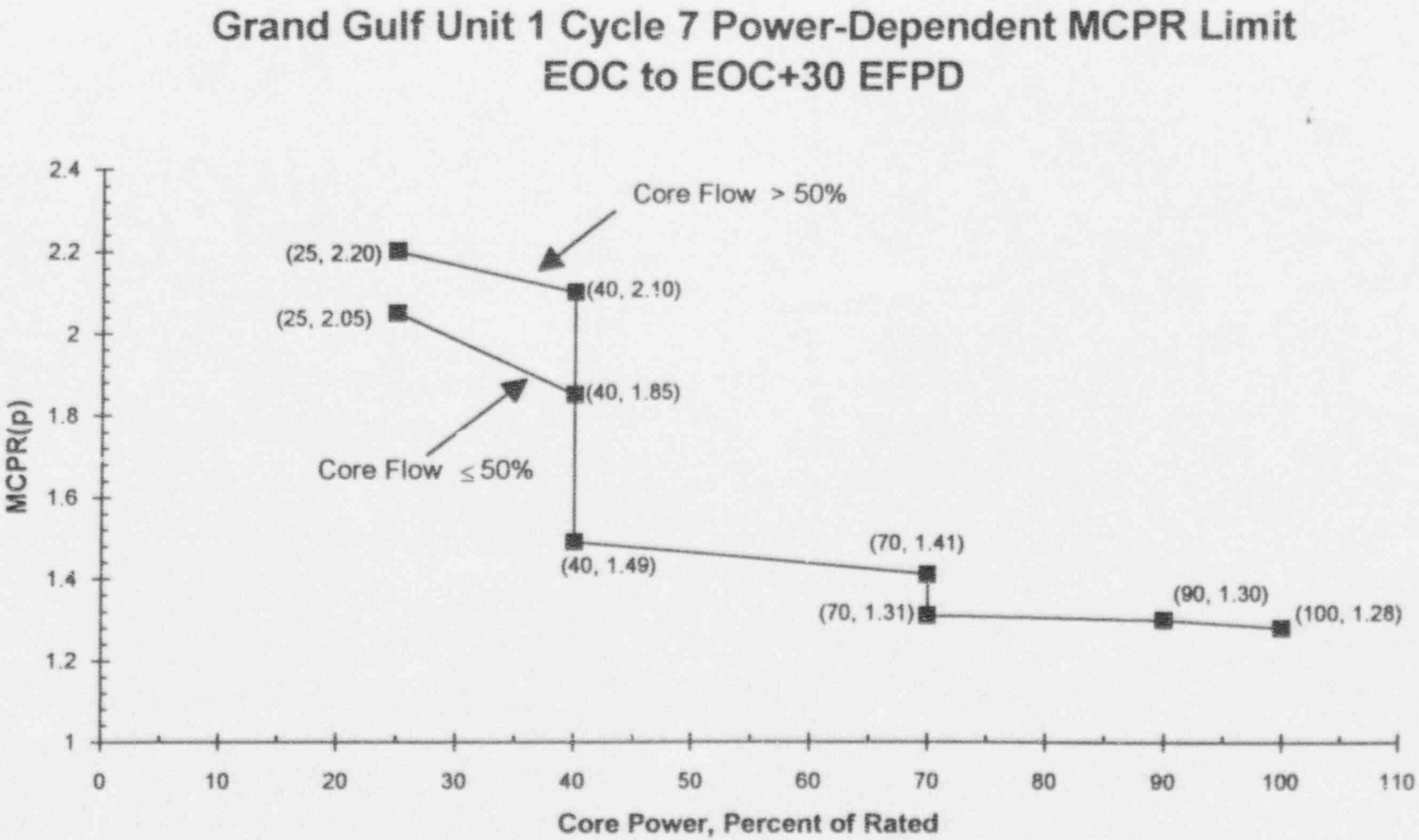

Figure 5.5 Power Dependent MCPR Limits for Grand Gulf Unit 1 Cycle 7 for Exposures From EOC To EOC+30 EFPD

Grand Gulf Unit 1 Cycle 7 Linear Heat Generation Rate

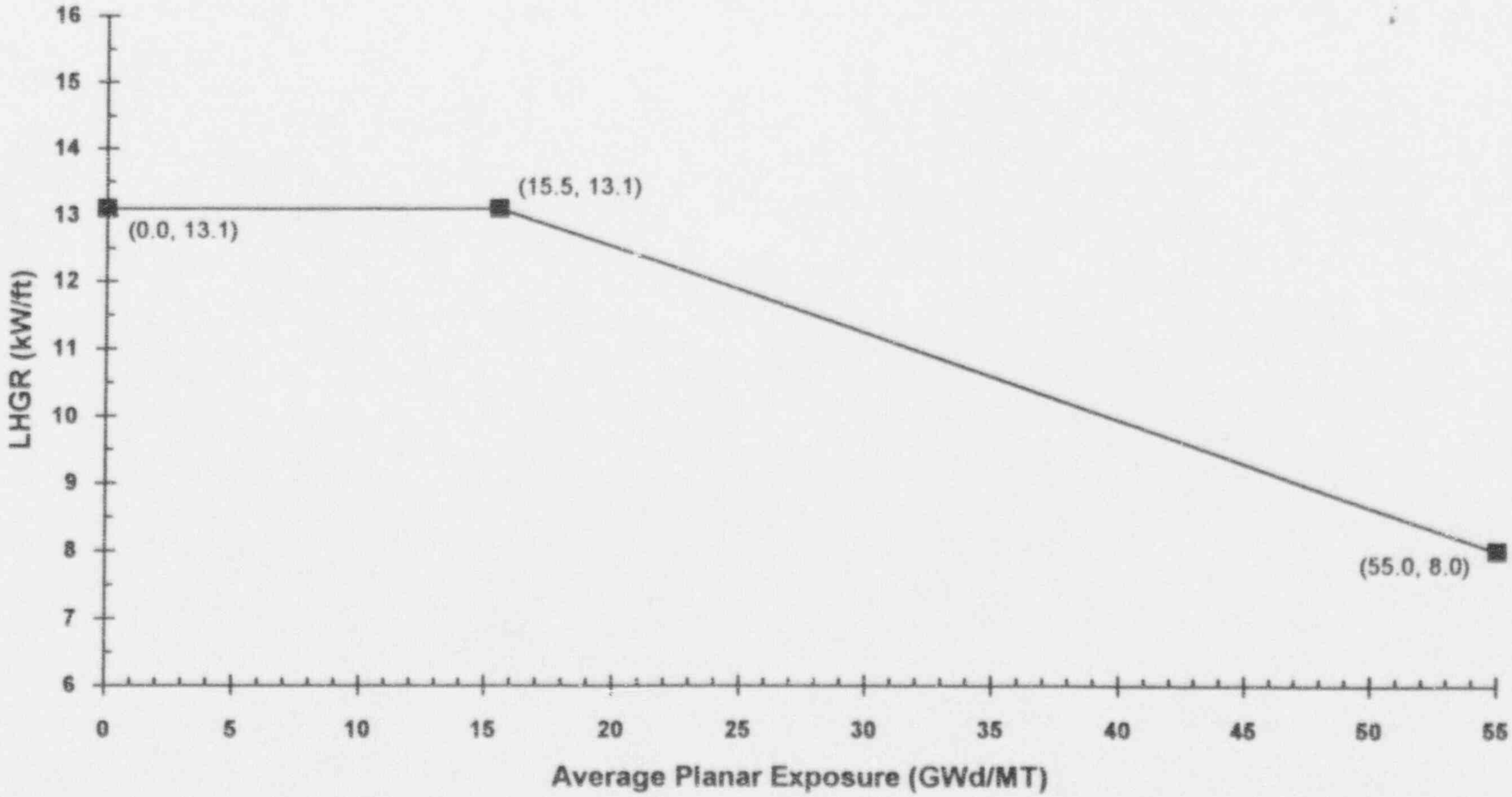

Figure 5.6 LHGR Limits for Grand Gulf Unit 1 Cycle 7

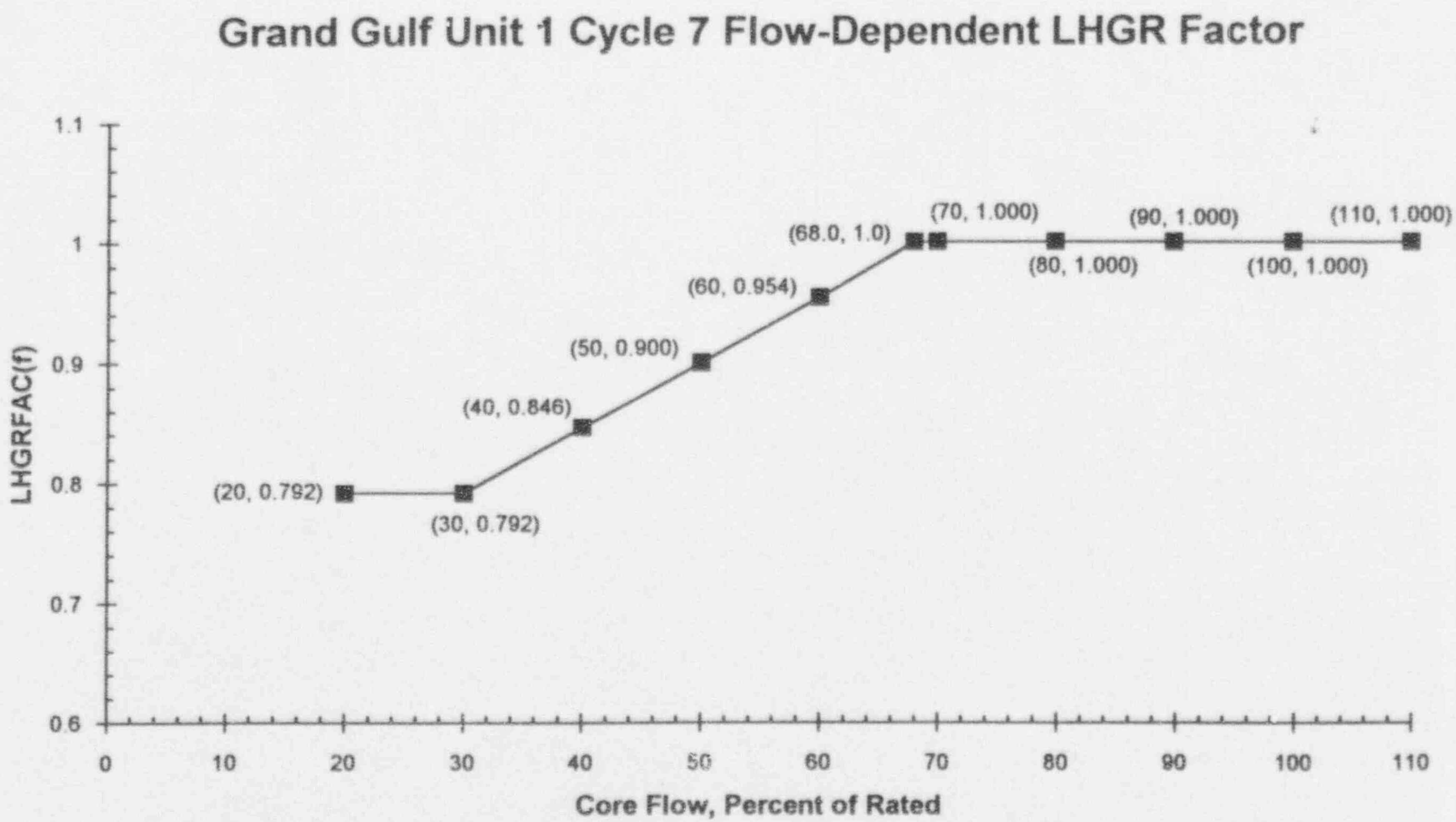

Figure 5.7 Flow Dependent LHGR Factors for Grand Gulf Unit 1 Cycle 7

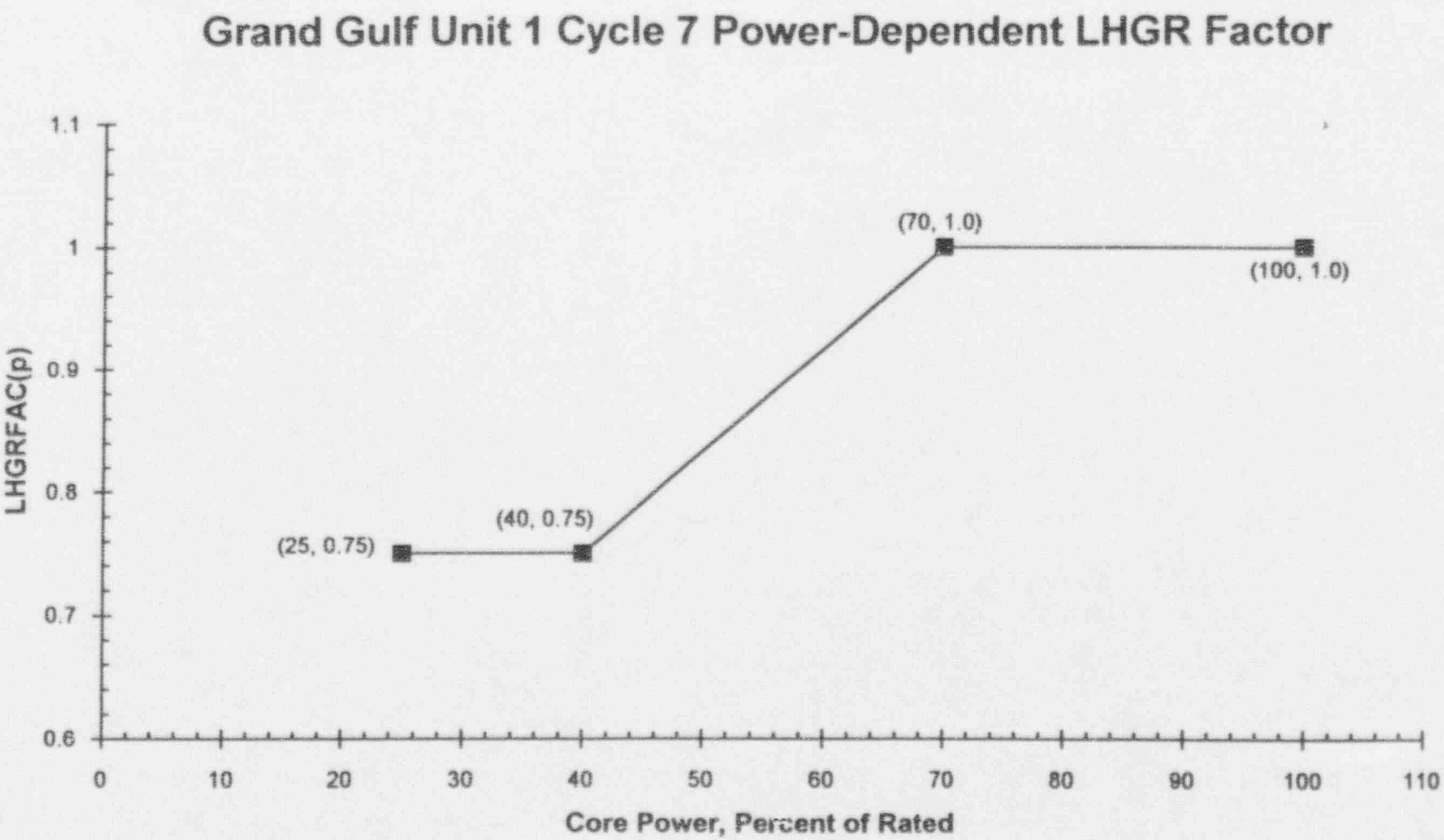

Figure 5.8 Power Dependent LHGR Factors for Grand Gulf Unit 1 Cycle 7

MARKED-UP IMPROVED TECHNICAL SPECIFICATIONS BASES PAGES
(ITS Submittal dated October 15, 1993)

CYCLE 7 RELOAD

Note: In this attachment, changes associated with this request are denoted by a circled "SE 93/100" adjacent to the marked-up change.

MINIMUM CRITICAL POWER RATIO (Continued)

was assumed to be constant. The generic flow control line is used to define several core power/flow states at which to perform steady-state core thermal-hydraulic evaluations.

Loop Manual mode of operation was analyzed. Consistent with the single failure/single operator error criterion, one loop runout was postulated for Loop Manual operation. The maximum core flow at loop runout was assumed to be 110% of rated flow. Peaking factors were selected such that the MCPR for the bundle with the least margin of safety would not decrease below the MCPR Safety Limit.

The $MCPR_p$ is established to protect the core from plant transients other than core flow increase including the localized rod withdrawal error event. The $MCPR_p$ limits are established for a set of exposure intervals. The limiting transients are analyzed at the limiting exposure for each interval. Core power dependent setpoints are incorporated (incremental control rod withdrawal limits) in the Rod Withdrawal Limiter (RWL) System Specification (3.3.6). These setpoints allow greater control rod withdrawal at lower core powers where core thermal margins are large. However, the increased rod withdrawal requires higher initial MCPR's to assure the MCPR safety limit Specification (2.1.2) is not violated. The analyses that establish the power dependent MCPR requirements that support the RWL system are presented in Reference 4. For core power below 40% of RATED THERMAL POWER, where the EOC-RPT and the reactor scrams on turbine stop valve closure and turbine control valve fast closure are bypassed, separate sets of $MCPR_p$ limits are provided for high and low core flows to account for the significant sensitivity to initial core flows. For core power above 40% of RATED THERMAL POWER, bounding power-dependent MCPR limits were developed. The abnormal operating transients analyzed for single loop operation are discussed in Reference 5 and the appropriate cycle-specific documents. No change to the MCPR operating limit is required for single loop operation.

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin.