May 28, 1992

Docket No. 50-416

Mr. William T. Cottle
Vice President, Operations GGNS
Entergy Operations, Inc.
Post Office Box 756
Port Gibson, Mississippi 39150

Dear Mr. Cottle:

SUBJECT: ISSUANCE OF AMENDMENT NO. 99 TO FACILITY OPERATING LICENSE NO. NPF-29 - GRAND GULF NUCLEAR STATION, UNIT 1 (TAC NO. M82280)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 99 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment revises the Technical Specifications (TS) in response to your application dated December 5, 1991.

The amendment revises a) the Safety Limit Maximum Critical Power Ratio (MCPR) values for Two-Loop Operation and Single-Loop Operation (SLO), b) the SLO Maximum Average Planar Heat Generation Rate (MAPLHGR) multiplier, c) the flow-dependent MCPR operating limits, d) the power-dependent MCPR operating limits, e) the exposure-dependent MCPR operating limits, f) Linear Heat Generation Rate (LHGR) limits for 8X8 fuel types for average planar exposures beyond 40,000 MWd/MTU, and g) the flow-dependent and power-dependent LHGR multipliers.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by:

Paul W. O'Connor, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 99 to NPF-29
2. Safety Evaluation

cc w/enclosures:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 99 to NPF-29
2. Safety Evaluation

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See next page

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Grand Gulf Nuclear Station

Entergy Operations, Inc.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

ENTERGY OPERATIONS, INC.

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

MISSISSIPPI POWER AND LIGHT COMPANY

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 99
License No. NPF-29

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated December 5, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-29 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 99 , are hereby incorporated into this license. Entergy Operations, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John T Larkins

John T. Larkins, Director
Project Directorate IV-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 28, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 99

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE PAGES	INSERT PAGES
2-1	2-1
B 2-1	B 2-1
B 2-1a	B 2-1a
B 2-2	B 2-2
3/4 1-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-6a	3/4 2-6a
3/4 2-7a	3/4 2-7a
3/4 2-7b	3/4 2-7b
3/4 2-7c	3/4 2-7c
B 3/4 2-1	B 3/4 2-1

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.06 during both two loop operation and 1.07 during single loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than the above limits and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

BASES

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit for the MCPR. MCPR greater than the applicable Safety Limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The Siemens Nuclear Power Corporation (SNP) ANFB critical power correlation is applicable to the SNP core. The applicable range of the ANFB correlation is for pressures above 585 psig and bundle mass flux greater than 0.25Mlbs/hr-ft^2. For low pressure and low flow conditions, a THERMAL POWER safety limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig and below 10% RATED CORE FLOW was justified for Grand Gulf cycle 1 operation based on ATLAS test data and the GEXL correlation. The use of the GEXL correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit was established by other means. This was done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi.

THERMAL POWER, Low Pressure or Low Flow (Continued)

Analyses show that with a bundle flow of 28 x 10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative. Overall, because of the design thermal-hydraulic compatibility of the SNP fuel designs with the cycle 1 fuel, this justification and the associated low pressure and low flow limits remain applicable for future cycles of cores containing these fuel designs.

With regard to the low flow range, the core bypass region will be flooded at any flow rate greater than 10% RATED CORE FLOW. With the bypass region flooded, the associated elevation head is sufficient to assure a bundle mass flux of greater than 0.25 Mlbs/hr-ft^2 for all fuel assemblies which can approach critical heat flux. Therefore, the ANFB critical power correlation is appropriate for flows greater than 10% RATED CORE FLOW.

The low pressure range for cycle 1 was defined at 785 psig. Since the ANFB correlation is applicable at any pressure greater than 585 psig, the cycle 1 low pressure boundary of 785 psig remains valid for the ANFB correlation.

2.1.2 THERMAL POWER, High Pressure and High Flow

The onset of transition boiling results in a decrease in heat transfer from the clad, elevated clad temperature, and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR), which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR).

The Safety Limit MCPR assures sufficient conservatism such that, in the event of a sustained steady state operation at the MCPR safety limit, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (MCPR = 1.00) and the Safety Limit MCPR is based on a detailed statistical procedure which considers the uncertainties in monitoring the core operating state and includes the effects associated with channel bow. One specific uncertainty included in the safety limit is the uncertainty inherent in the ANFB critical power correlation. SNP report ANF-524 (P)(A), Rev. 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors," April 1989, including supplements, describes the methodology used in determining the Safety Limit MCPR.

The ANFB critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power as evaluated by the correlation is within a small percentage of the actual critical power being estimated. The assumed reactor conditions used in defining the safety limit introduce conservatism into the limit because bounding radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. Still further conservatism is induced by the tendency of the ANFB correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the ANFB correlation provide assurance that during sustained operation at the Safety Limit MCPR there would be essentially no transition boiling in the core.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

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 shown in Figure 3.2.1-1.

During single loop operation, the APLHGR for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limit shown in Figure 3.2.1-1 multiplied by 0.86.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

During two loop operation or single loop operation, with an APLHGR exceeding the limits of Figure 3.2.1-1 as corrected by the appropriate multiplication factor, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the required limits:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

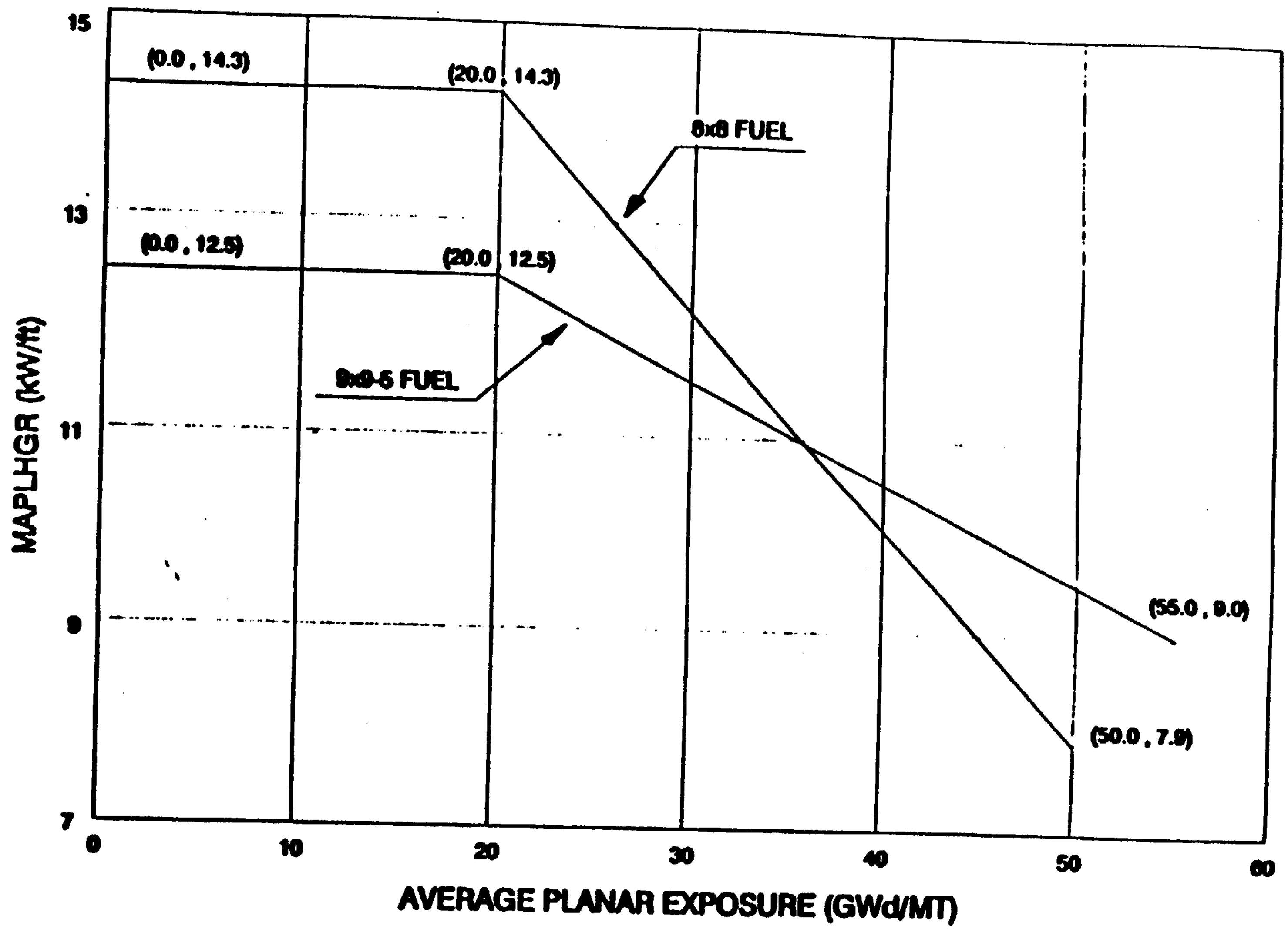

FIGURE 3.2.1-1 MAPLHGR vs AVERAGE PLANAR EXPOSURE

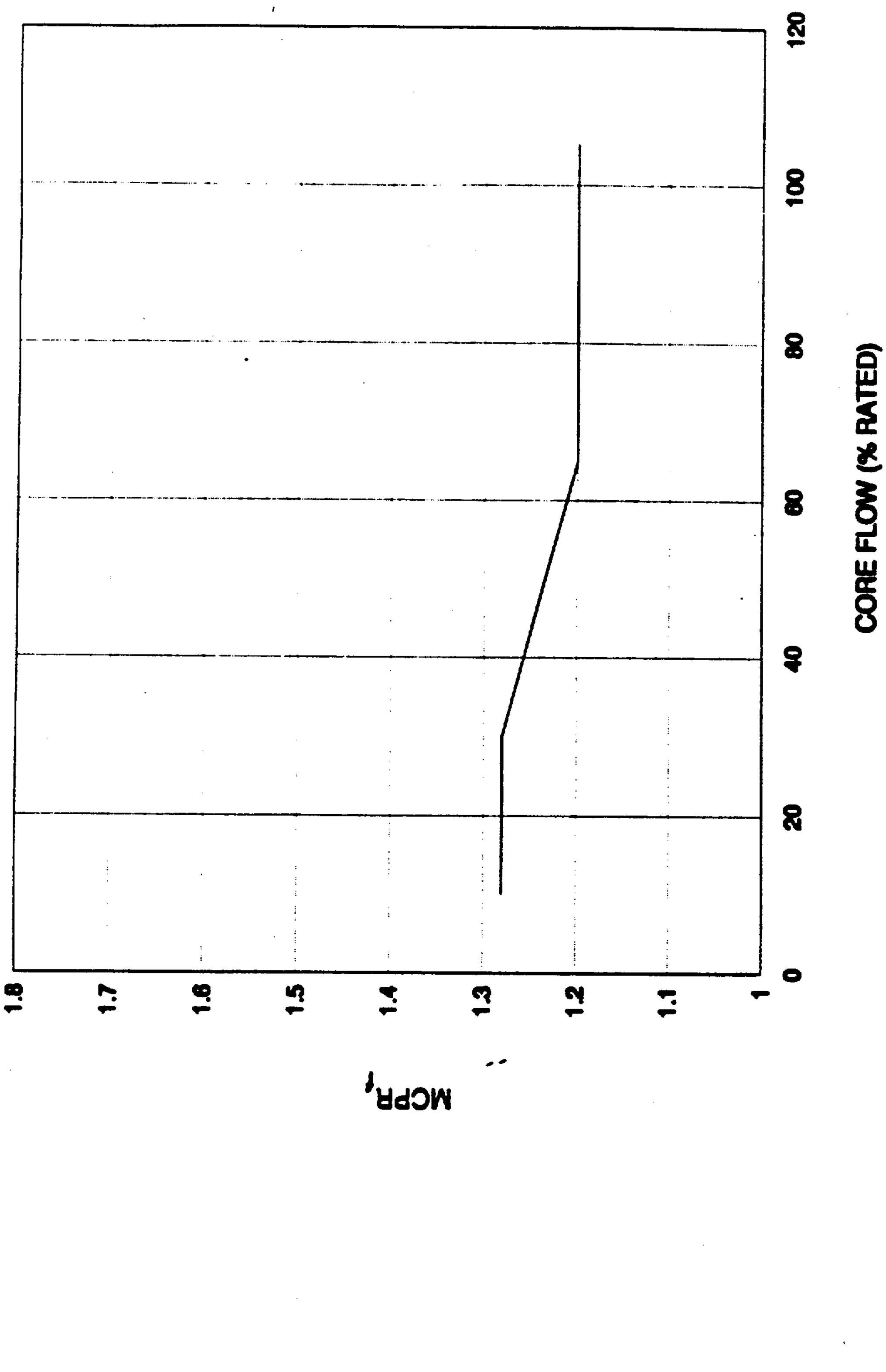

FIGURE 3.2.3-1 $MCPR_f$

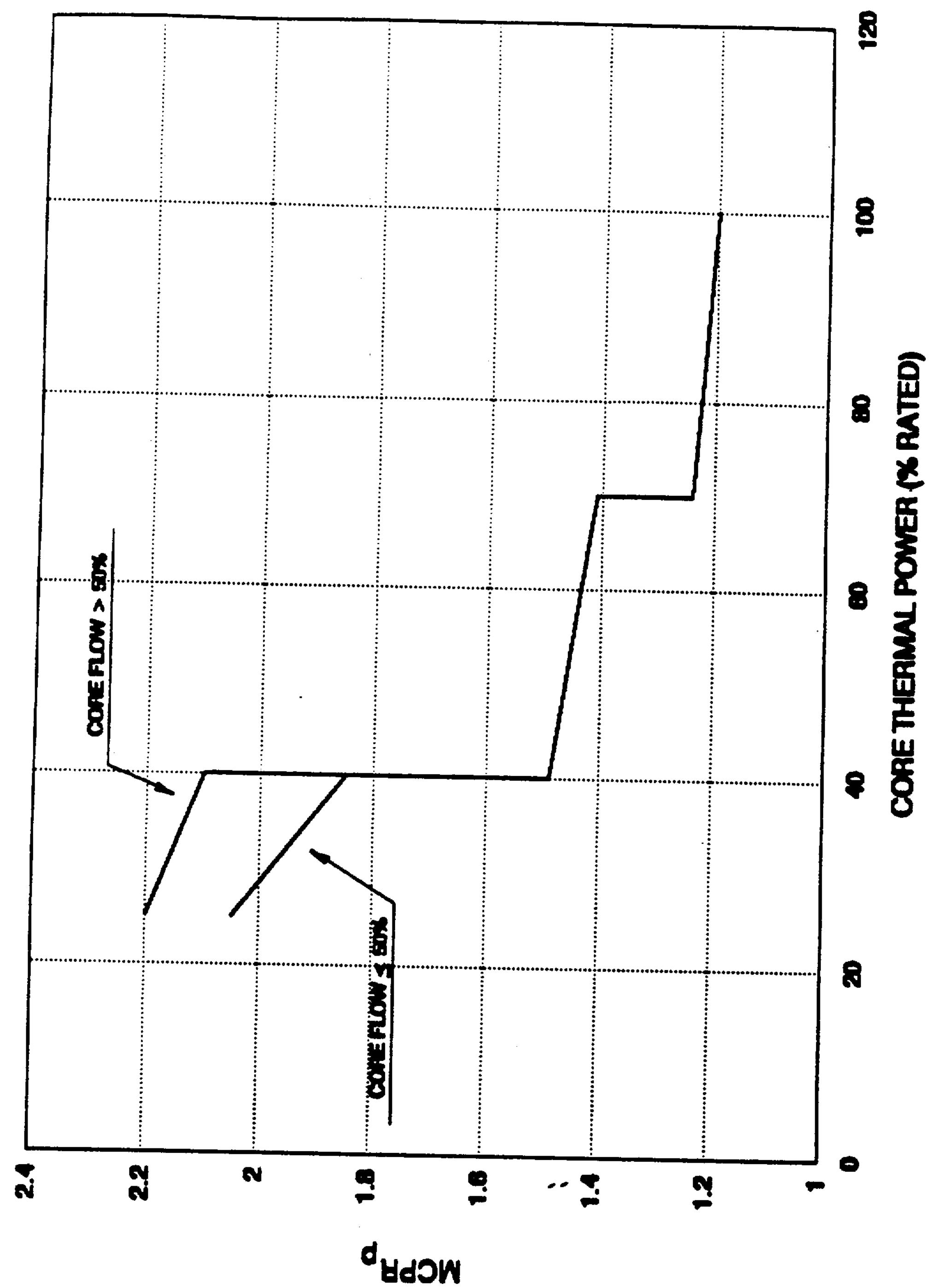

FIGURE 3.2.3-2 MCPR$_p$

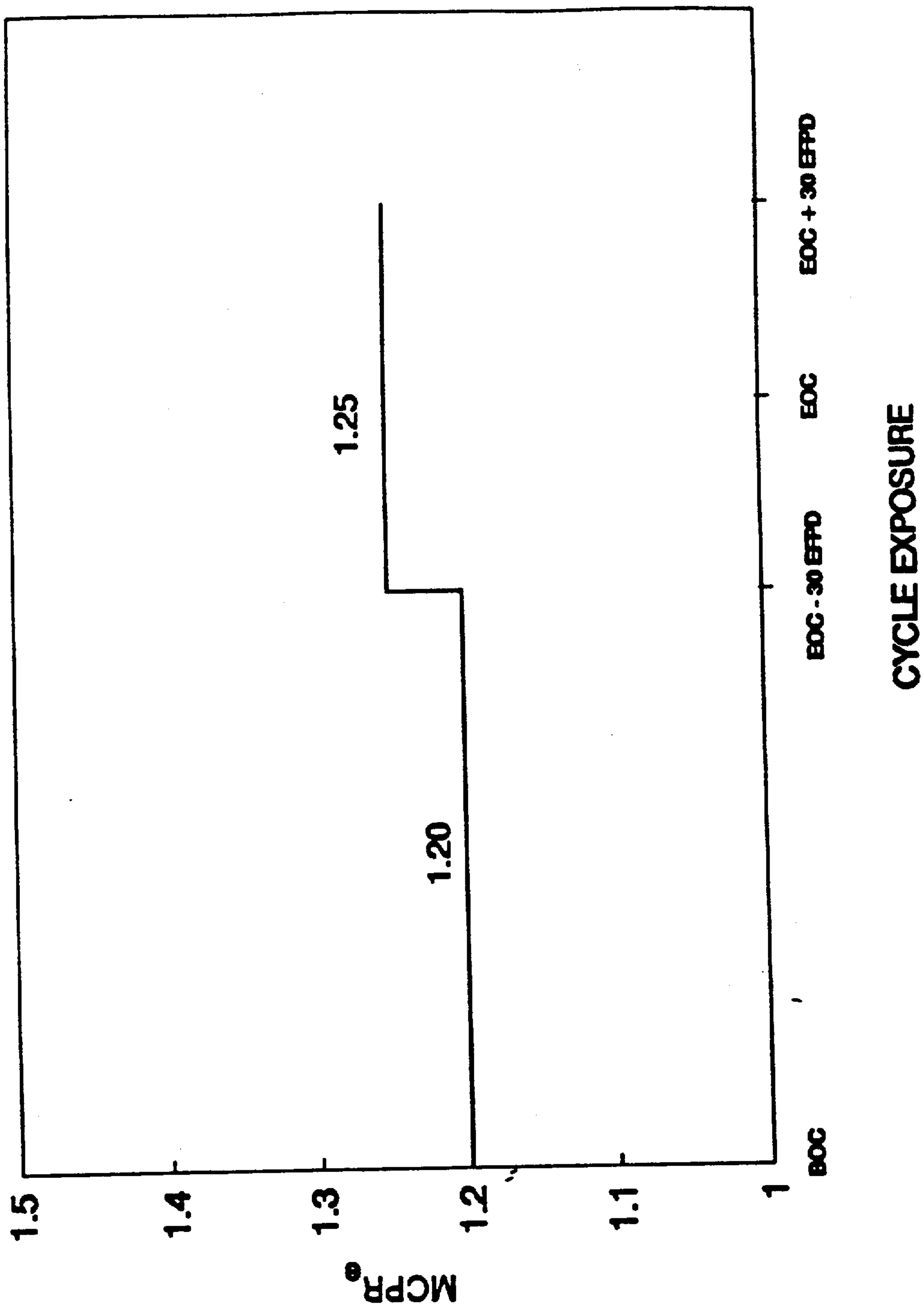

FIGURE 3.2.3-3 $MCPR_e$

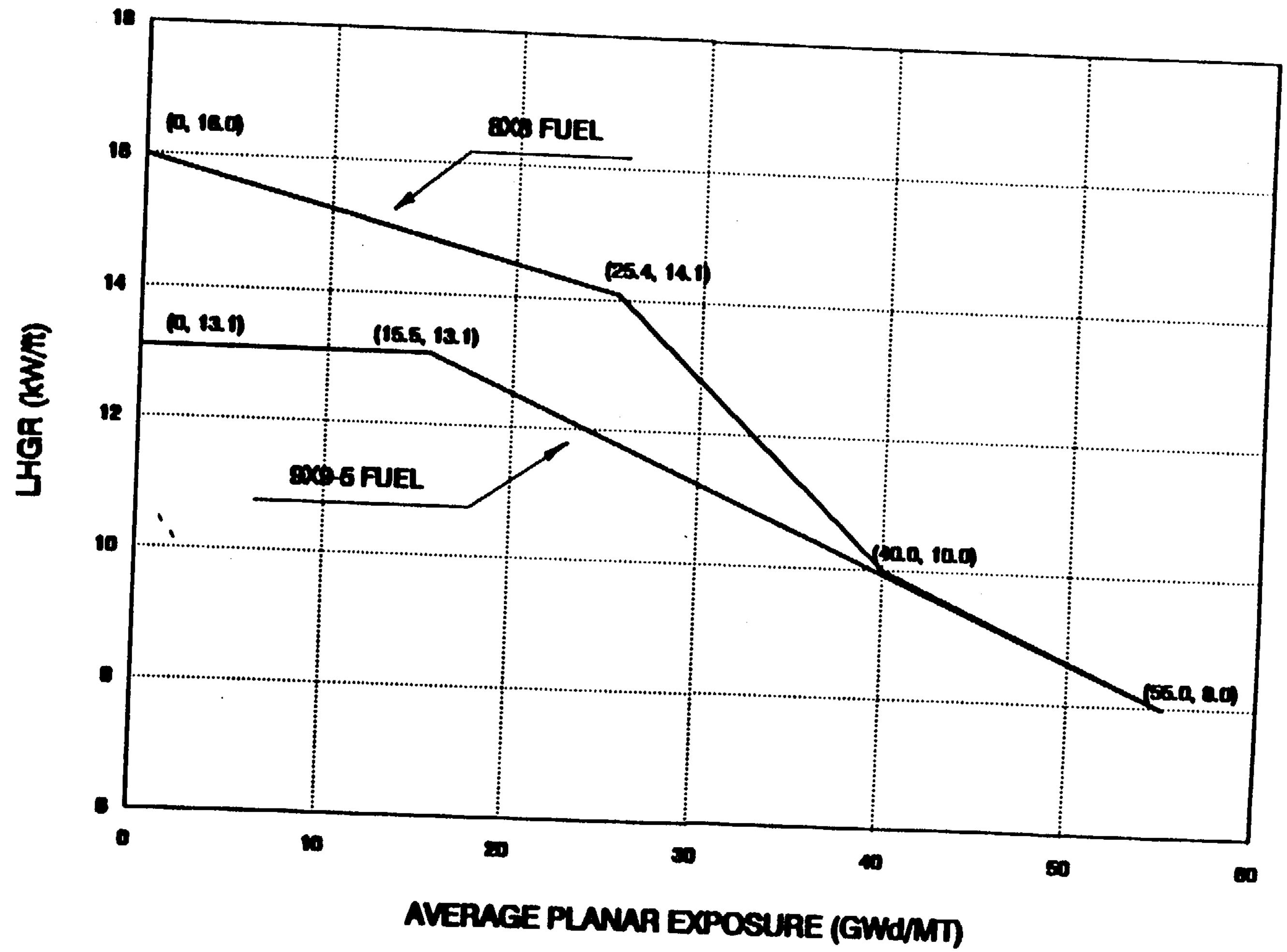

FIGURE 3.2.4-1 LHGR vs AVERAGE PLANAR EXPOSURE

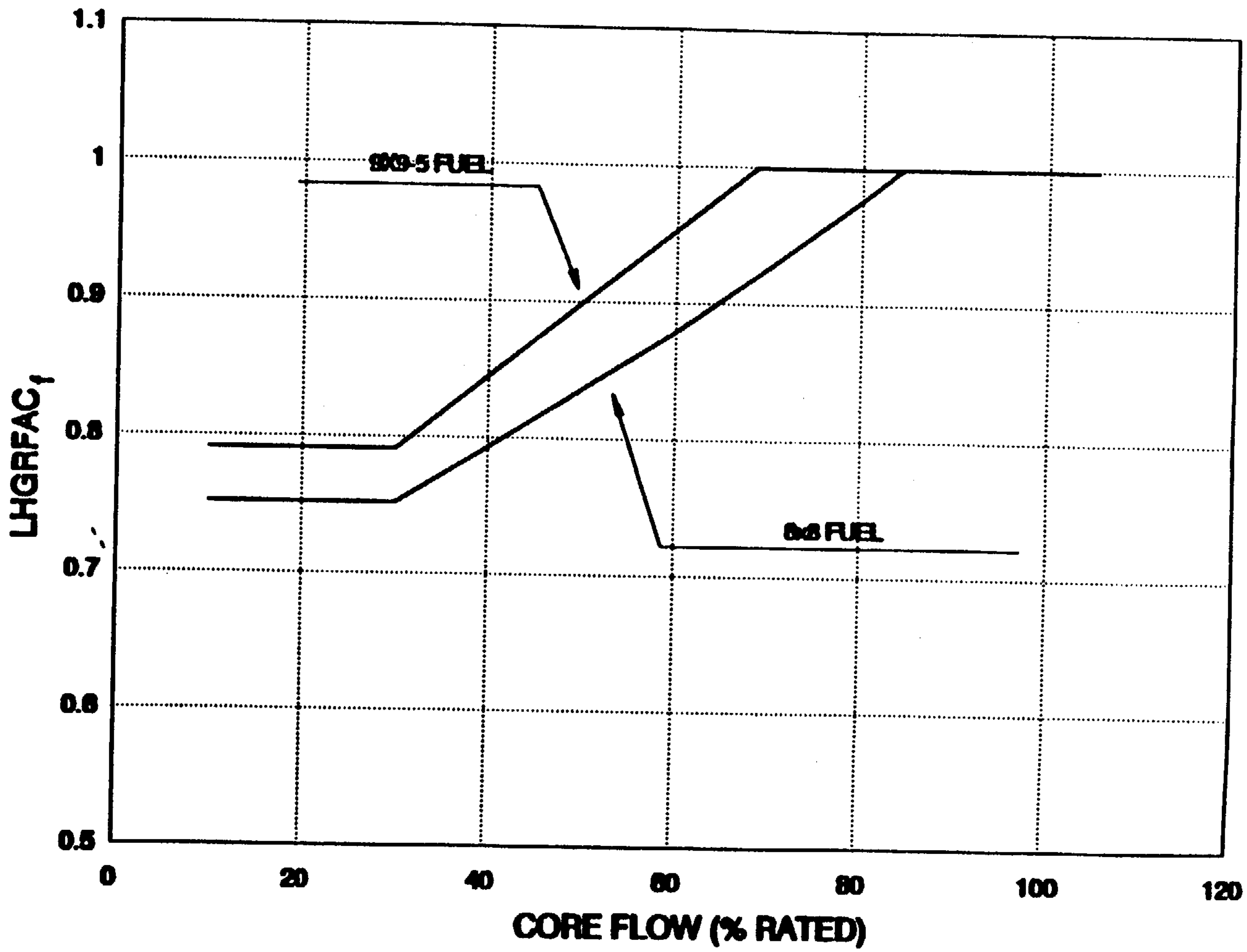

FIGURE 3.2.4-2 LHGRFAC$_f$

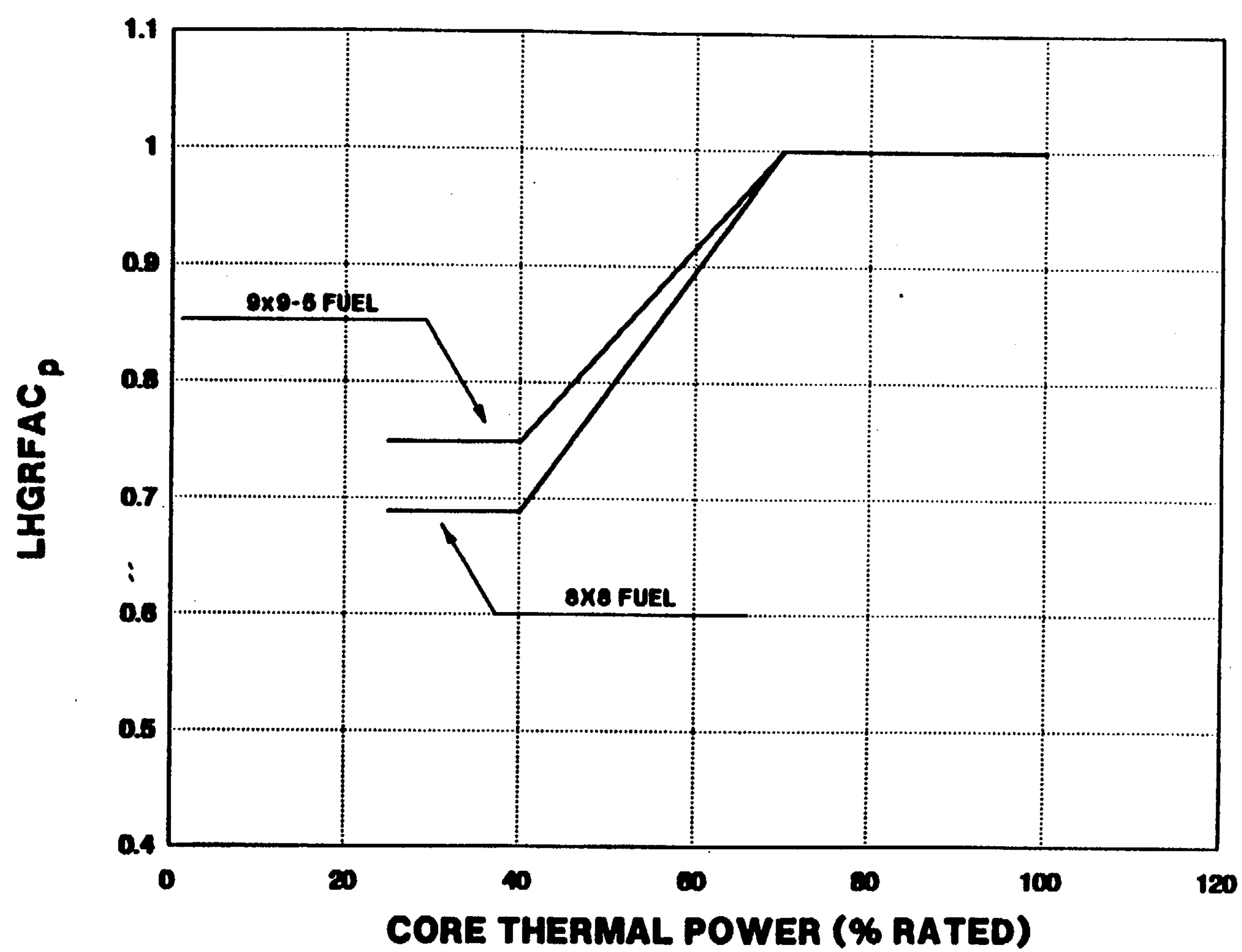

FIGURE 3.2.4-3 LHGRFAC$_p$

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits of Figure 3.2.1-1 are applicable to two loop operation.

For single-loop operation, a MAPLHGR limit corresponding to the product of the MAPLHGR, Figure 3.2.1-1, and 0.86 can be conservatively used to ensure that the PCT for single loop operation is bounded by the PCT for two loop operation.

The daily requirement for calculating APLHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 99 TO FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated December 5, 1991 (Ref. 1), the licensee (Entergy Operations, Inc.) submitted a request for revisions to the Grand Gulf Nuclear Station, Unit 1 (GGNS), Technical Specifications (TS). The revisions accommodate the core changes associated with Cycle 6 reload and operation.

The Cycle 6 reload will replace 272 SNP 8x8 fuel assemblies used in Cycle 5 with SNP 9x9-5 fuel assemblies. This is the second GGNS reload of this type; the GGNS Cycle 5 core consisted partly of SNP 9x9 fuel assemblies. In addition, a second batch of GE channels associated with the discharged 8x8 fuel will be replaced with CarTech channels. The core loading will retain 240 SNP 8x8 fuel assemblies and 4 lead test SNP 9x9-5 assemblies inserted in Cycle 4 and 284 SNP 9x9-5 fuel assemblies inserted in Cycle 5. Generally, the Cycle 6 reload is a normal reload with no unusual features other than the shift to a larger percentage of 9x9 fuel assemblies in the core. SNP 9x9 fuel has been used in other reactors; Susquehanna 2, for example, has been operating with an all SNP 9x9 fuel loading.

The Cycle 6 TS changes for GGNS are not extensive and are primarily related to the Minimum Critical Power Ratio (MCPR) safety limit, the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR), the Linear Heat Generation Rate (LHGR), and associated factors for Cycle 6 core operation as calculated by SNP. SNP performed the Cycle 6 reload analyses using methodologies that have been used for previous reload submittals and have been reviewed and approved by the staff.

2.0 EVALUATION

2.1 Fuel Design

The GGNS Cycle 6 reload will include 272 new SNP 9x9-5 Fuel assemblies. These contain 76 prepressurized fuel rods and 5 water rods. The rod enrichment distribution is described in the Cycle 6 Reload Analysis Report (Ref. 2). The mechanical design analyses for the SNP 8x8 and 9x9-5 fuel types are described

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in References 6 and 13 and in Supplement 1 of the GGNS Reload XN-1.3, "Cycle 4 Mechanical Design Report" (Ref. 5). Sections 2.1.1 through 2.1.6 evaluate the mechanical design analyses provided by the licensee in Reference 5 to support peak assembly discharge burnups of 40 GWd/MTU for both the 8x8 and the 9x9-5 fuel types. The fuel mechanical design is similar to that approved for Cycle 5. The fuel channels that will be used for Cycle 6 are manufactured by Carpenter Technology Corporation (CARTECH) and are of design similar to those used for Cycle 5.

SNP has analyzed the response of the SNP 9x9-5 fuel assemblies during seismic-LOCA events and has concluded that the response is essentially the same as for previous cycles because of the similarities of the physical properties and bundle natural frequencies. The licensee has demonstrated that the resultant loadings do not exceed the fuel design limits for either the 8x8 or the 9x9-5 fuel.

2.1.1 Stress and strain

The licensee used the approved methodology described in XN-NF-85-67(P)(A), Revision 1 (Ref. 6), for stress analysis and the approved RODEX2A code for strain analysis. The stress analysis showed that the cladding stress remained below the ASME code limits. The strain analysis showed that the cladding strain remained below the 1 percent strain limit. We conclude that the licensee's stress and strain analyses, based on the approved methodology and RODEX2A code, are acceptable for GGNS.

2.1.2 Rod Internal Pressure

The licensee's rod internal pressure criterion is that the rod pressure will be limited to a value below that which would cause (1) an increase of the diametral gap due to outward cladding creep and (2) extensive departure from nucleate boiling (DNB) propagation. The licensee used the approved methodology, described in Reference 6, to analyze the rod pressure. The results showed that the peak rod pressure was slightly above the system pressure but still met the design criterion. We conclude that the licensee's rod pressure analysis, based on the approved methodology, is acceptable for GGNS.

2.1.3 Fuel Temperature

The licensee's fuel temperature design criterion is that the maximum fuel temperature shall be less than the melting temperature of UO_2. The licensee used the approved methodologies described in Reference 6 for fuel temperature analysis. The result showed that the maximum fuel temperature remained below the melting temperature. We conclude that the licensee's fuel temperature analysis, based on the approved methodology, is acceptable for GGNS.

2.1.4 Cladding Fatigue

The licensee used the approved methodology, described in Reference 6, for fatigue analysis. The results showed that the fatigue usage factor was less than the acceptance criterion. We conclude that the licensee's cladding fatigue analysis, based on the approved methodology, is acceptable for GGNS.

2.1.5 Cladding Collapse

The licensee's cladding collapse criterion is that the cladding failure due to collapse should not occur. The design criterion also requires that the pellet-to-cladding gap remain open during the pellet densification. This requirement assures that axial gaps will not form in the fuel column. The licensee used the approved methodology, described in Reference 6, for collapse analysis. The results showed that the cladding collapse does not occur during the lifetime. We conclude that the licensee's collapse analysis, based on the approved methodology, is acceptable for GGNS.

2.1.6 Cladding Corrosion

The licensee used the approved methodology, described in Reference 6, for corrosion analysis. The results showed that the hydrogen pickup and clad oxidation were below the acceptance criterion. We conclude that the licensee's corrosion analysis, based on the approved methodology, is acceptable for GGNS.

Based on our review of the information presented and the similarities to previously approved design and analyses, we find the mechanical design of the ANF 9x9-5 fuel for GGNS Cycle 6 to be acceptable.

2.2 Nuclear Design

The SNP nuclear design methodology is presented in References 7 and 8, which have been reviewed and approved by the staff.

The beginning of cycle (BOC) shutdown margin is calculated to be 1.10 percent delta-K, and BOC + 500 MWd/MTU and BOC + 7500 MWd/MTU were determined to be most limiting conditions with a shutdown margin of 1.03 percent delta-K. Thus the cycle minimum shutdown margin is well above the required 0.38 percent delta-K. The Standby Liquid Control System also fully meets shutdown requirements. The GGNS high density spent fuel storage racks were reviewed and approved by the staff for the Cycle 5 reload (Ref. 9). The most reactive segment of the Cycle 6 fuel at its most reactive point in life is less reactive than was analyzed for Cycle 5. Therefore, it was concluded that the Cycle 5 analysis is bounding for the Cycle 6 fuel and that the storage racks can safely accommodate the Cycle 6 fuel.

The GGNS Cycle 6 nuclear characteristics have been calculated with approved methodologies, the results meet applicable criteria, and the review concludes that the design is acceptable.

2.3 Thermal-Hydraulic Design

This is the second reload of this type at GGNS. That the SNP 9x9 fuel is thermal-hydraulically compatible with the retained SNP 8x8 fuel has been demonstrated by approved methodologies, by the use of a partial 9x9 core for GGNS Cycle 5, and by the use of SNP 9x9 fuel at other BWRs.

The thermal-hydraulic methodology and criteria used for GGNS Cycle 6 are the same as used for the previous reload and are described in References 10 and 11. These methodologies are acceptable for Cycle 6 analysis.

The MCPR safety limit has been determined to be 1.06 for two-loop operation (TLO) and 1.07 for single-loop operation (SLO). The methodology and generic uncertainties used by SNP to perform the MCPR safety limit calculation are provided in Reference 12. This calculation included an evaluation of the effects of channel bow. The flow-dependent MCPR, power-dependent MCPR, and the exposure dependent MCPR limits were all revised for Cycle 6. These calculations were performed using approved methods and the limits are acceptable.

GGNS is currently operating under the BWR Owner's Group/General Electric Interim Recommendations for Stability Actions (IRSA) with stability boundary TS that were approved by the staff for the Cycle 5 core consisting partly of SNP 9x9-5 fuel. A comparative evaluation of the stability characteristics of the Cycle 5 and Cycle 6 cores, as well as of a full 9x9-5 core, was performed by SNP. The results of the SNP evaluation showed that the core decay ratios for the cycles were equivalent. The staff review concludes that continued use of the current stability TS boundaries is acceptable.

2.4 Anticipated Operational Occurrences and Accident Analyses

To provide the basis for the TS values of the various operating limits (MCPR and LHGR), SNP has analyzed the system Anticipated Operational Occurrence (AOO) events that could provide the most limiting conditions. This approach is in accordance with the approved methodology for operating limit analysis. The AOO events include Load Rejection Without Bypass (LRNB), Feedwater Controller Failure (FWCF), Loss of Feedwater Heating (LFWH), Flow Excursion (FE), Control Rod Withdrawal Error (CRWE), and the Fuel Loading Error (FLE). Previous analyses have shown that other events are non-limiting. Plant initial conditions for the analyses covered the full range of Maximum Extended Operating Domain (MEOD) approved for GGNS. Analyses were performed for End-of-Cycle (EOC), EOC-30 EFPD (Effective Full Power Days), and EOC+30 EFPD to provide exposure-dependent MCPR limits. Results of these analyses were used to provide the TS MCPR and LHGR limits as functions of power, flow, and exposure.

The analysis of the AOO events and the establishment of limiting operating values for MCPR and LHGR used approved methods and considered required events and reactor conditions. The analysis and the results are therefore acceptable.

SLO was also analyzed by SNP. The pump seizure event was analyzed and the MCPR safety limit and the MAPLHGR multiplier were determined for single-loop operations. The SLO MCPR was calculated to be 1.07, and the MAPLHGR multiplier was found to be 0.86. The analyses were performed with approved methods, and the results are therefore acceptable.

Compliance with overpressure criteria was demonstrated by analysis of the main steam isolation valve (MSIV) closure event, assuming failure of the direct scram signal on MSIV position. The analysis used approved methods and resulted in a pressure within the required limits and is therefore acceptable.

Accident analyses were performed by SNP for the Loss-of-Coolant Accident (LOCA) and the Rod Drop Accident (RDA) to demonstrate that the required limits are met for GG1C6. This analysis was performed with approved methods and is therefore acceptable.

2.5 <u>Technical Specification Changes</u>

The following TS changes have been proposed for operation of Cycle 6.

(1) TS 2.1.2

The MCPR safety limit is decreased from 1.09 to 1.06 for TLO and from 1.09 to 1.07 for SLO.

(2) TS 3/4.2.1

The SLO MAPLHGR multiplier is changed from 0.8 to 0.86. This change was based on SNP's detailed LOCA analysis for SLO.

(3) TS 3/4.2.3 - Figure 3.2.3-1

Flow-dependent MCPR limits have been revised. The lower $MCPR_f$ limits result from the lower MCPR safety limit and smaller delta-CPR values due to the improved transient response of the 9x9-5 fuel.

(4) TS 3/4.2.3 - Figure 3.2.3-2

Power-dependent MCPR limits have been revised to provide common $MCPR_p$ limits for both TLO and SLO.

(5) TS 3/4.2.3 - Figure 3.2.3-3

Exposure-dependent MCPR limits have been revised to reflect the revision to the MCPR safety limit and the improved transient response of the Cycle 6 core.

(6) TS 3/4.2.4 - Figure 3.2.4-1

LHGR limits for 8x8 fuel types have been increased for average planar exposures greater than 40,000 MWd/MTU in order to bound the LHGR performance for the 8x8 fuel near the end of Cycle 6.

(7) TS 3/4.2.4 - Figures 3.2.4-2 and 3

Off-rated mechanical limits have been revised to reflect the predominantly 9x9-5 fueled core and the higher LHGR limit for SNP 9x9-5 fuel.

All of the above changes are based upon analyses performed with approved methods and yielding results within prescribed safety limits. They are therefore acceptable. There are also changes to the Bases associated with the above TS to reflect the changes to the specifications or minor administrative changes. The changes reflect the TS changes and are acceptable. These include Bases 2.1.1, 2.1.2, and 3/4.2.1.

The staff has reviewed the reports submitted for the Cycle 6 operation of GGNS and concludes that appropriate material was submitted and that the fuel design, the nuclear design, the thermal-hydraulic design, and the transient and accident analyses are acceptable. The TS changes submitted for this reload reflect the necessary modifications for operation in this cycle.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Mississippi State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (57 FR 2593). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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