March 13, 1989

Docket No. 50-416

DISTRIBUTION
See attached sheet

Mr. W. T. Cottle
Vice President, Nuclear Operations
System Energy Resources, Inc.
Post Office Box 23054
Jackson, Mississippi 39205

Dear Mr. Cottle:

SUBJECT: ISSUANCE OF AMENDMENT NO. 57 TO FACILITY OPERATING LICENSE NO. NPF-29 - GRAND GULF NUCLEAR STATION, UNIT 1, REGARDING FUEL CYCLE 4 RELOAD (TAC NO. 71438)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 57 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment consists of changes to the Technical Specifications (TS) in response to your application dated December 6, 1988, as supplemented December 30, 1988 and January 31, 1989.

The amendment changes the TS as required to support the fuel reload for Cycle 4. Changes are made to the Bases for Section 2.1, "Safety Limits," the TS and Bases for Section 3/4.2, "Power Distribution Limits," and the TS for Section 5.3.1, "Fuel Assemblies."

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

/S/

Lester L. Kintner, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 57 to NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

[GGNS AMEND 71438]

Not SE Changes made LLK

with indicated revision to sec 2.4

OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	:BC:SRXB:DEST	:BC:PRPB:DREP	:D:PD21:DRPR	:	: : :
NAME	: PAnderson	: LKintner:jw	: MHodges	: LCunningham	: EReeves	:	: : :
DATE	:2/15/89	:2/15/89	:2/17/89	:2/23/89	:3/8/89	:	:

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System Energy Resources, Inc.

Grand Gulf Nuclear Station (GGNS)

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AMENDMENT NO. 57 TO FACILITY OPERATING LICENSE NO. NPF-29 - GRAND GULF

Docket File
NRC PDR
Local PDR
PDII-1 Reading
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ARM/LFMB

cc: Licensee/Applicant Service List

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

SYSTEM ENERGY RESOURCES, INC., et al.

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 57
License No. NPF-29

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

   A. The application for amendment by System Energy Resources, Inc., (the licensee), dated December 6, 1988, as supplemented December 30, 1988 and January 31, 1989, 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) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 57 , are hereby incorporated into this license. System Energy Resources, 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

Edward A. Reeves

Edward A. Reeves, Acting Director
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: March 13, 1989

not SE

OGC changes made

OFC	LA:PD21:DRPR	PM:PD21:DRPR	OGC	D:PD21:DRPR			
NAME	PAnderson	LKintner	MYoung	EReeves			
DATE	2/15/89	2/15/89	3/2/89	3/8/89			

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ATTACHMENT TO LICENSE AMENDMENT NO. 57

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.

Remove	Insert
B 2-1a	B 2-1a
B 2-2	B 2-2
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
	3/4 2-2d
	3/4 2-2e
3/4 2-3a	3/4 2-3a
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-7a	3/4 2-7a
B 3/4 2-1	B 3/4 2-1
B 3/4 2-4	B 3/4 2-4
B 3/4 2-6	B 3/4 2-6
5-5	5-5

2.1 SAFETY LIMITS

BASES

THERMAL POWER, Low Pressure or Low Flow (Continued)

The Advanced Nuclear Fuels Corporation (ANF) XN-3 critical power correlation is applicable to the mixed core beginning with cycle 2. The applicable range of the XN-3 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. Overall, because of the design thermal-hydraulic compatibility of the ANF 8x8 fuel design 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's 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 XN-3 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 XN-3 correlation is applicable at any pressure greater than 585 psig, the cycle 1 low pressure boundary of 785 psig remains valid for the XN-3 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. One specific uncertainty included in the safety limit is the uncertainty inherent in the XN-3 critical power correlation. ANF report XN-NF-524(A), Rev. 1, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors," Nov. 1983, describes the methodology used in determining the Safety Limit MCPR.

The XN-3 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 high 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 XN-3 correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the XN-3 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 Figures 3.2.1-1a, 3.2.1-1b, 3.2.1-1c, 3.2.1-1d, or 3.2.1-1e as multiplied by the smaller of either the flow-dependent MAPLHGR factor ($MAPFAC_f$) of Figure 3.2.1-2, or the power-dependent MAPLHGR factor ($MAPFAC_p$) of Figure 3.2.1-3.

During single loop operation, the APLHGR for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits as determined below:

a) for 8x8 ANF fuel types - the limit shown in Figure 3.2.1-1 as multiplied by the smaller of either $MAPFAC_f$, $MAPFAC_p$ or 0.86; and

b) for 9x9 ANF fuel type the limit determined in Figure 3.2.1-1e as multiplied by the smaller of either $MAPFAC_f$, $MAPFAC_p$ or 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 Figures 3.2.1-1, 3.2.1-1a, 3.2.1-1b, 3.2.1-1c, 3.2.1-1d or 3.2.1-1e as corrected by the appropriate multiplication factor for each type of fuel, 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.

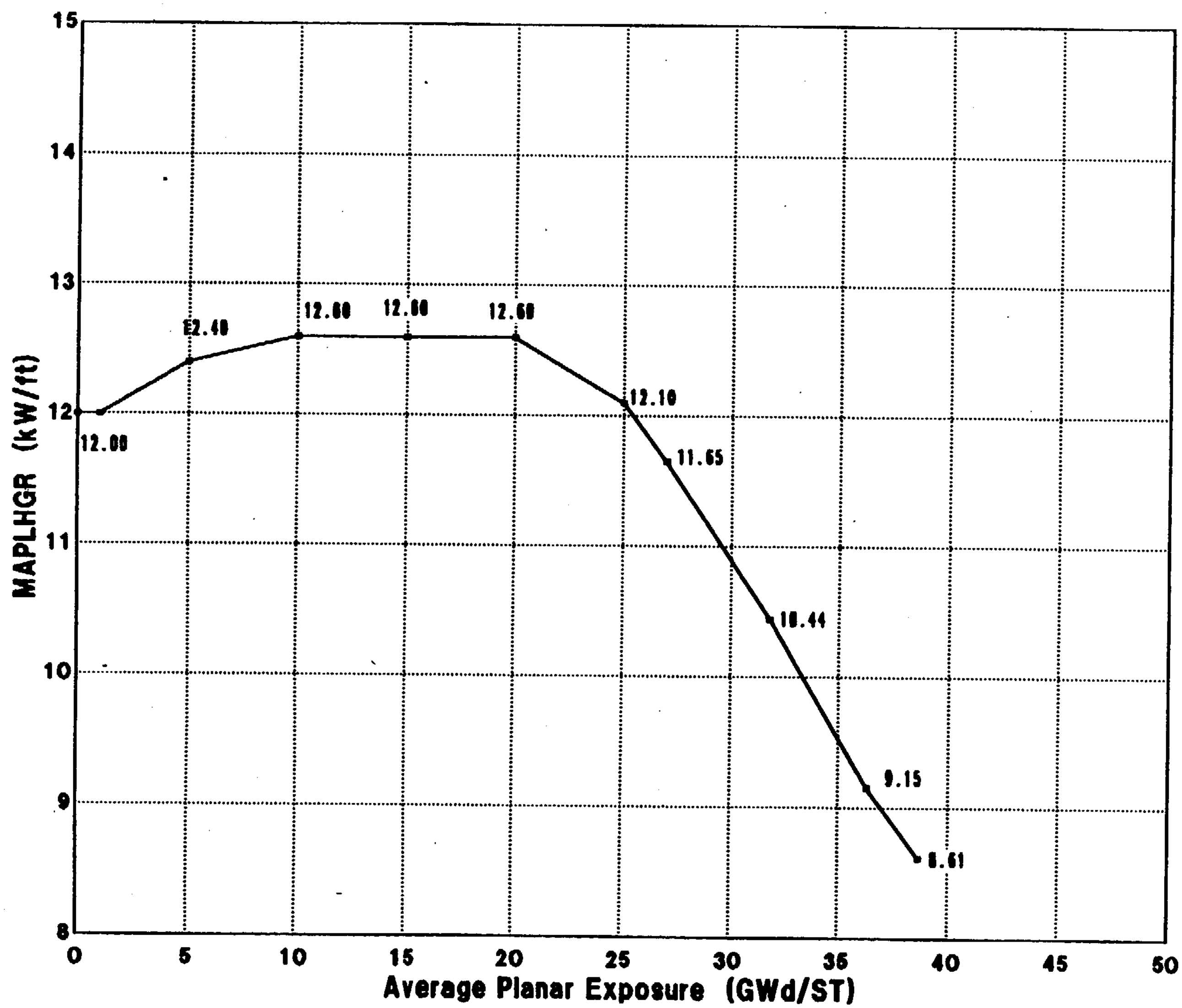

FIGURE 3.2.1-1 MAPLHGR vs AVERAGE PLANAR EXPOSURE FOR SINGLE LOOP OPERATION, 8X8 FUEL

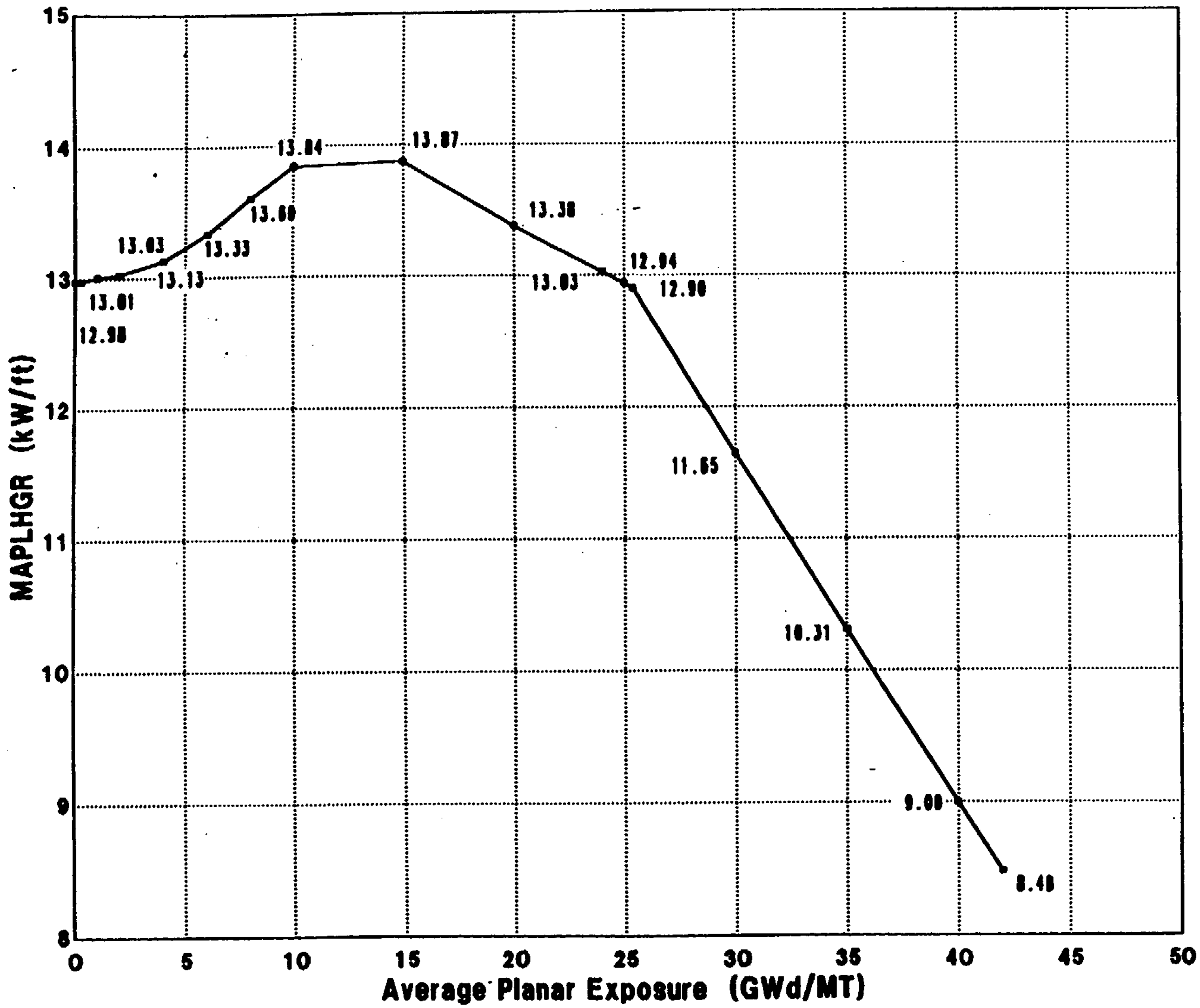

FIGURE 3.2.1-1d MAPLHGR vs AVERAGE PLANAR EXPOSURE
FOR ANF FUEL TYPE ANF361E8GZS8

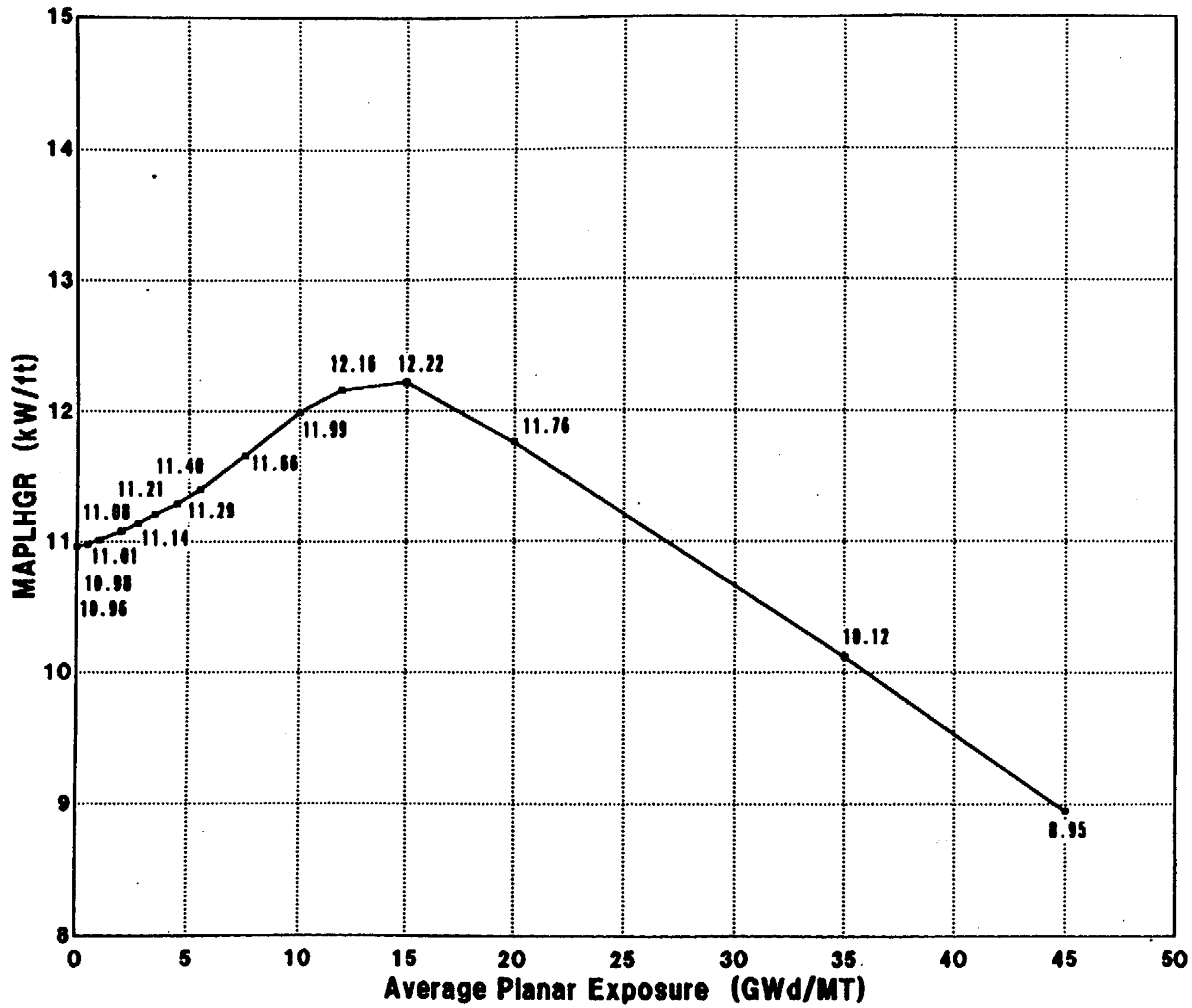

FIGURE 3.2.1-1e MAPLHGR vs AVERAGE PLANAR EXPOSURE FOR ANF 9x9-5 LEAD TEST ASSEMBLY

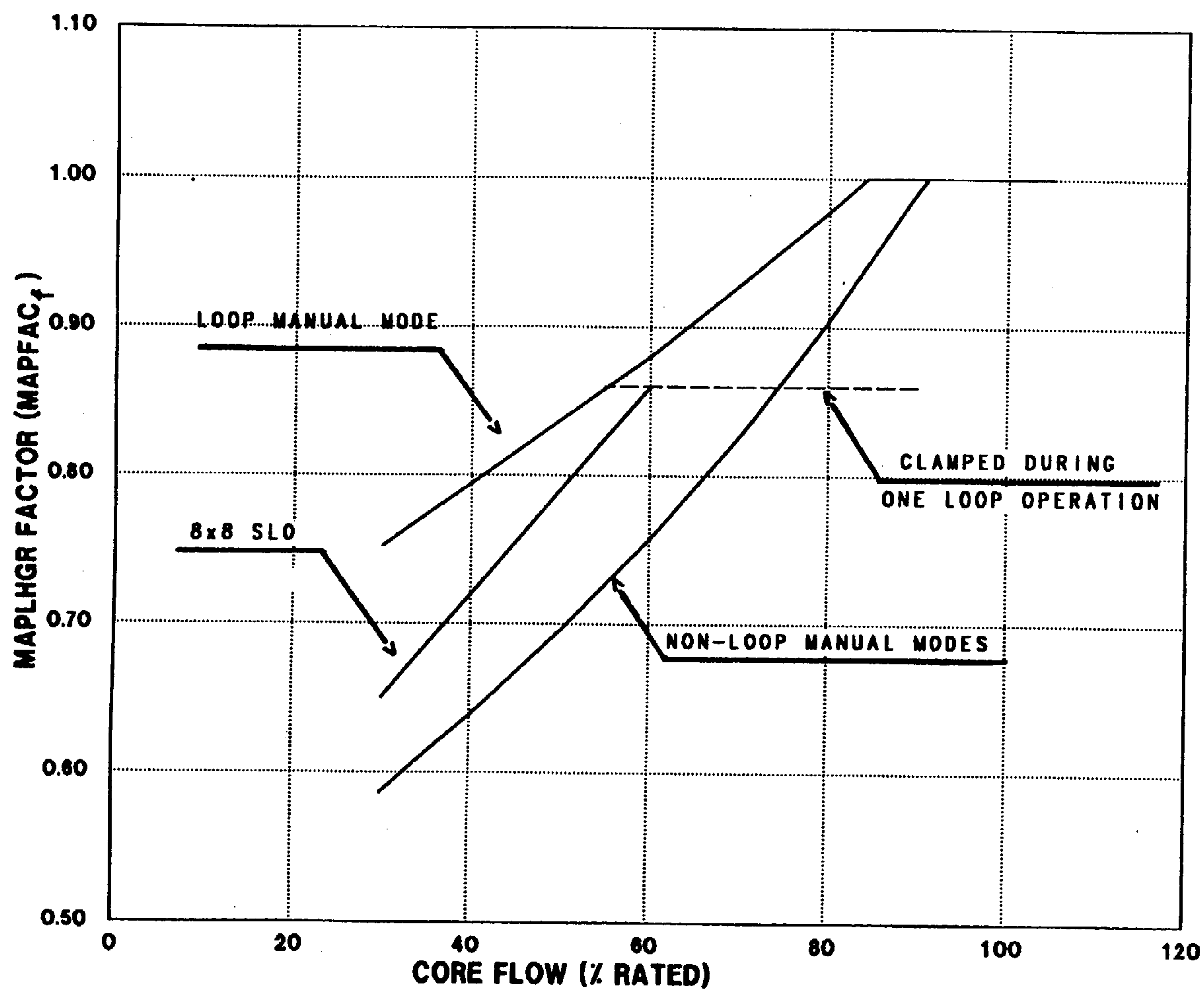

FIGURE 3.2.1-2 MAPFAC$_f$

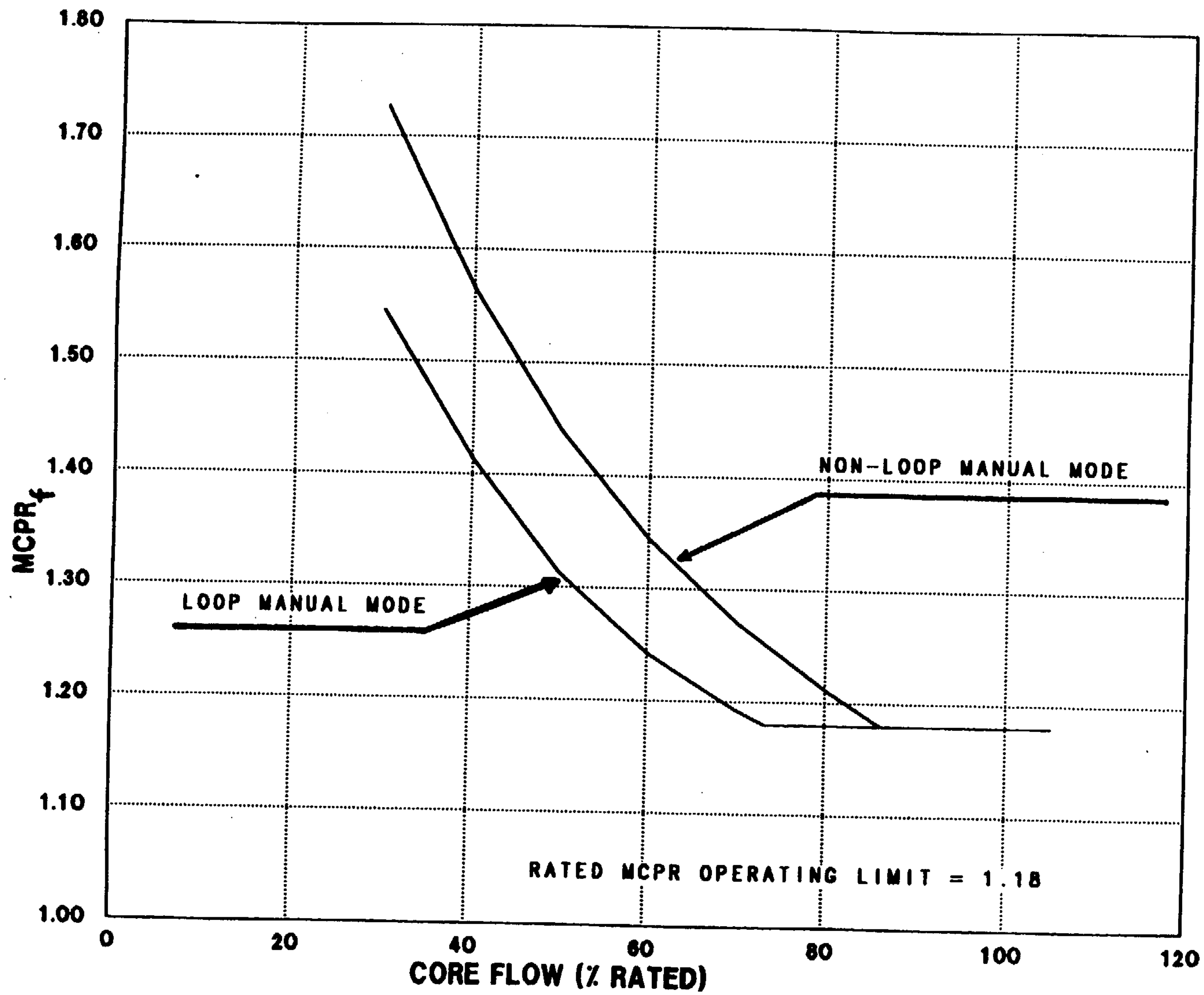

FIGURE 3.2.3-1 $MCPR_f$

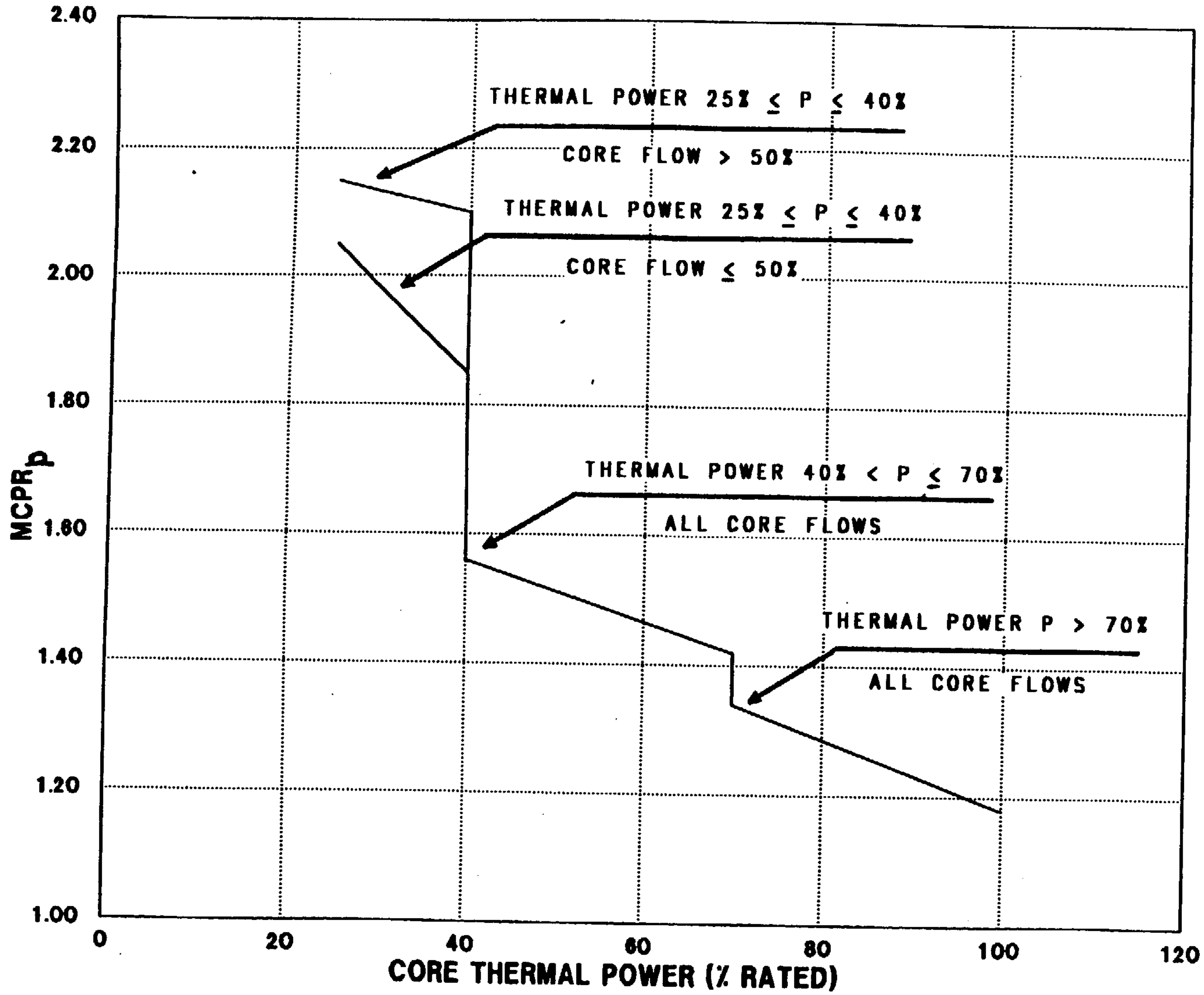

FIGURE 3.2.3-2 MCPR$_p$

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the limits shown in Figure 3.2.4-1.

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

ACTION:

With the LHGR of any fuel rod exceeding the limit of Figure 3.2.4-1, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGR's shall be determined to be equal to or less than their allowable 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,

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

d. The provisions of Specification 4.0.4 are not applicable.

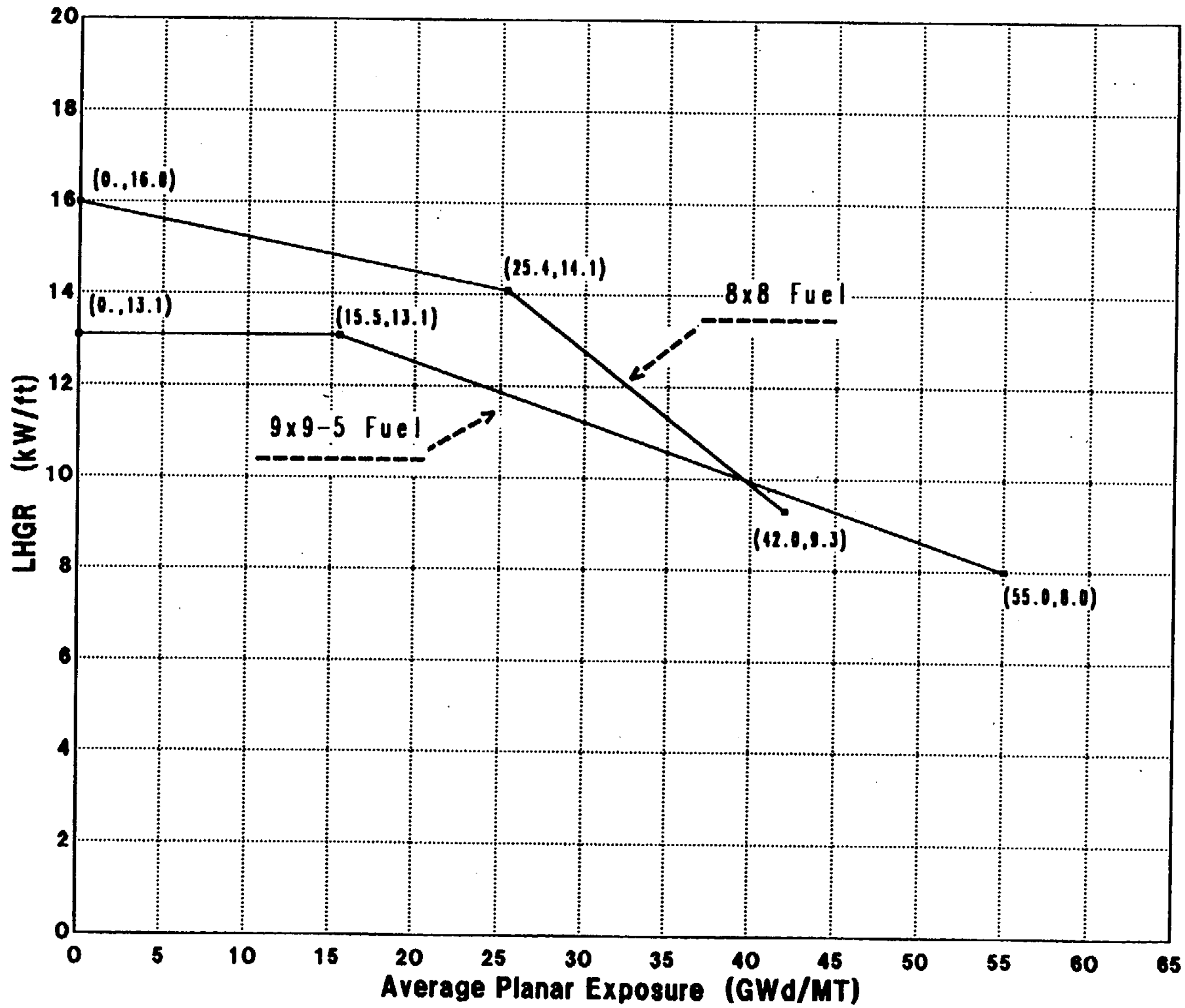

FIGURE 3.2.4-1 LINEAR HEAT GENERATION RATE (LHGR) LIMIT VERSUS AVERAGE PLANAR EXPOSURE FOR ANF FUEL

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 peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits of Figures 3.2.1-1, 3.2.1-1a, 3.2.1-1b, 3.2.1-1c, 3.2.1-1d or 3.2.1-1e are multiplied by the smaller of either the flow dependent MAPLHGR factor ($MAPFAC_f$) or the power dependent MAPLHGR factor ($MAPFAC_p$) corresponding to existing core flow and power state to assure the adherence to fuel mechanical design bases during the most limiting transient.

For single-loop operation with ANF 8x8 fuel, a MAPLHGR limit corresponding to the product of the MAPLHGR, Figure 3.2.1-1, and the appropriate MAPFAC, can be conservatively used. The allowable MAPLHGR shown in Figure 3.2.1-1 is a conservative bound during Cycle 4 for all 8x8 fuel types and the Cycle 3 SLO MAPLHGR (Reference 5). The MAPLHGR limit for ANF 9x9-5 fuel is the product of the MAPLHGR shown in Figure 3.2.1-1e and the appropriate MAPFAC. The maximum MAPFAC during single loop operation is 0.86 for all fuel types.

$MAPFAC_f$'s are determined using the three-dimensional BWR simulator code to analyze slow flow runout transients. Two curves are provided based on the maximum credible flow runout transient for ANF fuel for either Loop Manual or Non Loop Manual operation. The result of a single failure or single operator error during operation in Loop Manual is the runout of only one loop because both recirculation loops are under independent control. Non-Loop Manual operational modes allow simultaneous runout of both loops because a single controller regulates core flow.

$MAPFAC_p$'s are generated to protect the core from plant transients other than core flow increases.

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

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta CPR. When added to the Safety Limit MCPR, the required operating limit MCPR of Specification 3.2.3 is obtained. The power-flow map of Figure B 3/4 2.3-1 defines the analytical basis for generation of the MCPR operating limits (Reference 7).

The purpose of the $MCPR_f$ and $MCPR_p$ is to define operating limits at other than rated core flow and power conditions.

The $MCPR_f$s are established to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured. The reference core flow increase event used to establish the $MCPR_f$ is a hypothesized slow flow runout to maximum, that does not result in a scram from neutron flux overshoot exceeding the APRM neutron flux-high level (Table 2.2.1-1 item 2). Two flow rates have been considered. The maximum credible flow during a runout transient depends on whether the plant is in Loop Manual or Non Loop Manual operation. The result of a single failure or single operator error during Loop Manual operation is the runout of one loop because the two recirculation loops are under independent control. Runout of both loops is possible during Non Loop Manual operation because a single controller regulates core flow. With this basis, the $MCPR_f$ curves are generated from a series of steady state core thermal hydraulic calculations performed at several core power and flow conditions along the steepest flow control line. In the actual calculations a conservative highly steep generic representation of the 105% steam flow rodline flow control line has been used. Assumptions used in the original calculations of this generic flow control line were consistent with a slow flow increase transient duration of several minutes: (a) the plant heat balance was assumed to be in equilibrium, and (b) core xenon concentration

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 and Non Loop Manual modes of operation were analyzed. Consistent with the single failure/single operator error criterion, one loop runout was postulated for Loop Manual operation whereas two loop runout was postulated for Non 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 1.06.

The $MCPR_p$ is established to protect the core from plant transients other than core flow increase including the localized rod withdrawal error event. 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 ANF report, XN-NF-825 (P)(A), Supplement 2. 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. 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.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 800 fuel assemblies. Each fuel assembly shall contain fuel rods and water rods clad with Zircaloy cladding. Each fuel rod shall have a design nominal active fuel length of 150 inches. The initial core loading shall have a design nominal enrichment of 1.708 weight percent U-235. Reload fuel shall have mechanical, thermal-hydraulic and neutronic characteristics compatible with the initial core loading.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 193 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing a design nominal 143.7 inches of boron carbide, B_4C, powder surrounded by a cruciform shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pump.
2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.
3. 1550 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,000 cubic feet at a nominal T_{ave} of 533°F.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 57 TO FACILITY OPERATING LICENSE NO. NPF-29

SYSTEM ENERGY RESOURCES, INC., et al.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated December 6, 1988, as supplemented December 30, 1988 and January 31, 1989 (Reference 1), System Energy Resources, Inc. (the licensee), requested an amendment to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1 (GGNS-1). The proposed amendment would change the Technical Specifications (TS) as required for the reload and operation of Cycle 4. The requested TS changes and reports discussing the reload and analyses to support and justify Cycle 4 operation were enclosed in the December 6, 1988 submittal (References 2-4).

The January 31, 1989 submittal provided a non-proprietary version of a report previously submitted and did not alter the action noticed, or affect the initial determination published, in the Federal Register on February 8, 1989.

This reload has the following features: (a) it will make the reactor core the first total 8x8 Advanced Nuclear Fuels (ANF) fuel core, (b) the fresh fuel is designed for higher discharge exposures than the previous reloads, thus it has higher enrichment and higher gadolinia loading, and (c) it includes four lead test assemblies (LTA) of ANF 9x9-5 fuel. The reload methodology is based on the approved ANF topical reports XN-NF-80-19(A), Volumes 1-3 (References 9, 10, 13 and 14). Additional information has been submitted in plant specific reports. The scope of the proposed TS changes includes:

1. Addition of the maximum axial planar linear heat generation rate (MAPLHGR) curve for the new 8x8 fuel.

2. Revision of the single loop MAPLHGR curve.

3. Change of flow and power dependent thermal limits for off-rated condition transients.

4. Addition of linear heat generation rate (LHGR) and MAPLHGR curves for the 9x9-5 LTAs.

The specific TS changes are as follows:

1. Bases 2.1.1 and 2.1.2 - Editorial changes to substitue the name "Advanced Nuclear Fuels Corporation" for "Exxon Nuclear Company."

2. TS 3/4.2.1 - Revisions to delete references to the replaced fuel, to reference MAPLHGR curves and information for the new fuel and the 9x9-5 LTAs.

   A MAPLHGR curve for the new 8x8 and the 9x9-5 fuel has been added. The single loop operation limit is added for the new 8x8 fuel and the 9x9-5 lead assemblies. Figures 3.2.1-1, 3.2.1-2, 3.2.3-1, 3.2.3-2 and 3.2.4-1 reflect these changes.

3. TS 3/4.2.4 - Editorial changes to delete references to GE fuel which is no longer present in Cycle 4.

4. Bases 3/4.2.1 - References, text revisions and discussion were added to cover the revised MAPLHGR single loop operation. Description of flow runout was deleted. Text revision for $MAPFAC_p$ was made.

5. Bases 3/4.2.3 - The transient evaluation description for the GE core has been eliminated. The discussion of $MCPR_f$ for the loop manual and non-loop manual operation has been revised. The cycle specific $MCPR_p$ discussion is deleted. An editorial change was made to Figure B 3/4.2.3-1.

6. TS 5.3.1 - Editorial changes were made to generalize the description for all Cycle 4 fuel types.

2.0 EVALUATION

2.1 Reload Description

The Cycle 4 core will consist of 800 8x8 fuel assemblies of all ANF manufacture. Of these, 236 are twice burned (first reload), 288 are once burned (second reload), 272 are fresh (third reload) and 4 are 9x9-5 LTAs. The core arrangement is the conventional scatter load with the lowest reactivity bundles placed in the peripheral regions of the core. The loading pattern is designed to maximize the cycle energy while minimizing the peaking factors. The Cycle 4 core is estimated to provide 1,698 gigawatt days (GWd) of energy compared to an estimated 1,455 GWd for Cycle 3.

2.2 Fuel Design

The mechanical design of the fuel for the entire Cycle 4 loading is described in XN-NF-85-67(P)(A), Revision 1 (Reference 5). The 8x8 ANF fuel assembly contains 62 fuel rods and 2 water rods. The fuel rods are pressurized and use a diametral pellet to clad gap, which is smaller on the interior high enrichment rods than on the remaining rods in the bundle, to improve ECCS MARGIN. The scope of the mechanical design analyses included: cladding steady-state strain, transient stresses, fatigue damage, creep collapse, corrosion buildup, hydrogen absorption, fuel rod maximum internal pressure, differential fuel rod growth, creep bow and the grid spacer spring design. All parameters meet their respective design limits for a batch average burnup of 34,000 megawatt days per metric ton of uranium (MWd/MTU). This average burnup is about 4,000 MWd/MTU higher than the Cycle 3 burnup. The peak assembly exposure is 39,000 MWd/MTU (Reference 6).

The mechanical fuel design is essentially the same as the generic ANF design; thus, the majority of the specific features are covered by generic mechanical design reports. The few analyses which have been extended use approved methodology.

The mechanical response of the fuel assembly during loss of coolant accidents (LOCA) or seismic events is the same as the response of a GE assembly, because the physical properties and bundle natural frequencies are similar. The seismic-LOCA analyses for the GE fuel showed that the resultant loading would not exceed the fuel design limits (Reference 7). Seismic-LOCA analyses for the ANF fuel showed large design margins compared to the GE fuel (Reference 8). Because of the similarity of the fuel types and the large margin calculated for the ANF fuel, we find the mechanical design of the ANF fuel assemblies to be acceptable.

2.3 Thermal Hydraulic Design Analysis

The methodology used for the thermal hydraulic design analysis is described in the ANF approved topical report XN-NF-80-19(A), Volume 3, Revision 2 (Reference 9). The thermal hydraulic design criteria, which were used in the determination of the fuel cladding integrity safety limits and bypass flow, were defined as described in the approved ANF topical report XN-NF-80-19(A), Volume 4, Revision 1, (Reference 10).

The uncertainties used in the minimum critical power ratio (MCPR) safety limit calculation are provided in the approved topical report SN-NF-524(A) (Reference 11). The specific inputs for GGNS-1 and the results of the calculations for Cycle 4 are given in Reference 3.

The operating limit MCPR (OLMCPR) values are determined by the limiting transients. To confirm, and if needed revise, the thermal limits for the all-ANF Cycle 4 core, the following transients were analyzed: load rejection without bypass (LRNB), feedwater controller failure (FWCF) and loss of feedwater heating (LFWH). In addition, it was established that the generic analysis for the control rod withdrawal error is applicable to GGNS-1, Cycle 4. All these cases have shown that Cycle 4 is less restrictive than Cycle 3. Therefore, it is reasonable to conclude that the less restrictive transients will continue to be protected. The LFWH transient analysis covered the conditions from beginning to end of cycle and the maximum extended operating domain (MEOD). The calculation assumed a conservative reduction of 100°F in the feedwater temperature. The results showed that the LFWH OLMCPR for all operating conditions is 1.17. The LRNB event is the most limiting pressurization transient. In this transient, the load rejection causes fast closure of the turbine valves resulting in a vessel compression wave and reactor scram. In the analysis, condenser bypass is not allowed. The power spike due to void collapse is terminated by the scram and the recirculation pump. The maximum steam dome pressure is 1,280 psig, which is less than 1,325 psig the required limit.

A flow transient is also analyzed to determine the flow dependent thermal limits $MCPR_f$ and the $MAPFAC_f$. The transients analyzed assume failure of the recirculating control system which results in a flow increase equal to the maximum physically attainable flow. Two operational modes were assumed, i.e., flow excursion of one pump (designated loop manual) and of both pumps (designated non-loop manual). For both events, the recirculation system capacity was set at 110% of rated. For both cases, calculations show that the $MCPR_f$ and $MAPFAC_f$ values are conservative compared to the Cycle 4 operating limits that have been established for Cycle 4 (Reference 4). Core flows for the one loop were initialized from 30% to 100% of rated flow and for seven burnup values through Cycle 4. The MCPR safety limit for all types of fuel in Cycle 4 remain at 1.06 (References 10 and 11). However, the licensee decided to retain the existing 1.18 as the MCPR operating limit. The previous most limiting value was 1.17.

Finally, the power dependent $MCPR_p$ and $MAPFAC_p$ for off-rated condition operation during anticipated operational occurrences has been determined by adding the delta-CPR for the limiting event to the calculated safety limit MCPR. The $MAPFAC_p$ is used to protect against fuel melting and excessive clad strain by setting conservative LHGR limits consistent with the MAPLHGR and consideration of the maximum local peaking factors. The results showed that above 40% power the Cycle 3 $MCPR_p$ bounds the Cycle 4 results. Therefore, the existing limit remains unchanged. Similarly the $MAPFAC_p$ values are conservative with respect to the existing values (Reference 3) and require no change.

In summary, the thermal hydraulic design analysis has been performed with approved methods and conservative data. The resulting proposed TS power distribution limits are either within the limits of existing analyses or within the operating limits set for Cycle 4 and, therefore, are acceptable.

2.4 Core Stability Analysis

The 8x8-2 ANF fuel assemblies used in Cycle 4 are hydrodynamically similar to the previously used GE fuel assemblies. For Cycle 4, all fuel assemblies will be ANF assemblies, whereas for Cycle 3 about one third of the core was loaded with GE fuel assemblies. The licensee's analyses performed for Cycle 4 confirmed that stability analyses and tests performed for Cycle 3 are applicable to Cycle 4. The four 9x9-5 ANF lead test assemblies in the Cycle 4 core will be placed in low-power regions of the core. In this position the probability of channel instability in these assemblies is minimal. The overall stability characteristics of the core will, in any case, be determined by the 8x8 fuel and not by these four assemblies. We conclude that the core stability analysis for Cycle 4 is acceptable.

2.5 Nuclear Design Analysis

The methodology used for the nuclear design and analysis is contained in the NRC approved topical reports XN-NF-80-19A, Volume 1, and Supplements 1 and 2 (Reference 13). The core description and the results of the core reactivity characteristics are given in ANF-88-149 (Reference 2). The results are within the range of those usually encountered in BWR reloads. In particular, the shutdown margin is 1.094% delta-k/k at the beginning of the fuel cycle. This value is the minimum value because the reactivity defect is (0.0% delta-k/k) by a large margin. The shutdown margin is greater than the required value in the TS (0.38% delta-k/k) by a large margin. Similarly, the standby liquid control system reactivity at 660 ppm boron concentration, for cold xenon free condition is k effective equals 0.96215, which provides adequate shutdown margin. The end of Cycle 4 core exposure is estimated to be 22,308 MWd/MTU with a maximum value of 23,130 MWd/MTU. The nuclear design analysis was performed with previously approved methods and the results fall within expected ranges and with adequate margin. Therefore, we find the nuclear design acceptable.

2.6 Transient and Accident Analyses

For Cycle 4 the most limiting anticipated operational occurrences are: load rejection without bypass, feedwater controller failure and loss of feedwater heating. Our discussion and evaluation of these occurrences is provided in Section 2.3 above.

A fuel loading error analysis has been performed for Cycle 4 using the methodology described in the approved topical report XN-NF-80-19(A) (References 9, 10 and 13). The results of the analysis show that the maximum linear heat generation rate (LHGR) is less than the TS limit for Cycle 4 and the MCPR is 1.17; i.e., the same limiting value as for the limiting LFWH transient. Therefore, we find the fuel loading error analysis results are acceptable.

The control rod withdrawal error was analyzed generically (Reference 12) and found to be applicable for Cycle 4. Finally, the full ANF core was analyzed for reduced flow and power operation to establish $MCPR_f$, $MCPR_p$, $MAPFAC_f$ and $MAPFAC_p$ limits. These limits were established in Cycles 2 and 3 and have been revised for Cycle 4. We find these limits, as proposed in the TS changes, to be acceptable.

To support the Cycle 4 operation, the results of LOCA and rod drop accident analyses were provided. The LOCA methodology is based on approved ANF topicals in References 14-16. The analysis confirmed that the peak cladding temperature remains below the 2,200°F limit of 10 CFR 50.46 for all types of fuel present in Cycle 4. Similarly, the local zirconium-water reaction remains below 17% and the core wide hydrogen production below 1.0%, the required 10 CFR 50.46 limits (Reference 2). Accordingly, the analyses are acceptable.

The rod drop accident was analyzed using ANF's generic parametric methodology for the fuel enthalpy rise during a postulated rod drop accident (References 9, 10, and 13). The results listed in Reference 2 show that the maximum deposited fuel rod enthalpy is 172 cal/gm, which is much lower than the required limit of 280 cal/gm and are acceptable.

In summary, we conclude that the transient and accident analyses were performed with approved methodology, the results are within acceptable limits and are acceptable.

2.7 9x9-5 Lead Test Assemblies

As mentioned previously, Cycle 4 includes four ANF 9x9-5 lead test assemblies (LTA). The LTA are to be placed in low-power locations in the core and are designed to have improved thermal performance. Therefore, the LTA have ample margin to the operating limits.

The licensee analyzed the performance of the LTA. The analysis methodology used was the same as for the 8x8 fuel assemblies (References 9, 10, and 13). The licensee's analysis confirmed that the LTA mechanical design meets the no-centerline melting and the 1.0% clad strain criteria. The analysis also determined that the 9x9-5 LTAs are hydraulically compatible with the 8x8 regular assemblies over the full range of the expected operating conditions. Analyses showed that no reduction in thermal margin will take place.

The nuclear design of the 9x9-5 assemblies is similar to the 8x8-2 assemblies and, therefore, has similar reactivity characteristics. The transient and accident analyses, the shutdown margin, the liquid boron control and the LTA loading error analyses were explicitly modeled for the four LTAs and demonstrate that their power is conservatively predicted.

Because the LTA will be placed in a low-power region, the licensee's analysis showed that during anticipated operational occurrences the bundle power will be lower than that required to reach transition boiling. Therefore, we agree that operational limits are adequate for the LTA.

LOCA analyses for the LTA demonstrated that these assemblies perform better than the 8x8-2 assemblies, thus, meeting the 10 CFR 50.46 limits by a larger margin (Reference 17). The consequences of the rod drop accident are governed by the rod worth. In the vicinity of the LTA, the reactivity is not different from regions loaded with all 8x8-2 assemblies. Therefore, the results are similar to the previously analyzed regions and within the required limits.

In summary, approved analysis methods for the LTA performance in Cycle 4 showed that the LTA meet the operational and accident limits. This is because the LTA are located in positions of low power and because they are neutronically and hydraulically similar to the 8x8-2 assemblies. Therefore, we find the use of the four 9x9-5 LTAs in Cycle 4 is acceptable.

2.8 Fuel Handling Accident

The licensee has requested authorization to allow fuel burnup to 39,000 megawatt days per metric ton (MWD/MT) from 33,000 MWD/MT. The staff evaluated the potential impact of burnup up to 60,000 MWD/MT on the radiological assessment of design basis accidents (DBA), which were previously analyzed in the licensing of GGNS-1.

The staff reviewed the licensee's submittals and also reviewed a publication, "Assessment of the Use of Extended Burnup Fuel in Light Water Reactors," NUREG/CR-5009, February 1988, prepared for the NRC. The NRC contractor, the Pacific Northwest Laboratory (PNL) of Battelle Memorial Institute, examined the changes to NRC DBA assumptions (described in the various appropriate SRP sections and/or Regulatory Guides) that could result from the use of extended burnup fuel (up to 60,000 MWD/MT). The contractor concluded, and the NRC staff agrees, that the only DBA that could be affected by the use of extended burnup fuel, even in a minor way, would be the potential thyroid doses that could result from a fuel handling accident. PNL estimates that I-131 fuel gap activity in the peak fuel rod with 60,000 MWD/MTU burnup could be as high as 12%. This valve is 20% higher than the 10% value normally used by the staff in evaluating fuel handling accidents (Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors").

The staff, therefore, reevaluated the fuel handling accidents for GGNS-1 with an increase in iodine gap activity in the fuel damaged in a fuel handling accident. The maximum thyroid dose from a fuel handling accident within the secondary containment as shown in the operating licensing Safety Evaluation Report dated September 1981 is 2.3 rem at the exclusion area boundary. The recalculated thyroid dose (increased by 20%) possible with extended burnup fuel of 60,000 MWD/MT is 2.8 rem.

We conclude that the only potential increased dose potentially resulting from DBA with extended fuel burnup to 60,000 MWD/MT is the thyroid dose resulting from fuel handling accidents. This small calculated increase is insignificant, in that these doses remain well within the 300 rem thyroid exposure guideline value of 10 CFR Part 100. Therefore, the requested extended burnup to 39,000 MWD/MT is acceptable.

2.9 Summary

We have reviewed the material submitted by Systems Energy Resources, Inc., for the GGNS-1 Cycle 4 operation. Based on this review, we conclude that the fuel design analysis, the thermal hydraulic design analyses and the transient and accident analyses are acceptable. The TS changes requested for this reload (listed in Section 1.0 above) reflect the necessary modifications for the operation of Cycle 4 and are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the on February 13, 1989 in the Federal Register (53 FR 6629). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

4.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration, which was published in the Federal Register (54 FR 6196) on February 8, 1989, and consulted with the State of Mississippi. No public comments or requests for hearing were received, and the State of Mississippi did not have any comments.

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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and the security, or to the health and safety of the public.

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Dated:

REFERENCES

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2. ANF-88-149, "Grand Gulf Unit 1, Cycle 4, Reload Analysis" ANF, dated November 11, 1988. (Enclosure to Reference 1)

3. ANF-88-150, "Grand Gulf Unit 1, Cycle 4, Reload Analysis" ANF, dated November 11, 1988. (Enclosure to Reference 1)

4. NESDQ-88-003 Rev. 0, "Grand Gulf Nuclear Station, Unit 1, Revised Flow Dependent Thermal Limits" MSU Systems Services, Inc., dated November 1988. (Enclosure to Reference 1)

5. XN-NF-85-67(P)(A), Rev. 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Co., dated September 1986.

6. ANF-88-183(P), "Grand Gulf Unit 1, Reload XN-1,3 Cycle 4 Mechanical Design" ANF Corporation, November 1988.

7. "Grand Gulf Nuclear Station, Maximum Extended Operating Domain Analysis" General Electric Company, dated March 1986.

8. XN-NF-81-51(A), "LOCA-Seismic Structural Response of an ENC BWR Jet Pump Fuel Assembly" Exxon Nuclear Company, dated May 1986.

9. XN-NF-80-(A), Vol. 3, Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology" Exxon Nuclear Company, dated January 1987.

10. XH-NF-80-19(A), Vol. 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads" Exxon Nuclear Company, dated June 1985.

11. XN-NF-524(A), Revision 1, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors," Exxon Nuclear Company, dated November 1983.

12. XN-NF-825A, Supplement 2, "BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$ for Plant Operations Within the Extended Operation Domain," Exxon Nuclear Company, Inc., dated October 1986.

13. XN-NF-80-19PA, Volume 1, Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors: Neutronic Methods for Design and Analysis," Exxon Nuclear Company, dated March 1983.

14. XN-NF-80-19(A), Vols., 2, 2A, 2B, & 2C, "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," Exxon Nuclear Company, dated September 1982.

15. XN-NF-CC-33(A), Rev. 1, "HYXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option," Exxon Nuclear Company, dated November 1975.

16. XN-NF-82-07(A), Rev. 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, dated November 1982.

17. ANF-88-152, "Grand Gulf Unit 1 SN-1.3 Cycle 4 Design Report Mechanical, Thermal Hydraulic and Neutronic Design for Advanced Nuclear Tests 9x9-5 Leads," dated September 1988.