October 24, 1986

Docket No.: 50-416

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Mr. Oliver D. Kingsley, Jr.
Vice President, Nuclear Operations
Mississippi Power & Light Company
Post Office Box 23054
Jackson, Mississippi 39205

Dear Mr. Kingsley:

SUBJECT: LICENSE AMENDMENT REGARDING EXXON FUEL RELOAD

RE: GRAND GULF NUCLEAR STATION, UNIT 1

The Commission has issued the enclosed Amendment No. 23 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated July 14, 1986, as amended August 15, September 4, and September 5, and supplemented October 3, 1986.

This amendment changes the Technical Specifications for operation with new Exxon fuel assemblies replacing spent General Electric fuel assemblies in the core.

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

Sincerely,

Original signed by

Lester L. Kintner, Project Manager
BWR Project Directorate No. 4
Division of BWR Licensing

Enclosures:
1. Amendment No. 23 to License No. NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

Previously concurred*:

not required

PD#4/LA*	PD#4/PM*	~~FOB/D~~	OGC*	PD#4/D
MO'Brien	LKintner:lb	~~DVassallo~~	Young	WButler
10/22/86	10/17/86	~~/ /86~~	10/17/86	10/23/86

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PDR ADOCK 05000416
P PDR

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

October 24, 1986

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Vice President, Nuclear Operations
Mississippi Power & Light Company
Post Office Box 23054
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L L Kintner

Lester L. Kintner, Project Manager
BWR Project Directorate No. 4
Division of BWR Licensing

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2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Oliver D. Kingsley, Jr.
Mississippi Power & Light Company

Grand Gulf Nuclear Station

cc:

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

MISSISSIPPI POWER & LIGHT COMPANY
MIDDLE SOUTH ENERGY, INC.
SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION
DOCKET NO. 50-416
GRAND GULF NUCLEAR STATION, UNIT 1
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 23
License No. NPF-29

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

   A. The application for amendment by Mississippi Power & Light Company, Middle South Energy, Inc., and South Mississippi Electric Power Association, (the licensees) dated July 14, 1986 as amended August 15, September 4, and September 5, and supplemented October 3, 1986, 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:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 23, are hereby incorporated into this license. Mississippi Power & Light Company 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

Original signed by

Walter R. Butler, Director
BWR Project Directorate No. 4
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 24, 1986

PD#4/LA	PD#4/PM	OGC	PD#4/D
MO'Brien	LKintner:lb	MYoung	WButler
10/22/86	10/17/86	10/17/86	10/23/86

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

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
BWR Project Directorate No. 4
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 24, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 23

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. Overleaf page(s) provided to maintain document completeness.*

Remove	Insert
1-1	1-1*
1-2	1-2
B 2-1	B 2-1*
-	B 2-1a
B 2-2	B 2-2
B 2-3	B 2-3
B 2-4	B 2-4
3/4 1-1	3/4 1-1*
3/4 1-2	3/4 1-2
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-2a	3/4 2-2a
3/4 2-2b	3/4 2-2b*
3/4 2-7	3/4 2-7
-	3/4 2-7a
3/4 4-1b	3/4 4-1b*
3/4 4-1c	3/4 4-1c
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2*
B 3/4 2-1	B 3/4 2-1
B 3/4 2-1a	-
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4	B 3/4 2-4

Remove	Insert
B 3/4 2-5	B 3/4 2-5*
B 3/4 2-6	B 3/4 2-6
B 3/4 2-6a	-
B 3/4 2-7	B 3/4 2-7
B 3/4 3-7	B 3/4 3-7
B 3/4 3-8	B 3/4 3-8*
B 3/4 4-1	B 3/4 4-1
B 3/4 4-1a	B 3/4 4-1a*

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE PLANAR EXPOSURE

1.2 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.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (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.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

2.1 SAFETY LIMITS

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 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 is established by other means. This is 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. Analyses show that with a bundle flow of 28×10^3 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 28×10^3 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.

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. Once specific uncertainty included in the safety limit is the uncertainty inherent in the XN-3 critical power correlation. ENC 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.

Bases Table B 2.1.2-1

[DELETED]

Bases Table B 2.1.2-2

[DELETED]

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

a. 0.38% delta k/k with the highest worth rod analytically determined, or
b. 0.28% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS* and other activities that could reduce the SHUTDOWN MARGIN and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.1 The SHUTDOWN MARGIN shall be determined to be equal to or greater than specified at any time during the fuel cycle:

a. By measurement, prior to or during the first startup after each refueling.

b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.

c. Within 12 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

*Except movement of IRMs, SRMs or special moveable detectors.

REACTIVITY CONTROL SYSTEMS

3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITION FOR OPERATION

3.1.2 The reactivity difference between the monitored core k_{eff} and the predicted core k_{eff} shall not exceed 1% delta k/k.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the reactivity difference greater than 1% delta k/k:

a. Within 12 hours, perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

b. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.2 The reactivity difference between the monitored core k_{eff} and the predicted core k_{eff} shall be verified to be less than or equal to 1% delta k/k:

a. During the first startup following CORE ALTERATIONS, and

b. At least once per 1000 MWD/T during POWER OPERATION.

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 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 fuel types 8CR210 and 8CR160 - 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 fuel type XN-1 the limit determined in "a" above for fuel type 8CR210.

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 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.

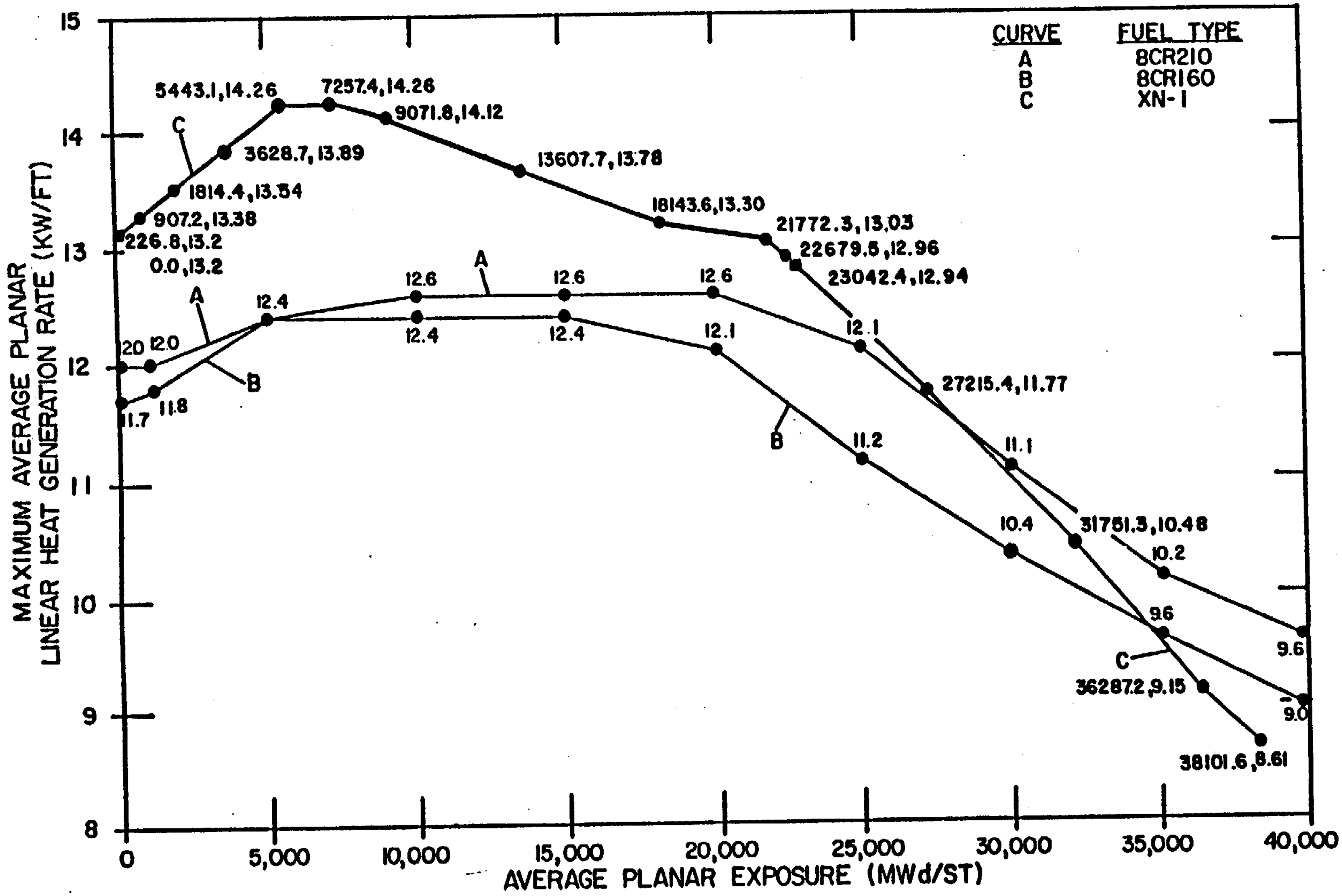


FIGURE 3.2.1-1 MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE FOR CORE FUEL TYPES 8CR210, 8CR160, AND XN-1

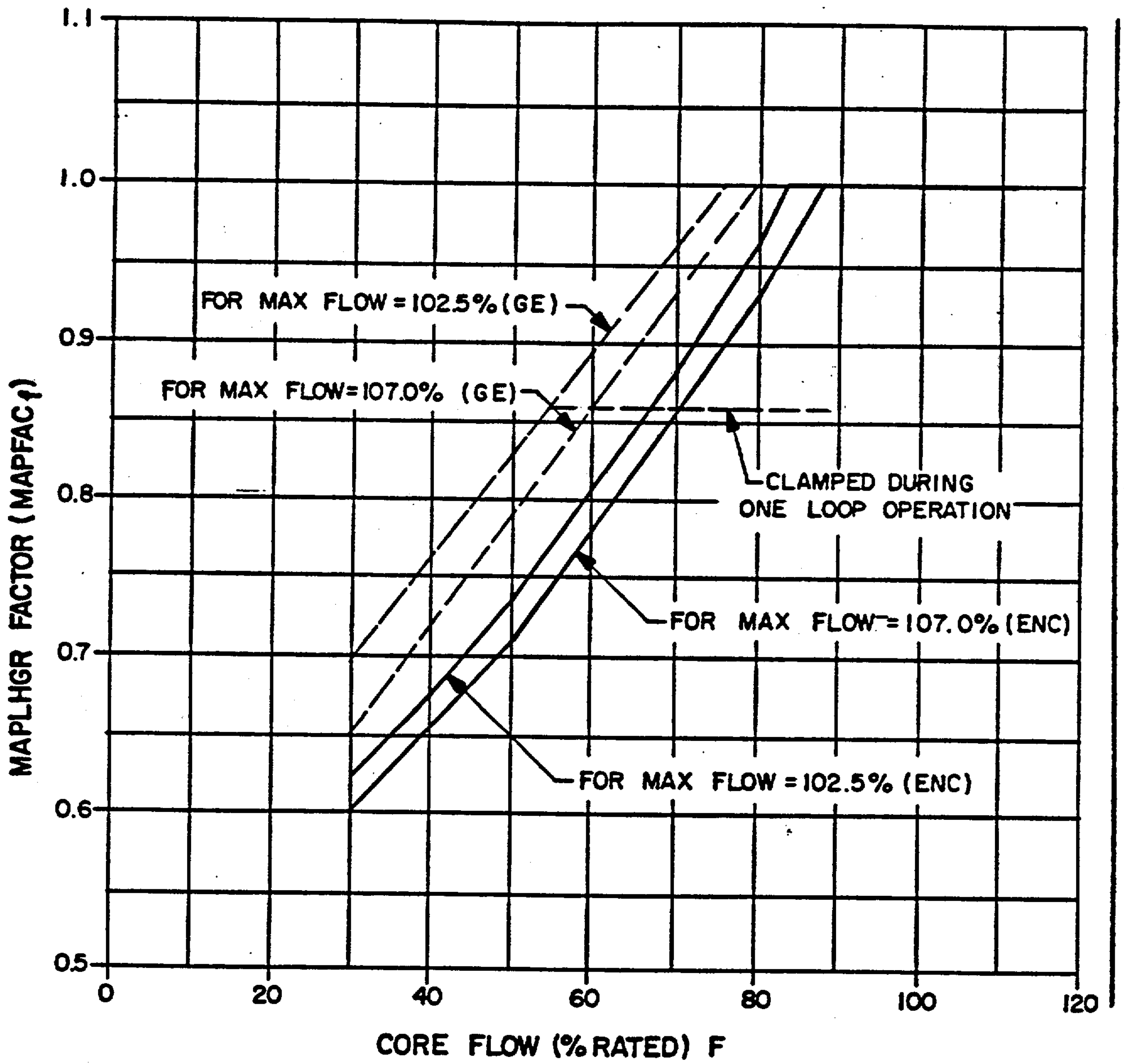


FIGURE 3.2.1-2 MAPFAC$_f$

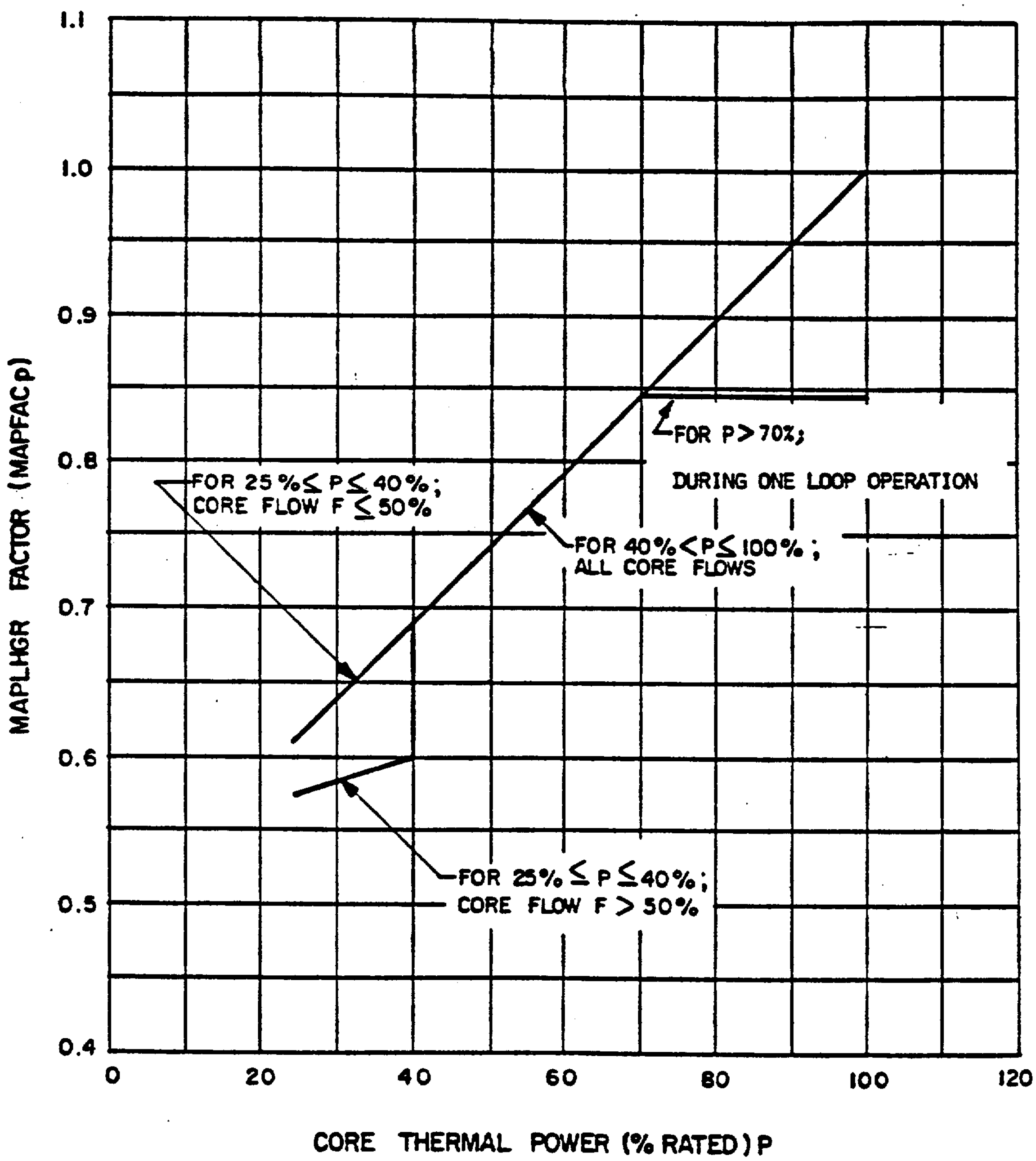


FIGURE 3.2.1-3 $MAPFAC_p$

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) for GE fuel shall not exceed 13.4 kw/ft.

The LINEAR HEAT GENERATION RATE (LHGR) for ENC fuel 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 GE fuel rod exceeding the 13.4 Kw/ft limit or with the LHGR of any ENC 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 of both GE fuel and ENC fuel 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.

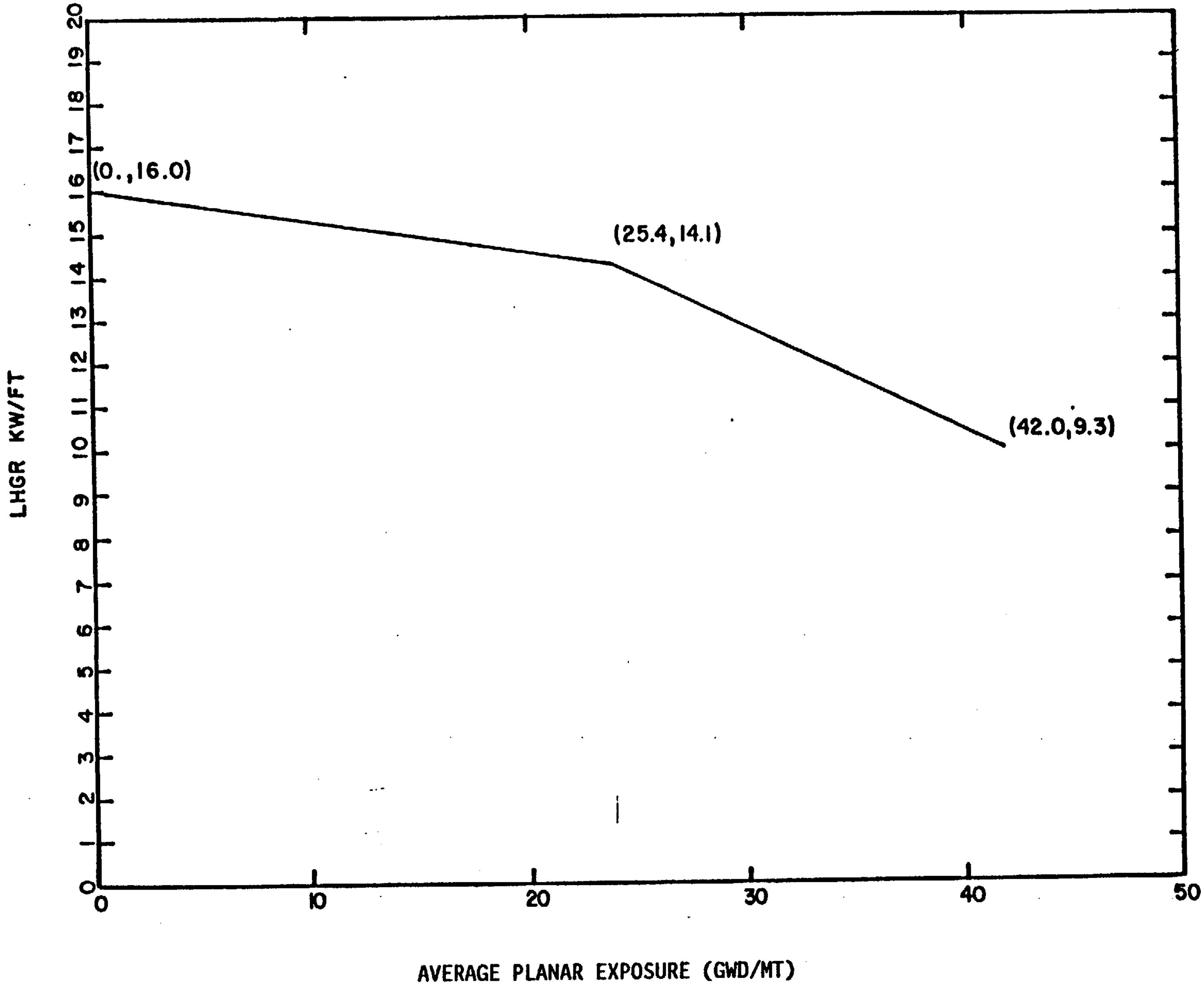


FIGURE 3.2.4-1 LINEAR HEAT GENERATION RATE (LHGR) LIMIT VERSUS AVERAGE PLANAR EXPOSURE EXXON 8x8 FUEL

SURVEILLANCE REQUIREMENTS (Continued)

The differential temperature requirements 4.4.1.1.5.b and c do not apply when the loop not in operation is isolated from the reactor pressure vessel.

4.4.1.1.6 The limits and setpoints of Specifications 2.2.1, 3.2.1, and 3.3.6 shall be verified to be within the appropriate limits within 8 hours of an operational change to either one or two loops operating.

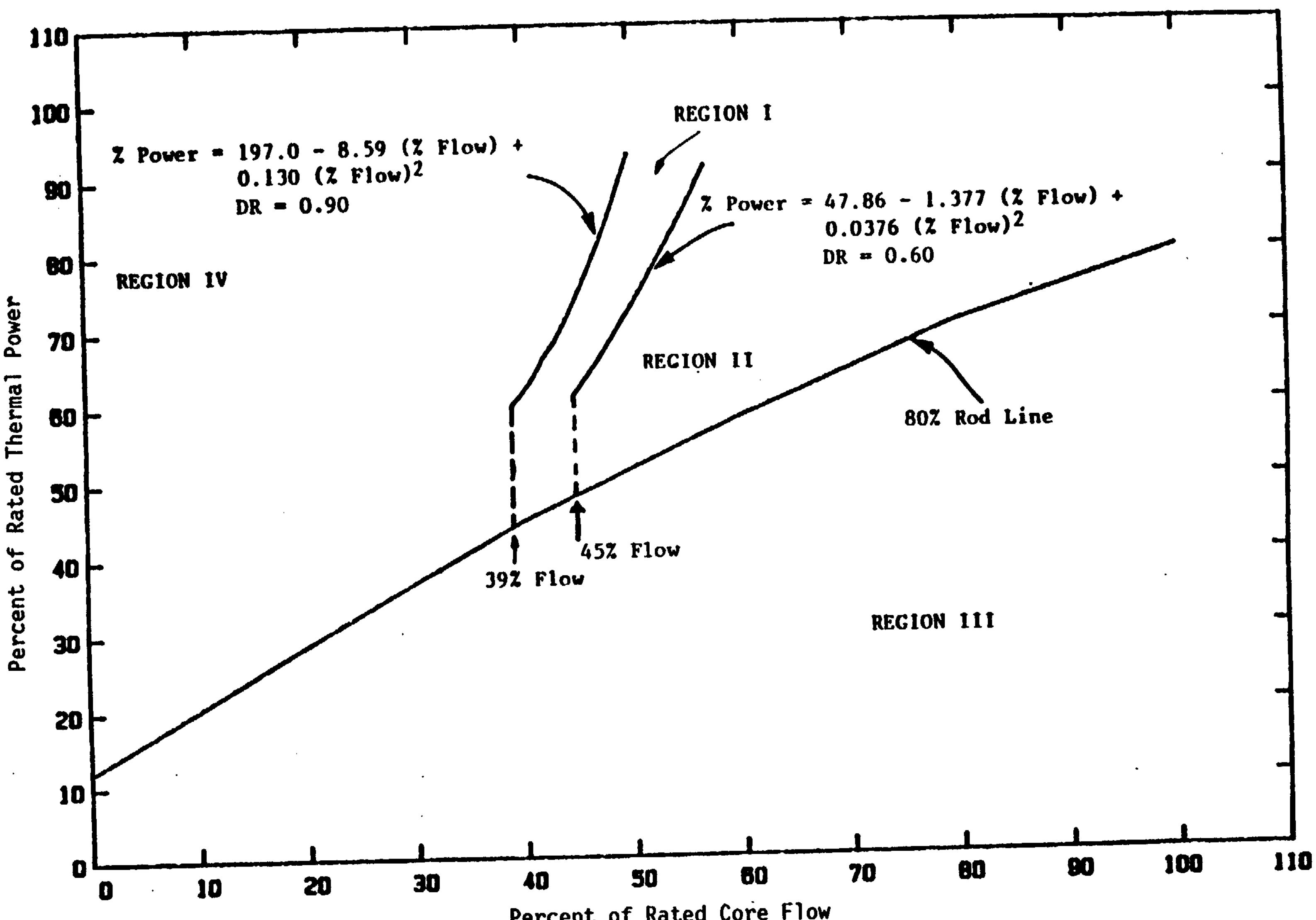


FIGURE 3.4.1.1-1 POWER-FLOW OPERATING MAP

3/4.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold, xenon-free condition and shall show the core to be subcritical by at least R + 0.38% delta k/k or R + 0.28% delta k/k, as appropriate. The value of R in units of % delta k/k is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of R must be positive or zero and must be determined for each fuel loading cycle.

Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of demonstration of the SHUTDOWN MARGIN. The highest worth rod may be determined analytically or by test. The SHUTDOWN MARGIN is demonstrated by an insequence control rod withdrawal at the beginning of life fuel cycle conditions, and, if necessary, at any future time in the cycle if the first demonstration indicates that the required margin could be reduced as a function of exposure. Observation of subcriticality in this condition assures subcriticality with the most reactive control rod fully withdrawn.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion.

3/4.1.2 REACTIVITY ANOMALIES

Since the SHUTDOWN MARGIN requirement is small, a careful check on actual reactor conditions compared to the predicted conditions is necessary. Any changes in reactivity from that of the predicted core k_{eff} can be determined from the monitored core k_{eff} using the core monitoring system. In the absence of any deviation in plant operating conditions or reactivity anomaly, these values should be essentially equal since the calculational methodologies are consistent. The predicted core k_{eff} is calculated by a 3D core simulation code as a function of cycle exposure. This is performed for projected or anticipated reactor operating states/conditions throughout the cycle and is usually done prior to cycle operation. The monitored core k_{eff} is that calculated by the core monitoring system for actual plant conditions.

A deviation in reactivity of more than 1% from that predicted is larger than expected for normal operation, and therefore, should be thoroughly evaluated.

3/4.1.3 CONTROL RODS

The specifications of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the accident, non-accident and transient analyses, and (3) the potential effects of the rod drop accident and rod withdrawal error event are limited. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable but trippable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shut down for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than the Safety Limit during the limiting power transient analyzed in Section 15.4 of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than the Safety Limit. The occurrence of scram times longer than those specified should be viewed as an indication of a systematic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be slowly scrammed via reactor pressure or inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

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 Figure 3.2.1-1 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. The maximum factor (MAPFAC) for single loop operation is 0.86.

For single-loop operation with ENC 8x8 fuel, a MAPLHGR limit corresponding to the product of the highest enriched GE fuel MAPLHGR, and the appropriate MAPFAC, can be conservatively used, provided that the average planar exposure is limited to 25,000 MWD/ST.

$MAPFAC_f$'s are determined using the three-dimensional BWR simulator code to analyze slow flow runout transients. Two curves for each fuel vendor are provided for use based on the existing setting of the core flow limiter in the Recirculation Flow Control System. The curve representative of a maximum core flow limit of 107.0% is more restrictive due to the larger potential flow runout transient.

$MAPFAC_p$'s are generated using the same data base as the $MCPR_p$ 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

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

rod changes. The requirement to calculate APLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating APLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that APLHGR will be known following a change in THERMAL POWER or power shape, that could place operation exceeding a thermal limit.

The calculational procedure used to establish the APLHGR limits is based on a loss-of-coolant accident analysis. The analysis was performed using calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. These models are described in references 1, 6, and 8.

3/4.2.2 [DELETED]

Bases Table B 3.2.1-1

[DELETED]

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.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-2 and in Table 15.C.3-1 of Reference 5 that are input to a GE-core dynamic behavior transient computer program. The evaluation of transients during operation in the MEOD begins with the system initial parameters shown in Tables 15.D.4.2 and 3 of Reference 7. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle. The principal result of this evaluation is the reduction in MCPR caused by the transient.

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). The maximum runout flow value is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System. Two flow rates have been considered, 102.5% core flow and 107.0% core flow (for Increased Core Flow operation). 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 rod-line 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

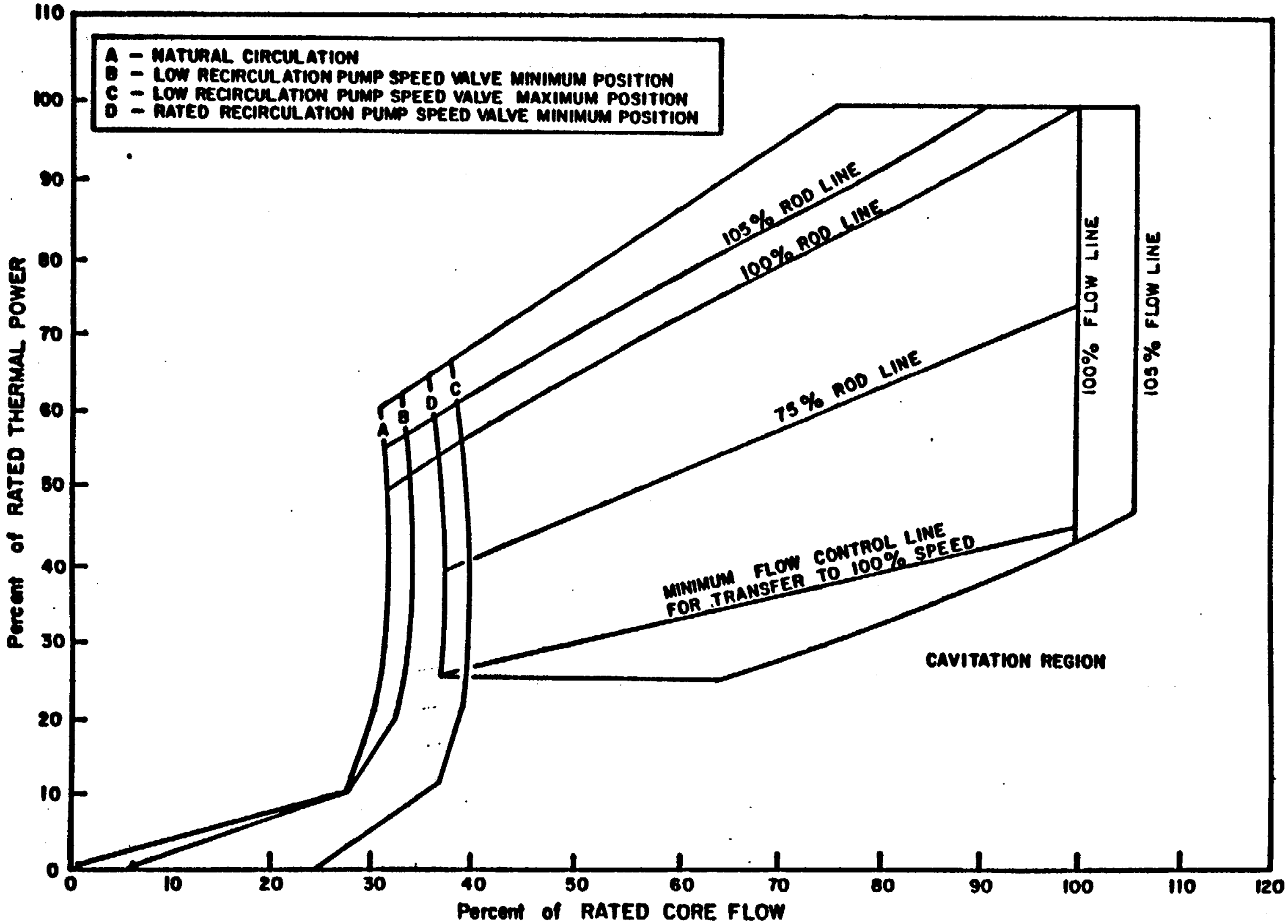


FIGURE B 3/4 2.3-1 POWER - FLOW OPERATING MAP

MINIMUM CRITICAL POWER RATIO (Continued)

to be in equilibrium, and (b) core xenon concentration 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.

The first state analyzed corresponded to the maximum core power at maximum core flow (either 102.5% for Rated Core Flow operation or 107% of rated for Increase Core Flow operation) after the flow runout. Several evaluations were performed at this state iterating on the normalized core power distribution input until the limiting bundle MCPR just exceeded the safety limit Specification (2.1.2). Next, similar calculations of core MCPR performance were determined at other power/flow conditions on the generic flow control line, assuming the same normalized core power distribution. The result is a definition of the $MCPR_f$ performance requirement such that a flow increase event to the maximum flow will not violate the safety limit. (The assumption of constant power distribution during the runout power increase has been shown to be conservative. Increased negative reactivity feedback in the high power limiting bundle due to doppler and voids would reduce the limiting bundle relative power in an actual runout.)

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 GESSAR II, Appendix 15B. 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. The current $MCPR_p$ limits were found to be bounding. These $MCPR_p$ limits have been validated for use during Cycle 2. 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.

MINIMUM CRITICAL POWER RATIO (Continued)

During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR 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 rod changes. The requirement to calculate MCPR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating MCPR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that MCPR will be known following a change in THERMAL POWER or power shape, that could place operation exceeding a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

The daily requirement for calculating LHGR 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 rod changes. The requirement to calculate LHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating LHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that LHGR will be known following a change in THERMAL POWER or power shape that could place operation exceeding a thermal limit.

References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, November 1975.
2. [DELETED]
3. [DELETED]
4. [DELETED]
5. GGNS Reactor Performance Improvement Program, Single Loop Operation Analysis, General Electric Final Report, February 1986.
6. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, Amendment 2, One Recirculation Loop Out-of-Service, NEDO-20566-2, Revision 1, July 1978.
7. General Electric Company, "Maximum Extended Operating Domain Analysis," March 1986.
8. XN-NF-80-19(A), Volume 2 "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," Exxon Nuclear Company, September 1982.

3/4.3.9 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety-related components, equipment or structures.

3/4.3.10 NEUTRON FLUX MONITORING INSTRUMENTATION

This specification is to ensure that neutron flux limit cycle oscillations are detected and suppressed.

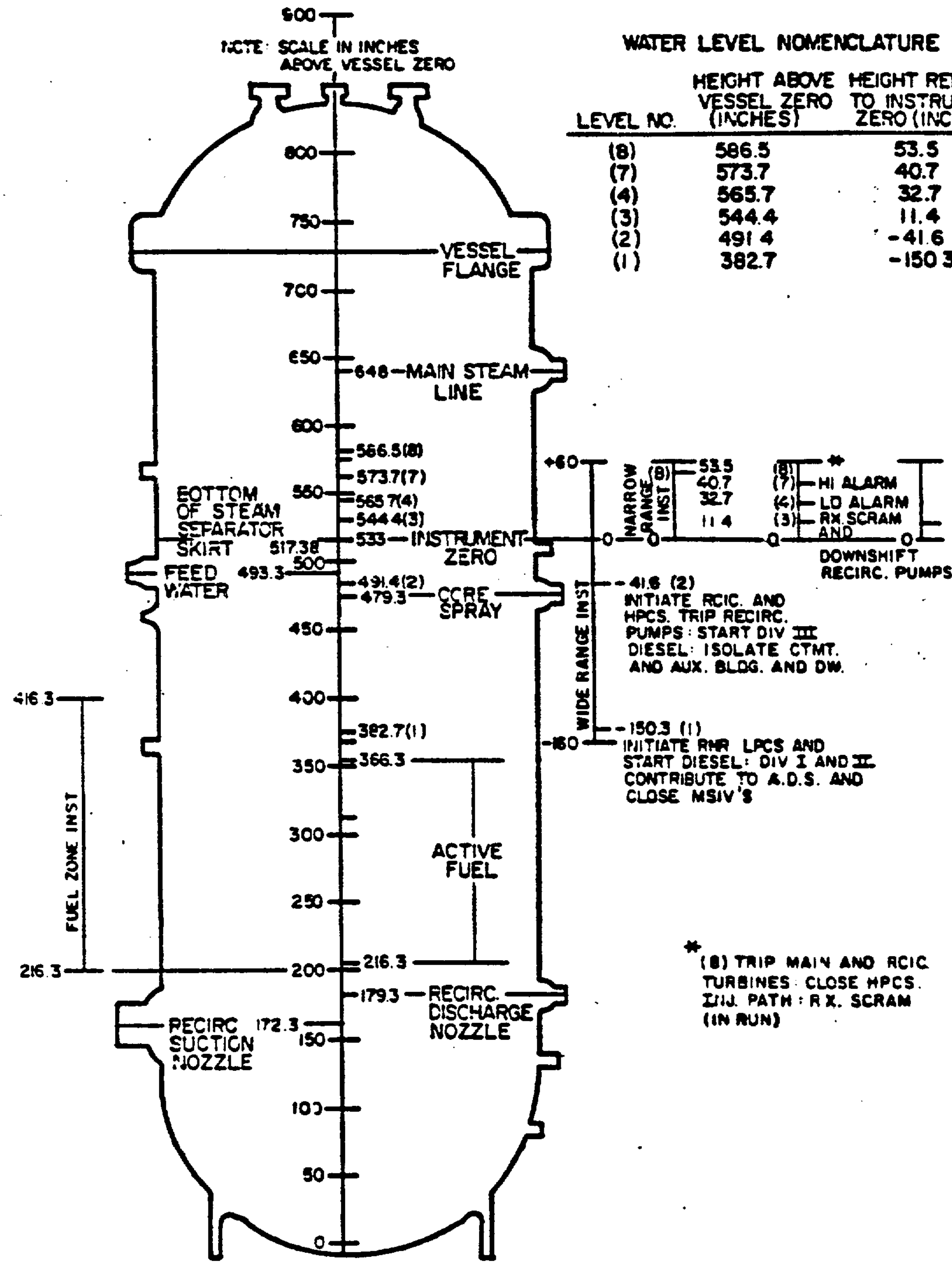
In order to identify a region of the operating map where surveillance should be performed, analyses were performed by Exxon Nuclear Company for the Grand Gulf reactor consistent with the USNRC approved methodology as described in XN-NF-691(P)(A) dated August 1984.

The surveillance region was established as that region for which the calculated decay ratio is greater than or equal to a value of 0.60 and less than 0.90, between the 39% and 45% flow line. The resulting region is illustrated in Figure 3.4.1.1-1 and is identified as Region I.

Region IV is restricted from operation. This is the region where either the calculated decay ratio is greater than 0.90 or flow is below the 39% flow line.

In Region III, below the 80% rod line, and Region II, where the decay ratio is below 0.60, no detect and suppress surveillance activities are required.

Neutron flux noise limits are also established to ensure the early detection of limit cycle oscillations. Typical APRM neutron flux noise levels at up to 12% of rated power have been observed. These levels are easily bounded by values considered in the thermal/mechanical fuel design. Stability tests have shown that limit cycle oscillations result in peak-to-peak magnitude of 5 to 10 times the typical values. Therefore, actions taken to suppress flux oscillations exceeding three (3) times the typical value are sufficient to ensure early detection of limit cycle oscillations. The specification includes the surveillance requirement to establish the requisite baseline noise data and prohibits operation in the region of potential instability if the appropriate baseline data is unavailable.



WATER LEVEL NOMENCLATURE

LEVEL NO.	HEIGHT ABOVE VESSEL ZERO (INCHES)	HEIGHT RELATIVE TO INSTRUMENT ZERO (INCHES)
(8)	586.5	53.5
(7)	573.7	40.7
(4)	565.7	32.7
(3)	544.4	11.4
(2)	491.4	-41.6
(1)	382.7	-150.3

BASES FIGURE B 3/4 3-1 REACTOR VESSEL WATER LEVEL

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor core coolant recirculation loop inoperable has been evaluated and found to remain within design limits and safety margins provided certain limits and setpoints are modified. The "GGNS Single Loop Operation Analysis" identified the fuel cladding integrity Safety Limit, MAPLHGR limit and APRM setpoint modifications necessary to maintain the same margin of safety for single loop operation as is available during two loop operation. Additionally, loop flow limitations are established to ensure vessel internal vibration remains within limits. A flow control mode restriction is also incorporated to reduce valve wear as a result of automatic flow control attempts and to ensure valve swings into the cavitation region do not occur.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation. During two loop operation, recirculation loop flow mismatch limits are in compliance with ECCS LOCA analysis design criteria. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. In cases where the mismatch limits cannot be maintained, continued operation is permitted with one loop in operation.

Figure 3.4.1.1-1 describes the boundaries of the detect and suppress region as discussed in bases 3/4.3.10

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 100°F. During single loop operation, the condition may exist in which the coolant in the bottom head of the vessel is not circulating. These differential temperature criteria are also to be met prior to power or flow increases from this condition.

The recirculation flow control valves provide regulation of individual recirculation loop drive flows; which, in turn, will vary the flow rate of coolant through the reactor core over a range consistent with the rod pattern and recirculation pump speed. The recirculation flow control system consists of the electronic and hydraulic components necessary for the positioning of the two hydraulically actuated flow control valves. Solid state control logic will generate a flow control valve "motion inhibit" signal in response to any one of several hydraulic power unit or analog control circuit failure signals. The "motion inhibit" signal causes hydraulic power unit shutdown and hydraulic isolation such that the flow control valve fails "as is." This design feature insures that the flow control valves do not respond to potentially erroneous control signals.

3/4.4.1 RECIRCULATION SYSTEM (Continued)

Electronic limiters exist in the position control loop of each flow control valve to limit the flow control valve stroking rate to 10±1% per second in the opening and closing directions on a control signal failure. The analysis of the recirculation flow control failures on increasing and decreasing flow are presented in Sections 15.3 and 15.4 of the FSAR respectively.

The required surveillance interval is adequate to ensure that the flow control valves remain OPERABLE and not so frequent as to cause excessive wear on the system components.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 23 TO FACILITY OPERATING LICENSE NO. NPF-29

MISSISSIPPI POWER & LIGHT COMPANY

MIDDLE SOUTH ENERGY, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated July 14, 1986, (Ref. 1), Mississippi Power & Light Company, (MP&L or the licensee) requested an amendment to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. The proposed amendment would change Technical Specifications to allow the operation of Grand Gulf Nuclear Station Unit 1 for Cycle 2 (GG1C2) with a reload using Exxon manufactured fuel assemblies and Exxon analyses and methodologies. Enclosed with the July 14, 1986 submittal were the requested changes to the Technical Specifications and reports discussing the reload and analyses made to support and justify the second fuel cycle operation with General Electric (GE) and Exxon fuel and the changes to the Technical Specifications. By letters dated August 15 and September 5, 1986, the licensee provided supplemental information describing additional analyses and providing results. By letters dated September 4 and October 3, 1986, the licensee proposed additional changes to some of the proposed Technical Specifications. These submittals are identified in References 1 through 9.

The notice of consideration of issuance of this license amendment was published in the Federal Register before the licensees October 3, 1986, submittal. The October 3, 1986, submittal contained clarifications to previously submitted proposed Technical Specifications to make them more specific and therefore less likely to be misinterpreted. The notice of consideration accurately described the license amendment request and the clarifications do not affect the substance of the requested amendment.

Cycle 2 will be the first use of Exxon fuel and analysis in this reactor. However, similar reloads with Exxon fuel have been done for Dresden 2 and 3, and more recently for Susquehanna 1 and 2 and Washington Public Power Supply 2 (WNP2). These reloads and the associated Exxon methodologies were extensively reviewed and approved (see for example Reference 10). These methodologies are generally applicable and were used for the most part for GG1C2 analyses.

Beyond the switch to Exxon-provided reload fuel, there is little that is unusual about GG1C2, and the proposed Technical Specification changes are primarily related to the use of Exxon fuel and accompanying analyses and methodology, terminology or related operational approaches. The reload related changes are similar to the corresponding changes for the Susquehanna 1 second cycle introduction of Exxon fuel (Ref. 10) and the other Exxon reloads mentioned above. During the first cycle Grand Gulf received approval and appropriate Technical Specifications for operation in the Maximum Extended Operating Domain (MEOD) and for single recirculation loop operation. This will continue in cycle 2 and appropriate analyses have been done for this reload. The reload and its analyses will be discussed in the following evaluation.

2.0 EVALUATION

2.1 Reload Description

The GG1C2 reload will retain 536 General Electric (GE) fuel assemblies from the first cycle and will add 264 Exxon manufactured XN-1 8x8, 2.81 percent average, 2.99 percent peak radial average U235 enriched fuel assemblies. The XN-1 fuel assemblies are similar to those used in the Susquehanna 1 second cycle (S1C2) reload. The loading pattern will be a conventional scatter pattern with low reactivity fuel on the periphery.

2.2 Fuel Design

The Exxon XN-1 fuel assembly used for GG1C2 is essentially the same as that used for the S1C2 reload. There are slight differences in the fuel enrichment and gadolinium placement patterns, but the significant mechanical and thermal-hydraulic design elements are the same and power distributions are similar. The methodologies used for the fuel design and analysis are the same as those developed and approved during the S1C2 reload review and then approved for the Susquehanna 1 Cycle 3 (S1C3) reload. The design and analyses of the XN-1 fuel assembly as used in GG1C2 are thus acceptable.

For GG1C2 the Technical Specifications will provide for a Linear Heat Generation Rate (LHGR) specification as a function of fuel burnup for the Exxon fuel. A similar specification was accepted for S1C3 as a result of discussions between the NRC staff and Exxon on the need for a LHGR specification. The specification is based on the approved fuel design methodology as discussed in the S1C3 review (Ref. 11) and is acceptable.

The mechanical response of Exxon fuel assemblies to design Seismic-LOCA events is essentially the same as for GE assemblies. The channel boxes for the new fuel are GE channels. Similar to the S1C2 and S1C3 reloads the analyses indicating that the design limits are not exceeded are acceptable.

The Exxon fuel has been analyzed for operation in the high flow region of MEOD (Ref. 8). This includes evaluation of vibration and assembly levitation. Similar calculations have been approved for WNP2. The results are acceptable.

2.3 Nuclear Design

The nuclear design for GG1C2 has been performed with Exxon methodologies previously reviewed and approved, and which were listed in the review for the S1C2 reload (Ref. 10). The fuel loading pattern is a normal type of scattered configuration. The beginning of cycle shutdown margin is 4.07 percent delta k and at minimum conditions is 2.73 percent delta k, well in excess of the required 0.38 percent delta k. The Standby Liquid Control System also fully meets shutdown requirements. These and other GG1C2 nuclear design parameters have been obtained with previously approved methods and fall within expected ranges. Thus the nuclear design is acceptable.

GG1C2 will use the Exxon POWERPLEX core monitoring system to monitor reactor parameters. We have not specifically reviewed details of this system (nor have we in the past reviewed details of the GE process computer monitoring system), but we have reviewed the principal methodologies involved in the system and consider them to be appropriate and acceptable. The system has been in use in Susquehanna and has provided suitable monitoring and predictive results.

The spent fuel pool has been previously analyzed with an infinite array of 8x8 fuel assemblies with an enrichment of 3.5 percent U235 and no burnable poison. The new fuel for cycle has an axial maximum enrichment of 2.99 percent and thus falls within the limits of previously reviewed and approved levels and is acceptable.

2.4 Thermal Hydraulic Design

The Exxon thermal-hydraulic methodology and criteria used for the GG1C2 design and analysis is the same as that used and approved in the S1C2, S1C3 and WNP2 reloads. The previous reviews concluded that hydraulic compatibility between GE and Exxon fuel is satisfactory and the calculation of core bypass flow and the Safety Limit Minimum Critical Power Ratio (MCPR) are acceptable. This is also the case for GG1C2. The Safety Limit MCPR continues to be 1.06 for two recirculation loop operation (the same value as for the first cycle GE methodology) as it is for Susquehanna and WNP2 and this is acceptable. As discussed in Section 2.6 below, the Operating Limit MCPR for GG1C2 (1.18) remains the same as for Cycle 1 since the analyses for Cycle 2 result in values no larger than this value.

GG1C2 already has Technical Specifications from Cycle 1 allowing and controlling one recirculation loop operation, including changes required on limits for Maximum Average Planar Linear Heat Generation Rate (MPLHGR) , Average Power Range Monitor (APRM) settings, and Safety Limit MCPR. Since the Exxon fuel is hydraulically compatible with the GE fuel, the previous analyses are also applicable to the Exxon XN-1 fuel loading. Similar to the approval for the Susquehanna one loop operation review (Ref. 10), the above first cycle one loop limit changes are also acceptable for this Exxon reload.

Grand Gulf also has Technical Specifications approved during the first cycle for Thermal-Hydraulic Stability surveillance and the subsequent suppression of possible oscillations. These specifications set up regions on the power-flow map in which operation is not permitted or regions in which detection of potential power oscillations must be performed using the incore neutron detector system. Because of the use of Exxon fuel and methodology for Cycle 2, changes have been made to the specification of these regions. There have been changes to Technical Specification Figure 3.4.1.1-1 and to relevant Bases (Ref. 6 and 7, with the figure in 7 superseding that of 6). These changes were the result of several discussions between the staff and Grand Gulf. The change provides for a Detect and Suppress region above the 80 percent load line and bounded by the 39 and 45 percent flow lines and lines representing analytical Decay Ratios of 0.90 and 0.60. Operation is not permitted at flows below 39 percent or above the Decay Ratio of 0.90. These limits are based on the staff interpretation of previous reviews of relevant Exxon methodology (Ref. 13) and approval of GE recommended surveillance mode of operation presented in References 14 and 15. This final version of the specification (Ref. 7) is acceptable. Grand Gulf will perform tests on stability during startup of Cycle 2 in cooperation with staff consultants from Oak Ridge. The above regions may be altered in the future as a result of these tests and corresponding analyses.

2.5 Transient and Accident Analyses

The GG1C2 core will have 800 fuel assemblies, including 264 unirradiated Exxon XN-1 8x8 assemblies and 536 previously irradiated GE assemblies from Cycle 1. The Exxon transient and accident analyses for Cycle 2 were based on the design and operational assumptions used for the analyses of Cycle 1.

References 1 through 5 and 16 through 21 describe the Exxon methodology used in the analysis of the plant transients and accidents for the Cycle 2 reload. The Exxon Cycle 2 Reload Analysis Report (Ref. 2) which describes plant and cycle specific analysis results is supplemented by Reference 16 which describes the Exxon approach to core reload analyses and references more detailed methods reports used in the safety analyses.

Core wide transients were analyzed with the COTRANSA computer code (Ref. 17) which includes a one-dimensional neutron kinetics model for evaluation of the axial power shape response during transient events. This reference has been reviewed by the staff. The methods for calculating the system transient response were found to be acceptable. Preparation of the Safety Evaluation Report (SER) and formal staff approval of this reference is in process.

Calculation of the change in Critical Power Ratio (CPR) during the core wide transient events involves the use of COTRANSA system results which serve as input to a hot channel analysis model used to calculate the delta-CPR values. The original submittal for core-wide transient results for Cycle 2 were based on a COTRANSA hot channel model. During an internal Exxon review, a potential nonconservatism was identified in the formulation of this model for delta-CPR calculations. To resolve the problem, this potential nonconservatism was evaluated by the licensee with the XCOBRA-T model (Ref. 18). The XCOBRA-T model has been reviewed by by the staff and found to be acceptable. Writing of the SER for this model is in process. As discussed below, the application of X-COBRA-T to the plant-specific GG1C2 results confirmed the COTRANSA hot channel model results. Hence the application of the COTRANSA results to Cycle 2 of Grand Gulf Unit 1 is acceptable.

In the initial submittal, the licensee referenced a generic report (Ref. 19) as the basis for the predicted response of Grand Gulf Unit 1 to a loss of feedwater heating (LFWH) event during Cycle 2. This generic report had not been reviewed by the staff. Hence, the staff requested formal submittal of a plant specific analysis of the event. This analysis, which was provided in Reference 8 is acceptable.

The rod withdrawal error (RWE) event for Cycle 2 of Grand Gulf Unit 1 was referenced to a generic study of this event provided in Reference 20. The staff has reviewed and evaluated this report which provides a statistical evaluation of the RWE and includes application to operation in the maximum extended operating domain (MEOD). The staff has found the report acceptable (Ref. 21).

The licensee evaluation of the Loss of Coolant Accident (LOCA) for Grand Gulf Unit during Cycle 2 is summarized in Reference 1. The analysis of the limiting break is provided in Reference 5. The LOCA analysis methodology used to obtain the results of Reference 5 has been approved by the staff. However, Reference 5 cites the Exxon generic jet pump BWR/6 LOCA break spectrum analysis of Reference 4 which was not reviewed by the staff. The staff requested a plant-specific break spectrum analysis that would be applicable to Cycle 2. This information which was provided in Reference 8, is acceptable.

2.6 Changes To Technical Specifications

The following changes have been requested for Grand Gulf Technical Specifications (TSs) and Bases (B) to accommodate the change to Exxon fuel, methodology and terminology. For the most part these changes are similar to those approved for Susquehanna and WNP2 on changing to Exxon methodology.

(1) The definition 1.8 for Critical Power Ratio is changed (see Ref. 9) to reflect the change of methodologies for GEXL to XN-3, and is acceptable.

(2) TS 3/4.1.2: The change to the definition of reactivity anomaly from control rod density to a monitored k_{eff} anomaly, reflects the use of a more direct parameter. POWERPLEX, which maintains a consistent methodology between active determination and prediction, can monitor k_{eff} directly. The change is acceptable.

(3) TS 3/4.2.1: The language of this specification has been changed (see both Ref. 1 and 9) to reference more explicitly and clearly Figures 3.2.1-1, 2 and 3 and indicate two and one loop operation limits and multiplication factors. This is acceptable.

(4) Figure 3.2.1-1: The MAPLHGR curve for the GE low enrichment fuel assembly from the previous cycle, which is removed for Cycle 2, has been eliminated and replaced by the curve for the new Exxon fuel assembly. This curve is based on the Exxon LOCA calcuations and is acceptable.

(5) Figure 3.2.1-2: $MAPFAC_f$ curves for the Exxon fuel assembly have been added to the existing curves for GE fuel. These are based on the Exxon calculations of flow increase transients and are acceptable.

(6) TS 3/4.2.4: This specification has been changed to add a reference to Figure 3.2.4-1 giving the Linear Heat Generation Rate (LHGR) curve for Exxon fuel. The GE fuel LHGR remains at 13.4 kw/ft. This is acceptable.

(7) Figure 3.2.4-1 has been added to give the Exxon fuel LHGR limit as a function of burnup. This is similar to previously approved curves for Susquehanna and WNP2. It is based on approved methods for Exxon fuel mechanical design analyses, and is compatible via the local peaking factor with the MAPLHGR limit of Figure 3.2.1-1 (Ref. 2). It is acceptable.

(8) Figure 3.4.1.1-1: As discussed in Section 2.4 on thermal-hydraulic stability, the power-flow plane regions requiring monitoring or no operation have been changed. The final version of the figure giving these regions was presented in Reference 7. It is acceptable.

(9) The following Bases have been changed to provide a description of Exxon methodology in addition to or in place of GE methodology. The changes are consistent with the Technical Specification changes and are similar to changes approved in the Susquehanna and WNP2 reviews. They are all acceptable.

B2.1, discusses the Exxon methodology relating to the Xn-3, critical power correlation, particularly at low flow and pressure, and the Exxon Safety Limit (MCPR).

B3/4.1.2, discusses Exxon methodology for MAPLHGR including MAPFAC.

B3/4.2.3, discusses Exxon methodology for MCPR.

B3/4.2 (References), deletes several GE references and adds an Exxon reference.

B3/4.3.10 and B3/4.4.1 discusses the surveillance regions of Figure 3.4.1.1-1 for thermal-hydraulics stability (see Ref. 6 and 7).

2.7 Summary

The NRC staff has reviewed the reports submitted for the Cycle 2 reload of Grand Gulf Unit 1 with Exxon fuel and with Exxon methodology and analysis. Based on this review the staff concludes that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design and transient and accident analyses are acceptable. The changes to the Technical Specifications submitted for this reload suitably reflect the changes from GE to Exxon methodology and the operating limits associated with these changes and reload parameters.

2.8 References

1. Letter from O. D. Kingsley, MPLC, to H. Denton, NRC, dated July 14, 1986, "Grand Gulf Nuclear Station Unit 1, Proposed Amendment to the Operating License, Cycle 2 Reload."

2. XN-NF-86-35 Rev. 3, dated August 1986, "Grand Gulf Unit 1 Cycle 2 Reload Analysis."

3. XN-NF-86-36 Rev. 3, dated August 1986, "GG1C2 Plant Transient Analysis."

4. XN-NF-86-37, dated April 1986, "Generic LOCA Break Spectrum Analysis for BWR/6 Plants."

5. XN-NF-86-38, dated June 1986, "Grand Gulf Unit 1 LOCA Analysis."

6. Letter from O. D. Kingsley, MPLC, to H. Denton, NRC, dated August 15, 1986, "Addendum to Cycle 2 Reload."

7. Letter from O. D. Kingsley, MPLC to H. Denton, NRC, dated September 4, 1986, "Addendum #2 to Cycle 2 Core Stability."

8. Letter from O. D. Kingsley, MPLC to H. Denton, NRC, dated September 5, 1986, "Additional Information (LOFWH, LOCA, Fuel Liftoff)."

9. Letter from O. D. Kingsley, MPLC to H. Denton, NRC, dated October 3, 1986, "Supplement to Cycle 2 Reload."

10. Letter from W. Butler, NRC, to N. W. Curtis, Pennsylvania Power & Light (PP&L), dated May 22, 1985.

11. Letter from E. Adensam, NRC, to H. W. Keiser, PP&L, dated April 11, 1986, "Amendment to Susquehanna Unit 2 for Cycle 3 Reload."

12. Letter from E. Adensam, NRC, to H. W. Keiser, PP&L, dated April 11, 1986, "Amendment Nos. 56 and 26 to Susquehanna Steam Electric Station Units 1 and 2."

13. Letter from C. Thomas, NRC, to J. Chandler, Exxon, dated May 10, 1984, "Stability Evaluation of Boiling Water Reactor Cores."

14. General Electric Service Information Letter No. 380, Revision 1, February 10, 1984.

15. Technical Resolution of Generic Issue B-9 Thermal Hydraulic Stability (Generic Letter No. 86-02), dated January 22, 1986.

16. XN-NF-79-71(P) Revision 2, dated November 1981, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors."

17. "Exxon Nuclear Methodology for Boiling Water Reactors: THERMEX Thermal Limits Methodology, Summary Description," XN-NF-80-19(P), Volume 3, Revision 1, Exxon Nuclear Company, Richland, Washington (April 1981).

18. XN-NF-84-105(P) dated May 1985, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis."

19. XN-NF-900(P) dated February 1986, "A Generic Analysis of the Loss of Feedwater Heating Transient for Boiling Water Reactors."

20. XN-NF-825(P) dated April 1985, "BWR/6 Generic Rod Withdrawal Error Analysis."

21. Letter from G. C. Lainas, NRC, to G. N. Ward, Exxon, dated October 14, 1986, "Acceptance for Referencing of Licensing Topical Report XN-NF-825(P), Supplement 2, BWR/6 Generic Rod Withdrawal Error Analysis, MCPRp for Plant Operations Within the Extended Operatign Domain."

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes to requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (51 FR 33955) on September 24, 1986, and consulted with the state of Mississippi. No public comments 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 nor to the health and safety of the public.

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