October 19, 1987

Docket No. 50-458

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Mr. James C. Deddens
Senior Vice President, (RBNG)
Gulf States Utilities
P. O. Box 220
St. Francisville, LA 70775
ATTN: Nuclear Licensing

Dear Mr. Deddens:

SUBJECT: RIVER BEND STATION, UNIT 1, AMENDMENT NO. 12 TO FACILITY OPERATING LICENSE NO. NPF-47 (TAC NO. 65881)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 12 to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated August 14, 1987.

This amendment modifies the TSs for Cycle 2 fuel reload and operation.

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

Sincerely,

/s/

Walter A. Paulson, Project Manager
Project Directorate - IV
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 12 to License No. NPF-47
2. Safety Evaluation

cc w/enclosures:
See next page

LTR NAME: RIVER BEND CYCLE 2 RELOAD PKG.

PD4/LA	PD4/PM	OGC-Bethesda	PD#4/D
PNoonan	WPaulson:		JCalvo
10/5/87	10/0V/87	10/15/87	10/19/87

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River Bend Nuclear Plant

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

GULF STATES UTILITIES COMPANY

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 12
License No. NPF-47

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

   A. The application for amendment filed by Gulf States Utilities Company, dated August 14, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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-47 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 12 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. GSU 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

José A. Calvo

Jose A. Calvo, Director
Project Directorate - IV
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 19, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 12

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

Replace the following page of the Appendix "A" Technical Specifications with the enclosed page. The revised page is identified by Amendment number and contains a vertical line indicating the area of change. Overleaf page provided to maintain document completeness.

REMOVE	INSERT
2-1	2-1
B2-1	B2-1
B2-2	B2-2
B2-3	-
B2-4	-
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
-	3/4 2-6a
B3/4 1-2	B3/4 1-2
B3/4 2-1	B3/4 2-1
B3/4 2-2	B3/4 2-2
B3/4 2-4	B3/4 2-4
B3/4 2-5	B3/4 2-5

ADMINISTRATIVE CONTROLS

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.07 and the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

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

SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

2.1.4 The reactor vessel water level shall be above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4 and 5

ACTION:

With the reactor vessel water level at or below the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required. Comply with the requirements of Specification 6.7.1.

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 such that the MCPR is not less than 1.07. MCPR greater than 1.07 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 GE Critical Power correlation (Reference 1) 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 28,000 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 28,000 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 fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using a statistical model that combines all of the uncertainties in the operating parameters and in the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical power correlation. Details of the fuel cladding integrity safety limit calculation are given in Reference 1. Reference 1 includes a tabulation of the uncertainties used in the determination of the Safety Limit MCPR and of the nominal values of parameters used in the Safety Limit MCPR statistical analysis.

Reference

1. "General Electric Standard Application for Reactor Fuel (GESTAR)," NEDE-24011-P-A-8.

(DELETED)

(DELETED)

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 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 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, and 3.2.1-6.

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

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5 or 3.2.1-6, 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 limits determined from Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5 and 3.2.1-6:

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.

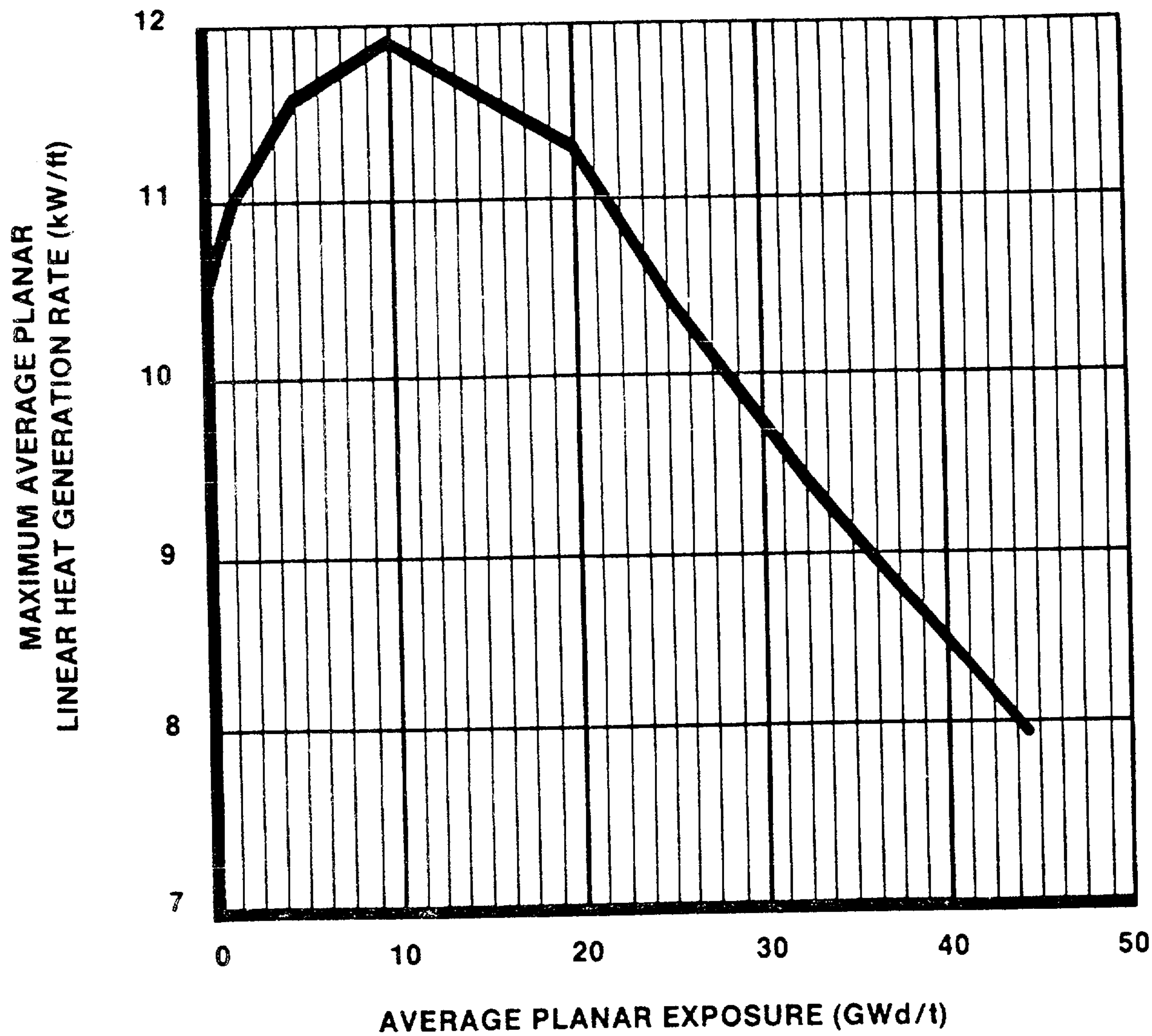


FIGURE 3.2.1-1
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE BP8SRB094

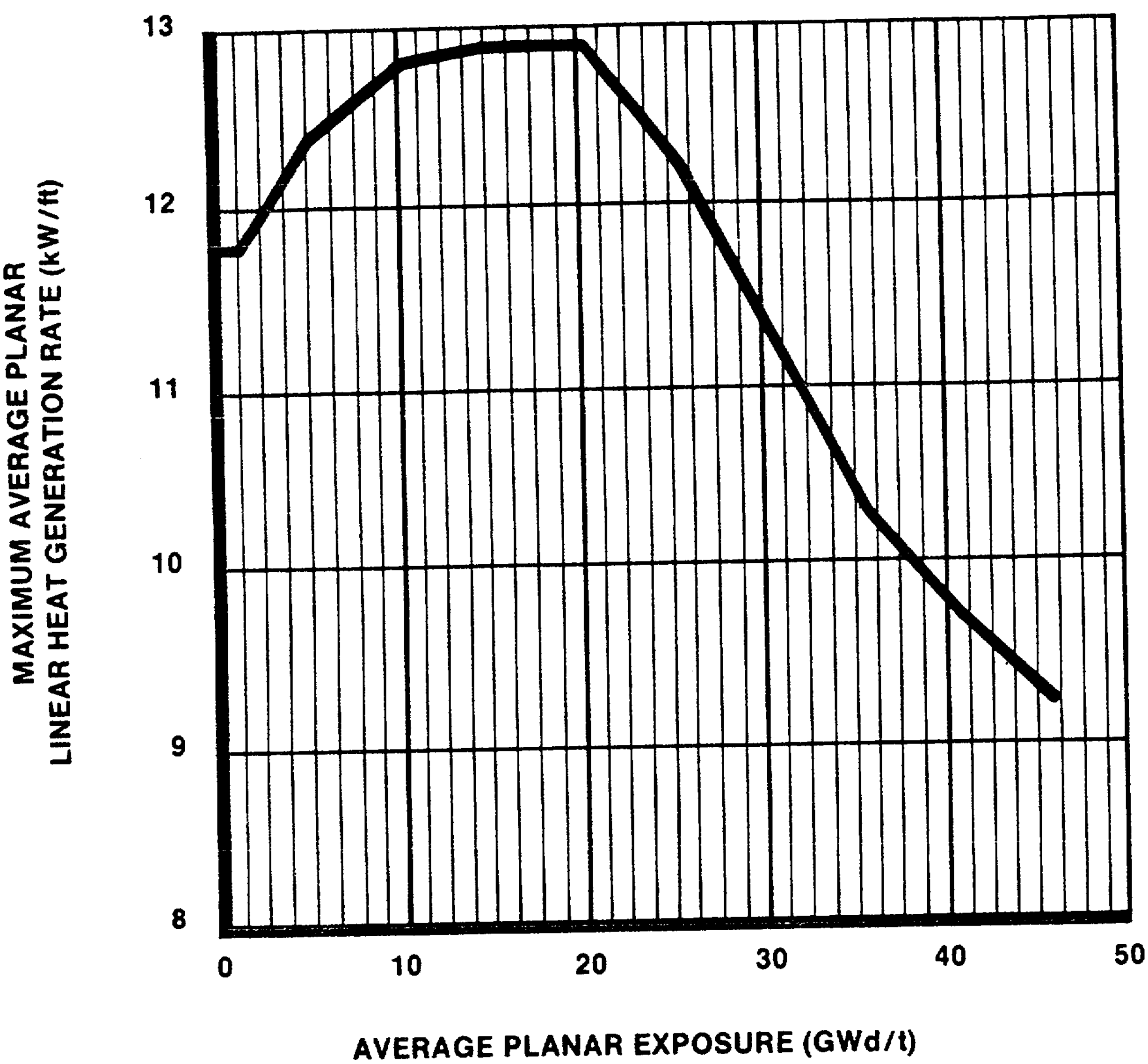


FIGURE 3.2.1-2
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE BP8SRB163

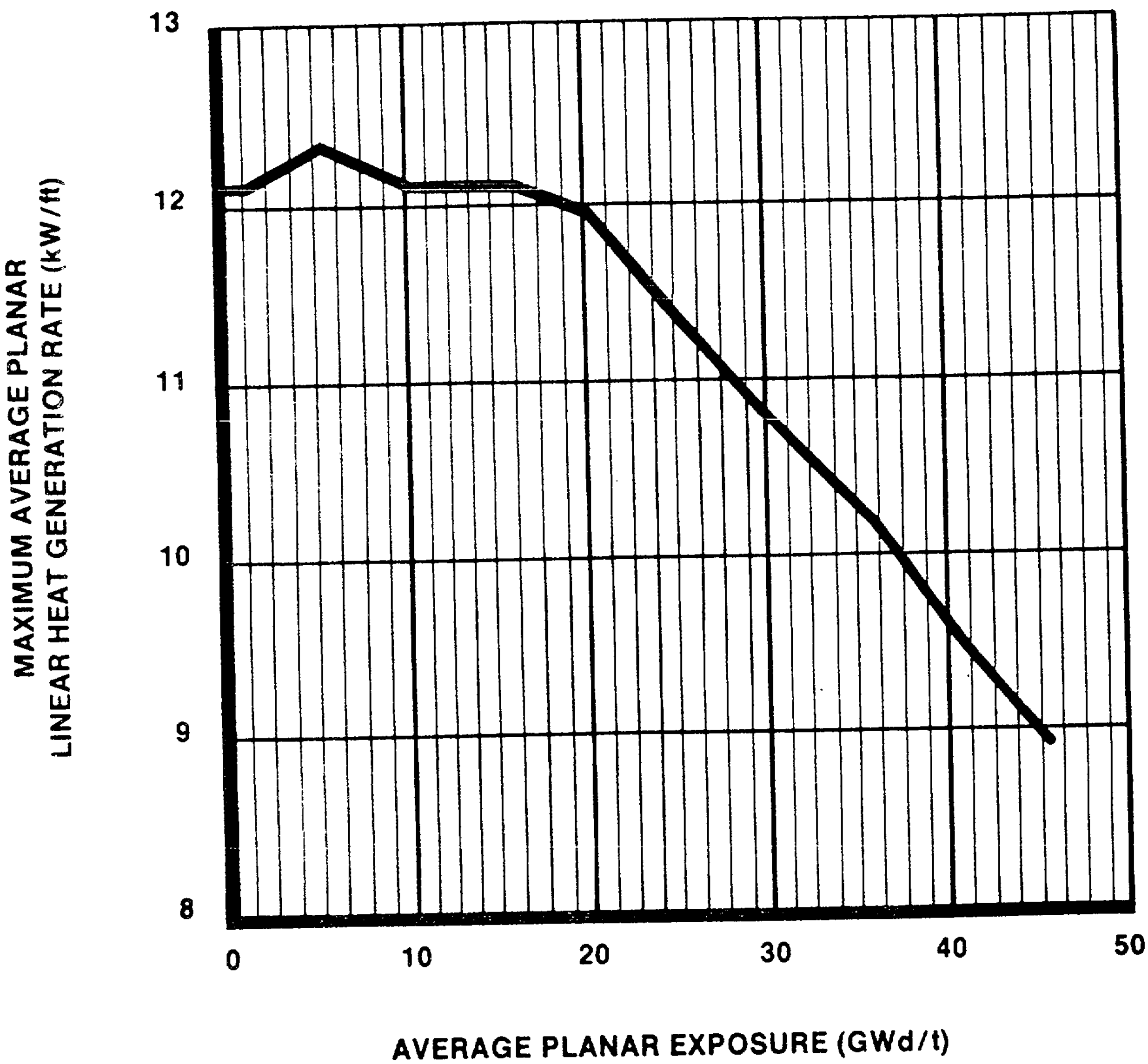


FIGURE 3.2.1-3
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE BP8SRB248

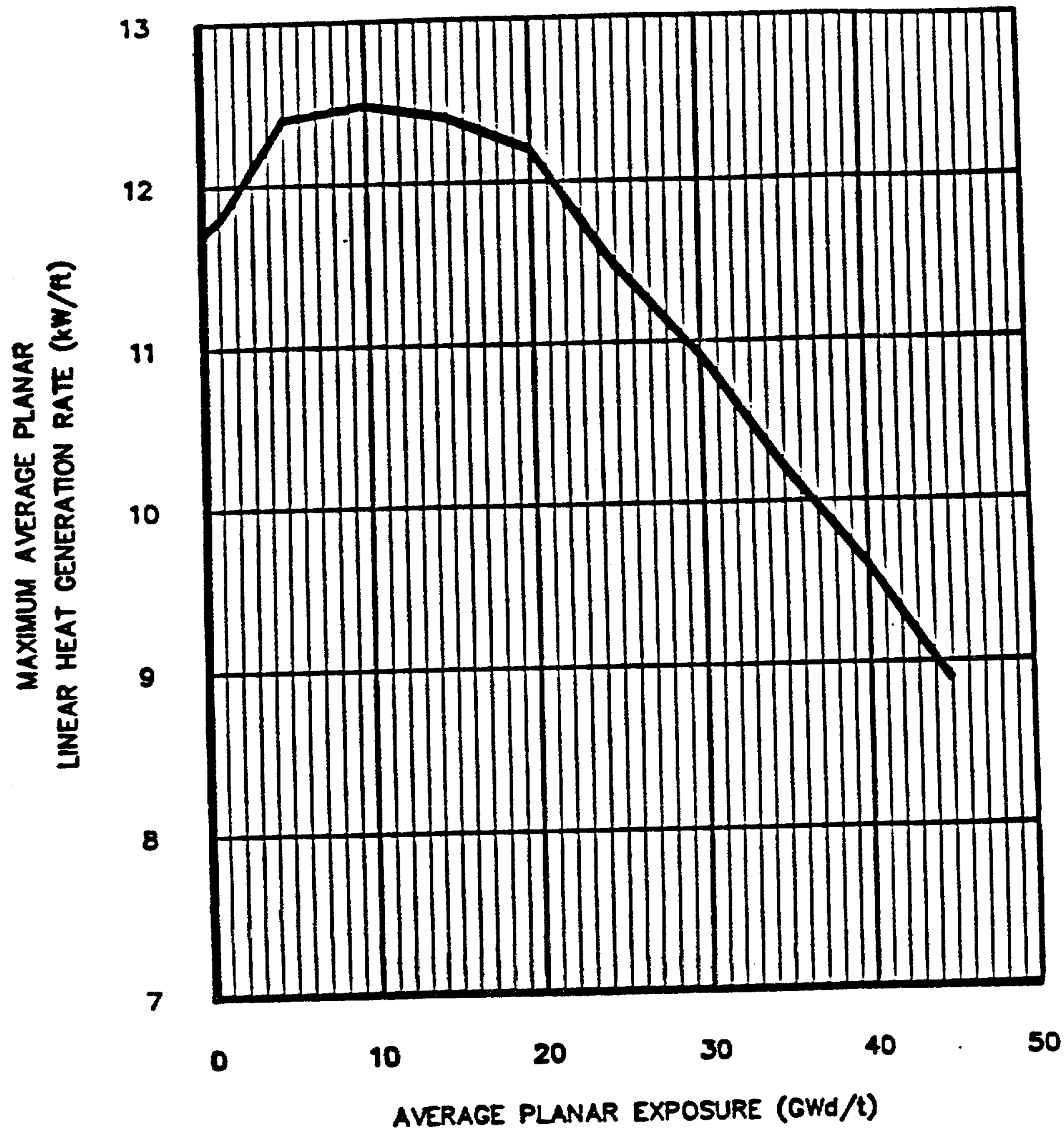


FIGURE 3.2.1–4

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE – BP8SRB278

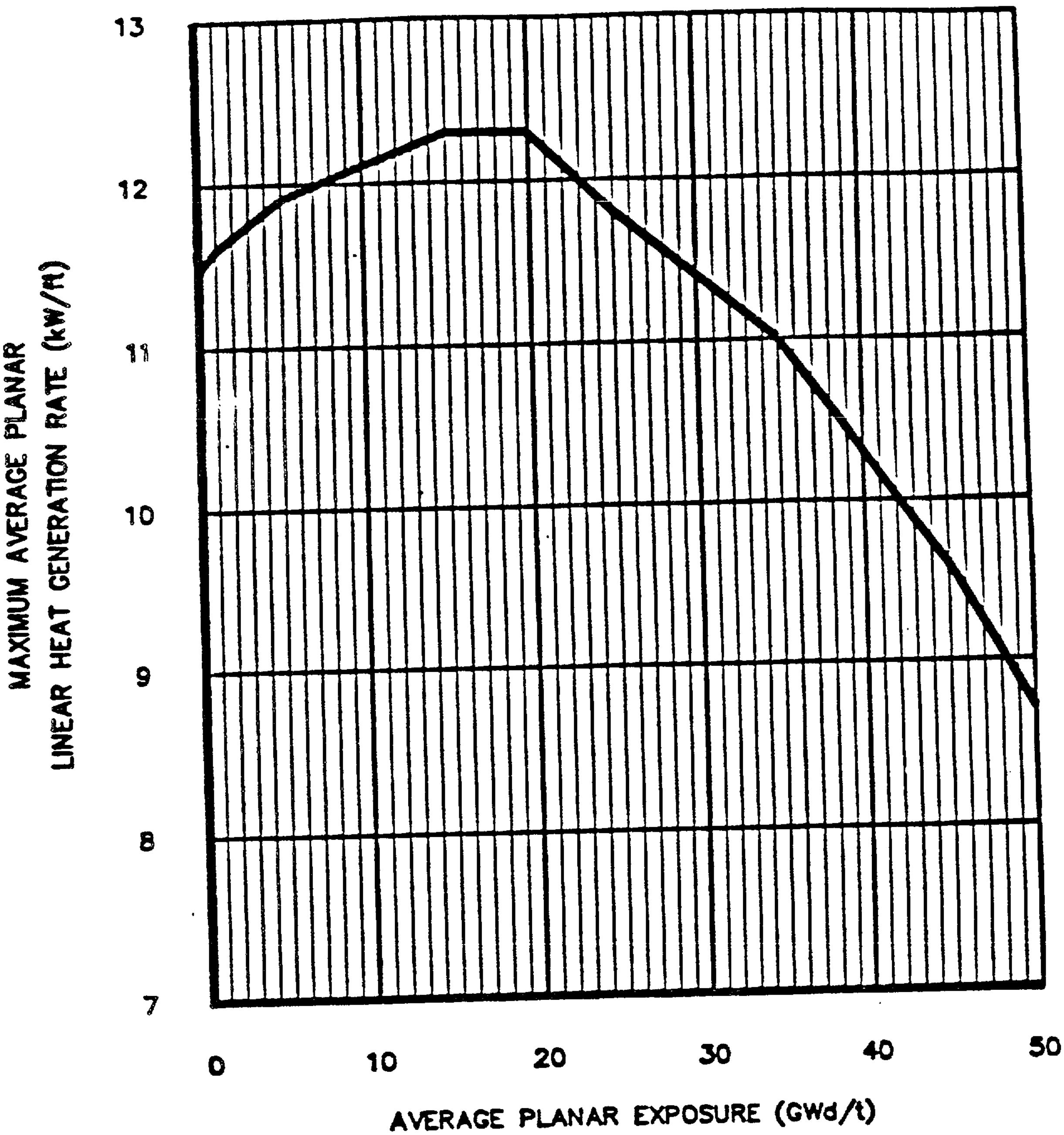


FIGURE 3.2.1-5

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE - BP8SRB299

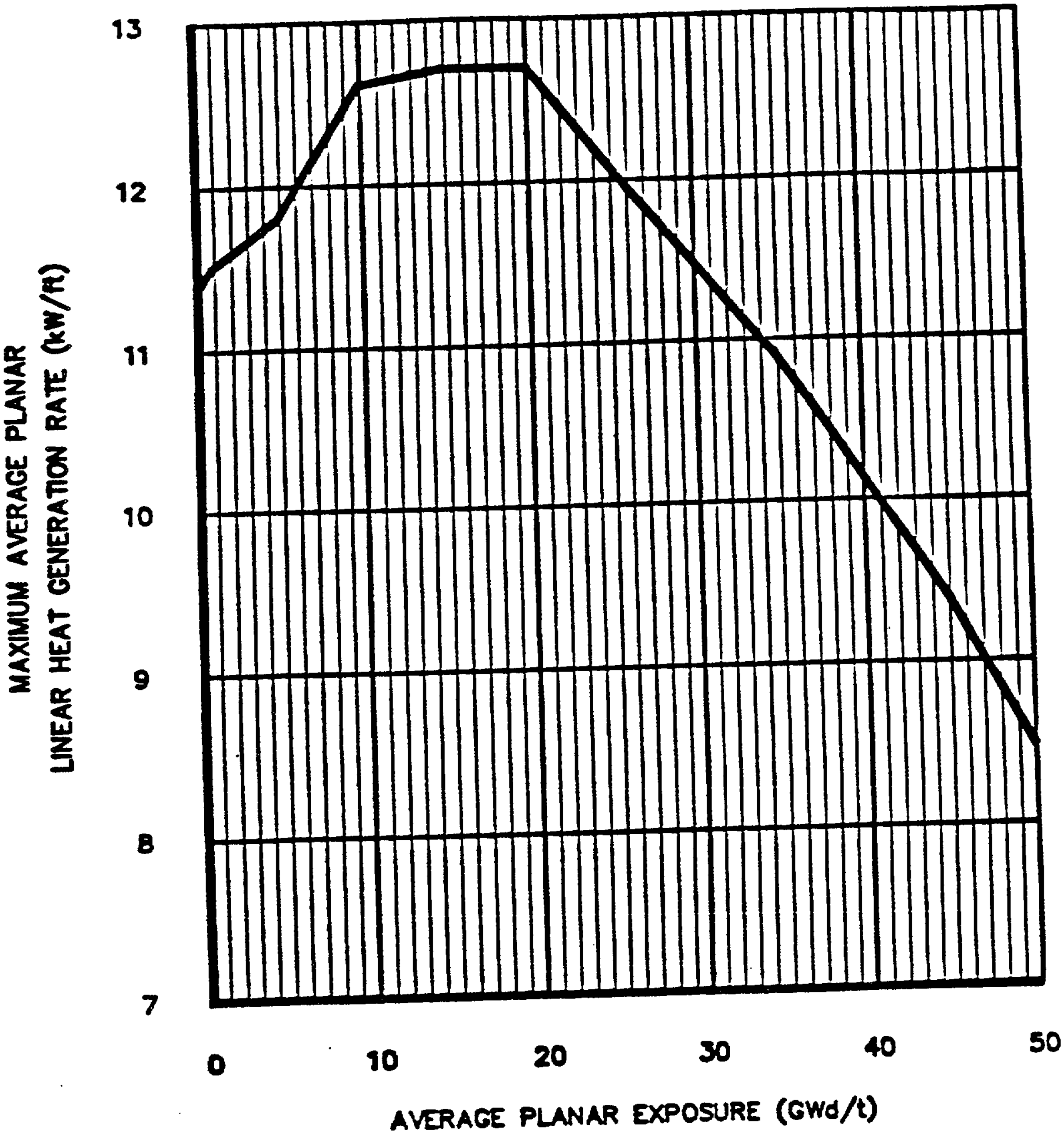


FIGURE 3.2.1-6

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE – BP8SRB305

BASES

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 in-sequence 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 any time a control rod is incapable of insertion.

3/4.1.2 REACTIVITY ANOMALIES

Since the SHUTDOWN MARGIN requirement for the reactor is small, a careful comparison of actual conditions to the predicted conditions is necessary, and the changes in reactivity can be inferred from these comparisons of rod patterns. Since the comparisons are easily done, frequent checks are not an imposition on normal operations. A 1% change is larger than is expected for normal operation so a change of this magnitude should be thoroughly evaluated. A change as large as 1% would not exceed the design conditions of the reactor and is on the safe side of the postulated transients.

3/4.1.3 CONTROL RODS

The specifications of this section (1) ensure that the minimum SHUTDOWN MARGIN is maintained and the control rod insertion times are consistent with those used in the safety analyses, and (2) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but 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 that is long enough to permit determining the cause of the inoperability yet 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 not fully inserted are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable 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 1.07 during the limiting power transient analyzed in Section 15.0 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 1.07. The occurrence of scram times longer then those specified should be viewed as an indication of a systemic 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 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

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 limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5 and 3.2.1-6.

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 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 into a condition exceeding a thermal limit.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5 and 3.2.1-6 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in NEDE-20566[1]. Differences in this analysis compared to previous analyses can be broken down as follows.

a. Input Changes

1. Corrected Vaporization Calculation - Coefficients in the vaporization correlation used in the REFLOOD code were corrected.

2. Incorporated more accurate bypass areas - The bypass areas in the top guide were recalculated using a more accurate technique.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

3. Corrected guide tube thermal resistance.

4. Correct heat capacity of reactor internals heat nodes.

b. Model Change

1. Core CCFL pressure differential - 1 psi - Incorporate the assumption that flow from the bypass to lower plenum must overcome a 1 psi pressure drop in core.

2. Incoporate NRC pressure transfer assumption - The assumption used in the SAFE-REFLOOD pressure transfer when the pressure is increasing was changed.

A few of the changes affect the accident calculation irrespective of CCFL. These changes are listed below.

a. Input Change

1. Break Areas - The DBA break area was calculated more accurately.

b. Model Change

1. Improved Radiation and Conduction Calculation - Incorporation of CHASTE-05 for heatup calculation.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B 3.2.1-1.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased simulated thermal power-high scram trip setpoint and the flow biased neutron flux-upscale control rod block trip setpoints of the APRM instruments must be adjusted to ensure that the MCPR does not become less than 1.07 or that $\geq$ 1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification, when the combination of THERMAL POWER and CMFLPD indicates a peak power distribution, to ensure that an LHGR transient would not be increased in degraded conditions.

Bases Table B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE

LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters;

Core THERMAL POWER	3015 Mwt* which corresponds to 105% of rated steam flow
Vessel Steam Output	13.08 x 10^6 lbm/hr which corresponds to 105% of rated steam flow
Vessel Steam Dome Pressure..............	1060 psia

Design Basis Recirculation Line Break Area for:

a. Large Breaks 2.2 ft^2.

b. Small Breaks 0.09 ft^2.

Fuel Parameters:

FUEL TYPE	FUEL ASSEMBLY GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kw/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	8 x 8	13.4	1.4	1.17

A more detailed listing of input of each model and its source is presented in Section II of NEDE 20566(1) and subsection 6.3.3 of the FSAR.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes an assembly power consistent with operation of the highest powered rod at 102% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

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 of 1.07 and an analysis of abnormal operational transients. For any abnormal operating transient analysis, 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 settings 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 MCPR. When added to the Safety Limit MCPR of 1.07, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and is presented in Figure 3.2.3-1. The power-flow map of Figure B 3/4 2.3-1 shows typical regions of plant operation.

The evaluation of a given transient begins with the system initial parameters identified in Reference 2 that are input to a GE core dynamic behavior transient computer program. The codes used to evaluate transients are described in Reference 2. The principal result of this evaluation is the reduction in MCPR caused by transient.

The purpose of the $MCPR_f$ and $MCPR_p$ of Figures 3.2.3-1 and 3.2.3-2 is to define operating limits at other than rated core flow and power conditions. At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_f$ and $MCPR_p$ at the existing core flow and power state. 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 $MCPR_f$s were calculated such that, for the maximum core flow rate and the corresponding THERMAL POWER along the 105%-of-rated steam flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the 105%-of-rated steam flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined as $MCPR_f$.

MINIMUM CRITICAL POWER RATIO (Continued)

The $MCPR_p$s are established to protect the core from plant transients other than core flow increases, including localized events such as rod withdrawal error. The $MCPR_p$s were calculated based upon the most limiting transient at the given core power level.

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 these low power levels, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. 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 for calculating 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 into a condition exceeding a thermal limit.

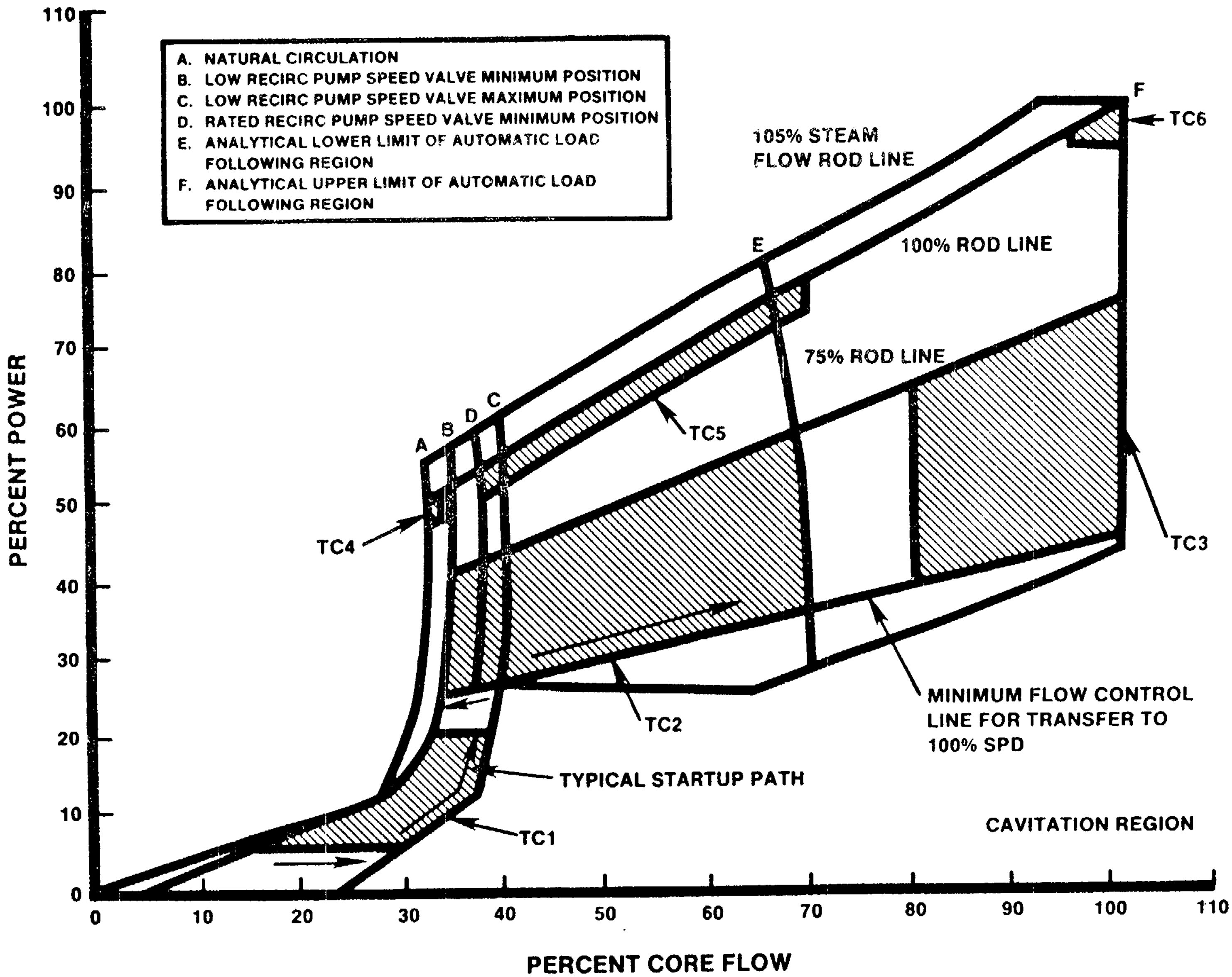
3/4.2.4 LINEAR HEAT GENERATION RATE

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

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. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A.



BASES FIGURE B3/4.2.3-1
POWER FLOW OPERATING MAP

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 12 TO FACILITY OPERATING LICENSE NO. NPF-47

GULF STATES UTILITIES COMPANY

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By letter dated August 14, 1987, Gulf States Utilities Company (GSU) (the licensee) requested an amendment to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The proposed amendment would amend the Technical Specifications for the Cycle 2 reload and operation (Refs. 1 & 2). The reload includes 164 new assemblies of GE manufacture. The reload design has no unusual features. The proposed Technical Specification changes are related to the Minimum Critical Power Ratio (MCPR), the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and updating and generalizing the bases and references associated with certain cycle dependent limits. The new fuel is of slightly increased enrichment designed for extended burnup.

The licensee provided plant-specific information used to determine reactor limits in a July 31, 1987 submittal (Ref. 2) that was referenced in the August 14, 1987 amendment application. Supplemental information clarifying the description of the new fuel for cycle 2 was provided in a September 18, 1987 submittal (Ref. 4).

2.0 EVALUATION

2.1 Reload Description

The licensee requests to be allowed to use GE fuel types BP8SRB299 and BP8SRB305 which have slightly higher enrichment than the present fuel types and will allow higher burnup. The core loading is the conventional new assembly scatter pattern, with low reactivity (old) assemblies located on the periphery. The new assembly types are not described in GESTAR II (Ref. 3).

2.2 Fuel Design

The new fuel for Cycle 2 is the GE fuel designated BP8SRB299 and BP8SRB305. This fuel is in the same class with approved designs but not for the enrichments used here. The specific description of this fuel is presented in Reference 4. This fuel description is acceptable.

For Cycle 2 operation, appropriate Maximum Average Planar Linear Heat Generation Rates (MAPLHGR) have been determined by approved thermal, mechanical and Loss-of-Coolant Accident (LOCA) analyses calculations. The most limiting MAPLHGR as a function of burnup for the new core loading are presented in the proposed Technical Specifications (Ref. 1) for the old and the new fuel types present in Cycle 2.

2.3 Nuclear Design

The nuclear design for Cycle 2 has been performed by GE using the approved GESTAR II methodology (Ref. 3). The results of these analyses are given in the GE reload report (Ref. 2) in the GESTAR II format. The results are within the acceptable reload range. The shutdown margin is 2.7% Δk at BOC with the strongest rod out and 1.2% Δk at the exposure with the minimum shutdown margin. Both meet the required 0.38% Δk margin required by the Technical Specifications. The standby liquid control system also meets the shutdown requirements with a shutdown margin of 4.0% Δk. Because these and other nuclear characteristics of the reload have been computed with previously approved methods (outlined in GESTAR II) and their values are within the allowed range, the nuclear design is acceptable.

2.4 Thermal-Hydraulic Design

The thermal-hydraulic design for Cycle-2 has been calculated using the approved methods described in GESTAR II. The results are given in the standard GESTAR II format in the reload report (Ref. 2). The parameters and initial values used for the calculations are those approved in GESTAR II for the BWR/6 class of reactors. The GEMINI set of methods (References 5 and 6) have been approved for the relevant transient analyses. In this method, the difference between the analyses and Technical Specification values of the scram speed is not taken into account. Only the Technical Specification values are used.

The operating limit of the Minimum Critical Power Ratio (MCPR) values are determined by the limiting transient among the following: local rod withdrawal error, feedwater controller failure, load rejection without bypass and loss of 100°F feedwater heating. The analyses of these events for Cycle 2 used approved methods. The loss of 100° F feedwater heating and the local rod withdrawal transient are limiting. The ΔCPR results of these analyses are reflected in the requested Technical Specification changes. The MCPR for Cycle 2 has been increased from 1.06 to 1.07 to account for Cycle 2 uncertainties. This value has been approved in the FSAR. For the analyses of the above transients, approved methods have been used. The results are within expected ranges and, hence, they are acceptable.

For the River Bend Cycle 2, no cycle specific stability analysis is required because the Technical Specifications have standard NRC approved provisions for incore neutron detector monitoring of thermal-hydraulic stability according to the recommendations of the General Electric SIL-380.

2.5 Transient and Accident Analyses

The accident and transient analysis methods used for Cycle 2 are described in GESTAR II. The GEMINI set of codes was used. The MCPR operating limit was determined from the rod withdrawal error transient $\Delta CPR=0.11$ added to the MCPR of 1.07 for a cycle operating MCPR limit of 1.18. The core wide transient analysis methodologies have been approved and the results fall within expected ranges and are acceptable.

The mislocated assembly event is not analyzed for reloads because studies indicated that there is a very small probability of an event exceeding the MCPR limits. The assembly misorientation event is not analyzed due to the symmetric water gap in type C lattices. This is acceptable.

The limiting overpressurization event analysis, i.e., main isolation valve closure with flux scram, was performed using the GEMINI methods (Refs. 5 and 6) at 102% of power level to account for the power level uncertainties specified in Regulatory Guide 1.49. The results show that the peak steam dome and vessel pressures of 1,210 and 1,247 psig to be under 1325 psig i.e., the required limit in the Technical Specifications. The methodology and the results of the overpressurization event analysis are acceptable (Ref. 2).

Loss of Coolant Accident (LOCA) analyses, using approved (SAFE/REFLOOD) methods and parameter values were performed to provide MAPLHGR values vs average planar exposure, peak clad temperature and oxidation fraction for both new fuel type assemblies for Cycle 2, i.e, BP8SRB299 and BP8SRB305. The results show compliance with 10 CFR 50.46, and the LHGR limit of 13.4 kW/ft. and, therefore, are acceptable.

2.6 Selected Margin Improvement and Operating Flexibility Options

River Bend has the following options:

- Recirculation Pump Trip
- Rod Withdrawal Limiter
- Thermal Power Monitor
- Single Loop Operation
- Feedwater Heater Out-of-Service

These options have been generically reviewed and approved.

2.7 Proposed Technical Specification Changes

The following Technical Specifications are proposed to be changed:

1. 2.1.2, Thermal Power, High Pressure and High Flow. The MCPR has been increased in the Technical Specification and the bases 2.1.2. Tables B2.1.2-1 and B2.1.2-2. These changes are acceptable as discussed in the evaluation.

2. 3/4 2.1, Average Planar Linear Heat Generation Rate. Modification of the MAPLHGR vs average exposure for each fuel type in Cycle 2. Figures 3.2.1-2 to 3.2.1-6 and bases 3/4 1.3 and 3/4 2.1 to 3/4 2.4. These changes have been discussed above and are acceptable.

3.0 SUMMARY

The NRC staff has reviewed the information submitted for the Cycle 2 operation of the River Bend reactor. Based on this review, the staff concludes that the fuel design, the nuclear design, the thermal hydraulic design and the accident and transient analyses are acceptable. The proposed Technical Specifications submitted for the Cycle 2 reload represent the necessary modifications for this cycle and they are acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 nor environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

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 security or to the health and safety of the public. The staff therefore concludes that the proposed changes are acceptable, and they are hereby incorporated into the River Bend Unit 1 Technical Specifications.

6.0 REFERENCES

1. Letter from J. C. Deddens, Gulf States Utilities Company, to USNRC, dated August 14, 1987.

2. "Supplemental Reload Licensing Submittal for River Bend Station Reload 1" GE Report 23A5819, dated July 1987.

3. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," GESTAR II, as amended, dated May 1986.

4. Letter from J. E. Booker, Gulf States Utilities Company, to USNRC, dated September 18, 1987 and GE Report 23A5819 "Supplement 1 to Supplemental Reload Licensing Submittal for River Bend Station Reload 1."

5. Letter from J. S. Charnley, General Electric, to M. W. Hodges, NRC, dated July 6, 1987.

6. Letter from G. C. Lainas to J. S. Charnley, General Electric, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, GE Generic Licensing Reload Report, Supplement to Amendment 11," March 22, 1986.

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Dated: October 19, 1987