November 15, 1990

Docket No. 50-416

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

Dear Mr. Cottle:

SUBJECT: ISSUANCE OF AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. NPF-29 - GRAND GULF NUCLEAR STATION, UNIT 1, REGARDING FUEL CYCLE 5 RELAOD (TAC NO. 76992)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 73 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment consists of changes to the Technical Specifications (TS) in response to your application dated June 8, 1990, as revised August 15, 1990.

The amendment revises the TS and Bases to reflect the Advanced Nuclear Fuels Corporation 8x8 and 9x9-5 fuel used in the fuel Cycle 5 Reload.

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

Sincerely,

ORIGINAL SIGNED BY:

Lester L. Kintner, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 73 to NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

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

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Lester L. Kintner

Lester L. Kintner, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

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ENTERGY OPERATIONS, INC.

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

MISSISSIPPI POWER AND LIGHT COMPANY

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 73
License No. NPF-29

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

   A. The application for amendment by the licensee dated June 8, 1990, as revised August 15, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 73 , are hereby incorporated into this license. Entergy Operations, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R Quay

Theodore R. Quay, Acting Director
Project Directorate IV-1
Division of Reactor Projects - III
IV, V, and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 15, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 73

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE PAGES	INSERT PAGES
1-2	1-2
2-1	2-1
B 2-1	B 2-1
B 2-1a	B 2-1a
B 2-2	B 2-2
3/4 1-16	3/4 1-16
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-2a	-
3/4 2-2b	-
3/4 2-2c	-
3/4 2-2d	-
3/4 2-2e	-
3/4 2-3a	-
3/4 2-3b	-
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
3/4 2-7	3/4 2-7
3/4 2-7a	3/4 2-7a
-	3/4 2-7b
-	3/4 2-7c
3/4 4-1	3/4 4-1
3/4 4-1a	3/4 4-1a
3/4 10-2	3/4 10-2
B 3/4 1-3	B 3/4 1-3
B 3/4 1-4	B 3/4 1-4
B 3/4 1-4a	B 3/4 1-4a
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-4	B 3/4 2-4
-	B 3/4 2-4a
B 3/4 2-5	B 3/4 2-5
B 3/4 2-6	B 3/4 2-6
B 3/4 2-7	B 3/4 2-7
-	B 3/4 2-7a
B 3/4 4-1	B 3/4 4-1

CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement of the SRMs, IRMs, LPRMs, TIPs, or special movable detectors is not considered to be CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CRITICAL POWER RATIO

1.8 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the ANFB correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.9 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

DRYWELL INTEGRITY

1.10 DRYWELL INTEGRITY shall exist when:

a. All drywell penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE drywell automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.4-1 of Specification 3.6.4.

b. The drywell equipment hatch is closed and sealed.

c. The drywell airlock is in compliance with the requirements of Specification 3.6.2.3.

d. The drywell leakage rates are within the limits of Specification 3.6.2.2.

e. The suppression pool is in compliance with the requirements of Specification 3.6.3.1.

f. The sealing mechanism associated with each drywell penetration; e.g., welds, bellows or O-rings, is OPERABLE.

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 1.09 during both two loop operation and single loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

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

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

THERMAL POWER, Low Pressure or Low Flow (Continued)

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

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

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

2.1.2 THERMAL POWER, High Pressure and High Flow

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

The Safety Limit MCPR assures sufficient conservatism such that, in the event of a sustained steady state operation at the MCPR safety limit, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (MCPR = 1.00) and the Safety Limit MCPR is based on a detailed statistical procedure which considers the uncertainties in monitoring the core operating state and includes the effects associated with channel bow. One specific uncertainty included in the safety limit is the uncertainty inherent in the ANFB critical power correlation. ANF report XN-NF-524(P), Rev. 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors," April 1989, including Supplement 1, describes the methodology used in determining the Safety Limit MCPR.

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

ROD PATTERN CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.4.2 The rod pattern control system (RPCS) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*#.

ACTION

a. With the RPCS inoperable or with the requirements of ACTION b, below, not satisfied and with:

   1. THERMAL POWER less than or equal to the Low Power Setpoint, control rod movement shall not be permitted, except by a scram.

   2. THERMAL POWER greater than the Low Power Setpoint, control rod withdrawal shall not be permitted.

b. OPERABLE control rod movement may continue by bypassing control rod(s) in the RPCS** provided that:

   1. With one control rod inoperable due to being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable, this inoperable control rod may be bypassed in the rod action control system (RACS) provided that the SHUTDOWN MARGIN has been determined to be equal to or greater than required by Specification 3.1.1.

   2. With up to eight control rods inoperable for causes other than addressed in ACTION b.1, above, these inoperable control rods may be bypassed in the RACS provided that:

      a) The control rod(s) to be bypassed is inserted and the directional control valves are disarmed either:

         1) Electrically, or

         2) Hydraulically by closing the drive water and exhaust water isolation valves.

      b) All inoperable control rods are separated from all other inoperable control rods by at least two control cells in all directions.

      c) There are not more than 3 inoperable control rods in any RPCS group.

   3. Control rods may be bypassed in the Rod Action Control System (RACS) at any time. However, if THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER:

*See Special Test Exception 3.10.2

#Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RPCS prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

**Bypassing control rod(s) in the RPCS shall be performed under administrative control.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 During two loop operation, all AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figure 3.2.1-1.

During single loop operation, the APLHGR for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limit shown in Figure 3.2.1-1 multiplied by 0.8.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 24 hours,

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

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

d. The provisions of Specification 4.0.4 are not applicable.

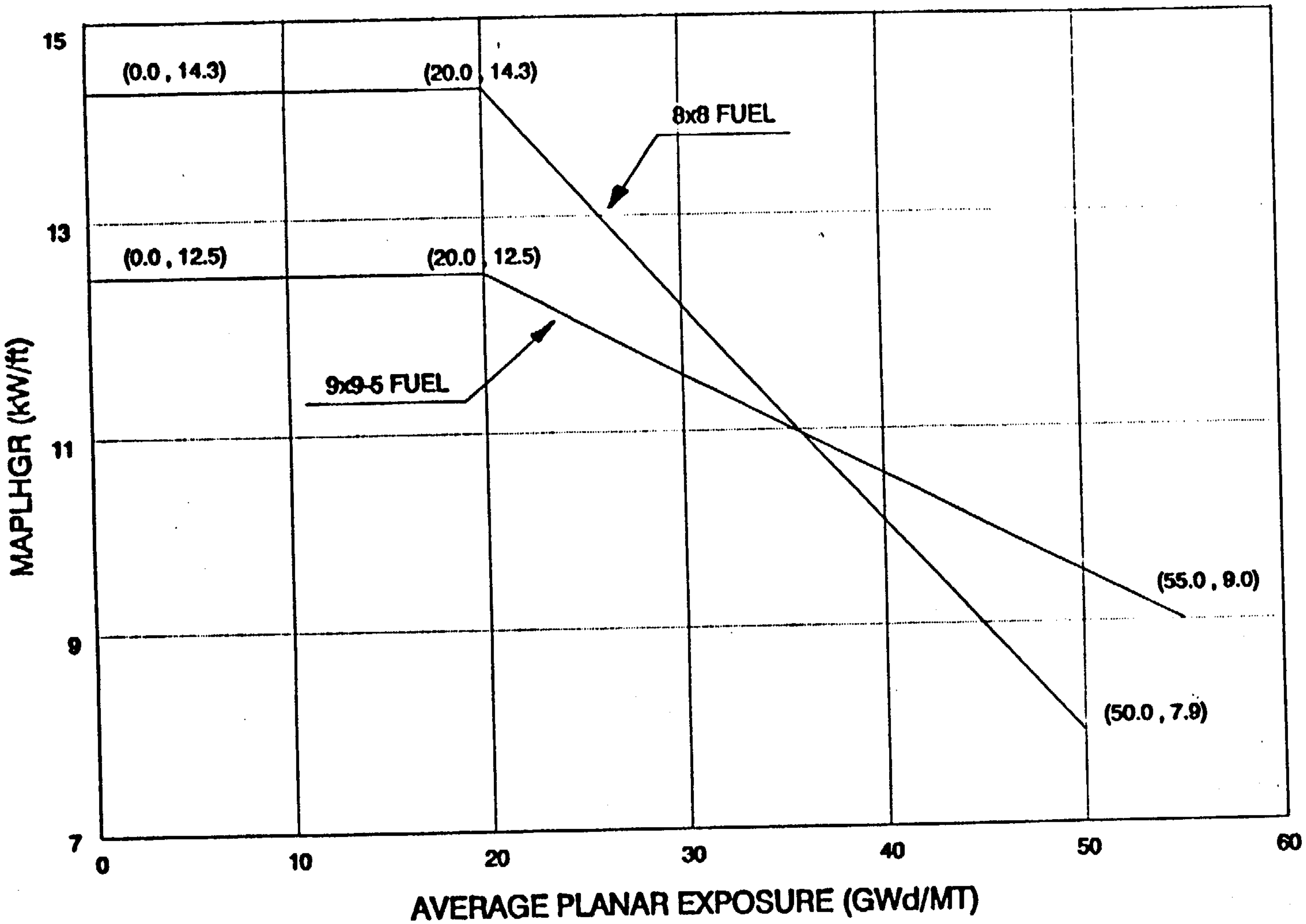

FIGURE 3.2.1-1 MAPLHGR vs AVERAGE PLANAR EXPOSURE FOR ANF FUEL

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the $MCPR_f$, $MCPR_p$, and $MCPR_e$ limits at indicated core flow, THERMAL POWER, and exposure, as shown in Figures 3.2.3-1, 3.2.3-2, and 3.2.3-3.

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

ACTION:

With MCPR less than the applicable MCPR limits determined from Figures 3.2.3-1, 3.2.3-2, and 3.2.3-3, initiate corrective action within 15 minutes and restore MCPR 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

4.2.3 MCPR shall be determined to be equal to or greater than the applicable MCPR limits determined from Figures 3.2.3-1, 3.2.3-2, and 3.2.3-3:

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 MCPR.

d. The provisions of Specification 4.0.4 are not applicable.

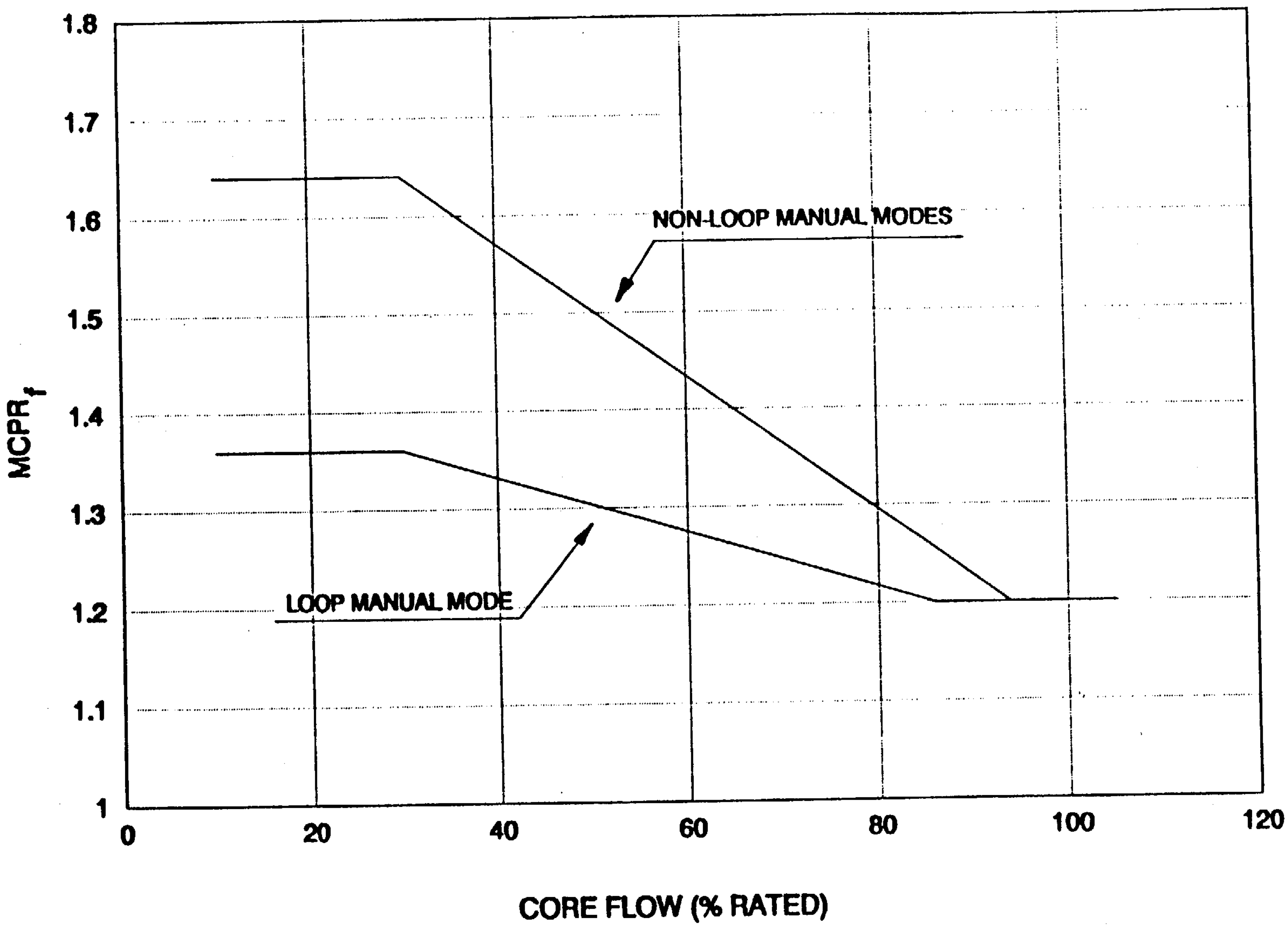

FIGURE 3.2.3-1 $MCPR_f$

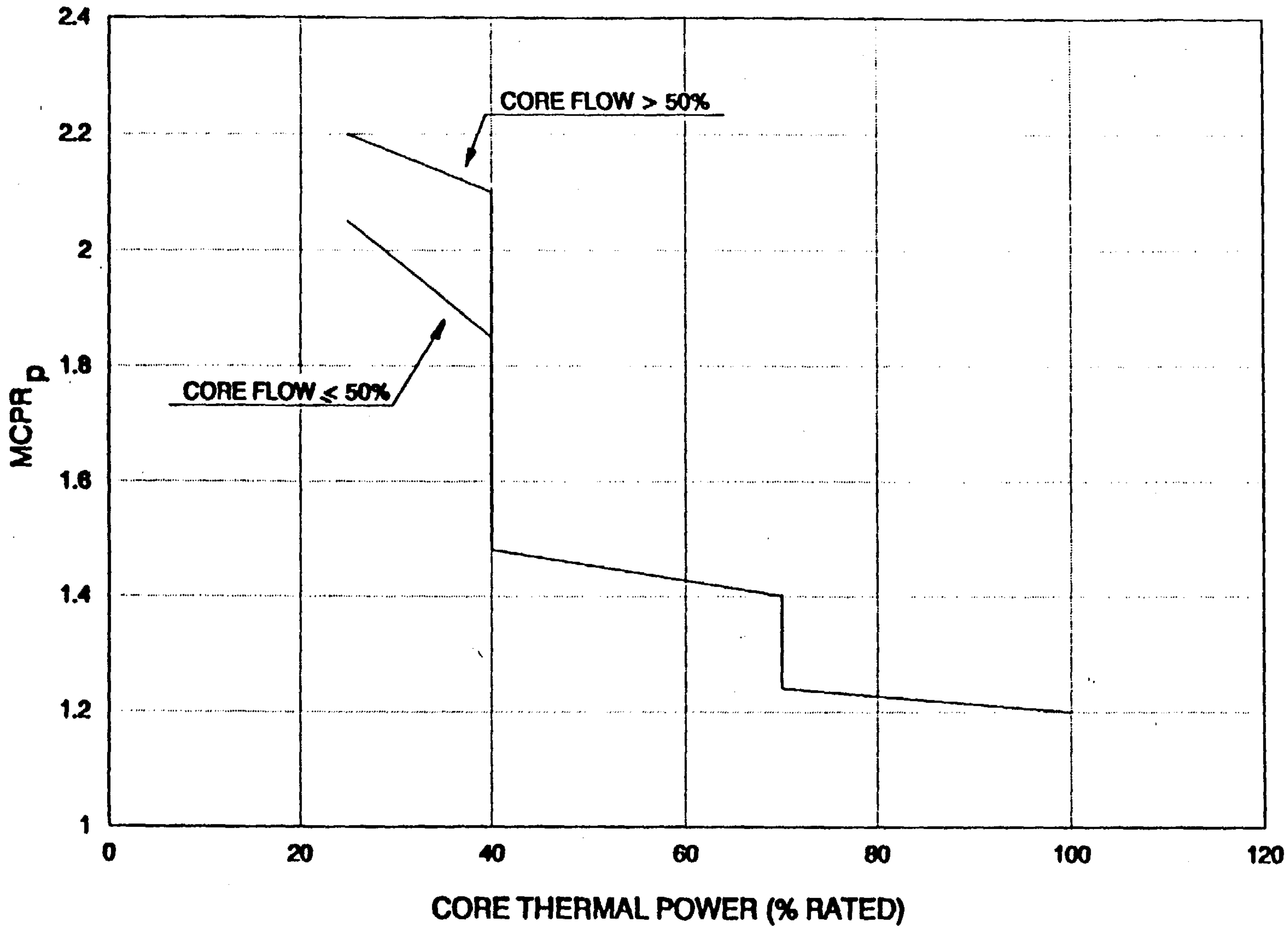

FIGURE 3.2.3-2 MCPR$_p$

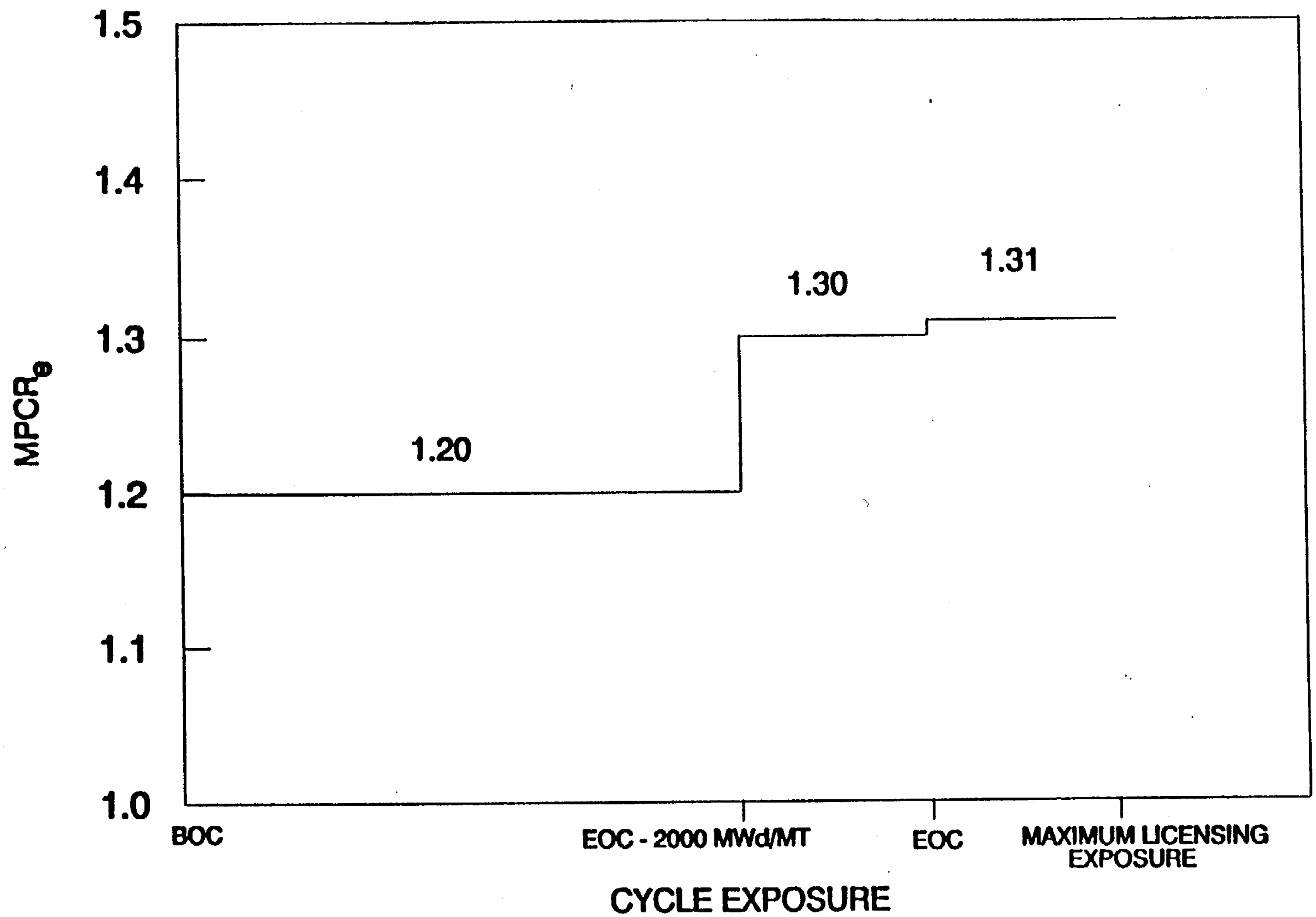

FIGURE 3.2.3-3 MCPR$_e$

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the limits shown in Figure 3.2.4-1 as multiplied by the smaller of either the flow-dependent LHGR factor ($LHGRFAC_f$) of Figure 3.2.4-2, or the power-dependent LHGR factor ($LHGRFAC_p$) of Figure 3.2.4-3.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 24 hours,

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

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

d. The provisions of Specification 4.0.4 are not applicable.

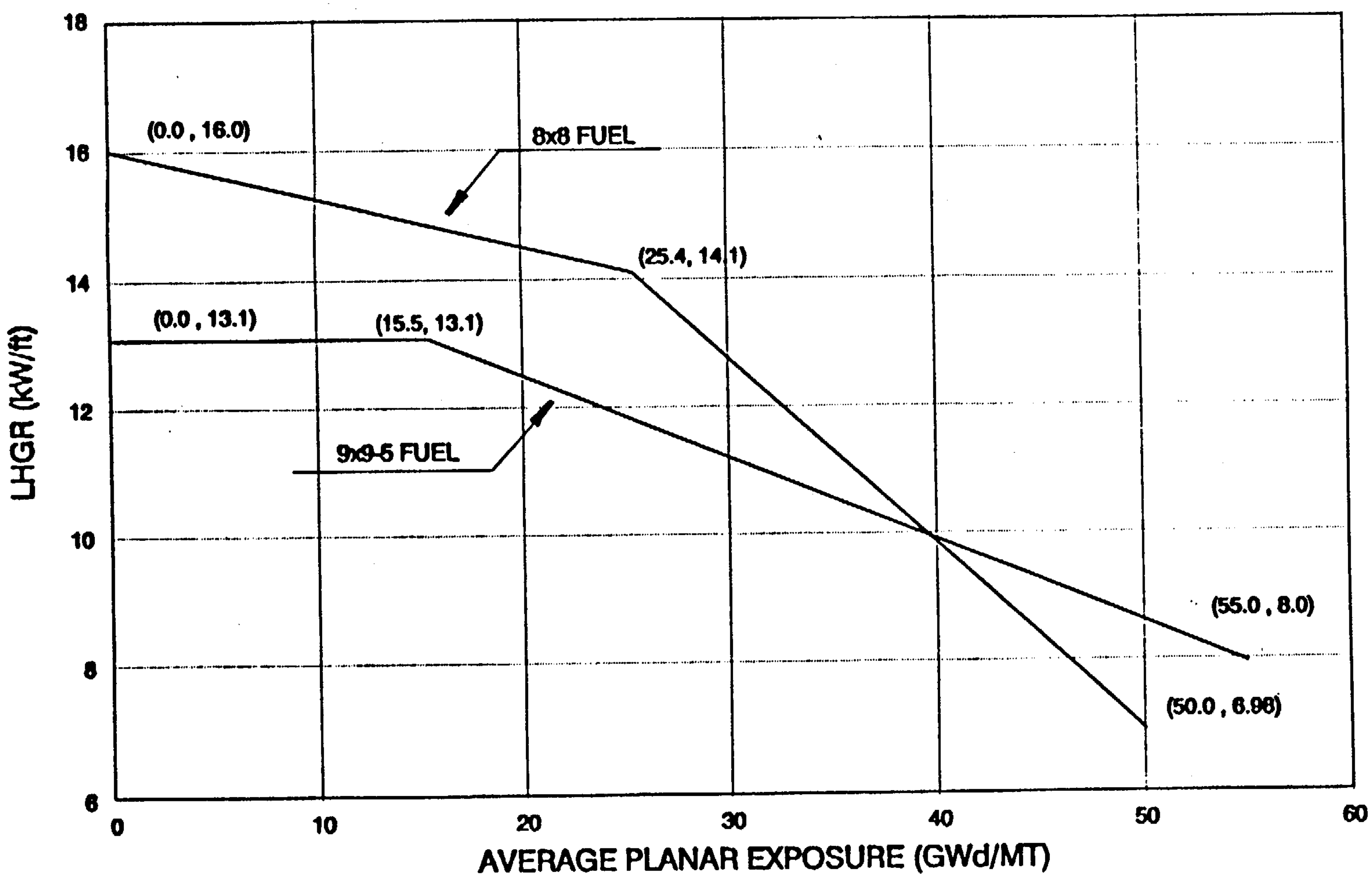

FIGURE 3.2.4-1 LHGR vs AVERAGE PLANAR EXPOSURE FOR ANF FUEL

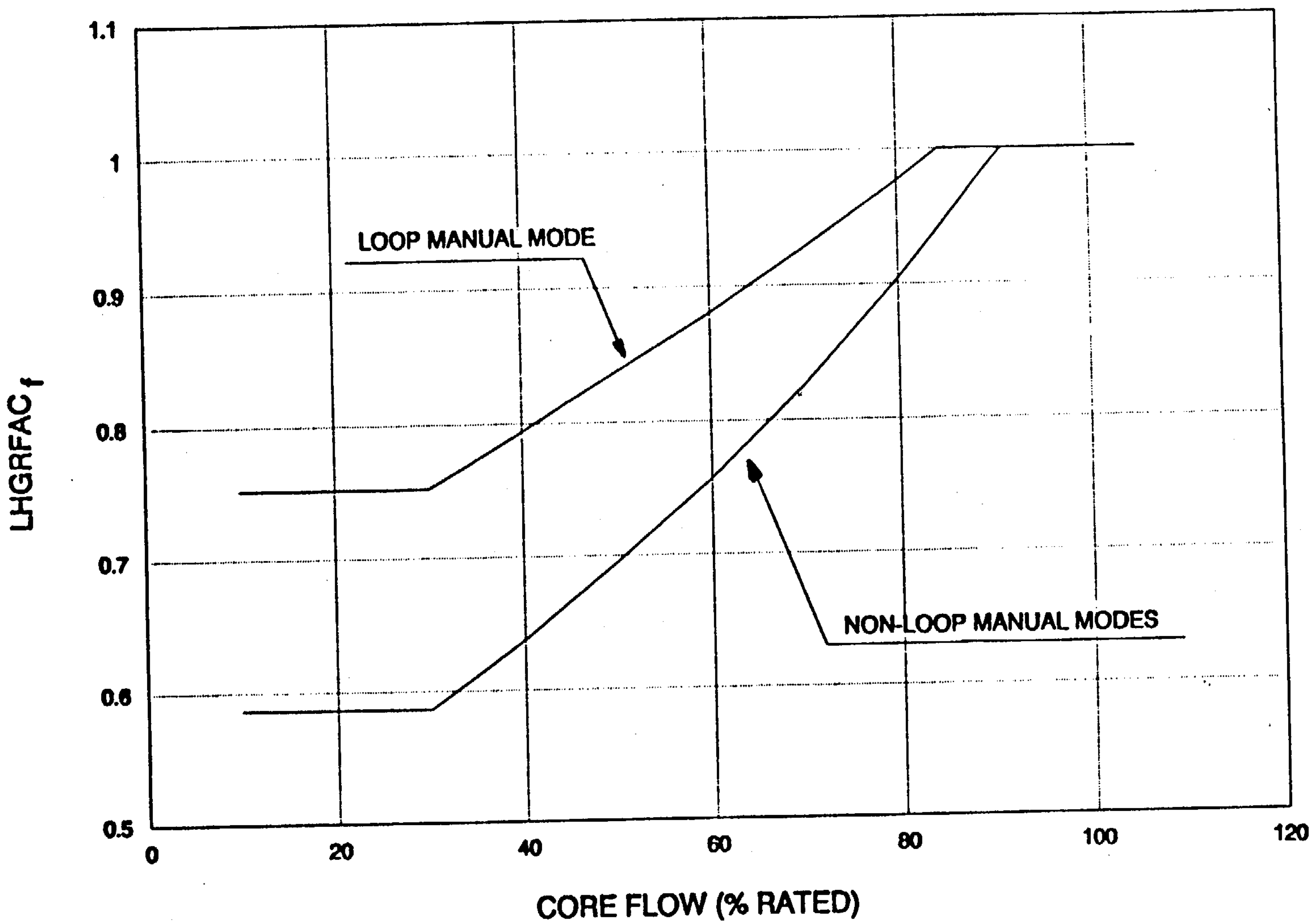

FIGURE 3.2.4-2 LHGRFAC$_f$

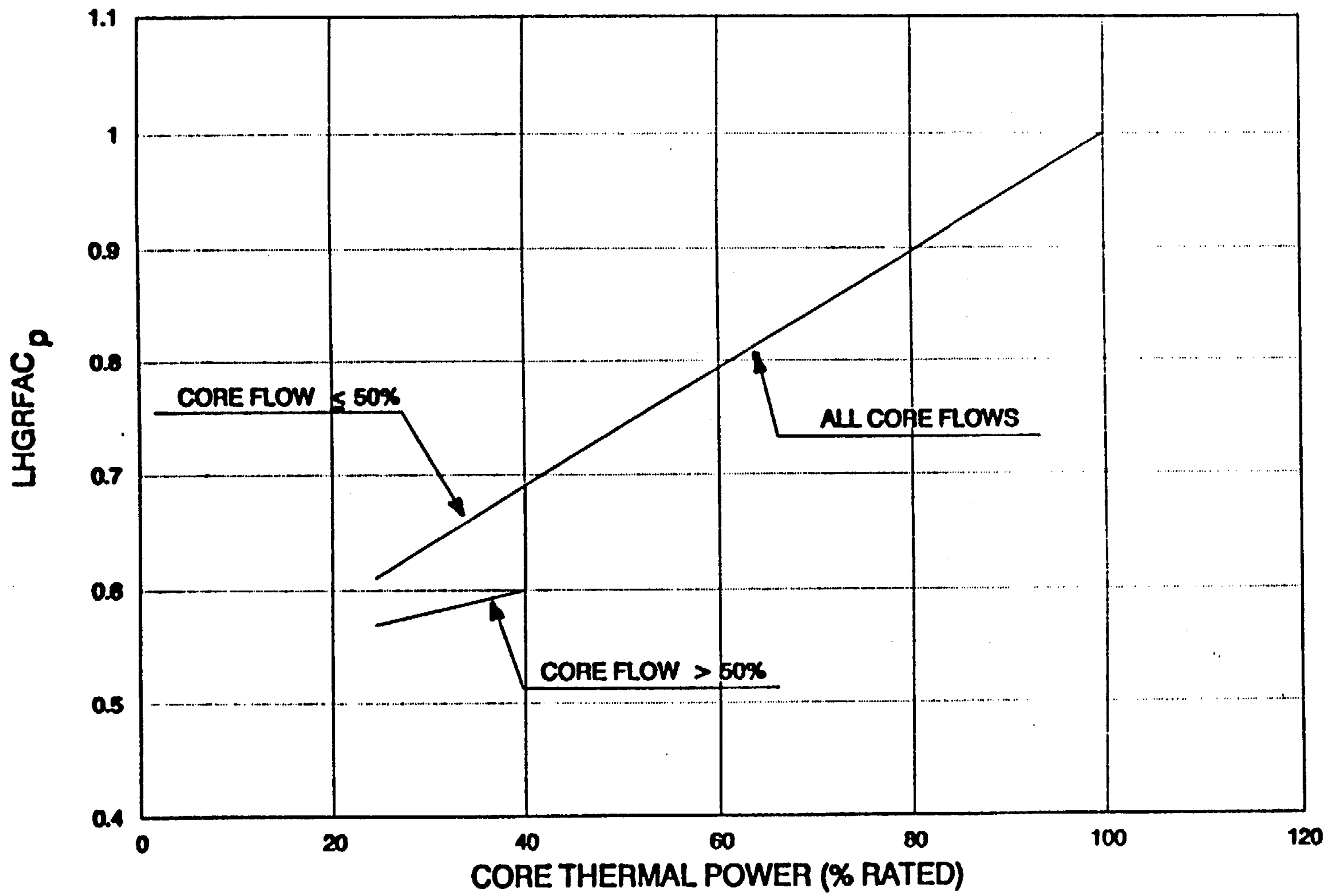

FIGURE 3.2.4-3 LHGRFAC$_p$

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 The reactor coolant recirculation system shall be in operation with either:

- a. Two recirculation loops operating with limits and setpoints per Specifications 2.2.1, 3.2.1, and 3.3.6, or
- b. A single recirculation loop operating with:
  1. A volumetric loop flow rate less than 44,600 gpm, and
  2. The loop recirculation flow control in the manual mode, and
  3. Limits and setpoints per Specifications 2.2.1, 3.2.1, and 3.3.6.

Operation is not permissible in Regions A, B or C as specified in Figure 3.4.1.1-1 except that operation in Region C is permissible during control rod withdrawals for startup.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

- a. With no reactor coolant system recirculation loops in operation and the reactor mode switch in the run position, immediately place the reactor mode switch in the shutdown position.
- b. With operation in Region A as specified in Figure 3.4.1.1-1, immediately place the reactor mode switch in the shutdown position.
- c. With operation in regions B or C as specified in Figure 3.4.1.1-1, observe the indicated APRM, neutron flux noise level. With a sustained APRM neutron flux noise level greater than 10% peak-to-peak of RATED THERMAL POWER, immediately place the reactor mode switch in the shutdown position.
- d. With operation in Region B as specified in Figure 3.4.1.1-1, immediately initiate action to either reduce THERMAL POWER by inserting control rods or increase core flow if one or more recirculation pumps are on fast speed by opening the flow control valve to within Region D of Figure 3.4.1.1-1 within 2 hours.
- e. With operation in Region C as specified in Figure 3.4.1.1-1, unless operation in this region is for control rod withdrawals during startup, immediately initiate action to either reduce THERMAL POWER or increase core flow to within Region D of Figure 3.4.1.1-1 within 2 hours.
- f. During single loop operation, with the volumetric loop flow rate greater than the above limit, immediately initiate corrective action to reduce flow to within the above limit within 30 minutes.

*See Special Test Exception 3.10.4.

LIMITING CONDITION FOR OPERATION (Continued)

g. During single loop operation, with the loop flow control not in the manual mode, place it in the manual mode within 15 minutes.

h. During single loop operation, with temperature differences exceeding the limits of SURVEILLANCE REQUIREMENT 4.4.1.1.5, suspend the THERMAL POWER or recirculation loop flow increase.

i. With a change in reactor operating conditions, from two recirculation loops operating to single loop operation, or restoration of two loop operation, the limits and setpoints of Specifications 2.2.1, 3.2.1, and 3.3.6 shall be implemented within 8 hours or declare the associated equipment inoperable (or the limits to be "not satisfied"), and take the ACTIONS required by the referenced specifications.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 At least once per 24 hours, the reactor coolant recirculation system shall be verified to be in operation and not in Regions A, B or C as specified in Figure 3.4.1.1-1 except that operation in Region C is permissible during control rod withdrawals for startup.

4.4.1.1.2 Each reactor coolant system recirculation loop flow control valve in an operating loop shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying that the control valve fails "as is" on loss of hydraulic pressure at the hydraulic unit, and

b. Verifying that the average rate of control valve movement is:

1. Less than or equal to 11% of stroke per second opening, and

2. Less than or equal to 11% of stroke per second closing.

4.4.1.1.3 During single loop operation, verify that the loop recirculation flow control in the operating loop is in the manual mode at least once per 8 hours.

4.4.1.1.4 During single loop operation, verify that the volumetric loop flow rate of the loop in operation is within the limit at least once per 24 hours.

3/4.10.2 ROD PATTERN CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups by the rod pattern control system (RPCS) per Specification 3.1.4.2 may be suspended by means of the individual rod position bypass switches* for the following tests:

a. Shutdown margin demonstrations, Specification 4.1.1.

b. Control rod scram, Specification 4.1.3.2.

c. Control rod friction measurements.

d. Startup Test Program with the THERMAL POWER less than 10% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the requirements of the above specification not satisfied, verify that the RPCS is OPERABLE per Specification 3.1.4.2.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints imposed on control rod groups by the RPCS are bypassed, verify;

a. Within 8 hours prior to bypassing any sequence constraint and at least once per 12 hours while any sequence constraint is bypassed, that movement of the control rods at less than or equal to 10% of RATED THERMAL POWER is limited to the established control rod sequence for the specified test, and

b. Conformance with this specification and test procedures by a second licensed operator or other technically qualified member of the unit technical staff.

*Bypassing control rod(s) in the RPCS shall be performed under administrative control.

CONTROL RODS (Continued)

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

The rod withdrawal limiter system input power signal orginates from the first stage turbine pressure. When operating with the steam bypass valves open, this signal indicates a core power level which is less than the true core power. Consequently, near the low power setpoint and high power setpoint of the rod pattern control system, the potential exists for nonconservative control rod withdrawals. Therefore, when operating at a sufficiently high power level, there is a small probability of violating fuel Safety Limits during a licensing basis rod withdrawal error transient. To ensure that fuel Safety Limits are not violated, this specification prohibits control rod withdrawal when a biased power signal exists and core power exceeds the specified level.

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 10% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the rod pattern controller function to be OPERABLE when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER provides adequate control.

CONTROL ROD PROGRAM CONTROLS (Continued)

The RPCS provides automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted. A rod is out of sequence if it does not meet the criteria of the Banked Position Withdrawal Sequence (Reference 1) as described in the FSAR. The RPCS function is allowed to be bypassed in the Rod Action Control System (RACS) if necessary, for example, to insert an inoperable control rod, return an out-of-sequence control rod to the proper in-sequence position or move an in-sequence control rod to another in-sequence position. The requirement that a second qualified individual verify such bypassing and positioning of control rods ensures that the bases for RPCS limitations are not exceeded. In addition, if THERMAL POWER is below the low power setpoint, additional restrictions are provided when bypassing control rods to ensure operation at all times within the basis of the control rod drop accident analysis.

The baseline analysis of the rod drop accident is presented in Section 15.4 of the FSAR and the techniques of the analysis are presented in Reference 1. Analyses applicable to the current cycle are addressed in the appropriate cycle-specific documentation.

The RPCS is also designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during higher power operation.

A dual channel system is provided that, above the low power setpoint, restricts the withdrawal distances of all non-peripheral control rods. This restriction is greatest at highest power levels.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core in approximately 90 to 120 minutes. A minimum available quantity of 4530 gallons of sodium pentaborate solution containing a minimum of 5800 lbs. of sodium pentaborate is required to meet a shutdown requirement of 3%. There is an additional allowance of 165 ppm in the reactor core to account for imperfect mixing and leakage. The time requirement was selected to override the reactivity insertion rate due to cooldown following the xenon poison peak and the required pumping rate is 41.2 gpm. The minimum storage volume of the solution is established to allow for the portion below the pump suction that cannot be inserted. The temperature requirement is necessary to ensure that the sodium pentaborate remains in solution.

1. C.J. Paone, "Banked Position Withdrawal Sequence," GE Topical Report, NEDO-21231, January 1977.

STANDBY LIQUID CONTROL SYSTEM (Continued)

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

Relief valves are provided on the SLCS pump discharge piping to protect the SLCS pump and piping from overpressure conditions. Testing of the relief valve setpoint and verifying that the relief valve does not open during steady state operation of the SLCS pumps demonstrates OPERABILITY of the relief valve. The relief valves are ASME Class 2 valves and, as such, have a ± 3% tolerance in the opening pressure from the set pressure, per the ASME Code (Section III - Division 1 Subsection NC-7614.2(b), 1974 Edition).

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

Compliance with the NRC ATWS Rule 10CFR50.62 has been demonstrated by means of the equivalent control capacity concept using the plant specific minimum parameters. This concept requires that each boiling water reactor must have a standby liquid control system with a minimum flow capacity and boron content equivalent in control capacity to 86 gpm for 13% weight sodium pentaborate solution (natural boron enrichment) used for the 251-inch diameter reactor vessel studied in NEDE-24222, Reference 2. The described minimum system parameters (82.4 gpm, 13.6% weight with natural boron enrichment) provides an equivalent control capacity to the 10CFR 50.62 requirement. The techniques of the analysis are presented in a licensing topical report NEDE-31096-P, Reference 3.

Only one subsystem is needed to fulfill the system design basis, and two subsystems are needed to fulfill ATWS rule requirements. An SLCS subsystem consists of the storage tank, one divisional pump, explosive type valve, and associated controls, and other valves, piping, instrumentation, and controls necessary to prepare and inject neutron absorbing solution into the reactor.

2. "Assessment of BWR Mitigation of ATWS, Volume II," NEDE-24222, December 1979.

3. L. B. Claasen et al., "Anticipated Transients Without Scram, Response to NRC ATWS Rule 10CFR50.62," G. E. Licensing Topical Report prepared for the BWR Owners' Group, NEDE-31096-P, December 1985.

BASES

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

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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

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

For single-loop operation, a MAPLHGR limit corresponding to the product of the MAPLHGR, Figure 3.2.1-1, and 0.8 can be conservatively used to ensure that the PCT for single loop operation is bounded by the PCT for two loop operation.

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

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 reference 1.

3/4.2.2 [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 (References 2 and 3).

MCPR operating limits are defined as functions of exposure ($MCPR_e$), flow ($MCPR_f$), and power ($MCPR_p$). The limit to be used at a given operating state is the highest of these three limits.

The purpose of the $MCPR_e$ is to define operating limits for all anticipated exposures during the Cycle. The $MCPR_e$ limits are established for a set of exposure intervals. The limiting transients are analyzed at the limiting exposure for each interval.

The $MCPR_e$ operating limits are established based on the largest delta-CPR calculated at the limiting exposure and ensure that the MCPR safety limit will not be exceeded during the most limiting transient in each of the exposure intervals.

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

The $MCPR_f$s are established to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured. The reference core flow increase event used to establish the $MCPR_f$ is a hypothesized slow flow runout to maximum, that does not result in a scram from neutron flux overshoot exceeding the APRM neutron flux-high level (Table 2.2.1-1 item 2). Two flow rates have been considered. The maximum credible flow during a runout transient depends on whether the plant is in Loop Manual or Non Loop Manual operation. The result of a single failure or single operator error during Loop Manual operation is the runout of one loop because the two recirculation loops are under independent control. Runout of both loops is possible during Non Loop Manual operation because a single controller

MINIMUM CRITICAL POWER RATIO (Continued)

regulates core flow. With this basis, the $MCPR_f$ curves are generated from a series of steady state core thermal hydraulic calculations performed at several core power and flow conditions along the steepest flow control line. In the actual calculations a conservative highly steep generic representation of the 105% steam flow rodline flow control line has been used. Assumptions used in the original calculations of this generic flow control line were consistent with a slow flow increase transient duration of several minutes: (a) the plant heat balance was assumed to be in equilibrium, and (b) core xenon concentration

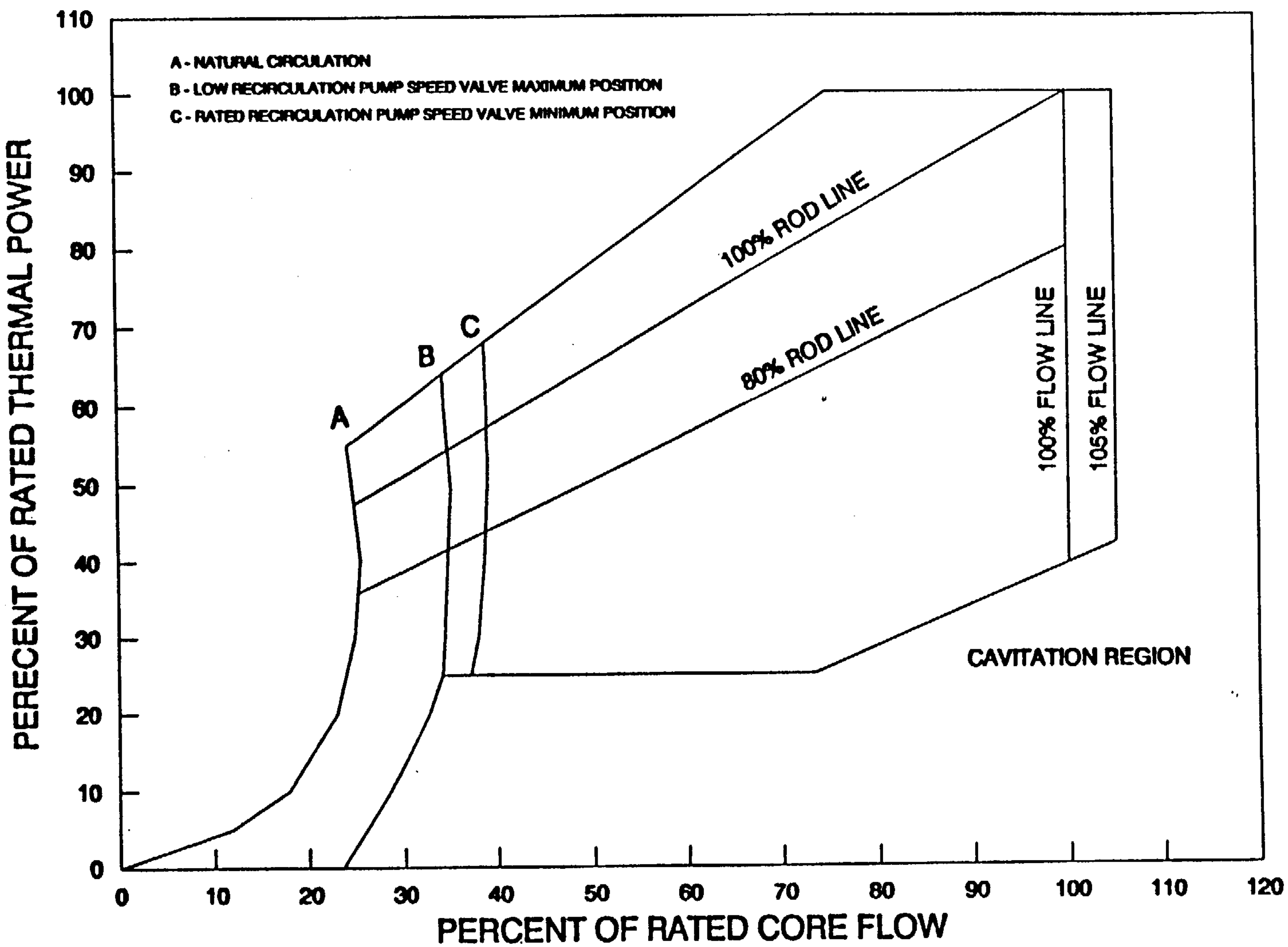

FIGURE B 3/4 2.3-1 POWER-FLOW OPERATING MAP

MINIMUM CRITICAL POWER RATIO (Continued)

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

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

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

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

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 LHGR limits of Figure 3.2.4-1 are multiplied by the smaller of either the flow dependent LHGR factor ($LHGRFAC_f$) or the power dependent LHGR factor ($LHGRFAC_p$) corresponding to the existing core flow and power state to ensure adherence to the fuel mechanical design bases during the limiting transient. $LHGRFAC_f$'s are generated to protect the core from slow flow runout transients. Two curves are provided based on the maximum credible flow runout transient for either Loop Manual or Non-Loop Manual operation. The result of a single failure or single operator error during operation in Loop Manual is the runout of only one loop because both recirculation loops are under independent control. Non-Loop Manual operational modes allow simultaneous runout of both loops because a single controller regulates core flow. $LHGRFAC_p$'s are generated to protect the core from plant transients other than core flow increases.

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.

BASES

MINIMUM CRITICAL POWER RATIO (Continued)

References:

1. XN-NF-80-19(A), Volume 2, "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," Exxon Nuclear Company, September 1982.

2. General Electric Company, "Maximum Extended Operating Domain Analysis," March 1986.

3. AECM-86/0066, "Final Summary Startup Test Report 12," Letter, O.D. Kingsley, MP&L, to J. N. Grace, NRC, February 1986.

4. XN-NF-825(P)(A), Supplement 2, "BWR/6 Generic Rod Withdrawal Analysis; $MCPR_p$ for All Plant Operations Within the Extended Operation Domain," Exxon Nuclear Company, October 1986.

5. GGNS Reactor Performance Improvement Program, Single Loop Operation Analysis, General Electric Final Report, February 1986.

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 applicable fuel thermal limits 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.

The power/flow operating map is divided into four (4) regions. Regions A and B are restricted from operations. They include the operating area above the 80% rod-line and below 40% core flow. Region C includes the operating area above the 80% rod-line and between 40% and 45% core flow. Operation in Region C is allowed only for control rod withdrawals during startup for required fuel preconditioning. Region D consists of the rest of the operating map. No core thermal-hydraulic stability related restrictions are applied to Region D since the potential onset of core thermal-hydraulic instabilities is not predicted within Region D.

The definition of Regions A, B and C is based on BWR stability operational data and required operator actions. Although a large margin to onset of instability was observed in Regions A, B and C during GGNS stability tests for typical operating configuration, a conservative approach is adopted in the specification.

With no reactor coolant system recirculation loops in operation, and the reactor mode switch in the Run position, an immediate reactor shutdown is required. Reactor shutdown is not required when recirculation pump motors are de-energized during recirculation pump speed transfers. Upon entry to Region A an immediate reactor shutdown is required. Upon entry to Region B or Region C, unless operation in Region C is for control rod withdrawals during startup, either a reduction of THERMAL POWER to below the 80% rod-line by control rod insertion or an increase in core flow to exit the region by opening the recirculation loop FCV is required.

Per the specification, the APRM neutron flux noise level should be observed while in Regions B and C. In the unlikely event in which a sustained

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated June 8, 1990 (Ref. 1), Entergy Operations, Inc. (the licensee), requested an amendment to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1 (GG1). Enclosed were five attachments providing the requested Technical Specifications (TS) changes and reports discussing the reload and analyses to support and justify Cycle 5 operation, including two reports by Advanced Nuclear Fuels Corporation (ANF) (Refs. 2 and 3).

By letter dated August 15, 1990 (Ref. 4), the licensee submitted revisions to the original request which presented changes to several of the initially proposed TS and accompanying analyses. Enclosed were revised versions of the five original attachments, including the ANF reports (Refs. 5 and 6). The revision addressed changes to ANF analyses as a result of a separate NRC review of the new CASMO/MICROBURN based methodology and the extended time required for NRC review of the new "TIP" uncertainty in that methodology. This caused ANF to revert (with NRC permission) to the currently accepted (more conservative) TIP values in order to complete, on time, the GG1 Cycle 5 (GG1C5) analyses. The primary result of the TIP change was an increase of 0.01 for the Minimum Critical Power Ratio (MCPR) safety limit and corresponding changes to the operating limit MCPR and associated factors. In addition, (1) the slow flow excursion, and (2) the loss of feedwater heating events were reanalyzed. The first incorporated the increased safety limit in the analysis and the second used the revised (and approved) MICROBURN methodology.

The Cycle 5 reload will replace 284 ANF 8x8 fuel assemblies used in Cycle 4 with ANF 9x9-5 fuel assemblies. The core loading will retain 512 ANF 8x8 fuel assemblies and 4 lead test ANF 9x9-5 assemblies from Cycle 4. The reload for Cycle 5 is generally a normal reload with no unusual features or characteristics other than the partial shift to a 9x9 loading pattern. ANF 9x9 fuel has been used in other reactors, and Susquehanna 2, for example, has been operating with an all ANF 9x9 fuel loading. The revised application did not significantly alter the action previously noticed or affect the initial no significant hazards consideration determination published in the Federal Register on July 25, 1990 (55 FR 30297).

GG1C5 TS changes are not extensive and are primarily related to Average Planar Linear Heat Generation Rate (APLHGR), Linear Heat Generation Rate (LHGR), and Minimum Critical Power Ratio (MCPR) limits and associated factors for Cycle 5 core operation as calculated by ANF. Several of these changes are the results of changes in the ANF methodologies. Some of this change has been indicated above as related to the CASMO/MICROBURN methodology and the NRC review. There has also been a change from the use of maximum APLHGR (MAPLHGR) multiplying factors (MAPFAC) to the use of LHGR multiplying factors (LHGRFAC) to provide for fuel design limits. Since LHGR limits are monitored directly, MAPLHGR limits need consider only the requirements for LOCA analyses. Exposure dependent MCPR limits are introduced. In addition to these thermal-hydraulic parameter changes, there is also a change to two Rod Pattern Control System (RPCS) TS reducing the setpoint for turning off the rod action control system (RACS) from 20 to 10 percent, as approved by the NRC staff in Amendment No. 17 to GESTAR II (Ref. 21).

The new methodologies used by ANF for GG1C5 involve the MCPR safety limit (Ref. 12), the ANFB critical power correlation (Ref. 13), the CASMO-3G/MICROBURN-B neutronic code (Ref. 14), and a revised COTRANSA2 (Ref. 15). These methodologies have all been reviewed and approved by the NRC staff (Refs. 10, 16, 17, and 18).

2.0 EVALUATION

2.1 Fuel Design

The GG1C5 reload will include 284 new ANF 9x9-5 fuel assemblies. These contain 76 fuel rods and 5 water rods. The fuel rods are enriched to an average of 3.42 without U-235 with eight to ten of the rods containing gadolinia as burnable poison. The fuel design and safety analysis are described in the GG1C5 Reload Summary Report (Ref. 7 and in Refs. 6 and 8). The methodologies and the application to ANF 9x9-5 have been reviewed and approved by the NRC staff. The fuel mechanical design is similar to ANF 9x9 fuel approved for use in other BWRs, e.g., Susquehanna 2 which currently has a complete ANF 9x9 core loading. The design analyses, using approved methodologies, were performed to support assembly discharge burnups of 39 GWd/MTU for the remaining 8x8 assemblies and 40 GWd/MTU for the 9x9-5 assemblies. The fuel channels to be used for ANF 9x9-5 fuel are manufactured by Carpenter Technology Corporation and are of a similar design and are equivalent to GE channels used in previous cycles. All channels, including those from previous cycles, to be used in GG1C5, and in future cycles, are being used for only a single bundle lifetime. MCPR effects from channel bowing have been included in the safety analyses.

ANF has analyzed the response of the ANF 9x9-5 fuel assemblies during seismic-LOCA events (Ref. 6, Appendix A) by comparison of characteristics to approved 8x8 fuel licensed to operate in GG1. Because of the similarity of the dynamic and hydraulic characteristics of the fuel assemblies and channel boxes, the 9x9-5 fuel will have essentially the same static and dynamic response.

Based on our review of the information presented, and the similarities to previously approved designs and analyses, we find the mechanical design of the ANF 9x9-5 fuel for GG1C5 to be acceptable.

2.2 Nuclear Design

The ANF nuclear design methodology is presented in References 9 and 14. The latter provides new methodology for nuclear design analysis and has recently been reviewed and approved by the NRC staff (Ref. 10).

The beginning of cycle (BOC) shutdown margin is calculated to be 1.06 percent delta-K, and BOC is the most limiting condition. Thus, the cycle minimum shutdown margin is well in excess of the required 0.38 percent delta-K. The standby Liquid Control System also fully meets shutdown requirements. The GG1 high density spent fuel storage racks have been reviewed separately for the acceptability of storing fuel from the Cycle 5 reload, and it was concluded (Ref. 11) that the storage racks can safety accommodate the Cycle 5 fuel.

The GG1C5 nuclear characteristics have been calculated with approved methodologies, the results are reasonable and fall within expected ranges and the review concludes that the design is acceptable.

2.3 Thermal-Hydraulic Design

The retained ANF 8x8 fuel and the new 9x9 fuel are thermal-hydraulically compatible as determined by approved methodologies and the combination has been approved for use in previous BWR reloads, e.g., Susquehanna 1 and 2. The use of the ANF 9x9 fuel in GG1C5 is acceptable from a thermal-hydraulic viewpoint.

The ANF thermal-hydraulic methodology and criteria used for GG1C5 design and analysis is for the most part the same as used for previous GG1 reloads. This is described in References 19 and 20. New aspects of the methodology were introduced in this reload, however. They are described in the ANF topical reports presented in References 12, 13, 14, and 15. They involve the MCPR safety limit, the ANFB critical power correlation, the CASMO-3G/MICROBURN-B neutronics code, and the revised thermal-hydraulic code COTRANSA2. These methodology changes have all been recently reviewed by the NRC staff and approved. The safety evaluation for these reports is presented in References 10, 16, 17, and 18. The methodologies used, including the approved changes, are acceptable to the NRC staff for GG1C5 analysis.

Changes were also made to the format for presenting MAPLHGR and LHGR limits. Formerly, the MAPLHGR limits had multipliers (MAPFAC) for off-rated conditions to provide LHGR protection at these conditions. In the revised format, this function has been transferred to LHGR multiplier ($LHGRFAC_F$ and $LHGRFAC_P$). The basic limits and methodology have not changed in this transfer. The MAPLHGR limit for ANF 8x8 fuel has been determined so as to cover all ANF 8x8 fuel types in Cycle 5, and similarly the MAPLHGR for ANF 9x9 covers all 9x9 fuel types. Also, the LHGR limits and multipliers have been determined to be applicable to all fuel types in the cycle. These changes in the LHGR and MAPLHGR format provide similar limit protection as the previously approved format and are acceptable.

The LHGR limits for the 8x8 fuel have been extended to include the expected end of Cycle 5 burnup. This extension falls within approved methodology (Ref. 3) and is acceptable.

The MCPR and LHGRFAC limits have been extended below 30 percent core flow. This covers both two loop and single loop operation (TLO and SLO). This has been justified by the analysis of the flow run out events, including consideration of both Loop Manual and Non-Loop Manual modes (corresponding to single and dual recirculation loop flow excursions). The analysis used approved methodology and is acceptable.

The MCPR safety limit has been determined to be 1.09 for both TLO and SLO using approved methodology. As discussed previously in this report, a conservative value for the "TIP" uncertainty factor was used as the result of the staff review of the methodology. ANF calculated MCPR operating limits at several cycle exposures and provided an exposure-dependent MCPR operating limit as well as the usual power and flow dependent MCPR limits ($MCPR_P$ and $MCPR_F$). The MCPR operating limits are based on analyses of plant transients to be discussed later. The development of these limits follows approved methodology and is acceptable.

The effect of channel bowing was included in the MCPR analysis by ANF. Channel use will involve only single bundle lifetime. ANF methodology for single bundle lifetime MCPR analysis has been reviewed by the NRC staff, and is acceptable for channel bow analysis of GG1C5.

GG1 is currently operating under the Interim Recommendations of Stability Actions with a TS approved by the NRC staff in a previous review of GG1 thermal-hydraulic stability. The boundaries of the TS designated operating regions were based on the interim recommendation boundaries. ANF has performed calculations of the stability characteristics of GG1 for Cycle 5, which will contain about a third of a core of ANF 9x9 fuel. The analysis indicated that the decay ratio has not changed significantly from Cycle 4 to 5 for equivalent conditions. Measurements by NRC consultants in the Susquehanna 2 reactor, with various amounts of ANF 9x9 fuel from succeeding reloads, including all 9x9 fuel, indicated no significant deterioration of the decay ratio. The NRC staff review thus concludes that continued use of the current stability TS boundaries is acceptable.

2.4 Anticipated Operational Occurrences and Accident Analyses

To provide the basis for the TS values of the various operating limits (MCPR, LHGR, and MAPLHGR), ANF has analyzed the system Anticipated Operational Occurrence (AOO) events which could provide the most limiting conditions. This follows the normal pattern of the approved methodology for operating limit analysis. This included Load Rejection Without Bypass (LRNB), Feedwater Controller Failure (FWCF), Loss of Feedwater Heating (LFWH), Flow Excursion (FE), Control Rod Withdrawal Error (CRWE) and the Fuel Loading Error (FLE). Previous analyses have shown that other events are non-limiting. Plant initial conditions for the analyses covered the full range of Maximum Extended Operating Domain (MEOD) approved for GG1. Analyses were done for End-of-Cycle (EOC), EOC-2000 MWd/MTU and EOC+30 EFPD (Effective Full Power Days) to provide burnup dependent MCPR limits. Results from these analyses were used to provide the TS MCPR and LHGR limits as a function of power, flow, and exposure.

The change to the "TIP" uncertainty, discussed previously, required reanalysis of the slow flow excursion and LFWH events. The former was rerun with the new safety limit MCPR of 1.09 and a complete new analysis was run for the LFWH. The LFWH was analyzed with the newly approved MCROBURN-B/ANFB following the approach previously approved for Cycle 4, using an expanded GG1 data base.

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

ANF also analyzed operation under single loop operation (SLO). The analyses included calculation of the pump seizure event and determination of MCPR and MAPLHGR limits. It was determined that the MCPR safety limit of 1.09 was applicable to SLO and the MAPLHGR limit reduction factor, from LOCA analyses, should be 0.80. The analyses were done with approved methods and are acceptable.

Compliance with overpressurization criteria was demonstrated by analysis of the main steam isolation valve (MSIV) closure event assuming MSIV position switch scram failure. The analyses used conservative parameters and resulted in pressure under required limits. The analysis used approved methods and is acceptable.

ANF analyzed the Loss of Coolant Accident (LOCA) and Rod Drop Accident (RDA) and determined that required limits are met for GG1C5. The analyses used approved methods and are acceptable.

2.5 Technical Specification Changes

The following Technical Specification (TS) changes have been proposed for operation of GG1C5.

(1) Definition 1.8 -- Administrative change of "XN-3" to "ANFB" to reference correct current correlation. The change is acceptable.

(2) TS 2.12 -- The MCPR safety limit is increased to 1.09 for both TLO and SLO. This has been determined with approved methods for the fuel in GG1C5 and is acceptable.

(3) TS 3/4.1.4.2 -- The power level above which the control rods may be bypassed in the Rod Action Control System (RACS) is reduced from 20 to 10 percent. This setpoint level is related to the control Rod Drop Accident (RDA) analysis. The problem area was reviewed generically by the NRC staff in connection with Amendment No. 17 to GESTAR II (Ref. 21), and permission to lower the setpoint to 10 percent received generic approval. It is acceptable for GG1.

(4) TS 3/4.2.1 -- The changes delete references to fuel type specific MAPLHGR curves, and delete references to MAPFAC curves. As previously discussed, the MAPFAC limits have been transferred to LHGR limits and the analysis of MAPLHGR limits for LOCA (only) have covered ANF 8x8 and AFN 9x9 fuel generically. These changes are acceptable. ANF has also determined that the SLO MAPLHGR multiplier is now 0.80. This change is also acceptable. Figure 3.2.1-1 is changed to show the new ANF 8x8 and 9x9 MAPLHGR values and Figures 3.2.1-1a through 3.2.1-1e, 3.2.1-2, and 3.2.1-3 are deleted because, as indicated above, the values are no longer used. This is acceptable.

(5) TS 3/4.2.3 -- Reference is now made to exposure dependent MCPR limits and minor administrative changes are made. These are acceptable. Changes are made to Figures 3.2.3-1 and 3.2.3-3 to reflect the changes to MCPR limits for the cycle which have been discussed previously. These changes are acceptable.

(6) TS 3/4.2.4 -- The text is changed to reflect the addition of the LHGRFAC multipliers as has been previously discussed. Figures 3.2.4-1, 3.2.4-2, and 3.2.4-3 have been revised or added to reflect these changes and the extended ANF LHGR curve (also previously discussed). The changes are acceptable.

(7) TS 3/4.4.1.1 -- Reference to TS 2.1.2 is deleted since the changes to TS 2.1.2, discussed above, make reference to this TS unnecessary. This is acceptable.

(8) TS 3/4.10.2 -- As was discussed above for TS 3/4.1.4.2, the setpoint for the RACS is reduced from 20 to 10 percent. The change is also acceptable for this specification.

There are also changes to the Bases associated with the above TS to reflect the changes to the specifications or minor administrative changes. The changes suitably reflect the basis for the changes and are acceptable. These include Bases 2.1.1 and 2.1.2; 3/4.1.4 and 3/4.1.5; 3/4.2.1, 3/4.2.3, and 3/4.2.4; and 3/4.4.1.

2.6 SUMMARY

The NRC staff has reviewed the reports submitted for the Cycle 5 operation of Grand Gulf Unit 1 and concludes that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design, and transient and accident analyses are acceptable. The Technical Specification changes submitted for this reload suitably reflect the necessary modifications for operation in this cycle.

2.7 References

1. Letter from W. T. Cottle, Entergy Operations, Inc. (EOI) to NRC, "Cycle 5 Reload," dated June 8, 1990.

2. ANF-90-21, Revision 1, "Grand Gulf Unit 1 Cycle 5 Plant Transient Analysis," dated May 23, 1990.

3. ANF-90-21, Revision 1, "Grand Gulf Unit 1 Cycle 5 Plant Transient Analysis," dated May 24, 1990.

4. Letter from W. T. Cottle, EOI, to NRC, "Cycle 5 Reload," dated August 15, 1990.

5. ANF-90-22, Revision 2, "Grand Gulf Unit 1 Cycle 5 Plant Transient Analysis," dated August 8, 1990.

6. ANF-90-21, Revision 2, "Grand Gulf Unit 1 Cycle 5 Plant Transient Analysis," dated August 8, 1990.

7. Attachment 2 to AECM-90/0146, "Grand Gulf Nuclear Station Unit 1 Cycle 5 Reload Summary Report," dated August 1990.

8. "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," XN-NF-85-67(P)(A), Revision 1, Exxon Nuclear Company, Richland, Washington, September 1986.

   "Generic Mechanical Design for Advanced Nuclear Fuels 9x9-5 BWR Reload Fuel," ANF-88-152(P), Amendment 1, September 1989, Advanced Nuclear Fuels Corporation, Richland, Washington.

9. XN-NF-80-19(A), Volume 1, Supplements 1 & 2, "Exxon Nuclear Methodology for Boiling Water Reactors: Neutronics Methods for Design and Analysis," Exxon Nuclear Co., March 1983.

10. Letter to R. Copeland, ANF, from A. Thadani, NRC, "Acceptance for Referencing of Topical Report XN-NF-80-19(P), Volume 1, Supplement 3, 'Advanced Nuclear Fuels Methodology for Boiling Water Reactors: Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology,'" dated August 13, 1990.

11. Letter to W. T. Cottle, EOI, from L. Kintner, NRC, "Criticality Analysis for Cycle 5 Fuel in Spent Fuel Storage Racks," dated July 16, 1990.

12. XN-NF-524(P), Revision 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors," including Supplements, Advanced Nuclear Fuels Corporation, April 1989.

13. ANF-1125(P), Supplement 1, "ANFB Critical Power Correlation," Advanced Nuclear Fuels Corporation, April 1989.

14. XN-NF-80-19(P), Volume 1, Supplement 3, "ANF Methodology for BWRs: Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology," Advanced Nuclear Fuels Corporation, February 1989, as supplemented by ANF letter RAC:083:90 dated July 20, 1990.

15. ANF-913, Volume 1, Supplements 1, 2, and 3, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analysis," Advanced Nuclear Fuels Corporation, June 1989.

16. Letter to R. Copeland, ANF, from A. Thadani, NRC, "Acceptance for Referencing of Topical Report ANF-524(P) Revision 2, 'ANF Critical Power Methodology for Boiling Water Reactors,'" dated August 8, 1990.

17. Letter to R. Copeland, ANF, from A. Thadani, NRC, "Acceptance for Referencing of Topical Report ANF-1125(P) and Supplement 1, 'ANFB Critical Power Correlation,'" dated March 8, 1990.

18. Letter to R. Copeland, ANF, from A. Thadani, NRC, "Acceptance for Referencing of Licensing Topical Report ANF-913, 'COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses'" (TAC No. 68356), dated May 23, 1990.

19. XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Co., June 1986.

20. XN-NF-80-19(P)(A), Volume 3, Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description," Exxon Nuclear Co., January 1987.

21. NEDE-24011-P-A-9, "General Electric Standard Application for Reactor Fuel" (GESTAR II), September 1988.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact was published in the Federal Register on October 2, 1990 (55 FR 40428).

Accordingly, based on the Environmental Assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

4.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration, which was published in the Federal Register on July 25, 1990 (55 FR 30297), and consulted with the State of Mississippi. No public comments or requests for hearing were received, and the State of Mississippi did not have any comments.

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

Dated: November 15, 1990

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