Attachment 2 to GNRO-92/00093

PROPOSED CHANGE TO THE OPERATING LICENSE

GENERIC LETTER 88-16

REMOVAL OF CYCLE SPECIFIC PARAMETERS FROM TECHNICAL SPECIFICATIONS

(GGNS PCOL-92/07)

A. SUBJECT: Generic Letter 88-16 - Removal of Cycle Specific Parameters from Technical Specifications

Technical Specifications: index, 1.7a (new), 2.1.2, 3/4.2.1, 3/4.2.3, 3/4.2.4, 5.3.1, 6.9.1 and bases.

Affected Pages: Index pages i, iv and xx. TS pages 1-2, 2-1, 3/4 2-1 through 3/4 2-7c, 5-5, 6-19, B 3/4 2-1, B 3/4 2-7.

B. DISCUSSION:

Cycle-specific parameters limits are evaluated every reload fuel cycle, which can result in the need for license amendments to the Grand Gulf Unit 1 Technical Specifications (TS). When the methodology for determining these limits is documented in an NRC-approved Topical Report or in a plant-specific submittal, the NRC review of the proposed TS changes is primarily limited to confirmation that the updated limits are calculated using an NRC-approved methodology and is consistent with all applicable limits of the safety analysis. For these situations, the NRC determined that this process is an unnecessary burden on licensee and NRC resources.

To simplify this process, Generic Letter (GL) 88-16 (Reference 1) provides specific guidance to remove the cycle specific parameter limits from the TS. This proposed amendment is consistent with GL 88-16 and revises the necessary specifications to remove the cycle specific information such as numerical values (or representation by curves) for certain core and fuel bundle type dependent power distribution limits. In place of the actual numerical value or curve, each specification references a formal report called the Core Operating Limits Report (COLR). The COLR contains the cycle-specific limits developed using methods previously reviewed and approved by the NRC. The COLR will be provided to the NRC for each reload cycle in accordance with the reporting requirements proposed in this request.

This proposed amendment to the Grand Gulf Nuclear Station (GGNS) Technical Specifications (TS) requests changes to the TS index; Specifications 1.0, Definitions; 2.1, Safety Limits; 3/4.2, Power Distribution Limits and bases; 5.3, Design Features; and 6.9, Reporting Requirements.

The proposed changes are described as follows:

1) A new specification is added after 1.7 as 1.7a. Specification 1.7a provides the definition of the CORE OPERATING LIMITS REPORT (COLR).

2) The word "both" is deleted from specification 2.1.2. This is an editorial change.

3) Limiting Condition for Operation (LCO) 3.2.1 and its Action statement are reworded to reference the Core Operating Limits Report for the specific Average Planar Linear Heat Generation Rate (APLHGR) limits. Figure 3.2.1-1 is deleted.

4) Limiting Condition for Operation (LCO) 3.2.3 and its Action statement are reworded to reference the Core Operating Limits Report for the specific Minimum Critical Power Ratio (MCPR) limits. Figures 3.2.3-1, 3.2.3-2 and 3.2.3-3 are deleted.

5) Limiting Condition for Operation (LCO) 3.2.4 is reworded to reference the Core Operating Limits Report for the specific Linear Heat Generation Rate (LHGR) limits. The 3.2.4 Action statement is revised to delete the source of the limits. Figures 3.2.4-1, 3.2.4-2 and 3.2.4-3 are deleted.

6) Section 3/4.2 page numbers and the index are revised to be consistent with the number of pages needed to contain specifications 3/4.2.1, 3/4.2.2, 3/4.2.3 and 3/4.2.4 as follows. This is an editorial change.

3/4.2.1	Page 3/4 2-1
3/4.2.2	Page 3/4 2-2
3/4.2.3	Page 3/4 2-3
3/4.2.4	Page 3/4 2-4

7) The bases of Specification 3/4.2.1 and 3/4.2.4 are revised to be more general and to reflect the application of Generic Letter 88-16.

8) The fuel assemblies design features in TS 5.3.1 is revised consistent with the use of a Core Operating Limits Report.

9) A new specification (6.9.1.11) is added to section 6.9.1, Routine Reports, that describes the administrative requirements for the Core Operating Limits Report.

10) The index is revised to add the CORE OPERATING LIMITS REPORT subsection and page numbers are revised/added to reflect changes described in items 1 and 8 above. This is an editorial change.

C. JUSTIFICATION:

The proposed amendment removes the values of the cycle-specific parameter limits from the Grand Gulf Technical Specifications and makes related administrative and format changes based on guidance provided in Generic Letter (GL) 88-16. As stated in GL 88-16, three separate actions are needed to modify the Technical Specifications (TS):

- the addition of the definition of a named formal report that includes the values of cycle-specific parameter limits established using an NRC-approved methodology and consistent with all applicable limits of the safety analysis,

- the addition of an administrative reporting requirement to submit the formal report on cycle-specific parameter limits to the Commission for information and,

- the modification of the individual TS to note that cycle-specific parameters shall be maintained within the limits provided in the defined formal report.

The cycle specific information from the applicable TS is assigned to separate entries within the COLR. Controlled copies of the COLR will be maintained at GGNS and revised in accordance with the requirements for future GGNS fuel cycles. Furthermore, the submittal of the COLR to the Commission will allow the Staff to continue to trend and review the values of these limits as stated in GL 88-16.

The cycle specific parameters to be located in the Core Operating Limits Report are as follows:

Average Planar Linear Heat Generation Rate (APLHGR) Limits

The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) power distribution limit curves are provided to address all fuel types in the core. These limits are determined by considering both thermal-hydraulic and emergency core cooling system performance. New limit values may have to be added to TS as new fuel designs are utilized. The Single Loop Operation (SLO) multiplier can be revised based on analyses performed each cycle. To eliminate license amendment requests to update MAPLHGR limits, the MAPLHGR curves and SLO multiplier should be relocated to the COLR.

Minimum Critical Power Ratio (MCPR) Limits:

MCPR limits are required to prevent exceeding fuel design and safety criteria in the event of Anticipated Operational Occurrences (AOO). The MCPR limits are determined for various power/flow/exposure conditions throughout a given cycle. The applicability of these MCPR limits for specific fuel designs and core configurations is evaluated for each fuel cycle [References 3 and 4]. Because of the potential for cycle to cycle variations in these limits, it is appropriate to remove them from the TS and relocate them to the COLR.

Linear Heat Generation Rate (LHGR)

The LHGR limits are fuel-design and core configuration dependent. These limits are required to prevent exceeding fuel design and safety criteria. The LHGR limits are determined for various power/flow/exposure conditions to protect the fuel during off-rated condition transient events [References 3 and 4]. The applicability of these LHGR limits for specific fuel designs and core configurations is evaluated for each fuel cycle. TS license amendment requests may be required to modify the LHGR power distribution limits or the off-rated factors for each cycle. Therefore, consistent with the relocation of the APLHGR and MCPR limits, the LHGR limits should also be relocated to the COLR.

As stated in GL 88-16, the proposed amendment will result in a resource savings for the licensee and the NRC by eliminating the majority of license amendment requests for changes in values of cycle-specific parameter limits in the TS. Indirectly, this is a safety improvement because these saved resources can be utilized on more important tasks.

The description of fuel assembly design features is revised to delete the reference to the initial core design and to require that designs be developed and analyzed using NRC-approved codes and methods. Additionally, the design description requires fuel assembly types to be identified in the COLR. This is consistent with the Technical Specification Improvement Program and Reference 6.

The editorial change to delete the word "both" from specification 2.1.2 clarifies this specification consistent with Reference 5. This word was inadvertently retained in Amendment 99 to the TS issued May 28, 1992. The remaining editorial changes are justified by the need to maintain the TS as an organized document.

D. NO SIGNIFICANT HAZARDS CONSIDERATIONS:

The Commission has provided standards for determining whether a no significant hazards consideration exists as stated in 10CFR50.92(c). A proposed amendment to an operating license involves a no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

Entergy Operations Inc. has evaluated the no significant hazards considerations in its request for a license amendment. In accordance with 10CFR50.91(a), Entergy Operations Inc. is providing the analysis of the proposed amendment against the three standards in 10CFR50.92(c). A description of the no significant hazards considerations determination follows:

1. The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated.

   The proposed amendment is in accordance with the guidance provided in Generic Letter (GL) 88-16 for removing the cycle-specific parameter limit values from Technical Specifications (TS). There will be no changes in the operation of the facility as a result of these changes. No safety-related equipment or function will be altered. The proposed amendment merely relocates cycle-specific parameter limits from the TS to the Core Operating Limits Report (COLR). The TS is revised to reference their inclusion in the COLR. NRC-approved analytical methodologies will continue to be used as the bases for the limits contained in the COLR. The editorial changes only clarify a specification or provide organizational notation in the TS.

   Removal of the cycle-specific parameter limits from the Grand Gulf Nuclear Station (GGNS) Technical Specifications has no influence or impact on the probability of any accident or malfunction evaluated in the GGNS Updated Final Safety Analysis Report (Reference 2). The cycle-specific operating limits, although not in the TS, will still be followed in operating GGNS. The proposed amendment requires exactly the same

actions to be taken if limits are exceeded as are required by the current TS. Therefore, the consequences of any accident previously evaluated have not increased.

Based on the above, these proposed changes cannot increase the probability or consequences of any accident previously evaluated.

2. These changes would not create the possibility of a new or different kind of accident from any previously analyzed.

   As stated above, no safety-related equipment, safety functions, or operating practices are altered as a result of these changes. No new accident modes are created. The proposed amendment is in accordance with the guidance provided in Generic Letter (GL) 88-16 for removing the cycle-specific parameter limit values from TS. The establishment of these limits in accordance with NRC approved methodologies and incorporating these limits in the Core Operating Limits Report will ensure that a new and different kind of accident is not created.

   The removal of the cycle-specific limits has no influence on, nor does it contribute in any way, to the possibility of a new or different kind of accident or malfunction from those previously analyzed. Cycle-specific limits are calculated using NRC approved methods. Technical Specifications will continue to require operation within the required core operating limits and appropriate actions will be taken when or if limits are exceeded. Therefore, the proposed changes do not create the possibility of a new or different type of accident from any accident previously analyzed.

3. The proposed changes do not involve a significant reduction in a margin of safety.

   The proposed changes do not alter the requirement that the plant operate within the limits specified for a given operating cycle, nor does it alter the required actions if these limits are not met. The margin of safety is not affected by relocating these limits from the Technical Specifications to the COLR. The margin of safety provided by the current technical specifications is unchanged. The proposed amendment still requires operation within the core limits determined using NRC approved reload design methodologies and appropriate actions when or if limits are violated. Therefore, the proposed changes do not result in a significant reduction in a margin of safety.

Based on the above evaluation, operation in accordance with the proposed amendment involves no significant hazards considerations.

E. REFERENCES:

1. USNRC Generic Letter 88-16, "Removal of Cycle Specific Parameter Limits from the Technical Specifications," October 4, 1988 (MAEC-88/0313).

2. Grand Gulf Nuclear Station Final Safety Analysis Report, Updated through Revision 6, Chapter 15.

3. EMF-91-169, Revision 1, "Grand Gulf Unit 1 Cycle 6 Reload Analysis," Siemens Nuclear Power Corporation, July 1992.

4. EMF-91-168, Revision 1, "Grand Gulf Unit 1 Cycle 6 Plant Transient Analysis," Siemens Nuclear Power Corporation, July 1992.

5. Safety Evaluation by the Office of NRR related to Amendment No. 99 to Facility Operating License No. NPF-29, May 28, 1992.

6. NUREG 1434, Standard Technical Specifications, General Electric BWR/6 Plants, Revision 0, September 28, 1992.

GGNS PCOL-92/07 - MARKED-UP TECHNICAL SPECIFICATION PAGES

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INDEX

DEFINITIONS

SECTION

1.0 DEFINITIONS

PAGE

Amendment No. ——

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION

INDEX

ADMINISTRATIVE CONTROLS

SECTION PAGE

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.

Insert

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

Relocate to New page 1-2a

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.

Insert for Page 1-2

CORE OPERATING LIMITS REPORT (COLR)

1.7a The COLR is the Grand Gulf Nuclear Station specific document that provides core operating limits for the current reload cycle. These cycle-specific operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.11. Plant operation within these operating limits is addressed in individual Specifications.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

THERMAL POWER, High Pressure and High Flow

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

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

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

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

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

Delete

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.

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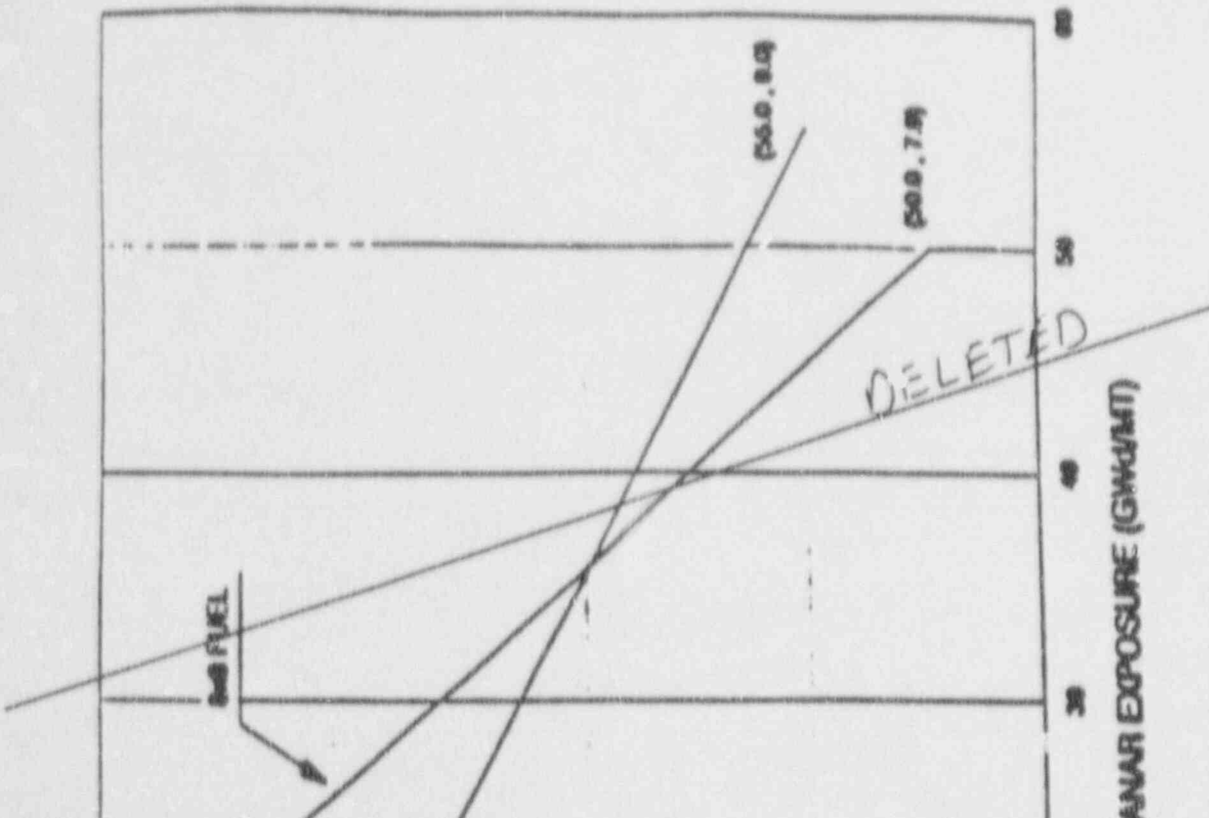

FIGURE 3.2.1-1 MAPLHGR vs AVERAGE PLANAR EXPOSURE

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3/4 2.2 [DELETED]

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, ~~MCPR$_{P}$, or MCPR$_{F}$~~ 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.~~ specified in the CORE OPERATING LIMITS REPORT.

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.

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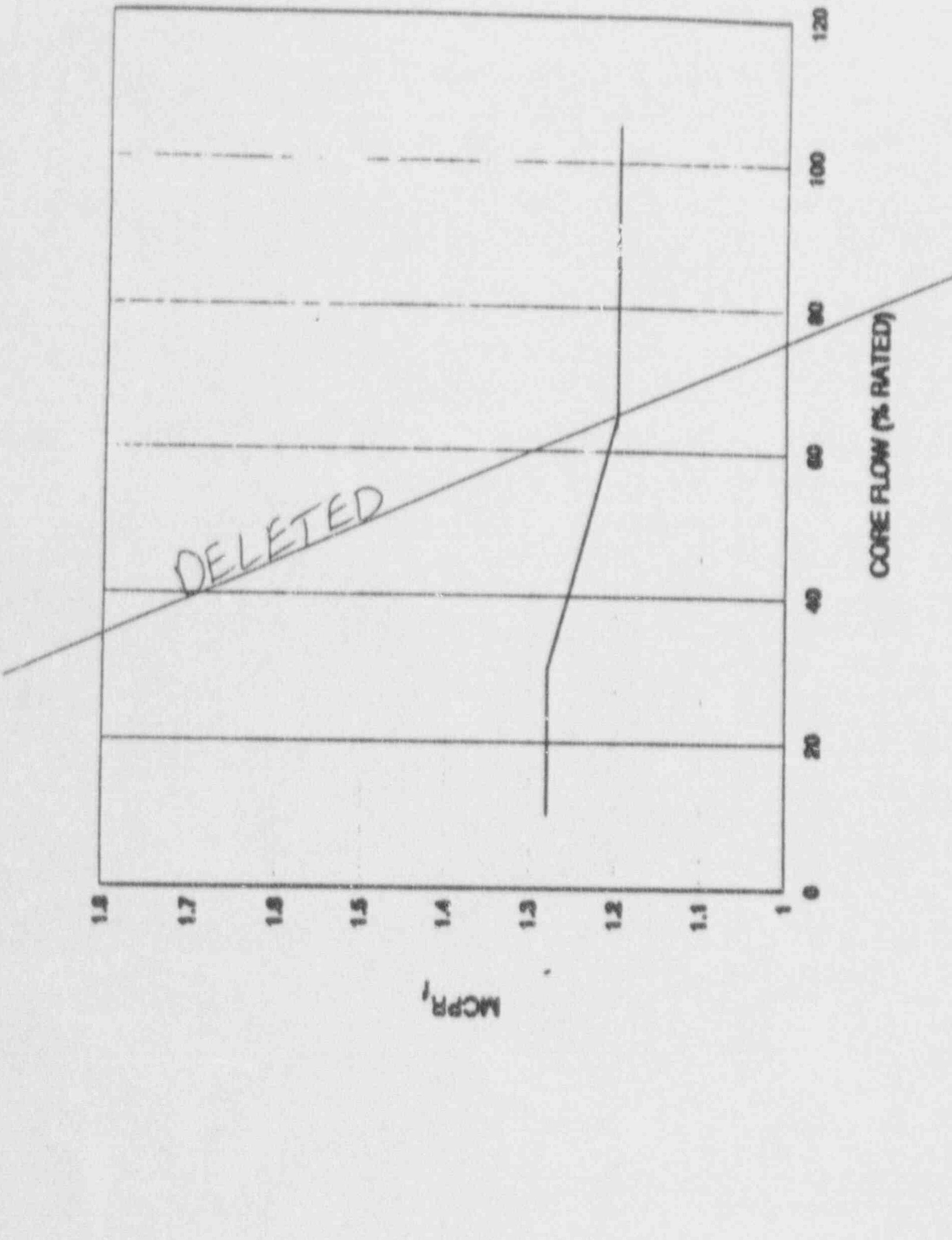

FIGURE 3.2.3-1 MCPR$_f$

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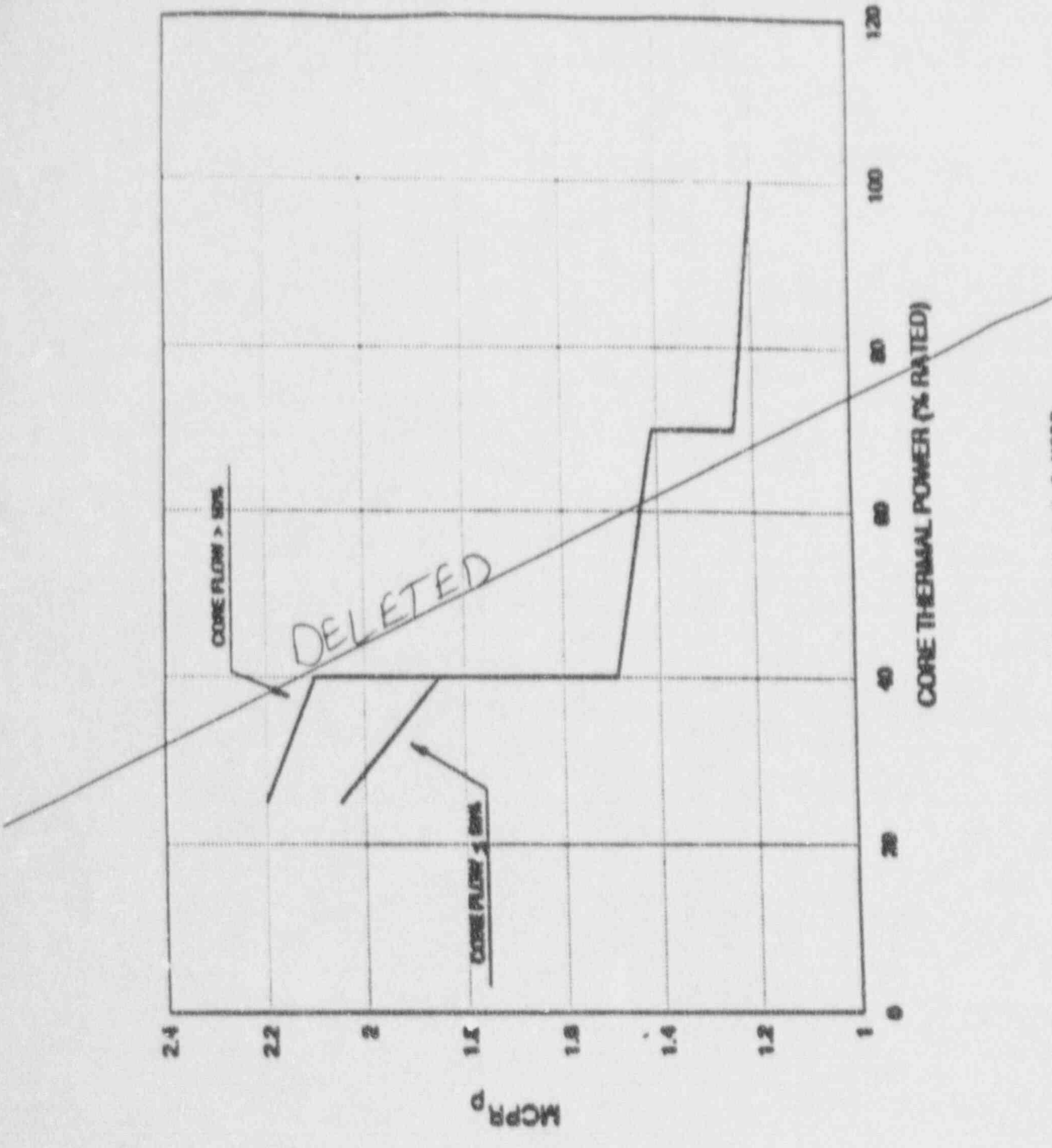

FIGURE 3.2.3-2 MCPR$_p$

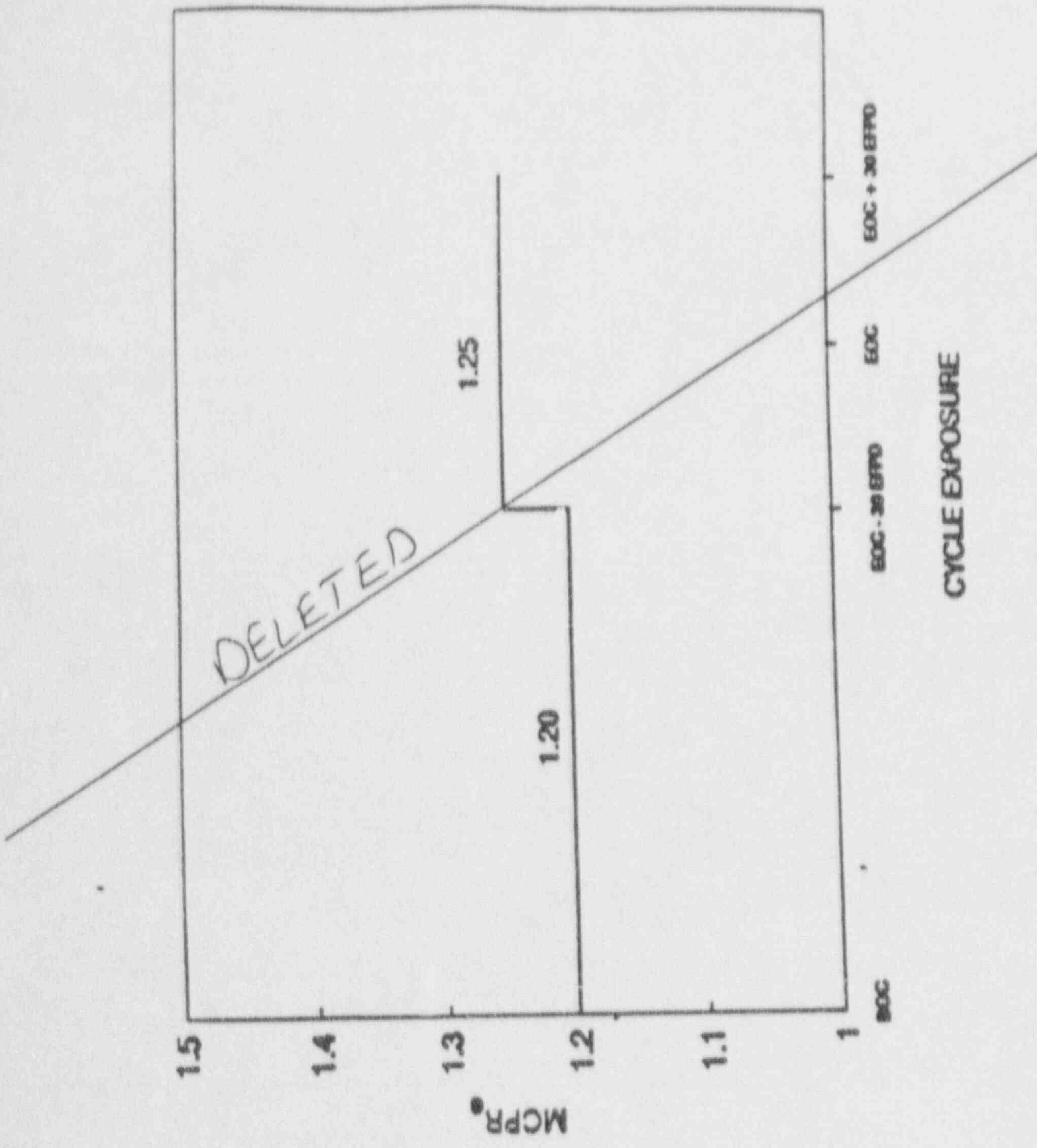

FIGURE 3.2.3-3 MCPR$_e$

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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. DELETE specified in the CORE OPERATING LIMITS REPORT.

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.

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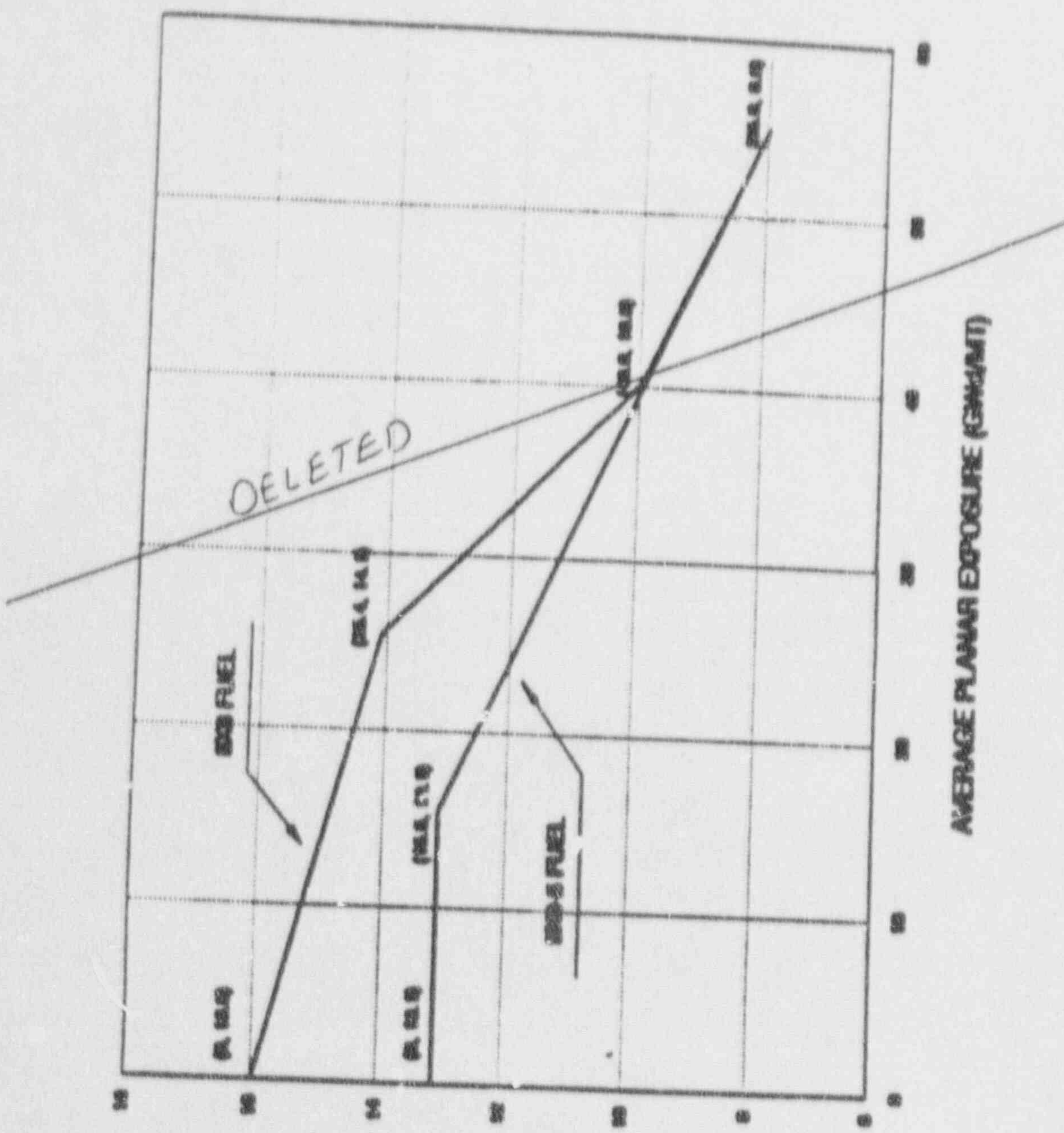

FIGURE 3.2.4-1 LHGR vs AVERAGE PLANAR EXPOSURE

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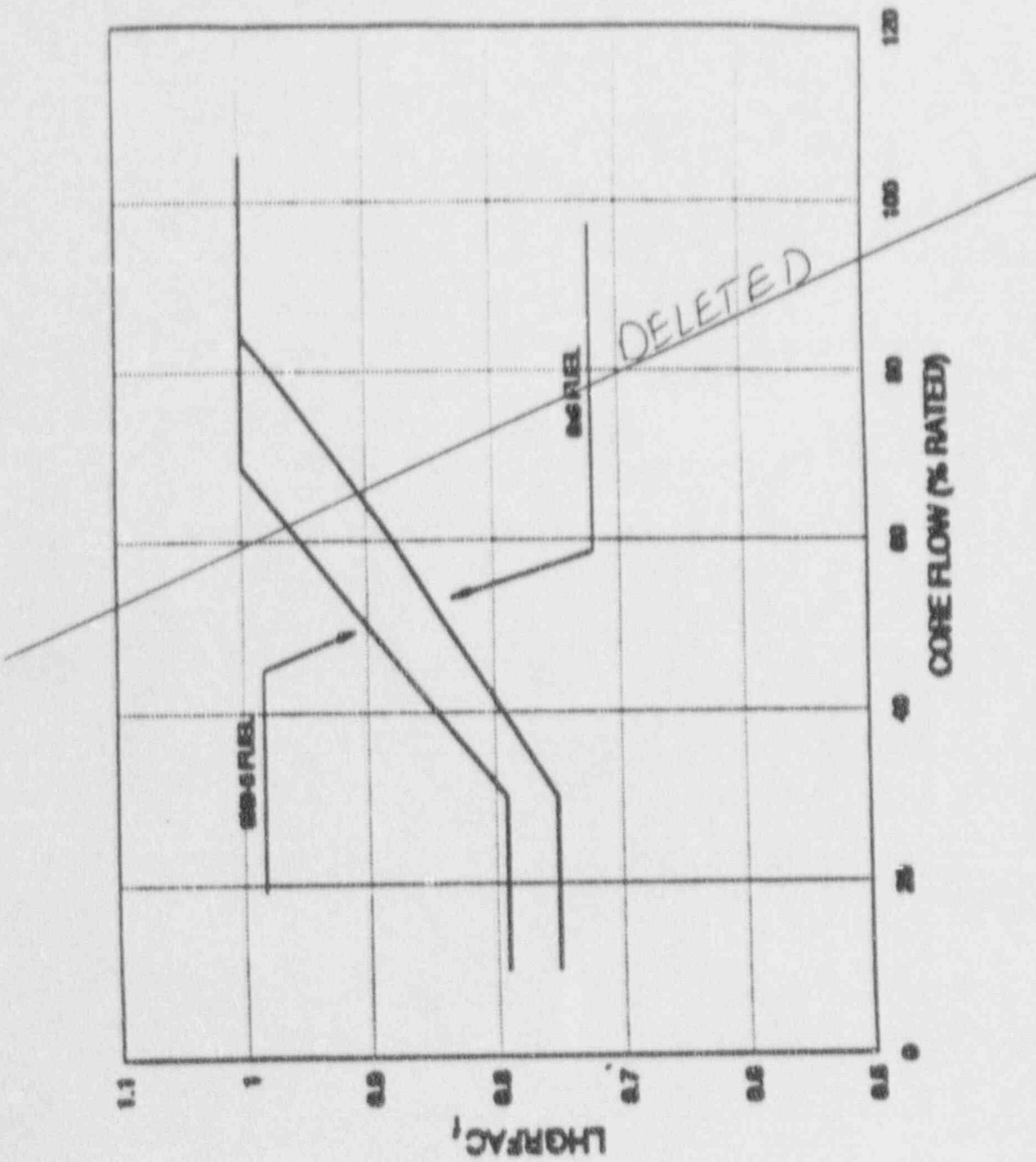

FIGURE 3.2.4-2 LHGRFAC$_f$

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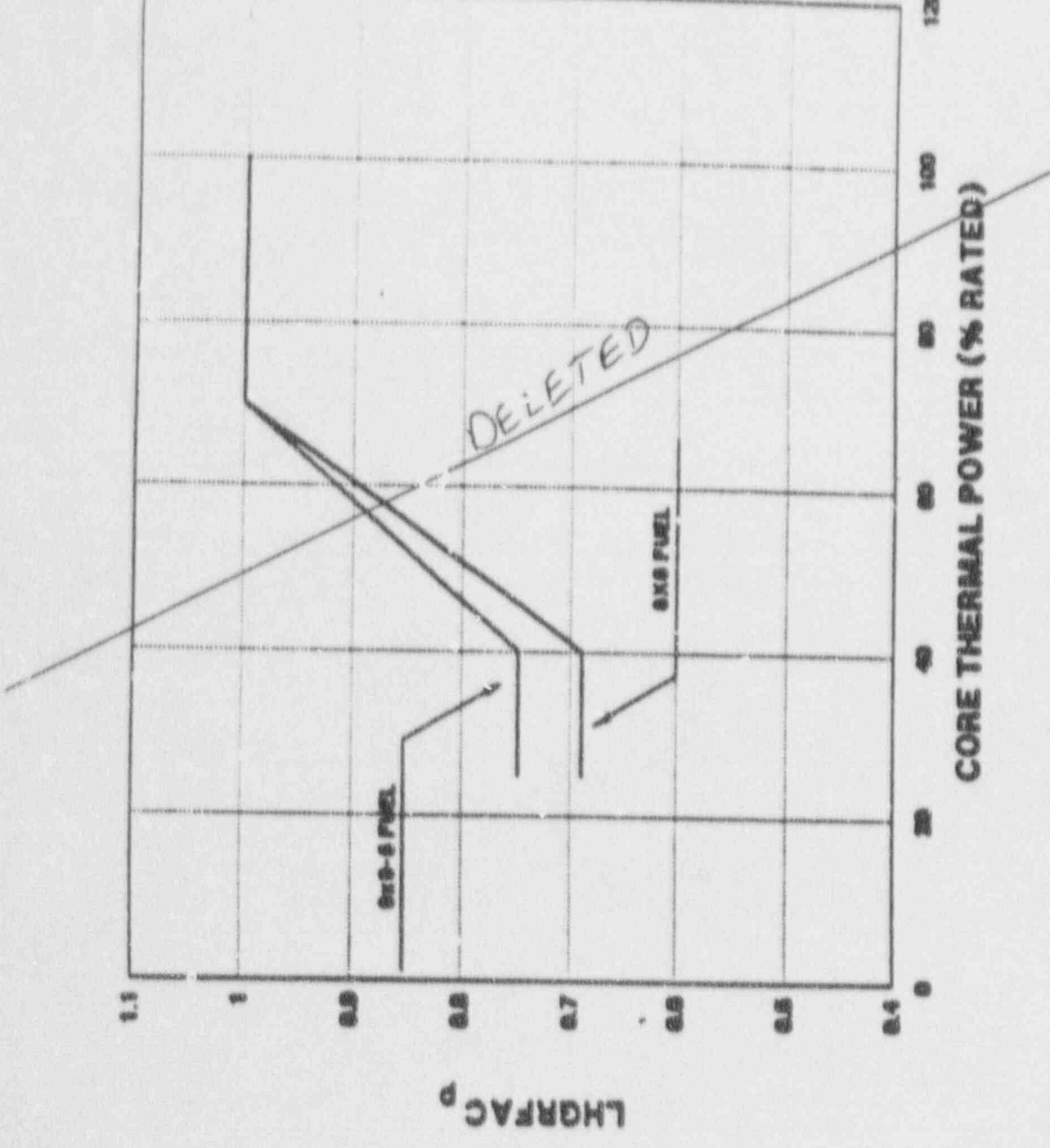

FIGURE 3.2.4-3 LHGRFAC$_p$

3/4.2 POWER DISTRIBUTION LIMITS

BASES

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

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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

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

For single-loop operation, a MAPLHGR limit corresponding to the product of the two loop MAPLHGR~~, Figure 3.2.1-1,~~ and ~~0.86~~ a reduction factor specified in the COLR 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

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. A curve is provided based on the maximum credible flow runout transient for 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. $LHGRFAC_p$'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.

5.3 REACTOR CORE

FUEL ASSEMBLIES

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

CONTROL ROD ASSEMBLIES

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

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

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

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

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pump.

2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.

3. 1550 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

VOLUME

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

Insert for Page 5-5, (end of 5.3.1)

Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC Staff-approved codes and methods, shown to comply with all safety design bases, and are identified in the Core Operating Limits Report.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

c. Principal radionuclide (specify whether determined by measurement or estimate),

d. Type of waste (e.g., spent resin, compact dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from the site to the UNRESTRICTED AREA of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP), OFFSITE DOSE CALCULATION MANUAL (ODCM) or radioactive waste systems made during the reporting period.

MONTHLY OPERATING REPORTS

Insert

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to main steam system safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

SPECIAL REPORTS

Relocate to new page 6-19b

6.9.2 Special reports shall be submitted to the Nuclear Regulatory Commission pursuant to Section 50.4 of 10 CFR Part 50 within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS.

Insert for Page 6-19

CORE OPERATING LIMITS REPORT (COLR)

6.9.1.11 Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the CORE OPERATING LIMITS REPORT (COLR) for the following:

a. The Average Planar Linear Heat Generation Rate (APLHGR) for Technical Specification 3.2.1.

b. The Minimum Critical Power Ratio (MCPR) for Technical Specification 3.2.3.

c. The Linear Heat Generation Rate (LHGR) for Technical Specification 3.2.4.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in the following documents. The appropriate revision/supplement number for each document shall be identified in the Core Operating Limits Report.

1) XN-NF-79-71(P), Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, Exxon Nuclear Company, Inc., Richland, WA.

2) XN-NF-80-19(P)(A), Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, Exxon Nuclear Company, Inc., Richland, WA.

3) XN-NF-80-19(P)(A), Volume 1, Advanced Nuclear Fuels Methodology for Boiling Water Reactors: Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology, Advanced Nuclear Fuels Corporation, Richland, WA.

4) XN-NF-80-19(P)(A), Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description," Exxon Nuclear Company, Inc., Richland, WA.

5) ANF-913 (P)(A) Volume 1, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analysis, Advanced Nuclear Fuels Corporation, Richland, WA.

6) ANF-1125(P)(A), ANFB Critical Power Correlation, Advanced Nuclear Fuels Corporation, Richland, WA.

7) XN-NF-84-105(P)(A), Volume 1, XCOBRA-T: A Computer Code for BWR Transient Thermal Hydraulic Core Analysis, Exxon Nuclear Company, Inc., Richland, WA.

8) XN-NF-573(P), RAMPEX Pellet-Clad Interaction Evaluation Code for Power Ramps, Exxon Nuclear Company, Inc., Richland, WA.

9) XN-NF-81-58(P)(A), RODEX2: Fuel Rod Thermal-Mechanical Response Evaluation Model, Exxon Nuclear Company, Inc., Richland, WA.

10) XN-NF-85-74(P)(A), RODEX2A (BWR): Fuel Rod Thermal-Mechanical Response Evaluation Model, Exxon Nuclear Company, Inc., Richland, WA.

11) XN-CC-33(P)(A), HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option, Exxon Nuclear Company, Inc., Richland, WA.

12) XN-NF-825(P)(A), BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$ for Plant Operation Within the Extended Operating Domain," Exxon Nuclear Company, Inc., Richland, WA.

13) XN-NF-81-51(P)(A), LOCA-Seismic Structural Response of an Exxon Nuclear Company BWR Jet Pump Fuel Assembly, Exxon Nuclear Company, Inc., Richland, WA.

14) XN-NF-84-97(P)(A), LOCA-Seismic Structural Response of an ENC 9x9 BWR Jet Pump Fuel Assembly, Advanced Nuclear Fuels Corporation, Richland, WA.

15) XN-NF-86-37(P), Generic LOCA Break Spectrum Analysis for BWR/6 Plants, Exxon Nuclear Company, Inc., Richland, WA.

16) XN-NF-82-07(P)(A), Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model, Exxon Nuclear Company, Inc., Richland, WA.

17) XN-NF-80-19(A), Volumes 2, 2A, 2B, & 2C, Exxon Nuclear Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model, Exxon Nuclear Company, Inc., Richland, WA.

18) XN-NF-79-59(P)(A), Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies, Exxon Nuclear Company, Inc., Richland, WA.

The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, thermal-hydraulic limits, Emergency Core Cooling System (ECCS) limits, Nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

GGNS PCOL-92/07 - PROPOSED TECHNICAL SPECIFICATIONS

(Information Only)

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DEFINITIONS

SECTION

1.0 DEFINITIONS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

ADMINISTRATIVE CONTROLS

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.

CORE OPERATING LIMITS REPORT (COLR)

1.7a The COLR is the Grand Gulf Nuclear Station specific document that provides core operating limits for the current reload cycle. These cycle-specific operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.11. Plant operation within these operating limits is addressed in individual Specifications.

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 1.06 during two loop operation and 1.07 during single loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

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

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) shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

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, 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.

3/4.2.2 [DELETED]

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 limits specified in the CORE OPERATING LIMITS REPORT.

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, 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.

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.

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 specified in the CORE OPERATING LIMITS REPORT.

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 exceeding the limit, 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.

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 for two loop operation are specified in the CORE OPERATING LIMITS REPORT (COLR).

For single-loop operation, a MAPLHGR limit corresponding to the product of the two-loop MAPLHGR and a reduction factor specified in the COLR 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

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 [illegible]ribution to stabilize. The requirement for calculating MCPR aft[illegible] determining a LIMITING CONTROL ROD PATTERN exists ensures that MCPR wil[illegible]n following a change in THERMAL POWER or power shape, that [illegible]ld p[illegible]ation exceeding a thermal limit.

[illegible]HEAT GENERATION RATE

[illegible]ification assures that the Linear Heat Generation Rate (LHGR) in any [illegible]ess than the design linear heat generation even if fuel pellet densification is postulated.

The LHGR limits 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. A curve is provided based on the maximum credible flow runout transient for 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. $LHGRFAC_p$'s are generated to protect the core from plant transients other than core flow increases.

The daily requirements 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.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 800 fuel assemblies. Each fuel assembly shall contain fuel rods and water rods clad with Zircaloy cladding. Each fuel rod shall have a design nominal active fuel length of 150 inches. Reload fuel shall have mechanical, thermal-hydraulic and neutronic characteristics compatible wi the initial core loading. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC Staff-approved codes and methods, shown to comply with all safety design bases, and are identified in the Core Operating Limits Report.

CONTROL ROD ASSEMBLIES

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

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

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

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

b. For a pressure of:

   1. 1250 psig on the suction side of the recirculation pump.

   2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.

   3. 1550 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

VOLUME

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

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

c. Principal radionuclide (specify whether determined by measurement or estimate),

d. Type of waste (e.g., spent resin, compact dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from the site to the UNRESTRICTED AREA of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP), OFFSITE DOSE CALCULATION MANUAL (ODCM) or radioactive waste systems made during the reporting period.

MONTHLY OPERATING REPORTS

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to main steam system safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT (COLR)

6.9.1.11 Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the CORE OPERATING LIMITS REPORT (COLR) for the following:

a. The Average Planar Linear Heat Generation Rate (APLHGR) for Technical Specification 3.2.1.

b. The Minimum Critical Power Ratio (MCPR) for Technical Specification 3.2.3.

c. The Linear Heat Generation Rate (LHGR) for Technical Specification 3.2.4.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in the following documents. The appropriate revision/supplement number for each document shall be identified in the COLR.

1) XN-NF-79-71(P), Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, Exxon Nuclear Company, Inc., Richland, WA.

2) XN-NF-80-19(P)(A), Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, Exxon Nuclear Company, Inc., Richland, WA.

CORE OPERATING LIMITS REPORT (COLR) (continued)

3) XN-NF-80-19(P)(A), Volume 1, Advanced Nuclear Fuels Methodology for Boiling Water Reactors: Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology, Advanced Nuclear Fuels Corporation, Richland, WA.

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6) ANF-1125(P)(A), ANFB Critical Power Correlation, Advanced Nuclear Fuels Corporation, Richland, WA.

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8) XN-NF-573(P), RAMPEX Pellet-Clad Interaction Evaluation Code for Power Ramps, Exxon Nuclear Company, Inc., Richland, WA.

9) XN-NF-81-58(P)(A), RODEX2: Fuel Rod Thermal-Mechanical Response Evaluation Model, Exxon Nuclear Company, Inc., Richland, WA.

10) XN-NF-85-74(P)(A), RODEX2A (BWR): Fuel Rod Thermal-Mechanical Response Evaluation Model, Exxon Nuclear Company, Inc., Richland, WA.

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12) XN-NF-825(P)(A), BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$ for Plant Operation Within the Extended Operating Domain," Exxon Nuclear Company, Inc., Richland, WA.

13) XN-NF-81-51(P)(A), LOCA-Seismic Structural Response of an Exxon Nuclear Company BWR Jet Pump Fuel Assembly, Exxon Nuclear Company, Inc., Richland, WA.

14) XN-NF-84-97(P)(A), LOCA-Seismic Structural Response of an ENC 9x9 BWR Jet Pump Fuel Assembly, Advanced Nuclear Fuels Corporation, Richland, WA.

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16) XN-NF-82-07(P)(A), Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model, Exxon Nuclear Company, Inc., Richland, WA.

CORE OPERATING LIMITS REPORT (COLR) (continued)

17) XN-NF-80-19(A), Volumes 2, 2A, 2B, & 2C, Exxon Nuclear Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model, Exxon Nuclear Company, Inc., Richland, WA.

18) XN-NF-79-59(P)(A), Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies, Exxon Nuclear Company, Inc., Richla , WA.

The core operating limits shall be determined such that a applicable limits (e.g., fuel thermal-mechanical limits, thermal-hydraulic limits, Emergency Core Cooling System (ECCS) limits, Nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Nuclear Regulatory Commission pursuant to Section 50.4 of 10 CFR 50 within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS.

CORE OPERATING LIMITS REPORT (COLR) - Example

(Information Only)

CORE OPERATING LIMITS REPORT (Example)
GRAND GULF NUCLEAR STATION - CYCLE 6

INTRODUCTION:

This Core Operating Limits Report for Grand Gulf Nuclear Station is prepared in accordance with Technical Specification (TS) 6.9.1.11. The core operating limits in this report were developed using NRC-approved methods in accordance with TS 6.9.1.11.

The cycle-specific core operating limits for the following Grand Gulf Nuclear Station, Unit 1 Technical Specifications are included in this report:

a. The Average Planar Linear Heat Generation Rate (APLHGR) limits for each fuel type for both two-loop and single-loop operation. (Technical Specification 3.2.1)

b. The Minimum Critical Power Ratio (MCPR) operating limit including the power, flow and exposure dependent curves. (Technical Specification 3.2.3)

c. The Linear Heat Generation Rate (LHGR) limit for each fuel type including the power and flow dependent parametric adjustment factor curves, $LHGRFAC_p$ and $LHGRFAC_f$, respectively. (Technical Specification 3.2.4)

GENERAL REFERENCES

1) MAEC-88/0313, Generic Letter 88-16, "Removal of Cycle Specific Parameter Limits from Technical Specifications"

2) EMF-91-169, Revision 1, Grand Gulf Unit 1 Cycle 6 Reload Analysis," Siemens Nuclear Power Corporation, July 1992.

3) EMF-91-168, Revision 2, Grand Gulf Unit 1 Cycle 6 Plant Transient Analysis," Siemens Nuclear Power Corporation, September 1992.

METHODOLOGY REFERENCES (per TS 6.9.11.1)

1) XN-NF-79-71(P), Revision 2, Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, Exxon Nuclear Company, Inc., Richland, WA, November 1981.

2) XN-NF-80-19(P)(A), Volume 1 and Supplements 1 and 2, Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, Exxon Nuclear Company, Inc., Richland, WA, March 1983.

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6) ANF-1125(P)(A) and Supplements 1 and 2, ANFB Critical Power Correlation, Advanced Nuclear Fuels Corporation, Richland, WA, April 1990.

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8) XN-NF-573(P), RAMPEX Pellet-Clad Interaction Evaluation Code for Power Ramps, Exxon Nuclear Company, Inc., Richland, WA, May 1982.

9) XN-NF-81-58(P)(A) and Supplements 1 and 2, Revision 2, RODEX2: Fuel Rod Thermal-Mechanical Response Evaluation Model. Exxon Nuclear Company, Inc., Richland, WA, March 1984.

10) XN-NF-85-74(P)(A), RODEX2A (BWR): Fuel Rod Thermal-Mechanical Response Evaluation Model, Exxon Nuclear Company, Inc., Richland, WA, August 1986.

11) XN-CC-33(P)(A), Revision 1, HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option, Exxon Nuclear Company, Inc., Richland, WA, November 1975.

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13) XN-NF-81-51(P)(A), LOCA-Seismic Structural Response of an Exxon Nuclear Company BWR Jet Pump Fuel Assembly, Exxon Nuclear Company, Inc., Richland, WA, May 1986.

14) XN-NF-84-97(P)(A), LOCA-Seismic Structural Response of an ENC 9x9 BWR Jet Pump Fuel Assembly, Advanced Nuclear Fuels Corporation, Richland, WA, August 1986.

15) XN-NF-86-37(P), Generic LOCA Break Spectrum Analysis for BWR/6 Plants, Exxon Nuclear Company, Inc., Richland, WA, April 1986.

16) XN-NF-82-07(P)(A), Revision 1, Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model, Exxon Nuclear Company, Inc., Richland, WA, November 1982.

17) XN-NF-80-19(A), Volumes 2, 2A, 2B, & 2C, Exxon Nuclear Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model, Exxon Nuclear Company, Inc., Richland, WA, September 1982.

18) XN-NF-79-59(P)(A), Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies, Exxon Nuclear Company, Inc., Richland, WA, November 1983.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (TS 3.2.1)

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

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

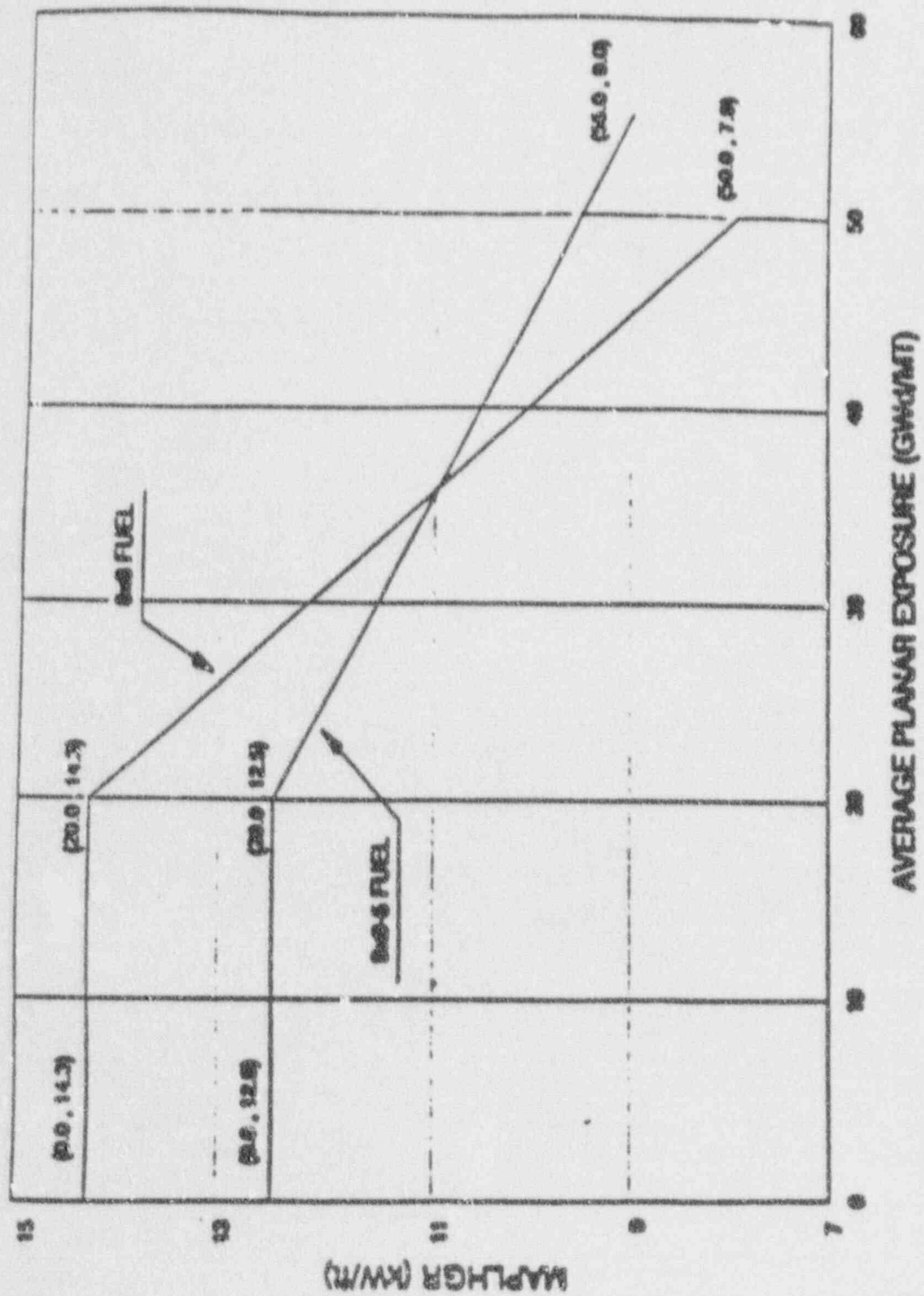

FIGURE 3.2.1-1 MAPLHGR vs AVERAGE PLANAR EXPOSURE

MINIMUM CRITICAL POWER RATIO (TS 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 the indicated core flow, thermal power, and exposure as shown in Figure 3.2.3-1, 3.2.3-2, and 3.2.3-3.

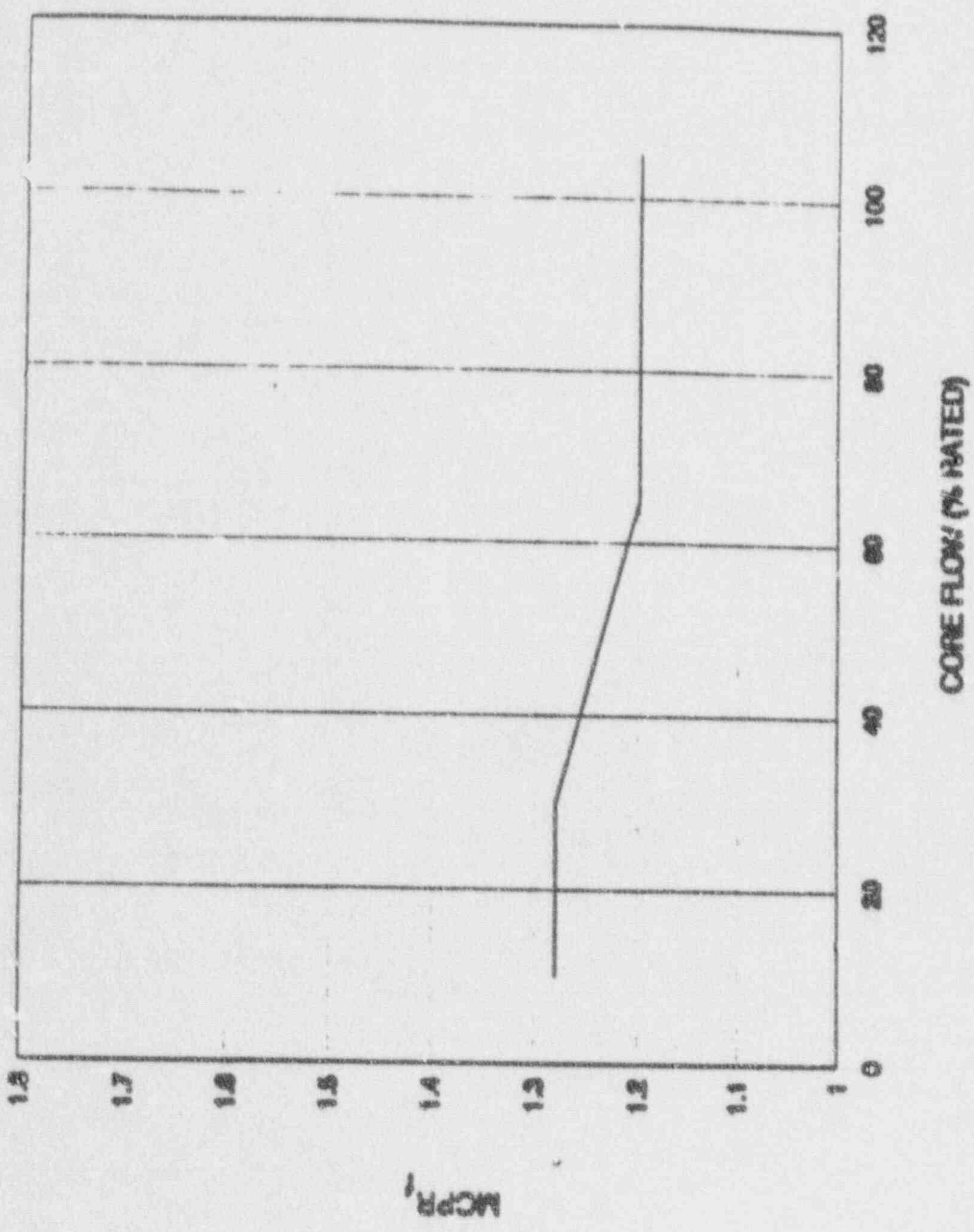

FIGURE 3.2.3-1 $MCPR_f$

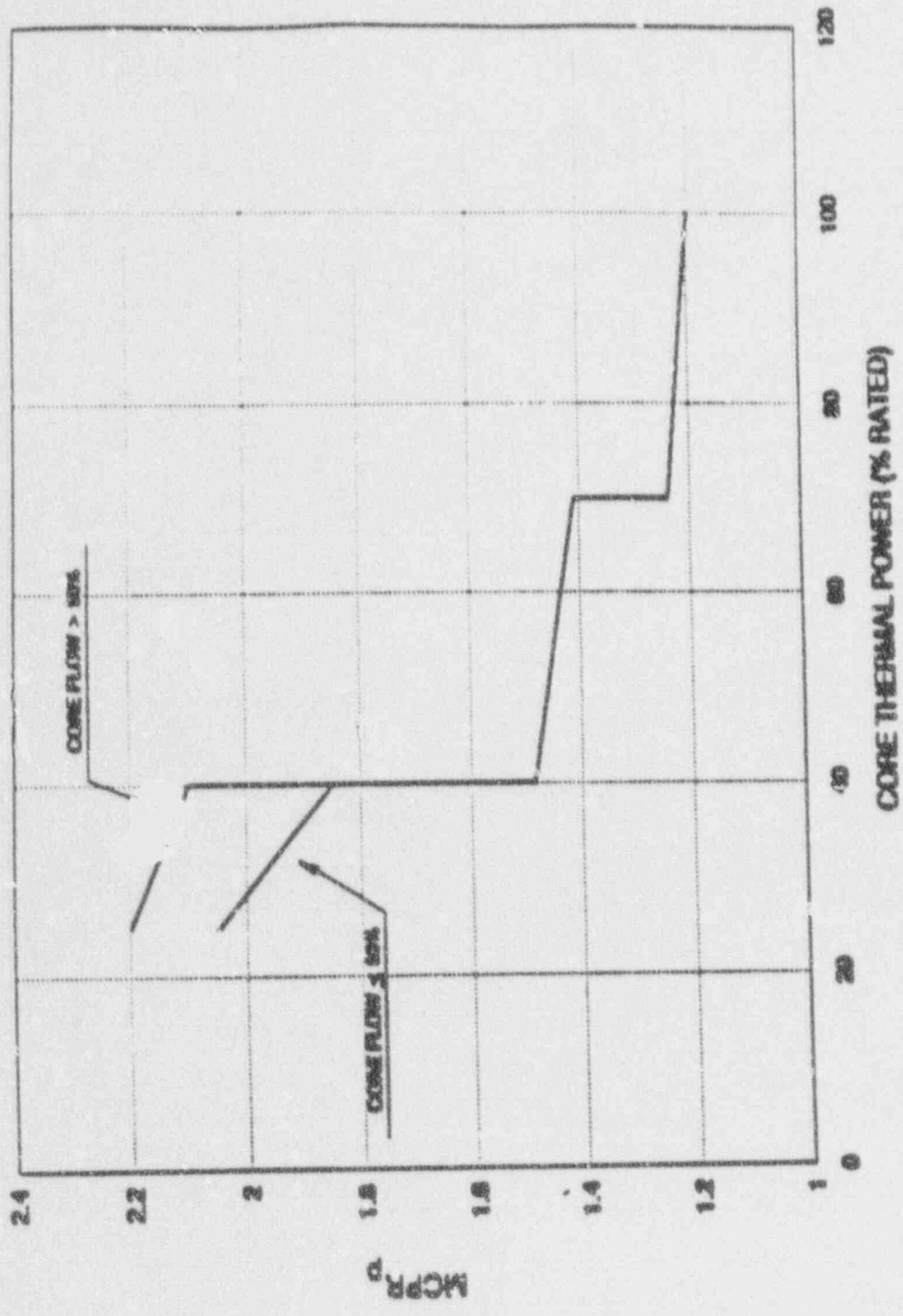

FIGURE 3.2.3-2 $MCPR_p$

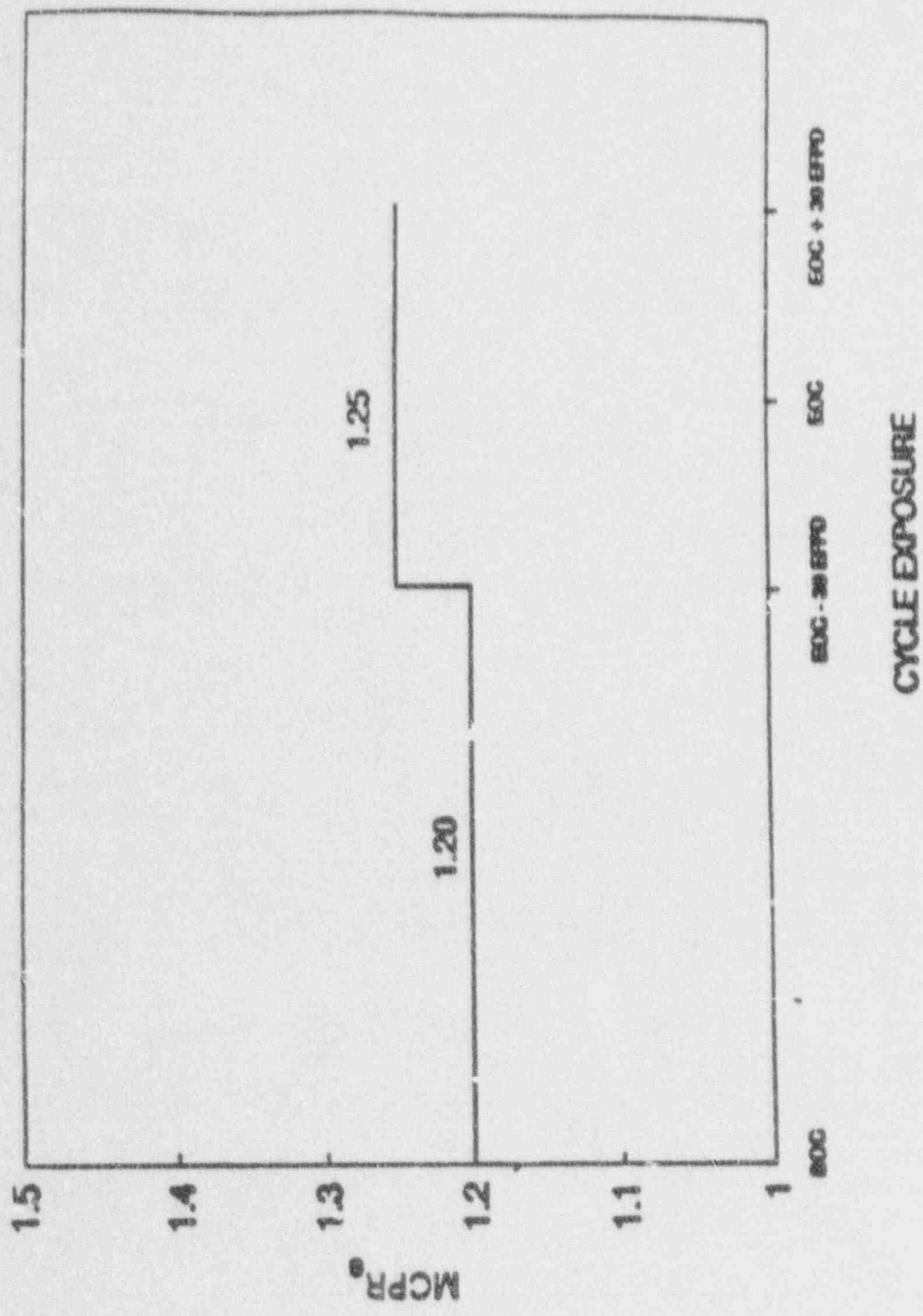

FIGURE 3.2.3-3 $MCPR_e$

LINEAR HEAT GENERATION RATES (TS 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.

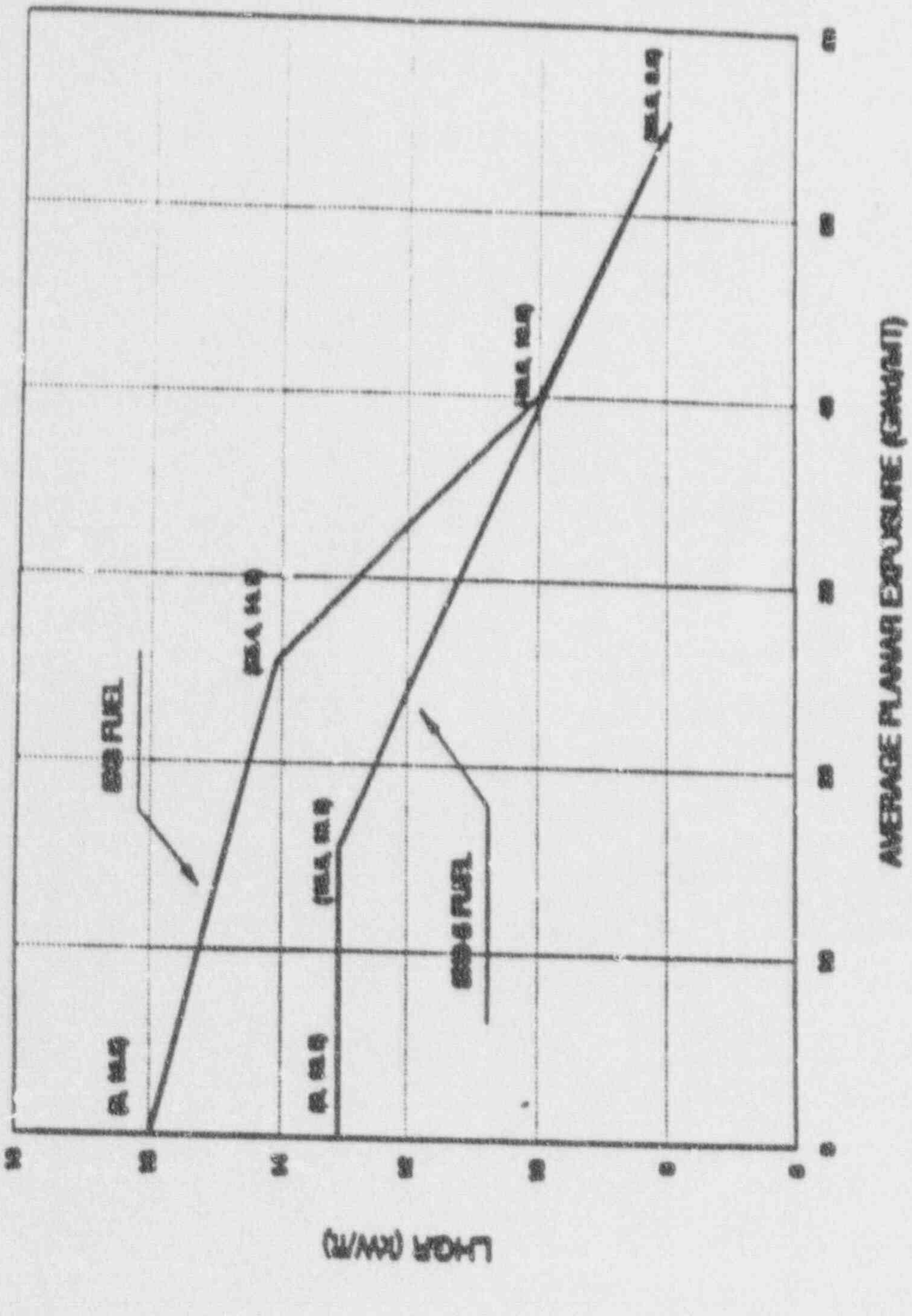

FIGURE 3.2.4-1 LHGR vs AVERAGE PLANAR EXPOSURE

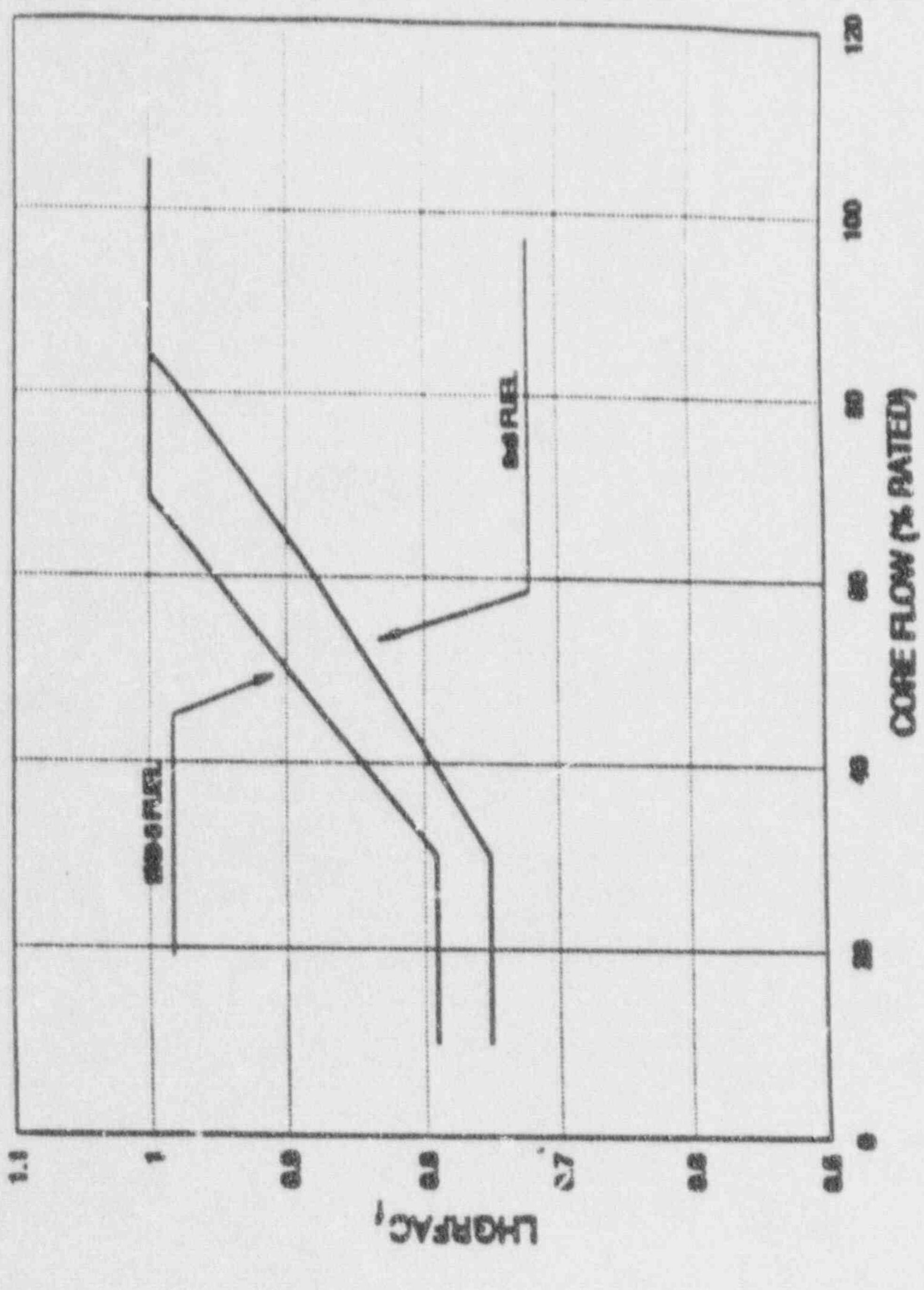

FIGURE 3.2.4-2 LHGRFAC$_f$

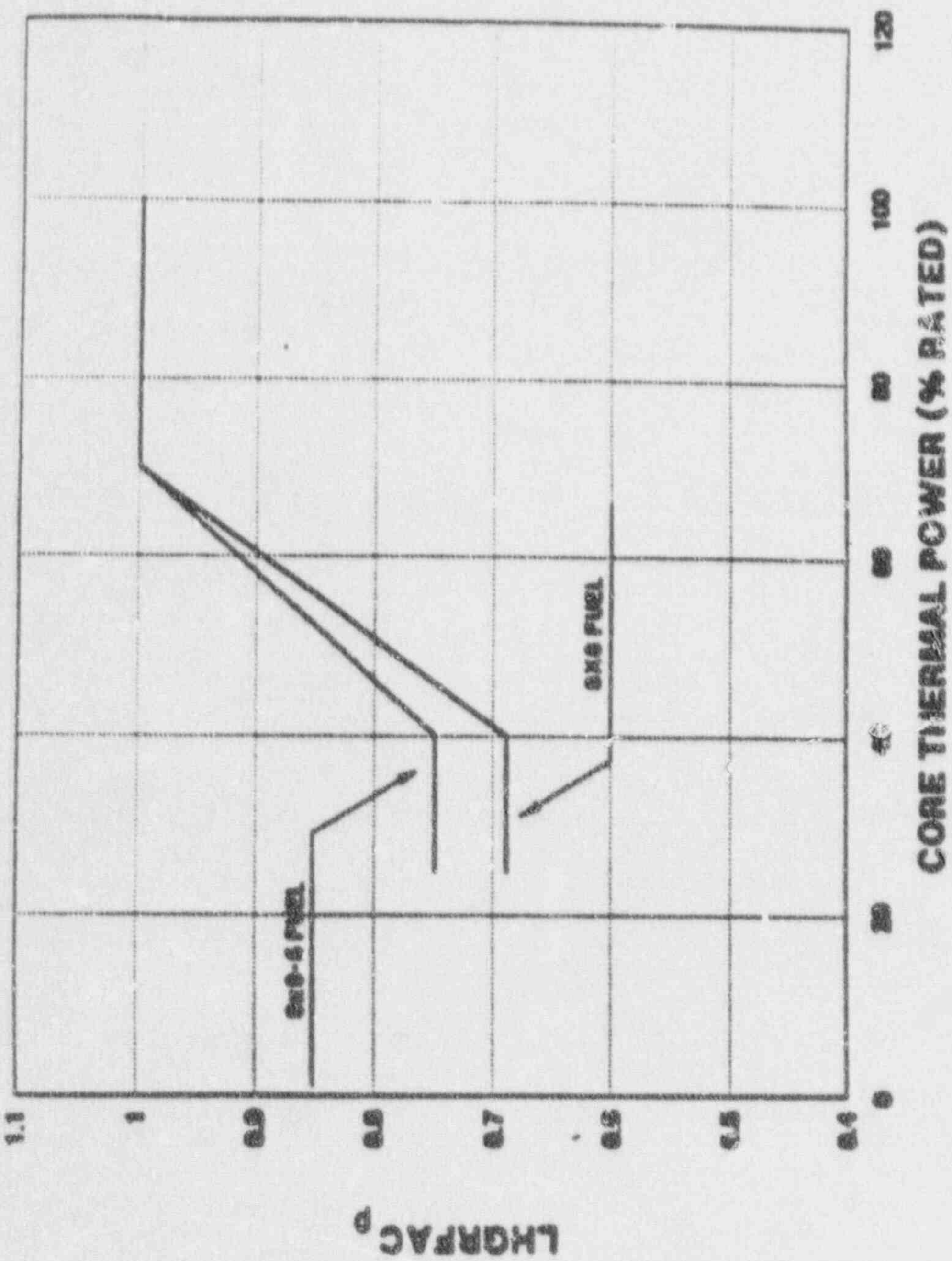

FIGURE 3.2.4-3 LHGRFAC$_p$