

April 26, 1989

Docket No. 50-440

Mr. Alvin Kaplan, Vice President
Nuclear Group
The Cleveland Electric Illuminating
Company
10 Center Road
Perry, Ohio 44081

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Dear Mr. Kaplan:

SUBJECT: AMENDMENT NO. 20 TO FACILITY OPERATING LICENSE NO. NPF-58
(TAC NO. 69525)

The Commission has issued the enclosed Amendment No. 20 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit No. 1. This amendment revises the Technical Specifications (TS) in response to your application dated November 28, 1988 as amended December 29, 1988.

This amendment increases the minimum critical power ratio (MCPR) from 1.06 to 1.07, adds two limiting lattice most-limiting average planar linear heat generation rate (MAPLHGR) curves to the TS to account for new fuel types being used this cycle, and deletes the MAPLHGR curve for natural uranium bundles.

Additionally, limiting conditions for operation and action statements for the APLHGR are being revised to reflect the lattice-dependent MAPLHGR limits in the GESTAR analysis and the default limits in the TS for hand calculations. Figure 3.2.2-1 of the TS is being revised to correct the extrapolated value for the flow-dependent MCPR and Figure 3.2.1-4 is being revised to extend the flow-dependent MAPLHGR factor down to the 20% rated core flow line. Curves A-A' and B-B' are being deleted from the current set of MCPR parametric curves and the TS for linear heat generation rate (LHGR) are being revised to reflect the higher LHGR associated with the new fuel. The definition of "critical power ratio" is being generalized and clarification of how power-dependent MAPLHGR factors are applied to lattice MAPLHGR's is being added. Various figures and pages are being renumbered and the associated bases for the above TS changes are being revised.

DF01
||

K/M

Mr. Alvin Kaplan

-2-

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

Sincerely,

/s/

Timothy G. Colburn, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects - III, IV, V
& Special Projects
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 20 to License No. NPF-58
2. Safety Evaluation

cc w/enclosures:
See next page

Office: LA/PDIII-3
Surname: PKreutzer
Date: 2/14/88

3/31/88

2/14/88
PM/PDIII-3
for: TColburn/tg
2/14/88

Rec 2/16/89
Rec 3/31/89
Rec 4/26/89

PD/PDIII-3
JHannon

2/16/88

at m for J.H.
4/8/89

OGC-WF1

4/18/88

Handwritten note: OGC-WF1 to SE; ask SPOB re issuance



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

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 20
License No. NPF-58

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by The Cleveland Electric Illuminating Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, and Toledo Edison Company (the licensees) dated November 20, 1988, as amended December 29, 1988 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-58 is hereby amended to read as follows:



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

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.

DOCKET NO. 50-440

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Amendment No. 20
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 - 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-58 is hereby amended to read as follows:

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The Cleveland Electric
Illuminating Company

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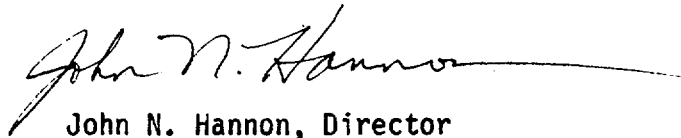
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(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 20 are hereby incorporated into this license. The Cleveland Electric Illuminating Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION



John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 26, 1989

ATTACHMENT TO LICENSE AMENDMENT NO.20

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Overleaf pages are provided to maintain document completeness.

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1.0 DEFINITIONS

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

ACTION

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

AVERAGE PLANAR EXPOSURE

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

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

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

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

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

DEFINITIONS

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 a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.8 The CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) shall be the highest value of the FLPD which exists in the core.

CRITICAL POWER RATIO

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

DOSE EQUIVALENT I-131

1.10 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

1.11 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 automatic isolation system, or
 2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position.
- b. The drywell equipment hatch is closed and sealed.
- c. The drywell head is installed and sealed.
- d. The drywell air lock is in compliance with the requirements of Specification 3.6.2.3.
- e. The drywell leakage rates are within the limits of Specification 3.6.2.2.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

THERMAL POWER, High Pressure and High Flow

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.07 and the reactor vessel steam dome pressure greater than 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 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

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

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4 and 5

ACTION:

With the reactor vessel water level at or below the top of the active irradiated fuel, manually initiate the ECCS to restore the water level. Depressurize the reactor vessel, as necessary for ECCS operation. Comply with the requirements of Specification 6.7.1.

2.1 SAFETY LIMITS

BASES

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.07. MCPR greater than 1.07 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the General Electric critical power correlations (Reference 1) are not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28×10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

SAFETY LIMITS

BASES

2.1.2 THERMAL POWER, High Pressure and High Flow

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

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB (Reference 1), which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using a GE critical power correlation. This correlation is valid over the range of conditions used in the tests of the data used to develop the correlation.

Details of the fuel cladding integrity safety limit calculation are given in Reference 2. Reference 2 provides the uncertainties used in the determination of the Safety Limit MCPR and of the nominal values of the parameters used in the Safety Limit MCPR statistical analysis.

1. "General Electric BWR Thermal Analysis Bases (GETAB) Data, Correlation and Design Application," NEDO-10958-A.
2. "General Electric Standard Application for Reactor Fuel, GESTAR-II," NEDE-24011-P-A (latest approved revision).

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POWER DISTRIBUTION LIMITS

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 result obtained from multiplying the applicable MAPLHGR values* by the smaller of either the flow dependent MAPLHGR factor ($MAPFAC_f$) of Figure 3.2.1-1 or the power dependent MAPLHGR factor ($MAPFAC_p$) of Figure 3.2.1-2.

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

ACTION:

If at any time during operation it is determined that an APLHGR is exceeding the result of the above multiplication, 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.

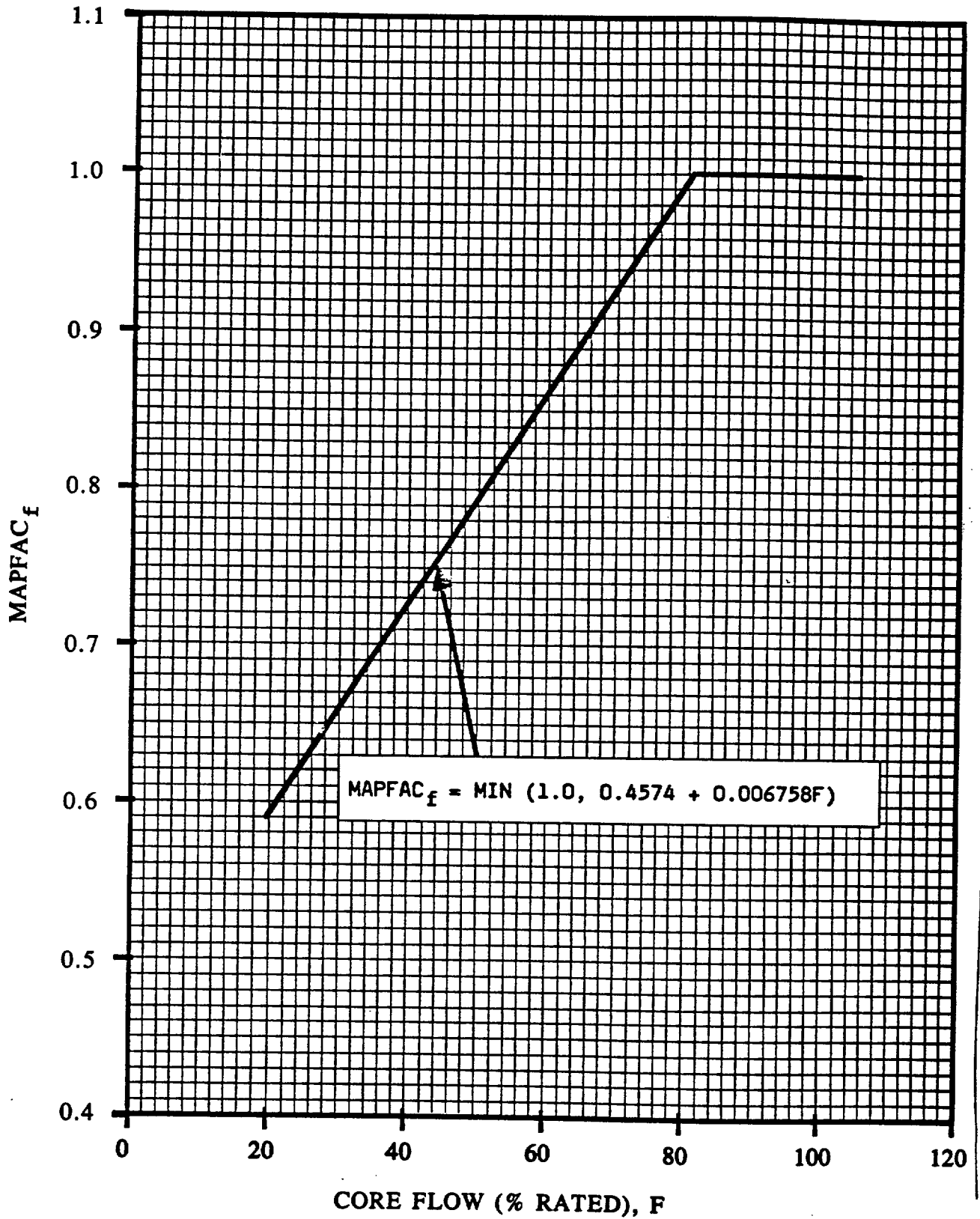
SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the above 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 in one hour, 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.

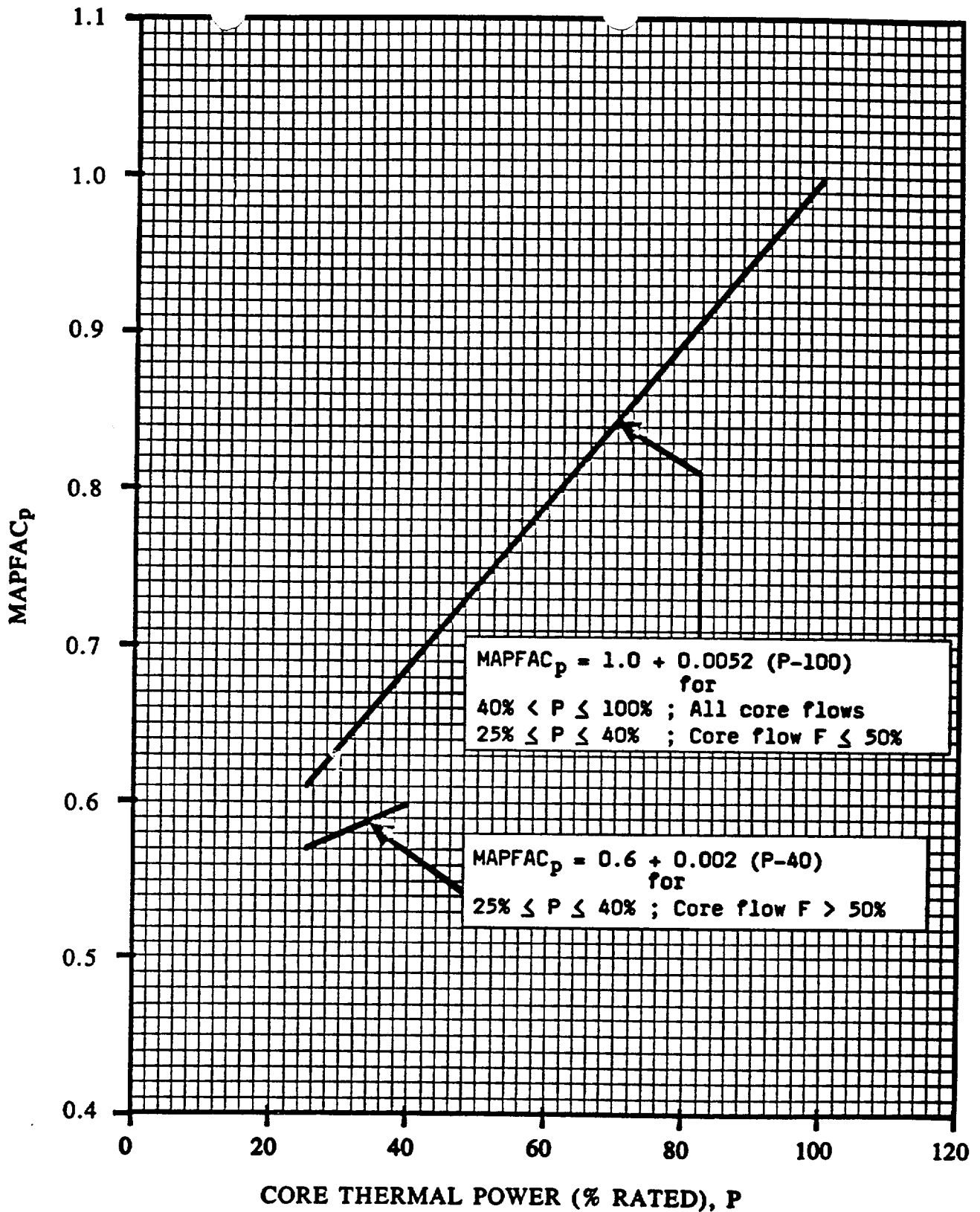
* These applicable MAPLHGR values are:

- 1) Those that have been approved for the respective fuel and lattice type as a function of the average planar exposure (as determined by the NRC approved methodology described in GESTAR-II)
or
- 2) When hand calculations are required, the MAPLHGR as a function of the average planar exposure for the most limiting lattice (excluding natural uranium) shown in the Figures 3.2.1-3, 3.2.1-4, 3.2.1-5, and 3.2.1.6 for the applicable type of fuel.



FLOW DEPENDENT MAPLHGR FACTOR
(MAPFAC_f)

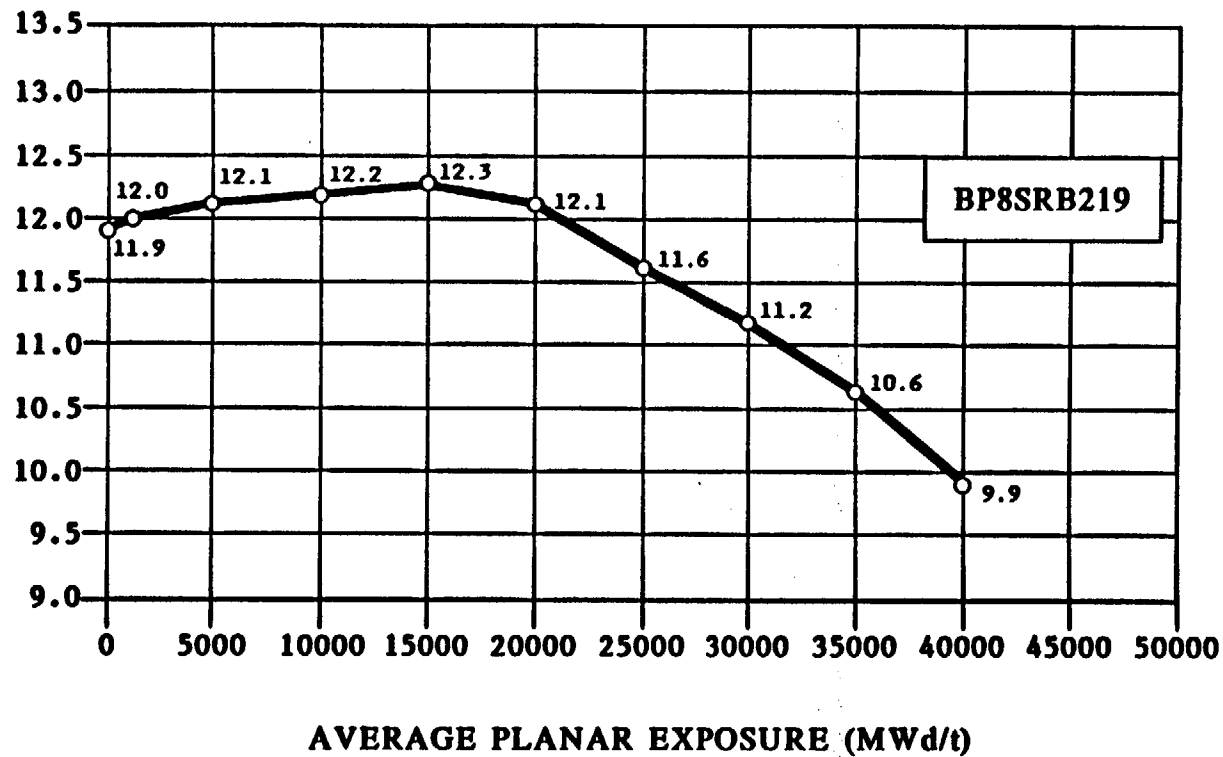
FIGURE 3.2.1-1



POWER DEPENDENT MAPLHGR FACTOR
(MAPFAC_p)

FIGURE 3.2.1-2

MAXIMUM AVERAGE PLANAR LINEAR
HEAT GENERATION RATE (kW/FT)

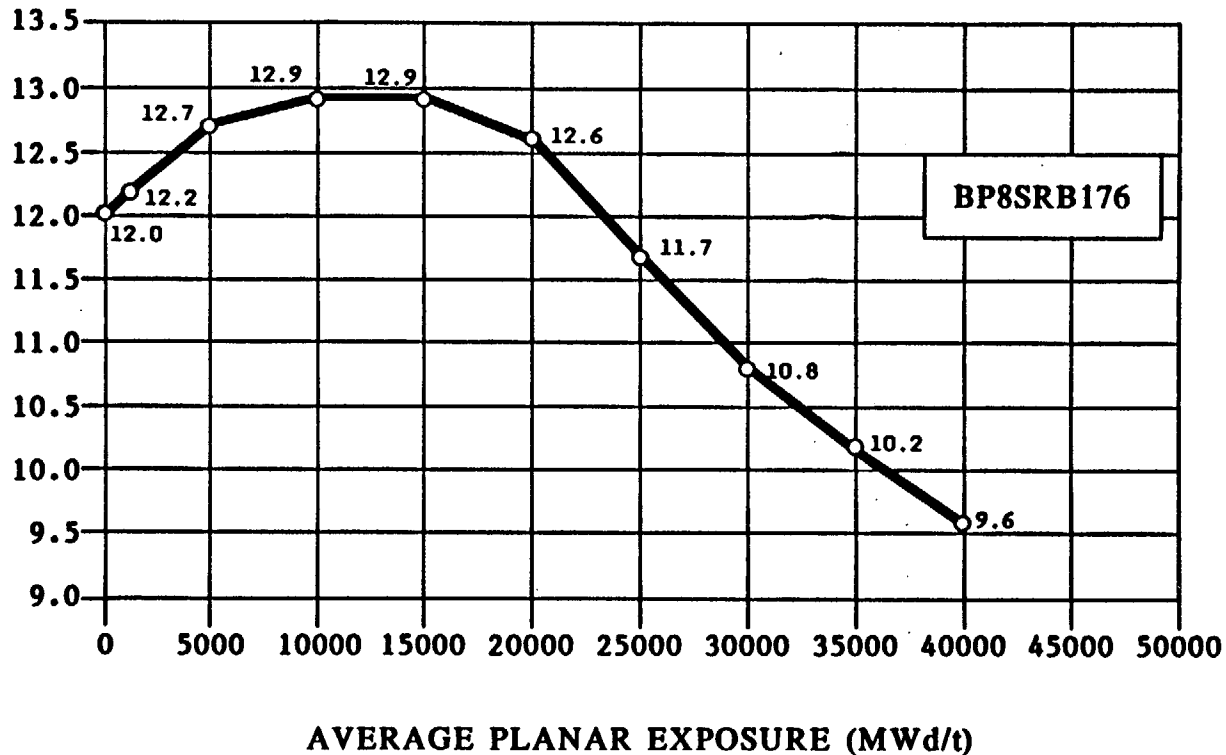


MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE

FUEL TYPE BP8SRB219

Figure 3.2.1-3

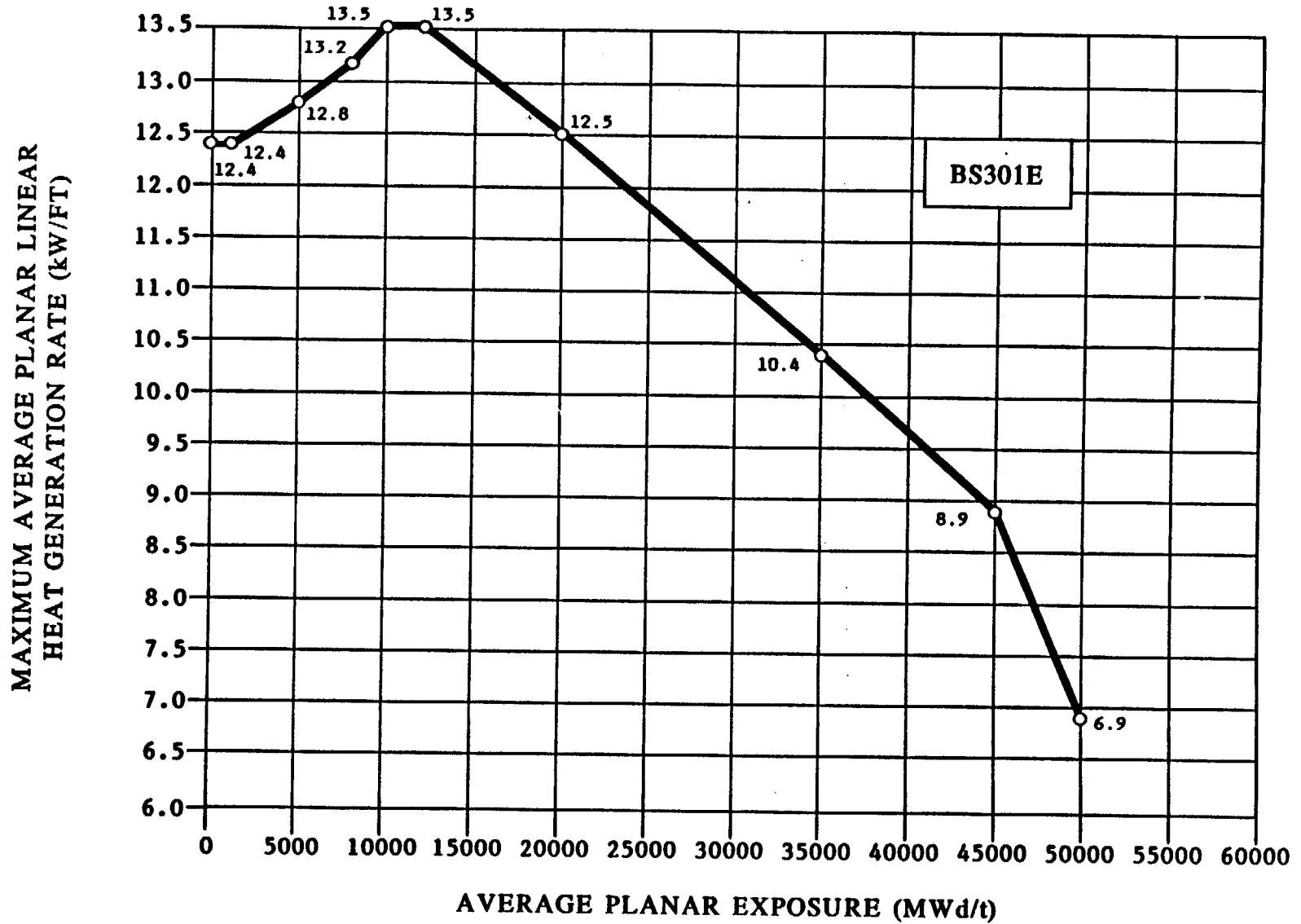
MAXIMUM AVERAGE PLANAR LINEAR
HEAT GENERATION RATE (kW/FT)



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

FUEL TYPE BP8SRB176

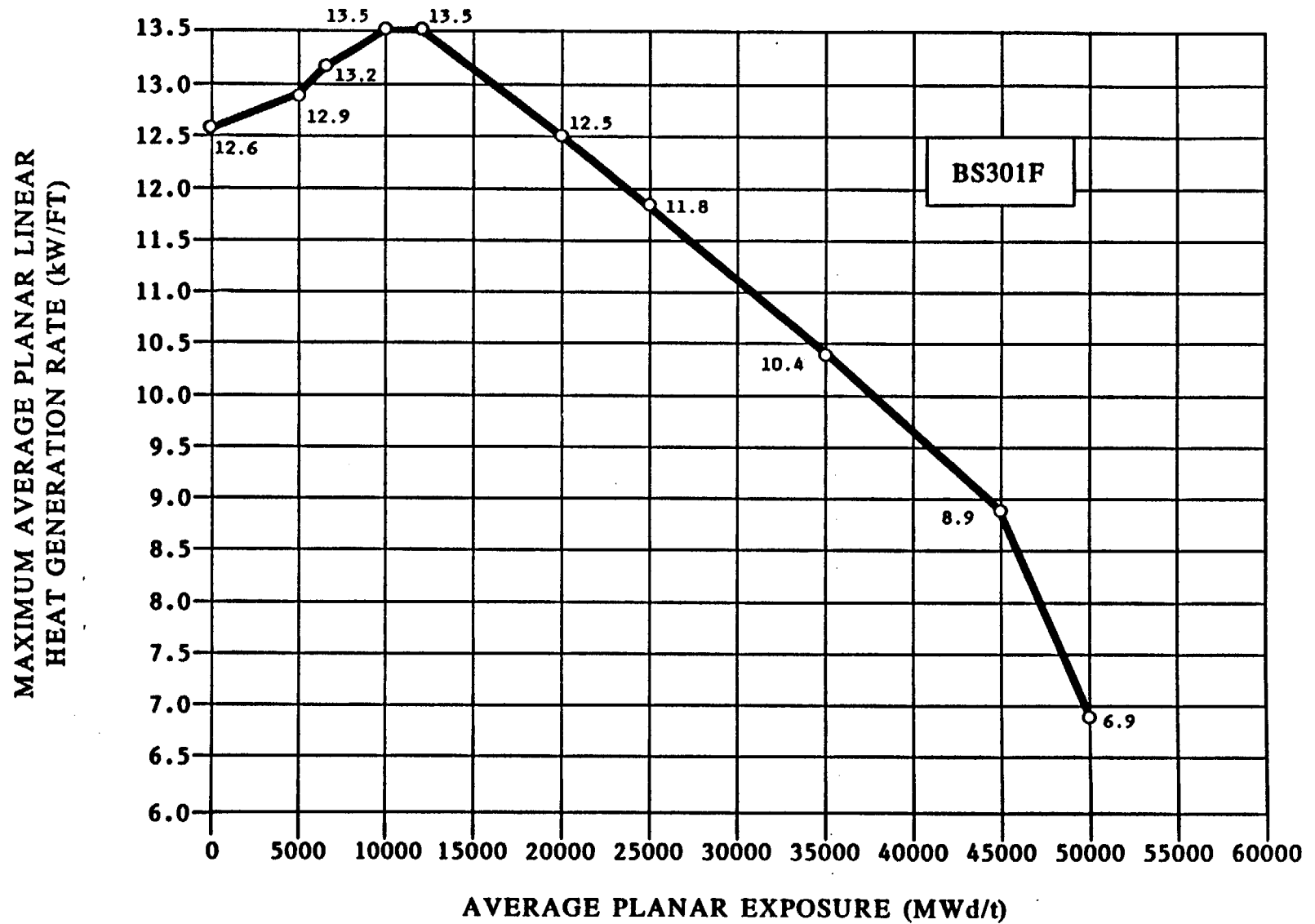
Figure 3.2.1-4



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE, GE8x8EB

FUEL TYPE BS301E

Figure 3.2.1-5



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE, GE8x8EB

FUEL TYPE BS301F

Figure 3.2.1-6

POWER DISTRIBUTION LIMITS

3/4.2.2 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than both $MCPR_f$ and $MCPR_p$ limits at indicated core flow, THERMAL POWER, ΔT^* and core average exposure compared to End of Cycle Exposure (EOCE)** as shown in Figures 3.2.2-1 and 3.2.2-2.

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 limit shown in Figures 3.2.2-1 and 3.2.2-2, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

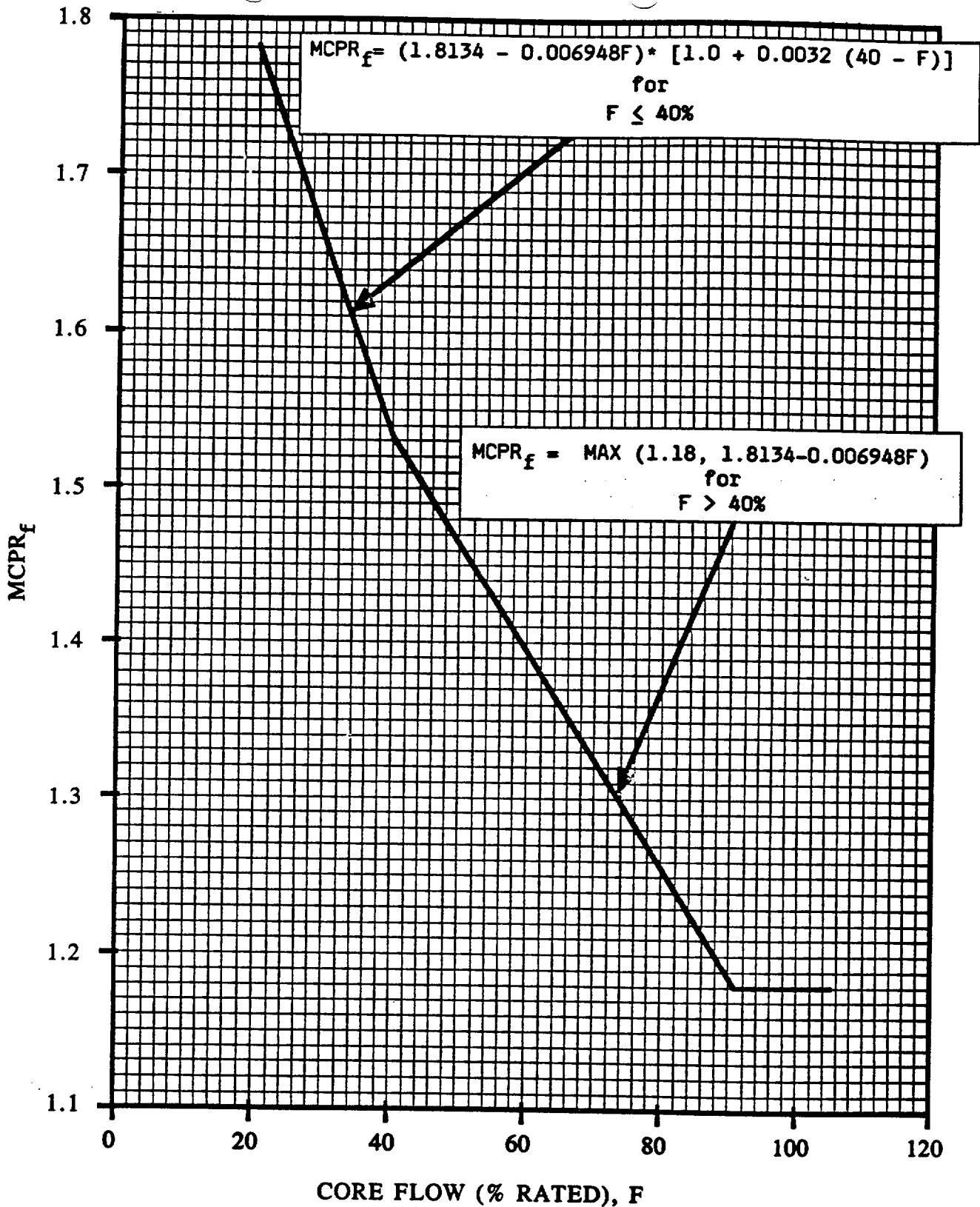
SURVEILLANCE REQUIREMENTS

4.2.2 MCPR shall be determined to be equal to or greater than the MCPR limit determined from Figures 3.2.2-1 and 3.2.2-2:

- 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 in one hour, 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.

*This ΔT refers to the planned reduction of rated feedwater temperature from nominal rated feedwater temperature (420°F), such as prolonged removal of feedwater heater(s) from service.

**End of Cycle Exposure (EOCE) is defined as 1) the core average exposures at which there is no longer sufficient reactivity to achieve RATED THERMAL POWER with rated core flow, all control rods withdrawn, all feedwater heaters in service and equilibrium Xenon, or 2) as specified by the fuel vendor.



FLOW DEPENDENT MCPR FACTOR
 (MCPR_f)

FIGURE 3.2.2-1

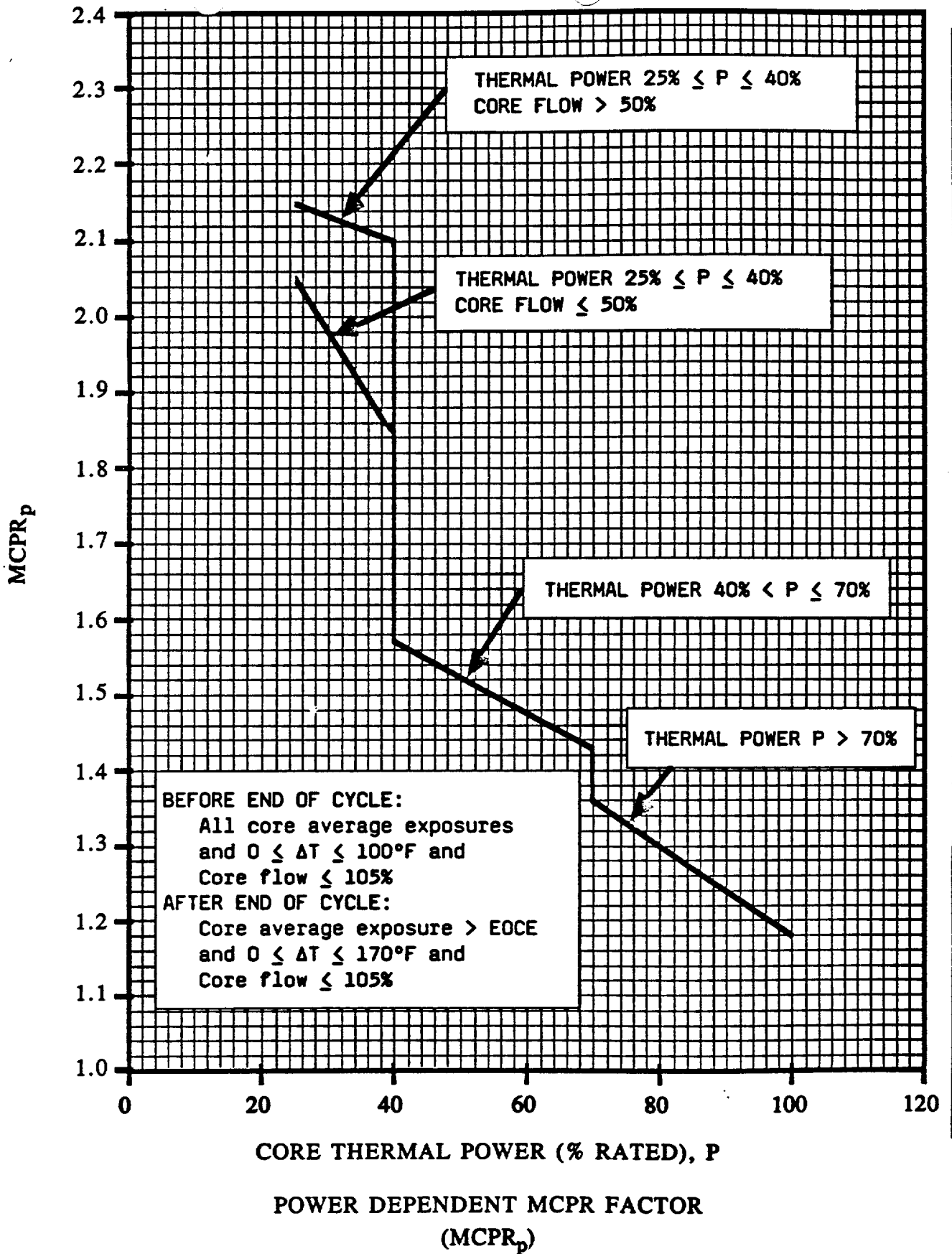


FIGURE 3.2.2-2

POWER DISTRIBUTION LIMIT

3/4.2.3 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.3 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed:

- a. 13.4 kw/ft for BP8x8R fuel.
- b. 14.4 kw/ft for GE8x8EB fuel.

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, 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

4.2.3 LHGR's shall be determined to be equal to or less than the limit:

- 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 in one hour, and
- c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.
- d. The provisions of Specification 4.0.4 are not applicable.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 SHUTDOWN MARGIN

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

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

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

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

3/4.1.2 REACTIVITY ANOMALIES

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

REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.3 CONTROL RODS

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

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

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

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

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than 1.07 during the limiting power transient analyzed in Chapter 15 of the USAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than 1.07. The occurrence of scram times longer than those specified should be viewed as an indication of a systematic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

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

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

3/4.2 POWER DISTRIBUTION LIMITS

BASES

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

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature (PCT) following the postulated design basis Loss-of-Coolant Accident (LOCA) will not exceed the limits specified in 10 CFR 50.46 and that the fuel design analysis limits specified in GESTAR-II (Reference 1) will not be exceeded.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The MAPLHGR limits of Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 are multiplied by the smaller of either the flow dependent MAPLHGR factor ($MAPFAC_f$) or the power dependent MAPLHGR factor ($MAPFAC_p$) corresponding to existing core flow and power state to assure the adherence to fuel mechanical design bases during the most limiting transient. $MAPFAC_f$'s are determined using the three-dimensional BWR simulator code to analyze slow flow runout transients. $MAPFAC_p$'s are generated using the same data base as the $M CPR_p$ to protect the core from plant transients other than core flow increases.

The Technical Specification MAPLHGR value is the most limiting composite of the fuel mechanical design analysis MAPLHGR and the ECCS MAPLHGR.

Fuel Mechanical Design Analysis: NRC approved methods (specified in Reference 1) are used to demonstrate that all fuel rods in a lattice, operating at the bounding power history, meet the fuel design limits specified in Reference 1. This bounding power history is used as the basis for the fuel design analysis MAPLHGR value.

LOCA Analysis: A LOCA analysis is performed in accordance with 10 CFR Part 50 Appendix K to demonstrate that the MAPLHGR values comply with the ECCS limits specified in 10 CFR 50.46. The analysis is performed for the most limiting break size, break location, and single failure combination for the plant.

POWER DISTRIBUTION LIMITS

BASES

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

Only the most limiting MAPLHGR values are shown in the Technical Specification figures for multiple lattice fuel. When hand calculations are required, these Technical Specification MAPLHGR figure values for that fuel type are used for all lattices in that bundle.

For some GE fuel bundle designs MAPLHGR depends only on bundle type and burnup. Other GE fuel bundles have MAPLHGRs that vary axially depending upon the specific combination of enriched uranium and gadolinia that comprises a fuel bundle cross section at a particular axial node. Each particular combination of enriched uranium and gadolinia, for these fuel bundle types, is called a lattice type by GE. These particular fuel bundle types have MAPLHGRs that vary by lattice type (axially) as well as with fuel burnup.

Approved MAPLHGR values (limiting values of APLHGR) as a function of fuel and lattice types, and as a function of the average planar exposure are provided in Technical Specification Figures 3.2.1-3 and 3.2.1-6.

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POWER DISTRIBUTION LIMITS

BASES

3/4.2.2 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.2 are derived from the established fuel cladding integrity Safety Limit MCPR of 1.07 and an analysis of the limiting 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 are documented in the USAR and Reference 1. The limiting transient yields the largest delta CPR. When added to the Safety Limit MCPR, the required operating limit MCPR of Specification 3.2.2 is obtained. The power-flow map of Figure B 3/4 2.2-1 defines the analytical basis for generation of the MCPR operating limits.

The evaluation of a given transient begins with the system initial parameters shown in USAR Chapter 15 and/or Reference 1, and Cleveland Electric's November 28 and December 29, 1988 submittals that are input to a GE-core dynamic behavior transient computer program. The codes used to evaluate these events are described in Reference 1.

The purpose of the $MCPR_f$ and $MCPR_p$ is to define operating limits at other than rated core flow and power conditions. At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_f$ and $MCPR_p$ at the existing core flow and power state. The $MCPR_f$ s are established to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

Figure 3.2.2-2 also reflects the required MCPR values resulting from the analysis performed to justify operation with the feedwater temperature ranging from 420°F to 320°F at 100% RATED THERMAL POWER steady state conditions, and also beyond the end of cycle with the feedwater temperature ranging from 420°F and 250°F.

The $MCPR_f$ s were calculated such that for the maximum core flow rate and the corresponding THERMAL POWER along a conservative steep generic power flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along this conservative steep power flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined as $MCPR_f$.

POWER DISTRIBUTION LIMITS

BASES

MINIMUM CRITICAL POWER RATIO (Continued)

The MCPR_ps are established to protect the core from plant transients other than core flow increases, including the localized event such as rod withdrawal error. The MCPR_ps were calculated based upon the most limiting transient at the given core power level. For core power less than or equal to 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 MRPR_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.

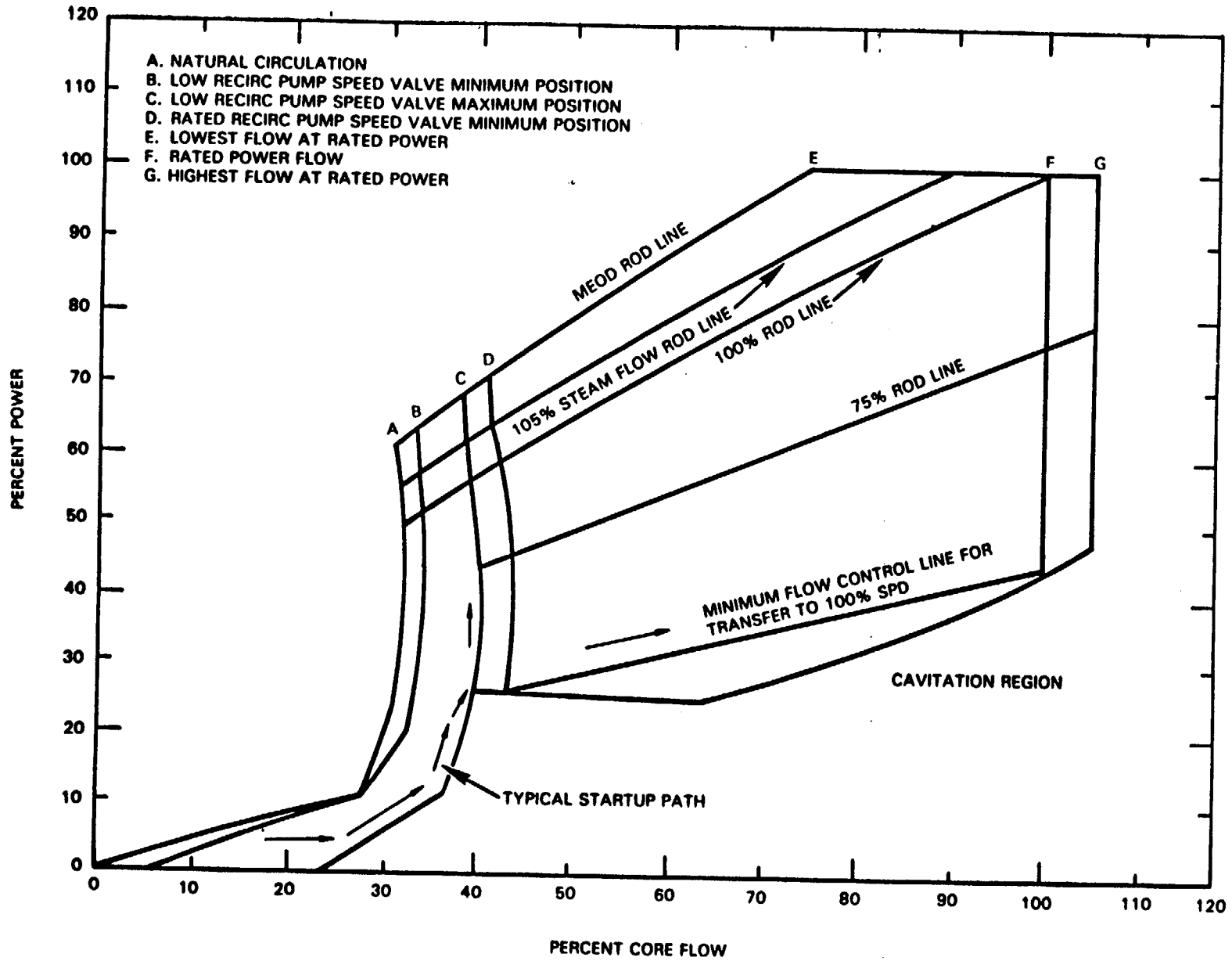
At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

3/4.2.3 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.

References:

1. GESTAR II, General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A, (latest approved revision).



**POWER-FLOW OPERATING MAP
BASES FIGURE B 3/4 2.2-1**



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 20 TO FACILITY OPERATING LICENSE NO. NPF-58

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-440

1.0 INTRODUCTION

By letters dated November 28, 1988 and December 29, 1988, the Cleveland Electric Illuminating Company, the licensee for the Perry Unit 1 Nuclear Generating Station, proposed to amend the Technical Specifications for the Cycle 2 reload and operation (Refs. 1, 2 and 7). The reload includes 272 new assemblies of GE manufacture. The reload design has no unusual features. The proposed Technical Specification changes are related to the Minimum Critical Power Ratio (MCPR), the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR), and updating the bases and references associated with certain cycle-dependent limits. The December 29, 1988 submittal also proposed TS changes to specify values for flow-dependent MAPLHGR factor (MAPFAC_f) and MCPR_f for off-rated conditions of operation. The new fuel is of slightly increased enrichment designed for extended burnup.

2.0 EVALUATION

2.1 Reload Description

The licensee requests to be allowed to use GE fuel types BS301E and BS301F which have slightly higher enrichment than the present fuel types and will allow higher burnup. The core loading is the conventional new assembly scatter pattern, with low reactivity (old) assemblies located on the periphery. The new assembly types are not described in GESTAR II (Ref. 3).

2.2 Fuel Design

The new fuel for Cycle 2 is the GE fuel designated BS301E and BS301F. This fuel is in the same class with approved designs but not for the enrichments used here. The specific description of this fuel is presented in Reference 4. This fuel description is acceptable.

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

2.3 Nuclear Design

The nuclear design for Cycle 2 has been performed by GE using the approved GESTAR II methodology (Ref. 3). The results of these analyses are given in the GE reload report (Ref. 2) in the GESTAR II format. The results are within the usual reload range. The shutdown margin is 2.9% delta-k/k at beginning of cycle (BOC) with the strongest rod out and 1.2% delta-k/k at the exposure with the minimum shutdown margin. Both meet the 0.38% delta-k/k margin required by the Technical Specifications. The standby liquid control system also meets the shutdown requirements with a shutdown margin of 4.0% delta-k/k. Because these and other nuclear characteristics of the reload have been computed with previously-approved methods (outlined in GESTAR II) and their values are within the allowed range, the nuclear design is acceptable.

2.4 Thermal-Hydraulic Design

The thermal-hydraulic design for Cycle 2 has been calculated using the approved methods described in GESTAR II. The results are given in the standard GESTAR II format in the reload report (Ref. 2). The parameters and initial values used for the calculations are those approved in GESTAR II for the BWR/6 class of reactors. The GEMINI set of methods (Refs. 5 and 6) have been approved for the relevant transient analyses. The Technical Specification values for scram speed, which are conservative, were used.

The operating limits of the MCPR values are determined by the limiting transient among the following: local rod withdrawal error, feedwater controller failure, load rejection without bypass and loss of 100°F feedwater heating. The analyses of these events for Cycle 2 used approved methods. The loss of 100°F feedwater heating transient is limiting. The delta-CPR results of these analyses are reflected in the requested Technical Specification changes. The MCPR for Cycle 2 has been increased from 1.06 to 1.07 to account for Cycle 2 uncertainties. The results are within expected ranges and, hence, they are acceptable.

For the Perry Unit 1, Cycle 2, no cycle-specific stability analysis is required because the Technical Specifications have standard NRC-approved provisions for incore neutron detector monitoring of thermal-hydraulic stability according to the recommendations of the General Electric SIL-380. Nevertheless, effective December 1, 1988, the licensee instituted procedures for the instance of loss of one or both recirculation pumps to prevent the reactor from entering an unstable mode of operation. This is responsive to Bulletin 88-07 and thus is acceptable.

2.5 Transient and Accident Analyses

The accident and transient analysis methods used for Cycle 2 are described in GESTAR II. The GEMINI set of codes was used. The MCPR operating limit was determined from the loss of 100°F feedwater heating transient, $\Delta\text{-CPR} = 0.11$ added to the MCPR of 1.07 for a cycle operating MCPR limit of 1.18. The core-wide transient analyses methodologies have been approved and the results fall within expected ranges and are acceptable.

The mislocated assembly event is not analyzed, because the NRC approved the non-applicability of loading errors to BWR/6 plants as documented in Ref. 3.

The limiting overpressurization event analysis, i.e., main isolation valve closure with flux scram, was performed using the GEMINI methods (Ref. 5 and 6) at 102% of power level to account for the power level uncertainties specified in Regulatory Guide 1.49. The results show that the peak steam dome and vessel pressures of 1,235 and 1,266 psig are under 1375 psig, i.e., the required limit. The methodology and the results of the overpressurization event analysis are acceptable (Ref. 2).

LOCA analyses, using approved (SAFE/REFLOOD) methods and parameter values were performed to provide MAPLHGR values versus average planar exposure, peak clad temperature and oxidation fraction for both new fuel type assemblies for Cycle 2, i.e., BS301E and BS301F. The results show compliance with 10 CFR 50.46 and the LHGR limits as listed in the Technical Specifications and, therefore, are acceptable.

2.6 Selected Margin Improvement and Operating Flexibility Options

The licensee has included in its reload analyses several assumptions regarding equipment inoperability which will allow operating flexibility. Equipment not credited in the analyses include recirculation pump trip, rod withdrawal limiter, and the thermal power monitor. In addition, the licensee has considered in its determination of operating limits and technical specifications the effects of feedwater heaters being out of service, single loop operation, maximum extended operating domain conditions, and increased core flow. These options have been approved on a generic basis and have been demonstrated as applicable to Perry in its reload submittal. The staff will evaluate technical specification changes related to single loop operation when they are submitted at a later date.

2.7 Evaluation of Changes to MCPR_f and MAPFAC_f Values

During off-rated power-flow operation, MCPR_f values of MCPR are required to ensure that the established safety limit value is met during inadvertent core flow increase. The MCPR_f were calculated as a function of flow. For each value of the flow the limiting bundle's relative power is adjusted until the MCPR is slightly above the safety limit MCPR. In this manner a power-flow line is defined to assure that the safety limit will be met.

The revision of these curves for Cycle 2 was necessitated by the non-conservative behavior at low flows of the GEXL-Plus critical power correlation used in the analysis of Cycle 2. The staff requires additional conservatism for flows below 40% of the rated flow (GESTAR, Amendment 15, Ref. 8). This conservatism is given in the form of a flow-dependent factor, $MAPFAC_f$, and the value of the flow dependent $MCPR_f$. These quantities are specified in Figures 3.2.1-1 and 3.2.2-1 which are part of the proposed Technical Specifications. These curves have been extended to 20% of rated core flow to cover potential core flow shortfall.

As pointed out above, the calculational methodology was based on GEXL-Plus and an NRC-approved code. The calculational results assure that the safety limit $MCPR$ is met, therefore, the proposed Technical Specification changes are acceptable.

2.8 Proposed Technical Specification Changes

The following Technical Specifications (and corresponding bases) are proposed to be changed:

1. 2.1.2, Thermal Power, High Pressure and High Flow

The $MCPR$ has been increased in the Technical Specification and the bases. Tables B2.1.2-1 and B2.1.2-2 are eliminated. These changes are acceptable as discussed in the evaluation.

2. 3/4.2.1, Average Planar Linear Heat Generation Rate

Modification of the $MAPLHGR$ versus average exposure for each fuel type in Cycle 2. Figures 3.2.1-1, 3.2.1-2, 3.2.1-4 and 3.2.1-5 were renumbered, Figure 3.2.1-3 was deleted. Figures 3.2.1-5 and 3.2.1-6 were added. These changes have been discussed above and are acceptable. Also, as discussed above, the new Figure 3.2.1-1 was modified to extend the curve to the 20% of rated core flow line. This change is acceptable.

3. 3/4.2.2, Minimum Critical Power Ratio

Changes in Figure 3.2.2-2 as discussed above reflect the revised $MCPR$. The changes are acceptable.

4. 3/4.2.3, Linear Heat Generation Rate

Changes to reflect explicitly the linear heat generation limits for all assemblies present in Cycle 2. This change is acceptable as discussed in the evaluation.

5. Figure 3.2.2-1 has been revised for the flow dependent minimum critical power ratio, $MCPR_f$, to correct the extrapolated value for Cycle 2 operation. As discussed above, this change is acceptable.

3.0 SUMMARY AND CONCLUSIONS

We have reviewed the information submitted for the Cycle 2 operation of the Perry Unit 1 plant. Based on this review, we conclude that the fuel design, the nuclear design, the thermal-hydraulic design and the accident and transient analyses are acceptable. The proposed Technical Specifications submitted for the Cycle 2 reload represent the necessary modifications for this cycle.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment. Comments were received dated February 22, 1988 with respect to the proposed issuance of this amendment and are addressed below.

5.0 PUBLIC COMMENTS RECEIVED

On February 22, 1989 Susan L. Hiatt, representing Ohio Citizens for Responsible Energy (OCRE), submitted comments with regard to the licensees' license amendment application dated November 28, 1988 as amended December 29, 1988. Notice of Consideration of Issuance had been published in the Federal Register on February 1, 1989. Ms. Hiatt stated that the amendment request is deficient in that a stability analysis had not been conducted by the licensees. She further stated that the licensees should be required to conduct a stability analysis for the second operating cycle demonstrating compliance with GDC-10 and -12 as a condition of restart. Ms. Hiatt identified her concerns as being related to the La Salle Unit 2 power oscillation event of March 9, 1988. The staff has evaluated Ms. Hiatt's comments and provides the following discussion.

Following the March 9, 1988 power oscillation event at La Salle Unit 2, the staff issued NRC Bulletin 88-07 - "Power Oscillations in Boiling Water Reactors (BWR's)" and Supplement 1 to that bulletin on June 22, 1988 and December 30, 1988, respectively. The bulletin states a modified staff position wherein stability analyses are no longer acceptable for demonstrating that a BWR core is stable. Instead, the staff requested that explicit modifications to operating procedures be implemented by licensees in order to ensure that power

oscillations are avoided or promptly detected and suppressed. The General Electric (GE) SIL-380 guidance regarding operating procedures was modified by the BWR Owners Group. The changes were further modified and endorsed by Supplement 1 to NRC Bulletin 88-07. By letter dated February 15, 1989, the licensees confirmed that actions requested in NRC Bulletin 88-07 Supplement 1 have been completed and implemented. Therefore, the staff has determined that the licensees have taken appropriate measures to avoid, detect and suppress power oscillations for Perry Unit 1 in accordance with NRC Bulletin 88-07 and Supplement 1. Further, the staff maintains that stability analyses are neither necessary nor sufficient for demonstrating that a BWR core is stable.

6.0 CONCLUSION

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

6.0 REFERENCES

1. Letter from A. Kaplan, Cleveland Electric Illuminating Company, to USNRC, dated November 18, 1988.
2. 23A5948 GE Report, "Supplemental Reload Licensing Submittal for Perry Nuclear Power Station, Unit 1, Reload 1, Cycle 2," dated November 1988.
3. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel, GESTAR II," as amended, dated May 1986 and NEDE-24011PA9-US dated September 1988.
4. 23A5948A, Rev. 0 Supplement 1, "Supplemental Reload Licensing Submittal for Perry Nuclear Power Plant Unit 1, Reload 1, Cycle 2," GE Report dated October 1988.
5. Letter from J. S. Charnley, General Electric, to M. W. Hodges, NRC, "GEMINI ODYN Adjustment Factors for BWR/6," dated July 6, 1987.
6. Letter from Ashok C. Thadani to J. S. Charnley, General Electric, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, GE Generic Licensing Reload Report, of Amendment 15," May 5, 1988.

7. Letter from A. Kaplan, Cleveland Electric Illuminating Company, to USNRC, "Technical Specification Change Request - Reload Submittal," dated December 29, 1988.
8. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel, GESTAR-II" Amendment 15, dated May 5, 1988.

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