ENCLOSURE I

CORE OPERATING LIMITS REPORT SUBMITTAL
89-03

TECHNICAL SPECIFICATION REVISION SUMMARY

Subject (Section)	Revision
INDEX	Add new definition item 1.9 "Core Operating Limits Report" (COLR)
List of Figures (1.0)	Delete reference to the MAPLHGR figures
Definitions (1.0)	Add new definition for Item 1.9 "Core Operating Limits Report (COLR)"
Average Planar Linear Heat Generation Rate (3/4.2.1)	Transfer MAPLHGR Figures 3.2.1-1, 2, 3, 4, 5, 6, 7, 8 to the COLR
Minimum Critical Power Ratio (3/4.2.3)	Revise to include reference to the COLR for MCPR Limits and transfer Figures 3.2.3-1 and 2
Linear Heat Generation Rate (3/4.2.4)	Revise to include reference to the COLR for the LHGR Limits
Reactor Protection System Response Times (3/4.3.1)	Revise to reference the thermal time constant to the COLR
Control Rod Block Instrument Setpoints (3/4.3.6)	Revise to reference the single and two loop recirculation loop flow biased setpoints from Table 3.3.6-2 item 2.a. to the COLR
Design Features - Fuel Assemblies (5.3)	Revise description to allow fuel designs approved per NEDE-24011-P-A-US to be used.
Administrative Controls (6.9.3)	Include requirement for the Core Operating Limits Report submittal.

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DELETE

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

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.

INSERT

CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current reload cycle. These cycle-specific core operating limits shall be determine for each reload cycle in accordance with Specifications 6.9.3.1, 6.9.3.2, 6.9.3.3 and 6.9.3.4. Pla operation within these core operating limits is addressed in individual specifications.

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 the GEXL 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 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 Specification 3.6.4.

b. All drywell equipment hatches are closed and sealed.

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

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits ~~shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, 3.2.1-7 and 3.2.1-8.* The limits of Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, 3.2.1-7 and 3.2.1-8~~ shall be reduced to a value of 0.84 times the two recirculation loop operation limit when in single loop operation. specified in the COLR.

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

ACTION:

specified in the COLR.

With an APLHGR exceeding the limits ~~of Figure 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, 3.2.1-7 or 3.2.1-8~~, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined ~~from Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, 3.2.1-7 and 3.2.1-8~~: specified in the COLR.

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.

~~*The limits on Figures 3.2.1-7 and 3.2.1-8 are to be used only for manual calculations.~~

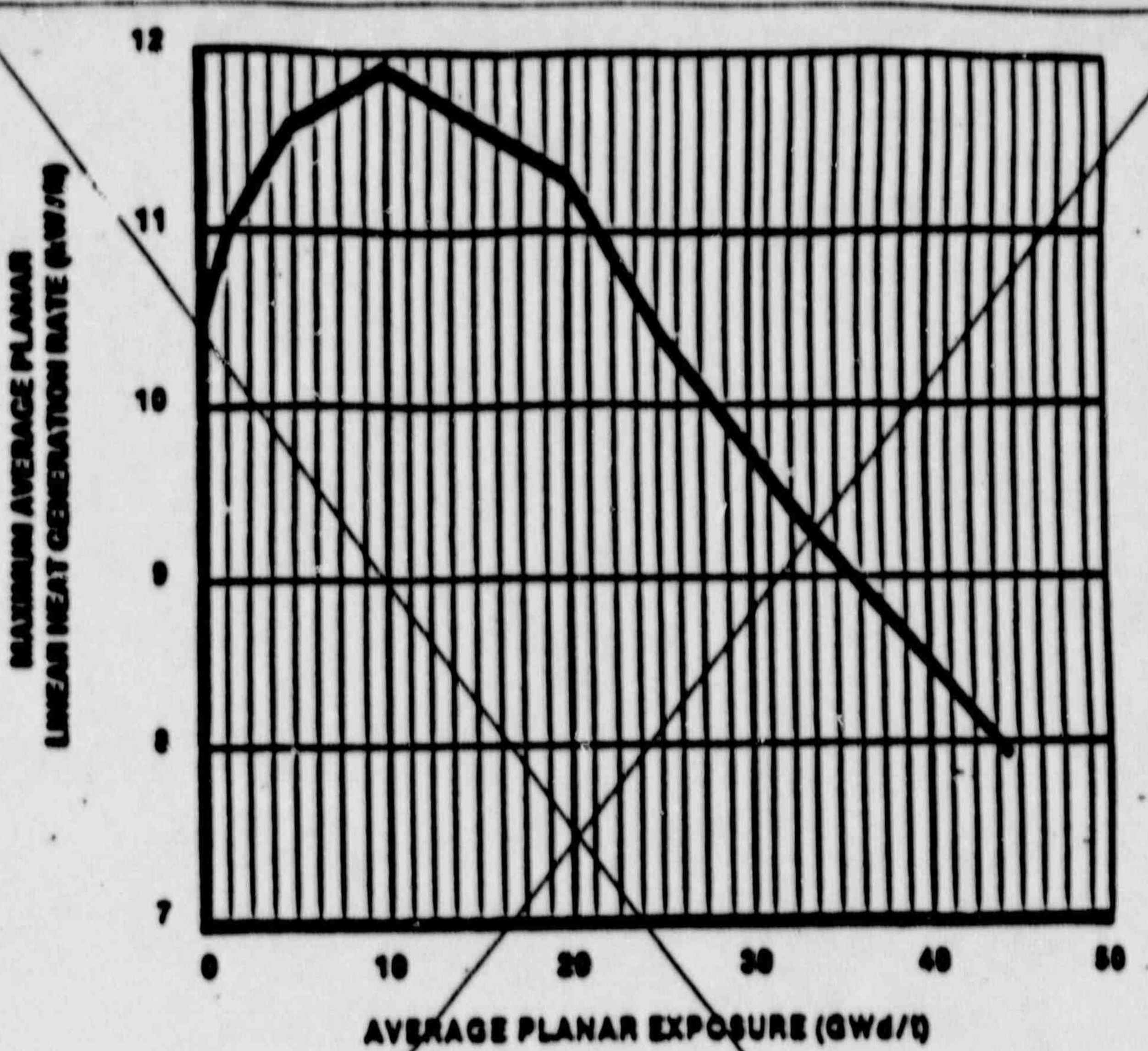

FIGURE 3.2.1-1
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE BP8SRB094

TRANSFER TO COLR

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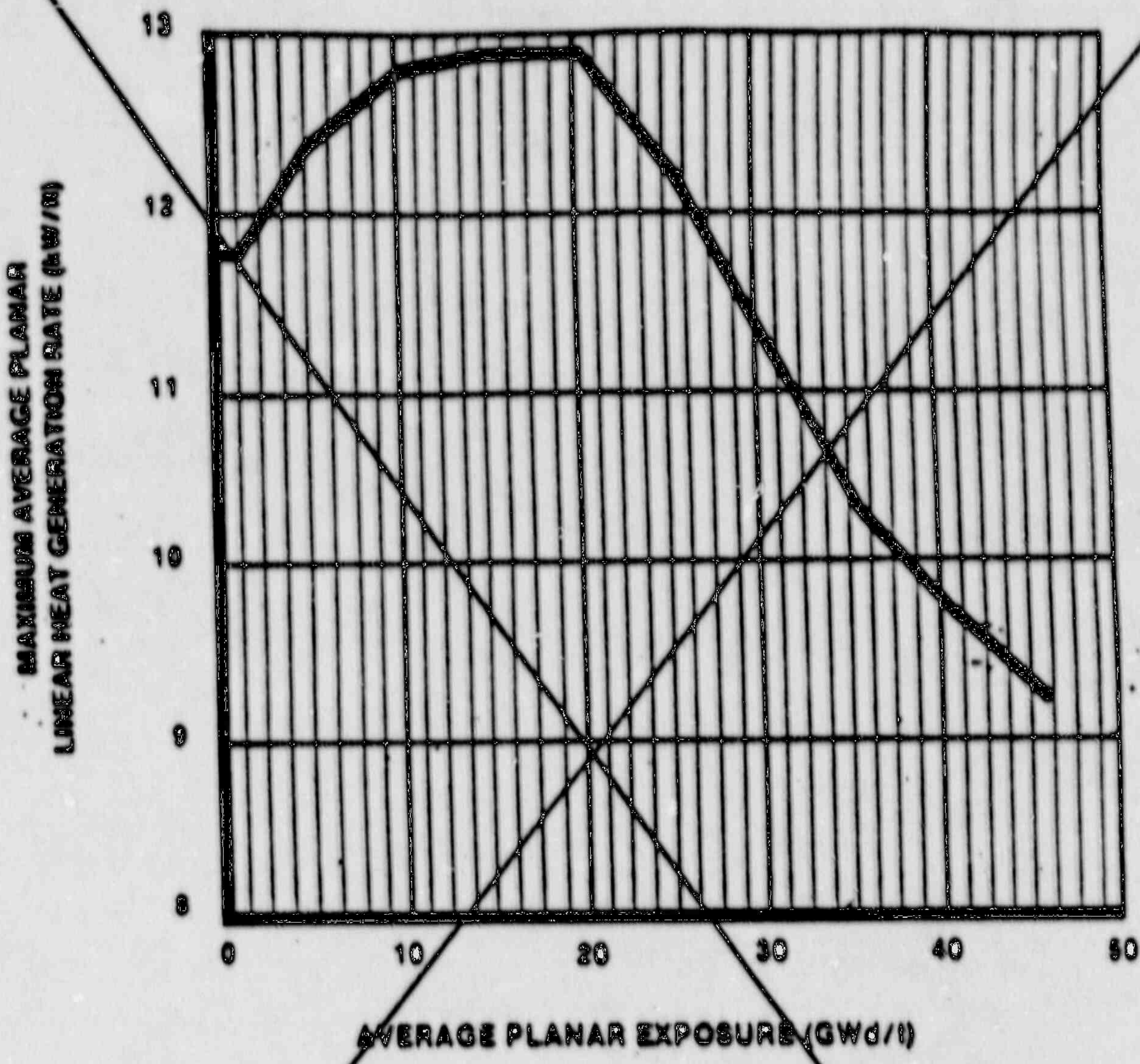

FIGURE 3.2.1-2
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE BP8SRB163

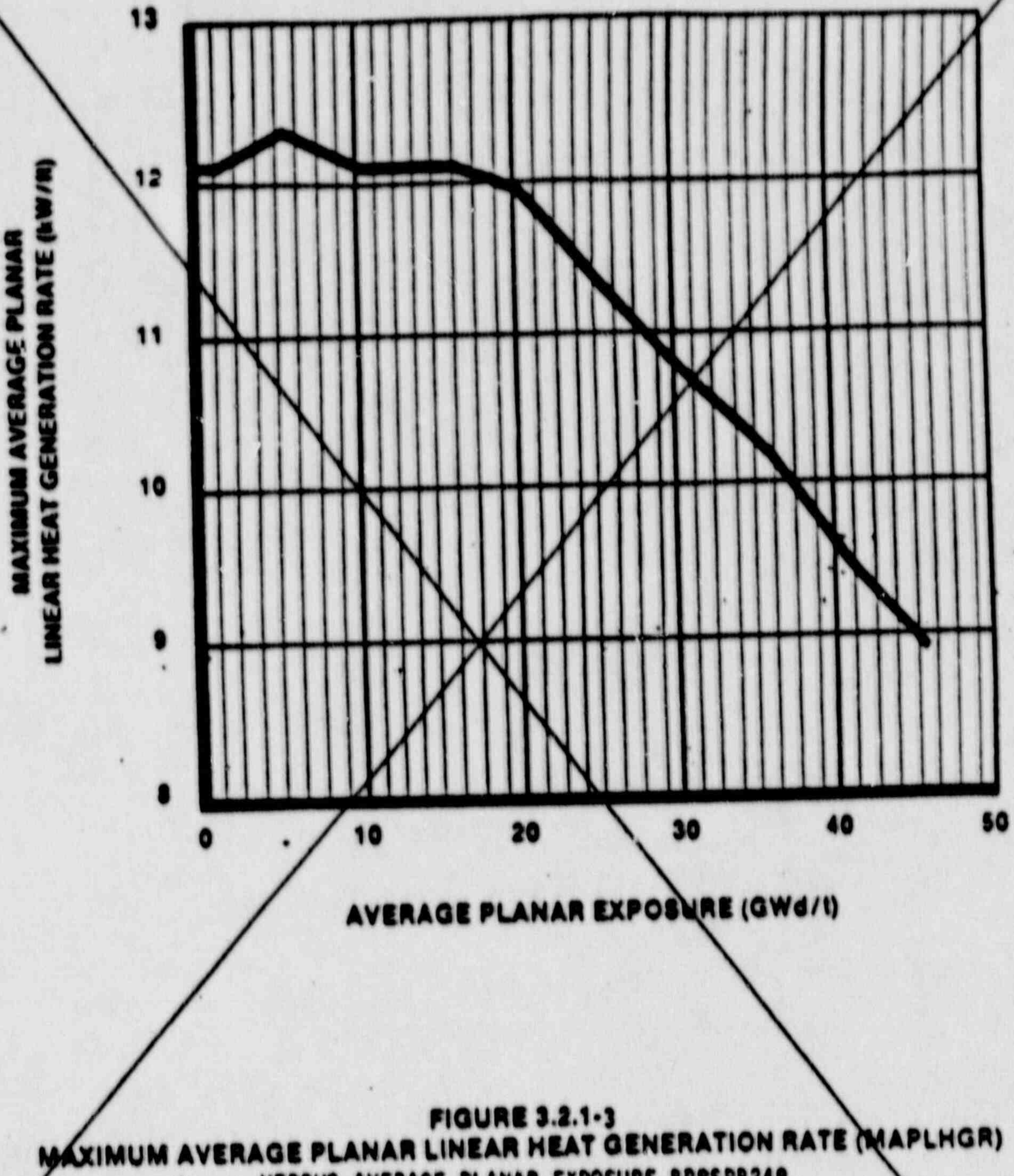

FIGURE 3.2.1-3
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE RP8SRB248

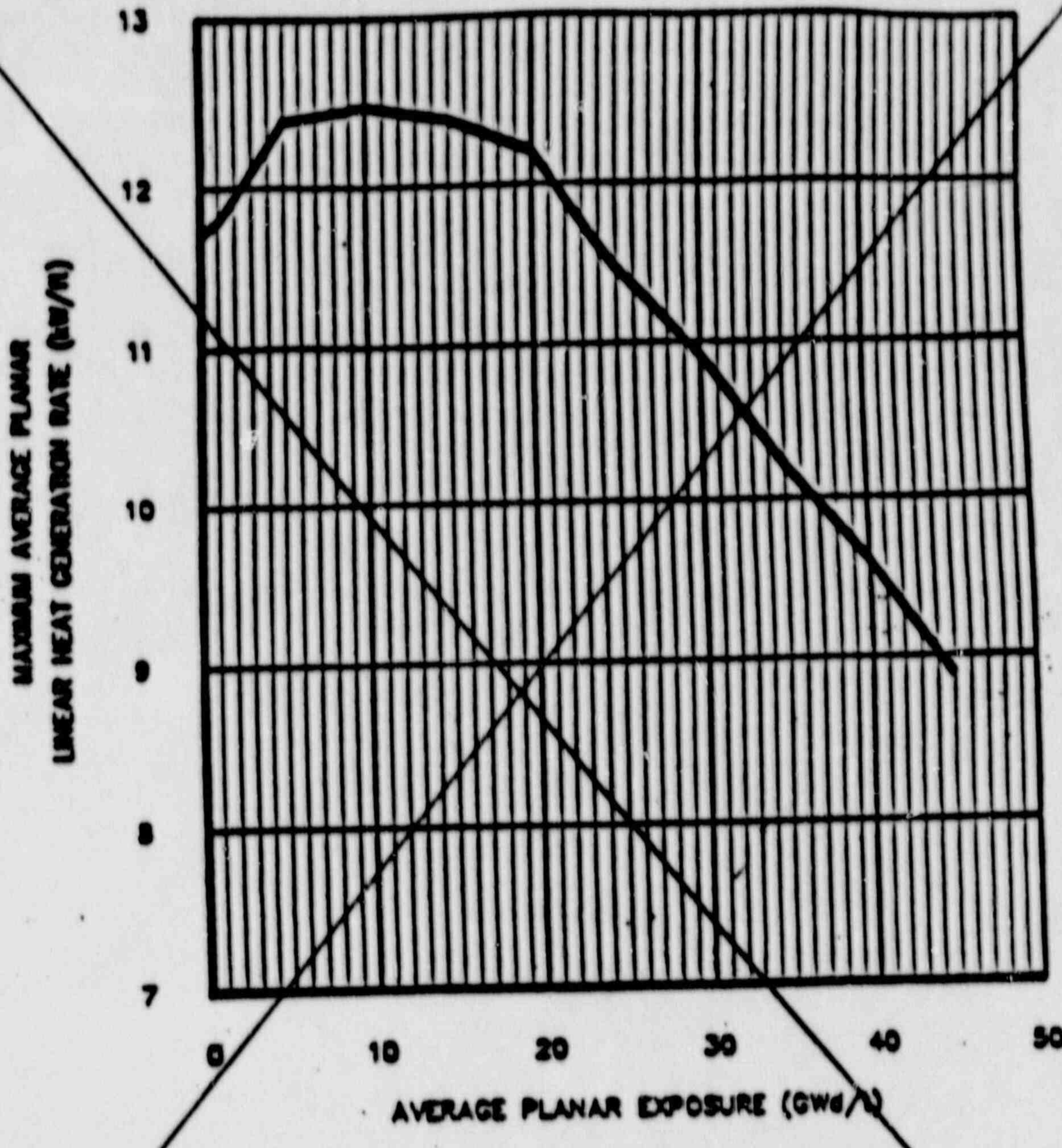

FIGURE 3.2.1-4

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

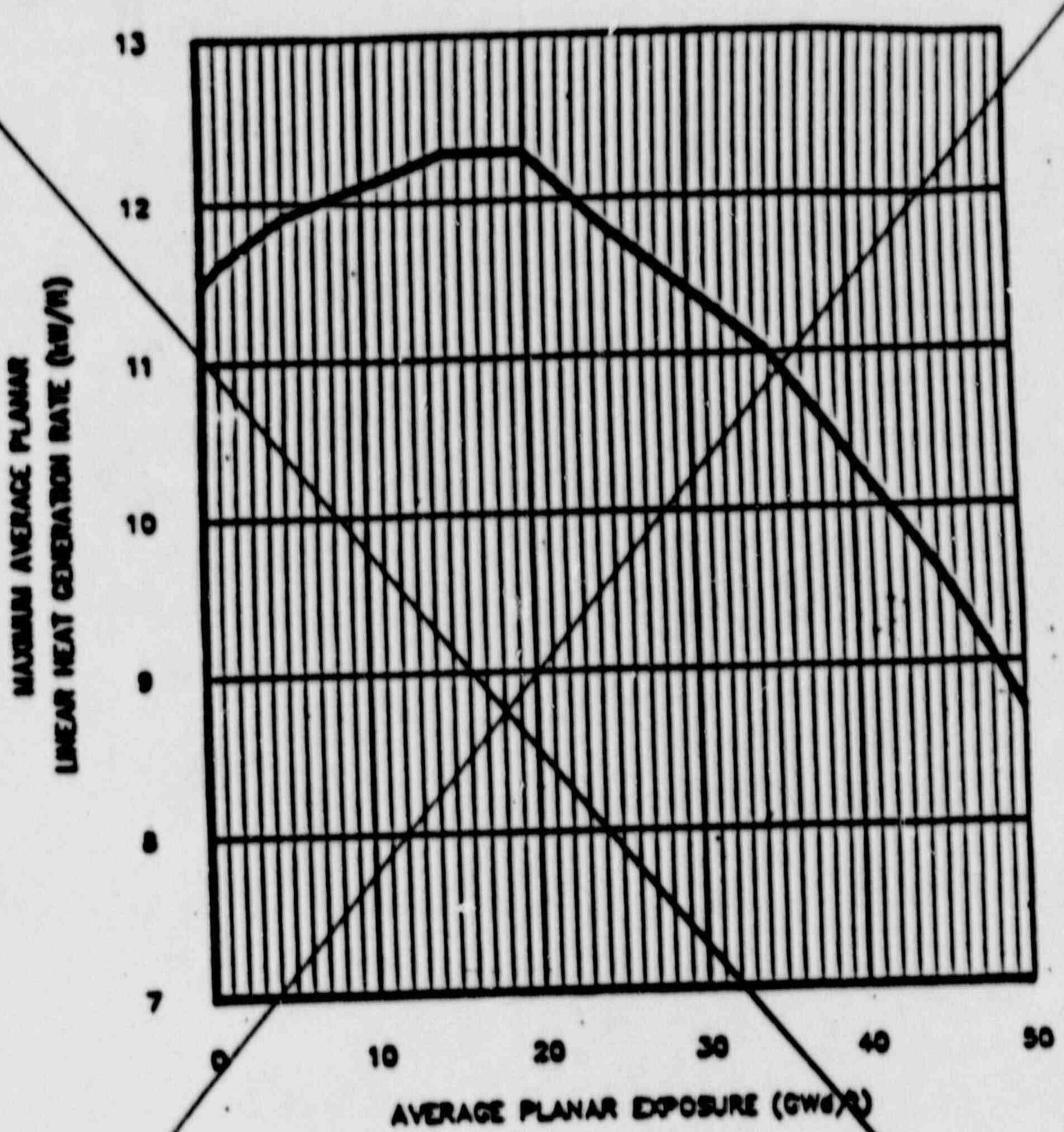

FIGURE 3.2.1-5

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

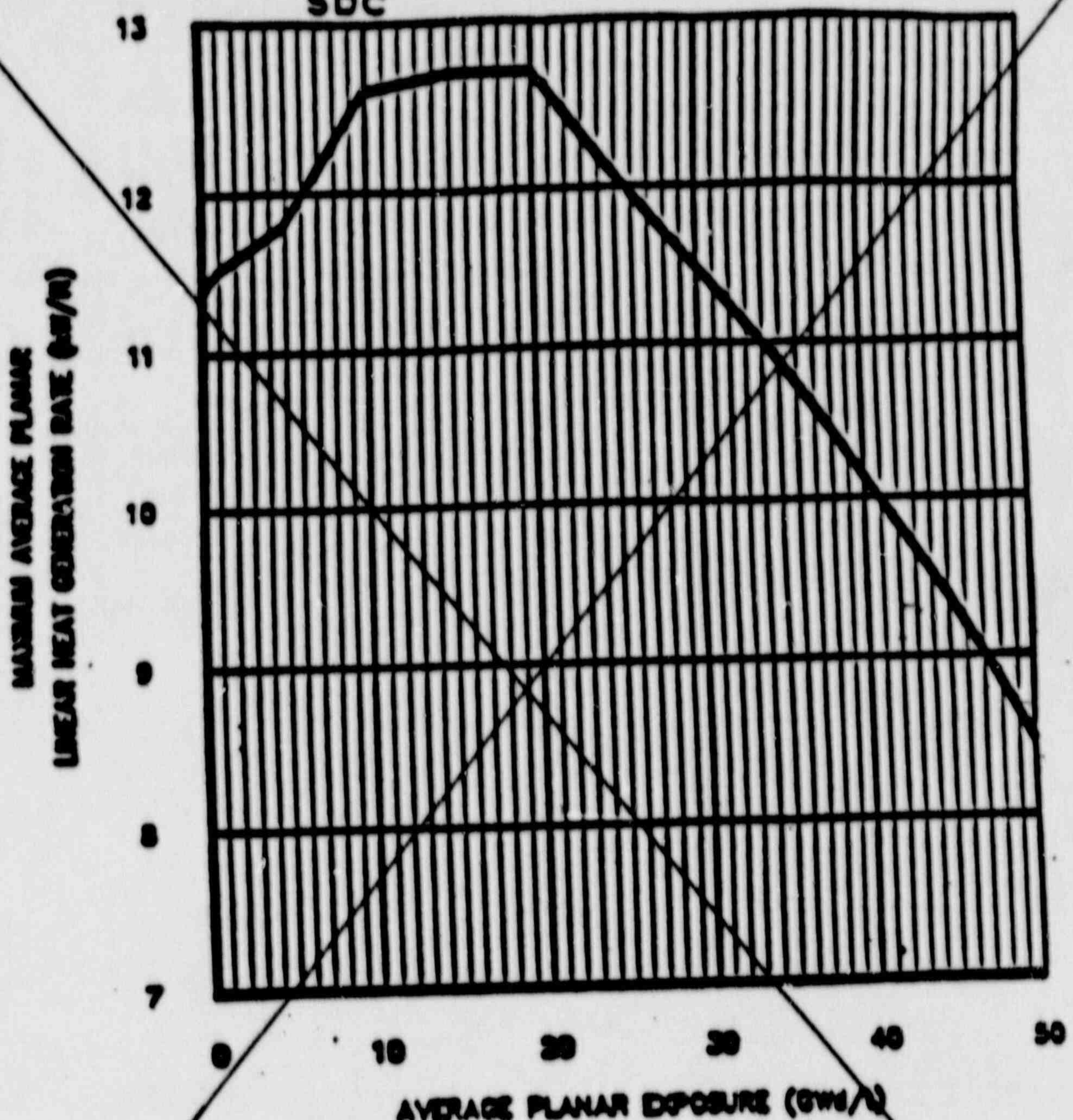

FIGURE 3.2.1-6

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE - 8P8SRB308

TRANSFER TO COLR

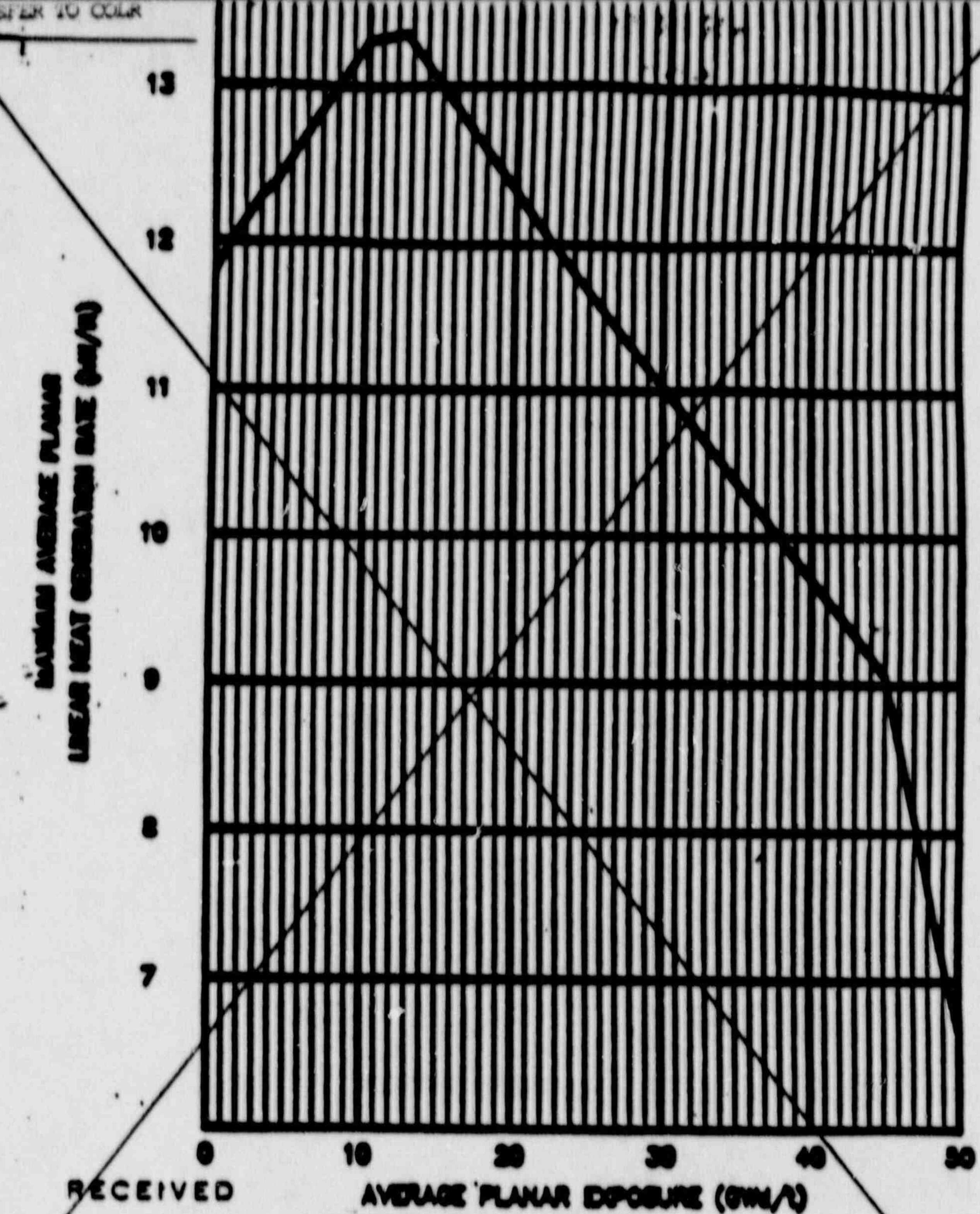

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FIGURE 3.2.1-7

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE - [illegible]

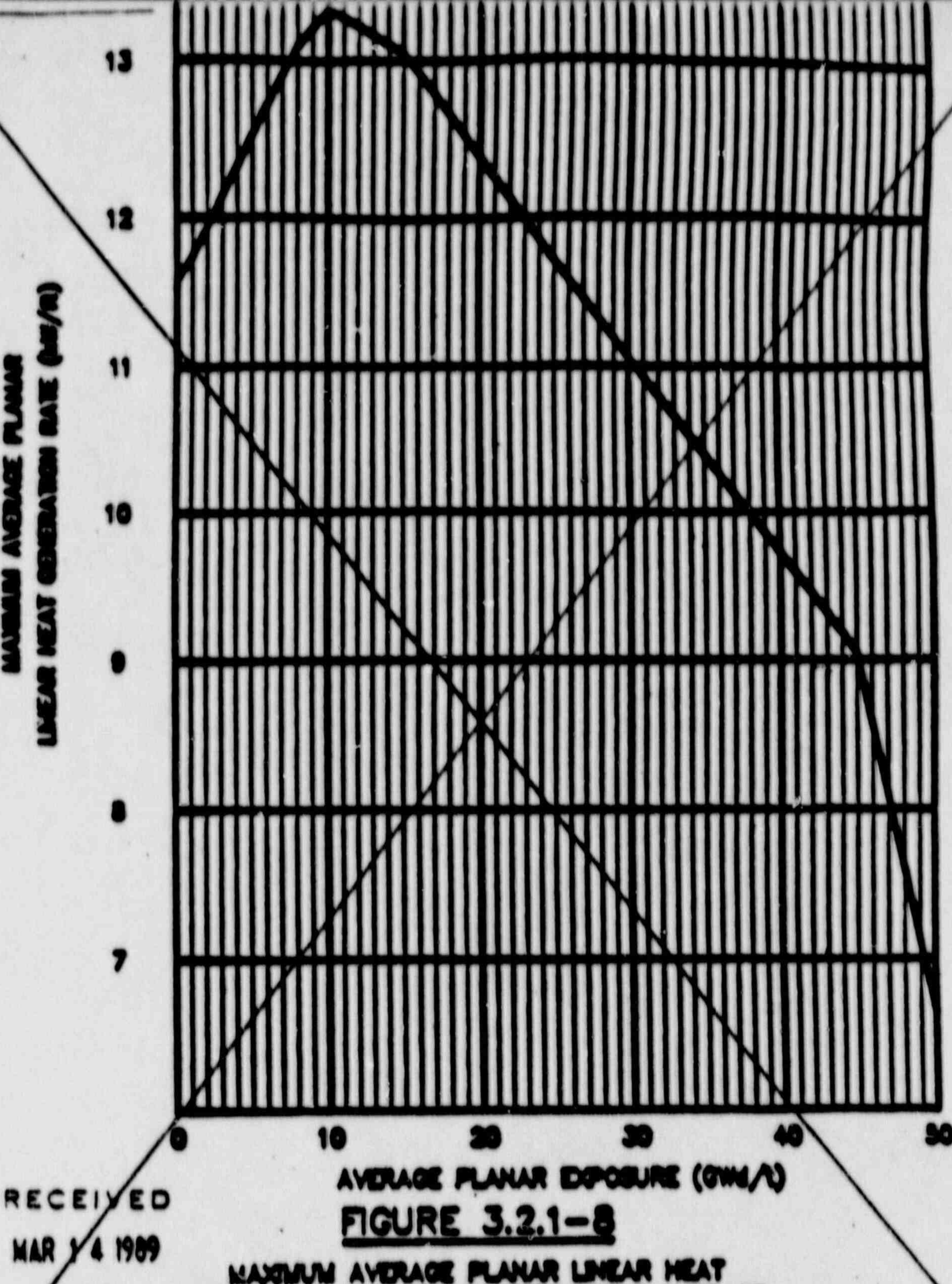

FIGURE 3.2.1-8

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE - BS322C

POWER DISTRIBUTION LIMITS

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 both $MCPR_f$ and $MCPR_p$ limits at indicated core flow and THERMAL POWER as ~~shown in Figures 3.2.3-1 and 3.2.3-2~~. specified in the COLR.

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 3.2.3-1 and 3.2.3-2~~, 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

4.2.3 MCPR shall be determined to be equal to or greater than the MCPR limit ~~determined from Figures 3.2.3-1 and 3.2.3-2~~. specified in the COLR.

a. At least once per 24 hours,

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

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

d. The provisions of Specification 4.0.4 are not applicable.

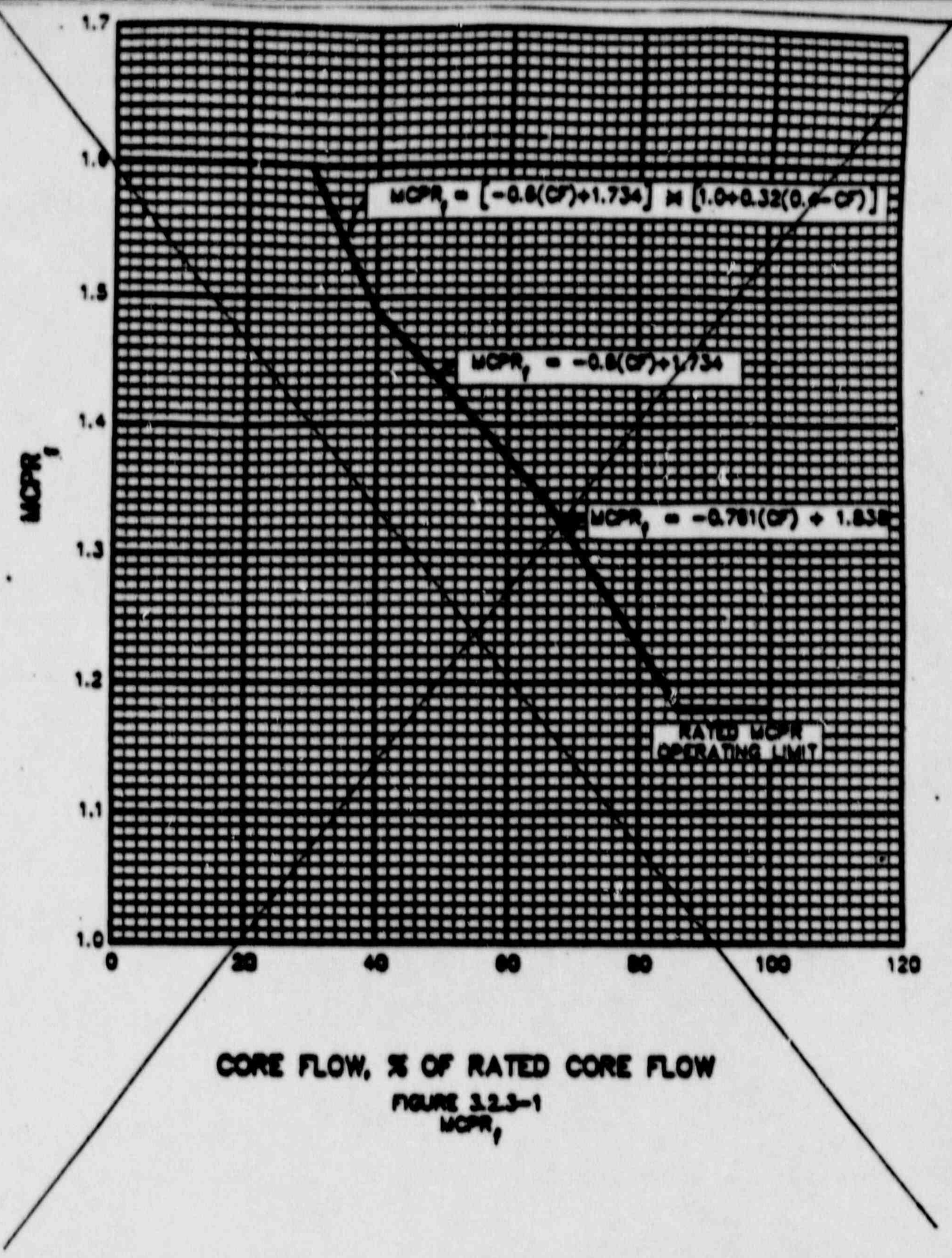

FIGURE 3.2.3-1
$MCPR_f$

TRANSFER TO COLR

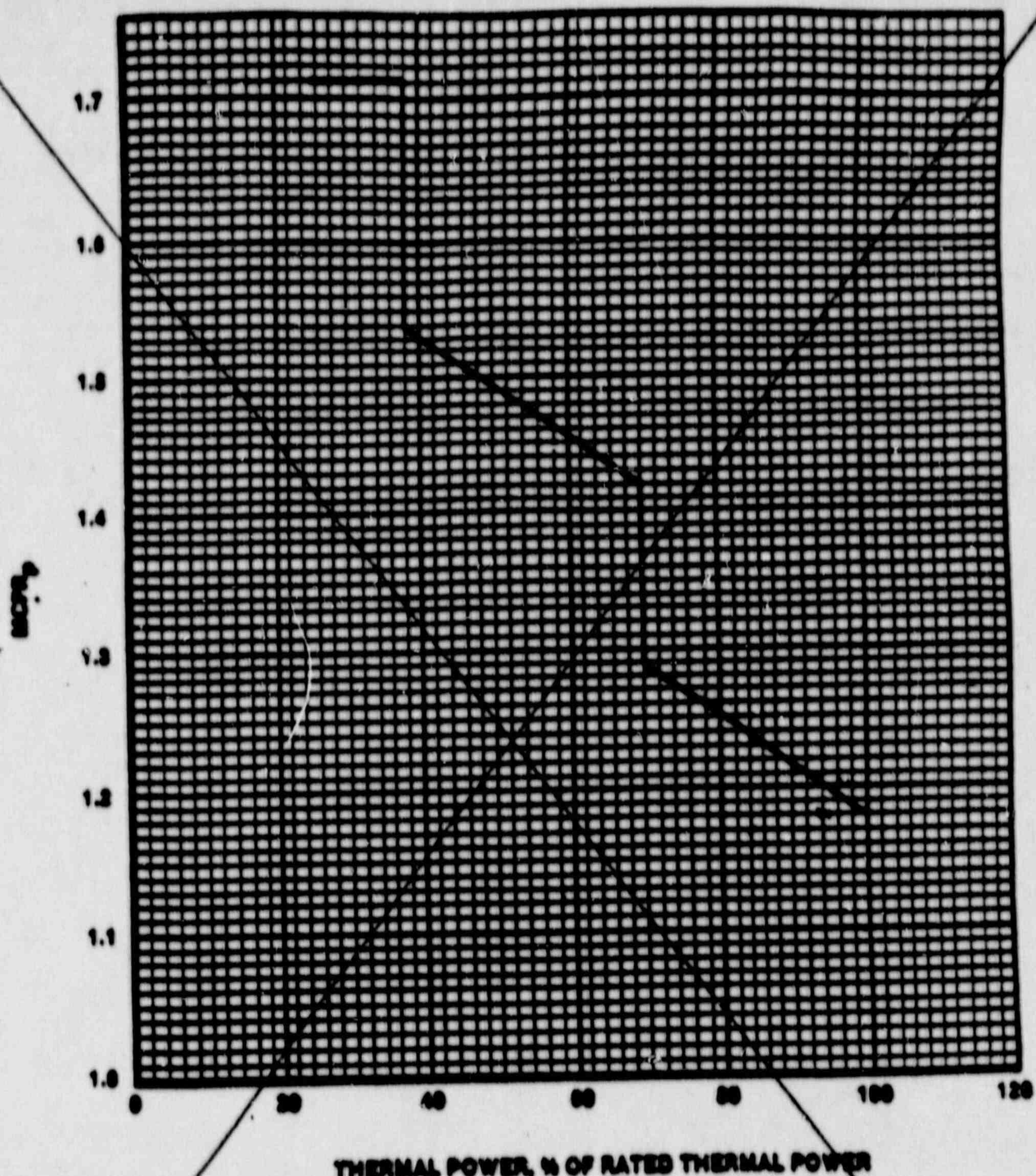

FIGURE 3.2.3-2
$MCPR_p$

The limits specified in the COLR

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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 ~~14.4 kw/ft for GE8X8EB* fuel and 13.4 kw/ft for all other 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

4.2.4 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, 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.

~~*GE8X8EB Fuel includes types BS322B and BS322C~~.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - High, Setdown	NA
b. Flow Biased Simulated Thermal Power - High	≤0.09**
c. Neutron Flux - High	≤0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	≤0.35
4. Reactor Vessel Water Level - Low, Level 3	≤1.05
5. Reactor Vessel Water Level - High, Level 8	≤1.05
6. Main Steam Line Isolation Valve - Closure	≤0.09
7. Main Steam Line Radiation - High	NA
8. Drywell Pressure - High	NA
9. Scram Discharge Volume Water Level - High	
a. Level Transmitter	NA
b. Float Switches	NA
10. Turbine Stop Valve - Closure	≤0.06
11. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	≤0.07#
12. Reactor Mode Switch Shutdown Position	NA
13. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.

**Not including simulated thermal power time constant, ~~6 ± 0.6 seconds~~. specified in the COLR.

#Measured from start of turbine control valve fast closure.

is within the limits specified in the COLR.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

(g) Calibrate Rosemount trip unit setpoint at least once per 31 days.

(h) Verify measured drive flow to be less than or equal to established drive flow at the existing flow control valve position.

(i) This calibration shall consist of verifying the simulated thermal power time constant to be less ~~than 6.6 seconds.~~

(j) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(k) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(l) This function is not required to be OPERABLE when DRYWELL INTEGRITY is not required per Specification 3.10.1

(m) Verify the Turbine Bypass Valves are closed when THERMAL POWER is greater than or equal to 40% RATED THERMAL POWER.

(n) The CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION shall include the turbine first stage pressure instruments.

(o) The CHANNEL CALIBRATION shall exclude the flow reference transmitters; these transmitters shall be calibrated at least once per 18 months.

(p) This period may be extended to the first refueling outage, not to exceed 9-15-87.

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5.3 REACTOR CORE

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FUEL ASSEMBLIES

~~5.3.1 The reactor core shall contain 624 fuel assemblies. Each assembly consists of zirconium alloy fuel and water rods arranged in a nominal 8x8 array. The fuel rods contain uranium dioxide fuel pellets with active lengths generally ranging between 144 and 150 inches. These fuel assemblies are limited to those that have been analyzed with NRC approved codes and methods and have been shown to comply with all of the criteria in the latest approved revision of GESTAR (NEDE-24011-P-A-US).~~

CONTROL ROD ASSEMBLIES

~~5.3.2 The reactor core shall contain 145 control rod assemblies, each consisting of a cruciform array of stainless steel tubes surrounded by a cruciform shaped stainless steel sheath. Each tube shall contain 143.7 inches of boron carbide (B_4C) powder.~~

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 18,000 cubic feet.

Replacement paragraphs for Technical Specifications 5.3.1, FUEL ASSEMBLIES and 5.3.2, CONTROL ROD ASSEMBLIES.

FUEL ASSEMBLIES

5.3.1 The reactor shall contain 624 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy clad fuel rods with an initial composition of slightly enriched uranium dioxide (UO_2) as fuel material. Fuel assemblies shall be limited to those fuel designs approved by the NRC Staff for use in BWR's.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 145 cruciform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C) and/or hafnium metal. The control rod assemblies shall be full length.

SEMIANNUAL EFFLUENT RELEASE REPORT (Continued)

to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1.3) during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. The assessment of radiation doses shall be performed in accordance with the methodology and parameters of the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most-exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the ODCM, as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2

SPECIAL REPORTS

6.9.2 Special reports shall be submitted in the following manner:

a. Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the Regional Office of the NRC and a copy to the NRC Resident Inspector, within the time period specified for each report.

b. Special reports in regard to Corbicula will be submitted to the NRC within 30 days of identification of infestation. In accordance with the settlement agreement dated October 10, 1984, these reports shall describe the level of infestation, affected systems and measures taken to prevent further infestation.

Insert, see attached.

6.10 RECORD RETENTION

6.10.1 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.

6.10.2 The following records shall be retained for at least 5 years:

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

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CORE OPERATING LIMITS REPORT

6.9.3.1 Core operating limits shall be established prior to startup from each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. The AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) for Specification 3.2.1.

b. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.3.

c. The LINEAR HEAT GENERATION RATE (LHGR) of Specification 3.2.4.

d. The REACTOR PROTECTION SYSTEM (RPS) response time for APRM thermal time constant for Specification 3.3.1.

and shall be documented in the CORE OPERATING LIMITS REPORT (COLR).

6.9.3.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).

6.9.3.3 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits and nuclear limits such as shutdown margins, and transient and accident analysis limits) of the safety analysis are met.

6.9.3.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revision 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.

ENCLOSURE II

CORE OPERATING LIMITS REPORT SUBMITTAL
89-03

BASES REVISION SUMMARY

Subject (Section)	Revision
Limiting Safety System Settings (B 2.0)	Remove reference to the 6 second simulated thermal time constant and setpoint formula.
Average Planar Linear Heat Generator Rate (B3/4.2.1)	Provide APLHGR references to COLR
Minimum Critical Power Ratio (B 3/4.2.3)	Provide MCPR reference to COLR

specified in the COLR

specified in Specification 2.1.2

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions ~~as specified in Specification 3.2.3~~ are derived from the established fuel cladding integrity Safety Limit MCPR ~~of 1.07~~ and an analysis of abnormal operational transients. For any abnormal operating transient analysis, with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip settings given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR ~~of 1.07~~, the required minimum operating limit MCPR ~~of Specification 3.2.3~~ specified in the COLR is obtained ~~and is presented in Figure 3.2.3-1~~. Analysis of transients occurring during single recirculation loop operation indicates that the maximum operating limit MCPR will be bounded by the limits ~~in Specification 3.2.3.~~ The power-flow map of Figure B 3/4 2.3-1 shows typical regions of plant operation.

specified in the COLR.

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

specified in the COLR

The purpose of the $MCPR_f$ and $MCPR_p$ ~~of Figures 3.2.3-1 and 3.2.3-2 is to~~ define operating limits at other than rated core flow and power conditions. At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_f$ and $MCPR_p$ at the existing core flow and power state. The $MCPR_f$s are established to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

The $MCPR_f$s were calculated such that, for the maximum core flow rate and the corresponding THERMAL POWER along the 105%-of-rated steam flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the 105%-of-rated steam flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined as $MCPR_f$.

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REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Neutron Flux-High setpoint; i.e., for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow Biased Simulated Thermal Power-High setpoint, a time constant ~~of 6 ± 0.6 seconds~~ is introduced into the flow biased APRM in order to simulate the fuel thermal transient characteristics. A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown. The flow referenced trip setpoint must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when CMFLPD is $\geq$ to FRTP.

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine control valve fast closure and turbine stop valve closure trips are bypassed. For a load rejection or turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint has been used in transient analyses dealing with coolant inventory decrease. The scram setting was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

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

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure-dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. ~~The limiting~~ value for APLHGR is ~~shown in Specification 3.2.1~~. specified in the CORE OPERATING LIMITS REPORT (COLR).

The daily requirement for calculating APLHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate APLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating APLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that APLHGR will be known following a change in THERMAL POWER or power shape that could place operation into a condition exceeding a thermal limit.

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

a. Input Changes

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

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

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AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

3. Corrected guide tube thermal resistance.

4. Correct heat capacity of reactor internals heat nodes.

b. Model Change

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

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

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

a. Input Change

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

b. Model Change

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

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

For plant operation with a single recirculation loop, the MAPLHGR limits ~~of figures 3.2.1-1 through 3.2.1-6~~ are multiplied by 0.84. The constant factor 0.84 is derived from LOCA analyses initiated from single recirculation loop operation to account for earlier boiling transition at the limiting fuel mode compared to the standard LOCA evaluations.

3/4.2.2 APRM SETPOINTS

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