Technical Specification Page No.	Technical Specification Section	Description of Change	Reason for Change
B3/4 2-7	Fig. B3/4.2.3-1	Replace the current operating map with a new operating map.	MEOD
B3/4 2-8	Fig. B3/4.2.3-2 (new)	Add an operating map for single recirculation loop operation.	Clarification
B3/4 6-5	B3/4.6.2.5	Revise the value for drywell peak calculated pressure from 18.9 psig to 19.7 psig.	MEOD

GSL/bjc
RFP3/GSL10

1.0 DEFINITIONS

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Insert "A" to Index

Editorial Note

Addition of the above figures will cause the page numbers of the index to change.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM RESPONSE TIME

1.7 The CONTAINMENT AND REACTOR VESSEL ISOLATION AND CONTROL SYSTEM (CRVICS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

CORE ALTERATION

1.8 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, or 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.

CRITICAL POWER RATIO

an approved General Electric Critical Power

1.9 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of ~~the General Electric Critical Quality Boiling Length (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 Table 3.6.4-1 of Specification 3.6.4.

b. The drywell equipment hatch is closed and sealed.

c. The drywell airlock is OPERABLE pursuant to Specification 3.6.2.3.

DRYWELL INTEGRITY (Continued)

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

e. The suppression pool is OPERABLE pursuant to Specification 3.6.3.1.

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

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.13 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function; i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.14 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker from initial movement of the associated:

a. Turbine stop valves and
b. Turbine control valves.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

[DELETED]

~~FRACTION OF LIMITING POWER DENSITY~~

1.15 ~~The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.~~

~~FRACTION OF RATED THERMAL POWER~~

1.16 ~~The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.~~

[DELETED]

FREQUENCY NOTATION

1.17 The FREQUENCY NOTATION specified for the performance of surveillance requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

1.18 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the main condenser evacuation system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.19 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank or

b. Leakage into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

LIMITING CONTROL ROD PATTERN

1.20 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.21 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.22 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc., of a logic circuit from sensor through and including the actuated device to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

[DELETED]

MAXIMUM FRACTION OF LIMITING POWER DENSITY

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

MEMBER(S) OF THE PUBLIC

1.24 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors or vendors. Also excluded from this category are

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~~ 1.07 with two recirculation loop operation and shall not be less than ~~1.07~~ 1.08 with single recirculation 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 ~~1.06~~ 1.07 with two recirculation loop operation or less than ~~1.07~~ 1.08 with single loop operation 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.

REACTOR VESSEL WATER LEVEL

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

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Intermediate Range Monitor		
a. Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
b. Inoperative	NA	NA
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-High		
1) During two recirculation loop operation:		
a. ~~1)~~ Flow Biased	≤ 0.66 (W-ΔW) + ~~48~~ 64%,(a) with a maximum of	≤ 0.66 (W-ΔW) + ~~51~~ 67%,(a) with a maximum of
b. ~~2)~~ High Flow Clamped	≤ 111.0% of RATED THERMAL POWER	≤ 113.0% of RATED THERMAL POWER
Insert W		
c. Neutron Flux-High	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1065 psig	≤ 1080 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 8.9 inches above instrument zero*	≥ 8.3 inches above instrument zero
5. Reactor Vessel Water Level-High, Level 8	≤ 52.0 inches above instrument zero*	≤ 52.6 inches above instrument zero
6. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
7. Main Steam Line Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background

Insert W for Page 2-3

2) During single recirculation loop operation:

a) Flow Biased	$\leq 0.66(W-\Delta W)+48\%$[a]	$\leq 0.66(W-\Delta W)+51\%$[a]
b) High Flow Clamped	Not Required OPERABLE	Not Required OPERABLE

2.1 SAFETY LIMITS

BASES

2.1.0 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 ~~1.06 for two recirculation loop operation and 1.07 for single recirculation loop operation.~~ the value given in Specification 2.1.2. MCPRs greater than these safety limits represent 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 GEXL correlation~~ an approved General Electric Critical Power correlation (Reference 1) is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28×10^3 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.

Insert A

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB[a], 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 the General Electric Critical Quality (X) Boiling Length (L), GEXL, correlation. The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation.

The required input to the statistical model are the uncertainties listed in Bases Table B2.1.2-1 and the nominal values of the core parameters listed in Bases Table B2.1.2-2.

The basis for the uncertainty in the GEXL correlation is given in NEDO-10958-A* and the bases for the uncertainties in the core parameters are given in NEDO-20340.** The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution during any fuel cycle would not be as severe as the distribution used in the analysis.

a

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The Safety Limit for the reactor coolant system pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the

a

*"General Electric BWR Thermal Analysis Bases (GETAB) Data, Correlation and Design Application," NEDO-10958-A.

**General Electric "Process Computer Performance Evaluation Accuracy," NEDO-20340 and Amendment 1, NEDO-20340-1, dated June 1974 and December 1974, respectively.

Insert "A" to page B 2-2

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

Reference

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

BASES TABLE B 2.1.2-1

UNCERTAINTIES USED IN THE DETERMINATION OF THE FUEL CLADDING SAFETY LIMIT

QUANTITY	STANDARD DEVIATION (% of Point)
Feedwater Flow	1.76
Feedwater Temperature	0.76
Reactor Pressure	0.5
Core Inlet Temperature	0.2
Core Total Flow	
Two Recirculation Loop Operation	2.5
Single Recirculation Loop Operation	6.0
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	
Two Recirculation Loop Operation	6.3
Single Recirculation Loop Operation	6.8
R Factor	1.5
Critical Power	3.6

Note: The uncertainty analysis used to establish the corewide Safety Limit MCPR is based on the assumption of quadrant power symmetry for the reactor core.

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BASES TABLE B 2.1.2-2

NOMINAL VALUES OF PARAMETERS USED IN
THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT

PARAMETER	VALUE
THERMAL POWER	3323 MW
Core Flow	108.5 Mlb/hr
Dome Pressure	1010.4 psig
Channel Flow Area	0.1089 ft^2
R-Factor	High enrichment - 1.043 Medium enrichment - 1.039 Low enrichment - 1.030

e

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LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 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 MFLPD is > 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 during operation 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 stop valve closure and turbine control valve fast closure trips are bypassed. For a turbine trip or load rejection 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.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod# from the fully withdrawn position, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed the following limits:

Reactor Vessel Dome Pressure (psig)*#	Maximum Insertion Times to Notch Position (Seconds) 43	29	13
950	0.31	0.81	1.44
1050	0.32	0.86	1.57

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the maximum scram insertion time of one or more control rods exceeding the maximum scram insertion time limits of Specification 3.1.3.2 as determined by Surveillance Requirement 4.1.3.2.a or b, operation may continue provided that:

   1. For all "slow" control rods, i.e., those which exceed the limits of Specification 3.1.3.2, the individual scram insertion times do not exceed the following limits:

Reactor Vessel Dome Pressure (psig)*#	Maximum Insertion Times to Notch Position (Seconds) 43	29	13
950	0.38	1.09	2.09
1050	0.39	1.14	2.22

   2. For "fast" control rods, i.e., those which satisfy the limits of Specification 3.1.3.2, the average scram insertion times do not exceed the following limits:

Reactor Vessel Dome Pressure (psig)*#	Maximum Average Insertion Times to Notch Position (Seconds) 43	29	13
950	0.30	0.78	1.40
1050	0.31	0.84	1.53

*For the initial fuel cycle only, up to 32 control rods, not occupying adjacent locations in any direction, including the diagonal, may be exempted from scram time testing under hot, pressurized conditions, provided they meet all other surveillance requirements, and that all scram reactivity requirements of the plant safety analysis are met with no scram contribution from these rods.

#For intermediate reactor vessel dome pressure, the scram time criteria are determined by linear interpolation at each notch position.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods* shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days,

b. For specifically affected individual control rods** following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

c. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

~~*For the initial fuel cycle only, up to 32 control rods, not occupying adjacent locations in any direction, including the diagonal, may be exempted from scram time testing under hot, pressurized conditions, provided they meet all other surveillance test requirements, and that all scram reactivity requirements of the plant safety analysis are met with no scram contribution from these rods.~~ e

**The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 2 provided this surveillance requirement is completed prior to entry into OPERATIONAL CONDITION 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) 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, and 3.2.1-3. The limits of Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 shall be reduced to a value of 0.85 times the two-recirculation loop operation limit when in single recirculation loop operation.

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

ACTION:

With an APLHGR exceeding the limits of Figures ~~3.2.1-1, 3.2.1-2, or 3.2.1-3,~~ 3.2.1-3 through 3.2.1-7, as multiplied 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: ~~determined from Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3.~~

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,

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

d. The provisions of Specification 4.0.4 are not applicable.

as determined below:

a. During two recirculation loop operation - the limits shown in Figures 3.2.1-3 through 3.2.1-7 multiplied 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.

b. During single recirculation loop operation - the limits shown in Figures 3.2.1-3 through 3.2.1-7 multiplied by the smallest of $MAPFAC_f$, $MAPFAC_p$ or 0.85.

New Figure

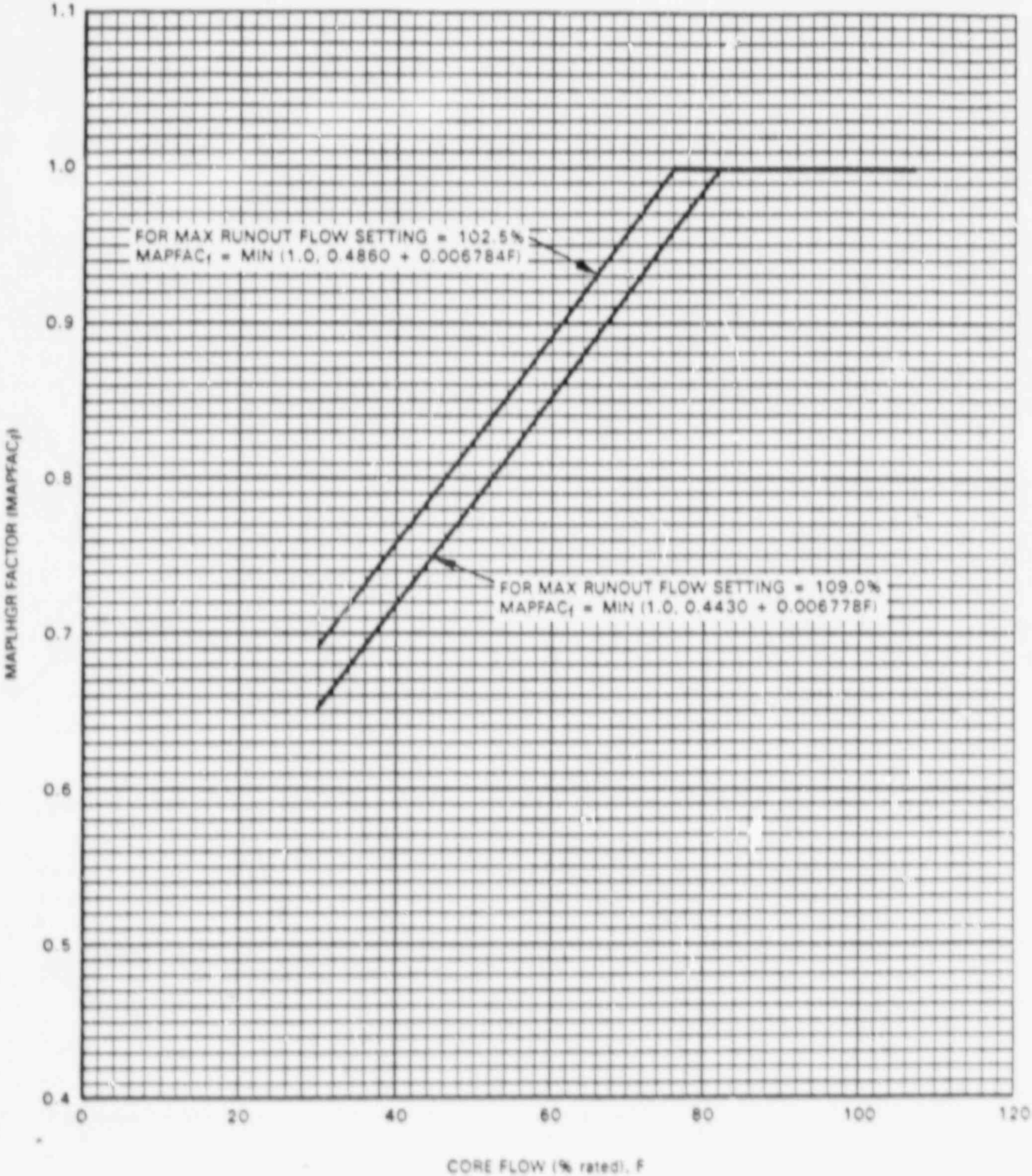


Figure 3.2.1-1 Flow-Dependent MAPLHGR Factors (MAPFAC$_f$)

New Figure

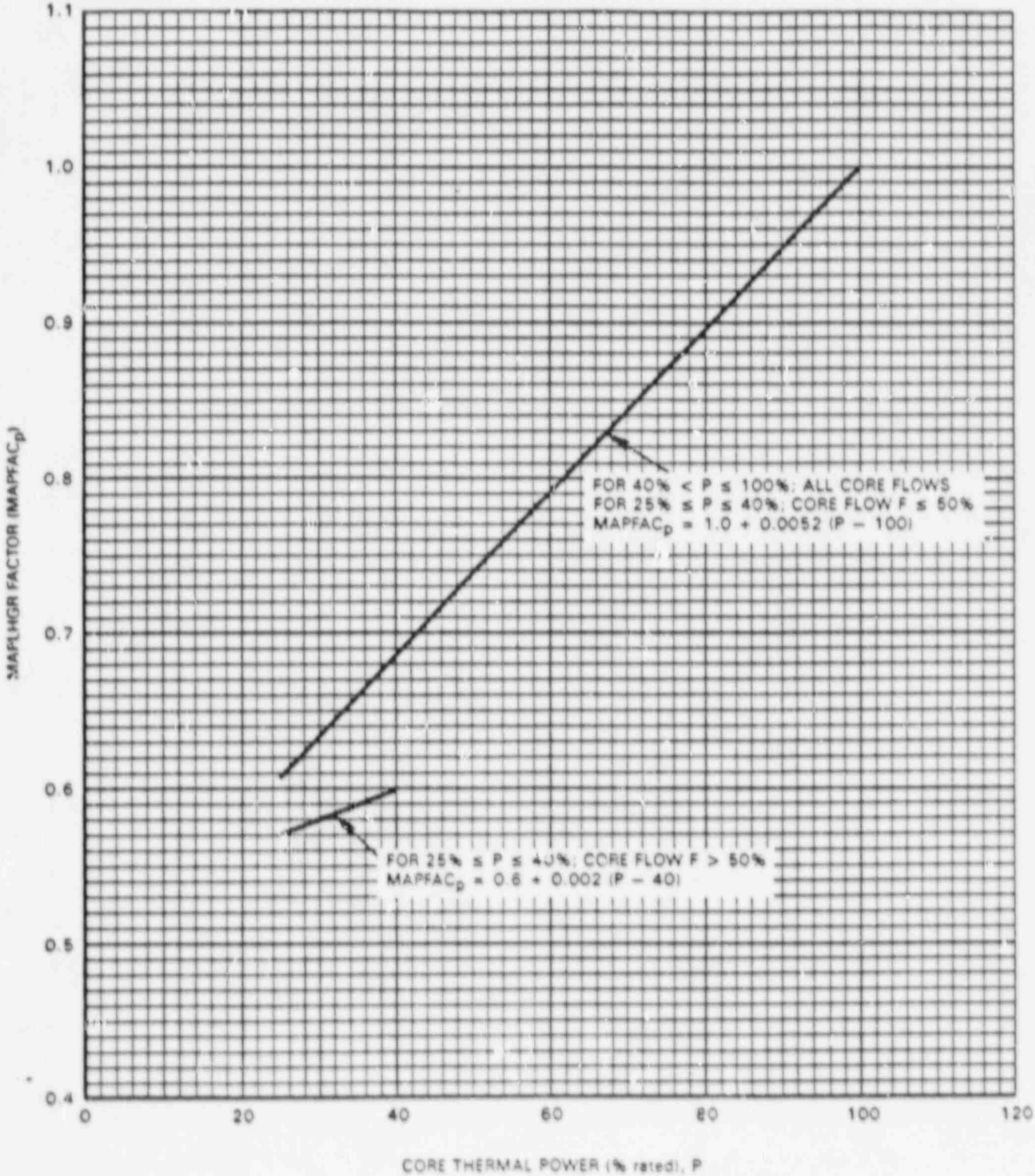


Figure 3.2.1-2 Power-Dependent MAPLHGR Factors (MAPFAC$_p$)

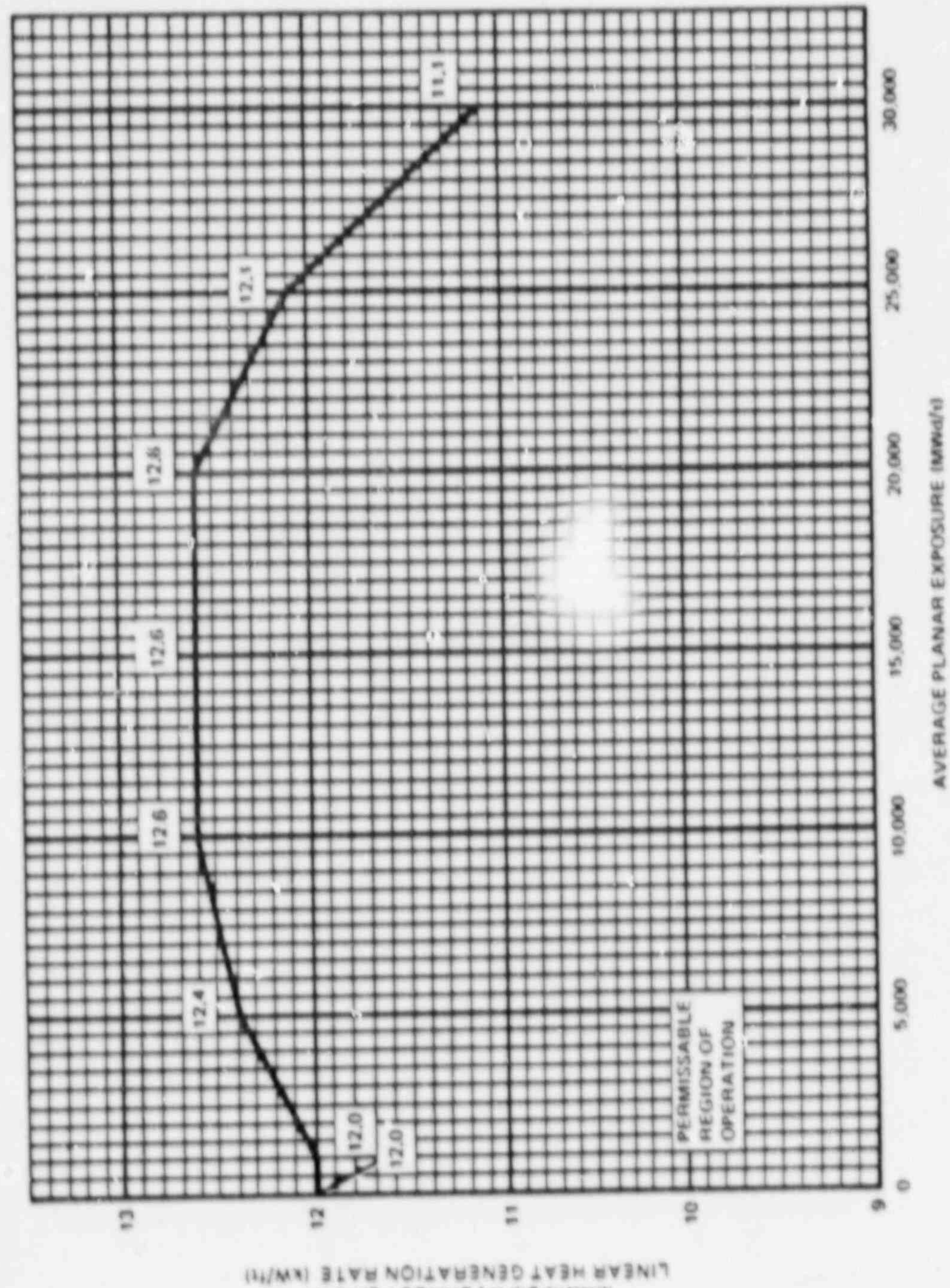


Figure 3.2.1-~~2~~3 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Initial Core Fuel Types - High Enrichment

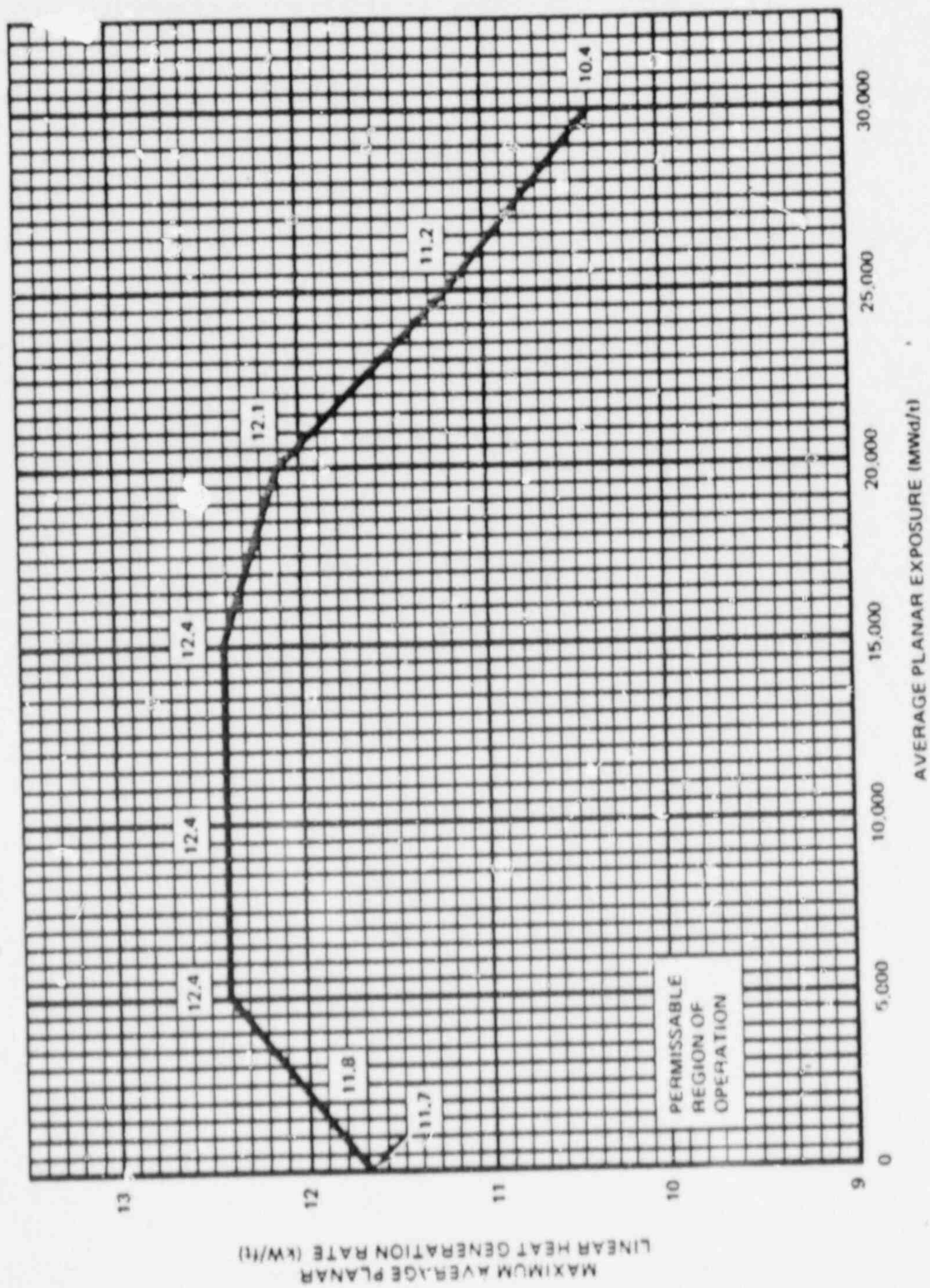


Figure 3.2.1-~~2~~4 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Initial Core Fuel Types - Medium Enrichment

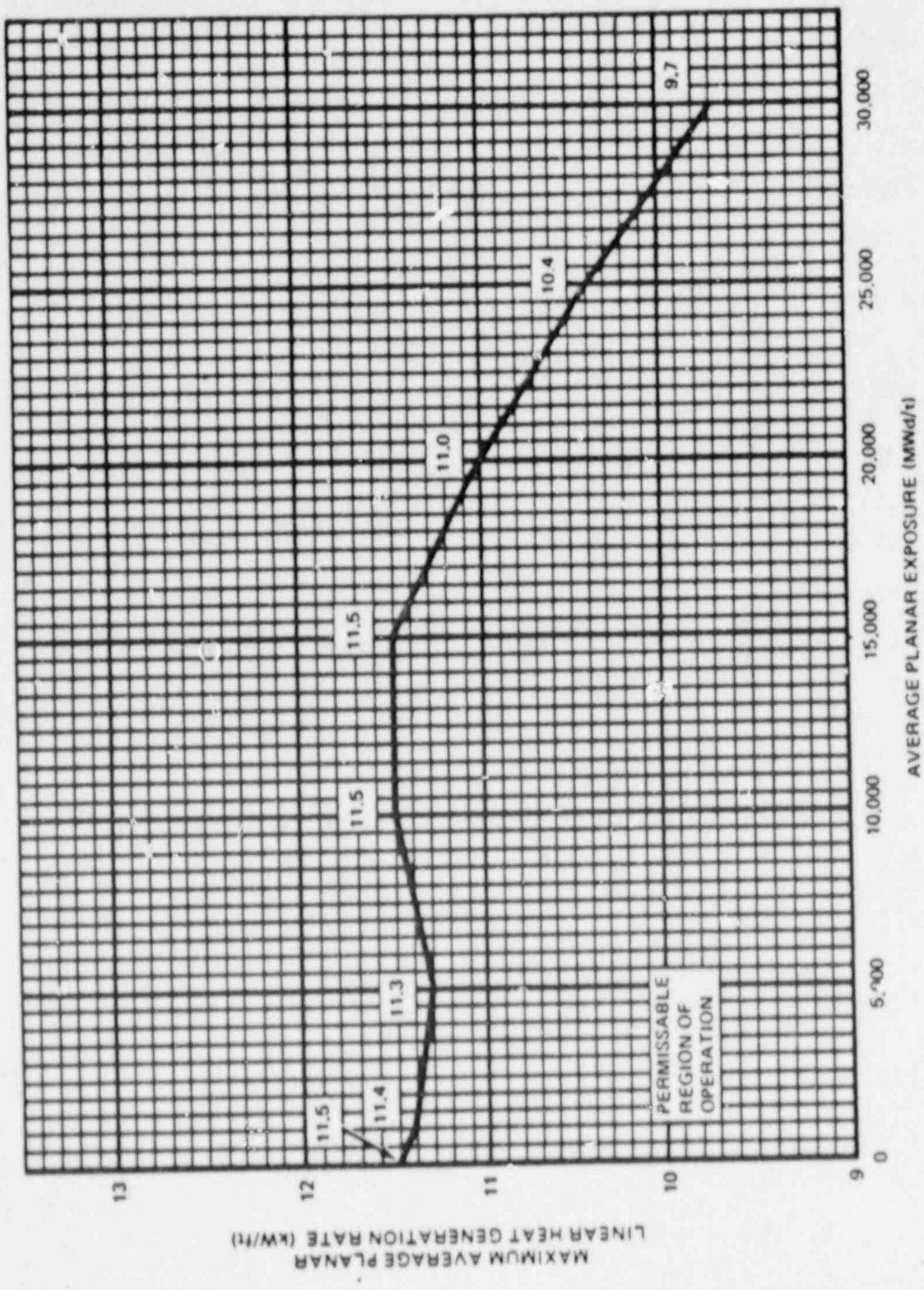


Figure 3.2.1-~~2~~5 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Initial Core Fuel Types - Natural Enrichment

New Figure

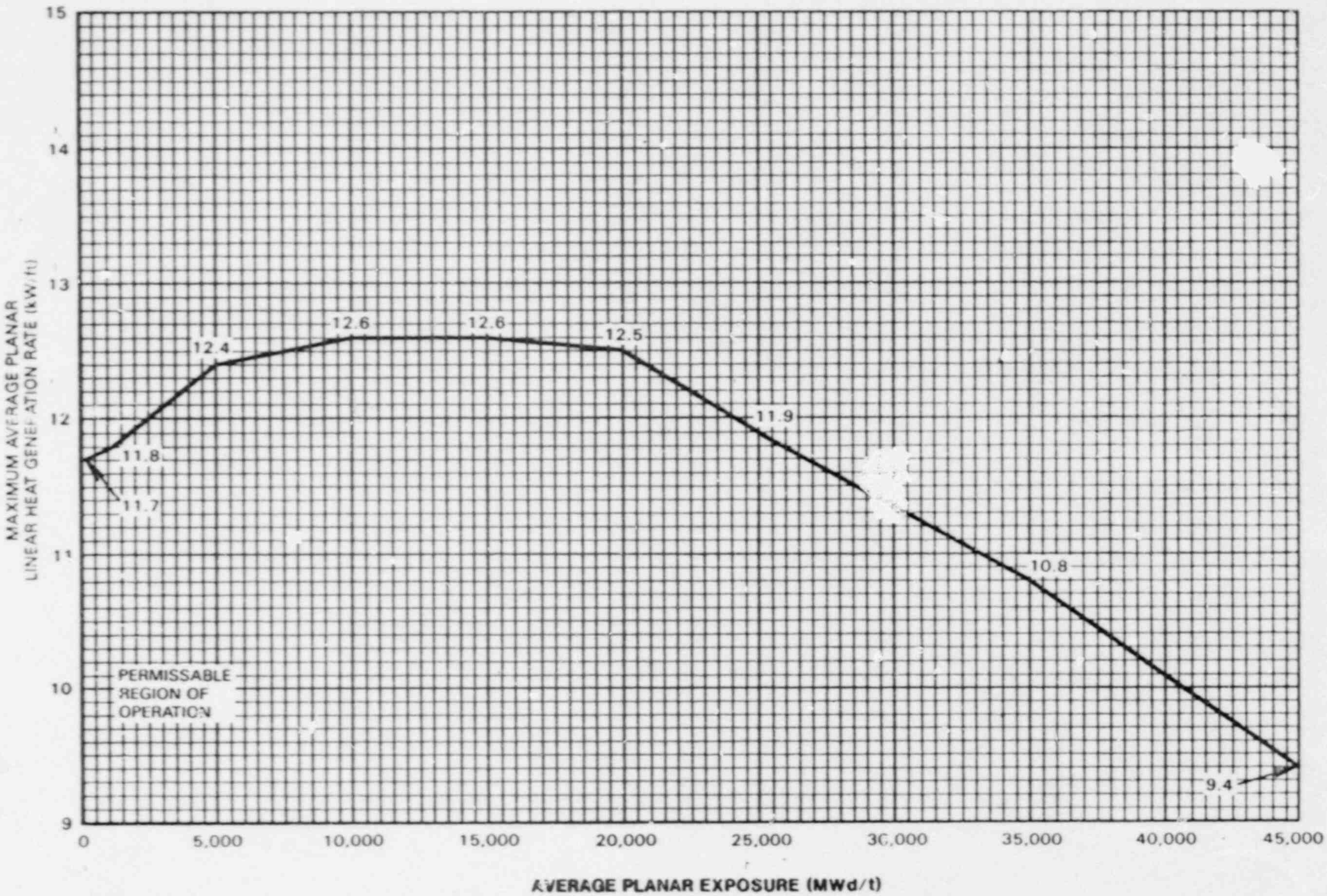
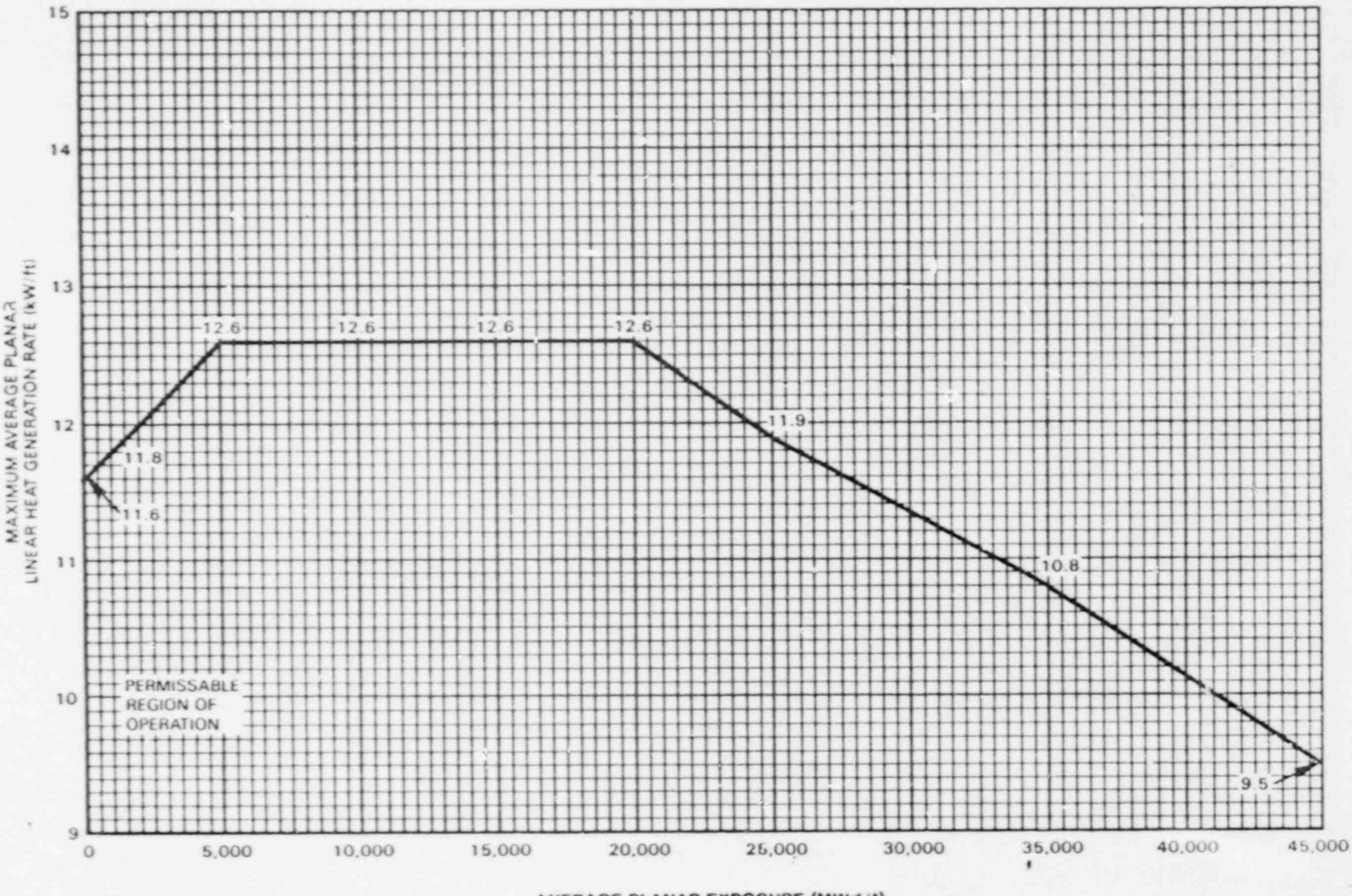


Figure 3.2.1-6 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)
Versus Average Planar Exposure – Reload 1 Fuel Type BP8SRB284L



New Figure

Figure 3.2.1-7 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure - Reload 1 Fuel Type BP8SRB284LC

POWER DISTRIBUTION LIMITS

3/4.2.2 APRM SETPOINTS [DELETED]

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LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow-biased simulated thermal power-high scram trip setpoint (S) and flow-biased neutron flux-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

TRIP SETPOINT	ALLOWABLE VALUE
$S \leq (0.66(W-\Delta W) + 48\%)T$	$S \leq (0.66(W-\Delta W) + 51\%)T$
$S_{RB} \leq (0.66(W-\Delta W) + 42\%)T$	$S_{RB} \leq (0.66(W-\Delta W) + 45\%)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER,

W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 84.5 million lbs/hr.

ΔW = Difference in indicated drive flow (in percent of drive flow which produces the same core flow) between two loop and single loop operation at the same core flow. See note (a) to Table 2.2.1-1.

T = The ratio of FRACTION OF RATED THERMAL POWER (FRTP) divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD). T is applied only if less than or equal to 1.0.

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

ACTION:

With the APRM flow-biased simulated thermal power-high scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as determined above, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint value* within 6 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP and MFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow-biased simulated thermal power-high scram and flow-biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

a. At least once per 24 hours,

*With MFLPD greater than the FRTP rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that the APRM readings are greater than or equal to 100% times MFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of the adjustment is posted on the reactor control panel.

POWER DISTRIBUTION LIMITS

APRM SETPOINTS [DELETED]

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2 (Continued)

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 MFLPD greater than or equal to FRTP.

The provisions of Specification 4.0.4 are not applicable.

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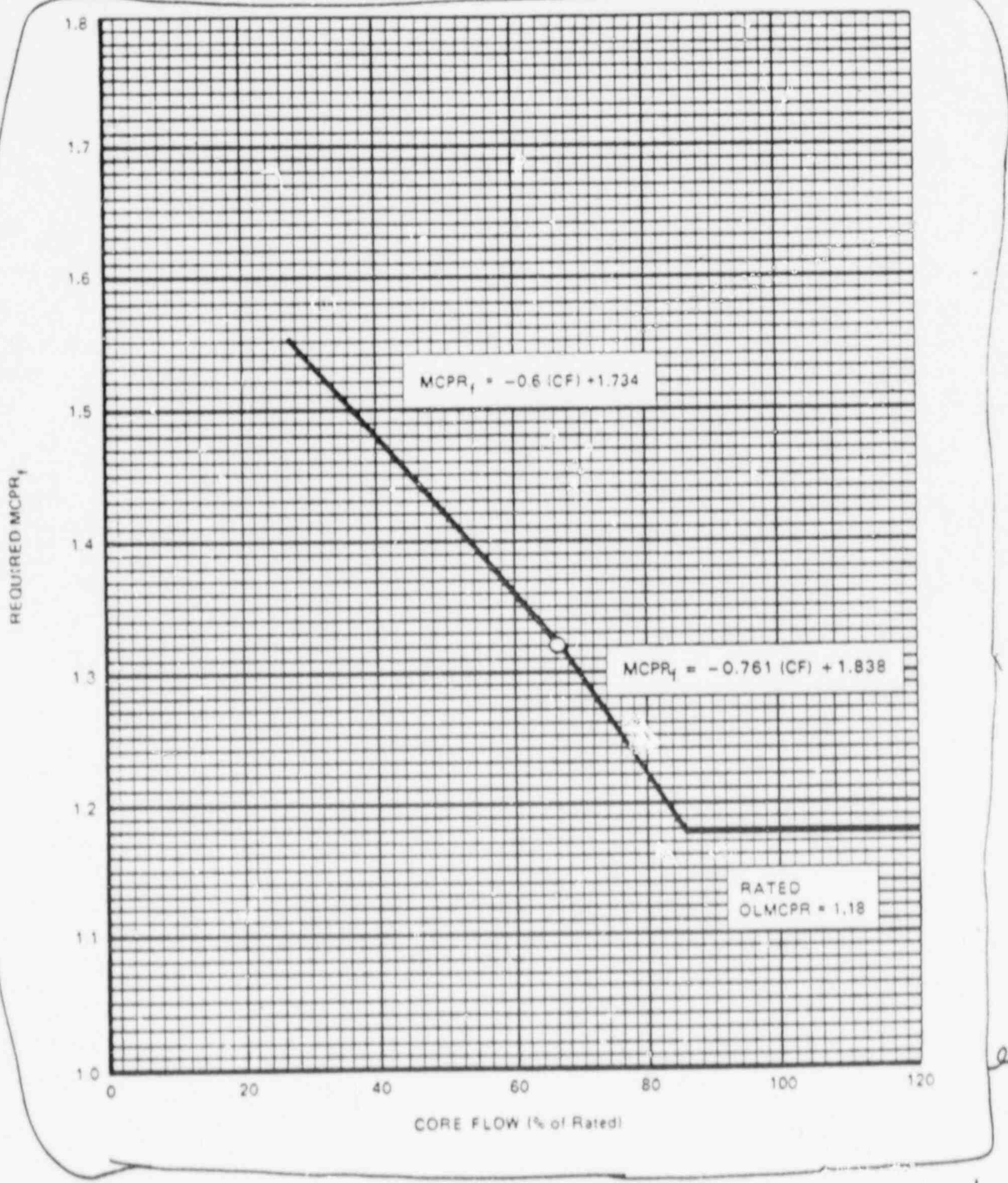


Replace with revised
Figure 3.2.3-1 attached

Figure 3.2.3-1 Clinton $MCPR_f$ Versus Core Flow

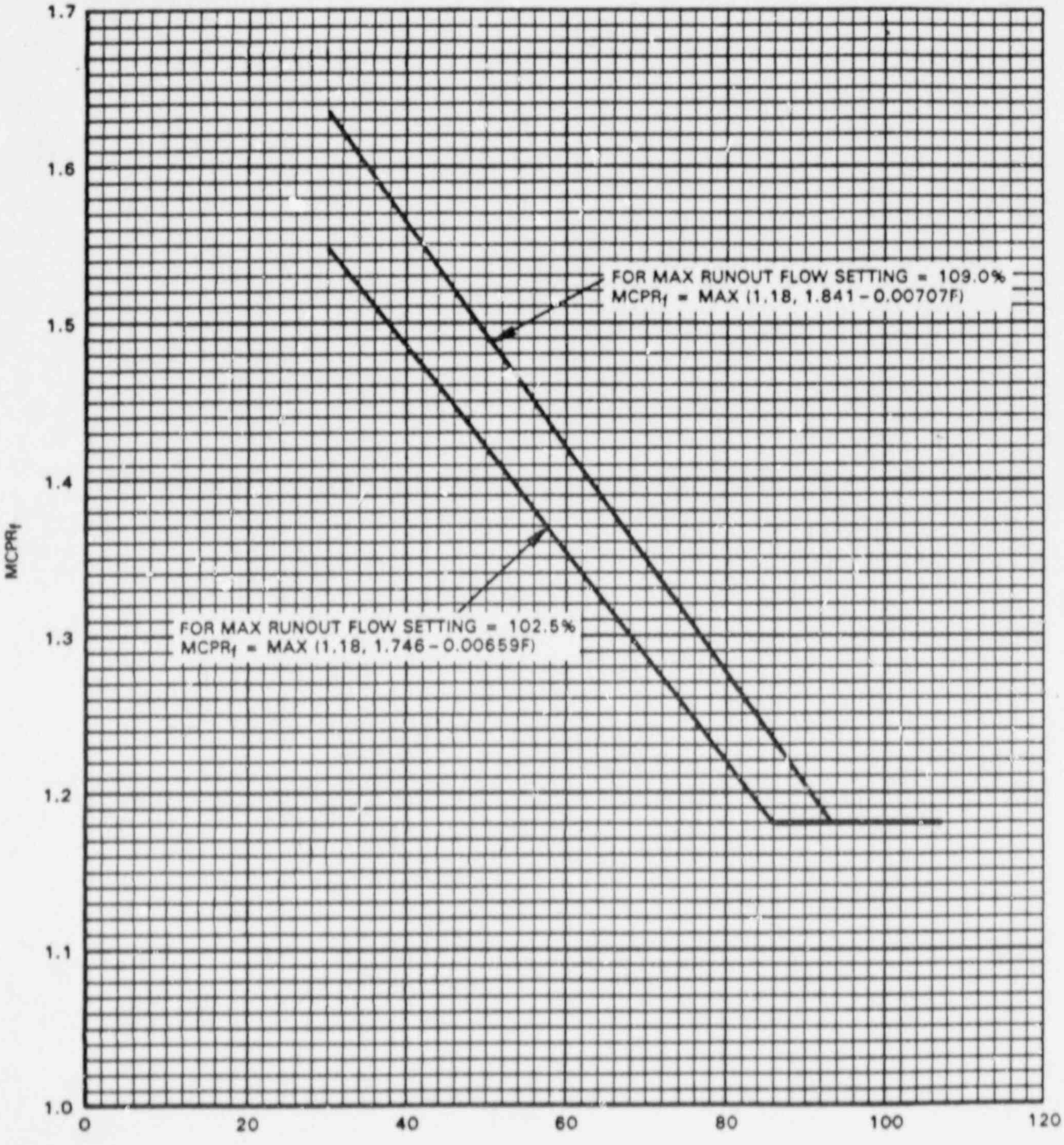
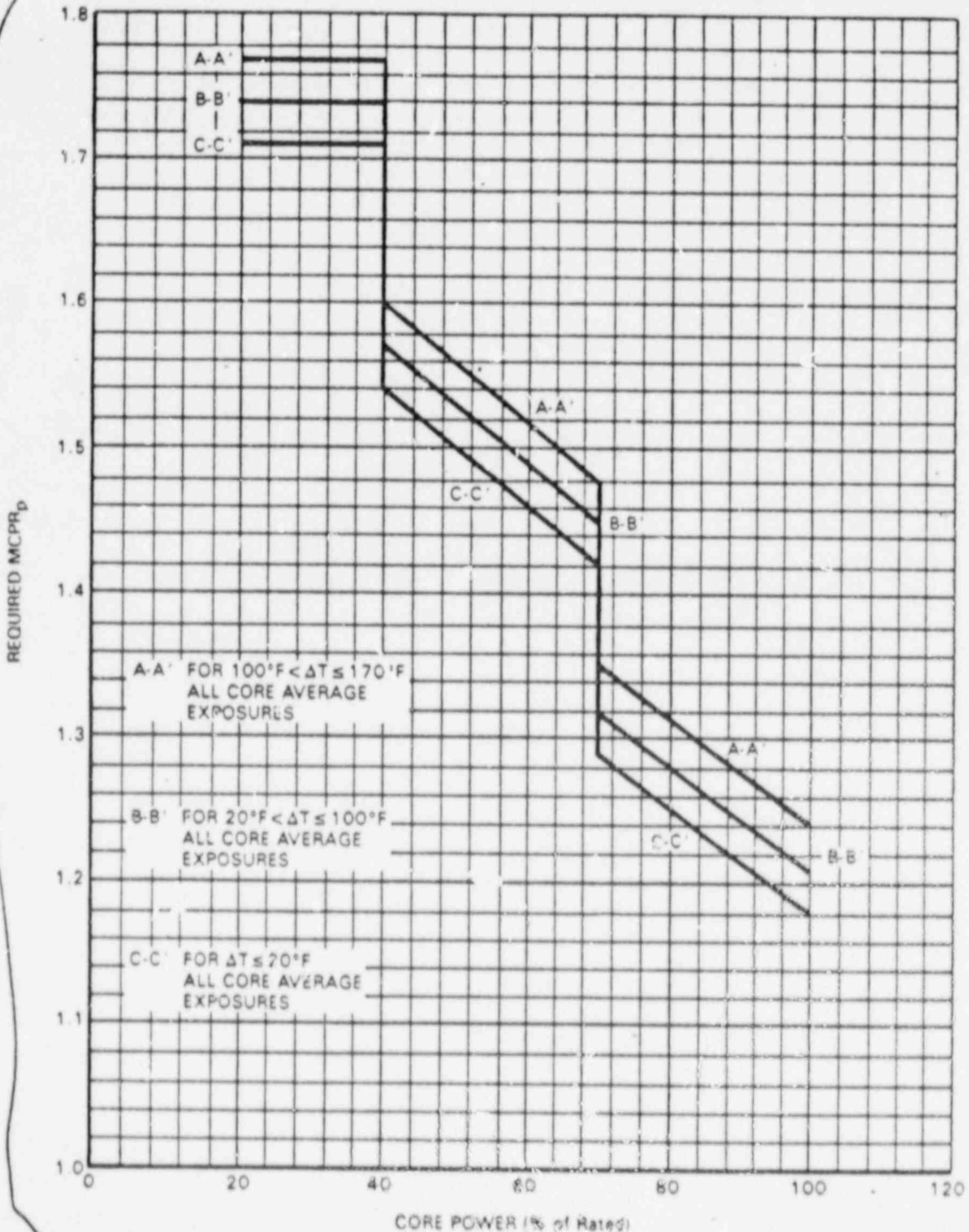


Figure 3.2.3-1 Clinton $MCPR_f$ Versus Core Flow



Clinton $MCPR_p$ Versus Power
Figure 3.2.3-2

Replace with revised Figure 3.2.3-2 attached.

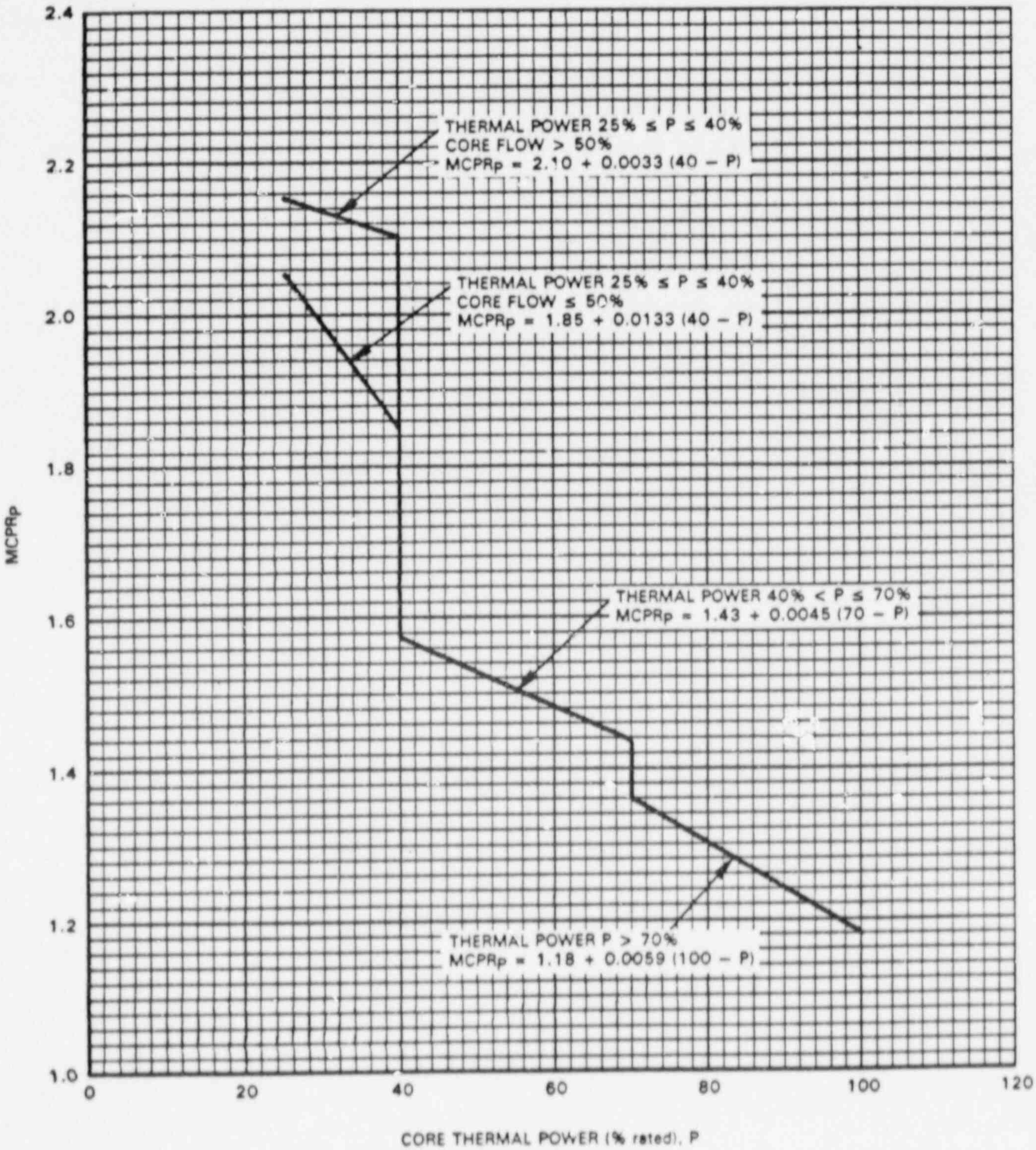


Figure 3.2.3-2 Clinton $MCPR_p$ Versus Power for $\Delta T \leq 50^{o}F$ and Core Flow $\leq$ 107%

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decade during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1 decade during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days.

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.

(e) This calibration shall consist of a setpoint verification of the Neutron Flux-High and the Flow Biased Simulated Thermal Power-High trip functions. The Flow Biased Simulated Thermal-High trip function is verified using a calibrated flow signal.

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

(g) Calibrate the analog trip module at least once per 31 days.

(h) Verify measured core (total core flow) flow to be greater than or equal to established core flow at the existing loop flow control (APRM % flow).

(i) This calibration shall consist of verifying the 6±0.6 second simulated thermal power time constant.

(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 to be OPERABLE per Special Test Exception 3.10.1.

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

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD PATTERN CONTROL SYSTEM		
a. Low Power Setpoint	(*)% of RATED THERMAL POWER	(*)% of RATED THERMAL POWER
b. RWL High Power Setpoint	(*)% of RATED THERMAL POWER	(*)% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - Upscale	~~≤ 0.66 (W-ΔW) + 42%**~~	~~≤ 0.66 (W-ΔW) + 45%**~~
Insert X → b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$< 1 \times 10^5$ cps	$< 1.6 \times 10^5$ cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps	≥ 1.8 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	< 108/125 division of full scale	< 110/125 division of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 division of full scale	≥ 3/125 division of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High, C11-N602A	≤ 12" #	≤ 19 7/8" #
b. Water Level-High, C11-N602B	≤ 12" ##	≤ 19 7/8" ##
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ ~~108%~~ 113% of rated flow	≤ ~~111%~~ 116% of rated flow
7. REACTOR MODE SWITCH		
a. Shutdown Mode	NA	NA
b. Refuel Mode	NA	NA

Insert X to page 3/4 3-66

1) During two recirculation loop operation:		
a) Flow Biased	≤0.66W + 58%** with a maximum of	≤0.66W + 61%** with a maximum of
b) High Flow Clamped	≤108.0% of RATED THERMAL POWER	≤110.0% of RATED THERMAL POWER
2) During single recirculation loop operation:		
a) Flow Biased	≤0.66(W-ΔW) + 42%**	≤0.66(W-ΔW) + 45%**
b) High Flow Clamped	Not required OPERABLE	Not required OPERABLE

TABLE 3.3.6-2 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TABLE NOTATIONS

* To be determined during startup test program. The actual setpoints are the corresponding values of the turbine first stage pressure for these power levels.

** The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with ~~Specification 3.2.2, and~~ note (a) of Table 2.2.1-1.

\# Instrument zero is 758' 5" msl.

\#\# Instrument zero is 758' 4 1/2" msl.

INSTRUMENTATION

TRAVERSING IN-CORE PROBE SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.7 The traversing in-core probe system shall be OPERABLE with:

a. Four movable detectors, drives and readout equipment to map the core and

b. Indexing equipment to allow all four detectors to be calibrated in a common location.

APPLICABILITY: When the traversing in-core probe is used for:

a. Recalibration of the LPRM detectors and

b. Monitoring the APLHGR, LHGR, or ~~,~~ MCPR ~~,~~ ~~or MFLPD~~.*

ACTION:

With the traversing in-core probe system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.7 The traversing in-core probe system shall be demonstrated OPERABLE by normalizing each of the above required detector outputs within 72 hours prior to use when required for the LPRM or calibration functions.

*Only the detector(s) in the location(s) of interest are required to be OPERABLE.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation with:

a. Total core flow greater than or equal to 45% of rated core flow, or

b. THERMAL POWER within the unrestricted zone of Figure 3.4.1.1-1, or

c. THERMAL POWER within the restricted zone† of Figure 3.4.1.1-1 and APRM or LPRM†† noise levels not larger than three times their established baseline noise levels.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

   1. Within 4 hours:

      a) Place the recirculation flow control system in the Local Manual (Position Control) mode, and

      b) Reduce THERMAL POWER TO ≤ 70% of RATED THERMAL POWER, and

      c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to ~~1.07~~ 1.08 per Specification 2.1.2, and

      d) Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) ~~limit to a value of 0.85 times the two-recirculation-loop operation~~ limit per Specification 3.2.1, and

      e) Reduce the Average Power Range Monitor (APRM) Scram and Rod Block Trip Setpoints and Allowable Values to those applicable for single-recirculation-loop operation per Specifications 2.2.1, ~~3.2.2,~~ and 3.3.6, and

*See Special Test Exception 3.10.4.

†The operating region for which monitoring is required. See Surveillance Requirement 4.4.1.1.2.

††Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

BASES

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

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is ~~shown in Figures 3.2.1-1, 3.2.1-2 and 3.2.1-3.~~ the MAPLHGR.

Insert Y

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-3 through 3.2.1-7 ~~3.2.1-1, 3.2.1-2 and 3.2.1-3~~ 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 Reference 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.

3. Corrected guide tube thermal resistance.

4. Correct heat capacity of reactor internals heat nodes.

Insert Y to page B 3/4 2-1

The MAPLHGR limits of Figures 3.2.1-3 through 3.2.1-7 are multiplied by the smaller of the flow-dependent MAPLHGR factor ($MAPFAC_f$) or the power-dependent MAPLHGR factor ($MAPFAC_p$) corresponding to existing core flow and power conditions to assure the adherence to fuel mechanical design bases during the most limiting transient (Reference 2). The $MAPFAC_f$ factors are determined using the three-dimensional BWR simulator code to analyze slow flow runout transients. The maximum runout flow settings of 102.5% and 109% include design allowances for recirculation flow instrument uncertainties (2.5% and 2.0% respectively) to ensure that the rated flow conditions of 100% and 107% can be achieved. The $MAPFAC_p$ factors are generated using the same data base as the $MCPR_p$ to protect the core from plant transients other than core flow runout.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

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

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

a. Input Change

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

b. Model Change

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

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

3.2.1-3 through 3.2.1-7

the smallest of $MAPFAC_f$, $MAPFAC_p$ or

For plant operation with a single recirculation loop, the MAPLHGR limits of Figures 3.2.1-1, 3.2.1-2 and 3.2.1-3 are multiplied by 0.85. The constant factor, 0.85, is derived from LOCA analyses initiated from single loop operation to account for earlier boiling transition at the limiting fuel node compared to standard LOCA evaluations.

(Reference 2)

3/4.2.2 APRM SETPOINTS [DELETED]

The fuel cladding integrity Safety Limits of Specific ion 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 setting and the flow biased neutron flux-upscale control rod block functions of the APRM instruments must be adjusted to ensure that the 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 and rod block setpoints are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and MFLPD indicates a peak power distribution to ensure than a LHGR transient would not be increased in degraded conditions.

BASES TABLE B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS*

Plant Parameters:

Core THERMAL POWER .... 3015 MWt** which corresponds to 105% of rated steam flow

Vessel Steam Output .... 13.08×10^6 lb_m/hr which corresponds to 105% of rated steam flow

Vessel Steam Dome Pressure.............. 1060 psia

Design Basis Recirculation Line Break Area for:

a. Large Breaks 2.2 ft^2.

b. Small Breaks 0.09 ft^2.

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kW/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial and Reload Cores	8 x 8	13.4	1.4	1.17***

*A more detailed listing of input of each model and its source is presented in Section II of Reference 1 and Section 6.3 of the FSAR.

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

***For single recirculation loop operation, loss of nucleate boiling is assumed at 0.1 seconds after a LOCA, regardless of initial MCPR. For core flows less than 85% of rated, the initial MCPR is taken from the $MCPR_f$ Curve.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

in Specification 2.1.2

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 1.06,~~ and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

for double or single recirculation loop operation respectively

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 power-flow map, of FiguresB 3/4.2.3-1, gives operational limits.

identified in Reference 3 Reference 3 or B 3/4.2.3-2 and non-pressurization

The evaluation of a given transient begins with the system initial parameters ~~shown in FSAR Tables 15.0-2 and 15.6.5-1~~ that are input to a GE-core dynamic behavior transient computer program. The codes used to evaluate pressurization events ~~is~~ are described in NEDO-24154[3] and the program used in non-pressurization events is described in NEDO-10802[2]. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic TASC code described in NEDE-25149[4]. The principal result of this evaluation is the reduction in MCPR caused by the transient.

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

The $MCPR_f$s were calculated such that for the maximum core flow rate and the corresponding THERMAL POWER along the ~~105% of rated steam~~ most limiting 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 the ~~105% of rated steam~~ most limiting power flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined as $MCPR_f$.

Insert A

Insert "A" to page B 3/4 2-4

The maximum runout flow settings (109% and 102.5%) include design allowances for recirculation flow instrument uncertainties (2% and 2.5% respectively) to ensure that the rated flow conditions (107% and 100%) can be achieved.

3/4.2.3 MINIMUM CRITICAL POWER RATIO (Continued)

The $MCPR_p$s are established to protect the core from plant transients other than core flow increases, including the localized event such as rod withdrawal error. The $MCPR_p$s were calculated based upon the most limiting transient at the given core power level~~.~~, including feedwater controller and load rejection transients. Insert Z

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 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 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 shift. 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.

REFERENCES:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, November 1975.

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For core power below 40% of RATED THERMAL POWER where the EOC-RPT and reactor scram on turbine stop valve closure and turbine control valve fast closure are bypassed, separate sets of $MCPR_p$ limits are provided for high and low core flows to account for the sensitivity to initial core flows. For core power above 40% of RATED THERMAL POWER, bounding $MCPR_p$ limits were developed.

REFERENCES (Continued):

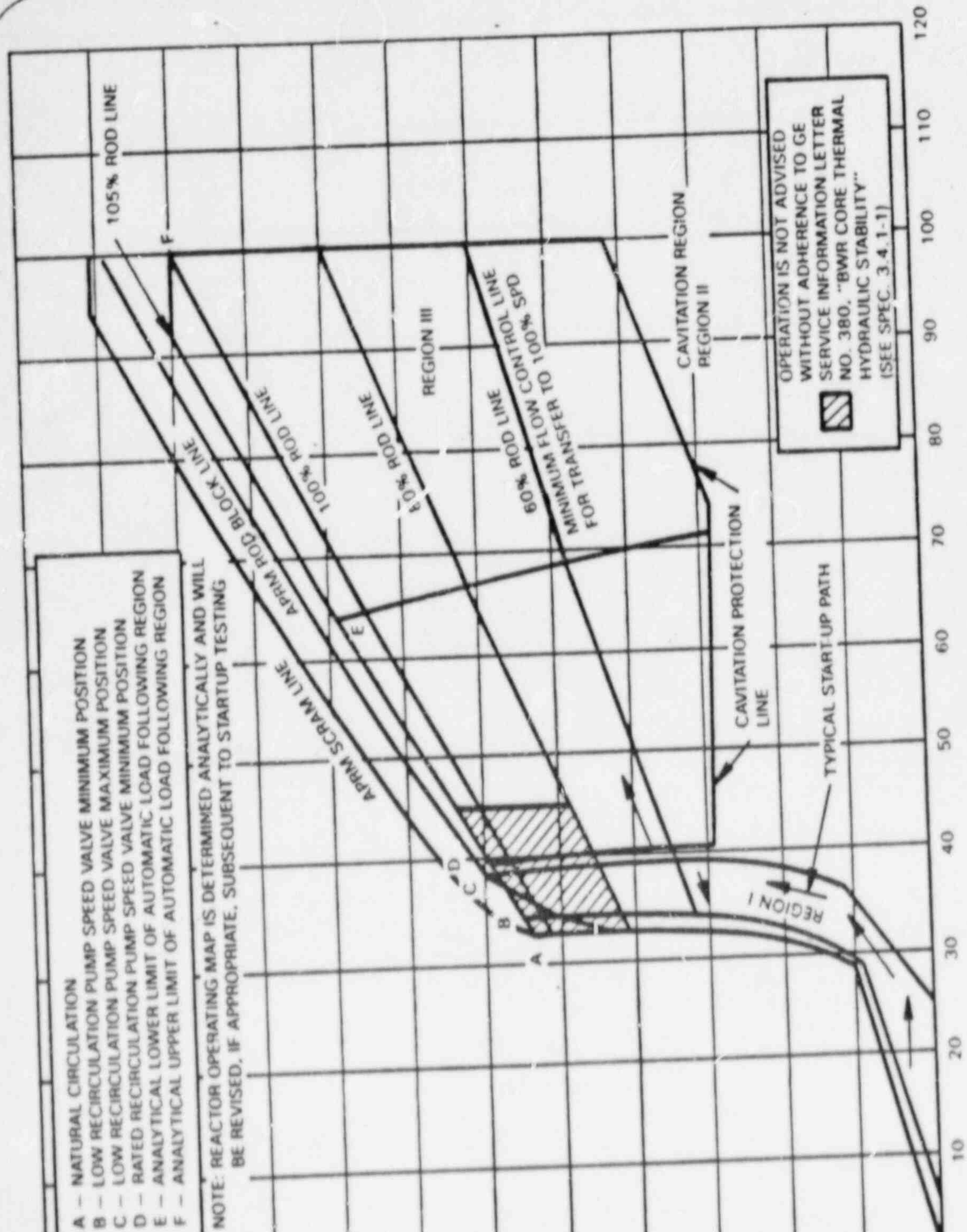
2. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

3. Qualification of the One Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154, October 1978.

4. TASC 01-A Computer Program for The Transient Analysis of a Single Channel, Technical Description, NEDE-25149, January 1980.

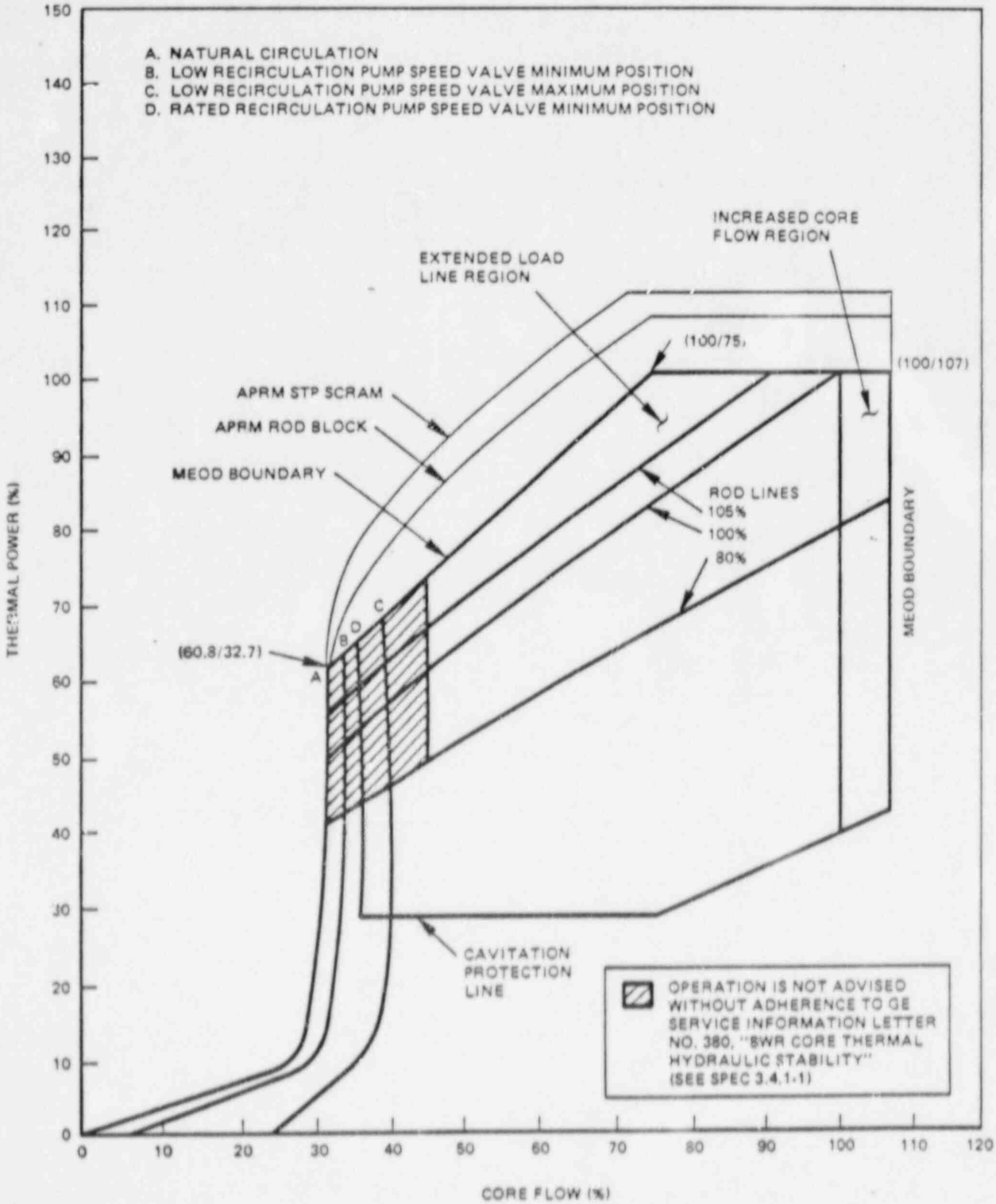
2. Maximum Extended Operating Domain and Feedwater Heater Out-of-Service Analysis for Clinton Power Station, NEDC-31546P, August 1988.

3. General Electric Standard Application for Reactor Fuel (GESTAR), NEDE-24011-P-A-8, as amended.

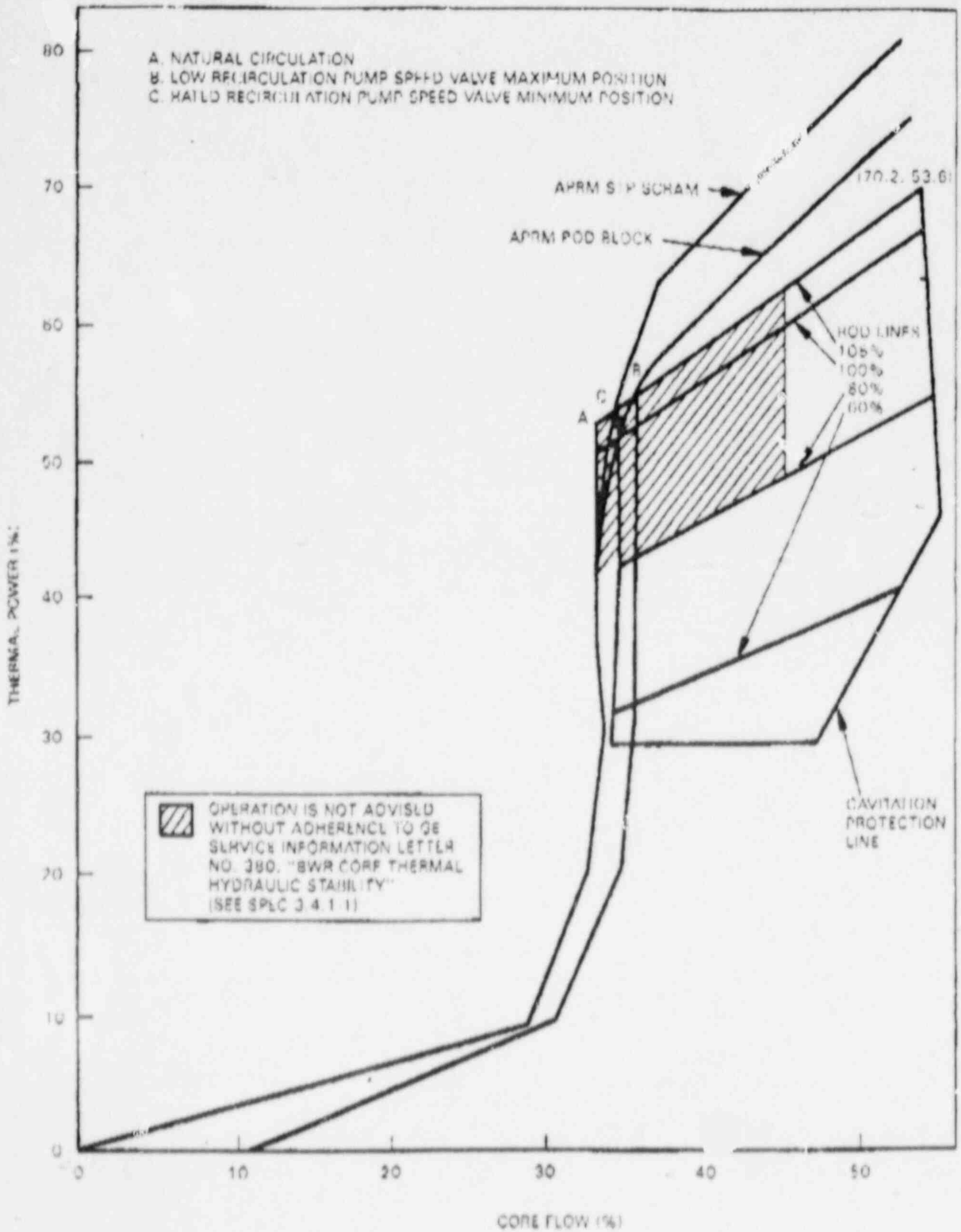


Bases Figure B 3/4.2.3-1 Reactor Operating Map

Replace with Figure B 3/4.2.3-1
Attached.



Bases Figure B 3/4.2.3-1 Reactor Operating Map for Two Recirculation Loop Operation



Bases Figure B 3/4.2.3-2. Reactor Operating Map for Single Recirculation Loop Operation

3/4.6.2.4 DRYWELL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the drywell will be maintained comparable to the original design specification for the life of the unit. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.2.5 DRYWELL INTERNAL PRESSURE

The limitations on drywell-to-containment differential pressure ensure that the drywell peak calculated pressure of ~~18.9~~ 19.7 psig does not exceed the design pressure of 30.0 psig and that the containment peak pressure of 9.0 psig does not exceed the design pressure of 15.0 psig during steam line break conditions. The maximum external drywell pressure differential is limited to 0.2 psid, well below the pressure at which suppression pool water will be forced over the wier wall and into the drywell. The limit of 1.0 psid for initial positive drywell to containment pressure will limit the drywell pressure to ~~18.9~~ 19.7 psid which is less than the design pressure and is consistent with the safety analysis to limit drywell internal pressure.

3/4.6.2.6 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that peak drywell temperature does not exceed the design temperature of 330°F during LOCA conditions and is consistent with the safety analysis.

3/4.6.2.7 DRYWELL VENT AND PURGE

The drywell purge system must be normally maintained closed to eliminate a potential challenge to containment structural integrity due to a steam bypass of the suppression pool Intermittent venting of the drywell is allowed for pressure control during OPERATIONAL CONDITIONS 1, 2, and 3, but the cumulative time of venting is limited to 5 hours per 365 days. Venting of the drywell is prohibited when the 12-inch continuous containment purge system or the 36-inch containment building ventilation system supply or exhaust valves are open. This eliminates any resultant direct leakage path from the drywell to the environment.

In OPERATIONAL CONDITIONS 1, 2 and 3, the drywell isolation valves (IVQ002, IVQ003) can be opened only if they are blocked so as not to open more than 50°. This assures that the valve would be able to close against drywell pressure buildup resulting from a LOCA.

Operation of the drywell vent and purge 24-inch supply and exhaust valves during plant operational conditions 4 and 5 is unrestricted; the 50° blocks may be removed to allow full opening of the valves, and the cumulative time for vent and purge operation is unlimited.

Technical Specification Changes Required to Support Refueling

The reload analysis provided in Attachment 6 "SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR CLINTON POWER STATION UNIT 1, RELOAD 1, CYCLE 2, 23A5921, Rev. 0" provides a licensing basis and the essential analyses for allowing CPS to perform its first reactor refueling (in which new types of nuclear fuel will be utilized) and to proceed with subsequent reactor operation with the reloaded core.

Two new G.E. BWR fuel types will be utilized for the CPS reload. These fuel types have specified MAPLHGR-vs-Core Exposure requirements. Additionally, as noted in the reload analysis, General Electric Standard Application for Reactor Fuel (GESTAR), NEDE-24011-P-A-8, which established the MCPR Safety Limit of 1.06 (1.07 for single recirculation loop operation) for the initial fuel cycle for CPS, requires this safety limit to be increased by 0.01 for reload cores. As the reload analysis includes an evaluation of plant operation (including postulated responses to design basis accidents or transients), the Technical Specifications must be changed to reflect the revised safety or power distribution limits.

Each of the proposed Technical Specification changes related to the reload analysis is briefly described in Attachment 2. Detailed justification for each of these changes is provided below. Some of the changes identified as reload-related are not unique to the first reload (for Cycle 2) as they apply to reloads in general. They should, however, minimize the number of Technical Specification changes that would otherwise be required for future reloads.

Technical Specification 1.9: This change replaced the reference to the GEXL correlation with reference to "an approved GE critical power correlation". This will allow future reloads to be analyzed using other approved critical power correlations, without requiring a change to the definition of CRITICAL POWER RATIO. For cycle 2 the approved GE critical power correlation is the GEXL correlation.

Technical Specification 2.1.2: The change to the MCPR Safety Limit from 1.06 to 1.07 for two recirculation loop operation (and from 1.07 to 1.08 for single recirculation loop operation) is a typical change made at the first refueling outage. The increase is due to increased uncertainties in power distribution, and local fuel rod power.

Technical Specification Bases 2.1.0: This change replaces the specific value for MCPR with a reference to Technical Specification 2.1.2 where the value for MCPR is specified. This change is being proposed to allow the value of MCPR to be changed in future reload applications without requiring a change to this section.

Technical Specification Bases 2.1.1: This change replaces reference to the GEXL correlation with reference to "an approved GE critical power correlation". This will allow future reloads to be analyzed using other approved critical power correlations, without requiring a change to this Technical Specification section.

Technical Specification Bases 2.1.2: This change revises the description of the determination of the MCPR safety limit by referencing the standard GE reload licensing document (GESTAR). GESTAR contains all of the assumptions, analysis methods, and other information used to perform this analysis for GE fuel at CPS.

Technical Specification Bases Tables B2.1.2-1 and B2.1.2-2: This change deletes the cycle specific reload information from the Technical Specification Bases. This information is contained in the GESTAR document. This change allows changes to the nominal values and uncertainties used to perform MCPR calculations to be updated without requiring a change to the Technical Specifications.

Technical Specification 3/4.1.3.2: The footnote which grants relief from scram time testing during the initial fuel cycle is being deleted because the initial fuel cycle will be completed at the beginning of the first refueling outage, therefore, the note is no longer applicable.

Technical Specification Bases 3/4.2.3 (paragraph 1): This change replaces the specific value for MCPR with a reference to Technical Specification 2.1.2 where the value for MCPR is specified. This change is being proposed to allow the value of MCPR to be changed in the future without requiring this section to be changed.

Technical Specification Bases 3/4.2.3 (paragraph 3): This change replaces references to specific analysis methods used to analyze pressurization and non-pressurization transients with a reference to the standard GE reload licensing document (GESTAR), because GESTAR contains these specific analysis methods and other information used to perform analyses for GE fuel at CPS.

Technical Specification Bases section 3/4.2 (References): This change revises the reference list to remove documents that have been removed from Technical Specification Bases section 3/4.2. The documents listed are contained within GESTAR which is now referenced in this section.

Technical Specification Changes Required to Support Operation Consistent with the "MAXIMUM EXTENDED OPERATING DOMAIN AND FEEDWATER HEATER OUT-OF-SERVICE ANALYSIS FOR CLINTON POWER STATION" (MEOD/FWHOS)

Purpose of MEOD/FWHOS Analysis

The "MAXIMUM EXTENDED OPERATING DOMAIN AND FEEDWATER HEATER OUT-OF-SERVICE ANALYSIS FOR CLINTON POWER STATION" provides the licensing basis and essential analyses for permitting reactor operation in an expanded domain. The current power flow operating domain is depicted in Technical Specification Bases Section 3/4.2.3 (Bases Figure B3/4.2.3-1, "Reactor Operating Map"). This operating map was developed based on restrictions such as recirculation pump NPSH, plant control characteristics and core thermal power and flow limits. Safe operation in this region is justified by the accident and transient analyses described in Final Safety Analysis Report (FSAR) Chapters 6 and 15. In order to improve the operating flexibility and the capacity factors for CPS, IP contracted General Electric to evaluate the accident and transient scenarios for the modified operating map in the regions of the Maximum Extended Operating Domain (MEOD).

The MEOD consists of two regions which supplement the current power flow map. One region expands the map to permit flows up to 107% of rated core flow; this region is termed the increased core flow region (ICFR). The second region, known as the extended load-line region (ELLR), permits operation at rated power levels with core flows less than 100%.

By expanding the operating domain allowed on the power flow map, significant benefits can result leading to greater operational flexibility and to improved unit capacity factor. From a core operations and fuel management standpoint the chief benefits are: 1) better power shaping and fuel preconditioning, 2) xenon compensation, and 3) compensation for reactivity reduction due to exposure.

The ability to increase power into the extended load-line region at low core flows allows the withdrawal of more control rod notches. As a result, the plant can attain, or closely approach, the full power target rod pattern. This enhances the capability to obtain optimum axial power shapes prior to encountering fuel preconditioning limitations. The net effect of this capability is an improvement in capacity factor brought about by optimized preconditioning ramps and the elimination of subsequent power reductions to attain the target control rod pattern. The MEOD additionally provides a fuel performance improvement through the reduction in thermal duty cycling on the fuel-cladding interface.

If the rated load line control rod pattern is maintained as core flow is increased, changing equilibrium xenon concentrations will result in less than rated power at rated core flow. Additional operating margin above the rated rod line on the power flow map (as permitted under the MEOD) allows for compensation for power reductions during plant startups due to transient xenon. The gross power reduction due to the reestablishment of equilibrium xenon conditions at rated power have been observed to be as great as 10%-12% during startups with peak xenon and 8%-10% during xenon-free startups. Excess flow capability will ensure, subsequent to attainment of equilibrium xenon, that the plant is capable of maintaining rated power.

In order to maintain a high capacity factor, continued operation at rated conditions is necessary. The effects of xenon buildup and fuel burnup reduce core thermal power and decrease the plant capacity factor. A significant benefit that MEOD offers during rated power operation lies in the fact that rated power conditions can be maintained for a longer period of time without maneuvering rods. This is made possible because rated power can be achieved at less than rated core flow. In the extended load-line region, 100% power can be achieved at 75% flow. Reactivity changes due to fuel burnup, burnable poison depletion, and increased xenon inventory can be countered with variations in core flow. Increased core flow above 100% is an additional aid provided by the increased core flow region of the MEOD. The ability to stay at full power can be extended by increases in core flow above rated core flow.

Included with the MEOD analysis is an evaluation which justifies eliminating the APRM flow-biased simulated thermal power-high scram setpoint adjustment (currently required under Technical Specification 3.2.2). More meaningful power and flow-dependent MAPLHGR limits, together with the new MCPR limits, supersede the need to manually adjust this setpoint. Additional details are provided in the next section of this attachment.

Also provided with the MEOD analysis in Attachment 7, is an analysis performed to evaluate the impact of reduced feedwater temperature on reactor operation at rated (worst case) conditions. It provides justification for operating the reactor with a feedwater temperature from 420°F (rated) down to 370°F at rated conditions with no required changes to the limits or setpoints established or assumed by the MEOD analysis (including the consideration given to eliminating the APRM flow biased simulated thermal power-high scram setpoint adjustment)

Evaluation of MEOD/FWHOS Analysis

As discussed previously, the MEOD analysis provides a basis for permitting reactor operation in an expanded domain. Operation within the new domain was therefore evaluated for impact on the accident and transient analyses. Although the details of this evaluation are a part of the analysis provided in Attachment 7, the key results obtained are summarized below.

1) LOCA analysis - A bounding BWR-6 analysis determined that the current MAPLHGR and $MCPR_f$ limits (FSAR Chapter 6) are adequate for the MEOD.

2) Containment response - A conservative containment analysis produced a peak drywell pressure of 19.7 psig. This pressure is greater then the drywell pressure determined in the FSAR Chapter 6 analysis but still well below the design pressure of 30 psig.

3) Abnormal Transients - A bounding BWR-6 analysis concluded that the delta-CPR results for all cases analyzed in the MEOD are enveloped by the current $MCPR_p$ limits. The $MCPR_f$ curve is revised based on the new analysis of the slow recirculation flow runout transient event to accommodate operation in the ICFR. In addition, the following limiting transients were analyzed in detail for CPS:

   a) Generator Load Rejection with Bypass Failure - As discussed in FSAR Section 15.2.2 this limiting vessel pressurization transient produced a peak vessel pressure of 1225 psig in the FSAR results. When evaluated for the MEOD, the peak pressure increased only slightly to 1226 psig.

   b) Feedwater Flow Controller Failure - Based on the MEOD evaluation, the existing $MCPR_p$ operating limits are adequate to ensure this transient will not violate the MCPR safety limit. ($MCPR_p$ has been revised, however, based on elimination of the APRM flow-biased simulated thermal power-high scram setpoint adjustment as discussed below.)

   c) As described in the attached report, the results of the FSAR Chapter 15 evaluations of both the 100°F Loss of Feedwater Heater Transient and the Rod Withdrawal Error Transient were found to provide adequate protection in the MEOD.

   d) Flow Runout Transient - The evaluation of this transient in the MEOD established the flow-dependent MCPR limits. This event, analyzed at two flow limiter settings, resulted in $MCPR_f$ values which were found to bound other flow dependent abnormal transient events.

4) Stability Evaluation - CPS Technical Specifications have implemented the recommendations of GE SIL-380. The stability compliance of all licensed GE BWR fuel designs, including those fuels described in General Electric Standard Application for Reactor Fuel (GESTAR) is demonstrated in NEDE-22277-P-1. The CPS cycle 2 reload contains GE BWR core fuel and therefore complies with 10CFR50, Appendix A, GDC 12.

   In addition, evaluation demonstrates that substantial thermal-mechanical margin is available for the GE BWR fuel designs even in the unlikely event of very large power oscillations.

5) The effects of increased Reactor Internal Pressure Differences, acoustic loads, flow induced loads, and fuel bundle lift forces have been evaluated and shown to not cause design limits to be exceeded.

   Based on actual flow-induced vibration testing at a valid prototype plant (Kuosheng 1) to 105% of rated flow, and extrapolations of this data to 107% of rated flow, it is predicted that the maximum alternating stresses on vessel internals will be approximately 70% of the acceptance criteria (10,000 psi).

6) Overpressure Protection - The MSIV closure transient was analyzed in the MEOD. The peak vessel pressure is 1245 psig, which is well within the design limit of 1375 psig.

The MEOD analysis also addresses the elimination of the APRM flow-biased simulated thermal power-high scram setpoint adjustment. The transient analyses performed for MEOD defines the appropriate thermal/operating limits which will assure that the criteria associated with thermal-mechanical fuel integrity and LOCA considerations are satisfied without requiring a setpoint adjustment.

With respect to the Feedwater-Heater-Out-of-Service (or Reduced Feedwater Temperature) analysis, which was included with the MEOD analysis in Attachment 7, an evaluation was performed which considered the following:

1) FSAR Chapter 15 abnormal operating transients

2) fuel mechanical design limits

3) LOCA and containment response as described in FSAR Chapter 6

4) fuel integrity thermal-hydraulic stability, and

5) effects of acoustic and flow induced loads.

The results of the evaluation indicate that operation with reduced feedwater temperatures (as low as 370°F at rated power) is acceptable because such operation is supported by the transient analyses performed for MEOD in which it was demonstrated that the acceptance criteria related to fuel integrity and LOCA considerations continue to be satisfied.

Justification for Individual Technical Specification Changes

Technical Specification 1.15: The definition of FRACTION OF LIMITING POWER DENSITY (FLPD) is being deleted because the only section of the CPS Technical Specifications that uses this parameter is Technical Specification 3/4.2.2 which is being deleted as part of this submittal.

Technical Specification 1.16: The definition of FRACTION OF RATED THERMAL POWER (FRTP) is being deleted because the only section of the CPS Technical Specifications that uses this parameter is Technical Specification 3/4.2.2 which is being deleted as part of this submittal.

Technical Specification 1.23: The definition of MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) is being deleted because the only section of the CPS Technical Specifications that uses this parameter is Technical Specification 3/4.2.2 which is being deleted as part of this submittal.

Technical Specification Table 2.2.1-1: This change increases the APRM flow biased scram setpoint and allowable value by 16%. This change is made to accommodate operation in the MEOD. Operation in the MEOD was analyzed as described in the attached MEOD analysis. It was found that operation in this region would not exceed design limits. This proposed change provides access to the extended load line region of the MEOD. The revised setpoints maintain the same slope, the same clamped setpoint and the same margin between the scram and rod block setpoints as the current technical specifications.

Technical Specification Bases 2.2.1: This change to the APRM scram function is an administrative change to delete the reference to Specification 3.2.2 which has been proposed for deletion in this change request. Justification for this deletion is provided below.

Technical Specification 3/4.2.1: As discussed in Section 2.6.3 of the attached MEOD analysis, the MAPLHGR reduction factors ($MAPFAC_f$ and $MAPFAC_p$) are derived from bounding BWR/6 analysis and the Clinton specific analysis where needed. Using these MAPLHGR reduction factors to reduce the rated MAPLHGR limits will ensure that the fuel thermal-mechanical limits will not be exceeded when the APRM flow-biased simulated thermal power-high scram setpoint adjustment (Technical Specification 3/4.2.2) is no longer required.

Technical Specification 3/4.2.2: The current CPS Technical Specifications require that the flow biased scram and rod block setpoints be lowered when the ratio of the Fraction of Rated Power to the Maximum Fraction of Limiting Power Density is less than 1.0. This requirement originated from a now obsolete Minimum Critical Heat Flux Ratio (MCHFR) criterion. The change to the General Electric BWR Thermal Analysis Basis (GETAB), NEDO-10958-A, as a licensing basis and a secondary reliance of flux scram for transient evaluations (for those transients terminated by a scram) now provides a more effective alternative to this requirement. With a revision in the power dependent MCPR limit and new flow and power dependent MAPLHGR reduction factors, it has been demonstrated that operation remains within design and regulatory limits.

Technical Specification Figures 3.2.3-1 and 3.2.3-2: As a result of the MEOD analyses of the slow recirculation flow run out transient a new flow dependent MCPR ($MCPR_f$) limit was established. The proposed curve is slightly greater than the existing curve but is not expected to unduly restrict normal operation.

A new set of power dependent MCPR ($MCPR_p$) limits has been developed based on the evaluation of the elimination of the APRM flow-biased simulated thermal power-high scram setpoint adjustment. The new limits are derived from the results of both CPS-specific and bounding BWR/6 analyses. These limits have been generated considering reduced feedwater temperature, and are therefore applicable to operation with reduced feedwater temperatures.

The operating limit MCPR at any power/flow condition is the larger of the new $MCPR_f$ and the $MCPR_p$. The new values are presented in the revised Figures 3.2.3-1 and 3.2.3-2.

Technical Specification Bases 3/4.2.1: A paragraph is being added to this section to discuss how $MAPFAC_f$ and $MAPFAC_p$ are to be used, and how $MAPFAC_f$ and $MAPFAC_p$ are determined. The source of this information is section 2.6.3 of the MEOD analysis.

The MAPLHGR figures in Technical Specification section 3.2.3 are referred to in two paragraphs of this section. These references are being revised to correctly reference the renumbered MAPLHGR figures.

The last paragraph in this section discusses MAPLHGR requirements while in single recirculation loop operation (SLO). This paragraph is being revised to require consideration of the MAPLHGR reduction factors ($MAPFAC_f$ and $MAPFAC_p$), as well as the SLO MAPLHGR multiplier (0.85) when determining the MAPLHGR limit while in SLO. The MAPLHGR reduction factors must be considered because the justification for deletion of Technical Specification 3/4.2.2 was based in part on the conservatisms gained by use of these MAPLHGR reduction factors.

Technical Specification Bases 3/4.2.2: This section is being deleted because Technical Specification 3/4.2.2 was deleted as discussed previously.

Technical Specification Bases Table B3.2.1-1: This change incorporates additional information into this section of the Technical Specification Bases regarding assumptions used to determine MCPR for core flows less than 85%. This information is from the MEOD analysis.

Technical Specification Bases 3/4.2.3 (paragraph 5): This change defines the new control rod line to be used when determining values for $MCPR_f$. The change is required because the MEOD region allows control rod lines (and core flow rates) in excess of the current limits.

Technical Specification Bases 3/4.2.3: This change adds a discussion of how the $MCPR_p$ limits are determined when core power is less than 40% of RATED THERMAL POWER. The new $MCPR_p$ limits are flow dependent at core power below 40% of RATED THERMAL POWER. Below 40% of rated power, the end of cycle-recirculation pump trip and the turbine stop valve closure and turbine control valve fast closure scrams are bypassed. Because of the bypass there is a significant MCPR sensitivity to initial core flows. At high core flows (i.e., greater than 50% of rated) the $MCPR_p$ is increased in order to maintain the margin of safety.

Technical Specification Bases Figure B 3/4.2.3-1: This change revises the CPS operating map. The map is being revised to show the boundary of operation allowed by the MEOD.

Technical Specification Bases Figure B 3/4.2/3-2 (new): This change adds a new operating map for single recirculation loop operation, to clarify what operating regions are acceptable for single recirculation loop operation.

Technical Specification Table 4.3.1.1-1 (note d): This change removes reference to Technical Specification 3.2.2 which is being deleted.

Technical Specification Table 3.3.6-2 (APRM flow biased rod block:) The increase in the APRM flow-biased rod block setpoint is similar to that for the flow-biased simulated thermal power scram justified above. The high flow clamp is added to maintain the same clamp setpoint at rated power/flow conditions as is currently available. That is, the maximum setpoint currently available is 108% (at W=100%). In the MEOD, the new setpoint (which could be calculated to be as high as 128%) is still to be clamped at this value (108%).

Technical Specification Table 3.3.6-2 (Reactor Coolant System Recirculation Flow): The Reactor Coolant System Recirculation Flow - High rod block setpoint is increased from 108% to 113%. Operation in the increased core flow region of the operating map (i.e., with core flow up to 107% of rated) has been evaluated as discussed in the attached MEOD analysis. Raising the rod block setpoint will minimize unnecessary rod block alarms when operating in the increased core flow region. While the allowable flow range has been extended from 100% to 107%, the rod block setpoint has been conservatively raised by only 5%.

Technical Specification Table 3.3.6-2 (** note): This change removes reference to Technical Specification 3.2.2 which is being deleted.

Technical Specification 3.3.7.7: This change removes the requirement to monitor MFLPD. The only reason for monitoring MFLPD is to ensure that Technical Specification 3/4.2.2 can be met. Since Technical Specification 3/4.2.2 is being deleted as part of this submittal, MFLPD no longer needs to be monitored.

Technical Specification 3.4.1.1 ACTION a.1.c: This change revises the value for the MCPR Safety Limit to 1.08 as discussed in the justification for the similar change to Technical Specification 2.1.2.

Technical Specification 3.4.1.1 ACTION a.1.d: This change removes the requirement to multiply the MAPLHGR limits by 0.83, and replaces it with a reference to Technical Specification 3.2.1 which contains a requirement to multiply the MAPLHGR limit by the smallest of $MAPFAC_f$, $MAPFAC_p$ or 0.85.

Technical Specification 3.4.1.1 ACTION a.1.e: This change removes reference to Technical Specification 3.2.2 which is being deleted as discussed previously.

Technical Specification Bases 3/4.6.2.5: The drywell peak calculated pressure is being changed from 18.9 psig to 19.7 psig. For a reactor recirculation piping break at the most limiting condition in the MEOD and with a reduced feedwater temperature, the predicted peak drywell pressure increases slightly. This increase is predominately due to the increased mass flow rate out of the break. The increased mass flow rate results from increased density of the reactor coolant. The new peak predicted value is well within the design limit of 30 psig.

Basis For No Significant Hazards Consideration (for the Technical Specification Changes Proposed as a Result of Operation with New Fuel Types and in the MEOD Region)

According to 10CFR50.92, a proposed change to the license (Technical Specifications) involves no significant hazards consideration if operation of the facility in accordance with the proposed change would not (1) involve a significant increase in the probability or consequences of any accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in the margin of safety.

The changes proposed in this submittal will allow continued operation during cycle 2 at CPS, provide for operation in the Maximum Extended Operating Domain (MEOD) and allow for elimination of the APRM flow-biased simulated thermal power-high scram setpoint adjustment requirements. All of these proposed changes have been evaluated and found to be appropriate for operation at RATED THERMAL POWER with feedwater temperature as low as 370°F (and equivalent operation at lower thermal power).

These changes have been evaluated as discussed in the attached analyses. Based on these analyses, the proposed changes do not involve a significant hazards consideration. Details of the basis for this conclusion are provided below by addressing the three concerns outlined in 10CFR50.92.

1) The proposed changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

   The changes to support the MEOD satisfy this concern because plant equipment and systems will continue to operate within their design limits. Changes to support operation in the MEOD involve a revision of the $MCPR_f$ limit and higher limits for the APRM scram and rod block setpoints. The increased Operating Limit for MCPR is needed to maintain the same margin of safety in the increased core flow region that was established for the current operating domain with respect to the recirculation flow run out transient. The revised limit ensures the consequences of this event are not increased. The revised APRM setpoints maintain the same scram/rod block-to-power margin in the MEOD as is currently provided. These conclusions are based on an evaluation (see attached MEOD analysis) which considered the following:

   - A bounding BWR/6 LOCA analysis was performed for the MEOD. It was determined that current MAPLHGR and $MCPR_p$ limits and the revised $MCPR_f$ limits are adequate to ensure LOCA consequences are not increased.

   - The containment response for a design basis accident in the MEOD, considering a feedwater temperature reduction due to feedwater heater(s) out of service, is slightly more severe than the analysis provided in Final Safety Analysis Report (FSAR) Section 6.2. As presented in the MEOD analysis, the differential peak drywell pressure of 19.7 psig is 0.8 psi above the current CPS FSAR Chapter 6 value, but it is still well below the design limit of 30 psig.

- Fuel thermal and mechanical performance for transients initiated in the MEOD is bounded by the fuel design bases.

- The effects of acoustic, flow induced, and reactor internal pressure differential induced load and of increased flow on the fuel bundle and reactor internals were found to be well within allowable design limits.

The elimination of the APRM flow-biased simulated thermal power-high scram setpoint-adjustment requirement involves revised MCPR and new MAPLHGR reduction factors. These limits are imposed to ensure that margins to fuel integrity limits are equal to or larger than those currently in existence. The criteria by which these changes were judged include the following:

- The MCPR safety limit shall not be violated,

- Fuel performance shall remain within design and licensing bases, and

- PCT and maximum cladding oxidation fractions shall remain within regulatory limits.

Based on the above criteria, elimination of the APRM flow-biased simulated thermal power-high scram setpoint-adjustment requirement is judged to meet the first concern for significant hazards consideration.

Operation at RATED THERMAL POWER with feedwater temperatures as low as 370°F (and equivalent operation at lower thermal power) was evaluated as described in the attached MEOD analysis. The evaluation considered the Chapter 15 transient evaluations, the Chapter 6 LOCA evaluation, fuel mechanical limits and thermal-hydraulic stability and the effects of flow-induced and acoustic loads on the vessel internals. All results remain within design and regulatory limits. Rated power operation with feedwater temperatures down to 370°F therefore does not involve a significant hazards consideration.

The reload analysis allows operation with two new fuel types. MAPLHGR curves for these new fuels are being added to the Technical Specifications to ensure that ECCS peak clad temperature (PCT) and LHGR limits are not exceeded as established by applicable analyses. Analysis shows that the PCT for the two new fuel types (for the postulated DBA LOCA) is 2078°F which is well below the 10CFR50.46 limit of 2200°F. Therefore, this change does not increase the probability or consequences of any accident previously analyzed.

2) The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The MEOD analysis effectively provides for normal plant operation in an increased area of the power-flow operating map. While the events previously analyzed may be initiated from a new operating point, these events were addressed in item 1 above. There are no new or different accidents created by the MEOD related changes.

The elimination of the APRM flow-biased simulated thermal power-high scram setpoint-adjustment requirement itself involves no physical design changes. With the incorporation of the new MCPR and MAPLHGR limits, plant operation does not change. Therefore, no new or different accident is created by these changes.

Operation with reduced feedwater temperatures involves normal plant operating practice, and no new or different accidents are created in this mode of operation.

Operation under the provisions of the reload analysis does not change any mode of plant operation and therefore does not create the possibility of a new or different kind of accident.

3) The proposed change does not involve a significant reduction in the margin of safety.

The provisions for reloading the reactor and operation within the MEOD (including operation with reduced feedwater temperature and without the APRM setpoint adjustment) have been completely evaluated. Revised limits and setpoints have been established which maintain or increase the margin of safety provided by the current values. As noted above, the containment response to a DBA initiated from the MEOD with reduced feedwater temperatures resulted in a slightly higher drywell differential pressure than was determined in the original FSAR evaluation. However, this does not constitute a significant reduction in the margin of safety.