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

January 31, 1989

Docket No. 50-461

Mr. Dale L. Holtzscher
Acting Manager - Licensing and Safety
Clinton Power Station
P. O. Box 678
Mail Code V920
Clinton, Illinois 61727

Dear Mr. Holtzscher:

SUBJECT: TECHNICAL SPECIFICATION CHANGE REQUEST RELATED TO THE FIRST REFUELING OF THE REACTOR WITH NEW FUEL TYPES AND TO SUPPORT SUBSEQUENT REACTOR OPERATION (CYCLE 2) IN THE MAXIMUM EXTENDED OPERATING DOMAIN (MEOD) AND WITH REDUCED FEEDWATER TEMPERATURE (TAC NO. 69308)

Re: CLINTON POWER STATION, UNIT NO. 1

The Commission has issued the enclosed Amendment No. 18 to Facility Operating License No. NPF-62 for the Clinton Power Station, Unit No. 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 6, 1988 and supplemented December 22, 1988.

The amendment consists of proposed changes to the Technical Specifications (TS) related to four issues. The first proposed change would allow the Clinton Power Station (CPS) to perform its first reactor refueling, in which new types of reactor fuel will be utilized, and to proceed with subsequent reactor operation with the reloaded core. The second proposed change would permit CPS operation in the maximum extended operating domain (MEOD) with (a) up to a 50°F reduction in feedwater temperature and (b) elimination of APRM setdown. The third proposed change would revise the Remote Shutdown Systems Controls to include additional control switches for valves 1E12-F068B and 1E12-F014B and circuit breaker 252-AT1AA1. The fourth proposed change consists of several changes to the surveillance requirements for the jet pumps.

c/P-1

DF01
1/1

A copy of our Safety Evaluation is also enclosed. Notice of Issuance is being filed with the Office of the Federal Register for publication.

Sincerely,

John B. Hickman, Project Manager
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. 18 to
   License No. NPF-62
2. Safety Evaluation

cc w/enclosures:
See next page

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A copy of our Safety Evaluation is also enclosed. Notice of Issuance is being filed with the Office of the Federal Register for publication.

Sincerely,

/S/

John B. Hickman, Project Manager
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. 18 to
   License No. NPF-62
2. Safety Evaluation

cc w/enclosures:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ILLINOIS POWER COMPANY, ET AL.

DOCKET NO. 50-461

CLINTON POWER STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 18
License No. NPF-62

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

   A. The application for amendment by Illinois Power Company* (IP), Soyland Power Cooperative, Inc., and Western Illinois Power Cooperative, Inc. (the licensees) dated September 6, 1988 and supplemented December 22, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-62 is hereby amended to read as follows:

*Illinois Power Company is authorized to act as agent for Soyland Power Cooperative, Inc. and Western Illinois Power Cooperative, Inc. and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 18, are hereby incorporated into this license. Illinois Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 31, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 18

FACILITY OPERATING LICENSE NO. NPF-62

DOCKET NO. 50-461

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

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i	i
iii	iii
iv	iv
v	v
xvi	xvi
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1-3	1-3
1-4	1-4
2-1	2-1
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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

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

DOSE EQUIVALENT I-131

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

DRYWELL INTEGRITY

1.11 DRYWELL INTEGRITY shall exist when:

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

   1. Capable of being closed by an OPERABLE 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.

1.15 [DELETED]

1.16 [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.

1.23 [DELETED]

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 1.07 with two recirculation loop operation and shall not be less than 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 1.07 with two recirculation loop operation or less than 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. Flow Biased	≤ 0.66 (W)+64%,[a] with a maximum of	≤ 0.66 (W)+67%,[a] with a maximum of
b. High Flow Clamped	≤ 111.0% of RATED THERMAL POWER	≤ 113.0% of RATED THERMAL POWER
2) During single recirculation loop operation:		
a. Flow Biased	≤ 0.66(W-ΔW)+48%[a]	≤ 0.66(W-ΔW)+51%[a]
b. High Flow Clamped	Not Required OPERABLE	Not Required OPERABLE
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**
8. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig

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 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 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 x 10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full-scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

2.1.2 THERMAL POWER, High Pressure and High Flow

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

The Safety Limit MCPR is determined using 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.

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 system is not endangered. The reactor pressure vessel is designed to Section III of the ASME Boiler and Pressure Vessel Code 1971 Edition, including Addenda through Summer 1973, which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The Safety Limit of 1325 psig, as measured by the reactor vessel steam dome pressure indicator, is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The reactor coolant system is designed to the ASME Boiler and Pressure Vessel Code, 1974 Edition, including Addenda through the Summer of 1974, for the reactor recirculation piping which permits a maximum pressure transient of 110% of design pressures of 1250 psig for suction piping and 1650 psig for discharge piping from the recirculation pump discharge to the outlet side of the discharge shutoff valve and 1550 psig from the discharge shutoff valve to the jet pumps. The pressure Safety Limit is selected to be the lowest transient overpressure allowed by the applicable codes.

Reference

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

BASES TABLE B 2.1.2-1

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BASES TABLE B 2.1.2-2

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2.1.4 REACTOR VESSEL WATER LEVEL

With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level became less than two-thirds of the core height. The Safety Limit has been established at the top of the active irradiated fuel to provide a point which can be monitored and also provide adequate margin for effective action.

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. In these flow biased equations, the variable W is the loop recirculation flow as a percentage of the total loop recirculation flow which produces a rated core flow of 84.5 million lbs/hr.

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.

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.

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

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.

*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 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.

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 3.2.1-3 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:

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.

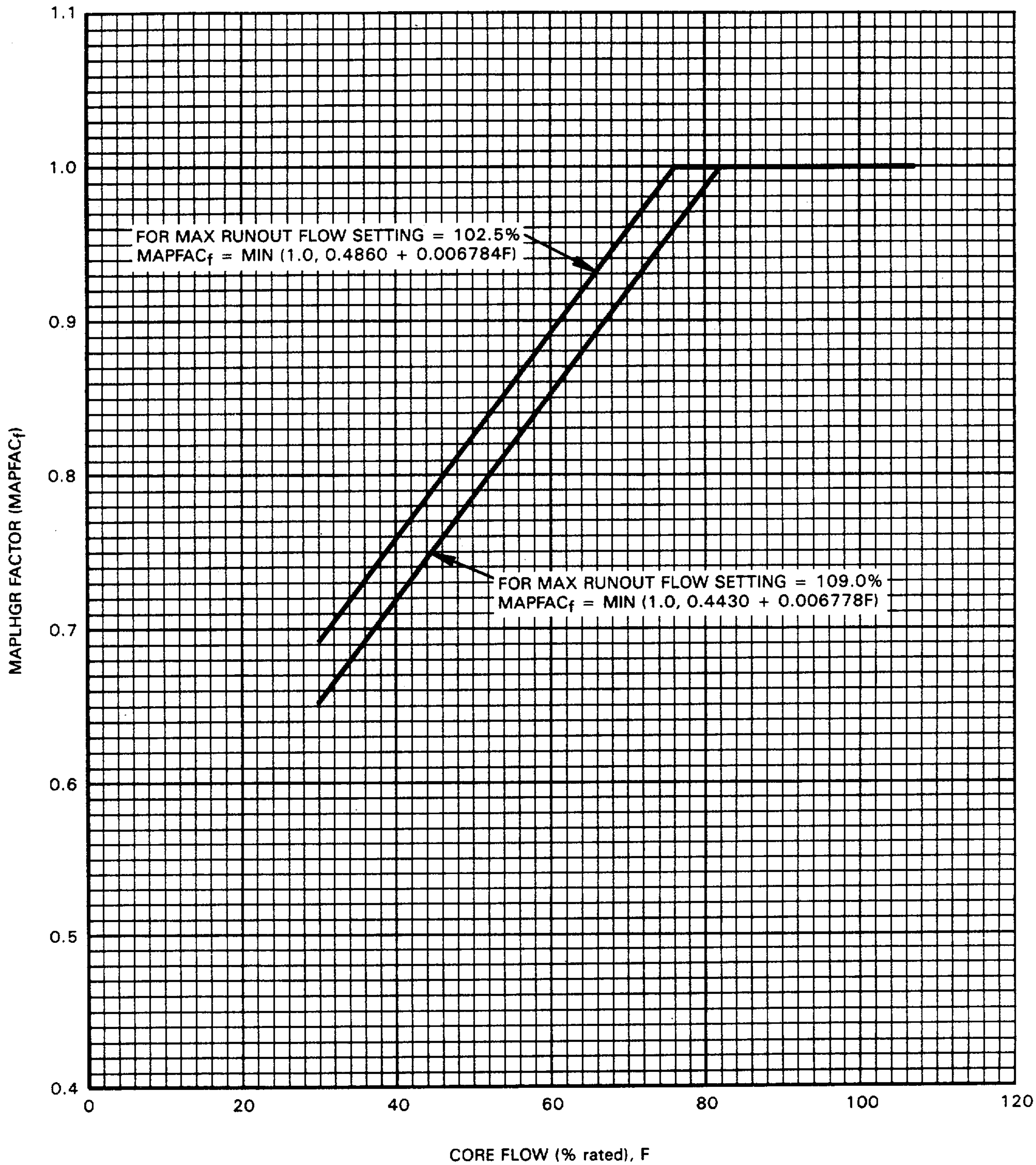

Figure 3.2.1-1 Flow-Dependent MAPLHGR Factors ($MAPFAC_f$)

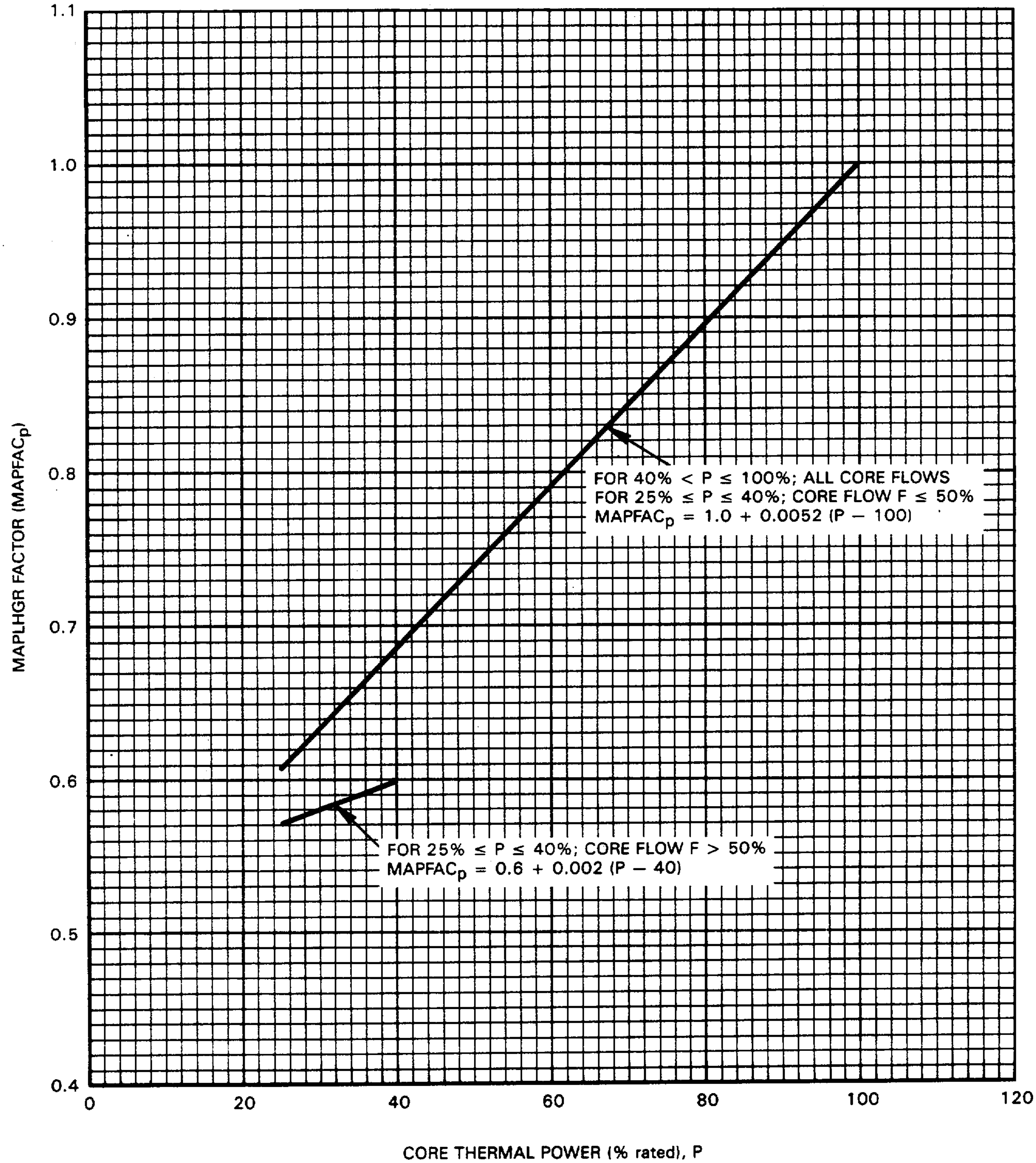

Figure 3.2.1-2 Power-Dependent MAPLHGR Factors ($MAPFAC_p$)

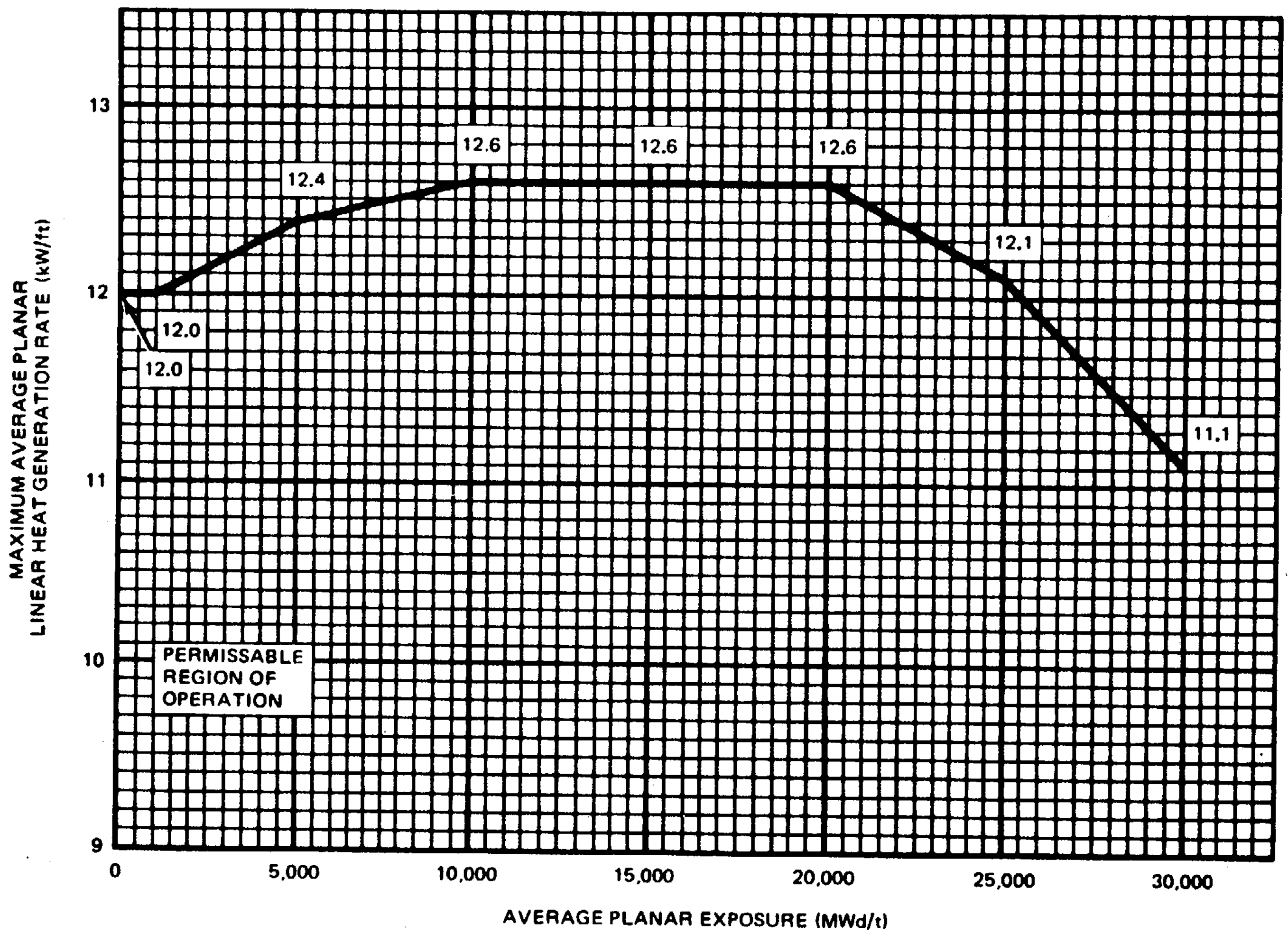

Figure 3.2.1-3 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Initial Core Fuel Types - High Enrichment

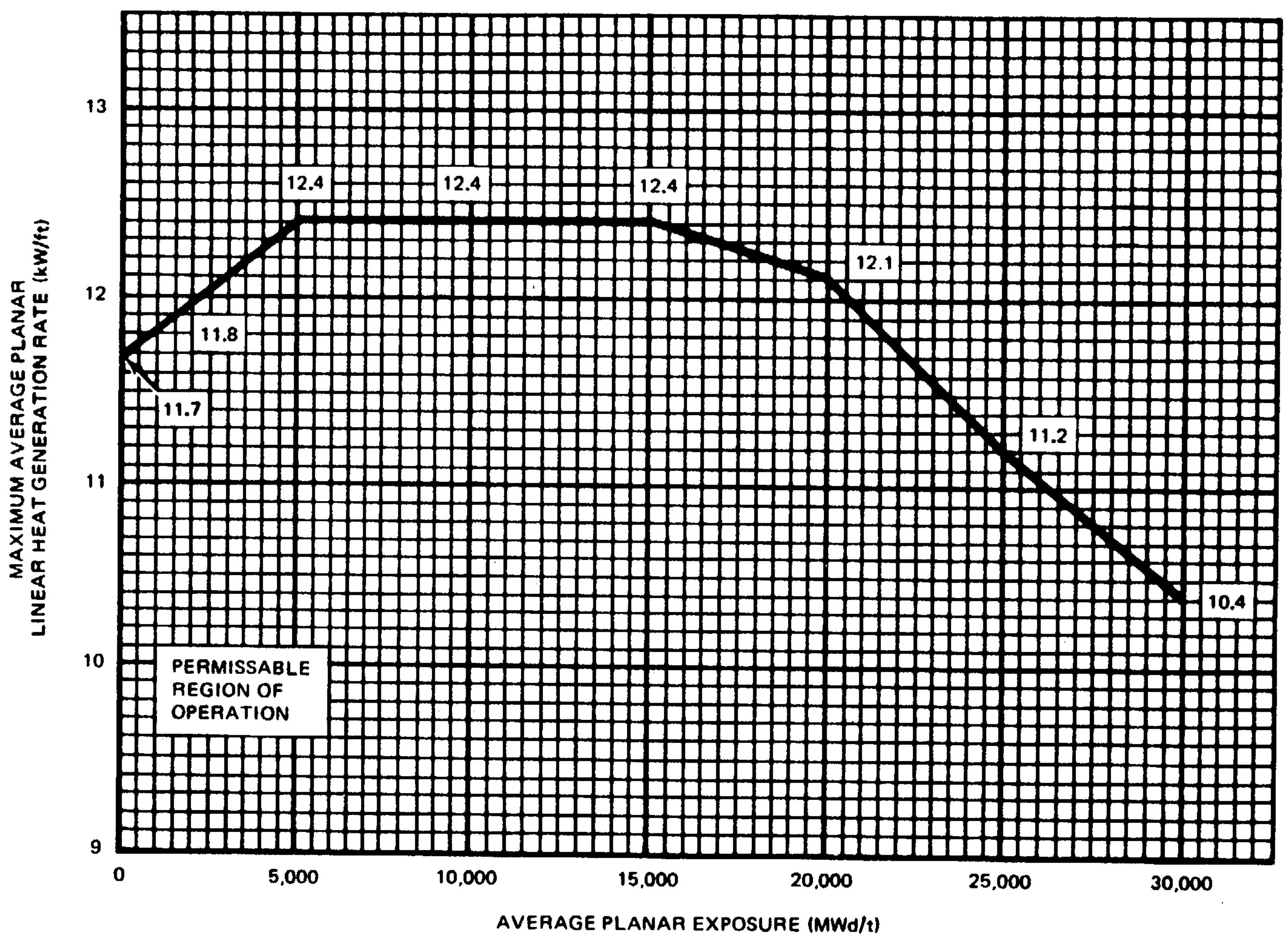

Figure 3.2.1-4 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Initial Core Fuel Types - Medium Enrichment

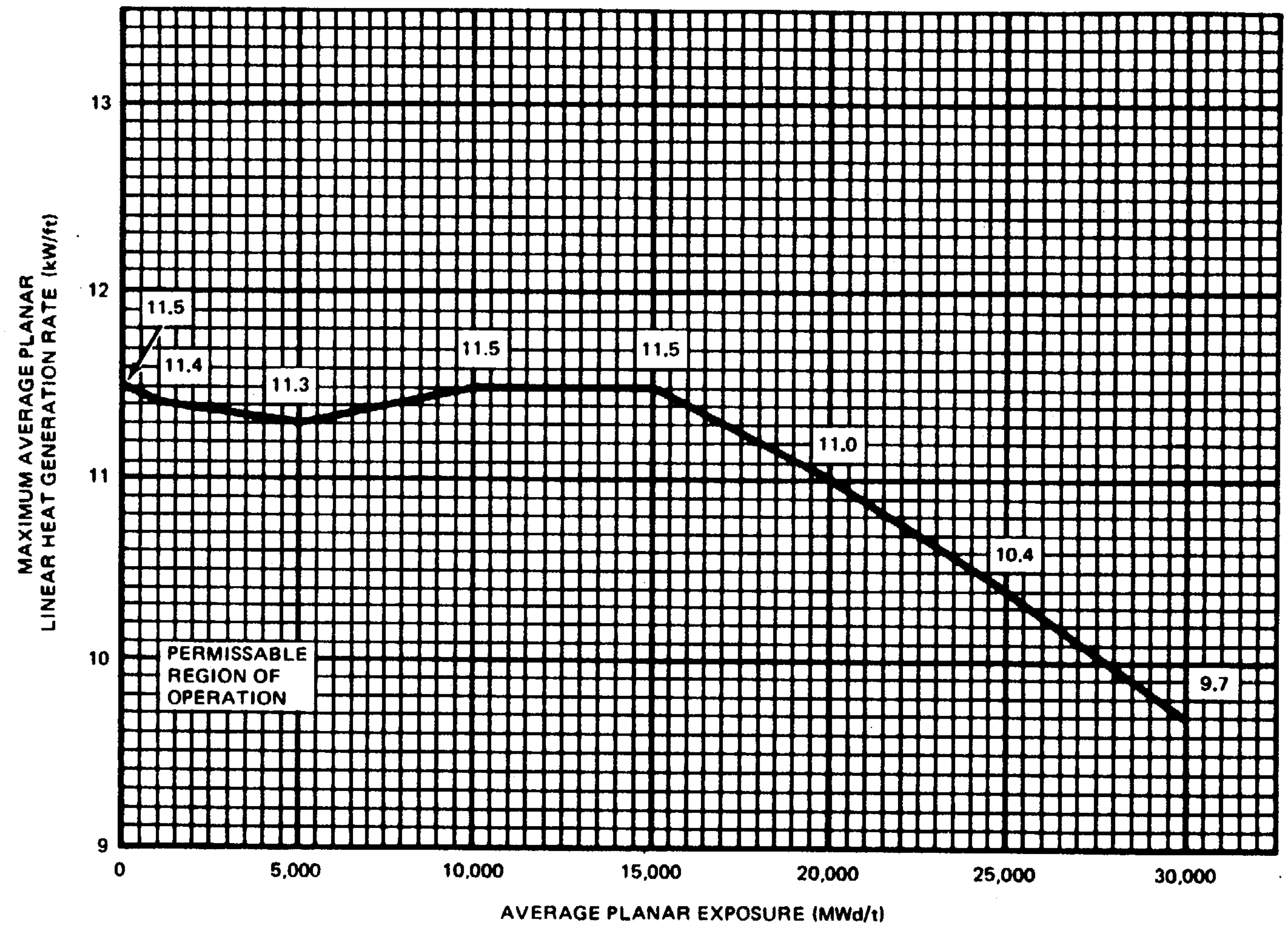

Figure 3.2.1-5 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Initial Core Fuel Types - Natural Enrichment

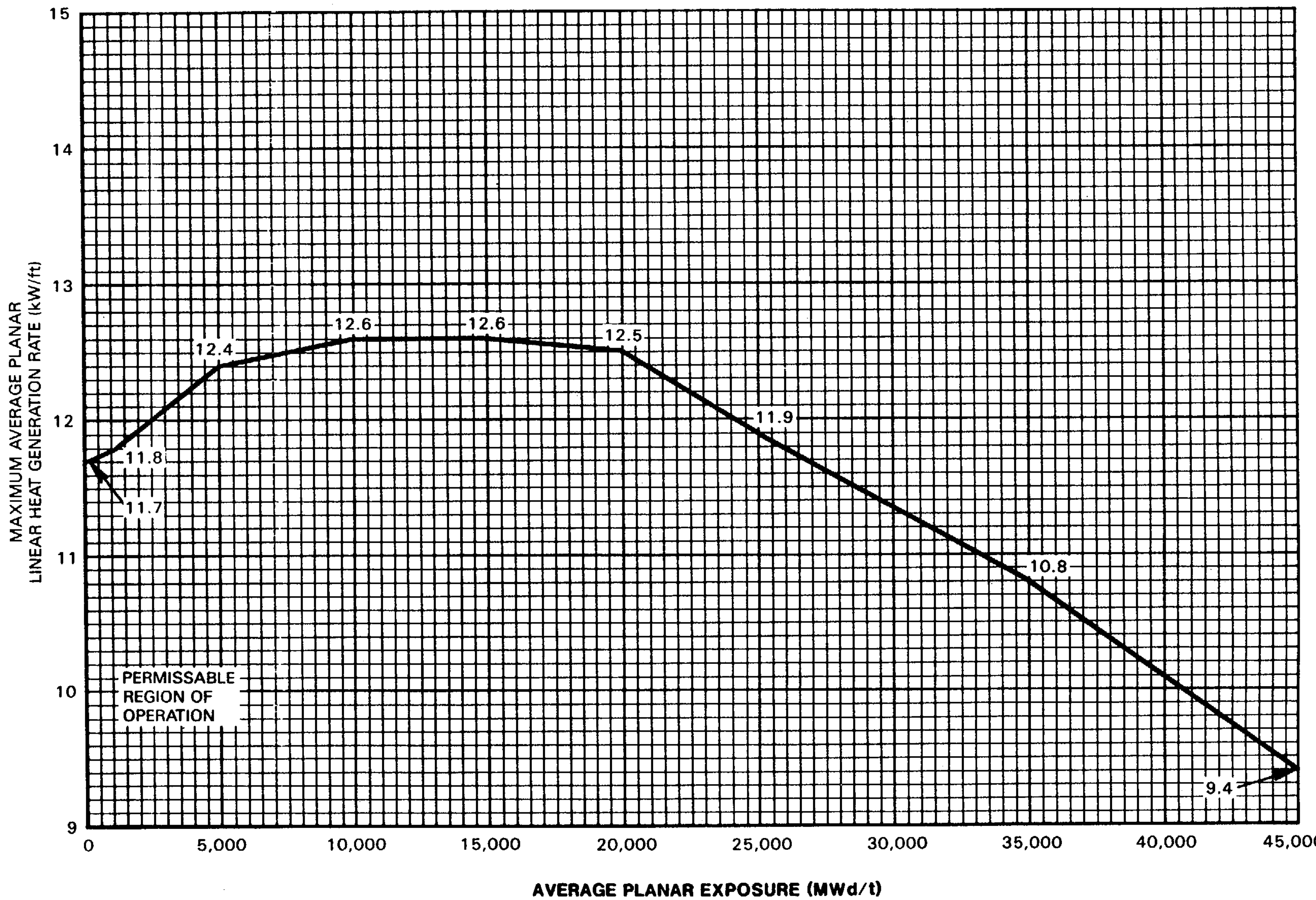

Figure 3.2.1-6 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure - Reload 1 Fuel Type BP8SRB284L

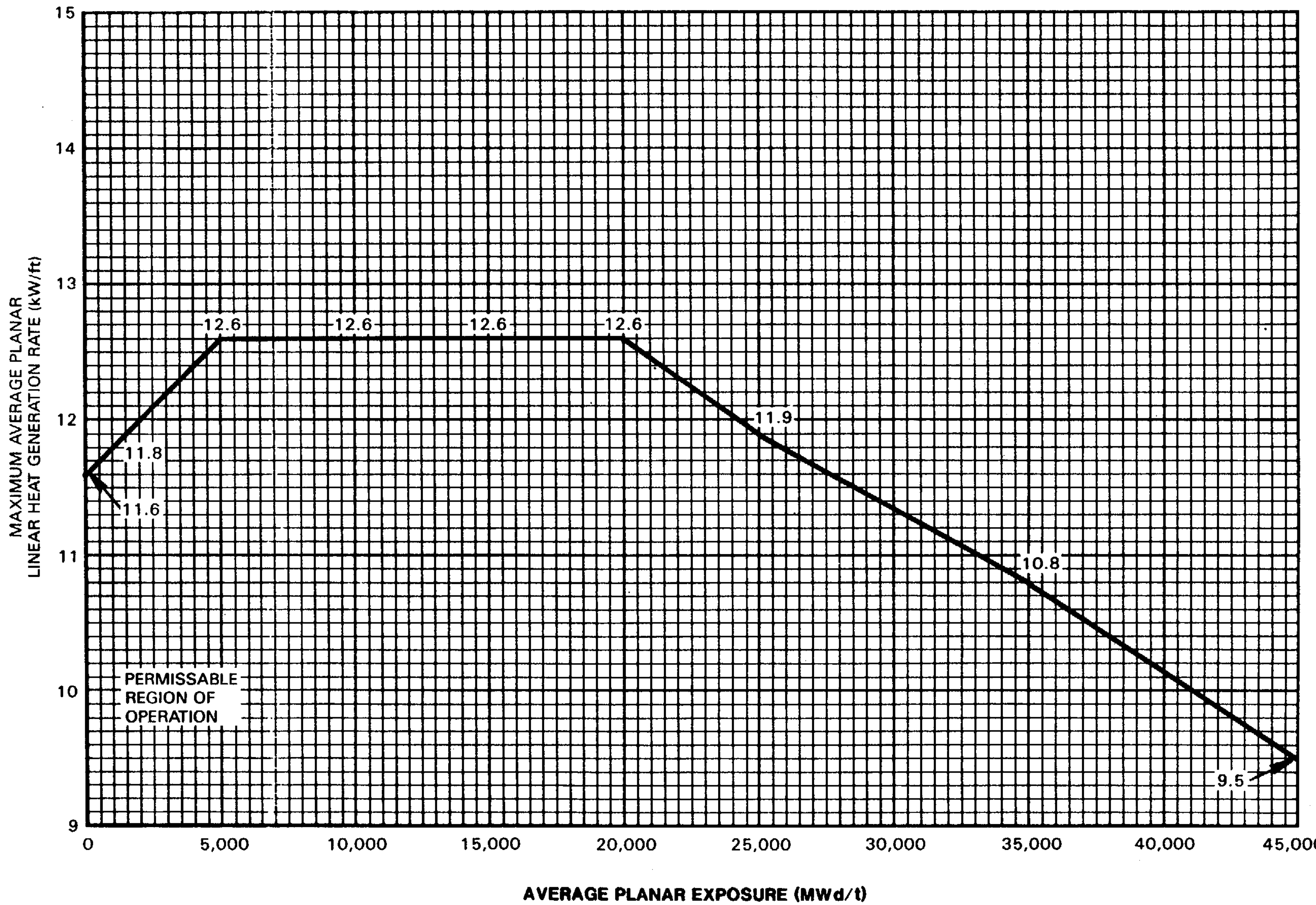

Figure 3.2.1-7 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure - Reload 1 Fuel Type BP8SRB284LC

3/4.2.2 APRM SETPOINTS [DELETED]

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APRM SETPOINTS [DELETED]

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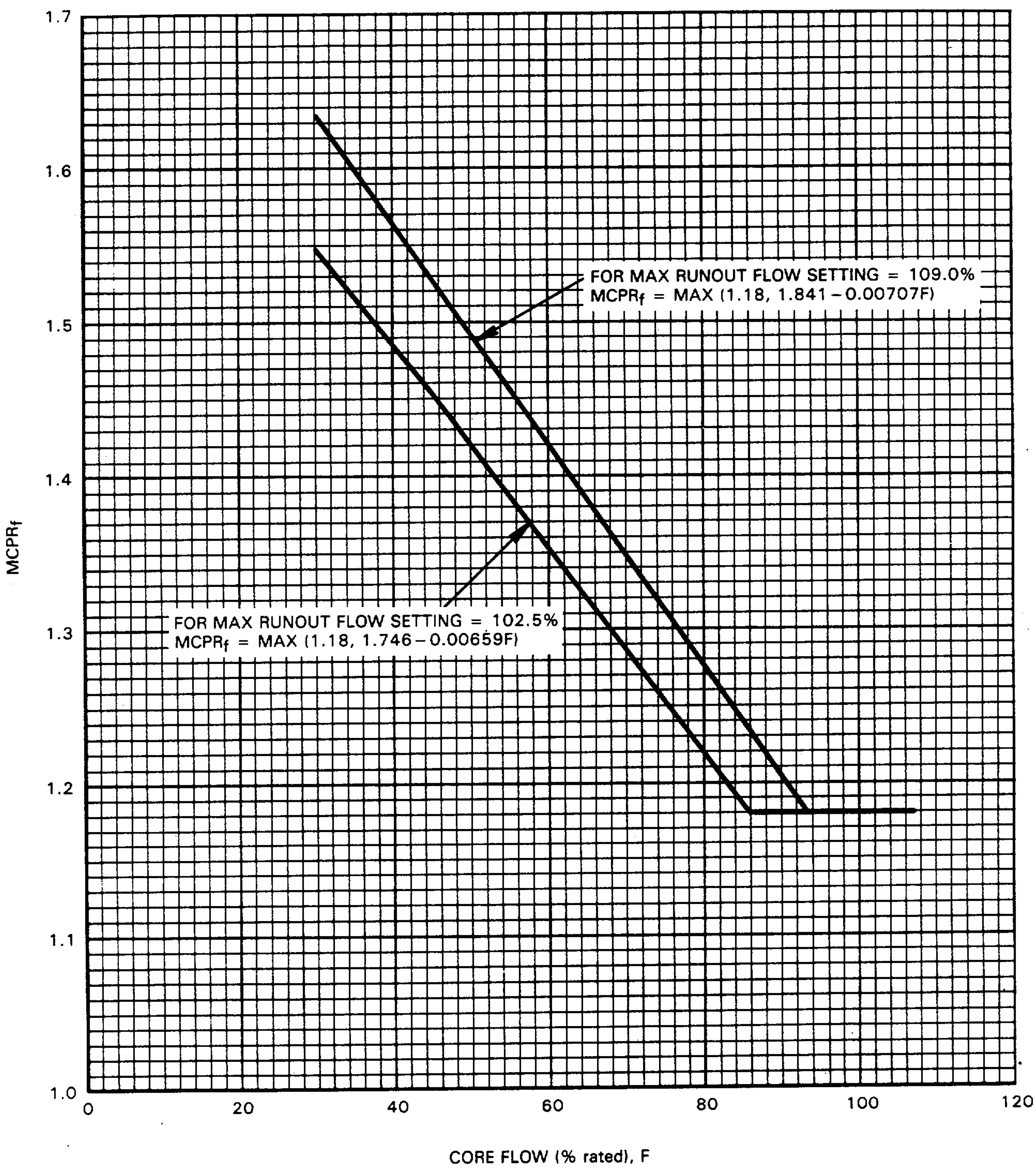


Figure 3.2.3-1 Clinton $MCPR_f$ Versus Core Flow

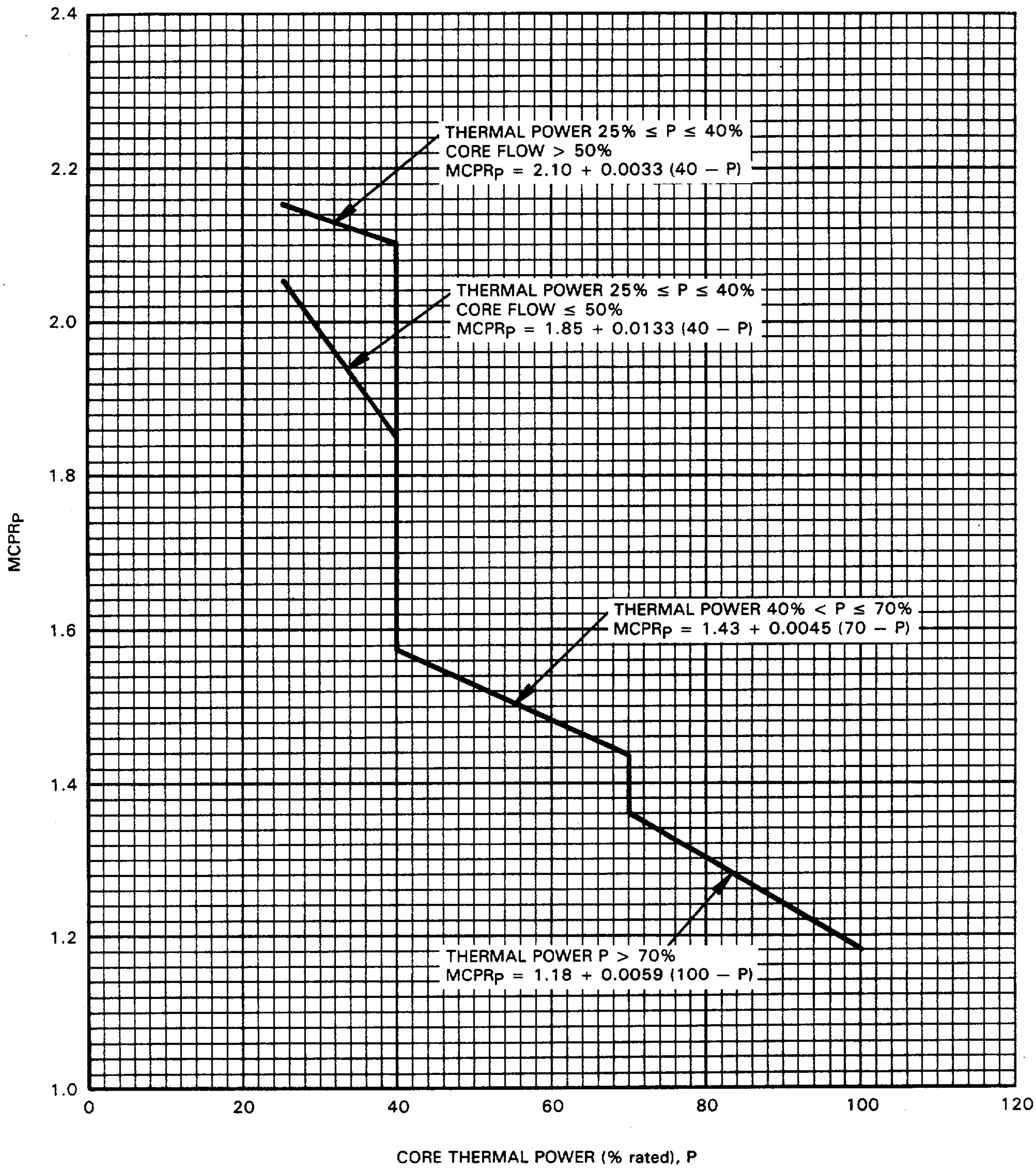


Figure 3.2.3-2 Clinton $MCPR_p$ Versus Power for $\Delta T \leq 50^oF$ and Core Flow $\leq$ 107%

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 $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER.

(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		
1) During two recirculation loop operation:		
a) Flow Biased	$\leq$ 0.66W + 58%** with a maximum of	$\leq$ 0.66W + 61%** with a maximum of
b) High Flow Clamped	$\leq$ 108.0% of RATED THERMAL POWER	$\leq$ 110.0% of RATED THERMAL POWER
2) During single recirculation loop operation:		
a) Flow Biased	$\leq$ 0.66(W-ΔW) + 42%**	$\leq$ 0.66(W-ΔW) + 45%**
b) High Flow Clamped	Not required OPERABLE	Not required OPERABLE
b. Inoperative	NA	NA
c. Downscale	$\geq$ 5% of RATED THERMAL POWER	$\geq$ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	$\leq$ 12% of RATED THERMAL POWER	$\leq$ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq 1 \times 10^5$ cps	$\leq 1.6 \times 10^5$ cps
c. Inoperative	NA	NA
d. Downscale	$\geq$ 3 cps	$\geq$ 1.8 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ 108/125 division of full scale	$\leq$ 110/125 division of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ 5/125 division of full scale	$\geq$ 3/125 division of full scale

TABLE 3.3.6-2 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

5. SCRAM DISCHARGE VOLUME		
a. Water Level-High, C11-N602A	$\leq$ 12" #	$\leq$ 19 7/8" #
b. Water Level-High, C11-N602B	$\leq$ 12" ##	$\leq$ 19 7/8" ##
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	$\leq$ 113% of rated flow	$\leq$ 116% of rated flow
7. REACTOR MODE SWITCH		
a. Shutdown Mode	NA	NA
b. Refuel Mode	NA	NA

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 note (a) of Table 2.2.1-1.

\# Instrument zero is 758' 5" msl.

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

TABLE 3.3.7.4-2

REMOTE SHUTDOWN SYSTEM CONTROLS

CONTROL	EQUIPMENT NUMBER	MINIMUM CHANNELS OPERABLE DIVISION I	DIVISION II
1. RHR Pmp	1E12-C002A/B	1	1
2. RHR Supp. Pool Suction Vlv	1E12-F004A/B	1	1
3. RHR Shutdown Cooling Supply Vlv	1E12-F006A	1	NA
4. RHR Shutdown Cooling Sply Otbd Isol Vlv	1E12-F008	1	NA
5. RHR HX Bypass Vlv	1E12-F048A/B	1	1
6. RHR Test Line Vlv to Supp. Pool	1E12-F024A/B	1	1
7. RHR HX Dsch Vlv	1E12-F003A/B	1	1
8. RCIC Steam Inlet Vlv to RHR HX	1E12-F052A/B	1	1
9. RHR HX Inlet Vlv	1E12-F047A/B	1	1
10. RHR HX SX Outlet Vlv	1E12-F068A/B	1	1
11. RHR Shutdown Cooling Return Vlv	1E12-F053A/B	1	1
12. RHR RPV Inboard Inject Vlv	1E12-F042A/B	1	1
13. RHR RPV Outboard Inject Vlv	1E12-F027A	1	NA
14. RHR Cnmt Spray Vlv	1E12-F028A	1	NA
15. RHR Supp Pool Cool Vlv	1E12-F011A	1	NA
16. RHR HX 1A RCIC Shutoff Vlv	1E12-F026A	1	NA
17. RHR FP/FC Sply Vlv	1E12-F037A	1	NA
18. RHR Pump Min Flow Recirc Vlv	1E12-F064A/B	1	1
19. RHR HX 1A SX Bypass Vlv	1SX173A	1	NA
20. RHR RR Sply Inbd Isol Vlv	1E12-F006B	NA	1
21. Shutdown Cooling Inbd Isol Vlv	1E12-F009	NA	1
22. RPV Head Spray Vlv	1E12-F023	NA	1
23. RCIC Stm Byps Vlv	1E51-F095	1	NA
24. RCIC Pump Cond Stg Tnk Suction Vlv	1E51-F010	1	NA
25. RCIC Supp Pool Suction Vlv	1E51-F031	1	NA
26. RCIC First Test Line Isol. Vlv to RCIC Storage Tank	1E51-F022	1	NA
27. RCIC Inject Vlv	1E51-F013	1	NA
28. RCIC Min Flow Recrc Vlv	1E51-F019	1	NA
29. RCIC Second Test Line Isol Vlv to RCIC Stg Tnk	1E51-F059	1	NA
30. RCIC Turbine L.O. Cool Wtr Sply Vlv	1E51-F046	1	NA
31. RCIC Gland Seal Air Cmpsr	1E51-C002F	1	NA
32. RCIC Outbd Vac Bkr Vlv	1E51-F077	1	NA
33. RHR RCIC Stm Sply Otbd Isol Vlv	1E51-F064	1	NA
34. RCIC Turb Stm Sply Vlv	1E51-F045	1	NA
35. RCIC Turb Xhst Stop Vlv	1E51-F068	1	NA
36. RCIC Trip/Throttle Vlv	1E51-C002E	1	NA
37. RCIC Turb Stm Supply Warm-up Vlv	1E51-F076	NA	1
38. SRV 51C	1B21-F051C	1	1
39. SRV 51D	1B21-F051D	1	1
40. SRV 51G	1B21-F051G	1	1
41. RCIC Stm Flow Cntrl	NA	1	NA
42. RCIC Turb Trip	NA	1	NA
43. DG 1A Vent Fan	1VD01CA	1	NA

TABLE 3.3.7.4-2 (Continued)

REMOTE SHUTDOWN SYSTEM CONTROLS

CONTROL	EQUIPMENT NUMBER	MINIMUM CHANNELS OPERABLE DIVISION I	MINIMUM CHANNELS OPERABLE DIVISION II
44. DG 1A Oil Rm A Xhst Fan	1VD02CA	1	NA
45. Div I Switchgear Heat Removal Vent Fan	1VX03CA	1	NA
46. Battery Rm 1A1 Xhst Fan	1VX05CA	1	NA
47. SX Pmp Rm Sply Fan	1VH01CA/B	1	1
48. RHR Pmp Rm 1A Sply Fan	1VY02C	1	NA
49. RHR Ht Xchg Rm A Sply Fan	1VY03C	1	NA
50. RCIC Pmp Rm Sply Fan	1VY04C	1	NA
51. DG 1A Ckt Brkr	252-DGKA	1	NA
52. DG 1A Fuel Oil Trnsfr Pmp	1D001PA	1	NA
53. SX Pmp	1SX01PA/B	1	1
54. SX/WS Isol Vlv	1SX014A/B	1	1
55. DG 1A Outlet Vlv	1SX063A	1	NA
56. SX 1A Strnr Inlet Vlv	1SX003A	1	NA
57. SX 1A Strnr Outlet Vlv	1SX004A		
58. SX 1A Strnr Bypass Vlv	1SX008A		
59. SX Xtie Vlv	1SX011A	1	NA
60. RHR Ht Xchg 1A Demin Wtr Sply Vlv	1SX082A	1	NA
61. Fuel Pool Ht Xchg 1A SX Sply Vlv	1SX012A	1	NA
62. Fuel Pool Ht Xchg 1A SX Dsch Vlv	1SX062A	1	NA
63. Fuel Pool M-U SX Sply Vlv	1SX016A	1	NA
64. SX-SGTS Charcoal Bed Train A Deluge Vlv	1SX073A	1	NA
65. Cntl Rm HVAC Recirc Unit A Deluge Vlv	1SX076A	1	NA
66. Cntl Rm HVAC M/U Unit A Deluge Vlv	1SX107A	1	NA
67. RHR HX Clg Wtr Sply Vlv	1E12-F014A/B	1	1
68. RCIC Inbd Vac Bkr Vlv	1E51-F078	NA	1
69. RCIC Stm Sply Inbd Isol Vlv	1E51-F063	NA	1
70. Remote Transfer Switch	1C61-HS501	NA	NA
71. Remote Transfer Switch	1C61-HS502	NA	NA
72. Remote Transfer Switch	1C61-HS508	NA	NA
73. Remote Transfer Switch	1C61-HS509	NA	NA
74. Remote Transfer Switch	1C61-HS510	NA	NA
75. Remote Transfer Switch	1C61-HS511	NA	NA
76. Remote Transfer Switch	1C61-HS527	NA	NA
77. Remote Transfer Switch	1C61-HS001	NA	NA
78. Remote Transfer Switch	1C61-HS002	NA	NA
79. Remote Transfer Switch	1C61-HS003	NA	NA
80. Remote Transfer Switch	1C61-HS004	NA	NA
81. Remote Transfer Switch	1C61-HS005	NA	NA
82. Remote Transfer Switch	1C61-HS006	NA	NA
83. Remote Transfer Switch	1C61-HS007	NA	NA
84. Remote Transfer Switch	1C61-HS008	NA	NA
85. Remote Transfer Switch	1C61-HS009	NA	NA
86. Remote Transfer Switch	1C61-HS010	NA	NA
87. Remote Transfer Switch	1C61-HS011	NA	NA
88. Remote Transfer Switch	1C61-HS012	NA	NA
89. Circuit Breaker 252-AT1AA1	1C61-HS565	1	NA

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.*

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 1.08 per Specification 2.1.2, and

   d) Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) 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 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.

REACTOR COOLANT SYSTEM

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.2 All jet pumps shall be demonstrated OPERABLE as follows:

Each of the above required jet pumps in an operating loop shall be demonstrated OPERABLE at least once per 24 hours when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER# by determining recirculation loop flow, total core flow and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur:

a. The indicated recirculation loop flow differs by more than 10% from the established flow control valve position-loop flow characteristics.

b. The indicated total core flow differs by more than 10% from the established total core flow value derived from recirculation loop flow measurements.

c. The indicated jet pump diffuser-to-lower plenum differential pressure (or jet pump flow) of any individual jet pump differs from established patterns by more than 20% (10% for flow).

*To be determined during the startup test program.
#The provisions of Specification 4.0.4 are not applicable.

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

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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 the MAPLHGR.

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.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-3 through 3.2.1-7 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.

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.

For plant operation with a single recirculation loop, the MAPLHGR limits of Figures 3.2.1-3 through 3.2.1-7 are multiplied by the smallest of $MAPFAC_f$, $MAPFAC_p$ or 0.85 (Reference 2). 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.

3/4.2.2 APRM SETPOINTS [DELETED]

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

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 in Specification 2.1.2, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The power-flow maps of Figures B 3/4.2.3-1 or B 3/4.2.3-2 give operational limits for double or single recirculation loop operation, respectively.

The evaluation of a given transient begins with the system initial parameters identified in Reference 3 that are input to a GE-core dynamic behavior transient computer program. The codes used to evaluate pressurization and non-pressurization events are described in Reference 3. 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 most limiting power flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. 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. Using this relative bundle power, the MCPRs were calculated at different points along the most limiting power flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined as $MCPR_f$.

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. 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.

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.

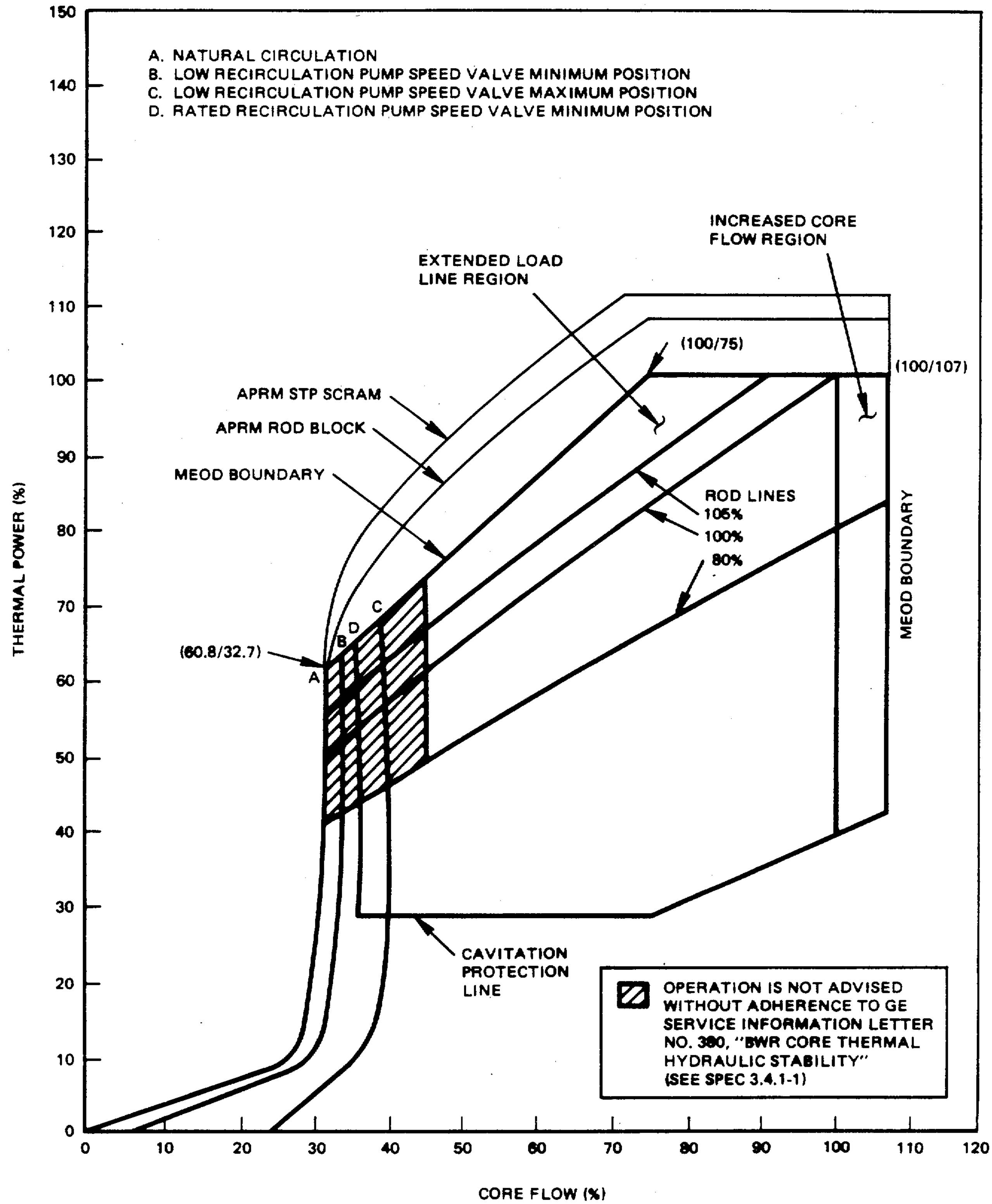
The daily requirement for calculating LHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate LHGR within 12 hours after the 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:

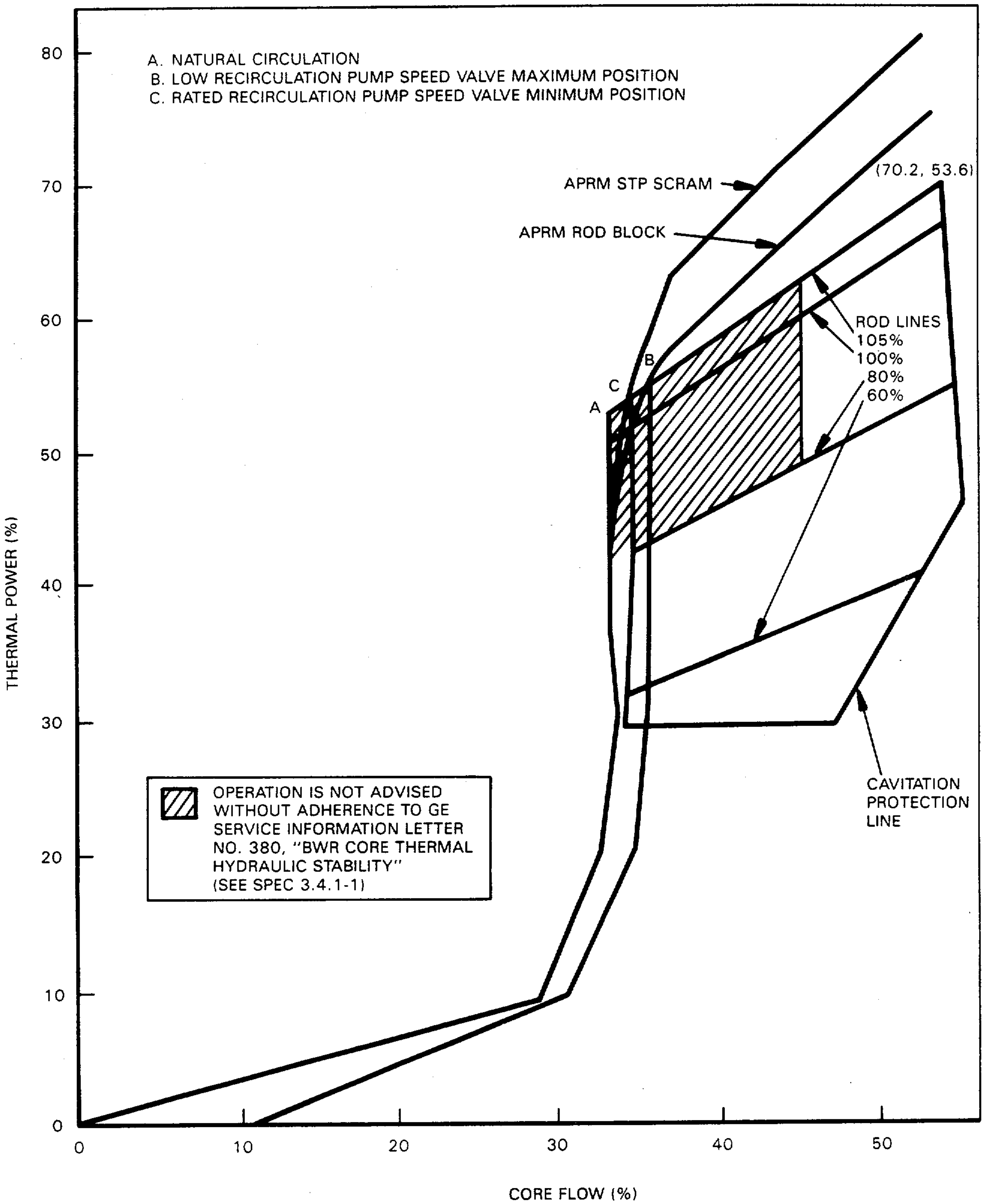
1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, November 1975.

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 for Two Recirculation Loop Operation



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

3/4.4.1 RECIRCULATION SYSTEM

The impact of single recirculation loop operation upon plant safety is assessed and shows that single-loop operation is permitted if the MCPR fuel cladding safety limit is increased as noted by Specification 2.1.2, APRM scram and control rod block setpoints are adjusted as noted in Tables 2.2.1-1 and 3.3.6-2, respectively, MAPLHGR limits are decreased by the factor given in Specification 3.2.1, and MCPR operating limits are adjusted per Section 3/4.2.3.

Additionally, surveillance on the volumetric flow rate of the operating recirculation loop is imposed to exclude the possibility of excessive core internals vibration. The surveillance on differential temperatures below (30%)* THERMAL POWER or (50%)* rated recirculation loop flow is to mitigate the undue thermal stress on vessel nozzles, recirculation pump, and vessel bottom head during the extended operation of the single recirculation loop mode.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation. Significant degradation is indicated if more than one of three specified surveillances performed confirms unacceptable deviations from established patterns or relationships. The surveillances, including the associated acceptance criteria, are in accordance with General Electric Service Information Letter No. 330, the recommendations of which are considered acceptable for verifying jet pump operability according to NUREG/CR-3052, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure." Performance of the specified surveillances, however, is not required when thermal power is less than 25% RATED THERMAL POWER because flow oscillations and jet noise precludes the collection of repeatable meaningful data during low flow conditions approaching the threshold response of the associated flow instrumentation.

Recirculation loop flow mismatch limits are in compliance with ECCS LOCA analysis design criteria for two recirculation loop operation. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. In the case where the mismatch limits cannot be maintained during two loop operation, continued operation is permitted in a single recirculation loop mode.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Sudden equilization of a temperature difference > 100°F between the reactor vessel bottom head coolant and the coolant in the upper region of the reactor vessel by increasing core flow rate would cause undue stress in the reactor vessel bottom head.

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*Initial Values. Final values to be determined during Startup Testing based on the threshold THERMAL POWER and recirculation loop flow which will sweep the cold water from the vessel bottom head preventing stratification.

3/4.4.1 RECIRCULATION SYSTEM (Continued)

The objective of GE BWR plant and fuel design is to provide stable operation with margin over the normal operating domain. However, at the high power/low flow corner of the operating domain, a small probability of neutron flux limit cycle oscillations exists depending on combinations of operating conditions (e.g., rod pattern, power shape). To provide assurance that neutron flux limit cycle oscillations are detected and suppressed, APRM and LPRM neutron flux noise levels should be monitored while operating in this region.

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservative decay ratio of 0.6 was chosen as the bases for determining the generic region for surveillance to account for the plant to plant variability of decay ratio with core and fuel designs. This generic region has been determined to correspond to a core flow of less than or equal to 45% of rated core flow and a THERMAL POWER greater than that specified in Figure 3.4.1.1-1.

Plant specific calculations can be performed to determine an applicable region for monitoring neutron flux noise levels. In this case the degree of conservatism can be reduced since plant to plant variability would be eliminated. In this case, adequate margin will be assured by monitoring the region which has a decay ratio greater than or equal to 0.8.

Neutron flux noise limits are also established to ensure early detection of limit cycle neutron flux oscillations. BWR cores typically operate with neutron flux noise caused by random boiling and flow noise. Typical neutron flux noise levels of 1-12% or rated power (peak-to-peak) have been reported for the range of low to high recirculation loop flow during both single and dual recirculation loop operation. Neutron flux noise levels which significantly bound these values are considered in the thermal/mechanical design of GE BWR fuel and are found to be of negligible consequence. In addition, stability tests at operating BWRs have demonstrated that when stability related neutron flux limit cycle oscillations occur they result in peak-to-peak neutron flux limit cycles of 5-10 times the typical values. Therefore, actions taken to reduce neutron flux noise levels exceeding three (3) times the typical value are sufficient to ensure early detection of limit cycle neutron flux oscillations.

Typically, neutron flux noise levels show a gradual increase in absolute magnitude as core flow is increased (constant control rod pattern) with two reactor recirculation loops in operation. Therefore, the baseline neutron flux noise level obtained at a specific core flow can be applied over a range of core flows. To maintain a reasonable variation between the low flow and high flow end of the flow range, the range over which a specific baseline is applied should not exceed 20% of rated core flow with two recirculation loops in operation. Data from tests and operating plants indicate that a range of 20% of rated core flow will result in approximately a 50% increase in neutron flux noise level during operation with two recirculation loops. Baseline data should be taken near the maximum rod line at which the majority of operation will occur. However, baseline data taken at low rod lines (i.e. lower power) will result in a conservative value since the neutron flux noise level is proportional to the power level at a given core flow.

3/4.4.1 RECIRCULATION SYSTEM (Continued)

The recirculation flow control valves provide regulation of individual recirculation loop drive flows; which, in turn, will vary the flow rate of coolant through the reactor core over a range consistent with the rod pattern and recirculation pump speed. The recirculation flow control system consists of the electronic and hydraulic components necessary for the positioning of the two hydraulically actuated flow control valves. Solid state control logic will generate a flow control valve "motion inhibit" signal in response to any one of several hydraulic power unit or analog control circuit failure signals. The "motion inhibit" signal causes hydraulic power unit shutdown and hydraulic isolation such that the flow control valve fails "as is." This design feature insures that the flow control valves do not respond to potentially erroneous control signals.

Electronic limiters exist in the position control loop of each flow control valve to limit the flow control valve stroking rate to 10±1% per second in opening and closing directions on a control signal failure. The analysis of the recirculation flow control failures on increasing and decreasing flow are presented in Sections 15.3 and 15.4 of the FSAR respectively.

The required surveillance interval is adequate to ensure that the flow control valves remain OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves (SRV) operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1375 psig in accordance with the ASME Code. A total of 11 OPERABLE safety-relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient. Any combination of 5 SRVs operating in the relief mode and 6 SRVs operating in the safety mode is acceptable.

Demonstration of the safety-relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

The low-low set system ensures that safety/relief valve discharges are minimized for a second opening of these valves, following any overpressure transient. This is achieved by automatically lowering the closing setpoint of 5 valves and lowering the opening setpoint of 2 valves following the initial opening. In this way, the frequency and magnitude of the containment blowdown duty cycle is substantially reduced. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS (Continued)

detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973. They provide the ability to measure leakage from fluid systems in the drywell.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shut down to allow further investigation and corrective action.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel.

The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

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 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 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) have permanently installed blocking devices 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, and the cumulative time for vent and purge operation is unlimited.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 18 TO FACILITY OPERATING LICENSE NO. NPF-62

CLINTON POWER STATION, UNIT NO. 1

ILLINOIS POWER COMPANY, ET AL.

DOCKET NO. 50-461

1.0 INTRODUCTION

By letter from D. P. Hall, Illinois Power Company (IP), to USNRC, dated September 6, 1988 (Ref. 1), Technical Specification changes were proposed for the operation of Clinton Power Station Cycle 2 (CPSC2) with a reload using General Electric (GE) manufactured fuel assemblies and GE analyses and methodologies. Enclosed were the requested Technical Specification (TS) changes and GE reload report (Ref. 2) discussing the reload and analyses to support and justify Cycle 2 operation.

The reload for Cycle 2 is generally a normal reload with no unusual core features or characteristics. Proposed TS changes relate to Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Linear Heat Generation Rate (LHGR) limits for the new fuel, and MAPLHGR and Minimum Critical Power Ratio (MCPR) limits for all of the fuel, using Cycle 2 core and transient parameters.

IP also requested changes to Clinton Power Station Technical Specifications to permit operation in the maximum extended operating domain (MEOD) with: (a) up to a 50°F reduction in feedwater temperature, and (b) elimination of APRM setdown. These proposed changes involve, among other factors, the development of new power and flow dependent relations for maximum average planar linear heat generation rate (MAPLHGR) and minimum critical power ratio (MCPR). A General Electric Company (GE) analysis of the consequences of operation in the MEOD (Ref. 3) was included in the submittal to justify the proposed changes. The MEOD includes expansion of the normal power/flow map into two new regions. One region, which involves operation at rated power at lower than rated core flow rates, is called the extended load line region (ELLR). The other region, which involves operation at core flows at up to 107% of rated flow is called the increased core flow region (ICFR). Operation in the ELLR and ICFR permits greater operational flexibility and an improved unit capacity factor. Reduced feedwater temperatures can arise from the inoperability or degraded performance of individual feedwater heaters or string(s) of feedwater heaters or by deliberate reduction of feedwater heating.

A third proposed change would revise the Remote Shutdown System Control to include additional control switches.

IP also requested changes to the Jet Pump surveillance requirements as specified in the Clinton Technical Specifications.

2.0 EVALUATION - RELOAD CYCLE 2

2.1 Reload Description

The CPSC2 will retain 96 P8SRB154 and 360 P8SRB200 GE fuel assemblies from Cycle 1, and add 168 new GE8x8 fuel assemblies (88 BP8SRB284LC and 80 BP8SRB284L). The reload is based on a Cycle 2 end of cycle core nominal average exposure of 12,393 MWd/MT.

2.2 Fuel Design

The new fuel for Cycle 2 is the GE fuel GE BP/P8x8R. The fuel designations are BP8SRB284LC and BP8SRB284L. This fuel type has been approved in the Safety Evaluation Report for Amendment 16 to GESTAR II (Refs. 4 and 5). The specific description of this fuel has been submitted in Amendment 16 to GESTAR II which has been accepted by the staff in Reference 5.

LOCA analyses have been performed for the retained and reload fuel using the SAFE/REFLOOD methods approved by the staff. Clinton still complies with 10 CFR 50.46, Appendix K.

2.3 Nuclear Design

The nuclear design for CPSC2 has been performed by GE with the approved methodology described in GESTAR II (Ref. 5). The results of these analyses are given in the GE reload report (Ref. 2) in standard GESTAR II format. The results are within the range of those usually encountered for BWR reloads. In particular, the shutdown margin is 6.6% delta-k at the beginning of cycle and 2.6% delta-k at the minimum conditions, thus fully meeting the required 0.38% delta-k shutdown margin. The standby liquid control system also meets shutdown requirement with a reasonable shutdown margin of 6.5% delta-k. Since the CPSC2 nuclear design parameters have been obtained with previously approved methods and fall within expected ranges, the nuclear design is acceptable.

2.4 Thermal-Hydraulic Design

The thermal-hydraulic design for CPSC2 has been performed by GE with the approved methodology described in GESTAR II (Ref. 5) and the results are given in the GE reload report (Ref. 2). The GEMINI/ODYN transient analysis methodology (Ref. 5) was used.

2.5 Thermal-Hydraulic Stability

IP implemented the recommendations of GE SIL 380 in CPS Technical Specifications. However, due to the LaSalle 2 instability event and the continuing investigation regarding stability concerns, the licensee informed the staff that IP will implement the interim recommendations of the BWR Owners Group on stability (Ref. 6). IP also responded to NRC Bulletin No. 88-07 "Power Oscillations in BWRs" (Ref. 7).

2.6 Transient and Accident Analysis

The transient and accident analysis methodologies used for CPSC2 are described and NRC approval indicated in GESTAR II (Ref. 5). The GEMINI/ODYN method was used for the core wide transient analysis which includes load rejection without bypass (LRNBP), loss of feedwater heating and feedwater controller failure. The rod withdrawal error (RWE) was analyzed on a generic basis for BWR/6 plant. The results are applicable to Clinton. The core wide and the local transient analysis methodologies and results are acceptable because they fall within expected ranges.

The limiting pressurization event, the main steam isolation valve closure with flux scram, analyzed with standard GESTAR II methods, gave results for peak vessel pressure of 1247 psig, which is below the ASME Section III limit of 1375 psig.

Banked position withdrawal sequence and rod patterns are used for Clinton. For plants using this system, the Rod Drop Accident (RDA) event has been statistically analyzed generically and it was found that with a high degree of confidence, the peak fuel enthalpy would not approach the NRC limit of 280 cal/gm for this event. This approach and analysis have been approved by NRC (Ref. 5). This approach is acceptable for CPSC2.

2.7 Technical Specification Changes for Reload - Cycle 2

T/S Section 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.

T/S Section 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. The change is acceptable.

T/S 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. The change is acceptable.

T/S 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. The change is acceptable.

T/S 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. The change is acceptable.

T/S 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. The change is acceptable.

T/S Section 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. The change is acceptable.

T/S 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. The change is acceptable.

T/S 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. The change is acceptable.

T/S 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.

3.0 EVALUATION - MEOD OPERATION

3.1 Evaluation of Operation in MEOD with Reduced Feedwater Temperature and Elimination of APRM Setdown

3.1.1 Operation in the MEOD with Reduced Feedwater Temperatures

The General Electric Company analysis of Reference 3, which was provided by IP as justification for the proposed changes in Technical Specifications, describes the results of an evaluation of the safety impact of operation in the MEOD with reduced feedwater temperatures. This evaluation included consideration of abnormal operating transients, LOCAs, containment pressures, load impact on vessel internals, flow induced vibration, and fuel mechanical performance.

3.1.2 Abnormal Operational Transients

Many transients of Chapter 15 of the FSAR were considered for operation in the MEOD. The transients investigated were generator load rejection without turbine bypass (LRNBP), feedwater controller failure to maximum demand (FWCF), and load rejection with bypass failure, no recirculation pump trip (RPT) on turbine trip, RPT on high pressure. It was concluded that the current power dependent MCPR limit ($MCPR_p$) bounded these cases in the MEOD.

At the staff's request, IP also provided a plant specific analysis of the loss of feedwater heating (LFWH) transient in the MEOD. This analysis indicated that the LFWH transient is bounded by the reload analysis (Ref. 2).

The flow dependent MCPR operating limit ($MCPR_f$) in the current Technical Specifications was based on slow recirculation flow runout transients. For operation in the MEOD, this event was reanalyzed on a generic basis (BWR/6) with approved methods to account for initial operation at low flows and a higher power rod line of the ELLR. The core flow conditions were extended to 112% for the generic analysis. Flow dependent MCPR and MAPLHGR curves are extrapolated from the BWR/6 analysis. Two new $MCPR_f$ relations were developed for two settings of the core flow limiter giving maximum core flows of 102.5 and 109 percent of rated flow. We find this acceptable.

3.1.3 LOCA Analysis

The licensee submitted consequences of a LOCA in the MEOD in Reference 3. The generic analysis results, obtained with approved methods, indicate that operation in the MEOD would result in less than a 5°F increase in the peak clad temperatures of Chapter 6 of the CPS FSAR and that the requirements of 10 CFR 50.46 are satisfied. We find this acceptable.

3.1.4 Containment Pressure Response

A conservative containment analysis for operation in the MEOD with feedwater heaters out-of-service (FWHOS) resulted in a peak drywell pressure of 19.7 psig. However, this is still below the design limit of 30 psig.

3.1.5 Overpressure Protection

The sizing of the main steam safety valves for the ASME overpressure protection analysis is obtained for an MSIV closure event with flux scram. Calculations of this event for operation in the MEOD indicated a peak vessel pressure of 1245 psig, well below the ASME code limit of 1375 psig. We find this acceptable.

3.2 Elimination of APRM Setdown

In the current CPS Technical Specifications, the flow-biased APRM trips are reduced (setdown when the core maximum total peaking factor exceeds the design total peaking factor). This requirement was associated with a now obsolete Hench-Levy Minimum Critical Heat Flux Ratio Criterion. The General Electric Company analysis (Ref. 3) supplied with the IP submittal includes results from analyses made to determine the new initial conditions of fuel thermal limits that would be needed to satisfy the pertinent licensing criteria if APRM setdown was eliminated. The new limits should (1) prevent violation of the MCPR safety limit, (2) keep the fuel thermal-mechanical performance within the design and licensing basis, and (3) keep peak cladding temperature and maximum cladding oxidation within allowable limits. The evaluation included operation in the MEOD with reduced feedwater temperature. The elimination of APRM setdown does affect the power dependent MCPR limit and the MAPLHGR limit. The results of the analysis with approved methods are as follows:

(1) New power dependent relations for MCPR and MAPLHGR limits are provided which include both high and low flow relations at powers below 40% where reactor scram on turbine control valve fast closure is bypassed. The MAPLHGR relation is a factor, $MAPFAC_p$, which is multiplied by the rated MAPLHGR limit to obtain the power dependent MAPLHGR limit.

(2) A new flow dependent MAPLHGR factor, $MAPFAC_f$, is provided. This factor was determined from analysis of slow flow runout transients with the requirement that peak transient MAPLHGR values not exceed the fuel design basis values. We find this acceptable.

3.3 Technical Specification Changes for MEOD

T/S Section 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. This is acceptable.

T/S Section 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. This is acceptable.

T/S Section 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. This is acceptable.

T/S Tables 2.2.1-1 and 3.3.6-2: 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 in Reference 3. 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. APRM flow biased scram setpoints are now given for two loop and single loop separately. Setpoints are newly added for single loop operations. The changes are acceptable.

The definitions of W and DW were deleted in T/S 3/4.2.2 and were not included in the proposed changes. The licensee agreed to include the definitions in the T/S as suggested by the staff.

T/S 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 deleted in this change request.

T/S Section 3/4.2.1: 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 with the deletion of the APRM flow-biased simulated thermal power-high scram setpoint adjustment (Technical Specification 3/4.2.2).

T/S Section 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 without this T/S requirement. The deletion of this section is acceptable.

T/S 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 temperature.

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. The changes are acceptable.

T/S 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 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 are 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. The proposed changes are acceptable.

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

T/S 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%. The changes are acceptable.

T/S 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. The changes are acceptable.

T/S 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. The proposed changes are acceptable.

T/S 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. The changes are acceptable.

T/S 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. The changes are acceptable.

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

T/S 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%. The changes are acceptable.

T/S Table 3.3.6-2 (**note): This change removes reference to Technical Specification 3.2.2 which is being deleted. The changes are acceptable.

T/S 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. The changes are acceptable.

T/S 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. The changes are acceptable.

T/S 3.4.1.1 ACTION a.1.d: This change removes the requirement to multiply the MAPLHGR Limits by 0.85, 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. The changes are acceptable.

T/S 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. The change is acceptable.

T/S 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 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. The change is acceptable.

4.0 EVALUATION - REMOTE SHUTDOWN SYSTEM CONTROLS

The proposed change is a revision to the Table 3.3.7.4-2, "Remote Shutdown System Controls," adding control switches used in the electrical operating circuit of division 2 RHR Heat Exchanger Valves (1E12-F068B and 1E12-F014B) and division 1 power supply 4 kV circuit breaker (252-AT1AA1). The design change to add these control switches in the RHR heat exchanger valves was approved in paragraph 7.4.3.1 of the staff Safety Evaluation Report (Ref. 8). The addition of a control switch at the remote shutdown panel for division 1 power supply circuit breaker control is a modification proposed to resolve similar concerns as those for the other components evaluated in reference 8.

CPS Remote Shutdown System (RSS) includes a single Remote Shutdown Panel (RSP) located in the auxiliary building. The function of the RSS is to achieve and maintain hot shutdown and a subsequent cold shutdown from the RSP or any other location remote from the main control room with a single failure including loss of electric power. Credit may be taken for manual actuation of systems from locations that are reasonably accessible from the RSP without using jumpers, rewiring or disconnecting circuits when effecting shutdown from outside the control room.

The original Clinton design provided only for the transfer to division 1 system from the control room to RSP. The design did not include transfer of control of the division 1 power supply circuit breaker to RSP. The required control of this circuit breaker remote from the control room was provided locally at the switchgear. The licensee has modified the design by adding a switch at the RSP to control the 4 kV division 1 power supply circuit breaker. This will provide an additional and more efficient control capability for the circuit breaker remote from the control room.

Similarly the division 2 RHR heat exchanger valves can only be operated manually in case the normal control at the main control room is not available. The licensee's proposal to add control switches for these valves at their respective motor control centers (MCCs) was found acceptable in reference 2. The addition of control switches to the division 2 RHR heat exchanger valves and division 1 power supply circuit breaker controls provides a diverse and redundant means of controlling the remote shutdown systems and thus will facilitate remote operation of these components. Reference 1 submitted that the control switches meet the same quality standards and will be installed in accordance with the same requirements as the existing components on the MCCs and the RSP. Reference 1 also indicated that the normal control and operation of the circuit breaker and the valves will not be affected by the addition of these switches, and the ability to manually operate these components will remain unchanged.

Based on the above evaluation, the staff concludes that the proposed changes to the Technical Specification represent the modification that was previously approved by the staff, do not involve an unreviewed safety question and therefore, is acceptable.

5.0 EVALUATION - JET PUMP OPERABILITY, TECHNICAL SPECIFICATION 3/4.4.1.2

A revision is proposed to the subject specifications to allow present Surveillance Requirement 4.4.1.2 to be performed with THERMAL POWER in excess of 25% of RATED THERMAL POWER instead of the present prior to exceeding 25% of RATED THERMAL POWER. The proposed Surveillance Requirement 4.4.1.2 will allow entry into OPERATIONAL CONDITIONS 1 and 2 without having to perform present Surveillance Requirement 4.4.1.2; however, jet pump OPERABILITY is required to be determined after entering OPERATIONAL CONDITION 2 (OC-2) and at least once every 24 hours thereafter by verifying that the diffuser to lower plenum differential pressure is within specified limits. Entry into OC-2 is necessary in order to perform the surveillance required to demonstrate jet pump operability. OC-2 operation is needed to achieve power levels sufficient for meaningful measurements of flow and differential pressure (dp). When power and flow conditions are too low, the effects of natural circulation, moderator subcooling changes and varying core dp result in large data uncertainties. These large uncertainties also necessitate different criteria for demonstrating jet pump operability when power is less than 25% of rated thermal power (RTP). We have reviewed the licensee's proposed changes to Technical Specification 4.4.1.2. The changes proposed are acceptable because they are necessary for meaningful surveillance measurements.

This proposed change also revises Surveillance Requirement 4.4.1.2 by deleting the requirement that the recirculation flow control valves be in the same position when performing the surveillance.

The existing specification requires that Surveillance 4.4.1.2 be performed with the flow control valves (FCV) in the same position. Changing the position of one FCV relative to the other has the effect of changing the flow in one recirculation loop relative to the other loop. The proposed change would allow different FCV positions, and thus different flows, provided that the flow mismatch is within the normal mismatch limits of Specification 3.4.1.3.

We have reviewed the proposal and note that even at the same FCV position, relative loop flows are different because of differing flow path resistances and individual pump characteristics. The effect of the proposed change on measured parameters is, therefore, expected to be insignificant. We conclude that the proposed changes are acceptable.

General Electric Service Information Letter (SIL) No. 330 was recognized in NUREG/CR-3052 as providing acceptable guidelines for verifying jet pump operability and for closeout of IE Bulletin 80-07, "BWR Jet Pump Assembly Failure." Both NUREG/CR-3052 and SIL 330 recognize that a 10% deviation from the normal percent difference between jet pump flow and the average jet pump flow is an acceptable criterion. Both documents also recognize that either differential pressure or flow could be monitored, depending on a particular plant's instrumentation, and that a 10% change in jet pump flow corresponds to at least a 20% in differential pressure. Therefore, if jet pump diffuser-to-lower plenum differential pressure is the monitored variable, then a 20% deviation should be an acceptable criterion for that variable. The current Technical Specification 3.4.1.2.3, however, specifies 10% (deviation) as the acceptance criterion for jet pump diffuser-to-lower plenum differential pressure. The licensee claims that this value is too restrictive relative to differential pressure and is not the value specified in SIL 330 as an acceptance criterion for that variable.

The proposed change to the specified acceptable deviation from patterns established for individual jet pump diffuser-to-lower plenum differential pressure from 10% to 20% is consistent with the recommendations of both NUREG/CR-3052 and SIL 330 which established 20% as an acceptable indication of jet pump operability. The T/S changes in Section 3.4.1.2.3 are acceptable.

The proposed wording changes to Technical Specification 4.4.1.2, including combining the sections which currently and separately address single and double loop operation into a single section, do not change the intent of the specification as it is currently worded since the surveillance requirements are the same for either mode of operation. The changes are acceptable.

The statement at the bottom of T/S Section 3.4.1.2 is revised by deleting the words "provided that this surveillance is performed within 24 hours after exceeding 25% of rated thermal power." The words are no longer needed since they are redundant to the words proposed for the surveillance itself. The proposed changes are acceptable.

6.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published (54 FR 4925) in the Federal Register on January 31, 1989. Accordingly, based upon the environmental assessment, the Commission had determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

7.0 CONCLUSION

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

8.0 REFERENCES

1. Letter from D.P. Hall (IP) to USNRC dated September 6, 1988 "Clinton Power Station Proposed Amendment to Facility Operating License NPF-62."

2. GE-23A5921, Rev. 0, August 1988, "Supplemental Reload Licensing Submittal for Clinton Power Station Unit 1, Reload 1, Cycle 2."

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

4. Letter from J.S. Charnley (GE) to H.N. Berkow (NRC) "Proposed Amendment 16 to GE Licensing Topical Report NEDE-24011-P-A," August 8, 1986.

5. Letter from A.C. Thadani (NRC) to J.S. Charnley (GE) "Acceptance for Referencing of Amendment 16 to GE Licensing Topical Report NEDE-24011-P-A, 'General Electric Standard Application for Reactor Fuel,'" April 20, 1988.

6. Letter from D. L. Holtzscher (IP) to USNRC dated December 22, 1988 "Clarification of Information Provided in the Clinton Power Station Reload 1 Licensing Submittal".

7. Letter from D.P. Hall (IP) to USNRC "Clinton Power Station Response to NRC Bulletin No. 88-07," dated September 13, 1988.

8. Staff Safety Evaluation Report, NUREG-0853, Supplement No. 6 dated July 1986.

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