August 12, 1986

Docket No.: 50-416

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Mr. Oliver D. Kingsley, Jr.
Vice President, Nuclear Operations
Mississippi Power & Light Company
Post Office Box 23054
Jackson, Mississippi 39205

Dear Mr. Kingsley:

SUBJECT: TECHNICAL SPECIFICATION CHANGES TO ALLOW OPERATION WITH ONE RECIRCULATION LOOP AND EXTENDED OPERATING DOMAIN

RE: GRAND GULF NUCLEAR STATION, UNIT 1

The Commission has issued the enclosed Amendment No. 16 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 31, 1986 as amended by letters dated May 2, June 9, and June 20, 1986 and supplemented July 11, 1986.

This amendment changes Technical Specifications to allow plant operation with one recirculation loop operable and to allow operation with increased reactor core cooling water flow rates up to 105% of rated flow rate at reduced power levels and operation with increased reactor power levels at reduced reactor core cooling water flow rates. The amendment does not involve an increase in licensed power level. Technical Specification changes include elimination of the average power range monitor trip setdown and changes to allow rated power operation with feedwater temperatures reduced 50°F below rated feedwater temperature (420°F).

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

Sincerely,

Original signed by

Lester L. Kintner, Project Manager
BWR Project Directorate No. 4
Division of BWR Licensing

Enclosures:
1. Amendment No. 16 to License No. NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

*Previously concurred:

PD#4/LA	PD#4/PM	FOB/D	OGC	PD#4/D
*MO'Brien	*LKintner:lb	*DVassallo	*Young	*WButler WB
08/01/86	08/01/86	08/08/86	08/05/86	08/04/86 8/11/86

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by

Walter R. Butler, Director
BWR Project Directorate No. 4
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 12, 1986

PD#4/LA	PD#4/PM	OGC	PD#4/D
MO'Brien	LKintner:lb	MYoung	WButler
8/1/86	8/1/86	8/5/86	8/11/86

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

Docket No.: 50-416

August 15, 1986

Mr. Oliver D. Kingsley, Jr.
Vice President, Nuclear Operations
Mississippi Power & Light Company
Post Office Box 23054
Jackson, Mississippi 39205

Dear Mr. Kingsley:

SUBJECT: TECHNICAL SPECIFICATION CHANGES TO ALLOW OPERATION WITH ONE RECIRCULATION LOOP AND EXTENDED OPERATING DOMAIN

RE: GRAND GULF NUCLEAR STATION, UNIT 1

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L L Kintner

Lester L. Kintner, Project Manager
BWR Project Directorate No. 4
Division of BWR Licensing

Enclosures:
1. Amendment No. 16 to
   License No. NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

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

MISSISSIPPI POWER & LIGHT COMPANY
MIDDLE SOUTH ENERGY, INC.
SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION
DOCKET NO. 50-416
GRAND GULF NUCLEAR STATION, UNIT 1
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 16
License No. NPF-29

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

   A. The application for amendment by Mississippi Power & Light Company, Middle South Energy, Inc., and South Mississippi Electric Power Association, (the licensees) dated March 31, 1986 as amended by letters dated May 2, June 9, and June 20, 1986 and supplemented July 11, 1986, 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-29 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 16 , are hereby incorporated into this license. Mississippi Power & Light 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

Walter R. Butler

Walter R. Butler, Director
BWR Project Directorate No. 4
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 15, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 16

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

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

Remove	Insert
iii	iii*
iv	iv
v	v
vi	vi*
xi	xi*
xii	xii
xiii	xiii
xiv	xiv*
2-1	2-1
2-2	2-2*
2-3	2-3*
2-4	2-4
--	2-4a
B 2-1	B 2-1
B 2-2	B 2-2
B 2-3	B 2-3
B 2-4	B 2-4*
B 2-7	B 2-7
B 2-8	B 2-8*
B 2-9	B 2-9
--	B 2-10
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
--	3/4 2-2a
--	3/4 2-2b
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4*
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 3-5	3/4 3-5
3/4 3-6	3/4 3-6*
3/4 3-7	3/4 3-7*
3/4 3-8	3/4 3-8
3/4 3-43	3/4 3-43
3/4 3-44	3/4 3-44*
3/4 3-55	3/4 3-55
--	3/4 3-55a
3/4 3-56	3/4 3-56*
3/4 3-57	3/4 3-57
3/4 3-58	3/4 3-58*

Remove	Insert
--	3/4 3-111
--	3/4 3-112
3/4 4-1	3/4 4-1
--	3/4 4-1a
--	3/4 4-1b
--	3/4 4-1c
3/4 4-2	3/4 4-2
3/4 4-3	3/4 4-3
3/4 4-4	3/4 4-4*
B 3/4 1-1	B 3/4 1-1*
B 3/4 1-2	B 3/4 1-2
B 3/4 2-1	B 3/4 2-1
--	B 3/4 2-1a
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	B 3/4 2-5
B 3/4 2-6	B 3/4 2-6
--	B 3/4 2-6a
B 3/4 2-7	B 3/4 2-7
B 3/4 3-3	B 3/4 3-3
--	B 3/4 3-3a
B 3/4 3-4	B 3/4 3-4*
B 3/4 3-7	B 3/4 3-7
B 3/4 3-8	B 3/4 3-8*
B 3/4 4-1	B 3/4 4-1
--	B 3/4 4-1a
B 3/4 4-2	B 3/4 4-2*

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

BASES

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

THERMAL POWER, High Pressure and High Flow

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

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

SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

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

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4 and 5

ACTION:

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

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	$\leq$ 120/125 divisions of full scale	$\leq$ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	$\leq$ 15% of RATED THERMAL POWER	$\leq$ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-High		
1) During two recirculation loop operation:		
a) Flow Biased	$\leq$ 0.66 W+64%, with a maximum of	$\leq$ 0.66 W+67%, with a maximum of
b) High Flow Clamped	$\leq$ 111.0% of RATED THERMAL POWER	$\leq$ 113.0% of RATED THERMAL POWER
2) During single recirculation loop operation:		
a) Flow Biased	$\leq$ 0.66 W+40%	$\leq$ 0.66 W+43%
b) High Flow Clamped	Not Required OPERABLE	Not Required OPERABLE
c. Neutron Flux-High	$\leq$ 118% of RATED THERMAL POWER	$\leq$ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 1064.7 psig	$\leq$ 1079.7 psig
4. Reactor Vessel Water Level - Low, Level 3	$\geq$ 11.4 inches above instrument zero*	$\geq$ 10.8 inches above instrument zero*
5. Reactor Vessel Water Level-High, Level 8	$\leq$ 53.5 inches above instrument zero*	$\leq$ 54.1 inches above instrument zero*

TABLE 2.2.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. Main Steam Line Isolation Valve - Closure	$\leq$ 6% closed	$\leq$ 7% closed
7. Main Steam Line Radiation - High	$\leq$ 3.0 x full power background	$\leq$ 3.6 x full power background
8. Drywell Pressure - High	$\leq$ 1.23 psig	$\leq$ 1.43 psig
9. Scram Discharge Volume Water Level - High	$\leq$ 60% of full scale	$\leq$ 63% of full scale
10. Turbine Stop Valve - Closure	$\geq$ 40 psig**	$\geq$ 37 psig
11. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\geq$ 44.3 psig**	$\geq$ 42 psig
12. Reactor Mode Switch Shutdown Position	NA	NA
13. Manual Scram	NA	NA

*See Bases Figure B 3/4 3-1.

**Initial setpoint. Final setpoint to be determined during startup test program. Any required change to this setpoint shall be submitted to the Commission within 90 days of test completion.

2.0 INTRODUCTION

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

2.1.1 THERMAL POWER, Low Pressure or Low Flow

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

2.1.2 THERMAL POWER, High Pressure and High Flow

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

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB[a], which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) Boiling Length (L), GEXL, correlation. The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation.

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

The bases for the uncertainties in the core parameters are given in NEDO-20340[b] and the basis for the uncertainty in the GEXL correlation is given in NEDO-10958-A[a]. The bases for the changes in uncertainties resulting from a single loop operation are given in the GGNS Single Loop Operation Analysis dated February 1986. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution during any fuel cycle would not be as severe as the distribution used in the analysis.

---

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

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

Bases Table B2.1.2-1

UNCERTAINTIES USED IN THE DETERMINATION OF THE FUEL CLADDING SAFETY LIMIT*

Quantity	Standard Deviation (% of Point)
Feedwater Flow	1.76
Feedwater Temperature	0.76
Reactor Pressure	0.5
Core Inlet Temperature	0.2
Core Total Flow	2.5[(a)]
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	6.3[(b)]
R Factor	1.5
Critical Power	3.6

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

(a) This value increases to 6.0 for single recirculation loop operation.

(b) This value increases to 6.8 for single recirculation loop operation.

Bases Table B2.1.2-2

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

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

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

amount, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

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 118% 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 ± 1 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 that produces a rated core flow of 112.5 million lb/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 while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine stop valve closure trip is bypassed. For a turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint 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.

5. Reactor Vessel Water Level-High

A reactor scram from high reactor water level, approximately two feet above normal operating level, is intended to offset the addition of reactivity effect associated with the introduction of a significant amount of relatively cold feedwater. An excess of feedwater entering the vessel would be detected by the level increase in a timely manner. This scram feature is only effective when the reactor mode switch is in the Run position because at THERMAL POWER levels below 10% to 15% of RATED THERMAL POWER, the approximate range of power level for changing to the Run position, the safety margins are more than adequate without a reactor scram.

6. Main Steam Line Isolation Valve-Closure

The main steam line isolation valve closure trip was provided to limit the amount of fission product release for certain postulated events. The MSIV's are closed automatically from measured parameters such as high steam flow, high steam line radiation, low reactor water level, high steam tunnel temperature and low steam line pressure. The MSIV's closure scram anticipates the pressure and flux transients which could follow MSIV closure and thereby protects reactor vessel pressure and fuel thermal/hydraulic Safety Limits.

7. Main Steam Line Radiation-High

The main steam line radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected, a trip is initiated to reduce the continued failure of fuel cladding. At the same time the main steam line isolation valves are closed to limit the release of fission products. The trip setting is high enough above background radiation levels to prevent spurious trips yet low enough to promptly detect gross failures in the fuel cladding.

8. Drywell Pressure-High

High pressure in the drywell could indicate a break in the primary pressure boundary systems or a loss of drywell cooling. The reactor is tripped in order to minimize the possibility of fuel damage and to reduce the amount of energy being added to the coolant and to the primary containment. The trip setting was selected to be as low as possible to minimize heat loads of equipment located in the primary containment and to avoid spurious trips. Negative barometric pressure fluctuations are accounted for in the trip setpoints and allowable values specified for drywell pressure-high.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

9. Scram Discharge Volume Water Level-High

The scram discharge volume receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this volume fill up to a point where there is insufficient volume to accept the displaced water at pressures below 65 psig, control rod insertion would be hindered. The reactor is therefore tripped when the water level has reached a point high enough to indicate that it is indeed filling up, but the volume is still great enough to accommodate the water from the movement of the rods at pressures below 65 psig when they are tripped. The trip setpoint for each scram discharge volume is equivalent to a contained volume of 26 gallons of water.

10. Turbine Stop Valve-Closure

The turbine stop valve closure trip anticipates the pressure, neutron flux, and heat flux increases that would result from closure of the stop valves. With a trip setting of 40 psig, the resultant increase in heat flux is such that adequate thermal margins are maintained during the worst case transient assuming the turbine bypass valves fail to operate. As indicated in Table 3.3.1-1, this function is automatically bypassed below the turbine first-stage pressure value equivalent to THERMAL POWER less than 40% of RATED THERMAL POWER.

The automatic bypass setpoint is feedwater temperature dependent as a result of the subcooling changes that affect the turbine first-stage pressure-reactor power relationship. For RATED THERMAL POWER operation with feedwater temperature greater than or equal to 420°F, an allowable setpoint of ≤ 26.9% of control valve wide-open turbine first-stage pressure is provided for the bypass function. This setpoint is also applicable to operation at less than RATED THERMAL POWER with the correspondingly lower feedwater temperature. The allowable setpoint is reduced to ≤ 22.5% of control valve wide-open turbine first-stage pressure for RATED THERMAL POWER operation with a feedwater temperature between 370°F and 420°F. Similarly, the reduced setpoint is applicable to operation at less than RATED THERMAL POWER with the correspondingly lower feedwater temperature.

11. Turbine Control Valve Fast Closure, Trip Oil Pressure-Low

The turbine control valve fast closure trip anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection coincident with failure of the turbine bypass valves. The Reactor Protection System initiates a trip when fast closure of the control valves is initiated by a low EHC fluid pressure in the control valve and in less than 100 milliseconds after the start of control valve fast closure. This loss of pressure is sensed by pressure transmitters which output to trip units whose contacts form the one-out-of-two twice logic

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

input to the Reactor Protection System. The trip setpoint is 43.3 psig. This trip setting and a different valve characteristic from that of the turbine stop valve combine to produce transients which are very similar to that for the stop valve. Relevant transient analyses are discussed in Section 15.2.2 of the Final Safety Analysis Report. As with the Turbine Stop Valve Closure, this function is also bypassed below 40% of RATED THERMAL POWER. The basis for the setpoint is identical to that described for the Turbine Stop Valve Closure.

12. Reactor Mode Switch Shutdown Position

The reactor mode switch Shutdown position provides trip signals into system trip channels which are redundant to the automatic protective instrumentation channels and provides additional manual reactor trip capability.

13. Manual Scram

The Manual Scram pushbutton switches introduce trip signals into system trip channels which are redundant to the automatic protective instrumentation channels and provides manual reactor trip capability.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figure 3.2.1-1 as multiplied by the smaller of either the flow-dependent MAPLHGR factor ($MAPFAC_f$) of Figure 3.2.1-2, or the power-dependent MAPLHGR factor ($MAPFAC_p$) of Figure 3.2.1-3.

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

ACTION:

With an APLHGR exceeding the applicable limits, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the applicable limits:

a. At least once per 24 hours,

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

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

d. The provisions of Specification 4.0.4 are not applicable.

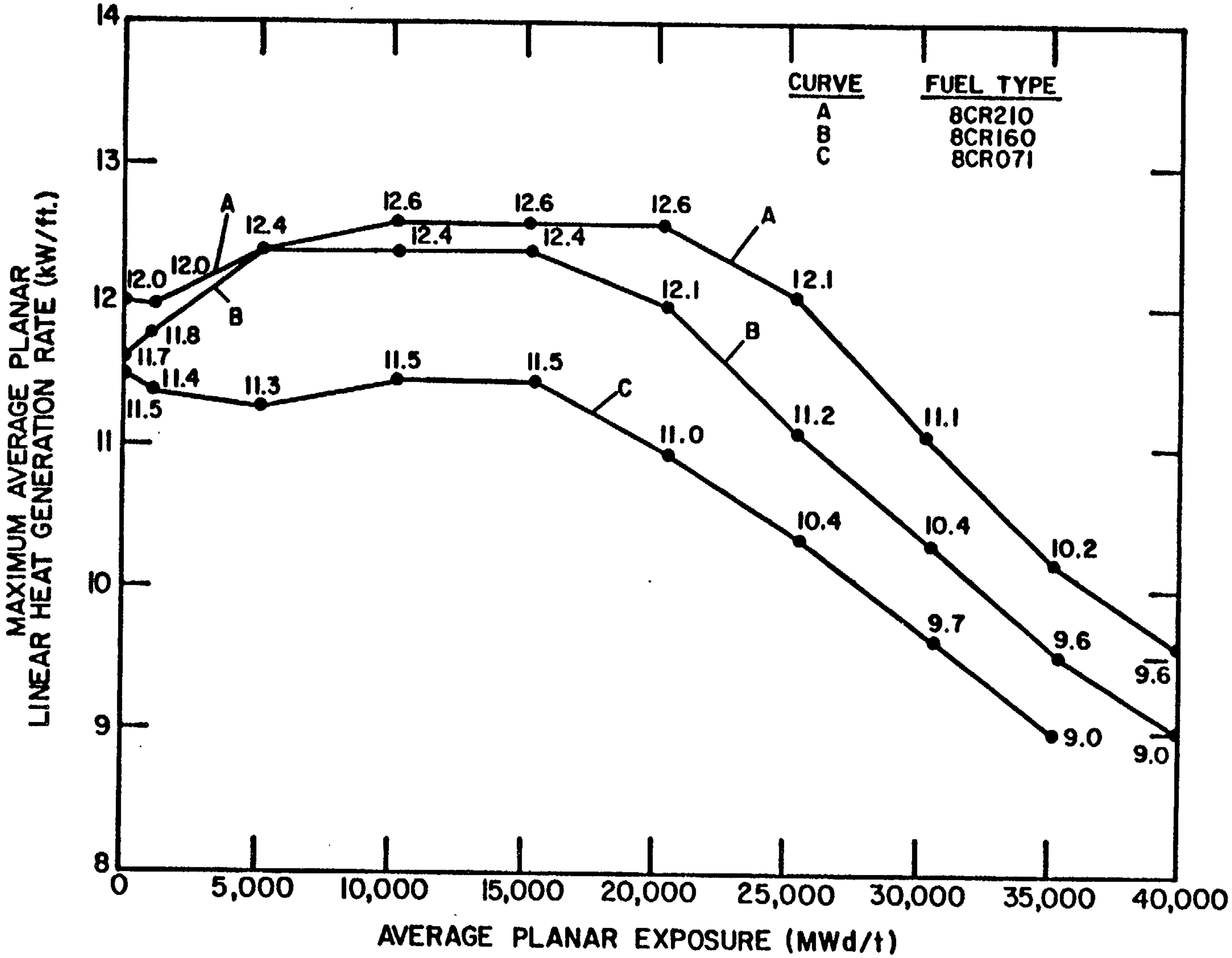

FIGURE 3.2.1-1 MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE FOR TWO LOOP OPERATION INITIAL CORE FUEL TYPES 8CR210, 8CR160 and 8CR071

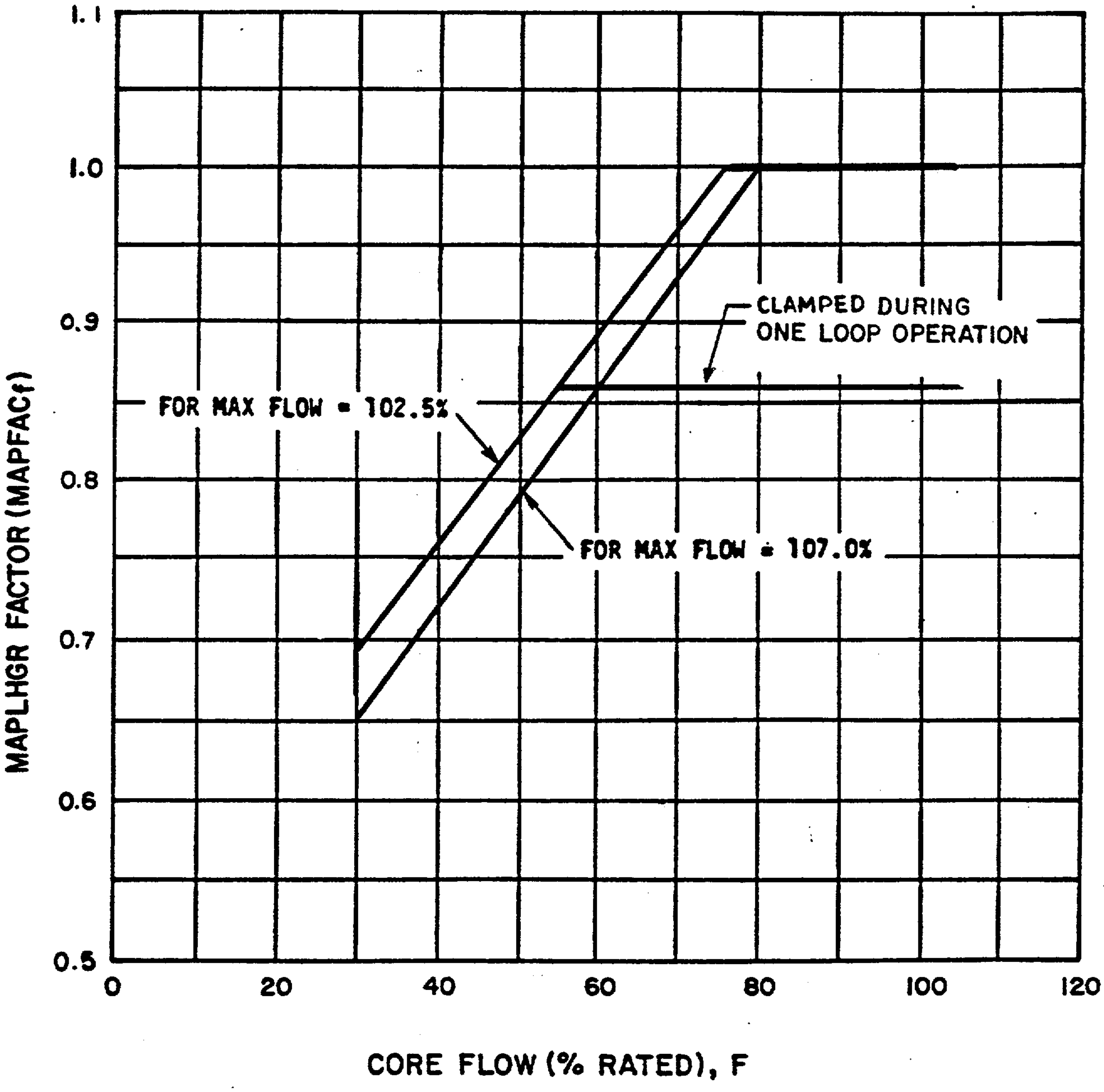

FIGURE 3.2.1-2 $MAPFAC_f$

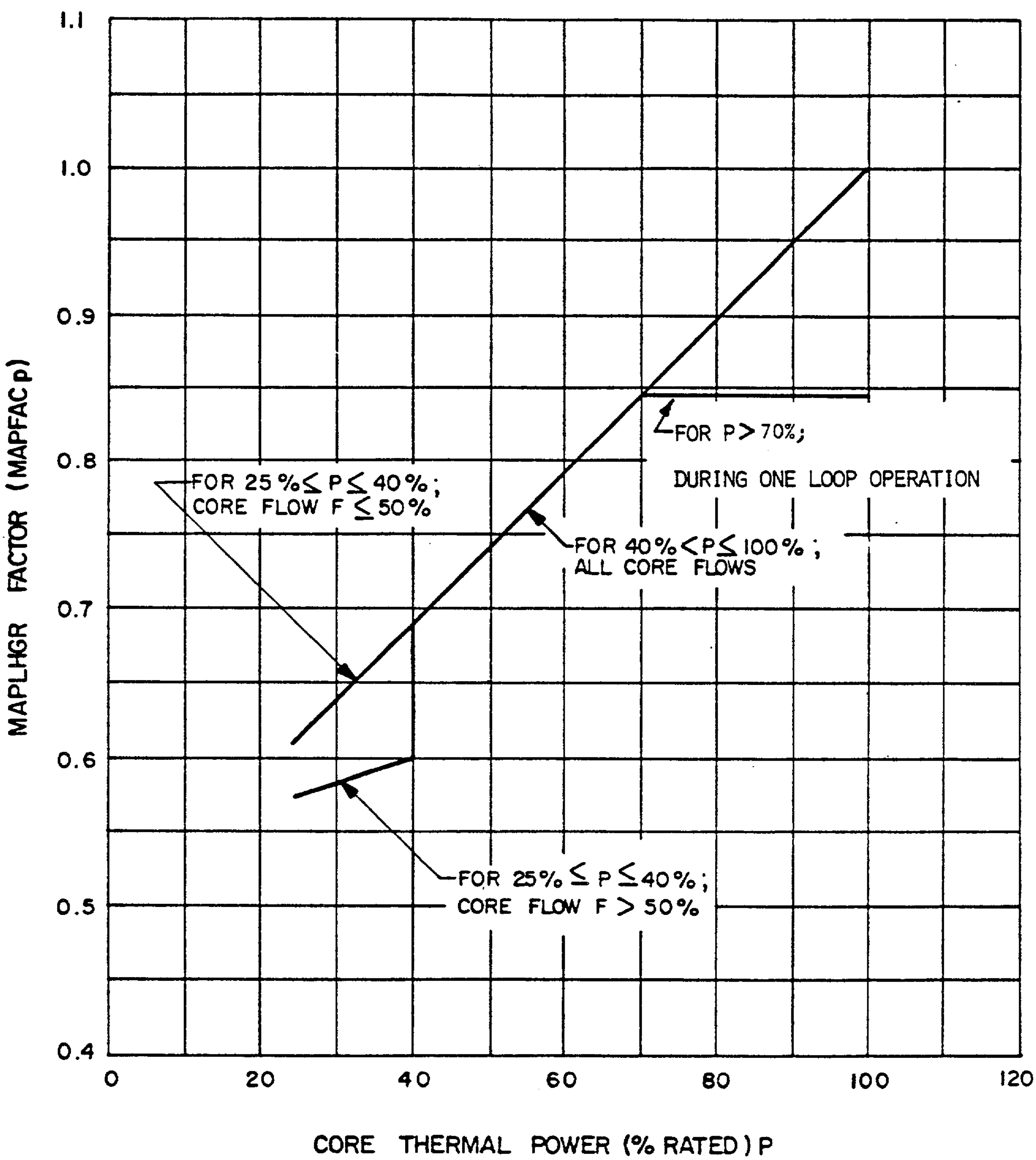

FIGURE 3.2.1-3 $MAPFAC_p$

POWER DISTRIBUTION LIMITS

3/4.2.2 [DELETED]

POWER DISTRIBUTION LIMITS

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than both $MCPR_f$ and $MCPR_p$ limits at indicated core flow and THERMAL POWER as shown in Figures 3.2.3-1 and 3.2.3-2.

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

ACTION:

With MCPR less than the applicable MCPR limits determined from Figures 3.2.3-1 and 3.2.3-2, initiate corrective action within 15 minutes and restore MCPR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR shall be determined to be equal to or greater than the applicable MCPR limits determined from Figures 3.2.3-1 and 3.2.3-2:

a. At least once per 24 hours,

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

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

d. The provisions of Specification 4.0.4 are not applicable.

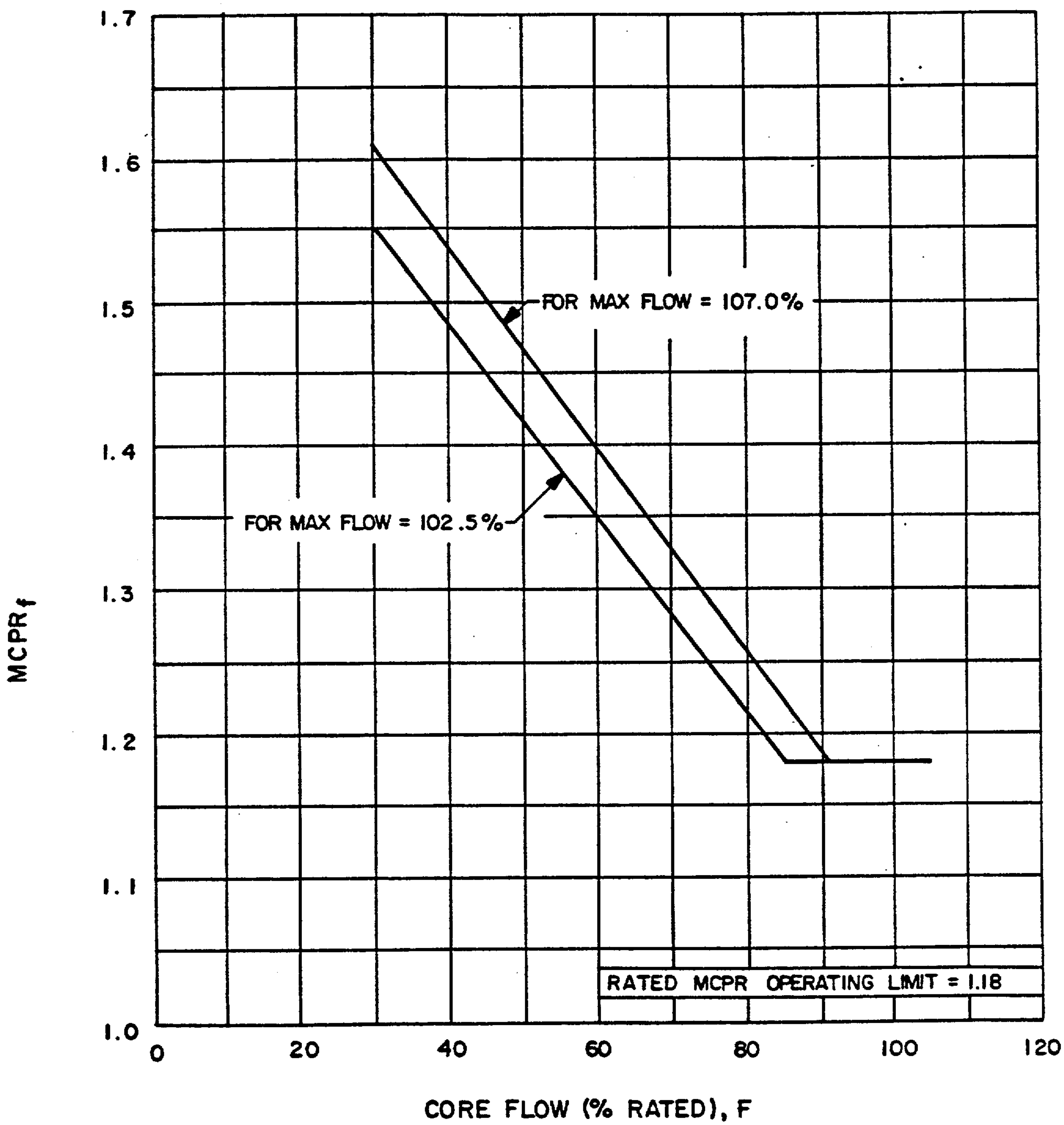

FIGURE 3.2.3-1 MCPR$_f$

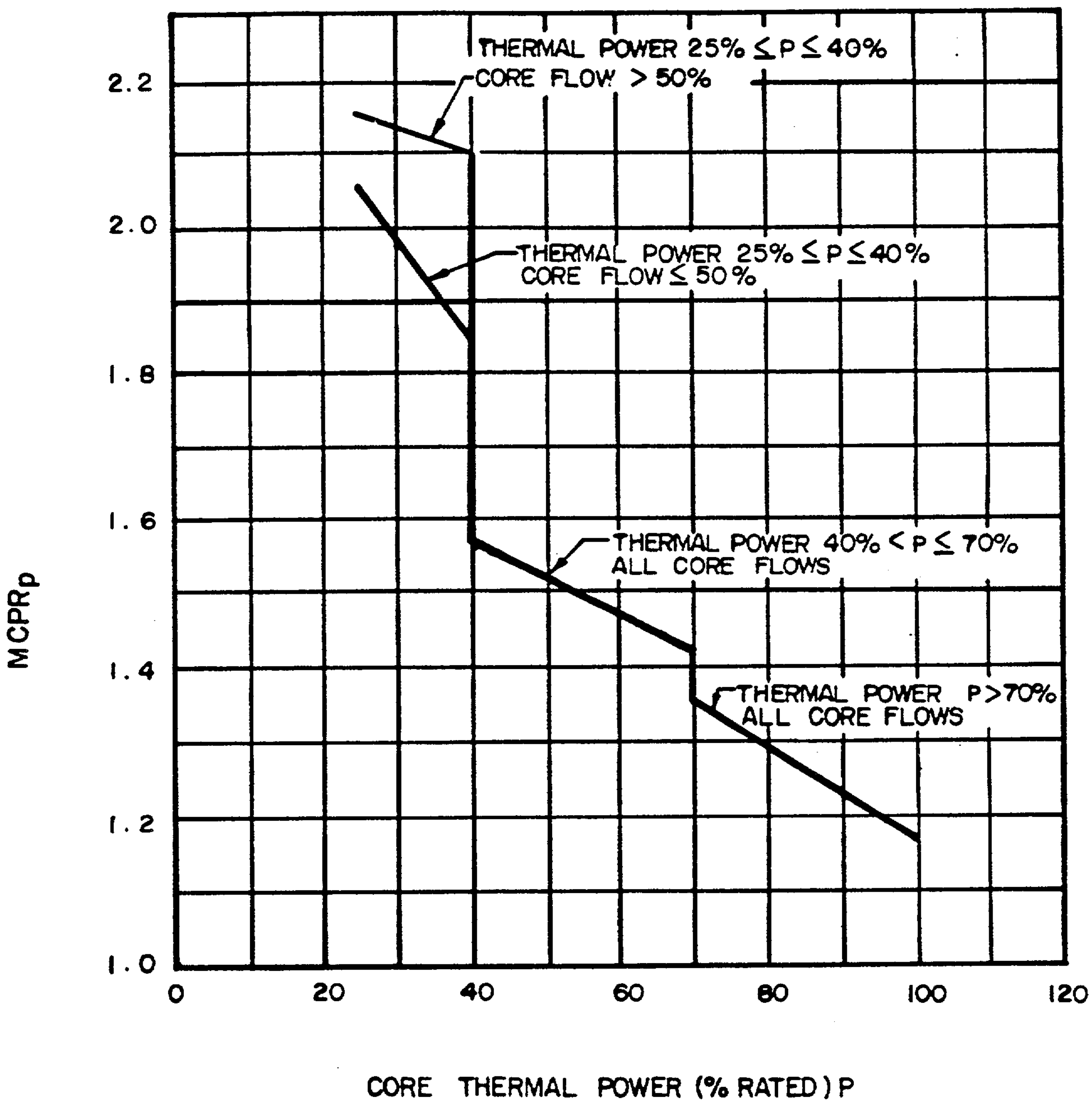

FIGURE 3.2.3-2 $MCPR_p$

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* per Specification 3.9.2 and shutdown margin demonstrations performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

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

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when DRYWELL INTEGRITY is not required.

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

(h) This function shall be automatically bypassed when operating below the appropriate turbine first stage pressure setpoint of:

(1) $\leq$ 26.9%** of the value of turbine first-stage pressure at valves wide open (VWO) steam flow when operating with rated feedwater temperature of greater than or equal to 420°F, or

(2) $\leq$ 22.5%** of the value of turbine first-stage pressure at VWO steam flow when operating with rated feedwater temperature between 370°F and 420°F.

---

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**Allowable setpoint values of turbine first-stage pressure equivalent to THERMAL POWER less than 40% of RATED THERMAL POWER.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - High, Setdown	NA
b. Flow Biased Simulated Thermal Power - High	$\leq$ 0.09**
c. Neutron Flux - High	$\leq$ 0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 0.35
4. Reactor Vessel Water Level - Low, Level 3	$\leq$ 1.05
5. Reactor Vessel Water Level - High, Level 8	$\leq$ 1.05
6. Main Steam Line Isolation Valve - Closure	$\leq$ 0.06
7. Main Steam Line Radiation - High	NA
8. Drywell Pressure - High	NA
9. Scram Discharge Volume Water Level - High	NA
10. Turbine Stop Valve - Closure	$\leq$ 0.10
11. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	$\leq$ 0.10#
12. Reactor Mode Switch Shutdown Position	NA
13. Manual Scram	NA

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

**Not including simulated thermal power time constant.

#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S,[b]	S/U, W	R	2
	S	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor:[f]				
a. Neutron Flux - High, Setdown	S/U,S,[b]	S/U, W	SA	2
	S	W	SA	3, 5
b. Flow Biased Simulated Thermal Power - High	S, D[h]	W	W[d][e], SA, R[i]	1
c. Neutron Flux - High	S	W	W[d], SA	1
d. Inoperative	NA	W	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	S	M	R[g]	1, 2[j]
4. Reactor Vessel Water Level - Low, Level 3	S	M	R[g]	1, 2
5. Reactor Vessel Water Level - High, Level 8	S	M	R[g]	1
6. Main Steam Line Isolation Valve - Closure	NA	M	R	1
7. Main Steam Line Radiation - High	S	M	R	1, 2[j]
8. Drywell Pressure - High	S	M	R[g]	1, 2[k]

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Scram Discharge Volume Water Level - High	S	M	R(g)	1, 2, 5(l)
10. Turbine Stop Valve - Closure	S	M	R(g)	1
11. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	S	M	R(g)	1
12. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
13. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

(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 1/2 decade during each controlled shutdown, if not performed within the previous 7 days.
(c) [DELETED]
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER > 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 MWD/T using the TIP system.
(g) Calibrate trip unit at least once per 31 days.
(h) Verify measured drive flow to be less than or equal to established drive flow at the existing flow control valve position.
(i) This calibration shall consist of verifying the 6 ± 1 second simulated thermal power time constant.
(j) Not applicable when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.
(k) Not applicable when DRYWELL INTEGRITY is not required.
(l) Applicable with any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

	TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]
1.	Turbine Stop Valve - Closure	2[b]
2.	Turbine Control Valve - Fast Closure	2[b]

(a) A trip system may be placed in an inoperable status for up to 2 hours for required surveillance provided that the other trip system is OPERABLE.

(b) This function shall be automatically bypassed when operating below the appropriate turbine first stage pressure setpoint of

(1) $\leq$ 26.9% of the value of turbine first-stage pressure at valves wide open (VWO) steam flow when operating with rated feedwater temperature of greater than or equal to 420°F; or

(2) $\leq$ 22.5% of the value of turbine first-stage pressure at VWO steam flow when operating with rated feedwater temperature between 370°F and 420°F.

These represent allowable setpoint values of turbine first-stage pressure equivalent to THERMAL POWER less than 40% of RATED THERMAL POWER.

TABLE 3.3.4.2-2

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Turbine Stop Valve - Closure	$\geq$ 40 psig*	$\geq$ 37 psig
2. Turbine Control Valve - Fast Closure	$\geq$ 44.3 psig*	$\geq$ 42 psig

*Initial setpoint. Final setpoint to be determined during startup test program. Any required change to this setpoint shall be submitted to the Commission within 90 days of test completion.

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	20 + 15, -0% of RATED THERMAL POWER	20 + 15, -0% of RATED THERMAL POWER
b. High Power Setpoint	≤ 70% of RATED THERMAL POWER	≤ 70% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux-Upscale		
1) During two recirculation loop operation		
a) Flow Biased	≤ 0.66 W+58%, with a maximum of	≤ 0.66 W+61%, with a maximum of
b) High Flow Clamped	≤ 108.0% of RATED THERMAL POWER	≤ 110.0% of RATED THERMAL POWER
2) During single recirculation loop operation:		
a) Flow Biased	≤ 0.66 W+34%	≤ 0.66 W+37%
b) High Flow Clamped	Not required OPERABLE	Not required OPERABLE
b. Inoperative	NA	NA
c. Downscale	≥ 4% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 1×10^5 cps	≤ 1.5×10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 0.7 cps	≥ 0.5 cps

TABLE 3.3.6-2 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 of full scale	≤ 110/125 of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 of full scale	≥ 3/125 of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	≤ 32 inches	≤ 33.5 inches
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 111% of rated flow	≤ 114% of rated flow
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	NA

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD PATTERN CONTROL SYSTEM				
a. Low Power Setpoint	NA	S/U[b], D[c], M[d]	Q	1, 2
b. High Power Setpoint	NA	S/U[b], D[c], M[d]	Q	1, 2
2. APRM				
a. Flow Biased Neutron Flux-Upscale	NA	W	W[f][g], SA	1
b. Inoperative	NA	S/U,W	NA	1, 2, 5
c. Downscale	NA	W	W[h], SA	1
d. Neutron Flux - Upscale, Startup	NA	S/U[b],M	Q	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U,W	NA	2, 5
b. Upscale	NA	S/U,W	Q	2, 5
c. Inoperative	NA	S/U,W	NA	2, 5
d. Downscale	NA	S/U,W	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U,W	NA	2, 5
b. Upscale	NA	S/U,W	Q	2, 5
c. Inoperative	NA	S/U,W	NA	2, 5
d. Downscale	NA	S/U,W	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	M	R	1, 2, 5*
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	M	Q	1
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	R	NA	3, 4

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. Within 7 days prior to startup.

c. Within 24 hours prior to control rod movement and as each power range above the RPCS low power setpoint is entered for the first time during any 24 hour period during power increase or decrease.

d. At least once per 31 days while operation continues within a given power range above the RPCS low power setpoint.

e. [Deleted]

f. 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 is greater than or equal to 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER.

g. This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

h. This calibration shall consist of verifying the trip setpoint only.

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

3/4.3.7 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.1 The radiation monitoring instrumentation channels shown in Table 3.3.7.1-1 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3.7.1-1.

ACTION:

a. With a radiation monitoring instrumentation channel alarm/trip setpoint exceeding the value shown in Table 3.3.7.1-1, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION required by Table 3.3.7.1-1.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.1 Each of the above required radiation monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the conditions and at the frequencies shown in Table 4.3.7.1-1.

3/4.3.10 NEUTRON FLUX MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.10 The APRM and LPRM* neutron flux noise levels shall not exceed three (3) times their established baseline value.

APPLICABILITY: OPERATIONAL CONDITION 1 with operation in Region I as specified in Figure 3.4.1.1-1.

ACTION:

a. With no established baseline flux noise levels, immediately initiate action to either reduce THERMAL POWER to within Region III as specified in Figure 3.4.1.1-1 or increase flow to within Region II as specified in Figure 3.4.1.1-1 within 2 hours.

b. With the flux noise levels greater than three (3) times their established baseline noise levels, initiate corrective action within 15 minutes to reduce the noise levels to within the required limits within 2 hours; if unsuccessful, either reduce THERMAL POWER to within Region III as specified in Figure 3.4.1.1-1 or increase flow to within Region II as specified in Figure 3.4.1.1-1 within the next 2 hours.

SURVEILLANCE REQUIREMENTS

4.3.10.1 The APRM and LPRM* neutron flux noise levels shall be determined to be less than or equal to the limit of Specification 3.3.10:

a. Within 2 hours after entering the applicable region, and

b. At least once per 8 hours, and

c. Within 30 minutes after completion of a change in THERMAL POWER of at least 5% of RATED THERMAL POWER.

The provisions of Specification 4.0.4 are not applicable.

4.3.10.2 Two loop baseline APRM and LPRM neutron flux noise levels shall be established at a point in Region II less than 60% of rated total core flow prior to operation in Region I of Figure 3.4.1.1-1 provided the baseline has not been established since the last CORE ALTERATION.

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*Detector A and C of one LPRM string per core octant plus detector A and C of one LPRM string in the central region of the core shall be monitored.

4.3.10.3 Single loop baseline APRM and LPRM neutron flux noise levels shall be established at a point in Region II less than 60% of rated core flow prior to single loop operation in Region I of Figure 3.4.1.1-1 provided the baseline has not been established since the last CORE ALTERATION; or in lieu of establishing single loop baseline data, the baseline established in 4.3.10.2 may be utilized for single loop operation in Region I of Figure 3.4.1.1-1.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 The reactor coolant recirculation system shall be in operation and not in Region IV as specified in Figure 3.4.1.1-1 with either:

a. Two recirculation loops operating with limits and setpoints per Specifications 2.1.2, 2.2.1, 3.2.1, and 3.3.6, or

b. A single recirculation loop operating with:

   1. A volumetric loop flow rate less than 44,600 gpm, and

   2. The loop recirculation flow control in the manual mode, and

   3. Limits and setpoints per Specifications 2.1.2, 2.2.1, 3.2.1, and 3.3.6.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. During single loop operation, with the volumetric loop flow rate greater than the above limit, immediately initiate corrective action to reduce flow to within the above limit within 30 minutes.

b. During single loop operation, with the loop flow control not in the manual mode, place it in the manual mode within 15 minutes.

c. With no reactor coolant system recirculation loops in operation, immediately initiate an orderly reduction of THERMAL POWER to within Region III as specified in Figure 3.4.1.1-1, and initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

d. During single loop operation, with temperature differences exceeding the limits of SURVEILLANCE REQUIREMENT 4.4.1.1.5, suspend the THERMAL POWER or recirculation loop flow increase.

e. With operation in Region IV as specified in Figure 3.4.1.1-1, initiate corrective action within 15 minutes to either reduce power to within Region III of Figure 3.4.1.1-1 or increase flow to within Region I or Region II of Figure 3.4.1.1-1 within 4 hours.

*See Special Test Exception 3.10.4.

LIMITING CONDITION FOR OPERATION (Continued)

f. With a change in reactor operating conditions, from two recirculation loops operating to single loop operation, or restoration of two loop operation, the limits and setpoints of Specifications 2.1.2, 2.2.1, 3.2.1, and 3.3.6 shall be implemented within 8 hours or declare the associated equipment inoperable (or the limits to be "not satisfied"), and take the ACTIONS required by the referenced specifications.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 The reactor coolant recirculation system shall be verified to be in operation and not in Region IV of Figure 3.4.1.1-1 at least once per 24 hours.

4.4.1.1.2 Each reactor coolant system recirculation loop flow control valve in an operating loop shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying that the control valve fails "as is" on loss of hydraulic pressure at the hydraulic unit, and

b. Verifying that the average rate of control valve movement is:

   1. Less than or equal to 11% of stroke per second opening, and

   2. Less than or equal to 11% of stroke per second closing.

4.4.1.1.3 During single loop operation, verify that the loop recirculation flow control in the operating loop is in the manual mode at least once per 8 hours.

4.4.1.1.4 During single loop operation, verify that the volumetric loop flow rate of the loop in operation is within the limit at least once per 24 hours.

4.4.1.1.5 During single loop operation, and with both THERMAL POWER less than 36% of RATED THERMAL POWER and the operating recirculation pump not on high speed, verify that the following differential temperature requirements are met within 15 minutes prior to beginning either a THERMAL POWER increase or a recirculation loop flow increase and within every hour during the THERMAL POWER or recirculation loop flow increase:

a. Less than 100°F, between the reactor vessel steam space coolant and the bottom head drain line coolant, and

b. Less than 50°F, between the coolant of the loop not in operation and the coolant in the reactor vessel, and

c. Less than 50°F, between the coolant in the operating loop and the coolant in the loop not in operation.

SURVEILLANCE REQUIREMENTS (Continued)

The differential temperature requirements 4.4.1.1.5.b and c do not apply when the loop not in operation is isolated from the reactor pressure vessel.

4.4.1.1.6 The limits and setpoints of Specifications 2.2.1, 3.2.1, and 3.3.6 shall be verified to be within the appropriate limits within 8 hours of an operational change to either one or two loops operating.

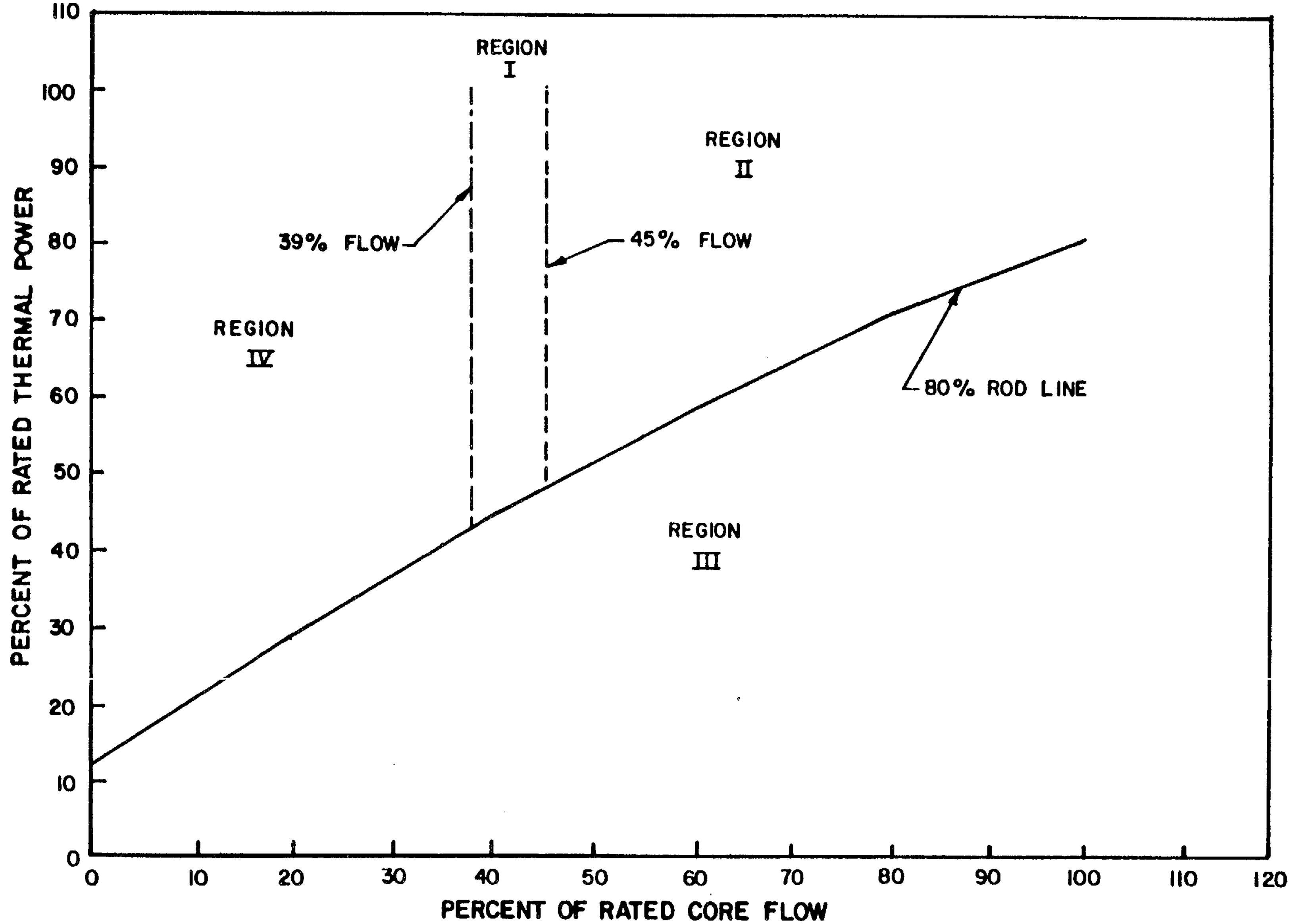


FIGURE 3.4.1.1-1 POWER-FLOW OPERATING MAP

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

4.4.1.2.1 Each of the above required jet pumps in an operating loop shall be demonstrated OPERABLE with THERMAL POWER in excess of 25% of RATED THERMAL POWER and at least once per 24 hours 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 diffuser-to-lower plenum differential pressure of any individual jet pump differs from established patterns by more than 10%.

4.4.1.2.2 The provisions of Specification 4.0.4 are not applicable provided the diffuser-to-lower plenum differential pressures of the individual jet pumps are determined to be within 50%* of the loop average within 72 hours after entering OPERATIONAL CONDITION 2 and at least once per 24 hours thereafter.

*Initial value. Final value to be determined during startup test program. Any required changes to the value shall be submitted to the Commission within 90 days of test completion.

RECIRCULATION LOOP FLOW

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation loop flow mismatch, when two loops are in operation, shall be maintained within:

a. 5% of rated recirculation flow with core flow greater than or equal to 70% of rated core flow.

b. 10% of rated recirculation flow with core flow less than 70% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With recirculation loop flows different by more than the specified limits, restore the recirculation loop flows to within the specified limit within 2 hours. If unsuccessful, either:

a. Shut down one recirculation loop and comply with the requirements of Specification 3.4.1.1, or

b. Be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation loop flow mismatch shall be verified to be within the limits at least once per 24 hours.

*See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.4 An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is less than or equal to 100°F,* and:

a. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is less than or equal to 50°F, or

b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is less than or equal to 50°F and the operating loop flow rate is less than or equal to 50% of rated loop flow.

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

ACTION:

With temperature differences and/or flow rates exceeding the above limits, suspend startup of any idle recirculation loop.

SURVEILLANCE REQUIREMENTS

4.4.1.4 The temperature differentials and flow rate shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.

*Below 25 psig, the temperature differential is not applicable.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 SHUTDOWN MARGIN

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

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

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

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

3/4.1.2 REACTIVITY ANOMALIES

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

3/4.1.3 CONTROL RODS

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

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

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

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

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than the Safety Limit during the limiting power transient analyzed in Section 15.4 of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than the Safety Limit. The occurrence of scram times longer than those specified should be viewed as an indication of a systematic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

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

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

3/4.2 POWER DISTRIBUTION LIMITS

BASES

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

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The MAPLHGR limits of Figure 3.2.1-1 are multiplied by the smaller of either the flow-dependent MAPLHGR factor ($MAPFAC_f$) or the power-dependent MAPLHGR factor ($MAPFAC_p$) corresponding to existing core flow and power state to ensure the adherence to fuel mechanical design bases during the most limiting transient. The maximum factor for single loop operation is 0.86.

$MAPFAC_f$'s are determined using the three-dimensional BWR simulator code to analyze slow flow runout transients. Two curves are provided for use based on the existing setting of the core flow limiter in the Recirculation Flow Control System. The curve representative of a maximum core flow limit of 107.0% is more restrictive due to the larger potential flow runout transient.

$MAPFAC_p$'s are generated using the same data base as the $MCPR_p$ to protect the core from plant transients other than core flow increases.

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

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

The calculational procedure used to establish the APLHGR limits 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 References 1 and 6. 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.

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

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

a. Input Change

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

b. Model Change

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

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

3/4.2.2 [DELETED]

Bases Table B 3.2.1-1

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

Plant Parameters;

Core THERMAL POWER .... 3993 MWt* which corresponds to 105% of rated steam flow

Vessel Steam Output .... 17.3×10^6 lbm/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 3.1 ft^2.

b. Small Breaks 0.1 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 Core	8 x 8 RP	13.4	1.4	$MCPR_f$**

A more detailed listing of input of each model and its source is presented in Section II of Reference 1 and subsection 6.3.3 of the FSAR.

*This power level meets the Appendix 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.

**During single loop operation, departure from nucleate boiling is assumed to occur 0.1 second following the LOCA regardless of initial MCPR.

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, 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 limiting transient yields the largest delta CPR. When added to the Safety Limit MCPR, the required operating limit MCPR of Specification 3.2.3 is obtained. The power-flow map of Figure B 3/4 2.3-1 defines the analytical basis for generation of the MCPR operating limits.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-2 and in Table 15.C.3-1 of Reference 5 that are input to a GE-core dynamic behavior transient computer program. The evaluation of transients during operation in the MEOD begins with the system initial parameters shown in Tables 15.D.4.-2 and 3 of Reference 7. The code used to evaluate pressurization events is described in NEDO-24154(3) and the program used in non-pressurization events is described in NEDO-10802(2). The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic TASC code described in NEDE-25149(4). The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the $MCPR_f$ and $MCPR_p$ is to define operating limits at other than rated core flow and power conditions.

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 reference core flow increase event used to establish the $MCPR_f$ is a hypothesized slow flow runout to maximum, that does not result in a scram from neutron flux overshoot exceeding the APRM neutron flux-high level (Table 2.2.1-1 item 2). The maximum runout flow value is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System. Two flow rates have been considered: 102.5% core flow and 107.0% core flow (for increased Core Flow operation). With this basis, the $MCPR_f$ curves are generated from a series of steady state core thermal hydraulic calculations performed at several core power and flow conditions along the steepest flow control line. In the actual calculations a conservative highly steep generic representation of the 105% steam flow rod-line flow control line has been used. Assumptions used in the original calculations of this generic flow control line were consistent with a slow flow increase transient duration of several minutes: (a) the plant heat balance was assumed

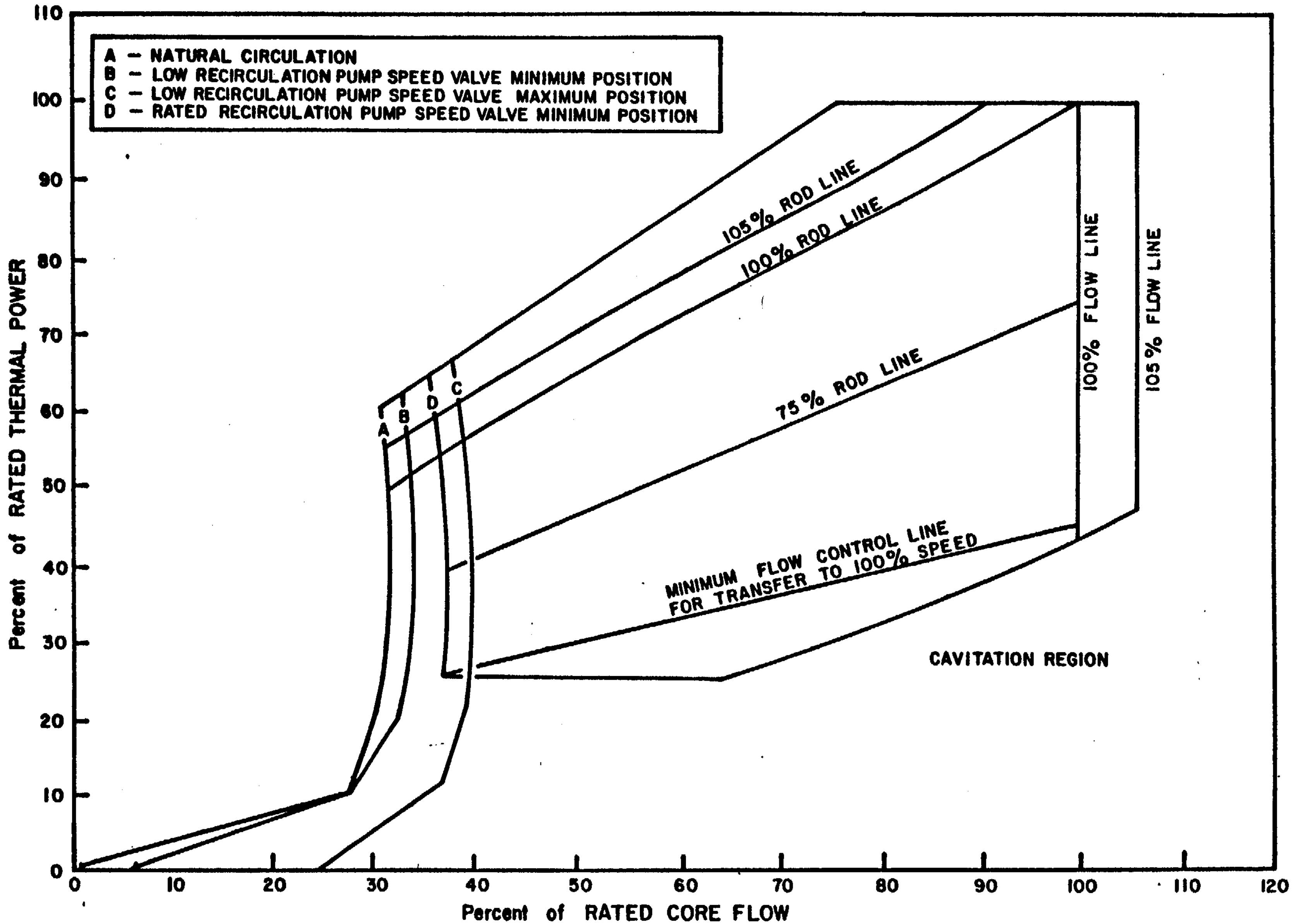


FIGURE B 3/4 2.3-1 POWER - FLOW OPERATING MAP

MINIMUM CRITICAL POWER RATIO (Continued)

to be in equilibrium, and (b) core xenon concentration was assumed to be constant The generic flow control line is used to define several core power/flow states at which to perform steady-state core thermal-hydraulic evaluations.

The first state analyzed corresponded to the maximum core power at maximum core flow (either 102.5% for Rated Core Flow operation or 107% of rated for Increased Core Flow operation) after the flow runout. Several evaluations were performed at this state iterating on the normalized core power distribution input until the limiting bundle MCPR just exceeded the safety limit Specification (2.1.2). Next, similar calculations of core MCPR performance were determined at other power/flow conditions on the generic flow control line, assuming the same normalized core power distribution. The result is a definition of the $MCPR_f$ performance requirement such that a flow increase event to maximum will not violate the safety limit. (The assumption of constant power distribution during the runout power increase has been shown to be conservative. Increased negative reactivity feedback in the high power limiting bundle due to doppler and voids would reduce the limiting bundle relative power in an actual runout.)

The $MCPR_p$ is established to protect the core from plant transients other than core flow increase including the localized rod withdrawal error event. Core power dependent setpoints are incorporated (incremental control rod withdrawal limits) in the Rod Withdrawal Limiter (RWL) System Specification (3.3.6). These setpoints allow greater control rod withdrawal at lower core powers where core thermal margins are large. However, the increased rod withdrawal requires higher initial MCPR's to assure the MCPR safety limit Specification (2.1.2) is not violated. The analyses that establish the power dependent MCPR requirements that support the RWL system are presented in GESSAR II, Appendix 15B. For core power below 40% of RATED THERMAL POWER, where the EOC-RPT and the reactor scrams on turbine stop valve closure and turbine control valve fast closure are bypassed, separate sets of $MCPR_p$ limits are provided for high and low core flows to account for the significant sensitivity to initial core flows. For core power above 40% of RATED THERMAL POWER, bounding power-dependent MCPR limits were developed. The abnormal operating transients analyzed for single loop operation are discussed in Reference 5. The current $MCPR_p$ limits were found to be bounding. No change to the operating MCPR limit is required for single loop operation.

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.

MINIMUM CRITICAL POWER RATIO (Continued)

During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate MCPR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met

MINIMUM CRITICAL POWER RATIO (Continued)

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 shifts while still allotting time for the power distribution to stabilize. The requirement for calculating LHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that LHGR will be known following a change in THERMAL POWER or power shape that could place operation exceeding a thermal limit.

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

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

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

5. GGNS Reactor Performance Improvement Program, Single Loop Operation Analysis, General Electric Final Report, February 1986.

6. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, Amendment 2, One Recirculation Loop Out-of-Service, NEDO-20566-2, Revision 1, July 1978.

7. General Electric Company, "Maximum Extended Operating Domain Analysis," March 1986.

RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION (Continued)

feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a closure sensor for each of two turbine stop valves provides input to one EOC-RPT system; a closure sensor from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 40% of RATED THERMAL POWER are annunciated in the control room. The automatic bypass setpoint is feedwater temperature dependent due to the subcooling changes that affect the turbine first-stage pressure-reactor power relationship. For RATED THERMAL POWER operation with feedwater temperature greater than or equal to 420°F, an allowable setpoint of $\leq$ 26.9% of control valve wide open turbine first-stage pressure is provided for the bypass function. This setpoint is also applicable to operation at less than RATED THERMAL POWER with the correspondingly lower feedwater temperature. The allowable setpoint is reduced to $\leq$ 22.5% of control valve wide open turbine first-stage pressure for RATED THERMAL POWER operation with a feedwater temperature between 370°F and 420°F. Similarly, the reduced setpoint is applicable to operation at less than RATED THERMAL POWER with the correspondingly lower feedwater temperature.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 190 ms, less the time allotted from start of motion of the stop valve or turbine control valve until the sensor relay contact supplying the input to the reactor protection system opens, i.e., 70 ms, and less the time allotted for breaker arc suppression determined by test, as correlated to manufacturer's test results, i.e., 50 ms, and plant pre-operational test results.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

The OPERABILITY of the control rod block instrumentation in OPERATIONAL CONDITION 5 is to provide diversity of rod block protection to the one-rod-out interlock.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION (Continued)

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that: (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980.

3/4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit.

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February, 1972.

3/4.3.7.4 REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

The OPERABILITY of the remote shutdown system instrumentation and controls ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

3/4.3.9 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety-related components, equipment or structures.

3/4.3.10 NEUTRON FLUX MONITORING INSTRUMENTATION

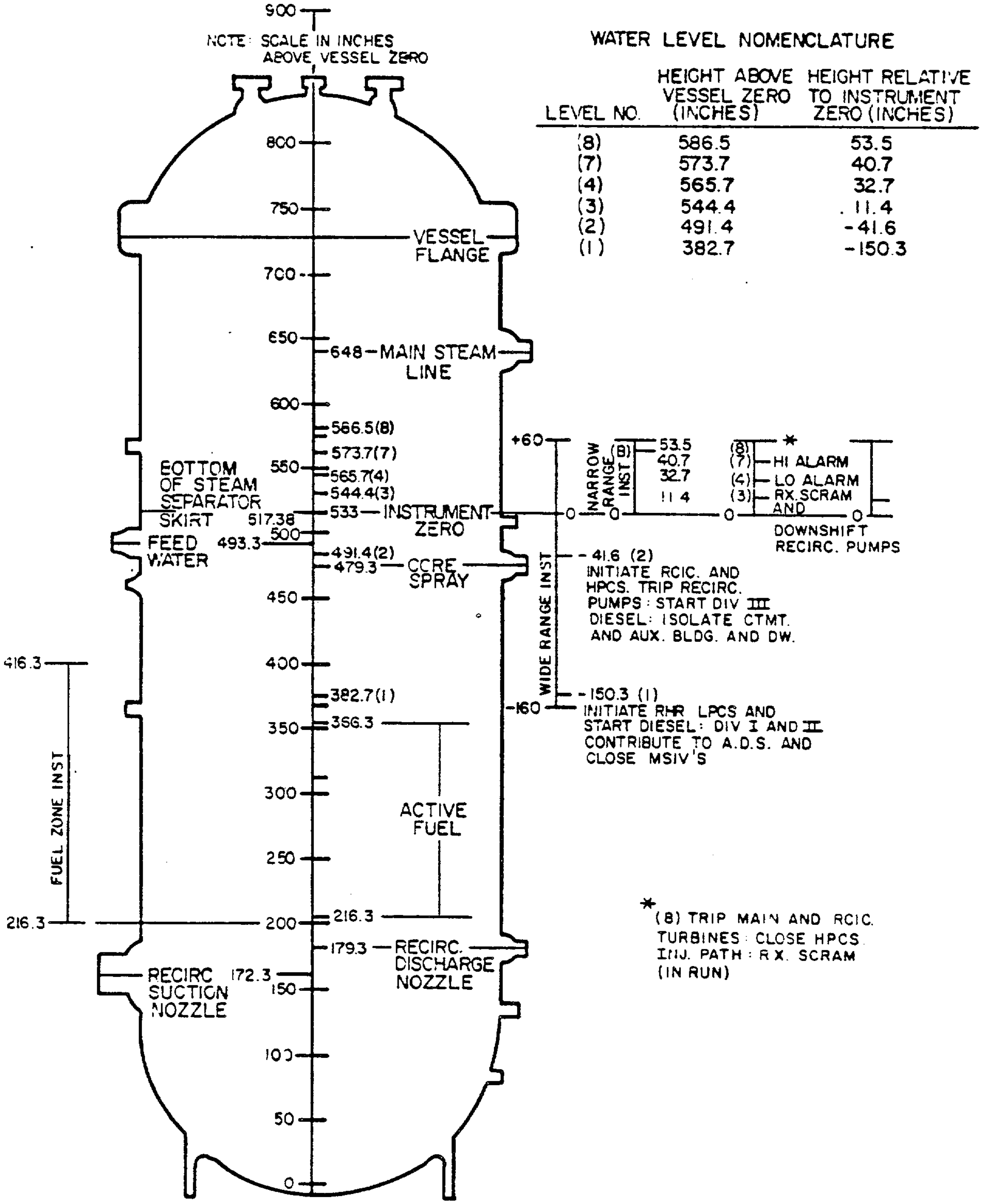
This specification is to ensure that neutron flux limit cycle oscillations are detected and suppressed.

Stability tests at operating plants were reviewed to identify a region of the operating map where surveillance should be performed. To account for variability, a conservative decay ratio of 0.6 was chosen as the basis for defining the region of potential instability. The resulting region corresponds to core flow less than 45% of rated and THERMAL POWER greater than the 80% rod line. The 80% rod line is illustrated in Figure 3.4.1.1-1.

Neutron flux noise limits are also established to ensure the early detection of limit cycle oscillations. Typical APRM neutron flux noise levels at up to 12% of rated power have been observed. These levels are easily bounded by values considered in the thermal/mechanical fuel design. Stability tests have shown that limit cycle oscillations result in peak-to-peak magnitude of 5 to 10 times the typical values. Therefore, actions taken to suppress flux oscillations exceeding three (3) times the typical value are sufficient to ensure early detection of limit cycle oscillations. The specification includes the surveillance requirement to establish the requisite baseline noise data and prohibits operation in the region of potential instability if the appropriate baseline data is unavailable.

WATER LEVEL NOMENCLATURE

LEVEL NO.	HEIGHT ABOVE VESSEL ZERO (INCHES)	HEIGHT RELATIVE TO INSTRUMENT ZERO (INCHES)
(8)	586.5	53.5
(7)	573.7	40.7
(4)	565.7	32.7
(3)	544.4	11.4
(2)	491.4	-41.6
(1)	382.7	-150.3



BASES FIGURE B 3/4 3-1 REACTOR VESSEL WATER LEVEL

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor core coolant recirculation loop inoperable has been evaluated and found to remain within design limits and safety margins provided certain limits and setpoints are modified. The "GGNS Single Loop Operation Analysis" identified the fuel cladding integrity Safety Limit, MAPLHGR limit and APRM setpoint modifications necessary to maintain the same margin of safety for single loop operation as is available during two loop operation. Additionally, loop flow limitations are established to ensure vessel internal vibration remains within limits. A flow control mode restriction is also incorporated to reduce valve wear as a result of automatic flow control attempts and to ensure valve swings into the cavitation region do not occur.

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

In accordance with BWR thermal hydraulic stability recommendations, operation above the 80% rod line with flow less than 39% of rated core flow is restricted.

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. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 100°F. During single loop operation, the condition may exist in which the coolant in the bottom head of the vessel is not circulating. These differential temperature criteria are also to be met prior to power or flow increases from this condition.

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.

3/4.4.1 RECIRCULATION SYSTEM (Continued)

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 the 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 1325 psig in accordance with the ASME Code. A total of 13 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 6 SRVs operating in the relief mode and 7 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 6 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 systems 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. Service sensitive reactor coolant system Type 304 and 316 austenitic stainless steel piping, i.e., those that are subject to high stress or that contain relatively stagnant, intermittent, or low flow fluids, requires additional surveillance and leakage limits.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO AMENDMENT NO.16 TO FACILITY OPERATING LICENSE NPF-29

GRAND GULF NUCLEAR STATION, UNIT 1

MISSISSIPPI POWER & LIGHT COMPANY

MIDDLE SOUTH ENERGY, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated May 2, 1986, (reference 1) the Mississippi Power and Light Company (MP&L) requested changes to the Grand Gulf Nuclear Station (GGNS) Unit 1 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 average power range monitor (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 (reference 2) 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 105% 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. The operation of GGNS with reduction in feedwater temperature while in the normal power/flow regime was evaluated by MP&L in accordance with 10 CFR 50.59. On the basis of this evaluation MP&L concluded that operation with up to a 50° F reduction in feedwater temperature (e.g., rated power feedwater temperature reduction from 420° F to 370° F) would not affect any safety limit and would not increase the consequences of any postulated accident. The conclusions of this evaluation and expansion of operating conditions to include this region were reported in reference 3. A brief discussion of the consequences of this reduction in feedwater temperature is also given in Section 15.1.7 "Feedwater Heaters Out of Service" which was included in the December 1985 update of the GGNS FSAR.

The average power range monitor (APRM) setdown requirement in the current GGNS Technical Specifications requires that the flow-biased APRM trips be 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. With the elimination of APRM setdown, a revision of the power dependent MCPR limit and development of new flow and power dependent MAPLHGR limits is provided to give fuel protection for power peaking effects at low core flows for those transients terminated by scram. The elimination of the APRM setdown does not affect the results of the loss of coolant accident (LOCA) calculations since setdown was not included in the LOCA calculations for the current Technical Specifications.

During the review of the MP&L submittal on MEOD, the staff requested additional information concerning Section 15.D.4.1 "Abnormal Operating Transients", Section 15.D.6 "Loss of Coolant Accidents" and Section 15.D.4.3. "Flow Runout Transients" of reference 2. This information was provided in reference 4. The analysis in the submittal also used the results of a generic analysis for the loss of feedwater heating (LOFWH) transient. This LOFWH analysis had not been reviewed by the staff. At the staff's request, the licensee provided a GGNS plant specific analysis of this event (reference 5). Finally, the staff requested a copy of the GE report on Feedwater Heating Out of Service (reference 6) which had been used by MPL in making the 10 CFR 50.59 evaluation of reference 3. This report was also referenced in the GE analysis (reference 2) that was provided in the MP&L submittal on MEOD.

By letters dated March 31, 1986, May 2, 1986 , and June 2, 1986 (reference 7), MP&L also requested changes to the Technical Specifications to (1) permit operation with one recirculation loop out of service, and (2) to include the (GE) Service Information Letter (SIL) No. 380, Revision 1, recommendations regarding thermal-hydraulic stability concerns for single loop operation. Presently, the GGNS operating license requires a unit to be in hot shutdown immediately if an idle recirculation loop cannot be returned to service within 12 hours. The recent resolution of Generic Issues B-19 and B-59 regarding thermal-hydraulic stability has provided a basis to permit operation in the single loop mode with appropriate restrictions relating to stability concerns (references 8 and 9). GE, in SIL No. 380 Revision 1, addressed these concerns by providing the boiling water reactor licensees generic guidance for actions which suppress thermal-hydraulic instability induced neutron flux oscillations. The licensee has proposed Technical Specifications in accordance with the guidance provided by GE in SIL No. 380, Revision 1. Specifically, the following changes are requested by the licensee:

(1) Revision of the Technical Specifications for average power range monitor (APRM) flux scram trip and rod block settings, an increase in the safety limit Minimum Critical Power Ratio (MCPR) value, and a revision to the allowable Average Planar Linear Heat Generation Rate (APLHGR) values.

(2) Incorporation of requirements in the Technical Specifications which should result in the detection and suppression of thermal-hydraulic instability induced neutron flux oscillations if they should occur.

2.0 EVALUATION

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

2.1.1 Operation in the MEOD with Normal Feedwater Temperatures

The General Electric Company analysis of reference 2, which was provided by MP&L 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 normal feedwater temperatures. This evaluation included consideration of abnormal operating transients, LOCAs, containment pressures, load impact on vessel internals, flow induced vibration, anticipated transients without scram (ATWS), and fuel mechanical performance.

Abnormal Operational Transients

All abnormal operational transients of Chapter 15 of the FSAR were considered for operation in the MEOD. A bounding analysis was performed using a standard BWR/6 plant at the end of an equilibrium cycle with a highly enriched GE6 fuel type. The transients investigated were generator load rejection without turbine bypass (LRNBP), feedwater controller failure to maximum demand (FWCF), cold loop startup (CLDLP) and flow control valve opening (FCVO). It was concluded that the current power dependent MCPR limit ($MCPR_p$) bounded these cases in the MEOD. Additional GGNS plant specific calculations were made for the LRNBP and FWCF transients to demonstrate that the bounding analysis for the standard plant equilibrium cycle is bounding for GGNS. At the staff's request, MP&L also provided a GGNS plant specific analysis of the loss of feedwater heating (LFWH) transient in the MEOD. This analysis indicated that the LFWH transient is less severe than the FWCF transient. Finally, it is noted that the rod withdrawal error (RWE) analysis in Chapter 15 of the GGNS FSAR included the MEOD region. Hence, the current Technical Specifications power dependent MCPR limit is protection against the RWE for operation in the MEOD. From the above considerations, it was concluded that the current power dependent MCPR is adequate for operation in the MEOD. We find this acceptable.

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 with approved methods to account for initial operation at low flows and a higher power rod line of the ELLR. Two new $MCPR_f$ relations were developed for two settings of the core flow limiter giving maximum core flows of 102.5 and 107.0 percent of rated flow. We find this acceptable.

Thermal-Hydraulic Stability

The staff has completed the generic review related to the thermal-hydraulic stability of BWR cores. In the evaluation report (reference 12), the staff concluded that GE fuel designs, including those fuels loaded in the GGNS core, meet the stability criteria set forth in 10 CFR Part 50, General Design Criteria 10 and 12, provided that the BWR has in place operating procedures and Technical Specifications which are consistent with the recommendations of GE SIL-380 to assure detection and suppression of global and local instabilities. This evaluation considered operation in the MEOD with reduction in feedwater heating. Since the licensee is implementing the SIL-380 recommendations, we conclude that this concern is satisfactorily resolved for GGNS during cycle 1 for operation in the MEOD with reduced feedwater heating.

LOCA Analysis

At the staff's request, the licensee submitted a revised justification of the consequences of a LOCA in the MEOD, in reference 4. The 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 GGNS FSAR and that the requirements of 10 CFR 50.46 are satisfied. We find this acceptable.

Containment Pressure Response

A conservative containment analysis for operation in the MEOD with FWHOS resulted in a peak drywell pressure 1.3 psi higher than the value of 22.0 psi in Chapter 6 of the FSAR. However, this is still below the design limit of 30 psig. It was also stated that the peak suppression pool temperatures, chugging loads, condensation oscillations and pool swell bounding loads were all found to be bounded by the rated power analysis in Chapter 6. We find this acceptable.

Load Impact on Internals and Flow-Induced Vibration

In reference 2 it was stated that the effects of increased reactor internal pressure differences, acoustic loads, flow-induced loads and fuel bundle lift forces were evaluated and that design limits were not exceeded. This evaluation included the effect of FWHOS. With respect to flow-induced vibrations, GGNS Unit 1 was the prototype BWR/6 251 plant for the testing to demonstrate that the flow-induced vibration response of the reactor internals is acceptable. In reference 11 the staff concluded that the tests demonstrate that the GGNS internals are adequately designed for flow induced vibration effects. We find this acceptable.

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 1262 psig, well below the ASME code limit of 1375 psig. We find this acceptable.

2.1.2 Operation in the MEOD with Reduced Feedwater Heating

The safety impact of operating GGNS Unit 1 in the normal power/flow region at reduced feedwater temperatures was evaluated previously by MP&L (reference 3). On the basis of a General Electric Company analysis (reference 6), they concluded that a reduction in rated power feedwater temperature of up to 50° F (down from 420° F to 370° F) would not affect any safety limits and would not increase the consequences of any postulated accident. Hence they modified the GGNS procedures to expand the operational feedwater temperature band to include this 50° F reduction. This evaluation and action by MP&L was reported to NRR in reference 3 in accordance with 10 CFR 50.59 requirements.

The present submittal by MP&L requests approval for operation in the MEOD with this same reduction in feedwater temperature. In the General Electric Company analyses of reference 2 which were provided as justification for this request, it is concluded that for operation in the MEOD, a reduction in rated feedwater temperatures from 420° F to 370° F would not result in changes to the current MCPR and MAPLHGR limits. We find this acceptable.

2.1.3 Elimination of APRM Setdown

In the current GGNS Technical Specifications the flow-biased APRM trips are reduced (setdown) when the core maximum total peaking factor exceeds the design total peaking factor. The General Electric Company analysis (reference 2) supplied with the MP&L 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 were 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. It was concluded that current MAPLHGR limits protect against a LOCA even without APRM setdown since the current LOCA analyses do not take credit for setdown. The flow dependent MCPR limit is also not affected by elimination of APRM setdown since the design basis flow runout event is a slow flow/power increase not terminated by scram. However, 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.

2.2 Changes to Technical Specifications to Permit operation in the Maximum Extended Operating Domain

The following changes to the Technical Specifications are proposed to permit operation of GGNS Unit 1 in the MEOD with feedwater temperatures reduced up to 50° F and with elimination of APRM setdown:

(1) Table 2.2.1-1(2). The proposed increase of the flow biased APRM setpoint and allowable values of 16% for two loop operation is made to permit operation in the ELLR part of the MEOD. The GE analysis of reference 2 shows that operation in the MEOD would not exceed design limits. This is acceptable.

(2) Specification 3/4.2.1. The proposed change to this specification dealing with MAPLHGR limits results from the proposed elimination of APRM setdown in Specification 3/4.2.2. The current specifications provide for reduction in the flow-biased APRM trips when the core maximum total peaking factor exceeds the design total peaking factor. With the proposed elimination of APRM setdown, this peaking effect is covered by revision to the MAPLHGR limits. The revised limits are presented as graphs for both a flow dependent and power dependent MAPLHGR factor in Figures 3.2.1-2 and 3.2.1-3. As discussed in reference 2, the revised limits provide equal or increased margins to fuel integrity limits relative to those obtained with APRM setdown. We find the proposed changes acceptable.

(3) Specification 3/3.2.2. The proposed change is to eliminate this specification which involves the APRM setdown. As discussed under Specification 3/4.2.1, this proposed change is acceptable.

(4) Figures 3.2.3-1 and 3.2.3-2. The slow recirculation flow runout analysis of reference 2 for the proposed operation in the MEOD

results in new flow dependent MCPR limit curves. The new curves, shown in Figure 3.2.3-1, are slightly above the curve in the current Technical Specifications. The new set of power dependent MCPR limits shown in Figure 3.2.3-2 result from elimination of APRM setdown. The new limits include the effect of operation at feedwater temperature reductions up to 50° F. The operating limit MCPR at any power/flow condition is the larger of the new flow and power dependent values. We find the proposed changes acceptable.

(5) Table 3.3.1-1. The proposed change involves Note h of Table 3.3.1-1 which deals with bypassing the turbine stop valve closure and turbine control valve fast closure scram when thermal power is less than 40% of rated thermal power. The high pressure turbine first stage pressure is used to measure thermal power. New setpoints for the first stage pressure are provided for feedwater temperatures greater than 420° F and between 370° F and 420° F. The proposed change clarifies the current requirement and incorporates the results of the startup tests on power versus first stage pressure. We find the changes acceptable.

(6) Table 3.3.4.2-1. This revision to Note b of Table 3.3.4.2-1 is a proposed change which clarifies the current requirement and is based on results of the startup test on thermal power versus first stage pressure. This change for the end of cycle recirculation pump trip (EOC-RPT) is identical to that for the turbine stop valve and turbine control valve fast closure scram (see item 5) and is acceptable.

(7) Table 3.3.6-2. The proposed changes are made to permit operation in the MEOD. Increase in the APRM flow-biased rod block setpoint is proposed to permit operation in the ELLR. However, the high flow clamp to this setpoint value for rod block is added to maintain the same clamp setpoint of 108% as in the current Technical

Specifications. These changes provide the same margin between the simulated thermal power monitor scram and rod block setpoints as the current Technical Specifications. In addition, the recirculation flow-high rod block setpoint is increased from 108% to 111% to decrease unnecessary rod block alarms when operating in the ICFR. We find these changes acceptable.

(8) Administrative changes made to the Technical Specifications include elimination of references to the deleted Specification 3.2.2 in Bases 2.2.1 and 3/4.2.2, Specifications 3/4.2.2, and 3/4.1.1, Tables 4.3.1.1-1, 3.3.4.2-1, 3.3.6-2 and 4.3.6-1, the Index, and Figures 3.2.3-1 and 3.2.3-2.

(9) Bases 2.2.1, 3/4.2.1, 3/4.2.2, 3/4.2.3, Bases Table B 3.2.1-1 and Bases Figure B 3/4.2.3-1. Proposed changes to the Bases were those modifications and additions provided to reflect the changes to the Technical Specifications needed for the proposed operation in the MEOD. We find these changes acceptable.

2.3 Single Loop Operation (SLO)

2.3.1 Accidents (Other Than Loss of Coolant Accidents) and Transients Affected by One Recirculation Loop Out of Service

One Pump Seizure Accident

A plant specific analysis was performed for this event. The analysis has shown that the event results in a MCPR value significantly above the SLO safety limit MCPR.

2.3.2 Abnormal Operational Transients

The licensee discussed the effects of SLO on the course of operational transients. Pressurization and cold water increase events, as well as rod withdrawal error, were addressed. Flow decrease is covered by the pump seizure accident already described. The results of calculations for the limiting event for each category were also presented. Initial operating conditions were conservatively assumed to be 70.6% of rated power and 54.1% of rated core coolant flow.

a) Pressurization Events

The limiting pressurization event is the generator load rejection without bypass transient. For single loop operation, the licensee has calculated that the maximum vessel pressure is 1179 psig and the MCPR is 1.41. Each of the values satisfies its respective safety limit.

b) Cold Water Increase Event

The limiting cold water increase event is the feedwater controller failure to maximum demand transient. The reactor is conservatively assumed to be in single loop operation at 70.6% of rated power and 54.1% of rated core coolant flow when failure of the feedwater control system instantaneously increases the feedwater flow to the pump runout capacity of 130% of rated flow. The peak pressure is calculated to be 1059 psig and the MCPR is 1.34, each satisfying its respective safety limit.

c) Rod Withdrawal Error

The rod withdrawal error at rated power is given in the FSAR for the initial core and in cycle dependent reload supplemental submittals.

These analyses are performed to demonstrate that, even if the operator ignores all instrument indications and the alarms which could occur during the course of the transient, the rod block system will stop rod withdrawal at a minimum critical power ratio which is higher than the fuel cladding integrity safety limit. Correction of the rod block equation for single-loop operation assures that the MCPR safety limit is not violated.

One-pump operation results in backflow through 12 of 24 jet pumps while flow is being supplied to the lower plenum from the active jet pumps. Because of this backflow through the inactive jet pumps, the present rod-block equation and APRM settings must be modified. The licensee has modified the two-pump rod block equation and APRM settings that exist in the Technical Specification for one-pump operation and the staff has found them acceptable.

The staff finds that one loop transients and accidents other than LOCA, are bounded by the two loop operation analyses and are therefore acceptable. Loss of coolant accidents are discussed in Section 2.3.4 below.

2.3.3 MCPR Uncertainties

For single-loop operation, the MCPR fuel cladding integrity safety limit is increased by 0.01 to account for increased uncertainties in the total core coolant flow and Traversing In-core Probe (TIP) readings. The limiting transients were analyzed to verify that there is more than enough margin during SLO to compensate for this increase in safety limit.

A feedwater controller failure initiating at 70.6% of rated power and 54.1% of rated core coolant flow results in a transient delta critical power ratio (CPR) of 0.07. A generator load rejection with bypass failure initiated at the same initial conditions resulted in a transient delta CPR of 0.002. Since the initial operating limit in SLO is equal to or greater than that at rated power and the transient delta CPR is less in SLO, there is more margin to the safety limit in SLO than at rated power. For single loop operation at lower power or at lower core coolant flows, the steady-state operating MCPR limit is established by the $MCPR_p$ and $MCPR_f$ curves. This ensures the 99.9% statistical limit requirement is always satisfied for any postulated abnormal operational occurrence. Since the maximum core coolant flow runout during single loop operation is only about 54.1% of rated core coolant flow, the current flow dependent MCPR limits which are generated based on the flow runout up to rated core flow are also adequate to protect the flow runout events during single loop operation. Since the SLO transient analysis is bounded by the two-loop transient analysis, power dependent MCPR curves used for two-loop operation are also applicable for SLO. The staff finds the licensee's consideration of MCPR uncertainties to be acceptable.

2.3.4 Loss of Coolant Accident (LOCA)

The licensee has performed analyses of a spectrum of recirculation suction line breaks for single loop operation conditions. The licensee states that evaluation of these calculations, which are performed according to the procedure outlined in NEDO-20556-2, Rev. 1, indicates that a multiplier of 0.86 should be applied to the MAPLHGR limits for single loop operation of GGNS. This evaluation methodology has been approved by the staff (reference 10).

The principal LOCA concern associated with single-loop operation is the possibility of the LOCA break occurring in the operating loop, in which case there is no coastdown of an intact loop recirculation pump to sustain jet pump and core coolant flow during the early portion of the system blowdown. An early boiling transition may result from this early loss of coolant flow capability.

To account for this possibility, GE derived a single-loop operation MAPLHGR multiplier of 0.86 to be used with calculated two-loop MAPLHGR limits during single-loop operation. The analyses which determined this multiplier assumed a near instantaneous boiling transition (0.1 second) even though a longer boiling transition time may have been calculated using approved models. This assumption is very conservative when applied to the GE fuel.

The MAPLHGR limits developed for MEOD, FWHOS and the APRM setdown elimination are more conservative than those for which SLO was analyzed. The flow dependent MAPLHGR reduction factor is clamped for SLO flows above 59% of rated core coolant flow in order to limit the factor to its analyzed value of 0.86 for SLO. Similarly, the power-dependent MAPLHGR reduction factor is clamped at the 70% of rated power value for SLO, because SLO is only permitted up to this power level. At 70% power, power dependent MAPLHGR reduction factor is 0.845 which is conservatively below the factor of 0.86. The staff finds that the consequences of LOCA for SLO are acceptable with the proposed reduction in MAPLHGR factors.

2.3.5 Thermal-Hydraulic Stability in Single Loop Operation

We have evaluated the licensee's proposed Technical Specification changes to assure that the changes provide adequate detection and suppression of potential thermal-hydraulic instabilities.

GE recently presented the staff with stability test data which demonstrated the occurrence of limit cycle neutron flux oscillations at natural circulation and several percent above the rated control rod line. The oscillations were observable on the APRMs and were suppressed with control rod insertion. It was predicted that limit cycle oscillations would occur at the operating condition tested; however, the characteristics of the observed oscillations were different from those previously observed during other stability tests. Namely, the test data showed that some local power range monitor (LPRM) indications oscillated out of phase with the APRM signal and at amplitude as great as six times the core average. GE has prepared and released a service information letter, SIL No. 380, to alert the BWR owners of these new data and to recommend actions to avoid and control abnormal neutron flux oscillations.

The General Electric recommendations were reviewed by the staff and found to be prudent recommendations which provide adequate detection and suppression of potential thermal-hydraulic instabilities as required by General Design Criteria (GDC) 10 and 12 of 10 CFR Part 50. The staff compared these recommendations with the GGNS technical specifications for operation with a recirculation loop out of service. The staff found that the proposed changes are in conformance with the SIL No. 380, Revision 1, recommendations and are therefore acceptable.

2.3.6 Jet Pump Surveillance

Some general questions have arisen regarding the adequacy of surveillance methods which have been used in some plants to monitor jet pump operability during SLO. These methods are used in accordance with NUREG/CR-3052 to close out problems presented in IE Bulletin 80-07, "BWR Jet Pump Assembly Failure." Since all the hold down beams at GGNS Unit 1 are an improved, acceptable design, the questions regarding SLO jet pump surveillance adequacy are not applicable.

2.3.7 SUMMARY ON SINGLE LOOP OPERATION

The staff concludes for GGNS Unit 1 that with the provisions given below, transient and accident bounds will not be exceeded during SLO operation.

1. Minimum Critical Power Ratio (MCPR) Safety Limit will be Increased by 0.01 to 1.07

The MCPR Safety Limit will be increased by 0.01 to account for increased uncertainties in traveling in-core probe (TIP) readings. The licensee has determined that the change conservatively bounds the uncertainties introduced by single loop operation.

2. Minimum Critical Power Ratio (MCPR) Limiting Condition for Operation (LCO)

The licensee proposed that the operating limit MCPR be established by the $MCPR_p$ and $MCPR_f$ curves. This LCO is acceptable.

3. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Limits will be Reduced by Appropriate Multipliers

The licensee proposed to reduce the flow dependent MAPLHGR by a factor of 0.86 and the power dependent MAPLHGR by a factor of 0.84 for Single Loop Operation. These reductions are acceptable.

4. The APRM Scram and Rod Block Setpoints will be Reduced

The licensee proposed to modify the two loop APRM Scram, Rod Block and Rod Block Monitor (RBM) setpoints to account for back flow through half the jet pumps. These setpoint equations will be changed in the GGNS TS. The changes are similar to other plant TS changes and are acceptable to the staff.

5. The Recirculation Control will be in Manual Control

The licensee proposed to operate the recirculation system in the manual mode to eliminate the need for control system analyses and to reduce the effects of potential flow instabilities. This change is acceptable.

2.4 Changes to Technical Specifications to allow Operation with One Recirculation Loop out of Service

(1) Specification 2.1.2 and Bases 2.0. The MCPR Safety Limit has been increased by 0.01 to 1.07 for SLO to compensate for the uncertainties introduced by SLO. This change is acceptable.

(2) Table 2.2.1-1. APRM flow biased scram function equations have an added term to account for the difference between single and two loop recirculation pump (drive) flow for the same core coolant flow. This adjustment accounts for the difference between actual and indicated coolant flow and preserves the original relation between limits and effective drive flow. These changes are acceptable.

(3) Bases Table B2.1.2-1

The standard deviations for total core flow and TIP readings are increased to 6% and 6.8% respectively to account for uncertainties during SLO. This change is acceptable.

(4) Specification 3/4.2.1. MAPLHGR will be multiplied by the smaller of either the flow-dependent MAPLHGR factor ($MAPFAC_f$) of Figure 3.2.1-2, or the power dependent MAPLHGR factor ($MAPFAC_p$) of Figure 3.2.1-3. This is acceptable.

(5) Table 3.3.6-2. APRM rod block (flow biased) equations have an added term to account for the difference between single and two loop drive flow for the same core flow. This adjustment preserves the original relation between actual and indicated flow and preserves the original relation between limits and effective drive flow. These changes are acceptable.

(6) Specification 3/4.3.10. This specification will assure that neutron flux limit cycle oscillations are detected and suppressed. This new specification is added to implement the guidance of SIL 380 to detect and suppress limit cycle power oscillations in the high power/low flow region of the power-flow map. The LIMITING CONDITION FOR OPERATION (LCO), APPLICABILITY, ACTION, and SURVEILLANCE REQUIREMENTS in the Technical Specifications are consistent with the recommendations in SIL 380, Rev-1. This is acceptable.

(7) Specification 3/4.4.1. This section is modified to permit operation with either one or two loops in operation. The LCO is expanded to address operation while one recirculation loop is out of service. This change is based on and justified by the GE analysis of SLO and is acceptable.

(8) Specification 4.4.1.2.1. Jet pump surveillance is only required for the operating loop. This is acceptable as described in section 2.3.6 of this SER.

(9) Specification 3.4.1.3. The LCO is changed to reflect that recirculation loop flow mismatch is only of concern when both recirculation loops are in operation. This is acceptable.

2.5 REFERENCES

1. Letter from O. D. Kingsley, Jr., Mississippi Power & Light Company, to H. Denton, NRC, "Proposed Amendment to the Operating License (PCOL-86-07) - Maximum Extended Operating Domain", May 2, 1986.

2. General Electric Company Report "GGNS Maximum Extended Operating Domain Analysis" March, 1985.

3. Letter from O. D. Kingsley, Jr., Mississippi Power & Light Company to J. N. Grace, NRC, Region II, November 15, 1985.

4. Letter from O. D. Kingsley, Jr., Mississippi Power & Light Company, to H. Denton, NRC, "Addendum to MEOD Submittal", July 11, 1986.

5. Letter from O. D. Kingsley, Mississippi Power & Light Company, to H. Denton, NRC, "Addendum to MEOD Submittal", June 9, 1986.

6. General Electric Company Report "GGNS Feedwater Heater(s) Out of Service Analysis", March, 1986.

7. Letters from O. D. Kingsley, Jr., Mississippi Power & Light Company, to H. Denton, NRC, AECM-86/0092, AECM-86/0129, AECM-86/0160, dated March 31, May 2, June 2, 1986 respectively.

8. Generic Letter No. 86-02 "Technical Resolution of Generic Issue B-19 Thermal Hydraulic Stability," January 23, 1986.

9. Generic Letter No. 86-09 "Technical Resolution of Generic Issue B-59 (N-1) Loop Operation in BWRs and PWRs," March 31, 1986.

10. Letter from H. N. Berkow (NRC) to J. F. Quirk (GE) dated March 5, 1986. Acceptance for Referencing of Licensing Topical Report NEDO-20566-2, Rev-1, "General Electric Analytical Model for Loss of Coolant Accident Analysis in Accordance with 10 CFR 50 Appendix K, Amendment No. 2, One Recirculation Loop-out-of service."

11. Memorandum from G. Lainas to W. Butler, "Grand Gulf Unit 1 Reactor Internals Vibration Measurements and Inspection Program", May 28, 1986.

12. Letter from L. S. Rubenstein to D. Crutchfield, "Safety Evaluation of GE Topical Report NEDE-24011 (GESTAR) Amendment 8", April 17, 1985.

3.0 ENVIRONMENTAL CONSIDERATION

The amendment involves a change to requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR 20 and changes to the surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (51 FR 24258) on July 2, 1986, and consulted with the state of Mississippi. No public comments were received, and the state of Mississippi did not have any comments.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security nor to the health and safety of the public.

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Dated: August 15, 1986