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

during two loop operation

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

the above limits

With MCPR less than ~~1.06~~ 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.

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.

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TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-High		
~~1) Flow Biased~~	~~≤ 0.66 W+48%, with a maximum of~~	~~≤ 0.66 W+51%, with a maximum of~~
~~2) High Flow Clamped~~	~~≤ 111.0% of RATED THERMAL POWER~~	~~≤ 113.0% of RATED THERMAL POWER~~
INSERT FOR TABLE 2.2.1-1 ITEM 2.b		
c. Neutron Flux-High	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1064.7 psig	≤ 1079.7 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 11.4 inches above instrument zero*	≥ 10.8 inches above instrument zero*
5. Reactor Vessel Water Level-High, Level 8	≤ 53.5 inches above instrument zero*	≤ 54.1 inches above instrument zero*
6. Main Steam Line Isolation Valve - Closure	≤ 6% closed	≤ 7% closed
7. Main Steam Line Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background
8. Drywell Pressure - High	≤ 1.23 psig	≤ 1.43 psig
9. Scram Discharge Volume Water Level - High	~~≤ 60% of full scale~~	~~≤ 63% of full scale~~ Submittal on Jan 29, 1986
a. Transmitter/Trip Unit	≤ 60% of full scale	≤ 63% of full scale
b. Float Switch	≤ 64"	≤ 65"
10. Turbine Stop Valve - Closure	≥ 40 psig**	≥ 37 psig
11. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 44.3 psig**	≥ 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.

INSERT for Table 2.2.1-1, item 2.b

2.b: 1)	During two recirculation loop operation:		
	a) Flow Biased b) High Flow Clamped	0.66 W+48%, with a maximum of 111.0% of RATED THERMAL POWER	0.66 W+51%, with a maximum of 113.0% of RATED THERMAL POWER
2)	During single recirculation loop operation:		
	a) Flow Biased	0.66 W+40%,	0.66 W+43%,
	b) High Flow Clamped	Not required operable	Not required operable

the applicable Safety Limit

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, for a step-back approach is used to establish a Safety Limit ~~such that~~ the MCPR ~~is not less than 1.06~~. MCPR greater than ~~1.06~~ 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 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.

The bases for the changes in uncertainties due to a single loop operation are given in the GGNS Single Loop Operation Analysis dated February, 1986.

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

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 during two loop operation or in Figure 3.2.1-2 during single loop operation.

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

ACTION:

With an APLHGR exceeding the applicable limits ~~of Figure 3.2.1-1~~, 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 : ~~determined from Figure 3.2.1-1.~~

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.

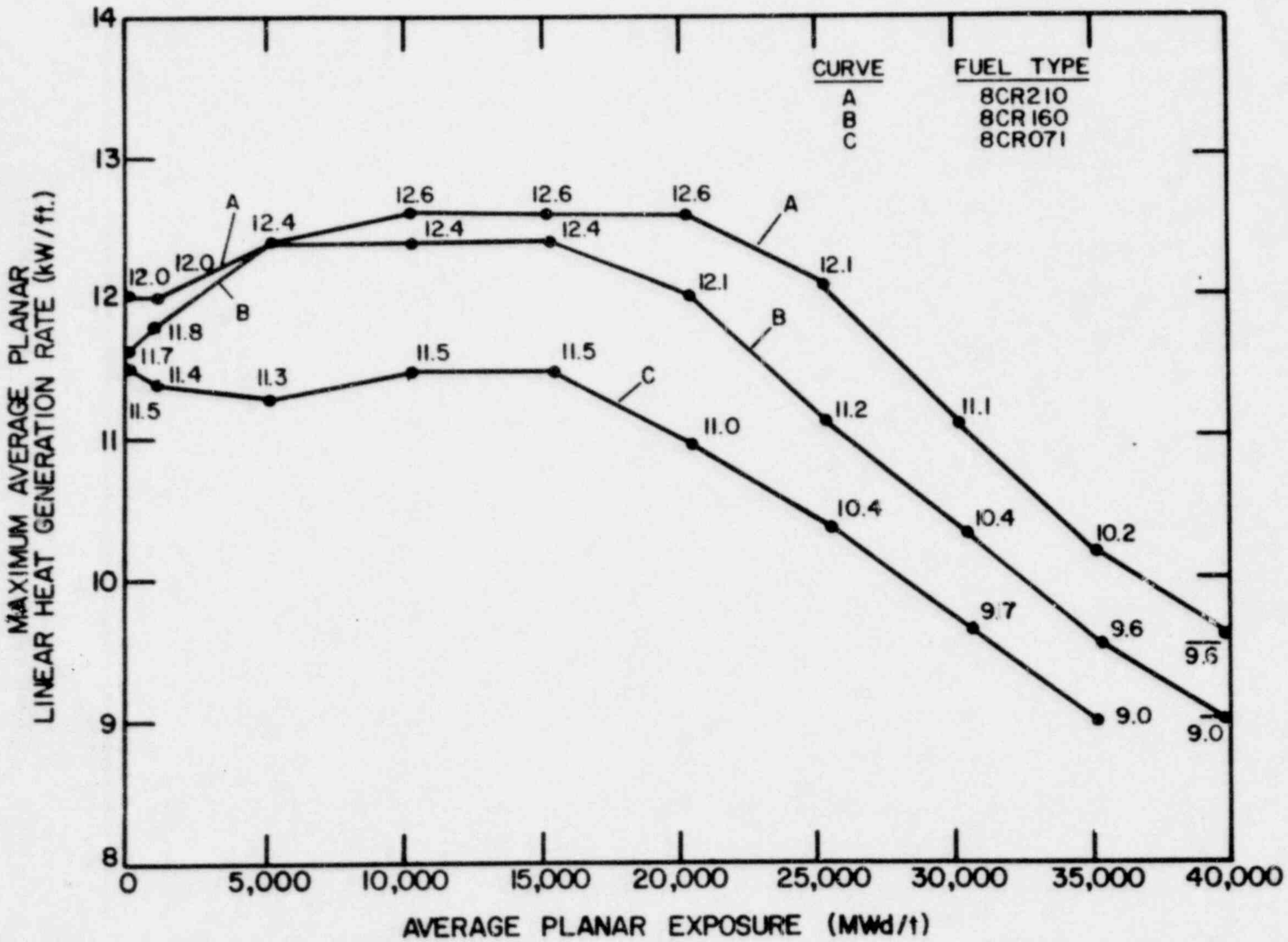

FIGURE 3.2.1-1 MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPES 8CR210, 8CR160 AND 8CR071

FOR TWO LOOP OPERATION

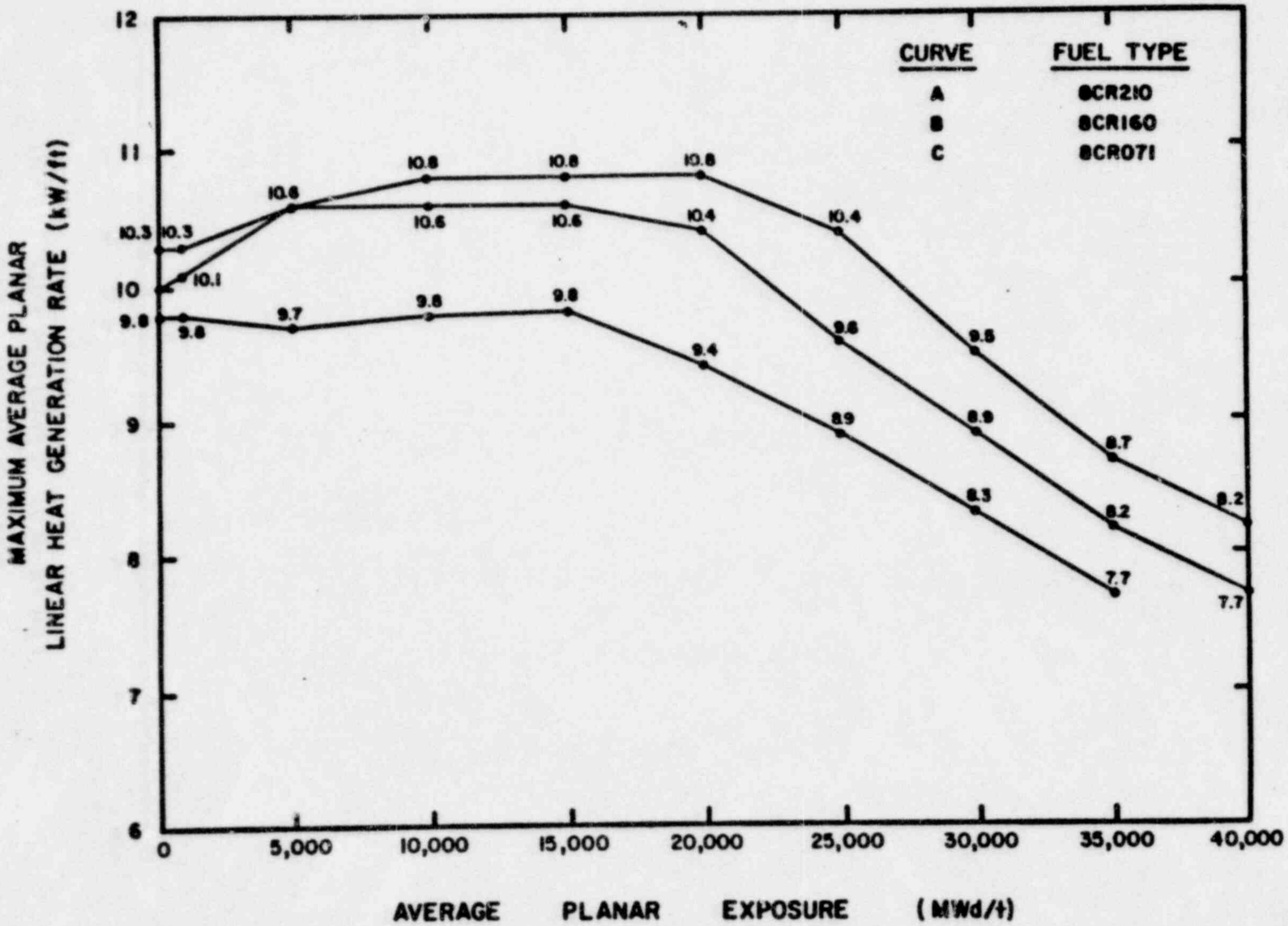

FIGURE 3.2.1-2 MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE FOR SINGLE LOOP OPERATION INITIAL CORE FUEL TYPES 8CR210, 8CR160 AND 8CR071.

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

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

a) During two recirculation loop operation:

Trip Setpoint	Allowable Value
$S \leq (0.66W + 48\%)T$	$S \leq (0.66W + 51\%)T$
$S_{RB} \leq (0.66W + 42\%)T$	$S_{RB} \leq (0.66W + 45\%)T$

b) During single recirculation loop operation:

Trip Setpoint	Allowable Value
$S \leq (0.66W + 40\%)T$	$S \leq (0.66W + 43\%)T$
$S_{RB} \leq (0.66W + 34\%)T$	$S_{RB} \leq (0.66W + 37\%)T$

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

W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 112.5 million lbs/hr.

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

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

ACTION:

With the APRM flow biased simulated thermal power-high scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the allowable value-column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and restore S and/or S_{RB} to within the required limits* within 8 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

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

a. At least once per 24 hours,

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

c. Initially and at least once per 12 hours when the reactor is operating with MFLPD greater than or equal to FRTP.

d. The provisions of Specification 4.0.4 are not applicable.

*With MFLPD greater than the FRTP during power ascension up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times MFLPD provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER, and a notice of adjustment is posted on the reactor control panel.

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		
1) During two loop operation	≤ (0.66W + 42%)T*	≤ (0.66W + 45%)T*
2) During single loop operation	≤ (0.66W + 34%)T*	≤ (0.66W + 37%)T*
a. Flow Biased Neutron Flux-Upscale	~~≤ (0.66 W + 42%)T*~~	~~≤ (0.66 W + 45%)T*~~
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
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	≤ 108% of rated flow	≤ 111% of rated flow
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	NA

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W) and the ratio of FRACTION of RATED THERMAL POWER to the MAXIMUM FRACTION of LIMITING POWER DENSITY (T factor). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

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 Establish two loop baseline APRM and LPRM neutron flux noise levels 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.

---

* 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 a. Establish single loop baseline APRM and LPRM neutron flux noise levels 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

b. In lieu of establishing single loop baseline data, the baseline established in 4.3.10.2 may be utilized for single loop operation in Regions I of Figure 3.4.1.1-1.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

INSERT 3/4.4-1

LIMITING CONDITION FOR OPERATION

~~3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.~~

~~APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.~~

~~ACTION:~~

~~a. With one reactor coolant system recirculation loop not in operation, immediately initiate an orderly reduction of THERMAL POWER to less than or equal to 80% of the 100% Rod Line as specified in Figure B 3/4 2.3-1, and be in at least HOT SHUTDOWN within the next 12 hours.~~

~~b. With no reactor coolant system recirculation loops in operation, immediately initiate an orderly reduction of THERMAL POWER to less than or equal to 80% of the 100% Rod Line as specified in Figure B 3/4 2.3-1, and initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.4.1.1.1 Both reactor coolant system recirculation loops shall be verified to be in operation at least once per 24 hours.~~

~~4.4.1.1.2 Each reactor coolant system recirculation loop flow control valve 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.~~

~~*See Special Test Exception 3.10.4.~~

Insert 3/4.4-1 page 1 of 2

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, 3.2.2 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, 3.2.2, 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.

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, 3.2.2, and 3.3.6, shall be implemented within 12 hours or declare the associated equipment inoperable, (or the limits to be "not satisfied") and take the ACTIONS required by the referenced specifications.

*See Special Test Exception 3.10.4.

Insert 3/4.4-1 page 2 of 2

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 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 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 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.

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, 3.2.2 and 3.3.6 shall be verified to be within the appropriate limits within 12 hours of an operational change to either one or two loops operating.

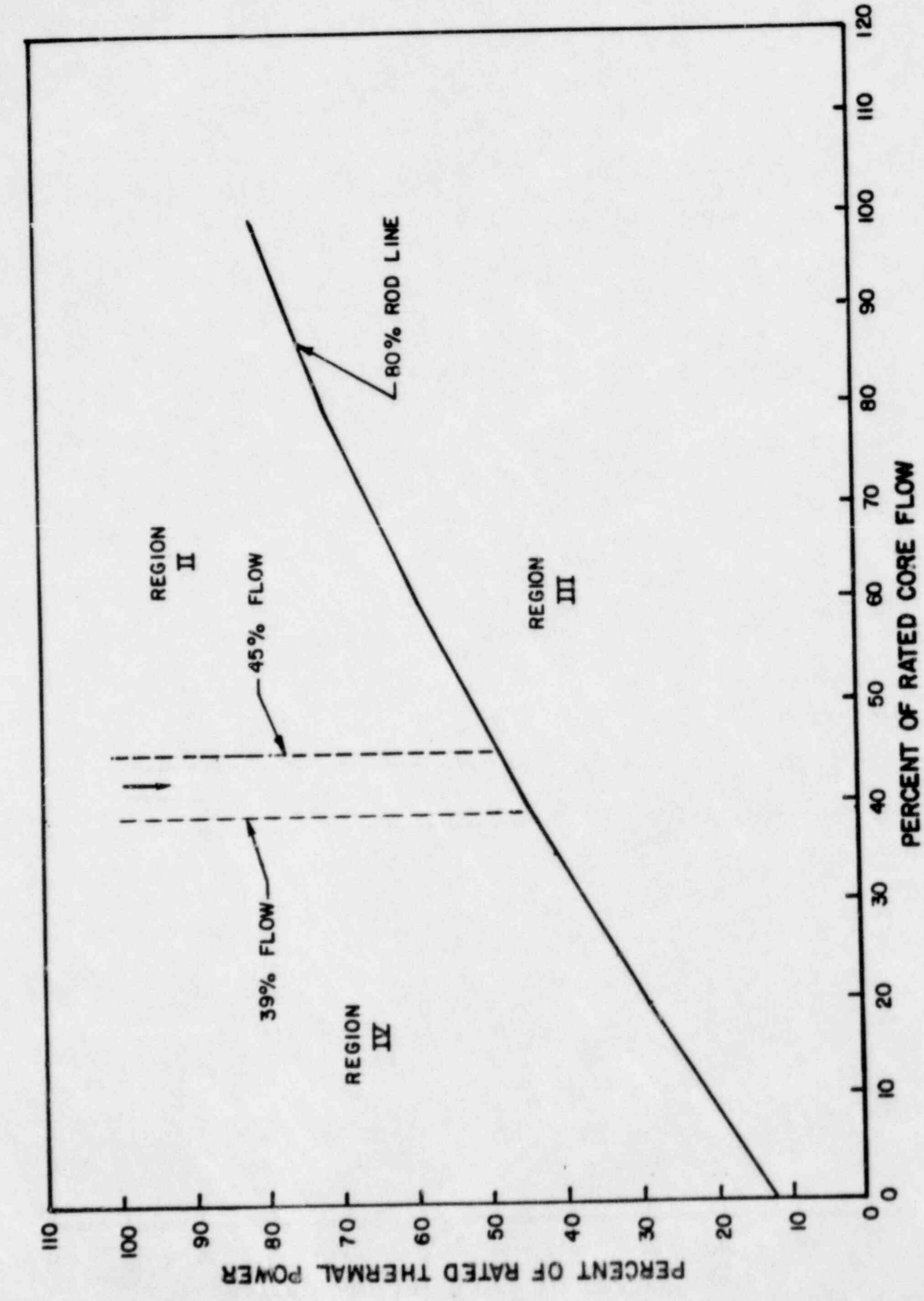

FIGURE 3.4.1.1-1 POWER - FLOW OPERATING MAP

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: ~~when both indicated recirculation loop flows are in compliance with Specification 3.4.1.3.~~

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 CONDITON 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.

REACTOR COOLANT SYSTEM

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, ~~either:~~

~~a.~~ Restore the recirculation loop flows to within the specified limit within 2 hours. ~~or~~ If unsuccessful, either:

a. ~~b.~~ Shutdown one ~~Declare the~~ recirculation loop and comply with the requirements of ~~with the lower flow not in operation and take the ACTION require by~~ Specification 3.4.1.1ℓ, 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.

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 ~~1.06~~ 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 ~~1.06.~~ the Safety Limit. The occurrence of scram times longer than those specified should be viewed as an indication of a systematic ~~systemic~~ 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 limiting value for APLHGR is shown in Figure 3.2.1-1~~.~~

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 stablize. 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.

The calculational procedure used to establish the APLHGR limits ~~shown on Figure 3.2.1-1~~ is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1 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.

for two loop operation and Figure 3.2.1-2 for single loop operation. The single loop operation limits are lower to account for earlier boiling transition assumed in the single loop operation LOCA analysis.

AVERAGE PLANAR LINEAR HEAT CENERATION 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 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased simulated thermal power-high scram setting and flow biased simulated thermal power-upscale control rod block functions of the APRM instruments must be adjusted to ensure that the MCPR does not become less than the Safety Limit ~~1.06~~ or that > 1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and MFLPD indicates a peak power distribution to ensure than an LHGR transient would not be increased in degraded conditions.

The daily requirement to verify the APRM control rod block and scram setpoints 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 verify the APRM setpoints 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 to verify the APRM setpoints once per 12 hours after initially determining MFLPD to be greater than FRTP ensures that the consequences of an LHGR transient would not be increased in degraded conditions.

AMENDMENT No. ___

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	1.17**

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

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR ~~of 1.06,~~ the required minimum operating limit MCPR of Specification 3.2.3 is obtained ~~and presented in Figure 3.2.3-1.~~ 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 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 $MCPR_p$ of Figures 3.2.3-1 and 3.2.3-2 is to define operating limits at other than rated core flow and power conditions. At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_f$ and $MCPR_p$ at the existing core flow and power state. The $MCPR_f$s are established to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

The 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). With this basis the $MCPR_f$ curve is 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. This corresponds to the 105% steamflow flow control line (Figure B 3/4 2.3-1). In the actual calculations a conservative highly steep generic representation of the 105% steamflow 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

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 (102.5% of rated) 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 (102.5%) 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. Since the severity of other (core-wide) transients at off-rated conditions is limited by the requirement to setdown the APRM flow biased simulated thermal power-high scram trip setpoint, Specification (3.2.2), the rod withdrawal error is the limiting transient and establishes $MCPR_p$ requirements. ← [ADD INSERT B2-6]

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

INSERT B2-6

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.

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 10CFR50, Appendix K-Amendment 2, One Recirculation Loop Out-of-Service, NEDO-20566-2, Revision 1, July, 1978.

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

INSERT 'D'

INSERT "D"

BASES

3/4.3.10 NEUTRON FLUX MONITORING INSTRUMENTATION

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

In order to identify a region of the operating map where surveillance should be performed, stability tests at operating plants were reviewed. 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 magnitudes 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.

3/4.4.1 RECIRCULATION SYSTEM

INSERT "A" BASES 3/4.4.1

Operation with one reactor core coolant recirculation loop inoperable ~~is prohibited until an evaluation of the performance of the ECCS during one loop operation has been performed, evaluated and determined to be acceptable.~~

During two loop operation, 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. 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.

INSERT "C" (NEW PARAGRAPH)

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. INSERT "B"

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

Electronic limiters exist in the position control loop of each flow control valve to limit the flow control valve stroking rate to 10±1% per second in 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.

INSERT "A"

BASES 3/4.4.1

...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 assure vessel internal vibration remains within limits. A flow control mode restriction is also incorporated to reduce valve wear due to automatic flow control attempts and to ensure valve swings into the cavitation region do not occur.

INSERT "B"

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.

INSERT "C"

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.