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* with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.06* 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.

*See Specification 3.4.1.1.2.a for single loop operation requirement.

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-Upscale, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-Upscale		
1) Flow Biased	≤ 0.58 W+59%#, with a maximum of	≤ 0.58 W+62%#, with a maximum of
2) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
c. Neutron Flux-Upscale	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1037 psig	≤ 1057 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 13.0 inches above instrument zero*	≥ 11.5 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 10% closed	≤ 11% closed
6. Main Steam Line Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background
7. Drywell Pressure - High	< 1.72 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High	≤ 88 gallons	≤ 88 gallons
9. Turbine Stop Valve - Closure	≤ 5.5% closed	≤ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 500 psig	≥ 460 psig
11. Reactor Mode Switch Shutdown Position	NA	NA
12. Manual Scram	NA	NA

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

See Specification 3.4.1.1.2.a for single loop operation requirement.

2.1 SAFETY LIMITS

BASES

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 such that the MCPR is not less than ~~1.06~~ the limit specified in Specification 2.1.2 for both GE and Exxon fuel. MCPR greater than ~~1.06~~ the specified 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. INSERT

2.1.1 THERMAL POWER, Low Pressure or Low Flow

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

The MCPR fuel cladding integrity Safety Limit assures that during normal operation and during anticipated operational occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling (ref. XN-NF-524(A)).

DELETE THIS TABLE

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
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	6.3
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.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

for GE fuel and AVERAGE BUNDLE EXPOSURE for Exxon fuel

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3.*

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3:

a. At least once per 24 hours,

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

* See Specification 3.4.1.1.2.a for single loop operation requirements.

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

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

Trip Setpoint#	Allowable Value#
$S \leq (0.58W + 59\%)T$	$S \leq (0.58W + 62\%)T$
$S_{RB} \leq (0.58W + 50\%)T$	$S_{RB} \leq (0.58W + 53\%)T$

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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 100 million lbs/hr,
(GE fuel) T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY. T is always less than or equal to 1.0.
T (Exxon fuel) = 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-upscale 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 adjust S and/or S_{RB} to be consistent with the Trip Setpoint value* within 2 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 the MFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale 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

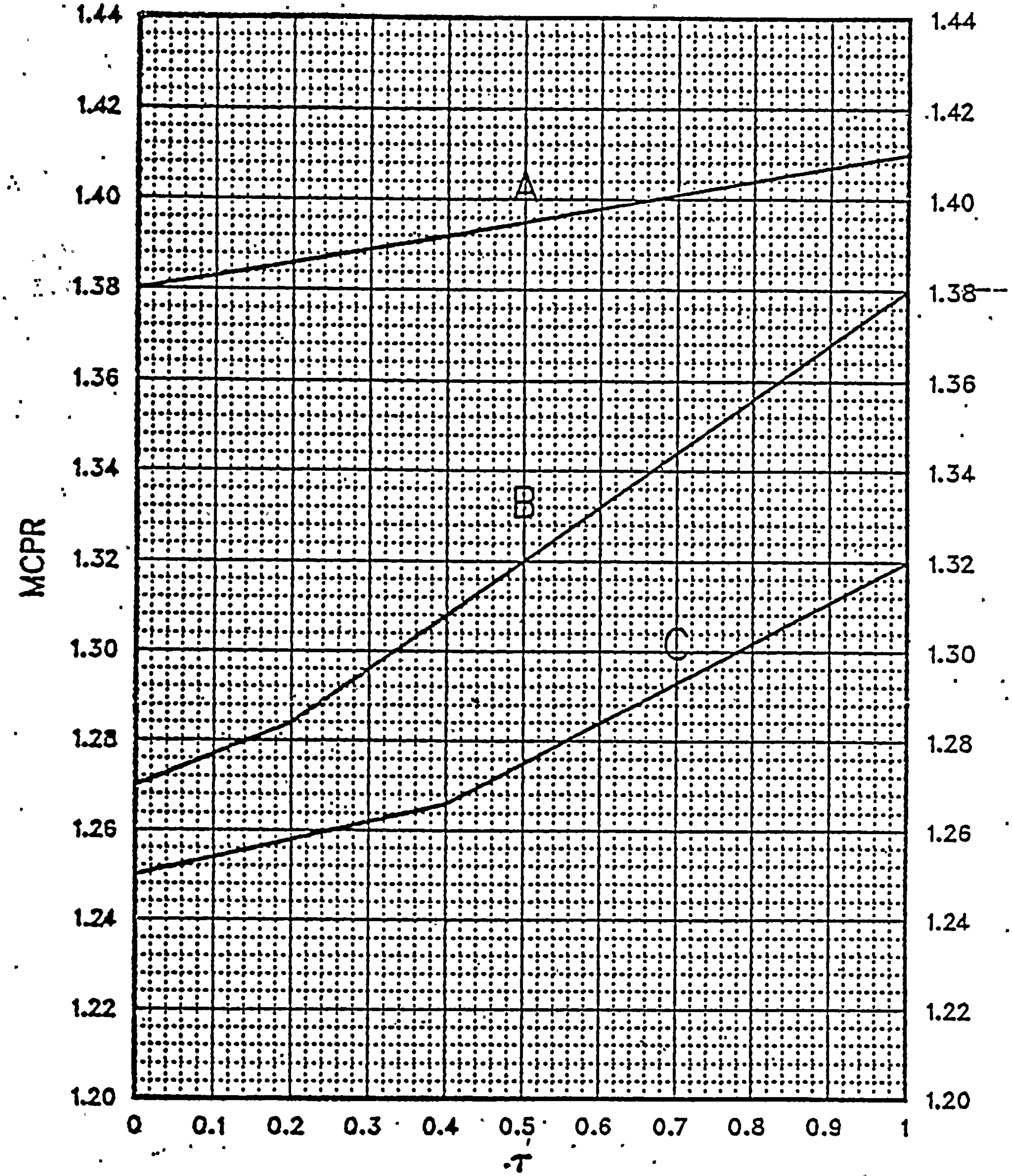
c. Initially and at least once per 12 hours when the reactor is operating with MFLPD greater than or equal to FRTP.

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, the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER, and a notice of the adjustment is posted on the reactor control panel.

See Specification 3.4.1.1.2.a for single loop operation requirements.

MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS τ AT RATED FLOW*



Curve A: Main Turbine Bypass Inoperable; EOC – RPT Operable
Curve B: EOC – RPT Inoperable; Main Turbine Bypass Operable
Curve C: EOC–RPT and Main Turbine Bypass Operable

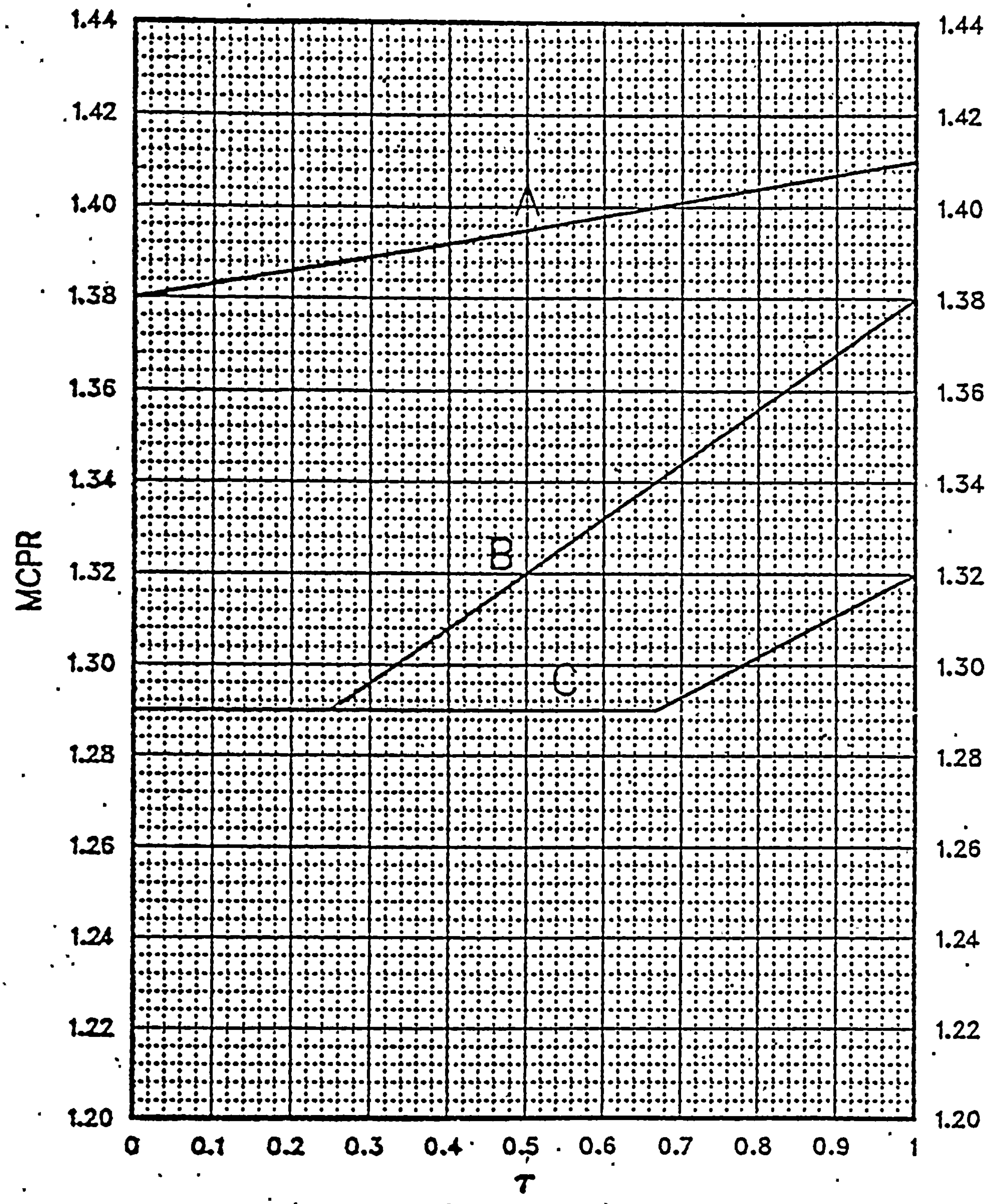
SET

* RBM ~~AT 106%~~ PER TABLE 3.3.6–2

TRIP FUNCTION 1.a.1

FIGURE 3.2.3–1a

MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS τ AT RATED FLOW*



Curve A: Main Turbine Bypass Inoperable; EOC — RPT Operable
Curve B: EOC — RPT Inoperable; Main Turbine Bypass Operable
Curve C: EOC — RPT and Main Turbine Bypass Operable

SET

* RBM ~~AT 100%~~ PER TABLE 3.3.6-2

) TRIP FUNCTION 1.a.2

FIGURE 3.2.3-1b

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR	~~≤ 0.66 W + 42%~~	~~≤ 0.66 W + 45%~~
INSERT a. ~~Upscale~~ ~~1) 100%~~ ~~2) 108% #~~	~~≤ 0.66 W + 40%~~	~~≤ 0.66 W + 43%~~
b. Inoperative	NA	NA
c. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 of divisions full scale
2. APRM		
a. Flow Biased Neutron Flux - Upscale ##	≤ 0.58 W + 50%*	≤ 0.58 W + 53%*
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq 2 \times 10^5$ cps	$\leq 4 \times 10^5$ cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps**	≥ 2 cps**
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level - High	≤ 44 gallons	≤ 44 gallons
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 108/125 divisions of full scale	111/125 divisions of full scale
b. Inoperative	NA	NA
c. Comparator	≤ 10% flow deviation	11% flow deviation

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

** - initial loading and startup the count rate may > 0.5 cps.

~~# May be used when the associated ... requirements in Specification 3.2.3 are satisfied~~

INSERT

Upscale ##

1 ***	$\leq 0.66\,W + 40\%$	$\leq 0.66\,W + 43\%$
2 ***	$\leq 0.66\,W + 42\%$	$\leq 0.66\,W + 45\%$

*** Trip functions 1.a.1 and 1.a.2 shall be used in conjunction with the MCPR limits specified in Figures 3.2.3-1a and 3.2.3-1b, respectively.

See Specification 3.4.1.1.2.a for single loop operation requirements.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS - TWO LOOP OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1.1 Two reactor coolant system recirculation loops shall be in operation and:

a. Total core flow shall be greater than or equal to 45 million lbs/hr, or

b. THERMAL POWER shall be less than or equal to the limit specified in Figure ~~3.4.1.1-1~~. 3.4.1.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*, except during single loop operation.#

ACTION: Within the next 12 hours, either comply with the single loop operation requirements of Specification 3.4.1.1.2, or

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 the limit specified in Figure 3.4.1.1-1, 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 the limit specified in Figure ~~3.4.1.1-1~~ 3.4.1.1.1-1, and initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

c. With two reactor coolant system recirculation loops in operation and total core flow less than 45 million lbs/hr and THERMAL POWER greater than the limit specified in Figure ~~3.4.1.1-1~~ 3.4.1.1.1-1:

   1. Reduce THERMAL POWER to less than or equal to the limit specified in Figure ~~3.4.1.1-1~~ 3.4.1.1.1-1, or

   2. Increase core flow to greater than 45 million lbs/hr, or

   3. Determine the APRM and LPRM*** neutron flux noise levels within 1 hour, and:

      a) If the APRM and LPRM*** neutron flux noise levels are less than three times their established baseline levels, continue to determine the noise levels at least once per 8 hours and within 30 minutes after the completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER, or

      b) If the APRM or LPRM*** neutron flux noise levels are greater than or equal to three times their established baseline levels, immediately initiate corrective action and restore the noise levels to within the required limits within 2 hours by increasing core flow to greater than 45 million lbs/hr, and/or by initiating an orderly reduction of THERMAL POWER to less than or equal to the limit specified in Figure ~~3.4.1.1-1~~ 3.4.1.1.1-1.

*See Special Test Exception 3.10.4.

***Detectors A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

#See Specification 3.4.1.1.2 for single loop operation requirements.

4.4.1.1.1. Each pump discharge valve and bypass valve shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each startup** prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.

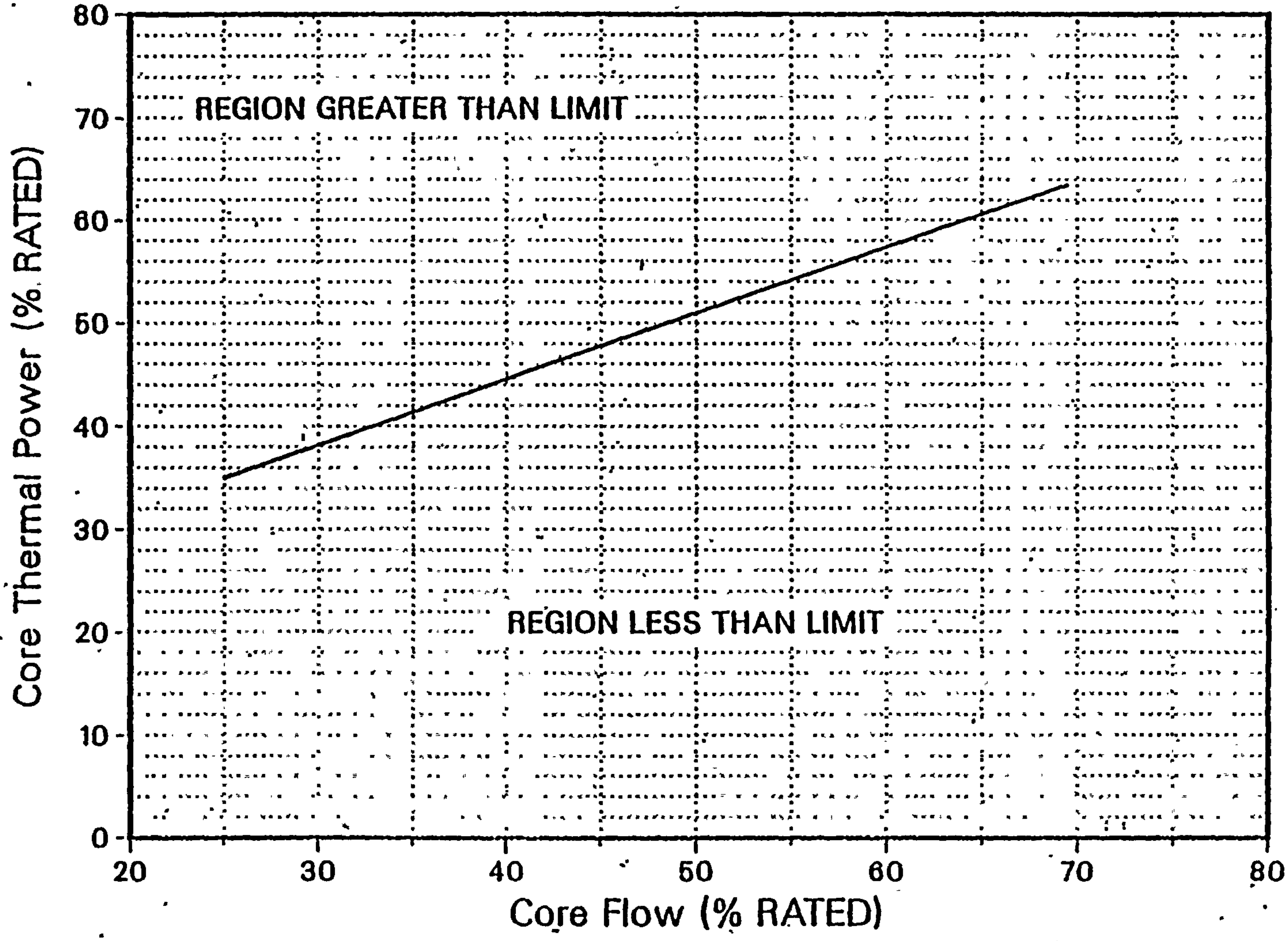
4.4.1.1.2
~~4.4.1.1.2~~ Each pump discharge bypass valve, if not OPERABLE, shall be verified to be closed at least once per 31 days.

4.4.1.1.3
~~4.4.1.1.3~~ Each pump MG set scoop tube electrical and mechanical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 102.5 and 105%, respectively, of rated core flow, at least once per 18 months.

4.4.1.1.4
~~4.4.1.1.4~~ Establish a baseline APRM and LPRM~~***~~ neutron flux noise value at a point within 5% RATED THERMAL POWER of the 100% rated rod line with total core flow between 35% and 50% of rated total core flow during startup testing following each refueling outage.

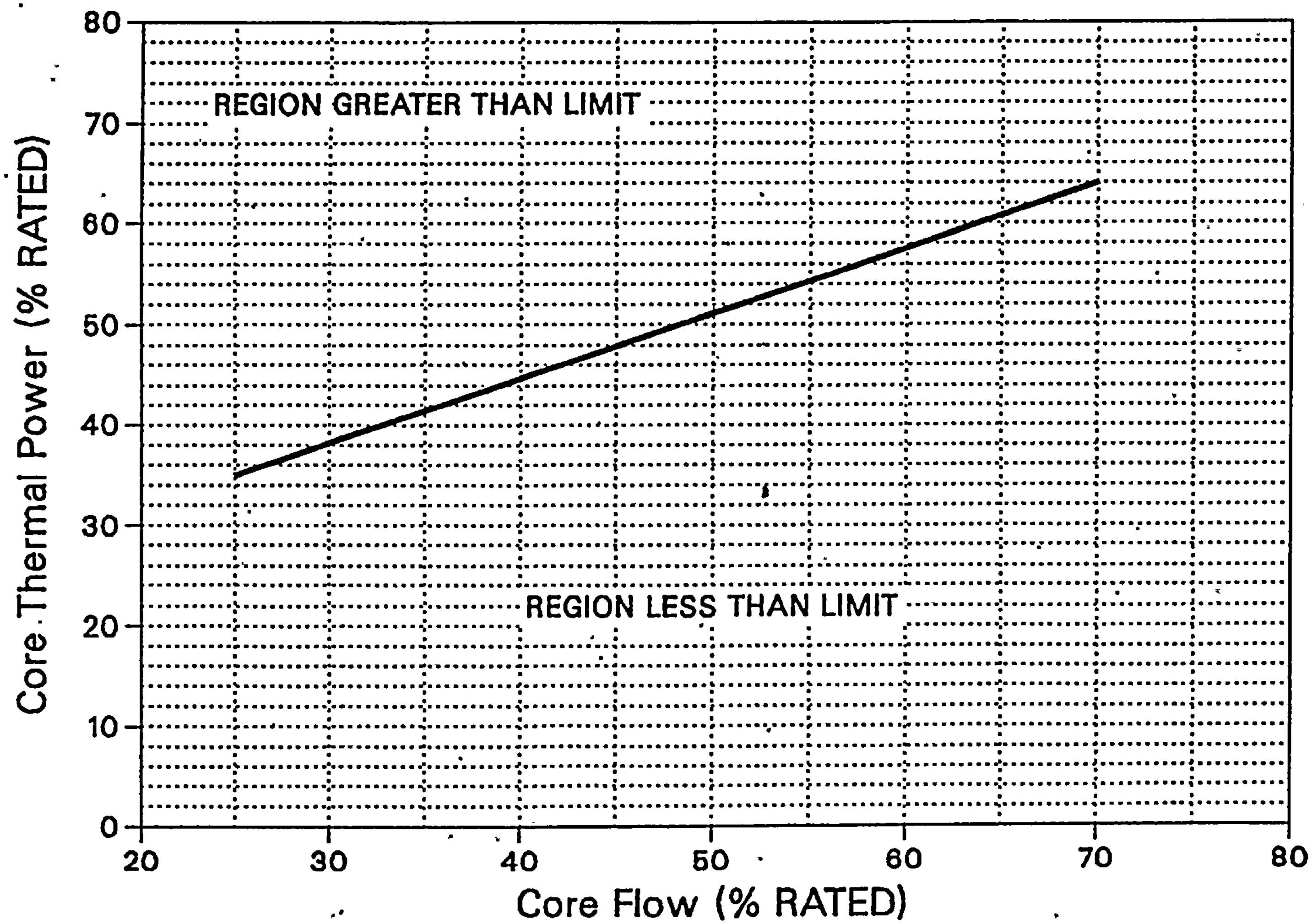
**If not performed within the previous 31 days.
~~***Detectors A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.~~

Figure 3.4.1.1-1
THERMAL POWER LIMITATIONS



REPLACE WITH NEW FIGURE 3.4.1.1-1

Figure 3.4.1.1.1-1
THERMAL POWER LIMITATIONS



REACTOR COOLANT SYSTEM

RECIRCULATION LOOPS - SINGLE LOOP OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1.2 One reactor coolant recirculation loop shall be in operation with the pump speed $\leq$ 80% of the rated pump speed, and

a. the following revised specification limits shall be followed:

1. Specification 2.1.2: the MCPR Safety Limit shall be increased to 1.07.

2. Table 2.2.1-1: the APRM Flow-Biased Scram Trip Setpoints shall be as follows:

Trip Setpoint	Allowable Value
$\leq$0.58W + 55%	$\leq$0.58W + 58%.

3. Specification 3.2.1: the MAPLHGR limits shall be as follows:

   a. GE fuel: the limits specified in Figures 3.2.1-1 and 3.2.1-2, multiplied by 0.81.

   b. Exxon fuel: the limits specified in Figure 3.2.1-2 multiplied by 0.81.

4. Specification 3.2.2: the APRM Setpoints shall be as follows:

Trip Setpoint	Allowable Value
$S \leq (0.58W + 55\%)T$	$S \leq (0.58W + 58\%)T$
$S_{RB} \leq (0.58W + 46\%)T$	$S_{RB} \leq (0.58W + 49\%)T$

5. Table 3.3.6-2: the RBM/APRM Control Rod Block Setpoints shall be as follows:

a. RBM - Upscale	Trip Setpoint	Allowable Value
1.	$\leq$ 0.66W + 35%	$\leq$0.66W + 38%
2.	$\leq$0.66W + 37%	$\leq$0.66W + 40%

   5.a.1 and 5.a.2 shall be used in conjunction with the MCPR limits specified in Figures 3.2.3-1a and 3.2.3-1b, respectively.

b. APRM-Flow Biased	Trip Setpoint	Allowable Value
	$\leq$0.58W + 46%	$\leq$ 0.58W + 49%

b. APRM and LPRM*** neutron flux noise levels shall be less than three times their established baseline levels when THERMAL POWER is greater than the limit specified in Figure 3/4.1.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*, except during two loop operation.#

ACTION:

a. With no reactor coolant system recirculation loops in operation, take the ACTION required by Specification 3.4.1.1.1.

b. With any of the limits specified in 3.4.1.1.2a not satisfied, take the ACTION required by the referenced Specification.

c. With the APRM or LPRM*** neutron flux noise levels greater than or equal to three times their established baseline levels when THERMAL POWER is greater than the limit specified in Figure 3/4.1.1.1-1, immediately initiate corrective action and restore the noise levels to within the required limits within 2 hours by initiating an orderly reduction of THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1.1-1. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

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

SURVEILLANCE REQUIREMENTS

4.4.1.1.2.1 Upon entering single loop operation and at least once per 24 hours thereafter, verify that the pump speed in the operating loop is ≤ 80% of the rated pump speed.

4.4.1.1.2.2 With THERMAL POWER greater than the limit specified in Figure 3.4.1.1.1-1, determine the APRM and LPRM*** neutron flux noise levels within 1 hour. Continue to determine the noise levels at least once per 8 hours and within 30 minutes after the completion of a THERMAL POWER increase ≥ 5% of RATED THERMAL POWER.

4.4.1.1.2.3 Within 15 minutes prior to either THERMAL POWER increase resulting from a control rod withdrawal or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is ≤ 30%**** of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is ≤ 50%**** of rated loop flow:

a. ≤ 145°F between reactor vessel steam space coolant and bottom head drain line coolant,

b.## ≤ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c.## ≤ 50°F between the reactor coolant within the loop not in operation and operating loop.

4.4.1.1.2.4 a. Establish a baseline APRM and LPRM neutron flux noise value at a point within 5% RATED THERMAL POWER of the 100% rated

rod line with total core flow between 35% and 50% of rated total core flow during startup testing following each refueling outage, or

b. In lieu of establishing a single loop operation baseline value, utilize the value established pursuant to Specification 4.4.1.1.1.4 if a baseline value is needed to meet the requirements of Specification 3.4.1.1.2.

4.4.1.1.2.5 The pump discharge valve and bypass valve in both loops shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each startup** prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.

4.4.1.1.2.6 The pump discharge bypass valve in the OPERABLE loop, if not OPERABLE, shall be verified to be closed at least once per 31 days.

4.4.1.1.2.7 The pump MG set scoop tube electrical and mechanical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 102.5 and 105%, respectively, of rated core flow, at least once per 18 months.

4.4.1.1.2.8 The pump discharge valve and bypass valve in the inoperable loop, if not OPERABLE, shall be verified to be closed at least once per 31 days.

4.4.1.1.2.9 During single recirculation loop operation, all jet pumps, including those in the inoperable loop, shall be demonstrated OPERABLE at least once per 24 hours by verifying that no two of the following conditions occur:###

a. The indicated recirculation loop flow in the operating loop differs by more than 10% from the established single recirculation pump speed-loop flow characteristics.

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

c. The indicated difference-to-lower plenum differential pressure of any individual jet pump differs from established single recirculation loop patterns by more than 10%.

4.4.1.1.2.10 The SURVEILLANCE REQUIREMENTS associated with the specifications referenced in 3.4.1.1.2a shall be followed.

* See Special Test Exception 3.10.4.

** If not performed within the previous 31 days.

*** Detectors A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

**** Initial value. Final value to be determined based on startup testing. Any required change to this value shall be submitted to the Commission within 90 days of test completion.

See Specification 3.4.1.1.1 for two loop operation requirements.

This requirement does not apply when the loop not in operation is isolated from the reactor pressure vessel.

During startup testing following each refueling outage, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the performance of subsequent required surveillances.

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*~~, when both recirculation loops are in operation.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.1.2** Each of the above required jet pumps shall be demonstrated OPERABLE prior to THERMAL POWER exceeding 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 the recirculation pumps are operating at the same speed:

a. The indicated recirculation loop flow differs by more than 10% from the established pump speed-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%.

** See Specification 4.4.1.1.2.9 for single loop operation requirements.

REACTOR COOLANT SYSTEM

RECIRCULATION PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation pump speed mismatch shall be maintained within:

a. 5% of each other with core flow greater than or equal to 75% of rated core flow.

b. 10% of each other with core flow less than 75% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*~~#~~, when both recirculation loops are in operation.

ACTION:

With the recirculation pump speeds different by more than the specified limits, either:

a. Restore the recirculation pump speeds to within the specified limit within 2 hours, or

b. Declare the recirculation loop of the pump with the slower speed not in operation and take the ACTION required by Specification 3.4.1.1.1.

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exception 3.10.4.

3/4.10.4 RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.4.1.1 and 3.4.1.3 may be suspended for up to 24 hours for the performance of:

a. PHYSICS TESTS, provided that THERMAL POWER does not exceed 5% of RATED THERMAL POWER, or

b. The Startup Test Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2, during PHYSICS TESTS and the Startup Test Program.

ACTION:

a. With the above specified time limit exceeded, insert all control rods.

b. With the above specified THERMAL POWER limit exceeded during PHYSICS TESTS, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The time during which the above specified requirement has been suspended shall be verified to be less than 24 hours at least once per hour during PHYSICS TESTS and the Startup Test Program.

4.10.4.2 THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

3/4.1.3 CONTROL RODS

The specification 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 analysis, and (3) limit the potential effects of the rod drop accident. 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 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 shutdown for investigation and resolution of the problem.

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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~~ during the ~~limiting power~~ transient analyzed in ~~Section 15.1 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 occurrence of scram times longer then those specified should be viewed as an indication of a ~~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 limit specified in Specification 2.1.2

Systematic core-wide

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 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.

3/4.2.3 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~~ Table 3.2.3-1.

The evaluation of a given transient begins with the Exxon system initial parameters shown in ~~FSAR Table 15.0-2~~ XN-NF-84-118 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 nonpressurization 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]~~. & INSERT (G) The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the ~~K_f factor~~ MCPR$_f$ of Figure 3.2.3-1 is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow the required MCPR is INSERT (H) ~~the product of the MCPR and the K_f factor. The K_f factors assure~~ that the Safety Limit MCPR will not be violated during a flow increase transient resulting from a motor-generator speed control failure. INSERT (I) ~~The K_f factors may be applied to both manual and automatic flow control modes.~~

~~The K_f factor values shown in Figure 3.2.3-1 were devloped generically and are applicable to all BWR/2, BWR/3 and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to RATED THERMAL POWER at rated core flow.~~

~~For the manual flow control mode, the K_f factors were calculated such that for the maximum flow rate, as limited by the pump scoop tube set point and the corresponding THERMAL POWER along the rated flow control line, the limiting bundle's relative power was adjusted until the MCPR changes with different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the K_f.~~

INSERT G

The codes and methodology to evaluate pressurization and non-pressurization events are described in XN-NF-79-71.

INSERT H

the maximum of the rated flow MCPR determined from Table 3.2.3-1 and the reduced flow MCPR determined from Figure 3.2.3-1. $MCPR_f$ assures

INSERT E

$MCPR_f$ is only calculated for the manual flow control mode. Therefore, automatic flow control operation is not permitted.

3/4.4.1 RECIRCULATION SYSTEM

INSERT X

~~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.~~

THERMAL POWER, core flow, and neutron flux noise level limitations are prescribed in accordance with the recommendations of General Electric Service Information Letter No. 380, Revision 1, "BWR Core Thermal Hydraulic Stability," dated February 10, 1984. 24

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 pump speed for two loop operation mismatch limits are in compliance with the ECCS LOCA analysis design criteria~~s~~. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. INSERT Y

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 145°F.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves 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 10 OPERABLE safety-relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient.

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.

X. Operation with one reactor recirculation loop inoperable has been evaluated and found acceptable, provided that the unit is operated in accordance with Specification 3.4.1.1.2.

For single loop operation, the MAPLHGR limits are multiplied by a factor of 0.81. Use of this factor in conjunction with the GE high enriched fuel MAPLHGR is conservative to use as a limit for the Exxon fuel. The multiplication factor is derived from LOCA analyses initiated from single loop operation; it maintains the same peak clad temperature margin to the licensing limit of 2200°F as in two loop operation.

For single loop operation, the RBM and APRM setpoints are adjusted by a 7% decrease in recirculation drive flow to account for the active loop drive flow that bypasses the core and goes up through the inactive loop jet pumps.

Surveillance on the pump speed of the operating recirculation loop is imposed to exclude the possibility of excessive reactor vessel internals vibration. Surveillance on differential temperatures below the threshold limits on THERMAL POWER or recirculation loop flow mitigates undue thermal stress on vessel nozzles, recirculation pumps and the vessel bottom head during extended operation in the single loop mode. The threshold limits are those values which will sweep up the cold water from the vessel bottom head.

Y. In the case where the mismatch limits cannot be maintained during the loop operation, continued operation is permitted in the single loop mode.

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* with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.06* 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.

* See Specification 3.4.1.1.2.a for single loop operation requirement.

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-Upscale, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-Upscale		
1) Flow Biased	≤ 0.58 W+59%,# with a maximum of	≤ 0.58 W+62%,# with a maximum of
2) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
c. Neutron Flux-Upscale	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1037 psig	≤ 1057 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 13.0 inches above instrument zero*	≥ 11.5 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 10% closed	≤ 11% closed
6. Main Steam Line Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background
7. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Level Transmitter	≤ 88 gallons	≤ 88 gallons
b. Float Switch	≤ 88 gallons	≤ 88 gallons
9. Turbine Stop Valve - Closure	≤ 5.5% closed	≤ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 500 psig	≥ 460 psig
11. Reactor Mode Switch Shutdown Position	NA	NA
12. Manual Scram	NA	NA

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

See Specification 3.4.1.1.2.a for single loop operation requirements.

2.1 SAFETY LIMITS

BASES

2.0 INTRODUCTION

the limit specified in Specification 2.1.2

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than ~~1.06~~. 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.

the specified limit

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.

DELETE THIS TABLE

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
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	6.3
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.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3.*

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3:

a. At least once per 24 hours,

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

* See Specification 3.4.1.1.2.a for single loop operation requirements.

POWER DISTRIBUTION LIMITS

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

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

Trip Setpoint #	Allowable Value #
$S \leq (0.58W + 59\%)T$	$S \leq (0.58W + 62\%)T$
$S_{RB} \leq (0.58W + 50\%)T$	$S_{RB} \leq (0.58W + 53\%)T$

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 100 million lbs/hr,
T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY. T is always 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-upscale 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 adjust S and/or S_{RB} to be consistent with the Trip Setpoint value* within 2 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 the MFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale 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.

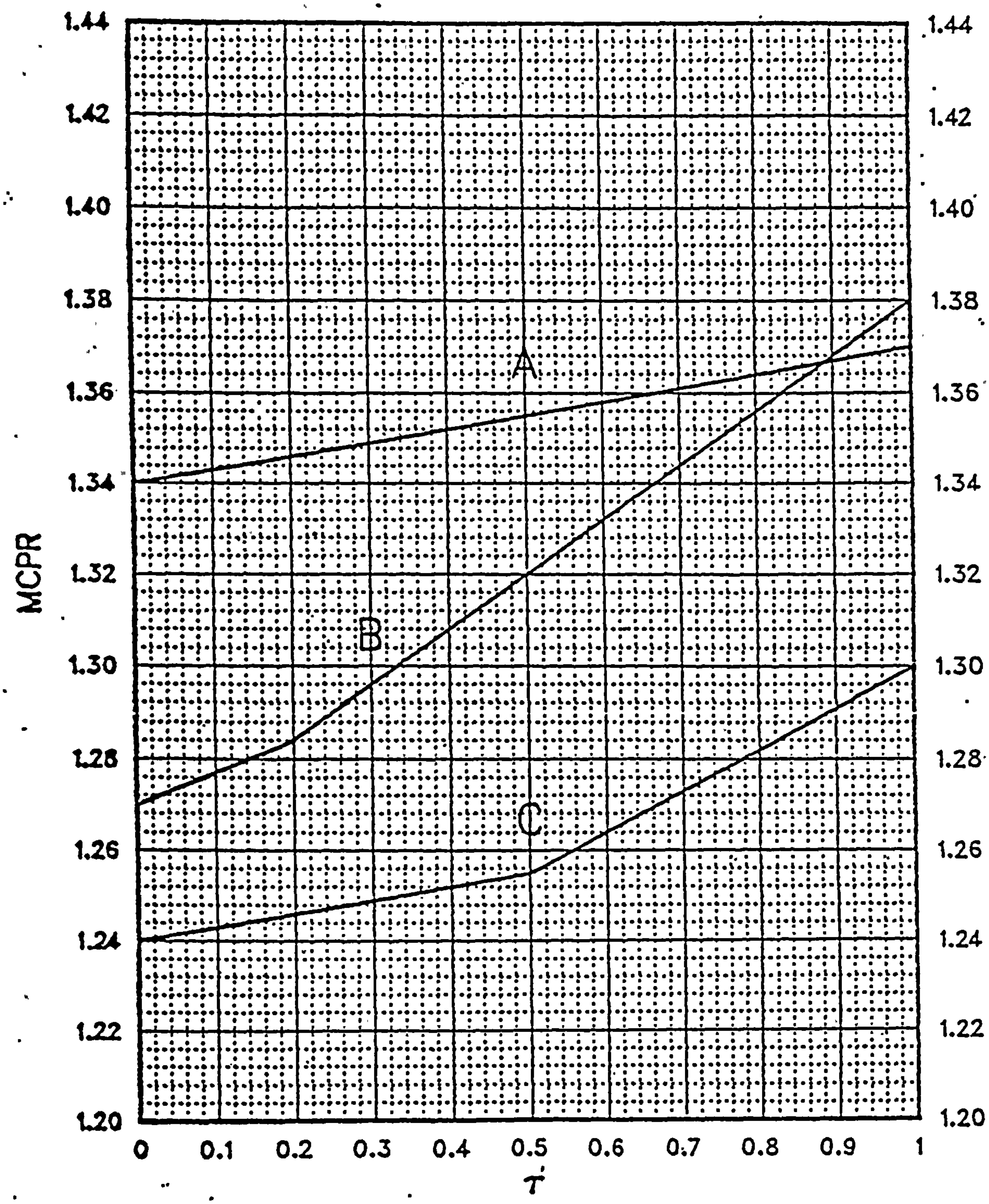
d. The provisions of Specification 4.0.4 are not applicable.

---

*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, the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER, and a notice of the adjustment is posted on the reactor control panel.

See Specification 3.4.1.1.2a for single loop operation requirements.

MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS τ AT RATED FLOW*

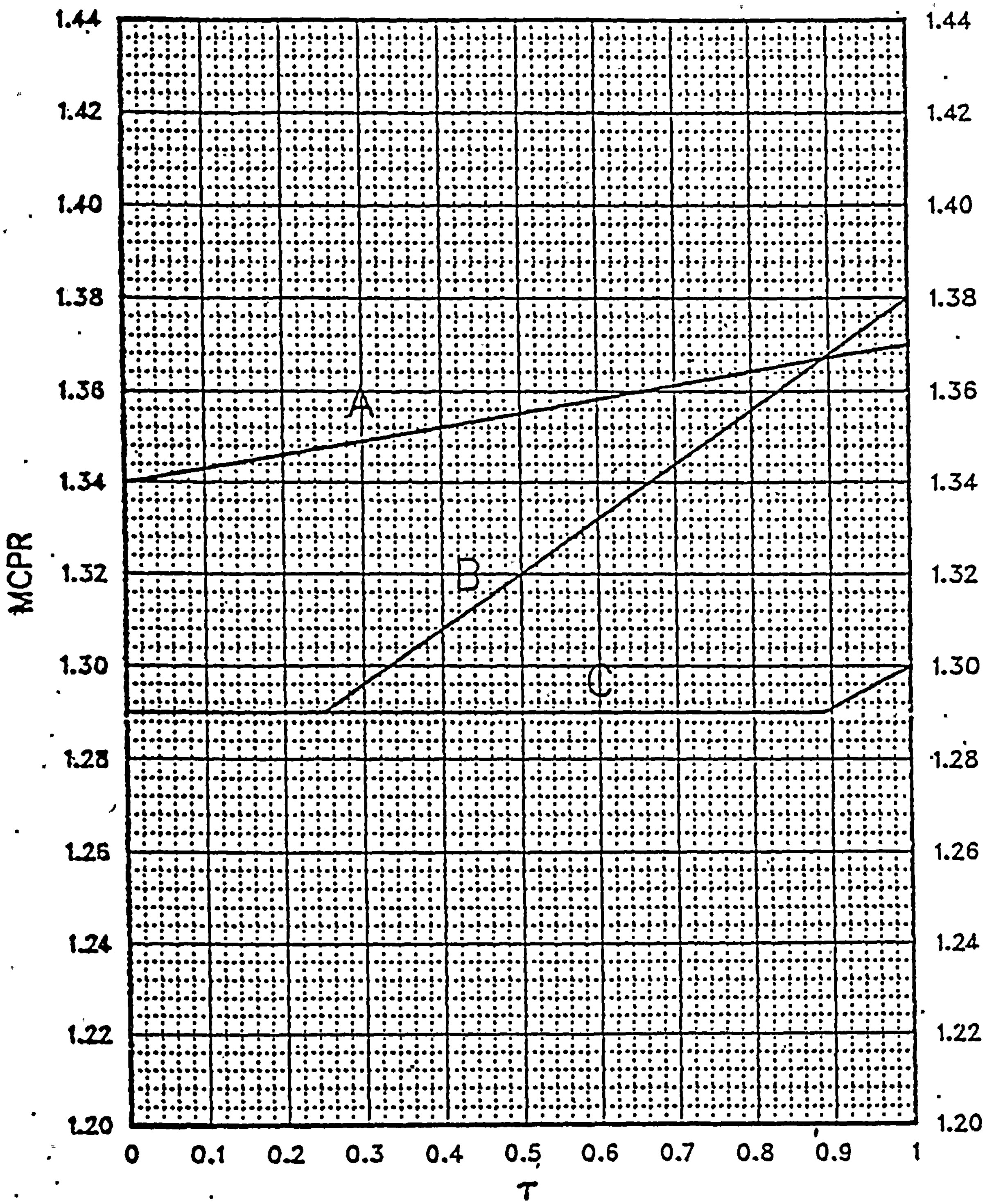


Curve A: Main Turbine Bypass Inoperable; EOC – RPT Operable
Curve B: EOC – RPT Inoperable; Main Turbine Bypass Operable
Curve C: EOC – RPT And Main Turbine Bypass Operable

SET
* RBM ~~AT 106%~~ PER TABLE 3.3.6–2
, TRIP FUNCTION 1.a.1

FIGURE 3.2.3–1a

MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS τ AT RATED FLOW*



Curve A: Main Turbine Bypass Inoperable; EOC – RPT Operable
Curve B: EOC – RPT Inoperable; Main Turbine Bypass Operable
Curve C: EOC – RPT And Main Turbine Bypass Operable

* RBM ~~AT 108%~~ SET PER TABLE 3.3.6–2, TRIP FUNCTION 1.a.2

FIGURE 3.2.3–1b

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR	~~≤ 0.66 W + 42%~~	~~≤ 0.66 W + 45%~~
INSERT a. ~~Upscale~~ ~~1) 106%~~	~~≤ 0.66 W + 40%~~	~~≤ 0.66 W + 43%~~
b. Inoperative ~~2) 108%#~~	NA	NA
c. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 of divisions full scale
2. APRM		
a. Flow Biased Neutron Flux - Upscale ~~##~~	< 0.58 W + 50%*	< 0.58 W + 53%*
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$< 2 \times 10^5$ cps	$< 4 \times 10^5$ 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 divisions of full scale	< 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level - High	≤ 44 gallons	≤ 44 gallons
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	< 108/125 divisions of full scale	< 111/125 divisions of full scale
b. Inoperative	NA	NA
c. Comparator	< 10% flow deviation	< 11% flow deviation

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

**Provided signal-to-noise ratio is ≥ 2. Otherwise, 3 cps as trip setpoint and 2.8 cps for allowable value.

~~# May be used when the associated MCPR requirements in Specification 3.2.3 are satisfied.~~

INSERT

Upscale ##

1 ***	$\leq 0.66 W + 40\%$	$\leq 0.66W + 43\%$
2 ***.	$\leq 0.66 W + 42\%$	$\leq 0.66W + 45\%$

*** Trip functions 1.a.1 and 1.a.2 shall be used in conjunction with the MCPR limits specified in Figures 3.2.3-1a and 3.2.3-1b, respectively.

See Specification 3.4.1.1.2.a for single loop operation requirements.

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS - TWO LOOP OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1.1 Two reactor coolant system recirculation loops shall be in operation. and:

a. Total core flow shall be greater than or equal to 45 million lbs/hr, or

b. THERMAL POWER shall be less than or equal to the limit specified in Figure ~~3.4.1.1-1~~. 3.4.1.1.1-1

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*, except during single loop operation.#

ACTION: within the next 12 hours, either comply with the single loop operation requirements of Specification 3.4.1.1.2, or

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 the limit specified in Figure 3.4.1.1-1, 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 the limit specified in Figure ~~3.4.1.1-1~~ 3.4.1.1.1-1, and initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

c. With two reactor coolant system recirculation loops in operation and total core flow less than 45 million lbs/hr and THERMAL POWER greater than the limit specified in Figure ~~3.4.1.1-1~~ 3.4.1.1.1-1:

1. Reduce THERMAL POWER to less than or equal to the limit specified in Figure ~~3.4.1.1-1~~ 3.4.1.1.1-1, or

2. Increase core flow to greater than 45 million lbs/hr, or

3. Determine the APRM and LPRM*** neutron flux noise levels within 1 hour, and:

   a) If the APRM and LPRM*** neutron flux noise levels are less than three times their established baseline levels, continue to determine the noise levels at least once per 8 hours and within 30 minutes after the completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER, or

   b) If the APRM or LPRM*** neutron flux noise levels are greater than or equal to three times their established baseline levels, immediately initiate corrective action and restore the noise levels to within the required limits within 2 hours by increasing core flow to greater than 45 million lbs/hr, and/or by initiating an orderly reduction of THERMAL POWER to less than or equal to the limit specified in Figure ~~3.4.1.1-1~~. 3.4.1.1.1-1

*See Special Test Exception 3.10.4.

***Detectors A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

See Specification 3.4.1.1.2 for single loop operation requirements.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 Each pump discharge valve and bypass valve shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each startup** prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.

4.4.1.1.2
~~4.4.1.1.2~~ Each pump discharge bypass valve, if not OPERABLE, shall be verified to be closed at least once per 31 days.

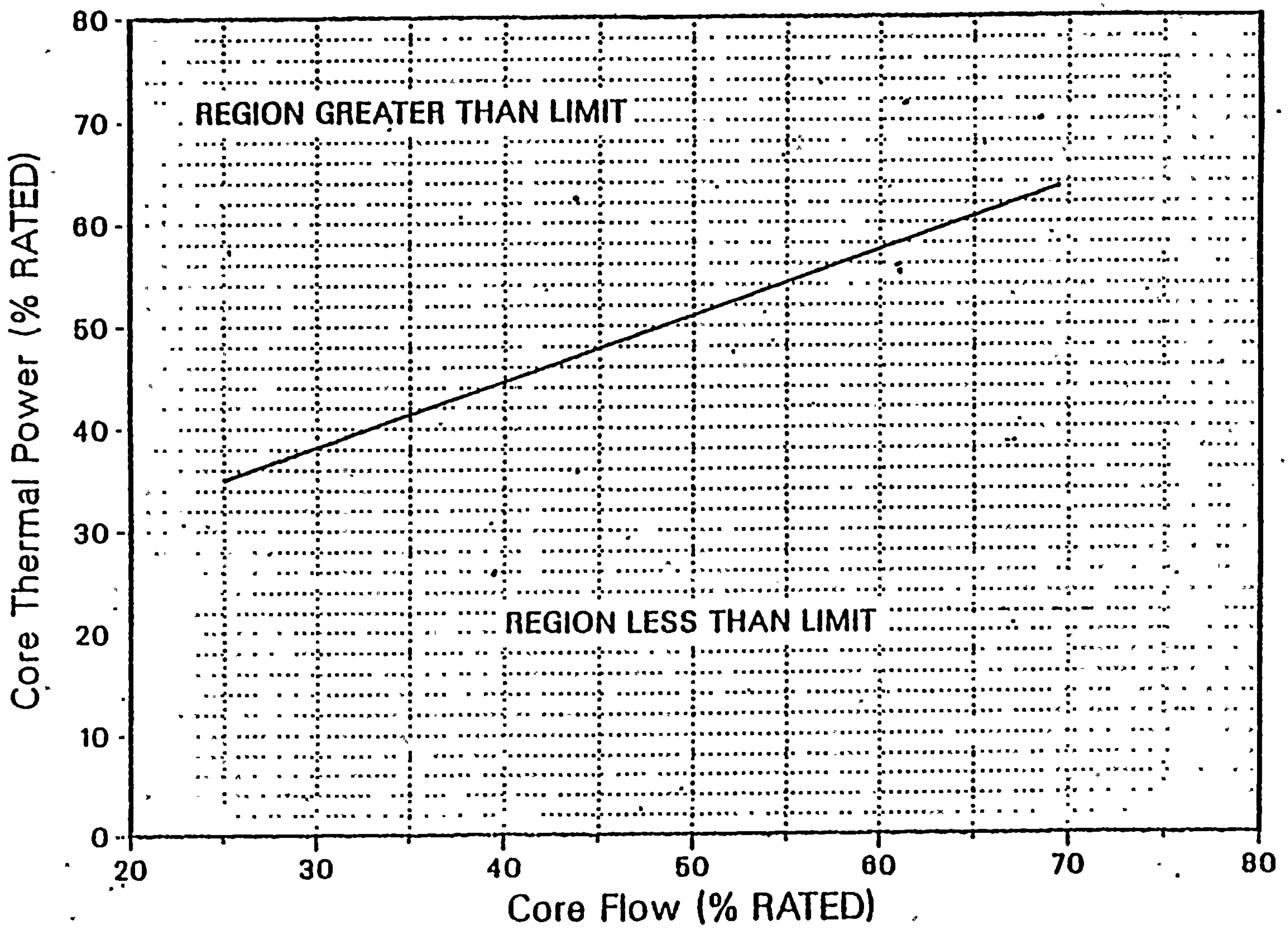
4.4.1.1.3
~~4.4.1.1.3~~ Each pump MG set scoop tube electrical and mechanical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 102.5 and 105%, respectively, of rated core flow, at least once per 18 months.

4.4.1.1.4
~~4.4.1.1.4~~ Establish a baseline APRM and LPRM~~***~~ neutron flux noise value at a point within 5% RATED THERMAL POWER of the 100% rated rod line with total core flow between 35% and 50% of rated total core flow during startup testing following each refueling outage.

**If not performed within the previous 31 days.

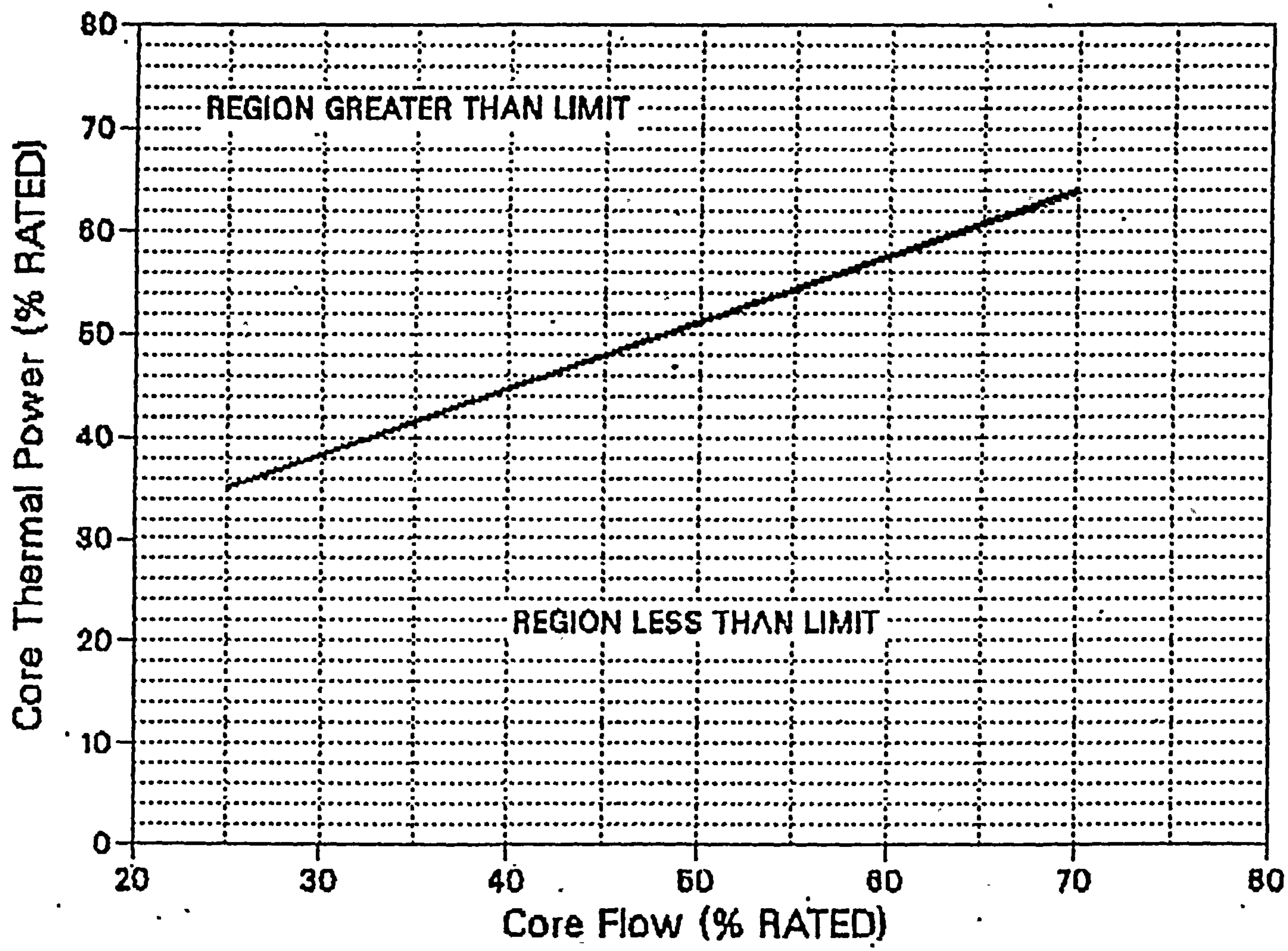
~~***Detectors A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.~~

Figure 3.4.1.1-1
THERMAL POWER LIMITATIONS



REPLACE WITH NEW FIGURE 3.4.1.1-1

Figure 3.4.1.1.1-1
THERMAL POWER LIMITATIONS



REACTOR COOLANT SYSTEM

RECIRCULATION LOOPS - SINGLE LOOP OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1.2 One reactor coolant recirculation loop shall be in operation with the pump speed $\leq$ 90% of the rated pump speed, and

a. the following revised specification limits shall be followed:

1. Specification 2.1.2: the MCPR Safety Limit shall be increased to 1.07.

2. Table 2.2.1-1: the APRM Flow-Biased Scram Trip Setpoints shall be as follows:

Trip Setpoint	Allowable Value
$\leq$ 0.58W + 55%	$\leq$ 0.58W + 58%.

3. Specification 3.2.1: the MAPLHGR limits shall be the limits specified in Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3, multiplied by 0.81.

4. Specification 3.2.2: the APRM Setpoints shall be as follows:

Trip Setpoint	Allowable Value
$S \leq (0.58W + 55\%)T$	$S \leq (0.58W + 58\%)T$
$S_{RB} \leq (0.58W + 46\%)T$	$S_{RB} \leq (0.58W + 49\%)T$

5. Table 3.3.6-2: the RBM/APRM Control Rod Block Setpoints shall be as follows:

a. RBM - Upscale	Trip Setpoint	Allowable Value
1.	$\leq$ 0.66W + 35%	$\leq$ 0.66W + 38%
2.	$\leq$ 0.66W + 37%	$\leq$ 0.66W + 40%

5.a.1 and 5.a.2 shall be used in conjunction with the MCPR limits specified in Figures 3.2.3-1a and 3.2.3-1b, respectively.

b. APRM-Flow Biased	Trip Setpoint	Allowable Value
	$\leq$ 0.58W + 46%	$\leq$ 0.58W + 49%

b. APRM and LPRM*** neutron flux noise levels shall be less than three times their established baseline levels when THERMAL POWER is greater than the limit specified in Figure 3/4.1.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*, except during two loop operation.#

ACTION:

a. With no reactor coolant system recirculation loops in operation, take the ACTION required by Specification 3.4.1.1.1.

b. With any of the limits specified in 3.4.1.1.2a not satisfied, take the ACTION required by the referenced Specification.

c. With the APRM or LPRM*** neutron flux noise levels greater than or equal to three times their established baseline levels when THERMAL POWER is greater than the limit specified in Figure 3/4.1.1.1-1, immediately initiate corrective action and restore the noise levels to within the required limits within 2 hours by initiating an orderly reduction of THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1.1-1. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

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

SURVEILLANCE REQUIREMENTS

4.4.1.1.2.1 Upon entering single loop operation and at least once per 24 hours thereafter, verify that the pump speed in the operating loop is ≤90% of the rated pump speed.

4.4.1.1.2.2 With THERMAL POWER greater than the limit specified in Figure 3.4.1.1.1-1, determine the APRM and LPRM*** neutron flux noise levels within 1 hour. Continue to determine the noise levels at least once per 8 hours and within 30 minutes after the completion of a THERMAL POWER increase ≥5% of RATED THERMAL POWER.

4.4.1.1.2.3 Within 15 minutes prior to either THERMAL POWER increase resulting from a control rod withdrawal or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is ≤30%**** of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is ≤ 50%**** of rated loop flow:

a. ≤145°F between reactor vessel steam space coolant and bottom head drain line coolant,

b.## ≤50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c.## ≤50°F between the reactor coolant within the loop not in operation and operating loop.

4.4.1.1.2.4. a. Establish a baseline APRM and LPRM neutron flux noise value at a point within 5% RATED THERMAL POWER of the 100% rated rod line with total core flow between 35% and 50% of rated total core flow during startup testing following each refueling outage, or

b. In lieu of establishing a single loop operation baseline value, utilize the value established pursuant to Specification 4.4.1.1.1.4 if a baseline value is needed to meet the requirements of Specification 3.4.1.1.2.

4.4.1.1.2.5 The pump discharge valve and bypass valve in both loops shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each startup** prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.

4.4.1.1.2.6 The pump discharge bypass valve in the OPERABLE loop, if not OPERABLE, shall be verified to be closed at least once per 31 days.

4.4.1.1.2.7 The pump MG set scoop tube electrical and mechanical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 102.5 and 105%, respectively, of rated core flow, at least once per 18 months.

4.4.1.1.2.8 The pump discharge valve and bypass valve in the inoperable loop, if not OPERABLE, shall be verified to be closed at least once per 31 days.

4.4.1.1.2.9 During single recirculation loop operation, all jet pumps, including those in the inoperable loop, shall be demonstrated OPERABLE at least once per 24 hours by verifying that no two of the following conditions occur:###

a. The indicated recirculation loop flow in the operating loop differs by more than 10% from the established single recirculation pump speed-loop flow characteristics.

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

c. The indicated difference-to-lower plenum differential pressure of any individual jet pump differs from established single recirculation loop patterns by more than 10%.

4.4.1.1.2.10 The SURVEILLANCE REQUIREMENTS associated with the specifications referenced in 3.4.1.1.2a shall be followed.

* See Special Test Exception 3.10.4.

** If not performed within the previous 31 days.

*** Detectors A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

**** Initial value. Final value to be determined based on startup testing. Any required change to this value shall be submitted to the Commission within 90 days of test completion.

See Specification 3.4.1.1.1 for two loop operation requirements.

This requirement does not apply when the loop not in operation is isolated from the reactor pressure vessel.

During startup testing following each refueling outage, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the performance of subsequent required surveillances.

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 when both recirculation loops are in operation.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.1.2** Each of the above required jet pumps shall be demonstrated OPERABLE prior to THERMAL POWER exceeding 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 the recirculation pumps are operating at the same speed:

a. The indicated recirculation loop flow differs by more than 10% from the established pump speed-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%.

*During the startup test program, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the conclusion of the startup test program.

** See Specification 4.4.1.1.2.9 for single loop operation requirements.

REACTOR COOLANT SYSTEM

RECIRCULATION PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation pump speed mismatch shall be maintained within:.

a. 5% of each other with core flow greater than or equal to 75% of rated core flow.

b. 10% of each other with core flow less than 75% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2* when both recirculation loops are in operation.

ACTION:

With the recirculation pump speeds different by more than the specified limits, either:

a. Restore the recirculation pump speeds to within the specified limit within 2 hours, or

b. Declare the recirculation loop of the pump with the slower speed not in operation and take the ACTION required by Specification 3.4.1.1.1.

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exception 3.10.4.

SPECIAL TEST EXCEPTIONS

3/4.10.4 RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.4.1.1 and 3.4.1.3 may be suspended for up to 24 hours for the performance of:

a. PHYSICS TESTS, provided that THERMAL POWER does not exceed 5% of RATED THERMAL POWER, or

b. The Startup Test Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2, during PHYSICS TESTS and the Startup Test Program.

ACTION:

a. With the above specified time limit exceeded, insert all control rods.

b. With the above specified THERMAL POWER limit exceeded during PHYSICS TESTS, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The time during which the above specified requirement has been suspended shall be verified to be less than 24 hours at least once per hour during PHYSICS TESTS and the Startup Test Program.

4.10.4.2 THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

3/4.1.3 CONTROL RODS

The specification 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 analysis, and (3) limit the potential effects of the rod drop accident. 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 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 shutdown for investigation and resolution of the problem.

the limit specified in Specification 2.1.2

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~~ during the limiting power transient analyzed in Section 15.1 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 occurrence of scram times longer then those specified should be viewed as an indication of a ~~systemic~~ 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 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.

3/4.2.3 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 evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-2 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 nonpressurization 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 K_f factor of Figure 3.2.3-1 is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow the required MCPR is the product of the MCPR and the K_f factor. The K_f factors assure that the Safety Limit MCPR will not be violated during a flow increase transient resulting from a motor-generator speed control failure. The K_f factors may be applied to both manual and automatic flow control modes.

The K_f factor values shown in Figure 3.2.3-1 were devloped generically and are applicable to all BWR/2, BWR/3 and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to RATED THERMAL POWER at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that for the maximum flow rate, as limited by the pump scoop tube set point and the corresponding THERMAL POWER along the rated flow control line, the limiting bundle's relative power was adjusted until the MCPR changes with different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the K_f.

3/4.4.1 RECIRCULATION SYSTEM

INSERT X

~~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.~~

THERMAL POWER, core flow, and neutron flux noise level limitations are prescribed in accordance with the recommendations of General Electric Service Information Letter No. 380, Revision 1, "BWR Core Thermal Hydraulic Stability," dated February 10, 1984.

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 pump speed for two loop operation. mismatch limits are in compliance with the ECCS LOCA analysis design criteria~~.~~ ~~The~~ limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. INSERT Y

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 145°F.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves 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 10 OPERABLE safety/ relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient.

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.

X. Operation with one reactor recirculation loop inoperable has been evaluated and found acceptable, provided that the unit is operated in accordance with Specification 3.4.1.1.2.

For single loop operation, the MAPLHGR limits are multiplied by a factor of 0.81. The multiplication factor is derived from LOCA analyses initiated from single loop operation; it maintains the same peak clad temperature margin to the licensing limit of 2200°F as in two loop operation.

For single loop operation, the RBM and APRM setpoints are adjusted by a 7% decrease in recirculation drive flow to account for the active loop drive flow that bypasses the core and goes up through the inactive loop jet pumps.

Surveillance on the pump speed of the operating recirculation loop is imposed to exclude the possibility of excessive reactor vessel internals vibration. Surveillance on differential temperatures below the threshold limits on THERMAL POWER or recirculation loop flow mitigates undue thermal stress on vessel nozzles, recirculation pumps and the vessel bottom head during extended operation in the single loop mode. The threshold limits are those values which will sweep up the cold water from the vessel bottom head.

Y. In the case where the mismatch limits cannot be maintained during the loop operation, continued operation is permitted in the single loop mode.