ATTACHMENT TO WNP-2 SINGLE LOOP OPERATION SUMMARY REPORT
TECHNICAL SPECIFICATION CHANGES (INCLUDING JUSTIFICATION)

The following Technical Specification pages are attached:

- Justification for Changes

v	Revise Index to add new 3/4.2.6 and 3/4.2.7
vi	Revise Index to delete 3/4.3.10
xii	Revise (Bases) Index to add new 3/4.2.6 and 3/4.2.7
xiii	Revise (Bases) Index to delete 3/4.3.10
xx	Revise List of Figures
2-4	Revise Table 2.2.1-1
3/4 2-4	Revise Figure 3.2.1-3
3/4 2-4C	Delete Figure 3.2.1-6
3/4 2-5	Revise as indicated
3/4 2-6	Revise as indicated
3/4 2-7	Revise MCPR Operating Limits
3/4 2-8	Revise Figure 3.2.3-1
New Page (-1-)	Add new Section 3/4.2.6 Power/Flow Instability
3/4 3-104	Adds new Figure 3.2.6-1
New Page (-2-)	Add new Section 3/4.2.7 Neutron Flux Noise Monitoring
3/4 3-104	Adds new Figure 3.2.7-1
3/4 3-55	Revise Table 3.3.6-2
3/4 3-102	Delete Section 3/4.3.10
3/4 3-103	Delete Section 3/4.3.10
3/4 3-104	Delete Figure 3.3.10-1
3/4 4-1	Revise as indicated
New Page (-4-)	Add new Action Statement to LCO 3.4.1.1
3/4 4-2	Revise as indicated
B3/4 2-1	Revise as indicated
New Page (-5-)	Add new Bases Section 3/4.2.6
New Page (-6-)	Add new Bases Section 3/4.2.6
B3/4 3-7	Delete Bases Section 3/4.3.10
B3/4 3-7a	Delete Bases Section 3/4.3.10
B3/4 3-7	Add new Bases Section 3/4.2.7
B3/4 3-7a	Add new Bases Section 3/4.2.7
B3/4 4-1	Revise as indicated

JUSTIFICATION FOR CHANGES TO TECHNICAL SPECIFICATION 3.3.10

1) Move this LCO to the POWER DISTRIBUTION LIMITS section of Tech. Specs. The intent of the LCO is to monitor neutron flux noise levels to detect the approach of an unstable region of operation. The LCO has little or nothing to do with instrument calibration, and thus, will be more appropriately located in the POWER DISTRIBUTION LIMITS section of Tech. Specs, immediately adjacent to related LCO 3/4.2.6, Power/Flow Stability.

2) Modify wording in the LCO and APPLICABILITY sections to more clearly define the region of applicability where noise monitoring is required.

3) Clarify ACTION based upon whether baselining has been performed or not.

4) Incorporate Surveillance Requirements 4.3.10.2 and 4.3.10.3 (old) into the ACTION statement (where they belong, since they constitute action statements).

5) Modify Figure 3.3.10-1 (old) to more clearly identify the separate regions where 1) noise monitoring is required, and 2) operation is prohibited.

6) Remove ACTION statement b. from 3.3.10 and incorporate it in a separate LCO, 3/4.2.6, Power/Flow Stability. Currently, this ACTION statement exists with no stated LCO.

JUSTIFICATION TO CHANGES to 3/4.1.1

1) Gather 15 minute actions and remove them from within the 4 hour criteria.

2) Scram and rod block trip setpoints do not need to be changed as the analysis envelopes operation up to 75% thermal power with single loop core flows and drive flows.

3) 34% core flow limit is in accordance with the SIL 380 recommendations of Minimum Forced Circulation. The true minimum forced circulation at WNP-2 is 24%, however, the normal flow line followed during reactor startup is the 2 pump, 15-Hz, FCV full open line, which is the 34% line. The previous value, 39%, was based on the flow that Duane Arnold, a non-flow control valve plant, could achieve with the recirc pump motor generator scoop tubes set at minimum.

4) Move power/flow instability requirements under power distribution.

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

SECTION | PAGE

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LIST OF FIGURES

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) Two Recirculation Loop Operation~~		
1) ~~a)~~ Flow Biased	≤ 0.66W + 51%, with a maximum of	≤ 0.66W + 54%, with a maximum of
2) ~~b)~~ High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
~~2) Single Recirculation Loop Operation~~		
~~a) Flow Biased~~	~~≤ 0.66W + 47.7%, with a maximum of~~	~~≤ 0.66W + 50.7%, with a maximum of~~
~~b) High Flow Clamped~~	~~≤ 113.5% of RATED THERMAL POWER~~	~~≤ 115.5% of RATED THERMAL POWER~~
c. Fixed Neutron Flux - High	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	N.A.	N.A.
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*	≥ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 10.0% closed	≤ 12.5% closed
6. Main Steam Line Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background

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

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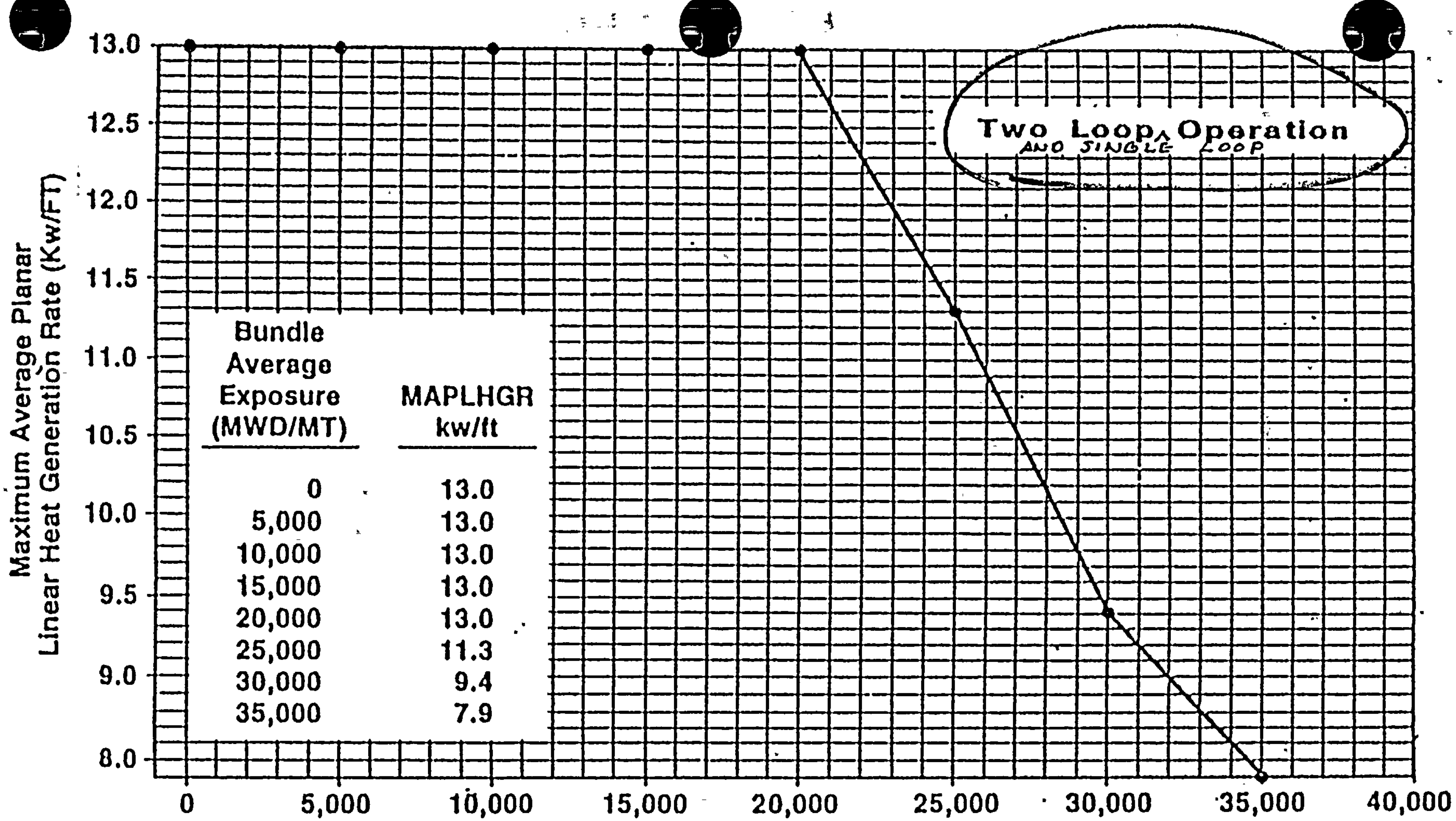

Bundle Average Exposure (MWD/MT)	MAPLHGR kw/ft
0	13.0
5,000	13.0
10,000	13.0
15,000	13.0
20,000	13.0
25,000	11.3
30,000	9.4
35,000	7.9

Bundle Average Exposure (MWD/MT)

ANF 8x8 Reload Fuel
Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Bundle Average Exposure

Figure 3.2.1-3

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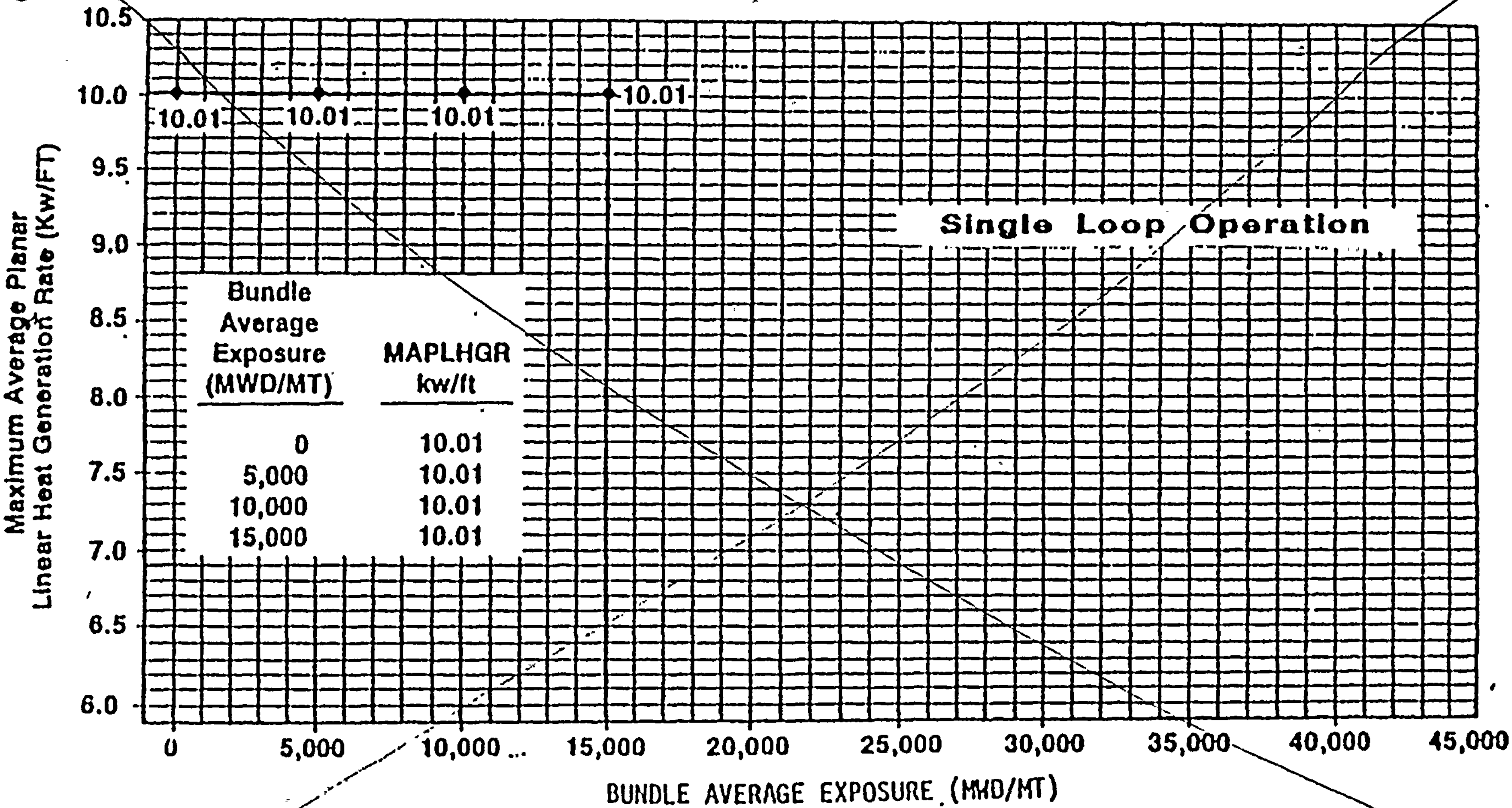

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus
BUNDLE AVERAGE EXPOSURE
ANF 8x8 Reload Fuel
Figure 3.2.1-6

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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:

~~a. Two Recirculation Loop Operation~~

TRIP SETPOINT	ALLOWABLE VALUE
$S \leq (0.66W + 51\%)T$	$S \leq (0.66W + 54\%)T$
$S_{RB} \leq (0.66W + 42\%)T$	$S_{RB} \leq (0.66W + 45\%)T$

~~b. Single Recirculation Loop Operation~~

~~TRIP SETPOINT~~	~~ALLOWABLE VALUE~~
~~$S \leq (0.66W + 47.7\%)T$~~	~~$S \leq (0.66W + 50.7\%)T$~~
~~$S_{RB} \leq (0.66W + 38.7\%)T$~~	~~$S_{RB} \leq (0.66W + 41.7\%)T$~~

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 108.5 million lbs/h.

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.

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 for each class of fuel 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.

*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.

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POWER DISTRIBUTION LIMITS

4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be:

a. Greater than or equal to the applicable MCPR limit determined from Table 3.2.3-1 during steady state operation at or above rated core flow, ~~or~~ in two loop operation, or when in single loop operation, or

b. Greater than or equal to the greater of the two values determined from Table 3.2.3-1 and Figure 3.2.3-1 during steady state operation at ~~other~~ less than rated core flow, when in two recirculation loop operation.

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

ACTION: With MCPR less than the applicable MCPR limit determined from Table 3.2.3-1 and Figure 3.2.3-1, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25 percent of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

2.3.1 MCPR shall be determined to be greater than or equal to the applicable MCPR limit determined from Table 3.2.3-1 and Figure 3.2.3-1.

a. At least once per 24 hours,

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

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

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Table 3.2.3-1

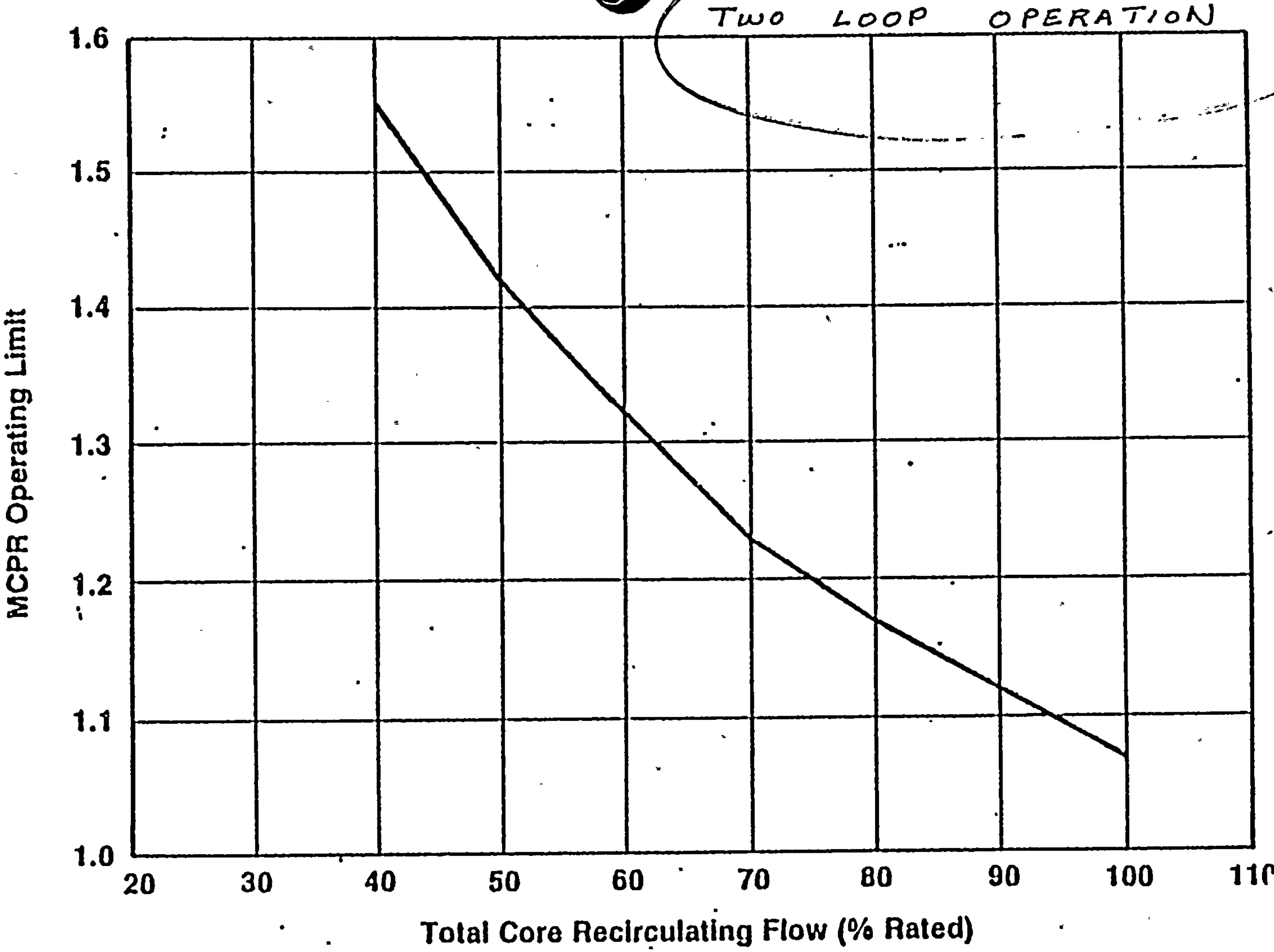
MCPR OPERATING LIMITS

	Cycle Exposure	Equipment Status	MCPR Operating Limit Up to 106% Core Flow	
			GE Fuel	ANF Fuel
1.	0 MWD/MTU - 4150 MWD/MTU	*	1.29	1.26
2.	4150 MWD/MTU - EOC MWD/MTU	Normal scram times**	1.32	1.30
3.	4150 MWD/MTU - EOC MWD/MTU	Control rod insertion bounded by Tech. Spec. limits (3.1.3.4 - p 3/4 1-7)	1.39	1.35
4.	4150 MWD/MTU - EOC MWD/MTU	RPT inoperable Normal scram times	1.37	1.35
5.	4150 MWD/MTU - EOC MWD/MTU	RPT inoperable Control rod insertion bounded by Tech. Spec. limits (3.1.3.4 - p 3/4 1-7)	1.43	1.39
	0 MWD/MTU - EOC MWD/MTU	SINGLE LOOP OPERATION NORMAL SCRAM TIMES	1.35	1.35

*In this portion of the fuel cycle, operation with the given MCPR operating limits is allowed for both normal and Tech. Spec. scram times and for both RPT operable and inoperable.

**These MCPR values are based on the ANF Reload Safety Analysis performed using the control rod insertion times shown below (defined as normal scram). In the event that surveillance 4.1.3.2 shows these scram insertion times have been exceeded, the plant thermal limits associated with normal scram times default to the values associated with Tech. Spec. scram times (3.1.3.4-p 3/4 1-7), and the scram insertion times must meet the requirements of Tech. Spec. 3.1.3.4.

Position Inserted From Fully Withdrawn	Slowest measured average control rod insertion times to specified notches for all operable control rods for each group of 4 control rods arranged in a a two-by-two array (seconds)
Notch 45	.404
Notch 39	.660
Notch 25	1.504
Notch 5	2.624



Reduced Flow MCPR Operating Limit
Figure 3.2.3-1

IN GRAPHICS

POWER DISTRIBUTION LIMITS

3/4.2.6 POWER/FLOW INSTABILITY

LIMITING CONDITION FOR OPERATION

3.2.6 Operation with THERMAL POWER/core flow conditions which lay in the crosshatched region of Figure 3.2.6-1 is prohibited.

APPLICABILITY: OPERATIONAL CONDITION 1

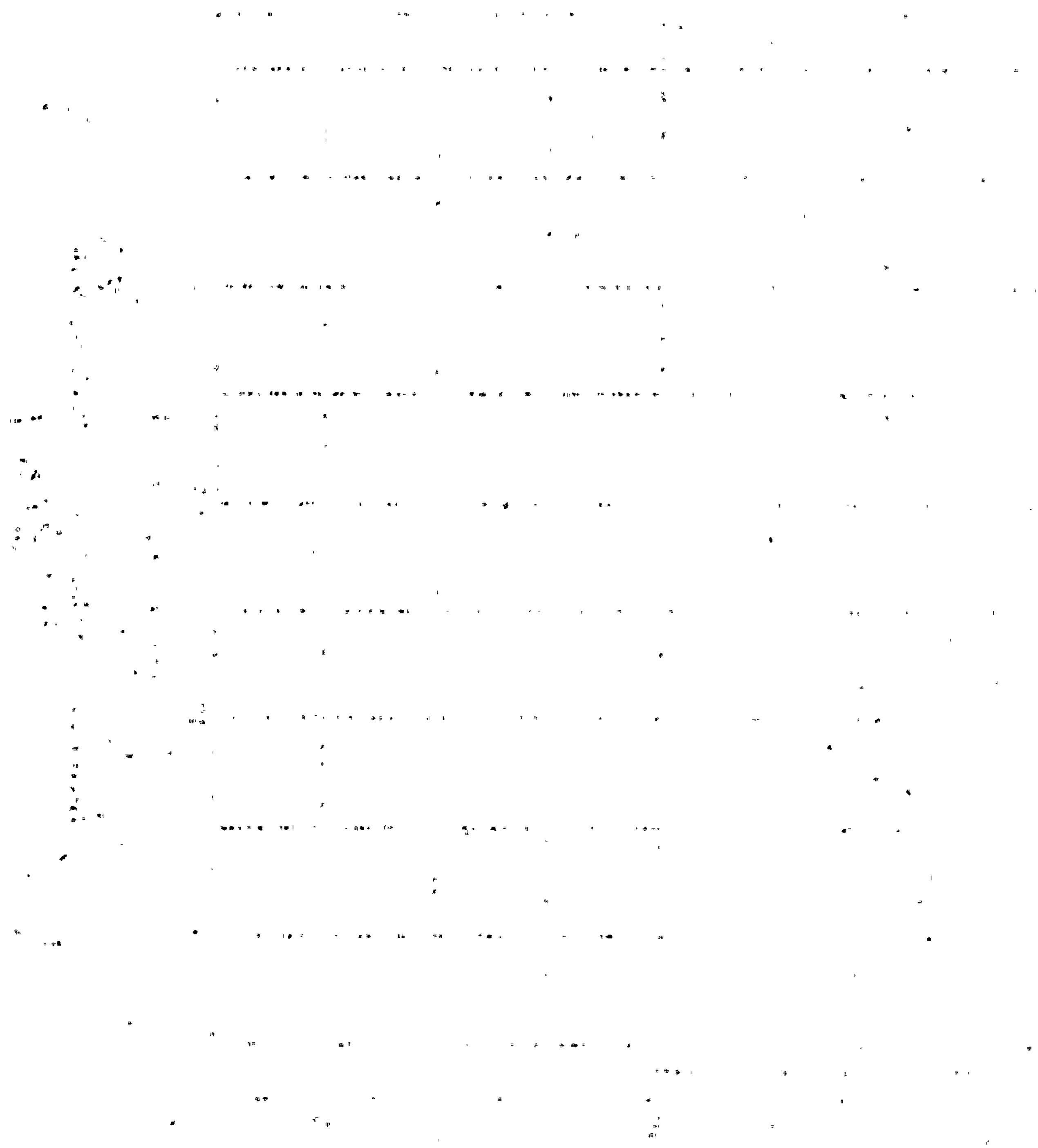
When THERMAL POWER is greater than 39% of RATED THERMAL POWER and core flow is less than or equal to 45% of rated core flow.

ACTION:

With THERMAL POWER/core flow conditions which lay in the crosshatched region of Figure 3.2.6-1, initiate corrective action within 15 minutes to establish a THERMAL POWER/core flow condition which lays outside the crosshatched region within 2 hours.

SURVEILLANCE REQUIREMENTS

4.2.6 The THERMAL POWER/core flow conditions shall be verified to lay outside the crosshatched region of Figure 3.2.6-1 once per 24 hours.



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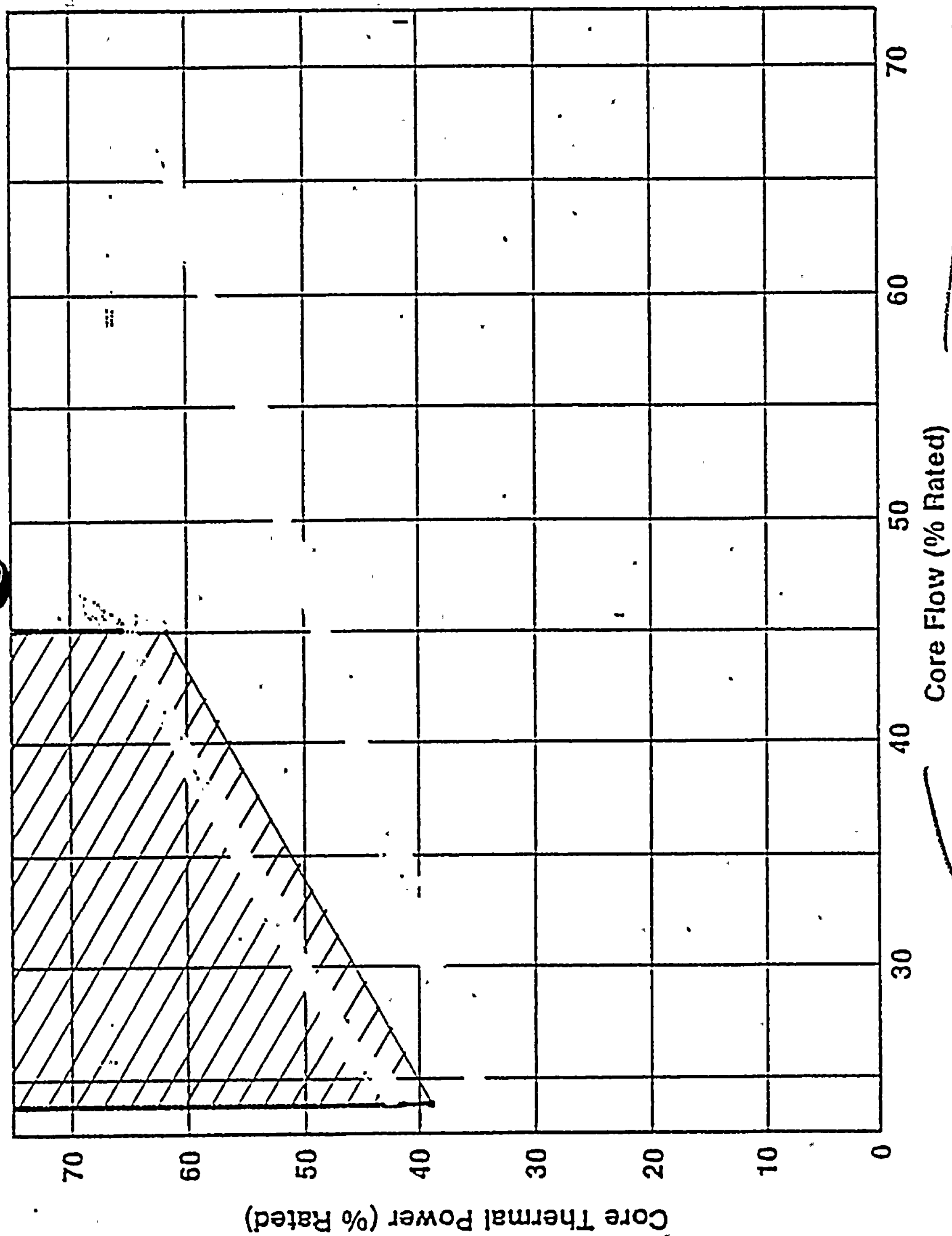

Figure ~~3/4.10-X~~ 3.2.6-1
~~Thermal Power~~ Operating Region Limits of Specification ~~3/4.10.X~~ 3.2.6

INSTRUMENTATION

3/4.2.7 NEUTRON FLUX NOISE MONITORING

ITING CONDITION FOR OPERATION

3.2.7 The APRM and LPRM neutron flux noise levels shall not exceed three (3) times their established baseline values when operating in the region of APPLICABILITY.

APPLICABILITY: OPERATIONAL CONDITION 1 with THERMAL POWER/core flow in Region B of Figure 3.2.7-1, with two reactor coolant system recirculation loops in operation and total core flow less than 45% of rated total core flow, or with one reactor coolant system recirculation loop not in operation.

ACTION:

a. If baseline APRM and LPRM neutron flux noise levels have not been established for the appropriate reactor coolant system condition (one or two loop operation) since the most recent CORE ALTERATION, then:

   Within 2 hours exit the region of APPLICABILITY. Establish baseline APRM and LPRM neutron flux noise levels prior to re-entering Region B of Figure 3.2.7-1.

b. If baseline APRM and LPRM neutron flux noise levels have been established for the appropriate reactor coolant system condition (one or two loop operation) since the most recent CORE ALTERATION, then:

   With the APRM or LPRM neutron flux noise levels greater than three (3) times their established noise levels, initiate corrective action within 15 minutes to restore the noise levels to within the required limits within 2 hours or reduce THERMAL POWER to below the region of APPLICABILITY within the next 2 hours.

SURVEILLANCE REQUIREMENTS

4.2.7.1 The provisions of Specification 4.0.4 are not applicable.

4.2.7.2 The APRM and LPRM neutron flux noise levels shall be determined to be less than or equal to three (3) times their established baseline values:

   a. At least once per 8 hours, and

   b. Within 30 minutes after completion of a THERMAL POWER increase of greater than or equal to 5% of rated THERMAL POWER.

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tector levels A and C of one LPRM string per core octant plus detector levels and C of one LPRM string in the center of the core should be monitored.

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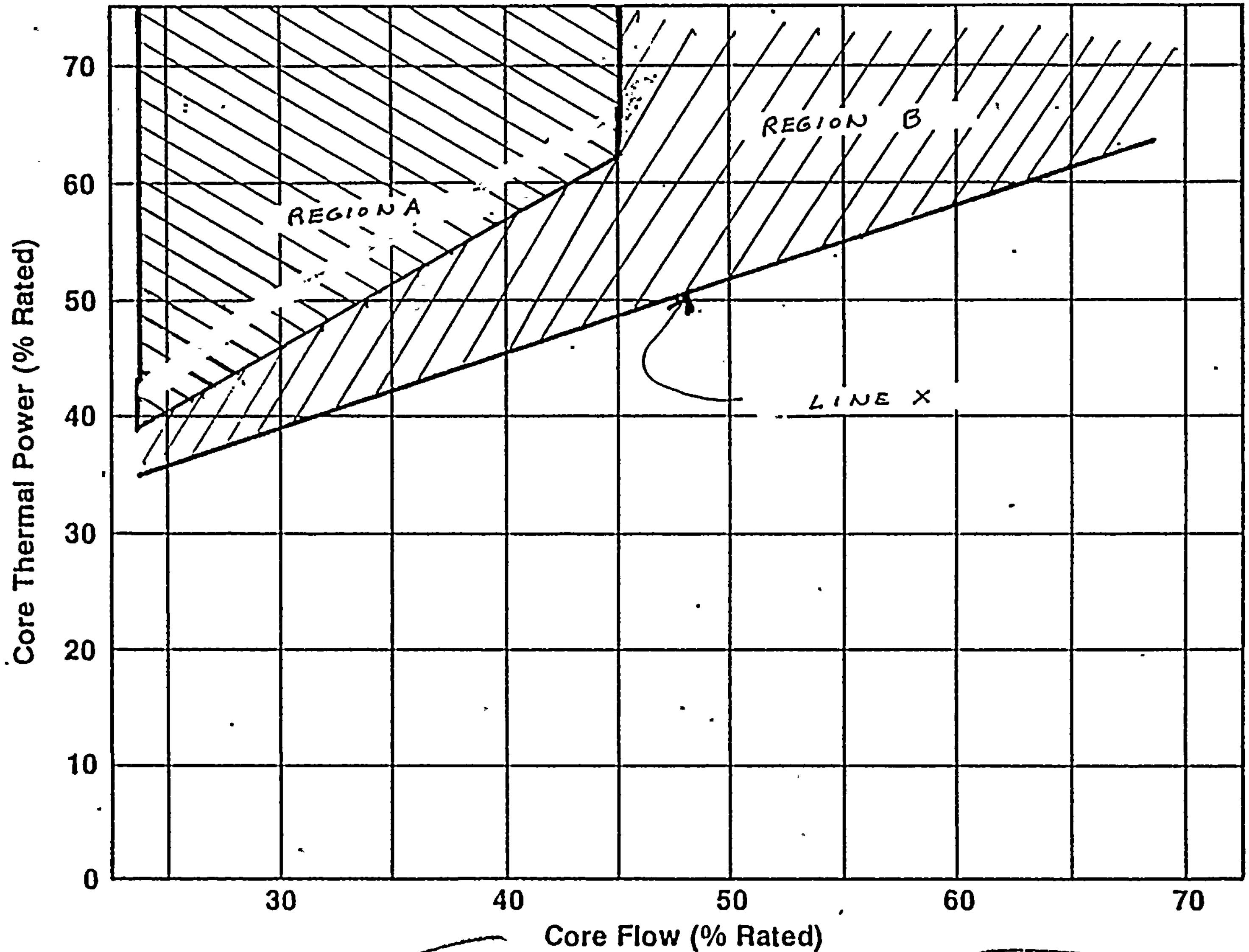

OPERATING REGION ~~Thermal Power~~ Limits of Specification ~~3.3.10-1~~ 3.2.~~8~~7
Figure ~~3.3.10-1~~ 3.2.7.-1

IN GRAPHICS

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale		
~~1) Two Recirculation Loop Operation~~	≤ 0.66 W + 40%	≤ 0.66 W + 43%
~~2) Single Recirculation Loop Operation~~	~~≤ 0.66 W + 36.7%~~	~~≤ 0.66 W + 39.7%~~
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux Upscale		
~~1) Two Recirculation Loop Operation~~	≤ 0.66 W + 42%*	≤ 0.66 W + 45%*
~~2) Single Recirculation Loop Operation~~	~~≤ 0.66 W + 38.7%~~	~~≤ 0.66 W + 41.7%~~
b. Inoperative	N.A.	N.A.
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	N.A.	N.A.
b. Upscale	≤ 1×10^5 cps	≤ 1.6×10^5 cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 0.7 cps	≥ 0.5 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	≤ 527 ft 2 in. elevation	≤ 527 ft 4 in. elevation
b. Scram Trip Bypass	N.A.	N.A.
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 108/125 divisions of full scale	≤ 111/125 divisions of full scale
b. Inoperative	N.A.	N.A.
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.

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3/4.3.10 NEUTRON FLUX MONITORING INSTRUMENTATION

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LIMITING CONDITION FOR OPERATION

3.3.10 The APRM and LPRM* neutron flux noise levels shall not exceed three (3) times their established baseline values when operating in the allowable region of Figure 3.3.10-1.

APPLICABILITY: OPERATIONAL CONDITION 1 with two reactor coolant system recirculation loops in operation with THERMAL POWER greater than the limit specified in Figure 3.3.10-1 and total core flow less than 45% of rated total core flow or with one reactor coolant system recirculation loop not in operation with THERMAL POWER greater than the limit specified in Figure 3.3.10-1.

ACTION:

a. With the APRM or LPRM* neutron flux noise level greater than three (3) times their established baseline noise levels, initiate corrective action within 15 minutes to restore the noise levels to within the required limits within 2 hours or reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.3.10-1 within the next 2 hours.

b. With reactor power/core flow in the crosshatched region of Figure 3.3.10-1, initiate corrective action within 15 minutes to reduce power by control rod insertion to a reactor power/core flow below the crosshatched region within 2 hours.

SURVEILLANCE REQUIREMENTS

4.3.10.1 The provisions of Specification 4.0.4 are not applicable.

4.3.10.2 With two reactor coolant system recirculation loops in operation, establish a baseline APRM and LPRM* neutron flux noise level value within 2 hours upon entering the APPLICABLE OPERATIONAL CONDITION of Specification 3.3.10 provided that baselining has not been performed since the most recent CORE ALTERATION.

4.3.10.3 With one reactor coolant system recirculation loop not in operation, establish a baseline APRM and LPRM* neutron flux noise level value with THERMAL POWER less than or equal to the limit specified in Figure 3.3.10-1 prior to entering the APPLICABLE OPERATIONAL CONDITION of Specification 3.3.10 provided baselining has not been performed with one reactor coolant system recirculation loop not in operation since the most recent CORE ALTERATION.#

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SURVEILLANCE REQUIREMENTS (Continued)

4.3.10.4 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 and the reactor power/core flow shall be verified to lie outside the crosshatched region of Figure 3.3.10-1 when operating within the APPLICABLE OPERATIONAL CONDITION of Specification 3.3.10:

a. At least once per 8 hours, and

b. Within 30 minutes after completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER.

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

#The baseline data obtained in Specification 4.3.10.3 is applicable to operation with one reactor coolant system recirculation loop not in operation and THERMAL POWER greater than the limits specified in Figure 3.3.10-1.

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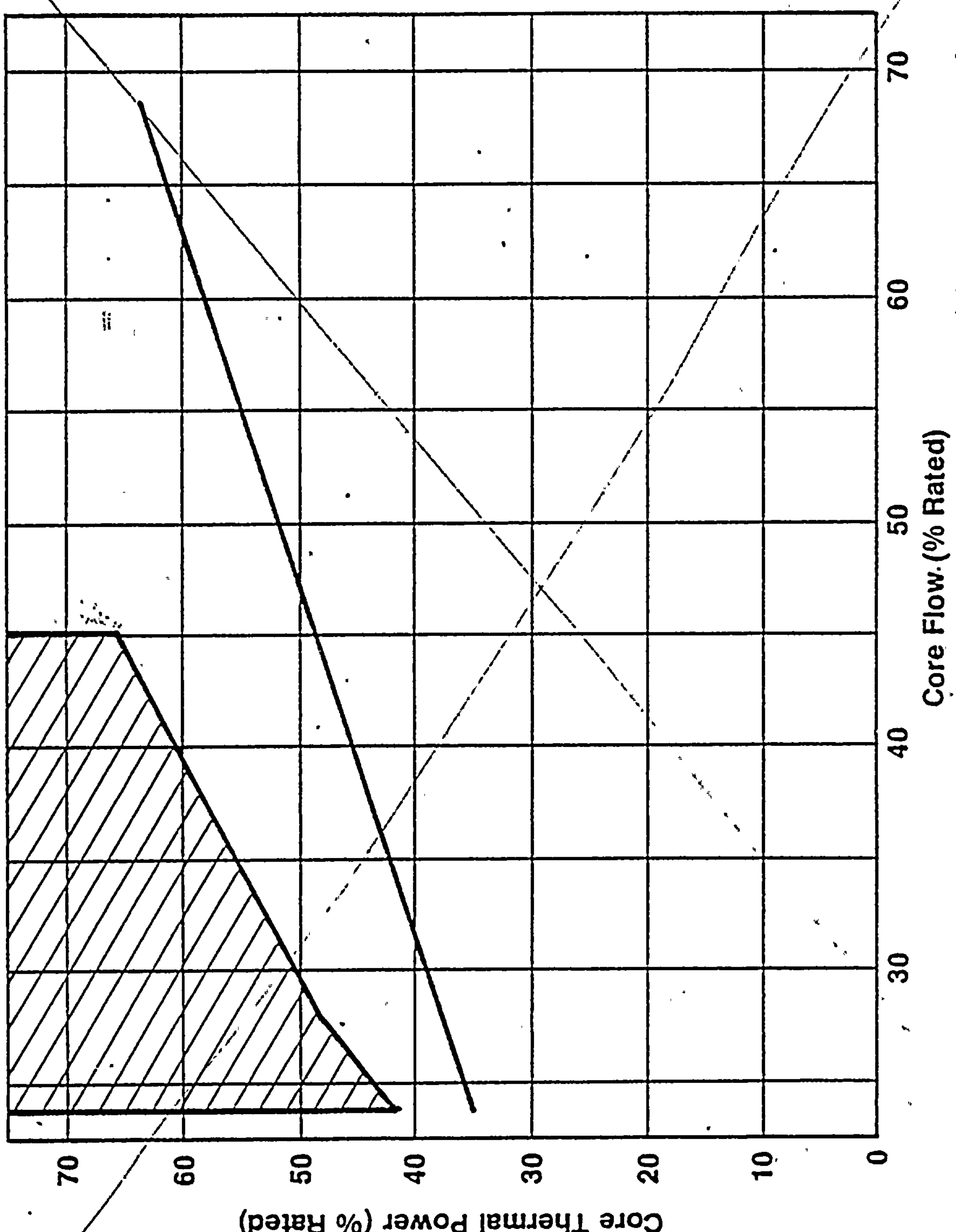

Figure 3.3.10-1
Thermal Power Limits of Specification 3.3.10-1

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3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

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

INSERT 1 → 2 ~~1.~~ Within 4 hours:

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

~~b) The THERMAL POWER shall be less than or equal to the limit specified in Figure 3.4.1.1-1 or the provisions of Specification 4.3.10.3 are satisfied. With one reactor coolant system recirculation loop not in operation and with THERMAL POWER greater than the limit specified in Figure 3.4.1.1-1, and the provisions of Specification 4.3.10.3 having not been satisfied, initiate action within 15 minutes to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 4 hours. The provisions of Specification 4.3.10.3 must be satisfied prior to resuming power operation above the limit specified in Figure 3.4.1.1-1.~~

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

c ~~d)~~ Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit FOR GENERAL ELECTRIC FUEL to a value of 0.84 times the two recirculation loop operation limit per Specification 3.2.1, and,

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

d ~~f)~~ Reduce the volumetric flow rate of the operating recirculation loop to ≤ 41,725** gpm.

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*See Special Test Exception 3.10.4.

**This value represents the actual volumetric recirculation loop flow which produces 100% core flow at 100% THERMAL POWER. This value was determined during the Startup Test Program.

INSERT

1. Within 15 minutes:

   a. Verify that core flow is 39% of rated core flow or that THERMAL POWER/core flow conditions lay below the line in Figure 3.4.1.1-1. With core flow 39% of rated core flow and THERMAL POWER/core flow conditions above the line in Figure 3.4.1.1-1, initiate action to reduce THERMAL POWER to below the line in Figure 3.4.1.1-1 or increase core flow to 39% of rated core flow within the next 4 hours.

   b. Verify that the requirements of LCO 3.2.7 are met, or comply with the associated ACTION statement within the specified time limits.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

e. ~~g)~~ Perform Surveillance Requirement 4.4.1.1.2 if THERMAL POWER is ≤ 25%*** of RATED THERMAL POWER or the recirculation loop flow in the operating loop is ≤ 10%*** of rated loop flow.

f ~~h)~~ Reduce recirculation loop flow in the operating loop until the core plate ΔP noise does not deviate from the established core plate ΔP noise patterns by more than 100%.

~~i) With one reactor coolant system recirculation loop not in operation and THERMAL POWER greater than the limit specified in Figure 3.4.1.1-1 and core flow less than 39% of rated core flow, initiate action within 15 minutes to reduce THERMAL POWER to less than or equal to the limit specified in Fig. 3.4.1.1-1 or increase core flow to greater than or equal to 39% of rated core flow within 4 hours.~~

~~2.~~ 3 The provisions of Specification 3.0.4 are not applicable.

~~3.~~ 4 Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant system recirculation loops in operation, immediately initiate measures to place the unit in at least HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 With one reactor coolant system recirculation loop not in operation, at least once per 8 hours verify that:

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

b. The volumetric flow rate of the operating loop is ≤ 41,725 gpm.**

**This value represents the actual volumetric recirculation loop flow which produces 100% core flow at 100% THERMAL POWER. This value was determined during the Startup Test Program.

***Final values were determined during Startup Testing based upon actual THERMAL POWER and recirculation loop flow which will sweep the cold water from the vessel bottom head preventing stratification.

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BASES

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

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. For GE fuel, 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) for GE fuel is this LHGR of the highest powered rod divided by its local peaking factor which results in a calculated LOCA PCT much less than 2200°F. The Technical Specification APLHGR for ANF fuel is specified to assure the PCT following a postulated LOCA will not exceed the 2200°F limit. The limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 for two recirculation loop operation. ~~These values shall be multiplied by a factor of 0.84 for single recirculation loop operation. This multiplier is determined from comparison of the limiting analysis between two recirculation loop and single recirculation loop operation.~~ AND FIGURES 3.2.1-4 AND 3.2.1-5 FOR SINGLE LOOP OPERATION. FIGURE 3.2.1-3 APPLIES TO BOTH SINGLE AND TWO LOOP OPERATION

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 is based on a loss-of-coolant accident analysis. The analysis was performed using calculational models which are consistent with the requirements of Appendix K to 10 CFR Part 50. These models are described in Reference 1 or XN-NF-80-19, Volumes 2, 2A, 2B and 2C, Rev. 1.

BASES

3/4.2.6 POWER/FLOW INSTABILITY

At the high power/low flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., power shape, bundle power, and bundle flow).

In February, 1984, GE issued SIl 380 addressing boiling instability and supplying several recommendations. In this SIL, the power/flow map was divided into several regions of varying concern. It also discussed the objectives and philosophy of "detect and suppress," coining the phrase.

The ANF topical report for COTRAN (XN-NF-691P) discusses boiling instability. The SER written on this topical (dated May 10, 1984) interprets the topical to require that the detect and suppress surveillance be used in regions which have code calculated decay ratios .75 or greater and that operation is forbidden in regions having calculated decay ratios of .9 and greater.

The NRC Generic Letter 86-02 addressed both GE and ANF (then EXXON) stability calculation methodology and stated that due to uncertainties, General Design Criterias 10 and 12 could not be met using analytic procedures on a BWR 5 design. The letter espoused GE SIL 380 and stated that General Design Criterias 10 and 12 could be met by imposing the SIl 380 recommendations in operating regions of potential instability. The NRC concluded that regions of potential instability constituted calculated decay ratios of .8 and greater by the GE methodology and .75 and greater by the EXXON methodology.

Predicated on the SIL 380 endorsement, WNP-2 has divided the power/flow map on the following boundary lines:

1. 80% rod line
2. 45% core flow line
3. APRM rod block line minus 3% power
4. Natural Circulation flow line
5. Minimum Forced Circulation for normal recirculation lineup.

This division conforms to the SIL 380 recommendations with a 3% power penalty on the APRM rod block line. For LCO 3.2.6, the region of concern is bounded by the APRM rod block line, minus 3% power, the natural circulation flow line, and the 45% core flow line. Calculated decay ratios between the two flow lines and on the APRM rod block line minus 3% must be less than .9. Operation in the region between the two flow lines and above the rod block line minus 3% is forbidden due to the potential for boiling instabilities.

For the ease of annual licensing submittals, a 3% margin from the rod block line is taken to avail the opportunity to submit with no Technical Specification changes under the provisions of 10CFR50.59. This 3% provides margin to ensure that vendor stability calculations can easily support the allowable operating region. For calculational ease the power boundary is linearized between two points, (24% Flow, 39% Power) and (45% Flow, 62% Power).

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring and controlling the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

This specification is provided to ensure that the turbine overspeed protection system 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.

3/4.3.9 FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

The feedwater system/main turbine trip system actuation instrumentation is provided to initiate the feedwater system/main turbine trip system in the event of reactor vessel water level equal to or greater than the level 8 setpoint associated with a feedwater controller failure.

DELETE

~~3/4.3.10 NEUTRON FLUX MONITORING INSTRUMENTATION~~

~~At the high power/low flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., rod patterns, power shape). To provide assurance that neutron flux limit cycle oscillations are detected and suppressed, APRM and LPRM neutron flux noise levels should be monitored while operating in this region.~~

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

DELETE

MONITORING INSTRUMENTATION (Continued)

NEUTRON FLUX MONITORING INSTRUMENTATION (Continued)

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

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

In the case of single loop operation (SLO), the normal neutron flux noise may increase more rapidly when reverse flow occurs in the inactive jet pumps. This justifies a smaller flow range under high flow SLO conditions. Baseline data should be taken at flow intervals which correspond to less than a 50% increase in APRM neutron flux noise level. If baseline data are not specifically available for SLO, than baseline data with two recirculation loops in operation can be conservatively applied to SLO since for the same core flow SLO will exhibit higher neutron flux noise levels than operation with two loops. However, because of reverse flow characteristics of SLO, the core flow/drive flow relationship is different than the two loop relationship and therefore the baseline data for SLO should be based on the active loop recirculation drive flow, and not the core flow. Because of the uncertainties involved in SLO at high reverse flows, baseline data should be taken at or below the power specified in Figure 3.4.1.1-1. This will result in approximately a 25% conservative baseline value if compared to baseline data taken near the rated rod line and will therefore not result in an overly restrictive baseline value, while providing sufficient margin to cover uncertainties associated with SLO

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BASES

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring and controlling the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

This specification is provided to ensure that the turbine overspeed protection system 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.

3/4.3.9 FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

The feedwater system/main turbine trip system actuation instrumentation is provided to initiate the feedwater system/main turbine trip system in the event of reactor vessel water level equal to or greater than the level 8 setpoint associated with a feedwater controller failure.

3/4.~~3.10~~ 2.7 NEUTRON FLUX NOISE MONITORING ~~INSTRUMENTATION~~

At the high power/low flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., rod patterns, power shape). To provide assurance that neutron flux limit cycle oscillations are detected and suppressed, APRM and LPRM neutron flux noise levels should be monitored while operating in this region.

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservative decay ratio of ~~0.8~~ 0.75 was chosen as the bases for determining the generic region for surveillance to account for the plant to plant variability of decay ratio with core and fuel designs. This generic region has been determined to correspond to a core flow of less than or equal to 45% of rated core flow and a thermal power greater than that specified in Figure 3.4.1.1-1 (Reference).

MONITORING INSTRUMENTATION (Continued)

NEUTRON FLUX NOISE MONITORING ~~INSTRUMENTATION~~ (Continued)

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

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

In the case of single loop operation (SLO), the normal neutron flux noise may increase more rapidly when reverse flow occurs in the inactive jet pumps. This justifies a smaller flow range under high flow SLO conditions. Baseline data should be taken at flow intervals which correspond to less than a 50% increase in APRM neutron flux noise level. If baseline data are not specifically available for SLO, then baseline data with two recirculation loops in operation can be conservatively applied to SLO since for the same core flow SLO will exhibit higher neutron flux noise levels than operation with two loops. However, because of reverse flow characteristics of SLO, the core flow/drive flow relationship is different than the two loop relationship and therefore the baseline data for SLO should be based on the active loop recirculation drive flow, and not the core flow. Because of the uncertainties involved in SLO at high reverse flows, baseline data should be taken at or below the power specified in Figure 3.4.1.1-1. This will result in approximately a 25% conservative baseline value if compared to baseline data taken near the rated rod line and will therefore not result in an overly restrictive baseline value, while providing sufficient margin to cover uncertainties associated with SLO.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor recirculation loop inoperable has been evaluated and been found to be acceptable ~~during the first fuel cycle only,~~ provided the unit is operated in accordance with the single recirculation loop operation Technical Specifications herein.

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

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. 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 capacity is designed to limit the primary system pressure, including transients, in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, 1971, Nuclear Power Plant components (up to and including Summer 1971 Addenda). The Code allows a peak pressure of 110% of design pressure (1250 (design) X 1.10 = 1375 psig maximum) under upset conditions. In addition, the Code specifications require that the lowest valve setpoint be at or below design pressure and the highest valve setpoint be set so that total accumulated pressure does not exceed 110% of the design pressure.

The safety valve sizing evaluation assumes credit for operation of the scram protective system which may be tripped by one of two sources; i.e., a direct position switch or neutron flux signal. The direct scram signal is derived from position switches mounted on the main steamline isolation valves (MSIV's) or the turbine stop valve, or from pressure switches mounted on the dump valve of the turbine control valve hydraulic actuation system. The position switches are actuated when the respective valves are closing, and following 10% travel of full stroke. The pressure switches are actuated when a fast closure of the control valves is initiated. Further, no credit is taken for power operation of the pressure relieving devices. Credit is only taken f r

DVANCED NUCLEAR FUELS CORPORATION

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February 22, 1988 R- 2/24/88
JBE:067:88
ANFWP-88-0020

Washington Public Power Supply System
3000 George Washington Way
P. O. Box 968
Richland, WA 98669-0968

Attn: Manager, Central Contracts

Gentlemen:

In response to a telephone request by the Supply System on February 4, 1988, the single loop control rod withdrawal error results have been revised and are enclosed.

Very truly yours,

J. B. Edgar tm

J. B. Edgar
Contract Administrator

tlm

Enclosure

WNP-2 SINGLE LOOP CONTROL ROD WITHDRAWAL ERROR (REVISED)

The limiting Cycle 3 control rod withdrawal case has been rerun at the single loop conditions provided by the Supply System in Reference 1. The single loop power and flow conditions for the analysis were 2492 MWt (75%) and 57.8 Mlb/hr (53.3%). The initial control rod pattern for the analysis is shown on Figure 5.1 of the Cycle 3 reload analysis report XN-NF-87-25. The calculated Cycle 3 ΔCPR and CPR values for single loop and two loop are as follows:

Single Loop Result

Rod Block Monitor Setting	Distance Withdrawn (ft)	Single Loop ΔCPR ANF	Single Loop ΔCPR GE	Single Loop CPR ANF	Single Loop CPR GE
	0.0	---	---	1.544	1.819
106%	5.0	0.22	0.26	1.326	1.564
107%	5.0	0.22	0.26	1.326	1.564
108%	6.0	0.24	0.29	1.302	1.526

Two Loop Result (from XN-NF-87-25)

Rod Block Monitor Setting	Distance Withdrawn (ft)	Two Loop ΔCPR ANF	Two Loop ΔCPR GE	Two Loop CPR ANF	Two Loop CPR GE
	0.0	---	---	1.369	1.605
106%*	4.5	0.20	0.23	1.173	1.376
107%	4.5	0.20	0.23	1.173	1.376
108%	5.0	0.22	0.25	1.154	1.352

* The Cycle 3 setting is 106% and the CRWE based MCPR operating limit is 1.26 for the ANF fuel.

The above reported single loop control rod withdrawal error (CRWE) calculation was performed to demonstrate that the 1.35 CPR value used to initialize the single loop ECCS analysis could be used as the single loop CPR limit. The single loop CRWE results presented in this letter show that the single loop CPR values at the reduced power and flow conditions are significantly higher than the two loop CPR values (more margin to the CPR safety limit). The single loop ΔCPR values are slightly larger than the two loop ΔCPR values as expected for the high starting CPR values. Based on the ANF experience with the CRWE analysis, a lower starting CPR results in a smaller calculated ΔCPR value.

The original basis for the flow dependent CPR limit for the two loop operation is the pump runup event. In the single loop configuration, however, the additional constraint of the reduced flow MCPR operating limits is no longer required (Reference 2). A two pump flow runup is not possible as the pump in

the idle loop is not running. An inadvertent start of the idle pump cannot affect flow appreciably as the pump is interlocked to prevent starting unless its associated flow control valve is at the minimum position (Reference 3).

For future operation, a constant single loop CPR limit of 1.35 is considered adequate for the single loop operational mode. This single loop limit is related to the two loop CRWE limit being equal to or less than 1.26. If the two loop CPR limit based on CRWE for the limiting fuel type is greater than 1.26, a cycle specific review of the single loop CPR limit may be required.

REFERENCES

1) Letter WPANF-2B-87-0101 from RA Vopalensky to JB Edgar dated November 20, 1987.

2) JE Krajicek, "WNP-2 Single Loop Operation Analysis", ANF-87-119, Advanced Nuclear Fuels Corporation, Richland, WA 99352, September 1987.

3) WNP-2 FSAR, Chapter 4, Section 4.4.3.3.3, pages 4.4-5 and 4.4-6 (Design Features for Power Flow Control).