Enclosure 3

Technical Specifications Page Changes with Justification

A description of the Technical Specification changes to allow Fermi 2 single loop operation and their respective justification and reasons are given below:

1. Page 2-1, Specification 2.1.2

   Increase in MCPR Safety Limit from 1.06 to 1.07 for Single Loop Operation (SLO) to account for additional uncertainty in establishing Safety Limits. With one recirculation loop only in operation, there is an increase in core flow measurement uncertainty. Additional uncertainty is experienced also with TIP measurements.

2. Page 2-3 Specification 2.2.1

   Added footnote to specify time allowance to comply with SLO requirements consistent with Specification 3.4.1.1.

3. Page 2-4, Table 2.2.1-1

   Change of Functional Unit 2.b setpoint to include SLO. Provision is included to adjust APRM gains in lieu of changing setpoints. This provides an expeditious method of complying with single loop requirements.

4. Page B2-1, Bases 2.0

   Bases correction consistent with Specification 2.1.2.

5. Page B2-3, Bases Table B2.1.2-1

   Bases correction consistent with Specification 2.1.2.

6. Page B 2-7, Bases 2.2.1

   Bases addition to include ΔW value for SLO and explanation of the non-applicability of the High Flow Clamped Flow Biased Neutron Flux-High setpoint in SLO.

7. Page 3/4 2-1, Specification 3.2.1

   A MAPLHGR multiplier is provided for SLO. This multiplier is used primarily to allow for the marginal difference in uncovered time and reflood time associated with a LOCA during SLO.

8. Page 3/4 2-5, Specification 3/4.2.2

Change of APRM Scram and Rod Block setpoint to include SLO.

Footnote * modified and # added to include provisions for APRM gain adjustment as discussed above.

9. Page 3/4 3-41 Specification 3.3.6

Added footnote to specify time allowance to comply with SLO requirements consistent with Specification 3.4.1.1.

10. Page 3/4 3-44, Table 3.3.6-2

Change of Rod Block Monitor and APRM Rod Block setpoint to include SLO. Provision for adjusting APRM gains included.

11. New pages 3/4 3-90, 3/4 3-91 and 3/4 3-92; New Specification 3.3.10.

Consolidated stability related requirements in single specification. Additional requirements for single loop operation are included.

12. Page 3/4 4-1 and Two Additional Pages, Specification 3.4.1.1

Included actions for SLO including stratification limits. Stability requirements moved to new Specification 3.3.10.

13. Page 3/4 4-2 Specification 4.4.1.1.1, 4.4.1.1.2, 4.4.1.1.3, 4.4.1.1.4

Additional surveillance requirements added to reflect limits imposed for SLO, and to reflect stratification limits. Stability surveillances moved to new Specification 3.3.10.

14. Page 3/4 4-4 Specification 4.4.1.2

Change of surveillance requirement 4.4.1.2 to reflect SLO.

Added Specification 4.0.4 allowance to overcome difficulty in performing this surveillance below 25% THERMAL POWER.

15. Page 3/4 4-5 Specification 3.4.1.3

    a. Change APPLICABILITY to allow SLO.

    b. Change to ACTION b. to improve flexibility in operation.

    c. Inclusion of ACTION to reach HOT SHUTDOWN.

16. Page B 3/4 1-2, Bases 3/4.1.3

    Corrections made to properly reflect safety limit which is 1.07 for SLO.

17. Pages B 3/4 2-1 Bases 3/4 2.1

    Bases of reduced MAPLHGR limits for SLO included.

18. Page B 3/4 2-2, Bases 3/4.2.2

    Correction made to paragraph 3/4.2.2 to properly reflect the safety limit which is 1.07 for SLO.

19. Page B 3/4 2-3 Bases Table B 3.2.1-1

    Footnote added to indicate conservatism introduced for SLO.

20. Page B 3/4 2-4 Bases 3/4.2.3

    a. Reference to 1.06 removed to properly reflect the safety limit which is 1.07 for SLO.

    b. Deletion of last two sentences of second paragraph to properly reflect safety limit of 1.07 and remove misleading statement on determination of MCPR operating limit.

21. Pages B 3/4 3-8, Bases 3/4.3.10

    Add Bases for new Specification 3/4.3.10.

22. Page B 3/4 4-1, Bases 3/4.4.1

    a. Deletion of first sentence - no longer applicable. Replace with "Insert G," justifying and describing new conditions for SLO. Also provides reasons for coolant stratification and vibration limits.

b. Extra words added to third paragraph plus "Insert H." Included to amplify reason for flow mismatch requirement.

c. Last sentence of fourth paragraph deleted and "Insert I" added to further describe stratification limit and reason for concern.

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 for two recirculation loop operation and shall not be less than 1.07 for single loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.06 for two recirculation loop operation or less than 1.07 for single loop operation and with 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.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

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

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

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

* The APRM Flow biased instrumentation need not be declared inoperable upon entering single recirculation loop operation provided the setpoints are adjusted within 4 hours per Specification 3.4.1.1.

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~~	≤ 0.66 W+51%, with ~~a maximum of~~	≤ ~~0.66 W+54%, with~~ a maximum of
2) High Flow Clamped	≤ ~~113.5% of RATED~~ THERMAL POWER	≤ 115.5% of RATED ~~THERMAL POWER~~
c. Fixed Neutron Flux-Upscale	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	N.A.	N.A.
3. Reactor Vessel Steam Dome Pressure - High	≤ 1068 psig	≤ 1088 psig
4. Reactor Vessel Low Water Level - Level 3	≥ 173.4 inches*	≥ 171.9 inches
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
6. Main Steam Line Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background
7. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Float Switch	≤ 594'8"	≤ 596'0"
b. Level Transmitter	≤ 592'6"	≤ 596'0"
9. Turbine Stop Valve - Closure	≤ 5% closed	≤ 7% closed
10. Turbine Control Valve Fast Closure	Initiation of fast closure	N.A.
11. Reactor Mode Switch Shutdown Position	N.A.	N.A.
12. Manual Scram	N.A.	N.A.
13. Backup Manual Scram	N.A.	N.A.

Replace with Insert A

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

Insert A

	TRIP SETPOINT	ALLOWABLE VALUE
1) During two recirculation loop operation:		
a. Flow Biased	≤0.66 W+51%, with a maximum of	≤0.66W+54%, with a maximum of
b. High Flow Clamped	≤113.5% RATED THERMAL POWER	≤115.5% RATED THERMAL POWER
2) During single recirculation loop operation:		
a. Flow Biased	≤0.66W+45.7%,**	≤0.66W+48.7%,**
b. High Flow Clamped	NA	NA

** During single recirculation loop operation, rather than adjusting the APRM Flow Biased Setpoints to comply with the single loop values, the gain of the APRMs may be adjusted for a period not to exceed 72 hours such that the final APRM readings are at least 5.3% of rated power greater than 100% times FRTP, provided that the adjusted APRM readings do not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

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. 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.

For two recirculation loop operation and 1.07 for single recirculation loop operation

2.1.1 THERMAL POWER, Low Pressure or Low Flow

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

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

Two recirculation loop operation 2.5
Single recirculation loop operation 6.0

Two recirculation loop operation 6.3
Single recirculation loop operation 6.8

* 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. The values herein apply to both two recirculation loop operation and single recirculation loop operation, except as noted.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Fixed Neutron Flux-Upscale setpoint; i.e, for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow Biased Neutron Flux-High setpoint, a time constant of 6 ± 1 seconds is introduced into the flow biased APRM in order to simulate the fuel thermal transient characteristics. A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown. The flow referenced trip setpoint must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when MFLPD is greater than or equal to FRTP. Add Insert.

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine stop valve closure trip is bypassed. For a turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

INSERT PAGE B 2-7

For single recirculation loop operation, the reduced APRM setpoints are based on a ΔW value of 8%. The ΔW value corrects for the difference in indicated drive flow (in percentage of drive flow which produces rated core flow) between two loop and single loop operation of the same core flow. The 5.3% decrease in setpoint is derived from 0.66 x 8%. The High Flow Clamped Flow Biased Neutron Flux-High setpoint is not applicable to single loop operation as core power levels which would require this limit are not achievable in a single loop configuration.

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 during two recirculation loop operation. The limits of Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 shall be reduced to a value of 0.90 times the two recirculation loop operation limit when in single recirculation loop operation.

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

ACTION:

With an APLHGR exceeding the 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.

POWER DISTRIBUTION LIMITS

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

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

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

Replace with Insert B

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, at 100% of RATED THERMAL POWER
T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY. T is applied only if less than or equal to 1.0

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

ACTION:

With the APRM flow biased neutron flux-high scram trip setpoint and/or the flow biased neutron flux-high 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 6 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 neutron flux-high scram and flow biased neutron flux-high control rod block trip setpoints verified to be within the above limits or adjusted, or the APRM gain readings shall be verified as indicated below,** 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.

Replace with footnote insert

~~*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.~~

Insert B

1. During two recirculation loop operation:

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

2. During single recirculation loop operation:

$S \leq (0.66W+45.7\%)T$	$S \leq (0.66W+48.7\%)T$
$S_{RB} \leq (0.66W+36.7\%)T$	$S_{RB} \leq (0.66W+39.7\%)T$

Footnote Insert Page 3/4 2-5

* 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 readings do not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel. With MFLPD greater than FRTP and a single recirculation loop in operation, if the APRM flow biased setpoints have not been adjusted to their single loop values then the minimum required APRM reading must be increased by an additional 5.3% of rated power.

\# During single recirculation loop operation with FRTP greater than or equal to MFLPD, rather than adjusting the APRM setpoints to comply with the single loop values, the APRM gain may be adjusted for a period not to exceed 72 hours such that the final APRM readings are at least 5.3% of rated power greater than 100% times FRTP, provided that the adjusted APRM readings do not exceed 100% RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod block instrumentation channel trip setpoint* less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

*The APRM Flow Biased Neutron Flux-High and Rod Block Monitor instrumentation need not be declared inoperable upon entering single reactor recirculation loop operation provided the setpoints are adjusted within hours per Specification 3.4.1.1.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
~~a. Upscale~~	~~< 0.66 W + 40%~~	~~< 0.66 W + 43%~~ Replace with Insert C
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
~~a. Flow Biased Neutron Flux - High~~	~~< 0.66 W + 42%*~~	~~< 0.66 W + 45%*~~ Replace with Insert D
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Setdown	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	< 1.0×10^5 cps	< 1.6×10^5 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	< 589'11½"	< 591'0"
b. Scram Trip Bypass	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	< 108/125% of rated flow	< 111/125% of rated flow
b. Inoperative	NA	NA
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	NA

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

**The downscale rodblock setpoint count rate may be reduced to 0.3 cps prior to achieving a burnup of 2000 MWD/T on the first core provided the signal-to-noise ratio is >2. After a burnup of 2000 MWD/T on the first core, the count rate may be reduced to 0.7 cps provided the signal-to-noise ratio is ≥2.

Insert C:

a. Upscale

1) During two recirculation loop operation	$\leq$0.66W+40%	$\leq$0.66W+43%
2) During single recirculation loop operation	$\leq$0.66W+34.7%#	$\leq$0.66W+37.7%#

Insert D:

a. Flow Biased Neutron Flux-High

1) During two recirculation loop operation	$\leq$0.66W+42%*	$\leq$0.66W+45%*
2) During single recirculation loop operation	$\leq$0.66W+36.7%#*	$\leq$0.66W+39.7%#*

During single recirculation loop operation, rather than adjusting the APRM and RBM Flow Biased Setpoints to comply with the single loop values, the gain of the APRMs may be adjusted for a period not to exceed 72 hours such that the final APRM readings are at least 5.3% of rated power greater than 100% times FRTP, provided that the adjusted APRM readings do not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

3/4.3.10 NEUTRON FLUX MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITION 1 with 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.

ACTION: With the APRM or LPRM* neutron flux noise level greater than three (3) times their established baseline noise levels, immediately initiate corrective action 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.

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.#

#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.

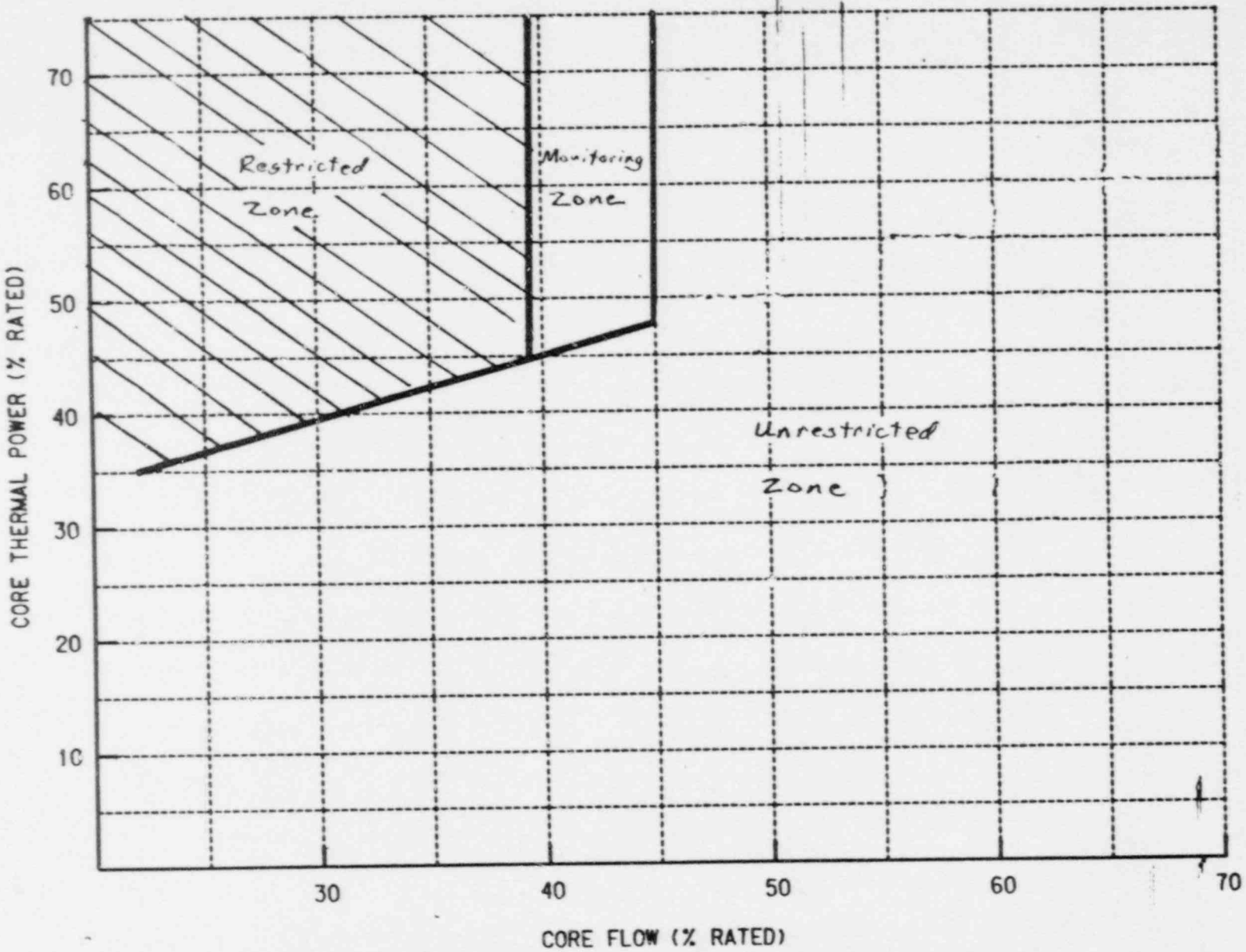
*Detector levels A and C of one LPRM string per core octant plus detector levels A and C of one LPRM string in the center of the core should be monitored.

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 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.



THERMAL POWER VERSUS CORE FLOW

FIGURE 3.3.10-1

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. ~~When total core flow is less than 45% of rated core flow, then THERMAL POWER must be less than or equal to the limit specified in Figure 3.4.1.1-1.~~

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

Replace with Insert E

~~a. With one reactor coolant system recirculation loop not in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours and initiate measures to place the unit in at least HOT SHUTDOWN within 12 hours.~~

b. With no reactor coolant system recirculation loops in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours 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% of rated core flow and THERMAL POWER greater than the limit specified in Figure 3.4.1.1-1:~~

~~1. Monitor the APRM and LPRM** noise levels (Surveillance 4.4.1.1.3):~~

~~a) Within 8 hours of entry into this condition and at least once per 24 hours thereafter while in this condition and,~~

~~b) Within 30 minutes after the completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER in an hour by control rod movement.~~

~~2. With the APRM or LPRM** neutron flux noise levels greater than three times their established baseline noise levels, immediately initiate corrective action to restore the noise levels to within the required limits within 2 hours by increasing core flow to greater than 45% of rated core flow or by reducing THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1.~~

---

*See Special Test Exception 3.10.4.

~~**Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored when operating with a nonsymmetric control rod pattern. Only the center of the core LPRM string detectors A and C and two other LPRM string detectors A and C need be monitored for operations with a symmetric control rod pattern.~~

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

1. Within 4 hours:

a) Place the individual recirculation pump flow controller for the operating recirculation pump in the Manual mode.

b) Reduce THERMAL POWER to less than or equal to 70% of RATED THERMAL POWER.

c) Limit the speed of the operating recirculation pump to less than or equal to 75% of rated pump speed.

d) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to 1.07 per Specification 2.1.2.

e) Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit to a value of 0.90 times the two recirculation loop operation limit per Specification 3.2.1.

f) 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.

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

---

* APRM gain adjustments may be made in lieu of adjusting the APRM and RBM Flow Biased Setpoints to comply with the single loop values for a period of up to 72 hours.

\# Value to be determined during Startup Test Program (Core flow with both recirculation pumps at minimum speed). Final value to be provided within 90 days of completion of Startup Test Program.

h) The provisions of Specification 4.3.10.3 must be satisfied unless THERMAL POWER is less than or equal to the limit specified in Figure 3.4.1.1-1 or total core flow is greater than or equal to 45% of rated core flow. 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 total core flow less than 45% of rated core flow, and the provisions of Specification 4.3.10.3 having not been satisfied, immediately initiate action to, within the next 4 hours, reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 or increase total core flow to greater than or equal to 45% of rated core flow.

i) Perform Surveillance Requirement 4.4.1.1.4 if THERMAL POWER is less than or equal to 30% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is less than or equal to 50% of rated loop flow.

2. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

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

~~4.4.1.1.3 Establish a baseline APRM and LPRM** neutron flux noise value within the regions for which monitoring is required (Specification 3.4.1.1, ACTION c) within 2 hours of entering the region for which monitoring is required unless baselining has previously been performed in the region since the last refueling outage~~

Add Insert F

*If not performed within the previous 31 days.

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

Insert F:

4.4.1.1.3 With one reactor coolant system recirculation loop not in operation, at least once per 12 hours verify that:

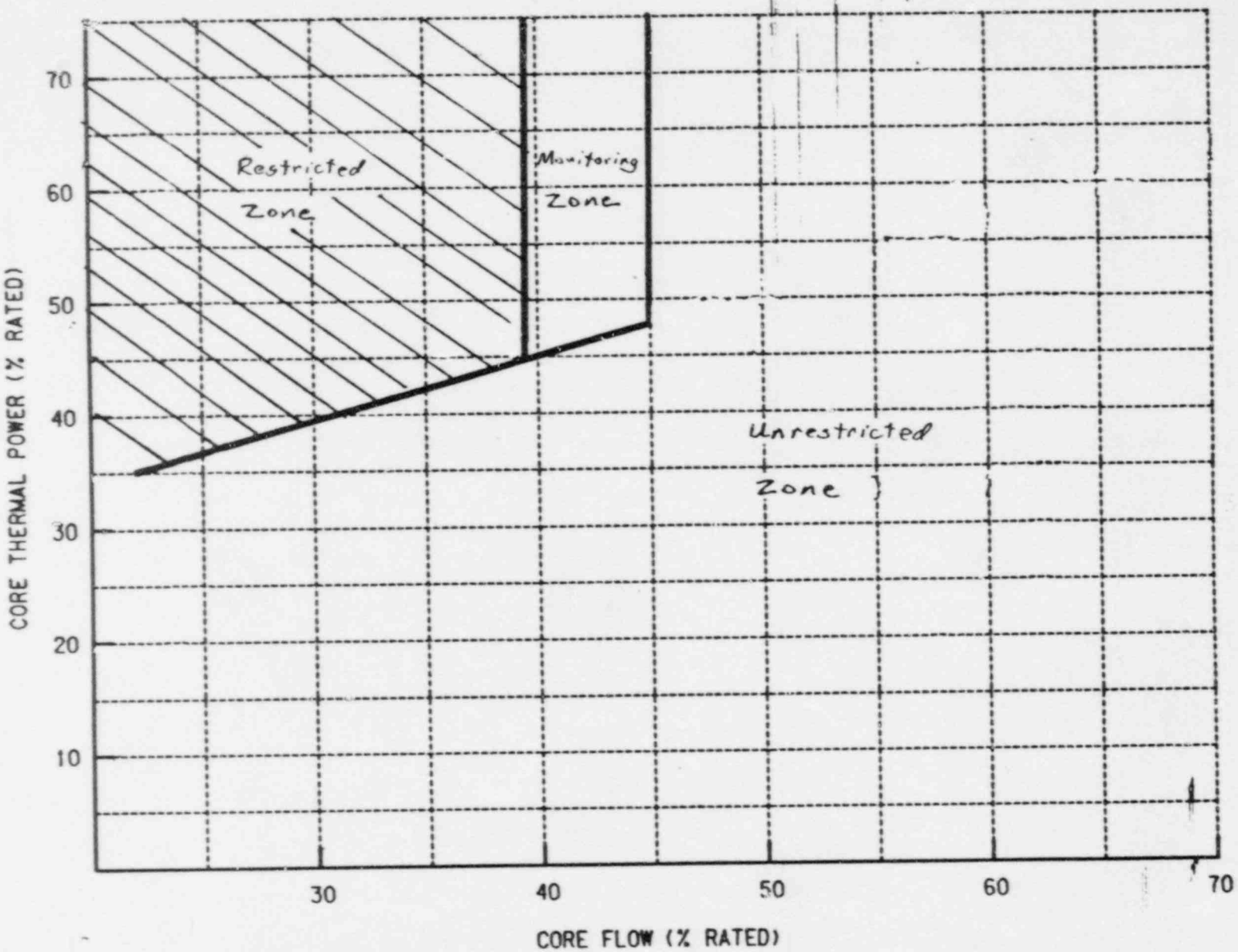
a. THERMAL POWER is less than or equal to 70% of RATED THERMAL POWER, and
b. The individual recirculation pump flow controller for the operating recirculation pump is in the Manual mode, and
c. The speed of the operating recirculation pump is less than or equal to 75% of rated pump speed, and
d. Core flow is greater than 39%# of rated core flow when THERMAL POWER is greater than the limit specified in Figure 3.4.1.1-1.

4.4.1.1.4 With one reactor coolant system loop not in operation with THERMAL POWER less than or equal to 30% of RATED THERMAL POWER or with recirculation loop flow in the operating loop less than or equal to 50% of rated loop flow, verify the following differential temperature requirements are met within no more than 15 minutes prior to either THERMAL POWER increase or recirculation flow increase:

a. Less than or equal to 145°F between reactor vessel steam space coolant and bottom head drain line coolant, and
b. Less than or equal to 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel***, and
c. Less than or equal to 50°F between the reactor coolant within the loop not in operation and the operating loop.***

*** Requirement does not apply when the recirculation loop not in operation is isolated from the reactor pressure vessel.

Value to be established during Startup Test Program (coreflow with both recirculation pumps at minimum speed). Final Value to be provided with 90 days of completion of Startup Test Program.



THERMAL POWER VERSUS CORE FLOW

FIGURE 3.4.1.1-1

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.1.2 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* with THERMAL POWER greater than 25% of RATED THERMAL POWER by determining operating recirculation loop flow(s), 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 operating recirculation loop flow(s) 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 the mean of all jet pump differential pressures in the same loop by more than 20% deviation from its normal* deviation.

The provisions of Specification 4.0.4 are not applicable provided that this surveillance is performed within 12 hours after exceeding 25% of RATED THERMAL POWER.

*During the start[illegible]t program, data shall be recorded for the parameters listed to provi[illegible]sis for establishing the specified relationships. Comparisons of the [illegible]tual data in accordance with the criteria listed shall commence upon the conclusion of the startup test program.

For both single recirculation loop and two recirculation loop operation.

RECIRCULATION PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation pump speed shall be maintained within:

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

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

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*x during two recirculation loop 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. Shutdown one of the recirculation loops ~~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.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exception 3.10.4.

3/4.1.3 CONTROL RODS

The specifications of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the safety analyses, 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 shut down for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than the fuel cladding safety limit ~~1.06~~ during the limiting power transient analyzed in Section 15B 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 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 POWER DISTRIBUTION LIMITS

BASES

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

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2 and 3.2.1-3.

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 General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared to previous analyses can be broken down as follows.

a. Input Changes

1. Corrected Vaporization Calculation - Coefficients in the vaporization correlation used in the REFLOOD code were corrected.

2. Incorporated more accurate bypass areas - The bypass areas in the top guide were recalculated using a more accurate technique.

3. Corrected guide tube thermal resistance.

4. Correct heat capacity of reactor internals heat nodes.

For plant operation with a single operating recirculation loop, the MAPLHGR limits of Figure 3.2.1-1, 3.2.1-2, and 3.2.1-3 are multiplied by 0.90. The constant factor of 0.90 is derived from LOCA analysis initiated from single loop operation to account for earlier boiling transition at the limiting fuel node compared to the standard LOCA analysis.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

b. Model Change

1. Core CCFL pressure differential - 1 psi - Incorporate the assumption that flow from the bypass to lower plenum must overcome a 1 psi pressure drop in core.

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

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

a. Input Change

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

b. Model Change

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

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

3/4.2.2 APRM SETPOINTS

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

BASES TABLE B 3.2.1-1

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

Plant Parameters:

Core THERMAL POWER.....	3430 MWt* which corresponds to 105% of rated steam flow
Vessel Steam Output.....	14.86 x 10^6 lbm/hr which corresponds to 105% of rated steam flow
Vessel Steam Dome Pressure.....	1055 psia

Design Basis Recirculation Line Break Area for:

a. Large Breaks 4.1 ft^2

b. Small Breaks 0.1 ft

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kW/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	8 x 8	13.4	1.4	1.18 **

A more detailed listing of input of each model and its source is presented in Section II of Reference 1 and subsection 6.3 of the FSAR.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

** For single recirculation loop operation, loss of nucl ate boiling is assumed at 0.1 second after LOCA regardless of initial MCPR.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady-state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR ~~of 1.06,~~ and an analysis of abnormal operational transients. For any abnormal operating transients 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 15B.0-1 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 3.2.3-2 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 3.2.3-2 were developed 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 setpoint 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.

3/4.3.10 NEUTRON FLUX MONITORING INSTRUMENTATION

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

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservative decay ratio of 0.6 was chosen as the bases for determining the generic region for surveillance to account for the plant to plant variability of decay ratio with core and fuel designs. This generic region has been determined to correspond to a core flow of less than or equal to 45% of rated core flow and a thermal power greater than that specified in Figure 3.3.10-1 (Reference 1).

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. Neutron flux noise levels which significantly bound these values are considered in the thermal/mechanical design of GE BWR fuel and are found to be of negligible consequence (Reference 2). In addition, stability tests at operating BWRs have demonstrated that when stability related neutron flux limit cycle oscillations occur they result in peak-to-peak neutron flux limit cycles of 5-10 times the typical values. Therefore, actions taken to reduce neutron flux noise levels exceeding three (3) times the typical value are sufficient to ensure early detection of limit cycle neutron flux oscillations.

Baseline data with two reactor recirculation loops in operation 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. Because of the uncertainties involved in SLO at high reverse flows, baseline data should be taken at or below the power specified in Figure 3.3.10-1. 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.

References

(1) "BWR Core Thermal-Hydraulic Stability" Service Information Letter 380, Revision 1, February 1984.
(2) G. A. Watford, "Compliance of the General Electric Boiling Water Reactor Fuel Designs to Stability Licensing Criteria," December 1982 (NEDE 22277-P).

3/4.4.1 RECIRCULATION SYSTEM

~~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.~~ Insert G

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 mismatch limits are in compliance with the ECCS LOCA analysis design criteria for two recirculation loop operation. Insert H

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.~~ Insert I

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 11 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.

The low-low set system ensures that a potentially high thrust load (designated as load case C.3.3) on the SRV discharge lines is eliminated during subsequent actuations. This is achieved by automatically lowering the closing setpoint of two valves and lowering the opening setpoint of two valves following the initial opening. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

Insert G:

The impact of single recirculation loop operation upon plant safety is assessed and shows that single-loop operation is permitted if the MCPR fuel cladding safety limit is increased as noted by Specification 2.1.2. APRM scram and control rod block setpoints (or APRM gains) are adjusted as noted in Tables 2.2.1-1 and 3.3.6-2, respectively. MAPLHGR limits are decreased by the factor given in Specification 3.2.1. A time period of 4 hours is allowed to make these adjustments following the establishment of single loop operation since the need for single loop operation often cannot be anticipated. MCPR operating limits adjustments in Specification 3.2.3 for different plant operating situations are applicable to both single and two recirculation loop operation.

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

Insert H:

The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA.

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

Insert I:

Sudden equalization of a temperature difference >145°F between the reactor vessel bottom head coolant and the coolant in the upper region of the reactor vessel by increasing core flow rate would cause undue stress in the reactor vessel bottom head.