UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 61
License No. NPF-58

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

   A. The application for amendment by The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, and Toledo Edison Company (the licensees) dated June 28, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-58 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 61 are hereby incorporated into this license. The Cleveland Electric Illuminating Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented by January 31, 1995.

FOR THE NUCLEAR REGULATORY COMMISSION

Jon B. Hopkins

Jon B. Hopkins, Senior Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of issuance: June 2, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 61

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

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

Remove	Insert
2-1	2-1
2-3	2-3
2-4	2-4
2-5	2-5
B 2-1	B 2-1
B 2-2	B 2-2
B 2-3	-
B 2-4	-
B 2-7	B 2-7
3/4 3-8	3/4 3-8
3/4 3-55	3/4 3-55
3/4 3-58	3/4 3-58
3/4 4-1	3/4 4-1
3/4 4-2	3/4 4-2
-------	3/4 4-2a*
-------	3/4 4-2b*
3/4 4-4	3/4 4-4
3/4 4-5	3/4 4-5
B 3/4 1-2	B 3/4 1-2
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	B 3/4 2-5
B 3/4 3-4	B 3/4 3-4
B 3/4 4-1	B 3/4 4-1
-------	B 3/4 4-1a*
-------	B 3/4 4-1b*
--------	B 3/4 4-1c*
B 3/4 4-2	B 3/4 4-2

* Denotes new page

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 1.07 (1.08 during single recirculation 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 the above Safety Limit and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

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

SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

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

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4 and 5

ACTION:

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

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

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

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

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

* The APRM flow biased instrumentation need not be declared inoperable upon entering single recirculation loop operation provided the setpoints are adjusted within 8 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:		
a. Neutron Flux - High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
b. Inoperative	NA	NA
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) During two recirculation loop operation:		
a) Flow Biased	≤ 0.66 W + 64%[a], with a maximum of	≤ 0.66W + 67%[a], with a maximum of
b) High Flow Clamped	≤ 111.0% of RATED THERMAL POWER	≤ 113.0% of RATED THERMAL POWER
2) During single recirculation loop operation:		
a) Flow Biased	≤ 0.66W + 42.7%[a][b]	≤ 0.66W + 45.7%[a][b]
b) High Flow Clamped	Not Required OPERABLE	Not Required OPERABLE
c. Neutron Flux-High	≤ 118.0% of RATED THERMAL POWER	≤ 120.0% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1064.7 psig	≤ 1079.7 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 177.7 inches above top of active fuel[c]	≥ 177.1 inches above top of active fuel[c]
5. Reactor Vessel Water Level-High, Level 8	≤ 219.5 inches above top of active fuel[c]	≤ 220.1 inches above top of active fuel[c]
6. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
7. Deleted		
8. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig

TABLE 2.2.1-1 (Continued)
REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Scram Discharge Volume Water Level - High		
a. Level Transmitter	≤ 37.9 inches[d] (626' 6.6" elevation)	≤ 38.87 inches[d] (626' 7.56" elevation)
b. Float Switches		
C11N013A	≤ 626' 9.87" elevation	≤ 626' 11.5" elevation
C11N013B	≤ 626' 10.25" elevation	≤ 626' 11.5" elevation
C11N013C	≤ 626' 10.87" elevation	≤ 626' 11.5" elevation
C11N013D	≤ 626' 11.18" elevation	≤ 626' 11.5" elevation
10. Turbine Stop Valve - Closure	≤ 5% closed	≤ 7% closed
11. Turbine Control Valve Fast Closure, Valve Trip Oil Pressure - Low	≥ 530 psig	≥ 465 psig
12. Reactor Mode Switch Shutdown Position	NA	NA
13. Manual Scram	NA	NA

(a) The Average Power Range Monitor flow biased scram function varies as a function of recirculation loop drive flow W. During single loop operation W is adjusted to account for the difference in indicated drive flow as described in the Bases.

(b) To functionally implement this protective function during entry into single loop operation, APRM gain adjustments may be made in lieu of adjusting the APRM Flow Biased Simulated Thermal Power-High Trip Setpoint and Allowable Value equations for a period not to exceed 72 hours provided that the APRM gains are adjusted to a value at least 21.3% of RATED THERMAL POWER higher than the actual thermal power.

(c) See Bases Figure B 3/4 3-1.

(d) Level zero is 623' 4.69" elevation.

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 the value given in Specification 2.1.2. MCPR greater than this limit (for the corresponding mode of operation) represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the General Electric critical power correlations (Reference 1) are not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28×10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

2.1.2 THERMAL POWER, High Pressure and High Flow

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

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB (Reference 1), which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using a GE critical power correlation. This correlation is valid over the range of conditions used in the tests of the data used to develop this correlation.

Details of the fuel cladding integrity Safety Limit calculation are given in Reference 2. Reference 2 provides the uncertainties used in the determination of the Safety Limit MCPR for both two loop and single recirculation loop operation, and of the nominal values of the parameters used in the Safety Limit MCPR statistical analysis.

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

2. "General Electric Standard Application for Reactor Fuel, GESTAR-II," NEDE-24011-P-A (latest approved revision).

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (continued)

Average Power Range Monitor (Continued)

5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Neutron Flux-High setpoint; i.e., for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow Biased Simulated Thermal Power-High setpoint, a time constant specified in the COLR 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. For single recirculation loop operation, the reduced APRM setpoints are based on a delta W value of 8%. The delta W 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 at the same core flow. The decrease in setpoint from MEOD to SLO conditions is derived by multiplying the slope of the flow biased setpoint equations by 8%, which results in a reduction in setpoint of 5.3%, and then decreasing an additional 16% to account for the difference between the MEOD and standard power-flow map rodlines for a total decrease in setpoint of 21.3% for SLO. APRM gain adjustments of at least 21.3% (to be consistent with the above basis) may be made to functionally implement the single loop equation setpoints (while these equations are being changed over) for a limited time period.

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 control valve fast closure and turbine stop valve closure trips are bypassed. For a load rejection or turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint has been used in transient analyses dealing with coolant inventory decrease. The scram setting was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

5. Reactor Vessel Water Level-High

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

6. Main Steam Line Isolation Valve-Closure

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

7. Deleted

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Within 24 hours prior to startup, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reaching 25% of RATED THERMAL POWER. To functionally implement this protective function during entry into single loop operation, APRM channel gain adjustments may be made in lieu of adjusting the APRM Flow Biased Simulated Thermal Power-High Trip Setpoint and Allowable Value equations for a period not to exceed 72 hours, provided the criteria in Note b to Table 2.2.1-1 are met. Any APRM channel gain adjustments made in compliance with Specifications 2.2.1 and 3.3.1 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 MWD/T using the TIP system.
(g) Calibrate trip unit setpoint at least once per 31 days.
(h) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing loop flow (APRM % flow).
(i) This calibration shall consist of verifying that the simulated thermal power time constant is within the limits specified in the COLR.
(j) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
(k) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
(l) This function is not required to be OPERABLE when Drywell Integrity is not required.
(m) The CHANNEL CALIBRATION shall exclude the flow reference transmitters, these transmitters shall be calibrated at least once per 18 months.

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

4.3.6.1 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.

4.3.6.2 The provisions of Specification 4.0.4 are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for the Intermediate Range Monitors and Source Range Monitors for entry into their applicable OPERATIONAL CONDITIONS (as shown in Table 4.3.6-1) from OPERATIONAL CONDITION 1 provided the surveillances are performed within 12 hours after such entry.

*The ARM flow biased instrumentation need not be declared inoperable upon entering single recirculation loop operation provided the setpoints are adjusted within 8 hours per Specification 3.4.1.1.

TABLE 3.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD PATTERN CONTROL SYSTEM			
a. Low Power Setpoint	2	1, 2	60
b. RWL - High Power Setpoint	2	1	60
2. APRM			
a. Flow Biased Neutron Flux - Upscale	6	1	61
b. Inoperative	6	1, 2, 5	61
c. Downscale	6	1	61
d. Neutron Flux - Upscale, Startup	6	2, 5	61
3. SOURCE RANGE MONITORS			
a. Detector not full in(a)	3 2**	2# 5	61 61
b. Upscale(b)	3 2**	2# 5	61 61
c. Inoperative(b)	3 2**	2# 5	61 61
d. Downscale(c)	3 2**	2# 5	61 61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale(d)	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5*	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	6	1	62
7. REACTOR MODE SWITCH SHUTDOWN POSITION	2	3, 4	63

TABLE 3.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Declare the RPCS inoperable and take the ACTION required by Specification 3.1.4.2.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

ACTION 63 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, initiate a rod block.

NOTES

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

**OPERABLE channels must be associated with SRMs required OPERABLE per Specification 3.9.2.

#With IRMs on range 2 or below.

(a)This function is automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range 3 or higher.

(b)This function is automatically bypassed when the associated IRM channels are on range 8 or higher.

(c)This function is automatically bypassed when the IRM channels are on range 3 or higher.

(d)This function is automatically bypassed when the IRM channels are on range 1.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD PATTERN CONTROL SYSTEM		
a. Low Power Setpoint	20 + 15, - 0% of RATED THERMAL POWER[a]	20 + 15, - 0% of RATED THERMAL POWER[a]
b. RWL - High Power Setpoint	70 + 0, - 15% of RATED THERMAL POWER[a]	70 + 0, - 15% of RATED THERMAL POWER[a]
2. APRM		
a. Flow Biased Neutron Flux - Upscale		
1) During two recirculation loop operation:		
a. Flow Biased	$\leq$ 0.66W + 58%[b], with a maximum of	$\leq$ 0.66W + 61%[b], with a maximum of
b. High Flow Clamped	$\leq$ 108.0% of RATED THERMAL POWER	$\leq$ 110.0% of RATED THERMAL POWER
2) During single recirculation loop operation:[b][c]		
a. Flow Biased	$\leq$ 0.66W + 36.7%[b][c]	$\leq$ 0.66W + 39.7%[b][c]
b. High Flow Clamped	Not required OPERABLE	Not required OPERABLE
b. Inoperative	NA	NA
c. Downscale	$\geq$ 4% of RATED THERMAL POWER	$\geq$ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	$\leq$ 12% of RATED THERMAL POWER	$\leq$ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq 1 \times 10^5$ cps	$\leq 1.6 \times 10^5$ cps
c. Inoperative	NA	NA
d. Downscale	$\geq$ 0.7 cps[d]	$\geq$ 0.5 cps[d]
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ 108/125 division of full scale	$\leq$ 110/125 division of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ 5/125 division of full scale	$\geq$ 3/125 division of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level - High	$\leq$ 16.6 inches[e] (624' 3.3" elevation)	$\leq$ 17.48 inches[e] (623' 4.17" elevation)
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	$\leq$ 111% of rated flow	$\leq$ 114% of rated flow
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	NA

(a) The actual setpoints are the corresponding values of the turbine first stage pressure for these power levels.
(b) The Average Power Range Monitor flow biased control rod block function varies as a function of recirculation loop drive flow W. During single loop operation W is adjusted to account for the difference in indicated drive flow as described for Specification 2.2.1 in the Bases.
(c) To functionally implement this protective function during entry into single loop operation, APRM gain adjustments may be made in lieu of adjusting the APRM Flow Biased Neutron Flux - Upscale Trip Setpoint and Allowable Value equations for a period not to exceed 72 hours provided that the APRM gains are adjusted to a value at least 21.3% of RATED THERMAL POWER higher than the actual thermal power.
(d) Provided the signal to noise ratio $\geq$ 2.
(e) Level zero is 622' 10.69" elevation; level transmitter readout.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 The reactor coolant system recirculation loop(s) shall be in operation with the total core flow greater than or equal to 45% of rated core flow, or THERMAL POWER less than or equal to the limit specified in Figure 3.4.1.1-1 and either:

a. Two recirculation loops operating, or

b. A single recirculation loop operating with the following limits and conditions:

   1. a) The MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit adjusted for single recirculation loop operation per Specification 2.1.2, and

      b) The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits adjusted for single recirculation loop operation per the CORE OPERATING LIMITS REPORT in accordance with Specification 3.2.1, and

      c) The Average Power Range Monitor (APRM) Scram and Rod Block Trip Setpoint and Allowable Value equations adjusted to those values applicable for single recirculation loop operation** per Specifications 2.2.1 and 3.3.6, and

   2. A volumetric recirculation loop drive flow less than or equal to 48,500 gpm, and

   3. The recirculation flow control system in the Loop Manual (Position Control) mode, and

   4. THERMAL POWER less than or equal to 2500 Megawatts-thermal.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. Upon initial entry into single loop operation, adjustments to the limits and setpoints of Specifications 2.1.2, 2.2.1**, 3.2.1, and 3.3.6** shall be implemented within 8 hours, or declare the associated equipment inoperable (or declare the associated limits to be "not satisfied"), and take the ACTIONS required by the applicable specifications.

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* See Special Test Exception 3.10.4.

** To functionally implement these protective functions during entry into single loop operation, APRM gain adjustments may be made in lieu of adjusting the APRM Scram and Rod Block Flow Biased Setpoints for an interim period of 72 hours.

b. During single loop operation, with the volumetric recirculation loop drive flow greater than the above limit, immediately initiate corrective action to reduce flow to less than or equal to the above limit within 1 hour.

c. During single loop operation, with the recirculation flow control system not in the Loop Manual mode, immediately initiate corrective action to place the recirculation flow control system in the Loop Manual mode within 1 hour.

d. During single loop operation, with THERMAL POWER greater than the above limit, immediately initiate corrective action to reduce THERMAL POWER to less than or equal to the above limit within 1 hour.

e. During single loop operation, with either:

   1. THERMAL POWER $\leq$ 30% of RATED THERMAL POWER and temperature differences exceeding the limits in Surveillance Requirement 4.4.1.1.4, or

   2. recirculation loop jet pump flow in the operating loop $\leq$ 50%# of rated (two loop) core flow and temperature differences exceeding the limits in Surveillance Requirement 4.4.1.1.4,

   suspend THERMAL POWER and recirculation loop flow increases.

f. 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.

A conservative initial value. A lower recirculation loop jet pump flow value may be determined during SLO and submitted for approval, based upon the threshold flow which will sweep the cold water from the vessel bottom head preventing stratification.

LIMITING CONDITION FOR OPERATION (Continued)

g. With one or 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. Determine the APRM and LPRM## noise levels (Surveillance 4.4.1.1.2):

   a) At least once per 8 hours, and

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

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.

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

4.4.1.1.2 Establish a baseline APRM and LPRM## neutron flux noise value within the regions for which monitoring is required (Specification 3.4.1.1, ACTION g) 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.

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.

4.4.1.1.3 Initially, within 1 hour upon entry into single recirculation loop operation and once per 12 hours thereafter, verify that:

a. The volumetric recirculation loop drive flow of the operating loop is less than or equal to the limit stated in Specification 3.4.1.1.b.2, and

b. The recirculation flow control system for the operating loop is in the Loop Manual (Position Control) mode, and

c. THERMAL POWER is less than or equal to the limit stated in Specification 3.4.1.1.b.4.

4.4.1.1.4 With one reactor coolant system recirculation loop not in operation, and either THERMAL POWER less than or equal to 30% of RATED THERMAL POWER or the recirculation loop jet pump flow in the operating loop less than or equal to 50%# of rated (two loop) core flow, verify within 15 minutes prior to an increase in THERMAL POWER or recirculation loop jet pump flow that the following differential temperature requirements are met:

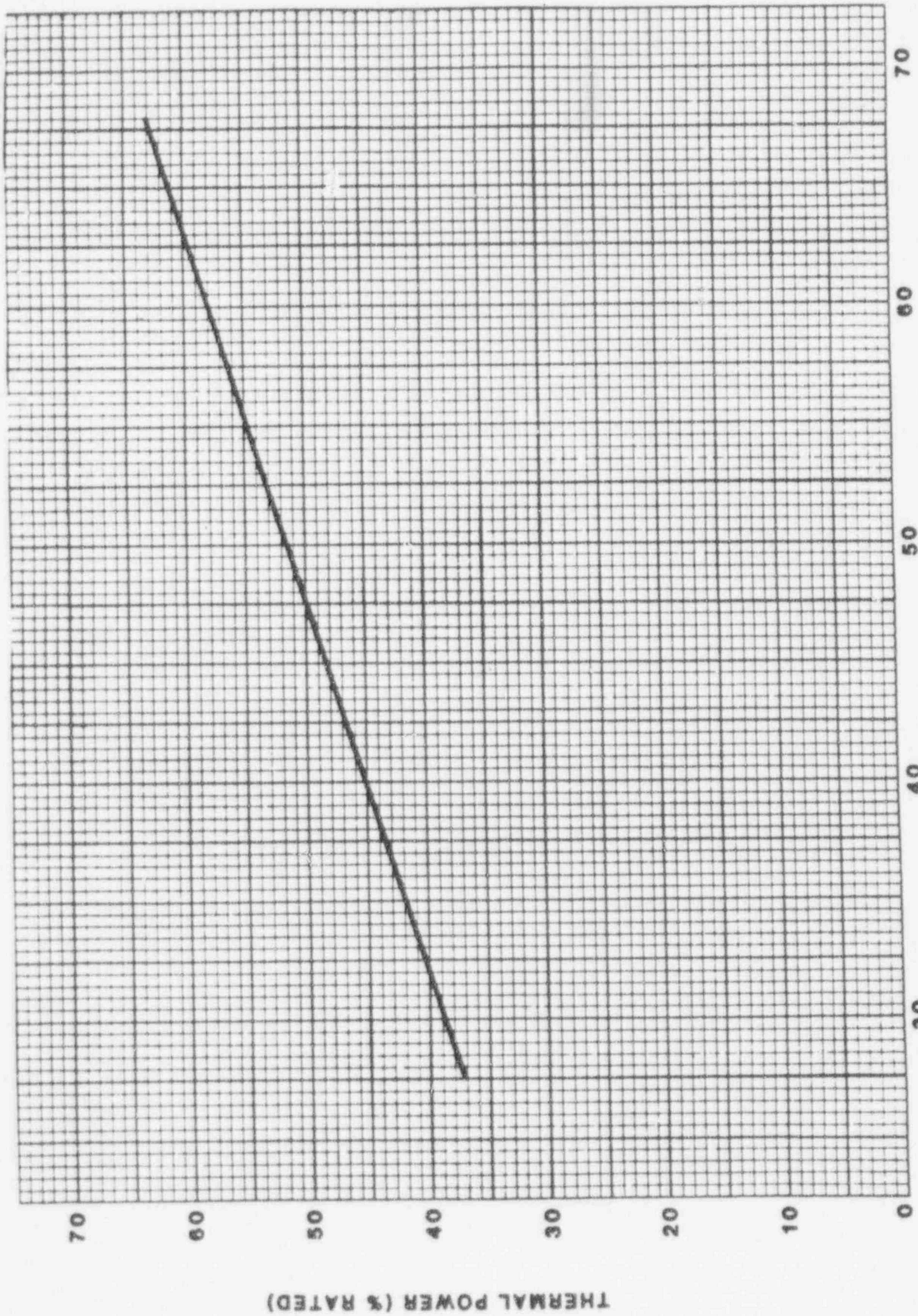
a. $\leq 100^{\circ}F$ between reactor vessel steam space coolant and bottom head drain line coolant, and

b. $\leq 50^{\circ}F$ between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c. $\leq 50^{\circ}F$ between the reactor coolant within the loop not in operation and the operating loop.

The differential temperature requirement of 4.4.1.1.4.a does not apply when the reactor pressure vessel is below 25 psig. The differential temperature requirements of 4.4.1.1.4.b and c do not apply when there is no flow through the loop not in operation due to either one or both the loop suction/discharge valve(s) being closed.

A conservative initial value. A lower recirculation loop jet pump flow value may be determined during SLO and submitted for approval, based upon the threshold flow which will sweep the cold water from the vessel bottom head preventing stratification.



CORE FLOW (% RATED)

THERMAL POWER VERSUS CORE FLOW

FIGURE 3.4.1.1-1

REACTOR COOLANT SYSTEM

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.1.2 Each jet pump in an operating recirculation loop shall be demonstrated OPERABLE at least once per 24 hours when THERMAL POWER is greater than 25% of RATED THERMAL POWER, by determining recirculation loop drive flow, recirculation loop jet pump flow and diffuser-to-lower plenum differential pressure (or flow), for each jet pump and verifying that no two of the following conditions occur:

a. The indicated recirculation loop drive flow differs by more than 10% from the established* flow control valve position-drive flow characteristics.

b. The indicated recirculation loop(s) jet pump flow differs by more than 10% from the established* recirculation loop(s) jet pump flow value derived from recirculation loop drive flow measurements.

c. The indicated jet pump diffuser-to-lower plenum differential pressure (or jet pump flow) of any individual jet pump differs from established* patterns by more than 20% (or 10% for flow).

The provisions of Specification 4.0.4 are not applicable provided that this surveillance is performed within 24 hours after exceeding 25% of RATED THERMAL POWER.

* Data shall be recorded following each refueling outage or upon first entry into single loop operation during an operating cycle, in order to establish the specified relationships for that cycle/mode of operation. Comparisons of the actual data shall commence upon the establishment of the specified relationships for that cycle/mode of operation.

RECIRCULATION LOOP FLOW

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation loop jet pump flow mismatch shall be maintained within:

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

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

APPLICABILITY: OPERATIONAL CONDITIONS 1* AND 2* during two recirculation loop operation.

ACTION:

With recirculation loop jet pump flows different by more than the specified limits, either:

a. Restore the recirculation loop jet pump flows to within the specified limit within 2 hours, or
b. Shutdown one of the recirculation loops and take the ACTION required by Specification 3.4.1.1.

Otherwise, 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 the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation loop jet pump flow mismatch shall be verified to be within the limits at least once per 24 hours. The provisions of Specification 4.0.4 are not applicable provided that this surveillance is performed within 12 hours after starting an idle recirculation loop.

*See Special Test Exception 3.10.4.

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

BASES

3/4.1.1 SHUTDOWN MARGIN

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

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

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

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

3/4.1.2 REACTIVITY ANOMALIES

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

3/4.1.3 CONTROL RODS

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

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than the Safety Limit MCPR during the limiting power transient analyzed in Chapter 15 of the USAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than the Safety Limit MCPR. The occurrence of scram times longer than those specified should be viewed as an indication of a systematic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

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

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be 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.

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

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature (PCT) following the postulated design basis Loss-of-Coolant Accident (LOCA) will not exceed the limits specified in 10 CFR 50.46 and that the fuel design analysis limits specified in GESTAR-II (Reference 1) will not be exceeded.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The MAPLHGR limits specified in the CORE OPERATING LIMITS REPORT are multiplied by the smaller of either the flow dependent MAPLHGR factor ($MAPFAC_f$) or the power dependent MAPLHGR factor ($MAPFAC_p$) corresponding to existing core flow and power state to assure the adherence to fuel mechanical design bases during the most limiting transient. A variety of PNPP specific Feedwater Controller Failure and Load Rejection with Bypass Failure transient events (with and without feedwater temperature reduction) together with a large data base of transient event results for other operating plants were used by General Electric to establish the $MAPFAC_p$ and $MAPFAC_f$ limits, with suitable conservatism for operation in the Maximum Extended Operating Domain with and without partial feedwater heating (Reference 2). $MAPFAC_f$'s are determined using the three-dimensional BWR simulator code to analyze slow flow runout transients. $MAPFAC_p$'s are generated using the same data base as the $MCPR_p$ to protect the core from plant transients other than core flow increases.

For single recirculation loop operation the MAPLHGR limits contained in the CORE OPERATING LIMITS REPORT are multiplied by a single loop operation MAPLHGR reduction factor determined each cycle as part of the reload safety analyses. The single loop operation MAPLHGR reduction factor is derived from the LOCA analyses initiated from single loop operational conditions to account for the earlier boiling transition at the limiting fuel assembly node compared to the standard (two loop) LOCA evaluations.

The Technical Specification MAPLHGR value is the most limiting composite of the fuel mechanical design analysis MAPLHGR and the ECCS MAPLHGR.

Fuel Mechanical Design Analysis: NRC approved methods (specified in Reference 1) are used to demonstrate that all fuel rods in a lattice, operating at the bounding power history, meet the fuel design limits specified in Reference 1. This bounding power history is used as the basis for the fuel design analysis MAPLHGR value.

LOCA Analysis: A LOCA analysis is performed in accordance with 10 CFR Part 50 Appendix K to demonstrate that the MAPLHGR values comply with the ECCS limits specified in 10 CFR 50.46. The analysis is performed for the most limiting break size, break location, and single failure combination for the plant.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

Only the most limiting MAPLHGR values are shown in the CORE OPERATING LIMITS REPORT figures for multiple lattice fuel. When hand calculations are required, these CORE OPERATING LIMITS REPORT MAPLHGR figure values for that fuel type are used for all lattices in that bundle.

For some GE fuel bundle designs MAPLHGR depends only on bundle type and burnup. Other GE fuel bundles have MAPLHGRs that vary axially depending upon the specific combination of enriched uranium and gadolinia that comprises a fuel bundle cross section at a particular axial node. Each particular combination of enriched uranium and gadolinia, for these fuel bundle types, is called a lattice type by GE. These particular fuel bundle types have MAPLHGRs that vary by lattice type (axially) as well as with fuel burnup.

Approved MAPLHGR values (limiting values of APLHGR) as a function of fuel and lattice types, and as a function of the average planar exposure are provided in the CORE OPERATING LIMITS REPORT.

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3/4.2.2 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.2 are derived from the established fuel cladding integrity Safety Limit MCPR and an analysis of the limiting operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated are documented in the USAR and Reference 1. The limiting transient yields the largest delta CPR. When added to the Safety Limit MCPR, the required operating limit MCPR of Specification 3.2.2 is obtained. The Maximum Extended Operating Domain (MEOD) power-flow map of Figure B 3/4 2.2-1 portrays the allowable operating region for two recirculation loop operation. A similar power-flow map for single recirculation loop operation is described in USAR Appendix 15F. The analytical basis for generation of the MCPR operating limits is described below.

The evaluation of a given transient begins with the system initial parameters shown in USAR Chapter 15 and/or Reference 1, that are input to a GE-core dynamic behavior transient computer program. The codes used to evaluate these events are described in Reference 1.

The purpose of the $MCPR_f$ and $MCPR_p$ is to define operating limits at other than rated core flow and power conditions. At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_f$ and $MCPR_p$ at the existing core flow and power state. The $MCPR_f$s are established to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

A variety of PNPP specific Feedwater Controller Failure and Load Rejection with Bypass Failure transient events (with and without feedwater temperature reduction) together with a large data base of transient event results for other operating plants were used by General Electric to establish the $MCPR_f$ and $MCPR_p$ limits, with suitable conservatism for operation in the MEOD with and without partial feedwater heating (Reference 2). For single recirculation loop operation the MCPR Safety Limit is increased by 0.01 as described in Section 2.0 of the BASES. No increase in the rated Operating Limit MCPR and no changes in the flow and power dependent MCPR limits are required for single recirculation loop operation because the limiting operational transients analyzed indicated that there is more than enough MCPR margin to compensate for the increase in the MCPR Safety Limit (Reference 3).

The $MCPR_p$ figure contained in the CORE OPERATING LIMITS REPORT also reflects the required MCPR values resulting from the analysis performed to justify operation with the feedwater temperature ranging from 420°F to 320°F at 100% RATED THERMAL POWER steady state conditions, and also beyond the end of cycle with the feedwater temperature ranging from 420°F and 250°F.

The $MCPR_f$s were calculated such that for the maximum core flow rate and the corresponding THERMAL POWER along a conservative steep generic power flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along this conservative steep power flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined as $MCPR_f$.

MINIMUM CRITICAL POWER RATIO (Continued)

The $MCPR_p$s are established to protect the core from plant transients other than core flow increases, including the localized event such as rod withdrawal error. The $MCPR_p$s were calculated based upon the most limiting transient at the given core power level. For core power less than or equal to 40% of RATED THERMAL POWER, where the EOC-RPT and the reactor scrams on turbine stop valve closure and turbine control valve fast closure are bypassed, separate sets of $MRPR_p$ limits are provided for high and low core flows to account for the significant sensitivity to initial core flows. For core power above 40% of RATED THERMAL POWER, bounding power dependent MCPR limits were developed.

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

3/4.2.3 LINEAR HEAT GENERATION RATE

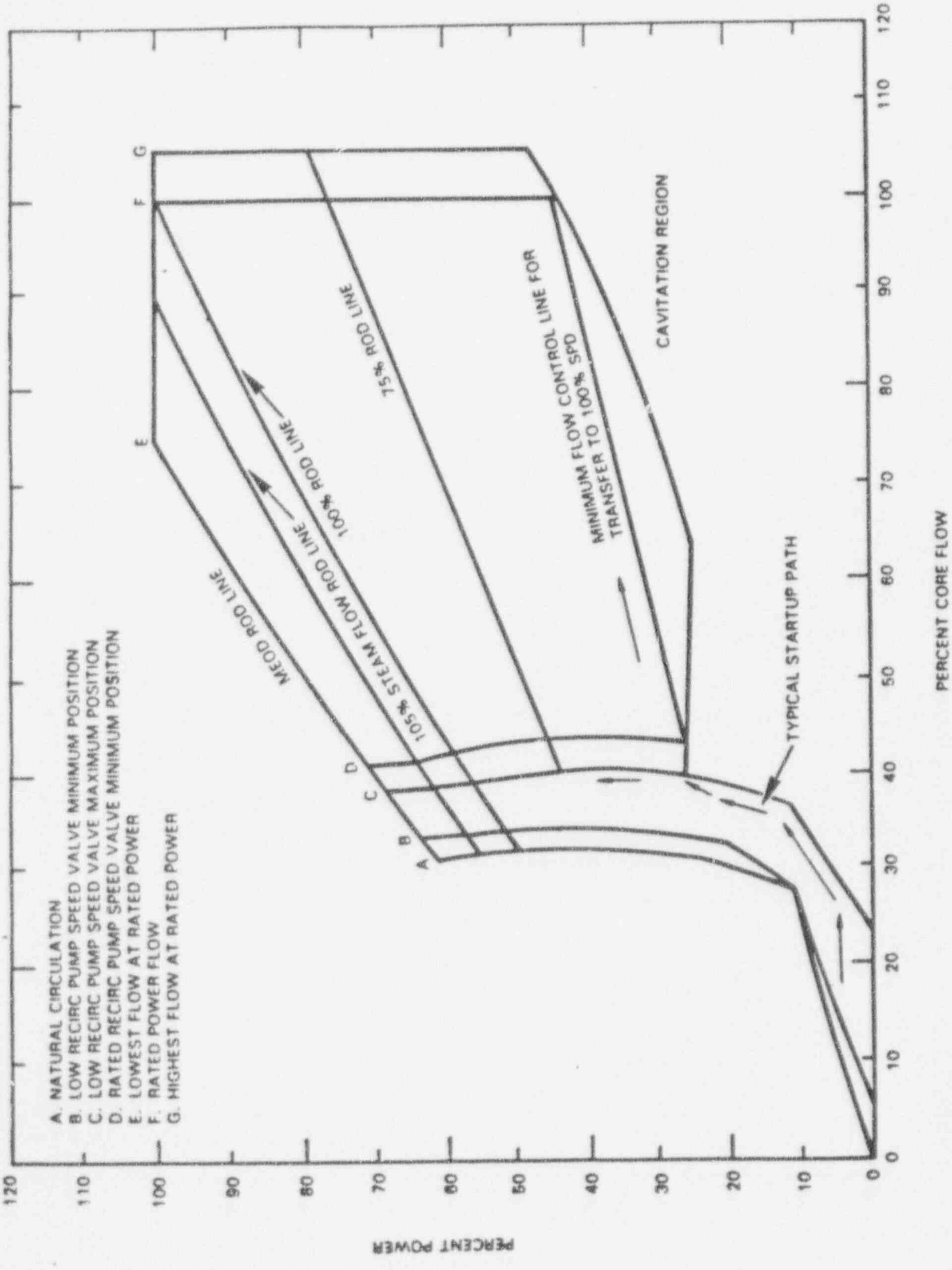
This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

References:

1. GESTAR II, General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A, (latest approved revision).

2. T. C. Lee, "Technical Specification Operating Limits with the Elimination of the APRM Trip Setdown Requirements - Perry Nuclear Power Plants," April 1985 (NEDM-30963).

3. USAR Appendix 15F, PNPP Single Loop Operation Analysis.



POWER-FLOW OPERATING MAP
BASES FIGURE B 3/4 2.2-1

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December 1979, and Section 15.8 of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is an essential safety supplement to the Reactor Protection System. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

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

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 40% of RATED THERMAL POWER are annunciated in the control room.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 140 ms. Included in this time are: the time from initial valve movement to reaching the trip setpoint, the response time of the sensor, the response time of the system logic, and the time allotted for breaker arc suppression.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

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

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses. Adjustments for single recirculation loop operation are made as discussed in the BASES for the APRM setpoints, Section 2.2.1 Item 2.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with 10 CFR Part 50, Appendix A, General Design Criteria 19, 41, 60, 61, 63 and 64.

3/4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit. This instrumentation is consistent with the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes", April 1974.

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

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

3/4.4.1 RECIRCULATION SYSTEM (Continued)

pump operability following a change from two to single loop operation or vice versa is considered acceptable for the duration needed to establish the new specified relationships based on having met the relationships for the former mode of operation.

During two recirculation loop operation, recirculation loop jet pump flow mismatch limits are specified to ensure compliance with the two loop ECCS LOCA analysis design criteria. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. In the case where the flow mismatch limits cannot be maintained during two recirculation loop operation, continued operation is permitted in the single recirculation loop mode. A 4.0.4 exception is provided to permit the startup of an idle recirculation loop when in single loop operation (to return to two loop operation).

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 recirculation 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. Sudden equalization of a temperature difference greater than 100°F between the reactor vessel bottom head coolant and the coolant in the reactor vessel by increasing core flow rate could cause undue stress in the reactor vessel bottom head.

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.

3/4.4.1 RECIRCULATION SYSTEM (Continued)

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 the 80% rodline in accordance with SIL-380 (Reference 1) and Supplement 1 to NRC Bulletin 88-07 (Reference 4). A generic 80% rodline is represented in Figure 3.4.1.1-1 (the actual 80% rodline varies slightly cycle to cycle with changes in fuel type, core hydraulic resistance etc.).

Plant specific calculations can be performed to determine an applicable region for monitoring neutron flux noise levels. In this case the degree of conservatism can be reduced since plant to plant variability would be eliminated. In this case, adequate margin will be assured by monitoring the region which has a decay ratio greater than or equal to 0.8.

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor core coolant recirculation loop out of service has been evaluated (Reference 3) and found to remain within design limits and safety margins provided certain limits and setpoints are modified. Single loop operation is permitted at power levels up to 2500 Megawatts-thermal (MWt) (slightly less than 70% of RATED THERMAL POWER), if the MCPR Fuel Cladding Integrity Safety Limit is increased as noted by Specification 2.1.2, the APRM flow biased scram and control rod block setpoints (or APRM gains) are adjusted as noted in Tables 2.2.1-1, 4.3.1.1-1 and 3.3.6-2, respectively, and the MAPLHGR limits are decreased in accordance with the value specified in the CORE OPERATING LIMITS REPORT. A time period of 8 hours is allowed to make these adjustments following establishment of single loop operation since the need for single loop operation often cannot be anticipated.

Additionally, a limitation on the volumetric drive flow of the operating recirculation loop (during SLO) is imposed to exclude the possibility of excessive core internals vibration. Recirculation loop drive flow is the discharge flow from the recirculation pumps. Recirculation loop jet pump flow is the summation of all the individual jet pump flows for a particular recirculation loop. Total core flow for two recirculation loop operation is the sum of the recirculation loop jet pump flows for both loops. Total core flow for single recirculation loop operation is the recirculation loop jet pump flow for the operating loop minus the reverse flow through the non-operating loop. Rated core flow as used in the Recirculation System Specifications corresponds to the rated (100%) core flow value for two recirculation loop operation (104 Mlb/hr).

To prevent potential control system oscillations from occurring in the recirculation flow control system (and to eliminate the need for flow control system failure analyses), the operating mode of the recirculation flow control system is restricted to the Loop Manual (Position Control) mode for single loop operation.

The surveillance on differential temperatures below 30% THERMAL POWER or 50% of rated (two loop) core flow is to prevent undue thermal stress on vessel nozzles, recirculation pump, and vessel bottom head during a power or flow increase during extended operation in the single recirculation loop mode, similar to the restrictions described below for idle recirculation loop startup. The current 50% of rated core flow value has been determined by analysis and operating experience at other GE BWRs to provide margin to the onset of stratification. To provide operational flexibility a threshold core flow (below which thermal stratification might occur) may be determined during testing in SLO and a new limit established with appropriate margin to this value.

3/4.4.1 RECIRCULATION SYSTEM (Continued)

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation. During single loop operation the jet pump operability surveillance is only performed for the jet pumps on the operating recirculation loop, as the loads on the jet pumps on the inactive loop are expected to be very low due to the low flow in the reverse direction through them. This has been demonstrated through operating experience at other GE BWRs. The jet pumps in the non-operating recirculation loop during SLO are considered operable based on this low expected loading, acceptable surveillance results obtained during two recirculation loop operation prior to entering SLO, or by visual inspection of the jet pumps during refueling/shutdowns. Upon startup of an idle recirculation loop when THERMAL POWER is greater than 25% of RATED THERMAL POWER, the specified jet pump surveillances are required to be performed for the previously idle loop within the 24 hour surveillance interval specified in Surveillance Requirement 4.4.1.2.

Significant degradation is indicated if more than one of the three specified surveillance requirements performed confirms unacceptable deviations from established patterns or relationships. The surveillances, including the associated acceptance criteria, are in accordance with General Electric Service Information Letter No. 330, the recommendations of which are considered acceptable for verifying jet pump operability according to NUREG/CR-3052, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure." Performance of the specified surveillance requirements, however, is not required when thermal power is less than 25% RATED THERMAL POWER because flow oscillations and jet pump noise precludes collection of repeatable data during low flow conditions, approaching the threshold response of the associated flow instrumentation.

Jet pump operability is considered acceptable prior to startup of the plant due to acceptable results obtained during the past cycle, or by visual inspection of the jet pumps. Initial data collection will be performed to establish the specified relationships during post-refuel performance testing for a new operating cycle, or upon first entry into single loop operation during an operating cycle. This satisfies the Surveillance Requirements of Specification 3.4.1.2, since taking the data establishes the relationships for that cycle or mode of operation, and there are no valid prior relationships to compare against. Jet

RECIRCULATION SYSTEM (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. 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.

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.

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) USAR Appendix 15F, PNPP Single Loop Operation Analysis.

(4) NRC Bulletin No. 88-07 Supplement 1, "Power Oscillations in Boiling Water Reactors (BWRs)."

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves (SRV) operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. A total of 13 OPERABLE safety-relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient. Any combination of 6 SRVs operating in the relief mode and 7 SRVs operating in the safety mode is acceptable.