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

June 30, 1989

Docket No.: 50-352

Mr. George A. Hunger, Jr.
Director-Licensing
Philadelphia Electric Company
Correspondence Control Desk
P. O. Box 7520
Philadelphia, Pennsylvania 19101

Dear Mr. Hunger:

SUBJECT: SINGLE LOOP OPERATION (TAC NOS. 71162, 72910, 72786, 72787)

RE: LIMERICK GENERATING STATION, UNIT 1

The Commission has issued the enclosed Amendment No. 30 to Facility Operating License No. NPF-39 for the Limerick Generating Station, Unit 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated November 4, 1988 as supplemented March 29, 1989.

This amendment revises the TSs to permit operation of the reactor with one of two reactor recirculation loops in service under certain specified conditions.

We have also reviewed the analysis you submitted with your letter of March 29, 1989 for operation of Limerick, Unit 2 with a single operable recirculation loop. As discussed in the enclosed Safety Evaluation, we concluded that appropriate documentation was submitted for both Limerick, Units 1 and 2 to satisfy the staff positions and requirements with respect to both single loop operation and thermal hydraulic stability. Extended single loop operation is approved for Limerick, Units 1 and 2: 1) with the monitoring for thermal hydraulic stability described in your submittal, 2) compliance to the monitoring requirements described in your responses of September 5, 1988 and March 7, 1989 to NRC Bulletin No. 88-07 and Supplement 1, respectively, 3) operation of Unit 1 with the TSs approved by this amendment and 4) operation of Unit 2 with TSs similar to those approved for Unit 1.

Your submittals of September 5, 1988 and March 7, 1989 acceptably respond to Bulletin 88-07 issued June 15, 1988 and to Supplement 1 to this Bulletin issued December 30, 1988. For purposes of our Safety Issues Management System (SIMS), the requirement was to submit an acceptable response. We consider this action to be implemented as of the date of this letter.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Richard J. Clark

Richard J. Clark, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 30 to License No. NPF-39
2. Safety Evaluation

cc w/enclosures:
See next page

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by
Richard J. Clark

Richard J. Clark, Project Manager
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Division of Reactor Projects I/II
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DISTRIBUTION w/enclosures:

Docket File	ACRS (10)	JCalvo
NRC PDR	GPA/PA	BGrimes
Local PDR	OGC	Brent Clayton
PDI-2 Rdg File	RDiggs, ARM/LFMB	EWenzinger
SVarga		CSchulten
OTSB		
BBoger	TMeek(4)	
WButler	EJordan	
RClark	DHagan	
RMartin	Wanda Jones	
MO'Brien	HRichings	

DF01
1/1

PM put his own set of TSs in pkg. LA did not review.

[LIM HUNGER]

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
for MO'Brien	RClark	MYoung	WButler
6/29/89	05/22/89	5/31/89	6/30/89

No note/comments expected by issuance

Mr. George A. Hunger, Jr.
Philadelphia Electric Company

Limerick Generating Station
Units 1 & 2

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

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 30
License No. NPF-39

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

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated November 4, 1988 as supplemented March 29, 1989, 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-39 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 30 , are hereby incorporated into this license. Philadelphia Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective within 30 days of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 30, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 30

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

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
2-2	2-2*
2-3	2-3*
2-4	2-4
B 2-1	B 2-1
B 2-2	B 2-2*
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2*
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8*
3/4 2-9	3/4 2-9*
3/4 2-10	3/4 2-10
3/4 2-10a	3/4 2-10a
3/4 2-10b	3/4 2-10b
3/4 2-10c	3/4 2-10c
-	-
3/4 3-59	3/4 3-59*
3/4 3-60	3/4 3-60
3/4 3-60a	3/4 3-60a
-	-
3/4 4-1	3/4 4-1
3/4 4-2	3/4 4-1a
-	3/4 4-2
-	-
3/4 4-3	3/4 4-3
3/4 4-4	3/4 4-4
-	3/4 4-4a
-	-

ATTACHMENT TO LICENSE AMENDMENT NO. 30

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

Remove	Insert
3/4 4-5	3/4 4-5
3/4 4-6	3/4 4-6*
B 3/4 1-1	B 3/4 1-1*
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 2-3	B 3/4 2-3*
B 3/4 2-4	B 3/4 2-4
B 3/4 4-1	B 3/4 4-1
B 3/4 4-2	B 3/4 4-2

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 for two recirculation loop operation and shall not be less than 1.08 for 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 1.07 for two recirculation loop operation or less than 1.08 for single recirculation loop operation 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, after depressurizing the reactor vessel, if required. 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.

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. Neutron Flux-Upscale		
1) During two recirculation loop operation:		
a) Flow Biased	≤ 0.58 W+ 59%, with a maximum of ≤ 116.5% of RATED THERMAL POWER	≤ 0.58 W+ 62%, with a maximum of ≤ 118.5% of RATED THERMAL POWER
b) High Flow Clamped		
2) During single recirculation loop operation:		
a) Flow Biased	≤ 0.58 W+ 54%,	≤ 0.58 W+ 57%,
b) High Flow Clamped	Not Required OPERABLE	Not Required OPERABLE
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 4% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	≤ 1037 psig	≤ 1057 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11.0 inches above instrument zero
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. Level Transmitter	≤ 260' 9 5/8" elevation**	≤ 261' 5 5/8" elevation
b. Float Switch	≤ 260' 9 5/8" elevation**	≤ 261' 5 5/8" elevation
9. Turbine Stop Valve - Closure	≤ 5% closed	≤ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 500 psig	≥ 465 psig
11. Reactor Mode Switch Shutdown Position	N.A.	N.A.
12. Manual Scram	N.A.	N.A.

*See Bases Figure B 3/4.3-1.
**Equivalent to 25.45 gallons/scram discharge volume.

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 1.07 for two recirculation loop operation and 1.08 for single recirculation loop operation. MCPR greater than 1.07 for two recirculation loop operation and 1.08 for single recirculation loop 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 (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 lb/h, 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 lb/h. 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 a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. Calculation of the Safety Limit MCPR is described in Reference 1.

Reference:

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

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 axial location and AVERAGE PLANAR EXPOSURE shall be within limits based on applicable APLHGR limit values which have been approved for the respective fuel and lattice types for two recirculation loop operation. When hand calculations are required, the APLHGR for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) as shown in the applicable figures for BP/P8X8R and GE8X8EB fuel types. The limits shall be reduced to a value of 0.89 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 limiting value, 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

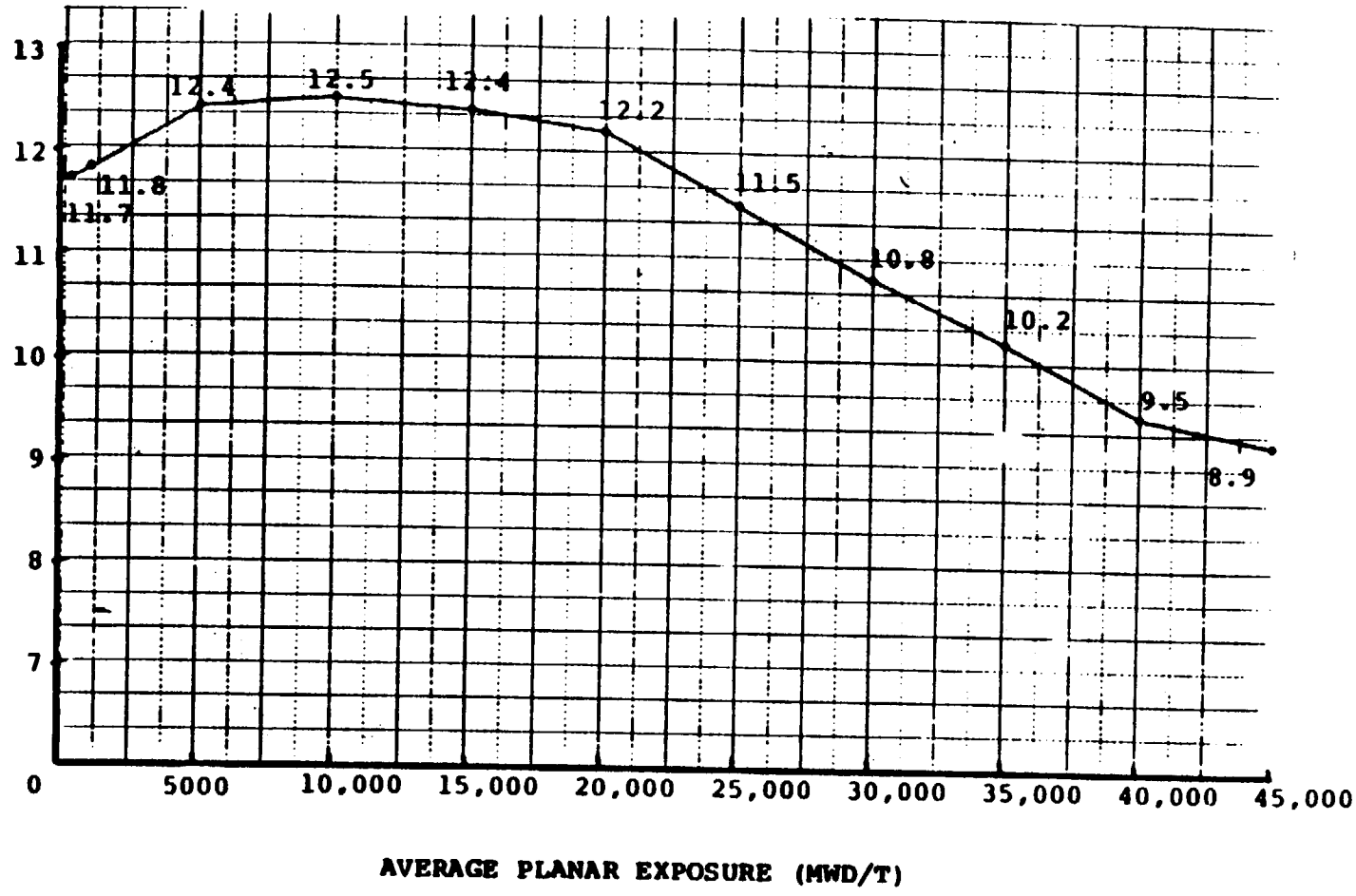
4.2.1 All APLHGRs shall be verified to be equal to or less than the limiting value

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.



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPES P8CIB278

FIGURE 3.2.1-1

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

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

	TRIP SETPOINT	ALLOWABLE VALUE
During two recirculation loop operation	$S \leq (0.58W + 59\%)T$ $S_{RB} \leq (0.58W + 50\%)T$	$S \leq (0.58W + 62\%)T$ $S_{RB} \leq (0.58W + 53\%)T$
During single recirculation loop operation	$S \leq (0.58W + 54\%)T$ $S_{RB} \leq (0.58W + 45\%)T$	$S \leq (0.58W + 57\%)T$ $S_{RB} \leq (0.58W + 48\%)T$

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

W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 100 million lbs/hr,

T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the CORE 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-upscale scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint values* 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 shall be determined, the value of T calculated, and the most recent actual APRM flow biased neutron flux-upscale scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

a. At least once per 24 hours,

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

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

d. The provisions of Specification 4.0.4 are not applicable.

---

*With MFLPD greater than the FRTP during power ascension up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that the 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.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit shown in Figure 3.2.3-1a (BP/P8X8R fuel), Figure 3.2.3-1b (BP/P8X8R fuel), Figure 3.2.3-1c (GE 8X8EB fuel) and Figure 3.2.3-1d (GE 8X8EB fuel) times the K_f shown in Figure 3.2.3-2, provided that the end-of-cycle recirculation pump trip (EOC-RPT) system is OPERABLE per Specification 3.3.4.2, with:

$$\tau = \frac{(\tau_{ave} - \tau_B)}{\tau_A - \tau_B}$$

where:

τ_A = 0.86 seconds, control rod average scram insertion time limit to notch 39 per Specification 3.1.3.3,

$$\tau_B = 0.672 + 1.65\left[\frac{N_1}{\sum_{i=1}^{n} N_i}\right]^{\frac{1}{2}}(0.016),$$

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i},$$

n = number of surveillance tests performed to date in cycle,

N_i = number of active control rods measured in the ith surveillance test,

τ_i = average scram time to notch 39 of all rods measured in the ith surveillance test, and

N_1 = total number of active rods measured in Specification 4.1.3.2.a.

APPLICABILITY:

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

LIMITING CONDITION FOR OPERATION (Continued)

ACTION

a. With the end-of-cycle recirculation pump trip system inoperable per Specification 3.3.4.2, operation may continue provided that, within 1 hour, MCPR is determined to be greater than or equal to the MCPR limit as a function of the average scram time shown in Figure 3.2.3-1a (BP/P8X8R fuel), Figure 3.2.3-1b (BP/P8X8R fuel), Figure 3.2.3-1c (GE8X8EB fuel) and Figure 3.2.3-1d (GE8X8EB fuel), EOC-RPT inoperable curve, times the k_f shown in Figure 3.2.3-2.

b. With MCPR less than the applicable MCPR limit shown in Figures 3.2.3-1a, 3.2.3-1b and 3.2.3-2, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR, with:

a. τ = 1.0 prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.1.3.2, or

b. τ as defined in Specification 3.2.3 used to determine the limit within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2,

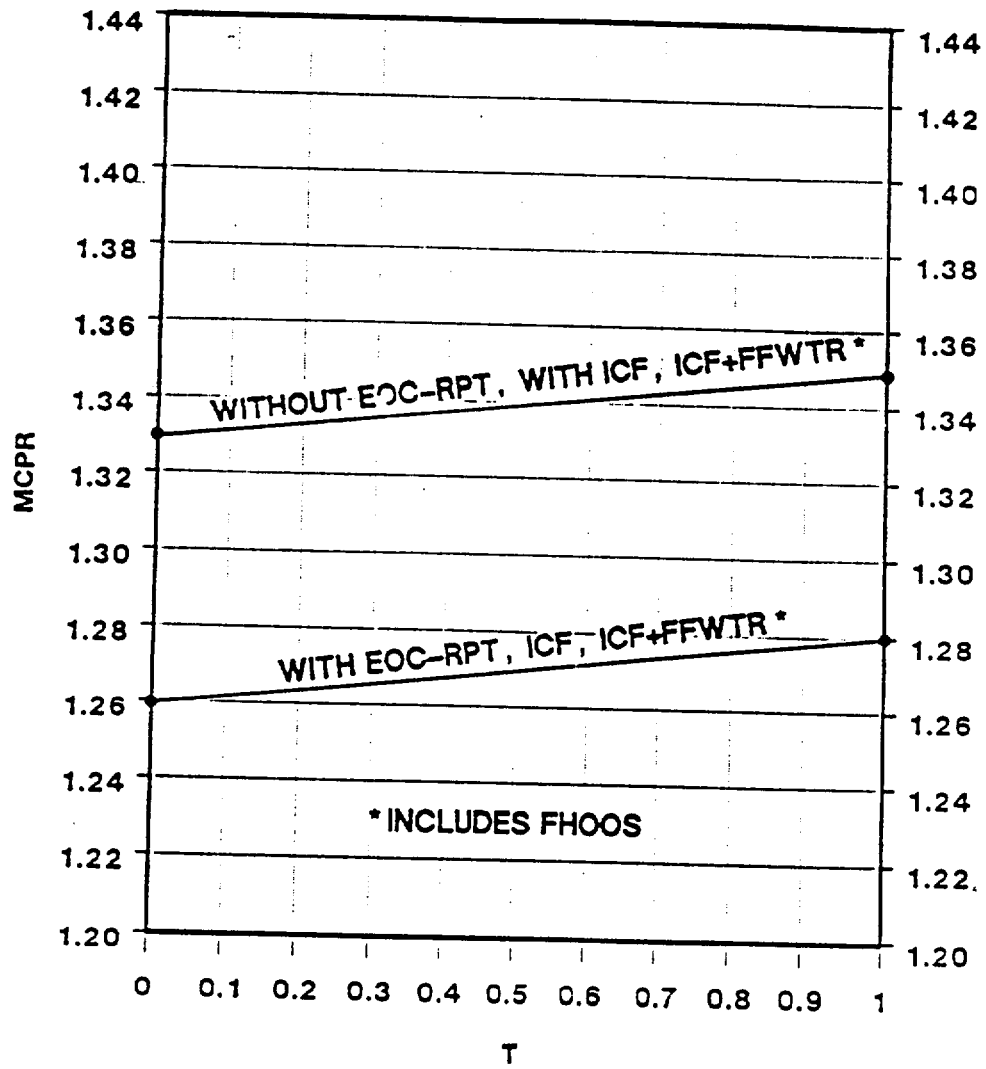
shall be determined to be equal to or greater than the applicable MCPR limit determined from Figures 3.2.3-1a, 3.2.3-1b and 3.2.3-2.

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 MCPR.

d. The provisions of Specification 4.0.4 are not applicable.



Note: These limits apply to Both Two Recirculation Loop and Single Recirculation Loop Operation.

DEFINITIONS:

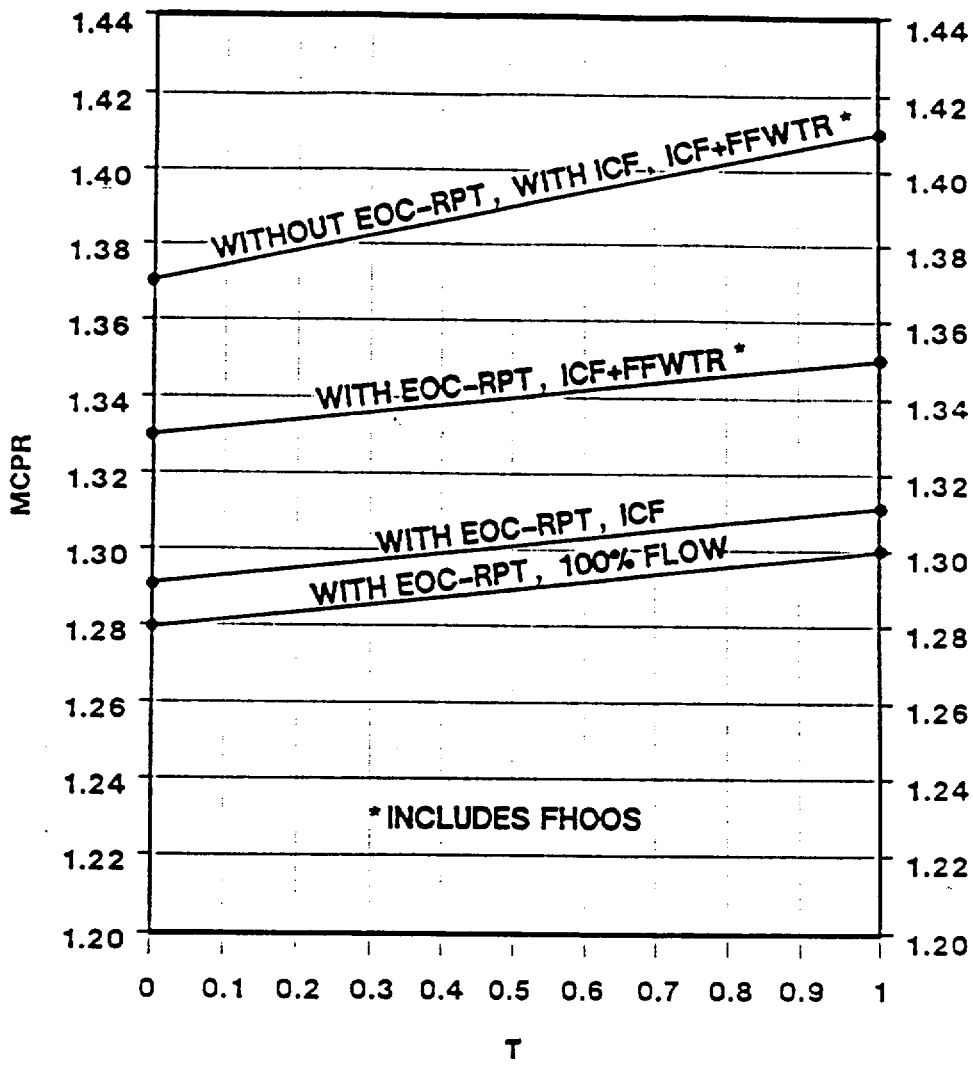
ICF - INCREASED CORE FLOW (UP TO 105% RATED)

FHOOS - FEEDWATER HEATING OUT OF SERVICE THROUGHOUT CYCLE (UP TO 60°F TEMP. REDUCTION; ACHIEVED BY REMOVAL OF FEEDWATER HEATER(s))

FFWTR - FINAL FEEDWATER TEMPERATURE REDUCTION AT END OF CYCLE (UP TO 60°F TEMP. REDUCTION: ACHIEVED BY REMOVAL OF ALL 6TH STAGE HEATERS)

MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS τ (P8x8R/BP8x8R FUEL)
(BOC TO EOC - 2000 MWD/ST)

FIGURE 3.2.3-1a



Note: These limits apply to Both Two Recirculation Loop and Single Recirculation Loop Operation.

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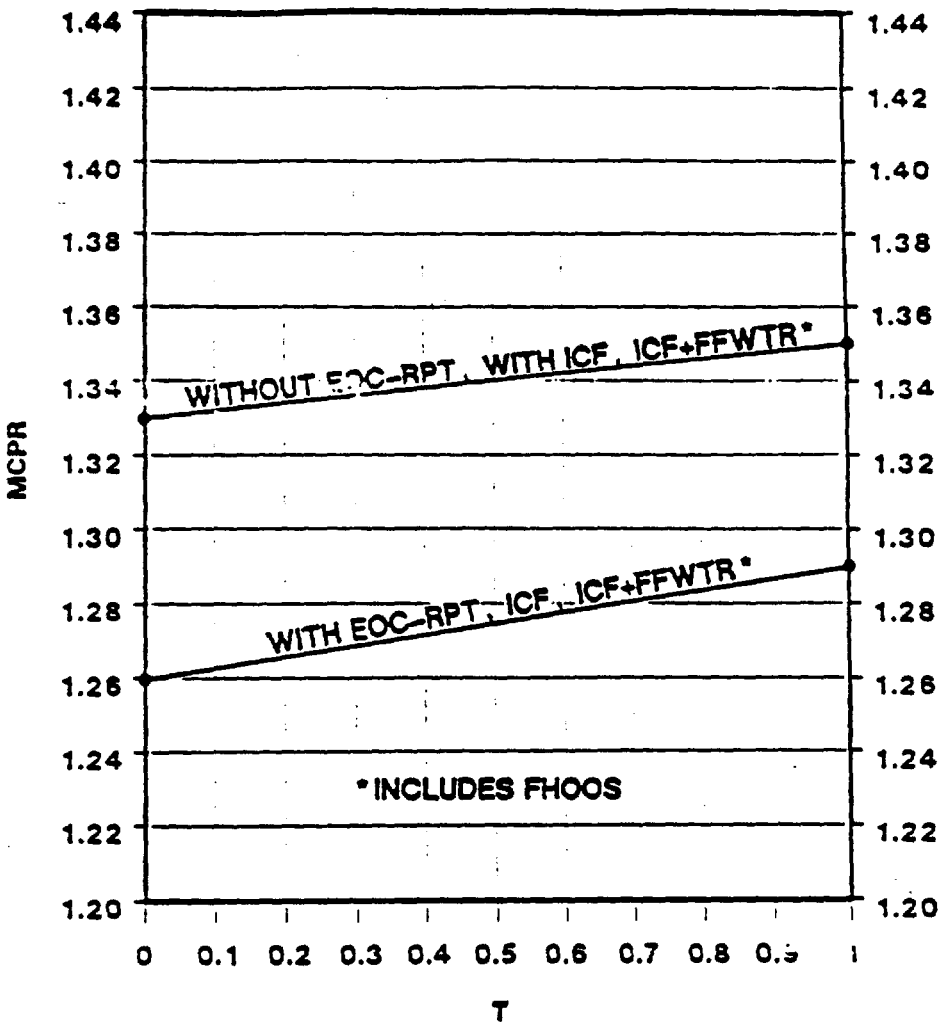
ICF - INCREASED CORE FLOW (UP TO 105% RATED)

FHOOS - FEEDWATER HEATING OUT OF SERVICE THROUGHOUT CYCLE (UP TO 60°F TEMP. REDUCTION; ACHIEVED BY REMOVAL OF FEEDWATER HEATER(S))

FFWTR - FINAL FEEDWATER TEMPERATURE REDUCTION AT END-OF-CYCLE (UP TO 60°F TEMP. REDUCTION: ACHIEVED BY REMOVAL OF ALL 6TH STAGE HEATERS)

MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS τ (P8x8R/BP8x8R FUEL)
(EOC - 2000 MWD/ST TO EOC)

FIGURE 3.2.3-1b



Note: These Limits Apply To Both Two Recirculation Loop and Single Recirculation Loop Operation.

DEFINITIONS:

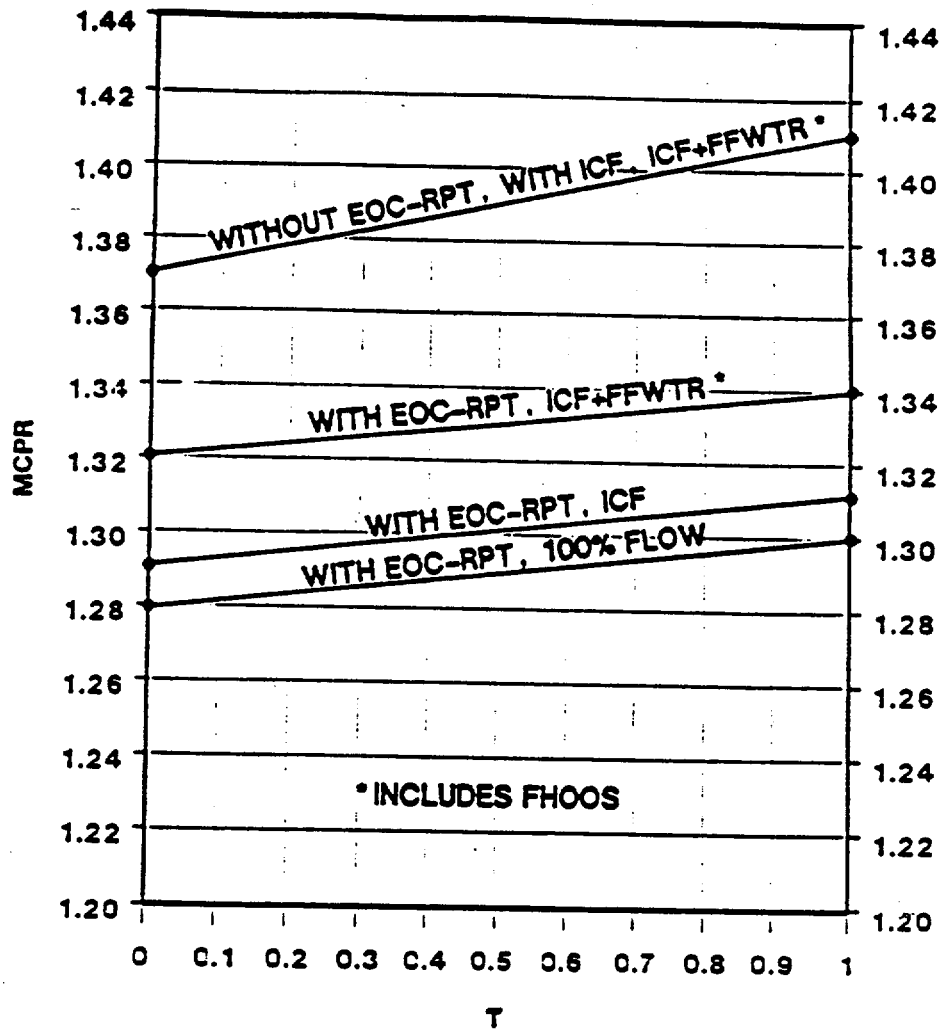
ICF - INCREASED CORE FLOW (UP TO 105% RATED)

FHOOS - FEEDWATER HEATING OUT OF SERVICE THROUGHOUT CYCLE (UP TO 60°F TEMP. REDUCTION; ACHIEVED BY REMOVAL OF FEEDWATER HEATER(S))

FFWTR - FINAL FEEDWATER TEMPERATURE REDUCTION AT END-OF-CYCLE (UP TO 60°F TEMP. REDUCTION; ACHIEVED BY REMOVAL OF ALL 6th STAGE HEATERS)

MINIMUM CRITICAL POWER RATIO (MPCR) VERSUS τ (GE8x8EB FUEL)
(BOC TO EOC - 2000 MWD/ST)

FIGURE 3.2.3-1c



Note: These Limits Apply To Both Two Recirculation Loop and Single Recirculation Loop Operation.

DEFINITIONS:

ICF - INCREASED CORE FLOW (UP TO 105% RATED)

FHOOS - FEEDWATER HEATING OUT OF SERVICE THROUGHOUT CYCLE (UP TO 60°F TEMP. REDUCTION; ACHIEVED BY REMOVAL OF FEEDWATER HEATER(S))

FFWTR - FINAL FEEDWATER TEMPERATURE REDUCTION AT END-OF-CYCLE (UP TO 60°F TEMP. REDUCTION; ACHIEVED BY REMOVAL OF ALL 6th STAGE HEATERS)

MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS τ (GE8x8EB FUEL)
(EOC - 2000 MWD/ST TO EOC)

FIGURE 3.2.3-1d

TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION STATEMENTS

ACTION 60 - Declare the RBM inoperable and take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE channels one 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 THERMAL POWER ≥ 30% of RATED THERMAL POWER.

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

*** These channels are not required when sixteen or fewer fuel assemblies, adjacent to the SRMs, are in the core.

(a) The RBM shall be automatically bypassed when a peripheral control rod is selected or the reference APRM channel indicates less than 30% of RATED THERMAL POWER.

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

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

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

(e) 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 BLOCK MONITOR		
a. Upscale		
1) During two recirculation loop operation		
a. Flow Biased*	≤ 0.66 W + 40%, with a maximum of,	≤ 0.66 W + 43%, with a maximum of,
b. High Flow Clamped	≤ 106%	≤ 109%
2) During single recirculation loop operation		
a. Flow Biased	≤ 0.66 W + 34%, with a maximum of,	0.66 W + 37%, with a maximum of,
b. High Flow Clamped	≤ 106%	≤ 109%
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - Upscale		
1) During two recirculation loop operation	≤ 0.58 W + 50%*	≤ 0.58 W + 53%*
2) During single recirculation loop	≤ 0.58 W + 45%*	≤ 0.58 W + 48%*
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 4% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 1×10^5 cps	≤ 1.6×10^5 cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 3 cps**	≥ 1.8 cps**
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High		
a. Float Switch	≤ 257' 5 9/16" elevation***	≤ 257' 7 9/16" elevation

TABLE 3.3.6-2 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 111% of rated flow	≤ 114% of rated flow
b. Inoperative	N.A.	N.A.
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation
7. REACTOR MODE SWITCH SHUTDOWN POSITION	N.A.	N.A.

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

**May be reduced to 0.7 cps provided the signal-to-noise ratio is ≥ 2.

***Equivalent to 13 gallons/scram discharge volume.

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 with:

a. Total core flow greater than or equal to 45% of rated core flow, or

b. THERMAL POWER within the unrestricted zone of Figure 3.4.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

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

1. Within 4 hours:

a. Place the recirculation flow control system in the Local Manual mode, and

b. Reduce THERMAL POWER to ≤ 70% of RATED THERMAL POWER, and,

c. Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit to a value of 0.89 times the two recirculation loop operation limit per Specification 3.2.1, and,

d. Limit the sped of the operating recirculation pump to less than or equal to 90% of rated pump speed, and

e. Verify that the differential temperature requirements of Surveillance Requirement 4.4.1.1.5 are met if THERMAL POWER is ≤ 30% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is ≤ 50% of rated loop flow, or suspend the THERMAL POWER or recirculation loop flow increase.

*See Special Test Exception 3.10.4.

ACTION: (Continued)

2. Within 6 hours:

   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, or declare the associated channel(s) inoperable and take the actions required by the referenced specifications, and,

3. The provisions of Specification 3.0.4 are not applicable.

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

b. With no reactor coolant system recirculation loops in operation, immediately initiate action to reduce THERMAL POWER such that it is not within the restricted zone of 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 one or two reactor coolant system recirculation loops in operation and total core flow less than 45% but greater than 39% of rated core flow and THERMAL POWER within the restricted zone of Figure 3.4.1.1-1:

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

      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, within 15 minutes initiate corrective action to restore the noise levels within the required limits within 2 hours by increasing core flow or by reducing THERMAL POWER.

d. With one or two reactor coolant system recirculation loops in operation and total core flow less than or equal to 39% and THERMAL POWER within the restricted zone of Figure 3.4.1.1-1, within 15 minutes initiate corrective action to reduce THERMAL POWER to within the unrestricted zone of Figure 3.4.1.1-1 or increase core flow to greater than 39% within 4 hours.

**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.

4.4.1.1.1 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 109% and 107%, 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.

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

a. Reactor THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER,

b. The recirculation flow control system is in the Local Manual mode, and

c. The speed of the operating recirculation pump is $\leq$ 90% of rated pump speed.

d. Core flow is greater than 39% when THERMAL POWER is within the restricted zone of Figure 3.4.1.1-1.

4.4.1.1.5 With one reactor coolant system recirculation loop not in operation, within 15 minutes prior to either THERMAL POWER increase or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is $\leq$ 30% of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is $\leq$ 50% of rated loop flow:

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

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

c. $\leq$ 50°F between the reactor coolant within the loop not in operation and the operating loop.

The differential temperature requirements of Specification 4.4.1.1.5b. and c. do not apply when the loop not in operation is isolated from the reactor pressure vessel.

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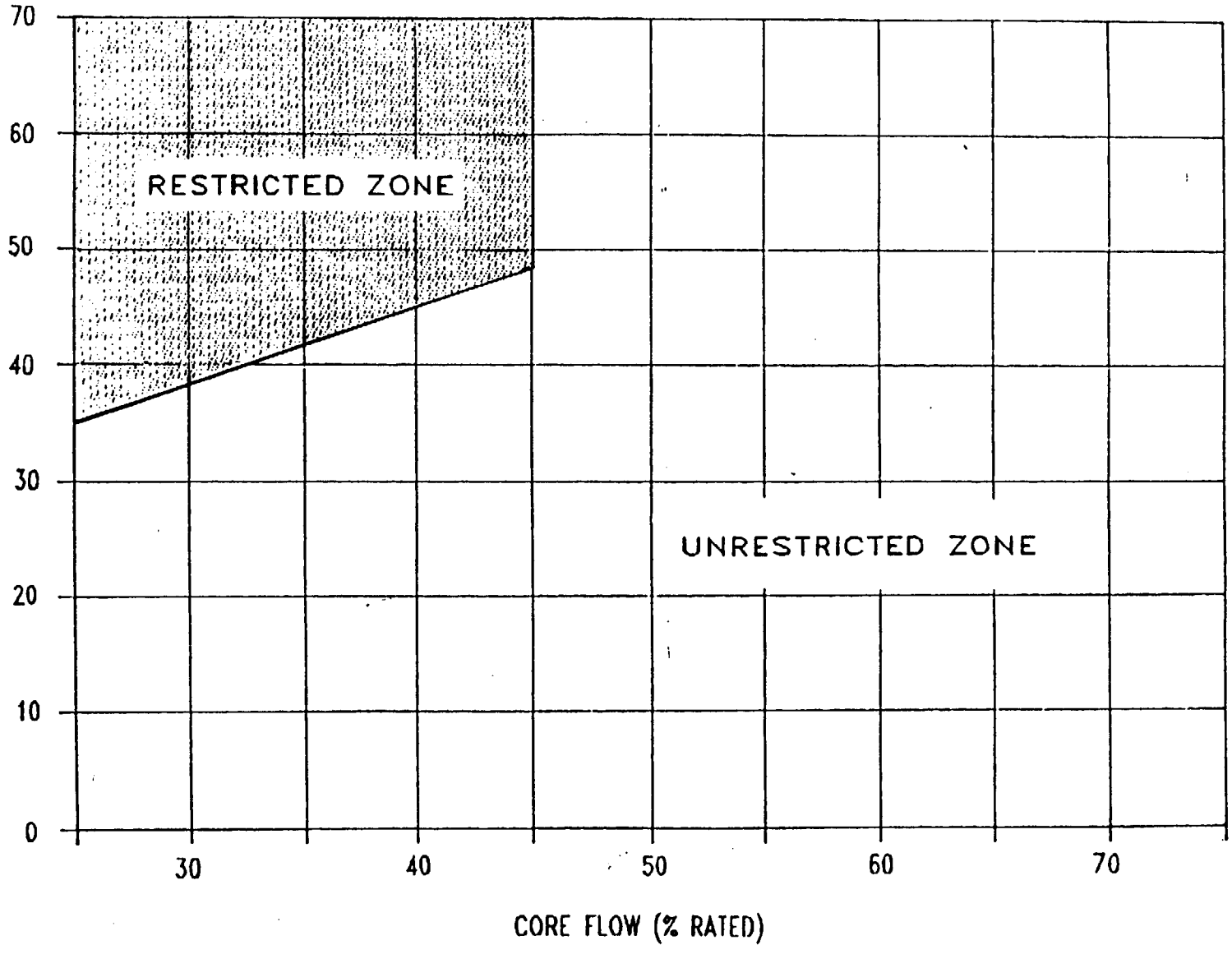
*If not performed within the previous 31 days.

**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.



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 All jet pumps shall be demonstrated OPERABLE as follows:

a. During two recirculation loop operation, 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 while greater than 25% of RATED THERMAL POWER by determining recirculation loop flow, total core flow and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur when both recirculation loop indicated flows are in compliance with Specification 3.4.1.3.

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

   2. The indicated total core flow differs by more than 10% from the established* total core flow value derived from recirculation loop flow measurements.

   3. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from the established* patterns by more than 10%.

*To be determined from the startup test program data.

SURVEILLANCE REQUIREMENTS (Continued)

b. During single recirculation loop operation, each of the above required jet pumps shall be demonstrated OPERABLE at least once per 24 hours by verifying that no two of the following conditions occur:

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

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

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

c. 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 and upon entering single recirculation loop operation.

*To be determined from the startup test program data.

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* during two recirculation loop operation.

ACTION:

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

a. Restore the recirculation loop flows to within the specified limit within 2 hours, or

b. Shutdown one of the recirculation loops within the next 8 hours and take the ACTION required by Specification 3.4.1.1.

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.

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 145°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.

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% Δ k/k or R + 0.28% Δ k/k, as appropriate. The 0.38% Δ k/k includes uncertainties and calculation biases. The value of R in units of % Δ k/k is the difference between the calculated value of minimum shutdown margin during the operating cycle and the calculated shutdown margin at the time of the shutdown margin test at the beginning of cycle. 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 accident analysis, and (3) the potential effects of the rod drop accident are limited. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable 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 fuel cladding safety limit during the limiting power transient analyzed in Section 15.2 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 the fuel cladding safety limit. The occurrence of scram times longer then those specified should be viewed as an indication of a systemic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The 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.

BASES

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 NEDE-24011-P-A (Reference 2) will not be exceeded.

Mechanical Design Analysis: NRC approved methods (specified in Reference 2) 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 2. No single fuel rod follows, or is capable of following, this bounding power history. This bounding power history is used as the basis for the fuel design analysis MAPLHGR limit.

LOCA Analysis: A LOCA analysis is performed in accordance with 10 CFR 50 Appendix K to demonstrate that the permissible planar power (MAPLHGR) limits 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.

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

Only the most and least limiting MAPLHGR values are shown in the Technical Specifications for multiple lattice fuel. Compliance with the specific lattice MAPLHGR operating limits, which are available in Reference 3, is ensured by use of the process computer.

The MAPLHGR limits shall be reduced to a value of 0.89 times the two recirculation loop operation limit when in single recirculation loop operation. The constant factor 0.89 is derived from LOCA analyses initiated from single loop operation to account for earlier boiling transition at the limiting fuel node compared to the standard LOCA evaluations.

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 neutron flux-upscale scram trip setpoint and flow biased neutron flux-upscale control rod block functions of the APRM instruments must be adjusted to ensure that the MCPR does not become less than the Safety Limit MCPR or that $\geq$ 1% plastic strain does not occur in the degraded situation. The scram and rod block setpoints are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and CMFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

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

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and presented in Figures 3.2.3-1a, 3.2.3-1b, 3.2.3-1c and 3.2.3-1d.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-2 that are input to a GE-core dynamic behavior transient computer program. The codes used to evaluate transients are discussed in Reference 2.

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 factors 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 set point and the corresponding THERMAL POWER along the rated flow control line, the limiting bundle's relative power was adjusted until the MCPR changes with different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the K_f.

3/4.4.1 RECIRCULATION SYSTEM

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 are adjusted as noted in Tables 2.2.1-1 and 3.3.6-2, respectively, and MAPLHGR limits are decreased by the factor given in Specification 3.2.1.

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

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. 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.

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. 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.

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 3.4.1.1-1.

RECIRCULATION SYSTEM (Continued)

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.

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. 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 end 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.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operates 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. The safety/relief valves will be removed and either set pressure tested or replaced with spares which have been previously set pressure tested and stored in accordance with manufacturers recommendations in the specified frequency.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 30 TO FACILITY OPERATING LICENSE NO. NPF-39

AND SUPPORTING SINGLE LOOP OPERATION OF LIMERICK, UNIT 2

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION, UNITS 1 AND 2

DOCKET NOS. 50-352/353

1.0 INTRODUCTION

By letter dated November 4, 1988, (Ref. 1) Philadelphia Electric Company (the licensee or PECo) requested an amendment to Facility Operating License No. NPF-39 for the Limerick Generating Station, Unit 1 (LI). The proposed amendment would revise the Technical Specifications (TSs) to permit operation of the reactor with one of two reactor recirculation loops in service under certain specified conditions. Included with the application was a report analyzing Single Recirculation Loop Operation (SLO) for extended periods of time for LI prepared by General Electric Company (GE) (Ref. 2).

Construction of Limerick, Unit 2 (L2) has been completed. On June 22, 1989, the Commission issued Facility Operating License NPF-83, together with TSs, authorizing fuel loading and precriticality testing of L2. To support SLO of L2 following licensing, PECo submitted by letter dated March 29, 1989 an analysis for L2 and a report prepared by GE (Refs 3 and 4) similar to those submitted for L1. The TSs which were issued with the fuel load license for L2 do not contain the changes and additions being approved by this amendment for L1, since the amendment had not been approved at the time of licensing. The TSs which we expect to issue with any future license for L2 authorizing low power or full power operation will include changes and additions parallel to those approved herein for L1. Limerick, Units 1 and 2 are identical BWR4 reactors. The SLO analyses and results are the same for both reactors, and this evaluation, conclusions and TSs approval apply equally to both reactors. In addition to the proposed changes to the TSs for SLO, PECo has also proposed several administrative TSs changes which are a necessary condition to approval of SLO to resolve potential concerns related to thermal hydraulic stability in this mode of operation.

Current Limerick TSs require shutdown of the reactor within 12 hours when only one recirculation loop is in operation. Proposals for TS changes to allow extended operating time under SLO conditions have been accepted in recent years for a large number of BWRs with similar restrictions. A previously existing primary problem area for SLO relating to thermal hydraulic stability (THS) concerns, has been largely resolved for SLO.

This has occurred by the introduction of TS requiring avoidance of potentially unstable regions of the power flow map and surveillance in neighboring regions, and more recently by interim operating procedures requirements described in NRC Bulletin No. 88-07 and Supplement 1 to that bulletin (Ref. 5). NRC Generic Letters 86-02 and 86-09 (Refs. 11 and 12) present staff positions in the areas of SLO and related THS surveillance requirements.

In addition to the THS changes, it is necessary to reexamine the analyses of abnormal operational transient and accident events under SLO conditions and provide for required changes, including TS changes, of trip setpoints and operating and safety limits resulting from the changed reactor conditions. The necessary analyses are provided in the GE reports (Refs. 2 and 4), and PECO has proposed the necessary changes to the TS. These analyses and changes are similar to those approved in previous reviews of SLO operations.

The March 29, 1989 submittal provided additional analyses to support the proposed TS changes and did not alter the action noticed in the Federal Register on March 8, 1989 or affect the intended no significant hazards determination.

2.0 EVALUATION

GE has provided (Refs. 2 and 3) the results of the reexamination, and where required, reanalysis of transients and accidents relevant to SLO. The events examined are the same as those considered and approved in previous staff reviews of SLO. These include the abnormal operational transients involving flow increase, flow decrease, cold water injection, pressurization and rod withdrawal events. Events requiring analysis have been analyzed with standard, staff approved methodology as described in GESTAR II (Ref. 6). For SLO these events begin at a maximum power level about 30 percent less than that for two loop operation (TLO). Thus maximum transient conditions are for the most part less severe than those analyzed for TLO. Average Power Range Monitor (APRM) and Rod Block Monitor (RBM) trip setpoint equations require adjustment for these analyses and for operations under SLO conditions to account for the changes in actual core flow versus measured flow as a result of backflow in idle jet pumps. These changes in trip equations in turn require TS changes for SLO.

As is normally the case, the reexamination of these transient events by GE resulted in substantial margin to safety limits. The operating limit minimum critical power ratio (MCPR) remains unchanged as determined by TLO. These results are expected and acceptable.

The safety limit MCPR does change, however. Two of the uncertainties involved in the determination of this safety limit are increased by SLO. These are (1) the random noise for the neutron flux detector readings (called the TIP readings) used for power determination, and (2) core flow measurement uncertainty. The analyses for these increased uncertainties are the same as have been presented and accepted in previous SLO reviews.

The result is an increase of 0.01 in the safety limit MCPR. This is a reasonable and acceptable change.

The only accident event, other than LOCA, relevant to SLO is the recirculation pump seizure accident. It was not specifically reanalyzed for Limerick. However, previous reviews for a range of reactors (including a large BWR4) requesting SLO (e.g., Refs. 7, 8 and 9) have provided analyses, approved by the staff, that have shown that the event results in a MCPR significantly above the SLO safety limit. It is concluded that this result is also applicable to Limerick.

SLO affects LOCA calculations primarily by decreasing core flow more rapidly than for TLO and thus decreasing the time to departure from nucleate boiling. To examine this and other effects of SLO, LOCA analyses were performed using standard, approved methodology and covering a full spectrum of large break sizes. These analyses result in a required reduction factor of 0.89 for L1 and 2 current fuel assembly maximum average planar linear heat generation rate (MAPLHGR) limits. With this limit reduction factor the large and small break LOCA results remain within required limits. This type of analysis and the reduction factor (of similar magnitude) have been reviewed and approved in previous SLO reviews and is acceptable for L1 and 2.

In addition to relevant transients and accidents, GE has examined several other areas possibly associated with SLO. These include containment analysis, ATWS, fuel mechanical performance and pressure vessel internal vibrations.

As in previous SLO reviews, because of the more limited power flow region of SLO, the reactor conditions associated with these areas for Limerick SLO generally fall within previously analyzed TLO bounds, with results within required limits. However, to assure conservatism, and in keeping with previous SLO submittals and approvals, there are proposed TS requiring (1) limitations on power (70 percent of rated) and flow (one pump speed 90 percent of rated), (2) additional surveillance on recirculation loop differential temperature to prevent stratification and associated stresses, (3) surveillance for the jet pumps to assure that increased vibration does not adversely affect jet pump performance, and (4) manual flow control to prevent possible control oscillations. These are measures approved in previous SLO reviews and are acceptable for L1 and 2.

It is required for approval of SLO that TS providing for suitable surveillance for monitoring THS be in place. L1 already has TS providing for surveillance related to THS concerns for TLO. PECO has now proposed that these specifications be modified to apply to SLO. There are proposed changes to details of these specifications to insert SLO requirements and to improve the clarity of the specifications and the surveillance regions. These revised TS are similar to others approved for SLO and are acceptable for Limerick. In addition to the THS TS, PECO has responded (Ref. 10) to NRC Bulletin 88-07 and to Supplement 1 to that bulletin which requests action with regard to utility operator training and procedure improvement following the LaSalle instability event and subsequent GE interim recommendations (also adopted by the BWR Owners Group) for operations

related to THS. These responses indicate that L1 and 2 are (or will be) in full compliance with the NRC request to improve training and implement the GE operating recommendations. It is not necessary to alter the TS to implement the recommendations. The THS TS may be changed in the future when final solutions for THS concerns are approved by the NRC.

3.0 TECHNICAL SPECIFICATIONS

PECO has proposed that the following TS be changed to provide for SLO requirements. For the most part the reasons for these changes have already been discussed and staff approval indicated. These changes are applicable to both L1 and 2. There are several administrative changes included to provide clarification of language or intention.

Specification 2.1.2. - The safety limit MCPR is changed to 1.08 for SLO. It remains at 1.07 for TLO. This increase of 0.01 because of increased power and flow noise and uncertainty, as previously discussed, is acceptable.

Table 2.2.1-1. - The APRM trip setpoint change for SLO is added. This change to account for the difference in measured and actual core flow, as previously discussed, is acceptable. No high flow clamp is required since such flow levels are not attainable.

Specification 3.2.1. - The reduction of 0.89 for the SLO MAPLHGRs is provided. As previously discussed, this is acceptable.

Specification 3.2.2. - This also changes the APRM trip setpoint equation for SLO as in Table 2.2.1-1 and is acceptable.

Figures 3.2.3-1a, b, c and d. - These are administrative changes adding a note to indicate that the figures apply to both SLO and TLO. They are acceptable. (Figures 3.2.3-1c and 3.2.3-1d were added to the TSs by Amendment No. 19 issued April 24, 1989. These figures did not exist at the time of the licensee's application. The figures all relate to MCPR limits and all were revised by Amendment 19 which imposed revised limits for the Cycle 3 reload. The note advises the reader that the curves are applicable for both dual and single recirculation loop operation. The addition is administrative and has no safety significance since it does not change anything.)

Table 3.3.6-2. - Along with the scram trip setpoints the rod block type setpoints are changed for both the APRM and RBM to add SLO trips. They are acceptable. This includes the note referring to TS 3.2.2.

Specification 3.4.1.1. - This specification receives a number of additions and changes to account for the majority of requirements for SLO. These include, in the Action section, requirements for flow control in Local Manual mode, power level not above 70 percent, MAPLHGR limit reduced by 0.89, pump speed not above 90 percent, differential temperature

surveillance for power not above 30 percent or for recirculation flow above 50 percent, and reduction of APRM and RBM trip setpoints. These action items have been previously discussed and are acceptable. The associated action times are similar to those previously reviewed and approved for SLO and are reasonable and appropriate for L1 and 2. There are also additions and changes to the Action and Surveillance sections requiring THS surveillance when operating in restricted regions of the power-flow map. The regions are better defined (with a new Figure 3.4.1.1-1) and SLO requirements added. These changes have been previously discussed and are acceptable. There are also additions to the Surveillance section, adding 4.4.1.1.4 and 5 (and footnote) which provide for surveillance requirements for the above actions for SLO. The required surveillance and frequencies are reasonable, generally in accord with previous reviews and acceptable.

Specification 3.4.1.2. - The jet pump specification has been changed to provide for surveillance in SLO conditions. The SLO surveillance is similar to that for TLO in terms of deviations from normal established (in startup measurements) characteristics. This is acceptable.

Specification 3.4.1.3. - There is an administrative change limiting applicability (recirculation loop mismatch) to TLO and indicating that, if one loop is shutdown because of a mismatch, the Action of TS 3.4.1.1 should be followed. This is acceptable.

There are minor changes to Bases 2.0, 3/4.1.3, 3/4.2.1 through .3 and 3/4.4.1 addressing the above TS changes. They suitably reflect the TS changes and are acceptable.

4.0 SUMMARY

We have reviewed the reports submitted by PECO for L1 and 2 proposing TS changes relating to SLO and THS. Based on this review we conclude that appropriate documentation was submitted and that the proposed changes satisfy staff positions and requirements in these areas. Extended SLO operation and THS monitoring in the manner thus described, and as augmented by compliance with the requests of NRC Bulletin 88-07 and Supplement 1, are acceptable.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has

been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (54 FR 9919) on March 8, 1989 and consulted with the State of Pennsylvania. No public comments were received and the State of Pennsylvania did not have any comments.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and the security nor to the health and safety of the public.

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Dated: June 30, 1989

REFERENCES

1. Letter and enclosures from E. Bradley, PECO, to T. Murley, NRR, dated November 4, 1988, "Limerick Generating Station, Unit 1."

2. GE report NEDC-31300-P, "Single Loop Operation Analysis for Limerick 1," October 1988.

3. Letter from J. Kemper, PECO, to NRC, dated March 29, 1989, "Limerick, Unit 2 Single Loop Operation Analyses."

4. GE report NEDC-31629-P, "Single Loop Operation Analysis for Limerick 2," September 1988.

5. NRC Bulletin No. 88-07: Power Oscillations in Boiling Water Reactors (BWRs), June 15, 1988 and NRC Bulletin No. 88-07, Supplement 1, December 30, 1988.

6. GE report GESTAR II, NEDE-24011-P, Revision 9, dated November 3, 1988.

7. Letter from R. W. Capstick, Vermont Yankee, to V. L. Rooney, NRC dated May 9, 1986, Subject: Single Loop Operation and Thermal-Hydraulic Stability: Clarification of Proposed Technical Specification Change No. 132.

8. Letter from D. Muslof, NSP, to Director of NRR, dated March 24, 1986, "Request for Amendment to Operating License No. DPR-22."

9. Letter from T. Riley, Clinton Power Station, dated December 15, 1986, "Pump Seizure During Single Loop Operation," L30-86(12-15)-6.

10. Letters and enclosures from J. Gallager to NRC, dated September 5, 1988 and March 7, 1989, "NRC Bulletin No. 88-07" and "Response to NRC Bulletin No. 88-07, Supplement 1."

11. Generic Letter No. 86-02, "Technical Resolution of Generic Issue B-19-Thermal Hydraulic Stability," January 23, 1986.

12. Generic Letter No. 86-09, "Technical Resolution of Generic Issue No. B-59-(N-1) Loop Operation in BWRs and PWRs," March 31, 1986.