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. 19
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 January 27, 1989, as supplemented by letter dated March 22, 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. 19 , 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 as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 24, 1989

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	RClark:tr		WButler
4/21/89	04/12/89	4/18/89	4/24/89

3. This license amendment is effective as 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: April 24, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 19

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

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

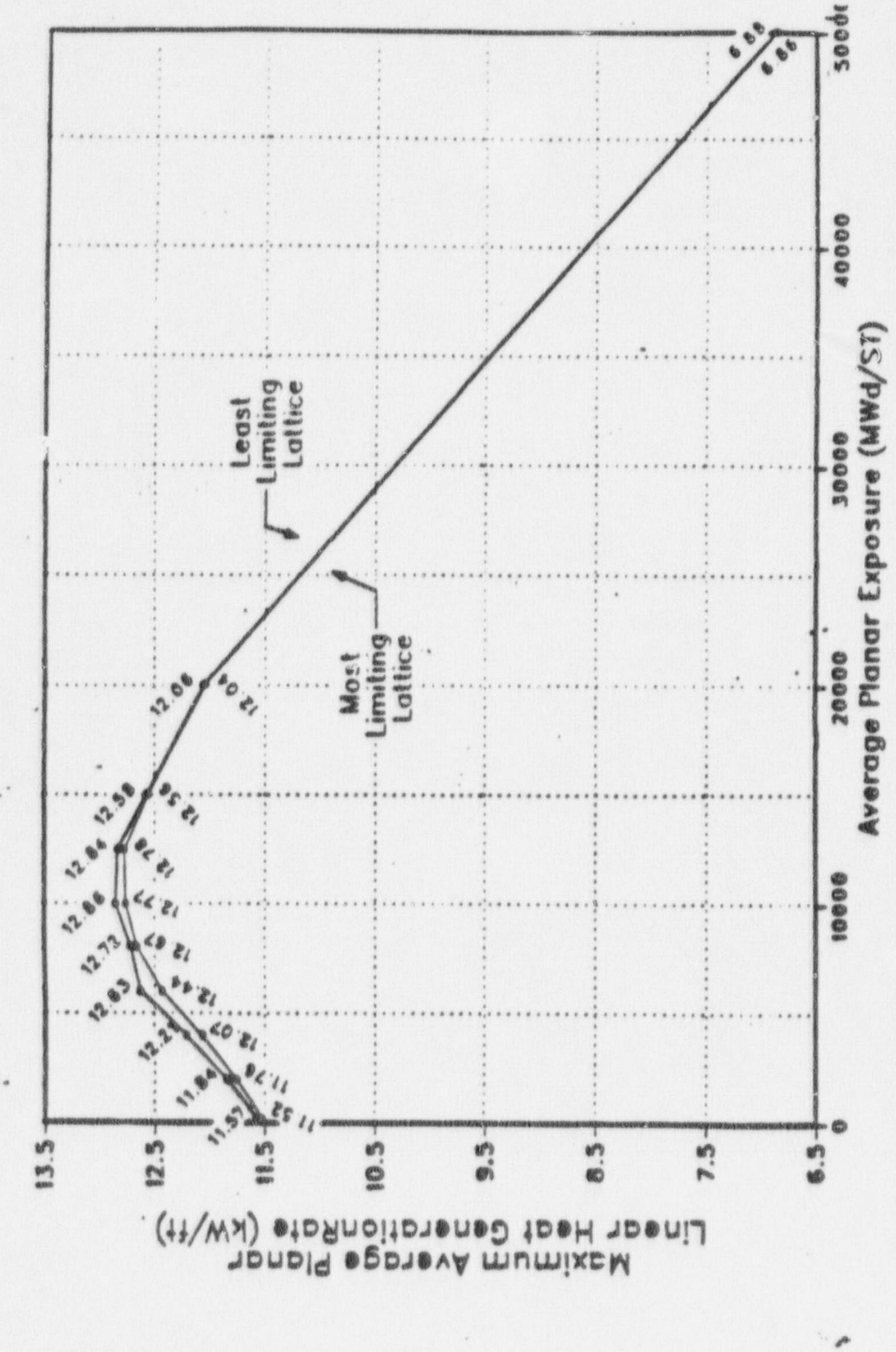
Remove	Insert
v	v*
vi	vi
-	via
3/4 2-6a	3/4 2-6a*
-	3/4 2-6b
-	3/4 2-6c
3/4 2-7	3/4 2-7*
3/4 2-8	3/4 2-8
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
3/4 2-10a	3/4 2-10a
-	3/4 2-10b
-	3/4 2-10c
3/4 3-59	3/4 3-59*
3/4 3-60	3/4 3-60
B 3/4 2-3	B 3/4 2-3*
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	B 3/4 2-5
5-7	5-7
5-8	5-8*

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

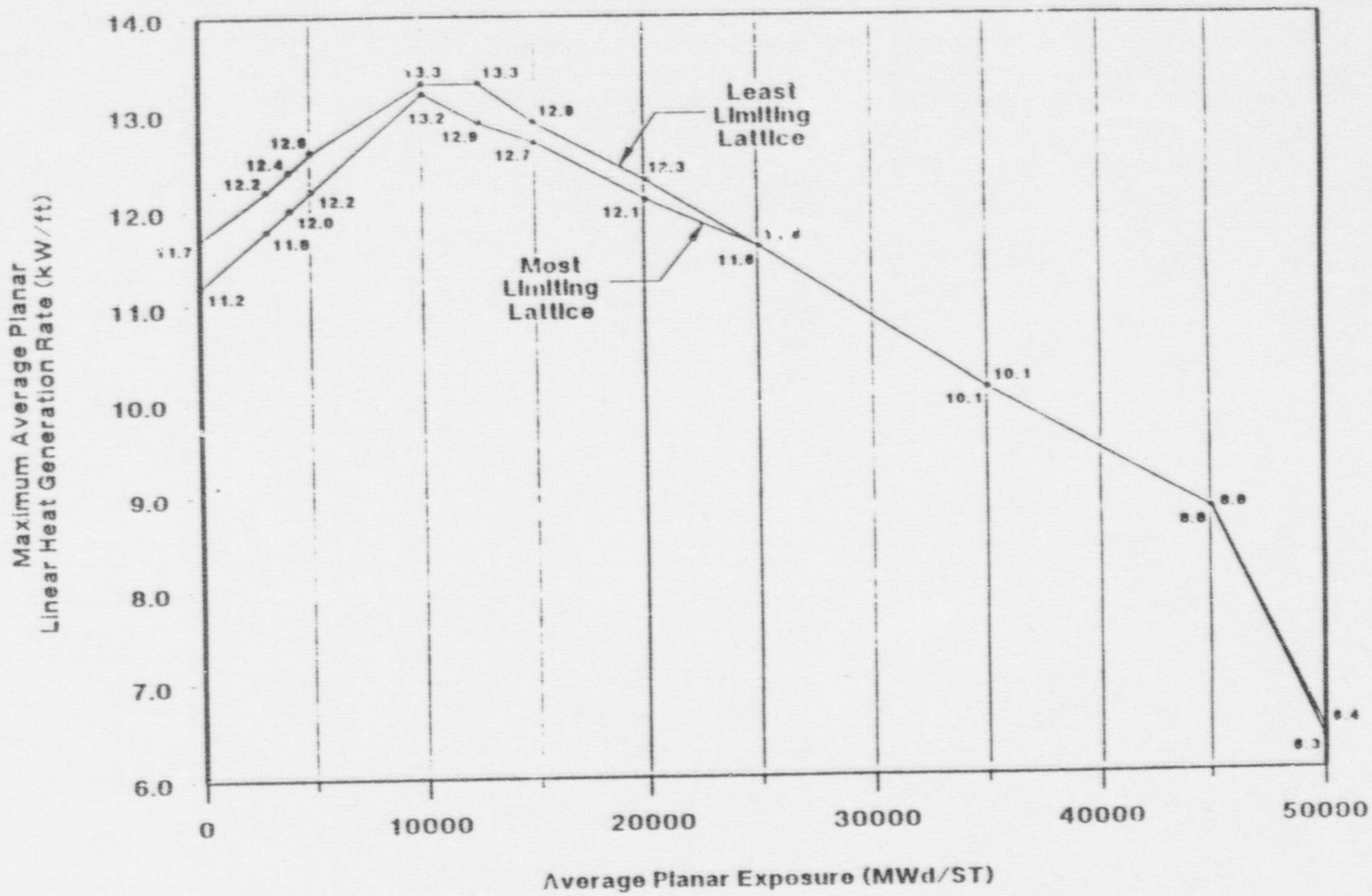
SECTION PAGE

POWER DISTRIBUTION LIMITS (Continued)



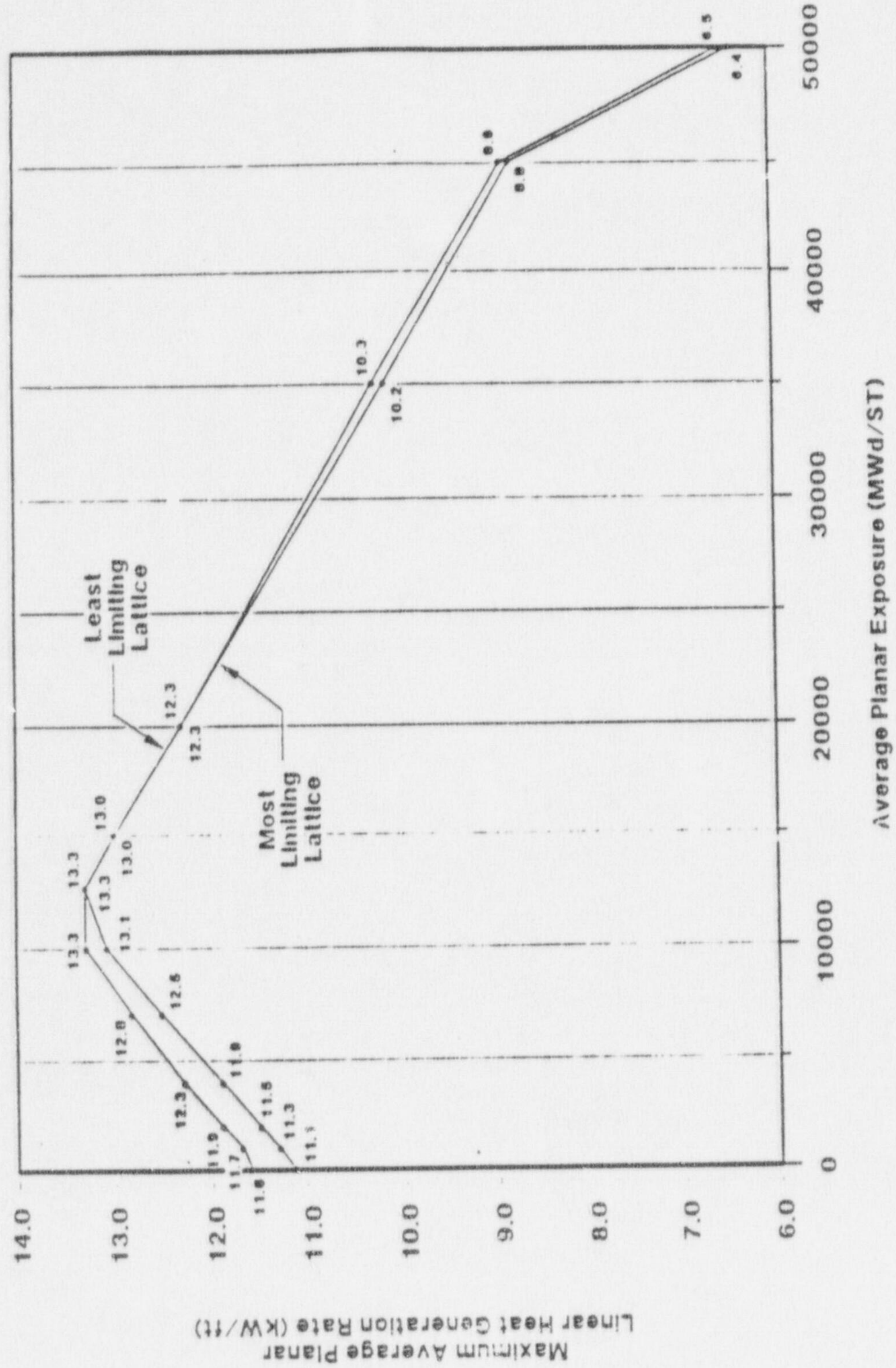
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE FOR FUEL TYPE BC320A (GE8X8EB)

FIGURE 3.2.1-6



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE FOR FUEL TYPE BC318A (GE8x8EB)

FIGURE 3.2.1-7



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE FOR FUEL TYPE BC322A (GE8x8EB)

FIGURE 3.2.1-8

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
$S \leq (0.58W + 59\%)T$	$S \leq (0.58W + 62\%)T$
$S_{RB} \leq (0.58W + 50\%)T$	$S_{RB} \leq (0.58W + 53\%)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER,
W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 100 million lbs/hr,
T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the 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.

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*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 i^{th} surveillance test,

τ_i = average scram time to notch 39 of all rods measured in the i^{th} 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,

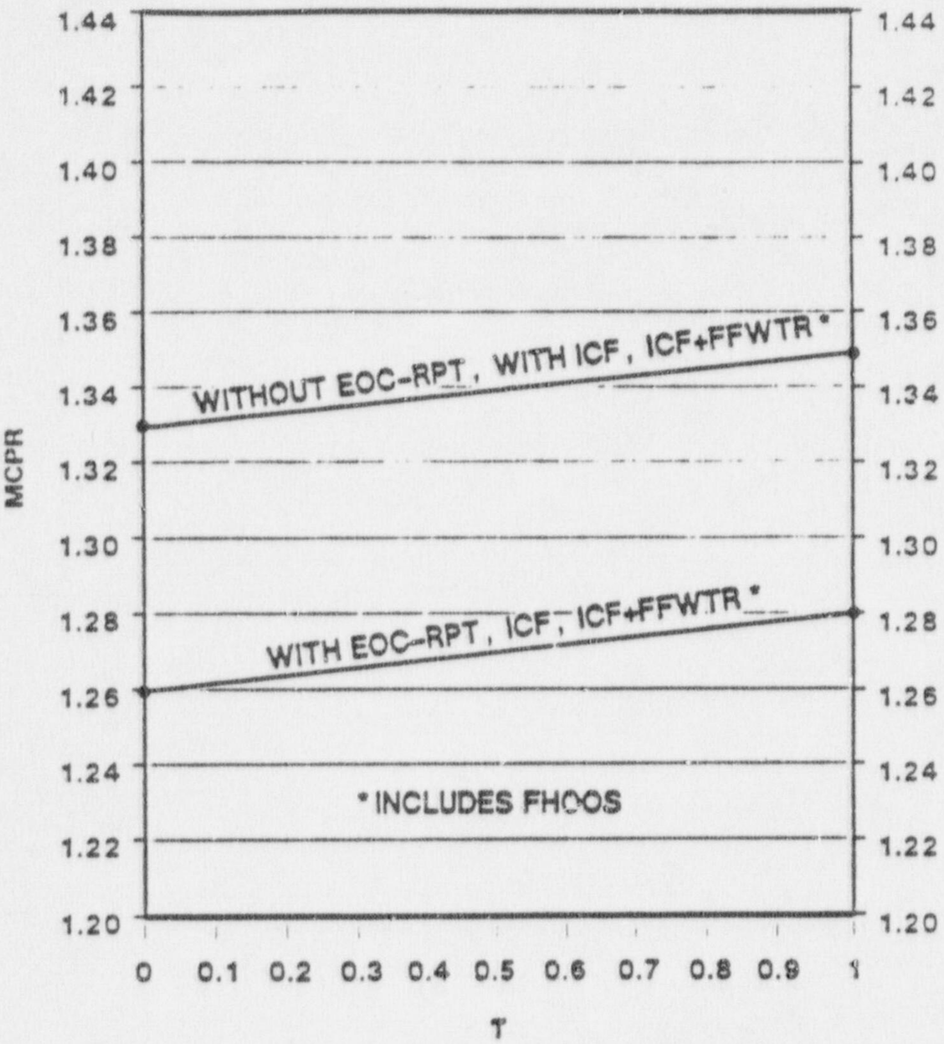
shall be determined to be equal to or greater than the applicable MCPR limit determined from Figures 3.2.3-1a, 3.2.3-1b and 3.2.3-2.

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.



DEFINITIONS:

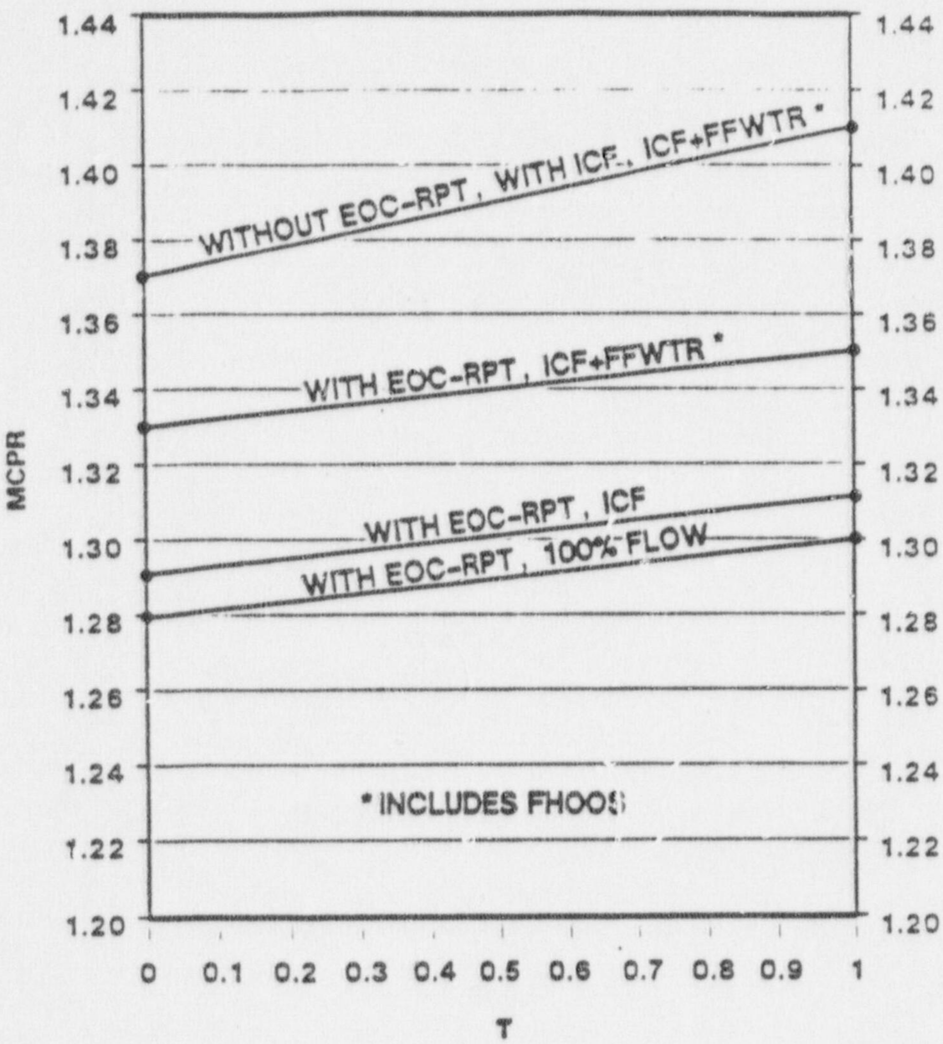
ICF - INCREASED CORE FLOW (UP TO 105% RATED)

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



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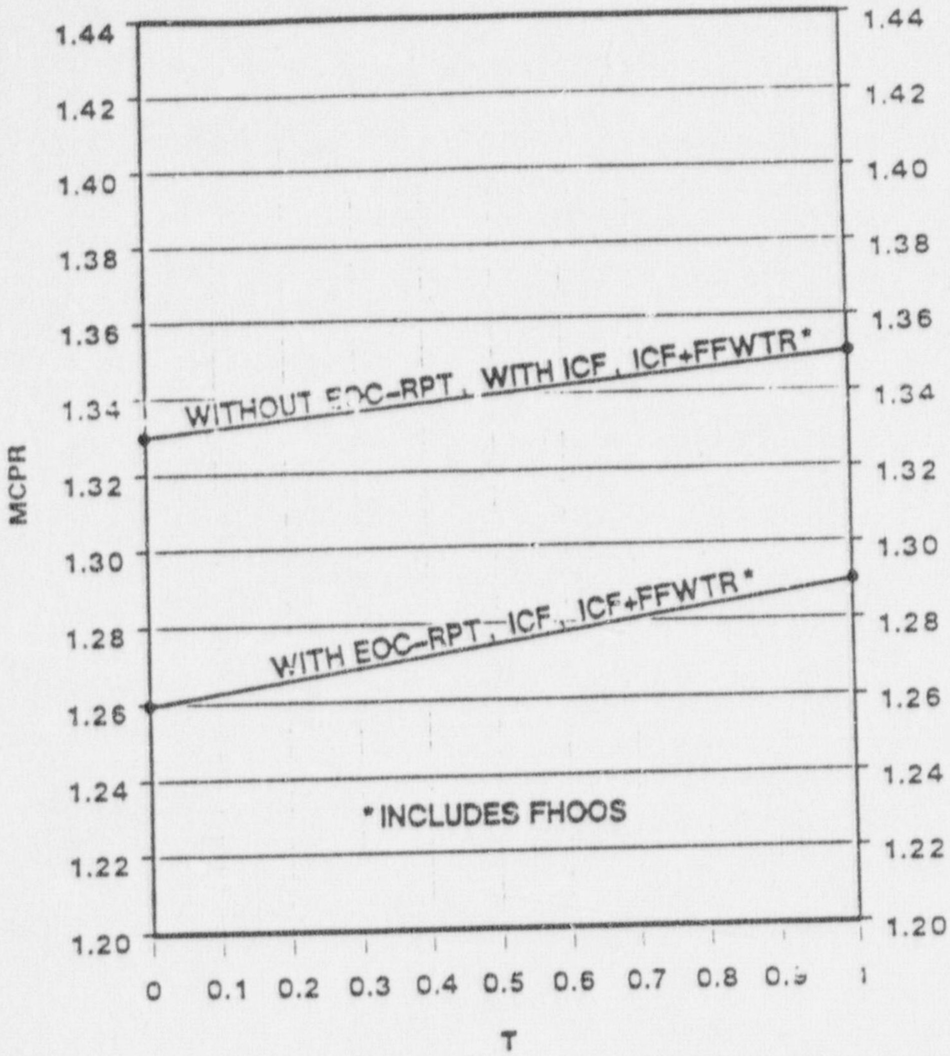
MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS τ (P8x8R/BP8x8R FUEL)
(EOC - 2000 MWD/ST TO EOC)

FIGURE 3.2.3-1b



DEFINITIONS:

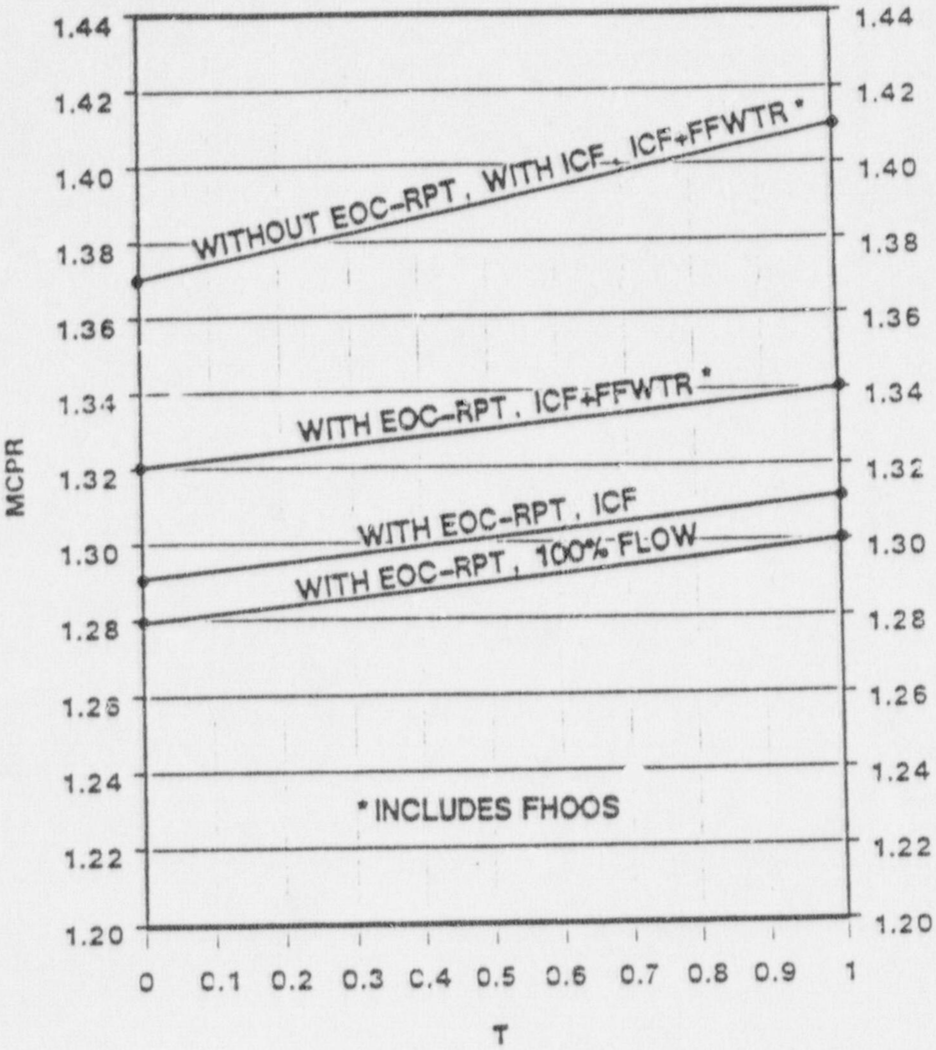
ICF - INCREASED CORE FLOW (UP TO 105% RATED)

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)
(BOC TO EOC - 2000 MWD/ST)

FIGURE 3.2.3-1c



<u>DEFINITIONS:</u>

<u>ICF</u> - INCREASED CORE FLOW (UP TO 105% RATED)

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

<u>FFWTR</u> - 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 $\geq$ 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 referance 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		
i. flow biased	$\leq$ 0.66 W + 40%, with a maximum of,	$\leq$ 0.66 W + 43%, with a maximum of,
ii. high flow clamped	$\leq$ 106%	$\leq$ 109%
b. Inoperative	N.A.	N.A.
c. Downscale	$\geq$ 5% of RATED THERMAL POWER	$\geq$ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - Upscale	$\leq$ 0.58 W + 50%*	$\leq$ 0.58 W + 53%*
b. Inoperative	N.A.	N.A.
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	N.A.	N.A.
b. Upscale	$\leq 1 \times 10^5$ cps	$\leq 1.6 \times 10^5$ cps
c. Inoperative	N.A.	N.A.
d. Downscale	$\geq$ 3 cps**	$\geq$ 1.8 cps**
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	$\leq$ 108/125 divisions of full scale	$\leq$ 110/125 divisions of full scale
c. Inoperative	N.A.	N.A.
d. Downscale	$\geq$ 5/125 divisions of full scale	$\geq$ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High		
a. Float Switch	$\leq$ 257' 5 9/16" elevation***	$\leq$ 257' 7 9/16" elevation

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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 specified in Reference 2, 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.

MINIMUM CRITICAL POWER RATIO (Continued)

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at RATED THERMAL POWER and rated thermal flow.

The K_f factors shown in Figure 3.2.3-2 are conservative for the General Electric Boiling Water Reactor plant operation because the operating limit MCPRs of Specification 3.2.3 are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

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.

3/4.2.4 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. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, November 1975.

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

3. "Basis of MAPLHGR Technical Specifications for Limerick Unit 1," NEDO-31401, February 1987 (as amended).

4. Deleted.

5. Increased Core Flow and Partial Feedwater Heating Analysis for Limerick Generating Station Unit 1 Cycle 1, NEDC-31323, October 1986 including Errata and Addenda Sheet No. 1 dated November 6, 1986.

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of three distinct isolatable zones. Zones I and II are the Unit 1 and Unit 2 reactor enclosures respectively. Zone III is the common refueling area. Each zone has an independent normal ventilation system which is capable of providing secondary containment zone isolation as required.

Each reactor enclosure (Zone I or II) completely encloses and provides secondary containment for its corresponding primary containment and reactor auxiliary or service equipment, and has a minimum free volume of 1,800,000 cubic feet.

The common refueling area (Zone III) completely encloses and provides secondary containment for the refueling servicing equipment and spent fuel storage facilities for Units 1 and 2, and has a minimum free volume of 2,200,000 cubic feet.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall consist of not more than 764 fuel assemblies and shall be limited to those fuel assemblies which have been analyzed with NRC approved codes and methods and have been shown to comply with all Safety Design Bases in the Final Safety Analysis Report (FSAR).

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 cruciform-shaped control rod assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements, DESIGN PRESSURE AND TEMPERATURE (Continued)

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pump.
2. 1500 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.
3. 1500 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,400 cubic feet at a nominal steam dome saturation temperature of 547°F.

5.5 FUEL STORAGE

CRITICALITY

5.5.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section 9.1.2 of the FSAR.

b. A nominal 6.625 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.5.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.5.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 346'0".

CAPACITY

5.5.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2040 fuel assemblies.

5.6 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.6.1 The components identified in Table 5.6.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.6.1-1.