April 24, 19

Docket No.: 50-352

DISTRIBUTION w/enclosures:

Docket File	ACRS (10)	EButcher
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	Tech Branch	

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

Dear Mr. Hunger:

SUBJECT: RELOAD AMENDMENT (TAC NO. 71952)

RE: LIMERICK GENERATING STATION, UNIT 1

The Commission has issued the enclosed Amendment No. 19 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 January 27, 1989 as supplemented by your letter of March 22, 1989.

This amendment changes the Technical Specifications (TSs) to accommodate the second refueling of Limerick 1 with new, previously irradiated and reconstituted fuel assemblies.

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
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 19 to License No. NPF-39
2. Safety Evaluation

cc w/enclosures:
See next page

c/R/ ect

[HUNGER LETT]

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

DF01 /1

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

April 24, 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: RELOAD AMENDMENT (TAC NO. 71952)

RE: LIMERICK GENERATING STATION, UNIT 1

The Commission has issued the enclosed Amendment No. 19 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 January 27, 1989 as supplemented by your letter of March 22, 1989.

This amendment changes the Technical Specifications (TSs) to accommodate the second refueling of Limerick 1 with new, previously irradiated and reconstituted fuel assemblies.

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. 19 to
   License No. NPF-39
2. Safety Evaluation

cc w/enclosures:
See next page

cc:

Troy B. Conner, Jr., Esquire
Conner and Wetterhahn
1747 Pennsylvania Ave., N.W.
Washington, D. C. 20006

Mr. Rod Krich S7-1
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Mr. David Honan N2-1
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Mr. Graham M. Leitch, Vice President
Limerick Generating Station
Post Office Box A
Sanatoga, Pennsylvania 19464

Mr. James Linville
U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Mr. Thomas Kenny
Senior Resident Inspector
US Nuclear Regulatory Commission
P. O. Box 596
Pottstown, Pennsylvania 19464

Mr. Joseph W. Gallagher
Vice President, Nuclear Services
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Mr. John S. Kemper
Senior Vice President-Nuclear
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Mr. Ted Ullrich
Manager - Unit 2 Startup
Limerick Generating Station
P. O. Box A
Sanatoga, Pennsylvania 19464

Mr. John Doering
Superintendent-Operations
Limerick Generating Station
P. O. Box A
Sanatoga, Pennsylvania 19464

Thomas Gerusky, Director
Bureau of Radiation Protection
PA Dept. of Environmental Resources
P. O. Box 2063
Harrisburg, Pennsylvania 17120

Single Point of Contact
P. O. Box 11880
Harrisburg, Pennsylvania 17108-1880

Mr. Philip J. Duca
Superintendent-Technical
Limerick Generating Station
P. O. Box A
Sanatoga, Pennsylvania 19464

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:ef		WButler
4/21/89	04/12/89	4/16/89	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

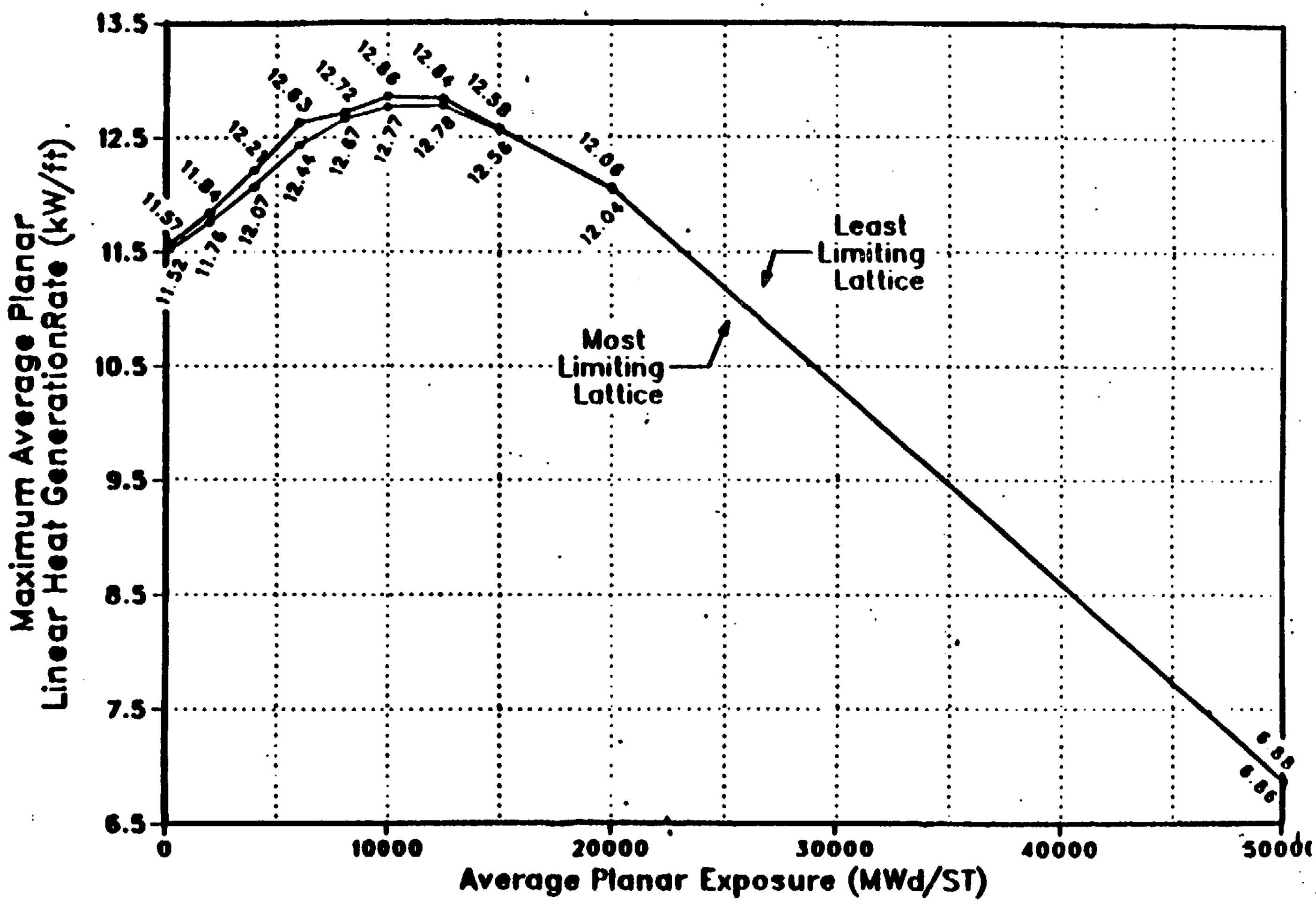
FACILITY OPERATING LICENSE NO. NPF-39

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.*

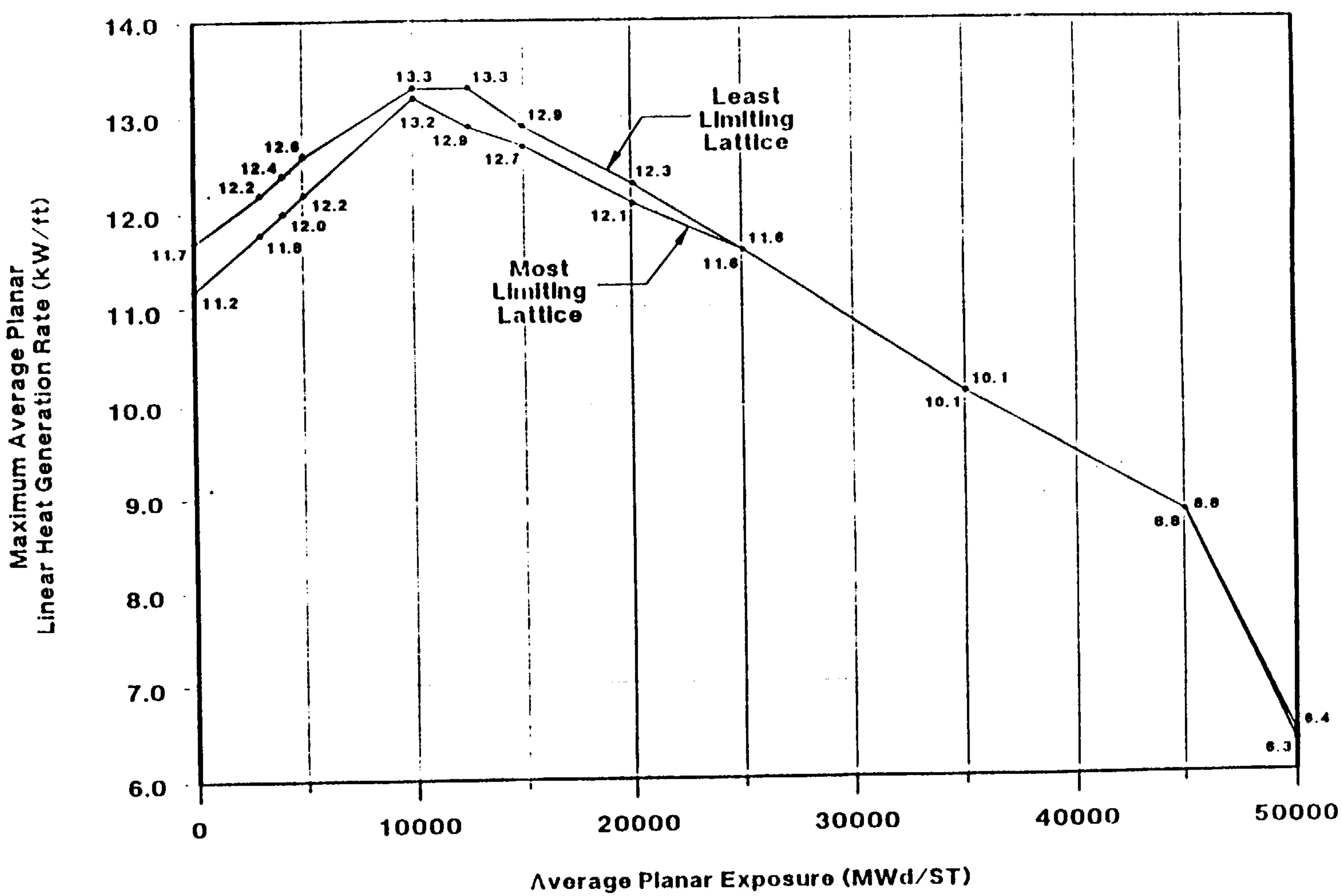
Remove	Insert
v	v*
vi	vi
-	via
3/4 2-6a	3/4 2-6a*
-	3/4 2-6b
-	3/4 2-6c
3/4 2-7	3/4 2-7*
3/4 2-8	3/4 2-8
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
3/4 2-10a	3/4 2-10a
-	3/4 2-10b
-	3/4 2-10c
3/4 3-59	3/4 3-59*
3/4 3-60	3/4 3-60
B 3/4 2-3	B 3/4 2-3*
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	B 3/4 2-5
5-7	5-7
5-8	5-8*

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS



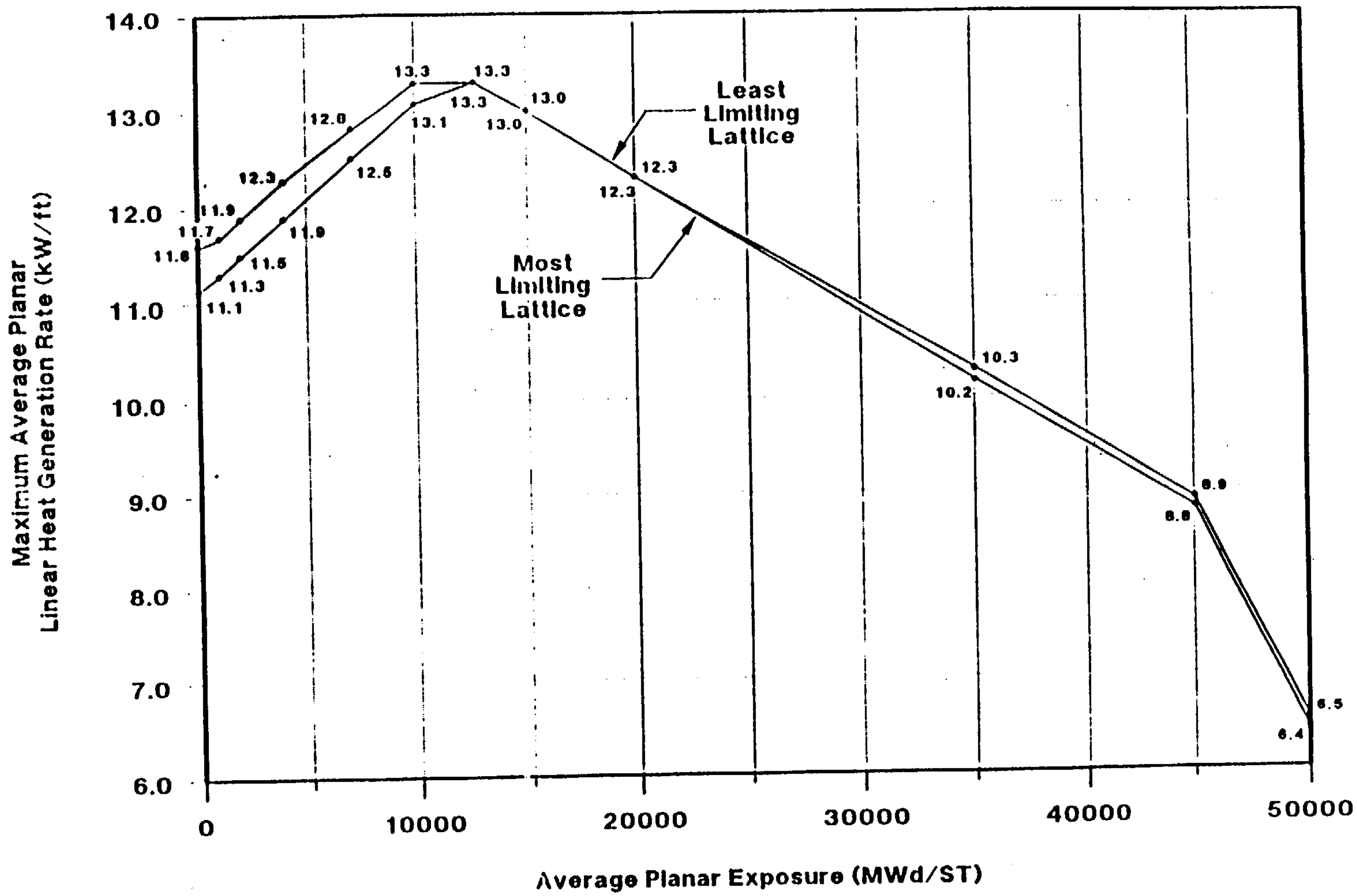
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE FOR FUEL TYPE BC320A (GE8X8EB)

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.

*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,

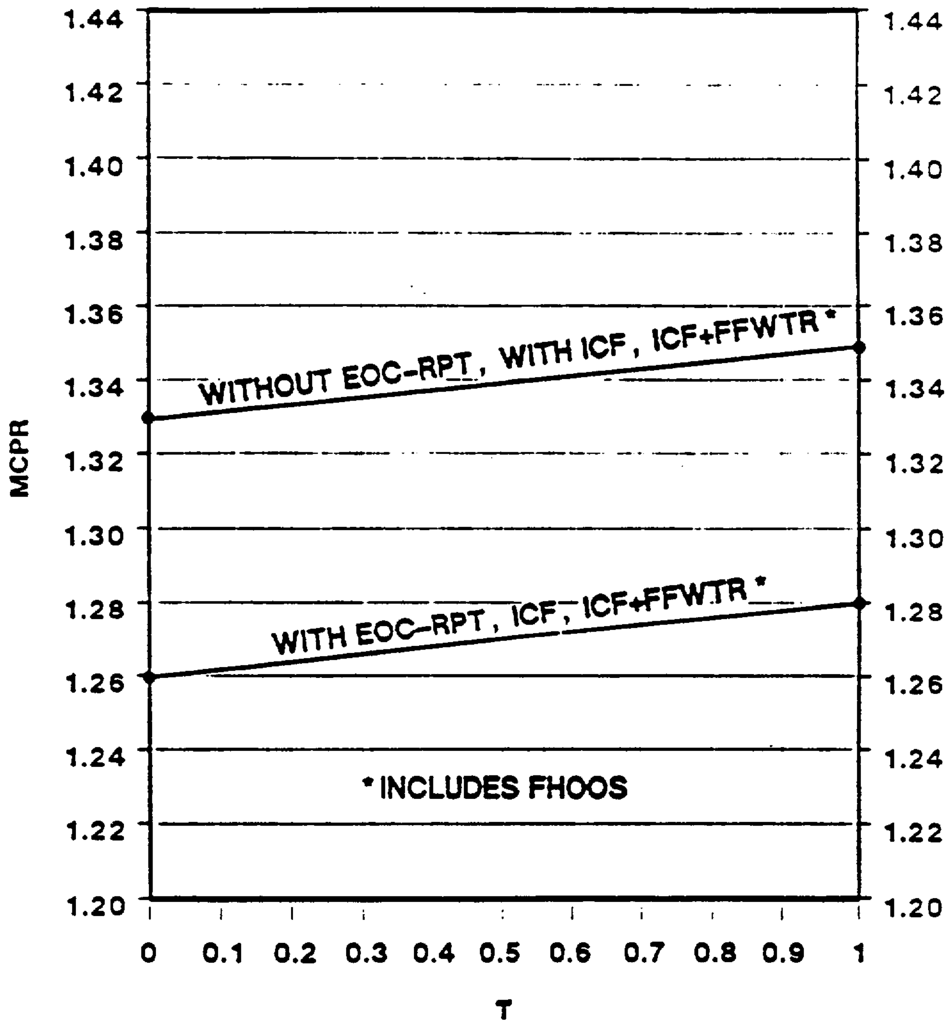
shall be determined to be equal to or greater than the applicable MCPR limit determined from Figures 3.2.3-1a, 3.2.3-1b and 3.2.3-2.

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:

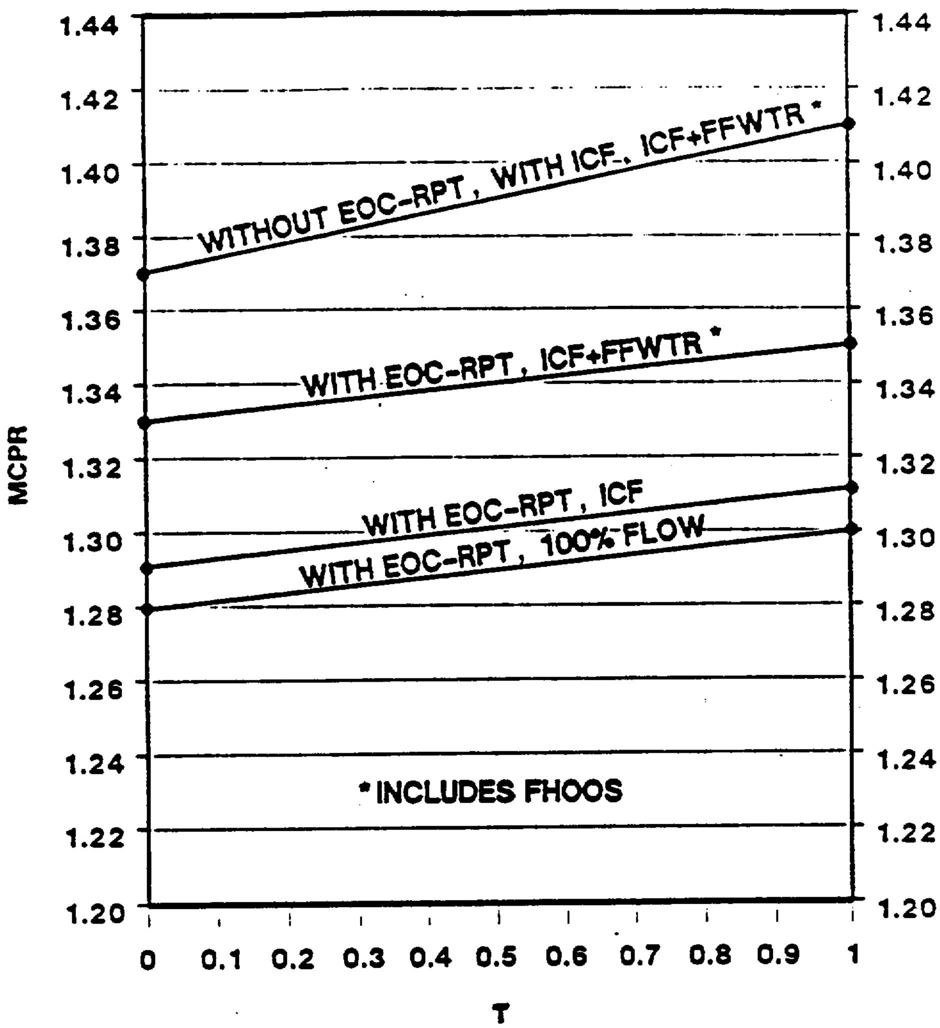
ICF - INCREASED CORE FLOW (UP TO 105% RATED)

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



DEFINITIONS:

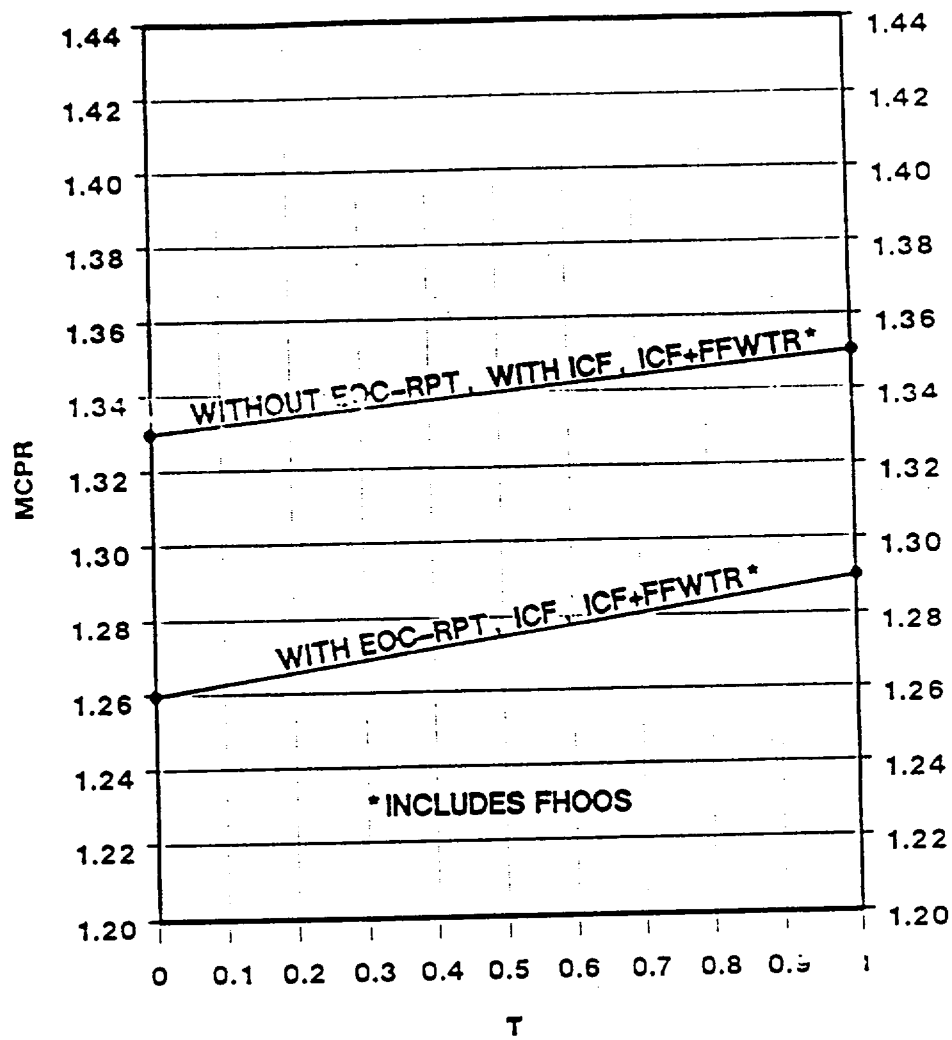
ICF - INCREASED CORE FLOW (UP TO 105% RATED)

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)
(EOC - 2000 MWD/ST TO EOC)

FIGURE 3.2.3-1b



DEFINITIONS:

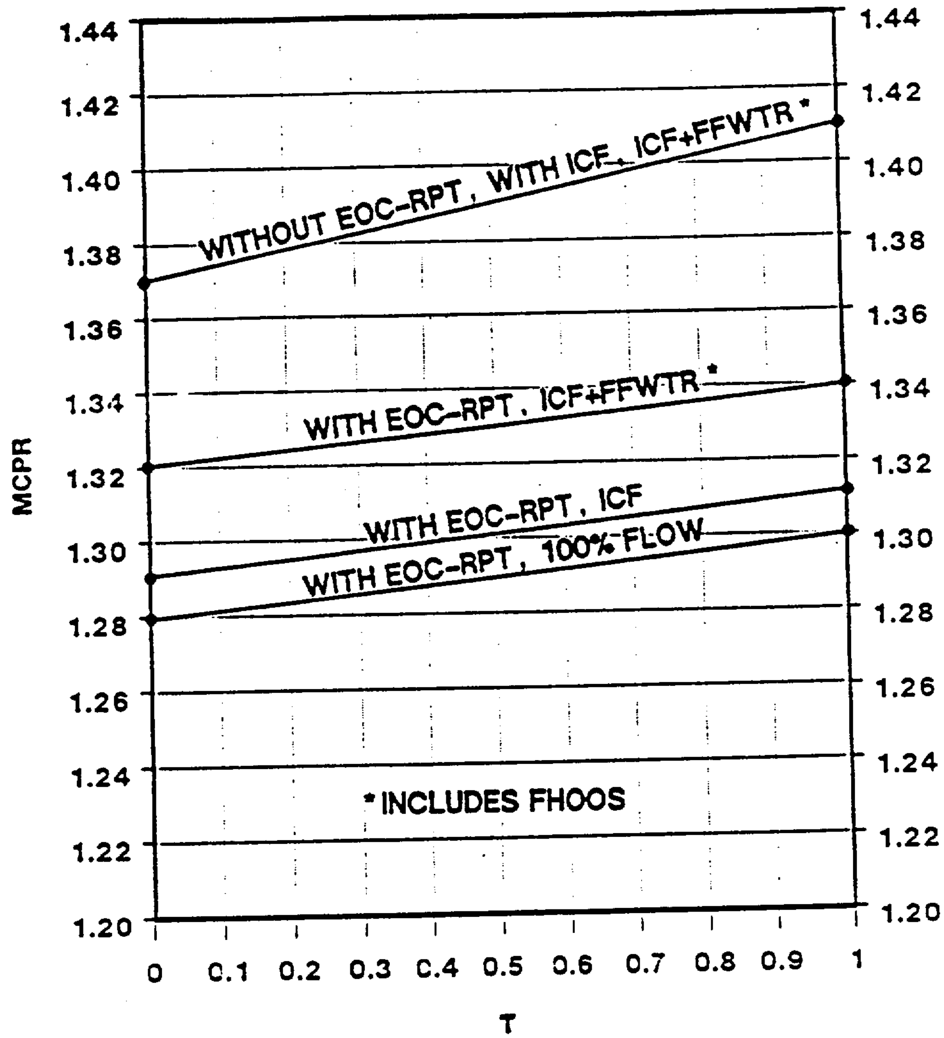
ICF - INCREASED CORE FLOW (UP TO 105% RATED)

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



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 $\geq$ 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 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		
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

LEFT INTENTIONALLY BLANK

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 19 TO FACILITY OPERATING LICENSE NO. NPF-39

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION, UNIT 1

DOCKET NO. 50-352

1.0 INTRODUCTION

By letter dated January 27, 1989 as supplemented by letter dated March 22, 1989, Philadelphia Electric Company (the licensee) requested an amendment to Facility Operating License No. NPF-39 for the Limerick Generating Station, Unit 1. The proposed amendment would change the Technical Specifications (TSs) to accommodate the second refueling of the reactor with new, previously irradiated and reconstituted fuel assemblies.

2.0 DISCUSSION

Limerick Unit 1 was shutdown on January 11, 1989 for the second refueling outage. Starting in late March 1988, the facility experienced an increasing number of fuel failures which caused gradual power reductions to limit offgas releases. At the time of shutdown, the plant was operating at 41 percent rated power. Prior to the fuel failures, the licensee had planned to replace 224 of the original cycle 1 fuel assemblies for cycle 3 operation, compared to the 268 fuel assemblies that were replaced during the first reload in the summer of 1987. All of these new fuel assemblies contain four rather than two water rods.

In the January 1989 shutdown, there was an intentional delay of several weeks before opening the primary system or turbine to permit decay of radioactive noble gases and sweeping of potential gas pockets out of the system through filtered purge systems. Initial sipping of 296 fuel assemblies from the reactor disclosed 5 leaking fuel assemblies from the initial core load and 13 leakers from the fuel assemblies of the first Unit 1 reload (reload-1) that had only operated for one fuel cycle. As a result of the initial inspection of the reload-1 fuel assemblies, PECo announced on February 15, 1989, that because of the cladding failures in some reload-1 fuel assemblies, almost all of the 268 reload-1 fuel assemblies also would be replaced with new fuel assemblies purchased for Unit 2. This is the first incident of Crud Induced Localized Corrosion (CILC) failures in heat-treated cladding. Previously, CILC attack had not occurred until the fuel exceeded at least 15,000 MWD/T exposure. Because Unit 1 was operated at reduced power to limit off-gas activity, the peak exposures were only about 10,500 MWD/T. In the reload-1 fuel assemblies that were only exposed for one cycle, CILC proceeded at a much faster rate in some fuel rods than that experienced in other plants with this type of fuel failure.

Before Unit 1 was shut down in January 1989, the licensee had planned to reconstitute 84 fuel assemblies discharged during the first refueling outage, replacing non-heat-treated rods with heat-treated rods. The licensee had conservatively established the criteria that only fuel meeting visual standards 1 or 2 would be reinserted for Cycle 3 restart. There were not sufficient donor pins to meet these criteria along with the requisite nuclear characteristics. The licensee revised its criteria so that non-heat-treated rods could be used, all rods would be inspected, and heat-treated rods that did not meet visual standards 1 or 2 would be replaced. As a result, the licensee only reconstituted 48 of the 2.48-percent-enriched initial core bundles and two reload-1 bundles. After a thorough inspection, the licensee cleared 42 of the reload-1 bundles (out of 268) for reinsertion into the core for Cycle 3. The other initial core and reload-1 bundles that were in Cycle 2 will be discharged. The remainder of the core will consist of 152 fuel bundles previously discharged from the initial core, the 224 fresh fuel bundles originally scheduled to be inserted, and an additional 296 lower enriched fresh fuel bundles that are on site and that originally were planned to have been loaded in the Unit 2 initial core.

As noted above, the core configuration that will be used for cycle 3 will be different from that projected when the licensee submitted the proposed TS changes. The core will still consist of new fuel assemblies, previously irradiated fuel assemblies and reconstituted fuel assemblies as discussed in the staff's proposed no significant hazards consideration determination (February 22, 1989, 54 FR 7642). In that notice, the staff had specifically stated that core configuration was going to be different from that described in the January 27, 1989 application and that the final core configuration would be dependent on the results of the fuel inspections. The staff also specifically stated that "to account for the above arrangement, the licensee has submitted a bounding reload analysis for cycle 3." Once the final core configuration was determined, the staff requested the licensee to perform a reevaluation and reanalysis of the arrangement. By letter dated March 22, 1989 the licensee provided additional documentation to support their conclusion: "that the revised reload 2 core configuration is bounded by the analysis supporting our TS change request and does not constitute an unreviewed safety question."

In the letter of March 22, 1989, the licensee also advised us that one of the proposed Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus average planar exposure curves that had been submitted as a contingency option would not be needed. The proposed curve was for a reconstituted initial core fuel type to account for the possible replacement of up to four original irradiated gadolinia fuel rods with fresh natural uranium fuel rods. The licensee decided not to employ this particular reconstitution option and requested that the optional MAPLHGR curve be deleted from the TS changes requested in the January 27, 1989 application. Deletion of this curve does not affect any of the other proposed TS changes.

The March 22 submittal does not substantially alter the action as noticed nor affect the staff's initial determination, in that the notice indicated that the precise configuration would be determined based on inspection results and the final configuration comports with the bounding reload analysis in the initial submittal.

3.0 EVALUATION

The reload for Limerick Unit 1 cycle 3 (L1C3) uses General Electric (GE) manufactured fuel assemblies and GE analyses and methodologies. Enclosed with the application of January 27, 1989 (Ref. 1), which requested proposed changes to the TSs, were reports (References 2 through 4) discussing the reload and analyses performed to support and justify cycle 3 operation and an extended power operating region.

The reload for L1C3 is a scheduled reload with special consideration given to the requirements resulting from a fuel inspection and reconstitution effort. This effort was necessary because of a potential fuel failure mechanism resulting from Crud Induced Localized Corrosion (CILC). TS changes are few and primarily related to Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Linear Heat Generation Rate (LHGR) limits for the new fuel and Minimum Critical Power Ratio (MCPR) limits for all of the fuel using Cycle 3 core and transient parameters and extended operating regions and conditions. The new fuel is the extended burnup type which has been used in several recent GE reloads and for which particular attention has been given to special aspects of the TS. The areas of change were the same as the prior reload 1 for Limerick Unit 1 (Ref. 5).

The submittal contains a previously approved extension of the original allowed operating region on the reactor power-flow map via an extended load line limit analysis (ELLLA). Increased core flow (ICF), final feedwater temperature reduction (FFWTR) and feedwater heaters out of service (FHOOS) modes of extended operation along with changes to the flow biased neutron flux scram and rod block setpoints necessary for ELLLA and some changed or additional MCPR limits were approved for the previous cycle (Ref. 5). Revised MCPR limits for operation in Cycle 3 are proposed in this amendment request.

A revision of the Design Features section of the TS to describe the use of hybrid hafnium control blade assemblies was included in the request.

3.1 Reload Description

The L1C3 reload will add 224 new GE 8x8EB fuel assemblies (extended burnup type identified in Reference 1) and 296 fresh bundles taken from the Limerick Unit 2 inventory. Those bundles taken from Unit 2 are unirradiated bundles of the same type (BP8x8R) as those previously used in prior reloads of Unit 1. Thus, a total of 520 (over 2/3) of the 764 fuel assemblies will be new fuel. Based on visual exams of the initial core and reload 1 fuel, there will be 48 reconstituted 2.48 wt% enriched initial core bundles and two reload 1 reconstituted bundles. The reconstituted bundles have the same nuclear and mechanical characteristics as the original bundles and are considered equivalent to the fresh 2.48 wt% assemblies from the Unit 1 inventory. There are 42 reload 1 bundles that were only irradiated for one cycle that were inspected and cleared for reinsertion into the core for cycle 3. The remainder of the core will be comprised of 152 initial core fuel bundles that were discharged during the first refueling and that have been inspected and cleared for reinsertion. The loading will be a conventional scatter pattern with low reactivity fuel on the periphery. The assemblies chosen for the periphery are reinserted assemblies from Cycle 1. The reconstituted assemblies will be distributed symmetrically inside the peripheral region in the revised loading pattern.

The March 22 submittal (Reference 9) was reviewed by staff to the extent necessary to confirm that the modeling assumptions used in the reload analyses are valid for the core reconfiguration. The inspection and reconstitution process was necessary because of fuel reliability concerns as a result of a corrosion mechanism which can cause fuel rod cladding degradation (crud-induced localized corrosion). The objective of the program was to provide a sufficient number of reload fuel assemblies to ensure reliable operation of the Limerick Unit 1 core within its licensing basis. Those considerations relative to the proposed Amendment included nuclear design characteristics, the transient and accident safety analysis results, and the proposed operating thermal limits.

The revised loading pattern uses an increased number of fresh fuel assemblies than normally associated with a typical reload. This creates an insufficient hot excess reactivity during the initial period after startup which necessitates a power derate to maintain the licensing basis thermal limits (Reference 10).

3.2 Fuel Design

The new fuel for Cycle 3 is the GE extended burnup fuel GE8x8EB. The fuel designations are BC 318A and BC 322A. This fuel type has been approved in the Safety Evaluation Report for Amendment 10 to GESTAR II (Refs. 6 and 7) and was used in the prior Cycle 2 reload for Limerick Unit 1. The specific description of this fuel has been submitted in Amendment 18 to GESTAR II which has been accepted in Reference 8.

In operation the GE8x8EB fuel will be assigned a number of axial lattice regions and appropriate MAPLHGR limits, which have been determined by approved thermal-mechanical and loss of coolant analyses (LOCA) calculations, will be applied to each of these regions. Staff approval for this approach has been granted for the prior Cycle 2 for Limerick 1 (Ref. 5) and remains acceptable.

3.3 Nuclear Design

The nuclear design for L1C3 has been performed by GE with the approved methodology described in GESTAR II (Ref. 7). The results of these analyses are given in the GE reload report (Ref. 2) in standard GESTAR II format. The results are within the range of those usually encountered for BWR reloads. In particular, the shutdown margin is 4.8% and 1.2% delta-k at Beginning of Life (BOL) and at the exposure of minimum shutdown margin respectively, thus fully meeting the required 0.38% delta-k. The Standby Liquid Control System also meets shutdown requirements with a shutdown margin of 3.4% delta-k. Since these and other L1C3 nuclear design parameters have been obtained with previously approved methods and fall within expected ranges, the nuclear design is acceptable.

3.4 Thermal-Hydraulic Design

The thermal-hydraulic design for L1C3 has been performed by GE with the approved methodology described in GESTAR II and the results are given in the GE reload report (Ref. 2). The parameters used for the analyses are those approved in Reference 7 for the Limerick class BWR 4. The GEMINI system of methods (approved in Ref. 8) was used for relevant transient analyses. The revised constants used to calculate the mean scram time, and which are a part of the TS changes for L1C3 (TS 3.2.3), were also approved in Reference 8. These methods and parameters are acceptable.

The Operating Limit MCPR (OLMCPR) values are determined by the limiting transients, which are usually Rod Withdrawal Error (RWE), Feedwater Controller Failure (FWCF) and Load Rejection Without Bypass (LRWBP). The analyses of these events for LIC3 using the standard, approved (Ref. 7) ODYN Option A and B approach for pressurization transients provide new Cycle 3 TS values of OLMCPR as a function of average scram time, for operation in both standard and extended operating regions. For all standard operating conditions FWCF is controlling at both option A and B limits. With the selected rod block setting of 106% the RWE is not limiting. These OLMCPR results are reflected in TS changes. Approved methods (Ref. 7) were used to analyze these events (and others which could be limiting) and the analyses and results are acceptable and fall within expected ranges.

The Limerick 1 TS have standard staff approved provisions for incore neutron detector monitoring of thermal-hydraulic stability according to the recommendations of GE SIL-380. Thus cycle specific stability calculations are not required, either for standard conditions or the extended temperature and power-flow conditions proposed for Cycle 3 operation.

3.5 Transient and Accident Analyses

The transient and accident analysis methodologies used for LIC3 are described and NRC approval indicated in GESTAR II. The GEMINI system of methods (Ref. 8) option was used for transient analyses. The limiting MCPR events for LIC3 are indicated in Section 3.4. The core wide transient analysis methodologies and results are acceptable and fall within expected ranges.

The RWE was analyzed on a plant and cycle specific basis (as opposed to the statistical approach) and a rod block setpoint of 106% was selected to provide an OLMCPR of 1.25 for all fuel types. The proposed TS change to lower the RBM setpoint from 107% to 106% ensures that the RWE analysis results are bounded by the limiting transient analysis (see Section 3.6). The mislocated assembly event is not analyzed for reload cores on the basis of NRC approved (see Reference S.2-59 of Ref. 7) studies indicating the small probability of an event exceeding MCPR Limits. The misorientation event is not analyzed for (symmetric gap) C lattices. The local transient event analyses are thus acceptable.

The limiting pressurization event, the main steam isolation valve closure with flux scram, analyzed with standard GESTAR II methods gave results for peak steam dome and vessel pressures well under required limits. These are acceptable methodologies and results.

LOCA analyses, using approved methodologies (SAFE/REFLOOD) and parameters were performed to provide MAPLHGR values for the new reload fuel assemblies. These analyses and results are acceptable.

Since Banked Position Withdrawal Sequence rod patterns are used for Limerick 1, a cycle specific control rod drop accident analysis is not required. The basis for this position and NRC approval is presented in Amendment 9 in Reference 7.

3.6 ELLLA, ICF, FFWTR, FHOOS and Inoperable RPT Extensions

The staff evaluation for the previous reload approved extensions to standard operating regions in the GESTAR II standard category of "Operating Flexibility or Margin Improvement Options." The selected options are ELLLA (Extended Load Line Limit Analysis), ICF (Increased Core Flow), FFWTR (Final Feedwater Temperature Reduction), FHOOS (Feedwater Heater Out of Service), and RPT (Recirculation Pump Trip). These have become commonly selected and approved options for a number of reactors in recent years.

The approved ELLLA changes the Average Power Range Monitor (APRM) rod block and scram lines on the power-flow map, and permits operation up to the new APRM rod block line (0.58W + 50%) up to the intersection with the 100 percent power line occurring at a flow of 87 percent. These are standard changes for ELLLA. For ICF the approved flow increase is to 105 percent of rated core flow at 100 percent power. The increased flow is allowed throughout the cycle and after normal end-of-cycle (with or without FFWTR) with reactivity coast down. FFWTR involves feedwater temperature reduction up to 60°F (to 360°F at full power) and is proposed only for operation after normal end-of-cycle. Limiting events have been analyzed for cycle extension to the exposure attainable using ICF and FFWTR at full power. For L1C3 the limiting MCPR is bounded by the limits for ICF plus FFWTR and may be used throughout the cycle, including cycle extension.

The GE L1C3 reload report (Ref. 2) presents additional calculations of limiting MCPR transients specifically for L1C3. The transient analyses demonstrate that the licensing basis results (e.g., 100 percent flow, 100 percent power for pressurization transients) bound the ELLLA region results (e.g., 87 percent flow, 100 percent power). These conclusions apply to all relevant MCPR events such as pressurization, rod withdrawal and flow runout events. Changes to MCPR TS are not required because of ELLLA adoption. Other relevant areas such as over pressure protection, LOCA and containment analysis have also been examined, and the analyses indicate that results are within allowable design limits. Thermal-hydraulic stability will be provided for by appropriate surveillance. The analyses have been done with approved methodologies and the results are similar to previously approved ELLLA extensions. Thus operation within the ELLLA region is acceptable for L1C3.

The standard (Ref. 7) relevant limiting transients and resulting OLMCPR values were calculated for L1C3 for normal operations for appropriate limiting ELLLA conditions, and for the approved ICF, FFWTR and FHOOS extension conditions. These calculations included the cycle extension conditions. The results provide OLMCPR values for the TS (MCPR versus scram speed based on ODYN option A and B limits). They use a standard approved methodology and are acceptable.

It was assumed for these transients that the RPT is operable. The limiting MCPR event (FWCF) was also calculated for limiting extension conditions assuming an inoperable RPT. This resulted in increased MCPR limits. Operation with inoperable RPT is proposed for L1C3 throughout the cycle and for various extension conditions using these increased limits. These calculations follow standard procedures for the inoperable RPT extension and operation within these limits is acceptable for L1C3.

3.7 Technical Specifications

The TS changes for L1C3 are primarily to provide for:

(a) MAPLHGR limits for the new fuel were determined by approved methods. The additions are Figure 3.2.1-7 and 3.2.1-8 and are acceptable.

(b) Revised MCPR limits for Cycle 3, for extended operation and for the new (GE8x8EB) fuel. The changes are to TS 3.2.3 and Figures 3.2.3-1a, 1b, 1c, and 1d and are acceptable.

(c) The Rod Block Monitor maximum trip setting in Table 3.3.6-2 is set at 106% to correspond to the selection in the RWE analysis. This is acceptable.

The acceptability of each of the above changes has been previously discussed in this review. There is also a change to the listed constants in TS 3.2.3 used to calculate the mean scram time. These constants were approved in the review of Amendment 11 to GESTAR II, (Ref. 9) and are acceptable. Finally, there are administrative changes to the Design Features, to a reference and to the Bases corresponding to the altered TS. These changes are also acceptable.

3.8 Considerations Related to Fuel Inspection and Reconstitution Effort

"At the request of the staff, the licensee in Reference 9 provided an evaluation of the effects of the inspection and reconstitution effort on the safety analysis results provided in the original Amendment request (Ref. 1). The staff has reviewed the additional information and confirmed that the core configuration is bounded by the analyses discussed in the prior sections of this SE.

The specific areas reviewed, with the staff conclusions, were:

a. Fuel design - The 48 rebuilt bundles were reconstituted by replacing up to four rods with donor rods having the same initial U-235 and Gadolinia concentration as the replaced rods. For a few selected replacement rods, no Gadolinia was used. Lattice calculations for each reconstituted bundle type showed an insignificant change in local peaking and k-infinity from the original analysis.

b. Nuclear design - The cold, xenon-free shutdown margin and standby liquid control margin were reanalyzed for the revised core configuration and it was demonstrated that with the modeling of the reconstituted fuel in the reanalysis, the shutdown margin remained within required limits.

c. Transient and accident analyses - The analyses of core-wide pressurization transients, non-pressurization events and the loss-of-coolant accident were evaluated for the revised core configuration. These events are affected primarily by the moderator void coefficient and scram reactivity since the thermal, mechanical, and hydraulic characteristics of the reconstituted assemblies are equivalent to those used in the original analyses. Comparisons of the nuclear parameters for the revised core loading and the reference core loading using the same approved GEMINI methods produced a less negative moderator coefficient for the revised loading pattern compared to the analysis provided in the original Amendment request (Ref. 1). A conservative determination of the power shape resulted in an improved scram response for the revised core configuration. These results, when considered in the reanalyses, make the reference analyses bounding for the core reconfiguration.

Since the above considerations showed the original analyses bound the revised core configuration, no TS changes different from those originally proposed for the Limerick 1 Cycle 3 reload are required.

4.0 SUMMARY

We have reviewed the reports submitted for the Cycle 3 operation of Limerick Unit 1 with extended operating regions. Based on this review, we conclude that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design, and transient and accident analyses are acceptable. The Technical Specification changes submitted for this reload suitably reflect the necessary modifications for operation in this cycle. The core reload configuration resulting from fuel inspection and reconstitution efforts was reviewed and it was determined that no additional Technical Specification changes are required since the analyses provided in Reference 1 and supplements are bounding for the revised core configuration.

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 7642) on February 22, 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.

Principal Contributors: Dick Clark, Mike McCoy

Dated: April 24, 1989

References

1. Letter, and Enclosure 3, from G. A. Hunger, PECo, to Document Control Desk, NRC, dated January 27, 1989. Technical Specification Change Request No. 88-07.

2. GE Report 23A5926, Rev. 0, dated October 1988, "Supplemental Reload Licensing Submittal for Limerick Generating Station, Unit 1, Reload 2, Cycle 3."

3. NEDE-31401, October 1988, "Basis of MAPLHGR TS for Limerick Unit 1."

4. NEDE-31401, November 1988, "Basis of MAPLHGR TS for Limerick Unit 1" Errata and Addenda Sheet No. 2.

5. Amendment No. 7 to Facility Operating License NPF-39 for Limerick Generating Station Unit 1, August 14, 1987.

6. Letter (and attachment) from C. Thomas, NRC, to J. Charnley, GE, dated May 28, 1985, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A-6, Amendment 10."

7. GESTAR II, NEDE-24011, Revision 8, "General Electric Standard Application for Reactor Fuel."

8. Letter (and attachment) from A. C. Thadani, NRC, to J. Charnley, GE, dated May 12, 1988, "Acceptance for Referencing of Amendment 18 to Licensing Topical Report NEDE-24011-P-A, 'General Electric Standard Application for Reactor Fuel.'"

9. Letter, with enclosure, from G. Hunger, PECo, to Document Control Desk, NRC, dated March 22, 1989. Subject: "Limerick Generating Station Unit 1 - Additional Information in Support of the Reload 2 Technical Specification Change Request.

10. Memorandum, M. Restaino, PECo, to R. Clark, NRC, dated March 30, 1989. Reply to verbal request for additional information on Limerick reload Technical Specification changes.