August 14, 1987

Docket No.: 50-352

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Mr. Edward G. Bauer, Jr.
Vice President and General Counsel
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Dear Mr. Bauer:

SUBJECT: CORE RELOAD (TAC NO. 65145)

RE: LIMERICK GENERATING STATION, UNIT 1

The Commission has issued the enclosed Amendment No. 7 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 April 3, 1987.

This amendment revises the Technical Specifications (TSs) to: 1) incorporate the operating limits (e.g., core physics, thermal and hydraulic limits, etc.) for all fuel types for Cycle 2 operation of Limerick, Unit 1; 2) incorporate a change in slope of the flow-biased Average Power Range Monitor (APRM) scram and rod block setpoints for operating flexibility or margin improvement options and 3) modify the Bases associated with core reloads.

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

Enclosures:
1. Amendment No. 7 to License No. NPF-39
2. Safety Evaluation

cc w/enclosures:
See next page

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PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	RClark:ca		WButler
8/14/87	08/05/87	8/10/87	8/12/87

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

August 14, 1987

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2301 Market Street
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2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Edward G. Bauer, Jr
Philadelphia Electric Company

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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. 7
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 April 3, 1987, 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. 7 , are hereby incorporated into this license. Philadelphia Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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P PDR

3. This license amendment is effective prior to startup of Limerick Unit 1 in Cycle 2.

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: August 14, 1987

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	RClark:ca	RMW	WButler
8/4/87	08/05/87	8/10/87	8/12/87

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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: August 14, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 7

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
iii*	iii*
iv	iv
v*	v*
vi	vi
xvii*	xvii*
xviii	xviii
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
B 2-3	B 2-3
B 2-4	B 2-4
3/4 2-1	3/4 2-1
3/4 2-2*	3/4 2-2*
-	3/4 2-6a
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 2-11*	3/4 2-11*
3/4 2-12	3/4 2-12

3/4 3-59*	3/4 3-59*
3/4 3-60	3/4 3-60
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 2-5	B 3/4 2-5

INDEX

DEFINITIONS

SECTION

DEFINITIONS (Continued) PAGE

1.45 UNRESTRICTED AREA..... 1-8

1.46 VENTILATION EXHAUST TREATMENT SYSTEM..... 1-8

1.47 VENTING..... 1-8

Table 1.1, Surveillance Frequency Notation..... 1-9

Table 1.2, Operational Conditions..... 1-10

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

SECTION	PAGE
2.1 SAFETY LIMITS	
THERMAL POWER, Low Pressure or Low Flow	2-1
THERMAL POWER, High Pressure and High Flow	2-1
Reactor Coolant System Pressure	2-1
Reactor Vessel Water Level	2-2
2.2 LIMITING SAFETY SYSTEM SETTINGS	
Reactor Protection System Instrumentation Setpoints	2-3
Table 2.2.1-1 Reactor Protection System Instrumentation Setpoints	2-4

BASES

2.1 SAFETY LIMITS	
THERMAL POWER, Low Pressure or Low Flow	B 2-1
THERMAL POWER, High Pressure and High Flow	B 2-2
Left Intentionally Blank	B 2-3
Left Intentionally Blank	B 2-4
Reactor Coolant System Pressure	B 2-5
Reactor Vessel Water Level	B 2-5
2.2 LIMITING SAFETY SYSTEM SETTINGS	
Reactor Protection System Instrumentation Setpoints	B 2-6

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		PAGE
3/4.0 APPLICABILITY		3/4 0-1
3/4.1 REACTIVITY CONTROL SYSTEMS		
3/4.1.1	SHUTDOWN MARGIN	3/4 1-1
3/4.1.2	REACTIVITY ANOMALIES	3/4 1-2
3/4.1.3	CONTROL RODS	
	Control Rod Operability	3/4 1-3
	Control Rod Maximum Scram Insertion Times	3/4 1-6
	Control Rod Average Scram Insertion Times	3/4 1-7
	Four Control Rod Group Scram Insertion Times	3/4 1-8
	Control Rod Scram Accumulators	3/4 1-9
	Control Rod Drive Coupling	3/4 1-11
	Control Rod Position Indication	3/4 1-13
	Control Rod Drive Housing Support	3/4 1-15
3/4.1.4	CONTROL ROD PROGRAM CONTROLS	
	Rod Worth Minimizer	3/4 1-16
	Rod Sequence Control System	3/4 1-17
	Rod Block Monitor	3/4 1-18
3/4.1.5	STANDBY LIQUID CONTROL SYSTEM	3/4 1-19
	Figure 3.1.5-1 Sodium Pentaborate Solution Temperature/Concentration Requirements	3/4 1-21
	Figure 3.1.5-2 Sodium Pentaborate Solution Volume/Concentration Requirements	3/4 1-22
3/4.2 POWER DISTRIBUTION LIMITS		
3/4.2.1	AVERAGE PLANAR LINEAR HEAT GENERATION RATE	3/4 2-1
	Figure 3.2.1-1 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Initial Core Fuel Types P8CIB278	3/4 2-2

SECTION PAGE

POWER DISTRIBUTION LIMITS (Continued)

Figure 3.2.1-2 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Initial Core Fuel Types P8CIB248..... 3/4 2-3

Figure 3.2.1-3 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Initial Core Fuel Types P8CIB163..... 3/4 2-4

Figure 3.2.1-4 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Initial Core Fuel Types P8CIB094..... 3/4 2-5

Figure 3.2.1-5 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Initial Core Fuel Types P8CIB071..... 3/4 2-6

Figure 3.2.1-6 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure For Fuel Type BC320A (GE8X8EB)..... 3/4 2-6a

3/4 2.2 APRM SETPOINTS..... 3/4 2-7

3/4 2.3 MINIMUM CRITICAL POWER RATIO..... 3/4 2-8

Table 3.2.3-1 Deleted

Figure 3.2.3-1a Minimum Critical Power Ratio (MCPR) Versus (P8X8R/BP8X8R Fuel)..... 3/4 2-10

Figure 3.2.3-1b Minimum Critical Power Ratio (MCPR) Versus (GE8X8EB Fuel)..... 3/4 2-10a

Figure 3.2.3-2 K_f Factor..... 3/4 2-11

3/4.2.4 LINEAR HEAT GENERATION RATE..... 3/4 2-12

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION..... 3/4 3-1

Table 3.3.1-1 Reactor Protection System Instrumentation..... 3/4 3-2

Table 3.3.1-2 Reactor Protection System Response Times..... 3/4 3-6

Table 4.3.1.1-1 Reactor Protection System Instrumentation Surveillance Requirements..... 3/4 3-7

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION	PAGE
RADIOACTIVE EFFLUENTS (Continued)	
Table 4.11.2.1.2-1 Radioactive Gaseous Waste Sampling and Analysis Program	3/4 11-9
Dose - Noble Gases	3/4 11-12
Dose - Iodine-131, Iodine-133, Tritium, and Radionuclides in Particulate Form	3/4 11-13
Ventilation Exhaust Treatment System	3/4 11-14
Explosive Gas Mixture	3/4 11-15
Main Condenser	3/4 11-16
Venting or Purging	3/4 11-17
3/4.11.3 SOLID RADWASTE TREATMENT	3/4 11-18
3/4.11.4 TOTAL DOSE	3/4 11-20
3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING	
3/4.12.1 MONITORING PROGRAM	3/4 12-1
Table 3.12.1-1 Radiological Environmental Monitoring Program	3/4 12-3
Table 3.12.1-2 Reporting Levels For Radioactivity Concentrations In Environmental Samples	3/4 12-9
Table 4.12.1-1 Detection Capabilities For Environmental Sample Analysis	3/4 12-10
3/4.12.2 LAND USE CENSUS	3/4 12-13
3/4.12.3 INTERLABORATORY COMPARISON PROGRAM	3/4 12-14

SECTION PAGE

3/4.0 APPLICABILITY..... B 3/4 0-1

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN..... B 3/4 1-1

3/4.1.2 REACTIVITY ANOMALIES..... B 3/4 1-1

3/4.1.3 CONTROL RODS..... B 3/4 1-2

3/4.1.4 CONTROL ROD PROGRAM CONTROLS..... B 3/4 1-3

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM..... B 3/4 1-4

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE..... B 3/4 2-1

LEFT INTENTIONALLY BLANK..... B 3/4 2-3

3/4.2.2 APRM SETPOINTS..... B 3/4 2-2

3/4.2.3 MINIMUM CRITICAL POWER RATIO..... B 3/4 2-4

3/4.2.4 LINEAR HEAT GENERATION RATE..... B 3/4 2-5

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION..... B 3/4 3-1

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION..... B 3/4 3-2

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION..... B 3/4 3-2

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION..... B 3/4 3-3

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION..... B 3/4 3-4

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION..... B 3/4 3-4

3/4.3.7 MONITORING INSTRUMENTATION

Radiation Monitoring Instrumentation..... B 3/4 3-4

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 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 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) Flow Biased	≤ 0.58 W+ 59%, with a maximum of	≤ 0.58 W+ 62%, with a maximum of
2) High Flow Clamped	≤ 116.5% of RATED THERMAL POWER	≤ 118.5% of RATED THERMAL POWER
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. MCPR greater than 1.07 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).

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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. 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.

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

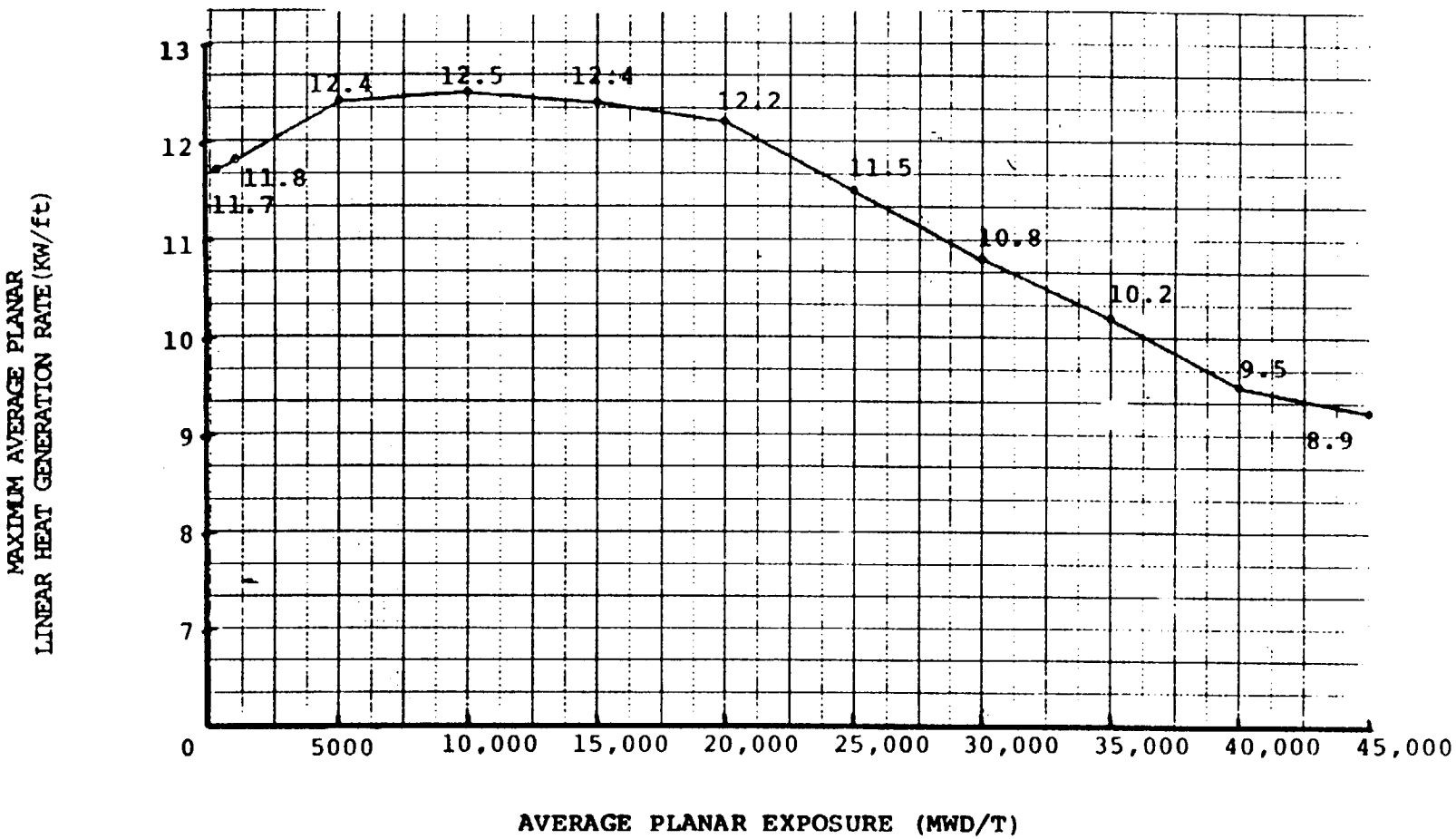
4.2.1 All APLHGRs shall be verified to be equal to or less than the limiting value

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

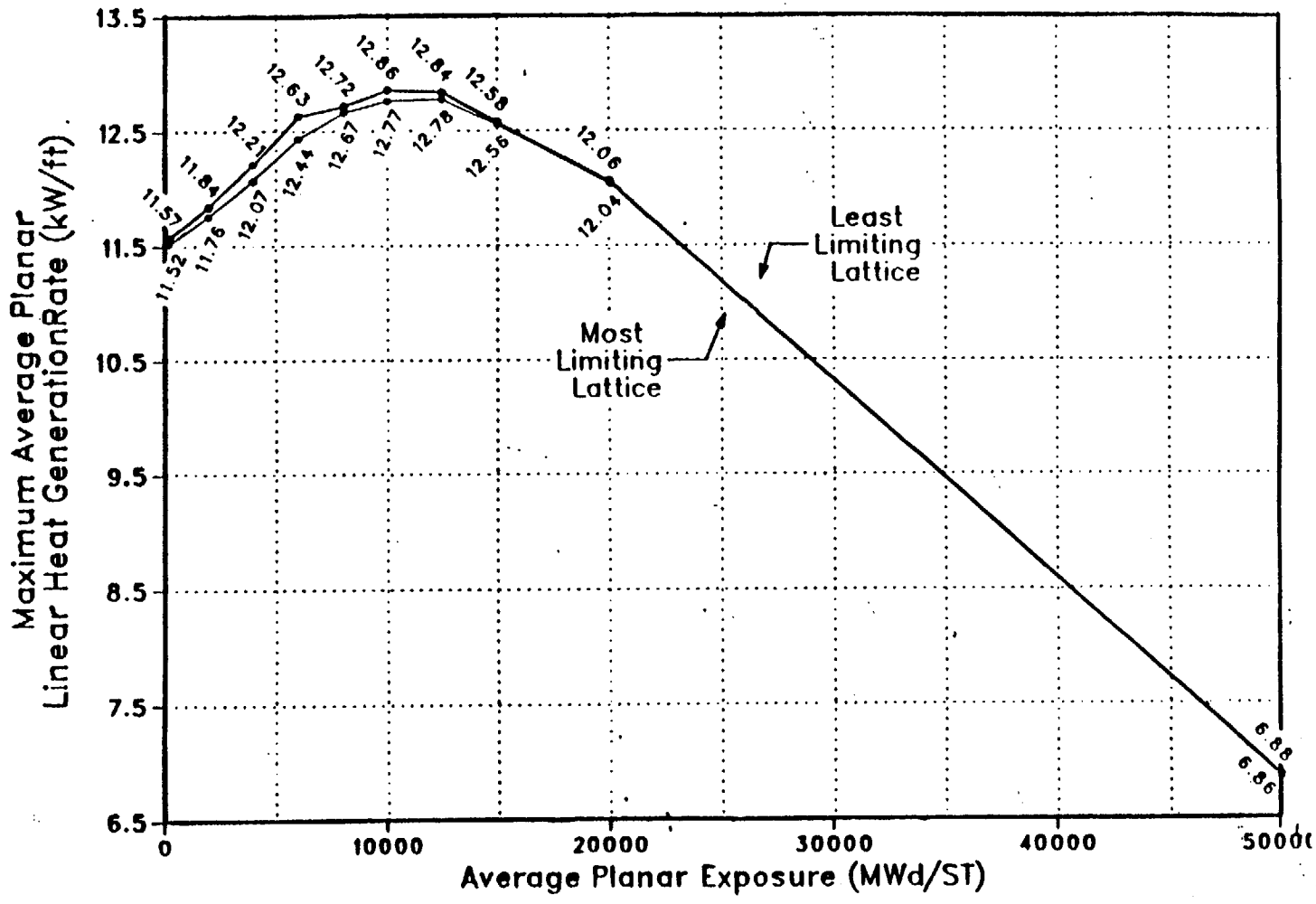
c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

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



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

FIGURE 3.2.1-6

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) and Figure 3.2.3-1b (GE8X8EB 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,

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

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

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 and the provisions of Specification 3.0.4 are not applicable 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) and Figure 3.2.3-1b (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. $\tau = 1.0$ 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,

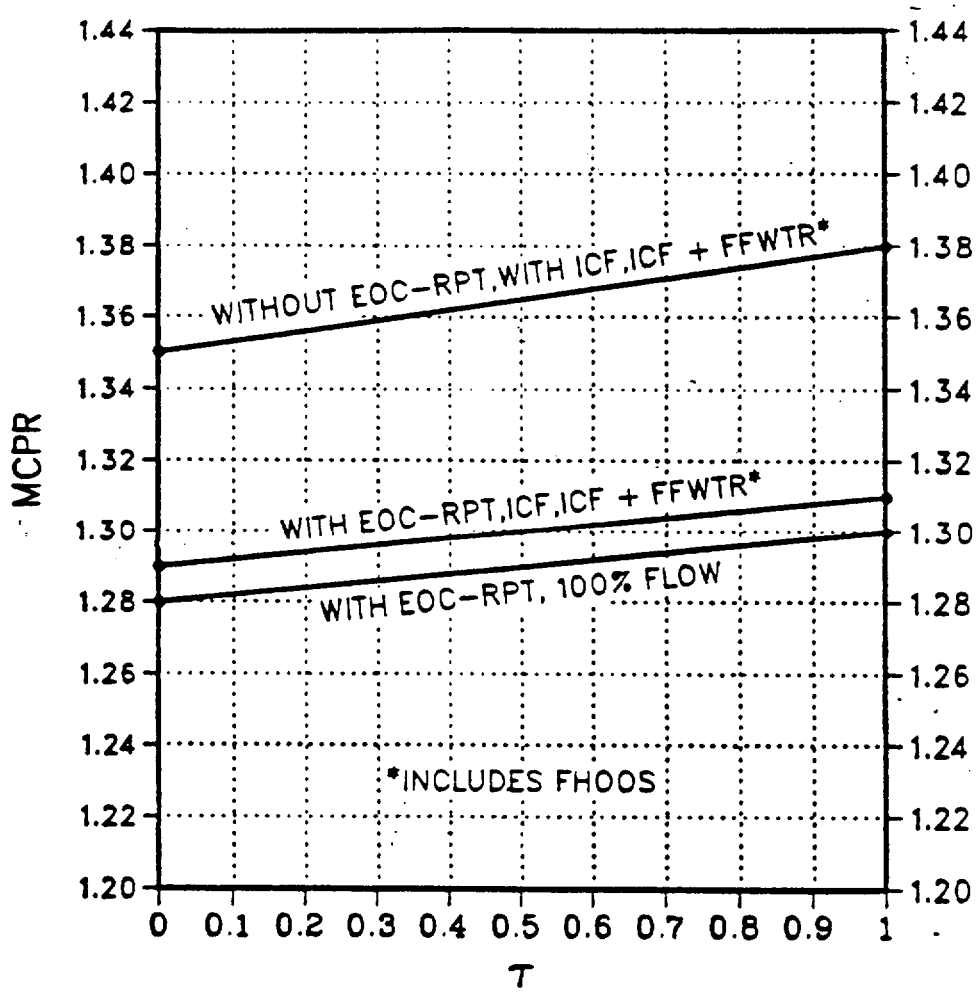
shall be determined to be equal to or greater than the applicable MCPR limit determined from Figures 3.2.3-1a, 3.2.3-1b and 3.2.3-2.

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:

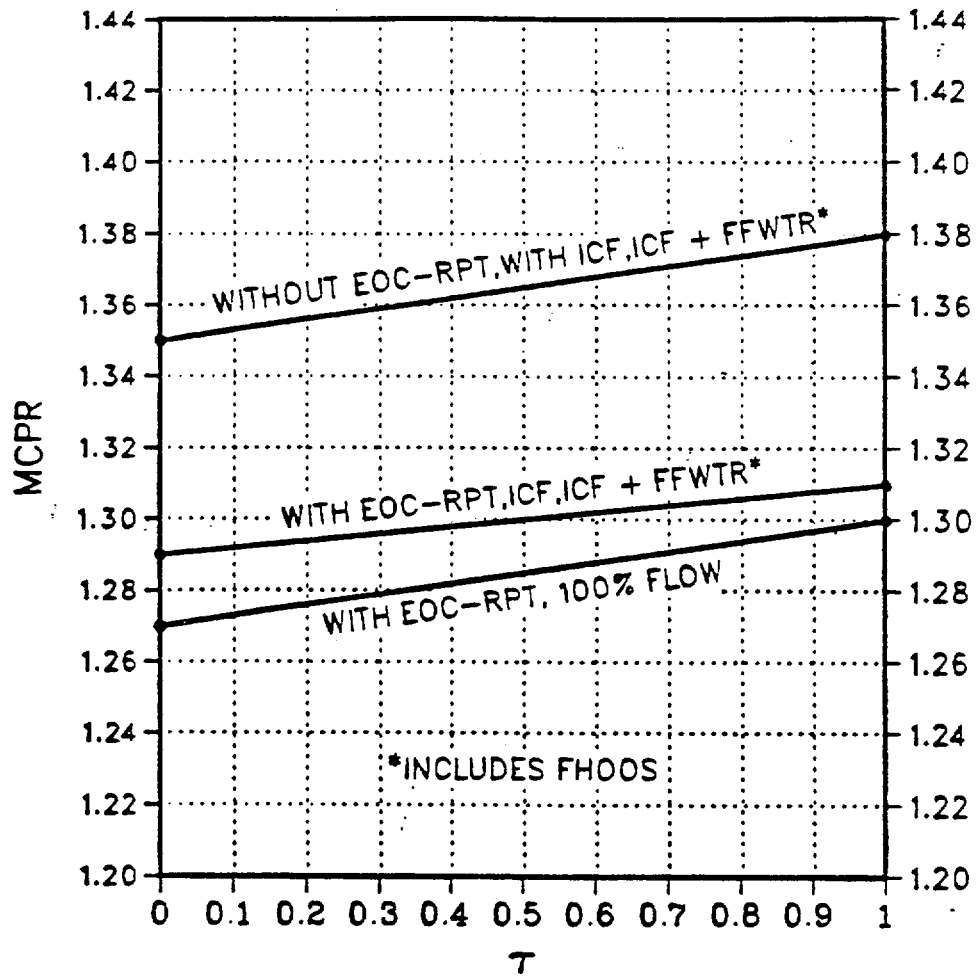
ICF – INCREASED CORE FLOW (UP TO 105% RATED)

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

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

MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS τ (P8X8R/BP8X8R FUEL)

FIGURE 3.2.3-1a



DEFINITIONS:

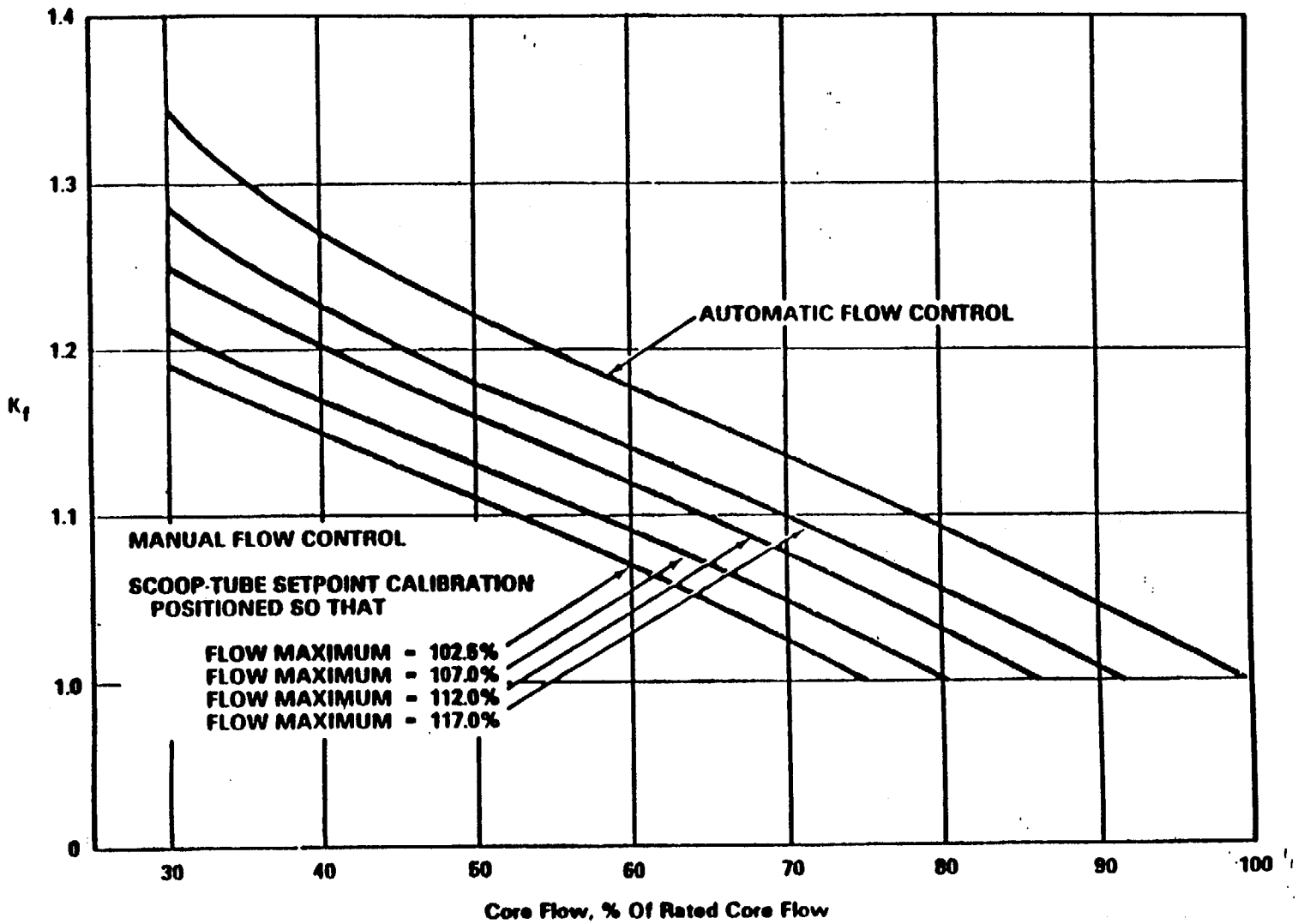
ICF – INCREASED CORE FLOW (UP TO 105% RATED)

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

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

MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS τ (GE8X8EB FUEL)

FIGURE 3.2.3-1b



K_f FACTOR

FIGURE 3.2.3-2

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed 13.4 kW/ft for BP/P8X8R fuel and 14.4 kw/ft for GE8X8EB fuel.

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

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGRs shall be determined to be equal to or less than the limit:

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 on a LIMITING CONTROL ROD PATTERN for LHGR.

d. The provisions of Specification 4.0.4 are not applicable.

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		
i. flow biased	≤ 0.66 W + 41%, with a maximum of,	≤ 0.66 W + 44%, with a maximum of,
ii. high flow clamped	≤ 107%	≤ 110%
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	≤ 0.58 W + 50%*	≤ 0.58 W + 53%*
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

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.

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 specified in Reference 2 or that ≥ 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 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 Figure 3.2.3-1.

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.
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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 7 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 April 3, 1987, Philadelphia Electric Company (PECo or the licensee) requested an amendment to Facility Operating License No. NPF-39 for the Limerick Generating Station, Unit 1. The proposed amendment would revise the Technical Specifications (TSs) to: 1) incorporate the operating limits (e.g., core physics, thermal and hydraulic limits, etc.) for all fuel types for Cycle 2 operation of Limerick, Unit 1; 2) incorporate a change in slope of the flow-biased Average Power Range Monitor (APRM) scram and rod block setpoints for operating flexibility or margin improvement options and 3) modify the Bases associated with core reloads.

Limerick, Unit 1 shutdown May 15, 1987 for the first refueling following coastdown. During the refueling, 268 of the 764 fuel assemblies will be replaced with General Electric Company (GE) manufactured 8X8 fuel bundles. The analyses for the reload were performed by GE for PECo using NRC approved procedures and methodologies. The April 3, 1987 submittal (Reference 1) included the necessary changes to the operating limits in the TSs as determined by the analyses for Cycle 2 operation and other supporting documents (References 2, 3 and 4).

The reload for Cycle 2 is generally a normal reload with no unusual core features or characteristics. 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 2 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.

The submittal proposes an extension of the current allowed operating region on the reactor power-flow map via an extended load line analysis (ELLLA). Increased core flow (ICF), final feedwater temperature reduction (FFWTR) and feedwater heaters out service (FHOOS) modes of extended operation have been approved previously (Ref. 5). Except for changes to the flow biased neutron flux scram and rod block setpoints necessary for ELLLA and some changed or additional MCPR limits, these extensions require no other changes to Cycle 2 TS. Revised MCPR limits for operation with inoperable recirculation pump trip (RPT) are also proposed.

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2.0 EVALUATION

2.1 Reload Description

The L1C2 reload will retain 496 BP8x8R GE fuel assemblies from the previous cycle and add 268 new GE8x8EB fuel assemblies. The reload is based on a previous cycle core nominal average exposure of 8.6 GWD/ST and Cycle 2 end of cycle exposure of 16.5 GWD/ST. The loading will be a conventional scatter pattern with low reactivity fuel on the periphery. This loading is acceptable.

2.2 Fuel Design

The new fuel for Cycle 2 is the GE extended burnup fuel GE8x8EB. The fuel designation is BC320A. This fuel type has been approved in the Safety Evaluation Report for Amendment 10 to GESTAR II (Ref. 6 and 7). The specific descriptions of this fuel have been submitted in Amendment 18 to GESTAR II, but since this amendment has not as yet been accepted, the fuel description has also been presented for L1C2 in Reference 3. This fuel description is acceptable.

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. There was extensive interaction between the staff, GE and utilities in deciding on an acceptable format for presentation of this information, suitable for plant use and staff requirements for TS. The process computer contains, and acts on, full details of the MAPLHGR information. The agreed upon TS present the least and most limiting lattice MAPLHGR as a function of burnup. When hand calculations of MAPLHGR are required (process computer inoperative) the most limiting values are used for all limits. These TS are acceptable. A proprietary report, reviewed by the staff, available to the Limerick engineering staff provides complete details of the lattice definitions and MAPLHGR limits.

The proposed LHGR limit for the GE8x8EB fuel is 14.4 kW/ft (rather than the 13.4 for other GE fuel). This LHGR has been reviewed and accepted for this fuel in the GE extended burnup fuel review (Ref. 6). (See the referrals in Reference 6 to References 18 and 19. These references are responses to questions and presentations relating to the GE8x8EB fuel which provide information on the 14.4 kW/ft LHGR.) This LHGR is acceptable for the fuel in L1C2.

2.3 Nuclear Design

The nuclear design for L1C2 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 2.0 and 1.0% delta k at 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 3.6% delta k. Since these and other L1C2 nuclear design parameters have been obtained with previously approved methods and fall within expected ranges, the nuclear design is acceptable.

2.4 Thermal-Hydraulic Design

The thermal-hydraulic design for L1C2 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 L1C2 (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 L1C2 using the standard, approved (Ref. 7) ODYN Option A and B approach for pressurization transients provide new Cycle 2 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 LRWBP is controlling at both option A and B limits. With the selected rod block setting of 107% 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 2 operation.

2.5 Transient and Accident Analyses

The transient and accident analysis methodologies used for L1C2 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 L1C2 are indicated in Section 2.4. 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 107% was selected to provide an OLMCPR of 1.26 for all fuel types. 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 (symetric 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.

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

The L1C2 reload submittal proposes extensions to standard operating regions in the GESTAR II standard category of "Operating Flexibility or Margin Improvement Options". The selected options are ELLLA, ICF, FFWTR, FHOOS and RPT. These have become commonly selected and approved options for a number of reactors in recent years. These options are described and discussed in Appendices A and B of the GE reload report (Ref. 2) and in a GE topical report for Limerick (Ref. 4) on the ELLA extension. The topical report provides generic analyses of transients and accidents. A similar topical report was previously reviewed for the ICF, FFWTR and FHOOS extensions approval (Ref. 5).

The proposed 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. Operation with FHOOS throughout the cycle has previously been approved (Ref. 5). For L1C2 the limiting MCPR is bounded by the limits for ICF plus FFWTR and may be used throughout the cycle, including cycle extension.

For the ELLLA extension the topical report (Ref. 3) discusses a full range of transient and accident events relevant to the region extension, and presents results of calculations or previously approved conclusions. In addition the GE L1C2 reload report (Ref. 2) presents additional calculations of limiting MCPR transients specifically for L1C2. 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 L1C2.

The standard (Ref. 7) relevant limiting transients and resulting OLMCPR values were calculated for L1C2 for normal operations and for appropriate limiting ELLLA conditions. They were also calculated for the previously approved ICF, FFWTR and FHOOS extension conditions. These calculations included the cycle extension conditions. The results provide OLMCPR values for the TS (MCPR vs 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 L1C2 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 L1C2.

2.7 Technical Specifications

The specific changes to the TS for the Limerick Cycle 2 operation and operation in extended power-flow operating regions are listed below. These have been discussed above along with the basis for their acceptability.

(a) The new ELLLA APRM scram and rod block flow dependent setpoints. The changes are to Table 2.2.1-1 of TS 2.2.1, TS 3.2.2 and Table 3.3.6-2 of TS 3.3.6 and are acceptable.

(b) The new MCPR limits for Cycle 2, for extended operation and for the new (GE8x8EB) fuel. The changes are to TS 3.2.3 and Figures 3.2.3-1a and 1b and are acceptable.

(c) The 14.4 KW/ft LHGR limit for the new (GE8x8EB) fuel. The changes are to TS 3.2.4 and are acceptable.

(d) MAPLHGR limits for the new fuel. The changes are to TS 3/4.2.1 and Figure 3.2.1-6 and are acceptable.

(e) The Rod Block Monitor maximum trip setting in Table 3.3.6-2 is set at 107% to correspond to the selection in the RWE analysis. This is acceptable.

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. 8) and are acceptable.

The safety limit MCPR presented in TS 2.1.2 has been increased from 1.06 to 1.07. The latter is the standard safety limit in BWR-4s after the initial cycle of operation. The increased limit provides additional conservatism to account for possible uncertainties in power distribution in reload cores.

There are changes to pages iv, vi and viii of the table of contents to reflect the changes associated with the reload. These are administrative changes that have no safety significance.

There are changes to the Bases for Sections 2 and 3/4 of the TSs. Page B 2-1 is revised to state that the Safety Limit MCPR will be 1.07 rather than the 1.06 that applied to the first fuel cycle. Pages B 2-2, B 2-3 and B 2-4 presently discuss how the Safety Limit MCPR is determined, summarizing information in two GE documents - NEDO-10958-A and NEDO-20340. These documents have been reviewed (and as the "A" indicates) approved by NRC; the results are incorporated in the GE Generic Reload topical report (Reference 8). The latter report will continue to be referenced in the Limerick TS Bases (e.g., page B 2-2), but the discussion on uncertainties in the MCPR analyses, which is redundant to that in the basic GE licensing reload reports, is being deleted from pages B 2-2, B 2-3 and B 2-4.

Pages B 3/4 2-1 through B 3/4 2-5 presently discuss how the MCPR, MAPLHGR and LHGR limits are calculated for Limerick fuel. The Bases presently cite the SLMCPR of 1.06 which was only valid for cycle one. As part of the reload application, PECo submitted GE document NEDE-31401-P dated February 1987, entitled "Bases of MAPLHGR Technical Specifications for Limerick Unit 1" (Reference 3). The references on page B 3/4 2-5 are being revised by this amendment to include the GE report. Much of the discussion presently on pages B 3/4 2-1 through B 3/4 2-3 on input parameters is redundant to or changed by the referenced reports and is being modified or deleted. Our review of the Bases was inherent in our review of the TS limits developed from the Bases.

3.0 SUMMARY

We have reviewed the reports submitted for the Cycle 2 operation of Limerick 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.

4.0 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.

5.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (52 FR 23104) on June 17, 1987 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 security or to the health and safety of the public.

Prinicipal Contributor: H. Richings
R. Clark

Dated: August 14, 1987

6.0 REFERENCES

1. Letter, and enclosures, from E. J. Bradley, PECo, to H. Denton, NRC, dated April 3, 1987. Application requesting changes to the Limerick TS for the first refueling.

2. GE Report 23A5801, dated February 1987, "Supplemental Reload Licensing Submittal for Limerick Generating Station, Unit 1, Reload 1".

3. NEDE-31401, February 1987, "Basis of MAPLHGR TS for Limerick Unit 1".

4. NEDC-31139, April 1986, "GE BWR Extended Load Line Limit Analysis for Limerick Unit 1".

5. Letter (and enclosure) for R. Martin, NRC, to E. Bauer, PECo, dated February 17, 1987, "Technical Specification Changes to Allow Operation with Increased Core Flow and Partial Feedwater Heating".

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 G. Lainas, NRC, to J. Charnley, GE, dated March 22, 1986, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, 'GE Generic Licensing Reload Report', Supplement to Amendment 11".