February 17, 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: TECHNICAL SPECIFICATION CHANGES TO ALLOW OPERATION WITH INCREASED CORE FLOW AND PARTIAL FEEDWATER HEATING

RE: LIMERICK GENERATING STATION, UNIT 1

The Commission has issued the enclosed Amendment No. 3 to Facility Operating License No. NPF-39 for the Limerick Generating Station, Unit 1. This amendment consists of changes to the Technical Specifications (TSs) and deletion of a license condition in response to your application dated November 17, 1986 as amended on December 22, 1986 and as supplemented on January 2 and January 29, 1987.

This amendment changes Technical Specifications to allow plant operation with partial feedwater heating and with increased reactor core cooling water flow rates up to 105% of rated flow. This amendment also deletes License Condition 2.C(13) which prohibited operation with partial feedwater heating for the purpose of extending the normal fuel cycle. This amendment does not authorize an increase in the licensed power level of 3293 MWt. Technical Specification changes include changes to the Minimum Critical Power Ratio limits, the control rod block instrumentation setpoints and the reactor coolant system surveillance requirements.

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

Sincerely,

/s/
Robert E. Martin, Project Manager
BWR Project Directorate No. 4
Division of BWR Licensing

Enclosures:
1. Amendment No. 3 to License No. NPF-39
2. Safety Evaluation

cc w/enclosures:
See next page

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PD#4/LA	PD#4/PM	FOB/D	OGC	PD#4/D
MO'Brien	RMartin:lb	DVassallo	Vogler	WButler
2/11/87	2/11/87	/ /87	2/13/87	2/17/87

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

February 17, 1987

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See next page

Mr. Edward G. Bauer, Jr
Philadelphia Electric Company

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Units 1 & 2

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

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 3
License No. NPF-39

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

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated November 17, 1986 as amended on December 22, 1986 and as supplemented on January 2 and January 29, 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. 3 , 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. The license is further amended by deleting paragraph 2.C(13).

4. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/ Robert E. Martin for
Walter R. Butler, Director
BWR Project Directorate No. 4
Division of BWR Licensing

Attachments:
1. Change to the License
2. Changes to the Technical
   Specifications

Date of Issuance: February 17, 1987

PD#4/LA
MO'Brien
2/11/87

PD#4/PM
RMartin:lb
2/11/87

OGC
B. VOGLER
2/13/87

PD#4/D
WButler
2/17/87

3. The license is further amended by deleting paragraph 2.C(13).

4. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert E Martin
for Walter R. Butler, Director
BWR Project Directorate No. 4
Division of BWR Licensing

Attachments:
1. Change to the License
2. Changes to the Technical Specifications

Date of Issuance: February 17, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 3

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

1. Revised Page 6 of Facility Operating License No. NPF-39

2. 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 page(s) provided to maintain document completeness.*

Remove	Insert
v	v*
vi	vi
3/4 2-7	3/4 2-7*
3/4 2-8	3/4 2-8
--	3/4 2-8a
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
--	3/4 2-10a
3/4 3-59	3/4 3-59*
3/4 3-60	3/4 3-60
--	3/4 3-60a
3/4 4-1	3/4 4-1*
3/4 4-2	3/4 4-2
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

(10) Reactor Enclosure Cooling Water and Chilled Water Isolation Valves (Section 6.2.4.2, SER and SSER-3)

The licensee shall, prior to startup following the first refueling outage, provide automatic and diverse isolation signals to the reactor enclosure cooling water inboard and outboard isolation valves in the supply and return lines to the recirculation pumps and the drywell chilled water outboard isolation valves in the supply and return lines.

(11) Hydrogen Recombiner Isolation (Section 6.2.4.2, SER and SSER-1 and SSER-3)

The licensee shall, prior to startup following the first refueling outage, install and test an additional automatic isolation valve in each of the hydrogen recombiner lines penetrating the primary containment.

(12) Remote Shutdown System (Sections 7.1.4.4, 7.4.2.3, SER and Section 7.4.2.3, SSER-3 and SSER-5)

The licensee shall, prior to startup following the first refueling outage, have completed modifications to the existing remote shutdown system to provide a redundant safety-related method of achieving safe shutdown conditions without lifting leads or adding jumpers.

The modifications to be completed shall be those described in the licensee's letters dated April 18 and 22, 1985 which allow for the operation of the B RHR pump, the B RHR SW pump and the B ESW pump from the respective pump breaker compartments by the installation of transfer switches. The licensee shall perform necessary tests prior to startup following the first refueling outage to demonstrate the operability of the modified system.

(13) (Deleted)

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

POWER DISTRIBUTION LIMITS

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 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
$S \leq (0.66W + 51\%)T$	$S \leq (0.66W + 54\%)T$
$S_{RB} \leq (0.66W + 42\%)T$	$S_{RB} \leq (0.66W + 45\%)T$

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 determined using the appropriate figure taken from Table 3.2.3-1, 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,

$$\tau_B = 0.688 + 1.65\left[\frac{N_1}{\sum_{i=1}^{n} N_i}\right]^{\frac{1}{2}}(0.052),$$

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

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

N_i = number of active control rods measured in the 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.

TABLE 3.2.3-1

Minimum Critical Power Ratio (MCPR)
Versus Plant Operating Condition

Rated Feedwater Temperature Reduction From the Nominal, delta T* (°F)	Maximum Core Flow (% of rated)	MCPR Figure #
0	$\leq$ 100	3.2.3-1a
$\leq$ 60	$\leq$ 105	3.2.3-1b

*This delta T refers to the planned reduction of feedwater temperature at rated conditions from nominal rated feedwater temperature during the prolonged removal of feedwater heaters from service.

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 the appropriate figure taken from Table 3.2.3-1 for EOC-RPT inoperable curve times the K_f shown in Figure 3.2.3-2.

b. With MCPR less than the applicable MCPR limit as identified in ACTION a above, 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,

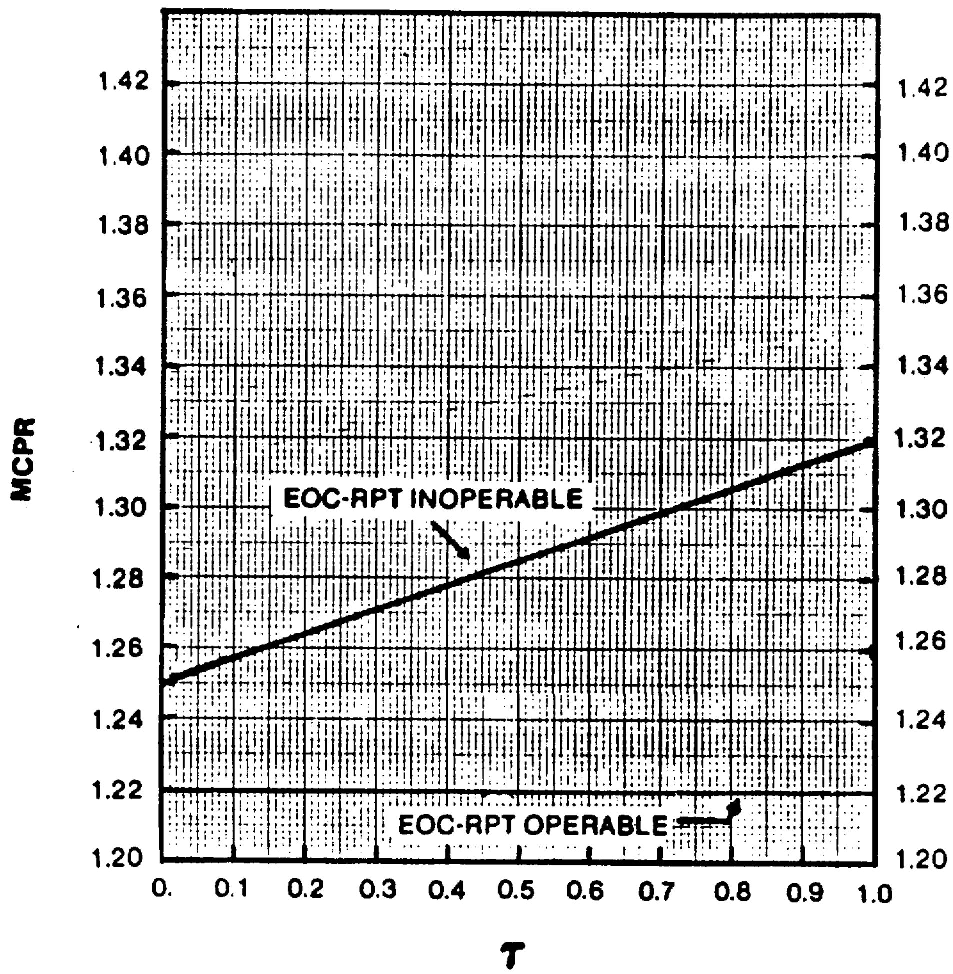
shall be determined to be equal to or greater than the applicable MCPR limit determined from the appropriate figure taken from Table 3.2.3-1 times the K_f shown in Figure 3.2.3-2.

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

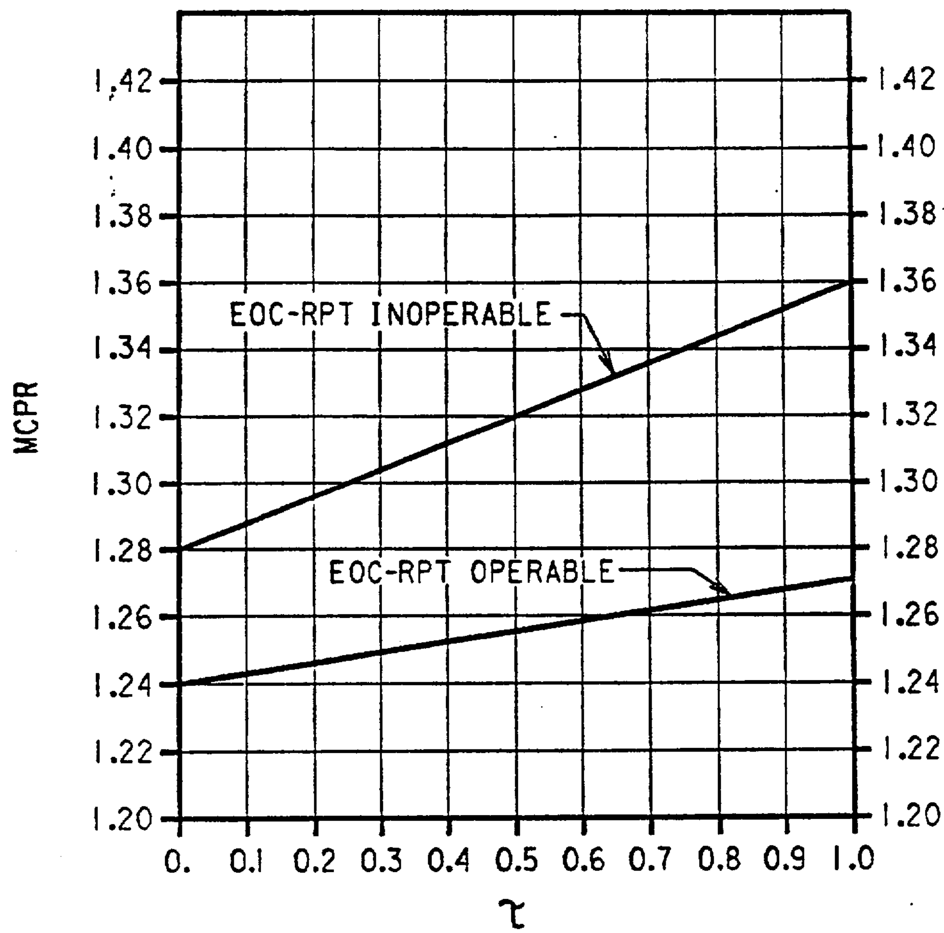
c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

d. The provisions of Specification 4.0.4 are not applicable.



MINIMUM CRITICAL POWER RATIO (MCPR)
VERSUS τ AT MAXIMUM CORE
FLOW ≤ 100% RATED
(RATED FEEDWATER TEMPERATURE)

FIGURE 3.2.3-1a



MINIMUM CRITICAL POWER RATIO (MCPR)
VERSUS τ AT MAXIMUM CORE FLOW $\leq$ 105% RATED
AND
MAXIMUM FEEDWATER TEMPERATURE REDUCTION $\leq$ 60°F
AT RATED CONDITIONS

FIGURE 3.2.3-1b

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.

(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 + 40%, with a maximum of,	≤ 0.66 W + 43%, with a maximum of,
ii. high flow clamped	≤ 106%	≤ 109%
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - Upscale	≤ 0.66 W + 42%*	≤ 0.66 W + 45%*
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 x 10^5 cps	≤ 1.6 x 10^5 cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 3 cps**	≥ 1.8 cps**
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High		
a. Float Switch	≤ 257' 5 9/16" elevation***	≤ 257' 7 9/16" elevation

TABLE 3.3.6-2 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

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

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

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

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

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation with:

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

b. THERMAL POWER less than or equal to the limit specified in Figure 3.4.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours and initiate measures to place the unit in at least HOT SHUTDOWN within 12 hours.

b. With no reactor coolant system recirculation loops in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours and initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

c. With two reactor coolant system recirculation loops in operation and total core flow less than 45% of rated core flow and THERMAL POWER greater than the limit specified in Figure 3.4.1.1-1:

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

      a) At least once per 8 hours, and

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

   2. With the APRM or LPRM** neutron flux noise levels greater than three times their established baseline noise levels, immediately initiate corrective action to restore the noise levels to within the required limits within 2 hours by increasing core flow to greater than 45% of rated core flow or by reducing THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1.

---

*See Special Test Exception 3.10.4.

**Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 Each pump discharge valve shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each startup* prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.

4.4.1.1.2 Each pump MG set scoop tube mechanical and electrical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 109% and 107%, respectively, of rated core flow, at least once per 18 months.

4.4.1.1.3 Establish a baseline APRM and LPRM** neutron flux noise value within the regions for which monitoring is required (Specification 3.4.1.1, ACTION c) within 2 hours of entering the region for which monitoring is required unless baselining has previously been performed in the region since the last refueling outage.

*If not performed within the previous 31 days.

**Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

BASES TABLE B 3/4.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters:

Core THERMAL POWER .... 3430 MWt* which corresponds to 105% of rated steam flow

Vessel Steam Output .... 14.86×10^6 lbm/h which corresponds to 105% of rated steam flow

Vessel Steam Dome Pressure.............. 1055 psia

Design Basis Recirculation Line Break Area for:

a. Large Breaks $4.1\ ft^2$, $1.0\ ft^2$

b. Small Breaks $1.0\ ft^2$, $0.07\ ft^2$, $0.09\ ft^2$, $0.02\ ft^2$

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kW/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	8 x 8	13.4	1.4	1.20

A more detailed listing of input of each model and its source is presented in Section II of Reference 1 and subsection 15.0.2 of the FSAR.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

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 of 1.06, 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 of 1.06, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and presented in Figures 3.2.3-1a and 3.2.3-1b.

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 code used to evaluate pressurization events is described in NEDO-24154[3] and the program used in non-pressurization events is described in NEDO-10802[2]. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic TASC code described in NEDE-25149[4]. The principal result of this evaluation is the reduction in MCPR caused by the transient.

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. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, NEDO-10802, February 1973.
3. Qualification of the One Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154, October 1978.
4. TASC 01-A Computer Program for the Transient Analysis of a Single Channel, Technical Description, NEDE-25149, January 1980.
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. 3 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 November 17, 1986 as amended on December 22, 1986 and as supplemented on January 2 and 29, 1987, Philadelphia Electric Company (the licensee) requested an amendment to Facility Operating License No. NPF-39 for the Limerick Generating Station (LGS), Unit 1. The proposed amendment would change the Technical Specifications (TS) to permit operation of the unit with partial feedwater heating (PFH) and increased core flow (ICF) limits and would delete License Condition 2.C(13) which presently prohibits the use of PFH. Specifically, TS 3/4.2.3 "Minimum Critical Power Ratio (MCPR)," TS Table 3.3.6-2 "Control Rod Block Instrumentation Setpoints," and TS 4.4.1.1.2 "Reactor Coolant System Surveillance Requirements" would be revised to permit operation of Unit 1 with a reduction of incoming feedwater temperature (partial feedwater heating, PFH) of up to 60°F and an increase in reactor core flow rate up to 105% of rated flow. License Condition 2.C(13), "Operation With Partial Feedwater Heating at End-of-Cycle" would be satisfied since the basis for the condition, namely that the applicable safety analyses to permit operation with PFH had not been performed, has been satisfied by the submittal of such analyses by the licensee. Near the end of a fuel cycle the depletion of fissionable material from prior power production results in a condition wherein the 100% of rated power condition (3293 MWt) can no longer be maintained. From this point, as the licensee states, a "coastdown mode" of operation with all power control rods "full out" may be followed to extend the fuel cycle. This would result in a gradually decreasing rate of power production until an optimum point is reached at which the licensee would remove the unit from service for refueling. The changes approved by this amendment, partial feedwater heating and increased core flow, take advantage of the boiling water reactor operating characteristics to allow an extension of the fuel cycle. The partial feedwater heating provisions also provide increased operational capability by providing for operations with some of the feedwater heaters removed from service.

As support for the proposed modifications, the licensee provided a General Electric Company report, NEDC-31323, "Increased Core Flow and Partial Feedwater Heating Analysis for Limerick Generating Station Unit 1 Cycle 1" dated October 1986. In response to staff requests for information on this report and on the licensee's application, the licensee submitted additional information by letters dated December 22, 1986, January 2, 1987 and January 29, 1987.

2.0 EVALUATION

License condition 2.C(13) for LGS contains, in part, a requirement that analyses of operation in the partial feedwater heating (PFH) mode must be provided to the staff for review and approval prior to operation in that mode. The analyses for PFH and increased core flow (ICF) were provided by the licensee in Reference 1. These analyses provide the required minimum critical power ratio (MCPR) operating limits for the proposed operation of LGS up to a maximum feedwater temperature reduction of less than or equal to 60° F at rated power and up to a maximum core flow of 105% of rated flow. Certain abnormal transients and accidents analyzed in the LGS FSAR have been examined for the effect of the proposed operational mode. The staff's evaluation of these considerations is discussed below.

2.1 Anticipated Operational Occurrences

The most limiting anticipated operational occurrences are:

a) Generator Load Rejection with Bypass Failure (LRNBP)

b) Feedwater Controller Failure (FWCF)

The evaluations were performed at 104.5% power (consistent with the original FSAR analysis input assumptions), 105% core flow with rated feedwater temperature of 360° F at end of cycle (EOC1). Nuclear transient data consistent with the original FSAR analysis were developed and the combination of the extended load line limit analysis (ELLLA) and PFH were incorporated in the analyses. Based on the limiting transients identified by previous analyses, the proposed MCPR operating limits are developed to include the cases of turbine bypass inoperable and end-of-cycle recirculation pump trip inoperable. The new required operating limit MCPRs shall be 1.24 and 1.28 (based on ODYN Option B results for the feedwater controller failure without bypass transient with reactor pump trip and without reactor pump trip, respectively) for a maximum core flow 105% of rated and a maximum feedwater reduction of 60° F. The new calculated operating limit MCPR values are incorporated into the proposed technical specifications.

Lower initial operating pressure and steam flow rate (due to lower feedwater temperature) provide more overpressure margin for the limiting MSIV closure flux scram event. Hence, it is concluded that the pressure barrier integrity is maintained under PFH conditions. The licensee has analyzed the overpressurization limiting transient (MSIV closure) for increased core flow (ICF) without PFH. The analysis of this bounding transient predicted a peak vessel pressure of 1273 psig which is below the ASME code limit of 1375 psig and the analysis results are therefore acceptable.

The fuel loading error accident, rod drop accident, and rod withdrawal error have been evaluated by the licensee for ICF and/or PFH operation. The rod withdrawal error transient is limited by a rod block system. The addition of a "high flow clamped" trip setpoint limit of 106 percent and allowable value of 109 percent of rated flow for the rod block monitor upscale alarm in TS Table 3.3.6-2 ensures that the rod blocks currently in the TS cannot be exceeded. This is the same requirement that has been in effect since initial plant operation. The reactor coolant system recirculation flow upscale trip setpoints and allowable values are increased and the values for the recirculation pump motor-generator (MG) set scoop tube mechanical and electrical stops are increased. These changes are necessary to accommodate the increased core flow operation and are acceptable. The licensee has stated that the fuel loading error and rod drop accident are not adversely affected by the proposed changes. For the fuel loading error, the licensee has reported in Reference 3 a maximum increase in CPR of 0.04 from the value of 0.11 stated in the FSAR for this event at rated conditions. Thus the fuel loading error remains a non-limiting event. With regard to the rod drop accident, the LGS utilizes a banked position withdrawal sequence (BPWS) for control rod movement. Based on prior staff review of BPWS as presented by General Electric (Ref. 5, Section S.2.5.1.3), the staff agrees that this event is not adversely affected by the proposed changes.

2.2 Loss of Coolant Accident Analysis

A loss of coolant accident (LOCA) with ICF and PFH was addressed by the licensee in Reference 2. The LOCA analyses with ICF alone bound operation with ICF and PFH. Since the peak clad temperature for ICF increases by less than 10° F for the limiting break compared to the rated core flow condition, the calculated peak clad temperature (PCT) of approximately 2100° F remains below the 10 CFR 50.46 cladding temperature limit. No changes to the current maximum average planar linear heat generation rates (MAPLHGR) are required. In Reference 2, GE stated that PCT changes throughout the remainder of the large break spectrum will be of a similar magnitude (less than 10° F). At the request of the staff, the licensee provided additional information (References 4 and 8) on the effect of increased core flow (ICF) and reduced feedwater temperature on the LGS LOCA analysis. Consideration was given to the break spectrum range of 60 to 100 percent of the design basis accident (DBA) for the separate effect of ICF for several classes of BWR plants with the resulting conclusion that increased core flow results in a peak clad temperature increase of less than 10 degrees F throughout the large break spectrum. The separate effect of reduced feedwater temperature is to reduce the calculated peak clad temperature. A discussion was presented for both reduced feedwater temperature and increased core flow conditions which bounds the conditions described in the proposed amendment. Based on the staff's review of the additional information provided by the licensee in References 4 and 8, which discusses the LGS specific LOCA analyses, the staff agrees with the conclusion in NEDC-31323 that the effect of ICF will not alter the limiting break size. The calculated peak clad temperature remains below the 10 CFR 50.46 cladding temperature limit and is acceptable.

The impact of the proposed operating mode on containment LOCA response was considered by the licensee. A conservative analysis resulted in a peak drywell deck downward differential pressure 2.6 psi higher than the value of

26.0 psid in the LGS FSAR. However, this is still below the design limit of 30.0 psid reported in the FSAR . The licensee stated that the drywell and suppression chamber temperatures, external pressures and maximum allowable leakage rates are bounded by the results reported in the FSAR. It was also stated that the chugging loads, condensation oscillations and pool swell loads were found to be bounded by the appropriate design loads. We find this acceptable.

2.3 Thermal-Hydraulic Stability

Reference 2 included a discussion of thermal-hydraulic stability (THS) for the LGS. The current LGS technical specifications implement a generic set of operating recommendations (Ref. 6) to assure acceptable plant performance in the least stable portion of the power/flow map and to provide operator instructions for the detect-and-suppress mode of operation. The THS compliance for all licensed GE BWR core fuel is demonstrated on a generic basis by Reference 7 and has been approved by the staff (NRC Safety Evaluation Report Approving Amendment B to NEDE-24011-P contained in Appendix US-C to Reference 5). The staff concludes that acceptable THS provisions have been made to cover the proposed modifications.

2.4 Flow Effects

NEDC-31323 (Reference 2) presents the results of a safety and impact evaluation of the limiting normal operational transients, loss-of-coolant accidents, fuel loading error accidents, rod drop accidents, and rod withdrawal error events. In addition, the effect of increased pressure differences on the reactor internals components, fuel channels and fuel bundles was also analyzed to show that the design limits will not be exceeded. The effect of ICF on the flow-induced vibration response of the reactor internals was also evaluated to ensure that the response is within acceptance limits. The thermal-hydraulic stability was evaluated for ICF/PFH operation, and the increase in the feedwater nozzle and feedwater sparger usage factors due to the feedwater temperature reduction was determined. The impact of ICF/PFH operation on the containment LOCA response was also analyzed.

This evaluation in section 2.4 of this safety evaluation addresses only those portions of Section 3.1, 3.2.1, 4 and 5 of Reference 2 which are pertinent to load impact of reactor internals flow-induced vibration and feedwater nozzle and feedwater sparger fatigue usage. Subsequent to its review of Reference 2, the staff requested clarification from the licensee regarding those three areas. In response to the staff's request, the licensee submitted a letter from J. W. Gallagher (PECo) to W. R. Butler (NRC) dated January 29, 1987 (Reference 8).

2.4.1 Reactor Internals Load Impact

All the reactor internals (e.g., core plate, shroud support, shroud, top guide, shroud head, steam dryer, control rod guide tube, control rod drive housing and jet pump) were evaluated under the consideration of additional loads imposed by the ICF and PFH operations. The conclusion, as stated in NEDC-31323, is that the stresses produced in those components are within the ASME Code, Section III, Subsection NG allowables, the criteria referenced in the FSAR. Based on the reported results, the staff finds this acceptable since the FSAR design limits were satisfied.

2.4.2 Flow-Induced Vibration

To ensure that the flow-induced vibration response of the reactor internals for plant operation with ICF up to 105% rated flow is acceptable, the prototype (Browns Ferry 1) plant test data were used as the bases for this vibration assessment. The Browns Ferry 1 test results are described in NEDE-24057-P-A which was accepted by the staff in a letter from R. Tedesco (NRC) to G. Sherwood (GE) dated October 28, 1980. Results from the Browns Ferry 1 test program include data up to 113% core flow. Test measurements of all sensors in all instrumented reactor internal components were reviewed, evaluated and compared with the acceptance criteria. The absolute sum of the peak alternating stresses of all the vibration modes was obtained. Using this method the maximum vibration level of 61% of the acceptance criteria was determined from a jet pump strain gauge at 113% of rated flow. Hence, the data showed that reactor internals response to flow induced vibration is within acceptable limits up to 113% core flow. As shown in NEDE-24057-P-A, Fitzpatrick is the only plant with instrumented fuel channels. Since the Limerick rated flow per fuel bundle is less than the Fitzpatrick rated flow per fuel bundle, the Fitzpatrick fuel channel test results can be applied to Limerick. An assessment based on Fitzpatrick test data described in NEDE-24057-P-A shows that the maximum recorded vibration of the fuel channels was less than 2% of the allowable for conditions corresponding to at least 128% of rated flow. Therefore, the operation of Limerick Unit 1 at 105% of rated core flow will not result in unacceptable fuel rod or fuel channel vibration.

2.4.3 Feedwater Nozzle and Feedwater Sparger Fatigue Usage

At the end of the 1970's, inspections at 22 of 23 boiling water reactor plants identified cracking in the feedwater nozzle and sparger at 18 reactor vessels. The NRC staff studied the issue and recommended hardware modifications, analysis methods, and inspection schedules for nozzles and spargers in NUREG-0619. Partial feedwater heating will affect the fatigue usage of the feedwater nozzle and sparger. The staff reviewed this request using the guidelines described in NUREG-0619 and the associated Generic Letter 81-11.

The licensee uses a GE designed triple thermal sleeve sparger to prevent the thermal cycling phenomena, thus reducing the likelihood of crack initiation at the feedwater nozzle. One end of the sparger consists of the three concentric sleeves with two piston ring seals that are fitted to the nozzle safe end, and the other end of the sparger consists of the arms that run along the vessel wall. The first seal has a clearance fit with the nozzle safe end and forms the primary seal between the innermost sleeve and the nozzle bore. The innermost sleeve conducts feedwater from the nozzle to the sparger arms. The primary seal prevents the mixing of relatively cooler feedwater with the hotter reactor coolant. Attached to the middle sleeve is an outer sleeve which is fitted tightly in the nozzle bore. The secondary seal at that tight interference joint reduces potential bypass flow. The staff's original concern about this GE design, as expressed in NUREG-0619, is that wear or corrosion would eventually reduce the sealing ability. GE also mentioned that corrosion of carbon steel safe-ends under the piston-ring seal posed a potential problem.

The use of PFH changes the number of cycles of reactor thermal transients, specifically rapid cycling. The rapid cycling is caused by small high frequency temperature changes by mixing of reactor coolant with colder incoming feedwater at the nozzle annulus. Because of PFH, the feedwater temperature will be lower than the original design basis and this temperature reduction will increase fatigue usage due to an increase in thermal stresses. The licensee studied two cases of PFH that affect the fatigue usage of the feedwater sparger and nozzle: the final feedwater temperature reduction (FFWTR) and feedwater heaters out-of-service (FWHOS). Thermal stresses due to temperature differentials are calculated using the conduction and convention heat transfer method and stress analysis. Fatigue usage is calculated by dividing the total number of cycles corresponding to thermal stresses by the number of ASME Code allowable cycles. The total fatigue factor is the sum of all of the fatigue usage factors for all transients. This analysis method was described in GE report NEDE 21821-02 and was approved by the staff in NUREG-0619. The licensee has shown that the total fatigue usage factor for the feedwater nozzle and sparger for the FFWTR Case and FWHOS Case can be kept below the required value of 1.0 by seal refurbishment after a 28 year period, based on a postulated number of thermal cycles. Although the refurbishment interval would be reduced, only one refurbishment would be required, as was the case for operation without PFH and ICF. We find this to be acceptable.

2.5 Technical Specification Changes

The proposed technical specification changes deal with the MCPR operating limits and certain trip setpoints which are identified below:

(a) Minimum Critical Power Ratio

As discussed in Section 2.1 of this Safety Evaluation (SE), changes to the limiting conditions of operation (LCO) are identified for the proposed operational mode. Based on the staff's review, the operating limit MCPRs of 1.24 and 1.28 are found acceptable. The changes are contained on technical specification (TS) pages 3/4 2-8, 3/4 2-9, and 3/4 2-10; TS pages 2/4 2-8a and 3/4 2-10a are added to reflect the modified operation mode.

(b) Instrumentation Setpoints

As discussed in Section 2.1 of this SE, changes to the values for the recirculation pump MG set scoop tube mechanical and electrical stops and the control rod block recirculation flow upscale trip setpoints and allowable values are made to accommodate ICF operation. Also, the rod block monitor upscale trip setpoint and allowable values are changed to accommodate the addition of a high flow-clamped rod block. Based on the results of our review the staff finds the proposed changes acceptable. The changes are contained on TS pages 3/4 4-2, 3/4 3-60a and 3/4 3-60, respectively.

(c) Administrative Changes

The index was updated to reflect the additional pages and the General Electric analyses document (Ref. 2) was added as a reference. A reference to Figure 3.2.3-1 on TS Basis page B 3/4 2-4 has been changed to reflect the division of Figure 3.2.3-1 into Figures 3.2.3-1a and 3.2.3-1b.

2.6 Conclusion

The NRC staff has reviewed the information provided by the Philadelphia Electric Company relative to the proposed license amendment to allow operation of the Limerick Generating Station Unit 1 with partial feedwater heating and increased core flow. Based on the results of the evaluation contained in this section the staff concludes that the proposed technical specification changes are acceptable.

3.0 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.

4.0 CONCLUSION

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

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Dated: February 17, 1987

REFERENCES

1. Letter, E. J. Bradley (PECo) to H. R. Denton (NRC) dated November 17, 1986 transmitting application for Amendment to Facility Operating License NPF-39.

2. NEDC-31323, "Increased Core Flow and Partial Feedwater Heating Analysis for Limerick Generating Station Unit 1 Cycle 1", dated October 1986.

3. Letter, J. W. Gallagher(PECo) to W. R. Butler(NRC) dated December 22, 1986.

4. Letter, M. J. Cooney(PECo) to W. R. Butler(NRC) dated January 2, 1987.

5. "General Electric Standard Application for Reactor Fuel (Supplement for US)", May 1986 (NEDE-24011-P-A-8-US, as amended).

6. General Electric Service Information Letter No. 380, Revision 1, February 10, 1984.

7. G. A. Watford, "Compliance of the General Electric Boiling Water Reactor Fuel Designs to Stability Licensing Criteria," October 1984 (NEDE-22277-P-1).

8. Letter, J. W. Gallagher (PECo) to W. R. Butler (NRC) dated January 29, 1987.