July 30, 1990

Docket Nos. 50-352
and 50-353

Mr. George A. Hunger, Jr.
Director-Licensing, MC 5-2A-5
Philadelphia Electric Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
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GSuh	Wanda Jones	
MO'Brien	SNewberry,C/SICB	

Dear Mr. Hunger:

SUBJECT: APRM OPERABILITY DURING OPCON 5 (TSCR NO. 90-02-0), LIMERICK GENERATING STATION, UNITS 1 AND 2 (TAC NOS. 76958/76959)

The Commission has issued the enclosed Amendment No. 41 to Facility Operating License No. NPF-39 and Amendment No. 7 to Facility Operating License No. NPF-85 for the Limerick Generating Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated June 14, 1990.

These amendments revise the TSs to remove the operability requirements for the Average Power Range Monitors (APRMs) in Operational Condition 5 (Refueling) except while performing a shutdown margin demonstration.

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

Sincerely,

Original signed by
Richard J. Clark

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

Enclosures:
1. Amendment No. 41 to License No. NPF-39
   Amendment No. 7 to License No. NPF-85
2. Safety Evaluation

cc w/enclosures:
See next page

CP-1

[LI LETTER]

PDI-2/LA	PDI-2/PE	PDI-2/PM	OGC	PDI-2/D	C/SICB
MO'Brien:tr	SDembek	RClark	RBachmann	WButler	SNewberry
7/11/90	7/10/90	07/09/90	7/17/90	7/30/90	7/13/90

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

Limerick Generating Station
Units 1 & 2

cc:

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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. 41
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 June 14, 1990, 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. 41 , are hereby incorporated into this license. Philadelphia Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

ORIGINAL SIGNED BY:

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance:
July 30, 1990

PDI-2/LA	PDI-2/PE	PDI-2/PM	OGC	PDI-2/D
MO'Brien:tr	SDembek	RClark	RBachmann	WButler
7/11/90	7/10/90	07/09/90	7/17/90	7/30/90

ATTACHMENT TO LICENSE AMENDMENT NO. 41

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

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

Remove	Insert
3/4 3-1	3/4 3-1*
3/4 3-2	3/4 3-2
3/4 3-5	3/4 3-5
3/4 3-6	3/4 3-6*
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
3/4 3-57	3/4 3-57*
3/4 3-58	3/4 3-58
3/4 3-59	3/4 3-59
3/4 3-60	3/4 3-60*
3/4 3-61	3/4 3-61
3/4 3-62	3/4 3-62

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within 1 hour. The provisions of Specification 3.0.4 are not applicable.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within 1 hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

---

*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

**The trip system need not be placed in the tripped condition if this would cause the Trip Function to occur. When a trip system can be placed in the tripped condition without causing the Trip Function to occur, place the trip system with the most inoperable channels in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors[b]:			
a. Neutron Flux - High	2	3	1
	3, 4	3	2
	5(c)	3(d)	3
b. Inoperative	2	3	1
	3, 4	3	2
	5	3(d)	3
2. Average Power Range Monitor[e]:			
a. Neutron Flux - Upscale, Setdown	2	2	1
	3	2	2
	5(c)(l)	2(d)	3
b. Neutron Flux - Upscale			
1) Flow Biased	1	2	4
2) High Flow Clamped	1	2	4
c. Inoperative	1, 2	2	1
	3	2	2
	5(c)(l)	2(d)	3
d. Downscale	1(g)	2	4
3. Reactor Vessel Steam Dome Pressure - High	1, 2(f)	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	1(g)	1/valve	4

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) This function shall be automatically bypassed when the reactor mode switch is in the Run position and the associated APRM is not downscale.

(c) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations performed per Specification 3.10.3.

(d) The noncoincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 4 APRMs, 6 IRMs and 2 SRMs.

(e) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(g) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) This function shall be automatically bypassed when turbine first stage pressure is equivalent to a THERMAL POWER of less than 30% of RATED THERMAL POWER.

(k) Also actuates the EOC-RPT system.

(l) Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	N.A.
b. Inoperative	N.A.
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, Setdown	N.A.
b. Neutron Flux - Upscale	
1) Flow Biased	≤0.09
2) High Flow Clamped	≤0.09
c. Inoperative	N.A.
d. Downscale	N.A.
3. Reactor Vessel Steam Dome Pressure - High	≤ 0.55
4. Reactor Vessel Water Level - Low, Level 3	≤ 1.05
5. Main Steam Line Isolation Valve - Closure	≤ 0.06
6. Main Steam Line Radiation - High	N.A.
7. Drywell Pressure - High	N.A.
8. Scram Discharge Volume Water Level - High	
a. Level Transmitter	N.A.
b. Float Switch	N.A.
9. Turbine Stop Valve - Closure	≤ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≤ 0.08**
11. Reactor Mode Switch Shutdown Position	N.A.
12. Manual Scram	N.A.

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.
**Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S(b)	S/U(c), W	R	2
	S	W(j)	R	3, 4, 5
b. Inoperative	N.A.	W(j)	N.A.	2, 3, 4, 5
2. Average Power Range Monitor[f]:				
a. Neutron Flux -	S/U,S(b)	S/U(c), W	SA	2
Upscale, Setdown	S	W(j)	SA	3, 5[k]
b. Neutron Flux - Upscale				
1) Flow Biased	S,D(g)	S/U(c), W	W(d)(e), SA	1
2) High Flow Clamped	S	S/U(c), W	W(d)(e), SA	1
c. Inoperative	N.A.	W(j)	N.A.	1, 2, 3, 5[k]
d. Downscale	S	W	SA	1
3. Reactor Vessel Steam Dome Pressure - High	S	M	R	1, 2(h)
4. Reactor Vessel Water Level - Low, Level 3	S	M	R	1, 2
5. Main Steam Line Isolation Valve - Closure	N.A.	M	R	1
6. Main Steam Line Radiation - High	S	M	R	1, 2(h)
7. Drywell Pressure - High	S	M	R	1, 2
8. Scram Discharge Volume Water Level - High				
a. Level Transmitter	S	M	R	1, 2, 5[i]
b. Float Switch	N.A.	M	R	1, 2, 5[i]

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Turbine Stop Valve - Closure	N.A.	M	R	1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	N.A.	M	R	1
11. Reactor Mode Switch Shutdown Position	N.A.	R	N.A.	1, 2, 3, 4, 5
12. Manual Scram	N.A.	M	N.A.	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least ½ decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least ½ decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Within 24 hours prior to startup, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.
(g) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing loop flow (APRM % flow). During the startup test program, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the conclusion of the startup test program.
(h) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
(j) If the RPS shorting links are required to be removed per Specification 3.9.2, they may be reinstalled for up to 2 hours for required surveillance. During this time, CORE ALTERATIONS shall be suspended, and no control rod shall be moved from its existing position.
(k) Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2..

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

TABLE 3.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD BLOCK MONITOR[a]			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. APRM			
a. Flow Biased Neutron Flux - Upscale	4	1	61
b. Inoperative	4	1, 2, 5[f]	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Startup	4	2, 5[f]	61
3. SOURCE RANGE MONITORS ***			
a. Detector not full in[b]	3	2	61
	2	5	61
b. Upscale[c]	3	2	61
	2	5	61
c. Inoperative[c]	3	2	61
	2	5	61
d. Downscale[d]	3	2	61
	2	5	61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale[e]	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5**	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. Comparator	2	1	62
7. REACTOR MODE SWITCH SHUTDOWN POSITION	2	3, 4	63

TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION STATEMENTS

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

ACTION 61 - With the number of OPERABLE channels one or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

ACTION 63 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, initiate a rod block.

NOTES

* With THERMAL POWER $\geq$ 30% of RATED THERMAL POWER.

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

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

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

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

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

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

(e) This function is automatically bypassed when the IRM channels are on range 1.

(f) Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR***		
a. Upscale		
1) During two recirculation loop operation		
a. Flow Biased*	≤ 0.66 W + (N-66)%, with a maximum of,	≤ 0.66 W + (N-63)%, with a maximum of,
b. High Flow Clamped	≤ N%	≤ (N+3)%
2) During single recirculation loop operation		
a. Flow Biased*	≤ 0.66 W + (N-72)%, with a maximum of,	≤ 0.66 W + (N-69)%, with a maximum of,
b. High Flow Clamped	≤ N%	≤ (N+3)%
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - Upscale		
1) During two recirculation loop operation	≤ 0.58 W + 50%*	≤ 0.58 W + 53%*
2) During single recirculation loop operation	≤ 0.58 W + 45%*	≤ 0.58 W + 48%*
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 4% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 1 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 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	N.A.	S/U[b][c],M[c]	SA	1*
b. Inoperative	N.A.	S/U[b][c],M[c]	N.A.	1*
c. Downscale	N.A.	S/U[b][c],M[c]	SA	1*
2. APRM				
a. Flow Biased Neutron Flux - Upscale	N.A.	S/U[b],M	SA	1
b. Inoperative	N.A.	S/U[b],M	N.A.	1, 2, 5***
c. Downscale	N.A.	S/U[b],M	SA	1
d. Neutron Flux - Upscale, Startup	N.A.	S/U[b],M	SA	2, 5***
3. SOURCE RANGE MONITORS				
a. Detector not full in	N.A.	S/U[b],W	N.A.	2, 5
b. Upscale	N.A.	S/U[b],W	SA	2, 5
c. Inoperative	N.A.	S/U[b],W	N.A.	2, 5
d. Downscale	N.A.	S/U[b],W	SA	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	N.A.	S/U[b],W	N.A.	2, 5
b. Upscale	N.A.	S/U[b],W	SA	2, 5
c. Inoperative	N.A.	S/U[b],W	N.A.	2, 5
d. Downscale	N.A.	S/U[b],W	SA	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	N.A.	M	R	1, 2, 5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	N.A.	S/U[b],M	SA	1
b. Inoperative	N.A.	S/U[b],M	N.A.	1
c. Comparator	N.A.	S/U[b],M	SA	1
7. REACTOR MODE SWITCH SHUTDOWN POSITION	N.A.	R	N.A.	3, 4

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) Within 24 hours prior to startup, if not performed within the previous 7 days.

(c) Includes reactor manual control multiplexing system input.

* 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.

*** Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

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

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 7
License No. NPF-85

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

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated June 14, 1990, 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-85 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

ORIGINAL SIGNED BY:

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance:
July 30, 1990

PDI-2/LA	PDI-2/PE	PDI-2/PM	OGC	PDI-2/D
MO'Brien:tr	SDembek	RClark	RBachmann	WButler
7/11/90	7/10/90	07/09/90	7/17/90	7/30/90

ATTACHMENT TO LICENSE AMENDMENT NO. 7

FACILITY OPERATING LICENSE NO. NPF-85

DOCKET NO. 50-353

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

Remove	Insert
3/4 3-1	3/4 3-1*
3/4 3-2	3/4 3-2
3/4 3-5	3/4 3-5
3/4 3-6	3/4 3-6*
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
3/4 3-57	3/4 3-57*
3/4 3-58	3/4 3-58
3/4 3-59	3/4 3-59
3/4 3-60	3/4 3-60*
3/4 3-61	3/4 3-61
3/4 3-62	3/4 3-62

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within 1 hour. The provisions of Specification 3.0.4 are not applicable.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within 1 hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

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*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

**The trip system need not be placed in the tripped condition if this would cause the Trip Function to occur. When a trip system can be placed in the tripped condition without causing the Trip Function to occur, place the trip system with the most inoperable channels in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors[b]:			
a. Neutron Flux - High	2	3	1
	3, 4	3	2
	5(c)	3(d)	3
b. Inoperative	2	3	1
	3, 4	3	2
	5	3(d)	3
2. Average Power Range Monitor[e]:			
a. Neutron Flux - Upscale, Setdown	2	2	1
	3	2	2
	5(c)(1)	2(d)	3
b. Neutron Flux - Upscale			
1) Flow Biased	1	2	4
2) High Flow Clamped	1	2	4
c. Inoperative	1, 2	2	1
	3	2	2
	5(c)(1)	2(d)	3
d. Downscale	1(g)	2	4
3. Reactor Vessel Steam Dome Pressure - High	1, 2(f)	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	1(g)	1/valve	4

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) This function shall be automatically bypassed when the reactor mode switch is in the Run position and the associated APRM is not downscale.

(c) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations performed per Specification 3.10.3.

(d) The noncoincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 4 APRMs, 6 IRMs and 2 SRMs.

(e) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(g) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) This function shall be automatically bypassed when turbine first stage pressure is equivalent to a THERMAL POWER of less than 30% of RATED THERMAL POWER.

(k) Also actuates the EOC-RPT system.

(l) Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

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*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	N.A.
b. Inoperative	N.A.
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, Setdown	N.A.
b. Neutron Flux - Upscale	
1) Flow Biased	≤0.09
2) High Flow Clamped	≤0.09
c. Inoperative	N.A.
d. Downscale	N.A.
3. Reactor Vessel Steam Dome Pressure - High	≤ 0.55
4. Reactor Vessel Water Level - Low, Level 3	≤ 1.05
5. Main Steam Line Isolation Valve - Closure	≤ 0.06
6. Main Steam Line Radiation - High	N.A.
7. Drywell Pressure - High	N.A.
8. Scram Discharge Volume Water Level - High	
a. Level Transmitter	N.A.
b. Float Switch	N.A.
9. Turbine Stop Valve - Closure	≤ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≤ 0.08**
11. Reactor Mode Switch Shutdown Position	N.A.
12. Manual Scram	N.A.

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.
**Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S(b)	S/U(c), W	R	2
	S	W(j)	R	3, 4, 5
b. Inoperative	N.A.	W(j)	N.A.	2, 3, 4, 5
2. Average Power Range Monitor[f]:				
a. Neutron Flux -	S/U,S(b)	S/U(c), W	SA	2
Upscale, Setdown	S	W(j)	SA	3, 5[k]
b. Neutron Flux - Upscale				
1) Flow Biased	S,D(g)	S/U(c), W	W(d)(e), SA	1
2) High Flow Clamped	S	S/U(c), W	W(d)(e), SA	1
c. Inoperative	N.A.	W(j)	N.A.	1, 2, 3, 5[k]
d. Downscale	S	W	SA	1
3. Reactor Vessel Steam Dome Pressure - High	S	M	R	1, 2(h)
4. Reactor Vessel Water Level - Low, Level 3	S	M	R	1, 2
5. Main Steam Line Isolation Valve - Closure	N.A.	M	R	1
6. Main Steam Line Radiation - High	S	M	R	1, 2(h)
7. Drywell Pressure - High	S	M	R	1, 2
8. Scram Discharge Volume Water Level - High				
a. Level Transmitter	S	M	R	1, 2, 5[i]
b. Float Switch	N.A.	M	R	1, 2, 5[i]

TABLE 4.3.1.1-1 (Continued)
REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Turbine Stop Valve - Closure	N.A.	M	R	1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	N.A.	M	R	1
11. Reactor Mode Switch Shutdown Position	N.A.	R	N.A.	1, 2, 3, 4, 5
12. Manual Scram	N.A.	M	N.A.	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least ½ decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least ½ decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Within 24 hours prior to startup, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.
(g) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing loop flow (APRM % flow). During the startup test program, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the conclusion of the startup test program.
(h) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
(j) If the RPS shorting links are required to be removed per Specification 3.9.2, they may be reinstalled for up to 2 hours for required surveillance. During this time, CORE ALTERATIONS shall be suspended, and no control rod shall be moved from its existing position.
(k) Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

TABLE 3.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD BLOCK MONITOR[a]			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. APRM			
a. Flow Biased Neutron Flux - Upscale	4	1	61
b. Inoperative	4	1, 2, 5[f]	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Startup	4	2, 5[f]	61
3. SOURCE RANGE MONITORS ***			
a. Detector not full in[b]	3	2	61
	2	5	61
b. Upscale[c]	3	2	61
	2	5	61
c. Inoperative[c]	3	2	61
	2	5	61
d. Downscale[d]	3	2	61
	2	5	61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale[e]	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5**	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. Comparator	2	1	62
7. REACTOR MODE SWITCH SHUTDOWN POSITION	2	3, 4	63

TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION STATEMENTS

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

ACTION 61 - With the number of OPERABLE channels one or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

ACTION 63 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, initiate a rod block.

NOTES

* With THERMAL POWER $\geq$ 30% of RATED THERMAL POWER.

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

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

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

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

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

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

(e) This function is automatically bypassed when the IRM channels are on range 1.

(f) Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR ****		
a. Upscale		
1) During two recirculation loop operation		
a. Flow Biased*	≤ 0.66 W + (N-66)%, with a maximum of,	≤ 0.66 W + (N-63)%, with a maximum of,
b. High Flow Clamped	≤ N%	≤ (N-3)%
2) During single recirculation loop operation		
a. Flow Biased*	≤ 0.66 W + (N-72)%, with a maximum of,	≤ 0.66 W + (N-69)%, with a maximum of,
b. High Flow Clamped	≤ N%	≤ (N+3)%
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - Upscale		
1) During two recirculation loop operation	≤ 0.58 W + 50%*	≤ 0.58 W + 53%*
2) During single recirculation loop operation	≤ 0.58 W + 45%*	≤ 0.58 W + 48%*
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 4% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 1×10^5 cps	≤ 1.6×10^5 cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 3 cps**	≥ 1.8 cps**

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	N.A.	S/U[b][c],M[c]	SA	1*
b. Inoperative	N.A.	S/U[b][c],M[c]	N.A.	1*
c. Downscale	N.A.	S/U[b][c],M[c]	SA	1*
2. APRM				
a. Flow Biased Neutron Flux - Upscale	N.A.	S/U[b],M	SA	1
b. Inoperative	N.A.	S/U[b],M	N.A.	1, 2, 5***
c. Downscale	N.A.	S/U[b],M	SA	1
d. Neutron Flux - Upscale, Startup	N.A.	S/U[b],M	SA	2, 5***
3. SOURCE RANGE MONITORS				
a. Detector not full in	N.A.	S/U[b],W	N.A.	2, 5
b. Upscale	N.A.	S/U[b],W	SA	2, 5
c. Inoperative	N.A.	S/U[b],W	N.A.	2, 5
d. Downscale	N.A.	S/U[b],W	SA	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	N.A.	S/U[b],W	N.A.	2, 5
b. Upscale	N.A.	S/U[b],W	SA	2, 5
c. Inoperative	N.A.	S/U[b],W	N.A.	2, 5
d. Downscale	N.A.	S/U[b],W	SA	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	N.A.	M	R	1, 2, 5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	N.A.	S/U[b],M	SA	1
b. Inoperative	N.A.	S/U[b],M	N.A.	1
c. Comparator	N.A.	S/U[b],M	SA	1
7. REACTOR MODE SWITCH SHUTDOWN POSITION	N.A.	R	N.A.	3, 4

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) Within 24 hours prior to startup, if not performed within the previous 7 days.

(c) Includes reactor manual control multiplexing system input.

* 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.

*** Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 41 AND 7 TO FACILITY OPERATING

LICENSE NOS. NPF-39 AND NPF-85

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION, UNITS 1 AND 2

DOCKET NOS. 50-352 AND 50-353

1.0 INTRODUCTION

By letter dated June 14, 1990, Philadelphia Electric Company (the licensee) requested an amendment to Facility Operating License Nos. NPF-39 and NPF-85 for the Limerick Generating Station, Units 1 and 2. These proposed amendments would revise the Technical Specifications (TSs) to remove the operability requirements for the Average Power Range Monitors (APRMs) in Operational Condition 5 (OPCON 5), except while performing a shutdown margin demonstration in accordance with TS Section 3.10.3. By definition, Operational Condition 5 is the Refueling Condition, with the reactor mode switch in the shutdown or refuel position and with the reactor coolant temperature less than 140°F.

The requirement for the APRMs to be operable during a shutdown margin demonstration when the mode switch is in Startup as allowed by TS Section 3.10.3 will remain unchanged. TS Section 3.10.3 is a Special Test Exception which allows operators to change the reactor mode switch from Refuel to Startup to perform a shutdown margin demonstration. The licensee is proposing to add the following qualification to the OPCON 5 APRM TS operability requirements, "Required to be OPERABLE only prior to and during shutdown margin demonstrations as performed per Specification 3.10.3." This note is proposed to be added to TS Tables 3.3.1-1, "Reactor Protection System Instrumentation," and 4.3.1-1, "Reactor Protection System Instrumentation Surveillance Requirements," and TS Tables 3.3.6-1, "Control Rod Block Instrumentation," and 4.3.6-1 "Control Rod Block Instrumentation Surveillance Requirements."

2.0 EVALUATION

The Neutron Monitoring System (NMS) is composed of the following subsystems: Source Range Monitors (SRMs), Intermediate Range Monitors (IRMs), Local Power Range Monitors (LPRMs), Average Power Range Monitors (APRMs), Rod Block Monitor, and Traversing Incore Probe. The purpose of the SRM, IRM, and APRM subsystems is to monitor local and core average neutron flux levels and provide trip signals to the Reactor Protection

System (RPS) and control rod block portion of the Reactor Manual Control System (RMCS) as required. The NMS provides local and core average power information to the reactor operator. The IRM and APRM are safety-related subsystems and provide safety functions.

The SRM subsystem is composed of four detectors that are inserted into the core during shutdown conditions. Although the subsystem is not safety-related, it is important to plant safety. The SRMs are required by the TS to be operational in OPCON 5. During refueling operations, the plant operators use the SRMs to ensure that neutron flux remains within an acceptable range. Also, plant operators can monitor the SRMs for increases in neutron flux which may indicate that the reactor is approaching criticality.

The IRM subsystem is composed of eight incore detectors that are inserted into the core. The IRM is a five-decade instrument with ten ranges that are ranged up during normal power increases. The IRMs are designed to monitor neutron flux levels at a local core location and provide protection against local criticality events caused by control rod withdrawal errors. The IRMs monitor neutron flux levels from the upper portion of the SRM range to the lower portion of the APRM range. In terms of rated reactor power, the IRMs range from about 10^{-4}% of full reactor power to greater than 15% of full reactor power. The IRMs provide control rod block and scram functions at 108 and 120, respectively, of a 125 division scale.

The APRMs do not have incore detectors of their own but receive input from the LPRM detectors which are located at various levels throughout the core. The APRMs monitor core power from about 1% of full reactor power to 125% of full reactor power. The APRMs represent a core average power level while the IRMs and SRMs indicate a local power level. In OPCON 5, the APRMs operate in the setdown mode to provide a control rod block and scram function at 12% and 15% core average power, respectively.

The safety design bases of the IRM subsystem is to generate trip signals to prevent fuel damage resulting from anticipated or abnormal operational transients that could possibly occur while operating in the intermediate power range. The safety design bases of the APRM subsystem is to generate trip signals in response to average neutron flux increases in time to prevent fuel damage while the plant is in the operating power range. The independence and redundancy incorporated in the design of the IRM and APRM subsystems are consistent with the safety design bases of the NMS and RPS.

There are various levels of control to prevent inadvertent reactor criticality and fuel damage during refueling operations.

1) Licensed plant operators are trained to operate equipment and follow approved procedures.

2) Plant approved refueling and maintenance procedures specify core alteration steps.

3) SRMs indicate the potential for reactor criticality and generate a control rod block signal on high neutron flux levels. When shutdown margin has not been demonstrated, TS Section 3.9.2 requires the shorting links be removed so that the SRMs will operate in the noncoincident scram mode to cause a reactor scram as necessary.

4) Refueling interlocks prevent the removal of more than one control rod and prevent the insertion of fuel bundles into the core unless all control rods are fully inserted.

5) The IRMs and APRMs provide an indication of local power and average power, respectively. IRMs and APRMs will provide rod blocks and scram signals on high neutron flux levels.

The APRMs are not necessary for safe operation of the plant during OPCON 5 because the IRMs will generate an RPS scram or control rod block if neutron flux increases to the applicable setpoint. The IRMs are required by TS to be operational in OPCON 5. The IRMs are a safety-related subsystem of the NMS and are designed to indicate and respond to neutron flux increases at local core locations. The APRMs are designed to monitor and respond (scram and/or control rod block) to a core average neutron flux level. The most likely reactivity insertion transient expected during refueling would be a core alteration type event, e.g., control rod withdrawal or fuel assembly insertion into the core. A core alteration event would result in a local core criticality transient readily detected by the IRMs and/or SRMs.

The IRM subsystem is designed and calibrated to respond to a neutron flux level that is significantly less than the flux level monitored by the APRMs. For example, during refueling, when the IRMs are on their most sensitive range, the IRMs will generate a scram signal at less than 0.01% core average power while the APRMs will generate a scram signal at 15% core average power. The IRM subsystem acts as a backup protection system to the Refueling Interlocks (RIs) during refueling.

RIs are required to be operational during refueling operations in OPCON 5. They are not safety-related but are designed such that a single component failure does not cause an interlock failure. The purpose of the RIs is to restrict the movement of the control rods and the operation of the refueling equipment to reinforce operational procedures that prevent the reactor from becoming critical during refueling operations. RIs require that all control rods be fully inserted into the core prior

to allowing reactor operators to select and withdraw a single control rod. Other RIs will prevent the withdrawal of a control rod if the fuel loaded refueling platform is over the core. Also, the RIs require an "all-rods-in" signal before allowing a fuel loaded refueling platform to go over the core.

TS and plant operating procedures allow only one control rod to be withdrawn or removed at a time while the plant is in OPCON 5 and the mode switch is in "Refuel." The core loading pattern is designed to ensure that the core is subcritical by a specified margin with the most reactive control rod at the full out position. Withdrawal of one control rod would not cause criticality and the event would not register on the APRMs.

The design of the control rod drive system reduces the probability of a control rod error during refueling. For example, the latching action of the collet finger assembly serves to block the index tube in place. The velocity limiter physically prevents the control blade from being removed from the core with fuel in place.

The licensee concluded that the APRMs are not necessary for safe operation of the plant while operating in OPCON 5 with the mode switch in "Refuel" for the following reasons.

- The IRMs are a safety-related subsystem of the NMS and are required by TS to be operable in OPCON 5. The IRMs will generate an RPS Scram or control rod block if neutron flux increased to the applicable setpoint.

- The IRMs and SRMs are designed and calibrated to be more sensitive to neutron flux than the APRMs.

- The IRMs are designed to monitor local core events while the APRMs provide a measure of core average power condition. The IRMs can monitor and react to the most probable reactivity events expected during refueling, i.e., control rod withdrawal or fuel insertion.

- The IRMs would detect and respond (control rod block or reactor scram) to an inadvertent criticality event before the APRMs would provide a trip function.

- The withdrawal of only one control rod in OPCON 5 is permitted by the "one-rod-out" interlock while in "Refuel." The core is designed to be subcritical with one rod out.

- The withdrawal of a second control rod or inadvertent addition of a fuel bundle in OPCON 5 is precluded by refueling interlocks, refueling procedures, and administrative controls.

- The APRMs will still be required to be operational during a shutdown margin demonstration performed in OPCON 5 (a special test exception in the TS).
- The SRMs are required to be operational in OPCON 5.
- The transient analysis discussed in the FSAR does not require the APRMs to be operational in OPCON 5 to mitigate an undesirable operational or transient condition.

The licensee also performed a review of Sections 7.0 and 15.0 of the FSAR, the results of which were discussed in the application. The intent of the review was to identify conditions when the APRMs and IRMs were required to be operable to mitigate unacceptable consequences of inadvertent operational or transient conditions. The licensee reviewed various postulated scenarios. The results of the review concluded that should assumed operator errors occur, followed by postulated equipment malfunctions, there were adequate systems and interlocks without the APRMs to preclude potential inadvertent criticality or violation of a safety limit.

We have reviewed the licensee's analyses and agree with their evaluations. We conclude that monitoring of neutron flux levels, administrative controls, plant procedures, refueling interlocks, and SRM and IRM protective features provide and maintain the defense-in-depth design and operation which precludes the need for the APRMs to be operable in OPCON 5 with the mode switch in "Refuel." The proposed changes to the TSs are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve 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 these amendments involve 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 these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet 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 these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that these amendments involve no significant hazards consideration which was published in the Federal Register (55 FR 26291) on June 27, 1990 and consulted with the Commonwealth of Pennsylvania. No public comments were received and the Commonwealth 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 these amendments will not be inimical to the common defense and the security nor to the health and safety of the public.

Dated: July 30, 1990

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