February 16 1995

Mr. George A. Hunger, [illegible]
Director-Licensing, M[illegible]2A-1
PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, Pennsylvania 19087-0195

SUBJECT: MAIN STEAMLINE RADIATION MONITORING, LIMERICK GENERATING STATION, UNITS 1 AND 2 (TAC NOS. M87288 AND M87289)

Dear Mr. Hunger:

The Commission has issued the enclosed Amendment No. 89 to Facility Operating License No. NPF-39 and Amendment No. 52 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 October 29, 1993.

These amendments eliminate the main steamline radiation monitoring system high radiation trip function for initiating 1) an automatic reactor scram and automatic closure of the main steamline isolation valves, and 2) automatic closure of the main steamline drain valves, and main steam and reactor water sample line valves. The amendments also approve the relocation of portions of the information contained in the bases section.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice. You are requested to notify the NRC, in writing, when this amendment has been implemented at LGS, Units 1 and 2

Sincerely,
/S/
Frank Rinaldi, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-352/353

Enclosures:
1. Amendment No. 89 to
   License No. NPF-39
   Amendment No. 52 to
   License No. NPF-85
2. Safety Evaluation

cc w/enclosures:
See next page

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DF01

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 16, 1995

Mr. George A. Hunger, Jr.
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PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, Pennsylvania 19087-0195

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cc w/enclosures:
See next page

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 89
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 October 29, 1993, 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. 89 , 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 and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the
Technical Specifications

Date of Issuance: February 16, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 89

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
2-4	2-4
B 2-8	B 2-8
3/4 3-3	3/4 3-3
3/4 3-6	3/4 3-6
3/4 3-7	3/4 3-7
3/4 3-11	3/4 3-11
3/4 3-18	3/4 3-18
3/4 3-23	3/4 3-23
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-31	3/4 3-31
3/4 6-19	3/4 6-19
3/4 6-22	3/4 6-22
3/4 6-24	3/4 6-24
3/4 6-31	3/4 6-31
B 3/4 3-1	B 3/4 3-1
B 3/4 3-2	B 3/4 3-2

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Neutron Flux-Upscale		
1) During two recirculation loop operation:		
a) Flow Biased	≤ 0.66 W+ 66%, with a maximum of	≤ 0.66 W+ 68%, with a maximum of
b) High Flow Clamped	≤ 115% of RATED THERMAL POWER	≤ 117% of RATED THERMAL POWER
2) During single recirculation loop operation:		
a) Flow Biased	≤ 0.66 W+ 61%,	≤ 0.66 W+ 63%,
b) High Flow Clamped	Not Required OPERABLE	Not Required OPERABLE
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 4% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	≤ 1037 psig	≤ 1057 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
6. DELETED	DELETED	DELETED
7. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Level Transmitter	≤ 260' 9 5/8"elevation**	≤ 261' 5 5/8"elevation
b. Float Switch	≤ 260' 9 5/8"elevation**	≤ 261' 5 5/8"elevation
9. Turbine Stop Valve - Closure	≤ 5% closed	≤ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 500 psig	≥ 465 psig
11. Reactor Mode Switch Shutdown Position	N.A.	N.A.
12. Manual Scram	N.A.	N.A.

* See Bases Figure B 3/4.3-1.
**Equivalent to 25.45 gallons/scram discharge volume.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint has been used in transient analyses dealing with coolant inventory decrease. The scram setting was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

5. Main Steam Line Isolation Valve-Closure

The main steam line isolation valve closure trip was provided to limit the amount of fission product release for certain postulated events. The MSIVs are closed automatically from measured parameters such as high steam flow, low reactor water level, high steam tunnel temperature, and low steam line pressure. The MSIVs closure scram anticipates the pressure and flux transients which could follow MSIV closure and thereby protects reactor vessel pressure and fuel thermal/hydraulic Safety Limits.

6. DELETED

7. Drywell Pressure-High

High pressure in the drywell could indicate a break in the primary pressure boundary systems or a loss of drywell cooling. The reactor is tripped in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant and to the primary containment. The trip setting was selected as low as possible without causing spurious trips.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
6. DELETED	DELETED	DELETED	DELETED
7. Drywell Pressure - High	1, 2(h)	2	1
8. Scram Discharge Volume Water Level - High			
a. Level Transmitter	1, 2	2	1
	5(i)	2	3
b. Float Switch	1, 2	2	1
	5(i)	2	3
9. Turbine Stop Valve - Closure	1(j)	4(k)	6
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	1(j)	2(k)	6
11. Reactor Mode Switch Shutdown Position	1, 2	2	1
	3, 4	2	7
	5	2	3
12. Manual Scram	1, 2	2	1
	3, 4	2	8
	5	2	9

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. DELETED	DELETED
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	S/U(c),W W(j)	R R	2 3, 4, 5
b. Inoperative	N.A.	W(j)	N.A.	2, 3, 4, 5
2. Average Power Range Monitor(f):				
a. Neutron Flux - Upscale, Setdown	S/U,S(b) S	S/U(c),W W(j)	SA SA	2 3, 5(k)
b. Neutron Flux - Upscale				
1) Flow Biased	S,D(g)	S/U(c),Q	W(d)(e),SA	1
2) High Flow Clamped	S	S/U(c),Q	W(d)(e), SA	1
c. Inoperative	N.A.	Q(j)	N.A.	1, 2, 3, 5(k)
d. Downscale	S	Q	SA	1
3. Reactor Vessel Steam Dome Pressure - High	S	Q	R	1, 2(h)
4. Reactor Vessel Water Level-Low, Level 3	S	Q	R	1, 2
5. Main Steam Line Isolation Valve - Closure	N.A.	Q	R	1
6. DELETED	DELETED	DELETED	DELETED	DELETED
7. Drywell Pressure - High	S	Q	R	1, 2
8. Scram Discharge Volume Water Level - High				
a. Level Transmitter	S	Q	R	1, 2, 5(i)
b. Float Switch	N.A.	Q	R	1, 2, 5(i)

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Low-Level 2	B	2	1, 2, 3	21
2) Low, Low, Low-Level 1	C	2	1, 2, 3	21
b. DELETED	DELETED	DELETED	DELETED	DELETED
c. Main Steam Line Pressure - Low	P	2	1	22
d. Main Steam Line Flow - High	E	2/line	1, 2, 3	20
e. Condenser Vacuum - Low	Q	2	1, 2**, 3**	21
f. Outboard MSIV Room Temperature - High	F(f)	2	1, 2, 3	21
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	F(f)	14	1, 2, 3	21
h. Manual Initiation	NA	2	1, 2, 3	24
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level Low - Level 3	A	2	1, 2, 3	23
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	V	2	1, 2, 3	23
c. Manual Initiation	NA	1	1, 2, 3	24

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Low - Level 2	≥ - 38 inches*	≥ - 45 inches
2) Low, Low, Low - Level 1	≥ - 129 inches*	≥ - 136 inches
b. DELETED	DELETED	DELETED
c. Main Steam Line Pressure - Low	≥ 756 psig	≥ 736 psig
d. Main Steam Line Flow - High	≤ 108.7 psid	≤ 111.7 psid
e. Condenser Vacuum - Low	10.5 psia	≥10.1 psia/≤ 10.9 psia
f. Outboard MSIV Room Temperature - High	≤ 192°F	≤ 200°F
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	≤ 165°F	≤ 175°F
h. Manual Initiation	N.A.	N.A.
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Water Level Low - Level 3	≥ 12.5 inches*	≥ 11.0 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 75 psig	≤ 95 psig
c. Manual Initiation	N.A.	N.A.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Low - Level 2	$\leq$ 13(a)**
2) Low, Low, Low - Level 1	$\leq$ 1.0*/$\leq$ 13(a)**
b. DELETED	DELETED
c. Main Steam Line Pressure - Low	$\leq$ 1.0*/$\leq$ 13(a)**
d. Main Steam Line Flow - High	$\leq$ 0.5*/$\leq$ 13(a)**
e. Condenser Vacuum - Low	N.A.
f. Outboard MSIV Room Temperature - High	N.A.
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	N.A.
h. Manual Initiation	N.A.
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	
a. Reactor Vessel Water Level Low - Level 3	$\leq$ 13(a)
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	N.A.
c. Manual Initiation	N.A.
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. RWCS Δ Flow - High	$\leq$ 13##
b. RWCS Area Temperature - High	N.A.
c. RWCS Area Ventilation Δ Temperature - High	N.A.
d. SLCS Initiation	N.A.
e. Reactor Vessel Water Level - Low, Low - Level 2	$\leq$ 13(a)
f. Manual Initiation	N.A.

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
f. Outside Atmosphere To Refueling Area △ Pressure - Low	N.A.
g. Reactor Enclosure Manual Initiation	N.A.
h. Refueling Area Manual Initiation	N.A.

TABLE NOTATIONS

(a) Isolation system instrumentation response time specified includes 10 seconds diesel generator starting and 3 seconds for sequence loading delays.

(b) DELETED

* Isolation system instrumentation response time for MSIV only. No diesel generator delays assumed for MSIVs.

** Isolation system instrumentation response time for associated valves except MSIVs.

\# Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time shown in Tables 3.6.3-1, 3.6.5.2.1-1 and 3.6.5.2.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

\#\# With 45 second time delay.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Low, Level 2	S	Q	R	1, 2, 3
2) Low, Low, Low - Level 1	S	Q	R	1, 2, 3
b. DELETED	DELETED	DELETED	DELETED	DELETED
c. Main Steam Line Pressure - Low	S	Q	R	1
d. Main Steam Line Flow - High	S	Q	R	1, 2, 3
e. Condenser Vacuum - Low	S	Q	R	1, 2**, 3**
f. Outboard MSIV Room Temperature - High	S	Q	R	1, 2, 3
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	S	Q	R	1, 2, 3
h. Manual Initiation	N.A.	R	N.A.	1, 2, 3
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level## Low - Level 3	S	Q	R	1, 2, 3
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	S	Q	R	1, 2, 3
c. Manual Initiation	N.A.	R	N.A.	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
7. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level Low, Low - Level 2	S	Q	R	1, 2, 3
b. Drywell Pressure## - High	S	Q	R	1, 2, 3
c.1. Refueling Area Unit 1 Ventilation Exhaust Duct Radiation - High	S	Q	R	*#
2. Refueling Area Unit 2 Ventilation Exhaust Duct Radiation - High	S	Q	R	*#
d. Reactor Enclosure Ventilation Exhaust Duct Radiation - High	S	Q	R	1, 2, 3
e. Outside Atmosphere To Reactor Enclosure Δ Pressure - Low	N.A.	M	Q	1, 2, 3
f. Outside Atmosphere To Refueling Area Δ Pressure - Low	N.A.	M	Q	*
g. Reactor Enclosure Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. Refueling Area Manual Initiation	N.A.	R	N.A.	*

*Required when (1) handling irradiated fuel in the refueling area secondary containment, or (2) during CORE ALTERATIONS, or (3) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

**When not administratively bypassed and/or when any turbine stop valve is open.

#During operation of the associated Unit 1 or Unit 2 ventilation exhaust system.

##These trip functions (2a, 6b, and 7b) are common to the RPS actuation trip function.

TABLE 3.6.3-1

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX.ISOL. TIME.IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
003B	CONTAINMENT INSTRUMENT GAS SUPPLY - HEADER 'B'	59-1005B (CK)		NA			59
			HV59-129B	7	C,H,S		
003D-2	CONTAINMENT INSTRUMENT GAS SUPPLY TO ADS VALVES E & K	59-1112(CK)		NA			
			HV59-151B	45	M		59
007A(B,C,D)	MAIN STEAM LINE 'A'(B,C,D)	HV41-1F022A (B,C,D)		5*	C,E,F,P,Q	6	41
			HV41-1F028A (B,C,D)	5*	C,E,F,P,Q	6	
			HV40-1F001B (F,K,P)	45	EA	6	
			(XV40-101B (F,K,P) SEE PART B, THIS TABLE)	NA		6,1	
008	MAIN STEAM LINE DRAIN	HV41-1F016		30	C,E,F,P,Q	4	41
			HV41-1F019	30	C,E,F,P,Q		
009A	FEEDWATER	41-1F010A(CK)		NA			41
			HV41-1F074A(CK)	NA			
			41-1036A(CK)	NA			
			HV41-130B	45			
			HV41-133A	45			
			HV41-109A	NA		32	
			HV41-1F032A(CK)	NA			
			HV55-1F105	30		7	
			HV44-1F039(CK) (X-9B)	NA			
			41-1016(X-9B, X-44)	NA		31	

TABLE 3.6.3-1 (Continued)

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX.ISOL. TIME.IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
025	DRYWELL PURGE SUPPLY	HV57-121(X-201A)		5**	B,H,S,U,W,R,T	3,11,14	57
		HV57-123		5**	B,H,S,U,W,R,T	3,11,14	
			HV57-109 (X-201A)	6**	B,H,S,U,W,R,T	11	
			HV57-131 (X-201A)	5**	B,H,S,U,W,R,T	11	
			HV57-135	6**	B,H,S,U,W,R,T	11	
	HYDROGEN RECOMBINER "B" INLET	HV57-163		9	B,H,R,S	3,11,14	
			FV-C-DO-101B	90	B,H,R,S	11	
026	DRYWELL PURGE EXHAUST	HV57-114		5**	B,H,S,U,W,R,T	3,11,14,33	57
		HV57-111		15**	B,H,S,U,R,T	11	
		SV57-139		5		10	
			HV57-115	6**	B,H,S,U,W,R,T	11,33	
			HV57-117	5**	B,H,S,U,R,T	11	
			SV57-145	5	B,H,R,S	11	
	HYDROGEN RECOMBINER "A" INLET	HV57-161		9	B,H,R,S	3,11,14	
			FV-C-DO-101A	90	B,H,R,S	11	
027A	CONTAINMENT INSTRUMENT GAS SUPPLY TO ADS VALVES H,M,&S	59-1128(CK)		NA			59
			HV59-151A	45	M		
028A-1	RECIRC LOOP SAMPLE	HV43-1F019		10	B		43
			HV43-1F020	10	B		
028A-2	DRYWELL H2/02 SAMPLE	SV57-132		5	B,H,R,S	11	57
			SV57-142	5	B,H,R,S	11	
028A-3	DRYWELL H2/02 SAMPLE	SV57-134		5	B,H,R,S	11	57
			SV57-144	5	B,H,R,S	11	

TABLE 3.6.3-1 (Continued)

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX.ISOL. TIME.IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
040G-1	ILRT DATA ACQUISITION	60-1057		NA		11	60
			60-1058	NA		11	
040G-2	ILRT DATA ACQUISITION	60-1071		NA		11	60
			60-1070	NA		11	
040H-1	CONTAINMENT INSTRUMENT GAS SUPPLY - HEADER 'A'	59-1005A(CK)		NA			59
			HV59-129A	7	C,H,S		
042	STANDBY LIQUID CONTROL	48-1F007(CK) (X-116)		NA			48
			HV48-1F006A	60		29	
043B	MAIN STEAM SAMPLE	HV41-1F084		10	B		41
			HV41-1F085	10	B		
044	RWCU ALTERNATE RETURN	41-1017		NA		5,31	41
			41-1016(X-9A, X-9B)	NA			
			PSV41-112	NA			
045A(B,C,D)	LPCI INJECTION 'A'(B,C,D)	HV51-1F041A(B,C,D)(CK)		NA	9,22	51	
		HV51-142A(B,C,D)		7		9,22	
			HV51-1F017A (B,C,D)	38			
050A-1	DRYWELL PRESSURE INSTRUMENTATION		HV42-147B	45		10	42
053	DRYWELL CHILLED WATER SUPPLY - LOOP 'A'	HV87-128		60	C,H	11	87
			HV87-120A	60	C,H	11	
			HV87-125A	60	C,H	11	

TABLE 3.6.3-1 (Continued)

PART B - PRIMARY CONTAINMENT ISOLATION EXCESS FLOW CHECK VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX.ISOL. TIME.IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
003A-1	INSTRUMENTATION - 'D' MAIN STEAM LINE FLOW	--	XV41-1F070D XV41-1F073D			1	41
003A-2	INSTRUMENTATION - 'A' RECIRC PUMP SEAL PRESSURE	--	XV43-1F003A			1	43
003C-1	INSTR. - HPCI STEAM FLOW	--	XV55-1F024A			1	55
003C-2	INSTR. - HPCI STEAM FLOW	--	XV55-1F024C			1	55
003D-1	INSTR. - 'A' MAIN STEAM LINE FLOW	--	XV41-1F070A XV41-1F073A			1	41
007A(B,C,D)	INSTR. - 'A'(B,C,D) MAIN STEAM LINE PRESSURE	(HV41-1F022A(B, C,D) SEE PART A THIS TABLE)		5*	C,E,F,P,Q	6	41
			(HV41-1F028A (B,C,D) SEE PART A THIS TABLE)	5*	C,E,F,P,Q	6	
			(HV40-1F001B (F,K,P) SEE PART A THIS TABLE)	45	EA	6	
			XV40-101B(F, K,P)			1,6	
020A-1	INSTR - RPV LEVEL	--	XV42-1F045B			1	42
020A-2	INSTR - 'B' LPCI DELTA P	--	XV51-102B			1	51
020A-3	INSTR - 'D' LPCI DELTA P	--	XV51-103B			1	51
020B-1	INSTR - RPV LEVEL	--	XV42-1F045C			1	42
020B-2	INSTR - 'C' LPCI DELTA P	--	XV51-102C			1	51

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279 for nuclear power plant protection systems. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P, "Technical Specification Improvement Analyses for BWR Reactor Protection System," as approved by the NRC and documented in the NRC Safety Evaluation Report (SER) (letter to T. A. Pickens from A. Thadani dated July 15, 1987. The bases for the trip settings of RPS are discussed in the bases for Specification 2.2.1.

Automatic reactor trip upon receipt of a high-high radiation signal from the Main Steam Line Radiation Monitoring System was removed as the result of an analysis performed by General Electric in NEDO-31400A. The NRC approved the results of this analysis as documented in the SER (letter to George J. Beck, BWR Owner's Group from A. C. Thadani, NRC, dated May 15, 1991).

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the safety analyses. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance.

Specified surveillance intervals and maintenance outage times have been determined in accordance with NEDC-30851P, Supplement 2, "Technical Specification Improvement Analysis for BWR Instrumentation Common to RPS and ECCS Instrumentation" as approved by the NRC and documented in the NRC Safety Evaluation Report (SER) (letter to D. N. Grace from C. E. Rossi dated January 6, 1989) and NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," as approved by the NRC and documented in the NRC SER (letter to S. D. Floyd from C. E. Rossi dated June 18, 1990).

Automatic closure of the MSIVs upon receipt of a high-high radiation signal from the Main Steam Line Radiation Monitoring System was removed as the result of an analysis performed by General Electric in NEDO-31400A. The NRC approved the results of this analysis as documented in the SER (letter to George J. Beck, BWR Owner's Group from A. C. Thadani, NRC, dated May 15, 1991).

Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 10-second diesel startup and the 3 second load center loading delay. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 13-second delay. It follows that checking the valve speeds and the 13-second time for emergency power establishment will establish the response time for the isolation functions.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 52
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 October 29, 1993, 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. 52, 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 and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the
Technical Specifications

Date of Issuance: February 16, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 52

FACILITY OPERATING LICENSE NO. NPF-85

DOCKET NO. 50-353

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.

Remove	Insert
2-4	2-4
B 2-8	B 2-8
3/4 3-3	3/4 3-3
3/4 3-6	3/4 3-6
3/4 3-7	3/4 3-7
3/4 3-11	3/4 3-11
3/4 3-18	3/4 3-18
3/4 3-23	3/4 3-23
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-31	3/4 3-31
3/4 6-19	3/4 6-19
3/4 6-22	3/4 6-22
3/4 6-24	3/4 6-24
3/4 6-31	3/4 6-31
B 3/4 3-1	B 3/4 3-1
B 3/4 3-2	B 3/4 3-2

TABLE 2.2.1-1
REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Neutron Flux-Upscale		
1) During two recirculation loop operation:		
a) Flow Biased	≤ 0.66 W+ 62%, with a maximum of	≤ 0.66 W+ 64%, with a maximum of
b) High Flow Clamped	≤ 115% of RATED THERMAL POWER	≤ 117% of RATED THERMAL POWER
2) During single recirculation loop operation:		
a) Flow Biased	≤ 0.66 W+ 57%,	≤ 0.66 W+ 59%,
b) High Flow Clamped	Not Required OPERABLE	Not Required OPERABLE
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 4% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	≤ 1096 psig	≤ 1103 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
6. DELETED	DELETED	DELETED
7. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Level Transmitter	≤ 261' 1 1/4" elevation**	≤ 261' 9 1/4" elevation
b. Float Switch	≤ 261' 1 1/4" elevation**	≤ 261' 9 1/4" elevation

* See Bases Figure B 3/4.3-1.
**Equivalent to 25.58 gallons/scram discharge volume.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint has been used in transient analyses dealing with coolant inventory decrease. The scram setting was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

5. Main Steam Line Isolation Valve-Closure

The main steam line isolation valve closure trip was provided to limit the amount of fission product release for certain postulated events. The MSIVs are closed automatically from measured parameters such as high steam flow, low reactor water level, high steam tunnel temperature, and low steam line pressure. The MSIVs closure scram anticipates the pressure and flux transients which could follow MSIV closure and thereby protects reactor vessel pressure and fuel thermal/hydraulic Safety Limits.

6. DELETED

7. Drywell Pressure-High

High pressure in the drywell could indicate a break in the primary pressure boundary systems or a loss of drywell cooling. The reactor is tripped in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant and to the primary containment. The trip setting was selected as low as possible without causing spurious trips.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
6. DELETED	DELETED	DELETED	DELETED
7. Drywell Pressure - High	1, 2(h)	2	1
8. Scram Discharge Volume Water Level - High			
a. Level Transmitter	1, 2	2	1
	5(i)	2	3
b. Float Switch	1, 2	2	1
	5(i)	2	3
9. Turbine Stop Valve - Closure	1(j)	4(k)	6
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	1(j)	2(k)	6
11. Reactor Mode Switch Shutdown Position	1, 2	2	1
	3, 4	2	7
	5	2	3
12. Manual Scram	1, 2	2	1
	3, 4	2	8
	5	2	9

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. DELETED	DELETED
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	S/U(c),W W(j)	R R	2 3, 4, 5
b. Inoperative	N.A.	W(j)	N.A.	2, 3, 4, 5
2. Average Power Range Monitor(f):				
a. Neutron Flux - Upscale, Setdown	S/U,S(b) S	S/U(c),W W(j)	SA SA	2 3, 5(k)
b. Neutron Flux - Upscale				
1) Flow Biased	S,D(g)	S/U(c),Q	W(d)(e),SA	1
2) High Flow Clamped	S	S/U(c),Q	W(d)(e),SA	1
c. Inoperative	N.A.	Q(j)	N.A.	1, 2, 3, 5(k)
d. Downscale	S	Q	SA	1
3. Reactor Vessel Steam Dome Pressure - High	S	Q	R	1, 2(h)
4. Reactor Vessel Water Level-Low, Level 3	S	Q	R	1, 2
5. Main Steam Line Isolation Valve - Closure	N.A.	Q	R	1
6. DELETED	DELETED	DELETED	DELETED	DELETED
7. Drywell Pressure - High	S	Q	R	1, 2
8. Scram Discharge Volume Water Level - High				
a. Level Transmitter	S	Q	R	1, 2, 5(i)
b. Float Switch	N.A.	Q	R	1, 2, 5(i)

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Low-Level 2	B	2	1, 2, 3	21
2) Low, Low, Low-Level 1	C	2	1, 2, 3	21
b. DELETED	DELETED	DELETED	DELETED	DELETED
c. Main Steam Line Pressure - Low	P	2	1	22
d. Main Steam Line Flow - High	E	2/line	1, 2, 3	20
e. Condenser Vacuum - Low	Q	2	1, 2**, 3**	21
f. Outboard MSIV Room Temperature - High	F(f)	2	1, 2, 3	21
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	F(f)	14	1, 2, 3	21
h. Manual Initiation	NA	2	1, 2, 3	24
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level Low - Level 3	A	2	1, 2, 3	23
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	V	2	1, 2, 3	23
c. Manual Initiation	NA	1	1, 2, 3	24

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Low - Level 2	≥ - 38 inches*	≥ - 45 inches
2) Low, Low, Low - Level 1	≥ - 129 inches*	≥ - 136 inches
b. DELETED	DELETED	DELETED
c. Main Steam Line Pressure - Low	≥ 756 psig	≥ 736 psig
d. Main Steam Line Flow - High	≤ 122.1 psid	≤ 123 psid
e. Condenser Vacuum - Low	10.5 psia	≥10.1 psia/≤ 10.9 psia
f. Outboard MSIV Room Temperature - High	≤ 192°F	≤ 200°F
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	≤ 165°F	≤ 175°F
h. Manual Initiation	N.A.	N.A.
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Water Level Low - Level 3	≥ 12.5 inches*	≥ 11.0 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 75 psig	≤ 95 psig
c. Manual Initiation	N.A.	N.A.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Low - Level 2	≤ 13(a)**
2) Low, Low, Low - Level 1	≤ 1.0*/≤ 13(a)**
b. DELETED	DELETED
c. Main Steam Line Pressure - Low	≤ 1.0*/≤ 13(a)**
d. Main Steam Line Flow - High	≤ 0.5*/≤ 13(a)**
e. Condenser Vacuum - Low	N.A.
f. Outboard MSIV Room Temperature - High	N.A.
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	N.A.
h. Manual Initiation	N.A.
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	
a. Reactor Vessel Water Level Low - Level 3	≤ 13(a)
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	N.A.
c. Manual Initiation	N.A.
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. RWCS Δ Flow - High	≤ 13##
b. RWCS Area Temperature - High	N.A.
c. RWCS Area Ventilation Δ Temperature - High	N.A.
d. SLCS Initiation	N.A.
e. Reactor Vessel Water Level - Low, Low - Level 2	≤ 13(a)
f. Manual Initiation	N.A.

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
f. Outside Atmosphere To Refueling Area Δ Pressure - Low	N.A.
g. Reactor Enclosure Manual Initiation	N.A.
h. Refueling Area Manual Initiation	N.A.

TABLE NOTATIONS

(a) Isolation system instrumentation response time specified includes 10 seconds diesel generator starting and 3 seconds for sequence loading delays.

(b) DELETED

* Isolation system instrumentation response time for MSIV only. No diesel generator delays assumed for MSIVs.

** Isolation system instrumentation response time for associated valves except MSIVs.

\# Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time shown in Tables 3.6.3-1, 3.6.5.2.1-1 and 3.6.5.2.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

\## With 45 second time delay.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Low, Level 2	S	Q	R	1, 2, 3
2) Low, Low, Low - Level 1	S	Q	R	1, 2, 3
b. DELETED	DELETED	DELETED	DELETED	DELETED
c. Main Steam Line Pressure - Low	S	Q	R	1
d. Main Steam Line Flow - High	S	Q	R	1, 2, 3
e. Condenser Vacuum - Low	S	Q	R	1, 2**, 3**
f. Outboard MSIV Room Temperature - High	S	Q	R	1, 2, 3
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	S	Q	R	1, 2, 3
h. Manual Initiation	N.A.	R	N.A.	1. 2, 3
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level## Low - Level 3	S	Q	R	1, 2, 3
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	S	Q	R	1, 2, 3
c. Manual Initiation	N.A.	R	N.A.	1, 2, 3

TABLE 4.3.2.1-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRE
7. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level Low, Low - Level 2	S	Q	R	1, 2, 3
b. Drywell Pressure## - High	S	Q	R	1, 2, 3
c.1. Refueling Area Unit 1 Ventilation Exhaust Duct Radiation - High	S	Q	R	*#
2. Refueling Area Unit 2 Ventilation Exhaust Duct Radiation - High	S	Q	R	*#
d. Reactor Enclosure Ventilation Exhaust Duct Radiation - High	S	Q	R	1, 2, 3
e. Outside Atmosphere To Reactor Enclosure Δ Pressure - Low	N.A.	M	Q	1, 2, 3
f. Outside Atmosphere To Refueling Area Δ Pressure - Low	N.A.	M	Q	*
g. Reactor Enclosure Manual Initiation	N.A.	R	N.A.	1, 2, 3
h. Refueling Area Manual Initiation	N.A.	R	N.A.	*

*Required when (1) handling irradiated fuel in the refueling area secondary containment, or (2) during CORE ALTERATIONS, or (3) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

**When not administratively bypassed and/or when any turbine stop valve is open.

#During operation of the associated Unit 1 or Unit 2 ventilation exhaust system.

##These trip functions (2a, 6b, and 7b) are common to the RPS actuation trip function.

TABLE 3.6.3-1

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX.ISOL. TIME.IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
003B	CONTAINMENT INSTRUMENT GAS SUPPLY - HEADER 'B'	59-2005B (CK)		NA			59
			HV59-229B	7	C,H,S		
003D-2	CONTAINMENT INSTRUMENT GAS SUPPLY TO ADS VALVES E & K	59-2112(CK)		NA			
			HV59-251B	45	M		59
007A(B,C,D)	MAIN STEAM LINE 'A'(B,C,D)	HV41-2F022A (B,C,D)		5*	C,E,F,P,Q	6	41
			HV41-2F028A (B,C,D)	5*	C,E,F,P,Q	6	
			HV40-2F001B (F,K,P)	45	EA	6	
			(XV40-201B (F,K,P) SEE PART B, THIS TABLE)	NA		6,1	
008	MAIN STEAM LINE DRAIN	HV41-2F016		30	C,E,F,P,Q	4	41
			HV41-2F019	30	C,E,F,P,Q		
009A	FEEDWATER	41-2F010A(CK)		NA			41
			HV41-2F074A(CK)	NA			
			41-2036A(CK)	NA			
			HV41-230B	45			
			HV41-233A	45			
			HV41-209A	NA		32	
			HV41-2F032A(CK)	NA			
			HV55-2F105	30		7	
			HV44-2F039(CK) (X-9B)	NA			
			41-2016(X-9B, X-44)	NA		31	

TABLE 3.6.3-1 (Continued)

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX.ISOL. TIME.IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
025	DRYWELL PURGE SUPPLY	HV57-221(X-201A)		5**	B,H,S,U,W,R,T	3,11,14	57
		HV57-223		5**	B,H,S,U,W,R,T	3,11,14	
			HV57-209 (X-201A)	6**	B,H,S,U,W,R,T	11	
			HV57-231 (X-201A)	5**	B,H,S,U,W,R,T	11	
			HV57-235	6**	B,H,S,U,W,R,T	11	
	HYDROGEN RECOMBINER "B" INLET	HV57-263		9	B,H,R,S	3,11,14	
			FV57-DO-201B	90	B,H,R,S	11,34	
026	DRYWELL PURGE EXHAUST	HV57-214		5**	B,H,S,U,W,R,T	3,11,14,33	57
		HV57-211		15**	B,H,S,U,R,T	11	
		SV57-239		5		10	
			HV57-215	6**	B,H,S,U,W,R,T	11,33	
			HV57-217	5**	B,H,S,U,R,T	11	
			SV57-245	5	B,H,R,S	11	
	HYDROGEN RECOMBINER "A" INLET	HV57-261		9	B,H,R,S	3,11,14	
			FV57-DO-201A	90	B,H,R,S	11,34	
027A	CONTAINMENT INSTRUMENT GAS SUPPLY TO ADS VALVES H,M,&S	59-2128(CK)		NA			59
			HV59-251A	45	M		
028A-1	RECIRC LOOP SAMPLE	HV43-2F019		10	B		43
			HV43-2F020	10	B		
028A-2	DRYWELL H2/02 SAMPLE	SV57-232		5	B,H,R,S	11	57
			SV57-242	5	B,H,R,S	11	
028A-3	DRYWELL H2/02 SAMPLE	SV57-234		5	B,H,R,S	11	57
			SV57-244	5	B,H,R,S	11	

TABLE 3.6.3-1 (Continued)

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX.ISOL. TIME.IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
040G-1	ILRT DATA ACQUISITION	60-2057		NA		11	60
			60-2058	NA		11	
040G-2	ILRT DATA ACQUISITION	60-2071		NA		11	60
			60-2070	NA		11	
040H-1	CONTAINMENT INSTRUMENT GAS SUPPLY - HEADER 'A'	59-2005A(CK)		NA			59
			HV59-229A	7	C,H,S		
042	STANDBY LIQUID CONTROL	48-2F007(CK) (X-116)		NA			48
			HV48-2F006A	60		29	
043B	MAIN STEAM SAMPLE	HV41-2F084		10	B		41
			HV41-2F085	10	B		
044	RWCU ALTERNATE RETURN	41-2017		NA		5,31	41
			41-2016(X-9A, X-9B)	NA			
			PSV41-212	NA			
045A(B,C,D)	LPCI INJECTION 'A'(B,C,D)	HV51-2F041A(B,C, D)(CK)		NA		9,22	51
		HV51-242A(B,C, D)		7		9,22	
			HV51-2F017A (B,C,D)	38			
050A-1	DRYWELL PRESSURE INSTRUMENTATION		HV42-247B	45		10	42
053	DRYWELL CHILLED WATER SUPPLY - LOOP 'A'	HV87-228		60	C,H	11	87
			HV87-220A	60	C,H	11	
			HV87-225A	60	C,H	11	

TABLE 3.6.3-1 (Continued)

PART B - PRIMARY CONTAINMENT ISOLATION EXCESS FLOW CHECK VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX.ISOL. TIME.IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	P&ID
003A-1	INSTRUMENTATION - 'D' MAIN STEAM LINE FLOW	--	XV41-2F070D XV41-2F073D			1	41
003A-2	INSTRUMENTATION - 'A' RECIRC PUMP SEAL PRESSURE	--	XV43-2F003A			1	43
003C-1	INSTR. - HPCI STEAM FLOW	--	XV55-2F024A			1	55
003C-2	INSTR. - HPCI STEAM FLOW	--	XV55-2F024C			1	55
003D-1	INSTR. - 'A' MAIN STEAM LINE FLOW	--	XV41-2F070A XV41-2F073A			1	41
007A(B,C,D)	INSTR. - 'A'(B,C,D) MAIN STEAM LINE PRESSURE	(HV41-2F022A(B, C,D) SEE PART A THIS TABLE)		5*	C,E,F,P,Q	6	41
			(HV41-2F028A (B,C,D) SEE PART A THIS TABLE)	5*	C,E,F,P,Q	6	
			(HV40-2F001B (F,K,P) SEE PART A THIS TABLE)	45	EA	6	
			XV40-201B(F, K,P)			1,6	
020A-1	INSTR - RPV LEVEL	--	XV42-2F045B			1	42
020A-2	INSTR - 'B' LPCI DELTA P	--	XV51-202B			1	51
020A-3	INSTR - 'D' LPCI DELTA P	--	XV51-203B			1	51
020B-1	INSTR - RPV LEVEL	--	XV42-2F045C			1	42
020B-2	INSTR - 'C' LPCI DELTA P	--	XV51-202C			1	51

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279 for nuclear power plant protection systems. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P, "Technical Specification Improvement Analyses for BWR Reactor Protection System," as approved by the NRC and documented in the NRC Safety Evaluation Report (SER) (letter to T. A. Pickens from A. Thadani dated July 15, 1987. The bases for the trip settings of RPS are discussed in the bases for Specification 2.2.1.

Automatic reactor trip upon receipt of a high-high radiation signal from the Main Steam Line Radiation Monitoring System was removed as the result of an analysis performed by General Electric in NEDO-31400A. The NRC approved the results of this analysis as documented in the SER (letter to George J. Beck, BWR Owner's Group from A. C. Thadani, NRC, dated May 15, 1991).

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the safety analyses. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance.

Specified surveillance intervals and maintenance outage times have been determined in accordance with NEDC-30851P, Supplement 2, "Technical Specification Improvement Analysis for BWR Instrumentation Common to RPS and ECCS Instrumentation" as approved by the NRC and documented in the NRC Safety Evaluation Report (SER) (letter to D. N. Grace from C. E. Rossi dated January 6, 1989) and NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," as approved by the NRC and documemnted in the NRC SER (letter to S. D. Floyd from C. E. Rossi dated June 18, 1990).

Automatic closure of the MSIVs upon receipt of a high-high radiation signal from the Main Steam Line Radiation Monitoring System was removed as the result of an analysis performed by General Electric in NEDO-31400A. The NRC approved the results of this analysis as documented in the SER (letter to George J. Beck, BWR Owner's Group from A. C. Thadani, NRC, dated May 15, 1991).

Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 10-second diesel startup and the 3 second load center loading delay. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 13-second delay. It follows that checking the valve speeds and the 13-second time for emergency power establishment will establish the response time for the isolation functions.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 89 AND 52 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 October 29, 1993, the Philadelphia Electric Company (PECo or the licensee) submitted a request for changes to the Limerick Generating Station, Units 1 and 2, Technical Specifications (TS). The requested changes would eliminate the Main Steamline Radiation Monitoring (MSLRM) system high radiation trip function for initiating an automatic reactor scram and automatic closure of the Main Steamline Isolation Valves (MSIV) and automatic closure of the Main Steamline Drain Valves, and Main Steam and Reactor Water Sample line valves.

PECO has stated that elimination of this trip function would result in reduced potential for unnecessary reactor shutdowns caused by spurious MSLRM actuation trips and would increase plant operational flexibility without compromising plant safety. The licensee referenced General Electric (GE) Topical Report NEDO-31400A, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steamline Isolation Valve Closure Function and Scram Function of the Main Steamline Radiation Monitor", which was approved by the NRC by letter dated May 15, 1991, as justification for the removal. The GE report (NEDO-31400A) does not address the TS changes to eliminate the automatic closure of the Main Steamline Drain Valves and Main Steam and Reactor Water Sample line valves on a MSLRM system high radiation signal. The licensee has provided additional information justifying these proposed changes. The amendments also include adding information to the applicable Bases sections (i.e., 3/4.3.1 and 3/4.3.2) to reflect the elimination of the automatic reactor scram and automatic MSIV closure functions on high radiation.

2.0 BACKGROUND

The MSLRM system is designed to monitor radiation levels in the Main Steam lines. This system monitors any gross release of fission products from the fuel and initiates system isolations and a reactor trip to limit fuel damage and contain released fission products. It consists of four gamma-sensitive instrument channels that monitor gamma radiation levels in the four Main Steam lines. These detectors are geometrically arranged to detect significant increases in gamma radiation levels. They allow for the earliest practicable detection of a gross fuel failure. The detectors' high radiation trip setting accounts for full reactor power background radiation levels and potential releases of fission products from the fuel. However, no consideration was given for the potential of spurious reactor trips from Nitrogen-16 spikes, instrument instabilities and other operational occurrences. If significant increase in radiation level is detected, the MSLRM system transmits signals to the Reactor Protection System (RPS) and the Primary Containment and Reactor Vessel Isolation Control System (PCRVICS). The RPS initiates an automatic reactor scram and the PCRVICS initiates an automatic closure of all MSIVs, Main Steam line drain valves, and Main Steam and Reactor Water Sample line valves.

The proposed TS changes defeat portions of MSLRM high radiation trip function logic circuitry in the RPS and PCRVICS and have no impact on the operation of the RPS or PCRVICS with respect to other intended safety functions. The MSLRM system high radiation trip function for the Mechanical Vacuum Pump (MVP) will be retained. The justification for the elimination of the MSLRM system high radiation trip function for initiating an automatic reactor scram and automatic closure of the MSIVs is based on GE's Topical Report NEDO-31400A and its applicability to LGS, Units 1 and 2.

3.0 EVALUATION

The MSLRM consists of four redundant radiation detectors located on the outside of the main steam lines and external to the primary containment. The MSLRM was designed to provide an early indication of gross fuel failures. The original intent of this monitor was to mitigate the releases of the detected fuel failure by providing a scram signal to terminate the initiating event and a MSIV closure signal to assure containment of the release. PECo has referenced Topical Report NEDO-31400A in support of its request to eliminate the MSLRM scram and group isolation functions. In the topical report, GE analyzes a control rod drop accident where the Main Steamline high radiation isolation is eliminated. NEDO-31400A, states that there have been a number of spurious actuations of the MSLRM system at other plants causing unnecessary automatic reactor shutdowns. These shutdowns were caused by instrument failures, reactor coolant chemistry excursions, radiation monitor maintenance errors, etc., but none from the detection of failed fuel. The report also indicates that removing the MSLRM system high radiation reactor scram and MSIV closure functions will reduce the potential for unnecessary reactor shutdowns and will increase plant operational flexibility since the main condenser will remain available for decay heat removal. In addition, it demonstrates that the Offgas Treatment System provides significant holdup times for radionuclides, and that use of this system is an acceptable method for controlling unexpected radioactive material releases. Further, eliminating the MSLRM system trip functions in conjunction with proper operation of the Offgas Treatment system will ensure that any radioactive material released to the environment is a small fraction of 10 CFR Part 100 limits.

The MSLRM system high radiation Main Control Room (MCR) alarms and trip function for isolating the MVP will be retained. This will ensure that any radioactive material released from a fuel failure will be contained in the main condenser and processed through the Offgas Treatment system, that continuously removes non-condensable gases from the main condenser by the Steam Jet Air Ejectors (SJAEs) during plant operation. The Offgas Treatment system reduces offgas radioactivity levels to permissible levels for release under all site atmospheric conditions. The system uses catalytic recombination for volume reduction and control of hydrogen concentration and activated charcoal filters to adsorb fission product and activation gases prior to release to the environment. Instrumentation permits system operation and monitoring from the MCR. To be consistent with Section 15.4.9 of the Standard Review Plan, all of the postulated radioactive material is assumed to be released to the condenser and turbine before the isolation occurs. Hence, the automatic isolation resulting from the MSLRM provides no benefits, since the resultant dose consequences from the control rod drive accident analysis will remain unchanged.

In the NRC Safety Evaluation of GE's NEDO-31400A, dated May 15, 1991, the staff stated that participating boiling water reactor utilities, listed in Table 1 of the Topical Report, may reference NEDO-031400 in support of their license amendment application if they meet the following criteria:

1. The licensee must demonstrate that the assumptions with regard to input values (including power per assembly, Chi/Q, and decay times) that are made in the generic analysis bound those for the plant.

2. The licensee must provide reasonable assurance that increased levels of radioactivity in the main steam lines will be controlled expeditiously to limit both occupational doses and environmental releases; and,

3. The MSLRM and offgas radiation monitor alarm setpoints must be set at 1.5 times the nominal background including N-16 at the monitor locations and the licensee must promptly sample the reactor coolant to determine possible contamination levels if the setpoint of either monitor is exceeded.

PECo has demonstrated that the assumption with regard to input values that are made in the generic analysis bound those for the plant. The power per assembly, Chi/Q, and decay time assumptions used in the NEDO-31400A analysis are all conservative with respect to the LGS Units 1 and 2, as stated in LGS Updated Final Safety Analysis Report (UFSAR). The licensee has indicated that plant procedures will be in place to implement the appropriate mitigative measures in response to a MSLRM system high radiation alarm signal. The procedures will maintain off-gas release rates within TS requirements, otherwise an orderly plant shutdown will be implemented. The MSLRM alarm setpoint is set at 1.5 times the expected full reactor power background radiation level, and the offgas treatment system radiation monitor high radiation setpoint is set at 2,100 mR/hr, per GE's recommendations. Further, samples will be taken to check the reactor coolant chemistry conditions and to assess the adequacy of the two setpoints per NRC's recommendations.

The licensee has proposed the elimination of the MSLRM system high radiation automatic closure function for the Main Steamline drain valves and Main Steam and Reactor Water Sample line valves, because the flow ultimately discharges into the main condenser, just as the flow from the MSIVs. Therefore, any radioactive material passing through the Main Steamline drain valves to the main condenser and through the Offgas Treatment system is treated identically to any radioactive material that would pass through the MSIVs. NEDO-31400A evaluated removing the MSLRM system high radiation trip function for closing the MSIVs. This same analysis applies for closure of the Main Steamline drain valves. The elimination of the Main Steam and Reactor Water Sample line valves is based on the fact that the effects are negligible, since these lines are small in comparison to the size of the lines associated with the MSIVs and Main Steamline drains. The sample lines are routed to a sample sink where inlet valves on the sample lines are normally closed, and flow downstream of these valves is controlled and limited by needle valves. These inlet valves must be opened periodically to allow for chemistry sampling. The sample sink is located in the Reactor Enclosure. It is enclosed, vented, and filtered prior to release to the environment. The Reactor Enclosure ventilation duct radiation monitor samples air from the sample sink hood exhaust, and will isolate the Reactor Enclosure ventilation system if the radiation levels exceed the monitor's setpoint. In the unlikely event that the Main Steamline inlet valve is open, and the throttling needle valve is fully open, a minimal amount of radioactive material would be released to the environment. The whole body dose received at the exclusion area boundary through this flowpath over a 30 day period would be 0.0005 sieverts (0.05 rems) as a result of the release of radioactive isotopes of Xenon and Krypton (i.e., radioactive isotopes of Iodine are adsorbed in the vent duct charcoal filter). This represents less than 1% of the whole body dose of six rem as stipulated in NUREG-0800, Standard Review Plan (SRP), Section 15.4.9, Appendix A, "Radiological Consequences of Control Rod Drop Accident (BWR)," and assures that any release would be a small fraction of the dose limit requirements specified in 10 CFR Part 100.

Based upon implementation of the revised procedures described in PECo's submittal, the staff finds that the requirements of the NRC's generic safety evaluation on NEDO-31400A are satisfied for LGS Units 1 and 2. Also, the changes to eliminate the automatic closure of the identified drain valves and sample line valves have been properly justified. Therefore, the staff finds the proposed revisions to LGS, Units 1 and 2 TS acceptable based on the above evaluation.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the 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 the amendments involve no significant hazards consideration, and there has been no public comment on such finding (59 FR 624). Accordingly, the 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 or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: February 16, 1995