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Docket No.: 50-387

Mr. Norman W. Curtis
Vice President
Engineering and Construction - Nuclear
Pennsylvania Power & Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Dear Mr. Curtis:

Subject: Amendment No. 6 to Facility Operating License No. NPF-14 - Susquehanna Steam Electric Station, Unit 1

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 6 to Facility Operating License No. NPF-14 for the Susquehanna Steam Electric Station, Unit 1. The amendment is in response to your letter dated August 18, 1982, as amended by your letter dated August 23, 1982. This amendment changes surveillance requirements for vacuum breakers to be consistent with manufacturer's test procedure, establishes minimum discharge pressure for the low pressure coolant injection pump for testing purposes, changes ECCS actuation instrumentation setpoint allowable values, revises applicability of SRV's when the vessel is not pressurized, corrects errors in the fire detection instrumentation table, and corrects typographical errors in various technical specifications.

A copy of the related safety evaluation supporting Amendment No. 6 to Facility Operating License NPF-14 is enclosed. Also enclosed is a copy of a related notice which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

Original signed by:
Robert L. Perch

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

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P PDR

Enclosures:
1. Amendment No. 6 to NPF-14
2. Safety Evaluation
3. Federal Register Notice

cc w/encls.: See next page

No legal objection to issue of amendment or FR notice SER review not requested

OFFICE	DL:LB#2/PM	DL:LB#2/LA	DL:LB#2/BC	OELD	DL:AD/L		
SURNAME	RPerch:pt	EHylton	ASchwencer	Cutchin	T Novak		
DATE	12/6/82	12/6/82	12/ /82	12/8/82	12/16/82		

NRC FORM 318 (10-80) NRCM 0240

OFFICIAL RECORD COPY

USGPO: 1981—335-960

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Attn: Coordinator, State Clearinghouse
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Attn: EIS Coordinator
Region III Office
Curtis Building
6th and Walnut Streets
Philadelphia, Pennsylvania 19106

PENNSYLVANIA POWER & LIGHT COMPANY
ALLEGHENY ELECTRIC COOPERATIVE, INC.
DOCKET NO. 50-387
SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1
AMENDMENT TO FACILITY OPERATING LICENSE

License No. NPF-14
Amendment No. 6

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

   A. The application for amendment filed by the Pennsylvania Power & Light Company, dated August 18, 1982, as amended by Pennsylvania Power & Light Company letter dated August 23, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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 the Facility Operating License No. NPF-14 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 6, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PP&L shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

OFFICE ▶							
SURNAME ▶							
DATE ▶							

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: DEC 1 6 1982

No legal objection to form of amendment. SER review not requested.

OFFICE	DL:LB#2/PM	DL:LB#2/LA	DL:LB#2/BC	OELD			
SURNAME	RPerch:pt	EHylton	ASchwencer	Wilhin			
DATE	12/6/82	12/6/82	12/6/82	12/8/82	12/14/82		

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ATTACHMENT TO LICENSE AMENDMENT NO. 6
FACILITY OPERATING LICENSE NO. NPF-14
DOCKET NO. 50-387

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

REMOVE	INSERT
3/4 1-9	3/4 1-9
3/4 1-10	3/4 1-10
3/4 3-11	3/4 3-11
3/4 3-12	3/4 3-12
3/4 3-15	3/4 3-15
3/4 3-16	3/4 3-16
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32
3/4 3-71	3/4 3-71
3/4 3-72	3/4 3-72
3/4 3-79	3/4 3-79
3/4 3-80	3/4 3-80
3/4 3-87	3/4 3-87
3/4 3-88	3/4 3-88
3/4 3-89	3/4 3-89
3/4 3-90	3/4 3-90
3/4 5-3	3/4 5-3
3/4 5-4	3/4 5-4
3/4 6-29	3/4 6-29
3/4 6-30	3/4 6-30
3/4 8-33	3/4 8-33
3/4 10-1	3/4 10-1
3/4 10-2	3/4 10-2
B3/4 3-7	B3/4 3-7
B3/4 3-8	B3/4 3-8

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.5 All control rod scram accumulators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITIONS 1 or 2:

   1. With one control rod scram accumulator inoperable within 8 hours:

      a) Restore the inoperable accumulator to OPERABLE status, or

      b) Declare the control rod associated with the inoperable accumulator inoperable.

      Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

   2. With more than one control rod scram accumulator inoperable, declare the associated control rod inoperable and:

      a) If any control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one control rod drive pump is operating by inserting at least one withdrawn control rod at least one notch or place the reactor mode switch in the Shutdown position.

      b) Insert the inoperable control rods and disarm the associated directional control valves either:

         1) Electrically, or
         2) Hydraulically by closing the drive water and exhaust water isolation valves.

      Otherwise, be in at least HOT SHUTDOWN within 12 hours.

b. In OPERATIONAL CONDITION 5*:

   1. With one withdrawn control rod with its associated scram accumulator inoperable, insert the affected control rod and disarm the associated directional control valves within one hour, either:

      a) Electrically, or

      b) Hydraulically by closing the drive water and exhaust water isolation valves.

   2. With more than one withdrawn control rod with the associated scram accumulator inoperable or with no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

*At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

a. At least once per 7 days by verifying that the indicated pressure is greater than or equal to 940 psig unless the control rod is inserted and disarmed or scrammed.

b. At least once per 18 months by:

1. Performance of a:

a) CHANNEL FUNCTIONAL TEST of the leak detectors, and

b) CHANNEL CALIBRATION of the pressure detectors, and verifying an alarm setpoint of 940 + 30, - 0 psig on decreasing pressure.

2. Verifying that each individual accumulator check valve maintains the associated accumulator pressure above the alarm set point for greater than or equal to 10 minutes with no control rod drive pump operating.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL(s)[a]	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Level 3	A	2	1, 2, 3	20
2) Low Low, Level 2	B	2	1, 2, 3	20
3) Low Low Low, Level 1	X	2	1, 2, 3	20
b. Drywell Pressure - High	Y,Z	2	1, 2, 3	20
c. Manual Initiation	NA	1	1, 2, 3	24
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	Y[c]	2	1, 2, 3 and *	25
b. Drywell Pressure - High	Y,Z[c]	2	1, 2, 3	25
c. Refuel Floor High Exhaust Duct Radiation - High	**	2	1, 2, 3 and *	25
d. Railroad Access Shaft Exhaust Duct Radiation - High	**	2	1, 2, 3 and *	25
e. Refuel Floor Wall Exhaust Duct Radiation - High	**	2	1, 2, 3 and *	25
f. Manual Initiation	NA	1	1, 2, 3 and *	24

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL(s)[(a)]	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
3. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low, Low, Level 2	B	2	1, 2, 3	21
b. Main Steam Line Radiation - High	C	2	1, 2, 3	21
c. Main Steam Line Pressure - Low	P	2	1	22
d. Main Steam Line Flow - High	D	2/line	1, 2, 3	20
e. Condenser Vacuum - Low	UA	2	1, 2, 3	21
f. Main Steam Line Tunnel Temperature - High	E	2/line	1, 2, 3	21
g. Main Steam Line Tunnel Δ Temperature - High	E	2	1, 2, 3	21
h. Manual Initiation	NA	1	1, 2, 3	24
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. RWCS Δ Flow - High	J	1	1, 2, 3	23
b. RWCS Area Temperature - High	W	3	1, 2, 3	23
c. RWCS Area Ventilation Δ Temp. - High	W	3	1, 2, 3	23
d. SLCS Initiation	(d)	NA	1, 2, 3	23
e. Reactor Vessel Water Level - Low Low, Level 2	B	2	1, 2, 3	23
f. RWCS Δ Pressure - High	J	1	1, 2, 3	23
g. Manual Initiation	NA	1	1, 2, 3	24

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	ISOLATION SIGNAL(s)(a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	A	2	1, 2, 3	26
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	UB	1	1, 2, 3	26
c. RHR Equipment Area Δ Temperature - High	M	1	1, 2, 3	26
d. RHR Area Cooler Temperature - High	M	1	1, 2, 3	26
e. RHR Flow - High	M	1	1, 2, 3	26
f. Manual Initiation	NA	1	1, 2, 3	24

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Be in at least STARTUP within 6 hours.

ACTION 23 - Close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION 24 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

ACTION 26 - Lock the affected system isolation valves closed within one hour and declare the affected system inoperable.

NOTES

* When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

** Actuates valves shown in Table 3.6.5.2-1.

(a) See Specification 3.6.3, Table 3.6.3-1 for valves which are actuated by these isolation signals.

(b) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the channel or trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter. In addition, for the HPCI system and RCIC system isolation, provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is OPERABLE and all required actuation instrumentation for that valve is OPERABLE, one channel may be placed in an inoperable status for up to 8 hours for required surveillance without placing the channel or trip system in the tripped condition.

(c) Also starts the standby gas treatment system.

(d) Closes only RWCU system inlet outboard valve.

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. **CORE SPRAY SYSTEM**		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥-129 inches*	≥-136 inches
b. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
c. Reactor Vessel Steam Dome Pressure - Low	≥ 436 psig, decreasing	≥ 416 psig, decreasing
d. Manual Initiation	NA	NA
2. **LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM**		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥-129 inches*	≥-136 inches
b. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
c. Reactor Vessel Steam Dome Pressure - Low	≥ 436 psig, decreasing	≥ 416 psig, decreasing
d. Manual Initiation	NA	NA
3. **HIGH PRESSURE COOLANT INJECTION SYSTEM**		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -38 inches*	≥ -45 inches
b. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
c. Condensate Storage Tank Level - Low	≥ 36.0 inches above tank bottom	≥ 36.0 inches above tank bottom
d. Reactor Vessel Water Level - High, Level 8	≤ 54 inches	≤ 55.5 inches
e. Suppression Pool Water Level - High	≤ 23 feet 9 inches	≤ 24 feet
f. Manual Initiation	NA	NA

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. AUTOMATIC DEPRESSURIZATION SYSTEM		
a. Reactor Water Level - Low Low Low, Level 1	≥ -129 inches*	≥-136 inches
b. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
c. ADS Timer	≤ 102 seconds	≤ 114 seconds
d. Core Spray Pump Discharge Pressure - High	145 ± 10 psig	145 ± 20 psig
e. RHR LPCI Mode Pump Discharge Pressure - High	125 ±4 psig	125 ± 10 psig
f. Reactor Vessel Water Level-Low, Level 3	≥ 13 inches	≥ 11.5 inches
g. Manual Initiation	NA	NA
5. LOSS OF POWER		
a. 4.16 kv ESS Bus Undervoltage (Loss of Voltage, <20%)	a. 4.16 kv Basis - 840 ± 16.8 volts b. 120 v Basis - 24 ± 0.48 volts c. 0.5 ± 0.1 second time delay	840 ± 59.6 volts 24 ± 1.7 volts 0.5 ± 0.1 second time delay
b. 4.16 kv ESS Bus Undervoltage (Degraded Voltage, <65%)	a. 4.16 kv Basis - 2695 ± 53.9 volts b. 120 v Basis - 77 ± 1.54 volts c. 3.0 ± 0.3 second time delay	2695 ± 191.3 volts 77 ± 5.5 volts 3 ± 0.3 second time delay
c. 4.16 kv ESS Bus Undervoltage (Degraded Voltage, <84%)	a. 4.16 kv Basis - 3483 ± 69.7 volts b. 120 v Basis - 99.5 ± 1.99 volts c. 5 minute ± 30 second time delay without LOCA 10 ± 1.0 second time delay with LOCA	3483 + 247.3, - 69.7 volts 99.5 ± 7.1 volts 5 minutes ± 30 second time delay without LOCA 10 ± 1.0 second time delay with LOCA

* See Bases Figure B 3/4 3-1.

TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1. Reactor Vessel Steam Dome Pressure	2	1	80
2. Reactor Vessel Water Level	2	1	80
3. Suppression Chamber Water Level	2	1	80
4. Suppression Chamber Water Temperature	8, 6 locations	6, 1/location	80
5. Suppression Chamber Air Temperature	2	1	80
6. Primary Containment Pressure	2/range	1/range	80
7. Drywell Temperature	2	1	80
8. Drywell Oxygen/Hydrogen Analyzer	2	1	80
9. Safety/Relief Valve Position Indicators	1/valve*	1/valve*	80
10. Containment High Radiation	2	1	81
11. Noble gas monitors			
a. Reactor Bldg. Vent	1	1	81
b. SGTS Vent	1	1	81
c. Turbine Bldg. Vent	1	1	81

*Acoustic monitor.

TABLE 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION STATEMENT

ACTION 80 -

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 81 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

a. Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

TABLE 3.3.7.9-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION				INSTRUMENTS OPERABLE					
				HEAT		IONIZATION		PHOTO-ELECTRIC	
FIRE ZONE	ROOM OR AREA	ROOM NO.	ELEV.	TOTAL	MIN.	TOTAL	MIN.	TOTAL	MIN.
b.	Reactor Building								
1-1B	Core Spray Pump Room	I-10	645'-0"	NA	NA	5	2	NA	NA
1-1A	Core Spray Pump Room	I-17	645'-0"	NA	NA	7	3	NA	NA
1-1E	RHR Pump Room	I-13	645'-0"	NA	NA	NA	NA	13	7
1-1F	RHR Pump Room	I-14	645'-0"	NA	NA	NA	NA	15	8
1-1D	RCIC Pump Room	I-12	645'-0"	2	1	NA	NA	5	2
1-1C	HPCI Pump Room	I-11	645'-0"	2	1	NA	NA	7	3
1-1G	Sump Room	I-15	645'-0"	NA	NA	2	1	NA	NA
1-2D	Remote Shutdown Panel Rm.	I-109	670'-0"	NA	NA	1	1	NA	NA
1-4C	Switchgear Room	I-406	719'-0"	NA	NA	2	1	NA	NA
1-4D	Switchgear Room	I-407	719'-0"	NA	NA	2	1	NA	NA
1-4A	Containment Access Area	I-401	719'-0"	NA	NA	17	8	NA	NA
1-5F	Load Center Room	I-507	749'-1"	NA	NA	2	1	NA	NA
1-5G	Load Center Room	I-510	749'-1"	NA	NA	2	1	NA	NA
1-2A	Access Area	I-105	670'-0"	NA	NA	4	2	NA	NA
1-3A	Access Area	I-203	683'-0"	NA	NA	4	2	NA	NA
1-3B	Access Area	I-200	683'-0"	NA	NA	9	5	NA	NA
1-3C	Access Area	I-202	683'-0"	NA	NA	NA	NA	13	6
1-4B	Pipe Penetration Room	I-403	719'-1"	NA	NA	2	1	NA	NA
1-4G	Main Steam Piping	I-411	719'-1"	NA	NA	NA	NA	4	2
1-5B	Valve Access Area	I-515	761'-10"	NA	NA	NA	NA	2	1
1-5D	RWCU Pumps & Heat Exchangers	I-501	749'-1"	NA	NA	NA	NA	2	1
1-5E	Penetration Room	I-506	749'-1"	NA	NA	NA	NA	2	1
1-6A	Access Area	I-606	779'-1"	NA	NA	9	4	NA	NA

TABLE 3.3.7.9-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION				INSTRUMENTS OPERABLE					
FIRE ZONE	ROOM OR AREA	ROOM NO.	ELEV.	HEAT TOTAL	HEAT MIN.	IONIZATION TOTAL	IONIZATION MIN.	PHOTO-ELECTRIC TOTAL	PHOTO-ELECTRIC MIN.
Reactor Building (Continued									
1-6D	H&V Equipment Room	I-612	779'-1"	NA	NA	10	5	NA	NA
1-6E	Recirculation Fans Area	I-615	779'-1"	NA	NA	2	1	NA	NA
0-6G	Surge Tank Vault	I-601	779'-4"	NA	NA	2	1	NA	NA
1-7A	H&V Fan and Filter Rooms	I-703	799'-1"	4	2	9	5	2	1
1-7B	Recirculation Fan Room	I-701	799'-1"	NA	NA	2	1	NA	NA
0-8A	Refueling Floor	-	818'-1"	NA	NA	NA	NA	30	15
c. ESSW Pumphouse									
0-51	Pump Room	E-1	685'-6"	NA	NA	6	3	NA	NA
0-52	Pump Room	E-2	685'-6"	NA	NA	6	3	NA	NA
						INFRA-RED (FLAME) TOTAL	INFRA-RED (FLAME) MIN.		
d. Diesel Generator Building									
0-41A	Diesel Generator Rooms and	DG-1 DG-16	660'-0" 677'-0"	22	11	2	1	15	7
0-41C	Diesel Generator Rooms and	DG-2 DG-17	660'-0" 677'-0"	22	11	2	1	15	7
0-41B	Diesel Generator Rooms and	DG-3 DG-18	660'-0" 677'-0"	22	11	2	1	15	7
0-41D	Diesel Generator Rooms and	DG-4 DG-19	660'-0" 677'-0"	22	11	2	1	15	7

*Not accessible.

TABLE 3.3.7.11-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Hydrogen Monitor	2	**	110
2. REACTOR BUILDING VENTILATION MONITORING SYSTEM			
a. Noble Gas Activity Monitor	1##	*	111
b. Iodine Monitor	1	*	112
c. Particulate Monitor	1	*	112
d. Effluent System Flow Rate Monitor	1	*	113
e. Sampler Flow Rate Monitor	1	*	113

TABLE 3.3.7.11-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
3. TURBINE BUILDING VENTILATION MONITORING SYSTEM			
a. Noble Gas Activity Monitor	1##	*	114
b. Iodine Monitor	1	*	112
c. Particulate Monitor	1	*	112
d. Effluent System Flow Rate Monitor	1	*	113
e. Sampler Flow Rate Monitor	1	*	113
4. MAIN CONDENSER OFFGAS PRE-TREATMENT RADIOACTIVITY MONITOR (Prior to Input to Holdup System)			
a. Noble Gas Activity Monitor	1	***	115
5. STANDBY GAS TREATMENT SYSTEM MONITOR			
a. Noble Gas Activity Monitor	1##	#	116
b. Iodine Monitor	1	#	112
c. Particulate Monitor	1	#	112
d. Effluent System Flow Rate Monitor	1	#	113
e. Sampler Flow Rate Monitor	1	#	113

TABLE 3.3.7.11-1 (Continued)

TABLE NOTATION

#During operation of the standby gas treatment system.

##Low-range and mid-range channels of monitor only

* At all times.

** During main condenser offgas treatment system operation.

*** During operation of the main condenser air ejector.

ACTION 110 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of main condenser offgas treatment system may continue for up to 30 days provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

ACTION 111 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 112 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11.2.1.2-1.

ACTION 113 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 114 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours and provided the mechanical vacuum pumps are not operated.

ACTION 115 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, releases to the environment may continue for up to 72 hours provided:

a. The offgas system is not bypassed, and

b. The Turbine Building vent noble gas activity monitor is OPERABLE;

Otherwise, be in at least HOT STANDBY within 12 hours.

ACTION 116 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, suspend release of radioactive effluent via this pathway.

TABLE 4.3.7.11-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor	D	N/A	Q(3)	M	**
2. REACTOR BUILDING VENTILATION MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(2)	Q(1)	*
b. Iodine Monitor	W	M	R(2)	Q(1)	*
c. Particulate Monitor	W	M	R(2)	Q(1)	*
d. Effluent System Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

LIMTING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. For the ADS:

1. With one of the above required ADS valves inoperable, provided the HPCI system, the CSS and the LPCI system are OPERABLE, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to $\leq$ 100 psig within the next 24 hours.

2. With two or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to $\leq$ 100 psig within the next 24 hours.

e. With a CSS header delta P instrumentation channel inoperable, restore the inoperable channel to OPERABLE status with 72 hours or determine the ECCS header delta P locally at least once per 12 hours; otherwise, declare the CSS inoperable.

f. In the event an ECCS system is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

4.5.1 The emergency core cooling systems shall be demonstrated OPERABLE by:

a. At least once per 31 days:

1. For the CSS, the LPCI system, and the HPCI system:

a) Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water by:

1. Venting at the high point vents

2. Performing a CHANNEL FUNCTIONAL TEST of the condensate transfer pump discharge low pressure alarm instrumentation.

b) Verifing that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

2. For the CSS, performance of a CHANNEL FUNCTIONAL TEST of the core spray header delta P instrumentation.

3. For the LPCI system, verifying that at least one LPCI system subsystem cross-tie valve is closed with power removed from the valve operator.

4. For the HPCI system, verifying that the pump flow controller is in the correct position.

b. Verifing that, when tested pursuant to Specification 4.0.5:

1. The two CSS pumps in each subsystem together develop a total flow of at least 6350 gpm against a test line pressure of greater than or equal to 269 psig, corresponding to a reactor vessel steam dome pressure of $\geq$ 105 psig.

2. Each LPCI pump in each subsystem develops a flow of at least 12,200 gpm against a test line pressure of $\geq$ 204 psig, corresponding to a reactor vessel to primary containment differential pressure $\geq$ 20 psid.

3. The HPCI pump develops a flow of at least 5000 gpm against a test line pressure of $\geq$ 1266 psig when steam is being supplied to the turbine at $\overline{920}$, +140, - 20 psig.*

c. At least once per 18 months:

1. For the CSS, the LPCI system, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3/4.6.4 VACUUM RELIEF

LIMITING CONDITION FOR OPERATION

3.6.4 Each pair of suppression chamber - drywell vacuum breakers shall be OPERABLE and closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more vacuum breakers in one pair of suppression chamber - drywell vacuum breakers inoperable for opening but known to be closed, restore the inoperable pair of vacuum breakers to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one suppression chamber - drywell vacuum breaker open, verify the other vacuum breaker in the pair to be closed within 2 hours. Restore the open vacuum breaker to the closed position within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one position indicator of any suppression chamber - drywell vacuum breaker inoperable:

   1. Verify the other vacuum breaker in the pair to be closed within 2 hours and at least once per 15 days thereafter, or

   2. Verify the vacuum breaker(s) with the inoperable position indicator to be closed by conducting a test which demonstrates that the ΔP is maintained at greater than or equal to 0.7 psi for one hour without makeup within 24 hours and at least once per 15 days thereafter.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.4 Each suppression chamber - drywell vacuum breaker shall be:

a. Verified closed at least once per 7 days.

b. Demonstrated OPERABLE:

1. At least once per 31 days and within 2 hours after any discharge of steam to the suppression chamber from the safety-relief valves, by cycling each vacuum breaker through at least one complete cycle of full travel.

2. At least once per 31 days by verifying both position indicators OPERABLE by observing expected valve movement during the cycling test.

3. At least once per 18 months by;

a) Verifying the opening setpoint, from the closed position, to be 0.5 psid $\pm$ 5%, and

b) Verifying both position indicators OPERABLE by performance of a CHANNEL CALIBRATION.

REACTOR PROTECTION SYSTEM ELECTRIC POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.3 Two RPS electric power monitoring assemblies for each inservice RPS MG set or alternate power supply shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one RPS electric power monitoring assembly for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring assembly to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

b. With both RPS electric power monitoring assemblies for an inservice RPS MG set or alternate power supply inoperable, restore at least one electric power monitoring assembly to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

SURVEILLANCE REQUIREMENTS

4.8.4.3 The above specified RPS electric power monitoring assemblies shall be determined OPERABLE:

*a. By performance of a CHANNEL FUNCTIONAL TEST when in COLD SHUTDOWN if not performed within the previous 6 months.

b. At least once per 18 months by demonstrating the OPERABILITY of over-voltage, under-voltage and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints:

	RPS Division A	RPS Division B
1. Over-voltage	$\leq$ 128.3 VAC	$\leq$ 129.5 VAC
2. Under-voltage	$\geq$ 110.7 VAC	$\geq$ 111.9 VAC
3. Under-frequency	$\geq$ 57 Hz	$\geq$ 57 Hz

*Prior to startup after the first refueling shutdown, this Surveillance Requirement is deleted and replaced as follows:

a. At least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.10.1 The provisions of Specifications 3.4.2, 3.6.1.1. 3.6.1.3 and 3.9.1 and Table 1.2 may be suspended to permit the safety valve function of the safety/relief valve to be inoperable, the reactor pressure vessel closure head and the drywell head to be removed and the primary containment air lock doors to be open when the reactor mode switch is in the Startup position during low power PHYSICS TESTS with THERMAL POWER less than 1% of RATED THERMAL POWER and reactor coolant temperature less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during low power PHYSICS TESTS.

ACTION:

With THERMAL POWER greater than or equal to 1% of RATED THERMAL POWER or with the reactor coolant temperature greater than or equal to 200°F, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.1 The THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during low power PHYSICS TESTS.

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups by the rod sequence control system (RSCS) per Specification 3.1.4.2 may be suspended by means of bypass switches for the following tests provided that the rod worth minimizer is OPERABLE per Specification 3.1.4.1:

a. Shutdown margin demonstrations, Specification 4.1.1.

b. Control rod scram, Specification 4.1.3.2.

c. Control rod friction measurements.

d. Startup Test Program with the THERMAL POWER less than 20% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the requirements of the above specification not satisfied, verify that the RSCS is OPERABLE per Specification 3.1.4.2.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints imposed by the RSCS are bypassed, verify;

a. That the RWM is OPERABLE per Specification 3.1.4.1.

b. That movement of control rods from 75% ROD DENSITY to the RSCS low power setpoint is blocked or limited to the approved control rod withdrawal sequence during scram and friction tests.

c. That movement of control rods during shutdown margin demonstrations is limited to the prescribed sequence per Specification 3.10.3.

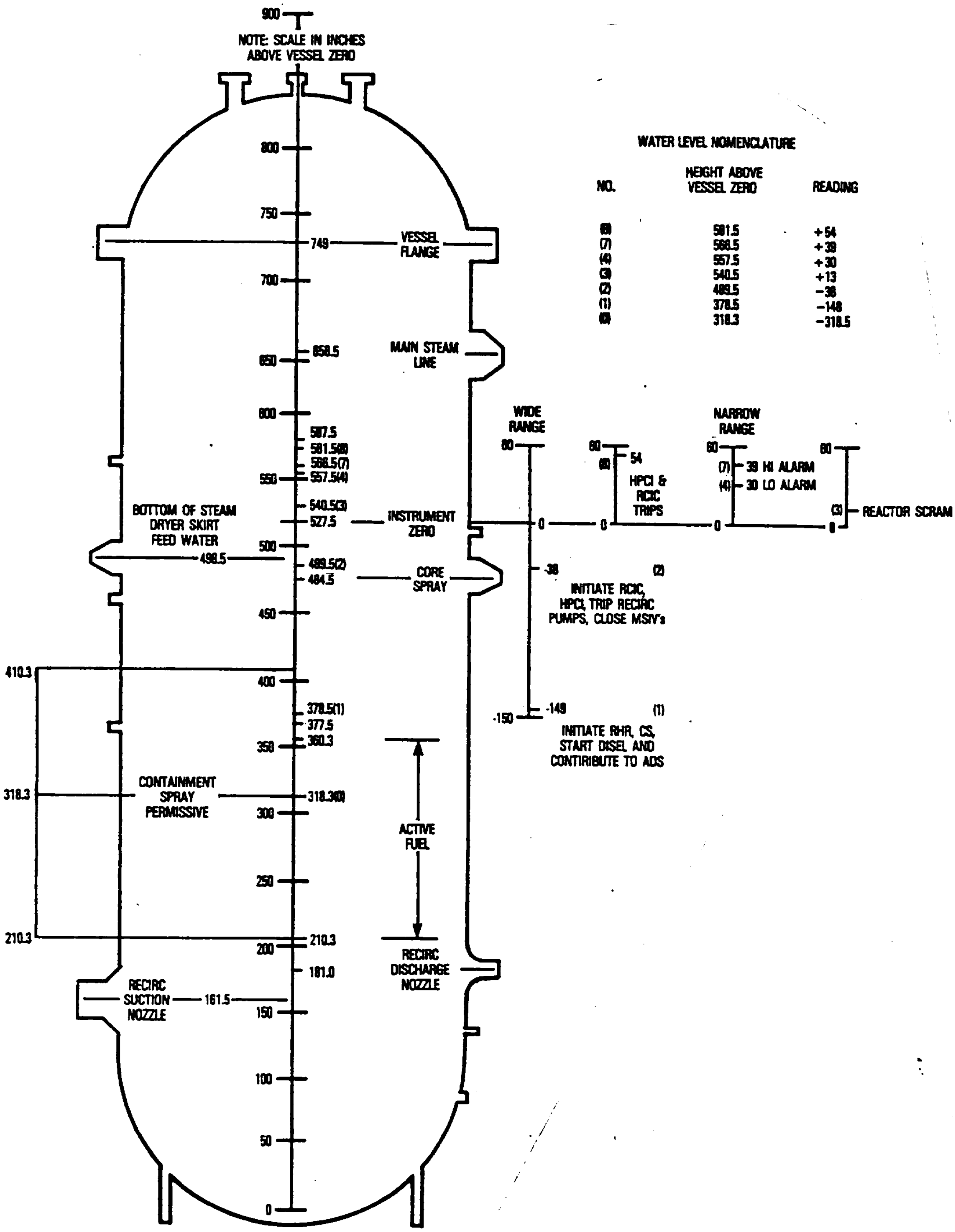
d. Conformance with this specification and test procedures by a second licensed operator or other technically qualified member of the unit technical staff.

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

This specification is provided to ensure that the turbine overspeed protection system instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

The feedwater/main turbine trip system actuation instrumentation is provided to initiate action of the feedwater system/main turbine trip system in the event of failure of feedwater controller under maximum demand.



WATER LEVEL NOMENCLATURE

NO.	HEIGHT ABOVE VESSEL ZERO	READING
(8)	581.5	+54
(7)	566.5	+39
(4)	557.5	+30
(3)	540.5	+13
(2)	489.5	-38
(1)	378.5	-149
(0)	318.3	-318.5

Bases Figure B 3/4 3-1

REACTOR VESSEL WATER LEVEL

SAFETY EVALUATION

AMENDMENT NO. 6 TO LICENSE NPF-14

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

DOCKET NO. 50-387

INTRODUCTION

The licensee proposed changes to the Technical Specifications of the operating license for Susquehanna Steam Electric Station, Unit 1 which are as follows:

a) changes surveillance requirements for vacuum breakers to be consistent with manufacturer's test procedure,

b) establishes minimum discharge pressure for the low pressure coolant injection pump for test purposes,

c) changes ECCS actuation instrumentation setpoint allowable values,

d) revises applicability of SRV's when the vessel is not pressurized,

e) corrects errors in the fire detection instrumentation table, and

f) corrects typographical errors in various technical specifications.

EVALUATION

a) Containment Systems Surveillance Requirement

In Specification 4.6.4.b.3.a), the licensee requested a change in the requirement from "to be less than or equal to 0.5 psid" to read "to be 0.5 psid $\pm$ 5%". The proposed change allows the surveillance requirement to be consistent with the manufacturer's test procedure used to adjust and verify the vacuum breaker opening setpoint. The highest opening setpoint allowed by the proposed change is 0.525 psid. The staff has reviewed the licensee's justification and finds the change acceptable since the containment depressurization analysis assumes the vacuum breakers remain closed until the wetwell to drywell P reaches 3.0 psid, and the impact of higher stress on the vacuum breaker shaft and shaft key from a setpoint of 0.525 psid are below the allowable stresses.

b) Emergency Core Cooling Systems Surveillance Requirements

In Specification 4.5.1.b.2, the licensee requested a change in the requirement from "a test line pressure of 204 psig" to read "a test line pressure of greater than or equal to 204 psig". The proposed change would allow a greater discharge pressure for testing. The staff has reviewed the licensee's justification and finds the change acceptable since the change increases the conservatism in the design system flow rate.

c) Emergency Core Cooling Systems Actuation Instrumentation Setpoints

In Table 3.3.3-2, the licensee requested changes to the trip setpoint and allowable value for the drywell pressure-high, ADS timer, and RHR LPCI mode pump discharge pressure-high trip functions for the automatic depressurization system. The proposed changes to allowable values, resulting from analytic limits, are more conservative than the currently listed allowable values. The proposed changes to the trip setpoints for the above function remain within the allowable ranges. The staff has reviewed the licensee's justification and finds the changes acceptable.

d) Safety/Relief Valves

In Specification 3.4.2, the licensee requested a change of applicability from "OPERATIONAL CONDITIONS 1, 2, and 3." to read "OPERATIONAL CONDITIONS 1, 2 (except during low power physics testing with the reactor pressure vessel closure head removal) and 3.". The change would exempt the SRV's from being required when the vessel is not pressurized, allowing operational flexibility without impacting plant safety. The staff has reviewed the licensee's justification and finds it acceptable, however, the change will be incorporated by changing Specification 3.10.1 under Special Test Exceptions from "The provisions of Specifications 3.6.1.1, 3.6.1.3 and 3.9.1 and Table 1.2 may be suspended to permit the reactor pressure vessel closure head..." to read "The provisions of Specifications 3.4.2, 3.6.1.1, 3.6.1.3 and 3.9.1 and Table 1.2 may be suspended to permit the safety valve function of the safety/relief valve to be inoperable, the reactor pressure vessel closure head...".

e) Fire Detection Instrumentation

In Table 3.3.7.9-1, the licensee requested a change in instruments operable in access area 1-3C from ionization 13 total, 6 minimum to read NA for both columns, and from photoelectric NA total, NA minimum to read 13 total, 6 minimum. The proposed change corrects typographical errors in these columns. The detectors in Fire Zone 1-3C are photoelectric per the original design and installation of the system. The staff finds this change acceptable.

f) Typographical Errors

The licensee has proposed changes to various technical specifications/ tables/figures to correct typographical errors. The staff has reviewed these changes and finds them acceptable. The changes have no impact on plant safety. The proposed changes are:

Specification 3.1.3.5b.1.
Change "withdrawn control rod scram with..." to read "withdrawn control rod with..."

Table 3.3.2-1, item 2.b
Change "Drywall Pressure" to read "Drywell Pressure".

Table 3.3.2-1, items 3.g, 4.c, 7.c
Change "W Temperature" to read " ΔTemperature".

Table 3.3.2-1, item 4.a
Change "W Flow" to read " ΔFlow".

Table 3.3.2-1, item 4.f
Change "W Pressure" to read " ΔPressure".

Table 3.3.7.5-1, item 11.c.
Change "Gldg." to read "Bldg.".

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Change "TABLE 3.3.7.12-1" to read "TABLE 3.3.7.11-1".

Page 3/4 3-89
Change "TABLE 4.3.7.11.1 (Continued)" to read "TABLE 3.3.7.11-1 (Continued)".

Specification 4.8.4.3.b.2
Change Under-voltage RPS Division A " >110.7 VAC" to read " $\geq$ 110.7 VAC".

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Under Water Level Nomenclature, change "WEIGHT ABOVE VESSEL ZERO" to read "HEIGHT ABOVE VESSEL ZERO".

ENVIRONMENTAL CONSIDERATION

We have determined that this amendment does not authorize a change in effluent types or total amount nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that this amendment involves action which is insignificant from the standpoint of environmental impact, and, pursuant to 10 CFR Section 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this statement.

CONCLUSION

We have concluded, based on the considerations discussed above, that; (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: DEC 1 6 1982

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-387

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

NOTICE OF ISSUANCE OF AMENDMENT OF FACILITY

OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 6 to Facility Operating License No. NPF-14, issued to Pennsylvania Power & Light Company and Allegheny Electric Cooperative, Inc., for Susquehanna Steam Electric Station, Unit 1 (the facility) located in Luzerne County, Pennsylvania. This amendment grants changes to Technical Specifications to change surveillance requirements for vacuum breakers to be consistent with manufacturer's test procedure, to establish minimum discharge pressure for the low pressure coolant injection pump for testing purposes, to change ECCS actuation instrumentation setpoint allowable values, to provide establishment of SRV acoustic monitoring setpoints based on the startup test program, to revise applicability of SRV's when the vessel is not pressurized, to correct errors in the fire detection instrumentation table, and to correct typographical errors in various technical specifications. This amendment is effective as of the date of issuance.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations. The Commission has made appropriate findings as required by the Act and the Commission's regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

OFFICE ▶							
SURNAME ▶							

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for the amendment dated August 18, 1982, as amended by licensee letter dated August 23, 1982; (2) Amendment No. 6 to License NPF-14 dated December 16, 1982; and (3) the Commission's evaluation dated December 16, 1982. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. 20555, and at the Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701. A copy of items (1), (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 16th day of December 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert L. Perch, Acting Chief
Licensing Branch No. 2
Division of Licensing

*SEE PREVIOUS CONCURRENCES

OFFICE	DL:LB#2/PM	DL:LB#2/LA	DL:LB#2/BC	OELD			
SURNAME	RLPerch:kw*	EHylton*	ASchwencer*	MCutchin*			
DATE	12/6/82	12/6/82	12/6/82	12/8/82			

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for the amendment dated August 18, 1982, as amended by licensee letter dated August 23, 1982; (2) Amendment No. 6 to License NPF-14 dated 1982; and (3) the Commission's evaluation dated , 1982. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. 20555, and at the Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701. A copy of items (1), (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this day of December 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

No legal objection to filing of FR notice SER review not requested

OFFICE	DL:LB#2/PM	DL:LB#2/LA	DL:LB#2/BC	OELD			
SURNAME	RPerch:pt	EHylton	ASchwencer	Cutchin			
DATE	12/6/82	12/6/82	12/ /82	12/8/82			

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