August 4, 1[illegible]

Docket Nos. 50-387
and 50-388

Mr. Robert G. Byram
Senior Vice President-Nuclear
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Dear Mr. Byram:

SUBJECT: ELIMINATION OF TEMPERATURE LEAK DETECTION ISOLATION FUNCTION IN RHR ROOM, SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (PLA-3485 AND PLA-3732) (TAC NOS. M80105 AND M80106)

The Commission has issued the enclosed Amendment No. 127 to Facility Operating License No. NPF-14 and Amendment No. 96 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Units 1 and 2. These amendments are in response to your letter dated January 14, 1991, as supplemented by your letter of March 3, 1992.

These amendments revise the Technical Specifications to delete the temperature leak detection system isolation function (ambient and differential) in the Residual Heat Removal pump rooms.

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

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

Enclosures:
1. Amendment No. 127 to License No. NPF-14
2. Amendment No. 96 to License No. NPF-22
3. Safety Evaluation

cc w/enclosures:
See next page

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

August 4, 1993

Docket Nos. 50-387
and 50-388

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Senior Vice President-Nuclear
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

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SUBJECT: ELIMINATION OF TEMPERATURE LEAK DETECTION ISOLATION FUNCTION IN RHR ROOM, SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (PLA-3485 AND PLA-3732) (TAC NOS. M80105 AND M80106)

The Commission has issued the enclosed Amendment No. 127 to Facility Operating License No. NPF-14 and Amendment No. 96 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Units 1 and 2. These amendments are in response to your letter dated January 14, 1991, as supplemented by your letter of March 3, 1992.

These amendments revise the Technical Specifications to delete the temperature leak detection system isolation function (ambient and differential) in the Residual Heat Removal pump rooms.

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

Sincerely,

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

Enclosures:
1. Amendment No. 127 to License No. NPF-14
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3. Safety Evaluation

cc w/enclosures:
See next page

Mr. Robert G. Byram
Pennsylvania Power & Light Company

Susquehanna Steam Electric Station,
Units 1 & 2

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

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-387

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 127
License No. NPF-14

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

   A. The application for the amendment filed by the Pennsylvania Power & Light Company, dated January 14, 1991, as supplemented by letter dated March 3, 1992, 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 set forth in 10 CFR Chapter I;

   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.

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P PDR

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. 127 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Michael L. Boyle, Acting 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: August 4, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 127

FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

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

REMOVE	INSERT
3/4 3-15	3/4 3-15
3/4 3-16	3/4 3-16*
3/4 3-19	3/4 3-19*
3/4 3-20	3/4 3-20
3/4 3-21	3/4 3-21*
3/4 3-22	3/4 3-22
3/4 3-25	3/4 3-25*
3/4 3-26	3/4 3-26

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/HEAD SPRAY 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 Flow - High	M	1	1, 2, 3	26
d. Manual Initiation	NA	1	1, 2, 3	24
e. Drywell Pressure - High	Z	2	1, 2, 3	26

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION STATEMENTS

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 1 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 1 hour.

ACTION 26 - Lock the affected system isolation valves closed within 1 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 dampers shown in Table 3.6.5.2-1.

*** When VENTING or PURGING the drywell per Specification 3.11.2.8.

(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.

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION Continued)		
d. RCIC Equipment Room Temperature - High	≤ 167°F**	≤ 174°F**
e. RCIC Equipment Room Δ Temperature - High	≤ 89°F	≤98°F*
f. RCIC Pipe Routing Area Temperature - High	< 167°F##	< 174°F##
g. RCIC Pipe Routing Area Δ Temperature - High	< 89°F##	< 98°F##*
h. RCIC Emergency Area Cooler Temperature - High	≤ 147°F	≤ 154°F
i. Manual Initiation	NA	NA
j. Drywell Pressure - High	≤1.72 psig	≤1.88 psig
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION		
a. HPCI Steam Line Flow - High	≤ 350 inches H_2O	≤ 367 inches H_2O
b. HPCI Steam Supply Pressure - Low	> 104 psig	> 90 psig
c. HPCI Turbine Exhaust Diaphragm Pressure - High	< 10 psig	< 20 psig
d. HPCI Equipment Room Temperature - High	< 167°F	< 174°F
e. HPCI Equipment Room Δ Temperature - High	< 89°F	< 98°F
f. HPCI Emergency Area Cooler Temperature - High	< 147°F	< 154°F
g. HPCI Pipe Routing Area Temperature - High	< 167°F##	< 174°F##

*These trip functions need not be OPERABLE from October 19, 1989 to January 19, 1990.

Effective Date: October 20, 1989
Amendment No. 94 DEC 6 1989

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
h. HPCI Pipe Routing Area Δ Temperature - High	≤ 89°F##	≤ 98°F## **
i. Manual Initiation	NA	NA
j. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
7. RHR SYSTEM SHUTDOWN COOLING/HEAD SPRAY MODE ISOLATION		
a. Reactor Vessel Water Level - Low, Level 3	≥ 13.0 inches*	≥ 11.5 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 98 psig	≤ 108 psig
c. RHR Flow - High	≤ 25,000 gpm	≤ 26,000 gpm
d. Manual Initiation	NA	NA
e. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig

* See Bases Figure B 3/4 3-1.
\# Lower setpoints for TSH-G33-1N600 E, F and TDSH-G33-1N602 E, F.
\## 15 minute time delay.
** This trip function need not be OPERABLE from October 19, 1989 to January 19, 1990.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Level 3	≤10[a]
2) Low Low, Level 2	≤1.0*/≤10[a]**
3) Low Low Low, Level 1	≤ 10[a]
b. Drywell Pressure - High	≤10[a]
c. Manual Initiation	NA
d. SGTS Exhaust Radiation - High[b]	≤10[a]
e. Main Steam Line Radiation - High[b]	≤10[a]
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level-Low Low, Level 2	≤10[a]
b. Drywell Pressure - High	≤10[a]
c. Refuel Floor High Exhaust Duct Radiation - High[b]	≤10[a]
d. Railroad Access Shaft Exhaust Duct Radiation - High[b]	≤10[a]
e. Refuel Floor Wall Exhaust Duct Radiation -High[b]	≤10[a]
f. Manual Initiation	NA
3. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level- Low Low Low, Level 1	≤10[a]
b. Main Steam Line Radiation - High[b]	≤1.0*/≤10[a]**
c. Main Steam Line Pressure - Low	≤1.0*/≤10[a]**
d. Main Steam Line Flow-High	≤0.5*/≤10[a]**
e. Condenser Vacuum - Low	NA
f. Reactor Building Main Steam Line Tunnel Temperature - High	NA
g. Reactor Building Main Steam Line Tunnel Δ Temperature - High	NA
h. Manual Initiation	NA
i. Turbine Building Main Steam Line Tunnel Temperature - High	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. RWCU Δ Flow - High	≤10[a]##
b. RWCU Area Temperature - High	NA
c. RWCU Area Ventilation Temperature ΔT - High	NA
d. SLCS Initiation	NA
e. Reactor Vessel Water Level - Low Low, Level 2	≤10[a]
f. RWCU Flow - High	NA
g. Manual Initiation	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Δ Pressure - High	≤10[a]##
b. RCIC Steam Supply Pressure - Low	≤10[a]
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
d. RCIC Equipment Room Temperature - High	NA

effective upon startup following the first refueling outage

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (seconds)#
e. RCIC Equipment Room Δ Temperature - High	NA
f. RCIC Pipe Routing Area Temperature - High	NA
g. RCIC Pipe Routing Area Δ Temperature - High	NA
h. RCIC Emergency Area Cooler Temperature - High	NA
i. Manual Initiation	NA
j. Drywell Pressure - High	≤10[a]
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION	
a. HPCI Steam Flow - High	≤10[a]####
b. HPCI Steam Supply Pressure - Low	≤10[a]
c. HPCI Turbine Exhaust Diaphragm Pressure - High	NA
d. HPCI Equipment Room Temperature - High	NA
e. HPCI Equipment Room Δ Temperature - High	NA
f. HPCI Emergency Area Cooler Temperature - High	NA
g. HPCI Pipe Routing Area Temperature - High	NA
h. HPCI Pipe Routing Area Δ Temperature - High	NA
i. Manual Initiation	NA
j. Drywell Pressure - High	≤10[a]
7. RHR SYSTEM SHUTDOWN COOLING/HEAD SPRAY MODE ISOLATION	
a. Reactor Vessel Water Level - Low, Level 3	≤10[a]
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	NA
c. RHR Flow - High	NA
d. Manual Initiation	NA
e. Drywell Pressure - High	≤ 10[a]

(a) The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes the delay for diesel generator starting assumed in the accident analysis.

(b) Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

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

** 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 Table 3.6.3-1 and 3.6.5.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

\#\# With time delay of 45 seconds.

\#\#\# With time delay of 3 seconds.

\#\#\#\# With time delay of 3 seconds.

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
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Δ Pressure - High	NA	M	Q	1, 2, 3
b. RCIC Steam Supply Pressure - Low	NA	M	Q	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA	M	Q	1, 2, 3
d. RCIC Equipment Room Temperature - High	NA	M	Q	1, 2, 3
e. RCIC Equipment Room Δ Temperature - High	NA	M	Q	1, 2, 3*
f. RCIC Pipe Routing Area Temperature - High	NA	M	Q	1, 2, 3
g. RCIC Pipe Routing Area Δ Temperature - High	NA	M	Q	1, 2, 3*
h. RCIC Emergency Area Cooler Temperature - High	NA	M	Q	1, 2, 3
i. Manual Initiation	NA	R	NA	1, 2, 3
j. Drywell Pressure - High	NA	M	R	1, 2, 3
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Δ Pressure - High	NA	M	Q	1, 2, 3
b. HPCI Steam Supply Pressure - Low	NA	M	Q	1, 2, 3
c. HPCI Turbine Exhaust Diaphragm Pressure - High	NA	M	Q	1, 2, 3

*These trip functions need not be OPERABLE from October 19, 1989 to January 19, 1990.

Effective Date: October 20, 1989
Amendment No. 94

DEC 6 1989

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
HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION (Continued)				
d. HPCI Equipment Room Temperature - High	NA	M	Q	1, 2, 3
e. HPCI Equipment Room Δ Temperature - High	NA	M	Q	1, 2, 3
f. HPCI Emergency Area Cooler Temperature - High	NA	M	Q	1, 2, 3
g. HPCI Pipe Routing Area Temperature - High	NA	M	Q	1, 2, 3
h. HPCI Pipe Routing Area Δ Temperature - High	NA	M	Q	1, 2, 3 ****
i. Manual Initiation	NA	R	NA	1, 2, 3
j. Drywell Pressure - High	NA	M	R	1, 2, 3
7. RHR SYSTEM SHUTDOWN COOLING/HEAD SPRAY MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	S	M	R	1, 2, 3
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA	M	Q	1, 2, 3
c. RHR Flow - High	S	M	R	1, 2, 3
d. Manual Initiation	NA	R	NA	1, 2, 3
e. Drywell Pressure - High	NA	M	R	1, 2, 3

* When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.
** When reactor steam dome pressure > 1043 psig and/or any turbine stop valve is open.
*** When VENTING or PURGING the drywell per Specification 3.11.2.8.
**** This trip function need not be OPERABLE from October 19, 1989 to January 19, 1990.

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

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 96
License No. NPF-22

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

   A. The application for the amendment filed by the Pennsylvania Power & Light Company, dated January 14, 1991, as supplemented by letter dated March 3, 1992, 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 set forth in 10 CFR Chapter I;

   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-22 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. 96 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Michael L. Boyle, Acting 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: August 4, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 96

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

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

REMOVE	INSERT
3/4 3-15	3/4 3-15
3/4 3-16	3/4 3-16*
3/4 3-19	3/4 3-19*
3/4 3-20	3/4 3-20
3/4 3-21	3/4 3-21*
3/4 3-22	3/4 3-22
3/4 3-25	3/4 3-25*
3/4 3-26	3/4 3-26

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/HEAD SPRAY 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 Flow - High	M	1	1, 2, 3	26
d. Manual Initiation	NA	1	1, 2, 3	24
e. Drywell Pressure - High	Z	2	1, 2, 3	26

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION STATEMENTS

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 1 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 1 hour.

ACTION 26 - Lock the affected system isolation valves closed within 1 hour and declare the affected system inoperable.

TABLES NOTATIONS

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

** Actuates dampers shown in Table 3.6.5.2-1.

*** When VENTING or PURGING the drywell per Specification 3.11.2.8.

(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.

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION (Continued)		
d. RCIC Equipment Room Temperature - High	≤ 167°F	≤ 174°F
e. RCIC Equipment Room Δ Temperature - High	≤ 89°F	≤ 98°F*
f. RCIC Pipe Routing Area Temperature - High	≤ 167°F##	≤ 174°F##
g. RCIC Pipe Routing Area Δ Temperature - High	≤ 89°F##	≤ 98°F##*
h. RCIC Emergency Area Cooler Temperature - High	≤ 147°F	≤ 154°F
i. Manual Initiation	NA	NA
j. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION		
a. HPCI Steam Line Flow - High	≤ 275 inches H_2O	≤ 292 inches H_2O
b. HPCI Steam Supply Pressure - Low	≥ 104 psig	≥ 90 psig
c. HPCI Turbine Exhaust Diaphragm Pressure - High	≤ 10 psig	≤ 20 psig
d. HPCI Equipment Room Temperature - High	≤ 167°F	≤ 174°F
e. HPCI Equipment Room Δ Temperature - High	≤ 89°F	≤ 98°F
f. HPCI Emergency Area Cooler Temperature - High	≤ 147°F	≤ 154°F
g. HPCI Pipe Routing Area Temperature - High	≤ 167°F##	≤ 174°F##
h. HPCI Pipe Routing Area Δ Temperature - High	≤ 89°F##	≤ 98°F##*
i. Manual Initiation	NA	NA
j. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig

*These trip functions need not be OPERABLE from October 19, 1989 to January 19, 1990.

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TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
7. RHR SYSTEM SHUTDOWN COOLING/HEAD SPRAY MODE ISOLATION		
a. Reactor Vessel Water Level - Low, Level 3	≥ 13.0 inches *	≥ 11.5 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 98 psig	≤ 108 psig
c. RHR Flow - High	≤ 25,000 gpm	≤ 26,000 gpm
d. Manual Initiation	NA	NA
e. Drywell Pressure - High	≤ 1.72 psig	≤ 1.88 psig

* See Bases Figure B 3/4 3-1.
\# Lower setpoints for TSH-G33-2N600 E, F and TDSH-G33-2N602 E, F.
\## 15 minute time delay.

TABLE 3.3.2-3
ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Level 3	≤10[a]
2) Low Low, Level 2	≤1.0°/≤10[a]°°
3) Low Low Low, Level 1	≤10[a]
b. Drywell Pressure - High	≤10[a]
c. Manual Initiation	NA
d. SGTS Exhaust Radiation - High[b]	≤10[a]
e. Main Steam Line Radiation - High[b]	≤10[a]
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low Low, Level 2	≤10[a]
b. Drywell Pressure - High	≤10[a]
c. Refuel Floor High Exhaust Duct Radiation - High[b]	≤10[a]
d. Railroad Access Shaft Exhaust Duct Radiation - High (b)	≤10[a]
e. Refuel Floor Wall Exhaust Duct Radiation - High[b]	≤10[a]
f. Manual Initiation	NA
3. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level - Low Low Low, Level 1	≤10[a]
b. Main Steam Line Radiation - High[b]	≤1.0°/≤10[a]°°
c. Main Steam Line Pressure - Low	≤1.0°/≤10[a]°°
d. Main Steam Line Flow - High	≤0.5°/≤10[a]°°
e. Condenser Vacuum - Low	NA
f. Reactor Building Main Steam Line Tunnel Temperature - High	NA
g. Reactor Building Main Steam Line Tunnel Δ Temperature - High	NA
h. Manual Initiation	NA
i. Turbine Building Main Steam Line Tunnel Temperature - High	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. RWCU Δ Flow - High	≤10[a]##
b. RWCU Area Temperature - High	NA
c. RWCU Area Ventilation Temperature ΔT - High	NA
d. SLCS Initiation	NA
e. Reactor Vessel Water Level - Low Low, Level 2	≤10[a]
f1. RWCU Flow - High	NA
f2. Non-Regenerative Heat Exchanger Discharge Temperature - High	NA
g. Manual Initiation	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Δ Pressure - High	≤10[a]###
b. RCIC Steam Supply Pressure - Low	≤10[a]
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
d. RCIC Equipment Room Temperature - High	NA
e. RCIC Equipment Room Δ Temperature - High	NA
f. RCIC Pipe Routing Area Temperature - High	NA
g. RCIC Pipe Routing Area Δ Temperature - High	NA
h. RCIC Emergency Area Cooler Temperature - High	NA
i. Manual Initiation	NA
j. Drywell Pressure - High	≤10[a]

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (seconds)#
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION	
a. HPCI Steam Flow - High	≤10[(a)]####
b. HPCI Steam Supply Pressure - Low	≤10[(a)]
c. HPCI Turbine Exhaust Diaphragm Pressure - High	NA
d. HPCI Equipment Room Temperature - High	NA
e. HPCI Equipment Room Δ Temperature - High	NA
f. HPCI Emergency Area Cooler Temperature - High	NA
g. HPCI Pipe Routing Area Temperature - High	NA
h. HPCI Pipe Routing Area Δ Temperature - High	NA
i. Manual Initiation	NA
j. Drywell Pressure - High	≤10[(a)]
7. RHR SYSTEM SHUTDOWN COOLING/HEAD SPRAY MODE ISOLATION	
a. Reactor Vessel Water Level - Low, Level 3	≤10[(a)]
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	NA
c. RHR Flow - High	NA
d. Manual Initiation	NA
e. Drywell Pressure - High	≤10[(a)]

(a) The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes the delay for diesel generator starting assumed in the accident analysis.

(b) Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

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

** 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 Table 3.6.3-1 and 3.6.5.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

\#\# With time delay of 45 seconds.

\#\#\# With time delay of 3 seconds.

\#\#\#\# With time delay of 3 seconds.

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
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Δ Pressure - High	NA	M	Q	1, 2, 3
b. RCIC Steam Supply Pressure - Low	NA	M	Q	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA	M	Q	1, 2, 3
d. RCIC Equipment Room Temperature - High	NA	M	Q	1, 2, 3
e. RCIC Equipment Room Δ Temperature - High	NA	M	Q	1, 2, 3*
f. RCIC Pipe Routing Area Temperature - High	NA	M	Q	1, 2, 3
g. RCIC Pipe Routing Area Δ Temperature - High	NA	M	Q	1, 2, 3*
h. RCIC Emergency Area Cooler Temperature - High	NA	M	Q	1, 2, 3
i. Manual Initiation	NA	R	NA	1, 2, 3
j. Drywell Pressure - High	NA	M	R	1, 2, 3
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Δ Pressure - High	NA	M	Q	1, 2, 3
b. HPCI Steam Supply Pressure - Low	NA	M	Q	1, 2, 3
c. HPCI Turbine Exhaust Diaphragm Pressure - High	NA	M	Q	1, 2, 3

*These trip functions need not be OPERABLE from October 19, 1989 to January 19, 1990.

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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
HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION (Continued)				
d. HPCI Equipment Room Temperature - High	NA	M	Q	1, 2, 3
e. HPCI Equipment Room Δ Temperature - High	NA	M	Q	1, 2, 3
f. HPCI Emergency Area Cooler Temperature - High	NA	M	Q	1, 2, 3
g. HPCI Pipe Routing Area Temperature - High	NA	M	Q	1, 2, 3
h. HPCI Pipe Routing Area Δ Temperature - High	NA	M	Q	1, 2, 3****
i. Manual Initiation	NA	R	NA	1, 2, 3
j. Drywell Pressure - High	NA	M	R	1, 2, 3
7. RHR SYSTEM SHUTDOWN COOLING/HEAD SPRAY MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	S	M	R	1, 2, 3
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA	M	Q	1, 2, 3
c. RHR Flow - High	S	M	R	1, 2, 3
d. Manual Initiation	NA	R	NA	1, 2, 3
e. Drywell Pressure - High	NA	M	R	1, 2, 3

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

** When reactor steam dome pressure > 1043 psig and/or any turbine stop valve is open.

*** When VENTING or PURGING the drywell per Specification 3.11.2.8.

**** This trip function need not be OPERABLE from October 19, 1989 to January 19, 1990.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.127TO FACILITY OPERATING LICENSE NO. NPF-14

AMENDMENT NO. 96 TO FACILITY OPERATING LICENSE NO. NPF-22

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-387 AND 388

1.0 INTRODUCTION

By letter dated January 14, 1991, as supplemented by letter dated March 3, 1992, the Pennsylvania Power and Light Company (the licensee) submitted a request for changes to the Susquehanna Steam Electric Station, Units 1 and 2, Technical Specifications (TS). The requested changes would remove the automatic isolation function of the Residual Heat Removal System (RHRS) based on high RHR room ambient and differential temperatures. This isolation function is part of the Primary Containment and Reactor Vessel Isolation Control System (PCRVICS) and is intended to isolate the RHRS in the event of a steam leak within the RHRS rooms.

2.0 BACKGROUND

The RHRS rooms at SSES, Units 1 and 2, are equipped with instrumentation to sense parameters indicative of a small steam leak in the RHRS and provide alarms and in some cases an isolation signal to secure the leak, thereby preventing a line break. One such system utilizes sensors which can detect high ambient and differential temperatures within the room. In the event of a line break within an RHRS room, a hot water/steam mixture would elevate the temperature of the room. The temperature sensors within the room are designed to be sensitive to the ensuing temperature rise as an indication of a leak in the RHRS. If the temperature increase within the room reaches the detector setpoint for the system, the RHRS is automatically isolated and an alarm in the control room alerts operators to the apparent system leak. The automatic isolation function of this system minimizes the coolant loss to outside primary containment.

The basis for the temperature leak detection instrumentation is described in the Final Safety Analysis Report (FSAR) Section 5.2.5.1.3(1). It states that

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the design of this system should be capable of detecting a leak rate which is less than the technical specification leakage rate limit of 5 gpm. Consideration is given with regard to the isolation temperature setpoints to prevent inadvertent isolation during post-LOCA conditions. The margin between the isolation setpoints and the room temperature during post-LOCA conditions must also account for additional factors such as instrument drift, instrument tolerances, and worst-case seasonal conditions (i.e., winter or summer). The technical specification bases section does not explain the purpose of the RHRS temperature isolation function.

As part of the licensee's commitment to resolve an NRC inspection finding at SSES, Units 1 and 2, an analysis found that the temperature setpoints for the RHRS room isolation function were inadequate for the detection of a small steam leak. The RHRS room temperature calculated by analysis did not increase sufficiently for a 5 gpm leak rate since the steam condensing mode of the RHRS has been eliminated and there are no high energy lines in the RHRS room. The findings of the licensee indicate that RHRS room temperatures do not provide sufficient sensitivity to serve as a system isolation parameter. Consequently, PP&L proposes deleting the isolation function from the RHRS room temperature instrumentation.

3.0 EVALUATION

The initial design of the RHRS included the capability to operate in the steam condensing mode. However, this mode is currently disabled as the piping associated with this mode of operation was abandoned in-place several years ago. The RHRS is equipped with an interlock to prevent exposing the RHR system piping to elevated RCS pressures. Operability of this interlock is required when the average reactor coolant temperature exceeds 200°F as per plant Technical Specification 3/4.3.2, Isolation Actuation Instrumentation. The isolation function of the interlock automatically closes or prevents opening the RHRS suction isolation valves with the RCS pressure greater than 98 psig (336°F). The discharge lines for the RHR system are protected from excessive RCS pressure by check valves and safety valves. The safety valves have the capacity to relieve any bypass around the discharge check valves.

Without the steam condensing mode of operation, the only possible period over which the RHRS could operate with reactor coolant of sufficient energy to elevate the room temperature by release of a mixture of steam and saturated water is during a short interval following the initiation of shutdown cooling. After the RHRS is placed in this mode of operation, the coolant temperature will decrease to below 212°F within a few hours. Therefore, the length of time over which the temperature sensors could adequately detect and isolate the RHRS is limited to this brief period.

The licensee evaluated the possibility of adjusting the design basis leak rate and/or lowering the isolation setpoint temperatures. By analysis, it was

determined that a leak rate of 25 gpm could be detected within a period of 4 hours provided the current setpoints were lowered 24 to 33 degrees Fahrenheit. Lowering the setpoint by this extent would eliminate the margin between the calculated post-LOCA temperatures and the isolation setpoint. Eliminating this margin is unacceptable as it would result in inadvertent isolation of the RHR system following a LOCA. Based on the analysis results, the thermal response within the RHRS rooms is unsuitable for providing the necessary input to the temperature sensors for isolating the RHRS.

Several other systems are installed to detect a leak in the RHRS. Flood detection instrumentation included in this room has the capability of sensing coolant leaked to the room floor to a height of approximately 3 inches. Although a cumulative leak of this magnitude corresponds to several thousand gallons of coolant, the total release to outside the containment remains bounded by the analysis of FSAR Section 15.6.4, Steam System Piping Break Outside Containment. The flood detectors do not automatically isolate the RHRS. However, the alarm response procedure includes steps to verify the existence of a leak within the room and instructs operators to isolate the system upon confirmation of a leak.

In addition to the flood detection instrumentation, there are several other methods for RHRS line break detection and isolation. The following parameters provide an automatic isolation signal.

- Reactor vessel water level of 13 inches or less will isolate the system.
- Reactor coolant pressure signal in excess of 98 psig will automatically isolate the RHRS.
- A high RHRS flow rate in excess of 25,000 gpm will isolate the system.

In accordance with plant operating procedures, a plant operator must enter the RHR pump room every 6 hours in order to check bearing oil levels. If a leak were to occur, operator rounds may discover the leak prior to detection by any of the other leak detection systems within the room. Upon confirmation of a leak the operators could then take steps to isolate the break.

The removal of the isolation function from the temperature sensors does not prevent this instrumentation from detecting a leak. The alarm capability of the ambient and differential room temperature detectors will be retained. Deletion of the isolation function from these detectors may allow for improving the sensitivity of this system by modifying the alarm setpoint. When inadvertent isolation is not a major consideration, the margin between the post-LOCA room temperature and the alarm setpoint can be reduced to provide improved detector sensitivity to a system line break. The licensee plans to reevaluate the RHRS room thermal response to a line break and develop revised alarm setpoints for these temperature detectors.

Elimination of the steam condensing mode has significantly reduced the chances for a steam leak to occur in the RHR system at Susquehanna. Further, without the steam condensing mode of operation, the Susquehanna RHR system is similar to RHR systems typically found on pressurized water reactors, and elimination of the RHR leak detection function is consistent with the requirements that have been imposed by the staff on PWRs. Therefore, the staff finds the proposed deletion of the automatic isolation function from the RHRS room ambient and differential temperature detectors acceptable.

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 (58 FR 34084). Accordingly, the amendments meet 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.

Principal Contributor: P. Rush

Date: August 4, 1993