Docket Nos. 50-387/388

December 30, 1991

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Mr. Harold W. Keiser
Senior Vice President-Nuclear
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Dear Mr. Keiser:

SUBJECT: TECHNICAL SPECIFICATION CHANGES TO REVISE SURVEILLANCE TEST INTERVALS FOR THE REACTOR PROTECTION SYSTEM, SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (TAC NOS. M71185 AND M71186)

The Commission has issued the enclosed Amendment No. 115 to Facility Operating License No. NPF-14 and Amendment No. 84 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 October 27, 1988, as supplemented by letters dated November 9, 1988, January 9, 1989, July 5, 1989, February 22, 1990, March 20, 1991 and July 31, 1991. The November 9, 1988, and January 9, 1989 letters requested that information be treated as proprietary. The information provided in the July 5, 1989 and March 20, 1991 letters related to the clarification of the information provided in the October 27, 1988 letter and did not change the initial proposed no significant hazards determination.

In your letter you requested changes to the Technical Specifications, related to the reactor protection system, revising surveillance test intervals and allowed outage times. Based on the staff review as outlined in the enclosed safety evaluation, we found the proposed changes to the surveillance testing intervals and the allowable outage times to be acceptable.

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

Sincerely,

/S/

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

Enclosures:
1. Amendment No. 115 to License No. NPF-14
2. Amendment No. 84 to License No. NPF-22
3. Safety Evaluation

cc w/enclosures:
See next page

OFC	:PDI-2/LA	:PDI-2/PM	:PDI-2/D	:OGC	:
NAME	:MO'Brien	:JRaleigh:rb	:CMiller	:RBachmann	:
DATE	:12/ /91	:12/13/91	:12/30/91	:12/18/91	:

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

December 30, 1991

Docket Nos. 50-387/388

Mr. Harold W. Keiser
Senior Vice President-Nuclear
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

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Sincerely,

James J. Raleigh

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

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1. Amendment No. 115 to
   License No. NPF-14
2. Amendment No. 84 to
   License No. NPF-22
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. Harold W. Keiser
Pennsylvania Power & Light Company

Susquehanna Steam Electric Station
Units 1 & 2

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

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. 115
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 October 27, 1988, as supplemented November 9, 1988, January 9, 1989, July 5, 1989, February 22, 1990, March 20, 1991, and July 31, 1991, 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-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. 115 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 and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

for James J Raleigh

Charles L. Miller, 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: December 30, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 115

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 3-1	3/4 3-1
3/4 3-2*	3/4 3-2*
3/4 3-5	3/4 3-5
3/4 3-6*	3/4 3-6*
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
B 3/4 3-1	B 3/4 3-1
B 3/4 3-2*	B 3/4 3-2*

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system:
   1. If placing the inoperable channel(s) in the tripped condition would cause a scram, the inoperable channel(s) shall be restored to OPERABLE status within 6 hours or the ACTION required by Table 3.3.1-1 for the affected Functional Unit shall be taken; or
   2. If placing the inoperable channel(s) in the tripped condition would not cause a scram, place the inoperable channel(s) and/or that trip system in the tripped condition within 12 hours.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE channels per Trip System requirement for both trip systems, place at least one trip system* in the tripped condition within 1 hour and take the ACTION required by Table 3.3.1-1. The provisions of Specification 3.0.4 are not applicable for entry into Operational Condition 2 or 3 from Operational Condition 1 for the IRMs or the Neutron Flux - Upscale, Setdown function of the APRMs.

SURVEILLANCE REQUIREMENTS

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

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

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

4.3.1.4 The provisions of Specification 4.0.4 are not applicable for entry into Operational Condition 2 or 3 from Operational Condition 1 for the IRMs or the Neutron Flux - Upscale, Setdown function of the APRMs.

---

* If more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause a scram to occur.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

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

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TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter. Upon determination that a trip setpoint cannot be restored to within its specified value during performance of the CHANNEL CALIBRATION, the appropriate ACTION, 3.3.1a or 3.3.1b, shall be followed.

(b) This function is automatically bypassed when the reactor mode switch is in the Run position.

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

(d) The non-coincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 4 APRMS and 6 IRMS.

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

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

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

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

(i) With any control rod withdrawn.*

(j) This function shall be automatically bypassed when turbine first stage pressure is less than 108 psig or 17% of the value of first stage pressure in psia at valves wide open (V.W.O.) steam flow, equivalent to THERMAL POWER of about 24% of RATED THERMAL POWER.

(k) Also actuates the EOC-RPT system.

---

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

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, Setdown	NA
b. Flow Biased Simulated Thermal Power - Upscale	≤ 0.09**
c. Fixed Neutron Flux - Upscale	≤ 0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 0.55
4. Reactor Vessel Water Level - Low, Level 3	≤ 1.05
5. Main Steam Line Isolation Valve - Closure	≤ 0.06
6. Main Steam Line Radiation - High	NA
7. Drywell Pressure - High	NA
8. Scram Discharge Volume Water Level - High	
a. Level Transmitter	NA
b. Float Switch	NA
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	NA
12. Manual Scram	NA

*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.
**Not including simulated thermal power time constant.
#Measured from actuation of fast-acting solenoid.

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	SA SA	2 3, 4, 5
b. Inoperative	NA	S/U[c], W	NA	2, 3, 4, 5
2. Avg. Power Range Monitor[f]:				
a. Neutron Flux - Upscale, Setdown	S/U,S,[b] S	S/U[c], W W	SA SA	2 3, 5
b. Flow Biased Simulated Thermal Power-Upscale	S,D[g]	S/U[c], Q	W[d][e], SA, R[h]	1
c. Fixed Neutron Flux - Upscale	S	S/U[c], Q	W[d], SA	1
d. Inoperative	NA	S/U[c], Q	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	NA	Q	Q	1, 2
4. Reactor Vessel Water Level - Low, Level 3	S	NA	Q	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
6. Main Steam Line Radiation - High	S	Q	R	1, 2[i]
7. Drywell Pressure - High	NA	Q	R	1, 2

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High				
a. Level Transmitter	NA	Q	R	1, 2, 5[j]
b. Float Switch	NA	Q	R	1, 2, 5[j]
9. Turbine Stop Valve - Closure	NA	Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	Q	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	NA	1, 2, 3, 4, 5

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

(b) The IRM and SRM channels shall be determined to overlap for at least 0.5 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 0.5 decades during each controlled shutdown, if not performed within the previous 7 days.

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

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

(g) Verify measured core flow to be greater than or equal to established core flow at the existing loop flow.

(h) This calibration shall consist of verifying the 6 ± 1 second simulated thermal power time constant.

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

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

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. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

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 accident analysis. 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) in place, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

CHANNEL FUNCTIONAL TEST frequencies and allowed outage times (AOTs) for repair and surveillance testing are based on General Electric report NEDC-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," dated March, 1988. The conclusion of this report is that fewer challenges to safety-related equipment, due to less frequent testing of the RPS, conservatively results in a decrease in core damage frequency. The 6 hour AOT for testing and the 12 hour AOT for repair of one trip system provide enough margin so as not to create an undue stress on personnel. The more restrictive 6 hour repair AOT (Action 1.a) reflects the potential that both trip systems are degraded.

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. 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. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 10 second delay. It follows that checking the valve speeds and the 10 second time for emergency power establishment will establish the response time for the isolation functions. However, to enhance overall system reliability and to monitor instrument channel response time trends, the isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME.

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 equal to or less than the drift allowance assumed 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.

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 equal to or less than the drift allowance assumed for each trip in the safety analyses.

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

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. 84
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 October 27, 1988, as supplemented November 9, 1988, January 9, 1989, July 5, 1989, February 22, 1990, March 20, 1991, and July 31, 1991, 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. 84 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 and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

for James J Raleigh

Charles L. Miller, 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: December 30, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 84

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

REMOVE	INSERT
3/4 3-1	3/4 3-1
3/4 3-2*	3/4 3-2*
3/4 3-5	3/4 3-5
3/4 3-6*	3/4 3-6*
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
B 3/4 3-1	B 3/4 3-1
B 3/4 3-2*	B 3/4 3-2*

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system:
   1. If placing the inoperable channel(s) in the tripped condition would cause a scram, the inoperable channel(s) shall be restored to OPERABLE status within 6 hours or the ACTION required by Table 3.3.1-1 for the affected Functional Unit shall be taken; or
   2. If placing the inoperable channel(s) in the tripped condition would not cause a scram, place the inoperable channel(s) and/or that trip system in the tripped condition within 12 hours.
b. With the number of OPERABLE channels less than required by the Minimum OPERABLE channels per Trip System requirement for both trip systems, place at least one trip system* in the tripped condition within 1 hour and take the ACTION required by Table 3.3.1-1. The provisions of Specification 3.0.4 are not applicable for entry into Operational Condition 2 or 3 from Operational Condition 1 for the IRMs or the Neutron Flux - Upscale, Setdown function of the APRMs.

SURVEILLANCE REQUIREMENTS

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

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

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

4.3.1.4 The provisions of Specification 4.0.4 are not applicable for entry into Operational Condition 2 or 3 from Operational Condition 1 for the IRMs or the Neutron Flux - Upscale, Setdown function of the APRMs.

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* If more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause a scram to occur.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

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

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter. Upon determination that a trip setpoint cannot be restored to within its specified value during performance of the CHANNEL CALIBRATION, the appropriate ACTION, 3.3.1a or 3.3.1b, shall be followed.

(b) This function is automatically bypassed when the reactor mode switch is in the Run position.

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

(d) The non-coincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 4 APRMS and 6 IRMS.

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

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

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

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

(i) With any control rod withdrawn.*

(j) This function shall be automatically bypassed when turbine first stage pressure is less than 108 psig or 17% of the value of first stage pressure in psia at valves wide open (V.W.O.) steam flow, equivalent to THERMAL POWER of about 24% of RATED THERMAL POWER.

(k) Also actuates the EOC-RPT system.

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

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

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

*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.

**Not including simulated thermal power time constant.

#Measured from actuation of fast-acting solenoid.

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	SA SA	2 3, 4, 5
b. Inoperative	NA	S/U[c], W	NA	2, 3, 4, 5
2. Avg. Power Range Monitor[f]:				
a. Neutron Flux - Upscale, Setdown	S/U,S,[b] S	S/U[c], W W	SA SA	2 3, 5
b. Flow Biased Simulated Thermal Power-Upscale	S,D[g]	S/U[c], Q	W[d][e], SA, R[h]	1
c. Fixed Neutron Flux - Upscale	S	S/U[c], Q	W[d], SA	1
d. Inoperative	NA	S/U[c], Q	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	NA	Q	Q	1, 2
4. Reactor Vessel Water Level - Low, Level 3	S	NA	Q	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
6. Main Steam Line Radiation - High	S	Q	R	1, 2[i]
7. Drywell Pressure - High	NA	Q	R	1, 2

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High				
a. Level Transmitter	NA	Q	R	1, 2, 5[j]
b. Float Switch	NA	Q	R	1, 2, 5[j]
9. Turbine Stop Valve - Closure	NA	Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	Q	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	NA	1, 2, 3, 4, 5

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

(b) The IRM and SRM channels shall be determined to overlap for at least 0.5 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 0.5 decades during each controlled shutdown, if not performed within the previous 7 days.

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

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

(g) Verify measured core flow to be greater than or equal to established core flow at the existing loop flow.

(h) This calibration shall consist of verifying the 6 ± 1 second simulated thermal power time constant.

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

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

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. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

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 accident analysis. 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) in place, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

CHANNEL FUNCTIONAL TEST frequencies and allowed outage times (AOTs) for repair and surveillance testing are based on General Electric report NEDC-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," dated March, 1988. The conclusion of this report is that fewer challenges to safety-related equipment, due to less frequent testing of the RPS, conservatively results in a decrease in core damage frequency. The 6 hour AOT for testing and the 12 hour AOT for repair of one trip system provide enough margin so as not to create an undue stress on personnel. The more restrictive 6 hour repair AOT (Action 1.a) reflects the potential that both trip systems are degraded.

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. 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 set-points 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. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 10 second delay. It follows that checking the valve speeds and the 10 second time for emergency power establishment will establish the response time for the isolation functions. However, to enhance overall system reliability and to monitor instrument channel response time trends, the isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME.

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 equal to or less than the drift allowance assumed 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.

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 equal to or less than the drift allowance assumed for each trip in the safety analyses.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 115 TO FACILITY OPERATING LICENSE NO. NPF-14 AND

AMENDMENT NO. 84 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 50-388

1.0 INTRODUCTION

By letter dated October 27, 1988, as supplemented by letters dated November 9, 1988, January 9, 1989, July 5, 1989, February 22, 1990, March 20, 1991 and July 31, 1991, the Pennsylvania Power and Light Company (PP&L) and the Allegheny Electric Cooperative, Inc. (the licensees), submitted a request for changes to the Susquehanna Steam Electric Station, Units 1 and 2 (SSES), Technical Specifications (TS). The November 9, 1988, and January 9, 1989 letters requested that information be treated as proprietary. The information provided in the July 5, 1989 and March 20, 1991 letters related to the clarification of the information provided in the October 27, 1988 letter and did not change the initial proposed no significant hazards determination. The requested changes would change the surveillance test intervals and allowable outage times for the reactor protection system.

As part of the BWR Owners' Group Technical Specification Improvement Program, General Electric (GE) performed a reliability analysis to identify improvements to the reactor protection system (RPS) surveillance test intervals (STIs) and allowed outage time (AOTs) as provided in the topical report, NEDC-30851P. NRC found that it provides an acceptable generic basis for supporting plant-specific technical specification (TS) changes related to the RPS. As noted in the SER for the GE Topical Report (NEDC-30851P) dated May 27, 1987, GE determined that if the proposed RPS TS changes are implemented, there would be no significant increase of RPS failure frequency for the reviewed BWR plants. This determination is based on use of the GE procedure given in Appendix K of NEDC-30851P for evaluating specific plants against the generic RPS design and analysis.

The GE report does not confirm that calibration of the analog trip units can be extended from monthly to quarterly without creating excessive drift. As a result, the staff has given a list of plant-specific conditions that applicants for proposed TS changes for individual plants must:

(1) Confirm the applicability of the generic analysis for NEDC-30851P to its plant.

(2) Demonstrate by use of current drift information provided by the equipment vendor or plant-specific data, that the drift characteristics for instrumentation used in the RPS channels in the plant are bounded by the assumption used in NEDC-30851P when functional test interval is extended from monthly to quarterly.

(3) Confirm that the differences between the parts of the RPS that perform the trip functions in the plant and those of the base case plant were included in the analysis using the procedures of Appendix K of NEDC-30851P, or provide plant-specific analyses to demonstrate that there is not appreciable change in RPS availability or public risk.

In accordance with the plant-specific conditions that each licensee must meet to make any proposed technical specification changes fully acceptable, proposes changes to the TS related to the RPS for the SSES Units 1 and 2 by letter dated October 27, 1988. The information provided in the October 27, 1988 letter was clarified via subsequent letters dated November 9, 1988, January 9, 1989, July 5, 1989, February 22, 1990, and supported by RPS surveillance test results provided by letter and telecopy dated March 20, 1991 and July 31, 1991, respectively. The information supplied by the supplemental letters did not affect the Commission's determination regarding no significant hazards consideration reported in the Federal Register (53 FR 50333) dated December 14, 1988.

2.0 EVALUATION

The generic study in NEDC-30851P provides a technical basis to modify the surveillance test frequencies and allowable out-of-service time of the RPS from the generic TS. The generic study also provides additional analyses of various known different RPS configurations to support the application of the generic basis on the plant-specific basis. The generic basis and the supporting analyses were utilized in the Susquehanna plant-specific evaluation. The plant-specific evaluation of the RPS for SSES was provided to the staff and contained several major design differences along with the applicable justifications. The staff reviewed these design differences and concluded that they did not effect the applicability of the generic TS improvements for SSES.

A plant-specific evaluation of modifying the surveillance test frequencies and allowable out-of-service time of the RPS from the TS of SSES has been performed by GE for PP&L. The evaluation utilized the generic basis and the additional analyses documented in NEDC-30851P that was approved by NRC. The results indicated that the RPS configuration for SSES is different when compared to the RPS configuration in the generic evaluation. These differences do not affect plant safety due to the changes in the TS based on the generic analysis. Therefore, the generic analysis in NEDC-30851P is applicable to SSES.

The staff SER of May 27, 1987, on GE Topical Reports NEDC-30844 and NEDC-30851P states the NRC's requirement for confirmation of instrument setpoint drift allowance. Licensees should examine plant and/or generic data from representative instrument channels over a sufficient period to demonstrate that the setpoint drift expected with the extended STIs is within the margins established using their current methodology. By letter to BWR Owners' Group from C. Rossi (NRC) dated April 27, 1988, the NRC requests the licensees to confirm that the setpoint drift which could be expected under the extended STIs has been studied and either (1) has been shown to remain within the existing allowance in the RPS and ESFAS instrument setpoint calculation or (2) that the allowance and setpoint have been adjusted to account for the additional expected drift.

PP&L has recently completed a review of RPS surveillance test results. By letter and telecopy dated March 20, 1991 and July 31, 1991, respectively, the licensee provided the results of a review for several years of pertinent RPS surveillance test setpoint drift data for 4 of the 12 RPS functions. The licensee demonstrated that setpoint drift for the instrumentation associated with the 3 of the 4 functions remained within the existing allowance in the RPS instrument setpoint calculation when considered over an extended period of time.

For the fourth function, Reactor Vessel Water Level - Low Level 3, an anomaly was found. The licensee proposed to change the channel calibration frequency for the RPS Level 3 switches from an 18-month frequency (i.e., "R") to quarterly frequency (i.e., "Q"). The quarterly channel functional test is to be deleted for this channel since the channel calibration by definition must include the channel functional test.

Based on the discovery of the need for these changes to the original proposal, the licensee has voluntarily gone back and determined that no other RPS instruments need to be tested on a more frequent basis than required by the Technical Specifications for other than short-term trouble shooting purposes. The licensee concluded that the RPS Level 3 problem was an isolated problem that is correct by this proposal.

Summary

Based on our evaluation of the setpoint drift analysis we find that the licensee has provided acceptable justification to extend the channel functional surveillance intervals as proposed from monthly to quarterly except for RPS Level 3 channel. Also, the proposed change to the channel calibration interval of the RPS Level 3 from 18 months to a quarterly interval is acceptable.

3.0 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.

4.0 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 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 (53 FR 50333). 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.

5.0 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 Contributors: SRhow
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Date: December 30, 1991