Docket Nos. 50-387
and 50-388

June 24, 1992

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

Dear Mr. Keiser:

SUBJECT: TYPE "A" OVERALL INTEGRATED CONTAINMENT LEAKAGE RATE TESTING TECHNICAL SPECIFICATION CHANGES, SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (TAC NOS. M81321 AND M81322)

The Commission has issued the enclosed Amendment No. 121 to Facility Operating License No. NPF-14 and Amendment No. 89 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 August 16, 1991 and supplemented by letter dated May 29, 1992.

These amendments revise the Technical Specification 4.6.1.2.a and the associated bases to incorporate an exemption to appendix J of 10 CFR part 50 that removes the requirement that the third Type "A" Overall Integrated Containment Leakage Rate test required in each 10-year service period is to be conducted at the 10-year inservice inspection interval.

A copy of our Safety Evaluation and Notice of Partial Withdrawal is also enclosed. Notice of Issuance will be published in the Commission's Biweekly Federal Register Notice and the Notice of Partial Withdrawal will be published separately in the Federal Register.

Sincerely,
/S/
George F. Maxwell, Acting Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

9207020249 920624
PDR ADOCK 05000387
P PDR

Enclosures:
1. Amendment No. 121 to License No. NPF-14
2. Amendment No. 89 to License No. NPF-22
3. Safety Evaluation
4. Notice of Partial Withdrawal

cc w/enclosures:
See next page

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

June 24, 1992

Docket Nos. 50-387
and 50-388

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

Dear Mr. Keiser:

SUBJECT: TYPE "A" OVERALL INTEGRATED CONTAINMENT LEAKAGE RATE TESTING TECHNICAL SPECIFICATION CHANGES, SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (TAC NOS. M81321 AND M81322)

The Commission has issued the enclosed Amendment No. 121 to Facility Operating License No. NPF-14 and Amendment No. 89 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 August 16, 1991 and supplemented by letter dated May 29, 1992.

These amendments revise the Technical Specification 4.6.1.2.a and the associated bases to incorporate an exemption to appendix J of 10 CFR part 50 that removes the requirement that the third Type "A" Overall Integrated Containment Leakage Rate test required in each 10-year service period is to be conducted at the 10-year inservice inspection interval.

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Sincerely,

George F. Maxwell, Acting Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 121 to License No. NPF-14
2. Amendment No. 89 to License No. NPF-22
3. Safety Evaluation
4. Notice of Partial Withdrawal

cc w/enclosures:
See next page

Mr. Harold W. Keiser
Pennsylvania Power & Light Company

Susquehanna Steam Electric Station,
Units 1 & 2

cc:

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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. 121
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 August 16, 1991 and supplemented by letter dated May 29, 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-14 is hereby amended to read as follows:

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PDR ADOCK 05000387
P PDR

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 121 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 is to be implemented 30 days after its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

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: June 24, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 121

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 6-3	3/4 6-3
3/4 6-4	3/4 6-4*
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2*

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 45.0 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with main steam line isolation valves and main steam line drain valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix "J" of 10 CFR Part 50 with the exception of exemptions granted for main steam isolation and drain valve leak testing and testing the airlocks after each opening.

The frequency for performing the Type A tests is consistent with the requirements of 10CFR50 Appendix "J" with the exception of the exemption granted to the schedular requirements of Section III.D.1(a).

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIV's such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIVs when isolation of the primary system and containment is required.

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 45.0 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression in chamber internal pressure ensure that the containment peak pressure of 45.0 psig does not exceed the design pressure of 53 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 5 psid. The limit of 2.0 psig for initial positive containment pressure will limit the total pressure to 45.0 psig which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The drywell and suppression chamber purge supply and exhaust isolation valves are required to be closed during plant operation except as required for inerting, de-inerting and pressure control. Until these valves have been demonstrated capable of closing during a LOCA or steam line break accident, they shall be blocked so as not to open more than 50°. Until these valves have been demonstrated capable of closing within the times assumed in the safety analysis, they shall not be open more than 90 hours in any consecutive 365 days.

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

restore:

a. The overall integrated leakage rate to less than or equal to 0.75 L_a, and

b. The combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves*, main steam line drain valves* and valves which are hydrostatically leak tested per Table 3.6.3-1, subject to Type B and C tests to less than or equal to 0.60 L_a, and

c. The leakage rate to less than or equal to 46 scf per hour for all four main steam lines through the isolation valves, and

d. The leakage rate to less than or equal to 1.2 scf per hour for any one main steam line drain valve, and

e. The combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 3.3 gpm,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 - 1972:

a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at P_a, 45.0 psig, during each 10-year service period.#

b. If any periodic Type A test fails to meet .75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet .75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet .75 L_a, at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a.

2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage at P_a, 45.0 psig.

Exemption to Appendix J of 10CFR50.

SURVEILLANCE REQUIREMENTS (Continued)

d. Type B and C tests shall be conducted with gas at P_a, 45.0 psig,* at intervals no greater than 24 months except for tests involving:

1. Air locks,

2. Main steam line isolation valves and main steam line drain valves,

3. Containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and

4. Purge supply and exhaust isolation valves with resilient material seals.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line isolation valves and main steam line drain valves shall be leak tested at least once per 18 months.

g. Containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months.

h. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.8.2.

i. The provisions of Specification 4.0.2 are not applicable to Specifications 4.6.1.2.a, 4.6.1.2.b, 4.6.1.2.c, 4.6.1.2.d, and 4.6.1.2.e.

*Unless a hydraulic test is required per Table 3.6.3-1.

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. 89
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 August 16, 1991 and supplemented by letter dated May 29, 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. 89 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 is to be implemented 30 days after its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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: June 24. 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 89

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 6-3	3/4 6-3
3/4 6-4	3/4 6-4*
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2*

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 45.0 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with main steam line isolation valves and main steam line drain valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix "J" of 10 CFR Part 50 with the exception of exemptions granted for main steam isolation and drain valve leak testing and testing the airlocks after each opening.

The frequency for performing the Type A tests is consistent with the requirements of 10CFR50 Appendix "J" with the exception of the exemption granted to the schedular requirements of Section III.D.1(a).

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIV's such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIVs when isolation of the primary system and containment is required.

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 45.0 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression in chamber internal pressure ensure that the containment peak pressure of 45.0 psig does not exceed the design pressure of 53 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 5 psid. The limit of 2.0 psig for initial positive containment pressure will limit the total pressure to 45.0 psig which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The drywell and suppression chamber purge supply and exhaust isolation valves are required to be closed during plant operation except as required for inerting, deinerting and pressure control. The 90 hours per 365 day limit on purge valve operation is imposed to protect the integrity of the SGTS filters. Analysis indicates that should a LOCA occur while this pathway is being utilized, the associated pressure surge through the (18 or 24") purge lines will adversely affect the integrity of SGTS. This limit is not imposed, however, on the subject valves when pressure control is being performed through the 2-inch bypass line, since a pressure surge through this line does not threaten the OPERABILITY of SGTS.

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

restore:

a. The overall integrated leakage rate to less than or equal to 0.75 L_a, and

b. The combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves*, main steam line drain valves* and valves which are hydrostatically leak tested per Table 3.6.3-1, subject to Type B and C tests to less than or equal to 0.60 L_a, and

c. The leakage rate to less than or equal to 46 scf per hour for all four main steam lines through the isolation valves, and

d. The leakage rate to less than or equal to 1.2 scf per hour for any one main steam line drain valve, and

e. The combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 3.3 gpm,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 - 1972:

a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at P_a, 45.0 psig, during each 10-year service period.#

b. If any periodic Type A test fails to meet .75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet .75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet .75 L_a, at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a.

2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage at P_a, 45.0 psig.

Exemption to Appendix J of 10CFR50.

SURVEILLANCE REQUIREMENTS (Continued)

d. Type B and C tests shall be conducted with gas at P_a, 45.0 psig,* at intervals no greater than 24 months except for tests involving:

1. Air locks,

2. Main steam line isolation valves and main steam line drain valves,

3. Containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and

4. Purge supply and exhaust isolation valves with resilient material seals.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line isolation valves and main steam line drain valves shall be leak tested at least once per 18 months.

g. Containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months.

h. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.8.2.

i. The provisions of Specification 4.0.2 are not applicable to Specifications 4.6.1.2.a, 4.6.1.2.b, 4.6.1.2.c, 4.6.1.2d and 4.6.1.2e.

*Unless a hydraulic test is required per Table 3.6.3-1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 121 TO FACILITY OPERATING LICENSE NO. NPF-14

AMENDMENT NO. 89 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 August 16, 1991 and supplemented by letter dated May 29, 1992, the Pennsylvania Power and Light Company and Allegheny Electric Cooperative, Inc. (the licensees) submitted a request for changes to the Susquehanna Steam Electric Station (SSES), Units 1 and 2, Technical Specifications (TS). The requested changes would revise the Technical Specification 4.6.1.2.a and the associated bases to incorporate an exemption to appendix J of 10 CFR part 50 that removes the requirement that the third Type "A" Overall Integrated Containment Leakage Rate test required in each 10-year service period is to be conducted at the 10-year inservice inspection interval. The May 29, 1992 supplemental letter did not change the initial proposed no significant hazards consideration.

2.0 EVALUATION

Technical Specification Surveillance Requirement 4.6.1.2.a states: "Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown, at P_a 45.0 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant in-service inspection."

This requirement is in compliance with Appendix J of 10 CFR Part 50, Section III.D.1(a), which states that a set of three Type A tests shall be performed at "approximately equal intervals during each 10-year service period" with the third test of each set conducted when the plant is shutdown for the 10-year in-service inspections (ISI).

With a current operating cycle of 18 months, integrated leak rate testing (ILRT) is required every other outage (nominally 36 months) to meet the 40 ± 10 month interval. Because this test schedule does not match up exactly with the 10 year ISI outage, back-to-back ILRTs would have to be performed in back-to-back outages.

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The 10 CFR Part 50, Appendix J requirements provide for periodic verification by tests of the leak-tight integrity of the primary reactor containment, and systems and components which penetrate the primary containment, and establish the acceptance criteria for these tests. The purposes of these tests are to assure that (1) leakage through the primary containment, and systems and components penetrating primary containment do not exceed allowable leakage rates specified in the Technical Specifications and associated Bases, and (2) periodic surveillance of primary containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment.

The ASME Boiler and Pressure Vessel Code, Section XI, provides requirements for the inservice inspection and test of ASME Code Class 1, 2 and 3 components, pumps, and valves. These component inspections, inservice tests for verifying operational readiness of pumps and valves whose function is required for safety, and piping system pressure tests, conducted at the 10-year intervals, comply with the requirements of 10 CFR 50.55a(g).

The requirement to perform the third Type A ILRT concurrent with the 10-year ISI stems from the 10 CFR Part 50, Appendix J. The apparent basis for coupling the two types of tests is to assure that three Type A tests are not grouped together during the first 90 months of each 10-year operating cycle.

Based upon the evaluation provided above, the change will not affect the safe operation of SSES since it does not reduce any requirement for primary containment integrity as imposed by the Technical Specifications or 10 CFR Part 50, Appendix J, nor any 10-year ISI requirements as imposed by Section XI of the ASME Code and 10 CFR 50.55a(g).

In conjunction with the exemption to Appendix J, it eliminates unnecessary testing in successive plant outages while providing the desired level of testing as required by the Technical Specifications and 10 CFR Part 50, Appendix J. The change affects the scheduling of only one of the three Type A tests during each 10-year service period. The scheduling and performance of the remaining tests would not be affected. The manner in which the Type A tests are performed and the applicable acceptance criteria would remain unchanged. The 10 CFR Part 50, Appendix J requirements will continue to be met with an exception to the schedular requirements of Section III.D.1(a).

By letter dated May 29, 1992, the licensees requested a withdrawal of a portion of the proposed changes. PP&L requested that the Technical Specification 4.6.1.2a testing frequency remain 40 $\pm$ 10 months and we find this acceptable. A Notice of Partial Withdrawal will be published in the Federal Register.

Summary

The staff considers the requirement that the third type "A" test occur during the 10-year ISI outage to be of minimal safety significance when compared to

the actual interval between tests. The licensees proposal maintains an appropriate interval between tests for ensuring containment leakage integrity. Further it should be noted that the staff has proposed a revision to Appendix J, Section III.D.1(a) regarding Type "A" test frequency (51 FR 39538) that would eliminate the requirement that the third Type "A" test per 10-year service period coincide with the 10-year ISI interval. The staff therefore finds the requested exemption and associated technical specification changes to be 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 and changes the surveillance requirements. 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 (56 FR 43812). 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.

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 Contributor: J. Raleigh

Date: June 24, 1992

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

PENNSYLVANIA POWER AND LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NOS. 50-387 AND 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

NOTICE OF PARTIAL WITHDRAWAL OF APPLICATION FOR

AMENDMENTS TO FACILITY OPERATING LICENSES

The United States Nuclear Regulatory Commission (the Commission) has granted the request by Pennsylvania Power and Light Company (PP&L) and Allegheny Electric Cooperative, Inc., (the licensees) to withdraw a portion of their August 16, 1991 application, for proposed amendments to Facility Operating Licenses DPR-14 and DPR-22 for the Susquehanna Steam Electric Station, Units 1 and 2, located in Luzerne County, Pennsylvania.

The proposed amendments involved changes to the Technical Specification 4.6.1.2a and the associated bases to incorporate an exemption from Appendix J of 10 CFR Part 50 that removes the requirement that the third Type "A" Overall Integrated Containment Leakage Rate test required in each 10-year service period is to be conducted at the 10-year inservice inspection interval.

On May 29, 1992, the licensee submitted a letter to the NRC requesting withdrawal of a proposed change. PP&L requests that the Technical Specification 4.6.1.2a testing frequency remain 40 $\pm$ 10 months.

The Commission has previously issued a Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing which was published in the FEDERAL REGISTER on September 4, 1991 (56 FR 43812).

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For further details with respect to this action, see the application for amendment dated August 16, 1991 and the licensee's letter dated May 29, 1992, which withdrew this portion of the application for license amendment. The above documents are available for public inspection at the Commission's Public Document Room, 2120 L Street, N.W., Washington, D.C., and the Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701.

Dated at Rockville, Maryland, this 24th day of June 1992.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Maxwell, Acting Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation