August 1 _ 1995

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

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (TAC NOS. M91013 and M91014)

Dear Mr. Byram:

The Commission has issued the enclosed Amendment No. 151 to Facility Operating License No. NPF-14 and Amendment No. 121 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 November 21, 1994, as supplemented by letters dated February 21, 1995, March 28, 1995, April 10, 1995, May 24, 1995, and June 23, 1995.

These amendments change the Technical Specifications for the two units by deleting reference to the main steamline isolation valve (MSIV) leakage control system and its associated primary containment isolation valves, and increase the allowable leakage rate for any MSIV and the total maximum pathway leakage for all four main steam lines.

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 J. Stolz for C. Poslusny
Chester Poslusny, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-387/50-388

Enclosures:
1. Amendment No. 151 to License No. NPF-14
2. Amendment No. 121 to License No. NPF-22
3. Safety Evaluation

cc w/encls: See next page

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

August 15, 1995

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

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (TAC NOS. M91013 and M91014)

Dear Mr. Byram:

The Commission has issued the enclosed Amendment No. 151 to Facility Operating License No. NPF-14 and Amendment No. 121 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 November 21, 1994, as supplemented by letters dated February 21, 1995, March 28, 1995, April 10, 1995, May 24, 1995, and June 23, 1995.

These amendments change the Technical Specifications for the two units by deleting reference to the main steamline isolation valve (MSIV) leakage control system and its associated primary containment isolation valves, and increase the allowable leakage rate for any MSIV and the total maximum pathway leakage for all four main steam lines.

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

Sincerely,

John F. Stolz for

Chester Poslusny, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-387/50-388

Enclosures:
1. Amendment No. 151 to License No. NPF-14
2. Amendment No. 121 to License No. NPF-22
3. Safety Evaluation

cc w/encls: See next page

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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. 151
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 November 21, 1994, as supplemented by letters dated February 21, 1995, March 28, 1995, April 10, 1995, May 24, 1995, and June 23, 1995, 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. 151 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 upon the restart of the unit after its 9th refueling and inspection outage.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 15, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 151

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.

REMOVE	INSERT
vii	vii
xiv	xiv
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-7	---
3/4 6-22	3/4 6-22
3/4 6-27	3/4 6-27
3/4 8-31	3/4 8-31
B 3/4 6-1	B 3/4 6-1

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 1.0 percent by weight of the containment air per 24 hours at P_a, 45.0 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a 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 when pressurized to P_a, 45.0 psig.

c. *Less than or equal to 100 scf per hour for any one main steam isolation valve and a combined maximum pathway leakage rate of ≤ 300 scf per hour for all four main steam lines through the isolation valves when tested at P_t, 22.5 psig.

d. *Less than or equal to 1.2 scf per hour for any one main steam line drain valve when tested at P_a, 45.0 psig.

e. A combined leakage rate of less than or equal to 3.3 gpm for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 P_a, 49.5 psig.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a, or

b. The measured 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 testes exceeding 0.60 L_a, or

c. The measured leakage rate exceeding 100 scf per hour for any one main steam isolation valve or a total maximum pathway leakage rate of > 300 scf per hour for all four main steam lines through the isolation valves, or

d. The measured leak rate exceeding 1.2 scf per hour for any one main steam line drain valve, or

e. The measured combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 3.3 gpm,

---

* Exemption to Appendix "J" of 10 CFR 50.

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 11.5 scf per hour for any main steam isolation valve that exceeds 100 scf per hour, and restore the combined maximum pathway leakage rate to ≤300 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.

TABLE 3.6.3-1 (Continued) PRIMARY CONTAINMENT ISOLATION VALVES		
Valve Function and Number	**Maximum Isolation Time (Seconds)**	**Isolation Signal(s)[a]**
Automatic Isolation Valves (Continued)		
SUPPRESSION POOL CLEANUP[b]		
HV-15766	30	B,Z
HV-15768	30	B,Z
HPCI VACUUM BREAKER		
HV-155F075	15	LB,Z
HV-155F079	15	LB,Z
RCIC VACUUM BREAKER		
HV-149F062	10	KB,Z
HV-149F084	10	KB,Z
TIP BALL VALVES[d]		
C51-J004 A,B,C,D,E	5	A,Z
CONTAINMENT RADIATION DETECTION SYSTEM		
SV-157100 A,B	N/A	B,Y
SV-157101 A,B	N/A	B,Y
SV-157102 A,B	N/A	B,Y
SV-157103 A,B	N/A	B,Y
SV-157104	N/A	B,Y
SV-157105	N/A	B,Y
SV-157106	N/A	B,Y
SV-157107	N/A	B,Y
b. MANUAL ISOLATION VALVES		
FEEDWATER[e]		
HV-141F032 A,B		
RWCU RETURN		
HV-14182 A,B		
RCIC INJECTION		
HV-149F013		
1-49-020		

TABLE 3.6.3-1 (Continued) PRIMARY CONTAINMENT ISOLATION VALVES
Valve Function and Number
Excess Flow Check Valves (Continued)
<u>REACTOR RECIRCULATION</u> XV-143F003 A,B XV-143F004 A,B XV-143F009 A,B,C,D XV-143F010 A,B,C,D XV-143F011 A,B,C,D XV-143F012 A,B,C,D XV-143F040 A,B,C,D XV-143F057 A,B, <u>NUCLEAR BOILER VESSEL INSTRUMENT</u> XV-142F041 XV-142F043 A,B XV-142F045 A,B XV-142F047 A,B XV-142F051 A,B,C,D XV-142F053 A,B,C,D XV-142F055 XV-142F057 XV-142F059 A,B,C,D,E,F,G,H,L,M,N,P,R,S,T,U, XV-142F061 XV-14201 XV-14202 <u>NUCLEAR BOILER</u> XV-141F070 A,B,C,D XV-141F071 A,B,C,D XV-141F072 A,B,C,D XV-141F073 A,B,C,D XV-141F009

TABLE 3.8.4.2.1-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION CONTINUOUS

Valve Number	System(s) Affected
HV-E11-1F028B	RHR
HV-E11-1F047B	RHR
HV-E11-1F016B	RHR
HV-E11-1F003B	RHR
HV-E11-1F017B	RHR
HV-E21-1F031B	CS
HV-E21-1F001B	CS
HV-E11-1F103B	RHR
HV-E11-1F075B	RHRSW
HV-E11-1F073B	RHRSW
HV-E11-1F006D	RHR
HV-E11-1F004D	RHR
HV-E11-1F024B	RHR
HV-E21-1F015B	CS
HV-E21-1F004B	CS
HV-E21-1F005B	CS
HV-E51-1F045	RCIC
HV-E51-1F012	RCIC
HV-E51-1F013	RCIC
HV-15012	RCIC
HV-E51-1F046	RCIC
HV-E51-1F008	RCIC
HV-E51-1F031	RCIC
HV-E51-1F010	RCIC

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 airlocks after each opening.

The frequency for performing the Type A tests is consistent with the requirements of 10 CFR 50 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.

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. 121
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 November 21, 1994, as supplemented by letters dated February 21, 1995, March 28, 1995, April 10, 1995, May 24, 1995, and June 23, 1995, 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. 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 within 60 days after its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment: Changes to the Technical Specifications

Date of Issuance: August 15, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 121

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.

REMOVE	INSERT
vii	vii
xiv	xiv
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-7	---
3/4 6-22	3/4 6-22
3/4 6-27	3/4 6-27
3/4 8-33	3/4 8-33
B 3/4 6-1	B 3/4 6-1

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION		PAGE

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.6 CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 1.0 percent by weight of the containment air per 24 hours at P_a, 45.0 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a 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 when pressurized to P_a, 45.0 psig.

c. *Less than or equal to 100 scf per hour for any one main steam isolation valve and a combined maximum pathway leakage rate of ≤ 300 scf per hour for all four main steam lines through the isolation valves when tested at P_t, 22.5 psig.

d. *Less than or equal to 1.2 scf per hour for any one main steam line drain valve when tested at P_a, 45.0 psig.

e. A combined leakage rate of less than or equal to 3.3 gpm for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 P_a, 49.5 psig.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a, or

b. The measured 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 testes exceeding 0.60 L_a, or

c. The measured leakage rate exceeding 100 scf per hour for any one main steam isolation valve or a total maximum pathway leakage rate of > 300 scf per hour for all four main steam lines through the isolation valves, or

d. The measured leak rate exceeding 1.2 scf per hour for any one main steam line drain valve, or

e. The measured combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 3.3 gpm,

---

* Exemption to Appendix "J" of 10 CFR 50.

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 11.5 scf per hour for any main steam isolation valve that exceeds 100 scf per hour, and restore the combined maximum pathway leakage rate to ≤300 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.

TABLE 3.6.3-1 (Continued) PRIMARY CONTAINMENT ISOLATION VALVES		
Valve Function and Number	**Maximum Isolation Time (Seconds)**	**Isolation Signal(s)[a]**
Automatic Isolation Valves (Continued)		
SUPPRESSION POOL CLEANUP[b]		
HV-25766	35	B,Z
HV-25768	30	B,Z
HPCI VACUUM BREAKER		
HV-255F075	15	LB,Z
HV-255F079	15	LB,Z
RCIC VACUUM BREAKER		
HV-249F062	10	KB,Z
HV-249F084	10	KB,Z
TIP BALL VALVES[d]		
C51-J004 A,B,C,D,E	5	A,Z
CONTAINMENT RADIATION DETECTION SYSTEM		
SV-257100 A,B	N/A	B,Y
SV-257101 A,B	N/A	B,Y
SV-257102 A,B	N/A	B,Y
SV-257103 A,B	N/A	B,Y
SV-257104	N/A	B,Y
SV-257105	N/A	B,Y
SV-257106	N/A	B,Y
SV-257107	N/A	B,Y
b. MANUAL ISOLATION VALVES		
FEEDWATER[e]		
HV-241F032 A,B		
RWCU RETURN		
HV-24182 A,B		
RCIC INJECTION		
HV-249F013		
2-49-020		

TABLE 3.6.3-1 (Continued) PRIMARY CONTAINMENT ISOLATION VALVES
Valve Function and Number
Excess Flow Check Valves (Continued)
<u>REACTOR RECIRCULATION</u> XV-243F003 A,B XV-243F004 A,B XV-243F009 A,B,C,D XV-243F010 A,B,C,D XV-243F011 A,B,C,D XV-243F012 A,B,C,D XV-243F040 A,B,C,D XV-243F057 A,B, <u>NUCLEAR BOILER VESSEL INSTRUMENT</u> XV-242F041 XV-242F043 A,B XV-242F045 A,B XV-242F047 A,B XV-242F051 A,B,C,D XV-242F053 A,B,C,D XV-242F055 XV-242F057 XV-242F059 A,B,C,D,E,F,G,H,L,M,N,P,R,S,T,U, XV-242F061 XV-24201 XV-24202 <u>NUCLEAR BOILER</u> XV-241F070 A,B,C,D XV-241F071 A,B,C,D XV-241F072 A,B,C,D XV-241F073 A,B,C,D XV-241F009

TABLE 3.8.4.2.1-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION CONTINUOUS

Valve Number	System(s) Affected
HV-E11-2F021B	RHR
HV-E11-2F010B	RHR
HV-E11-2F004B	RHR
HV-E11-2F007B	RHR
HV-E11-2F104B	RHR
HV-E11-2F026B	RHR
HV-E11-2F028B	RHR
HV-E11-2F047B	RHR
HV-E11-2F016B	RHR
HV-E11-2F003B	RHR
HV-E11-2F017B	RHR
HV-E21-2F031B	CS
HV-E21-2F001B	CS
HV-E11-2F103B	RHR
HV-E11-2F075B	RHRSW
HV-E11-2F073B	RHRSW
HV-E11-2F006D	RHR
HV-E11-2F004D	RHR
HV-E11-2F024B	RHR
HV-E21-2F015B	CS
HV-E21-2F004B	CS
HV-E21-2F005B	CS
HV-E51-2F045	RCIC
HV-E51-2F012	RCIC
HV-E51-2F013	RCIC
HV-25012	RCIC

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 airlocks after each opening.

The frequency for performing the Type A tests is consistent with the requirements of 10 CFR 50 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.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 151 TO FACILITY OPERATING LICENSE NO. NPF-14

AMENDMENT NO. 121 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 November 21, 1994, (Reference 1) as supplemented by letters dated February 21, 1995, March 28, 1995, April 10, 1995, May 24, 1995, and June 23, 1995, 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 change the TS for the two units by deleting reference to the main steamline isolation valve (MSIV) leakage control system and its associated primary containment isolation valves, to reflect a design change and use of an alternate leakage pathway and increase the allowable leakage rate for any MSIV and the total maximum pathway leakage for all four main steam lines.

Specifically, the licensee requested that:

1. The allowable leakage rate specified in TS 3.6.1.2 be modified from the current 11.5 scfh for any one MSIV when tested at 22.5 psig to 100 scfh for any one MSIV with a total maximum pathway leakage of 300 scfh through all four main steam lines when tested at 22.5 psig;

2. TS 3/4.6.1.4 and its Bases, Tables 3.6.3-1 and 3.8.4.2, be amended to permit the deletion of the MSIV LCS from the TSs;

3. A new requirement be added to TS 3.6.1.2 related to the restoration of acceptable leak rates if any of the proposed limits are exceeded, such that if any MSIV exceeds 100 scfh, it will be repaired and retested to meet a leak rate limit of 11.5 scfh per valve;

4. The Index, TS 3/4.6.1.4, and Bases 3/4.6.1.4 be administratively revised to reflect the above requested changes.

The licensee proposes these changes as an alternative to Regulatory Guide 1.96 (RG 1.96), "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor (BWR) Nuclear Power Plants," by utilizing the main steam lines and condenser, as an alternate method for MSIV leakage treatment. The proposed changes are a result of extensive work performed by the Boiling Water Reactor Owners Group (BWROG) in support of the resolution of Generic Issue C-8, "MSIV Leakage and Leakage Failure." In addition to the licensee's submittals, General Electric (GE) Report NEDC-31858P, Revision 2, "BWROG Report for Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems," dated September 1993 (Reference 2), also provided technical justification for the proposed changes.

The February 21, 1995, March 28, 1995, April 10, 1995, May 24, 1995, and June 23, 1995 letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

The main steam lines (MSLs) contain dual quick-closing MSIVs. These valves function to isolate the reactor system in the event of a break in a steam line outside the primary containment, a design basis loss of coolant accident (LOCA), or other events requiring containment isolation. Although the MSIVs are designed to provide a leak-tight barrier, it is recognized that some leakage through the valves will occur. Operating experience at various BWR plants has indicated that degradation has occasionally occurred in the leak-tightness of MSIVs, and the specified low leakage has not always been maintained.

Because of recurring problems with excessive leakage of MSIVs, RG 1.96 recommended the installation of a supplemental LCS to ensure that the isolation function of the MSIVs complies with the specified limits. To meet this requirement, the licensee installed a safety-related MSIV LCS that is designed to eliminate the release of fission products. This is accomplished by developing a negative pressure in the sections of the MSLs between the inboard and outboard MSIVs, and between the outboard MSIVs and the turbine stop valves. This negative pressure is developed by a series of blowers that discharge the leakage to an area where it is treated by the standby gas treatment system (SGTS).

Due to design limitations, the LCS would be unavailable if the MSIV leak rate was greatly in excess of the allowable value in the TSs. Hence, Generic Issue C-8 was initiated in 1983 to assess: (1) the causes of MSIV failures, (2) the effectiveness of the LCS and alternative leakage paths, and (3) the need for regulatory action to limit public risk. The resolution of C-8 (see NUREG-1372, Regulatory Analysis for the Resolution of Generic Issue C-8, "Main Steam Isolation Valve Leakage and LCS Failure," dated June 1990, concluded that no backfit requirements to reduce public risk were warranted and that no action should be taken. However, one of the alternative resolutions of C-8 showed that several non-seismic Category I paths gave off lower doses than the LCS and could handle larger MSIV leak rates.

In a parallel effort the BWROG formed an MSIV Leakage Committee in 1982 to identify and resolve the causes of high MSIV leakage rates. The BWROG then formed a follow-on MSIV Leakage Closure Committee to address alternate actions to resolve on-going but less severe MSIV leakage problems and to address the limited capability of the LCS. The results of these committee activities were submitted to the NRC in several GE proprietary reports, the latest of which is NEDC-31858P, Revision 2, (Reference 2). This report concludes that the proposed increase of the MSIV leakage limit will reduce radiation exposures to maintenance personnel, reduce outage durations, and extend the effective service life of the MSIVs. The report also concludes that the proposed elimination of the LCS will similarly reduce exposures to maintenance personnel, reduce outage durations, and that the LCS can be replaced with an alternate method for MSIV leakage treatment using the MSLs and condenser. The licensee referred to this report as a basis to delete the TS requirements for the MSIV LCS and requested a substantially higher (100 scfh per steam line and a total of 300 scfh for all four MSLs) MSIV leak rate limit.

The proposed alternative treatment method recommended in the BWROG report, and as proposed by the licensee, takes advantage of the large volume in the main steam lines and main condenser to provide hold-up and plate-out of fission products that may leak from closed MSIVs. This method uses the main steam drain lines to direct leakage to the main condenser. In this approach, the main steam piping, the bypass/drain piping, and the main condenser are used to mitigate the consequences of an accident which could result in potential offsite exposures comparable to 10 CFR Part 100. Therefore, as required by Appendix A to Part 100, the components and piping systems used in the alternative treatment path must be capable of performing their function during and following a safe shutdown earthquake (SSE). The BWROG report and the licensee's submittals provide the technical justification for the seismic capability of the alternate treatment path and also provide the dose calculations to demonstrate the acceptability of the system.

3.0 EVALUATION

This evaluation has been performed in several parts. Section 3.1 provides the radiological assessment; Section 3.2 provides the seismic evaluation; Section 3.3 provides the support evaluation; and Section 3.4 provides the plant systems evaluation.

3.1 Radiological Assessment

3.1.1 Background

In order to demonstrate the adequacy of the SSES Units 1 and 2 engineered safety features designed to mitigate the radiological consequences of the design basis accidents (DBAs) with a maximum MSIV leak rate of 300 scfh total from all four main steam lines and without the MSIV-LCS, the licensee assessed the offsite and control room radiological consequences which could result from the occurrence of a postulated loss of coolant accident (LOCA) and presented the results of that assessment in their submittal.

During the SSES Units 1 and 2 licensing review, the staff previously assessed the offsite radiological consequences of a LOCA using 46 scfh MSIV total leak rate from four main steam lines with the MSIV-LCS. The calculated results are shown in Table 15.1 of NUREG-0776, "Safety Evaluation Report Related to the Operation of Susquehanna Steam Electric Station, Units 1 and 2" (OL-SER), April 1981. In this OL-SER, the staff considered the following sources and radioactivity transport paths to the environment following a postulated LOCA:

(1) containment leakage
(2) main steam line isolation valve leakage
(3) post-LOCA leakage from engineered safety features outside containment

In this evaluation, the staff recalculated the radiological consequences associated with main steam isolation valve leakage path. It is assumed that the radiological consequences associated with the other radioactivity transport paths would be negligibly affected by the proposed amendments, therefore, they were not recalculated. The procedures used in the staff's calculation of the radiological consequences associated with MSIV valve leakage were based upon (1) the TID-14844 source term, consistent with the guidelines provided in the applicable sections of the Standard Review Plan (SRP, NUREG-0800) and Regulatory Guides and (2) assumptions and parameters used in the SSES Units 1 and 2 OL-SER, except for the following two deviations. The staff has accepted credit for radioactive iodine removal in the main steam lines, drain lines and main condenser by hold-up, decay and deposition. The staff has also accepted the deletion of the TS requirements for the MSIV-LCS. Dose contributions to the whole body from the increased MSIV leakage were recalculated based upon the ratio of the proposed leakage rate limit of 300 scfh to the current limit of 46 scfh. No credit was given for holdup and decay of noble gases in the main steam lines and condenser. The staff's recalculated offsite and control room operator doses resulting from a postulated LOCA and the parameters and assumptions used in the staff's recalculation are provided in Tables 1 and 2 of this safety evaluation, respectively.

The main steam lines in boiling water reactor plants, including SSES Units 1 and 2, contain dual quick-closing MSIVs. These valves function to isolate the reactor system in the event of a break in a steam line outside the primary containment, a design basis LOCA, or other events requiring containment isolation. Although the MSIVs are designed to provide a leak-tight barrier, it is recognized that some leakage through the valves will occur. The current SSES Units 1 and 2 technical specification limit for MSIV leakage is 11.5 scfh for any one MSIV. Operating experience at various BWR plants has indicated that degradation has occasionally occurred in the leak-tightness of MSIVs, and the specified low leakage has not always been maintained.

Because of recurring problems with excessive leakage of MSIVs, Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," recommended the installation of a supplemental main steam leakage control system to ensure that the isolation function of the MSIVs complies with the specified limits. In order to meet this requirement, the licensee installed a safety-related MSIV leakage Control System which is designed to eliminate the release of fission products through the MSIVs that would bypass the reactor building and filtration by the Standby Gas Treatment System following a postulated LOCA.

In response to the MSIV leakage concerns, the BWR Owners Group (BWROG) in 1986 commissioned a program of studies to determine the causes of high leak rates and the means to eliminate them. The results of these studies were submitted to the NRC in several revisions of a General Electric proprietary report, all titled, "Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems." (See Reference 2)

These reports conclude that the proposed increase of the MSIV leakage limit will reduce radiation exposures to maintenance personnel, reduce outage durations, and extend the effective service life of the MSIVs. The reports also conclude that the proposed elimination of the LCS will similarly reduce exposures to maintenance personnel and reduce outage durations and that the LCS can be replaced with an alternate method for MSIV leakage treatment utilizing the main steam lines and condenser. The licensee referenced these reports as a basis to delete the TS requirements for the MSIV Leakage Control System and to allow a substantially higher (100 scfh per valve) MSIV leak limit.

The MSIVs generally have not provided a leak-tight containment pressure boundary to the extent intended in the plant design. Although substantial progress has been made in recent years to identify the causes of the leakage and to reduce the amount of leakage, the current typical BWR Technical Specification limit of 11.5 scfh per valve used at SSES Unit Nos. 1 and 2 is still difficult to achieve when the valve is rapidly closed against full flow conditions at reactor operating pressure and temperature.

The current assumption used by the staff for operating plants in calculating radiological consequences of potential DBAs is based upon a conservative assumption that the leakage limit allowed by the Technical Specification is released directly into the environment. No credit is currently taken for the integrity and leaktightness of the main steam piping and condenser to provide holdup and plateout of fission products. The proposal developed by the BWROG and adopted by the licensee would allow higher leakage limits (300 scfh total from four steam lines) and delete the TS requirements for the main steam LCS.

3.1.2 Iodine Release Pathways

Following a LOCA, three potential release pathways exist for main steam leakage through the MSIVs:

(1) Main steam drain lines to the condenser with delayed release to the environment through the low pressure turbine seals.

(2) Turbine bypass lines to the condenser with delayed release to the environment through the low pressure turbine seals.

(3) Main steam lines through the turbine stop and control valves and through high pressure turbine seals to the environment bypassing the condenser.

The consequences of leakage from pathways 1 and 2 will be essentially the same because the condenser can be used to process MSIV leakage. The condenser iodine removal efficiency will vary depending on the inlet location of the bypass or drainline piping, but in either case, iodine will be removed. For pathway 3, MSIV leakage through the closed turbine stop and control valves will not be processed via the condenser. For this case, the high-pressure turbine (having a large internal surface area associated with the turbine blades) will remove iodine.

The staff believes that as long as either the turbine bypass or drainline leakage pathway is available, MSIV leakage through the closed turbine stop and control valves (pathway 3) will be negligible. Essentially all of the releases will be through the main condenser because there will be no differential pressure in the MSL downstream of the MSIVs following the closure of the valves.

Furthermore, MSIV leakage through pathway 3, if any, will have been subjected to the same iodine-removal processes in the MSLs (up to turbine stop valves) as the other pathways. The leakage will be further subjected to iodine removal by deposition on the high-pressure turbine internal surfaces. Removal by the main condenser is not applicable in pathway 3.

The licensee has selected to utilize pathway 1 using the main steam piping and its drain piping and the condenser to mitigate the radiological consequences of an accident which could result in potential offsite exposures comparable to the dose reference values specified in 10 CFR Part 100. The staff has accepted the licensee's proposed pathway. In the calculation of the contribution to the LOCA dose, the staff assumed that one of the inboard isolation MSIVs failed to close, thus allowing contaminated steam to travel to the outboard valve. The leakage through this outboard valve and the valve pairs in the other three steamlines were assumed to have a total leak rate of 300 scfh.

3.1.3 Iodine Transport Model

Basic chemical and physical principles predict that gaseous iodine and airborne iodine particulate material will deposit on surfaces. Several laboratory and in-plant studies have demonstrated that gaseous iodine deposits by chemical adsorption and that particulate iodine deposits through a combination of sedimentation, molecular diffusion, turbulent diffusion, and impaction. Gaseous iodine exists in nuclear power plants in several forms: elemental (I_2), hypoiodous acid (HOI), organic (CH_3I), and particulate. In accordance with RG 1.3, the staff assumed 91 percent of iodine is in the elemental form (includes hypoiodous acid), 5 percent in the particulate form, and 4 percent in the form of organic iodides.

Each of these forms deposits on surfaces at a different rate, described by a parameter known as the deposition velocity. The elemental iodine form, being the most reactive, has the largest deposition velocity, and organic iodide has the smallest. Further, studies of in-plant airborne iodine show that iodine (elemental and particulate) deposited on the surface undergoes both physical and chemical changes and can either be resuspended as an airborne gas or become permanently fixed to the surface. The data also show that the iodine can change its form so that iodine deposited as one form (usually elemental) can be resuspended in the same or in another form (usually organic). Conversion can be described in terms of resuspension rates that are different for each iodine species. Chemical surface fixation can similarly be described in terms of a surface fixation rate constant.

The transport of gaseous iodine in elemental and particulate forms has been studied for many years and several groups proposed different models to describe the observed phenomena (References 12 through 16). The staff used the model specifically developed by an NRC contractor (Reference 17) for iodine removal in BWR main steam lines and the main condenser following a LOCA.

The staff model treats the MSIV leakage pathway as a sequence of small segments for which instantaneous and homogeneous mixing is assumed, the mixing computed for each segment is passed along as input to the next segment. The number of segments depends upon the parameters of the line and flow rate and can be as many as 100,000 for a long, large-diameter pipe and a low flow. Each line segment is divided into five compartments that represent the concentrations of the three airborne iodine species, the surface that contains iodine available for resuspension, and surface iodine that has reacted and is fixed on the surface.

The staff's model considers three iodine species: elemental, particulate, and organic. A fourth species, hypoiodous acid, was considered for the purpose of the staff's model to be a form of elemental iodine. All iodine in the segment undergoes radioactive decay. The resulting concentration from each segment of the deposition compartment serves as the input to the next segment.

The GE model, as well as the one developed and used by the staff, is based on time-dependent temperature adsorption phenomena with instantaneous and perfect mixing in a given volume. Both models use the same MSIV leakage pathways. However, they differ in the treatment of buildup of iodine in the main steam lines and condenser. GE assumed steady state iodine in equilibrium in a large volume while the staff model assumed transient buildup of iodine in a finite number of small volumes. The staff does not consider these differences to be significant since the staff finds that the resulting iodine deposition and removal rates in the main steam lines and condenser are in good agreement.

The staff's transport model also assumed iodine transport through the condenser as a dilution flow rather than the plug flow as in the steam lines. The staff assumed that the iodine input into the condenser mixes instantaneously with a volume of air in the condenser and that the diluted air exhausts at the same time and same rate as the input air (MSIV leakage) flows into the condenser.

The staff developed the equations for iodine deposition velocities, resuspension rates, and surface fixation rates as a function of temperature using published data found in the literature. The equations and data are contained in the contractor's report (Reference 17). The equation for the deposition velocity of elemental iodine is based on the least-squares fit to the available data. Deposition velocity equations for HOI and organic iodine are based on the values at 30 °C; due to the lack of data at elevated temperatures, their temperature dependence is assumed to be similar to elemental iodine. Resuspension and fixation equations as a function of temperature are based on measurements available in the literature at ambient temperature. The staff assumed that resuspension and fixation rates will increase with increasing temperature.

The technical references and the GE and staff models indicate that particulate and elemental iodine would be expected to deposit on surfaces with rates of deposition varying with temperature, pressure, gas composition, surface material, and particulate size. Therefore, the staff believes that an appropriate credit for the removal of iodine in the MSLs and main condensers should be provided in the radiological consequence assessment following a design-basis accident. Consequently, the staff accepted the licensee's proposed elimination of the LCS and allowed a higher MSIV leakage providing an appropriate credit for the removal of iodine in the MSLs and condenser.

Sections III(c) and VI of Appendix A to 10 CFR Part 100 require that structures, systems, and components necessary to ensure the capability to mitigate the radiological consequences of accidents that could result in exposures comparable to the dose guidelines of Part 100 be designed to remain functional during and after a safe-shutdown earthquake (SSE). Thus, the MSL, portions of its associated piping, and the main condenser are required to remain functional if credit is taken for deposition of iodine and if the SSE occurs.

Consequently, the staff's past practice has been to classify these components as safety-related and seismic Category I. In addition, Appendix A to 10 CFR Part 100 requires that the engineering method used to ensure that the safety functions are maintained during and after an SSE involve the use of either a suitable dynamic analysis or a suitable qualification test.

For the purpose of providing a credit for iodine holdup and plateout, the staff's model requires that the main steam piping (including its associated piping to the condenser) and the condenser remain structurally intact following an SSE, so they can act as a holdup volume for fission products. By the term "structurally intact," the staff assumes the steamline will retain sufficient structural integrity to transport the relatively low flow rate

($\leq$ 2 ft^3/min) of MSIV bypass leakage throughout the steam lines and condenser. The staff considers, in its radiological consequence assessment, that the condenser is open to the atmosphere via leakage through the low pressure turbine seals. Thus, it is only necessary to ensure that gross structural failure of the condenser will not occur.

3.1.4 Control Room Habitability

The staff has previously evaluated the control room operator doses following a postulated LOCA in accordance with SRP Section 6.4 and found the calculated doses were within the guidelines of SRP Section 6.4 (OL-SER Section 6.4). In this evaluation, the staff considered the fission product releases from the low pressure turbine seal due to the MSIV leakage (up to 300 scfh total) through the MSIV drain lines and the main condensers. The staff assumed a ground level release of airborne fission products from the turbine building as a fission product diffusion source and the control room emergency air intake as a single point receptor.

The staff's recalculated control room operator doses following a postulated LOCA are listed in Table 1 and the staff finds that the recalculated whole-body and equivalent organ doses (thyroid) are still within the guidelines of SRP Section 6.4, and therefore, the staff's conclusions stated in the OL-SER Section 6.4 are not affected and remain the same.

3.1.5 Variations Between Staff and Licensee Calculated Doses

The following table shows both the staff's and the licensee's calculated dose contributions to the thyroid (in Rem) from an MSIV leak rate of 300 scfh:

	2-hour EAB	30-day LPZ	30-day Control Room
Licensee	0.11	12.1	4.95
Staff	10.3	51.6	3.03

The staff's calculated 2-hour dose at the exclusion area boundary is significantly higher than that calculated by the licensee. The main reason for this discrepancy is that the staff gave no credit for deposition and holdup in the main steam piping between the reactor vessel and the outboard MSIV. Crediting that section of the main steam system would significantly increase radionuclide transport time to the low pressure turbine seals. The increased transport time would delay any releases from the low pressure turbine seals and, thus, significantly reduce the 2-hour dose at the EAB.

The increased radionuclide transport time from crediting the additional main steam piping will have little impact on the 30-day dose at the low population zone boundary by virtue of the long duration of assumed exposure. However, the atmospheric dispersion factors utilized by the staff for calculating the 30-day dose at the LPZ were more conservative than those used by the licensee. Applying the licensee's atmospheric dispersion factors to the staff's calculated iodine releases would yield a 30-day dose at the LPZ of approximately 8 rem. Therefore, good correlation can be shown between the staff's and the licensee's calculated iodine releases over 30 days. The correlation is also supported by the control room operator doses which were calculated by the staff and the licensee utilizing the same atmospheric dispersion factors.

3.1.6 Findings

Several technical references (Reference 12-16) including an NRC contractor's report (Reference 17) indicate that particulate and elemental iodine would be expected to deposit on surfaces with rates of deposition varying with temperature, pressure, gas composition, surface material, and particulate size. The staff, therefore, concludes that an appropriate credit for the removal of iodine in the main steam lines and main condensers should be provided in the radiological consequence assessment following a DBA.

The staff has reviewed the licensee's analysis and has performed an independent reassessment of the radiological consequences resulting from the MSIV leakage transport pathway described in this SER. The calculated thyroid and whole-body dose are listed in the revised Table 1. Based on the above evaluation and the calculated radiological consequences shown in Table 1, the staff concludes that the MSIV leak rate limit of 300 scfh total from four main steam lines and the proposed deletion of the TS requirements for the MSIV Leakage Control System are acceptable.

The staff further concludes that the existing distances to the exclusion area and to the low population zone boundaries of SSES, in conjunction with the remaining engineered safety features provided at the Susquehanna Steam Electric Station are still sufficient to provide reasonable assurance that the radiological consequences of a postulated LOCA will be within the dose reference values set forth in 10 CFR Part 100 and the control room operator dose limits specified in GDC-19 of Appendix A to 10 CFR Part 50.

Table 1 Radiological Consequences of Loss-of-Coolant Accident (rem)

	Thyroid EAB	Whole Body LPZ	Thyroid EAB	Whole Body LPZ
Containment Leakage*				
00- 02 hours	218	5.8		
00- 08 hours			43.8	1.42
08- 24 hours			32.6	1.16
24- 96 hours			66.5	0.79
96-720 hours			54.4	0.40
Total containment leakage	218	5.8	197	3.77
ECCS component leakage*	0.2	<0.1	0.16	<0.01
MSIV leakage	10.3	3.9	51.6	0.65
Total	229	9.8	249	4.4

	Thyroid	Whole Body
Control Room Operator Doses (rem)	14.8	0.9

* From Table 15.1 of NUREG-0776, Susquehanna OL-SER

Table 2 Assumptions Used to Evaluate the Loss-of-Coolant Accident, MSIV Leakage Contribution

Core Thermal Power (MWt):	3441
Core Radionuclide Fractions Released to Drywell (%)	
Noble Gases:	100
Iodines:	25
Forms of Iodine Species (%)	
Elemental:	91
Organic:	4
Particulate:	5

Iodine Dose Conversion Factors:	ICRP-30
MSIV Total Leak Rate:	300 scfh
Containment Free Volume (ft^3):	3.89×10^5
Control Room Free Volume (ft^3):	1.10×10^5
Control Room Intake Flow (cfm):	5810
Control Room Intake Filter Iodine Removal Efficiencies (%):	
Elemental:	99
Organic:	99
Particulate:	99
Unfiltered Control Room Inleakage (cfm):	10
Control Room Geometry Factor:	24
Control Room Iodine Protection Factor:	85
Atmospheric Dispersion Factors (sec/m^3)	
0 - 2 hours, Exclusion Area Boundary:	1.1×10^{-3}
0 - 8 hours, Low Population Zone:	5.2×10^{-5}
8 - 24 hours, Low Population Zone:	3.6×10^{-5}
1 - 4 days, Low Population Zone:	1.6×10^{-5}
4 - 30 days, Low Population Zone:	5.3×10^{-6}
Effective Control Room Atmospheric Dispersion Factors (sec/m^3)	
0 - 8 hours:	3.32×10^{-4}
8 - 24 hours:	1.96×10^{-4}
1 - 4 days:	7.64×10^{-5}
4 - 30 days:	2.19×10^{-5}

3.2 Seismic/Mechanical Evaluation

3.2.1 Background

As discussed above, PP&L proposed to use the main steam piping, drain lines, and main condenser as an alternate means for MSIV leakage treatment. Because the original design basis of certain main steam piping and components is not Seismic Category I, PP&L has performed evaluations and seismic verification walkdowns to demonstrate that the main steam system piping and components which comprise the alternate leakage treatment (ALT) system are seismically rugged and are able to perform the safety function of an MSIV leakage treatment system.

The licensee also performed a design evaluation of the seismic adequacy of the turbine building (TB) which houses the ALT system. The structural integrity of the TB is important to the issue of MSIV leakage because a non-seismically designed TB should be capable of withstanding the earthquake and not degrade the capability of the ALT system.

The BWROG report, NEDC-31858P, Revision 2, has not been approved by the staff. However, based on a preliminary review to date, the staff has found the BWROG approach of utilizing the earthquake experience-based methodology, supplemented by plant-specific seismic adequacy evaluation, an acceptable basis to demonstrate the seismic ruggedness of non-seismically analyzed main steam system piping and condensers. The staff, therefore, has relied upon portions of the earthquake experience data, in the BWROG Report, for piping and main condenser in support of this safety evaluation.

It should be noted that there are no provisions in the Susquehanna Final Safety Analysis Report (FSAR) and the staff's safety evaluation associated with the facility operating license that would permit the use of experience data as a means of seismic qualification for piping systems and components. However, requiring the non-seismically analyzed portions of the main steam system piping and components to meet Seismic Category I requirements would not be practical because modifications required to upgrade the system to Seismic Category I requirements can not be justified from the cost-benefit standpoint.

The BWROG has retained Earthquake Engineering, Inc. (EQE) as a consultant to conduct a review of the earthquake experience data on the performance of facility piping and condensers. The study summarized the data on the performance of main steam system piping and condensers in primarily non-nuclear facilities which experienced strong motion earthquakes. In addition, it compared these piping systems and condensers with the piping systems and condensers typically used in GE boiling water reactors (BWRs) in the United States. The result of the comparison appears to support the BWROG contention that main steam piping and condensers employed in GE BWRs would maintain pressure boundary integrity during a Design Basis Earthquake (DBE). According to EQE, based on past earthquake experiences, welded steel piping and condensers designed and constructed to normal industrial practices (e.g., ANSI B31.1 and Heat Exchange Institute (HEI) Standard, respectively) have been found to be seismically rugged and not susceptible to a primary collapse mode of failure as a result of the seismic vibratory motions experienced at sites examined in the earthquake database. The earthquake experience is derived from a database that includes the seismic performance of power plant units and industrial facilities in actual earthquakes. The above BWROG Report notes that a relatively small number of seismically-induced piping failures have occurred due to excessive relative support movements or seismic interactions.

The primary components to be relied upon for the proposed ALT system are the main turbine condenser and the primary drain pathway piping which consists of the drain lines that originate in the steam tunnel just downstream of the outboard MSIVs and terminate at the High Pressure (HP) Condenser. Other potential drain pathways, which are not credited in the radiological dose calculation, are those originating from the main steam lines in the turbine building just upstream of the Main Stop Valves (MSVs) and discharging into the HP Condenser. The condenser forms the ultimate boundary of the ALT system.

Boundaries upstream of the condenser were established by utilizing existing valves, and were used to limit the extent of the seismic verification walkdown. These valves were selected using the criteria outlined in the BWROG Report and documented in PP&L Engineering Studies, Analyses, and Evaluations (SEA), SEA-ME-423, "MSIV Leakage Seismic Verification Boundary Determination Study, SSES Unit 1" and SEA-ME-424, "MSIV Leakage Seismic Verification Boundary Determination Study, SSES Unit 2."

3.2.2 Reliability of Boundary Valves

In Reference 6, PP&L identified that there are two motor-operated valves associated with the primary drain line pathway of which one is normally open and the other is designed to fail safe upon loss of offsite power. The F020 valve is the safety-related normally open valve that is designed to fail safe, while the F021 valve is normally closed and is required to be open to establish the primary flow path to the condenser. The F021 valve will be powered from a bus that is supplied from two independent offsite power sources, and a reliable diesel generator. The F021 valve will also be included in the IST program and stroke tested (open) once per cycle. The F021 valve operator has been evaluated to function under postulated accident conditions. In Reference 11, PP&L identified that, in addition to the F020 and F021 valves, there are three normally open motor-operated boundary valves that will need to be closed when the condenser pathway is used to treat the MSIV leakage. These are HV-10107 to Steam jet Air Ejector, HV-10109 to Steam Seal Evaporator, and HV-10111 to Reactor Feed Pump Turbines. PP&L stated that these latter valves are also boundary valves and their reliability is established based on the same factors addressed for valve F021 (i.e., independent power sources, inclusion in the IST program, and evaluation of the operators to function under accident conditions). Based on the above, the staff believes the licensee has provided sufficient basis to ensure that the valves and the pathway will function as required to establish the primary MSIV leakage pathway to the condenser.

3.2.3 Seismic Verification Walkdowns

The ALT system consists of the main steam piping (beyond the outboard MSIVs), the steam drain lines, the condenser, and interconnected piping. The ALT system, in general, is not seismically analyzed as this analysis was not required in the original licensing basis of either unit at Susquehanna.

In order to confirm the functional capability of the ALT system, the licensee has performed seismic verification walkdowns for Susquehanna Units 1 & 2. The purpose of the walkdowns was to ensure that the ALT system falls within the bounds of the design characteristics of the seismic experience database as discussed in Section 6.7 of the BWROG Report. Specifically, the walkdowns were performed to (1) verify that Susquehanna plant features have attributes similar to those in the earthquake experience database that have demonstrated good seismic performance, (2) verify general conformance of pipe support spans to the requirements of ANSI B31.1, and (3) examine the ALT system from the outboard MSIVs to the condenser to identify potential seismic vulnerabilities considering those structural details and causal factors that resulted in component damage at database plants.

The walkdowns focused on piping systems which were not seismically analyzed; however, those systems which are seismically analyzed, were also examined to identify any anomalies that may have gone undetected during the original construction. The potential vulnerabilities which were identified as "outliers" include categories such as support failure, failure of non-seismically designed plant features (II/I), proximity and impact, and differential seismic anchor motion on piping systems. PP&L's March 28, 1995 (Reference 3) submittal presents a complete list of the outliers identified during the walkdowns and their resolution and modification status.

These outliers have been either evaluated or analyzed by PP&L to demonstrate acceptability as-is, or to implement plant modifications to resolve the concerns. Where analysis was used to resolve the outliers, the evaluation for seismic loads was based on 5% of critical damping floor response spectral curves that were extrapolated from the existing 0.5% and 1.0% of critical damping spectra curves derived from the SSES DBE, anchored at 0.1g peak ground acceleration.

In addition to the resolution of the outliers, seismic margin assessment of a representative sample of pipe supports on the main drain line has also been conducted, as part of the bounding seismic analysis performed by the licensee for a representative main steam drain piping in the ALT system.

As a result of the walkdowns and the subsequent evaluations (Reference 3), PP&L identified the need for the following modifications:

<u>Unit 1</u>

(1) Main Steam from MSIV to Stop Valve: install restraints as necessary for the hoists above MSIV A to D;

(2) Main Steam Drip Leg Drains: add a new dead load support;

(3) Stop Valve Seat Drains to Condenser: modify Springs SP-MSV-100-H1, H2, H3, & H4; and

(4) HPCI Steam Drain to Main Steam Drain Header: modify Supports SP-EBD-114-H24 & H25.

Unit 2

(1) Main Steam Drain to Condenser: modify Supports H17 & H19;

(2) Main Steam Bypass to Condenser: modify platform;

(3) Main Steam Drip Leg Drains: modify insulation on 4" GBD and 10" HBD Line;

(4) Main Steam Drip Leg Level Instrumentation: modify insulation on 1" DBB & 16" HFD & 14" HBD Line;

(5) Main Steam Averaging Manifold to Pressure Transducer Panel: modify Supports SP-DCD-212-H2603 & H2604; and

The licensee has committed to complete the above modifications prior to the restart of each unit of the plant from its respective upcoming refueling outages. In addition, the licensee also indicated that further walkdowns will be performed on Main Steam Pressure Sensing Lines, and that duct supports will be installed prior to startup, if none exist.

3.2.4 Validation of Earthquake Experience Database

The staff reviewed the earthquake data to assure that the vibratory ground motion, experienced at each of the facilities with piping and equipment being used as a surrogate for that at SSES, did indeed exceed the SSES DBE. The ideal case for the estimation of ground motions would be to have actual recordings of the earthquake ground motion made at each of the facilities. PP&L has indicated (Reference 4) that only about 25 percent of the ground motion estimates in the database in the BWROG Report are from actual instrument recordings at (or near) a facility site. For the other facility-earthquake pairs in the database, the ground motion estimates were extrapolated from instruments located at some distance from the facilities or were made by speculation based on nearby damage or other arguments (Reference 4).

The ground motion from an earthquake at a particular site is a function of the earthquake source characteristics such as the magnitude, focal mechanism, radiation pattern, stress drop, location of asperities and fault rupture history, and depth and orientation of the fault. It is also a function of the distance of the facility to the fault and the propagation properties of the rocks between them. The geology immediately under the facility site can also have a large effect on the amplitude and frequency content of the ground motion. Two of the more appropriate methods of estimating earthquake ground motion, where there are no nearby recordings are: by calibrated numerical modeling of the fault rupture and the wave propagation process, and by the use of empirical attenuation relationships obtained from the statistical analysis of large sets of earthquake data. It has been observed from numerous earthquakes that the variation of peak ground motion values within short distances can be substantial. Consequently, applicability of inferred ground motion parameters needs to be carefully evaluated.

PP&L has stated that the SSES condenser design is typical of those at the Moss Landing Steam Plant, which experienced the Loma Prieta 1989 earthquake, and the Ormond Beach Generating Station, which experienced the Point Mugu 1973 earthquake. They also stated that the SSES main drain and its associated piping as well as the interconnected piping systems are similar to the commercial piping at the Moss Landing Steam Plant, which experienced the Loma Prieta 1989 earthquake, the Ormond Beach Generating Station, which experienced the Point Mugu 1973 earthquake, the El Centro Steam Plant, which experienced the Imperial Valley 1979 earthquake, the Valley Steam Plant, which experienced the San Fernando 1971 earthquake, and the PALCO Co-generation Plant which experienced the Cape Mendocino earthquakes of 1992. In order to allow the NRC staff to evaluate the adequacy of the ground motion estimates made for each of the above facilities, the staff requested additional information from PP&L (Reference 5). The following information was requested for each of the facilities in the earthquake experience database used to demonstrate the seismic adequacy of the Susquehanna ALT system.

1. The name, location, and foundation geology (i.e. rock, deep soil, shallow soil) of the facility.

2. The name, date, time, epicenter, magnitude and distance to the facility of the earthquake.

3. The five percent of critical damping response spectra of the ground motion estimated for the facility due to the earthquake.

4. The method used to estimate the ground motion at the facility. If the ground motion is based on actual ground motion recordings, provide the location and foundation geology of the recording station and its distance from the facility and its distance to the closest part of the fault rupture. If the estimation is based on a method other than an actual recording of the earthquake ground motion or if the recording station is not collocated with the facility, describe the method used to estimate the ground motion in detail and provide any ground motion attenuation equations which may have been used to obtain the estimate.

The PP&L's response (Reference 6) to the above RAI provided some of the information requested. A subsequent telephone conference with PP&L resulted in a letter from PP&L (Reference 4) supplementing the response. This allowed the staff to make independent analyses of the suitability of the ground motion for the earthquake-facility pairs which PP&L has indicated it relied on to demonstrate the seismic adequacy of the Susquehanna ALT system.

The EQE, Inc. ground motion estimate at the El Centro Steam Plant from the Imperial Valley 1979 earthquake was based on a recording made at a U.S. Geological Survey (USGS) strong ground motion station about 1 kilometer from the facility. Because of the density of seismic recordings in that area and the distribution of the ground motion, the staff concluded that the estimated ground motion for the site is significantly larger than the SSES DBE and is, therefore, appropriate for use in establishing the seismic capacity of the SSES piping and equipment similar to that in the El Centro Steam Plant.

The EQE, Inc. ground motion estimate at the Valley Steam Plant from the San Fernando 1971 earthquake was based on an extrapolation of data from a relatively distant location (8 km from the plant). In 1988, the USGS performed studies to estimate the ground motion at selected sites from the San Fernando 1971 earthquake in support of the NRC's resolution of the USI A-46 program (Reference 7). Figure 1 is a plot of the SSES DBE response spectrum, and the EQE, Inc. and USGS estimates of the Valley Steam Plant ground motion response spectrum from San Fernando 1971 earthquake. The USGS spectrum, while lower than the EQE, Inc. spectrum, is significantly higher than the SSES DBE spectrum. As in the resolution of A-46 ground motion issue, the staff considered the USGS estimate to be the characterization of the ground motion at the Valley Steam Plant from the San Fernando 1971 earthquake and is, therefore, the appropriate ground motion estimate for use in establishing the seismic capacity of the SSES piping and equipment similar to that in the Valley Steam Plant.

The EQE, Inc. ground motion estimate at the Moss Landing Steam Plant from the Loma Prieta 1989 earthquake is based on a study performed by Pacific Gas and Electric Company (PG&E) (Reference 8), the owner of the Moss Landing Steam Plant. PG&E provided a copy of the report for the NRC staff's use. The analysis performed by PG&E was found to be technically sound and comprehensive. The staff, therefore, concluded that its estimate of the ground motion is appropriate for use in establishing the seismic capacity of the SSES piping and equipment similar to that in the Moss Landing Steam Plant.

The ground motion estimate at the PALCO Co-generation Plant from the Cape Mendocino magnitude 7 earthquake of 1992 is based on a recording at a California Department of Mines and Geology station in Rio Dell at some distance from the facility. The NRC staff, using two ground motion estimation formulas which are based on the statistical analyses of large sets of empirical data, made its own estimate of the ground motion at PALCO Co-generation Plant from the Cape Mendocino 1992 earthquake. Figure 2 contains a plot of the SSES DBE response spectrum, the EQE, Inc. and the two NRC estimates of the PALCO Co-generation Plant response spectrum from the Cape Mendocino 1992 earthquake. The NRC spectra, while lower than the EQE, Inc. spectrum, are higher than the SSES DBE spectrum. The staff considered the lower bound envelope of the NRC estimates to be the appropriate characterization of the ground motion at the PALCO Co-generation Plant from the Cape Mendocino 1992 earthquake for use in establishing the seismic capacity of the SSES piping and equipment similar to that in the PALCO Co-generation Plant.

In Reference 4, PP&L stated "The EQE ground motion estimate at the Ormond Beach Generating Station from the Point Mugu 1973 earthquake is based on typical California attenuation relationships and independently based on observation and measurement of pipe displacements. The data provided establishes that the documented peak ground accelerations for the Ormond Beach Generating Station are substantially less than the other plants included in this evaluation." Figure 6 of Reference 1 shows PP&L's comparison of the SSES ground response spectrum to PP&L's characterization of the database spectra. PP&L's spectrum for the Ormond Beach Generating Station is a straight line between the frequencies of 25 and 30 Hertz at an acceleration level of 0.15 g, which is higher than the SSES design ground motion of 0.1 g. The NRC staff, using two ground motion estimation formulas which are based on the statistical analyses of large sets of empirical data, made its own estimate of the ground motion at the Ormond Beach Generating Station from the Point Mugu 1973 earthquake. Figure 3 contains a plot of the SSES DBE response spectrum, the PP&L and the two NRC estimates of the ground motion at the Ormond Beach Generating Station from the Point Mugu 1973 earthquake. The staff considered the lower bound envelope of the NRC estimates to be the appropriate characterization of the ground motion at the Ormond Beach Generating Station from the Point Mugu 1973 earthquake for use in establishing the seismic capacity of the SSES piping and equipment similar to that in the Ormond Beach Generating Station. The NRC estimate is lower than the SSES DBE spectrum at frequencies less than 1.1 Hertz. Therefore, the Ormond Beach Generating Station can only be used as an analog for SSES for structures, systems, or components that do not have vibrational modes with resonances below 1.1 Hertz.

Based on the independent analysis of the earthquake experience database, the staff concluded that SSES DBE demand is well below the seismic ground motion which was experienced at the facilities discussed above except as noted for the estimated ground motion at the Ormond Beach Generating Station from the 1973 Point Mugu earthquake. Consequently, the use of the database with the exceptions noted is acceptable for this SSES license amendment request.

3.2.5 Comparison of SSES and Experience Data

The staff reviewed the information provided in the licensee's November 21, 1994 submittal (Reference 1), and found that additional information related to piping and pipe supports would be required from the licensee in order for the staff to complete its review. During the January 24, 1995, meeting with PP&L, the staff discussed the extent of the additional information that would be required. The staff also requested the licensee to perform a bounding dynamic analysis of a representative drain line to address compliance to Appendix A of 10 CFR Part 100. The staff's RAIs were subsequently sent to the licensee on

February 3 (Reference 9), and March 3, 1995 (Reference 3). The licensee, in turn, provided its responses to the above staff requests in the letters of February 21 (Reference 11), March 28 (Reference 3), and April 10, 1995 (Reference 6).

In Reference 3, the licensee provided a database for main steam and process piping at the above mentioned plants, i.e., Valley Steam Plant, Ormond Beach Power Plant, El Centro Steam Plant, and Moss Landing Power Plant. Data provided included pipe diameter, schedule, wall thickness and pipe diameter-to-thickness (D/t) ratio. The licensee also provided a data comparison between the SSES ALT system piping and the piping of the above selected database facilities. The SSES data were presented on a system-by-system breakdown, through the ALT path, that included both seismically and non-seismically analyzed systems. The data were presented categorically for pipe diameter, schedule, wall thickness and pipe diameter-to-thickness ratio. The staff found the above database of the facilities' process piping to be an adequate upgrade of the original database provided in the BWROG Report (Reference 2). The staff also found that the Susquehanna ALT piping data are mostly enveloped by the experience database discussed in Reference 3. For the cases where the D/t ratios of the Susquehanna ALT piping are not enveloped by the database piping, the Susquehanna D/t ratios are favorably smaller than the respective values of the database piping. For the 1/2" diameter main steam drain line at Susquehanna, no counterpart seismic experience data is available. However, the associated D/t ratio was found to be favorably smaller than those of the other database pipe sizes. The staff determined that the upgraded database provided by the licensee is adequate for comparison with the corresponding Susquehanna ALT piping and is acceptable.

As indicated in Reference 6, the licensee has compared the structural characteristics of the SSES main condenser to those of similar database condensers which have experienced significant earthquakes as addressed in the BWROG Report. The SSES condenser is made up of three shells: high pressure, intermediate pressure, and low pressure units. Each condenser shell is specifically compared to the database condensers from Moss Landing, Units 6 and 7, and Ormond Beach, Units 1 and 2. The licensee stated that these condensers have physical arrangements and construction details similar to the SSES condenser and would function similarly in their responses to seismic excitations. The licensee also provided the overall dimensions and weights for each of the three condenser shells. By comparison to the database condensers, the licensee stated that most of the physical features of the SSES condenser structure are either enveloped by, or less critical than, the database condensers, with one possible exception.

The SSES condenser is generally higher than the database condensers. The larger ratio of height to base width is likely to cause larger overturning moments and, hence, larger stresses in the shell. The licensee did not consider the effects of this greater height to be critically significant, based on (1) the operating weight of each Susquehanna condenser shell in comparison to the shell side area is comparable to that of the database condensers, and, as a result, the shear stress in the shell plate would not be significantly greater than that of the database condensers under the same seismic load, and (2) the anchors for the SSES condenser assembly have been determined to have more than enough capacity to prevent overturning caused by the combined DBE seismic forces and operating loads.

Based on the above, the staff determined that the SSES MSIV ALT system components are generally enveloped by the database components, and that they possess sufficient capability to withstand the combined DBE seismic forces and operating loads.

3.2.6 Analyses for Alternate Leakage Treatment Pathway

As indicated in Reference 1, the main steam lines from containment isolation valves to the turbine stop valves, the bypass piping from the main steam lines to the main condenser, the main steam drain line header from containment isolation valves to in-line pipe anchors, and portions of main steam branch connection lines to in-line pipe anchors were seismically analyzed. These piping systems at SSES were designed in accordance with the requirements of ASME Code Section III, Class 2, 1971 Edition including Winter 1972 Addenda, and ANSI B31.1, 1973 Edition. These piping systems were designed using reactor building and turbine building response spectra inputs of Operating Basis Earthquake (OBE) and DBE, in combination with other applicable design loads. The analysis results satisfy the allowable limits specified for Class 2 pipes in the ASME Section III Code.

The licensee stated that the remaining portion of main steam drain and associated piping were analyzed for dead weight and thermal loads using a combination of piping analysis and spacing criteria, without consideration of seismic loads. These non-safety related pipes are generally composed of welded steel piping and standard support components, and are similar to piping found in the seismic experience database. The system is predominantly supported for dead weight utilizing rod hangers, constructed from standard support catalog parts typically consisting of clamps, threaded rods, weldless eye nuts, turnbuckles, welding legs and are attached to either concrete or structure steel. The objective of the assessment of the non-seismic main steam drain piping is to demonstrate that piping position retention will be maintained during a seismic event and hence provide assurance that the pipe supports will behave in a ductile manner and that all lines are free of known seismic hazards. Ultimately, it will ensure that these SSES piping systems will perform in a manner similar to piping and supports that have been observed to demonstrate good seismic performance.

As stated by PP&L in the above discussion, the non-seismically analyzed main steam drain and associated piping are generally bounded in diameter and diameter-to-thickness ratio by those installed in the earthquake experience database facilities, as evidenced in the BWROG Report and the supplemental updated earthquake performance data discussed above. PP&L has stated that upon completion of all necessary modifications, piping position retention and pressure boundary integrity will be maintained by the deadweight supports under normal and earthquake loadings.

Based on the above, the staff determined that the SSES non-seismically analyzed main steam system piping and condenser that will be used for the ALT system compared well with the earthquake experience database, and that the seismic verification walkdowns of the system and subsequent evaluations have addressed characteristics associated with the limited component damage situations observed at the database facilities. The staff also determined that PP&L has taken proper measures to ensure resolution for all of the identified outliers.

3.2.7 Bounding Seismic Analysis

During the January 24, 1995, meeting, and subsequently in the March 3, 1995, RAI (Reference 5), the staff requested that a bounding seismic analysis be performed for a representative main steam drain piping in the ALT system which had not been seismically analyzed. This analysis is necessary in order to supplement the BWROG's earthquake experience methodology and to further demonstrate analytically that the proposed ALT piping system will maintain its functionality under the Susquehanna DBE.

The piping selected consists of the main steam, reactor core isolation cooling (RCIC), and high pressure coolant injection (HPCI) turbine steam line drains from valves F019, F026, and F029, respectively, to the HP condensers. The Unit 1 model envelopes approximately 283 feet of 4-inch schedule 120 piping which is supported by 23 hangers. These hangers include anchor, spring, structural members, hanger rod and strut. Four pipe supports were selected from each unit for the seismic margin evaluation. The selection was based on an overview of the support configurations and the original design qualification. The selected population included both vertical and lateral restraint type supports.

Dead weight and operating mechanical loads, in addition to the seismic DBE loads, are accounted for in the dynamic analysis. The actual piping stress and pipe support loads due to these loading combinations are calculated by performing ME-101 computer analysis.

The methodology utilized to demonstrate the seismic adequacy of the non-seismically designed main steam drain lines is called Conservative Deterministic Failure Margin (CDFM), as described in the EPRI report, EPRI NP-6041, dated August 1991. Although this methodology has not been approved by the NRR staff for licensing reviews involving Seismic Category I systems, the staff determined that, in consideration of the available safety margins demonstrated by PP&L, its employment to demonstrate the functional operability of the ALT system to be adequate.

During the original design of the plant, seismic floor response spectra were generated, for 0.5% and 1.0% of critical damping, to determine seismic anchor forces and displacements for the piping systems that are attached to the turbine building. For the bounding analysis, 5% of critical damping floor response spectra were recommended by the above EPRI report. These were obtained by extrapolation from the existing floor response spectra of 0.5% and

1.0% values. The results of the bounding analysis were provided by PP&L in Reference 10. It indicated that the pipe stresses are within the allowable limits for sustained loads, occasional loads as well as thermal expansion, with safety margins of 3.34, 2.47 and 3.95, respectively. Supports, which include steel support members, anchor plates, welds, and anchor bolts, were also evaluated and found to be within the allowable limits. The staff found these seismic analysis results acceptable for providing confirmation to the seismic adequacy of the ALT system that was established on the basis of the earthquake database.

3.2.8 Seismic Dynamic Analysis of Turbine Building

The SSES turbine building (TB) is entirely supported on competent rock with reinforced concrete retaining walls extending up to grade level (Reference 1). As stated in the SSES FSAR section 3.7b.2.4 and in the staff's April 1981 safety evaluation report (SER (NUREG-0776)), the seismic dynamic analysis of the TB was done assuming a fixed base. The superstructure is framed with structural steel and reinforced concrete (RC), with exterior walls made of pre-cast RC panels except for the upper 30 feet (ft) which is metal siding. Each of the two turbine generator units housed in the TB is supported on a free standing RC pedestal extending down to the bedrock. The DBE peak ground acceleration at SSES is 0.1g, and the OBE peak ground acceleration is 0.05g. For the analysis, the ground motion was applied at the foundation level of the TB.

As stated in Reference 6, PP&L did not perform a seismic re-analysis of the TB in connection with this license amendment request. Instead, PP&L utilized the results of the original seismic analysis of the plant. The dynamic analysis of the TB was performed by the response spectrum method. Separate analyses were made for the vertical and two horizontal directions to calculate shears, moments, and deflections. Time history analyses were performed utilizing the two horizontal and one vertical models in order to calculate the floor response spectra (FRS). During the original design of SSES, FRS curves were generated for 0.5% and 1% of critical damping. The FRS curves were broadened by +/-20 percent to account for the parametric variations associated with the structure frequency, structure damping and the soil moduli. For the purpose of the MSIV license amendment request, PP&L extrapolated the existing 0.5% and 1% of critical damping FRS curves to generate the 5% of critical FRS curves by mainly using what is called the power method. This method enables the extrapolation of the spectral acceleration values, frequency by frequency, using the spectral values from both the 0.5% and the 1% of critical damping curves to determine the spectral values for the 5% FRS curves. For the few locations where only one set of FRS curves (either for 0.5% or 1% of critical damping) was available, PP&L used the square root method to generate the 5% curves. The staff considered the licensee's procedures to generate the 5% of critical damping FRS curves for piping and equipment using the existing FRS curves reasonable and adequate for this MSIV license amendment.

Reference 6 states that the structural acceptance criterion for the building did not permit the material to reach its yield limit under the loading combination including the DBE. The use of dynamic analyses in conjunction with the no-yielding criterion ensures that the turbine building will remain elastic during a DBE. The staff concluded that the turbine building will withstand a DBE and is adequate for this license amendment request.

3.2.9 Anchorage for piping

As part of the previously stated bounding seismic analysis, PP&L performed piping anchorage evaluation for the selected eight representative supports. PP&L stated in Reference 6 that pipe support loads resulting from the loading combination of dead weight, thermal expansion, and DBE were calculated and then compared with the anchorage capacity which was evaluated using Appendix C to Generic Implementation Procedure (GIP) criteria established by the Seismic Qualification Utility Group (SQUG). The comparison results indicated that the anchorage capacities were greater than the seismic demand. The licensee also stated that no concrete cracking around anchor bolts was found during recent walkdowns. For the evaluation of the adequacy of equipment anchorage at older operating plants, the staff has accepted the GIP criteria for anchorage. Based on this, the staff concluded that the anchorage for the ALT piping at Susquehanna is adequate.

3.2.10 Anchorage for Condenser

As stated previously, the condenser is made up of three shells: high pressure, intermediate pressure, and low pressure shells. Each shell is independently supported on the concrete base slab of the turbine pedestal by six embedded plate assemblies. Positive attachment is provided by anchor bolts and welds to the embedded plate assemblies. Forces in the anchoring systems of these condenser shells were calculated for the DBE loading. Capacities of the anchoring systems were also calculated. The results indicated that the tension capacities of bolts in the three condenser shells are about two times the values of the demand capacities due to the DBE, and that the shear capacities of bolts are about four times the demand capacities. Based on the above, the licensee determined that the overall anchor system of the SSES condenser was capable of withstanding the calculated DBE loads, in combination with operating loads. Based on its review of the licensee's information, the staff concluded that the condenser anchorage is adequate.

3.2.11 Findings

Based on the above evaluation, the staff concludes that upon completion of the plant modifications necessary for the identified outliers, there is reasonable assurance that the Susquehanna main steam lines, main steam drain lines, condenser, and associated interconnected piping and supports will be seismically adequate for the proposed MSIV ALT system. The staff's conclusion is based on (1) the staff's independent analysis of the earthquake experience database confirmed that the DBE demand at SSES is well below the seismic ground motion that was experienced at the facilities in the earthquake experience database except for the estimated ground motion at the Ormond Beach Generating Station from the 1973 Point Mugu earthquake at frequencies below 1.1 Hertz, (2) the Susquehanna main condenser is generally enveloped by the condensers in the earthquake experience database, and that the condenser assembly has sufficient anchorage capacity, (3) the majority of the main steam system piping was seismically analyzed as part of the initial design of the plant, (4) the non-seismically analyzed ALT pipes are represented by those in the earthquake experience database that demonstrated good seismic performance, (5) the bounding seismic analysis performed for the non-seismic portion of main steam drain lines indicated adequate safety margins for piping stresses and support loads, and (6) the turbine building has adequate capability to withstand the DBE loads. The staff, therefore, concludes that the licensee's proposed ALT system complies with Appendix A of 10 CFR Part 100 and is acceptable.

It should be noted that the staff's acceptance of the experience-based methodology as presented by the BWROG and PP&L is restricted to its application for ensuring the pressure boundary integrity and functionality of the MSIV alternate leakage treatment system. The staff's acceptance of the methodology for this application is not an endorsement for the use of the experience-based methodology for other applications at Susquehanna.

3.4 Plant Systems Evaluation

The proposed MSIV leakage alternate (alternate to the existing MSIV-LCS and alternate to RG 1.96) drain pathway is considered the primary success (credited) pathway for "treating" MSIV leakage following a LOCA and employs a MSL drain line downstream of the MSIVs. There are two motor operated valves (MOVs) in series in this line between the MSLs and the main condenser. Both valves must be open to establish the required drain path. The first (upstream) MOV, F020, is normally open and will fail "as-is" on a loss of power. The second (downstream) MOV, F021, is normally closed (with a small bypass orifice around the valve to allow drainage during normal operation) and is required to be opened following the design basis LOCA to establish a drain path to support the radiological analysis. Both valves are powered from Class IE sources. The staff requested the licensee to address the single failure of this downstream valve to open on demand, due to a valve or power supply failure.

In its May 24, 1995 submittal, the licensee stated that the downstream valve is powered from a bus which is supplied from two independent offsite sources and a diesel generator providing a highly reliable power source to the valve. To increase the reliability of the MOV itself, the valve will be included in the Susquehanna inservice test (IST) program. The valve will be stroke tested (open) once per cycle in accordance with the program. Engineering evaluations of the F021 valve operator have also been performed to verify the valve's capability to function under the postulated accident conditions.

However, given that the valve may be highly reliable, the licensee; nonetheless, evaluated the effects of a failure of the valve to open and demonstrated that other adequate (secondary) flow paths would still be available. The licensee verified there are a number of different orificed pathways that are included in the boundary of the MSIV leakage alternate drain pathway that would be available to convey MSIV leakage to the isolated condenser if the downstream valve (F021) fails to open. None of these secondary drain paths require the opening of any valves. These "backup" or secondary drain paths provide orificed flow pathways, which ensure that even with the failure of a valve in the primary flow path, flow will be directed to the main condenser at the same or lower elevation as that assumed in the radiological dose calculation. The radiological analysis did not take credit for these open pathways. Therefore, these backup pathways will ensure sufficient flow to the main condenser and will act to reduce the radiological impact to within the regulatory limits. Thus the backup paths will convey essentially all of the MSIV leakage to the main condenser. Consequently, the radiological dose assessment for these backup pathways should be equivalent to the dose assessment for the primary path. Additionally, the licensee has committed to update the Operating and/or Emergency Operating Procedures as necessary to address the alternate and backup leakage treatment methods. Based on the above, the staff concludes that the proposed design provides a reliable leakage treatment method which, overall, satisfies the single failure criterion of GDC 41, "Containment Atmosphere Cleanup." The staff; therefore, concludes the proposed design is acceptable and the TSs associated with the LCS can be deleted from the plant TSs.

The licensee further proposed new requirements in TS Section 3.6.1.2 related to restoration of acceptable leak rates if any of the proposed limits are exceeded. The new requirements basically require that if any single MSIV leakage rate exceeds 100 scfh, it will be repaired and retested to meet a leak rate limit of 11.5 scfh per valve (the current criterion for leakage) and that the maximum total leak rate will be restored to less than or equal to 300 scfh whenever the 300 scfh limit is exceeded. The staff concludes that this new requirement will restore the leakage rates to values that are consistent with the revised radiological analysis and is; therefore, 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 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 (60 FR 503). 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 and findings discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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