February 27, 1998

Mr. Harold W. Keiser
Executive Vice President -
Nuclear Business Unit
Public Service Electric & Gas
Company
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: 10 CFR PART 50, APPENDIX J, OPTION B (TAC NOS. MA0289 AND MA0290)

Dear Mr. Eliason:

The Commission has issued the enclosed Amendment Nos. 207 and 188 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 15, 1997.

These amendments revise the TSs to adopt Option B, of 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to implement a performance-based approach for Type B and C testing. Additionally, the wording in the TSs would be modified for the previous adoption of Option B on Type A testing and a section added on the primary containment leakage rate testing program.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
/S/
Patrick D. Milano, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-272/311

Enclosures: 1. Amendment No. 207 to License No. DPR-70
2. Amendment No. 188 to License No. DPR-75
3. Safety Evaluation

cc w/encls: See next page

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

February 27, 1998

Mr. Harold W. Keiser
Executive Vice President -
Nuclear Business Unit
Public Service Electric & Gas
Company
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: 10 CFR PART 50, APPENDIX J, OPTION B (TAC NOS. MA0289 AND MA0290)

Dear Mr. Keiser:

The Commission has issued the enclosed Amendment Nos. 207 and 188 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 15, 1997.

These amendments revise the TSs to adopt Option B, of 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to implement a performance-based approach for Type B and C testing. Additionally, the wording in the TSs would be modified for the previous adoption of Option B on Type A testing and a section added on the primary containment leakage rate testing program.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Patrick D. Milano, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-272/311

Enclosures: 1. Amendment No. 207 to License No. DPR-70
2. Amendment No. 188 to License No. DPR-75
3. Safety Evaluation

cc w/encls: See next page

Mr. Harold W. Keiser
Public Service Electric & Gas
Company

Salem Nuclear Generating Station,
Units 1 and 2

cc:

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 207
License No. DPR-70

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

   A. The application for amendment filed by the Public Service Electric & Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated December 15, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and 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 rules and 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 Facility Operating License No. DPR-70 is hereby amended to read as follows:

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(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 207 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance to be implemented within 60 days.

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: February 27, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 207

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 6-2	3/4 6-2
3/4 6-5	3/4 6-5
3/4 6-5a	3/4 6-5a
3/4 6-8	3/4 6-8
6-19	6-19
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate (Type A test) in accordance with the Containment Leakage Rate Testing Program.

b. A combined leakage rate in accordance with the Containment Leakage Rate Testing Program for all penetrations and valves subject to Type B and C tests.

APPLICABILITY: MODES 1, 2, 3 and 4. * *

ACTION:

With either (a) the measured overall integrated containment leakage (Type A test) not in accordance with the Containment Leakage Rate Testing Program, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests not in accordance with the Containment Leakage Rate Testing Program, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated as follows:

a. Type A tests shall be in accordance with the Containment Leakage Rate Testing Program.

b. Type B and C tests shall be conducted in accordance with the Containment Leakage Rate Testing Program.

c. Air locks shall be tested and demonstrated OPERABLE in accordance with the Containment Leakage Rate Testing Program.

* * A one-time change is granted to have the containment purge supply and/or exhaust isolation valves open in Modes 3 and 4 following the steam generator replacement outage (1R13). The cumulative time for having the valves open in Modes 3 and 4 is limited to fourteen (14) days. Each valve will be immobilized in the shut position prior to initial entry into Mode 2. The one-time exemption expires with initial entry into Mode 2 following 1R13.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage in accordance with the Containment Leakage Rate Testing Program.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With an air lock inoperable, restore the air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By verifying seal leakage rate in accordance with the Containment Leakage Rate Testing Program.

b. By conducting an overall air lock leakage test in accordance with the Containment Leakage Rate Testing Program.

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITIONS FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Surfaces The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined in accordance with the Containment Leakage Rate Testing Program.

4.6.1.6.2 Reports Any abnormal degradation of the containment structure detected during the above required inspections shall be evaluated for reportability pursuant to 10CFR50.72 and 10CFR50.73. The evaluation shall be documented and shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective action taken.

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System Subcooling Margin. This program shall include the following:

(i) Training of personnel, and

(ii) Procedures for monitoring

e. Postaccident Sampling

A program* which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(i) Training of personnel

(ii) Procedures for sampling and analysis,

(iii) Provisions for maintenance of sampling and analysis equipment.

6.8.4.f. Primary Containment Leakage Rate Testing Program

A program shall be established, implemented, and maintained to comply with the leakage rate testing of the containment as required by 10CFR50.54(o) and 10CFR50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 47.0 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of primary containment air weight per day.

Leakage Rate Acceptance Criteria are:

a. Primary containment leakage rate acceptance criterion is less than or equal to 1.0 L_a. During the first unit startup

*It is acceptable if the licensee maintains details of the program in plant operation manuals.

following testing in accordance with this program, the leakage rate acceptance criteria are less than or equal to 0.6 L_a for Type B and Type C tests and less than or equal to 0.75 L_a for Type A tests;

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is less than or equal to 0.05 L_a when tested at greater than or equal to P_a,

2) Seal leakage rate less than or equal to 0.01 L_a per hour when the gap between the door seals is pressurized to 10.0 psig.

Test frequencies and applicable extensions will be controlled by the Primary Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 will be applied to the Primary Containment Leakage Rate Testing Program.

3/4.6 CONTAINMENT SYSTEMS

BASES

==========

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 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 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the accident pressure, Pa. As an added conservatism, the measured overall integrated leakage rate(Type A test)is further limited to $\leq 0.75\ L_a$ or $\leq 0.75\ L_t$, as applicable, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is consistent with the Containment Leakage Rate Testing Program.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and the Containment Leakage Rate Testing Program. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.5 psig and 2) the containment peak pressure does not exceed the design pressure of 47 psig during the limiting pipe break conditions. The pipe breaks considered are LOCA and steam line breaks.

The limit of 0.3 psig for initial positive containment pressure is consistent with the accident analyses initial conditions.

The maximum peak pressure expected to be obtained from a LOCA or steam line break event is ≤ 47 psig.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a LOCA or steam line break. In order to determine the containment average air temperature, an average is calculated using measurements taken at locations within containment selected to provide a representative sample of the overall containment atmosphere.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the design pressure. The visual inspections of the concrete and liner and the Type A leakage test both in accordance with the Containment Leakage Rate Testing Program are sufficient to demonstrate this capability.

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 188
License No. DPR-75

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

   A. The application for amendment filed by the Public Service Electric & Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated December 15, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and 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 rules and 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 Facility Operating License No. DPR-75 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No.188 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

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: February 27, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 188

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-4	3/4 6-4
3/4 6-5	3/4 6-5
3/4 6-8	3/4 6-8
6-19	6-19
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except for valves that may be opened under Administrative control as permitted by Specification 3.6.3.1, and all equipment hatches are closed and sealed.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

c. After each closing of a penetration subject to Type B testing, except containment air locks, if opened following a Type A or B test, by leak rate testing in accordance with the Containment Leakage Rate Testing Program.

* Except vents, drains, test connections, etc. which are (1) one inch nominal pipe diameter or less, (2) located inside the containment, and (3) locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed at least once per 92 days.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate (Type A test) in accordance with the Containment Leakage Rate Testing Program.

b. A combined leakage rate in accordance with the Containment Leakage Rate Testing Program for all penetrations and valves subject to Type B and C tests.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate (Type A test) not in accordance with the Containment Leakage Rate Testing Program or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests not in accordance with the Containment Leakage Rate Testing Program, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated as follows:

a. Type A tests shall be in accordance with the Containment Leakage Rate Testing Program.

b. Type B and C tests shall be conducted in accordance with the Containment Leakage Rate Testing Program.

c. Air locks shall be tested and demonstrated OPERABLE in accordance with the Containment Leakage Rate Testing Program.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate in accordance with the Containment Leakage Rate Testing Program

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT STANDBY within the next six hours and in COLD SHUTDOWN within the following 30 hours.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next six hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

CONTAINMENT AIR LOCKS

SURVEILLANCE REQUIREMENTS (Continued)

a. By verifying seal leakage rate in accordance with the Containment Leakage Rate Testing Program

b. By conducting an overall air lock leakage test in accordance with the Containment Leakage Rate Testing Program

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITIONS FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Surfaces The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined in accordance with the Containment Leakage Rate Testing Program

4.6.1.6.2 Reports Any abnormal degradation of the containment structure detected during the above required inspections shall be evaluated for reportability pursuant to 10CFR50.72 and 10CFR50.73. The evaluation shall be documented and shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective action taken.

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System Subcooling Margin. This program shall include the following:

(i) Training of personnel, and

(ii) Procedures for monitoring

e. Postaccident Sampling

A program* which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(i) Training of personnel

(ii) Procedures for sampling and analysis,

(iii) Provisions for maintenance of sampling and analysis equipment.

6.8.4.f. Primary Containment Leakage Rate Testing Program

A program shall be established, implemented, and maintained to comply with the leakage rate testing of the containment as required by 10CFR50.54(o) and 10CFR50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 47.0 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of primary containment air weight per day.

Leakage Rate Acceptance Criteria are:

a. Primary containment leakage rate acceptance criterion is less than or equal to 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate

*It is acceptable if the licensee maintains details of the program in plant operation manuals.

acceptance criteria are less than or equal to 0.6 L_a for Type B and Type C tests and less than or equal to 0.75 L_a for Type A tests;

b. Air lock testing acceptance criteria are:
   1) Overall air lock leakage rate is less than or equal to 0.05 L_a when tested at greater than or equal to P_a,

   2) Seal leakage rate less than or equal to 0.01 L_a per hour when the gap between the door seals is pressurized to 10.0 psig.

Test frequencies and applicable extensions will be controlled by the Primary Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 will be applied to the Primary Containment Leakage Rate Testing Program.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 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 CONTAINMENT LEAKAGE

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

The surveillance testing for measuring leakage rates are consistent with the Containment Leakage Rate Testing Program.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and the Containment Leakage Rate Testing Program. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.5 psig, and 2) the containment peak pressure does not exceed the design pressure of 47 psig during the limiting pipe break conditions. The pipe breaks considered are LOCA and steam line breaks.

The limit of 0.3 psig for initial positive containment pressure is consistent with the accident analyses initial conditions.

The maximum peak pressure expected to be obtained from a LOCA or steam line break event is ≤ 47 psig.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a LOCA or steam line break. In order to determine the containment average air temperature, an average is calculated using measurements taken at locations within containment selected to provide a representative sample of the overall containment atmosphere.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the design pressure. The visual inspections of the concrete and liner and the Type A leakage test, both in accordance with the Containment Leakage Rate Testing Program, are sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 207 AND 188 TO FACILITY OPERATING

LICENSE NOS. DPR-70 AND DPR-75

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-272 AND 50-311

1.0 INTRODUCTION

On September 12, 1995, the U.S. Nuclear Regulatory Commission (NRC) approved issuance of a revision of Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to Title 10 of Part 50 of the Code of Federal Regulations (10 CFR 50, Appendix J) which was published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. With that revision, the NRC added Option B, "Performance-Based Requirements," to allow licensees to voluntarily replace the prescriptive testing requirements of 10 CFR 50, Appendix J, with testing requirements based on both overall leakage rate performance and the performance of individual components.

By letter dated December 15, 1997, the Public Service Electric & Gas Company (the licensee) submitted a request for changes to the Salem Nuclear Generating Station, Unit Nos. 1 and 2, Technical Specifications (TSs). The requested changes would revise the Technical Specifications (TSs) to adopt Option B, of 10 CFR Part 50, Appendix J, to implement a performance-based approach for Type B and C testing. The licensee has established a "Containment Leakage Rate Testing Program" and proposed adding this program to the TSs. The program references Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, which specifies a method acceptable to the NRC for complying with Option B. Additionally, the licensee proposed to modify the wording in the TSs related to the previously approved adoption of Option B on Type A testing.

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2.0 BACKGROUND

Compliance with 10 CFR 50, Appendix J, provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TSs and its Bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. Appendix J of 10 CFR 50 was considered for this initiative, and the NRC staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program."

Based on the results of this study, the staff developed a perfomance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B, "Performance-Based Requirements," to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

RG 1.163 was developed as a method acceptable to the NRC staff for implementing Option B. This RG states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that the RG, or other implementation document used by a licensee to develop a performance-based leakage rate testing program, be included, by general reference, in the plant TSs. The licensee has referenced RG 1.163 in the Salem Unit Nos. 1 and 2 TSs.

RG 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests, and Type C tests may be extended up to 5 years based upon two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TSs to implement Option B. After some discussion, the NRC staff and NEI agreed on final TSs which were attached to a letter from C. Grimes (NRC) to D. Modeen (NEI) dated November 2, 1995. These TSs are to serve as a model for licensees to develop plant-specific TSs in preparing amendment requests to implement Option B.

For a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation.

Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TSs requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires the licensee to maintain records to show that the criteria for Type A, B, and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adquate. These records are subject to NRC inspection.

3.0 EVALUATION

Option B permits a licensee to choose Type A; or Type B and C; or Type A, B, and C testing to be done on a performance basis. In amendments 184 and 166 dated July 11, 1996, the NRC has previously approved Option B for Type A testing at Salem Unit Nos. 1 and 2, respectively. The licensee is now electing to perform Type B and C testing on a performance basis.

In its letter of December 15, 1997, the licensee proposed to establish a "Primary Containment Leakage Rate Testing Program" and add this program to the TSs. The program references RG 1.163, which specifies a method acceptable to the NRC staff for complying with Option B. This requires changes to the TSs through the addition of new TS 6.8.4.f, "Primary Containment Leakage Rate Testing Program," and the addition of the references to the "Primary Containment Leakage Rate Testing Program" in TSs 4.6.1.1 (Unit 2 only), 3/4.6.1.2, 3/4.6.1.3, 4.6.1.6.1 and to the applicable Bases sections.

The TSs changes proposed by the licensee are in comformance with the requirements of Option B and consistent with the guidance of RG 1.163, and the generic TS of the November 2, 1995, letter and are, therefore, acceptable to the staff.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (63 FR 2281). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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