September 18, 1

Mr. Leon R. Eliason
Chief Nuclear Officer & President-
Nuclear Business Unit
Public Service Electric & Gas
Company
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION (TAC NO. M98318)

Dear Mr. Eliason:

The Commission has issued the enclosed Amendment No. 104 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station. This amendment consists of changes to the Technical Specifications (TSs) and License in response to your application dated April 1, 1997, as supplemented by letter dated May 30, 1997.

This amendment provides changes to Technical Specifications (TSs) 4.6.1.1, "Primary Containment Integrity," 3/4.6.1.2, "Primary Containment Leakage," 3/4.6.1.3, "Primary Containment Air Locks," 4.6.1.5.1, "Primary Containment Structural Integrity," and 4.6.1.8.2, "Drywell and Suppression Chamber Purge System." This amendment also provides changes to the Bases for 3/4.6.1.2, "Primary Containment Leakage," 3/4.6.1.3, "Primary Containment Air Locks," 3.4.6.1.5, "Primary Containment Structural Integrity," Section 6, "Administrative Controls," and License Condition 2.D of Facility Operating License NPF-57. A new TS, 6.8.4.f, "Primary Containment Leakage Rate Testing Program," is added. These changes modify the TSs and the Facility Operating License to adopt the performance based containment leak rate testing requirements (Option B) of 10 CFR Part 50, Appendix J.

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/
David H. Jaffe, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures: 1. Amendment No.104 to License No. NPF-57
2. Safety Evaluation

cc w/encls: See next page

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

September 18, 1997

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Public Service Electric & Gas
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Enclosures: 1. Amendment No. 104 to
License No. NPF-57
2. Safety Evaluation

cc w/encls: See next page

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 104
License No. NPF-57

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

   A. The application for amendment filed by the Public Service Electric & Gas Company (PSE&G) dated April 1, 1997, as supplemented by letter dated May 30, 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 License Condition 2.D on page 6 of Facility Operating License No. NPF-57*, and the license is also 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. NPF-57 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. 104 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSE&G shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance, to be implemented within 30 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

Attachments: 1. Changes to page 6 of Lciense No. NPF-57
2. Changes to the Technical Specifications

Date of Issuance: September 18, 1997

*Page 6 of the license is attached, for convenience, for the composite license to reflect this change.

ATTACHMENT TO LICENSE AMENDMENT NO. 104

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

1.

Remove	Insert
Page 6 of License	Page 6 of License

2. Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
3/4 6-5	3/4 6-5
3/4 6-6	3/4 6-6
3/4 6-8	3/4 6-8
3/4 6-11	3/4 6-11
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2
-	6-16b

(13) Safety Parameter Display System (Section 18.2, SSER No. 5)

Prior to the earlier of 90 days after restart from the first refueling outage or July 12, 1988, PSE&G shall add the following parameters to the SPDS and have them operational:

a. Primary containment radiation
b. Primary containment isolation status
c. Combustible gas concentration in primary containment
d. Source range neutron flux

(14) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. are hereby incorporated into this license. Public Service Electric and Gas Company shall operate the facility in accordance with the Additional Conditions.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. An exemption from the criticality alarm requirements of 10 CFR 70.24 was granted in Special Nuclear Material License No. 1953, dated August 21, 1985. This exemption is described in Section 9.1 of Supplement No. 5 to the SER. This previously granted exemption is continued in this operating license. An exemption from certain requirements of Appendix A to 10 CFR Part 50, is described in Supplement No. 5 to the SER. This exemption is a schedular exemption to the requirements of General Design Criterion 64, permitting delaying functionality of the Turbine Building Circulating Water System-Radiation Monitoring System until 5 percent power for local indication, and until 120 days after fuel load for control room indication (Appendix R of SSER 5). Exemptions from certain requirements of Appendix J to 10 CFR Part 50, are described in Supplement No. 5 to the SER. These include an exemption from the requirement of Appendix J, exempting main steam isolation valve leak-rate testing at 1.10 Pa (Section 6.2.6 of SSER 5); an exemption from Appendix J, exempting Type C testing on traversing incore probe system shear valves (Section 6.2.6 of SSER 5); an exemption from Appendix J, exempting Type C testing for instrument lines and lines containing excess flow check valves (Section 6.2.6 of SSER 5); and an exemption from Appendix J, exempting Type C testing of thermal relief valves (Section 6.2.6 of SSER 5). These exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. These exemptions are hereby granted. The special circumstances regarding each exemption are identified in the referenced section of the safety evaluation report and the supplements thereto. These exemptions are granted pursuant to 10 CFR 50.12. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

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

b. At least once per 31 days by verifying that all primary containment 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 position, except as provided in Table 3.6.3-1 of Specification 3.6.3.

c. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. By verifying the suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

---

*See Special Test Exception 3.10.1

**Except valves, blind flanges, and deactivated automatic valves which are located inside the primary containment, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been de-inerted since the last verification or more often than once per 92 days.

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 (Type A test) in accordance with the Primary Containment Leakage Rate Testing Program.

b. A combined leakage rate in accordance with the Primary Containment Leakage Rate Testing Program for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves*, valves which form the boundary for the long-term seal of the feedwater lines, and other valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests.

c. *Less than or equal to 46.0 scfh combined through all four main steam lines when tested at 5 psig (seal system ΔP).

d. A combined leakage rate of less than or equal to 10 gpm for all containment isolation valves which form the boundary for the long-term seal of the feedwater lines in Table 3.6.3-1, when tested at 1.10 Pa, 52.9 psig.

e. A combined leakage rate of less than or equal to 10 gpm for all other penetrations and containment isolation valves in hydrostatically tested lines in Table 3.6.3-1 which penetrate the primary containment, when tested at 1.10 Pa, 52.9 psig Δp.

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

ACTION:

With:

a. The measured overall integrated primary containment leakage rate (Type A test) not in accordance with the Primary Containment Leakage Rate Testing Program, 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*, valves which form the boundary for the long-term seal of the feedwater lines, and other valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests not in accordance with the Primary Containment Leakage Rate Testing Program, or

c. The measured leakage rate exceeding 46.0 scfh combined through all four main steam lines, or

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

ACTION (Continued)

d. The measured combined leakage rate for all containment isolation valves which form the boundary for the long-term seal of the feedwater lines in Table 3.6.3-1 exceeding 10 gpm, or

e. The measured combined leakage rate for all other penetrations and containment isolation valves in hydrostatically tested lines in Table 3.6.3-1 which penetrate the primary containment exceeding 10 gpm,

restore:

a. The overall integrated leakage rate(s) (Type A test) to be in accordance with the Primary Containment Leakage Rate Testing Program, 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*, valves which form the boundary for the long-term seal of the feedwater lines, and other valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests to be in accordance with the Primary Containment Leakage Rate Testing Program, and

c. The leakage rate to less than or equal to 46.0 scfh combined through all four main steam lines, and

d. The combined leakage rate for all containment isolation valves which form the boundary for the long-term seal of the feedwater lines in Table 3.6.3-1 to less than or equal to 10 gpm, and

e. The combined leakage rate for all other penetrations and containment isolation valves in hydrostatically tested lines in Table 3.6.3-1 which penetrate the primary containment to less than or equal to 10 gpm,

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

SURVEILLANCE REQUIREMENTS

4.6.1.2.a The primary containment leakage rates shall be demonstrated in accordance with the Primary Containment Leakage Rate Testing Program for the following:

1. Type A test.

2. Type B and C tests (including air locks).

b. DELETED.

c. DELETED.

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

SURVEILLANCE REQUIREMENTS (Continued)

d. DELETED.

e. DELETED.

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

g. Containment isolation valves which form the boundry for the long-term seal of the feedwater lines in Table 3.6.3-1 shall be hydrostatically tested at 1.10 P_a, 52.9 psig, at least once per 18 months.

h. All containment isolation valves in hydrostatically tested lines in Table 3.6.3-1 which penetrate the primary containment shall be leak tested at least once per 18 months.

i. DELETED.

j. DELETED.

PRIMARY CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each primary 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 Primary Containment Leakage Rate Testing Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

a. With one primary 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 SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the primary containment air lock inoperable, except as a 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 SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each primary containment air lock shall be demonstrated OPERABLE:

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

b. By conducting an overall air lock leakage test in accordance with the Primary 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.**

**Except that the inner door need not be opened to verify interlock OPERABILITY when the primary containment is inerted, provided that the inner door interlock is tested within 8 hours after the primary containment has been de-inerted.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment shall be determined in accordance with the Primary Containment Leakage Rate Testing Program.

4.6.1.5.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.2 within 30 days. This report shall include a description of the condition of the containment, the inspection procedure, and the corrective actions taken.

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.8 The drywell and suppression chamber purge system, including the 6-inch nitrogen supply line, may be in operation for up to 500 hours each 365 days with the supply and exhaust isolation valves in one supply line and one exhaust line open for containment prepurge cleanup, inerting, deinerting, or pressure control.*

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With a drywell or suppression chamber purge supply and/or exhaust isolation valve and/or the nitrogen supply valve open, except as permitted above, close the valves(s) or otherwise isolate the penetration(s) within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With a drywell purge supply or exhaust isolation valve, or a suppression chamber purge supply or exhaust isolation valve or the nitrogen supply valve, having a measured leakage rate exceeding the limit of Surveillance Requirement 4.6.1.8.2, restore the inoperable valve(s) to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.8.1 Before being opened, the drywell and suppression chamber purge supply and exhaust, and nitrogen supply butterfly isolation valves shall be verified not to have been open for more than 500 hours in the previous 365 days.*

4.6.1.8.2 At least once per 24 months, the 26-inch drywell purge supply and exhaust isolation valves and the 24-inch suppression chamber purge supply and exhaust isolation valves and the 6-inch nitrogen supply valve shall be demonstrated OPERABLE in accordance with the Primary Containment Leakage Rate Testing Program.

* Valves open for pressure control are not subject to the 500 hours per 365 days limit, provided the 2-inch bypass lines are being utilized.

3/4.6 CONTAINMENT SYSTEMS

BASES

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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 design basis LOCA maximum peak containment accident pressure of 48.1 psig, 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 during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation 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 Primary containment Leakage Rate Testing Program.

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 Testing Program. Only one closed door in each air lock is required to maintain the integrity of the containment.

3/4.6.1.4 MSIV SEALING SYSTEM

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

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 48.1 psig in the event of a LOCA. A visual inspection in accordance with the Primary Containment Leakage Rate Testing Program is sufficient.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the containment peak pressure of 48.1 psig does not exceed the design pressure of 62 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 3 psid. The limit of -0.5 to +1.5 psig for initial positive containment pressure will limit the total pressure to 48.1 psig which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the safety analysis. The 135°F average temperature is conducive to normal and long term operation.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The 500 hours/365 days limit for the operation of the purge valves and the 6" nitrogen supply valve during plant Operational Conditions 1, 2 and 3 is intended to reduce the probability of a LOCA occurrence during the above operational conditions when the applicable combination of the above valves are open.

Blow-out panels are installed in the CPCS ductwork to provide additional assurance that the FRVs will be capable of performing its safety function subsequent to a LOCA.

6.8.4.f <u>Primary Containment Leakage Rate Testing Program</u>

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 48.1 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.5% 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 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) Door seal leakage rate less than or equal to 5 scf per hour when the gap between the door seals is pressurized to greater than or equal to 10.0 psig.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. NPF-57

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated April 1, 1997, as supplemented by letter dated May 30, 1997, the Public Service Electric & Gas Company (PSE&G, the licensee) submitted a request for changes to the Hope Creek Generating Station, Technical Specification (TSs) and the license. The requested changes would provide changes to Technical Specifications (TSs) 4.6.1.1, "Primary Containment Integrity;" 3/4.6.1.2, "Primary Containment Leakage;" 3/4.6.1.3, "Primary Containment Air Locks;" 4.6.1.5.1, "Primary Containment Structural Integrity;" and 4.6.1.8.2, "Drywell and Suppression Chamber Purge System." This amendment would also change the Bases for 3/4.6.1.2, "Primary Containment Leakage;" 3/4.6.1.3, "Primary Containment Air Locks;" 3.4.6.1.5, "Primary Containment Structural Integrity;" Section 6, "Administrative Controls;" and License Condition 2.D of Facility Operating License NPF-57. A new TS, 6.8.4.f, "Primary Containment Leakage Rate Testing Program," would be added. These changes modify the TSs and the Facility Operating License to adopt the performance based containment leak rate testing requirements (Option B) of 10 CFR Part 50, Appendix J and reference 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.

2.0 BACKGROUND

Compliance with 10 CFR Part 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 TS and 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 Part 50 was considered for this initiative and the 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 performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 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 another implementation document used by a licensee to develop a performance-based leakage testing program must be included, by general reference, in the plant TSs. The licensee has referenced RG 1.163 in the proposed Hope Creek 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 on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TSs to implement Option B. After some discussion, the staff and NEI agreed on final TSs, which were attached to a letter to D. Modien, NEI, from C. Grimes, NRC, 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 TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee 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 adequate. These records are subject to NRC inspection.

3.0 EVALUATION

The licensee's April 1 and May 30, 1997, letters to the NRC propose to establish a "Primary Containment Leakage Rate Testing Program" and propose to add this program to the TSs. The program references RG 1.163, which specifies methods acceptable to the NRC for complying with Option B. This requires changes to License Condition 2.D and to existing TS 4.6.1.1, 3/4.6.1.2, 3/4.6.1.3, 4.6.1.5.1, and 4.6.1.8.2, and the addition of the "Primary Containment Leakage Rate Testing Program" as TS 6.8.4.f. Section e, "Diesel Fuel Oil Testing Program," was added by Amendment No. 100, dated July 24, 1997; therefore, this new section is now Section f. Corresponding bases were also modified.

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. The licensee has elected to perform Type A, B, and C testing on a performance basis.

The changes to License Condition 2.D are made to expunge any references to specific paragraphs within Option A of Appendix J. The staff finds that these references are no longer valid since the licensee will be complying with Option B. Also deleted is the exemption from the requirement of Paragraph III.D.2(b)(ii) of Appendix J, exempting overall containment air lock leakage testing unless maintenance has been performed on the air lock that could affect air lock sealing capability. The staff finds that this exemption would no longer be necessary with the adoption of Option B, since Option B does not have the requirement, put forth in Paragraph III.D.2(b)(ii) of Appendix J, for testing of air locks opened during periods when containment integrity is not required.

The changes to existing TS 4.6.1.1, 3/4.6.1.2, 3/4.6.1.3, 4.6.1.5.1, and 4.6.1.8.2, delete the prescriptive test requirements, criteria, and surveillance intervals taken from Option A of Appendix J, and reference the Primary Containment Leakage Rate Testing Program, in accordance with Option B, and are therefore acceptable.

The new TS 6.8.4.f adds by reference the Primary Containment Leakage Rate Testing Program, and gives the peak calculated containment internal pressure for the design basis loss of coolant accident, the maximum allowable primary containment leakage rate, and acceptance criteria for leakage rate testing and air lock testing. TS 6.8.4 also states that the provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program, therefore making the deletion of existing TS 4.6.1.2.j acceptable. The new TS 6.8.4.f is in compliance with the requirements of Option B and consistent with the guidance of RG 1.163, and is therefore acceptable.

The TS changes proposed by the licensee are in compliance with the requirements of Option B and consistent with the guidance of RG 1.163. Further, despite the different format of the licensee's current TSs, all of the elements of the model TS guidance provided in the NRC letter to NEI dated November 2, 1995, are included in the proposed TSs. The staff therefore concludes that the proposed TS changes are acceptable.

The changes to TS Bases 3/4.6.1.2, 3/4.6.1.3, and 3/4.6.1.5 replace references to the requirements of Appendix J, Option A, with references to the Primary Containment Leakage Rate Testing Program. There are also clarifications made to Bases 3/4.6.1.2. These changes are acceptable with regard to the adoption of Option B.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State Official was notified of the proposed issuance of the amendment. By letter dated July 31, 1997, the State official indicated that they had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (62 FR 43375). The amendment also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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