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

June 9, 1987

Docket No. 50-354

Mr. Corbin A. McNeill, Jr.
Senior Vice President - Nuclear
Public Service Electric & Gas Company
P.O. Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. McNeill:

SUBJECT: TECHNICAL SPECIFICATION CHANGE REGARDING LEAK RATE TESTS AND GRANTING OF RELIEF FROM ASME CODE SECTION XI (TAC NO. 65066)

Re: HOPE CREEK GENERATING STATION

The Commission has issued the enclosed Amendment No. 4 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station. It grants relief from an ASME Boiler and Pressure Vessel Code, Section XI, valve leakage test requirement for the Hope Creek Generating Station. The amendment and the code relief are in response to your letter, NLR-N87047, dated April 3, 1987, as supplemented May 8, 1987. The amendment also extends the current Technical Specification surveillance intervals for 27 reactor coolant system pressure isolation valves and primary containment isolation valves, on a one-time-only basis, from once every 18 months and once every 24 months, until the first refueling outage (currently scheduled to begin on February 1, 1988). The code relief allows leak tests of the 27 valves, required by the code to be performed no less than once every two years, to be deferred until the first refueling outage.

We have determined pursuant to 10 CFR Part 50.55a(g)(6)(i), that the granting of this relief is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. In making this determination, we have given due consideration to the burden that could result if these requirements were imposed on the facility.

The approval of the amendment also requires a one-time exemption from certain Type C local leakage rate test requirements of 10 CFR Part 50, Appendix J. In response to your letter, NLR-N87055, dated April 3, 1987, such an exemption is being issued separately.

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

Sincerely,

/s/

George Rivenbark, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 4 to
   License No. NPF-57
2. Safety Evaluation

cc w/enclosures:
See next page

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PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	GRivenbark:ca	E. Chan	WButler
5/27/87	5/27/87	6/4/87	6/8/87

6/5/87

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

Sincerely,

George W Rivenbark

George Rivenbark, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 4 to
   License No. NPF-57
2. Safety Evaluation

cc w/enclosures:
See next page

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Hope Creek Generating Station

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 4
License No. NPF-57

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 (PSE&G) dated April 3, 1987, as supplemented May 8, 1987, 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. 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. 4 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PSE&G 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.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 9, 1987

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	GRivenbark	E. CHAN	WButler
5/27/87	5/27/87	6/4/87	6/8/87

6/4/87

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 9, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 4

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

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. Overleaf pages are provided to maintain document completeness.

Remove	Insert
3/4 4-11	3/4 4-11 (overleaf)
3/4 4-12	3/4 4-12
3/4 6-3	3/4 6-3 (overleaf)
3/4 6-4	3/4 6-4

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm IDENTIFIED LEAKAGE averaged over any 24-hour period.

d. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1, at rated pressure.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one other closed manual or deactivated automatic or check* valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-2 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*Which have been verified not to exceed the allowable leakage limit at the last refueling outage or the after last time the valve was disturbed, whichever is more recent.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the drywell atmospheric gaseous radioactivity at least once per 12 hours (not a means of quantifying leakage),

b. Monitoring the drywell floor and equipment drain sump flow rate at least once per 12 hours, and

c. Monitoring the drywell air coolers condensate flow rate at least once per 12 hours, and

d. Monitoring the drywell pressure at least once per 12 hours (not a means of quantifying leakage), and

e. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours (not a means of quantifying leakage), and

f. Monitoring the drywell temperature at least once per 24 hours (not a means of quantifying leakage).

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months,** and

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with alarm setpoints per Table 3.4.3.2-2 by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

b. CHANNEL CALIBRATION at least once per 18 months.

**P.I.V. leak test extension to the first refueling outage is permissible for each RCS P.I.V. listed in Table 3.4.3.2-1, that is identified in Public Service Electric & Gas Company's letter to the NRC (letter No. NLR-N87047), dated April 3, 1987, as needing a plant outage to test. For this one time test interval, the requirements of Section 4.0.2 are not applicable.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

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 less than or equal to 0.60 L_a, 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 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 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, 48.1 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet 0.75 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 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 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 between 0.75 L_a and 1.25 L_a.

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

SURVEILLANCE REQUIREMENTS (Continued)

The formula to be used is: $[L_o + L_{am} - 0.25\ L_a] \leq L_c \leq [L_o + L_{am} + 0.25\ L_a]$ where $L_c \equiv$ supplement test result; $L_o \equiv$ superimposed leakage; and $L_a \equiv$ measured Type A leakage.

d. Type B and C tests shall be conducted with gas at P_a, 48.1 psig*, at intervals no greater than 24 months** except for tests involving:

1. Air locks,
2. Main steam line isolation valves,
3. Valves pressurized with fluid from a seal system,
4. All containment isolation valves in hydrostatically tested lines in Table 3.6.3-1 which penetrate the primary containment, and
5. Purge supply and exhaust isolation valves with resilient material seals.

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

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

g. Containment isolation valves which form the boundary 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. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirements 4.6.1.8.2 and 4.6.1.8.3.

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

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

**A Type C test interval extension to the first refueling outage is permissible for primary containment isolation valves listed in Table 3.6.3-1, which are identified in Public Service Electric & Gas Company's letter to the NRC (letter No. NLR-N87047), dated April 3, 1987, as needing a plant outage to test. For this one time test interval, the requirements of Section 4.0.2 are not applicable.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 4 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 3, 1987, Public Service Electric & Gas Company proposed changes to the Technical Specifications (TSs) that would on a one-time-only bases, defer the 18 and 24 month valve leakage test requirements for the 27 valves listed below in Table 1 until the first refueling outage, currently scheduled to begin in February 1988. The licensee also requested relief from an ASME Boiler and Pressure Vessel Code, Section XI requirement that these 27 valves (Code Category A valves) be leak tested no less than once every two years. This request became necessary as a result of delays in attaining full power operation. Under existing requirements, the tests will become overdue beginning June 11, 1987. The licensee stated that the temporary extension of the surveillance intervals is needed to avoid a forced shutdown when the distribution system's need for power is high (June-July 1987). In a supplemental letter, dated May 8, 1987, the licensee provided additional information concerning its forced and planned outage plans. A request for exemption from the Appendix J requirements was also submitted, but under separate cover. The related exemption request is the subject of a separate Safety Evaluation.

TABLE 1

VALVES DESCRIPTION

	System	Valve Number	Size Type	Overdue Date
1.-	"B" Shutdown Cooling Return	BC-V014(HV-F050B)	12" Stop Check	6/23/87
3.	To "B" Recirc Loop, (3	BC-V013(HV-F015B)	12" Globe	6/23/87
	valves)	BC-V118(FV-F122B)	2" Globe	6/23/87

TABLE 1 Cont'd

VALVES DESCRIPTION

	System	Valve Number	Size Type	Overdue Date
4.	Torus Spray Supply	BC-V015(HV-F027B)	6" Gate	8/12/87
5.-	"A" LPCI Injection, (3	BC-V113(HV-F017A)	12" Gate	7/16/87
7.	valves)	BC-V114(HV-F041A)	12" Stop Check	7/16/87
		BC-V117(HV-F146A)	2" Globe	7/16/87
8.-	"A" Shutdown Cooling Return	BC-V110(HV-F015A)	12" Globe	7/25/87
10.	to "A" Recirc, Loop,	BC-V111(HV-F050A)	12" Stop Check	7/25/87
	(3 valves)	BC-V117(HV-F122A)	2" Globe	7/25/87
11.	RHR "A" Loop Containment Spray Supply	BC-V116(HV-F021A)	16" Gate	6/11/87
12.-	"B" Core Spray Injection	BE-V003(HV-F005B)	12" Gate	8/6/87
14.	(3 valves)	BE-V002(HV-F006B)	12" Stop Check	8/6/87
		BE-V072(HV-F039B)	2" Globe	8/6/87
15.-	"A" Core Spray Injection	BE-V007(HV-F005A)	12" Gate	8/5/87
18.	(4 valves)	BE-V006(HV-F006A)	12" Stop Check	8/5/87
		BE-V07a(HV-F039A)	12" Globe	8/5/87
		BE-V001(HV-F006)	14" Gate	8/5/87
19.	"B" Instrument Gas Header	KL-V026(HV-51528)	2" Globe	8/7/87

TABLE 1 Cont'd

VALVES DESCRIPTION

System	Valve Number	Size Type	Overdue Date
20.- Head Spray (2 valves)	BC-V021(HV-F023)	6" Gate	9/3/87
21.	BC-V020(HV-F022)	6" Globe	9/3/87
22.- HPCI Steam Supply	FD-V001(HV-F002)	12" Gate	9/6/87
24. (3 valves)	FD-V051(HV-F100)	2" Globe	9/6/87
	FD-V002(HV-F003)	12" Gate	9/6/87
25.- RCIC Steam Supply	FC-V001(HV-F007)	4" Gate	9/18/87
27. (3 valves)	FC-V048(HV-F076)	2" Globe	9/18/87
	FC-V002(HV-F008)	4" Gate	9/18/87

2.0 EVALUATION

In order to meet the requirements of Technical Specifications Sections 4.0.2, 4.4.3.2.2.a, and 4.6.1.2.d and Section XI of the ASME Code on which the Inservice Testing Program is based, a plant shutdown would be necessary prior to the first refueling outage. A shutdown is necessary because a containment entry would be required to test the 27 valves affected by this proposal. Entry into containment during power operations would expose personnel to the hazards of high air temperature, radiation exposure that is high relative to as-low-as reasonably-achievable (ALARA) standards, and the nitrogen environment of the inerted containment atmosphere for which a self-contained breathing apparatus is required. The licensee stated that additional factors which preclude testing these valves at power include the need to drain the "A" and "B" RHR loops and the "A" core spray loop.

The surveillance requirements of TS 4.4.3.2.2.a and TS 4.6.1.2.d require leak tests to be performed nominally every 18 or 24 months. Additionally, Section XI of the Boiler and Pressure Vessel Code calls for the leak testing of Category A valves not less than once every two years. The TS 4.0.2 allows the 18-month and 24-month test intervals to be extended by 25 percent to allow flexibility in operations scheduling. The overdue dates in Table 1 correspond to the 24-month limit and the 18-month limit extended by the allowable 25 percent. For the valves in Table 1 which are classified as Pressure Isolation Valves (PIVs), the Hope Creek Safety Evaluation Report, Supplement No. 5, indicates that Type C (gas) testing in lieu of water testing will satisfy the TS 4.4.3.2.2.a test requirements. Consequently, the PIV testing is also used to meet the applicable containment isolation valve (Type C) test requirement of TS 4.6.1.2.d.

The Type C tests have already been performed by the licensee for those valves that did not require plant shutdown. If an unplanned outage of greater than 30 days is encountered prior to the first refueling outage, the licensee has committed to perform the testing of the valves that require plant shutdown with the exception of the valves (BC-V020 and BC-V021), which can only be tested during a refueling outage.

The justification provided by the licensee for this amendment request includes analysis of the previous test result, the operating experience, and the consequence of any degradation during the requested extension. In the review of the information provided by the utility, the staff made the following findings:

1. All of the containment isolation valves listed in Table 1 were tested successfully in late 1985. The Type C test leakage for the valves constituted only 4.9% of the allowable limit. Moreover, these valves do not have a significant amount of operating time on them and have exhibited favorable operating experience. The degradation of the valves is therefore expected to be negligible.

2. There is ample margin between the leakage previously measured during the Type C tests and the limiting leakage values in the Technical Specifications and in Appendix J to accommodate any degradation likely to be experienced by these 27 valves during the extension period. Therefore, the consequence of leakage past these isolation valves is bounded by safety analyses previously performed which were based on the limiting leakage values in the Technical Specifications and in Appendix J.

   NUREG/CR-4330, Review of Light Water Reactor Regulatory Requirements, has shown that containment leakage is a relatively minor contributor to overall risk. Its review of the BWRs in the study indicates that an increase in the containment leakage rate by an order of magnitude would only have marginal affect on the public risk (i.e., on the order of few percent or less of overall risk).

The staff has determined that the deferred testing has little safety significance and that the proposed amendment will not alter any of the accident analyses.

Based on the considerations discussed above, the staff concludes that the one-time extension of the testing interval for a period not to exceed 34 weeks (based on February 1, 1988 refueling outage commencement) is acceptable. If an outage of greater than 30 days is encountered prior to the first refueling outage at Hope Creek, the licensee will be required to perform the testing, which is the subject of this amendment request, except for those tests that can only be performed during a refeuling outage. The staff finds that this one-time extension will have no adverse affect on the health and safety of the public. Imposing these requirements would result in hardship or unusual difficulties without a compensating increase in the level of quality or safety.

Accordingly, we have also conlcuded that 1) the proposed change to the Technical Specification to permit this one-time extension is acceptable and 2) the proposed temporary and one-time relief until the first refueling outage from the ASME Boiler and Pressure Vessel Code, Section XI requirement to leak test Category A valves not less than once every two years should be granted.

3.0 ENVIRONMENTAL CONSIDERATION

The amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The 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. Accordingly, this amendment meets 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 this amendment.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (52 FR 16954) on May 6, 1987, and consulted with the State of New Jersey. No public comments were received and the State of New Jersey did not have any comments.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: K. Dempsey

Dated: June 9, 1987