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.16
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   A. The application for amendment filed by the Public Service Electric & Gas Company (PSE&G) dated June 4, 1986, as superseded and supplemented by letters dated November 21 and December 18, 1986 and February 20, March 19, May 15 and July 13, 1987, complies with the standards and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.16 , 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: March 30, 1988

LA:PDI-2:DRPI/II	PM:PDI-2:DRPI/II	OGC	D:PDI-2:DRPI/II
MO'Brien	GRivenbark	E. Chan	WButler
7/21/87	7/21/87	7/29/87	3/30/87

7/29/87

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
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Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 30, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 16

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 page(s) provided to maintain document completeness.*

Remove	Insert
3/4 6-11	3/4 6-11
3/4 6-12*	3/4 6-12*
3/4 6-23*	3/4 6-23*
3/4 6-24	3/4 6-24
3/4 6-25	3/4 6-25
3/4 6-26*	3/4 6-26*
B 3/4 6-1*	B 3/4 6-1*
B 3/4 6-2	B 3/4 6-2
B 3/4 6-3	B 3/4 6-3
B 3/4 6-4*	B 3/4 6-4*

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 120 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 valve(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, with resilient material seals 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 120 hours in the previous 365 days.*

4.6.1.8.2 At least once per 6 months**, but no more than once per 92 days***, 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 with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a per penetration when pressurized to P_a 48.1 psig.

*Valves open for pressure control are not subject to the 120 hours per 365 days limit, provided the 2-inch bypass lines are being utilized.

**Provided that the valve has not been operated since the previous test.

***Applies only to a valve which has been operated since the previous test.

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between 118,000 ft^3 and 122,000 ft^3, equivalent to an indicated level between 74.5" and 78.5" and a

2. Maximum average temperature of 95°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

a) 105°F during testing which adds heat to the suppression chamber.

b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

3. Maximum average temperature of 95°F during OPERATIONAL CONDITION 3, except that the maximum average temperature may be permitted to increase to 120°F with the main steam line isolation valves closed following a scram.

b. A total leakage between the suppression chamber and drywell of less than the equivalent leakage through a 1-inch diameter orifice at a differential pressure of 0.80 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the suppression chamber average water temperature greater than 95°F and THERMAL POWER greater than 1% of RATED THERMAL POWER, restore the average temperature to less than or equal to 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than 105°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the average temperature to less than 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
8. Group 8 - Torus Water Cleanup (TWC) System				
(a) TWC Suction Isolation Valves				M-53-1
Outside:				
HV-4680 (EE-V003)	P223	45	4	
HV-4681 (EE-V004)	P223	45	4	
(b) TWC Return Isolation Valves				M-53-1
Outside:				
HV-4652 (EE-V002)	P222	45	4	
HV-4679 (EE-V001)	P222	45	4	
9. Group 9 - Drywell Sumps				
(a) Drywell Floor Drain Sump Discharge Isolation Valves				M-61-1
Inside: HV-F003 (HB-V005)	P25	30	3	
Outside: HV-F004 (HB-V006)	P25	30	3	
(b) Drywell Equipment Drain Sump Discharge Isolation Valves				M-61-1
Inside: HV-F019 (HB-V045)	P26	30	3	
Outside: HV-F020 (HB-V046)	P26	30	3	
10. Group 10 - Drywell Coolers				
(a) Chilled Water to Drywell Coolers Isolation Valves				M-87-1
Inside:				
Loop A: HV-9531B1 (GB-V081)	P8B	60	3	
Loop B: HV-9531B3 (GB-V083)	P38A	60	3	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
Outside:				
Loop A: HV-9531A1 (GB-V048)	P8B	60	3	
Loop B: HV-9531A3 (GB-V070)	P38A	60	3	
(b) Chilled Water from Drywell Coolers Isolation Valves				M-87-1
Inside:				
Loop A: HV-9531B2 (GB-V082)	P8A	60	3	
Loop B: HV-9531B4 (GB-V084)	P38B	60	3	
Outside:				
Loop A: HV-9531A2 (GB-V046)	P8A	60	3	
Loop B: HV-9531A4 (GB-V071)	P38B	60	3	
11. Group 11 - Recirculation Pump System				
(a) Recirculation Pump Seal Water Isolation Valves				M-43-1
Outside:				
Loop A: HV-3800A (BF-V098)	P19	45	3	
Loop B: HV-3800B (BF-V099)	P20	45	3	
12. Group 12 - Containment Atmosphere Control System				
(a) Drywell Purge Supply Isolation Valves				M-57-1
Outside:				
HV-4956 (GS-V009)	P22	5	3, 8	
HV-4979 (GS-V021)	P22/220	5	3, 8	
(b) Drywell Purge Exhaust Isolation Valves				M-57-1
Outside:				
HV-4951 (GS-V025)	P23	15	3	
HV-4950 (GS-V026)	P23	5	3, 8	
HV-4952 (GS-V024)	P23	5	3, 8	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
(c) Suppression Chamber Purge Supply Isolation Valves				M-57-1
Outside:				
HV-4980 (GS-V020)	P22/P220	5	3, 8	
HV-4958 (GS-V022)	P220	5	3, 8	
(d) Suppression Chamber Purge Exhaust Isolation Valves				M-57-1
Outside:				
HV-4963 (GS-V076)	P219	15	3	
HV-4962 (GS-V027)	P219	5	3, 8	
HV-4964 (GS-V028)	P219	5	3, 8	
(e) Nitrogen Purge Isolation Valves				M-57-1
Outside:				
HV-4974 (GS-V053)	J7D/J202	45	3	
HV-4978 (GS-V023)	P22/P220	5	3, 8	
13. Group 13 - Hydrogen/Oxygen (H2/O2) Analyzer System				
(a) Drywell H2/O2 Analyzer Inlet Isolation Valves				M-57-1
Outside:				
Loop A: HV-4955A (GS-V045)	J9E	45	3	
HV-4983A (GS-V046)	J9E	45	3	
HV-4984A (GS-V048)	J10C	45	3	
HV-5019A (GS-V047)	J10C	45	3	
Outside:				
Loop B: HV-4955B (GS-V031)	J3B	45	3	
HV-4983B (GS-V032)	J3B	45	3	
HV-4984B (GS-V034)	J7D/J202	45	3	
HV-5019B (GS-V033)	J7D	45	3	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
(b) Suppression Chamber H2/O2 Analyzer Inlet Isolation Valves				M-57-1
Outside:				
Loop A: HV-4965A (GS-V050)	J212	45	3	
HV-4959A (GS-V049)	J212	45	3	
Outside:				
Loop B: HV-4965B (GS-V041)	J210	45	3	
HV-4959B (GS-V040)	J210	45	3	
(c) H2/O2 Analyzer Return to Suppression Chamber Isolation Valves				M-57-1
Outside:				
Loop A: HV-4966A (GS-V051)	J201	45	3	
HV-5022A (GS-V052)	J201	45	3	
Outside:				
Loop B: HV-4966B (GS-V042)	J202	45	3	
HV-5022B (GS-V043)	J202/J7D	45	3	
14. Group 14 - Containment Hydrogen Recombination (CHR) System				
(a) CHR Supply Isolation Valves				M-58-1
Outside:				
Loop A: HV-5050A (GS-V002)	P23	45	3	
HV-5052A (GS-V003)	P23	45	3	
Outside:				
Loop B: HV-5050B (GS-V004)	P22	45	3	
HV-5052B (GS-V005)	P22	45	3	
(b) CHR Return Isolation Valves				M-58-1
Outside:				
Loop A: HV-5053A (GS-V008)	P220	45	3	
HV-5054A (GS-V010)	P220	45	3	

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

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

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 48.1 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with 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 requirements of Appendix "J" of 10 CFR Part 50 with the exception of exemptions granted for main steam isolation valve leak testing and testing the airlocks after each opening.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

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

3/4.6.1.4 MSIV 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 conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression 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 120 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.

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM (Continued)

The use of the drywell and suppression chamber purge exhaust lines for pressure control during plant Operational Conditions 1, 2 and 3 is unrestricted provided 1) only the inboard purge exhaust isolation valves on these lines and the vent valves on the 2-inch vent paths are used and 2) the outboard purge exhaust isolation valves remain closed. This is because in such a situation, the vent valves will sufficiently choke the flow and additionally the applicable valves will close in a timely manner during a LOCA or steam line break accident and therefore the control room and the site boundary dose guidelines of applicable 10 CFR dose limits will not be exceeded in the event of an accident. The design of the purge supply and exhaust isolation valves and the 6-inch nitrogen supply valve meets the requirements of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations".

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

3/4.6.2. DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 62 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1020 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid must not exceed 62 psig, the suppression chamber maximum internal design pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant to be considered is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in this specification, containment pressure during the design basis accident is approximately 48.1 psig which is below the design pressure of 62 psig. Maximum water volume of 122,000 ft^3 results in a downcomer submergence of 3.33 ft and the minimum volume of 118,000 ft^3 results in a submergence of approximately 3.0 ft. The majority of the Bodega tests were run with a submerged length of four feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown

DEPRESSURIZATION SYSTEMS (Continued)

tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a water temperature of approximately 135°F immediately following blowdown which is below the 200°F used for complete condensation via mitered T-quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak local temperature of the suppression pool is maintained below 200°F during any period of relief valve operation. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve to assure mixing and uniformity of energy insertion to the pool.

In conjuction with the Mark I containment Long Term Program, a plant unique analysis was performed which demonstrated that the containment, the attached piping and internal structures meet the applicable structural and mechanical acceptance criteria for Hope Creek. The evaluation followed the design basis loads defined in the Mark I Load Definition Report, NEDO-21888, December 1978, as modified by NRC SER NUREG 0661, July 1980 and Supplement 1, August 1982, to ensure that hydrodynamic loads, appropriate for the life of the plant, were applied.