Docket No. 50-354

April 18, 1994

Mr. Steven E. Miltenberger
Vice President and Chief Nuclear
Officer
Public Service Electric & Gas
Company
Post Office Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. Miltenberger:

SUBJECT: INSERVICE LEAK AND HYDROSTATIC TESTING, HOPE CREEK GENERATING STATION (TAC NO. 88941)

The Commission has issued the enclosed Amendment No. 69 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 4, 1994.

This amendment adds a new TS 3/4.10.8, "Inservice Leak and Hydrostatic Testing," to the Hope Creek Generating Station TSs. The amendment also includes corresponding changes to the TS Index, Table 1.2, "OPERATIONAL CONDITIONS," and provides Bases for TS 3/4.10.8. The added TS 3/4.10.8 permits the unit to remain in OPERATIONAL CONDITION 4 with the average reactor coolant temperature being increased above 200°F, but not to exceed 212°F, and certain OPERATIONAL CONDITION 3 Limiting Conditions for Operation for secondary containment isolation, secondary containment integrity and filtration, recirculation and ventilation system (FRVS) operability being met.

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

You are requested to inform the NRC, in writing, when this amendment has been implemented.

Sincerely,
/S/
James C. Stone, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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P PDR

Enclosures:
1. Amendment No. 69 to License No. NPF-57
2. Safety Evaluation

cc w/enclosures:
See next page

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| DATE | :4/11/94 | :4/11/94 | :4/11/94 | :03/23/94 | :04/04/94 | :4/15/94 |

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

April 18, 1994

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1. Amendment No. 69 to
   License No. NPF-57
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Steven E. Miltenberger
Public Service Electric & Gas
Company

Hope Creek Generating Station

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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. 69
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 March 4, 1994, 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. 69 , 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 and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

Charles L. Miller, 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: April 18, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 69

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 provided to maintain document completeness.*

| Remove | Insert |
|---|---|
| xv | xv |
| xvi | xvi* |
| xxi | xxi |
| xxii | xxii* |
| 1-11 | 1-11 |
| - | - |
| 3/4 10-7 | 3/4 10-7* |
| - | 3/4 10-8 |
| B 3/4 5-1 | B 3/4 5-1* |
| B 3/4 5-2 | B 3/4 5-2 |
| - | B 3/4 5-3 |
| - | - |
| B 3/4 10-1 | B 3/4 10-1* |
| - | B 3/4 10-2 |

# INDEX

## LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

| SECTION | | PAGE |
|---|---|---|

INDEX

BASES

SECTION PAGE

TABLE 1.2

OPERATIONAL CONDITIONS

| CONDITION | MODE SWITCH POSITION | AVERAGE REACTOR COOLANT TEMPERATURE |
|---|---|---|
| 1. POWER OPERATION | Run | Any temperature |
| 2. STARTUP | Startup/Hot Standby | Any temperature |
| 3. HOT SHUTDOWN | Shutdown#,*** | > 200°F |
| 4. COLD SHUTDOWN | Shutdown#,##,*** | ≤ 200°F+ |
| 5. REFUELING* | Shutdown or Refuel**,# | ≤ 140°F |

---

#The reactor mode switch may be placed in the Run, Startup/Hot Standby, or Refuel position to test the switch interlock functions and related instrumentation provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff. If the reactor mode switch is placed in the Refuel position, the one-rod-out interlock shall be OPERABLE.

##The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.9.10.1.

*Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

**See Special Test Exceptions 3.10.1 and 3.10.3.

***The reactor mode switch may be placed in the Refuel position while a single control rod is being recoupled or withdrawn provided that the one-rod-out interlock is OPERABLE.

+See Special Test Exception 3.10.8.

## 3/4.10.7 SPECIAL INSTRUMENTATION - INITIAL CORE LOADING

### LIMITING CONDITION FOR OPERATION

3/4.10.7 The material originally contained in Section 3/4.10.7 was deleted with the issuance of Amendment No. 14. However, to maintain the historical reference to this section, Section 3/4.10.7 is intentionally left blank.

## 3/4.10.8 INSERVICE LEAK AND HYDROSTATIC TESTING

### LIMITING CONDITION FOR OPERATION

3.10.8 When conducting inservice leak or hydrostatic testing, the average reactor coolant temperature specified in Table 1.2 for OPERATIONAL CONDITION 4 may be increased to 212°F, and operation considered not to be in OPERATIONAL CONDITION 3, to allow performance of an inservice leak or hydrostatic test provided the following OPERATIONAL CONDITION 3 LCO's are met:

a. 3.3.2, "ISOLATION ACTUATION INSTRUMENTATION", Functions 2.a, 2.b, 2.c, 2.d and 2.e of Table 3.3.2-1;

b. 3.6.5.1, "SECONDARY CONTAINMENT INTEGRITY";

c. 3.6.5.2, "SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS"; and

d. 3.6.5.3, "FILTRATION, RECIRCULATION AND VENTILATION SYSTEM."

APPLICABILITY: OPERATIONAL CONDITION 4, with average reactor coolant temperature > 200°F.

ACTION:

With the requirements of the above specification not satisfied, immediately enter the applicable condition of the affected specification or immediately suspend activities that could increase the average reactor coolant temperature or pressure and reduce the average reactor coolant temperature to ≤ 200°F within 24 hours.

### SURVEILLANCE REQUIREMENTS

4.10.8 Verify applicable OPERATIONAL CONDITION 3 surveillances for specifications listed in 3.10.8 are met.

## 3/4.5 EMERGENCY CORE COOLING SYSTEM

## BASES

### 3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN

The core spray system (CSS), together with the LPCI mode of the RHR system, is provided to assure that the core is adequately cooled following a loss-of-coolant accident and provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the ADS.

The CSS is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining.

The surveillance requirements provide adequate assurance that the CSS will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Four subsystems, each with one pump, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

The surveillance requirements provide adequate assurance that the LPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The high pressure coolant injection (HPCI) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCI system continues to operate until reactor vessel pressure is below the pressure at which CSS operation or LPCI mode of the RHR system operation maintains core cooling.

The capacity of the system is selected to provide the required core cooling. The HPCI pump is designed to deliver greater than or equal to 5600 gpm at reactor pressures between 1120 and 200 psig. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor, but no credit is taken in the safety analyses for the condensate storage tank water.

ECCS-OPERATING and SHUTDOWN (Continued)

With the HPCI system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the CSS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the safety analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCI out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems and the RCIC system.

The surveillance requirements provide adequate assurance that the HPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor to be in HOT SHUTDOWN with vessel pressure not less than 200 psig. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCI system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 100 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls five selected safety-relief valves although the safety analysis only takes credit for four valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION CHAMBER

The suppression chamber is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCI, CSS and LPCI systems in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression chamber in OPERATIONAL CONDITIONS 1, 2 or 3 is also required by Specification 3.6.2.1.

Repair work might require making the suppression chamber inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression chamber must be made inoperable, including draining, in OPERATIONAL CONDITION 4 or 5.

===============================================================================

3/4.5.3 SUPPRESSION CHAMBER (Continued)

In OPERATIONAL CONDITION 4 and 5 the suppression chamber minimum required water volume is reduced because the reactor coolant is maintained at or below 200°F.+ Since pressure suppression is not required below 212°F, the minimum water volume is based on NPSH, recirculation volume and vortex prevention plus a safety margin for conservatism.

+ See Special Test Exception 3.10.8.

## 3/4.10 SPECIAL TEST EXCEPTIONS

## BASES

### 3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

The requirement for PRIMARY CONTAINMENT INTEGRITY is not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS.

### 3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

In order to perform the tests required in the technical specifications it is necessary to bypass the sequence restraints on control rod movement. The additional surveillance requirements ensure that the specifications on heat generation rates and shutdown margin requirements are not exceeded during the period when these tests are being performed and that individual rod worths do not exceed the values assumed in the safety analysis.

### 3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

Performance of shutdown margin demonstrations during open vessel testing requires additional restrictions in order to ensure that criticality is properly monitored and controlled. These additional restrictions are specified in this LCO.

### 3/4.10.4 RECIRCULATION LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain PHYSICS TESTS while at low THERMAL POWER levels.

### 3/4.10.5 OXYGEN CONCENTRATION

The material originally contained in this Technical Specification was deleted with the issuance of Amendment No. 35. However, to maintain the historical reference to this specification, this section has been intentionally left blank.

### 3/4.10.6 TRAINING STARTUPS

This special test exception permits training startups to be performed with the reactor vessel depressurized at low THERMAL POWER and temperature while controlling RCS temperature with one RHR subsystem aligned in the shutdown cooling mode in order to minimize contaminated water discharge to the radioactive waste disposal system.

### 3/4.10.7 SPECIAL INSTRUMENTATION - INITIAL CORE LOADING

The material originally contained in Bases Section 3/4.10.7 was deleted with the issuance of Amendment No. 14. However, to maintain the historical reference to this section, Bases Section 3/4.10.7 is intentionally left blank.

## 3/4.10.8 INSERVICE LEAK AND HYDROSTATIC TESTING

This special test exception allows reactor vessel inservice leak and hydrostatic testing to be performed in OPERATIONAL CONDITION 4 with reactor coolant temperatures ≤ 212°F. The additionally imposed OPERATIONAL CONDITION 3 requirement for SECONDARY CONTAINMENT operability provides conservatism in the response of the unit to an operational event. This allows flexibility since temperatures approach 200°F during the testing and can drift higher because of decay and mechanical heat.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 69 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 March 4, 1994, the Public Service Electric & Gas Company (the licensee) submitted a request for changes to the Hope Creek Generating Station, Technical Specification (TS). The requested changes would add a new TS 3/4.10.8, "Inservice Leak and Hydrostatic Testing." The proposed changes would also include corresponding changes to the TS Index, Table 1.2, "OPERATIONAL CONDITIONS," and provides Bases for TS 3/4.10.8. The proposed changes would permit Hope Creek to remain in OPERATIONAL CONDITION 4 with average reactor coolant temperature being increased above 200°F, but not to exceed 212°F, and OPERATIONAL CONDITION 3 Limiting Conditions for Operation (LCO) for secondary containment isolation, secondary containment integrity and filtration, recirculation and ventilation system (FRVS) operability being met.

## 2.0 EVALUATION

The Hope Creek TS define five OPERATIONAL CONDITIONS. OPERATIONAL CONDITION 4 requires the reactor mode switch to be in the shutdown position and the average reactor coolant temperature to be less than or equal to 200°F. OPERATIONAL CONDITION 3 also requires the reactor mode switch to be in the shutdown position but with the average reactor coolant temperature greater than 200°F.

The Hope Creek TS require that various TSs be applicable in one or more of the five OPERATIONAL CONDITIONS. Additional TSs become applicable when the plant enters OPERATIONAL CONDITION 3 from OPERATIONAL CONDITION 4. This change in OPERATIONAL CONDITIONS occurs when the average reactor coolant temperature is increased above 200°F. TSs of particular concern for entry into OPERATIONAL CONDITION 3 are TS 3.3.2, "Isolation Actuation Instrumentation," TS 3.6.5.1, "Secondary Containment Integrity," TS 3.6.5.2, "Secondary Containment Automatic Isolation," and TS 3.6.5.3, "Filtration, Recirculation, and Ventilation System." TS 3.3.2, "Emergency Core Coolant System (ECCS) - Operating," requires two core spray (CS) system loops, four low pressure coolant injection (LPCI) modes of the residual heat removal (RHR) system, the high pressure coolant injection system, and the automatic depressurization system to be OPERABLE in OPERATIONAL CONDITION 3. TS 3.5.2, "ECCS - Shutdown," requires one CS loop and one LPCI mode of RHR to be OPERABLE in

OPERATIONAL CONDITION 4 and 5, thereby permitting outage-related maintenance to be performed on the ECCS systems not required to be OPERABLE.

TS 3.6.1.1 requires PRIMARY CONTAINMENT INTEGRITY to be maintained in OPERATIONAL CONDITION 3 but PRIMARY CONTAINMENT INTEGRITY is not required in OPERATION CONDITION 4. The requirements of TS 3.6.1.1 significantly restrict unobstructed access within the primary containment during operations in OPERATIONAL CONDITION 3. The licensee desires to be able to perform certain outage activities on various systems while remaining consistent with OPERATIONAL CONDITION 4 applicable requirements that are in effect immediately prior to and immediately following inservice leak and hydrostatic testing.

The reactor coolant system (RCS) is isolated during leak or hydrostatic tests. This isolation makes RCS temperature control difficult since the RCS is isolated from its heat sinks and heat input to the RCS is caused by both decay heat and mechanical heat from the recirculation pumps. TS 3.4.6, "Reactor Coolant System Pressure/Temperature Limits," currently requires reactor pressure vessel temperatures approaching 200 °F when the RCS is pressurized for leak or hydrostatic testing. This minimum temperature for performing leak or hydrostatic tests will increase over time as fast neutron fluence to the reactor vessel increases with operating time. The leak or hydrostatic tests require several hours of completion; operating experience has shown that the RCS temperature slowly increases during these tests and dependent upon the amount of decay heat present, the RCS may approach the 200°F limit of OPERATIONAL CONDITION 4. Therefore, the licensee has proposed to increase the OPERATIONAL CONDITION 4 temperature limit to provide some additional margin within which to complete the leak or hydrostatic tests.

Permitting the average reactor coolant temperature to be increased above 200°F and limiting the maximum reactor coolant temperature to 212°F while performing leak or hydrostatic tests will not substantially affect the results of potential accidents which might occur with the increased average reactor coolant temperature since the leak and hydrostatic tests are performed with the RCS near water solid and with all control rods fully inserted. Therefore, the stored energy in the reactor core would be very low and the potential for causing fuel failures with a subsequent increase in coolant activity is minimal. The restrictions provided in the proposed new TS 10.8 would require secondary containment integrity as well as OPERABLE automatic isolation dampers, OPERABLE FRVS, and OPERABLE isolation actuation instrumentation for this equipment. Therefore, any leakage of radioactive materials from the RCS would be filtered by the FRVS prior to release to the atmosphere. Furthermore, since the reactor coolant temperature would be limited to a maximum of 212°F, there would be no flashing of coolant to steam and therefore, any releases of radioactive materials from the coolant would be minimized.

In the event of a large loss-of-coolant accident during a leak or hydrostatic test, the RCS would rapidly depressurize thereby permitting the low pressure

ECCS equipment, required by TS 3.5.2, to actuate and thereby keep the core flooded. This action would prevent the fuel from overheating and releasing radioactive materials. The RCS inspections required to be performed as part of the leak or hydrostatic tests would be expected to detect small leaks before a significant inventory coolant was lost.

Based on the foregoing analyses, the staff concludes that the proposed TS changes will ensure acceptable consequences of any postulated accidents, are enveloped by the previously accepted analyses, and are, therefore, acceptable.

## 3.0 STATE CONSULTATION

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

## 4.0 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 a surveillance requirement. 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 (59 FR 12384). Accordingly, the 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 the amendment.

## 5.0 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.

Principal Contributor: J. Zimmerman

Date: April 18, 1994