November 6, 1997

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. M99665)

Dear Mr. Eliason:

The Commission has issued the enclosed Amendment No. 109 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 September 24, 1997.

This amendment adds a Surveillance Requirement to Technical Specification 3/4.5.1, "Emergency Core Cooling Systems", to perform a monthly valve position verification for the four residual heat removal cross-tie valves.

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. 109 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

November 6, 1997

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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. 109
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 September 24, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. 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. 109, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Chester Poslusny for

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 6, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 109

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.

Remove	Insert
3/4 5-4	3/4 5-4
B 3/4 5-1	B 3/4 5-1
-	B 3/4 5-1a

SURVEILLANCE REQUIREMENTS

4.5.1 The emergency core cooling systems shall be demonstrated OPERABLE by:

a. At least once per 31 days:

1. For the core spray system, the LPCI system, and the HPCI system:

   a) Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

   b) Verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

   c) Verify the RHR System cross tie valves on the discharge side of the pumps are closed and power, if any, is removed from the valve operators.

2. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position.

b. Verifying that, when tested pursuant to Specification 4.0.5:

1. The two core spray system pumps in each subsystem together develop a flow of at least 6350 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥105 psi above suppression pool pressure.

2. Each LPCI pump in each subsystem develops a flow of at least 10,000 gpm against a test line pressure corresponding to a reactor vessel to primary containment differential pressure of ≥20 psid.

3. The HPCI pump develops a flow of at least 5600 gpm against a test line pressure corresponding to a reactor vessel pressure of 1000 psig when steam is being supplied to the turbine at 1000, +20, -80 psig.**

c. At least once per 18 months:

1. For the core spray system, the LPCI system, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

*Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

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.

Verification every 31 days that each RHR System cross tie valve on the discharge side of the RHR pumps is closed and power to its operator, if any, is disconnected ensures that each LPCI subsystem remains independent and a failure in the flow path in one subsystem will not affect the flow path of the other LPCI subsystem. Acceptable methods of removing power to the operator include de-energizing breaker control power or racking out or removing the breaker. For the valves in high radiation areas, verification may consist of verifying that no work activity was performed in the area of the valve since the last verification was performed. If one of the RHR System cross tie valves is open or power has not been removed from the valve operator, both associated LPCI subsystems must be considered inoperable. The 31 day frequency is acceptable, considering that these valves are under strict administrative controls that will ensure that the valves continue to remain closed with either control or motive power removed.

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.

3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN (Continued)

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.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 109 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 September 24, 1997, the Public Service Electric & Gas Company (the licensee) submitted a request for a change to the Hope Creek Generating Station (HCGS) Technical Specifications (TSs). The requested change would add a surveillance requirement (SR) to TS 3/4.5.1, "Emergency Core Cooling Systems", to perform a monthly valve position verification for the four residual heat removal (RHR) cross-tie valves.

2.0 DISCUSSION

The RHR System functions during normal reactor shutdown conditions to remove heat from the reactor. During reactor operation, the RHR System is in standby and functions as part of the emergency core cooling system (ECCS) in the event of a loss-of-coolant accident. The RHR System is described in the HCGS Updated Final Safety Analysis Report (UFSAR), Section 5.4.7, "Residual Heat [Removal] System," as follows:

> The Residual Heat Removal (RHR) System consists of four independent loops A, B, C and D as shown in Figure 5.4-13. Each loop contains a motor driven pump, piping, valves, instrumentation, and controls. Each loop takes suction from the suppression pool and is capable of discharging water to the reactor vessel via separate low pressure coolant injection (LPCI) nozzles, or back to the suppression pool via a full flow test line. Loops A and B have heat exchangers that are each cooled by an independent loop of the Safety Auxiliaries Cooling System (SACS). In addition, the loops A and C pump discharge headers and the loops B and D pump discharge headers are each cross-tied via two manual isolation valves. The purpose of these cross-ties is to permit the use of C pump with RHR heat exchanger A and the use of D pump with RHR heat exchanger B for alternate decay heat removal.

The cross-ties referenced above were added during the fifth and sixth refueling outages and are the subject of the TS change proposed by the licensee in the September 24, 1997 application. The proposed TS change would be added to the monthly surveillances of TS 4.5.1.a.1 as follows:

c) Verify the RHR System cross tie valves on the discharge side of the pumps are closed and power, if any, is removed from the valve operators

The SR is necessary to assure that the separation of the RHR loops is not compromised such that the ECCS function of RHR might become degraded. The specification of valves "...on the discharge side of the pumps..." is made to distinguish these cross ties from those located on the suction side of the pumps, which were a part of the initial RHR design. In addition, while the subject cross-tie valves are manual, the requirement to verify that "...power, if any, is removed..." refers to the licensee's option to modify the cross-ties via installation of motor-operated valves.

3.0 EVALUATION

The subject RHR cross-ties are major flow paths consisting of 18-inch piping. As such, the mispositioning of the cross-tie valves could result in significant changes in RHR flow distribution. During an ECCS injection, this flow distribution change could lead to significant degradation of the ECCS function. The design of the RHR cross-ties includes two manual isolation valves per cross-tie which decreases the likelihood that the degradation of ECCS would occur via the cross-tie line(s) since both valves in a cross-tie line would have to be mispositioned. The addition of the proposed TS surveillance is justified to further reduce the likelihood of degraded ECCS, considering the potentially, significant nature of such degradation. The proposed TS surveillance is consistent with the NRC staff guidance in NUREG-1433, Rev. 1, "Standard Technical Specifications, General Electric Plants, BWR/4", April 1995. Based upon the above, the proposed change to TS 4.5.1.a.1 is acceptable. The change to the TS also includes a change to the Bases.

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. The State Official 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 52162). 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.

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.

Principal Contributor: D. H. Jaffe

Date: November 6, 1997