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.68
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 23, 1993, and supplemented by letters dated November 10, 1993 and January 13, 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. 68 , 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 15, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 68

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
xix	xix*
xx	xx
3/4 7-5	3/4 7-5
3/4 7-6	3/4 7-6*
B 3/4 7-1	B 3/4 7-1
-	B 3/4 7-1a
-	B 3/4 7-1b
B 3/4 7-2	B 3/4 7-2*

BASES

BASES

PLANT SYSTEMS

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The ultimate heat sink (Delaware River) shall be OPERABLE with:

a. A minimum river water level at or above elevation -13'0 Mean Sea Level, USGS datum (76'0 PSE&G datum), and

b. An average river water temperature of less than or equal to 88.6°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5 and *.

ACTION:

With the river water temperature in excess of 88.6°F, but at or below 89.9°F, continued plant operation is permitted for 6 hours provided that both loops of SACS/SSWS are verified to be OPERABLE; otherwise, with the requirements of the above specification not satisfied:

a. In OPERATIONAL CONDITIONS 1, 2 or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITIONS 4 or 5, declare the SACS system and the station service water system inoperable and take the ACTION required by Specification 3.7.1.1 and 3.7.1.2.

c. In Operational Condition *, declare the plant service water system inoperable and take the ACTION required by Specification 3.7.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The ultimate heat sink shall be determined OPERABLE:

a. By verifying the river water level to be greater than or equal to the minimum limit at least once per 24 hours.

b. By verifying river water temperature to be within its limit:

1) at least once per 24 hours when the river water temperature is less than or equal to 85°F.

2) at least once per 2 hours when the river water temperature is greater than 85°F.

* When handling irradiated fuel in the secondary containment.

3/4.7.2 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 Two independent control room emergency filtration system subsystems shall be OPERABLE with each subsystem consisting of:

a) One control room supply unit,
b) One filter train, and
c) One control room return air fan.

APPLICABILITY: All OPERATIONAL CONDITIONS and *.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3 with one control room emergency filtration subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4, 5 or *:

1. With one control room emergency filtration subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or initiate and maintain operation of the OPERABLE subsystem in the pressurization/recirculation mode of operation.

2. With both control room emergency filtration subsystems inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

c. The provisions of Specification 3.0.3 are not applicable in Operational Condition *.

SURVEILLANCE REQUIREMENTS

4.7.2 Each control room emergency filtration subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 85°F#.

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, the control area chilled water pump, flow

*When irradiated fuel is being handled in the secondary containment.

#This does not require starting the non-running control emergency filtration subsystem.

BASES

3/4.7.1 SERVICE WATER SYSTEMS

The OPERABILITY of the station service water and the safety auxiliaries cooling systems ensures that sufficient cooling capacity is available for continued operation of the SACS and its associated safety-related equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

If average river water temperature is greater than 85°F and a Loss of Offsite Power (LOP) concurrent with a loss of a SSWS/SACS loop occurs, operator actions must be taken to increase the heat removal of the SACS heat exchangers and minimize the total heat duty. These actions and the conditions under which they must be taken are contained in approved station operating procedures.

Although the sustained six hour temperature requirement would permit the temperature to rise above the new UHS limit for short durations, this allowance is justified based on the probabilistic risk assessment (PRA) results, transient nature of the UHS temperature excursions and the conservative nature of the temperature limit calculation.

3/4.7.2 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

The OPERABILITY of the control room emergency filtration system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all design basis accident conditions. Continuous operation of the system with the heaters and humidity control instruments OPERABLE for 10 hours during each 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR Part 50.

3/4.7.3 FLOOD PROTECTION

The requirement for flood protection ensures that facility flood protection features are in place in the event of flood conditions. The limit of elevation 10.5' Mean Sea Level is based on the elevation at which facility flood protection features provide protection to safety related equipment.

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

The reactor core isolation cooling (RCIC) system is provided to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without requiring actuation of any of the Emergency Core Cooling System equipment. The RCIC system is conservatively required to be OPERABLE whenever reactor steam dome pressure exceeds 150 psig. This pressure is substantially below that for which the RCIC system can provide adequate core cooling for events requiring the RCIC system.

The RCIC system specifications are applicable during OPERATIONAL CONDITIONS 1, 2 and 3 when reactor vessel steam dome pressure exceeds 150 psig because RCIC is the primary non-ECCS source of emergency core cooling when the reactor is pressurized.

With the RCIC system inoperable, adequate core cooling is assured by the OPERABILITY of the HPCI system and justifies the specified 14 day out-of-service period.

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REACTOR CORE ISOLATION COOLING SYSTEM (Continued)

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

3/4.7.5 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety related system.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip, and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Plant Operations Review Committee. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guide 8.8 and 8.10. The addition or deletion of any snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to each safety-related system. Therefore, the required inspection interval is based on the number of unacceptable snubbers found during the previous inspection in proportion to the sizes of the various snubber populations or categories. This inspection schedule is based on the guidance provided in Generic Letter 90-09. In order to establish the inspection frequency for each type of snubber on a safety-related system, it was assumed that the frequency of snubber failures and initiating events is constant with time and that the failure of any snubber on that system could cause the system to be unprotected and to result in failure during an assumed