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. 14
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 November 9, 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. 14, 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 February 1, 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

Walter R. Butler, 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: January 19, 1988

Previously concurred*

LA:PDI-2:DRPI/II	PM:PDI-2:DRPI/II*	OGC	D:PDI-2:DRPI/II	BC:RSB*
MO'Brien	GRivenbark:mr	MYoung	WButler	WHodges
1/13/88	12/11/87	1/5/88	1/14/88	12/11/87

February 1, 1988.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

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Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 19, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 14

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
i*	i*
ii	ii
iii	iii
iv*	iv*
1-7*	1-7*
1-8	1-8
1-9	1-9
1-10*	1-10*
3/4 9-3	3/4 9-3
3/4 9-4	3/4 9-4
3/4 10-7	3/4 10-7
3/4 10-8*	3/4 10-8*
B 3/4 3-5	B 3/4 3-5
B 3/4 3-6*	B 3/4 3-6*
B 3/4 9-1	B 3/4 9-1
B 3/4 9-2*	B 3/4 9-2*
B 3/4 10-1	B 3/4 10-1
-	-

SECTION

1.0 DEFINITIONS

SECTION

DEFINITIONS (Continued)

SECTION

DEFINITIONS (Continued)

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

SECONDARY CONTAINMENT INTEGRITY

1.38 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All secondary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, or deactivated automatic valve or damper, as applicable secured in its closed position, except as provided in Table 3.6.5.2-1 of Specification 3.6.5.2.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The filtration, recirculation and ventilation system is in compliance with the requirements of Specification 3.6.5.3.

d. For double door arrangements, at least one door in each access to the secondary containment is closed.

e. For single door arrangements, the door in each access to the secondary containment is closed, except for normal entry and exit.

f. The sealing mechanism associated with each secondary containment penetration, e.g., welds, bellows or O-rings, is OPERABLE.

g. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1.a.

SHUTDOWN MARGIN

1.39 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SITE BOUNDARY

1.40 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled, by the licensee.

SOLIDIFICATION

1.41 SOLIDIFICATION shall be the immobilization of wet radioactive wastes such as evaporator bottoms, spent resins, sludges, and reverse osmosis concentrates as a result of a process of thoroughly mixing the water type with a solidification agent(s) to form a free standing monolith with chemical and physical characteristics specified in the PROCESS CONTROL PROGRAM (PCP).

SOURCE CHECK

1.42 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

SPIRAL RELOAD

1.43 A SPIRAL RELOAD is a core loading methodology employed to refuel the core after a complete core unload. During a SPIRAL RELOAD the fuel is to be loaded into individual control cells (four bundles surrounding a control blade) in a spiral fashion centered on an SRM moving outward. Before initiating a SPIRAL RELOAD, up to four bundles may be loaded in the four bundle locations immediately surrounding each of the four SRMs to obtain the required channel count rate.

1.44 A SPIRAL UNLOAD is a core unloading methodology employed to defuel when the complete core is to be unloaded. The core unload is performed by first removing the fuel from the outermost control cells (four bundles surrounding a control blade). Unloading continues in a spiral fashion by removing fuel from the outermost periphery to the interior of the core, symmetric about the SRMs, except for the four bundles around each of the four SRMs. When sixteen or less fuel bundles are in the core, four around each of the four SRMs, there is no need to maintain the required channel count rate.

STAGGERED TEST BASIS

1.45 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.46 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TURBINE BYPASS SYSTEM RESPONSE TIME

1.47 The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two separate time intervals: a) time from initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established, and b) the time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve. Either response time may be measured by any series of sequential, overlapping, or total steps such that the entire response time is measured.

UNIDENTIFIED LEAKAGE

1.48 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

UNRESTRICTED AREA

1.49 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.50 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.51 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

TABLE 1.1

SURVEILLANCE FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
A	At least once per 366 days.
R	At least once per 18 months (550 days).
S/U	Prior to each reactor startup.
P	Prior to each radioactive release.
N.A.	Not applicable.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 At least 2 source range monitor* (SRM) channels shall be OPERABLE and inserted to the normal operating level with:##

a. Annunciation and continuous visual indication in the control room,

b. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and

c. Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, the "shorting links" removed from the RPS circuitry prior to and during the time any control rod is withdrawn.#

d. During a SPIRAL UNLOAD, the count rate may drop below 3 cps when the number of assemblies remaining in the core drops to sixteen or less.

e. During a SPIRAL RELOAD, up to four fuel assemblies may be loaded in the four bundle locations immediately surrounding each of the four SRMs prior to obtaining 3 cps. Until these assemblies have been loaded, the 3 cps count rate is not required.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS and insert all insertable control rods.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. At least once per 12 hours:

1. Performance of a CHANNEL CHECK,

*The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

#Not required for control rods removed per Specification 3.9.10.1 and 3.9.10.2.

##Three SRM channels shall be OPERABLE for critical shutdown margin demonstrations. An SRM detector may be retracted provided a channel indication of at least 100 cps is maintained.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying the detectors are inserted to the normal operating level, and

3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and another is located in an adjacent quadrant.

b. Performance of a CHANNEL FUNCTIONAL TEST:

1. Within 24 hours prior to the start of CORE ALTERATIONS, and

2. At least once per 7 days.

c. Verifying that the channel count rate is at least 0.7 cps:*

1. Prior to control rod withdrawal,

2. Prior to and at least once per 12 hours during CORE ALTERATIONS***, and

3. At least once per 24 hours***.

d. Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, verifying that the RPS circuitry "shorting links" have been removed, within 8 hours prior to and at least once per 12 hours during the time any control rod is withdrawn.**

*Provided signal-to-noise is $\geq$ 2. Otherwise, 3 cps.
**Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.
***Except as noted in Specifications 3.9.2.d and 3.9.2.e.

SPECIAL TEST EXCEPTIONS

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.

SURVEILLANCE REQUIREMENTS (Continued)

4.10.7. (Continued)

3. The RPS "shorting links" are removed.

4. The reactor mode switch is locked in the REFUEL position.

b. Performance of a CHANNEL FUNCTIONAL TEST within 24 hours prior to the start and at least once per 7 days during CORE ALTERATIONS.

c. Verifying for at least one SRM channel that the count rate is at least 0.7 cps*:

1. Immediately following the loading of the first 16 fuel bundles.

2. At least once per 12 hours thereafter during CORE ALTERATIONS.

*Provided signal-to-noise is $\geq$ 2. Otherwise, 3 cps.

MONITORING INSTRUMENTATION (Continued)

radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February, 1972.

3/4.3.7.4 REMOTE SHUTDOWN MONITORING INSTRUMENTATION AND CONTROLS

The OPERABILITY of the remote shutdown monitoring instrumentation and controls ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions shall not be made without this flux level information available to the operator. For a discussion of SPIRAL RELOAD and SPIRAL UNLOAD and the associated flux monitoring requirements, see Technical Specification Bases Section 3/4.9.2. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution of the reactor core.

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.8

The material originally contained in Section 3/4.3.7.8 was deleted with the issuance of the Full Power License. However, to maintain numerical continuity between the succeeding sections and existing station procedural references to those Technical Specifications Sections, 3/4.3.7.8 has been intentionally left blank.

3/4.3.7.9 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.7.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.7.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM. This will ensure the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring and controlling the concentrations of potentially explosive gas mixtures in the main condenser offgas treatment system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

BASES

3/4.9.1 REACTOR MODE SWITCH

Locking the OPERABLE reactor mode switch in the Shutdown or Refuel position, as specified, ensures that the restrictions on control rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radiation.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core. The flux need not be monitored for the first sixteen bundles loaded before a SPIRAL RELOAD or for the last sixteen bundles unloaded during a SPIRAL UNLOAD. In the case of the SPIRAL RELOAD, the sixteen bundles loaded may be different from the bundles scheduled to occupy the bundle locations for the next cycle provided; (i) the cold reactivity of any unscheduled bundle temporarily loaded is individually less than the cold reactivity of the respective bundle scheduled for the subject location, (ii) the uncontrolled k-infinity of the lattice is less than 1.31, and (iii) the bundles are arranged in four two-by-two arrays surrounding an SRM with each array having a minimum of 12 inches between it and an adjacent array.

3/4.9.3 CONTROL ROD POSITION

The requirement that all control rods be inserted during other CORE ALTERATIONS minimizes the possibility that fuel will be loaded into a cell without a control rod, although one rod may be withdrawn under control of the reactor mode switch refuel position one-rod-out-interlock.

3/4.9.4 DECAY TIME

The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during movement of fuel within the reactor pressure vessel.

3/4.9.6 REFUELING PLATFORM

The OPERABILITY requirements ensure that (1) the refueling platform will be used for handling control rods and fuel assemblies within the reactor pressure vessel, (2) each crane and hoist has sufficient load capacity for handling fuel assemblies and control rods, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL

The restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 and 3/4.9.9 WATER LEVEL - REACTOR VESSEL and WATER LEVEL - SPENT FUEL STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.10 CONTROL ROD REMOVAL

These specifications ensure that maintenance or repair of control rods or control rod drives will be performed under conditions that limit the probability of inadvertent criticality. The requirements for simultaneous removal of more than one control rod are more stringent since the SHUTDOWN MARGIN specification provides for the core to remain subcritical with only one control rod fully withdrawn.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be OPERABLE or that an alternate method capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during REFUELING, and (2) sufficient coolant circulation would be available through the reactor core to assure accurate temperature indication and to distribute and prevent stratification of the poison in the event it becomes necessary to actuate the standby liquid control system.

The requirement to have two shutdown cooling mode loops OPERABLE when there is less than 22 feet 2 inches of water above the reactor vessel flange ensures that a single failure of the operating loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 22 feet 2 inches of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event a failure of the operating RHR loop, adequate time is provided to initiate alternate methods capable of decay heat removal or emergency procedures to cool the core.

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 startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 OXYGEN CONCENTRATION

Relief from the oxygen concentration specifications is necessary in order to provide access to the primary containment during the initial startup and testing phase of operation. Without this access the startup and test program could be restricted and delayed.

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.