August 21, 1989

Docket No. 50-354

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: ADD CLARIFICATION AND CONSISTENCY TO THE REFUELING SPECIFICATIONS AND RAISE THE MINIMUM ALLOWABLE SOURCE RANGE MONITOR COUNT RATE (TAC NO. 72698)

Re: HOPE CREEK GENERATING STATION

The Commission has issued the enclosed Amendment No. 31 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 February 27, 1989 and supplemented on April 3, 1989.

This amendment adds clarification and consistency to the refueling specifications with respect to reference measurements, load setpoints and travel limits. Additional changes conservatively raise the minimum allowable source range monitor (SRM) count rate in Sections 4.9.2 and 4.3.7.6 of the Technical Specifications to agree with SRM requirements imposed elsewhere in the Specifications.

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/

Clyde Shiraki, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 31 to
   License No. NPF-57
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION w/enclosures:

| | | | | | |
|---|---|---|---|---|---|
| Docket File | MO'Brien (2) | Wanda Jones | SVarga | NRC PDR | OGC |
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| Brent Clayton | PDI-2 Reading | EJordan | ACRS (10) | EWenzinger | WButler |
| BGrimes | CMiles, GPA/PA | CShiraki(3)/SBrown | TMeek (4) | RDiggs, ARM/LFMB | |

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| PDI-2/LA | PDI-2/PM | PDI-2/D | OGC |
| MO'Brien | CShiraki:tr | WButler | Bachmann |
| 7/21/89 | 7/26/89 | 8/18/89 | 8/3/89 |

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

August 21, 1989

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2. Safety Evaluation

cc w/enclosures:
See next page

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

Hope Creek Generating Station

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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. 31
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 February 27, 1989 and supplemented on April 3, 1989 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. 31, 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, but implementable within sixty days to allow for any germane procedural revisions.

FOR THE NUCLEAR REGULATORY COMMISSION

Mohan Thadani for
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: August 21, 1989

| | | | |
|---|---|---|---|
| PDI-2/LA | PDI-2/PM | PDI-2/D | OGC |
| MO'Brien | CShiraki:tr | WButler | EBachmann |
| 7/21/89 | 7/26/89 | 8/18/89 | 8/13/89 |
| | | | w/chg to SE p 2 |

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FOR THE NUCLEAR REGULATORY COMMISSION

Mohan C. Thadani
for

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: August 21, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 31

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 |
|---|---|
| 3/4 3-87* | 3/4 3-87* |
| 3/4 3-88 | 3/4 3-88 |
| 3/4 9-1 | 3/4 9-1 |
| 3/4 9-2* | 3/4 9-2* |
| 3/4 9-3* | 3/4 9-3* |
| 3/4 9-4 | 3/4 9-4 |
| 3/4 9-7* | 3/4 9-7* |
| 3/4 9-8 | 3/4 9-8 |
| 3/4 9-9 | 3/4 9-9 |
| 3/4 9-10* | 3/4 9-10* |
| B 3/4 9-1* | B 3/4 9-1* |
| B 3/4 9-2 | B 3/4 9-2 |
| - | B 3/4 9-3 |
| - | - |

TABLE 4.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| INSTRUMENT | CHANNEL CHECK | CHANNEL CALIBRATION | APPLICABLE OPERATIONAL CONDITIONS |
|---|---|---|---|
| 1. Reactor Vessel Pressure | M | R | 1,2,3 |
| 2. Reactor Vessel Water Level | M | R | 1,2,3 |
| 3. Suppression Chamber Water Level | M | R | 1,2,3 |
| 4. Suppression Chamber Water Temperature | M | R | 1,2,3 |
| 5. Suppression Chamber Pressure | M | R | 1,2,3 |
| 6. Drywell Pressure | M | R | 1,2,3 |
| 7. Drywell Air Temperature | M | R | 1,2,3 |
| 8. Primary Containment Hydrogen/Oxygen Concentration Analyzer and Monitor | M | Q* | 1,2,3 |
| 9. Safety/Relief Valve Position Indicators | M | R | 1,2,3 |
| 10. Drywell Atmosphere Post-Accident Radiation Monitor | M | R** | 1,2,3 |
| 11. North Plant Vent Radiation Monitor# | M | R | 1,2,3 |
| 12. South Plant Vent Radiation Monitor# | M | R | 1,2,3 |
| 13. FRVS Vent Radiation Monitor# | M | R | 1,2,3 |
| 14. Primary Containment Isolation Valve Position Indication | M | R | 1,2,3 |

*Using sample gas containing:
a. Five volume percent oxygen balance nitrogen (oxygen analyzer channel).
b. Five volume percent hydrogen, balance nitrogen (hydrogen analyzer channel).

**CHANNEL CALIBRATION shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source.

#High range noble gas monitors.

INSTRUMENTATION

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.7.6 At least the following source range monitor channels shall be OPERABLE:

a. In OPERATIONAL CONDITION 2*, three.

b. In OPERATIONAL CONDITION 3 and 4, two.

APPLICABILITY: OPERATIONAL CONDITIONS 2*, 3 and 4.

ACTION:

a. In OPERATIONAL CONDITION 2* with one of the above required source range monitor channels inoperable, restore at least 3 source range monitor channels to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4 with one or more of the above required source range monitor channels inoperable, verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

SURVEILLANCE REQUIREMENTS

4.3.7.6 Each of the above required source range monitor channels shall be demonstrated OPERABLE by:

a. Performance of a:

1. CHANNEL CHECK at least once per:

a) 12 hours in CONDITION 2*, and

b) 24 hours in CONDITION 3 or 4.

2. CHANNEL CALIBRATION** at least once per 18 months.

b. Performance of a CHANNEL FUNCTIONAL TEST:

1. Within 24 hours prior to moving the reactor mode switch from the Shutdown position, if not performed within the previous 7 days, and

2. At least once per 31 days.

c. Verifying, prior to withdrawal of control rods, that the SRM count rate is at least 3 cps with the detector fully inserted.

*With IRM's on range 2 or below.
**Neutron detectors may be excluded from CHANNEL CALIBRATION.

## 3/4.9 REFUELING OPERATIONS

### 3/4.9.1 REACTOR MODE SWITCH

#### LIMITING CONDITION FOR OPERATION

3.9.1 The reactor mode switch shall be OPERABLE and locked in the Shutdown or Refuel position. When the reactor mode switch is locked in the Refuel position:

a. A control rod shall not be withdrawn unless the Refuel position one-rod-out interlock is OPERABLE.

b. CORE ALTERATIONS shall not be performed using equipment associated with a Refuel position interlock unless at least the following associated Refuel position interlocks are OPERABLE for such equipment.

   1. All rods in.
   2. Refuel platform position.
   3. Refuel platform main hoist fuel-loaded.
   4. Service platform hoist fuel-loaded.

APPLICABILITY: OPERATIONAL CONDITION 5* #.

ACTION:

a. With the reactor mode switch not locked in the Shutdown or Refuel position as specified, suspend CORE ALTERATIONS and lock the reactor mode switch in the Shutdown or Refuel position.

b. With the one-rod-out interlock inoperable, lock the reactor mode switch in the Shutdown position.

c. With any of the above required Refuel position equipment interlocks inoperable, suspend CORE ALTERATIONS with equipment associated with the inoperable Refuel position equipment interlock.

* See Special Test Exceptions 3.10.1 and 3.10.3.

# The reactor shall be maintained in OPERATIONAL CONDITION 5 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The reactor mode switch shall be verified to be locked in the Shutdown or Refuel position as specified:

a. Within 2 hours prior to:

1. Beginning CORE ALTERATIONS, and

2. Resuming CORE ALTERATIONS when the reactor mode switch has been unlocked.

b. At least once per 12 hours.

4.9.1.2 Each of the above required reactor mode switch Refuel position interlocks* shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST within 24 hours prior to the start of and at least once per 7 days during control rod withdrawal or CORE ALTERATIONS, as applicable.

4.9.1.3 Each of the above required reactor mode switch Refuel position interlocks* that is affected shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST prior to resuming control rod withdrawal or CORE ALTERATIONS, as applicable, following repair, maintenance or replacement of any component that could affect the Refuel position interlock.

*The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that all control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

REFUELING OPERATIONS

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 3 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.**

**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.

## 3/4.9.5 COMMUNICATIONS

### LIMITING CONDITION FOR OPERATION

3.9.5 Direct communication shall be maintained between the control room and refueling floor personnel.

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.

ACTION:

When direct communication between the control room and refueling floor personnel cannot be maintained, immediately suspend CORE ALTERATIONS.

### SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the control room and refueling floor personnel shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

## 3/4.9.6 REFUELING PLATFORM

### LIMITING CONDITION FOR OPERATION

3.9.6 The refueling platform shall be OPERABLE with the main hoist to be used for handling fuel assemblies or control rods within the reactor pressure vessel and the frame-mounted or monorail-mounted auxiliary hoists to be used for handling control rods within the reactor pressure vessel.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel.

ACTION:

With the requirements for refueling platform OPERABILITY not satisfied, suspend use of any inoperable refueling platform equipment from operations involving the handling of control rods and fuel assemblies within the reactor pressure vessel after placing the load in a safe condition.

### SURVEILLANCE REQUIREMENTS

4.9.6. The refueling platform hoists used for handling of control rods or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the start of such operations by:

a. Demonstrating operation of the overload cutoff on the main hoist when the load exceeds 1200 + 0, -50 pounds.

b. Demonstrating operation of the overload cutoff on the frame-mounted and monorail-mounted auxiliary hoists when the load exceeds 500 ± 50 pounds.

c. Demonstrating operation of the main hoist uptravel stop when uptravel brings the point where the grapple attaches to the fuel bundle to 6 feet 6 inches, +3, -0 inches below the normal water level.

d. Demonstrating operation of the frame-mounted and monorail-mounted auxiliary hoists' uptravel stops when uptravel brings the point where the grapple attaches to a control rod to 6 feet, +1, -0 feet below the normal water level.

e. Demonstrating operation of the slack cable cutoff on the main hoist when the load is less than 50 ± 10 pounds.

f. Demonstrating operation of the loaded rod block interlock on the main hoist when the load exceeds 535 ± 50 pounds.

g. Demonstrating operation of the redundant loaded interlock on the main hoist when the load exceeds 550 ± 50 pounds.

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## 3/4.9.7 CRANE TRAVEL-SPENT FUEL STORAGE POOL

### LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1200 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks unless handled by a single failure proof handling system.

APPLICABILITY: With fuel assemblies in the spent fuel storage pool racks.

ACTION:

With the requirements of the above specification not satisfied, place the polar crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.9.7.1 Interlocks and physical stops which prevent polar crane main hoist travel over fuel assemblies in the spent fuel storage pool racks shall be demonstrated OPERABLE within 7 days prior to and at least once per 7 days during polar crane operation.

4.9.7.2 The single failure proof handling system shall be visually inspected and verified OPERABLE within 7 days prior to and at least once per 7 days during polar crane operation.

## 3/4.9 REFUELING OPERATIONS

### 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 fuel assemblies and control rods, with limits placed upon auxiliary hoists' usage, within the reactor pressure vessel, (2) each crane and hoist has sufficient load capacity for handling the loads within its permitted usage, (3) the core internals are protected from excessive lifting force in the event that they are inadvertently engaged during lifting operations, (4) the core internals are protected from a fuel bundle or control rod drop with more impact energy than that assumed in the accident analyses, (5) refueling interlocks and rod blocks are initiated to prevent conditions that could result in criticality during refueling 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.

RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION (Continued)

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.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## SUPPORTING AMENDMENT NO. 31 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 February 27, 1989 and supplemented on April 3, 1989, Public Service Electric & Gas Company requested an amendment to Facility Operating License No. NPF-57 for the Hope Creek Generating Station. The proposed amendment would add clarification and consistency to the refueling specifications with respect to reference measurements, load setpoints and travel limits. The changes would also conservatively raise the minimum allowable Source Range Monitor (SRM) count rate to agree with SRM requirements imposed elsewhere in the specifications.

## 2.0 EVALUATION

(a) The changes to TS 3.9.1.b.3 and 3.9.6 clarify the use of the main hoist as the only hoist permitted for fuel movement within the RPV. Section 9.1.4.2.10.2 of the final safety analysis report (FSAR) states that fuel handling is performed using the main hoist fuel grapple. This requirement is implied by the wording in the TS surveillance requirement section; however, it is not clearly stated in the TS limiting condition of operation (LCO) section. The proposed change will clarify the required use of only the main hoist for fuel movement within the RPV.

(b) The change to TS 4.9.6.1.b (which becomes TS 4.9.6.c) specifies a shift of reference from the "top of active fuel" to the "point of attachment of the grapple to the fuel bail handle" and also revises the main hoist up-travel stop setpoint with regard to the distance between the refueling cavity normal water level and the new point of reference. In support of this change, the licensee provided an "Analysis of Refueling Bridge Up-travel Limits." This analysis shows that the new value of 6'6" from the point of attachment of the grapple to the fuel bail handle corresponds to the present TS limit of 8'0" from the top of active fuel. The tolerance (+3'/-0") assigned to the main hoist up-travel stop limit defines a range in which to set the stop that provides adequate clearance for transporting loads over the fuel transfer "cattle chute." The licensee's analysis indicates that no fuel bundle is raised higher than the height assumed in the fuel handling accident analyses provided in Section 15.7.4 of the FSAR. Furthermore, the licensee's analysis concluded that the radiological doses at the refueling platform are not increased by this change because the combined effect of the revised reference point and the dimensional change to the up-travel limit results in no change in the actual height of the fuel bundle with respect to the surface of the refueling cavity water when withdrawn to the normal up-travel stop. On the basis of the above consideration, the staff agrees with the licensee that the change to TS 4.9.6.1.b does not allow a fuel bundle to be lifted higher than previously permitted by the analysis and adds a minimum height-hoisted tolerance to ensure that a bundle will have adequate clearance over the fuel transfer chute floor.

(c) The change to TS 4.9.6.1.d (which becomes TS 4.9.6.f) raises the loaded rod block setpoint from 485 pounds to 535 pounds, thereby allowing a blade guide to be hoisted by the main hoist without tripping the rod block interlock. The licensee stated that this problem was encountered in the past during control rod replacements. The setpoint at which the fuel-loaded rod block is actuated is below the redundant interlock setpoint 550 pounds. Therefore, the basis for the fuel-loaded rod block (to ensure that no control rod is removed while fuel is being handled) satisfies the criteria.

(d) The changes to TS 4.9.6.2.b and TS 4.9.6.3.b involve the combination of duplicative specifications for both auxiliary hoists in TS 4.9.6.d. The licensee stated that the point of reference and the dimension for the up-travel stops for the two hoists are changed to permit transport of control rods out of the RPV and through the "cattle chute" without the use of special rigging arrangements currently required to prevent interface problems in the chute. The licensee's analysis demonstrates that the proposed increase in height to which a control rod can be lifted is within the bounds of the previous analysis for a fuel handling (bundle drop) accident provided in Section 15.7.4 of the FSAR. The radiation dose rates associated with a control rod assembly at the up-travel limit would be slightly increased by this change; however, the calculated radiation dose rate (3.6 mr/hr at the surface of the water in the refueling cavity) from a control rod withdrawn to the height requested by the amendment will remain less than the calculated dose rate from a fuel bundle withdrawn to its current up-travel limit by the main hoist (7.6 mr/hr one foot above the surface of the water in the refueling cavity). The staff has reviewed the licensee's analysis and concurs with the licensee's conclusion.

(e) The changes to TS 4.9.6.2.c and 4.9.6.3.c involve an elimination of surveillance requirements for rod block interlocks on the monorail and the frame-mounted auxiliary hoists. The rod block is not required for these two auxiliary hoists as these hoists are precluded from lifting any fuel bundle as indicated in the proposed TS 3.9.6. Therefore, elimination of surveillance requirements for rod block interlocks on auxiliary hoists is acceptable.

(f) The changes to TS 4.3.7.6.c and 4.9.2.c conservatively raise the minimum allowable Source Range Monitor (SRM) count rate to agree with SRM requirements imposed elsewhere in the specifications.

## 3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

## 4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (54 FR 29410) on July 12, 1989 and consulted with the State of New Jersey. No public comments were received. The comments from the Bureau of Nuclear Engineering of the State of New Jersey and the NRC staff response appear below:

Comment No. 1) Can a fuel assembly be raised on the auxiliary hoists?

NRC Staff Response: Although the weight of a fuel assembly is within the lifting capacity of the auxiliary hoists, procedural controls, Technical Specifications load limits and restrictions, and an overload cutoff preclude their use for this purpose.

Comment No. 2) What radiation monitor protection is or will be provided on the refueling bridge since approval of this License Change Request will allow a control rod to be raised higher than previously?

NRC Staff Response: There is a permanent area monitor that alarms in the control room, and, during refueling, there is a portable radiation monitor on the bridge that alarms locally.

Comment No. 3) Can a fuel assembly and a control rod be raised simultaneously? For instance, one on an auxiliary hoist and one on the main hoist? If this can be done, has the radiation dose rate been calculated for this lift?

NRC Staff Response: This simultaneous lift is precluded by a three position selector switch with positions for Monorail Mounted Auxiliary Hoist, Frame Mounted Auxiliary Hoist, and Main Hoist. Administratively, the licensee's refueling procedures do not allow the lift.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security nor to the health and safety of the public.

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