July 24, 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. M98106)

Dear Mr. Eliason:

The Commission has issued the enclosed Amendment No. 101 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 3, 1997, as supplemented by letter dated May 5, 1997.

This amendment changes the Hope Creek TSs as follows: (1) TS 3/4.3.1, "Reactor Protection System Instrumentation," TS 3/4.3.2, "Isolation Actuation Instrumentation," and TS 3/4.3.3, "Emergency Core Cooling System Actuation Instrumentation," to include additional information concerning response time testing; (2) TS 4.0.5 to reference inservice inspection and test requirements; (3) TS 3/4.6.1, "Primary Containment," and associated Bases to reflect a design modification; (4) TS 3/4.7.7, "Main Turbine Bypass System," to specify a new operability requirement; and (5) the Bases for TS 3/4.8, "Electrical Power Systems."

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. 101 to License No. NPF-57
2. Safety Evaluation

cc w/encls: See next page

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Add: Dave Jaffe

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

July 24, 1997

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2. Safety Evaluation

cc w/encls: See next page

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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. 101
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 3, 1997, as supplemented by letter dated May 5, 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. 101, 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 to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

J. Ce. Dyer 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: July 24, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 101

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 0-2 | 3/4 0-2 |
| 3/4 3-1 | 3/4 3-1 |
| 3/4 3-10 | 3/4 3-10 |
| 3/4 3-32 | 3/4 3-32 |
| 3/4 6-4 | 3/4 6-4 |
| 3/4 6-11 | 3/4 6-11 |
| 3/4 7-21 | 3/4 7-21 |
| B 3/4 6-3 | B 3/4 6-3 |
| B 3/4 8-1 | B 3/4 8-1 |
| - | B 3/4 8-1a |

## APPLICABILITY

## SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within its specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50 Sections 50.55a(f) and 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(f)(6)(i) or Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and

## 3/4.3 INSTRUMENTATION
## 3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within twelve hours. The provisions of Specification 3.0.4 are not applicable.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shall be demonstrated to be within its limit at least once per 18 months. Neutron detectors are exempt from response time testing. For the Reactor Vessel Steam Dome Pressure - High Functional Unit and the Reactor Vessel Water Level - Low, Level 3 Functional Unit, the sensor is eliminated from response time testing for RPS circuits. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

4.3.1.4 The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 2 or 3 from OPERATIONAL CONDITION 1 for the Intermediate Range Monitors.

*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 6 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

**If more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause the Trip Function to occur.

## SURVEILLANCE REQUIREMENTS

4.3.2.1 Each isolation actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.2.1-1.

4.3.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation trip function shall be demonstrated to be within its limit at least once per 18 months. Radiation detectors are exempt from response time testing. The sensor is eliminated from response time testing for MSIV isolation logic circuits of the following trip functions: Reactor Vessel Water Level - Low Low Low, Level 1; Main Steam Line Pressure - Low; Main Steam Line Flow - High. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months, where N is the total number of redundant channels in a specific isolation trip system.

## 3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2.

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

### SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.3.1-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function shall be demonstrated to be within the limit at least once per 18 months. ECCS actuation instrumentation is eliminated from response time testing. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS trip system.

SURVEILLANCE REQUIREMENTS (Continued)

2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be between 0.75 $L_a$ and 1.25 $L_a$.

The formula to be used is: $[L_o + L_{am} - 0.25\ L_a] \leq L_c \leq [L_o + L_{am} + 0.25\ L_a]$ where $L_c$ = supplement test result; $L_o$ = superimposed leakage; and $L_a$ = measured Type A leakage.

d. Type B and C tests shall be conducted with gas at $P_a$, 48.1 psig*, at intervals no greater than 24 months except for tests involving:

1. Air locks,

2. Main steam line isolation valves,

3. Valves pressurized with fluid from a seal system,

4. All containment isolation valves in hydrostatically tested lines in Table 3.6.3-1 which penetrate the primary containment, and

5. DELETED.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line isolation valves shall be leak tested at least once per 18 months.

g. Containment isolation valves which form the boundary for the long-term seal of the feedwater lines in Table 3.6.3-1 shall be hydrostatically tested at 1.10 $P_a$, 52.9 psig, at least once per 18 months.

h. All containment isolation valves in hydrostatically tested lines in Table 3.6.3-1 which penetrate the primary containment shall be leak tested at least once per 18 months.

i. DELETED.

j. The provisions of Specification 4.0.2 are not applicable to Specifications 4.6.1.2.a, 4.6.1.2.b, 4.6.1.2.c, 4.6.1.2.d, and 4.6.1.2.e.

*Unless a hydrostatic test is required per Table 3.6.3-1.

CONTAINMENT SYSTEMS

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.8 The drywell and suppression chamber purge system, including the 6-inch nitrogen supply line, may be in operation for up to 500 hours each 365 days with the supply and exhaust isolation valves in one supply line and one exhaust line open for containment prepurge cleanup, inerting, deinerting, or pressure control.*

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With a drywell or suppression chamber purge supply and/or exhaust isolation valve and/or the nitrogen supply valve open, except as permitted above, close the valves(s) or otherwise isolate the penetration(s) within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With a drywell purge supply or exhaust isolation valve, or a suppression chamber purge supply or exhaust isolation valve or the nitrogen supply valve, having a measured leakage rate exceeding the limit of Surveillance Requirement 4.6.1.8.2, restore the inoperable valve(s) to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.8.1 Before being opened, the drywell and suppression chamber purge supply and exhaust, and nitrogen supply butterfly isolation valves shall be verified not to have been open for more than 500 hours in the previous 365 days.*

4.6.1.8.2 At least once per 24 months, the 26-inch drywell purge supply and exhaust isolation valves and the 24-inch suppression chamber purge supply and exhaust isolation valves and the 6-inch nitrogen supply valve shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 $L_a$ per penetration when pressurized to $P_a$ 48.1 psig.

* Valves open for pressure control are not subject to the 500 hours per 365 days limit, provided the 2-inch bypass lines are being utilized.

## 3/4.7.7 MAIN TURBINE BYPASS SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.7 The main turbine bypass system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1 when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION: With the main turbine bypass system inoperable, restore the system to OPERABLE status within 2 hours or reduce THERMAL POWER to less than or equal to 25% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.7.7 The main turbine bypass system shall be demonstrated OPERABLE at least once per:

a. 31 days by cycling each turbine bypass valve through at least one complete cycle of full travel, and

b. 18 months by:

1. Performing a system functional test which includes simulated automatic actuation and verifying that each automatic valve actuates to its correct position.

2. Demonstrating TURBINE BYPASS SYSTEM RESPONSE TIME meets the following requirements when measured from the initial movement of the main turbine stop or control valve:

a) 80% of turbine bypass system capacity shall be established in less than or equal to 0.3 second.

b) Bypass valve opening shall start in less than or equal to 0.1 second.

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM (Continued)

The use of the drywell and suppression chamber purge exhaust lines for pressure control during plant Operational Conditions 1, 2 and 3 is unrestricted provided 1) only the inboard purge exhaust isolation valves on these lines and the vent valves on the 2-inch vent paths are used and 2) the outboard purge exhaust isolation valves remain closed. This is because in such a situation, the vent valves will sufficiently choke the flow and additionally the applicable valves will close in a timely manner during a LOCA or steam line break accident and therefore the control room and the site boundary dose guidelines of applicable 10 CFR dose limits will not be exceeded in the event of an accident. The design of the purge supply and exhaust isolation valves and the 6-inch nitrogen supply valve meets the requirements of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations".

The 0.60 $L_a$ leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2. DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 62 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1020 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid must not exceed 62 psig, the suppression chamber maximum internal design pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant to be considered is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in this specification, containment pressure during the design basis accident is approximately 48.1 psig which is below the design pressure of 62 psig. Maximum water volume of 122,000 $ft^3$ results in a downcomer submergence of 3.33 ft and the minimum volume of 118,000 $ft^3$ results in a submergence of approximately 3.0 ft. The majority of the Bodega tests were run with a submerged length of four feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown

## 3/4.8 ELECTRICAL POWER SYSTEMS

### BASES

### 3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criteria 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one of the onsite A.C. and the corresponding D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. or D.C. source.

The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources", December 1974 as modified by plant specific analysis and diesel generator manufacturer recommendations. When two diesel generators are inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generators as a source of emergency power, are also OPERABLE. This requirement is intended to provide assurance that a loss of offsite power event will not result in a complete loss of safety function of critical systems during the period two or more of the diesel generators are inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component. The 14 day AOT for the "C" and "D" EDGs is based upon the following conditions being met:

1. Hope Creek should verify through Technical Specifications, procedures or detailed analyses that the systems, subsystems, trains, components and devices that are required to mitigate the consequences of an accident are available and operable before removing an EDG for extended preventative maintenance (PM). In addition, positive measures should be provided to preclude subsequent testing or maintenance activities on these systems, subsystems, trains, components and devices while the EDG is inoperable.

2. The overall unavailability of the EDG should not exceed the performance criteria developed for implementation of 10CFR50.65 requirements as described in NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", as endorsed by Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", June 1993.

3. When the "C" or "D" EDG is removed from service for an extended 14 day AOT, any two of the remaining EDGs must be capable, operable and available to mitigate the consequences of a LOOP condition.

4. The removal from service of safety systems and important non-safety equipment, including offsite power sources, should be minimized during the extended 14 day AOT.

5. Entry into this LCO should not be abused by repeated voluntary entry into and exit from the LCO. The primary intent of the extended EDG AOT is that the extended EDG AOT from 72 hours to 14 days may be needed to perform preplanned EDG maintenance such as teardowns and modifications that would otherwise extend beyond the original 72 hour AOT.

6. Any component testing or maintenance that increases the likelihood of a plant transient should be avoided. Plant operation should be stable during the extended 14 day AOT.

7. Voluntary entry into this LCO action statement should not be scheduled if adverse weather conditions are expected.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that (1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The surveillance requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies", March 10, 1971, Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants", Revision 1, August 1977 and Regulatory Guide 1.137 "Fuel-Oil Systems for Standby Diesel Generators", Revision 1, October 1979 as modified by plant specific analysis, diesel generator manufacturer's recommendations, and Amendment 59, to the Facility Operating License, issued November 22, 1993.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## RELATED TO AMENDMENT NO. 101 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 3, 1997, as supplemented by letter dated May 5, 1997, the Public Service Electric and Gas Company (PSE&G) requested an amendment to the Hope Creek Generating Station (HCGS) Facility Operating License No. NPF-57. The proposed change to HCGS Technical Specifications (TSs) would: (1) Change TS 3/4.3.1, "Reactor Protection System Instrumentation," TS 3/4.3.2, "Isolation Actuation Instrumentation," and TS 3/4.3.3, "Emergency Core Cooling System Actuation Instrumentation," to include additional information concerning response time testing; (2) Change TS 4.0.5 to reference inservice inspection and test requirements; (3) Change TS 3/4.6.1, "Primary Containment," and associated Bases to reflect a design modification; (4) Change TS 3/4.7.7, "Main Turbine Bypass System," to specify a new operability requirement; and (5) Change the Bases for TS 3/4.8, "Electrical Power Systems."

## 2.0 DISCUSSION AND EVALUATION

### 2.1 Response Time Testing

With regard to Reactor Protection System (RPS), Isolation System (IS), and Emergency Core Cooling System (ECCS) instrumentations, the NRC staff issued License Amendment No. 85, on October 24, 1995, to eliminate the requirement for response time testing (RTT) of certain classes of instrumentation, including sensors, and transfer other RTT requirements to the HCGS Updated Final Safety Analysis Report (UFSAR). The amended TS, however, did not reflect the NRC staff position (as stated in the Safety Evaluation for License Amendment No. 85) that certain instrumentation channel initiation sensors need not be part of the RTT requirements. This situation results from the definition of RTT in TS 1.36, "Reactor Protection System Response Time," which states, in part, "REACTOR PROTECTION RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor [emphases added] until the de-energization of the scram pilot valve solenoids." Similar definitions for ECCS RTT and IS RTT are contained in TSs 1.13 and 1.19, respectively. Accordingly, the licensee has proposed adding clarifications to TSs 4.3.1.3, 4.3.2.3, and 4.3.3.3, to clearly reflect the NRC staff position on RTT of the subject equipment as follows:

* TS 4.3.1.3, for RPS RTT, "For the Reactor Vessel Steam Dome Pressure - High Functional Unit and the Reactor Vessel Water Level - Low, Level 3 Functional Unit, the sensor is eliminated from response time testing for the RPS circuits."

* TS 4.3.2.3, for IS RTT, "The sensor is eliminated from response time testing for [main steam isolation valve] MSIV isolation logic circuits of the following trip functions: Reactor Vessel Water Level - Low Low Low, Level 1; Main Steam Line Pressure - Low; Main Steam Line Flow - High."

* TS 4.3.3.3, for ECCS RRT - "ECCS actuation instrumentation is eliminated from response time testing."

The NRC staff agrees with the licensee that elimination of RTT, for the instrumentation addressed in the March 3, 1997 application, was approved in License Amendment No. 85. This elimination of RTT, for selected instrumentation, would have been addressed in various tables in the TSs but these tables were eliminated as part of Amendment No. 85 and subsequently transferred to the HCGS UFSAR. The UFSAR has the following tabular notations:

* Table 7.2-3, "Reactor Protection System Response Times," notes that the RPS sensors for "Reactor Vessel Steam Dome - High," and "Reactor Vessel Water Level - Low, Level 3," are eliminated from the RRT.

* Table 7.3-16, "Isolation System Instrumentation Response Time," notes that the sensor of MSIV actuation logic, for "Reactor Vessel Water Level - Low Low Low, Level 1," "Main Steam Line Pressure - Low," and "Main Steam Line Flow - High," are eliminated from the RRT.

* Table 7.3-17, "Emergency Core Cooling System Response Times," notes that ECCS actuation instrumentation is eliminated from the RRT.

Based upon the above, the proposed changes to the TSs, which reiterate the elimination of RTT for selected instrumentation, are acceptable.

## 2.2 Inservice Testing (IST)/Inservice Inspection (ISI)

At the current time, TS 4.0.5a, requires that ISI of American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components, and IST of ASME Boiler and Pressure Vessel Code (ASME Code) Class 1, 2, and 3 pumps and valves be undertaken in accordance with Section XI of the ASME Code and applicable addenda in accordance with 10 CFR 50.55a(g), except where specific relief has been granted in accordance with 10 CFR 50.55a(g)(6)(i). The licensee's application draws attention to the fact that the two referenced citations, 10 CFR 50.55a(g) and 10 CFR 50.55a(g)(6)(i), apply solely to ISI and not to IST. The licensee has requested that the corresponding references to IST, 10 CFR 50.55a(f) and 10 CFR 50.55a(f)(6)(i), be added to TS 4.0.5a.

The NRC staff agrees with the licensee that the two referenced citations, 10 CFR 50.55a(g) and 10 CFR 50.55a(g)(6)(i), apply solely to ISI and not to IST. Accordingly, it is appropriate and acceptable to add the corresponding references to IST, 10 CFR 50.55a(f) and 10 CFR 50.55a(f)(6)(i) to TS 4.0.5a.

2.3 Primary Containment - Drywell and Suppression Chamber Purge System

The licensee has undertaken a modification, which resulted in the replacement of valves with resilient seats, in the Drywell and Suppression Chamber Purge System (DSCPS), with valves utilizing metal seats. The valves that were replaced were as follows: the drywell purge supply and exhaust isolation valves, the suppression chamber (torus) purge supply and exhaust isolation valves and the nitrogen supply valve. Accordingly, the licensee has proposed revised surveillance intervals for the replacement, metal-seated, valves. In addition, the licensee has proposed the deletion of references, in TS 3/4.6.1, to the specified DSCPS valves with resilient seats.

The HCGS Operating License Safety Evaluation Report, NUREG-1048, dated October 1984, expresses concerns related to the use of valves with resilient seats in containment purge and vent lines. Section 6.2.4.1, "Containment Purge System," of NUREG-1048 states:

> As a result of the numerous reports on unsatisfactory performance of the resilient seats for the isolation valves in containment purge and vent lines (addressed in IE Circular 77-11, dated September 6, 1977), Generic Issue B-20, "Containment Leakage Due to Seal Deterioration," was established to evaluate the matter and establish an appropriate testing frequency for the isolation valves. Excessive leakage past the resilient seats of isolation valves in purge/vent lines is typically caused by severe environmental conditions and/or frequent use. Consequently, the leakage test frequency for these valves should be keyed to the occurrence of severe environmental conditions and the use of the valves.

The applicant had committed to add the following provisions to the Technical Specifications for the leak testing of purge/vent line isolation valves:

> Leakage integrity tests shall be performed on the containment isolation valves with resilient material seals in (a) active purge/vent systems (i.e., those which amy be operated during plant operating Modes 1 through 4) at least once every three months and (b) passive purge systems (i.e., those which must be administratively controlled closed during reactor operating 1 Modes through 4) at least once every 6 months.

The above requirements for leak testing of resilient-seated isolation valves in the purge and vent lines were incorporated in TS 4.6.1.8.2 with surveillance intervals specified according to whether the valves were operated as "active" or "passive." The licensee has proposed replacing the active/passive surveillance requirements (and associated 6-month, but not more than 92 days, leakage test frequency) with a single test frequency of 24 months. The proposed change to TS 4.6.1.8.2 is acceptable since the existing TSs were only established to address the use of resilient-seated valves in the purge and vent systems, which are no longer in use at HCGS. Moreover, the proposed 24-month surveillance interval for the metal-seated valves, which replaced the resilient seated valves, is consistent with established practice at HCGS as required by TS 4.6.1.2d. In addition, the proposed deletion of references to testing of resilient-seated purge supply and exhaust isolation valves in TSs 4.6.1.2d.5 and 4.6.1.2i is acceptable in that these references apply only to this type of valve, which is no longer utilized at HCGS.

The licensee has also deleted those portions of the TS Bases that are only applicable to valves with resilient-seated valves in the purge and vent systems, which are no longer in use at HCGS.

2.4 Main Turbine Bypass System

At the present time, TS 3/4.7.7 requires that the Main Turbine Bypass System (MTBS) be operable in Operational Condition 1. If the MTBS becomes inoperable, the associated action statement requires that the MTBS be restored to operable status within 2 hours or reduce rated thermal power (RTP) to less than 25% of RTP within the next 4 hours. Unlimited operation below 25% of RTP, with an inoperable MTBS, is permitted by TS 3/4.7.7. The licensee has proposed that the "Applicability" statement be changed to require the MTBS to be operable in Operational Condition 1 when the RTP is greater than, or equal to, 25% of RTP.

The MTBS is described in Section 10.4.4, "Turbine Bypass System," of the UFSAR. The MTBS consists of nine globe-type valves which open sequentially to route steam directly to the main condenser when main turbine demand is less than the steam supply (e.g., following the trip of the main turbine). The capacity of the MTBS is 25% of the main turbine valve's wide open flow. Section 10.4.4.3, "Safety Evaluation," of the UFSAR indicates that, "The Turbine Bypass System has no safety-related function. Failure of the system does not compromise any safety-related system or component or prevent a safe shutdown of the plant." The UFSAR does, however, provide an analysis of the consequences of a main turbine trip, below 30% of rated power, with failure of the MTBS. The results of the analysis, presented in UFSAR Section 15.2.3.3.3.3, "Turbine Trip with Bypass Valve Failure, Low Power," show that fuel and systems performance are well within acceptable ranges following the transient.

The NRC staff agrees that the applicability statement of TS 3/4.7.7 can be changed to require that the MTBS only be required operable at, or above, 25% of RTP. Considering that the current remedial action required by the TS for an inoperable MTBS allows unlimited operation at, or below 25% of RTP, it appears that operability of the MTBS is not required in this power range. This is the approach used in the NRC's applicable Improved Standard Technical Specifications, NUREG-1433, Rev. 1. Accordingly, the licensee's proposed change to TS 3/4.7.7 is acceptable.

2.5 Change the Bases for 3/4.8, "Electrical Power Systems"

The NRC staff notes that the licensee has proposed changes to the Bases for TS 3/4.8. No NRC approval is needed to change the TS Bases.

## 3.0 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 effluent 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 33131). 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.

## 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 indicated that there were no comments.

## 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.

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