Mr. Harold W. Keiser
Chief Nuclear Officer & President
Nuclear Business Unit
Public Service Electric & Gas
Company
Post Office Box 236
Hancocks Bridge, NJ 08038

March 9, 1999

SUBJECT: HOPE CREEK GENERATING STATION, ISSUANCE OF AMENDMENT, SAFETY LIMIT MINIMUM CRITICAL POWER RATIO (TAC NO. MA3484)

Dear Mr. Keiser:

The Commission has issued the enclosed Amendment No.117 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station (HCGS). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated August 25, 1998, as supplemented January 27, 1999.

This amendment revises TS 2.1.2, "THERMAL POWER, High Pressure and High Flow," and the Bases for TS 2.1, "Safety Limits." These changes are being made to implement appropriately conservative Safety Limit Minimum Critical Power Ratio values for the Cycle 9 HCGS core and fuel designs. An administrative revision is also being made to TS 6.9.1.9 to reflect these changes for Cycle 9.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:

Richard B. Ennis, Project Manager
Project Directorate I-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures: 1. Amendment No. 117 to License No. NPF-57
2. Safety Evaluation

cc w/encls: See next page

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

March 9, 1999

Mr. Harold W. Keiser
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RB Ennis

Richard B. Ennis, Project Manager
Project Directorate I-2
Division of Licensing Project Management
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Mr. Harold W. Keiser
Public Service Electric & Gas
Company

Hope Creek Generating Station

cc:

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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. 117
License No. NPF-57

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by the Public Service Electric & Gas Company (PSE&G) dated August 25, 1998, as supplemented January 27, 1999, 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. 117, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSE&G shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days after the completion of Cycle 8.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Director
Project Directorate I-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment Changes to the Technical Specifications

Date of Issuance: March 9, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 117

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
2-1	2-1
B 2-1	B 2-1
6-21	6-21

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.09 with two recirculation loop operation and shall not be less than 1.11 with single recirculation loop operation, in both cases with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.*

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.09 with two recirculation loop operation or less than 1.11 with single recirculation loop operation and in both cases with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

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

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

*Values applicable to Cycle 9 operation only.

2.1 SAFETY LIMITS

BASES

==

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.09 for two recirculation loop operation and 1.11 for single recirculation loop operation. MCPR greater than 1.09 for two recirculation loop operation and 1.11 for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the applicable NRC-approved critical power correlation is not valid for all critical power calculations performed at reduced pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

CORE OPERATING LIMITS REPORT (Continued)

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in NEDE-24011-P-A (the latest approved revision)*, General Electric Standard Application for Reactor Fuel (GESTAR II).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Administrator, Region 1, within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Administrator, Region 1, via the Licensee Event Report System within 30 days.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

SPECIAL REPORTS

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

*For Cycle 9, as evaluated in the Safety Evaluation dated March 9, 1999 to support License Amendment No. 117 .

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 117 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 August 25, 1998, as supplemented January 27, 1999, the Public Service Electric & Gas Company (PSE&G or the licensee) submitted a request for changes to the Hope Creek Generating Station (HCGS), Technical Specifications (TSs). The proposed amendment would revise TS 2.1.2, "THERMAL POWER, High Pressure and High Flow," and the Bases for TS 2.1, "Safety Limits." These changes are being made to implement appropriately conservative Safety Limit Minimum Critical Power Ratio (SLMCPR) values for the HCGS Cycle 9 core and fuel designs. An administrative change would also be made to TS 6.9.1.9 to reflect the proposed changes for Cycle 9. The January 27, 1999, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

As discussed in Section 3.1.2.2.1 of the HCGS Updated Final Safety Analysis Report and Section 4.4 of the Hope Creek Safety Evaluation Report (NUREG-1048), the thermal and hydraulic design of the reactor core conforms to the requirements of 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 10. This GDC states that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

The SLMCPR is a specified acceptable fuel design limit as defined by GDC 10. As described in the Bases for TS 2.1.2, the SLMCPR is set to ensure that no fuel damage is calculated to occur if the limit is not violated. The SLMCPR defines the minimum allowable critical power ratio at which 99.9% of the rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The SLMCPR is determined using a statistical model that combines all of the uncertainties in operating parameters and in the procedures used to calculate critical power. The SLMCPR is determined for each fuel design under the conditions specified by General Electric (GE) topical report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (GESTAR II).

As discussed in a 10 CFR Part 21 notification by GE dated May 24, 1996, it was reported that the GESTAR II generic fuel product line calculated SLMCPR may be non-conservative when applied to some actual core and fuel designs. As a result of the issues discussed in the 10 CFR Part 21 notification, PSE&G issued Licensee Event Report (LER) 96-014-00 dated May 14, 1996, as supplemented by LER 96-014-01, dated September 30, 1996, to document the issue for HCGS. In a letter to the Nuclear Regulatory Commission (NRC) dated October 9, 1996, GE committed to discontinue the practice of specifying the SLMCPR based on a bounding calculation for a given fuel product line (e.g., GE9B). Since that time, GE has calculated revised plant-specific SLMCPR values for HCGS Cycles 7, 8, and 9 as part of the reload licensing analyses. The proposed amendment provides the revised SLMCPR values based on the GE analyses for HCGS Cycle 9. The Cycle 9 SLMCPR values are based on a full core of 764 GE9B fuel assemblies, of which there are 196 fresh bundles, 236 once burnt bundles, 232 twice burnt bundles, 72 thrice burnt bundles, and 28 bundles burned 4 cycles.

3.0 EVALUATION

3.1 Evaluation of Proposed Changes to TS 2.1.2

The licensee has proposed to change the SLMCPR values in TS 2.1.2 for Cycle 9 from 1.10 to 1.09 for two recirculation loop operation and from 1.12 to 1.11 for single recirculation loop operation. These values are for reactor pressure greater than 785 psig and core flow greater than 10% of rated flow. In addition, the footnote to TS 2.1.2 would be changed to reflect that the values are applicable to Cycle 9 operation only.

The licensee described the methodology used to calculate the new SLMCPR values for the TSs in the submittal. The Cycle 9 SLMCPR analysis was performed by GE using the plant- and cycle-specific fuel and core parameters, NRC-approved methodologies including GESTAR II (NEDE-24011-P-A-11, Sections 1.1.5 and 1.2.5) and the proposed Amendment 25 to GESTAR II. The proposed Amendment 25 to GESTAR II provides cycle-specific analysis for the SLMCPR that replaces the former generic, bounding SLMCPR analysis.

The staff has reviewed the following: (1) the justification for the SLMCPR value of 1.09 for the two recirculation loop operation and 1.11 for single recirculation loop operation for Cycle 9, and (2) the relevant information provided in the proposed Amendment 25 to GESTAR II (NEDE-24011-P-A), which is currently under staff review (reference TAC No. M97491).

Based on our review, the staff has concluded that the Cycle 9 SLMCPR analysis for HCGS using the plant-specific calculation in conjunction with the approved method is acceptable. The Cycle 9 SLMCPR will ensure that 99.9% of the fuel rods in the core will not experience boiling transition which satisfies the requirements of GDC 10 of Appendix A to 10 CFR Part 50 regarding acceptable fuel design limits. Therefore, the staff has concluded that the justification for analyzing and determining the SLMCPR value of 1.09 for two recirculation loop operation and 1.11 for single recirculation loop operation for HCGS Cycle 9 operation is acceptable since approved methodologies were used. The proposed change to the footnote to TS 2.1.2 is also acceptable to reflect the applicability of the proposed TS change to the upcoming Cycle 9 operation for HCGS.

3.2 Evaluation of Proposed Changes to TS 6.9.1.9

The proposed change to TS 6.9.1.9 includes a note marked with an asterisk to identify that the analytical methods evaluated in this Safety Evaluation are approved for Cycle 9 for determining the cycle-specific parameters. The TSs currently state that the latest approved revision of NEDE-24011-P-A may be used. The staff has reviewed the methodology proposed in Amendment 25 to GESTAR II (NEDE-24011-P-A), which includes cycle-specific analysis for the SLMCPR. This analytical method for determining cycle-specific limits will ensure that applicable MCPR safety limits of the safety analysis for HCGS Cycle 9 operation are met. The methods specified in the latest approved version of GESTAR II, as supplemented by the relevant information in Amendment 25 to GESTAR II, is acceptable for use in determining the HCGS Cycle 9 SLMCPR values. Therefore, the proposed change to TS 6.9.1.9 is acceptable. However, the staff notes that this footnote will not be needed after Amendment 25 to GESTAR II is approved by the NRC.

3.3 Evaluation of Proposed Changes to Bases for TS 2.1

The proposed changes to the Bases are acceptable since they reflect the new 1.09 MCPR limit for two recirculation loop operation and the 1.11 MCPR limit for single recirculation loop operation based on approved methodologies.

3.4 Summary

Based on our review, the staff concludes that the above described TS changes are acceptable for HCGS Cycle 9 operation because the changes were analyzed based on the NRC-approved methods using HCGS cycle-specific inputs including the specific fuel bundle design for Cycle 9 operation.

4.0 <u>STATE CONSULTATION</u>

In accordance with the Commission's regulations, the New Jersey State Official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 <u>ENVIRONMENTAL CONSIDERATION</u>

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. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (63 FR 50938). The amendment also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: T. Huang
R. Ennis

Date: March 9, 1999