Template #NRR-058

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

May 1, 2000

SUBJECT: HOPE CREEK GENERATING STATION, ISSUANCE OF AMENDMENT, RE: SAFETY LIMIT MINIMUM CRITICAL POWER RATIO AND FUEL VENDOR CHANGE (TAC NO. MA6771)

Dear Mr. Keiser:

The Commission has issued the enclosed Amendment No. 126 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 September 30, 1999, as supplemented March 27, 2000.

This amendment revises the TSs associated with the Safety Limit Minimum Critical Power Ratios (SLMCPRs) in order to support the operation of HCGS in the upcoming Cycle 10 with a mixed core of General Electric (GE) and Asea Brown Bovieri/Combustion Engineering (ABB/CE) fuel. In addition, administrative changes have been made to the TSs to reflect the change in fuel vendor from GE to ABB/CE.

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

Sincerely,

/RA/

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

Docket No. 50-354

Enclosures: 1. Amendment No. 126 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

May 1, 2000

Mr. Harold W. Keiser
Chief Nuclear Officer & President -
Nuclear Business Unit
Public Service Electric & Gas
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Post Office Box 236
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RB Ennis

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

Docket No. 50-354

Enclosures:
1. Amendment No. 126 to License No. NPF-57
2. Safety Evaluation

cc w/encls: See next page

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. 126
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 September 30, 1999, as supplemented March 27, 2000, 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. 126 , 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 completion of Cycle 9.

FOR THE NUCLEAR REGULATORY COMMISSION

James W. Clifford

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 1, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 126

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following pages of the Appendix "A" Technical Specifications with the attached revised pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
iv	iv
xvii	xvii
xxv	xxv
2-1	2-1
B 2-1	B 2-1
B 2-2	B 2-2
B 2-3	B 2-3
3/4 2-1	3/4 2-1
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 4-1	3/4 4-1
3/4 4-2	3/4 4-2
3/4 4-2a	3/4 4-2a
3/4 4-2b	3/4 4-2b
B 3/4 1-3	B 3/4 1-3
B 3/4 1-5	B 3/4 1-5
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4	B 3/4 2-4
B 3/4 4-1	B 3/4 4-1
B 3/4 4-2	B 3/4 4-2
6-21	6-21
6-26	6-26

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

SECTION PAGE

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 With reactor steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow:

The MINIMUM CRITICAL POWER RATIO (MCPR) for GE fuel shall be ≥ 1.10 for two recirculation loop operation and shall be ≥ 1.12 for single recirculation loop operation. The MCPR for ABB/CE fuel shall be ≥ 1.10 for two recirculation loop operation and shall be ≥ 1.13 for one recirculation loop operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With reactor steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow and the MCPR below the values for the fuel stated in LCO 2.1.2, 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.

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 for GE fuel is ≥ 1.10 for two recirculation loop operation and ≥ 1.12 for single recirculation loop operation; and the MCPR for ABB/CE fuel is ≥ 1.10 for two recirculation loop operation and ≥ 1.13 for single recirculation loop operation. These MCPR values represent 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 correlations are 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 28×10^3 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 28×10^3 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.

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using a statistical model that combines all of the uncertainties in operating parameters and in the procedures used to calculate critical power. Calculation of the Safety Limit MCPR is defined in Reference 1 for GE fuel and Reference 2 for ABB/CE fuel.

Reference:

1. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A (The approved revision at the time the reload analyses are performed. The approved revision number shall be identified in the CORE OPERATING LIMITS REPORT.)

2. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactors Reload Fuel" (The approved revision at the time the reload analyses are performed. The approved revision number shall be identified in the CORE OPERATING LIMITS REPORT.)

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3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

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

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit specified in the CORE OPERATING LIMITS REPORT.

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

ACTION:

a. With the end-of-cycle recirculation pump trip system inoperable per Specification 3.3.4.2, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that, within 1 hour, MCPR is determined to be greater than or equal to the EOC-RPT inoperable limit specified in the CORE OPERATING LIMITS REPORT.

b. With MCPR less than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR, shall be determined to be equal to or greater than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

d. The provisions of Specification 4.0.4 are not applicable.

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3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation with:

a. Total core flow greater than or equal to 45% of rated core flow, or
b. THERMAL POWER less than or equal to the limit specified in Figure 3.4.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

1. Within 4 hours:
   a) Place the recirculation flow control system in the Local Manual mode, and
   b) Reduce THERMAL POWER to ≤ 70% of RATED THERMAL POWER, and
   c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit per Specification 2.1.2, and
   d) Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit to a value specified in the CORE OPERATING LIMITS REPORT for single loop operation, and
   e) DELETED.
   f) Limit the speed of the operating recirculation pump to less than or equal to 90% of rated pump speed, and
   g) Perform surveillance requirement 4.4.1.1.2 if THERMAL POWER is ≤ 38% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is ≤ 50% of rated loop flow.

2. Within 4 hours, reduce the Average Power Range Monitor (APRM) Scram Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specifications 2.2.1 and 3.2.2; otherwise, with the Trip Setpoints and Allowable Values associated with one trip system not reduced to those applicable for single recirculation loop operation, place the affected trip system in the tripped condition and within the following 6 hours, reduce the Trip Setpoints and Allowable Values of the affected channels to those applicable for single recirculation loop operation per Specifications 2.2.1 and 3.2.2.

3. Within 4 hours, reduce the APRM Control Rod Block Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specifications 3.2.2 and 3.3.6; otherwise, with the Trip Setpoint and Allowable Values associated with one trip function not reduced to those applicable for single recirculation loop operation, place at least one affected channel

* See Special Test Exception 3.10.4.

in the tripped condition and within the following 6 hours, reduce the Trip Setpoints and Allowable Values of the affected channels to those applicable for single recirculation loop operation per Specifications 3.2.2 and 3.3.6.

4. Within 4 hours, reduce the Rod Block Monitor Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specification 3.3.6; otherwise, with the Trip Setpoints and Allowable Values associated with one trip function not reduced to those applicable for single recirculation loop operation, place at least one affected channel in the tripped condition and within the following 6 hours, reduce the Trip setpoints and Allowable Values of the remaining channels to those applicable for single recirculation loop operation per Specification 3.3.6.

5. The provisions of Specification 3.0.4 are not applicable.

6. Otherwise be in at least HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant system recirculation loops in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours and initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

c. With one or two reactor coolant system recirculation loops in operation and total core flow less than 45% but greater than 40% of rated core flow and THERMAL POWER greater than the limit specified in Figure 3.4.1.1-1:

1. Determine the APRM and LPRM* noise levels (Surveillance 4.4.1.1.4):

   a) At least once per 8 hours, and
   b) Within 30 minutes after the completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER.

2. With the APRM or LPRM* neutron flux noise levels greater than three times their established baseline noise levels, within 15 minutes initiate corrective action to restore the noise levels to within the required limits within 2 hours by increasing core flow to greater than 45% of rated core flow or by reducing THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1.

d. With one or two reactor coolant system recirculation loops in operation and total core flow less than or equal to 40% and THERMAL POWER greater than the limit specified in Figure 3.4.1.1-1, within 15 minutes initiate corrective action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 or increase core flow to greater than 40% within 4 hours.

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* Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 With one reactor coolant system recirculation loop not in operation at least once per 12 hours verify that:

a. Reactor THERMAL POWER is ≤ 70% of RATED THERMAL POWER, and

b. The recirculation flow control system is in the Local Manual mode, and

c. The speed of the operating recirculation pump is less than or equal to 90% of rated pump speed, and

d. Core flow is greater than 40% when THERMAL POWER is greater than the limit specified in Figure 3.4.1.1-1.

4.4.1.1.2 With one reactor coolant system recirculation loop not in operation, within no more than 15 minutes prior to either THERMAL POWER increase or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is ≤ 38% of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is ≤ 50% of rated loop flow:

a. ≤ 145°F between reactor vessel steam space coolant and bottom head drain line coolant, and

b. ≤ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c. ≤ 50°F between the reactor coolant within the loop not in operation and the operating loop.

The differential temperature requirements or Specifications 4.4.1.1.2b and 4.4.1.1.2c do not apply when the loop not in operation is isolated from the reactor pressure vessel.

4.4.1.1.3 Each pump MG set scoop tube mechanical and electrical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 109% and 107%, respectively, of rated core flow, at least once per 18 months.

4.4.1.1.4 Establish a baseline APRM and LPRM* neutron flux noise value within the regions for which monitoring is required (Specification 3.4.1.1, ACTION c) within 2 hours of entering the region for which monitoring is required unless baselining has previously been performed in the region since the last refueling outage.

* Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

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3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 10% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RWM to be OPERABLE when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER provides adequate control.

The RWM provides automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.4.9 of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

rate, solution concentration or boron equivalent to meet the ATWS Rule must not invalidate the original system design basis. Paragraph (c)(4) of 10 CFR 50.62 states that:

> "Each boiling water reactor must have a Standby Liquid Control System (SLCS) with a minimum flow capacity and boron control equivalent in control capacity to 86 gallons per minute of 13 weight percent sodium pentaborate solution (natural boron enrichment)."

The described minimum system parameters (82.4 gpm, 13.6 percent concentration and natural boron equivalent) will ensure an equivalent injection capability that exceeds the ATWS Rule requirement. The stated minimum allowable pumping rate of 82.4 gallons per minute is met through the simultaneous operation of both pumps.

1. CENPD-284-P-A, "Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification," July, 1996.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications in this section help assure that the fuel can be operated safely and reliably during normal operation. In addition, the limits specified in these specifications help ensure that the fuel does not exceed specified safety and regulatory limits during anticipated operational occurrences and design basis accidents. Specifically, these limits:

1. Ensure that the limits specified in 10CFR50.46 are not exceeded following the postulated design basis loss of coolant accident.

2. Ensure reactor operations remains within licensed, analyzed power/flow limits.

3. Ensure that the MCPR Safety Limit is not violated following any anticipated operational occurrence.

4. Ensure fuel centerline temperatures remain below the melting temperature and peak cladding strain remains below 1% during steady state operation.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is a measure of the average Linear Heat Generation Rate (LHGR) of all the fuel rods in a fuel assembly at any axial location. The Technical Specification APLHGR is the LHGR of the highest-powered rod divided by its local peaking factor. Limits on the APLHGR are specified to ensure that the fuel design limits are not exceeded. The limiting value of the APLHGR limit is specified in the CORE OPERATING LIMITS REPORT. The calculation procedure used to establish the APLHGR is based on a loss-of-coolant accident analysis. The post LOCA peak cladding temperature (PCT) is primarily a function of the APLHGR and is dependent only secondarily on the rod to rod power distribution within an assembly. The analytical modes used in evaluating the postulated loss-of-coolant accidents are described in References 1 and 2. These models are consistent with the requirements of Appendix K to 10CFR50.

For plant operation with a single recirculation loop, a lower value for the APLHGR limit is specified in the CORE OPERATING LIMITS REPORT. This lower value accounts for an earlier transition from nucleate boiling which occurs following a loss-of-coolant accident in the single loop operation compared to two loop operation.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased simulated thermal power-upscale scram setting and the flow biased neutron flux-upscale control rod block trip setpoints must be adjusted to ensure that the MCPR does not become less than the fuel cladding Safety Limit or that > 1% plastic strain does not occur in the degraded situation. The scram setpoints and rod block setpoints are adjusted in accordance with the formula in Specification 3.2.2 whenever it is known that the existing power distribution would cause the design LHGR to be exceeded at RATED THERMAL POWER.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is obtained.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-3 that are input to an ABB/CE core dynamic behavior transient computer program. The codes used to evaluate transients are discussed in Reference 2.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$ respectively) to ensure adherence to fuel design limits during the worst transient with moderate frequency that is postulated in Chapter 15.

Flow dependent MCPR limits ($MCPR_f$) are determined by steady state methods using a three dimensional BWR simulator code (Reference 2). $MCPR_f$ curves are provided based on the maximum credible flow runout transient (i.e., runout of both loops).

A three dimensional BWR simulator code and a one dimensional transient code (Reference 2) determine power dependent MCPR limits ($MCPR_p$). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scram limits are bypassed, high and low $MCPR_p$ operating limits are provided for operation between 25% of RATED THERMAL POWER and the bypass power levels.

MINIMUM CRITICAL POWER RATIO (Continued)

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump-speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, November 1975.

2. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactors Reload Fuel" (The approved revision at the time the reload analyses are performed. The approved revision number shall be identified in the CORE OPERATING LIMITS REPORT.)

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3/4.4.1 RECIRCULATION SYSTEM

The impact of single recirculation loop operation upon plant safety is assessed and shows that single loop operation is permitted if the MCPR fuel cladding Safety Limit is increased as noted by Specification 2.1.2, APRM scram and control rod block setpoints are adjusted as noted in Tables 2.2.1-1 and 3.3.6-2 respectively. MAPLHGR limits are decreased by the factor given in Specification 3.2.1, and MCPR operating limits are adjusted per Specification 3/4.2.3.

Additionally, surveillance on the pump speed of the operating recirculation loop is imposed to exclude the possibility of excessive core internals vibration. The surveillance on differential temperatures below 38% THERMAL POWER or 50% rated recirculation loop flow is to mitigate the undue thermal stress on vessel nozzles, recirculating pump and vessel bottom head during the extended operation of the single recirculation loop mode.

An inoperable jet pump is not in itself a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core, thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation.

Recirculation loop flow mismatch limits are in compliance with the ECCS LOCA analysis design criteria for two recirculation loop operation. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. In the case where the mismatch limits cannot be maintained during two loop operation, continued operation is permitted in a single recirculation loop mode.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Sudden equalization of a temperature difference > 145°F between the reactor vessel bottom head coolant and the coolant in the upper region of the reactor vessel by increasing core flow rate would cause undue stress in the reactor vessel bottom head.

The objective of BWR plant and fuel design is to provide stable operation with margin over the normal operating domain. However, at the high power/low flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., rod pattern, power shape). To provide assurance that neutron flux limit cycle oscillations are detected and suppressed, APRM and LPRM neutron flux noise levels should be monitored while operating in this region.

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservation decay ratio of 0.6 was chosen as the bases for determining the generic region for surveillance to account for the plant to plant variability of decay ratio with core and fuel designs. This generic region has been determined to correspond to a core flow of less than or equal to 45% of rated core flow and a THERMAL POWER greater than that specified in Figure 3.4.1.1-1.

Plant specific calculations can be performed to determine an applicable region for monitoring neutron flux noise levels. In this case the degree of conservatism can be reduced since plant to plant variability would be eliminated. In this case, adequate margin will be assured by monitoring the region which has a decay ratio greater than or equal to 0.8.

Neutron flux noise limits are also established to ensure early detection of limit cycle neutron flux oscillations. BWR cores typically operate with neutron flux noise caused by random boiling and flow noise. Typical neutron flux noise levels of 1-12% of rated power (peak-to-peak) have been reported for the range of low to high recirculation loop flow during both single and dual recirculation loop operation. Neutron flux noise levels which significantly bound these values are considered in the thermal/mechanical design of BWR fuel and are found to be of negligible consequence. In addition, stability tests at operating BWRs have demonstrated that when stability related neutron flux limit cycle oscillations occur they result in peak-to-peak neutron flux limit cycles of 5-10 times the typical values. Therefore, actions taken to reduce neutron flux noise levels exceeding three (3) times the typical value are sufficient to ensure early detection of limit cycle neutron flux oscillations.

Typically, neutron flux noise levels show a gradual increase in absolute magnitude as core flow is increased (constant control rod pattern) with two reactor recirculation loops in operation. Therefore, the baseline neutron flux noise level obtained at a specific core flow can be applied over a range of core flows. To maintain a reasonable variation between the low flow and high flow end of the flow range, the range over which a specific baseline is applied should not exceed 20% of rated core flow with two recirculation loops in operation. Data from tests and operating plants indicate that a range of 20% of rated core flow will result in approximately a 50% increase in neutron flux noise level during operation with two recirculation loops. Baseline data should be taken near the maximum rod line at which the majority of operation will occur. However, baseline date taken at lower rod lines (i.e., lower power) will result in a conservative value since the neutron flux noise level is proportional to the power level at a given core flow.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operates to prevent the reactor coolant system from being pressurized above the Safety Limit of 1375 psig in accordance with the ASME Code. A total of 13 OPERABLE safety/relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case transient.

Demonstration of the safety relief valve lift settings occurs only during shutdown. The safety relief valve pilot stage assemblies are set pressure tested in accordance with the recommendations of General Electric SIL No. 196, Supplement 14 (April 23, 1984), "Target Rock 2-Stage SRV Set-Point Drift." Set pressure tests of the safety relief valve main (mechanical) stage are conducted at least once every 5 years.

The low-low set system ensures that safety/relief valve discharges are minimized for a second opening of these valves, following any overpressure transient. This is achieved by automatically lowering the closing setpoint of two valves and lowering the opening setpoint of two valves following the initial opening. In this way, the frequency and magnitude of the containment blowdown duty cycle is substantially reduced. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

CORE OPERATING LIMITS REPORT (Continued)

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC, as applicable in References 1 and 2.

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.

REFERENCES

1. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactors Reload Fuel," (latest approved revision)

2. NEDE-24011-P-A (latest approved revision), "General Electric Standard Application for Reactor Fuel (GESTAR-II)"

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 126 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 September 30, 1999, as supplemented March 27, 2000, 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 the TSs associated with the Safety Limit Minimum Critical Power Ratios (SLMCPRs) in order to support the operation of HCGS in the upcoming Cycle 10 with a mixed core of General Electric (GE) and Asea Brown Bovieri/Combustion Engineering (ABB/CE) fuel. The Cycle 10 core will consist of 764 fuel assemblies, of which there will be 232 fresh SVEA-96+ bundles, 196 once burned GE9B bundles, 236 twice burned GE9B bundles, and 100 thrice burned GE9B bundles.

In addition to the above described TS changes, administrative changes would be made to the TSs to reflect the change in fuel vendor from GE to ABB/CE. The March 27, 2000, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

2.1 Evaluation of Proposed Changes to TS 2.1.2

The licensee proposed TS 2.1.2 would read as follows:

THERMAL POWER, High Pressure and High Flow

2.1.2 With the reactor steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow:

The MINIMUM CRITICAL POWER RATIO (MCPR) for GE fuel shall be ≥ 1.10 for two recirculation loop operation and shall be ≥ 1.12 for single recirculation loop operation. The MCPR for ABB/CE fuel shall be ≥ 1.10 for two recirculation loop operation and shall be ≥ 1.13 for one recirculation loop operation.

APPLICABILITY : OPERATIONAL CONDITIONS 1 AND 2

ACTION:

With reactor steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow and the MCPR below the values for the fuel stated in LCO 2.1.2, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

The licensee's submittals described the analysis used to calculate the new SLMCPR values for Cycle 10 for a mixed core of GE GE9B and ABB/CE SVEA-96+ fuel. The Cycle 10 SLMCPR analysis was performed by ABB/CE using the plant-specific and cycle-specific fuel and core parameters and NRC-approved methodologies including CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," UR-89-210-P-A, "SVEA-96 Critical Power Experiments on Full Scale 24-rod Sub-Bundle", NEDO-10958-A (GETAB), and NEDE-24011-P-A-13-US. Based on our review of the submittals, the staff concluded that the SLMCPR analysis for HCGS Cycle 10 operation using the plant-specific and cycle-specific fuel and core parameters in conjunction with the approved methodologies is acceptable. The proposed Cycle 10 SLMCPR TS values have been established using methods that will ensure that 99.9 percent of the fuel rods in the core will not experience boiling transition which satisfies the requirements of Generic Design Criterion 10 of Appendix A to 10 CFR Part 50 regarding acceptable fuel design limits. Therefore, the staff finds that the proposed TS 2.1.2 changes to reflect SLMCPR values of 1.10 for both GE fuel and ABB/CE fuel for two recirculation loop operation, 1.12 for GE fuel for single recirculation loop operation, and 1.13 for ABB/CE fuel for single recirculation loop operation for HCGS Cycle 10 operation are acceptable.

The staff has also reviewed the proposed changes to the Bases for TS 2.0, TS 2.1.1 and TS 2.1.2 and found they appropriately reflect the new MCPR limits and change in fuel vendor.

2.2 Evaluation of Proposed Changes to TS 3.2.1

For TS 3.2.1, the licensee proposed to delete the second sentence, "The limits specified in the CORE OPERATING LIMITS REPORT shall be reduced to a value of 0.86 times the two recirculation loop operation limit when in single recirculation loop operation." This sentence explains how to obtain the average planar linear heat generation rates (APLHGRs) for single recirculation loop operation. The staff has reviewed this change and found it acceptable since the APLHGRs limits for each type of fuel are specified in the Core Operating Limits Report (COLR). The staff has also reviewed the associated proposed changes to the Bases for TS 3/4.2 and TS 3/4.2.1 and found they appropriately clarify the existing descriptions and are administrative in nature.

2.3 Evaluation of Proposed Changes to TS 3.2.3

For TS 3.2.3, the licensee proposed to delete the formula for "τ" associated with the GE GEMINI/ODYN methodology for operating limit MCPR calculation. The staff has reviewed this change and finds it acceptable since the fuel vendor has changed from GE to ABB/CE.

2.4 Evaluation of Proposed Changes to TS 4.2.3

For TS 4.2.3, the licensee proposed to delete text associated with the value of "τ" consistent with the changes to TS 3.2.3. The staff has reviewed this change and finds it acceptable since the new fuel vendor is no longer using the GE methodology. The staff has also reviewed the associated proposed changes to the Bases for TS 3/4.2.3 and found they appropriately reflect the new approved methodology used for the new ABB/CE fuel.

2.5 Evaluation of Proposed Changes to TS 3.4.1.1

The current wording for TS 3.4.1.1 Action Statement a.1.d reads as follows:

> d) Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit to a value of 0.86 times the two recirculation loop limit per Specification 3.2.1, and

The licensee proposed to revise TS 3.4.1.1 Action Statement a.1.d to read as follows:

> d) Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit to a value specified in the CORE OPERATING LIMITS REPORT for single loop operation, and

The staff has reviewed and found this change acceptable because the MAPLHGR limit is specified in the COLR. The staff has also reviewed the associated proposed changes to the Bases for TS 3/4.4.1 and found they appropriately reflect the new methodology used for the new fuel.

2.6 Evaluation of Proposed Changes to TS 6.9.1.9

The licensee has proposed to revise TS 6.9.1.9 to reflect that ABB/CE methodology CENPD-300-P-A is being used in addition to the existing GE methodology to determine the core operating limits. The staff has reviewed the proposed changes and found them acceptable because the addition of an approved methodology to the COLR specification is administrative in nature.

2.7 Evaluation of Proposed Changes to Bases for TS 3/4.1.4

The licensee has proposed to revise the Bases for TS 3/4.1.4 to replace references to GE topical reports with ABB/CE Topical Report CENPD-284-P-A, "Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification." This change is required to reflect the NRC-approved ABB/CE methodology due to the change in fuel vendor from GE to ABB/CE. The staff has reviewed the proposed change and finds it acceptable since it is administrative in nature.

2.8 Summary

Based on our review, the staff concludes that the above described TS changes are acceptable for HCGS Cycle 10 operation because the changes: (1) were analyzed based on the NRC-approved methods using HCGS cycle-specific inputs and the fuel bundle design for Cycle 10 operation or; (2) are administrative in nature.

3.0 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 had no comments.

4.0 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. 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 (64 FR 59805). 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.

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.

Principal Contributor: T. Huang

Date: May 1, 2000

Mr. Harold W. Keiser
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, ISSUANCE OF AMENDMENT, RE: SAFETY LIMIT MINIMUM CRITICAL POWER RATIO AND FUEL VENDOR CHANGE (TAC NO. MA6771)

Dear Mr. Keiser:

The Commission has issued the enclosed Amendment No. 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 September 30, 1999, as supplemented March 27, 2000.

This amendment revises the TSs associated with the Safety Limit Minimum Critical Power Ratios (SLMCPRs) in order to support the operation of HCGS in the upcoming Cycle 10 with a mixed core of General Electric (GE) and Asea Brown Bovieri/Combustion Engineering (ABB/CE) fuel. In addition, administrative changes have been made to the TSs to reflect the change in fuel vendor from GE to ABB/CE.

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

Sincerely,

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

Docket No. 50-354

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

cc w/encls: See next page

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