Docket No. 50-354

January 25, 1994

Mr. Steven E. Miltenberger
Vice President and Chief Nuclear
Officer
Public Service Electric & Gas
Company
Post Office Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. Miltenberger:

SUBJECT: SINGLE LOOP OPERATION INSTRUMENTATION, HOPE CREEK GENERATING STATION (TAC NO. M85771)

The Commission has issued the enclosed Amendment No. 63 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated February 2, 1993, and supplemented by letter dated November 16, 1993.

This amendment extends the period of time to reduce the setpoints of the Average Power Range Monitors and the Rod Block Monitor when the plant enters single-loop operations. Additionally, the change incorporates updated core values relative to single-loop operations and the addition of a new Specification 3.0.5 and its associated Bases.

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

You are requested to inform the NRC, in writing, when this amendment has been implemented.

Sincerely,
/s/
James C. Stone, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 63 to License No. NPF-57
2. Safety Evaluation

cc w/enclosures:
See next page

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DISTRIBUTION

Docket File	CMiller	GHill(2),P1-22	OC/LFDCB
NRC & Local PDRs	MO'Brien(2)	JDonahue	EWenzinger, RGN-I
PDI-2 Reading	JStone/JZimmerman	CGrimes, 11E-21	JWhite, RGN-I
SVarga	OGC	ACRS(10)	RJones
JCalvo	DHagan, 3206	OPA	

#94-004

OFC	:PDI-2/LA	:PDI-2/PE	:PDI-2/PM	:SRXB/BC	:OTSB	:OGC	:PDI-2/D :
NAME	:MO'Brien	:JZimmerman	:JStone	:RJones	:CGrimes	:Cukahen	:CMiller :
DATE	:1/ /93 94	:1/4/94	:1/ /94	:1/ /94	:1/4/94	:1/5/94	:1/ /93:

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#94-004

OFC	:PDI-2/LA	:PDI-2/PE	:PDI-2/PM	:SRXB/BC	:OTSB	:OGC	:PDI-2/D
NAME	:MO'Brien	:JZimmerman	:JStone	:RJones	:CGrimes	:Cutchin	:CMiller
DATE	:1/ /93 94	:1/4/94	:1/6/94	:1/4/94	:1/4/94	:1/5/94	:1/21/93

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

Hope Creek Generating Station

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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. 63
License No. NPF-57

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

   A. The application for amendment filed by the Public Service Electric & Gas Company (PSE&G) dated February 2, 1993 and supplemented by letter dated November 16, 1993, 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. 63 , 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 of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

Charles L. Miller, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 25, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. ___

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Overleaf pages provided to maintain document completeness.*

Remove	Insert
2-3	2-3*
2-4	2-4
3/4 0-1	3/4 0-1
3/4 0-2	3/4 0-2*
3/4 2-1	3/4 2-1*
3/4 2-2	3/4 2-2
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 0-3	B 3/4 0-3*
B 3/4 0-4	B 3/4 0-3a
-	B 3/4 0-4
-	-
B 3/4 4-1	B 3/4 4-1
B 3/4 4-2	B 3/4 4-2*

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-Upscale		
1) Flow Biased	≤ 0.66(w-Δw)+51%** with a maximum of	≤ 0.66(w-Δw)+54%** with a maximum of
2) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux-Upscale	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1037 psig	≤ 1057 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed

*See Bases Figure B 3/4 3-1.

**The Average Power Range Monitor Scram function varies as a function of recirculation loop drive flow (w). Δw is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow. Δw = 0 for two recirculation loop operation. Δw = 9% for single recirculation loop operation.

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL CONDITIONS or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the Action requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in an OPERATIONAL CONDITION in which the Specification does not apply by placing it, as applicable, in:

1. At least STARTUP within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

This Specification is not applicable in OPERATIONAL CONDITIONS 4 or 5.

3.0.4 Entry into an OPERATIONAL CONDITION or other specified condition shall not be made when the conditions for the Limiting Condition for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL CONDITION or other specified condition may be made in accordance with the ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

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, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, 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

3/4.2 POWER DISTRIBUTION LIMITS

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. 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.

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.

POWER DISTRIBUTION LIMITS

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-upscale scram trip setpoint (S) and flow biased neutron flux-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

TRIP SETPOINT	ALLOWABLE VALUE
$S \leq (0.66(w-\Delta w)^{**} + 51\%)T$	$S \leq (0.66(w-\Delta w)^{**} + 54\%)T$
$S_{RB} \leq (0.66(w-\Delta w)^{**} + 42\%)T$	$S_{RB} \leq (0.66(w-\Delta w)^{**} + 45\%)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER,

W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 100 million lbs/hr.

T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER (FRTP) divided by the CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD). T is applied only if less than or equal to 1.0.

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

ACTION:

With the APRM flow biased simulated thermal power-upscale scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint values* within 6 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP and the CMFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

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 CMFLPD greater than or equal to FRTP.
d. The provisions of Specification 4.0.4 are not applicable.

---

* With CMFLPD greater than the FRTP, rather than adjusting the APRM setpoints, the APRM may be adjusted such that the APRM readings are greater than or equal to 100% times CMFLPD provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

**The Average Power Range Monitor Scram function varies as a function of recirculation loop drive flow (w). Δw is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow. $\Delta w = 0$ for two recirculation loop operation. $\Delta w = 9\%$ for single recirculation loop operation.

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 by 0.01 to 1.08 per Specification 2.1.2, and
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
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

*See Special Test Exception 3.10.4.

ACTION (Continued)

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 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

* 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.

ACTION (Continued)

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.

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.

SURVEILLANCE REQUIREMENTS (Continued)

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.

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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.

a lower CONDITION of operation is reached in less time than allowed, the total allowable time to reach COLD SHUTDOWN, or other OPERATIONAL CONDITION, is not reduced. For example, if STARTUP is reached in 2 hours, the time allowed to reach HOT SHUTDOWN is the next 11 hours because the total time to reach HOT SHUTDOWN is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to POWER operation, a penalty is not incurred by having to reach a lower CONDITION of operation in less than the total time allowed.

The same principle applies with regard to the allowable outage time limits of the ACTION requirements, if compliance with the ACTION requirements for one specification results in entry into an OPERATIONAL CONDITION or condition of operation for another specification in which the requirements of the Limiting Condition for Operation are not met. If the new specification becomes applicable in less time than specified, the difference may be added to the allowable outage time limits of the second specification. However, the allowable outage time limits of ACTION requirements for a higher CONDITION of operation may not be used to extend the allowable outage time that is applicable when a Limiting Condition for Operation is not met in a lower CONDITION of operation.

The shutdown requirements of Specification 3.0.3 do not apply in CONDITIONS 4 and 5, because the ACTION requirements of individual specifications define the remedial measures to be taken.

Specification 3.0.4 establishes limitations on a change in OPERATIONAL CONDITIONS when a Limiting Condition for Operation is not met. It precludes placing the facility in a higher CONDITION of operation when the requirements for a Limiting Condition for Operation are not met and continued noncompliance to these conditions would result in a shutdown to comply with the ACTION requirements if a change in CONDITIONS were permitted. The purpose of this specification is to ensure that facility operation is not initiated or that higher CONDITIONS of operation are not entered when corrective action is being taken to obtain compliance with a specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with ACTION requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a change in OPERATIONAL CONDITIONS. Therefore, in this case, entry into an OPERATIONAL CONDITION or other specified condition may be made in accordance with the provisions of the ACTION requirements. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 3.0.4 do not apply because they would delay placing the facility in a lower CONDITION of operation.

Specification 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of testing required to restore and demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the testing required to restore and demonstrate the OPERABILITY of the equipment. This Specification does not provide time to perform any other preventative or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the testing required to restore and demonstrate OPERABILITY.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of testing required to restore OPERABILITY of another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of testing required to restore and demonstrate the OPERABILITY on another channel in the same trip system.

LCO 3.0.5 is applicable to all Technical Specifications; however, the intent of LCO 3.0.5 is not to supersede more specific guidance contained within any individual specification.

Specifications 4.0.1 through 4.0.5 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establishes the requirement that surveillances must be performed during the OPERATIONAL CONDITIONS or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in an OPERATIONAL CONDITION or other specified condition for which the individual Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL CONDITION for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.

Specification 4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Specification 4.0.3 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to

3/4.4 REACTOR COOLANT SYSTEM

BASES

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 GE 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 GE 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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 63 TO FACILITY OPERATING LICENSE NO. NPF-57

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated February 2, 1993, as supplemented by letter dated November 16, 1993, the Public Service Electric & Gas Company (the licensee) submitted a request for changes to the Hope Creek Generating Station, Technical Specification (TS). The proposed changes extend the period of time permitted to reduce the average power range monitor (APRM) and rod block monitor (RBM) setpoints when the plant enters single loop operations (SLO). The November 16, 1993, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

Other changes incorporate core values determined since startup testing: (1) flow differential for SLO, (2) the minimum thermal power corresponding to no thermal stratification, and (3) core flow with both recirculation pumps at minimum speed.

Long-term single recirculation loop operation at Hope Creek Generating Station was approved by Amendment 3 (See Reference 2), so these changes are updates to a permitted mode of operation.

2.0 EVALUATION

2.1 Single Loop Operations

Technical Specification 3.4.1.1, Action (a)(1)(e) currently requires reduction of the APRM scram and rod block setpoints to their single loop values within 4 hours of entering single loop operation. Otherwise the plant must be in hot shutdown in 12 hours. The proposed change still allows 4 hours to reduce the APRM scram trip setpoints, the APRM control rod block setpoints, and the RBM setpoints to the applicable single loop values. However, this change requires that after the initial 4 hour period, instruments be placed in the tripped condition if their setpoints have not been reduced. Six additional hours are then allowed to adjust the affected channels and place them in operation. Essentially, the change permits an additional 6 hours to effect the setpoint reductions after entering single loop operation.

The proposed change separately addresses the APRM scram, APRM control rod block, and RBM trip requirements. There is a slight difference in the actions prescribed following the initial 4-hour period for these instruments. The APRM scram actions require that the affected trip system be put in the tripped condition if setpoints are not reduced after the first 4 hours. The APRM rod block and RBM sections specify that at least one affected channel be placed in the tripped condition if the 4-hour period is exceeded.

Chapter 7 of the Hope Creek Final Safety Analysis Report (FSAR) describes the reactor protection system (RPS) and the rod block portion of the rod control system. The RPS consists of two trip systems. Each receives inputs from two channels, with APRM instrumentation assigned to each of these trip channels. Putting one trip system in a tripped condition effectively reduces the RPS coincidence requirement to one-out-of-one from two-out-of-two. This ensures that a channel with the adjusted setpoints is providing protection. The APRM rod block and RBM action places at least one of the affected channels in the tripped condition, thereby imposing a rod block since any tripped condition input to the rod motion inhibit logic prevents rod movement.

Therefore, by requiring the setpoints of at least one trip system or function to be adjusted to the single loop values, a scram or rod block will be imposed if safety setpoints are reached.

Certain instrumentation guidelines require automatic changes to more restrictive setpoints if operation under conditions requiring them is a likely planned mode of operation (see Paragraph 4.15 of IEEE Standard 279-1971, Reference 3). However, the staff position for boiling water reactors is that automatic transfer is not required to ensure a safe and expedient implementation of the proper setpoints. The basis of this staff position allowing manual setpoint switching requires administrative controls to ensure that the more restrictive setpoints are activated in the time specified by the TS.

The proposed change seeks to extend the setpoint switching period by 6 hours. However, it does so by ensuring that at least one protective channel is available with the reduced setpoints after the initial 4 hour period. Channels not adjusted after the initial period are placed in the tripped condition, thereby ensuring that plant protection is afforded by the more restrictive setpoints.

Thus, plant safety is not adversely affected by the proposed change since more restrictive setpoints are in effect in the time currently required. Positive means of implementing the lower setpoints are still provided by the proposed TS requirements. As presented, the changes to provide additional time to reduce protective system setpoints for single recirculation loop operation are acceptable.

2.2 Power Ascension Program Data

The values in the TS for delta-w for single-loop operation, minimum thermal power without temperature stratification, and core flow with both recirculation pumps at minimum speed are initial values established prior to startup testing. The proposed updates seek to incorporate final values.

a. Delta-w for single-loop operation

Delta-w is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two-loop and single-loop operation. A conservative value has been determined based upon calculations following General Electric's (GE) guidelines using plant test data. This value is acceptable.

b. Minimum power corresponding to conditions in which no temperature stratification occurs

Power ascension testing verified power and flow conditions during single-loop operation at which no temperature stratification occurred. Although the test data did not establish actual minimum power and flow values for the onset of stratification, the data demonstrated that the surveillances required to check for stratification are not required above the proposed TS values of 38 percent thermal power and 50 percent of rated core flow. These proposed updated values of single loop power and flow limits of thermal stratification are conservative relative to actual plant test data and are acceptable.

c. Core flow with both recirculation pumps at minimum speed

TS Figure 3.4.1.1-1 is the power to flow limit used to specify actions needed to detect or avoid limit cycle neutron flux oscillations. TS 3.4.1.1 contains Actions and Surveillance Requirements (SR) dealing with operation in the power to flow region attributed to these oscillations. The current TS stipulates neutron monitoring noise level surveillances to be performed between core flow at minimum recirculation pump speed up to the maximum flow defining the limit region. Corrective action to avoid oscillations is taken when flow is equal to or below that value. The proposed change revises this threshold from 39 percent to 40 percent of rated core flow, consistent with NRC Bulletin 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors" (Reference 4). As this is the value specified in the Bulletin for the required actions, and the change to a higher flow is more conservative, the change is acceptable.

2.3 <u>Technical Specification 3.0.5</u>

The licensee proposed the addition of Limiting Condition for Operation (LCO) 3.0.5 and the associated bases of NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4," to the Hope Creek Generating Station TSs. The licensee requested this change in order to clarify the intent of LCO 3.0.2 to prevent misinterpretation of the LCO and associated Bases. The new LCO states, "Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its