December 2, 1993

Docket No. 50-354

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

Dear Mr. Miltenberger:

SUBJECT: REVISE SURVEILLANCE TEST INTERVALS AND ALLOWED OUT-OF-SERVICE TIMES FOR EMERGENCY CORE COOLING SYSTEM AND REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION, HOPE CREEK GENERATING STATION (TAC NO. M86104)

The Commission has issued the enclosed Amendment No. 62 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 April 1, 1993 and supplemented on July 2, 1993.

This amendment revises technical specifications surveillance requirements to extend the surveillance test intervals and allowed out-of-service times for the emergency core cooling system and reactor core isolation cooling system actuation instrumentation.

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

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Enclosures:
1. Amendment No. 62 to License No. NPF-57
2. Safety Evaluation

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P PDR

cc w/enclosures:
See next page

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

December 27, 1993

Docket No. 50-354

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

Dear Mr. Miltenberger:

SUBJECT: REVISE SURVEILLANCE TEST INTERVALS AND ALLOWED OUT-OF-SERVICE TIMES FOR EMERGENCY CORE COOLING SYSTEM AND REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION, HOPE CREEK GENERATING STATION (TAC NO. M86104)

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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,

FOR 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. 62 to
License No. NPF-57
2. Safety Evaluation

cc w/enclosures:
See next page

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

Hope Creek Generating Station

cc:

M. J. Wetterhahn, Esquire
Winston & Strawn
1400 L Street, N.W.
Washington, DC 20005-3502

R. Fryling, Jr., Esquire
Law Department - Tower 5E
80 Park Place
Newark, New Jersey 07101

Hope Creek Resident Inspector
U.S. Nuclear Regulatory Commission
Drawer I
Hancocks Bridge, New Jersey 08038

Mr. J. Hagan
Vice President - Nuclear Operations
Nuclear Department
P.O. Box 236
Hancocks Bridge, New Jersey 08038

Mr. R. Hovey
General Manager - Hope Creek Operations
Hope Creek Generating Station
P.O. Box 236
Hancocks Bridge, New Jersey 08038

Mr. Frank X. Thomson, Jr., Manager
Licensing and Regulation
Nuclear Department
P.O. Box 236
Hancocks Bridge, New Jersey 08038

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Dr. Jill Lipoti, Asst. Director
Radiation Protection Programs
NJ Department of Environmental
Protection and Energy
CN 415
Trenton, New Jersey 08625-0415

Mr. J. A. Isabella
MGR. - Generation Department
Atlantic Electric Company
Post Office Box 1500
1199 Black Horse Pike
Pleasantville, New Jersey 08232

Richard Hartung
Electric Service Evaluation
Board of Regulatory Commissioners
2 Gateway Center, Tenth Floor
Newark, NJ 07102

Lower Alloways Creek Township
c/o Mary O. Henderson, Clerk
Municipal Building, P.O. Box 157
Hancocks Bridge, NJ 08038

Mr. S. LaBruna
Vice President - Nuclear Engineering
Nuclear Department
P.O. Box 236
Hancocks Bridge, New Jersey 08038

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. 62
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 April 1, 1993, and supplemented on July 2, 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. 62 , 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

For 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: December 27, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 62

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 areas of change. Overleaf pages provided to maintain document completeness.*

Remove	Insert
3/4 3-33	3/4 3-33*
3/4 3-34	3/4 3-34
3/4 3-35	3/4 3-35
3/4 3-36	3/4 3-36*
3/4 3-39	3/4 3-39
3/4 3-40	3/4 3-40
3/4 3-51	3/4 3-51*
3/4 3-52	3/4 3-52
3/4 3-53	3/4 3-53
3/4 3-54	3/4 3-54*
3/4 3-55	3/4 3-55
3/4 3-56	3/4 3-56*
B 3/4 3-1	B 3/4 3-1*
B 3/4 3-2	B 3/4 3-2
B 3/4 3-3	B 3/4 3-3*
B 3/4 3-4	B 3/4 3-4

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. CORE SPRAY SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(b)(e)	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2(b)(e)	1, 2, 3	30
c. Reactor Vessel Pressure - Low (Permissive)	4/division(f)	1, 2, 3 4*, 5*	31 32
d. Core Spray Pump Discharge Flow - Low (Bypass)	1/subsystem	1, 2, 3, 4*, 5*	37
e. Core Spray Pump Start Time Delay - Normal Power	1/subsystem	1, 2, 3, 4*, 5*	31
f. Core Spray Pump Start Time Delay - Emergency Power	1/subsystem	1, 2, 3, 4*, 5*	31
g. Manual Initiation	1/division(b)(g)	1, 2, 3, 4*, 5*	33
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2/valve	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2/valve	1, 2, 3	30
c. Reactor Vessel Pressure - Low (Permissive)	1/valve	1, 2, 3 4*, 5*	31 32
d. LPCI Pump Discharge Flow - Low (Bypass)	1/pump	1, 2, 3, 4*, 5*	37
e. LPCI Pump Start Time Delay - Normal Power	1/pump(i)	1, 2, 3, 4*, 5*	31
f. Manual Initiation	1/subsystem	1, 2, 3, 4*, 5*	33
3. HIGH PRESSURE COOLANT INJECTION SYSTEM#			
a. Reactor Vessel Water Level - Low Low Level 2	4	1, 2, 3	34
b. Drywell Pressure - High	4	1, 2, 3	34
c. Condensate Storage Tank Level - Low	2(c)	1, 2, 3	35
d. Suppression Pool Water Level - High	2(c)	1, 2, 3	35
e. Reactor Vessel Water Level - High, Level 8	4(d)	1, 2, 3	31
f. HPCI Pump Discharge Flow - Low (Bypass)	1	1, 2, 3	37
g. Manual Initiation	1/system	1, 2, 3	33
4. AUTOMATIC DEPRESSURIZATION SYSTEM##			
a. Reactor Vessel Water Level - Low Low Low, Level 1	4	1, 2, 3	30
b. Drywell Pressure - High	4	1, 2, 3	30
c. ADS Timer	2	1, 2, 3	31
d. Core Spray Pump Discharge Pressure - High (Permissive)	1/pump	1, 2, 3	31

TABLE 3.3.3-1 (Cont'd)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION		MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
4.	AUTOMATIC DEPRESSURIZATION SYSTEM##			
	e. RHR LPCI Mode Pump Discharge Pressure - High (Permissive)	2/pump	1, 2, 3	31
	f. Reactor Vessel Water Level - Low, Level 3 (Permissive)	2	1, 2, 3	31
	g. ADS Drywell Pressure Bypass Timer	4	1, 2, 3	31
	h. ADS Manual Inhibit Switch	2	1, 2, 3	31
	i. Manual Initiation	4	1, 2, 3	33

		TOTAL NO. OF CHANNELS(h)	CHANNELS TO TRIP(h)	MINIMUM CHANNELS OPERABLE(h)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
5.	LOSS OF POWER					
	1. 4.16 kv Emergency Bus Under-voltage (Loss of Voltage)	4/bus	2/bus	3/bus	1, 2, 3, 4**, 5**	36
	2. 4.16 kv Emergency Bus Under-voltage (Degraded Voltage)	2/source/ bus	2/source/ bus	2/source/ bus	1, 2, 3, 4**, 5**	36

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) Also actuates the associated emergency diesel generators.

(c) One trip system. Provides signal to HPCI pump suction valve only.

(d) Provides a signal to trip HPCI pump turbine only.

(e) In divisions 1 and 2, the two sensors are associated with each pump and valve combination. In divisions 3 and 4, the two sensors are associated with each pump only.

(f) Division 1 and 2 only.

(g) In divisions 1 and 2, manual initiation is associated with each pump and valve combination; in divisions 3 and 4, manual initiation is associated with each pump only.

(h) Each voltage detector is a channel.

(i) Start time delay is applicable to LPCI Pump C and D only.

* When the system is required to be OPERABLE per Specification 3.5.2.

** Required when ESF equipment is required to be OPERABLE.

Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or declare the associated system inoperable.

b. With more than one channel inoperable, declare the associated system inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare the associated ECCS inoperable within 24 hours.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within 24 hours.

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 24 hours or declare the associated ECCS inoperable.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. For one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or declare the HPCI system inoperable.

b. With more than one channel inoperable, declare the HPCI system inoperable.

ACTION 35 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within 24 hours or declare the HPCI system inoperable.

ACTION 36 - With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the tripped condition within 1 hour; operation may then continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACITON 37 - With the number of OPERABLE channels less than required by the Minimum OPERABLE channels per Trip Function requirement, open the minimum flow bypass valve within one hour. Restore the inoperable channel to OPERABLE status within 7 days or declare the associated ECCS inoperable.

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. CORE SPRAY SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Reactor Vessel Pressure - Low	461 psig	≤ 481 psig and ≥ 441 psig
d. Core Spray Pump Discharge Flow - Low (Bypass)	≥ 775 gpm	≥ 650 gpm
e. Core Spray Pump Start Time Delay - Normal Power	10 seconds	≥ 9 seconds and ≤ 11 seconds
f. Core Spray Pump Start Time Delay - Emergency Power	6 seconds	≥ 5 seconds and ≤ 7 seconds
g. Manual Initiation	NA	NA
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Reactor Vessel Pressure - Low (Permissive)	450 psig	≤ 460 psig and ≥ 440 psig
d. LPCI Pump Discharge Flow - Low (Bypass)	≥ 1250 gpm	≥ 1100 gpm
e. LPCI Pump Start Time Delay - Normal Power	5 seconds	≥ 4 seconds and ≤ 6 seconds
f. Manual Initiation	NA	NA
3. HIGH PRESSURE COOLANT INJECTION SYSTEM		
a. Reactor Vessel Water Level - (Low Low, Level 2)	≥ -38 inches*	≥ -45 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Condensate Storage Tank Level - Low	≥ 22,558 gallons	≥ 19,174 gallons
d. Suppression Pool Water Level - High	≤ 78.5 inches	≤ 80.3 inches
e. Reactor Vessel Water Level - High, Level 8	≤ 54 inches	≤ 61 inches
f. HPCI Pump Discharge Flow - Low (Bypass)	≥ 550 gpm	≥ 500 gpm
g. Manual Initiation	NA	NA

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. CORE SPRAY SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	Q	R	1, 2, 3
c. Reactor Vessel Pressure - Low	S	Q	R	1, 2, 3, 4*, 5*
d. Core Spray Pump Discharge Flow - Low (Bypass)	S	Q	R	1, 2, 3, 4*, 5*
e. Core Spray Pump Start Time Delay - Normal Power	NA	Q	R	1, 2, 3, 4*, 5*
f. Core Spray Pump Start Time Delay - Emergency Power	NA	Q	R	1, 2, 3, 4*, 5*
g. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	Q	R	1, 2, 3
c. Reactor Vessel Pressure - Low (Permissive)	S	Q	R	1, 2, 3, 4*, 5*
d. LPCI Pump Discharge Flow - Low (Bypass)	S	Q	R	1, 2, 3, 4*, 5*
e. LPCI Pump Start Time Delay - Normal Power	NA	Q	R	1, 2, 3, 4*, 5*
f. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
3. HIGH PRESSURE COOLANT INJECTION SYSTEM#				
a. Reactor Vessel Water Level - Low Low, Level 2	S	Q	R	1, 2, 3
b. Drywell Pressure - High	S	Q	R	1, 2, 3
c. Condensate Storage Tank Level - Low	S	Q	R	1, 2, 3
d. Suppression Pool Water Level - High	S	Q	R	1, 2, 3
e. Reactor Vessel Water Level - High, Level 8	S	Q	R	1, 2, 3
f. HPCI Pump Discharge FLow - Low (Bypass)	S	Q	R	1, 2, 3
g. Manual Initiation	NA	R	NA	1, 2, 3

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
4. AUTOMATIC DEPRESSURIZATION SYSTEM##				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	Q	R	1, 2, 3
b. Drywell Pressure - High	S	Q	R	1, 2, 3
c. ADS Timer	NA	Q	Q	1, 2, 3
d. Core Spray Pump Discharge Pressure - High	S	Q	R	1, 2, 3
e. RHR LPCI Mode Pump Discharge Pressure - High	S	Q	R	1, 2, 3
f. Reactor Vessel Water Level - Low, Level 3	S	Q	R	1, 2, 3
g. ADS Drywell Pressure Bypass Timer	NA	Q	Q	1, 2, 3
h. ADS Manual Inhibit Switch	NA	R	NA	1, 2, 3
i. Manual Initiation	NA	R	NA	1, 2, 3
5. LOSS OF POWER				
a. 4.16 kv Emergency Bus Under-voltage (Loss of Voltage)	NA	NA	R	1, 2, 3, 4**, 5**
b. 4.16 kv Emergency Bus Under-voltage (Degraded Voltage)	S	M	R	1, 2, 3, 4**, 5**

* When the system is required to be OPERABLE per Specification 3.5.2.
** Required OPERABLE when ESF equipment is required to be OPERABLE.
# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.
Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5 The reactor core isolation cooling (RCIC) system actuation instrumentation channels shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.5-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more RCIC system actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.5-1.

SURVEILLANCE REQUIREMENTS

4.3.5.1 Each RCIC system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.5.1-1.

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

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	ACTION
a. Reactor Vessel Water Level - Low Low, Level 2	4(b)	50
b. Reactor Vessel Water Level - High, Level 8	4(b)	50
~~c. Condensate Storage Tank Water Level - Low(e)~~	~~2(c)~~	~~51~~
d. Manual Initiation	1(d)	52

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided all other channels monitoring that parameter are OPERABLE.
(b) One trip system with one-out-of-two twice logic.
(c) One trip system with one-out-of-two logic.
(d) One trip system with one channel.
(e) Initiates RCIC suction switchover from the condensate storage tank to the torus.

TABLE 3.3.5-1 (Continued)

REACTOR CORE ISOLATION COOLING SYSTEM

ACTUATION INSTRUMENTATION

ACTION 50 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within 24 hours or declare the RCIC system inoperable.

b. With more than one channel inoperable, declare the RCIC system inoperable.

ACTION 51 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within 24 hours or declare the RCIC system inoperable.

ACTION 52 - With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 24 hours or declare the RCIC system inoperable.

TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -38 inches*	≥ -45 inches
b. Reactor Vessel Water Level - High, Level 8	≤ 54 inches*	≤ 61 inches
c. Condensate Storage Tank Level - Low	≥ 22,558 gallons	≥ 19,174 gallons
d. Manual Initiation	NA	NA

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

TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level - Low Low, Level 2	S	Q	R
b. Reactor Vessel Water Level - High, Level 8	S	Q	R
c. Condensate Storage Tank Level - Low	NA	Q	R
d. Manual Initiation	NA	Q(a)	NA

(a) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 92 days as part of circuitry required to be tested for automatic system actuation.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1. The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 2 from OPERATIONAL CONDITION 1 for the Source Range Monitors or the Intermediate Range Monitors.

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279 for nuclear power plant protection systems. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P, "Technical Specification Improvement Analyses for BWR Reactor Protection System," as approved by the NRC and documented in the SER (letter to T. A. Pickens from A. Thadani dated July 15, 1987). The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the safety analyses. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 10 second diesel startup. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 13 second delay. It follows that checking the valve speeds and the 13 second time for emergency power establishment will establish the response time for the isolation functions.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30936P-A, "BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)," Parts 1 and 2. The safety evaluation reports documenting NRC approval of NEDC-30936P-A are contained in letters to D. N. Grace from A. C. Thadani (Part 1) and C. E. Rossi (Part 2) dated December 9, 1988. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971, NEDO-24222, dated December 1979, and Section 15.8 of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., $\leq$ 175 ms. Included in this time are: the response time of the sensor, the time allotted for breaker arc suppression (135 ms @ 100% RTP), and the response time of the system logic.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30936P-A, "BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)," Parts 1 and 2 and GENE-770-06-2-A, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications." The safety evaluation reports documenting NRC approval of NEDC-30936P-A and GENE-770-06-2-A are contained in letters to D. N. Grace from A. C. Thadani dated December 9, 1988 (Part 1), D. N. Grace to C. E. Rossi dated December 9, 1988 (Part 2), and G. J. Beck from C. E. Rossi dated September 13, 1991.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits and Section 3/4.3 Instrumentation. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with 10 CFR Part 50, Appendix A, General Design Criteria 19, 41, 60, 61, 63 and 64.

3.4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit. This instrumentation is consistent with the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes," April 1974.

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 62 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 April 1, 1993, as supplemented July 2, 1993, the Public Service Electric & Gas Company (the licensee) submitted a request for changes to the Hope Creek Generating Station (HCGS), Technical Specifications (TSs). This amendment request would revise the TSs surveillance requirements to extend the surveillance test intervals (STIs) and allowed out-of-service times (AOTs) for the emergency core cooling system (ECCS) and reactor core isolation cooling (RCIC) system actuation instrumentation. The July 2, 1993 letter forwarded a non-proprietary version (the original letter forwarded the proprietary version) of General Electric Company report RE-018 (Revision 1), "TS Improvement Analysis for ECCS Actuation Instrumentation for Hope Creek Generating Station," and did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The licensee has proposed changes to TS Sections 3/4.3.3 (ECCS) and 3/4.3.5 (RCIC) and their associated bases. The proposed changes are based on the NRC staff's previous approvals of the following GE licensing topical reports (LTRs):

1. NEDC-30936P-A, "BWR Owner's Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation) Part 1," dated December 1988. This LTR was approved by letter and safety evaluation dated December 9, 1988, from A. C. Thadani (NRC) to D. N. Grace (BWR Owners Group).

2. NEDC-30936P-A, "BWR Owner's Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation) Part 2," dated December 1988. This LTR was approved by letter and safety evaluation dated December 9, 1988, from C. E. Rossi (NRC) to D. N. Grace (BWR Owners Group).

3. GENE-770-06-2, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation

Technical Specifications," dated February 1991. This LTR was approved by letter and safety evaluation dated September 13, 1991, from C. E. Rossi (NRC) to G. J. Beck (BWR Owners Group).

Each of the above LTRs was prepared and approved on a generic basis by the staff as noted above. The staff's safety evaluations required that two conditions be satisfied to justify the applicability of the generic analysis to individual plants. These conditions were: 1) confirmation of the applicability of the generic analyses to the specific plant, and 2) confirmation that any increase in instrument drift due to the extended STIs is properly accounted for in the setpoint calculation methodology. The licensee's responses to these conditions are evaluated in the following discussion.

In order to discuss plant-specific differences between the GE LTRs and HCGS, the licensee referenced GE Report RE-018, Revision 1. This report extended the generic analyses developed in the LTRs to the HCGS and discussed the plant-specific differences in accordance with the NRC safety evaluations of the LTRs. Based on its own analysis and the results of the GE plant-specific analysis, the licensee stated that the proposed changes are consistent with the changes proposed and approved in the referenced GE LTRs and associated NRC safety evaluations with one exception. The licensee stated that the proposed wording relative to the maintenance AOTs in NEDC-30936P-A, Parts 1 and 2, imply an allowance of 24 hours <u>before</u> taking the action of TS 3.3.3-1. To clarify this TS, the licensee used the wording in a letter (GE Document No. OG90-319-32D) from W. P. Sullivan (GE) to NRC, "Clarification of Technical Specification Changes Given in ECCS Actuation Instrumentation Analysis," dated March 22, 1990. The staff reviewed this exception and agrees that it acceptably clarifies the TS wording. Therefore, based on the licensee's statements and the staff's review of the licensee's submittal (including the plant-specific analysis contained in GE Report RE-018, Revision 1), the staff agrees that the generic analyses apply to the HCGS.

In order to evaluate the acceptability of instrument drift, the licensee utilized a two-fold approach. The first approach determined that the existing setpoint drift calculations were based on an 18-month interval; therefore, the proposed STI extensions from monthly to quarterly are well bounded by the existing setpoint calculations. The second approach used actual plant data to confirm the vendor's drift expectations. The licensee's data showed that over a 12-month period, the observed drift was found to be conservatively bounded by the total loop allowance for a six-month period. This verifies that a quarterly STI schedule is acceptable. Based on the staff's review of the licensee's analysis, the staff agrees that instrument drift due to the extended STIs is properly accounted for in the setpoint calculation methodology.

The licensee has satisfied the two conditions that the staff imposed on licensees that request these TS amendments. Therefore, the licensee's proposal is acceptable.

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 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any 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 (58 FR 25864). 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: S. Dembek

Date: December 27, 1993