December 18, 1989

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: MODIFICATION OF TECHNICAL SPECIFICATIONS TO DELETE REFERENCES TO CONDITIONS PERMITTED DURING STARTUP TESTING PROGRAM (TAC NO. 75095)

Re: HOPE CREEK GENERATING STATION

The Commission has issued the enclosed Amendment No. 35 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 September 25, 1989.

This amendment revises the Technical Specifications by deleting references to conditions permitted during the startup testing program.

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

Sincerely,

Clyde Shiraki, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 35 to License No. NPF-57
2. Safety Evaluation

cc w/enclosures:
See next page

9001020099 891218
PDR ADOCK 05000354
P PNU

DISTRIBUTION w/enclosures:

Docket File	MO'Brien (2)	Wanda Jones	SVarga
NRC PDR	OGC	JCalvo	BBoger
Local PDR	DHagan	Tech Branch	JDyer
PDI-2 Reading	EJordan	ACRS (10)	JLinville
WButler	GPA/PA	CShiraki(3)	SBrown
GHill (4)	Rita Jaques, ARM/LFMB		

[TAC 75095]

PDI-2/LA	PDI-2/PM	For PDI-2/D	OGC
MO'Brien	CShiraki:mj	WButler	BMB
/89	12/14/89	12/18/89	11/30/89

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

December 18, 1989

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: MODIFICATION OF TECHNICAL SPECIFICATIONS TO DELETE REFERENCES TO CONDITIONS PERMITTED DURING STARTUP TESTING PROGRAM (TAC NO. 75095)

Re: HOPE CREEK GENERATING STATION

The Commission has issued the enclosed Amendment No. 35 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 September 25, 1989.

This amendment revises the Technical Specifications by deleting references to conditions permitted during the startup testing program.

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

Sincerely,

Clyde Shiraki, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 35 to
License No. NPF-57
2. Safety Evaluation

cc w/enclosures:
See next page

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

Hope Creek Generating Station

cc:

M. J. Wetterhahn, Esquire
Conner & Wetterhahn
Suite 1050
1747 Pennsylvania Avenue
Washington, D.C. 20006

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

Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 241
Hancocks Bridge, New Jersey 08038

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

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

Mr. B. A. Preston, 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, Ph.D
New Jersey Department of Environmental
Protection
Radiation Protection Program
State of New Jersey
CN 415
Trenton, New Jersey 08625

Mr. Scott B. Ungerer, Manager
Joint Generation Projects Department
Atlantic Electric Company
Post Office Box 1500
Pleasantville, New Jersey 08232

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 35
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 September 25, 1989 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.35 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PSE&G shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James C. Stone

For Walter R. Butler, 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 18, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 35

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
3/4 3-5	3/4 3-5
3/4 3-6	3/4 3-6*
3/4 3-21	3/4 3-21*
3/4 3-22	3/4 3-22
3/4 3-23	3/4 3-23
3/4 3-24	3/4 3-24
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26*
3/4 3-47	3/4 3-47
3/4 3-48	3/4 3-48*
3/4 3-65	3/4 3-65
3/4 3-66	3/4 3-66*
3/4 4-7	3/4 4-7*
3/4 4-8	3/4 4-8
3/4 4-9	3/4 4-9*
3/4 4-10	3/4 4-10
3/4 4-17	3/4 4-17*
3/4 4-18	3/4 4-18
3/4 5-5	3/4 5-5
3/4 5-6	3/4 5-6*
3/4 10-1	3/4 10-1*
3/4 10-2	3/4 10-2
3/4 10-3	3/4 10-3*
3/4 10-4	3/4 10-4
3/4 10-5	3/4 10-5
3/4 10-6	3/4 10-6*

ATTACHMENT TO LICENSE AMENDMENT NO. 35

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Remove	Insert
3/4 10-7	3/4 10-7*
3/4 10-8	-
B 3/4 10-1	B 3/4 10-1
-	-

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(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) This function shall be automatically bypassed when the reactor mode switch is in the Run position.

(c) Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn*.

(d) The non-coincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per the Trip System are 4 APRMS, 6 IRMS and 2 SRMS.

(e) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(g) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) This function shall be automatically bypassed when turbine first stage pressure is $\leq$ 159.7 psig equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER. To allow for instrument accuracy, calibration, and drift, a setpoint of $\leq$ 135.7 psig is used.

(k) Also actuates the EOC-RPT system.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, Setdown	NA
b. Flow Biased Simulated Thermal Power - Upscale	$\leq$ 0.09**
c. Fixed Neutron Flux - Upscale	$\leq$ 0.09
d. Inoperative	NA
e. Downscale	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 0.55
4. Reactor Vessel Water Level - Low, Level 3	$\leq$ 1.05
5. Main Steam Line Isolation Valve - Closure	$\leq$ 0.06
6. Main Steam Line Radiation - High, High	NA
7. Drywell Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
a. Float Switch	NA
b. Level Transmitter/Trip Unit	NA
9. Turbine Stop Valve - Closure	$\leq$ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\leq$ 0.08#
11. Reactor Mode Switch Shutdown Position	NA
12. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.
**Not including simulated thermal power time constant, 6 ± 0.6 seconds.
#Measured from start of turbine control valve fast closure.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATION

TRIP FUNCTION	VALVES CLOSED BY SIGNAL
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	
a. Reactor Vessel Water Level Low, Level 3	3 (HV-F008, HV-F009, HV-F015A & B, HV-F022, HV-F023)
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	3 (HV-F008, HV-F009, HV-F015A & B, HV-F022, HV-F023)
c. Manual Initiation	3 (HV-F008, HV-F009, HV-F015A & B, HV-F022, HV-F023)

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1) Low Low, Level 2	≥ -38.0 inches*	≥ -45.0 inches
2) Low Low Low, Level 1	≥ -129.0 inches*	≥ -136.0 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Reactor Building Exhaust Radiation - High	≤ $1x10^{-3}$μCi/cc	≤ $1.2x10^{-3}$μCi/cc
d. Manual Initiation	NA	NA
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -38.0 inches*	≥ -45.0 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Refueling Floor Exhaust Radiation - High	≤ $2x10^{-3}$μCi/cc	≤ $2.4x10^{-3}$μCi/cc
d. Reactor Building Exhaust Radiation - High	≤ $1x10^{-3}$μCi/cc	≤ $1.2x10^{-3}$μCi/cc
e. Manual Initiation	NA	NA
3. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129.0 inches*	≥ -136.0 inches
b. Main Steam Line Radiation - High, High###	≤ 3.0 X full power background	≤ 3.6 X full power background
c. Main Steam Line Pressure - Low	≥ 756.0 psig	≥ 736.0 psig
d. Main Steam Line Flow - High	≤ 108.7 psid	≤ 111.7 psid

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
MAIN STEAM LINE ISOLATION (Continued)		
e. Condenser Vacuum - Low	$\geq$ 8.5 inches Hg vacuum	$\geq$ 7.6 inches Hg vacuum
f. Main Steam Line Tunnel Temperature - High	$\leq$ 160°F	$\leq$ 172°F
g. Manual Initiation	NA	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. RWCU Δ Flow - High	$\leq$ 56.3 gpm	$\leq$ 61.3 gpm
b. RWCU Δ Flow - High, Timer	45.0 seconds $\leq$ t $\leq$ 47.0 seconds	45.0 seconds $\leq$ t $\leq$ 47.0 seconds
c. RWCU Area Temperature - High	$\leq$ 160°F, 140°F or 135°F***	$\leq$ 172°F, 152°F or 147°F***
d. RWCU/Area Ventilation Δ Temperature - High	$\leq$ 60°F	$\leq$ 70°F
e. SLCS Initiation	NA	NA
f. Reactor Vessel Water Level - Low Low, Level 2	$\geq$ -38.0 inches*	$\geq$ -45.0 inches
g. Manual Initiation	NA	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Δ Pressure (Flow) - High	$\leq$ 598" H_2O	$\leq$ 611" H_2O
b. RCIC Steam Line Δ Pressure (Flow) - High, Timer	3.0 seconds $\leq$ t $\leq$ 13.0 seconds	3.0 seconds $\leq$ t $\leq$ 13.0 seconds
c. RCIC Steam Supply Pressure - Low	$\geq$ 64.5 psig	$\geq$ 56.5 psig
d. RCIC Turbine Exhaust Diaphragm Pressure - High	$\leq$ 10.0 psig	$\leq$ 20.0 psig

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION		TRIP SETPOINT	ALLOWABLE VALUE
REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION (Continued)			
e.	RCIC Pump Room Temperature - High	$\leq$ 160°F	$\leq$ 172°F
f.	RCIC Pump Room Ventilation Duct Δ Temperature - High	$\leq$ 70°F	$\leq$ 80°F
g.	RCIC Pipe Routing Area Temperature - High	$\leq$ 160°F#	$\leq$ 172°F#
h.	RCIC Torus Compartment Temperature - High	$\leq$ 128°F#	$\leq$ 140°F#
i.	Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
j.	Manual Initiation	NA	NA
6.	HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION		
a.	HPCI Steam Line Δ Pressure (Flow) - High	$\leq$ 1032 inches H_2O	$\leq$ 1064 inches H_2O
b.	HPCI Steam Line Δ Pressure (Flow) - High, Timer	3.0 seconds $\leq t \leq$ 13.0 seconds	3.0 seconds $\leq t \leq$ 13.0 seconds
c.	HPCI Steam Supply Pressure - Low	$\geq$ 100.0 psig	$\geq$ 90.0 psig
d.	HPCI Turbine Exhaust Diaphragm Pressure - High	$\leq$ 10.0 psig	$\leq$ 20.0 psig
e.	HPCI Pump Room Temperature - High	$\leq$ 160°F	$\leq$ 172°F
f.	HPCI Pump Room Ventilation Ducts Δ Temperature - High	$\leq$ 70°F	$\leq$ 80°F
g.	HPCI Pipe Routing Area Temperature - High	$\leq$ 160°F##	$\leq$ 172°F##
h.	HPCI Torus Compartment Temperature - High	$\leq$ 128°F##	$\leq$ 140°F##
i.	Drywell Pressure High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
j.	Manual Initiation	NA	NA

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Water Level - Low, Level 3	$\geq$ 12.5 inches*	$\geq$ 11.0 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	$\leq$ 82.0 psig	$\leq$ 102.0 psig
c. Manual Initiation	NA	NA

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

***These setpoints are as follows:
- 160°F - RWCU pipe chase room 4402
- 140°F - RWCU pump room and heat exchanger rooms
- 135°F - RWCU pipe chase room 4505

#30 minute time delay.

##15 minute time delay.

###The hydrogen water chemistry (HWC) system shall not be placed in service until reactor power reaches 20% of RATED THERMAL POWER. After reaching 20% of RATED THERMAL POWER, and prior to operating the HWC system, the normal full power background radiation level and associated trip setpoints may be increased to levels previously measured during full power operation with hydrogen injection. Prior to decreasing below 20% of RATED THERMAL POWER and after the HWC system has been shutoff, the background level and associated setpoint shall be returned to the normal full power values. If a power reduction event occurs so that the reactor power is below 20% of RATED THERMAL POWER without the required setpoint change, control rod motion shall be suspended (except for scram or other emergency actions) until the necessary setpoint adjustment is made.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1) Low Low, Level 2	NA
2) Low Low Low, Level 1	NA
b. Drywell Pressure - High	NA
c. Reactor Building Exhaust Radiation - High	NA
d. Manual Initiation	NA
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level-Low Low, Level 2	NA
b. Drywell Pressure - High	NA
c. Refueling Floor Exhaust Radiation - High(b)	$\leq$ 4.0
d. Reactor Building Exhaust Radiation - High(b)	$\leq$ 4.0
e. Manual Initiation	NA
3. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\leq$ 1.0*/$\leq$ 13(a)**
b. Main Steam Line Radiation - High, High(a)(b)	$\leq$ 1.0*/$\leq$ 13(a)**
c. Main Steam Line Pressure - Low	$\leq$ 1.0*/$\leq$ 13(a)**
d. Main Steam Line Flow-High	$\leq$ 0.5*/$\leq$ 13(a)**
e. Condenser Vacuum - Low	NA
f. Main Steam Line Tunnel Temperature - High	NA
g. Manual Initiation	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. RWCU Δ Flow - High	NA
b. RWCU Δ Flow - High, Timer	NA
c. RWCU Area Temperature - High	NA
d. RWCU Area Ventilation Δ Temperature - High	NA
e. SLCS Initiation	NA
f. Reactor Vessel Water Level - Low Low, Level 2	NA
g. Manual Initiation	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Δ Pressure (Flow) - High	NA
b. RCIC Steam Line Δ Pressure (Flow) - High, Timer	NA
c. RCIC Steam Supply Pressure - Low	NA
d. RCIC Turbine Exhaust Diaphragm Pressure - High	NA

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]
1. Turbine Stop Valve - Closure	2[(b)]
2. Turbine Control Valve-Fast Closure	2[(b)]

(a) A trip system may be placed in an inoperable status for up to 2 hours for required surveillance provided that the other trip system is OPERABLE.

(b) This function shall be automatically bypassed when turbine first stage pressure is ≤ 159.7 psig equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER. To allow for instrument accuracy, calibration and drift, a setpoint of ≤ 135.7 psig is used.

TABLE 3.3.4.2-2

END-OF-CYCLE RECIRCULATION PUMP TRIP SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Turbine Stop Valve-Closure	$\leq$ 5% closed	$\leq$ 7% closed
2. Turbine Control Valve-Fast Closure	$\geq$ 530 psig	$\geq$ 465 psig

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

ACTION

ACTION 71 -

a. With one of the required monitors inoperable, place the inoperable channel in the tripped condition within one hour; restore the inoperable channel to OPERABLE status within 7 days, or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the pressurization mode of operation.

b. With both of the required monitors inoperable, initiate and maintain operation of the control room emergency filtration system in the pressurization mode of operation within one hour.

ACTION 72 - With the required monitor inoperable, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 73 - With the required monitor inoperable, obtain and analyze at least one sample of the monitored parameter at least once per 24 hours.

ACTION 74 - With the number of channels OPERABLE less than required by Minimum Channels OPERABLE requirement, release(s) via this pathway may continue for up to 30 days provided:

a. The offgas system is not bypassed, and

b. Grab samples are taken at least once per 8 hours and analyzed within the following 4 hours;

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Control Room Ventilation Radiation Monitor	S	M	R	1, 2, 3, 5 and *
2. Area Monitors				
a. Criticality Monitors				
1) New Fuel Storage Vault	S	M	R	#
2) Spent Fuel Storage Pool	S	M	R	##
b. Control Room Direct Radiation Monitor	S	M	R	At all times
3. Reactor Auxiliaries Cooling Radiation Monitor	S	M	R	At all times
4. Safety Auxiliaries Cooling Radiation Monitor	S	M	R	At all times
5. Offgas Pre-treatment Radiation Monitor	S	M	R	**

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 The safety valve function of at least 13 of the following reactor coolant system safety/relief valves shall be OPERABLE*# with the specified code safety valve function lift settings:**

4 safety-relief valves @ 1108 psig ±1%
5 safety-relief valves @ 1120 psig ±1%
5 safety-relief valves @ 1130 psig ±1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the safety valve function of two or more of the above listed fourteen safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 110°F, close the stuck open safety relief valve(s); if unable to close the stuck open valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more of the above required safety/relief valve acoustic monitors inoperable, restore the inoperable monitors to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*SRVs which perform as ADS function must also satisfy the OPERABILITY requirements of Specification 3.5.1, ECCS-Operating.

**The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

#SRVs which perform a low-low set function must also satisfy the OPERABILITY requirements of Specification 3.2.2, Safety/Relief Valves Low-Low Set Function.

SURVEILLANCE REQUIREMENTS

4.4.2.1 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be $\leq$ 30% of full open noise level** by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and a

b. CHANNEL CALIBRATION at least once per 18 months*.

4.4.2.2 At least 1/2 of the safety relief valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations at least once per 18 months, and they shall be rotated such that all 14 safety relief valves are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations at least once per 40 months.

---

*The provisions of Specification 4.0.4 are not applicable provided the Surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

**Initial setting shall be in accordance with the manufacturer's recommendations. Adjustment to the valve full open noise level shall be accomplished after the initial noise traces have been analyzed.

REACTOR COOLANT SYSTEM

SAFETY/RELIEF VALVES LOW-LOW SET FUNCTION

LIMITING CONDITION FOR OPERATION

3.4.2.2 The relief valve function and the low-low set function of the following reactor coolant system safety/relief valves shall be OPERABLE with the following settings:

Valve No.	Low-Low Set Function Setpoint* (psig) ±2% Open	Close
F013H	1017	905
F013P	1047	935

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the relief valve function and/or the low-low set function of one of the above required reactor coolant system safety/relief valves inoperable, restore the inoperable relief valve function and low-low set function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the relief valve function and/or the low-low set function of both of the above required reactor coolant system safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2.1 The relief valve function and the low-low set function pressure actuation instrumentation shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days.

b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system (excluding actual valve actuation) at least once per 18 months.

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

REACTOR COOLANT SYSTEM

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The drywell atmosphere gaseous radioactivity monitoring system,
b. The drywell floor and equipment drain sump monitoring system,
c. The drywell air cooler condensate flow rate monitoring system,
d. The drywell pressure monitoring system, and
e. The drywell temperature monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With only four of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required drywell atmosphere gaseous radioactivity monitoring system, the drywell pressure monitoring system, the drywell temperature monitoring system and/or the drywell air cooler condensate flow rate monitoring system is inoperable; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

a. Drywell atmosphere gaseous radioactivity monitoring system-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

b. The drywell pressure shall be monitored at least once per 12 hours and the drywell temperature shall be monitored at least once per 24 hours.

c. Drywell floor and equipment drain sump monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.

d. Drywell air coolers condensate flow rate monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

TABLE 3.4.4-1

REACTOR COOLANT SYSTEM CHEMISTRY LIMITS

OPERATIONAL CONDITION	CHLORIDES	CONDUCTIVITY (μmhos/cm @25°C)	PH
1	$\leq$ 0.2 ppm	$\leq$ 1.0	5.6 $\leq$ pH $\leq$ 8.6
2 and 3	$\leq$ 0.1 ppm	$\leq$ 2.0	5.6 $\leq$ pH $\leq$ 8.6
At all other times	$\leq$ 0.5 ppm	$\leq$ 10.0	5.3 $\leq$ pH $\leq$ 8.6

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.5 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.2 microcuries per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microcuries per gram.

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

ACTION:

a. In OPERATIONAL CONDITIONS 1, 2 or 3 with the specific activity of the primary coolant;

 1. Greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or greater than 4.0 microcuries per gram DOSE EQUIVALENT I-131, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

 2. Greater than $100/\bar{E}$ microcuries per gram, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

b. In OPERATIONAL CONDITIONS 1, 2, 3 or 4, with the specific activity of the primary coolant greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries per gram, perform the sampling and analysis requirements of Item 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.

c. In OPERATIONAL CONDITION 1 or 2, with:

 1. THERMAL POWER changed by more than 15% of RATED THERMAL POWER in one hour, or

 2. The off-gas level, at the SJAE, increased by more than 10,000 microcuries per second in one hour during steady state operation at release rates less than 75,000 microcuries per second, or

 3. The off-gas level, at the SJAE, increased by more than 15% in one hour during steady state operation at release rates greater than 75,000 microcuries per second,

 perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.

SURVEILLANCE REQUIREMENTS (Continued)

2. For the HPCI system, verifying that:
   a) The system develops a flow of at least 5600 gpm against a test line pressure corresponding to a reactor vessel pressure of $\geq$ 200 psig, when steam is being supplied to the turbine at 200 + 15, -0 psig.**
   b) The suction is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber - water level high signal.
3. Performing a CHANNEL CALIBRATION of the CSS, and LPCI system discharge line "keep filled" alarm instrumentation.
4. Performing a CHANNEL CALIBRATION of the CSS header ΔP instrumentation and verifying the setpoint to be $\leq$ the allowable value of 4.4 psid.
5. Performing a CHANNEL CALIBRATION of the LPCI header ΔP instrumentation and verifying the setpoint to be $\leq$ the allowable value of 1.0 psid.

d. For the ADS:
   1. At least once per 31 days, performing a CHANNEL FUNCTIONAL TEST of the Primary Containment Instrument Gas System low-low pressure alarm system.
   2. At least once per 18 months:
      a) Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.
      b) Manually opening each ADS valve when the reactor steam dome pressure is greater than or equal to 100 psig** and observing that either:
         1) The control valve or bypass valve position responds accordingly, or
         2) There is a corresponding change in the measured steam flow.
      c) Performing a CHANNEL CALIBRATION of the Primary Containment Instrument Gas System low-low pressure alarm system and verifying an alarm setpoint of 85 $\pm$ 2 psig on decreasing pressure.

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3/4 5.2 ECCS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.2 At least two of the following shall be OPERABLE:

a. Core spray system subsystems with a subsystem comprised of:

1. Two OPERABLE core spray pumps, and

2. An OPERABLE flow path capable of taking suction from at least one of the following water sources and transferring the water through the spray sparger to the reactor vessel:

a) From the suppression chamber, or

b) When the suppression chamber water level is less than the limit or is drained, from the condensate storage tank containing at least 135,000 available gallons of water.

b. Low pressure coolant injection (LPCI) system subsystems each with a subsystem comprised of:

1. One OPERABLE LPCI pump, and

2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITION 4 and 5*.

ACTION:

a. With one of the above required subsystems inoperable, restore at least two subsystems to OPERABLE status within 4 hours or suspend all operations with a potential for draining the reactor vessel.

b. With both of the above required subsystems inoperable, suspend CORE ALTERATIONS and all operations with a potential for draining the reactor vessel. Restore at least one subsystem to OPERABLE status within 4 hours or establish SECONDARY CONTAINMENT INTEGRITY within the next 8 hours.

---

*The ECCS is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, the spent fuel pool gates are removed, and water level is maintained within the limits of Specification 3.9.8 and 3.9.9.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.10.1 The provisions of Specifications 3.6.1.1, 3.6.1.3 and 3.9.1 and Table 1.2 may be suspended to permit the reactor pressure vessel closure head and the drywell head to be removed and the primary containment air lock doors to be open when the reactor mode switch is in the Startup position during low power PHYSICS TESTS with THERMAL POWER less than 1% of RATED THERMAL POWER and reactor coolant temperature less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during low power PHYSICS TESTS.

ACTION:

With THERMAL POWER greater than or equal to 1% of RATED THERMAL POWER or with the reactor coolant temperature greater than or equal to 200°F, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.1 The THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during low power PHYSICS TESTS.

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups by the rod worth minimizer (RWM) per Specification 3.1.4.1 and by the rod sequence control system (RSCS) per Specification 3.1.4.2 may be suspended by means of bypass switches for the following tests provided that control rod movement prescribed for this testing is verified by a second licensed operator or other technically qualified member of the unit technical staff present at the reactor console:

a. Shutdown margin demonstrations, Specification 4.1.1.

b. Control rod scram, Specification 4.1.3.2.

c. Control rod friction measurements.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the requirements of the above specification not satisfied, verify that the RWM and/or the RSCS is OPERABLE per Specifications 3.1.4.1 and 3.1.4.2, respectively.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints imposed by the RSCS and/or RWM are bypassed, verify:

a. That movement of the control rods from 75% ROD DENSITY to the RSCS low power setpoint is limited to the approved control rod withdrawal sequence during scram and friction tests.

b. That movement of control rods during shutdown margin demonstrations is limited to the prescribed sequence per Specification 3.10.3.

c. Conformance with this specification and test procedures by a second licensed operator or other technically qualified member of the unit technical staff.

SPECIAL TEST EXCEPTIONS

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

3.10.3 The provisions of Specification 3.9.1, Specification 3.9.3 and Table 1.2 may be suspended to permit the reactor mode switch to be in the Startup position and to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

a. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.9.2.

b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1 and is programmed for the shutdown margin demonstration, or conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.

c. The "rod-out-notch-override" control shall not be used during out-of-sequence movement of the control rods.

d. No other CORE ALTERATIONS are in progress.

APPLICABILITY: OPERATIONAL CONDITION 5, during shutdown margin demonstrations.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown or Refuel position.

SURVEILLANCE REQUIREMENTS

4.10.3 Within 30 minutes prior to and at least once per 12 hours during the performance of a shutdown margin demonstration, verify that;

a. The source range monitors are OPERABLE per Specification 3.9.2,

b. The rod worth minimizer is OPERABLE with the required program per Specification 3.1.4.1 or a second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown demonstration procedures, and

c. No other CORE ALTERATIONS are in progress.

SPECIAL TEST EXCEPTIONS

3/4.10.4 RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.4.1.1 and 3.4.1.3 that recirculation loops be in operation with matched pump speed may be suspended for up to 24 hours for the performance of:

a. PHYSICS TESTS, provided that THERMAL POWER does not exceed 5% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2, during PHYSICS TESTS.

ACTION:

a. With the above specified time limit exceeded, insert all control rods.

b. With the above specified THERMAL POWER limit exceeded during PHYSICS TESTS, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The time during which the above specified requirement has been suspended shall be verified to be less than 24 hours at least once per hour during PHYSICS TESTS.

4.10.4.2 THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.5 OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

The material originally contained in this Technical Specification was deleted with the issuance of Amendment No.35. However, to maintain the historical reference to this specification, this section has been intentionally left blank.

SPECIAL TEST EXCEPTIONS

3/4.10.6 TRAINING STARTUPS

LIMITING CONDITION FOR OPERATION

3.10.6 The provisions of Specification 3.5.1 may be suspended to permit one RHR subsystem to be aligned in the shutdown cooling mode during training startups provided that the reactor vessel is not pressurized, THERMAL POWER is less than or equal to 1% of RATED THERMAL POWER and reactor coolant temperature is less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during training startups.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.6 The reactor vessel shall be verified to be unpressurized and the THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during training startups.

3/4.10.7 SPECIAL INSTRUMENTATION - INITIAL CORE LOADING

LIMITING CONDITION FOR OPERATION

3/4.10.7 The material originally contained in Section 3/4.10.7 was deleted with the issuance of Amendment No. 14. However, to maintain the historical reference to this section, Section 3/4.10.7 is intentionally left blank.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

The requirement for PRIMARY CONTAINMENT INTEGRITY is not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS.

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

In order to perform the tests required in the technical specifications it is necessary to bypass the sequence restraints on control rod movement. The additional surveillance requirements ensure that the specifications on heat generation rates and shutdown margin requirements are not exceeded during the period when these tests are being performed and that individual rod worths do not exceed the values assumed in the safety analysis.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

Performance of shutdown margin demonstrations during open vessel testing requires additional restrictions in order to ensure that criticality is properly monitored and controlled. These additional restrictions are specified in this LCO.

3/4.10.4 RECIRCULATION LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 OXYGEN CONCENTRATION

The material originally contained in this Technical Specification was deleted with the issuance of Amendment No. 35. However, to maintain the historical reference to this specification, this section has been intentionally left blank.

3/4.10.6 TRAINING STARTUPS

This special test exception permits training startups to be performed with the reactor vessel depressurized at low THERMAL POWER and temperature while controlling RCS temperature with one RHR subsystem aligned in the shutdown cooling mode in order to minimize contaminated water discharge to the radioactive waste disposal system.

3/4.10.7 SPECIAL INSTRUMENTATION - INITIAL CORE LOADING

The material originally contained in Bases Section 3/4.10.7 was deleted with the issuance of Amendment No. 14. However, to maintain the historical reference to this section, Bases Section 3/4.10.7 is intentionally left blank.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 35 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 25, 1989, Public Service Electric & Gas Company requested an amendment to Facility Operating License No. NPF-57 for the Hope Creek Generating Station. The proposed amendment would modify the Technical Specification by deleting references to conditions permitted during the startup testing program.

2.0 EVALUATION

The changes requested in this submittal revise TS which permitted various conditions and initial values during the Startup Test Program. These changes generally consist of specifications which either: (i) reference conditions that do not exist or are no longer applicable, (ii) do not contain a specific value but rather reference the fact that the value will be determined later, or (iii) contain a specific value but identify the value as preliminary.

The Startup Test Program has been completed and the original TS values can now be finalized. These changes would aid operational use of the TS and assure that information contained within the document is accurate, even if more conservative.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (54 FR 46156) on November 1, 1989 and consulted with the State of New Jersey. No public comments were received and the State of New Jersey did not have any comments.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: C. Shiraki

Dated: December 18, 1989