August 17, 1987

Docket No. 50-354

Mr. Corbin A. McNeill, Jr.
Senior Vice President - Nuclear
Public Service Electric & Gas Company
P.O. Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. McNeill:

SUBJECT: MAIN STEAM LINE RADIATION-HIGH, HIGH TRIP SET POINTS(TAC# 65588)

Re: HOPE CREEK GENERATING STATION

The Commission has issued the enclosed Amendment No. 8 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 May 22, 1987, as supplemented June 30, 1987.

This amendment permits a temporary increase in the Main Steam Line Radiation-High-High scram and isolation trip setpoints to allow operation with expected higher radiation levels resulting from hydrogen injection testing.

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

Sincerely,

/s/

George Rivenbark, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II

8708240171
870817
PDR ADOCK 05000354
P PDR

Enclosures:
1. Amendment No. 8 to License No. NPF-57
2. Safety Evaluation

cc w/enclosures:
See next page

With changes noted to SE - J Santo gave oral concurrence on 8/6/87 GRivenbark

No Legal Objection 8/4/87

DISTRIBUTION:
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GRivenbark:mr
7/29/87

OGC
M. Karman
7/30/87

D:PDI-2:DRPI/II
WButler
8/10/87

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: **August 17, 1987**

LA:PDI-2:DRPI/II	PM:PDI-2:DRPI/II	OGC	D:PDI-2:DRPI/II
MO'Brien	GRivenbark:mr	M. Karman	WButler
8/17/87	7/24/87	7/30/87	8/10/87

with changes noted to OGC, J. Scinto gave his oral concurrence on 8/6/87

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8/ /87

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7/ /87

OGC
M. Karman
7/30/87

D:PDI-2:DRPI/II
WButler
8/10/87

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

August 17, 1987

Docket No. 50-354

Mr. Corbin A. McNeill, Jr.
Senior Vice President - Nuclear
Public Service Electric & Gas Company
P.O. Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. McNeill:

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Re: HOPE CREEK GENERATING STATION

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Sincerely,

George W Rivenbark

George Rivenbark, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 8 to
   License No. NPF-57
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. C. A. McNeill
Public Service Electric & Gas Co.

Hope Creek Generating Station

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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. 8
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 May 22, 1987, as supplemented June 30, 1987 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. 8, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
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Date of Issuance: **August 17, 1987**

LA:PDI-2:DRPI/II	PM:PDI-2:DRPI/II	OGC	D:PDI-2:DRPI/II
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8/17/87	7/24/87	7/ /87	8/10/87

with changes noted to ST, J. Scinto gave his oral concurrence on 8/6/87

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FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 17, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 8

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 page(s) provided to maintain document completeness.*

Remove	Insert
2-3*	2-3*
2-4	2-4
2-5	2-5
3/4 3-11*	3/4 3-11*
3/4 3-12	3/4 3-12
3/4 3-15*	3/4 3-15*
3/4 3-16	3/4 3-16
-	3/4 3-16a
3/4 3-21*	3/4 3-21*
3/4 3-22	3/4 3-22
3/4 3-25	3/4 3-25
3/4 3-26*	3/4 3-26*

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	$\leq$ 120/125 divisions of full scale	$\leq$ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	$\leq$ 15% of RATED THERMAL POWER	$\leq$ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-Upscale		
1) Flow Biased	$\leq$ 0.66(w-Δw)+51%** with a maximum of	$\leq$ 0.66(w-Δw)+54%** with a maximum of
2) High Flow Clamped	$\leq$ 113.5% of RATED THERMAL POWER	$\leq$ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux-Upscale	$\leq$ 118% of RATED THERMAL POWER	$\leq$ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
e. Downscale	$\geq$ 4% of RATED THERMAL POWER	$\geq$ 3% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 1037 psig	$\leq$ 1057 psig
4. Reactor Vessel Water Level - Low, Level 3	$\geq$ 12.5 inches above instrument zero*	$\geq$ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	$\leq$ 8% closed	$\leq$ 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 = "To be determined at a later date" for single recirculation loop operation.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS
(continued)

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. Main Steam Line Radiation - High, High#	≤ 3.0 x full power background	≤ 3.6 x full power background
7. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Float Switch	Elevation 110' 10.5"	Elevation 111' 0.5"
b. Level Transmitter/Trip Unit	Elevation 110' 10.5"*	Elevation 111' 4.5"*
9. Turbine Stop Valve - Closure	≤ 5% closed	≤ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 530 psig	≥ 465 psig
11. Reactor Mode Switch Shutdown Position	NA	NA
12. Manual Scram	NA	NA

*80.5" above instrument zero EL 104' 2" for Level Transmitter/Trip Unit A&B (South Header) 83.25" above instrument zero EL 103' 11.25" for Level Transmitter/Trip Unit C&D (North Header)

#Within 24 hours prior to the planned start of the hydrogen injection test, with reactor power at greater than 22% of RATED THERMAL POWER, the normal full power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test program based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and the associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of the hydrogen injection test or within 12 hours of establishing reactor power levels below 22% of RATED THERMAL POWER, while these functions are required to be OPERABLE.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE ACTUATION GROUPS OPERATED BY SIGNAL(d)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM(a)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low Low, Level 2	1, 2, 8, 9, 12, 13, 14, 15, 17, 18	2	1, 2, 3	20
2) Low Low Low, Level 1	10, 11, 15, 16	2	1, 2, 3	20
b. Drywell Pressure - High	1, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18	2	1, 2, 3	20
c. Reactor Building Exhaust Radiation - High	1, 8, 9, 12 13, 14, 15, 17, 18	3	1, 2, 3	28
d. Manual Initiation	1, 8, 9, 10 11, 12, 13, 14, 15, 16, 17, 18	1	1, 2, 3	24
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	19(c)	2	1, 2, 3 and *	26
b. Drywell Pressure - High	19(c)	2	1, 2, 3	26
c. Refueling Floor Exhaust Radiation - High	19(c)	3	1, 2, 3 and *	29
d. Reactor Building Exhaust Radiation - High	19(c)	3	1, 2, 3 and *	28
e. Manual Initiation	19(c)	1	1, 2, 3 and *	26

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE ACTUATION GROUPS OPERATED BY SIGNAL(d)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
3. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low Low Low, Level 1	1	2	1, 2, 3	21
b. Main Steam Line Radiation - High, High	1, 2(b)	2	1, 2, 3##	21
c. Main Steam Line Pressure - Low	1	2	1	22
d. Main Steam Line Flow - High	1	2/line	1, 2, 3	20
e. Condenser Vacuum - Low	1	2	1, 2**, 3**	21
f. Main Steam Line Tunnel Temperature - High	1	2/line	1, 2, 3	21
g. Manual Initiation	1, 2, 17	2	1, 2, 3	25
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. RWCU Δ Flow - High	7	1/Valve(e)	1, 2, 3	23
b. RWCU Δ Flow - High, Timer	7	1/Valve(e)	1, 2, 3	23
c. RWCU Area Temperature - High	7	6/Valve(e)	1, 2, 3	23
d. RWCU Area Ventilation Δ Temperature-High	7	6/Valve(e)	1, 2, 3	23
e. SLCS Initiation	7(f)	1/Valve(e)	1, 2, 5#	23
f. Reactor Vessel Water Level - Low Low, Level 2	7	2/Valve(e)	1, 2, 3	23
g. Manual Initiation	7	1/Valve(e)	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE ACTUATION GROUPS OPERATED BY SIGNAL[d]	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]	APPLICABLE OPERATIONAL CONDITION	ACTION
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	3	2/Valve[e]	1, 2, 3	27
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	3	2/Valve[e]	1, 2, 3	27
c. Manual Initiation	3	1/Valve[e]	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Be in at least STARTUP within 6 hours.

ACTION 23 - Close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION 24 - Restore the manual initiation function to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 25 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable.

ACTION 26 - Establish SECONDARY CONTAINMENT INTEGRITY with the Filtration, Recirculation and Ventilation System (FRVS) operating within one hour.

ACTION 27 - Lock the affected system isolation valves closed within one hour and declare the affected system inoperable.

ACTION 28 - Place the inoperable channel in the tripped condition or close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION 29 - Place the inoperable channel in the tripped condition or establish SECONDARY CONTAINMENT INTEGRITY with the Filtration, Recirculation, and Ventilation System (FRVS) operating within one hour.

NOTES

* When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

** When any turbine stop valve is greater than 90% open and/or when the key-locked bypass switch is in the Norm position.

\# Refer to Specification 3.1.5 for applicability.

\## Within 24 hours prior to the planned start of the hydrogen injection test, with reactor power at greater than 22% of RATED THERMAL POWER, the normal full power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test program based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and the associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of the hydrogen injection test or within 12 hours of establishing reactor power levels below 22% of RATED THERMAL POWER, while these functions are required to be OPERABLE.

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(b) Also trips and isolates the mechanical vacuum pumps.

(c) Also starts the Filtration, Recirculation and Ventilation System (FRVS).

TABLE 3.3.2-1 (Continued)

ACTION (Continued)

(d) Refer to Table 3.3.2-1 table notation for the listing of which valves in an actuation group are closed by a particular isolation signal. Refer to Tables 3.6.3-1 and 3.6.5.2-1 for the listings of all valves within an actuation group.
(e) Sensors arranged per valve group, not per trip system.
(f) Closes only RWCU system isolation valve(s) HV-F001 and HV-F004.
(g) Requires system steam supply pressure-low coincident with drywell pressure-high to close turbine exhaust vacuum breaker valves.
(h) Manual isolation closes HV-F008 only, and only following manual or automatic initiation of the RCIC system.
(i) Manual isolation closes HV-F003 and HV-F042 only, and only following manual or automatic initiation of the HPCI system.

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	$\geq$ -38.0 inches*	$\geq$ -45.0 inches
2) Low Low Low, Level 1	$\geq$ -129.0 inches*	$\geq$ -136.0 inches
b. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
c. Reactor Building Exhaust Radiation - High	$< 1 \times 10^{-3}$ μCi/cc**	$< 1.2 \times 10^{-3}$ μCi/cc**
d. Manual Initiation	NA	NA
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	$\geq$ -38.0 inches*	$\geq$ -45.0 inches
b. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
c. Refueling Floor Exhaust Radiation - High	$\leq 2 \times 10^{-3}$ μCi/cc**	$\leq 2.4 \times 10^{-3}$ μCi/cc**
d. Reactor Building Exhaust Radiation - High	$\leq 1 \times 10^{-3}$ μCi/cc**	$\leq 1.2 \times 10^{-3}$ μCi/cc**
e. Manual Initiation	NA	NA
3. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$ -129.0 inches*	$\geq$ -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	$\geq$ 756.0 psig	$\geq$ 736.0 psig
d. Main Steam Line Flow - High	$\leq$ 108.7 psid	$\leq$ 111.7 psid

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.
**Initial setpoint. Final setpoint to be determined during startup test program.
***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.
###Within 24 hours prior to the planned start of the hydrogen injection test, with reactor power at greater than 22% of RATED THERMAL POWER, the normal full power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test program based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and the associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of the hydrogen injection test or within 12 hours of establishing reactor power levels below 22% of RATED THERMAL POWER, while these functions are required to be OPERABLE.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 8 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 May 22, 1987, as supplemented June 30 1987, Public Service Electric & Gas Company (the licensee) requested an amendment to Facility Operating License No. NPF-57 for the Hope Creek Generating Station. The proposed amendment would modify the Technical Specifications to permit a temporary increase in the Hope Creek Main Steam Line Radiation-High-High scram and isolation setpoints to allow operation with expected higher radiation levels resulting from hydrogen injection testing. The purpose of the hydrogen injection testing is to determine the feasibility of hydrogen water chemistry control as a means of reducing intergranular stress corrosion cracking (IGSCC) of stainless steel piping.

2.0 EVALUATION

2.1 High Radiation Scram and Radiation Setpoints

The Main Steam Line Radiation Monitors (MSLRMs) provide reactor scram as well as Main Steam Isolation Valve (MSIV) closure signals upon detection of high radioactivity levels in the main steam lines. The closure of the MSIVs limits the release of fission products in the event of fuel failures. The proposed Technical Specification changes (Tables 2.2.1-1, 3.3.2-1, and 3.3.2-2) would allow adjustments to the normal background radiation level and associated trip setpoints for the MSLRMs at reactor power levels greater than 22 percent of rated power. The adjustments will be based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. Hydrogen water chemistry results in increased N-16 levels in the main steam lines.

The licensee states that the only design basis accident which takes credit for the main steam line high radiation scram and isolation set point is the Control Rod Drop Accident (CRDA). Generic analysis of the consequences of a CRDA have shown that fuel failures are not expected from a CRDA occurring at greater than 10 percent power. This is primarily a result of analyses which show that as power increases, the severity of the CRDA decreases due to the effects of increased void formation and increased Doppler reactivity feedback. Since hydrogen injection during the test will be limited to above 22 percent of rated power and the MSLRM setpoint adjustments will not be altered below this power level, the staff concludes that the currently approved CRDA analysis for Hope Creek is bounded appropriately and remains valid. Therefore, the proposed Technical Specification changes are acceptable.

2.2 Radiation Protection

The staff has reviewed the licensee's submittal regarding the radiological implications of the dose rate increases associated with N-16 activity increases during hydrogen injections into the reactor system. The review addresses the radiation protection/ALARA measures for the course of the planned test, in accordance with 10 CFR 20.1(c) and Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as low as is Reasonably Achievable." (ALARA)

One objective of the hydrogen injection test is to determine general in-plant and site boundary dose rate increases due to hydrogen addition. The licensee has stated that radiation protection/ALARA practices will be implemented during the test. Additionally, the licensee has stated that data will be obtained for shielding design should additional shielding be necessary for a permanent hydrogen water chemistry installation.

The staff has reviewed the licensee's proposed dose control measures and surveillance efforts planned for the hydrogen addition test. Tests of this type have been conducted at other operating BWRs, following staff review of similar Technical Specification changes. These test conditions, as identified by the licensee, as well as the measures proposed for radiation protection/ALARA at Hope Creek, are consistent with those utilized at the other BWRs during their hydrogen addition tests. None of these tests involved any significant, unanticipated, radiological exposures or releases.

Dresden 2, which has operated with hydrogen-water chemistry for two cycles, did not experience a measurable increase to site exposure. The staff does not expect a significant site boundary dose rate increase during the Hope Creek test. The conduct of the test and radiological surveys during the test will ensure ALARA in accordance with Regulatory Guide 8.8 and is, therefore, acceptable.

2.3 Hydrogen Storage and Distribution System

By letters dated May 22 and June 30, 1987, the licensee provided information on the hydrogen and oxygen storage and distribution system to facilitate the Hydrogen Water Chemistry pre-implementation test. The licensee's hydrogen and oxygen storage and distribution system is designed to minimize the potential hazard to safety related systems and meets the applicable parts of the BWR Owners Group, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations," 1987 Revision. The staff has reviewed this submittal to the guidance provided in NRC Branch Technical Position CMEB 9.5-1 of NUREG-0800, Standard Review Plan.

The pre-implementation test will be conducted with the guidance of General Electric (GE) taking into consideration the lessons learned from approximately eight other hydrogen injection tests previously performed with GE assistance. Compressed hydrogen will be supplied and stored onsite in a gaseous tube trailer (130,000 scf). The separation distance of the hydrogen tube trailer and safety related structures meets the BWR Owners Group (BWROG) Guidelines. The hydrogen distribution system contains an excess flow check valve to limit the release of hydrogen in the event of a pipe break. Hydrogen will be injected into the feedwater system at the suction side of the condensate feed pumps. To prevent the accumulation of combustible levels of hydrogen at the condensate booster pumps, near the control valves and/or at various locations along the supply lines, the hydrogen supply lines will be leak tested prior to the test and will be monitored for hydrogen concentrations during the test. The monitors will alarm when hydrogen concentrations exceed two percent and isolate the hydrogen supply line when the hydrogen concentration reaches four percent in order to prevent reaching an explosive concentration.

Oxygen will be injected upstream of the off-gas recombiner to ensure that all excess hydrogen in the off-gas stream is recombined.

The Hope Creek Plant uses sodium hypochlorite for cooling water system treatment. This eliminates the potential hazard associated with a simultaneous chlorine (also used for cooling water treatment) and hydrogen release.

We find that the proposed handling of the hydrogen for the test is in accordance with the BWROG (1987 Revision) "Guidelines for Permanent Hydrogen Water Chemistry Installations," meet General Design Criteria 3, and Branch Technical Position CMEB 9.5.1 of NUREG 0800 and are, therefore, acceptable.

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. 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 (52 FR 24557) on July 1, 1987, and a second notice (52 FR 26596) on July 15, 1987, 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: F. Witt

Dated: August 17, 1987