June 5, 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: INCREASE THE SURVEILLANCE TEST INTERVALS AND ALLOWABLE OUT-OF-SERVICE TIMES FOR THE REACTOR PROTECTION SYSTEM (TAC NO. 72699)

Re: HOPE CREEK GENERATING STATION

The Commission has issued the enclosed Amendment No. 26 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated February 6, 1989 supplemented by letter dated May 4, 1989. The supplemental information clarifies, and does not change, the technical content of the original change request.

This amendment increases the surveillance test intervals (STIs) and allowable out-of-service times (AOTs) for the reactor protection system.

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

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Enclosures:
1. Amendment No. 26 to License No. NPF-57
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION w/enclosures:

Docket File	MO'Brien (2)	Wanda Jones	SVarga
NRC PDR	OGC	EButcher	BBoger
Local PDR	DHagan	Brent Clayton	EWenzinger
PDI-2 Reading	EJordan	ACRS (10)	JMauck
WButler	BGrimes	CMiles, GPA/PA	CShiraki(3)
SBrown	TMeek (4)	RDiggs, ARM/LFMB	

DFol

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5/10/89	5/ /89	6/5/89	5/31/89	5/18/89

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

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2. Safety Evaluation

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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

Mr. David M. Scott, Chief
Bureau of Nuclear Engineering
Division of Environmental Quality
Department of Environmental Protection
State of New Jersey
CN 411
Trenton, New Jersey 08625

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. 26
License No. NPF-57

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

   A. The application for amendment filed by the Public Service Electric & Gas Company (PSE&G) dated February 6, 1989, as supplemented by letter dated May 4, 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. 26, 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

Mohan C Thadani

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: June 5, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 26

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 3-1	3/4 3-1
3/4 3-2*	3/4 3-2*
3/4 3-5	3/4 3-5
3/4 3-6*	3/4 3-6*
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
B 3/4 3-1	B 3/4 3-1
B 3/4 3-2*	B 3/4 3-2*

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within twelve hours. The provisions of Specification 3.0.4 are not applicable.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel 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 4.3.1.1-1.

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

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 6 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

**If more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause the Trip Function to occur.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors(b):			
a. Neutron Flux - High	2 3, 4 5(c)	3 2 3(d)	1 2 3
b. Inoperative	2 3, 4 5	3 2 3(d)	1 2 3
2. Average Power Range Monitor(e):			
a. Neutron Flux - Upscale, Setdown	2 3, 4 5(c)	2 2 2(d)	1 2 3
b. Flow Biased Simulated Thermal Power - Upscale	1	2	4
c. Fixed Neutron Flux - Upscale	1	2	4
d. Inoperative	1, 2 3, 4 5(c)	2 2 2(d)	1 2 3
e. Downscale	1(g)	2	4
3. Reactor Vessel Steam Dome Pressure - High	1, 2(f)	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	1(g)	4	4

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 < 153.3 psig** equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER. To allow for instrument accuracy, calibration, and drift, a setpoint of ≤ 132.4 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.

**Initial setpoint. Final setpoint to be determined during the startup test program.

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	≤ 0.09**
c. Fixed Neutron Flux - Upscale	≤ 0.09
d. Inoperative	NA
e. Downscale	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 0.55
4. Reactor Vessel Water Level - Low, Level 3	≤ 1.05
5. Main Steam Line Isolation Valve - Closure	≤ 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	≤ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≤ 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 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U[b],S	S/U[c], W	R	2
	S	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor[f]:				
a. Neutron Flux - Upscale, Setdown	S/U[b],S	S/U[c], W	SA	2
	S	W	SA	3, 4, 5
b. Flow Biased Simulated Thermal Power - Upscale	S,D[g]	S/U[c], Q	W[d][e],SA,R[h]	1
c. Fixed Neutron Flux - Upscale	S	S/U[c], Q	W[d], SA	1
d. Inoperative	NA	Q	NA	1, 2, 3, 4, 5
e. Downscale	S	W	SA	1
3. Reactor Vessel Steam Dome Pressure - High	S	Q[k]	R	1, 2
4. Reactor Vessel Water Level - Low, Level 3	S	Q[k]	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
6. Main Steam Line Radiation - High, High	S	Q	R	1, 2[i]
7. Drywell Pressure - High	S	Q[k]	R	1, 2

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High				
a. Float Switch	NA	Q	R	1, 2, 5(j)
b. Level Transmitter/Trip Unit	S	Q(k)	R	1, 5(j)
9. Turbine Stop Valve - Closure	NA	Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	Q	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least ½ decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least ½ decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days.

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

(g) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing recirculation loop flow (APRM % flow).

(h) This calibration shall consist of verifying the 6 ± 0.6 second simulated thermal power time constant.

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

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

(k) Verify the tripset point of the trip unit at least once per 92 days.

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. 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.26 TO FACILITY OPERATING LICENSE NO. NPF-57

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated February 6, 1989 and supplemented on May 4, 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 increase the surveillance test intervals (STIs) and allowable out-of-service times (AOTs) for the Reactor Protection System in accordance with General Electric Company Licensing Topical Report (LTR) NEDC-30851P-A. The supplemental information clarifies, and does not change, the technical content of the original change request. Thus it did not alter the action noticed, or affect the initial determination published, in the Federal Register on May 3, 1989.

2.0 EVALUATION

The proposed changes reflect those standard TS revisions contained in NEDC-30851P-A which, based upon probabilistic analyses, justify the identified time extensions by reducing the potential for: 1) unnecessary plant scrams; 2) excessive equipment test cycles; and 3) diversion of personnel and resources on unnecessary testing. The NRC staff has reviewed and approved this Licensing Topical Report in the letter, and accompanying Safety Evaluation Report (SER), from A. C. Thadani (NRC) to T. A. Pickens (BWR Owners Group) dated July 15, 1987.

PSE&G has extended the generic analysis completed by the BWR Owners Group to HCGS by completing the required plant specific analysis. As stated in the NRC's SER for Licensing Topical Report NEDC-30851P-A, three issues must be addressed to justify the applicability of the generic analysis to individual plants when specific facility Technical Specifications are considered for revision.

1. Confirm the applicability of the generic analysis to the specific facility.

Licensing Topical Report NEDC-30851P-A, Appendix L identifies PSE&G as a participating utility in the development of the RPS Technical Specification Improvement Analysis. Section 7.4 specifies that although "the evaluation found various differences between the RPS configuration of various plants and the generic plant....the generic results can be applied to plants in the BWROG Technical Specifications Improvement Program." Therefore, the generic analysis contained within the referenced report is applicable to HCGS.

2. Demonstrate that the drift characteristics for RPS channel instrumentation are bounded by the assumptions used in NEDC-30851P-A when the functional test interval is extended from monthly to quarterly.

PSE&G utilizes a setpoint calculation methodology for the RPS instrumentation which calculates the Technical Specification Trip Setpoint by subtracting the Loop Drift from the Technical Specification Allowable Value. This drift varies with time and hence the length of the surveillance interval is a factor in determining total instrument drift. Therefore, the current drift information provided by the equipment vendors and the applicable setpoint calculations for the HCGS RPS have been reviewed using the revised STIs as an input function. Even with the proposed increase in the surveillance interval, all instrumentation remain within their current Technical Specification Allowable Values. As a result it can be concluded that the proposed increases in the surveillance intervals do not require any corresponding changes to the RPS setpoints because the drift characteristics for RPS channel instrumentation are bounded by the assumptions used in NEDC-30851P-A when the functional test interval is extended from weekly/monthly to quarterly.

3. Confirm that the differences between the parts of the RPS that perform the trip functions in the plant and those of the base case plant were included in the specific analysis done using the procedures of Appendix K to NEDC-30851P-A.

In the General Electric (GE) Company Report MDE-85-0485 Revision 1 dated August 1988, the Technical Specification Improvement Analysis for the Reactor Protection System for Hope Creek Generating Station, the generic study completed in Licensing Topical Report NEDC-30851 for modifying the RPS was extended to HCGS. The GE report utilizes the procedures of Licensing Topical Report NEDC-30851P, Appendix K to identify and evaluate the differences between the parts of RPS that perform the trip functions at HCGS and those of the base case plant. The results indicate that while the RPS configuration for HCGS has several differences compared to the configuration in the base case, the differences and their impact do not significantly affect the applicability of the Technical Specifications changes developed by the generic efforts of Licensing Topical Report NEDC-30851P. Therefore, the conclusions reached in NEDC-30851P apply to HCGS and the plant-specific changes contained in this request are bounded by both the generic analysis and the NRC's SER.

3.0 RESULTS OF EVALUATION

Based on the evaluation above, the staff finds that HCGS has met the plant specific conditions to apply the results of General Electric Company's Topical Report NEDC-30851P-A to the Hope Creek Generating Station.

4.0 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.

5.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (54 FR 18956) on May 3, 1989 and consulted with the State of New Jersey. No public comments were received. The comment from the Bureau of Nuclear Engineering of the State of New Jersey and its resolution appear below:

Comment: If Public Service Electric & Gas (PSE&G) uses a method for calculating set point drift that is different from that used by the Boiling Water Reactor Owners Group (BWROG), are their method and their conclusions valid?

Resolution: As noted by PSE&G License Change Request 89-02, Attachment 1, paragraph 2, General Electric Company's Licensing Topical Report NEDC-30851P does not contain quantitative instrument drift assumptions. Hence, additional guidance was provided in an NRC letter dated April 27, 1988. The letter specifies that licensees must confirm that the setpoint drift which could be expected under the extended Surveillance Test Intervals (STIs) has been studied and either (1) has been shown to remain within the existing allowance in the Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) setpoint calculation or (2) that the allowance and setpoint have been adjusted to account for the additional expected drift. In License Change Request 89-02, paragraph 2 of Attachment 1, PSE&G confirmed that all instrumentation remained within their current Technical Specification allowable values. This confirmation satisfies the requirements of the NRC letter dated April 27, 1988.

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. Y. Shiraki

Dated: June 5, 1989