Mr. Harold W. Keiser
Chief Nuclear Officer & P[illegible]dent-
Nuclear Business Unit
Public Service Electric & Gas
Company
Post Office Box 236
Hancocks Bridge, NJ 08038

F[illegible]uary 10, 1999

SUBJECT: HOPE CREEK GENERATING STATION, ISSUANCE OF AMENDMENT, SAFETY RELIEF VALVE SETPOINT TOLERANCE CHANGE (TAC NO. MA1674)

Dear Mr. Keiser:

The Commission has issued the enclosed Amendment No. 115 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 28, 1998, as supplemented September 29, 1998, and December 8, 1998.

This amendment revises Technical Specification (TS) 3.4.2.1 to replace the ±1% setpoint tolerance limit for safety/relief valves (SRVs) with a ±3% setpoint tolerance limit. In addition, the amendment revises TS 4.4.2.2 to state that all SRVs will be re-certified to meet a ±1% tolerance prior to returning the valves to service after setpoint 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/

Richard B. Ennis, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures: 1. Amendment No. 115 to License No. NPF-57
2. Safety Evaluation

cc w/encls: See next page

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DFOI

Mr. Harold W. Keiser
Chief Nuclear Officer & President-
Nuclear Business Unit
Public Service Electric & Gas
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Post Office Box 236
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February 10, 1999

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*See previous concurrence

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

February 10, 1999

Mr. Harold W. Keiser
Chief Nuclear Officer & President-
Nuclear Business Unit
Public Service Electric & Gas
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Post Office Box 236
Hancocks Bridge, NJ 08038

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Sincerely,

RB Ennis

Richard B. Ennis, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures: 1. Amendment No. 115 to License No. NPF-57
2. Safety Evaluation

cc w/encls: See next page

Mr. Harold W. Keiser
Public Service Electric & Gas
Company

Hope Creek Generating Station

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 115
License No. NPF-57

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

   A. The application for amendment filed by the Public Service Electric & Gas Company (PSE&G) dated April 28, 1998, as supplemented September 29, 1998, and December 8, 1998, 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. 115, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSE&G shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance, to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

William M. Dean

William M. Dean, 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: February 10, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 115

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.

Remove	Insert
3/4 4-7	3/4 4-7
3/4 4-8	3/4 4-8

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 ±3%
5	safety-relief valves @ 1120 psig ±3%
5	safety-relief valves @ 1130 psig ±3%

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 3.4.2.2, 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 ½ of the safety relief valve pilot stage assemblies 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 valve pilot stage assemblies 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. All safety relief valves will be re-certified to meet a ±1% tolerance prior to returning the valves to service after setpoint testing.

4.4.2.3 The safety relief valve main (mechanical) stage assemblies shall be set pressure tested, reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations at least once every 5 years.

---

* 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 115 TO FACILITY OPERATING LICENSE NO. NPF-57

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated April 28, 1998 (Reference 1), as supplemented September 29, 1998, (Reference 5), and December 8, 1998, (Reference 6), the Public Service Electric & Gas Company (the licensee) submitted a request for changes to the Hope Creek Generating Station (HCGS), Technical Specifications (TSs). The requested changes would revise TS 3.4.2.1 to replace the ±1% setpoint tolerance limit for safety/relief valves (SRVs) with a ±3% setpoint tolerance limit. In addition, the amendment revises TS 4.4.2.2 to state that all SRVs will be re-certified to meet a ±1% tolerance prior to returning the valves to service after setpoint testing. The September 29, 1998, and December 8, 1998, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original Federal Register Notice.

Reference 1 included General Electric proprietary report NEDC-32511P, "Safety Review for Hope Creek Generating Station Safety/Relief Valve Tolerance Analyses." A revised version of this report and a non-proprietary version of the same report (NEDO-32511) were submitted by a letter from the licensee dated October 29, 1998 (Reference 7).

2.0 BACKGROUND

The proposed change does not alter the SRV lift setpoints, the SRV lift setpoint test frequency, or the number of SRVs required to be operable. Also, the proposed change requires the as-left safety valve function settings to be within ±1% of the specified nominal lift setpoints prior to installation after testing. The staff has previously granted approval to individual BWRs to increase the as-found SRV tolerance to ±3%. The basis for the approval was a staff safety evaluation report (SER) for a licensing topical report (Reference 3) evaluating the setpoint tolerance increase. The staff's SER (Reference 2) included six conditions which must be addressed on a plant-specific basis for licensees applying for the increased SRV setpoint tolerance:

(a) Transient analysis of all abnormal operational occurrences as described in NEDC-31753P (Reference 3), should be performed utilizing a ±3% tolerance for the safety mode of spring safety valves (SSVs) and SRVs. In addition, the standard reload methodology (or other method approved by the staff) should be used for this analysis.

(b) Analysis of the design basis overpressurization event using the 3% tolerance limit for the SRV setpoint is required to confirm that the vessel pressure does not exceed the ASME Boiler and Pressure Vessel Code upset limit.

(c) The plant-specific analysis described in items (a) and (b) should assure that the number of SSVs, SRVs, and relief valves (RVs) included in the analyses correspond to the number of valves required to be operable in the technical specification.

(d) Reevaluation of the performance of high pressure systems (pump capacity, discharge pressure, etc.), motor operated valves, and vessel instrumentation and associated piping must be completed, considering the 3% tolerance limit.

(e) Evaluation of the 3% tolerance on any plant-specific alternate operating modes (e.g., increased core flow, extended operating domain, etc.) should be completed.

(f) Evaluation of the effect of the 3% tolerance limit on the containment response during loss of coolant accidents and the hydrodynamic loads on the SRV discharge lines and containment should be completed.

3.0 EVALUATION

The safety objective of the SRVs is to prevent overpressurization of the nuclear system. This protects the nuclear system process barrier from failure which could result in the uncontrolled release of fission products. The pressure relief system at HCGS includes 14 SRVs, arranged into 3 setpoint groupings: 1 group of 4 SRVs set at 1108 psig, the second group of 5 SRVs set at 1120 psig, and the third group of 5 SRVs set at 1130 psig. Two valves of the total 14 are also part of the low-low set relief logic function. The low-low settings are 1017 psig and 1047 psig. The low-low set arming signal seals in lower automatic reopening and closing pressure setpoints for the selected two SRVs. The lower close setpoint pressure valve ensures that the low-low set SRVs remain open longer than any other SRV that may have opened. The low-low set relief logic feature reduces the probability of opening more than one valve and thereby reduces the probability of a stuck open SRV.

The existing TSs provide a ±1% as-found tolerance and ±1% as-left setpoint tolerance. The proposed modifications would provide a ±3% as-found tolerance and ±1% as-left setpoint tolerance. The licensee's submittal was evaluated against the generic SER described above.

3.1 Transient Analysis / Reload Methodology

The licensee must consider the impact of the tolerance increase on abnormal operational transients (AOTs). For HCGS, analysis of AOTs has been conducted utilizing the 3% tolerance and with 13 of the total 14 SRVs in service. The transient which generates the limiting decrease in critical power ratio (CPR) for current cycle 7 is the turbine trip without bypass.

The analysis showed that the thermal limits of the limiting transient would not be affected by the relaxation of SRV setpoint tolerance. Further, other transient events remain non-limiting and bounded by the above event. The NRC-approved licensing analysis methodology was used for the analysis (Reference 4).

The licensee evaluated the uncertainty associated with the testing of the SRVs and its effect on the transient analysis to support this TS change (Reference 5). The licensee stated that the ANSI/ASME OM-1987, requires a setpoint testing uncertainty of +2% to -1% with an overall combined accuracy not to exceed +1% to -2% for the indicated (measured) setpoint. The licensee stated that testing vendors who perform the testing typically use instruments with accuracies of ±0.15% to ensure that this requirement is met. The licensee further stated that this uncertainty equates to approximately 1.7 psi for any SRV, which is small compared to the allowed positive as-found setpoint drift of approximately 33 psi. The licensee also stated that the setpoint testing instrument inaccuracy will be accounted for in the determination of the as-found SRV setpoints. Therefore, the licensee concluded that the safety analysis remains bounding even though the setpoint testing instrument inaccuracy is not specifically accounted for. The staff finds that the setpoint testing uncertainty is small compared to both the allowable tolerance and the margin in the analysis and that this is acceptable.

3.2 Analysis of the Design Basis Overpressurization Event

The licensee is required to reevaluate the limiting design basis pressurization transient using the 3% tolerance limit to confirm that the vessel pressure does not exceed the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code upset limit. The ASME Boiler and Pressure Vessel Code Section III permits pressure transients up to 10% over design pressure (110% x 1250 psig = 1375 psig). The limiting pressurization AOT analyzed is a Main Steam Isolation Valve (MSIV) closure event occurring at the end of full power life without credit for a reactor trip on MSIV position sensing. An NRC-approved code was used for the analyses. The licensee analyzed the MSIV closure event with the 3% tolerance and calculated the maximum vessel pressure to be 1331 psig. This is within the 1375 psig ASME limit, and is acceptable to the staff.

3.3 TS Operability Statement for SRVs

The licensee has stated that all plant-specific analyses have been conducted with the number of SRVs included in the analyses corresponding to the number of valves required to be operable in TS. The analysis took credit only for 13 of the 14 SRVs required by the TS. This is acceptable to the staff.

As reported in Licensee Event Report 97-024, the HCGS SRVs have experienced several occurrences of positive setpoint drift in excess of the +3% used in the licensee's analysis to support this TS change. The cause of the drift has been determined to be corrosion bonding between the pilot disk and its seat. As a corrective action, the licensee has stated that it has installed platinum ion-beam implanted pilot valve disks in 7 of the 14 plant SRVs. The licensee is planning to install additional platinum ion-beam implanted pilot disks in the remaining 7 SRVs during the next refueling outage (RFO8) such that all 14 plant SRVs would have platinum ion-beam implanted pilot disks. In discussions held on October 15, 1998, the licensee informed the staff that it actually has 11 spare SRVs with the platinum ion-beam implanted pilot disks, in the event some of the existing SRV pilot disks require refurbishing. The licensee believes this modification will effectively reduce SRV setpoint drift resulting from corrosion bonding. The platinum ion-beam implanted pilot valve disks have significantly reduced the SRV setpoint drift experienced at another BWR plant site having SRVs of the same design as those at HCGS.

The staff finds that the licensee's corrective action is appropriate and consistent with the proposed TS change to the SRV setpoint tolerance.

3.4 Reevaluation of the Performance of High Pressure Systems

3.4.1 System Performance

The licensee must also reevaluate performance of high pressure systems (pump capacity, discharge pressure, etc.), considering the 3% tolerance limit. HCGS has three systems which are required to inject to the vessel at high pressure conditions: High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC), and Standby Liquid Control (SLC). The most significant impact is the increased reactor pressure specified for system operation. The systems performances were evaluated for the new reactor pressure of 1141 psig from 1120 psig. The HPCI system was determined to have the capability to inject its design flow of 5600 gpm to the vessel at the new maximum pressure of 1141 psig without any changes. The HPCI system design injection time of 35 seconds is unchanged since the maximum rated speed for the pump has not changed and there is no change in the pump total dynamic head (TDH). The HPCI system was found to have the capability to deliver the required flow.

The RCIC system performance was evaluated for the new reactor pressure of 1141 psig from 1120 psig. The ability of the RCIC pump to inject its design flow into the reactor is not affected by this change. The pump TDH required for system injection with the SRV setpoint change is within the design capability of the pump. The maximum steam flow is changed from 33,000 lb/hr to 33,700 lb/hr. The RCIC turbine has sufficient pressure margin at the higher reactor pressures to compensate for the expected increase in a steam line pressure drop with the higher steam flow rates. The RCIC turbine/pump maximum speed is unchanged in order for the RCIC system to perform at the new maximum reactor operating pressure. The RCIC was found to have the capability to deliver the required flow of 600 gpm at the increased reactor pressure resulting from relaxation of the SRV setpoint tolerance.

The standby liquid control system (SLCS) is an automatically operated system that will pump a sodium pentaborate solution into the reactor pressure vessel for the safe shutdown of the reactor. The SLCS pumps are positive displacement type pumps which have a nearly constant flow characteristic with increasing discharge pressure. The design pressure of the system is 1400 psig which is well above the calculated maximum system operating pressure. The increased operating pressure of the system results in a SLCS relief valve margin of 103 psi compared with the required margin of 75 psi for the system to reliably provide full pump discharge flow to the reactor. The SLCS system was determined to have the capability to inject boron into the vessel at its design flow rate.

3.4.2 Evaluation of Motor-Operated Valves

In support of the SRV tolerance increase from ±1% to ±3%, the licensee reviewed the motor-operated valves (MOVs) in the HPCI, RCIC and SLCS systems. This evaluation concluded that the plant high pressure systems will perform their required functions satisfactorily with the potential increase in differential pressure.

3.5 Alternate Operating Modes

The licensee must also evaluate the increased tolerance on any plant-specific alternate operating modes (e.g., increased core flow, extended operating domain, etc.) The analyses included evaluations for the currently approved operating domains: Extended Load Line Limit Analysis, Increased Core Flow, and Single Loop Operation. As discussed in Reference 6, these plant-specific alternate operating modes were considered in determining the most restrictive analytical conditions (i.e., the most limiting operating mode) for performing the analyses associated with the proposed TS changes. The staff finds that the consideration of these alternate operating modes in the analyses is acceptable.

3.6 Containment Response/Hydrodynamic Loads

The original licensing basis for Hope Creek was three distinctive setpoints with a tolerance of + or -1%. This tolerance level resulted in three distinctive bands. It can be seen by looking at the minimum and maximum values of acceptable lifting SRV pressures. The following is this information for each of the three distinctive setpoints.

1108 psig with a band of 1096.92 psig to 1119.08 psig

1120 psig with a band of 1108.80 psig to 1131.20 psig

1130 psig with a band of 1119.90 psig to 1142.50 psig

Between the lowest and the mid-range bands there is a slight overlap of about 11 psig and a slightly greater overlap for the higher set. However, for the most part one can see that banding is retained. But, the licensee's proposal to increase the SRV tolerance from ±1% to ±3% mixes the setpoint bands for each setpoint with the other two. A similar presentation of the banding setpoints including the tolerances produces the following results.

1108 psig with an acceptable range of 1074.76 psig to 1141.24 psig

1120 psig with an acceptable range of 1086.40 psig to 1153.60 psig

1130 psig with an acceptable range of 1096.10 psig to 1163.90 psig

This 3% tolerance removes the three distinctive setpoints for the SRV to lift. In fact, over an acceptable range of 67 psi only about 12 psi is not overlapped by the next higher band and 22 psi for the third band overlap. In other words, the overlap is so significant that an individual SRV setpoint could be within the band range of all individual SRV lifting pressures.

The original design for the Hope Creek SRVs allowed for a limited number of SRV actuations for any given sequence. The current request results in one band for all SRVs. There is a possibility that this may result in the actuation of all SRVs simultaneously. When the tolerance level is changed from ±1% to ±3% therefore the concern is whether or not the increased number of SRV actuations would violate the original licensing basis. Specifically, whether the limiting structural loading analysis assumed that all 14 SRVs opened simultaneously. The licensee has confirmed that the limiting structural loading calculations have been based on 14 SRVs opening at the same time and in phase.

3.7 Emergency Core Cooling System (ECCS)/LOCA Performance

The HCGS LOCA analysis in the Updated Final Safety Analysis Report was reviewed by General Electric for the licensee to determine the effect of an increase in SRV opening pressures on ECCS performance. The limiting break LOCA (the DBA recirculation break), the small break and the steam line break outside containment events were evaluated to determine the effects of the increased SRV setpoint tolerance. Peak cladding temperatures for the small break and the steam line breaks outside containment are not changed significantly and they are non-limiting. The acceptance criteria given in 10 CFR 50.46 are still satisfied for all break sizes and locations and hence the setpoint tolerance change for LOCA considerations is acceptable.

3.8 Effect on Anticipated Transient Without Scram (ATWS) Events

The Main Steam Isolation Valve closure under ATWS conditions was reevaluated to support the current condition of 13 of the 14 SRVs operable with the requested relaxation in SRV setpoint tolerance to ± 3%. The results of the analysis, using the REDY code, show that the vessel pressure reaches a maximum of 1425 psig, which is within the vessel overpressure criterion of 1500 psig for ATWS events. The long-term effect on suppression pool temperature due to 3% SRV tolerance is negligible because there is little change in the total energy discharged to the pool. The staff concludes that the results of the analyses are acceptable.

3.9 Technical Specification Changes

In TS 3.4.2.1 , TS Page 3/4 4-7, the setpoint tolerance is changed from ±1% to ±3%. This is acceptable as described in this SE.

In TS 4.4.2.2, TS Page 3/4 4-8, the following note is added: "All safety relief valves will be re-certified to meet a ±1% tolerance prior to returning the valves to service after setpoint testing." This is acceptable since it meets the intent of the Staff SER (Reference 2).

3.10 Conclusion

Based on the information provided by the licensee, the staff concludes that the plant will continue to satisfy the acceptance criteria for the limiting pressurization transient, AOTs and design-basis accidents. Therefore, the staff concludes that the proposed changes are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State Official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (63 FR 33108). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: G. Hammer
G. Thomas
A. Gill

Date: February 10, 1999

7.0 REFERENCES

1. Letter from E. C. Simpson (PSE&G) to USNRC, "Request for Change to Technical Specifications, Safety Relief Valve Setpoint Tolerances," dated April 28, 1998.

2. Letter from A. C. Thadani (NRC) to C. L. Tully (BWROG), "Acceptance for Referencing of Licensing Topical Report NEDC-31753P, 'BWROG In-Service Pressure Relief Technical Specification Licensing Topical Report,' " dated March 8, 1993.

3. General Electric Report NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," dated February 1990.

4. Letter from E. C. Simpson (PSE&G) to USNRC, Hope Creek Generating Station COLR, Cycle 7/Reload 6," dated October 17, 1997.

5. Letter from E. C. Simpson (PSE&G) to USNRC, "Supplement to a Request for Change to Technical Specifications, Safety Relief Valve Setpoint Tolerances," dated September 29, 1998.

6. Letter from E. C. Simpson (PSE&G) to USNRC, "Supplement to a Request for Change to Technical Specifications, Safety Relief Valve Setpoint Tolerances," dated December 8, 1998.

7. Letter from D. R. Powell (PSE&G) to USNRC, "Transmittal of General Electric Report NEDC-32511P and NEDO-32511," dated October 29, 1998.