November 1_ 1988

Docket No. 50-311

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

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Docket File
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Brent Clayton
OGC TMeek (4)
RGallo
HBalukjian
EJordan
WButler
CMiles, GPA/PA
RDiggs, ARM/LFMB

Dear Mr. Miltenberger:

SUBJECT: EMERGENCY TECHNICAL SPECIFICATION CHANGE, REACTOR COOLANT SYSTEM FLOW MEASUREMENT UNCERTAINTIES (TAC NO. 69804)

RE: SALEM GENERATING STATION, UNIT NO. 2

The Commission has issued the enclosed Amendment No. 64 to Facility Operating License No. DPR-75 for the Salem Generating Station, Unit No. 2. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 19, 1988, as supplemented on October 26, 1988. It was prepared and issued on an emergency basis to avoid an unnecessary shutdown following calorimetric calibrations after unit 2 restart.

This amendment consists of a change to the Technical Specifications to reduce the correction for flow measurement uncertainty from 3.5% to 2.2%.

The staff reviewed the circumstances associated with your request and concluded that you provided a sufficient basis for finding that the situation could not have been avoided by prior application. Therefore, in accordance with 10 CFR 50.91(a)(5), a valid emergency existed.

This amendment was authorized by telephone on November 3, 1988, and confirmed by letter on November 3, 1988.

A copy of our safety evaluation is also enclosed. Notice of Issuance and Final Determination of No Significant Hazards Consideration and Opportunity for Hearing will be included in the Commission's biweekly Federal Register notice.

Sincerely,
/s/
Bruce A. Boger, Assistant Director
for Region I Reactors
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 64 to License No. DPR-75
2. Safety Evaluation

cc w/enclosures:
See next page

*Previously Concurred

DF01
1/1

For JW per tel con P. Swetland

PDI-2/LA	PDI-2/PM*	OGC*	PDI-2/D*	Region I	ADR1
MO'Brien	JStone:mr	MYoung	WButler		BBoger
11/ /88	10/28/88	11/03/88	11/03/88	11/7/88	11/17/88

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PDI-2/LA	PDI-2/PM	OGC	PDI-2/D	Region I
MO'Brien	JStone:mr		WButler	
/ /88	10/28/88	11/3/88	11/3/88	11/2/88

w/ noted
per tel con P. Swetland
for

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 64 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment became effective on November 3, 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

Bruce A. Boger, Assistant Director
for Region I Reactors
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 17, 1988

*Previously Concurred

PDI-2/LA	PDI-2/PM*	OGC*	PDI-2/D*	Region I	ADRI
MO'Brien	JStone:mr	MYoung	WButler	per tel con P. Swetland	BBoger
11/14/88	10/28/88	11/03/88	11/03/88	for 11/2/88	11/17/88

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment ~~is~~ became effective on November 3, 1988 ~~as of its date of issuance.~~

K

FOR THE NUCLEAR REGULATORY COMMISSION

Bruce A. Boger, Assistant Director
for Region I Reactors
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance:

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D	Region I
MO'Brien	JStone:mr		WButler	
/ /88	10/28/88	11/3 /88	11/03/88	/ /88

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

November 17, 1988

Docket No. 50-311

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: EMERGENCY TECHNICAL SPECIFICATION CHANGE, REACTOR COOLANT SYSTEM FLOW MEASUREMENT UNCERTAINTIES (TAC NO. 69804)

RE: SALEM GENERATING STATION, UNIT NO. 2

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This amendment consists of a change to the Technical Specifications to reduce the correction for flow measurement uncertainty from 3.5% to 2.2%.

The staff reviewed the circumstances associated with your request and concluded that you provided a sufficient basis for finding that the situation could not have been avoided by prior application. Therefore, in accordance with 10 CFR 50.91(a)(5), a valid emergency existed.

This amendment was authorized by telephone on November 3, 1988, and confirmed by letter on November 3, 1988.

A copy of our safety evaluation is also enclosed. Notice of Issuance and Final Determination of No Significant Hazards Consideration and Opportunity for Hearing will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Bruce A. Boger

Bruce A. Boger, Assistant Director
for Region I Reactors
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 64 to
License No. DPR-75
2. Safety Evaluation

cc w/enclosures:
See next page

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

Salem Nuclear Generating Station

cc:

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Wilmington, DE 19899

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-311

SALEM GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 64
License No. DPR-75

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, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated October 19, 1988, as supplemented on October 26, 1988, 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. DPR-75 is hereby amended to read as follows:

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 64 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment became effective on November 3, 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

Bruce A. Boger

Bruce A. Boger, Assistant Director
for Region I Reactors
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 17, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 64

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 2-9	3/4 2-9
3/4 2-11	3/4 2-11
B 3/4 2-5	B 3/4 2-5

3/4.2.3 RCS FLOW RATE AND R

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R_1, R_2 shall be maintained within the region of allowable operation shown on Figure 3.2-3 for 4 loop operation.

where:

a. $R_1 = \dfrac{F^N_{\Delta H}}{1.49\ [1.0 + 0.3\ (1.0-P)]}$

b. $R_2 = \dfrac{R_1}{[1-RBP(BU)]}$

c. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

d. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3 includes penalties for undetected feedwater venturi fouling of 0.1% and for measurement uncertainties of 2.2% for flow and 4% for incore measurement of $F^N_{\Delta H}$.

e. RBP(BU) = Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-4, where a region is defined as those assemblies with the same loading date (reloads) or enrichment (first core).

APPLICABILITY: MODE 1

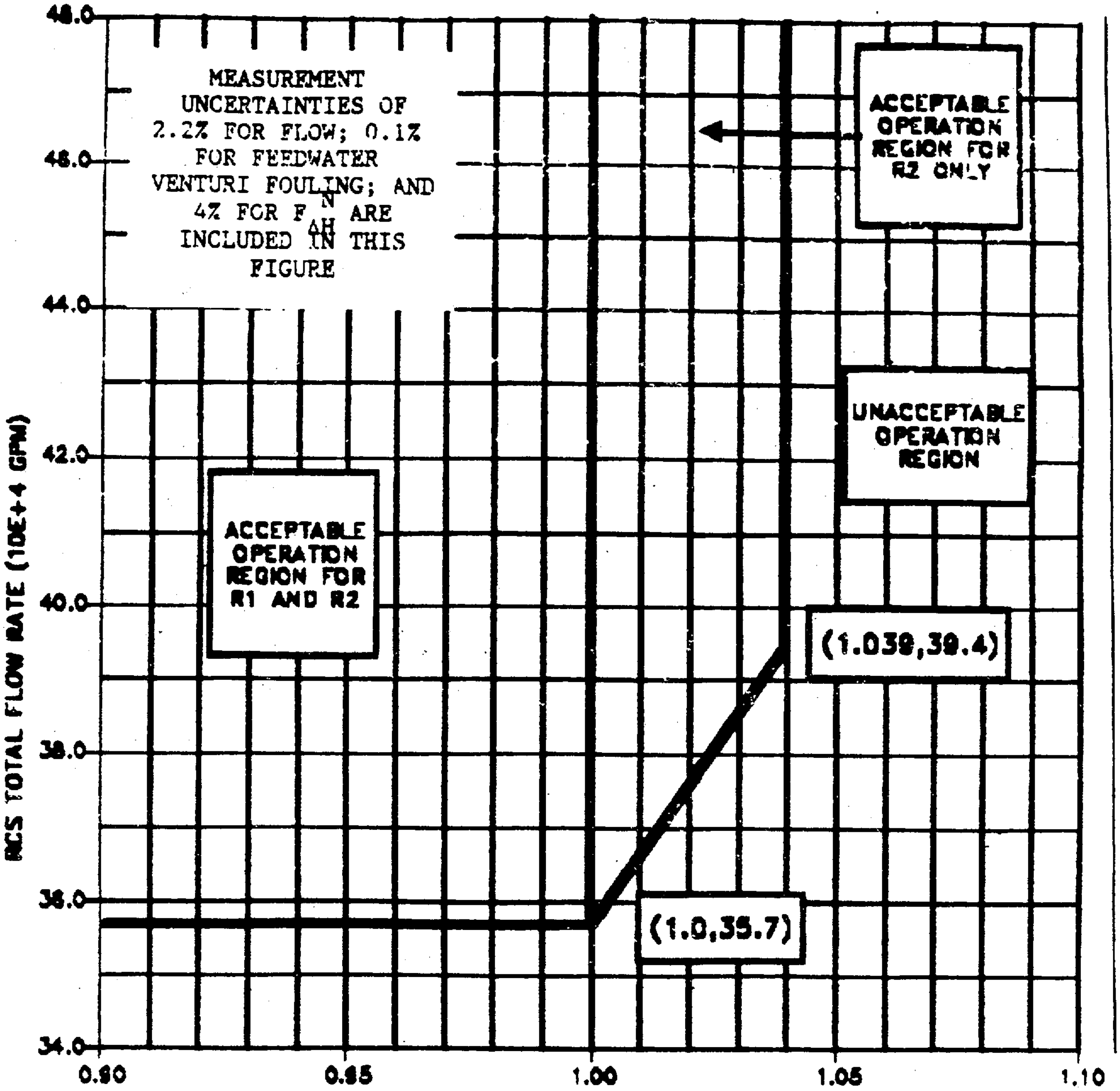
ACTION:

With the combination of RCS total flow rate and R_1, R_2 outside the region of acceptable operation shown on Figure 3.2-3:

a. Within 2 hours:

1. Either restore the combination of RCS total flow rate and R_1, R_2 to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High trip setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.



$$R1 = F^N_{\Delta H}/1.49[1.0+0.3(1.0 - P)]$$

$$R2 = R1 /[1 - RBP(BU)]$$

Figure 3.2-3
RCS TOTAL FLOWRATE VERSUS R - FOUR LOOPS IN OPERATION

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR $F_Q(Z)$, RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

event. The penalties applied to $F^N_{\Delta H}$ to account for Rod Bow (Figure 3.2-4) as a function of burnup are consistent with those described in Mr. John F. Stolz's (NRC) letter to T. M. Anderson (Westinghouse) dated April 5, 1979 and W 8691 Rev. 1 (partial rod bow test data).

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figure 3.2-3. Measurement errors of 2.2% for Reactor Coolant System total flow rate, 0.1% for feedwater venturi fouling, and 4% for $F^N_{\Delta H}$ have been allowed for in the determination of the design DNBR value.

The measurement error for Reactor Coolant System total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is included in Figure 3.2-3. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12 hour period surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown in Figure 3.2-3.

The radial peaking factor $F_{xy}(Z)$ is measured periodically to provide assurance that the hot channel factor, FQ(Z), remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F^{RTP}_{xy}), as provided in the Radial Peaking Factor Limit Report per specification 6.9.1.10, was determined from expected power control maneuvers over the full range of burnup conditions in the core.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 64 TO FACILITY OPERATING LICENSE NO. DRP-75

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

SALEM GENERATING STATION, UNIT NO. 2

DOCKET NO. 50-311

1.0 INTRODUCTION

By letter dated October 19, 1988 (Ref. 6) and supplemented by letter dated October 26, 1988 (Ref. 6) Public Service Electric and Gas Company (PSE&G), the licensee, requested an Emergency Licensing Amendment to Facility Operating License DPR-75 for Salem Generating Station Unit 2. The proposed amendment presented changes to the Technical Specifications due to a reduction of the flow measurement uncertainty value from 3.5% to 2.2% (not including a 0.1% feedwater venturi fouling penalty).

2.0 BACKGROUND

During a Salem Unit 2 steam generator tube eddy current test completed during the current refueling outage, some defective tubes were found. PSE&G committed to plugging 100% of row 1 tubes and additional tubes in row 2 identified as defective. The additional resistance from plugging the steam generator tubes results in a reduction in RCS flow. PSE&G estimates that the RCS flow will fall below, or be only marginally above the minimum allowable flow (MAF) value. The Limiting Condition for Operation (LCO) of Specification 3/4.2.3 requires the combination of RCS flow and R1 and R2 to be restored to within the allowable operating region of Figure 3.2-3 in 24 hours or that reactor power be reduced to less than 5% within the next 2 hours. Since Salem Unit 2 is scheduled to be synchronized with the grid on October 28, 1988 an Emergency License Amendment was requested. The MAF rate in Figure 3.2-3 is determined by applying the flow measurement uncertainty (FMU) in the following equation to the Thermal Design Flow (TDF) rate:

$$MAF = TDF * (1 + FMU)$$

The value for TDF is 87,300 gpm per loop or 349,200 gpm total and the current value of FMU is 0.035 (3.5%). The reduced value of FMU presented in the analysis (Ref. 2) is 0.023 (2.3%) including a 0.1% feedwater venturi fouling penalty. The current indicated flow from the latest calorimetric heat balance is 368,665 gpm (Ref. 6). However, with the increased resistance due to steam

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generator tube plugging, this value will decrease. By reducing the Technical Specifications value of FMU from 3.5% to 2.3%, the corresponding MAF will be reduced from 361,422 gpm to 357,232 gpm, respectively, for a difference of 4,190 gpm. The implementation of this 4,190 gpm reduction in MAF by means of a reduction of the FMU to 2.3% in the Technical Specifications will provide assurance that the reduced flow from steam generator tube plugging will not result in a Tech Spec violation for MAF.

3.0 ANALYSIS

The analysis for flow measurement uncertainty (Ref. 2.) was previously supplied for Salem Unit 2 as part of their submission for RTD bypass system removal. However, a reduction in the flow measurement uncertainty value was not requested at that time and was therefore not reviewed by the NRC. This analysis (Ref. 2) used the same methodology as previously approved for the Shearon Harris plant (Ref. 3) but differed slightly as the hot leg streaming error differs and because 3 RTDs are used in each hot leg for the Salem plant rather than 1. This is because the Salem plant has the RTD bypass system removed while the Shearon Harris plant analysis included a RTD bypass system. The licensee stated that the values used in the analysis are plant specific for Salem on the primary side and are bounding for the secondary side. To check on this, the licensee reevaluated the uncertainties for Feed Water Temperature and Differential Pressure, Steam Pressure and Pressurizer Pressure and made a Salem plant specific comparison with the values in reference 2 (WCAP-11579). This comparison confirmed that the WCAP-11579 analysis values were bounding.

The flow measurement analysis did not include any errors in the calibration of RTDs as the RTDs are recalibrated at the beginning of each reload using a cross calibration that is common to Westinghouse PWRs. The licensee stated (Ref. 6) that cross calibration is performed on each of the 8 RTDs in each RCS loop (i.e. 3 T-hot/3 spare and 1 T-cold/1 spare). The averaged value of the RTD readings is then compared to the individual RTD reading. Any RTD reading deviating from the average by greater than ± 0.5 °F is failed. Additionally, a cross comparison of the 4 loops is made using the averaged value of all T-hot RTDs. Any RTD reading deviating by greater than ± 2.0°F is failed. Any failed RTDs are either switched to a spare RTD, or if a spare is unavailable, the RTD is replaced.

The licensee stated that the dual element Weed RTD has a total uncertainty of + 0.7° F (Ref. 7). This value includes a drift (for 22.5 months) of + 0.4° F (Ref. 6 and 7) on top of the normal + 0.3° F accuracy (includes hysteresis and repeatability). NUREG/CR-4928 (Ref. 4) has addressed concern on the accuracy of RTDs and calibration. The licensee has confirmed (Ref. 6 and 7) their intent to replace 2 RTDs per refueling from Salem Unit 1 at each of the next two refueling outages. They will review the recalibration results from the RTDs removed, as well as other data anticipated to become available on the drift of the Weed RTDs, prior to making any subsequent long-range periodic RTD replacements. Since the replacement RTD would have to be within the allowable deviation from the averaged reading, verification of no significant systematic drift will be obtained.

The licensee stated (Ref. 1) that the reduced measurement uncertainty value would not affect any previous variable which inputs into a process control or reactor protection system control function.

The FMU analysis included the effect of measuring flow from the cold leg elbow taps. However, the Technical Specifications did not include the 0.1% penalty to account for feedwater venturi fouling. In response to questions, the licensee modified the Technical Specifications (Ref. 6) to include this correction. The staff has found the analysis for a flow measurement uncertainty of 2.2% (not including the 0.1% penalty for feedwater venturi fouling) to be acceptable.

References

1. Letter from S.E. Miltenberger, Public Service Electric and Gas Company (PSE&G), to USNRC, dated October 19, 1988.

2. WCAP-11579, "RTD Bypass Elimination Licensing Report for Salem Units 1 and 2," C.R. Tuley et. al., September 1987.

3. WCAP-11168 Rev. 1, "RCS Flow Uncertainty for Shearon Harris Unit 1," C.R. Tuley, W.N. Mooman, October 1986.

4. NUREG/CR-4428, "Degradation of Nuclear Plant Temperature Sensors," June 1987.

5. Letter from C.A. McNeill, Jr., PSE&G, to USNRC, dated October 1, 1987.

6. Letter from S.E. Miltenberger, PSE&G, to USNRC, dated October 26, 1988.

7. Letter from C.A. McNeill, Jr., PSE&G, to USNRC, dated September 2, 1987.

4.0 EMERGENCY CIRCUMSTANCES

During required Salem Unit 2 steam generator tube eddy current testing and the identification of numerous defective tubes, PSE&G committed to plugging 100% of row 1 tubes on all Salem, Unit 2 steam generators and any row 2 tubes identified as defective during supplemental testing. The result of plugging these tubes, as well as those non-row 1 tubes identified as defective during the testing, is a net reduction in available RCS flow. As a result, it is anticipated that upon performance of the RCS flow calorimetric following startup from the present refueling outage, RCS flow will fall below, or at best be only marginally above the minimum allowable value. The Limiting Condition for Operation of Specification 3/4.2.3 requires the combination of RCS flow and R1 and R2 be restored to within the allowable operating region of Figure 3.2-3 in 24 hours or that reactor power be reduced to less than 5% within the next 2 hours.

Without the proposed change Salem, Unit 2 could be forced into an unnecessary shutdown. Additionally, this condition could not have been reasonably foreseen prior to this time as it is a direct result of work accomplished during the present refueling outage.

Salem, Unit 2 is scheduled to be synchronized with the grid on November 4, 1988. It is therefore concluded that this change satisfies the criteria of 10 CFR 50.91(a)(5).

This amendment was authorized by telephone on November 3, 1988, and confirmed by letter on November 3, 1988.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated or;

(3) Involve a significant reduction in a margin of safety.

The licensee has determined and the staff agrees with the following:

1. The reduction in the uncertainty value is attributed to the reduced error associated with the modified RCS narrow range temperature monitoring system. The Chapter 15 accident analyses impacted by this modification were previously reviewed and approved by the NRC as amendments 84 and 56 to Salem, Units 1 and 2 licenses, respectively.

   The requested change does not result in a reduction of the RCS thermal design flow which was assumed for the purpose of accident analysis. The requested change therefore does not result in a value of DNBR which is less than the minimum design DNBR value identified in the Updated Final Safety Analysis Report.

   Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The correction is based on an analysis of flow measurement uncertainties. The correction does not affect any process variable which inputs to a process control or reactor protection system control function. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated.

3. An RCS Flow uncertainty error of 3.5% was assumed for the purpose of calculating a minimum allowable RCS flow rate for safe plant operation. The uncertainty correction provides a reference point from which the relative magnitude of the safety margin between measured flow rate and design thermal flow rate can be inferred. WCAP-11579 demonstrates that the total uncertainty associated with the modified RCS narrow range temperature monitoring system could be reduced to a conservative value of 2.3% from the existing value of 3.5%. If the uncertainty is not reduced, then a net increase in the margin of safety associated with the present thermal design flow value is achieved.

   The licensee does not intend to revise the value of thermal design flow used in the DNBR analysis. The requested change is only being applied to Figure 3.2-3 and the resultant value of minimum allowable RCE flow. The net result of the requested change is to redefine the reference point but maintain the magnitude of the existing margin of safety.

   It is therefore concluded that operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Based on the above discussion the staff concludes that this amendment meets the criteria and therefore does not involve a significant hazards consideration.

6.0 STATE CONSULTATION

The State of New Jersey was consulted on this matter and had no comments on the determination.

7.0 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 made a final no significant hazards finding with respect to this amendment. 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 staff has concluded, based on the considerations discussed above, that: (1) the amendment does not (a) significantly increase the probability or consequences of an accident previously evaluated, (b) increase the possibility of a new or different kind of accident from any previously evaluated or (c) significantly reduce a safety margin and, therefore, the amendment does not involve significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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 Contributors: H. Balukjian and J. Stone

Dated: November 17, 1988