December 22, 1997

Mr. Leon R. Eliason
Chief Nuclear Officer & President-
Nuclear Business Unit
Public Service Electric & Gas
Company
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NO. 1 (TAC NO. M99760)

Dear Mr. Eliason:

The Commission has issued the enclosed Amendment No. 202 to Facility Operating License No. DPR-70 for the Salem Nuclear Generating Station, Unit No. 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 6, 1997.

This amendment increases the allowable band for the control and shutdown rod demanded position versus indicated position from ± 12 steps to ± 18 steps when the power level is not greater than 85% rated thermal power. The same change was approved for Salem Unit 2 in Amendment No. 183, issued September 10, 1997, as an exigent amendment.

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/ J. Stolz for

Patrick D. Milano, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-272

Enclosures: 1. Amendment No. 202 to License No. DPR-70
2. Safety Evaluation

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

December 22, 1997

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John F. Stolz Jr

Patrick D. Milano, Senior Project Manager
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Office of Nuclear Reactor Regulation

Docket No. 50-272

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License No. DPR-70
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 202
License No. DPR-70

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 6, 1997, 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-70 is hereby amended to read as follows:

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(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 202 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: December 22, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 202

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 1-18	3/4 1-18
3/4 1-18a	3/4 1-18a
3/4 1-19	3/4 1-19
3/4 1-19a	3/4 1-19a
B 3/4 1-4	B 3/4 1-4
B 3/4 2-4	B 3/4 2-4

LIMITING CONDITION FOR OPERATION

==========

3.1.3.1 All full length (shutdown and control) rods, shall be OPERABLE and positioned within ± 18 steps (indicated position) when reactor power is ≤ 85% RATED THERMAL POWER, or ± 12 steps (indicated position) when reactor power is > 85% RATED THERMAL POWER, of their group step counter demand position within one hour after rod motion.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or mis-aligned from the group step counter demand position by more than ± 18 steps (indicated position) at ≤ 85% RATED THERMAL POWER or ± 12 steps (indicated position) at > 85% RATED THERMAL POWER, be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or mis-aligned from its group step counter demand position by more than ± 18 steps (indicated position) at ≤ 85% RATED THERMAL POWER or ± 12 steps (indicated position) at > 85% RATED THERMAL POWER, POWER OPERATION may continue provided that within one hour either:

1. The rod is restored to OPERABLE status within the above alignment requirements, or

2. The remainder of the rods in the bank with the inoperable rod are aligned to within ± 18 steps (indicated position) at ≤ 85% RATED THERMAL POWER or ± 12 steps (indicated position) at >85% RATED THERMAL POWER, of the inoperable rod while maintaining the rod sequence and insertion limits of Figures 3.1-1 and 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.5 during subsequent operation, or

3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

*See Special Test Exceptions 3.10.2 and 3.10.3.

a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours.

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER. THERMAL POWER shall be maintained less than or equal to 75% of RATED THERMAL POWER until compliance with ACTIONS 3.1.3.1.c.3.a and 3.1.3.1.c.3.c above are demonstrated.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the limits established in the limiting condition for operation at least once per 12 hours (allowing for one hour thermal soak after rod motion) except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2.1 The shutdown and control rod position indication systems shall be OPERABLE and capable of determining the actual and demanded rod positions as follows:

a. Analog rod position indicators, within one hour after rod motion (allowance for thermal soak);

All Shutdown Banks: ± 18 steps at ≤85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 200-228 steps.

Control Bank A: ± 18 steps at ≤85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 200-228 steps.

Control Bank B: ± 18 steps at ≤85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 160-228 steps.

Control Banks C and D: ± 18 steps at ≤85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal range of 0-228 steps.

b. Group demand counters; ± 2 steps of the pulsed output of the Slave Cycler Circuit over the withdrawal range of 0-228 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one analog rod position indicator per bank inoperable either:

1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and within one hour after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

b. With two or more analog rod position indicators per bank inoperable, within one hour restore the inoperable rod position indicator(s) to OPERABLE status or be in HOT STANDBY within the next 6 hours. A maximum of one rod position indicator per bank may remain inoperable following the hour, with Action (a) above being applicable from the original entry time into the LCO.

c. With a maximum of one group demand position indicator per bank inoperable either:

1. Verify that all analog rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 18 steps when reactor power is ≤ 85% RATED THERMAL POWER or if reactor power is > 85% RATED THERMAL POWER, 12 steps of each other at least once per 8 hours, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

==========

4.1.3.2.1.1 Each analog rod position indicator shall be determined to be OPERABLE by verifying that the demand position indication system and the rod position indication system agree within 18 steps when reactor power is ≤ 85% RATED THERMAL POWER or if reactor power is > 85% RATED THERMAL POWER, 12 steps (allowing for one hour thermal soak after rod motion) at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indication system at least once per 4 hours.

4.1.3.2.1.2 Each of the above required rod position indicator(s) shall be determined to be OPERABLE by performance of a CHANNEL calibration at least once per 18 months.

BASES

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% delta k/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2,600 gallons of 6,560 ppm borated water from the boric acid storage tanks or 7,100 gallons of 2,300 ppm borated water from the refueling water storage tank.

The 37,000 gallons limit in the refueling water storage tank for Modes 5 and 6 is based upon 21,210 gallons that is undetectable due to lower tap location, 8,550 gallons for instrument error, 7,100 gallons required for shutdown margin, and an additional 140 gallons due to rounding up.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod mis-alignment on associated accident analyses. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. OPERABLE condition for the analog rod position indicators is defined as being capable of indicating rod position to within the allowed rod misalignment relative to the bank demand position for a range of positions. For the Shutdown Banks, and Control Bank A this range is defined as the group demand counter indicated position between 0 and 30 steps withdrawn inclusive, and between 200 and 228 steps withdrawn inclusive. This permits the operator to verify that the control rods in these banks are either fully withdrawn or fully inserted, the normal operating modes for these banks. Knowledge of these banks positions in these ranges satisfies all accident analysis assumptions concerning their position. The range for control Bank B is defined as the group demand counter indicated position between 0 and 30 steps withdrawn inclusive, and between 160 and 228 steps withdrawn inclusive. For Control Banks C and D the range is defined as the group demand counter indicated position between 0 and 228 steps withdrawn. Comparison of the group demand counters to the bank insertion limits with verification of rod position with the analog rod position indicators (after thermal soak after rod motion) is sufficient verification that the control rods are above the insertion limits. The full out position will be specifically established for each cycle by the Reload Safety Analysis for that cycle. This position will be within the band established by "FULL WITHDRAWN" and will be administratively controlled. This band is allowable to minimize RCCA wear, pursuant to Information Notice 87-19.

BASES

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL AND RADIAL PEAKING FACTORS -

$F_Q(Z)$ AND $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors and RCS flow rate ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rod in a single group move together with no individual rod insertion differing from the group demand position by more than the allowed rod misalignment.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.

c. The control rod insertion limits of Specifications 3.1.3.4 and 3.1.3.5 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a through d above, are maintained.

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

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq 1.55/1.08$. The 8% allowance is based on the following considerations:

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 202 TO FACILITY OPERATING LICENSE NO. DPR-70

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

DOCKET NO. 50-272

1.0 INTRODUCTION

By letter dated October 6, 1997, the Public Service Electric & Gas Company (the licensee) submitted a request for changes to the Salem Nuclear Generating Station, Unit No. 1, Technical Specifications (TSs). The requested changes would increase the allowable band for control and shutdown rod demanded position versus indicated position from ± 12 steps to ± 18 steps when the power level is not greater than 85% rated thermal power. The changes have already been approved for Salem Unit 2 in Amendment No. 183, issued September 10, 1997, as an exigent amendment.

2.0 EVALUATION

The analog rod position indication system (ARPI) system is designed to an accuracy of 12 steps. Therefore, in order to guarantee a rod misalignment of less than 24 steps (12 steps misalignment plus 12 steps ARPI uncertainty), the individual ARPI readings must be no larger than 12 steps. In order to justify changing the misalignment limit to ± 18 steps, the licensee did evaluations for misalignments of up to 30 steps (18 steps indicated plus 12 steps uncertainty). The TS limits on peaking factors F_Q and F8H increase as the power level lowers. The increase in the limit for F_Q and F8H was used to accommodate the larger than ±12 steps misalignment at the reduced power levels. To justify the increase in allowable rod misalignment at a reduced power level, the following were evaluated:

1. reactivity control
2. control rod misoperation (dropped rods and static rod misalignments)
3. rod ejection
4. power operation with misaligned rods.

The principal tool used in the analysis was the Westinghouse PHOENIX-P/ANC core design system documented in References 2 and 3. For this analysis the changes in peaking factors rather than the absolute values of the peaking

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factors were of interest. For each case, calculations were performed for misalignments of ±24 and ±30 steps and compared to the corresponding non-misaligned reference case. The $F_{\Delta H}$ and F_Q for these cases were calculated and compared as a function of axial offset (AO) throughout the anticipated allowable range of operation. All calculations supporting this report used a Hot Full Power (HFP) AO band of ±15%.

The analysis was performed with two different models of the Salem core, the Unit 2 Cycle 10 core model and a "bounding" future cycle model. Applicability for each future cycle will be determined during the reload design process.

2.1 Reactivity Control

To demonstrate that reactivity control was acceptable with the additional allowed misalignment, the reactivity effect of a misaligned bank by an additional six steps was calculated for both core models at Hot Zero Power (HZP), HFP and part-power conditions. The change was found to be less than 100 pcm. These calculations were performed for End of Cycle (EOC) conditions since that represents the point in a cycle with the least available shutdown margin. For future cycles, if a cycle-specific calculation is not performed, the rod insertion allowance calculated as part of the reload safety evaluation will be conservatively increased by 120 pcm.

2.2 RCCA Misoperation Events

The RCCA misoperation events (dropped RCCAs and statically misaligned RCCAs) are initiated by the movement or displacement of one RCCA rod or bank from its normal position. These events result in reactivity and power distribution anomalies. A change in the number of steps of misalignment allowed does not affect the results of these events since these events bound the misalignment cases.

2.3 Rod Ejection

The rod ejection analysis is performed at HZP and HFP, Beginning of Cycle (BOC) and EOC conditions. The physics parameters of interest are the available trip worth following a rod ejection, the ejected rod worth, and the post-ejection F_Q. Calculations were performed for both core models. The results of these calculations showed that the maximum increases in F_Q and ejected rod worth were well within the margin on these parameters. For future cycles, if a cycle-specific analysis is not performed, the calculated ejected rod peak F_Q will be multiplied by 1.085 to bound the additional six steps of rod misalignment any time in the cycle. Likewise, the ejected rod worth will be multiplied by 1.065. In addition, the available trip worth following an ejected rod will be reduced by 100 pcm, which bounds the calculated values.

2.4 Power Operation with Misaligned Rod

Power distributions with control rod misalignment of 30 steps (18 steps misalignment plus 12 steps for ARPI uncertainty) were evaluated. To determine the misalignment cases to be analyzed for this TS change, an evaluation of the rod control system was performed, drawing from the Failure Mode and Effects Analysis. These analyses were performed to evaluate the impact of RCCA misalignment on steady state power distribution. Calculations were performed for both inward and outward misalignments from the demand counter position. Multiple misalignments as well as single misalignments were analyzed. The cases analyzed included BOC, MOC and EOC cases for both core models. A total of over 200 cases were examined for AOs from -15% to +15%.

Comparisons were made between the peaking factors assuming the 18 step misalignment, the 12 step misalignment and the base case (control bank D at rod insertion limit (RIL)). The results indicate that the maximum incremental increase in F_Q and $F\Delta H$ due to an additional misalignment of six steps is 3.6% and 2.4% respectively. Since the technical specification limits on F_Q and $F\Delta H$ for 85% power are 18% and 4.5% greater than those at 100% power, the small changes in F_Q and $F\Delta H$ due to the larger misalignments are adequately accommodated.

2.5 Summary

The proposed TS changes modify TS 3.1.3.1, 4.1.3.1, 3.1.3.2, and 4.1.3.2 and associated bases. The changes replace the rod misalignment value of ±12 steps with ±18 steps if RTP is not above 85%. The bases have been modified to reflect the new allowed rod misalignment.

RCCA misalignments of up to 30 steps (18 steps indicated plus 12 steps ARPI uncertainty) have been evaluated for impact on peaking factors and reactivity worth. The results with respect to reactivity control, RCCA misoperation events and rod ejection events have been shown to be acceptable. For power operation with misalignment of ±18 steps the results of the analysis showed that the incremental increases in the peaking factors were only a small fraction of the increase in the peaking factor limits for power levels less than 85%. Thus, it has been shown that the increase in peaking factors will be accommodated at or below 85% of RTP and the change to the TS to allow misalignment of up to 18 steps is acceptable.

3.0 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.

4.0 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 and changes surveillance requirements. 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 found that the amendment involves no significant hazards consideration. 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.

5.0 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 Contributor: M. Chatterton

Date: December 22, 1997

6.0 REFERENCES

1. E. C. Simpson, Public Service Electric and Gas Company, to NRC, dated August 19, 1997.

2. T. Q. Nguyen, et al., Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Cores, WCAP-11596-P-A, June 1988.

3. Y. S. Liu, et al., ANC: A Westinghouse Advanced Nodal Computer Code, WCAP-10965-P-A, December 1985.