Mr. David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036

September 25 1997

Dear Mr. Lochbaum:

Your letter to me dated September 3, 1997, expressed your concern with the staff's exigent handling of an amendment for the Salem Nuclear Generating Station, Unit 2, which increased the allowable band for control and shutdown rod demanded position versus indicated position.

The licensee requested that the amendment be processed on an exigent basis by letter dated August 19, 1997. Following discussions with the staff, the licensee submitted clarification for the exigent request in its August 20, 1997, follow-up letter.

Your letter cited two telephone discussions held with Mr. Leonard Olshan of my staff on September 2, 1997, regarding your comments on this matter. Your comments, as well as comments from other members of the public, have been addressed in Section 4.0 of the Safety Evaluation (SE) supporting the amendment, which was issued on September 10, 1997. Section 3.0 of the SE provides a statement of the exigent circumstances. We have enclosed a copy of the amendment.

Mr. Olshan and I also spoke with you on September 18, 1997, regarding our decision process related to the exigent handling of this amendment. We explained that the depth of our expedited review for the exigent amendment was the same as it would have been if the amendment had been processed in the normal manner. We also explained why we believed that the circumstances at Salem met the exigent circumstances of 10 CFR 50.91(a)(6) and, as explained in Section 3.0 of the SE, why the startup tests at Salem precluded using the methods used at Sequoyah. We understand that we have addressed your concerns on this matter and we thank you for your comments and interest.

If you have any further questions on this amendment, please contact Mr. Leonard Olshan at (301) 415-1419.

Sincerely,

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

Enclosure: As stated

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

September 25, 1997

Mr. David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036

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The licensee requested that the amendment be processed on an exigent basis by letter dated August 19, 1997. Following discussions with the staff, the licensee submitted clarification for the exigent request in its August 20, 1997, follow-up letter.

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Sincerely,

John F. Stolz

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

Enclosure: As stated

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

September 10, 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. 2 (TAC NO. M99414)

Dear Mr. Eliason:

The Commission has issued the enclosed Amendment No. 183 to Facility Operating License No. DPR-75 for the Salem Nuclear Generating Station, Unit No. 2. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated August 19, 1997, as supplemented by letter dated August 20, 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.

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

Sincerely,

Leonard N. Olshan, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-311

Enclosures: 1. Amendment No. 183 to License No. DPR-75
2. Safety Evaluation

cc w/encls: See next page

Enclosure

Mr. Leon R. Eliason
Public Service Electric & Gas
Company

Salem Nuclear Generating Station,
Units 1 and 2

cc:

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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. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 183
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 August 19, 1997, as supplemented by letter dated August 20, 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-75 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 183, 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 7 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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: September 10, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 183

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 1-13	3/4 1-13
3/4 1-14	3/4 1-14
3/4 1-16	3/4 1-16
3/4 1-16a	3/4 1-16a
B 3/4 1-4	B 3/4 1-4
B 3/4 2-4	B 3/4 2-4

REACTIVITY CONTROL SYSTEMS
3/4.1.3 MOVABLE CONTROL ASSEMBLIES
GROUP HEIGHT

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.

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.

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.

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. 183 TO FACILITY OPERATING LICENSE NO. DPR-75

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

DOCKET NO. 50-311

1.0 INTRODUCTION

By letter dated August 19, 1997, as supplemented by letter dated August 20, 1997, the Public Service Electric & Gas Company (the licensee) submitted a request for changes to the Salem Nuclear Generating Station, Unit No. 2, 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.

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 $F_{\Delta H}$ increase as the power level lowers. The increase in the limit for F_Q and $F_{\Delta H}$ 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 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 6 steps was calculated for both core models at Hot Zero Power (HZP), Hot Full Power (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 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 events 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 effect 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 6 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 100pcm, 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 technical specification 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 axial offsets 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 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 technical specification to allow misalignment of up to 18 steps is acceptable.

3.0 STATEMENT OF EXIGENT CIRCUMSTANCES

In the August 19, 1997 submittal, the licensee requested that the amendment be reviewed on an exigent basis to provide additional operational flexibility, to allow the orderly resumption of startup and preclude unwarranted power transients. As a result of the rod position indication being at minus 13 steps for demanded position for two rods, Salem Unit 2 completed a TS required shutdown on August 19, 1997.

In the August 20, 1997, submittal, the licensee stated that, in early August 1997, the licensee, in conjunction with vendor recommendations and participation, revised the calibration procedures to more closely reflect the original Westinghouse calibration procedures. The rod position indication system was successfully calibrated and Salem Unit 2 went critical on August 17, 1997. On August 18, during performance of reactor physics testing (rod swap), two control rods deviated from their group demand counter by 13 steps, one step over the limit. As a result, Salem Unit 2 entered TS Limiting Condition for Operation 3.3.2.1 and shutdown on August 20, 1997.

Investigation into this apparent misalignment did not indicate any deficiencies with the calibration or circuitry. Therefore, prior to August 19, 1997, the licensee could not have foreseen the need to expedite this change. Salem Unit 2 is expected to restart and a similar problem could arise that would necessitate a shutdown.

Based on the above, the Commission finds that exigent circumstances exist and that the provisions of 10 CFR 50.91(a)(6) apply. The licensee and the Commission must act quickly and time does not permit publication of a Federal Register notice allowing 30 days for prior public comment. Instead, as detailed below, notice was published in local media in the area surrounding the plant. As discussed in Section 4.0, the Commission has determined that the amendment involves no significant hazards considerations. The Commission also finds, pursuant to 10 CFR 50.91(a)(6)(vi), that the licensee did not create the exigency to avoid the normal notice and comment process.

Accordingly, the Commission published a public notice of the proposed amendment, issued a proposed finding of no significant hazards consideration and requested that any comments on the proposed no significant hazards consideration be provided to the staff by the close of business on September 3, 1997, pursuant to 10 CFR 50.91(a)(6). This notice was published in the Wilmington News Journal on August 22, 1997, and will be in Today's Sunbeam on August 24, 1997.

4.0 COMMENTS

During the comment period, the Commission received telephone calls from two individuals and a letter and telephone call from a third individual. The following is a summary of the comments that were received.

One individual asked several questions: (1) How many steps are there from fully out to fully in?; (2) Have any other plants received a similar amendment?; (3) What is the basis for the 12 step difference that is currently allowed in the Technical Specifications; and (4) What is the rush to process the amendment?

The staff provided the individual with the following responses: (1) Full out to full in is 228 steps. (2) Similar amendments have been granted for Turkey Point, Units 3 and 4, and North Anna, Units 1 and 2. (3) The 12 steps is the allowed misalignment at 100% power. At lower power levels, there is more margin available and therefore a larger misalignment is permitted. The amendment allows a misalignment of 18 steps at power levels not greater than 85%. (Additional discussion is provided in Section 2.0) (4) The amendment was processed on an exigent basis to preclude an unnecessary plant shutdown. Section 3.0 provides additional discussion on the need for the exigent amendment.

A second individual commented that the amendment appeared to be as though the NRC was "slackening the rules." The staff responded that the amendment was carefully reviewed by the staff's technical experts and it was found that granting the amendment would not have an adverse impact to the health and safety of the public.

The third individual telephoned and sent a letter with his comments. He concurred with the staff's assessment that the amendment will not adversely affect safety margins at Salem, but disagreed with the need to process the amendment on an exigent basis. On September 3, 1997, the staff spoke to the individual regarding his concerns.

As explained in Section 3.0, the licensee expected that the rod position misalignments would be within the Technical Specification limit after achieving critically since they had been within the limit in the last calibration performed on August 17. Accordingly, the licensee could not have foreseen the need to expedite the amendment prior to August 18, 1997, when it was discovered that two rods were outside the Technical Specification limit.

The third individual also referred to LER 50-327/96011, submitted for Sequoyah Nuclear Plant, Unit 1. This LER describes a situation where the rod position indication system was more than 12 steps different than the demand step counter for two control rods. Sequoyah dealt with the situation by dilution of the reactor coolant system and insertion of the two rods to the point where the non-linear response of the rod position indication system was less pronounced. By doing this, Sequoyah was able to return within the 12 step difference allowed by the Technical Specifications. The individual asked why Salem could not take the same approach and therefore not need the amendment on an exigent basis.

Sequoyah had been at 15% power and toward the end of the fuel cycle, while Salem was conducting low power physics testing at the beginning a new fuel cycle. Low power physics testing involves determining the worth of each rod in which the position of each rod is important. Dilution of the reactor coolant system, as done at Sequoyah, would not be permitted during this phase of the testing at Salem. Therefore, it was not appropriate for Salem to reposition the rods as did Sequoyah, to the point where the non-linearity of the rod position system was less pronounced.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant consideration if operation of the facility in accordance with the proposed 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 following evaluation was provided by the licensee:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change to the rod misalignment criteria of [plus or minus] 18 steps for core powers equal to or below 85% of RATED THERMAL POWER (RTP) does not increase the probability of previously evaluated accidents. Increasing the magnitude of the allowed control rod misalignment is not a contributor to the mechanistic cause of an accident evaluated in any accident analysis. The magnitude of control rod indicated misalignment is a parameter used to establish the initial conditions for accident evaluation.

   The proposed increase in the allowable rod misalignment from the current [plus or minus] 12 steps for reactor powers equal to or less than 85% RTP does not involve a significant increase in the consequence of any previously evaluated accident. Rod misalignment affects power distribution, shutdown margin and the ejected rod accident. An extension of the allowable rod misalignment above and below 85% RTP has been analyzed in Westinghouse WCAP-14672. As provided in WCAP-14672, above 85% the allowable misalignment is governed by the available peaking factor margins as determined by flux maps. PSE&G is simplifying the proposed change by keeping the currently allowed [plus or minus] 12 step misalignment in Technical Specifications 3.1.3.1 and 3.1.3.2.1 for reactor power greater than 85% RTP.

   The PSE&G proposed change is to allow [plus or minus] 18 steps misalignments in Technical Specifications 3.1.3.1 and 3.1.3.2.1 for reactor power less than or equal to 85% RTP. As demonstrated in WCAP-14672, for reactor powers less than 85% RTP, the available peaking factor margin increases faster than any penalty associated with a [plus or minus] 18 step misalignment.

   As described in Section 4.0 of the Westinghouse WCAP, a conservative penalty factor has been applied to the rod insertion allowance (RIA) of the shutdown margin calculation to account for rods misaligned an additional [plus or minus] 6 steps (for a total of [plus or minus] 18 steps). This conservative penalty factor is applied as part of the reload analysis in order to satisfy Technical Specification 3.1.1.1.

   In addition to the normal, or Condition 1, operational transients, the impacts of increased rod misalignment on Condition II, III and IV accident analysis have also been evaluated. The proposed increase in rod misalignment does not have a significant effect on any moderator or Doppler reactivity coefficients or defects, boron worth or reactor kinetics parameters.

To account for the potential increase in ejected rod parameters, conservative penalty factors have been applied to the reload safety evaluation to cover the additional [plus or minus] 6 step misalignment. Margin is available in the reload safety analysis to accommodate this impact.

Therefore, the proposed amendment does not increase the probability or consequences of any accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed change to the rod misalignment criteria of [plus or minus] 18 steps below 85% RTP. The implementation of the proposed rod misalignment criteria will have no adverse effect on the performance of any other safety related system. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

   Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety. The Technical Specifications allowed increase in peaking factors as power is reduced accommodates the peaking factor penalty associated with the additional [plus or minus] 6 step misalignment for core powers equal to or less than 85% RTP. Therefore, there is no change to the peaking factors assumed in the safety analysis. In addition to peaking factors, there is no change in any other current limit input into the safety analysis. As the input, or initial conditions, of the safety analysis have not changed, there is no reduction in the margin to safety.

In addition, the staff concludes, with respect to the second standard, that no physical modifications are being implemented in the facility.

The NRC staff has reviewed the licensee's analysis and based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the Commission finds that the amendment request involves no significant hazards consideration.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendment. By telephone call on August 21, 1997, the State official asked whether power measurement uncertainties had been considered since the amendment only changes the allowable band to ± 18 steps when power level is not greater than 85% rated thermal power. As explained in Section 2.0, there is adequate margin in the analysis at 85% rated thermal power to account for power measurement uncertainties.

7.0 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.

8.0 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: September 10, 1997

8.0 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.

UNION OF
CONCERNED
SCIENTISTS

September 3, 1997

Mr. John F. Stolz
Director, Project Directorate I-2
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: INAPPROPRIATE AMENDMENT REQUEST, SALEM GENERATING STATION UNIT 2

Dear Mr. Stolz:

By letter dated August 21, 1997, Mr. Leonard N. Olshan of your staff notified the licensee for the Salem Generating Station regarding the public notice for a proposed amendment to the facility operating license. As indicated in this letter, the Salem licensee had requested the amendment request on an exigent basis by letter dated August 19, 1997.

UCS reviewed the staff's determination with respect to the issue of no significant hazards consideration. We concur with the staff's assessment that the proposed changes will not adversely affect safety margins at Salem. However, we strongly disagree with the staff's determination that "the licensee has provided adequate justification for the staff to process this amendment in an exigent manner, as provided in 10 CFR 50.91(a)(6)." The facts do not support such a conclusion.

I spoke with Mr. Olshan by telephone on two occasions yesterday regarding this amendment request. Mr. Olshan informed me that the licensee's August 20, 1997, followup letter was in response to the staff's reluctance to consider this amendment request as qualifying for exigent handling. Mr. Olshan indicated that several other licensees had sought, and had been granted, comparable Technical Specification changes. Mr. Olshan had no idea if these other licensees had obtained the changes on an exigent basis. Mr. Olshan informed me that Salem Unit 2 restarted and did not re-experience the problems, which prompted the expedited amendment request.

I asked Mr. Olshan about a recent similar event at Sequoyah Unit 1 (see LER 50-327/96011 submitted by letter dated December 18, 1996) in which TVA encountered the same problem as experienced at Salem, but handled it via dilution and control rod insertion. TVA indicated that they would submit a license amendment request, presumably under the normal process, to remedy the very problem that this Salem license amendment request addresses.

It is troubling that the NRC staff would move so swiftly down the exigent pathway simply because this licensee requested it. From my discussions with Mr. Olshan, it is not apparent that the NRC reviewed prior events such as the one at Sequoyah to determine whether the Salem licensee truly had no option other than this "emergency" Technical Specification relief. Salem Unit 2 received its full power operating license in May 1981. If memory serves me correctly, the plant has started up more than once since 1981 giving this licensee amply opportunity to identify overly restrictive Technical Specifications. The fact that this

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"emergency" request was supported by a previously prepared and issued Westinghouse safety analysis (WCAP-14672) suggests that this problem has been around for some time and did not develop in the past few days. Clearly, this licensee had sufficient opportunity to be cognizant of this potential problem and its ready fix. That this licensee neglected the opportunity and was "surprised" by the problem during a startup is simply not proper grounds for the NRC to consider this "emergency" license amendment under exigent processing.

Despite our considered opinion that the NRC staff is being inappropriately accommodating to the Salem licensee with respect to this license amendment, UCS will not intervene in this matter. The technical justification for the proposed changes is solid and safety margins at the facility will not be compromised if these changes are approved. Therefore, we see no reason to oppose this amendment even though it is being pursued through improper administrative methods.

We respectfully ask that the NRC staff seriously review the process under which license amendment requests are processed under exigent conditions. The process appears fundamentally flawed in that it does not, or at least it did not in this case, determine whether the licensee has options other than the requested Technical Specification relief. The availability of other options could eliminate the need for exigent handling. In addition, the process does not, or at least it did not in this case, review previous staff actions on comparable Technical Specification changes. The routine handling of similar amendment requests could eliminate the justification for exigent treatment. Basically, the process appears to rely almost exclusively on information provided by the licensee in support of the amendment request. One of the primary lessons learned from the Maine Yankee RELAP5 code issue was that the NRC staff should not rely solely on licensees but should conduct independent verifications. Proper independent verification demands more than merely reading the material submitted by the licensee and reformatting it in a no significant hazards consideration statement.

If there are any questions or comments, please do not hesitate to contact me.

Sincerely,

David A. Lochbaum

David A. Lochbaum
Nuclear Safety Engineer

cc: Chairman Shirley A. Jackson
Mr. Samuel J. Collins
Mr. Hubert Bell
Senator Joseph R. Biden