May 25, 2001

Mr. Harold W. Keiser
Chief Nuclear Officer & President
PSEG Nuclear LLC - X04
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 - EXEMPTION FROM THE REQUIREMENTS OF 10 CFR 50.60, AND 10 CFR PART 50, APPENDIX G (TAC NOS. MB0606 AND MB0607)

Dear Mr. Keiser:

The Commission has granted the enclosed exemption from specific requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.60(a) and 10 CFR Part 50, Appendix G, for the Salem Nuclear Generating Station, Unit Nos. 1 and 2. This action is in response to your letter dated November 10, 2000, as supplemented by letters dated March 28 and April 2, 2001, that requested an exemption from U.S. Nuclear Regulatory Commission regulations in order to revise the methodology used to determine the reactor pressure vessel pressure-temperature (P/T) limit curves. Specifically, the exemption allows the use of American Society of Mechanical Engineers (ASME) Code Case N-640, "Alternative Reference Fracture Toughness for Development of P/T Limit Curves for ASME Section XI, Division 1," in lieu of certain specific requirements of 10 CFR 50.60 and Appendix G.

A copy of the exemption is enclosed. The exemption has been forwarded to the Office of the Federal Register for publication

Sincerely,

/RA/

Robert J. Fretz, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-272 and 50-311

Enclosure: Exemption

cc w/encl: See next page

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ACCESSION NO. ML011100257 TEMPLATE = NRR-048 * See SE dated 04/19/01. No major changes made

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

PSEG NUCLEAR LLC

SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-272 and 50-311

EXEMPTION

1.0 BACKGROUND

PSEG Nuclear LLC (PSEG or the licensee) is the holder of Facility Operating License Nos. DPR-70 and DPR-75 that authorize operation of the Salem Nuclear Generating Station, Unit Nos. 1 and 2. The licenses provide, among other things, that the facility is subject to all rules, regulations, and orders of the U.S. Nuclear Regulatory Commission (NRC, the Commission) now or hereafter in effect.

The facility consists of two pressurized water reactors located at the licensee's site on the southern end of Artificial Island in Lower Alloways Creek Township, Salem County, New Jersey. Salem, New Jersey, is located approximately 7.5 miles northeast of the site.

2.0 PURPOSE

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G requires that pressure-temperature (P-T) limits be established for reactor pressure vessels (RPVs) during normal operating and hydrostatic or leak rate testing conditions. Specifically, 10 CFR Part 50, Appendix G states that "[t]he appropriate requirements on . . . the pressure-temperature limits and minimum permissible temperature must be met for all conditions." Appendix G to 10 CFR Part 50 also specifies that the requirements for these limits are the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G Limits. In Generic Letter 88-11, the NRC staff advised licensees that the staff would use Regulatory Guide (RG) 1.99, Revision 2, to review P-T limit curves. RG 1.99, Revision 2, provides guidance for implementing 10 CFR Part 50, Appendix G, and contains conservative methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation.

In order to address provisions of amendments to the Technical Specifications' (TS) P-T limit curves, the licensee requested in its application dated November 10, 2000, that the staff exempt, as permitted by 10 CFR 50.60(b), Salem, Unit Nos. 1 and 2, from application of specific requirements of 10 CFR 50.60(a) and 10 CFR Part 50, Appendix G, and substitute use of ASME Code Case N-640. Code Case N-640 provides an alternate reference fracture toughness methodology for reactor vessel materials for use in determining the P-T limits. The proposed action is in accordance with PSEG's application for exemption contained in its November 10, 2000, letter, as supplemented by letters dated March 28 and April 2, 2001. The proposed action is needed to support PSEG's license amendment request to increase thermal power levels by 1.4% submitted under the same application (the final revision of the proposed P-T limit curves was submitted by the licensee by letter dated March 28, 2001). The proposed license amendment will, in part, revise the P-T limits for heatup, cooldown, core criticality, and hydrostatic/leak test limitations for the reactor coolant system (RCS) to 32 effective full power years (EFPYs).

<u>Code Case N-640</u>

The licensee has proposed an exemption to allow the use of Code Case N-640, in conjunction with ASME Section XI, Appendix G, 10 CFR 50.60(a), and 10 CFR Part 50, Appendix G, to determine the P-T limits, and stated that this proposed alternative meets the underlying intent of the NRC's regulations.

Standard Review Plan (NUREG-0800) Section 5.3.2 provides an acceptable method for determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 on the same stresses for hydrostatic testing curves. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

The methodology provided in Appendix G to Section XI of the ASME Code requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term by application of RG 1.99, Revision 2. The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Revision 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor (CF) was determined using the tables in RG 1.99, Revision 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence and the calculational procedures. RG 1.99, Revision 2, describes the methodology to be used in calculating the margin term.

The Pressurized Thermal Shock (PTS) rule, 10 CFR 50.61, requires that licensees demonstrate that facility RPV materials will continue to possess an adequate level of fracture resistance to protect the RPV from potential failure as a result of PTS events. Each material's PTS reference temperature, RT_{PTS}, is determined in a manner like that used to determine ART, except that the neutron fluence at the clad-to-base metal interface at end of license (EOL) conditions is used in lieu of either the 1/4T or 3/4T fluence. Each material's RT_{PTS} value is then compared to the screening limits given in 10 CFR 50.61, 270 °F for plates, forging, and axial welds, and 300 °F for circumferential welds. Provided that all RPV materials' RT_{PTS} values remain below these screening limits, the fracture resistance of the RPV is demonstrated to be adequate to meet the requirements of 10 CFR 50.61 through end of life.

The proposed license amendments to revise the P-T limits for Salem, Unit Nos. 1 and 2, rely in part on the requested exemption. These revised P-T limits have been developed using the K_{Ic} fracture toughness curve shown in ASME Section XI, Appendix A, Figure A-2200-1, in lieu of the K_{Ia} fracture toughness curve of ASME Section XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. The other margins involved with the ASME Section XI, Appendix G process for establishing P-T limit curves remain unchanged.

Use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of P-T operating limit curves is more technically correct than the K_{Ia} curve. The K_{Ic} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel. The licensee stated that the use of the K_{Ia} curve, with its initial conservatism, was justified when the curve was codified in 1974. This initial conservatism was necessary due to the limited knowledge of RPV materials. Since 1974, additional knowledge has been gained about RPV materials, that demonstrates that the lower bound on fracture toughness provided by the K_{Ia} curve is well beyond the margin of safety required to protect the public health and safety from potential RPV failure. In addition, P-T curves based on the K_{Ic} curve will enhance overall plant safety by opening the P-T operating window with the greatest safety benefit in the region of low temperature operations. The operating window through which the operator heats up and cools down the RCS is determined by the difference between the maximum allowable pressure determined by Appendix G of ASME Section XI, and the minimum required pressure for the reactor coolant pump (RCP) seals adjusted for instrument uncertainties.

Since the RCS P-T operating window is defined by the P-T operating and test limit curves developed in accordance with the ASME Section XI, Appendix G procedure, continued operation of Salem, Unit Nos. 1 and 2, with these P-T curves without the relief provided by ASME Code Case N-640 may unnecessarily restrict the P-T operating window, especially at low temperature conditions. The operating window becomes more restrictive with continued reactor vessel service. Implementation of the proposed P-T curves, as allowed by ASME Code Case N-640, does not significantly reduce the margin of safety. Thus, pursuant to 10 CFR 50.12(a)(2)(ii), the underlying purpose of the regulation will continue to be served.

In summary, the ASME Section XI, Appendix G procedure was conservatively developed based on the level of knowledge existing in 1974 concerning RPV materials and the estimated effects of operation. Since 1974, the level of knowledge about these topics has been greatly expanded. The NRC staff concurs that this increased knowledge permits relaxation of the ASME Section XI, Appendix G requirements by application of ASME Code Case N-640, while maintaining, pursuant to 10 CFR 50.12(a)(2)(ii), the underlying purpose of the ASME Code and NRC regulations to ensure an acceptable margin of safety.

3.0 DISCUSSION

Pursuant to 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 50, when (1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security; and (2) when special circumstances are present. Special circumstances are present whenever, according to 10 CFR 50.12 (a)(2)(ii), "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule." The staff accepts the licensee's determination that an exemption would be required to approve the use of Code Case N-640. The staff examined the licensee's rationale to support the exemption request and concurred that the use of the code case would also meet the underlying intent of these regulations. Based upon a consideration of the conservatism that is explicitly incorporated into the methodologies of 10 CFR Part 50, Appendix G; Appendix G of the ASME Code; and RG 1.99, Revision 2, the staff concluded that application of the code case as described would provide an acceptable margin of safety against brittle failure of the RPV.

Therefore, since strict compliance with the requirements of 10 CFR Part 50, Appendix G, is not necessary to serve the underlying purpose of the regulation, the staff concludes that application of Code Case N-640 to the P-T limit calculations meets the special circumstance provisions stated in 10 CFR 50.12(a)(2)(ii), for granting this exemption to the regulation, and that the methodology of Code Case N-640 may be used to revise the P-T limits for Salem, Unit Nos. 1 and 2.

4.0 CONCLUSION

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12(a), the exemption is authorized by law, will not endanger life or property or common defense and security, and is, otherwise, in the public interest. Therefore, the Commission hereby grants

PSEG Nuclear LLC an exemption from the requirements of 10 CFR Part 50, Section 50.60(a) and 10 CFR Part 50, Appendix G, for Salem, Unit Nos. 1 and 2.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will not have a significant effect on the quality of the human environment (66 FR 24410).

This exemption is effective upon issuance.

Dated at Rockville, Maryland, this 25th day of May 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

John A. Zwolinski, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation