March 26, 2001

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

SUBJECT: HOPE CREEK GENERATING STATION AND SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 - EVALUATION OF RELIEF REQUEST RR-B6-1 (TAC NOS. MB1401, MB1399, AND MB1400)

Dear Mr. Keiser:

By letter dated March 7, 2001, PSEG Nuclear LLC (PSEG) submitted relief request RR-B6-1 seeking relief from certain requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (the Code), for inservice inspection (ISI) of reactor pressure vessel welds at the Hope Creek Generating Station (Hope Creek) and Salem Nuclear Generating Station, Unit Nos. 1 and 2 (Salem). Specifically, PSEG proposed to use the depth sizing requirement contained in 10 CFR 50.55a(b)(2)(xv)(C)(1), in lieu of the requirements contained in Section XI, 1995 Edition, 1996 Addenda, Appendix VIII, Supplement 4, Subparagraph 3.2(c) of the Code.

The U.S. Nuclear Regulatory Commission (NRC) staff has completed the review of the subject relief request. The NRC staff's Safety Evaluation (SE) is enclosed. Our SE concludes that the proposed alternative in relief request RR-B6-1 will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the staff authorizes the proposed alternative for the second 10-year inservice inspection intervals at Hope Creek and Salem.

The staff considers your request to be a non-timely submittal, given both the short time period in which you requested a response, and in that there was no prior notification nor expedited transmission of your request to the staff. This is especially significant since you did not provide a discussion in your submittal of either the safety or operational implications of not receiving prompt staff review of your request. This has resulted in a significant reduction in the efficient use of staff resources since we had to stop ongoing reviews to complete action on your request.

Sincerely,

/RA/

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-354, 50-272, and 50-311

Enclosure: Safety Evaluation

cc w/encl: See next page

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Enclosure: Safety Evaluation

cc w/encl: See next page

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ACCESSION NUMBER: ML010850168 *See previous concurrence

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Salem Nuclear Generating Station, Unit Nos. 1 and 2, and Hope Creek Generating Station

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO RELIEF REQUEST RR-B6-1

FOR SECOND 10-YEAR INSERVICE INSPECTION INTERVAL AT

HOPE CREEK GENERATING STATION AND

SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

PSEG NUCLEAR LLC

DOCKET NOS. 50-354, 50-272, AND 50-311

1.0 INTRODUCTION

The inservice inspection of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME Code and applicable edition and addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). Pursuant to 10 CFR 50.55a(a)(3), alternatives to the requirements of paragraph (g) may be used, when authorized by the U.S. Nuclear Regulatory Commission (NRC), if the licensee demonstrates that: (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to Commission approval.

ENCLOSURE

By letter dated March 7, 2001, PSEG Nuclear LLC (PSEG or the licensee) submitted relief request RR-B6-1 seeking relief from certain requirements of the ASME Code, for ISI of reactor pressure vessel (RPV) welds at the Hope Creek Generating Station (Hope Creek) and Salem Nuclear Generating Station, Unit Nos. 1 and 2 (Salem). Specifically, PSEG proposed to use the 0.15-inch root mean square (RMS) depth sizing requirement contained in 10 CFR 50.55a(b)(2)(xv)(C)(1), in lieu of the requirements contained in Section XI, 1995 Edition, 1996 Addenda, Appendix VIII, Supplement 4, Subparagraph 3.2(c) of the Code.

In a related submittal dated July 28, 2000, the licensee submitted relief request RR-B6, "Alternative to Length Sizing Criterion for ASME XI Code Inspections." The NRC Safety Evaluation dated September 14, 2000, authorized the use of a length sizing tolerance of 0.75-inch RMS error in lieu of the requirements in Appendix VIII, Supplement 4, Subparagraph 3.2(b) of the Code.

2.0 BACKGROUND

2.1 Component Description

RPV longitudinal and circumferential shell welds and RPV head welds in Examination Category B-A, Item nos. B1.10 and B1.20 of ASME Code, Section XI.

2.2 Examination

Ultrasonic Testing (UT) in accordance with Appendix VIII, Supplement 4, ASME Code, Section XI.

2.3 Code Requirement for which Relief is Requested

Section 50.55a(g)(6)(ii)(C) of Title 10 of the *Code of Federal Regulations* imposes implementation of Appendix VIII to the 1995 Edition with the 1996 Addenda of Section XI of the ASME Code. The imposed implementation schedule for Supplement 4 to Appendix VIII is November 22, 2000. ASME Section XI, 1995 Edition, 1996 Addenda, Appendix VIII, Supplement 4, Subparagraph 3.2(c), requires that the UT performance demonstration results be plotted on a two-dimensional plot with the measured depth plotted along the ordinate axis and the true depth plotted along the abscissa axis. For qualification, the plot must satisfy the following statistical parameters: (1) the slope of the linear regression line is not less than 0.7; (2) the mean deviation of flaw depth is less than 0.25 inches; and (3) the correlation coefficient is not less than 0.70.

2.4 Licensee's Proposed Alternative to ASME Code

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee proposed to use the RMS values of 10 CFR 50.55a(b)(2)(xv)(C)(1) which modifies the depth sizing criterion of Appendix VIII, Supplement 4, Subparagraph 3.2(a), in lieu of Subparagraph 3.2(c). These examinations will be performed during the second inspection intervals at Hope Creek and Salem.

2.5 Licensee's Basis for Relief

The licensee's submittal provided the following as the basis for relief:

> The NRC granted relief to PSEG Nuclear's request RR-B6 to use a length sizing tolerance of 0.75-inch root mean square error (RMSE) in lieu of the requirements of Subparagraph 3.2(b) to Supplement 4 to Appendix VIII of the Code (ref. 2). Relief request RR-B6, however, failed to request eliminating the use of Supplement 4, Subparagraph 3.2(c) for depth sizing.
>
> In a public meeting on October 11, 2000, the Performance Demonstration Initiative (PDI) identified the discrepancy between Subparagraph 3.2(c) and the PDI program. After review, the NRC has agreed that 10 CFR 50.55a(b)(2)(xv)(C)(1) should have excluded Subparagraph 3.2(c) as a requirement.
>
> Subsequently, the NRC granted similar relief for Millstone Nuclear Power Station Nos. 2 and 3 (TAC Nos. MA9857 and MA9858) in a Safety Evaluation dated January 26, 2001 (ref. 3).

3.0 EVALUATION

The U.S. nuclear utilities created the PDI to implement performance demonstration requirements contained in Appendix VIII of Section XI of the ASME Code. To this end, PDI has developed a performance demonstration program for qualifying UT equipment, procedures, and personnel. During the development of the performance demonstration for Supplement 4, PDI determined that the Code criteria for flaw sizing was unworkable.

In relief request RR-B6-1, the licensee proposed to eliminate the use of the requirement in Supplement 4, Subparagraph 3.2(c) which imposes three statistical parameters for depth sizing. The first parameter, 3.2(c)(1), pertains to the slope of a linear regression line. The linear regression line is the difference between actual versus true value plotted along a through-wall thickness. For Supplement 4 performance demonstrations, a linear regression line of the data is not applicable because the performance demonstrations are performed on test specimens with flaws located in the inner 15% through-wall. The differences between actual versus true value produce a tight grouping of results which resemble a shotgun pattern. The slope of a regression line from such data is extremely sensitive to small variations, thus making the parameter of Subparagraph 3.2(c)(1) a poor and inappropriate acceptance criterion. The second parameter, 3.2(c)(2), pertains to the mean deviation of flaw depth. The value used in the code is too lax with respect to evaluating flaw depths within the inner 15% of wall thickness. Therefore, the licensee proposed to use the more appropriate criterion of 0.15-inch RMS of 10 CFR 50.55a(b)(2)(xv)(C)(1), which modifies Subparagraph 3.2(a), as the acceptance criterion. The third parameter, 3.2(c)(3), pertains to a correlation coefficient. The value of the correlation coefficient in Subparagraph 3.2(c)(3) is inappropriate for this application since it is based on the linear regression from Subparagraph 3.2(c)(1).

PDI was aware of the inappropriateness of Subparagraph 3.2(c) early in the development of their program. They brought the issue before the appropriate ASME committee which formalized eliminating the use of Supplement 4, Subparagraph 3.2(c) in Code Case N-622. The NRC staff representatives participated in the discussions and consensus process of the code case. Based on the above, the NRC staff finds that the use of Subparagraph 3.2(c) requirements in this context is inappropriate and that the proposed alternative to use the RMS value of 10 CFR 50.55a(b)(2)(xv)(C)(1), namely 0.15-inch RMS, which modifies the criterion of Appendix VIII, Supplement 4, Subparagraph 3.2(a), in lieu of Subparagraph 3.2(c), will provide an acceptable level of quality and safety.

4.0 CONCLUSION

Based on the discussion above, the staff has concluded that the proposed alternative in relief request RR-B6-1 will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the staff authorizes the proposed alternative for the second 10-year inservice inspection intervals at Hope Creek and Salem.

Principal Contributor: P. Patnaik

Date: March 26, 2001