September 14, 2000

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, UNITS 1 AND 2 - EVALUATION OF RELIEF REQUEST; ALTERNATIVE TO LENGTH SIZING CRITERION FOR ASME SECTION XI CODE INSPECTIONS (TAC NOS. MA9606, MA9607, AND MA9718)

Dear Mr. Keiser:

By letter dated July 28, 2000, Public Service Electric and Gas Company (PSE&G) submitted Relief Request No. RR-B6 seeking relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (the Code), Section XI, 1995 Edition, 1996 Addenda, Appendix VIII, Supplement 4, Subparagraph 3.2(b), for inspection of the Hope Creek Generating Station (Hope Creek) and Salem Nuclear Generating Station, Unit Nos. 1 and 2, (Salem) reactor pressure vessel welds. On August 21, 2000, the licenses for Salem and Hope Creek, to the extent held by PSE&G, were transferred to PSEG Nuclear Limited Liability Company (PSEG Nuclear). By letter dated September 6, 2000, PSEG Nuclear stated that it has assumed responsibility, as of the date of the transfer, for the active items on the Salem and Hope Creek dockets previously submitted by PSE&G, including the subject relief request. The July 28, 2000, relief request proposed to use a length sizing qualification criteria of 0.75 inch root mean square error. The proposed alternative is to be incorporated into the Salem and Hope Creek second interval inservice inspection programs.

The 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 will provide an acceptable level of quality and safety for ensuring the pressure boundary integrity of the Hope Creek and Salem reactor pressure vessels. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i).

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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<u>DISTRIBUTION</u>

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PDI-2 Reading	EAdensam	RFretz	ESullivan	GHill
OGC	JClifford	TClark	DNaujock	GMeyer, RGN-I

ACCESSION NUMBER: ML003737678 TEMPLATE = NRR-028 * See previous concurrence

OFFICE	PDI-2/PM	PDI-2/PM	PDI-2/LA	EMCB/SC*	OGC*	PDI-2/SC
NAME	RFretz	JHarrison	TClark	ESullivan	SHom	JClifford
DATE	09/07/00	9/7/00	9/7/00	08/20/00	08/22/00	9/12/00

OFFICIAL RECORD COPY

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO RELIEF REQUESTS FOR REACTOR VESSEL INSPECTIONS

PSEG NUCLEAR LLC

HOPE CREEK GENERATING STATION

SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2

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

1.0 INTRODUCTION

By letter dated July 28, 2000, Public Service Electric and Gas Company (PSE&G) submitted Relief Request No. RR-B6 seeking relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (the Code), Section XI, 1995 Edition, 1996 Addenda, Appendix VIII, Supplement 4, Subparagraph 3.2(b), for inspection of the Hope Creek Generating Station (Hope Creek) and Salem Nuclear Generating Station, Unit Nos. 1 and 2, (Salem) reactor pressure vessel (RPV) welds. On August 21, 2000, the licenses for Salem and Hope Creek, to the extent held by PSE&G, were transferred to PSEG Nuclear Limited Liability Company (PSEG Nuclear). By letter dated September 6, 2000, PSEG Nuclear stated that it has assumed responsibility, as of the date of the transfer, for the active items on the Salem and Hope Creek dockets previously submitted by PSE&G, including the subject relief request. The relief request proposed to use a length sizing qualification criteria of 0.75 inch root mean square (RMS) error. The proposed alternative is to be incorporated into the Salem and Hope Creek second interval inservice inspection programs. The licensee proposed using the qualification tolerance for length sizing from Appendix IV, “Qualification Requirements for the Clad-to-Base Metal Interface of Reactor Vessel,” to Code Case (CC) N-622, “Ultrasonic Examination of RPV and Piping and Bolts and Studs, Section XI, Division 1,” as an alternative to the Code. Code Case N-622 provides criteria for ultrasonic testing (UT) performance-based qualifications of procedures, equipment, and personnel.

2.0 BACKGROUND

The inservice inspection of the ASME Code Class 1, Class 2, and Class 3 components shall be performed in accordance with Section XI of the ASME Code and applicable addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(g), except where alternatives have been authorized by the Commission pursuant to 10 CFR 50.55a(a)(3). Section 50.55a(a)(3) states in part that alternatives to the requirements may be used providing 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.

ENCLOSURE

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 preservice examination requirements, set forth in the ASME Code, Section XI, “Rules for Inservice Inspection 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 third ten-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) on the date twelve 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.

2.1 Component Description

ASME Section XI, Class 1, Examination category B-A, Item no. B1.10 reactor pressure vessel (RPV) longitudinal and circumferential shell welds, and B1.20 RPV head welds subject to Appendix VIII, Supplement 4 examination.

2.2 Code Requirements

Title 10 of the *Code of Federal Regulations*, Section 50.55a(b)(2) was amended to reference Section XI of the Code through the 1995 Edition with the 1996 Addenda (64 FR 51370).

As amended, 10 CFR 50.55a(b)(2)(xv)(C)(1) requires a depth sizing acceptance criterion of 0.15 inch RMS be used in lieu of the requirements of Subparagraph 3.2(b) to Supplement 4 to Appendix VIII of Section XI of the 1995 Edition with 1996 Addenda of the Code.

2.3 Basis for Alternative

On January 12, 2000, NRC staff held discussions with representatives from the Electric Power Research Institute (EPRI) Nondestructive Examination Center, and representatives from the Performance Demonstration Initiative (PDI). The discussions included the differences between Supplement 4, “Qualification Requirements for the Clad/Base Metal Interface of Reactor Vessel,” to Appendix VIII, “Performance Demonstration for Ultrasonic Examination Systems,” Paragraph 10 CFR 50.55a(b)(2)(xv)(C)(1) in the rule (*Federal Register*, 64 FR 51370), and the implementation of Supplement 4 by the PDI program. Supplement 4, Subparagraph 3.2(b) imposed a flaw sizing tolerance of -¼ inch, +1.0 inch of true length to the performance demonstration qualification criteria. The rule changed Subparagraph 3.2(b) to a depth sizing requirement of 0.15 inch RMS, and the PDI program uses a length sizing tolerance of 0.75 inch RMS for Subparagraph 3.2(b). The NRC staff recognized that 10 CFR 50.55a(b)(2)(xv)(C)(1) in the rule was in error and should actually be a length sizing tolerance of 0.75 inch RMS, the same tolerance that was being implemented by the PDI program.

3.0 EVALUATION

3.1 Proposed Alternative Examination

PSE&G proposed that the staff accept implementation of a change in Subparagraph 3.2(b) to Supplement 4 of Appendix VIII to a flaw length sizing tolerance of 0.75 inch RMS as estimated by UT.

3.2 Discussion

U.S. nuclear utilities created the PDI to implement demonstration requirements contained in Appendix VIII. PDI developed a performance demonstration program for qualifying UT techniques. In 1995, the NRC staff performed an assessment of the PDI program and reported that PDI was using a length sizing tolerance of 0.75 inch RMS for RPV performance demonstrations. This criterion was introduced to reduce testmanship (passing the test based on manipulation of results rather than skill). The staff noted in the assessment report dated March 6, 1996, that the length sizing tolerance was not according to Appendix VIII but did not take exception to PDI's implementation of the 0.75 inch RMS length sizing tolerance. The staff requested that the length sizing difference between PDI and the Code be resolved.

The solution for resolving the differences between the PDI program and the Code was for PDI to participate in the development of a code case. The code case was presented to ASME for discussion and consensus building. NRC representatives participated in this process. ASME approved the code case and published it as Code Case N-622, "Ultrasonic Examination of RPV and Piping and Bolts and Stubs, Section XI, Division 1."

Operating in parallel with these actions, the staff incorporated most of Code Case N-622 criteria in 10 CFR 50.55a(b)(2)(xv). On January 12, 2000, PDI identified the omission of the length sizing tolerance in 10 CFR 50.55a(b)(2)(xv)(C). The staff agreed that the omission of the length sizing tolerance of 0.75 inch RMS in the rule, similar to 10 CFR 50.55a(b)(2)(xv)(E)(3), was an oversight, and that the inclusion of Subparagraph 3.2(b), Supplement 4 to Appendix VIII of Section XI of the Code in the depth sizing tolerance provided in 10 CFR 50.55a(b)(2)(xv)(C)(1) was an error. The staff considers that the proposed alternative to use a length sizing tolerance of 0.75 inch RMS in lieu of the requirements in Subparagraph 3.2(b) will provide an acceptable level of quality and safety, since it intended to use this value when it incorporated most of the Code Case N-622 criteria in 10 CFR 50.55a(b)(2)(xv); is consistent with the length sizing tolerance of 0.75 inch prescribed in 10 CFR 50.55a(b)(2)(xv)(E)(3); is of similar magnitude as the current flaw sizing tolerance of -¼ inch, +1.0 inch in the Code; and will also minimize the potential for testmanship.

4.0 CONCLUSION

Based on its review, the staff finds that the proposed alternative to use a length sizing tolerance of 0.75 inch RMS in lieu of the requirements in Subparagraph 3.2(b) to Supplement 4 to Appendix VIII of the Code will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative is authorized.

Principal Contributor: R. Fretz

Date: September 14, 2000