March 22, 2004

Mr. Roy A. Anderson
President & Chief Nuclear Officer
PSEG Nuclear LLC - X04
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK NUCLEAR GENERATING STATION - EVALUATION OF RELIEF REQUEST HC-RR-B12 (TAC NO. MB8407)

Dear Mr. Anderson:

By letter dated April 14, 2003, as supplemented by letters dated September 3, 2003, and January 30, 2004, PSEG Nuclear LLC (PSEG) submitted a request for relief from Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) under the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(a)(3)(ii) for the Hope Creek Nuclear Generating Station (Hope Creek). Specifically, Relief Request HC-RR-B12 proposed an alternative visual examination for selected nozzle inner radii in lieu of the ultrasonic testing examination requirements of the ASME Code. The request for relief is for the second 10-year inservice inspection interval, which started on December 13, 1997, and ends December 12, 2007.

Based on the information provided, the U.S. Nuclear Regulatory Commission (NRC) staff concludes that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, as described in Relief Request HC-RR-B12. The NRC staff also concludes that the proposed visual examination will provide reasonable assurance of structural integrity. Therefore, the NRC staff authorizes PSEG to use the proposed visual examination, pursuant to 10 CFR 50.55a(a)(3)(ii), for the second 10-year interval at Hope Creek.

The NRC staff's Safety Evaluation is enclosed. If you have any questions, please contact your Project Manager, John Boska, at 301-415-2901.

Sincerely,

/RA/

Darrell J. Roberts, Acting Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosure: As stated

cc w/encl: See next page

March 22, 2004

Mr. Roy A. Anderson
President & Chief Nuclear Officer
PSEG Nuclear LLC - X04
Post Office Box 236
Hancocks Bridge, NJ 08036

SUBJECT: HOPE CREEK NUCLEAR GENERATING STATION - EVALUATION OF RELIEF REQUEST HC-RR-B11 (TAC NO. MB8408)

Dear Mr. Anderson:

By letter dated April 14, 2003, as supplemented by letters dated September 3, 2003, and January 30, 2004, PSEG Nuclear LLC (PSEG) submitted a request for relief from Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) under the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(a)(3)(ii) for the Hope Creek Nuclear Generating Station (Hope Creek). Specifically, Relief Request HC-RR-B12 proposed an alternative visual examination for selected nozzle inner radii in lieu of the ultrasonic testing examination requirements of the ASME Code. The request for relief is for the second 10-year inservice inspection interval, which started on December 13, 1997, and ends December 12, 2007.

Based on the information provided, the U.S. Nuclear Regulatory Commission (NRC) staff concludes that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, as described in Relief Request HC-RR-B12. The NRC staff also concludes that the proposed visual examination will provide reasonable assurance of structural integrity. Therefore, the NRC staff authorizes PSEG to use the proposed visual examination, pursuant to 10 CFR 50.55a(a)(3)(ii), for the second 10-year interval at Hope Creek.

The NRC staff's Safety Evaluation is enclosed. If you have any questions, please contact your Project Manager, John Boska, at 301-415-2901.

Sincerely,

/RA/
Darrell J. Roberts, Acting Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-354
Enclosure: As stated
cc w/encl: See next page
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ACRS	DRoberts	DNaujock	GHill (2)	JBoska	PDI-2 R/F

ADAMS ACCESSION NUMBER: ML040570222 * See SE input dated February 13, 2004

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DATE	03/10/04	03/15/04	2/13/04	02/17/04	03/21/04

OFFICIAL RECORD COPY

Hope Creek Generating Station

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR RELIEF HC-RR-B12

SECOND 10-YEAR INSERVICE INSPECTION INTERVAL

HOPE CREEK NUCLEAR GENERATING STATION

PSEG NUCLEAR LLC

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated April 14, 2003, as supplemented by letters dated September 3, 2003, and January 30, 2004, PSEG Nuclear LLC (PSEG or the licensee) submitted a request for relief from Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) under the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(a)(3)(ii) for the Hope Creek Nuclear Generating Station (Hope Creek). Specifically, Relief Request HC-RR-B12 proposed an alternative visual examination for selected nozzle inner radii in lieu of the ultrasonic testing (UT) examination requirements of the ASME Code. The request for relief is for the second 10-year inservice inspection (ISI) interval, which started on December 13, 1997, and ends December 12, 2007.

2.0 BACKGROUND

The ISI of ASME Code Class 1, 2, and 3, components is to be performed in accordance with Section XI of the ASME Code and applicable addenda as required by 10 CFR 50.55a(g), except where specific relief has been granted by the U.S. Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 50.55a(g)(6)(i). As stated in 10 CFR 50.55a(a)(3), alternatives to the requirements of paragraph (g) may be used, when authorized by the 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 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 the ISI of components and system pressure tests conducted during the first 10-year interval, and subsequent intervals, complies with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in

Enclosure

10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. For Hope Creek, the applicable edition of Section XI of the ASME Code for the second 10-year ISI interval is the 1989 Edition, without Addenda.

3.0 EVALUATION

3.1 ASME Code Components Affected

The specific inner nozzle radii are identified in Table 1 of the licensee’s letter dated September 3, 2003. Reactor pressure vessel (RPV) inner nozzle radii subject to this request for relief are:

(2) jet pump instrumentation nozzles (RPV1-N8A, RPV1-N8B),
(10) jet pump riser inlet nozzles (RPV1-N2A, RPV1-N2B, RPV1-N2C, RPV1-N2D, RPV1-N2E, RPV1-N2F, RPV1-N2G, RPV1-N2H, RPV1-N2J, RPV1-N2K),
(2) core spray inlet nozzles (RPV1-N5A, RPV1-N5B), and
(4) low pressure coolant injection (LPCI) inlet nozzles (RPV1-N17A, RPV1-N17B, RPV1-N17C, RPV1-N17D).

3.2 Code Requirements for Which Relief Is Requested

1989 Edition of ASME Code, Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.100, “Nozzle Inside Radius Section,” and Figures IWB-2500-7(a) through (d). Relief is requested from the requirements to perform volumetric examinations for the second 10-year ISI interval at Hope Creek.

3.3 Licensee's Proposed Alternative

The licensee proposes to perform an enhanced VT-1 visual examination technique of the surface M-N shown in ASME Code, Section XI, Figures IWB-2500-7(a) through (d) as an alternative to ASME Code, Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.100 volumetric examination requirement. The enhanced remote visual examination will achieve essentially 40 to 60% coverage using 8x magnification video equipment to examine the inner radii. The resolution sensitivity for this remote examination will be established using a 1-mil diameter wire standard similar to that used for other RPV internal examinations intended to detect cracking.

The qualification process used to demonstrate resolution sensitivity will represent the surface texture (reflectivity, color, and finish) of the item to be examined. Targets of sufficient length will be used to demonstrate the required resolution across the entire field of view of the camera system. At least one such target will be oriented in the horizontal direction and another target oriented in the vertical direction. The target is a wire of less than, or equal to, 1-mil width.

The requirements for the video equipment used in the examination will be consistent with Boiling Water Reactor Vessel and Internals Project-03, Revision 3, “Reactor Pressure Vessel and Internals Examination Guidelines,” Section 2.5, “Generic Standards for Visual Inspection of Reactor Pressure Vessel Internals, Components and Associated Repairs.” Parameters considered in the nozzle inner radii visual examinations include lighting, depth-of-field, field-of-view, magnification, and speed-of-camera movement.

Crack-like surface flaws exceeding the acceptance criteria of Table IWB-3512-1 are unacceptable for continued service unless the RPV meets the requirements of IWB-3142.2, IWB-3142.3, or IWB-3142.4.

3.4 Licensee's Bases for Relief

The specific inner nozzle radii subject to this relief request are identified in Table 1 of the licensee's letter dated September 3, 2003. The licensee stated:

> Visual examination of the inner radius region for the subject nozzles is limited because reactor internal piping configuration prevents placement of the camera in all positions necessary to examine surface M-N over the full circumference. The specific nozzle limitations and estimated coverage are indicated in [licensee's letter of September 3, 2003] Table 1.
>
> Performance of the volumetric examination results in significant personnel radiation exposure and does not result in a significant increase in the level of plant quality or safety. The redesign and construction of these nozzles and associated thermal sleeves would result in a hardship and unusual difficulty without compensating increase in level of quality or safety.
>
> Performance of the volumetric examination requires the examiner to enter and remain inside the biological shield penetration area around the nozzle for the duration of the ultrasonic examination that takes approximately 1.5 hours. Dose rates for the specified RPV nozzles are in the range of 200 mr/hr to 250 mr/hr with shielding in place. Performance of these examinations results in an estimated personnel exposure of about 3.5 Rem per inspection interval.
>
> Performance of a visual examination using remote cameras essentially eliminates any personnel exposure. The implementation of this relief request should reduce vessel examination time by approximately 18 hours, which translates to significantly reduced personnel radiation exposure and cost savings.
>
> All nozzle forgings were nondestructively examined during fabrication and have been previously examined using ultrasonic techniques specific to the nozzle configuration. No indications of fabrication or service related cracking have been observed as a result of these exams.
>
> Nozzle inner radius examinations are the only non-welded areas requiring examination on the RPV. This requirement was deterministically made early in the development of ASME Section XI, and applied to 100 percent of nozzles welded with full penetration welds. Fatigue cracking is the only applicable degradation mechanism for the nozzle inner radius region. For all nozzles other than feedwater, there is no significant thermal cycling during operation. Therefore from a risk perspective there is no need to perform volumetric examination on any nozzles other than feedwater or operational Control Rod Drive (CRD) returns. No service related cracking has been discovered in any of the Boiling Water Reactor (BWR) fleet plant nozzles other than feedwater and

> operational CRD returns. The six feedwater nozzle inner radius sections will continue to be examined in accordance with UT techniques developed and qualified with GE-NE-523-A71-0594-A Revision 1 (the NRC has approved this report under TAC No. MA6787). PSEG Nuclear [the licensee] believes that application of a visual examination alternative for the listed nozzle inner radius regions ensures an acceptable level of quality and safety.

3.5 NRC Staff's Evaluation

The licensee proposed an enhanced visual examination method for examining the accessible portion of the nozzle inner radius sections in lieu of the UT method required by Section XI, Table IWB-2500-1. The licensee stated that the estimated coverages obtainable with the proposed alternative are 60 percent for the jet pump instrumentation nozzles, 50 percent for the jet pump riser inlet nozzles, 40 percent for the core spray inlet nozzles, and 50 percent for the LPCI inlet nozzles. The licensee attributed the less than essentially 100 percent coverage to restrictions imposed by thermal sleeves, instrumentation lines, and adjacent nozzles on camera movement.

Although the subject nozzles were initially examined essentially 100 percent in the first 10-year interval with UT, the personnel participating in the examinations were exposed to a high dosage of radiation. The licensee stated that the dosage rates for the subject nozzles are in the range of 200 mr/hr to 250 mr/hr with protective shielding in place, which resulted in an estimated 3.5 Rem per inspection interval. By comparison, the radiation exposure to personnel using remotely-controlled cameras and data recording devices is essentially eliminated. The licensee will perform the examinations on the subject inner radii with remotely-controlled cameras.

The primary degradation mode in RPV nozzles is fatigue, which produces a network of hairline surface indications along the circumference of the nozzle at the inner radius section. Given the resolution sensitivity that will be used to perform the enhanced VT-1 alternative examination, it is highly unlikely that the licensee would not detect such flaws. The high magnification cameras and data recording equipment provide for real-time viewing and follow-up viewing, as necessary. The NRC staff has determined that the high-resolution images from the cameras provide adequate assurance of structural integrity and may be used in lieu of the Code-required UT examinations of the inner nozzle radii. The licensee has also committed to adhere to the acceptance criteria for flaw length in Table IWB-3512-1 of Section XI of the 1989 Edition of the ASME Code.

While the proposed visual examination on the components will be limited from 40 to 60-percent of the required coverage, the NRC staff believes the coverage will provide reasonable assurance that flaws of significant size will be detected. When flaws are initiated by fatigue mechanism, they typically are encountered over a significant portion of the nozzle circumference, as was the case for cracking of the feedwater nozzles addressed in NUREG-0619, "Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking." The NRC staff also recognizes that industry experience has shown no cracking in the subject nozzle inner radii sections, and that the subject nozzles are not subjected to significant thermal cycling. All subject nozzle forgings were nondestructively examined during fabrication and have been previously examined using UT techniques specific for the configuration.

In addition, the licensee recently received authorization to use the same enhanced visual technique on other RPV nozzles in an NRC letter to Roy A. Anderson, PSEG Nuclear LLC, dated June 9, 2003. These nozzles were examined over the entire surface section (100-percent coverage) and no indications or service-related cracking were found. Therefore, the NRC staff has determined that imposing the Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

4.0 CONCLUSION

Based on the above evaluation, the NRC staff has determined that the enhanced visual examination to be performed with remotely-controlled cameras having demonstrated capabilities of 1-mil width resolution will provide reasonable assurance of the continued structural integrity of the subject inner nozzle radii. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative HC-RR-B12 is authorized for the subject inner radii at Hope Creek for the second 10-year ISI interval which ends December 12, 2007.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this relief request remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: D. Naujock

Date: March 22, 2004