June 9, 2003

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 on May 9 and 15, 2003, 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)(i) for the Hope Creek Nuclear Generating Station (Hope Creek). Specifically, Relief Request HC-RR-B11 proposed an alternative examination using enhanced remote visual equipment that is capable of a 1-mil (0.001 inch) wire resolution. The visual examination will be performed on essentially 100% of the Hope Creek reactor vessel inner nozzles radii. The request for relief is for the second 10-year inservice inspection interval, which commenced on December 13, 1997.

Based on the information provided, the U.S. Nuclear Regulatory Commission (NRC) staff concludes that your proposed alternative enhanced visual inspection, as described in Relief Request HC-RR-B11, and provides an acceptable level of quality and safety. Therefore, the NRC staff authorizes you to use the proposed alternatives pursuant to 10 CFR 50.55a(a)(3)(i) 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, George Wunder, at 301-415-1494.

Sincerely,

/RA/

James W. Clifford, 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

June 9, 2003

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 on May 9 and 15, 2003, 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)(i) for the Hope Creek Nuclear Generating Station (Hope Creek). Specifically, Relief Request HC-RR-B11 proposed an alternative examination using enhanced remote visual equipment that is capable of a 1-mil (0.001 inch) wire resolution. The visual examination will be performed on essentially 100% of the Hope Creek reactor vessel inner nozzles radii. The request for relief is for the second 10-year inservice inspection interval, which commenced on December 13, 1997.

Based on the information provided, the U.S. Nuclear Regulatory Commission (NRC) staff concludes that your proposed alternative enhanced visual inspection, as described in Relief Request HC-RR-B11, and provides an acceptable level of quality and safety. Therefore, the NRC staff authorizes you to use the proposed alternatives pursuant to 10 CFR 50.55a(a)(3)(i) 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, George Wunder, at 301-415-1494.

Sincerely,

/RA/

James W. Clifford, 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

DISTRIBUTION:

PUBLIC	RLaufer	TChan	GMeyer, RGN-1	OGC	CRaynor
ACRS	JClifford	ZFu	GHill (2)	GWunder	PDI-2 R/F

ADAMS ACCESSION NUMBER: ML031600421 * SE input provided on May 16, 2003

OFFICE	PDI-2/PM	PDI-2/LA	EMCB/SC*	OGC	PDI-2/SC
NAME	REnnis for GWunder	CRaynor	TChan	RHoefling	JBoska forJClifford
DATE	5/21/03	5/23/03	05/15/03	6/2/03	6/6/03

OFFICIAL RECORD COPY

cc:

Mr. Timothy J. O'Connor
Vice President - Operations
PSEG Nuclear - X15
P.O. Box 236
Hancocks Bridge, NJ 08038

Mr. John T. Carlin
Vice President - Engineering
PSEG Nuclear - N10
P.O. Box 236
Hancocks Bridge, NJ 08038

Mr. David F. Garchow
Vice President - Projects and Licensing
PSEG Nuclear - N28
P.O. Box 236
Hancocks Bridge, NJ 08038

Mr. Gabor Salamon
Manager - Nuclear Safety and Licensing
PSEG Nuclear - N21
P.O. Box 236
Hancocks Bridge, NJ 08038

Jeffrie J. Keenan, Esquire
PSEG Nuclear - N21
P.O. Box 236
Hancocks Bridge, NJ 08038

Ms. R. A. Kankus
Joint Owner Affairs
PECO Energy Company
Nuclear Group Headquarters KSA1-E
200 Exelon Way
Kennett Square, PA 19348

Lower Alloways Creek Township
c/o Mary O. Henderson, Clerk
Municipal Building, P.O. Box 157
Hancocks Bridge, NJ 08038

Dr. Jill Lipoti, Asst. Director
Radiation Protection Programs
NJ Department of Environmental
Protection and Energy
CN 415
Trenton, NJ 08625-0415

Brian Beam
Board of Public Utilities
2 Gateway Center, Tenth Floor
Newark, NJ 07102

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Senior Resident Inspector
Hope Creek Nuclear Generating Station
U.S. Nuclear Regulatory Commission
Drawer 0509
Hancocks Bridge, NJ 08038

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR RELIEF HC-RR-B11

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 on May 9 and 15, 2003, 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)(i) for the Hope Creek Nuclear Generating Station (Hope Creek). Specifically, Relief Request HC-RR-B11 proposed an alternative examination using enhanced remote visual equipment that is capable of a 1-mil (0.001 inch) wire resolution. The visual examination will be performed on essentially 100% of the Hope Creek reactor pressure vessel (RPV) inner- nozzles radii. The May 9, 2003, letter superceded information provided in PSEG's April 14, 2003, request for relief. Relief Request HC-RR-B11 applies to the second 10-year inservice inspection (ISI) interval, which commenced on December 13, 1997.

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)(i), 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

Enclosure

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. 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

This request covers a total of 10 RPV inner-nozzle radii. The components are from the following systems:

- Reactor vessel head vent nozzle (1)
- Main steam nozzles (4)
- Control Rod Drive (CRD) return nozzle (1)
- Reactor recirculation outlet nozzles (2)
- Spray head nozzles (2)

The specific nozzle designations are: RPV1-NIAIR, -NIBIR, -N3AIR, -N3BAIR, -N3CAIR, -N3DAIR, -N6AIR, -N6BIR, -N7IR, and -N9A.

3.2 Code Requirements for Which Relief Is Requested

The 1989 Edition of ASME Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.100 requires a volumetric examination of the RPV inner-nozzle radius section. Relief is requested from the requirements to perform the volumetric examination of the inner-nozzle radii for the nozzles listed in Section 3.1.

3.3 Licensee's Proposed Alternative

The licensee proposes to perform a visual examination per the requirements of the approved Hope Creek ISI Non-destructive Examination (NDE) Program. The required visual coverage will be essentially 100% (greater than 90% for each nozzle) of the surface M-N as shown in Figure IWB-2500-7 (a) through (d) of the 1989 Edition of ASME Section XI, in lieu of the volumetric examinations required by Table IWB-2500-1, Examination Category B-D, Item B3.100 of ASME Section XI.

The components identified in Section 3.1 will receive enhanced visual examinations. The examinations will be performed remotely using 8x magnification video equipment. The resolution sensitivity for this remote examination will be established using a 1-mil diameter wire standard similar to that used for other reactor pressure vessel internal examinations intended to detect cracking. Crack-like surface flaws exceeding the acceptance criteria of Table IWB-3512-1 are unacceptable for continued service unless the reactor vessel meets the requirements of IWB-3142.2, IWB-3142.3 or IWB-3142.4.

3.4 Licensee's Bases for Alternative (as stated):

> All nozzle forgings were nondestructively examined during fabrication and have previously been examined using inservice ultrasonic techniques specific to the nozzle configuration. No indication of fabrication defects or service related cracking has been detected by these examinations.
>
> 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 and operational CRD [control rod drive] returns.
>
> No service related cracking has ever been discovered in any of the BWR (boiling water reactor) fleet plant nozzles other than on feedwater or operational CRD returns. The six feedwater nozzle inner radius sections will continue to be examined with UT [ultrasonic testing] techniques developed and qualified with GE-NE-523-A71-0594-A, Revision 1 (the NRC has approved this report under TAC No. MA6787).

PSEG further stated that application of a visual examination alternative for the listed nozzle inner-radius regions ensures an acceptable level of quality and safety. The licensee also stated that for Table IWB-3512-1, the depth of a crack initiation is assumed to be one-half of the measured length of the crack initiation. Crack-like surface flaws found exceeding the acceptance criteria of Table IWB-3512-1 will be unacceptable for continued service unless the reactor vessel meets the requirements of IWB-3142.2, IWB-3142.3 or IWB-3142.4.

3.5 NRC Staff's Evaluation

In the mid 1970s, fatigue-initiated cracking was discovered in the nozzle inner-radius section of feedwater nozzles of 18 BWR vessels. The cracks were found using visual examinations. UT failed to reveal the presence of these cracks. The shortcomings with UT prompted the NRC to issue NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," which modified inspection requirements for these components.

In NUREG-0619, the NRC staff concluded that UT of the vessel nozzle inner-radius section involves complex geometries, long examination metal paths, and inherent UT beam spread, scatter, and attenuation. During the intervening years, improvements in UT technologies were introduced (e.g., computer modeling, tip diffraction, and phased array scanning) which improved the quality of the examination for this component. However, the area remains difficult to examine completely.

The NRC staff finds that even with vessel examinations using improved NDE technology from the outside surface, the complex geometry of the RPV nozzle inner-radius sections prevents complete UT coverage. For the RPV nozzle inner-radii, the licensee proposed to perform an enhanced visual examination with “essentially 100% coverage” in lieu of UT. The enhanced aspect of the examination is to use 8x magnification video equipment to examine the inner-radii. The resolution sensitivity for this remote, in-vessel exam will be established using a 1-mil diameter wire.

The demonstration provides assurance that an examiner would recognize a crack if one were to exist. In a letter dated May 15, 2003, the licensee indicated that examination conditions, including lighting, field of view, magnification, depth of field and speed of camera movement, will be consistent with the conditions used for the demonstration of examiner competency.

The primary degradation mechanism in RPV nozzles is fatigue, which produces hairline surface indications that form networks along the circumference of the nozzle at the inner-radius section. The licensee will be using high magnification cameras that have demonstrated resolution capability of detecting a 1-mil wire or equivalent and will be performing the examination over essentially 100% (greater than 90%) of the nozzle inner-radius surface area. Given the 1-mil resolution capability of the EVT system, it is highly unlikely that the licensee would not detect detrimental flaws. The staff has determined that the high resolution image from the camera, as demonstrated, will provide adequate assurance of structural integrity and may be used in lieu of UT for the inner-nozzle radius region.

4.0 CONCLUSION

Based on its review, the NRC staff has determined that the proposed Relief Request HC-RR-B11 submitted on April 14, 2003, as supplemented on May 9 and 15, 2003, 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 remainder of the second 10-year ISI interval at Hope Creek. The NRC staff’s authorization is limited to those components described in Section 3.1.

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: Z. Fu

Date: June 9, 2003