May 7, 2008

Mr. William Levis
President & Chief Nuclear Officer
PSEG Nuclear LLC - N09
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION, REQUEST FOR ADDITIONAL INFORMATION RELATED TO RELIEF REQUESTS HC-I3R-01 AND HC-I3R-02 (TAC NOS. MD7503 AND MD7504)

Dear Mr. Levis:

By letter dated December 12, 2007, PSEG Nuclear LLC (PSEG) requested relief from certain requirements specified in Section XI of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* for the third inservice inspection interval at Hope Creek Generating Station.

The Nuclear Regulatory Commission (NRC) staff is reviewing your submittal and has determined that additional information is needed to complete its review. The specific questions are found in the enclosed request for additional information (RAI). The RAI questions were previously provided in draft form to PSEG via e-mail on April 11, 2008. The draft questions were sent to ensure that the questions were understandable, the regulatory basis for the questions was clear, and to determine if the information was previously docketed. A conference call between the NRC staff and the PSEG staff was held on April 15, 2008, to discuss the questions.

We request that the additional information be provided by June 11, 2008. The response timeframe was discussed with Mr. Paul Duke of your staff on April 21, 2008. If circumstances result in the need to revise your response date, or if you have any questions, please contact me at (301) 415-1420.

Sincerely,

/ra/

Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosure: RAI

cc w/encl: See next page

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Division of Operating Reactor Licensing
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Docket No. 50-354
Enclosure: RAI
cc w/encl: See next page

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DATE	5/1/08	5/1/08	5/7/08

OFFICIAL RECORD COPY

Hope Creek Generating Station

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REQUEST FOR ADDITIONAL INFORMATION

RELATED TO RELIEF REQUESTS HC-I3R-01 AND HC-I3R-02

FOR THIRD TEN-YEAR INSERVICE INSPECTION INTERVAL

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

By letter dated December 12, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML073531254), PSEG Nuclear LLC (PSEG or the licensee) requested relief from certain requirements specified in Section XI of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code) for the third inservice inspection (ISI) interval at Hope Creek Generating Station (HCGS).

The Nuclear Regulatory Commission (NRC) staff has reviewed the information the licensee provided that supports the proposed relief requests and would like to discuss the following issues to clarify the submittal.

Relief Request HC-I3R-01

1. Per Regulatory Guide (RG) 1.193, Revision 2 (October 2007), Code Case N-578-1 is listed as an unacceptable Section XI Code Case. Please provide a justification for the application of Subarticle -2430 of Code Case N-578-1. Explain how the use of Code Case N-578-1 provides a "more refined methodology for implementing necessary additional examinations."

2. Have any welds that were selected for inspection in the Risk-Informed Inservice Inspection (RI-ISI) program that was approved by the NRC staff in Reference 1 been removed from the population of welds that will be inspected during the third ten-year interval? If so, why were the welds removed from the population of welds to be inspected?

3. Have any welds that were not selected for inspection in the RI-ISI program that was approved by the NRC staff in Reference 1 been selected for inspection during the third ten-year interval? If so, why were the welds added to the population of welds to be inspected?

4. The relief request states that:

> The Risk Impact Assessment completed as part of the original baseline RISI Program was an implementation/transition check on the initial impact of converting from a traditional ASME Section XI program to the new RISI methodology. For the Third Interval ISI update, there is no transition occurring between two different methodologies, but rather, the currently approved RISI methodology and evaluation will be maintained for the new interval. As such, the original risk impact assessment process is not impacted by the new interval and does not require update.

Enclosure

The NRC staff does not concur with the implication that, if there is no change in methodology, the change in risk assessment is not part of the living process. RG 1.178, Standard Review Plan (SRP) 3.9.8, and Electric Power Research Institute (EPRI) Topical Report TR-112657 (References 2, 3, and 4) require an evaluation of the change in risk arising from the proposed change in the ISI program. Please provide a discussion on the potential change in risk between the RISI program proposed for implementation in the third interval and the ASME Section XI requirements from which relief was granted in Reference 1. If inspections were discontinued or relocated between the second and third intervals' RISI programs, please provide an estimate of the change in risk.

5. The relief request states that:

> As an added measure of assurance, any new systems, portions of systems, or components being included in the RISI Program for the Third Interval will be added to the Risk Impact Assessment performed during the previous interval. These components will be addressed within the evaluation at the start of the new interval to assure that the new Third Interval RISI element selection provides an acceptable overall change-in-risk...

The results of the evaluations described above should be part of a request for relief to support the required finding that the proposed program provides an acceptable level of quality and safety. Please provide a brief description of these evaluations and an overview of the results.

6. The relief request states that:

> These portions of the RISI Program have been and will continue to be reevaluated and revised as major revisions of the site PRA [probabilistic risk assessment] occur and modifications to plant configuration are made. The Consequence Evaluation, Degradation Mechanism Assessment, Risk Ranking, and Element Selection steps encompass the complete living program process...

Please provide the date of the last reevaluation and revision that is described above and a brief description of the results of the reevaluations and revisions undertaken at that date.

References:

1) Letter from Darrell J. Roberts (NRC) to A. Christopher Bakken, III (PSEG) dated December 8, 2004, "Hope Creek Generating Station - Implementation of a Risk-Informed Inservice Inspection Program" (ADAMS Accession No. ML043080161)

2) Regulatory Guide 1.178, September 2003, "An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping" (ADAMS Accession No. ML032510128)

3) NUREG-0800, SRP Chapter 3.9.8, September 2003, "Standard Review Plan for the Review of Risk-Informed Inservice Inspection of Piping" (ADAMS Accession No. ML032510135)

4) EPRI Topical Report TR-112657, Revision B-A, January 2000, "Revised Risk-Informed Inservice Inspection Evaluation Procedure" (ADAMS Accession No. ML013470102)

Relief Request HC-I3R-02

1. Article IWF-5000, Subsections IWF-5200(c) and IWF-5300(c) clearly state that integral and non-integral attachments for snubbers (including lugs, bolting, pins, and clamps), shall be examined in accordance with the requirements of Subsection IWF. Please explain whether and how these requirements will be met.

2. The relief request and HCGS Technical Specification (TS) 3/4.7.5 do not address the requirements of ASME/American Nuclear Standards Institute (ANSI) *Code for Operation and Maintenance of Nuclear Power Plants* (OM), Part 4 (OM-4), Section 3.2.4, specifically Section 3.2.4.2, "Test Failure Mode Groups." Please explain how TS 3/4.7.5 meets these requirements.

3. Surveillance Requirement (SR) 4.7.5.e.2, provides an optional functional testing of snubbers in accordance with TS Figure 4.7.5-1. Please verify that: (1) this plan is equivalent to the 37 testing sample plan of OM-4; and (2) explain whether and how the requirement of additional sampling of at least one-half the size of the initial sample lot as required by OM-4, Paragraph 3.2.3.2(b), will be met while using TS Figure 4.7.5-1.

4. SR 4.7.5.c, "Visual Inspection Acceptance Criteria," states, in part, that "the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specifications 4.7.4.f." Please confirm that the reference to "Specifications 4.7.4.f" is in error and should actually be "Specification 4.7.5.f." If SR 4.7.5.c is in error, please enter this item into the corrective action program and provide a regulatory commitment to request a license amendment to correct the error.