June 19, 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 - ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT IMPACT FOR EXEMPTION FROM THE REQUIREMENTS OF 10 CFR PART 50, APPENDIX G (TAC NOS. MB0645 AND MB0646)

Dear Mr. Keiser:

Enclosed is a copy of the Environmental Assessment and Finding of No Significant Impact related to your application dated December 1, 2000, as supplemented on February 12, May 7, and May 14, 2001, for an exemption from the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G, for the Hope Creek Generating Station. Your application requested the exemption in order to revise the methodology used to determine the reactor pressure vessel pressure-temperature (P-T) limits. Specifically, the proposed exemption would allow the use of American Society of Mechanical Engineers (ASME) Code Cases N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1," and N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1," in lieu of some of the specific requirements in 10 CFR Part 50, Appendix G.

The assessment is being forwarded to the Office of the Federal Register for publication.

Sincerely,

/RA/

Richard B. Ennis, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosure: Environmental Assessment

cc w/encl: See next page

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Richard B. Ennis, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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cc w/encl: See next page

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Hope Creek Generating Station

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UNITED STATES NUCLEAR REGULATORY COMMISSION

PSEG NUCLEAR LLC

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

ENVIRONMENTAL ASSESSMENT AND FINDING OF

NO SIGNIFICANT IMPACT

The U.S. Nuclear Regulatory Commission (NRC) is considering issuance of an exemption from certain requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G, for Facility Operating License No. NPF-57, issued to PSEG Nuclear LLC, (the licensee) for operation of the Hope Creek Generating Station (HCGS), located in Salem County, New Jersey.

ENVIRONMENTAL ASSESSMENT

Identification of the Proposed Action:

Title 10 of the *Code of Federal Regulations*, Part 50, Appendix G, requires that pressure-temperature (P-T) limits be established for reactor pressure vessels (RPVs) during normal operating and hydrostatic or leak rate testing conditions. Specifically, 10 CFR Part 50, Appendix G, states, “The appropriate requirements on both the pressure-temperature limits and the minimum permissible temperature must be met for all conditions.” The purpose of 10 CFR Part 50, Appendix G, is to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. This is accomplished through these regulations that, in part, specify fracture toughness requirements for ferritic materials of the reactor coolant pressure boundary. Appendix G of 10 CFR Part 50 specifies that the requirements for these limits are the American

Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Appendix G Limits.

The proposed action would exempt HCGS from application of specific requirements of 10 CFR Part 50, Appendix G, and would substitute use of ASME Code Cases N-588 and N-640 as alternatives pursuant to 10 CFR 50.60(b).

The proposed action is in accordance with the licensee's application for exemption dated December 1, 2000, as supplemented by letters dated February 12, May 7, and May 14, 2001.

The Need for the Proposed Action:

The proposed action is needed to allow the licensee to implement ASME Code Cases N-588 and N-640 in order to revise the method used to determine the P-T limits.

Code Case N-588, “Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1,” amends the provisions of the 1989 Edition of ASME Section XI, Appendix G, by permitting the postulation of a circumferentially oriented reference flaw as the limiting flaw in a RPV circumferential weld for the purpose of establishing RPV P-T limits. The 1989 Edition of ASME Section XI, Appendix G, would require that such a reference flaw be postulated as an axially oriented flaw in the circumferential weld. The licensee addressed the technical justification for this exemption by citing industry experience and aspects of RPV fabrication which support the postulation of circumferentially oriented flaws for these welds. The reference flaw is a postulated flaw that accounts for the possibility of a prior existing defect that may have gone undetected during the fabrication process. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic and overly conservative, because the length of the flaw is 1.5 times the vessel wall, which is much longer than the width of the circumferential weld. Industry experience with the repair of weld indications found during preservice inspection, inservice

nondestructive examinations, and data taken from destructive examination of actual vessel welds confirms that any remaining defects are small, laminar in nature, and do not cross transverse to the weld bead. Therefore, any postulated defects introduced during the fabrication process, and not detected during subsequent nondestructive examinations, would only be expected to be oriented in the direction of weld fabrication. ASME Code Case N-588 also provides appropriate procedures for determining the stress intensity factors for use in developing RPV P-T limits per ASME Code, Section XI, Appendix G, procedures. The procedures allowed by ASME Code Case N-588 are conservative and provide a margin of safety in the development of RPV P-T operating and pressure test limits that will prevent nonductile fracture of the vessel.

Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1," amends the provisions of ASME Section XI, Appendix G, by permitting the use of the K_{Ic} equation as found in Appendix A in ASME Section XI, in lieu of the K_{Ia} equation as found in Appendix G in ASME Section XI. Use of the K_{Ic} equation in determining the lower bound fracture toughness in the development of the P-T operating limits curve is more technically correct than the use of the K_{Ia} equation since the rate of loading during a heatup or cooldown is slow and is more representative of a static condition than a dynamic condition. Use of K_{Ia} was justified by the initial conservatism of the K_{Ia} equation since 1974 when the equation was codified. This initial conservatism was necessary due to the limited knowledge of RPV materials. Since 1974, additional knowledge has been gained about RPV materials, which demonstrates that the lower bound on fracture toughness provided by the K_{Ia} equation is well beyond the margin of safety required to protect the public health and safety from potential RPV failure. The lower bound K_{Ic} fracture toughness provides an adequate margin of safety to protect the public health and safety from potential RPV failure.

The staff has determined that, pursuant to 10 CFR 50.12(a)(2)(ii), the underlying purpose of the regulation to protect the integrity of the reactor coolant pressure boundary will continue to be served with the implementation of Code Cases N-588 and N-640.

Environmental Impacts of the Proposed Action:

The NRC has completed its evaluation of the proposed action and concludes that the exemption and implementation of the proposed alternatives as described above are consistent with the intent of the applicable regulations and would provide an acceptable margin of safety against brittle failure of the HCGS RPV. Therefore, the proposed action will not have a significant impact on the environment.

The proposed action will not significantly increase the probability or consequences of accidents, no changes are being made in the types of any effluents that may be released offsite, and there is no significant increase in occupational or public radiation exposure. Therefore, there are no significant radiological environmental impacts associated with the proposed action.

With regard to potential nonradiological environmental impacts, the proposed action does not involve any historic sites. It does not affect nonradiological plant effluents and has no other environmental impacts. Therefore, there are no significant nonradiological impacts associated with the proposed action.

Accordingly, the NRC concludes that there are no significant environmental impacts associated with the proposed action.

Alternatives to the Proposed Action:

As an alternative to the proposed action, the staff considered denial of the proposed action (i.e., the “no-action” alternative). Denial of the application would result in no change in current environmental impacts. The environmental impacts of the proposed action and the alternative action are similar.

Alternative Use of Resources:

This action does not involve the use of any resources not previously considered in the Final Environmental Statement for the HCGS.

Agencies and Persons Consulted:

In accordance with its stated policy, on June 7, 2001, the staff consulted with the New Jersey State official, Mr. Dennis Zannoni, of the New Jersey Department of Environmental Protection, regarding the environmental impact of the proposed action. The State official had no comments.

FINDING OF NO SIGNIFICANT IMPACT

On the basis of the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letter dated December 1, 2000, as supplemented by letters dated February 12, May 7, and May 14, 2001. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the Internet at the NRC web site, http://www.nrc.gov/NRC/ADAMS/index.html. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public

Document Room (PDR) Reference staff at 1-800-397-4209, 301-415-4737 or by email to pdr@nrc.gov.

Dated at Rockville, Maryland, this 19th day of June 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard B. Ennis, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation