April 16, 2003

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

SUBJECT: HOPE CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: ONE-TIME EXTENSION OF TYPE A INTEGRATED LEAK RATE TEST INTERVAL (TAC NO. MB6551)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 147 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 9, 2002, as supplemented November 22, 2002, and December 6, 2002. The amendment revises the TSs to allow a one-time extension of the Type A Integrated Leak Rate Test interval to 15 years.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

George F. Wunder, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures: 1. Amendment No. 147 to License No. NPF-57
2. Safety Evaluation

cc w/encls: See next page

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* Safety Evaluation dated February 7, 2003
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*** See Previous Concurrence

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PSEG NUCLEAR LLC

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 147
License No. NPF-57

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by PSEG Nuclear LLC dated October 9, 2002, as supplemented November 22, and December 6, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-57 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 147, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by JBoska for/

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 16, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 147

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
6-16b	6-16b

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 147 TO FACILITY OPERATING LICENSE NO. NPF-57

PSEG NUCLEAR LLC

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated October 9, 2002, as supplemented on November 22, and December 6, 2002, PSEG Nuclear LLC (PSEG, or the licensee) submitted a request for changes to the Hope Creek Generating Station (Hope Creek) Technical Specifications (TSs). The requested changes would revise the TSs to allow a one-time extension of the Type A Integrated Leak Rate Test interval to 15 years. A Type A test is an overall (integrated) leakage rate test of the containment structure. The leak rate testing requirements of Option B of Appendix J, and the containment inservice inspection (ISI) requirements mandated by Title 10 of the *Code of Federal Regulations* (CFR) Part 50.55a complement each other in ensuring the leak-tightness and structural integrity of the containment.

2.0 REGULATORY EVALUATION

Part 50, Appendix J, Option B of 10 CFR requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. Hope Creek TS 6.8.4.f, “Primary Containment Leakage Rate Testing Program,” requires that leakage rate testing be performed as required by 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, “Performance-Based Containment Leak-Test Program,” dated September 1995. This RG endorses, with certain exceptions, Nuclear Energy Institute (NEI) report 94-01, Revision 0, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” dated July 26, 1995.

A Type A test is an overall (integrated) leakage rate test of the containment structure. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months in certain circumstances. The most recent two Type A tests at Hope Creek have been successful, so the current interval requirement is 10 years.

The licensee is requesting additions to TS 6.8.4.f, which would indicate that they are allowed to take an exception from the guidelines of RG 1.163 regarding the Type A test interval. Specifically, the proposed TS states that the first Type A test performed after April 12, 1994 (the date of the latest test), shall be performed no later than April 12, 2009.

3.0 TECHNICAL EVALUATION

3.1 Risk Impact

The licensee performed a risk impact assessment of extending the Type A test interval to 15 years. The assessment was provided to the staff in the October 9, 2002, application for license amendment. Additional analysis and information were provided by the licensee in letters dated November 22, and December 6, 2002. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute (EPRI) technical review (TR)-104285, “Risk Impact Assessment of Revised Containment Leak Rate Testing,” and RG 1.174, “An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.”

The basis for the current 10-year test interval is provided in Section 11 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11 of NEI 94-01 states that NUREG-1493, “Performance-Based Containment Leak-Test Program,” dated September 1995, provided the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the Nuclear Regulatory Commission’s (NRC or Commission) rulemaking basis, NEI undertook a similar study. The results of that study are documented in EPRI Research Project Report TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The EPRI study estimated that relaxing the test frequency from 3-in-10 years to 1-in-10 years will increase the average time that a leak detectable only by a Type A test goes undetected from 18 to 60 months. Since Type A tests only detect about three percent of leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a 10 percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage for the PWR and BWR representative plants confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from 3-in-10 years to 1-in-20 years leads to an “imperceptible” increase in risk on the order of 0.2 percent and a fraction of one-person rem per year.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem/year frequency. The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak was present. Since the Option B rulemaking in 1995, the staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in risk-informed changes to a plant’s licensing basis. The licensee has proposed using RG 1.174 to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6}/year and increases in large early release frequency (LERF) less than 10^{-7}/ year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee has estimated the change in LERF for the proposed change and the cumulative change from the original 3-in-10 year interval. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. The licensee estimated the change in the conditional containment failure probability for the proposed change to demonstrate that the defense in depth philosophy is met.

The licensee provided an analysis which estimated all of these risk metrics and whose methodology is consistent with previously approved submittals. The staff drew the following conclusions from the licensee's analysis associated with extending the Type A test frequency:

1. A slight increase in risk is predicted when compared to that estimated from current requirements. Given the change from a 3-in-10 year test interval to a 1-in-15 year test interval, the increase in the total integrated plant risk, in person-rem/year, is estimated to be about 0.2 percent. This increase is comparable to that estimated in NUREG-1493, in which it was concluded that a reduction in the frequency of tests from 3-in-10 years to 1-in-20 years leads to an "imperceptible" increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. The increase in LERF resulting from a change in the Type A test interval from the original 3-in-10 years to 1-in-15 years is estimated to be 5.2×10^{-8}/year. However, there is some likelihood that the flaws in the containment estimated as part of the Class 3b frequency would be detected as part of the IWE/IWL visual examination of the containment surfaces as identified in American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Subsections IWE/IWL. The most recent visual examination of the Hope Creek containment was performed in spring 2000. The next scheduled containment visual examination is spring 2003. Visual examinations are expected to be effective in detecting large flaws in the visible regions of the containment, and would reduce the impact of the extended test interval on LERF. The licensee performed additional risk analysis to consider the potential impact of corrosion in inaccessible areas of the containment shell on the proposed change. The risk analysis considered the likelihood of an age-adjusted flaw that would lead to a breach of the containment. The risk analysis also considered the likelihood that the flaw was not visually detected but could be detected by a Type A test. The increase in LERF associated with corrosion events is estimated to be about 1×10^{-9}/year. The staff concludes that increasing the Type A interval to 15 years results in only a small change in LERF and is consistent with the acceptance guidelines of RG 1.174.

3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The licensee estimates the change in the conditional containment failure probability to be an increase of about 0.6 percentage points for the cumulative change of going from a test interval of 3-in-10 years to 1-in-15 years. The staff finds that the defense-in-depth philosophy is maintained based on the change in the conditional containment failure probability for the proposed amendment.

Based on these conclusions, the staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines while maintaining the defense-in-depth philosophy of RG 1.174 and, therefore, is acceptable.

3.2 Inservice Inspection for Primary Containment Integrity

Hope Creek is a General Electric boiling water reactor (BWR) enclosed within in a Mark I-type containment. Hope Creek's Updated Final Analysis Report (UFSAR) Section 3.8.2 describes the primary containment. The steel containment is an ASME Code Class MC vessel designed to house the Nuclear Steam Supply System (NSSS). The steel containment is a part of the Primary Containment System, which limits the release of fission products following postulated design basis accidents.

In addition to Type A leak tests, containment integrity is ensured through both visual inspection and local leak rate testing. The reactor containment inspections will continue to be conducted in accordance with the requirements of ASME Section XI, Subsection IWE and IWL. The existing Type B and C containment penetration-testing program will continue to be performed in accordance with regulatory requirements.

The licensee provided information related to the ISI of the containment and discussed potential areas of weakness in the containment that may not be apparent in the risk assessment. In addition, in a letter dated December 6, 2002, the licensee provided responses to the staff's request for additional information (RAI) to address five specific issues. These five issues describe in more detail how visual inspection and local leak rate testing complement the ILRT.

1. ISI Program at Hope Creek - Methods and Schedule

The licensee stated that a general visual inspection is performed on the Hope Creek containment in accordance with IWE of the ASME Code, Section XI, 1998 Edition with the 1998 Addenda. The use of this edition of the Code was requested through PSEG's relief request RR-E1, via letter No. LR-N99-0409, dated October 7, 1999, and approved by NRC's letter dated June 6, 2000. The inspection interval was established such that it coincided with the Hope Creek inspection for Subsections IWA, IWB, IWC, IWE and IWF of ASME Code, Section XI. The interval start date was December 13, 1997, and the interval ends after refueling outage (RFO) 13 which is scheduled for spring 2006. After RFO13, the program will be upgraded to reflect the latest Edition and Addenda to the ASME Code, Section XI incorporated by reference in the regulations in accordance with 10 CFR 50.55a for the next 120-month inspection interval (Third Inspection Interval).

All required IWE examinations were completed during RFO9; this marked the completion of the first inspection period for the 10-year interval. The areas and items subject to examination included the accessible containment surface areas, including structural attachments and penetrations, pressure-retaining bolting, and Class MC supports.

The NRC technical staff reviewed the licensee's request for amendment and their December 6, 2002, response to the RAI. The staff has concluded that the ISI program at Hope Creek has been established and is being conducted in accordance with Section XI of the ASME Code and that the methods and schedule employed are acceptable.

2. Implementing IWE-1240 at Hope Creek - Augmented Examination

Certain areas of containment may be more susceptible than others to corrosion. To address this potential problem, IWE-1240 requires licensees to identify any areas of the containment that might require augmented examination. In their December 6, 2002, letter, the licensee stated that the Hope Creek containment ISI program has not identified any surface areas classified as augmented examination areas in accordance with Table IWE-2500-1, Examination Category E-C.

The licensee developed a formal technical position to evaluate the results of the inspections that are conducted against the requirements of paragraph IWE-1240 of the ASME Code, Section XI, 1998 Edition, including 1998 Addenda. The technical position was based upon a review of industry and plant-specific experience, two formal engineering evaluations, identification and consideration of the design corrosion allowances, and review of a specific evaluation documenting the condition of the applied surface coatings within the primary containment. This technical position supports the licensee's contention that initially no areas of the containment should have been identified as requiring augmented inspection. The visual examinations performed to date have not identified any areas requiring re-classification in accordance with Paragraph IWE-1240 as augmented examination areas subject to the requirements of Table IWE 2500-1, Examination Category E-C.

The NRC staff has reviewed both the licensee's application for amendment and their response to the RAI. The staff has concluded that the licensee has adequately applied the requirements of the ASME Code in making their determination that no areas of the containment require augmented inspection.

3. Schedule for Examination and Testing of Seals, Gaskets and Bolts Providing Containment Pressure Boundary Integrity

Type A containment leak tests evaluate the integrity of the entire containment; however, the most likely source of a containment leak is through a penetration. To address this under the 10 CFR 50 Appendix J, Option B program, those Type B penetrations that use resilient seals, gaskets, etc., are tested within the guidelines provided by Option B and RG 1.163. Most of the Type B penetrations at Hope Creek are on an extended test frequency based on performance with a percentage of the total population tested each refueling outage. The testing program is set up so that 100 percent of all components are tested during each 10-year interval. Those components that do not fall under extended test frequencies are tested at least once every 30 months. Components that are not under extended test frequencies are either penetrations that are disassembled and reassembled each outage or are components that have not demonstrated acceptable performance history per the Primary Containment Leakage Rate Program. Components that are disassembled and reassembled during an outage receive an as-found test prior to any work and an as-left test after all work is completed.

All the pressure-retaining bolting is included in Examination Category E-A, in the 1998 Edition of the ASME Code, Section XI, Subsection IWE. All pressure-retaining bolting is examined at least once each inspection period (40 months), as scheduled in the ISI Long-Term Plan, either in place or removed.

The staff has evaluated the licensee's application for amendment and their December 6, 2002, response to the RAI. The staff has concluded that the licensee has established a schedule for the examination of seals and gaskets and for the examination and testing of bolted connections that is consistent with the applicable sections of both the Code and the regulations and is, therefore, acceptable.

4. Integrity of Stainless Steel Bellows

The staff has determined that two-ply stainless steel bellows can be susceptible to trans-granular stress corrosion cracking, and that the leakage through them may not be detectable by Type B testing (see NRC Information Notice (IN) 92-20, “Inadequate Local Leak Rate Testing”). The licensee stated that the bellows assemblies are examined visually, per the requirements of the IWE program. These bellows assemblies are also local leak rate tested as part of the 10 CFR 50, Appendix J, Option B program. Although most of the bellows assemblies are eligible for extended frequency testing, the licensee has elected to conduct bellows testing every refueling outage with approximately one-half of the assemblies tested each outage. This frequency was established for two reasons: the first was due to the potential concerns in IN 92-90, and the second was to ensure that the assemblies met the ASME Section XI, Subsection IWC system pressure test requirement of being examined at least once each inspection period.

There are two bellows assemblies at Hope Creek that have been identified with minor leakage. The first one was discovered in March 1994. Extensive examination and troubleshooting was performed on this assembly to determine the source of the leakage but no source of external leakage was noted. It is suspected that the leakage is minor, through wall leak(s) on the inner ply of this assembly. This same bellows assembly was also tested as part of the ILRT performed at the end of this outage and no new source of external leakage was located. This assembly, because of a previously identified leak, is not eligible for extended frequency testing and will continue to be tested each refueling outage. The leakage has remained unchanged since 1994. A second bellows assembly was found with a minor leak in December 1995. No source of external leakage could be located on this assembly. This bellow is also tested each refueling outage and results have also remained unchanged since the 1995 test. A design change package was developed to encapsulate the affected bellows if a significant change in leakage were to occur. This same package would be utilized if a general and detailed visual examination revealed degradation that made the leak-tight integrity of the assembly suspect.

The staff has reviewed the licensee’s submittals and has determined that, with respect to the leakage of two-ply bellows, the licensee is meeting the requirements of ASME Code Subsections IWE and IWC and is following good engineering practices. Under these circumstances, the staff does not believe that a Type A leak test is likely to identify any bellows leakage that would not be otherwise identified through inspections required by the Code.

5. Degradation of Inaccessible Side of the Containment Steel Shell

Under its IWE program, the licensee will evaluate the acceptability of inaccessible areas of the containment steel shell if conditions exist in the accessible areas that could indicate the presence of, or result in, degradation to the inaccessible area. Section 50.55a(b)(2)(viii) of 10 CFR requires that this evaluation include the description of the type, estimated extent, and cause of the degradation as well as the examination results of each area and description of necessary corrective actions. In its submittal of October 9, 2002, the licensee did not address potential leakage due to age-related degradation in inaccessible areas and did not take any credit for the visual inspection (IWE) in the basic risk assessment. However, in its submittal of December 6, 2002, the licensee stated that to account for the inaccessible area, the following assumptions are made in its risk analysis to establish the likelihood of non-detected containment leakage:

a. The success data was tied to the initiation of IWE inspections;
b. The likelihood of developing a liner flaw is doubled every five years;
c. 90% of the containment is accessible for IWE inspection.

The staff has found that the licensee is evaluating the acceptability of the inaccessible region of the containment shell in accordance with 10 CFR 50.55a, that the assumptions made in calculating the increase in LERF based on potential impact of corrosion in the inaccessible region are consistent with the area of the uninspectable region, and that the area of the uninspectable region is small (about 10% of the shell). The staff has concluded, therefore, that the containment ISI inspections, together with the treatment of flaw development in the risk assessment, provide reasonable assurance that the risks associated with degradation of the containment will be low during the extended ILRT period.

Based on its review of the information provided in the licensee’s TS change request and responses to the staff’s questions, the staff finds that: (1) the structural integrity of the containment vessel is verified through the periodic ISI conducted as required by Subsections IWE and IWL of the ASME Code, Section XI and no augmented inspection is required; (2) the integrity of the penetrations, containment isolation valves, and bellows is periodically verified through Type B and Type C tests as required by 10 CFR Part 50, Appendix J, and Hope Creek TSs; (3) the licensee is aware of the potential problems associated with the degradation of two-ply stainless steel bellows and is taking prudent action, and that a Type A leak test is not likely to detect problems with these bellows that would otherwise go undetected; and (4) the licensee is employing an IWE program that requires them to estimate any potential degradation of inaccessible areas of the containment; furthermore, the licensee made conservative risk assumptions regarding such degradation. The staff concludes, therefore, that the licensee’s ISI program, as currently implemented, supports the one-time extension of the ILRT from 10 to 15 years.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State Official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or which changes an inspection or a surveillance requirement. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (68 FR 7819). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: G. Bedi
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Date: April 16, 2003

Hope Creek Generating Station

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