

November 1, 2004

Mr. A. Christopher Bakken, III
President & Chief Nuclear Officer
PSEG Nuclear LLC - X15
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION - ISSUANCE OF AMENDMENT
RE: CHANGE OF PRESSURE-TEMPERATURE LIMITS AND EXTENSION OF
VALIDITY TO 32 EFFECTIVE FULL-POWER YEARS (TAC NO. MC2534)

Dear Mr. Bakken:

The Commission has issued the enclosed Amendment No. 157 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 March 31, 2004, as supplemented by letter dated July 30, 2004. The amendment revises the reactor pressure vessel pressure-temperature limits and extends their validity to 32 effective full-power 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/

Daniel S. Collins, Sr. 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. 157 to
License No. NPF-57
2. Safety Evaluation

cc w/encls: See next page

Hope Creek Generating Station

cc:

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PSEG NUCLEAR LLC

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 157
License No. NPF-57

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment filed by the PSEG Nuclear LLC dated March 31, 2004, as supplemented by letter dated July 30, 2004, 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. 157, 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 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by Richard J. Laufer 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: November 1, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 157

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove

xviii
3/4 4-23
3/4 4-23a
3/4 4-23b
B 3/4 4-5
B 3/4 4-7
B 3/4 4-8
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Insert

xviii
3/4 4-23
3/4 4-23a
3/4 4-23b
B 3/4 4-5
B 3/4 4-7
B 3/4 4-8
B 3/4 4-9
B 3/4 4-10

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 157 TO FACILITY OPERATING LICENSE NO. NPF-57

PSEG NUCLEAR LLC

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated March 31, 2004, as supplemented by letter dated July 30, 2004, PSEG Nuclear LLC (PSEG, or the licensee) requested changes to the Hope Creek Generating Station (HCGS) Technical Specifications. The supplement dated July 30, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards determination as published in the Federal Register on June 8, 2004 (69 FR 32076).

The amendment would revise the reactor pressure vessel (RPV) pressure/temperature (P-T) limits and extend their validity to 32 effective full-power years (EFPYs) of operation.

The current P-T limits, as approved in License Amendment No. 131, were calculated for 32 EFPYs of operation; however, the NRC staff limited their applicability to the end of cycle 11 because the vessel fluence calculation did not comply with the guidance in Regulatory Guide (RG) 1.190. At that time, the NRC staff did not have the requisite assurance to approve the P-T limits' applicability through 32 EFPYs of operation due to issues the NRC staff had with PSEG's fluence methodology. Subsequently, the NRC staff found that there was adequate margin to allow their use for cycle 12 (the current cycle). The applicability of the P-T limits was extended to the end of cycle 12 by License Amendment No. 139. In their submittal for what became License Amendment No. 139, PSEG committed to revise the vessel fluence, using a methodology that adheres to the guidance in RG 1.190.

The currently proposed P-T curves were calculated using the General Electric vessel fluence methodology which has been reviewed and approved by the NRC staff. Additionally, the proposed changes are based on the methodology specified in the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Code Cases N-588 and N-640, the 1989 ASME Code, Section XI, Appendix G, and Appendix G to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50.

2.0 REGULATORY EVALUATION

The NRC staff evaluates the acceptability of a facility's proposed P-T limits based on the following regulations and guidance:

10 CFR Part 50.60(a) states:

Except as provided in paragraph (b) of this section, all light-water nuclear power reactors, other than reactor facilities for which the certifications under §50.82(a)(1) have been submitted, must meet the fracture toughness and material surveillance program requirements for the reactor coolant program pressure boundary set forth in appendices G and H to this part.

Appendix H to 10 CFR Part 50, "Reactor Vessel Material Surveillance Requirements," establishes requirements related to facility RPV material surveillance programs. RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," describes methods and assumptions acceptable to the NRC staff for determining the pressure vessel neutron fluence. RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," contains methodologies for determining the increase in transition temperature resulting from neutron radiation.

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," requires that facility P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Code. The most recent version of Appendix G to Section XI of the ASME Code, which has been endorsed in 10 CFR 50.55a, and therefore by reference in 10 CFR Part 50, Appendix G, is the 1998 Edition through the 2000 Addenda of the ASME Code. This edition of the Appendix incorporates the provisions of ASME Code Cases N-588 and N-640. Additionally, Appendix G to 10 CFR Part 50 imposes minimum head flange temperatures when system pressure is at or above 20% of the preservice hydrostatic test pressure.

Generic Letter (GL) 92-01, Revision 1, requested that licensees submit their RPV data for their plants to the NRC staff for review, and GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations.

NUREG-0800, "Standard Review Plan," Section 5.3.2, "Pressure Temperature Limits," provides guidance on using these regulations and documents in the NRC staff's review. Additionally, Section 5.3.2 provides guidance to the NRC staff in performing check calculations of the licensee's submittal.

3.0 TECHNICAL EVALUATION

The proposed amendment would replace the following figures with P-T limit curves valid to 32 EFPYs of operation:

- | | |
|----------------|---|
| Fig. 3.4.6.1-1 | Hydrostatic Pressure and Leak Tests Pressure/Temperature Limits |
| Fig. 3.4.6.1-2 | Non-Nuclear Heatup and Cooldown Pressure/Temperature Limits |
| Fig. 3.4.6.1-3 | Core Critical Heatup and Cooldown Pressure/Temperature Limits |

Additionally, the bases for these limits would be updated to reflect the methodologies and assumptions used in calculating the new P-T limits.

3.1 HCGS Information in Support of Proposed Changes

The licensee identified the limiting material for the HCGS RPV as the intermediate plate fabricated from heat number 5K3025/1. The licensee calculated the adjusted reference temperature (ART) values for only the 1/4T location because, as described in their July 30, 2004, letter, the cooldown curves were used for the heatup and cooldown assessments for the HCGS P-T limits. Additionally, in that letter, the licensee stated that the cooldown event is more severe and, therefore, bounds the heatup assessment. The critical parameters for the licensee's ART determination for the 1/4T location are shown in the table below.

Material	Location	Initial RT _{NDT} (EF)	Fluence (n/cm ²)	Chemistry Factor (EF)	ΔRT _{NDT} (EF)	Margin (EF)	ART (EF)
Intermediate Plate Heat No. 5K3025/1	1/4T	19	3.7E17	113	28	13.9	75

The licensee applied the methodologies of the 1989 Edition of Appendix G to the ASME Code, as modified by Code Cases N-588 and N-640. ASME Code Case N-588 provides an alternative procedure for assuming axially oriented reference defects in all axial welds and base metal and circumferentially-oriented reference defects in all circumferential welds. ASME Code Case N-640 allows the use of K_{IC} (the material toughness property measured in terms of stress intensity factor, K_I, which will lead to non-ductile crack propagation) instead of K_{IA} (which is the critical value of the stress intensity factor, K_I, for crack arrest as a function of temperature) in the development of P-T limit curves.

3.2 NRC Staff Evaluation of Proposed Changes

The licensee submitted information on the through-wall temperature gradients resulting from heatup and cooldown transients and their determination of the applied stress intensity (K_I) at the tip of the postulated 1/4T flaw due to thermal loading in an enclosure to its submittal. This information, along with the knowledge of the applied stress intensity at the tip of the postulated 1/4T and 3/4T flaws due to pressure loads, and the material property information cited in Section 3.1 of this document, allowed the NRC staff to independently evaluate the acceptability of the proposed HCGS P-T limit curves.

The calculated fluence was based on operation for 32 EFPYs of operation with 12 EFPYs at 3293 megawatt thermal (MWt) (the original licensed power), 3 EFPYs at 3339 MWt (the current licensed power), and the remaining 17 years at 3952 MWt (the planned extended power uprate power). The vessel fluence was calculated using the NRC-approved methodology contained in NEDC-32983P-A, "Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations." The licensee stated that the fluence component due to the original 12 EFPYs of operation ignored the presence of the jet pumps.

This is a conservative assumption, because the presence of the jet pumps provides some shielding from the neutron radiation, thus reducing embrittlement of the RPV in those locations. For 32 EFPYs, the peak inside the surface value is 1.1×10^{18} n/cm², and the limiting beltline material peak 1/4T fluence was 3.7×10^{17} n/cm². The values were derived in accordance with the methodology and guidance contained in RG 1.190, utilizing conservative assumptions and, therefore, meet the requirements of 10 CFR Part 50 Appendix H. Thus, the NRC staff finds the use of the calculated neutron fluence values in calculation of the ART to be acceptable.

The licensee developed beltline P-T curves for core critical, core not critical, and P-T conditions. For the normal operating conditions with the core not critical and P-T condition curves, individual P-T curves were proposed for the lower head and upper vessel, in addition to the composite curves proposed for the beltline regions of the RPV. To confirm the validity of the HCGS's proposed curve, the NRC staff performed an independent assessment of the licensee's submittal. The staff applied the methodologies of the 1989 Edition of Appendix G to the ASME Code and 10 CFR Part 50, Appendix G, as modified by the methodology of ASME Code Cases N-588 and N-640, as the bases for its independent assessment.

The NRC staff's assessment included an independent calculation of the ART values for the 1/4T location of the HCGS RPV beltline regions, based on the neutron fluence specified in the submittal for HCGS to 32 EFPYs. For the evaluation of the limiting beltline materials, the NRC staff confirmed that the material ART values were based on the methodology of RG 1.99, Revision 2. For the evaluation of the limiting material in the limiting nozzle and lower head evaluations, the NRC staff applied the plant-specific design basis data provided by the licensee. Based on the calculations, the staff verified that the licensee's limiting beltline material for the HCGS RPV is the intermediate plate, heat number 5K3025/1. The NRC staff's calculated ART values for the limiting material and the other beltline materials were within 1 EF of the licensee's calculated ART values.

Using the information submitted by the licensee, the NRC staff evaluated the licensee's P-T limit curves for acceptability by performing a finite set of check calculations in accordance with the methodology referenced in the ASME Code (as indicated by NUREG-0800 Section 5.3.2). Further, the staff compared information submitted by the licensee (in particular, the information related to the evaluation of thermal loading conditions) to information submitted previously for other, similar RPVs and determined that the information submitted by the licensee was consistent. The NRC staff verified that the licensee's proposed P-T limits satisfy the requirements in Paragraph IV.A.2 of Appendix G to 10 CFR Part 50. Specifically, the NRC staff concluded that the P-T limit curves submitted by the licensee were as conservative as those which would be generated by the staff's application of the methodology specified in Appendix G to Section XI of the 1998 Edition through 2000 Addenda of the ASME Code.

In addition, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the head flange, based on the reference temperature for the flange material. Section IV.A.2 of Appendix G requires that when the system pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 160 EF for core critical operation, 120 EF for normal, non-critical core operation, and by 90 EF for hydrostatic pressure tests and leak tests. Given that the limiting flange RT_{NDT} is 19 EF for HCGS, Appendix G requires a minimum closure flange region temperature of 179 EF, 139 EF, and 109 EF, respectively, for the three conditions. The preservice system hydrostatic test

pressure for HCGS was 1563 psig, which makes the requirements of Appendix G applicable at 312.6 psig. The P-T limits proposed by the licensee set a minimum upper-vessel temperature requirement of 188 EF for core critical operation, 148 EF for normal, non-critical core operation, and 118 EF for hydrostatic pressure and leak tests when the reactor coolant system pressure is 312.6 psig or greater. Therefore, the NRC staff finds that the curves satisfy the minimum head flange temperature requirements.

Based on the above discussion, the NRC staff has determined that the proposed P-T limits meet the requirements of 10 CFR Part 50, Appendices G and H, and thus the requirements of 10 CFR 50.60(a). The NRC staff, therefore, finds the proposed P-T limits to be acceptable.

Additionally, PSEG proposed changes to the bases to reflect the revised P-T limits. The NRC staff reviewed the proposed bases and finds that they adequately represent the new P-T limits and the methods used to create them. The NRC staff does not object to the proposed bases changes.

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. 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 (69 FR 32076). 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 Contributor: L. Lois
M. Khanna

Date: November 1, 2004