Dated at Rockville, Maryland, this 26th day of April, 2007.

For the Nuclear Regulatory Commission.

Margaret A. Janney,

NRC Clearance Officer, Office of Information Services.

[FR Doc. E7–8439 Filed 5–2–07; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

[Docket No. 50–354]

PSEG Nuclear LLC; Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The U.S. Nuclear Regulatory Commission (NRC or the Commission) is considering issuance of an amendment to Facility Operating License No. NPF–57 issued to PSEG Nuclear (the licensee) for operation of the Hope Creek Generating Station (Hope Creek) located in Salem County, New Jersey.

The proposed amendment would increase the authorized maximum power level from 3339 megawatts thermal (MWt) to 3840 MWt, an increase of approximately 15 percent.

Before issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in Title 10 of the Code of Federal Regulations (10 CFR), Section 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The CPPU [Constant Pressure Power Uprate] analyses, which were performed at or above CPPU power levels, included a review and evaluation of the structures, systems, and components (SSCs) that could be affected by the proposed change. The proposed amendment does not change the design function or operation of the affected SSCs.

Plant specific analyses were performed in the following areas: Reactor Core and Reactor Internals (e.g., steam dryer), Reactor Coolant System and associated systems, Containment, Emergency Core Cooling Systems, Control and Instrumentation Systems, Electrical Systems, Balance of Plant Systems, and Radwaste Systems. The results of the analyses, which included evaluating the increase in the likelihood of an SSC malfunction, concluded that the SSCs are capable of performing their design functions at CPPU conditions.

Comprehensive evaluations were performed on the steam dryer and other reactor internals for both operational and structural performance. Predicted steam dryer peak and alternating stress ratios remain within allowable levels. The existing margins to steam dryer alternating stress limits and the steam dryer monitoring program during power ascension provide assurance that steam dryer integrity will be maintained.

Vibration evaluations at CPPU conditions were performed on the Reactor Internal components and Reactor Coolant and associated system piping. These included the Main Steam, Feedwater and Reactor Recirculation systems piping and supports. The results of the vibration analyses demonstrate that operation at CPPU conditions will not result in any detrimental effects. System values will remain within allowable American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) limits. In addition, the ASME Code and regulatory guidelines require vibration test data be taken on high-energy piping during initial CPPU startup. The vibration start-up test program will validate the vibration analyses that were performed, demonstrating adequate performance of the SSCs.

Engineered Safety Features (ESF) were evaluated at CPPU conditions using NRC-approved methods. The Emergency Core Cooling Systems (ECCS) were evaluated to ensure they are capable of performing their design function during loss-of-coolant-accidents (LOCA). Adequate net positive suction head is maintained without reliance on post-accident containment pressure. CPPU does not result in an increase or decrease in the available water sources, and does not result in any change in the maximum nominal reactor operating pressure. The CPPU evaluations demonstrate that the ECCS performance satisfy the requirements of 10 CFR 50.46 and 10 CFR [Part] 50 Appendix K.

Balance-of-plant (BOP) systems and equipment were also evaluated for CPPU operation. The resulting evaluations demonstrate adequate performance with limited modifications that were or will be made to BOP components.

These analyses, which included evaluating the increased likelihood of an SSC malfunction, confirm acceptable performance of plant SSCs under CPPU conditions. On this basis, PSEG concludes that there is no significant change in the ability of the SSCs to preclude or mitigate the consequences of accidents.

The probability (frequency of occurrence) of postulated Design Basis Accidents (DBA), and other Updated Final Safety Analysis Report (UFSAR) evaluated accidents, occurring is not affected by the increased power level, and Hope Creek continues to comply with the regulatory and design basis criteria established for plant equipment. The changes in consequences of hypothetical accidents, which are assumed to occur at 102% of the CPPU RTP [Rated Thermal Power], compared to those previously evaluated, are in all cases insignificant. The CPPU accident evaluations do not exceed any of the NRC-approved acceptance limits. The spectrum of hypothetical accidents and transients has been investigated, and is shown to meet the plant's currently licensed regulatory criteria. Consequently, there is no significant increase in the probability or consequences of an accident previously evaluated.

The impact of CPPU on the radiological consequences of postulated DBAs, operational transients and other UFSAR accidents was evaluated. The magnitude of the potential consequences is dependent upon the quantity of fission products released to the environment, the atmospheric dispersion factors and the dose exposure pathways. The atmospheric dispersion factors and the dose exposure pathways are not changed by CPPU operation. The only factor which could influence the magnitude of the consequences is the quantity of activity released to the environment. For CPPU, the Control Rod Drop Accident (CRDA), Loss-of-Coolant Accident (LOCA), Fuel Handling Accident (FHA), Main Steamline Break Accident (MSLBA) and instrument line break accident (ILBA) were reanalyzed.

The DBA that has historically been limiting from a radiological criterion is the LOCA, for which USNRC Regulatory Guide 1.183, Appendix A guidance was applied. Adherence to the guidance in RG 1.183, and the use of the specific values/limits contained in the Technical Specifications with as-tested post-accident performance of the safety grade engineered safety functions (ESF), provide the assurance for sufficient safety margin, including a margin to account for analysis uncertainties. The CPPU LOCA evaluation results include the 2% power uncertainty factor from Regulatory Guide 1.49.

The results of the CPPU radiological analyses remain below the allowable limits of 10 CFR 50.67 and Table 6 in Regulatory Guide 1.183; the CPPU impact is minimal and all radiological limits are met at CPPU conditions. Therefore, the proposed change does not involve a significant increase in the radiological consequences of an accident previously evaluated.

While the proposed CPPU amendment is not being submitted as a risk-informed licensing action, it was evaluated from a risk perspective using the NRC guidelines established in Regulatory Guide 1.174. Level 1 and Level 2 Probabilistic Risk Assessments (PRAs) were performed for the CPPU. When compared to the risk-acceptance guidelines presented in Regulatory Guide 1.174, the calculated changes in core damage frequency (CDF) and large early release frequency (LERF) are insignificant. Based on these results, PSEG concludes that the proposed amendment would not involve a significant increase in the probability of an accident previously evaluated.

The impact of CPPU operation on plant operator actions and procedures was also evaluated. The operator action response times credited in the safety analyses in the UFSAR are not changed by CPPU. In addition, there is no change in Emergency Operating Procedure (EOP) strategy for CPPU operation.

Based on the above, PSEG concludes that the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

As discussed above, the evaluation of the proposed amendment included review of the SSCs that could be affected by the proposed change. The proposed amendment does not change the design function or operation of the affected SSCs. The proposed amendment does not introduce any new or different plant safety-related equipment, and only involves instrument set-point changes for CPPU conditions, and minimal modifications to plant BOP power generation equipment. The proposed amendment does not significantly impact the manner in which the plant is operated, and does not have any significant impact on the capability the SCCs involved to perform their design function.

No new operating mode, safety-related equipment lineup, accident scenario or equipment failure mode was identified. The CPPU evaluations also addressed the impact to postulated accidents, accident radiological consequences and operator response. No significant impacts were identified. The full spectrum of accident considerations has been evaluated, and no new, different, or limiting kind of accident has been identified. CPPU uses developed technology, and applies it within the capabilities of existing plant equipment in accordance with presently existing regulatory criteria to include NRC approved codes, standards and methods. The CPPU analyses results confirm acceptable performance of plant SSCs under CPPU conditions. Consequently, there are no new credible failure mechanisms, malfunctions, or accident initiators that were not previously evaluated in the plant design and licensing bases.

Based on the preceding, PSEG concludes that the proposed change would not introduce any new or different kind of accident, or failure mode, not previously analyzed.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

Safety margins are applied to plant parameters to account for various uncertainties and to avoid exceeding regulatory and licensing limits. The proposed change does not involve a significant reduction in any margin of safety. First, due to continuing improvements in the analytical techniques (computer codes and data) based on several decades of BWR safety technology, plant performance feedback, and improved fuel and core designs, a significant increase has resulted in the design and operating margins between calculated safety analysis results and the licensing limits. These available safety analyses differences, combined with the excess as-designed equipment, system and component capabilities, provide BWR plants the capability to achieve an increase in their thermal power ratings within the existing design and licensing basis. The proposed CPPU will reduce some of the existing design and operational margins. However, safety margins are considered to not be significantly reduced if: (1) Applicable regulatory requirements, codes and standards or their alternatives approved for use by the NRC, are met, and (2) if safety analysis acceptance criteria in the licensing basis are met, or if proposed revisions to the licensing basis provide sufficient margin to account for analysis and data uncertainty. This is the case for the proposed CPPU amendment.

Safety margin is related to the ability of the fission product barriers to limit the level of radiation dose to the public. The impact of the proposed CPPU amendment on the: (1) Fuel cladding barrier, (2) reactor coolant pressure boundary (RCPB) barrier, and (3) containment fission product barrier is discussed below.

To assure that fuel cladding damage limits are not exceeded, the impact of the proposed amendment on fuel system design, nuclear system design, thermal and hydraulic design, accident and transient analyses, and fuel design limits was evaluated. No new fuel design, or change in the specified fuel design limits, is required for CPPU. The current fuel and core design limits will continue to be met; both the Safety Limit Minimum Critical Power Ratio (SLMCPR) and other applicable Specified Acceptable Fuel Design Limits (SAFDLs) are still met. Analyses for each fuel reload will continue to meet the criteria accepted by the NRC. Continued compliance with the SLMCPR and other SAFDLs will be confirmed on a cycle specific basis consistent with the criteria accepted by the NRC as specified in NEDO–24011, ''General Electric Standard Application for Reactor Fuel, GESTAR II.'' The ECCS evaluation for CPPU demonstrates the continued conformance to the acceptance criteria of 10 CFR 50.46, for peak cladding temperature (PCT) and the other 10 CFR 50.46 parameters. The increased PCT consequences for CPPU are insignificant and remain substantially below the regulatory criteria. Therefore, the ECCS safety margin and fuel cladding margin (PCT) are not significantly impacted by CPPU.

Challenges to the Reactor Coolant Pressure Boundary were evaluated at CPPU conditions (pressure, temperature, flow, and radiation) and were found to meet their acceptance criteria for allowable stresses and overpressure margin. These evaluations included (1) overpressure protection, (2) structural integrity of the RCPB piping, components, and supports, and (3) structural integrity of the reactor vessel.

For the most limiting pressurization event, the peak calculated pressure remains below the ASME Code allowable peak pressure. The structural integrity of the RCPB piping, components, and supports was evaluated using NRC-approved methodology. The changes in flow, pressure and temperature associated with CPPU do not result in load limits being exceeded. Sufficient margin remains between the calculated stresses and ASME Code limits. In addition, the ASME Code and regulatory guidelines require vibration test data be taken on high-energy piping during initial CPPU startup. The vibration start-up test program will validate the vibration analyses that were performed, demonstrating adequate performance.

The structural integrity of the reactor vessel was evaluated. The neutron fluence was re-analyzed in accordance with the requirements of 10 CFR [Part] 50 Appendix G. The existing Pressure-Temperature (P–T) limit curves have been revised for CPPU conditions (a previous amendment to the Hope Creek license changed the P–T curves and included CPPU conditions). The reactor vessel materials surveillance program is unchanged by CPPU. The maximum normal operating reactor dome pressure for CPPU is unchanged and the vessel remains in compliance with regulatory requirements. Consequently, CPPU operation does not have an adverse effect on the reactor vessel fracture toughness. The structural evaluation of the vessel demonstrates that ASME Code requirements are met for normal, upset, emergency and accident conditions.

Based on the preceding, PSEG concludes that the RCPB structural integrity will be maintained and the licensing basis requirements will continue to be met following implementation of the proposed CPPU.

The impact of the proposed CPPU on the Containment was evaluated. The effect of CPPU on the peak values for containment pressure and temperature confirms the suitability of the plant for operation at CPPU RTP. Also, the effects of CPPU on the conditions that affect the containment dynamic loads were determined to be satisfactory for CPPU operation. Where plant conditions with CPPU are within the range of conditions used to define the current dynamic loads, current safety criteria are met and no further structural analysis was required. The change in short-term containment response is negligible. Because there will be more residual heat with CPPU, the containment long-term response slightly increases. However, containment pressures and temperatures remain below their design limits following any design basis accident, and thus, the containment and its cooling systems are satisfactory for CPPU operation. The small increase in the calculated post LOCA suppression pool temperature above the currently assumed peak temperature was evaluated and determined to be acceptable. Based on the use of conservative assumptions in these evaluations, PSEG concludes that containment structural integrity will be maintained under the proposed CPPU conditions, and the containment parameters will remain below design limits. Therefore there is no significant reduction in safety margin.

In summary, challenges to the fuel, RCPB, and containment were evaluated for CPPU conditions. The structural integrity of the fission product barriers will be maintained under CPPU conditions. As such, the proposed amendment would not degrade confidence in the ability of the barriers to limit the level of radiation dose to the public. Fuel integrity is maintained by meeting existing design and regulatory limits. The calculated loads on all affected structures, systems and components, including the reactor coolant pressure boundary, will remain within their design allowables for all design basis event categories. The containment parameters remain below design limits. No NRC acceptance criterion will be exceeded. Because the Hope Creek configuration and responses to transients and hypothetical accidents do not result in exceeding the presently approved NRC acceptance limits, CPPU does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of 60 days after the date of publication of this notice. The Commission may issue the license amendment before expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. In addition, the Commission may issue the amendment prior to the expiration of the 30-day comment period should circumstances change during the 30-day comment period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility. Should the Commission take action prior to the expiration of either the comment period or the notice period, it will publish in the **Federal Register** a notice of issuance. Should the Commission make a final No Significant Hazards Consideration Determination, any hearing will take place after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rulemaking, Directives and Editing Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D59, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, Public File Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland.

The filing of requests for hearing and petitions for leave to intervene is discussed below.

Within 60 days after the date of publication of this notice, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.309, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following general requirements: (1) The name, address and telephone number of the requestor or petitioner; (2) the nature of the requestor's/petitioner's right under the Act to be made a party to the proceeding; (3) the nature and extent of the requestor's/petitioner's property, financial, or other interest in the proceeding; and (4) the possible effect of any decision or order which may be entered in the proceeding on the requestors/petitioner's interest. The petition must also identify the specific contentions which the petitioner/requestor seeks to have litigated at the proceeding.

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner/requestor shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner/requestor must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. The petition must include sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner/requestor who fails to satisfy these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held. If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment. If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

Nontimely requests and/or petitions and contentions will not be entertained absent a determination by the Commission or the presiding officer of the Atomic Safety and Licensing Board that the petition, request and/or the contentions should be granted based on a balancing of the factors specified in 10 CFR 2.309(c)(1)(i)–(viii).

A request for a hearing or a petition for leave to intervene must be filed by: (1) First class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff; (2) courier, express mail, and expedited delivery services: Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff; (3) e-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, *HEARINGDOCKET@NRC.GOV*; or (4) facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC, *Attention:* Rulemakings and Adjudications Staff at (301) 415–1101, verification number is (301) 415–1966. A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov*. A copy of the request for hearing and petition for leave to intervene should also be sent to Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038, attorney for the licensee.

For further details with respect to this action, see the application for amendment dated September 18, 2006, as supplemented by letters dated October 10, 2006, October 20, 2006, February 14, February 16, February 28, March 13, and April 18, 2007 which is available for public inspection at the Commission's PDR, located at One White Flint North, File Public Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the ADAMS Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html*. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov*.

Dated at Rockville, Maryland, this 27th day of April 2007.

For the Nuclear Regulatory Commission.

James J. Shea,

Project Manager, Plant Licensing Branch I–2, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation.

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NUCLEAR REGULATORY COMMISSION

Sunshine Federal Register Notice

DATE: Weeks of April 30, May 7, 14, 21, 28, June 4, 2007.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of April 30, 2007

There are no meetings scheduled for the Week of April 30, 2007.

Week of May 7, 2007—Tentative

Monday, May 7, 2007

1:30 p.m. Briefing on Office of Federal and State Materials and Environmental Management Programs (FSME) Programs, Performance, and Plans (Public Meeting) (Contact: George Deegan, 301–415–7834).

This meeting will be Web cast live at the Web address—*http://www.nrc.gov.*

Week of May 14, 2007—Tentative

There are no meetings scheduled for the Week of May 14, 2007.