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U. S. Nuclear Regulatory Commission
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Hope Creek Generating Station
Facility Operating License No. NPF-57 NRC
Docket No. 50-354

Subject: **Supplement to License Amendment Request for Extended Power Uprate, Revised No Significant Hazards Consideration**

Reference:
1) PSEG letter LR-N06-0286, Request for License Amendment: Extended Power Uprate, September 18, 2006
2) PSEG letter LR-N07-0034, Supplement to License Amendment Request for Extended Power Uprate, February 27, 2007
3) PSEG letter LR-N07-0055, Supplement to License Amendment Request for Extended Power Uprate, March 13, 2007
4) PSEG letter LR-N06-0413, Supplement to License Amendment: Request for Extended Power Uprate, October 10, 2006
5) PSEG letter LR-N06-0418, Supplement to License Amendment: Request for Extended Power Uprate, October 20, 2006
6) PSEG letters LR-N07-0035, LR-N07-0056, LR-N07-0060, LR-N07-0069, and LR-N07-0070, Response to Request for Additional Information, Request for License Amendment - Extended Power Uprate, dated March 13, March 22, March 30, March 30, and April 13, 2007

In Reference 1, PSEG Nuclear LLC (PSEG) requested an amendment to Facility Operating License NPF-57 and the Technical Specifications (TS) for the Hope Creek Generating Station (HCGS) to increase the maximum authorized power level to 3840 megawatts thermal (MWt).

References 2 and 3 provided the results of evaluations performed to facilitate staff review, demonstrating the conservatism in the predicted loads on the HCGS steam dryer for extended power uprate (EPU) operation. References 4 and 5

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provided additional documentation of evaluations performed in support of the requested amendment. Reference 6 provided PSEG responses to NRC Requests for Additional Information on the EPU amendment request.

Following discussions with the NRC staff, PSEG has revised the proposed No Significant Hazards Consideration (NSHC) determination that was provided in Reference 1. The revised NHSC (Attachment 1 to this submittal) is more detailed reflecting the additional information provided in References 2 through 6 and is generally more comprehensive consistent with the regulatory guidance provided in NRC Regulatory Issue Summary (RIS) 2001-22, "Attributes of a Proposed No Significant Hazards Consideration," dated November 20, 2001.

Should you have any questions regarding this submittal, please contact Mr. Paul Duke at 856-339-1466.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 4/18/07
(Date)

Sincerely,

George Barnes

George Barnes
Site Vice President
Hope Creek Generating Station

Attachments (1)

S. Collins, Regional Administrator - NRC Region I
J. Shea, Project Manager - USNRC
NRC Senior Resident Inspector - Hope Creek
K. Tosch, Manager IV, NJBNE

No Significant Hazards Consideration

This proposed extended power uprate (EPU) amendment increases the Hope Creek Generating Station (HCGS) licensed thermal power level to 3840 megawatts thermal (MWt), approximately 15% above the current rated thermal power (RTP) of 3339 MWt and 16.6% above the original RTP of 3293 MWt. The requested increase in reactor thermal power level will allow operational changes to generate higher steam flow to the turbine generator, in turn permitting an increase in the electrical output of the plant.

A higher steam flow is achieved by increasing the reactor power along specified control rod and core flow lines. A limited number of operating parameters are changed, some setpoints are adjusted and instruments are recalibrated. Plant procedures are revised, and tests similar to some of the original startup tests are performed.

The technical bases for this request follow the guidelines contained in NRC-approved GE Nuclear Energy (GENE) Licensing Topical Reports (LTRs) for a Constant Pressure Power Uprate (CPPU). CPPU operation does not involve an increase in the maximum normal operating reactor dome pressure, or the maximum licensed core flow. This is possible because the plant, following modifications to non-power generating equipment, has sufficient capabilities to control turbine generator inlet pressure.

Detailed evaluations of the reactor, engineered safety features, emergency power, support systems, environmental issues, and design basis accidents have been performed. These evaluations demonstrate that Hope Creek can safely operate at 3840 MWt.

PSEG Nuclear LLC (PSEG) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment" as discussed below:

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

Response: No.

The CPPU 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 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, 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 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.

Based on the above, PSEG concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.