**PSEG Nuclear LLC**
P.O. Box 236, Hancocks Bridge, New Jersey 08038-0236

PSEG
*Nuclear LLC*

10 CFR 50.90

LR-N07-0070
LCR H05-01, Rev. 1
April 13, 2007

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Hope Creek Generating Station
Facility Operating License No. NPF-57
NRC Docket No. 50-354

Subject: Response to Request for Additional Information
Request for License Amendment - Extended Power Uprate

Reference:
1) Letter from George P. Barnes (PSEG Nuclear LLC) to USNRC, September 18, 2006
2) Letter from USNRC to William Levis, PSEG Nuclear LLC, February 16, 2007
3) Letter from George P. Barnes (PSEG Nuclear LLC) to USNRC, March 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).

In Reference 2, the NRC requested additional information concerning PSEG's request. In Reference 3, PSEG provided responses to each question with the exception of questions 3.18 and 3.29. Attachment 1 to this letter provides the responses to questions 3.18 and 3.29.

PSEG has determined that the information contained in this letter and attachment does not alter the conclusions reached in the 10CFR50.92 no significant hazards analysis previously submitted.

A001

There are no regulatory commitments contained within this letter.

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 4/13/07
(date)

Sincerely,

George P. Barnes

George P. Barnes
Site Vice President
Hope Creek Generating Station

Attachment (1)

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

Hope Creek Generating Station
Facility Operating License NPF-57
Docket No. 50-354

Extended Power Uprate

Response to Request for Additional Information

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).

In Reference 2, the NRC requested additional information concerning PSEG's request. In Reference 3, PSEG provided responses to each question with the exception of questions 3.18 and 3.29, which are restated below followed by PSEG's responses.

**3) BWR Systems Branch (SBWB)**

Reactor Core Isolation Cooling (RCIC) System

3.18 RCIC turbine steam exhaust trip is currently set at 25 psig. EPU analysis used a revised analytical model which predicted a maximum pressure of 38 psig. Please address the affect of the predicted maximum pressure will have on the system components. Has this revised analytical model been used in any other EPU analysis?

Response
The station blackout (SBO) analysis described in the PUSAR used the SHEX code to calculate the wetwell backpressure of 38 psig. Conservative analysis assumptions included an initial power level of 3952 MWt, no credit for heat sinks in the containment, and higher than predicted post-scram containment heat loads. This analysis was recently revised to provide a more realistic, but still bounding value. The revised wetwell backpressure value is 26.2 psig.

For EPU implementation, PSEG plans to increase the RCIC turbine exhaust pressure trip setpoint to at least 30 psig to provide adequate margin to maintain the RCIC system available for the revised SBO analysis. Affected piping calculations will be revised; and piping support modifications will be performed if required. The BWR Owners Group evaluated RCIC system operation with exhaust pressure as high as 50 psig and concluded that raising the exhaust pressure setpoint can be accomplished with existing RCIC system hardware (NEDE-22017, "BWR Owners Group Evaluation of RCIC Turbine Exhaust Pressure Trip for LOCA Application", November 1981).

PUSAR Table 1-2 indicates where the SHEX code is used in other EPU analyses.

Transients and Accidents

3.29 In section 9.3.2 second paragraph, you stated "Decay heat was conservatively evaluated assuming end-of-cycle and GE-14 fuel." Why is the SVEA 96 fuel not more limiting? In station blackout (SBO) transient, please provide more details about "coping capabilities" and how they are justified in constant pressure power uprate operation. Please also provide the sequence of events and operation of the safety features (i.e., RCIC) for the entire transient.

Response
The core decay heat for HCGS extended power uprate was generated based on a GE14 equilibrium core, with safety margins added to the cycle parameters that are relevant for decay heat assessments. The safety factors are intended as additional conservatism to bound future cycles with slight variations in fuel design and operational parameters.

In general, decay heat is not a function of fuel product line or fuel manufacturer. A comparison study had been performed to compare the ANS 5.1-1979 Standard decay heat result of a SVEA 96+ bundle with that of a GE14 bundle of comparable enrichment. The resulting differences were well within the calculation uncertainties, hence providing justification that the HCGS EPU decay heat bases remain valid for a mixed core of SVEA 96+/GE14. Furthermore, since the comparison was made independent of power level, it is applicable for operation at CPPU as well.

Methods
The SBO event evaluation performed for the HCGS EPU evaluated the effect of the EPU on the current (or pre-EPU) design and licensing basis for the Station Blackout event at HCGS and the current bases were validated for HCGS for EPU conditions using the approved SHEX code. The application of the methodology in the SHEX code to the containment response is approved by NRC in the letter to G. L. Sozzi (GE) from A. Thadani (NRC), "Use of the SHEX Computer Program and ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," July 13, 1993. The current bases, determined to be acceptable for HCGS to cope with the SBO event, has not changed for the EPU.

For the Station Blackout event the Hope Creek Generating Station (HCGS) has committed to meet the guidance of Regulatory Guide (RG) 1.155 dated August 1988 and NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," Rev. 1, dated August 1991, as endorsed by the regulatory guide.

The characteristics that affect the ability to cope with a station blackout event are identified as:

1. Condensate inventory for decay heat removal
2. Class 1E battery capacity
3. Compressed air capacity
4. Effects of loss of ventilation
5. Containment isolation

Per NUMARC 87-00 Section 7 the "AC Independent" approach is used in the HCGS SBO assessment. In the AC Independent approach the plant relies on available process steam, DC power and compressed air to operate equipment necessary to achieve and maintain hot shutdown. By satisfying the criteria used in assessing the above characteristics, the plant is able to show satisfactory response to an SBO event for EPU conditions.

Coping Duration
The four-hour coping duration criteria for AC-Independent plants apply to HCGS.

Condensate inventory for decay heat removal
Analyses have shown that the HCGS condensate inventory is adequate to meet the SBO coping requirement for EPU conditions.

DC power
Evaluation of the HCGS Class 1E Battery Capacity has shown that HCGS has adequate battery capacity to support decay heat removal during a station blackout for the required coping duration for EPU conditions.

Compressed Air
An evaluation has shown that the HCGS air operated valves required for decay heat removal have sufficient compressed air for the required automatic and manual operation during the station blackout event for EPU conditions.

Effects of Loss of Ventilation
An assessment was done of the average steady state temperature in dominant areas containing equipment necessary to achieve and maintain safe shutdown during a station blackout for EPU conditions. The assessment determined that the steady state temperature for these areas met the equipment operability requirements.

These areas for HCGS included:

- HPCI and RCIC rooms;
- Battery, Inverter and Electrical Access rooms;

- Containment; and
- Control and Control Equipment rooms.

Containment Isolation
An assessment of the HCGS containment isolation system determined that containment isolation is ensured during a station blackout event for EPU conditions.

Sequence of Events
The SBO containment analyses were performed using the approved SHEX code. The sequence of event is provided in Table 3.29-1.

**Table 3.29-1**

| HCGS SBO Sequence of Events | | |
|---|---|---|
| **Item** | **Time (Seconds)** | **Event** |
| 1 | 0 | Station blackout initiated. Reactor scrams, feedwater pumps trip and MSIVs begin closure. |
| 2 | 3.5 | MSIVs closure complete. |
| 3 | 5 | Feedwater flow reduced to zero. |
| 4 | 0 - 2725 | Automatic SRV operation on high reactor pressure. |
| 5 | 298 | Vessel level falls to L2. RCIC initiates with suction from CST. |
| 6 | 10928 | Vessel level increases to L8 resulting in RCIC isolation. |
| 7 | 2725 - 9090 | Initiate cooldown at 100°F/hr when pool temperature reaches 120°F. Operator control of SRVs; RCIC on level control. |
| 8 | ~9090 | Terminate depressurization and maintain reactor pressure at 180 psia. |
| 9 | 14,400 | Offsite power restored or diesel generator(s) started. Suppression pool cooling started via RHR system. Proceed to cold shutdown. |

SBO Response Description
In the following discussion, noted times are "after event initiation." The SHEX code models RPV water level with an equivalent vessel volume. L8 is about 586.5 inches above vessel zero, and the corresponding SHEX equivalent RPV volume is approximately 15,898 $ft^3$.

The RPV isolates shortly after the SBO event initiation. Pressurization of the RPV is relieved by SRV cycling which maintains the RPV pressure near the initial value. The SRV steam discharge from the vessel is primarily responsible for the

initial RPV water level drop. There is also a secondary effect due to the RPV leakage of 66 gpm as modeled in the SHEX analysis.

RCIC automatically starts to restore the water level after reaching L2 (approximately 469.5 inches and corresponding to the SHEX equivalent volume of 12,650 $ft^3$). This action continues to raise the water level until just before 1 hour. At this time, controlled vessel depressurization commences (as pool temperature of 120°F is reached) and depressurization continues until about 2.5 hours into the event. The integrated SRV steam flow due to vessel depressurization offsets the RCIC injection and results in a near constant vessel volume between 1 and 2.5 hours.

Depressurization is halted at approximately 2.5 hours into the event as the RPV pressure reaches 180 psia. RPV pressure is subsequently maintained at 180 psia and SRV steam flow from the vessel is reduced. After 2.5 hours, the constant RCIC inflow is greater than the combined inventory losses from SRV operation, used to maintain RPV pressure near 180 psia, and the RPV leakage flow and RPV water level begins to again increase.

At approximately 3 hours, the RPV water level reaches L8 and RCIC injection terminates. The RPV water level subsequently begins to drop again but remains above L2 through the end of the coping period (4 hours), and RCIC is not restarted. At the end of the coping period, the SHEX equivalent RPV volume is about 12,830 $ft^3$ and is above L2.

**References**

1. PSEG letter LR-N06-0286, Request for License Amendment: Extended Power Uprate, September 18, 2006

2. NRC letter, Hope Creek Generating Station - Request for Additional Information Regarding Request for Extended Power Uprate (TAC NO. MD3002), February 16, 2007

3. PSEG letter, LR-N07-0035, Response to Request for Additional Information, March 13, 2007