UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 19, 2011

Mr. Thomas Joyce
President and Chief Nuclear Officer
PSEG Nuclear LLC
P.O. Box 236, N09
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION - CORRECTION TO AMENDMENT NO. 187 REGARDING RELOCATION OF SPECIFIC SURVEILLANCE FREQUENCIES TO A LICENSEE-CONTROLLED PROGRAM BASED ON TECHNICAL SPECIFICATION TASK FORCE (TSTF) CHANGE TSTF-425 (TAC NO. ME5981)

Dear Mr. Joyce:

On February 25, 2011, the Nuclear Regulatory Commission (NRC) issued Amendment No. 187 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station (HCGS) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML103410243). This amendment consisted of changes to the Technical Specifications (TSs) and Facility Operating License in response to your application dated March 19, 2010, as supplemented by letters dated July 28, 2010, and January 10, 2011 (ADAMS Accession Nos. ML100900224, ML102230417, and ML110200059, respectively). The amendment modified the TSs by relocating specific surveillance frequencies to a licensee-controlled program based on TS Task Force (TSTF) Change TSTF-425.

By letter dated April 4, 2011 (ADAMS Accession No. ML110950150), PSEG Nuclear LLC (PSEG) informed the NRC staff that some of the TS camera-ready pages prepared by PSEG to support issuance of the amendment contained format and typographical errors. Corrected pages were provided in Attachment 1 to PSEG's letter. The NRC staff has reviewed the corrected TS pages and determined they are consistent with the proposed TS markups included in the application dated March 19, 2010. The specific corrections are as follows:

1) TS page 3/4 2-1 - the word "thereafter" was added to Surveillance Requirement 4.2.1.a.

2) TS pages 3/4 3-7, 3/4 3-29, 3/4 3-30, and 3/4 3-39 - spaces between some of the rows in each table were deleted so the information in the table aligns properly with the respective Functional Unit or Trip Function.

3) TS page 3/4 6-10 - the degree symbol "°" was added after the value "270" in Surveillance Requirement 4.6.1.7.d.

Enclosed are the corrected TS pages for HCGS Amendment No. 187. Please replace the affected pages in the HCGS TSs with the enclosed pages. If you have any questions, please contact me at 301-415-1420.

Sincerely,

Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosure:
As stated

cc w/encl: Distribution via Listserv

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) shall be less than or equal to the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 24% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 24% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT:

a. Once within 12 hours after THERMAL POWER is greater than or equal to 24% of RATED THERMAL POWER and in accordance with the Surveillance Frequency Control Program thereafter.

b. Initially and in accordance with the Surveillance Frequency Control Program when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK [m]	CHANNEL FUNCTIONAL TEST [m]	CHANNEL CALIBRATION [a] [m]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	[b]			2 3, 4, 5
b. Inoperative	NA		NA	2, 3, 4, 5
2. Average Power Range Monitor[f]:				
a. Neutron Flux – Upscale, Setdown	[b]	[l]		2 3, 4, 5
b. Flow Biased Simulated Thermal Power-Upscale	[g]		[d] [e] [h]	1
c. Fixed Neutron Flux - Upscale			[d]	1
d. Inoperative	NA		NA	1, 2, 3, 4, 5
3. Reactor Vessel Steam Dome Pressure - High		[k]		1, 2
4. Reactor Vessel Water Level - Low, Level 3		[k]		1, 2
5. Main Steam Line Isolation Valve - Closure	NA			1
6. This item intentionally blank				
7. Drywell Pressure - High		[k]		1, 2

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK (c)	CHANNEL FUNCTIONAL TEST (c)	CHANNEL CALIBRATION (c)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. RWCU Δ Flow - High				1, 2, 3
b. RWCU Δ Flow – High, Timer	NA			1, 2, 3
c. RWCU Area Temperature - High	NA			1, 2, 3
d. RWCU Area Ventilation Δ Temperature - High	NA			1, 2, 3
e. SLCS Initiation	NA	(b)	NA	1, 2
f. Reactor Vessel Water Level - Low Low, Level 2				1, 2, 3
g. Manual Initiation	NA	(a)	NA	1, 2, 3
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Δ Pressure (Flow) - High	NA			1, 2, 3
b. RCIC Steam Line Δ Pressure (Flow) – High, Timer	NA			1, 2, 3
c. RCIC Steam Supply Pressure - Low	NA			1, 2, 3
d. RCIC Turbine Exhaust Diaphragm Pressure - High	NA			1, 2, 3
e. RCIC Pump Room Temperature - High	NA			1, 2, 3
f RCIC Pump Room Ventilation Ducts Δ Temperature - High	NA			1, 2, 3
g. RCIC Pipe Routing Area Temperature - High	NA			1, 2, 3
h. RCIC Torus Compartment Temperature -High	NA			1, 2, 3
i. Drywell Pressure - High				1, 2, 3
j. Manual Initiation	NA		NA	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK [c]	CHANNEL FUNCTIONAL TEST [c]	CHANNEL CALIBRATION [c]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Δ Pressure (Flow) - High	NA			1, 2, 3
b. HPCI Steam Line Δ Pressure (Flow) – High, Timer	NA			1, 2, 3
c. HPCI Steam Supply Pressure - Low	NA			1, 2, 3
d. HPCI Turbine Exhaust Diaphragm Pressure - High	NA			1, 2, 3
e. HPCI Pump Room Temperature - High	NA			1, 2, 3
f HPCI Pump Room Ventilation Ducts Δ Temperature - High	NA			1, 2, 3
g. HPCI Pipe Routing Area Temperature - High	NA			1, 2, 3
h. HPCI Torus Compartment Temperature -High	NA			1, 2, 3
i. Drywell Pressure - High	NA			1, 2, 3
j. Manual Initiation	NA		NA	1, 2, 3
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3				1, 2, 3
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA			1, 2, 3
c. Manual Initiation	NA	[a]	NA	1, 2, 3

TABLE 4.3.3.1-1
EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK [(a)]	CHANNEL FUNCTIONAL TEST [(a)]	CHANNEL CALIBRATION [(a)]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. CORE SPRAY SYSTEM				
a. Reactor Vessel Water Level – Low Low Low, Level 1				1, 2, 3, 4*, 5*
b. Drywell Pressure - High				1, 2, 3
c. Reactor Vessel Pressure - Low				1, 2, 3, 4*, 5*
d. Core Spray Pump Discharge Flow - Low (Bypass)				1, 2, 3, 4*, 5*
e. Core Spray Pump Start Time Delay - Normal Power	NA			1, 2, 3, 4*, 5*
f. Core Spray Pump Start Time Delay - Emergency Power	NA			1, 2, 3, 4*, 5*
g. Manual Initiation	NA		NA	1, 2, 3, 4*, 5*
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM				
a. Reactor Vessel Water Level – Low Low Low, Level 1				1, 2, 3, 4*, 5*
b. Drywell Pressure - High				1, 2, 3
c. Reactor Vessel Pressure – Low (Permissive)				1, 2, 3, 4*, 5*
d. LPCI Pump Discharge Flow - Low (Bypass)				1, 2, 3, 4*, 5*
e. LPCI Pump Start Time Delay - Normal Power	NA			1, 2, 3, 4*, 5*
f. Manual Initiation	NA		NA	1, 2, 3, 4*, 5*
3. HIGH PRESSURE COOLANT INJECTION SYSTEM#				
a. Reactor Vessel Water Level – Low Low, Level 2				1, 2, 3
b. Drywell Pressure - High				1, 2, 3
c. Condensate Storage Tank Level - Low				1, 2, 3
d. Suppression Pool Water Level - High				1, 2, 3
e. Reactor Vessel Water Level - High, Level 8				1, 2, 3
f. HPCI Pump Discharge Flow – Low (Bypass)				1, 2, 3
g. Manual Initiation	NA		NA	1, 2, 3

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Drywell average air temperature shall not exceed 135°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the drywell average air temperature greater than 135°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The drywell average air temperature shall be the volumetric average of the temperatures at the following locations and shall be determined to be within the limit in accordance with the Surveillance Frequency Control Program:

	Elevation Zone	Approximate Azimuth*
a.	86'11"-112'8" (under vessel)	90°, 225°, 90°, 270°
b.	86'11"-111'10" (outside of pedestal)	135°, 300°, 100°, 190°
c.	111'10"-139'2"	55°, 240°, 155°, 315°
d.	139'2"-168'0"	45°, 215°, 0°, 90°,180°, 270°
e.	168'0"-192'7"	95°, 130°, 300°, 355°, 45°, 225°

* At least one reading from each elevation zone is required for a volumetric average calculation.

Enclosed are the corrected TS pages for HCGS Amendment No. 187. Please replace the affected pages in the HCGS TSs with the enclosed pages. If you have any questions, please contact me at 301-415-1420.

Sincerely,

/ra/

Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
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Docket No. 50-354

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