June 25, 2010

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE HOPE CREEK GENERATING STATION LICENSE RENEWAL APPLICATION FOR SECTION 4.3 (TAC NO ME1832)

Dear Mr. Joyce:

By letter dated August 18, 2009, as supplemented by letter dated January 23, 2009, Public Service Enterprise Group Nuclear, LLC, submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54 (10 CFR Part 54) for renewal of Operating License No. NPF-57 for the Hope Creek Generating Station. The staff of the U.S. Nuclear Regulatory Commission (NRC or the staff) is reviewing this application in accordance with the guidance in NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants." During its review, the staff has identified areas where additional information is needed to complete the review. The staff's request for additional information is included in the Enclosure. Further requests for additional information may be issued in the future.

Items in the enclosure were provided to John Hufnagel and other members of your staff, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me by telephone at 301-415-2981 or by e-mail at bennett.brady@nrc.gov.

Sincerely,

/RA/

Bennett M. Brady, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosure:
As stated

cc w/encl: See next page

June 25, 2010

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Bennett M. Brady, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-354
Enclosure:
As stated
cc w/encl: See next page
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OFFICE	PM:DLR:RPB1	LA:DLR	BC:DLR:RPB1	PM:DLR:RPB1
NAME	BBrady	SFigueroa	BPham	BBrady
DATE	6/22/10	6/9/10	6/25/10	6/25/10

OFFICIAL RECORD COPY

Letter to T. Joyce from B. Brady dated June 25, 2010

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE HOPE CREEK GENERATING STATION LICENSE RENEWAL APPLICATION FOR SECTION 4.3 (TAC NO ME1832)

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Hope Creek Generating Station

cc:

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Hancocks Bridge, NJ 08038

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Township Clerk
Lower Alloways Creek Township
Municipal Building, P.O. Box 157
Hancocks Bridge, NJ 08038

Mr. Paul Bauldauf, P.E., Asst. Director
Radiation Protection Programs
NJ Department of Environmental Protection and Energy, CN 415
Trenton, NJ 08625-0415

Mr. Brian Beam
Board of Public Utilities
2 Gateway Center, Tenth Floor
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Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
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Mr. Paul Davison
Vice President, Operations Support
PSEG Nuclear LLC
One Alloway Creek Neck Road
Hancocks Bridge, NJ 08038

cc:

Ms. Christine Neely
Director – Regulator Affairs
PSEG Nuclear LLC
One Alloway Creek Neck Road
Hancocks Bridge, NJ 08038

Senior Resident Inspector
Hope Creek Generating Station
U.S. Nuclear Regulatory Commission
Drawer 0509
Hancocks Bridge, NJ 08038

Mr. Earl R. Gage
Salem County Administrator
Administration Building
94 Market Street
Salem, NJ 08079

REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE HOPE CREEK GENERATING STATION (HCGS) LICENSE RENEWAL APPLICATION (LRA) FOR SECTION 4.3 (TAC NO ME1832)

RAI 4.3-01

Background:

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue (Part 1):

LRA Table 4.3.1-1 states that the limiting number of cycles for loss of feed water (FW) heaters (turbine trip with 100% steam bypass and partial FW heater bypass) is 23. In UFSAR Table 3.9-1a, the loss FW heaters transient is separated into two transients for turbine trip with 100% steam bypass and for partial FW heater bypass with three and 20 limiting numbers of cycles, respectfully. It is not clear to the staff whether (i) in the fatigue analyses for the FW nozzles these transients were accounted for as two separate transients and (ii) they should be included into the Metal Fatigue of Reactor Coolant Pressure Boundary Program as two transients with three and 20 limiting numbers of cycles.

Request (Part 1):

Clarify whether (i) in the fatigue analyses for the FW nozzles, the loss of FW heaters transients were accounted for as two separate transients and (ii) they should be included in the Metal Fatigue of Reactor Coolant Pressure Boundary Program as two transients with three and 20 limiting numbers of cycles.

Issue (Part 2):

LRA Table 4.3.1-1 states that the limiting number of cycles for scram (turbine generator trip-feedwater on-isolation valves stay open and all other) is 136. In UFSAR Table 3.9-1, the scram transient is separated into two transients for turbine generator trip-feedwater on-isolation valves stay open and other scrams with 40 and 140 limiting numbers of cycles respectfully. It is not clear to the staff whether (i) in the fatigue analyses for the reactor vessel (RV) and its components other than FW nozzles these transients were accounted for as two separate transients and (ii) should be included into the Metal Fatigue of Reactor Coolant Pressure Boundary Program as two transients.

Request (Part 2):

Clarify whether (i) in the fatigue analyses for the reactor vessel (RV) and its components other than FW nozzles, scrams transients were accounted for as two separate transients and (ii) should be included in the Metal Fatigue of Reactor Coolant Pressure Boundary Program as two transients.

RAI 4.3-02

Background:

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue:

LRA Section 4.3.1 states that 40- and 60-year cycles projections are based on the number of transients experienced at HCGS from plant startup and up to 12/31/2007 and the trends from the last twelve years of plant operation. However, LRA Section 4.3.1 does not provide sufficient information for the staff to conclude that the projection methodology used by the applicant is acceptable and would produce conservative values for 40- and 60-year cycle projections.

Request:

Clarify whether the applicant has been tracking (counting) the number of design basis transient occurrences at Hope Creek from the time when the initial operations of the unit had commenced. For those transients that are required to be tracked and monitored in accordance with Technical Specifications, or applicable design basis criteria, identify how many times the transients have occurred during the time period between December 31, 1995 and December 31, 2007. Clarify whether your assumption that the number of transients occurring during those 12 years of operations remains as a valid basis for calculating the 60-year transient projections for this time limiting aging analysis (TLAA). For these transients, identify all technical bases and assumptions that have been used to inform your conclusion that the cycle accumulation trends during the last twelve years of operation provide a conservative basis for projecting the cycle occurrences for these transients through the expiration of the period of extended operation.

RAI 4.3-03

Background:

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue:

LRA Section B.3 states that as a result of the high pressure coolant injection (HPCI) event experienced in October 2004, the number of injection cycles exceeded the assumed number of cycles in the core spray nozzle fatigue analysis. The corrective action was invoked to evaluate this event, resulting in an analysis indicating that the core spray nozzle cumulative fatigue usage (CUF) was 0.815. LRA Section 4.3.1 states that the applicant performed re-analysis for the core spray nozzle in accordance with ASME B&PV, Section III, 2001 Edition including 2003 Addenda. This re-analysis resulted in the core spray nozzle 40-year CUF of 0.063. It is not clear to the staff what assumptions used in the core spray nozzle re-analysis resulted in reduction of CUF by a factor of 13. Further, the staff identified an inconsistency in the reported 60-year projected CUF value for the core spray nozzle in that the LRA Table 4.3.1-2 lists the value as 0.065 and LRA Table 4.3.5-1 reports the value as 0.0202.

Request:

Explain and justify why the LRA lists two different 60-year projected CUF values for the core spray nozzles. Identify and justify which 60-year non-environmental effects CUF value should be used for the core spray nozzles in LRA Table 4.3.1-2 and 4.3.5-1. Identify the assumptions that were used to reduce the CUF for the core spray nozzle by a factor of 13 in the reanalysis of the component.

RAI 4.3-04

Background:

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue:

UFSAR Section 3.9.1.1.4 states that the transients and the number of cycles specified in UFSAR Table 3.9-1 were considered in 40-year fatigue analyses of the reactor pressure vessel internals. LRA Section 4.3.2 states that the applicant derived 60-year CUF values for reactor pressure vessel components by multiplying 40-year CUFs values by a factor of 1.5, which represent an increase in the plant life from 40 to 60 years. However, for some transients used in the reactor pressure vessel components fatigue analyses, the 40-year cycle projection summarized in LRA Table 4.3.1-1 exceed the values reported in UFSAR Table 3.9-1.

Therefore, to project the reactor pressure vessel internals CUFs to 60 years, the fatigue analyses for these components need to be updated based on the 60-year cycle projections. However, LRA Section 4.3.2 does not provide sufficient information for staff to determine whether the 60-year reactor CUF values for the core support plate, top guide beams, and core differential pressure sensing line (as based on a simple multiplication of the design basis values by a factor of 1.5) are conservative relative to those that would be calculated for these components if the 60-year projections were based on the 60-year cycle projections for the transients that were analyzed in the design basis CUF calculations of these components.

Request:

Provide the basis and justify why the 60-year projected CUF values that have been provided for these components (as based on a simple multiplication of the design basis CUF values by a factor of 1.5) are considered to be conservative relative to those that would be calculated using the actual 60-year cycle projections for the transients that are within the scope of the CUF calculations for these components.

RAI 4.3-05

Background:

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue (1):

LRA Section 4.3.5 states that the applicant identified the plant-specific components for the NUREG/CR-6260 sample locations and equivalent HCGS locations that bound those of NUREG/CR-6260 sample locations (LRA Table 4.3.5-1). Further, the applicant performed environmentally assisted fatigue (EAF) calculation for these equivalent HCGS locations to evaluate the effects of the reactor coolant system environment on fatigue life. However, LRA does not provide sufficient information on the methodology used in determining equivalent HCGS locations that bound those of NUREG/CR-6260 sample locations and the basis for performing EAF calculations for these locations in place of EAF calculation for identified NUREG/CR-6260 plant-specific components.

Request (1):

Explain the methodology used in determining equivalent HCGS locations that bound those of NUREG/CR-6260 sample locations. Identify the technical bases that have been used to confirm that the equivalent HCGS locations bound those of NUREG/CR-6260 sample locations. Provide all technical bases for performing EAF calculation for these locations in place of EAF calculation for identified NUREG/CR-6260 plant-specific components.

Issue (2):

The locations identified and analyzed in NUREG/CR-6260 include typical limiting locations but does not consider all plant-specific components and configurations.

Request (2):

Clarify whether any other plant-specific locations at HCGS are more limiting than those identified in NUREG/CR-6260. If other HCGS plant-specific locations exceed those from NUREG/CR-6260, provide EAF calculations for those locations.

RAI 4.3-06

Background:

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue:

LRA Section 4.3.5 does not provide sufficient information on the basis for assumptions used in the environmental fatigue multipliers (Fen) calculations for the NUREG/CR-6260 sample locations.

Request:

Identify all assumptions used (e.g., sulfur content, temperature, strain rate) in the Fen calculations for the NUREG/CR-6260 sample locations. Provide the basis why the assumptions applied to the Fen calculations (including those for dissolved oxygen level) are reasonable or conservative for determining Fen factors for application to the Hope Creek environmentally-assisted fatigue calculations.

RAI 4.3-07

Background:

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following: (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the extended period of operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Issue:

LRA Section 4.3.5 states that Fen factor of 1.49 was used for the Alloy 600 component (control rod drive penetration and core spray nozzle). Also, the LRA does not provide sufficient information to determine what methodology was used in obtaining Fen. Note that NUREG/CR-6909 incorporates more recent fatigue data using a larger database than prior reports for determining the Fen factor of nickel alloys. The basis methodology for the Fen of nickel alloys described in NUREG/CR-6909 is considered by the staff to represent the most up-to-date method for determining the Fen factor for nickel alloys for license renewal considerations.

Request:

(a) Justify using the value of 1.49 for the Fen factor if it is not a bounding/conservative value for the Alloy 600 component when compared to the Fen factor calculated based on NUREG/CR-6909 for nickel alloys.

(b) Describe the current or future planned actions to update the CUF calculation with Fen factor for the Alloy 600 component only, consistent with the methodology in NUREG/CR-6909. If there are no current or future planned actions to update the CUF calculation with Fen factor for the Alloy 600 component consistent with the methodology in NUREG/CR-6909, provide a justification for not performing the update.