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JUL 22 2010

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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 NRC Request for Additional Information, dated June 25, 2010, Related to Section 4.3 of the Hope Creek Generating Station License Renewal Application

Reference: Letter from Ms. Bennett Brady (USNRC) to Mr. Thomas Joyce (PSEG Nuclear, LLC) "REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE HOPE CREEK GENERATING STATION LICENSE RENEWAL APPLICATION FOR SECTION 4.3 (TAC NO. ME1832)", dated June 25, 2010

In the referenced letter, the NRC requested additional information related to Section 4.3 of the Hope Creek Generating Station License Renewal Application (LRA). Enclosed is the response to this request for additional information.

There are no new or revised regulatory commitments contained in this letter.

If you have any questions, please contact Mr. Ali Fakhar, PSEG Manager - License Renewal, at 856-339-1646.

A142
NRR

JUL 22 2010

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

Executed on 7/22/10

Sincerely,

RC

Robert C. Braun
Senior Vice President Nuclear
PSEG Nuclear LLC

Enclosure: Response to Request for Additional Information

cc: Regional Administrator – USNRC Region I
B. Brady, Project Manager, License Renewal – USNRC
R. Ennis, Project Manager - USNRC
NRC Senior Resident Inspector – Hope Creek
P. Mulligan, Manager IV, NJBNE
L. Marabella, Corporate Commitment Tracking Coordinator
T. Devik, Hope Creek Commitment Tracking Coordinator

Enclosure

Response to Request for Additional Information related to the Hope Creek Generating Station License Renewal Application

RAI 4.3-01
RAI 4.3-02
RAI 4.3-03
RAI 4.3-04
RAI 4.3-05
RAI 4.3-06
RAI 4.3-07

Note: For clarity, portions of the original LRA text are repeated in this Enclosure. Added text is shown in ***Bold Italics***, and deletions are shown with strikethrough text.

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.

<u>PSEG Response</u>:

<u>Part 1</u>

<u>Confirmation of Separate Transient Use</u>

(i) In the fatigue analyses for the FW nozzles, the turbine trip with 100% steam bypass and the partial FW heater bypass were accounted for as two separate transients.
(ii) These transients are included in the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program (Hope Creek LRA Appendix B, Section B.3.1.1) and are counted as two separate transients per the current design basis. As stated in the LRA section 4.3.1, page 4-24, the number of design basis cycles does not represent a design limit. The fatigue usage for a component is normally the result of several different thermal and pressure transients. Exceeding the number of cycles for one transient does not necessarily imply the fatigue usage will exceed an acceptance limit. As such, the two transients will not have limits set for them, since the calculated fatigue usage factor will be the limiting value monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program.

In the case of the FW nozzles, fatigue usage is not calculated directly as a result of specific transient cycles using cycle-based fatigue (CBF). As part of the enhanced program (Enhancement No. 2), FW nozzle fatigue monitoring will be performed using fatigue monitoring software, incorporating a stress-based fatigue (SBF) approach.

As described in LRA section 4.3.1, page 4-24, SBF consists of computing a "real time" stress history for a given component from actual temperature, pressure, and flow histories. The cumulative usage factor (CUF) is then computed from the stress history using appropriate cycle counting techniques and fatigue analysis methodology. A confirmatory evaluation has been performed to verify the conservatism of the Green's Function and associated SBF methodology.

<u>Part 2</u>

(i) The fatigue analyses for all the reactor vessel component locations other than the feedwater nozzle locations, listed in LRA Table 4.3.1-2, combine the two transients (turbine generator trip-feedwater on-isolation valves stay open and all other scrams).

(ii) The two scram transients are included in the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program (Hope Creek LRA Appendix B, Section B.3.1.1). These two scram transients are combined by the fatigue monitoring software and used to compute fatigue usage for the reactor vessel component locations listed in LRA Table 4.3.1-2, other than the feedwater nozzle locations.

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.

PSEG Response

Tracking (counting) the number of design basis transient occurrences
Since the time when initial operations of the unit commenced, Hope Creek has been tracking those transients required by the Hope Creek Technical Specifications. Enhancement 1 for the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program (Hope Creek LRA Section B.3.1.1) is being made to add transients to the program beyond those defined in the Technical Specifications. Those transients added to the program include the design basis transients necessary to monitor those components determined to be within the scope of license renewal, and having a cumulative fatigue usage Time-Limited Aging Analyses (TLAA).

Transient occurrences between December 31, 1995 and December 31, 2007
The transients required to be tracked and monitored for the Hope Creek Technical Specifications or applicable design basis criteria are contained in the LRA Table 4.3.1-1 with a Note 8. Calculation of the projected cycles in Table 4.3.1-1 was performed using the cycles experienced between November 11, 1995 and December 31, 2007, as opposed to those between December 31, 1995 and December 31, 2007 in order to coincide with the beginning of the seventh operating cycle. The method of obtaining the projections was as follows:

Forward projections of transient counts for 40 and 60 years of plant operation (using trending from the 1995 – 2007 time period of plant operation) are calculated as follows:

$$N_{40} = N_{2007} + [\ (N_{2007} - N_{1995}) / Time_{1995\ to\ 2007}\ ] * Time_{2007\ to\ 2026}$$
$$N_{60} = N_{2007} + [\ (N_{2007} - N_{1995}) / Time_{1995\ to\ 2007}\ ] * Time_{2007\ to\ 2046}$$

Note: N_{40} and N_{60} are rounded UP to the nearest integer.

where:	N_{40}	=	the projected number of cycles for 40 years of operation.
	N_{60}	=	the projected number of cycles for 60 years of operation.
	N_{1995}	=	the number of cycles experienced as of RFO6 (assumed to be 11/11/1995).
	N_{2007}	=	the number of cycles experienced as of (12/31/2007)
	$Time_{1995\ to\ 2007}$	=	elapsed time from RFO6 cycle counts (11/11/1995) to the date of the most recent cycle counts (12/31/2007).
	$Time_{2007\ to\ 2026}$	=	elapsed time from most recent cycle counts (12/31/2007) to the end of 40-year operating period (4/11/2026).
	$Time_{2007\ to\ 2046}$	=	elapsed time from most recent cycle counts (12/31/2007) to the end of 60-year operating period (4/11/2046).

The values for the number of cycles experienced between 11/11/1995 and 12/31/2007 (12.14 years) for the transients listed in Table 4.3.1-1 are presented in the table which follows:

Transient Occurrences Between 11/11/1995 and 12/31/2007

Transient	Occurrences
Boltup	10
Design Hydrostatic Test (1,250 psig)	10
Startup	30
Turbine Roll and Increase to Rated Power	30
Loss of Feedwater Heaters (Turbine Trip with 100% Steam Bypass and Partial Feedwater Heater Bypass)	4
SCRAM (Turbine Generator Trip-Feedwater On-Isolation Valves Stay Open and All Other)	28
Reduction to 0% Power	30
Hot Standby	30
Shutdown	30
Vessel Flooding	30
Hydrostatic Test (1,563 psig)	0
Unbolt	10
Pre-Op Blowdown	0
Loss of Feedwater Pumps, Isolation Valves Close	2
Reactor Overpressure with Delayed Scram, Feedwater Stays On, Isolation Valves Stay Open	0
Single Relief or Safety Valve Blowdown	0
Automatic Blowdown	0
Improper Start of Cold Recirc. Loop	0
Sudden Start of Pump in Cold Recirc. Loop	0
Improper Startup with Recirculation Pumps Off & Drain Shut Off	0
Pipe Rupture and Blowdown	0
Natural Circulation Startup	0
Loss of AC Power Natural Circulation Restart	0
Operating Basis Earthquake (OBE)	0
Safe Shutdown Earthquake (SSE) at Rated Operating Conditions	0
Safety Relief Valve (SRV) Actuations + Single + Multiple	 65 1
Core Spray Injection	0
High Pressure Coolant Injection (HPCI)	2
RWCU Pump Trip	37
Standby Liquid Control (SLC) Injection	0
Control Rod Drive (CRD) Events: + CRD Isolation	CRD Isolations (valving out and isolating individual CRD's) are not a normal practice and not tracked by the existing fatigue monitoring program
+ Single CRD Scram	3
+ Single CRD Scram During Refueling	17
Low Pressure Coolant Injection (LPCI)	0
Reactor Recirculation Single Loop Operation	5
Alternate Flood-up Event	2

Valid basis for calculating the 60-year transient projections

The assumption that the number of transients occurring during the last twelve years of operations remains as a valid basis for calculating the 60-year transient projections because these twelve years provide the most representative data for current plant operation. When compared to the initial 9 years of operation, the last 12 years of operation provides data which indicates changes in operation which should be considered in making projections for 60 years of operation.

This assumption is based on the review of the operating history (e.g., unit capacity factor) during these twelve years as compared to that of previous years. Capacity factor increases during the last 12 years when compared to the first 9 years are supported by the improvements made in operation of Hope Creek. Improvements have been made in operation, maintenance, and equipment reliability through company and industry improvement initiatives. The process of continuous improvement through the corrective action process and alliances with INPO, EPRI, and others is expected to improve the operation of the facility to levels even above those experienced to date. Using the number of transients occurring during the last twelve years therefore provides a reasonable input in calculating the 60-year transient projections.

The 60-year transient projections done for license renewal are representative of what is expected based on the transients experienced during the last twelve years. The projections do not take the place of the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program (Hope Creek LRA Appendix B Section B.3.1.1) implementing procedures which will include monitoring these transients to compute the cumulative usage factor (CUF). As stated in the "Disposition" section on LRA page 4-25, "The Metal Fatigue of Reactor Coolant Pressure Boundary program will monitor the numbers of cycles of the design transients and the corresponding CUF for critical reactor pressure vessel components. All necessary plant transient events, as shown in Table 4.3.1-1, will be tracked to ensure that the CUF remains less than the allowable CUF limit for all monitored components." In this section of the LRA it also states that "...the Metal Fatigue of Reactor Coolant Pressure Boundary program will manage the effects of aging due to fatigue on the reactor pressure vessel in accordance with 10 CFR 54.21(c)(1)(iii)."

Technical bases and assumptions for accumulation trends during the last twelve years

The accumulation trends during the last twelve years of operation used in the 60-year transient projections are based on data from the present cycle counting program, and supplemented by retrieval of plant data and operating history to determine a conservative cycle count for these transients.

Assumptions were included to provide a conservative basis for accumulation trends during the last twelve years. One example is when a degree of uncertainty existed, additional cycles were assumed to assure the accumulation of cycles to date were conservative. Another example is when no cycles were found during the last twelve years of operation for a transient, those transients were assigned additional cycles for the projections. This was done to appropriately consider the possibility of the cycles occurring in future years even though no cycles had occurred during the last twelve years of operation. Additionally, for HPCI injection transients, whose rate of occurrence may be affected by Extended Power Uprate (EPU), additional cycles were added to the projections to account for the possibility of a change in the rate of occurrence.

Note that some transients are shown in LRA Table 4.3.1-1 with zero number of cycles projected for 40 and 60 years of operation. However, should these transients occur in the future, they will be counted by the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program (Hope Creek LRA Appendix B Section B.3.1.1).

Therefore, the cycle accumulation trends during the last twelve years of operation, with the Hope Creek assumptions as described above, provide a conservative basis for projecting the cycle occurrences for these transients through the period of extended operation. The Metal Fatigue of Reactor Coolant Pressure Boundary aging management program (Hope Creek LRA Appendix B Section B.3.1.1) provides for monitoring and maintaining transient cycle counts 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.

PSEG Response:

Core Spray Nozzle – Difference in 60-year Projected CUF Values

The CUF values for the Core Spray Nozzle (Safe End/Thermal Sleeve and Nozzle Body) in LRA Table 4.3.1-2, "Fatigue Monitoring Locations for HCGS Reactor Pressure Vessel Components and Estimated CUFs", in the "Design Basis 40-Year CUF" column (with Note 6) were inadvertently not updated to reflect the final results of the calculation revision completed during preparation of the LRA. The updated final 60-Year CUF values are 0.0202 and 0.1063 for the Core Spray Nozzle (Safe End/Thermal Sleeve) and the Core Spray Nozzle (Nozzle Body), respectively. Note 6 in Table 4.3.1-2 indicates the values are not the design basis 40-Year CUF, but are the design basis 60-Year CUF. Therefore, the values presented in Table 4.3.1-2 in the "Design Basis 40-Year CUF" column with Note 6 should be consistent with those presented in Table 4.3.5-1, "Environmental Fatigue Results for HCGS for NUREG/CR-6260 Components", in the "60-Year Fatigue Usage Factor" column. The design basis 60-Year CUF values presented in Table 4.3.5-1 for the Core Spray Nozzle (Safe End/Thermal Sleeve and Nozzle Body) are based on the final results of the revised calculation, which is the current design analysis record. Therefore, LRA Table 4.3.1-2 is revised as shown below.

Table 4.3.1-2
Fatigue Monitoring Locations for HCGS Reactor Pressure Vessel Components and Estimated CUFs

Component	Design Basis 40-Year CUF (1)	Estimated CUF as of 12/31/07 (2)	Estimated 40-Year CUF (3)	Estimated 60-Year CUF (3)	Monitoring Technique (4, 5)
Core Spray Nozzle (Safe End/Thermal Sleeve)	~~0.067~~ (6) ***0.0202***	0.038	0.047	0.065	CBF (NUREG/CR-6260 component)
Core Spray Nozzle (Nozzle Body)	~~0.107~~ (6) ***0.1063***	0.040	0.063	0.087	CBF (NUREG/CR-6260 component)

Core Spray Nozzle – Assumptions Used in the Reanalysis which Reduced CUF

Prior to the most recent calculations performed for 60 years of operation in support of the LRA, the previous analyses performed to evaluate the October 2004 HPCI injection used the original Core Spray Nozzle safe end design. The original safe end design used a threaded-in thermal sleeve, and the analysis applied a stress concentration factor of 5 at this location. Applying the factor of 5 resulted in the primary plus secondary stress intensity range significantly exceeding $3S_m$ (three times the design stress intensity) and a resulting K_e (simplified elastic-plastic strain correction factor) value of 3.33. This threaded location became the bounding location which was evaluated in subsequent analyses. The original analysis design basis CUF value at the bounding location for 40 years was 0.796. This bounding location was evaluated in the Operating Experience example presented in the LRA Appendix B, Section B.3.1.1 on page B-224. Using the original analysis, the CUF value of 0.815 cited in that example was calculated based on accumulated transients up to the date of the operating experience example.

Although the safe end was replaced prior to initial plant operation, this configuration change was not incorporated into the previous fatigue analyses. The new configuration, an integral safe end without threads, was included in the finite element model, and used to perform the fatigue analysis in support of the LRA. Differences between the original safe end design and the installed design evaluated for the LRA include the threaded in thermal sleeve being replaced with an integral safe end and thermal sleeve. The fatigue analysis performed for the LRA considered the integral safe end as fabricated of Alloy 600 instead of stainless steel, with a stainless steel thermal sleeve welded to the integral safe end, plus the addition of a new weld at the safe end to nozzle location.

In addition to the changes in geometry and material, the fatigue analysis performed for the LRA refined the transient parameters, as compared to the simplified transient parameters used in the original analysis. The refinements included more detail with respect to time steps, and nozzle and vessel temperatures and flows, as well as use of actual lower flow rates associated with HPCI injection events, as compared to flow rates shown in the Thermal Cycle Diagram. The Thermal Cycle Diagram assumed all HPCI flow was injected through the Core Spray nozzle even though the system is designed to split the flow between the core spray and feedwater nozzles. A review of the fatigue summary from the previous fatigue analyses shows that the alternating stress values for all transient load set pairs were multiplied by the K_e multiplier of 3.33, whereas only a few load set pairs in the current fatigue analysis are affected by K_e.

The previous fatigue analyses were based on an older design configuration with less design margin for fatigue, and the analyses included significant over conservatism in the analyzed transient parameters. In the finite element model used to perform the new fatigue analysis in support of the LRA, these differences in design, including geometry and material, in addition to refined transient parameters with respect to time steps, temperatures and flows, were incorporated into the analysis. Collectively these differences resulted in a considerable reduction in the calculated design basis 60-year CUF for the Core Spray Nozzle (Safe End/Thermal Sleeve).

With regard to the nozzle body location, the original design basis 40-year CUF was 0.071. It did not experience a similar significant reduction in resultant calculated fatigue usage, but instead is comparable based on the "Estimated 40-Year CUF" shown in LRA Table 4.3.1-2.

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.

PSEG Response:

A review of the design basis CUF calculations was performed for the Core Support Plate (at stud), the Top Guide (at beam slot), and the Core Differential Pressure Sensing Line (at elbow). The 60-year projections for the transients which were paired within the scope of the design basis CUF calculations for these components were determined to increase no greater than 1.5 times the 40-year design input pairing values in the original design basis calculations. Therefore, the use of the simple multiplication of the design basis CUF by a factor of 1.5 is conservative relative to the CUF which 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.

PSEG Response:

(1) Methodology Used to Establish NUREG/CR-6260 Equivalent Locations

NUREG/CR-6260 includes the results of environmental fatigue evaluations of selected reactor coolant pressure boundary components at a sample of existing light water reactor facilities, including one older vintage and one newer vintage General Electric (GE) plant. The evaluations assessed the impact of interim environmental fatigue curves (from NUREG/CR-5999) on existing fatigue analyses at the sample plants.

A review of NUREG/CR-6260 was performed to determine whether the older or newer vintage General Electric (GE) plant was more appropriate for comparison to Hope Creek Generating Station (HCGS), since the HCGS plant design has attributes of both vintages. The newer vintage GE plant evaluated in NUREG/CR-6260 is the appropriate comparison to HCGS, since the HCGS piping design is ASME Section III.

NUREG/CR-6260, page 5-79, Section 5.6 provides the list of sample components to be evaluated for environmentally-assisted fatigue. For each of the sample components, the location having the highest design CUF was identified from the sample plant licensee's design basis calculations. This limiting location was then evaluated for environmentally assisted fatigue (EAF) using the NUREG/CR-5999 interim fatigue curves.

The equivalent HCGS plant-specific bounding locations were determined based on the most limiting (highest design CUF) location for each of the same six identical reactor pressure vessel (RPV) and Class 1 piping components identified in NUREG/CR-6260 for the newer vintage GE plant. The HCGS locations with the highest CUF were identified from HCGS design basis calculations.

The following table lists in the first column, each of the NUREG/CR-6260 components and associated limiting locations for the newer vintage GE plant. In the second column, the table lists the HCGS component that is equivalent to the NUREG/CR-6260 component. For each HCGS component, the table provides the HCGS plant-specific locations evaluated and determined to be the locations of highest design CUF from the HCGS design basis calculations. Therefore, these locations are bounding HCGS locations for the equivalent NUREG/CR-6260 sample components. These bounding locations were then evaluated for environmentally assisted fatigue using NUREG/CR-6583 for carbon and low-alloy steels, and NUREG/CR-5704 for austenitic stainless steels, or an approved technical equivalent.

The methodology used in determining the equivalent bounding HCGS locations as described above, is consistent with the NUREG/CR-6260 methodology, and meets the recommendations of NUREG-1801 aging management program X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary. The use of this methodology with design information from HCGS design basis calculations provides the technical basis for the equivalent HCGS location determinations and for performing the EAF calculations for these locations.

Newer Vintage GE Plant NUREG/CR-6260 Component and Location of Highest Design CUF	Equivalent HCGS Component / Location of Highest Design CUF/Technical Basis
NUREG/CR-6260 Section 5.6.1: Reactor Vessel shell and lower head. Highest design CUF is at the weld region between the low alloy steel shell and the Alloy 600 CRDM penetration.	Reactor Vessel shell and lower head component design fatigue analyses were evaluated and determined the stainless steel CRD penetration drive housing and the Alloy 600 CRD penetration with weld excavation were the locations with the highest design CUF and were used to perform the EAF calculations. Although the weld excavation location had a higher CUF, both locations were evaluated due to differing F_{en} multipliers. The CRD penetration drive housing was determined to be the most limiting location on this NUREG/CR-6260 component.
NUREG/CR-6260 Section 5.6.2: Reactor Vessel Feedwater nozzle. Highest design CUF is at the thermal sleeve and safe end.	The Reactor Vessel Feedwater Nozzle component design fatigue analyses were evaluated and determined locations with the highest design CUF are the Feedwater nozzle safe end and nozzle forging (blend radius or nozzle corner) and were used to perform the EAF calculations. The safe end was determined to be the most limiting location for this NUREG/CR-6260 component. The thermal sleeve at HCGS is not welded to the safe end as compared to the thermal sleeve configuration evaluated in NUREG/CR-6260, therefore the thermal sleeve does not experience high fatigue usage.
NUREG/CR-6260 Section 5.6.3: Reactor Recirculation piping (including the inlet and outlet nozzles). Highest design CUF is at the 304 stainless steel suction piping tee.	The Reactor Recirculation piping (including the inlet and outlet nozzles) component design fatigue analyses were evaluated and determined the Reactor Recirculation inlet and outlet nozzle blend radius locations had the highest design CUF for the recirculation nozzles and were used to perform the EAF calculations. The Reactor Recirculation piping design fatigue analyses were evaluated and determined the stainless steel tee on the suction piping that interfaces with the RHR system was the NUREG/CR-6260 location having the highest design CUF and was used to perform the EAF calculations.
NUREG/CR-6260 Section 5.6.4: Core Spray line reactor vessel nozzle and associated Class 1 piping. Highest design CUF is at the nozzle thermal sleeve and safe end.	The Core Spray line reactor vessel nozzle and associated Class 1 piping design fatigue analyses were evaluated and determined the Core Spray nozzle safe end (including thermal sleeve) and blend radius (nozzle corner) had the highest design CUF and were used to perform the EAF calculations. The nozzle corner was determined to be the location on this NUREG/CR-6260 component to have the highest design CUF.
NUREG/CR-6260 Section 5.6.5: Feedwater line Class 1 piping. Highest design CUF is at a 12 inch long radius low alloy steel elbow.	The Feedwater line Class 1 piping design fatigue analyses were evaluated and determined the Tee on the Loop B Feedwater piping was the NUREG/CR-6260 location with the highest design CUF and was used to perform the EAF calculations.
NUREG/CR-6260 Section 5.6.6: Residual Heat Removal (RHR) Nozzles and associated Class 1 piping. Highest design CUF is in the carbon steel piping.	The Residual Heat Removal (RHR) Nozzles and associated Class 1 piping design fatigue analyses were evaluated and determined the locations with the highest design CUF for this NUREG/CR-6260 component were on the Class 1 carbon steel supply piping and the stainless steel piping and were used to perform the EAF calculations.

Using the newer vintage GE plant components presented in NUREG/CR-6260, HCGS selected the same components for the evaluation of environmentally assisted fatigue. The location selected on the component to perform the EAF calculation is the location of highest design CUF. This forms the technical bases to confirm the equivalent HCGS locations on the component, presented in the Hope Creek LRA Table 4.3.5-1, bound those selected as NUREG/CR-6260 sample locations. The equivalent HCGS locations determined to have the highest design CUF and their associated EAF calculations, therefore, bound those EAF calculations for identified NUREG/CR-6260 plant specific components.

(2) Identify and Evaluate other More Limiting Locations than NUREG/CR-6260 Locations

As described in the response to Request(1), the methodology used in determining the equivalent HCGS locations is consistent with the NUREG/CR-6260 methodology and meets the recommendations contained in NUREG-1801, Revision 1 for Aging Management Program X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary". The plant-specific locations at HCGS that are the more limiting locations and equivalent to those identified in NUREG/CR-6260 are identified in the above table and in LRA Table 4.3.5-1, and are the most limiting for the NUREG/CR-6260 component. EAF calculations have been performed for these most limiting locations. Therefore, there are no other HCGS plant specific locations more limiting for the NUREG/CR-6260 components and no additional calculations are required.

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.

PSEG Response:

Basis for Assumptions used in Environmental Fatigue Multiplier (Fen) Calculations

There are several parameters which are used in calculating the environmental fatigue multiplier (F_{en}). These include sulfur content, temperature, strain rate, and dissolved oxygen. In all cases, values used for sulfur content were those which maximized the F_{en} multiplier. In all cases except for the Feedwater nozzle, the temperature value used was the design temperature of 550°F. For the Feedwater nozzle, the temperature used was the maximum temperature for each load pair in the detailed 60-Year fatigue calculation. In all cases, values used for strain rate were those which maximized the F_{en} multiplier.

For load pairs that are subject to dynamic loading, F_{en} = 1.0. This is based on the premise that the cycling due to dynamic loading occurs too quickly for environmental effects to be significant. Accordingly, only those transient pairs which do not have a dynamic loading transient require application of environmental fatigue multipliers.

Values for dissolved oxygen in the Feedwater and Recirculation systems, recorded over more than 18 years of plant operation, were used in determining the dissolved oxygen input to the F_{en} calculations. Dissolved oxygen measurements included both normal water chemistry (NWC) and hydrogen water chemistry (HWC) conditions. In order to correctly account for the effects of reactor coolant environment, F_{en} values were calculated for periods where dissolved oxygen values changed due to the water chemistry conditions in effect (NWC or HWC) during those operating periods. This was determined by calculating a weighted average of the actual and projected availability of HWC. Over the projected 60 year operating life, NWC accounts for 15% of the operating time and HWC accounts for the remaining 85% of operating time. Measured values of dissolved oxygen were averaged over periods of time for each of the operating conditions (NWC and HWC) and one standard deviation was applied to the data to obtain the dissolved oxygen values to be used in the F_{en} calculations. Dissolved oxygen data was not readily available for the period of time from startup until approximately three years prior to inception of hydrogen water chemistry (approximately 3 years). This represents approximately one half of the NWC period. During this period, chemistry controls were similar to those during the period for which data was available, and therefore dissolved oxygen levels were assumed to have been similar to those experienced during the time period for which data was available.

With measured data as an input, the Boiling Water Reactor Vessel and Internals Applications (BWRVIA) model was used to determine the dissolved oxygen values at the other NUREG/CR-6260 locations. The BWRVIA model software is a recognized industry and EPRI-sponsored modeling program used to obtain dissolved oxygen levels at locations when direct sample measurements for dissolved oxygen cannot be obtained.

Values for dissolved oxygen obtained as a result of these methodologies are presented in LRA Table 4.3.5-1, "Environmental Fatigue Results for HCGS for NUREG/CR-6260 Components", Note 4. The dissolved oxygen inputs to the F_{en} calculations are considered to be reasonable, based on 1) the use of more than 18 years of actual measured dissolved oxygen data that includes all HWC operating periods, 2) the use of NWC values representative of what chemistry controls would provide for the operating period prior to measured values being available, and 3) the use of industry recognized modeling software to obtain dissolved oxygen levels at locations when direct sample measurements cannot be obtained.

Based on the bases above, all the assumptions applied to the F_{en} calculations for the Hope Creek NUREG/CR-6260 locations are reasonable or conservative.

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.

PSEG Response:

(a) Basis for Use of 1.49 F_{en} Multiplier for Alloy 600 Materials

The value of 1.49 for F_{en} as shown in LRA Table 4.3.5-1, "Environmental Fatigue Results for HCGS for NUREG/CR-6260 Components", for the Control Rod Drive (CRD) Penetration with Excavation and Core Spray Nozzle Safe End locations is less than the value that would be calculated from NUREG/CR-6909 for nickel alloys. The F_{en} value of 1.49 for Alloy 600 components (Control Rod Drive Penetration with Excavation and Core Spray Nozzle Safe End) is determined based on the Alloy 600 methodology documented in NUREG/CR-6335.

A review was performed which indicates that using NUREG-6909 methodology would result in a conservative value for F_{en} of 3.56 for these two Alloy 600 locations. Using Equations A.14 through A.17 contained in Appendix A to NUREG/CR-6909, this conservative review was based on a reactor maximum temperature of 550°F (288°C), and a value for transformed strain rate to maximize F_{en}. The following relationship was used to account for overall hydrogen water chemistry (HWC) availability of 85% (as presented in Table 4.3.5-1, note 4) for the overall 60-year operating period:

Overall F_{en} = 0.85*F_{en} HWC + (1-0.85)*F_{en} NWC
Overall F_{en} = 0.85*(3.81) + (1-0.85)*(2.12) = 3.56

Using the conservative NUREG/CR-6909 methodology, the resultant environmentally assisted fatigue CUF for the CRD Penetration with Excavation is 0.80. For the Core Spray nozzle safe end, the resultant environmentally assisted fatigue CUF is 0.10. Using the NUREG/CR-6335 methodology, the environmentally assisted fatigue CUF values are 0.4119 for the CRD Penetration with Excavation, and 0.0301 for the Core Spray nozzle safe end, as shown in LRA Table 4.3.5-1. While the calculated environmentally assisted CUF values are higher using the conservative NUREG/CR-6909, they remain below the allowable value of 1.0 using either methodology.

NUREG-1800, Rev 1, Section 4.3.3.2 Generic Safety Issue, states that formulas for calculating the environmental life correction factors are those contained in NUREG/CR-6583 for carbon and low-alloy steels, and in NUREG/CR-5704 for austenitic stainless steels, or an approved technical equivalent. NUREG/CR-6335 is a previously approved technical equivalent for determining environmental life correction factors for Alloy 600 components. Therefore, the value of 1.49 for F_{en} that was calculated based on the Alloy 600 methodology documented in NUREG/CR-6335 is justified for license renewal.

In addition, Regulatory Guide 1.207 endorses the NUREG/CR-6909 methodology specifically for new plants, stating that "Because of significant conservatism in quantifying other plant-related variables (such as cyclic behavior, including stress and loading rates) involved in cumulative fatigue life calculations, the design of the current fleet of reactors is satisfactory." Finally, if the conservative F_{en} value of 3.56 is used and the NUREG/CR-6909 methodology is applied to calculate the 60 year environmentally assisted CUFs for the Hope Creek Alloy 600 locations, the environmentally assisted CUF values remain below 1.0 and are, therefore, acceptable for the period of extended operation.

(b) <u>Describe Planned Actions to Update the CUF Calculation Using NUREG/CR-6909</u>

As presented above, a conservative application of the NUREG/CR-6909 methodology for the Hope Creek Alloy 600 locations (Control Rod Drive Penetration with Excavation and Core Spray Nozzle Safe End) determined the 60-Year CUF values with F_{en} factor remain below 1.0, and are acceptable for the period of extended operation. Therefore, there are no planned actions to update the CUF calculations with F_{en} factor consistent with the methodology in NUREG/CR-6909.