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10 CFR 50
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SEP 20 2010

LR-N10-0356

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: Revised RAI 4.3-01 response associated with the Hope Creek Generating Station License Renewal Application

References: Letter from Mr. Robert C. Braun (PSEG Nuclear, LLC) to USNRC "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," dated July 22, 2010

In the referenced letter, PSEG Nuclear responded to NRC RAI 4.3-01, related to fatigue monitoring associated with the Hope Creek Generating Station License Renewal Application. The NRC staff reviewed that response and requested follow-up discussions. As a result of the subsequent discussions between NRC Staff and PSEG Nuclear representatives, PSEG Nuclear is providing a replacement RAI 4.3-01 Part 1 response. The replacement RAI 4.3-01 Part 1 response is contained within Enclosure A to this letter.

This revised response results in a change to the license renewal commitment for Metal Fatigue of the Reactor Coolant Pressure Boundary aging management program (Commitment # 46). This updated commitment is provided within Enclosure B to this letter.

This submittal has been discussed with the NRC License Renewal Project Manager for the Hope Creek License Renewal project.

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

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

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

Executed on 9/20/10

Sincerely,

Robert C. Braun
Senior Vice President, Operations
PSEG Nuclear LLC

Enclosure A: Replacement Response to NRC RAI 4.3-01 Part 1 related to the Hope Creek Generating Station License Renewal Application

Enclosure B: Revised Commitment associated with Replacement Response to NRC RAI 4.3-01 Part 1, Metal Fatigue of the Reactor Coolant Pressure Boundary

cc: Regional Administrator – USNRC Region I
B. Brady, Project Manager, License Renewal – USNRC
R. Ennis, Project Manager - USNRC
NRC Senior Resident Inspector - Hope Creek
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L. Marabella, Corporate Commitment Tracking Coordinator
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Enclosure A

Replacement Response to NRC RAI 4.3-01 Part 1 related to the Hope Creek Generating Station License Renewal Application

Note: Using the original 4.3-01 RAI Part 1 response as a reference and to provide clarity, added text is shown in ***Bold Italics***, and deletions are shown with strikethrough text.

RAI 4.3-01 Part 1

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.

PSEG Response:

Part 1

Confirmation of Separate Transient Use

(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.~~

In addition to the above revision to the response to this RAI, and based on discussions held with NRC Staff after their review of the response, a revision to the Hope Creek LRA is provided with this revised response. Due to NRC Staff concerns with the fatigue monitoring program confirmatory evaluation performed for Hope Creek to address RIS-2008-30, stress based fatigue (SBF) monitoring will not be used at this time. Instead, Hope Creek will only use cycle based fatigue (CBF) monitoring. CBF monitoring uses the design basis fatigue calculations which consider the six stress terms in accordance with the methodology from ASME Section III, Subsection NB, Subarticle NB-3200 for the reactor pressure vessel components. Should SBF monitoring be used for the fatigue monitoring program in the future, it will consider the six stress terms in accordance with the methodology from ASME Section III, Subsection NB, Subarticle NB-3200.

Therefore, the following portions of the Hope Creek LRA are revised as part of the response to this RAI. The pages which follow provide the necessary changes indicated by bold and italicized for additions, and strikethrough for deletions. Portions of the LRA affected by previous docketed responses have been incorporated and are shown in normal font.

LRA Section 4.3.1 (beginning on page 4-22):

4.3.1 REACTOR PRESSURE VESSEL FATIGUE ANALYSES

Summary Description

Reactor pressure vessel fatigue analyses, including the vessel support skirt, shell, upper and lower heads, closure flanges and studs, nozzles and penetrations, nozzle safe ends, and refueling bellows support depend on the assumed numbers and the severity of normal and upset event pressure and thermal operating cycles to predict end-of-life fatigue usage factors in accordance with Section III of the ASME Code. These assumed cycle counts used to determine fatigue usage factors are based on the original 40-year design life of the plant. The calculation of fatigue usage factors is part of the current licensing basis and is used to support safety determinations. As such the reactor pressure vessel fatigue analyses meet the requirements of 10 CFR 54.3(a) for a TLAA.

Analysis

The original HCGS reactor pressure vessel stress report included fatigue analyses for the reactor pressure vessel components based on a set of design basis duty cycles. These duty cycles are listed in Table 3.9-1 and 3.9-1a of the HCGS UFSAR, and are shown in Table 4.3.1-1. The original 40-year analyses demonstrated that the cumulative usage factors (CUFs) for all critical components would remain below the allowable fatigue usage value of 1.0 specified in Section III of the ASME Code. The HCGS reactor pressure vessel was designed in accordance with ASME Code, Section III, 1968 Edition, with addenda up to and including Winter 1969, as documented in Section 5.3.3.1 of the HCGS UFSAR.

Revised CUF evaluations were also performed to assess the impact of Extended Power Uprate (EPU) on the reactor pressure vessel for HCGS. These evaluations revised the CUF values for some reactor pressure vessel components based on the change in reactor operating conditions resulting from EPU. The revised CUF evaluations were approved by the NRC as a part of the EPU approval process (Reference 4.8.18).

In addition, revised fatigue evaluations were performed for the recirculation inlet nozzle for containment or "new" loads in 1988, for the top head and vessel flanges for asymmetric head spray cooling in 1990, for the main closure region to support reduced-pass stud tensioning in 2000, for the reactor pressure vessel support skirt to evaluate cumulative fatigue loadings in 1996, and for the core spray nozzles to evaluate the impact of additional HPCI injections in 2008.

The list of design transients used in the reactor pressure vessel fatigue analyses was intended to envelope all foreseeable thermal and pressure cycles that could be expected to occur within a nominal 40-year operating period for the plant. The list of controlling transients for HCGS is shown in Table 4.3.1-1. This list encompasses all transients listed in Tables 3.9-1 and 3.9-1a of the HCGS UFSAR for the reactor pressure vessel, but also includes all transients relevant to fatigue accumulation in all reactor pressure vessel, Class 1 piping, and containment components where a fatigue basis exists. The actual numbers of events experienced to-date (12/31/2007) are also listed in Table 4.3.1-1. The number of transients experienced to-date for the reactor pressure vessel and other analyzed components was compiled from the HCGS Cycle Counting program, which has been in place since plant startup. The sources of data used by that program include operator log books, plant instrument data, event reports, NRC correspondence, surveillance test results, equipment logs and operating experience reports. The numbers of occurrences expected for 40 and 60 years of operation were obtained by extrapolating the numbers of occurrences actually incurred to-date, and using the rate of occurrence experienced during the last twelve years of operation (nine operating cycles). The frequency of events, such as scrams and shutdowns, experienced in the last twelve years is significantly less than that experienced during the first ten years of operation, and is expected to remain equal to or less than the trend over the past twelve years through the period of extended operation by maintaining careful attention to good operating practices. Conservatism was added beyond the mathematically projected number of cycles to accommodate potential variation in plant performance late in plant life, as well as to allow for additional events where the projected number of cycles was very low and the likelihood of additional events could not be ruled out.

The projected numbers of occurrences for each event for 40 and 60 years are also included in Table 4.3.1-1, as are the numbers of cycles assumed in the design basis 40-year fatigue analyses. There are several transients in this table whose 60-year Assumed Number of Cycles exceed the Designed Analyzed Cycles for 40 years. The number of design basis cycles does not represent a design limit. The fatigue 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. Those transients that are important to determining a CUF for all critical reactor coolant pressure boundary locations by the Metal Fatigue of Reactor Coolant Pressure Boundary (B.3.1.1) aging management program are noted in Table 4.3.1-1.

The CUFs of the reactor pressure vessel, including the vessel support skirt, shell, upper and lower heads, closure flanges and studs, nozzles and penetrations, nozzle safe ends, and refueling bellows support will be managed by the Metal Fatigue of Reactor Coolant Pressure Boundary (B.3.1.1) aging management program. This program will monitor critical reactor pressure vessel CUFs through the use of a fatigue monitoring software application using ~~either stress-based fatigue (SBF) monitoring or~~ cycle-based fatigue (CBF) monitoring versus the allowable value.

~~Stressed-based fatigue monitoring consists of computing a "real time" stress history for a given component from actual temperature, pressure, and flow histories via a finite element-based Green's Function approach. CUF is then computed from the computed stress history using appropriate cycle counting techniques and appropriate ASME Code, Section III fatigue analysis methodology. The NRC concern regarding the simplified input to the Greens' function of only one value of stress expressed in RIS 2008-30 has been addressed in completion of the work done for Hope Creek by completing a detailed stress analysis using the six stress components as discussed in ASME Code, Section III, Subsection NB, Subarticle NB-3200. SBF monitoring is intended to duplicate the methodology used in the governing ASME Code stress report for the component in question, but uses actual transient severity in place of design basis transient severity. Confirmatory evaluation has been performed to verify the conservatism of the HCGS Green's Function and associated SBF methodology.~~

Cycle-based fatigue monitoring consists of a two-step process: (a) automated cycle counting, and (b) CUF computation based on the counted cycles. Automated cycle counting evaluates each transient that is defined in the plant licensing basis based upon the mechanistic process or sequence of events experienced by the plant (as determined from monitored plant instruments). The approach is conservative because it assumes each actual transient has a severity equal to that assumed in the design basis. The unique severity of any transient identified by the aging management program software is captured for each monitored component for ready comparison to design basis transient severity. Transients defined in the HCGS UFSAR are identified and implemented into the fatigue monitoring software. CUF computation calculates fatigue directly from counted transients and parameters for the monitored components. CUF is computed via a design-basis fatigue calculation where the actual numbers of counted cycles are substituted for the assumed design basis number of cycles using the governing stress report methodology.

All locations with CUF ratios (i.e., CUF/allowable) predicted to exceed 0.4 (or 40% of allowable) in the original design basis fatigue analysis will be included in the program. In addition, the locations identified in NUREG/CR-6260 for the newer-vintage General Electric plant, which have been evaluated for environmental fatigue effects as discussed in Section 4.3.5 below, have been included in the program. The list of monitored reactor pressure vessel locations is listed in Table 4.3.1-2. ***All of the CUF values reported for the reactor pressure vessel components in Table 4.3.1-2 and 4.3.5-1 were computed based on the fatigue tables from the design basis calculations which consider the six stress terms in accordance with the methodology from ASME Section III, Subsection NB, Subarticle NB-3200. The design basis calculations were done using the code of record or updated to a later code edition pursuant to 10 CFR 50.55a. Therefore the concerns of RIS 2008-30 have been addressed in performing the work for Hope Creek.***

One of the reactor pressure vessel components, the core spray nozzles, indicated fatigue usage over the allowable value prior to 40 years of operation when using actual numbers of cycles accumulated to-date and projected to 40 years of plant operation. The CUF for this component was re-analyzed in accordance with ASME Code, Section III, 2001 Edition, including addenda up to and including 2003, which have been accepted for use by the NRC through 10 CFR 50.55(a). Reconciliation was performed to justify the use of the later edition of the ASME Code compared to the edition originally used to design the HCGS reactor pressure vessel). The updated CUF value for the core spray nozzles is listed in Table 4.3.1-2.

Disposition: Aging Management, 10 CFR 54.21(c)(1)(iii)

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 the event the monitored CUF is predicted to exceed the allowable value for any component prior to 60 years of operation, appropriate corrective action will be taken in accordance with the corrective action process prior to the allowable limits being exceeded. HCGS has an existing program in place to track operating thermal and pressure cycles and to assess their effect on vessel fatigue. The requirements from this program will be incorporated into the Metal Fatigue of Reactor Coolant Pressure Boundary (B.3.1.1) aging management program. The required implementing actions will be completed prior to the period of extended operation. As such, 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).

Table 4.3.1-1
HCGS Reactor Pressure Vessel Design Transients and 60-Year Cycle Projections

Transient	Included in Table 3.9-1 or 3.9-1a of UFSAR?	Design Number of Cycles (1)	Cycles as of 12/31/07	40-Year Projected Number of Cycles (9)	60-Year Projected Number of Cycles (9)	60-Year Number of Cycles Assumed for Analysis
Boltup (8)	Y	44	18	34	50	55
Design Hydrostatic Test (1,250 psig) (8)	Y	44	18	34	50	55
Startup (8)	Y	117	79	125	174	180
Turbine Roll and Increase to Rated Power (8)	Y	117	79	125	174	180
Daily Reduction to 75% Power	Y	6,667	See Note 10			6,667
Weekly Reduction to 50% Power	Y	1,233	See Note 10			1,233
Rod Pattern Change	Y	400	See Note 10			400
Loss of Feedwater Heaters (Turbine Trip with 100% Steam Bypass and Partial Feedwater Heater Bypass) (8)	Y	23	10	16	22	25
SCRAM (Turbine Generator Trip-Feedwater On-Isolation Valves Stay Open and All Other) (8)	Y	136	80	124	169	175
Reduction to 0% Power (8)	Y	111	79	125	174	180
Hot Standby (8)	Y	111	79	125	174	180
Shutdown (8)	Y	111	79	125	174	180
Vessel Flooding (8)	Y	111	79	125	174	180
Hydrostatic Test (1,563 psig) (8)	Y	1	2	2	2	3
Unbolt (8)	Y	44	18	34	50	55
Pre-Op Blowdown (8)	Y	10	1	1	1	2
Loss of Feedwater Pumps, Isolation Valves Close (2,8)	Y	5	6	10	13	15
Reactor Overpressure with Delayed Scram, Feedwater Stays On, Isolation Valves Stay Open (8)	Y	1	0	0	0	See Note 3
Single Relief or Safety Valve Blowdown (8)	Y	8	2	4	6	See Note 3
Automatic Blowdown (8)	Y	1	0	0	0	See Note 3
Improper Start of Cold Recirc. Loop (8)	Y	1	0	0	0	See Note 3
Sudden Start of Pump in Cold Recirc. Loop (8)	Y	1	0	0	0	See Note 3
Improper Startup with Recirculation Pumps Off & Drain Shut Off (8)	Y	1	0	0	0	See Note 3
Pipe Rupture and Blowdown (8)	Y	1	0	0	0	See Note 4
Natural Circulation Startup (8)	Y	3	0	0	0	3
Loss of AC Power Natural Circulation Restart (8)	Y	5	See Note 6			5
RPV Drain Line Flow Transient	N	480	See Note 7			480
Operating Basis Earthquake (OBE) (8)	Y	10 / 50 (5)	0	0	0	10 / 50 (5)
Safe Shutdown Earthquake (SSE) at Rated Operating Conditions (8)	Y	1	0	0	0	See Note 4
Safety Relief Valve (SRV) Actuations (8): + Single + Multiple	N	966 596 370	390 380 10	505 487 18	618 592 26	966 596 370
Core Spray Injection (8)	N	10	3	3	5	5
High Pressure Coolant Injection (HPCI) (8)	N	15	21	25	34	35
RWCU Pump Trip (8)	N	240	85	140	200	240

For notes, see next page.

Table 4.3.1-1 (continued)
HCGS Reactor Pressure Vessel Design Transients and 60-Year Cycle Projections

Transient	Included in Table 3.9-1 or 3.9-1a of UFSAR?	Design Analyzed Number of Cycles (1)	Cycles as of 12/31/07	40-Year Projected Number of Cycles (9)	60-Year Projected Number of Cycles (9)	60-Year Number of Cycles for Assumed for Analysis
Standby Liquid Control (SLC) Injection (8)	N	10	0	0	0	10
Control Rod Drive (CRD) Events (8):	N	360	30	61	94	205
+ CRD Isolation		50	Note 11	Note 11	Note 11	100
+ Single CRD Scram		10	7	12	17	20
+ Single CRD Scram During Refueling		300	23	49	77	85
Low Pressure Coolant Injection (LPCI) (8)	N	11	3	4	5	5
Reactor Recirculation Single Loop Operation (8)	N	50	10	18	26	29
Alternate Flood-up Event (8)	Y	28	2	18	34	38

Notes:

1. Minimum number of events reported from all UFSAR sources.
2. Reclassified from an Emergency event to an Upset event.
3. Emergency event, so not included in fatigue analysis.
4. Faulted event, so not included in fatigue analysis..
5. 50 peak OBE cycles for the NSSS piping; 10 peak OBE cycles for other NSSS equipment and components.
6. Event no longer considered relevant to HCGS, as it is procedurally prevented; events will be included in the Fatigue Monitoring program for any components where this event was included in the fatigue analysis.
7. Event specified on design basis Thermal Cycle Diagram; insignificant impact on fatigue for all critical RPV Class 1 component fatigue analyses, so not tracked in Fatigue Monitoring program.
8. Transient will be tracked by the Metal Fatigue of Reactor Coolant Pressure Boundary (B.3.1.1) aging management program.
9. Projected for 40 and 60 years based on the number of events as of 12/31/07 and the trends from the past twelve years (nine operating cycles) of actual plant operation.
10. This event has an insignificant impact on fatigue, and load following is not practiced at HCGS; events will be included in the Fatigue Monitoring Program for any components where this event was included in the fatigue analysis.
11. CRD isolations are not a normal practice at HCGS and are not tracked as part of the existing fatigue monitoring program. A conservative and bounding number of cycles has been provided for analysis purposes, and an initial bounding number of cycles will be assigned as part of the enhanced Metal Fatigue of Reactor Coolant Pressure Boundary (B.3.1.1) aging management program prior to the period of extended operation.

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)
Main Closure Studs	0.872	0.505	0.854	1.221 (8)	CBF
Shroud Support (Shroud Cylinder)	0.672	0.205	0.333	0.465	CBF
CRD Penetrations (CRD Housing @ Weld)	0.021	0.011	0.019	0.034	CBF (NUREG/CR-6260 component)
Core Spray Nozzle (Safe End/Thermal Sleeve)	0.0202 (6)	0.038	0.047	0.065	CBF (NUREG/CR-6260 component)
Core Spray Nozzle (Nozzle Body)	0.1063 (6)	0.040	0.063	0.087	CBF (NUREG/CR-6260 component)
Top Head Lifting Lug Bracket	0.688	0.260	0.410	0.565	CBF
CRD Penetrations with Excavation (B)	0.456	0.093	0.154	0.216	CBF (NUREG/CR-6260 component)
Recirculation Outlet Nozzle (Nozzle Body)	0.086	0.022	0.036	0.051	CBF (NUREG/CR-6260 component)
Recirculation Inlet Nozzle (Nozzle Body)	0.116	0.036	0.058	0.081	CBF (NUREG/CR-6260 component)
Feedwater Nozzle (Safe End)	0.014	0.084	0.121	0.198	~~SBF~~ ***CBF*** (NUREG/CR-6260 component)
Feedwater Nozzle (Nozzle Forging)	0.118 (7)	0.066 (7)	0.115 (7)	0.168 (7)	~~SBF~~ ***CBF*** (NUREG/CR-6260 component)

Notes:

1. Based on the currently governing design basis 40-year fatigue analysis. Allowable CUF = 1.0 for all components.
2. Estimated CUF as of 12/31/07 is based on the cycles accumulated to-date as of 12/31/07 from Table 4.3.1-1. For the feedwater nozzle ~~(SBF)~~ locations, the estimated CUF as of 12/31/07 is based on a linear ratio of the Design Basis 40-year CUF.
3. Estimated CUF for 40 years or 60-years based on the CUF as of 12/31/07 and the trends from the past twelve years (nine operating cycles) of actual plant operation.
4. CBF = Cycle-Based Fatigue ~~and SBF = Stress-Based Fatigue~~.
5. All locations with 40-year design basis CUF ratios (i.e., CUF/allowable) expected to exceed 0.4 (or 40% of allowable) based on the original analysis will be included in the program. In addition, the locations identified in NUREG/CR-6260 for the newer-vintage General Electric plant, which have been evaluated for environmental fatigue effects as discussed in Section 4.3.5, have been included in the program.
6. CUF value shown is for 60 years of operation based on updated fatigue analysis.
7. CUF is the sum of system cycling CUF plus rapid cycling CUF.
8. Estimated 60-year CUF exceeds the allowable value of 1.0. As discussed in Section 4.3.1, if the Metal Fatigue of Reactor Coolant Pressure Boundary program predicts the CUF will reach the allowable value prior to 60 years of operation, appropriate action will be taken in accordance with the corrective action process prior to the allowable limit being exceeded, including replacement of the main closure studs.

LRA Appendix A, Section A.3 (beginning on page A-34)

A.3.1.1 Metal Fatigue of Reactor Coolant Pressure Boundary

The Metal Fatigue of Reactor Coolant Pressure Boundary Program is an existing program that manages cumulative fatigue damage in the selected reactor coolant components subject to the reactor coolant and treated water environments.

The Metal Fatigue of Reactor Coolant Pressure Boundary Program is a preventive program that monitors and tracks the number of critical thermal and pressure transients to ensure that the cumulative usage factors for selected reactor coolant pressure boundary components remain less than 1.00 through the period of extended operation. The program determines the number of transients that occur and updates the 60-year projections as required on an annual basis. A software program, FatiguePro, computes cumulative usage factors for select locations.

The effect of the reactor coolant environment on fatigue usage, known as environmental fatigue, has been evaluated for the period of extended operation using the formulae contained in NUREG/CR-6583 for carbon and low-alloy steels and NUREG/CR-5704 for austenitic stainless steels. The fatigue usage associated with the effects of the reactor coolant environment will be included into the ongoing monitoring program.

The program requires the generation of a periodic fatigue monitoring report, including a listing of transient events, cycle summary event details, cumulative usage factors, a detailed fatigue analysis report, and a cycle projection report. If the fatigue usage for any location has had an unanticipated increase based on cycle accumulation trends or if the number of cycles is approaching their limit, the corrective action program is used to evaluate the condition and determine the corrective action. Acceptable corrective actions include repair of the component, replacement of the component, and a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded during the period of extended operation. Corrective actions include a review of additional affected reactor coolant pressure boundary locations.

There are several enhancements identified for this existing program as follows.

1. The Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to include additional transients beyond those defined in the Technical Specifications and the UFSAR, and expanding the fatigue monitoring program to encompass other components identified to have fatigue as an analyzed aging effect, which require monitoring.
2. The Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to use a software program to automatically count transients and calculate cumulative usage on select components. ***At this time only cycle based fatigue monitoring will be used. If stress based fatigue monitoring is used in the future, it will consider the six stress terms in accordance with the methodology from ASME Section III, Subsection NB, Subarticle NB-3200.***

3. The Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to address the effects of the reactor coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant identified in NUREG/CR-6260.

4. The Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to require a review of additional reactor coolant pressure boundary locations if the usage factor for one of the environmental fatigue sample locations approaches its design limit.

These enhancements will be implemented prior to the period of extended operation.

LRA Appendix B, (beginning on page B-223):

NUREG-1801 Consistency

The Metal Fatigue of Reactor Coolant Pressure Boundary program is consistent with the ten elements of aging management program X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary", specified in NUREG-1801.

Exceptions to NUREG-1801

None.

Enhancements

Prior to the period of extended operation, the following enhancements will be implemented in the following program elements:

1. The Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to include additional transients beyond those defined in the Technical Specifications and the UFSAR, and expanding the fatigue monitoring program to encompass other components identified to have fatigue as an analyzed aging effect, which require monitoring. **Program Elements Affected: Parameters Monitored or Inspected (Element 3) and Monitoring and Trending (Element 5)**

2. The Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to use a software program to automatically count transients and calculate cumulative usage on select components. ***At this time only cycle based fatigue monitoring will be used. If stress based fatigue monitoring is used in the future, it will consider the six stress terms in accordance with the methodology from ASME Section III, Subsection NB, Subarticle NB-3200.*** **Program Elements Affected: Scope of Program (Element 1), Preventive Actions (Element 2), Parameters Monitored or Inspected (Element 3), Monitoring and Trending (Element 5) and Acceptance Criteria (Element 6)**

3. The Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to address the effects of the reactor coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant identified in NUREG/CR-6260. **Program Elements Affected: Preventive Actions (Element 2), Parameters Monitored or Inspected (Element 3), Monitoring and Trending (Element 5) and Acceptance Criteria (Element 6)**

4. The Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to require a review of additional reactor coolant pressure boundary locations if the usage factor for one of the environmental fatigue sample locations approaches its design limit. **Program Elements Affected: Corrective Actions (Element 7)**

Operating Experience

Demonstration that the effects of aging are effectively managed is achieved through objective evidence that shows that aging effects and mechanisms are being adequately managed. The following examples of operating experience provide objective evidence that the Metal Fatigue of Reactor Coolant Pressure Boundary program will be effective in assuring that intended function(s) would be maintained consistent with the CLB for the period of extended operation:

1. Hope Creek experienced an Emergency Core Cooling System (ECCS) actuation on May 29, 2007. As a result, there was a High Pressure Coolant Injection (HPCI) injection through the Core Spray nozzle. The corrective action program was invoked to update the fatigue usage analysis for the Core Spray nozzle. The analysis consisted of the latest information from the Metal Fatigue of Reactor Coolant Pressure Boundary program, and confirmed that the cumulative usage factor was still less than 1.00 for the Core Spray nozzle. Therefore, this example provides objective evidence that the program's confirmation process was successfully used and that the program engineer was able to verify that the design basis was maintained for a reactor pressure boundary component.

2. In October 2004, Hope Creek experienced a High Pressure Coolant Injection (HPCI) event. As a result, the actual cumulative injection cycles exceeded the number of events that was assumed in the Core Spray (CS) nozzle fatigue analysis. The HPCI event is considered an Emergency Core Cooling System (ECCS) injection event, which was not previously monitored in the program. The corrective action program was used to evaluate the condition, resulting in an analysis indicating that the cumulative usage factor was 0.815, which is less than the design limit of 1.0. Another corrective action was implemented to update the cycle counting procedure to specifically include ECCS injections to ensure future ECCS injections are counted. The extent of condition indicated that this condition was an isolated case. Therefore, this example provides objective evidence that the existing Metal Fatigue of Reactor Coolant Pressure Boundary program is capable of evaluating conditions that have exceeded original design limits to ensure that the design basis of the reactor coolant boundary is maintained, and to take corrective action to prevent a recurrence of the condition.

3. The Metal Fatigue of Reactor Coolant Pressure Boundary program tracks component cycles and transients. The program has a corrective action process where if the Ratio of Lifetime Cycles (RLC40) exceeds 1.0 for any cycle limit, a SAP Notification is generated to evaluate the condition for future actions. The 2006 annual review of plant transients indicated that the Heatup and Cooldown transients will exceed the 40-year lifetime ratio if the current trend of transients continues. The evaluation of this condition indicated that there were a large number of heatups and cooldowns early in plant life. From 1986 through 1995 there were 5 transients per year. From 1996 through 2002 the trend was down to 1.6 per year. The current trend (2003 through 2006) is 3.5 transients per year. The 40-year life limit for both categories (heatups and cooldowns) is 120 cycles, or equivalent to an average of 3 transients of each category per year. As of 12/31/2007, the cycle count for these categories is 79, thus there is adequate

margin to prevent cracking due to fatigue. The corrective action for this condition is to continue to trend the transients in accordance with the program.

4. To support the TLAAs associated with metal fatigue of the reactor coolant system pressure boundary components, Hope Creek analyzed the projected cumulative usage factor, incorporating the environmental fatigue effects for the six (6) NUREG/CR-6260 locations; Reactor Vessel (CRD penetrations), Reactor Recirculation Piping (recirculation inlet and outlet nozzles), Reactor Vessel Feedwater Nozzle (nozzle safe end and corner), Core Spray Line Reactor Vessel Nozzle and associated Class 1 Piping (nozzle safe end and corner), Feedwater Line Class 1 Piping (tee), and the RHR Class 1 Piping (RHR supply and return piping). The detailed analyses found the cumulative usage factors, with the environmental factor added, had met the acceptance criteria of < 1.0 for all locations except for the reactor pressure vessel feedwater nozzle safe end. Hope Creek will implement a computer-based program, which will continually monitor plant data and provide current ~~stress-based and~~ cycle-based fatigue calculations for the six NUREG/CR-6260 locations to ensure that the RCS pressure boundary design basis is maintained. Corrective action will be taken prior to exceeding the environmental assisted fatigue CUF value of 1.0. Therefore, this example provides objective evidence that the Metal Fatigue of Reactor Coolant Pressure Boundary program completed the 60-year environmental fatigue assessments required for license renewal, a computer-based program will continue to monitor the fatigue usage at the select locations and that corrective actions will be taken prior to the environmental assisted fatigue cumulative usage factors exceeding the acceptance criteria of 1.0.

Problems identified would not cause significant impact to the safe operation of the plant, and adequate corrective actions were taken to prevent recurrence. There is sufficient confidence that the implementation of Metal Fatigue of Reactor Coolant Pressure Boundary program will effectively identify degradation prior to failure. Appropriate guidance for re-evaluation, repair, or replacement is provided for locations where degradation is found. Assessments of Metal Fatigue of Reactor Coolant Pressure Boundary program are performed to identify the areas that need improvement to maintain the quality performance of the program.

Conclusion

The enhanced Metal Fatigue of Reactor Coolant Pressure Boundary program will provide reasonable assurance that the cumulative fatigue damage aging effects will be adequately managed so that the intended functions of components within the scope of license renewal will be maintained consistent with the current licensing basis during the period of extended operation.

Enclosure B

Revised Commitment associated with Replacement Response to NRC RAI 4.3-01 Part 1, Metal Fatigue of the Reactor Coolant Pressure Boundary

Note: To provide clarity, added text is shown in ***Bold Italic*** *font.*

Update to LRA Appendix A, Section A.5, Commitment #46

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA APP. A)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
46	Metal Fatigue of the Reactor Coolant Pressure Boundary	Metal Fatigue of the Reactor Coolant Pressure Boundary is an existing program that will be enhanced to include: 1. Adding transients beyond those defined in the Technical Specifications and the UFSAR, and expanding the fatigue monitoring program to encompass other components identified to have fatigue as an analyzed aging effect, which require monitoring. 2. Using a software program to automatically count transients and calculate cumulative usage on select components. ***At this time only cycle based fatigue monitoring will be used. If stress based fatigue monitoring is used in the future, it will consider the six stress terms in accordance with the methodology from ASME Section III, Subsection NB, Subarticle NB-3200.*** 3. Addressing the effects of the reactor coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant identified in NUREG/CR-6260. 4. Requiring a review of additional reactor coolant pressure boundary locations if the usage factor for one of the environmental fatigue sample locations approaches its design limit.	A.3.1.1	Program to be enhanced prior to the period of extended operation.	Section B.3.1.1 ***Hope Creek Letter*** ***LR-N10-0356*** ***RAI 4.3-01***