February 16, 2012

Mr. Alex Javorik, Vice President
of Engineering
Columbia Generating Station
Energy Northwest
MD PE23
P.O. Box 968
Richland, WA 99352

SUBJECT: AUDIT REPORT ON THE METAL FATIGUE CALCULATIONS IN THE COLUMBIA GENERATING STATION, LICENSE RENEWAL APPLICATION (TAC NO. ME3058)

Dear Mr. Javorik:

By letter dated January 19, 2010, Energy Northwest (EN), submitted an application pursuant to Title 10 of the *Code of Federal Regulation,* Part 54 (10 CFR Part 54) for renewal of Operating License NPF-21 for Columbia Generating Station (Columbia). On November 29, 2011 to December 1, 2011, the NRC audit team completed an audit of the metal fatigue calculations for analyzing metal fatigue limiting component. The audit report is enclosed.

If you have any questions, please contact me by telephone at 301-415-3897 or by e-mail at Arthur.Cunanan@nrc.gov.

Sincerely,

/RA/

Arthur Cunanan, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosure:
As stated

cc w/encl: Listserv

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DISTRIBUTION: Letter to Energy Northwest from A.Cunanan, dated February 16, 2012

SUBJECT: AUDIT REPORT ON THE METAL FATIGUE CALCULATIONS IN THE COLUMBIA GENERATING STATION, LICENSE RENEWAL APPLICATION (TAC NO. ME3058)

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U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION –
DIVISION OF LICENSE RENEWAL

Metal Fatigue Audit Report

Docket No:	50-397
License No:	NPF-21
Licensee:	Energy Northwest
Facility:	Columbia Generating Station
Location:	Nuclear Energy Institute Building, Washington, DC
Dates:	November 29, 2011 – December 1, 2011
Reviewers:	A. Cunanan, Project Manager, Division of License Renewal (DLR), Office of Nuclear Reactor Regulation
	C. Ng, Mechanical Engineer, DLR
	O. Yee, Mechanical Engineer, DLR

Applicant representatives:
A. Mosta, Energy Northwest
S. Gosselin, Lucius Pitkin
J. Cole, Becht Engineering

Executive Summary

The applicant evaluated environmentally-assisted fatigue (EAF) for 60-year operation to assess the effects of the reactor coolant environment on additional locations that are potentially more limiting than those described in NUREG/CR-6260, “Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components.”

The staff noted that the applicant's plant-specific configuration may contain additional locations that may need to be analyzed for the effects of the reactor coolant environment other than those identified in NUREG/CR-6260. The applicant performed additional analyses and presented the results of the data in its response.

On November 28, 2011, to December 1, 2011, the staff conducted an audit of Columbia’s methodology for selecting additional locations to address the effects of reactor coolant environment on component fatigue life. The staff selected a sample set of components and piping systems to perform its audit of the applicant’s methodology. Based on its three day audit, the staff’s objectives were met.

Introduction

On January 19, 2010, Energy Northwest (EN) submitted a license renewal application (LRA) for the Columbia Generating Station (Columbia). Section 4.3.5 of the LRA describes the applicant’s evaluation of the effect of reactor water coolant environment on fatigue usage for the period of extended operation. The applicant evaluated EAF for 60-year operation to assess the effects of the reactor coolant environment on additional locations that are potentially more limiting than those described in NUREG/CR-6260, “Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components.”

In its original application, the applicant stated that the minimum set of components for a boiling water reactor (BWR) of its vintage is derived from NUREG/CR-6260 as follows:

- reactor vessel shell and lower head
- reactor vessel feedwater nozzle
- reactor recirculation cooling (RRC) piping (including inlet and outlet nozzles)
- core spray line reactor vessel nozzle and associated Class 1 piping
- residual heat removal (RHR) return line Class 1 piping
- feedwater line Class 1 piping

During its review of the EAF analyses for these six NUREG/CR-6260 component locations, the staff noted that the applicant’s analyses did not consider other reactor pressure vessel (RPV) or Class 1 piping components with high design basis cumulative usage factor (CUF) values for environmental effects. For example, the Control Rod Drive (CRD) tube and CRD housing were selected and the design basis CUF values for these component locations were 0.083 and 0.196, respectively. However, in LRA Table 4.3-3, the shroud support (0.399), main steam nozzle shell (0.47), or low pressure coolant injection (LPCI) thermal sleeve (0.430) all have existing design basis CUF values that are greater than those reported for the CRD tubes and CRD housings. By letter dated August 26, 2010, the staff issued a request for additional information (RAI) 4.3-06, asking the applicant to provide the basis for selecting RPV and Class 1 piping locations as the EAF analysis locations in the LRA. Also, the staff asked the applicant to justify its basis for not selecting core shroud supports, main steam shell nozzles, and LPCI nozzle thermal sleeves as additional EAF assessment locations.

In its response dated November 11, 2010, the applicant stated that, consistent with the GALL Report AMP X.M1, "Fatigue Monitoring," it addressed the effects of the coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components identified in NUREG/CR-6260. Section 4.1 of NUREG/CR-6260 states that, for both pressurized water reactor (PWR) and BWR plants, these components are not necessarily the locations with the highest design CUFs in the plant, but they were chosen to give a representative overview of components that had higher CUFs or were important from a risk perspective or both. The applicant added that it analyzed 14 site-specific locations that represent the six components identified in NUREG/CR-6260 for a BWR of Columbia's vintage, and these locations contain all the different materials used in the Columbia pressure vessel and attached piping. The applicant further stated that the main steam shell nozzle is exposed to dry steam. The environmental life correction factors apply to components exposed to reactor coolant and not to surfaces exposed to gaseous environments such as dry steam. Additionally, the applicant indicated that it did not evaluate the LPCI nozzle thermal sleeve because it was located on the thermal sleeve extension within the RPV nozzle, in a non-pressure boundary portion of the sleeve.

However, the staff still noted that the applicant's plant-specific configuration may contain additional locations (including, but not limited to, those provided in LRA Tables 4.3-3 and 4.3-5) that may need to be analyzed for the effects of the reactor coolant environment other than those identified in NUREG/CR-6260.

By letter dated February 3, 2011, the staff issued RAI 4.3-09, asking the applicant to confirm and justify that the locations selected for EAF analyses in LRA Table 4.3-6 consist of the most limiting locations for the plant (beyond the generic components identified in the NUREG/CR-6260 guidance). If these locations are not bounding, the staff asked the applicant to clarify the locations that require an EAF analysis and the actions that will be taken for these additional locations. If the identified limiting location consists of nickel alloy, the staff asked the applicant to state whether the methodology used to perform the EAF calculation for nickel alloy is consistent with NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," and, if not, to justify the method chosen.

In its response dated November 4, 2011, the applicant stated that the locations selected for EAF analyses in LRA Table 4.3-6 correspond to the locations in NUREG/CR-6260. The nickel-alloy components in Table 4.3-6 were calculated consistent with NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels." The applicant also stated that since the EAF locations selected for analyses in the LRA Table 4.3-6 do not necessarily contain the limiting locations for Columbia, the applicant performed additional analyses and presented the results of the data in its response.

In a conference call dated November 14, 2011, the staff discussed its need to verify the information presented in the applicant's RAI response. The applicant proposed making the information available for the NRC staff to audit the methodology and metal fatigue calculations submitted in its response.

On November 28, 2011, to December 1, 2011, the staff conducted an audit of Columbia's methodology for selecting additional locations to address the effects of reactor coolant environment on component fatigue life. The audit was conducted at the Nuclear Energy Institute Office in Washington, DC. The staff selected a sample set of components and piping systems to perform its audit of the applicant's methodology.

The objectives of the audit were:

1) To review the applicant's methodology for selecting additional plant-specific reactor pressure vessel locations and reactor pressure boundary piping and piping component locations.

2) To confirm the locations screened for review of the effects of reactor coolant on fatigue life were appropriate (non-wetted, non-pressure boundary, previously evaluated NUREG/CR-6260 locations, and locations that bound other similar locations).

3) To review the applicant's methodology for a sample set of EAF calculations.

Objective No. 1

The staff selected the following components from the reactor pressure vessel to perform its audit of the applicant's methodology: core spray nozzle, reactor feedwater nozzle, and reactor recirculation inlet and outlet nozzle. In addition, the staff selected the following reactor pressure boundary piping systems to perform its audit of the applicant's methodology: reactor water clean-up piping system, reactor feedwater piping system, reactor pressure vessel head spray piping, and standby liquid control piping system.

During its audit, the staff asked the applicant to clearly describe the methodology it used to select additional locations to address the effects of reactor coolant environment. The staff also asked the applicant to justify any assumptions that were made when applying this methodology for selecting or evaluating additional Class 1 components or Class 1 piping systems. This was identified as part of Audit Question #1. The applicant stated that the selection of additional EAF locations were based on identifying the highest CUF in air for all Class 1 piping systems connected to the RPV and all of the remaining RPV components (i.e., those not already addressed by NUREG/CR-6260). In making the selection of these additional EAF locations, the summary tables that listed all maximum usages for the systems tabulated in LRA Tables 4.3-3 and 4.3-5 were reviewed to help guide the selection of the locations to be evaluated. The applicant discussed its methodology related to identifying the additional locations for the RPV. In particular, LRA Table 4.3-3 lists all CUF values that were evaluated in the Columbia Generating Station RPV Stress Report. The staff noted that the list of locations were then screened to eliminate non-wetted locations such as nozzles exposed to dry steam or components that were not exposed to reactor coolant, such as the vessel skirt, RPV flange, and RPV studs. The applicant stated that the remaining population of locations included all materials used in RPV components subject to the reactor coolant environment; the applicant performed EAF evaluations for all of these remaining components.

For each reactor vessel component and reactor pressure boundary piping system the staff's review involved ensuring that the applicant considered all applicable material types, the effects of transients on the particular component/system configuration, and the appropriate water chemistry conditions.

Core Spray Nozzle

During its review of the applicant's vessel stress report, the staff confirmed that the core spray nozzle consists of a low-alloy steel forging, nickel alloy safe-end, and carbon steel safe-end extension. The staff noted that the nozzle consists of a thermal sleeve and it was not considered for the environmental effects of reactor coolant on fatigue life because it is not a pressure boundary component.

The applicant performed EAF calculations for the following locations: stress point No.19, which corresponds to the low-alloy steel forging; stress point No. 7, which corresponds to the nickel alloy safe-end; and stress point No. 1, which corresponds to the carbon steel safe-end extension. The staff reviewed the applicant's evaluation and confirmed that the applicant selected the stress points on the nozzle safe-end, forging, and safe-end extension that corresponded to the highest CUF for each location. The staff noted that it is reasonable to assume that the effects from transients and water chemistry are consistent at the different locations of the nozzle. Therefore, the staff finds it appropriate that the applicant selected the stress points in contact with reactor coolant with the highest cumulative usage factor. The staff confirmed that the applicant considered the effects of reactor coolant environment on fatigue life on all applicable material types for the core spray nozzle.

The staff noted that in the original vessel stress report, the transients that affect this component were bundled into "event group 1" and "event group 2." The staff reviewed the applicant's EAF evaluation for the nozzle safe-end and confirmed that the applicant used the same method and approach for calculating fatigue usage as the original vessel stress report. The only update to the cumulative usage factor calculation was incorporation of the 60-year projected cycles. The staff noted that the applicant did not remove any conservatism from the calculation because the method and approach used to calculate fatigue usage were not changed. The staff finds it reasonable that the applicant used the 60-year projected cycles to calculate the CUF because a more realistic value is obtained based on how the applicant's site has been operated.

The staff noted that for the nozzle safe-end and safe-end extension, the applicant assumed normal water chemistry (NWC) when calculating the environmental factor (F_{en}). The staff reviewed updated final safety analysis report (UFSAR) Section 6.3.2.2.1 and finds that it is appropriate to assume NWC because the inventory for this nozzle is provided from the condensate storage tanks and the suppression pool. The staff also noted that for the nozzle forging, the applicant assumed NWC and hydrogen water chemistry (HWC) based on its plant operation history. The staff finds it reasonable that the applicant considered the water chemistry in different locations of the nozzle.

Reactor Recirculation Inlet Nozzle

During its review of the applicant's vessel stress report the staff confirmed that the reactor recirculation inlet nozzle consists of a low-alloy steel forging and stainless steel safe-end. The staff noted that this nozzle consists of a thermal sleeve and it was not considered for the environmental effects of reactor coolant on fatigue life because it is not a pressure boundary component. The staff noted that the applicant did not take credit for the stainless steel cladding because it was less than 10 percent of the nozzle forging thickness, consistent with ASME Code III, Subsection NB-3122.

The applicant performed EAF calculations for the following locations: stress point No. 2, which corresponds to the low-alloy steel forging; and stress point No.13, which corresponds to the stainless steel safe-end. The staff reviewed the applicant's evaluation and confirmed that the applicant selected the stress points on the nozzle safe-end and forging that corresponded to the highest CUF for each location. The staff noted that it is reasonable to assume that the effects from transients and water chemistry are consistent at the different locations of the nozzle. Therefore, the staff finds it appropriate that the applicant selected the stress points in contact with reactor coolant with the highest cumulative usage factor. The staff confirmed that the applicant considered the effects of reactor coolant environment on fatigue life on all applicable material types for the reactor recirculation inlet nozzle.

The staff reviewed the applicant's EAF evaluation and confirmed that the applicant used the same method and approach for calculating fatigue usage as the original vessel stress report. The only update to the cumulative usage factor calculation was incorporation of the 60-year projected cycles. The staff noted that the applicant has not removed any conservatism from the calculation because the method and approach used to calculate fatigue usage were not changed. The staff finds it reasonable that the applicant used the 60-year projected cycles to calculate the CUF because a more realistic value is obtained based on how the applicant's site has been operated.

The staff noted that for the reactor recirculation inlet nozzle the applicant considered plant operation under NWC and HWC when calculating the F_{en} factor. The staff finds it appropriate that the applicant considered both NWC and HWC because the reactor coolant in contact with this nozzle would be consistent with the bulk reactor coolant water chemistry conditions, which has undergone both NWC and HWC. The staff reviewed the applicant's technical memorandum (TM-2162) on determining oxygen concentration and noted that the applicant HWC used data of the dissolved oxygen for the reactor and feedwater system when calculating the F_{en} factor. The staff noted that the applicant continues to monitor dissolved oxygen at its sample points with its BWR Water Chemistry Program. The staff finds it appropriate that the applicant considered NWC and HWC because the plant-specific data taken for dissolved oxygen was used.

Reactor Feedwater Nozzle

During its review of the applicant's vessel stress report, the staff confirmed that the reactor feedwater nozzle consists of a low-alloy steel forging and nickel alloy safe-end. The staff noted that the nozzle consists of a thermal sleeve and it was not considered for the environmental effects of reactor coolant on fatigue life because it is not a pressure boundary component.

The staff noted that in 1990, Energy Northwest revised the design transient definitions for the reactor feedwater piping, which is connected to the reactor feedwater nozzle. In addition, these changes resulted in transient definitions that better represented the anticipated nozzle temperature than those considered in the original vessel stress report. The staff also noted that the stress locations that were used in the original vessel stress report were not changed due to the good representation of the fatigue distribution which these locations provide. The staff finds it reasonable that the applicant used the revised design transient definitions for the reactor feedwater piping for the reactor feedwater nozzle because it better represents the actual conditions experienced by the nozzle during transients. The staff also noted that the applicant incorporated the 60-year projected cycles to calculate the CUF value. The staff finds it reasonable that the applicant used the 60-year projected cycles to calculate the CUF because a more realistic value is obtained based on how the applicant's site has been operated.

The applicant performed EAF calculations for the following locations: stress point No. 511 (stress location No. 34), which corresponds to the low-alloy steel forging; stress point No. 840 (stress location No. 8), which corresponds to the nickel alloy safe-end; and stress point No. 253 (stress location No. 36), which corresponds to the low-alloy steel nozzle to shell junction. The staff reviewed the applicant's evaluation and confirmed that the applicant selected the stress points on the nozzle safe-end, nozzle to shell junction, and forging that corresponded to the highest CUF for each material type and location. The staff noted that it is reasonable to assume that the effects from transients and water chemistry are consistent at the different locations of the nozzle. Therefore, the staff finds it appropriate that the applicant selected the stress points in contact with reactor coolant with the highest cumulative usage factor. The staff confirmed that the applicant considered the effects of reactor coolant environment on fatigue life on all applicable material types for the reactor feedwater nozzle.

The staff noted that for the reactor feedwater nozzle the applicant considered plant operation under NWC and HWC when calculating the F_{en} factor. The staff reviewed TM-2162, on determining oxygen concentration, and noted that the applicant used data of the dissolved oxygen for the feedwater system in the F_{en} factor. The staff noted that the applicant continues to monitor dissolved oxygen at its sample points with its BWR Water Chemistry Program. The staff finds it appropriate that the applicant considered NWC and HWC because the plant-specific data taken for dissolved oxygen was used.

Reactor Pressure Vessel Head Spray Piping System

During its review of the design specification for American Society of Mechanical Engineers (ASME) piping systems, the staff confirmed that the entire reactor pressure vessel head spray piping system is fabricated from carbon steel (ASME SA-106 Grade B). The staff also confirmed that within the boundaries of the piping system, as originally analyzed in the design specification, the analyzed transients are applicable to the entire piping system and these were the same transients that the applicant considered in its EAF calculations.

The applicant performed a fatigue calculation for node point No. 6, which corresponds to the piping location downstream of the reactor core isolation cooling (RCIC) head spray check valve (RCIC-V-66). The staff reviewed the ADLPIPE [digital computer program used for stress analysis of complex piping systems] results as documented in the applicant's calculation and confirmed that the highest cumulative usage factor was at node point No 6. The staff noted that it is reasonable to assume that the effects from transients and water chemistry are consistent at different locations of the piping system as originally analyzed in the design specification. Therefore, the staff finds it appropriate that the applicant selected the node point in contact with reactor coolant with the highest CUF.

The staff reviewed the applicant's evaluation for the reactor pressure vessel head spray piping system and confirmed the applicant used the same methods and approach for calculating usage as the original design. The only update to the cumulative usage factor calculation was incorporation of the 60-year projected cycles. The staff noted that the applicant has not removed any conservatism from the calculation because the method and approach used to calculate fatigue usage were not changed. The staff finds it reasonable that the applicant used the 60-year projected cycles to calculate the CUF because a more realistic value is obtained based on how the applicant's site has been operated.

The staff noted that for the reactor pressure vessel head spray piping system, the applicant assumed NWC when calculating the F_{en} factor. The staff reviewed UFSAR Section 5.4 and finds that it is appropriate to assume NWC because the inventory for the reactor pressure vessel head spray piping system is provided from the condensate storage tanks.

Reactor Water Clean-Up Piping System

During its review of the design specification for ASME piping systems, the staff confirmed that the entire reactor water clean-up piping system is fabricated from carbon steel (ASME SA-106 Grade B). The staff also confirmed that within the boundaries of the piping system, as originally analyzed in the design specification, the analyzed transients are applicable to the entire piping system and these were the same transients that the applicant considered in its EAF calculations.

The applicant performed an EAF calculation for node point No. 404 with a 40-year usage of 0.152, which corresponds to a 2" sockolet. The staff reviewed the results of the ADLPIPE results as documented in the applicant's calculation and noted that node points No. 4110 and No. 113 with 40-year usage of 0.278 and 0.230, respectively. The staff requested that the applicant clarify why node point No. 404 was selected instead of node point No. 4110 or No.113. The applicant explained that the node points were considered as part of the reactor recirculation piping system and are addressed in a separate calculation. The staff reviewed drawing No. M200-51 and confirmed that node No. 4100 and No.113 are included in the reactor recirculation piping system. The staff reviewed the ADLPIPE results as documented in the applicant's calculation and confirmed that node point No. 404 had the highest CUF for the reactor water clean-up piping system. The staff noted that it is reasonable to assume that the effects from transients and water chemistry are consistent at different locations of the piping system as originally analyzed in the design specification. Therefore, the staff finds it appropriate that the applicant selected the node point in contact with reactor coolant with the highest CUF. The staff reviewed the applicant's EAF evaluation for the reactor water clean-up piping system and confirmed the applicant used the same method and approach for calculating usage as the original design. The only update to the cumulative usage factor calculation was incorporation of the 60-year projected cycles. The staff noted that the applicant has not removed any conservatism from the calculation because the methods and approach used to calculate fatigue usage were not changed. The staff finds it reasonable that the applicant used the 60-year projected cycles to calculate the CUF because a more realistic value is obtained based on how the applicant's site has been operated.

The staff noted that for the reactor water clean-up piping system the applicant assumed NWC when calculating the F_{en} factor. The staff reviewed UFSAR Section 5.4 and finds that it is appropriate to assume NWC because the inventory for the reactor pressure vessel head spray piping system is provided from the condensate storage tanks.

The staff noted that for the reactor water clean-up piping system, the applicant considered plant operation under NWC and HWC when calculating the F_{en} factor. The staff finds it appropriate that the applicant considered both NWC and HWC because the reactor coolant in contact with this piping system would be consistent with the bulk reactor coolant water chemistry conditions, which has undergone both NWC and HWC. The staff reviewed TM-2162, on determining oxygen concentration, and noted that the applicant used data of the dissolved oxygen for the reactor and feedwater system when calculating the F_{en} factor. The staff noted that the applicant continues to monitor dissolved oxygen at its sample points with its BWR Water Chemistry Program.

Reactor Feedwater Piping System (Loop A)

During its review of the design specification for ASME piping systems, the staff confirmed that the entire reactor feedwater piping system is fabricated from carbon steel (ASME SA-106 Grade B). The staff noted that the boundary of this piping system, as analyzed in the design specification, is within containment and that portions outside of containment were analyzed together with reactor water cleanup system piping. The staff also confirmed that within the boundaries of the piping system, as originally analyzed in the design specification, the analyzed transients are applicable to the entire piping system and these were the same transients that the applicant considered in its EAF calculations.

The applicant only performed EAF calculation for loop A of the reactor feedwater piping system because loop A bounds loop B. During the staff's review, the staff requested the applicant to justify why loop A bounds loop B. This was identified as part of Audit Question #1. The applicant stated that for piping systems such as RRC and RFW that have multiple loops or trains with similar geometric configuration and materials, the maximum fatigue usage was only evaluated from one of the loops, because the thermal transients are the same for each loop or train for these systems; thus, evaluation of a bounding location on one loop would envelop the conditions of the other loops. Although the staff noted that there were differences in fatigue usage associated with pipe support and restraint locations between the two loops, it also finds the applicant's explanation acceptable as the applicant has considered the maximum usage between multiple loops or trains of the same system and the thermal transients considered between the loops were consistent with each other.

The applicant performed an EAF calculation for node point No. 47. The staff reviewed the ADLPIPE results as documented in the applicant calculation and confirmed that the highest CUF was at node point No. 47 for carbon steel piping. The staff noted that it is reasonable to assume that the effects from transients and water chemistry are consistent at different locations of the piping system as originally analyzed in the design specification. Therefore, the staff finds it appropriate that the applicant selected the node point in contact with reactor coolant with the highest cumulative usage factor. The staff also noted that the CUF values presented in the applicant's calculation and LRA Tables for the reactor feedwater piping include the effect of the power uprate in 1995.

The staff reviewed the applicant's evaluation for the reactor feedwater piping system and confirmed the applicant used the same methods and approach for calculating usage as the original design. The only update to the cumulative usage factor calculation was incorporation of the 60-year projected cycles. The staff noted that the applicant has not removed any conservatism from the calculation because the method and approach used to calculate fatigue usage were not changed. The staff finds it reasonable that the applicant used the 60-year projected cycles to calculate the CUF because a more realistic value is obtained based on how the applicant's site has been operated.

The staff noted that for the reactor feedwater piping system the applicant considered plant operation under NWC and HWC when calculating the F_{en} factor. The staff finds it appropriate that the applicant considered both NWC and HWC because the reactor water in contact with this piping system would be consistent with the bulk reactor coolant water chemistry conditions, which has undergone both NWC and HWC. The staff reviewed TM-2162, on determining oxygen concentration, and noted that the applicant used data of the dissolved oxygen for the reactor and feedwater system when calculating the F_{en} factor. The staff noted that the applicant continues to monitor dissolved oxygen at its sample points with its BWR Water Chemistry Program.

Standby Liquid Control Piping System

During its review of the design specification for ASME piping systems, the staff noted that the standby liquid control piping contains carbon steel and stainless steel segments.

The applicant performed EAF calculations for node point No. 51 with a 60-year CUF of 0.424, which corresponds to the carbon steel segment of the piping system. The staff reviewed the ADLPIPE results as documented in the applicant's calculation and confirmed that the highest CUF was at node point No. 51 for the carbon steel piping. The staff noted that it is reasonable to assume that the effects from transients and water chemistry are consistent at different locations of the piping system as originally analyzed in the design specification. Therefore, the staff finds it appropriate that the applicant selected the node point in contact with reactor coolant with the highest cumulative usage factor.

However, the staff noted that the stainless steel segment of the piping system was not evaluated for the effects of reactor coolant environment on fatigue life; therefore, the staff requested the applicant to clarify why the stainless steel segment was not evaluated. This was identified as Audit Question #8. The staff noted that the 60-year CUF for the stainless steel segment was at node point No. 25 with a usage of 0.054. The applicant determined that the CUF of dissimilar metal weld and the remaining stainless steel portion were sufficiently low that even with bounding F_{en} penalty on a 60-year life, the CUF_{en} would not be limiting. The staff finds the applicant's response to Audit Question #8 acceptable because even with a bounding F_{en} factor based on the applicant's site for the stainless steel portion, it will not be more limiting than the CUF_{en} of 0.737 for the carbon steel portion.

The staff noted that for all design transients that were considered as part of the standby liquid control piping system in the design specification for ASME piping systems, the temperature does not exceed 150^{o}F; therefore, based on the F_{en} equations from NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," which were used by the applicant, the environmental effects from strain rate, dissolved oxygen, and sulfur content are not applicable because the temperature threshold was not reached.

During its audit, the staff noted that the applicant considered the effects of reactor coolant environment on its plant-specific locations that correspond to NUREG/CR-6260 and additional critical plant-specific locations beyond NUREG/CR-6260. It is not clear to the staff if the applicant, during the period of extended operation, will ensure on an on-going basis that all critical locations in these ASME Class 1 components and piping systems have been evaluated for the effects of reactor coolant environment. This was identified as Audit Question #3. By letter dated December 16, 2011, the applicant amended the UFSAR supplement in LRA Section A.1.3.4 to state that "[f]or the period of extended operation, on an on-going basis, [the applicant would] ensure that all the limiting locations in Class 1 components and Class 1 systems have been evaluated for the effect of reactor coolant environment." The staff noted that regardless of any modifications or changes to the applicant's site that may occur in the future, the applicant's Fatigue Monitoring Program will ensure that for Class 1 components and piping systems, on an on-going basis during the period of extended operation, the effects of reactor coolant environment will be evaluated for the limiting locations. The staff finds the applicant's response to Audit Question #3 acceptable because for these EAF evaluations, that are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii), the applicant will continually ensure that the limiting locations for its site have been addressed for the effects of reactor coolant environment.

During its audit, the staff noted that the applicant did not update LRA Section B.2.24, A.1.2.24 and A.1.3.4 to indicate that additional locations had been evaluated to address the effects of reactor coolant environment on fatigue life and the staff asked the applicant why these updates were not made (Audit Question #5). By letter dated December 16, 2011, the applicant amended these LRA Sections to clarify that other limiting components beyond those locations identified in NUREG/CR-6260 had been evaluated for the effects of reactor coolant environment. The staff noted that those locations identified in LRA Tables 4.3-6 and 4.3-7 have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii), the effects of EAF will be adequately managed for the period of extended operation using the Fatigue Monitoring Program. The staff finds the applicant's response to Audit Question #5 acceptable because the applicant amended its LRA to clearly identify the disposition of these additional EAF evaluations, as required by 10 CFR 54.21(c)(1), and that the applicant is managing the effects of reactor coolant environment on component fatigue life with its Fatigue Monitoring Program by ensuring the assumptions in these evaluations remain valid during the period of extended operation on an on-going basis.

Objective No.2

The staff reviewed LRA Table 4.3-3 and confirmed that, unless the component was non-pressure boundary, non-wetted or previously evaluated as part of NUREG/CR-6260, the applicant considered the effects of reactor coolant environment for all applicable material types (e.g., SS, CS, LAS, NiCrFe) for ASME Class 1 components with an existing fatigue analysis (CUF value) in the original vessel stress report. As discussed above, for a sample set of components from LRA Table 4.3-3, the staff confirmed that the applicant considered the effects of reactor coolant environment for the highest usage locations for a particular component for all applicable material types. The staff reviewed LRA Table 4.3-5 and confirmed that all Class 1 piping and piping components were considered for the effects of reactor coolant environment, unless it was non-pressure boundary, non-wetted, or previously evaluated as part of NUREG/CR-6260. However, during its audit, the staff noted in the applicant's basis document for "TLAA-Metal Fatigue" that some ASME Class 1 valves were not addressed in LRA Table 4.3-5, specifically, the reactor feedwater valves and shutdown cooling valves. The staff requested the applicant clarify whether any of these should have been considered when addressing the effects of reactor coolant environment on fatigue life. This was identified as Audit Question #2. By letter dated December 16, 2011, the applicant stated that valves HPCS-V-51, LPCS-V-5, LPCS-V-51, RHR-V-112A and 112B are all evaluated in the same design report and all of these other valves were bounded when evaluating HPCS-V-51 for EAF. The staff noted that for the input parameters to the F_{en} calculation, the HPCS-V-51 valve also bounds the other valves. The staff finds it reasonable that if the evaluation in the original design report bounded all five valves by the evaluation of HPCS-V-51, than the EAF evaluation is also bounding for these same five valves. The staff noted that the applicant amended LRA

Table 4.3-5 to clarify that these five valves are represented by the “12-inch containment isolation valves” with a cumulative usage factor of 0.6599.

The applicant also stated that in a similar manner valves RHR-V-53A and 53B are bounded by the evaluation of HPCS-V-51 because they are similar material (carbon steel), have similar geometry (i.e., same size and pressure rating), and the transients for HPCS are more or equally severe compared to the RHR temperature change and pressure. The staff finds the applicant’s response to Audit Question #2 acceptable and it is reasonable that the EAF evaluation for the HPCS-V-51 valve bounds the RHR-V-53A and 53B valves because these two valves are also carbon steel that have the same size and pressure rating as the HPCS-V-51 valve. In addition, the transients that affect the RHR-V-53A and 53B valves are no more severe than those analyzed in the EAF evaluation for the HPCS-V-51 valve. The staff noted that for the input parameters to the F_{en} calculation, the HPCS-V-51 valve also bounds the other valves. The staff noted LRA Table 4.3-5 has been amended and that these RHR valves are also represented by the “12-inch containment isolation valves” with a cumulative usage factor of 0.6599.

During its audit, the staff noted that for the feedwater nozzle-shell junction the reported CUF value in LRA Table 4.3-3 was not consistent with the reported value in the original vessel stress report. The staff asked the applicant to clarify the discrepancy. This was identified as Audit Question #7. During its audit, the staff noted that the CUF for the feedwater nozzle-shell junction is 0.709 after the applicant’s power uprate, which is their current licensing basis. By letter dated December 16, 2011, the applicant amended LRA Table 4.3-3 to state that the CUF value for the feedwater nozzle-shell junction is 0.709. By letter dated January 4, 2012, the applicant clarified that the original value listed in the table for the nozzle-shell junction usage was taken from the original Chicago Bridge & Iron (CB&I) vessel stress report, with a value of 0.650. Since that time, General Electric issued a report in May 2009 that changed the usage value resulting from the 1995 Power Uprate to 0.709. The staff finds the applicant’s revision and response to Audit Question #7 acceptable.

The staff noted that the LRA Table 4.3-6, as amended by letter dated November 4, 2011, states that the revised CUF in air is 0.097. However, during its audit, the staff noted that the revised CUF in air was 0.210. The staff asked the applicant to clarify the discrepancy between the LRA and the applicant’s calculation. This was identified as Audit Question #6. By letter dated December 16, 2011, the applicant amended LRA Table 4.3-6 to identify the revised CUF in air for the RFW/RWCU tee as 0.210. By letter dated January 4, 2012, the applicant clarified that the original value of 0.097 represented a 40-year air usage value while the column required a 60-year value. Thus, the value of 0.097 is replaced with the recalculated 60-year air usage value of 0.210 and the 60-year CUF_{en} originally listed as 0.389 was changed to 0.4333.

The staff noted that the applicant’s 60-year projection of the Reactor Startup transient is 250 while the 40-year analyzed cycle for the Reactor Startup transient is 110. The staff finds the applicant’s revision and response to Audit Question #6 acceptable because the LRA has been revised to be consistent with the EAF calculation.

During its audit, the staff noted that LRA Table 4.3-2, as updated by letter dated July 16, 2011, provides the 60-year projected number of occurrences of design transients. Column 3 of LRA Table 4.3-2 indicates the actual cycles accumulated from December 1984 to February 2010. The values of these actual cycles were used as baselines to project the number of occurrences of the design transients to 60 years. However, it is not clear to the staff how the values of the actual cycles were determined. The staff asked the applicant in Audit Question #4 to clarify whether the occurrences of all transients listed in LRA Table 4.3-2 have been systemically accounted for since the plant operation in December 1984, and to identify the transients that have not been systemically accounted for since the plant operation in December 1984. By letter dated December 16, 2011, the applicant stated that plant cycle counting has been done since plant startup and that plant Technical Specification 5.5.5 has required counting of plant thermal cycles listed in UFSAR Table 3.9-1. The applicant clarified that this required cycle counting is performed once every year per its plant procedure, "Tracking of Fatigue Cycles." The applicant stated that the latest summary tabulation of plant cycles was updated on August 26, 2011, and the update includes all events/cycles that have occurred going back to initial plant start up. The staff finds the applicant's response to Audit Question #4 acceptable because the applicant has complete records of the number of transient events that have occurred at its site since initial plant startup, which provides an accurate gauge of the margin between the assumptions in its fatigue evaluations and the calculated CUF value.

The staff noted in LRA Table 4.3-7 for vessel head spray nozzle, the October 6, 2011, letter stated that the location is a "dry steam environment," however in the November 4, 2011, letter the applicant provided a CUF and CUF_{en} value. During its audit, the staff asked the applicant to clarify the revision (Audit Question #9). By letter dated December 16, 2011, the applicant amended LRA Table 4.3-7 to indicate that the vessel head spray nozzle is exposed to dry steam and is not subject to environmental effects on fatigue life. By letter dated January 4, 2012, the applicant clarified that credit was taken for the thermal sleeve and spray nozzle inserted into the vessel nozzle to direct RCIC spray onto the steam dryer and that the nozzle is at the top of the RPV head and is exposed to dry steam. During the audit, the staff reviewed UFSAR Figure 5.4-11 and confirmed that the vessel head spray nozzle is exposed to a dry steam environment; therefore, the nozzle is not subject to the effect of reactor coolant environment on fatigue life and the staff finds the applicant's revision and response to Audit Question #9 acceptable.

Objective No. 3

The staff reviewed the EAF calculations to verify the methods used for refining CUF values and the input parameters used to calculate F_{en} factors. The staff selected the following components to perform its audit of the applicant's EAF calculation methodology: core spray nozzle, RPV head spray check valve, HPCS/LPCS valve, reactor feedwater nozzle forging, reactor recirculation inlet nozzle, and reactor recirculation outlet nozzle.

For the core spray nozzle, the applicant performed an EAF evaluation. The applicant utilized the same design transient groupings as in its original design reports. In particular, the 60-year projection of the loss of feedwater pump/cold injection events was used as the bounding transient for event group 1 and seismic event during normal steady state plant operation is the bounding event for event group 2. The applicant used the F_{en} formulation in NUREG/CR-6583 for the carbon steel safe end extension and the low alloy steel nozzle forging, and the integrated F_{en} method to calculate F_{en}. The staff noted that F_{en} formulation in NUREG/CR-6583 is appropriate for the carbon steel and low-alloy steel locations. The staff also noted that the integrated F_{en} approach, which computes F_{en} value over the entire range of temperature, gives a more refined F_{en} value to account for the environmental effects of reactor coolant on component fatigue life. For the inconel safe end, the applicant used the F_{en} formulation in NUREG/CR-5704 and assumed maximum temperature and lower-bound strain rate. The staff confirmed that the applicant conservatively used the worst-case strain-rate and maximum temperature assumption as input to the formula to calculate F_{en} for the safe end.

The staff noted that for the core spray nozzle safe-end and safe-end extension, the applicant assumed NWC when calculating F_{en}. The staff reviewed UFSAR Section 6.3.2.2.1 and finds that it is appropriate to assume NWC because the inventory for this nozzle is provided from the condensate storage tanks and the suppression pool. For the nozzle forging, the applicant considered both NWC and HWC based on its plant operation in the F_{en} formulation. The staff finds it appropriate that the applicant considered both NWC and HWC because the reactor coolant in contact with this nozzle area would be consistent with the bulk reactor coolant water chemistry conditions, which has undergone both NWC and HWC.

For the RPV head spray check valve (RCIV-V-66), the applicant performed an updated EAF calculation for two locations, which are the crotches up-stream and down-stream of the valve gate. The staff confirmed that ASME Code Section III, Section NB-3500 focus on the crotch region of the valve. The applicant utilized 60-year projected cycles of a single bundled transient of the loss of feedwater pump as in the original design of the valve. The applicant also used NUREG/CR-6583 assumed maximum temperature and lower-bound strain rate for the calculation of F_{en}. The staff confirmed that the applicant conservatively used the worst case strain-rate and maximum temperature assumption as input to the formula to calculate F_{en} . The staff noted that F_{en} formulation in NUREG/CR-6583 is appropriate for the carbon steel valve. The staff noted that the applicant's assumption NWC when calculating the F_{en} is acceptable because the inventory for the reactor pressure vessel head spray piping system is provided from the condensate storage tanks.

For the HPCS/LPCS valves, the applicant performed an EAF evaluation for the HPCS/LPCS-V-51. The applicant utilized the same design transient as in its original design reports for the valves. Using the F_{en} formulation in NUREG/CR-6583, the applicant assumed the temperature for the transients is less than 150^{o}C, which means the dissolved oxygen level and strain-rate is irrelevant. The staff confirmed that, in the F_{en} formulation for carbon steel, the dissolved oxygen level and strain rate would not affect the calculation when the temperature is below 150^{o}C. The staff also noted that the applicant determined the F_{en} factor is 1.74 and CUF_{en} is 0.568. However, LRA Table 4.3-7 indicated that the CUF_{en} is 0.558. The staff requested the applicant to reconcile the difference between the two reported values. By letter dated January 4, 2012, the applicant clarified that the correct CUF_{en} is 0.568 and revised LRA Table 4.3-7. The staff finds the applicant's revision acceptable because the LRA has been revised to be consistent with the environmentally assisted calculation. The staff noted that F_{en} formulation in NUREG/CR-6583 is appropriate for the carbon steel valve.

For the reactor feedwater nozzle forging, the applicant performed an updated EAF calculation. The applicant used the formulation in NUREG/CR-6583 and the integrated F_{en} method to calculate F_{en}. The applicant considered 21 years of NWC and 39 years of HWC and used 60-year projected cycles. The staff finds it appropriate that the applicant considered both NWC and HWC because the reactor coolant in contact with this nozzle would be consistent with the bulk reactor coolant water chemistry conditions, which has undergone both NWC and HWC. The staff noted that F_{en} formulation in NUREG/CR-6583 is appropriate for the low-alloy steel valve. The staff also noted that the integrated F_{en} approach, which computes F_{en} value over the entire range of temperature, gives a more refined F_{en} value to account for the environmental effects of reactor coolant on component fatigue life.

For the reactor recirculation inlet nozzle, the applicant performed an EAF evaluation of the nozzle forging. The applicant used the formulation in NUREG/CR-6583 and utilized 60-year projected cycles of a single bundled transient as in the original design. For the reactor recirculation outlet nozzle, the applicant performed an EAF evaluation of the nozzle forging. The applicant used the formulation in NUREG/CR-5704 and utilized 60-year projected cycles of a single bundled transient as in the original design. The staff noted that the cladding thickness is less than 10 percent of the nozzle forging thickness and therefore would not be considered when satisfying cyclic stress requirements, consistent with ASME Code III, Subsection NB-3122. However, an EAF evaluation was performed because the applicant considered it bounding for all other cladding locations. For both nozzles, the applicant also considered 21 years of NWC and 39 years of HWC. The staff finds it appropriate that the applicant considered both NWC and HWC because the reactor coolant in contact with this nozzle would be consistent with the bulk reactor coolant water chemistry conditions, which has undergone both NWC and HWC. The staff noted that the applicant conservatively used the worst-case strain-rate assumption. The staff noted that the applicant appropriately used the F_{en} formulation in NUREG/CR-6583 for the carbon steel forging of the inlet nozzle and the F_{en} formulation in NUREG/CR-5704 for the stainless steel cladding of the outlet nozzle.

Conclusion:

Based on its three-day audit, the staff's objectives were met. Specifically,

1) The audit provided the staff reasonable assurance that the applicant's methodology for selecting additional reactor pressure vessel locations and reactor coolant pressure boundary piping and piping component locations to assess the impact of environmentally effect on fatigue was appropriate for plant-specific configuration.

2) The audit provided the staff reasonable assurance that the locations screened out for review of effects of reactor coolant environmental on fatigue life were appropriate.

3) The audit provided the staff reasonable assurance that the methodologies used to refine the cumulative usage factors and calculate the environmental correction factors were appropriate.

The NRC staff concluded that the objectives of the audit have been met and the audit was closed.

Title	Proprietary
Chicago Bridge and Iron Design Report, “Stress Report, 251, BWR Vessel, Hanford II Reactor, AED CAL 72-2647,” Revision 0, October 1976	No
Energy Northwest, “ASME Boiler and Pressure Vessel Code Section III Class 1 Piping Analysis of RFW(1)-4 Line ‘A’ (Inside Containment) Anchor Group 2,” Revision 3, August 1990	No
Energy Northwest, “Class 1 Stress Report, Reactor Feedwater Piping Inside Containment Line ‘B’ for Washington Public Power Supply System Nuclear Project No. 2,” Revision 3, August 1990	No
Energy Northwest, “Columbia License Renewal Project Document, TLAA – Metal Fatigue, LRPD-03,” Revision 3, April 2010	No
Energy Northwest, “Design Specification for Division 100 Section 1 ASME Piping Systems,” Rev. 17, January 2011	No
Energy Northwest, “Reactor Vessel Thermal Cycles Diagrams,” July 1979	No
Energy Northwest, “Stress Report 251” BWR Vessel Hanford II Reactor,” Revision 0, May 1976	No
Energy Northwest Columbia Generating Station Plant Procedure Manual, “Tracking of Fatigue Cycles, TSP-RPV-A101,” Revision1, April 2008.	Yes
Energy Northwest Interoffice Memorandum, “Tracking Columbia Fatigue Cycles,” August 26, 2011	Yes
Energy Northwest Manual Calculation (ME-02-09-17), Appendix A, “Evaluation of Environmental Fatigue Effects at the CRD Penetration Mechanism Housing and Stub Tube,” Revision 0, September 2009	Yes
Energy Northwest Manual Calculation (ME-02-09-17), Appendix B, “Evaluation of Environmental Fatigue Effects at the RRC Inlet Nozzle,” Revision 1, October 2009	Yes
Energy Northwest Manual Calculation (ME-02-09-17), Appendix C, “Evaluation of Environmental Fatigue Effects at the RRC Outlet Nozzle,” Revision 0, September 2009	Yes
Energy Northwest Manual Calculation (ME-02-09-17), Appendix H, “Evaluation of Environmental Fatigue Effects at the HPCS Nozzle,” Revision 0, October 2009	Yes
Energy Northwest Manual Calculation (ME-02-09-17), Appendix I, “Evaluation of Environmental Fatigue Effects for the Class 1 RRC Piping,” Revision 0, October 2009	Yes

Title	Proprietary
Energy Northwest Manual Calculation (ME-02-09-17), Appendix J, “Evaluation of Environmental Fatigue Effects at the Feedwater Nozzle,” Revision 0, October 2009	Yes
Energy Northwest Manual Calculation (ME-02-09-17), Appendix K, “Evaluation of Environmental Fatigue Effects for the Class 1 RWCU Piping,” Revision 1, October 2009	Yes
Energy Northwest Manual Calculation (ME-02-09-17), Appendix M, “Evaluation of Environmental Fatigue Effects at the HPCS Nozzle Safe End Extension and Nozzle Forging Locations,” Revision 3, October 2011	Yes
Energy Northwest Manual Calculation (ME-02-09-17), Appendix O, “Evaluation of Environmental Fatigue Effects at the RRC Inlet Nozzle Forging,” Revision 3.	Yes
Energy Northwest Manual Calculation (ME-02-09-17), Appendix P, “Evaluation of Environmental Fatigue at the RRC Nozzle Cladding,” Revision 3.	Yes
Energy Northwest Manual Calculation (ME-02-09-17), Appendix R, “Evaluation of Environmental Fatigue in Loop ‘A’ Reactor Feedwater Piping,” Revision 3, September 2011	Yes
Energy Northwest Manual Calculation (ME-02-09-17), Appendix S, “Evaluation of Environmental Fatigue Effects RWCU Piping,” Revision. 3, September 2011	Yes
Energy Northwest Manual Calculation (ME-02-09-17), Appendix T, “Evaluation of Environmental Fatigue Effects RCIC Head Spray Piping,” Revision 3, September 2011	Yes
Energy Northwest Manual Calculation (ME-02-09-17), Appendix V, “Evaluation of Environmental Fatigue Effects Standby Liquid Control Piping,” Revision 3, September 2011	Yes
Energy Northwest Technical Memorandum (TM-2162), “Determination of Average Oxygen Concentration in the Reactor and Feedwater for License Renewal,” Revision 0, December 2007	No
Energy Northwest Technical Memorandum (TM-2172), “Reactor Vessel Thermal Cycles for 60 Years,” Revision 0, July 2009	No
GE Drawing 76E120, “Reactor Vessel Thermal Cycle Curves,” CVI No. 02B13-04, 79, 1	Yes
GE Drawing 76E120, “Reactor Vessel Thermal Cycle Curves,” CVI No. 02B13-04, 79, 2	Yes
GE Nuclear Energy Report, “WNP-2 Recirculation System Loop ‘A’ Piping and Equipment Loads,” Revision 1, April 1988	No
GE Nuclear Energy Report, “WNP-2 Recirculation System Loop ‘A’ Piping and Equipment Loads,” Revision 2, April 1988	No
GE Nuclear Energy Report, “WNP-2 Recirculation System Loop ‘B’ Piping and Equipment Loads,” Revision 1, March 1993	No
NEDC 32153, Rev.1	Yes
Velan Engineering Design Report, DR-1041, May 1975	Yes