Alex L. Javorik
Columbia Generating Station
P.O. Box 968, PE04
Richland, WA 99352-0968
Ph. 509-377-8555 F. 509-377-2354
aljavorik@energy-northwest.com

December 16, 2011
GO2-11-202

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 RESPONSE TO NRC AUDIT QUESTIONS, LICENSE RENEWAL APPLICATION**

References:
1) Letter, GO2-10-11, dated January 19, 2010, WS Oxenford (Energy Northwest) to NRC, "License Renewal Application"

2) Letter dated February 3, 2011, NRC to SK Gambhir (Energy Northwest), "Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application, for Metal Fatigue (TAC NO, ME3058" (ADAMS Accession No. ML110240426)"

3) Letter, GO2-11-046, dated March 3, 2011, SK Gambhir (Energy Northwest) to NRC, "Response to Request for Additional Information, License Renewal Application"

4) Letter, GO2-11-177, dated November 4, 2011, BJ Sawatzke (Energy Northwest) to NRC, "Response to Request for Additional Information, License Renewal Application"

Dear Sir or Madam:

By Reference 1, Energy Northwest requested the renewal of the Columbia Generating Station (Columbia) operating license.

A request for additional information (RAI) was transmitted to Energy Northwest via Reference 2. Reference 3 provided the initial response to RAI 4.3-09. Reference 4 provided the final response to this RAI.

A143
NRR

The analyses results of additional locations for limiting cumulative usage factor (CUF) were provided to the NRC in References 3 and 4. Following the issuance of Reference 4, the NRC conducted an audit the week of November 28 – December 2, 2011, to review time limited aging analysis for metal fatigue calculations and other supporting documents. During the audit the NRC requested Energy Northwest to provide clarification on the selection criteria for other limiting locations. Responses to the NRC audit questions and request for clarifications are included in the attachment. Additionally, updates and corrections to the LRA and previous amendments are provided in Amendment 49.

No new or revised commitments are included in this letter.

If you have any questions or require additional information, please contact John Twomey at (509) 377-4678.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the date of this letter.

Respectfully,

AL Javorik For

AL Javorik
Vice President, Engineering

Attachment: Response to NRC Audit Questions

Enclosure: License Renewal Application Amendment 49

cc: NRC Region IV Administrator
NRC NRR Project Manager
NRC Senior Resident Inspector/988C
EFSEC Manager
RN Sherman – BPA/1399
WA Horin – Winston & Strawn
AD Cunanan - NRC NRR (w/a)
MA Galloway – NRC NRR
RR Cowley – WDOH

Supplemental Letter for the NRC Audit Questions/Clarifications on RAI 4.3-09

Selection Criteria for Additional Locations Beyond NUREG/CR-6260:

The selection of additional locations beyond NUREG/CR-6260 for evaluation of the impact of reactor coolant environment on the metal fatigue usage was based upon identification of the highest air usage locations for all of the Class 1 piping systems connected to the reactor pressure vessel (RPV) and all of the remaining RPV components. In making the selection of the additional locations, the summary tables that listed all maximum usages for the systems that were tabulated in LRA tables 4.3-3 and 4.3-5, were reviewed to help guide the selection of the locations to evaluate. The LRA tables collect the bounding usages for all of the systems. Thus, the first cut used the highest usage locations.

The individual design reports tabulate few high usage locations. The remaining evaluated locations are generally very low usages, less than 0.1, such that it is a relatively straight forward activity to identify controlling fatigue locations for a component in the applicable design report. Since large sections of piping systems are all affected by the same fluid flow conditions, the highest usage locations will normally occur at structural discontinuities such as branch connections, tee's, reducers, and tapered transitions. For systems with multiple loops such as reactor feedwater (RFW) loops A and B, the transients for both loops are the same. Thus, using the maximum usage from either loop was considered acceptable.

Reactor Pressure Vessel (RPV) Additional Locations

For the RPV, Table 4.3-3 lists all fatigue usage values that were evaluated in the Columbia Generating Station RPV Stress Report. (Note: design report and stress report are synonymous terms since ASME changed the term in the late 70's from stress report to design report.) When selecting additional locations for evaluation of environmental fatigue, all locations evaluated were selected for assessment to determine if they were more limiting than the NUREG/CR-6260 locations. The list of locations was 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. This left a population that included all materials used in components subjected to the environment. A further screening was done to eliminate non-pressure boundary components such as thermal sleeves that extend into the vessel from the nozzles. Environmental fatigue evaluations were completed for all of the remaining components using the design basis analyses as a starting point for the evaluation.

The RPV stress report design analysis evaluated fatigue for various portions of the vessel nozzles, e.g., safe end, safe end extensions, nozzle forging, thermal sleeves, etc. Thus, the locations for evaluation of environmental effects were taken directly from the design analysis. The transients for the vessel nozzles are influenced by the vessel

transients and the transients that occur within the attached piping. For a condition of flow into the vessel, the pipe transient will be consistent with the nozzle transient. In some nozzles, a thermal sleeve is installed to protect the nozzle and vessel material. Under those conditions, the portion of the nozzle protected by the thermal sleeve is subject to the vessel transient for the given event. If flow is out of the vessel through the nozzle, then the vessel transients apply uniformly to the nozzle components and to the attached piping.

In some cases, the design basis analysis used a nozzle evaluation to envelope a similar nozzle based upon conservatism. For example, the high pressure core spray (HPCS), and low pressure core spray (LPCS) nozzles are the same size, material, and configuration. These nozzles were addressed as one nozzle, core spray, in the RPV stress report. The HPCS nozzle has more transient cycles, the cycles are more extreme than the LPCS nozzle, and the HPCS nozzle has a greater range of temperature and pressure change than the LPCS. The stress report evaluated the HPCS transients and qualified the LPCS by comparison. Thus for environmental fatigue evaluation, the same approach was used.

Piping System Additional Locations:

The additional piping locations beyond NUREG/CR-6260 that were selected for evaluation are taken from LRA Table 4.3-5. The table lists the maximum usages for all of the Class 1 piping systems. Table 4.3-5 was developed from the tabulation of all system fatigue usages as part of license renewal project basis document that documented review of all Class 1 piping systems. This document includes all piping that connects to the RPV nozzles. A screening of these piping systems similar to the RPV nozzles was completed. Piping systems that carry dry steam, such as main steam, were eliminated from further evaluation.

For piping systems such as reactor recirculation cooling (RRC) and reactor feedwater (RFW) that have multiple loops of similar geometric configuration and materials, the maximum fatigue usage from only one of the loops was evaluated. For these systems, the thermal transients are the same for each loop, thus evaluation of a bounding location on one loop would envelop the conditions of the other loop.

The piping systems for boiling water reactors (BWR's), such as reactor core isolation cooling (RCIC), RFW, residual heat removal (RHR), reactor water clean-up (RWCU), HPCS, and LPCS, are primarily SA-106 Gr. B carbon steel. The BWR Class 1 piping that is stainless steel is primarily the RRC system, short segments of the RHR and RWCU systems that connect to the RRC, the standby liquid control system (SLC), small bore, and the reactor vessel level instrument condensing chambers. In addition small bore instrumentation piping is stainless steel, but uses the Class 1 exemption for design of 1 inch and under piping. The highest usage locations for piping include both carbon and stainless steel materials. When combined with the vessel nozzle evaluations, all materials used in pressure boundary components are evaluated for the effect of reactor coolant environment.

Piping system high usage locations are generally at fittings such as reducers, tapered transitions, tees and branch connections. These fittings with structural discontinuities were avoided when making dissimilar metal weld joints to keep fatigue usage low. Columbia used butt weld joints in straight pipe for dissimilar metal welds between carbon and stainless steel in the RRC to RHR, RWCU to RRC, and SLC to HPCS connections. These locations were screened out of environmental fatigue evaluation because the usage factors were extremely low.

The piping systems tabulated contained a mixture of systems that provide injection to the vessel or draw supply from the vessel. This provides a variety of thermal transient conditions that give slow and fast heat up and cool-down of piping systems. Thus all transients that are experienced by the pressure boundary components were evaluated for the impact on environmental fatigue. In systems with dissimilar metal welds mentioned above there was no need to evaluate both materials due to the low usages in the non-evaluated segment.

All Class 1 piping systems that are required to supply emergency core cooling system (ECCS) functions and normal operation were included in the environmental fatigue evaluation. (Note: ECCS systems are HPCS, LPCS, and RHR)

Multiple Material Considerations:

Several of the limiting locations selected for evaluation were part of a piping anchor group that had a dissimilar metal weld and thus a portion of the piping was another material. A question raised during the audit was that when a high usage location evaluated was one material (e.g., stainless steel), if another material such as carbon steel is included, would it have a higher environmental impact. There are several locations where there are dissimilar metal welds between carbon and stainless steel piping. These welds occur in straight runs of piping. As noted above, the highest usage location was evaluated for the piping system thermal transient conditions. For other portions of the piping, including the dissimilar metal welds, the usages were reviewed to determine if an environmental assessment should be done. Carbon to stainless steel interfaces occur in the following systems:

- RRC to RHR on RRC Loops A and B: The RRC Stainless Steel Usage was evaluated for environmental effect. The applicable Design Report reports were reviewed for carbon steel usage values that were not evaluated for environmental effects. All usages were sufficiently low that when projected for 60 years and using a bounding environmentally assisted fatigue correction factor (F_{en}) penalty, the environmental usages would not be limiting.

- SLC to HPCS: The SLC piping is stainless steel and transitions to carbon steel before it connects into the HPCS system. The limiting usage that was evaluated for environment was on the carbon steel portion of the piping system. The dissimilar metal weld and the remaining stainless steel portion were reviewed and usages were sufficiently low that even with bounding F_{en} penalty on a 60 year life, the environmental usage would not be limiting.

- RWCU to RRC: The RWCU to RRC piping dissimilar metal weld connections were reviewed. The limiting location for evaluation selected was carbon steel. The review of the stainless steel portion of the piping that was subject to the same transients showed that the stainless steel usage factors when projected to 60 years and using a conservative F_{en} would not be limiting.

In conclusion, for piping systems where interfaces between carbon and stainless steel materials occur the use of the maximum usage provided the limiting location for evaluation.

Thermal Cycle Counts:

Plant cycle counting at Columbia has been done since plant startup. Plant Technical Specification 5.5.5 has required counting of plant thermal cycles listed in FSAR Table 3.9-1. This required cycle counting is completed once per year per plant procedure "Tracking of Fatigue Cycles". The latest summary tabulation of plant cycles was updated August 26, 2011. The update includes all events/cycles that have occurred going back to initial plant start up.

Class 1 Valves:

Valves HPCS-V-51, LPCS-V-5, LPCS-V-51, RHR-V-112A and 112B are all evaluated in the same Design Report. In evaluating HPCS-V-51 for environmental fatigue effects all of the other valves were bounded. In a like 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 than for RHR temperature change and pressure. Thus, all seven valves are covered by the limiting evaluation done for HPCS-V-51.

The 40 year usage factors for the RFW and RWCU valves are added to table 4.3-5. The 60 year environmentally assisted usage factors for the RFW and RWCU valves will be provided in table 4.3-7 at a later date.

With the addition of RFW, RWCU, and RHR valves to table 4.3-7, all the other limiting piping and valve locations will now be included.

FSAR Supplement:

An FSAR supplement is provided in the enclosure to ensure that all the limiting locations in class 1 components and class 1 systems have been evaluated for the effect of reactor water environment during the period of extended operation.

LICENSE RENEWAL APPLICATION AMENDMENT 49

Section Number	Page Number	RAI Number
Table 4.3-3	4.3-6	4.3-09
Section 4.3.3	4.3-10	4.3-09
Table 4.3-5	4.3-11	4.3-09
Section 4.3.5.2	4.3-13	4.3-09
Section 4.3.5.2	4.3-14	4.3-09
Table 4.3-6	4.3-16	4.3-09
Table 4.3-7	4.3-16a	4.3-09
Table 4.3-7	4.3-16b	4.3-09
Section A.1.2.24	A-16	4.3-09
Section A.1.3.4	A-35	4.3-09
Section A.1.3.4	A-36	4.3-09
Section B.2.24	B-104	4.3-09
Section B.2.24	B-105	4.3-09

Table 4.3-3
Fatigue Usage for Reactor Vessel Locations

Location:	CUF of Record
Base plate	0.003
Core DP nozzle stub tube	0.125
Core spray nozzle forging	0.018
Core spray nozzle safe end	0.801
Core spray nozzle sleeve	0.005
Core spray nozzle stub	0.187
CRD housing	0.196
CRD return nozzle safe end	0.543
CRD return nozzle forging	0.330
CRD stub tube	0.083
Drain nozzle	NA
FW nozzle forging	0.000
FW nozzle safe end	0.696
FW nozzle thermal sleeve	0.013
FW nozzle-shell junction	0.709 ~~0.650~~
Instrument Nozzles (N12, N13, N14)	NA
Jet pump instrumentation nozzle (N9)	NA
MS nozzle forging	0.340
MS nozzle safe end	0.030

Location:	CUF of Record
MS nozzle shell	0.470
Refueling bellows support	0.453
RHR/LPCI nozzle forging	0.116
RHR/LPCI safe end	0.157
RHR/LPCI safe end ext.	0.189
RHR/LPCI thermal sleeve	0.430
RRC inlet nozzle forging	0.22
RRC inlet nozzle safe end	0.214
RRC inlet nozzle thermal sleeve	0.0013
RRC outlet nozzle clad	0.005
RRC outlet nozzle forging	0.24
RRC outlet nozzle safe end	0.005
Shroud support - Inconel	0.399
Shroud support – low-alloy steel	0.102
Stabilizer bracket	0.678
Steam dryer brackets	0.064
Support skirt	0.064
Top head flange	0.855
Vessel head spray nozzle	0.249
Vessel studs	0.985

FSAR Section 3.6.2 indicates that potential intermediate high energy line break locations can be eliminated based on CUFs being less than 0.1 if other stress criteria are also met. The usage factors, as calculated in the design fatigue analyses, account for the design transients assumed for the original 40-year life of the plant. Therefore, the determination of cumulative usage factors used in the selection of postulated high energy line break locations are TLAAs The Fatigue Monitoring Program will identify when the transients for piping systems are approaching their analyzed numbers of cycles. Prior to any transient exceeding its analyzed number of cycles for a piping system, the design calculations for that system will be reviewed to determine if any additional locations should be designated as postulated high energy line breaks, under the original criteria of FSAR Section 3.6. If other locations are determined to require consideration as postulated break locations, actions will be taken to address the new break locations.

During initial plant startup, an induction heating stress improvement (IHSI) process was used on various RPV nozzles to safe end and safe end to pipe welds. In the 1994 refueling outage, Columbia performed a mechanical stress improvement process (MSIP) for multiple RPV nozzles to safe end and safe end to pipe welds. No credit is taken for MSIP or IHSI in the calculation of CUFs for the Columbia vessel nozzles and safe ends.

All Class 1 piping was reviewed for the power uprate. The power uprate evaluation scaled existing fatigue analyses based on the changes in stress expected from the power uprate. This evaluation showed that there was adequate margin in each system to accommodate the power uprate (the increased CUF after the power uprate was approximated by the report). The maximum CUFs for Class 1 piping are shown in Table 4.3-5. The Fatigue Monitoring Program uses the systematic counting of plant transient cycles to ensure that component design fatigue usage limits are not exceeded. Design fatigue usage for 40 years of operation is provided in Table 4.3-5 for the limiting reactor coolant pressure boundary components.

A review of Columbia's documentation found several fatigue analyses for Class 1 valves that were TLAAs. The fatigue usage for those valves is based on transients that are tracked by the Fatigue Monitoring Program. The maximum CUFs for any Class 1 valves is 0.84 for the head spray inside containment check valve and 0.6599 for ~~five~~ 12 inch containment isolation valves. These CUFs are included in Table 4.3-5. seven

Metal fatigue for all Class 1 reactor coolant pressure boundary piping and in-line components (as listed in Table 4.3-5) is managed by the Fatigue Monitoring Program. The Fatigue Monitoring Program will identify when the transients for piping systems are approaching their analyzed numbers of cycles. Prior to any transient exceeding its analyzed number of cycles for a piping system, the design calculations for that system will be reviewed and appropriate actions will be taken.

Disposition: 10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended functions of the reactor coolant pressure boundary piping and components will be adequately managed for the period of extended operation by the Fatigue Monitoring Program.

Table 4.3-5
CUFs for Reactor Pressure Boundary Piping and Piping Components

System or Component	Max CUF
Reactor Feedwater Line A	0.250
Reactor Feedwater Line B	0.137
Reactor Feedwater / RWCU	0.588
Main Steam Line A	0.446
Main Steam Line B	0.7225
Main Steam Line C	0.222
Main Steam Line D	0.647
Main Steam Isolation Valves	0.0093
Reactor Recirculation Loop A	0.850
Reactor Recirculation Loop B	0.920
Reactor Recirculation Isolation Valves	0.0036
Reactor Water Cleanup	0.152
High Pressure Core Spray	0.237
Low Pressure Core Spray	0.145
Residual Heat Removal	0.001
Reactor Core Isolation Cooling	0.487
Reactor Vessel Head Spray	0.209
Reactor Vessel Head Vent to Main Steam	0.940
Reactor Vessel Level Instrument Lines and Condensing Pots	0.49
Standby Liquid Control System	0.262
Head spray check valve (RCIC) (7)	0.84
12 inch containment isolation valves ~~(5)~~ (HPCS, LPCS, RHR)	0.6599
24 inch containment isolation valves (8) (RFW)	0.637
6 inch isolation valves (3) (RWCU)	0.5183

4.3.5 Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping

4.3.5.1 Background

The NRC requires applicants for license renewal to address the reactor coolant environmental effects on fatigue of plant components (NUREG-1800 Section 4.3). The minimum set of components for a BWR of Columbia's vintage is derived from NUREG/CR-6260 (Reference 4.8-10), as follows:

1. Reactor vessel shell and lower head
2. Reactor vessel feedwater nozzle
3. Reactor recirculation piping (including inlet and outlet nozzles)
4. Core spray line reactor vessel nozzle and associated Class 1 piping
5. Residual heat removal return line Class 1 piping
6. Feedwater line Class 1 piping

In NUREG-1800, the NRC mentions using the calculational approach whereby the fatigue life adjustment factor (F_{en}) is determined for each fatigue-sensitive component and applying those environmental fatigue correction factors to the component CUFs to verify acceptability of the components for the period of extended operation. In NUREG-1800, the NRC further points out equations for calculating F_{en} values as being those contained in NUREG/CR-6583 (Reference 4.8-11) for carbon steel and low alloy steel components and in NUREG/CR-5704 (Reference 4.8-12) for austenitic stainless steel components. Nickel alloy components were also analyzed using the stainless steel equations in NUREG/CR-5704.

Environmentally assisted fatigue (EAF) evaluations are not applied during the current licensing basis. EAF evaluations done for the period of extended operation apply the EAF correction factors per NUREG-6260.

4.3.5.2 Columbia Evaluation

Using projected cycles from the Fatigue Monitoring Program and methodology accepted by the NRC, as noted above, the limiting locations (a total of 14 component locations corresponding to the six NUREG/CR-6260 components) for the material for each component location were evaluated. None of the 14 locations evaluated have an environmentally adjusted CUF of greater than 1.0 (see Table 4.3-6).

Values for dissolved oxygen, before and after the adoption of Hydrogen Water Chemistry (HWC), were used in the F_{en} determination. The plant operated with Normal Water Chemistry (NWC) for 20.9 years from January 19, 1984 (initial startup) until November 28, 2004. The plant has operated with HWC from November 28, 2004, and is assumed to continue operating with HWC until January 13, 2044; a combined time of

For other limiting locations beyond NUREG/CR-6260 component results are listed in Table 4.3-7. The F_{en} for nickel alloy components in table 4.3-7 were analyzed in accordance to NUREG/CR-6909.

39.1 years. The time Columbia has operated under both NWC (21 years) and HWC (39 years) conditions was considered in the estimation of an effective F_{en} based on a time weighted average of the HWC and NWC F_{en} values over 60 years of operation. The cumulative fatigue usage factor incorporating the effects of reactor coolant environment is obtained by multiplying the usage factor by F_{en}.

Original fatigue usage calculations were reviewed, and the transient groupings and load pairs used in those analyses were carried over to the EAF analyses. This ranged from a single transient grouping with a single load pair for the RRC inlet nozzle safe end to nearly a dozen load pairs and individual transients for the feedwater nozzle and RRC piping. For each load pair, a value of F_{en} was calculated. The environmentally adjusted usage factor for each load pair was then obtained by multiplying the usage factor by the F_{en} for that load pair. The environmentally adjusted cumulative usage factor for each location was obtained by summing the individual environmentally adjusted usage factors for each load pair.

The environmentally-adjusted CUFs for Columbia are shown in ~~Table 4.3-6~~ Tables 4.3-6 and 4.3-7. The minimum F_{en} for any load pair, the maximum F_{en} for any load pair, and an "average F_{en}" for each location is given. The average F_{en} is simply the final environmentally assisted CUF divided by the non-environmentally assisted CUF.

Columbia will manage the aging effect of fatigue for the period of extended operation, with consideration of the environmental effects using the Fatigue Monitoring Program in accordance with 10 CFR 54.21(c)(1)(iii).

Disposition: 10 CFR 54.21(c)(1)(iii) – The effects of environmentally-assisted fatigue will be adequately managed for the period of extended operation using the Fatigue Monitoring Program.

Table 4.3-6 (continued)
CUFs for NUREG/CR-6260 Locations

NUREG/CR-6260 generic locations		Columbia plant-specific locations	Material type	Revised CUF in air (2)	Per NUREG/CR-5704 and NUREG/CR-6583			
					Min. F_{en} (3)	Average F_{en} (3)	Max. F_{en} (3)	Environmentally assisted CUF
4	Core spray line reactor vessel nozzle and associated Class 1 piping	LPCS piping	CS	0.155	1.74	5.22	7.33	0.809
4	Core spray line reactor vessel nozzle and associated Class 1 piping	HPCS piping	CS	0.321	1.74	2.25	2.49	0.723
5	Residual Heat Removal (RHR) nozzles and associated Class 1 piping	RHR/LPCI nozzle safe end	Nickel Alloy	0.139	2.55	6.16	6.94	0.856
5	Residual Heat Removal (RHR) nozzles and associated Class 1 piping	RHR/LPCI nozzle safe end extension	CS	0.190	1.74	2.39	2.75	0.455
5	Residual Heat Removal (RHR) nozzles and associated Class 1 piping	RHR/LPCI piping	CS	0.001	20.49	20.49	20.49	0.02
6	Feedwater line Class 1 piping	RFW/RWCU Tee (1)	CS	~~0.210~~ 0.097 0.210	1.74	1.85	2.85	~~0.389~~ 0.4333

Note: CS is carbon steel, LAS is low alloy steel, SS is stainless steel

(1) Assumed NWC dissolved oxygen concentration equaled to 150 ppb for the RFW nozzle and RFW/RWCU Tee F_{en} calculation.

(2) ~~CUF of record previously identified in Table 4.3-3 and Table 4.3-5.~~

(3) Effective F_{en} determined for each load pair based on a time weighted average for HWC and NWC for 60 years of operation. Average F_{en} is the reported environmentally assisted CUF divided by the non-environmentally assisted CUF.

Replace footnote 2 with the following footnote:

(2) The "Revised CUF in air" is the maximum computed CUF (in air) for the wetted surface of interest for the evaluation of the effect of the reactor water environment. The CUF of record was previously identified in Table 4.3-3 and Table 4.3-5.

Insert A from pages 4.3-16a through 4.3-16b

Insert A:

Table 4.3-7
CUFs for components beyond NUREG/CR-6260 locations

LRA Table 4.3-3 or 4.3-5 Component	CGS Specific Location	Component Material	60-year U_{air}	Fen	Environmentally assisted CUF
Core DP Cell	Stub Tube	NiCrFe	0.218	2.259	0.494
~~HPCS~~ Core Spray Nozzle [3]	Forging	LAS	0.0008 0.0008	3.979 2.455	0.005
	Safe End Extension	CS	0.1043[1] 0.0063[1]	3.584 1.740	0.385
CRD Return Nozzle	Forging	LAS	0.093	3.565	0.330
	Safe End	CS	0.162	2.527	0.410
FW Nozzle	Forging	LAS	0.0398	Min[2] = 2.4 Max = 5.34	0.140
RHR/LPCI Nozzle	Forging	LAS	0.001	10.51	0.0103
RRC Inlet Nozzle	Forging	LAS	0.0351	4.363	0.153
RRC Outlet Nozzle	Cladding	SS	0.00487	12.902	0.063
Vessel Head Spray	Nozzle	LAS	~~0.0013~~	~~3.106~~	~~0.004~~
RFW Piping	Line A	CS	0.284	Min[2] = 1.0 Max = 1.897	0.385
RFW Piping	Line B	CS	Bounded by RFW Line A Calculation		

Dry steam environment - No environmental effects

[1] For event group 1 and 2
[2] Highest and lowest Fen for multiple load pairs
[3] Includes HPCS and LPCS

Table 4.3-7 (continued)
CUFs for components beyond NUREG/CR-6260 locations

LRA Table 4.3-3 or 4.3-5 Component	CGS Specific Location	Component Material	60-year U_{air}	Fen	Environmentally assisted CUF
RWCU	Piping	CS	0.164	Min[3] = 1.0 Max = 4.266	0.193
RCIC	Piping	CS	Dry steam environment – No environmental effects		
RPV Head Spray	Piping	CS	0.259	1.74	0.451
RPV Vent to MS	Piping	CS	Dry steam environment – No environmental effects		
RPV Level	Condensing Pot	SS	0.245	2.547	0.624
SLC	Piping	CS /SS	0.424[4]	Min[3] = 1.0 Max = 1.74	0.737
RPV Head Spray Zone 1	Check Valve	CS	0.386	2.439	0.941
Zone 2			0.331	2.503	0.828
HPCS/LPCS /RHR	Valve	CS	0.326	1.74	~~0.558~~ 0.568

[3] Highest and lowest Fen for multiple load pairs

[4] A portion of the SLC system is stainless steel. For evaluation of environment the carbon steel portion was assessed because its usage was over 5 times the maximum stainless steel usage, while the default maximum Fen for SS was only 1 1/2 times larger than CS. Thus the CS location was limiting.

A.1.2.24 Fatigue Monitoring Program

Fatigue evaluations for mechanical components are identified as TLAAs; therefore, the effects of fatigue have been addressed for license renewal.

Columbia monitors fatigue of various components (including ASME Class 1 reactor coolant pressure boundary, high energy line break locations, and Primary Containment) via the Fatigue Monitoring Program, which tracks transient cycles and calculates fatigue usage. Columbia has assessed the impact of the reactor coolant environment on the sample of critical components identified in NUREG/CR-6260. Calculation of fatigue usage values is not required for non-Class 1 SSCs. Instead, stress intensification factors and lower stress allowables are used to ensure components are adequately designed for fatigue. and other limiting components beyond those locations identified in NUREG/CR-6260

In accordance with 10 CFR 54.21(c)(1)(iii), the Fatigue Monitoring Program will be used to manage the effects of aging due to fatigue on the intended functions of the components associated with fatigue TLAAs for the period of extended operation.

The Fatigue Monitoring Program is an existing program that requires enhancement prior to the period of extended operation.

A.1.2.25 Fire Protection Program

The Fire Protection Program is an existing program, described in Appendix F of the FSAR, that detects degradation of components in the scope of license renewal that have fire barrier functions. Periodic visual inspections and functional tests are performed of fire dampers, fire barrier walls, ceilings and floors, fire-rated penetration seals, fire wraps, fire proofing, and fire doors to ensure that functionality and operability are maintained. In addition, the Fire Protection Program supplements the Fuel Oil Chemistry Program and External Surfaces Monitoring Program through performance monitoring of the diesel-driven fire pump fuel oil supply components and testing and inspection of the ~~halon suppression system~~, respectively. The Fire Protection Program is a condition monitoring program, comprised of tests and inspections based on National Fire Protection Association (NFPA) recommendations.

Insert A from Page A-16a

A.1.2.26 Fire Water Program

The Fire Water Program (sub-program of the overall Fire Protection Program) is described in Appendix F of the FSAR, and is credited with managing loss of material due to corrosion, erosion, macrofouling, and selective leaching, cracking due to SCC/IGA of susceptible water-based fire suppression components in the scope of license renewal. Periodic inspection and testing of the water-based fire suppression systems provides reasonable assurance that the systems will remain capable of performing their intended function. Periodic inspection and testing activities include hydrant and hose station inspections, fire main flushing, flow tests, and sprinkler

7,000 thermal cycles. The allowable stress range is reduced by the stress range reduction factor if the number of thermal cycles exceeds 7,000. If fewer than 7,000 cycles are expected through the period of extended operation, then the fatigue analysis (stress range reduction factor) of record will remain valid through the period of extended operation.

Because none of the non-Class 1 vessels, heat exchangers, storage tanks, or pumps were designed to ASME Section VIII, Division 2 or ASME Section III, Subsection NC-3200, no fatigue evaluation is required. Therefore, there are no fatigue TLAAs for these components.

The fatigue evaluation of non-Class 1 piping and in-line components evaluated the associated operating temperature against the threshold temperature value for fatigue of the material. If the threshold temperature value was exceeded, then the number of transient cycles for the piping or in-line component was projected. In each case, the number of projected cycles for 60 years was found to be less than 7,000 for piping and in-line components whose temperatures exceed threshold values. Therefore, fatigue for non-Class 1 piping and in-line components remains valid for the period of extended operation.

Disposition

The TLAA for non-Class 1 component fatigue analyses remains valid for the period of extended operation.

A.1.3.4 Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping

Applicants for license renewal are required to address the reactor coolant environmental effects on fatigue of plant components. The minimum set of components for a BWR of Columbia's vintage is derived from NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," as follows:

1. Reactor vessel shell and lower head
2. Reactor vessel feedwater nozzle
3. Reactor recirculation piping (including inlet and outlet nozzles)
4. Core spray line reactor vessel nozzle and associated Class 1 piping
5. Residual heat removal return line Class 1 piping
6. Feedwater line Class 1 piping

Columbia has analyzed these locations for the effects of the reactor coolant environment on fatigue in support of license renewal. Original fatigue usage calculations were reviewed, and the transient groupings and load pairs used in those

Columbia has also analyzed other limiting components beyond those locations identified in NUREG/CR-6260 for the effects of the reactor coolant environment.

analyses were carried over to the environmentally-assisted fatigue analyses, with revised non-environmentally assisted usage factors determined.

An effective fatigue life adjustment factor, F_{en}, that considers a time weighted average of operation with normal water chemistry and hydrogen water chemistry over 60 years of operation, was determined for each load pair analyzed for the components ~~at the NUREG/CR-6260 locations~~. The fatigue life adjustment factors were applied to the revised component load pair usage factors, and the environmentally-adjusted usage factors were summed to obtain environmentally-adjusted CUFs to verify acceptability of the components for the period of extended operation.

Using fatigue data projected by the Fatigue Monitoring Program and the methodology summarized above, the limiting locations ~~(a total of 14 locations corresponding to the six NUREG/CR-6260 components)~~ were evaluated. None of the ~~14~~ locations evaluated have an environmentally adjusted CUF of greater than 1.0 during the period of extended operation.

The aging effect of fatigue, including consideration of the environmental effects, will be adequately managed for the period of extended operation using the Fatigue Monitoring Program.

For the period of extended operation, on an ongoing basis, ensure that all the limiting locations in class 1 components and class 1 systems have been evaluated for the effect of reactor water environment.

Disposition

The effects of environmentally-assisted fatigue on the intended functions of the ~~limiting~~ NUREG/CR-6260 and other limiting locations will be adequately managed for the period of extended operation using the Fatigue Monitoring Program.

A.1.3.5 Environmental Qualification of Electrical Equipment

Environmental qualification analyses for electrical equipment are identified as TLAAs. NRC regulation 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," requires licensees to identify electrical equipment covered under this regulation and to maintain a qualification file demonstrating that the equipment is qualified for its application and will perform its safety function up to the end of its qualified life. The EQ Program implements the requirements of 10 CFR 50.49 and will be used to manage the effects of aging on the intended functions of the components associated with environmental qualification TLAAs for the period of extended operation.

Disposition

The effects of aging on the intended functions of the environmentally qualified components will be adequately managed for the period of extended operation by the EQ Program.

If the additional locations other than those identified in NUREG/CR-6260 consists of nickel alloy, the environmentally assisted fatigue calculation is consistent with NUREG/CR-6909.

B.2.24 Fatigue Monitoring Program

Program Description

The Fatigue Monitoring Program manages fatigue of the reactor pressure vessel by tracking thermal cycles as required by Technical Specification 5.5.5, "Component Cyclic or Transient Limit." The Fatigue Monitoring Program also manages fatigue of other components (including the ASME Class 1 reactor coolant pressure boundary, high energy line break locations, and Primary Containment) by tracking transient cycles. The Fatigue Monitoring Program is a combination of time-limited aging analyses (cumulative usage factor calculations) and transient counting procedures.

The Fatigue Monitoring Program uses the systematic counting of plant transient cycles to ensure that the numbers of analyzed cycles are not exceeded, thereby ensuring that component fatigue usage limits are not exceeded.

The BWR Vessel Internals Program contributes to managing fatigue of the jet pumps by checking the jet pump set screw gaps each outage. If any out of specification gaps are found, Columbia will calculate the additional fatigue accumulated by the jet pumps due to those gaps.

The Fatigue Monitoring Program acceptance criteria are to maintain the number of counted transient cycles below the analyzed number of cycles for each transient. The Columbia program periodically updates the cycle counts. When the accumulated cycles approach the analyzed design cycles, corrective action is required to ensure the analyzed number of cycles is not exceeded. Corrective action may include update of the fatigue usage calculation. Any re-analysis will use an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) to determine a valid CUF.

and other limiting components beyond those identified in NUREG/CR-6260

Columbia has assessed the impact of the reactor coolant environment on the sample of critical components identified in NUREG/CR-6260. These components were evaluated by applying environmental life correction factors to ASME Code fatigue analyses. Formulae for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low alloy steels and in NUREG/CR-5704 for austenitic stainless steel. The austenitic stainless steel formulae are also applied to nickel alloys. Columbia will enhance the Fatigue Monitoring Program to include the cycles analyzed for the effects of the reactor coolant environment on fatigue prior to the period of extended operation. The enhancement is explained in detail under *Required Enhancements* below.

NUREG-1801 Consistency

The Fatigue Monitoring Program is an existing Columbia program that, with enhancement, will be consistent with the 10 elements of an effective aging management

program as described in NUREG-1801, Section X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary."

Exceptions to NUREG-1801

None.

Required Enhancements

Columbia has also analyzed other limiting components beyond those locations identified in NUREG/CR-6260 for the effects of the reactor coolant environment.

Prior to the period of extended operation the enhancements listed below will be implemented in the identified program elements:

- **Preventive Actions, Monitoring and Trending, Acceptance Criteria –**

  Columbia has analyzed the effects of the reactor coolant environment on fatigue for the six locations recommended by NUREG\CR-6260. These analyses are based on the projected cycles for 60 years of operation (plus some conservatism) rather than the original design cycles in FSAR Table 3.9-1. The Fatigue Monitoring Program will be enhanced to ensure that action will be taken when the lowest number of analyzed cycles is approached.

- **Acceptance Criteria –**

  For each location that may exceed a cumulative usage factor (CUF) of 1.0 (due to projected cycles exceeding analyzed, or due to as-yet undiscovered industry issues), the Fatigue Monitoring Program will implement one or more of the following:

  (1) Refine the fatigue analyses to determine valid CUFs less than 1.0.

  This includes refining the analysis to increase accuracy and reduce conservatism. Any re-analysis will use an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) to determine a valid CUF less than 1.0.

  (2) Manage the effects of aging due to fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC).

  Should Columbia select the option to manage the aging effects due to fatigue, the inspection program will meet the following criteria: (1) the inspection program will be based on the 10 elements for an effective aging management program, as defined in NRC Branch Position RLSB-1, (2) the aging management program will be submitted for NRC review and approval