ENERGY NORTHWEST

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September 27, 2010
GO2-10-142

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

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION LICENSE RENEWAL APPLICATION**

References: 1) Letter, GO2-10-011, dated January 19, 2010, WS Oxenford (Energy Northwest) to NRC, "License Renewal Application"

2) Letter dated August 3, 2010, NRC to SK Gambhir (Energy Northwest), "Request for Additional Information for the Review of the Columbia Generating Station, License Renewal Application," (ADAMS Accession No. ML 102020129)

Dear Sir or Madam:

By Reference 1, Energy Northwest requested the renewal of the Columbia Generating Station (Columbia) operating license. Via Reference 2, the Nuclear Regulatory Commission (NRC) requested additional information related to the Energy Northwest submittal.

Transmitted herewith in the Attachment is the Energy Northwest response to the Request for Additional Information (RAI) contained in Reference 2. No new commitments are included in this response. Amendment 8 to the License Renewal Application is provided in the enclosure to this letter.

If you have any questions or require additional information, please contact Abbas Mostala at (509) 377-4197.

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NRR

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

Respectfully,

SK Gambhir
Vice President, Technical Services

Attachment: Response to Request for Additional Information

Enclosure: License Renewal Application Amendment 8

cc: NRC Region IV Administrator
NRC NRR Project Manager
NRC Senior Resident Inspector/988C
EJ Leeds - NRC NRR
EFSEC Manager
RN Sherman – BPA/1399
WA Horin – Winston & Strawn
EH Gettys - NRC NRR (w/a)
BE Holian - NRC NRR
RR Cowley – WDOH

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

RAI 3.3.2.3-2

Background:

The Generic Aging Lessons Learned (GALL) Report, under items III.B2-7 and III.B4-7, identifies that aluminum support members exposed to outdoor air can experience loss of material due to pitting and crevice corrosion and recommends GALL aging management program (AMP) XI.S6, "Structures Monitoring Program" to manage the effects of aging. The GALL Report, under item VII.G-8, also identifies that aluminum piping, piping components, and piping elements exposed to raw water can experience loss of material due to pitting and crevice corrosion, and recommends GALL AMP XI.M26. "Fire Protection Program" to manage the effects of aging.

In license renewal application (LRA) Tables 3.3.2-18 and 3.3.2-22, the applicant stated that aluminum alloy flame arrestors exposed to outdoor air (internal and external) have no aging effects requiring management and no AMP is proposed. The LRA cites generic Note G, indicating that the environment is not in the GALL Report for this component and material.

Issue:

It is unclear to the staff why the applicant identified that aluminum components exposed to outdoor air do not have any aging effects requiring management given that the GALL Report identifies the potential for loss of material for similar aluminum components.

Request:

Provide justification for why loss of material due to pitting and crevice corrosion for aluminum alloy flame arrestors exposed to outdoor air (internal and external) is not considered a significant aging effect requiring aging management during the extended period of operation, or provide an AMP to manage this aging effect.

Energy Northwest Response:

The outdoor environment at Columbia does not provide the elements that support the aging effect of loss of material due to pitting or crevice corrosion and no plant specific operating experience has been identified that invalidates this assumption. Therefore, no aging management of the aluminum flame arrestors is required. As noted in EPRI TR-1010639, "Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools Revision 4" (The "Mechanical Tools"), loss of material due to crevice and pitting corrosion in aluminum and aluminum alloy components is a concern in wetted locations of outdoor environments and in outdoor environments when plant operating experience has shown an aggressive environment such as salt air, sulfur dioxide and acid rain as found in marine (seashore) or industrial areas. As discussed in LRA section 3.5.2.2.2

and LRA Table 3.0-2 the outdoor environment at Columbia neither provides this aggressive environment nor the potential for a continuously wetted surface as the annual precipitation amounts to less than seven inches.

During the process of responding to this request, an error was identified in Table 3.3.2-18 (Line 16) on page 3.3-208 of the LRA. This line references plant specific note 0324. However, this note does not apply to this line in the table. Amendment 8 to the LRA is in the Enclosure to this letter.

RAI B.2.26-5

Background:

GALL AMP XI.M26, "Fire Water System," recommends that loss of material due to corrosion be managed by performing volumetric wall thickness evaluations, or as an alternative, visual inspections may be performed provided they are capable of detecting (1) wall thickness to ensure against catastrophic failure and (2) the inner diameter of the piping such that design flow is maintained. GALL AMP XI.M26 does not address management of loss of material due to erosion.

LRA Section 8.2.26, Fire Water Program, states that it manages loss of material due to corrosion, erosion, and macrofouling and that it includes periodic inspections and testing and will be enhanced to perform either ultrasonic testing or visual inspections of representative portions of above ground water suppression piping that are exposed to water but do not normally experience flow.

During the audit, the staff noted that condition report (CR) 2-05-01670, dated March 22, 2005, stated that ultrasonic testing of the 10-inch piping downstream of two valves in the fire water system used to throttle flow during annual fire pump performance testing showed significant internal pipe wall thinning at two separate locations due to cavitation erosion. The follow-up actions stated in the CR included periodic nondestructive evaluation (NDE) of the piping downstream from the two valves, and establishment of a data base to track and trend the wall thickness of piping downstream of the throttle valves in the fire protection system piping.

Issue:

The staff noted that loss of material due to erosion is potentially a much more aggressive aging effect than loss of material due to corrosion and therefore requires specific inspection and testing techniques and frequencies. The staff also noted that although the applicant's Fire Water Program includes activities capable of managing the aging effects of erosion due to cavitation (e.g., volumetric examinations of piping), there is no supporting information in the LRA regarding how cavitation erosion is being managed by the Fire Water Program (e.g., inspection technique and frequency). Without this information, it is unclear to the staff whether this plant specific loss of material aging effect is being adequately managed by the Fire Water Program.

Request:

1. Describe the follow-up corrective actions taken to mitigate cavitation erosion damage in the fire protection system piping addressed in CR 2-05-01670, including the NDE technique that is being used to manage cavitation erosion for those components and the basis for the inspection frequency.

2. If volumetric testing is not being performed, describe how wall thickness reference points are established.

3. Based on plant-specific operating experience for other systems within the scope of license renewal, describe where cavitation erosion has been identified and what programs are being used to manage cavitation erosion.

Energy Northwest Response:

1. The follow-up corrective actions taken to mitigate cavitation erosion damage in the fire protection system were to perform periodic ultrasonic testing (UT) examinations to manage cavitation erosion. A repetitive preventative maintenance task in the work management system ensures that UT examinations occur on the established frequency. The basis for the examination frequency is recorded in a corrective action plan as:

   The pipe downstream of FP-V-33 is an open ended pipe that is open to the Circulation Water Basin. This pipe has degraded approximately 0.130" since its installation circa 1985. The erosion rate is approximately 0.0065" per year and it is estimated that the pipe will reach minimum wall thickness in 2021. However, to prevent the pipe from introducing an industrial safety hazard or from falling into the Circulation Water Basin, this should continue to be monitored every five years via ultrasonic examination.

   The pipe downstream of FP-V-172 circulates back into the bladder tank (i.e., FP-TK-110). This pipe has degraded approximately 0.060" since its installation circa 1985. The erosion rate is approximately 0.003" per year. Based on this erosion rate, it is estimated that the date the pipe reaches the minimum wall thickness is 2064. The ultrasonic examination should continue at ten year intervals.

2. Volumetric testing is being performed on a periodic basis as described in response to question 1 above.

3. As a result of a pinhole leak on the weld downstream of SW-V-12B in 2002, a root cause and extent of condition was performed to determine susceptibility for cavitation erosion. A piping system cavitation guide was developed using the following guidance documents:

   a. Electric Power Research Institute (EPRI) guidance documents to predict cavitation and the evaluate the extent of damage;

b. Energy Northwest plant specific specifications and calculations; and

c. *Cavitation Erosion Model*; Chemical Engineers' Handbook, Fifth Edition; ISA-RP75.23-1995, *Considerations for Evaluating Control Valve Cavitation*, Approved 6/2/1995; NUREG/CR-6031.

The cavitation guide is a generic guide to aid Energy Northwest system engineers in identifying/predicting locations in their water system piping with potential for sustaining cavitation damage. Once potential locations of cavitation were identified, vibration or acoustic measurements and piping wall thickness measurements were to be taken to confirm the presence of cavitation and/or cavitation damage. Users of the system cavitation guide were directed to refer to the reference documents for guidance and additional information. This cavitation guide is now an appendix to the Columbia Pipe Sizing Guide.

The following table shows the systems evaluated for cavitation potential and the systems in the Flow Accelerated Corrosion (FAC) Program.

Extent of condition Systems	FAC Program Systems
CST	AS[1]
CW	BD
DW	BS[1]
FPC	CO
FP	COND
HPCS*	ES[1]
LPCS*	HD
RCC	HPCS*
RCIC*	HS
RHR*	HV[1]
RRC*	LPCS*
RWCU*	MD
SCW	MS[1]
SW	MSLC
TMU	MWR
TSW	RCIC*
	RFW
	RHR*
	RRC*
	RWCU*
	SS[1]

* indicates the system is on both lists. [1] Indicates a steam system

The root cause analysis concluded that "high energy systems are currently addressed by the FAC program." However, the FAC Program in effect at that time indicated that it does not specifically check for cavitation. Therefore, the assumption made in 2002 that the FAC Program would manage cavitation in high energy systems was incorrect. Energy Northwest does not intend to provide justification of the extent of condition review and acknowledges that the 2002 review was not complete. This current license issue has been entered into the Columbia corrective action process (CAP) in order that water systems that were not evaluated for cavitation after the 2002 through-wall leakage event because they were in the FAC program get a proper cavitation screening. Resolution of the CR will include an extent of condition review.

Locations that were mitigated with the installation of stainless steel pipe to resolve previously identified cavitation erosion issues should have the inspection frequency reinstated to ensure that the material change was an adequate design change to address the issue. Energy Northwest recognizes that the UT of the locations on the service water system should not have been discontinued after the design change to stainless steel. This issue has been entered into the Columbia CAP to reinstate an appropriate inspection frequency for locations that have been replaced with stainless steel and the periodic inspection was cancelled. The CR resolution will include an extent of condition review to look for any other cases where UT inspections for cavitation were retired and reevaluate the basis.

RAI B.2.42-3

Background:

The program description for the GALL Report AMP XI.M20 Open-Cycle Cooling Water System states that the program relies on implementation of the recommendations for U.S. Nuclear Regulatory Commission (NRC) Generic Letter 89-13, and includes surveillance and control techniques to manage aging effects caused by various mechanisms including erosion in the open-cycle cooling water system.

Issue:

The LRA states there have been repeated instances of leaks and failures related to cavitation erosion in the standby service water system, where design and operational adjustments had not fully precluded subsequent cavitation-related failures. The LRA basis document for operating experience indicates that an extent of condition review was performed to ensure that no other systems were affected by cavitation issues. Other LRA basis documents indicate that components susceptible to this aging mechanism will be monitored and that cavitation erosion was not evaluated for systems that are managed by the Flow Accelerated Corrosion (FAC) Program. However the FAC program specifically excludes consideration of cavitation erosion. In addition, although the Columbia Generating Station (Columbia) Inservice Inspection (ISI) Program includes consideration of Risk-Informed ISI based on EPRI Topical Report

TR-112657, which contains criteria for mechanism-specific examination volumes for erosion-cavitation, the criteria used in the extent of condition review were not intuitively comparable. It is unclear to the staff how the extent of condition for cavitation erosion was conducted in order to ensure all susceptible areas were evaluated.

In addition, the LRA basis documents indicate that cavitation erosion for some previously identified susceptible locations has been mitigated by using stainless steel and that these areas are no longer inspected for cavitation erosion. Although stainless steel is more resistant to cavitation damage than carbon steel, it is still susceptible to this degradation mechanism. It is unclear to the staff how it was determined that replacement of the piping with stainless steel will manage aging through the period of extended operation.

Request:

1) Describe the extent of condition review performed to determine the susceptibility of systems to erosion cavitation, including those systems being managed by the FAC Program.

2) Provide the basis for not needing to inspect, during the period of extended operation, the locations which were mitigated with stainless steel to resolve previously identified cavitation erosion issues.

Energy Northwest Response:

1. As a result of a pinhole leak on the weld downstream of SW-V-12B in 2002, a root cause and extent of condition was performed to determine susceptibility for cavitation erosion. A piping system cavitation guide was developed using the following guidance documents:

   a. Electric Power Research Institute (EPRI) guidance documents to predict cavitation and the evaluate the extent of damage;

   b. Energy Northwest plant specific specifications and calculations; and

   c. *Cavitation Erosion Model*; Chemical Engineers' Handbook, Fifth Edition; ISA-RP75.23-1995, *Considerations for Evaluating Control Valve Cavitation*, Approved 6/2/1995; NUREG/CR-6031.

   The cavitation guide is a generic guide to aid Energy Northwest system engineers in identifying/predicting locations in their water system piping with potential for sustaining cavitation damage. Once potential locations of cavitation were identified, vibration or acoustic measurements and piping wall thickness measurements were to be taken to confirm the presence of cavitation and/or cavitation damage. Users of the system cavitation guide were directed to refer to the reference documents for guidance and additional information. This cavitation guide is now an appendix to the Columbia Pipe Sizing Guide.

The following table shows the systems evaluated for cavitation potential and the systems in the Flow Accelerated Corrosion (FAC) Program.

Extent of condition Systems	FAC Program Systems
CST	AS[1]
CW	BD
DW	BS[1]
FPC	CO
FP	COND
HPCS*	ES[1]
LPCS*	HD
RCC	HPCS*
RCIC*	HS
RHR*	HV[1]
RRC*	LPCS*
RWCU*	MD
SCW	MS[1]
SW	MSLC
TMU	MWR
TSW	RCIC*
	RFW
	RHR*
	RRC*
	RWCU*
	SS[1]

* indicates the system is on both lists. [1] Indicates a steam system

The root cause analysis concluded that "high energy systems are currently addressed by the FAC program." However, the FAC Program in effect at that time indicated that it does not specifically check for cavitation. Therefore, the assumption made in 2002 that the FAC Program would manage cavitation in high energy systems was incorrect. Energy Northwest does not intend to provide justification of the extent of condition review and acknowledges that the 2002 review was not complete. This current license issue has been entered into the Columbia CAP in order that water systems that were not evaluated for cavitation after the 2002 through-wall leakage event because they were in the FAC program get a proper cavitation screening. Resolution of the CR will include an extent of condition review.

2. Locations that were mitigated with the installation of stainless steel pipe to resolve previously identified cavitation erosion issues should have the inspection frequency reinstated to ensure that the material change was an adequate design change to address the issue. Energy Northwest recognizes that the UT of the locations on the service water system should not have been discontinued after the design change to stainless steel. This issue has been

entered into the Columbia CAP to reinstate an appropriate inspection frequency for locations that have been replaced with stainless steel and the periodic inspection was cancelled. The CR resolution will include an extent of condition review to look for any other cases where UT inspections for cavitation were retired and reevaluate the basis.

RAI B.2.46 -Reactor Vessel (RV) Surveillance AMP

1. Please state when the last Columbia RV surveillance capsule or applicable Integrated Surveillance Program (ISP) capsule was pulled and tested in accordance with 10 CFR Part 50, Appendix H requirements. Provide a reference for this surveillance capsule test report.

2. LRA Section B.2.46 states that the Columbia RV Surveillance Program requires that untested capsules either be returned to the RV or maintained in storage for possible future re-insertion in the RV. LRA Section B.2.46 further states that "[a]s no Columbia capsules are scheduled for testing, the disposition of tested capsules is not applicable to Columbia." Please clarify the meaning of the statement quoted above, with respect to the "disposition of tested capsules."

Note: The NRC staff notes that Columbia has two standby RV surveillance capsules. If these capsules are pulled from the RV and remain untested they shall either be returned to the RV or maintained in storage for possible future re-insertion. The Columbia RV Surveillance Program must comply with this requirement.

Energy Northwest Response:

1. The last Columbia RV applicable ISP capsules pulled and tested in accordance with 10 CFR 50, Appendix H requirements are provided in the table below.

Columbia Target Vessel Material		ISP Representative Material	Last Capsule	ISP Capsule BWRVIP (EPRI) Report No.	Date Capsule Pulled	Capsule Report Date
Weld	5P6756	5P6756	SSP C	BWRVIP-169 (1013399)	March 2003	2007
Plate	C1272-1	B0673-1	SSP F	BWRVIP-111 (Revision 1) (1015001)	October 2000	2007

The last Columbia specific capsule was pulled in the Spring of 1996. The capsule analysis report, GE-NE-B1301809-01 dated March, 1997, was submitted to the NRC via letter GO2-97-077, dated 24 April 1997 from J. V. Parrish (EN) to the USNRC, "WNP-2, Operating License NPF-21, Submittal of 'WNP-2 RPV Surveillance

Materials Testing and Analysis' Report." (This letter transmitted GE-NE-B1301809-01, Rev 0, "Washington Public Power Supply System WNP-2 RPV Surveillance Materials Testing and Analysis," March 1997).

2. As described in LRA Appendix B.2.46, the Columbia Reactor Vessel Surveillance Program is part of the BWRVIP Integrated Surveillance Program (as described in BWRVIP-86-A and BWRVIP-116). The NRC approved the use of the BWRVIP ISP in place of a unique plant program for Columbia in letter dated April 28, 2005, NRC to J.V. Parrish (EN), "Columbia Generating Station -Issuance Of Amendment Re: Implementation Of The Boiling Water Reactor Vessel and Internal Project Reactor Pressure Vessel Integrated Surveillance Program To Address The Requirements of Appendix H To Title 10 Of The Code Of Federal Regulations Part 50 (Tac No. Mc4484)".

   The only Columbia RV surveillance capsules are those currently in place in the RV. The Columbia RV Surveillance Program requires these capsules to remain in place as deferred/standby capsules for possible future use by the BWRVIP ISP. The program further requires that the NRC is to be notified if surveillance capsules are removed from the RV and not reinstalled before reactor start-up. This is consistent with BWRVIP ISP guidance. The only Columbia RV surveillance capsule that has been pulled and tested (1996-97) had the tested irradiated materials reconstituted and reinstalled in the RV.

   The statement that disposition of tested capsules is not applicable at Columbia was in reference to NUREG-1801, Section XI.M31, Item 4, which states "All pulled and tested capsules, unless discarded before August 31, 2000, are placed in storage. (Note: These specimens are saved for future reconstitution use, in case the surveillance program is reestablished.)" As there are no capsules at Columbia scheduled for testing, storage of capsules post-testing is not applicable. Energy Northwest acknowledges that if standby capsules are scheduled for testing, then the tested capsules will be stored by the ISP for possible re-constitution in the future.

<u>RAI 4.2.1 -Neutron Fluence and Beltline Evaluation</u>

1. The ASME Code, Section XI, Appendix G, Paragraph G-2223, "Toughness Requirements for Nozzles," states that fracture toughness analysis to demonstrate protection against nonductile failure is not required for portions of nozzles or appurtenances having a thickness of 2.5 inches (in.) or less, provided the lowest service temperature is not lower than the adjusted RT_{NDT} (ART) plus 60 °F.

   a. Specify the lowest service temperature for the "N12" instrumentation nozzles.

   b. Confirm that all portions of the "N12" instrumentation nozzles have a thickness of less than 2.5 in.

Energy Northwest Response:

Per telephone conversation on 8/12/20010 with Evelyn Gettys and Chris Sydnor of the NRC and Abbas Mostala and Scott Wood of Energy Northwest, Evelyn reiterated that no response was to be made to RAI 4.2.1 that was received in the letter of August 3, 2010. MI102020129. The NRC will issue a new RAI 4.2.1.

RAI 4.2.2 -Upper Shelf Energy (USE) Evaluation

1. LRA Section 4.2.2, "Upper Shelf Energy Evaluation," includes an equivalent margin analysis (EMA) for RV Beltline Plate Heats C1337-1 and C1337-2 and RV Beltline Weld Heat 624039/D205A27A. The EMA calculations for these components are provided in LRA Tables 4.2-3 and 4.2-4 for the beltline plate and beltline weld, respectively. These tables also provide EMA data for several RV surveillance materials.

   State whether the EMA/USE data for the RV surveillance materials in LRA Table 4.2-3 and 4.2-4 was used for adjusting the EMA data for the corresponding beltline materials, in accordance with Boiling Water Reactor (BWR)VIP-74-A, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74)," Appendix B. and Regulatory Position (RP) 2.2 of Regulatory Guide (RG) 1.99, Revision (Rev.) 2.

2. Table 4.2-4 provides the results of the USE EMA for the limiting beltline weld (Heat 624039/D205A27A) at 54 EFPY. This table depicts two percentage decreases in the USE for this weld -a "RG 1.99 predicted decrease" of 13.2% and an "adjusted decrease" of 21.6%. Provide the following additional information concerning these values for the percentage decrease in the USE for this weld. Clarify which of these values represents the accurate value for the actual reported percentage USE decrease for this weld.

   a. Explain whether the "adjusted" USE decrease for this weld was calculated based on the use of BWR Integrated Surveillance Program (ISP) RV surveillance program data for this weld, in accordance with Regulatory Position (RP) 2.2, "Charpy Upper-Shelf Energy," of RG 1.99, Rev. 2.

Energy Northwest Response:

1. As stated in LRA Section 4.2.2, "the USE calculation of record for the existing licensed period (33.1 EFPY) is Appendix F of GE NEDO-33144 (Reference 4.8-5)." LRA Tables 4.2-3 and 4.2-4 are based on Tables F-1 and F-2 of NEDO-33144; which are in turn based on Tables B-4 and B-5 of Appendix B to BWRVIP-74A ("BWR Vessel and Internals Project BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines for License Renewal," EPRI Report 1008872, June 2003). The NRC SER for BWRVIP-74-A is in NRC letter from Christopher I. Grimes

to Mr. Carl Terry, "Acceptance For Referencing of EPRI Proprietary Report TR-113596, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74)" and Appendix A, "Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21)," 18 Oct 2001.

An additional surveillance data point, Supplemental Surveillance Program (SSP) capsule C is available from BWRVIP-169. This data point will be added to LRA Table 4.2-4 for completeness. This data point has less measured decrease in USE (it actually measured an increase in USE) than that predicted by RG 1.99. Consequently, this is not the limiting data point and does not affect the EMA projections in License Renewal Table 4.2-4. The revised LRA table 4.2-4 is included in LRA Amendment 8 provided in the enclosure to this letter.

For the limiting beltline plate USE (Heats C1337-1 and C1337-2), surveillance data is not available, as shown by the "Adjusted Decrease" being N/A in LRA Table 4.2-3. The percent decrease in USE is that predicted by RG 1.99, Rev. 2, Position 2.1 (Figure 2).

The USE decrease for the limiting beltline weld Heat 624039/D205A27A was adjusted based on the representative weld (5P6756) in the BWRVIP ISP surveillance program, as done in Appendix F of GE NEDO-33144. Therefore, the limiting weld heat was extrapolated by applying the bounding correction of the four noted 5P6756 capsules, as per RP 2.2 of RG 1.99, Rev.2. This extrapolation confirms that the limiting beltline weld Heat 624039/D205A27A meets the acceptance criteria in BWRVIP-74-A and is therefore bounded by the EMA.

2. The adjusted decrease is the RG 1.99 Rev. 2, Figure 2 projection adjusted for BWR Integrated Surveillance Program surveillance data for beltline weld heat 5P6756 applying RG 1.99 Rev. 2, Position 2.2, as described in Response 1 above. The adjusted decrease at 54 EFPY is 21.6% and is the number compared to the 39.0% acceptance criteria at the bottom of LRA Table 4.2-4.
   Ref: BWRVIP-169, "BWR Vessel and Internals Project Testing and Evaluation of BWR Supplemental Surveillance Program (SSP) Capsules A, B, and C" EPRI Report 1013399, March 2007

<u>RAI 4.2.3 -Adjusted Reference Temperature (ART) Analysis</u>

The following questions concern the applicant's application of surveillance data to the ART calculations in LRA Section 4.2.3.

1. Indicate which of the RV beltline material ART values from LRA Table 4.2-5 utilize chemistry factor (CF) values that are calculated based on the application of credible surveillance data from Columbia surveillance capsules or BWR ISP surveillance capsules, in accordance with Regulatory Position (RP) 2.1 of RG 1.99, Rev. 2.

Provide references for any surveillance capsule test reports that were used for determining CF values for the RV beltline materials. (There are no Columbia or other ISP surveillance capsule test reports referenced in LRA Section 4.8.) State which of the RV beltline material ART values utilize CF values that are calculated based on RP 1.1 from RG 1.99, Rev. 2 (the CF tables).

2. Note (2) in LRA Table 4.2-5 states that the "adjusted chemistry factor" for Lower-to-Lower Intermediate Shell Circumferential Weld Heat 5P6756/0342-3477 was determined per General Electric (GE) Report NEDO-33144, "Pressure-Temperature Curves for Energy Northwest Columbia," April 2004. Section 4.2.1.1, which was approved by the NRC in a safety evaluation report (SER) and updated per Columbia-specific ISP data.

   Clarify whether the CF value listed in LRA Table 4.2-5 for this weld heat (153.97 °F) is based on the application of credible surveillance data from Columbia or another applicable ISP plant in accordance with RP 2.1 from RG 1.99, Rev. 2. The staff notes that Tables 4-5b and 4-6b in GE Report NEDO-33144 list a CF value of 157.68 °F for this weld. Explain whether the discrepancy between the LRA CF value and the NEDO-33144 CF value for this weld heat is due to the application of Columbia-specific or other ISP surveillance data to the CF calculation subsequent to the issuance of the license amendment referenced in LRA Section 4.8 (Reference 4.8-2). Provide a reference for the surveillance data used for determining the CF value listed on LRA Table 4.2-5 (153.97 °F).

The following questions concern discrepancies between LRA Table 4.2-5 and GE NEDO-33144:

3. Table 4-3 of GE NEDO-33144 lists two initial RT_{NDT} data points for weld heat 5P6756/0342-3477, one for single wire and one for tandem wire. LRA Table 4.2-5 lists only a single data point for this weld heat. Clarify whether the single data point for this weld heat in LRA Table 4.2-5 is representative of both the single wire and tandem wire properties.

4. LRA Table 4.2-5 lists the standard deviation for the initial RT_{NDT} value, sigma-i, as 1.4 for the Residual Heat Removal/Low Pressure Coolant Injection (RHR/LPCI) N6 Nozzles. Tables 4-5a and 4-6a of GE NEDO-33144 list the sigma-i value as 0 °F for these nozzles. Explain this discrepancy.

Energy Northwest Response:

1. The ART value for Columbia RV beltline material weld heat 5P6756 is calculated utilizing a chemistry factor based on the application of credible surveillance data from the BWRVIP ISP in accordance with RG 1.99 Rev. 2, Position 2.1. The relevant surveillance capsule test report references are as follows:

Capsule	BWRVIP Report Number (EPRI Doc. No.)
River Bend 183°F	113 (1003345)
SSP Capsule F	111 (1015001)
SSP Capsule H	87 (1015000) and 128 (1010997)
SSP Capsule C	169 (1013399)

   All other ART values for Columbia RV beltline material, plate, nozzle and weld heats, are calculated based on CF values obtained from RG 1.99 Rev. 2, Position 1.1 (Chemistry Tables).

   Upon review, referencing of footnote 2 in LRA Table 4.2-5 is not correct. Footnote 2 is correctly attached to weld Heat/Lot 5P6756/0342-3477 on page 4.2-10 of the LRA and should not have been attached to plate heat number B5301-1. Weld 5P6756 is the only entry on Table 4.2-5 with a Chemistry Factor adjusted by surveillance data.

   A revised LRA Table 4.2-5 with footnote 2 removed from plate B5301-1 is provided in the enclosure to this letter as Amendment 8.

2. The adjusted CF of 153.97°F listed in LRA Table 4.2-5 for beltline weld heat 5P6756/0342-3477 is based on the application of credible surveillance data from applicable BWRVIP ISP capsule data in accordance with RP 2.1 from RG 1.99, Rev. 2. As stated in Note 2 of Table 4.2-5, the CF from NEDO-33144 has been updated by Energy Northwest based on BWRVIP ISP data applicable to Columbia that became available subsequent to the issuance of NEDO 33144. The BWRVIP ISP surveillance capsule test reports that include the data used for determining the new adjusted CF are BWRVIP-128 and 169 (see table in response to 1 above).

   Footnote 2 is modified on the revised LRA Table 4.2-5 provided as Amendment 8 in the enclosure to this letter to more accurately reflect the explanation provided with this RAI response.

3. The two data points (single wire and tandem wire) for weld heat 5P6756/0342-3477 on Table 4-3 of NEDO-33144 both have RT_{NDT} values of -50°F. Weld chemistry is not affected by the single or tandem wire process. The fluence bounds the entire weld. As such, both the single and tandem wire weld portions are represented by the same line entry in LRA Table 4.2-5.

4. This is a typographical error in the LRA. The correct annotation is zero (0). The margin value calculated in LRA Table 4.2-5 is the correctly calculated margin for a σ_i

of zero. If a σ_i of 1.4 were to be used in the margin calculation, the margin would be 21.186 rather than 21.095. This margin has been corrected in the Amendment 8 to the LRA in Table 4.2-5 provided as an enclosure to this letter.

RAI 4.2.5 -RV Circumferential Weld Inspection Relief

1. BWRVIP-74A, Section A.4.5, "Circumferential Weld Inspection Relief," states that in order to obtain relief from circumferential RV weld examination requirements, each licensee must submit a plant-specific relief request. In that submittal, licensees have to demonstrate that (1) at the expiration of the license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds specified in the July 28, 1998 SER for BWRVIP-05, and (2) the applicants have implemented operator training and established procedures that limit the frequency of cold overpressure events to the frequency specified in this SER. The LRA addressed condition (1) for this time-limited aging analysis (TLAA). However, the LRA did not address condition (2). Address condition (2), as it relates to the proposed extended period of operation.

2. The NRC staff requires that a request for relief from the RV circumferential shell weld examination requirements for the extended licensed operating period be submitted prior to the beginning of the extended period of operation. State whether Energy Northwest intends to apply for relief from the RV circumferential weld examination requirements for the extended licensed operating period. State when Energy Northwest plans to submit this relief request.

3. In the July 28, 1998 SER for BWRVIP-05, the NRC staff concluded that the failure frequency of RV circumferential shell welds in BWRs was sufficiently low to justify elimination of the ISI requirements for these welds, provided that certain conditions are met. However, the staff also indicated that examination of the RV circumferential shell welds would need to be performed if the corresponding volumetric examinations of the RV axial shell welds revealed the presence of an age-related degradation mechanism. Confirm whether or not previous volumetric examinations of the Columbia RV axial shell welds have shown any indication of cracking or other age-related degradation mechanisms in the unit's RV axial welds.

Energy Northwest Response:

1. Energy Northwest has implemented operator training and established procedures that limit the frequency of cold overpressure events. The training and procedures were explained in the original relief request (Reference 1) and accepted by the NRC in the SER approving this relief request (Reference 2). Energy Northwest will continue the operator training and procedures through the period of extended operation. Energy Northwest will submit a relief request in accordance with 10CFR50.12 to officially request relief from these inspections. That relief request will address both Condition (1) and Condition (2). Condition (2) was not addressed in the LRA because condition 2 is not a TLAA.

Ref 1. Energy Northwest Letter to USNRC Document Control Desk "Columbia Generating Station, Docket 50-397 Request for Permanent Relief From Inservice Inspection Requirements of 10 CFR 50.55a(g) for the Volumetric Examination of Circumferential Reactor Pressure Vessel Welds," Dated July 15, 2004. (GO2-04-125),

Ref 2. NRC to J.V. Parrish (EN), "Safety Evaluation for Columbia Generating Station – Relief Request for Alternatives to Volumetric Examination of Reactor Pressure Vessel Circumferential Shell Welds in Accordance with BWRVIP-05 (TAC No. MC3916)," Dated June 1, 2005. (GI2-05-090)

2. Energy Northwest intends to apply for relief from the reactor vessel circumferential weld examination requirements of 10 CFR 50. Relief will be requested for each 10-year interval of the Inservice Inspection Program. Relief requests are typically submitted 12 or more months prior to the beginning of the 10-year interval, along with the program plan for that interval. Relief requests must be approved (or the inspections must be performed) by the end of the 10-year interval to remain in compliance with 10 CFR 50.

   The first relief requests for the extended licensed operating period will be submitted with the first licensed operating period 10-year inspection interval program plan. This plan will be submitted at least 12 months prior to the end of the 4th inspection interval, December 2023.

3. Inservice Inspections of the Columbia reactor vessel have found no age-related degradation. See the response to RAI 4.7.1 for details of what indications have been found.

RAI 4.2.6 -RV Axial Weld Failure Probability

1. LRA Section 4.2.6 states that mean RT_{NDT} value for the limiting RV axial shell weld at the end of the extended period of operation (54 EFPY) is significantly less than the NRC limiting plant-specific mean RT_{NDT} value established in Table 1 of the staff's SER on BWRVIP-74A, and, therefore, the Columbia axial weld failure probability is well below the acceptable limit of 5×10^{-6} per reactor-year. However, the limiting axial weld failure probability calculated by the NRC is based on the assumption that "essentially 100 percent" (e.g. greater than 90 percent) examination coverage of all reactor vessel axial welds is achieved in accordance with ASME Code, Section XI requirements.

   State the extent of volumetric examination coverage obtained for the RV axial welds during the current 10-year interval ISI program at Columbia. If less than 90% examination coverage is obtained for the RV axial welds for the current 10-year ISI interval program, provide a reference for the NRC SE authoring relief for the reduced volumetric examinations of the RV axial welds. If less than 90 percent overall examination coverage is achieved for the RV axial welds, revise this TLAA to account for the effects of the limited scope examination coverage.

2. State whether the ISI examination of the RV axial welds covers all the intersections with the RV circumferential welds.

Energy Northwest Response:

No axial welds have been examined during the current inspection interval. They are scheduled to be examined during the last inspection period of the current interval. During the previous (2nd Inspection) interval, Energy Northwest inspected greater than 90% of each of the axial welds. Approximately 94% of the total length of the 14 axial welds (BA through BR) was inspected. The examination coverage for the current 10-year interval of the ISI Program will inspect greater than 90% of each of the axial welds. The total reactor vessel weld examination volumes were given in the response to RAI B.2.33-4, letter, GO2-10-117, dated August 19, 2010, SK Gambhir (Energy Northwest) to NRC, "Columbia Generating Station Docket No. 50-397 Response to Request for Additional Information License Renewal Application."

All the intersections of the axial welds and the circumferential welds are inspected. Approximately two to three percent of circumferential welds will continue to be examined at their points of intersection with the axial welds.

RAI 4.7.1-RV Shell Indications

1. LRA Section 4.7.1 discusses two indications (flaws) found in the RV shell. State: (1) whether the flaws were found in weld material, in plate material adjacent to welds, or in plate material away from any weld; (2) whether these flaws were found in or near the circumferential or axial welds; and (3) the Columbia RV weld designations (e.g.,

welds "BG," "BM," etc.) where the flaws were found.

2. Are the flaws discussed in LRA Section 4.7.1 subsurface flaws (completely embedded in the weld or plate metal) or are they surface-breaking flaws?

3. LRA Section 4.7.1 references a flaw evaluation report which documents an analytical evaluation of the flaws in accordance with IWB-3600. Please state whether this flaw evaluation found that the flaws were caused by service-induced aging degradation or whether the flaws were found to be fabrication defects.

   Note: Section 4.0 of the NRC staff's safety evaluation for the BWRVIP-05 report states that examination of the RV circumferential shell welds shall be performed if axial weld examinations reveal that an active mechanistic mode of degradation exists. The timing and scope of these examinations are to be proposed by the licensee and approved by the NRC. The applicant is expected to comply with this requirement.

4. LRA Section 4.7.1 states that these flaws were found during ISI examinations in 2005 and that the flaws were also identified during previous ISI examinations, but "became rejectable under current ASME Section XI, IWB-3610 requirements." Explain why these flaws did not become rejectable until this time, given that they were identified during previous ISI examinations.

5. LRA Section 4.7.1 states that the flaw evaluation used two time-limited assumptions based on the original 40-year life of the plant. The first assumption concerns the projected neutron fluence used in the flaw evaluation and is as stated in LRA Section 4.7.1:

   > The 1/4T neutron fluence at weld BG (5.11×10^{17} n/cm^2 at 33.1 EFPY) was used for both welds. This fluence was used to calculate the material properties of the cracked area, and hence the crack propagation. As can be seen from [LRA] Table 4.2-1, the projected 1/4T fluence for Weld BG at 54 EFPY is 8.10×10^{17} n/cm^2

   a. State why the flaw evaluation report referenced in LRA Section 4.7.1 did not utilize projected neutron fluence values that are valid for the end of the period of extended operation (54 EFPY).

   b. State why the flaw evaluation report referenced in LRA Section 4.7.1 did not utilize more conservative neutron fluence values at the RV inner diameter (ID) location for determining the limiting fracture toughness (K_{IC}) value, as opposed to neutron fluence values calculated at the *1/4T* location, which are normally used for RV pressure-temperature limits and upper shelf energy evaluations.

c. Explain why the 1/4T neutron fluence at weld BG was used for both welds, as stated in assumption (1) above.

6. The second time-limited assumption used in the flaw evaluation concerns projected transient cycles (from LRA Section 4.3) and assumed transient cycles used in the flaw evaluation for projecting flaw growth. This assumption is as stated in LRA Section 4.7.1:

   > 500 significant thermal transients were assumed (SRV [Safety Relief Valve] blowdown cycles being the worst case thermal cycle). From [LRA] Table 4.3-2, it can be seen that no SRV blowdown cycles are expected through 60 years of operation; furthermore, only 409 significant thermal transients are expected (233 heatup/cooldowns,166 scrams, and 10 HPCS [High Pressure Core Spray] actuations).

   Clarify whether the flaw evaluation report referenced in LRA Section 4.7.1 analyzed plant cycles for projecting flaw acceptability out to the end of the current 40-year licensed operating period (33.1 EFPY) or to the end of the period of extended operation (54 EFPY).

7. The Columbia site corrective action / condition reporting program should document the identification of the flaws discussed in LRA Section 4.7.1 and immediate corrective actions taken to address these flaws. The NRC staff identified a site condition report (CR), Columbia Action Request (AR) Number (No.) 00031237, dated August 5,2006, documenting an indication associated with RV axial weld "BM," that was determined to be unacceptable for continued service (without repair or evaluation under IWB-3600) per the ASME Code, Section XI, Table IWB-3510-1 acceptance criteria. This report states that "[t]he analytical evaluation path will be followed." The date of the flaw evaluation report submittal referenced in LRA Section 4.7.1 (September 15, 2005) precedes the date of the AR (August 5,2006).

   a. Please state whether the flaw documented in Columbia AR No. 00031237 is identical to one of the two flaws documented in LRA Section 4.7.1. If this report addresses another unacceptable flaw not discussed in LRA Section 4.7.1, please revise LRA Section 4.7.1 to include documentation of a TLAA for this flaw, and provide a reference for an IWB-3600 analytical evaluation for this flaw.

   b. If the flaw documented in Columbia AR No. 0031237 corresponds to one the flaws discussed in LRA Section 4.7.1, please explain why the date of the flaw evaluation report submittal (September 15, 2005) precedes the date of AR No. 00031237.

8. LRA Section 4.7.1 states that, "[t]his indication is currently scheduled for re-inspection in 2015. Columbia will re-evaluate the indication based on the results of the 2015 inspection and either project this analysis through the period of extended operation or continue augmented inspections as required by the ASME Code."

   a. Please clarify whether this statement only applies to just one of the flaws discussed in LRA Section 4.7.1 or to both flaws.

   b. The NRC staff requests the applicant add the above statement to the Columbia LRA Commitment Table, given that the flaw evaluation referenced in LRA Section 4.7.1 will apparently only remain valid through the end of the current licensed operating period (33.1 EFPY).

9. Were any other flaws discovered in the RV plates or welds that required screening in accordance with the ASME Code, Section XI, IWB-3500? If so, indicate whether any of these flaws (other than the flaws discussed in LRA Section 4.7.1) were found to be unacceptable for continued service under IWB-3500.

Energy Northwest Response:

1. The two indications are planar subsurface indications. The first is located in the base material adjacent to the beltline axial weld BG. The second is located in non-beltline axial weld BM.

2. The two indications are planar subsurface indications. Both indications are located approximately midpoint between the ID and outside diameter (OD) surfaces. The indication adjacent to the beltline vertical weld BG has a depth of 0.39 inch, a length of 3.0 inches, and a surface separation of 2.68 inches. The indication located in non-beltline weld BM has a depth of 0.38 inches, a length of 3.75 inches, and a surface separation of 2.78 inches.

3. Both indications are approximately in the middle of the wall thickness. Service induced flaws usually initiate on the ID or OD surface, not in the middle of the wall. A flaw evaluation was performed because the flaws were beyond code acceptable. The evaluation of these two indications concluded that they were due to fabrication and were not service induced.

4. Both indications (BG and BM) were identified in the Refueling Outage R8 (1993) examination. Under the ASME Section XI recording criteria in effect during the 1993 examinations the indications did not require further evaluation and were acceptable. When the welds were examined during the 2005 outage the ASME Section XI recording and evaluation criteria had changed. This change required recording and evaluation at a lower signal level. In addition to the Code change the UT and evaluation techniques had improved. The indications will be re-inspected in 2015, the third 10-year interval, to ASME Section XI 2001 edition and 2003 addenda.

5. a. The flaw evaluation report referenced in LRA Section 4.7.1 did not utilize projected neutron fluence values for 54 EFPY because at the time of the analysis (2005) the design lifetime of the plant was only 40 years.

   b. The flaw evaluation report referenced in LRA Section 4.7.1 utilized neutron fluence values at the 1/4T location because that is still a conservative estimate of the fluence where the indications were located. Weld BG is nominally 6.44 inches thick with the indication 3.37 inches from the ID, 0.39 inches deep, and 2.68 inches from the OD. Weld BM is nominally 6.56 inches thick with the indication 3.40 inches from the ID, 0.38 inches deep, and 2.78 inches from the OD. Both indications start at over ½T into the vessel wall.

   c. The 1/4T neutron fluence at weld BG was used for both welds because weld BG is in the beltline and thus will have much higher fluence than weld BM, which is above the beltline. Rather than perform an additional fluence analysis specific to weld BM, the bounding fluence associated with weld BG was used.

6. The flaw growth evaluation for the RPV shell indication is neither a "40-year" nor a "60-year" analysis as it is not based on either the 40-year design cycles nor the 60-year cycle projections. As stated in LRA Section 4.7.1, this evaluation analyzed 500 "significant" thermal cycles. Significant thermal cycles were considered to include safety relief valve lifts, heatups/cooldowns, scrams, and high pressure core spray actuations. As the most limiting transient was the safety/relief valve lift, 500 cycles of that transient were analyzed. At the time of the analysis (2005), this was expected to bound all transients that would be incurred for the life of the plant. Based on the 60-year projections done for license renewal, this number is still bounding for 60-years as only 409 significant thermal transients are expected (0 safety/relief valve actuations, 233 heatup/cooldowns,166 scrams, and 10 high pressure core spray actuations).

7. a. The flaw documented in Columbia AR 00031237 is one of the two flaws documented in LRA Section 4.7.1. CR 2-05-03679, PER 205-0348, and AR 00031237 are the same corrective action activity for the indication in weld BM.

   b. The date of the flaw evaluation report submittal (September 15, 2005) precedes the date of AR 00031237 because of a change in the CAP data base. During the move to the new software, all previous electronic CRs and PERs from the old software database were migrated to the new database and given a new AR number. Therefore, the date of AR 0031237 reflects the conversion date to the new database. AR 0031237 is the conversion of the original CR from 2005.

   In 2005 the process of entering items into the corrective action program involved writing a condition report (CR). In this case the CR was upgraded to a second document in the corrective action database called a problem evaluation report (PER). In late 2007 the electronic CR system was moved to a new software database and all previous electronic documents converted to the new database and given an AR number.

8. a. All axial welds are scheduled for re examination during the 2015 outage. This statement applies to both of the flaws discussed in LRA Section 4.7.1.

   b. Energy Northwest will re-examine all axial welds, including the portions of welds BG and BM with indications, as part of the NRC approved program plan for the current 10-year ISI interval. These inspections are required to be completed by 2015, well before the beginning of the PEO. These re-inspections are required for the currently licensed term of operation regardless of whether or not the license is extended. As such, it is not a license renewal commitment to repeat these inspections.

9. The two indications addressed in LRA section 4.7.1 are the only two shell welds that required screening per IWB-3500. These are the only unacceptable indications that have been discovered in the reactor vessel plates or welds at Columbia.

RAI 4.7.3-1

Background:

In LRA Section 4.7.3, the applicant states that Columbia has projected the erosion of the main steam flow restrictors for the period of extended operation. The restrictor is designed to limit coolant flow rate from the reactor vessel (before the MSIVs are closed) to less than 200 percent of normal flow in the event a main steam line break occurs outside the containment. It was further stated that the projections concludes that after 60 years of erosion, the chocked flow from the main steam flow restrictors will be less than 200 percent of normal flow in the event of a main steam line break outside of containment.

Issue:

The LRA does not contain information regarding the analysis that demonstrates that the chocked flow will remain less than the 200 percent of normal flow in the event of a main steam line break. Continued extended wear could cause erosion that may prevent the restrictor from continuing to perform its safety function during the period of extended operation.

Request:

Please provide the results of the analysis that demonstrates that the main steam flow restrictor will perform satisfactorily for the period of extended operation.

Energy Northwest Response:

The original Energy Northwest analysis was based on a conservative wear rate of 0.004 inches per year. This was not an empirically determined wear rate, but was basically the largest wear rate that gave acceptable results. The result was a steam flow rate of 198.8% at 40 years. Extrapolating this wear rate to 60 years gives a flow rate of 201.3%, slightly over the acceptable flow. To resolve this issue, Energy Northwest considered technical reports and operating experience to establish a more realistic, yet still bounding wear rate.

To determine a bounding wear rate, Energy Northwest considered the following:

Material and Environment: The environment of the main steam lines, at the location of the flow restrictors, is treated water in the form of steam with only 0.1% to 0.2% moisture (FSAR Amendment 59 section 5.4.4.3 "Safety Evaluation"). Virtually no water droplets exist in the steam in the main steam lines to cause erosion. The venturi inlet and throat material was chosen for its excellent performance in high velocity steam.

Columbia Operating Experience: Energy Northwest considered wall loss data from a carbon steel elbow upstream of the main steam line flow restrictors between Refueling outages 5 and 9 which was an average of 0.00091 in/year. Erosion was measured on the outside radius of the elbow. The change in flow direction of the 90° elbow causes some droplets to impact the elbow wall; conversely, the venturi throat is parallel to the flow direction so no impacts occur on the throat. Stainless steels have at least twice the erosion resistance of carbon steels so this offsets the need to double the wear rate seen on the elbow. Therefore, the wear rate on the throat diameter would not be expected to exceed 0.00091 in/year. The throat diameter determines the ability of the venturi to perform its safety function of limiting flow.

Industry Data: An EPRI publication regarding turbine steam path damage shows that the normalized erosion resistance of 300 series stainless steels is at least 2 times greater than that of carbon steel.

Industry Operating Experience: Inspection of the venturi at Quad Cities after 30 years of operation did not show any impact erosion on the entrance which is subject to liquid droplet impact due to the angle to the steam flow. The Quad Cities inspection came after 34 days of operation with a significant increase in moisture carryover due to a damaged steam dryer.

The Energy Northwest analysis concluded that the main steam line flow restrictor throat will experience very little if any erosion over the life of the plant. Energy Northwest determined that a wear rate of 0.003 in/yr (over three times the wear rate observed on carbon steel piping near the flow restrictor) was conservative. The flow rate analysis based on the 0.003 in/yr wear rate gives a 60-year maximum flow rate of 199.4% and is acceptable.

License Renewal Application
Amendment 8

Section No.	Page No.	RAI Response
Table 4.2-4	4.2-7	RAI 4.2.2
Table 4.2-4	4.2-7a	RAI 4.2.2
Table 4.2-5	4.2-9	RAI 4.2.3
Table 4.2-5	4.2-10	RAI 4.2.3
Table 3.3.2-18	3.3-208	RAI 3.3.2.3-2

Table 4.2-4
RPV Beltline Weld USE Equivalent Margin Analysis for 54 EFPY

Surveillance Weld USE (Heat 3P4966):	
% Cu =	0.03
Unirradiated USE =	98.0 ft-lb
1st Capsule Measured USE =	108.0 ft-lb
1st Capsule Fluence =	1.55E+17 n/cm^2
1st Capsule Measured Decrease =	-10.2 %
1st Capsule RG 1.99 Predicted Decrease =	6.0 %
ISP Surveillance Weld USE (Heat 5P6756):	
% Cu =	0.06
Unirradiated USE =	104.4 ft-lb
River Bend 183° Capsule Measured USE =	84.4 ft-lb
River Bend 183° Capsule Fluence =	1.16E+18 n/cm^2
SSP Capsule F Measured USE =	79.3 ft-lb
SSP Capsule F Fluence =	1.94E+18 n/cm^2
SSP Capsule H Measured USE =	84.6 ft-lb
SSP Capsule H Fluence =	1.36E+18 n/cm^2
Insert A from Page 4.2-7a	
River Bend 183° Capsule Measured Decrease =	19.2 %
River Bend 183° Capsule RG 1.99 Predicted Decrease =	12.5 %
SSP Capsule F Measured Decrease =	24.0 %
SSP Capsule F RG 1.99 Predicted Decrease =	14.0 %
SSP Capsule H Measured Decrease =	19.0 %
SSP Capsule H RG 1.99 Predicted Decrease =	13.0 %
Insert B from Page 4.2-7a	
Limiting Beltline Weld USE (Heat 624039/D205A27A):	
% Cu =	0.10
54 EFPY ¼T Fluence =	8.10E+17 n/cm^2
RG 1.99 Predicted Decrease =	13.2 %
Adjusted Decrease =	21.6 % (1)
21.6 % (54 EFPY) ≤	39.0 % (bounding value from SER for BWRVIP-74-A)
Therefore, the vessel welds are bounded by this Equivalent Margin Analysis.	

(1) The 54 EFPY adjusted decrease was evaluated for license renewal using the formulas for the curves in Figures 1 and 2 of RG 1.99, rather than by reading values off the curves. This resulted in a larger adjustment based on surveillance data than was used for the 33.1 EFPY projections.

Insert A to Page 4.2-7

SSP Capsule C Measured USE = 110.7 ft-lb
SSP Capsule C Fluence = 2.93E+17 n/cm_2

Insert B to Page 4.2-7

SSP Capsule C RG 1.99 Predicted Decrease = -6.0 %
SSP Capsule C RG 1.99 Predicted Decrease = 8.7 %

Table 4.2-5
ART Values for 54 EFPY

Sub-Component [1]	Heat or Heat/Lot [1]	% Cu	% Ni	Chemistry Factor	Initial RT_{NDT} °F	¼T Fluence n/cm^2	ΔRT_{NDT} °F	σ_1	σ_Δ	Margin °F	ART °F
PLATES:											
Lower Shell	C1272-1	0.15	0.60	110	28	2.71E+17	22.8	0	11.4	22.8	73.6
(Course #1)	C1273-1	0.14	0.60	100	20	2.71E+17	20.7	0	10.4	20.7	61.4
	C1273-2	0.14	0.60	100	4	2.71E+17	20.7	0	10.4	20.7	45.4
	C1272-2	0.15	0.60	110	0	2.71E+17	22.8	0	11.4	22.8	45.6
Lower-Intermediate Shell	B5301-1~~[2]~~	0.13	0.50	88	-20	8.10E+17	33.1	0	16.5	33.1	46.2
(Course #2)	C1336-1	0.13	0.50	88	-8	8.10E+17	33.1	0	16.5	33.1	58.2
	C1337-1	0.15	0.51	105	-20	8.10E+17	39.5	0	17.0	34.0	53.5
	C1337-2	0.15	0.51	105	-20	8.10E+17	39.5	0	17.0	34.0	53.5
NOZZLES:											
N6 (RHR / LPCI)	Q2Q55W 790S	0.11	0.76	76.4	-20	4.48E+17	21.1	~~1.4~~ 0	10.5	21.1	22.2

Table 4.2-5 (continued)
ART Values for 54 EFPY

Sub-Component [(1)]	Heat or Heat/Lot [(1)]	% Cu	% Ni	Chemistry Factor	Initial RT_{NDT} °F	¼T Fluence n/cm^2	ΔRT_{NDT} °F	σ_1	σ_Δ	Margin °F	ART °F
WELDS											
Lower Vertical (Axial/Longitudinal)	04P046 / D217A27A	0.06	0.9	82	-48	2.71E+17	17.0	0	8.5	17.0	-14.0
	07L669 / K004A27A	0.03	1.02	41	-50	2.71E+17	8.5	0	4.2	8.5	-33.0
	3P4966/ 1214-3482 (S) 3P4966 / 1214-3482 (T)	0.025	0.913	34	-30 -48	2.71E+17	7.0	0	3.5	7.0	-15.9 -33.9
	C3L46C / J020A27A	0.02	0.87	27	-20	2.71E+17	5.6	0	2.8	5.6	-8.8
	08M365 / G128A27A	0.02	1.10	27	-48	2.71E+17	5.6	0	2.8	5.6	-36.8
	09L853 / A111A27A	0.03	0.86	41	-50	2.71E+17	8.5	0	4.2	8.5	-33.0
Lower-Intermediate Vertical (Axial/Longitudinal)	3P4966 / 1214-3481 (S) 3P4966 / 1214-3481 (T)	0.025	0.913	34	-20 -6	8.10E+17	12.8	0	6.4	12.8	5.6 19.7
	04P046 / D217A27A	0.06	0.90	82	-48	8.10E+17	30.8	0	15.4	30.8	13.7
	05P018 / D211A27A	0.09	0.90	122	-38	8.10E+17	45.9	0	22.9	45.9	53.8
	624063 / C228A27A	0.03	1.00	41	-50	8.10E+17	15.4	0	7.7	15.4	-19.2
	624039 / D224A27A	0.07	1.01	95	-36	8.10E+17	35.7	0	17.9	35.7	35.5
	624039 / D205A27A	0.10	0.92	134	-50	8.10E+17	50.4	0	25.2	50.4	50.8
Lower to Lower-Intermediate Girth (Circumferential)	492L4871 / A422B27AF	0.03	0.98	41	-50	3.30E+17	9.5	0	4.8	9.5	-31.0
	04T931 / A423B27AG	0.03	1.00	41	-50	3.30E+17	9.5	0	4.8	9.5	-31.0
	5P6756 / 0342-3477	0.08	0.936	153.97[(2)]	-50	3.30E+17	35.7	0	17.9	35.7	21.4
	3P4955 / 0342-3443 (S) 3P4955 / 0342-3443 (T)	0.027	0.921	37	-16 -20	3.30E+17	8.6	0	4.3	8.6	1.2 -2.8

(1) For weld materials, (S) = Single Wire, (T) = Tandem Wire.

(2) ~~Adjusted chemistry factor determined per NEDO-33144, Section 4.2.1.1 (Reference 4.8-5), which was approved by the NRC in an SER (Reference 4.8-2), and updated per Columbia specific Integrated Surveillance Program (ISP) data.~~

The chemistry factor for weld 5P6756/0342-3477 has been modified from the NRC approved (Reference 4.8-2)) chemistry factor in NEDO-33144, Section 4.2.1.1 (Reference 4.8-5), per a Columbia specific analysis incorporating recent surveillance data from the Integrated Surveillance Program (ISP).

Table 3.3.2-18 Aging Management Review Results – Diesel Fuel Oil System

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
10	Filter Body	Pressure boundary	Gray Cast Iron	Fuel oil (Internal)	Loss of material	Chemistry Program Effectiveness Inspection	VII.H1-10	3.3.1-20	A
11	Filter Body	Pressure boundary	Gray Cast Iron	Fuel oil (Internal)	Loss of material	Selective Leaching Inspection	N/A	N/A	G
12	Filter Body	Pressure boundary	Gray Cast Iron	Air-indoor uncontrolled (External)	Loss of material	External Surfaces Monitoring	VII.I-8	3.3.1-58	A
13	Filter Body	Pressure boundary	Steel	Fuel oil (Internal)	Loss of material	Chemistry Program Effectiveness Inspection	VII.H1-10	3.3.1-20	A
14	Filter Body	Pressure boundary	Steel	Fuel oil (Internal)	Loss of material	Fuel Oil Chemistry	VII.H1-10	3.3.1-20	B
15	Filter Body	Pressure boundary	Steel	Air-indoor uncontrolled (External)	Loss of material	External Surfaces Monitoring	VII.I-8	3.3.1-58	A
16	Flame Arrestor	Structural integrity	Aluminum Alloy	Air-outdoor (Internal)	None	None	N/A	N/A	G ~~0324~~
17	Flame Arrestor	Structural integrity	Aluminum Alloy	Air-outdoor (External)	None	None	N/A	N/A	G
18	Flexible Connection	Pressure boundary	Stainless Steel	Fuel oil (Internal)	Loss of material	Chemistry Program Effectiveness Inspection	VII.H1-6	3.3.1-32	A