UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 3, 2010

Mr. S.K. Gambhir
Vice President Technical Services
Columbia Generating Station
Energy Northwest
MD PE04
P.O. Box 968
Richland, WA 99352-0968

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE COLUMBIA GENERATING STATION, LICENSE RENEWAL APPLICATION CONCERNING SECTION 2.4

Dear Mr. Gambhir:

By letter dated January 19, 2010, Energy Northwest submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54 (10 CFR Part 54), to renew Operating License NPF-21 for Columbia Generating Station, for review by the U.S. Nuclear Regulatory Commission (NRC or the staff). The staff is reviewing the information contained in the license renewal application and has identified, in the enclosure, areas where additional information is needed to complete the review. Further requests for additional information may be issued in the future.

Items in the enclosure were discussed with Mr. Abbas Mostala and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-4029 or by e-mail at evelyn.gettys@nrc.gov.

Sincerely,

For

Evelyn Gettys, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosure:
As stated

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COLUMBIA GENERATING STATION
LICENSE RENEWAL APPLICATION
REQUEST FOR ADDITIONAL INFORMATION

RAI 3.3.2.3-1

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.

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 B.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

ENCLOSURE

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.

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.

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.

RAI 4.2.1 – Neutron Fluence and Beltline Evaluation

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.

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.

RAI 4.2.3 – Adjusted Reference Temperature (ART) Analysis

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:

4. 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.

5. 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.

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.

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.

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, Columbia Action Request Report (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.

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.

August 3, 2010

Mr. S.K. Gambhir
Vice President Technical Services
Columbia Generating Station
Energy Northwest
MD PE04
P.O. Box 968
Richland, WA 99352-0968

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Dear Mr. Gambhir:

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Sincerely,
/RA/ B. Pham for
Evelyn Gettys, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosure:
As stated

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Letter to W. Oxenford from E. Gettys dated August 3, 2010

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE COLUMBIA GENERATING STATION, LICENSE RENEWAL APPLICATION CONCERNING SECTION 2.4

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