March 27, 2007

Mr. J. V. Parrish
Chief Executive Officer
Energy Northwest
P.O. Box 968 (Mail Drop 1023)
Richland, WA 99352-0968

SUBJECT: COLUMBIA GENERATING STATION - CORRECTION TO SAFETY EVALUATION FOR AMENDMENT NO. 199, "ALTERNATIVE SOURCE TERM" (TAC NO. MC4570)

Dear Mr. Parrish:

On November 27, 2006, the U.S. Nuclear Regulatory Commission (NRC) issued Amendment No. 199 to Facility Operating License No. NPF-21 for the Columbia Generating Station. The amendment (Agencywide Documents Access and Management System (ADAMS) Accession No. ML062610429) replaced the accident source term used in design-basis radiological analyses with an alternative source term pursuant to Title 10 of the *Code of Federal Regulations*, Section 50.67, "Accident source term."

The following editorial errors were found in the safety evaluation (SE) after the issuance of Amendment No. 199, as documented by letter from Energy Northwest (the licensee) dated January 31, 2007 (ADAMS Accession No. ML070430470).

1. In SE Section 3.1.1.2.1, the first sentence should refer to leakage as a percent of atmospheric content by "volume" instead of by "weight."

2. In SE Section 3.1.1.2.2, the words "condensate filter demineralizers" should be changed to "MSIVs [main steam isolation valves]."

3. In SE Section 3.1.1.2.3, the last paragraph should refer to calculation number NE-02-04-05 instead of calculation LM-0646.

4. In SE Section 3.6, the first paragraph, second sentence, is incomplete. The sentence should read, "The radiation doses used for the EQ analyses at the current licensed core power level bound the doses used in the AST evaluations."

5. In the SE Table 1 listing of parameters for the fuel handling accident, the release location should refer to "the point on the RB [reactor building] wall closest to the local intake," instead of the SGT stack for "SGTS operating," and instead of the RB vent stack for "no SGTS."

6. In the SE Table 1 listing of parameters for the fuel handling accident, the atmospheric dispersion factors should refer to Submittal Table 4.7-1 instead of Tables 4.4-2 and 4.4-3.

In addition, during the NRC staff's evaluation of your application for Amendment No. 199, you stated in your letter dated September 11, 2006 (ADAMS Accession No. ML062620329), that regarding Surveillance Requirement (SR) 3.6.4.1.1:

> The use of 0.25 inches of vacuum water gauge includes margin to account for uncertainties. It is also understood that this SR is considered to be met during momentary perturbations of the secondary containment pressure (i.e., a transient that momentarily causes the pressure to be less than the SR acceptance criteria of 0.25 inches of vacuum water gauge) that are anticipated during routine operation of the plant (e.g., realignment of ventilation trains, opening and closing doors and hatches, and extreme wind gusts).

The NRC staff does not consider the intent of the SR to be met if the acceptance criteria are exceeded during its performance, even for "momentary perturbations of the secondary containment pressure that are anticipated during routine operation of the plant." Allowing performance outside the SR acceptance criteria is tantamount to changing the requirement.

The corrected SE pages are enclosed. The changes are identified by a vertical bar on the right. Please replace the affected pages in the November 27, 2006, SE with these revised pages. If you have any questions concerning this matter, please call me at 301-415-2296.

Sincerely,

/RA/

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosure: Corrected SE pages 6, 8, 22, and Table 1 (partial)

cc w/encl: See next page

In addition, during the NRC staff's evaluation of your application for Amendment No. 199, you stated in your letter dated September 11, 2006 (ADAMS Accession No. ML062620329), that regarding Surveillance Requirement (SR) 3.6.4.1.1:

> The use of 0.25 inches of vacuum water gauge includes margin to account for uncertainties. It is also understood that this SR is considered to be met during momentary perturbations of the secondary containment pressure (i.e., a transient that momentarily causes the pressure to be less than the SR acceptance criteria of 0.25 inches of vacuum water gauge) that are anticipated during routine operation of the plant (e.g., realignment of ventilation trains, opening and closing doors and hatches, and extreme wind gusts).

The NRC staff does not consider the intent of the SR to be met if the acceptance criteria are exceeded during its performance, even for "momentary perturbations of the secondary containment pressure that are anticipated during routine operation of the plant." Allowing performance outside the SR acceptance criteria is tantamount to changing the requirement.

The corrected SE pages are enclosed. The changes are identified by a vertical bar on the right. Please replace the affected pages in the November 27, 2006, SE with these revised pages. If you have any questions concerning this matter, please call me at 301-415-2296.

Sincerely,

/RA/
Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosure: Corrected SE pages 6, 8, 22, and Table 1 (partial)

cc w/encl: See next page

DISTRIBUTION:

PUBLIC	LPLIV Reading	RidsNrrDorlLpl4	RidsNrrPMFLyon
RidsNrrLALFeizollahi	RidsOgcRp	RidsNrrDirsItsb	
RidsAcrsAcnwMailCenter	RidsRegion4MailCenter		

ACCESSION NO.: ML070780388

OFFICE	LPL4/PM	LPL4/LA	ITSB/BC	LPL4/BC
NAME	FLyon	LFeizollahi	TKobetz	DTerao
DATE	3/21/07	3/21/07	3/26/07	3/27/07

OFFICIAL RECORD COPY

3.1.1.2.1 Containment Leakage Pathway

The drywell and wetwell are projected to leak at their design leakage of 0.5 percent of their atmospheric contents by volume per day for the first 24 hours, and 0.25 percent of their atmospheric contents by volume for the remainder of the 30-day accident duration. Leakage from the drywell and wetwell will collect in the free volume of the secondary containment and be released to the environment via ventilation system exhaust or leakage. Following a LOCA, the standby gas treatment system (SGTS) fans start and draw down the secondary containment to create a negative pressure with reference to the environment. The SGTS exhaust is processed through high-efficiency particulate air filter media before being released to the environment. EN states that, prior to the completion of the secondary containment drawdown, the containment leakage is assumed to go directly to the environment. After the 20-minute drawdown period, filtration of the leakage is credited; however, no credit is taken for the holdup in secondary containment.

3.1.1.2.2 Secondary Containment Bypass Leakages

Two sources of containment leakage that bypass secondary containment are MSIV leakage and miscellaneous leakages. The models for these leakages are discussed below. A new limit of 16 standard cubic feet per hour (scfh) per valve, or 64 scfh for four steam lines, at a test pressure of 39.7 pounds per square foot absolute (psia) (25 pounds per square inch gauge (psig)) is proposed in the TS change submitted with the license amendment request (LAR). Since the TS allowable leakage is assessed in units of scfh, and the steam lines are not at standard conditions of temperature and pressure, EN adjusted the assigned flow rates appropriately for the assumed accident conditions.

Credit was taken for natural deposition within the main steam lines. The main steam lines are seismically qualified up to the turbine stop valves. The main steam line piping between the two MSIVs is also credited for natural deposition. MSIV leakage was reduced by a factor of two at 24 hours. No credit was taken for the main steam line leakage control system. The operability requirements for this system would be removed as part of the proposed TS changes.

To accommodate a postulated single failure of an MSIV to close, credit for natural deposition was taken for only three of the four steam lines. For the three credited lines, natural deposition was calculated according to AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," dated December 9, 1998. A modified Bixler approach for gaseous iodine removal was used. The Bixler model is taken from NUREG/CR-6605, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," dated April 1998. The Bixler model was modified by adopting the AEB-98-03 well-mixed flow expression for gaseous iodine removal. Proposed credit for organic iodine removal is discussed below.

The second source of bypass leakage, miscellaneous leakage paths, was assumed to equal the proposed TS limit of 0.04 percent primary containment volume per day at

The second source of potential ESF leakage is into the condensate storage tanks (CSTs). During the operation of high-pressure core spray or reactor core isolation cooling systems aligned to the suppression pool, radiological impact of leakage into the CSTs through the CSTs suctions and test returns has been evaluated.

EN determined the dose contribution from the CSTs to the CR and offsite locations to be approximately 1 percent and 2 percent of the total dose, respectively. EN judged that these doses are not significant and did not include them in the total dose from a LOCA. Based upon EN's analysis for the CST pathways, the NRC staff agrees that they are not significant to the conclusion that 10 CFR 50.67(b)(2) regulatory criteria are met for the current analysis. However, in the EN LOCA calculation, NE-02-04-05, EN states that they conform to RG 1.183, Appendix A, Regulatory Position 5.2. This regulatory position considers the impact of design leakage through valves isolating ESF recirculation systems from tanks vented to the atmosphere. Should circumstances change (for example an increase in the operational or assumed leakage to the CSTs) such that the dose from these pathways either increases or becomes significant to the conclusions derived from the total dose from a postulated LOCA, the analyses should be updated to include the CST dose pathway.

3.1.1.2.2.2 Main Steam Isolation Leakage Control System

The original design function of the main steam isolation valve leakage control (MSIVLC) system was to minimize the release of fission products via the main steam lines that could potentially bypass containment and the SGTS after a LOCA. The MSIVLC system performed this function by directing MSIV leakage to the SGTS. This leakage was directed to the SGTS by a blower that served to maintain the pressure in the steam lines negative, with respect to atmosphere. The routing of this leakage to the SGTS provided for filtration of MSIV leakage and its exhaust via the plant stack.

The MSIV leakage in the AST LOCA dose model is assumed to flow directly to the environment without credit for SGTS filtration. Additionally, the MSIV leakage was analyzed as a release from the turbine generator building exhaust, and this provides a conservative χ/Q compared to a release via the plant stack. EN is planning to deactivate the MSIVLC system during the implementation of the approved AST LAR.

3.1.1.3 Secondary Containment Drawdown

Containment integrity is ensured in part by TS 3.6.4.1, “Secondary Containment” and its associated surveillance requirements. SR 3.6.4.1.4 demonstrates that the SGT has the capability of drawing down the secondary containment to negative 0.25 inches w.g. in a two minute period of time. Changes which reduce containment integrity would be detected by increases in draw down time in successive testing until ultimately the test would fail. SR 3.6.4.1.5 demonstrates that the SGT is capable of maintaining a negative pressure in the secondary containment for a period of time at a reduced SGT fan flow rate of 2240 cfm. An increase in inleakage above the fan flow rate would cause the test to fail and indicate that integrity had degraded. The licensee states in its current TS Bases document “[t]he internal pressure of the SGT system boundary [secondary containment] is maintained at a negative pressure of 0.25 inches water gauge when the system is in operation, which represents the internal pressure required to ensure zero exfiltration of air from the building using the 95%

3.6 Equipment Qualification

The licensee has elected to retain TID-14844 assumptions for performing the required EQ analyses. The radiation doses used for the EQ analyses at the current licensed core power level bound the doses used in the AST evaluations.

The equipment exposed to the containment atmosphere, the TID-14844 source term and the gap and in-vessel releases in the AST produced similar integrated doses. For the equipment exposed to sump water, the integrated doses calculated with the AST exceeded those calculated with TID-14844 after 145 days for a BWR, because of the 30 percent vs. 1 percent release of cesium according to NUREG-1465. The continued use of the TID-14844 source term provides integrated doses for equipment, which envelop those that would be calculated using AST. Therefore, following implementation of AST, CGS will continue to meet 10 CFR 50.49 by using a radiation environment associated with the most severe DBA.

The NRC staff considered the applicability of the revised source terms to operating reactors and determined that the current analytical approach based on the TID-14844 source term would continue to be adequate to protect public health and safety, and that operating reactors licensed under this approach would not be required to reanalyze accidents using the revised source term.

Based on the above information, the NRC staff finds the licensee's position to retain the TID-14844 assumptions for performing the required EQ analyses is acceptable.

4.0 TECHNICAL SPECIFICATION CHANGES

The proposed changes to the TSs are contained in Attachment 1, pages 5-15 of the September 30, 2004, submittal and its supplements. The TS changes that are supported by the radiological design-basis analyses discussed in Section 3.1 are summarized below:

4.1 Table of Contents

Technical Specification	Description
1.1	"Definitions"
Table 3.3.6.2-1	"Secondary Containment Isolation Instrumentation"
Table 3.3.7.1-1	"Control Room Emergency Filtration (CREF) System Instrumentation"
3.6.1.3	"Primary Containment Isolation Valves (PCIVs)"
3.6.1.8	"Main Steam Isolation Valve Leakage Control (MSLC) System"
3.6.4.1	"Secondary Containment"
3.6.4.2	"Secondary Containment Isolation Valves (SCIVs)"
3.6.4.3	"Standby Gas Treatment (SGT) System"
3.7.3	"Control Room Emergency Filtration (CREF) System"
3.9.7	"Reactor Pressure Vessel (RPV) Water Level - New Fuel or Control Rods"
3.9.10	"Decay Time"

Fuel Handling Accident

Peaking factor	1.7
Fuel rods damaged, rods[9]	250
Decay period, hrs	24
Pool decontamination factor	
Iodine	200
Noble Gases	1
Particulate	Infinite
Fraction of core in gap	
I-131	0.08
Kr-85	0.10
Other iodines	0.05
Other noble gases	0.05
Alkali Metals[10]	0.12
Release period, hr	2
Release location	The point on the RB wall closest to the local intake
Control Room Emergency Filtration (CREF) initiation	Not Credited
Control room normal unfiltered intake, cfm	1,100
Water Depth, ft	≥ 23
Atmospheric Dispersion Factors χ/Q values, sec/m^3	Submittal Table 4.7-1

Main Steam Line Break

MSIV closure time, sec	6
Reactor coolant system pressure, psia	1,060
Reactor coolant system temperature, degrees F	552
Reactor coolant activity, µCi/gm dose equivalent I-131	
Normal	0.2
Spike	4.0

[9]Based upon fuel assembly with an 8x8 fuel pin array, but applied to all fuel types.

[10]Cesium and rubidium are present and were considered, but were not included in the calculation because the DF for particulate is assumed to be infinite.

cc:

Mr. W. Scott Oxenford (Mail Drop PE04)
Vice President, Technical Services
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Mr. Albert E. Mouncer (Mail Drop PE01)
Vice President, Corporate Services/
General Counsel/CFO
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Chairman
Energy Facility Site Evaluation Council
P.O. Box 43172
Olympia, WA 98504-3172

Mr. Douglas W. Coleman (Mail Drop PE20)
Manager, Regulatory Programs
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Mr. Gregory V. Cullen (Mail Drop PE20)
Supervisor, Licensing
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-4005

Chairman
Benton County Board of Commissioners
P.O. Box 190
Prosser, WA 99350-0190

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 69
Richland, WA 99352-0069

Mr. Dale K. Atkinson (Mail Drop PE08)
Vice President, Nuclear Generation
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Mr. William A. Horin, Esq.
Winston & Strawn
1700 K Street, N.W.
Washington, DC 20006-3817

Mr. Matt Steuerwalt
Executive Policy Division
Office of the Governor
P.O. Box 43113
Olympia, WA 98504-3113

Ms. Lynn Albin
Washington State Department of Health
P.O. Box 7827
Olympia, WA 98504-7827

Technical Services Branch Chief
FEMA Region X
130 228th Street, S.W.
Bothell, WA 98201-9796

Ms. Cheryl M. Whitcomb (Mail Drop PE03)
Vice President, Organizational
Performance & Staffing/CKO
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Assistant Director
Nuclear Safety and Energy Siting Division
Oregon Department of Energy
625 Marion Street, NE
Salem, OR 97301-3742

Special Hazards Program Manager
Washington Emergency Management Div.
127 W. Clark Street
Pasco, WA 99301