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

August 18, 2009

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

SUBJECT: COLUMBIA GENERATING STATION - ISSUANCE OF AMENDMENT RE: CHANGE IN GROUP 1 PRIMARY ISOLATION VALVES REACTOR WATER LEVEL ISOLATION SIGNAL FROM LEVEL 2 TO LEVEL 1 (TAC NO. MD9598)

Dear Mr. Parrish:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 214 to Facility Operating License No. NPF-21 for the Columbia Generating Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 9, 2008, as supplemented by letter dated April 24, 2009.

The amendment revises TS 3.3.6.1, "Primary Containment Isolation Instrumentation." The change lowers the Group 1 isolation valves reactor water level isolation signal from Level 2 to Level 1.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

Nicholas J. DiFrancesco, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures:
1. Amendment No. 214 to NPF-21
2. Safety Evaluation

cc w/encls: Distribution via Listserv

ENERGY NORTHWEST

DOCKET NO. 50-397

COLUMBIA GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 214
License No. NPF-21

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Energy Northwest (licensee), dated September 9, 2008, as supplemented by letter dated April 24, 2009, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-21 is hereby amended to read as follows:

    (2) <u>Technical Specifications and Environmental Protection Plan</u>

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 214 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented prior to entry into Mode 2 during startup from refueling outage 20.

FOR THE NUCLEAR REGULATORY COMMISSION

Mohan C. Thadani /for

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility
Operating License No. NPF-21
and Technical Specifications

Date of Issuance: August 18, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 214

FACILITY OPERATING LICENSE NO. NPF-21

DOCKET NO. 50-397

Replace the following pages of the Facility Operating License No. NPF-21 and Appendix A, Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Facility Operating License

REMOVE	INSERT
-3-	-3-

Technical Specification

REMOVE	INSERT
3.3.6.1-5	3.3.6.1-5

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source of special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

(6) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to store byproduct, source and special nuclear materials not intended for use at Columbia Generating Station. The materials shall be no more than 9 sealed neutron radiation sources designed for insertion into pressurized water reactors and no more than 40 sealed beta radiation sources designed for use in area radiation monitors. The total inventory shall not exceed 24 microcuries of strontium-90, 20 microcuries of uranium-235, 30 curies of plutonium-238, and 3 curies of americium-241.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at reactor core power levels not in excess of full power (3486 megawatts thermal). Items in Attachment 1 shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 214 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(a) For Surveillance Requirements (SRs) not previously performed by existing SRs or other plant tests, the requirement will be considered met on the implementation date and the next required test will be at the interval specified in the Technical Specifications as revised in Amendment No. 149.

Table 3.3.6.1-1 (page 1 of 4)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ -142.3 inches
b. Main Steam Line Pressure - Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 804 psig
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 124.4 psid
d. Condenser Vacuum - Low	1,2[(a)], 3[(a)]	2	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 7.2 inches Hg vacuum
e. Main Steam Tunnel Temperature - High	1,2,3	2	D	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 170°F
f. Main Steam Tunnel Differential Temperature - High	1,2,3	2	D	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 90°F
g. Manual Initiation	1,2,3	4	G	SR 3.3.6.1.6	NA
2. Primary Containment Isolation					
a. Reactor Vessel Water Level - Low, Level 3	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 9.5 inches
b. Reactor Vessel Water Level - Low Low, Level 2	1,2,3	2[(e)]	H	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ -58 inches
c. Drywell Pressure - High	1,2,3	2[(e)]	H	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 1.88 psig

(continued)

(a) With any turbine throttle valve not closed.

(e) Also required to initiate the associated LOCA Time Delay Relay Function pursuant to LCO 3.3.5.1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 214 TO

FACILITY OPERATING LICENSE NO. NPF-21

ENERGY NORTHWEST

COLUMBIA GENERATING STATION

DOCKET NO. 50-397

1.0 INTRODUCTION

By application dated September 9, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML082610230, Reference 1), as supplemented by letter dated April 24, 2009 (ADAMS Accession No. ML091320471, Reference 5), Energy Northwest (the licensee) requested changes to the Technical Specifications (TSs) (Appendix A to Facility Operating License No. NPF-21) for the Columbia Generating Station (CGS). The requested change would revise TS Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," Function 1.a to lower the Group 1 isolation valves reactor water level isolation signal from Level 2 (L2) to Level 1 (L1). Changing the isolation signal water level from L2 to L1 allows more energy to be transferred to the main condenser rather than the suppression pool via the safety relief valves (SRVs).

Specifically, the licensee proposes to change TS Table 3.3.6.1-1, Function 1.a from:

1. Main Steam Line Isolation
   a. Reactor Vessel Water Level - Low Low, Level 2

to:

1. Main Steam Line Isolation
   a. Reactor Vessel Water Level - Low Low Low, Level 1

The allowable value of Function 1.a would correspondingly be changed from "≥ 68 inches" (Level 2) to "≥ -142.3 inches" (Level 1).

The proposed change is consistent with NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4," Volume 1, Revision 3 (Reference 2), and NUREG-1434, "Standard Technical Specifications General Electric Plants, BWR/6," Volume 1, Revision 3

(Reference 3). The NRC staff has previously approved the proposed change at several nuclear power plants, including: Amendment No. 83 for Cooper Nuclear Station, dated May 5, 1983; Amendment Nos. 112, 106, and 80 for Browns Ferry, Units 1, 2, and 3, dated September 19, 1984; Amendment Nos. 111 and 115 for Peach Bottom Atomic Power Station, Units 2 and 3, dated October 2, 1985; Amendment No. 25 for Susquehanna Steam Electric Station, Unit 2, dated April 1, 1986; Amendment No. 103 for FitzPatrick Nuclear Power Plant, dated December 19, 1986; and Amendment Nos. 50 and 33 for LaSalle County Station, Units 1 and 2, dated May 6, 1987.

In the supplemental letter dated April 24, 2009, the licensee proposed that the implementation period be such as to support implementation prior to entry into Mode 2 during startup from Refueling Outage 20, the change to the implementation date does not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 2, 2008 (73 FR 73353).

2.0 REGULATORY EVALUATION

The NRC staff evaluates the acceptability of the CGS application based on adherence to the following applicable Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A General Design Criteria (GDCs):

- GDC-16, "Containment design," which requires the reactor containment provide an "essentially leak-tight barrier against uncontrolled release of radioactivity to the environment..."

- GDC-33, "Reactor coolant makeup," which requires "a system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary... The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary..."

- GDC-35, "Emergency core cooling," which requires "a system to provide abundant emergency core cooling ... The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts..."

- GDC-54, "Piping systems penetrating containment," which requires that "piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems..."

Furthermore, the NRC staff has evaluated the application based on its adherence to the following regulations:

- Section 50.46 of 10 CFR, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors"

- Section 50.36 of 10 CFR, "Technical specifications." The last paragraph of 10 CFR 50.36(c)(1)(ii)(A) states:

> Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.

3.0 TECHNICAL EVALUATION

The licensee proposes to change the Group 1 primary containment isolation valves reactor water level isolation setpoint from L2 to L1. This allows energy to continue to be transferred to the main condenser for a longer period of time after a scram instead of to the suppression pool via the SRVs. The cycling of the SRVs creates the potential for an SRV to stick or leak and poses additional challenges to the plant. The proposed change has been implemented at several operating boiling-water reactors (BWRs) and has been included in NUREG-1433, Revision 3, and NUREG-1434, Revision 3.

The function of the reactor pressure vessel (RPV) water level initiation signals are to progressively and automatically actuate functions in response to changes in the RPV water level. The current function of RPV water L2 initiation includes the Group 1 isolations which include the main steam isolation valves (MSIVs) and the main steamline drain valves. When L2 is reached, the Group 1 isolation signal is activated and the MSIVs and the main steamline drain valves close. This closure isolates the main turbine and main condenser from the reactor. The main condenser is then no longer available to relieve pressure. The SRVs and suppression pool are utilized for pressure relief. If the Group 1 isolation setpoint is changed from L2 to L1, the need to use the SRVs for pressure relief would become less common.

After the Three Mile Island accident in 1979, NUREG-0737, "TMI Action Plan Requirements" (Reference 4), Item II.K.3.16 required licensees to reduce the potential load on SRVs so that the SRVs do not fail. One of the recommendations was to change the Group 1 isolation setpoint from L2 to L1. Due to this recommendation, the plants mentioned in Section 1.0 of this safety evaluation submitted applications to the NRC to lower the Group 1 isolation setpoint from L2 to L1. CGS did not submit an application at the time due to the installation of SRVs with improved reliability. The NRC subsequently incorporated the change to Group 1 isolation setpoint from L2 to L1 into Standard Technical Specifications.

As part of the submittal, the licensee evaluated the anticipated operational occurrences (AOOs), off-design abnormal transients, and postulated accidents that are discussed in Chapter 15 of the CGS final safety analysis report (FSAR). The evaluation was performed to see if those plant transients were affected by the change in the Group 1 isolation setpoint from L2 to L1. Most of the transients were not affected by the change because either the L2 setpoint was not challenged during the transient, or the Group 1 isolation was initiated by another signal (such as high steam flow), or the transient was bounded by a more limiting event.

The licensee identified three events that were affected by the Group 1 isolation setpoint change from L2 to L1. These events were: loss of feedwater flow (LOFF), loss-of-coolant accident (LOCA), and anticipated transient without scram (ATWS) with LOFF.

In a LOFF, it is assumed that the feedwater trips and reactor vessel water level drops quickly to the Level 3 (L3) setpoint which initiates a reactor scram. The level continues to drop until L2 is reached, at which point the high pressure core spray (HPCS) and the reactor core isolation cooling (RCIC) systems initiate. The current L2 setpoint would also initiate a signal to shut the Group 1 primary containment isolation valves. With the licensee's proposed change, the Group 1 isolation is not anticipated to occur. The reactor vessel water level will not reach the L1 setpoint in this event and the core will remain adequately covered and cooled using the RCIC system. The heat is carried to the main condenser through the bypass valves. No temperature, pressure, or level transient would occur and challenge the design of the fuel, RPV, or containment.

In a LOCA, it is assumed that RPV water will be flowing through a break in the primary pressure boundary and into the containment, lowering the RPV level and pressure. During a LOCA, several conditions may close the MSIVs prior to the L1 setpoint being reached. For CGS's analysis, lowering the Group 1 isolation setpoint from L2 to L1 would not affect the design-basis accident (DBA) LOCA analysis because the isolation is assumed to occur at the time of LOCA initiation. The licensee stated in its application that closing the MSIVs at the LOCA initiation results in a bounding peak cladding temperature. Therefore, the change in the Group 1 isolation setpoint from L2 to L1 would not challenge the LOCA acceptance criteria for peak cladding temperature or change the radiological consequences of a DBA-LOCA.

In an ATWS-LOFF, it is assumed that the reactor has failed to scram in conjunction with LOFF. When this happens, the level will decrease to L2 where the reactor recirculation pumps trip which will increase the voiding in the core that will decrease power. HPCS and RCIC also initiate when the L2 setpoint is reached and maintain reactor water level. The main condenser will still be available to condense steam; therefore, SRV cycling will not occur and the suppression pool heatup will be reduced during the event.

Based on the above, the NRC staff finds that the proposed change is acceptable. The licensee has provided adequate justification for the proposed change and the NRC staff concludes that the licensee's evaluation of the licensing basis events provides reasonable assurance that the proposed change to the Group 1 isolation setpoint from the current L2 to L1 will not adversely affect the plant's ability to comply with the regulatory requirements.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Washington State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding published in the *Federal Register* on December 2, 2008 (73 FR 73353). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. S. K. Gambhir, Energy Northwest, to U.S. Nuclear Regulatory Commission, "License Amendment Request for Proposed Changes to Columbia Technical Specification 3.3.6.1; Change Group 1 Primary Containment Isolation Valves Reactor Water Level Isolation Signal from Level 2 to Level 1," dated September 9, 2008, ADAMS Accession No. ML090330141.

2. U.S. Nuclear Regulatory Commission, "Standard Technical Specifications General Electric Plants, BWR/4," NUREG-1433, Vol. 1 and 2, Revision 3, June 30, 2004, ADAMS Accession Nos. ML041910194, ML041910212, and ML041910214.

3. U.S. Nuclear Regulatory Commission, "Standard Technical Specifications General Electric Plants, BWR/6," NUREG-1434, Vol. 1 and 2, Revision 3, June 30, 2004, ADAMS Accession Nos. ML041910204, ML041910223, and ML041910224.

4. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980, ADAMS Accession No. ML051400209.

5. S. K. Gambhir, Energy Northwest, to U.S. Nuclear Regulatory Commission, "Revised Implementation Date for License Amendment Request for Proposed Changes to Columbia Technical Specification 3.3.6.1; Change Group 1 Primary Containment Isolation Valves Reactor Water Level Isolation Signal from Level 2 to Level 1," dated April 24, 2009, ADAMS Accession No. ML091320471.

Principal Contributor: J. Miller

Date: August 18, 2009

August 18, 2009

Mr. J. V. Parrish
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P.O. Box 968 (Mail Drop 1023)
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SUBJECT: COLUMBIA GENERATING STATION - ISSUANCE OF AMENDMENT RE: CHANGE IN GROUP 1 PRIMARY ISOLATION VALVES REACTOR WATER LEVEL ISOLATION SIGNAL FROM LEVEL 2 TO LEVEL 1 (TAC NO. MD9598)

Dear Mr. Parrish:

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A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,
/RA/

Nicholas J. DiFrancesco, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-397

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1. Amendment No. 214 to NPF-21
2. Safety Evaluation

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