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

June 30, 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: CONTROL ROD NOTCH TESTING (TAC NO. ME0391)

Dear Mr. Parrish:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 212 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 January 14, 2009.

The amendment revises the TS surveillance requirement frequency in TS 3.1.3, "Control Rod OPERABILITY," and revises Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. These changes are in accordance with NRC-approved TS Task Force (TSTF) change traveler TSTF-475, Revision 1, "Control Rod Notch Testing Frequency and SRM [Source Range Monitor] Insert Control Rod Action," as part of the consolidated line item improvement process.

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

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. 212 to NPF-21
2. Safety Evaluation

cc w/encls: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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ENERGY NORTHWEST

DOCKET NO. 50-397

COLUMBIA GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 212
License No. NPF-21

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

   A. The application for amendment by Energy Northwest (licensee), dated January 14, 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.

Enclosure 1

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. 212 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 within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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: June 30, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 212

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
1.4-5	1.4-5
1.4-6	1.4-6
1.4-7	1.4-7
1.4-8	1.4-8
3.1.3-2	3.1.3-2
3.1.3-5	3.1.3-5
3.1.3-6	--
3.1.4-4	3.1.4-4

(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) Maximum Power Level

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) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 212 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.

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-3 (continued)

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day interval (plus the extension allowed by SR 3.0.2), but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----NOTE----- Only required to be met in MODE 1. ----------	
Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in

(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-4 (continued)

Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
------------------NOTE------------------ Only required to be performed in MODE 1. --	
Perform complete cycle of the valve.	7 days

The interval continues, whether or not the unit operation is in MODE 1, 2, or 3 (the assumed Applicability of the associated LCO) between performances.

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1.

(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-5 (continued)

Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

Once the unit reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance were not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
------------------NOTE------------------ Not required to be met in MODE 3. -- Verify parameter is within limits.	24 hours

Example 1.4-6 specifies that the requirements of this Surveillance do not have to be met while the unit is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1, 2, and 3). The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an “otherwise stated” exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the

(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-6 (continued)

24 hour interval (plus the extension allowed by SR 3.0.2), and the unit was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming agin that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Perform SR 3.1.3.2 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
	<u>AND</u>		
	A.4	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1	Verify the total number of "slow" and inoperable control rods is $\leq$ eight.	Immediately
	<u>AND</u>		
			(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2	-------------------NOTE------------------- Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. -- Insert each withdrawn control rod at least one notch.	 31 days
SR 3.1.3.3	Verify each control rod scram time from fully withdrawn to notch position 5 is $\leq$ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4
SR 3.1.3.4	Verify each control rod does not go to the withdrawn overtravel position.	Each time the control rod is withdrawn to "full out" position AND Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling

Table 3.1.4-1
Control Rod Scram Times

-----NOTES-----

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 5. These control rods are inoperable, in accordance with SR 3.1.3.3, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES(a)(b) (seconds) WHEN REACTOR STEAM DOME PRESSURE ≥ 800 psig
45	0.528
39	0.866
25	1.917
5	3.437

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure, when < 800 psig, are within established limits.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 212 TO

FACILITY OPERATING LICENSE NO. NPF-21

ENERGY NORTHWEST

COLUMBIA GENERATING STATION

DOCKET NO. 50-397

1.0 INTRODUCTION

By application dated January 14, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090330141, Reference 1), 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 revises the frequency for notch testing of fully withdrawn control rods from 7 to 31 days. In addition, the amendment proposes to clarify one Example in Section 1.4, "Frequency," to clarify that the 1.25 interval in Surveillance Requirement (SR) 3.0.2 is applicable to time periods discussed in the NOTE in the "SURVEILLANCE" column in addition to the time periods in the "FREQUENCY" column. The licensee's letter states,

> Energy Northwest has concluded that the justifications presented in the TSTF [Technical Specifications Task Force] proposal and the SE [safety evaluation] prepared by the NRC [U.S. Nuclear Regulatory Commission] staff are applicable to Columbia and justify this amendment to the Columbia TS.

The amendment request is based on the NRC-approved Technical Specifications Task Force (TSTF) change traveler TSTF-475, Revision 1, "Control Rod Notch Testing Frequency and SRM [Source Range Monitor] Insert Control Rod Action" (Reference 2), and proposes to modify the CGS TS by:

(1) revising the frequency of SR 3.1.3.2 notch testing of each fully withdrawn control rod, from "7 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP [low power setpoint] of the RWM [rod worth minimizer]" to "31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM" (NUREG-1433, Revision 3, and NUREG-1434, Revision 3, References 3 and 4), and

(2) revising Example 1.4-3 in Section 1.4, "Frequency," to clarify that the 1.25 surveillance test interval extension in SR 3.0.2 is applicable to time periods

discussed in NOTES in the "SURVEILLANCE" column in addition to the time periods in the "FREQUENCY" column.

The purpose of the surveillances is to confirm control rod insertion capability which is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. Control rods and the control rod drive (CRD) mechanism (CRDM), by which the control rods are moved, are components of the CRD system (CRDS), which is the primary reactivity control system for the reactor. By design, the CRDM is highly reliable with a tapered design of the index tube which is conducive to control rod insertion.

A stuck control rod is an extremely rare event and industry review of plant operating experience did not identify any incidents of stuck control rods while performing a rod notch surveillance test. The purpose of the change in surveillance frequency is to reduce the number of control rod manipulations and thereby reduce the opportunity for reactivity control events. The purpose of the change to Example 1.4-3 in Section 1.4, "Frequency," is to clarify the applicability of the 25 percent allowance of SR 3.0.2 to time periods discussed in NOTES in the "SURVEILLANCE" column as well as to time periods in the "FREQUENCY" column.

2.0 REGULATORY EVALUATION

In Section 50.36, "Technical specifications," of Title 10 of the *Code of Federal Regulations* (10 CFR), the Commission established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCO); (3) SRs; (4) design features; and (5) administrative controls.

In 10 CFR, Part 50, Appendix A, General Design Criterion (GDC) 26, "Reactivity control system redundancy and capability," states that:

> Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

GDC 29, "Protection against anticipated occurrence," states that:

> The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

The purpose of the rod notch test surveillance is to ensure that the CRDS maintains an extremely high probability of accomplishing its specified safety functions in the event of an anticipated operational occurrence (AOO) in accordance with GDC 29 (e.g., that any degradation in the system is identified early).

The design relies on the CRDS to function in conjunction with the protection systems under AOOs, including loss of power to all recirculation pumps, tripping of the turbine generator, isolation of the main condenser, and loss of all offsite power. The CRDS provides an adequate means of inserting sufficient negative reactivity to shut down the reactor and prevent exceeding acceptable fuel design limits during AOOs. Compliance with GDCs 26 and 29 for the CRDS prevents occurrence of mechanisms that could result in fuel cladding damage such as severe overheating, excessive cladding strain, or exceeding the thermal margin limits during AOOs. Preventing excessive cladding damage in the event of anticipated transients ensures maintenance of the integrity of the cladding as a fission product barrier. Per CGS's Final Safety Analysis Report, the plant's principal design criteria is in compliance with GDCs 26 and 29.

10 CFR 50.36(c)(3) states that TSs shall contain SRs "relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." As discussed in Section 3.0 of this safety evaluation (SE), revising the SR frequency for notch testing of each fully withdrawn control rod from 7 days to 31 days, as well as clarifying in a TS example that the 1.25 surveillance test interval extension in SR 3.0.2 is also applicable to time periods discussed in SR notes, still assures that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

3.0 TECHNICAL EVALUATION

3.1 Proposed Changes to TSs

In its application, the licensee proposed the following changes to TS 3.1.3, "Control Rod Operability," of the CGS TSs:

1. Remove the reference to SR 3.1.3.3 from TS LCO 3.1.3 Required Action A.3 for the Condition of one withdrawn control rod stuck,

2. Delete SR 3.1.3.2 and renumbering of SR 3.1.3.3 – 3.1.3.5 to SR 3.1.3.2 – 3.1.3.4,

3. Revise SR 3.1.3.3 by deleting the word "partially" and extending the time to perform the surveillance to 31 days, and

4. Revise the second note in Table 3.1.4-1, "Control Rod Scram Times," to reflect the revised reference to SR 3.1.3.3.

To be clear that the allowance of the 1.25 surveillance test interval extension in SR 3.0.2 is applicable to time periods discussed in notes for SRs (e.g., in revised SR 3.1.3.2), the Example

1.4-3 in TS Section 1.4, "Frequency," is also revised to add the phrase in two places: "(plus the extension allowed by SR 3.0.2)."

3.2 Revising the SR Frequency for Notch Testing of Each Fully Withdrawn Control Rod from 7 days to 31 days

The CRDS is the primary reactivity control system for the reactor. The CRDS, in conjunction with the reactor protection system, provides the means for the reliable control of reactivity changes to ensure under all conditions of normal operation, including AOOs that specified acceptable fuel design limits are not exceeded. Control rods are components of the CRDS that have the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRDS.

The CRDS consists of a CRDM, by which the control rods are moved, and a hydraulic control unit (HCU) for each control rod. The CRDM is a mechanical hydraulic latching cylinder that positions the control blades. The collet piston mechanism design feature ensures that the control rod will not be inadvertently withdrawn. This is accomplished by engaging the collet fingers, mounted on the collet piston, in notches located on the index tube. Due to the tapered design of the index tube notches, the collet piston mechanism will not impede rod insertion under normal insertion or scram conditions.

The collet retainer tube (CRT) is a short tube welded to the upper end of the CRD which houses the collet mechanism which consist of the locking collet, collet piston, collet return spring and an unlocking cam. The collet mechanism provides the locking/unlocking mechanism that allows the insert/withdraw movement of the control rod. The CRT has three primary functions: (a) to carry the hydraulic unlocking pressure to the collet piston, (b) to provide an outer cylinder, with a suitable wear surface for the metal collet piston rings, and (c) to provide mechanical support for the guide cap, a component which incorporates the cam surface for holding the collet fingers open and also provides the upper rod guide or bushing.

The NRC staff approved TSTF-475, which revised the TS SR 3.1.3.2, “Control Rod OPERABILITY,” in the STS (NUREG-1433, Revision 3 and NUREG-1434, Revision 3) from 7 days to 31 days, based on the following: (1) slow crack growth rate of the CRT; (2) the improved CRT design; (3) a higher reliable method (scram time testing) to monitor CRD scram system functionality; (4) GE Nuclear Energy (GE) chemistry recommendations; and (5) no known CRD failures have been detected during the notch testing exercise. The NRC staff concluded that the changes would reduce the number of control rod manipulations thereby reducing the opportunity for potential reactivity events while having a very minimal impact on the extremely high reliability of the CRDS. The following paragraphs describe the bases for the NRC staff's approval of TSTF-475, Revision 1 (Reference 5):

CRT cracking was first discovered in 1975. It was determined that during scrams, the CRT temperature distribution changes substantially at reactor operating conditions. Relatively cold water moves upward through the inside of the CRT and exits via the flow holes into the annulus on the outside. At the same time, hot water from the reactor vessel flows downward on the outside surface of the CRT. There is very little mixing of the cold water flowing from the three flow holes into the annulus and the hot water flowing downward. Thus, there are substantial through-wall and circumferential temperature gradients during scrams which contribute to the observed CRT cracking. It was recognized that notch testing provided a method to demonstrate the integrity of the CRT. Each partially or fully withdrawn operable control rod was required to be exercised one notch at least once each week. Control rod insertion capability was demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal.

Subsequently, many boiling-water reactors (BWRs) have reduced the frequency of notch testing for partially withdrawn control rods from 7 days to 31 days. The notch test frequency for fully withdrawn control rods was still performed every 7 days. The change, for partially withdrawn control rods, was made because of the potential power reduction required to allow control rod movement for partially withdrawn control rods, the desire to coordinate scheduling with other plant activities, and the fact that a large sample of control rods are still notch tested every 7 days. The operating experience related to the changes in CRD performance also provided additional justification to reduce the notch test frequency for the partially withdrawn control rods. Current operating experience now provides justification to reduce the notch test frequency for the fully withdrawn control rods as well. A review of industry operating experience did not identify any incidents of stuck control rods identified via performance of a rod notch surveillance for either partially or fully withdrawn control rods. Therefore, increasing the CRD notch testing frequency for fully withdrawn control rods from 7 days to 31 days will have minimal impact on the reliability of the CRDS.

The BWR scram system has extremely high reliability. In addition to notch testing, other SRs are performed to verify the operability of the CRDS. Scram time testing can identify failure of individual CRD operation resulting from intergranular stress-corrosion cracking (IGSCC)-initiated cracks and mechanical binding. Unlike the CRD notch tests, these single rod scram tests cover the other mechanical components such as scram pilot solenoid operated valves, the scram inlet and outlet air operated valves, and the scram accumulator, as well as operation of the control rods. Thus, the primary assurance of scram system reliability is provided by the scram time testing since it monitors the system scram operation and the complete travel of the control rod. Also, the HCUs, CRD drives, and control rods are also tested during refueling outages, approximately every 18-24 months. Based on the data collected during the preceding cycle of operation, selected control rod drives, are inspected and, as required, their internal components are replaced. As a result, increasing the CRD notch testing frequency of fully withdrawn rods from 7 days to 31 days will have minimal impact on the reliability of the CRDS since additional SRs are performed that verify the operability of the system.

Based on no known CRD failures having been detected during the notch testing SR, as well as the performance of other diverse SRs used to verify the operability of the CRDS, the NRC staff concludes that revising the SR frequency for notch testing of each fully withdrawn control rod from 7 days to 31 days is acceptable. Furthermore, the proposed revision to SR 3.1.3.2, which states "Insert each withdrawn control rod at least one notch" is acceptable to the NRC staff since it applies to both fully withdrawn and partially withdrawn control rods.

It should be noted that approval to relax the SR frequency for notch testing of each fully withdrawn control rod is based on, in part, operational experience that has demonstrated no known CRD failures having been detected during the notch testing SR. Should the SR frequency relaxation result in a noticeable trend in failures, the licensee is expected to consider the need for revising the TS to include a more conservative testing frequency in accordance with NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications That Are Insufficient to Assure Plant Safety" (Reference 6). Administrative Letter 98-10 states that,

> Occasionally, as a result of licensee design-basis reconstitution efforts or NRC inspection efforts, licensees determine that specific values or required actions in TS may not assure safety. When this occurs, licensees typically conduct an evaluation and, if necessary, institute administrative controls that instruct the operators to maintain a more restrictive value for a particular parameter or to take a more conservative required action.

Administrative Letter 98-10 also states that,

> Imposing administrative controls in response to an improper or inadequate TS is considered an acceptable short-term corrective action. The staff expects that, following the imposition of administrative controls, an amendment to the TS, with appropriate justification and schedule, will be submitted in a timely fashion.

Although not a basis for approving the frequency extension of notch testing for partially withdrawn control rods, GE [General Electric] Nuclear Energy report, "CRD Notching Surveillance Testing for Limerick Generating Station" (Reference 7), provides additional insight as to why a review of industry operating experience may not have identified any incidents of stuck control rods identified via performance of a rod notch surveillance. The GE report is discussed in TSTF-475, Revision 1. The GE report provides a description of the cracks noted on the original design CRT surfaces. These cracks, which were later determined to be intergranular, were generally circumferential, and appeared with greatest frequency below and between the cooling water ports, in the area of the change in wall thickness. Subsequently, cracks associated with residual stresses were also observed in the vicinity of the attachment weld. Continued circumferential cracking could lead to 360 degree severance of the CRT that would render the CRD inoperable which would prevent insertion, withdrawal or scram. Such failure would be detectable in any fully or partially withdrawn control rod during the surveillance notch testing required by the TSs. To a lesser degree, cracks have also been noted at the welded joint of the interim design CRT but no cracks haven been observed in the final improved CRT design. No collet housing failures have been noted since 1975. In addition, the IGSCC growth rates were evaluated using GE's PLEDGE model with the assumption that the water chemistry condition is based on GE recommendations. The model is based on fundamental principles of stress-corrosion cracking which can evaluate crack growth rates as a function of water oxygen level, conductivity, material sensitization, and applied loads. It was determined that the additional time of 24 days represented an additional 10 mils of growth in total crack length. The small difference in growth rate would have little effect on the behavior between one notch test and the next subsequent test. Therefore, from the materials perspective based on low crack growth rates, a decrease in the notch test frequency would not affect the reliability of detecting a CRDM failure due to crack growth.

The licensee stated in its submittal that the justifications presented in TSTF-475 and the SE prepared by the NRC staff are applicable to CGS. The NRC staff found that the proposed changes are consistent with the changes approved by the NRC staff in TSTF-475, Revision 1. The NRC staff, therefore, concludes that the changes are acceptable.

3.3 Clarify in TS Example that the 1.25 Surveillance Test Interval Extension in SR 3.0.2 is also Applicable to Time Periods Discussed in SR Notes

The NRC staff has reviewed the proposal to amend Example 1.4-3 in Section 1.4, "Frequency," to clarify that the 1.25 provision in SR 3.0.2 is equally applicable to time periods specified in the notes of the "Surveillance" column. The definition of specified "Frequency" provided in the second paragraph of TS Section 1.4, states that,

> The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, "Surveillance Requirement (SR) Applicability." The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.

The NRC staff concludes that this change is acceptable since it clarifies the example to make it consistent with the definition of specified "Frequency" provided in TS Section 1.4.

3.4 TS Bases

The licensee proposed changes to the TS Bases are consistent with the proposed changes to the TSs. The licensee proposed some variations in the TS Bases from those proposed in TSTF-475. The variations were administrative in nature; therefore, NRC staff has no objections to the proposed changes to the TS Bases.

3.5 Conclusion

The licensee stated in its application that it had reviewed the basis for the NRC staff's acceptance of TSTF-475, Revision 1, and concluded that the basis is applicable to CGS, and supports its adoption of the TSTF-475 changes into the CGS TS. The staff also reviewed the TSTF-475, Revision 1 basis, and concluded that the basis for the TSTF is applicable to CGS and, therefore, the TSTF is appropriate for adoption by the licensee. In addition, the NRC staff reviewed the licensee's proposed changes against the corresponding changes made to the STS by TSTF-475, Revision 1, which the staff has found to satisfy applicable regulatory requirements, as described above. The proposed changes would: (1) revise the TS control rod notch surveillance frequency in TS 3.1.3, "Control Rod OPERABILITY," and (2) revise one Example in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. The NRC staff found that the proposed changes are consistent with the changes approved by the NRC staff in TSTF-475, Revision 1 and the CGS TSs meet the requirements of 10 CFR 50.36.

Based on the NRC staff's review of the licensee's proposed changes, the staff also concluded that the TS revisions will have a minimal effect on the reliability of the CRDS while reducing the opportunity for potential reactivity events, thus, meeting the requirement of 10 CFR Part 50, Appendix A, GDC 26 and 29, and will clarify the applicability of the 1.25 provision in SR 3.0.2. The NRC staff, therefore, concludes the proposed changes acceptable.

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 February 10, 2009 (74 FR 6665). 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. W. S. Oxenford, Energy Northwest, to U.S. Nuclear Regulatory Commission, "Columbia Generating Station, Docket No. 50-397 License Amendment Request to Revise Control Rod Notch Testing Frequency using the Consolidated Line Item Improvement Process (CLIIP) for TSTF-475," dated January 14, 2009, ADAMS Accession No. ML090330141.

2. Technical Specifications Task Force to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0, 'Control Rod Notch Testing Frequency and SRM Insert Control Rod Action,' dated February 28, 2007" (TSTF-475 Revision 1 is an enclosure), TSTF-07-19, dated May 22, 2007, ADAMS Accession No. ML071420428.

3. 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.

4. 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.

5. U.S. Nuclear Regulatory Commission, "Notice of Availability of Model Application Concerning Technical Specification Improvement To Revise Control Rod Notch Surveillance Frequency, Clarify SRM Insert Control Rod Action, and Clarify Frequency Example," Federal Register, Vol. 72, No. 218, pp. 63935 - 63942, dated November 13, 2007.

6. U.S. Nuclear Regulatory Commission, "NRC Administrative Letter 98-10: Dispositioning of Technical Specifications that are insufficient to Assure Plant Safety," dated December 29, 1998, ADAMS Accession No. ML031110108.

7. BWR Owners Group to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0," BWROG-06036, dated November 16, 2006, with Enclosure of the GE Nuclear Energy Report, "CRD Notching Surveillance Testing for Limerick Generating Station," dated November 2006, ADAMS Accession No. ML063250258.

8. Specifications Task Force to U.S. Nuclear Regulatory Commission, "TSTF-475 Revision 0, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action," TSTF-04-07, dated August 30, 2004, ADAMS Accession No. ML042520035.

9. Technical Specifications Task Force to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0," TSTF-06-13, dated July 3, 2006, ADAMS Accession No. ML061840342.

Principal Contributor: R. P. Grover

Date: June 30, 2009

June 30, 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: CONTROL ROD NOTCH TESTING (TAC NO. ME0391)

Dear Mr. Parrish:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 212 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 January 14, 2009.

The amendment revises the TS surveillance requirement frequency in TS 3.1.3, "Control Rod OPERABILITY," and revises Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. These changes are in accordance with NRC-approved TS Task Force (TSTF) change traveler TSTF-475, Revision 1, "Control Rod Notch Testing Frequency and SRM [Source Range Monitor] Insert Control Rod Action," as part of the consolidated line item improvement process.

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
Enclosures:
1. Amendment No. 212 to NPF-21
2. Safety Evaluation

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R. Grover, NRR/DIRS/ITSB

ADAMS Accession No.: ML091550803 *SE Memo dated

OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	DIRS/ITSB/BC	OGC	NRR/LPL4/BC	NRR/LPL4/PM
NAME	NDiFrancesco FLyon for	JBurkhardt	RElliott*	JSuttenberg	MMarkley	NDiFrancesco
DATE	6/9/09	6/9/09	6/3/09	6/11/09	6/30/09	6/30/09

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