February 20, 2001

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: TECHNICAL SPECIFICATIONS SURVEILLANCE REQUIREMENT 3.6.1.3.8 (TAC NO. MB0421)

Dear Mr. Parrish:

The Commission has issued the enclosed Amendment No. 170 to Facility Operating License No. NPF-21 for the Columbia Generating Station. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated October 30, 2000.

The amendment revised Surveillance Requirement 3.6.1.3.8 to allow a representative sample of reactor instrument line excess flow check valves (EFCVs) to be tested every 24 months such that each reactor instrument EFCV will be tested at least once every 10 years. The amendment also limited the surveillance requirement to only the reactor instrument line EFCVs.

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/

Jack Cushing, Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures: 1. Amendment No. 170 to NPF-21
2. Safety Evaluation

cc w/encls: See next page

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NRR-058

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

February 20, 2001

Mr. J. V. Parrish
Chief Executive Officer
Energy Northwest
P.O. Box 968 (Mail Drop 1023)
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Sincerely,

J Cushing

Jack Cushing, Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures: 1. Amendment No. 170 to NPF-21
2. Safety Evaluation

cc w/encls: See next page

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

ENERGY NORTHWEST

DOCKET NO. 50-397

COLUMBIA GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 170
License No. NPF-21

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

   A. The application for amendment by Energy Northwest dated October 30, 2000, 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) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 170and 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 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen Dembek, Chief, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 20, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 170

FACILITY OPERATING LICENSE NO. NPF-21

DOCKET NO. 50-397

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains vertical lines indicating the areas of change. The overleaf page is provided to maintain document completeness.

REMOVE	INSERT
3.6.1.3-8	3.6.1.3-8

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.1.3.3	-----------------NOTES----------------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. --------------------------------------- Verify each primary containment isolation manual valve and blind flange that is located inside primary containment and is required to be closed during accident conditions is closed.	Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days
SR 3.6.1.3.4	Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.	31 days
SR 3.6.1.3.5	Verify the isolation time of each power operated and each automatic PCIV, except MSIVs, is within limits.	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is $\geq$ 3 seconds and $\leq$ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify a representative sample of reactor instrument line EFCVs actuate to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify the combined leakage rate for all secondary containment bypass leakage paths is $\leq$ 0.74 scfh when pressurized to $\geq P_a$.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE NO. NPF-21

ENERGY NORTHWEST

COLUMBIA GENERATING STATION

DOCKET NO. 50-397

1.0 INTRODUCTION

By letter dated October 30, 2000, Energy Northwest, the licensee for the Columbia Generating Station, submitted a technical specification (TS) change request for excess flow check valve (EFCV) surveillance testing. Currently, the Columbia Generating Station TS requires verification of the actuation capability of each EFCV every 24 months. The proposed change revises TS Surveillance Requirement (SR) 3.6.1.3.8 to test only the reactor instrument line EFCVs and to relax its surveillance frequency by limiting the number of tests to a "representative sample" of EFCVs every 24 months, such that each EFCV will be tested at least once every 10 years (nominal). The proposed change fully adopts the staff's approved Technical Specifications Task Force (TSTF) Traveler TSTF-334, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," dated October 31, 2000. The basis for the request is the high degree of reliability shown by the EFCVs and the low consequences of an EFCV failure. As part of the change, the licensee proposed that EFCVs in the containment atmosphere and suppression pool instrumentation lines will no longer be covered by the TS and that their testing provisions will instead be controlled by plant processes governed by 10 CFR Part 50.59.

The supporting analysis for the licensee's conclusion is based on General Electric Nuclear Energy Topical Report B21-00658-01, "Excess Flow Check Valve Testing Relaxation" dated November 1998 and a response to request for additional information (RAI) dated January 8, 2000. The staff accepted the generic applicability of the Topical report by safety evaluation (SE) dated March 14, 2000. This report provided: (1) estimate of steam release frequency (into the reactor building) due to a break in an instrument line concurrent with an EFCV failure to close, and (2) assessment of the radiological consequences of such a release.

2.0 BACKGROUND

2.1 Purpose and Function of EFCVs

EFCVs are installed in boiling water reactor (BWR) instrument lines that penetrate the primary containment boundary to limit the release of fluid in the event of an instrument line break. As discussed in Regulatory Guide 1.11 "Instrument Lines Penetrating Primary Reactor

Containment," the use of EFCVs satisfies the requirements of General Design Criteria 55 and 56 for automatic isolation capability of lines penetrating containment, while maintaining a highly reliable capability to monitor important parameters inside containment. Examples of EFCV installations include reactor pressure vessel level and pressure instrumentation, main steam line flow instrumentation, recirculation pump suction pressure, and reactor core isolation cooling steam line flow instrumentation and, in the case of the Columbia Generating Station, in containment atmosphere and suppression pool instrument lines. EFCVs are not required to close in response to a containment isolation signal and are not required to operate under post loss-of-coolant accident (LOCA) conditions.

Currently, the Columbia Generating Station TS Surveillance Requirement (SR) 3.6.1.3.8 requires that each EFCV be tested, by actuation to the isolation position, on a simulated instrument line break every 24 months. The proposed change revises SR 3.6.1.3.8 to require verification that a representative sample of reactor instrument line EFCVs actuates to the isolation position during a simulated instrument line break signal every 24 months, such that each EFCV will be tested at least once every 10 years (nominal). Prior to implementing the Improved Standard Technical Specifications at the Columbia Generating Station, the EFCV surveillance requirement was under TS 4.6.3.4. The surveillance required that "Each reactor instrumentation line excess flow check valve shown in Table 3.6.3-1 shall be demonstrated Operable...." Because Columbia Generating Station is a BWR/5 plant and there are no Improved Standard Technical Specifications written specifically for BWR/5s, NUREG-1434 (BWR/6) was used for the conversion during the implementation of the Improved Standard Technical Specifications.

2.2 Topical Report NEDO-32977-A

The supporting analysis for the licensee's application is based on General Electric Nuclear Energy Topical Report NEDO-32977-A, DRF B21-00658-01, "Excess Flow Check Valve Testing Relaxation" dated June 2000. This report provided: (1) an estimate of steam release frequency into the reactor building due to a break in an instrument line concurrent with an EFCV failure to close, and (2) an assessment of the radiological consequences of such a release. In Topical Report NEDO-32977-A, the Boiling Water Reactors Owners Group (BWROG) concluded that EFCVs should be tested on a staggered group test basis on a performance-based schedule not to exceed 10 years. The conclusion was based on a risk and consequences evaluations described in the topical report. The staff reviewed the topical report and issued its evaluation on March 14, 2000. In its evaluation, the staff agreed that the test interval could be extended to as much as 10 years. In conjunction with this finding, the staff noted that each licensee that adopts the relaxed test interval program for EFCVs must have a feedback mechanism and corrective action program to ensure that good performance of EFCVs is maintained. Also, each licensee is required to perform a plant-specific radiological dose assessment, EFCV failure rate analysis, and release frequency analysis to confirm that they are bounded by the generic analyses of the topical report.

2.3 TSTF-334

The proposed change implements Technical Specifications Task Force (TSTF) Traveler TSTF 334, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," dated

October 31, 2000. TSTF-334 was received by the staff on June 23, 1999. It proposed specific changes to the Standard Technical Specifications (STS) to provide guidance and facilitate licensees' applications to implement the extended EFCV test intervals proposed in the topical report. It was approved by the staff by W.D. Beckner's letter to A. R. Pietrangelo (NEI) of October 31, 2000. In the final revision of TSTF-334, applicability was limited to those facilities that are encompassed by the analyses performed in support of the topical report, one of which was the Columbia Generating Station, and are subject to performance and corrective action criteria to be developed by the licensee.

3.0 EVALUATION

3.1 Relocation of Containment Atmosphere and Suppression Pool Instrumentation Lines EFCVs from SR

The testing done on the reactor instrument line EFCVs is meant to simulate a line break outside containment during normal operation. The reactor instrument line is opened up outside containment and high-pressure water begins to flow out from the reactor coolant system. The high flow closes the EFCV, demonstrating that it works to limit the loss of reactor coolant into the environment.

The containment atmosphere and suppression pool instrument line EFCVs are not tested in this way, nor were they meant to be. A break of one of these instrument lines during normal operation would not establish conditions that would cause the EFCV to operate, as neither containment pressure nor suppression pool head would be sufficient to cause its actuation. Further, the containment atmosphere instrument lines are not postulated to ever contain water. The containment atmosphere and suppression pool instrument line EFCVs are not designed to limit the loss of reactor coolant. The Improved Standard Technical Specifications testing requirements are intended to apply only to the reactor instrument line EFCVs, and the containment atmosphere and suppression pool instrument line EFCVs should not have been included.

Before the Columbia Generating Station TSs were converted to the Improved Standard Technical Specifications, the containment atmosphere and suppression pool instrument line EFCVs were not included in the SR. The Improved Standard Technical Specifications also do not include them. The staff finds that including the containment atmosphere and suppression pool instrumentation line EFCVs in the Columbia Generating Station TSs during the licensee's implementation of the Improved Standard Technical Specifications was not necessary and went beyond the guidelines of the Improved Standard Technical Specifications and the requirements of 10 CFR 50.36(c)(3).

Therefore, the staff finds that the proposed removal of containment atmosphere and suppression pool instrument line EFCVs from proposed SR 3.6.1.3.8 with their testing controlled by plant processes governed by 10 CFR Part 50.59 to be acceptable.

3.2 Relaxation of SR Surveillance Frequency

The postulated break of an instrument line attached to the reactor coolant boundary is discussed and evaluated in the Columbia Generating Station, Final Safety Analysis Report (FSAR), Subsection 15.6.2, "INSTRUMENT LINE PIPE BREAK." The analysis assumes the break occurs at a point where the instrument line is not isolated, that immediate detection is not automatic or apparent, and there is a continuous discharge of reactor water through the instrument line until the primary system is cooled down and depressurized. The failure is assumed to occur outside the drywell but inside secondary containment. The line size and the restricting orifice in the line minimize leakage from a break postulated to occur upstream of the EFCV. Previous licensee evaluation of such an instrument line rupture did not take credit for the mitigating action of the EFCV and is bounded by the steam line break analysis. The integrity and functional performance of the secondary containment and standby gas treatment system are not impaired by this event, and the calculated potential offsite exposures are substantially below the requirements of 10 CFR Part 100. Therefore, a failure of a reactor instrument line EFCV, though not expected as a result of this TS change, is bounded by the previous evaluation of an instrument line break. The radiation dose consequences of such a break are not impacted by the proposed change.

The Columbia Generating Station SR 3.6.1.3.8 currently requires a demonstration that each EFCV is operable by verifying that the valve checks flow on a simulated instrument line break downstream of the valve every 24 months. The sentence in TS SR 3.6.1.3.8 will be revised to read, "Verify a representative sample of reactor instrument line EFCVs actuate to the isolation position on an actual or simulated instrument line break signal."

The term "representative sample," as proposed by the topical report and TSTF-334, is not defined in the TS itself. However, the BWROG in response to the staff RAI stated that the term "representative sample" with an accompanying explanation in the TS Bases is identical to the current usage in the STS, NUREG-1433, Revision 1. Specifically, NUREG-1433 uses the term "representative" in TS SR 3.8.6.3 in reference to battery cell testing, and "representative sample" in SR 3.1.4.2 for verification of control rod scram times. The criterion for "representative sample" and the basis for the nominal 10 year testing interval are provided in the licensee submittal, which are similar to Insert 1 and Insert 2 stated in the staff's approved TSTF-334, Revision 2. Therefore, the application of a "representative sample" for the EFCVs testing SR, with its accompanying explanation in the TS Bases, is consistent with TSTF-334, Revision 2 to the STS usage and is acceptable to the staff.

Licensees make changes to their Bases without need for prior NRC review or approval. Nevertheless, the licensee included in its submittal, for information, revised Bases for SR 3.6.1.3.8. The revised Bases state:

> This SR requires a demonstration that a representative sample of reactor instrument lines excess flow check valves (EFCVs) are operable by verifying that each tested valve actuates to the isolation position on an actual or simulated instrument line break condition. The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). In addition, the EFCVs in the sample are representative of the various plant configurations, models, sizes and operating

> environments. This ensures that any potentially common problem with a specific type or application of EFCV is detected at the earliest possible time. This SR provides assurance that the reactor instrument line EFCVs will perform as designed. The excess flow check valves in reactor instrument lines are tested by providing an instrument line break signal with pressure at 85 psig to 110 psig. Testing within this pressure range provides a high degree of assurance that these valves will close during an instrument line break while at normal operating pressure.
>
> The 24-month frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The nominal 10 year interval is based on performance testing. Furthermore, EFCV failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint. (Reference 5). In addition, due to operational concerns, the Surveillance should not be performed during Modes 1, 2, or 3. This restriction has been established to limit the thermal cycles at the containment penetration.

The staff noted that the topical report does not provide a specific failure feedback mechanism, but does state that a plant's corrective action program must evaluate equipment failures and establish appropriate corrective actions. The BWROG responded to the staff RAI question concerning failure feedback by stating that each licensee who adopts the relaxed surveillance intervals recommended by the topical report should ensure that an appropriate feedback mechanism responsive to EFCV failures is in place.

The licensee stated that any future reactor instrument line EFCV failure will be evaluated in the Columbia Generating Station Corrective Action Program. Also, the Columbia Generating Station 10 CFR 50.65 Maintenance Rule Program was revised to provide a means to track the performance of EFCVs under specific categories. To ensure EFCV performance remains consistent with the extended test interval, minimum acceptance criterion has been established by the licensee. The "Maintenance Preventable Functional Failure Reliability" criterion for reactor instrument line EFCVs has been established as no more than one failure per two year rolling period to ensure that the EFCV performance remains consistent with the extended surveillance interval assumptions and adverse trends in EFCV performance are identified. The staff considers the licensee proposed EFCV performance criterion and basis for a 10-year interval to be in conformance with TSTF-334, Revision 2 and that they will provide a meaningful feedback for appropriate corrective actions and, thus, are acceptable.

To estimate the release frequency initiated by an instrument line break, two factors are considered: (1) the instrument line break frequency downstream of the EFCV, and (2) the probability of the EFCV failing to close. The topical report calculated an instrument line break frequency based on a WASH-1400 small pipe break failure rate of 6.1E-12 per hour/per foot of line. The topical report assumed 100 feet for each instrument line, which resulted in a frequency of 5.34E-06 breaks per year for a single instrument line. The topical report provided

an EFCV composite failure rate based on BWR plant data. The data represented 12,424.5 valve years of operation with a total of 11 failures noted. The EFCV composite failure rate was 1.67E-07/hour and was referenced as the "upper limit" failure rate in the topical report.

In the review of the topical report, the staff noted the BWROG assumed the EFCV failure rate was constant over time and did not account for potential age-related degradation in the EFCV failure rate. Additionally, the staff questioned the use of an instrument line break frequency based on WASH-1400 and not on more current data. The BWROG RAI response included an updated instrument line failure frequency of 35.2E-06 failures/year based on the Electric Power Research Institute's (EPRI) Technical Report No. 100380, "Pipe Failures in U.S. Commercial Nuclear Power Plants," July 1992. This value is 6.6 times greater than the value calculated in the topical report using WASH-1400 data. The BWROG RAI response also assumed the observed EFCV failures were five times the actual observed number (55 vs.11) listed in the topical report. The additional impact of an increase in instrument line failure frequency and a fivefold increase in EFCV failures assumed by the BWROG response demonstrated that the topical report EFCV release frequency remained low with limited impact on release frequencies.

The Columbia Generating Station reactor EFCV failure rates were updated from those listed in the topical report based on additional testing performed since the topical report was issued. Specifically the data now represents 15 years of operation with one EFCV failure noted. Employing the updated EPRI instrument line failure rate to the Columbia Generating Station specific data (one EFCV failure, 96 valves, and 1.26E+07 hours operating time) the 24-month and 10-year total plant release frequency is estimated at 1.11E-05 release/year and 5.58E-05 release/year respectively. The 10-year release frequency shows an increase of 4.47E-05 over the 24-month value. This represents the increase in the total plant release frequency for a random break of any of the 96 instrument lines in Columbia Generating Station and a concurrent failure of the EFCV to isolate the break. The Columbia Generating Station release frequencies compare favorably with the adjusted topical report total plant composite release frequencies. Additionally, if the topical report composite industry failure/operating times are applied to the EFCVs installed at Columbia Generating Station, the resulting release frequencies are consistent with the adjusted release frequency results of the topical report and staff SE. Based on the above, the staff considers the increase in estimated EFCV release frequency for a 10-year surveillance interval to be sufficiently low. This is based on the qualitative analysis that an instrument line break with a concurrent failure of an EFCV to close is not a significant contributor to core damage accidents. Based on the above, the estimated increase in the 10-year release frequency is not considered significant. Therefore, the Columbia Generating Station plant results are consistent with the topical report results and staff SE conclusions and are therefore acceptable.

The methodology used by the topical report for assessing the impact of an EFCV surveillance test interval increase to 10 years is consistent with industry practice, accounts for potentially unknown changes in EFCV failure rates, and is therefore acceptable to the staff. The staff notes that the use of observed industry data for instrument line break and plant specific EFCV failure data is adequate for assessing the proposed surveillance interval revisions. The Columbia Generating Station EFCV failure rates are consistent with the industry data and with the results noted by the staff in the topical report. Based on the topical report results the staff did not consider the estimated increase in release frequency for Columbia Generating Station to be significant.

The operational impact of an EFCV failing to close during the rupture of an instrument line connected to the reactor pressure vessel (RPV) boundary is based on environmental effects of a steam release in the vicinity of the instrument racks. The environmental impact of the failure of instrument lines connected to the RPV pressure boundary is the released steam into the reactor building. The topical report stated that the magnitude of release through an instrument line would be within the pressure control capacity of reactor building ventilation systems and that the integrity and functional performance of secondary containment following an instrument line break would be met. The licensee's analysis confirmed that an instrument line rupture outside primary containment will not result in overpressurizing secondary containment. The separation of instrument lines and equipment in the reactor building is expected to minimize the operational impact of an instrument line break on other equipment due to jet impingement. The licensee's analysis assumes plant shutdown, depressurization, and cooldown occur after the line break.

The radiological consequences for an instrument line break evaluated by the licensee do not credit the EFCVs for isolating the break. The evaluation assumed a discharge of reactor water through an instrument line with a ½ inch restricting orifice during the detection and cooldown sequence. The assumptions of the accident analysis do not change as a result of the licensee's proposed EFCV surveillance intervals. As a result, a failure of an EFCV is bounded by the licensee's previous analysis and is consistent with the topical report results. The radiation dose consequences for an instrument line break are not impacted by the proposed change. Therefore, the NRC staff finds the proposed amendment will not involve a significant increase in the consequences of an accident previously evaluated.

As demonstrated in General Electric Nuclear Energy Topical Report B21-00658-01, the impact of an increase in EFCV surveillance test interval to 10 years results in an instrument line release frequency considered by the staff to be sufficiently low, especially since the consequences of an EFCV failure are bounded by previous licensee analysis and therefore are highly unlikely to lead to core damage. Additionally, the licensee's evaluation results including the plant specific EFCV failure data and release frequency is consistent with the topical report composite results. The staff concludes that the release frequency associated with the Columbia Generating Station request for relaxation of ECFV surveillance testing is sufficiently low and therefore acceptable.

The consequences of steam release from the failure of the EFCVs is not significant, as shown by the topical report, and previous licensee analysis. Based on the acceptability of the methods applied to estimate the release frequency, the licensee's relatively low release frequency estimate, the negligible consequence of a release in the reactor building, in conjunction with a highly unlikely impact on core damage, the staff concludes that the impact on risk associated with the Columbia Generating Station request for relaxation of EFCV surveillance testing is also sufficiently low and is acceptable.

The topical report established that each plant's corrective action program must evaluate equipment failures and establish appropriate corrective actions. These programs ensure that meaningful feedback data is acquired so that appropriate corrective action may be taken with regard to EFCV performance. The licensee's EFCV performance criterion and EFCV corrective

action program are in conformance with staff-approved TSTF-334, Revision 2, and are, therefore, acceptable by the staff.

Based on the above, the staff finds the change to SR 3.6.1.3.8 reactor instrument line EFCV surveillance frequency by allowing a representative sample of EFCVs to be tested every 24 months with all EFCVs being tested at least once every 10 years (nominal) and the removal of the containment atmosphere and suppression pool instrument line EFCVs from SR 3.6.1.3.8 and their testing controlled by plant processes governed by 10 CFR Part 50.59 to be 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 surveillance requirements. 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 (65 FR 71135). 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.

Principal Contributors: Cliff Doutt
James Pulsipher
N. Le

Date: February 20, 2001