June 19, 2002

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: (TAC NO. MB1777)

Dear Mr. Parrish:

The Commission has issued the enclosed Amendment No. 176 to Facility Operating License No. NPF-21 for the Columbia Generating Station. The amendment consists of changes to the description of the facility in the Final Safety Analysis Report (FSAR) in response to your application dated April 16, 2001, as supplemented by letters dated November 8, 2001, and February 11, 2002.

The amendment authorizes changes to the FSAR to allow an unisolable drain line between the reactor core isolation cooling and the control rod drive/condensate pump rooms and identify the pump room doors and penetration seals that are not watertight. In addition, the change documents the minimum acceptable safe shutdown equipment.

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/

John Hickman, Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures:
1. Amendment No. 176 to NPF-21
2. Safety Evaluation

cc w/encls: See next page

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Enclosures: 1. Amendment No. 176 to NPF-2
2. Safety Evaluation

cc w/encls: See next page

*SE dated 4/30/02

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Columbia Generating Station

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ENERGY NORTHWEST

DOCKET NO. 50-397

COLUMBIA GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 176
License No. NPF-21

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

   A. The application for amendment by the Energy Northwest (licensee) dated April 16, 2001, as supplemented by letters dated November 8, 2001, and February 11, 2002, 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, by Amendment No. 176, the license is amended to authorize revision of the Final Safety Analysis Report (FSAR) as set forth in the application for amendment by Energy Northwest dated April 16, 2001, and supplements dated November 8, 2001, and February 11, 2002. Energy Northwest shall update the FSAR as authorized by this amendment in accordance with 10 CFR 50.71(e).

3. The license amendment is effective as of its date of issuance and shall be implemented in the next periodic update to the FSAR in accordance with 10 CFR 50.71(e). Implementation of this amendment is the incorporation into the FSAR the changes to the description of the facility as described in the licensee's application dated dated April 16, 2001, and supplements dated November 8, 2001, and February 11, 2002, and evaluated in the staff's safety evaluation attached to this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Date of Issuance: June 19, 2002

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 176 TO FACILITY OPERATING LICENSE NO. NPF-21

ENERGY NORTHWEST

COLUMBIA GENERATING STATION

DOCKET NO. 50-397

1.0 INTRODUCTION

By application dated April 16, 2001, as supplemented by letters dated November 8, 2001 and February 11, 2002 (References 1, 2, and 3, respectively), Energy Northwest (the licensee) requested changes to the Columbia Generating Station (CGS) Final Safety Analysis Report (FSAR). The proposed changes would allow an unisolable drain line between the reactor core isolation cooling (RCIC) and the control rod drive/condensate (CRD/COND) pump rooms and permit pump room doors and penetration seals that are not watertight. In addition, the change establishes the minimum compliment of safe shutdown equipment that is necessary for flood mitigation.

The supplements dated November 8, 2001, and February 11, 2002, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on May 16, 2001 (66 FR 27175).

The NRC had identified in Inspection Report 50-397/00-10, that the RCIC and CRD/COND pump rooms were not consistent with the FSAR description. This identification resulted in the licensee submitting this license amendment request to resolve the issue.

Specifically, the proposed changes would revise the FSAR to:

1. Allow an unisolable drain line between the RCIC and the CRD/COND pump rooms and permit pump room doors and penetration seals that are not watertight. This change involved a revision to the flooding analysis.

2. Document the minimally acceptable safe shutdown equipment that is necessary for flood mitigation.

2.0 EVALUATION

The licensee in its application identified 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 4, "Environmental and Dynamic Effects Design Bases," as the applicable regulatory requirement, which requires flooding protection for equipment important to safety. The licensee evaluated this change against the guidance provided in the acceptance criteria of Branch Technical Position (BTP) ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment" (Reference 4). The regulatory requirement for which the staff based its acceptance is GDC 4.

2.1 Revised Flooding Analysis

In response to a request for additional information (RAI) dated September 24, 2001 (Reference 5), the licensee in a letter to the NRC dated November 8, 2001, provided detailed information regarding the flooding analysis performed by them that was revised to take into account the connection (unisolable 3-inch equipment drain line) between the RCIC pump room and the CRD/COND pump room and 4 gallons per minute (gpm) door seal leakage. The licensee has sought to demonstrate that the facility is in conformance with BTP ASB 3-1, and therefore is in compliance with GDC 4. BTP ASB 3-1 states that postulated piping failures in fluid systems should not cause a loss of function of essential safety-related systems. BTP ASB 3-1 defines essential systems (and components) as, "Systems and components required to shut down the reactor and mitigate the consequences of a postulated piping failure, without offsite power."

The licensee’s revised flooding analysis states that the limiting flooding event inside either the RCIC or CRD/COND pump room results from a postulated moderate energy crack in an 18-inch turbine service water (TSW) line located in the RCIC pump room, with a flow rate of 363.70 gpm. The flooding in the RCIC pump room floods into the CRD/COND pump room through the 3-inch unisolable equipment drain line connecting the two rooms. The licensee stated that the connecting drain line effectively increases the floor area subject to flooding which reduces the rate of water level increase in the rooms, allowing for more time for operator action to terminate the event.

In addition to the direct connection (3-inch unisolable drain line) between the RCIC pump room and the CRD/COND pump room, the RCIC pump room contains a floor drain radioactive (FDR) line that is connected directly to the residual heat removal (RHR) A pump room through automatic isolation valve FDR-V-607. Similarly, the floor drain located in the CRD/COND pump room is connected to a sump located in the high pressure core spray (HPCS) pump room through a drain line containing automatic isolation valve FDR-V-608. Single failure of a drain line isolation valve to close results in additional flooding in the connected room. The revised flooding analysis indicates that flooding from the TSW line pipe crack in the RCIC pump room affects equipment and cabling in the RCIC pump room and potentially affects equipment and cabling located in or passing through the CRD/COND pump room, as well as rooms that are adjacent as a result of door seal leakage. The licensee provided detailed information on the components located inside the pump rooms and equipment connected to cables passing through these rooms that can potentially be affected by the TSW pipe crack.

The staff reviewed the revised flooding analysis and determined that it incorporated appropriate single failure assumptions as well as other pertinent assumptions regarding the short-circuiting of equipment which conservatively results in the worst case flood. The revised flooding analysis incorporates: worst case single failures, flood detection method, event termination time, resultant water level flood height in adjacent or connected pump rooms, major equipment directly or indirectly affected by the flood that could be used for safe shutdown, and the safe shutdown equipment that remains unaffected.

The revised flooding analysis evaluates four scenarios which result from the postulated moderate energy pipe crack of the TSW line in the RCIC pump room. The scenarios each assume a different single failure and a different flood detection method. Pathways from the RCIC pump room are through equipment drain lines (with assumed single failure of the associated isolation valve) and/or leakage through pump room door seals which are designed to minimize leakage but not to be completely watertight. In each case RCIC, HPCS, LPCS, and RHR C systems are assumed to be rendered inoperable due to short circuiting of cabling for these systems located in the RCIC pump room. In each case the automatic depressurization system/safety relief valves (ADS/SRVs) remain unaffected, as does supporting service water for remaining RHR equipment. The most limiting case leaves only ADS/SRV, RHR A or RHR B, supporting service water, and other necessary supporting systems unaffected by the flooding.

The licensee has stated that the terms "water-resistant" and "watertight," as used in the FSAR to describe barriers such as pump room penetrations and doors located in the reactor building, 422 foot elevation, are not an accurate description of the performance of these components during a flooding event. The reactor building pump room penetrations and doors, in fact, have seals that are designed to minimize leakage, i.e., minimize flooding between rooms to approximately 4 gpm even with significant hydrostatic pressure generated from flooding water levels. The water leakage past these seals has been modeled in the licensee’s revised flooding analysis described above. The licensee intends to replace the terms "watertight" and "water-resistant" in the FSAR with discussion that more accurately describes the function of the seals as one that inhibits and minimizes leakage, and thereby serves to minimize the effects of flooding.

The staff's review of the revised flooding analysis indicates that it appropriately incorporates conservative assumptions concerning equipment lost due to flooding pathways and the shorting of cables located within or passing through flooded rooms. It is the staff’s understanding that these assumptions, made in addition to pertinent single failures, combine to conservatively analyze all appropriate scenarios. The staff concludes therefore, that the analysis is appropriate and acceptable for determining the worst case flood. The staff evaluated the equipment remaining after the worst case flood in the following section of this SE to confirm that it can in fact provide a viable safe shutdown pathway.

2.2 Safe Shutdown Path

The following list, furnished in the licensee’s application, represents the minimum complement of equipment that will remain available for flood mitigation. This list reflects the results of the

CGS flooding safe shutdown analysis for all high and moderate energy line breaks outside containment, assuming the worst case single active component failure.

- pipe break/crack detection
- automatic, high-energy line break isolation equipment
- reactor protection system for scram function
- main steam isolation valves
- five safety relief valves for reactor vessel depressurization
- a single residual heat removal loop with a heat exchanger (loop A or B) in the alternate shutdown cooling mode providing reactor vessel inventory makeup (short term cooling) and reactor vessel and suppression pool (long term) cooling
- supporting [reactor building] service water [for residual heat removal loops A or B]
- supporting heating, ventilation, and air conditioning
- supporting electrical power sources

The licensee is proposing that this list henceforth constitute the licensing basis acceptance limit for meeting the acceptance criteria of BTP ASB 3-1. The equipment remaining after the worst case flood evaluated in the previous section will be evaluated in this section to determine if it is sufficient to provide a safe shutdown path. This scenario was utilized by the licensee to formulate the above list along with other flooding events.

The method proposed to achieve safe shutdown (alternate shutdown cooling) with the equipment remaining from the RCIC pump room flood evaluated in the previous section is to depressurize the reactor with the use of safety relief valves and to provide adequate short- and long-term cooling with the use of one loop of RHR with a heat exchanger, i.e., RHR A or B. This method of alternate shutdown cooling, utilizing only one RHR pump for both core injection and suppression pool cooling, has been analyzed in the CGS FSAR Section 15.2.9, "Failure of Residual Heat Removal Shutdown Cooling." In the scenarios analyzed in this FSAR section, high pressure systems (either RCIC, HPCS, or both) are available during the early stage (approximately first two hours) of reactor shutdown. This is not the case with the flooding scenario in the RCIC and CRD/COND pump rooms. The analysis has shown the RCIC and HPCS systems to be unavailable as a result of the flood. However, the FSAR Section 15.2.9 analysis shows that alternate shutdown cooling utilizing only one train of RHR is a viable cooldown method once the reactor has been depressurized enough to inject with a low pressure pump. Analysis performed by GE Nuclear Energy entitled "Original Safe Shutdown Paths for the BWR," GE-NE-T43-00002-00-01-R01, Revision 1, August 1999 (Reference 6), has demonstrated the viability of a safe shutdown pathway using safety relief valves to depressurize the reactor without a high pressure system to provide makeup, to a point at which low pressure systems can inject and perform alternate shutdown cooling. This "GE Nuclear" document, and method of achieving safe shutdown therein, has been endorsed by the NRC by a memorandum dated September 21, 1999, entitled "SRXB Position On Use of SRVs and Low Pressure Systems as Appendix R 'Redundant' Systems" (Reference 7). Further, it was shown in another GE Nuclear document entitled "BWROG Position on the Use of Safety Relief Valves and Low Pressure Systems as Redundant Safe Shutdown Paths," GE-NE-T43-00002-00-03-R01, Revision 1, August 1999 (Reference 8) that the use of the above shutdown path was a viable pathway for achieving redundant and/or alternate safe shutdown in accordance with the requirements of 10 CFR 50 Appendix R, Sections III.G.1, 2 and 3. It is noted here that this method accepts a short period of core uncovery and that this document and shutdown path was accepted by the NRC in a letter to the BWR Owners Group dated December 12, 2000, entitled "BWR Owners Group Appendix R Fire Protection Committee Position on SRVs + Low Pressure Systems Used as 'Redundant' Shutdown Systems Under Appendix R (Topical Report GE-NE-T43-0002-00-03-R01) (TAC NO. MA8545)" (Reference 9).

These analyses show that the reactor can safely be depressurized using SRVs without a high pressure system for makeup to the reactor (GE Nuclear analysis), to a point at which one train of RHR with a heat exchanger (analyzed in FSAR Section 15.2.9) can cool the reactor to cold shutdown. These analyses, previously accepted by the NRC, demonstrate that the licensee's safe shutdown pathway, utilizing the equipment remaining after the worst case flood evaluated in the previous section, is viable. Also, the equipment used for this safe shutdown path is represented in the above list. Accordingly, the above list is hereby accepted as the Licensing Basis Acceptance Limit for all high and moderate energy flooding events outside containment.

The staff also finds that the 3-inch unisolable drain line connecting the RCIC and CRD/COND pump rooms, and door and penetration seals currently in-situ at the facility may remain because the guidance of BTP ASB 3-1 regarding the loss of function of safety-related systems (viable safe shutdown path available after accounting for equipment lost due to flood) has been met which demonstrates compliance with GDC 4.

2.3 Probabilistic Safety Assessment

The licensee evaluated the risk from flooding scenarios that initiate in the CRD/COND room and flood the RCIC room through the unisoable drain connecting the two rooms. The licensee did not evaluate the risk from flooding scenarios that initiate in the RCIC room and flood the CRD/COND room. As discussed below, the licensee stated that no systems, structures, and components (SSCs) that are expected to fail in the CRD/COND due to a flood are credited in the probabilistic risk analysis (PRA). Therefore, floods that initiate in the RCIC room and propagate to the CRD/COND room have no quantifiable risk implications and the identified scenario adequately addresses the change in risk.

In Reference 3, the licensee reported that flooding of the CRD/COND pump room will not fail any SSCs that are credited in the PRA. SSCs that are expected to fail, such as the control rod drive pumps, are not credited in the PRA. There are a variety of cables that pass through the CRD/COND room but PRA evaluation of flooding events does not require the assumption that cables will fail when submerged in water for the relatively short period of time of a flood in a power plant. The RCIC pump room contains cables and the RCIC pumps and valves. The RCIC SSCs are expected to fail when submerged. Reference 3 states that there are no other SSCs in the RCIC room that are credited in the PRA and are expected to fail due to flooding.

There are three sources of large floods in the CRD/COND room, two service water pipes and a condensate water system transfer pipe. Smaller flood sources will have the same eventual impact but over a much longer time interval. The unisolated drain line will allow water from a pipe rupture in the CRD/COND room to flow into the neighboring RCIC room. Consequently, flooding of the RCIC room through the drain line from the CRD/COND room will cause the

RCIC system to fail. The only other failure caused by the flood is loss of the function supported by the ruptured line itself.

The licensee estimated the core design frequency (CDF) due to flooding in the CRD/COND room caused by the rupture of the service water pipes to be 1.56E-9/year and 6.6E-10/year for train A and B respectively. Each rupture will fail the ruptured train and the RCIC system. The rupture of the condensate storage tank pipe segment causes the loss of backup water supply to the high pressure core spray system and fails the RCIC system. The CDF is estimated to be 1.18E-10/year. The CDFs are very small because flooding of the rooms fails only the ruptured train and the RCIC system.

The licensee used a likelihood of pipe rupture of 7.3E-6/yr for SW train A, 3.3E-6/year for SW train B, and 4.0E-4/year for the condensate storage tank (CST) pipe segments. The licensee provided estimates of the number of welds, length of piping, and potential degradation mechanisms in the CRD/COND room. With this information, rupture frequencies can be estimated using methods and parameters approved for use in risk-informed request for relief from ASME inspection requirements (Reference 10). Using the approved methods and parameters, rupture frequencies two to four times higher for the SW and about one order of magnitude lower for the condensate can be estimated. These differences have a negligible impact on the estimated risk given the absolute magnitude of risk estimates is in the 1E-9/year to 1E-10/year range. Consequently, there is a negligible impact on the change in risk estimates.

2.4 Summary

Very few SSCs credited in the PRA are lost due to flooding of these two rooms. The loss of such a small number of SSCs coupled with inherently low pipe rupture frequencies result in very small risk estimates and, consequently, negligible change in risk estimates. The staff finds that any increased risk associated with the drain remaining unisolated versus isolated is negligible and within the acceptable change in risk guidelines in Regulatory Guide 1.174 (Reference 11).

Further, the staff finds the existing configuration of the unisolable drain line and door and penetration seals which exhibit minimal leakage, to be acceptable because it has been shown that the facility meets the guidance of BTP ASB 3-1 and that GDC 4 therefore is met. Accordingly, the list of equipment represented in Section 2.2 of this safety evaluation is hereby accepted as the minimum licensing basis limit for all high and moderate energy flooding events outside containment.

3.0 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.

4.0 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 (66 FR 27175). 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.

5.0 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.

6.0 REFERENCES

1. Letter, R. L. Webring (Energy Northwest), to U. S. Nuclear Regulatory Commission, "Columbia Generating Station, Operating License NPF-21, Request for Amendment, Unisolable Piping Run Between the Control Rod Drive and Reactor Core Isolation Cooling Pump Rooms," dated April 16, 2001.

2. Letter, D. W. Coleman (Energy Northwest), to U. S. Nuclear Regulatory Commission, "Columbia Generating Station, Operating License NPF-21 Request for Additional Information Regarding the Unisolable Drain Line Between the Control Rod Drive/Condensate Pump and Reactor Core Isolation Cooling Pump Rooms," dated November 8, 2001.

3. Letter, D. W. Coleman (Energy Northwest), to U. S. Nuclear Regulatory Commission, "Columbia Generating Station, Operating License NPF-21 Request for Additional Information Regarding the Unisolable Drain Line Between the Control Rod Drive/Condensate Pump and Reactor Core Isolation Cooling Pump Rooms," dated February 11, 2002.

4. BTP/ASB 3-1, "Plant Design for Protection Against Postulated Piping Failures In Fluid Systems Outside Containment," USNRC NUREG-0800, Standard Review Plan Section 3.6.1.

5. Letter, Jack Cushing (NRC), to J.V. Parish (Energy Northwest), "Request for Additional Information (RAI) for the Columbia Generating Station (TAC NO. MB1777)," dated September 24, 2001.

6. GE Nuclear Energy document entitled, "Original Safe Shutdown Paths for the BWR, GE-NE-T43-00002-00-01-R01," Revision 1, August 1999.

7. Memorandum, Jared S. Wermiel (NRC) to John N. Hannon (NRC), "SRXB Position on Use of SRVs and Low Pressure Systems as Appendix R 'Redundant' Systems (TAC NO. MA4745)," dated September 21, 1999.

8. GE Nuclear Energy document entitled, "BWROG Position on the Use of Safety Relief Valves and Low Pressure Systems as Redundant Safe Shutdown Paths," GE-NE-T43-00002-00-03-R01, Revision 1, August 1999

9. Letter, Stuart A. Richards (NRC) to James M. Kenny (Chairman, BWR Owners Group), "BWR Owners Group Appendix R Fire Protection Committee Position on SRVs + Low Pressure Systems Used as 'Redundant' Shutdown Systems Under Appendix R, (Topical Report GE-NE-T43-0002-00-03-R01) (TAC NO. MA8545)," December 12, 2000.

10. EPRI TR-112657, Revision B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," January 2000.

11. NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

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Date: June 19, 2002