ENERGY NORTHWEST
People · Vision · Solutions

P.O. Box 968 • Richland, WA • 99352-0968

August 5, 2004
GO2-04-137

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397; REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATIONS FOR ONE-TIME EXTENSION OF CONTAINMENT LEAK RATE TEST INTERVAL**

Dear Sir or Madam:

Pursuant to the Code of Federal Regulations 10 CFR 50.90, Energy Northwest hereby requests an amendment to the Columbia Generating Station (Columbia) Technical Specifications.

Specifically, the proposed amendment would modify Technical Specification (TS) 5.5.12, "Primary Containment Leakage Rate Testing Program," to allow a one-time deferral of the Type A Containment Integrated Leak Rate Test (ILRT). The current 10-year interval between Type A tests would be extended to 15 years from the previous time a Type A test was performed on July 20, 1994. This amendment will result in significant savings of dose, cost and time during Columbia's next refueling outage.

The next refueling outage at Columbia (R-17) is currently scheduled to begin May 6, 2005. In order to facilitate scheduling and avoid preparatory costs associated with conducting a Type A test during R-17, approval of this submittal is requested by February 15, 2005.

This application represents a risk informed licensing change. The proposed changes meet the criteria of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk Informed Decisions on Plant Specific Changes to the Licensing Basis."

Attachment 1 provides a description of the proposed amendment, the supporting technical analysis, the no significant hazards determination and environmental consideration. Attachment 2 and 3 provide marked-up and revised Technical Specification pages, respectively. Attachment 4 provides a risk assessment that supports the proposed Technical Specification amendment.

A017

This request for amendment has been approved by Columbia's Plant Operations Committee and reviewed by the Energy Northwest Corporate Nuclear Safety Review Board. Pursuant to 10 CFR 50.91(b), the State of Washington has been provided a copy of this amendment request.

Should you have any questions or desire additional information regarding this matter, please call Mr. DW Coleman at (509) 377-4342.

Respectfully,

RL Webring
Vice President, Nuclear Generation
Mail Drop PE08

Attachments:

1. Evaluation of the Proposed Change
2. Marked-up Affected Pages from the Technical Specifications
3. Re-typed Affected Pages from the Technical Specifications
4. ERIN Engineering Report Number C106-04-0001-5801, "Columbia Generating Station Risk Assessment to Support ILRT (Type A) Interval Extension Request"

cc: BS Mallett - NRC - RIV
WA Macon - NRC - NRR
JO Luce - EFSEC
RRCowley - WDOH
NRC Sr. Resident Inspector - 988C
RN Sherman - BPA/1399
TC Poindexter - Winston & Strawn

STATE OF WASHINGTON)
)
COUNTY OF BENTON)

Subject: Request for Amendment, Technical Specification Surveillance Interval Extension

I, RL Webring, being duly sworn, subscribe to and say that I am the Vice President, Nuclear Generation, for ENERGY NORTHWEST, the applicant herein; that I have the full authority to execute this oath; that I have reviewed the foregoing; and that to the best of my knowledge, information, and belief that the statements made in it are true.

DATE 8/5/ 2004

RL Webring
Vice President, Nuclear Generation

On this date personally appeared before me RL Webring, to me known to be the individual who executed the foregoing instrument, and acknowledged that he signed the same as his free act and deed for the uses and purposes herein mentioned.

GIVEN under my hand and seal this 5th day of August 2004

Lori A. Walli
Notary Public in and for the
STATE OF WASHINGTON

LORI A. WALLI
COMMISSION EXPIRES
NOTARY
PUBLIC
3-29-05
STATE OF WASHINGTON

Residing at West Richland

My Commission expires 3-29-05

Evaluation of Proposed Change

1.0 DESCRIPTION

This letter is a request to amend Operating License Number NPF-21 for Columbia Generating Station (Columbia) in accordance with 10 CFR 50.90, "Application for amendment of license or construction permit." The proposed amendment would modify Technical Specification (TS) 5.5.12, "Primary Containment Leakage Rate Testing Program," to allow a one-time deferral of the Type A Containment Integrated Leak Rate Test (ILRT). The 10-year interval between Type A tests would be extended to 15 years from the previous time a Type A test was performed on July 20, 1994.

This application represents a risk informed licensing change. The proposed changes meet the criteria of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk Informed Decisions on Plant Specific Changes to the Licensing Basis" (Reference 4).

The next refueling outage at Columbia (R-17) is currently scheduled to begin May 6, 2005. In order to facilitate scheduling and avoid preparatory costs associated with conducting a Type A test during the next refueling outage at Columbia, approval of this submittal is requested by February 15, 2005.

2.0 PROPOSED CHANGES

The proposed changes are summarized below. The marked-up and retyped TS pages are shown in Attachments 2 and 3, respectively.

Technical Specification 5.5.12, "Primary Containment Leakage Rate Testing Program," currently states (in part):

> "The Primary Containment Leakage Rate Testing Program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception:
>
> Compensation for flow meter inaccuracies in excess of those specified in ANSI/ANS 56.8-1994 will be accomplished by increasing the actual instrument reading by the amount of the full scale inaccuracy when assessing the effect of local leak rates against the criteria established in Specification 5.5.12.a."

The proposed amendment would add the following exception to Technical Specification 5.5.12:

> "The Primary Containment Leakage Rate Testing Program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:
>
> **The next Type A test performed after the July 20, 1994 Type A test shall be performed no later than July 20, 2009, and** compensation for flow meter inaccuracies in excess of those specified in ANSI/ANS 56.8-1994 will be accomplished by increasing the actual instrument reading by the amount of the full scale inaccuracy when assessing the effect of local leak rates against the criteria established in Specification 5.5.12.a."

In summary, the proposed change would modify Columbia Technical Specification (TS) 5.5.12, "Primary Containment Leakage Rate Testing Program," to allow a one-time deferral of the Type A Containment Integrated Leak Rate Test. The 10-year interval between Type A tests would be extended to 15 years from the previous time a Type A test was performed on July 20, 1994. A marked-up version of TS 5.5.12 showing the proposed change is included in Attachment 2.

3.0 BACKGROUND

This application for amendment represents a risk informed licensing change. The proposed changes meet the criteria set forth in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk Informed Decisions on Plant Specific Changes to the Licensing Basis." The enclosed technical justification for this request utilizes methodologies that have been found acceptable for other similar requests from LaSalle County Station and Susquehanna Steam Electric Station (References 8-12). The supporting risk assessments are included in Attachment 4.

3.1 Containment Description

Columbia is a General Electric Boiling Water Reactor (BWR) design plant (BWR-5) with a Mark II primary containment. The primary containment is part of the overall containment system which provides the capability to reliably limit the release of radioactive materials to the environs subsequent to the occurrence of a postulated Loss of Coolant Accident (LOCA) so that offsite doses will be below the "reference values" stated in 10 CFR Part 100. Its design employs an over-and-under, steel pressure vessel which houses the reactor vessel, the reactor coolant recirculating loops, and other branch connections of the reactor primary system. The pressure suppression system consists of a drywell, a pressure suppression chamber which stores a large volume of water, a connecting submerged vent system between the drywell and water pool, isolation valves, containment cooling system, and other service equipment.

The primary containment vessel is a free-standing steel pressure vessel. It utilizes the pressure suppression technique through the Mark II over-under configuration. The primary containment vessel and its appurtenances comply with the requirement of the ASME Code, Section III, Subsection NE-Class MC Components, 1971 Edition through Summer 1972 Addenda. It is designed to resist all normal operating loads, loads resulting from the postulated design basis accident as well as those loads associated with the operating basis earthquake (OBE) and safe shutdown earthquake (SSE). The design also accounts for stresses induced by thermal expansion. The drywell floor, which divides the drywell and suppression chamber, is a reinforced-concrete slab supported by steel beams and concrete columns. The drywell floor to primary containment vessel gap is closed off by means of a floor seal. This configuration permits unrestrained expansion of the containment shell under differential thermal expansion and pressure loadings. The containment vessel is enclosed in a reinforced-concrete biological shield wall for shielding purposes and is separated from the reinforced concrete by an annulus of compressible isolation material, approximately 2 inches thick. The concrete wall thickness is governed by shielding requirements but also serves as a support for the reactor building floors. Shielding over the top of the drywell is provided by removable, segmented, reinforced-concrete shield plugs. The drywell is located directly above the suppression chamber. The drywell configuration is basically a frustum of a cone with removable ellipsoidal top closure head. The suppression chamber is cylindrical with an ellipsoidal base. The primary containment vessel is anchored to the concrete mat foundation.

The bottom of the suppression chamber is lined on the inside with reinforced concrete. The concrete mat foundation under the suppression chamber is a common foundation supporting the steel primary containment vessel, including all equipment and structures therein, and the reactor building of which the primary containment vessel is a part.

The physical dimensions of the steel primary containment vessel are as follows:

- The diameter of the cylindrical portion at the base of the cone is approximately 86 feet,
- The diameter at the top of the cone is approximately 39.5 feet and then narrows to 32 feet to carry the removable head,
- Ellipsoidal bottom head with a ratio of 2:1 has an inside height of approximately 21.5 feet,
- The removable ellipsoidal top closure head has an inside height of approximately 15.5 feet,
- The drywell shell height is approximately 99 feet,
- The suppression chamber shell height is approximately 72 feet, and
- Overall shell height is approximately 171 feet.

The primary containment vessel shell plate thicknesses vary. Typical thicknesses are as follows:

- Bottom ellipsoidal head: from 7/8 inches to 1-1/2 inches,
- The suppression chamber cylinder: from 1-5/16 inches to 1-1/2 inches,
- The drywell conical section: from 3/4 inches to 1-1/2 inches, and
- The removable top ellipsoidal head: 15/16 inches.

Material thicknesses meet requirements of the ASME Code Section III, Paragraphs NE-3133 and NE-3324.

The primary containment vessel is reinforced with internal vertical and horizontal stiffeners to satisfy design requirements of the various loading combinations and conditions. Circumferential rings are attached to the inside face of the primary containment vessel. The basic function of these rings is to support pipe whip protection framework and to adequately distribute pipe whip loading into the vessel.

A general description of the primary containment is contained in Columbia's Final Safety Analysis Report (FSAR) Section 1.2.2.5.9, "Primary Containment." A description of the structural design of the primary containment structure is contained in Columbia's FSAR Section 3.8.2.1, "Description of Primary Containment Vessel." A detailed description of the functional design of the primary containment is provided in Columbia's FSAR Section 6.2, "Containment Systems."

3.2 Primary Containment Leakage Rate Testing Requirements

Containment Integrated Leak Rate Tests (ILRTs) are required for US water-cooled power reactors to ensure the public health and safety in the event of an accident that would release radioactivity into the containment.

10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," requires verification of the integrity of primary containment penetrations and isolation valves through Type B and Type C local leak rate tests (LLRTs) and the verification of overall primary containment leak integrity through Type A integrated leak rate tests. These tests are periodically performed at Columbia to verify the essentially leak-tight characteristics of the primary containment at the design basis accident pressure.

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage through the primary containment does not exceed allowable leakage rate values. The allowable leakage rate is determined so that the leakage assumptions in the plant safety analyses are not exceeded. The limitation of primary containment leakage provides assurance that the primary containment would perform its design function following an accident, up to and including the design bases accident.

On October 26, 1995, 10 CFR 50 Appendix J, Option B, "Performance-Based Requirements," became effective allowing licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance. Incorporation of 10 CFR 50 Appendix J, Option B into Columbia's Technical Specification 5.5.12 was approved by the NRC on May 8, 1996, under License Amendment No. 144 (Reference 13).

10 CFR 50, Appendix J, Option B requires that a Type A test be conducted at a periodic interval based on historical performance of the overall primary containment system. Columbia's TS 5.5.12 requires that a Primary Containment Leakage Rate Testing (PCLRT) program be established to comply with the primary containment leakage rate testing requirements of 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. Additionally, this program is required to be established in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

Regulatory Guide 1.163 endorses, with certain exceptions, Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 26, 1995, for complying with the provisions of 10 CFR 50, Appendix J, Option B. Regulatory Guide 1.163 specifies that licensees intending to comply with Option B should establish test intervals based upon the criteria of NEI 94-01, rather than using the test intervals specified in ANSI/ANS-56.8-1994.

The required frequency for Type A testing in NEI 94-01 is at least once-per-10 years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart or refueling cycles where the calculated performance leakage rate was less than 1.0 L_a) and consideration of the performance factors in NEI 94-01. NEI 94-01 specifies an initial interval of 48 months for Type A tests and allows an extension of the interval to 10 years based on two consecutive successful tests. Columbia has performed four Type A tests to date and all have met their leakage acceptance criteria. In accordance with NEI 94-01, Columbia is currently on a 10-year Type A test interval based on the two most recent successful Type A tests completed on June 9, 1991, and July 20, 1994.

With the last Type A test performed on July 20, 1994, the 10-year anniversary date for the next Type A test is July 2004. Per Option B to Appendix J, the next Type A test due date is July 2004, and no later than October 2005, with the 15-month extension allowed by NEI 94-01. Without the requested extension, the next Type A test for Columbia must be performed during the R-17 refueling outage scheduled to begin in the spring of 2005. Therefore, Columbia desires to extend the current 10-year Type A test interval to 15 years.

Extension of the Type A test interval from 10 years to 15 years will eliminate the need to perform a Type A test for Columbia during the 2005 refueling outage. Extending the test interval would save a total of approximately 1.5 person-rem of personnel exposure. This would also result in an estimated monetary savings of approximately $200,000 associated with rental equipment, and vendor support in the preparation and performance of the test. This would also save approximately 36 hours of refueling outage critical path time, with associated replacement power cost savings of approximately $1.3 million.

In addition, discussions are in progress between the NRC and NEI with the objective of promulgating a permanent extension of the 10-year Type A test interval to 20 years. The requested one-time extension of the Type A test interval would allow adequate time for implementation of this industry-wide change to the test interval through a revision to NEI 94-01.

4.0 TECHNICAL ANALYSIS

The proposed amendment would authorize a one-time extension of the Type A test interval from 10 years to 15 years for Columbia. The proposed amendment is supported by both risk and non-risk considerations.

4.1 Containment Leakage Rate Test Interval

Exceptions to the requirements of RG 1.163 are permitted by 10CFR 50, Appendix J, Option B, as discussed in Section V.B, "Implementation." Therefore, this application does not require an exemption from 10CFR 50, Appendix J, Option B.

The adoption of an Option B performance-based primary containment leakage rate testing program by Columbia did not alter the basic method by which Appendix J leakage rate testing is performed or its acceptance criteria, but it did alter the frequency of primary containment Type A, B, and C leakage tests. Test frequency is based upon an evaluation which uses the "as found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained.

The allowed frequency for Type A testing is based, in part, upon a generic evaluation documented in NUREG-1493, "Performance-Based Leak-Test Program" (Reference 3). NUREG-1493, Section 10.1.2, "Leakage Testing Intervals," made the following observations regarding changing the Type A (ILRT) test intervals and Type B and C (LLRT) test intervals.

- Reducing the frequency of Type A tests (ILRTs) from the current three-per-10 years to one-per-20 years was found to lead to an "imperceptible increase in risk." The estimated increase in risk is very small because ILRTs identify only a few potential leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have only been marginally above the existing requirements.

- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk.

- While Type B and C tests identify the vast majority (i.e., greater than 95 percent) of all potential leakage paths, performance-based alternatives to current local leakage-testing requirements are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall accident risk under existing requirements, the overall impact is very small.

Based on the above, the proposed amendment to extend the Type A test interval from once-per-10 years to once-per-15 years does not require an exemption to 10 CFR 50 Appendix J and will have a minimal impact on public risk.

4.2 Columbia Generating Station's Integrated Leak Rate History

Type A testing is performed to verify the integrity of the containment structure under conditions representing its LOCA containment peak pressure. Industry test experience has demonstrated that Type B & C testing detect a large percentage of containment leakage paths and that the percentage of containment leakage paths detected only by Type A testing is very small.

The results of Columbia previous Type A tests are shown below. No Type A tests have failed to meet their acceptance criteria at Columbia. These results demonstrate the containment structure remains an essentially leak tight barrier and presents minimal risk to increased leakage.

Table 1-1
Columbia Generating Station
10 CFR 50, Appendix J, Integrated Leak Rate Test Information

Test Date	P_a (psig) (Note 1)	Total Leakage (Note 2)	Acceptance Limit (Note 2)
02/16/84	34.7	0.2758%	0.50%
06/17/87	34.7	0.3241%	0.50%
06/09/91	34.7	0.319%	0.50%
07/20/94	38.0	0.330%	0.50%

Note 1 The value of P_a for ILRT testing is 38.0 psig, which is conservative, as the actual calculated accident pressure for Columbia is 37.4 psig. The 38.0 psig was used in the 1994 ILRT. The three previous ILRTs used a P_a of 34.7. The higher 1994 P_a was the result of power up-rate.

Note 2 Leakage rates are expressed in units of containment air weight percent per day at test pressure (P_a). Calculated results are expressed at a 95% confidence level plus leakage attributed to non-vented penetrations. The maximum allowable primary containment leakage rate allowed by Option B during containment leak rate testing is 0.50% containment air weight percent per day ($1.0L_a$).

4.3 Type B and C Testing

Type B and C testing assures containment penetrations such as flanges, sealing mechanisms and containment isolation valves are essentially leak tight. Type B and C tests identify the vast majority (i.e., greater than 95%) of all potential leakage paths.

The Type B and C testing requirements will not be affected by this proposed change to the Type A test frequency.

4.4 10 CFR 50 Containment Inspection Programs

4.4.1 Appendix J Visual Inspections

The 10 CFR 50, Appendix J program requires visual inspections to be performed of the accessible interior and exterior surfaces of the containment system for structural problems that may affect either the containment structural leakage integrity or that might affect the performance of Type A testing. These examinations are currently required to be completed before each Type A test and during two other refueling outages before the next Type A test based on a 10-year frequency. These requirements will not be changed by this amendment.

4.4.2 10 CFR 50.55a Containment Inservice Inspections

Containment integrity is also verified through periodic in-service inspections conducted in accordance with the requirements of 10 CFR 50.55a (Reference 14). As identified in 10 CFR 50.55a, the requirements for containment inspections are contained in the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI (Reference 15). More specifically, Subsection IWE provides the rules and requirements for in-service inspection of Class MC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulation 10 CFR 50.55a(b)(2)(ix)(E) requires licensees to conduct a general visual inspection of the containment in accordance with ASME Section XI during each of the three inspection periods during the 10-year interval. These requirements will not be affected by this proposed change to the Type A test frequency.

The containment inservice inspection (IWE) program at Columbia is described in detail in Volume 4 of the Inservice Inspection (ISI) Program Plan. This plan was developed in accordance with the requirements of ASME Section XI, Subsection IWE, 1992 Edition with 1992 Addenda as modified by 10 CFR 50.55a and NRC approved 10 CFR 50.55(a) requests. The program requirements include a general visual examination of the containment shell each inspection period (each ten year interval is divided into three inspection periods of three to four years). The general visual examinations of the containment shell are conducted in accordance with Quality Control Inspection Procedure (QCI) 7-4, "Visual Examination of Containment." Any indications exceeding the screening criteria contained in QCI 7-4 are provided to a qualified engineer who compares the indication to the design requirements of the containment vessel. Any indications that exceed the design requirements are documented in the Corrective Action Program and are dispositioned in accordance with the ASME code requirements. In addition to providing screening criteria, QCI 7-4 also provides the qualification requirements for personnel conducting general visual examinations. These requirements will not be affected by this proposed change to the Type A test frequency.

Inspections performed to date in accordance with the IWE program have been evaluated as acceptable. There are no outstanding inspection issues. The Columbia IWE program currently has not identified any areas requiring augmented examinations in accordance with IWE-1240.

4.4.3 Section XI Examination Categories E-D and E-G

The ASME Section XI Subsection IWE Examination Program at Columbia for seals and gaskets is in accordance with ASME Section XI 1995 Edition, 1996 Addenda Table IWE-2500-1, Examination Category E-D. The NRC approved the plant specific use of this later Code for Columbia (Reference 17). Seals and gaskets are periodically pressure tested in accordance with ASME Section XI 1992 Edition 1992 Addenda, Table IWE-2500-1, Examination Category E-P, item E9.40. This test is performed in accordance with 10 CFR 50 Appendix J (Type B).

The IWE Examination Program at Columbia for bolted connections is in accordance with ASME Section XI 1992 Edition, 1992 Addenda Table IWE-2500-1, Examination Category E-G, as modified by Code Case N-604. Code Case N-604 provides an alternative to the bolt torque or tension test described in item E8.20. This alternative includes visual inspection per Item E8.10, and pressure testing (Type B) of bolting per Appendix J of 10 CFR 50. Regulatory Guide 1.147, Revision 13 (January 2004) lists Code Case N-604 as acceptable for use without conditions.

In accordance with the Columbia's Primary Containment Leakage Rate Testing (PCLRT) Program Plan, the initial test frequency for performing leak tests on seals, gaskets, and bolted connections, which are Type B components, is at least once every 30 months. If three consecutive as-found Type B tests are less than their administrative limit, then the test interval may be extended to 120 months.

If a test result is greater than the administrative limit for the component, the component test interval is re-established at 30 months. Regardless of the above test intervals, any repair or disassembly of a component with a seal, gasket, or bolted connection requires a post-maintenance Appendix J Type B test.

There are Type B penetrations at Columbia, such as the drywell head, drywell equipment hatch, suppression chamber (wetwell) hatch, and personnel airlock door seals, which are leak tested at least every refueling outage due to their required use during Primary Containment entries and/or Refueling Outages.

In addition, Section 3.3 of Columbia's PCLRT Program Plan, states that, "some of the Type B tests that may be extended to 120 month intervals should be scheduled to be performed at intermediate outages, depending on overall work loads from other outage activities." This staggered scheduling methodology increases the likelihood for the early detection of potential generic degradation/common mode failure mechanisms. This staggered schedule has been utilized at Columbia since adoption of Option B to Appendix J.

This approved Type B testing alternative for Categories E-D and E-G (seals, gaskets, and bolting) will be performed at least once during each 10-year containment inservice inspection interval. The proposed Technical Specification amendment does not affect the current examination schedule of these components.

4.4.4 Coatings Inspections

Containment coatings are periodically inspected for discontinuities. The containment coatings have been inspected during the last two Columbia refueling outages, which began during May 2001 and May 2003. The observed general condition of the primary containment boundary coatings was good. The inspection requirements of the containment coatings will not be changed as a result of the proposed amendment.

4.4.5 Maintenance Rule Inspections

An appropriate program has been developed and implemented to meet the requirements of 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," (the Maintenance Rule). The most recent periodic assessment, per paragraph (a)(3) of the Maintenance Rule, indicates that the program for monitoring the condition and effectiveness of structures is appropriate and meets the intent of the Maintenance Rule.

The Maintenance Rule Structural Baseline Inspection program is described in Technical Services Instruction (TI) 4.23, "Maintenance Rule Structural Baseline Inspections." Past inspections have taken credit for inspections performed in accordance with Plant Procedures Manual (PPM) 8.3.38, "Drywell/Wetwell Structural Inspection." The Maintenance Rule Program also reviews all Condition Reports. Condition Reports that may involve structural aspects of the containment are evaluated to determine if there has been a functional failure. No functional failures of containment have been identified. Columbia's primary containment is currently in (a)(2) status.

4.5 <u>Uninspectable Surfaces</u>

There are inaccessible areas of Columbia's primary containment, including essentially the entire outside surface and parts of the inner surface covered or blocked by concrete (for example the bottom head).

There are no programs that monitor the condition of the inaccessible areas of the containment shell directly. However, leak-tightness of containment shell is assessed periodically by measuring humidity in the sand pocket drains located at the base of the containment vessel.

Portions of the Columbia containment shell are submerged in the suppression pool. The submerged surfaces that are not covered by concrete are accessible and are examined at the end of the containment inservice inspection interval in accordance with ASME Section XI requirements.

Inspections of some reinforced and steel containments have indicated degradation from the uninspectable (embedded) side of the steel shell and liner of primary containment. In response to previous Type A test extension request submittals, the NRC has consistently requested licensees to perform a quantitative assessment of the impact on LERF due to age-related degradation of non-inspectable areas of the containment (References 9, 18). Therefore, a quantitative assessment using the same approach used by other plants (e.g., Calvert Cliffs) is included in Attachment 4 as a sensitivity case to this Type A test extension evaluation. Appendix D to Attachment 4 provides the analysis details.

The results of the sensitivity case indicate that the increase in LERF from the 10-year Type A test interval to the 15-year test interval is 2.28E-8/year, compared with 1.98E-8/year without corrosion effects. This is still well below the Regulatory Guide 1.174 acceptance criterion threshold for "very small" changes in risk of 1.0E-7/year. Additionally, the dose rate increase is negligible compared with the total of 1.34 person-rem/year. The increase in the CCFP is determined to be insignificant (70.9% for the 15-year interval case versus 70.6% for the 10-year interval case). The results demonstrate that including corrosion effects in the Type A test risk assessment do not alter the conclusions of the original analysis.

4.6 Plant Operational Performance

During power operation, the primary containment is inerted with nitrogen. The containment inerting system is used to purge the primary containment during plant start-up, and to provide a supply of makeup nitrogen to maintain primary oxygen concentration within Technical Specification limits during power operation.

During power operation, instrument air system (i.e., nitrogen) leaks occur from pneumatically operated valves inside the containment, which gradually increase pressure inside the primary containment. Primary containment pressure is monitored and trended during plant operation, and is periodically vented in order to maintain containment pressure within an acceptable operating range. This cycling of the primary containment pressure during power operation amounts to a periodic integrated pressure test of the containment at a low differential pressure. Although this cycling does not challenge the structural and leak tight integrity of the primary containment system at post-accident pressure, it provides assurance that a gross containment leak that may develop during power operation will be detected.

This feature is a complement to visual inspection of the interior and exterior of the containment structure for those areas that may be inaccessible for visual examination.

4.7 Stainless Steel Bellows

NRC Information Notice, 92-20, "Inadequate Local Leak Rate Testing," informed licensees that two-ply stainless steel bellows can be susceptible to transgranular stress corrosion cracking and the leakage through them may not be detectable by Type B testing. No such bellows exist in the Columbia containment pressure boundary.

4.8 Risk Information

The risk analysis performed to support this submittal is contained in Attachment 4. A summary of the risk analysis and its results are contained below. NOTE: Appendix E of Attachment 4 is not applicable to this license amendment request and has not been included. Appendix E of Attachment 4 was developed to justify a future license amendment request regarding drywell to suppression chamber bypass leak rate testing.

The risk impact of a one-time extension of the Columbia Type A test interval from the currently approved 10 years to 15 years was evaluated. The results demonstrate that a change in the Type A test interval from 10 years to 15 years represents a "very small" impact on risk, as defined by Reg. Guide 1.174.

The Columbia Type A test risk assessment uses Columbia specific information to calculate the changes to the risk profile due to changes in the Type A test interval. The evaluation approach for the assessment of the risk is based on EPRI-TR-104285 (Reference 6), NEI Interim Guidance (Reference 7), and previous Type A test risk assessment submittals. The full power internal events Probabilistic Risk Assessment (PRA) model for Columbia was used as the primary basis of the assessment.

Three risk measures were evaluated using the Columbia internal events PRA model (Revision 5) to characterize the reduction in Type A test frequency from 1-per-10 years to 1-per-15 years:

- The risk impact due to a change in Large Early Release Frequency (LERF) is an increase of 2.0 E-8/year. Regulatory Guide 1.174 characterizes this risk metric as "very small" (i.e., < 1E-7/year).
- The total integrated plant risk increase measured by person-rem/year is negligible (below significant figures).
- The risk change in conditional containment failure probability is an increase of 0.1 percent, which is considered to represent a very small impact on risk.

The change in LERF is considered by Regulatory Guide 1.174 to be a "very small" impact on risk. The other two risk measure changes do not have criteria in Regulatory Guide 1.174, but based on the past Type A test interval extension requests, these changes are also considered to represent "very small" impacts on risk.

In addition, several sensitivity cases were evaluated and documented in this analysis. These sensitivity cases demonstrate the following:

- Inclusion of long-term station blackout scenarios in the EPRI categories 3a and 3b frequencies increases the risk measures a negligible amount and does not change the conclusion of this report.
- LERF is not significantly impacted by the potential for containment leakage due to age-related degradation in non-inspectable areas; the ΔLERF remains within Region III as a "very small" risk change.
- The inclusion of external events increases LERF approximately four-fold; however, the ΔLERF remains within Region III as a "very small" risk change.

Based on the above, the proposed changes to TS 5.5.12 to decrease the frequency of the Type A test at Columbia from once-per-10 years to once-per-15 years would produce an insignificant impact to the existing plant safety margin.

4.9 Impact on Previous Submittals

There is no impact on any outstanding submittal from Columbia.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration Determination

Energy Northwest has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, Issuance of amendment," as discussed below:

1. **Does the operation of Columbia Generating Station in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No

   The proposed one-time extension to the Type A testing interval from once-per-10 years to once-per-15 years will not increase the probability of an accident previously evaluated. The performance of Type A tests is not an accident initiator. The primary containment Type A testing interval extension does not involve a plant modification and will not cause equipment failure or accident initiation.

   The proposed extension to the Type A testing interval does not involve a significant increase in the consequences of an accident. The NUREG 1493 generic study of the effects of extending containment leakage testing concluded that Type B and C testing can identify the vast majority (greater than 95 percent) of potential leakage paths and that reducing the Type A test interval to once-per-20 years leads to an "imperceptible increase in risk." Other testing and inspection programs, in addition to the Type A test, provide a high degree of assurance that the primary containment integrity will be maintained. Inspections required by the Maintenance Rule and ASME Code is periodically performed in order to identify indications of containment degradation that could affect containment leak tightness.

   Experience at Columbia demonstrates that excessive containment leakage paths are detectable by Type B and C local leak rate tests. Type B and C testing will identify containment openings, such as a valve, that would otherwise be detected by the Type A test. These factors show that a one-time Type A test interval extension from once-per-10 years to once-per-15 years will not involve a significant increase in the consequences of an accident.

   Previous Type A test results at Columbia show leakage has not exceeded acceptance criteria in the past, indicating a leak-tight containment and demonstrating the structural capability of the primary containment. The testing results have established that Columbia has had acceptable containment leakage rates with considerable margin.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the operation of Columbia Generating Station in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No

   The Columbia primary containment is designed to contain energy and fission products during and after a design basis accident. The proposed extension of the Type A testing interval will not create the possibility of a new or different type of accident from any previously evaluated. There are no changes being made to the physical plant or in operation of the plant that could introduce a new failure mode with the potential to create an accident or affect mitigation of an accident.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the operation of Columbia Generating Station in accordance with the proposed amendment involve a significant reduction in the margin of safety?**

   Response: No

   The proposed extension of the Type A testing interval will not significantly reduce the margin of safety. The NUREG 1493 generic study of the effects of extending containment leakage testing found that a 20 year interval in Type A leakage testing leads to an "imperceptible increase in risk." NUREG 1493 found that generically, the design containment leakage rate contributes less than 0.1 percent to the overall accident risk and that the increase in the Type A testing interval would have a minimal effect on risk because the vast majority (greater than 95 percent) of all potential leakage paths are detected by Type B and C leakage testing.

   A Columbia plant specific probabilistic risk assessment on the change in the Type A test interval from once-per-10 years to once-per-15 years determined:

   - The risk impact due to a change in Large Early Release Frequency (LERF) is an increase of 2E-8/year that is characterized by Regulatory Guide 1.174 as "very small."
   - The total integrated plant risk increase measured by person-rem/year is negligible.
   - The change in conditional containment failure probability is an increase of 0.1 percent, which is considered to represent a very small impact on risk.

Deferral of Type A testing for Columbia does not increase the level of risk to the public due to loss of capability to detect and measure containment leakage or loss of containment structural integrity. Other containment testing methods and inspections will assure all limiting conditions for operation will continue to be met. The margin of safety inherent in existing accident analyses will be maintained.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Energy Northwest concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements

The testing requirements of 10 CFR 50 Appendix J, provide assurance that leakage through the containment, including systems and components that penetrate the containment, does not exceed allowable leakage values specified in the plant's technical specifications. Limiting containment leakage following an accident provides assurance that the containment would perform its design function following an accident up to and including the plant design basis accident.

10 CFR 50, Appendix J, was revised, effective October 26, 1995, allows licensees to perform primary containment leakage testing under either Option A "Prescriptive Requirements" or Option B. Columbia License Amendment No. 144 (Reference 13) was approved in May 1996 to permit implementation of 10 CFR 50, Appendix J, Option B. Columbia Technical Specification 5.5.12 currently requires the establishment of a Primary Containment Leakage Rate Testing (PCLRT) program in accordance with 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The PCLRT program is required by TS 5.5.12 to implement the guidelines contained in RG 1.163, which specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01, subject to several regulatory positions stated in the RG.

Exceptions to the requirements of RG 1.163 are permitted by 10CFR 50, Appendix J, Option B, as discussed in Section V.B, "Implementation," which states:

"The regulatory guide or other implementing document used by a licensee, or applicant for an operating license, to develop a performance based leakage-testing program must be included, by general reference, in the plant's technical specifications. The submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from the methods approved by the Commission and endorsed in the regulatory guide."

Therefore, this proposed amendment to the Columbia Technical Specifications does not require an exemption from 10CFR 50, Appendix J, Option B. The incorporation of the 15-year interval for Type A containment leakage rate testing, which is an exception to the 10-year Type A containment leakage rate test required by NEI 94-01, into the Columbia Technical Specifications satisfies the requirements of 10 CFR 50, Appendix J, Option B, Section V.B.

The change in the Type A test interval from once-per-10 years to once-per-15 years has no impact on the CDF. The change in LERF of 2.0 E-8/yr is considered to be a "very small" impact on risk as defined by RG 1.174. The change in population dose rate is negligible. Therefore, the change meets RG 1.174 risk acceptance guidelines.

In conclusion, based on the considerations discussed above, (1) there is a 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 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve; (i) a significant hazards consideration; (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite; or, (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.
2. Nuclear Energy Institute (NEI) Report, NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 26, 1995.
3. NUREG-1493, "Performance-Based Leak-Test Program," dated September 1995.
4. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk Informed Decisions on Plant Specific Changes to the Licensing Basis," dated July 1998.
5. American National Standard ANSI/ANS 56.8-1994, "Containment System Leakage Requirements."

6. Electric Power Research Institute, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," EPRI TR-104285, August 1994.
7. Letter from A. Petrangelo (NEI) to NEI Administrative Points of Contact, "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leak Rate Test Surveillance Intervals," November 13, 2001.
8. Letter from KR Jury (Exelon) to NRC, "Request for Amendment to Technical Specification 5.5.13, Primary Containment Leakage Rate Testing Program," dated October 24, 2002.
9. Letter from TW Simpkin (Exelon) to NRC, "Response to Request for Additional Information to Support Request for Amendment to Technical Specification 5.5.13, Primary Containment Leakage Rate Testing Program," dated June 20, 2003.
10. Letter from WA Macon (NRC) to JK Skolds (Exelon), "LaSalle County Station, Units 1 and 2, Issuance of Amendment RE; Integrated Leak Rate Test Interval (TAC Nos. MB6574 and MB6575)," dated November 19, 2003.
11. Letter from GT Jones (PPL) to NRC, "Susquehanna Steam Electric Station Proposed Amendment No. 241 to License NPF-14 and Proposed Amendment No. 206 to License NPF-22: Request for a One Time Deferral of the Type A Containment Integrated Leak Rate Test (ILRT)," dated July 30, 2001.
12. Letter from TG Colburn (NRC) to RG Bryam (PPL), "Susquehanna Steam Electric Station, Units 1 and 2 - Issuance of Amendment RE: One-Time Deferral of Containment Integrated Leak Rate Test and Drywell-to-Suppression Chamber Bypass Leakage Test (TAC Nos. MB2894 and MB2895)," dated March 8, 2002.
13. Letter from JW Clifford (NRC) to JV Parrish (Energy Northwest), "Issuance of Amendment for the Washington Public Power Supply System Nuclear Project No. 2 (TAC No. M94573)," dated May 8, 1996.
14. 10 CFR 50.55a, Codes and standards.
15. American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," ASME Section XI, 1992 Edition through 1992 Addenda.
16. Not used.
17. Letter from S Dembeck (NRC) to JV Parrish (Energy Northwest), "NRC Staff Evaluation of Inservice Inspection Program (ISI) Relief Request 2ISI-18 and 2ISI-19 - WNP-2 (TAC No MA8626)," dated June 28, 2000.
18. Letter from RG Byram (PPL) to NRC, "Susquehanna Steam Electric Station, Supplement No. 2 to Proposed Amendment No. 241 to License NPF-14 and Proposed Amendment No. 206 to License NPF-22: Request for a One Time Deferral of the Type A Containment Integrated Leak Rate Test (ILRT)," dated October 16, 2001.

Marked-up Affected Pages from the Technical Specifications

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

4. Other appropriate limitations and remedial or compensatory actions.

b. A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems described in b.1 and b.2 above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

The Primary Containment Leakage Rate Testing Program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception; INSERT A
Compensation for flow meter inaccuracies in excess of those
C S

(continued)

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

specified in ANSI/ANS 56.8-1994 will be accomplished by increasing the actual instrument reading by the amount of the full scale inaccuracy when assessing the effect of local leak rates against the criteria established in Specification 5.5.12.a.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 38 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.5% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and Type C tests (except for main steam isolation valves) and $< 0.75\ L_a$ for Type A tests;

b. Primary containment air lock testing acceptance criteria are:

   1) Overall primary containment air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$; and

   2) For each door, leakage rate is $\leq 0.025\ L_a$ when pressurized to ≥ 10 psig.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

Insert A

The next Type A test performed after the July 20, 1994 Type A test shall be performed no later than July 20, 2009, and

Re-typed Affected Pages from the Technical Specifications

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

4. Other appropriate limitations and remedial or compensatory actions.

b. A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems described in b.1 and b.2 above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

The Primary Containment Leakage Rate Testing Program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions: The next Type A test performed after the July 20, 1994, Type A test

(continued)

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

shall be performed no later than July 20, 2009, and compensation for flow meter inaccuracies in excess of those specified in ANSI/ANS 56.8-1994 will be accomplished by increasing the actual instrument reading by the amount of the full scale inaccuracy when assessing the effect of local leak rates against the criteria established in Specification 5.5.12.a.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 38 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.5% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and Type C tests (except for main steam isolation valves) and $< 0.75\ L_a$ for Type A tests;

b. Primary containment air lock testing acceptance criteria are:

   1) Overall primary containment air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$; and

   2) For each door, leakage rate is $\leq 0.025\ L_a$ when pressurized to $\geq$ 10 psig.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

ERIN Engineering Report Number C106-04-0001-5801, "Columbia Generating Station Risk Assessment to Support ILRT (Type A) Interval Extension Request"