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

March 30, 2018

Mr. Mark E. Reddemann
Chief Executive Officer
Energy Northwest
76 North Power Plant Loop
P.O. Box 968 (Mail Drop 1023)
Richland, WA 99352-0968

SUBJECT: COLUMBIA GENERATING STATION - ISSUANCE OF AMENDMENT RE: REVISION OF PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM (CAC NO. MF9469; EPID L-2017-LLA-0197)

Dear Mr. Reddemann:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 247 to Renewed Facility Operating License No. NPF-21 for the Columbia Generating Station (CGS). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 27, 2017, as supplemented by letter dated January 2, 2018.

The amendment revises CGS TS 5.5.12, "Primary Containment Leakage Rate Testing Program," in accordance with Nuclear Energy Institute (NEI) 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR [Title 10 of the *Code of Federal Regulations*] Part 50, Appendix J," July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated October 2008, which serves as the guidance document for implementation of performance-based Option B of 10 CFR Part 50, Appendix J.

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

Sincerely,

L. John Klos, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures:
1. Amendment No. 247 to NPF-21
2. Safety Evaluation

cc: Listserv

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

ENERGY NORTHWEST

DOCKET NO. 50-397

COLUMBIA GENERATING STATION

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 247
License No. NPF-21

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

   A. The application for amendment by Energy Northwest (the licensee), dated March 27, 2017, as supplemented by letter dated January 2, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and 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 Renewed 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. 247 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed 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 120 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

R. J. Pascarelli

Robert J. Pascarelli, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: March 30, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 247

COLUMBIA GENERATING STATION

RENEWED FACILITY OPERATING LICENSE NO. NPF-21

DOCKET NO. 50-397

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

Renewed Facility Operating License

REMOVE	INSERT
-4-	-4-

Technical Specification

REMOVE	INSERT
5.5-9	5.5-9
5.5-10	5.5-10

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 247 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

(3) Deleted.

(4) Deleted.

(5) Deleted.

(6) Deleted.

(7) Deleted.

(8) Deleted.

(9) Deleted.

(10) Deleted.

(11) Shield Wall Deferral (Section 12.3.2, SSER #4, License Amendment #7)

The licensee shall complete construction of the deferred shield walls and window as identified in Attachment 3, as amended by this license amendment.

(12) Deleted.

(13) Deleted.

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*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

a. The SFDP shall contain the following:

1. Provisions for cross division checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

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 NEI 94-01, "Industry Guideline for Implementing Performance-Based option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008.

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

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.

5.5.13 Battery Monitoring and Maintenance Program

This Program provides for restoration and maintenance, which includes the following:

a. Actions to restore battery cells with float voltage < 2.13 V; and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates; and

c. Actions to verify that the remaining cells are ≥ 2.07 V when a cell or cells have been found to be < 2.13 V.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 247 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-21

ENERGY NORTHWEST

COLUMBIA GENERATING STATION

DOCKET NO. 50-397

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC, the Commission) dated March 27, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17086A586), as supplemented by letter dated January 2, 2018 (ADAMS Accession No. ML18002A501), Energy Northwest (the licensee) requested changes to the Technical Specifications (TSs) for Columbia Generating Station (CGS).

The proposed change would revise CGS TS 5.5.12, "Primary Containment Leakage Rate Testing Program," to allow for a permanent extension of the Type A test interval from one test every 10 years to one test in 15 years in accordance with Nuclear Energy Institute (NEI) 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR [Title 10 of the *Code of Federal Regulations*] Part 50, Appendix J," July 2012 (ADAMS Accession No. ML12221A202), and the conditions and limitations specified in NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated October 2008 (ADAMS Accession No. ML100620847), which serves as the guidance document for implementation of performance-based Option B of 10 CFR Part 50, Appendix J.

In addition, the proposed change would revise CGS TS 5.5.12 to allow for an extension of the Type C test interval from once in 60 months to once in 75 months for containment isolation valve (CIV) local leak rate tests (LLRTs) which is the maximum interval.

The supplemental letter dated January 2, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 6, 2017 (82 FR 26131).

2.0 REGULATORY EVALUATION

The regulation in 10 CFR 50.65(a), "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," states, in part, that the licensee "...shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components, as defined in paragraph (b) of this section, are capable of fulfilling their intended functions. These goals shall be established commensurate with safety and, where practical, take into account industrywide operating experience."

The regulation in 10 CFR 50.54(o) requires primary reactor containments for water cooled power reactors be subject to the requirements set forth in Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Appendix J to 10 CFR Part 50, includes two options: "Option A - Prescriptive Requirements," and "Option B - Performance-Based Requirements," either of which may be chosen by a licensee for meeting the requirements of the Appendix.

The overall (structural and leak-tight) integrity of the primary containment is verified by a Type A integrated leak rate test (ILRT) at design-basis accident pressure, and the integrities of the penetrations and isolation valves are verified by Type B and Type C LLRT as required by 10 CFR Part 50, Appendix J.

The testing requirements in 10 CFR Part 50, Appendix J ensure that (a) leakage through containments or systems and components penetrating containments does not exceed allowable leakage rates specified in the TS; and (b) integrity of the containment structure is maintained during the service life of the containment.

Appendix J to 10 CFR Part 50 was revised in 1995 with the addition of Option B to the original Option A requirements. Option B requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. A Type A test is an overall (integrated) leakage rate test of the containment structure. Since the most recent two Type A tests performed at CGS have been successful, the current test interval requirement is 10 years.

Option B of 10 CFR Part 50, Appendix J, specifies performance-based requirements and criteria for preoperational and subsequent leakage rate testing. These requirements are met by performing Type A tests to measure the containment system overall integrated leakage rate; Type B pneumatic tests to detect and measure local leakage rates across pressure-retaining leakage-limiting boundaries such as penetrations; and Type C pneumatic tests to measure CIV leakage rates. After the preoperational tests, these tests are required to be conducted at periodic intervals based on the historical performance of the overall containment system (for Type A tests), and based on the safety significance and historical performance of each penetration boundary and isolation valve (for Type B and C tests) to ensure integrity of the overall containment system as a barrier to fission product release. The leakage rate test results must not exceed the maximum allowable leakage rate (L_a) with margin, as specified in the TS.

Section V.B.3 of 10 CFR 50, Appendix J, Option B, requires that the regulatory guide or other implementation document used by a licensee to develop a performance-based leakage-testing program must be included, by general reference, in the plant TSs. Furthermore, the submittal for TS revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide.

As part of the development of Option B, the NRC also developed Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 (ADAMS Accession No. ML003740058) to specify a method acceptable to the NRC for complying with Option B. RG 1.163 endorses, with certain exceptions, NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995 (ADAMS Accession No. ML11327A025), as an acceptable method for complying with the provisions of Appendix J, Option B. NEI 94-01, Revision 0, specifies an initial test interval of 48 months for the Type A test, but allows an extended interval of 10 years based upon two consecutive successful tests and supported by a plant-specific risk assessment. It should be noted that Section 9.1 of NEI 94-01, Revision 0, allows the recommended 10 year Type A test interval to be extended by up to an additional 15 months, but with the restriction that this option should be used only in cases where refueling schedules have been changed to accommodate other factors.

Option B requires that the regulatory guide or another implementation document used by a licensee to develop a performance-based leakage rate testing program be included by general reference in the TS. The leakage rate testing requirements of 10 CFR Part 50, Appendix J, Option B (Type A, Type B, and Type C tests) and the containment inservice inspection (CISI) requirements mandated by 10 CFR 50.55a, "Codes and standards," assist in ensuring the continued leak-tight and structural integrity of the containment during its service life.

NEI 94-01, Revisions 2 and 3, have been reviewed by the NRC and approved for use. The final safety evaluation (SE) for NEI 94-01, Revision 2, issued by letter dated June 25, 2008 (ADAMS Accession No. ML081140105), documents the NRC's evaluation and acceptance of Revision 2, subject to six specific limitations and conditions listed in Section 4.1, "Limitations and Conditions for NEI TR [Topical Report] 94-01, Revision 2. The NRC SE of NEI 94-01, Revision 3, issued by letter dated June 8, 2012 (ADAMS Accession No. ML121030286), includes two specific limitations and conditions listed in Section 4.0, "Limitations and Conditions," for the Type C tests and NEI 94-01, Revisions 2-A and 3- A, include their corresponding SEs.

NEI 94-01, Revision 3-A, states, in part, that a confirmatory "analysis is to be performed by the licensee and retained in the plant documentation as part of the basis for extending the ILRT interval." The conditions for this confirmatory analysis stipulate that:

1. The licensee submits documentation indicating that the technical adequacy of its probabilistic risk assessment (PRA) is consistent with the requirements of RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated March 2009, Revision 2 (ADAMS Accession No. ML090410014), relevant to the ILRT extension application.

2. The licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is small, and consistent with the clarification provided in Section 3.2.4.6[1] of the SE for Electric Power Research Institute (EPRI) TR-1009325, Revision 2, dated June 25, 2008.

[1] The safety evaluation for EPRI TR-1009325, Revision 2, indicates that the clarification regarding small increases in risk is provided in Section 3.2.4.5; however, the clarification is actually provided in Section 3.2.4.6.

3. The methodology in EPRI TR-1009325, Revision 2, is acceptable provided the average leak rate for the pre-existing containment large leak accident case (i.e., accident case 3b) used by licensees is assigned a value of 100 times the L_a instead of 35 L_a.

4. A license amendment request (LAR) is required in instances where containment overpressure is relied upon for emergency core cooling system (ECCS) performance.

The regulations in 10 CFR 50.36, "Technical specifications" state, in part, that the TSs include items in five specific categories. These categories include: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls.

NUREG-1433, "Standard Technical Specifications – General Electric BWR [Boiling-Water Reactor]/4 Plants," Revision 4.0, Volume 1, Specifications (ADAMS Accession No. ML12104A192), incorporated Standard Technical Specifications Task Force (TSTF) Traveler TSTF-52, Revision 3 (ADAMS Accession No. ML040400371), that provided guidance for specific changes to TSs for implementation of 10 CFR Part 50, Appendix J, Option B.

The licensee is requesting a change to TS 5.5.12, which would add an exception from the normal requirements regarding the Type A test interval. Based on Option B requirements of 10 CFR Part 50, Appendix J, a Type A test (or ILRT) must be conducted (1) after a containment system has been completed and is ready for operation and (2) at a periodic interval based on historical performance of the overall containment system. A general visual inspection of the accessible interior and exterior surfaces of the containment system for structural deterioration that may affect the containment leak-tight integrity must be conducted prior to each test, and at a periodic interval between tests based on the performance of the containment system. CGS adopted 10 CFR Part 50, Appendix J, Option B, for Type A ILRTs and Type B and Type C LLRTs by Amendment No. 144 dated May 8, 1996 (ADAMS Accession No. ML022120330).

CGS TS 5.5.12 currently reflects a one-time 5-year extension of the Type A test by Amendment No. 191, dated April 12, 2005 (ADAMS Accession No. ML050620553) and 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 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.

The proposed changes to the CGS TS 5.5.12 will replace the reference to RG 1.163 with a reference to NEI 94-01, Revisions 2-A and 3-A. The licensee justified the proposed TS changes by providing historical plant-specific containment leakage testing program results and CISI program results, and a supporting plant-specific risk assessment, consistent with the guidance in NEI 94-01, Revision 2-A.

This LAR also proposes administrative changes to two exceptions in TS 5.5.12. The exception regarding the performance of the next CGS Type A test to be performed no later than July 20, 2009, will be deleted as this Type A test has already occurred. Additionally, the exception to compensate for flow meter inaccuracies in excess of those specified in American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-1994 is being deleted as new test equipment has been acquired with accuracies within the tolerances specified in ANSI/ANS 56.8-1994 and 2002.

The proposed change will revise TS 5.5.12 to state, 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 NEI 94-01, "Industry Guideline for Implementing Performance-Based option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008.

3.0 TECHNICAL EVALUATION

3.1 Plant-Specific Risk Evaluation

The licensee performed a risk impact assessment for extending the Type A containment ILRT interval from 10 to 15 years. The risk analyses for CGS was provided in Enclosure 4 of the LAR dated March 27, 2017. Additional information was provided by the licensee in its letter dated January 2, 2018, in response to the NRC's request for additional information (RAI).

In Section 3.5 of Enclosure 1 to the LAR, the licensee stated that the plant-specific risk assessment for CGS follows the guidance in:

- NEI 94-01, Revision 3-A, July 2012;
- EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," dated August 1994;
- The NEI document "Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals," Revision 4;
- RG 1.200, Revision 2;
- RG 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," May 2011 (ADAMS Accession No. ML100910006);
- EPRI TR-1009325, Revision 2-A (also known as EPRI 1018243); and
- Calvert Cliffs Nuclear Plant liner corrosion analysis described in a letter to the NRC dated March 27, 2002 (ADAMS Accession No. ML020920100).

3.1.1 Technical Adequacy of the PRA

The first condition for EPRI TR-1009325, Revision 2, stipulates that the licensee submits documentation indicating that the technical adequacy of its PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension application.

In NRC Regulatory Issue Summary (RIS) 2007-06, "Regulatory Guide 1.200 Implementation," dated March 22, 2007 (ADAMS Accession No. ML070650428), the NRC clarified that for all risk-informed applications received after December 2007, the NRC staff will use Revision 1 of RG 1.200 (ADAMS Accession No. ML070240001) to assess technical adequacy of the PRA used to support risk-informed applications. Revision 2 of RG 1.200 will be used for all risk-informed applications received after March 2010. In Section 3.2.4.1, of the NRC final SE for EPRI TR-1009325, Revision 2, the NRC staff states, in part, that:

> Licensee requests for a permanent extension of the ILRT surveillance interval to 15 years pursuant to NEI TR-94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, will be treated by NRC staff as risk-informed license amendment requests. Consistent with information provided to industry in Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation"...the NRC staff will expect the licensee's supporting Level 1/LERF [large early release frequency] PRA to address the technical adequacy requirements of RG 1.200, Revision 1.... Any identified deficiencies in addressing this standard shall be assessed further in order to determine any impacts on any proposed decreases to surveillance frequencies. If further revisions to RG 1.200 are issued which endorse additional standards, the NRC staff will evaluate any application referencing NEI TR-94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, to examine if it meets the PRA quality guidance per the RG 1.200 implementation schedule identified by the NRC staff.

In the same section of the SE, the NRC staff stated that Capability Category I of American Society of Mechanical Engineers (ASME) PRA standard shall be applied as the standard for assessing PRA quality for ILRT extension applications, as approximate values of core damage frequency (CDF) and LERF and their distribution among release categories are sufficient to support the evaluation of changes to ILRT frequencies.

In Enclosure 4 of the LAR, the licensee stated that the most current Level 1 and Level 2 internal events and internal flooding PRA model of record was Version 7.2.1. In Section A.2 of Enclosure 4 to the LAR, the licensee stated that it performed a review of plant modifications and procedure changes to assess whether the PRA reflects the as-built and as-operated plant. The licensee provided a summary of three plant modifications scheduled to be implemented in the next refueling outage, which are not reflected in the current PRA model, but would have an impact on the PRA. The licensee assessed that one change would improve the risk profile, making the current model conservative, and the other two have minimal quantitative impact. Further, in Enclosure 4 of the LAR, the licensee described the process used to ensure that the PRA model remains an accurate reflection of the as-built and as-operated plant. The licensee has a process for continued PRA maintenance and update, including procedures for regularly scheduled and interim PRA model updates and for tracking issues identified as potentially affecting the PRA model.

The licensee stated that the internal events PRA received a peer review in 2009 against the ASME/ANS PRA Standard ASME/ANS RA-Sa-2009, as clarified by RG 1.200, Revision 2, which, as further clarified in response to RAI 1a, in the letter dated January 2, 2018, included a full-scope peer review of the internal flooding PRA. In response to RAI 1b and 1c, in the letter dated January 2, 2018, the licensee provided an overview of PRA changes after the 2009 peer review and stated that none of these changes constituted a PRA upgrade and, therefore, no further peer reviews were required.

In Table A-2, "Resolution of CGS Internal Events Peer Review Findings not Meeting Category II," of Enclosure 4 to the LAR, the licensee provided the findings and observations (F&Os) from the 2009 peer review associated with supporting requirements that did not meet or exceed Capability Category II and the licensee's resolution or disposition for the application. The NRC staff reviewed the F&Os and its associated resolutions or dispositions and determined that they have no impact on the ILRT application results. The staff asked for further clarification of two F&Os, as discussed below. Additionally, the licensee identified F&O 2-2 and these three F&Os are further discussed below:

- F&O 1-42, related to supporting requirement HR-I2, identified that a Human Reliability Analysis (HRA) dependency analysis has not been conducted for the Level 2 model. In resolution to this F&O, the licensee stated that the Level 2 HRA dependency analysis has been performed and documented. In RAI 2a, the NRC staff requested the licensee to justify why no focused-scope peer review was necessary. In response to RAI 2a, in the letter dated January 2, 2018, the licensee explained that the Level 2 human error probabilities (HEPs) dependency analysis incorporates the same methodologies as in the Level 1 HEPs dependency analysis, which was peer reviewed. The licensee further stated that the Level 2 HEPs dependency analysis meets the supporting requirements for a HRA (i.e., those denoted as HR) in the PRA Standard ASME/ANS RA-Sa-2009. Therefore, the NRC staff finds that this F&O has no impact on the application.

- F&O 2-2, that has not been resolved related to supporting requirement DA-C6, found that the number of plant-specific demands on standby components was not based on the surveillance tests and maintenance acts as described in the supporting requirement. To disposition this F&O, the licensee performed a sensitivity study by replacing the baseline model failure data with data that was informed by the surveillance tests and maintenance acts, and concluded that the finding had a negligible impact on quantitative results associated with the permanent extension of the ILRT interval. This sensitivity was presented in Section 5.4.3 of Enclosure 4 to the LAR. Because the licensee performed a sensitivity study and showed minimal impact on the results, the NRC staff finds this F&O has no impact on the application.

- F&O 2-17, related to supporting requirement SY-A14, identified inadequate consideration of system operational history in the PRA system models. In resolution to this F&O, the licensee stated that operational history has been collected and will be added to the PRA documentation. In RAI 2b, the NRC staff requested the licensee to confirm that operational history has been reflected in the PRA models, or to justify why exclusion of operational history has no impact on the application. In response to RAI 2b, in the letter dated January 2, 2018, the licensee stated that actual system operational history, including industry significant operating experience and plant specific operational history was collected and has been added to the PRA documentation and that no PRA model changes were needed to resolve F&O 2-17. Therefore the NRC staff finds that this F&O has no impact on the application.

In Section 3.2.4.2 of the NRC final SE for NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2, the NRC staff states that:

> Although the emphasis of the quantitative evaluation is on the risk impact from internal events, the guidance in EPRI Report No. 1009325, Revision 2, Section 4.2.7, "External Events," states that: "Where possible, the analysis should include a quantitative assessment of the contribution of external events (e.g., fire and seismic) in the risk impact assessment for extended ILRT intervals." This section also states that: "If the external event analysis is not of sufficient quality or detail to directly apply the methodology provided in this document [(i.e., EPRI Report No. 1009325, Revision 2)], the quality or detail will be increased or a suitable estimate of the risk impact from the external events should be performed." This assessment can be taken from existing, previously submitted and approved analyses or other alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval."

Therefore, the NRC staff's review of the contribution of external events for this application is framed by the staff's prior determination that an order of magnitude estimate for the corresponding risk contribution is sufficient. The licensee performed an analysis of the impact of external events in Section 5.4.2 of Enclosure 4 to the LAR. For the evaluation of the acceptance criteria discussed in this SE, the licensee's analysis reflected the contribution from internal fire and seismic events. For other external events, the licensee determined its contribution to be negligible for this application as further discussed below.

The licensee stated that the current CGS fire PRA (FPRA) model was created based upon analysis performed in the Individual Plant Examination of External Events (IPEEE), with periodic updates to reflect the as-built and as-operated plant. The NRC staff notes that the licensee provided additional detail on the status of the FPRA in an LAR dated November 8, 2016, requesting a one-time extension of the Residual Heat Removal Train A Completion Time (ADAMS Accession No. ML16313A573). As evaluated in the NRC's SE dated October 30, 2017, for a one-time extension of the Residual Heat Removal Train A Completion Time (ADAMS Accession No. ML17290A127), the FPRA model has undergone a number of updates since the IPEEE. In January 2004, the FPRA's technical quality and adequacy were assessed based on industry and regulatory guidance on FPRAs, and considering process guidance in NEI 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance," Revision 3 (ADAMS Accession No. ML003728023). In 2006, the FPRA was updated using the methods within the EPRI FPRA Implementation Guide and NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology," September 2005 (ADAMS Accession Number ML052580118). Additionally, the 2006 FPRA update addressed issues identified during the 2004 FPRA assessment. Further, the FPRA has been updated to include the latest revision (i.e., Version 7.2.1) of the internal events PRA model, which is used to support this application. Because the licensee's FPRA is based on the most recent internal events PRA, it was peer reviewed and the findings from the peer review were addressed, and it incorporates the methodology in NUREG/CR-6850 as described above, the NRC staff finds that the licensee's assessment of fire risk provides an order of magnitude estimate, and therefore is acceptable for the ILRT application.

To estimate the seismic risk, the licensee provided in the LAR a seismic CDF and LERF taken from the IPEEE, with periodic updates to reflect the as-built and as-operated plant. Calculations were not performed for CGS as part of Generic Issue 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants: Safety/Risk Assessment" (ADAMS Accession No. ML11356A034), and therefore, an estimate of the seismic risk is unavailable from that source. Because the IPEEE estimate provided by the licensee in the LAR did not appear to take into account the higher reevaluated seismic hazard for CGS developed in response to the Near-Term Task Force (NTTF) Recommendation 2.1, and reviewed by the NRC (ADAMS Accession No. ML16285A410), the NRC staff requested the licensee in RAI 3 to justify the seismic estimate provided in the LAR or to provide an updated seismic risk estimate.

In response to RAI 3, in the letter dated January 2, 2018, the licensee provided updated seismic CDF and LERF estimates that were obtained from the licensee's seismic PRA (SPRA) that was not finalized. The licensee also provided an overview of the status of this SPRA under development, stating that it is based on the peer-reviewed internal events PRA model, it incorporates the reevaluated seismic hazard, and it uses draft representative fragilities for structures, components and contact chatter, which were developed based on the reevaluated seismic hazard. The licensee further stated that the seismic equipment list has been developed and added to the plant response model, walkdowns have been performed, and draft seismic HRA has been included. The licensee stated that there were two tasks that have not been completed for the SPRA: (1) development of detailed fragility curves and (2) development of detailed HRA. The licensee stated that these two remaining tasks were not expected to result in an increase in seismic CDF and LERF. The licensee explained that the draft representative fragilities used in the model have a conservative bias and the future expected refinements will produce higher seismic capacities and therefore, would lower the seismic risk estimates. The licensee also explained that the seismic HRA uses conservative screening HEPs for all post-initiator human failure events modeled in the SPRA, and those HEPs are expected to decrease when the detailed HEP analyses will be performed.

The NRC staff finds that the licensee's currently available SPRA includes the major technical elements necessary in the development of an SPRA; incorporates the reevaluated seismic hazard, with corresponding fragilities; and includes the development of the seismic equipment list. Further, the licensee has adequately justified how the incomplete tasks are not expected to result in an increase in the seismic risk estimates. Therefore, the staff finds that the licensee's seismic CDF and LERF values provided in response to RAI 3 are sufficient as an order of magnitude estimate of the seismic risk, and therefore, are acceptable for the ILRT application.

The licensee evaluated other external events, including high winds, tornadoes, external floods, transportation, and nearby facility accidents, based on CGS' IPEEE analysis performed in 1994. The licensee concluded that these events are considered negligible in estimation of the external events impact on the ILRT extension risk assessment. In response to RAI 4a, in the letter dated January 2, 2018, the licensee assessed the applicability of these conclusions to the current plant and its environs. The licensee explained that it evaluated results of new research on the topics in question, data revisions, new measures for risk management, and revised procedures for addressing external hazards. The licensee concluded that the risk from these hazards remains negligible.

In RAI 4b, the NRC staff asked the licensee to justify why the risk from external flooding is negligible, or to provide an update estimate of external flooding risk. In response to RAI 4b, in the letter dated January 2, 2018, the licensee stated that based on the flooding hazard reevaluation performed in response to the NTTF Recommendation 2.1, the risk from external flooding remains negligible. The licensee explained that the maximum water surface elevations validate the flood mitigation strategy of the current licensing basis, which states that the site can be maintained in a safe condition for water levels up to 441 feet. The licensee further stated that the reevaluated Local Intense Precipitation event will result in calculated maximum water depths between 0.03 feet and 0.79 feet, however, when water surface elevations exceed the mean sea level, there is no flooding in areas containing safety-related structures, systems and components. Because the licensee reassessed the external flooding risk using current information, the staff finds the licensee's assessment of external flooding risk acceptable for the application.

In summary, the licensee has evaluated its internal events PRA against the currently endorsed ASME PRA standard (i.e., ASME/ANS RA-Sa-2009) and the currently implemented version of RG 1.200 (i.e., Revision 2), resolved or dispositioned the findings developed during the peer review of its internal events PRA for applicability to the ILRT interval extension, and included a quantitative assessment of the contribution of external events. The NRC staff reviewed the internal events peer review findings and agrees that the findings have been adequately dispositioned for this application. Furthermore, the staff concludes that the impact from external events is appropriately considered by an order of magnitude estimate. Based on the above, the NRC staff concludes that the PRA used by the licensee is of sufficient technical adequacy to support the evaluation of changes to ILRT frequency. Accordingly, the first condition is met.

3.1.2 Estimated Risk Increase

The second condition for EPRI TR-1009325, Revision 2, stipulates that the licensee submit documentation indicating that the estimated risk increase associated with permanently extending the ILRT interval to 15 years is small, and consistent with the guidance in RG 1.174 and the clarification provided in Section 3.2.4.6 of the NRC SE for NEI 94-01, Revision 2-A, dated October 2008. Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem (roentgen equivalent man) per year or 1 percent of the total population dose, whichever is less restrictive. In addition, a small increase in conditional containment failure probability (CCFP) should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage points. Additionally, for plants that rely on containment overpressure for net positive suction for ECCS injection, both CDF and LERF will be considered in the ILRT evaluation and compared with the risk acceptance guidelines in RG 1.174. As discussed further in this SE, CGS does not credit containment overpressure. Thus, for this application, the associated risk metrics include LERF, population dose, and CCFP, in accordance with the guidance in EPRI TR-1009325, Revision 2, and the clarification provided in Section 3.2.4.6 of the NRC SE for NEI 94-01, Revision 2-A, and EPRI TR-1009325, Revision 2.

The licensee reported the results of the plant-specific risk assessment in Section 3.4.3 of Enclosure 1 to the LAR, and updated those results in the response to RAI 3. The reported risk impacts are risk impact from baseline, which estimates the impact of a change in test frequency from three tests in 10 years (the test frequency under 10 CFR Part 50, Appendix J, Option A, Section III, D.1(a)) to one test in 15 years. The following conclusions can be drawn based on the licensee's analysis associated with extending the Type A ILRT frequency:

1. The increase in LERF for internal events is 3.0E-08/year. As reported in the response to RAI 3, in the letter dated January 2, 2018, the increase in LERF for combined internal and external events is 2.7E-07/year. The risk contribution from external events includes the effects of internal fires and seismic events, as discussed in this SE. This change in risk is considered to be "small" (i.e., between 1E-06/year and 1E-07/year) per the acceptance guidelines in RG 1.174. An assessment of total baseline LERF is required to show that the total LERF is less than 1E-05/year. The licensee also stated that the estimated the total LERF for internal and external events as 9.0E-06 /year. The total LERF, given the increase in ILRT interval, is below the acceptance guideline of 1E-05/year in RG 1.174 for a "small" change.

2. The increase in population dose risk from changing Type A ILRT frequency from three in 10 years to once in 15 years is reported as 2.77E-4 person-rem/year or 0.00761 percent. The reported increase in total population dose is below the values provided in EPRI TR-1009325, Revision 2-A, and defined in Section 3.2.4.6 of the NRC SE for NEI 94-01, Revision 2. Thus, this increase in the total population dose for the proposed change is considered small and supportive of the proposed change.

3. The increase in CCFP due to change in test frequency from three in 10 years to once in 15 years is 0.51 percent. This value is below the acceptance guideline of 1.5 percentage points for a small increase in CCFP in Section 3.2.4.6 of the NRC SE for NEI 94-01, Revision 2.

Based on the risk assessment results, the NRC staff concludes that for CGS, the increases in CDF and LERF are small and consistent with the acceptance guidelines of RG 1.174, as the increase in the total population dose and the magnitude of the change in the CCFP for the proposed change are small and supportive of the proposed change. The defense-in-depth philosophy is maintained because the independence of barriers will not be degraded as a result of the requested change, and the use of quantitative risk metrics collectively ensures that the balance between prevention of core damage, prevention of containment failure, and consequence mitigation is preserved. Accordingly, the second condition is met.

3.1.3 Leak Rate for the Large Pre-existing Containment Leak Rate Case

The third condition of EPRI TR-1009325, Revision 2, stipulates that in order to make the methodology in EPRI TR-1009325, Revision 2, acceptable, the average leak rate for the pre-existing containment large leak rate accident case (i.e., accident case 3b) used by the licensees shall be 100 L_a instead of 35 L_a. As noted by the licensee in Section 3.5.1 of Enclosure 1 of the LAR, the methodology in EPRI TR-1009325, Revision 2, incorporates the use of 100 L_a as the average leak rate for the pre-existing containment large leak rate accident case, and this value has been used in the CGS plant-specific risk assessments. Accordingly, the third condition is met.

3.1.4 Applicability if Containment Overpressure is Credited for ECCS Performance

The fourth condition of EPRI TR-1009325, Revision 2, stipulates that in instances where containment overpressure is relied upon for ECCS performance, an LAR is required to be submitted. In Section 3.5.3 of Enclosure 1 to the LAR, the licensee states that containment overpressure is not required in the design basis or PRA model to support ECCS performance to mitigate accidents at CGS. Therefore, the effect of the ILRT extension on containment overpressure is not addressed in the LAR. Accordingly, the fourth condition is not applicable.

3.1.5 Conclusion

Based on the above, the NRC staff concludes that the LAR for a permanent extension of the Type A containment ILRT frequency to once in 15 years for CGS is acceptable. In accordance with the revised TS 5.5.12, the containment leakage rate testing program for CGS shall be in accordance with the guidelines contained in NEI 94-01, Revision 3-A, and conditions and limitations specified in NEI 94-01, Revision 2-A.

3.2 Plant-Specific Containment Inservice Inspection Program

3.2.1 Integrated Leakage Rate Testing History

The CGS TS 5.5.12 currently requires testing in accordance with RG 1.163, which endorses the methodology for complying with Option B of 10 CFR Part 50, Appendix J. Since the adoption of Option B, the performance leakage rates are calculated in accordance with NEI 94-01, Revision 2-A, Section 9.1.1, for Type A testing. In Section 3.4.4 of Enclosure 1 to the LAR, the licensee reported in Table 3.4.4-1, a historical summary of periodic Type A ILRT results performed for CGS that demonstrate the containment has a history of leak-tightness and structural integrity with leakage well under the acceptance limits. Since 1984, five operational Type A tests have been performed with considerable margin (56 percent to 68 percent of the TS acceptance limit of L_a (0.50 percent containment air weight per day)), thereby demonstrating that CGS has a low leakage containment. The licensee also stated that the CGS CISI program and maintenance rule monitoring provide confidence in containment integrity, and there have been no Type A ILRT failures over this period.

The range of total leakage rates over this 5-year period, in percent air weight per day, measured 0.2758 percent for the first test performed on February 16, 1984, to 0.3418 percent for the last of the five tests performed on June 14, 2009. Results for all five tests were below the TS acceptance limit of 0.50 percent weight per day. Additionally, the test data results in the LAR's Table 3.6.6-1, "Columbia Type B and C LLRT Combined As-Found/As-Left Trend Summary," of Enclosure 1, provides the LLRT data trend summaries since 2007 and documents that there have been no "as-found" aggregate Type B and Type C LLRT failures that resulted in exceeding the TS limit of less than 0.6 L_a, and that the aggregate results for all the Type B and Type C tests since 2007 were well below the acceptance limit. The results demonstrate a history of adequately managing the leakage rates of the Type B and Type C containment penetrations thereby continuing to provide a high degree of assurance that containment leak-tight integrity is maintained if the Type A test is extended as requested.

3.2.2 Description of Primary Containment System

CGS is designed with a General Electric Company BWR enclosed by a Mark II type pressure suppression containment system. The reactor building consists of a steel primary containment vessel, which provides secondary containment. The primary containment vessel contains the drywell, suppression chamber, structural floor separating the drywell from the suppression chamber, sacrificial shield wall, and reactor pedestal. The reactor building secondary containment encloses the biological shield wall, spent fuel storage pool, dryer-separator pool, and the reactor well pool.

The primary containment vessel at Columbia is a free-standing carbon steel pressure vessel, which utilizes the pressure suppression technique through the Mark II over-under configuration and has an overall height of 171 ft. 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 and safe shutdown earthquake. The primary containment consists of a free standing carbon steel pressure vessel with a drywell composed of a removable ellipsoidal upper head on top of a truncated cone with a lower cylindrical section and ellipsoidal bottom head containing the suppression chamber or wetwell, which is separated from the drywell by a concrete slab structural floor supported by concrete columns and steel beams. 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. The primary containment vessel is anchored to the concrete mat foundation. 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 is a part.

The drywell floor serves as a pressure barrier between the drywell and suppression chamber. The drywell houses the reactor vessel and its associated primary system. The drywell and wetwell spaces are connected via large vertical vent pipes originating just above the drywell floor and passing through the drywell floor to their lower ends submerged in the suppression pool. The primary function of the drywell is to contain the effects of a loss-of-coolant accident (LOCA), and to direct the steam released from a primary system pipe break into the suppression chamber pool to limit the total pressure rise during a LOCA. The steel primary containment vessel is designed to act as a structural component within the reactor building. The primary containment provides the "leak tight" barrier against the potential uncontrolled release of fission products during a design-basis accident-LOCA (DBA-LOCA). CGS TS 5.5.12 identifies the primary containment L_a as 0.5 weight percent of the contained free air volume per day at the calculated maximum DBA-LOCA pressure (P_a) of 38 pounds per square inch gauge (psig). The CGS containment is discussed in Section 6.2 of the Updated Final Safety Analysis Report.

3.2.3 CISI Program (Section XI, Subsection IWE/IWL)

Section 3.6.2 of Enclosure 1 to the LAR, the licensee stated that it is implementing its CISI Program in accordance with the applicable edition/addenda of Subsection IWE of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code (ASME Code), subject to the applicable regulatory conditions as required by 10 CFR 50.55a. Subsections IWE and IWL of ASME Section XI contain inservice inspection (ISI) and repair and replacement rules for metal containment vessels IWE (Class MC) and concrete containment vessels IWL (Class CC). Subsection IWE requires general visual examination of 100 percent of accessible metallic surfaces of the containment pressure boundary 3 times over a 10-year inspection interval, pursuant to 10 CFR 50.55a(b)(2)(ix)(E), "Metal containment examinations: Fifth provision." The CGS primary containment vessel is a free-standing metal containment to which the requirements of Subsection IWE apply. The ASME Code of record for the fourth 10-year interval CISI program plan began on December 13, 2015, and ends on December 12, 2025; and is in accordance with 10 CFR 50.55a(g)(4)(ii), "Applicable ISI [inservice inspection] Code: Successive 120-month intervals," the reference code for this interval is the ASME Section XI, 2007 Edition, 2008 Addenda. The licensee presented the inspection periods and interval dates for the fourth 10-year interval in Table 3.6.2-2, "Fourth Inspection Interval Ending 12/12/2025," of Enclosure 1 to the LAR. The three inspection period intervals in that table are to be conducted during 30-40 day periods over the course of five refueling outages (RFOs), from R23 to R27. Additional programs employed at CGS, which are part of the supporting basis of the LAR, include the following license renewal Aging Management Programs: 10 CFR Part 50, Appendix J Program, IWE Inservice Inspection Program, Structures Monitoring Program, and the Service Level 1 Protective Coatings Program.

The licensee stated that during RFO R21, the drywell containment liner, piping and coated steel surfaces, and concrete floor and pedestal were examined. The inspection results identified the drywell coatings in overall good condition and are acceptable for the next operating cycle; some areas of concern were noted during the inspection and added to the Degraded Coatings Log for future repair. During RFO R22, the coating inspection focused on the primary containment shield wall and pedestal, undervessel coatings, coated concrete, and drywell and wetwell coatings. The inspection results identified that the containment coatings are in an overall good condition and are acceptable for the next operating cycle, with no new areas of concern noted.

Section 3.7.7.3 of Enclosure 1 to the LAR, discussed the results of recent containment inspections performed in accordance with ASME Code requirements. The licensee stated that no IWE examinations were conducted during RFO R21, and that all ISI period containment inspection requirements were met during RFO R22. The inspections consisted of both General Visual and VT-3 qualified examinations utilizing ISI direct and remote techniques. Some degradation noted in non-pressure boundary components was entered into the CGS Corrective Action Program (CAP). The LAR stated that a review of inspection reports from the last two RFOs (R21 and R22) showed no recordable indications reported. Maintenance Rule Structural Inspections performed during RFOs R19 and R21, concluded that the general condition of the wetwell structural steel components and drywell were in good condition.

The licensee also indicated that it evaluates potential degradation in inaccessible areas in accordance with the regulatory conditions in 10 CFR 50.55a(b)(2)(viii)(E), "Concrete containment examinations: Fifth provision," and 10 CFR 50.55a(b)(2)(ix)(A), "Metal containment examinations: First provision." As an example, NRC RIS 2016-07, "Containment Shell or Liner Moisture Barrier Inspection," dated May 9, 2016 (ADAMS Accession No. ML16068A436), stated that instances have been identified in which containment shell or liner moisture barrier materials were not properly inspected in accordance with ASME Code Section XI, Table IWE-2500-1, Item E1.30. Note 4 for Item E1.30 states, "Examination shall include moisture barrier materials intended to prevent intrusion of moisture against inaccessible areas of the pressure retaining metal containment shell or liner at concrete-to-metal interfaces and at metal-to-metal interfaces which are not seal welded." The LAR also stated that CGS is in compliance with ASME Code, and 10 CFR 50.55a(b)(2)(ix)(A)(1) since a check for moisture between the containment vessel and concrete structure is through periodic operation of the sand pocket drain valves.
Section 3.9.2 of Enclosure 1 to the LAR, discusses aging management commitments associated with the sand pocket region of the CGS containment vessel, which were made part of the license renewal application submitted to the NRC on January 19, 2010.

As discussed in the LAR, the licensee committed to perform an inspection of the containment sand pocket drain lines using a borescope to confirm the absence of clogged drain lines and existence of a flow path for the identification of any potential leakage into the sand pocket region, to ensure no water present in inaccessible areas of containment. Water in inaccessible areas is of particular concern as corrosion could go undetected for years and possibly result in loss of material (wall thinning) to less than the minimum design thickness. All actions were performed to satisfy the commitment prior to the due date of December 31, 2015.

Based on the results of recent IWE, Maintenance Rule Structural Inspections, and other inspections discussed above, the NRC staff finds that the licensee is appropriately crediting the CISI inspections to meet the 10 CFR Part 50, Appendix J visual inspection requirements of the CGS primary containment, and that concerns noted during inspections have been entered into the CAP and appropriately managed. Additionally, the staff finds that the results of containment performance (structural and leak-tightness) from the licensee's containment leakage test program will continue to be maintained if the current Type A test frequency is extended as requested. Therefore, the NRC staff finds the licensee's proposed request to allow for a permanent extension of the Type A leak rate test frequency from 10 years to 15 years is acceptable.

3.2.4 Conclusion

In summary, the results of containment performance (structural and leak-tightness) from the licensee's containment leakage test program will continue to be maintained if the current Type A test frequency is extended as requested. Therefore, the NRC staff finds the licensee's proposed request to allow for a permanent extension of the Type A leak rate test frequency from 10 years to 15 years is acceptable and, accordingly, the staff finds it acceptable to revise TS 5.5.12 as proposed by the licensee in this amendment request.

3.3 Plant-Specific Primary Containment Leak Rate Testing Program

3.3.1 Licensee's Proposed Changes

The licensee's TS change for CGS will implement NEI 94-01, Revision 3-A, and the limitations and conditions of Section 4.1 of the NEI 94-01, Revision 2-A, SE. NEI 94-01, Revision 3-A, provides that extension of the Type A test interval to 15 years be based on two consecutive successful Type A tests (performance history) and other requirements stated in Section 9.2.3 of the SE. The basis for acceptability of extending the Type A test interval also includes implementation of robust Type B and Type C testing of the penetration barriers where most containment leakage has historically been shown to occur and are expected to continue to be the pathways for a majority of potential primary containment leakage; and of a robust containment visual inspection program where deterioration of the primary containment boundary away from penetrations can be detected and remediated before any significant leakage potential were to develop.

NEI 94-01, Revision 3-A, also provides that Type C test intervals may be extended to 75 months based on two consecutive successful tests (performance history) and meeting other specified conditions.

The existing CGS TS 5.5.12 exception to NEI 94-01, Revision 0, regarding a specific date by which the then next Type A test was required is no longer needed and may be removed as that date has passed with the test having been completed. The other existing exception regarding adjustment to flow meter readings to compensate for use of instruments that did not meet the accuracy requirement of ANSI/ANS 56.8-1994 may be removed since adoption of NEI 94-01, Revision 3-A, includes incorporation of instrument requirements from ANSI/ANS-56.8-2002, and no exception to those requirements is requested.

The reference to RG 1.163 is to be replaced with a reference to NEI 94-01, Revision 3-A, and the conditions and limitations of NEI 94-01, Revision 2-A. This is allowed by the provision in

10 CFR Part 50, Appendix J, Option B, Section V.B.3, regarding the TS referencing the NRC staff approved guidance document for program implementation.

3.3.2 Historical Type A Test (ILRT) Results

In Enclosure 1 of the LAR, Table 3.4.4-1, "Columbia Type A History," the licensee presented the historical results of ILRT as summarized below.

Test Date	Leakage Rate (Primary Containment Percent Weight per Day, Performance Criterion is L_a, which is 0.500 weight percent per day)
June 1987	0.3241
June 1991	0.297
July 1994	0.302
June 2009	0.3418

The licensee noted that 1987 and 1991 ILRTs were performed when P_a was 34.7 psig and after the power uprate in 1995 P_a increased to 38 psig.

The NEI 94-01, Revision 3-A (and Revision 2-A), requirement for allowing the extended ILRT interval is that the past two consecutive tests meet the performance criterion by showing a leakage of L_a or less. The CGS TS 5.5.12 performance criterion is L_a (0.5 percent by weight of containment free air volume per day), the acceptance criterion for reactor restart is 0.75 L_a (0.375 weight percent per day), as stated in TS 5.5.12. The 1994 and 2009 ILRT results both show leakage less than L_a, and thus meet the NEI 94-01 requirement for interval extension.

3.3.3 Historical Type B and Type C Combined Test (LLRT) Results

In Enclosure 1 of the LAR, Table 3.6.6-1, "Columbia Type B and C LLRT Combined As-Found / As-Left Trend Summary," the licensee presented the historical results of the CGS Type B and C test combined as-found minimum pathway leakage totals as summarized below:

RFO and Year of Tests	As-Found Minimum Pathway Leakage Rate (standard cubic centimeters per minute (sccm))	Percent (%) of TS 5.5.12a Combined Type B and C Total Criterion (0.6 L_a, which equates to 72,922 sccm)
R18, 2007	13,191	18.09
R19, 2009	20,482	28.08
R20, 2011	9,482	13.00
R21, 2013	6,387	8.76
R22, 2015	8,932	12.25

The CGS TS 5.5.12a criterion for combined Type B and C test total is 0.6 L_a. As detailed in NEI 94-01, Revision 3-A (and Revision 2-A), this criterion is evaluated minimum pathway for as-found values and maximum pathway for as-left values. The as-found minimum pathway total provides an assessment of the leakage testing and CAPs effectiveness for ensuring penetration leakage potential is kept acceptable throughout each operating cycle such that margin to L_a is maintained to accommodate some increase in non-penetration leakage potential between ILRTs. The as-left maximum pathway total criterion is a permissive for restoring primary containment operability and ensures margin is available to accommodate increases in leakage potential between outages were leakage testing is performed.

The last five combined Type B and Type C testing totals show substantial margin to the applicable performance criterion suggesting that both the ILRT and LLRT performance criteria are unlikely to be exceeded by allowing CGS ILRT maximum interval be extended to 15 years and also allowing the Type C testing maximum interval to be extended to 75 months.

3.3.4 NRC SE Conditions of Application Relative to NEI 94-01, Revision 2-A

In Section 4.1 of the NRC staff's SE, which was incorporated in NEI 94-01, Revision 2-A, the NRC staff concluded that the guidance stated is acceptable for reference by licensees proposing to amend their TSs in regards to containment leakage rate testing, subject to six conditions. The requirements of NEI 94-01 stayed essentially the same from the original version through Revision 2 except that the regulatory positions of RG 1.163 were incorporated and the maximum ILRT interval extended to 15 years. Table 3.10.1-1, "NEI 94-01, Revision 2-A, Limitations and Conditions," in Enclosure 1 to the LAR, described the licensee response to the six conditions identified in the final SE dated June 25, 2008. The NRC staff has evaluated these responses to determine whether the licensee adequately addressed these conditions below:

1. NRC Condition 1

> For calculating the Type A leakage rate, the licensee should use the definition in the NEI TR 94-01, Revision 2, in lieu of that in ANSI/ANS-56.8-2002. (Refer to SE Section 3.1.1.1).

The licensee stated in the LAR that CGS will utilize the definition in NEI 94-01, Revision 3-A, Section 5.0. This definition has remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01 and is the one identified as acceptable. Therefore, the licensee has addressed and satisfied NRC Condition 1.

2. NRC Condition 2

> The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests. (Refer to SE Section 3.1.1.3).

The LAR provided a schedule and discussion of containment inspections in the LAR Section 3.6.2, "Containment Inservice Inspection Program," and Section 3.6.3, "Supplemental Inspection Requirements." Therefore, the licensee addressed and satisfied NRC Condition 2.

3. NRC Condition 3

> The licensee addresses the areas of the containment structure potentially subjected to degradation. (Refer to SE Section 3.1.3).

The licensee described in the LAR, Section 3.6.2, inaccessible areas potentially subjected to degradation and the methods for monitoring this potential. Therefore, the licensee addressed and satisfied NRC Condition 3.

4. NRC Condition 4

> The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable. (Refer to SE Section 3.1.4).

The LAR described past and planned containment modifications in LAR Section 3.8 "Columbia Containment Modifications." The LAR indicated that there are no major modifications planned to the containment structure. Modifications to comply with NRC severe accident capable hardened vent Order EA-13-109, issued as the result of the Fukushima Dai-ichi event are described in Section 3.8 of the LAR and involve changes to penetrations and performance of LLRT that do not present a potential need for an integrated containment test. The licensee thus addressed and satisfied NRC Condition 4.

5. NRC Condition 5

> The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provision of Section 9.1 of NEI TR 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition. (Refer to SE Section 3.1.1.2).

The licensee response in the LAR indicates acknowledgement and acceptance of this NRC staff position. Therefore, the licensee addressed and satisfied NRC Condition 5.

6. NRC Condition 6

> For plants licensed under 10 CFR 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, including the use of past ILRT data.

This condition is not applicable to CGS as it was not licensed under 10 CFR Part 52.

3.3.5 NRC Conditions In NEI 94-01, Revision 3-A

In Section 4.0 of the NRC's SE, which was incorporated in NEI 94-01, Revision 3-A, it is concluded that the guidance stated there is acceptable for reference by licensees proposing to amend their TSs in regards to containment leakage rate testing, subject to two conditions.

1. NRC Condition 1

> NEI TR 94-01, Revision 3, is requesting that the allowable extended interval for Type C LLRTs be increased to 75 months, with a permissible extension (for non-routine emergent conditions) of nine months (84 months total). The staff is allowing the extended interval for Type C LLRTs be increased to 75 months with the requirement that a licensee's

post-outage report include the margin between the Type B and Type C leakage rate summation and its regulatory limit. In addition, a corrective action plan shall be developed to restore the margin to an acceptable level. The staff is also allowing the non-routine emergent extension out to 84-months as applied to Type C valves at a site, with some exceptions that must be detailed in NEI 94-01, Revision 3. At no time shall an extension be allowed for Type C valves that are restricted categorically (e.g. BWR MSIVs [main steam isolation valves]), and those valves with a history of leakage, any valves held to either a less than maximum interval or to the base refueling cycle interval. Only non-routine emergent conditions allow an extension to 84 months.

2. NRC Condition 2, states in part:

> When routinely scheduling any LLRT valve interval beyond 60-months and up to 75-months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B and C total, and must be included in a licensee's post-outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations.

The licensee indicated in the LAR that the CGS post-outage reports will include the margin between the Type B and Type C minimum pathway leak rate summation value adjusted for understatement and the acceptance criterion. Should the Type B and C combined totals exceed the CGS Maintenance Rule leakage summation limit of 0.45 L_a but be less than the TS acceptance value (performance criterion) of 0.6 L_a, then an analysis will be performed and a corrective action plan prepared to restore and maintain the leakage summation margin to less than the administrative limit. The LAR also stated that CGS will apply the 9-month grace period only to eligible Type C tested components and only for non-routine emergent conditions. The licensee acknowledges these two conditions and the likelihood that longer test intervals would increase the understatement of actual leakage potential given the method by which the totals are calculated, and will assign additional margin for monitoring acceptability of results via administrative limits and understatement contribution adjustments. Therefore, the licensee addressed and satisfied NRC Conditions 1 and 2 of NEI 94-01 Revision 3-A.

3.3.6 Conclusion

Based on the preceding regulatory and technical evaluations, the NRC staff finds that the licensee has adequately implemented its primary containment leakage rate testing program consisting of ILRT and LLRT. The results of the recent ILRTs and of LLRT combined totals demonstrate acceptable performance and support a conclusion that the structural and leak-tight integrity of the primary containment vessel is adequately managed and will continue to be periodically monitored and managed effectively by the licensee's Primary Containment Leakage Rate Testing Program. The staff finds that the licensee has addressed the NRC conditions to demonstrate acceptability of adopting NEI 94-01, Revision 3-A, and the limitations and conditions identified in the staff SE incorporated in NEI 94-01, Revision 2-A. Therefore, the staff concludes that the proposed changes to CGS TS 5.5.12 regarding the primary containment leakage rate testing program are 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 on February 15, 2018. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding, published in the *Federal Register* on June 6, 2017 (82 FR 26131). 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) there is reasonable assurance that 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.

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Date: March 30, 2018

SUBJECT: COLUMBIA GENERATING STATION - ISSUANCE OF AMENDMENT RE: REVISION OF PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM (CAC NO. MF9469; EPID L-2017-LLA-0197) DATED MARCH 30, 2018.

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