April 12, 2005

Mr. J. V. Parrish
Chief Executive Officer
Energy Northwest
P.O. Box 968 (Mail Drop 1023)
Richland, WA 99352-0968

SUBJECT: COLUMBIA GENERATING STATION - ISSUANCE OF AMENDMENT RE: ONE-TIME EXTENSION OF APPENDIX J TYPE A INTEGRATED LEAKAGE RATE TEST INTERVAL (TAC NO. MC3942)

Dear Mr. Parrish:

The Commission has issued the enclosed Amendment No. 191 to Facility Operating License No. NPF-21 for the Columbia Generating Station. This amendment is in response to your application dated August 5, 2004, as supplemented on January 17, 2005.

This amendment revises Technical Specification (TS) Section 5.5.12, "Primary Containment Integrity," to allow a one-time extension of its Appendix J, Type A, Containment Integrated Leak Rate Test interval from the current 10-year interval to a proposed 15-year interval.

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,

/RA by J. Donohew for/

Brian Benney, Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures: 1. Amendment No. 191 to License No. NPF-21
2. Safety Evaluation

cc w/encls: See next page

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ENERGY NORTHWEST

DOCKET NO. 50-397

COLUMBIA GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 191
License No. NPF-21

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

   A. The application for amendment by Energy Northwest (licensee) dated August 5, 2004, as supplemented on January 17, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter1;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-21 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 191, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by J. Donohew for/

Robert A. Gramm, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 12, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 191

FACILITY OPERATING LICENSE NO. NPF-21

DOCKET NO. 50-397

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove	Insert
5.5-11	5.5-11
5.5-12	5.5-12

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 191 TO FACILITY OPERATING LICENSE NO. NPF-21

ENERGY NORTHWEST

COLUMBIA GENERATING STATION

DOCKET NO. 50-397

1.0 INTRODUCTION

By letter dated August 5, 2004, as supplemented on January 17, 2005, Energy Northwest (licensee) submitted a request for changes to the Columbia Generating Station (CGS) Technical Specifications (TSs). The requested changes would revise TS Section 5.5.12, "Primary Containment Integrity," to allow a one-time extension of its Appendix J, Type A, Containment Integrated Leak Rate Test (ILRT) interval from the current 10-year interval to a proposed 15-year interval. The TS revision is based on the risk-informed approach developed using Regulatory Guide (RG) 1.174. The January 17, 2005 letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, Option B, requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. The CGS TS 5.5.12, "Primary Containment Integrity," requires that leakage rate testing be performed as required by 10 CFR Part 50, Appendix J, Option B, as modified by Nuclear Regulatory Commission (NRC)-approved exemptions, and in accordance with the guidelines contained in RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, with certain exceptions specified in the TS. This RG endorses, with certain exceptions, Nuclear Energy Institute (NEI) report NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995.

A Type A test is an overall (integrated) leakage rate test of the containment structure. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months in certain circumstances. The most recent two Type A tests at the CGS have been successful, so the current interval requirement is 10 years.

The licensee is requesting a change to TS 5.5.12 which would add an exception from the guidelines of RG 1.163 and NEI 94-01, Revision 0, regarding the Type A test interval. Specifically, the exception states that the first Type A test performed after the July 20, 1994 Type A test shall be performed no later than July 20, 2009.

The local leakage rate tests (Type B and Type C tests), including their schedules, are not affected by this request.

When implementing 10 CFR Part 50, Appendix J, Option B, "Performance-Based Leakage-Test Requirements," licensees typically follow the guidelines in RG 1.163. The RG 1.163, Section C, "Regulatory Position" states, "licensees intending to comply with the Option B in the amendment to Appendix J should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01, rather than using the test intervals specified in ANSI/ANS [American National Standards Institute/American Nuclear Society]-56.8-1994." The NEI 94-01, Section 11 states that Type A testing shall be performed at a frequency of at least once per 10 years. The licensee's proposed TS change is an extension of the currently specified 10-year interval for ILRTs to a 15-year interval on a one-time basis. There are no changes to any Code or regulatory requirement or acceptance criteria.

3.0 TECHNICAL EVALUATION

3.1 Containment Integrity

The CGS utilizes a General Electric boiling water reactor (BWR)-5, Mark II primary containment consisting of a drywell, a suppression chamber, vents connecting the drywell and the suppression chamber, primary containment access penetrations, and other process piping and electrical penetrations. The leak tight integrity of the penetrations and isolation valves are verified through Type B and Type C local leak rate tests (LLRTs) as required by 10 CFR Part 50, Appendix J, and the overall leak-tight integrity of the primary containment is verified through an ILRT. These tests are performed to verify the essentially leak-tight characteristics of the containment at the design basis accident pressure. The last ILRT for Columbia's primary containment was performed in July 1994. With the extension of the ILRT time interval, the next overall verification of the containment leaktightness will be performed no later than July 2009. Because the ILRT, the LLRTs, and inservice inspection (ISI) of the containment collectively ensure the leak-tight and structural integrity of the containment, normally, the staff requests information regarding the licensee's program for containment ISI and potential areas of weaknesses in the containment that may not be apparent in the risk assessment. In Section 4.4 of Attachment 1 of the amendment request, the licensee provided a summary of the containment inservice inspection. A review of the Attachment warranted certain clarifications and additional information. The staff's request for additional information (RAI), and licensee's response are discussed below.

The licensee is using the 1992 Edition and 1992 Addenda of Subsections IWE of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), with certain approved relief from some Code requirements, for conducting the ISI of the CGS's primary containment.

In RAI 1, the staff pointed out that in Section 4.5 (Attachment 1 of Reference 1), the licensee has stated, "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." The staff requested the following information:

a. How the measurement of humidity in the sand-pocket area provides information regarding leak tightness of the bottom head?

b. Provide a brief description of the provisions made to ensure that the water ingress from the reactor well pool cavity to the space between the containment shell and the biological shield wall would not corrode the containment shell.

c. Provide a summary of the operating experience related to the bottom head shell in the sand-pocket area, and the drainage condition of the sand-pocket area.

In its January 17, 2005 letter (Reference 2), the licensee provided the following responses to the above items:

a. The sand pocket area is located outside the primary containment between the 443 and 446 foot elevations (see Columbia Final Safety Analysis Report Figure 3.8-37). The inside portion of the containment shell below elevation 446 foot (bottom head) and the outside portion below elevation 443 foot is encased in concrete. This portion of the containment metal shell is not directly or indirectly accessible for inspection.

The containment shell between elevation 443 and 446 foot is bounded by concrete on the inside and bounded by the sand pocket area on the outside. Therefore, the sand pockets will capture any potential water leakage through the annulus between the containment shell and the biological shield wall above elevation 443 foot. The outer portion of the containment shell exposed to the sand pocket area has been found to be especially susceptible to corrosion at other plants if the sand pocket area is not maintained in a relatively dry condition (reference NRC Information Notice 86-99, Degradation of Steel Containments).

The humidity in the sand pocket area is measured prior to reactor cavity flood-up and after reactor cavity drain-down. This is to ensure that water has not been introduced into the sand pocket area during refueling activities. Measurement of sand pocket area humidity provides no information regarding leak-tightness of the portion of the containment bottom head below elevation 443 foot. Measurement of sand pocket area humidity does provide assurance that water is not accumulating in the sand pocket area which could cause corrosion of the outer containment shell between the 443 and 446 foot elevations. See Columbia Final Safety Analysis Report Figures 3.8-17 and 3.8-37 for details regarding the bottom head portion of the containment structure.

b. Columbia utilizes an inner and outer drywell refueling bellows seal assembly to prevent water from flowing out of the reactor well pool cavity when the reactor well pool cavity is filled with water. The inner refueling bellows seal, which is welded to both the reactor vessel and the bulkhead plate, serves to seal the gap between the reactor vessel and the primary containment vessel. The outer refueling bellows seal is welded between the primary containment vessel and the biological shield wall and seals the space between the primary containment vessel and the biological shield wall.
The outer drywell refueling bellows seal has six 4-inch seal rupture drains and two 2-inch liner drains on the non-immersed (dry) side of the seal. These drain lines drain to a common header. The common header is opened before the reactor well pool cavity is filled with water. The cumulative drain line flow is

monitored and flow in excess of one gallon per minute through the drain lines causes an alarm in the control room. This ensures that potential leakage through the bellows seal is collected and measured prior to leaking into the space between the primary containment and the sacrificial shield wall. A review of plant operating logs for the period between 1996 and the end of 2004 found no documented cases where a valid alarm was received.

Plant Procedure Manual (PPM) 10.24.206, "Containment Annulus Sand Pocket Humidity Measurement," describes methods to be employed for determining the relative humidity of air drawn from within the containment annulus sand-pocket region. Due to the possibility of containment shell degradation due to corrosion induced by a moist environment in the sand-pocket region, Columbia Generating Station measures humidity levels in the sand-pocket area prior to and after each refueling outage. An abnormally high humidity level change in the sand-pocket area might indicate that water has drained through the annulus between the containment and biological shield wall into the sand pocket area from a leak in the outer refueling bellows seal, drain lines from the reactor well pool cavity or a through-wall leak in the containment shell below the suppression pool water level. (Note: The annulus between the primary containment vessel and the biological shield wall is constructed of compressible material. The method of construction is described in Columbia FSAR Section 3.8.2.1 (pages 3.8-8 through 3.8-10) and shown in FSAR Figure 3.8-6. This method of construction would at least inhibit, and at most prevent, the free flow of moisture to the sand pocket drains.

The sand-pocket drains are also visually checked for the presence of water. The frequency of this visual check has varied in the past from monthly to semiannually. The frequency is currently monthly.

The results of visual checks and humidity measurements in the sand pocket region combined with no known cases of leakage through the bellows seals provides a reasonable assurance that there has been no water ingress from the reactor pool cavity during past refueling outages. This provides a reasonable confidence that there has been no corrosion of the containment shell due to water ingress between the containment shell and the biological shield wall.

c. A review of the associated completed work order notes associated with the visual sand-pocket drain inspections was performed dating back to July 1995. There was no information indicating the presence of water in the sand-pocket drains.

A review of PPM 10.24.206, "Containment Annulus Sand Pocket Humidity Measurement," results for Columbia refueling outages in 1998, 1999, 2001, and 2003 was performed. The results of these measurements indicate that humidity levels in the sand-pocket drains are low enough so that water condensation will not occur on the outer containment shell in this region. From the data reviewed, the worst case (maximum) dew point temperature of the sand pocket air was about 54 degrees Fahrenheit. The temperature of the containment shell in the sand pocket area is essentially the same temperature as the suppression pool

> water. The suppression pool water temperature is normally maintained above 54 degrees Fahrenheit.

The response to item "a" indicates that the licensee is monitoring the humidity in the sand-pocket areas to ensure that the sand-pocket areas are relatively dry. In conjunction with the responses to items "b" and "c" below, the staff finds the licensees efforts to monitor the condition in the sand-pocket areas acceptable.

In response to item "b," the licensee describes the preventive measures it takes to monitor the leakage from the refueling bellow seals (found to be problematic in some Mark 1 containments). The staff finds that the licensee's actions to monitor and prevent the leakage between the biological shield wall and the containment shell acceptable, as these actions will alert the licensee of degradation potential of the primary containment.

In response to item "c," the licensee describes its operating experience related to the water leakage in the sand-pocket area, and the way it monitors the humidity in those areas. Based on the operating experience, the staff believes that the continued monitoring of various aspects of degradation of the primary containment shell will ensure leak-tightness of containment inaccessible areas during the ILRT extended interval.

In RAI 2, the staff pointed out that for the examination of seals and gaskets, and examination and testing of bolts associated with the primary containment pressure boundary (Examination Categories E-D and E-G), the licensee had requested relief from the requirements of the Code. As an alternative, the licensee had indicated that it plans to examine these items during the leak rate testing of the primary containment. With the flexibility provided in Option B of Appendix J for Type B and Type C testing (as per NEI 94-01 and RG 1.163), and the extension requested in this amendment for Type A testing, the licensee was requested to provide its schedule for examination and testing of seals, gaskets, and bolts that provide assurance regarding the integrity of the containment pressure boundary. Furthermore, the staff clarified that this request pertained to the mechanical and electrical penetrations other than the penetrations and openings which are leak rate tested during each outage (i.e., drywell head, equipment hatches, and air-locks).

In its January 17, 2005 letter, the licensee provided the following information:

> In addition to the Type B penetrations which are opened and leak rate tested each refueling outage, Columbia has 67 Type B penetrations which are tested in accordance with 10 CFR 50, Appendix J, Option B, which is the approved alternative for Category E-D and E-G examinations (seals, gaskets, and bolting). Attachment 2 of Reference 2 provides a description of the 67 Type B penetrations, and the next scheduled or required test date.
>
> All Type B penetrations on the attached list have been tested at least once since 1999 except one, which was tested in 1998.
>
> Type B penetrations on the 120-month testing interval are tested on staggered frequencies to normalize outage testing work load and to monitor generic design leakage performance. Based on this scheduling methodology, Type B penetration leakage testing is commonly performed more frequently than the required 10 year

> interval. A typical refueling outage at Columbia includes about 30% of the Type B penetrations which are on the 120 month testing interval. Currently, 22 of these 67 Type B penetrations are planned and scheduled to be performed during Columbia's R17 Refueling Outage in 2005.

The staff reviewed the tabulated information regarding the schedule of Type B testing of the mechanical and electrical penetrations, and believes that the seals, gaskets, and pressure retaining bolts associated with these penetrations are not likely to be a significant source of leakage during the ILRT extended interval period.

In RAI 3, the staff inquired about the stainless steel bellows that could be susceptible to trans-granular, stress corrosion cracking, and pointed out that the leakages through them could not be detected during Type B testing (see Information Notice 92-20). The staff also acknowledged that the plant does not have any bellows similar to those described in the information notice. However, if other types of pressure retaining bellows exist, the licensee was requested to provide information regarding inspection, testing, and operating experience of such bellows.

In its January 17, 2005 letter, the licensee provided the following information:

> The Columbia primary containment utilizes no bellows type assemblies for containment pressure retaining purposes. Columbia does utilize an inner and outer drywell refueling bellows seal assembly to prevent water from flowing out of the reactor well pool cavity when the reactor well pool cavity is filled with water. These bellows seal assemblies are not used for containment pressure retaining purposes. The inner refueling bellows seal, which is welded to both the reactor vessel and the bulkhead plate, serves to seal the gap between the reactor vessel and the primary containment vessel. The outer refueling bellows seal is welded between the primary containment vessel and the biological shield wall which seals the space between the primary containment vessel and the biological shield wall. The bellows are 0.062 inches thick and constructed of SA 240-T304 Stainless Steel. The bellows are not of two-ply construction.

The staff agrees with the licensee that the drywell refueling seal bellows assemblies are not containment pressure retaining. However, as described in response to RAI 1, the licensee is monitoring the water leakage through them to prevent potential degradation of the primary containment shell.

Recognizing the NRC staff's standard question related to the effects of degradations in uninspectable areas of the steel shell which could not be identified by visual examinations, in Section 4.5, Attachment 1, Reference 1, the licensee has provided the following information:

> 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. Therefore, a quantitative assessment using the same approach used by other plants (e.g., Calvert Cliffs) is included in Attachment 4 (Reference 1) 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 10E-7/year. Additionally, the dose rate increase is negligible compared with the total of 1.34 person-rem/year. The increase in the CCFP [conditional containment failure probability] 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.

The staff evaluation of the above analysis is included in Section 3.2 of this SE.

In Section 4.6, Attachment 1, Reference 1, the licensee has described the following process during operation of the reactor:

> During power operation the primary containment atmosphere 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 operation will be detected.

The staff believes that this continuous monitoring at a slight positive pressure will provide some indications of gross leakage, if the pressure trend in the containment is properly monitored.

Considering the precautionary measures taken to monitor and prevent any degradation in the uninspectable areas of the primary containment structure, the ability to detect significant degradation due to slightly higher pressure level in the drywell, and that the risk analysis includes a large leakage up to $35L_a$ (where L_a is the acceptable leak rate), the staff finds that the licensee has appropriately addressed the potential degradation of the uninspectable areas of the drywell.
Based on the licensee's procedures related to the potential degradation of the pressure retaining primary containment components, the staff finds that granting the requested ILRT extension will not adversely affect the leak tight integrity of the primary containment. It should be noted that Subarticle IWE-5000 of the ASME Code, Section XI requires leak rate testing following a major repair, modification, or replacement of containment components. An ILRT might be required to confirm that these activities are adequate and that further degradation does not exist in other areas of the containment. The licensee is required to report serious degradation of the containment pressure boundary pursuant to 10 CFR 50.72 or 10 CFR 50.73.

3.2 Type A Test Interval Risk Assessment

The licensee has performed a risk impact assessment of extending the Type A test interval to 15 years. The risk assessment was provided in the August 5, 2004 application for the license amendment. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute (EPRI) TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing," and RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during the development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," provided the technical basis to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement this basis, industry undertook a similar study. The results of that study are documented in EPRI Research Project Report TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The Appendix J, Option A, requirements that were in effect for the CGS early in the plant's life required a Type A test frequency of three tests in 10 years. The EPRI study estimated that relaxing the test frequency from three tests in 10 years to one test in 10 years would increase the average time that a leak, that was detectable only by a Type A test, goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of the leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a 10 percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage for the pressurized water reactor and boiling water reactor representative plants in the EPRI study confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from three tests in 10 years to one test in 20 years leads to an "imperceptible" increase in risk that is on the order of 0.2 percent and a fraction of one person-rem per year in an increased public dose.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem per year frequency. The licensee quantified the risk from sequences that have the potential to result in large releases, if a pre-existing leak were present. Since the Option B rulemaking was completed in 1995, the staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in evaluating risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 guidance to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking.

RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per year and increases in large early release frequency (LERF) less than 10^{-7} per year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee has estimated the change in LERF for the proposed change and the cumulative change from the original frequency of three tests in a 10-year interval. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. The licensee estimated the change in the conditional containment failure probability for the proposed change to demonstrate that the defense-in-depth philosophy is met.

The licensee provided its analyses, as discussed below. The following comparisons of risk from a change in test frequency from three tests in 10 years to one test in 15 years are considered to be bounding for the CGS comparative frequencies of one test in 10 years to one test in 15 years. The following conclusions can be drawn from the analysis associated with extending the Type A test frequency:

1. Given the change from a three in 10-year test frequency to a one in 15-year test frequency, the increase in the total integrated plant risk is estimated to be less than 0.01 person-rem per year. This increase is comparable to that estimated in NUREG-1493, where it was concluded that a reduction in the frequency of tests from three in 10 years to one in 20 years leads to an "imperceptible" increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. The increase in LERF resulting from a change in the Type A test frequency from the original three in 10 years to one in 15 years is estimated to be 4.7×10^{-9} per year based on the internal events PRA, and 2.6×10^{-7} per year including both internal and external events. However, there is some likelihood that the flaws in the containment estimated as part of the Class 3b frequency would be detected as part of the IWE/IWL visual examination of the containment surfaces (as identified in ASME Code, Section XI, Subsections IWE/IWL). Visual inspections are expected to be effective in detecting large flaws in the visible regions of containment, and this would reduce the impact of the extended test interval on LERF. The licensee's risk analysis considered the potential impact of age-related corrosion/degradation in inaccessible areas of the containment shell on the proposed change. The increase in LERF associated with corrosion events is estimated to be less than 1×10^{-8} per year. The NRC staff concludes that increasing the Type A interval to 15 years results in only a small change in LERF and is consistent with the acceptance guidelines of RG 1.174.

3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The licensee estimates the change in the conditional containment failure probability to be an increase of less than one percentage point for the cumulative change of going from a test frequency of three in 10 years to one in 15 years. The NRC staff finds that the defense-in-depth philosophy is maintained based on the small magnitude of the change in the conditional containment failure probability for the proposed amendment.

Based on these conclusions, the NRC staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines of RG 1.174, while maintaining the defense-in-depth philosophy, and, therefore, is acceptable.

3.3 NRC Staff’s Conclusion

Based on its review, the NRC staff finds that the interval between Type A containment ILRT tests at the CGS may be extended to 15 years, and that the proposed changes to TS 5.5.12 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. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes the surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration (69 FR 53102, published on August 31, 2004), and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.229(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter from RL Webring (Energy Northwest) to NRC, "Request for Amendment to Technical Specification for One-time Extension of Containment Leak Rate Test Interval," August 5, 2004.

2. Letter from WS Oxenford (Energy Northwest) to NRC, "Response to Request for Additional Information Regarding One-time Extension of Containment Leak Rate Test Interval," January 17, 2004.

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Date: April 12, 2005