Dominion Energy Kewaunee, Inc.
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Dominion

September 11, 2008

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

Serial No. 08-0475
KPS/LIC/JF: R3:
Docket No. 50-305
License No. DPR-43

DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
LICENSE AMENDMENT REQUEST 242:
EXTENSION OF THE ONE-TIME FIFTEEN YEAR CONTAINMENT INTEGRATED LEAK RATE TEST INTERVAL

Pursuant to 10 CFR 50.90, Dominion Energy Kewaunee, Inc. (DEK) requests an amendment to Facility Operating License (OL) Number DPR-43 for Kewaunee Power Station (KPS). The proposed amendment would change KPS Technical Specification (TS) 4.4.a, *"Integrated Leak Rate Tests (Type A),"* to permit a one-time, six-month extension, to the currently approved 15-year interval between Type A containment integrated leak rate tests (ILRT). TS 4.4.a currently requires that the next Type A test be performed within 15 years of the previous test. The last Type A test performed at KPS was completed in April 1994. Therefore, the next test is currently required to be performed no later than the end of April 2009. However, since the next KPS refueling outage is scheduled for September/October 2009, DEK requests a one-time extension to the currently approved 15-year interval to avoid a mid-cycle shutdown solely to perform the test.

Kewaunee license amendment request (LAR) 198 (reference 1) requested a one-time extension of the containment Type A test interval from 10 to 15 years. LAR 198 was supplemented with a response to an NRC request for information (reference 2), which included a revised risk impact assessment for the requested amendment. On April 6, 2004, the NRC issued amendment 173 (reference 3) to the KPS operating license. Amendment 173 incorporated a one-time extension of the Type A test interval from 10 to 15 years. The safety evaluation (SE) for amendment 173 states, *"With the extension of the ILRT time interval, the next overall verification will be performed no later than April 2009."* At the time, a refueling outage was scheduled to occur at KPS in April 2009. However, due to subsequent unforeseen circumstances, the KPS refueling outage schedule was moved beyond April 2009 and the schedule for performing the next Type A test was likewise, although inappropriately, moved along with the refueling outage.

The proposed amendment would extend the currently approved 15-year interval between Type A tests by 6 months. Per the SE for KPS amendment 173, the current Type A test interval expires at the end of April 2009. The proposed amendment would extend the current Type A test interval, on a one-time basis, to October 2009 to coincide with completion of the next scheduled refueling outage (KR 30).

After discovery that the next Type A test was required to be performed prior to the start of the next scheduled refueling outage, DEK began to plan for a mid-cycle shutdown to perform the required Type A test in parallel with development of this LAR. Since vendor support is needed to perform the required Type A test, DEK has identified an available vendor to support testing in late-March 2009 if the schedule extension is not approved.

Attachment 1 provides a description, a safety evaluation, a significant hazards determination, and environmental considerations for the proposed amendment. Attachment 2 provides the marked-up KPS TS page.

Attachment 3 provides a risk assessment that supports the proposed amendment. This risk assessment is based on a Type A test interval of 15 years - 9 months and conservatively bounds the risk impact of the requested test interval of 15 years - 6 months.

The KPS Facility Safety Review Committee has approved the proposed amendment and a copy of this submittal has been provided to the State of Wisconsin in accordance with 10 CFR 50.91(b). DEK requests approval of the proposed amendment by April 30, 2009. Once approved, the amendment will be implemented within 60 days.

If you have any questions or require additional information, please contact Mr. Craig Sly at (804) 273-2784.

Sincerely,

Leslie N. Hartz
Vice President – Nuclear Support Services

VICKI L. HULL
Notary Public
Commonwealth of Virginia
140842
My Commission Expires May 31, 2010

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz, who is Vice President – Nuclear Support Services of Dominion Energy Kewaunee, Inc. She has affirmed before me that she is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of her knowledge and belief.

Acknowledged before me this 11TH day of September, 2008.

My Commission Expires: May 31, 2010.

Vicki L. Hull
Notary Public

Attachments:

1. Discussion of Change, Safety Evaluation, Significant Hazards Determination and Environmental Considerations
2. Marked-up Technical Specification Page
3. Evaluation of Increased Risk due to One-Time 9-Month Extension of Integrated Leak Rate Testing Interval

References:

1. Letter from T. Coutu (NMC) to Document Control Desk (NRC), "License Amendment Request 198 to the Kewaunee Nuclear Power Plant Technical Specifications for One-time Extension of Containment Integrated Leak Rate Test Interval," dated June 20, 2003. [ADAMS Accession No. ML031820613]
2. Letter from T. Coutu (NMC) to Document Control Desk (NRC), "License Amendment Request 198, 'ILRT 5-Year Extension,' NMC Response to NRC Request for Additional Information," dated December 12, 2003. [ADAMS Accession No. ML033570469]
3. Letter from J. Lamb (NRC) to T. Coutu (NMC), "Kewaunee Nuclear Power Plant – Issuance of Amendment (TAC No. MB9907)," dated April 6, 2004. [ADAMS Accession No. ML040340168]

Commitments made in this letter: None

cc:

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ATTACHMENT 1

LICENSE AMENDMENT REQUEST 242

<u>EXTENSION OF THE ONE-TIME FIFTEEN YEAR CONTAINMENT INTEGRATED LEAK RATE TEST INTERVAL</u>

DISCUSSION OF CHANGE, SAFETY EVALUATION, SIGNIFICANT HAZARDS DETERMINATION AND ENVIRONMENTAL CONSIDERATIONS

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

EXTENSION OF THE ONE-TIME FIFTEEN YEAR CONTAINMENT INTEGRATED LEAK RATE TEST INTERVAL

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EXTENSION OF THE ONE-TIME FIFTEEN YEAR CONTAINMENT INTEGRATED LEAK RATE TEST INTERVAL

1.0 DESCRIPTION

Pursuant to 10 CFR 50.90, Dominion Energy Kewaunee, Inc. (DEK) requests an amendment to Facility Operating License (OL) Number DPR-43 for Kewaunee Power Station (KPS). The proposed amendment would change KPS Technical Specification (TS) 4.4.a, *"Integrated Leak Rate Tests (Type A),"* to permit a one-time, six-month extension to the currently approved 15-year interval between Type A containment integrated leak rate tests (ILRT). TS 4.4.a currently requires that the next Type A test be performed within 15 years of the previous test. The last Type A test performed at KPS was completed in April 1994. Therefore, the next test is currently required to be performed no later than the end of April 2009. However, since the next KPS refueling outage is scheduled for September/October 2009, DEK requests a one-time extension to the currently approved 15-year interval to avoid a mid-cycle shutdown solely to perform the test.

Kewaunee license amendment request (LAR) 198 (reference 1), requested a one-time extension of the containment Type A test interval from 10 to 15 years. This LAR was supplemented with a response to an NRC request for information (reference 2), which included a revised risk impact assessment for the requested amendment. On April 6, 2004, the NRC issued amendment 173 (reference 3) to the KPS operating license. Amendment 173 incorporated a one-time extension of the Type A test interval from 10 to 15 years. The safety evaluation (SE) for amendment 173 states, *"With the extension of the ILRT time interval, the next overall verification will be performed no later than April 2009."* At that time, a refueling outage was scheduled for April 2009.

Subsequently, in 2005/2006 KPS experienced four forced shutdowns. The forced shutdowns resulted in the plant being shutdown for over 164 days. Due to these forced shutdowns, the start of the next refueling outage was rescheduled from April 2006 to September 2006. Rescheduling the 2006 refueling outage (KR 28) also resulted in modification of subsequent refueling outage schedules. An evaluation was performed to determine the effects of rescheduling KR 28 on required surveillance intervals. This evaluation identified several surveillance intervals requiring extension. An LAR extending these surveillances was submitted to and approved by the NRC (reference 11).

However, while evaluating the effects of rescheduling future refueling outages an error was made in determining the maximum allowable interval between Type A tests. TS 4.4.a discusses a Type A test frequency change from *"at least once per 10 years"* to *"at least once per 15 years"* and references NEI 94-01, revision 0 (reference 6). Based on the verbiage of TS 4.4.a, the plant staff incorrectly concluded that the other provisions of NEI 94-01, revision 0, also applied, including the 15-month extension (grace period) for Type A tests discussed in sections 9.1 and 11.3.

DEK identified this error in interpretation on June 20, 2008 after completion of the most recent refueling outage (KR 29) in May 2008. While reviewing the scope of the next refueling outage (KR 30), it was determined that previous conclusions regarding application of a grace period were wrong. Although the language of TS 4.4.a is not specific, the NRC SE that accompanied amendment 173 specifically states that the next Type A test will be performed no later than April 2009.

Therefore, a revision to the KPS TS is requested to increase the currently approved 15-year interval between Type A tests by 6 months. The current Type A test interval expires at the end of April 2009. The proposed amendment would extend the current Type A test interval, on a one-time basis, to October 2009 to coincide with completion of KR 30. This amendment is requested to avoid the hardship and risks associated with a forced mid-cycle shutdown solely to conduct a Type A test. The duration of a mid-cycle shutdown to perform a Type A test is estimated to be a minimum of ten days.

The proposed TS amendment qualifies for a no significant hazards consideration under the standards set forth in 10 CFR 50.92(c). The proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

2.0 PROPOSED CHANGE

The proposed amendment would modify KPS TS 4.4.a, "Integrated Leak Rate Test (Type A)."

The current KPS TS 4.4.a reads as follows:

a. *Integrated Leak Rate Tests (Type A)*

Perform required visual examinations and leakage rate testing in accordance with the Containment Leakage Rate Testing Program.

As a one-time change, the Type A test frequency specified in NEI 94-01, Revision 0, Paragraph 9.2.3, as "...at least once per 10 years based on acceptable performance history" is changed to "...at least once per 15 years based on acceptable performance history." This change applies only to the interval following the Type A test performed in April 1994.

Based on the discussion and justification provided in this license amendment request, DEK is proposing to add the following statement: "*The first ILRT performed after April 1994 shall be performed no later than October 2009.*"

When completed the modified TS 4.4.a would read as follows:

a. *Integrated Leak Rate Tests (Type A)*

Perform required visual examinations and leakage rate testing in accordance with the Containment Leakage Rate Testing Program.

As a one-time change, the Type A test frequency specified in NEI 94-01, Revision 0, Paragraph 9.2.3, as "...at least once per 10 years based on acceptable performance history" is changed to "...at least once per 15 years based on acceptable performance history." This change applies only to the interval following the Type A test performed in April 1994. <u>The first ILRT performed after April 1994 shall be performed no later than October 2009.</u>

3.0 BACKGROUND

The testing requirements of 10 CFR Part 50, Appendix J (reference 5) provide assurance that the primary containment, including those systems and components that penetrate the primary containment, do not exceed the allowable leakage rate assumed in the plant safety analyses. The main purpose of the reactor containment system is to mitigate the consequences of potential accidents by minimizing the release of radionuclides to the environment to assure the health and safety of the public. Appendix J requires performance of periodic ILRTs, (also known as a Type A tests) to measure the overall leakage rate of the primary containment.

10 CFR 50, Appendix J was revised in 1995 to allow use of Option B, "Performance Based Requirements." Regulatory Guide (RG) 1.163 (reference 4) concludes that NEI 94-01, revision 0 (reference 6), provides methods acceptable to the NRC staff for complying with the provisions of Option B. NEI 94-01, revision 0 permits an extended Type A test interval of up to 10 years based on two consecutive successful tests. In 1996, KPS implemented performance based containment leakage rate testing in accordance with Option B and RG 1.163 and established a ten-year frequency for Type A tests. Type A tests were completed in February 1977, June 1980, April 1984, March 1987, April 1990 and April 1994. In addition, a pre-operational Type A test was completed in June 1973. KPS met the established leakage rate acceptance criterion during all of the Type A tests performed (see Table 1).

In April 2004, the NRC approved Kewaunee Amendment 173 (reference 3) incorporating a one-time, five-year extension to the 10-year Type A test interval. The associated NRC safety evaluation (SE) concluded that the risk associated with increasing the interval between Type A tests was small and that the ability of the inservice inspection (ISI) program to detect containment degradation was unlikely to be affected. The SE also stated, *"With the extension of the ILRT time interval, the next overall verification will be performed no later than April 2009."*

This LAR would change KPS TS 4.4.a, *"Integrated Leak Rate Tests (Type A),"* to permit a one-time, six-month extension to the currently approved 15-year interval between Type A tests. The last Type A test at KPS was completed in April 1994. Therefore, the next test is currently required to be performed no later than the end of April 2009. However, since the next refueling outage (KR 30) is scheduled for September/October 2009[#], DEK requests a one-time extension to the currently approved 15-year interval to avoid a mid-cycle shutdown solely to perform the next test. If the requested extension

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[#] Shortly after shutdown in September 2009 for KR 30, containment system integrity will not be required per TS 1.0.g and TS 3.6.a (i.e. the plant will be in the Cold Shutdown with the reactor vessel head installed, or Refueling Mode). If the outage schedule is delayed or unforeseen events occur during the outage, completion of the Type A test could occur later than October 2009. However, containment system integrity is not required in the Cold Shutdown or Refueling Modes, and completion of the Type A test is not required until the plant reaches a mode where containment system integrity is required. The Type A test must be complete prior to declaring the containment operable and reaching greater than Cold Shutdown conditions coming out of KR 30. This meets the guidance provided in Section 9.2.2 of NEI 94-01, Revision 2.

is not obtained, DEK would be required to shutdown KPS for a minimum of 10 days to perform the next Type A test.

3.1 System Function and Design Requirements

3.1.1 Primary Containment

The KPS Primary Containment system consists of a steel structure and its associated-engineered safety features systems. The Primary Containment system is a freestanding carbon steel cylindrical pressure vessel with hemispherical dome and ellipsoidal bottom (the Reactor Containment Vessel), with an internal net free volume of 1,320,000 cubic feet, and its associated engineered safety features systems.

The Reactor Containment Vessel is capable of withstanding a design internal pressure of 46 psig and a temperature of 268°F. The Reactor Containment Vessel is supported on a grout base that was placed after vessel construction was completed and tested. The Reactor Containment Vessel is a low-leakage steel shell, including all its penetrations, designed to confine the radioactive materials that could be released by accidental loss of integrity of the Reactor Coolant system pressure boundary.

The Primary Containment associated engineered safety features systems include fan coil units and internal containment sprays capable of rapidly absorbing the energy released by a loss of coolant accident. The containment systems are described in detail in Chapter 5 of the KPS Updated Safety Analysis Report (USAR) (reference 7).

Primary Containment integrity is assumed in the following KPS design basis accidents (DBAs):

- The steam line break containment integrity analysis demonstrates that the energy release into containment does not cause the failure of the containment structure following a steam line break event. All cases analyzed result in a maximum containment pressure that is less than 46 psig and a containment vessel shell temperature that is less than 268 °F.
- The rupture of a Control Rod Drive Mechanism Housing (also called Rod Cluster Control Assembly (RCCA) Ejection accident) is the result of an extremely unlikely mechanical failure of a control rod drive mechanism pressure housing such that Reactor Coolant System (RCS) pressure would eject the RCCA and drive shaft. The consequences of this mechanical failure, in addition to being a minor Loss-of-Coolant Accident (LOCA), may also result in a rapid reactivity insertion and an adverse core power distribution, possibly leading to localized fuel rod damage. From a containment response standpoint this accident mimics the LOCA accident.
- By definition, LOCAs (includes both large-break (LB) and small-break (SB) LOCAs) are events that may happen very infrequently, but may occur during the life of the plant. A SBLOCA event will not, by itself, result in a consequential loss of function of the RCS or of containment barriers. Should a major break (LBLOCA) occur rapid depressurization of the RCS to a pressure nearly equal to the containment

pressure occurs with a nearly complete loss of RCS inventory. Containment pressure is evaluated in each LOCA event to ensure the containment safeguards systems are capable of limiting the peak containment pressure to less than 46 psig.

The KPS primary containment system is designed so that for all LOCA break sizes, up to and including the double-ended severance of a reactor coolant pipe, the containment peak pressure remains below its design pressure. The containment response analysis demonstrates the acceptability of the containment safeguards system to mitigate the consequences of a LOCA inside containment. The impact of LOCA mass and energy releases is evaluated to ensure that containment pressure remains below 46 psig at the licensed core power conditions. In support of equipment design and licensing criteria (for example, qualified operating life), with respect to post-accident environmental conditions, long-term containment pressure and temperature transients are generated to conservatively bound the potential post-LOCA containment conditions.

3.1.2 Shield Building

The Reactor Containment Vessel is surrounded by a cylindrical Shield Building constructed of reinforced concrete, which serves as radiation shielding for normal operation and in the event of a LOCA. The Shield Building is designed to protect the entire primary containment vessel from direct exposure to the outside atmosphere. The Shield Building has the shape of a right circular cylinder with a shallow dome roof. A five-foot annular space is provided between the Reactor Containment Vessel and the Shield Building. Clearance at the roof of the Shield Building is seven feet. The Reactor Containment Vessel is supported on a grout base that was placed after the Reactor Containment Vessel was constructed and tested. Both the Reactor Containment Vessel and the Shield Building are supported on a common foundation slab. A Shield Building Ventilation System serves the annulus between these structures, filtering any leakage from the inner Reactor Containment Vessel before it is released to the atmosphere.

The Shield Building concrete wall is 2'-6" thick and the dome is 2'-0" thick for biological shielding requirements. Because the Shield Building wall thickness was designed to meet radiation shielding requirements, the thickness is generally in excess of that necessary for structural requirements. The design basis of the Shield Building is to completely enclose the Reactor Containment Vessel, the access openings, the equipment door, and that portion of all penetrations that are associated with Primary Containment. The design of the Shield Building provides for:

- Biological shielding,
- Controlled releases of the annulus atmosphere under accident conditions,
- Protection of the Reactor Containment Vessel from the environment.

With this design, the KPS Reactor Containment Vessel is completely accessible except for those portions that interface with the foundation slab. Unlike most other plant containment structures, in which the outside of the containment vessel is located directly against a concrete barrier, both sides of the KPS Reactor Containment Vessel shell are entirely accessible, with the exception of the base. This affords greater access for leak testing and inspection of the KPS Reactor Containment Vessel.

3.2 Regulatory Requirements

10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," (reference 5) requires verification of the integrity of primary containment penetrations and isolation valves through Type B and Type C local leak rate tests (LLRTs) and the verification of overall primary containment leak integrity through Type A tests (Integrated Leak Rate Tests). These tests are periodically performed at KPS to verify the essentially leak-tight characteristics of the primary containment at the design basis accident pressure. The Type A test measures the overall leakage rate of the primary containment.

The testing requirements of 10 CFR Part 50, Appendix J provide assurance that the primary containment, including those systems and components that penetrate the primary containment, do not exceed the leakage rate assumed in the plant safety analyses. The main purpose of the reactor containment system is to mitigate the consequences of potential accidents by minimizing the release of radionuclides to the environment to assure the health and safety of the public.

On October 26, 1995, 10 CFR 50, Appendix J, Option B, "Performance-Based Requirements," became effective allowing licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance. Incorporation of 10 CFR 50, Appendix J, Option B into the KPS TS was approved by the NRC on May 28, 1998, under License Amendment No. 136 (reference 29) as corrected by NRC letter dated June 24, 1998 (reference 30).

Regulatory Guide 1.163 (reference 4) endorses, with certain exceptions, Nuclear Energy Institute (NEI) 94-01, revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 26, 1995 (reference 6), for complying with the provisions of 10 CFR 50, Appendix J, Option B. Regulatory Guide 1.163 specifies that licensees intending to comply with Option B should establish test intervals based upon the criteria of NEI 94-01, rather than using the test intervals specified in ANSI/ANS-56.8-1994 (reference 27).

The required frequency for Type A testing in NEI 94-01, revision 0 is at least once-per-10 years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart or refueling cycles where the calculated performance leakage rate was less than 1.0 L_a) and consideration of the performance factors in NEI 94-01. NEI 94-01, revision 0 specifies an initial interval of 48 months for Type A tests and allows an extension of the test interval to at least once per 10 years based on two consecutive successful tests.

Recently the NRC has reviewed and approved an update to NEI 94-01 by endorsing revision 2 (reference 9) as described in section 3.3. NEI 94-01, revision 2 describes an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J; including provisions for extending Type A ILRT intervals to up to 15 years. NEI 94-01, revision 2 also incorporates the regulatory positions provided in Regulatory Guide 1.163. NEI 94-01, revision 2 permits permanent 15-year intervals between Type A tests provided certain performance-based measures are met. This was a change to NEI 94-01, revision 0 which permitted only a one-time 15-year interval between Type A tests provided certain performance-based measures were met.

3.3 NRC Safety Evaluation on NEI 94-01 Extension Bases

The NRC Safety Evaluation (reference 8) that endorses use of NEI 94-01, revision 2 (reference 9) and EPRI Report No. 1009325, revision 2 (reference 23) states that:

> *"... extensions of the performance-based Type A test interval beyond the required 15 years should be infrequent and used only for compelling reasons. ...the licensee will have to demonstrate to the NRC staff that an unforeseen emergent condition exists."*

Based on these statements the following sections address the compelling reasons and unforeseen emergent conditions that necessitate the requested extension.

3.3.1 Compelling Reasons

Unnecessary Shutdown of Kewaunee Power Station

As previously discussed, the next Type A test is currently required to be performed no later than April 2009. However, the scheduled start of the next KPS refueling outage has been moved from April 2009 to the September/October 2009 due to the effects of a series of forced shutdowns in 2005/2006. Currently, there are no other issues that would require a plant shutdown before the end of the current refueling cycle. Therefore, if the required interval between Type A tests is not increased then a plant shutdown will be required in April 2009 solely to perform a Type A test.

A mid-cycle shutdown in April 2009 to perform a Type A test would be a costly evolution without commensurate safety benefit. Performance of the Type A test in April rather than conducting the Type A test in October 2009 during a scheduled refueling outage, would require an additional plant shutdown of approximately 10 days with the consequence of lost generation. An additional shutdown would also put the plant through an additional thermal cycle. In the long term, additional thermal cycles lead to a reduction in component reliability and are a recognized stressor on component aging.

There is also an inherent risk associated with placing the plant in a shutdown condition and returning it to full power operations. Evaluating the risk of performing plant shutdown and startup activities (called transition risk) involves assessing the impact of human error and equipment failure during this transition period. Transition risk estimates the risk from changing reactor power from 100% to 0% and returning back to 100% power. The risk is calculated in terms of core damage probability (CDP) and large early release probability (LERP) due to a transient event occurring during power reduction and power ascension operations. The CDP calculated for KPS in the transition period is 1.27E-6 and the LERP calculated for KPS is 6.35E-8. In addition, the probability of having a reactor trip during the transition period is 2 – 3 times higher (depending on the length of time in the transition period) than if the plant stays at full power conditions.

There are costs and risks associated with shutting down KPS. The costs include financial and component aging costs. The risks are associated with transitioning the plant from full power operations to shutdown operations and return to full power operations. Considering the insignificant risk benefit of Type A testing six months earlier, DEK has concluded that shutting down KPS for the sole purpose of performing a Type A test is not commensurate with the associated lost generation and transition risks to shutdown.

Historical Performance of Type A Tests

All previous Type A tests have passed the associated KPS leakage rate acceptance criterion. See Table 1 below for the KPS performance record.

TABLE 1

Summary of Kewaunee Type A Test Results

Date	As-Left Leak Rate (Wt. %/Day)	Acceptance Criteria (Wt. %/Day)
June 1973*	0.0484	0.375
February 1977	0.09998	0.375
June 1980	0.037	0.375
April 1984	0.0162	0.375
March 1987^	0.1634	0.375
April 1990^	0.0926	0.375
April 1994^	0.0610	0.375

* Pre-operational test

^ Test results obtained using BN-TOP-1. Includes LLRT penalties for valves/penetrations not in their normal post accident condition.

The leakage results of all Type A tests performed at KPS have been less than the associated acceptance criterion. Note that later results (1987, 1990 and 1994) reflect the addition of calculation conservatisms from use of the BN-TOP-1 methodology and addition of leak-rate penalties based on local leak rate test (LLRT) results for valves or penetrations that were not in their normal post-accident position during the Type A Test.

These results demonstrate the KPS containment vessel has a history of both leak tightness and structural integrity.

NEI 94-01, Revision 0 and Regulatory Guide 1.163 Recommendations

NEI 94-01, revision 0, section 11.3 (reference 6) states the following: *"An extension of up to 25 percent of the test interval (not to exceed 15 months) may be allowed on a limited basis for scheduling purposes only."*

The NRC later endorsed this statement in RG 1.163, section C (reference 4), which states: *"NEI 94-01, Revision 0, ...provides methods acceptable to the NRC staff for complying with the provisions of Option B in Appendix J to 10 CFR Part 50, subject to the following:*

1. *NEI 94-01 references ANSI/ANS-56.8-1994, "Containment System Leakage Testing Requirements," for detailed descriptions of the technical methods and techniques for performing Types A, B, and C tests under the amendment of Appendix J to 10 CFR Part 50. However, as stated in NEI 94-01, the test intervals in ANSI/ANS 56.8-1994 are not performance-based. Therefore, licensees intending to comply with 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-56.8-1994. ..."*

Therefore, RG 1.163 allows up to a 15-month extension of Type A tests for scheduling purposes* consistent with NEI 94-01, revision 0, section 11.0. The extension requested by this LAR is six months. This is less than the maximum extension of 15 months allowed by NEI 94-01, revision 0 and RG 1.163.

* Note that when RG 1.163 references section 11.0 of NEI 94-01 it means the entire section 11 of the document. NEI 94-01 section 11.0 is only a title to the section, therefore, it is concluded that RG 1.163 is endorsing all of section 11 of NEI 94-01, Revision 0.

NEI 94-01, Revision 2 Recommendation

NEI 94-01, revision 2, section 9.1 (reference 9), states the following with respect to extending Type A test intervals:

> *"Required surveillance intervals for recommended Type A testing given in this section may be extended by up to 9 months to accommodate unforeseen emergent conditions but should not be used for routine scheduling and planning purposes."*

DEK is requesting a one-time extension to the currently approved 15-year test interval based on the unforeseen emergent conditions described in section 3.3.2 below. The one-time extension requested by this LAR is six months, which is less than the maximum extension of nine months allowed by NEI 94-01, revision 2.

Minimal Change in Risk

In NUREG-1493 (reference 10) the NRC concluded that reducing the frequency of Type A tests from three-in-10 years to one-in-20 years leads to an imperceptible increase in risk. For example, in NUREG-1493 the incremental risk contribution of leakage associated with the difference between performing Type A tests three-times-in-10 years versus 1-time-in-20 years is computed to be a 0.08% risk increase for the baseline plant, demonstrating that the incremental risk is imperceptible. The estimated increase in risk is concluded to be very small because Type A tests identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.

The evaluations performed to support a one-time extension of the interval between Type A tests from 10 years to 15 years (references 1 and 2) for KPS indicated good correlation with the NRC conclusions above. The evaluations indicated acceptable (minimal) risk associated with extending the Type A test interval from 10 to 15 years on a one-time basis.

The previous evaluations performed for KPS also indicated good correlation with more recent industry results provided in EPRI Report 1009325 (reference 23). For example, the KPS change in conditional containment failure probability (CCFP) value of 0.0021, change in large early release frequency (LERF) value of 3.3E-07, and the change in person-rem/year value of 0.0113 provided in reference 2 are within the range of results provided in Table G-1 of EPRI Report 1009325. Table G-1 provides a summary of the one-time Type A test extension submittals from the industry to the NRC.

Attachment 3 provides a KPS site-specific risk evaluation for extending the Type A test interval to once-in-15 years - 9 months. The evaluation was performed for an additional 9 months to bound the 6-month extension requested in this LAR. This evaluation also indicates an imperceptible change in risk associated with the requested extension period. See section 4.3 for further details.

Type A Test Failures are Rare

In NUREG-1493, section 4.1 (reference 10) the NRC states the following:

> *"In the approximately 180 ILRT reports considered in this study, covering 110 individual reactors and approximately 770 years of operating history, only 5 ILRT failures were found which local leak rate testing could not and did not detect. These results indicate that Type A testing detected failures to meet current leak-tightness requirements in approximately 3 percent of all tests.... The percentage of containment leakage that can be detected only by integrated containment leakage testing is very small. Of note in the ILRT failures observed that were not detected by Type B and C testing, the actual leakage rates were very small, only marginally in excess of the current leak-tightness requirements."*

EPRI Report No. 1009325 (reference 23) verifies that the conclusions from NUREG-1493 are correct and conservative. EPRI found that recent data from 2001 to 2007 indicated that 35 more Type A tests had been performed at nuclear power plants and no failures were identified.

Based on the above, Type A tests provide limited benefit regarding detection of failed containment conditions independent of Type B and Type C test performance. Therefore, KPS considers that a one-time extension of six months of the current Type A test interval will not result in a significant increase in plant risk.

TS Surveillance Testing Approach

Neither the KPS TS nor Standard TS (NUREG-1431) permit extensions to the Type A test interval. Nevertheless, Standard TS contain a principal for TS surveillance performance that is applicable to Type A testing. NUREG-1431 (reference 22), states the following in the bases section for SR 3.0.3 regarding missed surveillances:

> *"SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2...*
>
> *The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, <u>the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.</u>"* [emphasis added]

This statement indicates the safety significance of the delay in performing a missed surveillance test is typically small. This statement is consistent with the conclusions in NUREG 1493 and EPRI Report No. 1009325 regarding the low frequency and significance of Type A test failures. Therefore, it is concluded there is little safety significance (in terms of risk significance) in delaying the performance of a Type A test.

In addition, the TS do not assume that equipment is inoperable if its specified surveillance frequency has expired. Rather, the standard TS recognize that the most probable result of performing a surveillance test is that the equipment will pass the surveillance test. In other words, it is not assumed that a component is inoperable or degraded at the point that its required surveillance test frequency is exceeded. This conclusion is well supported by the conclusions in NUREG 1493 and EPRI Report No. 1009325.

The interval between Type A tests is longer than the interval between typical surveillance test intervals. However, Type A tests are augmented by Type B and Type C leak rate testing which are required on a much more frequent bases.

No Performance-Based or Risk-Based Issues with Extending Intervals

Previously under NEI 94-01, Revision 0, Type A tests had an associated grace period provided with a specified frequency. For example, for licensees who were permitted up to 10 years between Type A tests, a grace period of up to 15 months was allowed for scheduling purposes without prior NRC approval. However, when the NRC approved a method that would allow licensees to justify permanent 15-year intervals between Type A tests (under NEI 94-01, revision 2), no grace period was allowed for scheduling purposes without prior approval. The basis for not allowing a grace period for permanent 15-year intervals between Type A tests appears to be that, "*NEI 94-01, Revision 2, allows this interval to be as much as 15 years which is a significant period of time between tests.*"

One goal of the "performance based" and "risk based" approach to regulation is to reduce the resources dedicated to activities, including surveillance tests, having limited or no risk benefit. It has been recognized within the industry that Type A test failures that are not detectable by Type B or Type C testing are rare (less than 3%) and that the increase in risk of performing a Type A test once every 20 years is "imperceptible" (NRC NUREG 1493). Based on the low frequency of Type A test failures within the industry and the low risk associated with increasing the period between Type A tests, it is reasonable that the NRC should entertain this one-time extension to the currently approved one-time 15-year test interval for KPS. Licensees have demonstrated and the NRC agreed that there are no performance-based or risk-based issues with extending Type A test intervals beyond 15 years. See section 3.7 for a discussion on past precedence.

3.3.2 Unforeseen Emergent Conditions

Unforeseen Shutdown Period

KPS LAR 198 (reference 1), dated June 20, 2003, requested a one-time extension of the Type A test interval from 10 to 15 years. This LAR was later supplemented with a response to an NRC request for information (reference 2), which included a revised risk impact assessment for the requested change. On April 6, 2004, the NRC issued amendment 173 (reference 3) to the KPS operating license incorporating the one-time extension of the Type A test interval from 10 to 15 years. The safety evaluation that accompanied the amendment specifically states, *"With the extension of the ILRT test interval, the next overall verification will be performed no later than April 2009."* At that time, a refueling outage (KR 30) was scheduled to begin in April 2009.

Subsequently, KPS experienced four forced shutdowns in 2005/2006. This resulted in the plant being shutdown for a total of over 164 days. Due to these shutdowns, the refueling outage schedule was modified. The new schedule moved refueling outage (KR 28) from April 2006 to September 2006. This in turn modified subsequent refueling outage schedules, including refueling outage KR 30, which was rescheduled from April 2009 to September/October 2009.

The forced shutdown period in 2005/2006 was unforeseen in 2003 when the LAR to modify the Type A test interval was submitted and in 2004 when NRC approval was obtained to increase the interval. The start date for KR 30 was moved beyond April 2009 due to the unforeseen extended forced shutdown period in 2005/2006.

Unforeseen Emergent Error Identification

When evaluating the impact of rescheduling the 2006 refueling outage (KR 28), an error was made in determining the next required performance of a Type A test. In August 2005, the plant staff performed a detailed review of TS testing requirements, preventive maintenance requirements, commitments and other outage related activities. This review was performed before changing the outage schedule. Based on this review, many TS surveillance tests were required to be extended. An LAR requesting extension of these surveillance tests was submitted to the NRC and subsequently approved (see license amendment 187 (reference 11) and KPS license condition 2.C.(9)). However, when the analysis for extending the Type A test interval was performed, KPS staff believed from review of RG 1.163 (reference 4) and NEI 94-01, revision 0 (reference 6) that the 15-month extension available for Type A tests, was applicable and prior NRC approval for extending the surveillance interval was not required. Based on this evaluation and since the outage schedule changed by only six months, the next Type A test continued to be planned for KR 30 and no NRC approval for the extension was sought.

DEK did not determine that this conclusion was in error until after completion of the most recent refueling outage (KR 29), which was completed in May 2008. The recent determination that an error was made during the 2005 review constitutes an unforeseen emergent condition.

3.4 Kewaunee Specific Considerations

Consistent with the defense-in-depth philosophy described in RG 1.174 (reference 12), DEK has assessed other considerations specific to KPS that are relevant to the proposed amendment. These considerations are discussed below.

3.4.1 Local Leakage Rate Testing (LLRT)

DEK has a comprehensive LLRT (Type B and C testing) program in place to meet the requirements of 10 CFR 50, Appendix J, Option B. The program tests 68 total containment penetrations. The Type B program tests 38 penetrations including: electrical penetrations, the fuel transfer tube (bellows and flange), equipment hatch (o-ring), personnel air lock, emergency airlock, mechanical spare penetrations with flanged closures (o-rings), pressure transmitters, expansion bellows, and ventilation penetrations with flanged enclosures. The Type C program tests 43 penetrations. There are 13 penetrations that receive both a Type B and a Type C test.

The historical LLRT test data provided below indicates that LLRT test results have been consistently below the maximum allowable combined leakage rate of 0.60 L_a (129,120 SCCM) for all penetrations and valves subject to Type B and C tests. In addition, DEK conservatively maintains an administrative leakage limit for each tested containment penetration subjected to Type B and C testing based on penetration size. Leakage to the special ventilation zone (Zone SV) is limited to 0.1 L_a (21,520 SCCM) and a limit of 0.01 L_a (2,152 SCCM) is applied to leakage that bypasses the special ventilation zone system. Table 2 provides the combined total as-found leakage of the LLRTs performed since the five-year extension request was made in 2003. The maximum and minimum pathway leakage calculation methods are defined in Section 2 of ANSI/ANS-56.8-1994 (reference 27).

TABLE 2

Year	Maximum Pathway Leakage All Penetrations	Minimum Pathway Leakage All Penetrations	Maximum Pathway Leakage To Zone SV	Minimum Pathway Leakage To Zone SV	Maximum Pathway Leakage Bypass Zone SV	Minimum Pathway Leakage Bypass Zone SV
2003	20,078	17,043	2,968	958	1,701	1,289
2004	9,135	8,290	1,059	638	605	1,273
2006	14,592	6,589	3,128	1,486	480	466
2008	27,788	13,582	16,612	3,267	701	656

Note: All values are in SCCM. L_a was 0.5 weight percent prior to 2006. L_a was changed in 2006 to 0.2 weight percent by License Amendment 190 (reference 28).

3.4.2 Containment ISI Program

DEK has established a containment ISI program that implements the examination and testing requirements of American Society of Mechanical Engineer's (ASME) Section XI and 10 CFR 50.55a Class MC components. The containment ISI program was developed pursuant to the requirements of ASME Subsection IWE◆, of Section XI, of the Boiler and Pressure Vessel Code (Code). DEK maintains the KPS containment ISI program as required by 10 CFR 50.55a(g)(6)(ii)(B).

Previously, 10 CFR 50.55a(g)(6)(ii)(B) required an expedited examination of the accessible portions of primary containment liner, penetrations, selected pressure retaining bolted connections, the moisture barrier at the liner-to-containment floor junction, and outer concrete surfaces. The required examinations were conducted during refueling outages in 1998, 2000, 2001, 2003, 2004 and 2006 as part of the first ten-year interval examinations, and met the applicable requirements of the ASME code.

The first ten-year interval of the containment ISI program was developed and implemented in accordance with the requirements of the 1992 Edition with the 1992 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Subsection IWE, as modified by NRC final rulemaking of 10 CFR 50.55a, published in the Federal Register on August 8, 1996. The first ten-year inspection interval was established from September 9, 1996, to September 9, 2006.

DEK is currently implementing the second ten-year interval of the containment ISI program at KPS. The second ten-year interval was established from September 9, 2006, to September 9, 2016. This interval was developed in accordance with the requirements of the 2001 Edition with the 2003 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Subsection IWE.

The scope of the KPS containment ISI program includes all the containment surfaces, pressure retaining welds, containment surfaces requiring augmented examination, seals, gaskets, moisture barriers, pressure retaining dissimilar metal welds, pressure retaining bolting and pressure retaining components that are required to be examined. The containment ISI program is unaffected by the proposed amendment, and will continue to provide a high degree of assurance that any containment degradation will be detected and corrected before it can result in a leakage path.

RG 1.163 (reference 4), position C.3, specifies performing visual examinations of the accessible interior and exterior surfaces of the containment system for structural problems prior to initiating a Type A test and during two other refueling outages prior to the next Type A test, if the interval for the Type A test has been extended to ten years. The purpose of this requirement is to allow for early discovery of evidence of structural deterioration.

◆ Subsection IWL is not applicable to KPS since KPS has a freestanding metal containment structure as described in section 3.1 of this attachment.

Results of KPS containment examinations are provided in section 3.4.4 below.

In addition to the examinations discussed above, periodic functional ISI examinations are performed on ASME piping to ensure weld structural integrity. These examinations comply with the requirements of the 1998 Edition 2000 Addenda of the ASME Section XI, Article IWA-5000, System Pressure Tests. The inspection of ASME piping and welds is performed through containment penetrations.

3.4.3 Approved Alternatives to Subsection IWE Requirements

No alternatives (approved relief requests) to ASME Section XI, Subsection IWE requirements are approved that credit the performance of integrated (Type A) or local (Type B and C) leak rate testing for the second ten-year interval of the KPS containment ISI program.

3.4.4 Containment Examination History

Containment ISI program examinations performed since the most recent Type A test (April 1994) include visual examinations of the containment vessel pursuant to ASME Section XI, Subsection IWE. These examinations were conducted during refueling outages that occurred from 1998 through 2008. Examination results of the containment inspection history from 1998 through and including 2001 were provided to the NRC in a letter dated June 20, 2003 (reference 1). All recordable indications were considered minor conditions and corrected or accepted.

A description of the recordable indications identified since 2001 is provided below. With the exception of the minor conditions described below, all containment ISI program examination results were within the established acceptance criteria.

The 2003 containment examinations identified the following:

- No recordable indications were noted.

The 2004 containment examinations identified the following:

- A General Visual and VT-3 indication was recorded consisting of a slight inward bulge on Plate 74 and Plate 75. This slight inward bulge was first recorded in the fall of 2001. The slight inward bulge was evaluated by engineering and accepted under KPS Specification TS-1052, Addenda No.4, Item No.19, Section 10.3 - Shell Tolerance.
- General Visual and VT-3 indications were recorded consisting of previously identified surface rust on Plate 64, Plate 65, Plate 66, Plate 67, and on the plates for Penetration No. 37EN, Penetration No. 38EN, and Penetration No. 38ES. The surface rust was removed, and the affected areas of the Reactor Building Containment Vessel were ultrasonically examined to verify wall thickness and

repainted. A General Visual and VT-3 re-examination was performed with satisfactory results.

- A General Visual and VT-3 indication was recorded consisting of a 3" x 4" scab-type indication located on Plate 158. This scab-type indication was accepted by engineering as a preservice surface imperfection.
- General Visual and VT-3 indications were recorded consisting of caulking between the concrete floor and the Personnel Airlock pulled away and/or cracked. The caulking was repaired/replaced. The repaired/replaced caulking was accepted by engineering for continued service.
- A VT-3 indication was recorded consisting of a gasket on the Personnel Airlock starting to peel away. The gasket was replaced.
- A VT-1 indication was recorded consisting of a missing cotter pin on the swing bolt of the Equipment Door. The missing cotter pin was replaced.

The 2006 containment examinations identified the following:

- A General Visual and VT-3 indication was recorded consisting of a 3" x 4" scab-type indication located on Plate 158. This scab-type indication was accepted by engineering as a preservice surface imperfection during the 2004 refueling outage. Reexamination of this indication during the 2006 refueling outage revealed no noticeable change and the indication was accepted by engineering.

The 2008 containment examinations identified the following:

- General Visual indications were recorded consisting of separation between the moisture barriers of two Plates (62 and 67). The moisture barriers were repaired, reexamined and accepted.
- VT-3 indications were recorded consisting of damaged threads on six bolts for Penetration No. 18, Refueling Transfer Canal Blind Flange. The six bolts were repaired, reexamined by VT-1, as required by 10 CFR 50.55a(b)(2)(ix)(G) and (H), and accepted. The repaired bolts were evaluated and accepted by engineering.

As described above, each of the minor indications were repaired/replaced or were accepted. In some cases, re-examination of previous flaws have been performed and it was determined that no further degradation has occurred.

3.4.5 Areas Requiring Augmented Examinations Per IWE-1240

The first ten-year interval of the ASME Section XI, Subsection IWE inspection plan began at KPS on September 9, 1996. All examinations have been completed for the first ten-year interval. KPS began the second ten-year interval on September 9, 2006. To date, the first refueling outage of the first period of the second ten-year surveillance interval has been completed. Currently no area of the KPS containment structure requires augmented inspection in accordance with IWE-1240.

3.4.6 Containment Penetration Bellows

The NRC has identified that containment penetration bellows have been susceptible to improper testing practices. As documented in NRC Information Notice 92-20 (reference 13), leakage through some bellows may not be readily detectable by LLRTs due to their design. KPS has nine penetration assemblies that incorporate two-ply mechanical bellows and one penetration assembly that incorporates a three-ply mechanical bellows. The two-ply assemblies are the two main feedwater, two main steam, two steam generator blowdown, two residual heat removal, and one letdown penetrations. The three-ply assembly is the fuel transfer tube penetration.

A review of plant drawings indicates that wire mesh is installed between the two-plies of each two-ply bellows assembly (and between each set of plies in the three-ply bellows), ensuring that an adequate gap exists to measure leakage when performing required Type B tests. The LLRT administrative acceptance criterion for measured leakage through these penetrations is very low at 100 standard cubic centimeters per minute (SCCM). Some of these penetrations are tested during each refueling outage per the KPS Containment Leak Rate Testing Program with satisfactory results. The LLRT test results for each bellows assembly (for tests conducted through 2003) were previously provided to the NRC in a letter dated December 12, 2003 (reference 2).

The LLRT test results for each bellows assembly tested since 2003 are provided below in Table 3:

TABLE 3

PENETRATION BELLOWS LLRT RESULTS

YEAR TESTED	PENETRATION NUMBER									
	6W		6E		7W		7E		8S	
	B1	B2	B1	B2	B1	B2	B1	B2	B1	B2
2004							<20	20.4		
2006			<20	<20						
2008	<20	<20	<20	<20	<20	<20			<20	<20

YEAR TESTED	PENETRATION NUMBER							
	8N		9		10		11	
	B1	B2	B1	B2	B1	B2	B1	B2
2004			<20	<20	<20	<20		
2006	<20	<20						
2008	<20	<20					<20	<20

YEAR TESTED	PENETRATION NUMBER		
	18		
	B1	B2	B3
2004	<20	<20	<20
2006			
2008			<20

Note: Leakage values are provided in SCCM. Values are reported as <20 SCCM if the measured value was smaller since 20 SCCM is the minimum value for instrument calibration.

Based on the bellows leakage information provided above, and those previously provided, leakage values measured since 2003 have not indicated degradation of the bellows assemblies.

3.4.7 Maintenance Rule – Containment Inspection

As required by 10 CFR 50.65 *"Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,"* DEK has established a program for monitoring the condition of structures. The primary containment building is within the scope of the program. The containment isolation function of limiting the release of radioactive fission products following an accident has been classified as "high risk significant." Therefore, the condition of the containment is monitored pursuant to 10 CFR 50.65 and in accordance with the KPS Maintenance Rule program.

Baseline structural inspections of the KPS containment were conducted in 1998. Additional inspections were performed in 2003 and 2008 (KR 29). By procedure, the frequency of these inspections does not exceed five years and will not be revised by the one-time extension of the Type A test interval. The concrete inspection program for the containment includes examination for the following defects:

- Surface Pitting and Scaling
- Leaching
- Delaminations
- Dissolution
- Water In-Leakage
- Cracking
- Peeling/Chipped Paint
- Spalling and Popout
- Honeycombs
- Discoloration
- Other defects

The results of the KPS containment structural inspections are as follows:

- 1998 - Found only cosmetic defects in coatings inside containment.
- 2003 - Found minimal rust accumulation from the containment vessel on the concrete foundation and on bolts and unistrut in floors. Found minor hairline cracks due to shrinkage in concrete. Discovered cracking of east and south wall faces of reactor refueling pool. Loose paint/coating on east and south wall removed. No sign of displacement was observed.
- 2008 - The base of the containment structure appeared in good condition. Inspected the intersection between the pressure vessel steel and concrete (this area was identified as degraded in the 2003 inspection). No corrosion was noted. Hairline cracks in base noted during the 2003 inspection had not degraded. No further loose paint/coating on the east and south wall faces of reactor refueling pool were observed. Various minor conditions were dispositioned or resolved.

Operability of the containment isolation equipment is controlled under KPS TS sections 1.0.g, 3.6, 3.8, 4.4, 5.2 and 6.20. The proposed amendment affects only the Type A test requirements found in TS sections 4.4 and 6.20 and has no impact on the KPS Maintenance Rule program.

3.4.8 Maintenance Rule - Shield Building Inspection

The secondary containment structure (Shield Building) shields the primary containment structure from the effects of weather, thereby providing additional assurance of its integrity. DEK conducts a periodic visual inspection of the KPS Shield Building every five years to ensure its integrity as required by the Maintenance Rule program.

Baseline structural inspections of the KPS Shield Building were conducted in 1998. Additional inspections were performed in 2003 and 2008 (KR 29). By procedure, the frequency of these inspections shall not exceed five years and will not be revised by the one-time extension of the Type A test interval. The concrete inspection program for the shield building includes surveillance elements for the containment discussed in section 3.4.7 above.

The results of the KPS Shield Building inspections are as follows:

- 1998 - Found minor concrete spalling in localized areas of roof trough.
- 2003 - Found minor concrete leaching, discoloration cracking around the shield building dome and shell interface. Shell interior and exterior appeared in good condition with no structural cracks, indications or flaws visible.

- 2008 - The joint between the roof and the upper wall of the shield building has numerous areas of minor staining and leaching due to water (rain and snow melt) intrusion. Overall, the shield building was determined to be in good condition.

3.5 Risk Impact of Extended Type A Testing Intervals

Attachment 3 contains a risk evaluation performed in accordance with the guidance provided in NEI 94-01, revision 2 (reference 9) and EPRI report 1009325, revision 2 (reference 23). The results of this analysis are discussed more fully in section 4.3 of this LAR.

Section 3.3.1 of this LAR discusses the NRC conclusions in NUREG-1493 (reference 10) regarding the risk impact of extending Type A test intervals. In NUREG-1493, the NRC concluded that reducing the frequency of Type A tests from three-in-10 years to one-in-20 years leads to an imperceptible increase in risk. The estimated increase in risk is very small because Type A tests identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.

EPRI report 1009325, revision 2 (reference 23) updated the work performed in NUREG-1493 (published in 1995) with the results of more recently performed Type A tests. EPRI report 1009325 (published in 2007) collected data from Type A tests performed across the industry. In summary, two NEI utility surveys and an examination of recent Type A test results provided data from 217 Type A tests that have been performed in the nuclear industry. Based on this data, the number of containment leakage events found during the performance of these tests is very small. In fact, no Type A test failures that would have resulted in a large early release have been found.

EPRI also reviewed industry data relating to the risks associated with extended Type A test intervals and determined the following:

- While no acceptance guidelines for these additional figures of merit are published, examinations of NUREG-1493 and Safety Evaluation Reports for one-time interval extensions indicate a range of incremental increases in population dose that have been accepted by the NRC. The range of incremental population dose increases is from < 0.01 to 0.2 person-rem/yr and/or 0.002 to 0.46% of the total accident dose. The total doses for the spectrum of all accidents result in health effects that are at least two orders of magnitude less than the NRC Safety Goal Risk. Given these perspectives, a very small population dose is defined as < 0.75 person-rem per year for the risk impact assessment of the extended Type A test intervals.
- Regarding CCFP, changes of up to 1.1% have been accepted by the NRC for the one-time requests for extension of Type A test intervals. In context, it is noted that a CCFP of 1/10 (10%) has been approved for application to evolutionary light water designs. Given these perspectives, a change in the CCFP of up to 10% is assumed to be small. The degree of small is dependent on the magnitude of the

initial CCFP since small initial CCFPs will result in larger acceptable percentage changes.

- Despite the very conservative assumptions, the submittals to date have been able to demonstrate that the revised Type A testing interval has little impact on risk. That is, the risk or the change in population dose and change in LERF are small.

The conclusions of EPRI report 1009325, revision 2 substantiate that the original bases for extended Type A test intervals provided in NUREG-1493, and the bases used for the original extension of the KPS Type A test interval have not changed.

3.6 Technical Adequacy of KPS Probabilistic Risk Assessment (PRA)

3.6.1 Scope of the PRA Model

The KPS PRA model used in this evaluation is a Level 1 and Level 2 PRA model that addresses internal, seismic, and fire events at full power. The LERF figure of merit is calculated using the full Level 2 PRA model.

The KPS PRA model was developed to support the Individual Plant Examination in 1992 and the Individual Plant Examination for External Events (IPEEE) in 1994.

3.6.2 Technical Adequacy

RG 1.177 (reference 31) provides a framework for the risk evaluation of proposed changes to surveillance intervals. RG 1.177 requires identification of the risk contribution from the impacted surveillances, determination of the risk impact due to the proposed change to the surveillance interval, and performance of sensitivity and uncertainty evaluations. Regulatory Position 2.3.1 of RG 1.177 states that PRA quality must be compatible with the safety implications of the TS change being requested. In the Final Safety Evaluation for NEI 94-01, revision 2 (reference 8), the NRC staff states that, *"Capability I of ASME RA-Sa-2003 shall be applied as the standard, since approximate values of CDF* [Core Damage Frequency] *and LERF and their contribution among release categories are sufficient for use in the EPRI methodology."*

Additionally, as noted by the Staff in the Final Safety Evaluation of NEI 94-01, revision 2 (reference 8, Page 13), the emphasis of the quantitative evaluation provided in NEI 94-01 is on the risk impact from the internal events. The PRA analysis performed in support of this one-time Type A test extension request is more comprehensive and includes a quantitative assessment of the contribution from external events (fire and seismic). However, the risk matrices for the seismic hazard and internal/fire hazards are reported separately on the basis that:

- Simple addition of contribution from these hazards is judged not to be appropriate.
- An alternative method for aggregating the risk matrices is not yet available.

The concern about simple addition of risk matrices has been recognized and documented in EPRI 1010068 (reference 34). This aggregation issue is one of several issues that are being addressed as part of an addendum to the Memorandum of Understanding between the NRC and EPRI as described in SECY-07-0176 (reference 35, Enclosure 1, page 8).

3.6.3 Internal Events

Since 1994, updates have been made to incorporate plant and procedure changes, update plant-specific reliability and unavailability data, improve the fidelity of the model, incorporate Westinghouse Owners Group (WOG) Peer Review comments, and support other applications, such as On-line Maintenance, Risk-Informed In-Service Inspection (RI-ISI), Maintenance Rule Risk Significance, and Mitigating System Performance Index (MSPI).

The enhancements to the KPS PRA model include a major internal flooding update and a number of updates to the Level 2 PRA model to allow a more realistic assessment of the LERF figure of merit.

Peer Review

Peer Review (Certification) of the KPS PRA model, using the WOG Peer Review Certification Guidelines, was performed in June 2002 (reference 32). The Peer Review found 54 significant issues (A and B Facts and Observations (F&Os)). On the basis of its evaluation, the Certification Team determined that, with A and B F&Os addressed, the technical adequacy of all elements of the PRA would be sufficient to support risk significant evaluations with defense-in-depth input. All of the F&Os except two B level F&Os have been addressed. The following is a summary of the open F&Os and their potential impact on the risk insights provided in support of this application.

- IE-1 - Loss of ventilation initiating event is not well discussed. This F&O is a documentation issue, does not impact the model logic, and does not have an impact on the proposed amendment.
- TH-4 - HVAC calculations need to be revised. The peer review identified that the HVAC calculation for AFW room was overly conservative in nature. This conservatism may result in an early failure assumption of components and result in a higher CDF. The overall effect is added conservatism to the calculations that support the proposed amendment.

Self Assessments

In addition to the Peer Review, KPS performed an assessment of PRA quality in support of the MSPI implementation. The following is a summary of section 3.2, "Technical Adequacy of the KPS PRA," from the MSPI basis document (Reference 33).

- The Kewaunee PRA results for the Emergency Diesel Generator, Auxiliary Feedwater, High Pressure Safety Injection, and Service Water systems were identified as having appropriate component importance with respect to similar plants, as assessed by Westinghouse in the MSPI Cross Comparison in WCAP-16464-NP.
- The Low Pressure Safety Injection (Residual Heat Removal) MSPI system was identified as an outlier in the MSPI cross-comparison because the Birnbaum importance is lower than that of similar plants. The reason for the lower Birnbaum importance is because initiation of containment sump recirculation requires manual alignment of valves by operators. Human error probabilities dominate over equipment failures. Additionally, containment sump strainer plugging, although a low probability, also has an impact on CDF. Thus, the probability of operators failing to align for containment sump recirculation or containment sump strainers plugging is higher than the probability of the monitored components failing. Since the human error and containment sump strainers plugging probabilities dominate, the monitored component Birnbaum values are appropriate for KPS.

Additionally, in January 2008, an independent assessment of the KPS PRA model against RG 1.200, revision 1 (reference 14) was performed. The assessment was conducted by a team of experts with experience in performing NEI PRA Certifications and ASME PRA Standard Reviews. The assessment included a review of the Dominion PRA procedures, current documentation notebooks, and other associated documentation.

The scope of this assessment was to compare the current PRA against the ASME standard (ASME RA-Sa-2003) to determine if each of the requirements of Capability Category II had been met and sufficiently documented. The assessment identified a number of Supporting Requirements (SRs) that did not meet Capability Category II requirements. The "NOT MET" characterization was conservatively assigned to an SR if one or more apparent documentation or modeling issue(s) could not be readily disposed of, even if the overall analysis had been found to be appropriate. Due to the scope of (i.e., focus on Capability Category II) and conservative nature of the initial assessment, the "NOT MET" SRs were reviewed to:

1. Determine if "NOT-MET" Category II SRs would meet Capability Category I requirements.
2. Identify those "NOT-MET" SRs that do not have an impact on the risk insights provided in support of this proposed amendment (e.g., documentation only issues).
3. Identify potential sensitivity studies that can be performed to ensure that the risk insights are not significantly affected by the "NOT MET' findings.

Based on this review, it was determined that:

1. A number of SRs that did not meet Capability Category II requirements may not meet Capability Category I requirements, since the applicable requirement applied to both capability categories.

2. Of those SRs that meet #1 above, none were determined to have an impact on the risk insights provided to support this proposed amendment based on one or more of the following:

   a. The potential issue was documentation only.

   b. The potential issue would result in an over-estimation of ILRT related change in CDF or change in LERF (i.e., conservative with respect to the application). For example, SR IE-IC2 was characterized as "NOT-MET" because, based on the documentation review, it was concluded that all potential inter-system (IS) LOCA pathways may not have been identified. Since the ISLOCA contribution to CDF or LERF is not affected by the proposed amendment, the potential underestimation of ISLOCA frequency would have a conservative impact on the change in LERF calculations.

   c. The potential issue was assessed to have no impact on the CDF/LERF estimate. For example, AS-A6 SR is characterized as "NOT-MET" because reviewers found that, although the sequence of top events shown on the KPS event trees appears to follow the expected accident sequence, the High Pressure Injection (HPI) node in the Station Blackout (SBO) event trees is placed immediately after the initiating event (prior to the secondary decay heat removal node).

      This issue was assessed to have a minimal impact on the CDF/LERF results on the basis that the ordering of the top events; 1) was determined by the original reviewers to be adequate in almost all cases, and; 2) in one instance the reviewers indicated that the order may not be justified. However, based on discussion with the plant PRA Engineer and a sensitivity run, the order does not change the CDF/LERF results.

   d. The potential issue impacts very low frequency events, such as Anticipated Transient Without Scram (ATWS)-LOCA sequences. As a result, the impact on the overall CDF/LERF estimate is assessed to be negligible.

   e. The potential issue could be eliminated based on formal inquiries to the ASME requirements. For example, AS-A6 SR is characterized as "NOT-MET" partially because reviewers found that the initiating event screening discounts initiators based on whether a plant trip occurs. The reviewers believe that the Failure Modes and Effects Analysis (FMEA) should instead use the screening criteria stated by AS-A6 SR, and a system or train loss that requires shutdown within a relatively short time-frame due to TS requirements (e.g., within six and potentially up to 24 hours), should be modeled as an IE. However, based on inquiry Record Number 07-213, the risk from such an event needs to be captured in a transition risk or low power risk.

It should be noted that the current plan is to disposition all the SRs that did not meet Capability Category II requirements by the end of 2009.

3.6.4 External Events

As stated above, the KPS PRA model used in this evaluation is a full scope Level 1 and Level 2 PRA model that addresses internal, seismic, and fire events at full power.

Seismic PRA

The main elements of a seismic PRA are the seismic hazard evaluation, structure and component fragility analysis, plant logic analysis, and event tree quantification. A summary of each of these risk assessment elements is provided below.

1. The seismic hazard evaluation provides KPS-specific seismic hazard levels and the probable frequency of occurrence.
2. The structure and component fragility analysis provides unique fragility curves for the components and structures assessed in the PRA model.
3. The seismic plant logic analysis determines the consequence of various structural and component failures. This logic is added to the general transient event trees developed for the internal events PRA, as used in the Individual Plant Examination (IPE) report.

Note that, compared with the internal events analysis, the seismic PRA is very conservative on the basis that:

1. Almost all non-safety related components and systems (e.g., Main Feedwater System) were assumed to fail during a seismic event with probability of 1.0.
2. Almost all elements of the internal events analysis have been updated several times whereas only some elements of the seismic PRA have been updated (e.g., fragility or hazard curves have not been updated). More recent hazard curves indicate significantly lower frequencies.
3. Recovery of damaged components is not considered in the KPS seismic PRA.
4. The correlation of damage between systems is not evaluated. The current model conservatively assumes multiple concurrent component failures given a specific seismic event.

Therefore, the estimate of the seismic hazard to the CDF and LERF figure of merit is considered conservative. However, the CDF and LERF estimates are also judged to be acceptable for the proposed amendment on the basis that a conservative estimate of CDF and LERF would result in a decrease in the allowable Type A test interval extension.

Fire PRA Model

The fire PRA model was developed to support the IPEEE study. The fire analysis was based on a combination of a PRA and an EPRI fire-induced vulnerability evaluation (FIVE) approach. A screening study, based on plant walkdowns and the FIVE approach was used, and for the areas that passed screening a full PRA was used. Based on a review of the Kewaunee IPEEE submittal, the NRC staff concluded that Kewaunee's IPEEE process was capable of identifying the most likely severe accidents and severe accident vulnerabilities, and that the IPEEE study met the intent of Supplement 4 to GL 88-20.

Currently, the KPS fire PRA model is scheduled to be upgraded based on the current state-of-the-art approaches and guidance to support transitioning the fire protection program to the NFPA-805 standard (reference 36).

3.6.5 PRA Maintenance and Update

The KPS PRA configuration management process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated plant. This process is defined in the configuration control program, which consists of a governing procedure and subordinate implementation procedures. The governing procedure delineates the responsibilities and guidelines for updating the PRA models. The overall objective is to define a process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, and industry operating experience), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plant, the following activities are routinely performed:

1. Design changes and procedure changes are reviewed for their impact on the PRA model.
2. New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
3. Maintenance unavailability is captured, and its impact on CDF is trended.
4. Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated on a regular basis.

In addition to these activities, KPS PRA configuration management procedures provide the guidance for particular risk management and PRA quality and maintenance activities. This guidance includes:

1. Documentation of the PRA model, PRA products, and bases documents.
2. The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.

3.6.6 Conclusions on PRA Technical Adequacy and Scope

As a result of the conservative IPE and IPEEE PRA models, as well as the considerable effort to incorporate the latest industry insights into the PRA, self-assessments, and Peer Reviews, DEK concludes that the current KPS PRA model meets the expectations for PRA technical adequacy set forth in the NRC SE (reference 8).

3.7 Past Precedence

The proposed license amendment would permit a one-time, six-month extension to the currently approved 15-year interval between performances of Type A tests at KPS. DEK has identified other similar requests which have been approved by the NRC as follows:

1. River Bend Station requested and received approval for an extension of the Type A test interval from once in 15 years to once in 15 years - 4 months (references 15, 16 and 17). River Bend Station then requested and received approval for a further extension of the Type A test interval from once in 15 years - 4 months to once in 15 years - 8 months (references 18 and 19).
2. Seabrook Station requested and received approval for an extension of the Type A Test interval from once in 15 years to once in 15 years - 6 months (references 20 and 21).
3. St. Lucie Unit 2 requested and received approval for an extension of the Type A Test interval from once in 15 years to once in 15 years - 6 months[#] (references 25 and 26).

Each of the license amendments above is similar to the proposed license amendment request for KPS. The broader similarities are as follows:

1. Each licensee had previously applied for and received permission from the NRC for a one-time extension of the Type A surveillance test interval to 15 years.
2. Each licensee was seeking to extend the one-time performance of the Type A surveillance test interval beyond 15 years.
3. Each licensee was seeking to extend the one-time performance of the Type A surveillance test interval beyond 15 years to prevent a premature shutdown of the reactor (extend the test interval until a scheduled refueling outage), thereby reducing costs and risks associated with a forced outage.
4. Each licensee generally performs Type A testing in accordance with 10 CFR 50, Appendix J, Option B; RG 1.163; NEI 94-01, revision 0; and ANSI/ANS 56.8-1994, with minor exceptions.

[#] This request was to change the TS to indicate that the next performance of the ILRT would occur prior to the end of an outage. The estimated duration of the extension in the referenced submittal was six months.

Regarding performance of Type A testing, there is little difference between Boiling Water Reactors (River Bend) and Pressurized Water Reactors (KPS, Seabrook and St. Lucie Unit 2). The differences between this amendment request and the precedence amendments cited above are as follows:

1. Methodology used to perform risk evaluation:
   a. River Bend performs Type A testing in conjunction with Drywell Bypass testing. Therefore, River Bend used a custom risk evaluation methodology (called a slightly modified version of the Grand Gulf Nuclear Station Methodology) to determine the risks associated with deferral of the testing (reference 15, sections 4.0 and 4.3, reference 18, sections 3.0 and 4.0). This methodology had been previously reviewed and approved by the NRC.
   b. Seabrook used the NUREG-1493/EPRI-104285 (references 10 and 24) methodology (reference 20, attachment 3), which is also reviewed and approved by the NRC.
   c. St. Lucie Unit 2 used the methodology in Combustion Engineering Owners Group Joint Applications Report, WCAP-15691, *"Joint Applications Report for Containment Integrated Leak Rate Test Interval Extension,"* revision 2, June 2002 for computation of risk. The NRC did not approve WCAP-15691, revision 2.
   d. DEK is using the NEI 94-01, revision 2/EPRI-1009325, revision 2 (references 9 and 23) methodology for evaluation of risk, since this is the latest approved guidance. NEI 94-01, revision 2 and EPRI-1009325, revision 2 were approved after River Bend, Seabrook and St. Lucie Unit 2 received NRC approval to extend their Type A test interval beyond 15 years.

2. Seabrook cited a TS surveillance requirement (SR) for performing containment inspections that is not found in the KPS TS. Seabrook TS 3.6.5.3 required the licensee to conduct a periodic visual inspection of the concrete containment enclosure building every other refueling outage to ensure its integrity. This TS comes from NUREG-1431 (reference 22) SR 3.6.1.1. KPS is currently in the process of converting to improved standard TS and adoption of these TS will be evaluated during the TS conversion project. At KPS, containment and shield building inspections are performed under the Maintenance Rule program (10 CFR 50.65) and ASME Code requirements (KPS TS 6.20). See sections 3.4.7 and 3.4.8 for details.

3. St. Lucie Unit 2 requested a Type A test interval extension based in part on the projected need to perform a Type A test two times in a short time period. The first performance would be necessary to meet the previously approved TS interval. The second performance would be necessary due to a steam generator replacement project which required cutting a hole in the containment structure to bring in and set the new steam generators. After repair of the hole in the containment structure another Type A test would be required to verify containment leakage rates were satisfactory.

The differences stated above do not constitute a significant departure from the bases for NRC approval of the referenced amendments. There are no significant differences in risk or in the methodologies employed for determining risk provided in these amendments. Therefore, it is acceptable to use these license amendment requests and NRC approvals as past precedence in the development of this license amendment request.

4.0 TECHNICAL ANALYSIS

The proposed amendment would change the KPS TS to permit a one-time, six-month extension, to the currently approved 15-year interval between performances of Type A tests. The proposed amendment to extend the Type A surveillance test interval by six months is justified based on previous Type A test results, Type B and Type C test results, containment inspection results, and risk evaluation results.

4.1 Previous Type A, Type B and C Testing Results

Previous Type A testing confirmed that the KPS containment structure leakage is acceptable, with considerable margin, with respect to the TS acceptance criterion of 0.20% of primary containment air weight per day (1.0 L_a). The Type A test results and methods used to determine containment leakage are shown in Section 3.3.1.

Previous Type B and Type C testing confirmed that KPS containment penetration and valve leakage is acceptable, with considerable margin, with respect to the TS acceptance criterion of 0.20% of primary containment air weight per day (0.6 L_a). Type B and C test methods and results are provided in Section 3.4.1.

As documented in NUREG-1493 (reference 10), industry experience has shown that most Type A test failures result from leakage that is detectable by LLRT (Type B and C testing as defined in 10 CFR 50, Appendix J). The KPS Type B and C testing requirements per the Containment Leak Rate Testing Program are unaffected by this proposed amendment. The Type B and C test results provide reasonable assurance that the most likely sources of leakage have been identified and repaired.

4.2 Containment Inspection Programs

Comprehensive containment inspection programs are in place at KPS. The containment ISI program has identified minor deficiencies that have been repaired upon discovery throughout the years. No major defects of the containment structure have been discovered during the first ten-year inspection interval or the second ten-year inspection interval. As a result of the continued high integrity of the containment structure, no augmented examinations have been required.

In addition, containment bellows leak rate testing provides assurance that the containment bellows are functioning satisfactorily and have very low leakage rates.

Finally, results of inspections performed as part of the Maintenance Rule program and shield building inspection program provide further assurance that a diversity of structures are available and functioning satisfactorily to contain any leakage past the reactor containment vessel.

The KPS containment inspection programs are unaffected by this proposed amendment. The containment inspection programs will, continue to provide assurance that flaws in the containment structure will be identified and repaired.

4.3 Risk Impact

Attachment 3 provides a KPS site-specific risk evaluation for extending the Type A test interval to once-in-15 years - 9 months. The risk evaluation was performed using the latest guidance reviewed and approved by the NRC. This guidance consists of NEI 94-01, revision 2 (reference 9), EPRI Report No. 1009325 (reference 23) and the NRC safety evaluation for these documents, including certain exceptions taken by the NRC (reference 8). The evaluation provides the following conclusions:

1. The 15 year – 9 month Type A test interval does not impact CDF. Therefore, the proposed test interval extension does not result in a change in the CDF figure of merit. The increase in LERF resulting from the evaluated change in the Type A test interval from one in 10 years to one in 15 years – 9 months is conservatively calculated as 8.3E-7/yr using the NEI guidance. Per Regulatory Guide (RG) 1.174 definition and acceptance criteria, this increase is defined as a "small increase" and is acceptable on the basis that:

    - RG 1.174 states that "when the calculated increase in LERF is in the range of 10^{-7} per reactor year to 10^{-6} per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 10^{-5} per reactor year (Region II)."
    - Regions are established by a measure of the baseline risk metric (CDF or LERF) and the change in those metrics (CDF or LERF). Acceptance guidelines are established for each region.
    - The KPS baseline LERF estimates are 4.86E-6/year, 3.18E-8/year, and 5.08E-6/year for internal event, fire, and seismic hazards, respectively. Although, these results show a total LERF of slightly less than 1.0E-5/year, a realistic total LERF can reasonably be shown to be much smaller than this estimate. For example, as can be seen from the above estimates, the largest contributor to the total base line LERF estimate is from the seismic hazard. This estimate is based on the IPEEE seismic model which has been shown to be very conservative. The 2005 Kewaunee flooding significance determination showed significantly lower risk when more recent hazard curves and spectra were used. Another significant conservatism in the seismic LERF estimate is that every containment failure is characterized as a contributor to the large early release frequency figure of merit.

    The NRC and the nuclear industry have identified that due to highly different level of details, applied conservatism, and uncertainty, it is not appropriate to add LERF estimates from different hazard types (i.e., internal events vs. external events). This concern has been recognized and documented in EPRI 1010068 (reference 34). The aggregation issue is one of several issues that are being addressed as part of an addendum to the Memorandum of Understanding between the NRC and EPRI as described in SECY-07-0176 (reference 35, Enclosure 1, page 8).

2. The evaluated change in Type A test frequency from once-in-10 years to once-in-15 years – 9 months, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 1.4E-2 person-rem/yr, which is 0.067% of the total accident dose. This meets the acceptance criteria of less than 1.0 person-rem/yr or 1% of the total accident dose provided by the NRC SE dated June 25, 2008 (reference 8) for approval of permanent 15-year Type A test intervals.

3. The increase in the CCFP from extending the Type A test interval from once-in-10-years to once-in-15 years – 9 months is 0.4%. This meets the acceptance criteria of less than 1.5%, provided in the NRC SE dated June 25, 2008 (reference 8) for approval of permanent 15-year Type A test intervals.

Therefore, increasing the Type A test interval to 15 years – 9 months is considered to be acceptable since it represents a small change to the Kewaunee risk profile and meets the acceptance criteria stated in the NRC SE dated June 25, 2008 (reference 8).

This conclusion is consistent with NUREG-1493, Section 10.1.2:

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

4.4 Conclusions

The proposed TS change to permit a six-month extension to the currently approved one-time 15-year interval between performances of Type A tests is acceptable. The elements for determining the acceptability of such amendments have been satisfied. The elements required to justify such amendments are as follows:

4.4.1 Satisfactory Risk Impact Assessment

A risk evaluation found acceptable results in three required areas. These areas are percentage change in maximum total dose risk, maximum change in LERF due to the Type A test interval extension, and maximum change in CCFP due to the Type A test interval extension.

DEK performed a risk impact assessment for extending the Type A test interval to 15 years – 9 months (Attachment 3) in support of this proposed amendment. The proposed extension of the Type A test interval does not impact core damage risk or the reliability of the containment isolation valves to close on demand. The results of the areas evaluated are as follows:

1. The 15 year – 9 month Type A test interval does not impact CDF. Therefore, the proposed test interval extension does not result in a change in the CDF figure of merit. The increase in LERF resulting from the evaluated change in the Type A test interval from one in 10 years to one in 15 years – 9 months is conservatively calculated as 8.3E-7/yr using the NEI guidance. Per Regulatory Guide (RG) 1.174 definition and acceptance criteria, this increase is defined as a "small increase" and is acceptable given the total LERF is less than 1.0E-5.

2. Extending the Type A test interval presents an extremely small increase in total population dose. The calculated increase in dose is 1.4E-2 person-rem/yr, which is 0.067% of the total population dose. This meets the acceptance criteria provided in the NRC SE dated June 25, 2008 (reference 8) for increase in population dose of less than 1.0 person-rem/yr or 1% of the total population dose.

3. The increase in the CCFP from extending the Type A test interval from once-in-10-years to once-in-15 years – 9 months is 0.4%. This meets the NRC acceptance criteria of less than 1.5%.

As discussed in section 3.6, KPS has spent considerable resources and effort to increase the quality of the KPS PRA analysis. A recently performed assessment indicates that while the current KPS PRA analysis does not meet all SRs at Capability Category II levels, overall the PRA meets the expectations for PRA technical adequacy as required by the NRC SE (reference 8). This ensures that the analysis performed in support of this license amendment is valid for KPS.

4.4.2 Adequate Assurance that the Containment Structural Integrity will be Satisfactorily Maintained During the Extended Type A Test Interval Period

The KPS containment is a large steel structure. The containment pressure boundary consists of the steel liner, containment access penetrations, and process piping and electrical penetrations. The KPS containment vessel is freestanding, and the fraction of the surface area that is inaccessible for inspection is significantly smaller than at plants with steel-lined concrete containment structures. As a result, the potential contribution to risk from corrosion in uninspectable areas for an extended Type A test interval is less.

DEK maintains a comprehensive containment ISI program pursuant to the requirements of Subsection IWE of Section XI of the 2001 Edition with the 2003 Addenda of the ASME Boiler and Pressure Vessel Code. This program examines the accessible pressure-retaining surfaces to periodically monitor the condition of the containment building. Historical inspections of containment performed under this program have not identified any suspect areas.

DEK performs inspections of the reinforced concrete and steel containment structures to identify degradation of the containment steel shell of the primary containment. This is performed in conjunction with scheduled containment ISI inspections. The moisture barrier areas on the periphery of the inaccessible portion of the containment vessel are also inspected. Appropriate action is taken for any indications of degradation in these areas.

The integrity of the containment penetrations is verified through Type B and Type C testing as required by 10 CFR 50, Appendix J and the overall integrity of the containment structure is verified through periodic Type A tests. The leak rate testing requirements of Option B of 10 CFR 50, Appendix J, and the containment ISI requirements mandated by 10 CFR 50.55a complement each other in ensuring the leak-tightness and structural integrity of the containment.

4.4.3 A Demonstrated Need for Extending the Type A Test Interval

DEK has demonstrated compelling reasons and unforeseen emergent conditions that necessitate the extension of the Type A test interval.

The compelling reasons include:

1. The safety benefit of performing the Type A test in April, rather than October 2009 is insignificant and not commensurate with the lost generation and risk associated with an additional shutdown of Kewaunee Power Station for the sole purpose of performing a Type A test.
2. Results of past Type A tests – No Type A test failures have occurred throughout KPS history.
3. Interval extensions are allowed as discussed in applicable industry documents that govern Type A testing – This includes NRC approved documents such as NEI 94-01, revision 0, RG 1.163, and NEI 94-01, revision 2.
4. Minimal change in risk – NUREG-1493 demonstrated that the change in risk associated with extending Type A test intervals to 20 years between performances was imperceptible.
5. Type A test failures are rare – Type A tests detect less than 3% of leakage paths as compared to Type B and Type C test detection rates.
6. TS surveillance testing approach – The NRC recognizes that there is very little risk significance in delaying a TS surveillance test.
7. There are no performance-based or risk-based reasons for not extending the Type A test interval.

The unforeseen emergent conditions are:

1. An unforeseen shutdown period - The series of extended forced shutdowns was unforeseen when the original request to extend the Type A test interval was made.
2. An evaluation error was made by the plant staff when moving outage dates following the extended forced outages. This error was not identified until after completion of the most recent refueling outage prior to the required performance of the Type A test.

Finally, the proposed amendment is very similar in nature, bases and substance to approved amendments from three other licensees.

Therefore, based on the considerations discussed previously, (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.

5.0 REGULATORY SAFETY ANALYSIS

5.1. No Significant Hazards Consideration

The proposed amendment would change KPS Technical Specification (TS) 4.4.a, *"Integrated Leak Rate Tests (Type A),"* to permit a one-time, six-month extension, to the currently approved 15-year interval between performances of Type A containment integrated leak rate tests (ILRT). TS 4.4.a currently requires that the next Type A test be performed within 15 years of the previous test. The last Type A test performed at KPS was completed in April 1994 and the next test is currently required to be performed no later than the end of April 2009. The proposed amendment would extend the current Type A test interval, on a one-time basis, to October 2009 to coincide with the next scheduled refueling outage (KR 30).

Dominion Energy Kewaunee, Inc. has evaluated the proposed amendment to determine if a significant hazards consideration is involved by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The probability or consequences of accidents previously evaluated in the Updated Safety Analysis Report are unaffected by this proposed change. There is no change to any equipment response or accident mitigation scenario, and this change results in no additional challenges to fission product barrier integrity. The proposed change does not alter the design, configuration, operation, or function of any plant system, structure, or component. As a result, the probabilities of previously evaluated accidents are unaffected. The proposed extension to the Type A test interval does not involve a significant increase in consequences because, as discussed in NUREG-1493, Performance Based Containment Leak Rate Test Program, Type B and C tests identify the vast majority (approximately 97 percent) of all potential leakage paths. Further, Type A tests identify only a few potential leakage paths that cannot be identified through Type B and C testing, and leaks found by Type A testing have been only marginally greater than existing requirements. The frequency and methods of performance of Type B and Type C testing are unaffected by this proposed change. In addition, periodic inspections of containment required by the ASME code and the maintenance rule, which are capable of detecting any significant degradation, are unaffected by the proposed change.

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

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The proposed change does not challenge the performance or integrity of any safety-related system. The proposed change does not install or remove any plant equipment. The proposed change does not alter the design, physical configuration, or mode of operation of any plant structure, system, or component. No physical changes are being made to the plant, so no new accident causal mechanisms are being introduced. The proposed change only changes the frequency of performing the next Type A test; the Type A test implementation and acceptance criteria are unchanged. Type B and Type C testing frequency and method of performance are not affected by this proposed change.

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

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The margin of safety associated with the acceptance criteria of any accident is unchanged. The proposed change will have no affect on the availability, operability, or performance of the safety-related systems and components. The proposed change does not alter the design, configuration, operation, or function of any plant system, structure, or component. The ability of operable structures, systems, and components to perform their designated safety function is unaffected by this proposed change. NUREG-1493 concluded that reducing the frequency of Type A tests to one-in-20 years resulted in an imperceptible increase in risk. Type B and Type C testing frequency and method of performance are unaffected by this proposed change. Also, inspections of containment, required by the ASME code and the maintenance rule, provide reasonable assurance that containment will not degrade in a manner that is only detectable by Type A testing. In addition, the inherent risk of an additional plant shutdown would be eliminated by the proposed amendment, further ensuring no significant reduction in safety margin.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

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

5.2. Applicable Regulatory Requirements/Criteria

The US Atomic Energy Commission (AEC) issued their Safety Evaluation (SE) of the Kewaunee Power Station (KPS) on July 24, 1972 with supplements dated December 18, 1972 and May 10, 1973. The SE, Section 3.1, "Conformance with AEC General Design Criteria," described the conclusions the AEC reached associated with the General Design Criteria in effect at the time. The AEC stated:

> *The Kewaunee plant was designed and constructed to meet the intent of the AEC's General Design Criteria, as originally proposed in July 1967. Construction of the plant was about 50% complete and the Final Safety Analysis Report (Amendment No. 7) had been filed with the Commission before publication of the revised General Design Criteria in February 1971 and the present version of the criteria in July 1971. As a result, we did not require the applicant to reanalyze the plant or resubmit the FSAR. However, our technical review did assess the plant against the General Design Criteria now in effect and we are satisfied that the plant design generally conforms to the intent of these criteria.*

The KPS containment and containment testing is designed to meet the following General Design Criteria as it applies to this license amendment request. Discussion of the General Design Criteria and how KPS is designed to meet these requirements are included in the KPS Updated Safety Analysis Report (reference 7).

Criterion 49 – Containment Design Basis
The containment structure including access openings, penetrations and any necessary containment heat removal systems, shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a Loss-of-Coolant Accident, including a considerable margin for effects from metal-water or other chemical reactions, that could occur as a consequence of failure of Emergency Core Cooling Systems.

Criterion 53 – Containment Isolation Valves
Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

Criterion 54 – Containment Leakage Rate Testing
Containment shall be designed so that an [sic] *integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period of time to verify its conformance with required performance.*

Criterion 55 – Containment Periodic Leakage Rate Testing
The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

Criterion 56 – Provisions For Testing Of Penetrations
Provisions shall be made for testing penetrations, which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at design pressure at any time.

Criterion 57 – Provisions For Testing Of Isolation Valves
Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

6.0 ENVIRONMENTAL CONSIDERATION

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

7.0 REFERENCES

1. Letter from T. Coutu (NMC) to Document Control Desk (NRC), "License Amendment Request 198 to the Kewaunee Nuclear Power Plant Technical Specifications for One-Time Extension of Containment Integrated Leak Rate Test Interval," dated June 20, 2003. [ADAMS Accession No. ML031820613]
2. Letter from T. Coutu (NMC) to Document Control Desk (NRC), "License Amendment Request 198, 'ILRT 5-Year Extension,' NMC Response to NRC Request for Additional Information," dated December 12, 2003. [ADAMS Accession No. ML033570469]
3. Letter from J. G. Lamb (NRC) to T. Coutu (NMC), "Kewaunee Nuclear Power Plant – Issuance of Amendment (TAC No. MB9907)," dated April 6, 2004. [ADAMS Accession No. ML040340168]
4. Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. [ADAMS Accession No. ML003740058]

5. 10 CFR 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.”

6. NEI 94-01, “Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J,” Revision 0, dated July 26, 1995.

7. Kewaunee Power Station Updated Safety Analysis Report, Revision 20.

8. Letter from M. J. Maxin (NRC) to J. C. Butler (NEI), “Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) 94-01, Revision 2, ‘Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J’ and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, August 2007, ‘Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals’ (TAC No. MC9663),” dated June 25, 2008. [ADAMS Accession No. ML081140105]

9. NEI 94-01, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” Revision 2, dated August 2007.

10. NUREG-1493, “Performance-Based Containment Leak-Test Program,” dated July 1995.

11. Letter from D. H. Jaffe, (NRC) to D. A. Christian (DEK), “Kewaunee Power Station - Issuance of Amendment RE: Surveillance Interval Extension (TAC No. MC9782),” dated July 12, 2006. [ADAMS Accession No. ML061640286]

12. Regulatory Guide 1.174, Revision 1, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” dated November 2002. [ADAMS Accession No. ML023240437]

13. NRC Information Notice 92-20, “Inadequate Local Leak Rate Testing,“ dated March 3, 1992.

14. Regulatory Guide 1.200, Revision 1, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” dated January 2007. [ADAMS Accession No. ML070240001]

15. Letter from R. J. King (Entergy – River Bend Station) to Document Control Desk (NRC), “License Amendment Request, LAR 2005-01, One-time Extension of the Integrated Leak Rate Test Interval,” dated March 8, 2005. [ADAMS Accession No. ML050740346]

16. Letter from R. J. King (Entergy – River Bend Station) to Document Control Desk (NRC), “Supplement to Amendment Request One Time Extension to the Integrated Leak Rate Test Interval,” dated January 17, 2006. [ADAMS Accession No. ML060230049]

17. Letter from B. Vaidya (NRC) to P. D. Hinnenkamp (Entergy – River Bend Station), “River Bend Station, Unit 1 - Issuance of Amendment RE: Additional Extension Of Appendix J, Type A Integrated Leakage Rate Test Interval (TAC No. MC6328),” dated February 9, 2006. [ADAMS Accession No. ML060410310]

18. Letter from J. C. Roberts (Entergy – River Bend Station) to Document Control Desk (NRC), “Subject: License Amendment Request, LAR 2007-10, One-time

Extension of the Integrated Leak Rate Test Interval," dated August 17, 2007. [ADAMS Accession No. ML072340544]

19. Letter from B. Vaidya (NRC) to J. E. Venable (Entergy – River Bend Station), "River Bend Station, Unit 1 - Issuance of Amendment RE: One-Time Extension Of The Integrated Leak Rate Test Interval (TAC No. MD6528)," dated December 3, 2007. [ADAMS Accession No. ML072880125]
20. Letter from G. St. Pierre (FPL Energy Seabrook, LLC.) to Document Control Desk (NRC), "License Amendment Request 05-06, Application for Amendment to Technical Specification 6.15 for a Six-Month Extension to the Containment Integrated Leak Rate Test Interval," dated September 29, 2005. [ADAMS Accession No. ML052770187]
21. Letter from G. E. Miller (NRC) to G. St. Pierre (FPL Energy Seabrook, LLC.), "Seabrook Station, Unit No. 1 - Issuance of Amendment RE: Six-Month Extension For The Containment Integrated Leakage Rate Test Interval (TAC No. MC8549)," dated March 24, 2006. [ADAMS Accession No. ML060520032]
22. NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 3.1, dated December 1, 2005.
23. EPRI Report No. 1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," dated August 2007.
24. EPRI Report No. TR-104285, Final Report, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, " dated August 1994.
25. Letter from W. Jefferson, Jr. (FPL) to Document Control Desk (NRC), "Proposed License Amendment Containment Leakage Rate Program One-Time Type A Test Interval Extension," dated March 31, 2005. [ADAMS Accession No. ML050950235]
26. Letter from B. T. Moroney (NRC) to J. A. Stall (FPL), "St. Lucie Plant, Unit No. 2 - Issuance of Amendment Regarding Type A Test Interval Extension (TAC No. MC6629)," dated December 23, 2005. [ADAMS Accession No. ML053190343]
27. ANSI/ANS-56.8-1994, "Containment System Leakage Testing Requirements," dated August 1994.
28. Letter from R. F. Kuntz (NRC) to D. A. Christian (Dominion), "Kewaunee Power Station – Issuance of Amendment RE: Radiological Accident Analysis and Associated Technical Specification Change (TAC No. MC9715)," dated March 8, 2007. [ADAMS Accession No. ML070430020]
29. Letter from W. O. Long (NRC) to M. L. Marchi (WPSC), "Amendment No. 136 to Facility Operating License No. DPR-43 – Kewaunee Nuclear Power Plant (TAC No. MA1271)," dated May 28, 1998. [ADAMS Accession No. ML020770280]
30. Letter from W. O. Long (NRC) to M. L. Marchi (WPSC), "Corrections to Amendment 136 to Facility Operating License No. DPR-43 – Kewaunee Nuclear Power Plant (TAC No. MA1271)," dated June 24, 1998. [ADAMS Accession No. ML020770256]

31. Regulatory Guide 1.177, "An Approach for Plant Specific, Risk Informed Decisionmaking: Technical Specifications," dated August, 1998. [ADAMS Accession No. ML003740176]

32. Kewaunee Nuclear Power Plant Probabilistic Risk Assessment Peer Review Report, December 2002.

33. NRC Mitigating System Performance Index (MSPI) Basis Document, Kewaunee Power Station, Revision F, June 2008.

34. EPRI Report No. 1010068, Final Report, "Aggregation of Quantitative Risk Assessment Results – Comparing and Manipulating Risk Metrics," dated December 2005.

35. SECY-07-0176, "Status of the Accident Sequence Precursor Program and the Development of Standardized Plant Analysis Risk Models," dated October 3, 2007. [ADAMS Accession No. ML072070469]

36. Letter from G. T. Bischoff (Dominion) to Document Control Desk (NRC), "Letter of Intent to Adopt NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants, 2001 Edition," dated July 21, 2008. [ADAMS Accession No. ML082040111]

ATTACHMENT 2

LICENSE AMENDMENT REQUEST 242

EXTENSION OF THE ONE-TIME FIFTEEN YEAR CONTAINMENT INTEGRATED LEAK RATE TEST INTERVAL

MARKED-UP TECHNICAL SPECIFICATIONS PAGE

TS 4.4-1

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

4.4 CONTAINMENT TESTS

APPLICABILITY

Applies to integrity testing of the steel containment, shield building, auxiliary building special ventilation zone, and the associated systems including isolation valves.

OBJECTIVE

To verify that leakage from the containment system is maintained within allowable limits in accordance with 10 CFR Part 50, Appendix J.

SPECIFICATION

a. Integrated Leak Rate Tests (Type A)

   Perform required visual examinations and leakage rate testing in accordance with the Containment Leakage Rate Testing Program.

   As a one-time change, the Type A test frequency specified in NEI 94-01, Revision 0, Paragraph 9.2.3, as "...at least once per 10 years based on acceptable performance history" is changed to "...at least once per 15 years based on acceptable performance history." This change applies only to the interval following the Type A test performed in April 1994. <u>The first ILRT performed after April 1994 shall be performed no later than October 2009.</u>

b. Local Leak Rate Tests (Type B and C)

   Perform required air lock, penetration, and containment isolation valve leakage testing in accordance with the Containment Leakage Rate Testing Program.

c. Shield Building Ventilation System

   1. At least once per operating cycle or once every 18 months, whichever occurs first, the following conditions shall be demonstrated:

      a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is < 10 inches of water and the pressure drop across any HEPA filter bank is < 4 inches of water at the system design flow rate (±10%).

      b. Automatic initiation of each train of the system.

      c. Operability of heaters at rating and the absence of defects by visual observation.

ATTACHMENT 3

LICENSE AMENDMENT REQUEST 242

EXTENSION OF THE ONE-TIME FIFTEEN YEAR CONTAINMENT INTEGRATED LEAK RATE TEST INTERVAL

EVALUATION OF INCREASED RISK DUE TO ONE-TIME 9-MONTH EXTENSION OF INTEGRATED LEAK RATE TESTING INTERVAL

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

Kewaunee Power Station
Probabilistic Risk Assessment Notebook

Part V PRA Risk Analysis
Volume KPS.RA.LI.3 – Evaluation of Increased Risk due to One-Time 9-Month Extension of Type A Containment Test Interval

Revision No. 3
Effective Date: September 2008

Purpose: The purpose of this analysis is to provide a risk assessment of a one-time extension of the currently allowed containment Type A leak rate test at the Kewaunee Power Station by 9 months. The extension would allow for substantial cost savings as the Type A Test could be deferred to the fall 2009 refueling outage (RF30). The risk assessment follows industry guidelines.		
Conclusion: Increasing the Type A test interval to 15 years, 9 months is considered to be insignificant since it represents a very small change to the Kewaunee risk profile.		
Prepared By:	**Signature**	**Date** September 9, 2008
Reviewed By:	**Signature**	**Date** September 9, 2008
Approved By:	**Signature**	**Date** September 9, 2008

Table of Contents

1.0 PURPOSE

The purpose of this analysis is to provide a risk assessment of a one-time extension of the currently allowed containment Type A leak rate test at the Kewaunee Power Station by 9 months. The extension would allow for substantial cost savings as the Type A test could be deferred to the fall 2009 refueling outage (RF30). The risk assessment follows the guidelines from NEI 94-01, Revision 1 [REPORT01], the methodology used in EPRI TR-104285 [REPORT02], the NEI, "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals," dated November 2001 [REPORT03], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of licensing basis changes as outlined in Regulatory Guide (RG) 1.174 [GUIDE01], the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval [LETTER01], and the methodology used in EPRI 1009325, Revision 2 [REPORT04].

2.0 INTRODUCTION

On April 6, 2004, Kewaunee received approval from the Nuclear Regulatory Commission allowing a one-time extension of the Type A containment test interval to 15 years [LETTER02]. The basis for this extension was the methodology from EPRI TR-104285. Approval of the 15 year Type A test interval was based on an acceptable performance history, defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated leakage was less than allowable containment leakage of 1La (0.2%/day at 46 psig [REPORT11]).

An NRC report on performance-based leak testing, NUREG-1493 [REPORT05], analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from containment leak rate testing. NUREG-1493 determined that for a representative PWR plant (i.e., Surry) that containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Consequently, it can be shown that extending the Type A test interval will not lead to a substantial increase in risk from containment isolation failures for Kewaunee.

In 2005-2006 a series of mid-cycle outages occurred. As a result, each subsequent outage was delayed approximately 6 months. The initial plan was to perform a Type A test during Refueling Outage RF-30, which was before the date that the 15 year Type A test interval was to expire, April 2009 [LETTER02]. Due to the delay, a 6-month extension to the currently approved 15 year Type A test interval is required to perform the ILRT during RF-30. This assessment uses a 15 year, 9 month interval, which bounds the 15 year, 6 month interval actually being requested. The last Type A test was performed in April of 1994.

This assessment is based on the methodology of EPRI Report No. 1009325, Revision 2, *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals.* The guidance provided in Appendix H of EPRI Report 1009325, Revision 2, for performing risk impact assessments in support of Type A test extensions builds on the EPRI risk assessment methodology of EPRI TR-104285. This methodology is used to determine the appropriate risk information for evaluating the impact of the proposed Type A test interval change.

It should be noted that the containment is also subjected to periodic inservice inspections in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI. More specifically, Subsection IWE provides the rules and requirements for in-service inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulation 10 CFR 50.55a(b)(2)(ix)(E) requires visual inspections of the accessible areas of the interior of the containment. The associated change to NEI 94-01 will require that visual examinations be conducted during at least three other outages, and during the outage in which the ILRT is conducted. These requirements are not changed as a result of the proposed Type A test interval extension. In addition the frequency of Appendix J, Type B local leak tests; which verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets; are also not affected by the change to the Type A test frequency.

2.1 Acceptance Guidelines

Based on a review of NRC SE [LETTER06], pages 17 and 18, the following acceptance guidelines are used in this report:

LERF Acceptance Guidelines

The acceptance guidelines in RG 1.174 are used to assess the acceptability of this one-time extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J. RG 1.174 defines very small changes in the risk acceptance guidelines as increases in core damage frequency (CDF) less than 1E-6 per reactor year and increases in large early release frequency (LERF) less than 1E-7 per reactor year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. RG 1.174 also defines small changes in LERF as below 1E-6 per reactor year.

Population Dose

An acceptable increase (defined as a small increase) in population dose is defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive.

Conditional Containment Failure Probability (CCFP)

An acceptable increase (defined as a small increase) is defined as the increase in CCFP which is less than or equal to 1.5 percentage point.

3.0 ANALYSIS

3.1 Inputs

This section summarizes the general resources available as input (Section 3.1.1) and the plant specific resources required (Section 3.1.2).

3.1.1 GENERAL RESOURCES AVAILABLE

Various industry studies on containment leakage risk assessment are briefly summarized here.

1. NUREG/CR-3539 [REPORT06]
2. NUREG/CR-4220 [REPORT07]
3. NUREG-1273 [REPORT08]
4. NUREG/CR-4330 [REPORT09]
5. EPRI TR-105189 [REPORT10]
6. NUREG-1493 [REPORT05]
7. EPRI TR-104285 [REPORT02]
8. NEI Interim Guidance [REPORT03, LETTER03]
9. Calvert Cliffs liner corrosion analysis [LETTER01]
10. EPRI report 1009325, Revision 2 [REPORT04], Appendix H

The first study is applicable because it provides a basis for the containment leakage size threshold in the Level 2 PRA that is considered significant. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from Type A test interval extensions. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the Type A test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending Types A, B and C test intervals on at-power public risk. The eighth study includes the NEI recommended methodology (promulgated in two letters) for evaluating the risk associated with obtaining a one-time extension of the Type A test interval. The ninth study addresses the impact of age related degradation of containment liners on Type A test interval extension evaluations. Finally, the tenth study builds on the previous work and includes a recommended methodology and template for evaluating the risk associated with Type A test interval extensions.

3.1.2 PLANT-SPECIFIC INPUTS

The plant-specific information used to perform the Kewaunee Type A test interval Extension Risk Assessment includes the following:

- Level 1 Model results [NB01]
- Level 2 Model results [NB05]
- Release category definitions used in the Level 2 Model [NB02]
- Dose Assessment [LETTER04]

Internal Events Level 1 Model

The Internal Events Level 1 PRA model that is used for Kewaunee [MODEL01] is characteristic of the as-built plant. The current Level 1 model is a linked fault tree model and was quantified with the total core damage frequency (CDF) = 4.21E-5/yr. When broken down by release category, additional cutsets that are non-minimal for core damage but minimal for the release category are introduced. Therefore, the sum of the release category frequencies is somewhat higher, 4.32E-5/yr.

Internal Events Level 2 Model

The Kewaunee Internal Events Level 2 Model was developed to calculate the LERF contribution as well as the other release categories evaluated in the model. Table 3.1.2-1 summarizes the pertinent Kewaunee results in terms of release category for internal and external events. Internal Events frequencies are from [NB05] and fire and seismic frequencies are from [LETTER04]. These categories are then mapped to the appropriate EPRI categories defined in Table 3.1.2-2

External Events

The fire and seismic release category frequencies in Table 3.1.2-1 were taken from the model used for the Type A test interval extension to 15 years [LETTER04, Enclosure 1, Page 8]. No significant changes have been made to these models since that time. Other external events screened out of the Individual Plant Examination for External Events as unimportant and no subsequent assessment has indicated that this has changed.

Table 3.1.2-1
Kewaunee Level 2 PRA Model Release Categories and Frequencies

Release Category [NB02]	Definition	Internal Freq/yr	Fire Freq/yr	Seismic Freq/yr	Total Freq/yr	EPRI Category
1	No In-Vessel Core Cooling, No Containment Failure	5.95E-7	3.80E-6	0	4.40E-6	1
2	No In-Vessel Core Cooling, Late Containment Failure, Continuous Spray	0	0	0	0	7
3	No In-Vessel Core Cooling, Late Containment Failure, Early Spray	0	0	0	0	7
4	No In-Vessel Core Cooling, Late Containment Failure, No Spray	2.72E-5	3.80E-7	0	2.76E-5	7
5	No In-Vessel Core Cooling, Basemat Meltthrough	1.07E-7	1.37E-4	4.31E-6	1.41E-4	7
6	No In-Vessel Core Cooling, Small Containment Isolation Failure	2.84E-9	6.05E-7	0	6.08E-7	2
7	No In-Vessel Core Cooling, Large Containment Isolation Failure (LERF)	1.52E-8	3.18E-8	0	4.70E-8	2
8	In-Vessel Cooling, Containment Isolated	9.93E-6	0	0	9.93E-6	1
9	In-Vessel Core Cooling, Small Containment Isolation Failure	0	0	0	0	2
10	In-Vessel Core Cooling, Large Containment Isolation Failure (LERF)	0	0	0	0	2
11	Interfacing System LOCA with Scrubbing	1.23E-7	0	0	1.23E-7	8
12	Interfacing System LOCA with no Scrubbing (LERF)	2.17E-7	0	5.08E-6	5.30E-6	8
13	Steam Generator Tube Rupture with no Scrubbing (LERF)	4.70E-6	0	0	4.70E-6	8
14	Steam Generator Tube Rupture with Scrubbing	3.07E-7	0	0	3.07E-7	8
	Total Release Category Frequency	4.32E-5	1.42E-4	9.39E-6	1.94E-4	
	LERF	4.86E-6*	3.18E-8#	5.08E-6#		

* From [NB05]
# From [LETTER04]

Table 3.1.2-2 defines the containment failure classifications used in the Type A test interval extension evaluation. These classifications are consistent with the EPRI/NEI methodology [REPORT02] and are used in this analysis to determine the risk impact of extending the Containment Type A test interval as described in Section 4 of this report.

Table 3.1.2-2
EPRI/NEI CONTAINMENT FAILURE CLASSIFICATIONS

Class	Description
1	Containment remains intact, including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values L_a, under Appendix J for that plant.
2	Containment isolation failures (as reported in the IPEs) include those accidents in which there is a failure to isolate the containment.
3	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress.
4	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures but is applicable to sequences involving Type B tests and their potential failures. These are the Type B-tested components that have isolated but exhibit excessive leakage.
5	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.
6	Containment isolation failures include those leak paths covered in the plant test and maintenance requirements or verified per in-service inspection and testing (ISI/IST) program.
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
8	Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in Class 8. Changes in Appendix J testing requirements do not impact these accidents.

The population dose in Table 3.1.2-3 is from the submittal for the one-time extension to 15 years [LETTER04, Enclosure 1, Page 13]. Use of the previous analysis is somewhat conservative in that it assumes a higher allowable leakage (La) than the current Technical Specifications allow. Technical Specification amendment 190 decreased La from 0.5 %/day to 0.2%/day at 46 psig [LETTER07]. The dose was calculated using the MACCS code for the Point Beach Nuclear Plant located 5 miles from Kewaunee. A comparison of the features and surrounding conditions for the two site locations indicates that use of the PBNP inputs for KNPP will result in population doses appropriate or slightly conservative for KNPP. The dose from each Kewaunee release category is shown in Table 3.1.2-3. The Kewaunee Level 2 model was revised extensively since the analysis in [LETTER04]. Therefore, the mapping of old and new release categories is provided below.

Table 3.1.2-3
RELEASE CATEGORY DOSES

[LETTER04] Release Category	Current Model Release Category	EPRI Accident Class	Population Dose (Person-REM)
1	1	1	1.20E+2
N/A	2	7a	Not Calculated
N/A	3	7a	Not Calculated
5	4	7a	4.04E+5
4	5	7b	7.51E+1
2	6	2a	2.01E+5
3	7	2b	2.97E+5
1	8	1	1.20E+2
2	9	2a	2.01E+5
3	10	2b	2.97E+5
7	11	8a	1.17E+6*
7	12	8a	1.17E+6
8	13	8b	6.35E+5
8	14	8b	6.35E+5*

* Release categories 11 and 14, crediting scrubbing in an interfacing systems LOCA and steam generator tube rupture, were developed after the Type A test interval extension to 15 years. The actual dose for these categories would be lower than the values for no scrubbing. Use of the larger dose is conservative in the calculation of the total dose and has no impact on the dose increase due to Type A test interval extension.

These categories are also mapped to the EPRI accident classes defined in Table 3.1.2-4. EPRI Categories 2, 7 and 8 have been split into two subcategories because there is more than one dose for each of these categories.

Table 3.1.2-4
ACCIDENT CLASSES

Accident Classes (Containment Release Type)	Description
1	No Containment Failure
2a	Large Isolation Failures (Failure to Close 2"- 5" Penetrations)
2b	Large Isolation Failures (Failure to Close >5" Penetrations)
3a	Small Isolation Failures (liner breach)
3b	Large Isolation Failures (liner breach)
4	Small Isolation Failures (Failure to seal –Type B)
5	Small Isolation Failures (Failure to seal—Type C)
6	Other Isolation Failures (for example, dependent failures)
7a	Failures Induced by Phenomena (Pressure Failure)
7b	Failures Induced by Phenomena (Basemat Meltthrough)
8a	Bypass (Interfacing System LOCA)
8b	Bypass (Steam Generator Tube Rupture)
CDF	All CET end states (including very low and no release)

Finally, the Kewaunee release category frequencies from Table 3.1.2-1 are mapped to EPRI Release categories and presented in Table 3.1.2-5.

Table 3.1.2-5
EPRI RELEASE CATEGORY FREQUENCIES AND POPULATION DOSES

EPRI Accident Class	Current Model Release Categories	Accident Class Frequency/yr	Dose Person-REM
1	1, 8	1.43E-5	1.20E+2
2a	6, 9	6.08E-7	2.01E+5
2b	7, 10	4.70E-8	2.97E+5
7a	2, 3, 4	2.76E-5	4.04E+5
7b	5	1.41E-4	7.51E+1
8a	11, 12	5.42E-6	1.17E+6
8b	13, 14	5.01E-6	6.35E+5
Total		1.94E-4	

3.1.3 IMPACT OF EXTENSION ON DETECTION OF COMPONENT FAILURES THAT LEAD TO LEAKAGE (SMALL AND LARGE)

Type A tests can detect a number of component failures, such as containment liner breach, failure of certain bellows arrangements, and failure of some containment sealing surfaces, which can lead to containment leakage. The proposed Type A test interval extension may influence the conditional probability of detecting these types of failures. To ensure that this effect is properly accounted for, the EPRI Class 3 accident class, as defined in Table 3.1.2-2, is divided into two sub-classes, Class 3a and Class 3b, representing small and large leakage failures, respectively.

The probability of the EPRI Class 3a and 3b failures is determined consistent with the EPRI Guidance [REPORT04]. For Class 3a, the probability is based on the maximum likelihood estimate of failure (arithmetic average) from the available data (i.e., two "small" failures in 217 tests leads to 2/217=0.0092). For Class 3b, Jeffreys Non-Informative Prior distribution is assumed for no "large" failures in 217 tests (i.e., 0.5/(217+1) = 0.0023).

In a follow-on letter [LETTER03] to their ILRT guidance document [REPORT03], NEI issued additional information concerning the potential that the calculated delta LERF values for several plants may fall above the "very small change" guidelines of the NRC Regulatory Guide 1.174. This additional NEI information includes a discussion of conservatisms in the quantitative guidance for delta LERF. NEI describes ways to demonstrate, using plant-specific calculations, that the delta LERF is smaller than that calculated by the simplified method. The supplemental information [LETTER03] states:

> "The methodology employed for determining LERF (Class 3b frequency) involves conservatively multiplying the CDF by the failure probability for this class (3b) of accident. This was done for simplicity and to maintain conservatism. However, some plant-specific accident classes leading to core damage are likely to include individual sequences that either may already (independently) cause a LERF or could never cause a LERF, and are thus not associated with a postulated large Type A containment leakage path (LERF). These contributors can be removed from Class 3b in the evaluation of LERF by multiplying the Class 3b probability by only that portion of CDF that may be impacted by Type A leakage."

The application of this additional guidance to the analysis for Kewaunee, as detailed in Section 5, involves the following:

- The Class 2 and Class 8 sequences are subtracted from the CDF that is applied to Class 3b.

To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF. Class 2 and Class 8 events refer to sequences with either large preexisting containment isolation failures or containment bypass events. These sequences are already considered to contribute to LERF in the Kewaunee Level 2 PRA analysis.

- A review of Class 1 accident sequences shows that none of these cases involve successful operation of containment sprays. Therefore, no correction for successful spray in applied

Consistent with the NEI guidance [REPORT03], the change in the leak detection probability can be estimated by comparing the average time that a leak could exist without detection. For example, the average time that a leak could exist without detection for a three-year test interval is 1.5 years (3 yr / 2), the average time that a leak could exist without detection for a 10-year interval is 5 years (10 yr / 2) and the average time that a leak could exist without detection for a 15-year 9-month interval is 7.875 years (15.75 yr / 2). The change to 10 years would lead to a non-detection probability that is a factor of 3.33 (5/1.5) higher for the probability of a leak that is detectable only by ILRT testing. Correspondingly, an extension of the ILRT interval from 10 years to 15 years 9 months can be estimated to lead to a factor of about a 1.58 (7.875/5) increase in the non-detection probability of a leak.

It should be noted that using the methodology discussed above is very conservative compared to previous submittals (for example, the IP3 request for a one-time ILRT extension that was approved by the NRC [LETTER05]) because it does not factor in the possibility that the failures could be detected by other tests (for example, the Type B local leak rate tests that will still occur.) Eliminating this possibility conservatively over-estimates the risk increases attributable to the Type A test interval extension.

3.1.4 IMPACT OF EXTENSION ON DETECTION OF STEEL LINER CORROSION THAT LEADS TO LEAKAGE

An estimate of the likelihood and risk implications of corrosion-induced leakage of the steel liners occurring and remaining undetected during the extended test interval is evaluated using the methodology from the Calvert Cliffs liner corrosion analysis [LETTER01]. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. Kewaunee's containment has a free-standing steel containment, but there are portions, such as the basemat, that are similar to Calvert Cliffs.

The following approach is used to determine the change in likelihood of detecting corrosion of the containment steel liner due to extending the Type A test interval. This likelihood is then used to determine the resulting change in risk. Consistent with the Calvert Cliffs analysis, the following issues are addressed:

- Differences between the containment basemat and the containment cylinder and dome
- The historical steel liner flaw likelihood due to concealed corrosion
- The impact of aging
- The corrosion leakage dependency on containment pressure

- The likelihood that visual inspections will be effective at detecting a flaw

Assumptions

- Consistent with the Calvert Cliffs analysis, a half-failure is assumed for basemat concealed liner corrosion due to the lack of identified failures. (See Table 3.1.4-1, Step 1.)

- The two corrosion events used to estimate the liner flaw probability in the Calvert Cliffs analysis are assumed to be applicable to this Kewaunee containment analysis. These events, one at North Anna Unit 2 and one at Brunswick Unit 2, initiated from the nonvisible (backside) portion of the containment liner.

- Consistent with the Calvert Cliffs analysis, the estimated historical flaw probability is also limited to 5.5 years to reflect the years since September 1996 when 10CFR 50.55a began requiring periodic visual inspections. Additional success data were not used to limit the aging impact of corrosion, even though inspections were being performed prior to this date (and have been performed since the time of the Calvert Cliffs analysis), and no evidence that additional liner corrosion has been identified. (See Table 3.1.4-1, Step 1.)

- Consistent with the Calvert Cliffs analysis, the steel liner flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increased likelihood of corrosion as the steel liner ages. (See Table 3.1.4-1, Steps 2 and 3.) Sensitivity studies are included that assume the steel liner flaw likelihood doubles every 10 years and every two years.

- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated as 1.1% for the cylinder and dome and 0.11% (10% of the cylinder failure probability) for the basemat. Conservative probabilities of 1% for the cylinder and dome and 0.1% for the basemat are used in this analysis, based on the methodology of [REPORT04] and sensitivity studies are included that increase and decrease the probabilities by an order of magnitude. (See Table 3.1.4-1, Step 4.)

- Consistent with the Calvert Cliffs analysis, the likelihood of leakage escape (due to crack formation) in the basemat region is considered to be less likely than leakage escape in the containment cylinder and dome region. (See Table 3.1.4-1, Step 4.)

- Consistent with the Calvert Cliffs analysis, a 5% visual inspection detection failure likelihood if the flaw is visible and a total detection failure likelihood of 10% are used. To date, all liner corrosion events have been detected through visual inspection. (See Table 3.1.4-1, Step 5.)

Sensitivity studies are included that evaluate total detection failure likelihood of 5% and 15%, respectively.

- Consistent with the Calvert Cliffs analysis, all non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

Table 3.1.4-1
Steel Liner Corrosion Base Case

<table>
<tr><th>Step</th><th>DESCRIPTION</th><th colspan="2">Containment Cylinder and Dome</th><th colspan="2">Containment Basemat</th></tr>
<tr><td>1</td><td>Historical Steel Liner Flaw Likelihood</td><td colspan="2">Events: 2</td><td colspan="2">Events: 0
(assume half a failure)</td></tr>
<tr><td></td><td>Failure Data: Containment location specific (consistent with Calvert Cliffs analysis).</td><td colspan="2">2/(70 * 5.5) = 5.2E-3</td><td colspan="2">0.5/(70 * 5.5) = 1.3E-3</td></tr>
<tr><td rowspan="2">2</td><td rowspan="2">Age-Adjusted Steel Liner Flaw Likelihood

During 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for 5th to 10th year is set to the historical failure rate (consistent with Calvert Cliffs analysis).</td><td>Year

1
avg 5-10
15
16</td><td>Failure Rate
2.1E-3
5.2E-3
1.4E-2
1.6E-2</td><td>Year

1
avg 5-10
15
16</td><td>Failure Rate
5.0E-4
1.3E-3
3.5E-3
4.0E-3</td></tr>
<tr><td colspan="2">15 year average = 6.27E-3
16 year average = 6.88E-3
16 years is chosen vs. 15.75 for simplicity. This is slightly conservative.</td><td colspan="2">15 year average = 1.57E-3
16 year average = 1.72E-3
16 years is chosen vs. 15.75 for simplicity. This is slightly conservative.</td></tr>
<tr><td>3</td><td>Flaw likelihood at 3, 10, and 15 years

Uses age-adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years (consistent with Calvert Cliffs analysis – see Table 6 of reference [LETTER01]).</td><td colspan="2">0.71% (1 to 3 years)
4.06% (1 to 10 years)
9.40% (1 to 15 years)
11.01% (1 to 16 years)
(Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3-, 15-, and 16-year intervals, consistent with the desired presentation of the results.</td><td colspan="2">0.18% (1 to 3 years)
1.02% (1 to 10 years)
2.35% (1 to 15 years)
2.75% (1 to 16 years)
(Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3-, 15-, and 16-year intervals consistent with desired presentation of the results.</td></tr>
</table>

Step	DESCRIPTION	Containment Cylinder and Dome	Containment Basemat
4	Likelihood of Breach in Containment Given Steel Liner Flaw The failure probability of the cylinder and dome is assumed to be 1% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 0.1% (compared to 0.11% in the Calvert Cliffs analysis).	**1%**	**0.1%**
5	**Visual Inspection Detection Failure Likelihood** Utilize assumptions consistent with Calvert Cliffs analysis.	**10%** 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT) All events have been detected through visual inspection. Visible failure detection of 5% is a conservative assumption.	**100%** Cannot be visually inspected.
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3 * 4* 5)	**0.00071% (at 3 years)** 0.71% * 1% * 10% **0.0041% (at 10 years)** 4.1% * 1% * 10% **0.0094% (at 15 years)** 9.4% * 1% * 10% **0.0110% (at 16 years)** 11.0% * 1% * 10%	**0.00018% (at 3 years)** 0.18% * 0.1% * 100% **0.0010% (at 10 years)** 1.0% * 0.1% * 100% **0.0024% (at 15 years)** 2.4% * 0.1% * 100% **0.0028% (at 16 years)** 2.8% * 0.1% * 100%

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum of Step 6 for the containment cylinder and dome and the containment basemat as summarized below for Kewaunee.

Total Likelihood of Non-Detected Containment Leakage Due to Corrosion for Kewaunee:

- At 3 years: 7.1E-6 + 1.8E-6 = 8.9E-6
- At 10 years: 4.1E-5 + 1.0E-5 = 5.1E-5
- At 16 years: 1.1E-4 + 2.8E-5 = 1.38E-4

The above factors are applied to those core damage accidents that are not already independently LERF or that could never result in LERF. The resulting increase is then added to the class 3b frequency. For example, the three-in-10 year case is calculated as follows:

For 3 in 10 years:

- From Table 4-2, the EPRI Class 3b frequency without considering corrosion is 4.21E-7/yr.
- As discussed in Section 4.1, the Kewaunee CDF associated with accidents that are not independently LERF or could never result in LERF is 1.94E-4 - 6.08E-7 - 4.70E-8 - 5.42E-6 - 5.01E-6 = 1.83E-4/yr
- The increase in the base case Class 3b frequency due to the corrosion-induced concealed flaw issue is calculated as 1.83E-4 * 8.9E-6 = 1.63E-9/yr, where 8.9E-6 was previously shown above to be the cumulative likelihood of non-detected containment leakage due to corrosion at 3 years. The three-in-10 year Class 3b frequency including the corrosion-induced concealed flaw issue is then calculated as 4.21E-7/yr + 1.63E-9/yr = 4.23E-7/yr.

For 1 in 10 years:

- From Table 4-3, the EPRI Class 3b frequency without considering corrosion is 1.40E-6/yr.
- The increase in the base case Class 3b frequency due to the corrosion-induced concealed flaw issue is calculated as 1.83E-4 * 5.1E-5 = 9.33E-9/yr, where 5.1E-5 was previously shown above to be the cumulative likelihood of non-detected containment leakage due to corrosion at 10 years.
The 1-in-10 year Class 3b frequency including the corrosion-induced concealed flaw issue is then calculated as 1.40E-6/yr + 9.33E-9/yr = 1.41E-6/yr.

For 1 in 15 years, 9 months:

- From Table 4-4, the EPRI Class 3b frequency without considering corrosion is 2.21E-6/yr.
- The increase in the base case Class 3b frequency due to the corrosion-induced concealed flaw issue is calculated as 1.83E-4 * 1.38E-4 = 2.53E-8/yr, where 1.38E-4 was previously shown above to be the cumulative likelihood of non-detected containment leakage due to corrosion at 16 years.
The 1-in-15.75 year Class 3b frequency including the corrosion-induced concealed flaw issue is then calculated as 2.21E-6/yr + 2.53E-8/yr = 2.24E-6/yr.

3.2 Assumptions

The following assumptions are used in the analysis:

- The Kewaunee Level 1 and Level 2 internal events PRA models provide representative results.

- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology [REPORT02] and are summarized in Table 3.1.2-4.
- The representative containment leakage for Class 1 sequences is 1La. Class 3 accounts for increased leakage due to Type A inspection failures.
- The representative containment leakage for Class 3a sequences is 10La. based on the previously approved methodology [LETTER06].
- The representative containment leakage for Class 3b sequences is 100La. based on the previously approved methodology [LETTER06].
- The Class 3b can be very conservatively categorized as LERF based on the previously approved methodology [LETTER06].
- The impact on population doses from containment bypass scenarios is not altered by the proposed Type A test interval extension but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the containment bypass contribution to population dose is fixed, no changes in the conclusions from this analysis will result from this separate categorization.
- Increasing the Type A test interval does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.
- The Kewaunee Seismic and Fire PRAs [NB03, NB04] are acceptable for this analysis. Both are very conservative. For example the seismic assessment assumes that every seismic containment failure is also a large early release and the fire assessment assumes offsite power is disabled in some situations in which it would not have to be. The calculated risk contribution would decrease if a more realistic assessment were used.

3.3 Methodology

A simplified bounding analysis approach consistent with the EPRI approach is used for evaluating the change in risk associated with a one-time increase of the test interval to 15 years, 9 months. The approach is presented in Appendix H of EPRI Report No. 1009325, Revision 2, *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals*, EPRI TR-104285, NUREG-1493, and the Calvert Cliffs liner corrosion analysis. The analysis uses results from a Level 2 analysis of core damage scenarios from the current Kewaunee PRA model and subsequent containment response, resulting in various fission product release categories (including no or negligible release). This risk assessment is applicable to Kewaunee.

The six general steps of this assessment are as follows:

1. Quantify the baseline risk in terms of the frequency of events (per reactor year) for each of the eight containment release scenario types identified in the EPRI report.

2. Develop plant-specific person-rem (population dose) per reactor year for each of the eight containment release scenario types from plant-specific consequence analyses.

3. Evaluate the risk impact (i.e., the change in containment release scenario type frequency and population dose) of extending the Type A test interval to 15 years, 9 months.

4. Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 [4] and compare with the acceptance guidelines of RG 1.174.

5. Determine the impact on the Conditional Containment Failure Probability (CCFP).

6. Evaluate the sensitivity of the results to assumptions in the liner corrosion analysis, external events, and to the fractional contribution of increased large isolation failures (due to liner breach) to LERF.

This approach is based on the information and approaches contained in the previously mentioned studies. Furthermore:

- Consistent with the other industry containment leak risk assessments, the Kewaunee assessment uses LERF and delta LERF in accordance with the risk acceptance guidance of RG 1.174. Changes in population dose and conditional containment failure probability are also considered to show that defense-in-depth and the balance of prevention and mitigation is preserved.

- This evaluation for Kewaunee uses the ground rules and methods to calculate changes in risk metrics used in Appendix H of EPRI Report No. 1009325, Revision 2, *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals.*

4.0 RESULTS AND CONCLUSIONS

The application of the approach based on the guidance contained in EPRI Report 1009325, Revision 2, Appendix H; EPRI TR-104285; and previous risk assessment submittals on this subject have led to the results discussed in this section. The results are displayed according to the eight accident classes defined in the EPRI report. Table 3.1.2-4 lists these accident classes, with subclasses defined for the Kewaunee analysis.

The analysis performed examined Kewaunee-specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the breakdown of the severe accidents contributing to risk were considered in the following manner:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR-104285 Class 1 sequences).

- Core damage sequences in which containment integrity is impaired due to random containment isolation failures of plant components other than those associated with Type B or Type C test components. For example, containment liner breach or bellows leakage (EPRI TR-104285 Class 3 sequences).

- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left open following a plant post-maintenance test. (For example, a valve failing to close following a valve stroke test.) (EPRI TR-104285 Class 6 sequences.) Consistent with the NEI guidance [REPORT01], this class is not specifically examined since it will not significantly influence the results of this analysis.

- Accident sequences involving containment bypassed (EPRI TR-104285 Class 8 sequences), large containment isolation failures (EPRI TR-104285 Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI TR-104285 Class 4 and 5 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency change.

- Class 4 and 5 sequences are impacted by changes in Type B and C test intervals; therefore, changes in the Type A test interval do not impact these sequences.

The steps taken to perform this risk assessment evaluation are as follows:

Step 1 - Quantify the baseline risk in terms of frequency per reactor year for each of the eight accident classes presented in Table 3.1.2-4.

Step 2 - Develop plant-specific person-rem dose (population dose) per reactor year for each of the eight accident classes.

Step 3 - Evaluate risk impact, in terms of population dose, of extending Type A test interval from three to 15 and 10 to 15 years, 9 months.

Step 4 - Determine the change in risk in terms of large early release frequency (LERF) in accordance with RG 1.174.

Step 5 - Determine the impact on the conditional containment failure probability (CCFP).

4.1 STEP 1 - QUANTIFY THE BASELINE RISK IN TERMS OF FREQUENCY PER REACTOR YEAR

As previously described, extension of the Type A interval does not influence those accident progressions that involve large containment isolation failures, Type B or Type C testing, or containment failures induced by severe accident phenomena.

For the assessment of Type A test impacts on the risk profile, the potential for pre-existing containment leaks is included in the model. (These events are represented by the Class 3 sequences in EPRI TR-104285.) The node on containment integrity was modified to include the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two failure modes were considered for the Class 3 sequences. These are Class 3a (small breach) and Class 3b (large breach).

The frequencies of the accident classes defined in Table 3.1.2-4 were developed for Kewaunee by first determining the frequencies for classes 1, 2, 7, and 8, determining the frequencies for classes 3a and 3b, and determining the remaining frequency for class 1. Furthermore, adjustments were made to the class 3b and hence class 1 frequencies to account for the impact of undetected corrosion of the steel liner per the methodology described in Section 3.1.4.

The total CDF is 1.94E-4, which is the sum of the frequencies from Table 3.1.2-5. The results are summarized in Table 4-1.

Class 1 Sequences. This group consists of all core damage accident progression bins for which the containment remains intact (modeled as technical specification leakage). The frequency per year is initially determined from the Level 2 release categories 1 and 8 listed in Table 4-1, minus the EPRI/NEI Class 3a and 3b frequency, calculated below.

Class 2 Sequences. This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. The frequency per year for these sequences is obtained from the release categories 6 and 7, listed in Table 4-1.

Class 3 Sequences. This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (for example, containment liner) exists. The containment leakage for these sequences can be either small ($2L_a$ to $35L_a$) or large ($>35L_a$).

The respective frequencies per year are as follows:

$PROB_{class\_3a}$ = probability of small pre-existing containment liner leakage
= 0.0092 (see Section 3.1.3)
$PROB_{class\_3b}$ = probability of large pre-existing containment liner leakage
= 0.0023 (see Section 3.1.3)

As described in section 3.1.3, these failure probabilities are not applied to those cases that are already LERF scenarios (i.e., the Class 2 and Class 8 contributions).

CLASS_3a_FREQUENCY = 0.0092 * (CDF-Class 2-Class 8)
= 0.0092*(1.94E-4 - 6.08E-7 - 4.70E-8 - 5.42E-6 – 5.01E-6)= 1.68E-6/yr

CLASS_3b_FREQUENCY = 0.0023 * (CDF-Class 2-Class 8)
=0.0023 * (1.94E-4 - 6.08E-7 - 4.70E-8 - 5.42E-6 – 5.01E-6) = 4.21E-7/yr

Class 4 Sequences. This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type B test components occurs. Because these failures are detected by Type B tests, which are unaffected by increasing the frequency of Type A tests, this group is not evaluated any further in the analysis.

Class 5 Sequences. This group consists of all core damage accident progression bins for which a containment isolation failure-to-seal of Type C test components occurs. Because the failures are detected by Type C tests, which are unaffected by increasing the frequency of Type A tests, this group is not evaluated any further in this analysis.

Class 6 Sequences. This group is similar to Class 2. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution. In accordance with the NEI Interim Guidance [LETTER03], this accident class is not explicitly considered since it has a negligible impact on the results.

Class 7 Sequences This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs (for example, overpressure). For this analysis, the frequency is determined from release categories 4 and 5 from the Kewaunee Level 2 results.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass occurs. For this analysis, the frequency is determined from release categories 11, 12, 13 and 14 from the Kewaunee Level 2 results.

Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to radionuclide release to the public have been derived according to the definitions of accident classes defined in EPRI-TR-104285 and the NEI Interim Guidance [LETTER03]. Table 4-1 summarizes these accident frequencies by accident class for Kewaunee.

Table 4-1
RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS (KEWAUNEE BASE CASE)

Accident Classes (Containment Release Category)	Description	Frequency (per Rx-yr)	
		NEI Methodology	NEI Methodology Plus Corrosion
1	No Containment Failure	1.43E-5	1.43E-5
2a	Large Isolation Failures (Failure to Close >5" Penetrations)	6.08E-7	6.08E-7
2b	Large Isolation Failures (Failure to Close 2" – 5" Penetrations)	4.70E-8	4.70E-8
3a	Small Isolation Failures (liner breach)	1.68E-6	1.68E-6
3b	Large Isolation Failures (liner breach)	4.21E-7	4.23E-7
4	Small Isolation Failures (Failure to seal — Type B)	0	0
5	Small Isolation Failures (Failure to seal — Type C)	0	0
6	Other Isolation Failures (for example, dependent failures)	0	0
7a	Failures Induced by Phenomena (Pressure Failure)	2.76E-5	2.76E-5
7b	Failures Induced by Phenomena (Basemat Meltthrough)	1.41E-4	1.41E-4
8a	Bypass (Interfacing System LOCA)	5.42E-6	5.42E-6
8b	Bypass (Steam Generator Tube Rupture)	5.01E-6	5.01E-6
CDF	All CET end states	1.94E-4	1.94E-4

4.2 STEP 2 - DEVELOP PLANT-SPECIFIC PERSON-REM DOSE (POPULATION DOSE) PER REACTOR YEAR

Plant-specific release analyses were performed to estimate the person-rem doses to the population within a 50-mile radius from the plant. The releases are based on plant-specific analysis performed for the Kewaunee Type A test interval extension to 15 years [LETTER04]. No plant changes have been made since this analysis which would cause it to be invalidated. The results of applying these releases to the EPRI/NEI containment failure classification are as follows:

Class 1 = 1.20E+2 person-rem (at 1.0La)
Class 2a = 2.01E+5 person-rem
Class 2b = 2.97E+5 person-rem
Class 3a = 1.20E+2 person-rem x 10 = 1.2E+3 person-rem
Class 3b = 1.20E+2 person-rem x 100 = 1.2E+4 person-rem
Class 4 = Not analyzed
Class 5 = Not analyzed
Class 6 = Not analyzed
Class 7a = 4.04E+5 person-rem

Class 7b = 7.51E+1 person-rem
Class 8a = 1.17E+6 person-rem
Class 8b = 6.35E+5 person-rem

The above dose estimates, when combined with the release frequencies presented in Table 4-1, yield the Kewaunee baseline mean consequence measures (i.e., population dose and LERF) for each accident class. These results are presented in Table 4-2.

Table 4-2
KEWAUNEE ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 3/10 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change in Dose Due to Corrosion Person-Rem/yr
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure	1.20E+2	1.22E-5	1.46E-3	1.22E-5	1.46E-3	0
2a	Large Isolation Failures (Failure to Close >5" Penetrations)	2.01E+5	6.08E-7	1.22E-1	6.08E-7	1.22E-1	0
2b	Large Isolation Failures (Failure to Close 2" – 5" Penetrations)	2.97E+5	4.70E-8	1.40E-2	4.70E-8	1.40E-2	0
3a	Small Isolation Failures (liner breach)	1.20E+3	1.68E-6	2.02E-3	1.68E-6	2.02E-3	0
3b	Large Isolation Failures (liner breach)	1.20E+4	4.21E-7	5.05E-3	4.23E-7	5.08E-3	3.00E-5
4	Small Isolation Failures (Failure to seal — Type B)	N/A	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0
5	Small Isolation Failures (Failure to seal — Type C)	N/A	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0
6	Other Isolation Failures (for example, dependent failures)	N/A	0.00E+0	0.00E+0	0.00E+0	0.00E+0	0
7a	Failures Induced by Phenomena (Pressure Failure)	4.04E+5	2.76E-5	1.12E+1	2.76E-5	1.12E+1	0
7b	Failures Induced by Phenomena (Basemat Meltthrough)	7.51E+1	1.41E-4	1.06E-2	1.41E-4	1.06E-2	0
8a	Bypass (Interfacing System LOCA)	1.17E+6	5.42E-6	6.34E+0	5.42E-6	6.34E+0	0
8b	Bypass (Steam Generator Tube Rupture)	6.35E+5	5.01E-6	3.18E+0	5.01E-6	3.18E+0	0
CDF	All CET end states		1.94E-4	2.09E+1	1.94E-4	2.09E+1	3.00E-5

4.3 STEP 3 - EVALUATE RISK IMPACT OF EXTENDING TYPE A TEST INTERVAL FROM 10 TO 15 YEARS 9 MONTHS

The next step is to evaluate the risk impact of extending the test interval from the nominal 10-year value to 15 years, 9 months. To do this, an evaluation must first be made of the risk associated with the 10-year interval since the base case applies to a 3-year interval (i.e., a simplified representation of a three-in-10 interval).

Risk Impact Due to 10-Year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in the Type A test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3a and 3b sequences is impacted. The risk contribution is changed based on the NEI guidance as described in Section 3.1.3 by a factor of 3.33 compared to the base case values. The results of the calculation for a 10-year interval are presented in Table 4-3.

Risk Impact Due to 15-Year 9-Month Test Interval

The risk contribution for a 15-year 9-Month interval is calculated in a manner similar to the 10-year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the value used in the analysis is a factor of 5.25, compared to the three-year interval value, as described in Section 3.1.3. The results for this calculation are presented in Table 4-4.

Table 4-3
KEWAUNEE ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/10 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure	1.20E+2	7.31E-6	8.77E-4	7.26E-6	8.71E-4	-1.00E-6
2a	Large Isolation Failures (Failure to Close >5" Penetrations)	2.01E+5	6.08E-7	1.22E-1	6.08E-7	1.22E-1	0
2b	Large Isolation Failures (Failure to Close 2" – 5" Penetrations)	2.97E+5	4.70E-8	1.40E-2	4.70E-8	1.40E-2	0
3a	Small Isolation Failures (liner breach)	1.20E+3	5.59E-6	6.71E-3	5.59E-6	6.71E-3	0
3b	Large Isolation Failures (liner breach)	1.20E+4	1.40E-6	1.68E-2	1.41E-6	1.69E-2	1.00E-4
4	Small Isolation Failures (Failure to seal — Type B)	N/A	0	0	0	0	0
5	Small Isolation Failures (Failure to seal — Type C)	N/A	0	0	0	0	0
6	Other Isolation Failures (for example, dependent failures)	N/A	0	0	0	0	0
7a	Failures Induced by Phenomena (Pressure Failure)	4.04E+5	2.76E-5	1.12E+1	2.76E-5	1.12E+1	0
7b	Failures Induced by Phenomena (Basemat Meltthrough)	7.51E+1	1.41E-4	1.06E-2	1.41E-4	1.06E-2	0
8a	Bypass (Interfacing System LOCA)	1.17E+6	5.42E-6	6.34E+0	5.42E-6	6.34E+0	0
8b	Bypass (Steam Generator Tube Rupture)	6.35E+5	5.01E-6	3.18E+0	5.01E-6	3.18E+0	0
CDF	All CET end states		1.94E-4	2.09E+1	1.94E-4	2.09E+1	9.9E-5

Table 4-4
KEWAUNEE ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/15.75 YEARS

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)	NEI Methodology		NEI Methodology Plus Corrosion		Change Due to Corrosion Person-Rem/yr
			Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	Frequency (per Rx-yr)	Person-Rem/yr (50 miles)	
1	No Containment Failure	1.20E+2	3.27E-6	3.92E-4	3.19E-6	3.83E-4	-9.00E-6
2a	Large Isolation Failures (Failure to Close >5" Penetrations)	2.01E+5	6.08E-7	1.22E-1	6.08E-7	1.22E-1	0
2b	Large Isolation Failures (Failure to Close 2" – 5" Penetrations)	2.97E+5	4.70E-8	1.40E-2	4.70E-8	1.40E-2	0
3a	Small Isolation Failures (liner breach)	1.20E+3	8.82E-6	1.06E-2	8.82E-6	1.06E-2	0
3b	Large Isolation Failures (liner breach)	1.20E+4	2.21E-6	2.65E-2	2.24E-6	2.69E-2	4.00E-4
4	Small Isolation Failures (Failure to seal — Type B)	N/A	0	0	0	0	0
5	Small Isolation Failures (Failure to seal — Type C)	N/A	0	0	0	0	0
6	Other Isolation Failures (for example, dependent failures)	N/A	0	0	0	0	0
7a	Failures Induced by Phenomena (Pressure Failure)	4.04E+5	2.76E-5	1.12E+1	2.76E-5	1.12E+1	0
7b	Failures Induced by Phenomena (Basemat Meltthrough)	7.51E+1	1.41E-4	1.06E-2	1.41E-4	1.06E-2	0
8a	Bypass (Interfacing System LOCA)	1.17E+6	5.42E-6	6.34E+0	5.42E-6	6.34E+0	0
8b	Bypass (Steam Generator Tube Rupture)	6.35E+5	5.01E-6	3.18E+0	5.01E-6	3.18E+0	0
CDF	All CET end states		1.94E-4	2.09E+1	1.94E-4	2.09E+1	3.91E-4

4.4 STEP 4 - DETERMINE THE CHANGE IN RISK IN TERMS OF LARGE EARLY RELEASE FREQUENCY (LERF)

The risk increase associated with extending the Type A test interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could in fact result in a larger release due to the increase in probability of failure to detect a pre-existing leak. With strict adherence to the NEI guidance, 100% of the Class 3b frequency would be considered a large early release.

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines "very small" changes in risk as changes resulting in increases of core damage frequency (CDF) below 1E-6/yr and increases in LERF below 1E-7/yr, and "small" changes as changes resulting in increases of CDF below 1E-5/yr and increases in LERF below 1E-6/yr. Because the Type A test interval does not impact CDF, the relevant metric is LERF.

For Kewaunee, 100% of the frequency of Class 3b sequences can be used as a very conservative first-order estimate to approximate the potential increase in LERF from the Type A test interval extension (consistent with the NEI guidance methodology). For the 10-year test interval (see Table 4-3), the Class 3b frequency is 1.41E-6/yr; and for the 15-year 9-month test interval (see Table 4-4), the Class 3b frequency is 2.24E-6. Thus, the increase in the frequency of Class 3b sequences resulting from increasing the Type A test interval from 3 to 10 years is 9.9E-7/yr. Similarly, the increase in LERF resulting from increasing the Type A test interval from 10 to 15 years 9 months is 8.3E-7/yr. As can be seen, even with the conservatisms included in the evaluation (per the NEI methodology), the estimated change in LERF for Kewaunee is below the Regulatory Guide 1.174 criterion for a "small" change when comparing the 15-year 9-month results to the nominal 10-year requirement.

Regulatory Guide 1.174 states that for plants with an overall LERF of greater than 1E-5/yr, a "very small" LERF increase (< 1E-7) is acceptable and for plants with an overall LERF of less than 1E-5/yr, a "small" LERF increase (< 1E-6) is acceptable. Kewaunee's internal events LERF, from Table 3.1.2-1 is 4.9E-6/yr. The total external events LERF, from Table 3.1.2-1 is 5.1E-6/yr, which is dominated by the seismic LERF. The external events models are not as fully developed and are more conservative than the internal events models. Therefore, it is not appropriate to add them directly. Additionally, there is inherent conservatism in the seismic assessment. One example of this is use of IPEEE vintage hazard curves and spectra. Seismic analysis performed for the 2005 Kewaunee flooding significance determination [LETTER08] showed significantly lower risk when more recent hazard curves and spectra were used. Another conservatism is that every containment failure is considered a large early release. Therefore, the total LERF is less than 1E-5/yr, so an increase in LERF of less than 1E-6 is acceptable.

4.5 STEP 5 – DETERMINE THE IMPACT ON THE CONDITIONAL CONTAINMENT FAILURE PROBABILITY (CCFP)

Another parameter that provides input into the decision making process is the change in the conditional containment failure probability (CCFP). The CCFP is the probability, given core damage, that containment failure occurs. The change in CCFP is indicative of the effect of the Type A test interval on all radionuclide releases, not just large early releases. The CCFP can be calculated from the results of this analysis. One of the difficult aspects of this calculation is providing a definition of *failed containment*. In this assessment, the CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state.
The change in CCFP can be calculated by using the method specified in the NEI interim guidance [LETTER04]. The NRC has accepted a threshold of less than 1.5% as an acceptance criterion for CCFP. [LETTER06]

$$CCFP = [1 - (\text{Class 1 frequency} + \text{Class 3a frequency}) / CDF] * 100\%$$

$$CCFP_3 = 92.8\%$$
$$CCFP_{10} = 93.4\%$$

$$CCFP_{15.75} = 93.8\%$$

$$\Delta\ CCFP = CCFP_{10} - CCFP_3 = 0.6\%$$

$$\Delta\ CCFP = CCFP_{15.75} - CCFP_{10} = 0.4\%$$

The change in CCFP of 0.4% due to extending the Type A test interval to 15 years 9 months from the nominal 10-year requirement meets the NRC acceptance criterion of less than 1.5%.

4.6 SUMMARY OF RESULTS

The results from this Type A test interval extension risk assessment for Kewaunee are summarized in Table 4-5.

Table 4-5
ILRT Risk Assessment Results

EPRI Class	DOSE Per-Rem	Base Case 3 in 10 Years		Extend to 1 in 10 Years		Extend to 1 in 15.75 Years	
		CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr	CDF/Yr	Per-Rem/Yr
1	1.20E+2	1.22E-5	1.46E-3	7.26E-6	8.71E-4	3.19E-6	3.83E-4
2a	2.01E+5	6.08E-7	1.22E-1	6.08E-7	1.22E-1	6.08E-7	1.22E-1
2b	2.97E+5	4.70E-8	1.40E-2	4.70E-8	1.40E-2	4.70E-8	1.40E-2
3a	1.20E+3	1.69E-6	2.03E-3	5.63E-6	6.76E-3	8.87E-6	1.06E-2
3b	1.20E+4	4.23E-7	5.08E-3	1.41E-6	1.69E-2	2.24E-6	2.69E-2
7a	4.04E+5	2.76E-5	1.12E+1	2.76E-5	1.12E+1	2.76E-5	1.12E+1
7b	7.51E+1	1.41E-4	1.06E-2	1.41E-4	1.06E-2	1.41E-4	1.06E-2
8a	1.17E+6	5.42E-6	6.34E+0	5.42E-6	6.34E+0	5.42E-6	6.34E+0
8b	6.35E+5	5.01E-6	3.18E+0	5.01E-6	3.18E+0	5.01E-6	3.18E+0
Total		1.94E-4	2.09E+1	1.94E-4	2.09E+1	1.94E-4	2.09E+1
ILRT Dose Rate from 3a and 3b		7.11E-3		2.37E-2		3.75E-2	
Delta Total Dose Rate	From 3 yr	---------------------------		1.66E-2		3.04E-2	
	From 10 yr	---------------------------		---------------------------		1.38E-2	
3b Frequency (LERF)		4.23E-7		1.41E-6		2.24E-6	
Delta LERF	From 3 yr	---------------------------		9.87E-7		1.82E-6	
	From 10 yr	---------------------------		---------------------------		8.30E-7	
CCFP %		92.8		93.4		93.8	
Delta CCFP %	From 3 yr	--------------------------		0.6		1.0	
	From 10 yr	--------------------------		--------------------------		0.4	

4.7 CONCLUSIONS

Based on the above results and the sensitivity calculations presented in Section 5, the following conclusions can be made:

- Reg. Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines "small" changes in risk as those changes resulting in increases of CDF below 1E-5/yr and increases in LERF below 1E-6/yr. Since the Type A test interval does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A test interval from one in 10 years to one in 15 years 9 months is very conservatively calculated as 8.3E-7/yr using the NEI guidance as written. As such, the change in LERF is acceptable using the acceptance guidelines of Reg. Guide 1.174.

- The change in Type A test frequency from one in 10 years to once per 15 years 9 months, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 1.4E-2 person-rem/yr, which is 0.067% of the total dose. This meets the acceptance guidelines stated in section 2.1.

- The increase in the conditional containment failure frequency from the one-in-10-years interval to a one-in-15.75-years interval is 0.4%. This is below the acceptance the CCFP acceptance guidelines stated in section 2.1..

Therefore, increasing the Type A test interval to 15 years, 9 months is acceptable since it results in a very small change to the Kewaunee risk profile.

Previous Assessments

The NRC in NUREG-1493 has previously concluded that:

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

- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between Type A tests is possible with minimal impact on public risk. The impact of increasing the Type A test frequency beyond one in 20 years has not been evaluated.

The results of this assessment confirm these general findings on a plant-specific basis by considering plant specific core damage frequencies, containment failure modes, and population characteristics.

5.0 SENSITIVITIES

5.1 Sensitivity to Corrosion Impact Assumptions

The results in Tables 4-2, 4-3, and 4-4 show that including corrosion effects as described in Section 3.1.4 does not significantly affect the results of the Type A test extension risk assessment.

Sensitivity cases were developed to gain an understanding of the sensitivity of the results to the key parameters in the corrosion risk analysis. The time for the flaw likelihood to double was adjusted from every five years to every two and every 10 years. The failure probabilities for the containment cylinder and dome and the basemat were increased and decreased by an order of magnitude. The total detection failure likelihood was adjusted from 10% to 15% and 5%. Details of this calculation are in Attachment C. The results are presented in Table 5-1. In every case, the impact from including the corrosion effects is minimal. Even the upper bound estimates with conservative assumptions for all of the key parameters yield increases in LERF due to corrosion of only 7.9E-7/yr. The results indicate that even with conservative assumptions, the conclusions from the base analysis would not change.

Table 5-1
Steel Liner Corrosion Sensitivity Cases

Age (Step 3 in the corrosion analysis)	Containment Breach (Step 4 in the corrosion analysis)	Visual Inspection and Non-Visual Flaws (Step 5 in the corrosion analysis)	Increase in Class 3b Frequency (LERF) for ILRT Extension 10 to 15 years 9 months (per Rx-yr)	
			Total Increase	**Increase Due to Corrosion**
Base case Doubles every 5 yrs	Base Case (1% Cylinder, 0.1% Basemat)	Base Case 10%	8.3E-7	1.6E-8
Doubles every 2 yrs	Base	Base	8.7E-7	5.7E-8
Doubles every 10 yrs	Base	Base	8.2E-7	1.1E-8
Base	Base	*15%*	8.3E-7	2.2E-8
Base	Base	*5%*	8.2E-7	9.7E-9
Base	10% Cylinder, 1% Basemat	Base	9.7E-7	1.6E-7
Base	0.1% Cylinder, 0.01% Basemat	Base	8.1E-7	1.6E-9
Lower Bound				
Doubles every 10 yrs	0.1% Cylinder, 0.01% Basemat	5% 1%	8.1E-7	6.4E-10
Upper Bound				
Doubles every 2 yrs	10% Cylinder, 1% Basemat	15% 100%	1.6E-6	7.9E-7

5.2 Sensitivity to Class 3b Contribution to LERF

All calculations in this document assume that 100% of the Class 3b sequences result in a large early release, which is conservative.

6.0 REFERENCES

[GUIDE01] *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,* Regulatory Guide 1.174, July 1998.

[LETTER01] *Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension,* Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to NRC Document Control Desk, Docket No. 50-317, March 27, 2002.

[LETTER02] *Kewaunee Nuclear Power Plant – Issuance of Amendment,* Letter from Mr. J. G. Lamb (NRC) to Mr. T. Coutu, (Nuclear Management Company), Docket No. 50-305, April 6, 2004.

[LETTER03] Anthony R. Pietrangelo, *One-Time Extensions of Containment Integrated Leak Rate Test Interval – Additional Information*, NEI letter to Administrative Points of Contact, November 30, 2001.

[LETTER04] *License Amendment Request 198, "ILRT 5-Year Extension," NMC Response to NRC Request for Additional Information"*, Letter from Mr. T. Coutu (Nuclear Management Company) to NRC Document Control Desk, Docket No. 50-305, December 12, 2003.

[LETTER05] United States Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No. 3 - Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB0178), April 17, 2001.

[LETTER06] *Final Safety Evaluation for NEI TR-94-01Revision 2 and EPRI TR-1009329 Revision 2,* Letter from Mr. M. J. Maxin (NRC) to Mr. J. C. Butler, NEI, June 25, 2008.

[LETTER07] *Kewaunee Power Station – Issuance of Amendment,* Letter from Mr. R. F. Kuntz (NRC) to Mr. D. A. Christian, (Dominion), Docket No. 50-305, March 8, 2007.

[LETTER08] *"Kewaunee Power Station Flooding Significance Determination Process Risk Assessment Update Report"*, Letter from Mr. M. G. Gaffney (Dominion) to NRC Document Control Desk, Docket No. 50-305, December 16, 2005.

[MODEL01] Kewaunee Power Station PRA Model K107Aa, July 15, 2008.

[NB01] Kewaunee Power Station Probabilistic Risk Assessment Model Notebook, Volume QU.2, Model Quantification Results, Revision 1, July 10, 2008.

[NB02] Kewaunee Power Station Probabilistic Risk Assessment Model Notebook, Volume LE.2, Model Quantification Results, Revision 1, May 31, 2007.

[NB03] Kewaunee Power Station Probabilistic Risk Assessment Model Notebook, Section 8, Seismic Analysis, Revision 0403A, October 26, 2004

[NB04] Kewaunee Power Station Probabilistic Risk Assessment Model Notebook, Section 9, Internal Fire Analysis, Revision 0403, September 23, 2004

[NB05] Kewaunee Power Station Probabilistic Risk Assessment Notebook, Volume KPS.RA.008, Determination of Source Term Category Frequencies in Support of Integrated Leak Rate Testing Interval Extension, August 13, 2008

[REPORT01] *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,* NEI 94-01, July 1995.

[REPORT02] *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals.* EPRI, Palo Alto, CA: 1994. TR-104285.

[REPORT03] *Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals*, Rev. 4, Developed for NEI by EPRI and Data Systems and Solutions, November 2001.

[REPORT04] *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, TR-1009325, Revision 2.* EPRI, Palo Alto, CA: November, 2007.

[REPORT05] *Performance-Based Containment Leak-Test Program,* NUREG-1493, September 1995.

[REPORT06] *Impact of Containment Building Leakage on LWR Accident Risk*, Oak Ridge National Laboratory, NUREG/CR-3539, ORNL/TM-8964, April 1984.

[REPORT07] *Reliability Analysis of Containment Isolation Systems*, Pacific Northwest Laboratory, NUREG/CR-4220, PNL-5432, June 1985.

[REPORT08] *Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3 'Containment Integrity Check,'* NUREG-1273, April 1988.

[REPORT09] *Review of Light Water Reactor Regulatory Requirements*, Pacific Northwest Laboratory, NUREG/CR-4330, PNL-5809, Vol. 2, June 1986

[REPORT10] *Shutdown Risk Impact Assessment for Extended Containment Leakage Testing Intervals Utilizing ORAM™.* EPRI, Palo Alto, CA: 1995. TR-105189.

[REPORT11] *Technical Specifications for Kewaunee Power Station* as revised through Amendment 197, Dominion Energy Kewaunee, May 7, 2008

ATTACHMENT A, JUSTIFICATION OF VOLUME CHANGE

Revision 0

This is the original revision.

Revision 1

This revision addresses several editorial comments made by licensing.

Revision 2

This revision replaces the post-change LERF with the pre-change LERF in the Regulatory Guide 1.174 LERF calculation.

Revision 3

This revision clarifies the acceptance guidelines in section 2.1.

ATTACHMENT B, REVIEWER COMMENTS/RESOLUTIONS

Comment Number	Section/Page	Review Comment	Response to Review Comment
1	Cover Sheet	Why 9 months?	This bounds the actual extension being asked for (April to October) and 9 months is called out in the SER on the methodology as an acceptable time to extend.
2	2.0	Add what the allowable leak rate is.	Done.
3	2.0	Need reference for need for 9-month extension.	Added reason for delay and clarity.
4	3.1.2	Is there a reference for Attachment C?	Attachment C has been converted to a standalone document KPS.RA-008
5	3.1.2	CDF and LERF are different.	Model used is indicated. This is different from the model used for the poster.
6	3.1.2	Does this include external?	No added some text to explain where external results came from.
7	3.1.2	Why didn't you use IE frequencies from LETTER04?	We would not be using the current mode, which is the one evaluated for the standard.
8	Table 3.1.2-1	Need reference for KPS release categories.	Done.
9	3.1.2	La decreased from what to what?	Added detail.
10	3.1.2	Mention that the MACCS runs were for Point Beach.	Done.
11	Table 3.1.2-3	Why are the release categories different from [LETTER04]	Explained.

Comment Number	Section/Page	Review Comment	Response to Review Comment
12	Table 3.1.2-3	Why is the order of 7a and 7b different.	An oversight. I did notice that 7a and 7b were previously defined. At this late date there is no need to change.
13	3.1.2	Delete words “rates” and “and LERF” in the paragraph below Table 3.1.2-3	Done.
14	3.1.2	Move Table 4-1 to above Table 3.1.2-4	Done.
15	Table 3.1.2-4	Add total frequency.	Done
16	3.1.3	Where does 217 come from.	As text states, there were 217 tests.
17	4.1	Reword to indicate where 1.94E-4 comes from.	Done.
18	4.1	Why is the total frequency different from that was in Attachment C.	Clarified that the attachment (now NB05]) is internal events only (see comment 6)
19	Table 4-7	Shouldn’t there be a column for 16 years?	Corrected typo. 15 should be 15.75.
20	3.1.3	Add reference for EPRI guidance in second paragraph.	Done
21	3.1.4	Below table 3.1.4-1, specify that the increase is applied to class 3b only.	Done
22	3.1.4	Typo in “3 in 10” section, 1.83E-6 should be 1.83E-4.	Corrected
23	3.2	Add Reference LETTER06	Added
24	3.2	The last sentence doesn’t make sense.	Corrected
25	4.1	The class 3a and 3b frequencies are calculated here. Show the others as well.	Section 4.1 is for the base case. The others are calculated in Section 4.3
26	Table 4-2	Numerous typos in this table.	Corrected
27	Table 4.3	Correct “Corrosion” number for 3a	Corrected

Comment Number	Section/Page	Review Comment	Response to Review Comment
27	Table 4-3 & others	Why include total LERF?	Deleted.
28	4.3	In the paragraph on the 10-year interval, there is a 15 that should be a 10.	Corrected
29	4.3	Show calculation of 3.33 and 5.25	The text points back to the calculation in 3.1.3.
30	4.4, 4.7, 5.2	The LERF calculation does not appear to include corrosion	Corrected.
31	4.7	In the 3rd paragraph add the interval that the change is from. (I in 10)	Corrected
32	4.7	Right before "previous Assessments" the new interval is 15 years, 9 months, not 15 years.	Corrected
33	5.1	Show sample calculation of corrosion sensitivity.	The calculation is detailed in Attachment C, but there was no reference. The reference has been added.
34	Title Page	Effective date should be August 15	Done
35	2.0	Mention that the most recent ILRT was performed in 1993.	Actually, it was 1994, but I added it.
36	Table 4-3	Why not subtract corrosion for class 3b from class 1 to preserve CDF.	I did. But since corrosion is only 2E-9 it truncated out for class 1.

ATTACHMENT C, CORROSION SENSITIVITY DATA

Section 5.1 of this analysis contains a series of sensitivity cases varying the parameters used in assessing the risk of containment failure due to corrosion. The formula for determining the LERF due to corrosion is as follows. The nominal values are from Table 3.1.4-1.

L(Corr) = [(PFC * PBC * PVC) + (PFB * PBB * PVB)] FCb, where:

L(Corr) = LERF due to corrosion

PFC = Probability of a flaw in the containment cylinder or dome (nominally 0.0071, 0.041, 0.11 at 3, 10 and 16 years, respectively)

PBC = Probability of a breach given a flaw in the containment cylinder or dome (nominally 0.01, low 0.001, high 0.1)

PVC = Probability of failure to identify visual flaws in the containment cylinder or dome (nominally 0.1, low 0.05, high 0.15)

PFB = Probability of a flaw in the containment basemat (nominally 0.0018, 0.01, 0.028 at 3, 10 and 16 years, respectively)

PBB = Probability of a breach given a flaw in the containment basemat (nominally 0.001, low 0.0001, high 0.01)

PVB = Probability of failure to identify visual flaws in the containment basemat (1 because the basemat is uninspectible)

F3b = Frequency of Class 3b sequences without corrosion = 4.21E-7/yr, 1.40E-6/yr and 2.21E-6/yr for 3, 10 and 16 years, respectively)

PBC, PBB – probability of a breach

The base case assumes a probability that continues to increase at a constant rate and doubles every five years. The failure rate for each year is calculated assuming an average of 5.2E-3 for 5 to 10 years for the cylinder 1.3E-3 for 5 to 10 years for the basemat, based on data. The low and high sensitivities assume a doubling every 10 years and 2 years, respectively. This means from 1 year to the next the failure rate increases by a factor of $2^{1/10}$ or 1.0718 for a 10-year doubling period and $2^{1/2}$ or 1.4142 for a 2-year doubling period. Using iteration to get an initial failure rate such that the average from 5 to 10 years remains at the nominal value, the values from Table C1 result: For each Type A test interval, the cumulative probability is used to determine a probability that a flaw would develop between Type A tests.

Table C1
FAILURE RATES GIVEN DOUBLING IN 10 OR 2 YEARS

Year	Cylinder/Dome				Basemat			
	10 Year Doubling		2 Year Doubling		10 Year Doubling		2 Year Doubling	
	Frequency	Cumulative	Frequency	Cumulative	Frequency	Cumulative	Frequency	Cumulative
0	3.07E-03		3.26E-04		7.68E-04		8.16E-05	
1	3.29E-03	3.29E-03	4.62E-04	4.62E-04	8.23E-04	8.23E-04	1.15E-04	1.15E-04
2	3.53E-03	6.82E-03	6.53E-04	1.11E-03	8.82E-04	1.70E-03	1.63E-04	2.79E-04
3	3.78E-03	1.06E-02	9.23E-04	2.04E-03	9.45E-04	2.65E-03	2.31E-04	5.09E-04
4	4.05E-03	1.47E-02	1.31E-03	3.34E-03	1.01E-03	3.66E-03	3.26E-04	8.36E-04
5	4.34E-03	1.90E-02	1.85E-03	5.19E-03	1.09E-03	4.75E-03	4.62E-04	1.30E-03
6	4.65E-03	2.36E-02	2.61E-03	7.80E-03	1.16E-03	5.91E-03	6.53E-04	1.95E-03
7	4.99E-03	2.86E-02	3.69E-03	1.15E-02	1.25E-03	7.16E-03	9.23E-04	2.87E-03
8	5.35E-03	3.40E-02	5.22E-03	1.67E-02	1.34E-03	8.49E-03	1.31E-03	4.18E-03
9	5.73E-03	3.97E-02	7.38E-03	2.41E-02	1.43E-03	9.93E-03	1.85E-03	6.02E-03
10	6.14E-03	4.59E-02	1.04E-02	3.45E-02	1.54E-03	1.15E-02	2.61E-03	8.64E-03
11	6.58E-03	5.24E-02	1.48E-02	4.93E-02	1.65E-03	1.31E-02	3.69E-03	1.23E-02
12	7.06E-03	5.95E-02	2.09E-02	7.02E-02	1.76E-03	1.49E-02	5.22E-03	1.75E-02
13	7.56E-03	6.71E-02	2.95E-02	9.97E-02	1.89E-03	1.68E-02	7.38E-03	2.49E-02
14	8.11E-03	7.52E-02	4.18E-02	1.42E-01	2.03E-03	1.88E-02	1.04E-02	3.54E-02
15	8.69E-03	8.39E-02	5.91E-02	2.01E-01	2.17E-03	2.03E-01	1.48E-02	5.01E-02
16	9.31E-03	9.32E-02	8.35E-02	2.84E-01	2.33E-03	2.33E-02	2.09E-02	7.10E-02