NMC
Committed to Nuclear Excellence

Kewaunee Nuclear Power Plant
Operated by Nuclear Management Company, LLC

NRC-03-065 10 CFR 50.90

June 20, 2003

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

KEWAUNEE NUCLEAR POWER PLANT
DOCKET 50-305
LICENSE No. DPR-43
LICENSE AMENDMENT REQUEST 198 TO THE KEWAUNEE NUCLEAR POWER PLANT TECHNICAL SPECIFICATIONS FOR ONE-TIME EXTENSION OF CONTAINMENT INTEGRATED LEAK RATE TEST INTERVAL

The Nuclear Management Company (NMC) in accordance with 10 CFR 50.90 is submitting this Licensing Amendment Request (LAR) to the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS) to revise the Surveillance Requirements for containment integrated leak rate testing in TS 4.4.a, Integrated Leak Rate Tests (Type A). This change will allow a one-time extension of the interval between integrated leakage rate tests (ILRTs) from 10 to 15 years. The proposed amendment would provide savings in radiation exposure to personnel, cost, and critical path time during the 2004 refueling outage.

The proposed amendment is risk-informed and follows the guidance in Regulatory Guide (RG) 1.174. NMC has performed an analysis showing that the increase in risk resulting from the proposed amendment is small and within established guidance. NMC has also determined that defense-in-depth principles will be maintained based on both risk and other considerations.

Attachment 1 to this letter contains a description, a safety evaluation, a significant hazards determination and environmental considerations for the proposed change. Attachment 2 contains the strikeout Technical Specification page: TS 4.4-1. Attachment 3 contains the affected Technical Specification page as revised: TS 4.4-1. Attachment 4 contains the calculation for the risk impact assessment for extending the containment Type A test interval.

There are no new commitments made by this submittal. This submittal contains no proprietary information.

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NMC requests approval of this submittal by December 19, 2003, to be implemented within 60 days. The short lead-time for approval is necessary to allow NMC adequate time for ILRT preparation work prior to the Fall 2004 refueling outage if the LAR is not approved.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on June 20, 2003.

for

Thomas Coutu
Site Vice-President, Kewaunee Plant

GHR

cc- US NRC, Region III
US NRC Senior Resident Inspector
Electric Division, PSCW

Attachments:

1. Description and Analysis of Proposed Change
2. Marked-up TS Page
3. Clean TS Page
4. Risk Impact Assessment

ATTACHMENT 1

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

June 20, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 198

Description of the Proposed Change

Safety Evaluation

Significant Hazards Determination

Environmental Considerations

1.0 INTRODUCTION

Pursuant to 10 CFR 50.90, Nuclear Management Company (NMC), the licensee for Kewaunee Nuclear Power Plant (KNPP), proposes to amend Appendix A, Technical Specifications (TS), of Facility Operating License DPR-43. NMC proposes to revise the Surveillance Requirements for containment integrated leak rate testing in TS 4.4.a to allow a one-time extension of the interval between reactor containment vessel integrated leakage rate tests (ILRTs) from 10 to 15 years. The proposed amendment would provide savings in radiation exposure to personnel, cost, and critical path time during the 2004 refueling outage.

The proposed amendment is risk-informed and follows the guidance in Regulatory Guide (RG) 1.174 (Reference 1). In accordance with RG 1.174, NMC has performed an analysis showing that the increases in risk resulting from the proposed amendment are small and within established guidance. NMC has also determined that defense-in-depth principles will be maintained based on both risk and other considerations.

2.0 DESCRIPTION OF PROPOSED CHANGE TO TECHNICAL SPECIFICATION (TS) 4.4.a, INTEGRATED LEAK RATE TESTS (TYPE A)

The proposed change would add a statement to TS 4.4.a stipulating that the ILRT (Type A test) frequency specified in the Nuclear Regulatory Commission (NRC)-endorsed industry guideline (NEI 94-01) 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" and will also state that the change applies only to the interval following the previous ILRT (April 1994). Attachment 2 provides the TS page marked to show the proposed change. Attachment 3 provides the TS page with the proposed change incorporated.

3.0 BACKGROUND

The KNPP 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, capable of withstanding a design internal pressure of 46 pounds per square inch gage and a temperature of 268 degrees Fahrenheit. The containment 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 KNPP Updated Safety Analysis Report (USAR).

Current Requirements

TS 4.4.a currently requires that integrated leak rate testing of the containment be performed in accordance with the Containment Leak Rate Testing Program (CLRTP). TS 6.20 requires the CLRTP to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, except as modified by NRC-approved exemptions, and in accordance with RG 1.163 (Reference 2). Regulatory Position C.I of RG 1.163 states that licensees should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01 (Reference 3). Section 11.0 of NEI 94-01 references Section 9.0 which allows ILRTs to be performed at a frequency of once per 10 years if the calculated leakage rate for two consecutive previous tests is less than 1.0 L_a. L_a is defined in KNPP TS 6.20 as 0.5 weight percent of the contained air per 24 hours at the peak test pressure, P_a, of 46.0 psig. The KNPP reactor containment vessel has met this criterion and therefore qualifies for the 10-year frequency.

Section 9.0 of NEI 94-01, however, also allows a 15-month extension of the ILRT test interval "...in cases where refueling schedules have been changed to accommodate other factors." NMC considers that the change to 18-month fuel cycles following the last ILRT performance satisfies this criteria and the 15-month extension allowance may be applied. Since an ILRT was last completed in April 1994, the current due date for the next ILRT is April 2004. Compliance with this due date would require that the ILRT be performed during the Spring 2003 refueling outage. Therefore, a portion of the 15-month extension is being used and the next ILRT is scheduled for the Fall 2004 refueling outage.

Basis for Current Requirements

The maximum allowable containment leakage rate, L_a, specified in TS 6.20, Containment Leakage Rate Testing Program, ensures that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure. As an added conservatism, TS 6.20 limits the measured overall integrated leakage rate to less than or equal to 0.75 L_a during performance of periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The performance-based ILRT requirements of Option B of 10 CFR 50, Appendix J, provide an alternative to the 3 tests per 10-year frequency specified by the prescriptive requirements of Option A of 10 CFR 50, Appendix J. As documented in RG 1.163, the NRC has endorsed NEI 94-01 as providing acceptable methods for complying with the requirements of Option B of 10 CFR 50, Appendix J. NEI 94-01 specifies an ILRT frequency of 1 test per 10 years if certain performance criteria are met. The basis for the 1 test per 10-year frequency is described in Section 11.0 of NEI 94-01, which states that NUREG-1493 (Reference 4) provides the technical basis to support rulemaking that established Option B. That basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. NEI undertook a similar study, the results of which are documented in Electric Power Research Institute (EPRI) report TR-104285 (Reference 5). The EPRI study determined a reduction in the frequency of ILRTs from 3 tests per 10 years to 1 test per 10 years would result in an incremental risk contribution of 0.035 percent. This value is comparable to the range of risk increases (0.002 percent to 0.14 percent) presented in NUREG-1493 for the same frequency reduction. Additionally, NUREG-1493 described the increase in risk resulting from an even lower frequency, 1 test per 20 years, as "imperceptible."

Reason for Requesting Amendment

Extension of the ILRT interval from 10 years to 15 years would eliminate the need to perform an ILRT for KNPP during the 2004 outage. This would save a total of approximately 0.5 person-rem exposure. This would also result in an estimated monetary savings of about $200,000, and save an estimated 30 hours of critical path time, at $15,000 per hour. The total monetary savings for KNPP would therefore be approximately $0.65 million. NMC is requesting this license amendment to obtain these personnel exposure and monetary savings.

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

4.0 TECHNICAL ANALYSIS

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

Risk Assessment

Methodology

The purpose of the risk assessment is to demonstrate that the requested extension of ILRT test interval has a negligible impact on risk at KNPP. To do this, the risk metrics evaluated were: population dose (person-rem per year), large early release frequency (LERF) and conditional containment failure probability (CCFP). The change in each of these risk metrics due to the requested increase in test interval from 10 to 15 years is determined along with the cumulative change from the original test interval (that corresponding to 3 tests in 10 years) to the requested 15 year test interval.

This analysis was performed in accordance with NEI 94-01 (Reference 3) guidelines, and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a licensee request for changes to a plant's licensing basis, Regulatory Guide RG 1.174 (Reference 1). This methodology is similar to that presented in NUREG-1493 (Reference 4) and EPRI TR-104285 (Reference 5) and incorporates the revised guidance and additional information of References 6 and 7.

The potential impact of age-related corrosion of the steel containment vessel on the risk associated with extending the ILRT interval has also been determined. The methodology used for this analysis is similar to the assessments performed for Calvert Cliffs Nuclear Power Plant (CCNPP) (Reference 8), however, the significantly lower potential for corrosion of the KNPP free-standing steel shell containment is considered. This is due to the significantly smaller surface area susceptible to corrosion resulting from foreign material imbedded in concrete contacting the steel containment.

The details of the analyses are provided in Attachment 4 to this submittal.

Input Information

The risk assessment utilizes the results of the latest update of the KNPP Level 1 and 2 PRA. KNPP maintains a living PRA. The PRA model used in this submittal reflects the as-built as-operated plant as of January 2003. There have been no changes to the plant since then that would affect the analysis in the submittal.

A peer review of the Kewaunee PRA was conducted June 10-14, 2002, using the Westinghouse Owners Group Peer Review process. The final Peer Review report was issued in December of 2002. Kewaunee received five category A (extremely important and necessary to address) Facts and Observations (F&Os). Three of these F&Os have been resolved in the update of the PRA used for this submittal. The resolution of the other two F&Os is underway. One involves the bases for time windows for human actions while the other involves internal flooding. A review of the details and potential impact of the resolution of these F&Os indicates that their resolution will not change the conclusions of the present analysis.

Results

The increase in ILRT test interval from 10 years to 15 years results in an increase in population dose of 0.00059 person-rem per year or 0.010% of the total population dose without considering corrosion and 0.00061 person-rem per year or 0.011% if corrosion is considered. The cumulative changes for the ILRT interval increase from that corresponding to 3 tests in 10 years to the requested 15 years are 0.00141 person-rem per year or 0.024%, without corrosion, and 0.00144 person-rem per year or 0.025%, with corrosion. These increases in risk are all very small and essentially negligible considering other risk contributions.

The overall baseline LERF for the Kewaunee Nuclear Power Plant is 1.69E-6 per year. The increase in ILRT test interval from 10 years to 15 years results in an increase in LERF (ΔLERF) of 3.4E-08 per year, without considering corrosion and 3.8E-08 per year, with corrosion. The cumulative changes for the ILRT interval increase from that corresponding to 3 tests in 10 years to the requested 15 years are 8.1E-08 per year, without corrosion, and 8.8E-08 per year, with corrosion. These increases in LERF are within the RG 1.174 definition of very small changes and are considered non-risk significant.

The increase in ILRT test interval from 10 years to 15 years results in an increase in conditional containment failure probability (CCFP) of 0.0012, without considering corrosion and 0.0014, with corrosion. The cumulative changes for the ILRT interval increase from that corresponding to 3 tests in 10 years to the requested 15 years are 0.0028, without corrosion, and 0.0031, with corrosion. These increases in CCFP are very small changes and are essentially negligible considering other risk contributions.
These results are summarized in the following table.

	Test Interval Extended	
	From 3 in 10 years to 1 in 15 years	From 1 in 10 years to 1 in 15 years
Total person-rem/year increase		
Without Corrosion	0.00141	0.00059
Including Corrosion	0.00144	0.00061
The percentage increase in person-rem/year risk		
Without Corrosion	0.024%	0.010%
Including Corrosion	0.025%	0.011%
Change in LERF (per year)		
Without Corrosion	8.1E-08	3.4E-08
Including Corrosion	8.8E-08	3.8E-08
Change in the Conditional Containment Failure Probability		
Without Corrosion	0.0028	0.0012
Including Corrosion	0.0031	0.0014

The above results demonstrate that the increases in risk and LERF resulting from the proposed amendment are within established guidelines and that defense-in-depth principles would be maintained.

Other Considerations

Consistent with the defense-in-depth philosophy provided in RG 1.174, NMC has assessed other considerations relevant to the proposed amendment. These are discussed below.

ILRT History

TS 4.4.a requires measurement of the containment leakage rate. TS 6.20 establishes the limit for the measured overall integrated containment leakage rate as 0.75 L_a (i.e., 0.375 weight percent) of the containment air per 24 hours at P_a. The results of all Type A tests for Kewaunee Nuclear Power Plant are reported below using the 95 percent upper confidence level estimate of leak rate.

KNPP ILRT Results Summary		
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&λ	0.1634	0.375
April, 1990&λ	0.0926	0.375
April, 1994&λ	0.0610	0.375

* Pre-operational test
& Test results obtained using BN-TOP-1
λ Includes LLRT additions for valves/penetrations not in their normal post-accident condition.

The results of all Type A tests performed at Kewaunee Nuclear Power Plant have been less than the acceptance criteria. Note that later results reflect the addition of calculation conservatism due to the 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 ILRT.

These results demonstrate a history of satisfactory performance for both leak tightness and structural integrity of the containment vessel.

Local Leakage Rate Testing

As documented in NUREG-1493, industry experience has shown that most ILRT failures result from leakage that is detectable by local leakage rate testing (Type B and C testing as defined in 10 CFR 50, Appendix J). The KNPP local leak rate testing requirements per the Containment Leak Rate Testing Program are unaffected by this proposed amendment. The local leak rate testing program will, therefore, provide continuing assurance that the most likely sources of leakage will be identified and repaired.

Containment Inservice Inspection Program

KNPP has established a containment inservice inspection program that implements the requirements for examination and testing of ASME Section XI and 10 CFR 50.55a Class MC components. This program was developed in accordance with the requirements of the 1992 Edition with the 1992 Addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Division 1, Subsections IWE and IWL, as modified by NRC final rulemaking of 10 CFR 50.55a, published in the Federal Register on August 8, 1996. The scope of the 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 first ten-year inspection interval has been established from September 9, 1996, to September 9, 2006. The containment inservice inspection program is unaffected by the proposed amendment, and will continue to provide a high degree of assurance that any degradation of the containment will be detected and corrected before it can result in a leakage path.

Approved Alternatives to Subsection IWE and IWL Requirements

There are no alternatives to Subsection IWE and IWL requirements approved for KNPP that credit the performance of integrated (Type A) or local (Type B and C) leak rate testing.

Containment Inspection History

ISI program examinations performed since the most recent ILRT (April 1994) include visual examinations of the containment vessel pursuant to Subsection IWE, which were conducted from 1998 through 2001. With the exception of the conditions described below, all results were within the established acceptance criteria.

In 1998, an inspection of the KNPP containment identified the following:

- Two (2) visual recordable indications on Reactor Building Containment Vessel Plate 98 and Plate 107 were recorded during performance of general visual examinations. The recordable visual indications were removed by grinding. Ultrasonic, magnetic particle and general visual examinations were performed following repair with no indications recorded.

- Two (2) visual recordable indications on Reactor Building Containment Vessel Plate 120 and Plate 155 were recorded during performance of general visual examinations. The recordable indications were apparent gouges in the base metal. Supplemental ultrasonic examination determined there was no violation of minimum wall and the gouges were accepted as is.

The 2000 inspection of the KNPP containment identified the following:

- Visual recordable indications on the Reactor Building Containment Vessel equipment door inner and outer gaskets were recorded during performance of VT -3 examinations. The recordable visual indications noted were damage to the gaskets and portions of the gaskets with tears. Both the inner and outer door gaskets were replaced. VT -3 Examinations were performed on the replacement equipment door inner and outer gaskets and found to be acceptable.

- Leak testing results obtained per Appendix J Type B test requirements on Penetration 41E (vacuum breaker) O-ring seals exceeded the administrative limits. The condition was repaired and subsequent reexamination measured acceptable leakage.

In 2001, an inspection of the KNPP containment identified the following:

- General visual indication consisting of a 4" x 8" surface defect was recorded on Plate 155. This surface defect was previously recorded and accepted in 1998 and showed no change in dimension or surface condition during the 2001 Refueling Outage.

- General visual indication consisting of a slight inward bulge on Plates 74, 75 and 83 was recorded. The slight inward bulge was evaluated by Engineering and accepted under Kewaunee Nuclear Power Plant Specification TS-1052, Addendum No.4, Item No. 19 Section 10.3 -Shell Tolerance.

- General visual indication consisting of slight outward bulges on Plates 144, 145, 146, 147 and 148 were recorded. The slight outward bulges were evaluated and accepted by engineering analysis.

- General visual indications consisting of weld deposits on Plate 62 and arc strikes on Plate 99 were recorded and are acceptable per ASME Boiler and Pressure Vessel Code Section XI 1992 Edition 1992 Addenda.

- VT-3 Indications consisting of lack of bonding and tears were recorded in the moisture barriers on Plates 62, 64, 65, 66 and 67 and were repaired.

- A VT-1 Indication on an Emergency Airlock bolt was recorded and was repaired.

Areas Requiring Augmented Examinations Per IWE-1240

As stated above, the ASME Section XI, Subsection IWE inspection plan was implemented for KNPP on September 9, 1996. All inspections have been completed through the second period, first outage, of the first 10-year surveillance interval. There are currently no identified areas at KNPP that require augmented inspection in accordance with IWE-1240.

Containment Penetration Bellows

In reviewing similar amendment requests from other licensees, the NRC has noted that stainless steel containment penetration bellows have been found to be susceptible to trans-granular stress corrosion cracking. As documented in NRC Information Notice 92-20 (Reference 9), leakage through such bellows may not be readily detectable by LLRTs. KNPP has nine penetration assemblies that incorporate two-ply mechanical bellows. These are the two main feedwater, two main steam, two steam generator blowdown, two residual heat removal, and one letdown penetrations. Review of plant drawings indicates that wire mesh is installed between the two-plies of each bellows assembly, ensuring that an adequate gap exists to measure leakage when performing the 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. These penetrations have been tested each outage per the KNPP Containment Leak Rate Testing Program with satisfactory results.

Maintenance Rule

The containment isolation function of limiting the release of radioactive fission products following an accident has been classified as high risk significant and its condition is monitored pursuant to 10 CPR 50.65 in accordance with the KNPP Maintenance Rule program. Operability of the containment isolation equipment is ensured by compliance with Technical Specifications, sections 1.0.g, 3.6, 3.8, 4.4, and 5.2. The proposed amendment affects only the ILRT requirements and has minimal impact.

5.0 REGULATORY SAFETY ANALYSIS

Significant Hazards Determination for Proposed Change to TS 4.4.a, Integrated Leak Rate Tests (Type A)

Nuclear Management Company (NMC), the licensee for Kewaunee Nuclear Power Plant (KNPP), proposes to amend Appendix A, Technical Specifications (TS), of Facility Operating License DPR-43. NMC proposes to revise the Surveillance Requirements for containment integrated leak rate testing in TS 4.4.a to allow a one-time extension of the interval between reactor containment vessel integrated leakage rate tests (ILRTs) from 10 to 15 years.

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

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

Response: No

Probability of Occurrence of an Accident Previously Evaluated –

The proposed change to extend the ILRT interval from 10 to 15 years does not affect any accident initiators or precursors. The containment vessel function is purely mitigative. There is no design basis accident that is initiated by a failure of the containment leakage mitigation function. The extension of the ILRT will not create any adverse interactions with other systems that could result in initiation of a design basis accident. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased.

Consequences of an Accident Previously Evaluated –

The potential consequences of the proposed change have been quantified by analyzing the changes in risk that would result from extending the ILRT interval from 10 to 15 years. The increase in risk in terms of person rem per year within 50 miles resulting from design basis accidents was estimated to be of a magnitude that NUREG-1493 indicates is imperceptible. NMC has also analyzed the increase in risk in terms of the frequency of large early releases from accidents. The increase in the large early release frequency resulting from the proposed extension was determined to be within the guidelines published in Regulatory Guide 1.174. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. NMC has determined that the increase in conditional containment failure probability from reducing the ILRT frequency from 1 test per 10 years to 1 test per 15 years would be small. Continued containment integrity is also assured by the history of successful ILRTs, and that established programs for local leakage rate testing and in-service inspections which are unaffected by the proposed change. Therefore, the consequences of an accident previously analyzed are not significantly increased.

In summary, the probability of occurrence and the consequences of an accident previously evaluated are not significantly increased.

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

   Response: No

   The proposed change to extend the ILRT interval from 10 to 15 years does not create any new or different accident initiators or precursors. The length of the ILRT interval does not affect the manner in which any accident begins. The proposed change does not create any new failure modes for the containment and does not affect the interaction between the containment and any other system. Thus, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

   Response: No

   The risk-based margins of safety associated with the containment ILRT are those associated with the estimated person-rem per year, the large early release frequency, and the conditional containment failure probability. NMC has quantified the potential effect of the proposed change on these parameters and determined that the effect is not significant. The non-risk-based margins of safety associated with the containment ILRT are those involved with its structural integrity and leak tightness. The proposed change to extend the ILRT interval from 10 to 15 years does not adversely affect either of these attributes. The proposed change only affects the frequency at which these attributes are verified. Therefore, the proposed change does not involve a significant reduction in margin of safety.

In summary, based upon the above evaluation, NMC has concluded that the proposed change involves 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.

Applicable Regulatory Requirements/Criteria

Regulations

TS 4.4.a currently requires that leakage rate testing of the containment be performed in accordance with the Containment Leak Rate Testing Program. TS 6.20 requires the CLRTP to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by NRC-approved exemptions, and in accordance with RG 1.163. Regulatory Position C.1 of RG 1.163 states that licensees should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01. Section 11.0 of NEI 94-01 references Section 9.0, which, as described above under "Background," would require that ILRTs be performed for KNPP within 10 years plus 15 months from the date of their last performance.

NMC is proposing a license amendment that would modify TS 4.4.a to allow a one-time extension of this interval to 15 years. The technical analysis for the proposed license amendment is based on risk related and non-risk related considerations. A risk analysis was performed for NMC showing that the increases in estimated person-rem and LERF are consistent with guidance provided in RG 1.174 and NUREG-1493. NMC has also demonstrated that defense-in-depth would be provided by the low increase in the conditional containment failure probability, and by non-risk based considerations such as the ILRT and containment inspection history, and the ongoing LLRT and ISI programs. The technical analysis provides the basis for NMC's determination that the proposed amendment does not involve significant hazards considerations as described in 10 CFR 50.92.

No other regulations or TS will be affected by the proposed amendment.

USAR

USAR Section 5.0, "Containment System," provides licensing basis information for the KNPP reactor containment vessel. Subsection 5.7.1 describes pre-operational and subsequent leakage rate testing of the containment. This subsection states that subsequent integrated (Type A) and local (Type B and C) leakage rate tests are in accordance with the requirements of 10 CFR 50, Appendix J and detailed in Technical Specifications, Section 4.4. The validity of this statement is unaffected by the proposed amendment since the proposed extension will only apply to the current ILRT interval and will not alter the accuracy of the statements as descriptions of normal requirements. Additionally, the proposed amendment does not affect any other aspect of the ILRT, such as test methodology, pressure, or acceptance criteria.

USAR Section 14, "Safety Analysis," provides descriptions of the licensing basis accident analyses for KNPP including the relevant parameters for the analyses. The proposed amendment only involves the ILRT interval and does not affect any parameters, such as pressure or leakage rate, that can affect the results of these analyses.

In conclusion, based on the considerations discussed above, (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 detrimental to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATIONS

NMC has evaluated this license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. NMC 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 needs to be prepared concerning the proposed amendment.

7.0 PRECEDENT LICENSING ACTIONS

The NRC has approved one-time extensions of the ILRT interval to 15 years based on risk and non-risk based considerations for Waterford Steam Electric Station, Unit 3 (Reference 10), Peach Bottom Atomic Power Station, Unit 3 (Reference 11), Crystal River Nuclear Plant, Unit 3 (Reference 12), Indian Point 3 Nuclear Power Plant (Reference 13), and D.C. Cook Nuclear Plant, Units 1 and 2 (Reference 14).

8.0 References

1. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," dated July 1998.
2. Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995.
3. Nuclear Energy Institute document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995.
4. NUREG-1493, "Performance-Based Containment Leak-Test Program," dated September 1995.
5. Electric Power Research Institute report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," dated August 1994.
6. EPRI document "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals," Rev. 4, dated November 2001.
7. NEI memo, "One-Time Extension of Containment Integrated Leak Rate Test Interval - Additional Information," dated November 30, 2001.
8. "Calvert Cliffs Nuclear Power Plant Unit No. 1; Docket No. 50-317, Response to Request for Additional Information Concerning the License Amendment Request for a One-time Integrated Leakage Rate Test Extension," Constellation Nuclear letter to USNRC, March 27, 2002. (ML020920100)
9. NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing," dated March 3, 1992.
10. Letter from N. Kalyanam, NRC, to J. E. Venable, Entergy Operations Incorporated, "Waterford Steam Electric Station, Unit 3, - Issuance of Amendment Re: Integrated Leakage Rate Testing Interval Extension (TAC No. MB2461)," dated February 14, 2002. (ML020460272)
11. Letter from J. P. Boska, NRC, to O .D. Kingsley, Exelon Generation Company, "Peach Bottom Atomic Power Station, Unit 3 - Issuance of Amendment Re: Extension of the Containment Integrated Leak Rate Testing Program (TAC No. MB2094)," dated October 4, 2001. (ML012210108)
12. Letter from J. M. Goshen, NRC, to D. E. Young, Florida Power Corporation, "Crystal River Unit 3 - Issuance of Amendment regarding Containment Leakage Rate Testing Program (TAC No. MB 1349)," dated August 30, 2001. (ML012190219)
13. Letter from G. F. Wunder, NRC, to M. Kansler, Entergy Nuclear Operations Incorporated, "Indian Point Nuclear Generating Unit 3 - Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB 0178)," dated April 17, 2001. (ML011021315)
14. Letter from J. F. Stang, NRC, to A. C. Bakken III, Indiana Michigan Power Company, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Issuance of Amendments (TAC Nos. MB4837 and MB4838)," dated February 25, 2003. (ML030160330)

ATTACHMENT 2

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

June 20, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 198

Strike Out TS Page:

TS 4.4-1

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.

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

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

June 20, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 198

Affected TS Page:

TS 4.4-1

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.

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 4

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

June 20, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 198

Risk Impact Assessment For Extending

Containment Type A Test Interval

Nuclear Management Company
Kewaunee Nuclear Power Plant

RISK IMPACT ASSESSMENT
FOR
EXTENDING CONTAINMENT TYPE A TEST INTERVAL

Analysis File 17547-0001-A3, Rev. 0

March 27, 2003

Prepared By: E R Schmidt **Date:** 3/28/03

Reviewed By: E A Krantz **Date:** 3/28/03

Accepted By: George H. Reuter **Date:** 4-10-03

SCIENTECH, Inc.

Gaithersburg, Maryland

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ANALYSIS FILE: 17547-0001-A3, Rev. 0

1.0 CLIENT **Nuclear Management Company – Kewaunee Nuclear Power Plant**

2.0 TITLE **Risk Informed/Risk Impact Assessment for Extending Containment Type A Test Interval**

3.0 AUTHOR **E. Robert Schmidt**

4.0 PURPOSE

The purpose of this calculation is to assess the risk impact for extending the Integrated Leak Rate Test (ILRT) interval for the Kewaunee Nuclear Power Plant (KNPP) from ten to fifteen years. In October 26, 1995, the Nuclear Regulatory Commission (NRC) revised 10 CFR 50, Appendix J. The revision to Appendix J allowed individual plants to select containment leakage testing frequency under Option A "Prescriptive Requirements" or Option B "Performance-Based Requirements". KNPP selected the requirements under Option B as its testing program.

The surveillance testing requirements (for Option B of Appendix J) as proposed in NEI 94-01 [Reference 1] for Type A testing is at least once per 10 years based on an acceptable performance history (defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage was less than 1.00La. KNPP will use this analysis to seek a one-time exemption from a 10 year test interval to a 15 year test interval.

5.0 INTENDED USE OF ANALYSIS RESULTS

The results of this calculation will be used to obtain NRC approval to extend the Integrated Leak Rate Test interval from one in ten years to one in fifteen years.

6.0 TECHNICAL APPROACH

The methodology used for this analysis is similar to the assessments originally performed for Crystal River 3 (CR3) [Reference 2] and Indian Point 3 (IP3) [Reference 3] with enhancements outlined in the EPRI Interim Guidance [Reference 4] and incorporated in numerous subsequent submittals, such as Salem [Reference 5] and D. C. Cook [Reference 6] The ILRT interval extensions requested by these submittals have been approved by the NRC. The impact of age-related degradation of the containment is also evaluated in a sensitivity study (see Appendix A) using methodology similar to that first employed in the Calvert Cliffs Nuclear Plant (CCNPP) response to an NRC Request for Additional Information (RAI) [Reference 7] and subsequently used in numerous other submittals including those for Comanche Peak and D. C. Cook [References 8 and 6].

This calculation was performed in accordance with NEI 94-01 [Reference 1] guidelines, and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a licensee request for changes to a plant's licensing basis, Regulatory Guide RG 1.174 [Reference 9]. This methodology is similar to that presented in EPRI TR-104285 [Reference 10] and NUREG-1493 [Reference 11] and incorporates the revised guidance and additional information of References 4 and 12. It uses a simplified bounding analysis approach to evaluate the risk impact of increasing the ILRT Type A interval from 10 to 15 years by using core damage and containment failure frequency information from the most recent update of the KNPP PRA [Reference 13]. Specifically, the following were considered:

- 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 pre-existing isolation failures of plant components other than those subjected to Type B or Type C tests. For example, this includes sequences with pre-existing liner breach or steam generator manway leakage (EPRI TR-104285 Class 3 sequences). Type B tests measure component leakage across pressure retaining boundaries (e.g., gaskets, expansion bellows and air locks). Type C tests measure component leakage rates across containment isolation valves.
- 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, this includes situations in which a valve fails to close following a valve stroke test (EPRI TR-104285 Class 6 sequences).
- Accident sequences involving containment failure induced by severe accident phenomena (EPRI TR-104285 Class 7 sequences), 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). The sequences of these classes are impacted by changes in Type B and C test intervals, not changes in the Type A test interval (Type A test measures the containment air mass and calculates the leakage from the change in mass over time).

Detailed descriptions of Classes 1 through 8 are excerpted from Reference 10 and provided in Table 1 of this analysis.

This calculation uses the following steps.

Step 1 – Quantify the baseline frequency per reactor year for each of the eight accident classes (See Table 2).

The KNPP Level 1 and 2 PRA analyses [Reference 13], and NUREG-1493 [Reference 11] were used to provide data to evaluate the annual frequencies for Classes 1,2,3,6,7 and 8. These frequencies are evaluated in detail in Section 11.1 of this analysis. Table 2 summarizes the results of this step. Class 4 and 5 sequences were not quantified because they are not impacted by the Type A test interval and are small contributors to the total. The containment failure modes modeled in the KNPP Level 2 analysis were based on important phenomena and system related events identified in NUREG-1335 [Reference 14].

Step 2 – Develop plant specific person-rem dose (population dose) per reactor year for each of the eight accident classes (See Table 4).

Reference 16 was used to develop person-rem for each of the classes described in Table 1 excluding Classes 4 and 5. Reference 15 is a calculation of the conditional person-rem dose to the population, within a 50-mile radius from the KNPP. The total population dose frequency in person-rem per year for each class is evaluated in detail in Section 11.2 of this analysis. Table 4 summarizes the results of this step.

Step 3 – Evaluate risk impact of extending Type A test interval.

This step evaluates potential increase in the population dose due to extending the ILRT test interval from that for 3 tests in 10 year (a 3 year interval) to a 10 year interval and to a 15 year interval. Section 11.3 of this calculation contains the detailed evaluation of this step. Section 13.0 and Tables 4, 5 and 6 summarize the results of this step.

Step 4 – Determine the change in risk in terms of Large Early Release Frequency (LERF) in Accordance with R.G. 1.174 [Reference 9].

This step evaluates the increase in the Large Early Release Frequency (LERF) due to extending the ILRT test interval from a 3 year test interval to a 15 year test interval and from a 10 year to a 15 year test interval. Section 11.4 of this calculation contains the detailed evaluation of this step while Section 13.0 summarizes the result of this step.

Step 5 – Determine the change in the Conditional Containment Failure Probability for the proposed and cumulative changes of Type A test interval.

This step evaluates the increase in the Conditional Containment Failure Probability (CCFP) due to extending the ILRT test interval from one test interval to another. CCFP is defined as: [1 – (Frequency Class1 + Frequency Class3a)/Core Damage Frequency (CDF)]. The changes in CCFP are evaluated in detail in Section 11.5 while Section 13.0 summarizes the results of this step.

The technical approach for the sensitivity study evaluating the potential impact of age-related corrosion of the steel containment is provided in Appendix A along with the detailed calculations and results.

7.0 INPUT INFORMATION

1. Updated PRA total Core Damage Frequency (CDF) and the frequency of various release categories from KNPP updated Level 2 PRA as calculated in Reference 13.

2. Population Doses for containment failure modes. Provided by "KNPP Year 2000 Offsite Dose Assessment", Calculation # 17547-0001-A1", dated 3/21/2003 [Reference 15].

3. Fraction of containment surface that cannot be inspected for Appendix J, ASME Section XI from "KNPP Calculation of Inspectable and Uninspectable Containment Vessel Surface Areas," Calculation # 17547-0001-A2, dated 3/24/2003 [Reference 16]

8.0 REFERENCES

1. NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10CFR Part 50, Appendix J, July 26, 1995, Revision 0.

2. "Crystal River – Unit 3 – License Amendment Request #267, Revision 2, Supplemental Risk-Informed Information in Support of License Amendment Request #267," Florida Power, 3F0601-06, June 20, 2001.

3. "Supplemental Information Regarding Proposed Change to Section 6.14 of the Administrative Section of the Technical Specification", Entergy, IPN-01-007, Indian Point 3 Nuclear Power Plant, January 18, 2001.

4. J. Haugh, J. M. Gisclon, W. Parkinson, K. Canavan, "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals", Rev. 4, EPRI, November, 2001.

5. "Request for Change to Technical Specifications, One-Time Extension to Increase the Interval of the Integrated Leak Rate Test from Ten to Fifteen Years, Salem Generating Station Unit 2," PSEG Nuclear LLC, March 22, 2002.

6. "Donald C. Cook Nuclear Plant Units 1 and 2, Response to Nuclear Regulatory Commission Request for Additional Information Regarding the License Amendment Request for a One-time Extension of Integrated Leakage Rate Test Interval," Indiana Michigan Power Company, November 11, 2002.

7. "Calvert Cliffs Nuclear Power Plant Unit No. 1; Docket No. 50-317, ," Constellation Nuclear letter to USNRC, March 27, 2002.

8. "Comanche Peak Steam Electric Station (CPSES), Docket Nos. 50-445 and 50-446, Response to Request for Additional Information Regarding License Amendment Request (LAR) 01-14 Revision to Technical Specification (TS) 5.5.16 Containment Leakage Rate Testing Program," TXU Energy letter to USNRC, June 12, 2002.

9. Regulatory Guide 1.174, " An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis" July 1998.

10. EPRI TR-104285, "Risk Assessment of Revised Containment Leak Rate Testing Intervals" August 1994.

11. NUREG-1493, "Performance-Based Containment Leak-Test Program, July 1995".

12. NEI Memo, "One-Time Extension of Containment Integrated Leak Rate Test Interval – Additional Information", Nuclear Energy Institute, November 30, 2001.

13. Edward Coen, "WinNUCAP Output.doc," KNPP, transmitted via E-mail, 3/11/03.

14. United States Nuclear Regulatory Commission, "Individual Plant Examination: Submittal Guidance," NUREG-1335, August 1989.

15. P.J. Fulford, "Risk Impact Assessment For Extending Containment Type A Test Interval," SCIENTECH, INC. Analysis File 17547-0001-A1, Rev. 0, March 21, 2003

16. S. E. Phillippi, "Calculation of Inspectable And Uninspectable Containment Vessel Surface Areas," SCIENTECH, INC. Analysis File 17547-0001-A2, Rev. 0 March 24, 2003

9.0 MAJOR ASSUMPTIONS:

1. The containment leakage for Class 1 sequences is assumed to be 1 La. [Reference 4]

2. The containment leakage for Class 3a sequences is assumed to be 10 La. [Reference 4]

3. The containment leakage for Class 3b sequences is assumed to be 35 La. [Reference 4]

4. Because Class 8 sequences are containment bypass sequences (e.g., Steam Generator Tube Rupture - SGTR, Isolation Loss of Coolant Accidents - ISLOCA), potential releases are primarily

directly to the environment. Therefore, the integrity of the containment structure will not significantly impact the release magnitude.

5. The probability of failure to detect a flaw during the visual inspection of the containment performed to satisfy the requirements of 10 CFR 50 Appendix J and ASME Section XI Subsection IWE is assumed to be 0.1 for the portion inspected.

10.0 IDENTIFICATION OF COMPUTER CODES

None used.

11.0 DETAILED ANALYSIS:

11.1 Step 1 – Quantify the baseline frequency per reactor year for each of the eight accident classes presented in Table 1.

As mentioned in the methods section above, step 1 quantifies the annual frequencies for the eight accident classes defined in Reference 11. Except for Class 1 and Class 7, the equations used in this quantification are very similar to those used in the Indian Point Unit 3 (IP3) Calculation [Reference 3]. Class 1 and Class 7 were evaluated based on the Crystal River Unit 3 (CR3) Calculation [Reference 2] where the term CI (CI is the sum of the frequencies for Classes 3a, 3b, and 6) is deducted from Class 1 as shown below. In the IP3 Calculation [Reference 3], the term CI was deducted from Class 7. Class 3 was evaluated based on Interim Guidance and Additional Information from EPRI and NEI [References 4 and 12].

Reference 13 provides the following results of the latest KNPP PRA update. Also included are the accident classes corresponding to the KNPP Release Categories (RCs).

KNPP Release Category	Description	Frequency (per year)	Accident Class
1	No Cont. Failure	2.026E-06	1
2	Isol. Failure	1.594E-05	2
3	LER - Isol. Failure	4.882E-10	2
4	Basemat Melthrough	6.959E-06	7
5	Press. Failure	1.707E-06	7
6	LER - Press. Failure	3.690E-10	7
7	LER - ISLOCA	1.976E-07	8
8	LER - SGTR	2.633E-06	8
	TOTAL CDF	2.947E-05	

The annual frequencies for each accident class are assessed as follows:

Class 1 Sequences. This group consists of all core damage accident progression bins for which the containment remains intact. For this analysis the associated maximum containment leakage for this group is 1 La. The frequency for these sequences is determined as follows:

Class_1_Frequency = No_Cont_Failure_Freq – CI

Where:

No-Cont_Failure_Freq = 2.026E-06/yr [From table above for RC 1]

CI = Class_3a_Frequency + Class_3b_Frequency + Class_6_Frequency

= 2.228E-07/yr + 2.014E-08/yr + 2.947E-08 /yr = 2.724E-07/yr
[These values are obtained from the Class 3 and 6 sequences sections below.]

or

Class_1_Frequency = 2.026E-06/yr – 2.724E-07/yr = 1.754E-06/yr

<u>Class 2 Sequences.</u> This group consists of all core damage accident progression bins for which pre-existing leakage due to failure to isolate the containment occurs. These sequences are dominated by failure to close of greater than 2 inches diameter but less than 5 inches diameter containment isolation valves (RC 2). Failure to close of very large isolation valves (greater than 5 inches) that could lead to a large early release (LER) (RC 3) have a much lower frequency.

The frequency for these sequences is determined as follows:

Class_2_Frequency = The sum of RC 2 and 3 frequencies [From table above]
Class_2_Frequency = 1.594E-05/yr + 4.882E-10/yr
Class_2_Frequency = 1.594E-05/yr

<u>Class 3 Sequences.</u> This group consists of all core damage accident progression bins for which pre-existing leakage in the containment structure (i.e., containment liner) exists. The containment leakage for these sequences can be either small (10 La for Class 3a) or large (35 La for Class 3b).

For this analysis, the question on containment analysis was modified to include the probability of a liner breach (due to excessive leakage) at the time of core damage. This class is divided into two classes (Class 3a and Class 3b). Class 3a is defined as small liner breach and Class 3b represents a large containment breach. Evaluation of these two classes is based on EPRI TR-104285 [Reference 10], the EPRI Interim Guidance [Reference 4] and the NEI Additional Information [Reference 12].

The frequency for this Class event is determined as follows:

Class_3a_Frequency = Prob(Class 3a)*CDF*(Probability that a pre-existing leak is not detected by visual examination)

Class_3b_Frequency = Prob(Class 3b)* (portion of CDF that may be impacted by Type A leakage and contribute to Class 3b) * (Probability that a pre-existing leak is not detected by visual examination)

Frequency of Class 3a Event (Small Containment Breach) –Class_3a_Frequency

To calculate the probability that a liner leak will be small (Class 3a), use was made of the data presented in NUREG-1493 [Reference 12] and the EPRI Interim Guidance [Reference 4]. NUREG-1493 states that 144 ILRTs have been conducted. The data reported that 23 of 144 tests had allowable leak rates in excess of 1 La. However, of these 23 'failures,' only 4 were found by an ILRT. The others were found by Type B and C testing or errors in test alignments. Therefore, the number of failures considered for 'small releases' are 4 of 144. The EPRI Interim Guidance stated that one failure found by an ILRT was found in 38 ILRTs performed after NUREG-1493. Thus, the best estimate of the probability of a small leak, Prob(Class 3a), is calculated as 5/182 = 0.027 [Reference 4].

The total updated CDF is 2.947E-05 / yr from Reference 13.

In addition to the above, there is the expectation that visual inspection in accordance with Appendix J of ASME Section XI will detect liner leaks. Probability that a pre-existing leak is not detected by visual examination = [Fraction of containment liner that cannot be inspected for Appendix J, ASME Section XI] + [Fraction of containment that can be inspected for Appendix J, ASME Section XI] * [Probability of failure to detect a flaw during a visual inspection]

Where:

Fraction of containment liner that cannot be inspected for Appendix J, ASME Section XI
= 0.20 [From Reference 16]

Fraction of containment liner that can be inspected for Appendix J, ASME Section XI
= 1- 0.20 = 0.80

Probability of failure to detect a flaw during a visual inspection = 0.1 [Assumption 5]

Probability that a pre-existing leak is not detected by visual examination =

= 0.20 + (0.80 * 0.1) = 0.28

Therefore the frequency of release due to Class 3a failures is calculated as:

Class_3a_Frequency = Prob(Class 3a) * CDF * (Probability that a pre-existing leak is not detected by visual examination)

= 0.027 * 2.947E-05/yr * 0.28 = 2.228E-07/yr

Frequency of Class 3b Event (Large Containment Breach) –Class_3b_Frequency

To calculate the probability that a liner leak will be large (Class 3b), use was made of the data presented in NUREG-1493 [Reference 11] and new data presented by the EPRI Interim Guidance [Reference 4]. One data set found in NUREG-1493 reviewed 144 ILRTs and the EPRI Interim Guidance reviewed additional 38 ILRTs. The largest reported leak rate from those 144 tests was 21 times the allowable leakage rate (La). Since 21 La does not constitute a large release, no large releases have occurred based on the 144 ILRTs reported in NUREG-1493. One failure was found in the 38 ILRTs discussed in the EPRI Interim Guidance and this failure was not considered large.

Because no Class 3b failures have occurred in 182 ILRT tests, the EPRI Interim Guidance suggested that the Jeffery's non-informative prior distribution would be appropriate for the Class 3b distribution. (The rationale for using the Jeffery's non-informative prior distribution was discussed in Reference 4.)

Prob(Class 3b) = Failure probability = (# of failures (0) + ½)/(Number of tests (182) + 1)

The number of large failures is zero and the probability is

Prob(Class 3b) = 0.5/183 = 0.0027

The use of this probability and the total core damage frequency (CDF) as the Class 3b frequency is very conservative since not all core damage sequences will contribute to releases equivalent to a Class 3b failure. A number of sequences (containment bypass sequences and those resulting in a early containment failure due to severe accident phenomena –hydrogen explosion, etc.) will lead to large risk-significant releases regardless if there is a preexisting leak or not and including them in Class 3b is not appropriate. Further, there are a number of sequences that would not lead to large risk-significant releases due to the presence of release mitigation or significant warning time before release. Therefore:

PCDF_TypeA = Portion of CDF that may be impacted by Type A leakage and contribute to Class 3b = Total CDF – (CDF of sequences that have a large release irrespective of Type A Leakage) - (CDF of sequences that cannot cause a large risk significant release)

Where:

CDF = 2.947E-05/yr [From Reference 13]

CDF of sequences that have a large release irrespective of Type A Leakage
= Sum of RC 3, RC 6, RC 7 and RC 8
= 4.882E-10 + 3.690E-10 + 1.976E-07 + 2.633E-06 [From table above]
= 2.831E-06/yr

CDF of sequences that cannot cause a large risk significant release
= 0
(No credit taken for containment spray in KNPP PRA due to dominate contributor to CDF being station blackout)

Therefore:

PCDF_TypeA = 2.947E-05 – 2.831E-06 - 0 = 2.664E-05/yr

Also, as discussed above for Class 3a, there is the expectation that visual inspection in accordance with Appendix J of ASME Section XI will detect liner leaks.

Probability that a pre-existing leak is not detected by visual examination = 0.28

Therefore the frequency of release due to Class 3b failures is calculated as:

Class_3b_Frequency = Prob(Class 3b) * PCDF_TypeA * (Probability that a pre-existing leak is not detected by visual examination)

= 0.0027 * 2.664E-05 * 0.28 = 2.014E-08 / yr

<u>Class 4 Sequences</u>. This group consists of all core damage accident progression bins for which a failure-to-seal containment isolation due to failure of Type B test components occurs. Because these failures are detected by Type B tests, this group is not evaluated further.

<u>Class 5 Sequences</u>. This group consists of all core damage accident progression bins for which a failure-to-seal containment isolation due to failure of Type C test components occurs. Because these failures are detected by Type C tests, this group is not evaluated further.

Class 6 Sequences. This group is similar to Class 2 and addresses additional failure modes not typically modeled in PRAs due to the low probability of occurrence. 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.

The low failure probabilities are based on the need for multiple failures, the presence of automatic closure signals, and control room indication. Based on the purpose of this calculation, and the fact that this failure class is not impacted by Type A testing, no further evaluation is needed. This is consistent with the EPRI guidance. However, in order to maintain consistency with the previously approved methodology, i.e., PROB(Class6) > 0, a conservative screening value of 1.0E-03 will be used to evaluate this class.

The annual frequency for these sequences is determined as follows:

Class_6_Frequency = (Screening Value) *CDF

Where:

Screening Value = 1.0×10^{-3} [Assumed Conservative Value]

CDF = 2.947-05/yr

Class_6_Frequency = 1.0E-03 * 2.947E-05/yr = 2.947E-08 /yr

Class 7 Sequences. This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs (i.e., H2 combustion). For this analysis the associated maximum containment leakage for this group is 35 La.
The annual frequency for these sequences is determined as follows:

Class_7_Frequency = Sum of RC 4, RC 5 and RC 6 Frequencies
= 6.959E-06 + 1.707E-06 + 3.690E-10 [From above table]
= 8.666E-06/yr

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass occurs. The failure frequency for this class is:

Class_8_Frequency = Sum of RC 7 and RC 8 Frequencies
= 1.976E-07 + 2.633E-06 [From above table]
= 2.831E-06/yr

Note for this class the maximum release is not based on normal containment leakage, because most of the releases are directly to the environment. Therefore, the integrity of the containment structure will not significantly impact the release magnitude.

The annual frequencies for the eight classes are summarized in Table 2.

11.2 Step 2 – Develop plant specific person-rem dose (population dose) per reactor year for each of the eight accident classes and quantify baseline risk

In accordance with guidance given by Reference 10, this step develops the KNPP population dose and evaluates the baseline risk impact for the eight accident classes defined in the previous sections of this calculation.

2a) Characterize accident scenarios into major groups (eight classes).

(See Class 1 through 8 sequences above)

2b) Develop plant specific person-rem dose (population dose) per reactor year.

Reference 15 documents an assessment of the KNPP site population dose consequences due to the accidental release of radiological materials resulting from several severe accident scenarios. This assessment utilizes the meteorology, year 2000 population distribution, geographic data, evacuation time estimates and other offsite data from a recent Level 3 analysis for the Point Beach Nuclear Plant (PBNP) which is located approximately 4 miles from KNPP. 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 source terms used for the KNPP consequence analysis are for a planned KNPP uprated power level of 1772 MWth and KNPP specific severe accident analysis for sequences representative of the 8 Release Categories. The 50-mile population dose (person-rem) for each RC are given below along with the RC frequency, the risk in person-rem/year (the product of the frequency and the population dose) and the EPRI accident class.

KNPP Release Category	Description	Frequency (per year)	Population Dose (person-rem)	Risk (person-rem/year)	Accident Class
1	No Cont. Failure	2.026E-06	1.200E+02	2.431E-04	1
2	Isol. Failure	1.594E-05	2.010E+05	3.204E+00	2
3	LER - Isol. Failure	4.882E-10	2.970E+05	1.450E-04	2
4	Basemat Melthrough	6.959E-06	7.510E+01	5.226E-04	7
5	Press. Failure	1.707E-06	4.040E+05	6.896E-01	7
6	LER - Press. Failure	3.690E-10	2.600E+05	9.594E-05	7
7	LER - ISLOCA	1.976E-07	1.170E+06	2.312E-01	8
8	LER - SGTR	2.633E-06	6.350E+05	1.672E+00	8
	TOTAL CDF	2.947E-05		5.798E+00	

The population dose for each accident class in the table is determined from the total risk for the class divided by the total frequency for the class, or

Class 2 = (3.204E+00 + 1.450E-04)/(1.594E-05 + 4.882E-10) = 2.010E+05 person-rem
Class 7 = (5.226E-04 + 6.896E-01 + 9.594E-05)/(6.959E-06 + 1.707E-06 + 3.690E-10)
= 7.965E+04 person-rem
Class 8 = (2.312E-01 + 1.672E+00)/(1.976E-07 + 2.633E-06) = 6.723E+05 person-rem

The population dose for Classes 3a and 3b are taken to be 10 and 35, respectively, times that for Class 1 based on the assumed leakage rates of 10 La and 35 La.

Class 1 = (1.20E+02) * 1 La = 1.20E+02 person-rem
Class 3a = (1.20E+02) * 10 La = 1.20E+03 person- rem
Class 3b = (1.20E+02) * 35 La = 4.20E+03 person-rem

The population dose for Class 6 is assumed to be the same as that for Class 2.

The above values are summarized in Table 3.

2c) Calculate and Review Baseline Risk for Each Accident Class

The baseline risk for each accident class is presented in Table 4. The baseline risk is defined as the product of the containment failure mode frequency and the conditional population dose. Table 4 is the product of Tables 2 and 3. The ILRT baseline risk is based on the test interval corresponding to 3 tests in 10 years or about a 3 year interval.

As mentioned in the method section of this calculation, only Classes 3a and 3b are impacted by the Type A ILRT test. Therefore, the percent risk contribution (%Base_Risk) for these classes is:

%Base_Risk = [(Class3a_Base + Class3b_Base) / Total_base)] * 100

Where:

Class3a_Base = 2.674E-04 person-rem/year

Class3b_Base = 8.458E-05 person-rem/year

Class_3_Base_Total = 2.674E-04 + 8.458E-05 = 3.519E-04 person-rem/yr

Total_base = 5.804 person-rem/year

%Base_Risk = (3.519E-04 / 5.804) * 100
%Base_Risk = 0.0061%

Therefore, the total baseline risk contribution of leakage, potentially impacted by the ILRT test interval, represented by Class 3 accident scenarios is 0.00035 person-rem/year or 0.0061% of the total population exposure risk.

11.3 <u>Step 3 – Evaluate risk impact of extending Type A test interval.</u>

Risk Impact due to 10-year test interval

According to NUREG-1493 [Reference 11], extending the Type A ILRT interval from that corresponding to 3 tests in 10 years to that for 1 test in 10 years will increase the average time that a leak detectable only by an ILRT goes undetected from 18 to 60 months. The average time that a pre-existing leak may go undetected is calculated by multiplying the test interval by 0.5 and multiplying by 12 to convert from "years" to "months." The recent EPRI Guidance suggested use the factor of 3.33 (60/18) to estimate the increase of Class 3 since Type A tests impact only Class 3 sequences. Also, as with the baseline case, the frequency of Class 1 has been reduced by the frequencies of Classes 3a, 3b, and Class 6 in order to preserve total CDF.

The results of this calculation are presented in Table 5.

Based on the above values, the Type A 10-year test frequency percent risk contribution (%Risk_10) for Class 3 is as follows:

%Risk_10 = [(Class3a_10 +Class3b_10) / Total_10] * 100

Where:

Class3a_10 = 8.903E-04 person-rem/year

Class3b_10 = 2.817E-04 person-rem/year

Class3_10_total = 8.903E-04 + 2.817E-04 = 1.172E-03 person-rem/year

Total_10 = 5.805E+00 person-rem/year

%Risk_10 = (1.172E-03 / 5.805E+00) * 100

%Risk_10 = 0.020%

Therefore, the total risk contribution of leakage for Type A 10-Year ILRT interval represented by Class 3 accident scenarios is 0.00117 person-rem/year or 0.020% of the total population risk.

Since the only change in risk is due to the change in Class 3 (conservatively neglecting the reduction in risk for Class 1), the percent risk increase due to extending the ILRT interval from that corresponding to 3 tests in 10 years (baseline case) to that corresponding to 1 test in 10 years is evaluated as follows:

[(Total_10 – Total_base) / Total_base] * 100 =
[(Class3_10_total - Class_3_Base_Total) / Total_base] * 100

Where:

Class_3_Base_Total = 3.519E-04 person-rem/yr	[From above]
Class3_10_total = 1.172E-03 person-rem/year	[From above]
Total_base = 5.804 person-rem/year	[From Table 4]

[(Class3_10_total - Class3_Base_total) / Total_base] * 100
= [(1.172E-03 - 3.519E-04) / 5.804] * 100 = (8.20E-04/5.804) * 100 = 0.014 %

Therefore, The total risk increase due to extending the ILRT interval from that corresponding to 3 tests in 10 years (baseline case) to that corresponding to 1 test in 10 years is 0.00082 person-rem/year or 0.014% of the total population risk.

Risk Impact due to 15-year test interval

The risk contribution for a 15-year interval is similar to the 10-year interval. The difference is in the increase in probability of leakage value. If the test interval is extended to 15 years, the mean time that a leak detectable only by an ILRT test goes undetected increases to 90 months (0.5 * 15 * 12). Reference 12 suggested to use a factor of 5 (90/18) to account for the increased likelihood of fail to detect, which will be implemented here. As with the baseline case, the PRA frequency of Class 1 has been reduced by the frequency of Class 3a, 3b, and Class 6 in order to preserve total CDF. The results for this calculation are presented in Table 5.

Based on the above values, the Type A 15-year test interval percent risk contribution (%Risk _15) for Class 3 is as follows:

%Risk_15 = [(Class3a_15 +Class3b_15) / Total_15] * 100

Where:

Class3a_15 = 1.337E-03 person-rem/year

Class3b_15 = 4.229E-04 person-rem/year

Class3_15_total = 1.337E-03 + 4.229E-04 = 1.760E-03 person-rem/year

Total_15 = 5.805 person-rem/year [From Table 5]

%Risk_15 = (1.760E-03 / 5.805) * 100
%Risk_15 =0.030%

Therefore, the total risk contribution of leakage for Type A 15-year ILRT interval represented by Class 3 accident scenarios is 0.00176 person-rem/year or 0.030% of the total population risk.

The percent risk increase due to extending the ILRT interval from that corresponding to 3 tests in 10 years (baseline case) to that corresponding to 1 test in 15 years is evaluated as follows:

[(Total_15 – Total_base) / Total_base] * 100 =
[(Class3_15_total - Class_3_Base_Total) / Total_base] * 100

Where:

Class3_15_total = 1.760E-03 person-rem/year [From above]
Class_3_Base_Total = 3.519E-04 person-rem/yr [From above]
Total_base = 5.804 person-rem/year [From Table 4]

[(Class3_15_total - Class_3_Base_Total) / Total_base] * 100
= [(1.760E-03 – 3.519E-04)/ 5.804] * 100 = (1.408E-03/5.804) * 100 = 0.024%

Therefore, the total risk increase due to extending the ILRT interval from that corresponding to 3 tests in 10 years (baseline case) to that corresponding to 1 test in 15 years is 0.00141 person-rem/year or 0.024% of the total baseline population risk.

The percent risk increase in terms of person-rem/year from a 10 year to a 15 year test interval for Classes 3a and 3b is:

% Risk (10-15PR) =[(Class3_15_total) – (Class3_10_Total) / (Class3_10_Total)]*100

Where:

Class3_15_total = 1.760E-03 person-rem/year [From above]
Class3_10_Total = 1.172E-03 person-rem/year [From above]

% Risk (10-15PR) = [(1.760E-03 – 1.172E-03) /1.172E-03] * 100 = 50%

The increase in person-rem/year for all accident classes (conservatively neglecting the reduction in Class 1 risk) from 1 in 10 years to 1 in 15 years test interval is:

(Class3_15_total - Class3_10_Total) = 1.760E-03 – 1.172E-03 = 5.88E-04 person-rem/year

The percent risk increase due to extending the ILRT interval from 1 in 10 years to 1 in 15 years is evaluated as follows:

[(Class3_15_total - Class3_10_Total) / Total_10] * 100

Where:

Class3_15_total = 1.760E-03 person-rem/year	[From above]
Class3_10_Total = 1.172E-03 person-rem/year	[From above]
Total_10 = 5.805 person-rem/year	[From Table 5]

[(Class3_15_total - Class3_10_Total) / Total_10] * 100 = [(1.760E-03 – 1.172E-03)/ 5.805] * 100
= (5.88E-04 /5.805) * 100 = 0.010%

Therefore, the total risk increase due to extending the ILRT interval from 10 years to 15 years is 0.00059 person-rem/year or 0.010% of the total baseline population risk.

11.4 Step 4 – Determine the change in risk in terms of Large Early Release Frequency (LERF)

This step evaluates the increase in the Large Early Release Frequency (LERF) due to extending the ILRT test interval from that corresponding to 3 tests in 10 years to that corresponding to 1 test in 15 years and from a 10 year interval to a 15 year interval.

The risk impact associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from containment could in fact result in large release due to failure to detect a pre-existing leak during the relaxation period. For this evaluation only Class 3b sequences, which have the potential to result in large releases if pre-existing leak were present, are impacted by the ILRT Type A test.

The previous methodology [References 2 and 3] employed for determining LERF (Class 3b frequency) involved multiplying the total CDF by the failure probability for this class (3b) of accident. This was done for simplicity and is conservative. 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. For instance, the CR3 [Reference 2] evaluation assumption number 7 states that "The containment releases for Classes 2, 6, 7, and 8 are not impacted by the ILRT Type A test frequency. These classes already include containment failure with release consequences equal or greater than those impacted by Type A."

These corrections have been accounted for in determining the Class 3b frequency in Section 11.1 above. Consequently the LERF values affected by the ILRT are equal to the Class 3b frequencies given above, or

The Baseline LERF affected by ILRT = 2.014E-08 per year	[Table 4]
The 1 in 10 years LERF affected by ILRT = 2.014E-08* 3.33 = 6.706E-08 per year	[Table 5]
The 1 in 15 years LERF affected by ILRT = 2.014E-08* 5 = 1.007E-07 per year	[Table 6]

Change in LERF due to test interval going from that corresponding to 3 tests in 10 years to that corresponding to 1 test in 15 years =

1.007E-07 – 2.014E-08 = 8.06E-08/year

Change in LERF due to test interval going from 1 in 10 years to 1 in 15 years =

1.007E-07 – 6.706E-08 = 3.36E-08/year

11.5 Step 5 – Determine the change in the Conditional Containment Failure Probability (CCFP) for the proposed and cumulative changes of Type A test interval

The change in Conditional Containment Failure Probability (CCFP) for the proposed and cumulative changes is estimated as follows:

1. Estimate the CCFP for each test interval (i.e., 3 years, 10 years, and 15 years)
2. Calculate the change in CCFP between the test intervals.

The Conditional Containment Failure Probability (CCFP) can be defined as:

[1 – (Class_1_ Frequency + Class_3a_ Frequency)/CDF]

Where

Class_1_ Frequency = Frequency per year of No Containment Failure.

Class_3a_ Frequency = Frequency per year of Small Isolation Failure.

Using the above equation and the data from Table 4 (i.e., Class 1 frequency is 1.754E-06 per year, the Class 3a frequency is 2.228E-07 year and CDF is 2.947E-05 per year),

the CCFP for 3 tests in 10 years =

1 – [(1.754E-06 + 2.228E-07)/2.947E-05] = 0.9329

Using the above equation and the data from Table 5 (i.e., Class 1 frequency is 1.188E-06 per year , the Class 3a frequency is 7.419E-07 per year and CDF is 2.947E-05 per year),

the CCFP for 1 test in 10 years =

1-[(1.188E-06 + 7.419E-07) / 2.947E-05] = 0.9345

Using the above equation and the data from Table 6 (i.e., Class 1 frequency is 7.819E-07 per year , the Class 3a frequency is 1.114E-06 per year and CDF is 2.947-05 per year),

the CCFP for 1 test in fifteen years =
1-[(7.819E-07 + 1.114E-06)/2.947E-05] = 0.9357

The change in CCFP due to the ILRT interval going from that corresponding to 3 tests in 10 years to that corresponding to 1 test in 15 years

= 0.9357 – 0.9329 = 0.0028

The change in CCFP due to the ILRT interval going from that corresponding to 1 test in 10 years to that corresponding to 1 test in 15 years

= 0.9357 – 0.9345 = 0.0012

12.0 COMPUTER INPUT AND OUTPUT

NONE

13.0 SUMMARY OF RESULTS

The table below summarizes the major results.

	Test Interval Extended	
	From 3 in 10 years to 1 in 15 years	From 1 in 10 years to 1 in 15 years
Total person-rem/year increase (See Section 11.3)	0.00141	0.00059
The percentage increase person-rem/year risk (See Section 11.3)	0.024%	0.010%
Change in LERF – per year (See Section 11.4)	8.1E-08	3.4E-08
Change in the Conditional Containment Failure Probability (See Section 11.5)	0.0028	0.0012

Other results are shown in the following table.

Class	Risk Impact		
	Baseline 3 in 10 years	1 in 10 years	1 in 15 years
3a and 3b. These classes are impacted by Type A test	0.0061% of integrated value based on 10 La for Class 3a and 35 La for Class 3b, which is equivalent to: 0.00035 person-rem/year	0.020 % of integrated value based on 10 La for Class 3a and 35 La for Class 3b, which is equivalent to: 0.0012 person-rem/year	0.030% of integrated value based on 10 La for Class 3a and 35 La for Class 3b, which is equivalent to: 0.0018 person-rem/year
Total Integrated Risk	5.804 person-rem/year	5.805 person-rem/year	5.805 person-rem/year

Appendix A provides an assessment of the sensitivity of the above results to age-related corrosion of the containment shell. The above major results are repeated below along with the results if the impact of age-related corrosion is included.

	Test Interval Extended	
	From 3 in 10 years to 1 in 15 years	From 1 in 10 years to 1 in 15 years
Total person-rem/year increase		
Without Corrosion	0.00141	0.00059
Including Corrosion	0.00144	0.00061
The percentage increase in person-rem/year risk		
Without Corrosion	0.024%	0.010%
Including Corrosion	0.025%	0.011%
Change in LERF (per year)		
Without Corrosion	8.1E-08	3.4E-08
Including Corrosion	8.8E-08	3.8E-08
Change in the Conditional Containment Failure Probability		
Without Corrosion	0.0028	0.0012
Including Corrosion	0.0031	0.0014

14.0 CONCLUSIONS:

The conclusions regarding the change in plant risk associated with extension of the Type A ILRT test frequency from ten-years to fifteen-years, based on the results in Section 13, are as follows:

The change in Type A test frequency from once per 10 years to once per 15 years increases the total integrated plant risk for those accident sequences influenced by Type A testing by only 0.00059 person-rem/year. This increase in person-rem/year is negligible when compared to other accident risks.

Reg.Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Very small changes in risk are defined in Reg. Guide 1.174 as increases of CDF below 1.0E-06/yr or increases in LERF of less than 1E-07/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test frequency from once per 10 years to once per 15 years is 3.4E-08/yr. Since guidance in Reg. Guide 1.174 defines very small changes in LERF as below 1.0E-7/yr, increasing the ILRT interval from 10 to 15 years is therefore considered very small and non-risk significant.

The change in conditional containment failure probability due to the requested change in ILRT frequency is 0.0012 and is also very small.

The cumulative impact of the change in ILRT frequency from 3 tests in 10 years to 1 test in 15 years is an increase in integrated risk of 0.0014 person-rem/year or 0.024% of the baseline risk, an increase in LERF of 8.1E-08 per year and an increase of 0.0028 in conditional containment failure probability. All of these changes meet the above criteria.

The impact of age-related corrosion of the steel containment has essentially a negligible impact on each of the risk measures associated with the extension of the Type A ILRT test frequency. The above conclusions remain valid.

Table 1- Detailed Description for the Eight Accident Classes as defined by EPRI TR-104285

Class	Detailed 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. The allowable leakage rates (L_a), are typically 0.1 weight percent of containment volume per day for PWRs .(all measured at P_a, calculated peak containment pressure related to the design basis accident). Changes to leak rate testing frequencies do not affect this classification.
2	Containment isolation failures (as reported in the IPEs) include those accidents in which the pre-existing leakage is due to failure to isolate the containment. These include those that are dependent on the core damage accident in progress (e. g., initiated by common cause failure or support system failure of power) and random failures to close a containment path. Changes in Appendix J testing requirements do not impact these accidents.
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. This accident class is applicable to sequences involving ILRTs (Type A tests) and potential failures not detectable by LLRTs.
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 not identified by the LLRTs. The type of penetration failures considered under this class includes those covered in the plant test and maintenance requirement or verified by in service inspection and testing (ISI/IST) program. This failure to isolate is not typically identified in LLRT. Changes in Appendix J LLRT test intervals do not impact this class of accidents.
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 typically impact these accidents, particularly for PWRs.

TABLE 2 - Containment Frequency Measures for a Given Accident Class

Class	Description	Frequency/yr.
1	No Containment Failure	1.754E-06
2	Large Containment Isolation Failure (Failure-To-Close)	1.594E-05
3a	Small Isolation Failures (Liner Breach)	2.228E-07
3b	Large Isolation Failures (Liner Breach)	2.014E-08
4	Small Isolation Failure – Failure-To-Seal (Type B test)	
5	Small Isolation Failure – Failure-To-Seal (Type C Test)	
6	Containment isolation Failures (Dependent failures, Personnel Errors)	2.947E-08
7	Severe Accident Phenomena Induced Failure (Early and Late Failures)	8.666E-06
8	Containment Bypassed (SGTR)	2.831E-06
Core Damage	All Containment Event Tree (CET) Endstates	2.947E-05

TABLE 3 – Conditional Person-Rem Measures for a Given Accident Class

Class	Description	Person-Rem (50-miles)
1	No Containment Failure	1.200E+02
2	Large Containment Isolation Failure (Failure-To-Close)	2.010E+05
3a	Small Isolation Failures (Liner Breach)	1.200E+03
3b	Large Isolation Failures (Liner Breach)	4.200E+03
4	Small Isolation Failure – Failure-To-Seal (Type B test)	
5	Small Isolation Failure – Failure-To-Seal (Type C Test)	
6	Containment isolation Failures (Dependent failures, Personnel Errors)	2.010E+05
7	Severe Accident Phenomena Induced Failure (Early and Late Failures)	7.965E+04
8	Containment Bypassed (SGTR)	6.723E+05

TABLE 4 – Baseline Mean Consequence Measures for a Given Accident Class

Class	Description	Frequency/yr	Person-Rem (50-miles)	Person-Rem/yr (50-miles)
1	No Containment Failure	1.754E-06	1.200E+02	2.104E-04
2	Large Containment Isolation Failure (Failure-To-Close)	1.594E-05	2.010E+05	3.204E+00
3a	Small Isolation Failures (Liner Breach)	2.228E-07	1.200E+03	2.674E-04
3b	Large Isolation Failures (Liner Breach)	2.014E-08	4.200E+03	8.458E-05
4	Small Isolation Failure – Failure-To-Seal (Type B test)			0.000E+00
5	Small Isolation Failure – Failure-To-Seal (Type C Test)			0.000E+00
6	Containment isolation Failures (Dependent failures, Personnel Errors)	2.947E-08	2.010E+05	5.924E-03
7	Severe Accident Phenomena Induced Failure (Early and Late Failures)	8.666E-06	7.965E+04	6.902E-01
8	Containment Bypassed (SGTR)	2.831E-06	6.723E+05	1.903E+00
	All CET End states	2.947E-05		5.804E+00

TABLE 5 Mean Consequence Measures for 10 – Year Test Interval for a Given Accident Class

Class	Description	Frequency/yr	Person-Rem (50-miles)	Person-Rem/yr (50-miles)
1	No Containment Failure	1.188E-06	1.200E+02	1.425E-04
2	Large Containment Isolation Failure (Failure-To-Close)	1.594E-05	2.010E+05	3.204E+00
3a	Small Isolation Failures (Liner Breach)	7.419E-07	1.200E+03	8.903E-04
3b	Large Isolation Failures (Liner Breach)	6.706E-08	4.200E+03	2.817E-04
4	Small Isolation Failure – Failure-To-Seal (Type B test)			0.000E+00
5	Small Isolation Failure – Failure-To-Seal (Type C Test)			0.000E+00
6	Containment isolation Failures (Dependent failures, Personnel Errors)	2.947E-08	2.010E+05	5.924E-03
7	Severe Accident Phenomena Induced Failure (Early and Late Failures)	8.666E-06	7.965E+04	6.902E-01
8	Containment Bypassed (SGTR)	2.831E-06	6.723E+05	1.903E+00
CDF	All CET Endstates	2.947E-05		5.805

TABLE 5 - Mean Consequence Measures for 15 – Year Test Interval for a Given Accident Class

Class	Description	Frequency/yr	Person-Rem (50-miles)	Person-Rem/yr (50-miles)
1	No Containment Failure	7.819E-07	1.200E+02	9.382E-05
2	Large Containment Isolation Failure (Failure-To-Close)	1.594E-05	2.010E+05	3.204E+00
3a	Small Isolation Failures (Liner Breach)	1.114E-06	1.200E+03	1.337E-03
3b	Large Isolation Failures (Liner Breach)	1.007E-07	4.200E+03	4.229E-04
4	Small Isolation Failure – Failure-To-Seal (Type B test)			0.000E+00
5	Small Isolation Failure – Failure-To-Seal (Type C Test)			0.000E+00
6	Containment isolation Failures (Dependent failures, Personnel Errors)	2.947E-08	2.010E+05	5.924E-03
7	Severe Accident Phenomena Induced Failure (Early and Late Failures)	8.666E-06	7.965E+04	6.902E-01
8	Containment Bypassed (SGTR)	2.831E-06	6.723E+05	1.903E+00
CDF	All CET End States	2.947-05		5.805

ANALYSIS FILE: 17547-0001-A3, Rev. 0, Appendix A

A.1.0 CLIENT Nuclear Management Company – Kewaunee Nuclear Power Plant

A.2.0 TITLE Effect of Age-Related Degradation on Risk Informed/Risk Impact Assessment for Extending Containment Type A Test Interval

A.3.0 AUTHOR E. Robert Schmidt

A.4.0 PURPOSE

The purpose of this calculation is to assess the effect of age-related degradation of the containment on the risk impact for extending the Kewaunee Nuclear Power Plant (KNPP) Integrated Leak Rate Test (ILRT or Containment Type A test) interval from ten to fifteen years.

A.5.0 INTENDED USE OF ANALYSIS RESULTS

The results of this calculation will be used to indicate the sensitivity of the risk associated with the extension in the ILRT interval to potential age-related degradation of the containment shell to support obtaining NRC approval to extend the Integrated Leak Rate Test (ILRT) interval at KNPP from 10 years to 15 years.

A.6.0 TECHNICAL APPROACH

The present analysis shows the sensitivity of the results of the assessment of the risk impact of extending the Type A test interval for the KNPP to age-related liner corrosion.

The prior assessment included the increase in containment leakage for EPRI Containment Failure Class 3 leakage pathways that are not included in the Type B or Type C tests. These classes (3a and 3b) include the potential for leakage due to flaws in the containment shell. The impact of increasing the ILRT interval for these classes included the probability that a flaw would occur and be detected by the Type A test that was based on historical data. Since the historical data includes all known failure events, the resulting risk impact inherently includes that due to age-related degradation.

The present analysis is intended to provide additional assurance that age-related liner corrosion will not change the conclusions of the prior assessment. The methodology used for this analysis is similar to the assessments performed for Calvert Cliffs Nuclear Power Plant (CCNPP - Reference A1), Comanche Peak Steam Electric Station (CPSES - Reference A2), D. C. Cook (CNP - Reference A3) and St. Lucie (SL - Reference A4) in responses to requests for additional information (RAIs) from the NRC staff. The CCNPP, CPSES and CNP extension request submittals have been approved by the NRC.

The significantly lower potential for corrosion of free-standing steel shell containments, such as that at KNPP, is considered. This is due to the significantly smaller surface area susceptible to corrosion resulting from foreign material imbedded in concrete contacting the steel containment. Because of this, the analysis is carried out separately for those portions of the containment not in potential contact with foreign material and those portions in potential contact with the foreign material. (This is considered more appropriate than the cylinder and dome portions and the basemat portions utilized in prior analyses.)

As in Reference A1, this calculation uses the following steps with KNPP values utilized where appropriate:

Step 1 – Determine a corrosion-related flaw likelihood

Historical data will be used to determine the annual rate of corrosion flaws for the containment. The significantly lower potential for corrosion in the free-standing KNPP containment will be included.

Step 2 – Determine an age-adjusted flaw likelihood

The historical flaw likelihood will be assumed to double every 5 years. The cumulative likelihood of a flaw is then determined as a function of ILRT interval.

Step 3 – Determine the change in flaw likelihood for an increase in inspection interval

The increase in the likelihood of a flaw due to age-related corrosion over the increase in time interval between tests is then determined from the results of Step 2.

Step 4 – Determine the likelihood of a breach in containment given a flaw

For there to be a significant leak from the containment, the flaw must lead to a gross breach of the containment. The likelihood of this occurring is determined as a function of pressure and evaluated at the KNPP ILRT pressure.

Step 5 – Determine the likelihood of failure to detect a flaw by visual inspection

The likelihood that the visual inspection will fail to detect a flaw will be determined considering the portion of the containment that is uninspectable at KNPP as well as an inspection failure probability.

Step 6 – Determine the likelihood of non-detected containment leakage due to the increase in test interval

The likelihood that the increase in test interval will lead to a containment leak not detected by visual examination is then determined as the product of the increase in flaw likelihood due to the increased test interval (Step 3), the likelihood of a breach in containment (Step 4) and the visual inspection non-detection likelihood (Step 5). The results of the above for the two regions of the containment are then added to get the total increased likelihood of non-detected containment leakage due to age-related corrosion resulting from the increase in ILRT interval.

The result of Step 6 is then used, along with the results of the prior risk analysis in the body of this analysis to determine the increase in LERF as well as the increase in person-rem/year and conditional containment failure probability due to age-related liner corrosion.

A.7.0 INPUT INFORMATION

1. General methodology and generic results from the Calvert Cliffs assessment of age-related liner degradation (Reference A1).

2. The KNPP ILRT test pressure of 45.4 to 46.0 psig (Reference A5).

3. KNPP containment failure pressure of 137 psia (Reference A6). This is a conservatively low value corresponding to a high confidence of a low probability of failure.

4. Fraction of containment shell that cannot be inspected for Appendix J, ASME Section XI of 0.20 (Reference A7).

5. Fraction of containment shell that is potentially in contact with foreign material either imbedded in the adjacent concrete or trapped in areas of limited access of 20% (Reference A7).

6. The number of containments, either free-standing steel shell or concrete with steel liners, is equal to the number of operating nuclear plants or 104 (Reference A11)

A.8.0 REFERENCES

1. "Calvert Cliffs Nuclear Power Plant Unit No. 1; Docket No. 50-317, Response to Request for Additional Information Concerning the License Amendment Request for a One-time Integrated Leakage Rate Test Extension," Constellation Nuclear letter to USNRC, March 27, 2002.

2. "Comanche Peak Steam Electric Station (CPSES), Docket Nos. 50-445 and 50-446, Response to Request for Additional Information Regarding License Amendment Request (LAR) 01-14 Revision to Technical Specification (TS) 5.5.16 Containment Leakage Rate Testing Program," TXU Energy letter to USNRC, June 12, 2002.

3. "Donald C. Cook Nuclear Plant Units 1 and 2, Response to Nuclear Regulatory Commission Request for Additional Information Regarding the License Amendment Request for a One-time Extension of Integrated Leakage Rate Test Interval," Indiana Michigan Power Company, November 11, 2002.

4. "St. Lucie Units 1 and 2, Docket Nos. 50-335 and 50-389, Proposed License Amendments, Request for Additional Information Response on Risk-Informed One Time Increase in Integrated Leak Rate Test Surveillance Interval,' Florida Power & Light Company letter to USNRC, December 13, 2003.

5. "Containment Building Integrated Leak Rate Test", SP 56A-088, Rev. F, KNPP.

6. Edward Coen, "Section 6.0 Level 2 Source Term And Sensitivity Analysis," KSEC6.doc, KNPP, transmitted by E-mail 2/12/03

7. S. E. Phillippi, "Calculation of Inspectable And Uninspectable Containment Vessel Surface Areas," SCIENTECH, INC. Analysis File 17547-0001-A2, Rev. 0 March 24, 2003

8. "Containment Liner Through Wall Defect due to Corrosion," Licensee Event Report, LER-NA2-99-02, North Anna Nuclear Power Station Unit 2.

9. "Brunswick Steam Electric Plant, Units 1 and 2, Dockets 50-325 and 50-324/License Nos. DPR-71 and DPR-62, Response to Request for Additional Information Regarding Request for License Amendments – Frequency of Performance Based Leakage Rate Testing," CP&L letter to USNRC, February 5, 2002.

10. "IE Information Notice No. 86-99: Degradation Of Steel Containments," USNRC, December 8, 1986.

11. "Number of Operating Power Reactors in the United States, 1973-2002," *Nuclear News*, March 2003, p.57

A.9.0 MAJOR ASSUMPTIONS:

1. As indicated in the NRC's RAIs (References A3 and A4, for example) there have been 4 instances of age-related corrosion leading to holes in steel containment liners or shells. Three of these instances (Cook - Reference A3, North Anna – Reference A8 and Brunswick – Reference A9) were in concrete containments with steel liners and due to foreign material imbedded in the concrete in contact with the steel liner. The fourth instance (Oyster Creek – Reference A10) was in a free standing steel containment and occurred in an area where sand fills the gap between the steel shell and the surrounding concrete and was attributed to water accumulating in this sand. This data is therefore considered to represent a corrosion induced failure rate only for the areas of the KNPP in contact with concrete or other areas where foreign material may be trapped. For the other areas where the containment steel shell is not likely to be in contact with foreign material, the corrosion induced failure rate is substantially lower and taken to be negligible.

2. The historical data of age-related corrosion leading to holes in the steel containment has occurred primarily (3 out of 4 instances) for steel lined concrete containments. For these containments the surface area in contact with the concrete comprises essentially the entire surface area of the containment. As indicated in Reference A7, this is true for only 20% of the KNPP containment surface area. Since the greater the surface area in contact with the concrete, the greater the chance of foreign material being in contact with steel containment and therefore the greater the chance of corrosion induced flaws, the containment failure rate due to corrosion will be taken to be proportional to the surface area in contact with the concrete. For KNPP, which has a smaller containment volume and surface area than the large dry containments, where the failures have primarily occurred, the corrosion induced flaw rate will conservatively be taken to be the historical values based on the 4 data points times the fraction of the surface area in contact with concrete.

3. The visual inspection data are conservatively limited to 5.5 years reflecting the time from September 1996, when 10 CFR 50.55a started requiring visual inspection, through March 2002, the cutoff date for this analysis. Additional success data were not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to September 1996 (and after March 2002) and there is no evidence that liner corrosion issues were identified. (Step 1)

4. As in Reference A1, the containment flaw likelihood is assumed to double every 5 years. This is included to address the increased likelihood of corrosion due to aging. (Step 2)

5. The likelihood of a significant breach in the containment due to a corrosion induced localized flaw is a function of containment pressure. At low pressures, a breach is very unlikely. Near the nominal failure point, a breach is expected. As in Reference A1, anchor points of 0.1% chance of cracking near the flaw at 20 psia and 100% chance at the failure pressure (137 psia for KNPP from Reference A6) are assumed with logarithmic interpolation between these two points. (Step 4)

6. In general, the likelihood of a breach in the lower head region of the containment occurring, and this breach leading to a large release to the atmosphere, is less then that for the cylindrical portion of the containment. The assumption discussed in item 5 above is, however, conservatively applied to the lower head region of the containment, as well as to the cylindrical portions.

7. All non-detected containment overpressure leakage events are assumed to be large early releases.

8. The interval between ILRTs at the original frequency of 3 tests in 10 years is taken to be 3 years.

A.10.0 IDENTIFICATION OF COMPUTER CODES

None used.

A.11.0 DETAILED ANALYSIS:

A.11.1 *Step 1 – Determine a corrosion-related flaw likelihood*

As discussed in Assumptions 1, 2 and 3, the likelihood of through wall defects due to corrosion for the areas of the containment potentially contacted by foreign material is based on 4 data points in 5.5 years.

> 4 failures * 0.20 KNPP relative area subject to corrosion/ (104 plants * 5.5 years/plant) = 1.40E-03 per year

For the areas of the containment where foreign material is not likely to contact the containment the defect likelihood is taken to be very much smaller and is therefore neglected.

A.11.2 *Step 2 – Determine an age-adjusted liner flaw likelihood*

Reference A1 provides the impact of the assumption that the historical flaw likelihood will double every 5 years on the yearly, cumulative and average likelihood that an age-related flaw will occur. For a flaw likelihood of 5.2E-03 per year, the 15 year average flaw likelihood is 6.27E-03 per year for the cylinder/dome region. This result of Reference A1 is generic in nature, as it does not depend on any plant specific inputs except the assumed historical flaw likelihood.

For the present assumption of 4 historical failures in 104 plants, the 15 year average flaw likelihood is 26.9% (1.40E-03/5.2E-03 = 0.269 or 26.9%) of the above value or 1.69E-03 per year, and in accordance with Assumption 1, is applicable to only the region of the containment potentially in contact with foreign material.

A.11.3 *Step 3 – Determine the change in flaw likelihood for an increase in inspection interval*

The increase in the likelihood of a flaw due to age-related corrosion over the increase in time interval between tests from 3 to 15 years is determined from the result of Step 2 in Reference A1 to be 8.7% for the cylinder/dome region based on assumed historical flaw likelihood and the resulting 6.27E-03 per year 15 year average flaw likelihood. This result of Reference A1 is generic in nature, as it does not depend on any plant specific inputs except the assumed historical flaw likelihood.

For the present assumption of 4 historical failures in 104 plants, the increase in the likelihood of a flaw due to age-related corrosion over the increase in time interval between tests from 3 to 15 years is 26.9% (as in Step 2) of that given in Reference A1 or 2.34% and in accordance with Assumption 1 is applicable to only the region of the containment potentially in contact with foreign material

A.11.4 *Step 4 – Determine the likelihood of a breach in containment given a liner flaw*

The likelihood of a breach in containment occurring is determined as a function of pressure as follows.

For a logarithmic interpolation on likelihood of breach

$$\text{Log (likelihood of breach)} = m\,(\text{pressure}) + a$$

Where: m = slope
a = intercept

The values of m and a are determined from solution of the two equations for the values of 0.1% at 20 psia and 100% at containment failure pressure of 137 psia (Reference A6),

$$\text{Log } 0.1 = m * 20 + a$$

$$\text{Log } 100 = m * 137 + a$$

or

$$m = (\text{Log } 100 - \text{Log } 0.1) / (137 - 20) = 0.02564$$

and

$$a = \text{Log } 0.1 - 0.02564 * 20 = -1.5128$$

The upper end of the range of KNPP ILRT pressures of 46.0 psig (Reference A5) gives the highest likelihood of breach.

At 60.7 psia (46.0 + 14.7), the above equation gives

$$\text{Log (likelihood of breach)} = 0.02564 * 60.7 - 1.5128 = 0.04355$$

$$\text{Likelihood of breach} = 10^{0.04355} = 1.11\%$$

In accordance with Reference A1, the above value is for the cylinder/dome portions of the containment. For this analysis, this value is assumed applicable to the region of the containment potentially in contact with foreign material.

A.11.5 *Step 5 – Determine the likelihood of failure to detect a flaw by visual inspection*

A review of the geometry of the containment shell and the relative areas that are not inspectable and those in potential contact with foreign material, indicates that these two areas are essentially the same, both comprising approximately 20% of the total surface area of the steel shell (Reference A7). Consequently, the portion of the containment not likely to be in contact with potential foreign material is 100% visually inspectable, while the portion that may be in contact with potential foreign material is not visually inspectable. While a 10% failure rate for that portion that is inspectable is assumed, this has no impact on the result since the corrosion-related flaw rate is considered negligible.

A.11.6 *Step 6 – Determine the likelihood of non-detected containment leakage due to the increase in test interval*

The likelihood of non-detected containment leakage in each region due to age-related corrosion of the liner considering the increase in ILRT interval is then given by

The increased likelihood of an undetected flaw because of the increased ILRT Interval (Step 3)	*	The likelihood of a containment breach given a liner flaw (Step 4)	*	The likelihood that visual inspection will not detect the flaw (Step 5)

= 0.0% * 0.0111 * 0.10 = 0.0% for the regions not potentially contacted by foreign material

= 2.34% * 0.0111 * 1.0 = 0.0260% for the regions potentially contacted by foreign material.

The total is then the sum of the values for the two regions or

Total Likelihood of Non-Detected Containment Leakage = 0.0% + 0.026% = 0.026%

for the ILRT interval increase from 3 years to 15 years.

A.11.7 Impact on Risk

The above indicates that there is a very small likelihood that corrosion will lead to undetected containment leakage over the increase in ILRT interval from 3 to 15 years. If it is assumed that this leakage is sufficient to lead to a large release and therefore could contribute to the Large Early Release Frequency (LERF), the above percent increase would be applied to the portion of the core damage frequency (CDF) whose release may be impacted by the leakage and could contribute to the LERF. Note that this is identified in the CCNPP submittal of Reference A1 as "The non-large early release frequency (LERF) containment over-pressurization failures...".

From the body of this analysis (PCDF_TypeA in Section 11.1) this value is 2.664E-05 per year. The resulting increase in LERF is

Delta LERF due to age-related corrosion = 0.00026 * 2.664E-05 = 6.93E-09 per year

The total increase in LERF due to the increase in ILRT interval from 3 years (or the equivalent 3 in10 years) to 15 years is the value from Section 11.4 plus the above or

Total Delta LERF = 8.06E-08 + 6.93E-09 = 8.75E-08 per year

The person-rem/year impact of the above age-related corrosion can be estimated by assuming that the delta LERF due to age-related corrosion contributes to the EPRI containment failure Class 3b leakage. From Section 11.2 of the body of this analysis, the population exposure (50 mile person-rem) given an accident of this class is 4.20E+03 person-rem. The increase in person-rem/year due to the above assessment of age-related corrosion is therefore

4.20E+03 * 6.93E-09 = 2.9E-05 person-rem/year

This increase is very small compared to the increase estimated in Section 11.3 of the body of this analysis of 1.41E-03 person-rem/year for the increase in ILRT interval from that corresponding to 3 tests in 10 years to that corresponding to 1 test in 15 years. The total increase in population risk is

1.41E-03 + 2.9E-05 = 1.44E-03 person-rem/year

This corresponds to an increase of

(1.44E-03 / 5.804) * 100 = 0.0025%

of the baseline total risk.

The increase in containment leakage due to age-related liner corrosion will also lead to an increase in the conditional containment failure probability (CCFP) equal to the total likelihood of non-detected containment leakage as calculated above or 0.026% (or 0.00026). This added to the increase estimated in Section 11.5 of the body of this analysis of 0.0028 gives a total increase in CCFP of 0.0031 for the increase in ILRT interval from that corresponding to 3 tests in 10 years to 1 test in 15 years including the effect of corrosion.

All of the above analysis and results are for the impact of increasing the ILRT interval from that corresponding to 3 tests in 10 years to that corresponding to 1 test in 15 years. The impact in going from 1 in 10 years to 1 in 15 years may be estimated from the information in Table 6 of Reference A1. The delta between 1 in 10 and 1 in 15 years can be obtained from this table as 5.3% compared to the delta of 8.7% for the delta between 3 in 10 years (or the equivalent 1 in 3 years) and 1 in 15 years. The delta risk values for increasing the ILRT interval from 10 years to 15 years is then 61% (5.3/8.7) of the above values. This relative increase from Reference A1 is generic in nature and equally applicable to the present analysis.

A.12.0 COMPUTER INPUT AND OUTPUT

None

A.13.0 SUMMARY OF RESULTS

Table 1 below summarizes the major steps of the analysis and the results for the increase in LERF due to age-related corrosion of the containment liner for an ILRT interval increase from that corresponding to 3 tests in 10 years to that corresponding to 1 test in 15 years. The impact of these results on the major results of the ILRT extension analysis from the body of this analysis is provided in Table 2.

Table 1: Liner Corrosion Analysis Steps and Results

Step	Description	Regions Not Potentially Contacted by Foreign Material (57,657 ft^2 or 80% of total)		Regions Potentially Contacted by Foreign Material (13,773 ft^2 or 20% of total)	
1	**Historical Flaw Likelihood** Failure Data: Assumed to be applicable to only region susceptible to accelerated corrosion Success Data: Based on 104 steel-lined or steel shell containments and 5.5 years since the 10 CFR 50.55a requirements for periodic visual inspection of containment surfaces.	Events: none applicable to this region. Negligible		Events: 4 through wall corrosion-related flaws. (Brunswick 2, North Anna 2, Cook and Oyster Creek) 4 * 0.20 / (104 * 5.5) **= 1.40E-03/year**	
2	**Age-Adjusted Liner Flaw Likelihood** During 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for the 5th to 10th year set equal to the historical failure rate.	Year	Failure Rate	Year	Failure Rate
		1	negligible	1	5.7E-04
		avg. 5 – 10	negligible	avg. 5 – 10	1.3E-03
		15	negligible	15	3.8E-03
		15-year avg = negligible ~ 0.0		**15-year avg = 1.69E-03/year**	
3	**Increase in Flaw Likelihood Between 3 and 15 Years** Uses age-adjusted liner flaw likelihood (step 2).	**NA**		**2.34%**	
4	**Likelihood of Breach in Containment Given Liner Flaw** The upper end pressure is consistent with the KNPP PRA Level 2 analysis. 0.1% is assumed for the lower end. Intermediate failure likelihood's are determined through logarithmic interpolation. Region potentially in contact with foreign material assumed to be the same as for the cylinder/dome region.	Pressure (psia)	Likelihood of Breach	Pressure (psia)	Likelihood of Breach
		20 60.7 (ILRT) 80 137	0.10% 1.11% 3.45% 100%	20 60.7 (ILRT) 80 137	0.1% 1.11% 3.45% 100%
5	**Visual Inspection Detection Failure Likelihood**	**10%** Assumed 10% failure rate inspection.		**100%** Cannot be visually inspected.	
6	**Likelihood of Non-Detected Containment Leakage** (Setps 3*4*5)	**0.0%** (0.0% * 1.11% * 10%)		**0.0260%** (2.34% * 1.11% * 100%)	
	Total Likelihood of Non-Detected Containment Leakage Sum of contributions from cylinder/dome and basemat regions	**0.026%** (0.0% + 0.026%)			

Step	Description	Regions Not Potentially Contacted by Foreign Material (57,657 ft^2 or 80% of total)	Regions Potentially Contacted by Foreign Material (13,773 ft^2 or 20% of total)
	Delta LERF Due to Age-Related Corrosion Total likelihood of non-detected containment leakage times portion of the CDF that could lead to LERF and that would not otherwise always be a LERF.	**6.93E-09 per year** (0.000260 * 2.664E-05/yr)	

Table 2: Major Results

	Test Interval Extended	
	From 3 in 10 years to 1 in 15 years	From 1 in 10 years to 1 in 15 years
Total person-rem/year increase		
Without Corrosion (Section 13)	0.00141	0.00059
Including Corrosion	0.00144	0.00061
The percentage increase in person-rem/year risk		
Without Corrosion (Section 13)	0.024%	0.010%
Including Corrosion	0.025%	0.011%
Change in LERF (per year)		
Without Corrosion (Section 13)	8.1E-08	3.4E-08
Including Corrosion	8.8E-08	3.8E-08
Change in the Conditional Containment Failure Probability		
Without Corrosion (Section 13)	0.0028	0.0012
Including Corrosion	0.0031	0.0014

A.14.0 CONCLUSIONS

For the above results it is concluded that age-related containment corrosion has essentially a negligible impact on the risk associated with the extension of the Type A ILRT test frequency from 1 test in 10 years to 1 test in 15 years as well the extension from a frequency of 3 tests in 10 years to 1 test in 15 years.

Age-related corrosion increases the LERF due to the change in the Type A ILRT interval from that corresponding to 1 test in 10 years to that corresponding to 1 test in 15 years from 3.4E-08/yr to

3.8-08/yr and that due to a change in interval from that corresponding to 3 tests in 10 years to that corresponding to 1 test in 15 years from 8.1E-08/yr to 8.8E-08/yr. Based on the guidance in Reg. Guide 1.174, the change in LERF for the requested change in Type A ILRT interval from the current 1 test in 10 years to 1 test in 15 years represents a very small change in LERF and is non-risk significant.