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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

Donald C. Cook Nuclear Plant Unit 1 and Unit 2
RESPONSE TO NUCLEAR REGULATORY COMMISSION REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST FOR ONE-TIME EXTENSION OF CONTAINMENT INTEGRATED LEAKAGE RATE TEST INTERVAL
(TAC Nos. MB4837 and MB4838)

Reference:

1. Letter from J. E Pollock, Indiana Michigan Power Company (I&M), to U. S. Nuclear Regulatory Commission (NRC) Document Control Desk, "License Amendment Request for One-time Extension of Containment Integrated Leakage Rate Test Interval" AEP:NRC:2612, dated April 11, 2002

2. Letter from J. F. Stang, NRC, to A. C. Bakken III, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Request for Additional Information Regarding License Amendment Request, 'One Time Extension Of Integrated Containment Leak Rate Test Interval,' dated Aprill 11, 2002 (TAC Nos. MB4837 and MB4838)," dated October 31, 2002

This letter provides Indiana Michigan Power Company's (I&M) response to a Nuclear Regulatory Commission (NRC) request for additional information regarding a proposed license amendment for a one-time extension of the containment integrated leakage rate test (ILRT) interval for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2.

By Reference 1, I&M proposed to amend Facility Operating Licenses DPR-58 and DPR-74 for CNP Unit 1 and Unit 2 to allow a one-time extension of the interval between ILRTs from 10 to 15 years. By Reference 2, the NRC staff

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requested additional information regarding the proposed license amendment. Attachment 1 to this letter provides I&M's response to the request for additional information. Attachments 2, 3, and 4 provide additional details pertaining to portions of the response provided in Attachment 1. Attachment 2 provides a summary of the Unit 1 and Unit 2 containment concrete examinations performed in accordance with Subsection IWL of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Attachment 3 provides a new risk assessment for extending the ILRT interval. Attachment 4 provides an assessment of the effect of age-related containment liner degradation on the new risk assessment for the proposed extension of the ILRT interval. There are no new regulatory commitments made in this letter.

The information provided in this letter consists of supporting information for the license amendment request submitted by Reference 1. The information provided in this letter does not alter the license amendment requested by Reference 1. Additionally, the information provided in this letter does not affect the validity of the evaluation of significant hazards considerations or environmental assessments that were performed in accordance with 10 CFR 50.92 and 10 CFR 51.21, and documented in Enclosure 2 of Reference 1.

Should you have any questions, please contact Mr. Brian A. McIntyre, Manager of Regulatory Affairs, at (269) 697-5806.

Sincerely,

J. E. Pollock
Site Vice President

JRW/jen

Attachments:

1. Response to Nuclear Regulatory Commission Request for Additional Information Regarding License Amendment Request for One-Time Extension of Containment Integrated Leakage Rate Test Interval
2. Summary of ASME Subsection IWL Containment Concrete Examinations
3. Risk Impact Assessment for Extending Containment Type A Test Interval
4. Effect of Age-Related Degradation on Risk Impact Assessment for Extending Containment Type A Test Interval

c: K. D. Curry, Ft. Wayne AEP
J. E. Dyer, NRC Region III
MDEQ - DW & RPD
NRC Resident Inspector
J. F. Stang, Jr., NRC Washington, DC
R. Whale, MPSC

AFFIRMATION

I, Joseph E. Pollock, being duly sworn, state that I am Site Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

J E Pollock

J. E. Pollock
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 11 DAY OF November, 2002

Jennifer L Kernosky
Notary Public

My Commission Expires 5/26/05

JENNIFER L KERNOSKY
Notary Public, Berrien County, Michigan
My Commission Expires May 26, 2005

bc: A. C. Bakken III
M. J. Finissi
S. A. Greenlee
D. W. Jenkins, w/o attachments
D. R. Hafer/J. T. Hawley
J. A. Kobyra, w/o attachments
B. A. McIntyre, w/o attachments
J. E. Newmiller
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M. K. Scarpello, w/o attachments
T. K. Woods, w/o attachments

ATTACHMENT 1 TO AEP:NRC:2612-01

RESPONSE TO NUCLEAR REGULATORY COMMISSION REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST FOR ONE-TIME EXTENSION OF CONTAINMENT INTEGRATED LEAKAGE RATE TEST INTERVAL

The documents referenced below are identified at the end of this attachment.

This attachment provides Indiana Michigan Power Company's (I&M) response to a Nuclear Regulatory Commission (NRC) request for additional information regarding a proposed license amendment for a one-time extension of the containment integrated leakage rate test interval for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2. The proposed amendment was transmitted to the NRC by Reference 1. The NRC request for additional information was transmitted to I&M by Reference 2.

NRC Question 1

Based on the review of the proposed amendment, the NRC staff understands that you are using the 1992 Edition and the 1992 Addenda of Subsections IWE and IWL of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. Reference 1 also describes the findings of corrosion and a thru-wall hole in the liner plate of Unit 2 containment. In addressing "containment inspection history", you indicate that there are no areas that require augmented examination. Please provide justification for not identifying the areas of the degraded liner plates and penetrations (accepted by engineering evaluation), and other suspect areas not requiring additional examination (as per IWE-2430), or augmented examination (as per IWE-1240) during the subsequent inspection periods.

Response to NRC Question 1

The requirements of IWE-2430 and IWE-1240 and their applicability to the through-wall hole and corrosion findings are described below.

Requirements of IWE-2430 and IWE-1240

IWE-2430 requires that examinations performed during any one inspection that reveal flaws or areas of degradation exceeding the specified acceptance standards be extended to include an additional number of examinations within the same category approximately equal to the initial number of examinations during the inspection. IWE-1240 requires augmented examination of surface areas likely to experience accelerated degradation and aging.

Inadequate Repair of Through-Wall Hole

As described in the technical analysis supporting the proposed amendment (Reference 1), the 3/16-inch diameter through-wall hole in the Unit 2 containment liner plate was the result of an inadequate repair of a hole drilled in error during plant construction. Following discovery of the inadequate repair in 1999, the repair material was dislodged, revealing a through-wall hole. The section of liner plate containing the hole was removed and a shallow hole of approximately the same diameter was discovered in the concrete behind the liner hole. The shallow hole in the concrete occurred when a drill bit penetrated the liner and continued into the concrete for a short distance. This provides evidence that the liner hole was man-made and not the result of degradation or aging, or contact with the wooden handle of a wire brush found embedded in the concrete near the hole. The affected section of liner plate was restored to an acceptable design configuration. The repair was vacuum box tested and subjected to a local leakage rate test (LLRT). The through-wall hole in the liner plate did not invoke the requirements for additional examination per IWE-2430 since all accessible areas of the containment liner were already being examined. The through-wall hole in the liner plate did not invoke the requirements for augmented examination per IWE-1240 since the hole was not the result of degradation or aging.

Liner Corrosion in the Area of the Through-Wall Hole

As noted in the technical analysis supporting the proposed amendment, some localized corrosion was identified on the concrete side of the liner in the vicinity of the through-wall hole. Ultrasonic readings were taken every 1/2-inch at 45-degree increments around the hole for a 6-inch radius. These readings identified an area approximately 3.5 inches in diameter in which the liner thickness was less than the nominal value for this location, 0.375 inches. This area was generally centered on the exposed portion of the wire brush handle rather than on the hole. With one exception, the minimum liner thickness found in this area was 0.303 inches. The exception consisted of a small corroded area on the concrete side of the liner approximately 0.5 inches below the hole. The minimum liner thickness measured in this location was 0.187 inches. Because this location was outside the contact area with the wire brush and directly below the through-wall hole, I&M considers that the likely cause of this corrosion was water intrusion through the inadequate hole repair during pressure washing.

The corrective action for the through-wall hole included removal of the wire brush and wooden handle to the maximum extent practical without cutting any stiffener steel, performing a repair of the concrete, and replacing the liner section that contained the hole. Therefore, augmented examination per IWE-1240 is not required because the area is no longer likely to experience accelerated degradation or aging since there is no wood in contact with the liner and the inadequate repair/hole has been eliminated. Since the liner served as the inner form during pouring of the concrete for the cylindrical portion of the containment walls, any water intrusion through the inadequate repair/hole would have remained localized. The additional examination requirements of IWE-2430 did not apply, since all accessible areas of the containment liner were already being examined in accordance with the IWE program.

Liner Corrosion in the Area of the Moisture Barrier Seal

The technical analysis supporting the proposed amendment also described liner corrosion that had been discovered in 1998 in the area behind the moisture barrier seal on both Unit 1 and Unit 2 at elevation 598 feet, 9-3/8 inches. As stated above, the nominal liner thickness at this location is 0.375 inches. The thickness of the remaining sound metal at several corrosion pits was less than 0.250 inches, the minimum thickness specified by I&M for acceptance of inspection results without a detailed engineering analysis. There were 61 locations in Unit 1 where the depth of the corrosion pits exceeded 0.125 inches, with pit depth ranging from 0.141 inches to a maximum of 0.172 inches at four locations. The distribution of the corrosion and the level of pitting were not uniform around the perimeter of the moisture barrier seal. The degree of corrosion decreased with depth below the moisture barrier. The corrosion was classified as general corrosion resulting from a failure of the seal to prevent moisture intrusion. Similar but less extensive corrosion was found behind the Unit 2 moisture barrier seal. There were two locations in Unit 2 where the depth of the corrosion pits exceeded 0.125 inches, with the deepest having a depth of 0.141 inches.

The area behind the moisture barrier seal is normally inaccessible. In accordance with IWE-1220(b), the area is exempt from the examination requirements of IWE-2000, including the additional examination requirements of IWE-2430. The corrective action for the liner corrosion included modifying the seal design to prevent moisture intrusion and re-coating the affected area. Therefore, augmented examination per IWE-1240 is not required because the area is no longer likely to experience accelerated degradation or aging. Additionally, IWE-1240 does not apply to inaccessible areas.

Although not required by IWE-2430 or IWE-1240, I&M has performed supplemental inspections of portions of the area behind the moisture barrier seal to verify the effectiveness of the new design. Sections of the redesigned moisture barrier seal were removed in both Unit 1 and Unit 2 approximately three years after installation, and a visual examination was performed on the liner area where corrosion was previously identified. The visual examination found no moisture intrusion and no active corrosion. Continued monitoring of the moisture barrier seal is performed with the scheduled VT-3 visual examination of the moisture barrier seal area each inspection period (3 inspections are required in a 10-year interval).

Additional Information

The NRC staff has also requested information as to the basis for I&M's conclusion that compliance with the requirements of 10 CFR 50.55a has been maintained regarding the as-left condition of the corroded areas behind the moisture barrier seal.

Regulation 10 CFR 50.55a(g)(4)(v)(B) requires that metallic shell and penetration liners which are pressure retaining components and their integral attachments in concrete containments meet the inservice inspection, repair, and replacement requirements applicable to components which are classified as American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class MC. Subsection IWE of the ASME Code contains the requirements for Class MC and metallic liners of Class CC components of light-water cooled plants. The acceptance standard for coated and non-coated areas are given in IWE-3510.2 and IWE-3510.3, which state that areas which are suspect must be accepted by engineering evaluation or corrected by repair/replacement activities.

Consistent with these requirements, the area of liner corrosion behind the moisture barrier seal was accepted by an engineering evaluation. That evaluation demonstrated that the liner can fulfill its leak prevention function with a wall thickness as low as 0.0625 inches. The evaluation was reviewed by NRC inspection personnel as documented in Section E8.2 of an NRC Inspection Report dated January 19, 2000 (Reference 3). In that inspection report, the engineering evaluation was described as comprehensive and thorough, and it was noted that the evaluation demonstrated the existence of a substantial safety margin. Based on the actions taken by I&M, the inspectors closed the associated Case Specific Checklist item established through NRC Manual Chapter 0350, "Staff Guidelines for Restart Approval."

NRC Question 1 (Redesignated as Question 2 by I&M)

Please provide the following information related to the finding of the through-wall hole in the Unit 2 liner plate:

NRC Question 2.a

Please provide location (elevation, azimuth), liner thickness, nearness to discontinuity areas (i.e. areas that would be subjected to bending under the postulated loadings, or thickness transition), size of the opening made to remove the wire brush, and corrective actions taken to ensure the integrity of the liner plate.

Response to NRC Question 2.a

As noted in the response to Question 1, the nominal liner thickness in this area is 0.375 inches. This thickness is based on constructability considerations and not on the structural design requirements for containment integrity. The through-wall hole was located at elevation 602 feet, 3 7/8 inches, and azimuth 112 degrees. As described above in the response to Question 1, the minimum measured liner plate thickness was 0.303 inches, except for a small area with a thickness of 0.187 inches, located approximately 0.5 inches directly below the hole. The nearest feature that may be considered to be a discontinuity was a stiffener weld that was approximately one inch away from the hole. The only liner function that is credited in accident analyses is that of a leakage barrier. The liner is not assumed to resist any bending load since it is not credited in the structural analysis of the containment. A section of liner approximately 4.75 inches by

4.375 inches was cut out to expose the concrete behind it. The wire brush and wooden handle were removed to the maximum extent practical, repairs were made to the concrete, and the section of liner plate was replaced. A vacuum test and an LLRT were performed on the repaired section.

NRC Question 2.b

You postulate that the through-liner hole was due to the inadequate repair of the liner hole drilled in error during construction. How did you verify that there are no such holes and repairs in other areas, and in the uninspectable areas of the containment liners in both Unit 1 and Unit 2?

Response to NRC Question 2.b

The inadequate repair of the liner hole drilled in error during construction was found during an inspection of 100 percent of the inspectable areas. No other such holes and repairs were found. Additionally, I&M inspects portions of areas that are classified as uninspectable when those portions become accessible during non-routine or special maintenance activities. These inspections have included:

- Portions of the containment liner behind the ice condenser that became accessible during repairs to the ice condenser end wall divider barrier seal.
- Portions of the containment liner behind other sections of the divider barrier seal when these areas were accessible during maintenance.
- Portions of the containment liner behind the ice condenser that became accessible when the ice condenser top deck vent curtain was removed.
- Portions of the containment liner behind the ice condenser that were visible when the three access port covers were removed.
- Portions of the containment liner behind and below the moisture barrier seal when all or part of that seal has been removed as described in the response to Question 1.

None of these inspections have identified similar holes and repairs, providing reasonable assurance that the condition was an isolated occurrence.

NRC Question 2.c

Investigation of other incidents of such through-wall hole in liner plates indicated the cause to be corrosion induced by the foreign elements stuck in the containment concrete. It appears that the Unit 2 through-wall hole in the liner was due to similar reason. In the 1992 integrated leak rate test (ILRT), the corrosion had not propagated to the extent that the ILRT would fail. However, if the ILRT were performed prior to this finding, the containment leakage rate could *have been unacceptable. Please provide specific discussion of this potential for each unit at D. C. Cook.*

Response to NRC Question 2.c

As described in the response to Question 1, the through-wall hole in Unit 2 was not the result of corrosion induced by foreign elements in the concrete. The hole was clearly the result of intentional drilling.

Even if the condition had gone undetected, it is unlikely that it would have caused unacceptable integrated leakage rate test (ILRT) results. The repair, although improper, had apparently remained in place since plant construction, including several acceptable ILRTs. Additionally, since concrete for the cylindrical portion of the containment was poured using the steel liner as the form for the inside wall, the concrete against the outside liner surface would provide a significant additional barrier for mitigating leakage even if the temporary repair material was completely dislodged.

As described above in the response to Question 1, liner thickness measurements in the area were all greater than or equal to 0.303 inches with one location immediately below the hole measuring 0.187 inches. As also described in the response to Question 1, an engineering evaluation has demonstrated that the liner can fulfill its leak prevention function with a wall thickness as low as 0.0625 inches.

Therefore, there is reasonable assurance that neither the improper hole repair or the local corrosion would have resulted in an unacceptable containment leakage rate even if the conditions were undetected. The preceding discussion would also apply if the condition had occurred in the Unit 1 containment liner.

NRC Question 2.d

Recognizing the discussion in "b," and "c" above, please provide justification for not performing ILRT after the through-liner hole finding, or in accordance with the present technical specification requirement.

Response to NRC Question 2.d

The response to Question 2.b provides the basis for I&Ms conclusion that the hole and inadequate repair was an isolated occurrence. The response to Question 2.c provides the basis for I&M's conclusion that the hole and inadequate repair would not have resulted in an unacceptable ILRT leakage rate. Neither of these conclusions indicated the need to perform an ILRT after finding the hole and inadequate repair.

The section of the CNP Technical Specifications that governs ILRTs is Section 4.6.1.2. This section requires that leakage rate testing be conducted in accordance with 10 CFR 50, Appendix J, Option B, and Regulatory Guide (RG) 1.163 (Reference 4). Section C of RG 1.163 states that NEI 94-01 (Reference 5) provides methods acceptable to the NRC staff for complying with the provisions of 10 CFR 50, Appendix J, Option B. Section 9.2.4 of NEI 94-01 states that repairs affecting the containment leakage integrity require an ILRT or an LLRT. In accordance with these requirements, the repair of the hole was tested by an LLRT that demonstrated no leakage.

NRC Question 3

Please provide a summary of findings of the examination of containment concrete performed in accordance with 10 CFR 50.55a and Subsection IWL including the acceptance criteria used for accepting concrete and reinforcing bar degradation.

Response to NRC Question 3

The requested summary is provided in Attachment 2 to this letter.

NRC Question 4

Inspections of some reinforced and steel containments (e.g., North Anna, Brunswick, D. C. Cook, Oyster Creek) have indicated degradation from the uninspectable side of the liner/steel shell of primary containments. The major uninspectable areas of the ice condenser containment include those behind the ice baskets and part of the shell (liner) embedded in the basemat. Please provide information as to how potential leakage due to age related degradation from these uninspectable areas are factored into the risk assessment in support of the requested ILRT interval extension.

Response to NRC Question 4

The amendment request included an assessment of the risk resulting from the proposed extension of the ILRT interval. In a telephone conference conducted on October 1, 2002, the NRC requested that I&M's response to Question 4 include information regarding the quality of the plant risk analysis that was used as the basis for that risk assessment. This information is provided below, followed by the information requested by Question 4.

Quality of Plant Risk Analysis

The risk assessment for extending the ILRT interval that was submitted with the amendment request was based on Revision 1 of the CNP Individual Plant Examination (IPE), which was transmitted to the NRC by Reference 6. I&M considers that use of Revision 1 of the CNP IPE to support the proposed amendment was appropriate for the following reasons:

- Revision 1 of the IPE was the most recent I&M approved plant risk analysis that was available at the time the amendment request was prepared.
- Revision 1 of the IPE provided resolution of NRC questions regarding the original IPE. Following receipt of Revision 1 of the IPE, the NRC stated, in Reference 7, that it did not intend to review Revision 1 of the IPE based on the acceptability of the original IPE in meeting the intent of Generic Letter 88-20 (Reference 8).
- Amendments similar to that requested by I&M have been approved for Crystal River Nuclear Plant (Reference 9) and Indian Point 3 Nuclear Power Plant (Reference 10) based on their respective IPEs.

Subsequent to preparation of the original CNP amendment request, I&M approved updated Level 1 and Level 2 Probabilistic Risk Assessment (PRA) models. The updated Level 1 model included a large early release frequency (LERF) model. The LERF model was integrated into the updated Level 1 PRA model to provide rapid solutions for Maintenance Rule evaluations pursuant to 10 CFR 50.65(a)(4). The Level 2 model was revised to obtain an updated risk profile for use in plant life extension applications. Both the LERF and Level 2 models were developed based on NUREG/CR-6595 (Reference 11), and extend the core damage sequences from the updated Level 1 model. The LERF and Level 2 models were developed with the same approach as the updated Level 1 model. A description of the quality and audits of the updated Level 1 PRA model is provided in Reference 12. As indicated in Reference 12, a Westinghouse Owners Group Peer Certification identified no significant (i.e., Level A) Facts and Observations related to the LERF and Level 2 models.

To assure that the proposed amendment is supported by the highest quality risk analysis available, I&M has performed a new assessment of risk resulting from the proposed extension of the ILRT interval. The new assessment uses the updated Level 1 and Level 2 PRA models described above. Consistent with the original assessment, the new assessment determines the risk from changing the containment ILRT frequency from once per 10 years to once per 15 years, and from three times per 10 years to once per 15 years. Also consistent with the original assessment, the new assessment:

- Follows NEI 94-01 guidelines and the NRC guidance in RG 1.174 (Reference 13).
- Uses methodology similar to that presented in EPRI TR-104285 (Reference 14) and NUREG-1493 (Reference 15).
- Incorporates revised guidance and additional information from EPRI (Reference 16) and NEI (Reference 17).

Similar to the original assessment, the new assessment uses a simplified bounding analysis approach to evaluate the change in risk associated with increasing the ILRT interval by examining the updated Level 1 and Level 2 PRA plant-specific accident sequences in which the containment remains intact or the containment is impaired. In addition to using the updated Level 1 and Level 2 PRA models, the new assessment includes the following significant differences from the original assessment:

- Justification is provided for not including undetected flaws in the basemat liner in LERF considerations. The treatment of such flaws has not changed from the original assessment. The new assessment includes justification for the treatment.
- The determination of the Class 3a and Class 3b core damage frequencies includes consideration of the probability of failure to detect a liner flaw during the visual inspection of the containment liner. There is some likelihood that the undetected flaw in the containment liner estimated as part of the Class 3a and Class 3b frequencies would be detected as part of the IWE visual examination process of the containment liner. The Unit 1 and Unit 2 containments were visually inspected between 1996 and 2000. Additional visual inspection of both units is planned for their respective refueling outages in 2003. Approximately 58.5 percent of the inner containment liner can be visually inspected. In the new assessment, it is assumed that the visual inspections are 90 percent effective in detecting large flaws in the visible regions of the containment (5 percent for failure to detect and 5 percent for flaw not detectable (not-through-wall)).
- The determination of the Class 3b core damage frequency (CDF) considers sequence progression and excludes the following sequences:

  - Sequences that progress to a large, early release end-state irrespective of Type A containment leakage,
  - Sequences in which there is substantial release mitigation due to operating systems, or,
  - Sequences in which there is significant warning time before release.

The new assessment is provided as Attachment 3 to this letter. The results are summarized below under "Summary of New Assessment Results."

In addition to performing a new assessment of the risk of extending the ILRT interval, I&M also determined values for total LERF. These values were used in evaluating the changes in LERF as determined by the new risk assessment. The total LERF was determined by estimating the contributions due to both internal events and external events (seismic, fire and flood). Based on the CNP Individual Plant Examination of External Events, the CDF values for these events are as follows:

Seismic CDF	3.17×10^{-6} /year
Fire CDF	3.76×10^{-6} /year
Flood-induced CDF	2.17×10^{-7} /year
Total External Events CDF	7.15×10^{-6} /year

Note that the Seismic CDF includes a "direct damage" contribution which involves early containment failure due to the seismic event. The portion of the Seismic CDF due to direct damage is 0.61×10^{-6} /year.

The external event models were not developed beyond CDF, so LERF information is not readily available. However, the effects of these initiators are typically similar to other trip initiators, such as loss of component cooling water, loss of offsite power, etc. Consequently, the updated Level 1 PRA model results were reviewed to determine the LERF-to-CDF ratios for the various internal initiating events. These LERF-to-CDF ratios were determined for each unit using the following equation:

LERF-to-CDF ratios = (Internal Event's LERF percentage x Internal Event's Total LERF) / (Internal Event's CDF percentage x Internal Event's Total CDF)

The LERF-to-CDF ratios ranged from the highest value of approximately 88 percent for interfacing systems loss of coolant accident (ISL), to the second-highest value of approximately 18 percent for dual-unit loss of offsite power events, to the lowest value of approximately 4 percent for loss of component cooling water sequences. Using the appropriate LERF-to-CDF ratios for the various types of external events sequences results in a best-estimate value for external event LERF of 1.14×10^{-6} /year. Given the internal events LERF value of 5.59×10^{-6} /year, a best-estimate total LERF value is 6.73×10^{-6} /year.

I&M also considered how more conservative assumptions regarding LERF-to-CDF ratios would affect the value for the external event LERF estimate. For external event initiators that do not contribute to the ISL category, a bounding estimate for the LERF-to-CDF ratio for the other initiators is about 20 percent. However, all of the direct damage events should be assumed to contribute to LERF for a bounding case. Using these bounding LERF-to-CDF ratios and the above CDF values yield:

Seismic CDF	$(3.17 \times 10^{-6} - 0.61 \times 10^{-6}) \times 0.2$/year =	5.12×10^{-7} /year
Direct Damage Seismic CDF	$0.61 \times 10^{-6} \times 1.0$/year =	6.10×10^{-7} /year
Fire CDF	$3.76 \times 10^{-6} \times 0.2$/year =	7.52×10^{-7} /year
Flood-induced CDF	$2.17 \times 10^{-7} \times 0.2$/year =	4.34×10^{-8} /year
Internal Events LERF		5.59×10^{-6} /year
Total LERF		7.51×10^{-6} /year

As described below, the values for total LERF were used to evaluate the results of the change in LERF resulting from the proposed increase in ILRT interval as determined by the new assessment.

Summary of New Assessment Results

The results of the new assessment are summarized below.

- The increase in the total integrated plant risk (person rem/year within 50 miles) resulting from reducing the ILRT frequency from 1 test per 10 years to 1 test per 15 years was calculated to be 0.012 percent. The increase in the total integrated plant risk resulting from reducing the ILRT frequency from 3 tests per 10 years to 1 test per 15 years was calculated to be 0.028 percent. These increases in risk are reasonable when compared to the range of increase, 0.02 to 0.14 percent, estimated in NUREG-1493 for reducing the ILRT frequency from 3 tests per 10 years to the 1 test per 10 years allowed by Option B to 10 CFR 50, Appendix J. NUREG-1493 indicates that such increases are "imperceptible."

- The increase in the LERF resulting from reducing the ILRT frequency from 1 test per 10 years to 1 test per 15 years was calculated to be 5.14 x 10^{-8}/year. The increase in the LERF resulting from reducing the ILRT frequency from 3 tests per 10 years to 1 test per 15 years was calculated to be 1.23 x 10^{-7}/year. RG 1.174 provides guidelines for acceptable changes in LERFs resulting from proposed changes to a plant's licensing basis. The LERF increase calculated for reducing the ILRT frequency from 1 test per 10 years to 1 test per 15 years is below the LERF acceptance guideline given in RG 1.174 of less than 10^{-7} for "very small" increases. The LERF increase calculated for reducing the ILRT frequency from 3 tests per 10 years to 1 test per 15 years is below the LERF acceptance guideline given in RG 1.174 of less than 10^{-6} for "small" increases. In accordance with RG 1.174, small LERF increases are acceptable for plants with a total LERF less than 10^{-5}. As described above, I&M determined values for total LERF using two methods. Both of these methods determined a total LERF of less than 10^{-5}.

  RG 1.174 also provides guidelines for acceptable changes in the CDF resulting from a proposed change to a plant's licensing basis. However, the CDF would not be affected by a change to the ILRT interval. Therefore, the change in LERF is the sole RG 1.174 numerical risk guideline applicable to this proposed change.

- The fractional increase in conditional containment failure probability from reducing the ILRT frequency from 1 test per 10 years to 1 test per 15 years was calculated to be 0.0011. The fractional increase in conditional containment failure probability from reducing the ILRT frequency from 3 tests per 10 years to 1 test per 15 years was calculated to be 0.0025. These small increases in conditional containment failure probability demonstrate that, consistent with the defense-in-depth guidelines provided in RG 1.174, the proposed amendment would not significantly affect the balance among prevention of core damage, prevention of containment failure, and consequence mitigation.

Information Requested by Question 4

Potential leakage due to age-related degradation from uninspectable areas is factored into both the original risk assessment for extending the ILRT interval and the new risk assessment provided as Attachment 3. Both these assessments include the increase in containment leakage from pathways that are not tested by LLRTs. These pathways include leakage due to liner failure. For these pathways, the impact of increasing the ILRT interval included the probability that a liner failure would occur and be detected by the ILRT based on historical data. Since the historical data includes all known liner failure events, both the original risk assessment and the new risk assessment provided as Attachment 3 inherently included the risk due to age-related degradation.

However, to provide added assurance regarding the NRC concern, I&M has performed an additional risk assessment to consider the impact of age-related degradation (corrosion) in inaccessible areas of the containment liner. The inaccessible areas included the containment liner behind the ice condenser. The additional risk assessment considered the likelihood of an age-adjusted liner flaw that would lead to a breach of containment. The additional risk assessment also considered the likelihood that the flaw was not visually detected but could be detected by an ILRT. The methodology used for this assessment is similar to that used in analyses performed to address the same concern at Calvert Cliffs Nuclear Power Plant and Comanche Peak Steam Electric Station in Reference 18 and Reference 19, respectively. These analyses supported ILRT interval extension amendments that were approved by the NRC in Reference 20 and Reference 21, respectively. Details of the CNP assessment are provided in Attachment 4 to this letter, and a summary of the assessment is provided below.

Consistent with the methodology used for Calvert Cliffs and Comanche Peak, the following steps were performed for CNP:

- The likelihood of a corrosion-related liner flaw was determined.
- The likelihood of a corrosion-related liner flaw was adjusted for age.
- The change in flaw likelihood for an increase in inspection interval was determined.
- The likelihood of a breach in containment for a given liner flaw was determined.
- The likelihood of failure to detect a flaw by visual inspection was determined considering the portion of the liner that is uninspectable.
- The likelihood of non-detected containment leakage due to the increase in test interval was determined.

The results of the above process were then used, along with the results of the new risk assessment, described above under "Quality of Plant Risk Analysis" and "Summary of New Assessment Results," to determine the effect on the predicted person-rem/year, LERF, and conditional containment failure probability due to age-related liner degradation. The results are summarized in the following table:

Change in Risk	Test Interval Extension	
	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		
New Risk Assessment	0.0294	0.0122
Including Liner Degradation	0.0297	0.0124
Percentage increase in person-rem/year		
New Risk Assessment	0.028 percent	0.012 percent
Including Liner Degradation	0.028 percent	0.012 percent
Change in LERF		
New Risk Assessment	1.23×10^{-7}	5.14×10^{-8}
Including Liner Degradation	1.32×10^{-7}	5.70×10^{-8}
Fractional Change in the Conditional Containment Failure Probability		
New Risk Assessment	0.0025	0.0011
Including Liner Degradation	0.0029	0.0013

The above results indicate the following:

- The effect of age-related liner degradation on the percentage increase in person-rem/year determined by the new risk assessment is imperceptible. Therefore, the percentage increase in person-rem/year remains reasonable when compared to the range of increase resulting from Option B to 10 CFR 50, Appendix J, as stated in NUREG-1493 and described above under "Summary of New Assessment Results."

- The effect of liner degradation on the increase in LERF determined by the new risk assessment is such that the changes in LERF remain below the acceptance guidelines in RG 1.174 for "very small" and "small" changes as described above under "Summary of New Assessment Results."

- The effect of liner degradation on the fractional increase in conditional containment failure probability determined by the new risk assessment is not significant. Therefore, defense-in-depth is maintained in accordance with the guidelines provided in RG 1.174, as described above under "Summary of New Assessment Results."

Additional Information

In a telephone conference conducted on October 1, 2002, the NRC identified certain deviations from the assumptions used in the Calvert Cliffs and Comanche Peak analyses that they considered appropriate for the CNP assessment. These deviations are addressed below.

Value Assumed for Historical Basemat Liner Failures

As documented in Reference 18 and Reference 19, a value of 0.5 was assumed in the Calvert Cliffs and Comanche Peak analyses for the number of historical basemat liner failures due to corrosion, and a value of 2 was assumed for the number of historical cylinder and dome liner failures due to corrosion. The value of 0.5 for basemat liner failures was a conservative assumption, since there have been no such failures identified in the industry. In the October 1, 2002, telephone conference, the NRC stated that I&M should provide additional justification for assuming a basemat liner failure rate due to corrosion that was lower than that assumed for the cylinder and dome liner. This justification is provided below. Notwithstanding this justification, I&M conservatively assumed a value for basemat liner failures that was equal to the value assumed for cylinder and dome liner failures in its risk assessment of the impact of age-related degradation in inaccessible areas of the containment liner. This assumption is shown in Table 1 of Attachment 4 to this letter.

I&M addressed the potential for basemat liner corrosion in its investigation of the liner corrosion that was found in the area behind the moisture barrier seal, as described in the response to Question 1. As part of that investigation, the basemat liner below the moisture barrier seal was examined using fiber optics. It was determined that the protective passive film caused by the alkalinity of the concrete was intact. Samples from the bottom liner indicated a high pH level, which promotes passivation, and a chloride level well below that necessary to break down the passivation film. The joints between sections of concrete above the basemat liner were designed with water stops to prevent moisture from reaching the basemat liner plate. The seals at all joints for floor level concrete sections were inspected and the seal material at all joints was replaced. There were no indications of subsurface degradation at any of these joints.

A specific corrosion mechanism of concern is that resulting from contact between the liner and moisture-retaining foreign material, such as wood, embedded in the concrete. I&M considers the likelihood of foreign material contacting the basemat liner to be much less than the likelihood of foreign material contacting the cylinder liner, due to the configuration of the respective concrete pours during construction. As previously noted, the cylindrical portion of the liner acts as the inner form during the pouring of the vertical concrete walls. The space between the outer form and the cylindrical liner (approximately three feet) is extensively occupied by reinforcing bar (rebar). This would have hampered full observation of the outer diameter of the liner prior to pouring the concrete. However, the concrete basemat was poured and the surface finished prior to installation of the basemat liner. Since the surface of the nominal 5 1/2-inch finish pour would have been visible prior to installing the liner, it is unlikely that it contains foreign material that could potentially contact the liner. During the subsequent pouring of concrete sections above the basemat liner, the top surface of the liner would have been visible through no more than 24 inches of rebar. Consequently, it is more likely that construction workers would have identified and removed foreign material that could contact the basemat liner than the cylinder liner.

Value Assumed for Historical Cylinder and Dome Liner Failures

As noted above, a value of 2 was assumed in the Calvert Cliffs and Comanche Peak analyses for the number of historical cylinder and dome liner failures due to corrosion. This value was based on the instances of liner corrosion identified at North Anna Power Station Unit 2 and at Brunswick Steam Electric Plant Unit 2. In the October 1, 2002, telephone conference, the NRC stated that I&M should provide additional justification for not assuming an additional historical failure based on the improper repair and through-wall hole in the CNP Unit 2 containment liner described in the response to Question 1 above. This justification is provided below. Notwithstanding this justification, I&M conservatively assumed a value of 3 for cylinder and dome liner failures in its risk assessment of the impact of age-related degradation in inaccessible areas of the containment liner. This assumption is shown in Table 1 of Attachment 4 to this letter.

As documented in Reference 22 and Reference 23, the North Anna Unit 2 and Brunswick Unit 2 liner failures consisted of through-wall holes that were clearly caused by corrosion. Therefore, it is appropriate that they be included in an assessment specifically addressing the risk due to age-related degradation. However, as described in the response to Question 1, the through-wall hole in the CNP Unit 2 liner was clearly not caused by corrosion. Although there was corrosion centered on the exposed portion of the wire brush handle, the remaining wall thickness was well above the minimum required for the liner to fulfill its function, as determined by an engineering analysis. Therefore, I&M considers that the through-wall hole at CNP Unit 2 need not be considered as a containment liner failure caused by age-related degradation.

References

1. Letter from J. E. Pollock, Indiana Michigan Power Company (I&M), to U. S. Nuclear Regulatory Commission (NRC) Document Control Desk, "License Amendment Request for One-time Extension of Containment Integrated Leakage Rate Test Interval" AEP:NRC:2612, dated April 11, 2002

2. Letter from J. F. Stang, NRC, to A. C. Bakken III, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Request for Additional Information Regarding License Amendment Request, 'One Time Extension Of Integrated Containment Leak Rate Test Interval,' dated April 11, 2002 (TAC Nos. MB4837 and MB4838)," dated October 31, 2002

3. Letter from J. A. Grobe, NRC, to R. P. Powers, I&M, "NRC Inspection Report 50-315/99026(DRS); 50-316/99026(DRS)," dated January 19, 2000

4. RG 1.163, "Performance-Based Containment Leak Test Program," dated September 1995

5. 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

6. Letter from E. E. Fitzpatrick, I&M, to NRC Document Control Desk, "Individual Plant Examination Response to NRC Audit Concerns and Request for Additional Information," dated October 26, 1995

7. Letter from J. B. Hickman, NRC, to E. E. Fitzpatrick, I&M, "Review of D. C. Cook Individual Plant Examination Submittal – Internal Events (TAC Nos. M74298 and M74399)," dated September 6, 1996

8. Generic Letter 88-20, "Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities-10 CFR 50.54(F)," dated November 23, 1988

9. 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

10. 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

11. NUREG/CR-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," dated January 1999

12. Letter from J. E. Pollock, Indiana Michigan Power Company, to Nuclear Regulatory Commission Document Control Desk, "License Amendment Request for One-Time Extension of Essential Service Water System Allowed Outage Time – Additional Information," AEP:NRC:2741-01, dated August 23, 2002

13. RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated July 1998

14. Electric Power Research Institute report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," dated August 1994

15. NUREG-1493, "Performance-Based Containment Leak-Test Program," dated September 1995

16. 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

17. NEI memo, "One-Time Extension of Containment Integrated Leak Rate Test Interval – Additional Information," dated November 30, 2001

18. Letter from C. H. Cruse, Constellation Nuclear, to NRC Document Control Desk, "Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension," dated March 27, 2002

19. Letter from C. L. Terry, TXU Energy, to NRC Document Control Desk, "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 (TAC Nos. MB3685 and MB3685)," dated June 12, 2002

20. Letter from D. Skay, NRC, to C. H. Cruse, Constellation Nuclear, "Calvert Cliffs Nuclear Power Plant, Unit No. 1 – Amendment Re: One-Time Extension of Appendix J, Type A, Integrated Leak Rate Test Interval and Exception from Performing a Post-Modification Type A Test (TAC No. MB3929)," dated May 1, 2002

21. Letter from D. H. Jaffee, NRC, to C. L. Terry, TXU Energy, "Comanche Peak Steam Electric Station (CPSES), Units 1 and 2 – Issuance of Amendments Re: One-Time Extension of Appendix J, Type A, Integrated Leak Rate Test Interval from Ten to Fifteen Years (TAC Nos. MB3685 and MB3686)," dated August 15, 2002

22. Letter from W. R. Matthews, Virginia Electric and Power Company, to NRC Document Control Desk, transmitting Licensee Event Report No. 50-339/99-002-00, "Containment Liner Through Wall Defect Due to Corrosion," dated October 21, 1999

23. Letter from J. S. Keenan, Carolina Power and Light Company, to U. S. NRC Document Control Desk, "Response to Request for Additional Information Regarding Request for License Amendments – Frequency of Performance Based Leakage Rate Testing (NRC TAC Nos. MB3470 and MB3471)," dated February 5, 2002

ATTACHMENT 2 TO AEP:NRC:2612-01

SUMMARY OF ASME SUBSECTION IWL
CONTAINMENT CONCRETE EXAMINATIONS

An examination of the Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2 containment concrete surfaces was performed in the fall of 2001 in accordance with 10 CFR 50.55a and American Society of Mechanical Engineers Boiler and Pressure Vessel Code Subsection IWL. The scope, inspection basis (acceptance criteria), and findings are summarized below.

Scope

In accordance with IWL-1220(b), the following portions of containment are exempt from the examination requirements of IWL-2000:

> "Portions of the concrete surface that are covered by liner, foundation material, or backfill, or are otherwise obstructed by adjacent structures, components, parts, or appurtenances."

Based on these exemption criteria, the surfaces described below were exempt from inspection.

Concrete Base Slab –The top of the base slab within the containment is lined with steel plate; the remainder of the concrete base slab is covered by backfill or obstructed by the containment concrete wall. Therefore, all surface areas of the concrete base slab were exempt from inspection.

Concrete Wall –The following portions of the concrete wall were exempt from inspection:

- The entire interior surface of the concrete wall. This surface is lined with a metallic liner.
- The exterior surfaces covered by backfill between elevations 596 feet 3½ inches and 608 feet.
- The exterior surfaces obstructed by the east and west main steam enclosures.
- The exterior surfaces obstructed by the Auxiliary Building structure.
- The exterior surfaces obstructed by the electrical penetration tunnel structure.
- The exterior surfaces obstructed by the containment building exhaust duct.

Concrete Dome – The following portions of the concrete dome were exempt from inspection:

- The entire interior surface of the concrete dome. This surface is lined with a metallic plate.
- The exterior surfaces obstructed by the containment building exhaust dome and duct.
- The exterior surfaces obstructed by the ice deflector assembly, located adjacent to the exhaust dome between elevation 709 feet and 715 feet 6 inches.

Inspection Basis

To assure a consistent approach during the performance of visual examinations, I&M developed an inspection basis to identify conditions that could indicate damage or degradation with a potential to affect the containment structural integrity. The conditions identified were consistent with those specified in ACI 201.1-68, "Guide for Making a Condition Survey of Concrete in Service, " as referenced by IWL-2510. These conditions were documented prior to the inspection. The following categories and sub categories were developed to envelope those conditions that may indicate concrete damage or degradation :

- Cracking
- Deterioration
  - Distortion
  - Leaching
  - Popout
  - Scaling
  - Spall
  - Corrosion
- Degradation Mechanisms
  - Exposed reinforcing steel

For each of the above categories and sub-categories, three threshold levels of severity (Recordable, Suspect, and Potentially Degraded or Damaged) were developed. The specific thresholds for each category or sub-category are described in the following table:

Threshold Level	Threshold
Cracking	
Recordable	Any crack that visually appears to be greater than 1 mm (0.04") in maximum width.
Suspect	Evidence of 1) active cracking, 2) corrosion staining, or 3) other degradation mechanisms at the site of the crack (e.g., bulging caused by corrosion buildup).
Potentially Degraded or Damaged	Confirmation of 1) active cracking, 2) corrosion staining, or 3) other degradation mechanisms.

Threshold Level	Threshold
	Distortion
Recordable	Any abnormal deformation of concrete from its original shape.
Suspect	Evidence of abnormal deformation of concrete from its original shape.
Potentially Degraded or Damaged	Confirmation of abnormal deformation of concrete from its original shape.
	Leaching
Recordable	Any leaching.
Suspect	Evidence of reinforcing steel degradation (either corrosion staining or bulging caused by corrosion build-up).
Potentially Degraded or Damaged	Confirmation of reinforcing steel degradation.
	Popout
Recordable	Any popout that visually appears to be greater than 50 millimeters (2.00 inches) in diameter (or equivalent surface area).
Suspect	Evidence of 1) popout beyond the concrete cover, 2) reinforcing steel degradation (either corrosion staining or bulging caused by corrosion build-up), or 3) exposed reinforcing steel.
Potentially Degraded or Damaged	Confirmation of reinforcing steel degradation or exposure.
	Scaling
Recordable	Any scaling that visually appears to be greater than 30 millimeters (1.125 inches) in depth.
Suspect	Evidence of 1) scaling beyond the concrete cover, 2) reinforcing steel degradation (either corrosion staining or bulging caused by corrosion build-up), or 3) exposed reinforcing steel.
Potentially Degraded or Damaged	Confirmation of reinforcing steel degradation or exposure.

Threshold Level	Threshold
	Spall
Recordable	Any spall that visually appears to be 20 millimeters (0.750 inches) or more in depth and 200 millimeters (8.00 inches) or greater in any dimension.
Suspect	Evidence of 1) spalling beyond the concrete cover, 2) reinforcing steel degradation (either corrosion staining or bulging caused by corrosion build-up), or 3) exposed reinforcing steel.
Potentially Degraded or Damaged	Confirmation of reinforcing steel degradation or exposure.
	Corrosion
Recordable	Indication of corrosion staining emerging from a concrete surface or any other evidence of corrosion (e.g., corrosion stains in pores, bulging caused by corrosion buildup, etc.).
Suspect	Evidence of 1) reinforcing steel degradation (either corrosion staining or bulging caused by corrosion build-up), or 2) exposed reinforcing steel.
Potentially Degraded or Damaged	Confirmation of reinforcing steel degradation or exposure.
	Exposed Reinforcing Bar
Recordable	Any exposed reinforcing steel.
Suspect	Evidence of exposed reinforcing steel.
Potentially Degraded or Damaged	Confirmation of exposed reinforcing steel.

The disposition processes for the three severity thresholds are described below. Where appropriate, the conditions identified were documented in the CNP corrective action program.

- Recordable: The location and conditions are recorded for further monitoring.
- Suspect: The results are forwarded to the Registered Professional Engineer (RPE) specified by IWL-2320 for disposition. In accordance with IWL-2320, the RPE is responsible for the final evaluation of defects and their potential damage to the structural integrity.
- Potentially Degraded or Damaged: The RPE investigates the condition to determine if further evaluation or repair is warranted.

Findings

The results for each category or sub-category are summarized below. No conditions that met the Potentially Degraded or Damaged threshold were identified.

Cracking Recordable surface cracks were identified in both units. The highest crack density was observed around the openings in the shell used for personnel and equipment access during construction. The primary cracks in this area are along the concrete interfaces exposed to the weather. There is a crack that extends around most of the Unit 2 containment circumference in the eighth section above the spring line. The crack is almost continuous and runs at an approximately constant elevation. However, the width of the crack at this time is not significant. In general, all cracks observed are confined within the construction joints, i.e., a crack will not cross a construction joint. The cracking observed is typical of a concrete structure and does not pose structural concern.

Leaching (Efflorescence) Efflorescence was observed at several locations on both units. The most common locations are areas in which repairs had been effected, either through patching or structural repairs. The efflorescence is most pronounced on the Unit 1 dome (top five sections), and along the joints of construction accesses of both units. Except for the efflorescence on top of the domes, previous inspections have noted the same efflorescent locations. The photographs of common efflorescence locations taken in the 1993 inspection were compared to those from the same locations identified in the 2001 inspection. No discernible differences were noted. It was concluded that efflorescence in the cylindrical walls is no longer ongoing.

Popout, Scaling, and Spall A certain amount of popout, scaling, and spalling was observed. The original surface repairs to the containment are becoming loose due to age and weathering, and are coming out as loose plate-like sheets. Because these sheets are very shallow, no reinforcing bars have been exposed and no telltale brown stain, indicating reinforcing bar (rebar) leaching, is present. The condition is more prevalent on the Unit 1 containment dome than on the Unit 2 containment dome.

Corrosion: There was no structural corrosion observed. At one Unit 1 location, rust stains were observed due to an exposed bolt. A few other similar observations from past intrusive activities on the containment were observed. Such stains do not affect structural integrity.

Exposed Reinforcing Steel An exposed rebar was found at one location on the Unit 1 containment. The exposed rebar was approximately one half inch diameter. The primary rebar used in the containment is greater than one inch in diameter. Since the exposed rebar is not a primary rebar, there is minimal effect on structural integrity. However, the exposed rebar violates the applicable construction specification. The discrepant condition was evaluated and accepted in accordance with the CNP corrective action program.

Miscellaneous Observations Outside the Inspection Basis The technicians performing the inspection were instructed to note any conditions that may be discrepant even if the condition was not within the defined inspection basis. The following instances of foreign material in the concrete were noted:

- Unit 2 - Piece of wood, 5 1/2 inches by 5 1/2 inches, located in the steam generator normal blowdown flash tank area.
- Unit 1 – Piece of wood, 3/8 inch by 2 inches, located at elevation 700 feet, azimuth 63°.
- Unit 1 – Piece of wood, 1 inch by 6 inches, located at elevation 694 feet, azimuth 80°.
- Unit 1 – Crack, from 1/32 inch to 1/16 inch, with 1 inch diameter piece of wood, located in the first section above the spring line (elevation 709 feet), azimuth 164°.
- Unit 1 –Foreign material, apparently plastic, located at elevation 662 feet, azimuth 270°.

All of the items were shallowly embedded in the concrete. The areas around these items did not indicate any significant leaching or any other distress conditions. Considering the size of the elements found, engineering judgement indicated that the structural integrity of the containment was unaffected by the presence of these foreign materials.

Overall Summary

The examination of the Unit 1 and Unit 2 containments did not reveal any conditions that could potentially affect the structural integrity or the calculated design safety margins of the containments.

Several surface conditions in the form of buried wood and plastic, and exposed rebar were identified. The presence of these materials does not affect the structural integrity of the containments.

Through the life of the plant and especially at the time of construction, the containments have undergone repairs to preserve the structure's surface and avoid the effects of long-term weathering. Some of these surface repairs are deteriorating and, in the case of the Unit 1 dome, these patches of repairs are coming off as plate-like chunks of mortar. These conditions do not impact the structural integrity of the concrete.

ATTACHMENT 3 TO AEP:NRC:2612-01

RISK IMPACT ASSESSMENT FOR
EXTENDING CONTAINMENT TYPE A TEST INTERVAL

American Electric Power Nuclear Generating Group
Donald C. Cook Nuclear Plant Units 1 and 2

RISK IMPACT ASSESSMENT
FOR
EXTENDING CONTAINMENT TYPE A TEST INTERVAL

Analysis File 17141-0007-A2, Rev. 2

October 21, 2002

Prepared By: (Original signed by R. Dremel, 10-21-02) **Date:**__________

Prepared By: (Original signed by R. Schmidt, 10-21-02) **Date:**__________

Reviewed By: (Original signed by T. Morgan, 10-22-02) **Date:**__________

SCIENTECH, Inc.

Gaithersburg, Maryland

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ANALYSIS FILE: 17141-0007-A2, Rev. 2

1.0 CLIENT: American Electric Service Power Corporation –D.C. COOK Nuclear Plant Units 1 and 2

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

3.0 AUTHOR:

Hassan Elrada (Rev. 0)
E. Robert Schmidt (Rev. 1)
E. Robert Schmidt and Raymond Dremel (Rev. 2)

4.0 PURPOSE:

The purpose of this calculation is to assess the risk impact for extending the D.C.Cook Integrated Leak Rate Test (ILRT) interval 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". D. C. Cook Nuclear Power Plant Units 1 and 2 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. D.C.Cook will use this analysis to seek a one-time exemption from 1 in 10 years test interval to 1 in 15 years test interval.

Revision 2 of this analysis file is identical to Revision 1 except that Revision 2 uses results from the June 2001 PRA update [Reference 12] and the September 2002 Level 2 PRA update [Reference 13]. In addition, the determination of Class 3a and 3b frequencies account for the failure to detect the leak by visual inspection and the determination of Class 3b frequency accounts for core damage sequence characteristics.

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 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 performed for Crystal River 3 (CR3) [Reference 9] and Indian Point 3 (IP3) [Reference 7] with enhancements outlined in the EPRI Interim Guidance [Reference 10]. The CR3 and IP3 submittals have been approved by the NRC.

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 3]. This methodology is similar to that presented in EPRI TR-104285 [Reference 2] and NUREG-1493 [Reference 5] and incorporates the revised guidance and additional information of References 10 and 11. It uses a simplified bounding analysis approach to evaluate the risk impact on increasing the ILRT Type A interval from 10 to 15 years by examining the June 2001 update of the D.C.Cook PRA [Reference 12] and the September 2002 Level 2 PRA update [Reference 13] plant specific accident sequences in which the containment integrity remains intact or the containment is impaired. 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 random isolation failures of plant components other than those associated with Type B or Type C test components. 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 2] and provided in Table 1A of this report.

This calculation uses the following steps.

Step 1 – Quantify the base-line risk in terms of frequency per reactor year for each of the eight accident classes presented in Table 1.

The D.C. Cook Level 1 and 2 PRA analyses [References 12 and 13], and NUREG-1493 [Reference 5] 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 1 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 D.C. Cook Level 2 analysis were based on important phenomena and system related events identified in NUREG-1335 [Reference 6].

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

Reference 8 was used to assign person-rem to each of the classes described in Table 1A excluding Classes 4 and 5. Reference 8 is a calculation of the conditional person-rem dose to the population, within a 50-mile radius from the D.C.Cook plant. The total population dose in person-rem for each class is evaluated in detail in Section 11.2 of this analysis. Table 2 summarizes the results of this step.

Step 3 – Evaluate risk impact of extending Type A test interval from 10 to 15 years.

This step evaluates potential increase in the population dose release due to extending the ILRT test interval from 3-in-10-year to 1-in-10 year and to 1-in-15-year. Section 11.3 of this calculation contains the detailed evaluation of this step. Section 13.0 and Tables 3, 4 and 5 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 3].

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

7.0 INPUT INFORMATION

1. Updated PRA total Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) as calculated in Reference 12.

2. Dose Rates for the eight classes. Provided by P.J. Fulford, "D.C.Cook Year 2000 Offsite Dose Assessment, Calculation # 17141-0007-A1", dated 11/13/01 [Reference 8].

3. Frequency of various release categories from D.C.Cook updated Level 2 PRA as calculated in Reference 13.

4. Core damage frequency that cannot lead to large early release given a containment flaw from, American Electric Power, "Core Damage Frequency That Cannot Lead to Large Early Release Given a Containment Flaw Unit 1," EVAL-PA-02-03, Revision 0 [Reference 15].

5. Fraction of containment liner that cannot be inspected for Appendix J, ASME Section XI from Reference 14

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

3. 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.

4. American Electric Power Corp., "Donald C. Cook Nuclear Plants Units 1 & 2 Individual Plant Examination, Revision 1", Transmitted to the NRC by letter from E. E. Fitzpatrick, Indiana Michigan Power Company, to NRC Document Control Desk, dated October 26, 1995.

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

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

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

8. P.J. Fulford, "D.C.Cook Year 2000 Offsite Dose assessment, Calculation # 17141-0007-A1", dated 11/13/01.

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

10. 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.

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

12. American Electric Power, "Unit 1 and 2 Core Damage and Large Early Release Quantification," EVAL-PA-01-02, Revision 0.

13. SCIENTECH, Inc., "Donald C. Cook Nuclear Plant Unit 1 Addendum to Level 2 Probabilistic Risk Assessment Update," Revision 0.

14. American Electric Power, "Inspectable Surface Area of the Containment Liner During Conduct of the Visual Examination Required by ASME Section XI Subsection IWE and 10CFR 50 Appendix J," DIT-S-01135-00.

15. American Electric Power, "Unit 1 Core Damage Frequency That Cannot Lead to Large Early Release Given a Containment Flaw," EVAL-PA-02-03, Revision 0.

9.0 MAJOR ASSUMPTIONS:

1. Containment leak rates less than twice the allowable leak rate (La) or 2 La indicates an intact containment. This leak rate is considered as "negligible".

2. The containment leakage for Class 1 sequences is assumed to be 1 La.

3. The containment leakage for Class 3a sequences is assumed to be 10 La.

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

5. 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.

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

7. An undetected failure of the liner in the basemat region of the containment cannot lead to a LERF. For such a failure to lead to a LERF, there must be a pre-existing failure in the two-foot-thick bottom slab concrete which overlays the liner. Should that unlikely event happen, the flaw must align with a pre-existing crack in the ten-foot basemat concrete. Even then, the release would need to penetrate the soil structure surrounding the concrete. Therefore, any release through a

pre-existing flaw in the basemat liner is unlikely to lead to a large release. Another mitigating feature for most sequences, given that ice from the ice condenser would melt is that water would cover the floor preventing direct release of fission products to the environment.

10.0 IDENTIFICATION OF COMPUTER CODES

None used.

11.0 DETAILED ANALYSIS:

11.1 Step 1 – Quantify the base-line risk in terms of 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 2. 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 7]. Class 1 and Class 7 were evaluated based on the Crystal River Unit 3 (CR3) Calculation [Reference 9] 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 7], the term CI was deducted from Class 7. Class 3 was evaluated based on interim Guidance [Reference 10]. 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 = 2.61E-05/yr [From Table 3 of Reference 13]

CI = Class_3a_Frequency + Class_3b_Frequency + Class_6_Frequency

= 6.198E-07/yr + 3.073E-08/yr + 4.848E-08 /yr = 6.990E-07/yr
[These values are obtained from the Class 3 and 6 sequences sections below.]

or

Class_1_Frequency = 2.61E-05/yr – 6.990E-07/yr = 2.540E-05/yr

Class 2 Sequences. 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 large, greater than 2 inch diameter, containment isolation valves. The frequency for these sequences is determined as follows:

Class_2_Frequency = The Sum of STC E, G, and U frequencies [From Table 3 or Reference 13]
Class_2_Frequency = 4.60E-09/yr + 8.64E-09/yr + 5.59E-10/yr [From Table 3 or Reference 13]
Class_2_Frequency = 1.38E-08/yr [From Table 3 or Reference 13]

Class 3 Sequences. 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 2], the EPRI Interim Guidance [Reference 10] and the NEI Additional Information [Reference 11].

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 5] and the EPRI Interim Guidance [Reference 10]. 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 10].

The total updated CDF is 4.848E-05 / yr from Reference 12.

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
= 0.415 [From Reference 14]

Fraction of containment liner that can be inspected for Appendix J, ASME Section XI
= 0.585 [From Reference 14]

Probability of failure to detect a flaw during a visual inspection = 0.1 [Assumption 6]

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

= 0.415 + (0.585 * 0.1) = 0.4735

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 * 4.848E-05/yr * 0.4735 = 6.198E-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 5] and new data presented by the EPRI Interim Guidance [Reference 10]. 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 10.)

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 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 even if there is a preexisting leak and including them in Class 3b in 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 = 4.848E-5/yr [From Reference 12]

CDF of sequences that have a large release irrespective of Type A Leakage
= 5.588E-06 [From Reference 12]

CDF of sequences that cannot cause a large risk significant release
= 1.885E-05 [From Reference 15]

Therefore:

PCDF_TypeA = 4.848E-05 – 5.588E-06 - 1.885E-05 = 2.404E-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.

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 * 2.404E-05 * 0.4735 = 3.073E-08 / yr

Class 4 Sequences. 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.

Class 5 Sequences. 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 x 10^{-3} [Assumed Conservative Value]

CDF = 4.848-05/yr

Class_6_Frequency = 1.0E-03 * 4.848E-05/yr = 4.848E-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 = CDF – (No-Cont_Failure_Freq + Class_8_Frequency + Class_2_Frequency)

Where:

CDF = 4.848E-05/yr	[From Reference 12]
No-Cont_Failure_Freq =2.61E-05/yr	[STC S from Table 3 of Reference 13]
Class_8_Frequency = Sum of STC C and T Frequencies	[From Table 3 of Reference 13]
Class_8_Frequency = 1.78E-06/yr + 1.29E-06/yr	
Class_8_Frequency = 3.07E-06/yr	
Class_2_Frequency = 1.38E-08/yr	[Calculated Above]

Class_7_Frequency = 4.848E-05/yr – (2.61E-05/yr + 3.07E-06/yr + 1.38E-08/yr) = 1.930E-05/yr

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

Class_8_Frequency = Sum of STC C and T Frequencies [From Table 3 of Reference 13]
Class_8_Frequency = 1.78E-06/yr + 1.29E-06/yr
Class_8_Frequency = 3.07E-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 1.

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 2, this step develops the D.C. Cook 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 one through eight sequences above)

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

Reference 8 documents an updated assessment of the D. C. Cook Power Plant off-site population dose consequences due to the accidental release of radiological materials resulting from several severe accident scenarios. This assessment utilizes a year 2000 population estimate and is an update in this respect of a previous Level III PRA consequence analysis prepared for the plant IPE for a year 1980 region population distribution. The results are for a 50-mile radius region surrounding the plant.

This calculation uses the 200% of 1980 increase data values from Table 2 of Reference 8 for sequences SB0181$ = 1.01+03 for Classes 1, 3a and 3b, LL08** = 3.84E+06 for Classes 2, 6 and 7 and SGR50** = 9.68E+6 for Class 8. These sequences and their source terms were those selected for the IPE Level III analysis and are defined in Reference 4. Sequence SBO181$ is a station blackout sequence with no containment failure. Sequence LL08** is a large LOCA sequence with early containment failure at the basemat. Sequence SGR50** is a steam generator containment bypass sequence with containment failure at the basemat. The resulting conditional population doses given the release are

Class 1 = (1.01E+03) * 1 La = 1.01E+03 person-rem
Class 2 = (3.84E+06) person-rem
Class 3a = (1.01E+03) * 10 La = 1.01E+04 person- rem
Class 3b = (1.01E+03) * 35 La = 3.535E+04 person-rem
Class 4 = Not Analyzed
Class 5 = Not Analyzed
Class 6 = (3.84E+06) person-rem
Class 7 = (3.84E+06) person-rem

Class 8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this class are expected to be released directly to the environment. Based on Table 2 of Reference 8, the value used is 9.68E+06 person-rem.
The above values are summarized in Table 2.

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

The baseline risk for each accident class is presented in Table 3. The baseline risk is defined as the product of the containment failure mode frequency and the conditional population dose. Table 3 is the product of Tables 1 and 2. The ILRT baseline risk is based on the test interval of 3 in 10 years or about 1 in 3 years.

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 = 6.260E-03 person-rem/year

Class3b_Base = 1.086E-03 person-rem/year

Class_3_Base_Total = 6.260E-03 + 1.086E-03 = 0.00735 person-rem/yr

Total_base = 1.0409E+02 person-rem/year

%Base_Risk = [(6.260E-03 + 1.086E-03 / 1.0409E+02] * 100
%Base_Risk = 0.0071%

Therefore, the total baseline risk contribution of leakage, potentially impacted by the ILRT test interval, represented by Class 3 accident scenarios is 0.0074 person-rem/year or 0.0071% of the total population exposure risk.

11.3 Step 3 – Evaluate risk impact of extending Type A test interval from 10 to 15 years.

Risk impact due to 10-year test interval

According to NUREG-1493 [Reference 7], extending the Type A ILRT interval from 3 in 10 years to 1 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 4.

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 = 2.085E-02 person-rem/year

Class3b_10 = 3.617E-03 person-rem/year

Class3_10_total = 2.085E-02 + 3.617E-03 = 0.0245 person-rem/year

Total_10 = 1.0410E+02 person-rem/year

%Risk_10 = [(2.085E-02 + 3.617E-03) / 1.0410E+02] * 100

%Risk_10 = 0.024%

Therefore, the total risk contribution of leakage for Type A 10-Year ILRT interval represented by Class 3 accident scenarios is 0.0245 person-rem/year or 0.024% 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 3 in 10 years (baseline case) to 1 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 = 0.00735 person-rem/yr	[From above]
Class3_10_total = 0.0245 person-rem/year	[From above]
Total_base = 1.0409E+02 person-rem/year	[From Table 3]

[(Class3_10_total - Class3_Base_total) / Total_base] * 100
= [(0.0245 – 0.00735) / 1.0409E+02] * 100 = (0.0172/1.0409E+02) * 100 = 0.017 %

Therefore, The total risk increase due to extending the ILRT interval from 3 in 10 years (baseline case) to 1 in 10 years is 0.0172 person-rem/year or 0.017% 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 1 in 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 11 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 PSA 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 = 3.130E-02 person-rem/year

Class3b_15 = 5.432E-03 person-rem/year

Class3_15_total = 3.130E-02 + 5.432E-03 = 0.0367 person-rem/year

Total_15 = 1.0412E+02 person-rem/year

%Risk_15 = [(3.130E-02 + 5.432E-03) / 1.0412E+02] * 100 [From Table 5]

%Risk_15 =0.035%

Therefore, the total risk contribution of leakage for Type A 15-year ILRT interval represented by Class 3 accident scenarios is 0.0367 person-rem/year or 0.035% of the total population risk.

The percent risk increase due to extending the ILRT interval from 3 in 10 years (baseline case) to 1 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 = 0.0367 person-rem/year [From above]
Class_3_Base_Total = 0.00735 person-rem/yr [From above]
Total_base = 1.0409E+02 person-rem/year [From Table 3]

[(Class3_15_total - Class_3_Base_Total) / Total_base] * 100
= [(0.0367 – 0.00735)/ 1.0409E+02] * 100 = (0.0294 /1.0409E+02) * 100 = 0.028%

Therefore, the total risk increase due to extending the ILRT interval from 3 in 10 years (baseline case) to 1 in 15 years is 0.0294 person-rem/year or 0.028% of the total baseline population risk.

The percent risk increase in terms of person-rem/year from 1 in 10 years to 1 in 15 years test interval for Classes 3a and 3b is:

% Risk (10-15PR) =[(Class3_15_total) – (Class_3_10_Total) / (Class_3_10_Total)]*100

Where:

Class3_15_total = 0.0367 person-rem/year [From above]
Class_3_10_Total = 0.0245 person-rem/yr [From above]

% Risk (10-15PR) = [(0.0367 – 0.0245) /0.0245] * 100 = 49.8%

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:

[(person-rem(Class3)_15) – (person-rem(Class3)_10)] = 0.0367 – 0.0245 = 0.0122 person rem

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 - Class_3_10_Total) / Total_10] * 100

Where:

Class3_15_total = 0.0367 person-rem/year [From above]
Class_3_10_Total = 0.0245 person-rem/yr [From above]
Total_10 = 1.0411E+02 person-rem/year [From Table 4]

[(Class3_15_total - Class_3_10_Total) / Total_10] * 100= [(0.0367 – 0.0245)/ 1.0411E+02] * 100 =
(0.0122 /1.0411E+02) * 100 = 0.012%

Therefore, the total risk increase due to extending the ILRT interval from 1 in 10 years (baseline case) to 1 in 15 years is 0.0122 person-rem/year or 0.012% 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 3 in 10 years to 1 in 15 years and from 1 in 10 years to 1 in 15 years .

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 7 and 9] 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 9] 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 = 3.073E-08 per year [Section 11.1]

The 1 in 10 years LERF affected by ILRT = 3.073E-08 * 3.33 = 1.023E-07 per year

The 1 in 15 years LERF affected by ILRT = 3.073E-08 * 5 = 1.537E-07 per year

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

1.537E-07 – 3.073E-08 = 1.23E-07/year

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

1.537E-07 – 1.023E-07 = 5.14E-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 tests in ten years, 1 test in ten years, and 1 test in fifteen 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 1 (i.e., Class 1 frequency is 2.540E-05, the Class 3a frequency is 6.198E-07 and CDF = 4.848E-05),

the CCFP for 3 tests in ten years =

1 – [(2.540E-05 + 6.198E-07)/4.848E-05] = 4.633E-01

Using the above equation and the data from Table 4 (i.e., Class 1 frequency is 2.389E-05 ,the Class 3a frequency is 2.064E-06 and CDF=4.848E-05),

the CCFP for 1 test in ten years =

1-[(2.389E-05 + 2.064E-06) / 4.848E-05] = 4.647E-01

Using the above equation and the data from Table 5 (i.e., Class 1 frequency is 2.280E-05 , the Class 3a frequency is 3.099E-06 and CDF = 4.848E-05),

the CCFP for 1 test in fifteen years =
1-[(2.280E-05 + 3.099E-06)/4.848E-05] = 4.658E-01

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

= 4.658E-01 – 4.633E-01 = 0.0025

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

= 4.658E-01 – 4.647E-01 = 0.0011

12.0 COMPUTER INPUT AND OUTPUT

NONE

13.0 SUMMARY OF RESULTS

Table R1 below summarizes the major results.

Table R1- 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)	0.0294	0.0122
The percentage increase person-rem/year risk (See Section 11.3)	0.028%	0.012%
Change in LERF (See Section 11.4)	1.23E-07	5.14E-08
Change in the Conditional Containment Failure Probability (See Section 11.5)	0.0025	0.0011

Other results are shown in the Table R2 below. It shows (for example), that the change in Type A test frequency from once per ten years to once per fifteen years increases the total integrated plant risk for those accident sequences influenced by Type A testing by only 0.0122 (i.e. 0.0367 - 0.0245 = 0.0122) person-rem/year.

Table R2 – Other Results

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	0.0071% of integrated value based on 10 La for Class 3a and 35 La for Class 3b, which is equivalent to: 0.0074 person-rem/year	0.017 % of integrated value based on 10 La for Class 3a and 35 La for Class 3b, which is equivalent to: 0.0245 person-rem/year	0.035% of integrated value based on 10 La for Class 3a and 35 La for Class 3b, which is equivalent to: 0.0367 person-rem/year
Total Integrated Risk	104.09 person-rem/year	104.11 person-rem/year	104.12 person-rem/year

The sensitivity of the above results to changes in the Level III results used in this evaluation have been considered. As indicated in Reference 8 the consequences for LOCA and Steam Generator Tube rupture sequences are higher for sensitivity study cases run with containment failure in the upper compartment rather than the base case assumption of failure at the basemat. If the higher numbers were used in the present analysis, the absolute value of the change in person-rem/year would remain the same while the percentage contribution to the total person-rem/year would decrease.

Also, as indicated in Reference 8 there is some uncertainty as to the accuracy of the original IPE population distribution. Two-hundred percent of the 1980 results were judged to be a conservative bounding estimate for the year 2000. If, however, the true consequences are higher than this, the total risk, as well as the incremental risk for the extended ILRT interval, will increase by the same proportion while the percentage contribution and percentage increase due to the requested change will remain the same.

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 ten years to once per fifteen years increases the total integrated plant risk for those accident sequences influenced by Type A testing by only 0.0122 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 interval from once per 10 years to once per 15 years is 5.14E-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.

Table 1A- 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 1 - Mean Containment Frequency Measures for a Given Accident Class

Class	Description	Frequency/yr.
1	No Containment Failure	2.540E-05
2	Large Containment Isolation Failure (Failure-To-Close)	1.380E-08
3a	Small Isolation Failures (Liner Breach)	6.198E-07
3b	Large Isolation Failures (Liner Breach)	3.073E-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)	4.848E-08
7	Severe Accident Phenomena Induced Failure (Early and Late Failures)	1.930E-05
8	Containment Bypassed (SGTR)	3.070E-06
Core Damage	All Containment Event Tree (CET) Endstates	4.848E-05

TABLE 2 – Conditional Person-Rem Measures for a Given Accident Class

Class	Description	Person-Rem (50-miles)
1	No Containment Failure	1.01E+03
2	Large Containment Isolation Failure (Failure-To-Close)	3.84E+06
3a	Small Isolation Failures (Liner Breach)	1.01E+04
3b	Large Isolation Failures (Liner Breach)	3.54E+04
4	Small Isolation Failure – Failure-To-Seal (Type B test)	N/A
5	Small Isolation Failure – Failure-To-Seal (Type C Test)	N/A
6	Containment isolation Failures (Dependent failures, Personnel Errors)	3.84E+06
7	Severe Accident Phenomena Induced Failure (Early and Late Failures)	3.84E+06
8	Containment Bypassed (SGTR)	9.68E+06

TABLE 3 – 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	2.540E-05	1.010E+03	2.566E-02
2	Large Containment Isolation Failure (Failure-To-Close)	1.380E-08	3.840E+06	5.299E-02
3a	Small Isolation Failures (Liner Breach)	6.198E-07	1.010E+04	6.260E-03
3b	Large Isolation Failures (Liner Breach)	3.073E-08	3.535E+04	1.086E-03
4	Small Isolation Failure – Failure-To-Seal (Type B test)		N/A	N/A
5	Small Isolation Failure – Failure-To-Seal (Type C Test)		N/A	N/A
6	Containment isolation Failures (Dependent failures, Personnel Errors)	4.848E-08	3.840E+06	1.862E-01
7	Severe Accident Phenomena Induced Failure (Early and Late Failures)	1.930E-05	3.840E+06	7.410E+01
8	Containment Bypassed (SGTR)	3.070E-06	9.680E+06	2.972E+01
	All CET End states	4.848E-05		1.0409E+02

TABLE 4 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	2.389E-05	1.010E+03	2.412E-02
2	Large Containment Isolation Failure (Failure-To-Close)	1.380E-08	3.840E+06	5.299E-02
3a	Small Isolation Failures (Liner Breach)	2.064E-06	1.010E+04	2.085E-02
3b	Large Isolation Failures (Liner Breach)	1.023E-07	3.535E+04	3.617E-03
4	Small Isolation Failure – Failure-To-Seal (Type B test)		N/A	N/A
5	Small Isolation Failure – Failure-To-Seal (Type C Test)		N/A	N/A
6	Containment isolation Failures (Dependent failures, Personnel Errors)	4.848E-08	3.840E+06	1.862E-01
7	Severe Accident Phenomena Induced Failure (Early and Late Failures)	1.930E-05	3.840E+06	7.410E+01
8	Containment Bypassed (SGTR)	3.070E-06	9.680E+06	2.972E+01
CDF	All CET Endstates	4.848E-05		1.0411E+02

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	2.280E-05	1.010E+03	2.303E-02
2	Large Containment Isolation Failure (Failure-To-Close)	1.380E-08	3.840E+06	5.299E-02
3a	Small Isolation Failures (Liner Breach)	3.099E-06	1.010E+04	3.130E-02
3b	Large Isolation Failures (Liner Breach)	1.537E-07	3.535E+04	5.432E-03
4	Small Isolation Failure – Failure-To-Seal (Type B test)		N/A	N/A
5	Small Isolation Failure – Failure-To-Seal (Type C Test)		N/A	N/A
6	Containment isolation Failures (Dependent failures, Personnel Errors)	4.848E-08	3.840E+06	1.862E-01
7	Severe Accident Phenomena Induced Failure (Early and Late Failures)	1.930E-05	3.840E+06	7.410E+01
8	Containment Bypassed (SGTR)	3.070E-06	9.680E+06	2.972E+01
CDF	All CET End States	4.848E-05		1.0412E+02

ATTACHMENT 4 TO AEP:NRC:2612-01

EFFECT OF AGE-RELATED DEGRADATION ON RISK IMPACT ASSESSMENT FOR EXTENDING CONTAINMENT TYPE A TEST INTERVAL

American Electric Power Nuclear Generating Group
Donald C. Cook Nuclear Plant Units 1 and 2

EFFECT OF AGE-RELATED DEGRADATION ON

RISK IMPACT ASSESSMENT
FOR
EXTENDING CONTAINMENT TYPE A TEST INTERVAL

Analysis File 17141-0007-A3, Rev. 0

October 23, 2002

Prepared By: (Original signed by R. Schmidt, 10-23-02) **Date:**____________

Reviewed By: (Original signed by T. Morgan, 10-23-02) **Date:**____________

SCIENTECH, Inc.

Gaithersburg, Maryland

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ANALYSIS FILE: 17141-0007-A3, Rev. 0

1.0 CLIENT

American Electric Service Power Corporation –D.C. COOK Nuclear Plant Units 1 and 2

2.0 TITLE

Effect of Age-Related Degradation on 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 effect of age-related degradation of the containment liner on risk impact for extending the D.C.Cook Integrated Leak Rate Test (ILRT or Containment Type A test) interval from ten to fifteen years. This is in response to a Request for Additional Information (RAI) from the Nuclear Regulatory Commission (NRC) (Reference 1) concerning the D. C. Cook License Amendment Request for One-time Extension of Containment Integrated Leakage Rate Test Interval (Reference 17). Results for the requested test interval increase are provided along with the cumulative results for the change from the original 3 tests in 10 years to the requested 1 test in 15 years.

5.0 INTENDED USE OF ANALYSIS RESULTS

The results of this calculation will be used to support obtaining NRC approval to extend the Integrated Leak Rate Test (ILRT) interval from 1 test in 10 years to 1 test in 15 years. Specifically it is to respond to RAI #4 of Reference 1.

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 Cook Nuclear Plant (CNP) to age-related liner corrosion. The analysis, "Risk Impact Assessment for Extending Containment Type A Test Interval" SCIENTECH, Inc. analysis file 17141-0007-A2, Rev. 2 (Reference 2) provides an assessment of the increase in risk (person-rem/year, large early release frequency – LERF and Conditional Containment Failure Probability - CCFP) due to the requested extension in Type A test interval based on accepted EPRI and NEI guidance (References 3, 4, 5 and 6) and NRC reports and guidance (References 7 and 8).

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 liner failure. The impact of increasing the ILRT interval for these classes included the probability that a liner failure would occur and be detected by the Type A test that was based on historical data. Since the historical data includes all known liner 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 9), Comanche Peak Steam Electric Station (CPSES - Reference 10), South Texas Project (Reference 11) and H. B. Robinson Steam Electric Plant (Reference 12) in responses to similar RAIs. The CCNPP and CPSES extension request submittals have been approved by the NRC.

The following issues are addressed in the present analysis:

- Differences between the containment basemat and the containment cylinder and dome;
- The historical liner flaw likelihood due to concealed corrosion;
- The impact of aging;
- The liner corrosion leakage dependency on containment pressure; and
- The likelihood that visual inspection will be effective at detecting flaws.

As in Reference 9, this calculation uses the following steps with D. C. Cook Nuclear Plant (CNP) values utilized where appropriate:

Step 1 – Determine a liner corrosion-related flaw likelihood

Historical data will be used to determine the annual rate of liner corrosion flaws for the containment cylinder and head and for the basemat.

Step 2 – Determine an age-adjusted liner flaw likelihood

The historical liner flaw likelihood will be assumed to double every 5 years. The cumulative likelihood of a flaw in the liner 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 liner flaw

For there to be a leak from the containment, the liner flaw must be connected to a pathway through the surrounding concrete. The likelihood of this occurring is determined as a function of pressure and evaluated at the CNP 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 liner flaw will be determined considering the portion of the liner that is uninspectable at CNP 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 liner 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 containment cylinder and dome and the basemat area 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 updated analysis (Reference 2) 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.

7.0 INPUT INFORMATION

1. Methodology and generic results from the Calvert Cliffs assessment of age-related liner degradation (Reference 9).

2. The CNP ILRT test pressure of 12 to 12.5 psig (References 13 and 14).

3. CNP containment failure pressure of 36 psig (Reference 15). This is a conservatively low value corresponding to a high confidence of a low probability of failure.

4. Fraction of containment liner that cannot be inspected for Appendix J, ASME Section XI of 0.415 (Reference 16).

5. CNP core damage frequency that may be impacted by Type A leakage and contribute to LERF of 2.404-05 per year (Reference 2).

6. Person-rem for EPRI Class 3b failure classes at CNP of 3.535E+04 person-rem (Reference 2)

8.0 REFERENCES

1. "Request for Additional Information, D. C. Cook Nuclear Plant, Units 1, 2, One Time Extension of Integrated Containment Leak Rate Test Interval," received via E-mail from John Stang, USNRC, Date - 08/15/2002 07:59 AM, to Joseph R Waters, AEP, Subject - Fwd: RAI on D.C. Cook - ILRT Extension Amendment.

2. SCIENTECH, Inc., "Risk Impact Assessment for Extending the Type A Test Interval," Analysis File Number 17141-0007-A2, Revision 2, October 21, 2002. (To be provided at Attachment 3 to AEP letter to USNRC AEP:NRC:2612-02.)

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

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

5. 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.

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

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

8. 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.

9. "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.

10. "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.

11. "South Texas Project Units 1 and 2, Docket Nos. STN 50-498, STN 50-499, Response to Request for Additional Information – South Texas Project Containment Integrated Leakage Rate Test Interval Extension," STP Nuclear Operating Company letter to USNRC, June 25, 2002.

12. "H. B. Robinson Steam Electric Plant, Unit No. 2, Docket No. 50-261/License No. DPR-23, Response to Request for Additional Information on Amendment Request Regarding One-time Extension of Containment Type A Test Interval," CP&L letter to USNRC, June 19, 2002.

13. 1-EHP-4030-STP-202, "Integrated Leak Rate Test," Rev. 0-CS13.

14. 2-EHP-4030-STP-202, "Integrated Leak Rate Test," Rev. 0-CS8.

15. "Donald C. Cook Nuclear Plants Units 1 & 2 Individual Plant Examination, Revision 1," Transmitted to the NRC by letter from E. E. Fitzpatrick, Indiana Michigan Power Company, to NRC Document Control Desk, October 26, 1995.

16. Design Information Transmittal (DIT) No. DIT-S-01135-00, "Inspectable Surface Area of the Containment Liner during Conduct of the Visual Examination required by ASME Section XI Subsection IWE and 10 CFR 50 Appendix J," September 27, 2002.

17. "Donald C. Cook Nuclear Plant Units 1 and 2, Docket Nos. 50-315 and 50-316, License Amendment Request for One-time Extension of Containment Integrated Leakage Rate Test Interval," Indiana Michigan Power Company letter to USNRC AEP:NRC:2612, April 11, 2002.

9.0 MAJOR ASSUMPTIONS:

1. Previous submittals (References 9 through 12) have assumed a half failure for the basemat concealed liner corrosion due to the lack of any identified failures in the applicable historical record. It has been postulated that the lack of visual inspection of the liner in the basemat region would explain the lack of failures rather than the lower liner corrosion rate and that the rate in the basemat region should be the same as in the cylinder region. The nature of the construction configuration of the basemat is, however, significantly different from that of the cylinder. The liner in the basemat region is laid down upon a previously poured concrete slab while for the cylinder the liner serves as the form for the concrete. It is believed that this difference would lead to a difference in the likelihood of foreign objects being left in the concrete adjacent to the liner and therefore a difference in the liner corrosion rate between the two regions. However, to show that liner corrosion has little impact on the risk associated with ILRT extension, the conservative assumption will be made that the liner failure rate due to corrosion rate in the basemat region is the same as the rate in the cylinder and dome regions. (Step 1)

2. 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)

3. As in Reference 9, the liner flaw likelihood is assumed to double every 5 years. This is included to address the increased likelihood of corrosion due to aging. (Step 2)

4. The likelihood of the containment atmosphere reaching the outside environment given a liner flaw occurs is a function of containment pressure. Even without a liner, the containment is an excellent barrier. As the containment pressure increases, cracks in the concrete will form. If a crack occurs in the same region as a liner flaw then the containment atmosphere can communicate to the outside environment. At low pressures, crack formation is very unlikely. Near the failure point, crack formation is expected. As in Reference 9, anchor points of 0.1% chance of cracking near the flaw at 20 psia and 100% chance at the failure pressure (50.7 psia for D. C. Cook from Reference 15) are assumed with logarithmic interpolation between these two points. (Step 4)

5. As in Reference 9, the likelihood of leakage escape, due to crack formation, in the basemat region is considered to be 10 times less likely than in the cylinder or dome regions. While the assessment in Reference 15 of containment failure pressure concludes that the most likely containment failure location is the basemat adjacent to the cylinder wall, for such a failure to lead to a LERF, there must be a pre-existing failure in the two-foot-thick bottom slab concrete which overlays the liner. Should that unlikely event happen, the flaw must align with a crack in the ten-foot basemat concrete. Even then, the release would need to penetrate the soil structure surrounding the concrete. Therefore, any release through a pre-existing flaw in the basemat liner is unlikely to lead to a large release. Another mitigating feature for most sequences, given that ice from the ice condenser would melt is that water would cover the floor preventing direct release of fission products to the environment. Considering all the above, the assumption that the likelihood of leakage escape, due to crack formation, in the basemat region is 10 times less likely than in the cylinder or dome regions is retained. (Step 4)

6. As in Reference 9, a total visual inspection failure likelihood of 10% for that fraction of the liner that is inspectable is assumed. (Step 5)

7. All non-detectable 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.

10.0 IDENTIFICATION OF COMPUTER CODES

None used.

11.0 DETAILED ANALYSIS:

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

As indicated in Reference 9, two occurrences (Brunswick 2 and North Anna 2) of through wall liner corrosion related defects have been found since the September 1996 implementation of the visual inspection requirements of 10 CFR 50.55a. Both of these defects were in the cylinder region of the liner. None were found in the basemat region.

While some liner corrosion has been observed at CNP (see the response to RAI #1) the through wall liner defect found is not believed to be due to corrosion. Nevertheless, it is assumed that the observation at CNP is a corrosion related liner failure.

The Oyster Creek event cited in Reference 1 is not applicable to CNP since the Oyster Creek containment is a free-standing steel shell containment.

The likelihood of through wall liner defects for both the cylinder/dome region and the basemat region is therefore:

3 / (70 plants * 5.5 years/plant) = 7.79E-03 per year

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

Reference 9 provides the impact of the assumption that the historical liner flaw likelihoods will double every 5 years on the yearly, cumulative and average likelihood that an age-related flaw will occur. For a liner flaw likelihood based on 2 liner failures, the 15 year average flaw likelihood is 6.27E-03 per year for the cylinder/dome region. This result of Reference 9 is generic in nature, as they do not depend on any plant specific inputs, and therefore, are applicable to CNP.

For the present assumption of 3 liner historical failures, the 15 year average flaw likelihood is 1.5 times the above value or 9.41E-03 per year and in accordance with assumption 1 is applicable to both the cylinder and dome region and the basemat region.

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 9 to be 8.7% for the cylinder/dome region based on 2 historical liner flaws and the resulting 6.27E-03 per year 15 year average flaw likelihood. This result of Reference 9 is generic in nature, as they do not depend on any plant specific inputs, and therefore, are applicable to CNP.

For the present assumption of 3 liner historical failures, 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 1.5 times that given in Reference 9 or 13.1% and in accordance with assumption 1 is applicable to both the cylinder and dome region and the basemat region.

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 50.7 psia (Reference 15),

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

$$\text{Log } 100 = m * 50.7 + a$$

or

$$m = (\text{Log } 100 - \text{Log } 0.1) / (50.7 - 20) = 0.09772$$

and

$$a = \text{Log } 0.1 - 0.09772 * 20 = -2.9544$$

The upper end of the range of CNP ILRT pressures of 12.5 psig (References 13 and 14) gives the highest likelihood of breach.

At 27.2 psia (12.5 + 14.7), the above equation gives

$$\text{Log (likelihood of breach)} = 0.09772 * 27.2 - 2.9544 = -0.2964$$

$$\text{Likelihood of breach} = 10^{-0.2964} = 0.505\ \%$$

The above values are for the cylinder/dome portions of the containment. If the basemat is assumed to have 1/10th the failure rate, the likelihood would be 0.0505%.

11.5 *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 liner flaw is given by the percentage that is uninspectable plus an assumed 10% failure rate for the portion that is inspectable. Reference 6 indicates that approximately 25,700 sq. ft. of the liner in the cylinder and dome portion of the containment is obstructed and cannot be visually inspected. The total liner area in the cylinder/dome regions is approximately 61,900 sq. ft. (Reference 6), therefore

25,700 / 61,900 = 0.415 or 41.5% of the liner in this region is uninspectable

The 10% assumed inspection failure in the inspectable area leads to an additional 5.9% (10 % of the 58.5% that is inspectable) failure likelihood, or a total likelihood of failure to detect a flaw in the cylinder/dome region of

41.5 % + 5.9 % = 47.4 %

Since the basemat liner is inaccessible, the likelihood of failure to detect a flaw is taken as 100%.

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)

= 13.1% * 0.00505 * 0.474 = 0.0314% for the cylinder/dome region

= 13.1% * 0.000505 * 1.0 = 0.00662% for the basemat region.

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

Total Likelihood of Non-Detected Containment Leakage = 0.0314% + 0.0066% = 0.0380%

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

11.7 *Impact on Risk*

The above indicates that there is a very small likelihood that liner 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 9 as "The non-large early release frequency (LERF) containment over-pressurization failures...".

From Reference 2 this value is 2.404E-05 per year. The resulting increase in LERF is

Delta LERF due to age-related corrosion = 0.000380 * 2.404E-05 = 9.14E-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 updated value from Reference 2 plus the above or

Total Delta LERF = 1.23E-07 + 9.14E-09 = 1.32E-07 per year

The person-rem/year impact of the above age-related corrosion can be estimated by assuming that the delta LERF contributes to the EPRI containment failure Class 3b leakage. From Reference 2, the population exposure (50 mile person-rem) given an accident of this class is 3.535E+04 person-rem. The increase in person-rem/year due to the above assessment of age-related corrosion is therefore

3.535E+04 * 9.14E-09 = 3.2E-04 person-rem/year

This is very small compared to the increase estimated in Reference 2 of 0.0294 person-rem/year for the increase in ILRT interval from 3 in 10 years to 1 in 15 years.

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 0.0380% (or 0.00038). This added to the increase estimated in Reference 2 of 0.0025 gives a total increase in CCFP of 0.0029 for the increase in ILRT interval from 3 in 10 years to 1 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 3 in 10 years to 1 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 9. 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 1 in 10 to 1 in 15 years is then 61% (5.3/8.7) of the above values.

12.0 COMPUTER INPUT AND OUTPUT

None

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 3 tests in 10 years to 1 test in 15 years. The impact of these results on the major results of the updated ILRT extension analysis (Reference 2) is provided in Table 2 along with the results of the original submittal based on the IPE (Reference 17).

Table 1: Liner Corrosion Analysis Steps and Results

Step	Description	Containment Cylinder and Dome		Containment Basemat	
1	**Historical Liner Flaw Likelihood** Failure Data: Assumed to be applicable to both regions Success Data: Based on 70 steel-lined containments and 5.5 years since the 10 CFR 50.55a requirements for periodic visual inspection of containment surfaces.	Events: 3 through liner corrosion-related flaws. (Brunswick 2, North Anna 2 and Cook) 3 / (70 * 5.5) = **7.79E-03/year**		Events: 3 through liner corrosion-related flaws. (Brunswick 2, North Anna 2 and Cook) 3 / (70 * 5.5) = **7.79E-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	3.2E-03	1	3.2E-03
		avg. 5 – 10	7.8E-03	avg. 5 – 10	7.8E-03
		15	2.1E-02	15	2.1E-02
		15-year avg = 9.41E-03/year		**15-year avg = 9.41E-03/year**	
3	**Increase in Flaw Likelihood Between 3 and 15 Years** Uses age-adjusted liner flaw likelihood (step 2).	**13.1%**		**13.1%**	

Step	Description	Containment Cylinder and Dome		Containment Basemat	
		Pressure (psia)	Likelihood of Breach	Pressure (psia)	Likelihood of Breach
4	**Likelihood of Breach in Containment Given Liner Flaw** The upper end pressure is consistent with the D. C. Cook PRA Level 2 analysis. 0.1% is assumed for the lower end. Intermediate failure likelihood's are determined through logarithmic interpolation. Basemat is assumed to be 1/10 of cylinder/dome region.	20 27.2 (ILRT) 40 50.7	0.10% 0.505% 9.00% 100%	20 27.2 (ILRT) 40 50.7	0.01% 0.0505% 0.900% 10.0%
5	**Visual Inspection Detection Failure Likelihood**	**47.4%** 41.5% of surface cannot be inspected plus assumed 10% failure rate for the 58.5% that can be inspected.		**100%** Cannot be visually inspected.	
6	**Likelihood of Non-Detected Containment Leakage** (Setps 3*4*5)	**0.0314%** (13.1% * 0.505% * 47.4%)		**0.00662%** (13.1% * 0.0505% * 100%)	
	Total Likelihood of Non-Detected Containment Leakage Sum of contributions from cylinder/dome and basemat regions	**0.0380%** (0 0314% + 0.0066%)			
	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.	**9.14E-09 per year** (0.000380 * 2.404E-05)			

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		
Original Submittal Table R1 (Reference 17)	0.105	0.0439
Table R1 Based on Updated PRA (Reference 2)	0.0294	0.0122
Including Liner Corrosion	0.0297	0.0124
The percentage increase in person-rem/year risk		
Original Submittal Table R1 (Reference 17)	0.06%	0.025%
Table R1 Based on Updated PRA (Reference 2)	0.028%	0.012%
Including Liner Corrosion	0.028%	0.012%
Change in LERF (per year)		
Original Submittal Table R1 (Reference 17)	8.2E-08	3.4E-08
Table R1 Based on Updated PRA (Reference 2)	1.23E-07	5.14E-08
Including Liner Corrosion	1.32E-07	5.70E-08
Change in the Conditional Containment Failure Probability		
Original Submittal Table R1 (Reference 17)	1.1% (0.011)	0.47% (0.0047)
Table R1 Based on Updated PRA (Reference 2)	0.0025	0.0011
Including Liner Corrosion	0.0029	0.0013

14.0 CONCLUSIONS

For the above results it is concluded that age-related liner corrosion has 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 1 test in 10 years to 1 test in 15 years from 5.14E-08/yr to 5.70-08/yr and that due to a change in interval from 3 tests in 10 years to 1 test in 15 years from 1.23E-7/yr to 1.32E-07/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 changes in LERF and is non-risk significant. The relative results of the sensitivity studies performed in Reference 9 provide further support for the above conclusions.