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A Progress Energy Company

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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING
ONE-TIME EXTENSION OF CONTAINMENT TYPE A TEST INTERVAL

Ladies and Gentlemen:

In accordance with the provisions of the Code of Federal Regulations, Title 10 (10 CFR), Part 50.90, Carolina Power & Light (CP&L) Company is submitting a request for an amendment to the Technical Specifications (TS) for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. The proposed amendment would permit a one-time, five-year extension of the 10-year performance-based Type A test interval established in NEI 94-01, "Nuclear Energy Institute Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0, July 26, 1995.

Specifically, the proposed change would revise TS 5.5.16, "Containment Leakage Rate Testing Program," to require the performance of a Type A test within 15 years from the last Type A test, performed on April 9, 1992. The proposed change is supported by a plant-specific risk assessment. This risk assessment considered the guidance provided by NEI 94-01, Electric Power Research Institute (EPRI) Topical Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals;" NRC Regulatory Guide (RG) 1.174, "An Approach For Using Probabilistic Risk Assessments In Risk-Informed Decisions On Plant-Specific Changes To The Licensing Basis," dated July 1998; and letters from NEI to NEI Administrative Points of Contact, dated November 13, 2001, and November 30, 2001, providing additional guidance on one-time extensions of containment integrated leak rate (Type A) test interval.

Attachment I provides an affidavit as required by 10 CFR 50.30(b).

A001

Robinson Nuclear Plant
3581 West Entrance Road
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Attachment II provides a description of the current condition, a description of the proposed change, a safety assessment of the proposed change, a discussion of a finding of no significant hazards, and an environmental impact determination.

Attachment III provides a markup of the current TS page.

Attachment IV provides retyped page for the proposed TS.

Attachment V provides RSC 02-12, "Risk Significance of ILRT Extension Based on NEI Guidance."

Attachment VI provides Appendix A of RSC 02-12, consisting of RSC 01-44, "Surrogate Level 3 Evaluation Methodology," proprietary version.

The information contained in Attachment VI is considered by the preparer to contain, in part, proprietary information. The preparer requests exemption from public disclosure in accordance with 10 CFR 2.790(b). Attachment VII contains an affidavit and application for withholding from public disclosure executed by Mr. Ricky Summitt, President of Ricky Summitt Consulting, Inc., who is authorized to apply for the withholding of the proprietary information for Ricky Summit Consulting, Inc.

Attachment VIII provides Appendix A of RSC 02-12, consisting of RSC 01-44NP, "Surrogate Level 3 Evaluation Methodology," non-proprietary version.

In accordance with 10 CFR 50.91(b), CP&L is providing the State of South Carolina with a copy of the proposed license amendment.

The current 10-year period for the performance of the Type A test ends on April 9, 2002. However, CP&L has applied the guidelines of NEI 94-01 to extend the test interval for HBRSEP, Unit No. 2. The next Type A test is scheduled to be performed during Refueling Outage (RO)-21, which is currently scheduled to begin on October 12, 2002. To support incorporation of the Type A testing changes into the schedule for the upcoming RO-21, CP&L requests approval of the proposed license amendment by August 1, 2002, with the amendment being implemented within 60 days of issuance.

If you have any questions concerning this matter, please contact Mr. C. T. Baucom.

Sincerely,

B. L. Fletcher III
Manager - Regulatory Affairs

CWS/cws

Attachments

I. Affidavit
II. Request For Technical Specifications Change Regarding One-Time Extension of Containment Type A Test Interval
III. Markup of Technical Specifications Page
IV. Retyped Technical Specifications Page
V. RSC 02-12, "Risk Significance of ILRT Extension Based on NEI Guidance"
VI. RSC 02-12, Appendix A, RSC 01-44, "Surrogate Level 3 Evaluation Methodology," proprietary version
VII. Affidavit and Application for Withholding from Public Disclosure
VIII. RSC 02-12, Appendix A, RSC 01-44NP, "Surrogate Level 3 Evaluation Methodology," non-proprietary version

c (w/o Attachment VI):

Mr. H. J. Porter, Director, Division of Radioactive Waste Management (SC)
Mr. R. M. Gandy, Division of Radioactive Waste Management (SC)
Mr. L. A. Reyes, NRC, Region II
Mr. R. Subbaratnam, NRC, NRR
NRC Resident Inspector, HBRSEP
Attorney General (SC)

Affidavit

State of South Carolina
County of Darlington

J. W. Moyer, having been first duly sworn, did depose and say that the information contained in letter RNP-RA/02-0028 is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, contractors, and agents of Carolina Power & Light Company.

JMoyer
____

Sworn to and subscribed before me

This 26 day of MARCH 2002

Peter N Yarter
____
Notary Public for South Carolina

My commission expires: Sept 13, 2009

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING ONE-TIME EXTENSION OF CONTAINMENT TYPE A TEST INTERVAL

Description of Current Condition

H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, Technical Specification (TS) 5.5.16 requires the Containment Leakage Rate Testing Program to be in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. RG 1.163 endorses NEI 94-01, "Nuclear Energy Institute Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0, July 26, 1995, as a method acceptable to the NRC staff for complying with the provisions of 10 CFR Part 50, Appendix J, Option B.

RG 1.163 also states that "...licensees intending to comply with Option B in the amendment to Appendix J should establish test intervals based on the criteria in Section 11.0 of NEI 94-01, rather than the test intervals specified in ANSI/ANS-56.8-1994." NEI 94-01, Section 11.0, refers to Section 9.0 for guidance on extended test intervals. Section 9.2.3 of NEI 94-01 states that "Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per 10 years based on acceptable performance history. Acceptable performance history is defined as completion of two consecutive periodic Type A tests where the calculated performance leakage rate was less than 1.0 L_a."

The current 10-year interval for the performance of the Type A test ends on April 9, 2002. However, Carolina Power and Light (CP&L) Company has applied the guidelines of NEI 94-01 to extend the test interval for HBRSEP, Unit No. 2. NEI 94-01, Section 9.1, provides guidance for extending the recommended Type A test interval by up to 15 months in cases where refueling schedules have been changed to accommodate other factors. The next Type A test is currently scheduled to be performed during Refueling Outage (RO)-21 which is scheduled to begin on October 12, 2002.

Description of the Proposed Change

The proposed change would revise TS 5.5.16 to allow a one-time exception to the 10-year interval of the performance-based leakage rate testing program for Type A tests as required by NEI 94-01. The one-time exception applies to the requirement of NEI 94-01, Section 9.2.3, to perform Type A testing at an interval of up to ten years, with allowance for a 15-month extension. The exception will require Type A testing within 15 years from the last Type A test, which was performed on April 9, 1992.

TS 5.5.16 is revised by the addition of ", as modified by the following exception:" to the end of the second sentence of the first paragraph. The first paragraph is reformatted and the following exception added: " a. NEI 94-01 - 1995, Section 9.2.3: The first Type A test performed after the April 9, 1992, Type A test shall be performed no later than April 9, 2007."

Safety Assessment

The HBRSEP, Unit No. 2, containment structure is a steel lined concrete shell in the form of a vertical right cylinder with a hemispherical dome, as described in the Updated Final Safety Analysis Report (UFSAR), Section 3.8.1. The containment structure is designed for an accident pressure based upon the pressure transients as described in the UFSAR, Section 15.6, and is designed to contain radioactive material which might be released from the core following a loss-of-coolant accident.

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage through the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in the TS. Containment structure testing is intended to assure the leak-tight integrity of the containment structure under all design basis conditions.

10 CFR 50, Appendix J, was revised, effective October 26, 1995, to allow licensees to choose containment leakage testing under Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." On May 28, 1996, the NRC issued Amendment 169 to the HBRSEP, Unit No. 2, TS to permit implementation of 10 CFR 50, Appendix J, Option B, for Type A tests; and 10 CFR 50, Appendix J, Option A, for Type B and C tests. The amendment added TS 6.12, "Containment Leakage Rate Testing Program," which required Type A testing in accordance with NRC Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. TS 6.12 was subsequently changed to TS 5.5.16 when HBRSEP, Unit No. 2, adopted the Improved Standard Technical Specifications (Amendment 176, dated October 24, 1997). RG 1.163 specifies a method acceptable to the NRC staff for complying with the provisions of Option B by approving the use of NEI 94-01 and ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements," subject to several regulatory positions in the guide.

Exceptions to the requirements of RG 1.163 are allowed by 10 CFR 50, Appendix J, Option B, Section V.B, "Implementation," which states "The Regulatory Guide or other implementing document used by a licensee, or applicant for an operating license, to develop a performance-based leakage-testing program must be included, by general reference, in the plant technical specifications. The submittal technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide." Therefore, this application does not require an exemption to Option B.

Adoption of the Option B performance-based containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing was performed. However, the frequency of measuring containment leakage in Type A tests was revised. This frequency is based on an evaluation of the "as-found" leakage history to determine whether the frequency for leakage testing provides assurance that leakage limits will be maintained.

The current HBRSEP, Unit No. 2, frequency for Type A testing was based on a generic evaluation documented in NUREG-1493, "Performance-Based Containment Leakage-Test Program." NUREG-1493 made the following observations with regard to decreasing the test frequency:

- Reducing the Type A (i.e., ILRT) testing frequency to one per twenty years was found to lead to an imperceptible increase in risk. The estimated increase in risk is small because ILRTs identify only a few potential leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above the existing requirements. Given the insensitivity of risk to containment leakage rate testing, and the small fraction of leakage detected solely by Type A testing, increasing the interval between ILRT tests had minimal impact on public risk.

- While Type B and C tests identify the vast majority (greater than 95%) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall risk under existing requirements, the overall effect is very small.

As previously stated, Type B and C testing is performed under Option A of 10 CFR 50, Appendix J. Type B testing is applied to air lock door seals which are part of the containment pressure boundary, doors with resilient seals or gaskets except for seal-welded doors, those containment penetrations whose design incorporates resilient seals or sealant compounds, piping penetrations fitted with expansion bellows, and electrical penetrations fitted with flexible metal seal assemblies. Type C testing is applied to those containment isolation valves that:

- Provide a direct connection between the inside and outside atmospheres of the containment vessel under normal operation,

- Are required to close automatically on receipt of a containment isolation signal in response to controls intended to effect containment isolation,

- Are required to operate intermittently under post accident conditions, or

- Are in penetrations that are part of the reactor coolant pressure boundary or communicate directly with containment atmosphere normally or as a result of a loss-of-coolant accident.

10 CFR 50, Appendix J, Option B, Test Information

The surveillance frequency for Type A testing in NEI 94-01 is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart where the calculated leakage rate was less than 1.0 L_a) as described in Section 9.2.3. Based on the April 1987 and April 1992 Type A tests, the current interval for HBRSEP, Unit No. 2, is once per ten years.

The results for the last two Type A tests for HBRSEP, Unit No. 2, are as follows:

Date	As-Found Leakage (% wt./day)		Test Pressure (psig)	Acceptance Limit (% wt./day)
	Mass Point Method	Total Time Method		
April 8, 1987	0.041	0.036	21.0	0.053
April 9, 1992	0.0602	0.0644	42.0	0.075

As discussed in the HBRSEP, Unit No. 2, letter dated January 31, 1996, that requested changes to the TS to allow implementation of performance-based Type A testing, previous Type A test results and conclusions substantiate that the reliability and performance of the containment design has not been challenged. The additional margin of safety from supporting containment isolation systems (i.e., Type B and Type C tests) provided additional justification for the implementation of 10 CFR 50, Appendix J, Option B, for Type A testing. The Type B and C testing will continue to be performed in accordance with 10 CFR 50, Appendix J, Option A, as required by TS 5.5.16.

Containment Inspection Program Activities

CP&L has implemented a Containment Inspection Program in conformance with 10 CFR 50.55a(g)(6)(ii)(B). The Containment Inspection Program has been established in accordance with Subsections IWE and IWL of the American Society of Mechanical Engineers (ASME), Section XI, 1992 Edition through the 1992 Addenda, to assure detection of deterioration affecting containment integrity. The first interval of the HBRSEP, Unit No. 2, Containment Inspection Program began in September 1998, and ends in September 2008.

10 CFR 50.55a(g)(6)(ii)(B) required that expedited examinations of containment be completed by September 9, 2001. Visual examinations of the containment structure were conducted on 100% of the accessible surfaces between 1998 and 2001, and were performed to meet the requirements of ASME Section XI, Subsections IWE and IWL. These examinations consisted of a general visual examination of accessible areas of the containment vessel liner (Pressure Boundary) for IWE and the reinforced concrete exterior (Structural Integrity) for IWL. Although the containment vessel liner between the floor and the containment vessel dome is insulated and not typically accessible, numerous sections of insulation were removed

over the last three refueling outages which allowed VT-3 examinations of portions of the containment vessel liner.

RG 1.163, Regulatory Position C.3, specifies that examinations of the accessible surfaces of the containment for detection of structural problems should be conducted prior to initiating a Type A test and during two other outages before the next Type A test if the interval for the Type A test has been extended to 10 years, in order to allow for early detection of evidence of structural deterioration. These visual examinations have been completed with no significant defects noted.

In accordance with IWE-1240, an engineering evaluation was developed to determine which containment surface areas required augmented examinations. The only component categorized as Category E-C (augmented examinations) at HBRSEP, Unit No. 2, was the equipment hatch cylinder. Corrosion was observed in 1999, during RO-19, on a small area of the bottom of the interior portion of the cylinder. The bottom portion of the equipment hatch was categorized as "augmented," because a majority of this portion of the cylinder is insulated and not visible. The insulation was removed to allow examination of the bottom portion of the cylinder interior in 2000, during RO-20. The maximum amount of corrosion observed on the cylinder that has a 1" nominal thickness was 1/16". The maximum amount of corrosion observed on the area of the cylinder that has 3-1/2" nominal thickness was 5/16". This degradation was determined to be acceptable without repair.

The NRC has granted nine requests for relief from certain requirements of the ASME Boiler and Pressure Code, Section XI, involving containment inspections at HBRSEP, Unit No. 2. Each of these relief requests was reviewed to assess the effect, if any, resulting from the proposed Type A test interval extension, as follows:

- Relief Request IWE/IWL-01 obtained relief from Section XI, 1992 Edition, 1992 Addenda, Table IWE-2500-1, "Examination Categories," Examination Category E-A, which requires a visual examination of 100% of the accessible surface areas of containment in accordance with paragraphs IWE-3510.2, "Visual Examinations on Coated Areas," and IWE-3510.3, "VT-3 Visual Examinations on Non-Coated Areas," for Class MC, and metallic liners of Class CC components. The relief permits an alternative which involves the performance of a VT-3 examination on those portions of the insulated containment liner that are exposed when a maintenance activity requires removal of the liner insulation. NRC letter dated July 26, 1999, granted this relief to HBRSEP, Unit No. 2. The proposed Type A testing interval change does not affect the visual inspections of the relief alternative. As a result, the relief request remains valid and unaffected by the proposed change.

- Relief Request IWE/IWL-02 obtained relief from Section XI, 1992 Edition, 1992 Addenda, Table IWE-2500-1, "Examination Categories," Examination Category E-D, which requires a visual examination of the containment moisture barrier, in accordance with Figure IWE-2500-2, "Examination Areas for Moisture Barriers," for

Class MC, and metallic liners of Class CC components. The relief permits an alternative which involves a VT-3 examination on those portions of the 228-foot elevation concrete-metal liner moisture barrier that are exposed when a maintenance activity requires removal of the associated barrier. NRC letter dated July 26, 1999, granted this relief to HBRSEP, Unit No. 2. The proposed Type A testing interval change does not affect the visual inspections of the relief alternative. As a result, the relief request remains valid and unaffected by the proposed change.

- Relief Request IWE/IWL-03 obtained relief from Section XI, 1992 Edition, 1992 Addenda, Table IWE-2500-1, "Examination Categories," Examination Category E-G, which requires a visual examination of the surfaces of bolted connections in accordance with the acceptance standard of IWE-3515, "Standards for Examination Category E-G, Pressure Retaining Bolting." The relief permits the use of an alternative acceptance standard for the examination of bolted connections through the use of a performance criterion based on conditions that may cause the bolted connection to violate either containment leak-tightness or structural integrity. NRC letter dated July 26, 1999, granted this relief to HBRSEP, Unit No. 2. The proposed Type A testing interval change does not affect the visual inspections of the performance criteria of the relief alternative. As a result, the relief request remains valid and unaffected by the proposed change.

- Relief Request IWE/IWL-04 obtained relief from Section XI, 1992 Edition, 1992 Addenda, Table IWE-2500-1, "Examination Categories," Examination Category E-D, which requires a visual examination of 100% of the containment seals and gaskets for Class MC pressure-retaining components and metallic shell and penetration liners of Class CC components. The relief permits functional testing of containment seals and gaskets through the performance of 10 CFR 50, Appendix J, testing, rather than by individual visual inspection. NRC letter dated July 26, 1999, granted this relief to HBRSEP, Unit No. 2. The proposed Type A testing interval change does not affect the Type B testing requirements of the relief alternative. As a result, the relief request remains valid and unaffected by the proposed change.

- Relief Request IWE/IWL-05 obtained relief from Section XI, 1992 Edition, 1992 Addenda, Paragraph IWE-2420, "Successive Inspections," Items (b) and (c), which require successive examinations of repaired areas in accordance with Table IWE-2500-1, Examination Category E-C. The relief permits the use of the process and acceptance examinations and evaluations required by the Code, in lieu of the successive examination requirements of IWE-2420, Items (b) and (c). NRC letter dated July 26, 1999, granted this relief to HBRSEP, Unit No. 2. The proposed Type A testing interval change does not affect the process and acceptance examinations and evaluations required by the relief alternative. As a result, the relief request remains valid and unaffected by the proposed change.

- Relief Request IWE/IWL-06 obtained relief from Section XI, 1992 Edition, 1992 Addenda, Table IWE-2500-1, "Examination Categories," Examination Category E-G, which requires a 100% VT-1 examination of the surfaces of pressure-retaining bolted connections in Class MC, and in the metallic liners of Class CC components. The relief permits continued acceptance of bolted connections associated with containment through the performance of general visual examinations and 10 CFR 50, Appendix J testing, rather than by VT-1 visual examination. NRC letter dated July 26, 1999, granted this relief to HBRSEP, Unit No. 2. The proposed Type A testing interval change does not affect the Type B testing requirements of the relief alternative. As a result, the relief request remains valid and unaffected by the proposed change.

- Relief Request IWE/IWL-07 obtained relief from Section XI, 1992 Edition, 1992 Addenda, Table IWE-2500-1, "Examination Categories," Examination Category E-G, which provides requirements for a torque or tension test of bolted connections that have not been disassembled and reassembled during the inspection interval. The relief permits the leaktight integrity of bolted connections that are required for containment vessel leaktight integrity to be verified in accordance with the performance of 10 CFR 50, Appendix J testing, rather than untorquing and re-torquing bolted connections. NRC letter dated July 26, 1999, granted this relief to HBRSEP, Unit No. 2. The proposed Type A testing interval change does not affect the Type B testing requirements of the relief alternative. As a result, the relief request remains valid and unaffected by the proposed change.

- Relief Request IWE/IWL-08 obtained relief from Section XI, 1992 Edition, 1992 Addenda, Table IWE-2500-1, "Examination Categories," Examination Category E-A, which requires a 100% VT-3 examination of accessible surface areas of the containment vessel. The relief permits the performance of a general visual examination in accordance with Paragraph IWE-3510.1 of the accessible surface areas of the containment, in lieu of the VT-3 examination. When evidence of degradation is detected, a detailed visual examination will be performed to determine the magnitude and extent of any distress and deterioration of suspect containment surfaces. NRC letter dated July 26, 1999, granted this relief to HBRSEP, Unit No. 2. The proposed Type A testing interval change does not affect the visual examination requirements of the relief alternative. As a result, the relief request remains valid and unaffected by the proposed change.

- Relief Request IWE/IWL-09 obtained relief from Section XI, 1992 Edition, 1992 Addenda, Table IWE-2500-1, "Examination Categories," Examination Category L-A, which requires visual examinations of concrete surfaces of the containment in accordance with the requirements of Paragraph IWL-2510. The relief permits the performance of a general visual examination of the concrete surfaces, in lieu of the requirements for a VT-1C and VT-3C examination. When evidence of degradation is detected, a detailed visual examination will be performed of the suspect area. If a detailed visual examination cannot be performed, the suspect area will be evaluated and approved by a Registered Professional Engineer (RPE). NRC letter dated July 26, 1999, granted this relief to HBRSEP, Unit No. 2. The proposed Type A testing interval change does not affect the visual examination requirements of the relief alternative. As a result, the relief request remains valid and unaffected by the proposed change.

For potential degradation of the uninspectible (embedded) side of the containment vessel liner, CP&L has performed limited Ultrasonic Testing (UT) on some of the liner panels. For those liner panels that have been visible after removal of insulation and sheathing, ultrasonic measurements have been made on approximately one foot centers for the approximately 3'8" by 7'8" panels. Approximately 100 panels have been examined in this manner. These measurements did not indicate degradation of the embedded side of the containment vessel liner. It should be noted that these measurements represent only a sampling of the overall number of containment liner panels.

The following table provides a history of IWE and IWL examinations since the last Type A test:

Outage	Examinations Performed
RO-18 (1998)	Portions of the containment vessel liner behind the insulation.
RO-19 (1999)	Portions of the containment vessel liner behind the insulation, electrical penetrations, airlock, and portions of the reinforced concrete exterior.
RO-20 (2001)	Portions of the containment vessel liner behind the insulation, the dome interior, mechanical penetrations, equipment hatch, and the remaining portions of the reinforced concrete exterior, including the dome exterior.

The conclusions of these IWE and IWL examinations, which were completed in May 2001 (RO-20), provide assurance that the structural integrity and leak-tightness of the HBRSEP, Unit No. 2, containment have not been compromised. Continued performance of these examinations provides a high degree of assurance that any degradation of the containment structure will be detected and corrected before it can produce a containment leakage path or impact structural integrity.

Plant-Specific Risk Assessment for the Extended Type A Test Interval

A plant-specific risk assessment has been performed considering the guidance provided by NEI 94-01, Electric Power Research Institute (EPRI) Topical Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals;" NRC RG 1.174, "An Approach For Using Probabilistic Risk Assessments In Risk-Informed Decisions On Plant-Specific Changes To The Licensing Basis," dated July 1998; and letters from NEI to NEI Administrative Points of Contact, dated November 13, 2001, and November 30, 2001, providing additional guidance on one-time extensions of the containment integrated leak rate test interval. The approach used in this assessment is also similar to assessments provided in support of similar Technical Specifications changes for the Indian Point 3 and Crystal River 3 plants, which were approved by the NRC in safety evaluations dated April 17, 2001, and August 30, 2001, respectively. A copy of the HBRSEP, Unit No. 2, risk assessment (RSC 02-12, Revision 0, February 2002) has been included as Attachment V. The release category and person-rem information are based on design basis leakage evaluations and extrapolation of the release category information using a modeling approach that is described in Appendix A of RSC 02-12 (RSC 01-44, Revision 0, dated August 2001). A proprietary copy of RSC 01-44 has been provided as Attachment VI. A non-proprietary version (RSC 01-44NP) has been provided as Attachment VIII.

The assessment explicitly includes the potential for containment leakage. By definition, the intact containment cases, EPRI Containment Failure Class 1, include a leakage term that is independent of the source of the leak. Similarly, the Containment Failure Class 3a and 3b cases model the potential leakage impact of the Type A test interval extension. These cases include the potential that the leakage is due to a containment liner failure.

Based on the assessment performed in accordance with the guidance provided in the NEI letter dated November 13, 2001, the following can be stated:

- The person-rem/year increase in risk contribution from the one-time extension of the Type A test interval from 10 years to 15 years is 4.13E-2 person-rem/year.

- The one-time change in Type A test frequency from once per 10 years to once per 15 years increases the risk impact on the total integrated plant risk by only 0.0464%. Therefore, the risk impact when compared to other severe accident risks is negligible.

- The risk increase in Large Early Release Frequency (LERF) from the one-time extension of the Type A test interval from 10 years to 15 years is 1.95E-7/year.

- The change in Conditional Containment Failure Probability (CCFP) from the one-time extension of the Type A test interval from 10 years to 15 years is 0.455%.

RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines "very small changes" in risk as resulting in increases of Core Damage Frequency (CDF) below 1E-6/year and increases in LERF below 1E-7/year. The relevant criterion is LERF, because the Type A containment leakage rate testing does not impact CDF. The increase in LERF associated with a one-time change in Type A test interval from 10 years to 15 years is 1.95E-7/year. Based on the guidance of RG 1.174, this change in LERF is considered to constitute a "small change" in risk. For the purposes of tracking cumulative changes in risk impact, the total change in LERF from the original Type A test frequency of three per 10 years to once per 15 years is 4.69E-7/year.

RG 1.174 states that applications involving an increase in the calculated LERF in the range of 1E-7/year to 1E-6/year will be considered only if it can be reasonably shown that the total LERF is less than 1E-5/year. The total LERF from internal events, including internal flooding, for HBRSEP, Unit No. 2, is 4.93E-6/year. Including the contribution for the LERF associated with the once per 10 year test interval and the contribution for the LERF associated with this one-time change results in a total internal LERF of 5.52E-6/year.

Another factor to consider in total LERF is the impact of other initiating events that are not included in the HBRSEP, Unit No. 2, Probabilistic Safety Assessment. Events at shutdown are not of consequence, because containment status is typically unisolated and any event would be relatively slow acting and provide opportunity for mitigation. Other external events, i.e., internal fire, seismic, high winds, could be of some consequence and should be considered. An evaluation of these other events is supplied in RSC 02-12. This evaluation showed that, considering the one-time change in Type A interval and these other external events, the total LERF would increase to 5.72E-6/year. This value indicates that the proposed change would not result in an unacceptable increase in total LERF as defined by RG 1.174.

RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with defense-in-depth philosophy is maintained by demonstrating that the balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in CCFP was estimated to be 0.455% for the proposed change. This change is small and demonstrates that the defense-in-depth philosophy is maintained.

The November 30, 2001, NEI letter provides additional guidance for those plants finding that the method described in the November 13, 2001, NEI letter results in ΔLERF impact above the "very small change" guidelines of RG 1.174. Additional evaluations incorporating the guidance of the November 30, 2001, NEI letter are provided in RSC 02-12. These evaluations concluded that the proposed one-time change in Type A test interval from 10 years to 15 years results in a ΔLERF of 9.88E-8/year. This results in a ΔLERF that is within the RG 1.174 "very small change" category.

Although the proposed change is justified by the assessment incorporating the improvements contained in NEI letters dated November 13, 2001, and November 30, 2001, an additional assessment consistent with that supplied for Crystal River 3 was performed for HBRSEP, Unit No. 2. The Crystal River 3 assessment methodology was found to be acceptable by the NRC in a safety evaluation dated August 30, 2001. This additional assessment determined the change in LERF associated with the one-time extension of the Type A test interval from 10 years to 15 years to be 5.10E-8/year, well within the "very small change" category of RG 1.174. This provides further assurance that the assessment of this change was performed using conservative assumptions and is therefore acceptable.

No Significant Hazards Consideration Determination

Carolina Power & Light (CP&L) Company is proposing a change to the Appendix A, Technical Specifications (TS), of Facility Operating License No. DPR-23, for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. This change will revise the requirements of TS 5.5.16, "Containment Leakage Rate Testing Program," to incorporate a one-time extension to the 10-year interval of the performance-based leakage rate testing program for Type A tests specified by Nuclear Energy Institute (NEI) 94-01, "Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0, July 26, 1995. The revised TS will allow a Type A test to be performed within 15 years from the last Type A test. The last HBRSEP, Unit No. 2, Type A test was performed on April 9, 1992. The proposed exception will require performance of the next HBRSEP, Unit No. 2, Type A test no later than April 9, 2007.

An evaluation of the proposed change has been performed in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards considerations using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this amendment request follows:

1. The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

   The proposed change to TS 5.5.16 provides a one-time extension to the testing interval for Type A (containment integrated leak rate) testing. The existing 10-year test interval is based on past test performance. The proposed TS change provides a one-time extension of the Type A test interval to 15 years for HBRSEP, Unit No. 2. The proposed TS change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The containment vessel is designed to provide a leak tight barrier against the uncontrolled release of radioactivity to the environment in the unlikely event of postulated accidents. As such, the containment vessel is not considered as the initiator of an accident. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

   The proposed change involves only a one-time change to the interval between Type A containment leakage tests. Type B and C leakage testing will continue to be performed at the interval specified in 10 CFR Part 50, Appendix J, Option A, as currently required by the HBRSEP, Unit No. 2, TS. As documented in NUREG-1493, "Performance-Based Containment Leakage-Test Program," industry experience has shown that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. In fact, an analysis of 144 integrated leak rate tests results, including 23 failures, found that none of the failures involved containment liner breach. NUREG-1493 also concluded, in part, that reducing the frequency of Type A containment leakage rate testing to once per 20 years was found to lead to an imperceptible increase in risk. The HBRSEP, Unit No.2, test history and risk-based evaluation of the proposed extension to the Type A test interval supports this conclusion. The design and construction requirements of the containment vessel, combined with the containment inspections performed in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI, and the Maintenance Rule (i.e., 10 CFR 50.65) provide a high degree of assurance that the containment vessel will not degrade in a manner that is detectable only by Type A testing. Therefore, the proposed Technical Specification change does not involve a significant increase in the consequences of an accident previously evaluated.

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

2. The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated.

The proposed change to TS 5.5.16 provides a one-time extension to the testing interval for Type A (containment integrated leak rate) testing. The existing 10-year test interval is based on past test performance. The proposed TS change will provide a one-time extension of the Type A test interval to 15 years for HBRSEP, Unit No. 2. The proposed change to the Type A test interval does not result in any physical changes to HBRSEP, Unit No. 2. In addition, the proposed test interval extension does not change the operation of HBRSEP, Unit No. 2, such that a failure mode involving the possibility of a new or different kind of accident from any accident previously evaluated is created.

Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety.

The proposed change to TS 5.5.16 provides a one-time extension to the testing interval for Type A (containment integrated leak rate) testing. The existing 10-year test interval is based on past test performance. The proposed TS change will provide a one-time extension of the Type A test interval to 15 years for HBRSEP, Unit No. 2. The NUREG-1493 generic study of the effects of extending containment leakage testing found that a 20 year extension for Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG-1493 found that, generically, the design containment leakage rate contributes a very small amount to the individual risk, and that the decrease in Type A testing frequency would have a minimal affect on this risk, because most potential leakage paths are detected by Type B and C testing.

The proposed change involves only a one-time extension of the interval for Type A containment leakage testing; the overall containment leakage rate specified by the HBRSEP, Unit No. 2, Technical Specifications is being maintained. Type B and C containment leakage testing will continue to be performed at the frequency required by the HBRSEP, Unit No. 2, Technical Specifications. The regular containment inspections being performed in accordance with ASME, Section XI, and the Maintenance Rule (i.e., 10 CFR 50.65) provide a high degree of assurance that the containment will not degrade in a manner that is only detectable by Type A testing. In addition, a plant-specific risk evaluation has demonstrated that the one-time extension of the Type A leakage test interval from 10 years to 15 years results in only a very small increase in risk for those accident sequences influenced by Type A testing.

Therefore, this change does not involve a significant reduction in a margin of safety.

Based on the above discussion, CP&L has determined that the requested change does not involve a significant hazards consideration.

Environmental Impact Consideration

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions for categorical exclusion for performing an environmental assessment. A proposed change for an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed change would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite; (3) result in an increase in individual or cumulative occupational radiation exposure. Carolina Power and Light (CP&L) Company has reviewed this request and determined that the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows:

Proposed Change

The H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, Technical Specification 5.5.16, "Containment Leakage Rate Testing Program," is revised to incorporate a one-time extension to the Type A containment leakage rate test interval. The proposed change will allow a Type A test to be performed within 15 years from the last Type A test. The last HBRSEP, Unit No. 2, Type A test was performed on April 9, 1992. The proposed exception will require performance of the next Type A test no later than April 9, 2007.

Basis

The proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

1. As demonstrated in the No Significant Hazards Consideration Determination, the proposed change does not involve a significant hazards consideration.

2. As demonstrated in the No Significant Hazards Consideration Determination, the proposed change does not result in a significant increase in the consequences of an accident previously evaluated and does not result in the possibility of a new or different kind of accident. Therefore, the proposed change does not result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite.

3. The proposed change does not involve physical plant changes, or introduce any new mode of plant operation. Therefore, the proposed change does not result in an increase in individual or cumulative occupational radiation exposures.

United States Nuclear Regulatory Commission
Attachment III to Serial: RNP-RA/02-0028
2 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING
ONE-TIME EXTENSION OF CONTAINMENT TYPE A TEST INTERVAL

MARKUP OF TECHNICAL SPECIFICATIONS PAGE

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program

This program provides controls for implementation of the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions for Type A testing. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. Type B and C testing shall be implemented in the program in accordance with the requirements of 10 CFR 50, Appendix J, Option A.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 40.5 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of the containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criteria is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and Type C tests, and $\leq$ 0.75 L_a for Type A tests.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

, as modified by the following exception:

a. NEI 94-01 - 1995, Section 9.2.3: The first Type A test performed after the April 9, 1992, Type A test shall be performed no later than April 9, 2007

United States Nuclear Regulatory Commission
Attachment IV to Serial: RNP-RA/02-0028
2 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING
ONE-TIME EXTENSION OF CONTAINMENT TYPE A TEST INTERVAL

RETYPED TECHNICAL SPECIFICATIONS PAGE

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program

This program provides controls for implementation of the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions for Type A testing. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception:

a. NEI 94-01 - 1995, Section 9.2.3: The first Type A test performed after the April 9, 1992, Type A test shall be performed no later than April 9, 2007.

Type B and C testing shall be implemented in the program in accordance with the requirements of 10 CFR 50, Appendix J, Option A.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 40.5 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of the containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

(continued)

United States Nuclear Regulatory Commission
Attachment V to Serial: RNP-RA/02-0028
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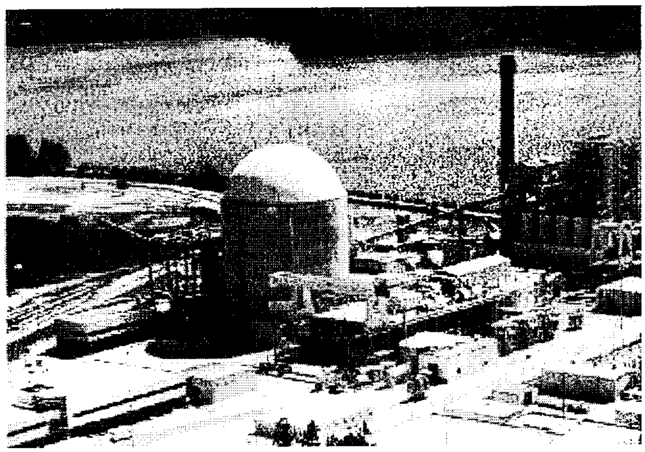
H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING
ONE-TIME EXTENSION OF CONTAINMENT TYPE A TEST INTERVAL

RSC 02-12, "RISK SIGNIFICANCE OF ILRT EXTENSION BASED ON NEI GUIDANCE"

Carolina Power and Light RSC 02-12

Robinson Nuclear Plant
Probabilistic Safety Assessment



Risk Significance of ILRT Extension based on NEI Guidance

Revision 0

February 2002

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Preparer:	Ricky Summitt
RSC Reviewer: J. Miller	Date: February 12, 2002
RSC Approver: R. Summitt	Date: February 13, 2002

Abstract (brief statement of purpose): Develop a risk informed analysis of the impact of extending the Robinson Nuclear Plant ILRT using the NEI interim guidance documented in revision 4

Documentation Retrieval Information:

Keywords: Special Analysis   Level 2 Analysis   Internal Events

☐ Amends / ☐ Supercedes / ☐ Supplements RSC Document(s):

Verification and Review Method:
☒ Detailed Review   ☐ Alternative Calculation
☐ Qualification Testing   ☐ Other (specify:)

General Documentation Requirements	Acceptable	Reviewer Comments
Introduction – provides summary of purpose, scope, and principle tasks required to meet objective	☒	Note format not standard to meet prior submittal format.
Methodology – description of process and supporting methodology that is sufficient to understand approach and to support peer review	☒	
Analysis and Results – detailed documentation of the implementation of the methodology and task steps that may be supported by report appendices and includes intermediate and final results	☒	
Conclusions and Recommendations – concise presentation of results of the analysis that answers the objectives of the study and should include any important assumptions and/or findings	☒	NA

Editorial Review:
☒ Spell Checked   ☒ Grammar Checked
☒ Tables and Figures Checked   ☒ Sections Checked

Sufficient References to Reproduce Results: Yes (with Excel sheets)

Resolved all Comments: Yes   Incorporated Resolutions From Review: Yes

Reviewer Comment	Resolution of Comment
1. The following PDS bins do not appear in the new data: 13P, 5E and 13Q. New bins appear: 12C, B5B, B10N, 3G, 12F, 4K, 2C, 2E, 2F and B10E.	Consistent with new data provided by CP&L.
2. The value for the adjusted PDS 2A in equation 21 is incorrect.	Value corrected.
3. The value for non-LERF is incorrect in equation 24.	Value corrected.
4. The value for PDS 2P in Table 2 does not match the value in the spreadsheet.	Value in the spreadsheet is incorrect and has been corrected.
5. Editorial comments.	Incorporated

Editorial or illustrative comments are attached to this review sheet to complete the review package.

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1.0 PURPOSE

The purpose of this calculation is to evaluate the risk of extending the Type A Integrated Leak Rate Test (ILRT) interval beyond the current 10 years required by 10 CFR 50, Appendix J at Robinson Nuclear Plant to 15 years.

1.1 SUMMARY OF THE ANALYSIS

10 CFR 50, Appendix J allows individual plants to extend Type A surveillance testing requirements and to provide for performance-based leak rate testing. This report documents a risk-based evaluation of the proposed change of the integrated leak rate test (ILRT) test interval for the Robinson Nuclear Plant (RNP). The proposed change would impact testing associated with the current surveillance test for Type A leakage (procedure EST-085)[1]. No change to Type B or Type C testing is proposed at this time.

The evaluation for RNP is performed consistent with interim guidance, Revision 4, provided by the Nuclear Energy Institute (NEI)[2]. The guidance refines the information provided in EPRI TR-104285[3] and NUREG-1493[4]. The NEI guidance also considers the submittals generated by other utilities. The assessment contained in this document utilizes the method set forth and utilizes metrics presented in Reference 2 supported by the metrics identified in Reference 5.

This calculation evaluates the risk associated with various ILRT intervals as follows:

- 3 years - Interval based on the original requirements of 3 tests per 10 years.
- 10 years – This is the current test interval required for RNP.
- 15 years – Proposed extended test interval.

The analysis utilizes the recent RNP probabilistic safety assessment (PSA) results. The PSA was initially developed for the RNP individual plant examination (IPE)[6] to estimate the baseline core damage and plant damage classes. Several updates to the RNP level 1 analysis have been incorporated since the IPE, including an update to the level 2 information.

The release category and person-rem information are based on design basis leakage evaluations and extrapolation of the release category information using a modeling framework that develops the person-rem estimates based on the relative release fractions of radionuclides. This approach has been utilized for other evaluations of risk significance[7]. The approach utilized to generate the population dose is consistent with and similar to the method used in the CR3 submittal[8]. The framework is described in detail in Appendix A.

1.2 SUMMARY OF RESULTS/CONCLUSIONS

The specific results are summarized in Table 1 below. The Type A contribution to LERF is defined as the contribution from Class 3b.

Table 1
Summary of Risk Impact on Extending Type A ILRT Test Frequency

	Risk Impact for 3-years (baseline)	Risk Impact for 10-years (current requirement)	Risk Impact for 15-years
Total Integrated Risk (Person-Rem/yr)	81.63	81.68	81.72
% Total Risk (Type A / Total)	0.030%	0.101%	0.152%
Type A LERF (Class 3b) (per year)	1.17E-7	3.91E-7	5.86E-7
Changes due to extension from 10 years (current)			
Δ Risk from current (Person-rem/yr)			4.13E-2
% Increase from current (Δ Risk / Total Risk)			0.0464%
Δ LERF from current (per year)			1.95E-7
Δ CCFP from current			0.455%
Changes due to extension from 3 years (baseline)			
Δ Risk from baseline (Person-rem/yr)			9.91E-2
% Increase from baseline (Δ Risk / Total Risk)			0.1115%
Δ LERF from baseline (per year)			4.69E-7
Δ CCFP from baseline			1.093%

The results are discussed below:

- The person-rem/year increase in risk contribution from extending the ILRT test frequency from the current once-per-ten-year interval to once-per-fifteen years is 4.13E-2 person-rem/yr. The change in Type A test frequency from once-per-ten-years to once-per-fifteen-years increases the risk impact on the total integrated plant risk by only 0.0464%. Also, the change in Type A test frequency from the original three-per-ten-years to once-per-fifteen-years increases the risk only 0.1115%. Therefore, the risk impact when compared to other severe accident risks is negligible.

- The risk increase in LERF from extending the ILRT test frequency from the current once-per-10-year interval to once-per-15 years is 1.95E-7/yr. Reg. Guide 1.174 (Reference 5) provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval from 10 to 15 years does not meet this criterion using the approach outlined in Reference 2.

- Reg. Guide 1.174 also encourages the use of risk analysis techniques to help ensure and to show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with defense-in-depth philosophy is maintained by demonstrating that the balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in conditional containment failure probability was estimated to be 0.455% for the proposed change. This change is small and demonstrates that the defense-in-depth philosophy is maintained.

In reviewing these results the RNP analysis demonstrates that the change in plant risk is small as a result of this proposed extension of ILRT testing. The change in LERF defined in the analysis is only slightly above the acceptance criterion and is a conservative estimate based on the NEI simplified analysis approach.

More detailed evaluations of the impact identify that the LERF increase from the current interval (10 years) to the proposed interval (15 years) does not exceed 1.0E-7/yr when the frequency for existing LERF contribution and the frequency for sequences that could not support a LERF release are removed from the calculation of LERF due to the increased probability of Type A leakage classified as Class 3B.

Additional analysis also indicates that the increase is not considered sufficient to increase the RNP total LERF above 1.0E-5/yr. Reference 5 identifies that a change may be considered if it can be reasonably shown that the increase from the change does not increase the total LERF above 1.0E-5/yr. The analysis indicates that the proposed change in the ILRT testing interval meets this criterion.

2.0 DESIGN INPUTS

The RNP PSA is a non-safety related tool and is intended to provide "best estimate" results that can be used as input when making risk informed decisions. The current RNP PSA[9] is an update to the IPE, which was a NRC submittal of the PRA provided in response to requests from Generic Letter 88-20. The PSA is not considered as design basis information.

The inputs for this calculation come from the information documented in the RNP PSA and the more recent update (Reference 9). The RNP plant damage classes are summarized in Table 2.

Table 2
RNP Plant Damage Classes

Damage Class	Representative Sequence	Frequency (/yr)
5D	Loss of component cooling water occurs, losing all cooling to the reactor coolant pumps and resulting in an RCP seal LOCA. Injection is successful, but recirculation fails due to lack of CCW cooling. The failure of recirculation fails containment sprays.	1.31E-5
10P	Loss of offsite power with failure of all ac power resulting in a RCP seal LOCA. Limited operation of the AFW steam-driven pump delays core damage.	5.88E-6
5J	Same as 5D with failure of fans.	4.92E-6
3P	A transient event without heat removal. Containment cooling is also failed. The containment is isolated.	2.90E-6
19A	Transient with failure of the reactor to scram resulting in an RCS overpressure condition which fails the RCS and safety injection.	2.83E-6
2A	Loss of offsite power, failure of AFW early, and failure of the operators to establish feed and bleed cooling.	2.43E-6
B5E	Steam generator tube rupture event with a failure of makeup during recirculation and a loss of containment sprays upon recirculation.	2.02E-6
12C (12O)[1]	Interfacing systems LOCA through the RHR hot leg suction line.	1.26E-6
6J	A large service water flood occurs which results in a reactor cooling pump seal LOCA and a failure of all injection. In addition, all containment safeguards are failed.	1.12E-6
4D	Small LOCA that is too small to remove decay heat occurs, failure of AFW early, successful feed and bleed cooling during injection, and failure of the operators to accomplish high-head recirculation results in containment spray failure.	6.52E-7
17D	Medium LOCA occurs with successful injection and failure of the operators to establish high-head recirculation. The failure to establish recirculation also fails containment sprays.	6.22E-7

Table 2
RNP Plant Damage Classes

Damage Class	Representative Sequence	Frequency (/yr)
20D	Large LOCA occurs with failure of the operators to establish recirculation. The failure of recirculation results in containment spray failure.	6.00E-7
10A	Although different break location and somewhat earlier timing, this sequence similar to RCP seal LOCA cases (10J).	4.97E-7
B9B	Similar to PDS B9E below except that containment sprays function.	4.88E-7
B9E	A steam generator tube rupture occurs with a failure of AFW when the CST empties (~4 hours). The operators fail to accomplish shutdown cooling or high-head recirculation. The failure of recirculation results in containment spray failure and the faulted steam generator results in a small bypass through a cycling steam generator SRV.	4.88E-7
20A	Similar to PDS 20D with the difference being the successful operation of containment sprays.	4.75E-7
2D	Loss of offsite power, failure of AFW early, and failure of the operators to establish feed and bleed cooling. The containment sprays initiate but fail during recirculation.	4.45E-7
B10B	Steam generator tube rupture with a failure of makeup. All containment safeguards available.	3.69E-7
5E	Loss of component cooling water occurs, losing all cooling to the reactor coolant pumps and resulting in an RCP seal LOCA. Injection is successful, but recirculation fails due to a lack of CCW cooling. The failure of recirculation fails containment sprays. A small isolation failure is also present.	3.66E-7
10J	A loss of service water occurs which fails all RCP seal cooling and safety injection. In addition, all containment safeguards are lost.	2.80E-7
4J	Small LOCA sequence with the loss of all safeguards after the failure of containment sprays in recirculation.	2.67E-7
5K	Similar to 5D with the failure of all safeguards.	1.69E-7
19B	Transient with failure of the reactor to scram resulting in an RCS overpressure condition which fails the RCS and safety injection. In addition, a small containment isolation failure exists.	8.90E-8
16A	Similar to 16D found in PSA submittal. Difference is that for this PDS the containment sprays are not failed by the recirculation failure. In both cases, fan coolers are available to provide containment heat removal and existing MAAP runs adequate.	6.65E-8

Table 2
RNP Plant Damage Classes

Damage Class	Representative Sequence	Frequency (/yr)
3J	A total loss of service water, occurs with a failure of AFW early, The loss of service water fails RHR, recirculation and containment sprays. The fans are failed due to the loss of service water.	5.66E-8
B5B	Steam generator tube rupture with failure of makeup to the reactor vessel. The steam generator tube rupture indicates a small leak with secondary-side heat removal available.	4.84E-8
B10N	Steam generator tube rupture with failure of makeup and containment sprays.	4.62E-8
10R	A loss of service water occurs which fails all RCP seal cooling and safety injection. In addition, all containment safeguards are lost early in the event with the failure of containment sprays prior to recirculation.	3.43E-8
2J	Similar to total loss of service water (3K).	3.25E-8
11J	Small LOCA or seal LOCA sequence with a failure of core heat removal and containment heat remo val.	3.04E-8
10Q	Similar to 10D except that all containment cooling is lost.	2.90E-8
10D	Small LOCA or seal LOCA with failure of high-pressure injection and low-pressure injection. Containment spray fails in recirculation.	2.63E-8
11P	A large service water flood occurs resulting in a seal LOCA and failure of all containment safeguards. Containment isolation, however, is successful.	2.14E-8
5F	Small LOCA or seal LOCA with a failure of low-pressure injection late in the sequence. A large isolation failure exists and sprays fail in recirculation.	1.91E-8
20B	Large LOCA sequence with a small isolation failure.	1.87E-8
3Q	Loss of heat removal sequence with the failure of all containment safeguards and a small isolation failure.	1.33E-8
3R	Similar to 3Q except a large isolation failure exists.	1.33E-8
20E	Large LOCA occurs with failure of the operators to establish recirculation. The failure of recirculation results in containment spray failure.	1.19E-8
5L	Small LOCA or seal LOCA with a failure of cooling late, no containment cooling and a large isolation failure.	1.05E-8
2B	Loss of offsite power, failure of AFW early, and failure of the operators to establish feed and bleed cooling. In addition, a small containment	9.98E-9

Table 2
RNP Plant Damage Classes

Damage Class	Representative Sequence	Frequency (/yr)
	isolation failure exists.	
5A	Similar to 5F except that all safeguards are functioning and the containment is intact.	9.63E-9
3G	Transient occurs with a loss of heat removal and failure of feed-and-bleed. The containment fan coolers are also failed.	8.47E-9
17A	A medium LOCA sequence with a failure of recirculation and all safeguards available.	7.86E-9
2P	Similar to 2A with a failure of containment sprays early in the sequence.	7.34E-9
17E	Medium LOCA occurs with successful injection and failure of the operators to establish high-head recirculation. The failure to establish recirculation also fails containment spray.	6.48E-9
10B	Similar to PDS 15F.	5.96E-9
4E	Similar to PDS 4J except that fans function and a small isolation failure exists.	5.88E-9
19D	Similar to PDS 19A except that containment sprays fail in recirculation.	4.92E-9
12F	Interfacing system LOCA with failure of containment sprays in recirculation. Similar to PDS 12C.	4.80E-9
6D	Similar to a small LOCA case (4D) with the failure of AFW and recirculation. RHR failure fails containment sprays.	4.71E-9
4K	Similar to PDS 4J except that a small isolation failure is present.	4.67E-9
2C	Similar to PDS 2A with the exception that a small isolation failure is present.	4.58E-9
2E	Similar to PDS 2C with the exception that the containment sprays are unavailable in recirculation.	4.58E-9
2F	Similar to 2E except that the isolation failure is large.	4.58E-9
B10E	A steam generator tube rupture that is similar to B10N except containment sprays function until recirculation.	4.17E-9
Total		4.29E-5

1. Revised plant damage state to reflect appropriate end state.

In order to develop the person-rem dose associated with each plant damage state it is necessary to associate each plant damage state with an associated release of radionuclides and from this information to calculate the associated dose.

The IP3 submittal[10] utilizes a multiplication factor to adjust the design basis leakage value (L_a) that is based on generic information that relates dose to leak size. The CR3 submittal (Reference 8) utilized plant-specific dose estimates based on the predicted level 2 analysis results.

The RNP PSA (Reference 9) contains the necessary information to convert the plant damage classes to release categories. Using this information the plant damage classes are mapped to the ten release categories.

Since the RNP PSA contains the necessary release fraction information, an approach similar to the CR3 submittal is utilized that better reflects the specific release conditions for RNP. The RNP PSA release categories are defined by the release fraction of major radionuclides. These are extrapolated to dose using the approach presented in Appendix A with the exception of the intact containment dose. The intact containment dose is based on the licensing design basis leakage rate and is developed in Reference 11. The release category dose information is presented in Table 3.

Table 3
Release Category Radionuclide Fraction and Total

Release Category	Frequency	Noble Gas	Iodine	Cesium	Tellurium	Strontium	Person-Rem
IC-1[1]	2.18E-5	NA[2]	NA	NA	NA	NA	1.56E+3
RC-1	1.01E-5	1.00E+0	1.80E-3	2.30E-3	5.40E-5	7.40E-6	1.71E+6
RC-1A	2.08E-7	1.00E+0	1.22E-1	7.21E-2	2.10E-5	0.00E+0	1.12E+7
RC-1B	4.83E-6	1.00E+0	7.50E-3	6.90E-3	4.89E-2	5.00E-4	2.47E+6
RC-1BA	2.46E-7	1.00E+0	4.71E-2	5.82E-2	7.69E-2	8.00E-4	7.15E+6
RC-2	3.51E-8	1.00E+0	2.63E-2	2.83E-2	4.30E-6	1.00E-4	4.23E+6
RC-2B	1.70E-7	1.00E+0	1.65E-1	1.90E-1	9.21E-2	3.90E-3	1.97E+7
RC-3	7.29E-7	2.00E-1	8.00E-4	5.00E-4	1.07E-1	8.60E-6	8.98E+5
RC-3B	2.97E-9	2.00E-1	8.00E-4	5.00E-4	1.07E-1	1.10E-3	9.04E+5
RC-4	0.00E+0	4.00E-1	1.70E-2	1.53E-2	0.00E+0	1.70E-5	2.22E+6
RC-4C	3.47E-6	4.00E-1	1.70E-2	1.53E-2	0.00E+0	1.70E-5	2.22E+6
RC-5	1.20E-6	1.00E+0	2.86E-1	2.61E-1	0.00E+0	8.00E-3	2.89E+7
RC-5C	6.31E-8	1.00E+0	2.86E-1	2.61E-1	0.00E+0	8.00E-3	2.89E+7
Total	4.29E-5						

1. Contributing fission product groups are discussed in Appendix A.
2. Release fractions not necessary for this calculation.
3. Intact containment representing design basis leakage (developed in Reference 11).

Other inputs to this calculation include the NEI guidance (Reference 2), ILRT test data from NUREG-1493 (Reference 4) and the EPRI report (Reference 3) and are referenced in the body of the calculation.

3.0 ASSUMPTIONS

1. The maximum containment leakage for EPRI Class 1 (Reference 3) sequences is 1 L_a (Type A acceptable leakage) because a new Class 3 has been added to account for increased leakage due to Type A inspections.

2. The maximum containment leakage for Class 3a (References 2, 8 and 10) sequences is 10 L_a based on the NEI guidance and previously approved methodology (References 2, 8 and 10).

3. The maximum containment leakage for Class 3b sequences is 35 L_a based on the NEI guidance and previously approved methodology (References 2, 8 and 10).

4. Class 3b is conservatively categorized LERF based on the NEI guidance and previously approved methodology (References 2, 8 and 10).

5. Containment leakage due to EPRI Classes 4 and 5 are considered negligible based on the NEI guidance and the previously approved methodology (References 2, 8 and 10).

6. The containment releases are not impacted with time.

7. The containment releases for EPRI 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.

8. Because EPRI Class 8 sequences are containment bypass sequences, potential releases are directly to the environment. Therefore, the containment structure will not impact the release magnitude.

4.0 CALCULATIONS

This calculation applies the RNP PSA release category information in terms of frequency and person-rem estimates to estimate the changes in risk due to increasing the ILRT test interval. The changes in risk are assessed consistent with the guidance provided in the NEI interim guidance document (Reference 2). This approach considers other similar analyses presented in EPRI TR-104285 (Reference 3) and NUREG-1493 (Reference 4).

4.1 CALCULATIONAL STEPS

The analysis is based on guidance provided in Reference 2 and uses risk metrics presented in Reference 5 to evaluate the impact of a proposed change on plant risk. References 2, 8 and 10 utilize several measures in their assessments. These measures are change in release frequency, change in risk as defined by the change in person-rem, the change in LERF and the change in the conditional containment failure probability.

Reference 5 also lists the change in core damage frequency as a measure to be considered. Since the testing addresses the ability of the containment to maintain its function, the proposed change has no measurable impact on core damage frequency. Therefore, this attribute remains constant and has no risk significance.

The overall process is outlined below:

- Define and quantify the baseline plant damage classes and person-rem estimates
- Calculate baseline leakage rates and estimate probability to define the analysis baseline
- Develop baseline population dose (person-rem) and population dose rate (person-rem/yr)

- Modify Type A leakage estimate to address extension of the Type A test frequency and calculate new population dose rates, LERF and conditional containment failure probability

- Compare analysis metrics to estimate the impact and significance of the increase related to those metrics

The first step in the analysis is to define the baseline plant damage classes and person-rem dose measures. Plant damage state information is developed using the RNP PSA (References 6 and 9) results. The plant damage state information and the results of the containment analysis are used to define the sequences. The population person-rem dose estimates for each key plant damage classes are based on the application of the method described in Appendix A and the calculation presented in Reference 11.

The product of the person-rem for the plant damage classes and the frequency of the plant damage state is used to estimate the annual person-rem for the plant damage state. Summing these estimates produces the annual person-rem dose based on the sequences defined in the PSA.

The PSA plant damage state definitions considered isolation failures due to Type B and Type C faults and examine containment challenges occurring after core damage and/or reactor vessel failure. These sequences are grouped into key plant damage classes. Using the plant damage state information, bypass, isolation failures and phenomena-related containment failures are identified. Once identified, the sequence was then classified by release category definitions specified in Reference 3. With this information developed, the PSA baseline model is completed.

The second step expands the baseline model to address Type A leakage. The PSA did not directly address Type A (liner-related) faults and this contribution must be added to provide a complete baseline. In order to define leakage that can be linked directly to the Type A testing, it is important that only failures that would be identified by Type A testing exclusively be included.

Reference 2 provides the estimate for the probability of a leakage contribution that could only be identified by Type A testing based on industry experience. This probability is then used to adjust the intact containment category of the RNP PSA to develop a baseline model including Type A faults.

The release, in terms of person-rem, is developed based on information contained in Reference 2 and is estimated as a leakage increase relative to allowable release L_a defined as part of the ILRT.

The predicted probability of Type A leakage is then modified to address the expanded time between testing. This is accomplished by a ratio of the existing testing interval and the proposed test interval. This assumes a constant failure rate and that the failures are randomly dispersed during the interval between the test.

The change due to the expanded interval is calculated and reported in terms of the change in release due to the expanded testing interval, the change in the population person-rem and the change in large early release frequency. The change in the conditional containment failure probability is also developed. From these comparisons, a conclusion is drawn as to the risk significance of the proposed change.

Using this process, the following performed:

1. Map the release categories into the 8 release classes defined by the EPRI Report (Reference 3).
2. Calculate the Type A leakage estimate to define the analysis baseline.
3. Calculate the Type A leakage estimate to address the current inspection frequency.
4. Modify the Type A leakage estimates to address extension of the Type A test interval.
5. Calculate increase in risk due to extending Type A inspection intervals.
6. Estimate the change in LERF due to the Type A testing.
7. Estimate the change in conditional containment failure probability due to the Type A testing.

4.2 SUPPORTING CALCULATIONS

Step 1: Map the Level 3 release categories into the 8 release classes defined by the EPRI Report

EPRI Report TR-104285 (Reference 3) defines eight (8) release classes as presented in Table 4.

Table 4
Containment Failure Classifications (from Reference 3)

Failure Classification	Description	Interpretation for Assigning RNP Release Category
1	Containment remains intact with containment initially isolated	Intact containment bins
2	Dependent failure modes or common cause failures	Isolation faults that are related to a loss of power or other isolation failure mode that is not a direct failure of an isolation component
3	Independent containment isolation failures due to Type A related failures	Isolation failures identified by Type A testing
4	Independent containment isolation failures due to Type B related failures	Isolation failures identified by Type B testing
5	Independent containment isolation failures due to Type C related failures	Isolation failures identified by Type C testing
6	Other penetration failures	Other faults not previously identified
7	Induced by severe accident phenomena	Early containment failure sequences as a result of hydrogen burn or other early phenomena
8	Bypass	Bypass sequence or SGTR

Table 5 presents the RNP release category mapping for these eight accident classes. Person-rem per year is the product of the frequency and the person-rem.

The NEI guidance document specifies that Class 6 is not a significant contributor and can be excluded from consideration and Class 2 is considered as a measurable contributor. However, for the RNP PSA, the potential for isolation failure is dominated by human errors failing to isolate containment lines or the personnel hatch and not by the failure of valves to close (Class 2 failures). In reviewing the EPRI guidance and the IP3 submittal, these failures were grouped into Class 6 since they are not failure of the isolation component. RNP has adopted this approach such that Class 6 is retained in the evaluation.

Table 5
RNP PSA Release Category Grouping to EPRI Classes (Reference 3)

Class	Description	Release Category	Frequency	Person-Rem	Person-Rem/yr
1	No Containment Failure	IC-1	2.18E-5	1.56E+3	3.401E-2
2	Large Containment Isolation Failures	None	ε[1]	0	0
3a	Small Isolation Failures (Liner breach)	None	0	0	0
3b	Large Isolation Failures (Liner breach)	None	0	0	0
4	Small isolation failures - failure to seal (type B)	None	0	0	0
5	Small isolation failures - failure to seal (type C)	None	0	0	0
6	Containment Isolation Failures (dependent failure, personnel errors)	RC-2, RC-2B, RC-3	9.34E-7	4.45E+6[2]	4.154E+0
7	Severe Accident Phenomena Induce Failure (Early and Late)	RC-1, RC-1A, RC-1B, RC-1BA, RC-3B	1.54E-5	2.16E+6[2]	3.328E+1
8	Containment Bypass	RC-4C, RC-5, RC-5C	4.73E-6	9.33E+6[2]	4.413E+1
		Total	4.29E-5		81.60

1. No results. Contributions are below quantification truncation value.
2. Based on weighted doses of individual contributors.

Step 2: Calculate the Type A leakage estimate to define the analysis baseline (3 year test interval)

As displayed in Table 5 the RNP PSA did not identify any release categories specifically associated with EPRI Classes 3, 4, or 5. Therefore each of these classes must be evaluated for applicability to this study.

Class 3:

Containment failures in this class are due to leaks such as liner breaches that could only be detected by performing a Type A ILRT. In order to determine the impact of the extended testing interval, the probability of Type A leakage must be calculated.

In order to better assess the range of possible leakage rates, the Class 3 calculation is divided into two classes. Class 3a is defined as a small liner breach and Class 3b is defined as a large liner breach. This division is consistent with the NEI guidance (Reference 2) and the previously approved methodology (References 8 and 10). The calculation of Class 3a and Class 3b probabilities is presented below.

Calculation of Class 3a Probability

The data presented in NUREG-1493 (Reference 4) is also used to calculate the probability that a liner leak will be small (Class 3a). The data found in NUREG-1493 states that 144 ILRTs were conducted. The data reported that 23 of 144 tests had allowable leak rates in excess of $1.0L_a$. However, of the 23 events that exceeded the test requirements, only 4 were found by an ILRT, the others were found by Type B and C testing or errors in test alignments.

Data presented in Reference 2, taken since 1/1/1995, increases this database to a total of 5 Type A leakage events in total of 182 events. Using the data a mean estimate for the probability of leakage is determined for Class 3a as shown in Equation 1.

$$p_{Class3a} = \frac{5}{182} = 0.027 \qquad \text{(eq. 1)}$$

This probability, however, is based on three tests over a 10-year period and not the one per ten-year frequency currently employed at RNP (Reference 1). The probability (0.027) must be adjusted to reflect this difference and is adjusted in step 3 of this calculation.

Multiplying the core damage frequency (CDF) times the probability of a Class 3a leak develops the Class 3a frequency contribution in accordance with guidance provided in Reference 2. This is conservative since part of the CDF already includes LERF sequences. The CDF for RNP is 4.29E-5/yr as presented in Table 5.

Therefore the frequency of a Class 3a failure is calculated as:

$$FREQ_{class3a} = PROB_{class3a} \times CDF = 0.027 \times 4.29E\text{-}5/yr = 1.18E\text{-}6/yr \qquad \text{(eq. 2)}$$

Calculation of Class 3b Probability

To calculate the probability that a liner leak will be large (Class 3b) use was made of the data presented in the calculation of Class 3a. Of the events identified in NUREG-1493 (Reference 4), the largest reported leak rate from those 144 tests was 21 times the allowable leakage rate (L_a). Since 21 L_a does not constitute a large release, no large releases have occurred based on the 144 ILRTs reported in NUREG-1493 (Reference 4). The additional data point was also not considered to constitute a large release.

To estimate the failure probability given that no failures have occurred, the guidance provided in Reference 2 suggests the use of a non-informative prior. This approach essentially updates a uniform distribution (no bias) with the available evidence (data) to provide a better estimation of an event.

A beta distribution is typically used for the uniform prior with the parameters α=0.5 and β=1. This is then combined with the existing data (no Class 3b events, 182 tests) using Equation 3.

$$p_{Class3b} = \frac{n+\alpha}{N+\beta} = \frac{0+0.5}{182+1} = \frac{0.5}{183} = 0.0027 \qquad \text{(eq. 3)}$$

where: N is the number of tests, n is the number of events (faults) of interest, α, β are the prior uninformed distribution. From this solution, the frequency for Class 3b is generated using Equation 4 and is adjusted appropriately in step 3.

$FREQ_{class3b} = PROB_{class3b}$ x CDF = 0.0027 x 4.29E-5/yr = 1.17E-7/yr (eq. 4)

Class 1:

Although Type A testing does not directly impact the frequency of this class, the PSA did not model Class 3 failures, and the frequency for Class 1 should be reduced by the estimated frequencies in the new Class 3a and Class 3b in order to preserve the total CDF. The revised Class 1 frequency is therefore:

$FREQ_{class1} = FREQ_{class1} - (FREQ_{class3a} + FREQ_{class3b})$ (eq. 5)

$FREQ_{class1}$ = 2.18E-5/yr – (1.18E-6/yr + 1.17E-7/yr) = 2.05E-5/yr (eq. 6)

The other classes are not impacted by the ILRT contribution and are based on the existing RNP model solution presented in Table 5.

Class 2:

The Table 5 does not identify any contribution to Class 2.

Class 4:

This group consists of all core damage accident accidents for which a failure-to-seal containment isolation failure of Type B test components occurs. By definition, these failures are dependent on Type B testing, and Type A testing will not impact the probability. Therefore this group is not evaluated any further, consistent with the approved methodology.

Class 5:

This group consists of all core damage accident accidents for which a failure-to-seal containment isolation failure of Type C test components occurs. By definition, these failures are dependent on Type C testing, and Type A testing will not impact the probability. Therefore this group is not evaluated any further, consistent with the approved methodology.

Class 6:

The Class 6 group is comprised of isolation faults that occur as a result of the accident sequence progression. The leakage rate is not considered large by the PSA definition and therefore it is placed into Class 6 to represent a small isolation failure and identified in Table 5 as Class 6.

$FREQ_{class6}$ =9.34E-7/yr

Class 7:

The frequency of Class 7 is the sum of those release categories identified in Table 5 as Class 7.

$FREQ_{class7}$ = 1.54E-5/yr

Class 8:

The frequency of Class 8 is the sum of those release categories identified in Table 5 as Class 8.

$FREQ_{class8}$ = 4.73E-6/yr

Table 6 summarizes the above information by the EPRI defined classes. This table also presents dose exposures calculated using the methodology described in Appendix A. For Class 1, 3a and 3b, the person-rem is developed based on the design basis assessment of the intact containment (Reference 11). The Class 3a and 3b doses are represented as $10L_a$ and 35 L_a respectively. Table 6 also presents the person-rem frequency data determined by multiplying the failure class frequency by the corresponding exposure.

Table 6
Baseline Risk Profile

Class	Description	Frequency (/yr)	Person-rem (from calculation)[1]	Person-rem (from L_a factors)	Person-rem (/yr)
1	No Containment Failure	2.05E-5		1.56E+3[2]	3.199E-2
2	Large Containment Isolation Failures	0	NA	NA	0
3a	Small Isolation Failures (Liner breach)	1.18E-6		1.56E+4[3]	1.838E-2
3b	Large Isolation Failures (Liner breach)	1.17E-7		5.46E+4[4]	6.397E-3
4	Small isolation failures - failure to seal (type B)	0	NA	NA	0
5	Small isolation failures - failure to seal (type C)	0	NA	NA	0
6	Containment Isolation Failures (dependent failure, personnel errors)	9.34E-7	4.45E+6		4.154E+0
7	Severe Accident Phenomena Induce Failure (Early and Late)	1.54E-5	2.16E+6		3.328E+1
8	Containment Bypass	4.73E-6	9.33E+6		4.413E+1
	Total	4.29E-5			81.6275

1. From Table 3 using the method presented in Appendix A.
2. 1 L_a dose value calculated in Reference 12.
3. 10 times L_a
4. 35 times L_a
5. Frequency weighted dose from contributing release categories.

The percent risk contribution due to Type A testing is as follows:

$\%Risk_{BASE} = [(Class3a_{BASE} + Class3b_{BASE}) / Total_{BASE}] \times 100$

Where:

$Class3a_{BASE}$ = Class 3a person-rem/year =1.838E-2 person-rem/year

$Class3b_{BASE}$ = Class 3b person-rem/year = 6.397E-3 person-rem/year

$Total_{BASE}$ = total person-rem year for baseline interval = 81.6275 person-rem/year (Table 6)

$\%Risk_{BASE} = [(1.838E\text{-}2 + 6.397E\text{-}3) / 81.6275] \times 100 = \mathbf{0.030\%}$ (eq. 7)

Step 3: Calculate the Type A leakage estimate to address the current inspection interval

The current surveillance testing requirements as proposed in NEI 94-01[12] for Type A testing and allowed by 10 CFR 50, Appendix J 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 $1.0L_a$).

According to References 2 and 13, 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. Multiplying the testing interval by 0.5 and multiplying by 12 to convert from "years" to "months" calculates the average time for an undetected condition to exist.

The increase for a 10-yr ILRT interval is the ratio of the average time for a failure to detect for the increased ILRT test interval (from 18 months to 60 months) multiplied by the existing Class 3a probability as shown in Equation 8.

$$p_{Class3a}(10y) = 0.027 \times \left(\frac{60}{18}\right) = 0.09 \qquad \text{(eq. 8)}$$

A similar calculation is performed for the Class 3b probability as presented in Equation 9.

$$p_{Class3b}(10y) = 0.0027 \times \left(\frac{60}{18}\right) = 0.009 \qquad \text{(eq. 9)}$$

Risk Impact due to 10-year Test Interval

Based on the previously approved methodology (References 8 and 10) and the NEI guidance (Reference 2), the increased probability of not detecting excessive leakage due to Type A tests directly impacts the frequency of the Class 3 sequences.

Consistent with Reference 2 the risk contribution is determined by multiplying the Class 3 accident frequency by the increase in the probability of leakage. Additionally the Class 1 frequency is adjusted to maintain the overall core damage frequency constant. The results of this calculation are presented in Table 7 below.

Table 7
Risk Profile for Once in Ten Year Testing

Class	Description	Frequency (/yr)	Person-rem [2]	Person-rem (/yr)
1	No Containment Failure [1]	1.75E-5	1.56E+3	2.728E-2
2	Large Containment Isolation Failures	0	NA	0
3a	Small Isolation Failures (Liner breach)	3.93E-6	1.56E+4	6.126E-2
3b	Large Isolation Failures (Liner breach)	3.91E-7	5.46E+4	2.132E-2
4	Small isolation failures - failure to seal (type B)	0	NA	0
5	Small isolation failures - failure to seal (type C)	0	NA	0
6	Containment Isolation Failures (dependent failure, personnel errors)	9.34E-7	4.45E+6	4.154E+0
7	Severe Accident Phenomena Induce Failure (Early and Late)	1.54E-5	2.16E+6	3.328E+1
8	Containment Bypass	4.73E-6	9.33E+6	4.413E+1
	Total	4.29E-5		81.6806

1. The PSA frequency of Class 1 has been reduced by the frequency of Class 3a and Class 3b in order to preserve total CDF.
2. From Table 6.

Using the same methods as for the baseline, and the data in Table 7 the percent risk contribution due to Type A testing is as follows:

$\%Risk_{10} = [(Class3a_{10} + Class3b_{10}) / Total_{10}] \times 100$

Where:

$Class3a_{10}$ = Class 3a person-rem/year = 6.126E-2 person-rem/year

$Class3b_{10}$ = Class 3b person-rem/year = 2.132E-2 person-rem/year

$Total_{10}$ = total person-rem year for current 10-year interval = 81.6806 person-rem/year (Table 7)

$\%Risk_{10}$ = [(6.126E-2 + 2.132E-2) / 81.6806] x 100 = **0.101%** (eq. 10)

The percent risk increase ($\Delta\%Risk_{10}$) due to a ten-year ILRT over the baseline case is as follows:

$\Delta\%Risk_{10} = [(Total_{10} - Total_{BASE}) / Total_{BASE}] \times 100.0$

Where:

$Total_{BASE}$ = total person-rem/year for baseline interval = 81.6275 person-rem/year (Table 6)

$Total_{10}$ = total person-rem/year for 10-year interval = 81.6806 person-rem/year (Table 7)

$\Delta\%Risk_{10} = [(81.6806 - 81.6275) / 81.6275] \times 100.0 = \mathbf{0.0650\%}$ (eq. 11)

Step 4: Calculate the Type A leakage estimate to address extended inspection intervals

If the test interval is extended to 1 in 15 years, the average time that a leak detectable only by an ILRT test goes undetected increases to 90 months (0.5 x 15 x 12). For a 15-yr-test interval, the result is the ratio (90/18) of the exposure times as was the case for the 10 year case. Thus, increasing the ILRT test interval from 10 years to 15 years results in a proportional increase in the overall probability of leakage.

The approach for developing the risk contribution for a 15-year interval is the same as that for the 10-year interval. The increase for a 15-yr ILRT interval is the ratio of the average time for a failure to detect for the increased ILRT test interval (from 18 months to 90 months) multiplied by the existing Class 3a probability as shown in Equation 12.

$$p_{Class3a}(10y) = 0.027 \times \left(\frac{90}{18}\right) = 0.135 \qquad \text{(eq. 12)}$$

A similar calculation is performed for the Class 3b probability as presented in Equation 13.

$$p_{Class3b}(10y) = 0.0027 \times \left(\frac{90}{18}\right) = 0.0135 \qquad \text{(eq. 13)}$$

As stated for the 10-year case, the increased probability of not detecting excessive leakage due to Type A tests directly impacts the frequency of the Class 3 sequences.

The increased risk contribution is determined by multiplying the Class 3 accident frequency by the increase in the probability of leakage. Additionally the Class 1 frequency is adjusted to maintain the overall core damage frequency constant. The results of this calculation are presented in Table 8 below.

Table 8
Risk Profile for Once in Fifteen Year Testing

Class	Description	Frequency (/yr)	Person-rem [2]	Person-rem (/yr)
1	No Containment Failure [1]	1.53E-5	1.56E+3	2.391E-2
2	Large Containment Isolation Failures	0	NA	0
3a	Small Isolation Failures (Liner breach)	5.89E-6	1.56E+4	9.189E-2
3b	Large Isolation Failures (Liner breach)	5.86E-7	5.46E+4	3.199E-2
4	Small isolation failures - failure to seal (type B)	0	NA	0
5	Small isolation failures - failure to seal (type C)	0	NA	0
6	Containment Isolation Failures (dependent failure, personnel errors)	9.34E-7	4.45E+6	4.154E+0
7	Severe Accident Phenomena Induce Failure (Early and Late)	1.54E-5	2.16E+6	3.328E+1
8	Containment Bypass	4.73E-6	9.33E+6	4.413E+1
	Total	4.29E-5		81.7185

1. The PSA frequency of Class 1 has been reduced by the frequency of Class 3a and Class 3b in order to preserve total CDF.
2. From Table 6.

Using the same methods as for the baseline, and the data in Table 10 the percent risk contribution due to Type A testing is as follows:

$\%Risk_{15} = [(Class3a_{15} + Class3b_{15}) / Total_{15}] \times 100$

Where:

$Class3a_{15}$ = Class 3a person-rem/year = 9.189E-2 person-rem/year

$Class3b_{15}$ = Class 3b person-rem/year = 3.199E-2 person-rem/year

$Total_{15}$ = total person-rem year for 15-year interval = 81.7185 person-rem/year (Table 8)

$\%Risk_{15} = [(9.189E-2 + 3.199E-2) / 81.7185] \times 100 =$ **0.152%** (eq. 14)

The percent risk increase ($\Delta\%Risk_{15}$) due to a fifteen-year ILRT over the baseline case is as follows:

$\Delta\%Risk_{15} = [(Total_{15} - Total_{BASE}) / Total_{BASE}] \times 100.0$

Where:

$Total_{BASE}$ = total person-rem/year for baseline interval = 81.6275 person-rem/year (Table 6)

$Total_{15}$ = total person-rem/year for 15-year interval = 81.7185 person-rem/year (Table 8)

$\Delta\%Risk_{15} = [(81.7185 - 81.6275) / 81.6275] \times 100.0 =$ **0.1115%** (eq. 15)

Step 5: Calculate increase in risk due to extending Type A inspection intervals

Based on the guidance in Reference 2, the percent increase on the total integrated plant risk for these accident sequences is computed as follows:

$\%Total_{10\text{-}15} = [(Total_{15} - Total_{10}) / Total_{10}] \times 100$

Where:

$Total_{10}$ = total person-rem/year for 10-year interval

= 81.6806 person-rem/year (Table 7)

$Total_{15}$ = total person-rem/year for 15-year interval

= 81.7185 person-rem/year (Table 8)

$\% Total_{10\text{-}15} = [(81.7185 - 81.6806) / 81.6806] \times 100 =$ **0.0464%** (eq. 16)

Step 6: Calculate the change in Risk in terms of Large Early Release Frequency (LERF)

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 a larger release due to failure to detect a pre-existing leak during the relaxation period.

From References 2, 8 and 10, the Class 3a dose is assumed to be 10 times the allowable intact containment leakage, L_a (or 15,600 person-rem) and the Class 3b dose is assumed to be 35 times L_a (or 54,600 person-rem). The dose equivalent for allowable leakage (L_a) is developed in Reference 11. This compares to a historical observed average of twice L_a. Therefore, the estimate is somewhat conservative.

Based on the NEI guidance (Reference 2) and the previously approved methodology (References 8 and 10), only Class 3 sequences have the potential to result in large releases if a pre-existing leak were present. Class 1 sequences are not considered as potential large release pathways because for these sequences the containment remains intact. Therefore, the containment leak rate is expected to be small (less than $2L_a$). A larger leak rate would imply an impaired containment, such as Classes 2, 3, 6 and 7.

Late releases are excluded regardless of the size of the leak because late releases are, by definition, not a LERF event. At the same time, sequences in the RNP PSA (Reference 9), that result in large releases, are not impacted because a LERF will occur regardless of the presence of a pre-existing leak. Therefore, the change in the frequency of Class 3b sequences is used as the increase in LERF for RNP, and the change in LERF can be determined by the differences. Reference 2 identifies that Class 3b is considered to be the contributor to LERF. Table 9 summarizes the results of the LERF evaluation that Class 3b is indicative of a LERF sequence.

Table 9
Impact on LERF due to Extended Type A Testing Intervals

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
Class 3b (Type A LERF)	1.17E-7/yr	3.91E-7/yr	5.86E-7/yr
ΔLERF (10 year baseline)			1.95E-7/yr
ΔLERF (3 year baseline)			4.69E-7/yr

Reg. Guide 1.174 (Reference 5) provides guidance for determining the risk impact of plant-specific changes to the licensing basis. The Reg. Guide 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 1E-6/yr and increases in LERF below 1E-7/yr. Since the ILRT does not impact CDF, the relevant metric is LERF. Calculating the increase in LERF requires determining the impact of the ILRT interval on the leakage probability.

Increasing the ILRT interval from the currently acceptable 10 years to 15 years results in an increase in LERF of **1.95E-7/yr**. This only slightly exceeds the guidance in Reg. Guide 1.174 defining very small changes in LERF. The LERF increase is measured from the original 3-in-10-year interval to the 15-year interval is **4.69E-7/yr**, which again exceeds the criterion presented in Regulatory Guide 1.174.

Step 7: Calculate the change in Conditional Containment Failure Probability (CCFP)

The conditional containment failure probability (CCFP) is defined as the probability of containment failure given the occurrence of an accident. This probability can be expressed using the following equation:

$$CCFP = 1 - \left[\frac{f(ncf)}{CDF} \right] \qquad \text{(eq. 17)}$$

Where *f(ncf)* is the frequency of those sequences which result in no containment failure. This frequency is determined by summing the Class 1 and Class 3a results, and CDF is the total frequency of all core damage sequences.

Therefore the change in CCFP for this analysis is the CCFP using the results for 15 years ($CCFP_{15}$) minus the CCFP using the results for 10 years ($CCFP_{10}$). This can be expressed by the following:

$$\Delta CCFP_{10-15} = CCFP_{15} - CCFP_{10} \qquad \text{(eq. 18)}$$

Using the data previously developed the change in CCFP from the current testing interval is calculated and presented in Table 10.

Table 10
Impact on Conditional Containment Failure Probability due to Extended Type A Testing Intervals

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
f(ncf) (/yr)	2.17E-5	2.14E-5	2.12E-5
f(ncf)/CDF	0.506	0.499	0.495
CCFP	0.494	0.501	0.505
ΔCCFP (3 year baseline)		0.638%	1.093%
ΔCCFP (10 year baseline)			0.455%

Alternative LERF Calculation Using Refined Data

The LERF development using the guidance in Reference 2 assumes that all sequences would be impacted by the potential for Type A leakage. However, the presence or absence of a Type A leak would not significantly alter those accident sequences that are already LERF contributors. Since the change in LERF for RNP exceeds the target value specified in Reference 5, a refined assessment is performed.

Reference 14 provides additional guidance and indicates that a more refined calculation can be performed if the calculated value exceeds this criterion. This more refined calculation is made to remove the existing LERF contribution from the estimation of Class 3b. This presents a more realistic impact of the ILRT extension. The calculation is presented below.

From Table 6 the baseline Class 3b frequency is 1.17E-7/yr and is derived by multiplying the total CDF by the probability of a Type A leak. The RNP analysis already includes LERF contributions from both interfacing systems LOCA and steam generator tube rupture events. From Reference 9, their contributions to LERF are captured in release categories RC-4, RC-4C, RC-5 and RC-5C. Early containment failures are also taken from RC-2 and RC-2B.

All of these contributions are currently considered LERF sequences and their contributions are listed in Table 11.

Table 11
LERF Release Category Contributions

Release Category	Frequency (/yr)	LERF Cumulative Frequency (/yr)
RC-4	ε^1	0
RC-4C	3.47E-6	3.47E-6
RC-5	1.20E-6	4.67E-6
RC-5C	6.31E-8	4.73E-6
RC-2	3.51E-8	4.76E-6
RC-2B	1.70E-7	4.93E-6

1. Contribution is less than truncation.

This contribution is already considered for LERF and can be excluded from the baseline CDF as presented in Equation 20.

Adjusted CDF = 4.29E-5/yr – 4.93E-6/yr = 3.79E-5/yr (eq. 19)

The Class 3b frequency is then determined by the following equation:

$FREQ_{class3b} = PROB_{class3b}$ x CDF = 0.0027 x 3.79E-5/yr = 1.042E-7/yr (eq. 20)

This can then be extrapolated using the methods presented earlier to determine the 10-year and 15-year contributions and to generate adjusted LERF values as presented in Table 12.

Table 12
Class 3b Contributions Using Adjusted CDF

Interval	Frequency (/yr)	Delta Frequency from Prior Period (/yr)
Baseline	1.04E-7	0
10-year (current)	3.46E-7	2.42E-7
15-year	5.18E-7	1.73E-7

As the results indicate, the value still slightly exceeds the figure of merit for the expansion of the testing interval from the current period (10 years) to the proposed period (15 years).

Revised LERF Based on Intact Containment

Reference 13 utilized only the intact containment contribution when calculating the Class 3 contributions. The analysis was based on the fact that the overall dose for any other case involving an impaired containment would be bounded by the existing dose rate such that the predicted Type A dose would be inconsequential. This is supported when the predicted doses are compared. Table 13 provides a comparison of the Type 3b dose to the Class 7 and Class 8 doses.

Table 13
Comparison of Class 3b, Class 7 and Class 8 Population Doses

Case	Population Dose (person-rem)	Normalized Dose to Class 3b
Class 3b	54,600	1.0
Class 7	2,160,000	39.6
Class 8	9,330,000	170.9

As the table indicates, the expected dose from the impaired containment is at least 40 times that from the Class 3b release. Only the intact containment case would, therefore, actually increase due to Type A leakage.

If this approach is utilized, the probability of Class 3b leakage (0.0027) is multiplied by the intact containment contribution (2.18E-5/yr) to generate the Class 3b frequency. This value is then extrapolated to the 10-year and 15-year cases using previously described multiplication factors. The results are then compared using the methods described earlier for determining the change in LERF. The results are summarized in Table 14.

Table 14
Class 3b Contributions Using Intact Containment Component Only

Interval	Frequency (/yr)	Delta Frequency from Prior Period (/yr)
Baseline	5.89E-8	0
10-year (current)	1.96E-7	1.37E-7
15-year	2.94E-7	9.81E-8

If only the intact containment contribution is considered, the results indicate that the increase for the period from the current interval (10 years) to the proposed interval (15 years) does not exceed the numeric requirement from Reference 5. This conclusion supports the extension of the ILRT testing interval.

Revised LERF Based on Elimination of LERF and non-LERF Contributions

Reference 14 indicates that the estimation of the impact to LERF can be refined by excluding that frequency already defined as LERF (see prior alternative calculation) and the frequency contribution from those sequences that would not result in a large early release regardless of the presence of a Type A failure due to scrubbing provided by containment sprays. The LERF contribution was presented in Table 11 as 4.93E-6/yr. This calculation examines the additional benefit of removing accident sequences that would have adequate scrubbing to preclude LERF releases.

The assessment assumes that containment spray must be available for both injection and recirculation to ensure scrubbing of released radionuclides that are released during initial reactor vessel failure and subsequent releases from radionculides released from the RCS after vessel failure. The end state (plant damage state) must also be an intact containment state since the unisolated containment states are already considered by the LERF fraction.

A review of the RNP containment safeguards event tree (CSET) identifies that this condition can be met for PDS states "A", "G" and "D" only.

PDSs with endstates "A" and "G" are associated with continuous operation of the containment sprays. PDS "D" is representative of a sequence where, based on the cutset results, recirculation fails, including a loss of component cooling water (CCW) cooling to the residual heat removal (RHR) heat exchangers. Since RHR is needed for containment spray operation during recirculation, a loss of RHR would entail a failure of containment sprays during recirculation. If the failure is due to a loss of cooling to the heat exchangers but the pumps continue to function then sprays would be available. This can happen if the fan coolers can remove adequate heat to maintain the water temperature below the maximum water temperature of the RHR pumps.

The PDS "D" contributing sequences involve reactor coolant pump (RCP) seal failures or a small LOCA. Secondary-side cooling provides a means to remove decay heat and this reduces the heat load to the containment prior to reactor vessel failure. A comparison of peak cavity water temperatures for cases with only RHR heat exchanger cooling to a case with only fan coolers indicates that continued operation is possible without heat exchanger cooling. Therefore, the assumption is made those contributions to PDS "D" can be excluded from the calculation of LERF since containment spray would be available to scrub any release. Table 15 lists the contributing PDSs that are from one of the three PDSs identified above.

Table 15
Contributing PDS States "A" and "G"

PDS	Frequency (/yr)
5D	9.83E-6
19A	2.12E-6
2A	1.96E-6
4D	4.88E-7
17D	4.66E-7
20D	4.50E-7
10A	3.72E-7
2D	3.58E-7
20A	3.56E-7
16A	4.98E-8
10D	1.97E-8
3G	6.75E-9
5A	7.21E-9
17A	5.89E-9
6D	3.53E-9
19D	3.69E-9
Total	1.65E-5

The first six PDS contributions in Table 15 represent 92.8% of the total and are reviewed to ensure their applicability and to ensure that the encompassed system failures would not result in a release of radionuclides sufficient to be classified as LERF.

PDS 5D is characterized in the cutsets as a loss of CCW initiating event with failure of the reactor coolant pump seals resulting in a small LOCA. The loss of CCW also precludes successful recirculation due to a loss of cooling water flow to the RHR heat exchangers. For the core damage model, credit is not taken for the cooling provided by the containment fan coolers. However, for the level 2 analysis, the containment fan coolers will provide sufficient recirculation cooling to maintain RHR pump operability and credit is taken for continued containment spray operation during recirculation[16]. The small LOCA leakage rate also delays the time to for the core to uncover and provides additional time for radionuclide transport and settling within the RCS piping. Therefore, the sequence would not provide a LERF contribution.

PDS 19A represents an ATWS overpressure sequence where the RCS is overpressurized resulting in failure. Injection is assumed to fail due to deformation of the discharge check valves. The containment sprays function. The sudden failure of the RCS at high pressure will significantly reduce the potential for deposition of the radionuclides within the RCS and could increase the environmental release given a Type A failure even with containment spray operation. To ensure that this is not the case, a MAAP comparison analysis was performed that examined the releases following a large LOCA without injection and a failure of all RCS cold legs[17]. The analysis assumed a 2-inch diameter isolation failure was present at the initiation of the event and tracked the time rate of releases of radionuclides. The results indicate that a somewhat higher release does occur, but it is less than a factor of two such that it would not be expected to lead to a LERF release and can be excluded.

PDS 2A involves a loss of offsite power with a failure of ac power to one of two emergency buses. This allows for one train of containment spray to function. However, due to the quantification modeling, a subsequent diesel generator failure that would fail the running containment spray pump is not addressed since this would be a non-minimal cutset. To address this occurrence, the probability of a subsequent diesel generator failure is added to the existing PDS evaluation to determine that fraction of the PDS that would not have containment spray operation.

The RNP database identifies that the most likely failure modes involve the diesel generator failing to run for 24 hours (2.70E-2), the failure of the diesel generator to start (6.52E-3) or the diesel generator being in maintenance (1.85E-2). The sum of these contributions is 5.2E-2. When this value is combined with the frequency of PDS 2A (1.96E-6/yr) the adjustment factor is determined (1.02E-7/yr).

The fraction of the PDS 2A frequency that would not lead to LERF, therefore is determined by Equation 21 below:

Adjusted PDS 2A = 1.96E-6/yr – 1.02E-7/yr = 1.85E-6/yr (eq. 21)

PDS 4D is similar to PDS 5D except that the first cutset involves a failure of the operators to initiate recirculation. The other contributors all involve CCW failure described for PDS 5D. The failure to perform recirculation would fail the containment sprays and cannot be credited. This contribution is removed as shown in Equation 22.

Adjusted PDS 4D = 4.88E-7/yr – 2.79E-7/yr = 2.09E-7/yr (eq. 22)

PDS 17D is defined by cutsets involving a small LOCA with failure of recirculation. Similar to PDS 4D, the first contributor involves an operator action to fail to accomplish recirculation. The remaining contributions are various CCW failures and are similar to those described earlier. This contribution is again adjusted to remove the operator action failure that would preclude containment spray operation (Equation 23).

Adjusted PDS 17D = 4.66E-7/yr – 2.31E-7/yr = 2.35E-7/yr (eq. 23)

Finally PDS 20D is represented by a failure of the reactor vessel in excess of makeup capacity. This sequence would be similar to the ATWS overpressure case and containment sprays would function. The lower radionuclide retention within the RCS would be bounded by the ATWS analysis discussed previously and this should not result in a sufficient increase in releases from the containment and the containment sprays should preclude any releases that would be approach those associated with LERF.

The reduction in PDSs 2A, 4D and 17D lowers the frequency of PDSs that will not result in a LERF release to 1.65E-5/yr. The adjusted CDF for calculating the Class 3b frequency is then calculated as shown in Equation 24 by removing both the LERF sequences and those PDSs that cannot result in a LERF release given a Type A failure.

Adjusted CDF = 4.29E-5/yr – (4.93E-6/yr + 1.59E-5/yr) = 2.21E-5/yr (eq. 24)

The Class 3b frequency calculation is presented in Equation 25:

$FREQ_{class3b} = PROB_{class3b}$ x CDF = 0.0027 x 2.21E-5/yr = 5.93E-8/yr (eq. 25)

The calculation indicates that using the assumptions listed above for LERF and non-LERF sequences results in a total LERF that is below the 1.0E-7/yr. This result is then extrapolated to 10 years and 15 years and the differences calculated in Table 16.

Table 16
Class 3b Contributions Using Adjusted CDF

Interval	Frequency (/yr)	Delta Frequency from Prior Period (/yr)
Baseline	5.93E-8	0
10-year (current)	1.98E-7	1.38E-7
15-year	2.97E-7	9.88E-8

As the table indicates, the change from the current testing interval (10 years) to the proposed interval (15 years) meets the 1.0E-7/yr criterion for an insignificant increase in risk. The increase from the baseline case (3 years) is only slightly above the 1.0E-7/yr criterion.

Impact on Total LERF Due to ILRT Extension

Reference 14 indicates that the proposed extension represents a one-time deferral and that Reference 5 provides for consideration of changes that may exceed 1.0E-7/yr so long as the total LERF remains under 1.0E-5/yr.

The total LERF from internal events, including internal flooding, for RNP is 4.93E-6/yr. This result does not consider the Class 3b contribution. Including this contribution for the 10-year case (3.91E-7/yr) yields a total LERF of 5.32E-6/yr.

The proposed change would increase the LERF contribution to by 1.95E-7/yr to 5.52E-6/yr. This equates to an increase of approximately 4%. This is a small increase relative to the total LERF. The total LERF, therefore, is still less than the 1.0E-5/yr.

Another factor to consider is the impact of other initiating events that are not included in the RNP PSA. Events at shutdown are not of consequence since the containment status is typically unisolated and any event would be slow acting and provide opportunity for mitigation.

Other external events, i.e., internal fire, seismic, high winds, could be of some consequence and should be considered. These events were addressed in the Individual Plant Examination for External Events[15] but have not been updated since this analysis. A further complication is that seismic events were addressed using seismic margins and no frequency information is available. Therefore a qualitative argument is provided based on comparisons to the internal events analysis to generate a quantitative estimate for LERF for these events.

The external events of importance typically result in plant challenges that are similar to transient events addressed within the PSA. This includes loss of heat removal, seal LOCA, PORV LOCA and ATWS. The accident progression is typically not altered by the event and the main impact is the propagation of common faults due to the impact of the external event on multiple systems. This is similar to support system failures that support multiple systems such as service water.

Since the impact is similar to a transient event with loss of one or more trains of equipment, a simple estimate for external event LERF can be generated by observing the contributions from transient internal events. Since the total frequency from external events not included in the PSA would be expected to be less than the total frequency of modeled transients, the contribution from these transient events is assumed to be bounding.

The LERF contributions from steam generator tube rupture and interfacing system LOCA initiating events are associated with RC-4, RC-4C, RC-5 and RC-5C are on the order of 4.73E-6/yr. The total LERF contribution, defined earlier, is 4.93E-6/yr and includes this contribution.

If the LOCA contribution is assumed to be small the total transient LERF can be estimated by subtracting the SGTR and interfacing systems LOCA LERF estimates from the total LERF (4.93E-6/yr – 4.73E-6/yr = 2.05E-7/yr). This value can be used as a bounding surrogate for the external event LERF.

When combined with the prior LERF contribution given an extended ILRT testing interval (5.52E-6/yr) the estimated total LERF increases to 5.72E-6/yr. This value remains below 1.0E-5/yr and indicates that the proposed change would not result in an unacceptable increase in total LERF as defined by the information in Reg. Guide 1.174.

Evaluating the proposed change using the simplified analysis based on the NEI approach did not yield a solution that meets the Reg. Guide 1.174 criterion for an insignificant change in risk. However, the performance of these alternative studies indicates that the proposed ILRT extension can meet the acceptance criteria when a more detailed assessment of possible LERF sequences due to Type A leakage is performed.

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17. Calculation RSC-CALKNX-2002-0201, RNP Source Term Release Comparison Between Small and Large LOCAs and ATWS Overpressurization Events, Revision 0, February 2002.

United States Nuclear Regulatory Commission
Attachment VII to Serial: RNP-RA/02-0028
4 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING
ONE-TIME EXTENSION OF CONTAINMENT TYPE A TEST INTERVAL

AFFIDAVIT AND APPLICATION FOR WITHHOLDING FROM PUBLIC DISCLOSURE

Affidavit and Request for Withholding of Information

Before me, the undersigned authority, personally appeared Ricky Summitt, President of Ricky Summitt Consulting (RSC), Inc., who, being me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of RSC, Inc. and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Ricky Summitt

Ricky Summitt, President

Sworn to and subscribed

Before me this 6th day

Of March ,2002

Notary Public

My Commission Expires 3-27-2002

I am the president of Ricky Summitt Consulting (RSC), Inc. and as such, I have the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of RSC, Inc, a Tennessee corporation.

I am making the Affidavit in conformance with the provisions of 10CFR Section 2.790 of the Commission's regulations.

I have personal knowledge of the criteria and procedures utilized by RSC, Inc in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

The information sought to be withheld from public disclosure is owned and has been held in confidence by RSC, Inc. The information is of a type customarily held in confidence by RSC, Inc. and not customarily disclosed to the public. RSC, Inc. has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes RSC, Inc. policy and provides the rational basis required. The policy identifies that information is held in confidence if any of the following apply:

a) The information reveals the distinguishing aspects of a process (or component, structure, tool method, etc.) where prevention of its use by any of RSC's competitors without license from RSC, Inc. constitutes a competitive economic advantage over other companies.

b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

The policy and request are based on the following:

a) The use of information such as that embodied in RSC 01-44 provides RSC, Inc. a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the RSC, Inc. competitive position.

b) The information contained in RSC 01-44 is marketable in many ways. The extent to which such information is available to competitors diminishes the RSC, Inc. ability to sell products and services involving the use of the information.

c) Use by our competitor of this information would put RSC, Inc. at a competitive disadvantage by reducing his expenditure of competitor resources at our expense.

d) Unrestricted disclosure would jeopardize the position of prominence of RSC, Inc. in this area and thereby give a market advantage to other competition.

e) The RSC, Inc. capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

The information is being transmitted to the Commission in confidence and, under the provisions of 10CFR Section 2.790, it is to be received in confidence by the Commission. The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief. The proprietary information sought to be withheld in this submittal is that which is approximately marked as RSC 01-44 titled "Surrogate Level 3 Evaluation Methodology", Revision 0.

This information is part of that which will enable RSC, Inc. to:

- Develop person-rem estimates in support of the ILRT extension request.

Further this information has substantial commercial value as follows:

- RSC, Inc. could sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation.

- RSC, Inc. can sell support and defense of the technology to its customers in the licensing process.

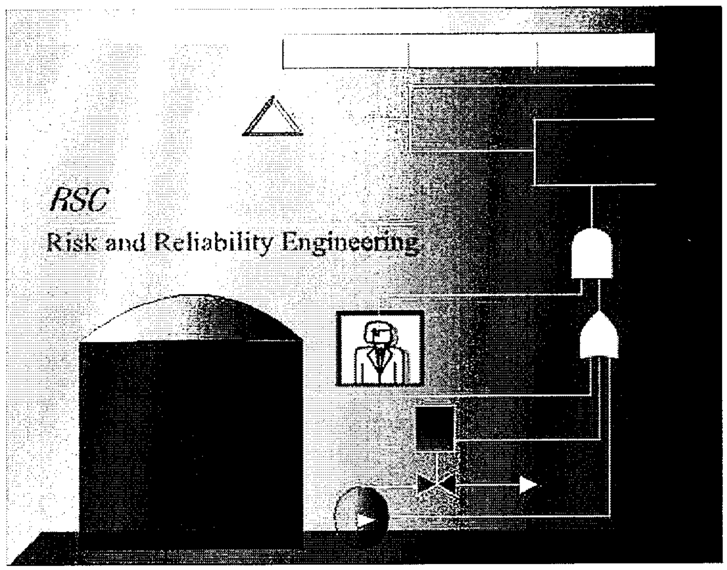
Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of RSC, Inc. because it would enhance the ability of competitors to provide similar licensing support documentation and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the RSC, Inc. collective memory and the expenditure of a considerable sum of money. In order for competitors of RSC, Inc. to duplicate this information, similar research efforts would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be extended for developing analytical methods and performing tests.

United States Nuclear Regulatory Commission
Attachment VIII to Serial: RNP-RA/02-0028
16 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING
ONE-TIME EXTENSION OF CONTAINMENT TYPE A TEST INTERVAL

RSC 02-12, APPENDIX A, RSC 01-44NP, "SURROGATE LEVEL 3 EVALUATION METHODOLOGY," NON-PROPRIETARY VERSION



Surrogate Level 3 Evaluation Methodology

Revision 0

August 2001

Principal Analyst

Ricky Summitt

NON-PROPRIETARY DOCUMENT
This document has been reviewed and proprietary information removed. It may be freely distributed as a complete document only.

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Report Review and Resolution Form
FORM NO.: RSC-RPT-RVR00-02Rev. 2

Preparer:		Ricky Summitt
RSC Reviewer: R. Summitt		Date: August 6, 2001
RSC Approver: R. Summitt		Date: August 6, 2001
Abstract (brief statement of purpose): Document methodology for converting radionuclide release fractions to dose. NOTE: Document grandfathered and does not require independent review since prior client review.		
Documentation Retrieval Information: Keywords: Level 2 Analysis Other Calculation MAAP Analysis		
☒ Amends / ☐ Superceeds / ☐ Supplements RSC Document(s): PSA Paper (Reference 1).		
Verification and Review Method: ☐ Detailed Review ☐ Alternative Calculation ☐ Qualification Testing ☒ Other (specify: Grandfathered)		
General Documentation Requirements	Acceptable	Reviewer Comments
Introduction – provides summary of purpose, scope, and principle tasks required to meet objective	☐	
Methodology – description of process and supporting methodology that is sufficient to understand approach and to support peer review	☐	
Analysis and Results – detailed documentation of the implementation of the methodology and task steps that may be supported by report appendices and includes intermediate and final results	☐	
Conclusions and Recommendations – concise presentation of results of the analysis that answers the objectives of the study and should include any important assumptions and/or findings	☐	NA
Editorial Review: ☒ Spell Checked ☒ Grammer Checked ☒ Tables and Figures Checked ☒ Sections Checked		
Sufficient References to Reproduce Results: Yes		
Resolved all Comments: NA		Incorporated Resolutions From Review: NA

Reviewer Comment	Resolution of Comment
1. NA	

Editorial or illustrative comments are attached to this review sheet to complete the review package.

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1.0 INTRODUCTION

The current industry emphasis is on applying the PSA to assist in plant operational decision-making. Most of the IPE submittals stop at the frequency of containment release and do not address offsite consequences. Since public safety is a primary consideration, it is important to have a tool that provides insights into how potential changes will impact public health risk.

Although a primary measure currently being proposed examines changes in the large early release fraction (LERF), the total effect should also be considered when evaluating changes.

The total whole body person-rem released is one measure to address the change in public health risk due to a proposed change to plant configuration. This quantity is considered one possible measure of merit and is traditionally calculated for the Level 3 PSA.

Given that most PSAs stop at containment release, additional effort is needed. To generate the person-rem release in order to expand the evaluation it is necessary to develop a model for extrapolating the existing information in the PSA to person-rem.

One approach to accomplish this task is to expand the existing PSA into a Level 3 PSA. This requires information on meteorological conditions, population densities, and evacuation planning. This information is then input into an offsite analysis code and results generated. The effort required to develop this detailed model may not be necessary for most cases.

A surrogate model can be used to estimate the change in whole body person-rem based on existing analyses[1]. The process used to develop the model is present in this report.

2.0 METHODOLOGY

The basis for the surrogate model is the development of a relationship between the radionuclide release fractions and the predicted whole body person-rem. To make the model useful, this relationship is developed at a release category level and in terms of a minimal set of radionuclide release fractions that, based on prior studies, can be shown to control the various aspects of offsite doses. This is accomplished by examining several prior studies that included measures of offsite consequences.

3.0 DEVELOPMENT OF RADIONUCLIDE RELEASE TO PERSON-REM RELATIONSHIP

The understanding that the dose values must be considered in terms of the "fence post" dose is key to the model development. In other words, the dose that the envelop around the plant would receive. This allows the results to be independent of evacuation and meteorological considerations. The result may be somewhat conservative, but it provides a measure that can be applied across plant sites uniformly.

3.1 DATA ASSESSMENT EXISTING

The results of the Level 2 IPE assessment are typically provided in terms of release category frequencies and radionuclide release fractions. Therefore, any method must utilize these two characteristics form the basis for estimating the offsite consequence from release sequences to be useful.

To determine this relationship, available published and unpublished Level 3 PSAs were reviewed to determine a range of release fractions and corresponding doses. The release fractions identified in these PSAs for the following radionuclides: noble gases, iodine, cesium, tellurium, strontium, ruthenium, lanthanum, cerium and barium. The relative release fractions for each were collected as identified in the PSAs.

These radionuclides are most reported in the literature and provide the majority of offsite dose. The release fractions for each of the release categories is cataloged (each release category is defined as a *case*) along with the associated whole body person-rem. Figures 1 through 4 graphically presents the results for four PSAs as examples of this effort.

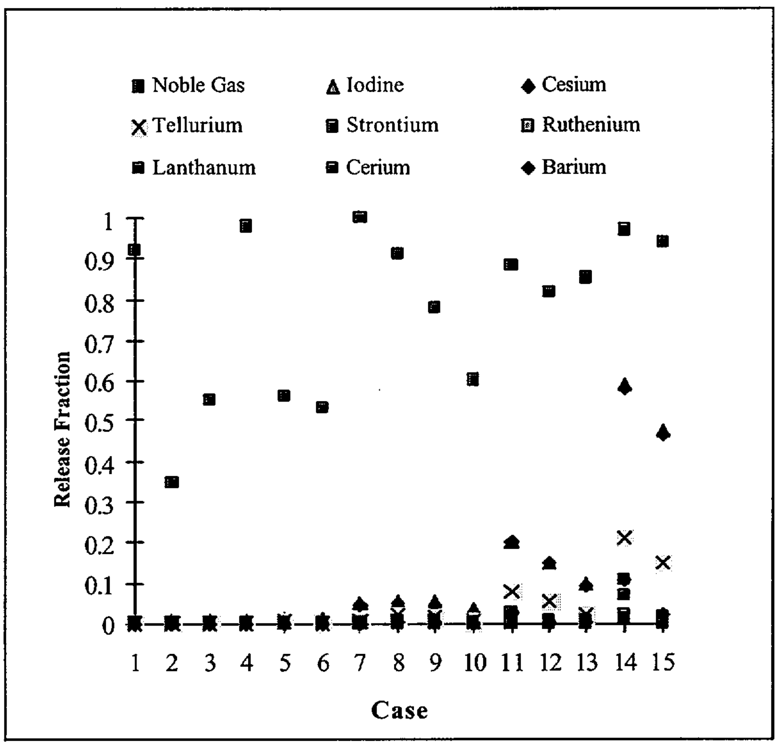

Figure 1 Sequoyah Release Fraction Cases (Reference 2)

[• • •]

Figure 2 Unpublished PWR Release Fraction Cases (Reference 3)

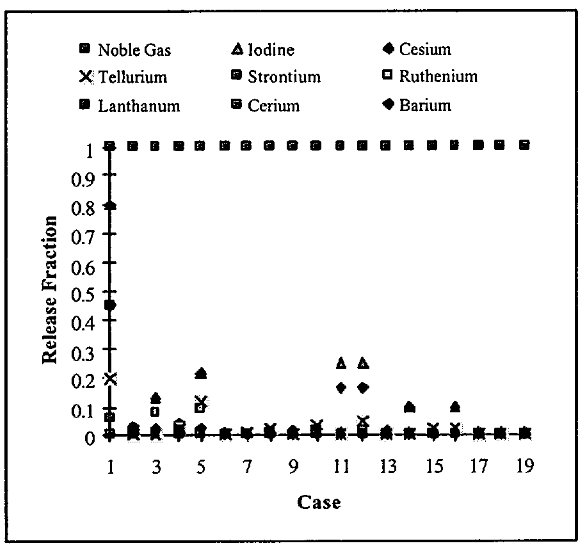

Figure 3 Oconee IPE Release Fraction Cases (Reference 4)

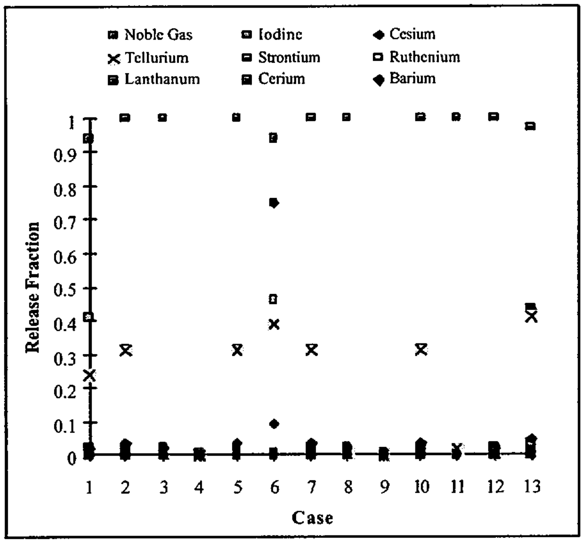

Figure 4 Seabrook Release Fraction Cases (Reference 5)

3.2 DATA INTERPRETATION

From these studies, a total of 56 unique release categories, defining radionuclide fractions and person-rem were plotted on a normalized plot to determine the type of relationship that existed between dose and release fractions. Five of the more important radionuclides were used to develop the release fraction value. These five radionuclides, noble gases, [···],[···],[···], and [···], are all considered important contributors to offsite dose.

Noble gas releases were chosen to represent the "baseline" dose. Most studies indicate that if a release occurs, the vast majority of noble gases will be released. The others were chosen based on their relatively important biological effects and tend to be significant release contributors. Figure 5 shows how the dose essentially maps the release fraction.

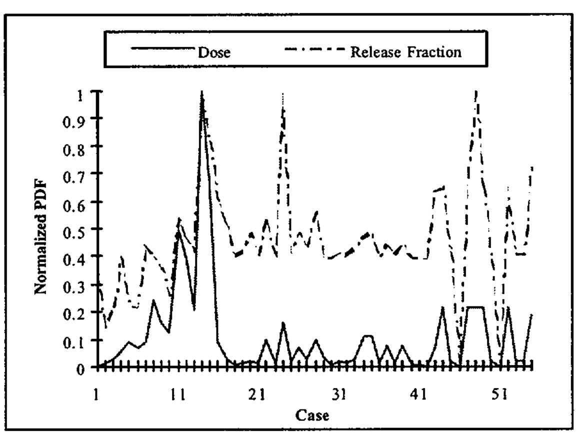

Figure 5 Relationship between Release Fraction and Dose

Although a clear linear relationship does not exist between the two functions, it is clear that a trend is found between the fraction released and the resulting dose. This is hardly a revelation since the dose exposure is a function of the radionuclides released. The simplicity of the relationship, [···], is somewhat of a surprise. Given this relationship, a set of 56 [···] equations was developed. For each case, the equation took the form:

[···]

where: D_i=dose for case i

Xn_i = the release fraction for the key radionuclide n and case i

A,B,C,D, and E are constants.

These equations were setup as a series of simultaneous equations and the constants varied until an optimal solution to all equalities was determined. The correlation was obtained by matching the values generated by the equation to the whole body dose reported in the literature. Figure 6 presents the correlation for the 56 cases obtained for the final solution.

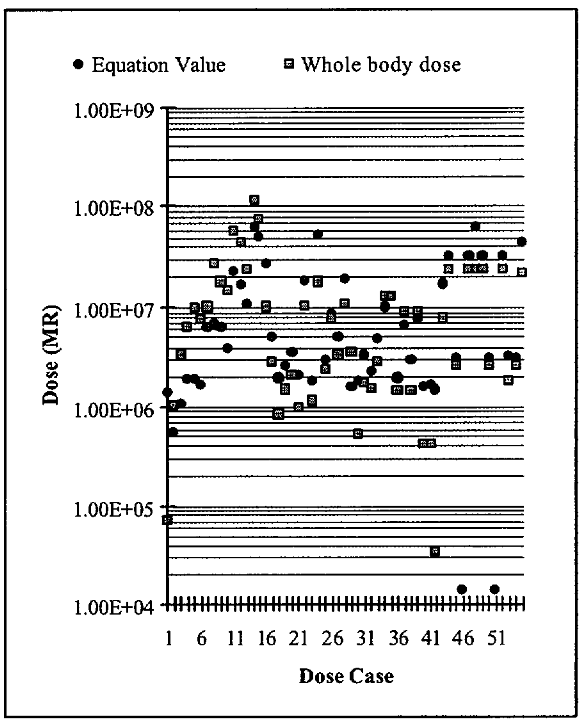

Figure 6 Comparison of Equation Results and Reported Dose Values

The factors used to serve as constants that provide the best solution are presented in Table 1.

Table 1
Release Split Fraction to Dose Conversion Factors

Constant	Radionuclide Group	Value
A	Noble gases	[···]
B	[···]	[···]
C	[···]	[···]
D	[···]	[···]
E	[···]	[···]

3.4 APPLICATION WITH MAAP

The MAAP code provides radionuclide release fractions for significant radionuclides given a failure of containment. The release fractions can be used along with the method presented in this document to estimate the person-rem release.

In order to perform the calculation it is necessary to define what radionculide categories, as defined by MAAP, are needed. Table 2 lists the radionuclide categories utilized and how these radionuclides are mapped to the variables in the methodology.

Table 2
Mapping of Method Variables to MAAP Output Variables

Equation Variable	MAAP Output Variables
X1	Noble gas
X2	[···]
X3	[···]
X4	[···]
X5	[···]

Several of the surveyed PSAs utilized MAAP results to define the release category source term and the correlation has shown to be applicable if these MAAP variables are utilized.

3.5 QUALITATIVE UNCERTAINTY ASSESSMENT

The objective of this activity is to develop a realistic tool for estimation of person-rem. The process must not introduce excessive or unpredictable uncertainty. Two aspects of uncertainty that impact the analysis are the uncertainty in the generated magnitude and the consistency of the overall predictions.

3.5.1 Qualitative Evaluation of Predictive Dose

In addition to choosing the best fit for the 56 cases, the variation of the result for each unique case was examined. Figure 7 plots the variation from the reported value for each of cases. The range represents a deviation of a factor of two (2) in either direction.

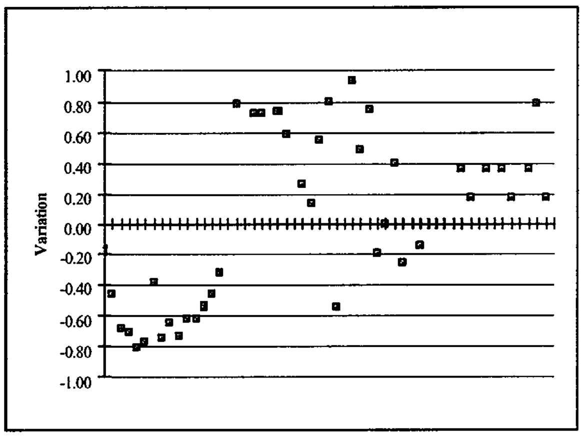

Figure 7 Variation of Equation to Reported Dose

As shown, most calculated values do not vary from the reported value by more than 50%. Given that the most likely use of this evaluation is to perform an assessment of relative change and that large uncertainties are already present in the PSA, errors of this magnitude (less than a factor of 2) are not significant.

The equation, however, was found to significantly over predict dose for cases involving intact containment leakage rates. In these cases, the offsite dose was less than 1.0E+5 person-rem and the variation approached a factor of 50. Thus, the equation may not be appropriate for intact containment cases. The cause of this error is the noble gas contribution. A basic assumption for impaired containment cases is that essentially 100% of noble gases are released such that the noble gas release is essentially a baseline dose as stated earlier. This is not the case for intact containments and the constant chosen for the noble gas contribution is significantly overestimated. This limitation, however, does not affect the use of this model since any assessment would be based on results for impaired containment events. Existing licensing basis analyses can cover intact containment doses and it is this data that is the support for the intact containment release category.

3.5.2 Results Predictability

To have confidence in the method it is necessary for the analysis to be internally consistent. This does not preclude generating conservative or non-conservative results. It does require that the results generated are not bimodal resulting in significant differences in the trend of the results. For example, if one release category is underestimated and another overestimated the importance of the two release categories will be incorrect. If both are slightly overestimated the relative importance will be maintained.

An evaluation of the results (see Figures 6 and 7) indicates that the model consisting estimates a value slightly greater than the reference value. For intact containment cases, however, this was not the case. The value was significantly overestimated and again this supports not using this approach for intact containment cases. Figure 7 also shows several cases when the values were slightly under predicted. This was a single plant with an older evaluation of source term not representative of the current state of knowledge and the underestimation is appropriate and more representative of expected source term. Again the analysis is internally consistent. The method is consistent to provide predictable results and the uncertainty from this aspect is small.

4.0 SUMMARY AND CONCLUSIONS

A simplified model for addressing offsite risk is possible using existing PSA information and can be based on relatively few radionuclides. The development of this model can provide a useful tool to evaluate potential plant configuration changes and improvements.

The use of this model to calculate the impact of proposed changes can be used to assess the impact of procedural changes, operating status, or other modifications on a relative change in whole body person-rem.

It is important to mention that person-rem is only one of the factors that should be considered and that it is not usually the most restrictive when evaluating total risk. The lost plant investment and replacement power costs must also be considered internally in the decision process. The use of a health risk measure such as person-rem, however, does provide a type of regulatory perspective on potential changes in plant status or configuration.

5.0 REFERENCES

1. Summitt, R., *Development of a Surrogate Risk Measure for Risk Benefit Assessment*, PSA 96, September 29-October 3, 1996.
2. Benjamin et al., Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Sequoyah Power Station, Unit 1, United State Regulatory Commission, NUREG/CR-4551, Vol. 2, in press.
3. Unpublished PSA.
4. W. Sugnet et al., Oconee PRA, A Probabilistic Risk Assessment of Oconee Unit 3, The Nuclear Safety Analysis Center and Duke Power Company, NSAC-60, June 1984.

5. Garrick et al., Seabrook Station Probabilistic Safety Assessment, Pickard, Lowe, and Garrick, Inc., PLG-0300, December 1983.