CP&L
A Progress Energy Company

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10 CFR 50.90

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS REGARDING FREQUENCY OF PERFORMANCE-BASED LEAKAGE RATE TESTING

Ladies and Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Parts 50.90 and 2.101, Carolina Power & Light (CP&L) Company is requesting a revision to the Technical Specifications for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2. The proposed license amendments will revise Technical Specification 5.5.12, "Primary Containment Leakage Rate Testing Program," to incorporate a one-time exception to the 10-year frequency of the performance-based leakage rate testing program for Type A tests as specified by Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and endorsed by 10 CFR Part 50, Appendix J, Option B. The new exception will allow performance of a Type A test within 15 years, one month from the last Type A test for Unit 1 and 15 years for Unit 2. The last BSEP, Unit 1 Type A test was performed on February 15, 1991; the last BSEP, Unit 2 Type A test was performed on February 28, 1993.

A plant-specific, risk-based evaluation (i.e., Enclosure 2, RSC 01-24, Revision 0, "Evaluation of Risk Significance of ILRT Extension") has been performed in support of the one-time exception to extend the Type A test from once in 10 years to once in 15 years. The guidance in NEI 94-01, Electric Power Research Institute (EPRI) Topical Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," and NRC Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes To The Licensing Basis," dated July 1998, was used in performing this risk assessment.

The plant-specific risk assessment uses the latest peer-reviewed BSEP Level 1 and Level 2 probabilistic safety assessment (PSA) models to estimate the changes in risk associated with increasing the Type A testing interval. This risk assessment is best-

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estimate, consistent with current PSA best practices. The release category and person-rem information is 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 01-24.

In a separate license amendment application dated August 1, 2001 (Serial: BSEP 01-0063), CP&L has requested approval for a full-scope implementation of an Alternative Radiological Source Term (AST). The offsite dose consequences resulting from the AST evaluations associated with a design basis loss of coolant accident have been used as input into this best-estimate risk assessment.

One input assumption used in certain AST calculations is the efficiency of the high efficiency particulate absorbers (HEPAs) in the Standby Gas Treatment System. Based on recent discussions with the NRC, CP&L is revising the value assumed for the efficiency of these HEPA filters in certain AST calculations. An initial, qualitative assessment of these AST-related changes indicates that the change in HEPA efficiency does not significantly change the offsite design basis loss of coolant accident dose inputs used to develop this best-estimate risk assessment. As such, CP&L has determined that the conclusions of the risk assessment will remain valid; that is, the proposed change to the Type A test frequency does not represent a risk significant change.

This license amendment application represents a cost-beneficial licensing action. The Type A test imposes significant expense on CP&L while the safety benefit of performing the Type A test within 10 years, versus 15 years, is minimal. This request is similar to a license amendment authorized by the NRC on August 30, 2001 (i.e., ADAMS Accession Number ML012190219) for the Crystal River Nuclear Plant, Unit 3.

As discussed in Enclosure 3, this license amendment application does not involve a significant hazard consideration in accordance with 10 CFR 50.92. Also, as discussed in Enclosure 4, CP&L has determined that this license amendment request meets the criteria of 10 CFR 51.22(c)(9) and 10 CFR 51.22(c)(10) for a categorical exclusion from the requirements for an Environmental Impact Statement.

To support incorporation of the Type A testing changes into the schedule for the upcoming BSEP, Unit 1 Refueling Outage 13 (i.e., designated as B114R1), CP&L requests issuance of the proposed license amendments by March 1, 2002. Currently, Unit 1 Refueling Outage 13 is scheduled to begin March 2, 2002. To support implementation of the Technical Specification changes associated with these proposed license amendments, CP&L requests an implementation period of 60 days following issuance of these license amendments.

In accordance with 10 CFR 50.91(b), CP&L is providing a copy of the license amendment application to Mr. Mel Fry of the State of North Carolina. In accordance with 10 CFR 50.4(b)(1), CP&L is providing a copy of the license amendment application to the NRC Region II Office and the NRC Resident Inspector.

Please refer any questions regarding this submittal to Mr. David C. DiCello, Manager - Regulatory Affairs, at (910) 457-2235.

Sincerely,

John S. Keenan

John S. Keenan

WRM/wrm

Enclosures:

1. Basis For Change Request
2. RSC 01-24, "Evaluation of Risk Significance of ILRT Extension"
3. 10 CFR 50.92 Evaluation
4. Environmental Considerations
5. Page Change Instructions
6. Typed Technical Specification Pages - Unit 1
7. Typed Technical Specification Pages - Unit 2
8. Marked-up Technical Specification Pages - Unit 1
9. Marked-up Technical Specification Pages - Unit 2

John S. Keenan, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, and agents of Carolina Power & Light Company.

Diane S. Mann

Notary (Seal)

My commission expires: August 29, 2004

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ENCLOSURE 1

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS REGARDING FREQUENCY OF
PERFORMANCE-BASED LEAKAGE RATE TESTING

Basis For Change Request

Introduction

In accordance with the Code of Federal Regulations, Title 10, Parts 50.90 and 2.101, Carolina Power & Light (CP&L) Company is requesting a revision to the Technical Specifications for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2. The proposed license amendments request a one-time exception to the 10-year frequency of the performance-based leakage rate testing program for Type A tests, as specified by Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and endorsed by 10 CFR Part 50, Appendix J, Option B. The one-time exception applies to the requirement of NEI 94-01 to perform an integrated leak rate test (ILRT) at a frequency of up to 10 years, with allowance for a 15 month extension. The exception is to allow performance of an ILRT within 15 years, one month for Unit 1 and 15 years for Unit 2. The last BSEP, Unit 1 ILRT was performed on February 15, 1991; the last BSEP, Unit 2 ILRT was performed on February 28, 1993.

Background

Technical Specification 5.5.12, "Primary Containment Leakage Rate Testing Program" currently states:

> A primary containment leakage rate testing program shall establish requirements to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995, as modified by the following exceptions:
>
> [Items "a" through "e" omitted].

Technical Specification 5.5.12 for BSEP, Unit 1 is being revised to add the following exception to the list of approved exceptions:

> f. NEI 94-01 – 1995, Section 9.2.3: The first Type A test performed after the February 15, 1991, Type A test shall be performed no later than March 21, 2006.

For BSEP, Unit 1, the proposed date by which the Type A test must be performed has been increased by approximately one month beyond the nominal 15 year period discussed in this license amendment application. This adjustment has been made based on the currently scheduled start date and planned duration for the 2006 Unit 1 refueling outage. Since the unit will be shut down during the approximately one month increase, the conclusions of the probabilistic safety assessment evaluation are not impacted.

Technical Specification 5.5.12 for BSEP, Unit 2 is being revised to add the following exception to the list of approved exceptions contained in BSEP, Unit 2 Specification 5.5.12:

f. NEI 94-01 – 1995, Section 9.2.3: The first Type A test performed after the February 28, 1993, Type A test shall be performed no later than February 28, 2008.

Basis For Proposed Change

Primary containment provides an essentially leak-tight barrier against the uncontrolled release of radioactivity into the environment following a design basis accident. The testing requirements of 10 CFR Part 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 Technical Specifications. The limitation of containment leakage provides assurance that the containment will perform its design function following a design basis accident.

Effective October 26, 1995, 10 CFR Part 50, Appendix J, was revised to allow licensees to choose to perform containment leakage testing under Option A, "Prescriptive Requirements" or Option B, "Performance-Based Requirements." On February 1, 1996, License Amendments 181 and 213 for BSEP, Units 1 and 2, respectively, were issued to permit implementation of 10 CFR Part 50, Appendix J, Option B. The license amendments added Technical Specification 6.8.3.4, "Primary Containment Leakage Rate Testing Program," which required Type A, B, and C testing in accordance with NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by certain exceptions. Technical Specification 6.8.3.4 was subsequently changed to Specification 5.5.12 when BSEP, Units 1 and 2 adopted the Improved Standard Technical Specifications. Regulatory Guide 1.163 specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01 and American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-1994, "Containment System Leakage Testing Requirements," subject to several regulatory positions in the guide.

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 is performed, but it did alter the frequency of measuring primary containment leakage in Type A, B, and C tests. 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 frequency for 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, and the small fraction of leakage detected solely by Type A testing, increasing the interval between ILRT testing had minimal impact on public risk.

- While Type B and C tests identify the vast majority (i.e., 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.

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 10 years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate was less than $1.0L_a$) and consideration of the performance factors in NEI 94-01, Section 11.3. Based on the results of the last two Type A tests performed in BSEP, Units 1 and 2, the current interval for both BSEP units is once every 10 years.

A Type A test can detect containment leakage due to a loss of structural capability. All other sources of containment leakage detected in Type A test analyses can be detected by the Type B and C tests, and the IWE/IWL inservice inspections.

The results for the last two Type A tests for each BSEP unit are reported in the following table:

Unit	Test Date	* Test Leakage (% wt./day)	Test Pressure (psig)	Performance Based Acceptance Criteria (% wt./day)
1	5/19/87	0.215	50.0	0.5
1	2/15/91	0.341	49.5	0.5
2	12/12/91	0.355	50.3	0.5
2	2/26/93	0.351	50.4	0.5

* The Total Time method values, including penalties, isolated leakage paths and water level changes that could have affected the free volume, are identified because they were more conservative (i.e., higher) than those utilizing the Mass Point Analysis technique. As identified in NEI 94-01, leakage savings were not added to the leakage total.

Previous Type A tests confirmed that both BSEP reactor containment structures have very low leakage and represent minimal risk to increased leakage. The risk is minimized by continued local leak rate testing (i.e., Type B and C), continued performance of the general visual inspection required by Regulatory Guide 1.163, the Maintenance Rule (i.e., 10 CFR 50.65) inspection program, and the implementation of the Containment Inservice Inspection Program. As a result, BSEP containment leakage that is present has been and will continue to be detected in Type B and C testing. The existing Type B and C testing will continue to be performed in accordance with 10 CFR 50, Appendix J, Option B.

Regulatory Guide 1.163 Containment Visual Examination

CP&L has established procedures for performing visual examination of the accessible surfaces of the containment for detection of structural problems. Regulatory Guide 1.163, Regulatory Position C.3 specifies that these examinations 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.

Containment Inspection Program Activities

CP&L has developed a Containment Inspection Program as stipulated by 10 CFR 50.55a(g)(6)(ii)(B). The Containment Inspection Program was established in 1998, in accordance with Subsections IWE and IWL of the American Society of Mechanical Engineers (ASME), Section XI, 1992 Edition, to assure detection of deterioration affecting containment integrity. The first interval of the BSEP Containment Inspection Program began in 1998 and ends in 2008.

10 CFR 50.55a(g)(6)(ii)(B) required expedited examination of the accessible portions of primary containment liner. The most recent visual inspections of the BSEP, Units 1 and 2 containments were performed in 2000 and 2001, respectively. These visual inspections were performed by qualified personnel under the direction of a Registered Professional Engineer. The results of these inspections identified no adverse conditions.

For BSEP, the containment surface areas that have been identified as requiring augmented examinations and classified as Category E-C are limited to the containment vent system. Small areas of the vent system have experienced accelerated degradation caused by accumulation of water and end-of-service life of the protective coating. None of these identified areas have challenged the leak tightness of the containment. The applicable areas were repaired and recoated, and corrective measures have been employed.

On August 10, 1999, the NRC authorized an alternative to Code-required VT-3 visual examinations of seals and gaskets (i.e., Relief Request CIP-04). The approved alternative involves verification, through the use of Type B testing, the leak-tight integrity of seals and gaskets used on penetrations. The proposed change to the Type A test frequency will not alter the integrity of these items because they are being subjected to Type B testing in accordance with

10 CFR 50, Appendix J. Containment bolting is being examined during each inspection period; these examinations will not be affected by the proposed change to the Type A test frequency.

The BSEP containment design employs a single-ply bellows. These containment bellows are located inside the suppression chamber and are insulated by a protective metal cover. The controlled atmosphere of the suppression chamber (i.e., a nitrogen atmosphere which is maintained during power operation), the protective cover over the bellows, and the location ensure an environment that is resistant to stress corrosion cracking.

To assure comprehensive inspection of the containment, the Containment Inspection Program has been integrated with visual inspection activities performed in conjunction with Type A testing, as well as with Maintenance Rule activities. The integration of these inspection activities provides a consistent and effective approach for assessing the condition of the containment and assuring detection of degradation. There will be no change to the schedule for the Containment Inspection Program activities as a result of this license amendment application.

Plant Operational Performance

BSEP, Units 1 and 2 are boiling water reactors contained in a Mark I containment. During power operation, the primary containment atmosphere is inerted with nitrogen to ensure that no external sources of oxygen are introduced into containment. The containment inerting system is used during the initial purging of the primary containment prior to power operation and provides a supply of makeup nitrogen to maintain primary containment oxygen concentration within Technical Specification limits. As a result, the primary containment is maintained at a slightly positive pressure during power operation. During power operation, instrument air system (i.e., nitrogen) leaks occur from pneumatically-operated valves inside the containment which gradually pressurize the primary containment. Primary containment pressure is monitored in the control room. The primary containment atmosphere is periodically vented in order to maintain containment pressure within an acceptable operating range. This cycling of the primary containment pressure during operation amounts to a periodic integrated pressure test of the containment at a low differential pressure. Although this cycling does not challenge the structural and leak tight integrity of the primary containment system at post-accident pressure, it provides assurance that a gross containment leakage that may develop during power operation will be detected. This feature is a complement to visual inspection of the interior and exterior of the containment structure for those areas that may be inaccessible for visual examination. In the event pressurization does not occur, a leakage path may be present. Plant operators are aware of the implications of lack of pressurization during power operation. Following approval of this license amendment application, administrative controls will be established to monitor containment depressurization activities and evaluate trends (e.g., frequency, duration) for indication of changes to containment leakage.

Plant-Specific Risk Assessment

A plant-specific risk assessment has been performed in support of the one-time exception to extend the Type A test from once in 10 years to once in 15 years. The results of the plant-specific risk assessment are documented in RSC 01-24, Revision 0, "Evaluation of Risk Significance of ILRT Extension." A copy of the plant-specific risk assessment is provided in Enclosure 2.

The plant-specific risk assessment uses the latest peer-reviewed BSEP Level 1 and Level 2 probabilistic safety assessment (PSA) models to estimate the changes in risk associated with increasing the Type A testing interval. The release category and person-rem information is 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 01-24, Revision 0. The assessment uses the methodology described in Electric Power Research Institute (EPRI) Topical Report TR-104285 to estimate plant risk on specific accident sequences impacted by Type A testing.

The guidance in NRC Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes To The Licensing Basis," dated July 1998, on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a licensee request for changes to a plant's licensing basis, was used in evaluating the results of this risk assessment.

The plant-specific risk assessment determined that a change in the Type A test frequency from 10 years to 15 years will have an extremely small change in population dose consequences. Specifically, the proposed Type A test frequency change from 10 years to 15 years will result in only a 0.00096 percent increase in total integrated plant risk. In comparison, the change in the Type A test frequency from 3 years to 15 years increases total integrated plant risk by only 0.00214 percent.

Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 1E-6 per year and increases in Large Early Release Frequency (LERF) below 1E-7 per year. The proposed extension of the Type A test interval does not have an impact on CDF; therefore, the change in LERF provides the appropriate assessment of the change in risk associated with the proposed change. The increase in LERF resulting from the proposed Type A test frequency change from 10 years to 15 years is 5.14E-8. In comparison, the increase in LERF resulting from the proposed Type A test frequency change from 3 years to 15 years is 1.54E-7 per year. Therefore, based on this risk assessment, the proposed change to the Type A test frequency does not represent a risk significant change.

Regulatory Guide 1.174 also encourages the use of risk analysis techniques to ensure and demonstrate that a proposed change is consistent with the defense-in-depth philosophy. Consistency with the 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. For the proposed Type A test frequency change from 10 years to 15 years, the change in conditional containment failure probability was determined to be 0.104 percent. In comparison, the proposed Type A test frequency change from 3 years to 15 years results in only a 0.312 percent increase in conditional containment failure probability. Thus, these changes are small and the defense-in-depth philosophy is maintained.

Benefits of the Proposed Change

The next BSEP, Unit 1 10-year Type A test is scheduled to be performed during Unit 1 Refueling Outage 13 (i.e., designated as B114R1) in March 2002. The next BSEP, Unit 2 10-year Type A test is scheduled to be performed during Unit 2 Refueling Outage 15 (i.e., designated as B216R1) in March 2003. By allowing the one-time exception, CP&L will:

- Perform the next Unit 1 Type A test no later than March 21, 2006. Unit 1 Refueling Outage 15 (i.e., designated as B116R1) is currently scheduled to begin in February 2006.

- Perform the next Unit 2 Type A test no later than February 28, 2008. Unit 2 Refueling Outage 17 (i.e., designated as B218R1) is currently anticipated to begin in Spring 2007.

- Realize a substantial cost savings by not performing the Type A test for an additional 5 years. The estimated savings for the next outage of each BSEP unit include saving $216,000 associated with performance of the test, elimination of up to 36 hours of critical path outage time with associated replacement power costs savings of $1.1 million, and saving 1.5 rem of personnel radiation exposure.

CP&L understands that NEI is planning to seek NRC acceptance of a change to the NEI 94-01 guidance document with respect to Type A testing frequencies. It is anticipated that approval of this license amendment application will provide sufficient time for NEI to obtain NRC concurrence with the revised Type A testing frequency.

Need For Regulatory Exemption

Exceptions to the requirements of Regulatory Guide 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 for 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 license amendment application does not require an exemption to 10 CFR 50, Appendix J, Option B.

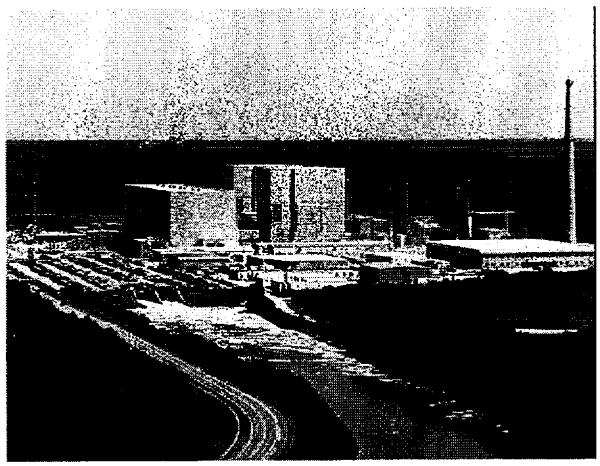
ENCLOSURE 2

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS REGARDING FREQUENCY OF
PERFORMANCE-BASED LEAKAGE RATE TESTING

RSC 01-24, "Evaluation of Risk Significance of ILRT Extension"

Carolina Power and Light RSC 01-24

Brunswick Steam Electric Plant Probabilistic Safety Assessment



Evaluation of Risk Significance of ILRT Extension

Revision 0

November 2001

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Preparer:	Ricky Summitt
RSC Reviewer: R. Young	Date: November 11, 2001
RSC Approver: R. Summitt	Date: November 11, 2001

Abstract (brief statement of purpose): Develop a risk informed analysis of the impact of extending the Brunswick Nuclear Plant ILRT

Documentation Retrieval Information:

Keywords: Special Analysis   Level 2 Analysis   Internal Events

☐ Amends / ☐ Superceeds / ☐ 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. Need to add calculation of base to 10 year CCFP.	Added to spreadsheet.
2. Class 6 and 2 dose fractions not listed in spreadsheet.	Removed from Table 3. Actually taken from other categories.

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 the Brunswick Steam Electric Plant (i.e., referred to hereafter as the Brunswick Nuclear Plant or BNP) for both Unit 1 and Unit 2.

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 testing. This report documents a risk-based evaluation of the proposed change of the ILRT test interval for BNP. The proposed change would impact testing associated with the current surveillance test for Type A leakage (procedure 0PT-20.5)[1]. No change to Type B or Type C testing is proposed at this time.

The evaluation for BNP is consistent with similar assessments performed for the Indian Point 3 (IP3) plant[2,3] and for the Crystal River 3 plant[4] that were approved by the NRC. This assessment utilizes the guidelines set forth in NEI 94-01[5], the methodology used in EPRI TR-104285[6] and the regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings in support of a licensee request to a plant's licensing basis, RG 1.174[7].

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 BNP.
- 15 years – Proposed extended test interval, similar to IP3 request.

The analysis utilizes the latest BNP probabilistic safety assessment (PSA) results. The PSA was initially developed for the BNP individual plant examination (IPE)[8] to estimate the baseline core damage and plant damage classes. Several updates to the BNP level 1 analysis have been incorporated since the IPE and a recent update to the Level 2 analysis has been completed. Therefore, this information represents the most recent analysis documented for BNP.

The release category and person-rem information is 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. The framework is described in Appendix A.

1.2 SUMMARY OF RESULTS/CONCLUSIONS

The specific results are summarized in Table 1 below. The Type A contribution to Large Early Release Fraction (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)	312.351	312.355	312.358
Type A Testing Risk (Person-Rem/yr)	4.755E-2	5.231E-2	5.468E-2
% Total Risk (Type A / Total)	0.01522%	0.01675%	0.01751%
Type A LERF (Class 3b) (per year)	1.03E-6	1.13E-6	1.18E-6
Changes due to extension from 10 years (current)			
Δ Risk from current (Person-rem/yr)			2.23E-3
% Increase from current (Δ Risk / Total Risk)			0.00096%
Δ LERF from current (per year)			5.14E-8
Δ CCFP from current			0.104%
Changes due to extension from 3 years (baseline)			
Δ Risk from baseline (Person-rem/yr)			6.69E-3
% Increase from baseline (Δ Risk / Total Risk)			0.00214%
Δ LERF from baseline (per year)			1.54E-7
Δ CCFP from baseline			0.312%

Based on the analysis and available data the following is stated:

- 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 2.23E-3 person-rem/yr.

- 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 5.14E-8/yr.

- The change in CCFP from the current once-per-10-year interval to once-per-15 years is 0.104%

- 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 0.00096%. Also, the change in Type A test frequency from the original three-per-ten-years to once-per-fifteen-years increases the risk only 0.00214% Therefore, the risk impact when compared to other severe accident risks is negligible.

- Reg. Guide 1.174 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 core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from an once-per-ten-years to an once per-fifteen-years is 5.14E-8. 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 is therefore considered non-risk significant. In addition, the change in LERF resulting from a change in the Type A ILRT test interval from a three-per-ten-years to an once per-fifteen-years is 1.54E-7/yr, is only slightly above the guidance.

- Reg.Guide 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 conditional containment failure probability was estimated to be 0.104% for the proposed change and 0.312% for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. These changes are small and demonstrate that the defense-in-depth philosophy is maintained.

2.0 DESIGN INPUTS

The BNP 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 BNP PSA (Reference 8) 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 BNP PSA and the level 2 update (Reference 9). The BNP plant damage classes are summarized in Table 2.

Table 2
BNP Plant Damage Classes

Damage Class	Representative Sequence	Frequency (/yr)
IA	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high.	1.89E-5
IBE	Accident sequences involving a station blackout and loss of coolant inventory makeup.	5.92E-6
IBL	Accident sequences involving a station blackout and loss of coolant inventory makeup.	6.48E-6
ID	Accident sequences involving a loss of coolant inventory makeup in which reactor pressure has been successfully reduced to 200 psi.	5.13E-6
IIA & IIL	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage (A) or with the RPV breached but no initial core damage (L); core damage induced post containment failure.	6.73E-7
IIIC	Accident sequences initiated or resulting in medium or large LOCAs for which the reactor is a low pressure and no effective injection is available.	4.87E-6
IVA	Accident sequences involving failure of adequate shutdown reactivity with the RPV initially intact; core damage induced post containment failure.	5.35E-6
IVL	Accident sequences involving a failure of adequate shutdown reactivity with the RPV initially breached (e.g. LOCA or SORV); core damage induced post containment failure.	1.75E-6
V	Unisolated LOCA outside containment.	3.49E-7
Total		4.95E-5

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 associate dose.

The IP3 submittal (Reference 2) 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 4) utilized plant-specific dose estimates based on the predicted level 2 analysis results.

The BNP PSA (Reference 9) contains the necessary information to convert the plant damage classes to release categories that contain the necessary information. Using this information the plant damage classes are mapped to the ten release categories. In addition, the fraction of intact containment cases is determined using the split fraction information contained in Reference 8.

Since the BNP PSA contains the necessary release fraction information, an approach similar to the CR3 submittal is utilized that better reflects the specific release conditions for BNP. The BNP PSA (Reference 8) 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 12. 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
Intact Containment[1]	6.21E-6	NA[2]	NA	NA	NA	NA	709[3]
BSEP IA-23	8.70E-6	7.09E-01	2.09E-01	3.81E-01	0.00E+00	1.22E-06	3.06E+7
BSEP IVA-14	7.48E-8	1.00E+00	1.45E-02	2.32E-02	0.00E+00	5.29E-07	3.39E+6
BSEP IA-20	2.77E-5	1.00E+00	1.11E-02	1.48E-02	0.00E+00	6.21E-09	2.80E+6
BSEP IVA-03	2.49E-7	1.00E+00	1.14E-03	1.07E-03	0.00E+00	6.08E-08	1.61E+6
BSEP IA-14 XFR5-5	3.30E-6	1.00E+00	2.73E-06	4.59E-05	0.00E+00	1.05E-09	1.50E+6
BSEP IA-03 XFR2-5	2.70E-6	1.00E+00	3.10E-06	4.07E-05	0.00E+00	0.00E+00	1.50E+6
BSEP V-1	3.49E-7	1.00E+00	9.50E-01	9.78E-01	0.00E+00	7.83E-04	9.79E+7

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 12).

Other inputs to this calculation include ILRT test data from NUREG-1493[10] and the EPRI report (Reference 6) and are referenced in the body of the calculation.

3.0 ASSUMPTIONS

1. The maximum containment leakage for EPRI Class 1 (Reference 6) sequences is $1L_a$ (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 and 4) sequences is $10L_a$ based on the previously approved methodology (References 2 and 3).
3. The maximum containment leakage for Class 3b sequences is $35L_a$ based on the previously approved methodology (References 2 and 3).
4. Class 3b is conservatively categorized LERF based on the previously approved methodology (References 2 and 3).
5. Containment leakage due to EPRI Classes 4 and 5 are considered negligible based on the previously approved methodology (References 2 and 3).
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 BNP 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 previously approved methodology used by Indian Point 3[2,3] and Crystal River 3[4]. This approach is similar to that presented in EPRI TR-104285[6] and NUREG-1493[10]. Namely, the analysis performed examined BNP PSA plant specific results in which the containment integrity remains intact or the containment is impaired.

4.1 CALCULATIONAL STEPS

The analysis is based on guidance provided in Reference 6 and uses risk metrics presented in Reference 7 to evaluate the impact of a proposed change on plant risk. References 2 and 4 utilize several measures in there 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 7 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 baseline plant damage classes and person-rem estimates
- Calculate baseline Type A leakage estimate to define the analysis baseline
- Modify Type A leakage estimate to address extension of the Type A test frequency
- 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 BNP PSA (References 8 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 design basis information[12].

The product of the person-rem for the key plant damage classes and the frequency of the key plant damage state estimates the annual person-rem estimate 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 6. 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 6 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 BNP 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 6 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 6)
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 defines eight (8) release classes as presented in Table 4.

Table 4
Containment Failure Classifications (from Reference 6)

Failure Classification	Description	Interpretation for Assigning BNP 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

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

Table 5
BNP PSA RC Grouping to EPRI Classes (Reference 6)

Class	Description	Release Category	Frequency	Person-Rem	Person-Rem/yr
1	No Containment Failure	Intact	6.21E-6	7.05E+2	4.376E-3
2	Large Containment Isolation Failures	Partial Contributions from IA, IB, ID	2.25E–8	3.06E+7	6.874E-1
3a	Small Isolation Failures (Liner breach)	None	0	NA	0
3b	Large Isolation Failures (Liner breach)	None	0	NA	0
4	Small isolation failures - failure to seal (type B)	None	0	NA	0
5	Small isolation failures - failure to seal (type C)	None	0	NA	0
6	Containment Isolation Failures (dependent failure, personnel errors)	Partial Contributions from IA, IB, ID, IIIC	1.25E-7	2.64E+7[1]	3.299E+0
7	Severe Accident Phenomena Induce Failure (Early and Late)	All other Release Categories	4.28E-5	6.41E+6[1]	2.741E+2
8	Containment Bypass	V	3.49E-7	9.79E+7	3.417E+1
	Total		4.95E-5		3.123E+2

1. 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 BNP 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.

Reference 3 states that a review of experience data finds that Type A testing identified only 4 leakage events of the 144 events identified. Thus about 3% (0.028) of containment leakage events are identified by the ILRT. The remaining events were identified by LLRT (Type B and C testing) and are not included in the analysis. This probability, however, is based on three tests over a 10-year period and not the one per ten-year frequency currently employed at BNP (Reference 1). The probability (0.028) must be adjusted to reflect this difference.

For this estimation, the question on containment isolation was modified consistent with the previously approved methodology (References 2 and 3), to include the probability of a liner breach (due to excessive leakage) at the time of core damage.

Class 3 is divided into two classes using this approach. Class 3a is defined as a small liner breach and Class 3b is defined as a large liner breach.

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 NUREG-1493 (Reference 10). One data set found in NUREG-1493 reviewed 144 ILRTs. The largest reported leak rate from those 144 tests was 21 times the allowable leakage rate (L_a). Since $21L_a$ does not constitute a large release, no large releases have occurred based on the 144 ILRTs reported in NUREG-1493.

To estimate the failure probability given that no failures have occurred, a conservative estimate is obtained from the 95th percentile of the χ^2 distribution. This is consistent with the Indian Point 3 (Reference 2) and Crystal River 3 (Reference 4) templates. In statistical theory, the χ^2 distribution can be used for statistical testing, goodness-of-fit tests (See Reference 11). The χ^2 distribution is really a family of distributions, which range in shape from that of the exponential to that of the normal distribution.

Each distribution is identified by the degrees of freedom, ν. For time-truncated tests (versus failure-truncated tests), an estimate of the probability of a large leak using the χ^2 distribution can be calculated using the following equation:

$$p(\alpha) = \frac{\chi^2(2F+2,\alpha)}{2N}$$

where: N is the number of events, F is the number of events (faults) of interest, α is the percentile distribution (typically assumed to be the 95%-tile). The result of 2F+2 defines the degree of freedom.

Given that there have been no large leaks (n = 0, therefore ν =2) in 144 events (N = 144) the value of $\chi^2(2, 0.05)$ is equal to 5.99. Solving for the 95th percentile estimate of the probability of a large leak yields 0.021 as presented below:

$$p_{Class3B} = \frac{\chi^2(2,0.05)}{2 \bullet 144} = \frac{5.99}{288} = 0.0208$$

Calculation of Class 3a Probability

The data presented in NUREG-1493 (Reference 10) 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.

Therefore, a best estimate for the probability of leakage is ~0.03 (4-of-144). However, the Class 3a probability is estimated using the conservative χ^2 distribution approach described previously. This is consistent with the approach taken in References 2, 3 and 4.

The χ^2 distribution is calculated by F = 4 (number of small leaks) and N = 144 (number of events) which yields a solution as shown below:

$$p_{Class3A} = \frac{\chi^2(10,0.05)}{2 \bullet 144} = \frac{18.307}{288} = 0.0636$$

Therefore, the 95th percentile estimate of the probability of a small leak (Class 3a) is calculated as 0.064.

The probability of liner failures must then be multiplied by an appropriate accident frequency to determine the Class 3a and Class 3b frequencies. The IP3 (Reference 2) and CR3 (Reference 4) submittals utilized the entire core damage frequency when developing the contributions for Classes 3a and 3b and then adjusted the Class 1 contribution.

This is somewhat conservative since it does provide the maximum possible contributions due to the extension of the ILRT testing interval. This approach is maintained for the BNP analysis, in order to be consistent with the approved methodology.

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

$FREQ_{class3b} = PROB_{class3b}$ x CDF = 0.0208 x 4.95E-5/yr = 1.03E-6/yr

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

$FREQ_{class3a} = PROB_{class3a}$ x CDF = 0.0636 x 4.95E-5/yr = 3.14E-6/yr

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 6 as Class 6.

$FREQ_{class6}$.=1.25E-7/yr

Class 1:

Although the frequency of this class is not directly impacted by Type A testing, 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:

$FREQ_{class1} = FREQ_{class1} – (FREQ_{class3a} + FREQ_{class3b})$

$FREQ_{class1}$ = 6.21E-6/yr – (1.03E-6/yr + 3.14E-6/yr) = 2.03E-6/yr

Class 2:

The frequency of Class 2 is the sum of those release categories identified in Table 6 as Class 2.

$FREQ_{class7}$ = 1.25E-7/yr

Class 7:

The frequency of Class 7 is the sum of those release categories identified in Table 6 as Class 7.

$FREQ_{class7}$ = 4.28E-5/yr

Class 8:

The frequency of Class 8 is the sum of those release categories identified in Table 6 as Class 8.

$FREQ_{class8} = 3.49E\text{-}7/yr$

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 12). The Class 3a and 3b doses are represented as $10L_a$ and $35L_a$ 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	2.03E-6		7.05E+2[2]	1.435E-3
2	Large Containment Isolation Failures	2.25E–8	3.06E+7		6.874E-1
3a	Small Isolation Failures (Liner breach)	3.14E-6		7.05E+3[3]	2.216E-2
3b	Large Isolation Failures (Liner breach)	1.03E-6		2.47E+4[4]	2.539E-2
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)	1.25E-7	2.64E+7		3.299E+0
7	Severe Accident Phenomena Induce Failure (Early and Late)	4.28E-5	6.41E+6		2.741E+2
8	Containment Bypass	3.49E-7	9.79E+7		3.417E+1
	Total	4.95E-5			3.12351E+2

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 = 2.216E-2 person-rem/year

$Class3b_{BASE}$ = Class 3b person-rem/year = 2.539E-2 person-rem/year

$Total_{BASE}$ = total person-rem year for baseline interval = 312.351 person-rem/year (Table 6)

$\%Risk_{BASE}$ = [(2.539E-2 + 2.216E-2) / 312.351] x 100 = **0.015%**

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 (Reference 5) 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 NUREG-1493 (Reference 10), 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.

Since ILRTs only detect about 3% of leaks (4/144) that are not detected by other local tests, 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 (60 months) to the baseline average time for a failure to detect of 18 months (i.e., 0.03 x 60/18 = 0.10). References 2 and 4 indicate this is a 10% increase in the likelihood of a Type A leak.

Risk Impact due to 10-year Test Interval

Based on the previously approved methodology (References 2 and 3), the increased probability of not detecting excessive leakage due to Type A tests directly impacts the frequency of the Class 3 sequences. Consistent with References 2 and 4, the risk contribution is determined by multiplying the Class 3 accident frequency by the increase in the probability of leakage (1.1 x Class 3 baseline). 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]	1.62E-6	7.05E+2	1.140E-3
2	Large Containment Isolation Failures	2.25E–8	3.06E+7	6.874E-1
3a	Small Isolation Failures (Liner breach)	3.46E-6	7.05E+3	2.438E-2
3b	Large Isolation Failures (Liner breach)	1.13E-6	2.47E+4	2.793E-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)	1.25E-7	2.64E+7	3.299E+0
7	Severe Accident Phenomena Induce Failure (Early and Late)	4.28E-5	6.41E+6	2.741E+2
8	Containment Bypass	3.49E-7	9.79E+7	3.417E+1
	Total	4.95E-5		3.12355E+2

1. The IPE 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:

$\%Risk10 = [(Class3a_{10} + Class3b_{10}) / Total_{10}] \times 100$

Where:

$Class3a_{10}$ = Class 3a person-rem/year = 2.438E-2 person-rem/year

$Class3b_{10}$ = Class 3b person-rem/year = 2.793E-2 person-rem/year

$Total_{10}$ = total person-rem year for current 10-year interval = 312.355 person-rem/year (Table 7)

$\%Risk_{10}$ = [(2.438E-2 + 2.793E-2) / 312.355] x 100 = **0.01675%**

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

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

Where:

$Total_{BASE}$ = total person-rem/year for baseline interval = 312.351 person-rem/year (Table 6)

$Total_{10}$ = total person-rem/year for 10-year interval = 312.355 person-rem/year (Table 7)

$\Delta\%Risk_{10}$ = [(312.355 – 312.351) / 312.351] x 100.0 = **0.001428%**

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 (0.03 x 90/18) of the exposure times. 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. References 2 and 4 indicate that the increase is a 50% increase from that for the 10-year interval or a 15% increase from the baseline. Different values are provided for the probability of leakage. In addition, the containment leakage used for the 10-year test interval for Class 3 is used in the 15-year interval evaluation (1.15 x Class 3 baseline). The results for this calculation are presented in Table 8.

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]	1.41E-6	7.05E+2	9.933E-4
2	Large Containment Isolation Failures	2.25E–8	3.06E+7	6.874E-1
3a	Small Isolation Failures (Liner breach)	3.62E-6	7.05E+3	2.549E-2
3b	Large Isolation Failures (Liner breach)	1.18E-6	2.47E+4	2.920E-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)	1.25E-7	2.64E+7	3.299E+0
7	Severe Accident Phenomena Induce Failure (Early and Late)	4.28E-5	6.41E+6	2.741E+2
8	Containment Bypass	3.49E-7	9.79E+7	3.417E+1
	Total	4.95E-5		3.12358E+2

1. The IPE 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 = 2.549E-2 person-rem/year

$Class3b_{15}$ = Class 3b person-rem/year = 2.920E-2 person-rem/year

$Total_{15}$ = total person-rem year for 15-year interval = 312.358 person-rem/year (Table 8)

$\%Risk_{15}$ = [(2.549E-2 + 2.920E-2) / 312.358] x 100 = **0.01751%**

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 = 312.351 person-rem/year (Table 6)

$Total_{15}$ = total person-rem/year for 15-year interval = 312.358 person-rem/year (Table 8)

$\Delta\%Risk_{15} = [(312.358 - 312.351) / 312.351] \times 100.0 =$ **0.00214%**

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

Based on the previously approved methodology (Reference 2 and 4), the percent increase in risk (in terms of person-rem/yr) of these associated specific sequences is computed as follows.

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

Where:

$PER\text{-}REM_{10}$ = person-rem/year of ten years interval (see Table 7, classes 1, 3a and 3b)

= 5.345E-2 person-rem/yr

$PER\text{-}REM_{15}$ = person-rem/year of fifteen years interval (Table 8, classes 1, 3a and 3b)

= 5.568E-2 person-rem/yr

$\%Risk_{10\text{-}15} = [(5.568E\text{-}2 - 5.345E\text{-}2) / 5.345E\text{-}2] \times 100 =$ **4.17%**

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

= 312.355 person-rem/year (Table 7)

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

= 312.358 person-rem/year (Table 8)

$\% Total_{10\text{-}15} = [(312.358 - 312.355) / 312.355] \times 100 =$ **0.00096%**

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 and 4, the Class 3a dose is assumed to be 10 times the allowable intact containment leakage, L_a (or 7,050 person-rem) and the Class 3b dose is assumed to be 35 times L_a (or 24,700 person-rem). The dose equivalent for allowable leakage (L_a) is developed in Reference 12. This compares to a historical observed average of twice L_a. Therefore, the estimate is somewhat conservative.

Based on the previously approved methodology (References 2 and 4), 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 BNP 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 BNP, and the change in LERF can be determined by the differences. References 2 and 4 identify 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)	1.03E-6/yr	1.13E-6/yr	1.18E-6/yr
ΔLERF (10 year baseline)			5.14E-8
ΔLERF (3 year baseline)			1.54E-7

Reg. Guide 1.174 (Reference 7) 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 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.

Since guidance in Reg. Guide 1.174 defines very small changes in LERF as below 1.0E-7/yr, increasing the ILRT interval to 15 years (**5.14E-8/yr**) is non-risk significant. It should be noted that if the risk increase is measured from the original 3-in-10-year interval, the increase in LERF is **1.54E-7/yr**, which is only slightly above the 1.0E-07/yr screening criterion 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:

$$CCFP = 1 - \left[\frac{f(ncf)}{CDF}\right]$$

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:

$$\Delta CCFP_{10-15} = CCFP_{15} - CCFP_{10}$$

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)	5.18E-6	5.08E-6	5.02E-6
f(ncf)/CDF	0.105	0.103	0.102
CCFP	0.895	0.897	0.898
ΔCCFP (3 year baseline)		0.208%	0.312%
ΔCCFP (10 year baseline)			0.104%

5.0 REFERENCES

1. Integrated Primary Containment Leak Rate Test (IPCLRT), Rev. 36, Carolina Power and Light Company, 0PT-20.5.

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

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10. Performance-Based Containment Leak-Test Program, USNRC, NUREG-1493, July 1995.

11. PRA Procedures Guide - A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants, American Nuclear Society and the National Institute of Electrical and Electronic Engineers, NUREG/CR-2300, January 1983.

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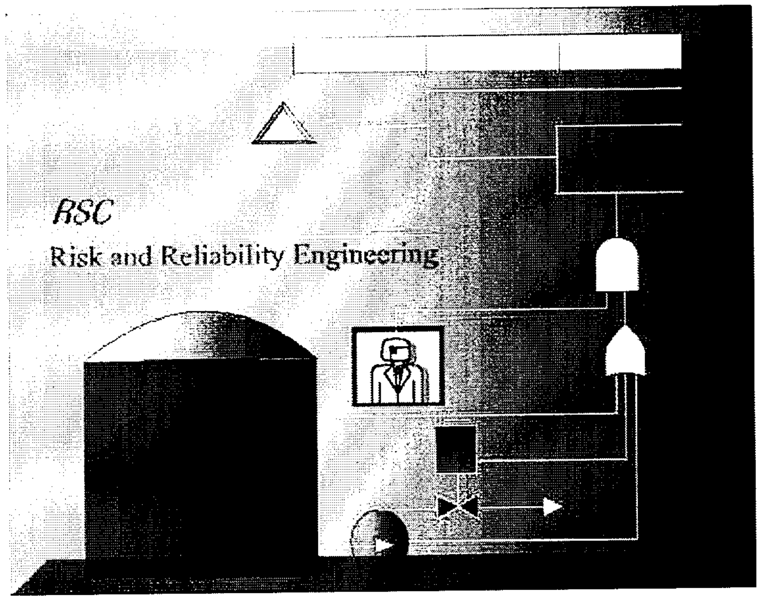
Appendix A

Surrogate Person-Rem Methodology
(RSC 01-44)

Ricky Summitt Consulting RSC 01-44



Surrogate Level 3 Evaluation Methodology

Revision 0

August 2001

Principal Analyst

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FORM NO.: RSC-RPT-STD99-04, Rev. 4

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Title: Surrogate Level 3 Evaluation Methodology

Revision: Revision 0

Author: Ricky Summitt

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Date of Record for all Models and Analysis	August 6, 2001
Software and Version Used: (bold all that apply)	Word (doc) Version 95, Version 97, **Version 2000** Excel (xls): Version 95, Version 97, **Version 2000** Access (mdb): Version 97, Version 2000 Designer (ds4/f): Version 7 CAFTA: Version 3.2b ETA Version 3.2b MAAP BWR Version 3.0B R9, R11, Version 4.0 MAAP PWR Version 18, 19, 20, Version 4.0 RSC Software: PRAMS, SIP, TIFA, BAYESUPDATE

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

RSC RSC STD RPT R3

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 to 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 A.1 through A.4 graphically presents the results for four PSAs as examples of this effort.

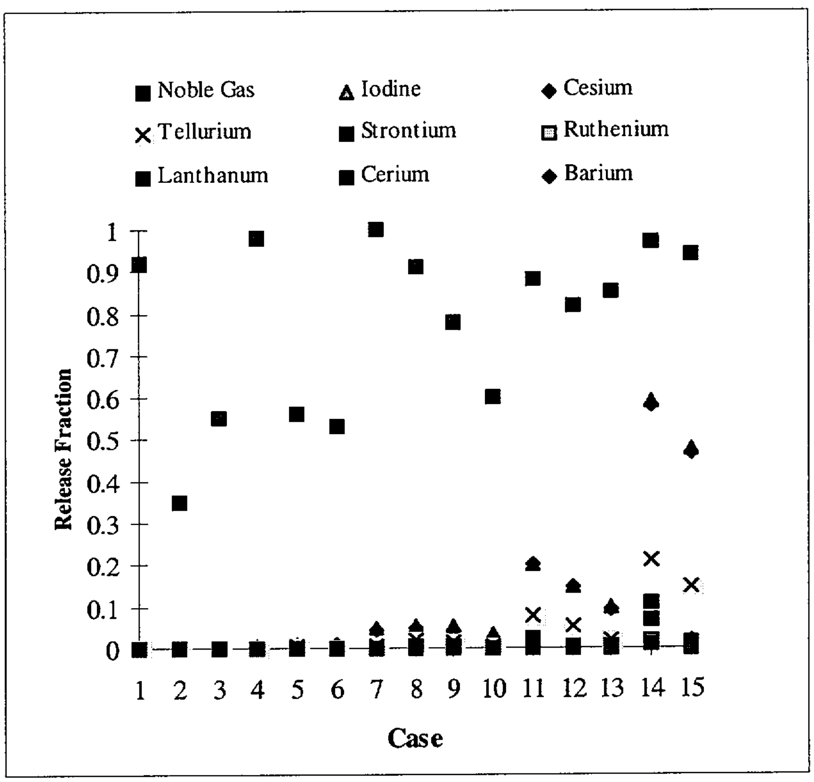

Figure 1 Sequoyah Release Fraction Cases (Reference 2)

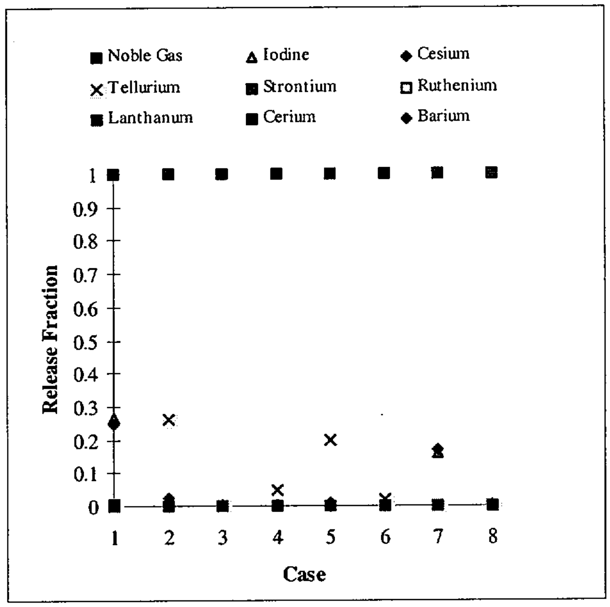

Figure 2 Unpublished PWR Release Fraction Cases (Reference 3)

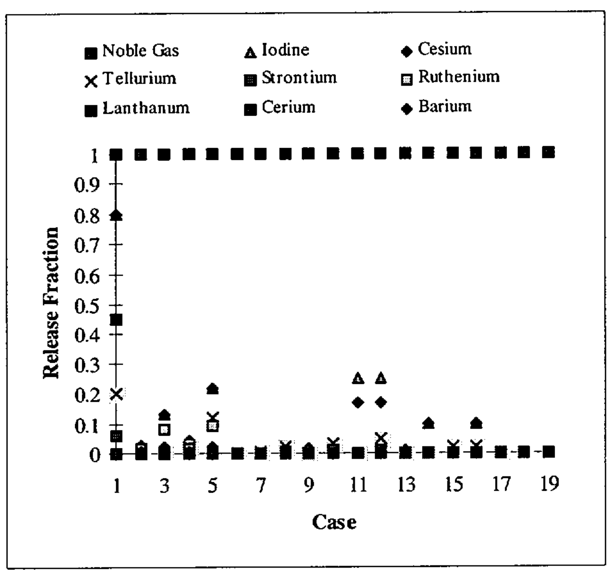

Figure 3 Oconee IPE Release Fraction Cases (Reference 4)

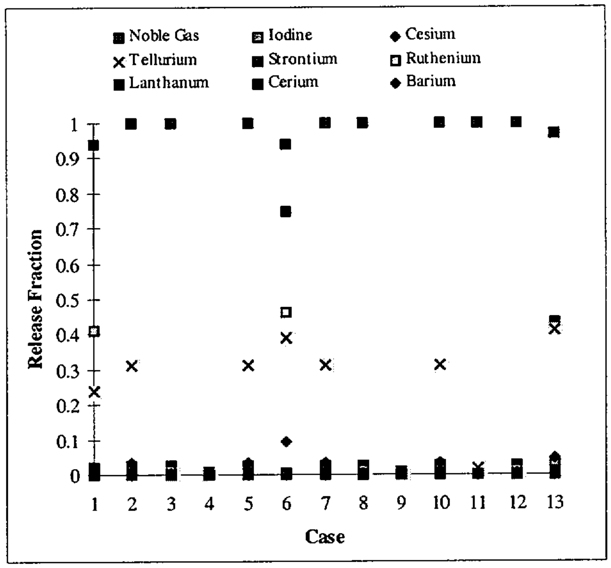

Figure 4 Seabrook Release Fraction Cases (Reference 5)

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, iodine, cesium, tellurium, and strontium, 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 A.5 shows how the dose essentially maps the release fraction.

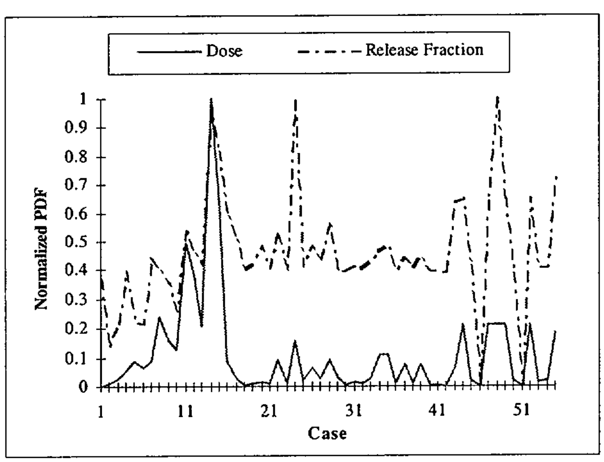

Figure 5 Relationship between Release Fraction and Dose

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, essentially linear, is somewhat of a surprise. Given this relationship, a set of 56 linear equations was developed. For each case, the equation took the form:

$$D_i = AX1_i + BX2_i + CX3_i + DX4_i + EX5_i$$

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 A.6 presents the correlation for the 56 cases obtained for the final solution.

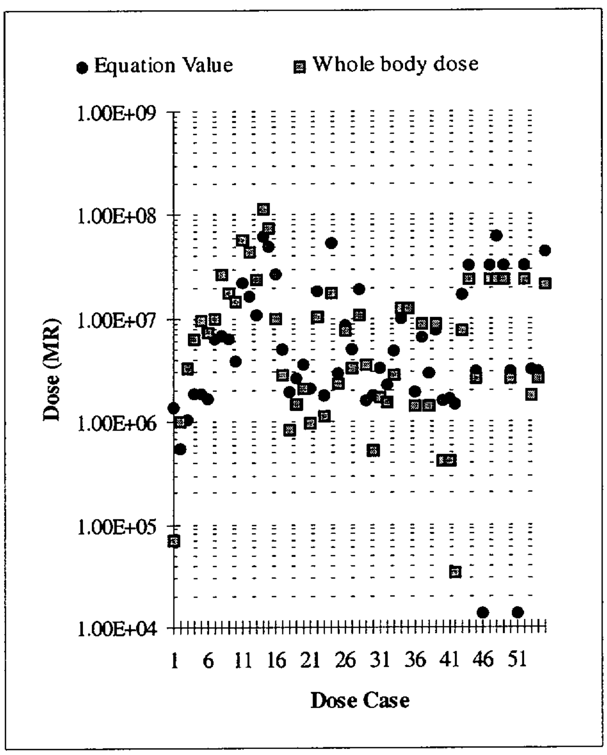

Figure 6 Comparison of Equation Results and Reported Dose Values

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	1500000
B	Iodine	50000000
C	Cesium	50000000
D	Tellurium	5000000
E	Strontium	5000000

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	CsI
X3	CsOH
X4	TeO_2 and Te
X5	SrO

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 A.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.

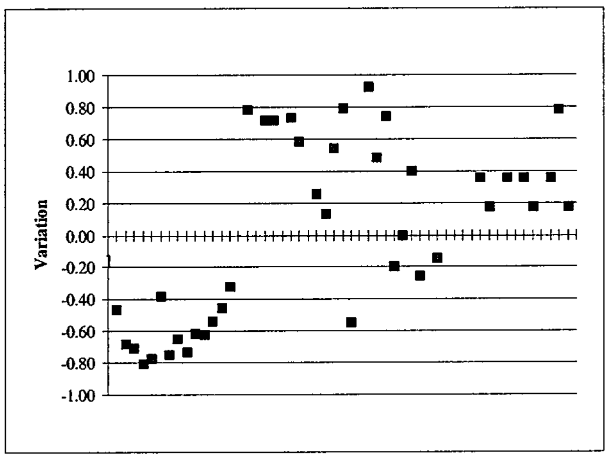

Figure 7 Variation of Equation to Reported Dose

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 A.6 and A.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 A.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.

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ENCLOSURE 3

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS REGARDING
FREQUENCY OF PERFORMANCE-BASED LEAKAGE RATE TESTING

10 CFR 50.92 Evaluation

Carolina Power & Light (CP&L) Company is requesting a revision to Technical Specification 5.5.12 for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2, to incorporate a one-time exception to the 10-year frequency of the performance-based leakage rate testing program for Type A tests specified by Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and endorsed by 10 CFR Part 50, Appendix J, Option B. The new exception will allow a Type A test to be performed within 15 years, one month from the last Type A test for Unit 1 and 15 years for Unit 2. The last BSEP, Unit 1 Type A test was performed on February 15, 1991; the last BSEP, Unit 2 Type A test was performed on February 28, 1993. The new exception will require performance of the next Type A test no later than March 21, 2006, for BSEP, Unit 1, and no later than February 28, 2008, for BSEP, Unit 2.

In support of the No Significant Hazards determination, an evaluation of each of the three (3) standards set forth in 10 CFR 50.92 is provided below.

1. The proposed license amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change to Technical Specification 5.5.12 provides a one-time extension to the testing frequency for containment integrated leakage rate (i.e., Type A) testing. The existing 10-year test interval is based on past test performance. The proposed Technical Specification change will extend the Type A testing frequency to 15 years, one month from the last Type A test for Unit 1 and to 15 years for Unit 2. The proposed Technical Specification 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 primary containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the primary containment does not involve the prevention or identification of any precursors of an accident. Therefore, the proposed Technical Specification 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 containment leakage testing will continue to be performed at the frequency currently required by the BSEP Technical Specifications. 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 no failures were due to 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 BSEP, Unit 1 and 2 test history and risk-based evaluation of the proposed extension to the Type A test frequency supports this conclusion. The design and construction requirements of the primary containment, 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 primary containment 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.

2. The proposed license amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change to the Technical Specification 5.5.12 involves a one-time extension to the testing interval for Type A containment leakage rate testing. The primary containment and the testing requirements invoked to periodically demonstrate the integrity of the primary containment exist to ensure the ability to mitigate the consequences of an accident. The primary containment and its associated testing requirements do not involve the prevention or identification of any precursors of an accident. The proposed change to the Type A leakage rate testing frequency does not involve any physical changes being made to the facility. In addition, the proposed changes to the Type A leakage rate testing frequency does not change the operation of the plants such that a new failure mode involving the possibility of a new or different kind of accident from any accident previously evaluated is created. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed license amendments do not involve a significant reduction in a margin of safety.

   The proposed extension to the Type A testing frequency will not significantly reduce the margin of safety. 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 since most potential leakage paths are detected by Type B and C testing. The proposed change involves only an extension of the frequency for Type A containment leakage testing; the overall primary containment leakage rate limit specified by Technical Specifications is being maintained. Type B and C containment leakage testing will continue to be performed at the frequency currently required by the BSEP Technical Specifications. The regular containment inspections being performed in accordance with the 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, the on-line containment monitoring capability that is inherit to boiling water reactor using an inert containment atmosphere allows for the detection of gross containment leakage that may develop during power operation. The combination of these factors ensures that the margin of safety is maintained. Therefore, the proposed license amendments do not involve a significant reduction in a margin of safety.

ENCLOSURE 4

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS REGARDING
FREQUENCY OF PERFORMANCE-BASED LEAKAGE RATE TESTING

Environmental Considerations

Carolina Power & Light (CP&L) Company is requesting a revision to Technical Specification 5.5.12 for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2, to incorporate a one-time exception to the 10-year frequency of the performance-based leakage rate testing program for Type A tests as specified by Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and endorsed by 10 CFR Part 50, Appendix J, Option B. The new exception will allow performance of a Type A within 15 years, one month from the last ILRT for Unit 1 and within 15 years from the last ILRT for Unit 2. The last BSEP, Unit 1 ILRT was performed on February 15, 1991; the last BSEP, Unit 2 ILRT was performed on February 28, 1993. The new exception will require performance of the next Type A test no later than March 21, 2006, for BSEP, Unit 1, and no later than February 28, 2008, for BSEP, Unit 2.

CP&L has concluded that the proposed changes to the Technical Specifications for BSEP, Units 1 and 2 are eligible for categorical exclusion from performing an environmental assessment. In support of this determination, an evaluation of each of the three (3) criteria set forth in 10 CFR 51.22(c)(9) is provided below.

1. The proposed license amendments do not involve a significant hazards consideration, as shown in Enclosure 3.

2. The proposed license amendments do not result in a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite.

   The proposed license amendments have no impact on the environment. The proposed license amendments do not involve installation of any new equipment or modification of any existing equipment that may affect the types or amounts of effluents that may be released offsite. Therefore, CP&L has concluded that there will not be a significant increase in the types or amounts of any effluent that may be released offsite and, as such, the changes do not involve irreversible environmental consequences beyond those already associated with normal operation.

3. The proposed license amendments do not result in a significant increase in individual or cumulative occupational radiation exposure.

   The proposed changes do not involve plant physical changes, or introduce any new mode of plant operation. Therefore, there is no significant increase in individual or cumulative occupational radiation exposure.

ENCLOSURE 5

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS REGARDING
FREQUENCY OF PERFORMANCE-BASED LEAKAGE RATE TESTING

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ENCLOSURE 6

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS REGARDING
FREQUENCY OF PERFORMANCE-BASED LEAKAGE RATE TESTING

Typed Technical Specification Pages – Unit 1

5.5 Programs and Manuals

Primary Containment Leakage Rate Testing Program (continued)

a. Compensation of instrument accuracies applied to the primary containment leakage total in accordance with ANSI/ANS 56.8-1987 instead of ANSI/ANS 56.8-1994;

b. Following air lock door seal replacement, performance of door seal leakage rate testing with the gap between the door seals pressurized to 10 psig instead of air lock testing at P_a as specified in Nuclear Energy Institute Guideline 94-01, Revision 0;

c. Reduced duration Type A tests may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1.

d. Performance of Type C leak rate testing of the hydrogen and oxygen monitor isolation valves is not required; and

e. Performance of Type C leak rate testing of the main steam isolation valves at a pressure less than P_a instead of leak rate testing at P_a as specified in ANSI/ANS 56.8-1994.

f. NEI 94-01 - 1995, Section 9.2.3: The first Type A test performed after the February 15, 1991, Type A test shall be performed no later than March 21, 2006.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 49 psig.

The maximum allowable primary containment leakage rate, L_a, shall be 0.5% of primary containment air weight per day at P_a.

Leakage rate acceptance criteria are:

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

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

(continued)

5.5 Programs and Manuals

Primary Containment Leakage Rate Testing Program (continued)

2) For each air lock door, leakage rate is ≤ 5 scfh when the gap between the door seals is pressurized to ≥ 10 psig.

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

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrem and their associated collective deep dose equivalent (reported in person-rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), electronic dosimeter or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report, covering the previous calendar year, shall be submitted by April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of Table 3 in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

------------------------------NOTE------------------------------
A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station.

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1;

2. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.2;

3. The Allowable Value for Function 2.b, APRM Flow Biased Simulated Thermal Power—High, for Specification 3.3.1.1; and

4. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).

2. NEDO-32339-A, "Reactor Stability Long Term Solution: Enhanced Option I-A," July 1995.

3. NEDC-32339-P Supplement 1, "Reactor Stability Long Term Solution: Enhanced Option I-A ODYSY Computer Code," March 1994 (Approved in NRC Safety Evaluation dated January 4, 1996).

4. NEDO-32339 Supplement 3, "Reactor Stability Long Term Solution: Enhanced Option I-A Flow Mapping Methodology," August 1995 (Approved in NRC Safety Evaluation dated May 28, 1996).

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates not exceeding 1.0 rem/hour (at 30 centimeters from the radiation sources or from any surface penetrated by the radiation)

a. Each accessible entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area ("radiation monitoring and indicating device"); or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached ("alarming dosimeter"), with an appropriate alarm setpoint; or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area; or

(continued)

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates not exceeding 1.0 rem/hour (at 30 centimeters from the radiation sources or from any surface penetrated by the radiation) (continued)

4. A self-reading dosimeter and,

(a) Be under the surveillance, as specified in the RWP or equivalent, of an individual at the work site, qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device who is responsible for controlling personnel radiation exposure within the area, or

(b) Be under the surveillance, as specified in the RWP or equivalent, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area.

e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been established and entry personnel are knowledgeable of them.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation)

a. Each accessible entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked door, gate, or guard that prevents unauthorized entry, and in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift superintendent or the radiation control supervisor or designated representative; and

2. Doors and gates shall remain locked or guarded except during periods of personnel or equipment entry or exit.

(continued)

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation) (continued)

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual (whether alone or in a group) entering such an area shall possess:

1. An alarming dosimeter with an appropriate alarm setpoint; or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area; or

3. A direct-reading dosimeter and,

(a) Be under the surveillance, as specified in the RWP or equivalent, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device who is responsible for controlling personnel exposure within the area, or

(b) Be under the surveillance, as specified in the RWP or equivalent, by means of closed circuit television, of personnel qualified in radiation

(continued)

5.7 High Radiation Area

5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation)</u> (continued)

protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area; or

4. A radiation monitoring and indicating device in those cases where the options of Specifications 5.7.2.d.2 and 5.7.2.d.3, above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle.

e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been established and entry personnel are knowledgeable of them.

f. Such individual areas that are within a larger area that is controlled as a high radiation area, where no enclosure exists for purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, but shall be barricaded and conspicuously posted as a high radiation area, and a conspicuous, clearly visible flashing light shall be activated at the area as a warning device.

ENCLOSURE 7

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS REGARDING
FREQUENCY OF PERFORMANCE-BASED LEAKAGE RATE TESTING

Typed Technical Specification Pages – Unit 2

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

a. Compensation of instrument accuracies applied to the primary containment leakage total in accordance with ANSI/ANS 56.8-1987 instead of ANSI/ANS 56.8-1994;

b. Following air lock door seal replacement, performance of door seal leakage rate testing with the gap between the door seals pressurized to 10 psig instead of air lock testing at P_a as specified in Nuclear Energy Institute Guideline 94-01, Revision 0;

c. Reduced duration Type A tests may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1.

d. Performance of Type C leak rate testing of the hydrogen and oxygen monitor isolation valves is not required; and

e. Performance of Type C leak rate testing of the main steam isolation valves at a pressure less than P_a instead of leak rate testing at P_a as specified in ANSI/ANS 56.8-1994.

f. NEI 94-01 - 1995, Section 9.2.3: The first Type A test performed after the February 28, 1993, Type A test shall be performed no later than February 28, 2008.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 49 psig.

The maximum allowable primary containment leakage rate, L_a, shall be 0.5% of primary containment air weight per day at P_a.

Leakage rate acceptance criteria are:

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

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

(continued)

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

2) For each air lock door, leakage rate is ≤ 5 scfh when the gap between the door seals is pressurized to ≥ 10 psig.

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

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrem and their associated collective deep dose equivalent (reported in person-rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), electronic dosimeter or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report, covering the previous calendar year, shall be submitted by April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of Table 3 in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

------------------------------NOTE------------------------------
A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station.

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1;

2. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.2;

3. The Allowable Value for Function 2.b, APRM Flow Biased Simulated Thermal Power—High, for Specification 3.3.1.1; and

4. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).

2. NEDO-32339-A, "Reactor Stability Long Term Solution: Enhanced Option I-A," July 1995.

3. NEDC-32339-P Supplement 1, "Reactor Stability Long Term Solution: Enhanced Option I-A ODYSY Computer Code," March 1994 (Approved in NRC Safety Evaluation dated January 4, 1996).

4. NEDO-32339 Supplement 3, "Reactor Stability Long Term Solution: Enhanced Option I-A Flow Mapping Methodology," August 1995 (Approved in NRC Safety Evaluation dated May 28, 1996).

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates not exceeding 1.0 rem/hour (at 30 centimeters from the radiation sources or from any surface penetrated by the radiation)

a. Each accessible entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area ("radiation monitoring and indicating device"); or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached ("alarming dosimeter"), with an appropriate alarm setpoint; or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area; or

(continued)

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates not exceeding 1.0 rem/hour (at 30 centimeters from the radiation sources or from any surface penetrated by the radiation) (continued)

4. A self-reading dosimeter and,

   (a) Be under the surveillance, as specified in the RWP or equivalent, of an individual at the work site, qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device who is responsible for controlling personnel radiation exposure within the area, or

   (b) Be under the surveillance, as specified in the RWP or equivalent, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area.

e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been established and entry personnel are knowledgeable of them.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation)

a. Each accessible entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked door, gate, or guard that prevents unauthorized entry, and in addition:

   1. All such door and gate keys shall be maintained under the administrative control of the shift superintendent or the radiation control supervisor or designated representative; and

   2. Doors and gates shall remain locked or guarded except during periods of personnel or equipment entry or exit.

(continued)

5.7 High Radiation Area

5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation)</u> (continued)

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual (whether alone or in a group) entering such an area shall possess:

1. An alarming dosimeter with an appropriate alarm setpoint; or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area; or

3. A direct-reading dosimeter and,

(a) Be under the surveillance, as specified in the RWP or equivalent, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device who is responsible for controlling personnel exposure within the area, or

(b) Be under the surveillance, as specified in the RWP or equivalent, by means of closed circuit television, of personnel qualified in radiation

(continued)

5.7 High Radiation Area

5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation)</u> (continued)

protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area; or

4. A radiation monitoring and indicating device in those cases where the options of Specifications 5.7.2.d.2 and 5.7.2.d.3, above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle.

e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been established and entry personnel are knowledgeable of them.

f. Such individual areas that are within a larger area that is controlled as a high radiation area, where no enclosure exists for purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, but shall be barricaded and conspicuously posted as a high radiation area, and a conspicuous, clearly visible flashing light shall be activated at the area as a warning device.

ENCLOSURE 8

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS REGARDING
FREQUENCY OF PERFORMANCE-BASED LEAKAGE RATE TESTING

Marked-up Technical Specification Pages – Unit 1

5.5 Programs and Manuals

Primary Containment Leakage Rate Testing Program (continued)

a. Compensation of instrument accuracies applied to the primary containment leakage total in accordance with ANSI/ANS 56.8-1987 instead of ANSI/ANS 56.8-1994;

b. Following air lock door seal replacement, performance of door seal leakage rate testing with the gap between the door seals pressurized to 10 psig instead of air lock testing at P_a as specified in Nuclear Energy Institute Guideline 94-01, Revision 0;

c. Reduced duration Type A tests may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1.

d. Performance of Type C leak rate testing of the hydrogen and oxygen monitor isolation valves is not required; and

e. Performance of Type C leak rate testing of the main steam isolation valves at a pressure less than P_a instead of leak rate testing at P_a as specified in ANSI/ANS 56.8-1994.

INSERT

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 49 psig.

The maximum allowable primary containment leakage rate, L_a, shall be 0.5% of primary containment air weight per day at P_a.

Leakage rate acceptance criteria are:

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

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2) For each air lock door, leakage rate is ≤ 5 scfh when the gap between the door seals is pressurized to ≥ 10 psig.

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

INSERT for TS 5.5.12

f. NEI 94-01 - 1995, Section 9.2.3: The first Type A test performed after the February 15, 1991, Type A test shall be performed no later than March 21, 2006.

ENCLOSURE 9

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS REGARDING
FREQUENCY OF PERFORMANCE-BASED LEAKAGE RATE TESTING

Marked-up Technical Specification Pages – Unit 2

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

a. Compensation of instrument accuracies applied to the primary containment leakage total in accordance with ANSI/ANS 56.8-1987 instead of ANSI/ANS 56.8-1994;

b. Following air lock door seal replacement, performance of door seal leakage rate testing with the gap between the door seals pressurized to 10 psig instead of air lock testing at P_a as specified in Nuclear Energy Institute Guideline 94-01, Revision 0;

c. Reduced duration Type A tests may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1.

d. Performance of Type C leak rate testing of the hydrogen and oxygen monitor isolation valves is not required; and

e. Performance of Type C leak rate testing of the main steam isolation valves at a pressure less than P_a instead of leak rate testing at P_a as specified in ANSI/ANS 56.8-1994.

INSERT

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 49 psig.

The maximum allowable primary containment leakage rate, L_a, shall be 0.5% of primary containment air weight per day at P_a.

Leakage rate acceptance criteria are:

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

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2) For each air lock door, leakage rate is ≤ 5 scfh when the gap between the door seals is pressurized to ≥ 10 psig.

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

INSERT for TS 5.5.12

f. NEI 94-01 - 1995, Section 9.2.3: The first Type A test performed after the February 28, 1993, Type A test shall be performed no later than February 28, 2008.