10 CFR 50.90

Progress Energy

Serial: RNP-RA/03-0101

AUG 2 0 2003

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

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

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

Ladies and Gentlemen:

In a letter dated June 11, 2003, Progress Energy Carolinas, Inc., submitted a request for an amendment to the Technical Specifications (TS) for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. The proposed amendment would modify the TS to allow a one-time extension of the containment Type A leak rate test interval to 15 years.

The purpose of this letter is to supersede the technical justification, No Significant Hazards Consideration Determination, and Environmental Impact Consideration provided in the June 11, 2003 letter, which were based partly on the risk assessment methodology described in WCAP-15691, "Joint Applications Report for Containment Integrated Leak Rate Test Interval Extension," Revisions 2 and 3. The revised evaluations provided with this supplement eliminate reliance on and reference to the WCAP-15691 report methodology, and replace the previous evaluations in their entirety. The revised risk evaluation is based on a method referred to as the Crystal River Unit 3 method, which was recently used as a basis for approval of a similar amendment for St. Lucie Units 1 and 2 (TAC Nos. MB6138 and MB6139).

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

Attachment II provides a description of the current condition, a description of the proposed change, and the revised technical justification of the proposed change, revised No Significant Hazards Consideration Determination, and revised Environmental Impact Consideration.

A markup of the proposed TS page and a retyped version of the proposed TS page were provided as Attachments III and IV of the June 11, 2003 letter. The revised justifications do not impact the proposed Technical Specifications wording. Therefore, Attachments III and IV of the June 11, 2003 letter remain valid.

Progress Energy Carolinas, Inc.
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville, SC 29550

A001

Attachment III provides a copy of calculation RNP-F/PSA-0020, "Evaluation of Risk Significance of ILRT Extension."

In accordance with 10 CFR 50.91(b), Progress Energy Carolinas, Inc., is providing the State of South Carolina with a copy of this supplement.

Progress Energy Carolinas, Inc., requests approval of this license amendment request by January 16, 2004, with the amendment being implemented within 30 days of approval. The approval date was selected to allow for effective planning for the refueling outage that is currently scheduled for April 2004.

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

Sincerely,

Jan Lucas

Jan F. Lucas
Manager - Support Services - Nuclear

Attachments:

I. Affirmation
II. Request for Technical Specifications Change Regarding One-Time Extension of Containment Type A Test Interval
III. Calculation RNP-F/PSA-0020, "Evaluation of Risk Significance of ILRT Extension"

RAC/rac

c: Mr. T. P. O'Kelley, Director, Bureau of Radiological Health (SC)
Mr. L. A. Reyes, NRC, Region II
Mr. C. P. Patel, NRC, NRR
NRC Resident Inspector, HBRSEP
Attorney General (SC)

AFFIRMATION

The information contained in letter RNP-RA/03-0101 is true and correct to the best of my information, knowledge, and belief; and the sources of my information are officers, employees, contractors, and agents of Progress Energy Carolinas, Inc. I declare under penalty of perjury that the foregoing is true and correct.

Executed On: AUG 2 0 2003

C L Burton

C. L. Burton
Director – Site Operations
HBRSEP, Unit No. 2

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

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

Description of Current Condition

Containment structure testing is intended to assure the leak-tight integrity of the containment structure under all design basis conditions. Conservative design and construction have led to very few containment Type A tests exceeding the leak test acceptance criteria. The NRC has extended the allowable Type A test period from three times in 10 years to once in 10 years based on past successful tests. NUREG-1493, "Performance-Based Containment Leak-Test Program," which supported that change, also states that test periods of up to 20 years would lead to an imperceptible increase in risk.

The current 10 year interval for performance of the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, Type A test ended on April 9, 2002. By letter dated September 16, 2002, the NRC issued Amendment No. 193 to the HBRSEP, Unit No. 2, Operating License (OL) and Technical Specifications (TS) that authorized an extension of the Type A test interval of 12.1 years based on a plant-specific risk assessment. As a result, a one-cycle deferral was implemented, such that the next Type A test is currently scheduled to be performed during Refueling Outage (RO)-22 in April 2004. This one-cycle deferral is currently reflected by TS 5.5.16, which requires that the next Type A test be performed "no later than May 9, 2004."

In order to take full advantage of the one-time extension of the Type A test from once in 10 years to once in 15 years, HBRSEP, Unit No. 2, has re-evaluated the risk basis for the previously approved TS change. This re-evaluated risk basis, when combined with the satisfactory results from previous tests and inspections, supports the proposed revision to TS 5.5.16 that would allow the next Type A test to be performed no later than April 9, 2007.

Description of the Proposed Change

The current HBRSEP, Unit No. 2, TS 5.5.16 requires that the next Type A test be performed "no later than May 9, 2004." The proposed change would revise TS 5.5.16 to allow the next Type A test for HBRSEP, Unit No. 2, to be performed "no later than April 9, 2007." This proposed change is intended to take full advantage of the one-time extension of the Type A test from once in 10 years to once in 15 years. As described within the Technical Justification below, the bases for the proposed change are the satisfactory results from previous tests and inspections, combined with the re-evaluation of the risk basis for the previously approved TS change.

Technical Justification

Summary of Test and Inspection Programs

Satisfactory results from previous Type A tests at HBRSEP, Unit No. 2, as well as continued satisfactory results from local leak rate tests and containment inspections, support the proposed one-time extension of the containment Type A test interval. The HBRSEP, Unit No. 2, reactor containment vessel (CV) will continue to be inspected under the requirements of the HBRSEP, Unit No. 2, programs for Subsections IWE and IWL of the American Society of Mechanical Engineers (ASME) Code, Section XI. The existing Type B and C containment penetration testing program will continue to be performed in accordance with previous regulatory approvals.

Previous Type A Test Results

As shown within Table 1 below, HBRSEP, Unit No. 2, has performed six operational Type A tests, with each test passing the as-found acceptance criteria. The design basis containment leak rate limit (L_a) is 0.1% weight per day.

Table 1
Type A Test Results

Test Date	Test Results [% weight per day]	Results Adjusted to P_a [% weight per day]
May 1974	0.013	0.013
Feb. 1978	0.035*	0.049
Mar. 1982	0.026*	0.037
Nov. 1984	0.011*	0.016
April 1987	0.041*	0.058
April 1992	0.0644	0.0644

* Test performed at ½ P_a and results are as calculated at ½ P_a

Subsection IWE and IWL Program Results

Detailed information regarding the Subsection IWE and IWL programs for containment inspections was provided by the HBRSEP, Unit No. 2, letter dated March 26, 2002 (Serial: RNP-RA/02-0028). Certain portions of that information are repeated herein for clarity and completeness.

The HBRSEP, Unit No. 2, Subsection IWE and IWL programs are fully implemented, and expedited examinations for the first period of the program interval have been completed. Visual examinations of 100% of the accessible surfaces of the CV liner were conducted between 1998 and 2001 in accordance with the 1992 Edition (with 1992 Addenda) of the ASME Code for Subsections IWE and IWL. Those examinations are summarized within Table 2, as follows:

Table 2
Subsection IWE and IWL Program Examinations

Examination Date	Examination Summary
1998 (RO-18)	Examination of portions of the CV liner behind the insulation.
1999 (RO-19)	Examination of portions of the CV liner behind the insulation, electrical penetrations, the personnel airlock, and portions of the reinforced concrete exterior.
2001 (RO-20)	Examination of portions of the CV liner behind the insulation, the dome interior, mechanical penetrations, equipment hatch, and the remaining portions of the reinforced concrete exterior, including the dome exterior.

Examinations consisted of a general visual examination of accessible areas of the CV liner (pressure boundary) and the reinforced concrete exterior (structural integrity). Although the CV liner between the floor and the CV dome is insulated and not readily accessible for visual examination, selected sections of insulation were removed over the last four refueling outages to allow VT-3 examinations of portions of the CV liner. HBRSEP, Unit No. 2, Relief Request IWE/IWL-01 was obtained to allow inspection of a portion of the CV liner.

In accordance with IWE-1240, "Surface Areas Requiring Augmented Examination," an engineering evaluation has been developed to determine areas that might require augmented examinations. No areas exist that are currently categorized as Examination Category E-C for augmented examinations.

For seals, gaskets, and examination of bolting, HBRSEP, Unit No. 2, has been granted relief from certain Code requirements as follows:

- Relief Request IWE/IWL-04 provides relief from examination of the containment seals and gaskets for Class MC and Class CC components. The approved alternative is performance testing of Type B and C penetrations in accordance with Option A of 10 CFR 50, Appendix J. HBRSEP, Unit No. 2, performs these tests each refueling outage to verify the functionality and integrity of the containment seals and gaskets.

- Relief Request IWE/IWL-03 and IWE/IWL-07 provide relief from examination and testing of pressure-retaining bolting:
  - o IWE/IWL-03 requires examination of pressure-retaining bolting for conditions that may cause the bolted connection to violate containment leak-tightness or structural integrity. Conditions identified during the visual examinations would be subsequently evaluated.
  - o IWE/IWL-07 provides relief from torque or tension testing of bolted connections. As an alternative to this testing, local leak rate testing is conducted in accordance with 10 CFR 50, Appendix J, to confirm leak-tight integrity.

For potential degradation of the uninspectable (embedded) side of the CV liner, HBRSEP, Unit No. 2, has performed limited ultrasonic testing (UT) on CV liner panels. For those CV liner panels that have been visible after removal of insulation and sheathing, the practice has been to take UT measurements on approximately one foot centers for the approximate 3' 8" by 7' 8" panels. Approximately 100 panels have been examined in this manner. These measurements did not indicate degradation of the embedded side of the CV liner.

These Subsection IWE and IWL program examinations, which were completed during RO-20 in May 2001, demonstrated that the structural integrity and leak-tightness of the HBRSEP, Unit No. 2, containment have not been compromised.

An additional inspection of the entire uninsulated CV dome liner was conducted during RO-21 with no indications of through-wall corrosion. This provides continued assurance of the integrity of the CV dome liner.

The above-described surveillances and inspections provide a high degree of assurance that degradation of the containment structure will be detected and corrected before it can produce a containment leak path or impact structural integrity.

Re-Evaluation of Plant-Specific Risk Basis

Attachment III to this letter provides the HBRSEP, Unit No. 2, calculation RNP-F/PSA-0020, "Evaluation of Risk Significance of ILRT Extension," which evaluates the risk impact associated with extending the Integrated Leak Rate Test (ILRT) interval. This analysis uses a method developed for the Crystal River Unit 3 ILRT extension. A sensitivity evaluation of certain conservative assumptions is also provided.

The following are the risk impact summary table and conclusions from the Attachment III calculation. The change in risk resulting from a proposed 15 year ILRT test interval is compared to three different test intervals. The comparisons with the 3 tests in 10 years and with the 1 test in 10 years are based on the intervals from 10 CFR 50, Appendix J. The comparison with the 12.1 year test interval is to show the change in risk from the current Technical Specifications interval as approved in Amendment No. 193.

SUMMARY OF RISK IMPACT				
	3 Tests in 10 Years	1 Test in 10 Years	1 Test in 12.1 Years	1 Test in 15 Years
Total Integrated Risk (Person-Rem/year)	90.687	90.697	90.699	90.700
Type A Testing Risk (Person-Rem/year)	0.092	0.101	0.103	0.105
% Total Risk (Type A/Total)	0.101%	0.111%	0.114%	0.116%
Type A Large Early Release Frequency (LERF) (Class 3b) (per year)	8.99E-7	9.89E-7	1.01E-6	1.03E-6
CHANGES DUE TO EXTENSION FROM 12.1 YEARS (CURRENT)				
Δ Risk from current (Person-Rem/year)				0.001
% Increase from current (Δ Risk / Total Risk)				0.001%
Δ LERF from current (per year)				2.00E-8
Δ Conditional Containment Failure Probability (CCFP) from current				0.089%
CHANGES DUE TO EXTENSION FROM 10 YEARS				
Δ Risk from 1-in-10 (Person-rem/year)				0.003
% Increase from 1-in-10 (Δ Risk / Total Risk)				0.003%
Δ LERF from 1-in-10 (per year)				4.10E-8
Δ CCFP from 1-in10				0.177%
CHANGES DUE TO EXTENSION FROM 3 YEARS (BASELINE)				
Δ Risk from baseline (Person-rem/year)				0.013
% Increase from baseline (Δ Risk / Total Risk)				0.014%
Δ LERF from baseline (per year)				1.31E-7
Δ CCFP from baseline				1.080%

The risk impact of the proposed change to the ILRT test interval has been shown to be very minor, by a number of measures. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," provides guidance for determining the risk impact of plant-specific changes. It defines "very small" changes in risk as those resulting in increases in core damage frequency (CDF) below 1E-6/year and increases in LERF below 1E-7/year. Since the ILRT does not impact CDF, the relevant metric is LERF.

Referring to the information in the table above, the change in LERF due to extending the test interval from the current once-per-12.1-year basis to a once-per-15-year basis is 1.03E-6 minus 1.01E-6, which is approximately 2.0E-8/year. This change is "very small." Comparing the change in LERF due to extending the test interval from the once-per-10-year basis to a once-per-15-year basis is 1.03E-6 minus 9.89E-7, which is approximately 4.1E-8/year. This change is "very small."

Comparing the proposed once-per-15-year basis to a 3-in-10-year basis results in a change in LERF of 1.31E-7/year, which is considered "small."

Note that this result was obtained using a method which contained substantial conservatisms. For example, the potential for a release due to the proposed change in test interval was estimated by multiplying the postulated probability of undetected containment failure by the total core damage frequency. However, some core damage accidents and releases will by definition occur late and cannot contribute to LERF. Therefore, the CDF value used to calculate the risk significance of the postulated undetected containment failure could be adjusted downward to account for this. Suitable corrections for accidents which by definition must result in LERF, and for accidents which cannot result in LERF, would result in a significantly lower applicable CDF value. As a result, a change in LERF that is "very small" is obtained, even for the comparison between a once-per-15-year test basis and a 3-per-10-year test basis.

No Significant Hazards Consideration Determination

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

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

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

   The proposed change to TS 5.5.16 provides a one-time extension of the containment Type A test interval to 15 years for HBRSEP, Unit No. 2. The proposed TS change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The containment vessel is designed to provide a leak-tight barrier against the uncontrolled release of radioactivity to the environment in the unlikely event of postulated accidents. As such, the containment vessel is not considered as the initiator of an accident. Therefore, the proposed TS 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 containment Type A tests. Type B and C leakage testing will continue to be performed at the intervals specified in 10 CFR Part 50, Appendix J, Option A, as required by the HBRSEP, Unit No. 2, TS. As documented in NUREG-1493, "Performance-Based Containment Leakage-Test

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

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

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

The proposed change to TS 5.5.16 provides a one-time extension of the containment Type A test interval to 15 years for HBRSEP, Unit No. 2. The proposed change to the Type A test interval does not result in any physical changes to HBRSEP, Unit No. 2. In addition, the proposed test interval extension does not change the operation of HBRSEP, Unit No. 2, such that a failure mode involving the possibility of a new or different kind of accident from any accident previously evaluated is created.

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

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

The proposed change to TS 5.5.16 provides a one-time extension of the containment Type A test interval to 15 years for HBRSEP, Unit No. 2. The NUREG-1493 study of the effects of extending containment leak rate testing found that a 20 year extension for Type A testing resulted in an imperceptible increase in risk to the public. NUREG-1493 found that, generically, the design containment leak 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 leak paths are detected by Type B and C testing.

The proposed change only involves a one-time extension of the interval for containment Type A testing; the overall containment leak rate specified by the HBRSEP, Unit No. 2, TS is being maintained. Type B and C testing will continue to be performed at the frequency required by the HBRSEP, Unit No. 2, TS. The regular containment inspections being performed in accordance with the ASME Code, Section XI, and the Maintenance Rule

(10 CFR 50.65) provide a high degree of assurance that the containment will not degrade in a manner that is only detectable by Type A testing. In addition, a plant-specific risk evaluation demonstrates that the extension of the Type A test interval from 10 years to 15 years results in a "very small" increase in risk for those accident sequences influenced by Type A testing and a "small" increase in risk when compared to the test frequency of 3 tests per 10 years.

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

Based on the above discussion, Progress Energy Carolinas, Inc., has determined that the requested change does not involve a significant hazards consideration.

Environmental Impact Consideration

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

Proposed Change

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

Basis

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

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

2. Extension of the allowable date for the next containment Type A test has no negative impact on effluent releases. Therefore, the proposed change does not result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite.

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

United States Nuclear Regulatory Commission
Attachment III to Serial: RNP-RA/03-0101
32 Pages (including cover page)

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

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

CALCULATION RNP-F/PSA-0020, "EVALUATION OF RISK SIGNIFICANCE OF ILRT EXTENSION"

SYSTEM # 9400

CALC. SUB-TYPE PSA

PRIORITY CODE 4

QUALITY CLASS D

NUCLEAR GENERATION GROUP

RNP-F/PSA-0020

(Calculation #)

Evaluation of Risk Significance of ILRT Extension

(Title including structures, systems, components)

☐ BNP UNIT ______

☐ CR3 ☐ HNP ☒ RNP ☐ NES ☐ ALL

APPROVAL

REV	PREPARED BY	REVIEWED BY	SUPERVISOR
0	Signature S/ Bruce A. Morgen	Signature S/ David N. Miskiewicz	Signature S/ Steven. A. Laur
	Name Bruce A. Morgen	Name David N. Miskiewicz	Name Steven A. Laur
	Date 8-7-03	Date 8-7-03	Date 8-7-03
1	Signature Bradley W. Dolan	Signature D.N. Miskiewicz	Signature S A Laur Via Telecon BWD
	Name Bradley W. Dolan	Name David N. Miskiewicz	Name Steven A. Laur
	Date 8/18/03	Date 8/18/03	Date 8/18/03

(For Vendor Calculations)

Vendor______________ Vendor Document No. ______________

Owner's Review By______________ Date______________

LIST OF EFFECTIVE PAGES

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				2	0	1
				AMENDMENTS		
				Letter	Rev	Number of Pages

REVISION SUMMARY

Rev. #	Date	Revision Summary (list ECs incorporated)
0	08/07/03	This calculation evaluates the risk significance of extending the ILRT test interval at the Robinson Nuclear Plant (RNP) to 15 years, using the method developed for the Crystal River 3 (CR3) ILRT extension.
1	8/14/03	Corrects typos and makes other administrative changes

TABLE OF CONTENTS

1.0 PURPOSE

This calculation evaluates the risk significance of extending the ILRT test interval at the Robinson Nuclear Plant (RNP) to 15 years, using the method developed for the Crystal River 3 (CR3) ILRT extension. A sensitivity evaluation to certain conservative assumptions is also provided.

2.0 LIST OF REFERENCES

2.1 Carolina Power & Light Company, H. B. Robinson Steam Electric Plant Unit No. 2, Individual Plant Examination Submittal, August 1992.
2.2 RNP-F/PSA-0001, "Updated Individual Plant Examination Probabilistic Safety Assessment Model", Revision 1, July 24, 2000.
2.3 F01-0001, "Evaluation of Risk Significance of ILRT Extension", Revision 2, June 19, 2001.
2.4 RNP-RA/02-0028, "Request for Technical Specification Change Regarding One-Time Extension of Containment Type A Test Interval", March 26, 2002.
2.5 EC 50759, "Model of Record MOR99 Summary of Results", October 29, 2002.
2.6 RNP-F/PSA-0048, "PSA Model Section 9.0 – Source Term Development", Revision 0, June 3, 2003.
2.7 RNP-F/PSA-0050, "Estimate of 50 Mile Population Dose from Design Basis Containment Leakage Following a Core Melt Accident", Revision 0, November 29, 2001.
2.8 RSC-01-44, "Surrogate Level 3 Evaluation Methodology", Revision 0, August 2001.
2.9 Excel Spreadsheet "RNP SUMMARY MOR99 r3f.xls", P:\Site\HNPApps\Control\ARCHIVE\PSA\RNP\Appendix S-Summary Document

3.0 ENGINEERING ANALYSIS SOFTWARE

3.1 Computer Codes Used

None.

3.2 Computers Used

None.

4.0 BODY OF CALCULATION

4.1 Design Inputs

The Probabilistic Safety Assessment (PSA) model does not provide plant design basis information nor is the PSA model used to modify design outputs. Therefore, no design inputs are used.

References 2.1 and 2.2 provide the updated RNP Individual Plant Evaluation (IPE) PSA model used to determine the baseline risk when evaluating the significance of an ILRT extension. The current PSA model is referred to as MOR99. Reference 2.3 provides the method submitted to and approved by the NRC to evaluate the risk significance of an ILRT extension at CR3 (referred to as the CR3 method). The inputs to the ILRT evaluation are documented in References 2.4, 2.5, 2.6 and 2.7. The Reference 2.4 calculation also provides a proprietary vendor methodology, RSC-01-44 (Surrogate Level 3 Evaluation Methodology) Reference 2.8. This was submitted to the NRC for a previous request by RNP for an extension to the ILRT interval. In addition, this reference provides details of the RNP plant damage classes. Reference 2.5 provides an update to the summary spreadsheet (Reference 2.9) for the current PSA model and together with the plant damage classes from Reference 2.4 provides information to map release categories into the EPRI classes. References 2.6 and 2.7 provide the source term input and release information for intact containment.

4.2 Assumptions

Consistent with the CR3 method (Reference 2.3), the following assumptions are applicable to the calculation of the risk significance of the ILRT extension.

1. The maximum containment leakage for EPRI Class 1 sequences is 1 L_a (Type A acceptable leakage) because a new Class 3 has been added to account for increased leakage due to Type A inspections.
2. The maximum containment leakage for Class 3a sequences is 10 L_a based on the previously approved methodology.
3. The maximum containment leakage for Class 3b sequences is 35 L_a based on the previously approved methodology.
4. Class 3b is conservatively categorized LERF based on the previously approved methodology.
5. Containment leakage due to EPRI Classes 4 and 5 are considered negligible based on the previously approved methodology.
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.

4.3 Calculations

4.3.1 Evaluation Methodology

This calculation uses the CR3 method (Reference 2.3) to determine the risk significance of an increase in the ILRT test interval for RNP containment to a total interval of fifteen (15) years. The method is summarized below:

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

4.3.2 Determination of Risk Significance of ILRT Extension

The current RNP PSA Level 2 results are developed within a spreadsheet (Reference 2.9) as updated per Reference 2.5. The results include the contribution to core damage frequency (CDF) by plant damage class and by the release category frequencies. These are used to develop the inputs required by the CR3 method. The person-rem inputs are based on the RNP PSA release category source term and the surrogate level 3 methodology developed by a contractor (Reference 2.8).

STEP 1 – MAP RELEASE CATEGORIES

The RNP plant damage classes are summarized in Table 1. The representative sequences are consistent with and reproduced from those documented by Reference 2.4, with the exception of damage classes X14C, X14F and 12R. These differences are a result of the update to the spreadsheet and are representative of large bypass sequences. The MOR99 CDF is 4.32E-5/yr.

TABLE 1 – DAMAGE CLASSES		
DAMAGE CLASS	**REPRESENTATIVE SEQUENCE**	**FREQUENCY /YR**
5D	Loss of component cooling water occurs, losing all cooling to the reactor coolant pumps and resulting in an RCP seal LOCA. Injection is successful, but recirculation fails due to lack of CCW cooling. The failure of recirculation fails containment sprays.	1.31E-5
10P	Loss of offsite power with failure of all ac power resulting in a RCP seal LOCA. Limited operation of the AFW steam-driven	5.88E-6

TABLE 1 – DAMAGE CLASSES		
DAMAGE CLASS	REPRESENTATIVE SEQUENCE	FREQUENCY /YR
	pump delays core damage.	
5J	Same as 5D with failure of fans.	4.92E-6
3P	A transient event without heat removal. Containment cooling is also failed. The containment is isolated.	2.90E-6
19A	Transient with failure of the reactor to scram resulting in an RCS overpressure condition which fails the RCS and safety injection.	2.83E-6
2A	Loss of offsite power, failure of AFW early, and failure of the operators to establish feed and bleed cooling.	2.43E-6
B5E	Steam generator tube rupture event with a failure of makeup during recirculation and a loss of containment sprays upon recirculation.	2.02E-6
12O	Interfacing systems LOCA through the RHR hot leg suction line.	1.26E-6
6J	A large service water flood occurs which results in a reactor cooling pump seal LOCA and a failure of all injection. In addition, all containment safeguards are failed.	1.12E-6
4D	Small LOCA that is too small to remove decay heat occurs, failure of AFW early, successful feed and bleed cooling during injection, and failure of the operators to accomplish high-head recirculation results in containment spray failure.	6.52E-7
17D	Medium LOCA occurs with successful injection and failure of the operators to establish high-head recirculation. The failure to establish recirculation also fails containment sprays.	6.22E-7
20D	Large LOCA occurs with failure of the operators to establish recirculation. The failure of recirculation results in containment spray failure.	6.00E-7
10A	Although different break location and somewhat earlier timing, this sequence similar to RCP seal LOCA cases (10J).	4.97E-7
B9B	Similar to PDS B9E below except that containment sprays function.	4.88E-7
B9E	A steam generator tube rupture occurs with a failure of AFW when the CST empties (~4 hours). The operators fail to accomplish shutdown cooling or high-head recirculation. The failure of recirculation results in containment spray failure and the faulted steam generator results in a small bypass through a cycling steam generator SRV.	4.88E-7
20A	Similar to PDS 20D with the difference being the successful operation of containment sprays.	4.75E-7
2D	Loss of offsite power, failure of AFW early, and failure of the operators to establish feed and bleed cooling. The containment sprays initiate but fail during recirculation.	4.45E-7
B10B	Steam generator tube rupture with a failure of makeup. All containment safeguards available.	3.69E-7

TABLE 1 – DAMAGE CLASSES		
DAMAGE CLASS	REPRESENTATIVE SEQUENCE	FREQUENCY /YR
5E	Loss of component cooling water occurs, losing all cooling to the reactor coolant pumps and resulting in an RCP seal LOCA. Injection is successful, but recirculation fails due to a lack of CCW cooling. The failure of recirculation fails containment sprays. A small isolation failure is also present.	3.66E-7
10J	A loss of service water occurs which fails all RCP seal cooling and safety injection. In addition, all containment safeguards are lost.	2.80E-7
X14C	Large Bypass	2.76E-7
4J	Small LOCA sequence with the loss of all safeguards after the failure of containment sprays in recirculation.	2.67E-7
5K	Similar to 5D with the failure of all safeguards.	1.69E-7
19B	Transient with failure of the reactor to scram resulting in an RCS overpressure condition which fails the RCS and safety injection. In addition, a small containment isolation failure exists.	8.90E-8
16A	Similar to 16D found in PSA submittal. Difference is that for this PDS the containment sprays are not failed by the recirculation failure. In both cases, fan coolers are available to provide containment heat removal and existing MAAP runs adequate.	6.65E-8
3J	A total loss of service water, occurs with a failure of AFW early, The loss of service water fails RHR, recirculation and containment sprays. The fans are failed due to the loss of service water.	5.66E-8
B5B	Steam generator tube rupture with failure of makeup to the reactor vessel. The steam generator tube rupture indicates a small leak with secondary-side heat removal available.	4.84E-8
B10N	Steam generator tube rupture with failure of makeup and containment sprays.	4.62E-8
10R	A loss of service water occurs which fails all RCP seal cooling and safety injection. In addition, all containment safeguards are lost early in the event with the failure of containment sprays prior to recirculation.	3.43E-8
2J	Similar to total loss of service water (3K).	3.25E-8
11J	Small LOCA or seal LOCA sequence with a failure of core heat removal and containment heat removal.	3.04E-8
X14F	Large Bypass	3.00E-8
10Q	Similar to 10D except that all containment cooling is lost.	2.90E-8
10D	Small LOCA or seal LOCA with failure of high-pressure injection and low-pressure injection. Containment spray fails in recirculation.	2.63E-8
11P	A large service water flood occurs resulting in a seal LOCA and failure of all containment safeguards. Containment isolation,	2.14E-8

TABLE 1 – DAMAGE CLASSES		
DAMAGE CLASS	REPRESENTATIVE SEQUENCE	FREQUENCY /YR
	however, is successful.	
5F	Small LOCA or seal LOCA with a failure of low-pressure injection late in the sequence. A large isolation failure exists and sprays fail in recirculation.	1.91E-8
20B	Large LOCA sequence with a small isolation failure.	1.87E-8
3Q	Loss of heat removal sequence with the failure of all containment safeguards and a small isolation failure.	1.33E-8
3R	Similar to 3Q except a large isolation failure exists.	1.33E-8
20E	Large LOCA occurs with failure of the operators to establish recirculation. The failure of recirculation results in containment spray failure.	1.19E-8
5L	Small LOCA or seal LOCA with a failure of cooling late, no containment cooling and a large isolation failure.	1.05E-8
2B	Loss of offsite power, failure of AFW early, and failure of the operators to establish feed and bleed cooling. In addition, a small containment isolation failure exists.	9.98E-9
5A	Similar to 5F except that all safeguards are functioning and the containment is intact.	9.63E-9
3G	Transient occurs with a loss of heat removal and failure of feed-and-bleed. The containment fan coolers are also failed.	8.47E-9
17A	A medium LOCA sequence with a failure of recirculation and all safeguards available.	7.86E-9
2P	Similar to 2A with a failure of containment sprays early in the sequence.	7.34E-9
17E	Medium LOCA occurs with successful injection and failure of the operators to establish high-head recirculation. The failure to establish recirculation also fails containment spray.	6.48E-9
10B	Similar to PDS 15F.	5.96E-9
4E	Similar to PDS 4J except that fans function and a small isolation failure exists.	5.88E-9
19D	Similar to PDS 19A except that containment sprays fail in recirculation.	4.92E-9
12R	Large Bypass.	4.80E-9
6D	Similar to a small LOCA case (4D) with the failure of AFW and recirculation. RHR failure fails containment sprays.	4.71E-9
4K	Similar to PDS 4J except that a small isolation failure is present.	4.67E-9
2C	Similar to PDS 2A with the exception that a small isolation failure is present.	4.58E-9
2E	Similar to PDS 2C with the exception that the containment sprays are unavailable in recirculation.	4.58E-9

TABLE 1 – DAMAGE CLASSES		
DAMAGE CLASS	REPRESENTATIVE SEQUENCE	FREQUENCY /YR
2F	Similar to 2E except that the isolation failure is large.	4.58E-9
B10E	A steam generator tube rupture that is similar to B10N except containment sprays function until recirculation.	4.17E-9
Total		4.32E-5

The person-rem associated with each release category is developed using the proprietary method documented by RSC 01-44 (Reference 2.8). This method assigns a dose conversion factor to each of five radionuclide groups. The dose conversion factors and release category source term assignments are listed in Table 2.

TABLE 2 – DOSE CONVERSION FACTORS
This Table contains proprietary data from Table 1 of calculation RSC 01-44. A proprietary version of RSC 01-44 was previously submitted to the NRC as Attachment VI of letter RNP-RA/02-0028, dated March 26, 2002

These source term assignments are taken from Reference 2.6 and grouped as described in Table 2. The person-rem is calculated by multiplying the release fraction for each radionuclide group by the dose conversion factor in Table 2, and summing for each release category.

The person-rem for the intact containment is the design basis leakage, L_a calculated as the 50 mile population dose (Reference 2.7). The RNP release category, frequency and person-rem are summarized in Table 3.

TABLE 3 – RNP RELEASE CATEGORIES							
RELEASE CATEGORY	FREQUENCY /YR	NOBLE GAS[4]	IODINE[4]	CESIUM[4]	TELLU-RIUM[4]	STRON-TIUM[4]	PERSON-REM
IC-1[1]	2.18E-5	NA[2]	NA	NA	NA	NA	1.56E+3[3]
RC-1	1.01E-5	1.00E+0	1.80E-3	2.30E-3	5.40E-5	7.40E-6	1.71E+6
RC-1A	2.08E-7	1.00E+0	1.48E-1	8.74E-2	2.10E-5	0.00E+0	1.33E+7
RC-1B	4.83E-6	1.00E+0	7.50E-3	6.90E-3	4.89E-2	5.00E-4	2.47E+6
RC-1BA	2.46E-7	1.00E+0	4.71E-2	5.82E-2	7.69E-2	8.00E-4	7.15E+6
RC-2	3.51E-8	1.00E+0	2.63E-2	2.83E-2	4.30E-6	1.00E-4	4.23E+6
RC-2B	1.70E-7	1.00E+0	1.65E-1	1.90E-1	9.21E-2	3.90E-3	1.97E+7
RC-3	7.29E-7	2.00E-1	8.00E-4	5.00E-4	6.00E-4	8.60E-6	3.68E+5
RC-3B	2.97E-9	2.00E-1	8.00E-4	5.00E-4	6.00E-4	1.10E-3	3.74E+5
RC-4	0.00E+0	4.00E-1	1.70E-2	1.53E-2	0.00E+0	1.70E-5	2.22E+6
RC-4C	3.47E-6	4.00E-1	1.70E-2	1.53E-2	0.00E+0	1.70E-5	2.22E+6
RC-5	1.20E-6	1.00E+0	2.86E-1	2.61E-1	0.00E+0	8.00E-3	2.89E+7
RC-5C	3.70E-7	1.00E+0	2.86E-1	2.61E-1	0.00E+0	8.00E-3	2.89E+7
Total	4.32E-5						

1. Fission product groups are from Reference 2.6 grouped per RSC 01-44 (Reference 2.3).
2. Release fractions not necessary for this calculation.
3. Intact containment representing design basis leakage (Reference 2.7).
4. see Table 9.4 of RNP Source Term Analysis (Reference 2.6)

The release categories are mapped into the eight (8) EPRI release classes as shown in Table 4.

TABLE 4 – EPRI CLASSES			
EPRI CLASS	**DESCRIPTION**	**RNP INTERPRETATION**	**RNP RELEASE CATEGORY ASSIGNMENT**
1	Containment remains intact with containment initially isolated.	Intact containment bins.	IC-1
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.	None
3	Independent containment isolation failures due to Type A related failures.	Isolation failures identified by Type A testing.	None
4	Independent containment isolation failures due to Type B related failures.	Isolation failures identified by Type B testing.	None
5	Independent containment isolation failures due to Type C related failures.	Isolation failures identified by Type C testing.	None
6	Other penetration failures.	Other faults not previously identified.	RC-2, RC-2B, RC-3
7	Induced by severe accident phenomena.	Early containment failure sequences as a result of hydrogen burn or other early phenomena.	RC-1, RC-1A, RC-1B, RC-1BA, RC-3B
8	Bypass.	Bypass sequences or SGTR.	RC-4C, RC-5, RC-5C

The frequencies from Table 3 are summed by release category comprising each EPRI class. The person-rem is the frequency-weighted person-rem for the release categories comprising the EPRI class. The person-rem per year is the category person-rem multiplied by the category frequency. The RNP PSA Release Category Groupings to EPRI Class are summarized in Table 5.

TABLE 5 – RNP RELEASE CATEGORY TO EPRI MAPPING					
EPRI CLASS	DESCRIPTION	RNP RELEASE CATEGORY	FREQUENCY /YR	PERSON -REM	PERSON-REM/YR
1	No Containment Failure	IC-1	2.18E-5	1.56E+3	0.034
2	Large Containment Isolation Failures	None	0[1]	0	0
3a	Small Isolation Failures (Liner breach)	None	0	0	0
3b	Large Isolation Failures (Liner breach)	None	0	0	0
4	Small isolation failures - failure to seal (type B)	None	-	-	-
5	Small isolation failures - failure to seal (type C)	None	-	-	-
6	Containment Isolation Failures (dependent failure, personnel errors)	RC-2	3.51E-8		0.148
		RC-2B	1.70E-7		3.349
		RC-3	7.29E-7		0.268
		Class 6	9.34E-7	4.03E+6	3.765
7	Severe Accident Phenomena Induce Failure (Early and Late)	RC-1	1.01E-5		17.271
		RC-1A	2.08E-7		2.766
		RC-1B	4.83E-6		11.930
		RC-1BA	2.46E-7		1.759
		RC-3B	2.97E-9		0.001
		Class 7	1.54E-5	2.19E+6	33.727
8	Containment Bypass	RC-4C	3.47E-6		7.703
		RC-5	1.20E-6		34.680
		RC-5C	3.70E-7		10.693
			5.04E-6	1.05E+7	53.076
		Total	4.32E-5		90.602
1. No result. Contributions are below quantification truncation value.					

STEP 2 – CALCULATE TYPE A LEAKAGE

As shown in Table 5, the RNP PSA does not associate any release categories with EPRI Classes 2, 3, 4, or 5. Therefore each of these classes must be evaluated for applicability.

<u>Class 3</u>

Containment failures in this class are due to leaks such as liner breaches which would only be detected by performing a Type A ILRT.

For this estimation, the question on containment isolation was modified consistent with the previously approved methodology to include the probability of a liner breach (due to excessive leakage) at the time of core damage. Using this methodology, Class 3 is divided into two classes. These are Class 3a (small liner breach) and Class 3b (large liner breach).

To calculate the probability that a liner leak will be large (Class 3b), use was made of the data presented in NUREG-1493. 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 (La). Since 21 L_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 $\chi 2$ distribution. In statistical theory, the $\chi 2$ distribution can be used for statistical testing, goodness-of-fit tests. The $\chi 2$ 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, v. For time-truncated tests (versus failure-truncated tests), an estimate of the probability of a large leak using the $\chi 2$ distribution can be calculated as $\chi 2_{95}(v = 2n+2)/2N$, where n represents the number of large leaks and N represents the number of ILRTs performed to date. With no large leaks (n = 0) in 144 events (N = 144) and $\chi 2_{95}(2) = 5.99$, the 95th percentile estimate of the probability of a large leak is calculated as 5.99/(2*144) = 0.0208.

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

$FREQ3b = PROB_{class3b} \times CDF$

FREQ3b = 0.0208 * 4.32E-5 = 8.99E-7/yr

To calculate the probability that a liner leak will be small (Class 3a), use was made of the data presented in NUREG-1493. 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.OLa. However, of these 23 'failures' only 4 were found by an ILRT, the others were found by Type B and C testing or errors in test alignments. Therefore, the number of failures considered for 'small releases' are 4-of-144. Similar to the Class 3b probability, the estimated failure probability for small release is found by using the $\chi 2$ distribution. The $\chi 2$ distribution is calculated by n=4 (number of small leaks) and N=144 (number of events) which yields a $\chi 2(10) = 18.3070$. Therefore, the 95th percentile estimate of the probability of a small leak is calculated as 18.3070/(2*144) = 0.0636.

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

$FREQ3a = PROB_{class3a} \times CDF = 0.0636 \times 4.32E\text{-}05/yr$

FREQ3a = 0.0636 * 4.32E-5 = 2.75E-6/yr

Note: Using the methodology discussed above is conservative compared to the typical mean estimates used for PRA analysis. The mean probability of a Class 3 failure would be the (number of failures)/(number of tests) or 4/144 = 0.03.

Class 1

Although Type A testing does not directly impact the frequency of this class, the PSA did not model Class 3 failures, and the frequency of 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:

FREQ1 = FREQ1 – (FREQ3a + FREQ3b)
FREQ1 = 2.18E-5 – (2.75E-6 + 8.99E-7) = 1.82E-5/yr

Class 2

Table 5 does not identify any contribution to Class 2.

Class 4

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

Class 6

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

FREQ6 = 9.34E-7

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

FREQ7 = 1.54E-5

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

FREQ8 = 5.04E-6

The person-rem per year is recalculated using these revised frequencies and the assumption concerning leakage, L_a from intact containment (EPRI Class 1).

The CR3 method calculates the Type A leakage to define the baseline (3 year test interval, also referred to as the 3-in-10). Table 5 results are revised to account for EPRI Class 3 and the new baseline is shown as Table 6.

TABLE 6 – (3-IN-10 OR BASELINE)					
EPRI CLASS	DESCRIPTION	RNP RELEASE CATEGORY	FREQUENCY /YR	PERSON -REM	PERSON-REM/YR
1	No Containment Failure	IC-1	1.82E-5	1.56E+3[1]	0.028
2	Large Containment Isolation Failures	None	0'	0	0
3a	Small Isolation Failures (Liner breach)	None	2.75E-6	1.56E+4[2]	0.043
3b	Large Isolation Failures (Liner breach)	None	8.99E-7	5.46E+4[3]	0.049
4	Small isolation failures - failure to seal (type B)	None	-	-	-
5	Small isolation failures - failure to seal (type C)	None	-	-	-
6	Containment Isolation Failures (dependent failure, personnel errors)	RC-2	3.51E-8		0.148
		RC-2B	1.70E-7		3.349
		RC-3	7.29E-7		0.268
		Class 6	9.34E-7	4.03E+6	3.765
7	Severe Accident Phenomena Induce Failure (Early and Late)	RC-1	1.01E-5		17.271
		RC-1A	2.08E-7		2.766
		RC-1B	4.83E-6		11.930
		RC-1BA	2.46E-7		1.759
		RC-3B	2.97E-9		0.001
		Class 7	1.54E-5	2.19E+6	33.727
8	Containment Bypass	RC-4C	3.47E-6		7.703
		RC-5	1.20E-6		34.680
		RC-5C	3.70E-7		10.693
		Class 8	5.04E-6	1.05E+7	53.076
		Total	4.32E-5		90.687

1. 1 L_a dose value per Reference 2.7
2. 10 times L_a.
3. 35 times L_a.

The percent risk contribution due to Type A testing (baseline value) is the ratio of the person-rem/yr from Classes 3a and 3b to the total.

%Risk (base) = [(0.043 + 0.049)/90.687] * 100 = 0.101%

EPRI Class 3b is considered to be LERF. Thus the baseline LERF from Class 3b is 8.99E-7/yr.

STEP 3A – CALCULATE TYPE A LEAKAGE FOR CURRENT INTERVALS

The current surveillance testing requirements as proposed in NEI 94-01 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, extending the Type A ILRT interval from 3-in-10 years to 1-in-10 years will increase the average time that a leak detectable only by an ILRT goes undetected from 18 to 60 months. (The average time for undetection is calculated by multiplying the test interval by 0.5 and multiplying by 12 to convert from "years" to "months"). Since ILRTs only detect about 3% of leaks (4/144), the result for a 10-yr ILRT interval is a 10% increase in the overall probability of leakage. This value is determined by multiplying 3% and the ratio of the average time for undetection for the increased ILRT test interval (60 months) to the baseline average time for undetection of 18 months (i.e., 3 * 60/18).

The CR3 Method calculates the change in %Risk for a 10-year ILRT interval (also referred to as 1-in-10) by determining the increase in probability of not detecting excessive leakage due to Type A testing. The increase in testing interval from 3-in-10 to 1-in-10 results in a 10% increase in the likelihood of a Type A leak.

Based on this, the frequencies for Classes 3a and 3b are obtained by multiplying the frequencies shown in Table 6 by a factor 1.1. This accounts for the increased probability of not detecting a Type A leak due to the ILRT extension:

FREQ3a = 2.75E-6 * 1.1 = 3.03E-6/yr
FREQ3b = 8.99E-7 * 1.1 = 9.89E-7/yr

The Class 1 frequency is reduced by the above to maintain total CDF.

FREQ1 = 2.18E-5 – (3.03E-6 + 9.89E-7) = 1.78E-5/yr

The person-rem per year is recalculated using these revised frequencies and the assumption concerning leakage, L_a from intact containment (EPRI Class 1). The Table 6 results are revised to account for the increase in Class 3 frequency and shown as Table 7.

TABLE 7 – (1-IN-10)					
EPRI CLASS	DESCRIPTION	RNP RELEASE CATEGORY	FREQUENCY /YR	PERSON -REM	PERSON-REM/YR
1	No Containment Failure	IC-1	1.78E-5	1.56E+3[1]	0.028
2	Large Containment Isolation Failures	None	0'	0	0
3a	Small Isolation Failures (Liner breach)	None	3.03E-6	1.56E+4[2]	0.047
3b	Large Isolation Failures (Liner breach)	None	9.89E-7	5.46E+4[3]	0.054
4	Small isolation failures - failure to seal (type B)	None	-	-	-
5	Small isolation failures - failure to seal (type C)	None	-	-	-
6	Containment Isolation Failures (dependent failure, personnel errors)	RC-2	3.51E-8		0.148
		RC-2B	1.70E-7		3.349
		RC-3	7.29E-7		0.268
		Class 6	9.34E-7	4.03E+6	3.765
7	Severe Accident Phenomena Induce Failure (Early and Late)	RC-1	1.01E-5		17.271
		RC-1A	2.08E-7		2.766
		RC-1B	4.83E-6		11.930
		RC-1BA	2.46E-7		1.759
		RC-3B	2.97E-9		0.001
		Class 7	1.54E-5	2.19E+6	33.727
8	Containment Bypass	RC-4C	3.47E-6		7.703
		RC-5	1.20E-6		34.680
		RC-5C	3.70E-7		10.693
		Class 8	5.04E-6	1.05E+7	53.076
		Total	4.32E-5		90.697

1. 1 L_a dose value per Reference 2.7
2. 10 times L_a.
3. 35 times L_a.

The percent risk contribution due to Type A testing is the ratio of the person-rem/yr from Classes 3a and 3b to the total.

%Risk (1-in-10) = [(0.047 + 0.054)/90.697] * 100 = 0.111%

The percent risk increase (delta%Risk10) due to a ten-year ILRT interval over the baseline case is the increase in person-rem/yr divided into the baseline person-rem/yr, or

delta%Risk10 = [(90.697 – 90.687) / 90.687] * 100

delta%Risk10 = 0.0110%

EPRI Class 3b is considered to be LERF. Thus the 1-in-10 LERF from Class 3b is 9.89E-7 /yr.

STEP 3B - CALCULATE TYPE A LEAKAGE FOR CURRENT INTERVALS
Cuurently RNP has been approved for a one cycle ILRT deferral. This increases the currently approved interval to 12.1 years. The change in %Risk for a 12.1-year ILRT interval (also referred to as 1-in-12.1) by determining the increase in probability of not detecting excessive leakage due to Type A testing. The increase in testing interval from 3-in-10 to 1-in-12.1 results in a 12.1% increase in the likelihood of a Type A leak.

Based on this, the frequencies for Classes 3a and 3b are obtained by multiplying the frequencies shown in Table 6 by a factor 1.121. This accounts for the increased probability of not detecting a Type A leak due to the ILRT extension:

FREQ3a = 2.75E-6 * 1.121 = 3.08E-6/yr
FREQ3b = 8.99E-7 * 1.121 = 1.01E-6/yr

The Class 1 frequency is reduced by the above to maintain total CDF.

FREQ1 = 2.18E-5 – (3.08E-6 + 1.01E-6) = 1.77E-5/yr

The person-rem per year is recalculated using these revised frequencies and the assumption concerning leakage, L_a from intact containment (EPRI Class 1). The Table 6 results are revised to account for the increase in Class 3 frequency and shown as Table 8.

TABLE 8 – (1-IN-12.1 OR CURRENT)					
EPRI CLASS	DESCRIPTION	RNP RELEASE CATEGORY	FREQUENCY /YR	PERSON -REM	PERSON-REM/YR
1	No Containment Failure	IC-1	1.77E-5	1.56E+3[1]	0.028
2	Large Containment Isolation Failures	None	0[1]	0	0
3a	Small Isolation Failures (Liner breach)	None	3.08E-6	1.56E+4[2]	0.048
3b	Large Isolation Failures (Liner breach)	None	1.01E-6	5.46E+4[3]	0.055
4	Small isolation failures - failure to seal (type B)	None	-	-	-
5	Small isolation failures - failure to seal (type C)	None	-	-	-
6	Containment Isolation Failures (dependent failure, personnel errors)	RC-2	3.51E-8		0.148
		RC-2B	1.70E-7		3.349
		RC-3	7.29E-7		0.268
		Class 6	9.34E-7	4.03E+6	3.765
7	Severe Accident Phenomena Induce Failure (Early and Late)	RC-1	1.01E-5		17.271
		RC-1A	2.08E-7		2.766
		RC-1B	4.83E-6		11.930
		RC-1BA	2.46E-7		1.759
		RC-3B	2.97E-9		0.001
		Class 7	1.54E-5	2.19E+6	33.727
8	Containment Bypass	RC-4C	3.47E-6		7.703
		RC-5	1.20E-6		34.680
		RC-5C	3.70E-7		10.693
		Class 8	5.04E-6	1.05E+7	53.076
		Total	4.32E-5		90.699

1. 1 L_a dose value per Reference 2.7
2. 10 times L_a.
3. 35 times L_a.

The percent risk contribution due to Type A testing is the ratio of the person-rem/yr from Classes 3a and 3b to the total.

%Risk (1-in-12.1) = [(0.048 +0.0 55)/90.699] * 100 = 0.0114%

The percent risk increase (delta%Risk12.1) due to a fifteen-year ILRT interval over the baseline case is the increase in person-rem/yr divided into the baseline person-rem/yr, or

delta%Risk12.1 = [(90.699 – 90.687) / 90.687] * 100

delta%Risk12.1 = 0.0132%

EPRI Class 3b is considered to be LERF. Thus the 1-in-12.1 LERF from Class 3b is 1.01E-6/yr.

STEP 4 – CALCULATE TYPE A LEAKAGE FOR EXTENDED INTERVAL

The CR3 Method calculates the change in %Risk for a 15-year ILRT interval (also referred to as 1-in-15) by determining the increase in probability of not detecting excessive leakage due to Type A testing. The increase in testing interval from 3-in-10 to 1-in-15 results in a 15% increase in the likelihood of a Type A leak.

Based on this, the frequencies for Classes 3a and 3b are obtained by multiplying the frequencies shown in Table 6 by a factor 1.15. This accounts for the increased probability of not detecting a Type A leak due to the ILRT extension:

FREQ3a = 2.75E-6 * 1.15 = 3.16E-6/yr
FREQ3b = 8.99E-7 * 1.15 = 1.03E-6/yr

The Class 1 frequency is reduced by the above to maintain total CDF.

FREQ1 = 2.18E-5 – (3.16E-6 + 1.03E-6) = 1.76E-5/yr

The person-rem per year is recalculated using these revised frequencies and the assumption concerning leakage, L_a from intact containment (EPRI Class 1). The Table 6 results are revised to account for the increase in Class 3 frequency and shown as Table 9.

TABLE 9 – (1-IN-15)					
EPRI CLASS	DESCRIPTION	RNP RELEASE CATEGORY	FREQUENCY /YR	PERSON -REM	PERSON-REM/YR
1	No Containment Failure	IC-1	1.76E-5	1.56E+3[1]	0.027
2	Large Containment Isolation Failures	None	0	0	0
3a	Small Isolation Failures (Liner breach)	None	3.16E-6	1.56E+4[2]	0.049
3b	Large Isolation Failures (Liner breach)	None	1.03E-6	5.46E+4[3]	0.056
4	Small isolation failures - failure to seal (type B)	None	-	-	-
5	Small isolation failures - failure to seal (type C)	None	-	-	-
6	Containment Isolation Failures (dependent failure, personnel errors)	RC-2	3.51E-8		0.148
		RC-2B	1.70E-7		3.349
		RC-3	7.29E-7		0.268
		Class 6	9.34E-7	4.03E+6	3.765
7	Severe Accident Phenomena Induce Failure (Early and Late)	RC-1	1.01E-5		17.271
		RC-1A	2.08E-7		2.766
		RC-1B	4.83E-6		11.930
		RC-1BA	2.46E-7		1.759
		RC-3B	2.97E-9		0.001
		Class 7	1.54E-5	2.19E+6	33.727
8	Containment Bypass	RC-4C	3.47E-6		7.703
		RC-5	1.20E-6		34.680
		RC-5C	3.70E-7		10.693
		Class 8	5.04E-6	1.05E+7	53.076
		Total	4.32E-5		90.700

1. 1 L_a dose value per Reference 2.7
2. 10 times L_a.
3. 35 times L_a.

The percent risk contribution due to Type A testing is the ratio of the person-rem/yr from Classes 3a and 3b to the total.

%Risk (1-in-15) = [(0.049 + 0.056)/90.700] * 100 = 0.0116%

The percent risk increase (delta%Risk15) due to a fifteen-year ILRT interval over the baseline case is the increase in person-rem/yr divided into the baseline person-rem/yr, or

delta%Risk15 = [(90.700 – 90.687) / 90.687] * 100

delta%Risk15 = 0.0143%

EPRI Class 3b is considered to be LERF. Thus the 1-in-15 LERF from Class 3b is 1.03E-6/yr.

STEP 5 – CALCULATE INCREASE IN RISK
The CR3 Method calculates the increase in risk in terms of increase in person-rem/yr and LERF increase for the ILRT extension. The increase in risk in person-rem/yr is shown in Table 10A.

TABLE 10A – CHANGE IN PERSON-REM/YR			
ILRT INTERVAL	PERSON-REM/YR	DELTA PERSON-REM/YR[1]	% INCREASE[1]
3-in-10 (Baseline)	90.687	-	-
1-in-10	90.697	0.010	0.011%
1-in12.1 (Current)	90.699	0.012	0.013%
1-in-15	90.700	0.013	0.014%
1. Relative to Baseline.			

STEP 6 – CALCULATE CHANGE IN 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 large release due to failure to detect a pre-existing leak during the relaxation period. Based on the previously approved methodology, 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 which result in large releases, are not impacted because a LERF will occur regardless of the presence of a pre-existing leak. Therefore, the frequency of Class 3b sequences is used as the increase in LERF and the change in LERF can be determined by the differences. Table 10B summarizes the results.

TABLE 10B – CHANGE IN LERF (EPRI CLASS 3b)			
ILRT INTERVAL	LERF	DELTA LERF	% INCREASE[1]
3-in-10 (Baseline)	8.99E-7	-	-
1-in-10	9.89E-7	9.89E-8	10%
1-in-12.1 (Current)	1.01E-6	1.11E-7	12.1%
1-in-15	1.03E-6	1.31E-7	15%
1. Relative to Baseline Class 3B LERF only.			

STEP 7 – CALCULATE CHANGE IN CCFP

The conditional containment failure probability (CCFP) is the probability of containment failure given the occurrence of an accident, expressed below where f(ncf) is the frequency of those sequences which result in no containment failure. This is the sum of the Class 1 and Class 3a result.

CCFP = 1 – [f(ncf)/CDF]

Table 11 provides the delta CCFP for the ILRT extension:

TABLE 11 – IMPACT ON CCFP				
ILRT INTERVAL	3-IN-10 (BASELINE)	1-IN-10	1-IN-12.1 (CURRENT)	1-IN-15
f(ncf)	2.10E-5	2.08E-5	2.078E-5	2.076E-5
f(ncp)/CDF	0.48611	0.48148	0.48102	0.48056
CCFP	0.51389	0.51852	0.51898	0.51944
Δ CCFP from baseline	-	0.00463	0.00509	0.00555
Δ CCFP from 1-in-10	-	-	0.00460	0.00092
Δ CCFP from current	-	-	-	0.00046

4.3.3 Sensitivity Evaluation

The currently approved methodology used above, assumes that the LERF contribution created by delaying the ILRT is a function of the total CDF. Arguments can be made that this is very conservative. First, it is unlikely that a liner breach would lead to a release path capable of a "large" release. In addition, some fraction of the total CDF is already classified as LERF or would never result in LERF using the rules defined by the containment event tree, and therefore, should not be a contributor to the ILRT LERF fraction.

The following sensitivty evaluation examines the LERF impact of deferring the ILRT if the LERF fraction due to ILRT Type A test intervals is only a function of the CDF portion which excludes previously defined LERF cases and cases where LERF could not occur.

The following release categories shown in Table 12 are defined as LERF in the RNP PSA.

TABLE 12		
RNP RELEASE CATEGORY	FREQUENCY (/YR)	CUMULATIVE LERF (/YR)
RC-4	0	0
RC-4C	3.47E-6	3.47E-6
RC-5	1.20E-6	4.67E-6
RC-5C	3.70E-7	5.04E-6
RC-2	3.51E-8	5.08E-6
RC-2B	1.70E-7	5.24E-6
LERF		5.24E-6

With the CDF is reduced by the LERF listed in Table 12 the baseline is recalculated as follows:

The frequency of a Class 3b failure is calculated as:

$FREQ3b = PROB_{class3b} \times (CDF - LERF)$

FREQ3b = 0.0208 * (4.32E-5 – 5.24E-6) = 7.90E-7/yr (baseline)
FREQ3b = 1.15 * FREQ3b (baseline) = 9.09E-7 (1-in-15)

With this change, the evaluation result of interest is:

Delta LERF = FREQ3b (1-in-15) – FREQ3b (baseline)
Delta LERF = (9.09E-7 – 7.90E-7) = 1.19E-7

As a further refinement, additional contributions may be excluded for sequences that would not result in large early release regardless of the presence of a Type A failure due to scrubbing provided by containment sprays. Reference 2.4 provides details the method used to select plant damage states with containment spray. Only Plant Damage States (PDSs) with endstates "A", "G", and "D" meet the requirement to be an intact containment state with containment sprays available for both injection and recirculation. The PDSs listed in Table 13 were identified by Reference 2.4 as applicable and contributing sufficient frequency to warrant review. These PDSs are confirmed to appear in the updated spreadsheet as well. Reference 2.4 reviews these PDSs in detail to assure that sequences not meeting this definition are not considered. Thus the PDSs determined to never go to LERF are reduced by those sequences that do not meet the above definition. The net frequency in Table 13 is that portion of CDF that will never be large and early.

TABLE 13 – NEVER LERF PDSs			
RNP PLANT DAMAGE STATE	**INTACT FREQUENCY (/YR) (NEVER LERF)**	**PROVIDE A LERF CONTRIBUTION**	**NET FREQUENCY (/YR)**
5D	9.83E-6	No.	9.83E-6
19A	2.12E-6	Not expected to lead to a LERF release and may be excluded.	2.12E-6
2A	1.96E-6	Yes. Reduce PDS frequency by 1.02E-7 to account for EDG failures that would fail spray.	1.86E-6
4D	4.88E-7	Yes. Reduce PDS contribution by 2.79E-7 to account for cutset with failure of recirculation.	2.09E-7
17D	4.66E-7	Yes. Reduce PDS contribution by 2.31E-7 to account for cutset with failure of recirculation.	2.35E-7
20D	4.50E-7	No.	4.50E-7
10A	3.72E-7	No.	3.72E-7
2D	3.58E-7	No.	3.58E-7
20A	3.56E-7	No.	3.56E-7
16A	4.98E-8	No.	4.98E-8
10D	1.97E-8	No.	1.97E-8
3G	6.75E-9	No.	6.75E-9
5A	7.21E-9	No.	7.21E-9
17A	5.89E-9	No.	5.89E-9
6D	3.53E-9	No.	3.53E-9
19D	6.69E-9	No.	6.69E-9
TOTAL	1.65E-5		
NEVER LERF			1.59E-5

The frequency of a Class 3b failure is re-calculated as:

FREQ3b = $PROB_{class3b}$ x [CDF – (LERF+NEVER LERF)]

FREQ3b = 0.0208 * [4.32E-5 – (5.24E-6 + 1.59E-5)] = 4.60E-7/yr (baseline)

FREQ3b = 1.15 * FREQ3b (baseline) = 5.29E-7 (1-in-15)

With this changes, the evaluation results are :

Delta LERF = FREQ3b (1-in-15) – FREQ3b (baseline)
Delta LERF = (5.29E-7 – 4.60E-7) = 6.90E-8

4.4 Precautions and Limitations

Use of this report by other organizations should be with the full knowledge of the PSA Unit.

5.0 CONCLUSIONS

Table 14 summarizes the results of this calculation.

TABLE 14 – SUMMARY OF RISK IMPACT				
	3-IN-10	1-IN-10	1-IN-12.1	1-IN-15
Total Integrated Risk (Person-Rem/yr)	90.687	90.697	90.699	90.700
Type A Testing Risk (Person-Rem/yr)	0.092	0.101	0.103	0.105
% Total Risk (Type A/Total)	0.101%	0.111%	0.114%	0.116%
Type A LERF (Class 3b) (per year)	8.99E-7	9.89E-7	1.01E-6	1.03E-6
CHANGES DUE TO EXTENSION FROM 12.1 YEARS (CURRENT)				
Δ Risk from current (Person-rem/yr)				0.001
% Increase from current (Δ Risk / Total Risk)				0.001%
Δ LERF from current (per year)				2.00E-8
Δ CCFP from current				0.089%
CHANGES DUE TO EXTENSION FROM 10 YEARS				
Δ Risk from 1-in-10 (Person-rem/yr)				0.003
% Increase from 1-in-10 (Δ Risk / Total Risk)				.003%
Δ LERF from 1-in-10 (per year)				4.10E-8
Δ CCFP from 1-in10				0.177%
CHANGES DUE TO EXTENSION FROM 3 YEARS (BASELINE)				
Δ Risk from baseline (Person-rem/yr)				0.013
% Increase from baseline (Δ Risk / Total Risk)				0.014%
Δ LERF from baseline (per year)				1.31E-7
Δ CCFP from baseline				1.080%

The risk impact of the proposed change to the ILRT test interval has been shown to be very minor, by a number of measures. Reg. Guide 1.174 provides guidance for determining the risk impact of plant specific changes. It defines very small changes in risk as those resulting in increases in core damage frequency (CDF) below 1E-6/yr and increase in LERF below 1E-7/yr. Since the ILRT does not impact CDF, the relevant metric is LERF.

Referring to the information in Table 14, the change in LERF due to extending the test interval from the current once-per-12.1-year basis to a once-per-15-year basis is 1.03E-6 – 1.01E-6 or 2.0E-8/yr. This change is very small. Comparing the change in LERF due to extending the test interval from the once-per-10-year basis to a once-per-15-year basis is 1.03E-6 – 9.89E-7 or 4.1E-8/yr. This change is very small. Comparing the proposed once-per-15-year basis to a 3-in-10-year basis results in a change in LERF of 1.31E-7/yr, which is considered small.

Note that this result was obtained using a method which contained substantial conservatisms. For example, the potential for a release due to the proposed change in test interval was estimated by multiplying the postulated probability of undetected containment failure by the total core damage frequency. However, some core damage accidents and releases will by definition occur late and can never contribute to LERF. Therefore the CDF value used to calculate the risk significance of the postulated undetected containment failure could be adjusted downward to account for this. Suitable corrections for accidents which by definition must result in LERF and for accidents which can never result in LERF result in a significantly lower applicable CDF value. As a result, a change in LERF which is "very small" will be obtained, even for the comparison between a once-per-15-year test basis and a 3-per-10-year test basis.

6.0 CROSS DISCIPLINE IMPACT

This calculation has no impact on any design documents outside of the PSA Unit of NFM&SA. Therefore no additional review is required.

7.0 LICENSING DOCUMENT/DESIGN BASIS IMPACT

This calculation provides input to a proposed licensing request to extend the ILRT interval.

8.0 PLANT DOCUMENT IMPACT

This calculation does not change any existing plant document.

9.0 SCOPE OF REVIEW

The following is the suggested minimum scope for this calculation:

Lead Reviewer: complete the EGR-NGGC-0003 Record of Lead Review (Engineering Review) and include in Attachment 2.