SYSTEM #	N/A
CALC. SUB-TYPE	N/A
PRIORITY CODE	4
QUALITY CLASS	E

NUCLEAR GENERATION GROUP

HNP-F/PSA-0066

(CALCULATION #)

EVALUATION OF RISK SIGNIFICANCE OF ILRT EXTENSION

(Title including structures, systems, components)

☐ BNP UNIT ____

☐ CR3 ☒ HNP ☐ RNP ☐ NES ☐ ALL

APPROVAL ☒ Electronically Approved

REV	PREPARED BY	REVIEWED BY	SUPERVISOR
1	Signature Original signed by Steven L. Mabe	Signature Original signed by Bruce A. Morgen	Signature Original signed by Robert I. Rishel
	Name Steven L. Mabe	Name Bruce A. Morgen	Name Robert I. Rishel
	Date 11-02-2005	Date 11-02-2005	Date 11-02-2005

(For Vendor Calculations)

Vendor________________________ Vendor Document No. ______________

Owner's Review By ______________________ Date ______________

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REVISION SUMMARY

Rev. #	Revision Summary (list of ECs incorporated)
0	Revision 0 evaluates the risk significance of extending the ILRT test interval at the Harris Nuclear Plant (HNP) to 15 years, using the method developed for the Crystal River 3 (CR3) ILRT extension. This calculation also evaluates the risk due to postulated concealed containment liner corrosion at HNP, using a method obtained from a relevant Calvert Cliffs RAI response.
1	Revision 1 incorporates a revised response for evaluating the risk significance of extending the ILRT test interval at the Harris Nuclear Plant to 15 years. This revision is based on revised methodology (referred to as the NEI Interim Guidance, November 2001) and provides a re-assessment of the risk impacts of the requested change including internal fire and seismic impacts. This calculation also evaluates the risk due to postulated concealed containment liner corrosion at HNP, using a method obtained from a relevant Calvert Cliffs RAI response.

DOCUMENT INDEXING TABLE

Document Type (e.g. CALC, DWG, TAG, PROCEDURE, SOFTWARE)	ID Number (e.g., Calc No., Dwg. No., Equip. Tag No., Procedure No., Software name and version)	Function (i.e. IN for design inputs or references; OUT for affected documents)	Relationship to Calc. (e.g. design input, assumption basis, reference, document affected by results)	Action (specify if Doc. Services or Config. Mgt. to Add, Deleted or Retain) (e.g., CM Add, DS Delete)
CALC	HNP-F/PSA-0001	IN	Used in the calculation	DS Retain
LICENSING	HO-930142	IN	Individual Plant Examination (IPE) Submittal, August 1993.	DS Retain
CALC	HNP-F/PSA-0067	IN	Used in the calculation	DS Retain

RECORD OF LEAD REVIEW

Design HNP-F/PSA-0066 **Revision** 1
RISK SIGNIFICANCE OF ILRT EXTENSION

The signature below of the Lead Reviewer records that:

- the review indicated below has been performed by the Lead Reviewer;
- appropriate reviews were performed and errors/deficiencies (for all reviews performed) have been resolved and these records are included in the design package;
- the review was performed in accordance with EGR-NGGC-0003.

☐ **Design Verification Review** ☒ **Engineering Review** ☐ **Owner's Review**
- ☐ Design Review
- ☐ Alternate Calculation
- ☐ Qualification Testing

☐ **Special Engineering Review**__________

☐ YES ☒ N/A **Other Records are attached.**

Bruce A. Morgen	PSA	11-02-05
Lead Reviewer	**Discipline**	**Date**

Item No.	Deficiency	Resolution
	None.	

FORM EGR-NGGC-0003-2-10
This form is a QA Record when completed and included with a completed design package. Owner's Reviews may be processed as stand alone QA records when Owner's Review is completed.

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Purpose

This calculation evaluates the risk significance of extending the ILRT test interval at the Harris Nuclear Plant (HNP) from the current once-per-ten-years to once-per-fifteen-years, using revised methodology (referred to as the NEI Interim Guidance, November 2001) and provides a re-assessment of the risk impacts of the requested change including internal fire and seismic impacts. This calculation also evaluates the risk due to postulated concealed containment liner corrosion at HNP, using a method obtained from a relevant Calvert Cliffs RAI response. A sensitivity evaluation to certain conservative assumptions is also provided.

References

1. HNP-93-835, "Shearon Harris Nuclear Power Plant Docket No. 50-400/License No. NFP-63, Submittal of Individual Plant Examination (IPE)", August 20, 1993.
2. HNP-F/PSA-0001, "HNP Probabilistic Safety Assessment Model", Revision 5.
3. HNP-F/PSA-0067, "Estimate of 50 Mile Population Dose from Design Basis Containment Leakage Following a Core Melt Accident", Revision 0.
4. RSC 04-03, "Evaluation of Risk Significance of ILRT Extension", Revision 0, April 2004.
5. Constellation Nuclear, Calvert Cliffs Nuclear Power Plant, Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension, 3/27/02. (Provided as Attachment 4).
6. EST-210, "Periodic Containment Integrated Leak Rate Testing (Type A Test)", Revision 11.
7. HNP-F/PSA-0059, "HNP PRA – Appendix L – Summary Document", Revision 2.
8. (Drawing CAR-216)8-G-0228, "CONTAINMENT BLDG - CONTAINMENT LINER-SH 1 UNIT 1", Revision 13.
9. Letter, Jan F. Lucas to USNRC, "Supplement to Amendment Request Regarding One-Time Extension of Containment Type A Test Interval", Serial: RNP-RA/03-0121, October 13, 2003.
10. RSC 05-08, "Evaluation of Risk Significance of ILRT Extension Based on the NEI Approach", Revision 0, October 2005.

Body of Calculation

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.

The inputs to and assumptions for the ILRT evaluation are documented in the Attachment 1 vendor report. The inputs to and assumptions for the evaluation of concealed containment liner corrosion are documented in Attachment 3.

Progress Energy acceptance of the Attachment 1 vendor report is documented by the enclosed Owners Review, shown as Attachment 2.

Conclusions

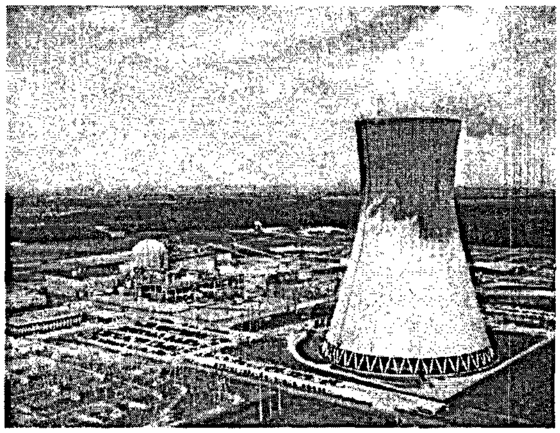
The risk impact of the proposed extension of the ILRT test interval as documented herein is very small. Reg. Guide 1.174 provides guidance in determining the risk impact of specific plant changes. It defines very small changes in risk as those resulting in increases in core damage frequency (CDF) below 1E-6/yr and increases in Large Early Release Frequency (LERF) below 1E-7/yr. Since the ILRT does not impact CDF, the relevant metric is LERF.

Referring to the information in Table A.1 of Attachment 1, the increase in LERF due to extending the Type A ILRT test interval, including internal fire and seismic impacts, from the current once-per-10 year basis to once-per-fifteen-years is 1.82E-8/yr. The increase in LERF resulting from a change in the test interval from a three-per-ten-years to a once-per-fifteen years is 4.37E-8/yr. Both these results are well below the Reg. Guide 1.174 guidance for a very small change in risk value of 1.0E-7/yr. Referring to the information in Table 1 of Attachment 3, the increase in LERF due to potential concealed corrosion is 3.11E-9/yr. This change is very small. The total combined increase in LERF from a three-per-ten-years to a once-per-fifteen years is 4.68E-8/yr and is very small.

The sensitivity evaluation documented herein (using a refined definition for sequences contributing to LERF due to the ILRT interval) demonstrates that the above risk impact contains a substantial amount of conservatism.

Progress Energy RSC 05-08

Harris Nuclear Plant
Probabilistic Safety Assessment



Evaluation of Risk Significance of ILRT Extension Based on the NEI Approach

Revision 0

October 2005

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RSC

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Risk and Reliability Engineering

Document Revision History

Document Revision	RSC Principle Analyst/Project Manager	RSC Internal Reviewer/Date Review Complete (initials/date)	RSC QC and Standards Reviewer/Date Review Complete (initials/date)	RSC Approval for Client Release/Date of Approval (initials/date)
Original Issue	R. Summitt	JM 11-1-05	NA	RS 11-1-05
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Main Report:
Evaluation of Risk Significance of ILRT Extension Based on the NEI Approach

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1.0 PURPOSE

The purpose of this report is to provide an alternative estimation of the change in risk associated with extending the Type A integrated leak rate test interval beyond the current 10 years required by 10 CFR 50, Appendix J, Option B at the Harris Nuclear Plant (HNP). Specifically, this report utilizes the methodology identified by the Nuclear Energy Institute (NEI)[1].

A completed assessment of the proposed change is documented in Reference 2 and serves as a basis for this document. The evaluation found in Reference 2 is consistent with similar assessments performed for the San Onofre Nuclear Generation Station[3] (SONGS), the Comanche Peak plant[4], the Indian Point 3 (IP3) plant[5, 6] and for the Crystal River 3 (CR3) plant[7].

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 integrated leak rate test (ILRT) interval for the HNP. The proposed change would impact testing associated with the current surveillance test for Type A leakage, procedure EST-210[8]. No change to Type B or Type C testing is proposed at this time.

This analysis utilizes the guidelines set forth in NEI 94-01[9], the methodology used in Electric Power Research Institute (EPRI) TR-104285[10] and NUREG-1493[11]. The NEI guidance also considers the submittals generated by other utilities. The assessment contained in this document utilizes the method set forth and metrics presented in Reference 1 supported by the metrics identified in Reference 9. The regulatory guidance on the use of probabilistic safety assessment (PSA) findings in support of a licensee request to a plant's licensing basis, RG 1.174[12], is also utilized.

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 HNP.
- 15 years – Proposed extended test interval, similar to prior industry requests.

The analysis utilizes the HNP PSA results utilized in Reference 2 in order to provide a consistent analysis and is based on information provided in References 13 and 14.

The release category and person-rem information is based on the analysis provided in Appendix A of Reference 2.

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)	110.261	110.323	110.367
Type A testing risk (person-rem/yr)	0.029	0.096	0.144
% total risk (Type A / total)	0.026%	0.087%	0.131%
Type A LERF (Class 3b) (per year)	6.22E-9	2.05E-8	3.08E-8
Changes due to extension from 10 years (current)			
Δ Risk from current (Person-rem/yr)			4.42E-2
% Increase from current (Δ Risk / Total Risk)			0.040%
Δ LERF from current (per year)			1.03E-8
Δ CCFP from current			0.455%
Changes due to extension from 3 years (baseline)			
Δ Risk from baseline (Person-rem/yr)			1.06E-1
% Increase from baseline (Δ Risk / Total Risk)			0.096%
Δ LERF from baseline (per year)			2.46E-8
Δ CCFP from baseline			1.093%

The results are discussed below:

- The person-rem/year increase in risk contribution from extending the ILRT test frequency from the current once-per-ten-year interval to once-per-fifteen years is 0.044 person-rem/year.

- The risk increase in LERF from extending the ILRT test frequency from the current once-per-10-year interval to once-per-15 years is 1.03E-8/yr.

- The change in conditional containment failure probability (CCFP) from the current once-per-10-year interval to once-per-15 years is 0.455%.

- 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 0.040%. Also, the change in Type A test frequency from the original three-per-ten-years to once-per-fifteen-years increases the risk only 0.096%. 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 a once-per-ten-years to a once-per-fifteen-years is 1.03E-8/yr. 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 and the results support this determination. In addition, the change in LERF resulting from a change in the Type A ILRT test interval from a three-per-ten-years to a once-per-fifteen-years is 2.46E-8/yr, is also below the guidance target value. The NEI approach is considered to yield conservative results and there is large uncertainty associated with the seismic and fire events. Therefore, this result is believed to be acceptable when these factors are considered.

- R.G. 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 0.455% for the proposed change and 1.093% 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.

In reviewing these results the HNP analysis demonstrates that the change in plant risk is small as a result of this proposed extension of ILRT testing. The change in LERF defined in the analysis for both the baseline and the current cases is within the acceptance criterion.

2.0 DESIGN INPUTS

The HNP PSA is intended to provide "best estimate" results that can be used as an input when making risk informed decisions. The PSA provides the most recent results for the HNP PSA. The inputs for this calculation come from the information documented in the HNP model of record (MOR) and the HNP individual plant examination (IPE) (References 13 and 14). The HNP plant damage states are summarized in Table 2. Since all core damage bin (CDB) state 16 contributions relate to bypass sequences they are grouped in the table.

Table 2
HNP Plant Damage States

Plant Damage State	Representative Sequence	Frequency (/yr)
10P	Loss of offsite power, RCP seal LOCA at diesel failure time, no RCS makeup.	5.53E-6
1P	Loss of offsite power, failure of AFW after battery depletion, RCP seal LOCA at 1.5 hours. Containment spray and fans are failed.	5.31E-6
2A	Loss of feedwater, failure of AFW, and operator fails to accomplish feed-and-bleed cooling. The containment is isolated and both containment sprays and fans function.	3.12E-6
X16C and all other X16n and B16n contributions	SGTR with failure of safety injection and one SRV associated with the faulted steam generator fails open. This results in a bypass sequence.	3.98E-6
4A	Loss of an ac bus and failure of both AFW and high-pressure recirculation. Containment isolation occurs and the containment sprays and fans are functioning.	1.48E-6
7A	ATWS occurs with overpressure due to insufficient moderator feedback. Containment isolation occurs and the containment sprays and fans are functioning.	1.01E-6
17A	S1 LOCA occurs with failure of recirculation due to loss of RHR pumps. Containment isolation occurs and the containment sprays and fans are functioning.	5.83E-7
13A	S2 LOCA with failure of recirculation. Containment isolation occurs and the containment sprays and fans are functioning.	2.45E-7

Table 2 (continued)
HNP Plant Damage States

Plant Damage State	Representative Sequence	Frequency (/yr)
15A	S2 LOCA occurs without injection. Containment isolation occurs and the containment sprays and fans are functioning.	3.97E-7
3P	Similar to 10P except operators fail to depressurize the RCS.	2.59E-7
8G	Loss of one emergency bus and failure of cooling to the RCPs resulting in a seal LOCA. Failure of all injection and shutdown cooling.	2.41E-7
1A	Similar to 2A with a pressurizer PORV opening late and failing to reclose. Containment isolation occurs and the containment sprays and fans are functioning.	1.72E-7
3A	Transient-induced LOCA with failure of long term cooling due to the failure of depressurization and cooldown of the RCS. A failure of the RHR system may also occur.	1.56E-7
5A	S1 LOCA occurs with failure of recirculation and cooldown. Containment isolation occurs and the containment sprays and fans are functioning.	1.31E-7
4P	S1 LOCA occurs with failure of heat removal and no feed-and-bleed cooling. Isolation of the containment is successful, but the containment sprays and fans are failed.	1.11E-7
12A	Medium LOCA occurs with a failure of recirculation. Containment isolation occurs and the containment sprays and fans are functioning.	1.20E-7
1Q	Similar to 1P except that the isolation failure is small.	1.12E-7
10Q	Similar to 10P except that the isolation failure is small.	1.08E-7
Other PDS Contributors		8.56E-7
Total		2.39E-5

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

The IP3 submittal (Reference 6) 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 7) utilized plant-specific dose estimates based on the predicted level 2 analysis results.

The HNP PSA (References 13 and 14) contains the necessary information to convert the plant damage states to release categories. Using this information, the plant damage states are mapped to one of the fourteen release categories. In addition, the fraction of intact containment cases is determined using the split fraction information contained in References 13 and 14.

Since the HNP PSA contains the necessary release fraction information, an approach similar to the CR3 submittal is utilized that better reflects the specific release conditions for HNP. The HNP PSA (References 13 and 14) release categories are defined by the release fraction of major radionuclides.

These are extrapolated to dose using the approach presented in Reference 2. This approach has been presented in other licensing submittals (References 3 and 4) and is consistent with the method used in the CR3 submittal (Reference 7). The intact containment dose is developed in Reference 2 and is consistent with the approach used in Reference 4. The release category dose information is presented in Table 3 for release categories containing frequency contributions.

Table 3
Release Category Radionuclide Percentage Release and Total Person-Rem

Release Category	Frequency	Noble Gas[1] (%)	Iodine[1] (%)	Cesium[1] (%)	Tellurium[1] (%)	Strontium[1] (%)	Person-Rem
IC-1	1.26E-5	NA[2]	NA	NA	NA	NA	3.26E+3[3]
RC-1	2.77E-7	100	0.162	0.553	0	1.9E-5	1.86E+6
RC-1A	8.55E-9	100	1.8E-4	1.7E-4	3.7E-6	6.5E-8	1.50E+6
RC-1B	6.03E-7	100	1.27	1.89	0	1.7E-5	3.08E+6
RC-1BA	5.01E-8	100	1.8E-4	1.7E-4	7.8E-2	2.1E-2	1.51E+6
RC-2	2.30E-8	100	0.846	1	0	1.1E-5	2.42E+6
RC-2B	6.69E-8	100	6.62	5.5	3.3E-2	3.8E-3	7.56E+6
RC-3	2.26E-7	100	9.2E-2	8.9E-2	1.6E-6	5.3E-5	1.59E+6
RC-3B	3.89E-8	100	0.185	0.186	0	3.3E-5	1.69E+6
RC-5C[4]	3.98E-6	100	77.5	80.8	0	10	8.12E+7
RC-6	1.35E-6	100	0.021	0.063	7.8E-3	2.1E-3	1.54E+6
RC-7	4.65E-6	100	0.21	0.63	7.8E-2	2.1E-2	1.92E+6

1. Contributing fission product groups are discussed in Reference 2.
2. Release fractions not necessary for this calculation.
3. Intact containment representing design basis leakage (developed in Reference 2).
4. Includes other bypass sequences from similar initiating events and sequences (RC-4, RC-4C and RC-5).

Other inputs to this calculation include ILRT test data from NUREG-1493 (Reference 11) and the EPRI report (Reference 10) and are referenced in the body of the calculation.

3.0 ASSUMPTIONS

1. The maximum containment leakage for EPRI Class 1 (Reference 10) 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 (Reference 1) sequences is 10 L_a based on the NEI guidance and previously approved methodology (References 3, 4 and 6).

3. The maximum containment leakage for Class 3b sequences is 35 L_a based on the NEI guidance (Reference 1) and previously approved methodology (References 3, 4 and 6).

4. Class 3b is conservatively categorized LERF based on the NEI guidance and previously approved methodology (References 3, 4 and 6).

5. Containment leakage due to EPRI Classes 4 and 5 are considered negligible based on the NEI guidance and the previously approved methodology (References 3, 4 and 6).

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 HNP PSA release category information in terms of frequency and person-rem estimates to estimate the changes in risk due to increasing the ILRT testing interval. The changes in risk are assessed consistent with the guidance provided in the NEI interim guidance document (Reference 1). This approach considers other similar analyses presented in EPRI TR-104285 (Reference 10) and NUREG-1493 (Reference 11).

The detailed calculations performed to support this report were of a level of mathematical significance necessary to calculate the results recorded. However, the tables and illustrational calculation steps presented may present rounded values to support readability.

4.1 CALCULATIONAL STEPS

The analysis employs the steps provided in Reference 1 and uses risk metrics presented in Reference 12 to evaluate the impact of a proposed change on plant risk. These measures are the change in release frequency, the 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 12 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 analysis process is outlined below:

- Define and quantify the baseline plant damage classes and person-rem estimates.
- Calculate baseline leakage rates and estimate probability to define the analysis baseline.
- Develop baseline population dose (person-rem) and population dose rate (person-rem/yr).
- Modify the Type A leakage estimate to address extension of the Type A test frequency and calculate new population dose rates, LERF and conditional containment failure probability.
- Compare analysis metrics to estimate the impact and significance of the increase related to those metrics.

The first step in the analysis is to define the baseline plant damage classes and person-rem dose measures. Plant damage state information is developed using the HNP PSA (References 13 and 14) results. The plant damage state information and the results of the containment analysis are used to define the representative sequences. The population person-rem dose estimates for the key plant damage classes are based on the application of the method described in Reference 2. The product of the person-rem for the plant damage classes and the frequency of the plant damage state is used to estimate the annual person-rem for the plant damage state. Summing these estimates produces the annual person-rem dose based on the sequences defined in the PSA.

The PSA plant damage state definitions considered isolation failures due to Type B and Type C faults and examine containment challenges occurring after core damage and/or reactor vessel failure. These sequences are grouped into key plant damage classes. Using the plant damage state information, bypass, isolation failures and phenomena-related containment failures are identified. Once identified, the sequence was then classified by release category definitions specified in Reference 10. With this information developed, the PSA baseline inputs are 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 1 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 HNP PSA to develop a baseline model including Type A faults. The release, in terms of person-rem, is developed based on information contained in Reference 2 and is estimated as a leakage increase relative to allowable dose (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 were performed:

1. Map the HNP release categories into the 8 release classes defined by the EPRI report (Reference 10).

2. Calculate the Type A leakage estimate to define the analysis baseline.

3. Calculate the Type A leakage estimate to address the current inspection frequency.

4. Modify the Type A leakage estimates to address extension of the Type A test interval.

5. Calculate increase in risk due to extending Type A inspection intervals.
6. Estimate the change in LERF due to the Type A testing.
7. Estimate the change in conditional containment failure probability due to the Type A testing.

4.2 SUPPORTING CALCULATIONS

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

EPRI Report TR-104285 (Reference 10) defines eight (8) release classes as presented in Table 4.

Table 4
Containment Failure Classifications (from Reference 10)

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

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

Table 5
HNP PSA Release Category Grouping to EPRI Classes (as described in Reference 10)

Class	Description	Release Category	Frequency	Person-Rem	Person-Rem/yr
1	No containment failure	IC-1	1.26E-5	3.26E+3	4.12E-2
2	Large containment isolation failures	None	ε[1]		
3a	Small isolation failures (liner breach)	None	Not addressed		0.00E+0
3b	Large isolation failures (liner breach)	None	Not addressed		0.00E+0
4	Small isolation failures - failure to seal (type B)	None	ε		
5	Small isolation failures - failure to seal (type C)	None	ε		
6	Containment isolation failures (dependent failure, personnel errors)	RC-3, RC-3B	2.65E-7	1.64E+6[2]	4.34E-1
7	Severe accident phenomena induced failure (early and late)	All other RCs	6.94E-6	1.90E+6[2]	1.32E+1
8	Containment bypass	RC-4, RC-4C, RC-5, RC-5C, RC-2, RC-2B	4.07E-6	2.37E+7[2]	9.66E+1
		Total	2.39E-5		1.1023E+02

1. ε represents a probabilistically insignificant value.
2. The value presented represents an average of the contributing release categories.

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

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

Class 3:

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

In order to better assess the range of possible leakage rates, the Class 3 calculation is divided into two classes. Class 3a is defined as a small liner breach and Class 3b is defined as a large liner breach. This division is consistent with the NEI guidance (Reference 1) and the previously approved methodology (References 3, 4 and 6). The calculation of the Class 3a and Class 3b probabilities is presented below.

Calculation of Class 3a Probability

The data presented in NUREG-1493 (Reference 11) 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 1.0 L_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 were identified as errors in test alignments.

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

$$p_{Class3a} = \frac{5}{182} = 0.0275 \qquad \text{(eq. 1)}$$

This probability, however, is based on three tests over a 10-year period and not the one per ten-year frequency currently employed at HNP (Reference 8). The probability (0.0275) must be adjusted to reflect this difference and is adjusted in step 3 of this calculation.

Multiplying the CDF times the probability of a Class 3a leak develops the Class 3a frequency contribution in accordance with guidance provided in Reference 1. This is conservative since part of the CDF already includes LERF sequences. The CDF for HNP is 2.39E-5/yr as presented in Table 5.

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

$$FREQ_{class3a} = PROB_{class3a} \times CDF = 0.0275 \times 2.39E\text{-}5/yr = 6.57E\text{-}7/yr \qquad \text{(eq. 2)}$$

Calculation of Class 3b Probability

To calculate the probability that a liner leak will be large (Class 3b) use was made of the data presented in the calculation of Class 3a. Of the events identified in NUREG-1493 (Reference 11), the largest reported leak rate from those 144 tests was 21 times the allowable leakage rate (L_a). Since 21 L_a does not constitute a large release, no large releases have occurred based on the

144 ILRTs reported in NUREG-1493. The additional data point was also not considered to constitute a large release.

To estimate the failure probability given that no failures have occurred, the guidance provided in Reference 1 suggests the use of a non-informative prior. This approach essentially updates a uniform distribution (no bias) with the available evidence (data) to provide a better estimation of an event.

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

$$p_{Class3b} = \frac{n+\alpha}{N+\beta} = \frac{0+0.5}{182+1} = \frac{0.5}{183} = 0.00273 \qquad \text{(eq. 3)}$$

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

$FREQ_{class3b} = PROB_{class3b}$ x CDF = 0.00273 x 2.39E-5/yr = 6.54E-8/yr (eq. 4)

Class 4:

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

Class 5:

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

Class 6:

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

$FREQ_{class6}$ = 2.65E-7/yr (eq. 5)

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:

$FREQ_{class1} = FREQ_{class1} - (FREQ_{class3a} + FREQ_{class3b})$ (eq. 6)

$FREQ_{class1} = 1.26E\text{-}5/yr - (6.57E\text{-}7/yr + 6.54E\text{-}8/yr) = 1.19E\text{-}5/yr$

Class 2:

The HNP PSA did not identify any contribution to this group above the quantification truncation.

Class 7:

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

$FREQ_{class7} = 6.94E\text{-}6/yr$ (eq. 7)

Class 8:

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

$FREQ_{class8} = 4.07E\text{-}6/yr$ (eq. 8)

Table 6 summarizes the above information by the EPRI defined classes. This table also presents dose exposures calculated using the methodology described in Reference 2. For Class 1, 3a and 3b, the person-rem is developed based on the design basis assessment of the intact containment as developed in Reference 2.

The Class 3a and 3b doses are represented as 10 L_a and 35 L_a respectively. Table 6 also presents the person-rem frequency data determined by multiplying the failure class frequency by the corresponding exposure.

Table 6
Baseline Risk Profile

Class	Description	Frequency (/yr)	Person-rem (calculated)[1]	Person-rem (from L_a factors)	Person-rem (/yr)
1	No containment failure	1.19E-5		3.26E+3[2]	3.89E-2
2	Large containment isolation failures	ε[3]			
3a	Small isolation failures (liner breach)	6.57E-7		3.26E+4[4]	2.14E-2
3b	Large isolation failures (liner breach)	6.54E-8		1.14E+5[5]	7.46E-3
4	Small isolation failures - failure to seal (type B)	ε			
5	Small isolation failures - failure to seal (type C)	ε			
6	Containment isolation failures (dependent failure, personnel errors)	2.65E-7	1.64E+6[6]		4.34E-1
7	Severe accident phenomena induced failure (early and late)	6.94E-6	1.90E+6[6]		1.32E+1
8	Containment bypass	4.07E-6	2.37E+7[6]		9.66E+1
	Total	2.39E-5			1.1026E+2

1. From Table 3 using the method presented in Reference 2.
2. 1 times L_a dose value calculated in Reference 2.
3. ε represents a probabilistically insignificant value.
4. 10 times L_a.
5. 35 times L_a.
6. The value presented represents an average of the contributing release categories.

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

$\%Risk_{BASE} = [(Class3a_{BASE} + Class3b_{BASE}) / Total_{BASE}] \times 100$ (eq. 9)

Where:

$Class3a_{BASE}$ = Class 3a person-rem/year =2.14E-2 person-rem/year

$Class3b_{BASE}$ = Class 3b person-rem/year = 7.46E-3 person-rem/year

$Total_{BASE}$ = total person-rem year for baseline interval = 1.1026E+2 person-rem/year (Table 6)

$\%Risk_{BASE}$ = [(2.14E-2 + 7.46E-3) / 1.1026E +2] x 100 = **0.026%** (eq. 10)

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 9) 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 1.0 L_a).

According to References 1 and 11, extending the Type A ILRT interval from 3-in-10 years to 1-in-10 years will increase the average time that a leak detectable only by an ILRT goes undetected from 18 to 60 months. Multiplying the testing interval by 0.5 and multiplying by 12 to convert from "years" to "months" calculates the average time for an undetected condition to exist.

The increase for a 10-yr ILRT interval is the ratio of the average time for a failure to detect for the increased ILRT test interval (from 18 months to 60 months) multiplied by the existing Class 3a probability as shown in Equation 11.

$$p_{Class3a}(10y) = 0.0275 \times \left(\frac{60}{18}\right) = 0.0916 \qquad \text{(eq. 11)}$$

A similar calculation is performed for the Class 3b probability as presented in Equation 12.

$$p_{Class3b}(10y) = 0.00273 \times \left(\frac{60}{18}\right) = 0.0091 \qquad \text{(eq. 12)}$$

<u>Risk Impact due to 10-year Test Interval</u>

Based on the previously approved methodology (References 3, 4 and 6) and the NEI guidance (Reference 1), the increased probability of not detecting excessive leakage due to Type A tests directly impacts the frequency of the Class 3 sequences.

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

Table 7
Risk Profile for Once in Ten Year Testing

Class	Description	Frequency (/yr)	Person-rem [2]	Person-rem (/yr)
1	No Containment Failure[1]	1.02E-5[1]	3.26E+3	3.34E-2
2	Large Containment Isolation Failures	ε[3]		
3a	Small Isolation Failures (Liner breach)	2.19E-6	3.26E+4	7.14E-2
3b	Large Isolation Failures (Liner breach)	2.18E-7	1.14E+5	2.49E-2
4	Small isolation failures - failure to seal (type B)	ε		
5	Small isolation failures - failure to seal (type C)	ε		
6	Containment Isolation Failures (dependent failure, personnel errors)	2.65E-7	1.64E+6[4]	4.34E-1
7	Severe Accident Phenomena Induce Failure (Early and Late)	6.94E-6	1.90E+6[4]	1.32E+1
8	Containment Bypass	4.07E-6	2.37E+7[4]	9.66E+1
	Total	2.39E-6		1.1032E+2

1. The PSA frequency of Class 1 has been reduced by the frequency of Class 3a and Class 3b in order to preserve total CDF.
2. From Table 6.
3. ε represents a probabilistically insignificant value.
4. The value presented represents a frequency weighted average of the contributing release categories.

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

$\%Risk_{10} = [(Class3a_{10} + Class3b_{10}) / Total_{10}] \times 100$ (eq. 13)

Where:

$Class3a_{10}$ = Class 3a person-rem/year = 7.14E-2 person-rem/year

$Class3b_{10}$ = Class 3b person-rem/year = 2.49E-2 person-rem/year

$Total_{10}$ = total person-rem year for current 10-year interval = 1.1032E+2 person-rem/year (Table 7)

$\%Risk_{10} = [(7.14E\text{-}2 + 2.49E\text{-}2) / 1.1032E{+}2] \times 100 =$ **0.087%** (eq. 14)

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

$\Delta\%Risk_{10} = [(Total_{10} - Total_{BASE}) / Total_{BASE}] \times 100.0$ (eq. 15)

Where:

$Total_{BASE}$ = total person-rem/year for baseline interval = 1.1026E+2 person-rem/year (Table 6)

$Total_{10}$ = total person-rem/year for 10-year interval = 1.1032E+2 person-rem/year (Table 7)

$\Delta\%Risk_{10} = [(1.1032E{+}2 - 1.1026E{+}2) / 1.1026E{+}2] \times 100.0 =$ **0.06%** (eq. 16)

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

If the test interval is extended to 1 per 15 years, the average time that a leak detectable only by an ILRT test goes undetected increases to 90 months (0.5 x 15 x 12). For a 15-yr-test interval, the result is the ratio (90/18) of the exposure times as was the case for the 10 year case. Thus, increasing the ILRT test interval from 3 years to 15 years results in a proportional increase in the overall probability of leakage.

The approach for developing the risk contribution for a 15-year interval is the same as that for the 10-year interval. The increase for a 15-yr ILRT interval is the ratio of the average time for a failure to detect for the increased ILRT test interval (from 18 months to 90 months) multiplied by the existing Class 3a probability as shown in Equation 17.

$$p_{Class3a}(15y) = 0.0275 \times \left(\frac{90}{18}\right) = 0.1375 \qquad \text{(eq. 17)}$$

A similar calculation is performed for the Class 3b probability as presented in Equation 18.

$$p_{Class3b}(15y) = 0.00273 \times \left(\frac{90}{18}\right) = 0.0137 \qquad \text{(eq. 18)}$$

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

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

Table 8
Risk Profile for Once in Fifteen Year Testing

Class	Description	Frequency (/yr)	Person-rem [2]	Person-rem (/yr)
1	No Containment Failure[1]	9.03E-6[1]	3.26E+3	2.95E-2
2	Large Containment Isolation Failures	ε[3]		
3a	Small Isolation Failures (Liner breach)	3.29E-6	3.26E+4	1.07E-1
3b	Large Isolation Failures (Liner breach)	3.27E-7	1.14E+5	3.73E-2
4	Small isolation failures - failure to seal (type B)	ε		
5	Small isolation failures - failure to seal (type C)	ε		
6	Containment Isolation Failures (dependent failure, personnel errors)	2.65E-7	1.64E+6[4]	4.34E-1
7	Severe Accident Phenomena Induce Failure (Early and Late)	6.94E-6	1.90E+6[4]	1.32E+1
8	Containment Bypass	4.07E-6	2.37E+7[4]	9.66E+1
	Total	2.39E-5		1.1037E+2

1. The PSA frequency of Class 1 has been reduced by the frequency of Class 3a and Class 3b in order to preserve total CDF.
2. From Table 6.
3. ε represents a probabilistically insignificant value.
4. The value presented represents a frequency weighted average of the contributing release categories.

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:

$$\%Risk_{15} = [(Class3a_{15} + Class3b_{15}) / Total_{15}] \times 100 \quad \text{(eq. 19)}$$

Where:

$Class3a_{15}$ = Class 3a person-rem/year = 1.07E-1 person-rem/year

$Class3b_{15}$ = Class 3b person-rem/year = 3.73E-2 person-rem/year

$Total_{15}$ = total person-rem year for 15-year interval = 1.1037E+2 person-rem/year (Table 8)

$$\%Risk_{15} = [(1.07E\text{-}1 + 3.73E\text{-}2) / 1.1037E\text{+}2] \times 100 = \mathbf{0.131\%} \quad \text{(eq. 20)}$$

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

$$\Delta\%Risk_{15} = [(Total_{15} - Total_{BASE}) / Total_{BASE}] \times 100.0 \quad \text{(eq. 21)}$$

Where:

$Total_{BASE}$ = total person-rem/year for baseline interval = 1.1026E+2 person-rem/year (Table 6)

$Total_{15}$ = total person-rem/year for 15-year interval = 1.1037E+2 person-rem/year (Table 8)

$$\Delta\%Risk_{15} = [(1.1037E+2 - 1.1026E+2) / 1.1026E+2] \times 100.0 = \mathbf{0.096\%} \quad \text{(eq. 22)}$$

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

Based on the guidance in Reference 1, the percent increase in the total integrated plant risk for these accident sequences is computed as follows:

$$\%Total_{10\text{-}15} = [(Total_{15} - Total_{10}) / Total_{10}] \times 100 \quad \text{(eq. 23)}$$

Where:

$Total_{10}$ = total person-rem/year for 10-year interval = 1.1032E+2 person-rem/year (Table 7)

$Total_{15}$ = total person-rem/year for 15-year interval = 1.1037E+2 person-rem/year (Table 8)

$$\%\ Total_{10\text{-}15} = [(1.1037E+2 - 1.1032E+2) / 1.1032E+2] \times 100 = \mathbf{0.040\%} \quad \text{(eq. 24)}$$

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

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 1, 3, 4 and 6, the Class 3a dose is assumed to be 10 times the allowable intact containment leakage, L_a (or 3.26E+4 person-rem) and the Class 3b dose is assumed to be 35 times L_a (or 1.14E+5 person-rem). The dose equivalent for allowable leakage (L_a) is developed in Reference 2. This compares to a historical observed average of twice L_a. Therefore, the estimate is somewhat conservative.

Based on the NEI guidance (Reference 1) and the previously approved methodology (References 3, 4 and 6), 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 2 L_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 HNP PSA (References 13 and 14) 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 HNP, and the change in LERF can be determined by the differences. Reference 1 identifies that Class 3b is considered to be the contributor to LERF. Table 9 summarizes the results of the LERF evaluation that Class 3b is indicative of a LERF sequence.

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

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
Class 3b (Type A LERF)	6.54E-8	2.18E-7	3.27E-7
ΔLERF (3 year baseline)		1.52E-7	2.61E-7
ΔLERF (10 year baseline)			1.09E-7

Reg. Guide 1.174 (Reference 12) provides guidance for determining the risk impact of plant-specific changes to the licensing basis. The Reg. Guide 1.174 defines very small changes in risk as resulting in increases of core damage frequency below 1E-6/yr and increases in LERF below 1E-7/yr. Since the ILRT does not impact CDF, the relevant metric is LERF. Calculating the increase in LERF requires determining the impact of the ILRT interval on the leakage probability.

Increasing the ILRT interval from the currently acceptable 10 year period to a new period of 15 years increases the LERF contribution of 1.09E-7/yr. This value does not meet the guidance in Reg. Guide 1.174 defining very small changes in LERF. The LERF increase is measured from the original 3-in-10-year interval to the 15-year interval is 2.61E-7/yr, which also exceeds the criterion presented in Regulatory Guide 1.174.

Reference 15 indicates that plants with a CDF in excess of 1.0E-5/yr may have difficulty demonstrating a change in LERF less than 1.0E-7/yr. It further states that the analysis as embodied in the NEI approach is conservative and provides additional guidance with respect to refining the initial analysis.

The change in LERF for both cases exceeds the limit and some refinement is necessary. The increase is explicitly tied to the Class 3b contribution which is generated by multiplying the total CDF by the defined split fraction (0.0027).

The estimated split fraction is conservative since some sequence frequencies comprising the total CDF already account for other LERF sequences which may occur due to interfacing system LOCA events or steam generator tube ruptures. The removal of this conservatism can have a significant impact on the split fraction.

The first refinement centers on the conservatism centers on using the whole CDF. Sequences which result in LERF contributions are not influenced (change in outcome) by the potential for Type A leakage and can be excluded from the calculation of Class 3b leakage.

The guidance in Reference 15 indicates that LERF sequences and those sequences where, based on Level 2 analysis, any release would be scrubbed should not be considered with regard to the Type 3b frequency.

The HNP Level 2 containment safeguards event tree (CSET) and the containment event tree (CET) models (References 13 and 14) were utilized to identify the characteristics necessary for determining the status of these aspects of the analysis. The existing containment event tree model provides adequate detail to define the endpoints associated with LERF. Additionally, it defines cases where the core debris would be flooded and/or where containment spray would be expected to function for a prolonged period of time such that the releases through any Type 3b leak would be scrubbed.

A review of the CSET specifically defines that CSET end state with a preceding "X" signifies a early release that is considered to be large and the preceding "B" indicates bypass to some degree. In addition, the 7C PDS is related to the interfacing systems LOCA which can also be classified as early. A final contribution is due to early containment failures that are associated with release categories 2 and 2B.

Selecting these attributes the LERF sequences are defined. Table 10 lists the contributing PDS results that contribute to LERF. From Table 10 the LERF contribution is 4.07E-6/yr.

Table 10
LERF Contributors by PDS

PDS	Frequency (/yr)	Cumulative Frequency (/yr)	Description of PDS Sequence
X and B16 series	3.98E-6[1]	3.98E-06	SGTR with failure of SI and a SRV on the faulted steam generator fails open or SGTR, failure of SI and shutdown cooling with a cycling SRV on the faulted steam generator
Release categories 2 and 2B	8.99E-8[1]	4.07E-6	Early containment failure
7C	4.91E-9	**4.07E-6**	Interfacing systems LOCA occurs in the RHR system

1. Represents a sum of several PDS results with similar description.

The second aspect defined in Reference 15 addresses the magnitude of the source term expected to be available for release during the accident sequence. If the debris escapes the reactor vessel but remains essentially covered with water (either due to large pools or continual containment sprays) the source term will be greatly reduced and a large source term would not be expected.

Therefore, if the accident sequence involves containment spray operation or coverage of the debris with large pools of water, the source term is not considered sufficient to support a LERF release and these contributions can be excluded.

Reference 15 suggests that one criterion is the status of the containment sprays. This criterion assumes that containment spray must be available for both injection and recirculation to ensure scrubbing of released radionuclides that are released during initial reactor vessel failure and subsequent releases from radionuclides released from the RCS after vessel failure. The end state (plant damage state) must also be an intact containment state since the unisolated containment states are already considered by the LERF fraction.

A review of the CSET defines several states where the containment spray system will function for both injection and recirculation. CSET endstates "A", "G" and "D" all represent accident sequences where the containment sprays are functioning for some period of time.

Endstates "A" and "G" are associated with continuous operation of the containment sprays in both injection and recirculation. Endstate "D" is representative of a sequence where containment spray fails in recirculation. The contribution to this sequence predominantly involves the loss of heat removal to the sprays and a failure of pressure control. The lack of header cooling would not impact the scrubbing function of the containment sprays. The containment spray would continue to function based on analysis developed for the PSA (Reference 13) and the system could maintain the scrubbing function. Containment pressure control could be maintained by the containment fan coolers. Therefore, the assumption is made to exclude PDS "D" from the calculation of Class 3b (LERF associated with the ILRT extension) since containment spray would be available to scrub any release.

Table 11 lists the summed frequency contribution for the PDSs that are from one of the three states identified above.

Table 11
Contributing PDSs with CSET States A, D or G

Endstate	Frequency (/yr)	CDF Contribution
A	7.66E-6	93.3%
D	1.08E-7	1.3%
G	4.45E-7	5.4%
Total	8.21E-6	100%

The PDS contributions in Table 11 represent 34% of the total CDF. The major contribution involves state "A" and the operation of the containment sprays is assured. The encompassed system failures would not result in a release of radionuclides sufficient to be classified as LERF.

From the solution presented above, a table of results is obtained and reproduced in Table 12.

Table 12
Source Term Outcomes

Source Term Outcome	Frequency (/yr)	Description
Source term refinement outcome 1	4.07E-6	Predefined LERF sequence (CSET X or B or early failure)
Source term refinement outcome 2	8.21E-6	Non-LERF sequence with the containment sprays functioning (endstates A, D or G)
Source term refinement outcome 3	1.16E-5	Non-LERF sequence without the containment sprays functioning (all other end states)

Only outcome 3 contributes to the potential for a Type A LERF. This value is then utilized to calculate the LERF contribution from Class 3b frequency by the following equation:

$FREQ_{class3b} = PROB_{class3b}$ x Adjusted CDF = 0.00273 x 1.16E-5/yr = 3.18E-8/yr (eq. 25)

This can then be extrapolated using the methods presented earlier to determine the 10-year and 15-year contributions and to generate adjusted LERF values as presented in Table 13.

Table 13
Class 3b Contributions Using Adjusted CDF

Test Interval	Frequency (/yr)	Delta Frequency from Prior Period (/yr)
Baseline	3.18E-8	
10-year (current)	1.05E-7	7.32E-8
15-year	1.57E-7	5.25E-8

Summing the last column provides the total increase from the baseline (3 year) to the proposed (15 year) interval (1.26E-7/yr). This increase is still slightly higher than the limit for a small change in risk.

The controlling plant damage states are 10P (5.53E-6/yr) and 1P (5.31E-6/yr). These sequences both involve the failure of the ac power system (both onsite and offsite) with a reactor coolant pump seal LOCA and no injection. PDS 1P also includes a loss of secondary-side heat removal.

The PDS definition neglects the potential for ac power restoration prior to significant fuel damage. This is addressed in the CET modeling for invessel recovery (Event IVR).

The HNP CET model includes modeling of the potential for recovery of ac power prior to significant releases following reactor vessel breach. Analysis performed for estimation of the time available for sequences involving a loss of ac power indicates that an incremental factor (time to recover between time "t" and the final failure time) is 0.09 for cases involving no secondary-side heat removal and 0.18 for cases with secondary-side heat removal. The major difference is that the recovery curve for the cases without heat removal is substantially steeper than the case with secondary-side heat removal and so a small increase in time has a more pronounced impact.

These adjustments can be made directly to the PDS results with the cases involving ac power recovery being assumed to have containment sprays available. Table 14 provides a summary of the refinement for these two PDS results.

Table 14
Refinement of PDS 1P and 10P to Address Late Ac Power Recovery

Plant Damage State	Frequency Contribution without Ac Recovery (/yr)	Frequency Contribution with Ac Recovery (/yr)
1P	4.78E-7	4.83E-6
10P	9.95E-7	4.53E-6
Total	1.47E-6	9.37E-6

Using this information an adjustment to the Class 3b frequency can be made. The earlier value of 1.16E-5/yr can be reduced to 2.28E-6/yr. This value is then utilized to calculate the LERF contribution from Class 3b frequency by the following equation:

$$FREQ_{class3b} = PROB_{class3b} \text{ x Adjusted CDF} = 0.00273 \text{ x } 2.28E\text{-}6/yr = 6.22E\text{-}9/yr \qquad \text{(eq. 26)}$$

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

Table 15
Class 3b Contributions Using Adjusted PDS After Ac Recovery

Test Interval	Frequency (/yr)	Delta Frequency from Prior Period (/yr)
Baseline	6.22E-9	
10-year (current)	2.05E-8	1.43E-8
15-year	3.08E-8	1.03E-8

Summing the last column provides the total increase from the baseline (3 year) to the proposed (15 year) interval (2.46E-8/yr). This LERF increase associated with the ILRT extension is substantially below the limit for a small change in risk.

As a sensitivity analysis, the offsite power recovery value was set at 0.5 and the results recalculated. Table 16 presents the results.

Table 16
Class 3b Contributions Using Conservative Ac Power Recovery (0.5)

Test Interval	Frequency (/yr)	Delta Frequency from Prior Period (/yr)
Baseline	1.70E-8	
10-year (current)	5.61E-8	3.91E-8
15-year	8.42E-8	2.81E-8

Summing the contributions yields a value of 6.72E-8/yr which is still below the significance threshold.

Step 7: Calculate the change in Conditional Containment Failure Probability

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

$$CCFP = 1 - \left[\frac{f(ncf)}{CDF}\right] \qquad \text{(eq. 27)}$$

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

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

$$\Delta CCFP_{10-15} = CCFP_{15} - CCFP_{10} \qquad \text{(eq. 28)}$$

Using the data previously developed the change in CCFP from the current testing interval is calculated and presented in Table 17.

Table 17
Impact on Conditional Containment Failure Probability due to Extended Type A Testing Intervals

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
f(ncf) (/yr)	1.258E-5	1.243E-5	1.232E-5
f(ncf)/CDF	0.526	0.520	0.515
CCFP	4.74E-1	4.80E-1	4.85E-1
ΔCCFP (3 year baseline)		0.638%	1.093%
ΔCCFP (10 year baseline)			0.455%

Appendix A:
External Events Sensitivity Study

A.0 EXTERNAL EVENTS SENSITIVITY STUDY

NEI guidance (Reference 15) suggests the need to address external initiating events when estimating the impact of the proposed ILRT extension in cases where additional refinements are made to the analysis. A sensitivity study using data for the plant damage state frequencies including seismic and fire contribution to release frequency is used to address this requirement.

A.1 SUMMARY OF THE ANALYSIS

This section is completed in the same manner as NEI baseline analysis. Information from References 1 through 14 are used in the same manner as NEI baseline analysis presented in the prior sections and the methodology steps outlined in Section 4.1. The section only addresses areas of deviation from the earlier results and includes a summary of the results.

A.2 SUMMARY OF RESULTS/CONCLUSIONS

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

Table A.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)	112.30	112.37	112.42
Type A testing risk (person-rem/yr)	0.032	0.107	0.161
% total risk (Type A / total)	0.029%	0.096%	0.143%
Type A LERF (Class 3b) (per year)	1.11E-8	3.65E-8	5.47E-8
Changes due to extension from 10 years (current)			
Δ risk from current (person-rem/yr)			4.93E-2
% increase from current (Δ risk / total risk)			0.044%
Δ LERF from current (per year)			1.82E-8
Δ CCFP from current			0.455%
Changes due to extension from 3 years (baseline)			
Δ risk from baseline (person-rem/yr)			1.18E-1
% increase from baseline (Δ risk / total risk)			0.105%
Δ LERF from baseline (per year)			4.37E-8
Δ CCFP from baseline			1.093%

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 0.049 person-rem/year.

- The risk increase in LERF from extending the ILRT test frequency from the current once-per-10-year interval to once-per-15 years is 1.82E-8/yr.

- The change in conditional containment failure probability (CCFP) from the current once-per-10-year interval to once-per-15 years is 0.455%.

- 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 0.044%. Also, the change in Type A test frequency from the original three-per-ten-years to once-per-fifteen-years increases the risk only 0.105%. 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 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 a once-per-ten-years to a once-per-fifteen-years is 1.82E-8/yr. 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. The value is below this guidance indicating that the change is not 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 a once-per-fifteen-years is 4.37E-8/yr, and is also below the guidance.

- R.G. 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 0.455% for the proposed change and 1.093% 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.

A.3 DESIGN INPUTS

The inputs for this calculation are similar to the information in the baseline analysis. The only change is that the input information includes not only internal event initiators but estimates for internal fire and seismic sequences. The HNP plant damage states are summarized in Table A.2. As before, all CDB state 16 contributions relate to bypass sequences they are grouped in the table.

Table A.2
HNP Plant Damage States

Plant Damage State	Representative Sequence	Frequency (/yr)
10P	Loss of offsite power, RCP seal LOCA at diesel failure time, no RCS makeup.	6.59E-6
1P	Loss of offsite power, failure of AFW after battery depletion, RCP seal LOCA at 1.5 hours. Containment spray and fans are failed.	5.31E-6
2A	Loss of feedwater, failure of AFW, and operator fails to accomplish feed-and-bleed cooling. The containment is isolated and both containment sprays and fans function.	3.33E-6
X16C and all other X16n and B16n contributions	SGTR with failure of safety injection and one SRV associated with the faulted steam generator fails open. This results in a bypass sequence.	3.98E-6
4A	Loss of an ac bus and failure of both AFW and high-pressure recirculation. Containment isolation occurs and the containment sprays and fans are functioning.	2.75E-6
7A	ATWS occurs with overpressure due to insufficient moderator feedback. Containment isolation occurs and the containment sprays and fans are functioning.	1.01E-6
17A	S1 LOCA occurs with failure of recirculation due to loss of RHR pumps. Containment isolation occurs and the containment sprays and fans are functioning.	5.83E-7
13A	S2 LOCA with failure of recirculation. Containment isolation occurs and the containment sprays and fans are functioning.	2.45E-7

Table A.2 (continued)
HNP Plant Damage States

Plant Damage State	Representative Sequence	Frequency (/yr)
15A	S2 LOCA occurs without injection. Containment isolation occurs and the containment sprays and fans are functioning.	3.97E-7
3P	Similar to 10P except operators fail to depressurize the RCS.	2.59E-7
8G	Loss of one emergency bus and failure of cooling to the RCPs resulting in a seal LOCA. Failure of all injection and shutdown cooling.	2.41E-7
1A	Similar to 2A with a pressurizer PORV opening late and failing to reclose. Containment isolation occurs and the containment sprays and fans are functioning.	1.72E-7
3A	Transient-induced LOCA with failure of long term cooling due to the failure of depressurization and cooldown of the RCS. A failure of the RHR system may also occur.	1.56E-7
5A	S1 LOCA occurs with failure of recirculation and cooldown. Containment isolation occurs and the containment sprays and fans are functioning.	1.31E-7
4P	S1 LOCA occurs with failure of heat removal and no feed-and-bleed cooling. Isolation of the containment is successful, but the containment sprays and fans are failed.	2.06E-7
12A	Medium LOCA occurs with a failure of recirculation. Containment isolation occurs and the containment sprays and fans are functioning.	1.20E-7
1Q	Similar to 1P except that the isolation failure is small.	1.12E-7
10Q	Similar to 10P except that the isolation failure is small.	1.10E-7
Other PDS Contributors		9.91E-7
Total		2.67E-5

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

The IP3 submittal (Reference 6) 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 7) utilized plant-specific dose estimates based on the predicted level 2 analysis results.

The HNP PSA (References 13 and 14) contains the necessary information to convert the plant damage states to release categories. Using this information, the plant damage states are mapped to one of the fourteen release categories. In addition, the fraction of intact containment cases is determined using the split fraction information contained in References 13 and 14.

Since the HNP PSA contains the necessary release fraction information, an approach similar to the CR3 submittal is utilized that better reflects the specific release conditions for HNP. The HNP PSA (References 13 and 14) release categories are defined by the release fraction of major radionuclides.

These are extrapolated to dose using the approach presented in Reference 2. This approach has been presented in other licensing submittals (References 3 and 4) and is consistent with the method used in the CR3 submittal (Reference 7). The intact containment dose is developed in Reference 2 and is consistent with the approach used in Reference 4. The release category dose information is presented in Table A.3 for categories with some frequency component.

Table A.3
Release Category Radionuclide Percentage Release and Total Person-Rem

Release Category	Frequency	Noble Gas[1] (%)	Iodine[1] (%)	Cesium[1] (%)	Tellurium[1] (%)	Strontium[1] (%)	Person-Rem
IC-1	1.44E-5	NA[2]	NA	NA	NA	NA	3.26E+3[3]
RC-1	2.92E-7	100	0.162	0.553	0	1.9E-5	1.86E+6
RC-1A	1.02E-8	100	1.8E-4	1.7E-4	3.7E-6	6.5E-8	1.50E+6
RC-1B	6.94E-7	100	1.27	1.89	0	1.7E-5	3.08E+6
RC-1BA	5.78E-8	100	1.8E-4	1.7E-4	7.8E-2	2.1E-2	1.51E+6
RC-2	2.45E-8	100	0.846	1	0	1.1E-5	2.42E+6
RC-2B	6.80E-8	100	6.62	5.5	3.3E-2	3.8E-3	7.56E+6
RC-3	2.29E-7	100	9.2E-2	8.9E-2	1.6E-6	5.3E-5	1.59E+6
RC-3B	3.89E-8	100	0.185	0.186	0	3.3E-5	1.69E+6
RC-5C[4]	3.98E-6	100	77.5	80.8	0	10	8.12E+7
RC-6	1.43E-6	100	0.021	0.063	7.8E-3	2.1E-3	1.54E+6
RC-7	5.49E-6	100	0.21	0.63	7.8E-2	2.1E-2	1.92E+6

1. Contributing fission product groups are discussed in Reference 2.
2. Release fractions not necessary for this calculation.
3. Intact containment representing design basis leakage (developed in Reference 2).
4. Combined results of RC-5C, RC-4, RC-4C and RC-5. Since all similar and involve bypass can be combined.

Other inputs to this calculation include ILRT test data from NUREG-1493 (Reference 11) and the EPRI report (Reference 10) and are referenced in the body of the calculation.

A.4 CALCULATIONS

Following the methodology presented in Section 4.1, the following calculation steps were performed:

1. Map the HNP release categories into the 8 release classes defined by the EPRI Report (Reference 10).

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.

A.5 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 A.4.

Table A.4
Containment Failure Classifications (from Reference 10)

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

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

Table A.5
PSA Release Category Grouping to EPRI Classes (as described in Reference 7)

Class	Description	Release Category	Frequency	Person-Rem	Person-Rem/yr
1	No containment failure	IC-1	1.44E-5	3.26E+3	4.69E-2
2	Large containment isolation failures	None	ε[1]		
3a	Small isolation failures (liner breach)	None	Not addressed		0.00E+0
3b	Large isolation failures (liner breach)	None	Not addressed		0.00E+0
4	Small isolation failures - failure to seal (type B)	None	ε		
5	Small isolation failures - failure to seal (type C)	None	ε		
6	Containment isolation failures (dependent failure, personnel errors)	RC-3, RC-3B	2.68E-7	1.64E+6[2]	4.40E-1
7	Severe accident phenomena induced failure (early and late)	All other RCs	7.97E-6	1.90E+6[2]	1.52E+1
8	Containment bypass	RC-4, RC-4C, RC-5, RC-5C, RC-2, RC-2B	4.07E-6	2.37E+7[2]	9.66E+1
		Total	2.67E-5		1.1227E+2

1. ε represents a probabilistically insignificant value.

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

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

Class 3:

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

In order to better assess the range of possible leakage rates, the Class 3 calculation is divided into two classes. Class 3a is defined as a small liner breach and Class 3b is defined as a large liner breach. This division is consistent with the NEI guidance (Reference 1) and the previously approved methodology (References 3, 4 and 6). The calculation of the Class 3a and Class 3b probabilities is presented below.

Calculation of Class 3a Probability

The data presented earlier from NUREG-1493 (Reference 11) and data presented in Reference 1 is used to calculate the probability that a liner leak will be small (Class 3a) as done earlier. Using the data a mean estimate for the probability of leakage is determined for Class 3a as shown in Equation 1.

$$p_{Class3a} = \frac{5}{182} = 0.0275 \qquad \text{(eq. 1)}$$

This probability, however, is based on three tests over a 10-year period and not the one per ten-year frequency currently employed at HNP (Reference 7). The probability (0.0275) must be adjusted to reflect this difference and is adjusted in step 3 of this calculation.

Multiplying the CDF times the probability of a Class 3a leak develops the Class 3a frequency contribution in accordance with guidance provided in Reference 2. This is conservative since part of the CDF already includes LERF sequences. The CDF for HNP is 2.67E-5/yr as presented in Table A.5.

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

$$FREQ_{class3a} = PROB_{class3a} \times CDF = 0.0275 \times 2.67E\text{-}5/yr = 7.33E\text{-}7/yr \qquad \text{(eq. 2)}$$

Calculation of Class 3b Probability

To calculate the probability that a liner leak will be large (Class 3b) use was made of the data presented in the calculation of Class 3a. Of the events identified in NUREG-1493 (Reference 11), the largest reported leak rate from those 144 tests was 21 times the allowable leakage rate (L_a). Since 21 L_a does not constitute a large release, no large releases have occurred based on the 144 ILRTs reported in NUREG-1493. The additional data point was also not considered to constitute a large release.

To estimate the failure probability given that no failures have occurred, the guidance provided in Reference 2 suggests the use of a non-informative prior. This approach essentially updates a uniform distribution (no bias) with the available evidence (data) to provide a better estimation of an event. A beta distribution is typically used for the uniform prior with the parameters α=0.5

and β=1. This is then combined with the existing data (no Class 3b events, 182 tests) using Equation 3.

$$p_{Class3b} = \frac{n+\alpha}{N+\beta} = \frac{0+0.5}{182+1} = \frac{0.5}{183} = 0.00273 \qquad \text{(eq. 3)}$$

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

$FREQ_{class3b} = PROB_{class3b}$ x CDF = 0.00273 x 2.67E-5/yr = 7.29E-8/yr (eq. 4)

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 A.5 as Class 6.

$FREQ_{class6}$ = 2.68E-7/yr (eq. 5)

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:

$FREQ_{class1} = FREQ_{class1} - (FREQ_{class3a} + FREQ_{class3b})$ (eq. 6)

$FREQ_{class1}$ = 1.44E-5/yr – (7.33E-7/yr + 7.29E-8/yr) = 1.36E-5/yr

Class 2:

The HNP PSA did not identify any contribution to this group above the quantification truncation.

Class 7:

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

$FREQ_{class7}$ = 7.97E-6/yr (eq. 7)

Class 8:

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

$FREQ_{class8}$ = 4.07E-6/yr (eq. 8)

Table A.6 summarizes the above information by the EPRI defined classes. This table also presents dose exposures calculated using the methodology described in Reference 2. For Class 1, 3a and 3b, the person-rem is developed based on the design basis assessment of the intact containment. The Class 3a and 3b doses are represented as 10 L_a and 35 L_a respectively.

Table A.6 also presents the person-rem frequency data determined by multiplying the failure class frequency by the corresponding exposure.

Table A.6
Baseline Risk Profile

Class	Description	Frequency (/yr)	Person-rem (calculated)[1]	Person-rem (from L_a factors)	Person-rem (/yr)
1	No containment failure	1.36E-5		3.26E+3[2]	4.43E-2
2	Large containment isolation failures	ε[3]			
3a	Small isolation failures (liner breach)	7.33E-7		3.26E+4[4]	2.39E-2
3b	Large isolation failures (liner breach)	7.29E-8		1.14E+5[5]	8.32E-3
4	Small isolation failures - failure to seal (type B)	ε			
5	Small isolation failures - failure to seal (type C)	ε			
6	Containment isolation failures (dependent failure, personnel errors)	2.68E-7	1.64E+6[6]		4.40E-1
7	Severe accident phenomena induced failure (early and late)	7.97E-6	1.90E+6[6]		1.52E+1
8	Containment bypass	4.07E-6	2.37E+7[6]		9.66E+1
	Total	2.67E-5			1.123E+2

1. From Table 17 using the method presented in Reference 2.
2. 1 times L_a dose value calculated in Reference 2.
3. ε represents a probabilistically insignificant value.
4. 10 times L_a.
5. 35 times L_a.
6. The value presented represents an average of the contributing release categories.

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

$\%Risk_{BASE} = [(Class3a_{BASE} + Class3b_{BASE}) / Total_{BASE}] \times 100$ (Eq. 9)

Where:

$Class3a_{BASE}$ = Class 3a person-rem/year = 2.39E-2 person-rem/year

$Class3b_{BASE}$ = Class 3b person-rem/year = 8.32E-3 person-rem/year

$Total_{BASE}$ = total person-rem year for baseline interval = 1.123E+2 person-rem/year (Table A.6)

$\%Risk_{BASE}$ = [(2.39E-2 + 8.32E-3) / 1.1230E+2] x 100 = **0.029%** (Eq. 10)

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 9) 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 1.0 L_a).

According to References 6 and 11, extending the Type A ILRT interval from 3-in-10 years to 1-in-10 years will increase the average time that a leak detectable only by an ILRT goes undetected from 18 to 60 months. Multiplying the testing interval by 0.5 and multiplying by 12 to convert from "years" to "months" calculates the average time for an undetected condition to exist.

The increase for a 10-yr ILRT interval is the ratio of the average time for a failure to detect for the increased ILRT test interval (from 18 months to 60 months) multiplied by the existing Class 3a probability as shown in Equation 11.

$$p_{Class3a}(10y) = 0.0275 \times \left(\frac{60}{18}\right) = 0.0916 \qquad \text{(eq. 11)}$$

A similar calculation is performed for the Class 3b probability as presented in Equation 12.

$$p_{Class3b}(10y) = 0.00273 \times \left(\frac{60}{18}\right) = 0.0091 \qquad \text{(eq. 12)}$$

Risk Impact due to 10-year test interval

Based on the previously approved methodology (References 3, 4 and 6) and the NEI guidance (Reference 1), the increased probability of not detecting excessive leakage due to Type A tests directly impacts the frequency of the Class 3 sequences.

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

Table A.7
Risk Profile for Once in Ten Year Testing

Class	Description	Frequency (/yr)	Person-rem[2]	Person-rem (/yr)
1	No containment failure[1]	1.17E-5	3.26E+3	3.81E-2
2	Large containment isolation failures	ε[3]		
3a	Small isolation failures (liner breach)	2.44E-6	3.26E+4	7.97E-2
3b	Large isolation failures (liner breach)	2.43E-7	1.14E+5	2.77E-2
4	Small isolation failures - failure to seal (type B)	ε		
5	Small isolation failures - failure to seal (type C)	ε		
6	Containment isolation failures (dependent failure, personnel errors)	2.68E-7	1.64E+6[4]	4.40E-1
7	Severe accident phenomena induced failure (early and late)	7.97E-6	1.90E+6[4]	1.52E+1
8	Containment bypass	4.07E-6	2.37E+7[4]	9.66E+1
	Total	2.67E-5		1.1237E+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 20.
3. ε represents a probabilistically insignificant value.
4. The value presented represents an average of the contributing release categories.

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

$$\%Risk_{10} = [(Class3a_{10} + Class3b_{10}) / Total_{10}] \times 100 \qquad \text{(eq. 13)}$$

Where:

$Class3a_{10}$ = Class 3a person-rem/year = 7.97E-2 person-rem/year

$Class3b_{10}$ = Class 3b person-rem/year = 2.77E-2 person-rem/year

$Total_{10}$ = total person-rem year for current 10-year interval = 1.1237E+2 person-rem/year (Table A.7)

$\%Risk_{10}$ = [(7.97E-2 + 2.77E-2) / 1.1237E+2] x 100 = **0.096%** (eq. 14)

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 (eq. 15)

Where:

$Total_{BASE}$ = total person-rem/year for baseline interval = 1.1230E+2 person-rem/year (Table A.6)

$Total_{10}$ = total person-rem/year for 10-year interval = 1.1237E+2 person-rem/year (Table A.7)

$\Delta\%Risk_{10}$ = [(1.1237E+2 – 1.1230E+2) / 1.1230E+2] x 100.0 = **0.062%** (eq. 16)

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

If the test interval is extended to 1 per 15 years, the average time that a leak detectable only by an ILRT test goes undetected increases to 90 months (0.5 x 15 x 12). For a 15-yr-test interval, the result is the ratio (90/18) of the exposure times as was the case for the 10 year case. Thus, increasing the ILRT test interval from 3 years to 15 years results in a proportional increase in the overall probability of leakage.

The approach for developing the risk contribution for a 15-year interval is the same as that for the 10-year interval. The increase for a 15-yr ILRT interval is the ratio of the average time for a failure to detect for the increased ILRT test interval (from 18 months to 90 months) multiplied by the existing Class 3a probability as shown in Equation 17.

$$p_{Class3a}(15y) = 0.0275 \times \left(\frac{90}{18}\right) = 0.1375 \qquad \text{(eq. 17)}$$

A similar calculation is performed for the Class 3b probability as presented in Equation 18.

$$p_{Class3b}(15y) = 0.00273 \times \left(\frac{90}{18}\right) = 0.0137 \qquad \text{(eq. 18)}$$

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

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

Table A.8
Risk Profile for Once in Fifteen Year Testing

Class	Description	Frequency (/yr)	Person-rem[2]	Person-rem (/yr)
1	No containment failure[1]	1.04E-5	3.26E+3	3.37E-2
2	Large containment isolation failures	ε[3]		
3a	Small isolation failures (liner breach)	3.67E-6	3.26E+4	1.20E-1
3b	Large isolation failures (liner breach)	3.65E-7	1.14E+5	4.16E-2
4	Small isolation failures - failure to seal (type B)	ε		
5	Small isolation failures - failure to seal (type C)	ε		
6	Containment isolation failures (dependent failure, personnel errors)	2.68E-7	1.64E+6[4]	4.40E-1
7	Severe accident phenomena induced failure (early and late)	7.97E-6	1.90E+6[4]	1.52E+1
8	Containment bypass	4.07E-6	2.37E+7[4]	9.66E+1
	Total	2.67E-5		1.1242E+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 20.
3. ε represents a probabilistically insignificant value.
4. The value presented represents an average of the contributing release categories.

Using the same methods as was described earlier, and the data in Table A.8, the percent risk contribution due to Type A testing is as follows:

$\%Risk_{15} = [(Class3a_{15} + Class3b_{15}) / Total_{15}] \times 100$ (eq. 19)

Where:

$Class3a_{15}$ = Class 3a person-rem/year = 1.2E-1 person-rem/year

$Class3b_{15}$ = Class 3b person-rem/year = 4.16E-2 person-rem/year

$Total_{15}$ = total person-rem year for 15-year interval = 1.1242E+2 person-rem/year (Table A.8)

$\%Risk_{15}$ = [(1.2E-1 + 4.16E-2) / 1.1242E+1] x 100 = **0.143%** (eq. 20)

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

$\Delta\%Risk_{15} = [(Total_{15} - Total_{BASE}) / Total_{BASE}] \times 100.0$ (eq. 21)

Where:

$Total_{BASE}$ = total person-rem/year for baseline (3 per 10 years) interval = 1.1230E+2 person-rem/year (Table A.6)

$Total_{15}$ = total person-rem/year for 15-year interval = 1.1242E+2 person-rem/year (Table A.8)

$\Delta\%Risk_{15} = [(1.1242E+2 - 1.1230E+2) / 1.1230E+2] \times 100.0 =$ **0.105%** (eq. 22)

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

Based on the guidance in Reference 1, the percent increase in the total integrated plant risk for these accident sequences is computed as follows:

$\%Total_{10\text{-}15} = [(Total_{15} - Total_{10}) / Total_{10}] \times 100$ (eq. 23)

Where:

$Total_{10}$ = total person-rem/year for 10-year interval = 1.1237E+2 person-rem/year (Table A.7)

$Total_{15}$ = total person-rem/year for 15-year interval = 1.1242E+2 person-rem/year (Table A.8)

$\% Total_{10\text{-}15} = [(1.1242E+2 - 1.1237E+2) / 1.1237E+2] \times 100 =$ **0.044%** (eq. 24)

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 1, 3, 4 and 6, the Class 3a dose is assumed to be 10 times the allowable intact containment leakage, L_a (or 3.26E+4 person-rem) and the Class 3b dose is assumed to be 35 times L_a (or 1.14E+5 person-rem). The dose equivalent for allowable leakage (L_a) is developed in Reference 2. This compares to a historical observed average of twice L_a. Therefore, the estimate is somewhat conservative.

Based on the NEI guidance (Reference 1) and the previously approved methodology (References 3, 4 and 6), 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 2 L_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 HNP PSA (Reference 13 and 14) 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 HNP, and the change in LERF can be determined by the differences. Reference 1 identifies that Class 3b is considered to be the contributor to LERF. Table A.9 summarizes the results of the LERF evaluation assuming that Type 3b is indicative of a LERF sequence.

Table A.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)	7.29E-8	2.43E-7	3.65E-7
ΔLERF (3 year baseline)		1.70E-7	2.92E-7
ΔLERF (10 year baseline)			1.22E-7

Reg. Guide 1.174 (Reference 12) 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 1.0E-6/yr and increases in LERF below 1.0E-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 increases the LERF contribution by 1.22E-7/yr which does not meet the criterion to be considered a non-risk significant change. It should be noted that if the risk increase is measured from the original 3-in-10-year interval, the increase in LERF is 2.92E-7/yr, which is also above the 1.0E-7/yr screening criterion in Reg. Guide 1.174. Given the large uncertainty associated with the seismic and fire events that tend to yield somewhat conservative results, this result is believed to be acceptable but additional analysis can be used to relax the inherent conservatism in the NEI approach.

Reference 15 indicates that plants with a CDF in excess of 1.0E-5/yr may have difficulty demonstrating a change in LERF less than 1.0E-7/yr. It further states that the analysis as embodied in the NEI approach is conservative and provides additional guidance with respect to refining the initial analysis.

The estimated split fraction is conservative since some sequence frequencies comprising the total CDF already account for other LERF sequences which may occur due to interfacing system LOCA events or steam generator tube ruptures. The removal of this conservatism can have a significant impact on the split fraction.

The first refinement suggested centers on the conservatism associated with using the whole CDF when estimating the potential impact of Type A leakage and specifically with class 3b leakage.

Sequences which result in LERF contributions are not influenced (change in outcome) by the potential for Type A leakage and can be excluded from the calculation of Class 3b leakage.

Also, guidance in Reference 15 indicates that LERF sequences and those sequences where, based on Level 2 analysis, any release would be scrubbed should not be considered with regard to the Class 3b frequency.

The HNP Level 2 containment safeguards event tree (CSET) and the containment event tree (CET) models (References 13 and 14) are utilized to assess the internal fire and seismic contributions and to identify the characteristics necessary for determining the status of these initiating events.

The existing containment event tree model provides adequate detail to define the endpoints associated with LERF. Additionally, it defines cases where the core debris would be flooded and/or where containment spray would be expected to function for a prolonged period of time such that the releases through any Type 3b leak would be scrubbed.

There are no bypass sequences of significant frequency identified for internal fire and seismic initiating events. Therefore, the LERF contribution is the same as for the main report. Table A.10 lists the contributing PDS results that contribute to LERF. The LERF contribution is 4.07E-6/yr.

Table A.10
LERF Contributors by PDS

PDS	Frequency (/yr)	Cumulative Frequency (/yr)	Description of PDS Sequence
X and B16 series	3.98E-6[1]	3.98E-06	SGTR with failure of SI and a SRV on the faulted steam generator fails open or SGTR, failure of SI and shutdown cooling with a cycling SRV on the faulted steam generator
Release categories 2 and 2B	8.99E-8	4.07E-6	Early containment failure and includes some internal fire contribution.
7C	4.91E-9	**4.07E-6**	Interfacing systems LOCA occurs in the RHR system

1. Represents a sum of several PDS results with similar description.

The second aspect defined in Reference 15 addresses the magnitude of the source term expected to be available for release during the accident sequence. If the debris escapes the reactor vessel but remains essentially covered with water (either due to large pools or continual containment sprays) the source term will be greatly reduced and a large source term would not be expected.

Therefore, if the accident sequence involves containment spray operation or coverage of the debris with large pools of water, the source term is not considered sufficient to support a LERF release and these contributions can be excluded.

Reference 15 suggests that one criterion is the status of the containment sprays. This criterion assumes that containment spray must be available for both injection and recirculation to ensure scrubbing of released radionuclides that are released during initial reactor vessel failure and subsequent releases from radionuclides released from the RCS after vessel failure. The end state (plant damage state) must also be an intact containment state since the unisolated containment states are already considered by the LERF fraction.

A review of the CSET defines several states where the containment spray system will function for both injection and recirculation. CSET endstates "A", "G" and "D" all represent accident sequences where the containment sprays are functioning for some period of time.

Endstates "A" and "G" are associated with continuous operation of the containment sprays in both injection and recirculation. Endstate "D" is representative of a sequence where containment spray fails in recirculation. The contribution to this sequence predominantly involves the loss of heat removal to the sprays and a failure of pressure control. The lack of header cooling would not impact the scrubbing function of the containment sprays. The containment spray would continue to function based on analysis developed for the PSA (Reference 14) and the system could maintain the scrubbing function. Containment pressure control could be maintained by the containment fan coolers. Table A.11 lists the summed frequency contribution for the PDSs that are from one of the three states identified above.

Table A.11
Contributing PDSs with CSET States A, D or G

Endstate	Frequency (/yr)	CDF Contribution
A	8.56E-6	93.6%
D	1.39E-7	1.5%
G	4.49E-7	4.9%
Total	9.14E-6	100%

The PDS contributions in Table A.11 represent 34.3% of the total CDF. The major contribution involves state "A" and the operation of the containment sprays is assured. The encompassed system failures would not result in a release of radionuclides sufficient to be classified as LERF.

From the solution presented above, a table of results is obtained and reproduced in Table A.12.

Table A.12
Source Term Outcomes

Source Term Outcome	Frequency (/yr)	Description
Source term refinement outcome 1	4.07E-6	Predefined LERF sequence (CSET X or B or early failure)
Source term refinement outcome 2	9.14E-6	Non-LERF sequence with the containment sprays functioning (endstates A, D or G)
Source term refinement outcome 3	1.35E-5	Non-LERF sequence without the containment sprays functioning (all other end states)

Only outcome 3 contributes to the potential for a Type A LERF. This value is then utilized to calculate the LERF contribution from Class 3b frequency by the following equation:

$FREQ_{class3b} = PROB_{class3b} \times \text{Adjusted CDF} = 0.00273 \times 1.35E\text{-}5/yr = 3.68E\text{-}8/yr$ (eq. 26)

This can then be extrapolated using the methods presented earlier to determine the 10-year and 15-year contributions and to generate adjusted LERF values as presented in Table A.13.

Table A.13
Class 3b Contributions Using Adjusted CDF

Test Interval	Frequency (/yr)	Delta Frequency from Prior Period (/yr)
Baseline	3.68E-8	
10-year (current)	1.21E-7	8.47E-8
15-year	1.82E-7	6.08E-8

Summing the last column provides the total increase from the baseline (3 year) to the proposed (15 year) interval (1.45E-7/yr). This increase is still higher than the criterion for a sufficiently small increase to meet the guidance for a non risk significant change in risk.

Examining the PDS contributors the controlling plant damage states are 10P (6.59E-6/yr) and 1P (5.31E-6/yr). These sequences both involve the failure of the ac power system (both onsite and offsite) with a reactor coolant pump seal LOCA and no injection.

PDS 1P also includes a loss of secondary-side heat removal. The PDS definition neglects the potential for ac power restoration prior to significant fuel damage. This is addressed in the CET modeling for invessel recovery (Event IVR).

The HNP CET model includes modeling of the potential for recovery of ac power prior to significant releases following reactor vessel breach. Analysis performed for estimation of the time available for sequences involving a loss of ac power indicates that an incremental factor (time to recover between time "t" and the final failure time) is 0.09 for cases involving no secondary-side heat removal and 0.18 for cases with secondary-side heat removal.

The major difference is that the recovery curve for the cases without heat removal is substantially steeper than the case with secondary-side heat removal and so a small increase in time has a more pronounced impact.

These adjustments cannot be made directly to the PDS results as was done in the main analysis. This would require ac power recovery being assumed in the seismic risk contribution. The potential for recovery of offsite ac power for seismic sequences is very uncertain and almost no credit should be applied to higher acceleration cases. Therefore, the seismic contribution is removed and then the recovery can be applied.

Table A.14 provides a summary of the refinement for these two PDS results.

Table A.14
Refinement of PDS 1P and 10P to Address Late Ac Power Recovery

Plant Damage State	Frequency Contribution without Ac Recovery (/yr)	Frequency Contribution with Ac Recovery (/yr)
1P	4.78E-7	4.83E-6
10P	1.01E-6	4.60E-6
10P (seismic)	8.04E-7	-
Total	1.49E-6	9.43E-6

Using this information an adjustment to the Class 3b frequency can be made. The earlier value of 1.35E-5/yr (Table A.12) can be reduced to 4.05E-6/yr. This value is then utilized to calculate the LERF contribution from Class 3b frequency by the following equation:

$FREQ_{class3b} = PROB_{class3b}$ x Adjusted CDF = 0.00273 x 4.05E-6/yr = 1.106E-8/yr (eq. 27)

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

Table A.15
Class 3b Contributions Using Adjusted PDS After Ac Recovery

Test Interval	Frequency (/yr)	Delta Frequency from Prior Period (/yr)
Baseline	1.106E-8	
10-year (current)	3.648E-8	2.543E-8
15-year	5.473E-8	1.824E-8

Summing the last column provides the total increase from the baseline (3 year) to the proposed (15 year) interval (4.37E-8/yr). This LERF increase associated with the ILRT extension is substantially below the limit for a small change in risk.

Similar to the main analysis, a sensitivity analysis related to the offsite power recovery value was performed. The recovery value was set at 0.5 and the results recalculated. Table A.16 presents the results.

Table A.16
Class 3b Contributions Using Conservative Ac Power Recovery (0.5)

Test Interval	Frequency (/yr)	Delta Frequency from Prior Period (/yr)
Baseline	2.19E-8	
10-year (current)	7.23E-8	5.04E-8
15-year	1.08E-7	3.61E-8

Summing the contributions yields a value of 8.65E-8/yr which is still below the significance threshold.

Step 7: Calculate the change in conditional containment failure probability (CCFP)

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

$$CCFP = 1 - \left[\frac{f(ncf)}{CDF}\right] \quad \text{(Eq. 28)}$$

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

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

$$\Delta CCFP_{10-15} = CCFP_{15} - CCFP_{10} \qquad \text{(Eq. 29)}$$

Using the data previously developed the change in CCFP from the current testing interval is calculated and presented in Table A.17.

Table A.17
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)	1.431E-5	1.414E-5	1.402E-5
f(ncf)/CDF	0.536	0.530	0.525
CCFP	0.464	0.470	0.475
ΔCCFP (3 year baseline)		0.638%	1.093%
ΔCCFP (10 year baseline)			0.455%

Appendix B:
References

B.0 REFERENCES

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7. Evaluation of Risk Significance of ILRT Extension, Revision 2, Florida Power Corporation, F-01-0001, June 2001.

8. Shearon Harris Nuclear Power Plant, Plant Operating Manual, Volume 6, Part 6, Engineering Surveillance Test, Periodic Containment Integrated Leak Rate Testing (Type A Test), Rev. 11, Progress Energy, EST-210.

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

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13. Updated Individual Plant Examination Probabilistic Safety Assessment Model, Rev 5, Progress Energy, HNP-F/PSA-0001, November 2003.

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Document Quality Configuration Control
FORM NO.: RSC-RPT-STD05-01, Rev. 0

Project Document and Software Configuration Control

RSC Project Number:	05-08
Project Folder on RSC Server:	\RSC 2005 Project Folders\HNP ILRT NEI Approach
Document Name and Last Saved:	RSC 05-08 R0.doc / November 01, 2005
Software Used for Report Creation: (place an X by all that apply)	X Word (doc) X Excel (xls) _ Access (mdb) _ Visio (vsd) _ CAFTA _ EOOS _ MAAP BWR _ MAAP PWR _ PRAMS _ SIP _ TIFA _ FATCAT _ DBM
Abstract (purpose of study): To determine the increase in risk from extended ILRT schedule using the NEI approach.	
List of keywords for RSC library document retrieval (not more than three): ILRT HNP NEI	

Review and Quality Configuration Control

Verification and Review of Methods and Results: (Place X to Left of Method Utilized)

_ Detailed Review
_ Alternative Calculation
_ Review Not Required
_ Other (describe other review method) ____________________

Reviewer Comment	Resolution of Comment
1. Editorial comments in text.	Incorporated.
2. Equation 7 and on in Appendix A of the report are numbered incorrectly as marked. Equation 27 in main report too.	Corrected.
3. Why is there not an entry in Table 9 for delta-LERF (10 year baseline) for the 10 year case when there is in Appendix A Table A.9?	Value added.
4. Should the values in Table A.10 be the same as Table 10 in the main report? Table A.11 and Table 11 differ.	Yes because the internal fire and seismic events do not contain bypass or LERF contributions.

Editorial or illustrative comments may be electronically provided (tracking) or attached to this review sheet. Reviewer is to approve all proposed resolutions prior to completing the review process. No review is complete until this step is accomplished.

ATTACHMENT 2 – OWNERS REVIEW OF VENDOR REPORT

Design Vendor Report RSC 05-08 Evaluation of Risk Significance of ILRT Extension Based on the NEI Approach **Revision** 0

The signature below of the Lead Reviewer records that:

- the review indicated below has been performed by the Lead Reviewer;
- appropriate reviews were performed and errors/deficiencies (for all reviews performed) have been resolved and these records are included in the design package;
- the review was performed in accordance with EGR-NGGC-0003.

☐ **Design Verification Review** ☐ **Engineering Review** ☒ **Owner's Review**

☐ Design Review
☐ Alternate Calculation
☐ Qualification Testing

☐ **Special Engineering Review**__________

☐ YES ☒ N/A **Other Records are attached.**

Steven L. Mabe	PSA	10/29/05
Lead Reviewer	**Discipline**	**Date**

Item No.	Deficiency	Resolution
1	Last sentence of page 1 indicates that Reference 2 had Appendices B and C when it only had Appendix A.	Corrected.
2	In 5th bullet on page 3 (that begins with Reg. Guide 1.174…), the following sentence should be reworded "In addition, the change in LERF resulting from a change in the Type A ILRT test interval from a three-per-ten-years to a once per-fifteen-years is 2.46E-8/yr, is also slightly above the guidance." The resulting value is below the guidance.	Corrected.
3	1st paragraph of Section 2 refers to References 14 and 15 but should refer to References 13 and 14.	Corrected.
4	In Table 5, Class 8 lists RC-4, RC-4C, and RC-5 under the Release Category column; however, they are not in Table 3 list. Remove these RC's from Table 5.	They are included in the RC-5C category since they are all bypass sequences. A footnote has been added to Table 4 to specify that RC-5C includes all the other cases.
5	The result of Equation 14 shows to be 0.080% but hand calculation shows it to be 0.087% which matches the value in Table 1.	Error in cell equation identified and corrected.

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6	The result of Equation 27 shows to be 3.11E-8/yr but hand calculation shows it to be 3.20E-8/yr. Whichever value is correct (understand rounding can impact results), the Baseline Frequency in Table 13 should be the same value and the corresponding 10-year and 15-year values will change depending on what the Baseline value is.	Should be 3.18E-8. Typographical error corrected. Calculations that are supported by this value are correct.
7	The second sentence following Table 13 reads "This increase is still slightly higher than the limit for a sufficiently small to meet the guidance for a small change in risk." Suggest wording be changes to something like "This increase is still slightly higher than the limit for a small change in risk."	Suggestion adopted.
8	The first sentence following Table 14 indicates 2.28E-6/yr as the revised Class 3b frequency; however, hand calculation of 1.16E-5/yr minus 9.37E-6/yr yields 2.23E-6/yr. Inputting this value into Equation 28 yields 6.09E-9/yr. This value is then used to develop Table 15 results and delta frequency values. What are correct values?	The values listed in the report for results are correct and the variation is due to rounding in the values. Please refer to cells Q11-16 of tab Revised Lerf Calculation of spreadsheet RSC 05-08 R0.xls.
9	The second sentence following Table 15 reads "This LERF increase associated with the ILRT extension is substantially below the limit for a sufficiently small to meet the guidance for a small change in risk." Suggest wording be changed to something like "This LERF increase associated with the ILRT extension is substantially below the limit for a small change in risk."	Suggested wording adopted.
10	Third paragraph following Table 2 states that the PDS's were mapped to one of fourteen release categories. However, Table 3 only contains **eleven** release categories.	A comment has been added to Table 2 to indicate that the RC—5C information also included RC-4, RC-4C, and RC-5. No change to text since that does equal 14 release categories.
11	Second sentence of Section A.1 of Appendix A discusses References 1 through 14; however, there are 15 References. Should this say References 1 through 15?	The omission of Reference 15 was intentional. It is identified later when the analysis is refined.
12	Third paragraph following Table A.2 states that the PDS's were mapped to one of fourteen release categories. However, Table A.3 only contains **eleven** release categories.	See response to #10.

Design Vendor Report RSC 05-08 **Revision** 0
Evaluation of Risk Significance of ILRT Extension Based on the NEI Approach

13	In Table A.5, Class 8 lists RC-4, RC-4C, and RC-5 under the Release Category column; however, they are not in Table 3 list. Remove these RC's from Table 5.	See response to #4. It is assumed that the table references are associated with Appendix A (references to Table 3 and Table 5 should be A.3 and A.5).
14	The Person-rem/yr total shown in Table A.6 is 1.123E+1; however, this value should be 1.123E+2.	Typo corrected in table. No change to follow on calculations.
15	Appendix A Equation 11 shows the Class 3a(base) person-rem/yr value to be 2.39E-3; however, this value should be 2.39E-2.	Typo corrected as is shown above in input description. No impact on calculations.
16	In Appendix A, the result of Equation 11 shows to be 0.026% but hand calculation shows it to be 0.0287% which matches the value in Table A.1.	See response to comment #5.
17	In Appendix A, the result of Equation 15 shows to be 0.088% but hand calculation shows it to be 0.0956% which matches the value in Table A.1.	See similar response for comment #5.
18	The 1st sentence of the second paragraph following Table A.9 reads "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 (1.22E-7/yr) and this does not meet the criterion to be considered immediately as a non-risk significant change." Suggest wording be something like "Since guidance in Reg. Guide 1.174 defines very small changes in LERF as below 1.0E-7/yr, changing the ILRT interval to 15 years increases the LERF contribution by 1.22E-7/yr which does not meet the criterion to be considered a non-risk significant change."	Suggested wording adopted.
19	The 2nd sentence following Table A.15 reads "This LERF increase associated with the ILRT extension is substantially below the limit for a sufficiently small to meet the guidance for a small change in risk." Suggest wording the changed to something like "This LERF increase associated with the ILRT extension is substantially below the limit for a small change in risk."	Suggested wording adopted.
20	HNP specific RSC calculations or spreadsheets developed to provide results and/or support conclusions presented in this report should be provided to PGN.	Calculations are developed an awaiting RSC internal review. They will be provided to PGN when approved.

FORM EGR-NGGC-0003-2-10
This form is a QA Record when completed and included with a completed design package. Owner's Reviews may be processed as stand alone QA records when Owner's Review is completed.

ATTACHMENT 3 – RISK FROM CONCEALED LINER CORROSION

References
See main section of the calculation.

Design Inputs
The evaluation of risk due to containment liner corrosion does not provide plant design basis information nor is the evaluation used to modify design outputs. Therefore, no design inputs are used.

The inputs to the evaluation are documented in the attached report and its references.

Assumptions
This calculation applies an analytical method developed by Calvert Cliffs in response to an NRC question about concealed containment liner corrosion (Reference 5). Assumptions associated with that method are provided in Reference 5. Information from the updated HNP PSA Model is used in this analysis. The PSA model and its associated assumptions are described in References 1 and 2.

The applicability of the Calvert Cliffs assumptions to the HNP analysis is discussed below:

ASSUMPTION	BASIS FOR APPLICABILITY TO HNP
A. Zero basemat corrosion failures are evaluated as if 0.5 failures.	Typical PRA assumption for cases with zero actual failures. Industry-wide data is employed. That data is applicable to HNP.
B. Success data limited to period since 10 CFR 50.55 requirements to inspect.	Industry-wide data is employed. That data is applicable to HNP.
C. Liner flaw likelihood doubles every five years.	Calvert Cliffs sensitivities addressing range of doubling period bracket this assumption and are applicable to HNP, as well.
D. Likelihood of liner breach is a function of pressure.	Calvert Cliffs sensitivities addressing range of failure probabilities at high and low range of pressure are applicable to HNP, as well.
E. Basemat leakage assumption.	HNP basemat (liner thickness and placement of concrete) is similar to Calvert Cliffs.
F. Visual detection likelihoods.	Calvert Cliffs sensitivities addressing range of detection likelihoods are applicable to HNP.
G. Non-detectable containment over-pressurization failures are assumed to be LERF.	This conservative assumption avoids need for detailed analysis of containment failure timing and operator recovery actions.

Calculations
This analysis provides information previously requested by the NRC for ILRT extension evaluations at the other Progress Energy nuclear plants. An analytical approach employed by Calvert Cliffs to estimate risk due to concealed containment corrosion is used, with adaptations for the Harris Nuclear Plant described below. Discussion of the method and its underlying assumptions is provided in Reference 5. Table 1 below documents the application of the method to HNP.

Step 0 is added to show the calculation of inaccessible areas of the containment liner. To account for areas near the fuel transfer tube and other obstructed locations, only 97% of the cylinder walls and dome is assumed to be accessible for inspection. A sensitivity calculation assuming an approximate 5,000 sq. ft. sump liner did not affect the conclusion. Liner dimensions are taken from Reference 8. This information is unchanged from Revision 0 of this calculation, as the HNP containment is unchanged.

Step 1 is updated from the Calvert Cliffs analysis to account for two additional failures recognized as applicable by the NRC (Reference 9). The applicable period since the 10 CFR 50.55a requirement is now 7.5 years (Sep-96 to Mar-04). Steps 2 and 3 are calculated with this updated information. Additional details for the Step 3 calculation are provided in Table 2. An increase in the number of failures by one (to a total of five) is included, to address emerging information from the March 2004 Brunswick Nuclear Plant liner inspection. The applicable period is unchanged from Revision 0 of this calculation (7.5 years) despite the additional time since Mar-04, to remain consistent with the previous analysis.

Plant specific information is input into Step 4. The upper end pressure (153 psig) is taken from the HNP IPE containment overpressure capacity for the limiting failure mode, basemat shear (Reference 1) and converted to psia. The ILRT test pressure (44 psig) is taken from Reference 6 and converted to psia with 1 psia added for conservatism. The Step 6 likelihood of non-detected containment leakage is weighted by the accessible and inaccessible percentage of the liner, calculated in Step 0. This information is unchanged from Revision 0 of this calculation, as the HNP containment overpressure analysis is unchanged.

The internal events CDF is taken from the current PSA model (Reference 2), as refined by Reference 10. Portions of CDF that are already LERF or that can never become LERF and thus are unaffected by the liner corrosion are taken from Reference 10 (Attachment 1). Reference 7 provided the input to the vendor report for plant damage states.

The Calvert Cliffs analysis provides a number of sensitivity calculations. Those calculations are illustrative of the impact of the assumptions and are not repeated here.

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum of Step 6 for the containment cylinder and dome and the containment basemat.

Conclusions

If all non-detectable containment leakage events are considered to be LERF, then the increase in LERF associated with containment liner corrosion based on going from an ILRT frequency of three times per ten years to once per fifteen years is 3.11E-9, which is a very small contribution.

If an additional failure is included in the historical liner flaw likelihood, the increase in LERF is 3.21E-9, which remains a very small contribution. If sequences that are already LERF and sequences that can never become LERF are excluded from the calculation,

then the increase in LERF associated with containment liner corrosion is 2.93E-10. These sensitivities demonstrate the substantial amount of conservatism in the above calculation.

Table 1

Step	Item	Inputs	Accessible Containment Cylinder and Dome	Inaccessible Cylinder and Dome	Basemat
0	Percent Cylinder Accessible for Inspection	97.0%			
	Percent Dome Accessible for Inspection	100.0%			
	Dome Surface Area	26,546	26,546	0	
	Cylinder Surface Area	65,345	63,385	1,960	
	Drywell Floor Surface Area (sq ft)	13,273			13,273
	Vertical Sides of Sump	0			0
	Total Surface Area (sq ft)	105,165	89,931	1,960	13,273
	Percentage Total		85.5%	1.9%	12.6%
	Percentage Accessible (to weight Step 6)		97.9%	2.1%	
	Historic Liner Flaw Likelihood Failure Data: Containment location specific Succcess Data: Based on 70 steel-lined containments and 7.5 years since the 10CFR50.55a requirement for periodic visual inspections of containment surfaces. (5 failures in 7.5 yr; assume 0.5 failure for basemat in 7.5 yr)				
1			9.524E-03	9.524E-03	9.524E-04
2	Age Adjusted Liner Flaw Likelihood (15 yr avg)		1.18E-02	1.18E-02	1.18E-03
3	Increase in Flaw Likelihood between 3 and 15 Years (See Table 2)		15.05%	15.05%	1.63%
	Upper End Pressure (100% likelihood), psia	168			
	Lower End Pressure (0.1% likelihood), psia	20			
	Test Pressure (psia)	60			
	Slope (*m*)	4.67E-02			
	Intercept (b)	3.93E-04			
4	Likelihood of Breach in Containment Given Liner Flaw		0.65%	0.65%	0.06%
5	Visual Inspection Detection Failure Likelihood		10%	100%	100%
6	Likelihood of Non-Detected Containment Leakage		0.00973%	0.09731%	0.00105%
	Total Likelihood of Non-Detected Containment Leakage (weighted)		0.0130%		
	Internal Events CDF (MOR2003, with adjustment)	2.39E-05			
	Already LERF (RC-2, 2B,and -5C = PDS Endstates 7C, X and B16 series)	4.07E-06			
	Never go to LERF (PDS Endstates A, D and G)	8.21E-06			
	Non-LERF Sequences with Cont Spray (PDS Endstates 1P and 10P refinement)	9.37E-06			
	Non-LERF CDF -- internal	2.25E-06			

Increase in LERF due to Liner Corrosion (Internal Events Only)	**3.11E-09**
Increase in LERF due to Liner Corrosion (Non-LERF CDF – internal)	**2.93E-10**

Table 2

During 15-year interval, assumed failure rate (FR) doubles every five years,
Yrs to Double = 5
(eq. 1) FR(n+5) = FR(n) * 2.0
Let FR increase at a constant rate (x)
FR(n+1) = FR(n) * (1+x)^n
FR(n) = FR0 * (1+x)^n
For t= 5 yrs, FR(5+5) = FR(5) * 2.0
FR(10) = FR(5) * (1+x)^5
FR(5) * (1+x)^5 = FR(5) * 2.0
(1+x)^5 = 2.0
1 + x = 2.0 ^(1/5)
x = 2.0^(1/5) - 1
Increase per year, x = 14.87%
The average for fifth to tenth year was set to the historic failure rate
Historic Failure Rate = 9.52E-03
(eq. 2) [FR(5) + FR(6) + FR(7) + FR(8) + FR(9) + FR(10)]/6 = 8.79E-3
FR(5) * [1+(1+x) + (1+x)^2 + (1+x)^3 + (1+x)^4 + (1+x)^5]/6 = 8.79E-3
FR(5) * [1 + 1.1487 + 1.1487^2 + 1.1487^3 + 1.1487^4 + 1.1487^5]/6 = 8.79E-3
FR(5) = 6.55E-03
FR0 = 3.27E-03

YR	AGE ADJUSTED FAILURE RATE (FR)	6-YR AVE	15-YR AVE	SUCCESS RATE (1-FR)	AVERAGE SUCCESS RATE FROM YR=1	AVERAGE FAILURE RATE FROM YR=1
0	3.27E-03			9.97E-01	1.00E+00	0.00%
1	3.76E-03			9.96E-01	9.96E-01	0.38%
2	4.32E-03			9.96E-01	9.92E-01	0.81%
3	4.96E-03			9.95E-01	9.87E-01	**1.30%**
4	5.70E-03			9.94E-01	9.81E-01	1.86%
5	6.55E-03	4.76E-03		9.93E-01	9.75E-01	2.50%
6	7.52E-03	5.47E-03		9.92E-01	9.68E-01	3.24%
7	8.64E-03	6.28E-03		9.91E-01	9.59E-01	4.07%
8	9.93E-03	7.22E-03		9.90E-01	9.50E-01	5.03%
9	1.14E-02	8.29E-03		9.89E-01	9.39E-01	6.11%
10	1.31E-02	**9.52E-03**		9.87E-01	9.27E-01	7.34%
11	1.50E-02	1.09E-02		9.85E-01	9.13E-01	8.73%
12	1.73E-02	1.26E-02		9.83E-01	8.97E-01	10.31%
13	1.99E-02	1.44E-02		9.80E-01	8.79E-01	12.09%
14	2.28E-02	1.66E-02		9.77E-01	8.59E-01	14.10%
15	2.62E-02	1.90E-02	**1.18E-02**	9.74E-01	8.37E-01	**16.35%**

Increase in Flaw Likelihood Between 3 and 15 Years **15.05%**
(Delta between ave failure rate for YR=15 and YR=3)

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Calvert Cliffs
Nuclear Power Plant

A Member of the
Constellation Energy Group

March 27, 2002

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit No. 1; Docket No. 50-317
Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension

REFERENCES:

(a) Telephone Conferences between Ms. D. J. Moeller, et al. (CCNPP) and Ms. D. M. Skay, et al., dated March 1, March 7, March 14, and March 19, 2002, same subject

(b) Letter from Mr. C. H. Cruse (CCNPP) to NRC Document Control Desk, dated January 31, 2002, "License Amendment Request: One-Time Integrated Leakage Rate Test Extension"

(c) Letter from Mr. C. H. Cruse (CCNPP) to NRC Document Control Desk, dated November 19, 2001, "License Amendment Request: Revision to the Containment Leakage Rate Testing Program Technical Specification to Support Steam Generator Replacement"

This letter provides the information requested in a series of teleconferences (Reference a) and supplements the information provided in Reference (b). Specifically, we were asked to provide information addressing how the potential leakage due to age-related degradation mechanisms were factored into the risk assessment for our requested Integrated Leakage Rate Test (ILRT) one-time extension. In addition, we are submitting a correction to the marked-up pages originally provided in Reference (b). This information does not change the conclusions of the significant hazards determination provided in Reference (b).

REQUESTED CHANGE

The final Technical Specification pages are included in Attachment (1). In Reference (b), the term "exempted" was used in the marked-up version of the Technical Specification pages. The correct term that should have been used was "excepted." The final Technical Specification pages reflect this correction. This correction should also be applied to the change requested in Reference (c).

A047

SUPPLEMENTAL INFORMATION

Structural Design

Walls

The Containment Structure is a post-tensioned, reinforced concrete cylinder and dome connected to and supported by a massive reinforced concrete slab (basemat). The liner plate is ¼-inch thick and is attached and anchored to the containment concrete structure. The concrete vertical wall thickness is 3-¾ feet. The concrete dome thickness is 3-¼ feet. Since the concealed side of the liner plate is in contact with the concrete, leakage requires a localized transmission path connecting a breach in the containment concrete with a flaw in the liner.

Floor

The containment basemat is a 10-foot thick base slab that was constructed monolithically with steel sections (H or W sections) laid out to match the liner plate joints and embedded such that one flange surface was flush with the finished concrete. The liner plates were then laid out on top of these sections and welded. The liner plates are full penetration welded to each other with a gap of sufficient thickness to allow the root of the weld to partially penetrate the embedded steel. This provides a segmented area under the floor liner plates where free communication from one area to the other is heavily constrained.

After welding was complete, the welds themselves were covered with channel sections (leak chases), seal welded to the plates, and ported to allow pressure testing of the liner welds. The floor liner plates were oiled and the interior slab was poured with the test connections left in place to provide for future weld testing during ILRTs.

The liner plates under the interior slab are in contact with the concrete on both sides except for a small area at the leak chases and at the edge of the concrete where an expansion material was used. Since concrete acts to protect steel in contact with it, we feel that there is little likelihood of corrosion occurring in the floor liner plates. During replacement of the moisture barrier, the area directly behind the old barrier material was determined to be the area most affected by corrosion. This area was evaluated on both units and has been incorporated into an augmented examination population required by the American Society of Mechanical Engineers (ASME) Code.

Inspectable Area

Approximately 85 percent of the interior surface of the liner is accessible for visual inspections. The 15 percent that is inaccessible for visual inspections includes the fuel transfer tube and area under the containment floor.

Liner Corrosion Events

Two events of corrosion that initiated from the non-visible (backside) portion of the containment liner have occurred in the industry. These events are summarized below:

- On September 22, 1999, during a coating inspection at North Anna Unit 2, a small paint blister was observed and noted for later inspection and repair. Preliminary analysis determined this to be a through-wall hole. On September 23, a local leak rate test was performed and was well below the allowable leakage. The corrosion appeared to have initiated from a 4"x4"x6' piece of lumber embedded in the concrete.

An external inspection of the North Anna Containment Structures was performed in September 2001. This inspection (using the naked eye, binoculars, and a tripod-mounted telescope) found several additional pieces of wood in both Unit 1 and Unit 2 Containments. No liner degradation associated with this wood was discovered.

- On April 27, 1999, during a visual inspection of the Brunswick 2 drywell liner, two through-wall holes and a cluster of five small defects (pits) in the drywell shell were discovered. The through-wall holes were believed to have been started from the coated (visible side). The cluster of defects was caused by a worker's glove embedded in the concrete.

Calvert Cliffs Inspection Program

To help assure continued containment integrity, the containment liners at Calvert Cliffs Nuclear Power Plant (CCNPP) are examined in accordance with the requirements of ASME Boiler and Pressure Vessel (B&PV) Code Section XI, Subsection IWE (as amended and modified by 10 CFR 50.55a) and the plant Protective Coatings Program, both as a natural consequence of maintenance activities and as planned events. Each will be discussed separately.

During the course of maintenance activities requiring repairs to the containment liner plate coatings, ASME XI Subsection IWE requires visual exams to evaluate the condition of the liner plate. Typically, these repairs are done to correct blisters, peeling, flaking, delamination, and mechanical damage of the coating system of the liner. To date, there have been over 500 exams of this nature (one repair generates multiple exams) performed at CCNPP since the requirements of Subsection IWE were imposed with no indication of liner base metal degradation.

The safety-related Protective Coatings Program at CCNPP requires a walkdown of the containment interior be performed at the beginning of each refueling outage to determine areas requiring repair. This walkdown, performed by engineering personnel, maintenance personnel, and National Association of Corrosion Engineers (NACE)-trained coatings examiners, looks at accessible coated structures in the Containment as well as the liner.

Repair of items found on these walkdowns is then planned, staged, and performed, with any postponement of repairs beyond the current outage requiring engineering approval. Liner coating repairs are witnessed and documented at the beginning stage and upon completion by a Certified Non-Destructive Examination (NDE) Examiner. This is to allow proper assessment of the cause of the damage prior to repair and to document the as-left condition. The specific goal of this approach is to identify any indication of liner damage. As stated above, over 500 documented exams have shown no evidence of liner degradation.

Scheduled inservice inspection (ISI) exams are performed in accordance with the scheduling requirements of the ASME Section XI, Subsection IWE, and 10 CFR 50.55a. These documents require visual examination of essentially 100% of the containment liner accessible surface area once per ISI period (three in ten years). This exam is performed and documented by Certified NDE Examiners during the outage and/or before an ILRT.

This exam is performed both directly and remotely, depending upon the accessibility to the various areas. Remote exams are performed with binoculars to provide a clear view of all areas. To date, this exam has been performed twice on Unit 1 and once on Unit 2 with no recordable indications of liner plate degradation.

Several areas were identified on both units as candidate areas for Augmented Examination, in accordance with IWE-1241. These included areas beneath the liner to floor slab moisture barriers, potential ponding areas at structural steel attachments, and several areas with photographic evidence of dark areas. Further evaluation of these areas yielded the following conclusions:

- No ponding areas were evident either as being presently wet or by the presence of watermarks.
- The dark areas were identified in both cases to be insulation at a penetration.
- The area beneath the moisture barrier on both units showed degradation that required engineering evaluation. The area beneath the moisture barrier was found to suffer from scaling, rust, and pitting. Areas visually representative of the worst of these were selected for detailed examination and documented using a combination of ultrasonic thickness measurement, pit depth measurement, and detailed visual examination. These areas are now designated as Augmented Examination in accordance with Subsection IWE, and are subject to repeat examination once per ISI period as required by Subsection IWE.

The bolting examinations required by Table IWE-2500-1, Category E8.10 and E8.20, are performed during preventive maintenance activities of certain components. These maintenance activities are scheduled to support replacement of the seals and gaskets used in the component connections. Additionally, some of these connections are routinely used during outages, and the examination and testing of these connections is performed to re-establish containment integrity at the end of the outage. Any parts (except for seals and gaskets, which are exempt) that are replaced are subject to compliance with our Repair and Replacement Program and receive the appropriate inspections at that time.

Non-destructive examination examiner qualifications are governed by Calvert Cliffs procedure MP-3-105, "Qualification of Non-Destructive Examination Personnel and Procedures." This procedure requires documenting the necessary experience, training, visual acuity, and certifications in accordance with American National Standards Institute/American Society for Nondestructive Testing CP-189. Additionally the CCNPP coating examiners are NACE trained.

Effectiveness of the CCNPP inspection programs is judged to be high. This is based on the use of both NACE and CP-189-certified examiners for the different exams that are conducted. The depth that is provided by this approach yields a level of redundancy due to the differing focus of each examination.

Rigor of the examinations is provided by compliance with our Protective Coatings, NDE, and ISI programs. The coatings program controls the initial walkdown and focuses on the condition of the safety-related Level 1 coatings. This effort provides an initial assessment of the gross liner condition. In addition, the NDE Program provides a CP-189 certified examiner when preparation is started on each area to be repaired. This is done to verify the condition of the base metal as the defective coating is removed. As noted previously, this activity has resulted in over 500 documented examinations with no indications of liner deterioration.

Further, the ISI Program for Subsections IWE and IWL requires examination of the accessible portions of the liner once per period. This exam is conducted using a mixture of direct and remote examination techniques. Both units have been examined completely through these joint programs at least one time each with no defects noted. We will perform an additional Subsection IWE visual exam during the 2004 Unit 1 refueling outage.

Liner Corrosion Analysis

The following approach was used to determine the change in likelihood, due to extending the ILRT, of detecting liner corrosion. This likelihood was then used to determine the resulting change in risk. The following issues are addressed:

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

Assumptions

A. A half failure is assumed for basemat concealed liner corrosion due to the lack of identified failures. (See Table 1, Step 1.)

B. The success data was limited to 5.5 years to reflect the years since September 1996 when 10 CFR 50.55a started requiring visual inspection. Additional success data was not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date and there is no evidence that liner corrosion issues were identified. (See Table 1, Step 1.)

C. The liner flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increase likelihood of corrosion as the liner ages. Sensitivity studies are included that address doubling this rate every 10 years and every two years. (See Table 1, Steps 2 and 3, and Tables 5 and 6.)

D. The likelihood of the containment atmosphere reaching the outside atmosphere given a liner flaw exists is a function of the pressure inside the Containment. Even without the liner, the Containment is an excellent barrier. But as the pressure in Containment increases, cracks will form. If a crack occurs in the same region as a liner flaw, then the containment atmosphere can communicate to the outside atmosphere. At low pressures, this crack formation is extremely unlikely. Near the point of containment failure, crack formation is virtually guaranteed. Anchored points of 0.1% at 20 psia and 100% at 150 psia were selected. Intermediate failure likelihoods are determined through logarithmic interpolation. Sensitivity studies are included that decrease and increase the 20 psia anchor point by a factor of 10. (See Table 4 for sensitivity studies.)

E. The likelihood of leakage escape (due to crack formation) in the basemat region is considered to be 10 times less likely than the containment cylinder and dome region. (See Table 1, Step 4.)

F. A 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 10% is used. To date, all liner corrosion events have been detected through visual inspection. (See Table 1, Step 5.) Sensitivity studies are included that evaluate total detection failure likelihoods of 5% and 15%. (See Table 4 for sensitivity studies.)

G. All non-detectable containment over-pressurization failures are assumed to be large early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

Analysis

Table 1

Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome 85%		Containment Basemat 15%	
1	**Historical Liner Flaw Likelihood** Failure Data: Containment location specific Success Data: Based on 70 steel-lined Containments and 5.5 years since the 10 CFR 50.55a requirement for periodic visual inspections of containment surfaces.	Events: 2 (Brunswick 2 and North Anna 2) 2/(70 * 5.5) = 5.2E-3		Events: 0 Assume half a failure 0.5/(70 * 5.5) = 1.3E-3	
2	**Aged Adjusted Liner Flaw Likelihood** During 15-year interval, assumed failure rate doubles every five years (14.9% increase per year). The average for 5th to 10th year was set to the historical failure rate. (See Table-5 for an example.)	Year 1 avg 5 – 10 15	Failure Rate 2.1E-3 5.2E-3 1.4E-2	Year 1 avg 5 – 10 15	Failure Rate 5.0E-4 1.3E-3 3.5E-3
		15 year avg = 6.27E-3		**15 year avg = 1.57E-3**	
3	**Increase in Flaw Likelihood Between 3 and 15 years** Uses aged adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years. See Tables 5 and 6.	**8.7%**		**2.2%**	
4	**Likelihood of Breach in Containment given Liner Flaw** The upper end pressure is consistent with the Calvert Cliffs Probabilistic Risk Assessment (PRA) Level 2 analysis. 0.1% is assumed for the lower end. Intermediate failure likelihoods are determined through logarithmically interpolation. The basemat is assumed to be 1/10 of the cylinder/dome analysis	Pressure (psia) 20 64.7 (ILRT) 100 120 150	Likelihood of Breach 0.1% **1.1%** 7.02% 20.3% 100%	Pressure (psia) 20 64.7 (ILRT) 100 120 150	Likelihood of Breach 0.01% **0.11%** 0.7% 2.0% 10.0%
5	**Visual Inspection Detection Failure Likelihood**	**10%** 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT) All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.		**100%** Cannot be visually inspected.	

Table 1

Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome 85%	Containment Basemat 15%
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3 * 4* 5)	0.0096% 8.7% * 1.1% * 10%	0.0024% 2.2% * 0.11% * 100%

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum of Step 6 for the containment cylinder and dome and the containment basemat.

Total Likelihood of Non-Detected Containment Leakage = 0.0096% + 0.0024% = 0.012%

The non-large early release frequency (LERF) containment over-pressurization failures for CCNPP Unit 1 are estimated at 8.6E-5 per year. This is based on the Revision 0 Unit 1 Model. This model includes both internal and external events. The external events portion of the model was recently finalized. External events represents 55% of the total core damage frequency (CDF) with fire being by far the largest external event contributor. The total CDF is 8.9E-5. This current CDF is used to re-generate the delta LERF/rem impacts for both the Crystal River (CR) method and Combustion Engineering Owners Group (CEOG) method. If all non-detectable containment leakage events are considered to be LERF, then the increase in LERF associated with the liner corrosion issue is:

Increase in LERF (ILRT 3 to 15 years) = 0.012% * 8.6E-5 = 1E-8 per year.

<u>Change in Risk</u>

The risk of extending the ILRT from 3 in 10 years to 1 in 15 years is small and estimated as being less than 1E-7. It is evaluated by considering the following elements:

1. The risk associated with the failure of the Containment due to a pre-existing containment breach at the time of core damage (Class 3 events).
2. The risk associated with liner corrosion that could result in an increased likelihood that containment over-pressurization events become LERF events.
3. The likelihood that improved visual inspections (frequency and quality) will be effective in discovering liner flaws that could lead to LERF.

These elements are discussed in detail below.

Pre-existing Containment Breach

The original submittal addressed Item 1. The submittal calculated the increase risk using a new CEOG methodology and a previously NRC-approved methodology. This supplement modifies, in Table 2, these values to reflect the recent update of the CCNPP Unit 1 PRA.

Table 2

Original Submitted with Updated Values

Method	LERF Increase	Person-rem/yr increase	Percentage Increase in Person-rem/yr
CEOG Method	5.4E-8	236	0.36%
NRC Approved Method	2.9E-7	19.4	0.24%

The numerical results for the previously-approved methodology shows an LERF increase that is greater than 1E-7. However, as noted in the original submittal, the calculated LERF would likely be lower than 1E-7 if conservatisms associated with the modeling of the steam generator tube rupture sequences were removed (note that this improvement was not incorporated into the modified values). In addition, the steam generators for Unit 1 are being replaced and should further reduce this likelihood.

Liner Corrosion

The original submittal also did not fully address the risk associated with liner corrosion. This supplement shows an additional small increase in LERF of 1E-8. Table 2 would be modified as follows:

Table 3

Updated Values with Corrosion Impact

Method	LERF Increase	Person-rem/yr increase	Percentage Increase in Person-rem/yr
CEOG Method	5.4E-8	236	0.36%
CEOG Method with Liner Corrosion	6.4E-8	250	0.38%
NRC-Approved Method	2.9E-7	19.4	0.24%
NRC-Approved Method with Liner Corrosion	3.0E-7	20.3	0.25%

Visual Inspections

The original submittal did not fully address the benefit of the Subsection IWE visual inspections. Visual inspections following the 1996 change in the ASME Code are believed to be more effective in detecting flaws. In addition, the flaws that are of concern for LERF are considerably larger than those of concern for successfully passing the ILRT. Integrated leakage rate test failures have occurred even though visual inspections have been performed. However, the recorded ILRT flaw sizes for these failed tests are much smaller than that for LERF. Therefore, it is likely that future inspections would be effective in detecting the larger flaws associated with a LERF.

An additional visual inspection is now planned for 2004 to further increase the likelihood for flaw detection.

Impact of Improved Visual Inspections

The raw data for both the CEOG method and the NRC-approved method is contained in NUREG-1493. This containment performance data is pre-1994. An amendment to 10 CFR 50.55a became effective September 9, 1996. This amendment, by endorsing the use of Subsections IWE and IWL of Section XI of the ASME B&PV Code, provides detailed requirements for ISI of Containment Structures. Inspection (which includes examination, evaluation, repair, and replacement) of the concrete containment liner plate, in accordance with the 10 CFR 50.55a requirements, involves consideration of the potential corrosion areas. Although the improvement gained by this requirement varies from plant to plant, it is believed that this requirement makes the detection of flaws post-September 1996 much more likely than pre-September 1996 using visual inspections.

Visual inspection improvements directly reduce the delta LERF increases as calculated in the CEOG method and NRC-approved method. The CCNPP Unit 1 Containment was visually inspected in 2000 and 2002. The Unit 1 containment is scheduled for inspection in 2004. This increased inspection frequency further reduces the delta LERF as calculated by both the CEOG and NRC-approved methods.

Table 7 illustrates the benefit of visual inspection improvements on the delta LERF calculations:

If the improved inspections (additional inspection, improved effectiveness, and larger flaw size) were 90% effective in detecting the flaws in the visible regions of the containment (5% for failure to detect and 5% for flaw not detectable [not-through-wall]), then the increase ILRT LERF frequency could be reduced by 23.5%. See Table 7 for additional sensitivity cases. This would result in a LERF increase of less than 1E-7 (without consideration of the LERF reduction due to PRA model improvements).

Sensitivity Studies

The following cases were developed to gain an understanding of the sensitivity of this analysis to the various key parameters.

Table 4

Liner Corrosion Sensitivity Cases

Age (Step 2)	Containment Breach (Step 4)	Visual Inspection & Non-Visual Flaws (Step 5)	Likelihood Flaw is LERF	LERF Increase
Base Case Doubles every 5 years	**Base Case** 1.1/0.11	**Base Case** 10%	**Base Case** 100%	**Base Case** 1E-8
Doubles every 2 years	Base	Base	Base	8E-8
Doubles every 10 years	Base	Base	Base	5E-9
Base	Base point 10 times lower (0.24/0.02)	Base	Base	2E-9
Base	Base point 10 times higher (4.9/0.49)	Base	Base	5E-8
Base	Base	5%	Base	6E-9
Base	Base	15%	Base	1E-8
Lower Bound				
Doubles every 10 years	Base point 10 times lower (0.24/0.02)	5%	10%	7E-11
Upper Bound				
Double every 2 years	Base point 10 times higher (4.9/0.49)	15%	100%	5E-7

Table 5

Flaw Failure Rate as a Function of Time

Year	Failure Rate (FR)	Success Rate (1-FR)
0	1.79E-03	9.98E-01
1	2.05E-03	9.98E-01
2	2.36E-03	9.98E-01
3	2.71E-03	9.97E-01
4	3.11E-03	9.97E-01
5	3.57E-03	9.96E-01
6	4.10E-03	9.96E-01
7	4.71E-03	9.95E-01
8	5.41E-03	9.95E-01
9	6.22E-03	9.94E-01
10	7.14E-03	9.93E-01
11	8.20E-03	9.92E-01
12	9.42E-03	9.91E-01
13	1.08E-02	9.89E-01
14	1.24E-02	9.88E-01
15	1.43E-02	9.86E-01

Table 6

Average Failure Rate

Years	Average Success Rate (SR)	Average Failure Rate (1-SR)
1 to 3	9.93E-1	0.71%
1 to 10	9.59E-1	4.06%
1 to 15	9.06E-1	9.40%

Δ = 9.40% - 0.71% = 8.7% (delta between 1 in 3 years to 1 in 15 years)

Table 7

Benefit of Visual Inspection Improvements

Factor Improvement due to Visual Inspections	Reduction in Delta LERF	NRC Approved Method Delta LERF	NRC Approved Method w/Liner Corrosion Considered Delta LERF	CEOG Method Delta LERF	CEOG Method w/Liner Corrosion Considered Delta LERF
Pre-1996 Inspection Approach (Base Case)	0%	3E-07	3E-07	5E-08	6E-08
Post-1996 with Visual Inspections Perfectly Accurate	85%	4E-08	5E-08	8E-09	2E-08
Post-1996 with Visual Inspections 95% Accurate	80.8%	6E-08	7E-08	1E-08	2E-08
Post-1996 with Visual Inspections 95% Accurate and 5% chance of Undetectable Leakage	76.5%	7E-08	8E-08	1E-08	2E-08
Post-1996 with Visual Inspections 80% accurate and a 5% Chance of Undetectable Leakage	63.8%	1E-07	1E-07	2E-08	3E-08

Conclusion

Considering increased frequency of visual inspections and the benefit of improved visual inspections post-1996, the increase in risk is considered to be less than 1E-7 for LERF. Changes less than 1E-7 are considered small per Regulatory Guide 1.174. The one-time extension of the ILRT interval from 3-in-10 years to 1-in-15 years is considered an acceptable risk increase.

Should you have questions regarding this matter, we will be pleased to discuss them with you.

Very truly yours,

STATE OF MARYLAND :
: **TO WIT:**
COUNTY OF CALVERT :

I, Charles H. Cruse, being duly sworn, state that I am Vice President - Nuclear Energy, Calvert Cliffs Nuclear Power Plant, Inc. (CCNPP), and that I am duly authorized to execute and file this License Amendment Request on behalf of CCNPP. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other CCNPP employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of Calvert, this 27th day of March, 2002.

WITNESS my Hand and Notarial Seal: Denise D. Inglis
Notary Public

My Commission Expires: 02/02/06
Date

CHC/DJM/dlm

Attachment: (1) Final Technical Specification Pages

cc: R. S. Fleishman, Esquire
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