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ATTN: Chief, Information Management Branch
Division of Program Management

Subject: Transmittal of WCAP-15691-NP (Non-Proprietary) Revision 5, "Joint Applications Report for Containment Integrated Leak Rate Test Interval Extension," March 2004

Ref.: 1. CEOG Letter, R. Bernier to NRC Document Control Desk, transmitting WCAP-15691 (Proprietary) and WCAP-15715 (Non-proprietary) for NRC Review, CEOG-01-184, July 11, 2001.

2. NRC Letter, D. Holland to G. Bischoff, "Request for Additional Information – WCAP-15691, Rev. 00, "Joint Applications for Containment Integrated Leak Rate Test Interval Extension', (TAC NO. MB6806)," June 16, 2003.

3. NRC Letter, D. Holland to G. Bischoff, "Status of the Review of WCAP-15691, "Joint Applications for Containment Integrated Leak Rate Test Interval Extension', (TAC NO. MB2554 and MB6806)," August 13, 2003.

4. EPRI TR-104285 Rev. 01, "Risk Assessment of Revised Containment Leak Rate Testing Intervals," October 2003.

By letter dated July 19, 2001, the Combustion Engineering Owners Group (CEOG) submitted WCAP-15691-P, Revision 0, "Joint Applications Report for Containment Integrated Leak Rate Test Interval Extension," for NRC review and approval (Ref. 1). On June 16, 2003, the NRC issued a Request for Addition Information (RAI) concerning WCAP-15691 (Ref.2). On August 13, 2003, the NRC notified the Westinghouse Owners Group (WOG) that the staff had stopped the review of WCAP-15691 until the latest revision is submitted (Ref. 3). Accordingly, please find enclosed the latest revision of WCAP-15691-NP (non-proprietary), Revision 5. This revision incorporates the responses to the NRC RAIs and utilizes risk-based expert elicitation methodologies as set forth in EPRI TR-104285 (Ref. 4). Specific responses to the NRC RAIs are provided in Attachment 1.

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The WOG hereby requests that the NRC resume the review of WCAP-15691 which provides the methodology and risk justification to extend, on a one-time basis, the Type A containment integrated leak rate test interval from 10 to 15 years for PWRs with large, dry containment structures. WCAP-15691 demonstrates that the increase in risk associated with extending the interval between Type A containment leak rate tests to 10, 15 and 20 years is very small when compared with the guidelines set forth in Regulatory Guide 1.174.

Please feel free to call Paul Hijeck at 860-731-6240 if you have any questions concerning this matter.

Sincerely yours,

Frederick. P. "Ted" Schiffley, II, Chairman
Westinghouse Owners Group

Enclosures: WCAP-15691-NP Rev. 5 (3 bound, 1 unbound)

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Attachment 1
WOG Responses to Request for Additional Information – WCAP-15691, Rev. 0, "Joint Applications for Containment Integrated Leak Rate Test Interval Extension'

Westinghouse has revised WCAP-15691, "Containment Integrated Leak Rate Test Interval Extension," to address concerns identified by the Staff through comments and requests for additional information. Responses to staff RAIs dated June 16, 2003 are documented below and are incorporated in Revision 5 of WCAP-15691. This revision of WCAP-15691 also incorporates EPRI methodology based on expert elicitation conducted by EPRI to extrapolate failure probabilities. The expert elicitation methodology is documented in EPRI topical report TR-104285 Rev 1, which has been submitted separately to the NRC.

Concerns 1 through 4

1. There is no statistical justification for using the tail probability of the log-normal distribution, or any other fitted distribution, to estimate the probability of a large leak. Because the largest observed leakage rate is 21 La and the leakage rate assumed for a large leak is >100 La, the calculated tail probability is extrapolated far beyond the observed data.

2. The parameters of the log-normal distribution fitted to the 23 observed leaks should have been estimated using the sample mean and standard deviation of the underlying normal distribution.

3. The weight that should be applied to the conditional probability of a large leak is 0.13 (= 23/180) but not 0.028 (= 5/180) as used. The correct weight is the estimated mixture fraction of the assumed log-normal distribution, which is the ratio of the observed number of leaks of 23, to which the log-normal distribution was fitted, to the total number of tests (180).

4. Using the conditional probability of a large leak from the fitted log-normal distribution of 0.006 and the corrected weight for the log-normal distribution, the probability of a large leak is estimated as 0.00077 (= 0.006*23/180). The corresponding confidence level of 13% is inappropriate for comparison against mean values.

Response to Concerns 1 through 4
Extrapolated tail failure probabilities are discussed in Section 5.2.3 of WCAP-15691. This report section has been revised to replace the statistical methodology with EPRI methodology based on "expert elicitation" conducted by EPRI. Details of the EPRI model are documented in TR-104285 Rev 1.

Concern 5

5. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy can be maintained if a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The increase in the conditional containment failure probability or a suitable alternative was not provided for the proposed change from a 1 in 10-year to a 1 in 15-year test interval or the cumulative change of going from a 3 in 10-year to a 1 in 15-year test interval.

Response to Concern 5

Section 5.3 of WCAP-15691 has been added to discuss defense in depth. An example plant evaluation is contained in Appendix X (See Section X3.0) which summarizes the impact on LERF for various increases in the test interval. The maximum increase in conditional containment unreliability is seen to be much less than 1%.

Concern 6

6. WCAP-15691 does not address corrosion events, which have been identified by visual examinations required by 10 CFR 50.55a, and how such events should be considered in the risk model. These events would include possible through-wall corrosion in the un-inspectable areas of the containment liner. Section 2.3 of RG 1.174 states that a monitoring plan should be developed. WCAP-15691 does not address such a monitoring plan nor does it address how indications identified as part of a licensee's 10 CFR 50.55a program would be considered as part of the applicable monitoring plan. An example is a through-liner indication that would have resulted in a failed Type A test had one been performed.

Response to Concern 6

Section 6.0 of WCAP-15691 has been added to describe a methodology for addressing corrosion events. The example plant evaluation contained in Appendix X (See Section X4.0) provides an evaluation of the potential impact of containment liner corrosion.

Concern 7

7. The topical report does not address PRA quality as discussed in Section 2.2.3.3 of RG 1.174.

Response to Concern 7

Section 5.4 of WCAP-15691 has been added to discuss PRA quality.

Westinghouse Non-Proprietary Class 3

WCAP-15691
Revision 5

March 2004

Joint Applications Report for Containment Integrated Leak Rate Test Interval Extension

WOG Task 2070

Westinghouse

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Westinghouse Non-Proprietary Class 3

WCAP 15691, Rev. 05

Joint Applications Report for Containment Integrated Leak Rate Test Interval Extension

WOG Task 2070

March 2004

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LIST OF ACRONYMS

ASME	American Society of Mechanical Engineers
CCNPP	Calvert Cliffs Nuclear Power Plant
CDF	Core Damage Frequency
CE	Combustion Engineering
CEOG	Combustion Engineering Owners Group
CET	Containment Event Tree
CIAS	Containment Isolation Actuation Signal
CILRT	Containment Integrated Leak Rate Test
CIV	Containment Isolation Valve
$F_{Class\ x}$	Frequency of Event Class x
FSAR	Final Safety Analysis Report
ILRT	Integrated Leak Rate Test
IPE	Individual Plant Examination
ISLOCA	Interfacing System Loss of Coolant Accident
ISTS	Improved Standard Technical Specifications
L_a	Containment Allowable Leak Rate
LDBA	Leakage Design Basis Accident
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
MSSV	Main Steam Safety Valve
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
ORNL	Oak Ridge National Laboratory
P_a	Internal Containment Pressure
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
TS	Technical Specification
UCL	Upper Confidence Limit
WOG	Westinghouse Owners Group
WSES	Waterford Steam Electric Station

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1.0 INTRODUCTION

The purpose of this report is to provide a risk informed methodology for justifying modification of the plant licensing basis for PWR containment Integrated Leak Rate Test (ILRT) intervals. The Owners Group previously requested that the methodology be reviewed and approved for a one time five year extension of the ILRT interval. (Reference letter CEOG-01-220 dated August 28, 2001 to NRC Document Control Desk, from R.A. Bernier, Chairman CE Owners Group).

This ILRT extension is sought to provide cost savings and increased plant availability by shortening refueling outages by approximately two critical path days. Justification of this ILRT modification is based on a review and assessment of plant operations, deterministic/design basis factors, and plant risk.

The ILRT extension was found to have a very small impact on the risk of events that may give rise to large early radionuclide releases. Therefore, any decrease in containment reliability due to the ILRT extension for the requested ILRT test interval modifications would result in a very small (negligible) impact on the large early release probability.

Specifically, the results of the evaluation provided herein demonstrate that the risk level associated with the proposed ILRT extension is below the regulatory guidelines set forth in Regulatory Guide 1.174 (Reference 3).

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2.0 SCOPE OF PROPOSED CHANGE

2.1 DEFINITION OF CONTAINMENT INTEGRATED LEAK RATE TEST

Containment structure testing is intended to assure leak-tight integrity of the containment structure under all design basis conditions. Containment leakage test methods include Integrated Leakage Rate Tests (ILRTs or Type A tests) and local leakage rate tests (LLRTs or Type B and Type C tests). The intention of this report is to justify modifying the test interval for Type A ILRT testing.

Type A tests are performed by pressurizing the primary containment to an internal pressure (P_a) derived from the Leakage Design Basis Accident (LDBA) and specified in the unit technical specifications or associated bases. The primary containment system is aligned, as closely as practical, to the configuration that would exist following a LDBA (e.g. systems are vented, drained, flooded, or in operation, as appropriate). At pressure P_a, the actual containment leakage rate (L_a) is derived from measurements. The derived leakage rate is expressed in percent per 24 hours by weight of the containment normal air inventory, with the leakage taking place at P_a. The parameters actually measured are pressure, temperature and humidity. Utilizing the Ideal Gas Law and placing a statistical boundary on the leakage rate calculated at 95% probability or upper confidence limit, a true leakage rate is calculated.

Type A tests measure very small leakage rates and require approximately two days of critical path time to complete.

2.2 PROPOSED EXTENSION OF ILRT INTERVAL

This report provides justifications for an extension in the containment ILRT interval from 10 years to 20 years. This is consistent with the conclusions of NUREG-1493 (Reference 4), Performance-Based Containment Leak-Test Program. NUREG-1493 conclusions are that "Reducing the frequency of Type A tests (ILRTs) from three per 10-year period to one per 20 years was found to lead to an imperceptible increase in risk."

The risk calculations included in this evaluation consider all significant impacts of the ILRT test interval modification, including:

- Change in Large Early Release Frequency
- Total impact in terms of change in person-rem/year.
- Altering the ILRT test interval has no impact on Core Damage Frequency (CDF)

The supporting analytical material contained within this document is considered applicable to PWRs with large dry containments, including all CE NSSS designed units.

For some plants, implementation of the ILRT interval change will require a change to the plant's Technical Specifications or other Licensing document. For other plants, the change can be made to administrative documents which define the approved ILRT interval.

3.0 BACKGROUND

This report provides a risk-informed technical basis for extending the containment integrated leak rate test interval. This change is warranted based on the low risk associated with the extended ILRT. This application is being pursued by the Westinghouse Owners Group as a risk informed plant modification in accordance with NRC Regulatory Guide 1.174, (Reference 3).

Implementation of the ILRT extension will save utilities approximately two critical path days per outage where an ILRT is performed, with a resulting savings in excess of $300,000 per day. This saving will be realized with negligible public risk impact.

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4.0 SYSTEM DESCRIPTION AND OPERATING EXPERIENCE

4.1 SYSTEM DESCRIPTION

The primary function of containment is to prevent the release of radioactive material from either the containment atmosphere or the reactor coolant system to the outside environment. The appendices to this report contain plant specific descriptions of the containment systems.

4.2 OPERATING EXPERIENCE

NUREG-1493, Performance-Based Containment Leak-Test Program, determined that, "In approximately 180 ILRT reports considered in this study, covering approximately 770 years of operating history, only five ILRT failures were found which local leakage-rate testing could not and did not detect. These results indicate that Type A testing detected failures to meet current leak-tightness requirements in approximately 3 percent of all tests. These findings clearly support earlier indications that Type B and C testing can detect a very large percentage of containment leakages. The percentage of containment leakages that can be detected only by integrated containment leakage testing is very small. Of note, in the ILRT failures observed that were not detected by Type B and C testing, the actual leakage rates were very small, only marginally in excess of the current leak-tightness requirements."

The current surveillance testing requirements, as outlined in NEI 94-01 (Reference 1) for Type A testing, is at least once per 10 years based on an acceptable performance history (define as two consecutive Type A tests at least 24 months apart in which the calculated performance leakage was less than $1.0L_a$). The appendices to this report discuss plant specific operating experience.

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5.0 ASSESSMENT OF RISK

The purpose of this Section is to provide a risk-informed assessment for extending a plant's Integrated Leak Rate Test (ILRT) interval from 10 to 20 years. The risk assessment is consistent with the methodologies set forth in NEI 94-01 (Reference 1), the methodology used in EPRI TR-104285 (Reference 2), Revision 1 of EPRI TR-104285 (Reference 7) and the NRC guidance in NUREG-1493 (Reference 4). In addition, the methodology incorporates Probabilistic Risk Assessment (PRA) findings and risk insights in support of risk informed licensee requests for changes to a plant's licensing basis, Regulatory Guide 1.174 (Reference 3).

Specifically, this approach combines the plant's PRA results and findings with the methodology described in EPRI TR-104285 to estimate public risk associated with extending the containment Type A test interval.

The change in plant risk is evaluated based on the change in the predicted releases in terms of person-rem/year and Large Early Release Frequency (LERF). Changes to Type A testing have no impact on plant CDF.

5.1 OVERVIEW

In October 26, 1995, the NRC revised 10 CFR 50, Appendix J. The revision to Appendix J allowed individual plants to select containment leakage testing under Option A "Prescriptive Requirements" or Option B "Performance-Based Requirements." Individual WOG members have selected the requirements under Option B as their testing program.

The current surveillance testing requirement, as outlined in NEI 94-01 (Reference 1) for Type A testing, is at least once per 10 years based on an acceptable performance history (define as two consecutive Type A tests at least 24 months apart in which the calculated performance leakage was less than $1.0L_a$). Experience has not shown these tests as being needed for identifying containment leakages, with more than 97% of all containment leakages in excess of L_a being identified by local tests. As a result of the small benefit, the risk impact of extending this test interval from 10 to 20 years will be negligible. This Section provides the risk assessment methodology for assessing the risk significance of this surveillance test interval change. Analysis presented in the following paragraphs is consistent with the NRC methodology used for their initial Appendix J change and considers risk impact in accordance with Regulatory Guide 1.174.

5.2 RISK ASSESSMENT METHODOLOGY

The risk of extending the ILRT interval for Type A tests from its current interval of 10 years to 20 years, is evaluated for potential public exposure impact (as measured in person-rem/year) and for impact on Large Early Release Frequency (LERF) as identified in Regulatory Guide 1.174 (Reference 3). The analysis employs a simplified approach similar to that presented in EPRI TR-104285 (Reference 2) and NUREG-1493 (Reference 4). The methodology explicitly accounts for large releases and specifically computes the LERF metric. The analysis performed examines each plant's IPE and subsequent PRA upgrades for plant specific accident sequences which may impact containment performance.

In the EPRI/NRC approaches, the core damage events are binned into eight containment classes including two intact containment states; one with containment leakage less than L_a, and one with containment leakage in excess of L_a. It is assumed that extending the ILRT will increase the likelihood of containment states with excess leakage. This Section contains an evaluation of the magnitude of the increase in probability of core damage events with significant containment leakage. This evaluation is performed using the methodology described below. The methodology for the risk calculations is summarized in Sections 5.2.1 through 5.2.4. These sections are divided as follows:

Section 5.2.1 defines the containment failure frequency and associated releases for each of eight accident classes used in this evaluation.

Section 5.2.2 develops the plant specific dose (population dose) per reactor year.

Section 5.2.3 provides an evaluation of the risk impact of extending Type A test interval from 10 years to 15 and 20 years.

Section 5.2.4 evaluates the risk impact of extending the Type A test interval based on the change in risk in terms of Large Early Release Frequency (LERF), in accordance with Regulatory Guide 1.174 (Reference 3)

5.2.1 Methodology for Assessment of Accident Class Frequency and Releases

Extension of the Type A interval does not influence those accident progressions that involve containment isolation failures associated with Type B or Type C testing or containment failure induced by severe accident phenomena. The CET containment isolation models are reviewed for applicable isolation failures and their impacts on the overall plant risk. Specifically, a simplified model to predict the likelihood of having a small or large pre-existing breach in the containment, that is undetected due to the extension of the Type A ILRT test interval, is developed.

For this present work, the EPRI accident Class designations (Reference 2) are used to define the spectrum of plant releases. Following the EPRI approach, the intact containment event was modified to include the probability of a pre-existing containment breach at the time of core damage. Two additional basic events are addressed. These are Event Class 3A (small leak) and Event Class 3B (large leak). (This addresses the 'Class 3' sequence discussed in EPRI TR-104285). Both event Class 3A and 3B are considered in estimating the public exposure impact of the ILRT extension. However, since leaks associated with event Class 3A are small (that is, marginally above normal containment leakage), only event Class 3B frequency change is considered in bounding the LERF impact for the proposed change.

The eight EPRI accidents Classes are discussed in the following paragraphs.

Class 1 Sequences: This sequence class consists of all core damage accident progression bins for which the containment remains intact with negligible leakage. Class 1 sequences arise from those core damage sequences where containment isolation is successful, and long term containment heat removal capability is available via containment sprays or fan coolers. The frequency of an intact containment is established based on the individual plant's PRA. For Class 1 sequences, it is assumed that the intact containment end state is subject to a containment leakage rate less than the containment allowable leakage (L_a). To obtain the Class 1 event frequency, intact containment events are parsed into three classes: Class 3A, Class 3B and Class 1. Class 1 represents containments with expected leakages less than L_a. Class 3A represents intact containments with leakages somewhat larger than L_a, and Class 3B represents intact containment endstates with large leaks.

The frequency for Class 1 events is related to the intact containment core damage frequency (CDF_{Intact}) and the Class 3 categories, as follows.

$$F_{Class\ 1} = CDF_{Intact} - F_{Class\ 3A} - F_{Class\ 3B}$$

Where:
CDF_{Intact} = the Core Damage Frequency for intact containment sequences from the plant specific PRAs.

The calculation of Class 3 frequencies is discussed below. Radiological releases for Class 1 sequences are established assuming a containment leakage rate equal to the design basis allowable leakage (L_a).

Class 2 Sequences: This group consists of all core damage accident progression bins for which a pre-existing leakage due to failure to isolate the containment occurs. These sequences are dominated by failure-to-close of large (>2-inch diameter) containment isolation valves. The frequency per year for these sequences is determined from the plant specific PRAs as follows:

$$F_{Class\ 2} = PROB_{large\ CI} * CDF_{Total}$$

Where:
$PROB_{large\ CI}$ = random containment large isolation failure probability (i.e. large valves), and
CDF_{Total} = Total plant specific CDF.

This value is obtained from plant specific PRAs.

For this analysis, the associated maximum containment leakage for this group is estimated at approximately 100 wt% per day (See Table 5-1).

Class 3 Sequences: Class 3 endstates are developed specifically for this application. The Class 3 endstates include all core damage accident progression bins with a pre-existing leakage in the containment structure in excess of normal leakage. The containment leakage for these sequences can be grouped into two categories: small leakage, or large.

The respective frequencies per year are determined as follows:

$F_{Class\ 3A} = PROB_{Class\ 3A} * CDF_{Intact}$

$F_{Class\ 3B} = PROB_{Class\ 3B} * CDF_{Intact}$

Where:
$PROB_{Class\ 3A}$ = the probability of small pre-existing containment leakage in excess of design allowable but less than 100 L_a. $PROB_{Class\ 3A}$ is presented as a function of ILRT test interval in Table 5-5, in Section 5.2.3,
$PROB_{Class\ 3B}$ = the probability of large (>100 L_a) pre-existing containment leakage. $PROB_{Class\ 3B}$ is a presented as a function of ILRT test interval in Table 5-5, in Section 5.2.3, and
CDF_{Intact} = the Core Damage Frequency for intact containment sequences from the plant specific PRAs.

No ILRT has identified a pre-existing leakage in excess of 21 L_a (See Section 5.2.3). The 21 L_a leakage was identified by an LLRT and thus would not have gone undetected even if the ILRT were not performed, and a 15 L_a discovery was a result of failed LLRT which would have been picked up during the next test (Reference 7). Class 3A releases were established based on ILRT testing history which indicated that approximately 3% of ILRT identified leaks may not have been picked up by LLRTs. Class 3A releases are conservatively estimated based on a leakage rate of 25 L_a. Class 3B release frequencies are approximated, consistent with Reference 7, as a containment leakage $\geq$ 100 L_a. For the purpose of computing public dose, Class 3B radiological releases are conservatively based on leakages of 1000 L_a. This corresponds to an equivalent containment leakage of about 6 in^2 (See Figure 5-1).

<u>Class 4 Sequences:</u> This group consists of all core damage accident progression bins for which a failure-to-seal containment isolation failure of Type B test components occurs. Because these failures are detected by Type B tests, and their frequency is very low compared with the other classes, this group is not evaluated any further. The frequency for Class 4 sequences is subsumed into Class 7 where it contributes insignificantly.

<u>Class 5 Sequences:</u> This group consists of all core damage accident progression bins for which a failure-to-seal containment isolation failure of Type C test components occurs. Because these failures are detected by Type C tests, and their frequency is very low compared with the other classes, this group is not evaluated any further. The frequency for Class 5 sequences is subsumed into Class 7 where it contributes insignificantly.

<u>Class 6 Sequences:</u> This group is similar to Class 2. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage, due to failure to isolate the containment, occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution typically resulting in a failure to close smaller containment isolation valves. All other failure modes are bounded by the Class 2 assumptions.

The frequency per year for these sequences is determined as follows:

$$F_{Class\ 6} = PROB_{largeT\&M} * CDF_{Total}$$

Where:
$PROB_{largeT\&M}$ = probability of random failure of containment to isolate due to valve misalignment (failure modes not otherwise include in Class 2).
CDF_{Total} = the Total plant specific CDF.

For this analysis the associated maximum containment leakage for this group is 35 wt%/day.

<u>Class 7 Sequences:</u> This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs (i.e. H_2 combustion, direct containment heating, etc.).

$$F_{Class\ 7} = CDF_{CFL} + CDF_{CFE}$$

Where:
CDF_{CFE} = the CDF resulting from accident sequences that lead to early containment failure, and
CDF_{CFL} = the CDF resulting from accident sequences that lead to late containment failure.

$F_{Class\ 7}$ can be determined by subtracting the intact, bypass (See Class 8 discussion) and loss of isolation CDFs from the total CDF.

These endstates include containment failure. For this analysis the associated containment leakage for this group is 280 wt%/day (based on 0.1 ft^2 failure, see NUREG 1493)

<u>Class 8 Sequences:</u> This group consists of all core damage accident progression bins in which containment bypass occurs. Each plant's PRA is used to determine the containment bypass contribution. Contributors to bypass events include ISLOCA events and SGTRs with an un-isolated steam generator.

$$F_{Class\ 8} = CDF_{ISLOCA} + CDF_{Unisolated\ SGTR}$$

The magnitude of bypass releases is plant specific and is typically considerably larger (two or more orders of magnitude) than releases expected for leakage events. The containment structure will not impact the release magnitude for this event class.

Table 5-1 summarizes the methodology for determining the event class frequency and associated releases.

Table 5-1
Mean Containment Frequency Measures and Representative Releases - by Accident Class

Class	Description	Frequency Relationships (NOTE 1)	Estimated Leakage
1	No Containment Failure	$F_{Class\ 1} = CDF_{Intact} - F_{Class\ 3A} - F_{Class\ 3B}$	L_a
2	Large Containment Isolation Failures (failure-to-close)	$F_{Class\ 2} = PROB_{large\ CI} * CDF_{Total}$	100 wt%/day
3A	Small Pre-existing Containment Leak	$F_{Class\ 3A} = PROB_{Class\ 3A} * CDF_{Intact}$ (Note 2)	25 L_a
3B	Large Pre-existing Containment Leak	$F_{Class\ 3B} = PROB_{Class\ 3B} * CDF_{Intact}$ (Note 2)	100 wt%/day
4	Small isolation failure - failure-to-seal (Type B test)	Not Analyzed	
5	Small isolation failure - failure-to-seal (Type C test)	Not Analyzed	
6	Containment Isolation Failures (dependent failures, personnel errors)	$F_{Class\ 6} = PROB_{largeT\&M} * CDF_{Total}$	35 wt%/day
7	Severe Accident Phenomena Induced Failure (early and late failures)	$F_{Class\ 7} = CDF_{CFL} + CDF_{CFE}$	280 wt%/day
8	Containment Bypassed (ISLOCA, SGTR with stuck open MSSVs)	$F_{Class\ 8}$ = [Plant Specific]/year	PRA defined large release
Total	All CET Endstates	From PRA (Sum of Classes 1 through 8)	

Note 1 - Parameters for an example plant are summarized in Table 5-2.
Note 2 - $PROB_{Class\ 3A}$ and $PROB_{Class\ 3B}$ are ILRT interval specific and are summarized in Table 5-5.

The appendices to this report include determination of the plant specific frequencies for each event class. Table 5-2 summarizes the plant specific frequencies for each event class for participating PWRs (See for example Appendix X).

Table 5-2
Event Class Frequencies (per year) for Example Plant – Baseline ILRT Interval

Class	Description	Example Plant[1]
1	No Containment Failure	4.73E-06
2	Large Containment Isolation Failures (failure-to-close)	7.43E-08
3A	Small Pre-existing Containment Leak	1.36E-07
3B	Large Pre-existing Containment Leak	1.20E-09
4	Small isolation failure – failure-to-seal (Type B test)	NA
5	Small isolation failure – failure-to-seal (Type C test)	NA
6	Containment Isolation Failures (dependent failures, personnel errors)	0.00E+00
7	Severe Accident Phenomena Induced Failure (early and late failures)	5.90E-06
8	Containment Bypassed (ISLOCA, SGTR with stuck open MSSVs)	2.53E-06
Total	All CET Endstates	1.34E-05

Note 1 – Values for the Example Plant are obtained from Appendix X.

5.2.2 Methodology for the Calculation of Plant Specific Population Dose (per reactor year)

Plant-specific release analyses are performed to evaluate the whole body dose to the population, within a 50-mile radius from the plant. The releases are based on the large Loss-Of-Coolant Accident (LOCA) associated with the maximum hypothetical accident.

The population dose is estimated assuming leakages for accident Classes are as defined in Table 5-1.

Since the containment release pathways are generally the same for containment Classes 1 through 7, doses are directly proportional to the ratio of the leakage rate to the nominal leakage value. Therefore, the Class 2 through 7 leakage related doses are ratioed upwards to account for the particular increased leakages associated with event Classes 2 through 7. In this methodology, the Class 1 leakage is represented by REL_{Intact}. Table 5-3 presents the releases for each class as a function of REL_{Intact} and L_a. Class 8 events are represented by bypass releases based on iodine and noble gas releases identified in the PRA for the dominant sequence. The population estimate can be based on FSAR siting projections.

The assessment of containment leakages for Classes 1 through 8 and associated releases are defined in Table 5-3. Intact containment release (REL_{Intact}) for Class 1 events and bypass releases for Class 8 events are obtained from plant specific assessments. Plant specific containment releases are summarized in Table 5-4.

Table 5-3
Containment Leakage Rates and Doses – for Accident Classes

Class	Description	Leakage (wt%/day)	Release (50 miles) (person-rem)	Basis
1	No Containment Failure	L_a (1)	REL_{Intact}	See Section 5.2.1
2	Large Containment Isolation Failures (failure-to-close)	100	$(100/L_a)^* REL_{Intact}$	Ratio from Class 1 baseline
3A	Small Pre-existing Containment Leak	25 L_a	$25^* REL_{Intact}$	Ratio from Class 1 baseline
3B	Large Pre-existing Containment Leak	100	$(100/L_a)^* REL_{Intact}$	Ratio from Class 1 baseline
4	Small isolation failure - failure-to-seal (Type B test)	Not analyzed	N/A	Ratio from Class 1 baseline
5	Small isolation failure - failure-to-seal (Type C test)	Not analyzed	N/A	Ratio from Class 1 baseline
6	Containment Isolation Failures (dependent failures, personnel errors)	35	$(35/L_a)^* REL_{Intact}$	Ratio from Class 1 baseline
7	Severe Accident Phenomena Induced Failure (early and late failures)	280	$(280/L_a)^* REL_{Intact}$	Ratio from Class 1 baseline
8	Containment Bypassed (ISLOCA, SGTR with stuck open MSSVs)	-	Plant Specific	No credit for containment

Note 1 - L_a is a Plant Specific parameter, typically 0.1 or 0.5 wt%/day.

Table 5-4, below, provides a summary of the plant specific releases for each of the eight event classes.

Table 5-4
Event Class Releases for Example Plant (person-rem - within 50 miles)

Class	Description	Example Plant[1]
1	No Containment Failure	3.77E+03
2	Large Containment Isolation Failures (failure-to-close)	1.06E+06
3A	Small Pre-existing Containment Leak	9.43E+04
3B	Large Pre-existing Containment Leak	3.77E+06
4	Small isolation failure – failure-to-seal (Type B test)	NA
5	Small isolation failure – failure-to-seal (Type C test)	NA
6	Containment Isolation Failures (dependent failures, personnel errors)	NA
7	Severe Accident Phenomena Induced Failure (early and late failures)	1.65E+05
8	Containment Bypassed (ISLOCA, SGTR with stuck open MSSVs)	2.54E+06

Note 1 – Values for Example Plant are obtained from Appendix X.

The above results can be combined with the class frequency results presented in Table 5-2 to yield the plant specific baseline mean risk measure for each accident class (calculated as the product of the frequencies in Table 5-2 and the releases in Table 5-4). The resulting doses for the i^{th} Class are represented by the parameter $Risk_{Class\ i}$.

Risk Contribution of Classes 1 and 3

In order to evaluate the impact of an ILRT extension on incremental doses, it is necessary to investigate the change in the expected doses on the "intact" containment classes. While other sequences contribute more significantly to risk, the other sequences are insensitive to changes in ILRT intervals.

Based on the parameters defined above, the percent risk contribution associated with the "intact" containment sequences for Class 1 and Class 3 (%Risk) is as follows:

$$\%Risk = [(Risk_{Class\ 1} + Risk_{Class\ 3A} + Risk_{Class\ 3B}) / Total] \times 100$$

Where:
$Risk_{Class\ 1}$ = Class 1 person-rem/year
$Risk_{Class\ 3A}$ = Class 3A person-rem/year
$Risk_{Class\ 3B}$ = Class 3B person-rem/year

Total = total person-rem/year

Thus, the total risk contribution of leakage, represented by Class 1 and Class 3 accident scenarios can be determined for the baseline ILRT interval (the 3 per 10 year ILRT interval that is represented in the PRA), the current 10 year ILRT interval, and for 15 and 20 year ILRT intervals. All of the parameters in the above equation are dependent on the ILRT interval.

5.2.3 Methodology for Evaluation of Risk Impact of Extending Type A Test Interval From 10 To 15 and 20 Years

In order to calculate the impact of the change in the ILRT interval, it is first necessary to define the probability that a Type A leakage test is required to detect a containment leak. This probability is then adjusted to account for the proposed change in testing interval.

Probability of ILRT Leak Detection

NUREG-1493 (Reference 4) states that a review of experience data finds that a review of approximately 180 ILRT Type A tests identified 5 leaks that would not otherwise be identified by the more frequent local leak tests (Types B and C). That is, approximately 3% (0.028) of containment leakage events would not be identified without a Type A ILRT. In all instances, the detected leaks exhibited leak rates marginally in excess of the design basis allowable leakage. Therefore the probability of finding a small Type A leak (Class 3A) at a given Type A ILRT test is 0.028.

This probability is based on a testing frequency of three tests over a ten-year period and is used to define the baseline for the analysis. A once per ten-year frequency is currently employed at most WOG plants. Therefore, it is necessary to adjust the baseline probability (0.028) to reflect the current testing interval, and alternative testing intervals.

Probability of ILRT Identifying a Large Leak

The data in Reference 4 indicates that in the conduct of the ILRTs discussed above, 23 leaks were detected; the largest leak was 21 L_a, the second largest was 15 L_a, and the third largest was less than 3 L_a. From this it is apparent that given a leak, the probability that the leak is a large leak is very small. In order to estimate the conditional probability that a given leak is a large leak, EPRI sponsored an expert elicitation to estimate the mean probability of occurrence of leaks exceeding various sizes. The expert elicitation determined the cumulative probability of pre-existing containment leaks. This work is documented in EPRI TR-104285, Revision 1 "Risk Assessment of Revised Containment Leak Rate Testing Intervals," October 2003 (Reference 7).

For frequency estimation purposes a large leak is conservatively defined to be 100 L_a. The result of the expert elicitation is an estimate that leaks of that size or larger have a probability of 2.47E-04. Therefore, the probability of finding a large Type A leak (Class 3B) at a given Type A ILRT test is 2.47E-04. Defining a large release as 100 L_a is a conservative position, as LERF releases are on the order of 1000 L_a. Use of the 100 L_a assumption conservatively over estimates the LERF contribution by a factor of about 60.

A once per ten-year frequency is currently employed at most WOG plants. Therefore, it is necessary to adjust the baseline probability (0.028) to reflect the current testing interval, and alternative testing intervals.

For purposes of dose estimation, large leak releases are conservatively based on a containment leakage of 100 wt% per day. The L_a multiplier will vary from plant to plant as the allowable leakage (L_a) will range from 0.1 wt% to 0.5 wt% and the risk impact of this leakage will vary. The impact of leakage on event consequences is presented in Figure 5-2. Note that leakages of the order of 100 L_a (or 10 wt% per day for a containment with a 0.1 wt% per day leakage) have a 1% impact on risk.

Impact of Test Interval Extension on Leak Probabilities

The same process as described above for the three tests per ten-year case is applied for the current interval of once per 10 years, and for 15 and 20-year intervals.

The impact of relaxing the Type A test interval will increase the average time that a leak, that could only be detected by the Type A test, could possibly be present. The increase is proportional to the increase in duration between containment tests. The historical data is based on testing three times per ten years. This equates to a mean time between tests of 3.3 years or 40 months. The current test interval is 10 years (120 months). The increase in exposure time will influence the probability of leakage. To calculate this impact, two assumptions are made: a constant rate for Type A leakage events, and the potential for leakage is equally distributed across the period of interest such that the average exposure time is one-half the test interval.

The increased probability can be determined as the ratio of the proposed to the prior exposure times multiplied by the known rate for the prior probability of failure. For the current ten year ILRT interval, the equation is:

$$P_{10} = P_{10/3}[(0.5Exp_{10}/0.5Exp_{10/3})]$$

Substituting for $P_{10/3}$ (0.028) and for the exposure times, Exp_{10} =120, and $Exp_{10/3}$ = 40, yields a value for the probability of leakage of 0.084. This value represents the likelihood of Type A leakage given a 10-year testing interval.

The proposed ILRT interval extensions would increase the duration between tests by increasing the time between tests from 10 years to 15 or 20 years. Therefore the total time between Type A testing will increase from 10 years (120 months) to 15 years (180 months) or 20 years (240 months). The above equation is used with these new values:

$$P_{15} = P_{10}\ [(0.5Exp_{15}/0.5Exp_{10})]$$

$$P_{20} = P_{10}\ [(0.5Exp_{20}/0.5Exp_{10})]$$

The same method was used to determine the probability of a small leak and of a large leak, as a function of ILRT test interval. Substituting yields the values shown in Table 5-5.

Table 5-5
Probability of Type A Leakage for a Given Test Interval

Test Interval	Probability	
	Small Leak (Class 3A) ($PROB_{Class\ 3A}$)	Large Leak (Class 3B) ($PROB_{Class\ 3B}$)
3 per 10 Years	0.028	2.47E-4
10 Years	0.084	7.41E-4
15 Years	0.126	1.11E-3
20 Years	0.168	1.48E-3

Definition of Large Leak

No large leaks have occurred. The largest reported leak rate out of the 23 'failures' identified in the NUMARC list in NUREG-1493 (Reference 4), was 21 times the allowable leakage rate (L_a). 21 L_a (or from 2.1 to 10.5 wt% per day) does not constitute a large release.

For the purpose of this calculation, a large leak is assumed to result in a containment failure with a leak rate of >100 L_a per day.

Risk Impacts due to Test Interval Extensions

Contribution of Class 1 and 3 to Risk - Type A tests impact only Class 1 and Class 3 sequences. The increased probability of not detecting excessive leakage does not increase the frequency of occurrence for Class 1 sequences. In fact, the frequency of occurrence decreases by the same amount that Class 3 frequency of occurrence increases. For Class 3 sequences, the frequency increases in proportion to the leak probabilities shown in Table 5-5.

Note that the release magnitude of a class is not impacted by the change in test interval. That is, the magnitude of a small leak remains the same, even thought the probability of not detecting the leak increases.

Thus, the only parameters that change for calculating the risk impacts of an N-year interval versus the baseline interval (3 per 10 year testing interval), are the frequencies for Class 1 and Class 3 events.

The impact of the interval extensions on the frequencies of Class 1, 3A and 3B events are presented in Table 5-6. Frequency values are shown for the initial baseline of 3 inspections in 10 years (3/10), the current once per ten years (1/10) and for once in 15 years (1/15) and once in 20 years (1/20).

Table 5-6
Mean Event Class Frequencies for Various ILRT Intervals
(Intact Sequences – events/yr)

Plant	ILRT Interval	$F_{Class\ 1}$	$F_{Class\ 3A}$	$F_{Class\ 3B}$	Total(1 and 3)
Example Plant	3/10	4.73E-6	1.36E-7	1.20E-9	4.87E-6
Example Plant	1/10	4.45E-6	4.09E-7	3.61E-9	4.87E-6
Example Plant	1/15	4.25E-6	6.13E-7	5.41E-9	4.87E-6
Example Plant	1/20	4.04E-6	8.18E-7	7.21E-9	4.87E-6

The impact of the interval extensions on Class 1, 3A and 3B doses, and the % risk impact of the intact sequences is presented in Table 5-7. The appendices to this report include determination of the plant specific risk measures for each event class. Table 5-7 summarizes the plant specific risk measures for each event class. Table 5-7 shows how risk contribution of Class 1 and Class 3 events changes as a function of ILRT interval for the Example Plant. Risk and %Risk values are shown for the initial baseline of 3 inspections in 10 years (3/10), the current once per ten years (1/10) and for once in 15 years (1/15) and once in 20 years (1/20).

Table 5-7
Mean Event Class Risk Measures for various ILRT Intervals
(Intact Sequences, person-rem/year)

Plant	ILRT Interval	$Risk_{Class\ 1}$	$Risk_{Class\ 3A}$	$Risk_{Class\ 3B}$	Total	% Risk
Example Plant	3/10	0.0178	0.0128	0.00453	7.51	0.47
Example Plant	1/10	0.0168	0.0385	0.0136	7.55	0.91
Example Plant	1/15	0.0160	0.0578	0.0204	7.57	1.24
Example Plant	1/20	0.0152	0.0771	0.0272	7.60	1.57

Note that the methodology for computing %Risk is defined in Section 5.2.2.

Increase in Total Risk vs. Baseline Interval - The percent risk increase ($\%\Delta Risk_N$) due to an N-year ILRT over the baseline case is as follows:

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

Where:
$Total_{BASE}$ = total person-rem/yr for baseline test interval
$Total_N$ = total person-rem/yr for N-year test interval

Thus, we can determine the total increase in risk contribution associated with relaxing the ILRT test frequency.

Table 5-8 shows %ΔRisk$_N$ as a function of ILRT interval for the Example Plant. %ΔRisk values are shown for the initial baseline of 3 inspections in 10 years (3/10), the current once per ten years (1/10) and for once in 15 years (1/15) and once in 20 years (1/20).

Table 5-8
Percent Change in Total Risk for ILRT Interval Extensions

Plant	ILRT Interval	%ΔRisk
Example Plant	from 3/10 to 1/10	0.45 %
Example Plant	from 1/10 to 1/15	0.34 %
Example Plant	from 1/10 to 1/20	0.67 %

Note that the methodology for computing %ΔRisk is described above.

5.2.4 Methodology for Evaluating Change in Risk in Terms of Large Early Release Frequency (LERF)

Regulatory Guide 1.174 (Reference 3) provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as those resulting in increases of Core Damage Frequency (CDF) of less than 1.0E-6/yr and increases in LERF of less than 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 test interval on the large leakage probability.

<u>Quantification of LERF</u>

Justifying the extension of the Type A test interval requires establishing the success criteria for a large release. The criteria are based on:

1) The containment leak rate versus breach size, and
2) The impact of leak rate on risk.

Type A tests have typically been used in the past to identify containment leaks that are on the order of the diameter of a quarter inch or less. An approximate assessment of the effect of containment leak size on the containment leak rate is presented in Figure 5-1. The assessment assumes that leakage occurs as a result of critical flow of a steam-air mixture from the containment through variously sized leak areas. The actual leak rate for a given containment failure is dependent on containment volume and assumptions regarding the specific constituents in the containment atmosphere. In addition, Oak Ridge National Laboratory (ORNL) (Reference 5) completed a study evaluating the impact of leak rates on public risk using information from WASH-1400 (Reference 6) as the basis for its risk sensitivity calculations (See Figure 5-2).

It is judged that small leaks resulting from a severe accident (those that are deemed not to dominate public risk) can be defined as those leaks that have a weighted impact of less than 5%. In general, this suggests that containment leaks of about 35 wt% per day are not dominant contributors to public risk. For this assessment, large releases are assumed to occur when the leakage rate exceeds 100 L_a (or 10 to 50 wt%/day depending on the plant L_a).

The actual 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 (intact containment with a leakage of $< L_a$) could in fact result in a large release due to failure to detect a pre-existing leak during the relaxation period. Experience indicates that leaks not detected by Type B or C (LLRT tests) are both infrequent and of low magnitude. Therefore, for this evaluation, only Class 3 sequences will have the potential to impact risk as a result of the inability to detect a containment leak. Class 3A events would increase the leakage a marginal amount. Class 3B events are those for which the containment release may be conservatively considered to be large. Class 1 sequences are not large release pathways because the containment leak rate is expected to be small (on the order of L_a). It should be noted that, in estimating the ΔLERF, only changes to Class 3B events will effect a change in the LERF metric. However, for the purpose of this evaluation, the baseline LERF consists of contributions due to Classes 2, 3B and 6, 7 (early release portion, assumed to be half the total), and 8.

Figure 5-1
Evaluated Impact of Containment Leak Size on Containment Leak Rate

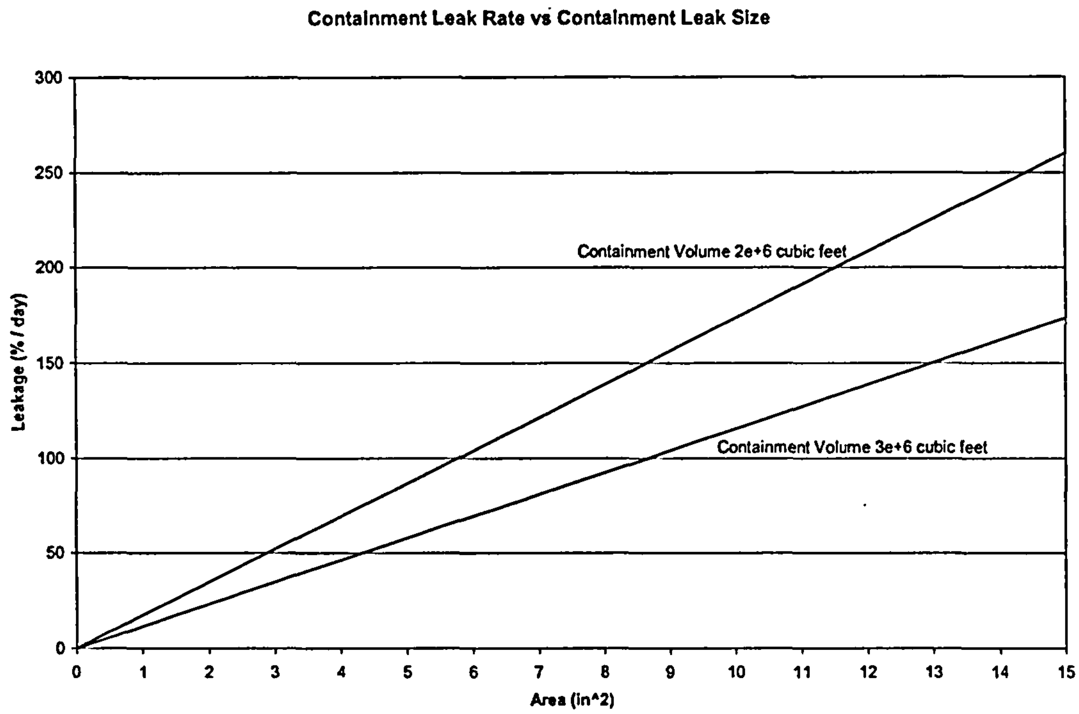
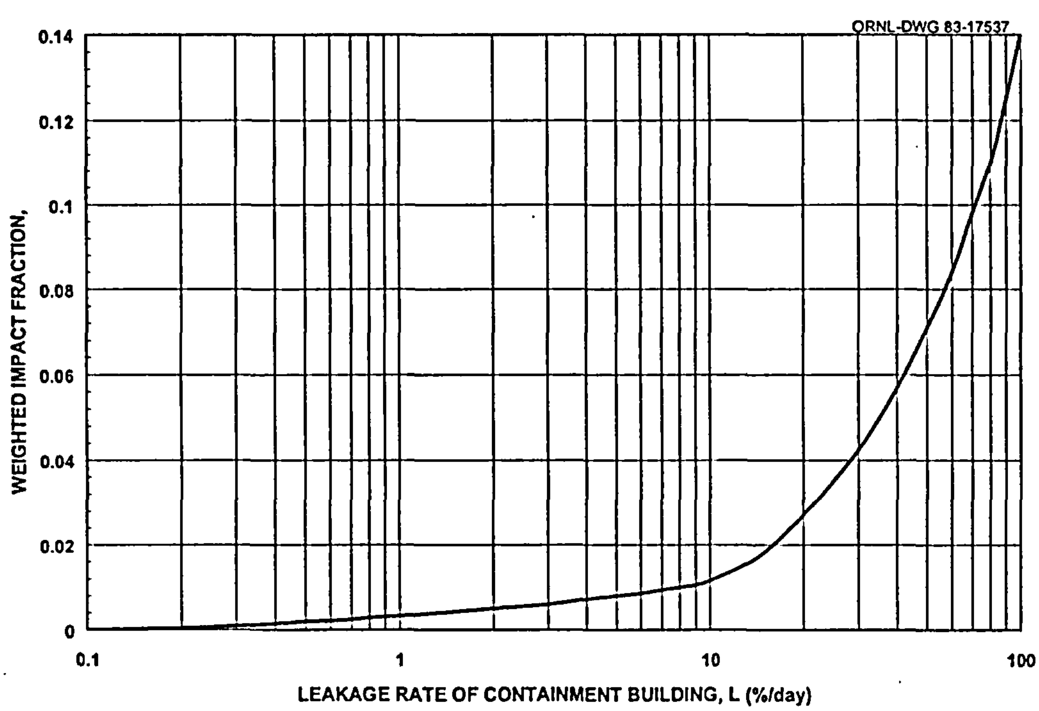


Figure 5-2
Fractional Impact on Risk Associated with Containment Leak Rates



Plant specific LERF frequency values are listed in Table 5-9a through 5-9d for the baseline, 10 year, 15 year and 20 year ILRT test intervals, respectively.

For the purpose of discussion, a generic estimate of the LERF increment may be readily estimated for a bounding PWR. As previously discussed in this Section, the only large release event class impacted by the increase in ILRT interval is that of Class 3B.

The relationship between the event Class 3B and ILRT test interval (INT_{ILRT}) is as follows:

$$F_{Class\ 3B}(INT_{ILRT}) = [\text{Probability of a Class 3B failure for a given inspection interval}]x[CDF_{Intact}]$$

(See Table 5-5 for the Probability of a Class 3B failure for a given inspection interval)

$$F_{Class\ 3B}(INT_{ILRT}) = [2.47E\text{-}4]\ x\ [\ INT_{ILRT}/3.33]x[CDF_{Intact}] = 7.41E\text{-}5\ x\ INT_{ILRT}\ x\ CDF_{Intact}$$

Where:
INT_{ILRT} is the inspection interval in years.

ΔLERF is defined as the increment in the large early release frequency. The ΔLERF is the difference between the Class 3B frequency established using the new inspection interval and the current Class 3B frequency.

For the bounding case of a PWR with a total CDF of 1.0E-4/year, and a 75% probability of an intact containment, the ΔLERF for a 20 year interval extension, compared with the current 10 year interval, is:

$$\Delta LERF = (7.41E\text{-}5/year)\ x\ (20\ yrs - 10\ yrs)\ x\ 0.75\ x\ 1.0E\text{-}4/year = 5.6E\text{-}8/year$$

LERF increments of this magnitude are considered to be very small (negligible). Plant specific LERF results are presented in Tables 5-9a through 5-9d.

Table 5-9a
LERF Frequencies for Example Plant - Baseline ILRT Interval

Class	Description	LERF (per year)
		Example Plant
2	Large Isolation Failures (failure to close)	7.43E-08
3B	Large Pre-existing Containment Leak	1.20E-09
6	Other Isolation Failures (e.g., dependent failures)	0.00E+00
7 (Early)	Failure Induced by Phenomena (early failures)	6.71E-08
8	Bypass (SGTR. ISLOCA)	2.53E-06
LERF	Total	2.673E-06

Table 5-9b
LERF Frequencies for Example Plant - 10 Year ILRT Interval

Class	Description	LERF (per year)
		Example Plant
2	Large Isolation Failures (failure to close)	7.43E-08
3B	Large Pre-existing Containment Leak	3.61E-09
6	Other Isolation Failures (e.g., dependent failures)	0.00E+00
7 (Early)	Failure Induced by Phenomena (early failures)	6.71E-08
8	Bypass (SGTR. ISLOCA)	2.53E-06
LERF	Total	2.675E-06
ΔLERF	Increase from Baseline LERF	2.404E-9
% ΔLERF	% Increase from Baseline LERF	0.09%

Table 5-9c
LERF Frequencies for Example Plant - 15 Year ILRT Interval

Class	Description	LERF (per year)
		Example Plant
2	Large Isolation Failures (failure to close)	7.43E-08
3B	Large Pre-existing Containment Leak	5.41E-09
6	Other Isolation Failures (e.g., dependent failures)	0.00E+00
7 (Early)	Failure Induced by Phenomena (early failures)	6.71E-08
8	Bypass (SGTR. ISLOCA)	2.53E-06
LERF	Total	2.677E-06
ΔLERF	Increase from Current LERF	1.803E-9
% ΔLERF	% Increase from Current LERF	0.07%

Table 5-9d
LERF Frequencies for Example Plant - 20 Year ILRT Interval

Class	Description	LERF (per year)
		Example Plant
2	Large Isolation Failures (failure to close)	7.43E-08
3B	Large Pre-existing Containment Leak	7.21E-09
6	Other Isolation Failures (e.g., dependent failures)	0.00E+00
7 (Early)	Failure Induced by Phenomena (early failures)	6.71E-08
8	Bypass (SGTR. ISLOCA)	2.53E-06
LERF	Total	2.679E-06
ΔLERF	Increase from Current LERF	3.606E-09
% ΔLERF	% Increase from Current LERF	0.13%

5.3 DEFENSE IN DEPTH CONSIDERATIONS

Since the ILRT does not impact CDF, the only impact of ILRT interval extension is on containment reliability. The above analyses estimate the incremental dose released to the public due to a potential increase in the containment leakage. To conservatively maximize the risk impact of these changes several conservative assumptions are included in the model. These are summarized below:

1. Release frequencies and radionuclide releases have been conservatively matched. By matching the probability of containment leakage greater than 100 L_a with an effective 1000 L_a dose, the effective public dose is conservatively estimated.

2. As the EPRI tail frequencies have been based on an expert elicitation that explicitly considers effects of corrosion on aging (see Reference 7), the addition of the incremental corrosion contribution identified in Section 6, below, effectively double counts the risk impact of this contribution.

3. Calculations of radionuclide release do not consider re-establishment of containment cooling following core damage. As these small releases occur over time, long term scrubbing will impact public release. Note that at a leakage rate of 0.1% per day, a 100 L_a leakage would take 10 days for one turnover of the containment atmosphere. Re-establishment of containment cooling would both decrease the actual release rate and scrub fission products from the containment atmosphere.

4. The current approach assumes that leakages are either detected by ILRTs or LLRTs. In practice containment leakages may also be identified by visual examinations of the containment liner or observable changes of the containment instrument air "in leakage". Visual inspections since the 1996 change in the ASME Code are believed to be more effective in detecting flaws. In addition the flaws of concern for LERF are considerably larger than those of concern for successfully passing the ILRT. It is likely that future inspections will be effective in detecting the larger flaws associated with LERF. In addition, many plants periodically experience "in leakage" of instrument air. This condition often results in plants periodically purging containment to maintain the containment pressure within the Technical Specification band. Therefore, for many plants, leakage areas sufficient to result in a LERF-like release could result in a reduced need for purging. Recognition of this change in containment performance would likely result in closer scrutiny of containment integrity.

5.4 PRA QUALITY

All WOG members have had PRA peer reviews in accordance with NEI-00-02 and either have resolved or are in the process of resolving high level peer review comments. The current application evaluates the impact of incremental intact containment releases. The goal of the current application is to demonstrate the potential impact of incremental increases in containment leakage would be very small. The ILRT extension, per se, has no direct impact on core damage frequency. In confirming PRA capability, plants pursuing the risk informed ILRT extension should discuss in the submittal, results of the PRA peer review, and actions taken to resolve high level (A and B) findings. Specifically, items that will significantly impact CDF and LERF predictions associated with ILRT extension should be discussed.

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6.0 POTENTIAL IMPACT OF CONTAINMENT LINER CORROSION

Containment ILRTs are intended to confirm the extent of leakage from a reactor containment. In many instances containment leakage can be detected by localized penetration tests. As the ILRT interval is extended there is increased potential that containment leak paths may develop as a consequence of liner corrosion. While ILRTs have not identified corrosion induced through wall cracks, past visual inspections have identified two events. The treatment of the potential risk impact of extending ILRT intervals with respect to liner corrosion is discussed below. This methodology is consistent with the approach used by CCNPP and previously reviewed and approved by NRC for a one time five year ILRT extension.

6.1 LINER CORROSION EVENTS

Two events of corrosion that initiated from the non-visible (backside) portion of the containment liner have occurred in the industry. Both conditions were detected via routine visual inspection. 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.

6.2 LINER CORROSION ANALYSIS

The following approach is used to determine the change in likelihood, due to extending the ILRT, of detecting liner corrosion. This likelihood is then used to determine the resulting change in risk (both LERF and Person-rem per year). 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.

6.2.1 Liner Corrosion Analysis Method

Differences between the containment basemat and containment cylinder and dome – in the current analysis it is assumed that the liner embedded within the basemat is uninspectable and therefore any corrosion induced leakages cannot be detected by visual inspection. The containment cylinder and dome are capable of being visually inspected. Plants with steel shell containment designs that can and will inspect both sides of the containment are not assumed to be impacted by corrosion in those areas as visual inspection is highly effective.

Historical Liner Flaw Likelihood – is determined based on failure data and success data. Failure data is containment location specific in that the historical events occurred in the cylinder and dome area which is subject to visual inspection rather than in the Basemat region which is not. Success data is based on 70 steel-lined containments with at least 6.0 years of operation since the 10 CFR 50.55a requirement for periodic visual inspections of containment surfaces was established. Two failure events have been observed for the containment cylinder and dome. And zero (assume half a failure) failures have been observed for the containment basemat area. Therefore the flaw likelihoods are determined by dividing the number of observed failure events (2 and 0.5 respectively) by the number of applicable plant years (approximately 70 x 6 = 420 containment-years).

Thus, the Frequency of liner flaws is calculated as:

$$F_{LF} = N_{Failures}/N_{Liner\text{-}Years}$$

Where:
$N_{Failures}$ is the number of observed liner flaws
$N_{Liner\text{-}Years}$ is the number of applicable liner years of operating experience

For an assumed six years since the 10CFR50.55a requirements for periodic visual inspections of containment surfaces, the value for NLF for the containment Cylinder and dome, and basemat are 4.76E-3 and 1.19E-3 respectively.

Age Adjusted Liner Flaw Likelihood - During a 15-year ILRT interval it is assumed that the failure rate doubles every five years. F_{LF} is taken as the failure rate at the mid point for the 5th through 10th (i.e., the 7.5th year). The cumulative failure probability is determined as follows:

F_P = Cumulative Failure Probability at year J = $F_0((1+\chi)^J - 1) / \ln(1+\chi)$

Where: $F_0 = F_{LF} / (1+\chi)^{7.5}$, and

χ is the % increase in failure rate per year

For example, the % increase, χ, such that the flaw failure rate doubles in 5 years is determined via the compound interest formula:

$\chi = \exp(0.693 / 5 \text{ yrs}) - 1$

$\chi = 0.149$, or 14.9%

The failure rate, FR, at year J is then $F_0(1+\chi)^J$. Tables 6-1 and 6-2 illustrate the yearly failure rates and failure probabilities for the base case of liner failure rate doubling every five years.

Table 6-1

Flaw Failure Rate as a Function of Time

Year	Failure Rate (FR)	Success Rate (1-FR)
0	1.68E-03	9.98E-01
1	1.93E-03	9.98E-01
2	2.22E-03	9.98E-01
3	2.55E-03	9.97E-01
4	2.93E-03	9.97E-01
5	3.37E-03	9.97E-01
6	3.87E-03	9.96E-01
7	4.44E-03	9.96E-01
8	5.10E-03	9.95E-01
9	5.86E-03	9.94E-01
10	6.73E-03	9.93E-01
11	7.74E-03	9.92E-01
12	8.89E-03	9.91E-01
13	1.02E-02	9.90E-01
14	1.17E-02	9.88E-01
15	1.35E-02	9.87E-01

The 3-year failure probability is estimated as 1 - the product of the three annual success probabilities. Similarly the 15-year failure probability is determined, as shown in Table 6-2.

Table 6-2

Failure Probability

Years	Success Probability (1-FP)	Failure Probability (FP)
1 to 3	9.94E-01	0.63%
1 to 10	9.64E-01	3.64%
1 to 15	9.15E-01	8.50%

To determine the increase in Flaw Likelihood between a 3-year period and a 15-year period, the failure probability for a three year period is subtracted from the failure probability for a 15 year period. Thus:

Δ = 8.50% - 0.63% = 7.87% (delta between 1 in 3 years to 1 in 15 years)

Liner Corrosion Leakage Dependency on Containment Pressure – To determine the likelihood of breach in containment given liner flaw:

- The containment fragility curve is represented by an exponential function P_F.

  $P_F = A\ e^{BPi}$

  Where P_i is the containment pressure, and A and B are fitting parameters. The parameters of the function are established assuming:

  1. The upper end pressure is set consistent with providing a bounding estimate for the plant containment fragility curve.
  2. The containment failure probability at 20 psia is conservatively assumed to be 0.001.

The basemat failure likelihood is assumed to be 1/10 of the cylinder/dome analysis.

Likelihood that Visual Inspections will be Effective at Detecting a Flaw – it should be noted that both corrosion events discovered were detected by use of visual inspection. In the baseline evaluation, it is assumed that for the containment cylinder and dome the probability of failing to detect an existing flaw is 10%. This is based on a 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. For the containment basemat the probability of failing to detect an existing flaw is 100%.

For plants with concrete containment designs, small liner cracks will not compromise the containment leak prevention capability unless a through wall containment leak path will result. As the liner is designed to withstand design basis pressures and remain integral, an environmental release must be accompanied by a flaw in the concrete. This failure probability represents the likelihood that the liner flaw and the containment flaw are coincident or at least are in sufficiently close proximity that a release path exists. This is a conservative approach as the mechanisms that will induce the leak are likely to be independent.

Likelihood of Non-Detected Containment Leakage

The likelihood of non-detected containment leakage is estimated from the above information as the product of:

1. Increase in Flaw Likelihood between 3 and 15 years,

2. The Likelihood of a breach in containment given a liner flaw,
3. The likelihood of visual detection failure.

6.2.2 Summary of Assumptions

A. Two corrosion events have been identified which could potentially result in liner corrosion. It is assumed that these events may be precursors for a larger containment leakage.

B. A half failure is assumed for basemat concealed liner corrosion due to the lack of identified failures.

C. The success data is limited to the time period since September 1996 when 10 CFR 50.55a started requiring visual inspection.

D. The liner flaw likelihood is assumed to double every five years.

E. 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 200 psia were selected based on conservative representation of the failure probabilities and pressures provided in the example plant IPE structural analysis section. 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.

F. 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.

G. A 5% visual inspection detection failure likelihood given the flaw is visible and a total detection failure likelihood of 10% is used.

H. All non-detectable containment over-pressurization failures are assumed to be large early releases.

I. Containment failure probabilities used for this calculation are based on the plant specific Containment Fragility Curves.

J. The ILRT test pressure of [plant specific design pressure] psia will bound the test failure condition for the example plant.

K. The Base methodology uses continuous fits to an exponential function to estimate containment failure probabilities.

6.2.2 Analysis Results

Table 6-3 presents the Example Plant results of the analysis of the likelihood of non-detected containment leakage because of liner corrosion. The analysis considers the inspectable portion of the liner and the uninspectable portion of the liner.

Table 6-3
Likelihood of Non-detected Containment Leakage due to Liner Corrosion for Example Plant.

Step	Parameter	Location	Example Plant
1	Historical Liner Flaw Likelihood	Containment Cylinder and Dome	4.76E-3
		Containment Basemat	1.19E-3
2	Age Adjusted Liner Flaw Likelihood (15 year avg.)	Containment Cylinder and Dome	5.55E-3
		Containment Basemat	1.39E-3
3	Increase in Flaw Likelihood Between 3 and 15 years	Containment Cylinder and Dome	7.87%
		Containment Basemat	1.97%
4	Likelihood of Breach in Containment given Liner Flaw	Containment Cylinder and Dome	0.83%
		Containment Basemat	0.083%
5	Visual Inspection Detection Failure Likelihood	Containment Cylinder and Dome	10%
		Containment Basemat	100%
6	Likelihood of Non-Detected Containment Leakage	Containment Cylinder and Dome	0.0065%
		Containment Basemat	0.0016%
		Total Likelihood of Non-Detected Containment Leakage	0.0081%

6.3 ANALYSIS OF INCREASE IN LERF DUE TO LINER CORROSION

In this calculation, one estimates the frequency that a corrosion induced through wall flaw exists. To estimate the increase in LERF due to liner corrosion, the plant PRA is used to determine the frequency of core damage events that are not LERFs. This non-LERF frequency (as defined by the sum of Class 1, Class 3a and the late release contribution from Class 7) is multiplied by the total likelihood of non-detected corrosion induced containment leakage (See step 6 in Table 6-3).

Note that in determining the likelihood of non-detected corrosion induced containment leakage, the total likelihood is taken as a weighted average of basemat and dome/cylinder contributions.

6.4 ANALYSIS OF INCREASE IN PERSON-REM PER YEAR

The estimate of Person-rem per year is established by multiplying the frequency of corrosion induced liner flaw times a representative large release.

Table 6-4
Updated Values with Corrosion Impact (from 3/10 years to 15 years) for Example Plant

	Example Plant
LERF Increase Without Liner Corrosion	4.21E-09
LERF Increase With Liner Corrosion	5.08E-09
Person-rem/yr Increase Without Liner Corrosion	0.060
Person-rem/yr Increase With Liner Corrosion	0.063
% Person-rem/yr Increase Without Liner Corrosion	0.80%
% Person-rem/yr Increase With Liner Corrosion	0.84%

6.5 SENSITIVITY STUDIES

The following cases were evaluated to gain an understanding of the sensitivity of this analysis to the various key parameters. The sensitivity analyses are intended to address issues of uncertainty. The base case analyses are considered conservative; however sensitivity analyses are used to help understand the impacts of the more significant assumptions.

Table 6-5

Liner Corrosion Sensitivity Cases

Age (Step 2)	Containment Breach (Step 4)	Visual Inspection & Non-Visual Flaws (Step 5)	Likelihood Flaw is LERF	Example Plant LERF Increase
Base Case Doubles every 5 years	$P_F = 0.000464159$ (Exp (+0.038376418 * Pi) (See footnote 1)	**Base Case** 10%	**Base Case** 100%	**Base Case** **8.69E-10**
Doubles every 2 years	Same as Base	Base	Base	**9.56E-09**
Doubles every 10 years	Same as Base	Base	Base	**4.28E-10**
Base	Base point 10 times lower	Base	Base	**1.76E-10**
Base	Base point 10 times higher	Base	Base	**4.30E-09**
Base	Same as Base	5%	Base	**5.22E-10**
Base	Same as Base	15%	Base	**1.22E-09**
Lower Bound				
Doubles every 10 years	Base point 10 times lower	5%	10%	**5.19E-12**
Upper Bound				
Double every 2 years	Base point 10 times higher	15%	100%	**6.62E-08**

1. P_F for example plant: $P_F = A\,e^{-BPi}$ where A = 0.00046, B = 0.03837 / psi

Key sensitivity studies include:

- Sensitivity to liner failure rate
- Sensitivity to development of fragility curve,
- Sensitivity to visual inspection of flaw

6.6 CONCLUSION FROM CONTAINMENT LINER CORROSION ANALYSIS

For most plants it is expected that the increase in LERF associated with increasing the ILRT interval to 15 years will be very small, and that the additional increase in LERF due to consideration of the potential for containment liner corrosion will also be very small such that the combined impact is a LERF increase that is less than 1.0E-07 per year.

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7.0 RESULTS AND CONCLUSIONS

7.1 SUMMARY OF RESULTS

The results of the plant specific evaluations of risk impacts of ILRT test interval extension are summarized in Table 7-1.

Table 7-1
Summary of Risk Impact of Extending Type A ILRT Test Interval

	Example Plant
BASELINE ILRT INTERVAL	
Baseline Risk Contribution of Class 1 and 3	0.47%
Baseline LERF (per year)	2.673E-6
10 YEAR ILRT INTERVAL	
10 Year Interval Risk Contribution of Class 1 and 3	0.91%
Increase in Total Risk from increasing from Baseline to 10 years	0.45%
10 Year Interval LERF (per year)	2.675E-6
Increase in LERF – Baseline to 10 years (per year)	2.404E-9
15 YEAR ILRT INTERVAL	
15 Year Interval Risk Contribution of Class 1 and 3	1.24%
Increase in Total Risk from increasing from 10 to 15 years	0.34%
15 Year Interval LERF (per year)	2.677E-6
Increase in LERF – 10 Years to 15 years (per year)	1.803E-9
% Increase in LERF – 10 Years to 15 years	0.07%
15 YEAR ILRT INTERVAL from 3/10 to 15 years Considering Liner Corrosion	
LERF Increase Without Liner Corrosion	4.21E-09
LERF Increase With Liner Corrosion	5.08E-09
Person-rem/yr Increase Without Liner Corrosion	0.060
Person-rem/yr Increase With Liner Corrosion	0.063
% Person-rem/yr Increase Without Liner Corrosion	0.80%
% Person-rem/yr Increase With Liner Corrosion	0.84%
20 YEAR ILRT INTERVAL	
20 Year Interval Risk Contribution of Class 1 and 3	1.57%
Increase in Total Risk from increasing from 10 to 20 years	0.67%
20 Year Interval LERF (per year)	2.679E-6
Increase in LERF – 10 Years to 20 years (per year)	3.606E-9
% Increase in LERF – 10 Years to 20 years	0.13%

7.2 CONCLUSIONS FROM RISK EVALUATION

Results are in agreement with the initial NRC/EPRI conclusions that there is a very small (negligible) increase in risk (in terms of person-rem per year) and that there is a very small (negligible) impact on LERF. The change in Type A test interval from 10 years to 20 years increases the risk of those associated specific accident sequences by a small percentage. However, the risk impact on the total integrated plant risk for those accident sequences influenced by Type A testing is a very small percentage (See Table 5-8 for Example Plant values). Therefore, the risk impact when compared to other severe accident risks is very small (negligible).

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 1.0E-6 per year, and increases in LERF below 1.0E-7 per year. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from an 10 years to an 20 years is <1.0E-7 /yr. Therefore, the risk for increasing the ILRT interval from 10 to 20 years is considered to be very small.

8.0 REFERENCES

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

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

3. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," July 1998.

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

5. Burns, T. J., "Impact of Containment Building Leakage on LWR Accident Risk," Oak Ridge National Laboratory, NUREG/CR-3539, April 1984.

6. United States Nuclear Regulatory Commission, Reactor Safety Study, WASH-1400, October 1975.

7. EPRI TR-104285, Revision 1, "Risk Assessment of Revised Containment Leak Rate Testing Intervals," October 2003.

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APPENDIX X

APPLICATION OF THE JOINT APPLICATION REPORT TO AN EXAMPLE PLANT

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X1.0 SYSTEM DESCRIPTION AND OPERATING EXPERIENCE

X1.1 SYSTEM DESCRIPTION

The Containment Structure forms the third and final fission product barrier to minimize the release of radioactivity to the environment in the event of a catastrophic failure of the Reactor Coolant System. The Containment Structure also provides biological shielding for the auxiliary plant and yard areas during normal and accident conditions, and provides a housing for the Nuclear Steam Supply System and certain engineered safeguard components. The following safety-related and non-safety-related functions apply to the Containment Structure:

Safety-Related Functions

The following basic safety-related functions are provided by the Containment Structure:

- Isolation of the containment interior from the environment to reduce the release of radioactive material to values less than those which would result in off-site radiation doses as set forth in 10CFR100 in the unlikely event of a failure in the Reactor Coolant System.
- Biological shielding to the adjacent auxiliary plant and yard areas during normal operation and accident situations.
- Protection for the Nuclear Steam Supply System and other engineered safeguards components from internal and external missiles.
- Protection of safety-related systems and components from the effects of natural phenomena.
- An anchor point and support structure for penetrations.

Non-Safety-Related Functions

The following basic non-safety-related functions are provided by the Containment Structure:

- Housing for various non-safety-related mechanical systems.
- A facility for refueling the reactor and transferring spent fuel to the Auxiliary Building.

Containment Structure

The Containment Structure is a vertical cylinder with a domed roof and a flat base. The cylinder and dome are constructed of post-tensioned concrete. The base is reinforced concrete. A depressed center at the bottom mat houses the reactor. The entire structure is lined with a carbon steel membrane that forms a continuous steel envelope to provide a vapor tight container. The carbon steel envelope encompasses internal reinforced concrete that is independent of the containment wall.

The internal concrete provides:

- Housing for the Reactor Coolant System and some engineered safeguards components

- Localized biological shielding
- Missile shielding
- Refueling cavity

The containment is supported on steel pipes driven into bedrock. The foundation mat forms the base of the Containment Structure, and is constructed of high-strength reinforcing steel and concrete with a permanent access gallery extending under the Containment Structure directly below the wall. The cylindrical concrete wall and dome utilize a post-tensioned construction with 616 cables in the wall and 210 cables in the dome. This network of steel cables (tendons) embedded in the concrete is placed under tension to produce an external force on the structure that will balance the internal forces during a loss-of-coolant accident. These tendons are installed in steel conduits, which are filled with waterproof grease to prevent corrosion of the tendons. Tendon anchors are located so that they are accessible for inspections, testing, and retensioning.

The lower end of the anchors is accessible via the stressing gallery. The stressing gallery is a tunnel, which encircles the Containment Structure, and is accessed via a non-fire-rated watertight door that is on the 972-foot elevation of the Auxiliary Building in Room 22 (safety injection pump room). An escape ladder is located on the south side of the containment for access into Room 66 (equipment hatch room). However, an access cover in the grating is normally locked by Security. Stressing gallery sump pumps were installed by the original design, but have since been removed. The tendon anchor system consists of a stressing head/shim nut combination on one end and a stressing head on the other end. The stressing head transmits the tension force via split tube shims to a bearing plate, which transfers the force to the concrete surface. Each tendon is comprised of 90 parallel, 1/4-inch diameter, high tensile, cold drawn, stress relieved wires with minimum ultimate strength of 1,060,000 lbs. and minimum yield strength of 848,000 lbs.

The containment liner is a 0.25-inch-thick carbon steel membrane welded in sections to form a complete envelope of the inner surface of the complete structure. The membrane forms a leak-tight barrier against the release of radioactive material. The liner is thickened at the penetrations to minimize stress concentration. The liner is protected from damage by the external containment wall, and the internal concrete structure protects the liner from internal missiles. The internal concrete structure is independent of the containment wall and foundation mat. The internal concrete structure provides a missile shield to protect the containment walls and liner against potential missiles such as instrument thimbles, valve stems, valve bonnets, nuts and bolts. The internal structure also provides shielding for radiation protection. The primary shield surrounds and supports the reactor vessel. The secondary shield comprises walls and floors inside the containment, which are built around the reactor loop and other equipment that contains radioactivity. The fuel transfer canal and refueling cavity are part of the secondary shield. Removable concrete slabs over the reactor block any missiles generated by the fracture of a CEDM. The slabs also provide protection against direct and air-scattered radiation from the reactor during operation. The containment sump is located in the floor of the Containment Structure and collects leakage from all floor elevations within the containment. A drainage annulus around the periphery of each floor collects leakage, and a network of 4-inch pipes carries the water from each annulus to the sump. During normal operations, two containment sump pumps remove water from the sump at the 974-foot elevation and pump it to the spent regenerant tank in the Auxiliary Building.

Following a loss-of-coolant accident (LOCA), large quantities of water will accumulate in the sump. Five mesh baskets of trisodium phosphate dodecahydrate (TSP) are located on the lower level of containment (995-foot elevation). The five baskets contain at least 129 cubic feet of trisodium phosphate in total: three baskets contain 23 cubic feet each, and two baskets contain 30 cubic feet each. Trisodium phosphate neutralizes the water that collects in the containment sump following safety injection to raise the pH of the water to greater than 7.0. Two suction strainers in the basement floor at the 994-foot elevation provide suction for emergency core cooling during the re-circulation phase after an accident. Each suction strainer leads to an independent suction header. Each suction header is provided with a motor-operated isolation valve controlled by a handswitch on panel AI-30A in the Control Room. The emergency sump suction valves automatically open on receipt of a re-circulation actuation signal. Each valve is contained within a protective enclosure to contain any leakage.

The Containment Structure is a domed cylinder with an outside diameter of 117 feet 9 1/2 inches, an outside height of 140 feet 4 3/4 inches, an inside diameter of 110 feet, and an inside height of 137 feet 4 1/2 inches. The cylinder wall is 3 feet 10 1/2 inches thick, the domed roof is 3 feet thick, the carbon steel liner plate is 0.25 inches thick, and the foundation slab is 12 feet thick. The slab is support by piling driven to bedrock, which is approximately 70 feet below grade. Each of the 800 piles has an outside diameter of 20 inches. The reinforcing steel of the Auxiliary Building, the Containment Structure, and the mat are connected to the plant grounding system (the steel piles) and thus if exposed to ground water are afforded the same protection as the piles. Containment integrity is defined to exist when all of the following are met:

- All non-automatic containment isolation valves that are not required to be opened during accident conditions and blind flanges are closed.
- The equipment hatch is properly closed and sealed.
- At least one door in the personnel air lock is properly closed and sealed.
- All automatic containment isolation valves are operable or locked closed (or isolated by locked closed valves or blind flanges as permitted by limiting condition for operation).
- The uncontrolled containment leakage is within limits.

X1.2 EXAMPLE PLANT OPERATING EXPERIENCE

Summary Type A Testing History

The example plant has an inspection program and procedure for visual inspection of all accessible areas of the steel containment liner and the concrete containment building. This inspection has been performed prior to each ILRT from Unit startup until the most recent ILRT in 1993. Subsequently, inspection was performed on the same interval of 3 times in 10 years. These inspections indicate no problems with structural integrity or materiel condition of the steel containment vessel and only minor coatings issues. In 2000 the ASME Section XI Subsection IWE inspection plan was approved and implemented at the example plant. Inspections have been completed for the first period of the first 10 year interval with results similar to those determined under the previously program.

X2.0 ASSESSMENT OF RISK FOR THE EXAMPLE PLANT

The purpose of this section is to provide an example of a risk informed assessment for extending the Integrated Leak Rate Test (ILRT) interval. The risk assessment is performed as described in the main body of this report.

In addition, the results and findings from the example plant Individual Plant Examination (IPE) (Reference X-1) and subsequent PRA upgrades are used for this risk assessment. Specifically the approach combines the use of the example plant PRA results with the methodology described in EPRI TR-104285 to estimate public risk associated with extending the containment Type A testing.

The change in plant risk is evaluated based on the change in the predicted releases in terms of person-rem/year and Large Early Release Frequency (LERF). Changes to Type A testing have no impact on CDF.

X2.1 OVERVIEW

In October 26, 1995, the NRC revised 10 CFR 50, Appendix J. The revision to Appendix J allowed individual plants to select containment leakage testing under Option A "Prescriptive Requirements" or Option B "Performance-Based Requirements." The example plant selected the requirements under Option B as its testing program.

The current surveillance testing requirement, as outlined in NEI 94-01 (Reference X-2) for Type A testing, is at least once per 10 years based on an acceptable performance history (define as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage was less than $1.0L_a$). However, the example plant seeks to extend the test interval for Type A testing from ten years to fifteen years based on the substantial cost savings from extending this test interval and the low risk impact.

X2.2 ASSESSMENT OF RISK

The risk impact of extending the ILRT (Type A) interval from its current interval of 10 years to 15 years, is evaluated from a potential public exposure impact (as measured in person-rem/year) and from a Large Early Release (LERF) perspective as identified in Regulatory Guide 1.174. The methodology used accounts for large releases and computes the LERF metric. The analysis examined the example plant IPE and subsequent PRA upgrades for plant specific accident sequences which may impact containment performance. Specifically, as discussed in the main body of this report, core damage sequences were considered with respect to which EPRI event class they are in (EPRI TR-104285 Class 1, 2, 3, 4, 5, 6, 7 or 8 events in terms of containment integrity – Reference X-3).

Table X2-2 presents the Example Plant PRA frequencies for these eight accident classes.

X2.2.1 Quantification of Base-Line Frequency for Accident Classes

The eight EPRI accident class frequencies were determined, using the methodology described in the main body of this report, as described in the following paragraphs:

<u>Class 1 Sequences:</u> This group consists of all core damage accident progression bins for which the containment remains intact. Class 1 sequences arise from those core damage sequences that have long term heat removal capability available via containment sprays or fan coolers. PRA upgrades performed over the past several years have resulted in an overall plant CDF estimate of 1.34E-5/year.

Based on a review of the core damage sequences, the intact containment frequency is estimated to be 4.87E-6 per year. For this analysis, it is assumed that the associated maximum containment leakage for this group is L_a (or 0.1 wt/% per day) (Reference X-4). For this analysis, the events that the PRA categorizes as intact containment events are parsed into three categories, Class 3A, Class 3B and Class 1. As discussed in the text of the main report, as Class 1 and Class 3 events are related, the frequency for Class 1 events is calculated as:

$$F_{Class\ 1} = CDF_{Intact} - F_{Class\ 3A} - F_{Class\ 3B}$$

Class 1 event frequencies are presented in the discussion of Class 3 events, below.

Releases from Class 1 events are calculated based on MACCS 2.0 analysis utilizing the design basis L_a. This is consistent with the assumption that the containment is intact.

<u>Class 2 Sequences:</u> This group consists of all core damage accident progression bins for which a pre-existing leakage due to failure to isolate the containment occurs. These sequences are dominated by failure-to-close large (>2-inch diameter) containment isolation valves. Such sequences contribute to the plant LERF. The frequency per year for these sequences is determined from the example plant PRA as the sum of those release classes that indicate core damage in the presence of an un-isolated containment.

$$F_{Class\ 2} = 7.43E\text{-}08\ /year$$

Class 2 releases for the example plant analyses are associated with loss of isolation failures resulting in a through containment equivalent leakage from a pipe greater than 2 inches in diameter.

<u>Class 3 Sequences:</u> Class 3 endstates are developed specifically for this application. The Class 3 endstates include all core damage accident progression bins for which a pre-existing leakage in the containment structure exists. The containment leakage for these sequences can be grouped into two categories, small leaks or large.

The respective frequencies per year are determined as follows:

$$F_{Class\,3A} = PROB_{Class\,3A} * CDF_{Intact}$$

$$F_{Class\,3B} = PROB_{Class\,3B} * CDF_{Intact}$$

Where:

CDF_{Intact} = the Core Damage Frequency for the intact containment sequences, and is 4.87E-06/year.
$PROB_{Class\,3A}$ = the probability of small pre-existing containment leakage in excess of design allowable.
$PROB_{Class\,3B}$ = the probability of large pre-existing containment leakage.

$PROB_{Class\,3A}$ and $F_{Class\,3B}$ are a function of inspection interval and are obtained from Section 5.2.3, using Table 5-5 (reproduced here for convenience) as follows:

Probability of Type A Leakage for a Given Test Interval

Test Interval	Probability	
	Small Leak (Class 3A) ($PROB_{Class\,3A}$)	Large Leak (Class 3B) ($PROB_{Class\,3B}$)
3 per 10 Years	0.028	2.47E-4
10 Years	0.084	7.41E-4
15 Years	0.126	1.11E-3
20 Years	0.168	1.48E-3

The resulting values for $F_{Class\ 1}$, $F_{Class\ 3A}$, and $F_{Class\ 3B}$ as a function of ILRT interval are presented in Table X2-1.

Table X2-1
Example Plant Frequency of Type A Leakage for a Given Test Interval

Test Interval	Release Class Frequency (per year)		
	$F_{Class\ 1}$	$F_{Class\ 3A}$	$F_{Class\ 3B}$
3 per 10 Years	4.73E-6	1.36E-7	1.20E-9
10 Years	4.45E-6	4.09E-7	3.61E-9
15 Years	4.25E-6	6.13E-7	5.41E-9
20 Years	4.04E-6	8.18E-7	7.21E-9

As Class 3A represents a small pre-existing containment leak, its value was set to bound the maximum quantified release identified in Table 4-2 of NUREG-1493. The largest identified release multiple was $21L_a$. Class 3A releases were therefore quantified as $25L_a$. For the example plant this results in a containment leakage rate of 2.5 wt% per day.

Class 3B releases are assumed to be greater than $100L_a$ (or 10 wt% per day). Releases in this category were represented by a 100 wt% per day release which is roughly equivalent to a release from a 2.5 inch orifice. This leakage is essentially equivalent to $1000L_a$ (for the example plant) and is considered a very conservative estimate of potential containment releases that may result from extension of Type A containment Testing. The specific man-rem estimate for this release was evaluated by multiplying the intact release calculated dose by 1000.

It should also be noted that in establishing the Class 3B frequency, large releases were based on the expert elicitation with a frequency associated with releases $\geq 100L_a$. This is a conservative estimate as LERF releases are on the order of $1000L_a$. This assumption conservatively over estimates the LERF contribution by a factor of about 60.

Class 4 Sequences: This group consists of all core damage accident progression bins for which a failure-to-seal containment isolation failure of Type B test components occurs. Because these failures are detected by Type B tests, this group is not evaluated any further.

Class 5 Sequences: This group consists of all core damage accident progression bins for which a failure-to-seal containment isolation failure of Type C test components occurs. Because these failures are detected by Type C tests, this group is not evaluated any further.

Class 6 Sequences: This group is similar to Class 2. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution, typically resulting in a failure to close smaller containment isolation valves. All other failure modes are bounded by the

Class 2 assumptions. All sequences in this category were subsumed in Class 2 releases and therefore this release is not evaluated any further.

<u>Class 7 Sequences:</u> This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs (e.g., H_2 combustion).

$$F_{Class\ 7} = CDF_{CFL} + CDF_{CFE}$$

Where:
CDF_{CFE} = the CDF resulting from phenomena that lead to early containment failure.
CDF_{CFL} = the CDF resulting from phenomena that lead to late containment failure.

This frequency was determined by subtracting the intact, bypass (See Class 8 discussion) and loss of isolation CDFs from the total CDF. This results in the following Class 7 frequency:

$$\mathbf{F_{Class\ 7} = 5.90E\text{-}6 / year}$$

These endstates include containment failure. It was determined from the PRA that the early component of $F_{Class\ 7}$, CDF_{CFE}, is 6.71E-08. The small contribution of early containment failures for the example plant is a result of the robust containment design. Detailed structural evaluation of the example plant containment indicates the mean failure pressure of t example plant containment is 195 psig (Reference X-1).

<u>Class 8 Sequences:</u> This group consists of all core damage accident progression bins in which containment bypass occurs.

Using the results of the most recent example plant PRA and including ISLOCA and SGTR sequences, the failure frequency for this class is 2.53E-6 / year.

$$\mathbf{F_{Class\ 8} = 2.53E\text{-}6 / year}$$

<u>**Comments on Calculation of Releases:**</u>

Releases for the example plant sequences and release classes are based on an update of the Level 3 PRA. This analysis was initially performed in support of the IPE (Reference X-1) and was recently extended to account for current site conditions and demographics. Analyses were performed using MACCS 2.0 (Reference X-5). Since release Class 3A and 3B were not previously considered in prior applications, the releases were established by scaling the intact releases upward by factors that reflect the increased leakage magnitudes. For the example plant Class 3A and 3B, the factors were 25 and 1000 respectively. Releases include consideration of all radionuclide classes considered in NUREG-1150.

Table X2-2 provides a summary of the example plant Release Class frequencies.

Table X2-2
Example Plant Mean Containment Frequencies (from the PRA)

Class	Description	Frequency (per Rx-year)
1	No Containment Failure	4.73E-06
2	Large Containment Isolation Failures (failure-to-close)	7.43E-08
3A	Small Pre-existing Containment Leak	1.36E-07
3B	Large Pre-existing Containment Leak	1.20E-09
4	Small isolation failure - failure-to-seal (Type B test)	NA
5	Small isolation failure - failure-to-seal (Type C test)	NA
6	Containment Isolation Failures (dependent failures, personnel errors)	NA
7	Severe Accident Phenomena Induced Failure (early and late failures)	5.90E-06
8	Containment Bypassed (SGTR / ISLOCA)	2.53E-06
Total	All CET Endstates	1.34E-05

X2.2.2 Example Plant Population Dose per Reactor Year

Plant-specific release analysis was performed for the example plant to evaluate the doses to the population, within a 50-mile radius from the plant. The releases for Classes 1 through 8 are based on population dose calculations obtained with MACCS2.0 (Reference X-5). Representative intact releases were obtained from a frequency-weighted average of the dose associated with intact containment core damage scenarios with and without operational containment sprays (see Table X2-3). Class 3A and 3B were determined by multiplying mean Class 1 calculated doses by multipliers reflective of the increase in fission product releases associated with the degraded containment conditions. Class 7 doses represent a frequency-weighted average of late and early releases. Class 8 (Bypass) calculated doses are established as the frequency-weighted average of ISLOCA and SGTR (both randomly initiated and thermally induced) events. The resulting mean population dose is summarized in Table X2-4.

In performing the above analyses offsite population estimates are based on the example plant demographics projections to 2030, from the example plant Severe Accident Mitigation Alternatives evaluation in the example plant License Renewal evaluation (Reference X-6). Atmospheric dispersions are based on representative meteorological data for a representative year. Impact of variations in weather data between the representative year and another five-year data span was found to be small.

Table X2-3
Example Plant Population Dose – Intact Containment

Example Plant Population Dose – Intact Containment			
Containment Status	Frequency (per year)	Release (person-rem/event)	Dose (person – rem/yr)
Intact Containment with Sprays Operating	3.58E-06	3.39E+02	1.25E-03
Intact Containment without Sprays	1.15E-06	1.40E+04	1.66E-02
Representative Release		3.77E+03	

Table X2-4
Example Plant Population Dose – Bypass Events

Example Plant Population Dose – Containment Bypassed			
Release Type	Frequency (per year)	Release (person-rem/event)	Dose (person – rem/yr)
SGTR	8.54E-07	8.61E+06	1.84E+00
ISLOCA	1.229E-06	6.54E+06	3.62E+00
TI-SGTR	4.42E-07	2.17E+06	9.59E-01
Representative Release		2.54E+06	

Table X2-5
Example Plant Containment Leakage Rate and Dose – for Accident Classes

Class	Description	Leakage (wt%/day)	Release (50 miles) (person-rem)	Basis
1	No Containment Failure	0.1 (L_a)	3.77E+03	Note 1
2	Large Containment Isolation Failures (failure-to-close)	Note 1	1.06E+06	Note 1
3A	Small Isolation Failures (containment leak)	2.5 (25 L_a)	9.43E+04	Ratio from class 1 baseline
3B	Large Isolation Failures (containment leak)	100	3.77E+06	Ratio from class 1 baseline
4	Small isolation failure - failure-to-seal (Type B test)	Not analyzed	NA	
5	Small isolation failure - failure-to-seal (Type C test)	Not analyzed	NA	
6	Containment Isolation Failures (dependent failures, personnel errors)	Not analyzed	NA	
7	Severe Accident Phenomena Induced Failure (early and late failures)	Note 1	1.65E+05	Note 1
8	Containment Bypassed (SGTR / ISLOCA)	-	2.54E+06	Note 1

Note 1 - From MACCS 2.0 calculations performed for the example plant Severe Accident Mitigation Alternative evaluation for License Renewal (Ref. X-6).

The above results when combined with the frequencies presented in Table X2-2 yields the example plant baseline mean consequence measures (risks, in terms of person-rem/yr) for each accident class. The resulting risks (in terms of person-rem/yr), for each accident class, are presented in Table X2-6 below.

Table X2-6
Example Plant Mean Baseline Risk - for Accident Classes

Class	Description	Frequency (per Rx-yr)	Person-Rem (50-Miles)	Person-Rem/yr (50-Miles)
1	No Containment Failure	4.73E-06	3.77E+03	1.78E-02
2	Large Isolation Failures (failure to close)	7.43E-08	1.06E+06	7.88E-02
3A	Small Pre-existing Containment Leak	1.36E-07	9.43E+04	1.28E-02
3B	Large Pre-existing Containment Leak	1.20E-09	3.77E+06	4.53E-03
4	Small Isolation Failure to Seal (Type B Test)	Not Analyzed	NA	N/A
5	Small Isolation Failure to Seal (Type C Test)	Not Analyzed	NA	N/A
6	Other Isolation Failures (e.g., dependent failures)	Not Analyzed	NA	N/A
7	Failure Induced by Phenomena (early and late failures)	5.90E-06	1.65E+05	9.74E-01
8	Bypass (SGTR / ISLOCA)	2.53E-06	2.54E+06	6.43E+00
Total	All CET End States	1.34E-05	N/A	7.51

Based on the above values, the percent risk contribution associated with the "intact" containment sequences for Class 1 and Class 3 ($\%Risk_{BASE}$) is as follows:

$$\%Risk_{BASE} = [(Risk_{Class\ 1\ BASE} + Risk_{Class\ 3A\ BASE} + Risk_{Class\ 3B\ BASE}) / Total_{BASE}] \times 100$$

Where:

$Risk_{Class\ 1\ BASE}$ = Class 1 person-rem/yr = 1.78E-02 person-rem/yr [Table X2-6]

$Risk_{Class\ 3A\ BASE}$ = Class 3A person-rem/yr = 1.28E-02 person-rem/yr [Table X2-6]

$Risk_{Class\ 3B\ BASE}$ = Class 3B person-rem/yr = 4.53E-03 person-rem/yr [Table X2-6]

$Total_{BASE}$ = total dose/year for baseline interval = 7.51 person-rem/year [Table X2-6]

$\%Risk_{BASE}$ = [(1.78E-02 + 1.28E-02 + 4.53E-03) / 7.51] x 100

$\%Risk_{BASE}$ = 0.47 %

Therefore, the total baseline risk contribution of leakage, represented by Class 1 and Class 3 accident scenarios is 0.47 %.

X2.2.3 Risk Impact of Extending Type A Test Interval From 10 To 15 And 20 Years

Using the methodology described in the main report that was used above to determine baseline risk values (see Table X2-6), the risk values were determined for the Current 10 year ILRT test interval, a 15 year ILRT test interval, and a 20 year ILRT test interval. These risk values are presented below in Table X2-7.

Table X2-7
Example Plant Risk Values vs. ILRT Interval (Person-Rem/yr to 50-Miles)

Class	Description	Current 10 year ILRT interval	15 year ILRT interval	20 year ILRT interval
1	No Containment Failure	1.68E-02	1.60E-02	1.52E-02
2	Large Isolation Failures (failure to close)	7.88E-02	7.88E-02	7.88E-02
3A	Small Pre-existing Containment Leak	3.85E-02	5.78E-02	7.71E-02
3B	Large Pre-existing Containment Leak	1.36E-02	2.04E-02	2.72E-02
4	Small Isolation Failure to Seal (Type B Test)	N/A	N/A	N/A
5	Small Isolation Failure to Seal (Type C Test)	N/A	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	NA	NA	NA
7	Failure Induced by Phenomena (early and late failures)	9.74E-01	9.74E-01	9.74E-01
8	Bypass (SGTR/ISLOCA)	6.43E+00	6.43E+00	6.43E+00
Total	All CET End States	7.55	7.57	7.60

Based on the above values, and using the methodology described in the main report, the percent risk contribution (%Risk$_N$, for values of N of 10, 15 and 20 years) for Class 1 and Class 3 is determined and yields the results summarized in Table X2-8, below. Also, the percent change in risk due to ILRT interval extensions is determined and presented in Table X2-8.

Table X2-8
Example Plant Percent Risk Increases from ILRT Interval Extensions

	Description	Current 10 year ILRT interval	15 year ILRT interval	20 year ILRT interval
%Risk$_N$	Percent risk contribution for Class 1 and Class 3	0.91%	1.24%	1.57%
Δ%Risk$_{\text{Base to N}}$	Percent increase in total risk due to an N-year ILRT over the baseline case	0.45%	N/A	N/A
Δ%Risk$_{10\text{-}N}$	Percent increase in risk due to an N-year ILRT over the 10 year case	N/A	0.34%	0.67%

X2.2.4 Change In Risk In Terms Of Large Early Release Frequency (LERF)

Section 5.2.4 of the main body of this report discusses the quantification of LERF. This analysis assumes that Class 2, 3B, 6, 7 and 8 lead to large leak rates. The baseline LERF frequency, for the 3 in 10 year inspection interval, is determined as shown in Table X2-9. The estimate for Class 7 includes only the portion of Class 7 identified in the PRA as representing early containment failure.

Table X2-9
Example Plant Baseline LERF Frequency Calculation

Class	Description	LERF
2	Large Isolation Failures (failure to close)	7.43E-08
3B	Large Pre-existing Containment Leak	1.20E-09
6	Other Isolation Failures (e.g., dependent failures)	0.00E+00
7 (Early)	Failure Induced by Phenomena (early failures)	6.71E-08
8	Bypass (SGTR / ISLOCA)	2.53E-06
LERF	(total)	**2.673E-06**

Impact of ILRT Test Interval Extensions on Large Early Release Frequency (LERF)

Table X2-10 presents the frequencies for each large release class, for each of four ILRT intervals. The total LERFs are also listed, along with the increase in LERF from the current LERF, and the percent increase from the current LERF.

As the only class contributor to the change in large early release is due to Class 3B events, the ΔLERF = $F_{Class\ 3B}$ (evaluated at the new inspection interval) – $F_{Class\ 3B}$ (of the baseline interval or the current interval, as appropriate).

The percent change in LERF is calculated as:

$$\%\Delta LERF = [\Delta LERF/LERF_{Total}] \times 100$$

Where:
$LERF_{Total}$ = The sum of the Frequencies of Sequences 2, 3B, 6, 8, and the "early" portion of Class 7, (6.71E-8).

Table X2-10
Example Plant LERF Variation as a Function of Change in Inspection Interval

Class	Description	3 per 10 Years	10 Years	15 Years	20 Years
2	Large Isolation Failures (failure to close)	7.43E-08	7.43E-08	7.43E-08	7.43E-08
3B	Large Pre-existing Containment Leak	1.20E-09	3.61E-09	5.41E-09	7.21E-09
6	Other Isolation Failures (e.g., dependent failures)	0.00E+00	0.00E+00	0.00E+00	0.00E+00
7 (Early)	Failure Induced by Phenomena (early failures)	6.71E-08	6.71E-08	6.71E-08	6.71E-08
8	Bypass (SGTR)	2.53E-06	2.53E-06	2.53E-06	2.53E-06
LERF	Total	2.673E-06	2.675E-06	2.677E-06	2.679E-06
ΔLERF	Increase from Current LERF	N/A	0.0	1.803E-09	3.606E-09
%ΔLERF	% Increase from Current LERF	N/A	0.0%	0.07%	0.13%

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X3.0 SUMMARY OF RESULTS

Baseline ILRT Interval Results (For this evaluation, the baseline risk contribution is taken as the original inspection interval at the time that the IPE was done; that is, three inspections per 10 year interval)

1. The baseline risk contribution of leakage, represented by Class 1 and Class 3 accident scenarios is 0.47 % of total risk.

2. The baseline LERF is 2.673E-06 per year.

Ten Year ILRT Interval Results

1. The current Type A 10-year ILRT interval risk contribution of leakage, represented by Class 1 and Class 3 accident scenarios is 0.91% of total risk.

2. The increase in total risk from extending the ILRT test interval from the baseline interval to current 10 year interval is 0.45%

3. The LERF with a 10 year ILRT interval is 2.675E-06 per year.

4. The increase in LERF from extending the ILRT test interval from the baseline interval to the current 10 year interval is 2.404E-09 per year.

5. The % increase in LERF from extending the ILRT test interval from the baseline interval to 10 years is 0.09 %. Since the CDF is not changed as a result of the extended ILRT interval, the increase in LERF is due only to the small increase (0.09 %) in conditional containment unreliability.

Fifteen Year ILRT Interval Results

1. Type A 15-year ILRT interval risk contribution of leakage, represented by Class 1 and Class 3 accident scenarios is 1.24 % of total risk.

2. The increase in total risk from extending the ILRT test interval from the current 10 year interval to 15 years is 0.34 %.

3. The LERF for the 15 year interval is 2.677E-06 per year.

4. The increase in LERF from extending the ILRT test interval from the 10 year interval to 15 years is 1.803E-09 per year.

5. The % increase in LERF from extending the ILRT test interval from the 10 year interval to 15 years is 0.07 %. Since the CDF is not changed as a result of the extended ILRT interval, the increase in LERF is due only to the small increase (0.07 %) in conditional

containment unreliability. The total the increase in conditional containment unreliability in going from the baseline 3-in10 year ILRT interval to a 15 year interval is also small (0.16%).

Twenty Year ILRT Interval Results

1. Type A 20-year ILRT interval risk contribution of leakage, represented by Class 1 and Class 3 accident scenarios is 1.57 % of total risk.

2. The increase in total risk from extending the ILRT test interval from the current 10 year interval to 20 years is 0.67 %.

3. The LERF for the 20 year interval is 2.679E-06 per year.

4. The increase in LERF from extending the ILRT test interval from the 10 year interval to 20 years is 3.606E-09 per year.

5. The % increase in LERF from extending the ILRT test interval from the 10 year interval to 20 years is 0.13 %. Since the CDF is not changed as a result of the extended ILRT interval, the increase in LERF is due only to the small increase (0.13 %) in conditional containment unreliability. The total the increase in conditional containment unreliability in going from the baseline 3-in10 year ILRT interval to a 20 year interval is also small (0.22%).

X4.0 POTENTIAL IMPACT OF CONTAINMENT LINER CORROSION

X4.1 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.

X4.2 EXAMPLE PLANT STRUCTURAL DESIGN

X4.2.1 Structural Design of Walls

The Containment Structure is a reinforced concrete pressure vessel partially pre-stressed, with cylindrical walls, domed roof and a bottom mat incorporating a depressed center portion for the reactor. This structure is lined with a steel membrane forming a continuous steel envelope located at the inner surface of cylinder, roof and mat. The liner plate is ¼-inch thick and is attached and anchored to the containment concrete structure. The rear face of the liner plate is unpainted; concrete of the containment shell was poured directly against it and protects it against corrosion. There is thickened liner plate at the polar crane supports. The concrete vertical wall thickness is 3' $10^{1}/_{2}$". The concrete dome thickness is 3 feet. The steel liner has test channels welded over all seams, which have not been accessible for inspection since placement of the interior concrete. 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.

X4.2.2 Structural Design of Floor

The containment basemat is a 12-foot thick foundation slab. The containment structure is supported on steel piles driven to bedrock located approximately 70 feet below grade. The piles are embedded into the foundation mat for a length of three feet. The concrete foundation mat is reinforced with high strength reinforcing steel and has a permanent access gallery extending under the containment structure directly below the cylindrical wall. The containment mat was designed as a conventionally reinforced foundation supported on piles subjected to the various combinations of loadings for the working stress, modified ultimate strength and no loss of function. For liner plates on nominally horizontal surfaces, such as the surface of the reactor cavity and the foundation mat, continuous anchor members were embedded flush with the concrete surface for plate attachment.

X4.2.3 Inspectable Area

Approximately 86 percent of the interior surface of the liner is accessible for visual inspections. The 14 percent that is inaccessible for visual inspections include the fuel transfer tube and area under the concrete containment floor.

Visual inspections following the 1996 change in the ASME Code are believed to be more effective in detecting containment liner flaws. In addition, the flaws 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.

The containment performance data is contained in NUREG-1493. This 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.

X4.3 EXAMPLE PLANT INSPECTION PROGRAM

Inspections of both the containment liner and the concrete structure are conducted in accordance with the example plant IWE/IWL Program Plan. A VT General inspection was completed on the liner during the 2001 Refueling Outage and was repeated during the 2003 Refueling Outage. A more thorough detailed visual examination is performed on any area with evidence of degradation. The liner inspection is repeated on a nominal 40-month interval. General (VT-3C) Inspection of the Concrete Containment using IWL requirements was completed in August 2001 and will be repeated at 5-year intervals.

The IWE liner inspection conducted in 2001 revealed only minor areas of corrosion, which were either repaired or noted for additional monitoring. The example plant has no history of significant liner corrosion.

X4.4 LINER CORROSION ANALYSIS FOR THE EXAMPLE PLANT

The approach discussed above in Section 6.0 was used to determine how liner corrosion affects the risk associated with extending the ILRT.

X4.4.1 Analysis

Table X4-1 presents the results of the analysis of the likelihood of non-detected containment leakage because of liner corrosion. The analysis considers the inspectable portion of the liner and the uninspectable portion of the liner. Approximately 86 percent of the interior surface of the containment liner is accessible for visual inspection. The 14 percent that are inaccessible for visual inspection include the fuel transfer tube shielded area, the area under the concrete floor, and the area behind the elevator shaft. The area under the concrete floor accounts for almost all of the inaccessible area.

Table X4-1
Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome 86%		Containment Basemat 14%	
1	**Historical Liner Flaw Likelihood** Failure Data: Containment location specific Success Data: Based on 70 steel-lined Containments and 6.0 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 * 6.0) = 4.76E-3		Events: 0 Assume half a failure 0.5/(70 * 6.0) = 1.19E-3	
2	Age Adjusted Liner Flaw Likelihood During 15-year interval, assumed failure rate doubles every five years (14.9% increase per year). The midpoint for 5th to 10th year was set to the historical failure rate. (See Table-X4-5 for an example.)	Year	Failure Rate	Year	Failure Rate
		1	1.93E-3	1	4.83E-4
		avg 5 – 10	4.76E-3	avg 5 – 10	1.19E-3
		15	1.35E-2	15	3.37E-3
		15 year avg = 5.55E-3		15 year avg = 1.39E-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 X4-5 and X4-6.	7.87%		1.97%	
4	Likelihood of Breach in Containment given Liner Flaw The upper end pressure is consistent with the Example Plant 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 failure likelihood is assumed to be 1/10 of the cylinder/dome analysis	Pressure (psia)	Likelihood of Breach	Pressure (psia)	Likelihood of Breach
		20	0.10%	20	0.01%
		75 (ILRT)	0.83%	75 (ILRT)	0.083%
		80	1.0%	80	0.10%
		120	4.6%	120	0.46%
		200	100%	200	10%
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.	
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3 * 4* 5)	0.0065% 7.87% * 0.83% * 10%		0.0016% 1.97% * 0.083% * 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 = 0.0065% + 0.0016% = 0.0081%

The non-Large Early Release Frequency (LERF) containment over-pressurization failures for the example plant are estimated, based on the PRA, at **1.07E-05** per year. The non-LERF frequency is obtained by adding the Class 1 (intact) and late releases contribution from Class 7 (severe accident). 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 (comparing a 3 in 10 year ILRT to a one in 15 year ILRT) = 0.0081% * 1.07E-5= 8.7E-10 per year.

Note that the current approved ILRT test interval at the example plant and at most WOG plants is one ILRT every ten years. The increase in LERF published here is greater than it would be when comparing to a one in ten year frequency currently in effect.

X4.4.2 Change in Risk from extending ILRT Interval

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.

These elements are discussed in detail below.

X4.4.2.1 Risk from Pre-existing Containment Breach

Item 1 (above) is addressed above and summarized in Section X3.0. Table X4-2 lists the key risk values.

Table X4-2

Risk Values without Liner Corrosion (from 3/10 years to 15 years)

Method	LERF Increase	Person-rem/yr increase	Percentage Increase in Person-rem/yr
Without Liner Corrosion	4.21E-09	0.06	0.80%

X4.4.2.2 Risk from Liner Corrosion

Table X4-3, below shows an additional small increase in LERF of 8.7E-10 (due to liner corrosion) from going from a 3/10 to a 15 year ILRT interval. Thus, Table X4-2 is modified as follows:

Table X4-3
Updated Risk Values with Corrosion Impact (from 3/10 years to 15 years)

Method	LERF Increase	Person-rem/yr increase	Percentage Increase in Person-rem/yr
Without Liner Corrosion	4.21E-09	0.060	0.80%
With Liner Corrosion	5.08E-09	0.063	0.84%

X4.4.3 Sensitivity Studies

The cases listed in Table X4-4 were developed to gain an understanding of the sensitivity of this analysis to the various key parameters.

Table X4-4
Liner Corrosion Sensitivity Cases

Age (Step 2)	Containment Breach (Step 4)	Visual Inspection & Non-Visual Flaws (Step 5)	Likelihood Flaw is LERF	LERF Increase due to Liner Corrosion
Base Case Doubles every 5 years	PF = 0.000464159 (Exp (+0.038376418 * Pi)	Base Case 10%	Base Case 100%	Base Case 8.69E-10
Doubles every 2 years	Same as Base	Base	Base	9.56E-09
Doubles every 10 years	Same as Base	Base	Base	4.28E-10
Base	Base point 10 times lower	Base	Base	1.76E-10
Base	Base point 10 times higher	Base	Base	4.30E-09
Base	Same as Base	5%	Base	5.22E-10
Base	Same as Base	15%	Base	1.22E-09
Lower Bound				
Doubles every 10 years	Base point 10 times lower	5%	10%	5.19E-12
Upper Bound				
Double every 2 years	Base point 10 times higher	15%	100%	6.62E-08

Table X4-5

Flaw Failure Rate as a Function of Time

Year	Failure Rate (FR)	Success Rate (1-FR)
0	1.68E-03	9.98E-01
1	1.93E-03	9.98E-01
2	2.22E-03	9.98E-01
3	2.55E-03	9.97E-01
4	2.93E-03	9.97E-01
5	3.37E-03	9.97E-01
6	3.87E-03	9.96E-01
7	4.44E-03	9.96E-01
8	5.10E-03	9.95E-01
9	5.86E-03	9.94E-01
10	6.73E-03	9.93E-01
11	7.74E-03	9.92E-01
12	8.89E-03	9.91E-01
13	1.02E-02	9.90E-01
14	1.17E-02	9.88E-01
15	1.35E-02	9.87E-01

Table X4-6

Failure Probability

Years	Success Rate (1-FP)	Failure Rate (FP)
1 to 3	9.94E-01	0.63%
1 to 10	9.64E-01	3.64%
1 to 15	9.15E-01	8.50%

Δ = 8.50% - 0.63% = 7.87% (delta between 1 in 3 years to 1 in 15 years)

X4.5 CONCLUSION FROM CONTAINMENT LINER CORROSION ANALYSIS

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

X5.0 REFERENCES

X-1 Example Plant Individual Plant Examination, December 1993.

X-2 NEI 94-01, Revision 0 "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50," Appendix J, July 26, 1995.

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

X-4 Example Plant UFSAR.

X-5 MACCS2: MELCORE Accident Consequence Code System for the calculation of Health and Economic Consequences of Accidental Radiological Releases, NUREG/CR-6613, May 1998.

X-6 Letter, to U.S. NRC, "Applicant's Environmental Report, Example Plant Application for Renewed Operating License," dated January 9, 2002.

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Non-Proprietary Class 3

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