COLUMBIA GENERATING STATION RISK ASSESSMENT TO SUPPORT ILRT (TYPE A) INTERVAL EXTENSION REQUEST

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EXECUTIVE SUMMARY

The risk impact of a one-time extension of the Columbia Generating Station (CGS) integrated leak rate test (ILRT) interval from the currently approved 10 years to 15 years has been evaluated. The results demonstrate that a change in the ILRT test interval from 10 years to 15 years represents a "very small" impact on risk, as defined by Reg. Guide 1.174.

The Columbia ILRT risk assessment uses CGS specific information to calculate the changes to the risk profile due to changes in the ILRT interval. The evaluation approach for the assessment of the risk is based on EPRI-TR-104285, NEI Interim Guidance (dated November 2001), and previous ILRT risk assessment submittals. The full power internal events PRA model for CGS is used as the primary basis of the assessment. External events are addressed by sensitivity evaluations. The ex-plant consequences are based on adjusting the ex-plant consequences from a surrogate plant (as allowed by the NEI Interim Guidance).

The consequence evaluation utilizes NUREG/CR-4551 50-mile dose risk for a Mark I plant (Peach Bottom). The total dose risk is subdivided into accident progression bins (APBs) based on NUREG/CR-4551. The dose risk for each APB is adjusted to account for population differences, containment leakage rate, and power level for applicability to CGS. The CGS adjusted dose for each APB is then applied to an equivalent EPRI category for consideration of the effects of ILRT interval changes. CGS Level 2 release sequences are used to determine the frequency of each EPRI category. Three of the EPRI categories are affected by ILRT interval changes (Categories 1, 3a, and 3b). Table ES-1 compares the results of various risk measures for the current 10-year ILRT interval with the proposed 15-year ILRT interval.

Three risk measures are evaluated using the CGS internal events PSA model (Rev. 5) to characterize the reduction in ILRT frequency from 1-per-10 years to 1-per-15 years:

Risk Metrics	Risk Increase
• Change in Large Early Release Frequency (LERF)	2.0E-8/yr[1]
• Change in conditional containment failure probability	0.1%
• Change in population dose rate (person-rem/yr)	Negligible[2]

The first risk measure change is considered by Reg. Guide 1.174 as a "very small" impact on risk. The other two risk measure changes do not have criteria in Reg. Guide 1.174, but based on past ILRT interval extension requests these changes are also considered to represent "very small" impacts on risk.

Additionally, several sensitivity cases are evaluated and documented in this ILRT analysis. These sensitivity cases demonstrate the following:

- Inclusion of long-term station blackout scenarios in the EPRI categories 3a and 3b frequencies increases the risk measures a negligible amount and does not change the conclusion of this report.
- LERF is not significantly impacted by the potential for containment leakage due to age-related degradation in non-inspectable areas; the ΔLERF remains within Region III as a "very small" risk change.
- The inclusion of external events increases LERF approximately four-fold; however, the ΔLERF remains within Region III as a "very small" risk change.

This ILRT assessment also evaluates the risk impact of extending the drywell to wetwell Bypass Leak Rate Test (BLRT) interval from 2 years to 10 years. The risk increase on the total integrated plant risk by extending the BLRT interval from 2 years to 10 years is negligible (<<1%). The increase in LERF is zero (i.e., because any associated changes in releases are late). Per RG 1.174, the LERF change and CDF change are not significant.

(1) Rounded up from the calculated value of 1.98E-8/yr.

(2) The change in dose rate from the 1-per-10 year ILRT frequency to the 1-per-15 year ILRT frequency is approximately 2E-3 person-rem/yr. This change is less than the number of significant figures being reasonably carried in the assessment.

The vacuum breaker leakage test (proposed for future) and stringent acceptance criteria, combined with the negligible non-vacuum breaker leakage, and thorough periodic visual inspection provide an equivalent level of assurance as the BLRT that the drywell-to-wetwell bypass leakage can be measured and any adverse condition detected prior to a Loss of Coolant Accident (LOCA). Additionally, operator action to use containment sprays or RPV depressurization will mitigate the consequences of a bypass area failure during a small break LOCA (the LOCA of most significance for bypass).

Table ES-1

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL

EPRI Category	Category Description	Dose (Person-Rem Within 50-miles)[1]	Quantitative Results as a Function of ILRT Interval			
			Current (10-year ILRT Internal)		Proposed (15-year ILRT Internal)	
			Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50-miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/ Year Within 50-miles)
1	No Containment Failure[2]	6.68E+2	1.76E-6	1.18E-3	1.55E-6	1.03E-3
2	Containment Isolation System Failure	3.46E+5	4.43E-9	1.53E-3	4.43E-9	1.53E-3
3a	Small Pre-Existing Failures[2], [3]	6.68E+3	3.97E-7	2.65E-3	5.95E-7	3.97E-3
3b	Large Pre-Existing Failures[2], [3]	2.34E+4	3.97E-8	9.27E-4	5.95E-8	1.39E-3
4	Type B Failures (LLRT)[4]	N/A	N/A	N/A	N/A	N/A
5	Type C Failures (LLRT)[4]	N/A	N/A	N/A	N/A	N/A
6	Other Containment Isolation System Failure	N/A	N/A	N/A	N/A	N/A
7	Containment Failure Due to Severe Accident	2.57E+5	4.97E-6	1.28	4.97E-6	1.28
8	Containment Bypass Accidents	3.46E+5	1.57E-7	5.43E-2	1.57E-7	5.43E-2
TOTALS:			7.33E-6	1.34	7.33E-6	1.34
Increase in Dose Rate						neg.[5]
Increase in LERF					1.98E-8	
Increase in CCFP (%)					0.1%	

Notes to Table ES-1:

(1) The population dose associated with the Technical Specification Leakage is based on scaling the population data, the power level, and allowable Technical Specification leakage compared to the NUREG/CR-4551 reference plant, Peach Bottom.

(2) Only EPRI categories 1, 3a, and 3b are affected by ILRT (Type A) interval changes.

(3) Dose estimates for categories 3a and 3b, per the NEI Interim Guidance, are calculated as 10xCategory 1 dose and 35xCategory 1 dose, respectively.

(4) An LLRT is a local leak rate test which is not affected by the ILRT frequency.

(5) The change in dose rate from the 1-per-10 year ILRT frequency to the 1-per-15 year ILRT frequency is approximately 2E-3 person-rem/yr. This change is less than the number of significant figures being reasonably carried in the assessment.

SECTION 1
INTRODUCTION

1.1 PURPOSE

The purpose of this analysis is to provide an assessment of the risk associated with implementing a one-time extension of the Columbia Generating Station (CGS) containment Type A integrated leak rate test (ILRT) interval from ten years to fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], the NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals [3], NEI Additional Information for ILRT Extensions [21], and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PSA) findings and risk insights in support of a request for a change in a plant's licensing basis as outlined in Regulatory Guide 1.174 [4].

This analysis also provides a risk assessment of extending the plant's Drywell to Wetwell Bypass Leak Rate Test interval (BLRT) from 2 to 10 years. This risk assessment is performed in Appendix E separate from the Type A Test assessment in the main body of the calculation. The BLRT risk assessment is performed in accordance with the guidelines set forth in NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a licensee request for changes to a plant's licensing basis, Reg. Guide 1.174 [4].

1.2 BACKGROUND

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing requirements from three-in-ten years to at least once per ten years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage is less than normal containment leakage of 1.0La (allowable leakage).

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Type A test frequency per Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493 [5], "Performance-Based Containment Leak Test Program," September 1995, provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285. [2]

The NRC report, Performance Based Leak Test Program, NUREG-1493 [5], analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined for a comparable BWR plant that increasing the containment leak rate from the nominal 0.5 percent per day to 5 percent per day leads to a barely perceptible increase in total population exposure, and increasing the leak rate to 50 percent per day increases the total population exposure by less than 1 percent. Consequently, extending the ILRT interval should not lead to any substantial increase in risk. The current analysis is being performed to confirm these conclusions based on CGS specific models and available data.

Earlier ILRT frequency extension submittals have used the EPRI TR-104285 methodology to perform the risk assessment. In November and December 2001, NEI issued enhanced guidance (hereafter referred to as the NEI Interim Guidance) that builds on the EPRI TR-104285 methodology and is intended to provide for more consistent submittals to the NRC. [3,21] The NEI Interim Guidance was developed for NEI by EPRI using personnel who also developed the EPRI TR-104285 methodology. This CGS ILRT interval extension risk assessment employs the NEI Interim Guidance methodology.

The NEI methodology utilizes pre-existing ILRT-detectable leakage probabilities reflective of a 3-per-10 year ILRT frequency. This 3-per-10 year frequency is utilized in this assessment and is termed the baseline case. Since 1996, the CGS plant has been operating under a 1-per-10 year ILRT testing frequency consistent with the performance based Option B of 10 CFR Part 50, Appendix J [16]. This 1-per-10 year frequency is referred to as the "current" frequency in this assessment. The 1-per-15 year frequency is referred to as the "proposed" frequency in this assessment. The risk impacts of primary interest in this study are those associated with increasing the ILRT test interval from the current 1-per-10 year frequency to the proposed 1-per-15 year frequency.

Consistent with the NEI methodology, long term station blackout (LT SBO) scenarios are not included in pertinent EPRI category frequency calculations in the evaluation of the ILRT frequency extension assessment. The Nuclear Regulatory Commission (NRC) has consistently asked licensees, via Requests for Additional Information (RAIs), to provide an assessment of the impact on risk results if long term station blackout sequences are retained in selected EPRI categories. This ILRT extension assessment includes a sensitivity case to demonstrate that retaining LT SBO sequences does not change the conclusions of the overall assessment.

It should be noted that, in addition to ILRT tests, containment leak-tight integrity is also verified through periodic in-service inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. More specifically, Subsection IWE provides the rules and requirements for in-service inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E), require licensees to conduct visual inspections of the accessible areas of the interior of the containment 3 times every 10 years. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency. Type C tests are also not affected by the Type A test frequency change.

In response to previous ILRT extension request submittals, the NRC has consistently requested licensees to perform a quantitative assessment of the impact on LERF due to age-related degradation of non-inspectable areas of the containment. Therefore, a quantitative assessment using the same approach used by other industry plants (e.g., Calvert Cliffs) is included as Appendix D to this ILRT extension evaluation. This sensitivity case demonstrates that age-related degradation of non-inspectable areas of the containment does not change the conclusions of the overall assessment.

1.3 CRITERIA

Based on previously approved ILRT extension requests, CGS uses the following risk metrics to characterize the change in risk associated with the one time ILRT extension:

- Change in Large Early Release Frequency (LERF)
- Change in conditional containment failure probability
- Change in population dose rate (person-rem/yr)

Consistent with the NEI Interim Guidance, the acceptance guidelines in Regulatory Guide 1.174 [4] are used to assess the acceptability of this one-time extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J.

RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (ΔCDF) less than 10^{-6} per reactor year and increases in large early release frequency (ΔLERF) less than 10^{-7} per reactor year. Since the Type A test does not impact the at-power CDF[(1)], the relevant criterion is the change in LERF. This approach is consistent with previous BWR ILRT submittals using the NEI Guidance and approved by the NRC. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to show that key principles, such as the defense-in-depth philosophy, are met. Therefore, the increase in the conditional containment failure probability, which helps to ensure that the defense-in-depth philosophy is maintained, will also be calculated. Figure 2 shows the acceptance guidelines for ΔLERF.

In addition, based on the precedent of other ILRT extension requests [6], the total annual risk (person-rem/yr population dose rate) and the conditional containment failure probability are examined to demonstrate the relative change in risk. (No threshold has been established for these parameter changes in RG 1.174.)

(1) It is noted that catastrophic containment failure would result in impacts on the mitigating systems, and therefore affect CDF. However, the containment failure sizes identified here that are prevented by ILRT are not sufficient to challenge the mitigating systems. Conversely, there is some probability that the leakage from containment could act as a beneficial influence by creating a self-relieving vent that removes decay heat and prevents containment overpressure conditions. This latter "benefit" is not credited with reducing the CDF in the CGS ILRT assessment presented in this report.

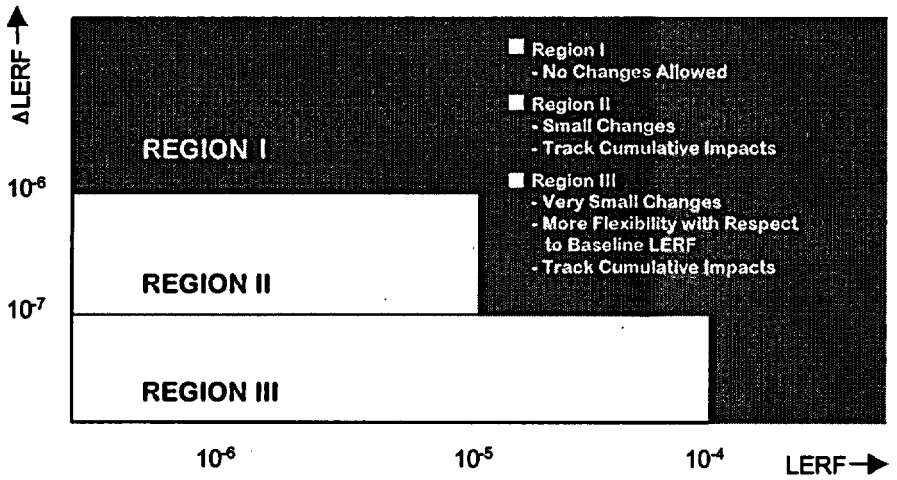

Figure 2 Acceptance Guidelines* for Large Early Release Frequency (LERF)

* The analysis will be subject to increased technical review and management attention as indicated by the darkness of the shading of the figure. In the context of the integrated decision-making, the boundaries between regions should not be interpreted as being definitive; the numerical values associated with defining the regions in the figure are to be interpreted as indicative values only.

SECTION 2
METHODOLOGY

This section provides the following methodology related items:

- A brief summary of available resource documents to support the methodology
- The NEI Interim Guidance for the analysis approach to be used
- The assumptions used in the evaluation
- The inputs required
  - Generic ex-plant consequence
  - Plant specific inputs

The following subsections address these items.

2.1 GENERAL RESOURCES AVAILABLE

This section summarizes the general resources available as input. Various industry studies on containment leakage risk assessment are briefly summarized here:

1) NUREG/CR-3539 [10]
2) NUREG/CR-4220 [11]
3) NUREG-1273 [12]
4) NUREG/CR-4330 [13]
5) EPRI TR-105189 [8]
6) NUREG-1493 [5]
7) EPRI TR-104285 [2]
8) NEI Interim Guidance [3, 21]
9) NUREG-1150 [14] and NUREG/CR-4551 [9]

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PSA for the size of containment leakage that is considered significant and to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh and eighth studies are EPRI studies of the impact of extending ILRT and LLRT test intervals on at-power public risk. The ninth study provides consequence evaluations that can be used as surrogate results when plant specific characteristics are not available.

NUREG/CR-3539 [10]

Oak Ridge National Laboratory (ORNL) documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [15] as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

NUREG/CR-4220 [11]

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories (PNL) for the NRC in 1985. The study reviewed over two thousand LERs, ILRT reports and other related records to calculate the unavailability of containment due to leakage. The study calculated unavailabilities for Technical Specification leakages and "large" leakages.

NUREG/CR-4220 assessed the "large" containment leak probability to be in the range of 1E-3 to 1E-2, with 5E-3 identified as the point estimate based on 4 events in 740 reactor years and conservatively assuming a one-year duration for each event.

NUREG-1273 [12]

A subsequent NRC study, NUREG-1273, performed a more extensive evaluation of the NUREG/CR-4220 database. This assessment noted that about one-third of the reported events were leakages that were immediately detected and corrected. In addition, this study noted that local leak rate tests can detect "essentially all potential degradations" of the containment isolation system.

NUREG/CR-4330 [13]

NUREG/CR-4330 is a study that examined the risk impacts associated with increasing the allowable containment leakage rates. The details of this report have no direct impact on the modeling approach of the ILRT test interval extension, as NUREG/CR-4330 focuses on leakage rate and the ILRT test interval extension study focuses on the frequency of testing intervals. However, the general conclusions of NUREG/CR-4330 are consistent with NUREG/CR-3539 and other similar containment leakage risk studies:

> "...the effect of containment leakage on overall accident risk is small since risk is dominated by accident sequences that result in failure or bypass of containment."

EPRI TR-105189 [8]

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because this EPRI study provides insight regarding the impact of containment testing on shutdown risk. This study performed a quantitative evaluation (using the EPRI ORAM

software) for two reference plants (a BWR/4 and a PWR) of the impact of extending ILRT and LLRT test intervals on shutdown risk.

The result of the study concluded that a small but measurable safety benefit is realized from extending the test intervals. For the BWR, the benefit from extending the ILRT frequency from 3 per 10 years to 1 per 10 years was calculated to be a reduction of approximately 1E-7/yr in the shutdown core damage frequency. This risk reduction is due to the following issues:

- Reduced opportunity for draindown events
- Reduced time spent in configurations with impaired mitigating systems

The study identified 7 shutdown incidents (out of 463 reviewed) that were caused by ILRT or LLRT activities. Two of the 7 incidents were RCS draindown events caused by ILRT/LLRT activities. The other 5 events involved loss of RHR and/or SDC due to ILRT/LLRT activities. This information was used in the EPRI study to estimate the safety benefit from reductions in testing frequencies. This represents a valuable insight into the improvement in the safety due to extending the ILRT test interval.

NUREG-1493 [5]

NUREG-1493 is the NRC's cost-benefit analysis for proposed alternatives to reduce containment leakage testing intervals and/or relax allowable leakage rates. The NRC conclusions are consistent with other similar containment leakage risk studies:

- Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk.
- Increasing containment leak rates several orders of magnitude over the design basis would minimally impact (0.2 – 1.0%) population risk.
- Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

EPRI TR-104285 [2]

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending Integrated Leak Rate Test (ILRT) and (Local Leak Rate Test) LLRT test intervals on at-power public risk. This study combined IPE Level 2 models with NUREG-1150 Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 used a simplified Containment Event Tree to subdivide representative core damage sequences into eight (8) categories of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures due to support system or active failures
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failure due to core damage accident phenomena
8. Containment bypass

Consistent with the other containment leakage risk assessment studies, this study concluded:

> *"These study results show that the proposed CLRT* [containment leak rate tests] *frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.02 person-rem per year . . ."*

NEI Interim Guidance [3, 21]

NEI "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions of Containment Integrated Leakage Rate Test Surveillance Intervals" [3] has been developed to provide utilities with revised guidance regarding licensing submittals. Additional information from NEI on the "Interim Guidance" was supplied in Reference [21].

A nine-step process is defined which includes changes in the following areas of the previous EPRI guidance [2]:

- Impact of extending surveillance intervals on dose
- Method used to calculate the frequencies of leakages detectable only by ILRTs
- Provisions for using NUREG-1150 dose calculations to support the population dose determination.

This NEI Guidance is used in the CGS ILRT analysis.

NUREG-1150 [14] and NUREG/CR-4551 [9]

NUREG-1150 and the technical basis, NUREG/CR-4551, provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Technical Specification leakage). The ex-plant consequences from NUREG-1150 have been used as surrogate results by other BWRs for ILRT evaluation. BWR consequence results are provided for Peach Bottom (Mark I containment) and Grand Gulf (Mark III containment). Although the NUREG-1150 series did not evaluate a BWR with a Mark II containment, such as that for CGS, Mark II containments are very similar to Mark I containments such that Mark I consequence results are judged applicable to Mark II containments. Use of Mark I consequence results for a Mark II plant was specifically performed in the EPRI TR-104285 study

discussed previously. The ex-plant consequences from NUREG/CR-4551 for Peach Bottom are therefore used for the CGS ILRT evaluation.

2.2 NEI INTERIM GUIDANCE

The CGS risk assessment analysis uses the approach outlined in the NEI Interim Guidance. [3,21] The nine steps of the methodology are:

1. Quantify the baseline (nominal three year ILRT interval) frequency per reactor year for the EPRI accident categories of interest. Note that EPRI categories 4, 5, and 6 are not affected by changes in ILRT test frequency.
2. Determine the containment leakage rates for EPRI categories 1, 3a and 3b.
3. Develop the baseline population dose (person-rem) for the applicable EPRI categories.
4. Determine the population dose rate (person-rem/year) by multiplying the dose calculated in Step (3) by the associated frequency calculated in Step (1).
5. Determine the change in probability of leakage detectable only by ILRT, and associated frequency for the new surveillance intervals of interest. Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rate are assumed not to change, however the probability of leakage detectable only by ILRT does increase.
6. Determine the population dose rate for the new surveillance intervals of interest.
7. Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases.
8. Evaluate the risk impact in terms of LERF.
9. Evaluate the change in conditional containment failure probability.

The first seven steps of the methodology calculate the change in dose. The change in dose is the principal basis upon which the Type A ILRT interval extension was previously granted and is a reasonable basis for evaluating additional extensions. The

eighth step in the interim methodology calculates the change in LERF and compares it to the guidelines in Regulatory Guide 1.174. Because there is no change in CDF, the change in LERF forms the quantitative basis for a risk informed decision per current NRC practice, namely Regulatory Guide 1.174. The ninth and final step of the interim methodology calculates the change in containment failure probability. The NRC has previously accepted similar calculations (Ref. [7], referred to as conditional containment failure probability, CCFP) as the basis for showing that the proposed change is consistent with the defense in depth philosophy. As such this last step suffices as the remaining basis for a risk informed decision per Regulatory Guide 1.174.

2.3 GROUND RULES

The following ground rules are used in the analysis:

- The CGS Level 1 and Level 2 internal events PSA model provides representative results for the analysis.
- It is appropriate to use the CGS internal events PSA model as a gauge to effectively describe the risk change attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose rate) from external events can be included in the calculations by sensitivity studies.
- An evaluation of the risk impact of the ILRT on shutdown risk is addressed using the generic results from EPRI TR-105189 [8] as augmented by NEI Interim Guidance. [3, 21]
- Radionuclide release categories are defined consistent with the EPRI TR-104285 methodology. [2]
- The CGS ex-plant consequence in terms of population dose results for the containment failures modeled in the PSA can be characterized by information provided in NUREG/CR-4551 [9]. They are estimated by scaling the NUREG/CR-4551 population dose results by power level, population, and Tech Spec leak rate differences for CGS compared to the NUREG/CR-4551 Mark I reference plant, Peach Bottom. Use of NUREG/CR-4551 [9] Mark I dose results for a Mark II plant is acceptable and is consistent with EPRI TR-104285 [2]. Use of dose results for the 50-mile radius around the plant as a figure of merit in the risk evaluation is consistent with NUREG-1150, past ILRT frequency extension submittals, and the NEI Interim Guidance.

- Per the NEI Interim Guidance [3], the representative containment leakage for EPRI Category 1 sequences is 1 L_a (L_a is the Technical Specification maximum allowable containment leakage rate).
- Per the NEI Interim Guidance [3], the representative containment leakage for EPRI Category 3a sequences is 10 L_a.
- Per the NEI Interim Guidance [3], the representative containment leakage for EPRI Category 3b sequences is 35 L_a.
- EPRI Category 3b is conservatively categorized as LERF based on the previously approved NEI methodology [3] despite the fact that the radionuclide release magnitude is significantly less than a large release.
- The impact on population doses from Interfacing System LOCAs is not altered by the proposed ILRT extension. The ISLOCA contribution to risk is accounted for in the EPRI methodology as a separate entry.
- The containment isolation valve test frequency is not altered. Therefore, the reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.

2.4 PLANT SPECIFIC INPUTS

The inputs to the risk assessment include the following:

- Past CGS ILRT results to demonstrate the adequacy of the administrative and hardware issues.
- Ex-plant consequence evaluation from NUREG/CR-4551 for a Mark I plant (Peach Bottom)
- CGS PSA Level 1 & 2, Rev. 5
- CGS specific adjustments to ex-plant consequence evaluation from NUREG/CR-4551

2.4.1 Prior CGS ILRT Results

The surveillance frequency for Type A testing in NEI 94-01 is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate was less than 1.0 L_a) and consideration of the performance factors in NEI 94-01, Section 11.3. Based

on the consecutive successful ILRTs performed in 1991[1] and 1994[2], the current ILRT interval for CGS is once per ten years. [16]

2.4.2 Ex-Plant Consequences

Consistent with the NEI Interim Guidance [3] and the supplemental information [21], ex-plant consequence evaluations from NUREG-1150 can be used in the ILRT evaluation to support the population dose estimate.

Figure 2-1 is a simplified flow chart that shows the process for determining the CGS specific population dose (person-rem) for comparable radionuclide release categories starting with the NUREG/CR-4551 (Peach Bottom) ex-plant consequence evaluation and correcting for key differences.

The surrogate plant consequence analysis for Peach Bottom is calculated for the 50-mile radial area surrounding Peach Bottom (A). The ex-plant calculation is delineated by total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551 (B). The best matching APB of NUREG/CR-4551 is assigned to each of the CGS Level 2 Model end state bins as a means to estimate population dose associated with the CGS Level 2 bins.

In order to convert the Peach Bottom APB population dose estimates for use in the CGS consequence evaluation, adjustments to these ex-plant consequences are needed to account for the following (D, F, G):

- Population differences
- Containment leakage rate
- Power level

Finally, the CGS specific ex-plant consequences are calculated (H).

[1] 1991 ILRT results = 0.319 weight %/day
[2] 1994 ILRT results = 0.330 weight %/day

The parameters that were used in the Peach Bottom analysis from NUREG/CR-4551 for comparison with CGS are the following:

- Peach Bottom Population out to 50 miles = 3.2E+6 persons
  (See Appendix A for derivation)
- Peach Bottom Power level = 3293 MWt
  (See Table 4.2-1 of Reference [9])
- Peach Bottom Containment leak rate = 0.5%/day[1]

While meteorology could play a role in the early health effects calculations, the meteorology and site topography for Peach Bottom and CGS are assumed to be sufficiently similar that these differences are assumed not to play a significant role in this evaluation of total population dose.

2.4.3 Plant Specific Inputs

The CGS specific information used to perform this ILRT interval extension risk assessment includes the following:

- CGS Level 1 PSA Model Rev. 5
- CGS Level 2 PSA LERF Model Rev. 5
- CGS Plant and Site Characteristics
  - Population out to 50 miles = 3.6E+5 persons (year 2000)
    (See Appendix A for derivation)
  - Power Level = 3486 MWt
    (Includes recent 5% power uprate)
  - Containment Leakage Rate = 0.5%/day
    (CGS Technical Specification Section 5.5.12)

(1) The analysis performed in NUREG/CR-4551 used a leakage of 0.5%/day (Vol. 4, Rev. 1, Part 2, page B.2-9 of Reference [9]). The current Peach Bottom Technical Specification leakage may differ.

2.4.3.1 CGS PSA

The CGS Level 1 and 2 PSA (Rev. 5) used as input to this analysis is characteristic of the as-built, as-operated plant. The Rev. 5 PSA model is the latest CGS model and includes internal flooding. The model is developed in WinNUPRA.

The CGS total core damage frequency (CDF) as reported in the CGS Level 1 Quantification Notebook is 7.33E-6/yr (mean value) at a truncation of 5E-12/yr[1] [18]. Table 2-1 summarizes the CGS Level 1 PSA frequency results by plant damage state (PDS). The primary categorization scheme used to bin the CGS Level 1 sequences is accident type.

The CGS Level 2 PSA is used to calculate the release frequencies for the accidents evaluated in this assessment. Table 2-2 summarizes the CGS Level 2 PSA results for containment failure. The total release frequency is 5.13E-6/yr; with a total CDF of 7.33E-6/yr. [22] The "No Release" frequency (i.e., containment leakage within Technical Specifications) for CGS is 2.20E-6/yr. [22] Table 2-3 summarizes the LERF contribution associated with each CGS plant damage state.

2.4.4. Adjustments to Ex-plant Consequence Calculations

This NUREG/CR-4551 ex-plant consequence analysis is calculated for the 50-mile radial area surrounding Peach Bottom, and is reported in total person-rem for discrete accident categories (termed Accident Progression Bins (APB) in NUREG/CR-4551). To use the NUREG/CR-4551 consequences in this ILRT risk assessment, the following steps should first be performed:

---

(1) WinNUPRA has a 60,000 cutset limit and utilizes three distinct truncation limits. The following truncation limits are used in PSA model Rev. 5:

- Fault Tree truncation: 2E-9 (used to derive functional equations)
- Event Tree truncation: 5E-12 (used to merge functional equations)
- Global Core Damage equation truncation: 5E-12 (used to quantify the CDF)

- Adjust the person-rem results to account for differences between the Peach Bottom analyses in NUREG/CR-4551 and the CGS plant and its demographics:
  - Population
  - Reactor Power Level
  - Technical Specification Allowed Containment Leakage Rate
- Assign the adjusted NUREG/CR-4551 APB consequences to the EPRI categories used in this risk assessment

2.4.4.1 Surrounding Population

The 50-mile radius population used in the Peach Bottom NUREG/CR-4551 consequence calculations is 3.2E+6 persons (refer to Appendix A of this report).

The year 2000 population within the 50-mile radius of CGS is estimated in Appendix A of this report at 3.6E+5 persons.

The ratio of the population surrounding CGS to that in the Peach Bottom analysis results in a factor of:

$$\frac{3.6E+5 \text{ persons}}{3.2E+6 \text{ persons}} = 0.11$$

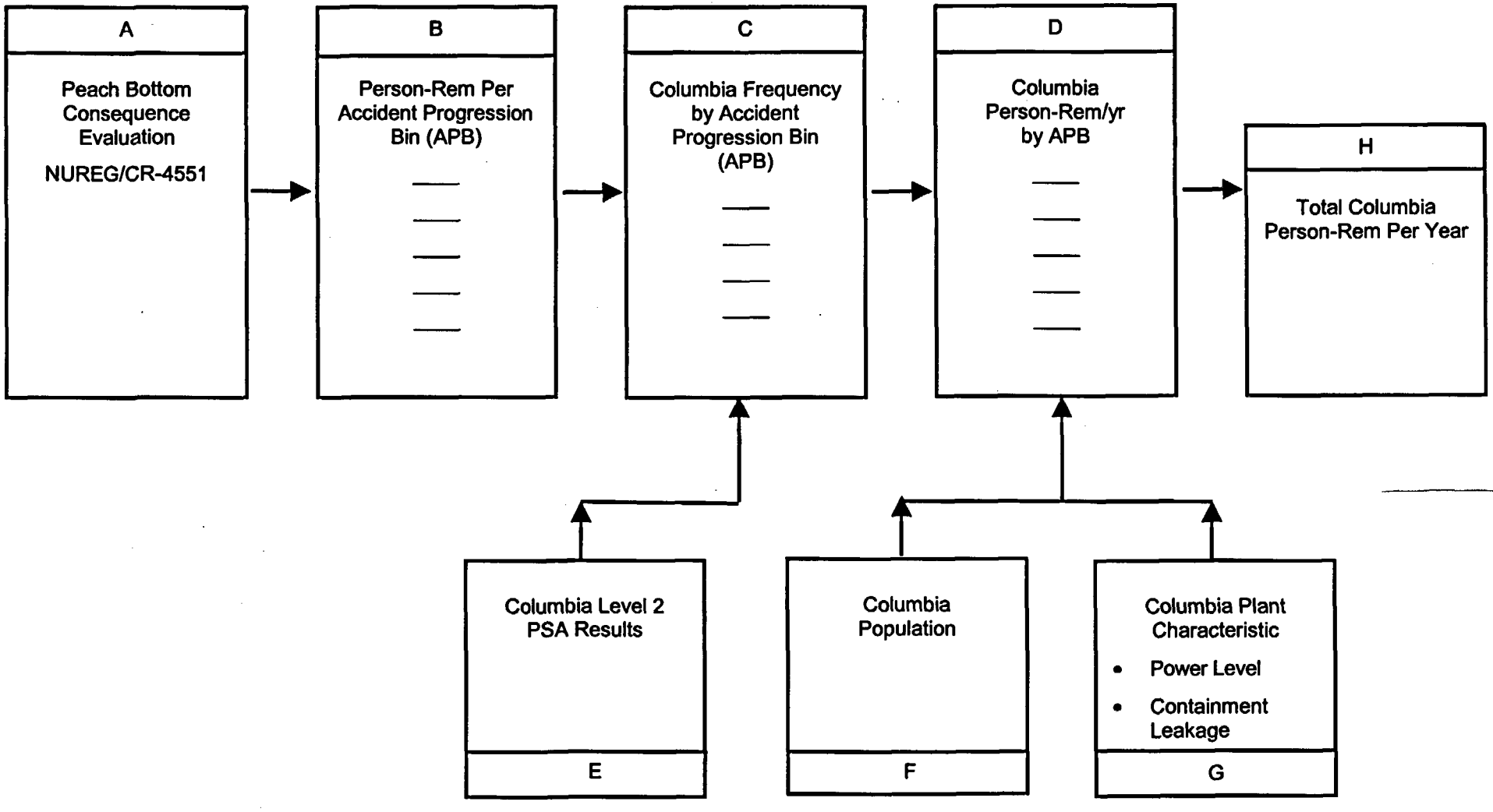

Figure 2-1 Process for Calculating Population Dose for CGS Using the Surrogate Plant Results from NUREG/CR-4551

2.4.4.2 Reactor Power Level

The Peach Bottom reactor power level used in the NUREG/CR-4551 consequence calculations is 3293 MWt. CGS recently performed a power uprate of 5% over the originally licensed thermal power; the CGS full power level is now 3486 MWt.

The CGS power level used in this ILRT evaluation is the power uprate power level of 3486 MWt. This represents a factor of 1.06 = (3486 MWt/3293 MWt) change in the population dose for each Peach Bottom APB.

2.4.4.3 Technical Specification Containment Leakage

The Peach Bottom analysis in NUREG/CR-4551 assumes containment leakage of 0.5%/day (see Vol. 4, Rev. 1, Part 2, page B.2-9 of Reference [9]).

The CGS Technical Specification leakage is 0.5%/day. Because the source term leakage is a function of containment volume, the following plant characteristics are also needed:

- Peach Bottom Containment Volume:

	min (ft^3)	max (ft^3)	average (ft^3)
Drywell free volume[1]	1.59E+5	1.76E+5	1.67E+5
Supp. Pool free volume[1]	1.28E+5	1.32E+5	1.30E+5
TOTAL	--	--	2.97E+5

[1] NUREG/CR-4551, Vol. 4, Part 2, Section A.3.1.

- CGS Containment Volume:

	min (ft^3)	max (ft^3)	average (ft^3)
Drywell free volume[2]	–	–	2.01E+5
Supp. Chamber free volume[2]	1.43E+5	1.47E+5	1.45E+5
TOTAL	–	–	3.46E+5

For this comparison, the following factor can be developed to relate the leakage impact between the two plants:

$$\frac{\text{Total Leakage CGS}}{\text{Total Leakage Peach Bottom}} = \frac{0.50^{CGS}\ \%/\text{day}}{0.50^{PB}\ \%/\text{day}} \times \frac{3.46\text{E+5 ft}^3\ \text{Vol.}^{CGS}}{2.97\text{E+5 ft}^3\ \text{Vol.}^{PB}}$$

$$\frac{\text{Total Leakage for CGS}}{\text{Total Leakage for Peach Bottom}} = 1.0 \times 1.16 = 1.16$$

This represents a factor of 1.16 increase in the person-rem consequence for the "intact" containment Accident Progression Bin (APB) of NUREG/CR-4551.

2.4.4.4 Summary

The factors that are calculated for use in adjusting the population dose (person-rem) of the surrogate plant (NUREG-1150 Peach Bottom) for the site and plant differences are as follows:

- Consequence categories dependent on the "intact" Tech Spec Leakage are adjusted as follows:

  $F_{INTACT} = F_{POPULATION} \times F_{POWER} \times F_{TS\ LEAK}$

  $F_{INTACT} = 0.11 \times 1.06 \times 1.16$

  $F_{INTACT} = 0.14$

[2] CGS UFSAR, Table 6.2-1 provides the minimum free volume. The maximum free volume is an estimate based on comparisons with other plants.

- Consequence categories not dependent on the Tech Spec Leakage are adjusted as follows:

  $F_C = F_{POPULATION} \times F_{POWER}$

  $F_C = 0.11 \times 1.06$

  $F_C = 0.12$

These factors are applied to the individual NUREG/CR-4551 bins in Table 3-4 of the analysis section of this report.

Table 2-1

CORE DAMAGE FREQUENCY CONTRIBUTIONS BY PDS[2] [22][3]

Contributing Plant Damage State		Core Damage Frequency[1]	% of CDF
PDS Class			
I - Transient and Small LOCA with Loss of RPV Injection Capability	IA1 - Short term TUX with loss of containment air	2.58E-8	0.4
	IA2 - Short term TUX with offsite power available	7.32E-7	10.0
	IA3 - Long term TUX for LOSP with at least 1 DG available	1.12E-7	1.5
	IB0 - Loss of containment heat removal with failure of HPCS	6.92E-7	9.4
	IC - Loss all ECCS due to flooding	1.88E-7	2.6
	IG - Long term TUV with offsite power available	1.38E-6	18.8
	IH -Long term TUV for LOSP with at least 1 DG available	1.80E-7	2.5
II - Transient with Loss of Containment Heat Removal	IIB - Long term TW with stuck open PORV	8.11E-9	0.1
	IID - Long term TW	1.11E-6	15.2
III - LOCAs	IIIC - Reactor Vessel Rupture	3.00E-7	4.1
	IIIE - Large LOCA with failure of containment pressure suppression	0.0	0.0
IV - ATWS	IVBA - ATWS with vessel intact at time of core uncovery	1.24E-7	1.7
	IVBL - ATWS with vessel failed at time of core uncovery	6.25E-8	0.9
V - LOCA (BOC)	V - Large LOCA Outside of Containment	1.57E-7	2.1
VI - Station Blackout	VIA1 - Short term (<2hr.) DC power and ADS available at time of core melt.	9.75E-7	13.3
	VIA2 - Long term (>6hr.) DC (and ADS) power not available at time of core melt. Stuck Open SRV.	3.72E-8	0.5
	VIB1 - Long term (>6hr.) DC power not available at time of core melt. HPCS recoverable with recovery of AC power.	1.03E-6	14.0
	VIB2 - Long term (>6hr.) DC power not available at time of core melt. HPCS not recoverable.	2.12E-7	2.9
Total		7.33E-6	100

(1) All frequencies in events per reactor year.

(2) CGS plant damage states (PDS) are based on accident type binning.

(3) Source: Table 4.2-2 and Table 4.2-3 of Reference [22]

Table 2-2

SUMMARY OF CGS LEVEL 2 PSA RESULTS [22][(2)]

Level 2 End State	Frequency (per year)[(1)]	% CDF
Containment Intact (Tech Spec leakage)	2.20E-6	30%
Containment Failure[(3)]		
Large Early Release - not scrubbed	6.9E-7	9.3%
Large Late Release - not scrubbed	3.80E-6	52%
Large Late Release - scrubbed	6.4E-7	8.7%
Total	7.33E-6	100

(1) All frequencies in events per reactor year.

(2) Source: Table 4.5.1.1 of Reference [22]

(3) Distinctions between the containment failure releases are as follows (Section 5.3 of Reference [22]):

- "Large" refers to the containment failure size. Previous versions of the CGS L2 PSA classified containment breaks as large (i.e., 28 in^2 leak area) or small (i.e., 6-inch diameter hole). This small size break has been shown in other studies to result in a large release in terms of LERF. Therefore, CGS presently classifies all break sizes as large for LERF analysis.
- "Early/Late" designation is based on an assumed evacuation time of 4 hours. If the release occurs prior to evacuation of the immediate population, the release is considered "early". If the release occurs after the immediate population evacuation, the release is considered "late".
- "Scrubbing" occurs if debris is flooded or the suppression pool is not bypassed. (Scrubbing by sprays is not credited). A release via a scrubbed pathway is considered "small" in magnitude, regardless of the containment break size.

Table 2-3

LERF SPLIT FRACTION FOR EACH PDS[2] [22][3]

Plant Damage State		LERF Split Fraction	LERF Contribution[1]
PDS Class			
I - Transient and Small LOCA with Loss of RPV Injection Capability	IA1	.108	2.78E-9
	IA2	.108	7.88E-8
	IA3A	.0062	2.30E-10
	IA3B	.0435	3.32E-9
	IB0	0.0	0.0
	IC	1.00	1.88E-7
	IG	7.88E-4	1.08E-9
	IHA	7.88E-4	6.06E-11
	IHB	7.88E-4	8.11E-11
II - Transient with Loss of Containment Heat Removal	IIB	0.0	0.0
	IID	0.0	0.0
III - LOCAs	IIIC	7.88E-4	2.31E-10
IV - ATWS	IVBA	1.0	1.24E-7
	IVBL	1.0	6.25E-8
V - LOCA (BOC)	V	1.0	1.57E-7
VI - Station Blackout	VIA1A	.068	1.99E-8
	VIA1B	.068	4.75E-8
	VIA2	0.00	0.00
	VIB1	0.00	0.00
	VIB2A	0.00	0.00
	VIB2B	0.00	0.00
TOTAL			6.9E-7

(1) All frequencies in events per reactor year.

(2) CGS plant damage states (PDS) are based on accident type binning.

(3) Source: Table 5.6-1 of Reference [22]

SECTION 3
ANALYSIS

This section provides a step-by-step summary of the NEI guidance as applied to the CGS ILRT interval extension risk assessment. Each subsection addresses a step or group of steps in the NEI guideline.

3.1 BASELINE ACCIDENT CATEGORY FREQUENCIES (STEP 1)

The first step of the NEI Interim Guidance is to quantify the baseline frequencies for each of the EPRI TR-104285 accident categories. This portion of the analysis is performed using the CGS Level 1 and Level 2 PSA results. The results for each EPRI category are described below.

Tables 2-1, 2-2 and 2-3 of Section 2 compiled from the CGS PSA [18, 22] are used for the inputs to the accident frequency assessment.

Frequency of EPRI Category 1

This group consists of all core damage accident sequences in which the containment is initially isolated and remains intact throughout the accident (i.e., containment leakage at or below maximum allowable Technical Specification leakage). The ILRT methodology artificially divides this category among the Tech Spec leakage case (Category 1) and two other categories that are used to simulate possible changes due to reduced ILRT frequencies (i.e., Categories 3a and 3b; see below for their definition). Per NEI Interim Guidance, the frequency per year for this category is calculated by subtracting the frequencies of EPRI Categories 3a and 3b (see below) from the sum of all severe accident sequence frequencies in which the containment is initially isolated and remains intact (i.e., accidents classified as "OK" in the CGS Level 2 PSA).

As discussed previously in Section 2.4, the frequency of the CGS Level 2 PSA "OK" or "No Release" accident bin is 2.20E-6/yr (see Table 2-2). As described below, the frequencies of EPRI Categories 3a and 3b are 1.19E-7/yr and 1.19E-8/yr, respectively. Therefore, the frequency of EPRI Category 1 is calculated as 2.20E-6/yr – 1.19E-7/yr - 1.19E-8/yr = 2.07E-6/yr for the assumed 3-year ILRT interval for which the ILRT data has been collected.

Frequency of EPRI Category 2

This group consists of all core damage accident sequences in which the containment isolation system function fails during the accident progression (e.g., due to failures-to-close of large containment isolation valves initiated by support system failures, or random or common cause valve failures).

The frequency of this EPRI category is estimated by multiplying the conditional probability of containment isolation failure from the CGS Level 2 PSA by the portion of the severe accident sequences (CDF) that would be challenged. The CGS plant damage state (PDS) sequences that have containment isolation already failed at the time of core damage are PDS Class IC, Class II, Class IIIE, Class IV, and Class V.[1] Therefore, the EPRI Category 2 CDF does not include these CGS Level 1 accident sequences. The following values are used for this calculation:

- Containment Isolation System failure probability = 7.80E-4 [22][2]
- Total CDF = 7.33E-6/yr [22]
- PDS Class IC sequences = 1.88E-7/yr [22]
- PDS Class II sequences = 1.12E-6/yr [22]
- PDS Class IIIE sequences = 0.0/yr [22]
- PDS Class IV sequences = 1.87E-7/yr [22]
- PDS Class V sequences = 1.57E-7/yr [22]

(1) Source: Table 4.2-1 of Reference [22].

(2) Containment isolation system failure probability based on nodal quantification of event node "ISOL" (7.80E-4) This value represents only active containment isolation failures (i.e., no pre-existing failure probabilities are included). Pre-existing containment failures are evaluated in other EPRI categories.

The frequency per year for this category is calculated as follows:

Frequency 2 = (containment isolation failure probability)
x (CDF - CDF of Class IC - CDF of Class II - CDF of Class IIIE - CDF of Class IV - CDF of Class V)[1]

Frequency 2 = (7.80E-4) X (7.33E-6/yr - 1.88E-7/yr - 1.12E-6/yr - 0.0/yr - 1.87E-7/yr - 1.57E-7/yr)

Frequency 2 = 4.43E-9/yr

Note that pre-existing isolation failures are included in Category 6.

The frequency of EPRI Category 2 is 4.43E-9/yr.

Frequency of EPRI Category 3a

This group consists of all core damage accident sequences in which the containment is failed due to a pre-existing "small" leak in the containment structure or shell that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). Consistent with NEI Interim Guidance [21], the frequency per year for this category is calculated as:

Frequency 3a = (3a conditional failure probability) x (CDF - CDF with independent LERF - CDF that cannot cause LERF)

The 3a conditional failure probability (2.7E-2) value is the conditional probability of having a pre-existing "small" containment leak that is detectable only by ILRTs. This value is derived in Reference [3] and is based on data collected by NEI from 91 plants. This value is also assumed reflective of ILRT testing frequencies of 3 tests in 10 years.

[1] See Table 2-1 of this report for accident class definitions for CGS.

The pre-existing leakage probability is multiplied by the residual core damage frequency (CDF) determined as the total CDF minus the CDF for those individual sequences that either may already (independently) cause a LERF or could never cause a LERF due to the delay time of the release (i.e., non-early). As discussed previously in Section 2.4.2, the CGS total internal events core damage frequency is 7.33E-6/yr. Of this total CDF, the following CGS plant damage state (PDS) sequences result in LERF directly (e.g., containment bypass) or will never result in LERF:

Always LERF[1] (= 5.32E-7/yr)

- Containment Bypass accidents
  PDS Class V = 1.57E-7/yr [22]
- Internal Flooding accidents
  PDS Class IC = 1.88E-7/yr [22]
- ATWS accidents
  PDS Class IVBA = 1.24E-7/yr [22]
  PDS Class IVBL = 6.25E-8/yr [22]
  ATWS Total = 1.87E-7/yr

Never LERF[2] (= 2.40E-6/yr)

- Long Term Station Blackout (SBO) scenarios
  PDS Class VIA2 = 3.72E-8/yr [22]
  PDS Class VIB1 = 1.03E-6/yr [22]
  PDS Class VIB2 = 2.12E-7/yr [22]
  LT SBO Total = 1.28E-6/yr
- Loss of Containment Heat Removal accidents
  PDS Class IIB = 8.11E-9/yr [22]
  PDS Class IID = 1.11E-6/yr [22]
  Class II Total = 1.12E-6/yr

(1) Identified in Table 2-3 of this report as those PDS with LERF split fractions = 1.0.
(2) Identified in Table 2-3 of this report as those PDS with LERF split fractions = 0.0. Fission product release for these scenarios occurs after the "early" time period.

Therefore, the frequency of EPRI Category 3a is calculated as:

2.70E-2 x [7.33E-6/yr - 5.32E-7/yr - 2.40E-6/yr] = 1.19E-7/yr

Frequency of EPRI Category 3b

This group consists of all core damage accident sequences in which the containment is failed due to a pre-existing "large" leak in the containment structure or shell that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). Similar to Category 3a, the frequency per year for this category is calculated as:

Frequency 3b = (3b conditional failure probability) x (CDF - CDF with independent LERF - CDF that cannot cause LERF)

The 3b failure probability (2.7E-3) value is the conditional probability of having a pre-existing "large" containment leak that is detectable only by ILRTs. This value is derived in Reference [3] and is based on data collected by NEI from 91 plants. This value is also assumed reflective of ILRT testing frequencies of 3 tests in 10 years.

Therefore, similar to EPRI Category 3a, the frequency of Category 3b is calculated as:

2.70E-3 x [7.33E-6/yr - 5.32E-7/yr - 2.40E-6/yr] = 1.19E-8/yr

Frequency of EPRI Category 4

This group consists of all core damage accident sequences in which the containment isolation function is failed due to a pre-existing failure-to-seal of Type B component(s) that would not be identifiable by an ILRT. Per NEI Interim Guidance, because this category of failures is only detected by Type B tests and not by the Type A ILRT, this group is not evaluated further in this analysis.

Frequency of EPRI Category 5

This group consists of all core damage accident sequences in which the containment isolation function is failed due to a pre-existing failure-to-seal of Type C component(s) that would not be identifiable by an ILRT. Per NEI Interim Guidance, because this category of failures is only detected by Type C tests and not by the Type A ILRT, this group is not evaluated further in this analysis.

Frequency of EPRI Category 6

This group consists of all core damage accident sequences in which the containment isolation function is failed due to "other" pre-existing failure modes (e.g., pathways left open or valves that did not properly seal following test or maintenance activities) that would not be identifiable by containment leak rate tests. Per NEI Interim Guidance, because this category of failures is not impacted by leak rate tests, this group is not evaluated further in this analysis.

Frequency of EPRI Category 7

This group consists of all core damage accident progression bins in which containment failure is induced by severe accident phenomena (e.g., overpressure). Other severe accidents such as intact containment leakage and containment bypass are accounted for in other EPRI categories. Per NEI Interim Guidance, the frequency per year for this category is based on the plant Level 2 PSA results.

For this analysis, the associated radionuclide releases are based on the application of the CGS Level 2 end state bins to the Peach Bottom Accident Progression Bins (APBs) from NUREG/CR-4551. The Peach Bottom APB definitions are reproduced in Table 3-1. The CGS Level 2 PSA containment event tree sequences could be correlated or binned into groups very similar to those characterized in NUREG/CR-4551. This additional level of differentiation, however, is not necessary. The four CGS bins of Table 2-2 are sufficient for the ILRT analysis since:

- Category 7 is not directly affected by the changes in ILRT interval (only Categories 1, 3a and 3b are affected).
- The doses associated with the Peach Bottom APBs applicable to Category 7 are very similar in magnitude (shown later in this report) such that a finer differentiation scheme would not appreciably alter the resulting dose associated with Category 7.

Therefore, EPRI Category 7 is divided into three sub-categories for mapping to the most appropriate APBs of NUREG/CR-4551. (The fourth CGS bin, Intact Containment, does not apply to EPRI Category 7).

The frequency of each Category 7 subgroup is as follows:

EPRI Category	CGS Bin[1]	CGS Frequency[1]
7a	Large, Early, Not Scrubbed	5.29E-7/yr[2]
7b	Large, Late, Not Scrubbed	3.80E-6/yr
7c	Large, Late, Scrubbed	6.40E-7/yr
	Total:	4.97E-6/yr

Frequency of EPRI Category 8

This group consists of all core damage accident progression bins in which the accident is initiated by a containment bypass scenario (i.e., Break Outside Containment LOCA or Interfacing Systems LOCA, ISLOCA). The frequency of Category 8 is the total frequency of the CGS Level 1 PSA containment bypass scenarios (Class V). Based on the CGS Level 1 PSA results summarized earlier in Table 2-1, the frequency of Category 8 is 1.57E-7/yr.

(1) From Table 2-2. The CGS bin of containment intact does not apply to EPRI Category 7.

(2) The release frequency associated with active containment isolation failure (4.43E-9/yr, EPRI Category 2) and containment bypass scenarios (1.57E-7/yr, EPRI Category 8), have been subtracted from the Table 2-2 total of 6.9E-7/yr of this CGS bin to prevent double counting.

Summary of Frequencies of EPRI Categories

In summary, per the NEI Interim Guidance, the accident sequence frequencies that can lead to radionuclide releases to the public have been derived for accident categories defined in EPRI TR-104285. The accident sequence frequency results by EPRI category are summarized in Table 3-2.

Table 3-1

COLLAPSED ACCIDENT PROGRESSION BIN (APB) DESCRIPTIONS [9][(1)]

Collapsed APB Number	Description
1	CD, VB, Early CF, WW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means Direct Containment Heating (DCH) is possible).
2	CD, VB, Early CF, WW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
3	CD, VB, Early CF, DW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means DCH is possible).
4	CD, VB, Early CF, DW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
5	CD, VB, Late CF, WW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the wetwell (i.e., after vessel breach during Molten Core-Concrete Interaction (MCCI)) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.
6	CD, VB, Late CF, DW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the drywell (i.e., after vessel breach during MCCI) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.

(1) Source: Section 2.4.3 of Reference [9] for NUREG-1150 BWR plant.

Table 3-1

COLLAPSED ACCIDENT PROGRESSION BIN (APB) DESCRIPTIONS [9][(1)]

Collapsed APB Number	Description
7	CD, VB, No CF, Vent, N/A Core Damage occurs followed by vessel breach. The containment never structurally fails, but is vented sometime during the accident progression. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH does not significantly affect the source term as the containment does not fail and the vent limits its effect.
8	CD, VB, No CF, N/A, N/A Core damage occurs followed by vessel breach. The containment never fails structurally (characteristic 4 is N/A) and is not vented. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH did not fail containment. Some nominal leakage from the containment exists and is accounted for in the analysis so that while the risk will be small it is not completely negligible.
9	CD, No VB, N/A, N/A, N/A Core damage occurs but is arrested in time to prevent vessel breach. There are no releases associated with vessel breach or MCCI. It must be remembered, however, that the containment can fail due to overpressure or venting even if vessel breach is averted. Thus, the potential exists for some of the in-vessel releases to be released to the environment.
10	No CD, N/A, N/A, N/A, N/A Core damage did not occur. No in-vessel or ex-vessel release occurs. The containment may fail on overpressure or be vented. The RPV may be at high or low pressure depending on the progression characteristics. The risk associated with this bin is negligible.

Legend

CD = Core Damage

VB = Vessel Breach

CF = Containment Failure

WW = Wetwell

DW = Drywell

RPV = Reactor Pressure Vessel

Table 3-2

SUMMARY OF CGS BASELINE RELEASE FREQUENCIES AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
1	No Containment Failure: Accident sequences in which the containment remains intact and is initially isolated. Only affected by ILRT leak testing frequency due to the incorporation of categories 3a and 3b.	Per NEI Interim Guidance: *[Total CGS "OK" release category frequency] – [Frequency EPRI Categories 3a and 3b]* *2.20E-6/yr – 1.19E-7/yr - 1.19E-8yr = 2.07E-6/yr*	2.07E-6
2	Containment Isolation System Failure: Accident sequences in which the containment isolation system function fails during the accident progression (e.g., due to failures-to-close of large containment isolation valves initiated by support system failures, or random or common cause failures). Not affected by ILRT leak testing frequency.	[CGS containment isolation failure probability] X [Total CDF – CDF of Class II - CDF of Class IIID - CDF of Class IV - CDF of Class V] [7.80E-4] X [7.33E-6/yr – 1.88E-7/yr - 1.12E-6/yr – 1.87E-7/yr - 1.57E-7/yr] = 4.43E-9/yr	4.43E-9
3a	Small Pre-Existing Failures: Accident sequences in which the containment is failed due to a pre-existing small leak in the containment structure or shell that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).	Per NEI Interim Guidance: *[Total CDF - CDF of Always LERF - CDF of Never LERF] x [2.7E-2]* *[7.33E-6/yr - 5.32E-7/yr - 2.40E-6/yr] x [2.70E-02] = 1.19E-7/yr*	1.19E-7
3b	Large Pre-Existing Failures: Accident sequences in which the containment is failed due to a pre-existing large leak in the containment structure or shell that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).	Per NEI Interim Guidance: *[Total CDF - CDF of Always LERF - CDF of Never LERF] x [2.7E-3]* *[7.33E-6/yr - 5.32E-7/yr - 2.40E-6/yr] x [2.70E-03] = 1.19E-8/yr*	1.19E-8
4	Type B Failures: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type B components that would not be identifiable from a ILRT (and thus not affected by ILRT testing frequency).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A

Table 3-2

SUMMARY OF CGS BASELINE RELEASE FREQUENCIES AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
5	Type C Failures: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type C components that would not be identifiable from a ILRT (and thus not affected by ILRT testing frequency).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A
6	Other Containment Isolation System Failure: Accident sequences in which the containment isolation system function fails due to "other" pre-existing failure modes not identifiable by leak rate tests (e.g., pathways left open or valves that did not properly seal following test or maintenance activities). Not affected by ILRT leak testing frequency.	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A
7	Containment Failure Due to Severe Accident: Vessel breach occurs and both the containment and the drywell have failed either before or at the time of vessel breach. The containment sprays do not operate before or at the time of vessel breach.	Assignment of CGS Level 2 sequences to NUREG/CR-4551 APBs. See Table 3-3 7a 5.29E-7/yr 7b 3.80E-6/yr 7c 6.40E-7/yr 4.97E-6/yr	4.97E-6
8	Containment Bypass Accidents: Accident sequences in which the containment is bypassed. Such accidents are initiated by LOCAs outside containment (i.e., Break Outside Containment LOCA, or Interfacing Systems LOCA). Not affected by ILRT leak testing frequency.	*[Total CGS Containment Bypass release frequency]* Class V = 1.57E-7/yr	1.57E-7
		TOTAL:	7.33E-6

3.2 CONTAINMENT LEAKAGE RATES (STEP 2)

The second step of the NEI Interim Guidance is to define the containment leakage rates for EPRI Categories 3a and 3b. As discussed earlier, EPRI Categories 3a and 3b are accidents with pre-existing containment leakage pathways ("small" and "large", respectively) that would only be identifiable from an ILRT.

The NEI Interim Guidance recommends containment leakage rates of $10L_a$ and $35L_a$ for Categories 3a and 3b, respectively. The NEI Interim Guidance describes these two recommended containment leakage rates as "conservative". These values are consistent with previous ILRT frequency extension submittal applications. L_a is the plant Technical Specification maximum allowable containment leak rate; for CGS L_a is 0.5% of containment air weight per day (per CGS Technical Specifications).

The NEI recommended values of $10L_a$ and $35L_a$ are used as is in this analysis to characterize the containment leakage rates for Categories 3a and 3b.

By definition, the containment leakage rate for Category 1 (i.e., accidents with containment leakage at or below maximum allowable Technical Specification leakage) is $1.0L_a$.

3.3 BASELINE POPULATION DOSE RATE ESTIMATES (STEPS 3-4)

The third and fourth steps of the NEI Interim Guidance are to estimate the baseline population dose (person-rem) for each EPRI category and to calculate the dose rate (person-rem/year) by multiplying the category frequencies by the estimated dose.

3.3.1 Population Dose Estimates (Step 3)

The NEI Interim Guidance recommends two options for calculating population dose for the EPRI categories:

- Use of generic NUREG-1150 dose calculations
- Use of plant-specific dose calculations

The NUREG-1150 [14] dose calculations were used in the EPRI TR-104285 study, as discussed previously in Section 2.1. The use of generic dose information, such as NUREG-1150, is recommended by NEI to make the ILRT risk assessment methodology more readily usable for plants that do not have a Level 3 PSA. As CGS does not have a Level 3 PSA or associated plant-specific dose calculations, this ILRT risk assessment employs use of NUREG-1150 dose results calculated using the MACCS (MELCOR Accident Consequence Code System) consequence code. The doses of the Peach Bottom study (as documented in NUREG/CR-4551, Vol. 4, Rev. 1, Parts 1 and 2) are used. The following discussion summarizes the population dose calculation and results.

Peach Bottom NUREG/CR-4551 Study Population Dose

The CGS population dose is calculated using data provided in NUREG/CR-4551 for Peach Bottom and adjusting the results for applicability to CGS. Each Peach Bottom accident sequence was assigned to an applicable Accident Progression Bin (APB) in NUREG/CR-4551. The definitions of the ten APBs are reproduced in Table 3-1.

Table 3-3 summarizes the calculated population dose associated with each Peach Bottom APB from NUREG/CR-4551, including the fraction of the population dose within 50 miles contributed by each APB and the frequency of release.

Adjustment of NUREG/CR-4551 Doses to CGS

As discussed in Section 2.4.2, the Peach Bottom NUREG/CR-4551 ex-plant consequence results are used as input to determine the population dose estimates of this CGS ILRT risk assessment consistent with guidance provided by NEI. The NUREG/CR-4551 consequences summarized in Table 3-3 are adjusted for use in this analysis to account for plant specific differences in the following parameters:

- Population
- Reactor Power Level
- Technical Specification Allowed Containment Leakage Rate

Population Adjustment

As discussed in Section 2.4.4.1, the 50-mile radius Peach Bottom population used in the NUREG/CR-4551 consequence calculations is estimated at 3.2E+6 persons, whereas the year 2000 population within the 50-mile radius of CGS is estimated at 3.6E+5 persons. This difference in population results in the adjustment factor to be applied to the NUREG/CR-4551 APB doses of 0.11.

Reactor Power Level Adjustment

As discussed in Section 2.4.4.2, the reactor power level used in the NUREG/CR-4551 Peach Bottom consequence calculations is 3293 MWth, whereas the CGS 5% Power Uprate full power level is 3486 MWth. This difference in reactor power level results in the adjustment factor to be applied to the NUREG/CR-4551 APB doses of 1.06.

Containment Leakage Rate Adjustment

As discussed in Section 2.4.4.3, the containment leakage rate used in the NUREG/CR-4551 consequence calculations for core damage accidents with the containment intact is 0.5% over 24 hours, the same as the CGS maximum allowable containment leakage per Technical Specifications. While use of a leakage rate below the maximum allowable may be reasonable, this analysis assumes that containment leakage is at the maximum allowable Technical Specification value. Additionally, a correction is required to account for differences in containment volumes. The containment volume of Peach Bottom is 2.97E+5 ft^3 while that of CGS is slightly larger, 3.46E+5 ft^3. These differences result in an adjustment factor of 1.16 to be applied to the NUREG/CR-4551 APB doses.

Population Dose by APB for CGS

Table 3-4 provides the translation of the surrogate analysis (Peach Bottom from NUREG/CR-4551) to the CGS plant and site based on APBs. This translation uses the adjustments to power, population, and containment leak rate to the NUREG/CR-4551 population dose results at 50 miles to obtain the adjusted population dose at 50 miles for CGS for each APB.

Population Dose by EPRI Category for CGS

Using the preceding information, the population dose as a function of EPRI category for the 50-mile radius surrounding CGS is summarized in Table 3-6. The following discussion provides the basis for the assignment of population dose for each EPRI category. Due to differences in the EPRI category scheme and the NUREG/CR-4551 APB scheme, conservative assumptions are utilized in the dose assignment process. For example, the highest dose of all the Peach Bottom "containment failure" APBs (i.e., APB #3) is assigned to the three EPRI categories (i.e., #2, 7a, and 8) reflective of a large, unscrubbed release. Note that all population doses are derived from the scaled dose estimates of the surrogate plant (see Table 3-4).

The dose for EPRI category #1 (core damage accident with isolated and intact containment, i.e., no containment failure) is based on NUREG/CR-4551 APB #8. This is the APB closest to the definition of an intact containment.

The dose for EPRI Category 2 for core damage accidents with containment isolation failure is based on NUREG/CR-4551 APB #3. This assignment is based on assuming that the containment isolation failure of EPRI Category 2 occurs in the drywell as an unscrubbed release. APB #3 results in the highest dose of all the Peach Bottom "containment failure" APBs (which is indicative of a containment failure with an unscrubbed release).

No separate assignment of NUREG/CR-4551 APBs is made for EPRI Categories 3a and 3b. Instead, per the NEI Interim Guidance, the doses for EPRI Categories #3a and #3b are taken as factors of 10 and 35, respectively, times the population dose of EPRI Category 1.

As EPRI Categories 4, 5, and 6 are not affected by ILRT frequency and not analyzed as part of this risk assessment (per NEI Interim Guidance), no assignment of NUREG/CR-4551 APBs is made for these categories.

The dose for EPRI Category 7 is based on the development of a weighted average person-rem dose representative of the EPRI Category 7 subcategories 7a, 7b, and 7c. This weighted average approach is acceptable since the total frequency and dose associated with EPRI Category 7 does not change as part of the ILRT extension. Table 3-5 summarizes the dose for subcategories 7a, 7b, and 7c and the representative (i.e., weighted average) Category 7 dose.

- The dose for EPRI Category 7a (CGS large, early, unscrubbed release) is based on NUREG/CR-4551 APB #3. The CGS accident scenarios resulting in LERF represent the most severe release category. Accordingly, the most severe NUREG/CR-4551 dose case is used to characterize this category (i.e., APB #3).
- The dose for EPRI Category 7b (CGS large, late, unscrubbed release) is based on NUREG/CR-4551 APB #4. APB #4 is the highest frequency bin for Peach Bottom Class II accident scenarios involving containment failure (generally late releases) and depicts a drywell (i.e., unscrubbed release). The dose of APB #4 is also significant.
- The dose for EPRI Category 7c (CGS large, late, scrubbed release) is based on NUREG/CR-4551 APB #5. The late containment failure and wetwell location of APB #5 are indicative of a large, late, scrubbed release.

The representative dose for EPRI Category 7 is 2.57E+5 person-rem.

Table 3-3

PEACH BOTTOM NUREG/CR-4551 50-MILE RADIUS POPULATION DOSE CALCULATION[1]

APB #	APB Definition	APB Frequency (per year) [2]	APB Fractional Contribution to 50-Mile Radius Total Dose Rate[3]	APB 50-Mile Radius Dose Rate (person-rem/year) [4]	APB 50-Mile Radius Dose (Person-rem) [5]
1	CD, VB, Early CF, WW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means Direct Containment Heating (DCH) is possible).	9.55E-8	0.021	0.166	1.74E+6
2	CD, VB, Early CF, WW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).	4.77E-8	0.0066	0.0521	1.09E+6
3	CD, VB, Early CF, DW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means DCH is possible).	1.48E-6	0.556	4.39	2.97E+6

Table 3-3

PEACH BOTTOM NUREG/CR-4551 50-MILE RADIUS POPULATION DOSE CALCULATION[1]

APB #	APB Definition	APB Frequency (per year) [2]	APB Fractional Contribution to 50-Mile Radius Total Dose Rate[3]	APB 50-Mile Radius Dose Rate (person-rem/year) [4]	APB 50-Mile Radius Dose (Person-rem) [5]
4	CD, VB, Early CF, DW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).	7.94E-7	0.226	1.79	2.25E+6
5	CD, VB, Late CF, WW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the wetwell (i.e., after vessel breach during Molten Core-Concrete Interaction (MCCI)) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.	1.30E-8	0.0022	0.0174	1.34E+6
6	CD, VB, Late CF, DW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the drywell (i.e., after vessel breach during MCCI) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.	2.04E-7	0.059	0.466	2.28E+6

Table 3-3

PEACH BOTTOM NUREG/CR-4551 50-MILE RADIUS POPULATION DOSE CALCULATION[1]

APB #	APB Definition	APB Frequency (per year) [2]	APB Fractional Contribution to 50-Mile Radius Total Dose Rate[3]	APB 50-Mile Radius Dose Rate (person-rem/year) [4]	APB 50-Mile Radius Dose (Person-rem) [5]
7	CD, VB, No CF, Vent, N/A Core Damage occurs followed by vessel breach. The containment never structurally fails, but is vented sometime during the accident progression. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH does not significantly affect the source term as the containment does not fail and the vent limits its effect.	4.77E-7	0.118	0.932	1.95E+6
8	CD, VB, No CF, N/A, N/A Core damage occurs followed by vessel breach. The containment never fails structurally (characteristic 4 is N/A) and is not vented. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH did not fail containment. Some nominal leakage from the containment exists and is accounted for in the analysis so that while the risk will be small it is not completely negligible.	7.99E-7	0.0005	3.95E-3	4.94E+3
9	CD, No VB, N/A, N/A, N/A Core damage occurs but is arrested in time to prevent vessel breach. There are no releases associated with vessel breach or MCCI. It must be remembered, however, that the containment can fail due to overpressure or venting even if vessel breach is averted. Thus, the potential exists for some of the in-vessel releases to be released to the environment.	3.86E-7	0.01	0.079	2.05E+5

Table 3-3

PEACH BOTTOM NUREG/CR-4551 50-MILE RADIUS POPULATION DOSE CALCULATION(1)

APB #	APB Definition	APB Frequency (per year) (2)	APB Fractional Contribution to 50-Mile Radius Total Dose Rate(3)	APB 50-Mile Radius Dose Rate (person-rem/year) (4)	APB 50-Mile Radius Dose (Person-rem) (5)
10	No CD, N/A, N/A, N/A, N/A Core damage did not occur. No in-vessel or ex-vessel release occurs. The containment may fail on overpressure or be vented. The RPV may be at high or low pressure depending on the progression characteristics. The risk associated with this bin is negligible.	4.34E-8	0	0	0
	Total:	4.34E-6	1.0	7.9	--

(1) This table is presented in the form of a calculation because NUREG/CR-4551 does not document dose results as a function of accident progression bin (APB); as such, the dose results as a function of APB must be back calculated from documented APB frequencies and APB dose results.

(2) The total (i.e., internal accident sequences) CDF of 4.34E-6/yr and the CDF subtotals by APB are taken from Figure 2.5-6 of NUREG/CR-4551 Vol. 4, Rev. 1, Part I.

(3) The individual APB contributions to the total (i.e., internal accident sequences) 50-mile radius dose rate are taken from Table 5.2-3 of NUREG/CR-4551 Vol. 4, Rev. 1, Part I.

(4) The APB 50-mile dose rate is calculated by multiplying the individual APB dose rate fractional contributions (column 4) by the total 50-mile radius dose rate of 7.9 person-rem/yr (taken from Table 5.1-1 of NUREG/CR-4551 Vol. 4, Rev. 1, Part I).

(5) The individual APB doses are calculated by dividing the individual APB dose rate (Column 5) by the APB frequencies (Column 3).

Table 3-4

CGS POPULATION DOSE BY APB:
ADJUSTED PEACH BOTTOM NUREG/CR-4551
50-MILE RADIUS POPULATION DOSES

APB #	Peach Bottom 50-Mile Radius Dose (Person-rem) [1]	Population Adjustment Factor	Reactor Power Adjustment Factor	Containment Leak Rate Adjustment Factor	CGS Population Dose Adjusted 50-Mile Radius Dose (Person-rem)
1	1.74E+6	0.11	1.06	n/a	2.03E+5
2	1.09E+6	0.11	1.06	n/a	1.27E+5
3	2.97E+6	0.11	1.06	n/a	3.46E+5
4	2.25E+6	0.11	1.06	n/a	2.62E+5
5	1.34E+6	0.11	1.06	n/a	1.56E+5
6	2.28E+6	0.11	1.06	n/a	2.66E+5
7	1.95E+6	0.11	1.06	n/a	2.27E+5
8	4.94E+3	0.11	1.06	1.16	6.68E+2
9	2.05E+5	0.11	1.06	n/a	2.39E+4
10	0	0.11	1.06	n/a	0

[1] Source: Last column of Table 3-3.

The dose for the containment bypass category, EPRI Category 8, is based on NUREG/CR-4551 APB #3. APB #3 results in the highest dose of all the NUREG/CR-4551 "containment failure" APBs, indicative of containment bypass scenarios.

3.3.2 Baseline Population Dose Rate Estimates (Step 4)

The baseline dose rates per EPRI accident category are calculated by multiplying the population dose estimates from Table 3-6 by the frequencies summarized in Table 3-2. The resulting baseline population dose rates by EPRI category are summarized in Table 3-7. As the conditional containment pre-existing leakage probabilities for EPRI Categories 3a and 3b are reflective of a 3-per-10 year ILRT frequency (refer to Section 3.1), the baseline results shown in Table 3-7 are indicative of a 3-per-10 year ILRT surveillance frequency.

3.4 IMPACT OF PROPOSED ILRT INTERVAL (STEPS 5-9)

Steps 5 through 9 of the NEI Interim Guidance assess the impact on plant risk due to the new ILRT surveillance interval in the following ways:

- Determine change in probability of detectable leakage (Step 5)
- Determine population dose rate for new ILRT interval (Step 6)
- Determine change in dose rate due to new ILRT interval (Step 7)
- Determine change in LERF risk measure due to new ILRT interval (Step 8)
- Determine change in CCFP due to new ILRT interval (Step 9)

Table 3-5

CGS EPRI CATEGORY 7 POPULATION DOSE RATE

EPRI Category	Assigned Peach Bottom APB	CGS Release Frequency per year[1]	CGS Population Dose (50 miles) Person-Rem [2]	Population Dose Rate (50 mile) Person-Rem/yr[3]
7a	#3	5.29E-7	3.46E+5	1.83E-1
7b	#4	3.80E-6	2.62E+5	9.96E-1
7c	#5	6.40E-7	1.56E+5	9.98E-2
Category 7 Total	--	4.97E-6	2.57E+5[4]	1.28

Notes:

(1) Section 3.1

(2) Table 3-4, Column 6

(3) Obtained by multiplying the release frequency (Column 3) by the population dose (Column 4)

(4) Weighted average population dose for Category 7 obtained by dividing the total population dose rate (1.28 Person-Rem/yr) by the total release frequency (4.97E-6/yr).

Table 3-6

CGS POPULATION DOSE ESTIMATES AS A FUNCTION OF EPRI CATEGORY WITHIN 50-MILE RADIUS

EPRI Category	Category Description	CGS Person-Rem Within 50 miles
1	No Containment Failure(1)	6.68E+2
2	Containment Isolation System Failure(2)	3.46E+5
3a	Small Pre-Existing Failures(3)	6.68E+3
3b	Large Pre-Existing Failure(4)	2.34E+4
4	Type B Failures (LLRT)(5)	n/a
5	Type C Failures (LLRT)(5)	n/a
6	Other Containment Isolation System Failure(5)	n/a
7	Containment Failure Due to Severe Accident(6)	2.57E+5
8	Containment Bypass Accidents(2)	3.46E+5

Notes:

(1) Based on APB #8 of Table 3-4
(2) Based on APB #3 of Table 3-4
(3) Factor of 10 times EPRI Category 1
(4) Factor of 35 times EPRI Category 1
(5) Not analyzed since not affected by ILRT frequency
(6) Weighted average of subcategories 7a-7c of Table 3-5

Table 3-7

CGS DOSE RATE ESTIMATES AS A FUNCTION OF EPRI CATEGORY FOR POPULATION WITHIN 50 MILES
(Baseline 3-per-10 year ILRT)

EPRI Category	Category Description	Person-Rem Within 50 miles[6]	Baseline Frequency (per year)[7]	Dose Rate (Person-Rem/yr)
1	No Containment Failure[1]	6.68E+2	2.07E-6	1.38E-3
2	Containment Isolation System Failure[2]	3.46E+5	4.43E-9	1.53E-3
3a	Small Pre-Existing Failures[3]	6.68E+3	1.19E-7	7.95E-4
3b	Large Pre-Existing Failures[3]	2.34E+4	1.19E-8	2.78E-4
4	Type B Failures (LLRT)	n/a	n/a	n/a
5	Type C Failures (LLRT)	n/a	n/a	n/a
6	Other Containment Isolation System Failure	n/a	n/a	n/a
7	Containment Failure Due to Severe Accident[4]	2.57E+5	4.97E-6	1.28
8	Containment Bypass Accidents[5]	3.46E+5	1.57E-7	5.43E-2
	Total		7.33E-6	1.34

Notes to Table 3-7

(1) The population dose associated with the Technical Specification Leakage is based on scaling the population data, the power level, and allowable Technical Specification leakage compared to the NUREG/CR-4551 reference plant. The release for this EPRI category is assigned from APB #8 from Table 3-4.

(2) EPRI Category 2 (Containment Isolation failures) may include drywell isolation failures. Therefore, the release associated with this category is assigned to be equivalent to the release associated with the highest dose (APB #3) from Table 3-4.

(3) Dose estimates for categories 3a and 3b, per the NEI Interim Guidance, are calculated as 10xCategory 1 dose and 35xCategory 1 dose, respectively.

(4) Dose estimate for category 7 is the weighted average of subcategories 7a-7c of Table 3-5.

(5) EPRI Category 8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this category are assumed to result in a direct path to the environment, and as such, are assigned to be equivalent to the highest release category from NUREG/CR-4551. APB #3 from Table 3-4 is therefore used.

(6) Table 3-6.

(7) Table 3-2.

3.4.1 Change in Probability of Detectable Leakage (Step 5)

Step 5 of the NEI Interim Guidance is the calculation of the change in probability of leakage detectable only by ILRT (and associated re-calculation of the frequencies of the impacted EPRI categories). Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rates are assumed not to change; however, the probability of pre-existing leakage detectable only by ILRT does increase.

Per the NEI Interim Guidance, the calculation of the change in the probability of a pre-existing ILRT-detectable containment leakage is based on the relationship that relaxation of the ILRT interval results in increasing the average time that a pre-existing leak would exist undetected. Using the standby failure rate statistical model, the average time that a pre-existing containment leak would exist undetected is one-half the surveillance interval. For example, if the ILRT frequency is 1-per-10 years, then the average time that a leak would be undetected is 60 months (surveillance interval of 120 months divided by 2). The impact on the leakage probability due to the ILRT interval extension is then calculated by applying a multiplier determined by the ratio of the average times of undetection for the two ILRT interval cases.

As discussed earlier in Section 3.1, the conditional probability of a pre-existing ILRT-detectable containment leakage is divided into two categories. The calculated pre-existing ILRT-detectable leakage probabilities are reflective of a 3-per-10 year ILRT frequency and are as follows:

- "Small" pre-existing leakage (EPRI Category 3a): 2.70E-2
- "Large" pre-existing leakage (EPRI Category 3b): 2.70E-3

Since 1996, the CGS plant has been operating under a 1-per-10 year ILRT testing frequency consistent with the performance-based Option B of 10 CFR Part 50,

Appendix J. [16] The baseline[1] leakage probabilities first need to be adjusted to reflect the current 1-per-10 year CGS ILRT testing frequency, as follows:

- "Small" : 2.70E-2 x [(120 months/2) / (36 months/2)] = 9.00E-2
- "Large" : 2.70E-3 x [(120 months/2) / (36 months/2)] = 9.00E-3

Note that a nominal 36 month interval (i.e., as opposed to 40 months, 120/3) is used in the above adjustment calculation to reflect the 3-per-10 year ILRT frequency. This is consistent with operational practicalities and the NEI Interim Guidance.

Similarly, the pre-existing ILRT-detectable leakage probabilities for the 1-per-15 year ILRT frequency currently being pursued by CGS (and the subject of this risk assessment) are calculated as follows:

- "Small" : 9.00E-2 x [(180 months/2) / (120 months/2)] = 1.35E-1
- "Large" : 9.00E-3 x [(180 months/2) / (120 months/2)] = 1.35E-2

Given the above adjusted leakage probabilities, the impacted frequencies of the EPRI categories are summarized below (refer to Table 3-2 for details regarding frequency calculations for the individual EPRI categories):

EPRI Category	EPRI Category Frequency as a Function of ILRT Interval		
	Baseline (3-per-10 year ILRT)	Current (1-per-10 year ILRT)	Proposed (1-per-15 year ILRT)
1	2.07E-6	1.76E-6	1.55E-6
3a	1.19E-7	3.97E-7	5.95E-7
3b	1.19E-8	3.97E-8	5.95E-8

Note that, per the definition of the EPRI categories, only the frequencies of Categories 1, 3a, and 3b are impacted by changes in ILRT testing frequencies.

[1] The baseline case uses data characteristic of the 3-per-10 year ILRT frequency of testing.

3.4.2 Population Dose Rate for New ILRT Interval (Step 6)

The dose rates per EPRI accident category as a function of ILRT interval are summarized in Table 3-8.

3.4.3 Change in Population Dose Rate Due to New ILRT Interval (Step 7)

As can be seen from the dose rate results summarized in Table 3-8, the calculated total dose rate increase from the current CGS 10-year ILRT interval amount of 1.34 person-rem/year is negligible. The total dose rate remains unchanged due to the dominance of the dose contributed by severe accident induced containment failure.

Per the NEI Interim Guidance, the change in percentage contribution to total dose rate attributable to EPRI Categories 3a and 3b is also investigated here. Using the results summarized in Table 3-8, for the current CGS 1-per-10 year ILRT interval, the percentage contribution to total dose rate from Categories 3a and 3b is shown to be very minor:

$$[(2.65E\text{-}3 + 9.27E\text{-}4) / 1.34] \times 100 = 0.27\%$$

For the proposed 1-per-15 year ILRT interval, the percentage contribution to total dose rate from Categories 3a and 3b increases slightly but remains very minor:

$$[(3.97E\text{-}3 + 1.39E\text{-}3) / 1.34] \times 100 = 0.40\%$$

The change in percentage contribution is therefore an insignificant 0.13% increase.

Table 3-8

DOSE RATE ESTIMATES BY EPRI CATEGORY FOR POPULATION WITHIN 50 MILES

EPRI Category	Category Description	Dose Rate as a Function of ILRT Interval (Person-Rem/Yr)		
		Baseline (3-per-10 year ILRT)	Current (1-per-10 year ILRT)	Proposed (1-per-15 year ILRT)
1	No Containment Failure	1.38E-3	1.18E-3	1.03E-3
2	Containment Isolation System Failure	1.53E-3	1.53E-3	1.53E-3
3a	Small Pre-Existing Failures	7.95E-4	2.65E-3	3.97E-3
3b	Large Pre-Existing Failures	2.78E-4	9.27E-4	1.39E-3
4	Type B Failures (LLRT)	N/A	N/A	N/A
5	Type C Failures (LLRT)	N/A	N/A	N/A
6	Other Containment Isolation System Failure	N/A	N/A	N/A
7	Containment Failure Due to Severe Accident	1.28	1.28	1.28
8	Containment Bypass Accidents	5.43E-2	5.43E-2	5.43E-2
	TOTAL	1.34	1.34	1.34[1]

[1] The change in dose rate from the 3-per-10 year ILRT frequency to the 1-per-15 year ILRT frequency is approximately 3E-3 person-rem/yr. This change is less than the number of significant figures being reasonably carried in the assessment.

3.4.4 Change in LERF Due to New ILRT Interval (Step 8)

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would not result in a radionuclide release from an intact containment could in fact result in a release due to the increase in probability of failure to detect a pre-existing leak. Per the NEI Interim Guidance, only Category 3b sequences have the potential to result in large releases if a pre-existing leak were present. As such, the change in LERF (Large Early Release Frequency) is determined by the change in the frequency of Category 3b.

Category 1 accidents are not considered as potential large release pathways because the containment remains intact. Therefore, the containment leak rate is expected to be small. Similarly, Category 3a is a "small" pre-existing leak. Other accident categories such as 2, 6, 7, and 8 could result in large releases but these are not affected by the change in ILRT interval. Late releases are excluded regardless of the size of the leak because late releases are, by definition, not LERF contributors.

The impact on the LERF risk measure due to the proposed ILRT interval extension is calculated as follows:

delta LERF = (Frequency of EPRI Category 3b for 1-per-15 year ILRT interval) – (Frequency of EPRI Category 3b for 1-per-10 year ILRT interval)

= 5.95E-8/yr – 3.97E-8/yr = 1.98E-8/yr

≈ 2.0E-8/yr

This delta LERF of 2.0E-8/yr falls into Region III, Very Small Change in Risk, of the acceptance guidelines in NRC Regulatory Guide 1.174. Therefore, increasing the ILRT interval at CGS from the currently allowed 10 years to 15 years represents a very small change in risk, and is an acceptable plant change from a risk perspective.

3.4.5 Impact on Conditional Containment Failure Probability (Step 9)

Another parameter that the NRC Guidance in Reg. Guide 1.174 states can provide input into the decision-making process is the consideration of change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT on all radionuclide releases, not just LERF. The conditional containment failure probability (CCFP) can be calculated from the risk calculations performed in this analysis. In this assessment, based on the NEI Interim Guidance, CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state (EPRI Category 1) and small failures (EPRI Category 3a). The conditional part of the definition is conditional given a severe accident (i.e., core damage).

Consequently, the change in CCFP can be calculated by the following equation:

$$CCFP_{\%} = [1 - (\text{Intact Containment Frequency} / \text{Total CDF})] \times 100\%, \text{ or}$$
$$= [1 - ((\#1 \text{ Frequency} + \#3a \text{ Frequency}) / \text{CDF})] \times 100\%$$

For the 10-year interval:

$$CCFP_{10} = [1 - ((1.76E\text{-}6 + 3.97E\text{-}7) / 7.33E\text{-}6)] \times 100\% = 70.6\%$$

For a 15-year interval:

$$CCFP_{15} = [1 - ((1.55E\text{-}6 + 5.95E\text{-}7) / 7.33E\text{-}6)] \times 100\% = 70.7\%$$

Therefore, the change in the conditional containment failure probability is:

$$\Delta\, CCFP = CCFP_{15} - CCFP_{10} = 0.1\%$$

This change in CCFP of 0.1% is insignificant from a risk perspective.

3.5 SENSITIVITIES

It is observed that the NRC, via Requests for Additional Information (RAIs), has previously requested additional quantitative assessments regarding age related degradation of non-inspectable areas of the containment and assumptions regarding long term station blackout sequences. This section summarizes the results of sensitivities performed to address these issues.

A sensitivity assessment of external event impacts is discussed in Section 5 of this report.

3.5.1 Containment Degradation Sensitivity

Inspections of some reinforced and steel containments (e.g., North Anna, Brunswick, D.C. Cook, and Oyster Creek) have indicated degradation from the uninspectable (embedded) side of the steel shell and liner of primary containment. In response to previous ILRT extension request submittals, the NRC has consistently requested licensees to perform a quantitative assessment of the impact on LERF due to age-related degradation of non-inspectable areas of the containment. Therefore, a quantitative assessment using the same approach used by other industry plants (e.g., Calvert Cliffs) is performed as a sensitivity case to this ILRT extension evaluation. Appendix D provides the analysis details.

The results of the sensitivity case indicate that the increase in LERF from the 10-year ILRT interval to the 15-year ILRT interval is 2.28E-8/year, compared with 1.98E-8/yr without corrosion effects. This is still well below the Regulatory Guide 1.174 acceptance criterion threshold for "very small" changes in risk of 1.0E-7/yr. This confirms that the proposed interval extension is acceptable from a risk-informed perspective. Additionally, the dose rate increase is negligible compared with the total of 1.34 person-rem/yr. The increase in the CCFP is determined to be insignificant (70.9% for the 15-year interval case versus 70.6% for the 10-year interval case). The results demonstrate

that including corrosion effects in the ILRT assessment do not alter the conclusions from the original analysis.

3.5.2 Long Term Station Blackout Sensitivity

The NRC has previously asked licensees, via RAIs, to provide technical justification for the assumption that long term station blackout (LT SBO) scenarios do not contribute to LERF, and to provide an assessment of the impact on risk results if long term station blackout sequences are retained in selected EPRI categories.

Appendix C provides the technical justification for not including LT SBO scenarios in the LERF assessment and also provides a sensitivity case to demonstrate that placing LT SBO sequences in the LERF category does not change the conclusions of the overall ILRT assessment.

The frequency of long term LT SBO core damage sequences in the CGS PRA is 1.28E-6/yr. [22] The subject sensitivity case in Appendix C repeats the calculations of the ILRT assessment performed in Section 3, with the exception that the long term SBO sequences are retained in the EPRI Category 3a and 3b frequency calculations.

Retaining the LT SBO sequences in the EPRI Categories 3a and 3b frequency calculations results in a LERF increase of 2.56E-8/yr for the change from the current 10-year ILRT interval to the 15-year interval. This represents an additional LERF increase of 5.8E-9/yr (a 29% increase) over the best estimate ILRT increase in LERF of 1.98E-8/yr based on the full power internal events PRA. Including the long term SBO contribution, however, still results in a LERF increase below the NRC Regulatory Guide 1.174 Region III criterion of 1.0E-7/yr indicating a "very small" risk change. The increase in the population dose rate remains negligible, the same as in the baseline analysis. The increase in the conditional containment failure probability (CCFP) is determined to increase slightly from 0.1% to 0.3% (with LT SBO sequences included).

The sensitivity case demonstrates that even if long term SBO scenarios are included in the EPRI Category 3a and 3b frequencies, the conclusion of the risk assessment does not change; that is, the CGS ILRT interval extension to 15 years has a minimal impact on plant risk.

3.5.3 Conclusions

The sensitivity cases evaluated demonstrate:

- LERF is not significantly impacted by the potential for containment leakage due to age-related degradation in non-inspectable areas.
- Inclusion of long-term station blackout scenarios in the EPRI Category 3a and 3b frequencies does not change the conclusion of this report.

SECTION 4
RESULTS SUMMARY

The application of the approach based on NEI Interim Guidance [3, 21], EPRI-TR-104285 [2] and previous risk assessment submittals on this subject [6] have led to the quantitative results summarized in this section. The CGS full power internal events Probabilistic Risk Assessment (PRA) is used in the quantification. These results demonstrate a very small impact on risk associated with the one time extension of the ILRT test interval to 15 years.

The analysis performed examined CGS specific accident sequences in which the containment remains intact or the containment is impaired. The accidents are analyzed and the results are displayed according to the eight (8) EPRI accident categories defined in Reference [2]:

1. Containment intact and isolated
2. Containment isolation failures due to support system or active failures
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failure due to core damage accident phenomena
8. Containment bypass

The quantitative results are summarized in Table 4-1. The key results to this risk assessment are those for the ten year interval (current CGS condition) and the fifteen year interval (proposed change). The 3-per-10 year ILRT is a baseline starting point for this risk assessment given that the pre-existing containment leakage probabilities (estimated based on industry experience -- refer to Section 3.1) are reflective of the 3-per-10 year ILRT testing.

The following is a brief summary of some of the key aspects of the ILRT test interval extension risk analysis:

- Increasing the current 10 year ILRT interval to 15 years results in a negligible increase in total population dose rate.
- The increase in the LERF risk measure is also insignificant, a 2.0E-8/yr[1] increase. This LERF increase is categorized as a "very small" increase per NRC Reg. Guide 1.174.
- Likewise, the conditional containment failure probability ($CCFP_{\%}$) increases insignificantly by 0.1%. This is judged to be a negligible change in the CCFP.

[1] Rounded up from the calculated value of 1.98E-8/yr.

Table 4-1

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL

EPRI Category	Dose (Person-Rem Within 50 miles)	Quantitative Results as a Function of ILRT Interval					
		Baseline (3-per-10 year ILRT)		Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
		Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	6.68E+2	2.07E-6	1.38E-3	1.76E-6	1.18E-3	1.55E-6	1.03E-3
2	3.46E+5	4.43E-9	1.53E-3	4.43E-9	1.53E-3	4.43E-9	1.53E-3
3a	6.68E+3	1.19E-7	7.95E-4	3.97E-7	2.65E-3	5.95E-7	3.97E-3
3b	2.34E+4	1.19E-8	2.78E-4	3.97E-8	9.27E-4	5.95E-8	1.39E-3
4	N/A	N/A	N/A	N/A	N/A	N/A	N/A
5	N/A	N/A	N/A	N/A	N/A	N/A	N/A
6	N/A	N/A	N/A	N/A	N/A	N/A	N/A
7	2.57E+5	4.97E-6	1.28	4.97E-6	1.28	4.97E-6	1.28
8	3.46E+5	1.57E-7	5.43E-2	1.57E-7	5.43E-2	1.57E-7	5.43E-2
TOTALS:		7.33E-6	1.34	7.33E-6	1.34	7.33E-6	1.34
Increase in Dose Rate (1)					neg.		neg. (4)
Increase in LERF (2)				2.78E-8		1.98E-8	
Increase in CCFP (%) (3)				0.4%		0.1%	

Notes to Table 4-1:

(1) The increase in dose rate (person-rem/year) is with respect to the results for the preceding ILRT interval, as presented in the table. For example, the increase in dose rate for the proposed 1-per-15 ILRT is calculated as: total dose rate for 1-per-15 year ILRT, minus total dose rate for 1-per-10 year ILRT. For each case, the dose rate increase is insignificant.

(2) The increase in Large Early Release Frequency (LERF) is with respect to the results for the preceding ILRT interval, as presented in the table. As discussed in Section 3.4.4 of the report, the change in LERF is determined by the change in the accident frequency of EPRI Category 3b. For example, the increase in LERF for the proposed 1-per-15 ILRT is calculated as: 3b frequency for 1-per-15 year ILRT, 5.95E-8/yr, minus 3b frequency for 1-per-10 year ILRT, 3.97E-8/yr, equals1.98E-8/yr.

(3) As discussed in Section 3.4.5, the conditional containment failure probability (CCFP) is calculated as:

$$CCFP_{\%} = [1 - ((\text{Category \#1 Frequency} + \text{Category \#3a Frequency}) / CDF)] \times 100\%$$

(4) The change in dose rate from the 1-per-10 year ILRT frequency to the 1-per-15 year ILRT frequency is approximately 2E-3 person-rem/yr. This change is less than the number of significant figures being reasonably carried in the assessment.

SECTION 5
CONCLUSIONS

5.1 QUANTITATIVE CONCLUSIONS

The conclusions from the risk assessment of the one time ILRT extension can be characterized by the risk metrics used in previously approved ILRT test interval extensions. These include:

- Change in LERF
- Change in conditional containment failure probability
- Change in population dose rate

5.1.1 LERF

Based on the results from Sections 3 and 4, the main conclusion regarding the impact on plant risk associated with extending the Type A ILRT test frequency from ten years to fifteen years is:

> Reg. Guide 1.174 [4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from once-per-ten years to once-per-fifteen years (using the change in the EPRI Category 3b frequency per the NEI Interim Guidance) is 2.0E-8/yr. Guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr. Therefore, increasing the CGS ILRT interval from 10 to 15 years results in a very small change in risk, and is an acceptable plant change from a risk perspective.

5.1.2 CCFP

The change in conditional containment failure probability (CCFP) is also calculated as an additional risk measure to demonstrate the impact on defense-in-depth. The ΔCCFP is found to be very small (0.1% increase) and represents a negligible change in the CGS defense-in-depth.

5.1.3 Population Dose Rate

The change in population dose rate is also reported consistent with previously approved ILRT interval extension requests. The change in population dose rate from the current 1/10 year ILRT frequency to 1/15 year frequency is negligible.

5.2 RISK TRADE-OFF

The performance of an ILRT introduces risk.. An EPRI study of operating experience events associated with the performance of ILRTs has indicated that there are real risk impacts associated with the setup and performance of the ILRT during shutdown operation [8]. While these risks have not been quantified for CGS, it is judged that there is a positive (yet unquantified) safety benefit associated with the avoidance of frequent ILRTs.

The safety benefits relate to the avoidance of plant conditions and alignments associated with the ILRT which place the plant in a less safe condition leading to events related to drain down or loss of shutdown cooling. Therefore, while the focus of this evaluation has been on the negative aspects, or increased risk, associated with the ILRT extension, there are, in fact, positive safety benefits associated with reducing the risk contribution from shutdown risk configurations.

5.3 EXTERNAL EVENTS IMPACT

External hazards were evaluated in the CGS Individual Plant Examination of External Events (IPEEE) Submittal in response to the NRC IPEEE Program (Generic Letter 88-20 Supplement 4). The IPEEE Program was a one-time review of external hazard risk and was limited in its purpose to the identification of potential plant vulnerabilities and the understanding of severe accident risks.

Seismic events were addressed through a Seismic Probabilistic Safety Assessment (SPSA) as part of the IPEEE. The seismic external event study provides adequate (but conservative) information to assess the impact of seismic hazards on the conclusions of the CGS ILRT interval extension risk assessment.

Internal fire events were addressed through a Fire Probabilistic Safety Assessment (FPSA). Its conclusions are considered a reasonable reflection of the current state of the technology and adequate for assessing the impact of fires on the conclusions of the ILRT interval extension risk assessment.

The proposed ILRT interval extension impacts plant risk in a limited way. Specifically, the probability of a pre-existing containment leak being the initial containment failure mode given a core damage accident is potentially higher when the ILRT interval is extended. This impact is manifested in the plant risk profile in a similar manner for both internal events and external events.

The quantitative consideration of external hazards is discussed in more detail in Appendix B of this report. As can be seen from Appendix B, if the external hazard risk results are included in this assessment (i.e., in addition to internal events), the change in LERF associated with the increase in ILRT interval from 10 years to 15 years will be 3.3E-8/yr. This delta LERF falls below the Region III boundary of <1E-7/yr and, therefore, is within the NRC RG 1.174 Region III ("Very Small Changes" in risk).

Therefore, incorporating external event accident sequence results into this analysis does not change the conclusion of this risk assessment (i.e., increasing the CGS ILRT interval from 10 to 15 years is an acceptable plant change from a risk perspective).

5.4 PREVIOUS ASSESSMENTS

The NRC in NUREG-1493 [5] has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.
- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk.

The findings for CGS confirm the above general findings on a plant specific basis when considering the following: (1) CGS severe accident risk profile, (2) the CGS containment failure modes, and (3) the local population surrounding the CGS site.

SECTION 6

REFERENCES

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[15] U.S. Nuclear Regulatory Commission, Reactor Safety Study, WASH-1400, October 1975.

[16] NRC letter to Washington Public Power Supply System (Columbia Generating Station) Issuing Technical Specification Amendment to implement the requirements of 10 CFR 50, Appendix J, Option B for performance-based primary reactor containment leakage testing; May 8, 1996.

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[18] "CGS PSA Quantification Notebook", CGS PSA-2-QU-0001, Rev. 2, May, 2004.

[19] Not used.

[20] Not used.

[21] Letter from A. Petrangelo (NEI) to NEI Administrative Points of Contact, *"One-Time Extension of Containment Integrated Leak Rate Test Interval – Additional Information"*, November 30, 2001.

[22] "CGS PSA Containment Performance Analysis Notebook", CGS PSA-2-L2-0001, Rev. 2, May, 2004.

APPENDIX A

POPULATION ESTIMATES

APPENDIX A
POPULATION ESTIMATES

This appendix includes the population estimates for the following:

- Appendix A.1: Peach Bottom 50-Mile Radius Population Data Used to Characterize Population Dose in NUREG/CR-4551
- Appendix A.2: Columbia Generating Station 50-Mile Radius Population

A.1 PEACH BOTTOM POPULATION DATA USED IN NUREG/CR-4551

Background

NEI Interim Guidance for the ILRT internal extension licensing request includes the option to use NRC Ex-Plant consequences from a surrogate plant (e.g., NUREG-1150) if a plant does not have a plant specific Level 3 PSA. This approach is used for the Columbia Generating Station (CGS) ILRT analysis using the ex-plant consequences of Peach Bottom Unit 2 as documented in NUREG/CR-4551 [A-1].

While the Peach Bottom NUREG/CR-4551 study reports population dose rate results for the 50-mile radius around the Peach Bottom site, the NUREG/CR-4551 documentation does not report the population total of the 50-mile radius used in the analysis. The purpose of this appendix is to estimate the 50-mile radius population total that was used in the NUREG/CR-4551 study, so that it may be used in the CGS ILRT risk assessment for scaling and estimating population dose rates.

Analysis

Table A-1 summarizes the population data around the Peach Bottom site as reported in the NUREG/CR-4551 study. As can be seen from Table A-1, NUREG/CR-4551 does not specify the 50-mile radius population for Peach Bottom. This section derives the population within 50 miles of Peach Bottom used to support the NUREG/CR-4551 risk estimates based upon population estimates for other radial distances.

Table A-1

PEACH BOTTOM POPULATION DATA REPORTED IN NUREG/CR-4551 [A-1]

Radius From Site		Population (persons) (1)
Miles	Kilometers	
1	1.6	118
3	4.8	1,822
10	16.1	28,647
30	48.3	989,356
100	160.9	14,849,112
350	563.3	68,008,584
1000	1609.3	154,828,144

(1) The NUREG/CR-4551 population estimates summarized from the MACCS demographic input based on 1980 census information (Vol. 4, Rev. 1, Part 1, Table 4.2-2).

Two methods are used to estimate the population within 50 miles of Peach Bottom:

- Method 1: Assume direct proportion of the population with area
- Method 2: Interpolate between estimates for 30 miles and 100 miles as a function of area.

Method 1

This method assumes a constant population density around the Peach Bottom site, thus calculating the population of one area as a direct proportion of another. This population estimation method is performed for both the Peach Bottom 30-mile radius data point and the 100-mile radius data point.

Using the population density of the 30-mile radius data point produces the following 50-mile radius population estimate:

$$\frac{\pi R_{30}^2}{9.89E+5} = \frac{\pi R_{50}^2}{Pop_{50}}$$

$$Pop_{50} = 9.89E+5 * (50^2/30^2) = 2.75E+6 \text{ persons}$$

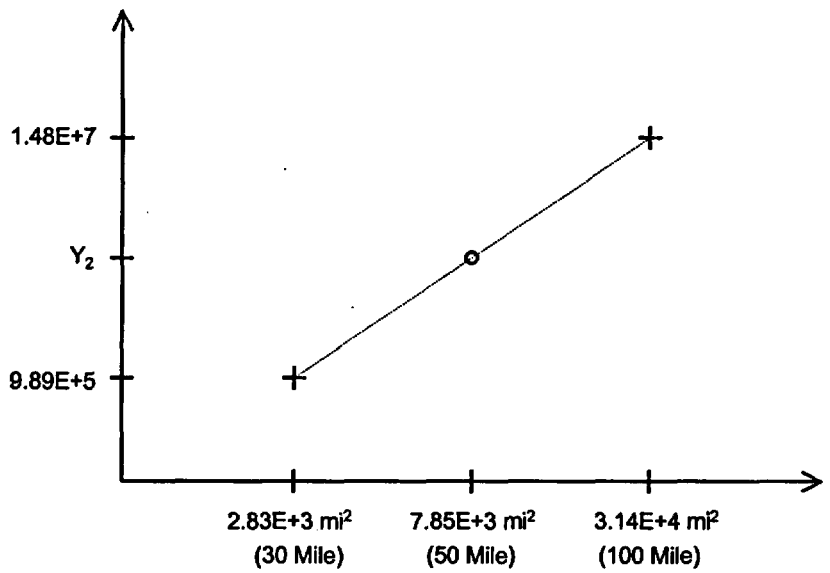
Using the population density of the 100-mile radius data point produces the following 50-mile radius population estimate:

$$Pop_{50} = 1.485E+7 * (50^2/100^2) = 3.71E+6 \text{ persons}$$

The relative closeness of these two estimates indicates that the assumptions of a constant population density around the Peach Bottom site is reasonable. The averaged 50-mile radius NUREG-1150 Peach Bottom estimate using the documented 30-mile and 100-mile points is 3.23E+6 persons.

Method 2

This population estimation method is an interpolation assuming a linearly increasing population with distance as a function of area (as shown in the graph below).



$Y = mx + b$ $\qquad$ $Y_2 - Y_1 = m(X_2 - X_1)$

$m = \dfrac{Y_3 - Y_1}{X_3 - X_1}$ $\qquad$ $Y_3 = 1.48E+7$ $\qquad$ $Y_1 = 9.89E+5$

$X_3 = 3.14E+4$ $\qquad$ $X_1 = 2.83E+3$

$$Y_2 = 9.89E+5 + \frac{(1.48E+7 - 9.89E+5)}{(3.14E+4 - 2.83E+3)} * (7.85E+3 - 2.83E+3)$$

$Y_2 = 3.42E+6$ persons

Therefore, this method estimates a 50-mile radius population of 3.42E+6 persons.

Summary of Peach Bottom NUREG-1150 50-mile Radius Population

The two population estimation methods yield estimates that are very close. The smaller estimate, 3.2E+6, is chosen for use in this risk assessment since this will lead to a more conservative estimate of the risk at CGS when the person-rem are scaled to the CGS site.

A.2 COLUMBIA 50-MILE RADIUS POPULATION

Year 2000 population information for the Columbia Generating Station (CGS) site has been documented in NUREG/CR-6525, Rev. 1 [A-2] as part of the verification of the SECPOP2000 population estimate program.

Table A-2 provides the CGS 50-mile radius population estimated by the SECPOP2000 computer code documented in NUREG/CR-6525. The SECPOP2000 code utilizes block-level data from the 2000 census to calculate population counts for user defined sector segments. This code estimates a total CGS 50-mile radius population of approximately 3.6E+5 people.

Table A-3 is the population data documented in Table 2.1-2 of the CGS UFSAR [A-3]. This Licensee reported population is also based on year 2000 census data. The total CGS 50-mile radius population estimate documented in the UFSAR is approximately 3.6E+5 people.

Based upon the agreement of these two sources, a population total of 3.6E+5 is chosen for use in this ILRT risk assessment.

It is noted that no additional population adjustment (i.e., increase) is judged warranted to account for projected growth in the 50-mile radius of CGS for the following reasons:

- Only modest population growth (i.e., 1-3%) is expected for the 50-mile radius area over the ILRT one-time extension period of 5 years.

- The population estimate primarily impacts the baseline dose estimate associated with severe accident containment failure scenarios (EPRI Category 7) Intact Containment Tech Spec leakage. The population dose increase associated with the proposed ILRT interval extension (as determined by comparison to the baseline) would be negligibly impacted by small population changes.

Table A-2[1]

Columbia 7 Rings to 50 Miles (WNP2)

SPATIAL DISTANCES

SECPOP2000 POPULATION

Ring:	3	5	10	20	30	40	50	Total	
N	0	0	103	126	848	1,044	30,189	32,310	N
NNE	0	0	44	805	10,941	3,702	679	16,171	NNE
NE	0	11	254	1,456	476	236	734	3,167	NE
ENE	4	38	360	916	3,133	370	186	5,007	ENE
E	0	0	262	541	72	134	103	1,112	E
ESE	0	0	517	511	248	1,001	206	2,483	ESE
SE	0	0	383	10,017	12,282	410	1,291	24,383	SE
SSE	0	0	174	63,802	42,898	265	222	107,361	SSE
S	0	0	203	33,479	881	6,495	25,288	66,346	S
SSW	0	0	498	6,392	149	191	6,275	13,505	SSW
SW	0	0	0	1,656	9,115	1,166	196	12,133	SW
WSW	0	0	0	24	1,893	36,252	3,646	41,815	WSW
W	0	0	0	0	92	1,204	20,963	22,259	W
WNW	0	0	0	0	190	2,757	0	2,947	WNW
NW	0	0	0	0	562	2,869	550	3,981	NW
NNW	0	0	0	0	478	3,257	1,819	5,554	NNW
Totals:	4	49	2,798	119,725	84,258	61,353	92,347	360,334	
Density	0	1	9	98	73	53	46	People/Sq Mile	

(1) Year 2000 population data, reproduced from Appendix F of Reference [A-2], using year 2000 U.S. Census data.

Table A-3[2]

2000 Population Distribution by Compass Sector and Distance from the Site

Distance (miles)	Direction (compass segment)	2000 Population	Distance (miles)	Direction (compass segment)	2000 Population	Distance (miles)	Direction (compass segment)	2000 Population
0-3	ALL	0	5-10	N	33	10-20	N	169
			5-10	NNE	71	10-20	NNE	680
3-4	NNE	0	5-10	NE	294	10-20	NE	1,535
3-4	ENE	2	5-10	ENE	281	10-20	ENE	912
3-4	E	4	5-10	E	312	10-20	E	567
3-4	ESE	3	5-10	ESE	369	10-20	ESE	479
3-4	SE-NNW	0	5-10	SE	471	10-20	SE	13,147
			5-10	SSE	118	10-20	SSE	65,247
4-5	N-NNE	0	5-10	S	391	10-20	S	27,095
4-5	NE	12	5-10	SSW	481	10-20	SSW	6,517
4-5	ENE	25	5-10	SW	17	10-20	SW	1,426
4-5	E	31	5-10	WSW-NW	0	10-20	WSW	21
4-5	ESE	24	5-10	NNW	3	10-20	WNW	0
4-5	SE	3				10-20	NNW	8
4-5	SSE-NNW	0						
0-5	TOTAL	104	0-10	TOTAL	2,945	0-20	TOTAL	120,748
20-30	N	1,158	30-40	N	1,077	40-50	N	30,168
20-30	NNE	10,663	30-40	NNE	3,643	40-50	NNE	713
20-30	NE	502	30-40	NE	251	40-50	NE	733
20-30	ENE	3,089	30-40	ENE	370	40-50	ENE	179
20-30	E	74	30-40	E	143	40-50	E	92
20-30	ESE	424	30-40	ESE	959	40-50	ESE	215
20-30	SE	14,781	30-40	SE	366	40-50	SE	2,915
20-30	SSE	42,124	30-40	SSE	408	40-50	SSE	3,876
20-30	S	841	30-40	S	5,494	40-50	S	19,644
20-30	SSW	143	30-40	SSW	186	40-50	SSW	3,857
20-30	SW	9,560	30-40	SW	1,398	40-50	SW	209
20-30	WSW	1,561	30-40	WSW	36,199	40-50	WSW	3,801
20-30	W	81	30-40	W	954	40-50	W	20,934
20-30	WNW	210	30-40	WNW	3,861	40-50	WNW	8
20-30	NW	531	30-40	NW	1,870	40-50	NW	577
20-30	NNW	406	30-40	NNW	3,290	40-50	NNW	1,707
0-30	TOTAL	206,896	0-40	TOTAL	267,365	0-50	TOTAL	356,993

[2] Source: Table 2.1-2 of UFSAR [A-3].

REFERENCES

[A-1] Sandia National Laboratories, Evaluation of Severe Accident Risk: Peach Bottom Unit 2, NUREG/CR-4551, SAND86-1309, Vol. 4, Rev. 1, Part 1, December 1990.

[A-2] Sandia National Laboratories, SECPOP2000: Sector Population, Land Fraction, and Economic Estimation Program, NUREG/CR-6525, Rev. 1, August 2003.

[A-3] Columbia Power Station (formerly Washington Nuclear Project (WNP-2)) Final Safety Analysis Report, Amendment 57.

APPENDIX B

EXTERNAL EVENTS ASSESSMENT

Appendix B
EXTERNAL EVENTS ASSESSMENT

This appendix discusses the external events assessment in support of the Columbia Generating Station (CGS) Integrated Leak Rate Test (ILRT) interval extension risk assessment.

External hazards were evaluated in the CGS Individual Plant Examination of External Events (IPEEE) Submittal in response to the NRC IPEEE Program (Generic Letter 88-20 Supplement 4). The IPEEE Program was a one-time review of external hazard risk and was limited in its purpose to the identification of potential plant vulnerabilities and the understanding of associated severe accident risks.

The results of the CGS IPEEE study are documented in the CGS IPEEE Main Report [B-1]. The primary areas of external event evaluation at CGS were internal fire, seismic, and volcanic activity. Adequate assurance regarding safe shutdown for volcanic events (i.e., design basis ash fall) was addressed via plant procedures and equipment modifications and no further examination (i.e., quantitative assessment) was performed for the IPEEE.

Seismic events were addressed through a Seismic Probabilistic Safety Assessment (SPSA) as part of the IPEEE. The seismic external event study provides adequate (but conservative) information to assess the impact of seismic hazards on the conclusions of the CGS ILRT interval extension risk assessment.

Internal fire events were addressed through a Fire Probabilistic Safety Assessment (FPSA). Its conclusions are considered a reasonable reflection of the current state of the technology and adequate for assessing the impact of fires on the conclusions of the ILRT interval extension risk assessment.

B.1 COLUMBIA INTERNAL FIRES ANALYSIS

The Columbia fire PSA was updated in 2003. The EPRI FIVE Methodology [B-2] and Fire PSA Implementation Guide (FPRAIG) [B-3] screening approaches, EPRI Fire Events Database [B-4] and plant specific data were used to perform the FPSA [B-5].

Based on the 2003 CGS fire PSA update, the CGS CDF contribution due to internal fires in the unscreened fire areas is calculated at 1.08E-5/yr.[1] As part of the impact assessment on possible large, early releases, the FPSA coupled with available generic insights offer the following conclusions:

- The FPSA investigated fire induced containment isolation failures and determined that scenarios with containment isolation failure were not likely containment failure modes.
- The FPSA does not quantify the LERF risk measure, however, review of NUREG-1742, Perspectives Gained from the IPEEE Program [B-7], indicates that the fire CDF for BWRs is primarily determined by plant transient type of events.

Given the above, it is judged reasonable to assume that the ratio of LERF to CDF for fire scenarios is comparable to the ratio determined for CGS internal events. For CGS internal events, the ratio of LERF (6.90E-7/yr) to CDF (7.33E-6/yr) is approximately 9.4%. As such, it is reasonable to assume here that fire-induced LERF is approximately 10% of the fire induced CDF (1.08E-5/yr), yielding a fire-induced LERF estimate of 1.1E-6/yr.

This information is used in Section B.4 of this appendix to provide insight into the impact of external hazard risk on the conclusions of this ILRT risk assessment.

[1] Table 4-1 of Reference [B-5].

B.2 COLUMBIA IPEEE SEISMIC ANALYSIS

The Columbia seismic PSA was performed as part of the Individual Plant Examination of External Events (IPEEE). The Columbia SPSA followed the guidance of NUREG-1407 [B-8], NUREG/CR-2300 [B-9] and EPRI NP-6041 [B-10]. The SPSA calculated an overall core damage frequency of 2.1E-5/yr[1]. The CGS IPEEE Seismic model was developed as a screening tool for one-time use in resolving the GL88-20 issues. The SPSA is not considered to be on the same best estimate basis as the CGS current Internal Events PSA (e.g., no recovery actions are modeled in the SPSA).

Similar to the Columbia fire PSA, the seismic PSA does not provide a detailed breakdown of the seismic risk profile by accident class. The CGS seismic PSA also does not distinguish between LERF and non-LERF accident sequence end states. The process used in this ILRT external events assessment for the determination of LERF and non-LERF end states included the following:

- An evaluation of the accident sequences to assess whether the timing of a projected release would be greater than 4 hours[2] following a declaration of a General Emergency (GE). This evaluation determined that approximately 9% of the seismic CDF is comprised of core damage events occurring in an early time frame (i.e., loss of all injection at or about t = 0)[3]. Conservatively assuming that all such seismic CDF accidents result in a large magnitude release, the CGS seismic LERF can be approximated by 9% of the seismic CDF. This results in an estimated seismic LERF of 1.9E-6/yr.
- An assessment of the ability to effectively evacuate people if such an event occurred. This assessment considered the following:
  - For relatively small magnitude seismic events (up to 0.3g), the ability to evacuate people is considered similar to the existing evacuation study for internal events.

(1) Page 1.0-7 of [B-1]
(2) Discussion of required CGS Emergency Plan evacuation timing is included in Appendix C.
(3) Sequences E(E)S10, ESS07, and "All Other Sequences" of Table 3.5-4 of [B-1].

- For relatively high magnitude seismic events (>0.5g), the containment is considered to be effectively bypassed regardless of the ILRT interval.[1] Therefore, it is always a LERF.
- For the narrow spectrum of seismic events in the 0.3g to 0.5g range, there is judged to be a combination of effects of low frequency within this narrow spectrum, and ability to effectively evacuate, such that those sequences with a delayed release (>4 hours from GE) are deemed to be non-LERF.

This information is used in Section B.4 of this appendix to provide quantitative insights into the impact of external hazard risk on the conclusions of this ILRT risk assessment.

B.3 OTHER EXTERNAL HAZARDS

In addition to internal fires and seismic events, the Columbia IPEEE Submittal analyzed a variety of other external hazards:

- Volcanic Activity
- High Winds/Tornadoes
- External Flooding
- Transportation and Nearby Facility Accidents
- Other External Hazards

The Columbia IPEEE analysis of volcanic activity, high winds, tornadoes, external floods, transportation accidents, nearby facility accidents, and other external hazards was accomplished by reviewing the plant environs against regulatory requirements regarding these hazards. Based upon this review, it was concluded that Columbia meets the applicable Standard Review Plan requirements and therefore has an

(1) The seismic PSA concluded that the High Confidence Low Probability of Failure (HCLPF) capacities of the reactor building, containment vessel, and containment internal structure were at least 0.50g peak ground acceleration, page 3.0-23 of Reference [B-1].

acceptably low risk with respect to these hazards. As such, these hazards were determined in the Columbia IPEEE to be negligible contributors to overall plant risk.

Accordingly, these other external event hazards are not included explicitly in this appendix and are reasonably assumed not to impact the results or conclusions of the ILRT interval extension risk assessment.

B.4 IMPACT OF EXTERNAL HAZARD RISK ON LERF

The NEI Interim Guidance calculation of delta LERF performed in Section 3 of this report is re-performed here including, in addition to internal event information, the Columbia IPEEE external event risk information discussed in the previous sections.

Per the NEI Interim Guidance, the impact on the LERF risk measure due to the proposed ILRT interval extension is calculated as follows:

delta LERF = (Frequency of EPRI Category 3b for 1-per-15 year ILRT interval) – (Frequency of EPRI Category 3b for 1-per-10 year ILRT interval)

As discussed in Section 3.1, the frequency per year for EPRI Category 3b is calculated as:

Frequency 3b = [3b conditional failure probability] x [CDF - (CDF with independent LERF + CDF that cannot cause LERF)]

Using the percentage of total CDF contributing to LERF developed in Sections B.1 through B.3 for fire and seismic as an approximation of the early CDF applicable to the frequency of Category 3b (i.e., potentially impacted by the ILRT test frequency) yields the following:

Frequency 3b = [2.70E-3] x [(1.08E-5/yr)(0.10) + (2.1E-5/yr)(0.09)]

Frequency 3b = 8.0E-9/yr

As discussed in the main report, this frequency is representative of the baseline (3-per-10 year) case.

Using the relationship described in Section 3.4.1 for the impact on 3b frequency due to increases in the ILRT surveillance interval, the EPRI Category 3b frequency for the 1-per-10 year and 1-per-15 year ILRT intervals are calculated as 2.7E-8/yr and 4.0E-8/yr, respectively. Therefore, the change in the LERF risk measure due to extending the ILRT from 1-per-10 years to 1-per-15 years, including both internal and external hazard risk, is estimated as:

	3b Frequency (3-per-10 yr ILRT)	3b Frequency (1-per-10 year ILRT)	3b Frequency (1-per-15 year ILRT)	LERF Increase[1]
External Events Contribution	8.0E-9/yr	2.7E-8/yr	4.0E-8/yr	1.3E-8/yr
Internal Events Contribution	1.19E-8/yr	3.97E-8/yr	5.95E-8/yr	1.98E-8/yr
Combined (Internal + External)	2.0E-8/yr	6.7E-8/yr	1.0E-7/yr	3.3E-8/yr

Thus, the increase in LERF due to the combined internal and external events contribution is estimated as 3.3E-8/yr.

B.5 COMPARISON TO RG 1.174 ACCEPTANCE GUIDELINES

NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", provides NRC recommendations for using risk information in support of applications requesting changes

[1] Associated with the change from the current 1-per-10 year frequency to the proposed 1-per-15 year frequency.

to the license basis of the plant. As discussed in Section 2 of this report, the risk acceptance criteria of RG 1.174 is used here to assess the ILRT interval extension.

The 3.3E-8/yr increase in LERF due to the combined internal and external events from extending the Columbia ILRT frequency from 1-per-10 years to 1-per-15 years falls into Region III ("Very Small Change" in risk) of the RG 1.174 acceptance guidelines. Per RG 1.174, when the calculated increase in LERF due to the proposed plant change is in the range of 1E-7 to 1E-6 per reactor year (Region II, "Small Change" in risk), the risk assessment must also reasonably show that the total LERF is less than 1E-5/yr. Although not required in this case (since the delta LERF is less than 1E-7 and falls in Region III), the total LERF from all hazards is calculated in this analysis for completeness.

Per the Columbia internal events Level 2 PRA (Rev. 5), the Columbia LERF due to internal event accidents is 6.9E-7/yr [B-6]. The LERF due to external events is estimated here based on the discussion of Sections B.1 and B.2:

Contributor	LERF
Fire	1.1E-6/yr
Seismic	1.9E-6/yr
Internal	6.9E-7/yr
Total	3.7E-6/yr

As can be seen, the external events LERF (fire and seismic) is estimated at 3.0E-6/yr. The internal events LERF is 6.9E-7/yr. Therefore, the total LERF for Columbia is estimated at 3.7E-6/yr, which is less than the RG 1.174 requirement to demonstrate that the total LERF of internal events and external events is less than 1E-5/yr. Therefore, per the guidance in RG 1.174, no further quantification of external event effects is necessary.

REFERENCES

[B-1] Washington Public Power Supply System (Columbia Generating Station), WNP-2 Individual Plant Examination for External Events, Main Report, June 1995.

[B-2] Professional Loss Control, Inc., Fire-Induced Vulnerability Evaluation (FIVE) Methodology Plant Screening Guide, EPRI TR-100370, Electric Power Research Institute, April 1992.

[B-3] W.J. Parkinson, et. al., Fire PRA Implementation Guide, EPRI TR-105928, Electric Power Research Institute, December 1995.

[B-4] NSAC/179L, Electric Power Research Institute, Fire Events Database for U.S. Nuclear Power Plants, Rev. 1, January, 1993.

[B-5] Columbia Generating Station, "CGS Fire PSA Quantification and Results Notebook," CGS FPSA-1-RE-0001, Rev. 1, 2003.

[B-6] Columbia Generating Station, "CGS PSA Containment Performance Analysis Notebook," CGS PSA-2-L2-0001, Rev. 2, May, 2004.

[B-7] U.S. Nuclear Regulatory Commission, Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program, NUREG-1742, Vol. 2, April 2002.

[B-8] U.S. Nuclear Regulatory Commission, Procedural and Submittal Guidance for the IPEEE for Severe Accident Vulnerabilities, June, 1991.

[B-9] J.W. Hickman, et. al., PRA Procedures Guide, NUREG/CR-2300, U.S. Nuclear Regulatory Commission, January, 1983.

[B-10] NTS Engineering, et. al., A Method for Assessment of Nuclear Power Plant Seismic Margin, EPRI NP-6041, Electric Power Research Institute, October 1988.

APPENDIX C

SENSITIVITY FOR LONG TERM STATION BLACKOUT

Appendix C
SENSITIVITY FOR LONG TERM STATION BLACKOUT

The definition of Large Early Release Frequency (LERF) requires the release to be both "large" and "early." The definition of early is tied to a release that occurs prior to effective protective actions for the public (e.g., evacuation). Station Blackout (SBO) events which result in radionuclide release at times after effective public protective actions have been implemented (sometimes referred to as long term SBO events) are not treated as LERF. The Nuclear Regulatory Commission (NRC) has previously requested licensees to provide (as part of ILRT RAIs) the technical justification for the assumption that long term station blackout scenarios do not contribute to LERF, and to provide an assessment of the impact on risk results if long-term station blackout sequences were retained in selected EPRI categories.

This appendix provides the technical justification for not including long term SBO scenarios in the LERF assessment and also provides a sensitivity case to demonstrate that retaining SBO sequences in the analysis does not change the conclusions of the overall ILRT assessment.

C.1 TECHNICAL JUSTIFICATION FOR EXCLUSION OF SBO SCENARIOS

Typical of many industry PRAs, the Columbia PRA uses a radionuclide release categorization scheme comprised of two factors: release timing and release magnitude. The Columbia long-term station blackout (LT SBO) core damage accidents are classified as non-LERF releases based on release timing rather than release magnitude (i.e., LT SBO core damage accidents have the potential to result in the entire spectrum of release magnitudes, including High magnitude releases; but, they cannot result in Early releases). The following describes the timing issues of Columbia LT SBO scenarios.

Three timing categories are typically used in a full Level 2 analysis, as follows:

1. Early (E) — Less than approximately 4 hours
2. Intermediate (I) — Greater than or equal to approximately 4 hours, but less than or equal to 24 hours
3. Late (L) — Greater than 24 hours.

The above accident release categories are based upon past experience concerning offsite accident response. The timing categories are relative to the declaration of the General Emergency Action Level.

- 0-4 hours is conservatively assumed to include cases in which minimal offsite protective measures have been observed to be performed historically in accidents requiring population evacuation.
- 4-24 hours is a time frame in which much of the offsite nuclear plant protective measures can be assured to be accomplished.
- >24 hours are times at which the offsite measures can be assumed to be effective and resources for repair and recovery can be implemented effectively.

Since Columbia Generating Station (CGS) Level 2 analysis is a LERF model as opposed to a full Level 2, CGS utilizes only two time categories, as follows:

1. Early (E) — Less than 4 hours.[1] Release occurs prior to completion of emergency zone evacuation.
2. Late (L) — Greater than 4 hours.[1] Release occurs after completion of emergency zone evacuation.

The Columbia LT SBO accident scenarios include only those sequences in which RPV injection is available until the time of battery depletion (4 hours minimum) and core

[1] Section 4.4.1.1, page 144 of Reference [C-2]. CGS has an unusually short evacuation completion time of 2 hours 30 minutes due to low population within the 10-mile emergency planning zone.

damage subsequently occurs due to AC recovery failure. The three plant damage state (PDS) long term SBO sequences for CGS are as follows:

PDS Class	Description[1]	CDF[2]
VIA2	LT SBO with SORV, RPV at low pressure after battery depletion due to SORV	3.72E-8/yr
VIB1	LT SBO with HPCS (independent diesel AC powered) for 12 hours, RPV repressurizes	1.03E-6/yr
VIB2	LT SBO with RCIC until battery depletion (early HPCS failure), RPV repressurizes	2.12E-7/yr

Section 4.3.1.2 of the CGS LERF analysis [C-2] notes that releases associated with these three SBO scenarios are classified as late because timing of core melt occurs in the 12-hour time frame relative to the time a general emergency is declared. This projected time of core damage is well beyond the time estimated to complete emergency zone evacuation (estimated at 2 hours 30 minutes per Reference [C-1]).

The Columbia Emergency Plan directs declaration of a General Emergency (i.e., the emergency classification with associated directives for evacuation) for the following station blackout conditions [C-3]:

1. Loss of all AC power to bus SM-7

   AND

2. Loss of all AC power to bus SM-8

   AND

(1) Descriptions from Section 4.3.1.2 of Reference [C-2]
(2) From Table 2-1 of the ILRT main report

3. Any of the following:
   - Restoration of power to EITHER bus SM-7 OR SM-8 within 4 hours is NOT likely.

     OR

   - RPV level less than 161 inches

Additionally, the CGS Emergency Plan directs declaration of a General Emergency based on "other" considerations as follows:

- Any event, in the judgement of the Emergency Director, that could lead or has led to a loss of any two fission product barriers and loss or potential loss of the third fission product barrier.

The loss of offsite and emergency power occurs at t=0 for SBO sequences. The Columbia PRA assumes that the determination that AC power is not likely to be restored within the 4-hour time frame is made within the first hour into the accident. As such, a General Emergency is assumed declared at 1 hour into the event. The evacuation process would be initiated within 30 minutes after the declaration and is estimated to be completed within 2 hours 30 minutes under worst assumed conditions based on site specific evacuation studies for weather and times of day variations[(1)] [C-1]. The earliest core damage for the long term SBO would not occur for many more hours (i.e., beyond the 4-hour total time). Therefore, the long term SBO core damage accidents are not an Early release.

(1) This fast evacuation time is primarily due to the low population within the 10-mile Emergency Planning Zone. The evacuation accounts for 3,674 permanent residents, 15,456 transients (e.g., local employees) and 568 special individuals (i.e., schools requiring bus evacuation), per Figure 5-3 of Reference [C-1].

C.2 SENSITIVITY FOR LONG TERM SBO

The NRC has previously asked licensees, via Requests for Additional Information (RAIs), to provide an assessment of the impact on risk results if long term station blackout sequences were retained in selected EPRI categories. This appendix provides that information for use by the NRC.

The frequency of long term SBO core damage sequences in the Columbia PRA LERF model is as follows (from Table 2-1 of the ILRT main report):

Class VIA2 - Long term DC (and ADS) power not available at time of core melt. Stuck Open SRV.	3.72E-8
Class VIB1 - Long term DC power not available at time of core melt. HPCS recoverable with recovery of AC power.	1.03E-6
Class VIB2 - Long term DC power not available at time of core melt. HPCS not recoverable.	2.12E-7
Total	1.28E-6

This sensitivity case repeats the calculations of the ILRT assessment performed in Section 3 of the main report, with the exception that the long term SBO sequences listed above are retained in the EPRI Category 3b frequency calculations as potential LERF contributors. Refer to Section 3.1 of the main report for the EPRI Category calculational methodology.

Retaining the SBO sequences in the EPRI Categories 3a and 3b frequency calculations results in the following new frequencies for the 3-per-10 year baseline case:

Category 3a = 1.53E-7/yr
Category 3b = 1.53E-8/yr
Category 1 = 2.03E-6/yr

The impact of the changes to these EPRI Category frequencies is shown in Table C-1.

The total increase in LERF using the full power internal events PRA due to the extension of the ILRT interval from 10 years to 15 years is determined to be 2.56E-8/yr when long term SBO scenarios are included in the EPRI Category 3a and 3b frequencies. This represents an additional LERF increase of 5.8E-9/yr (a 29% increase) over the best estimate ILRT increase in LERF of 1.98E-8/yr based on internal events. Including the long term SBO contribution, however, still results in a LERF increase below the NRC Regulatory Guide 1.174 criterion of 1.0E-7/yr for "very small" risk change. The population dose rate for the 10 year and 15 year ILRT intervals with long term SBO sequences included remains negligible. The increase in the conditional containment failure probability (CCFP) increases slightly from 0.1% to 0.3% (with LT SBO sequences included).

The sensitivity case demonstrates that even if long term SBO scenarios are included in the EPRI Category 3a and 3b frequencies, the conclusion of the risk assessment does not change; that is, the Columbia ILRT interval extension to 15 years has a minimal impact on plant risk.

Table C-1

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL

- Sensitivity Case to Include Long-Term SBO Contributions in Category 3a and 3b Frequencies -

EPRI Category	Dose (Person-Rem Within 50 miles)	Baseline (3-per-10 year ILRT)		Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
		Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	6.68E+2	2.03E-6	1.36E-3	1.64E-6	1.09E-3	1.36E-6	9.06E-4
2	3.46E+5	4.43E-9	1.53E-3	4.43E-9	1.53E-3	4.43E-9	1.53E-3
3a	6.68E+3	1.53E-7	1.02E-3	5.11E-7	3.41E-3	7.67E-7	5.12E-3
3b	2.34E+4	1.53E-8	3.59E-4	5.11E-8	1.20E-3	7.67E-8	1.79E-3
4	n/a	n/a	n/a	n/a	n/a	n/a	n/a
5	n/a	n/a	n/a	n/a	n/a	n/a	n/a
6	n/a	n/a	n/a	n/a	n/a	n/a	n/a
7	2.57E+5	4.97E-6	1.28	4.97E-6	1.28	4.97E-6	1.28
8	3.46E+5	1.57E-7	5.43E-2	1.57E-7	5.43E-2	1.57E-7	5.43E-2
TOTALS:		7.33E-6	1.34	7.33E-6	1.34	7.33E-6	1.34
Increase in Dose Rate (1)					neg.		neg. (4)
Increase in LERF (2)				3.59E-8		2.56E-8	
Increase in $CCFP_{\%}$ (3)				0.5%		0.3%	

(1) The Increase in Dose Rate (person-rem/year) is with respect to the preceding ILRT interval, and is calculated by subtracting the Dose Rate totals.

(2) The Increase in LERF is with respect to the preceding ILRT interval, and is calculated by subtracting the EPRI Category 3b frequencies.

(3) The Increase in CCFP% (units in percentage points) is with respect to the preceding ILRT interval. The CCFP% is calculated as:

$$CCFP_{\%} = [1 - ((\text{Category 1 Frequency} + \text{Category 3a Frequency}) / CDF)] \times 100$$

(4) The change in dose rate from the 1-per-10 year ILRT frequency to the 1-per-15 year ILRT frequency is approximately 2E-3 person-rem/yr. This change is less than the number of significant figures being reasonably carried in the assessment.

REFERENCES

[C-1] Columbia Generating Station, Emergency Plan, Rev. 37, October 2003.

[C-2] Columbia Generating Station, "Containment Performance Analysis Notebook", CGS PSA-2-L2-0001, Rev. 2, May, 2004.

[C-3] Washington Public Power Supply System, "Columbia Generating Station Plant Procedures Manual, Emergency Plan Implementing Procedures"; Section 13.1.1 Classifying the Emergency, Rev. 31, May 10, 2002; Section 13.1.1a Classifying the Emergency - Technical Bases, Rev. 11, May 2, 2003.

APPENDIX D

CONTAINMENT DEGRADATION SENSITIVITY

Appendix D
CONTAINMENT CORROSION SENSITIVITY

D.1 BACKGROUND

Inspections of some reinforced and steel containments (e.g., North Anna, Brunswick, D.C. Cook, and Oyster Creek) have indicated degradation from the uninspectable (embedded) side of the steel shell and liner of primary containment. In response to previous ILRT extension request submittals, the NRC has consistently requested licensees to perform a quantitative assessment of the impact on LERF due to age-related degradation of non-inspectable areas of the containment. Therefore, a quantitative assessment using the same approach used by other industry plants (e.g., Calvert Cliffs) is included as Appendix D to this ILRT extension evaluation.

The analysis described in Sections 3 and 4 of the main report was performed to evaluate the risk impact of extending the Integrated Leak Rate Test (ILRT) interval for the Columbia Generating Station (CGS). That analysis was performed using the recommended approach developed by NEI for performing assessments of one-time extensions for containment ILRT surveillance intervals [D-1]. The results of that analysis are summarized in Table D-1, which is a copy of Table 4-1 from the main report.

The risk increase from extending the ILRT interval from the present 1-in-10 year requirement to 1-in-15 years is quantified by the increase in LERF (the CDF is not impacted by the ILRT interval). The NRC Regulatory Guide 1.174 [D-2] defines very small changes in risk as resulting in increases in LERF below 1.0E-7/yr. The Regulatory Guide also states that when the calculated increase in LERF is in the range of 1.0E-6/yr to 1.0E-7/yr, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.0E-5/yr.

For Columbia the increase in LERF from the 1-in-10 year interval to the 1-in-15 year interval was calculated in the main report to be 1.98E-8/yr, which is well below the very small change threshold. Also, the dose rate increase was determined to be negligible compared with the total dose rate of 1.34 person-rem/yr. The increase in the containment failure probability (CCFP) was determined to be 0.1%, which is also judged to be insignificant.

Table D-1

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL (FROM MAIN REPORT)

EPRI Category	Dose (Person-Rem Within 50 miles)	Baseline (3-per-10 year ILRT)		Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
		Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	6.68E+2	2.07E-6	1.38E-3	1.76E-6	1.18E-3	1.55E-6	1.03E-3
2	3.46E+5	4.43E-9	1.53E-3	4.43E-9	1.53E-3	4.43E-9	1.53E-3
3a	6.68E+3	1.19E-7	7.95E-4	3.97E-7	2.65E-3	5.95E-7	3.97E-3
3b	2.34E+4	1.19E-8	2.78E-4	3.97E-8	9.27E-4	5.95E-8	1.39E-3
4	N/A	N/A	N/A	N/A	N/A	N/A	N/A
5	N/A	N/A	N/A	N/A	N/A	N/A	N/A
6	N/A	N/A	N/A	N/A	N/A	N/A	N/A
7	2.57E+5	4.97E-6	1.28	4.97E-6	1.28	4.97E-6	1.28
8	3.46E+5	1.57E-7	5.43E-2	1.57E-7	5.43E-2	1.57E-7	5.43E-2
TOTALS:		7.33E-6	1.34	7.33E-6	1.34	7.33E-6	1.34
Increase in Dose Rate (1)					neg.		neg.
Increase in LERF (2)				2.78E-8		1.98E-8	
Increase in CCFP(%)(3)				0.4%		0.1%	

Notes to Table D-1:

(1) The increase in dose rate (person-rem/year) is with respect to the results for the preceding ILRT interval, as presented in the table. For example, the increase in dose rate for the proposed 1-per-15 ILRT is calculated as: total dose rate for 1-per-15 year ILRT, minus total dose rate for 1-per-10 year ILRT. For each case, the dose rate increase is insignificant.

(2) The increase in Large Early Release Frequency (LERF) is with respect to the results for the preceding ILRT interval, as presented in the table. As discussed in Section 3.4.4 of the report, the change in LERF is determined by the change in the accident frequency of EPRI Category 3b. For example, the increase in LERF for the proposed 1-per-15 ILRT is calculated as: 3b frequency for 1-per-15 year ILRT, 5.95E-8/yr, minus 3b frequency for 1-per-10 year ILRT, 3.97E-8/yr, equals1.98E-8/yr.

(3) As discussed in Section 3.4.5, the conditional containment failure probability (CCFP) is calculated as:

$CCFP_{\%}$ = [1 – ((Category #1 Frequency + Category #3a Frequency) / CDF)] x 100%

D.2 CORROSION ANALYSIS

This containment corrosion sensitivity analysis utilizes the methods of the Calvert Cliffs liner corrosion analysis [D-3] to estimate the likelihood and risk-implications of degradation-induced leakage occurring undetected during the extended test interval. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. The Columbia containment is a pressure-suppression BWR/Mark II type with a steel shell in the drywell and wetwell regions, as shown in Figure D-1. The shell is surrounded by a concrete shield.

The following approach is used to determine the change in likelihood of detecting corrosion of the steel containment shell due to extending the ILRT. This likelihood is then used to determine the resulting change in risk. Consistent with the Calvert Cliffs analysis, the following issues are addressed:

- Differences between the containment basemat and other regions of the containment
- The historical steel liner/shell flaw likelihood due to concealed corrosion
- The impact of aging
- The likelihood that visual inspections will be effective at detecting a flaw

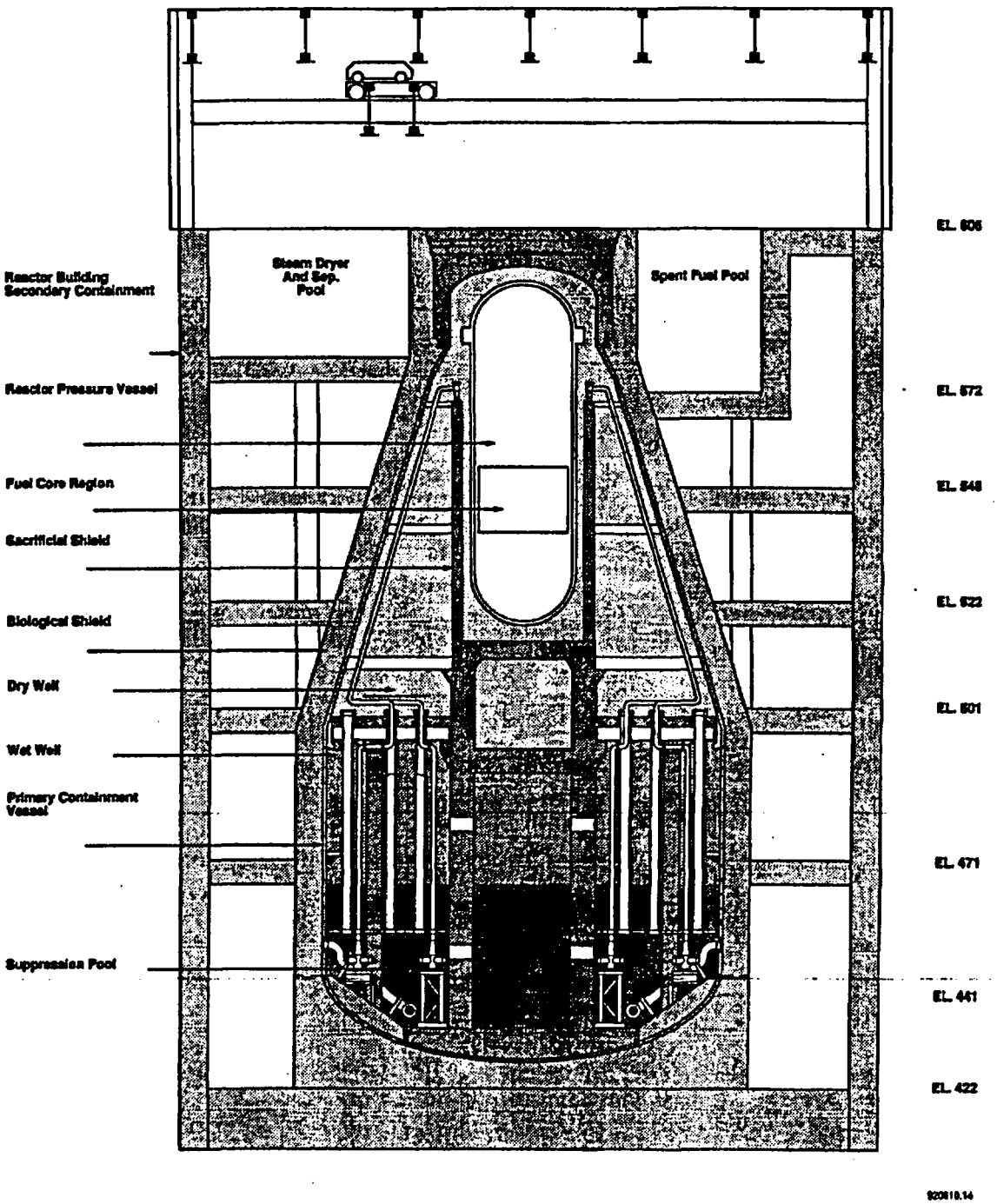


FIGURE D-1 CGS PRIMARY CONTAINMENT

D.2.1 ASSUMPTIONS

The following assumptions are utilized in the sensitivity analysis:

A. Consistent with the Calvert Cliffs analysis, a half failure is assumed for basemat concealed liner corrosion due to the lack of applicable industry events. Assuming a half failure when zero failures have ocurred is a typical PRA approach. (See Table D-2, Step 1.)

B. The two corrosion events used to estimate the wall liner flaw probability in the Calvert Cliffs analysis are assumed to be applicable to the Columbia containment analysis. These events, one at North Anna Unit 2 and one at Brunswick Unit 2, were initiated from the non-visible (backside) portion of the containment liner. (See Table D-2, Step 1.)

C. For consistency with the Calvert Cliffs analysis, the estimated historical flaw probability is calculated using a 5.5-year data period. This reflects the span from September 1996 when 10 CFR 50.55a started requiring visual inspection to the time of the Calvert Cliffs analysis. Additional success data were not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date (and have been performed since the time frame of the Calvert Cliffs analysis), and there is no evidence that additional corrosion issues were identified. (See Table D-2, Step 1.)

D. Consistent with the Calvert Cliffs analysis, the corrosion-induced steel liner/shell flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increased likelihood of corrosion as the steel shell ages. (See Table D-2,

Steps 2 and 3.) Sensitivity studies are included that address doubling this rate every ten years and every two years.

E. In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated (based on an assessment of the containment fragility curve versus the ILRT test pressure) as 1.1% for the containment walls and dome region and 0.11% (factor of ten less) for the basemat. For Columbia the containment failure probabilities are conservatively assumed to be 10% for the shell wall and 1% for the basemat. Because the basemat for the Columbia containment is in the suppression pool, it is judged that failure of this area would not lead to LERF. Hence, the assumed 1% probability is particularly conservative. Sensitivity studies are included that increase the probabilities by a factor of five and decrease them by an order of magnitude. (See Table D-2, Step 4.)

F. In the Calvert Cliffs analysis it is noted that approximately 85% of the interior wall surface is accessible for visual inspections. The Columbia interior wall surface accessible for visual inspections is estimated to be at least 90% (the majority of uninspectable wall surface being the area between the drywell floor slab and the DW-WW omega seal). Therefore, consistent with the Calvert analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a 5% likelihood of a non-detectable flaw are used. This results in a total undetected flaw probability of 10%, which is assumed in the base case analysis. (See Table D-2, Step 5.) Sensitivity studies are included that evaluate a total detection failure likelihood of 5% and 15%, respectively. (See Table D-4 for sensitivity studies.) Additionally, it should be noted that to date, all liner/shell corrosion events have been detected through visual inspection and repaired.

G. Consistent with the Calvert analysis, all non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

D.2.2 ANALYSIS

Table D-2
STEEL SHELL CORROSION BASE CASE

Step	Description	Containment Walls		Containment Basemat	
1	**Historical Steel Liner/Shell Flaw Likelihood** Failure Data: Containment location specific (consistent with the Calvert Cliffs analysis).	Industry Applicable Events: 2 (North Anna and Brunswick events assumed to be applicable to Columbia) 2/(70 * 5.5) = **5.2E-3** (Based on 70 units with liners over 5.5 years)		Industry Applicable Events: 0 (assume 0.5) 0.5/(70 * 5.5) = **1.3E-3** (Based on 70 units with liners over 5.5 years)	
2	**Age Adjusted Steel Liner/Shell Flaw Likelihood** During 15-year interval, assume failure rate doubles at the end of every five years (which equates to a 14.9% increase per year). The average over the 5th through 10th year period is set equal to the historical failure rate of Step 1 (consistent with Calvert Cliffs analysis). These assumptions are used to calculate the flaw likelihood for each year (for a 15 year period)	Year	Flaw Likelihood	Year	Flaw Likelihood
		0	1.79E-03	0	4.47E-04
		1	2.05E-03	1	5.13E-04
		2	2.36E-03	2	5.89E-04
		3	2.71E-03	3	6.77E-04
		4	3.11E-03	4	7.77E-04
		5	3.57E-03	5	8.93E-04
		6	4.10E-03	6	1.03E-03
		7	4.71E-03	7	1.18E-03
		8	5.41E-03	8	1.35E-03
		9	6.22E-03	9	1.55E-03
		10	7.14E-03	10	1.79E-03
		11	8.21E-03	11	2.05E-03
		12	9.43E-03	12	2.36E-03
		13	1.08E-02	13	2.71E-03
		14	1.24E-02	14	3.11E-03
		15	1.43E-02	15	3.57E-03

Table D-2
STEEL SHELL CORROSION BASE CASE

Step	Description	Containment Walls	Containment Basemat
3	**Flaw Likelihood at 3, 10, and 15 years** This cumulative probability uses the age adjusted liner/shell flaw likelihood of Step 2 (consistent with Calvert Cliffs analysis – See Table 6 of Reference [D-3]). For example, the 7.12E-03 (at 3 years) cumulative flaw likelihood is the sum of the year 1, year 2, and year 3 likelihoods of Step 2.	**7.12E-3 (at 3 years)** **4.14E-2 (at 10 years)** **9.66E-2 (at 15 years)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis the values are calculated based on the 3, 10, and 15 year intervals.)	**1.78E-3 (at 3 years)** **1.03E-2 (at 10 years)** **2.41E-2 (at 15 years)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis the values are calculated based on the 3, 10, and 15 year intervals.)
4	**Likelihood of Breach in Containment Given Steel Liner/Shell Flaw** The wall failure probability of the containment is assumed to be 10% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 1%, (compared to 0.11% in the Calvert Cliffs analysis).	**10%**	**1%**
5	**Visual Inspection Detection Failure Likelihood** Utilize assumptions consistent with Calvert Cliffs analysis.	**10%** 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-wall but could be detected by ILRT). All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	**100%** Cannot be visually inspected.

Table D-2
STEEL SHELL CORROSION BASE CASE

Step	Description	Containment Walls	Containment Basemat
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3 * 4 * 5)	**7.12E-5 (at 3 years)** 7.12E-3 * 10% * 10% **4.14E-4 (at 10 years)** 4.14E-2 * 10% * 10% **9.66E-4 (at 15 years)** 9.66E-2 * 10% * 10%	**1.78E-5 (at 3 years)** 1.78E-3 * 1% * 100% **1.03E-4 (at 10 years)** 1.03E-2 * 1% * 100% **2.41E-4 (at 15 years)** 2.41E-2 * 1% * 100%

Cumulative Likelihood of Non-Detected Containment Leakage Due to Corrosion

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum in Step 6 for the containment walls and the containment basemat:

At 3 years: 7.12E-5 + 1.78E-5 = 8.90E-5

At 10 years: 4.14E-4 + 1.03E-4 = 5.17E-4

At 15 years: 9.66E-4 + 2.41E-4 = 1.21E-3

Table D-3 summarizes the results of the revised ILRT assessment including the potential impact from non-detected corrosion-induced containment leakage scenarios, with the assumption that all of these scenarios result in EPRI Class 3b (i.e., LERF). The impact of including the potential for corrosion-induced leakages compared to the original analysis results presented in the main report is noted in parentheses.

The factors calculated above are applied to those core damage accidents that are not already independently LERF or that could never result in LERF. For example, the 3-in-10 year case is calculated as follows:

- Per Table D-1, the EPRI Class 3b frequency is 1.19E-8/yr.
- As discussed in Section 3.1, the Columbia CDF associated with accidents that are not independently LERF or that could never result in LERF is 7.33E-6/yr – 5.32E-7/yr – 2.40E-6/yr = 4.40E-6/yr.
- The increase in the base case Class 3b frequency due to the corrosion-induced concealed flaw issue is calculated as 4.40E-6/yr * 8.90E-5 = 3.92E-10/yr, where 8.90E-5 was previously shown to be the cumulative likelihood of non-detected containment leakage due to corrosion at 3 years.
- The 3-in-10-year Class 3b frequency including the corrosion-induced concealed flaw issue is then calculated as 1.19E-8/yr + 3.92E-10/yr = 1.23E-8/yr.

Table D-3

COLUMBIA ILRT CASES: BASE, 1-IN-10, AND 1-IN-15 YR EXTENSIONS (Including Age Adjusted Steel Shell Corrosion Likelihood) [1]

EPRI Category	Dose (Per-Rem)	Base Case 3-in-10 Years		Current 1-in-10 Years		Proposed 1-in-15 Years	
		Core Damage Frequency (/yr)	Dose Rate (person-Rem/yr)	Core Damage Frequency (/yr)	Dose Rate (person-Rem/yr)	Core Damage Frequency (/yr)	Dose Rate (person-Rem/yr)
1	6.68E+2	2.07E-6	1.38E-3	1.76E-6	1.18E-3	1.54E-6	1.03E-3
2	3.46E+5	4.43E-9	1.53E-3	4.43E-9	1.53E-3	4.43E-9	1.53E-3
3a	6.68E+3	1.19E-7	7.95E-4	3.97E-7	2.65E-3	5.95E-7	3.97E-3
3b	2.34E+4	1.23E-8	2.88E-4	4.20E-8	9.82E-4	6.48E-8	1.52E-3
7	2.57E+5	4.97E-6	1.28	4.97E-6	1.28	4.97E-6	1.28
8	3.46E+5	1.57E-7	5.43E-2	1.57E-7	5.43E-2	1.57E-7	5.43E-2
Total		7.33E-6	1.34	7.33E-6	1.34	7.33E-6	1.34
Dose Rate from 3a and 3b (person-Rem/yr)			1.08E-3 (9.17E-6)		3.63E-3 (5.34E-5)		5.49E-3 (1.25E-4)
Increase in Total Dose Rate (person-Rem/yr)		From 3-per-10 yr	---		2.35E-3 =0.2% (5.00E-5)		4.06E-3 =0.3% (1.10E-4)
		From 1-per-10 yr	---		---		1.71E-3 =0.1% (6.39E-5)
LERF from 3b (/yr)			1.23E-8 (3.92E-10)		4.20E-8 (2.30E-9)		6.48E-8 (5.30E-9)
Increase in LERF (/yr)		From 3-per-10 yr	---		2.97E-8 (1.91E-9)		5.25E-8 (4.93E-9)
		From 1-per-10 yr	---		---		2.28E-8 (3.04E-9)
CCFP %			70.2% (neg)		70.6% (neg)		70.9% (0.1%)
Increase in CCFP		From 3-per-10 yr	---		0.4% (neg)		0.7% (neg)
		From 1-per-10 yr	---		---		0.3% (neg)

[1] The numbers in parenthesis represent the incremental change (compared to Table D-1) due to inclusion of the impact from the corrosion analysis.

Results

Based on the results shown in Table D-3, it can be seen that including corrosion effects in the ILRT assessment does not alter the conclusions from the original analysis. The increase in LERF from the 1-in-10 year interval to the 1-in-15 year interval is 2.28E-8/year, compared with 1.98E-8/yr without corrosion effects. This is still well below the Regulatory Guide 1.174 [D-2] acceptance criterion threshold for very small changes in risk of 1.0E-7/yr. This confirms that the proposed interval extension is acceptable from a risk basis. Additionally, the dose rate increase is negligible compared to the total 1.34 person-rem/yr. The increase in the CCFP is determined to be insignificant (70.9% for the 1-in-15 year case versus 70.6% for the 1-in-10 year case).

D.3 SENSITIVITY STUDIES

Sensitivity cases were also developed to gain an understanding of the sensitivity of this analysis to the various key parameters. The time for the flaw likelihood to double was adjusted from every five years to every two and every ten years. The failure probabilities for the shell wall and the basemat were increased by a factor of five and decreased by an order of magnitude. The total detection failure likelihood was adjusted from 10% to 15% and 5%. The results of the sensitivity cases are summarized in Table D-4. For all cases, the increase in LERF (for 1-in-10 years to 1-in-15 years) due to corrosion is less than 1.0E-7/yr. The total increase in LERF is within the range of 1.99E-8/yr and 8.38E-8/yr.

Table D-4

COLUMBIA STEEL SHELL CORROSION SENSITIVITY CASES

Age (Step 3)	Containment Breach (Step 4)	Visual Inspection & Non-Visual Flaws (Step 5)	Increase in Class 3b Frequency (LERF) for ILRT Extension From 1-in-10 to 1-in-15 years (/yr)	
			Total Increase	Increase Due to Corrosion
Base Case (Doubles every 5 yrs)	Base Case (10% Walls, 1% Basemat)	Base Case (10%)	2.28E-8	3.04E-9
Doubles every 2 yrs	Base	Base	2.89E-8	9.14E-9
Doubles every 10 yrs	Base	Base	2.19E-8	2.09E-9
Base	Base	15%	2.41E-8	4.25E-9
Base	Base	5%	2.16E-8	1.82E-9
Base	50% Walls, 5% Basemat	Base	3.50E-8	1.52E-8
Base	1% Walls, 0.1% Basemat	Base	2.01E-8	3.04E-10
Lower Bound				
Doubles every 10 yrs	1% Walls, 0.1% Basemat	5%	1.99E-8	1.25E-10
Upper Bound				
Doubles every 2 yrs	50% Walls, 5% Basemat	15%	8.38E-8	6.40E-8

D.4 SUMMARY AND CONCLUSIONS

This sensitivity analysis provides a quantitative assessment of the impact on risk of the potential for undetected steel shell corrosion due to an extension of the ILRT interval. The increase in LERF due to extending the test interval from the present schedule of 1 in 10 years to 1 in 15 years is 2.28E-8/yr, of which 3.04E-9/yr is due to corrosion. This value is considerably less than the RG 1.174 "very small" change criterion of 1.0E-7/yr. This confirms that the proposed interval extension is acceptable from a risk basis. Additionally, a series of parametric sensitivity studies regarding the potential age-related corrosion effects on the steel shell indicate that even with very conservative assumptions, the conclusions from the original analysis would not change; that is, the ILRT interval extension is judged to have a minimal impact on public risk and is therefore acceptable.

REFERENCES

[D-1] *Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Intervals*, Developed for NEI by John M. Gisclon, EPRI Consultant, William Parkinson and Ken Canavan, Data Systems and Solutions, November 2001.

[D-2] U.S. Nuclear Regulatory Commission, *"An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,"* Regulatory Guide 1.174, Revision 1, November 2002.

[D-3] *Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension*, Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to NRC Document Control Desk, March 27, 2002.

NOTE:

Appendix E of this report is not applicable to this license amendment request and has not been included. Appendix E of this report was developed to justify a future license amendment request regarding drywell to suppression chamber bypass leak rate testing.