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10 CFR 50.90

Docket No. 50-278

License No. DPR-56

May 30, 2001

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: Peach Bottom Atomic Power Station, Unit 3
License Amendment Request 01-00430

Dear Sir/Madam:

In a letter dated May 30, 2001, Exelon Generation Company, LLC, submitted License Amendment Request 01-00430, in accordance with 10 CFR 50.90, requesting an amendment to the Technical Specifications (Appendix A) of Operating License No. DPR-56, for Peach Bottom Atomic Power Station (PBAPS), Unit 3. This proposed change will revise Technical Specifications (TS) Section 5.5.12 ("Primary Containment Leakage Rate Testing Program") to reflect a one-time deferral of the Type A Containment Integrated Leak Rate Test (ILRT) to no later than December, 2007.

In that letter, Exelon Generation Company, LLC, committed to developing detailed performance based, risk-informed information to support this request. Attached is the analysis.

We have determined that the additional information does not alter the Conclusions, the Information Supporting a Finding of No Significant Hazards Consideration, or the Information Supporting an Environmental Impact Assessment in LAR 01-00430.

If you have any questions, please do not hesitate to contact us.

Very truly yours,

James A. Hutton
Director - Licensing

Enclosures: Affidavit, Attachment

cc: H. J. Miller, Administrator, Region I, USNRC
A. C. McMurtray, USNRC Senior Resident Inspector, PBAPS
R. R. Janati, Commonwealth of Pennsylvania

A017

COMMONWEALTH OF PENNSYLVANIA :

: ss.

COUNTY OF CHESTER :

J. J. Hagan, being first duly sworn, deposes and says:

That he is Senior Vice President of Exelon Generation Company, LLC; the Applicant herein; that he has read the attached performance based, risk-informed information concerning License Amendment Request 01-00430, for Peach Bottom Facility Operating License DPR-56, and knows the contents thereof; and that the statements and matters set forth therein are true and correct to the best of his knowledge, information and belief.

JJHagan

Senior Vice President

Subscribed and sworn to

before me this 30th day

of May 2001.

Vivia V. Gallimore

Notary Public

Notarial Seal
Vivia V. Gallimore, Notary Public
Tredyffrin Twp., Chester County
My Commission Expires Oct. 6, 2003

Member, Pennsylvania Association of Notaries

ATTACHMENT 1

PEACH BOTTOM ATOMIC POWER STATION
UNIT 3

Docket No. 50-278

License No. DPR-56

LICENSE AMENDMENT REQUEST
01-00430

RISK IMPACT ASSESSMENT

PEACH BOTTOM ATOMIC POWER STATION

RISK IMPACT ASSESSMENT OF EXTENDING THE CONTAINMENT TYPE A TEST INTERVAL

P1050001-1792

Prepared by: Donald E. Vanover Date: 05/22/01

Reviewed by: [signature] Date: 05/23/01

Accepted by: [signature] Date: 05/25/01

Approved by: [signature] Date: 5/25/01

Revisions:

Rev.	Description	Preparer/Date	Reviewer/Date	Approver/Date

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ES EXECUTIVE SUMMARY

Revisions to 10CFR50, Appendix J 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. Consistent with the guidelines in NEI 94-01, 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 $1.0L_a$. Peach Bottom selected the revised requirements for its testing program with the next ten-year Type A test due during the upcoming refueling outage, 3R13, scheduled for the fall of 2001. Prior to the performance of that test, however, Peach Bottom is seeking an extension of the test interval to sixteen years. A substantial cost savings will be realized and unnecessary personnel radiation exposure will be avoided by deferring the Type A test for an additional six (6) years. Cost savings have been estimated for this outage at approximately $1.5 million, which includes labor, equipment and critical path outage time needed to perform the test. Personnel radiation exposure reduction is estimated at 2.0 rem.

A risk assessment of the proposed extension of the Containment Type A test interval from once-per-ten-years to once-per-sixteen-years for Peach Bottom has been completed. In performing the risk assessment, the methodology described in EPRI TR-104285 is implemented, and the NRC Regulatory Guide 1.174 guidance on evaluating findings and risk insights in support of a licensee request for changes to a plant's licensing basis is applied. The approach is also consistent with that presented in NUREG-1493.

The risk assessment performed uses more recent risk models and data to confirm previously published information that concludes extending the ILRT interval results in a very small increase in risk.

The analysis uses the current Peach Bottom internal events PRA model that includes a full Level 2 analysis of core damage scenarios and subsequent containment response resulting in various fission product release categories (including no release). The release category end states from the PBAPS Level 2 model are also applied to align with those used by the NRC in NUREG/CR-4551 for Peach Bottom. This categorization allows the dose information provided in NUREG/CR-4551 (adjusted by estimated changes in population since the publication of that document) to be used as a consequence model to provide an estimate of the person-rem dose per reactor year associated with various scenarios. The change in plant risk is then evaluated based on the changes from the consequence model end states and also from the Large Early Release Frequency (LERF) end states.

The methodology described in EPRI TR-104285 uses a simplified Containment Event Tree to subdivide representative core damage frequencies into eight (8) containment response scenario types to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures dependent upon the core damage accident
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 general steps of this risk assessment are as follows:

- Quantify the baseline risk in terms of frequency per reactor year for each of the eight containment release scenario types identified in the EPRI report.
- Develop plant-specific person-rem (population dose) per reactor year at 50 miles for each of the eight containment release scenario types from the EPRI report. Note that a 50-mile region is chosen because of availability of data. Similar percent changes in dose would be expected if the analysis focused on the entire region as in the EPRI report, and as such, using the 50-mile region dose results is judged to provide representative results for the analysis.
- Evaluate the risk impact (i.e. the change in containment release scenario type frequency and population dose) of extending the ILRT interval to sixteen years.
- Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with Regulatory Guide 1.174.

For each of the eight containment release scenario types identified in the EPRI report, the baseline risk and population dose information are shown in Table ES-1.

Table ES-1
Base Case Mean Frequencies and Consequence Measures

Release Type (1)	Description	Frequency (per Rx-yr)	Person-rem (50-miles)	Person-rem/yr (50-miles)
1	No Containment Failure (Including successful venting)	2.94E-6	1.11E5	0.328
2	Large Isolation Failures (Failure to Close)	Negligible (2)	4.98E6	Negligible
3, 4, 5	Small Isolation Failures (Failure to Seal)	2.87E-11	4.98E6	1.43E-4
6	Other Isolation Failures (e.g., dependent failures)	Negligible (2)	4.98E6	Negligible
7	Failures Induced by Phenomena (Early and Late)	1.59E-6	3.70E6	5.87
8	Bypass (Interfacing System LOCA)	2.30E-9	3.78E6	8.70E-3
CDF	All CET End states	4.53E-6		6.21

(1) EPRI TR-104285 Containment Response Class
(2) No contributing cutsets appeared in the Level 2 CET results at a truncation of 1.0E-11/yr.

The impact associated with extending the Type A test frequency interval is investigated in a series of sensitivity cases. The results from those cases, measured as percent change with respect to population dose and Large Early Release Frequency (LERF), are shown in Table ES-2.

Table ES-2
PBAPS ILRT Extension Summary of Results

Case: Description	LERF (per Rx-yr)	Person-Rem/yr (50 miles-2000)
Case 0: Base Case (No ILRT Extension)	6.167E-8	6.21
Case 1: Best Estimate (ILRT Extension to sixteen years leads to a 16% increase in the probability of a pre-existing undetected leak)	6.168E-8 (+0.016%)	6.21 (+0.003%)
Case 2: Best Estimate Upper Bound (Probability of pre-existing leak is at upper bound value of 1.0E-2 instead of 5.0E-3)	6.173E-8 (+0.097%)	6.21 (+0.007%)
Case 3: Pessimistic Upper Bound (ILRT extension leads to a hundred fold increase in the probability of a pre-existing undetected leak)	7.127E-8 (+15.6%)	6.26 (+0.91%)

As is shown in Table ES-2, the best estimate of the impact from an extension in the ILRT interval to sixteen years is calculated to result in a 16% increase in the probability that a pre-existing undetected leak exists at Peach Bottom. This in turn leads to very marginal increases in the calculated Large Early Release Frequency and Population Dose (0.016% and 0.003%, respectively). Results are also obtained for the upper bound of the best estimate case based on a Pacific Northwest Laboratory reported upper bound of a pre-existing leakage from containment. In this case, the calculated increases in Large Early Release Frequency and Population Dose are slightly higher, but still very low (0.097% and 0.007%, respectively). Finally, a pessimistic sensitivity case for the upper bound is performed by increasing the probability of a pre-existing undetected leak by a factor of one hundred compared to their current best estimate values in the PBAPS Level 2 model. In that case, the calculated increases in Large Early Release Frequency and Population Dose are 15.6% and 0.91%, respectively.

The risk-informed treatment of regulatory issues is addressed by a series of Regulatory Guides. These Regulatory Guides use CDF and LERF as two of the quantitative parameters that are compared with acceptance guidelines to assess the magnitude of the changes in the risk profiles. Regulatory Guide 1.174 provides acceptance guidelines for determining the risk impact of plant-specific changes to the licensing basis. In that Regulatory Guide, a very small increase in risk (non-risk significant) is defined as a core damage frequency (CDF) change below 10^{-6}/yr and a large early release frequency (LERF) change below 10^{-7}/yr. For the ILRT extension, the calculated CDF does not change and only LERF is impacted. Because the guidance in Regulatory Guide 1.174 defines very small changes in LERF as below 1.0E-7/yr, increasing the ILRT interval to sixteen years can be seen to have very low risk significance.

Based upon the leak detection capabilities of BWRs with inerted containments, the probability that a pre-existing undetected leak would exist is quite low. Because of this low probability, the calculated population dose from Peach Bottom is dominated by containment failures that result from phenomena induced failures (e.g., early drywell shell melt-through or late containment over-pressurization) rather than pre-existing containment isolation failures. The best estimate results from this analysis indicate that very marginal increases in calculated population dose and Large Early Release Frequency (LERF) would result by extending the ILRT interval to sixteen years. Even an upper bound sensitivity case that increases the probability of pre-existing failures by a factor of one hundred leads to less than 1% increase in population dose and less than 1E-8/yr increase in LERF. As such, the ILRT extension to sixteen years is found to be of very low risk significance per Regulatory Guide 1.174.

1.0 PURPOSE

Revisions to 10CFR50, Appendix J 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 was less than normal containment leakage of $1.0L_a$. Peach Bottom had selected the revised requirements for its testing program with the next ILRT due during the upcoming refueling outage, 3R13, scheduled for the fall of 2001.

The purpose of this calculation is to provide a risk assessment of extending the currently allowed containment Type A integrated leak rate test (ILRT) to sixteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for three additional scheduled refueling outages for each of the Peach Bottom units. The risk assessment follows the guidelines from 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 request for a plant's licensing basis as outlined in Regulatory Guide 1.174 [3].

Previously, the NRC published a report, Performance Based Leak Test Program, NUREG-1493 [4], which analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. Chapter 5 of that report presents results using NUREG/CR-4551 population dose information for Peach Bottom. In that analysis, it was determined 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 help confirm these conclusions based on more recent models and available data.

2.0 METHODOLOGY

A simplified bounding analysis approach is used for evaluating the change in risk associated with increasing the test interval to sixteen years. The approach is consistent with that presented in EPRI TR-104285 [2] and NUREG-1493 [4]. The analysis uses the current Peach Bottom PRA model that includes a full Level 2 analysis of core damage scenarios and subsequent containment response resulting in various fission product release categories (including no release). The release category end states from the PBAPS Level 2 model have also been applied to align with those used by the NRC in NUREG/CR-4551 for Peach Bottom [5]. This categorization allows the population dose information provided in NUREG/CR-4551 (adjusted by estimated changes in population since the publication of that document) to be used as a consequence model to provide an estimate of the person-rem dose per reactor year associated with various scenarios. The results from the consequence model end states as described in Appendix A and from the base Level 2 model Large Early Release Frequency (LERF) end states are presented.

The four general steps of this risk assessment are as follows:

1) Quantify the baseline risk in terms of frequency per reactor year for each of the eight containment release scenario types identified in the EPRI report.
2) Develop plant-specific person-rem (population dose) per reactor year at 50 miles for each of the eight containment release scenario types from the EPRI report. Note that a 50-mile region is chosen because of availability of data. Similar percent changes in dose would be expected if the analysis focused on the entire region as in the EPRI report, and as such, using the 50-mile region dose results is judged to provide representative results for the analysis.
3) Evaluate the risk impact (i.e. the change in containment release scenario type frequency and population dose) of extending the ILRT interval to sixteen years.

4) Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with Regulatory Guide 1.174 [3].

This approach is based on the information and approaches contained in the previously mentioned studies.

- Consistent with the other industry containment leak risk assessments, the PBAPS assessment uses population dose as one of the risk measures. The other risk measure used in the PBAPS assessment is LERF.

- Consistent with TR-104285, the PBAPS risk assessment combines the PBAPS Level 2 PRA models with the PBAPS NUREG/CR-4551 Level 3 population dose models to perform the analysis.

- Consistent with TR-104285 and NUREG-1493, the PBAPS assessment uses information from NUREG-1273 [6] regarding the low percentage of containment leakage events that would only be detected by an ILRT to calculate the increase in the pre-existing containment leakage probability due to the testing interval extension.

3.0 GROUNDRULES

The following groundrules are used in the analysis:

- The PBAPS Level 1 and Level 2 internal events PRA model for Unit 2 provides representative results for the analysis. A Unit 3 PRA model is available, but it is judged that it will not provide any unique or additional insights compared to the results from the Unit 2 model.

- It is appropriate to use the PBAPS internal events PRA model as a gauge to effectively describe the risk change attributable to the ILRT extension. Fire and Seismic PRA models are not available for PBAPS, but it is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if fire and seismic events were to be included in the calculations for the base case and sensitivity cases.

- The Population Dose from NUREG/CR-4551 can be scaled by the change in population since that time. That is, the calculated doses are directly proportional to the population and adjusted by a factor increase obtained by estimating the population increase from 2000 census data compared to the 1980 census data that was used in NUREG/CR-4551.

- The dose results for the accident bins from NUREG/CR-4551 can be estimated by dividing the reported dose per year by the bin frequency.

- Dose results for containment isolation failures can be conservatively characterized by the NUREG/CR-4551 Release Bin 3 results (i.e., early containment failure in the drywell with RPV pressure at the time of vessel breach greater than 200 psi). Release Bin 3 represents the highest dose category from NUREG/CR-4551.

- Dose results for Interfacing System LOCA scenarios can be characterized by the NUREG/CR-4551 Release Bin 4 results (i.e., early containment failure in the drywell with RPV pressure at the time of vessel breach less than 200 psi). This categorization may not be truly representative of releases that may be associated with ISLOCA scenarios but NUREG/CR-4551 did not separate this category out into its own Release Bin. The impact on population doses from Interfacing System LOCAs is not altered by the proposed ILRT extension, but is accounted for in the EPRI methodology as a separate entry for comparison purposes. Since the ISLOCA contribution to population dose is fixed, no changes on the conclusions from this analysis will result from this assumption.

- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal. Containment isolation valves that fail to close during an accident and in response to a containment isolation signal are not integral to this analysis.

- Recovery of pre-existing leakage during an accident is not credited in this analysis.

- The assumptions used in applying the current Level 2 end states to coincide with the release category bins used in NUREG/CR-4551 for PBAPS are detailed in Appendix A.

4.0 DATA

4.1 OVERVIEW

Various industry studies on containment leakage risk assessment are briefly summarized here:

1) NUREG/CR-3539 [7]
2) NUREG/CR-4220 [8]
3) NUREG-1273 [6]
4) NUREG/CR-4330 [9]
5) EPRI TR-105189 [10]
6) NUREG-1493 [2]
7) EPRI TR-104285 [4]

The first study is applicable because it provides the basis for the threshold used in the PBAPS Level 2 PRA 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 the basis of the probability used in the PBAPS Level 2 PRA 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 last study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk.

NUREG/CR-3539 [7]

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

The information of this study was used as the basis in the PBAPS Level 2 PRA for determining the size of containment leakage pathways significant to risk. The PBAPS Level 2 PRA defined non-significant leakage pathways as those that would modify risk by less than 5%. Based on the study results of NUREG/CR-3539, as containment leak pathways surpass rates of 35%/day the change in the public risk as analyzed in the study is approximately 5%. A leak rate of 35%/day at containment design pressure equates to an equivalent diameter leak of slightly greater than 2 inches. Therefore, the PBAPS Level 2 containment isolation fault tree logic did not explicitly model containment atmosphere penetrations 2" or less in diameter.

NUREG/CR-4220 [8]

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories 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. It is the latter category that is applicable to the PBAPS Level 2 containment isolation modeling that is the focus of this risk assessment. This information was used in the PBAPS Level 2 PRA to estimate the probability of pre-existing containment leakage at the time of a core damage accident.

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. It should be noted that all of the 4 identified large leakage events were PWR events, and the assumption of a one-year duration is not applicable to an inerted containment such as PBAPS. To account for the quick identification of such a leak in the PBAPS inerted containment, the PBAPS Level 2 PRA assumes a 3 day duration for each event and calculates the unavailability as (5E-3) x (3 days / 365 days) = 4E-5/yr. This calculation is presented in NUREG/CR-4220 as an "upper bound" estimate for BWRs (presumably meaning "inerted" BWR containment designs).

NUREG/CR-4220 also calculates the probability of very large leaks related to open airlocks. This probability was estimated at 5E-5. However, this pre-existing containment failure mode is not applicable to an inerted containment such as PBAPS. The PBAPS Level 2 containment isolation fault tree includes this failure mode for completeness sake but assigns a probability of 3E-6 (the standard probability in the PBAPS Level 2 PRA for "negligible" probability events).

As a final note, the PBAPS Level 2 PRA uses the terms "small" and "large" to characterize the NUREG/CR-4220 "large" leak and "airlock/hatch" failure modes, respectively.

<u>NUREG-1273 [6]</u>

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.

<u>NUREG/CR-4330 [9]</u>

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 PBAPS ILRT test interval extension, as NUREG/CR-4330 focuses on leakage rate and the PBAPS 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."

<u>EPRI TR-105189 [10]</u>

The EPRI study TR-105189 is useful to the PBAPS ILRT test interval extension risk assessment because this EPRI study answers the question regarding the impact 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 risk 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 risk 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, and the other 5 were events involving loss of RHR and/or SDC due to ILRT/LLRT activities. This information was used in the EPRI study to estimate the risk benefit from reductions in testing frequencies.

NUREG-1493 [4]

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.

NUREG-1493 used information from NUREG-1273 regarding the low percentage of containment leakage events that would only be detected by an ILRT in the calculation of the increase in the pre-existing containment leakage probability due to the testing interval extension. NUREG-1493 makes the following assumptions in this probability calculation:

- The average time that a pre-existing leakage may go undetected increases with the length of the testing interval (and is ½ the length of the test interval)
- Only 3% of all pre-existing leaks can be detected only by an ILRT (i.e., and not by LLRTs)

This same approach is used in the PBAPS ILRT test interval extension risk assessment.

EPRI TR-104285 [2]

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the TR-104285 EPRI study is a quantitative evaluation of the impact of extending ILRT and 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 frequencies into eight (8) classes of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures dependent upon the core damage accident
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 . . ."*

Also note that the containment response classes described above are referred to as containment response scenario types in this current analysis to avoid confusion with the Accident Class designator used in the standard PBAPS PRA model as described in Table A2-4 of Appendix A of this report.

4.2 APPLICATION TO CURRENT ANALYSIS

In NUREG-1493 [4], it is noted that based on a review of leakage-rate testing experience, a small percentage (3%) of leakages that exceed current requirements are detectable only by Type A testing (ILRT). Further, in NUREG-1493 it is noted that the leakage rates observed in these few Type A test failures were only marginally above currently prescribed limits and could be characterized by a leakage rate of about two times the allowable.

Also in NUREG-1493 [4], it was assumed that the characteristic magnitude of leakages detectable only by ILRTs would not change, but the probability of leakage would change due to the longer intervals between tests. The change in probability was estimated by comparing the average time that a leak could exist without detection. For example, the average time that a leak could go undetected with a three-year test interval is 1.5 years (3 yrs / 2), and the average time that a leak could exist without detection for a ten-year interval is 5 years (10 yrs / 2). This change would lead to a 3.33 factor increase (5.0/1.5) in the probability of a leak that is detectable only by ILRT testing. However, since ILRTs have been demonstrated to improve the residual leak detection by only 3%(1), the interval change noted above would only lead to about a 10% increase (3.33 x 3%) in the probability of an undetected leak. Correspondingly, an extension of the ILRT interval to sixteen years can be estimated to lead to about a 16% increase (8.0/1.5 x 3%) in the probability of an undetected leak.

The PBAPS Level 2 model uses an NRC-sponsored evaluation of containment isolation failures, a Pacific Northwest Laboratory (PNL) report [6], in estimating the probability of a pre-existing undetected large leak from containment. The estimate of containment bypass due to large leakage events in all LWRs is reported to range from 0.001 to 0.01 in that report with a calculated value of 5E-03 per year. In the PBAPS Level 2 model, the leak frequency developed for LWRs using observed failures (only PWR failures

(1) Assumes that the Local Leak Rate Tests (LLRT) will continue to provide leak detection for the other 97% of leakages.

occurred) of 5E-03 per year is applied. As in NUREG-1493 [4] and in the EPRI report [2], a three-day detection and correction time is assumed because of the inerted containment in BWR facilities. This results in an estimated unavailability of 4E-5 (5E-03/yr x 3 days / 365 days per year) for leaks.

In the PBAPS Level 2 model, the leakage probability of pre-existing failures is separated into two different basic events (IS-07-02 for "large" airlock/hatch failures and ISAV-CIVS24F for "small" leakage failures). The impact from these two separate terms, however, is the same in the quantification; they both reflect pre-existing leakage from containment sufficient to lead to large early release fractions. Changes to these events are made to investigate the impact of extending the Type A test interval to 16 years. A summary of the changes made to these events in a series of sensitivity cases is shown below in Table 4-1. Note that since the PNL data was developed at the time that the test interval was three years, the extension to sixteen years is assumed to lead to a 16% increase in the probability of a pre-existing undetected leak in the best estimate case.

Table 4-1
PBAPS ILRT Extension Sensitivity Cases

Case: Description	**"Large" Pre-existing Failure (IS-07-02)[1]**	**"Small" Pre-existing Failure (ISAV-CIVS24F)**
Case 0: Base Case (No ILRT Extension)	3.0E-6	4.0E-5
Case 1: Best Estimate (ILRT Extension to sixteen years leads to a 16% increase in the probability of a pre-existing undetected leak)	3.5E-6	4.6E-5
Case 2: Best Estimate Upper Bound (Probability of pre-existing leak is at upper bound value of 1.0E-2 instead of 5.0E-3)	6.0E-6	8.0E-5
Case 3: Pessimistic Upper Bound (ILRT extension leads to a hundred fold increase in the probability of a pre-existing undetected leak)	3.0E-4	4.0E-3

[1] The open airlock/hatch is included in the PBAPS Level 2 model for completeness even though a negligible probability of failure is assigned because of the inerted BWR containment. Also for completeness, changes to this basic event value are included in the sensitivity case results reported here.

5.0 RESULTS

The results are developed in the following subsections:

- Section 5.1 – The baseline risk for Peach Bottom is developed based on the current assumptions in the Level 2 model regarding containment isolation failure probabilities.
- Section 5.2 – A series of sensitivity studies are developed and presented.
- Section 5.3 – The results for the base case and sensitivity cases with respect to Large Early Release Frequency (LERF) are presented.

A summary of the results is then provided in Section 6.

5.1 BASELINE RISK RESULTS

The first step in the analysis is to quantify the baseline risk in terms of frequency per reactor year for each of the eight containment release scenario types identified in the EPRI report [2]. Table 5-1 summarizes the results of this first step with the individual contributions detailed below.

- The Type 1 frequency (No Containment Failure) is obtained by subtracting the total Containment Event Tree (CET) end state frequency from the Level 1 core damage frequency (i.e., the sum of the "OK" end states from the Level 2 analysis) and also adding the other end states that do not result in containment failure, including venting scenarios. This latter portion is the sum of the Accident Progression Bins 7, 8, 9, and 10 from NUREG/CR-4551 as applied to the current PBAPS Level 2 model as described in Appendix A.

- The Type 2 frequency is obtained by examining the percent contribution of the CET end states from large pre-existing isolation failures (i.e., the Level 2 Fussell-Vesely value of basic event IS-07-02).
- The Type 3, 4, 5 frequency is obtained by examining the percent contribution to the CET end states from small pre-existing isolation failures (represented by basic event ISAV-CIVS24F).
- The Type 6 frequency is obtained by examining the CET end state results for dependent failure combinations of the modeled isolation valves in the Level 2 model. It turns out that dependent failures do not contribute to the overall release frequency because all individual valve isolation failures are so small (even when common cause terms are accounted for) that they are truncated out of the final Level 2 results for PBAPS.
- The Type 8 frequency is obtained from the total of the ISLOCA CETs.
- And finally, the remainder of the Level 2 CET end state results (i.e. that not accounted for in Type 1, 2, 3, 4, 5, 6 or 8) is then assigned to Type 7.

Table 5-1
Base Case Release Frequencies from the PBAPS Model

Containment Release Type(1)	Description	Frequency (Per Reactor-year)
1	No Containment Failure (Including Successful Venting)	2.94E-6
2	Large Isolation Failures (Failure to Close)	Negligible (2)
3, 4, 5	Small Isolation Failures (Failure to Seal)	2.87E-11
6	Other Isolation Failures (e.g., dependent failures)	Negligible (2)
7	Failures Induced by Phenomena (Early and Late)	1.59E-6
8	Bypass (Interfacing System LOCA)	2.30E-9
CDF	All CET End states (including very low and no release)	4.53E-6

(1) EPRI TR-104285 Containment Response Class
(2) No contributing cutsets appeared in the Level 2 CET results at a truncation of 1.0E-11/yr.

The second step is to then develop plant-specific person-rem (population dose) per reactor year at 50 miles for each of the eight containment release types. The population dose information was obtained by extending the current Peach Bottom Level 2 PRA results into the format used in NUREG/CR-4551 [5] and scaling the consequence analysis output based on those Level 2 results and updated demographic information of the surrounding communities based on 2000 census data. Appendix A provides details of the application of the Peach Bottom NUREG/CR-4551 consequences to the current PBAPS Level 2 PRA sequences, and also provides an update of the NUREG/CR-4551 consequences based on current population estimates. The baseline results for person-rem within a 50-mile region are shown in Table 5-2.

The frequency results from Table 5-1 are then combined with the person-rem results from Table 5-2 to estimate the baseline mean consequence measures. The results of that combined calculation are shown in Table 5-3.

Table 5-2
Containment Release Type Assignment from the PBAPS Consequence Model

EPRI TR-104285 Containment Release		PBAPS NUREG/CR-4551	
Scenario Type	**Dose (Person-Rem)**	**Accident Progression Bin**	**2000 Dose (Person-Rem)**
1	1.11E+5 [(1)]	7 (VB, No CF, Vent)	3.28E+6
		8 (VB, No CF, No Vent)	8.30E+3
		9 (No VB, No CF, No Vent)	3.44E+5
		10 (No core damage)	0.00
2	4.98E+6	3 (VB, Early DW, Hi Press)	4.98E+6
3, 4, 5	4.98E+6		
6	4.98E+6		
7	3.70E+6 [(1)]	1 (VB, Early WW, Hi Press)	2.92E+6
		2 (VB, Early WW, Lo Press)	1.84E+6
		5 (VB, Late WW)	2.24E+6
		6 (VB, Late DW)	3.84E+6
8	3.78E+6 [(2)]	4 (VB, Early DW, Lo Press)	3.78E+6

[(1)] Given that multiple NUREG/CR-4551 discrete scenarios apply to the broader EPRI type, the EPRI type dose is based on a weighted average (weights based on PBAPS PRA scenario frequencies) of the applicable NUREG/CR-4551 APB doses.

This approach is the more appropriate for the PBAPS ILRT risk assessment, that is, than simply applying the worst case APB dose. This latter approach would potentially mask the minor delta risk being calculated by the PBAPS ILRT risk assessment.

[(2)] No specific Release Bin for this category exists in NUREG/CR-4551. Assigned to Release Bin 4 in this analysis, but will not impact the calculated change for the proposed ILRT extension.

Table 5-3
Base Case Mean Consequence Measures

Release Type	Description	Frequency (per Rx-yr)	Person-rem (50-miles)	Person-rem/yr (50-miles)
1	No Containment Failure (Including successful venting)	2.94E-6	1.11E5	0.328
2	Large Isolation Failures (Failure to Close)	Negligible [1]	4.98E6	Negligible
3, 4, 5	Small Isolation Failures (Failure to Seal)	2.87E-11	4.98E6	1.43E-4
6	Other Isolation Failures (e.g., dependent failures)	Negligible [1]	4.98E6	Negligible
7	Failures Induced by Phenomena (Early and Late)	1.59E-6	3.70E6	5.87
8	Bypass (Interfacing System LOCA)	2.30E-9	3.78E6	8.70E-3
CDF	All CET End states	4.53E-6		6.21

[1] No contributing cutsets appeared in the Level 2 CET results at a truncation of 1.0E-11/yr.

5.2 SENSITIVITY CASE RESULTS

The sensitivity calculations are performed using the Peach Bottom Unit 2 Level 1 / Level 2 PRA model with the Level 2 end state assignments calculated for both the default values and to match the collapsed accident class bins from NUREG/CR-4551 as described in Appendix A. In each case, the values for the containment isolation failure basic events were modified as indicated in Table 4-1, and a complete model re-quantification was performed using WinNUPRA v1.2. The results are obtained at a truncation level of 1.0E-11 for all sequences with the exception of LERF sequences that are obtained at a truncation level of 1.0E-12.

5.2.1 Sensitivity Case 1

The first sensitivity case provides a best estimate of the anticipated change from extending the existing PBAPS ILRT interval to sixteen years. As discussed previously, based on the change in the average duration that a leak may go undetected, this is estimated to lead to a 16% increase in the probability that a pre-existing leak would be undetected compared to a probability value that was based on an ILRT interval of about three years.

This change results in a slightly greater than 16% increase in the Type 3,4,5 frequency since a few more cutsets are brought into the final equation at this value. However, this release type represents only a small contribution to the total calculated consequences. Therefore, increasing the probability of pre-existing failures by 16% from their base values leads to a negligible increase in total person-rem/yr as shown by the mean consequence measures reported for this sensitivity case in Table 5-4.

Table 5-4
Sensitivity Case 1 Mean Consequence Measures

Release Type	Description	Frequency (per Rx-yr)	Person-rem (50-miles)	Person-rem/yr (50-miles)
1	No Containment Failure (Including successful venting)	2.94E-6	1.11E5	0.328
2	Large Isolation Failures (Failure to Close)	Negligible [(1)]	4.98E6	Negligible
3, 4, 5	Small Isolation Failures (Failure to Seal)	3.62E-11	4.98E6	1.80E-4
6	Other Isolation Failures (e.g., dependent failures)	Negligible [(1)]	4.98E6	Negligible
7	Failures Induced by Phenomena (Early and Late)	1.59E-6	3.70E6	5.87
8	Bypass (Interfacing System LOCA)	2.30E-9	3.78E6	8.70E-3
CDF	All CET End states	4.53E-6		6.21

[(1)] No contributing cutsets appeared in the Level 2 CET results at a truncation of 1.0E-11/yr.

5.2.2 Sensitivity Case 2

The second sensitivity case was set up to provide a best estimate upper bound of the anticipated change from extending the ILRT interval to sixteen years. In this case, it is assumed that the probability that a pre-existing leak would be undetected goes to its upper bound value of 0.01 per year as reported in the PNL report [6]. With a three-day detection period assumed here as well, this, in effect doubles the assumed probability of pre-existing failures compared to the base case results. The mean consequence measure results from this case are shown in Table 5-5. Again, while there is a calculated increase in the consequences associated with the Type 2,3,4, and 5 results, the contribution to the total risk is only very marginal increase compared to the base case results.

Table 5-5
Sensitivity Case 2 Mean Consequence Measures

Release Type	Description	Frequency (per Rx-yr)	Person-rem (50-miles)	Person-rem/yr (50-miles)
1	No Containment Failure (Including successful venting)	2.94E-6	1.11E5	0.328
2	Large Isolation Failures (Failure to Close)	Negligible (1)	4.98E6	Negligible
3, 4, 5	Small Isolation Failures (Failure to Seal)	8.05E-11	4.98E6	4.01E-4
6	Other Isolation Failures (e.g., dependent failures)	Negligible (1)	4.98E6	Negligible
7	Failures Induced by Phenomena (Early and Late)	1.59E-6	3.70E6	5.87
8	Bypass (Interfacing System LOCA)	2.30E-9	3.78E6	8.70E-3
CDF	All CET End states	4.53E-6		6.21

(1) No contributing cutsets appeared in the Level 2 CET results at a truncation of 1.0E-11/yr.

5.2.3 Sensitivity Case 3

In the third sensitivity case, it is assumed that the probability that a pre-existing leak would be undetected increases by a factor of 100 compared to the current default values. This assumes that over the interval extension, some unforeseen mechanism that has not been experienced in previous ILRT results will come into the realm of possibilities thereby dramatically increasing the probability of pre-existing failures compared to the best estimate calculations. The mean consequence measure results from this case are shown in Table 5-6. in this case, a marginal increase (<1%) in the calculated total person-rem per year is apparent compared to the base case results.

Table 5-6
Sensitivity Case 3 Mean Consequence Measures

Release Type	Description	Frequency (per Rx-yr)	Person-rem (50-miles)	Person-rem/yr (50-miles)
1	No Containment Failure (Including successful venting)	2.93E-6	1.11E5	0.328
2	Large Isolation Failures (Failure to Close)	4.48E-10	4.98E6	2.23E-3
3, 4, 5	Small Isolation Failures (Failure to Seal)	9.17E-9	4.98E6	4.57E-2
6	Other Isolation Failures (e.g., dependent failures)	Negligible (1)	4.98E6	Negligible
7	Failures Induced by Phenomena (Early and Late)	1.59E-6	3.70E6	5.87
8	Bypass (Interfacing System LOCA)	2.30E-9	3.78E6	8.70E-3
CDF	All CET End states	4.53E-6		6.26

(1) No contributing cutsets appeared in the Level 2 CET results at a truncation of 1.0E-11/yr.

5.3 LARGE EARLY RELEASE FREQUENCY RESULTS

The risk-informed treatment of regulatory issues is addressed by a series of Regulatory Guides. These Regulatory Guides use CDF or LERF as two of the quantitative parameters that are compared with acceptance guidelines to assess the magnitude of the changes in the risk profiles. Regulatory Guide 1.174 [3] provides acceptance guidelines for determining the risk impact of plant-specific changes to the licensing basis. In that Regulatory Guide, a very small increase in risk (non-risk significant) is defined as a core damage frequency (CDF) change below 10^{-6}/yr and a large early release frequency (LERF) change below 10^{-7}/yr. For the ILRT extension, the calculated CDF does not change and only LERF is impacted.

In the PBAPS Level 2 model, the end state results are assigned based on a combination of the magnitude and timing of fission product releases for given accident scenarios. There are four magnitude categories (High, Medium, Low, and Low-low) and three timing categories (Early, Intermediate, and Late). Details of these assignments are described in the IPE model documentation for PBAPS [11]. The sequences with an end state categorization of High and Early are consistent with the general definition of LERF. The results presented below for LERF are then based on those sequences from the PBAPS Level 2 model that are categorized as High and Early releases.

The calculated LERF results from the base case and the three sensitivity cases described previously are shown in Table 5-7. As can be seen, the worst-case assumption from Sensitivity Case 3 of a 100-fold increase in the probability of pre-existing failures leads to about a 15.6% increase in LERF. This 15.6% increase, however, is characterized by an absolute frequency change of approximately 1.0E-8/yr that is well below the demarcation for very low risk significance. Because the guidance in Regulatory Guide 1.174 defines very small changes in LERF as below 1.0E-7/yr, increasing the ILRT interval to sixteen years can be seen to have very low risk significance.

Table 5-7
PBAPS ILRT Extension Large Early Release Frequency (LERF) Results

Case: Description	LERF (per Rx-yr)	Percent Change
Case 0: Base Case (No ILRT Extension)	6.167E-8	N/A
Case 1: Best Estimate (ILRT Extension to sixteen years leads to a 16% increase in the probability of a pre-existing undetected leak)	6.168E-8	+0.016%
Case 2: Best Estimate Upper Bound (Probability of pre-existing leak is at upper bound value of 1.0E-2 instead of 5.0E-3)	6.173E-8	+0.097%
Case 3: Pessimistic Upper Bound (ILRT extension leads to a hundred fold increase in the probability of a pre-existing undetected leak)	7.127E-8	+15.6%

6.0 SUMMARY AND CONCLUSIONS

A summary of the results obtained from this analysis is shown in Table 6-1. The best estimate of the impact from an extension in the ILRT interval to sixteen years is calculated to result in a 16% increase in the probability that a pre-existing undetected leak exists at Peach Bottom. This in turn leads to very marginal increases in the calculated Large Early Release Frequency and Population Dose (0.016% and 0.003%, respectively). Results were also obtained for the upper bound of the best estimate case based on the PNL [6] reported upper bound of a pre-existing leakage from containment. In this case, the calculated increases in Large Early Release Frequency and Population Dose are slightly higher, but still very low (0.097% and 0.007%, respectively). Finally, a pessimistic sensitivity case for the upper bound was performed by increasing the probability of a pre-existing undetected leak by a factor of one hundred compared to their current best estimate values in the PBAPS Level 2 model. In that case, the calculated increases in Large Early Release Frequency and Population Dose are 15.6% and 0.91%, respectively.

It can be noted that in all of the cases reported here that the percentage increase in LERF is higher than the percentage increase in population dose resulting from potential increases in the probability of undetected pre-existing containment isolation failures. This difference is directly attributable to the low probability of large early releases in the base Level 2 model. The population dose at 50 miles consists of contributions from large early releases, and also from medium early, large late, etc. The changes in the assumed probability of pre-existing containment isolation failures only influences the large early release fraction. Therefore, because the large early release frequency (LERF) at PBAPS is small to begin with in the base case Level 2 results representing less than 1.5% of the core damage frequency, the percent change to LERF resulting from increasing the containment isolation failure probability will be larger than the percent change to the total population dose by definition.

Table 6-1
PBAPS ILRT Extension Summary of Results

Case: Description	LERF (per Rx-yr)	Person-Rem/yr (50 miles-2000)
Case 0: Base Case (No ILRT Extension)	6.167E-8	6.21
Case 1: Best Estimate (ILRT Extension to sixteen years leads to a 16% increase in the probability of a pre-existing undetected leak)	6.168E-8 (+0.016%)	6.21 (+0.003%)
Case 2: Best Estimate Upper Bound (Probability of pre-existing leak is at upper bound value of 1.0E-2 instead of 5.0E-3)	6.173E-8 (+0.097%)	6.21 (+0.007%)
Case 3: Pessimistic Upper Bound (ILRT extension leads to a hundred fold increase in the probability of a pre-existing undetected leak)	7.127E-8 (+15.6%)	6.26 (+0.91%)

Based upon the leak detection capabilities of BWRs with inerted containments, the probability that a pre-existing undetected leak would exist is quite low. Because of this low probability, the calculated population dose from Peach Bottom is dominated by containment failures that result from phenomena induced failures (e.g., early drywell shell melt-through or late containment over-pressurization) rather than pre-existing containment isolation failures. The best estimate results from this analysis indicate that very marginal increases in calculated population dose and large early release frequency (LERF) would result by extending the ILRT interval to sixteen years. Even an upper bound sensitivity case that increases the probability of pre-existing failures by a factor of one hundred leads to less than 1% increase in population dose and less than 1E-8/yr increase in LERF. As such, the ILRT extension to sixteen years is found to be of very low risk significance per Regulatory Guide 1.174.

7.0 REFERENCES

[1] *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,* NEI 94-01, July 1995.

[2] *Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals,* EPRI, Palo Alto, CA TR-104285, August 1994.

[3] *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,* Regulatory Guide 1.174, July 1998.

[4] *Performance-Based Containment Leak-Test Program,* NUREG-1493, September 1995.

[5] *Evaluation of Severe Accident Risks: Peach Bottom, Unit 2,* Main Report NUREG/CR-4551, SAND86-1309, Volume 4, Revision 1, Part 1, December 1990.

[6] *Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3 'Containment Integrity Check',* NUREG-1273, April 1988.

[7] *Impact of Containment Building Leakage on LWR Accident Risk*, Oak Ridge National Laboratory, NUREG/CR-3539, ORNL/TM-8964, April 1984.

[8] *Reliability Analysis of Containment Isolation Systems*, Pacific Northwest Laboratory, NUREG/CR-4220, PNL-5432, June 1985.

[9] *Review of Light Water Reactor Regulatory Requirements*, Pacific Northwest Laboratory, NUREG/CR-4330, PNL-5809, Vol. 2, June 1986.

[10] *Shutdown Risk Impact Assessment for Extended Containment Leakage Testing Intervals Utilizing ORAM™,* EPRI, Palo Alto, CA TR-105189, Final Report, May 1995.

[11] *Individual Plant Examination Peach Bottom Atomic Power Station Units 2 and 3,* Volumes 1 and 2, Philadelphia Electric Company, 1992.

8.0 LIST OF ACRONYMS

APB – Accident Progression Bin (used in NUREG/CR-4551)

BWR – Boiling Water Reactor

CDF – Core Damage Frequency

CET – Containment Event Tree

DCH – Direct Containment Heating

F-V – Fussell-Vesely

ILRT – Integrated Leak Rate Test

ISLOCA – Interfacing System Loss of Coolant Accident

LERF – Large Early Release Frequency

LLRT – Local Leak Rate Test

LOCA – Loss of Coolant Accident

LWR – Light Water Reactor

MCCI – Molten Core-Concrete Interactions

MFCR – Mean Fractional Contribution to Risk (used in NUREG/CR-4551)

NRC – Nuclear Regulatory Commission

PBAPS – Peach Bottom Atomic Power Station

PDR – Population Dose Risk

PNL – Pacific Northwest Laboratory

PRA – Probabilistic Risk Assessment

PWR – Pressurized Water Reactor

RAW – Risk Achievement Worth

RPV – Reactor Pressure Vessel

APPENDIX A

PBAPS CONSEQUENCE MODEL

APPENDIX A – PBAPS CONSEQUENCE MODEL

A1.0 OVERVIEW

One of the risk measures used in this study to estimate the risk impact due to extending the ILRT interval at PBAPS is population dose (person-rem/year). This risk measure information is obtained from a Level 3 PSA model. A Level 3 model was created for PBAPS as part of NUREG-1150 and NUREG/CR-4551 [A1, A2]; however, while the Level 1 and 2 PSA models have been updated and enhanced to continually reflect plant changes since the publication of these NUREGs, the Level 3 model has not been updated.

Version 1.5 of the MACCS code [A3] was used to perform the PBAPS Level 3 PSA in NUREG/CR-4551. The analysis was performed specifically for Peach Bottom Unit 2 and includes data unique to that site. While that report provides thorough documentation of the Level 3 analysis, the results are not directly applicable today. Some of the characteristics of the site data have changed since the performance of NUREG/CR-4551 in 1990. This appendix documents the update to the NUREG/CR-4551 Peach Bottom consequences, performed in support of this risk assessment, to address changes to the area surrounding Peach Bottom.

A1.1 POPULATION CHANGES

The population estimate for the area surrounding the site used in the NUREG/CR-4551 analysis was based on 1980 census information. The recently released population data from the 2000 US census is used to update the NUREG/CR-4551 population data. First, data from Table 4.2-2 of NUREG/CR-4551 was used to calculate the population within a 50-mile radius of the plant (assuming a linear growth in population density away from the plant). The 2000 population estimate was then compared to the 1980 estimate to determine the factor increase in population dose per year. The population data used for this estimate is shown in the Tables A1-1 and A1-2. Table A1-1 provides the information presented in Table 4.2-2 of NUREG/CR-4551 and Table A1-2 summarizes the 2000 US census information.

TABLE A1-1
NUREG/CR-4551 Population Data

Distance from Plant (miles)	Population
1	118
3	1822
10	28,647
30	989,356
100	14,849,112
350	68,008,584
1000	154,828,144

To estimate the population within 50 miles of the site in 1980, a population density, $PD_{50(1980)}$, is calculated as follows:

$$PD_{50(1980)} = \frac{\left[\frac{\text{pop. within 100 miles}}{\left(3.14*100^2\right)} - \frac{\text{pop. within 30 miles}}{\left(3.14*30^2\right)}\right]}{70\text{ miles}} * 20\text{ miles} + \frac{\text{pop. within 30 miles}}{(3.14*30^2)}$$

Using the data from Table A1-1 results in a $PD_{50(1980)}$ value of 385 people per square mile or approximately 3.02E6 people within a 50-mile radius of the plant.

For the updated population estimate, data is available for population by county from the US Census Bureau's web site (http://www.census.gov). This data is used to estimate the population within a 50-mile radius of the plant. If the entire county falls within the 50-mile radius based on a review of an atlas containing a mileage scale and county borders, then the entire population can be included in the population estimate. Otherwise, a fraction of the population is counted based on the percentage of the county within the 50-mile radius. The land area within the 50-mile radius is estimated based on visual inspection of the map and the population of that area is estimated assuming uniform distribution of the population within the county. The results of this updated population estimate are presented in Table A1-2.

Table A1-2
Population Within 50 Miles of PBAPS (2000 US Census)

County Name	County Population: Total	County Population: Percent Within 50 Miles of PBAPS	Population Within 50 Miles of PBAPS
Delaware, PA	550,864	85%	468234
Montgomery, PA	750,097	15%	112515
Berks, PA	373,638	50%	186819
Lebanon, PA	120,327	75%	90245
Adams, PA	91,292	40%	36517
Dauphin, PA	251,798	40%	100719
Cumberland, PA	213,674	10%	21367
Carroll, MD	150,897	85%	128262
Queen Anne's, MD	40,563	60%	24338
Anne Arundel, MD	489,656	30%	146897
Howard, MD	247,842	50%	123921
Salem, NJ	64,285	50%	32143
Gloucester, NJ	254,673	20%	50935
Kent, DE	126,697	25%	31674
York, PA	381,751	100%	381751
Lancaster, PA	470,658	100%	470658
Chester, PA	433,501	100%	433501
Baltimore, MD	754,292	100%	754292
Baltimore City, MD	651,154	100%	651154
Harford, MD	218,590	100%	218590
Cecil, MD	85,951	100%	85951
Kent, MD	19,197	100%	19197
New Castle, DE	500,265	100%	500265
		Total =	5069945

With an estimated updated population of approximately 5.07E6 people compared to an estimated 3.02E6 people used in the NUREG/CR-4551 analysis, the population dose for this analysis is increased by a factor of 1.68 (i.e. 5.07E6/3.02E6) from that reported in NUREG/CR-4551 for each of the release category bins.

A1.2 ECONOMY AND AGRICULTURE

As part of NUREG/CR-4551, site-specific data was collected on the economic and agricultural characteristics surrounding the Peach Bottom site. It is assumed that the relative distribution of these factors has remained constant and that the overall growth in "economy" and "agriculture" is represented by the growth in population. Therefore, no additional changes in the dose results are incorporated into this analysis based on changes in "economy" and "agriculture".

A1.3 OTHER PLANT SPECIFIC DATA

MACCS, as utilized in NUREG/CR-4551, implemented a large, plant specific input file to account for other site aspects. These factors include evacuation characteristics, meteorological data, and core inventories that affect the Level 3 analysis. This data is available, including the economic and agricultural demographics, in Volume 2, Part 7 of NUREG/CR-4551. It is assumed that this remaining plant specific data is constant or is treated by the application of the population growth ratio. No changes have been made to update the original input other than the scaling of the population estimates that is described above.

The Peach Bottom generating capacity has been increased from 3293 $MW_{thermal}$ per unit to 3458 $MW_{thermal}$ per unit since the time the NUREG/CR-4551 analysis was performed. The Peach Bottom PSA accounts for the power uprate in the application of success criteria and event timing. The Level 3 results have not been modified to account for the change in fuel design that accompanied the power uprate as the corresponding impact on core inventory is considered to be insignificant compared with the variation that occurs within the core during the course of a fuel cycle. Any such impacts are bounded by the sensitivity case quantifications performed in this risk assessment.

A2.0 APPLICATION OF PBAPS PSA MODEL RESULTS TO NUREG/CR-4551 LEVEL 3 OUTPUT

A major factor related to the use of NUREG/CR-4551 in this evaluation is that the PBAPS PSA has been enhanced to reflect plant changes and new information. While consistent with the IPE, the level of sophistication of the PSA model has increased and the results have changed as modeling techniques have improved. In addition, the results of the PBAPS PSA Level 2 model are not defined in the same terms as reported in NUREG/CR-4551. In order to use the Level 3 model presented in that document, it was necessary to apply the PBAPS PSA Level 2 model results into a format which allowed for the scaling of the Level 3 results based on current Level 2 output. Finally, as mentioned above, the Level 3 results were modified to reflect the change in the site demographics that have occurred since the publication of NUREG/CR-4551. This subsection provides a description of the process used to apply the PBAPS PSA Level 2 model results into a form that can be used to generate Level 3 results using the NUREG/CR-4551 documentation. The Unit 2 PSA model, which has a slightly higher CDF between the Unit 2 and Unit 3 models, is used for the calculations in this study.

A2.1 ASSIGNMENT OF PBAPS LEVEL 2 ENDSTATES TO THE COLLAPSED ACCIDENT PROGRESSION BINS USED IN NUREG/CR-4551

The basic process that was pursued to obtain Level 3 results based on the PBAPS PSA Level 2 model and NUREG/CR-4551 was to define a useful relationship between the Level 2 and Level 3 results. Consequently, each sequence of the PBAPS PSA Level 2 model was reviewed and assigned into one of the collapsed Accident Progression Bins (APBs) from NUREG/CR-4551. The Level 2 model contains a significantly larger amount of information about the accident sequences than what is used in the collapsed APBs in NUREG/CR-4551 and this assignment process required simplification of accident progression information and assumptions related to categorizations of certain items. The assumptions used for these assignments are discussed later and shown in Table A2-5.

In NUREG/CR-4551, the collapsed APBs are characterized by 5 attributes related to the accident progression. Unique combinations of the 5 attributes result in a set of 10 bins that are relevant to the analysis. Information from the PBAPS PSA Containment Event Trees (CETs) was used to classify each of the Level 2 sequences using these attributes. The definitions of the 10 collapsed APBs are provided in NUREG/CR-4551 and are reproduced in Table A2-3 for references purposes.

Table A2-3
Collapsed Accident Progression Bin (APB) Descriptions

Collapsed APB Number	Description
1	CD, VB, Early CF, WW Failure, V 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 DCH is possible).
2	CD, VB, Early CF, WW Failure, V 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, V 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, V 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 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.

Table A2-3
Collapsed Accident Progression Bin (APB) Descriptions

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, No CF, 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.

Acronyms in Table A2-3:

CD – Core Damage
VB – Vessel Breach
CF – Containment Failure
WW – Wetwell
DW – Drywell
Additional acronyms can be found in Section 8 of the main report.

Some general assumptions were made during the classification of the Level 2 CET sequences in order to categorize certain sequences that contained characteristics that did not directly fit into one of the 10 collapsed APBs. As it is possible for these assumptions to vary between each of the 5 accident classes, each accident class is associated with a unique set of assumptions on a node-by-node basis. The "nodes" in the CETs represent phenomenological events, operation of plant systems, and operator performance. Table A2-4 summarizes the accident class definitions and Table A2-5 summarizes the nodal assumptions used to group the PBAPS PSA Level 2 sequences into the collapsed bins.

Table A2-4

PBAPS Core Damage Accident Class Definitions

Accident Class Designator	Definition
1A	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high.
1B	Accident sequences involving a loss of offsite power and loss of inventory makeup.
1C	Accident sequences involving a mitigated ATWS scenario with subsequent loss of inventory makeup.
1D	Accident sequences involving a loss of coolant inventory makeup in which reactor pressure has been successfully reduced to 200 psi.
1E	Accident sequences resulting from a common mode failure of DC batteries or buses.
2A	Accident sequences involving a loss of containment heat removal (including no venting capability), but with a LOCA or stuck open relief valve preventing vessel re-pressurization prior to containment failure.
2F	Accident sequences involving a loss of containment heat removal and with injection lost following venting of containment.
2T	Accident sequences involving a loss of containment heat removal and no venting with injection terminated prior to containment failure.
3A	Accident sequences leading to core vulnerable conditions initiated by vessel rupture where the containment integrity is not breached in the initial time phase of the accident.
3B	Accident sequences initiated by or resulting in small or intermediate LOCAs for which the reactor is not fully depressurized.
3C	Accident sequences that are initiated by a LOCA or RPV failure and for which the vapor suppression system is inadequate challenging containment integrity.

Table A2-4

PBAPS Core Damage Accident Class Definitions

Accident Class Designator	Definition
4A	Accident sequences involving a failure to insert negative reactivity leading to a containment vulnerable condition due to high containment pressure.
5	Unisolated LOCA outside containment.

Table A2-5

Nodal Assumptions

Accident Class	PBAPS PSA Containment Event Tree Node	Assumption
1	IS – Containment Isolation	If the containment is not isolated, it is assumed that it will be open for the equivalent of an un-scrubbed release as soon as the vessel is breached. No depressurization is asked prior to this node; it is assumed that RPV pressure is >= 200 psi for these sequences. This is bin #3.
	OP – Operator depressurizes the RPV	It is assumed that success on this branch results in RPV pressure below 200 psi.
	RX – Core Melt Arrested in Vessel	A success on this branch signifies that there is no vessel breach. The sequences following this path are grouped in bin #9. However, there is one case in which combustible gas venting (GV) fails followed by containment failure (CZ); this is assumed to result in a high early release and is categorized as a bin #4 event for low pressure and #3 for high pressure.
	CX – Containment Intact During Flood, RPV Breach	Failure of containment during flood is assumed to result in an un-scrubbed release. The timing is technically later than vessel breach, but it is conservatively assumed to be "early" and is grouped in bins 3 or 4 depending on RPV pressure.
	NC – No Large Containment Failure	A large containment failure instigated by high containment pressure following vessel breach is assigned to the "late containment failure" bins. The sequences contributing to these bins need to be separated into either WW or DW failures. While the PB CETs distinguish between these types of failures, the NUREG/CR-4551 analysis takes credit for scrubbing for any WW release (with respect to the collapsed bins in Section 2.4.3). Not all WW failure in the CETs can be credited with successful scrubbing. Given a large containment failure, the only successful scrubbing path is that in which the WW fails in an area above the water level (success in node WW).
	MU – Coolant Inventory Makeup	Coolant inventory makeup is assumed only to provide cooling to the core debris. No credit is taken for any potential scrubbing effects that water coverage may yield.

Table A2-5

Nodal Assumptions

Accident Class	PBAPS PSA Containment Event Tree Node	Assumption
1 (Continued)	RB – Release Mitigated in Reactor Building	The RB node, release mitigated in reactor building, is not credited as a scrubbing mechanism. The only scrubbing accounted for in the collapsed bins is distinguished by indicating a WW release and the amount of scrubbing that the reactor building is capable of providing is not considered to be the equivalent a WW scrub.
2	RX – Core Melt Arrested in Vessel	A success on this branch signifies that there is no vessel breach. The sequences following this path are grouped in bin #9. However, for accident class 2T sequences in which core melt has been mitigated in the vessel, a failure in the CZ node is also assumed to result in bins 3 or 4 according to RPV pressure.
	CZ/SI – Containment Intact/Mark I Shell Failure	Given that the core melt has not been contained in the RPV, failure in node CZ is assumed to result in an un-scrubbed release through the drywell. Failure in node SI is also assumed to result in an un-scrubbed release due to fission product release through the gap between the liner and the concrete. No credit is given to reactor building scrubbing (RB) or to injection to the DW or RPV (TD). The sequences with failures in these nodes are assigned to bins 3 or 4 depending on RPV pressure.
	RB – Release Mitigated in Reactor Building	The RB node, release mitigated in reactor building, is not credited as a scrubbing mechanism. The only scrubbing accounted for in the collapsed bins is distinguished by indicating a WW release and the amount of scrubbing that the reactor building is capable of providing is not considered to be the equivalent a WW scrub.
	SP – Suppression Pool Not Bypassed	The suppression pool bypass node is considered in the PB CETs to determine whether the vent volume passes through the suppression pool or not. This node is currently only quantified for cases in which the core melt has been arrested in the RPV (no VB breach). These sequences are assigned to bin #9 and no further breakdown of the sequences is performed.
	MU – Coolant Inventory Makeup	Coolant inventory makeup is assumed only to provide cooling to the core debris. No credit is taken for any potential scrubbing effects that water coverage may yield.
	RB – Release Mitigated in Reactor Building	The RB node, release mitigated in reactor building, is not credited as a scrubbing mechanism. The only scrubbing accounted for in the collapsed bins is distinguished by indicating a WW release and the amount of scrubbing that the reactor building is capable of providing is not considered to be the equivalent a WW scrub.

Table A2-5

Nodal Assumptions

Accident Class	PBAPS PSA Containment Event Tree Node	Assumption
3 (Continued)	SP – Suppression Pool Not Bypassed	The suppression pool bypass node is considered in the PB CETs to determine whether the vent volume passes through the suppression pool or not. This node is quantified in Class 3 accidents for both vessel breach and "no breach" cases. For no vessel breach: Bin #9 is assigned unless there is a failure in the CZ node. A failure in the CZ node denotes early containment failure and these sequences are assigned to bin #4 (depressurization is always successful in the Class 3 trees, so there is no use of bin #3.) For vessel breach: If the WW is not bypassed, bin #7 is assigned, which is in accord with the bin definition of "vessel breach, vent". If the WW is bypassed, the conditions are assumed to be similar to bin #6 as the venting will take place late in time as would a late containment failure and the un-scrubbed vent volume will be vented directly to the atmosphere through the stack.
	CZ/SI – Containment Intact/Mark I Shell Failure	Given that the core melt has not been contained in the RPV, failure in node CZ is assumed to result in an un-scrubbed release through the drywell. Failure in node SI is also assumed to result in an un-scrubbed release due to fission product release through the gap between the liner and the concrete. No credit is given to reactor building scrubbing (RB) or to injection to the DW or RPV (TD). The sequences with failures in these nodes are assigned to bins 3 or 4 depending on RPV pressure.
4	RB – Release Mitigated in Reactor Building	The RB node, release mitigated in reactor building, is not credited as a scrubbing mechanism. The only scrubbing accounted for in the collapsed bins is distinguished by indicating a WW release and the amount of scrubbing that the reactor building is capable of providing is not considered to be the equivalent a WW scrub.
	SP – Suppression Pool Not Bypassed	The suppression pool bypass node is considered in the PB CETs to determine whether the vent volume passes through the suppression pool or not. This node is quantified in Class 4 accidents for only "no breach" cases. For no vessel breach Bin #9 is assigned.

Table A2-5

Nodal Assumptions

Accident Class	PBAPS PSA Containment Event Tree Node	Assumption
4 (Continued)	CZ/SI – Containment Intact/Mark I Shell Failure	Given that the core melt has not been contained in the RPV, failure in node CZ is assumed to result in an un-scrubbed release through the drywell. Failure in node SI is also assumed to result in an un-scrubbed release due to fission product release through the gap between the liner and the concrete. No credit is given to reactor building scrubbing (RB) or to injection to the DW or RPV (TD). The sequences with failures in these nodes are assigned to bins 3 or 4 depending on RPV pressure.
5	N/A	No collapsed bin is available for containment bypass scenarios. The closest match to a bypass scenario is assumed to be a vessel breach with early drywell failure (bins 3 and 4). These bins are assigned based on RPV pressure (failure to depressurize is set to 0.0, so all sequences with non-zero results will be assigned to bin #4).

A2.2 DETERMINATION OF POPULATION DOSE RISK (0-50 MILES)

NUREG/CR-4551 defines the fractional contribution of the 10 collapsed Accident Progression Bins (APBs) to the Population Dose Risk at 50 miles (PDR50). It was determined that the frequency of each collapsed APB could be calculated based on the information provided in NUREG/CR-4551. Given this relationship, it was possible to determine the PDR50 based on the results of the PBAPS PSA model with the results reported in terms of the same accident bins. For example, for a given collapsed APB:

$$PDR50_{(PBAPSPSA)} = \frac{PBAPS\,PSA\,Frequency}{NUREG/CR-4551\,Frequency} * \mathrm{Re}duced\,APB\,Fractional Contribution * Total\,PDR50_{(NUREG/CR-4551)}$$

If this is performed for each of the 10 collapsed APBs and the results are summed, the total is the PDR50 for the PBAPS PSA. Additionally, the PDR50 results for the PBAPS

PSA model are scaled by a factor of 1.68 to account for the estimated impact to account for the change in the surrounding demographics as described previously. Table A2-6 summarizes the results of this process.

Table A2-6
Calculation of PBAPS Population Dose Risk at 50 Miles

Collapsed Bin #	Fractional APB Contributions to Risk (MFCR)[1]	NUREG/CR-4551 Population Dose Risk at 50 miles (From a total of 7.9 person-rem, mean)[2]	NUREG/CR-4551 Collapsed Bin Frequencies[3] (per year)	PBAPS PSA Collapsed Bin Frequencies[4] (per year)	PBAPS PSA Population Dose Risk at 50 miles (MCFR) (1980 Pop Data)[5] (person-REM/yr)	Population Dose Risk at 50 miles (PBAPS PSA, scaled to 2000 population) (person-REM/yr)
1	0.021	0.1659	9.55E-08	0	0.00E+00	0.00E+00
2	0.0066	0.05214	4.77E-08	0	0.00E+00	0.00E+00
3	0.556	4.3924	1.48E-06	4.66E-08	1.38E-01	2.32E-01
4	0.226	1.7854	7.94E-07	1.42E-06	3.19E+00	5.38E+00
5	0.0022	0.01738	1.30E-08	1.17E-07	1.56E-01	2.63E-01
6	0.059	0.4661	2.04E-07	2.01E-09	4.59E-03	7.72E-03
7	0.118	0.9322	4.77E-07	2.25E-08	4.39E-02	7.38E-02
8	0.0005	0.00395	7.99E-07	1.42E-08	7.02E-05	1.18E-04
9	0.01	0.079	3.86E-07	7.38E-07	1.51E-01	2.54E-01
10	0	0	4.34E-08	0	0.00E+00	0.00E+00
Totals	1.0	7.9	4.34E-6	2.36E-6	3.70	6.21

Notes to Table A2-6:

1. Mean Fractional Contribution to Risk from Table 5.2-3 of NUREG/CR-4551
2. The total population dose risk at 50 miles from internal events in person-rem is provided in Table 5.1-1 of NUREG/CR-4551. The contribution for a given APB is the product of the total PDR50 and the fractional APB contribution.
3. NUREG/CR-4551 provides the conditional probabilities of the collapsed APBs in Figure 2.5-6. These conditional probabilities are multiplied by the total internal CDF to calculate the collapsed APB frequency.
4. Determined by re-grouping PBAPS PSA results into the 10 collapsed APBs.
5. This column is the ratio of the PBAPS PSA collapsed APB frequency to the NUREG/CR-4551 collapsed APB frequency multiplied by the NUREG/CR-4551 APB specific PDR50 contribution.

A3.0 SUMMARY

Given the change in the current PBAPS Level 2 PSA models compared to the NUREG/CR-4551 models (refer to Section A1.0), the APB frequencies differ between the two. The APBs with the most influence on the PDR50 from the NUREG/CR-4551 analysis are 3, 4, and 7. In the current analysis, the frequency for APB 3 dropped by about 2 orders of magnitude relative to NUREG/CR-4551 and as a result, this bin is no longer the dominant contributor to the PDR50. Conversely, the frequency of bin 4 increased by a factor of 2 and this bin now contributes about 87% of the PDR50. APB 7 was reduced in frequency by a factor of 20 and remains as a significant, but non-dominant contributor to the results. Bins 5 and 9 also increased in frequency compared to the NUREG/CR-4551 results and as such can also be considered as significant, but non-dominant contributors to the results.

It is also important to note that there were no Level 2 sequences categorized in APBs 1 or 2. This is primarily due to the assumption that failure on the SI node (shell melt through) results in an un-scrubbed release. The collapsed APBs treat a wetwell release as a scrubbed release. Thus, the SI failures (this node is set to 1.0) are binned with the drywell failures to prevent un-scrubbed sequences from being categorized with the scrubbed releases. An early failure of containment due to the effects of vessel breach (CZ) is also assumed to result in an un-scrubbed release and therefore is not binned in APBs 1 or 2.

As shown in Table A2-6, the end result is a baseline PDR50 of 6.2 person-rem per year based on the scaled population data for 2000. The majority of this value is contributed by the Bin 4 category that is comprised of scenarios involving Core Damage followed by vessel breach with early containment failure in the drywell (i.e., either before core damage, during core damage, or at vessel breach). A change to the PDR50 value of 6.2 person-rem per year is used in this analysis as one of the figures of merit in evaluating the effects of the proposed ILRT interval extension. Additionally, a change in the calculated Large Early Release Frequency (LERF) from the base PBAPS Level 2 model (i.e. – based on the CET end states prior to the re-categorization described here) is used as an additional figure of merit in this analysis.

A4.0 REFERENCES

[A1] *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*, U. S. Nuclear Regulatory Commission, NUREG-1150, Washington, D.C., June 1989.

[A2] A. C. Payne, R. J. Breeding, H. –N. Jow, J. C. Helton, L. N. Smith, A. W. Shiver, *Evaluation of Severe Accident Risks: Peach Bottom, Unit 2,* Main Report NUREG/CR-4551, SAND86-1309, Volume 4, Revision 1, Part 1, December 1990.

[A3] D. I. Chanin, J. L. Sprung, L. T. Ritchie and H. –N. Jow, *MELCOR Accident Consequence Code System (MACCS): User's Guide*, NUREG/CR-4691, SAND86-1562, Volumes 1-3, Sandia National Laboratories, February 1990.